STAFF EVALUATION REPORT OF
INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SUBMITTAL
ON DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

I. INTRODUCTION

On June 28, 1991, the NRC issued Generic Letter (GL) 88-20, Supplement 4 (with NUREG-1407, *Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities*) requesting all licensees to (1) perform individual plant examinations of external events to identify plant-specific vulnerabilities to severe accidents, and (2) report the results to the Commission together with any licensee-determined improvements and corrective actions. In a letter dated December 30, 1997, the licensee, Commonwealth Edison Company, submitted its response to the NRC.

The staff contracted with Brookhaven National Laboratory and Sandia National Laboratories (SNL) to conduct screening reviews of the licensee's IPEEE submittal in the seismic and fire areas, respectively. The NRC staff conducted a screening review in the high winds, floods, and other (HFO) external events area of the submittal. Based on the review, the staff sent a request for additional information (RAI) to the licensee on December 14, 1998, in the seismic and fire areas. There were no RAIs in the HFO events area. The licensee responded to the RAI in a letter dated March 30, 2000, which also included revisions to certain sections of the IPEEE submittal. In particular, the original executive summary for the licensee's IPEEE and the seismic and fire sections were completely replaced with revised sections including additional and updated information. In addition, the revised fire section included results for fire events based on the use of analytical models that have less conservatism than in the original analysis. As a result of the use of the revised fire models, the associated core damage frequency (CDF) contributions were smaller than those reported in the original submittal. After reviewing the licensee's RAI response, the staff concluded that additional information was needed to complete its review in the fire area, and a supplementary RAI was sent to the licensee on December 17, 2000. The licensee responded to the supplementary RAI in a letter dated January 31, 2001. Based on the results of the review of the submittal and the licensee's responses to the RAIs, the staff concludes that the aspects of seismic events, fires, and HFO events have been adequately addressed. The review findings are summarized in the evaluation section below. Details of the staff's and contractors' findings are presented in the three technical evaluation reports (TERs) attached to this staff evaluation report.

In accordance with Supplement 4 to GL 88-20, the licensee provided information to address the resolution of Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," Generic Safety Issue (GSI) 103, "Design for Probable Maximum Precipitation," GSI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment," and the Sandia Fire Risk Scoping Study (FRSS) issues. These issues were explicitly requested in Supplement 4 to GL 88-20 and its associated guidance in NUREG-1407. The licensee did not propose to resolve any additional USIs or GSIs as part of their IPEEE.

An IPEEE Senior Review Board (SRB) was established and meets on a regular basis. The purposes of the SRB are (1) for the contractor to present the findings and conclusions of its review and the bases for its conclusions, and (2) for the SRB members to provide their perspectives on the contractor's findings and conclusions and to make recommendations based on their technical expertise. In this manner, the SRB provides additional assurance that (1) the scope of the review meets the objectives of the program, and (2) critical issues that have the potential to mask vulnerabilities are not overlooked.

II. EVALUATION

Dresden Nuclear Power Station (DNPS), Units 2 and 3, consists of two similar units of the General Electric BWR/3 design which are housed in Mark I containments. The two units are both rated at an electrical power output of 809 MWe. DNPS is presently owned and operated by Exelon Corporation (previously Commonwealth Edison Company) and is located in northeastern Illinois near the town of Morris in Grundy County. The north and east boundaries of the plant site are formed by the Illinois and Kankakee rivers. DNPS Unit 2 and Unit 3 started commercial operation in June 1970 and November 1971, respectively.

For the seismic analysis, DNPS is categorized as a 0.3g focused-scope plant (per NUREG-1407, *Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities*). To perform the seismic evaluation, the licensee used the Electric Power Research Institute's (EPRI) seismic margins assessment methodology as described in EPRI NP-6041-SL, *A Methodology for Assessment of Nuclear Power Plant Seismic Margin*. The Dresden site structures are founded on rock, and the original design basis earthquake is a Housner-type spectrum with a peak ground acceleration of 0.20g for the horizontal safe shutdown earthquake (SSE). The operating basis earthquake (OBE) is equal to one-half of the SSE spectrum. The vertical SSE ground spectrum is two-thirds of the horizontal. For fire events, the licensee performed a fire probabilistic safety assessment based on EPRI's Fire-Induced Vulnerability Evaluation (FIVE) methodology (EPRI TR-100370) and the EPRI Fire PRA Implementation Guide (FPRAIG) (EPRI TR-105928). The licensee evaluated HFO events using the progressive screening approach consistent with the guidance in NUREG-1407, *Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities*. Since DNPS was designed and constructed prior to the issuance of the 1975 Standard Review Plan (SRP), the plant was not designed according to the SRP criteria. The licensee performed walkdowns to confirm that no plant changes had occurred since the plant was licensed that would impact on the IPEEE review.

Core Damage Frequency Estimates

The licensee did not quantitatively estimate a seismic core damage frequency (CDF) contribution, since a seismic margin assessment was performed. The licensee stated in the submittal that its seismic margin analysis indicated that the overall high-confidence-of-low-probability-of-failure (HCLPF) plant capacity for some components was less than the review level earthquake level (RLE) of 0.3g. In the response to the staff's RAI, the licensee identified some components whose HCLPF values ranged between 0.2 and 0.3, including the Condensate Storage Tank (0.20g, controlled by tank buckling), the Diesel Fuel Oil Storage Day Tank (0.26g, controlled by an adjacent masonry wall), and the Torus Suppression Chambers (0.28g, controlled by torus shell stress). The status of these components will be discussed further in the following section of this report regarding dominant contributors.

The licensee estimated that the contribution to plant CDF from fires was approximately 1.7E-5 per reactor-year (ry) for Unit 2 and 3.1E-5/ry for Unit 3 based on the use of improved fire risk models that have less conservatism than the licensee's original fire analysis. The licensee did not quantitatively estimate the CDF contribution from HFO events since these events were screened out using an approach consistent with NUREG-1407. In its Individual Plant Examination (IPE) submittal, the licensee estimated the total CDF from internal events to be approximately 1.9E-5/ry for both units, including internal flooding.

Dominant Contributors

As indicated above, the licensee stated that the DNPS plant's HCLPF capacities, based on the EPRI assessment methodology were, in some cases, less than the RLE value of 0.3g. The limiting (lowest) HCLPF values, as reported in the licensee's March 30, 2000, RAI response, Section 1.4.1, were electrical buses (0.20g), electrical distribution panels (0.20g), and condensate storage tanks (0.20g). A number of other components (including a diesel fuel oil storage day tank, a battery charger, switchgear, battery bus, torus suppression chambers, and motor control centers) ranged in HCPLF values from 0.26g to 0.29g. The staff reviewed the licensee's response to RAIs concerning the HCPLF values that were below the RLE value. In discussing the responses with the SRB, it was concluded that the licensee's method for evaluating the HCPLFs, which followed the procedure in EPRI NP-6041, is acceptable. The licensee stated that a number of improvements have been made in the seismic area, primarily in equipment anchorages, during the resolution of the USI A-46 program and that further improvements were being considered. The licensee stated that it is intended that all IPEEE components are to have a seismic capacity that meet design basis requirements (the SSE level of 0.2g). The licensee also stated that this activity will be completed with the resolution of USI A-46 requirements consistent with the USI A-46 resolution schedule.

For fire events, the licensee reported that the compartment fires that represented the main contributors to the fire-related CDF were the Unit 2/3 Control Room (approximately 7.1E-6/ry), the Unit 3 West Corridor and Trackway (6.9E-6/ry), the Unit 2 Trackway/Switchgear Area (5.4E-6/ry), the Unit 3 Reactor Building Mezzanine (3.5E-6/ry), the Unit 3 Mezzanine floor (3.4E-6/ry), and the Unit 2/3 Auxiliary Electric Equipment Room (2.5E-6/ry).

The contribution to CDF from the HFO-related events was not quantitatively estimated (since these events were screened), and so there was no relative ranking of contributors for HFO events.

The licensee's IPEEE assessment appears to have examined the significant initiating events and dominant accident sequences.

Containment Performance

The licensee addressed the seismic-related containment issues in Section 3.5 of the submittal. To evaluate potential seismic-induced containment failures, the licensee performed a specific plant walkdown. The licensee stated that the focus of the walkdown was to evaluate unusual conditions, such as potential spatial interactions, active seals, unique containment penetration configurations, and bypass systems. The licensee's submittal states that no vulnerabilities were found in this area.

The licensee addressed the fire-related containment issues in Section 4.8 of the submittal. The licensee stated that studies were performed to identify any vulnerability that could lead to early containment failure during a fire, including structures, systems and components needed to ensure containment integrity, containment isolation, and prevention of containment bypass. The submittal indicated that fire impact on the containment was expected to be minimal and that no fire-related vulnerabilities were identified that could cause early containment failure or containment bypass. It was noted that except for brief periods after startup, before a shutdown, or for infrequent drywell entries at power, the primary containment is inerted with nitrogen, further reducing the risk of a fire at power.

The licensee's containment performance analyses for seismic and internal fire events appear to have considered important containment performance issues 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. These safety issues were evaluated by the NRC's contractors. The results of these evaluations are contained in the attached TERs. For those safety issues that were not completely resolved by the contractors, the NRC staff performed additional reviews. The final resolution of these issues is provided below.

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

   In the seismic area, this subject was addressed in Section 3.7 of the submittal. The licensee stated that they had reviewed the safe shutdown equipment list (SSEL), which included plant systems used for decay heat removal (DHR), for seismic-related vulnerabilities under the Unresolved Safety Issue (USI) A-46 program, "Verification of Seismic Adequacy of Equipment in Operating Plants," and that all of the seismically related issues regarding DHR were resolved under that program. The licensee noted that, following a dam failure, the isolation condenser will be used as the means of DHR. According to the licensee's response to the staff's request for additional information (RAI), the licensee plans to include a seismic makeup path to the isolation condenser, and operator actions required for the proposed seismically qualified/verified makeup path to the isolation condenser will be submitted to the NRC when they are developed. For the fire area, the licensee discussed the means for providing long term DHR in Section 4.11.1 of the submittal. The licensee stated that they did not identify any fire-related vulnerabilities. The licensee stated that DHR will be available, with necessary manual actions, following a fire in any location at the plant. The staff finds that the licensee's evaluation of USI A-45 is consistent with the guidance provided in Section 6.3.3.1 of NUREG-1407 and, therefore, the staff considers this issue resolved contingent upon the licensee resolving the isolation condenser makeup seismic issues. Since the licensee plans to develop operator actions required for the proposed seismically qualified/verified makeup path to the isolation condenser, the need for any additional assessment or actions related to the follow-up actions of this issue for DNPS will be addressed by the NRR staff separately from the IPEEE program.

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

   Since DNPS is not a Westinghouse plant, this issue does not apply.

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

   The licensee addressed the new Probable Maximum Precipitation (PMP) criteria in Section 5.2.2 of its IPEEE submittal. In its submittal, the licensee stated that this issue was addressed previously for DNPS as SEP Topic II-3.B, and that as a result of the PMP issue, scuppers were installed in the roof parapets of the turbine and reactor buildings and in the crib house. The licensee stated that based on their evaluation, they concluded that the effects of site flooding and roof ponding at DNPS were acceptable (i.e., did not pose a significant risk to the plant). The staff finds that the licensee's GSI-103 evaluation is consistent with the guidance provided in Section 6.2.2.3 of NUREG-1407 and, therefore, the staff considers this issue resolved.

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

   The licensee has assessed the impact of inadvertent actuation of fire protection systems on safety systems which is one of the Fire Risk Scoping Study (FRSS) issues. The licensee's IPEEE submittal addresses this issue in Section 4.11.2. The licensee stated that this issue was previously evaluated during the plant's review of the IN 83-41 issues, *Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment*, and that several plant modifications had been made as a result of that review. The staff finds that the licensee's evaluation of GSI-57 is consistent with the guidance provided in the EPRI FIVE methodology which was accepted by the staff, and therefore the staff considers this issue resolved.

5. Fire Risk Scoping Study (FRSS) Issues

   The licensee has addressed the FRSS issues in Section 4.10 of the submittal following the FIVE guidance on these issues. These issues are: (1) seismic/fire interactions, (2) adequacy of fire barriers, (3) smoke control and manual fire fighting effectiveness, (4) equipment survival in a fire-induced environment, and (5) fire-induced alternate shutdown/control room panel interaction. The licensee stated in the submittal that no unacceptable risks or outliers were identified at DNPS due to the FRSS issues. The staff finds that the licensee's evaluation is consistent with the guidance provided in NUREG-1407 and, therefore, the staff considers these issues resolved.

<u>Other Generic Safety Issues</u>

In addition to those safety issues discussed above that were explicitly requested in Supplement 4 to GL 88-20, four GSIs 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. These GSIs were initially evaluated by the NRC's contractors, and the results of these evaluations are contained in the attached TERs. For those GSIs that were not completely resolved by the NRC's contractors, the NRC staff performed additional reviews. The final resolution of these issues is provided below.

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

   The licensee addressed this issue in Section 4.10.2.5 of the submittal and in the RAI response dated March 30, 2000, for question number 9 on control systems interactions. Regarding this issue, the licensee discussed plant design features and plant procedures stating that, as described in the Fire Protection Report (Appendix R Conformance/Safe Shutdown Report), safe shutdown circuits which are not independent of the control room are manually isolated in the event of a control room fire. Based on the results of the IPEEE submittal review, and the review of the licensee's RAI response, the staff considers that the licensee's process is capable of identifying potential vulnerabilities associated with this issue. On the basis that no vulnerability associated with this issue was identified in the IPEEE submittal, the staff considers this issue resolved.

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

   Section 4.10.2.3 of the licensee's IPEEE submittal contains information addressing this issue as a part of the discussion regarding the FRSS. The licensee discussed the reporting of fires, personnel and equipment requirements, and training. It is stated that the plant's fire brigade equipment includes portable smoke ejection equipment. Based on the results of the IPEEE submittal review, the staff considers that the licensee's process is capable of identifying potential vulnerabilities associated with this issue. On the basis that no vulnerability associated with this issue was identified in the IPEEE submittal, the staff considers this issue resolved for DNPS.

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

   The licensee's IPEEE submittal contains information to directly address the following external-event-related SEP issues: (1) settlement of foundations and buried equipment (not required since Dresden is situated on a rock site); (2) dam integrity and site flooding (Section 5.2.2); (3) seismic design of structures, systems, and components (Sections 3.1.3 and 3.4); (4) site hydrology and ability to withstand floods (Section 5.2.2); (5) industrial hazards (Section 5.2.3.5); (6) tornado missiles (Section 5.2.1.2); (7) severe weather effects on structures (Sections 5.2.1 and 5.2.2); (8) design codes, criteria and load combinations (Section 3.1); and (9) shutdown systems and electrical instrumentation and control features (Section 4.10.2.5).

   Based on the results of the IPEEE submittal review, the staff considers that the licensee's process is capable of identifying potential vulnerabilities associated with this issue. On the basis that no vulnerabilities associated with this issue were identified in the licensee's IPEEE submittal, the staff considers this issue resolved for DNPS. With respect to the external-event-related SEP issue on dam integrity and site flooding, since the licensee plans follow-up actions to identify a method of supplying make-up water to the shell of the isolation condenser through piping and components that are seismically qualified or verified, the need for any additional assessment or actions related to the follow-up actions of this issue for DNPS will be addressed by the NRR staff separately from the IPEEE program.

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

   The licensee's IPEEE submittal contains information addressing the following external-event-related MSRP issues: (1) effects of fire protection system actuation on non-safety related and safety-related equipment (Section 4.11.2); (2) seismically induced fire suppression system actuation (Section 3.4.7.2); (3) seismically induced fires (Section 3.7.4.2); (4) effects of hydrogen line rupture (Section 4.10); (5) the IPEEE-related aspects of common cause failures related to human errors (Sections 4.5.3 and 4.5.4); (6) non-safety-related control system/safety-related protection system dependencies (Section 4.10.2.5 and RAI response dated March 30, 2000); (7) effects of flooding and/or moisture intrusion on non-safety related and safety-related equipment (Section 4.11.2); (8) seismically induced spatial/functional interactions (addressed under USI A-46 program); (9) seismically induced flooding (Section 3.4.7.1); (10) seismically induced relay chatter (Section 3.6); and (11) evaluation of earthquake magnitudes greater than the safe shutdown earthquake (Section 3).

   Based on the results of the IPEEE submittal review, the staff considers that the licensee's process is capable of identifying potential vulnerabilities associated with this issue. On the basis that no vulnerabilities associated with this issue were identified in the licensee's IPEEE submittal, the staff considers this issue resolved for Dresden.

<u>Unique Plant Features, Potential Vulnerabilities, and Improvements</u>

There were no unique plant features noted in the licensee's IPEEE submittal. The licensee did not provide an explicit definition of a plant vulnerability, but stated that, based on the evaluation of DNPS, no vulnerabilities were identified in the seismic, fire, or HFO events areas. In the seismic area, the licensee stated that a number of improvements were implemented during the resolution of USI A-46, and that additional improvements were still under consideration. These improvements included modifications to anchorages of electrical buses and distribution panels. In a response dated March 30, 2000, to an NRC RAI regarding the licensee's seismic IPEEE evaluation, the licensee stated that the concept of providing a seismically qualified/verified make-up path to each plant unit's isolation condenser was being developed, and that the design changes required to implement this concept will be completed in conjunction with the approved schedule for resolution of the USI A-46 outliers. The operator actions required for the proposed seismically qualified/verified makeup path to the isolation condenser will be submitted to the NRC when they are developed. In addition, the licensee stated that a study will be performed to ensure that a small break LOCA, with no torus cooling but with the isolation condenser in operation, does not result in unacceptable torus temperature. Furthermore, the licensee indicated that the resolution of questions concerning the seismic capacity of the other IPEEE-related components including a group of relays associated with the isolation condensers is still planned as a part of the USI A-46 program. Regarding fires, in the RAI response dated March 30, 2000, the licensee stated that two hydrogen-related systems would be modified (seismically mounted) to reduce the risk associated with a seismic/fire event. No plant improvements were identified in the HFO events area that were a direct result of the IPEEE. However, two improvements that were related to HFO events were cited as resulting from the Systematic Evaluation Program (SEP) that was completed prior to the IPEEE program. These were the addition of scuppers to aid in draining water from roofs during heavy precipitation and revisions made to the site flood emergency plan.

III. CONCLUSION

On the basis of the above findings, the staff notes 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 DNPS design, operation, and history. Therefore, the staff concludes that (1) the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities from external events, and (2) the DNPS IPEEE has met the intent of Supplement 4 to Generic Letter 88-20 and the resolution of specific generic safety issues discussed in this evaluation report.

As indicated in Section II of this SER, the licensee plans to perform follow-up actions on two issues, namely, USI A-45 and the GSI-156 issue on "Dam Integrity and Site Flooding." Since no vulnerabilities associated with these issues were identified in the licensee's IPEEE submittal, the staff considers these issues resolved for the IPEEE review for DNPS; the need for any additional assessment or actions related to the follow-up actions of these two issues for DNPS will be addressed by the NRR staff separately from the IPEEE program.

It should be noted that the staff focused its review primarily on the licensee's ability to examine DNPS 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 quantitative estimates) that underlie or stem from the examination. Therefore, this evaluation report 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 staff evaluation report.

Attachment 1

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)

TECHNICAL EVALUATION REPORT

SEISMIC EVENTS

SUBMITTAL-ONLY SCREENING REVIEW OF THE DRESDEN NUCLEAR POWER STATION UNITS 2 AND 3 INDIVIDUAL PLANT EXAMINATION FOR EXTERNAL EVENTS

(Seismic Portion)

August 1998
(Updated, July 2000)
(Finalized October 2000)

Brookhaven National Laboratory

ACRONYMS

ADS	Automatic Depressurization System
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CCSW	Containment Cooling Service Water
CRD	Control Rod Drive
EPRI	Electric Power Research Institute
GIP	Generic Implementation Procedure
GL	Generic Letter
GSI	Generic Safety Issue
HCLPF	High Confidence of Low Probability of Failure
HPCI	High Pressure Coolant Injection
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
LOCA	Loss-of-Coolant Accident
LPCI	Low Pressure Coolant Injection
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
PGA	Peak Ground Acceleration
RAI	Request for Additional Information
RCS	Reactor Coolant System
SEP	Systematic Evaluation Program
SMA	Seismic Margins Assessment
SME	Seismic Margin Earthquake
SPLD	Success Path Logic Diagram
SQUG	Seismic Qualification Utilities Group
SRT	Seismic Review Team
SSE	Safe-Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
USI	Unresolved Safety Issue

1.0 INTRODUCTION

1.1 Purpose

In response to the the Nuclear Regulatory Commission (NRC) issued Supplement 4 to Generic Letter (GL) 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," ComEd performed an IPEEE for Dresden Nuclear Power Station, Units 2 and 3 and submitted the IPEEE results to the NRC (Reference 1). Brookhaven National Laboratory (BNL), as requested by the NRC, has performed a submittal-only screening review to verify the technical adequacy of the seismic portion of the IPEEE submittal. After an initial review, the NRC submitted a Request for Additional Information (RAI) to the licensee in a letter dated December 14, 1998. BNL's technical evaluation is based on the licensee's IPEEE submittal and the response to the RAI (ComEd letter dated March 30, 2000 [Reference 2]).

BNL's methodology utilized for the review followed the guidelines provided in the document titled, "Guidance for the Performance of Screening Reviews of Submittals in response to USNRC Generic Letter 88-20, Supplement 4," (Draft, October 24, 1996), and subsequent guidance for review of GSIs/USIs (August 21, 1997).

1.2 Background

Dresden Units 2 and 3 are similar generating units which include two boiling water reactor (BWR-3) nuclear steam supply systems (NSSSs) and turbine-generators furnished by General Electric Company (GE). The reactors are housed in Mark I containments.

Each NSSS is designed for a power output of 2,527 MWt, which is the license application rating. The net electrical output of each unit is 809 MWe. Units 2 and 3 were completed and went into commercial service in June 1970 and November 1971, respectively.

Dresden Station is located in northeastern Illinois, in the northeast quarter of the Morris Quadrangle (USGS designation), near the town of Morris in Grundy County (Goose Lake Township). The north and east boundaries are formed by the Illinois and Kankakee Rivers. The site is 953 acres, plus a 1,275-acre cooling lake. The lake was formed by construction of an impervious earth-fill dam and is connected to the intake and discharge flumes of Units 2 and 3 by two canals. Each canal is about 11,000 feet long.

Other than stating that the Dresden structures are founded on rock, the submittal does not describe the site geology/seismology. The original design basis earthquake is a "Housner-type" ground spectrum, anchored to a peak ground acceleration (PGA) of 0.20g for the horizontal safe-shutdown earthquake (SSE). The vertical SSE ground spectrum is 2/3 of the horizontal. The operating-basis earthquake (OBE) ground spectra are ½ times the SSE spectra. The design basis in-structure response spectra were developed from time history analysis, using the El Centro 1940 earthquake record (N-S component) scaled to a 0.10g PGA for the horizontal OBE. The resulting spectral values were then doubled to obtain the horizontal SSE spectra. The design basis vertical in- structure response spectra are equal to the vertical ground spectra; i.e., no vertical amplification.

For the IPEEE, Dresden is classified as a 0.3g focused-scope plant.

1.3 Licensee's IPEEE Process and Licensee's Insights

ComEd performed a seismic margins assessment (SMA) following the guidance of NUREG-1407 and Electric Power Research Institute (EPRI) Report NP-6041, Seismic Margins Assessment Methodology. Walkdowns were performed by a combination of ComEd plant engineers and outside contractors, all of whom were A-46 (Seismic Qualification Utilities Group [SQUG]) trained and certified, and a majority also had the EPRI Add-on Seismic IPE training course for SMA evaluations.

In this assessment, the seismic capacity is expressed in terms of the PGA of the seismic margin earthquake (SME). The assessed components included the structures, equipment, and distribution systems identified through the systems analysis. First, a component was evaluated based on the screening criteria presented in NP-6041 and summarized in Tables 2-3 and 2-4 of that reference. If a component met the requirements for the second earthquake level, it was assigned a capacity of 0.5g. If it could not meet the requirements for the second level, but could meet the requirements for the first level, it was assigned a capacity of 0.3g. If a component could not meet the requirements for the first level, a capacity was calculated.

The evaluation of major structures was based primarily on a review of the design bases, augmented by a walkdown to identify any anomalous conditions. Seismic capacities were explicitly calculated for masonry block walls, either by scaling existing IE Bulletin 80-11 calculations or by specific calculation. The evaluation of mechanical and electrical equipment relied heavily on the USI A-46 walkdowns. Equipment that met A-46 requirements (i.e., was not an outlier), was assigned an equipment seismic capacity of 0.3g or 0.5g, depending on the criteria in NP-6041. If the equipment was an A-46 outlier, a seismic capacity was calculated. If the equipment had anchorage that was not judged robust by the walkdown team, the A-46 anchorage evaluation was scaled to obtain an anchorage seismic capacity. During the walkdowns, the masonry block walls adjacent to the equipment were noted (if any). The final seismic capacity assigned to the equipment was the minimum of the equipment capacity, the anchorage capacity, and the capacity of any adjacent block walls.

IPEEE equipment which was not A-46 equipment was evaluated identically to A-46 equipment. Distribution systems included piping, electrical raceways, and ductwork. The seismic capacity of the raceways was based on the A-46 raceway evaluations. Piping and ductwork was evaluated based on a review of the design bases, augmented by walkdowns.

There were no significant concerns identified as a result of the seismic margins assessment. As a general observation, some electrical equipment anchorages have limited anchorage margin. This condition was also noted during the Systematic Evaluation Program (SEP) and led to Information Notice 80-21 that identified marginally anchored (or unanchored) equipment assemblies in older plants. Many of the equipment anchorages at Dresden were modified (improved) during the SEP. Since the IPEEE Seismic Margin effort considers a larger cross-section of systems, some other equipment not reviewed or considered in the SEP has been shown to have limited seismic anchorage capacity. That equipment has already been improved (anchorage upgrades completed) or is scheduled for design improvements.

In response to the RAI, the licensee submitted an update to the Executive Summary (Section 1.0) and the Seismic Event Assessment (Section 3.0) portions of its initial IPEEE submittal. This March 2000 update documents the results of the outlier resolution effort and the results of further evaluation of components initially identified to have high confidence of low probability of failure (HCLPF) capacities less than 0.2g PGA. Revised Section 1.4.1 lists the components found to have a seismic capacity less than 0.30g PGA; the lowest capacity is estimated at 0.20g PGA, which is the plant SSE design basis. Between 0.20g and 0.30g PGA, several notable components were listed:

- Condensate Storage Tank - 0.20g controlled by tank buckling
- Diesel Fuel Oil Storage Day Tank - 0.26g controlled by adjacent masonry wall
- Torus Suppression Chambers - 0.28g controlled by torus shell stress.

Approximately twenty electrical equipment anchorage capacities are also listed between 0.20g and 0.30g PGA. Four have capacities of 0.20g and the rest have capacities of at least 0.27g PGA.

Revised Section 3.8, titled "Resolution of Open Items," addresses those items previously identified as "Open Items Pending Resolution" in the licensee's initial IPEEE submittal. At the time of the initial submittal, walkdowns and assessments had not yet been performed. The licensee has determined that the open items have a capacity of 0.30g PGA or greater, with the exception of a group of relays that are part of the Isolation Condenser System. The licensee indicates that 65% of these relays still need to be walked down "as plant status permits".

The licensee summarizes its Seismic Margins Assessment in Section 8.1:

"As a results of this study, no programmatic issues were identified. No major weak links were identified among building, distribution systems which include piping and cable trays, or relays. The few issues that have been identified are associated with equipment anchorage. No single class of equipment emerged as being problematic with respect to base anchorage. Given the vintage of the Dresden plant, the fact that selected electrical equipment anchorage capacities govern the plant's seismic capacity is not unexpected. Since the anchorages identified in the aforementioned table are being improved, and based on experiences with actual industrial facilities in moderate to severe earthquakes, it is concluded that the Dresden plant possesses reasonable margin with respect to its design basis earthquake."

2.0 REVIEW FINDINGS

2.1 IPEEE Format and Methodology Documentation

The IPEEE format requested in NUREG-1407 was followed in the Dresden submittal. Documentation of the seismic methodology is somewhat sketchy; the USI A-46 program was heavily relied upon to address the IPEEE. The licensee's IPEEE submittal, as supplemented by the licensee's RAI responses, provided sufficient qualitative information to address the IPEEE issues such as: plant walkdowns, success path selection, structural and equipment seismic capacities, seismic-induced fire/flooding, and containment performance. Generic issues as specified in NUREG-1407 were also addressed in the submittal. Quantitative details of the licensee's seismic IPEEE evaluations are contained in separate reports and references to these reports were provided in the submittal.

2.2 Seismic Review Team (SRT) Selection

The IPEEE seismic review team was essentially the same as the A-46 team. A combination of in-house personnel and consultants was utilized. All are SQUG certified and most had EPRI IPEEE Add-On training. Five consultants from Stevenson and Associates and Dr. R.P. Kennedy participated in the A-46/IPEEE seismic walkdown. Four ComEd personnel also participated. Resumes and certification are included as an Appendix to the submittal.

The independent seismic peer review was conducted by Mr. Harry Johnson of Programmatic Solutions. Documentation of the peer review is supposed to be included in Appendix B to the submittal; however, these pages were not in the licensee's submittal.

The seismic review team and peer review appear to be in accordance with the guidelines of NUREG-1407.

2.3 Seismic Input

Section 3.3 of the submittal describes the seismic input used for the IPEEE. A 0.30g PGA NUREG/CR-0098 median rock spectrum was used as the SME. Reduction factors from p.4-6 of NP-6041 were applied. Based on the basemat dimension of 210', factors of 0.86 at 10Hz and 0.72 at 25 Hz were used.

SME in-structure response spectra were developed by Sargent and Lundy Engineers. A 3-dimensional horizontal model of the major structures was developed. The original design basis utilized two 2-D models. A vertical seismic model, including floor flexibility, was also developed. The original design basis assumed no vertical amplification of the ground motion.

Structural damping values consistent with the guidance in NP-6041 for structural stress above ½ yield stress were used in the re-analysis. A value of 7% damping was used for the structures, which is higher than the damping used in original design basis calculation. The drywell and reactor vessel were assigned 3% and 1% damping respectively.

SME in-structure response spectra were provided in response to the RAI. Design Basis response spectra were also provided. Review of the spectra indicates that they are reasonable.

2.4 Success Path Selection and Safe Shutdown Equipment List (SSEL)

The selection of the systems and the equipment required for system operations in an accident mitigation process is based on the EPRI methodology with enhancements as specified in NUREG-1407.

Plant-specific Success Path Logic Diagrams (SPLDs) are not presented in the submittal. The frontline systems that can be used for the safety functions required to establish and maintain a long-term safe shutdown condition (i.e., reactor reactivity control, reactor coolant system pressure control, reactor coolant system inventory control, and decay heat removal) are discussed in the submittal.

Consistent with the EPRI methodology, a preferred and an alternate success path are selected. The preferred path consists of the control rod drive (CRD) system for reactivity control, the relief valves[1] for reactor coolant system (RCS) pressure control, the high-pressure coolant injection (HPCI) system for RCS inventory control, and Division II of the low pressure coolant injection (LPCI) system in the torus cooling mode for decay heat removal (DHR). The alternate path consists of the CRD for reactivity control, the safety valves for RCS pressure control, the automatic depressurization system (ADS) and Division I of the LPCI system for RCS inventory control, and Division I of the LPCI system in the torus cooling mode for DHR. Both success paths can handle a small LOCA condition.

As discussed above, DHR is achieved in both the preferred and the alternate success paths by the use of LPCI system in the torus cooling mode. The Containment Cooling Service Water (CCSW) system, whose pumps take suction from Bay 13 of the cribhouse[2], provides cooling water to the LPCI heat exchangers for heat removal. This cooling mode may not be available upon a dam failure.

Following a dam failure, the level in the intake canal will drop to elevation 495'. Because the center line of the CCSW intake pipes is at elevation 498', stop logs must be placed where screens normally exist in the openings to the CCSW intake bay (i.e., Bay 13). The screen wash refuse pumps would then be used to reflood Bay 13 so that CCSW pumps could be started. The refuse pit pumps were included in the SSEL (called the success path equipment list, SPEL, in the submittal) and were evaluated as having a seismic capacity of 0.3g. However, according to the licensee's response to the RAI (Reference 2), the motor control centers and switchgear for these pumps were identified as outliers because of potential interactions with the cribhouse block wall, and because of the high cost to resolve these outliers, the refuse pit pumps, CCSW, and LPCI cooling mode will not be used for DHR for the specific case of a dam failure. For a dam failure, the isolation condenser for each unit will be used as the means for DHR.

The Isolation Condenser system is included in the SSEL. However, the components that are required to provide makeup water to the isolation condenser are not included in the SSEL[3]. The capacity of DHR for the isolation condenser is therefore limited. According to the response to the RAI, the development of a seismically qualified or verified makeup path to supply water from the ultimate heat sink to the shell side of the isolation condenser is being considered. And, also, according to the revision of the IPEEE submittal attached to the licensee's response to the RAI, "the Unit 2 Emergency Diesel Generator cooling water system is a seismically verified source of Isolation Condenser makeup for providing decay heat removal."

Although the use of the Isolation Condenser, with a verified makeup water supply source, provides a means of DHR for the intact reactor case, torus cooling may still be needed for the small loss-of-coolant accident (LOCA) case. According to the licensee's response to the RAI, "a study will be performed to ensure that a small break LOCA, with no torus cooling but with the isolation condenser in operation, does not result in unacceptable torus temperature." and "the design changes required to implement this concept will be completed in conjunction with the approved schedule for resolution of USI A-46 outliers."

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[1] Dresden has five relief valves (one Target Rock SRV and four electromatic relief valves) that discharge to the suppression pool and eight safety valves that discharge to the drywell. All are included in the SSEL

[2] Dreden Station has a dike surrounding the cooling lake and a dam on the Illinois River.

[3] Depending on plant configuration, makeup water can be provided by the Clean Demineralized Water system, the Service Water/Fire Protection system, or the Condensate Transfer system.

The systems selected for the success paths and their support systems are not discussed in the submittal. Dependency matrices are also not provided. It is simply stated in the submittal that, the SSEL "was generated to identify those components that are part of these systems as well as those components that are needed for support of the systems." The actual SSEL is provided in the submittal.

2.5 Plant Walkdown Approach

A combined A-46/IPEEE walkdown was conducted, in accordance with the requirements of the Generic Implementation Procedure (GIP). This is consistent with the walkdown procedures for an SMA in accordance with NP-6041. Anomalous conditions, anchorage inadequacies, spatial interactions, and A-46 outliers were identified. For structures included in the IPEEE scope, documented design details were verified during the walkdown. Potential interactions between masonry walls and safe shutdown equipment were identified. Seismic/fire interaction, seismic/internal flooding interactions and containment performance-related components were assessed by separate walkdowns.

The plant walkdown scope appears to be adequate, and was conducted by SQUG/IPEEE trained seismic capability engineers.

2.6 Structural Analysis and HCLPF Calculation

2.6.1 Structural Analysis and In-Structure Response Spectra

The development of the IPEEE RLE in-structure response spectra is discussed in Section 3.3.2 of the submittal. The IPEEE spectra were submitted in response to the RAI. Review of these spectra indicates that they represent a reasonable RLE demand for the evaluation of HCLPF capacities.

Sections 3.4.1, 3.4.2, and 3.4.3 of submittal provide the pertinent information for the analysis of structures. The structures evaluated include the containment (drywell), suppression chamber (torus), reactor building, turbine building complex, 310-foot main stack, and the cribhouse. No structure-to-structure interaction concerns were identified, based on the review of connectivity between the reactor and turbine buildings, and existing gaps. Based primarily on NP-6041, Table 2-3, all structures were screened at 0.3g PGA, except for the torus. A conservative HCLPF capacity of 0.28g was calculated. This was controlled by torus shell stress and was developed based on scaling SSE seismic stresses to obtain SME seismic stresses.

The 310-foot main stack at Dresden is identical to the Quad Cities stack, which has been seismically qualified to a 0.24g PGA for SSE. On this basis, Table 2-3 of NP-6041 was used to screen the stack at the 0.3g PGA SME level.

The turbine building complex is a reinforced concrete structure with a steel frame superstructure. It is designated a Class II structure. In the original design basis seismic analysis, however, it was coupled with the reactor building and evaluated as a Class I structure. Consequently, ComEd assigned it a 0.3g PGA SME capacity, based on NP-6041, Table 2-3.

The Cribhouse is a Class II structure, constructed of reinforced concrete below grade and masonry walls above grade,. The seismic review team determined that, except for the masonry walls, the cribhouse could be screened at 0.3g PGA SME. Potential seismic interactions resulting from masonry wall failure was designated as an open item/outlier to be resolved. These masonry walls were apparently not in I.E. Bulletin 80-11 scope.

However, subsequent systems analysis conducted by the licensee removed the Cribhouse from the IPEEE scope.

Also discussed under structures is the control room ceiling. Based on the details of the support system, ComEd assigned a 0.3g PGA SME capacity. However, beam clamps are used to attach the ceiling support grid to structural steel. The assigned capacity may be optimistic.

The last issue addressed in Section 3.4.2 of the submittal is potential failure of the dike surrounding the cooling lake and a dam on the Illinois River. Dresden 2 and 3 were included in the Systematic Evaluation Program (SEP). Under SEP Topic 11-4.E, it was concluded the failure of the dike or dam will not impact shutdown.

Masonry wall reevaluation is described in Section 3.4.3 of the submittal and the HCLPF capacities are tabulated in Table 3.1. Only those walls judged a potential interaction hazard were re-evaluated. In most cases, scaling of the I.E. Bulletin 80-11 results was employed to estimate HCLPF capacities. It is noted that the cribhouse masonry walls, whose interactions with the refuse pit pumps and the associated electrical equipments were important concerns for the DHR mode, were not evaluated at the time of the initial submittal. According to the licensee's response to the NRC seismic RAI, question No. (1a), subsequent to the submittal of the IPEEE report to NRC, the unit 2 diesel generator cooling water system was identified as a more viable and reliable success path for DHR in the event of the dam failure, and this path would eliminate the use of the refuse pit pumps and the associated electrical equipments. Therefore, the cribhouse masonry walls were removed from consideration in the IPEEE.

There is a major problem with the scaling process employed by ComEd for un-reinforced masonry walls. The licensee scaled the IE Bulletin 80-11 masonry wall evaluations with 2% damping, assuming 7% damping for the cracked un-reinforced block wall. This is not acceptable practice, because of brittleness of the un-reinforced block walls. However, since these walls were evaluated for the design basis under the IE Bulletin 80-11, they should have at least the design basis capacity, and there are other SSEL components having similar near design basis capacities. Therefore, no further review is recommended.

2.6.2 SSEL HCLPF Calculations

Sections 3.4.4 of the submittal addresses mechanical and electrical equipment. Table 3.2 is referenced for "SMA Seismic Capacity (PGA)" and "Outlier" identification. The SMA seismic capacity is determined as the lowest capacity of the equipment, its anchorage, or an interacting masonry wall. A confusing aspect of Table 3.2 is that most identified outliers have a minimum capacity of 0.3g or greater, while several items with capacity < 0.3g are not identified as outliers. No explanation is provided. The table of items with HCLPF capacities < 0.3g previously presented in Section 1.4.1 of the licensee's RAI response appears to contain more useful information about HCLPF. These items were previously discussed in Section 1.3 of this report. The licensee's RAI response documents the resolution of USI A-46 equipment outliers, including revised HCLPF capacities for the equipment anchorage. The revisions to the capacities are based on further detailed evaluation or plant improvements required for the A-46 program.

Section 3.4.5 and 3.4.6 addresses NSSS components and Distribution Systems, respectively. Per Table 2-4 of NP-6041, a minimum of 0.3g PGA capacity is assigned to all items except cable raceway systems. The A-46 cable raceway review identified a specific support configuration of potentially low capacity (0.15g PGA).

The licensee's RAI response indicates that, based on a detailed evaluation, the actual HCLPF capacity is > 0.30g PGA.

2.7 Soil Evaluation

No soil evaluation was conducted for Dresden; Supplement 5 to GL 88-20 removed the soil evaluation from the IPEEE for focused-scope plants.

2.8 Relay Chatter Evaluation

Section 3.6 of the submittal is entitled "IPEEE Relay Evaluation". ComEd conducted a joint A-46/IPEEE program. For relays common to both programs, the A-46 relay evaluation took precedence over the IPEEE evaluation. For IPEEE only relays, "bad actors" need to be identified and resolved, per NUREG-1407.

The licensee identified approximately 130 items of electrically controlled equipment exclusive to IPEEE. The IPEEE-only relay list was developed by examining the control circuits for the equipment. A significant volume of tabulated data is included in the submittal (as appendices) for relays. In the March 2000 update to Section 3.6, the licensee indicates that all relays have been evaluated except for those associated with the Isolation Condenser system. A relay walkdown is still needed, as the plant status permits. Also see Section 1.3 of this report.

2.9 Containment Performance

A brief discussion of containment performance is provided in Section 3.5 of the IPEEE submittal. Important issues raised in NUREG-1407 are addressed in the IPEEE.

According to the submittal, a specific walkdown for containment integrity was conducted for the Dresden station. The items investigated include the integrity of the containment itself, isolation systems such as valves, mechanical and electrical penetrations, bypass systems, and plant unique containment systems such as active seals. In the walkdown, no piping supports were observed to provide a "hard point" and all systems have sufficient flexibility to withstand differential displacement between the reactor building and the drywell containment. The personnel air locks and equipment access hatch do not require active isolation systems, and no credible seismic vulnerabilities were observed in the walkdown.

The walkdown did not identify any vulnerabilities associated with early containment failure due to a postulated seismic event.

2.10 Nonseismic Failures and Human Actions

On nonseismic failures and human actions, NUREG-1407 states that it is important that the failure modes and human actions are clearly identified and have low enough probabilities to not affect the seismic margins evaluation. Nonseismic failures and human actions are not discussed specifically in the submittal.

As discussed in Section 2.4 of this evaluation report, RCS inventory control for the preferred success path is provided by the HPCI system, which is a single train system with only moderate reliability. The use of the HPCI alone for inventory control in a success path, according to EPRI NP-6041-SL, is ,therefore, a nonseismic failure concern. Since HPCI is the only high pressure injection system for Dresden, the reliability of the preferred success path can not be improved by the inclusion of another high pressure system (e.g., RCIC) as suggested by EPRI NP-6041-SL. It is noted that the isolation condenser, although not originally selected in the SSEL for the success path consideration, is included in the SSEL due to other considerations. The availability of the isolation condenser eliminates some of the non-seismic concern of the HPCI system.

It should also be noted that the selected success path for some of the safety functions relies on a single train of a safety system. Different trains of the same system, instead of different systems, are used for different success paths. For example, while the preferred path relies on Division I of the LPCI system, the alternate success path relies on Division II of the same system. The core spray system, which is another low pressure injection system, is not included in the SSEL. Although this represents a lack of diversity and redundancy, it is not expected to cause a significant non-seismic failure concern because of the reliability of the LPCI system.

Human error probabilities are not specifically presented and discussed in the submittal. However, according to the submittal, the selection of the success paths is based on a review of the Dresden station operating procedures, and the Dresden Station Operations Department has chosen the systems to be utilized for the primary and backup safe shutdown paths. The success paths selected in the IPEEE are generally consistent with those the operators are likely to perform under accident conditions.

The only human action concern is that related to operator actions required to restore the water supply to the CCSW system following a dam failure. As discussed in Section 2.4 of this TER, the isolation condenser, in lieu of CCSW and LPCI, will be used as the means of DHR following a dam failure. According to the licensee's response to the RAI, "Operator actions required for the proposed seismically qualified/verified makeup path to the isolation condenser will be submitted to the NRC when they are developed."

2.11 Seismic-Induced Fires/Floods

Seismically-induced flooding is discussed in Section 3.4.7.1 of the submittal. The evaluation focused on first assembling an inventory of all potential internal flooding sources in areas containing SSEL equipment, then performing a walkdown to assess whether the sources are both significant hazards and seismically vulnerable.

Nonseismically designed piping, such as fire protection, noncritical main steam and noncritical service water piping, and large tanks with a capacity greater than 1000 gallons were compiled and evaluated by the walkdown. All areas of the Reactor, Turbine, and Cribhouse buildings were walked down.

Some concerns on seismically-induced floods were identified in the evaluation. Resolutions to these concerns are identified in Tables 3.3 and 3.4 of the submittal. In general, the evaluation as that described in the submittal seems to be adequate.

There is no discussion in the submittal on external flooding sources.

Seismic-induced fire issues are discussed in Section 3.4.7.2 of the submittal. Three issues were considered by the SRT during the seismic capability walkdown:

- Seismic-induced fires,
- Inadvertent actuation of fire suppression systems,

- Seismic degradation of fire suppression systems.

Seismic-induced fires were evaluated by first assembling a list of combustion sources, then performing a walkdown to assess whether the sources are both significant hazards and seismically vulnerable. Potential issues with respect to seismic-induced fire hazards identified by the SRT, such as the effect of the failure of the hydrogen seal oil panel and hydrogen monitors on the integrity of the hydrogen lines, are presented in Section 3.4.7.2 and resolutions to these issues are discussed in Section 4.9.3 of the submittal.

Seismic-actuation of fire suppression systems were examined by a walkdown and a relay functionality review to look for the potential of spray-down or release of fire suppression media due to seismic interaction. In addition, fire control equipment (panels and cabinets) were walked down to ensure they were properly anchored and not subject to potential seismic interactions. No problems were found. Detailed discussion of these issues is also provided in Chapter 4, Internal Fire Evaluation, of the submittal.

Seismic degradation of the fire suppression system was reviewed by walking down fire piping and looking for poor structural design features or potential interactions with a safe shutdown path component. No such potential interactions were noted except for one piping segment.

Resolutions to the potential problems identified in the evaluation are presented in Tables 3.3 and 3.4 of the submittal.

2.12 Unresolved Safety Issues (USIs) and Generic Safety Issues (GSIs)

Section 3.7 of the submittal addresses USI's and GSI's.

<u>USI A-45</u> Shutdown Decay Heat Removal Requirements

USI A-45 was subsumed in the USI A-46 program. All components are in the SSEL for A-46 and are dispositioned under that program.

<u>GSI-131</u> Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants

GSI-131 is not applicable to Dresden because it is not a Westinghouse plant.

<u>GSI-156</u> Systematic Evaluation Program

Since Dresden is a rock site, all soil related issues do not apply. The seismic design of structures, systems, and components were addressed in the submittal with respect to ground response spectra, and in-structure response spectra. The seismic input used in the IPEEE evaluations was based on a 0.3g NUREG/CR-0098 median rock spectrum. In addition, Dresden was included in the original SEP. Consequently, GSI-156 issues were addressed under SEP.

<u>GSI-172</u> Multiple System Response Program

GSI-172 issues were addressed to the IPEEE submittal as follows:

- Failures related to human error are not specifically addressed in the submittal. See Section 2.10 of this report.

- Seismically Induced Spatial and Functional Interactions are addressed, consistent with A-46 program requirements.

- Seismically-Induced Fires are addressed in Section 3.4.7.2 of the submittal. See Section 2.11 of this report.

- Seismically Induced Fire Suppression System Actuation is addressed in Section 3.4.7.2 of the submittal. See Section 2.11 of this report.

- Seismically Induced Internal Flooding is addressed in Section 3.4.7.1 of the submittal. External Flooding is not addressed. See Section 2.11 of this report.

- Seismically Induced Relay Chatter is addressed in Section 3.6 of the submittal. See Section 2.8 of this report.

2.13 Vulnerabilities/Plant Improvements

The term "vulnerability" is not used in the submittal. Items with estimated HCLPF capacities < 0.30g PGA are identified in Section 1.4.1 of the licensee's RAI response. Also, in this same section the licensee states "The above items meet or exceed the design basis requirement of 0.20g PGA, thereby meeting Dresden's intention to ensure that all IPEEE components have a seismic capacity that complies with design requirements [1.8]. Items 1 and 3 are the most limiting conditions and are controlled by anchorage capacity or tank buckling. However, based on experiences with actual industrial facilities in moderate to severe earthquakes, it is concluded that the Dresden plant possesses reasonable margin with respect to its design basis earthquake, and safe shutdown capability will not be lost."

The licensee does not plan to make improvements beyond those documented in the licensee's RAI response. The plant improvements implemented are those required for resolution of USI A-46. Consequently, it does not appear that any plant improvements can be attributed to the IPEEE. Resolution of a group of relays associated with the Isolation Condenser system is still pending.

One item worth noting is the licensee's commitment to search for a method to supply make-up water to the shell of the isolation condenser through piping and components that are seismically qualified or verified. The licensee will also perform, within this effort, a study to ensure that a small break LOCA, with no torus cooling but with isolation condenser in operation, does not result in unacceptable torus temperatures. According to the licensee's response to the RAI, the design change required to implement this concept will be completed in conjunction with the approved schedule for resolution of USI A-46 outliers. According to the A-46 SER (Reference 3), the licensee confirmed that "outliers will be resolved within two refueling outages per unit following receipt of NRC SER on USI A-46 submittal."

3.0 OVERALL EVALUATION AND CONCLUSIONS

The initial IPEEE submittal for Dresden Units 2 and 3 was incomplete. While the procedures described are consistent with NUREG-1407 and GL 88-20, a significant number of items related to the IPEEE only (i.e., not included in USI A-46) had not been evaluated. The substantive report information in the initial IPEEE submittal apparently resulted from the USI A-46 program.

In response to the RAI, the licensee submitted an update to its initial IPEEE submittal, which documents the resolution of outliers and open items, and revises the plant HCLPF capacity to 0.20g PGA. The licensee's evaluation of masonry walls for the IPEEE is questionable because of the use of 7% damping. However, since these walls were evaluated for the design basis under the IE Bulletin 80-11, they should have at least the design basis capacity, and there are other SSEL components having similar near design basis capacities. Therefore, no further review is recommended.

The IPEEE submittal for Dresden Units 2 and 3, as updated by the licensee's response to the RAI, appears to meet the objectives outlined in GL 88-20; the plant elements with the lowest seismic capacity apparently have been identified. It should be noted that while the IPEEE has identified a number of seismic issues for this plant, resolution of these issues has not taken place as of the writing of this report. Resolution must be verified in conjunction with resolution of the A-46 program issues. It should also be noted that, even with the issues resolved, the plant HCLPF is only at the level of the SSE.

4.0 REFERENCES

[1] Dresden Nuclear Power Station Units 2 and 3 Individual Plant Examination of External Events, Attachment to Letter dated December 30, 1997 from J. M. Heffley, Station Manager, Dresden Nuclear Power Station, Commonwealth Edison Company, to USNRC.

[2] Response to Request for Additional Information Regarding Report NSPLMMI-96001, Individual Plant Examination of External Events (IPEEE), Letter dated March 30, 2000, ComEd to NRC.

[3] Dresden - Plant-Specific Safety Evaluation for (USI) A-46 Program Implementation (TAC Nos. M69442 and M69443), February 23, 2000.

Attachment 2

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)

TECHNICAL EVALUATION REPORT

FIRES

Review of the Submittal in Response to U.S. NRC Generic Letter 88-20, Supplement 4: "Individual Plant Examination-External Events"

Fire Submittal Screening Review Technical Evaluation Report: Dresden Revision 5: May 7, 2001

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1.0 INTRODUCTION

This Technical Evaluation Report (TER) presents the results of the Step 0 review of the fire assessment contained in Attachment 1 of [3], " Revision to the Individual Plant Examination of External Events for Dresden Nuclear Power Station Units 2 and 3". This fire assessment superceded the original assessment [1] provided by the licensee. This TER also includes the review of the responses to requests for additional information (RAI) [3] issued to the licensee [2] based on questions raised during the review of original submittal [1] and the responses to supplemental RAIs (SRAIs) [12] issued to the licensee following the review of the responses to these RAIs [11]. The RAIs, the licensee responses, and the assessment of the responses are documented in Appendix A of this TER. The SRAIs, the licensee responses, and the assessment of the responses are documented in Appendix B of this TER.

1.1 Plant Description

The Dresden Nuclear Power Station consists of two BWR/3s - Unit 2 and Unit 3. Each produces 2527 MW_t from a two recirculation loop BWR nuclear steam supply system (NSSS) and turbine generator supplied by General Electric Company. Each unit has a Mark 1 primary containment consisting of a drywell and pressure suppression chamber. They have isolation condensers and are the only such plants with high pressure coolant injection (HPCI). Both units use the same reactor building which provides secondary containment. The building has a single operating floor with no separation barriers above that level. Beneath the operating floor the reactor building has a common wall which separates the operating and equipment areas of the two units. Shared buildings/systems include the Turbine Building, Service Water System, intertied Reactor Building Closed Cooling Water System, Radioactive Waste Systems, and Process Computer. The diesel generators (DGs) and HPCI Systems also share the same building. The control rooms for the two units are adjacent and open to each other. The Service Water System supplies water to safety-related equipment, in particular it cools the Reactor Building Closed Cooling Water (RBCCW) System, the Turbine Building Closed Cooling Water (TBCCW) System, and other heat loads. The Containment Cooling Service Water System provides cooling for the containment cooling heat exchangers. The ultimate heat sink for this system is a lake which is connected to the intake and discharge flumes of Units 2 and 3 by two canals, one intake and one discharge.

Engineered safety features for each unit include an Emergency Core Cooling System (ECCS) and the primary containment. The ECCS consists of four subsystems: HPCI system, Core Spray (CS) system, Low Pressure Coolant Injection (LPCI) system, and Automatic Depressurization System (ADS). The HPCI system contains one steam driven main pump and a booster pump. The CS system consists of two motor driven pumps. The LPCI system has four motor driven pumps. The ADS has five relief valves. Decay heat removal is provided by a separate three loop Shutdown Cooling System which is cooled by the RBCCW. A total of three EDGs are provided for the two units to supply emergency power to the ESF loads. One EDG, which can supply the ECCS power requirements or the power for safe shutdown, is dedicated to each unit. The third diesel generator normally supplies a division of ECCS for Unit 2, but can be used to supply one of the ECCS buses in Unit 3. The DGs are cooled by a DG service water system.

1.2 Review Objectives

The performance of an IPEEE was requested of all commercial U.S. nuclear power plants by the U.S. Nuclear Regulatory Commission (USNRC) in Supplement 4 of Generic Letter 88-20 [4]. Additional guidance on the intent and scope of the IPEEE process was provided in NUREG-1407 [5]. The objective of this Step 0 screening review is to help the USNRC determine if the Dresden submittal has met the intent of the generic letter and to also determine the extent to which the fire assessment addresses certain other specific issues and ongoing programs.

1.3 Scope and Limitations

The Step 0 review documented in this TER was limited to the material presented in the Dresden IPEEE revised submittal [3 (Attachment 1)], the responses to RAIs [3], and the responses to SRAIs [12]. The RAIs were submitted to the licensee based on an initial review of the original submittal [1] and SRAIs were issued based on the review of the revised submittal. The review of the revised submittal was limited to verifying that the critical elements of an acceptable fire analysis have been presented. An in-depth evaluation of the various inputs, assumptions, and calculations was not performed. The review was performed according to the guidance presented in Reference 6. The results of the review are presented in Section 2.0 and Appendices A and B. Conclusions and recommendations as to the adequacy of the Dresden IPEEE revised submittal with regard to the fire assessment and its use in supporting the resolution of other issues are presented in Section 3.0.

2.0 FIRE ASSESSMENT EVALUATION

The following subsections provide the results of the review of the revised Dresden fire assessment. The review compares the fire assessment against the requirements for performing the IPEEE and their use in addressing other issues. Both areas of weakness and strengths of the fire assessment are highlighted.

2.1 Compliance with USNRC IPEEE Guidelines

The USNRC guidelines for performance of the IPEEE fire analysis derive from two major documents. The first is NUREG-1407 [5], and the second is Supplement 4 to USNRC Generic Letter 88-20 [4]. In the current screening assessment, the adequacy of the utility treatment in comparison to these guidelines has been made as outlined in "Guidance for the Performance of Screening Review of Submittals in Response to U. S. NRC Generic Letter 88-20, Supplement 4: 'Individual Plant Examinations - External Events,'" Revision 3, March 21, 1997 [6]. The following sections discuss the revised IPEEE submittal in the context of the specific review objectives set forth in this Screening Review Guidance Document and assess the extent to which the utility document in conjunction with the RAI and SRAI responses has achieved the stated objectives.

2.1.1 Methodology Documentation

The revised Dresden fire assessment of Units 2 and 3 [3 (Attachment 1)] was performed using both the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology [7] and the EPRI Fire PRA Implementation Guide (FPRAIG) [8]. The FIVE methodology was partially used to perform the qualitative screening, determination of ignition source frequencies (the FPRAIG was also used), fire compartment boundary requirements, and plant walkdowns. The revised submittal stated that the FPRAIG was used "... to provide enhancements for the development of individual fire-induced scenarios and the multi-compartment analysis."

2.1.2 Plant Walkdown

Several sources of information were used to provide inputs to the plant walkdowns that were conducted. Documents utilized included:

- Dresden Fire Hazards Analysis (FHA). This was used to obtain plant layout for defining the fire areas and compartments, barrier information, and detection and suppression data for fire modeling.

- Appendix R Safe Shutdown Analysis (SSA). This was used to determine systems and components used for Appendix R shutdown, location and function of the Appendix R safe shutdown cables and circuits, and post-fire manual actions.

- Sargent & Lundy Interactive Cable Engineering (SLICE) Cable database. This was used in conjunction with plant electrical drawings to determine locations of safe shutdown equipment cables and related offsite power cables.

- Transient Combustible Control and Housekeeping Procedures. These were used in the process of selecting transient fire scenarios.

According to the submittal, walkdowns were conducted by the IPEEE team throughout the project to obtain and/or confirm data. The composition of the walkdown teams varied depending on the information to be collected or confirmed. The personnel who participated in the various walkdowns and their organizations were not identified. Detailed preparation preceded each walkdown followed by documentation of the data obtained. The following walkdowns were performed for the purposes indicated:

- Walkdowns to identify fire ignition sources.

- Control Room walkdown to determine (a) ignition source loading and separation, (b) damage due to a postulated fire event, (c) location of detectors, and (d) ventilation and smoke removal capabilities.

- Walkdowns of all unscreened compartments to determine the locations of fixed ignition sources with respect to potential targets, the locations of detection and suppression systems with respect to the source and targets, and placements of combustibles near fire barriers. This information was used in the preliminary (quantitative) screening of fire compartments and for detailed fire modeling.

- Additional walkdowns of unscreened compartments to evaluate transient fires.

- Walkdowns to support the multi-compartment analysis.

The walkdowns conducted appear to be adequate for their intended purposes.

2.1.3 Fire Area Screening

The revised analysis was based on the division of the plant into the 19 fire areas defined in the Appendix R Program. These areas were then evaluated using the FIVE screening criteria, i.e., area must have rated boundaries and any postulated fire not result in a safety challenge. Using these criteria, the Unit 2 and Unit 3 primary containments were screened, partially on the basis that they are normally inerted with nitrogen when the units are at power.

The remaining 17 fire areas were subdivided into 83 fire compartments using the Appendix R fire zone definitions. A table was provided in the revised submittal that listed the fire compartment designation, its associated fire area, and the fire compartment title. The Appendix R fire zone definitions were used because the existing data structure for cable location information was based on these definitions. The second level qualitative screening required that a fire compartment meet the FIVE boundary criteria and contain no Appendix R safe shutdown equipment or cables. Using these criteria, the Auxiliary Boiler House, Hydrogen Storage Facility, and Unit 1 were screened.

The second level of qualitative screening resulted in a total of five compartments being screened. The remaining 78 fire compartments were then subjected to quantitative screening. This process consisted of applying a single bounding scenario for each fire compartment. (Unlike the original submittal, no severity factors were used and automatic fire suppression was not credited.) Fire ignition frequencies were developed based on information obtained from walkdowns of the fire compartments as described in Section 2.1.4 of this report. CCDPs were developed using information about the equipment and cables in each compartment, assuming that all of these items were damage by the fire. The compartment screening CDF was then calculated by multiplying the compartment fire ignition frequency by the CCDP. Fire compartments with a CDF less than 1E-07/yr were screened.

The single compartment quantitative screening resulted in the screening of 62 of the 78 remaining fire compartments for Unit 2. For Unit 3, 57 of the 78 fire compartments were screened. A table was provided in the submittal which showed, for both units, the compartments screened and the screening CDF. The compartments which did not screen for either unit were also included in the table. The remaining areas included areas that might be expected to survive screening, including the Control Room and switchgear areas. No further analysis was performed for screened fire compartments other than those considered in the multi-compartment analysis (MCA) which evaluated the potential for fires to spread to or damage equipment in an adjacent compartment. The Dresden MCA process is discussed further below.

The screening analysis appears to be satisfactory and was an improvement over the screening that was conducted in the original fire assessment [1].

2.1.4 Fire Occurrence Frequency

Fire ignition sources were identified and fire compartment ignition frequencies developed based on guidance provided in FIVE and the FPRAIG. The process consisted of reviewing the FHA to identify likely ignition sources in each fire compartment. If accessible, a walkdown of the fire compartment was completed. However, some compartments were inaccessible to the walkdown team during plant operation. Ignition sources in these compartments were identified by:

- Using the information available from a mirror-image, accessible compartment in the other unit.
- Viewing videotapes, photographs, and/or plant drawings to identify ignition sources in compartments inaccessible due to radiological hazards.
- Interviewing plant personnel familiar with the area.

After all of the accessible fire compartments were walked down, a compartment fire ignition frequency was calculated using the FIVE methodology guidance. Both fixed and transient ignition sources were considered in developing fire frequencies. The submittal states that location weighting factors were calculated with the guidance provided in FIVE, but some apparent discrepancies (discussed below) were identified. Ignition source weighting factors (W_S) were, with some modifications, also calculated in accordance with FIVE. Transient fires due to maintenance activities and credible sources were considered.

The location weighting factors used for the analysis are provided in the submittal. With two exceptions they appear to be consistent with the guidance in FIVE. For the turbine building a location weighting factor (W_L) of 1 is used. Since there is only one Turbine Building for both units it appears that a value of 2 (units per site/number of buildings) should have been used. For switchyards a value of $W_L = 0.67$ was used. Per FIVE, W_L for a switchyard (transformer yard) is determined from: units per site/number of switchyards. The submittal identifies seven switchyard zones, including the Unit 1 switchyard, but does not explain how the value of 0.67 for W_L was determined based on two (or three) units and the seven identified zones.

The W_S values used for the switchyard areas were also not specified. To account for the unique distribution of high voltage switchgear at Dresden, W_S was calculated using a method that differs from that used in FIVE. W_S was calculated by dividing the number of switchgear cubicles in a switchgear area by the total number of switchgear cubicles in all of the switchgear areas in both units. Similarly, the FIVE methods for calculating W_L and W_S for battery rooms and intake structures were modified to account for the plant specific configurations at Dresden. The modified methods used appear to be reasonable.

Using the above approach, fire ignition frequencies were developed for the 78 compartments that remained after qualitative screening. All reported ComEd fires were reviewed and treated according to the EPRI Fire Events Database [9] to identify any data which would affect the calculated fire compartment frequencies. Two ignition sources were identified in the ComEd data (refuel hoists and isophase bus ducts) that do not appear in the EPRI database. Location-specific fire frequency terms were added to applicable locations to account for these plant-specific events.

Cabinet fire frequencies in the control room were determined by partitioning the total control room ignition frequency over panels and cabinets. A walkdown was used to determine a weighting factor based generally on panel or cabinet length. These resulted in "space units" being assigned to each panel/cabinet. The total number of space units (78) identified was used to determine an ignition frequency of 2.47E-04/yr for each cabinet or panel. Since the space units for each panel/cabinet were provided, this value enabled the associated panel/cabinet ignition frequency to be calculated. This value was used in the analysis of the fire scenario(s) associated with the specific panel/cabinet.

Cabinet fire frequencies in the Unit 2/3 Auxiliary Electric Equipment Room (AEER) were determined using a method similar to that used for the control room. The cabinets were divided into sections of approximately the same size. This resulted in the 99 cabinets being evaluated in terms of 147 sections, each of which was assigned an ignition frequency of 1.36E-4/yr, i.e., the total compartment frequency of 2.0E-2/yr divided by 147.

The methodology for determining the fire frequencies appears, in general, to be adequate. However, with the exception of the control room and AEER, no frequencies were presented in the revised submittal.

2.1.5 Fire Propagation and Suppression Analysis

According to the revised submittal, fire modeling based on the FIVE methodology [7] and FPRAIG [8] was used to analyze unscreened fire compartments and identify credible fire events. The modeling effort was based on extensive plant walkdowns, and review of controlled drawings and related fire protection documents. Ignition sources were examined to determine if they were capable of propagating a fire. Fire scenario geometries reflecting the locations of the ignition sources, PRA targets, and intervening combustibles were developed. Field data gathered also included identification of fire protection/mitigation features such as suppression, detection, and other features which protect PRA targets.

Fire exposure temperatures at the PRA targets were calculated based on the fire modeling correlations provided in FIVE, supplemented by fire modeling data presented in the FPRAIG. The following key inputs/assumptions were used in the FIVE fire modeling worksheets:

- A cable damage and ignition threshold of 425°F, consistent with non-IEEE 383 qualified cable. The cable ignition threshold used in the original fire assessment [1] was a subject of an RAI (see Section A.3 in Appendix A of this report).
- A heat loss factor (HLF) of 0.7 except in the MCA where a value of 0.85 was used. The HLFs used in the original analysis were the subject of RAI 5 which is summarized in Section A.5 of Appendix A of this report. (The section of the revised analysis which describes the MCA is referenced, but the section contains no discussion of the HLF used.)
- Critical radiant heat fluxes of 0.5 Btu/s/ft^2 and 1.0 Btu/s/ft^2 for non-qualified and all other components, respectively.
- A target thermal response parameter of 16 (PE/PVC cable) per FIVE.
- A damage temperature of 150°F for solid state electronics.
- Detector/suppression system actuation temperatures based on specific system installation.
- Detection system time constant based on specific system installation.

The burning characteristics of equipment and materials involved in the fire scenarios were based on heat release rate and heat content information provided in FIVE and the FPRAIG. Eighteen percent of the postulated oil fires were treated as "large". The remaining 82% were treated as "small". (Large and small refer to the size and type of oil spill considered for the specific component associated with the postulated fire, as discussed in the revised submittal.) The licensee noted that the FPRAIG recommends that only two percent of oil spills should be treated as large and states that the percentages used were consistent with the EPRI Fire Events Database [9].

Electrical fires were evaluated using methods consistent with FIVE and the FPRAIG. Five types of electrical fires were evaluated. The types considered were low voltage cabinets and panels, motor control centers (MCCs), low voltage buses, medium voltage switchgears, and transformers. Low voltage cabinets and panels are those containing circuits that operate at less than 600 V. Fire propagation is not considered to be credible if the panel is "substantially sealed", but this term was not defined in the submittal. If propagation is considered credible, a 400 Btu/s heat release rate and a 15 minute fire duration were used in the analysis.

Fires in MCCs were assumed to result in functional failure. Based on the MCC construction, propagation of a postulated fire outside the MCC is not considered likely. However, to address potential uncertainty, 10% of the postulated MCC fires were assumed to propagate vertically outside the MCC and potentially damage circuits above the MCC that are associated with safe shutdown functions. Similar damage and propagation assumptions were made for the low voltage (480 V AC & DC) buses. The treatment of the medium voltage (4.16 kV) switchgear was similar to the low voltage buses and MCCs except that it was assumed 20% of the postulated fires would propagate beyond the boundaries of the switchgear. The transformer classification addressed 4 kV, 480 VAC, and off-site transformers. Since a non-combustible material (Pyranol) is used to fill these transformers, the analysis did not consider transformers fires that could occur if the fluid was oil-based.

The majority of the cable used in the Dresden plant is not IEEE 383 qualified. The risk associated with this cable was examined by evaluating cable tray fires in the Unit 3 cable tunnel and those raceways containing circuits associated with the ADS. The scenarios involving self-initiated cable fires resulting in failure in all circuits in the raceway were analyzed and discussed in the revised submittal.

Transient ignition sources were not explicitly treated in the fire growth and damage scenario analyses. That is, only fires resulting from fixed ignition sources were actually modeled. This approach was based on several qualitative arguments. For example, transient ignition sources associated with hot work are controlled by plant procedures which include a requirement for a fire watch. Therefore, the likelihood that a significant fire could result from such a source is considered to be low. Other potential transient ignition sources such as those considered in the development of the compartment ignition frequencies were also screened. This was based on the specific modeling analyses and walkdowns that were performed which considered severe and non-severe fire events involving in-situ sources. The licensee argues that these events bound the impact of the screened transient ignition sources. However, a concern was raised that this treatment of transient fire sources could have under-estimated the CDF from fires at locations without fixed sources. Thus, the licensee was issued an SRAI requesting a reassessment of the CDF from transient fire sources in 6 fire areas (see Section B.1 of Appendix B of this report for a summary of the licensee's response).

The modeling generally assumed that non-transient fires would reach their peak heat release rate immediately upon ignition. This minimized the time available for suppression and therefore produced conservative results and is consistent with the FIVE methodology. Propagation of cable fires was based on the following assumptions:

- If vertical cable runs were ignited, it was assumed that the fire propagated up to the room boundary or until the cable direction changed to horizontal. (Subsequent propagation was not described.)
- Tray-to-tray propagation of fires in horizontal tray stacks (ladderback trays) was assumed to progress as described in Appendix I of the FPRAIG.
- Barriers are credited with limiting damage to PRA targets per Appendix J of the FPRAIG. Based on test data, it states that barriers did not delay cable damage for non-qualified cable. For qualified cable, the trays do not ignite until after the fire brigade reaches the scene which is assumed to be 20 minutes.

The above reference to Appendix I of the FPRAIG is not consistent with the licensee response to RAI #4. It states that "The upgraded fire analysis has no scenarios in which experimental data were used to estimate the rate and extent of fire propagation." The response to RAI #4 discussed in Section A.4 of Appendix A to this report is considered to be the definitive explanation of how cable tray fire propagation was treated. It is assumed that the description in the revised submittal reflects information in the original submittal that was not properly updated.

The potential for a given fire to generate a HGL within the compartment was not postulated for those spaces that have substantial ventilation. Only those openings that provide a vertical vent path were considered. Available ventilation pathways as well as fire-induced boundary failures were evaluated in the MCA discussed in Section 2.1.6 of this report. To ensure that the potential for localized heating outside the fire plume and ceiling jet region was considered for substantially ventilated compartments, a minimum margin of 50°F between the calculated target temperature (apparently based on plume temperature) and the damage threshold was used instead of the FIVE HGL analysis. A 50°F temperature margin may be optimistic for particular rooms because the fuel loading was apparently not considered. No results were provided in the submittal which showed that the 50°F temperature margin was valid for the compartments to which it was applied.

The licensee related the fire modeling discussed above to the plant response by examining the plant fire PRA model to determine the equipment and functions credited for post fire plant trip response. A review of plant drawings and cable databases, including Appendix R data, was performed by the licensee to identify those cable and circuit failures due to a fire which would result in the equipment failure(s) in the PRA model and their location in the plant. This information was then linked to the fire compartment designators and the fire PRA model. The methodology and data sources used to develop the cable/circuit and equipment location database required for the fire risk analysis are described in detail in the revised IPEEE submittal. The inventory control, decay heat removal, and support systems were considered in this process. Selected non-Appendix R systems, as well as Appendix R systems, were considered.

The revised fire analysis considered three possible fire-induced cable failure modes- open circuit, short circuit, and hot shorts. Spurious equipment actuation was considered for all three failure modes. The potential effects of open circuit failures and short circuits were described in the submittal. Both of these failures were conservatively assigned a conditional failure probability of 1.0. The description of the treatment of hot shorts implied that a failure probability of 1.0 was also used in the analysis for such events. However, the fact that cable-to-cable hot shorts were not analyzed is considered to be a weakness. The details of the analyses of these fire-induced cable failure modes were not provided.

A listing of the automatic fire detection and suppression systems (none, partial, or full) in all 83 compartments considered in the revised analysis was provided in the submittal. The suppression system response parameters and the automatic suppression system reliabilities provided in the FIVE methodology were assumed to be applicable. Automatic suppression was considered if it was possible for the system to actuate and extinguish a fire before damage would occur to a PRA target. Sprinkler and spray systems were also credited for cooling hot gases and mitigating HGL damage.

The original submittal did not indicate if the suppression systems are designed and maintained in accordance with NFPA standards. As a result, the use of the suppression system failure probabilities provided in FIVE may not have been appropriate. The resulting RAI #1 and licensee response is discussed in Section A.1 of Appendix A to this report.

The treatment of manual suppression in the original study was a subject of an RAI (#7). The response to this RAI is in Section A.7 of Appendix A of this report. In the revised assessment, manual fire suppression was generally not credited (two exceptions were noted) as being effective in preventing damage to critical targets. Credit was not taken due to the time delay between detection of a fire and the time required for the fire brigade to respond. Manual suppression was credited in the main control room because it is continuously manned. (The revised submittal states that the control room has full detection coverage.) Manual suppression was also credited during hot work activities because of the presence of a fire watch. The potential for manual suppression to cause collateral damage to nearby equipment was not discussed.

In the original submittal, manual recovery of failed automatic suppression systems was also credited, as described in the FPRAIG. This was the subject of RAI #6 which was submitted to the licensee. The RAI and licensee response are discussed in Section A.6 of Appendix A to this report.

For Control Room (CR) fires, two situations were considered. First, it was assumed that evacuation will be required if smoke obscures the control panels. In this case, the plant remote shutdown capability would be used for safe shutdown. For less severe fires, it was assumed that the fire would potentially damage the contents of one or more cabinets but suppression would be successful before control room abandonment was required. Functions still available in the control room or the plant remote shutdown capability would be used for safe shutdown. For each of these cases, the plant system functions associated with each of the control cabinets that would be impacted by the fires requiring abandonment and those successfully suppressed were identified.

Key assumptions in the control room analyses, which was based on the guidelines in Appendix M of the FPRAIG, included:

- The control room abandonment time was assumed to be similar to representative Sandia National Laboratories cabinet fire tests as discussed in the FPRAIG, i.e., 15 minutes per the FPRAIG interpretation.
- Each cabinet was assumed to contain sufficient material that a fire could generate enough smoke to require evacuation if suppression is not successful.
- Postulated fires can be screened as non-risk significant if the fire is suppressed before abandonment is required and the functional loss of the affected cabinets does not impact systems credited in the PRA or does not cause a plant trip.
- Fire propagation to an adjacent cabinet is prevented if suppressed within the time frame associated with control room abandonment and there is a double wall and intervening air gap between adjacent cabinets/panels.
- Fire damage to the reactor protection system (RPS) circuits results in a scram.

The basis for the above assumptions was provided in the revised submittal. The third assumption above is considered to be a weakness since any significant fire in the control room would very likely lead to a trip. The last assumption concerning RPS circuits neglects the possibility of multiple hot shorts and is also considered to be a weakness.

The submittal states that the control room ventilation system ducts are provided with smoke detectors which, when activated, switch to a smoke purge operating mode. The designed airflow pattern is such that air is exhausted from within the main control building area so the impact on of fires in this area on control room habitability is minimized.

For a bounding control room fire, a probability of non-suppression of 3.4E-3 was applied per the FPRAIG. The analysis also incorporates a panel severity factor of 0.20 and a CCDP of 0.50 for shutdown from outside the control room. The applicability of the severity factor and CCDP to the Dresden control room configuration is discussed in Section A.12 of Appendix A to this report.

Based on the answers to the questions provided in Section 2.2.4 of this report, no mention was made of potential damage that might be caused by the fire brigade or the time required for some specific actions, e.g., time required to assess the fire.

2.1.6 Fire-induced Initiating Events and Fire Scenarios

After the qualitative and quantitative screenings, 16 Unit 2 and 21 Unit 3 compartments remained to be analyzed in detail. For the fire scenarios in each of these compartments, including the control room, a CCDP was determined using the Dresden PRA model.

The Dresden fire PRA model was developed from the plant internal events PRA model. The development began with the selection of the potential fire initiated events. It was determined that the internal events PRA model structure was adequate to address all of these events except for an event involving spurious actuation of multiple ADS valves. Thus, a new initiating event was created for this event. The other initiating events included in the fire PRA model were turbine trip, loss of main condenser, loss of instrument air, spurious actuation of a single ADS valve, single unit LOSP, and dual unit LOSP. These events were based on an assessment of fire-induced failures which were based, in turn, on a determination of the affected circuits and equipment. The specific effects of each of the initiators was discussed in some detail in the revised submittal.

The front-line and support systems modeled in the fire PRA were described in the revised submittal. The performance of these systems was modeled by fault trees so that hardware failures, human errors, test and maintenance unavailabilities, and other events were included in the analysis. The system functions and associated equipment treated in the fire PRA model were linked to the spatial location information developed as described in Section 2.1.5 of this report. In particular, the cables whose damage would impact the functionality of a given piece of equipment was related in a fire risk analysis code (FRANC) data file that also identified the cable routing. This, in conjunction with the fire modeling that was done, allowed the analysis to identify the fire PRA model functions that would be adversely affected by a postulated fire and quantify the results. If fire modeling was not performed for a particular fire compartment, all basic events (functions) associated with that compartment were set to 'true' (failed) in the PRA model quantification.

The fire PRA model incorporated all of the operator actions included in the plant internal events PRA model. These actions were reviewed to determine those which occur in the control room and those which occur outside. The revised analysis took credit for Emergency Operating Procedures (EOPs) which were not fully credited in the original IPEEE analysis. All but one of the postulated fire scenarios in the revised analyses were quantified based only on the provisions of the EOPs. The original analysis was based primarily on the Appendix R post-fire safe shutdown procedures. In the revised analysis, the Appendix R post-fire safe shutdown procedures were credited only for the bounding control room fire event.

All of the operator actions in the fire PRA model were reviewed to determine the time line associated with each action. Based on these times, three groups of actions were defined for actions which occur in the control room and three groups for those actions which occur outside. Operator actions which are performed in the control room were not considered to be adversely affected by postulated fires outside the control room. Any operator action with a required response time of 30 minutes or less (the first group) was assumed to fail given a control room fire. Nine such actions were identified in the PRA model and identified in the revised submittal. No details concerning the timing of human responses to particular fires, specifically evacuation of the MCR and manning the remote shutdown panels, were provided.

For human actions taken outside the control room, it was recognized that smoke and heat from a fire may affect the success of the operator action. To assess these effects, a bounding quantification was performed with the human actions assigned an HEP of 1.0. This resulted in two operator actions being identified as risk significant. For these two actions, the internal events HEP was considered to be applicable if at least 30 minutes was available to complete the action and the fire would not affect the completion of the action. Fires in compartments that did not meet these conditions were analyzed using a HEP of 1.0 for these two operator errors. For all other human actions outside the control room, a conservative HEP of 1.0 was used in the analysis.

Single compartment fires were evaluated for both units. Each of the compartments analyzed was described in the revised submittal. Some compartments are common to both units so separate analysis were made to determine the effects on each unit. The description included the compartment CDF contribution, the general location of the compartment, a qualitative description of the compartment ignition sources and combustibles, and the fire scenarios/ignition sources that were considered in the analysis. In some cases it was concluded that the CDF calculated in the original analysis was so close to the screening criteria (1.0E-7/yr) that additional analysis was not necessary. For the other compartments the configuration, equipment, and cables in the compartment were reviewed to determine the fire scenarios that should be analyzed.

Due to the number of cabinets (99) in the Unit 2/3 Auxiliary Electric Equipment Room (AEER), the revised analysis identified each cabinet, determined the plant system functions associated with the cabinet based on the cables it contained, and assessed the likelihood for fire propagation between cabinets and to overhead raceways based on the cabinet structure. If a cabinet was "not considered to be sealed" or had ventilation louvers, it was assumed that a in-cabinet fire would propagate to the tray directly above but not to an adjacent cabinet. An SRAI (see Section B.3 in Appendix B of this report) was issued to the licensee concerning the propagation of cabinet fires in the AEER to adjacent cabinets. The results for the AEER were presented in a table in the revised submittal which identified the cabinet, the number of sections associated with it, the fire scenario ID, whether or not the fire propagated outside the cabinet, and the fire consequences. For Unit 2 and Unit 3 the percentage of fires which resulted in propagation were 16% and 14%, respectively. Self-initiated cable tray fires were also considered for the AEER.

The results of the single compartment analyses for Unit 2 and Unit 3 were presented in separate tables in the revised submittal. In some cases the same compartment was analyzed. For each scenario analyzed, the table contained the compartment ID, the scenario ID/description, compartment name, ignition frequency, probability of fire non-suppression, severity factor used, CCDP, and CDF contribution. These tables show that in some scenarios non-suppression probabilities were used in conjunction with severity factors. Taking credit for both of these items in the original analysis was the subject of RAI #2. The licensee response to this RAI, which referenced the revised analysis, is discussed in Section A.2 of Appendix A to this report.

A multi-compartment analysis (MCA) was also performed to evaluate the potential for a fire starting in one compartment spreading to or damaging equipment in an adjacent compartment. The analysis used a graded screening approach that considered the potential for severe fires that challenge the integrity of barriers, the frequency of occurrence, and, if necessary, the challenge to plant safe shutdown capability assuming loss of equipment in both compartments.

The MCA methodology was based on the FPRAIG, which was the source of the barrier failure probabilities used in the MCA. The analysis focuses on the physical boundaries (some unrated) that separate the Dresden fire areas. For each compartment considered in the revised fire analysis, the fire detection and suppression features present were determined, along with the adjacent fire compartments. As a first step, potential multi-compartment scenarios involving downward propagation pathways were screened. This is a step that is not consistent with the FIVE fire compartment interaction analysis (FCIA) methodology and could ignore potentially significant fires, e.g., those due to oil spills. Multi-compartment scenarios involving the drywell as the initiating fire compartment were also screened due to the inert atmosphere present.

The next MCA screening step screened scenarios if the exposing fire compartment does not contain credited fire PRA equipment. This was justified on the basis that the potential consequences of fire propagation to an adjacent area is bounded by the analysis for the adjacent compartment. This may be optimistic if the adjacent compartment has significant combustibles but limited ignition sources. Multi-compartment scenarios were also screened if the exposing fire compartment does not contain significant ignition sources.

For fire compartments with area-wide fire suppression system coverage, a multi-compartment scenario initiating event frequency was calculated. This frequency was determined based on the total fire ignition frequency of the compartment, a severity factor of 0.20, the automatic fire suppression system failure probability, and the barrier failure probability. Multi-compartment fire scenarios with frequencies below 1.0E-6/yr were screened. No justification for the 0.20 severity factor was provided.

Following the above screening, a multi-compartment scenario initiating event frequency was calculated for fire compartments with area-wide fire detection system. This frequency was determined based on the total fire ignition frequency of the compartment, a severity factor of 0.20, a fire brigade failure probability of 0.10, and the barrier failure probability. Multi-compartment fire scenarios with frequencies below 1.0E-6/yr were screened. As in the case of the fire suppression system treatment, no justification for the 0.20 severity factor was provided.

For those cases where the initiating (exposing) compartment has either an automatic suppression or detection system, and the adjacent (exposed) compartment has an independent fire detection or suppression systems, credit is taken for the redundant system. If fire modeling showed that a damaging compartment-wide HGL condition does not exist, the scenario was screened. In general, this is probably consistent with the simpler FIVE FCIA criteria which allows screening of a boundary if an automatic fire suppression system is installed over combustibles in the exposing compartment.

The final screening step treated those cases in which neither an automatic fire detection or suppression system is available. The multi-compartment scenario initiating event frequency was determined based on the total fire ignition frequency of the compartment, a severity factor of 0.20, and the barrier failure probability. Multi-compartment fire scenarios with frequencies below 1.0E-6/yr were screened. The revised submittal states that compartments which are unscreened following this step "require detailed examination for potential risk contribution."

The Dresden MCA results show, in Table 4-18 of the submittal, that the screening steps discussed above resulted in all of the multi-compartment fire scenarios being screened. Each of the 83 fire compartments considered in fire analysis are shown in the table as exposing fire compartments, along with the compartment name, and the MCA result which includes a reference to a one of the notes in a list at the end of the table. The table and notes did not identify the initiating event, the exposed compartment(s) considered in the fire scenarios, or the multi-compartment scenario initiating event frequencies that were used to screen some of the compartments per the steps discussed above. The basis for the severity factors used was not provided. It was optimistically assumed that compartments involving downward propagation pathways could be screened. Compartments were screened using screening criteria that were cited in the results, but were not described in the description of the steps used in the MCA methodology.

In the original Dresden IPEEE submittal, it appeared that unrated fire barriers had been credited in the MCA. As a result, RAI #8 requested the licensee to discuss the impact of not crediting such barriers in the MCA. The licensee was also asked to consider the impact resulting from all barriers in high hazard fire areas failing and describe the effect on the resulting CDFs for the associated fire scenarios. The response to RAI #8 is not complete as discussed in Section A.8 of Appendix A to this report and represents a weakness in the revised submittal. However, based on fire risk assessments at other plants, it is unlikely that a more complete multi-compartment analysis would result in dominant fire risk contributors.

The control room analysis followed the guidelines in Appendix M of the FPRAIG. Only cabinet fires were considered and no severity factors were apparently used except in the case of the bounding control room fire. To carry out the analysis, the plant system functions associated with each of the control room cabinets were identified. The control room ignition frequency was apportioned to the cabinets as described in Section 2.1.4 of this report. The functions impacted by postulated control room fires in each cabinet were then determined so that the CDF contributions for the fire scenarios could be determined. The panels analyzed were identified by ID, description, and fire-impacted PRA systems in Table 4-19 in the revised submittal. It also noted if the cabinet/panel fire would or would not cause a plant trip. The specific locations of these panels was not provided.

The control room was analyzed on the basis of the 78 space (panel) units identified as described in Section 2.1.4 of this report. Those scenarios where a postulated fire was determined to cause a plant trip were quantified unless no post fire safe shutdown functions are immediately disabled. If a postulated fire does not cause a plant trip and no post fire safe shutdown functions are immediately disabled, the scenario is screened. If a postulated fire does not cause a plant trip but post fire safe shutdown functions are impacted, the scenario was screened if the affected system is not risk significant relative to the scenario being considered. A bounding fire that was not suppressed and would require abandonment of the control room was also quantified using the total control room ignition frequency. A table was provided in the revised submittal which presented the results of the fire scenarios analyzed for each panel, including the effects of the fire and the CDF contribution.

No examples of the cabinet modeling were provided to show how the guidance in Appendix M of the FPRAIG was applied in the analysis. Identifying critical cabinets, determining remote shutdown capability, and defining control room fire damage sequences are some of specific steps in the Appendix M method. However, no activities related to these steps were mentioned in the control room analysis discussion. The lack of such information in the analysis is considered a weakness.

2.1.7 Quantification and Uncertainty Analysis

The results of the fire modeling and subsequent quantification of the PRA models indicates that three of the 16 unscreened Unit 2 fire compartments and six of the 21 Unit 3 fire compartments analyzed in detail have an estimated core damage frequency greater than 1.0E-6/yr due to fires. Compartment CDF contributions greater than 1E-7/yr for each unit are shown in the following table along with the total unit CDFs.

Fire Compartment	Description	Unit 2 CDF	Unit 3 CDF
1.1.1.2	U3 Reactor Bldg. Ground Floor	screened	7.16E-7/yr
1.1.1.3	U3 Reactor Bldg. Mezzanine	screened	3.54E-6/yr
1.1.2.1	U2 Torus Basement	1.10E-7/yr	screened
1.1.2.3	U2 Reactor Bldg. Mezzanine	1.65E-6/yr	screened
1.4.1	U3 Tip Room	screened	1.10E-7/yr
2.0	U2/3 Control Room	7.15E-6/yr	7.11E-6/yr
6.2	U2/3 Aux. Electric Equipment Room	5.36E-7/yr	2.53E-6/yr
8.2.1.B	U3 Cond. PP Area	screened	4.85E-7/yr
8.2.4	U3 Cable Tunnel	screened*	2.12E-6/yr
8.2.5.A	U2 No. Trackway/Switchgear Area	5.38E-6/yr	4.94E-7/yr
8.2.5.C	U2/3 Turbine Bldg. Corridor	2.52E-7/yr	8.36E-7/yr
8.2.5.E	U3 West Corridor and Trackway	1.17E-7/yr	6.85E-6/yr
8.2.6.A	Control Room Backup Ventilation	5.86E-7/yr	4.59E-7/yr
8.2.6.B	U2 Mezzanine	6.74E-7/yr	screened
8.2.6.C	U2/3 SBGT and TBCCW HX	<1E-8/yr	5.32E-7/yr
8.2.6.D	U3 Mezzanine Floor	screened	7.90E-7/yr
8.2.6.E	U3 Mezzanine Floor	<1E-7/yr	3.44E-6/yr
8.2.7	Vent Room Over NE Switchgear	screened	2.39E-7/yr
9.0.B	U3 D/G	screened	2.19E-7/yr
11.3	Cribhouse Upper	2.45E-7/ry	2.38E-7/yr
Total CDF		**1.69E-5/yr**	**3.08E-5/yr**

* The Unit 3 cable tunnel screened for Unit 2. The licensee was issued an SRAI (#2) requesting an explanation as to why no detailed assessment of a Unit 2 cable tunnel was presented especially in light of the fact that the cable tunnel is directly below the turbine building. The response is summarized in Section B.2 in Appendix B of this report.

The dominant scenarios for each unit involve control room fires. In both cases the dominant control room scenarios involve a severe fire requiring evacuation and shutdown from outside the control room. The second most important scenario for each unit involves a large reactor feedwater pump (RFP) fire with successful actuation of the suppression system. This fire involves fire compartment 8.2.5.A in Unit 2 with a CDF of 2.48E-6/yr and fire compartment 8.2.5.E in Unit 3 with a CDF of 3.4E-6/yr. An MCC fire in compartment 8.2.5.A in Unit 2 contributed 1.68E-6/yr to the CDF for this unit. The other Unit 2 scenario with a CDF greater than 1E-6/yr involves reactor water cleanup (RWCU) pump fires in compartment 1.1.2.3. The other Unit 3 fire scenarios with a CDF contribution of greater than 1E-6/yr are:

- A DC panel fire in compartment 8.2.6.E - CDF = 2.69E-6/yr
- A self-initiated cable fire in compartment 1.1.1.3 - CDF = 1.93E-6/yr
- Compressor fires with and without suppression in compartment 8.2.5.E - CDF = 1.9E-6/yr
- A fire in compartment 6.2 which propagates to another tray - CDF = 1.28E-6/yr

The most risk significant control room fire which does not require control room abandonment involves a postulated fire in panel 902-8/903-8. This fire results in LOSP and loss of Division II of the onsite AC power distribution system. It was also found that the lower damage threshold of the non-IEEE 383 qualified cables limited the effectiveness of the automatic fire suppression systems. Although not supported by analysis, the licensee expected that the impact of the RFP oil fire would be reduced if IEEE-383 qualified cables were installed.

As noted in Section 2.1.6 of this report, dual unit loss of offsite power was considered as an initiating event in the PRA model. An RAI on the original study requested that the licensee to address in more detail fires that could impact both units. The response to this RAI (#10) is summarized in Section A.10 in Appendix A of this report.

Based on the results of the fire analysis, it did not appear that there are postulated fires that lead directly to core damage. The MCA concluded that all of the postulated multi-compartment fire scenarios could be screened.

2.1.8 Sensitivity and Importance Ranking Studies

The revised Dresden submittal contain no information related to these topics.

2.2 Special Issues

As a part of the IPEEE fire submittal, the utilities were asked to address a number of fire-related issues identified in the Fire Risk Scoping Study (FRSS) [10] and USNRC Generic Safety Issues (GSI). Specific review guidance on these issues is found in Reference 6. Some, but not all, of these issues were discussed in the original Dresden IPEEE submittal [1] and the revised submittal

[3 (Attachment1)]. The responses to the FRSS issues appear to be the same in both documents. In the following paragraphs, the applicable sections in the revised submittal are referenced.

2.2.1 Decay Heat Removal (USI A-45)

As discussed in Generic Letter 88-20 [4] and NUREG-1407 [5], USI A-45 which is associated with the adequacy of decay heat removal (DHR) at nuclear power plants is subsumed into the IPE submittals. A submittal meeting the intent of Generic Letter 88-20, Supplement 4 is assumed to satisfy the requirements of USI A-45. Specifically, the fire assessment presented in the IPEEE submittal should address the adequacy of long-term decay heat removal in the event of fires.

DHR is addressed in Section 4.11.1 of the revised submittal [3 (Attachment1)]. It describes the various systems that are used for DHR for transient events and "feed and bleed" (as it is referred to in the submittal). The Isolation Condenser (IC) system and selected makeup sources are defined as a part of the safe shutdown paths that are used for a majority of the fire areas at Dresden. The HPCI system is used for the remaining fire areas. According to the submittal, the Dresden Safe Shutdown Analysis (SSA) and related procedures document DHR capability by demonstrating the availability of equipment needed for IC and HPCI operation. (However, an essential element of DHR, suppression pool cooling, is not discussed except as noted below.) The fire IPEEE models these safe shutdown methods as well as others, subject to the availability of offsite power. The submittal states that the results of the fire analysis demonstrate the availability of DHR for any fire leading to a non-LOCA event. That is, no fire scenario leads to a CCDP of 1.0. However, the revised submittal states that excluding a severe control room fire, the dominant fire-induced core damage sequence involves a loss of decay heat removal.

A fire-induced plant trip is postulated to occur on low reactor pressure vessel (RPV) water level due to an inadvertent open relief valve (IORV). For such an event, decay heat is removed by providing RPV makeup with HPCI and operating the LPCI and Containment Cooling Service Water (CCSW) systems in suppression pool cooling (SPC) mode, or by use of the IC. The SSA documents the availability of these systems as part of the safe shutdown methods. The fire IPEEE models these safe shutdown methods as well as others, subject to the availability of offsite power. The submittal states that the results of the fire analysis demonstrate the availability of DHR for any fire leading to an IORV event. That is, no ignition source has a corresponding CCDP of 1.0.

The submittal concludes that DHR will be available, with necessary manual actions, following a fire in any location at Dresden. This is based on the CDF results for the compartments containing DHR equipment, on the redundancy of the methods and equipment needed for DHR, and on the implementation time required to achieve various stages of the DHR functions.

No time period for successful DHR is provided. The effects of postulated component failures on mitigating systems are not specifically discussed. Other than the systems mentioned above, no list of components (safety or non-safety grade) is provided in the submittal.

2.2.2 Effects of Fire Protection System Actuation on Safety-related Equipment (FRSS, GSI 57)

This issue is associated with the concern that traditional fire PRA methods have generally considered only direct thermal damage effects. Other potential damage mechanisms have not been addressed, such as smoke and the potential that the activation of fire suppression systems, either as part of actual fire fighting or spuriously, might result in damage to plant systems and components. In general, this is an area where the database on equipment vulnerability is rather sparse. The analytical results obtained for resolution of the issue, subsumed by GSI-57, identified the dominant risk contributors as: (1) Seismic-induced fire plus seismic-induced suppressant diversion and (2) Seismic-induced actuation of the fire protection system (FPS). The NRC anticipated that the licensee would conduct seismic/fire walkdowns to assess (1) whether an actuated FPS would spray safety-related equipment, and (2) whether some protective measures to prevent the same could be instituted. The results could be documented in the IPEEE submittal.

GSI 57 is addressed in Section 4.11.2 of the revised submittal [3 (Attachment1)]. It states that GSI 57 was investigated in 1985 by examining the effects of fire suppression system actuation on nuclear safety related equipment in response to IN 83-41. Modifications were apparently made at that time to ensure that safety-related equipment was not subject to damage from the perils described in IN 83-41. In support of the IPEEE, inadvertent actuation of fire suppression systems was examined by performing a walkdown, as described in Section 4.10.2.1.2 of Reference 3, Attachment 1. During the walkdown, the potential for spraying of safety related equipment or the release of fire suppression media due to a seismic event was evaluated. No instances where such events could occur were observed. A review of relays which could potentially lead to inadvertent suppression system actuation found that no such relays exist in the plant.

Per Section 4.10.2.1.3 of Reference 3, Attachment 1, seismic degradation of fire suppression systems was reviewed by walking down fire piping and looking for poor structural design features or potential interactions with safe shutdown path components. No potential interactions were noted except for piping near two panels which were analyzed in the seismic portion of the IPEEE. The results were not noted in the fire portion of the submittal. However, it was concluded that fire protection system piping was not expected to fail due to a seismic event and safe shutdown path components will not be damaged.

Section 4.10.2.4.1 of the revised IPEEE mentions that plant staff should be aware of the potential impact of smoke and products of combustion on human performance during safe shutdown operations. Plant operators are trained annually on the use of self-contained breathing apparatus (SCBA). The potential effects of combustion products on the ability of safe shutdown equipment to continue to function in such environments or the effects of smoke transport through the plant buildings were mentioned in the submittal as problems. However, it was noted that they were not considered in the analysis.

Operator action effectiveness in relation to safe shutdown procedures and training is also mentioned in Section 4.10.2.4.3 of the revised submittal.

2.2.3 Fire-induced Alternate Shutdown/Control Room Panel Interactions (FRSS, GSI 147)

The issue of control systems interactions is associated primarily with the potential that a fire in the plant, i.e., main control room (MCR), might lead to potential control systems vulnerabilities. Given a fire in the plant, the likely sources of control systems interactions are between the control room, remote shutdown panel, and shutdown systems. Specific areas that should be addressed in the IPEEE fire analysis include 1) Electrical independence of the remote shutdown control systems; 2) Loss of control equipment or power before transfer; 3) Spurious actuation of components leading to component damage, LOCA, or interfacing LOCA; and 4) Total loss of system function. It is anticipated that the licensee's submittal will describe its remote shutdown capability including the nature and location of the shutdown station(s) and the types of control actions which can be taken from the remote panel(s).

Section 4.10.2.5 of the revised submittal states that, as described in the Fire Protection Report (Appendix R Conformance/Safe Shutdown Report), safe shutdown circuits which are not independent of the Control Room (CR) are manually isolated in the event of a CR fire. The procedure which defines the operator actions required is referenced but not discussed. The location(s) where this isolation is performed and the equipment required or systems involved were not provided in the submittal.

Loss of control equipment or power before transfer; spurious actuation of components leading to component damage, or interfacing LOCA; and total loss of system function are not discussed in the submittal. The processes used to verify electrical independence and evaluate the level of indication and control of remote shutdown control and monitoring circuits were also not described. The potential effects of ISLOCAs on containment bypass were discussed in the submittal as described in Section 2.3 of this report. Other possible fire-induced LOCAs were not discussed.

Additional information on control systems interactions is provided in the response to RAI 9 provided in Section A.9 in Appendix A of this report.

2.2.4 Smoke Control and Manual Fire Fighting Effectiveness (FRSS, GSI 148)

Smoke control and manual fire fighting effectiveness is associated with the concern that nuclear power plant ventilation systems are known to be poorly configured for smoke removal in the event of a fire, and hence, a significant potential exists for the buildup of smoke to hamper the efforts of the manual fire brigade to promptly and effectively suppress fires. Sensitivity studies have shown that prolonged fire fighting times can lead to a noticeable increase in fire risk. Smoke, identified as one of the major contributors to prolonged response times, can also cause misdirected suppression efforts and hamper the operator's ability to safely shut down the plant.

The effects of smoke were considered as discussed in Section 2.2.2 of this report. It was assumed that under some conditions smoke from a fire in the Control Room would require evacuation.

Manual fire suppression was not credited as being effective in preventing damage to critical targets. Manual suppression was credited in the main control room because it is continuously manned. However, the potential for manual suppression, even though not credited, to be misdirected and cause collateral damage to nearby equipment was not discussed.

The following six topics were covered in some detail in Section 4.10.2.3 of the revised submittal [3 (Attachment 1)]:

- Fire reporting, including the use and availability of portable fire extinguishers and plant procedures for reporting fires, including plant communication.

- Fire brigade makeup, equipment, and physical condition requirements.

- Fire brigade classroom training, including fire fighting plan, team member responsibilities, identification of fire hazards/types, location of fire fighting equipment, plant layout, use of equipment, and fire fighting strategies.

- Fire brigade hands-on equipment and structural fire training.

- Fire brigade periodic and unannounced drills, including execution of fire preplans.

- Fire brigade training records.

In terms of operating in a smoke environment, the submittal notes that SCBA equipment is available and that smoke ejectors are included in three of the fire equipment carts.

2.2.5 Seismic/Fire Interactions (FRSS, MSRP)

The issue of Seismic/Fire Interactions primarily involves three concerns. First is the potential that seismic events might result in fires internal to the plant. Such threats might be realized from inadequately secured liquid fuel or oil tanks, through breakage of fuel lines, or through the rocking of unanchored electrical panels (either safety or non-safety grade). The second concern is the potential that seismic events might render fixed fire suppression systems inoperable. This could include detection systems, fixed suppression systems, and fixed manual fire fighting support elements such as the plant fire water distribution system. The third concern is that a seismic event might spuriously actuate fixed fire detection and suppression systems. The spurious operation of detectors might both complicate operator response to the seismic event and/or cause the actuation of automatic fire suppression systems. Actuation of a suppression system may lead to flooding problems, habitability concerns (in the case of CO_2 systems), diversion of suppressants to non-fire areas rendering them unavailable in the event of a fire elsewhere, the potential over-dumping of gaseous suppressants resulting in an over-pressure of a compartment, and spraying of important plant components. It had been anticipated that a typical fire IPEEE submittal would provide for some treatment of these issues through a focused seismic/fire interaction walkdown.

The fire-seismic walkdowns are described in Section 4.10.2.1.1 of the revised submittal. The walkdowns evaluated fixed plant systems including piping and hydrogen and combustible liquid storage vessels and determined that, with few exceptions, they are not subject to leakage due to seismic events. The following items were found during the fire-seismic walkdowns and evaluated as noted:

1. Oil Filled Step-Down Transformers
   Some of these transformers, associated with switchgear, were found to be not anchored and therefore subject to tipping which could cause an oil spill. These transformers were modeled as potential ignition sources and as sources of oil which could be released to the surrounding bermed area.

2. Hydrogen Seal Oil Panel and Hydrogen Monitors
   A hydrogen seal oil control panel and a turbine generator hydrogen monitor were found to be unanchored or inadequately anchored. Hydrogen lines are routed through these cabinets so the potential for a gas release exists. The submittal did not provide a licensee response to these problems in terms of modifications or modeling. This was the subject of RAI #11 which was submitted to the licensee. The RAI and licensee response are discussed in Section A.11 of Appendix A to this report.

3. Flammable Liquid Storage Cabinets in Reactor Building
   These cabinets were determined not to be subject to tipping so they were not considered a source of exposed combustibles in the fire evaluation.

4. PCB Holding Tanks
   These tanks have a sight glass which could break. However, it was found that these tanks are normally empty so they are not considered to be a fire hazard during normal operations.

5. Hydrogen Tanks in Tank Farm
   The tank farm is substantially removed from safety-related structures and equipment. It therefore does not represent a fire risk to them.

The possibility of a seismic event displacing cabinets and causing cable damage which could lead to fires was not discussed in the submittal.

2.2.6 Adequacy of Fire Barriers (FRSS)

The common reliance on fire barriers to separate redundant components needed to achieve safe shutdown has elevated the risk sensitivity of fire barrier performance. Degraded fire barrier penetration seals and unsealed penetrations in some barriers can contribute to this source of fire risk, since fires in one area might impact other adjacent or connected areas through the spread of heat and smoke. In general, it is expected that a utility analysis would provide for some treatment of such potential by considering that (1) manual fire fighting activities might allow for the spread of heat and smoke through the opening of access doors, and (2) the failure of active fire barrier elements such as normally open doors, water curtains, and ventilation dampers might compromise barrier integrity. Resolution of the fire barrier issue is to verify that fire barriers are properly installed and maintained under a surveillance program.

According to Section 4.10.2.2 of the revised submittal, fire barriers and components such as fire dampers, penetration seals, and barrier fire doors are included in the plant surveillance program. Fire rated barriers are visually inspected every 18 months. Additionally, specific surveillance is performed on fire doors, penetration seals, fire dampers and structural steel fire proofing. Ten percent of the penetration seals are inspected every 18 months. The penetration seals have also been evaluated in connection with concerns identified in various NRC Information Notices, including IN 88-04. Fire damper installation was evaluated per NFPA code reviews and to respond to concerns such as those identified in IN 83-69 and IN 89-52.

Fire barriers credited in the fire analysis were visually inspected during the development of the analysis to verify that they can prevent the spread of fire. In addition, a detailed multi-compartment analysis was performed which postulated barrier failure and fire propagation into adjacent compartments. The CDF resulting from this analysis was included in the overall plant fire risk.

2.2.7 Effects of Hydrogen Line Ruptures (MSRP)

The use of flammable gases in the plant, including hydrogen, introduces the potential that a rupture of the gas flow lines might lead to the introduction of a serious fire hazard into plant safety areas. It had been anticipated that a typical fire IPEEE analysis would include the consideration of such sources in the analysis.

The effects of hydrogen line ruptures were considered as part of the fire-seismic walkdowns as noted in Section 2.2.5 of this report.

2.2.8 Common Cause Failures related to Human Errors (MSRP)

Common cause failures resulting from human errors include operator acts of omission or commission that could be initiating events or could affect redundant safety-related trains needed to mitigate other initiating events. It had been anticipated that a typical fire IPEEE analysis would include the consideration of such failures in the submittal.

Section 4.5.3.1.2 of the original Dresden IPEEE submittal [1] includes a discussion of the human recovery actions and methods used in the fire analysis. Operator actions are included in the fire PRA model based on the Dresden Safe Shutdown Procedures and Emergency Operating Procedures (EOPs) which are used to respond to accident initiators. These actions are generally performed from the Control Room and were considered to be unaffected by a fire outside the CR. The effects of stress caused by the accident are included in the operator action failure probabilities. In particular, the CR analysis considers the effects of smoke and the fact that some fires will require evacuation. In Section 4.5.3.2.3 of [1], it is stated that the fire analysts determined which components were capable of being manually restored in each of the fire zones evaluated in the screening analysis. Failure probabilities for these operator actions were estimated by performing a human reliability analysis (HRA) for each type of action and included the stress effects due to the fire.

Section 4.5.4.1 of the revised IPEEE submittal [3 (Attachment 1)] states that the original analysis results were based primarily on the implementation of the Appendix R based post-fire safe shutdown procedures. The PRA model developed for the upgraded analysis relied on the implementation of the EOPs which were not fully credited in the original analysis. Section 4.5.4.4 of [3 (Attachment 1)] notes that the model incorporated all of the operator actions included in the plant internal events PRA model. Operator actions were categorized based on whether they occur within the control room or outside it. Timelines were also determined for each action. Key operator actions in each of these areas that were used in the analysis were discussed. For actions outside the control room, the PRA model included events for human actions that could be taken to mitigate the effects of system failures. The effects of fires on these actions were considered or conservatively included by using an HEP value of 1.0

2.2.9 Non-safety Related Control System/Safety Related Protection System Dependencies (MSRP)

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 licensee's IPE process should provide a framework for systematic evaluation of interdependence between safety-related and non-safety related systems and identify potential sources of vulnerabilities. It had been anticipated that the fire IPEEE analysis would include the consideration of such dependencies in the submittal.

The submittal contained no information related to this issue except as described in Section 2.2.3 of this report.

2.2.10 Effects of Flooding and/or Moisture Intrusion on Non-Safety and Safety-Related Equipment (MSRP)

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 the fire suppression system, or backflow through part of the plant drainage system. It had been anticipated that the fire IPEEE analysis would include the consideration of such events in the submittal.

The effects of automatic fire suppression system actuation on safety/safe-shutdown related equipment are discussed in Section 2.2.2 of this report. The other possible effects due to flooding noted in the previous paragraph were not discussed in the submittal.

2.2.11 Shutdown Systems and Electrical Instrumentation and Control Features (SEP)

The issue of shutdown systems addresses the capacity of plants to ensure reliable shutdown using safety-grade equipment. The issue of electrical instrumentation and control addresses 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. It had been anticipated that the fire IPEEE analysis would include the consideration of this issue in the submittal.

The portions of this issue that were covered in the submittal are described in Section 2.2.3 of this report.

2.3 Containment Performance Issues Unique to Fire Scenarios

The submittal discusses containment performance in the context of the potential effects of fires. The submittal points out that the containment issues that need to be considered are bypass, isolation failures, and other failure modes. Fire impact on the containment itself is expected to be minimal. Hatches are not expected to sustain fire damage and fire is not expected to fail the steel piping and cable penetrations. The containment fire area was eliminated during the screening phase using the approach suggested in the FIVE methodology [7]. In addition, except for brief periods after startup, before a shutdown, or for infrequent drywell entries at power, the primary containment is inerted with nitrogen, further reducing the risk of a fire at power.

The high pressure/low pressure interfacing systems LOCA (ISLOCA) paths identified in the Dresden IPE were reviewed to determine the possible impacts due to fire. According to Appendix N of the FPRAIG, any path that contains two or more closed valves that are not susceptible to fire can be screened from further evaluation. Based on this criteria, all but two paths, the LPCI and core spray injection lines, were screened. The scenarios involving these lines include one random failure (leakage/rupture of the check valve) and one spurious actuation of a closed MOV. An analysis of these scenarios concluded that the upper bound CDF for the fire induced ISLOCA event is 1E-8/reactor-year per line. An analysis was not performed to determine if a temporary spurious actuation could be recovered or a seal-in circuit exists which would keep the valve open after circuit failure progressed to an open circuit. However, the licensee considers this estimate conservative because it was assumed that a fire anywhere in the zone would damage the cable(s) associated with the applicable MOV(s).

Per FIVE, the ten fire compartments that were not screened were reviewed for potential impact on containment performance. Additional accident sequences that could fail containment prior to core damage were also reviewed. These sequences are related to failure to scram and loss of decay heat removal. These heat removal "mismatch sequences" can be induced by fires. The failure to scram sequences were screened based on low frequency. The character of the loss of decay heat removal sequences was found to be the same as those identified in the internal events analysis. As a result, no new insights related to containment performance were identified.

Dresden normally operates with the containments inerted with nitrogen. The IPE concluded that containment isolation failures are not likely and they were not included in the IPE event trees. The scope of the containment isolation pathways considered in the revised submittal is the same as that evaluated in the Level 2 PRA. This scope includes containment penetration paths larger than two inches in diameter.

Fire-induced impacts on automatic primary containment isolation valves (PCIVs) may be postulated due to hot shorts. However, the Primary Containment Isolation System is equipped with design features (described in the submittal) which minimize the likelihood of containment isolation failure. The Appendix R analysis also concluded that the probability of both isolation valves in a line being affected by a fire such that they spuriously open was too low to warrant further consideration.

2.4 Plant Vulnerabilities and Improvements

In the original IPEEE submittal, the licensee stated that the NEI Severe Accident Closure guidelines were used to identify vulnerabilities. This statement did not appear in the revised submittal which did not specifically define what constituted a fire vulnerability. No required hardware modifications were identified in either submittal. However, the response to RAI #11 stated that two hydrogen related systems would be modified (seismically mounted) to reduce the risk associated with a seismic/fire event and provided a time frame for the modifications.

After screening in the revised fire assessment, 37 fire areas remained that required detailed analysis. The licensee states that the calculated CDF contribution due to postulated fire events is consistent with other BWR plants. Section 4.9.2 of the submittal stated that the upgraded analysis highlighted 11 insights, but only six were listed. None of the insights mentioned any potential or planned plant improvements. The insights did not refer to any of the dominant core damage contributors or sequences as vulnerabilities. However, the licensee noted that, as of the date of the original submittal, the licensee says no unresolved Appendix R modifications have been identified.

Based on the fire scenarios with the largest contribution to the Unit 2 CDF, a Control Room fire which requires evacuation and fires in the North Trackway/SWGR area (Fire Zone 8.2.5.A) are the dominant contributors to CDF (69%). The dominant core damage sequence for Fire Zone 8.2.5.A results in loss of decay heat removal. For Unit 3 the largest contribution to CDF is also a Control Room fire which requires evacuation. Fires in the West Corridor and Trackway (Fire Zone 8.2.5.E) and Mezzanine Floor (Fire Zone 8.2.6.E) are also dominant contributors to the Unit 3 CDF. Fire scenarios in these two zones plus the Control Room comprise 54% of the Unit 3 CDF.

The total CDF due to fires for was estimated to be 1.69E-5/yr for Unit 2 and 3.08E-5/yr for Unit 3. The multi-compartment analysis demonstrated that fires involving two or more compartments do not significantly impact risk at Dresden. Based on the analysis, oil fires are a primary risk.

3.0 CONCLUSIONS AND RECOMMENDATIONS

In most areas the submittal responds to the intent of the IPEEE study, including responses relevant to FRSS issues, Generic Issues, Unresolved Safety Issues, and Multiple Systems Response Program issues. Several areas relevant to fire could not be assessed in this submittal-only review. Some strengths of the analysis include:

- Calculation of CCDPs using a model developed from the Dresden IPE internal events PRA model to provide estimated CDFs for significant fire events.

- Performance of a well-structured walkdown effort in conjunction with a good review of plant documentation sources.

- Consideration of plant specific (ComEd) fires to identify any adjustments that were needed for fire compartment frequencies. (Effects were found to be minimal.)

- Treatment of operator actions, including consideration of the effects of fires on these actions.

The major weaknesses in the Dresden submittal were addressed either by the revised fire assessment or the responses to the RAIs and SRAIs which were issued to the licensee. Remaining weakness in the fire assessment in terms of meeting the requirements of the IPEEE process, but whose effects on the results are minimal, or understood, and do not require further elaboration include the following:

- The treatment of transient ignition sources in four fire areas discussed in the response to SRAI #1did not consider the resulting hot gas layer or the potential for damage to overhead cables.

- Two weaknesses were identified in the control room analysis. First, it was assumed that multiple hot shorts affecting the RPS circuits could not occur and thereby prevent a reactor scram. It was also assumed that postulated fires can be screened as non-risk significant if the fire is suppressed before abandonment is required and the functional loss of the affected cabinets does not cause a plant trip. This is optimistic since any significant fire in the control room would very likely lead to a trip.

- The control room analysis also did not provide examples of the cabinet modeling to show how the guidance in Appendix M of the FPRAIG was applied in the analysis. Identifying critical cabinets, determining remote shutdown capability, and defining control room fire damage sequences are some of specific steps in the Appendix M method. However, no activities related to these steps were mentioned in the control room analysis discussion.

- Fire severity factors were used in the analysis of many fire compartments. The possibility that the use of a fire severity factor when fire suppression is explicitly modeled credited suppression efforts twice and resulted in RAI #2. The response to the RAI was adequate; however, the lack of explanations for the non-suppression probabilities (NSP) used in two cases and the use of an

NSP value in a compartment that has no suppression are considered to be weaknesses.

- The potential effects of fire-induced multiple hot shorts and cable-to-cable hot shorts were not considered in the fire analysis.

- The compartments considered in the MCA were identified, but the results did not identify the initiating event, the exposed compartment(s) considered in the fire scenarios, or the multi-compartment scenario initiating event frequencies that were used to screen some of the compartments per the steps discussed above. The basis for the severity factors used was not provided. It was optimistically assumed that compartments involving downward propagation pathways could be screened. Compartments were screened using screening criteria that were cited in the results, but were not described in the description of the steps used in the MCA methodology. In addition, the response to RAI #8 discussed in Section A.8 of Appendix A to this report did not adequately address the issues raised concerning the credit for unrated fire barriers in the MCA.

- The location weighting factors used for the Turbine Building and the switchyards do not appear to be consistent with the guidance provided in FIVE. Also, the ignition source weighting factors for the switchyard zones were not discussed.

- Except in a few cases, compartment fire frequency values used in the analysis were not identified.

Based on the revised Dresden IPEEE submittal and the responses to RAIs, the reviewers recommend that a sufficient level of documentation and appropriate bases for analysis have been established to conclude that the subject licensee fire submittal has substantially met the intent of GL 88-20.

4.0 REFERENCES

1. "Individual Plant Examination of External Events for Seismic, Fire, High Winds/Tornadoes, External Floods, Transportation Accidents; Dresden Nuclear Power Station Units 2 and 3," Final Report, Vol. 1, ComEd, December 30, 1997.

2. Letter from USNRC to O. D. Kingsley (ComEd), "Request for Additional Information Regarding the Individual Plant Examination of External Events (IPEEE) for Dresden Nuclear Power Station, Units 2 and 3 (TAC NOS. M83616 and M83617)," dated December 14, 1998.

3. Letter from ComEd to USNRC with attachments, "Request for Additional Information Regarding Individual Plant Examination of External Events," (Atch 1: Revision to the Individual Plant Examination of External Events for Dresden Nuclear Power Station Units 2 and 3; Atch 3: Response to Fire Questions), dated March 30, 2000.

4. USNRC, "Individual Plant Examination of External Events for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," Generic Letter 88-20, Supplement 4, April 1991.

5. USNRC, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, May 1991.

6. S. Nowlen, M. Bohn, J. Chen, Guidance for the Performance of Screening Reviews of Submittals in Response to U.S. NRC Generic Letter 88-20, Supplement 4: 'Individual Plant Examination - External Events,'" Rev. 3, 21 Mar 1997.

7. EPRI TR-100370, "Fire Induced Vulnerability Evaluation (FIVE)," April 1992.

8. EPRI TR-105928, "Fire PRA Implementation Guide," Final Report, December 1995.

9. NSAC-178L, "Fire Events Data Base for U.S. Nuclear Power Plants," EPRI Report, December 1992.

10. NUREG/CR-5088, "Fire Risk Scoping Study", January 1989.

11. Letter from USNRC to O. D. Kingsley (ComEd), "Dresden Units 2 and 3 - Request for Additional Information Regarding the Individual Plant Examination of External Events (IPEEE)," dated December 7, 2000.

12. Letter from Exelon Nuclear to USNRC, "Dresden Nuclear Power Station, Units 2 and 3 - Request for Additional Information Regarding Individual Plant Examination of External Events," dated January 31, 2001.

Appendix A
Description and Evaluation of Dresden RAIs

A.1 **RAI #1**: Conformance of Dresden automatic fire suppression systems (AFSSs) with NFPA standards.

A.1.1 Synopsis of RAI

The automatic suppression failure analysis used reliability values from the FIVE methodology. This data is acceptable for systems that have been designed, installed, and maintained in accordance with appropriate industry standards, such as those published by National Fire Protection Agency (NFPA). RAI #1 asked for verification that automatic fire suppression systems at Dresden meet NFPA standards.

A.1.2 Synopsis of Licensee Response

The response stated that the plant AFSSs comply with "applicable NFPA codes of record except where deviations have been identified. Technical justifications have been provided for these deviations."

A.1.3 Assessment of RAI Response

The response to RAI #1 is considered satisfactory.

A.2 **RAI #2**: Credit in fire assessment for automatic suppression in conjunction with use of a fire severity factor.

A.2.1 Synopsis of RAI

Fire severity factors were used in the analysis of many fire compartments. There appears to be a significant possibility that the use of a fire severity factor when fire suppression is explicitly modeled credits suppression efforts twice. The licensee was asked to (1) describe the instances in the Dresden fire assessment in which automatic fire suppression was credited explicitly in conjunction with the use of a fire severity factor, (2) explain, for each case, why such credit does not constitute redundant credit for suppression, and (3) reanalyze the CDF contribution from each scenario where such credit is identified.

A.2.2 Synopsis of Licensee Response

The response stated that the revised fire risk assessment credits AFS systems in conjunction with the use of fire severity factors in several areas. Table 4-14 of the revised submittal shows that suppression was credited in five fire compartments in Unit 2. Table 4-16 shows that suppression was credited in three fire compartments in Unit 3. In two cases the sources of the non-suppression probabilities (NSPs) that appear in Table 4-14 are not explained. For fire compartment 1.1.1.2 (Unit 2 Reactor Building Ground Floor) an NSP value of 0.1 was used. This room has partial suppression. The NSP value used for fire compartment 7.0.A.1 (Unit 2 Battery Room) was 0.033. According to Table 4-4 of the submittal this compartment has no suppression. No NSP values were provided in the original submittal.

The application of the severity factors was based on a review of fire incidents in the EPRI Fire Events Database (presumably Reference 9). The methodology for partitioning the fire events in this database required that any fire that caused the actuation of an AFSS be treated as a large (severe) fire regardless of the actual consequences of the event. Therefore, fires that could become severe if the suppression system failed were properly classified as severe events. On this basis, the licensee argues no reanalysis is required.

A.2.3 Assessment of RAI Response

The response to RAI #2 is considered adequate. However, the lack of explanations for 1) the NSPs values used in two cases and 2) the use of an NSP value in a compartment that has no suppression are considered to be weaknesses.

A.3 **RAI #3:** Use of qualified cable ignition temperature for unqualified cable.

A.3.1 Synopsis of RAI

In the original Dresden IPEEE submittal it appeared that it was assumed that the plant cables were not IEEE-383 qualified. However, a cable ignition temperature of 932°F was also assumed and the *EPRI Fire PRA Implementation Guide* (FPRAIG) was cited as the basis for this value. This value is significantly optimistic in comparison to piloted ignition temperatures observed in tests by Sandia National Laboratories (SNL). The SNL tests show that the piloted ignition temperature for cables will be as low or lower than the thermal damage threshold; hence, use of a piloted ignition temperature of no greater than 425°F would be appropriate for unqualified cable.

If a cable ignition temperature of 932°F was used, the licensee was asked to (1) describe the fire scenarios, associated cables, and analysis results for those cases in which it was applied and (2) provide a specific basis for the assumption that the cables at Dresden are consistent with this temperature. Alternatively, the licensee was asked to provide an assessment of the impact on the analysis results (CDF) if it is assumed that the flammability and/or appropriate non-qualified cable damage properties, including a piloted ignition temperature of 425°F, are applicable.

A.3.2 Synopsis of Licensee Response

The licensee response stated that the revised Dresden fire assessment did not use the 932°F cable ignition temperature recommended in the FPRAIG. Instead, all fire modeling analysis was based on a temperature of 425°F.

A.3.3 Assessment of RAI Response

The response to RAI #3 is considered satisfactory.

A.4 RAI #4: Analysis of fire propagation and equipment damage based on experimental results.

A.4.1 Synopsis of RAI

Inappropriate use of experimental results (e.g., employing propagation times specific to a particular cable tray separation for fires involving cable trays with lesser separation) can lead to improper assessments of scenario importance. The original Dresden submittal apparently assumed a fixed fire spread geometry (35°) for at least one cable tray scenario and fixed propagation delay times between the involvement of subsequent stack trays in the fire. The submittal does not provide a basis for expecting the results of limited experimental observation to be reproduced in the plant fire scenarios.

For each fire scenario in which experimental data were used to estimate the rate and extent of fire propagation, the licensee was asked to describe the scenario and how the experimental results were used in the analysis. In those cases where the analysis that was used is found to be unjustified, the licensee was asked to analyze the scenario using FIVE (or a similar methodology) and provide the results (equipment damaged). The response was to indicate which experimental results were used, how they were utilized in the reanalysis, and justify the applicability of these experimental results to the scenario being analyzed. The discussion on results applicability was to compare the geometries, ignition sources, fuel type and loadings, ventilation characteristics, and compartment characteristics of the experimental setup(s) with those of the scenario of interest.

A.4.2 Synopsis of Licensee Response

The licensee response stated that the revised Dresden fire analysis has no scenarios in which experimental data were used to estimate the rate and extent of fire propagation. The extent of fire propagation considered in the revised fire analysis relied on the fire modeling relationships contained in FIVE. The analysis applied a simplified approach that assumed no delay in fire propagation from the source to targets, except in those cases where suppression system actuation was credited.

A.4.3 Assessment of RAI Response

The licensee response to RAI #4 is considered to be satisfactory.

A.5 RAI #5: Heat loss factors used in scenarios requiring calculation of hot gas layer temperatures.

A.5.1 Synopsis of RAI

Hot gas layer predictions are very sensitive to the assumed value of the heat loss factor (HLF). Also, large HLFs cannot be justified for a single room fire scenario based on the information referenced in the FPRAIG. As a result, the licensee was asked to describe, for each multi-compartment or single area fire scenario analyzed, the scenario and the HLF used in the analysis to determine the hot gas layer temperature. The response was to include (1) a justification for the HLF value used and a discussion of its effect on the identification of fire vulnerabilities or (2) a revised analysis using a more justifiable value. The resulting changes in the scenario contributions to CDF were also to be provided.

A.5.2 Synopsis of Licensee Response

The licensee response stated that all single area fire scenarios assumed an HLF of 0.7 as recommended in FIVE. The analysis of multi-compartment scenarios, as described in Section 4.7.3 of the revised Dresden submittal, did not require the calculation of a hot gas layer temperature.

A.5.3 Assessment of RAI Response

The response to RAI #5 is considered satisfactory. However, the information provided in the RAI response and the revised submittal do not appear to be consistent. Section 4.4.1.2 of the revised submittal states that an HLF of 0.85 was used in the MCA. This value is not mentioned in the RAI response or in the section of the submittal (4.7.3) that describes the MCA and was justified per the revised EPRI guidance for Generic RAI Question #2. Also, Note 5. on page 4-90 of the submittal gives "... lack of an ignition source of sufficient magnitude to cause HGL conditions" as a multi-compartment screening criteria. How this criteria was satisfied for the 13 compartments that it was used to screen was not explained in Section 4.7.3 of the revised submittal.

A.6 RAI #6: Credit for both manual recovery of automatic suppression systems and manual suppression of fires.

A.6.1 Synopsis of RAI

The *EPRI Fire PRA Implementation Guide* methodology for evaluating the effectiveness of suppression efforts treats manual recovery of automatic suppression systems as being independent of subsequent manual efforts to suppress the fire. This assumption is optimistic. Also, the Nuclear Regulatory Commission (NRC) staff's evaluation of the FIVE methodology [7] specifically stated that licensees need to assess the effectiveness of manual fire-fighting teams by using plant-specific data from fire brigade training to determine the response time of the fire fighters.

The RAI requested identification of those scenarios for which credit is taken for both manual recovery of automatic suppression systems and manual suppression of the fires and the plant equipment that may be affected by the fires. The licensee was also asked to describe and justify how the dependencies between manual actions were treated in the analysis of these scenarios.

A.6.2 Synopsis of Licensee Response

The licensee response stated that the upgraded (revised) fire risk assessment has no scenarios in which credit is taken for both manual recovery of automatic suppression systems and manual suppression of the fires. The upgraded assessment did not credit recovery of any automatic fire suppression systems failures.

A.6.3 Assessment of RAI Response

The licensee response to RAI #6 is considered satisfactory.

A.7 **RAI #7:** Treatment of manual suppression.

A.7.1 Synopsis of RAI

In the original Dresden submittal, the treatment of manual suppression appeared to be derived from curves that indicate manual suppression success as a function of fire-fighting time. The submittal did not provide a basis for a quantitative assessment of manual suppression effectiveness at Dresden. An acceptable approach to the assessment of manual suppression success compares the damage time to the time required for suppression. The suppression time includes the time to detect the fire, the brigade response time, fire assessment time, and the extinguishment time.

The licensee was asked to provide a comparison of the manual suppression time to the damage time for those compartments where manual suppression was credited. This assessment was to include any adjustments resulting from responses to the above questions addressing ignition and damage temperatures, propagation delay assumptions, or model parameters.

A.7.2 Synopsis of Licensee Response

The licensee response stated that, with one exception, the revised Dresden fire risk assessment did not credit manual suppression in the individual compartment fire assessments. The exception involved the postulated main control room fire where operator action to suppress the fire is credited. Sections 4.6.2 and 4.7.4 of the revised submittal were referenced for additional details.

A.7.3 Assessment of RAI Response

The licensee response to RAI #7 is considered to be adequate. For a bounding control room fire, a probability of non-suppression of 3.4E-3 was applied. Based on the FPRAIG, this apparently assumed that about 15 minutes would be available to manually suppress a fire before smoke would require CR evacuation. A panel severity factor of 0.20 and a CCDP of 0.50 for shutdown from outside the control room were also applied. The applicability of the non-suppression probability value to the Dresden control room configuration was not discussed. Justification for the use of the severity factor and CCDP values noted above was provided in the response to RAI #12 as discussed in Section A.12 of this appendix.

A.8 **RAI #8:** Effectiveness of unrated fire barriers.

A.8.1 Synopsis of RAI

Assumptions concerning the effectiveness of unrated fire barriers can have a major impact on the screening of multi-compartment fires. The potential for fire barrier failure due to fires in high-hazard areas (e.g., large spills of oil or other liquid fuel, oil filled transformers, large turbine fires) can also be important.

a) Section 4.7.3.3.2 of the original Dresden submittal implied that unrated fire barriers had been credited in the fire study's multi-compartment analysis. The licensee was asked to assess the impact of eliminating the credit for such barriers in the multi-compartment analyses. (In the analysis, a damage temperature of 425°F for unqualified cable should be used. If a higher temperature is used, such as the 700°F referenced in Section 4.7.3.3.2 of the (original) submittal, the licensee was asked to provide justification.)

b) Based on the discussions provided in the original Dresden submittal for multi-compartment fire scenarios, it could not be determined that the impact of the failure of barriers in high hazard fire areas had been considered. The licensee was asked to evaluate the effect of such barrier failures and describe the resulting CDF contributions from the associated scenarios.

A.8.2 Synopsis of Licensee Response

The licensee response stated that the multi-compartment fire assessment described in the revised Dresden submittal was based on the fire zone definitions defined in the plant Appendix R related studies. These zones were examined as part of the revised fire assessment and the adequacy of their boundaries with respect to minimizing the likelihood of fire propagation was considered.

A formal re-examination of the multi-compartment analysis to eliminate credit for unrated barriers was not performed. Instead the licensee provided the following three qualitative insights which he feels indicate that no significant change in the overall conclusions of the multi-compartment analysis would result.

- The presence of an unrated barrier does not necessarily mean that a fire will readily propagate across the boundary. In many cases barriers are unrated because of unsealed openings.
- Many of the plant fire compartments have combustible loadings and/or ignition sources which are insufficient to lead to the formation of a hot gas layer.
- Dresden has automatic fire detection and suppression in many areas of the plant. Thus, the likelihood of a severe fire is limited.

The revised Dresden fire assessment modified the multi-compartment analysis so that it includes an initiating event frequency estimate. In the licensee's opinion the treatment of multiple boundary failures is indirectly addressed by these frequency estimates.

A.8.3 Assessment of RAI Response

The licensee responded to RAI #8. However, the response is not considered complete and therefore a weakness in the submittal. While the three qualitative arguments apply to some (or most) compartments/barriers in the plant, exceptions undoubtedly exist. For example, a barrier with an unsealed opening could lead to fire propagation to another compartment. The licensee response admits that the unsealed barrier would need to be examined to determine if the barrier could be credited, but states that this evaluation was not done.

Similarly, compartments with concentrations of combustible materials and/or ignition sources were ignored because many (not all) of the plant fire compartments do not have sufficient concentrations to cause a HGL to form. The argument that fire modeling would "typically" show that a target in the adjacent fire compartment would not be affected by a fire does not provide a basis for ignoring the impacts that might result from fires involving a subset of the fire compartments considered.

While it is obvious that many areas of the plant have fire detection and suppression systems, the last qualitative argument does not address the issue of the effects of fires on barriers in those compartments that do not have such systems.

A.9 RAI #9: Control systems interactions.

A.9.1 Synopsis of RAI

The issue of control systems interactions is associated with the potential that a fire in the plant might lead to potential control systems vulnerabilities. Specific areas of concern are electrical independence of remote shutdown control systems, loss of control equipment or power before transfer, spurious actuation of components leading to component damage or LOCAs, and total loss of system function.

In light of the above issues, the licensee was asked to provide:

- A description of control and instrumentation functions that are provided at each remote shutdown station. For each such function, the licensee was asked to indicate whether or not it can be isolated from damage in the main control room.
- An identification of scenarios that might not be mitigated by the remote stations.
- An evaluation of the reliability of the remote shutdown stations that includes consideration of spurious component actuations or LOCAs that might result from fire–induced cable faults, hot shorts, or component failures.
- A description of how the plant procedures provide for transfer of control to the remote station(s).
- An evaluation of whether loss of control power due to hot shorts and/or blown fuses could occur prior to transferring control to the remote shutdown locations and the associated risk

contribution of such failures.

A.9.2 Synopsis of Licensee Response

The licensee responded to each part of the RAI as follows:

- The plant does not rely on a single remote shutdown panel or station. Given a control room fire, safe shutdown is addressed through operator actions in various areas of the plant. The procedure which governs these actions was referenced in the response. Control functions are provided for strategic remote shutdown stations, such as isolation switches at 4kV switchgear. The response says these functions can be isolated from damage in the control room. Other control and instrumentation functions would be isolated and operated per procedures after evacuation. The availability of the remaining equipment needed for safe shutdown is ensured by isolating it from potential spurious operations (method unspecified). Remaining equipment can be isolated by removing power and positioning it manually. Instrumentation functions are not provided at the remote shutdown stations. Safe shutdown monitoring is performed using SSA required local instrumentation.

- The RAI response says the fire analysis did not identify any scenarios that might not be mitigated by the remote shutdown stations. Failure of equipment controlled by these stations was considered. For the control scenario which required evacuation, a CCDP of 0.5 was assumed. For fires outside the control room, it was assumed that remote station actions would, with two exceptions, fail. These exceptions were for operator actions related to the Alternate 125 Vdc Batteries and the Alternate 125 Vdc Battery Charger. The response says these actions were modeled using realistic failure probabilities.

- According to the RAI response, operator response to fires was modeled in terms of the EOPs, except for control room fires. As noted above, the fire analysis did not, in general, credit local actions involving the remote shutdown stations. Fire induced hot shorts and spurious operations were assumed to occur. According to the licencee, the risk of spurious actions resulting from fire induced cable faults or component failures was very low. Issues discussed further in this portion of the RAI response included the risk of spurious breaker operation, fire induced cable faults, and non-safe shutdown loads actions in response to a fire. Based on the isolation of power and the ability to control safe shutdown operations, the reliability of equipment controlled at the at the safe shutdown remote locations is considered acceptable.

  The potential for faults leading to component damage, a LOCA or ISLOCA prior to taking control at the remote shutdown stations, spurious starting and running of pumps, and repositioning of valves are all considered by the licensee to have a low probability. To support this conclusion the response noted that the safe shutdown analysis considered cable faults and identified modifications and actions to mitigate potential spurious operations. Also, modifications were made to address the issues associated with IN 92-18. The portion of the revised IPEEE submittal that discusses containment bypass and the associated low CDF was also referenced.

- As pointed out earlier, Dresden does not use a single remote shutdown panel or station. Detailed procedures prescribe manual actions and local controls required for shutdown. The actions to be taken in the event of a control room fire were described and it was noted that local control panels are available to ensure components can be operated outside the control room. Other control is achieved by closing or opening breakers and manual operation of equipment.

- The operator actions (scram, etc.) in the event of a control room fire to achieve hot shutdown were described. The response states that it is realized that a blown fuse can interrupt local control. The section of the SSA which addresses blown fuses was referenced, and it was noted that safe shutdown procedures are designed to insure that breakers/fuses are operable and local control is obtainable. The response states that hot shorts are not a concern because, in the event of a fire, equipment and cable required for safe shutdown "remains independent of the concerned fire zones." The section of the SSA that addresses fire induced cable faults was referenced and quoted to support the contention that this issue had been considered. The section of the revised IPEEE submittal (reference Section 2.1.5 of this report) that discusses fire induced cable failures - open circuit, short circuit, and hot shorts - was also referenced and discussed in the RAI response to this item. The licensee says the risk contribution of these types of cable failures is not readily quantifiable, but is considered to be significantly less than the CDF contribution from the control room fire scenarios which were quoted. Despite the previous statement, the response said the contribution due to cable hot shorts is included in the deterministic failure of equipment due to fire through the mapping to equipment linked in the fault tree model.

A.9.3 Assessment of RAI Response

The extensive response to RAI #9 is considered to be satisfactory.

A.10 **RAI #10:** Effects on CDF of fires that impact both units.

A.10.1 Synopsis of RAI

Except for a LOOP, fires that could affect both units were not considered, even though some fire areas contain elements of both units. As a result, the licensee was asked to provide the following:

- An identification of all fire areas that are shared between units and the potentially risk important systems/components for each unit that are housed in each such area. Include in the response (1) an assessment of the multi-unit fire risk associated with each such area, with emphasis on fire or smoke-induced control room evacuations and (2) an estimate of the risk contribution of the multi-unit scenarios identified.

- An indication of whether any fire response safe shutdown procedures call for unit cross-connects and the operator actions associated with these procedures. If any such cross-connects are required, the impact on fire risk if the total unavailability of the sister unit equipment is included in the assessment.

- An analysis of potential scenarios involving propagation of smoke, fire, and suppressants to and from fire zones containing equipment for the other unit. For any such multi-unit scenarios, provide an assessment of the associated risk contribution.

A.10.2 Synopsis of Licensee Response

The licensee response stated that all fires that could affect both units had been considered. According to the licensee, the revised IPEEE submittal consisted of a comprehensive analysis of CDF contributors for both units. This included the potential that a fire in one unit could adversely affect the functionality of ability of an operator to perform actions in the opposite unit. The potential for a multi-unit trip and multi-unit scenarios due to a fire is addressed and therefore reported as part of the CDF for each unit. Section 4.7 of the revised submittal was referenced.

With respect to cross-connects, the licensee notes that there are a limited number of such features credited in the fire risk assessment. These cross-connects consist of designed common shared equipment and cases where a designated opposite unit system is credited. In both cases, the licensee says that the fire risk assessment model explicitly treats such failures, crediting the opposite unit system only if both redundant trains are available. Section 4.7 of the revised submittal was also referenced in regards to this issue.

In response to the third request in the RAI, the licensee simply stated that the revised fire assessment provides the risk contribution of all multi-unit scenarios.

A.10.3 Assessment of RAI Response

The licensee responded to RAI #10. However, Section 4.7 of the submittal does not contain the specific information requested in the first two items in Section A.10.1 above. The information in Section 4.7 of the submittal shows that the effects of fires in some compartments represent potential fire risks to both units. Also, the multi-compartment analysis shows that fires originating in one compartment can affect both units. However, all such areas and the risk important equipment they contain were not identified and the impact of cross-connect availabilities was not specifically discussed.

Instead of an analysis of potential scenarios involving propagation of smoke, fire, and suppressants to and from fire zones containing equipment for the other unit, the licensee simply stated, as noted above, that the revised fire assessment provides the risk contribution of all multi-unit scenarios. The information in the submittal shows that some multi-unit scenarios were analyzed. However, the information needed to evaluate the completeness of this analysis was not provided.

A.11 RAI #11: Potential risk associated with inadequately anchored hydrogen lines.

A.11.1 Synopsis of RAI

As a result of the seismic/fire walkdown, a hydrogen seal oil control panel and a turbine generator hydrogen monitor were found to be unanchored or inadequately anchored. Hydrogen lines are routed through these cabinets so the potential for a gas release exists. The submittal did not assess the potential risk associated with this situation.

This RAI requested that the risk associated with a seismic/fire event due the inadequate anchoring be provided. Alternatively, a description of the existing systems, procedures, or modifications which could mitigate a seismic-induced failure of these hydrogen systems was requested.

A.11.2 Synopsis of Licensee Response

The licensee response stated that the hydrogen seal oil control panel and a turbine generator hydrogen monitor are being modified so that they will be seismically mounted. One design change was to be implemented starting in April 2000 (the month following the date of the RAI response) and the other is to be installed prior to the completion of the next Unit 2 refueling outage (date unspecified).

A.11.3 Assessment of RAI Response

The response to RAI #11 is considered to be satisfactory.

A.12 RAI #12: Alternate shutdown procedures in the event of a fire-induced main control room abandonment.

A.12.1 Synopsis of RAI

The original IPEEE submittal seemed to indicate that alternate shutdown in the event of a main control room abandonment due to a fire relies on an EDG whether or not off-site power remains available. This could result in a station blackout so the licensee was asked to describe how the plant alternate shutdown procedures address this possibility. If this situation has not been addressed, the licensee was asked to provide an assessment of the risk significance of potential SBO scenarios associated with remote shutdown.

A.12.2 Synopsis of Licensee Response

The licensee response stated that the revised fire risk assessment credited the Dresden Safe Shutdown Procedures (DSSPs) only for the bounding main control room fire scenario which would require control room abandonment. This event would place the plant in a configuration wherein the designated safe shutdown success path would involve the systems and plant features addressed in the DSSPs. The overall reliability of this path, given the need to perform all of the actions outside of the main control room, is, according to the licensee, covered by the bounding CCDP estimate of 0.50 which is included in the analysis. The licensee considers this failure probability to be relatively high and a value which bounds (1) the potential uncertainty in the HRA estimation and (2) the failure probability of the available safe shutdown path equipment, including any potential SBO scenarios.

A.12.3 Assessment of RAI Response

The response to RAI #12 is considered to be acceptable.

Appendix B
Description and Evaluation of Dresden SRAIs

B.1 SRAI #1: Analysis of transient fire sources

B.1.1 Synopsis of SRAI

Licensee appeared to have included transient fire frequencies with fixed ignition source frequencies and the fire modeling only considered fires from the fixed sources. Concern was raised that CDF from transient fire sources could have been under-estimated because fires at all critical locations may not have been considered. Licensee was asked to reassess the CDF from transient fire sources in 6 fire areas.

B.1.2 Synopsis of Licensee Response

Evaluated fires from transient sources that involve vertical cable risers in 4 of the areas. Grouped cable risers with less than 6 feet separation and assumed each group would be damaged by a transient fire source. Area ratios (one for an individual riser and one for a group of risers) were used to partition transient fire source frequencies. Additional exposure factors allowed in FIVE and fire suppression were not credited. Total screening CDFs for all four areas are 5.17E-7/yr and 7.24E-6/yr for units 2 and 3, respectively. Licensee argues that application of severity factor and credit for manual suppression would reduce Unit 3 CDF for these areas to 3.36E-7/yr.

Cable tunnel was not reevaluated because licensee claims the existing evaluation considered a number of scenarios at different locations and thus bounds the potential CDF from transient fire sources. For the Auxiliary Electric Equipment Room (AEER) analysis, the licensee credited automatic fire suppression.

B.1.3 Assessment of Response

The transient fire frequencies used in the evaluation look reasonable. The floor ratios were also reasonable. The evaluation of four of the areas failed to consider the impact of hot gas layer formation. Thus, damage was always limited to the cables in a single riser or a group of risers located within 6 feet of each other. This was offset by some measure by the fact that no fire suppression was credited. It is thus difficult to conclude that the values presented are reasonable upper bounds. Examination of the information provided indicate that the total CDFs for these 4 areas could be on the order of 1E-6/yr for both units.

The existing cable spreading room evaluation appears to bound the contribution from transient fire sources. The AEER evaluation appears to have one weakness. The welding fire frequency is reduced by a factor of 5E-2 to account for failure to follow procedures for minimizing the ignition of combustible materials. This factor is probably inherent in the welding fire frequency and thus some double accounting may have occurred. However, a high probability for failure of the fire watch to suppress the fire probably counteracts this. The bounding CDF for this area is thus probably reasonable.

Overall, the SRAI response is adequate. However, the failure to consider hot gas layer is a weakness in the analysis of the four areas with vertical risers.

B.2 SRAI #2: Analysis of cable tunnel.

B.2.1 Synopsis of SRAI

Since the cable tunnel at Dresden is directly below the turbine building, there was a concern that the fire risk assessment did not consider the potential for large oil fires damaging this area. In addition, the licensee was requested to address why no analysis was presented for the Unit 2 cable tunnel.

B.2.2 Synopsis of Licensee Response

The response indicated that there is only one cable tunnel that extends the entire length of both units. Since there are no important cables for Unit 2 located in the tunnel, it screened for Unit 2. With regard to turbine building oil fires, the licensee indicated that each access to the cable tunnel is surrounded by a 3 inch high curb and is covered by steel plates (not completely sealed) that the licensee states would preclude the propagation of an oil spill.

B.2.3 Assessment of Response

The licensee resolved the question about the Unit 2 cable tunnel and provided a reasonable argument against the potential spread of an oil spill from the turbine building to the cable tunnel. The SRAI response is adequate.

B.3 SRAI #3: Fire propagation outside panels.

B.3.1 Synopsis of SRAI

The licensee assumed fires would not propagate outside panels that are "substantially sealed." The licensee was requested to analyze the risk resulting from panel-to-panel fire spread and damage to overhead cables for fires involving two groups of 3 panels in the AEER.

B.3.2 Synopsis of Licensee Response

The licensee stated that each cabinet in the AEER was examined for the potential for fire spread to other panels and to overhead cables. With regard to the 6 panels mentioned in the SRAI, the licensee provided a diagram that indicates there is physical separation (ranging between several inches to several feet) between the panels.

B.3.3 Assessment of Response

Although the licensee did not provide all the requested information (e.g., panel contents and proximity of cables to each panel), they did provide the requested information concerning the separation of the 6 panels in question. The implied answer to the risk implications of panel-to-panel propagation is that there is none. The licensee also indicated that in several instances, panel fires were assumed to damage overhead cables. The licensee response is less than desired but addressed the main question.

Attachment 3

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)

TECHNICAL EVALUATION REPORT

HIGH WINDS, FLOODS, AND OTHER (HFO) EXTERNAL EVENTS

Brad Hardin, USNRC
May 7, 2001

Dresden Nuclear Power Station, Units 2 and 3
IPEEE Technical Evaluation Report
High Winds, Floods and
Other (HFO) External Events

1.0 Introduction

Dresden consists of Unit 2 and Unit 3 with each unit having a General Electric boiling water reactor (BWR) with a MARK 1 containment. Dresden is currently owned and operated by Exelon Corporation (previous owner: Commonwealth Edison Company (ComEd)). The site is located in northeastern Illinois near the town of Morris in the county of Grundy. Each unit is rated at 2,527 MW(t). Unit 2 received its construction permit (CP) in 1966 and its operating license (OL) in 1970, whereas Unit 3 received its CP in 1966 and its OL in 1971.

Regarding the Dresden HFO analyses, ComEd used the progressive screening approach recommended in Supplement 4 to Generic Letter 88-20 and NUREG-1407. Major information resources used for Dresden HFOs were NUREG-0823, *Integrated Plant Safety Assessment Report (IPSAR) for Dresden Unit 2*, performed as part of the NRC's *Systematic Evaluation Program* (SEP) in 1983, the updated final safety analysis report (UFSAR), and the Dresden *Individual Plant Examination* (IPE). ComEd performed a review of the Dresden plant and found that there were no other plant-unique external events that would pose a significant threat of severe accidents within the context of the NUREG-1407 screening approach.

2.0 High Winds

ComEd provided discussions related to the high wind and tornado design at the Dresden site in its IPEEE submittal. The wind design criterion for Dresden structures as stated in its UFSAR was to have the structures capable to withstand a wind loading of 110 mph which was considered by NRC staff as appropriate in NRC's SEP evaluation (Supplement 1 to NUREG-0823) for the Dresden site. The tornado design basis for Dresden structural elements, as reported in NUREG-0823, was to have load capacities of 160 mph or greater; and the mean frequency of exceeding a tornado wind speed of 160 mph was approximately 3E-5/yr. ComEd stated that the frequency of the unit 2/3 chimney (ventilation stack) collapsing and causing damage to safety-related equipment was reported in NUREG-0823 to be less than 1E-6/yr. ComEd also stated that the results of the SEP and subsequent tornado missile evaluations, including tornado missile probabilistic analyses for a variety of targets, were summarized in the USFAR Section 3.5.4 indicating that safety systems were considered either adequately protected from the effects of tornado missiles or the tornado missile striking frequencies for those targets were on the order of 1E-7/yr. In addition, ComEd discussed the concern of a tornado causing a dual unit station blackout and concluded that such an event was subsumed in the IPE results for a dual unit loss of off-site power event.

3.0 External Floods

ComEd identified four SEP evaluations relating to external flooding as discussed in their IPEEE Submittal Section 5.2.2.

For SEP Topic II-3.B, "Flooding Potential and Protection Requirements," ComEd performed evaluations to assess the effects of local intense precipitation. As a result, scuppers were installed in the roof parapets of the turbine building, reactor building, and the crib house to deal with the roof ponding from the probable maximum precipitation (PMP). The local drainage configuration was considered adequate to protect the plant from the localized PMP.

The evaluations of SEP Topic II-3.B.1, "Capability of Operating Plants to Cope with Design-Basis Flooding Condition," and Topic II-3.C, "Safety-Related Water Supply (Ultimate Heat Sink)," led ComEd to revise the flood emergency plan related to flood protection procedure.

The evaluation of SEP Topic II-3.A, "Effects of High Water Level on Structures," led NRC to conclude in NUREG-0823 that Dresden 2 flood design criteria met "current (1982) criteria" and was acceptable. ComEd provided an additional discussion related to the effects of high water levels on the plant and the associated safe shutdown requirement (technical specification 3/4.8.E). NRC stated in a recent (1995) review of the new Technical Specification 3/4.8.E that "The proposed requirements are applicable to the Dresden ... plant design and provide an adequate level of protection for plant flood protection."

4.0 Transportation and Nearby Facility Accidents

ComEd stated that transportation accidents, toxic hazards, and explosive hazards were insignificant risk contributors for Dresden based on the evaluation performed under SEP Topic II-1.C, "Potential Hazards or Changes in Potential Hazards due to Transportation, Institutional, Industrial, and Military Facilities." Rail transportation, barge transportation, pipeline transportation, chemical release, nearby industrial, nearby military, aircraft hazards, and on-site hazardous material accidents were evaluated under that topic. NRC concluded in NUREG-0823 that Dresden "meets current (1983) criteria..."

ComEd stated that no significant changes were found to impact the plant's original design conditions. The licensee did not identify any potential vulnerabilities associated with HFO events.

5.0 Other External Events

ComEd stated that no other plant-unique external event is known that poses any significant threat of severe accident. Lightning strikes, severe temperature transients, external fires, meteor strikes, volcanic activity and abrasive windstorms were included in the other events that were considered by the licensee.

6.0 GENERIC SAFETY ISSUE (GSI) RESOLUTION

GSI-103, "Design for Probable Maximum Precipitation"

ComEd performed evaluations to assess the effects of PMP. As a result, the licensee installed scuppers in the roof parapets of the turbine building, reactor building, and the crib house to deal with the roof ponding from the PMP. The licensee considered its local drainage configuration as adequate to protect the plant from the PMP. The evaluation of GSI-103 was addressed by the licensee in Section 5.2.2 of the IPEEE submittal. Overall, the staff finds that the licensee's GSI-103 evaluation is consistent with the guidance provided in Section 6.2.2.3 of NUREG-1407 and, therefore, the staff considers this issue resolved.

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

ComEd performed specific SEP evaluations, and the NRC documented its review conclusions in NUREG-0823 indicating that the licensee's SEP was acceptable. The following external-event-related SEP issues were addressed during the review: settlement of foundations and buried equipment (not required for plants on rock sites); dam integrity and site flooding; site hydrology and ability to withstand floods; industrial hazards; tornado missiles; severe weather effects on structures; design codes, criteria and load combinations; and seismic design of structures, systems and components. Based on the results of the IPEEE submittal review, the staff considers the HFO-related GSI-156 issues resolved.

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

With respect to effects of flooding and/or moisture intrusion on non-safety related and safety-related equipment, the licensee provided a discussion in Section 4.11.2 of the IPEEE submittal. Based on the results of the staff's IPEEE submittal review, the staff considers that the licensee's process is capable of identifying potential vulnerabilities associated with this issue. On the basis that no potential vulnerability associated with this issue was identified in the IPEEE submittal, the staff considers the HFO-related aspects of this issue resolved.

7.0 Conclusions

The IPEEE submittal is judged to meet the intent of Supplement 4 to Generic Letter 88-20 for the high winds, floods, transportation, and other external events. The licensee found no vulnerabilities with respect to HFO events. No plant improvements were identified in the HFO events areas that were a direct result of the IPEEE. However, two improvements that were related to HFO events were cited as resulting from the Systematic Evaluation Program (SEP) prior to the IPEEE. These were the addition of scuppers to aid in draining water from roofs during heavy precipitation and revisions made to the site flood emergency plan.