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

February 29, 2000

Mr. Otto L. Maynard
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KA 66839

SUBJECT: REVIEW OF INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR WOLF CREEK GENERATING STATION (TAC NO. M83696)

Dear Mr. Maynard:

On June 28, 1991, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," Supplement 4 (with NUREG-1407, Procedural and Submittal Guidance), requesting licensees to perform individual plant examinations of external events (IPEEE) to identify plant-specific vulnerabilities to severe accidents from external events and to report the results to the Commission together with any licensee-determined improvements and corrective actions. In the letters dated December 24, 1991 (ET 91-0235), and September 18, 1992 (WM 92-0151), you provided the schedule to respond to GL 88-20, and in your submittal of December 24, 1991, you also provided the selected IPEEE methods for the Wolf Creek Generating Station (WCGS). We concluded that these methods were acceptable in our letter of July 10, 1992.

You submitted on June 27, 1995 (ET 95-0055), the results of the IPEEE review for WCGS in response to GL 88-20. In your letters of December 8, 1997 (WM 97-0142) and June 11, 1999 (ET 99-0028), you responded to the staff's requests for additional information (RAIs) of October 8, 1997, and April 6, 1999.

The staff's findings of the WCGS IPEEE submittals are summarized in the enclosed staff evaluation report (SER) (Enclosure 1). The staff used a contractor to review the submittals and the details of the contractor's findings are in the enclosed technical evaluation report (TER) (Enclosure 2).

Based on our review of your submittals, the staff concludes that (1) the WCGS IPEEE is complete with regard to the information requested by Supplement 4 to GL 88-20, including NUREG-1407, (2) the IPEEE results are reasonable given the design, operation, and history of WCGS, and (3) the IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities for external events for WCGS. Based on this, the staff also concludes that the WCGS IPEEE has met the intent of Supplement 4 to GL 88-20.

Although NUREG-1407 placed WCGS in the focused-scope bin for seismic evaluation, you elected to perform a reduced-scope assessment. This was addressed previously in your letter of May 20, 1994 (WM 94-0081), in the letter from the Nuclear Energy Institute of April 5, 1994, to Dr. Ashok Thadani, and in the staff's letter of August 15, 1994. The principal justification given for your decision was that the Lawrence Livermore National Laboratory (LLNL) seismic hazard curves provided a mean hazard at the WCGS site equivalent to that of some of the sites

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originally placed in the reduce-scope bin in the NUREG. You provided additional supporting information in response to staff RAIs regarding this issue. Based on the staff's and contractor's review of the information provided, the staff concluded that WCGS could be assigned to the reduced-scope category for IPEEE purposes. Consistent with NRC's guidance for reduced-scope plants, you did not provide seismic capacity calculations and did not estimate the seismic core damage frequency (CDF); however, you did perform an IPEEE seismic walkdown consistent with Electric Power Research Institute's (EPRI's) seismic margin analysis.

A probabilistic risk assessment (PRA) quantification for fire events that used the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology was performed in which the contribution to plant CDF from fires was estimated in your submittals to be 7.5E-6/reactor year (RY). The other external events (HFO) were evaluated using the progressive screening approach from NUREG-1407 and GL 88-20, Supplement 4. Because WCGS was designed to the 1975 standard review plan (SRP) criteria, the HFO evaluation focused on any changes in the plant design or operation that occurred since the plant's operating license was issued. The contribution to CDF from HFO was not estimated because it was stated in your submittals that no significant plant changes had occurred since licensing, and the contribution to CDF was considered in your submittals to be negligible. The overall CDF due to internal events was also stated to be about 4E-5/RY.

Your staff used the severe accident closure guidelines in Nuclear Energy Institute (NEI) 91-04 to identify IPEEE vulnerabilities. Based on these guidelines, no potential vulnerabilities associated with seismic, fire or HFO events were identified in your submittals and it was concluded that no improvements related to external events were needed for WCGS. The staff does not disagree with this conclusion.

A number of minor corrections were identified by you in the June 27, 1995, submittal, which was a result of the IPEEE seismic walkdowns. These corrections have been completed.

The WCGS IPEEE addressed generic safety issues (GSIs) GSI-131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants," GSI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment," GSI-103, "Design for Probable Maximum Precipitation," unresolved safety issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," and the Sandia Fire Risk Scoping Study issues which were explicitly requested in Supplement 4 to GL 88-20 and its associated guidance in NUREG-1407. Because the evaluations for WCGS are consistent with the guidance in NUREG-1407 or EPRI's FIVE and the results are acceptable, the staff considers these issues resolved.

In addition, the WCGS IPEEE contained specific information that addressed the external event aspects of the following safety issues: GSI-147, "Fire-Induced Alternate Shutdown/Control Room Panel Interactions," GSI-148, "Smoke Control and Manual Fire-Fighting Effectiveness," and GSI-172, "Multiple System Responses Program (MSRP)." The specific information associated with each of these issues is also discussed in the enclosed SER. Because no vulnerabilities associated with the external event aspects of these issues were identified, the staff also considers these issues resolved for WCGS.

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If you have any questions regarding the attached SER, please contact me at 301-415-1307 or at jnd@nrc.gov on the Internet.

Sincerely,

/RA/

Jack Donohew, Senior Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosure: Staff Evaluation Report
w/Technical Evaluation Report

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

STAFF EVALUATION REPORT

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

On June 28, 1991, the NRC issued Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," Supplement 4 (with NUREG-1407, Procedural and Submittal Guidance) requesting all licensees to perform individual plant examinations of external events (IPEEE) to identify plant-specific vulnerabilities to severe accidents and to report the results to the Commission together with any licensee-determined improvements and corrective actions. Wolf Creek Operating Corporation (the licensee) responded to the GL for Wolf Creek Nuclear Generating Station (WCGS) in letters dated December 24, 1991; September 18, 1992; June 27, 1995; December 8, 1997; and June 11, 1999.

In its letters dated December 24, 1991, and September 18, 1992, the licensee provided the schedule to respond to the GL and the selected IPEEE methods for WCGS. The staff concluded that these methods were acceptable in its letter of July 10, 1992. The licensee then submitted the results of the IPEEE review for WCGS in response to the GL in its letter of June 27, 1995 (the licensee's IPEEE submittal). The licensee's additional letters of December 8, 1997, and June 11, 1999, were its response to the staff's requests for additional information (RAIs) in letters of October 8, 1997, and April 6, 1999.

The staff contracted with Energy Research, Inc. to conduct a completeness and reasonableness "Step 1" review of the licensee's IPEEE submittal and sent an RAI to the licensee in October 1997. The licensee responded to the RAI in December 1997. As a result of the review of the licensee's initial RAI response, the staff concluded that additional information was needed to complete its review and a second RAI was sent in April 1999. The licensee responded to the second RAI in June 1999. The details of the contractor's findings are in the technical evaluation report (TER) attached to this staff evaluation report (SER).

Based on the results of the review of the licensee's submittals by the contractor and the staff, the staff concluded that the aspects of seismic, fires, and high winds, floods, transportation, and other external events were adequately addressed. The review findings are summarized in the evaluation section below.

In accordance with Supplement 4 to GL 88-20, the licensee provided information to address the resolution of the following:

- Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements,"
- Generic Safety Issue (GSI) -103, "Design for Probable Maximum Precipitation (PMP),"
- GSI-131, "Potential Seismic Interaction Involving Movable In-Core Flux Mapping System Used in Westinghouse Plants,"
- GSI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment," and
- 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. Staff and contractor review findings regarding these issues are included in this SER. The licensee did not propose to resolve any additional USIs or GSIs as part of the Wolf Creek IPEEE.

An IPEEE Senior Review Board (SRB) was established within NRC for these reviews 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 perspective 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.

2.0 EVALUATION

WCGS is a single unit Westinghouse four-loop pressurized-water reactor (PWR) with a licensed thermal power output of 3565 MWt. It is a sister unit to the Callaway Plant in Callaway County, Missouri. The WCGS plant site is in Coffey County approximately 3.5 miles northeast of Burlington, Kansas, and about 75 miles southwest of Kansas City. The main station buildings are located on a terrace above the flood plain, about 3.5 miles east of the Neosho River and the main dam at John Redmond Reservoir. The ultimate heat sink is a protected cooling pond. WCGS was given its full power license on June 4, 1985.

The contribution to plant core damage frequency (CDF) from seismic events was not estimated by the licensee because a reduced-scope seismic assessment was performed. Although NUREG-1407 placed WCGS in the focused-scope bin for seismic evaluation, the licensee elected to perform a reduced-scope assessment. The principle justification given for the licensee's decision was that the Lawrence Livermore National Laboratory (LLNL) seismic hazard curves provided a mean hazard at the WCGS site equivalent to that of some of the sites originally placed in the reduced-scope bin in NUREG-1407. The licensee provided additional supporting information in response to staff RAIs regarding this issue. Based on the staff's and contractor's review of the information provided, the staff concluded that WCGS could be assigned to the reduced scope category for IPEEE purposes. Therefore, consistent with NRC's reduced guidance for reduced-scope plants, the licensee did not provide any results of seismic capacity calculations or a seismic CDF estimate. The licensee, however, did perform an IPEEE seismic walkdown consistent with EPRI's seismic margin analysis.

A PRA quantification for fire events was performed by the licensee that utilized the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology. The licensee evaluated the other external events (HFO) using the progressive screening approach from NUREG-1407 and Supplement 4 to GL 88-20. WCGS was designed to the 1975 standard review plan (SRP) criteria, and hence, the HFO evaluation focused on any changes in the plant design or operation that occurred since the plant's operating license was issued.

The licensee used the severe accident closure guidelines in Nuclear Energy Institute (NEI) 91-04 to identify IPEEE vulnerabilities. Based on these guidelines, the licensee did not identify any potential vulnerabilities associated with seismic, fire or HFO events; thus no improvements related to external events were considered necessary by the licensee for WCGS.

2.1 Core Damage Frequency Estimates

As noted above for reduced-scope plants, the licensee did not estimate a seismic CDF contribution because a reduced-scope assessment was performed. For this assessment, a detailed walkdown was performed in which components were screened using an overall high confidence of low probability of failure (HCLPF) capacity of 0.3g, the review level earthquake (RLE) value for the plant, and the screening level that would be used for a focused-scope plant. A judgment was made that certain other components (including four batteries, 12 cabinets for the engineered safety features actuation system, strainers and screens) could not be screened at the RLE level, and for these components a HCLPF capacity of 0.2g was assigned which is the safe shutdown earthquake (SSE) level. The use of this level is consistent for reduced-scope plant seismic analyses. A quantification for fire events, that utilized the EPRI FIVE methodology, indicated that the contribution to plant CDF from fire was 7.5E-6/RY. The CDF contribution from HFO events was considered to be negligible. In earlier studies, the licensee estimated that the overall CDF due to internal events was about 4E-5/RY.

2.2 Dominant Contributors

As stated above, the licensee's analysis indicated that the overall HCLPF plant capacity was at least equal to the RLE (0.3g) except for certain batteries, cabinets, strainers and screens and that there were no weak components. For fire events, the licensee reported that the main contributors to the fire-related CDF were the control room and the two emergency safety features switchgear rooms that comprise about 90 percent of the total fire CDF. The contribution to CDF from the HFO-related events was not quantitatively estimated since this contribution was judged to be insignificant.

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

2.3 Containment Performance

The licensee has assessed containment performance under seismic conditions at WCGS by investigating (1) containment integrity, specifically focusing on the ruggedness of containment isolation equipment to protect against containment bypass (including the potential for a unique seismic-caused containment bypass), and (2) containment cooling systems, during seismic events. The licensee evaluated fires both inside and outside of containment and found them to be of little significance primarily because of the reactor coolant pump oil collection design, independence of penetrations and hatches from active systems, protection of high-to-low pressure boundaries, and the separation of redundant electrical cables. None of the fires outside of containment were found to adversely affect the containment functions.

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

2.4 Generic Safety Issues

As a part of the IPEEE, a set of unresolved and generic safety issues (U/GSIs) (i.e., USI A-45, GSI-131, GSI-103, GSI-57, and the Sandia 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 plant-specific IPEEE. These safety issues were evaluated by the NRC's contractor, and the results of these evaluations are contained in the attached TER. For those safety issues that were not completely resolved by the contractor, the NRC staff performed additional reviews. The final resolution of these issues is provided below.

2.4.1 USI A-45, "Shutdown Decay Heat Removal Requirements"

This issue was addressed in Section 3.7 of the WCGS IPEEE submittal for the seismic area and in Section 4.3.1 for fires. During the seismic margins assessment, the licensee evaluated the primary and alternate success paths which included decay heat removal as one of the safety functions required to mitigate the effects of a seismic event. The licensee stated in the submittal that many of the decay heat systems have low failure probabilities and that several of these systems would have to fail in combination to have a significant impact on system performance.

Regarding the fire aspects of USI A-45, the licensee used the same model in its IPEEE evaluation as was used in the individual plant evaluation (IPE) for GL 88-20, which included the entire array of heat removal capabilities for the plant. The licensee reported that, for all of the fire scenarios, at least two trains of auxiliary feedwater, or one train of auxiliary feedwater along with main feedwater will be unaffected by the fire.

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 the licensee's results are not unacceptable; therefore, the staff considers this issue resolved.

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

The licensee addressed this issue in Section 3.5.9.5 of its IPEEE submittal. The licensee reported that a seismic analysis pertaining to the interaction of the movable in-core flux mapping system indicated that the restraints for the flux mapping system were determined to be adequate considering that angles had been added (welded) to the wide flange beams to provide lateral support. The staff finds that the licensee's GSI-131 evaluation is consistent with the guidance provided in Section 6.2.2.1 of NUREG-1407 and the licensee's results are not unacceptable; therefore, the staff considers this issue resolved.

2.4.3 GI-103, "Design for Probable Maximum Precipitation"

In Section 5.2.1 of its IPEEE submittal, the licensee reported that the new probable maximum precipitation (PMP) criteria will not have any adverse impact on Wolf Creek due to the roof design and the ground drainage system. 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 the licensee's results are not unacceptable; therefore, the staff considers this issue resolved.

2.4.4 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 also one of the issues identified in the FRSS. The IPEEE submittal addresses this issue in Section 4.8.2. The licensee stated that this issue was addressed as part of the plant design and considered in the Appendix R program, and the licensee did not identify any improvements for WCGS. The staff finds that the licensee's GSI-57 evaluation is consistent with the guidance provided in EPRI's Fire-Induced Vulnerability Evaluation (FIVE) which was accepted by the NRC staff and the licensee's results are not unacceptable; therefore, the staff considers this issue resolved.

2.4.5 Fire Risk Scoping Study Issues

In Section 4.8.2 of its IPEEE submittal, the licensee has explicitly addressed the FRSS issues. The licensee states in its submittal that it has not identified any unacceptable risks or outliers at WCGS due to the FRSS issues. The staff finds that the licensee's evaluation is consistent with the guidance provided in NUREG-1407 and the licensee's results are not unacceptable; therefore, the staff considers these issues resolved.

2.5 Other Generic Safety Issues

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 GL, the NRC staff evaluated the scope and specific information requested in the GL 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 contractor, and the results of these evaluations are contained in the attached TER. For those GSIs that were not completely resolved by the NRC's contractor, the NRC staff performed additional reviews. The final resolution of these issues is provided below.

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

The licensee's IPEEE submittal contains a discussion addressing this issue in Sections 4.6.7.3.1 and 4.8.2.5 on FRSS issues. In the discussion, the licensee stated that WCGS has a safe shutdown facility that is independent of the control room and auxiliary shutdown panel. 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.

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

The licensee's IPEEE submittal contains information addressing this issue in Sections 4.6.7 and 4.8.2.4. The licensee addressed this issue and concluded that the WCGS fire protection systems and procedures provide adequate assurance that manual fire fighting effectiveness will not be significantly degraded from smoke and other fire effects. 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 WCGS.

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

The plant is not an SEP plant.

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

The licensee's IPEEE submittal contains information directly addressing the following external-event-related MSRP issues:

- Effects of fire protection system actuation on safety-related and non safety-related equipment (Section 4.8.2 of the submittal);
- Seismically induced fire suppression system actuation (Section 4.8.2);
- Seismically induced fires (Section 4.8.2);
- Effects of hydrogen line rupture (Section 4.8.2);
- IPEEE-related aspects of common cause failures associated with human errors, recovery actions for certain seismic scenarios (RAI response dated December 8, 1997);
- IPEEE-related aspects of common cause failures associated with human errors, recovery actions for fire scenarios (Section 4.3.5);
- Non safety-related control system/safety-related system dependencies (Sections 3.2.1., 3.5, 4.6.7.3.1, and 4.8.2);

- Effects of flooding and/or moisture intrusion on non safety-related and safety-related equipment (Sections 4.2.2.2 and 4.8.2);
- Seismically induced spatial interactions (Sections 3.2.1 and 3.5);
- Seismically induced flooding (Section 4.8.2);
- Seismically induced relay chatter (Section 3.3.3.2); and
- Evaluation of earthquake magnitude greater than safe shutdown earthquake (Section 3.2.2).

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

No other specific USIs or GSIs were proposed by the licensee for resolution as part of the Wolf Creek IPEEE.

2.6 Unique Plant Features, Potential Vulnerabilities, and Improvements

In Section 7 of its IPEEE submittal, the licensee reported a number of unique safety features at WCGS. Several features are reported to enhance the ability to withstand a seismic event (e.g., most large components (pumps) are located low in the structures to minimize amplifications of seismic ground motion). Regarding fire, the plant is reported to be extensively compartmentalized, and the control room has two separate cable spreading rooms.

The licensee stated that it utilized guidance given in EPRI documents NP-6041-SL and NP-6041 in its evaluation of seismic vulnerabilities and document EPRI-TR-100370 in evaluating fire vulnerabilities using the EPRI FIVE approach. In Section 7 of its IPEEE submittal, the licensee stated that no fundamental weaknesses or vulnerabilities with regard to external events were identified during their evaluation. This section did, however, discuss seven corrections that have been implemented or were planned in the seismic area as a result of the walkdowns which were done as a part of the IPEEE review. These corrections were identified by the licensee as miscellaneous housekeeping or equipment issues and are listed in Section 7 of its IPEEE submittal. The corrections include the following:

- Furniture or equipment found in the control room which could interact with nearby safety-related equipment,
- Replacement of two missing bolts on the transformer of an inverter,
- Tightening of loose screws, and
- Replacement of some missing shim plates.

The licensee informed the staff that these miscellaneous issues been completed (NRC ADAMS accession ML003686707).

The licensee did not identify any vulnerabilities or improvements in the fire area as a result of the IPEEE evaluation. Also, no vulnerabilities or needed enhancements were identified in the HFO area.

3.0 CONCLUSIONS

On the basis of the above findings, the staff concludes that: (1) the licensee's IPEEE for WCGS 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 Wolf Creek design, operation, and history. This conclusion is based on the findings as presented in the attached TER (and reviewed by the staff) and the additional reviews conducted by the staff. Based on this, the staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, the Wolf Creek IPEEE has met the intent of Supplement 4 to GL 88-20 and has resolved the specific generic safety issues discussed in this SER.

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

Attachment: Technical Evaluation Report

Principal Contributors: William Hardin
Alan Rubin

Date: February 29, 2000

ERI/NRC 97-504

TECHNICAL EVALUATION REPORT ON THE "SUBMITTAL-ONLY" REVIEW OF THE INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS AT THE WOLF CREEK GENERATING STATION

Completed: May 1998
Final: December 1999

M. Khatib-Rahbar
Principal Investigator

Author:

Michael V. Frank[1]

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

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EXECUTIVE SUMMARY

This preliminary technical evaluation report (TER) documents a "submittal-only" review of the individual plant examination of external events (IPEEE) conducted for the Wolf Creek Generating Electric Station (WCGS). Energy Research, Inc. (ERI) performed the review on behalf of the U.S. Nuclear Regulatory Commission (NRC). The submittal-only review process consists of the following tasks:

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

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

Wolf Creek Nuclear Operating Corporation (WCNOC) conducted the WCGS IPEEE with contractor assistance. The WCGS IPEEE submittal considers seismic; fire; and high winds, floods and other initiators for the external events analysis. The seismic IPEEE process was based on a reduced-scope seismic margin assessment (SMA); the fire IPEEE used the Electric Power Research Institute's (EPRI's) fire-induced vulnerability evaluation (FIVE) methodology for screening fire areas, and then a probabilistic risk analysis (PRA) of the most risk significant compartments; and high winds, floods and other (HFO) events were evaluated using the screening approach discussed in NUREG-1407 and GL 88-20, Supplement 4.

Licensee's IPEEE Process

Although NUREG-1407 has placed the WCGS in the focused-scope bin for seismic evaluation, the licensee elected to perform a reduced-scope assessment. The principle justification for this decision is that the new Lawrence Livermore National Laboratory (LLNL) hazard curves provide a mean hazard at the Wolf Creek site equivalent to that of some of the sites originally placed in the reduced-scope bin. Additional justification includes work funded by NEI which placed WCGS near the bottom with respect to mean probability of exceeding the SSE, which is 0.12g at this site and 0.2g on the power block. The licensee further argued that the most important part of the seismic IPEEE, namely the walkdown, was performed as a focused scope plant. Furthermore, additional work would not be cost effective because no additional seismic insight would be obtained. Subsequent independent reevaluation by the NRC concluded that the Wolf Creek plant can be assigned to the reduced bin category for IPEEE purposes.

The essential element of the WCGS IPEEE was the walkdown which was conducted in a manner consistent with an EPRI seismic margin analysis using a review level earthquake (RLE)of 0.3g. This is consistent with the requirements of a focused scope assessment. The guidance provided in Tables 2-3 and 2-4 of EPRI NP-6041, along with the judgment of the seismic review team (SRT), was relied upon , to screen out all structures and components at the SSE level. Consistent with reduced scope requirements, no seismic capacity calculations were performed except for four bounding anchorage calculations to assist in the walkdown. The following steps of an EPRI seismic margin assessment were undertaken for the reduced-scope seismic evaluation of WCGS:

- Selection of the seismic margin earthquake (SME)
- Selection of the SRT
- Preparatory work prior to walkdown
- Systems and element selection (development of the safe shutdown equipment list)
- Seismic capability walkdown
- Relay chatter evaluation
- Containment performance analysis
- Consideration of Unresolved Safety Issue (USI) A-45, Generic Issue (GI)-131, USI A-17, USI A-40, and Eastern U.S. Seismicity Issue
- Peer review
- Documentation

A soil failure assessment was not performed consistent with both the reduced scope requirements and the fact that Wolf Creek is a rock site.

The licensee has conducted an extensive and detailed analysis of potential fire events at WCGS. Starting with 133 Appendix R defined fire areas, a three phase FIVE screening process left eight compartments unscreened. The screening process included two separate methods for investigating the potential for inter-compartment fire propagation. COMPBRN was used, during the screening analysis, in a conservative manner. The FIVE database, guidance and ignition source data sheets were used to determine the fire ignition frequencies. The plant has been walked down several times to support the analysis and help resolve the Sandia fire risk scoping study (FRSS) issues.

The following steps were undertaken for the fire IPEEE of the WCGS:

- Identification of Appendix R fire areas and compartments, including inter-compartment propagation
- Qualitative screening
- Quantitative screening based on fire ignition frequency and complete damage to a compartment
- Quantitative screening based on COMPBRN IIIe to address the extent of potential damage
- Scenario-by-scenario PRA of unscreened compartments
- Containment performance review
- Plant walkdown
- Discussion of Sandia fire risk scoping study issues per FIVE guidance
- Resolution of USI A-45 and GI-57

The fire PRA of unscreened compartments was performed conservatively, but with a less rigorous approach than the screening study. Fire scenarios were developed for each compartment and each fire ignition source.

The WCGS HFO events analysis was performed in a manner consistent with the progressive screening method suggested in NUREG-1407. The examination process involved: (1) a review of plant-specific hazard data and plant licensing-basis information; (2) identification of changes to the plant and vicinity since issuance of the operating license; (3) walkdowns to validate the documentation review and help identify changes to the plant; and (4) comparison of the current as-built unit with the current Standard Review Plan (SRP) criteria.

Key IPEEE Findings

All equipment and structures were screened out with a high confidence of low probability of failure (HCLPF) capacity of more than the 0.3g review level earthquake (RLE), except for the following, each of which were assigned a HCLPF of 0.2g based on review of analyses performed for the FSAR:

- Refueling water storage tank (RWST)
- Turbine building
- Four 60-cell batteries and racks
- Twelve load shed/emergency load sequencer (LSELS)/engineered safety features actuation system (ESFAS) cabinets
- Strainers and screens

The submittal states that more detailed analysis could demonstrate that the above equipment have a capacity greater than 0.3g. The walkdown resulted in several minor work requests for adjustment of clearances, adjustment of bolts, and addition of shims.

Concerning fire events, eight compartments survived the FIVE screening and were subjected to detailed PRA. The compartments are: control room, north and south emergency safety feature (ESF) switchgear rooms, a portion of the auxiliary building general area at elevation 2000, a portion of the auxiliary building general area at elevation 2026, north and south electrical penetration rooms, and reactor trip switchgear room. The fire core damage frequency (CDF) of unscreened compartments was estimated to be 7.5×10^{-6} per reactor year. The main contributors to the fire CDF are the control room and the two ESF switchgear rooms. These compartments comprise about 82% of the total fire CDF. Each of the other unscreened compartments was far less significant to the overall fire risk. Two methods of evaluating inter-compartment fires assure that this possibility was adequately considered.

The HFO evaluation confirmed that the plant continues to conform to the SRP criteria, including the assessment of changes since the issuance of the operating license.

Generic Issues and Unresolved Safety Issues

For seismic events, the WCGS IPEEE specifically addressed the following safety issues: USI A-45, "Shutdown Decay Heat Removal Requirements;" GI-131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants;" seismic-fire interaction aspects of the Sandia FRSS issues; USI A-17, "System Interactions in Nuclear Power Plants;" and USI A-40, "Seismic Design Criteria." Resolution of these issues to the satisfaction of GL 88-20, Supplement 4 and NUREG-1407 awaits the decision on whether the WCGS is to be treated as a focused or reduced scope unit.

For the fire IPEEE, all of the Sandia FRSS issues and USI A-45 issues have been addressed and resolved.

The submittal discusses the effects of the probable maximum precipitation (PMP) on the plant (GI-103), and concludes that short-term on-site flood levels or roof ponding would not adversely affect safe operation.

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

Vulnerabilities and Plant Improvements

The submittal concludes that, relative to the SSE, WCGS equipment and structures are "seismically very rugged" because of a conservative design, and the plant "has minimal vulnerability to a seismic event." No vulnerabilities were identified as part of the IPEEE process. The licensee argued that because the WCGS design basis ground spectra are very close to the site specific 0.3g SME ground spectra and floor response spectra follow the results of the ground spectra, then the "seismic response spectra to be used for a site specific SME (0.3g) evaluation would be very similar to the seismic response spectra used for the WCGS design basis SSE." The licensee concludes as follows: "As a result of this considerable design margin, evaluation of a 0.3g SME at Wolf Creek would be nearly identical to evaluating the current design basis earthquake at WCGS." The small list of unscreened items relative to a hypothetical 0.3g RLE supports the licensee's contention at the SSE.

With respect to fires, the licensee has concluded "that the WCGS is not vulnerable to fire." The fire CDF is within the range of fire-induced CDFs obtained for other nuclear power plants. The submittal states that no plant modifications are required for adequate defense against fires.

The HFO evaluation, performed by the licensee, confirmed that the plant continues to conform to the SRP criteria. Accordingly, no vulnerabilities nor any corrective actions were reported.

Observations

The licensee performed a satisfactory reduced scope seismic IPEEE assessment with some features, notably the walkdown and a relay chatter evaluation, that are consistent with a focused scope assessment.

The walkdown was comprehensive and well documented, and the success paths were derived in a logical fashion as suggested in EPRI NP-6041. The walkdown was conducted against a 0.3g RLE screening criterion per a focused scope plant in order to demonstrate the seismic margin of the plant. Anchorage calculations were performed to help support screening judgments. Although a few items could not be screened out at the RLE, all equipment and structures were screened out during the walkdown at the SSE level. The study assigned a HCLPF capacity of 0.2g for the primary and alternate safe shutdown paths, with the caveat that more rigorous evaluation of the above components could allow the HCLPF to be increased to the RLE. Of significance is the fact that the WCGS design basis is a conservatively established 0.2g peak ground acceleration (PGA) for the power block, with a design basis seismic demand that exceeds, in many cases, the demands imposed by the RLE. Therefore, the plant might indeed have such margin.

With respect to fires, the licensee has performed an extensive and detailed fire screening study, followed by a fire probabilistic risk assessment. The licensee appears to be well aware of available methods, their uses, and their limitations. Assumptions, sensitivity studies and uncertainties are well presented. Accepted methods have been used and they have generally been applied using recognized data. The fire PRA of unscreened compartments was performed conservatively, but with a less rigorous approach than the screening study. The licensee demonstrated an understanding of fire propagation/damage methods. COMPBRN IIIe was used to address the extent of potential damage of each fire source, except for cabinets. Cabinet fires were treated by a series of assumptions, which appear reasonable. Although credit is taken for automatic and manual suppression, calculations of timing of damage versus suppression were not performed during the PRA of unscreened compartments.

One unusual finding was that hot shorts were an important contributor to core damage frequency. Upon further inquiry, however, this was attributed to a highly conservative approach which (1) generally assigned a probability of unity for a hot short type failure mode if it could occur at all, and (2) perhaps unrealistically included sustained hot shorts of motor-operated valves. Generally, the licensee demonstrated an in-depth knowledge of the fire characteristics of the plant, the fire risk database and state-of-the-art fire risk assessment techniques. It can be said that the licensee's fire IPEEE process is capable of identifying severe accident vulnerabilities, and none were found.

The HFO evaluation implemented the progressive screening method of NUREG-1407. The performance of verification walkdowns was appropriate. After identification of changes made since the operating license (OL), an evaluation to determine compliance with the current SRP criteria was performed. All aspects were found to comply. The screening methodology appeared to be followed correctly, and no significant assessment weaknesses were noted during this review.

PREFACE

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

Seismic; Fire; High Winds, Floods and Other External Events

M. V. Frank

Review Oversight, Coordination and Integration

M. Khatib-Rahbar, Principal Investigator/Review Coordination and Integration
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Mr. A. Kuritzky (formerly of Energy Research, Inc., now with the Nuclear Regulatory Commission) was the review coordinator for the draft version of this report.

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

ABBREVIATIONS

AFW	Auxiliary Feed Water
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CCW	Component Cooling Water
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Margin
CFR	Code of Federal Regulations
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
ERI	Energy Research, Inc.
ESF	Emergency Safety Features
ESW	Emergency Service Water
ESFAS	Engineered Safety Features Actuation System
FCIA	Fire Compartment Interaction Analysis
FIVE	Fire Induced Vulnerability Evaluation Method
FRSS	Fire Risk Scoping Study
FSAR	Final Safety Analysis Report
GI	Generic Issue
GL	Generic Letter
GSI	Generic Safety Issue
HCLPF	High Confidence of Low Probability of Failure (capacity)
HFO	High Winds, Floods and Other (external initiators)
HVAC	Heating, Ventilation, and Air Conditioning
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IRS	In-Structure Response Spectrum
LSELS	Load Shed/Emergency Load Sequencer
LLNL	Lawrence Livermore National Laboratory
LOCA	Loss of Coolant Accident
MCC	Motor Control Center
MOV	Motor-Operated Valve
MSRP	Multiple System Responses Program
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
NSSS	Nuclear Steam Supply System
OL	Operating License
PCCL	Probability of Critical Combustible Loading
PGA	Peak Ground Acceleration
P&ID	Piping and Instrument Diagrams
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PORV	Power-Operated Relief Valve

PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RLE	Review Level Earthquake
RPS	Reactor Protection System
RWST	Refueling Water Storage Tank
SCBA	Self-Contained Breathing Apparatus
SEWS	Seismic Evaluation Work Sheet
SI	Safety Injection
SLOCA	Small-break Loss of Coolant Accident
SMA	Seismic Margin Assessment
SME	Seismic Margin Earthquake
SMM	Seismic Margin Methodology
SNUPPS	Standardized Nuclear Unit Power Plant Sites
SPLD	Success Path Logic Diagram
SQUG	Seismic Qualification Utility Group
SRP	Standard Review Plan
SRT	Seismic Review Team
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
TER	Technical Evaluation Report
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
USI	Unresolved Safety Issue
WCGS	Wolf Creek Generating Electric Station
WCNOC	Wolf Creek Nuclear Operating Corporation

1 INTRODUCTION

This technical evaluation report (TER) documents the results of the "submittal-only" review of the individual plant examination of external events (IPEEE) for the Wolf Creek Generating Electric Station (WCGS), operated by the Wolf Creek Nuclear Operating Company (WCNOC) [1]. This technical evaluation review, conducted by Energy Research, Inc. (ERI), has considered various external initiators, including seismic events; fires; and high winds, floods, and other (HFO) external events.

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

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

The remainder of this section of the TER describes the plant configuration and presents an overview of the licensee's IPEEE process and insights, as well as the review process employed for evaluation of the seismic, fire, and HFO events sections of the WCGS IPEEE submittal. Sections 2.1 to 2.3 of this report present ERI's detailed findings related to the seismic, fire, and HFO event reviews, respectively. Section 2.4 identifies the location in the IPEEE submittal where information having potential relevance to generic safety issues (GSIs) 147, 148 and 172 may be found. Sections 3.1 to 3.3 summarize ERI's overall evaluation and conclusions from the seismic, fire, and HFO events reviews, respectively. Section 4 summarizes the licensee's IPEEE insights, improvements, and commitments. Section 5 includes completed IPEEE data summary and entry sheets. Finally, Section 6 provides a list of the references cited in the TER.

1.1 Plant Characterization

Wolf Creek is a single-unit nuclear power generating facility consisting of a 4-loop Westinghouse pressurized water reactor (PWR). Its licensed core power level is 3,565 MWt. The containment building at Wolf Creek is of the large dry variety, designed by Bechtel Power Corporation. The containment walls and dome are steel-lined, reinforced, post-tensioned concrete, founded on a containment base slab of reinforced, steel-lined concrete. The plant is located 3.5 miles northeast of Burlington in Coffey County, Kansas, which is about 75 miles southwest of Kansas City. The plant was licensed for full-power operation in June 1985, and it began commercial operation in September 1985.

Wolf Creek is a sister unit to the Callaway Plant; both are among three units (Callaway, Sterling, and Wolf Creek) that were planned for development under the Standardized Nuclear Unit Power Plant Sites (SNUPPS) concept, where power-block components are designed to demands determined from envelopes of individual demands for the three sites. Non-power-block structures and components are designed based on site-dependent demands. Power-block structures at Wolf Creek include the following: reactor containment building, auxiliary/control building, fuel buildings, and diesel generator building. Seismic Category I structures include the power-block, emergency service water pump house, refueling water storage tank and valve house, emergency fuel oil storage tanks, access vaults, and the ultimate heat sink pond. Seismic Category I structures are founded on shallow soil columns over bedrock. The soil overburden is less than 16 feet. Non-seismic Category I structures include the turbine building and radwaste building.

The safe shutdown earthquake (SSE) peak ground acceleration (PGA) for both horizontal and vertical motion is 0.2g for seismic Category I power block structures. An in-structure response spectrum, which enveloped the three units and was anchored at 0.25g, was used for much of the design. The SSE is 0.12g for seismic Category I non-power block structures. The design spectral shape in all cases is the Regulatory Guide 1.60 spectrum [4].

The WCGS unit has complied with Appendix R with respect to design and analysis. Fire retardant cable wrap is used in the plant.

The site design basis tornado is 360 mph. The main station buildings are located on a terrace above the flood plain, on the east bank of Wolf Creek. The ultimate heat sink is a cooling pond fed by upstream dams and drained by downstream dams. Non-seismic Category I structures are designed to withstand tornado loads such that they will not collapse on seismic Category I structures.

1.2 Overview of Licensee's IPEEE Process and Important Insights

1.2.1 Seismic

NUREG-1407 assigns the Wolf Creek Generating Station plant to the 0.3g focused-scope seismic review category. The WCNOC made the decision to perform a reduced-scope assessment that implements the Electric Power Research Institute's (EPRI's) seismic margins assessment (SMA) methodology for walkdowns and screening. The justification for this decision is described in the submittal [1]. It is based on a review of EPRI NP-6395-D [5], NUREG/CR-5250 [6], and NUREG-1488 [7]. The licensee argued that the seismic hazard at the WCGS, as indicated in NUREG-1488 and EPRI NP-6395-D, is comparable to the seismic hazard at the ten plant sites that were placed in the reduced-scope bin. Additional justification, offered by the licensee, includes work funded by NEI which placed WCGS near the bottom with respect to mean probability of exceeding the SSE [8].

As an original member of the SNUPPS consortium, its in-structure seismic design basis spectra for the power block were based on 0.25g with 3% damping, as opposed to the suggested 5% for seismic margin assessments [9]. This is conservative for the Wolf Creek site. Finally, the licensee states: "The mean probability of exceeding the SSE at the WCGS using the NUREG-1488 hazard estimates is significantly less than the mean probability of exceeding a NUREG/CR-0098 spectrum using the previously published NUREG/CR-5250 hazard estimates."

Of significance is the fact that the horizontal motion design-basis spectral shape for Wolf Creek is more severe than the spectral shape defining the seismic margin earthquake (SME), as defined in NUREG-1407, over the range of 1 to 10 hertz. The licensee states that accelerations above 10 hertz tend to be lower. The vertical motion design-basis spectral shape is more severe than the SME spectral shape over the entire frequency range. Hence, for power-block components, the design-basis seismic demands exceed, in many cases, the demands imposed by the SME.

A subsequent independent reevaluation by the NRC [10] noted the following:

(a) Using the 1993 LLNL hazard curves, the site specific hazard at Wolf Creek is in the mid-range of hazard predicted for other reduced bin sites.

(b) The Wolf Creek safety evaluation report (NUREG-0881) indicates staff approval of 0.12g for seismic Category I structures outside of the standard plant portion of the facility and a 0.2g SSE for the standard plant portion. The SSE level of 0.12g is similar to those in the reduced bin category.

(c) A comparison of uniform hazard spectra (UHS) for Wolf Creek with the UHS for selected reduced bin plant sites also show that the UHS for Wolf Creek are within the range for such sites.

The NRC evaluation [10] concluded that an assignment to the reduced bin category is acceptable for Wolf Creek for IPEEE purposes.

The essential element of the WCGS IPEEE was the detailed and well documented walkdown. A significant feature of the walkdown was that it was performed as if the plant was a focused scope plant using an RLE of 0.3g consistent with the guidance in for EPRI seismic margin assessments. The guidance provided in Tables 2-3 and 2-4 of EPRI NP-6041[9], along with the judgment of the seismic review team (SRT), was relied upon to screen all structures and components at the SSE level. Screening was aided by "screening" analysis of Category I and non-Category I structures. The screening guidance, notes, and caveats associated with spectral accelerations less than 0.8g were used to support the decisions to screen out components and structures as having a high confidence of low probability of failure (HCLPF) capacity of at least 0.3g PGA. Four anchorage calculations were performed to support screening decisions. In five cases, a judgment could not be made for screening at the 0.3g PGA level. These were screened out at the 0.2g SSE level. Consistent with a reduced scope assessment, no further seismic capacity calculations were performed to determine if unscreened items were capable of functioning at the review level earthquake (RLE).

The equipment that could not be screened out at 0.3g but were judged capable of a HCLPF of 0.2g are as follows:

- Refueling water storage tank (RWST)
- Turbine building
- Four 60-cell batteries and racks
- Twelve load shed/emergency load sequencer (LSELS)/engineered safety features actuation system (ESFAS) cabinets
- Strainers and screens

The submittal states that more detailed analysis could demonstrate that the above equipment had a capacity greater than 0.3g. However, these analyses were not performed and no corrective actions were taken.

The study assigned a HCLPF capacity of 0.2g for the primary and alternate safe shutdown paths, with the caveat that more rigorous evaluation of the above components could allow the HCLPF to be increased to the RLE.

The following steps were undertaken for the reduced-scope seismic evaluation of WCGS:

- Selection of the SME
- Selection of the SRT
- Preparatory work prior to walkdown
- Systems and element selection (development of the safe shutdown equipment list)
- Seismic capability walkdown
- Relay chatter evaluation
- Containment performance analysis
- Consideration of Unresolved Safety Issue (USI) A-45, Generic Issue (GI)-131, USI A-17, USI A-40, and Eastern U.S. Seismicity Issue
- Peer review
- Documentation

Note, that a separate system and element selection walkdown was not performed; however, the system and element selection was developed in a comprehensive manner from documentation. As Wolf Creek is a rock site, a soil failure assessment was not needed.

The walkdown also resulted in several minor work requests for adjustment of clearances, adjustment of bolts, and addition of shims.

The seismic IPEEE submittal concludes that WCGS equipment and structures are "seismically very rugged" because of a conservative design, and that the plant has adequate seismic capacity relative to the SSE. The licensee further argued that because the WCGS design basis ground spectra are very close to the site specific 0.3g SME ground spectra and the floor response spectra follow the results of the ground spectra, then the "seismic response spectra to be used for a site specific SME (0.3g) evaluation would be very similar to the seismic response spectra used for the WCGS design basis SSE." The licensee continues to argue that there is "considerable design margin with respect to seismic events at WCGS". The licensee concludes as follows: "As a result of this considerable design margin, evaluation of a 0.3g SME at Wolf Creek would be nearly identical to evaluating the current design basis earthquake at WCGS. As such no additional insights into seismic vulnerability would be expected."

1.2.2 Fire

The licensee has conducted an extensive and detailed analysis of potential fire events at WCGS. Starting with 133 Appendix R defined fire areas, a three phase Fire Induced Vulnerability Evaluation (FIVE) methodology screening process left eight compartments unscreened. These are the control room, two emergency safety features (ESF) switchgear rooms, two electrical penetration rooms, the reactor protection system (RPS) switchgear room, and two auxiliary building areas. The screening process included two separate methods for investigating the potential for inter-compartment fire propagation. Accepted fire damage propagation methods (e.g., COMPBRN and FIVE) were used, during the screening analysis, in a conservative manner to determine (1) the distance between fire source and target over which damage would occur, and (2) the room size below which a hot gas layer could damage overhead cables. Screening core damage frequencies were calculated as the product of the fire ignition frequency and the conditional core damage probability. The FIVE database, guidance and ignition source data sheets were used to determine the fire ignition frequencies. Conditional core damage probabilities were calculated using logic models from the individual plant examination (IPE). A variety of initiating events were identified, and the plant response to fires that cause these events was modeled. Several walkdowns of the plant have been conducted to support the analysis and help resolve the Sandia fire risk scoping study (FRSS) issues.

The fire probabilistic risk assessment (PRA) of unscreened compartments was performed conservatively, but with a less structured approach than the screening study. Fire scenarios based on individual fixed and transient combustible fire sources were developed for each compartment. The fire ignition frequency and conditional core damage probability were assessed for each scenario. COMPBRN IIIe was used to address the extent of potential damage of each fire source, except for cabinets. Cabinet fires were treated by a series of assumptions. This is significant because cabinets are the dominant ignition sources of the unscreened compartments. Control room cabinets are assumed to contain all fires started from within. Using data from Reference [11], cabinet seals and manual suppression are assumed to decrease the likelihood of fire damage from cabinets in other areas. Fires that propagate from cabinets are assumed to damage all equipment and cables within a 10 to 20 foot radius, and automatic suppression is assumed to prevent damage beyond this radius, unless the system is unavailable. Calculation of timing of damage versus suppression does not appear to have been performed during the PRA. However, the damage radius assumption is equivalent to an implicit time delay for the initiation of automatic suppression. This implicit delay was not quantified. Simple event trees have been used to explain the sequences associated with whether or not propagation or suppression occurs.

The licensee has assessed the overall fire core damage frequency (CDF) for unscreened compartments to be 7.5×10^{-6}/ry, which is 15% of the internal events CDF. The primary contributing locations to the fire CDF are the control room and the two ESF switchgear rooms.

The licensee has adequately addressed Sandia's FRSS issues, the USI A-45 issue, and the GI-57 issue. The submittal demonstrated that the containment and its functions are not vulnerable to fires occurring either inside or outside of containment.

The submittal states that no plant modifications are required for adequate defense against fires. The licensee states that the study reaffirms the value of divisional separation, maintenance of fire boundaries and good housekeeping.

1.2.3 HFO Events

The licensee has conducted an analysis in concert with the screening method recommended in NUREG-1407. The design basis was reviewed. Changes since the issuance of the operating license were identified by documentation reviews and verified via walkdowns. The WCGS was designed and constructed in compliance with the 1975 Standard Review Plan (SRP) criteria. The licensee demonstrated that changes since the issuance of the operating license did not affect the conclusion that the unit is still in compliance with the SRP with respect to HFO events. Accordingly, all HFO events were screened out.

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

The present qualitative review of the Wolf Creek IPEEE is focused on the study's completeness in reference to NUREG-1407 guidance; its ability to achieve the intent and objectives of GL 88-20, Supplement Nos. 4 and 5; its strengths and weaknesses with respect to the state-of-the-art; and the robustness of its conclusions. This review does not emphasize confirmation of numerical accuracy of submittal results; however, any numerical errors that were obvious to the reviewers are noted in the review findings. The review process includes the following major activities:

- Complete examination of the IPEEE
- Development of preliminary TER and RAIs
- Examination of responses to the RAIs
- Finalization of this TER and its findings

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

1.3.1 Seismic

Although the WCNOC elected to perform a reduced-scope seismic margin evaluation, this review evaluated the submittal for its adequacy with respect to a focused-scope seismic margin assessment. This is because NUREG-1407 [3] categorized the Wolf Creek site as belonging to the focused-scope bin for purposes of seismic margin reviews. In conducting the seismic review, ERI generally followed the emphasis and guidelines described in *Individual Plant Examination of External Events: Review Guidance* [12] for review of a seismic margin assessment, and the guidance provided in the NRC report, *IPEEE Step 1 Review Guidance Document* [13]. In addition, based on the Wolf Creek IPEEE submittal, ERI completed data entry tables developed in the Lawrence Livermore National Laboratory (LLNL) document entitled *"IPEEE Database Data Entry Sheet Package"* [14].

In its seismic review of the WCGS IPEEE, ERI examined Sections 1, 2, 3, 4.8, 6, 7, and 8 of the IPEEE submittal for WCGS [1], and the licensee's responses to RAIs [15]. The checklist of items identified in Reference [12] was generally consulted in conducting the seismic review. Some of the primary considerations in the seismic review have included (among others) the following items:

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

- Was the development of success paths performed in a manner consistent with prescribed practices?

- Were non-seismic hardware failures and human actions properly considered in such development?

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

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

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

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

Special consideration was given to review of assumptions, because the results of many studies are unduly influenced by assumptions made to simplify the analysis or introduce conservatism.

1.3.2 Fire

During this technical evaluation, ERI reviewed the fire events portion of the IPEEE for completeness and consistency with past experience. This review was based on Sections 1, 2, 4, 6, 7, and 8 of Reference [1] and Reference [15]. The guidance provided in References [12,13] was used to formulate the review process and to organize this technical evaluation report.

The process implemented for ERI's review of the fire IPEEE included an examination of the licensee's methodology, relevant data, and results. The review was conducted with an eye toward consistency with currently accepted methods as well as with the guidance in NUREG-1407. Special attention was given to the following items: (1) the screening methodology, because a tendency to prematurely screen out potentially significant areas, or to inadequately justify screening out an area, has emerged as a common problem among past fire PRAs and IPEEE analyses; and (2) assumptions, because the results of many studies are unduly influenced by key assumptions made to simplify the analysis or introduce conservatism. Other methodological elements include, for example, development of fire event trees, fire propagation modeling, treatment of suppression and detection, and systems modeling. Data elements include such items as: cable routing, fire area partitioning, fire initiation frequency, detection and non-suppression frequencies, and recovery probabilities. Results include such items as: minimal cut sets, core damage frequency and fractional contributions of cut sets, identification of important fire areas and scenarios, and effects of fire on early containment failure.

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

1.3.3 HFO Events

The review process for HFO events closely followed the guidance provided in the report entitled *IPEEE Step 1 Review Guidance Document* [13]. This process involved examinations of the methodology, the data used, and the results and conclusions derived in the submittal. The IPEEE methodology was reviewed for consistency with currently accepted practices and NRC recommended procedures. The validity of the licensee's conclusions, in consideration of the results reported in the IPEEE submittal, was assessed.

2 CONTRACTOR REVIEW FINDINGS

2.1 Seismic

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

2.1.1 Overview and Relevance of the Seismic IPEEE Process

a. Seismic Review Category and RLE

WCGS was originally assigned to the focused-scope seismic review category [3]. However, subsequent reevaluation by the NRC [10] allowed it to be binned in the reduced scope category. The WCGS SSE spectra were used for the assessment. The licensee showed comparisons between the SME anchored as 0.3g and the sites SSE, which indicated that the two sets of spectra are comparable, as was discussed in Section 1, of the present TER.

b. Seismic IPEEE Process

As described in Section 1.2.1, the WCNOC implemented a reduced-scope evaluation for conducting the seismic IPEEE of WCGS. The licensee generally followed the guidance for a reduced-scope assessment found in References [3] and [9] for a rock site, with the addition of a bad-actor relay screening analysis. In addition, the walkdown and screening study was performed using Reference [9] criteria suitable to the RLE. Because the WCGS is a single-unit PWR, no issues pertaining to multiple units were addressed in the seismic IPEEE.

c. Review Findings

In general, the seismic IPEEE addresses all major elements of a reduced-scope assessment, as identified by NUREG-1407. Extensive walkdown preparation, as well as a well-documented walkdown, were performed. The safe shutdown equipment list (SSEL) is reasonable and was logically derived from the IPE loss of coolant accident (LOCA) event tree. A comparison of the SSE and RLE spectra is presented in the submittal.

Although not required of reduced scope plants, finite element modeling of the site was performed, which included soil structure interactions. The relative motion of soil, piping, and structures was also included. In addition, a bad-actor relay chatter evaluation was performed. The containment assessment was simplified. It concentrated on identifying design weaknesses that would prevent screening out the containment structures, internals, heat removal systems, and isolation systems. Non-seismic failures and human actions were accounted for in a manner consistent with an EPRI seismic margin assessment. The evaluation of seismic degradation of fire protection equipment included the effects of such degradation on SSEL equipment and associated structures.

2.1.2 Success Paths and Component List

For the seismic IPEEE of the WCGS, the WCNOC developed a success path logic diagram (SPLD) that describes the plant systems needed to achieve and maintain a stable shutdown condition for at least 72 hours.

The SPLD identifies both preferred and alternate success paths. Reactivity control, reactor coolant system (RCS) inventory control, and decay heat removal are the essential functions that have been addressed in the development of success paths. Because both of the success paths were developed to accommodate a simultaneous loss of offsite power and a small loss of primary coolant (SLOCA), they were both adapted from the IPE SLOCA event tree. The preferred success path relies on high-head safety injection (SI), via the charging or safety injection pumps, with secondary side heat removal via the auxiliary feedwater (AFW) system and steam generator power-operated relief valves (PORVs). Long-term decay heat removal is achieved by loop recirculation within the residual heat removal (RHR) system using the RHR heat exchangers, or by feeding the steam generators with auxiliary feedwater pumps taking suction from the essential service water system. The condensate storage tank is assumed to be unavailable. The essential service water system takes suction from the ultimate heat sink which is considered a safety-related structure. The alternate success path also relies on high-head safety injection, via the safety injection pumps or charging pumps, with bleed and feed cooling for decay heat removal. Long-term inventory control and decay heat removal is achieved by RHR sump recirculation and the RHR heat exchangers. In both success paths, reactor trip and introduction of borated water are the means relied upon for reactivity control.

Once the success paths were determined, the frontline and support systems and components that comprise these paths were identified. The sources used for this identification include the IPE, emergency operating procedures (EOPs), piping and instrumentation diagrams (P&IDs), electrical diagrams, system notebooks, and Appendix R information. Non-safety-related cabinets were included in a few rooms with obvious proximity to safety- related cabinets. The "rule-of-the-box" approach [9] was used in reducing the number of separately identified components. In addition, a simulation of the accident was conducted using an operating crew. All instruments and controls used by the crew were added to the SSEL. Containment cooling and isolation systems were also included. The submittal states that the list of systems to be used was influenced by considerations of spatial interactions. A list of 473 SSEL components was developed to represent the success paths. In addition, another 253 components were identified as being associated with components found in the IPE in preparation for a seismic PRA that was later abandoned.

Table B.1, Appendix B of Reference [3] states that both a focused-scope and reduced-scope EPRI seismic margin assessment should include a system and element walkdown. Either this aspect of the assessment was ignored in this submittal, or it was combined with the seismic capability walkdown. Because spatial interactions did not emerge as a significant problem during the walkdown, the selection of success paths was demonstrated to be acceptable with respect to the spatial interactions. A review of the SSEL itself revealed a comprehensive list consistent with those of other similar units, except that only one 4160V switchgear bus was listed.

2.1.3 Non-Seismic Failures and Human Actions

The submittal states that non-seismic failures and human actions were considered in the development of the SSEL. Non-seismic failures were considered by incorporating high reliability equipment in the safe shutdown paths. The safe shutdown paths use auxiliary feedwater systems and safety injection systems. These systems are those suggested in EPRI NP-6041, and can be considered highly reliable by virtue of their redundancy. The submittal provides failure probabilities of equipment at the train level, but does not use a screening criteria [16] as suggested in the NUREG-1407 guidance. However, quantitative analysis of none-seismic and human errors are not required for a reduced scope assessment.

To account for human actions, the licensee has provided a list of operator actions, the location of the actions, their failure probabilities, associated time limits, and an evaluation of whether each would be affected by an earthquake [15]. All but one action takes place in the control room. Control room actions are in cut sets that are not likely to significantly influence the failure of success paths. This is because they generally would be associated with cut sets that include either previous failures of trains of equipment (e.g., switchover of component cooling water (CCW) after failure of a CCW loop), or for which redundant actions are in the procedures (e.g., feed and bleed initiated if steam generator level can not be maintained). There is one action which does not take place in the control room. This local action involves manual closure of a CCW heat exchanger bypass valve or bypass isolation valve. The submittal states that at least 30 minutes is available for this action and it is not likely to be affected by the SME.

2.1.4 Seismic Input

For purposes of the walkdown, it appears that the WCGS seismic IPEEE has used the NUREG/CR-0098 median 5%-damped spectrum for rock, anchored to the RLE PGA value of 0.3g, to define seismic margin earthquake demands. The licensee also performed a comparison of this spectrum with the SSE design basis spectrum, which followed Regulatory Guide 1.60 for a composite site composed of the SNUPPS units. It was found that the design basis horizontal ground motion spectrum enveloped the RLE spectrum in the 1 to 10 hertz interval, and the design basis vertical ground motion spectrum enveloped the RLE spectrum for the entire frequency range. For components that could not be screened out during the walkdown using the RLE spectrum, judgments were made to assess if these had capacities to withstand the SSE design basis spectrum. These judgments were founded on conformity of the equipment, as viewed during the walkdown, with the design specifications.

2.1.5 Structural Responses and Component Demands

a. Overall Approach

In the WCGS seismic IPEEE, structural responses and component demands associated with the seismic margin earthquake were obtained by linearly scaling existing design-basis responses and demands. For example, design-basis in-structure response spectra (IRS) were scaled by a factor that is the ratio of the RLE spectral acceleration to the 3% damped SSE at a building's dominant natural frequency. This factor was then applied to the design basis 5% damped in-structure spectra for a particular building in order to obtain the IRS for seismic margin evaluation (see EPRI NP-6041-SL [9]). The submittal provides a revealing comparison of the 5% damped SMA spectrum and the 3% damped design basis spectrum. The increase of the SMA demand over the design basis demand is always less than 50% in the horizontal direction, and there is no increased demand in the vertical direction. New in-structure response models were not generated for the IPEEE.

b. Structural Models

Dynamic finite element models were used to generate the original (design-basis) structural responses and component demands. Inputs included synthetic earthquake time-histories. The models considered damping, soil structure interactions, and spatial interactions among structures of the power block. New structural models were not generated for the IPEEE and are not required in a reduced scope assessment.

c. Soil Structure Interaction Analysis and Results

Floor response spectra were generated using the acceleration time histories generated from the FLUSH finite element analysis for power block structures. Other techniques were used for the RWST and buried equipment such as the emergency fuel oil storage tank. The design of all components was actually based on floor response spectra combined to envelop the floor response spectra of three of the SNUPPS units. Site dependent soil properties were used, taking into account non-linearity effects.

2.1.6 Screening Criteria

The screening criteria and guidelines contained in EPRI NP-6041-SL were used as the basis for conducting screening of components and structures for the WCGS seismic IPEEE. Components and structures were checked against the EPRI SMA screening caveats, in addition to being evaluated for anchorage adequacy and for potential seismic spatial interaction concerns. Judging by the references to the notes of Tables 2-3 and 2-4 of EPRI NP-6041, Rev. 1, contained in the submittal, the screening column applicable to spectral accelerations less than 0.8g was used. In a few cases, the submittal noted that the component would also be screened at an acceleration of 0.5g PGA. Using these criteria, the walkdown screened out all structures and components except for LSELS/ESFAS cabinets, battery racks, strainers/screens, the RWST and the turbine building. These unscreened items were judged to have HCLPF capacities in excess of 0.2g.

Since no seismic capacity calculations on the unscreened components were performed, the success path and, therefore, plant HCLPF, was judged to be 0.2g.

2.1.7 Plant Walkdown Process

a. Preparatory Work

The scope of preparatory work included review of equipment specifications and arrangement drawings, review of design bases and licensing bases, review of seismic qualification documentation, review of seismic qualification review team files, review of existing calculations, and review of references to understand potential seismic failure modes.

b. Systems and Element Selection Walkdown

A separate systems and element selection walkdown was not performed.

c. Seismic Capability Walkdown

The walkdown was the primary method of evaluation for this IPEEE. The approach and justifications for screening were well documented in the submittal. The SSEL and structures housing the components on the list were divided into 42 categories, generally corresponding to the categories in EPRI NP-6041 Tables 2-3 and 2-4. At least one example of each category was examined in detail. For example, a few cabinets were opened and scrutinized for the adequacy of fastenings for the internal components. They were further examined for anchorages and spatial interactions. The other equipment in each category were "walked by" to see if they generally conformed to the exemplar component(s). A limited sampling of distributed systems (e.g., piping, heating, ventilation and air conditioning [HVAC] ducts, and cable trays) was walked by. The walkdown investigation emphasized continued equipment functionality, anchorage, and spatial interactions in response to the RLE. Judgments regarding screening with respect to these three aspects, and in accordance with the above-described criteria, were generally made during the walkdown. If an item could not be screened at the RLE level, a judgment was made about whether it would have sufficient capacity for the 0.2g SSE.

With respect to functionality, at least one seismic evaluation work sheet (SEWS) was developed for each category. For some categories, the observations or items for which a walk-by was performed were documented using "abbreviated screening evaluation sheets." With respect to spatial interactions, attention was paid to the effect of non-seismic Category I equipment on Category I equipment, spray or flooding concerns, and close proximity of two or more seismic Category I items. General seismic housekeeping issues were also addressed.

With respect to anchorage evaluation, judgments were based on conformance with the design documentation, anchorage details, qualification reports, bounding calculations and existing "specific" calculations. Bounding anchorage calculations were performed, as part of the IPEEE, to demonstrate that the following anchorages could be screened out for the RLE: motor control centers (MCC), room coolers (other than containment fan coolers and AFW pump room coolers), 125 V DC panels, and 7.5 kVA inverters. Prior "specific" calculations were used for CCW heat exchangers, RHR pumps, containment spray pumps, and diesel generator panels. The MCC anchorage calculations involved using the highest elevation in-structure response spectra for calculation of demand. "Studs were checked for shear and tension, embedded channels checked for weak axis bending" [15]. All anchorages were screened out at the SME level. Although requested in the RAIs, example calculations were not provided in the submittal.

d. Subsequent Walkdowns

Subsequent walkdowns were not performed.

e. Treatment of Inaccessible Components

Only a few components listed in the SSEL were not accessible for inspection as part of the seismic walkdown. For these components, the evaluation was based on a review of existing drawings, qualification packages, or comparison with components that were viewed.

f. Walkdown Duration and Training of Seismic Review Team (SRT)

The seismic walkdown was performed by a single SRT consisting of seven trained members: five seismic capability engineers from Vectra Technologies Inc., one seismic capability engineer from WCNOC, and one systems engineer from WCNOC. Information about the walkdown duration was not provided.

WCGS personnel were involved in all aspects of the analysis, although they did not play dominant roles. The WCGS engineers who participated in the seismic evaluation received seismic qualification utility group (SQUG) training and EPRI Seismic IPE follow-on training. The submittal identifies all seismic IPEEE participants, their roles in conducting/reviewing the IPEEE, and their relevant qualifications.

2.1.8 Evaluation of Outliers

a. Overall Approach

The walkdown identified no outliers in that all but five sets of equipment items or structures were screened out at the SME level. The five sets are (1) eight LSELS/ESFAS cabinets, (2) four sixty-cell battery racks, (3) strainers/screens, (4) RWST and (5) turbine building. These items "were considered to be indeterminate beyond the plant design basis without additional detailed evaluations." The issue regarding the LSELS/ESFAS cabinets is that they are attached by bushing with the attachment points at or below the vertical mid-plane of the cabinets. The cabinets were tested and qualified to the SSE in that configuration. The effect of their interactions at higher levels cannot be easily evaluated without additional tests. Because the cabinets conformed to the design basis documents, a HCLPF capacity of 0.2g was assigned. The issue regarding the battery racks is that the batteries loosely fit inside the racks. The racks have side rails and they are laterally braced between adjacent frames. The gap between the batteries and the frame was shown to be within vendor specifications which, in turn, was based on qualification data. Therefore, the batteries and racks were assigned a HCLPF capacity of 0.2g. The issue concerning strainers/screens is that insufficient guidance is provided in EPRI NP-6041 to make a capacity judgment. Existing design basis seismic capacity analysis of the RWST allowed the judgment that it is adequate for the SSE. The submittal suggested that a higher capacity could be shown as a result of a conservative deterministic failure margin (CDFM) SMA per Appendix H of Reference [9]. Regarding the turbine building, the SRT judged that the margin for lateral loads (and base shears) introduced by the tornado design basis made the actual design basis shears equivalent to those of the 1985 Uniform Building Code Zone 4. On this basis, a HCLPF capacity of 0.2g was assigned. The original design basis corresponds to the Uniform Building Code Zone 3.

Four areas were noted in which the actual field installation did not conform to the seismic design basis. These areas are:

- A transformer inside an inverter was not bolted to the frame on one side,
- Structural members or fire protection material was in close proximity to electrical cabinets and motor control centers in one instance,
- A victaulic coupling on a drain line in the diesel generator building was in close proximity to a motor control center, and
- Loose and missing bolts, and shims missing, on chillers.

These issues were identified on work requests for corrective action. In addition, several housekeeping items with respect to trash barrels and cabinets near safety-related equipment were identified on a Plant Improvement Request.

b. HCLPF Calculations

HCLPF calculations are not required for a reduced scope assessment.

c. Other Calculations

No other calculations were performed with respect to outliers.

2.1.9 Relay Chatter Evaluation

The WCNOC conducted a computerized search for low ruggedness relay model numbers found in EPRI NP-7148, the results of which were confirmed by review of electrical schematic diagrams. Five models were found, only one of which was used for equipment within the SSEL. The configurations of this relay actually used at the WCGS is not the low ruggedness configuration. Therefore, the licensee performed a low ruggedness screening assessment consistent with focused-scope guidelines, and found none.

2.1.10 Soil Failure Analysis

As the WCGS is situated on a rock site, a soil failure analysis was not performed. The seismic IPEEE addressed relative displacement of buried piping between the power block and the emergency service water pump house. A review of design documents by the SRT led to the judgment that the interaction of the piping with the associated structures can accommodate the RLE.

2.1.11 Containment Performance Analysis

The seismic IPEEE for WCGS included an analysis of containment performance for identifying vulnerabilities that involve early containment failure. The evaluation included the following structural integrity issues: concrete structure, hatches, personnel air locks, and penetration cooling. The submittal discusses why the WCGS design does not have vulnerabilities in these areas. Isolation failure owing to failure of air operated valves is also not an issue because these valves do not need air to actuate for isolation. Containment heat removal is evaluated by considering the robustness of the containment air coolers and RHR heat exchangers. Both were screened out at the RLE level during the walkdown. The SRT performed a walkdown of the containment to investigate the potential of spatial interactions or other unusual circumstances. None were found. Hence, the containment was screened out at a capacity of at least 0.3g PGA.

2.1.12 Seismic-Fire Interaction and Seismically Induced Flood Evaluations

a. Seismic-Fire Evaluation and Walkdown

The WCGS IPEEE included a review of seismic-fire interaction concerns as part of the evaluation of the Sandia fire risk scoping study issues. This review included an assessment of: (1) seismically induced fires, (2) seismic actuation of fire suppression systems, and (3) seismically induced degradation of fire suppression systems. These issues were evaluated from the perspective of interfering with the operation of equipment or structures within the success paths [15]. A separate walkdown to address these issues was not conducted. The seismic capability walkdown was sufficiently comprehensive to address these issues.

b. Seismic-Induced Fires

The assessment focused on potential earthquake induced leakage or rupture of pipes, tanks and pumps which contain flammable materials. Only hydrogen, lube oil, and diesel oil are present in structures related to the SSEL. Reviews of the seismic qualification documents, as well as walkdown observations and judgments, determined that all equipment containing flammable materials is seismically robust.

c. Seismic Actuation of Fire Protection System Equipment

The evaluation for seismic actuation of fire suppression systems considered the potential effects of flooding or wetting of safety systems. The submittal states that this was considered as part of the plant design basis, as described in the WCGS USAR. Drip-proof pumps are used in the plant, and wet-pipe sprinklers are not used in electric motor driven safety-related pump rooms and electrical equipment rooms. The submittal notes that drainage is adequate in rooms that contain fire suppression equipment. Because low ruggedness relay failure modes were not found, earthquake-induced inadvertent actuation of fire suppression systems does not appear to be a concern at the WCGS.

d. *Seismically Induced Degradation of Fire Protection System Equipment*

The licensee's review of seismically induced degradation of fire suppression systems included investigating dislodgment of fire protection equipment, such that they could affect safe shutdown equipment, as suggested in Reference [17]. The submittal states that the portions of the fire protection equipment that pass through safety-related areas are seismically analyzed and supported. These systems are also designed to preclude flooding of safety-related areas. The submittal noted, in addition, that the walkdown findings supported this statement. Note, that seismic failure of fire protection equipment, as pertains to its operability, was not addressed. The focus was on whether degradation of fire protection equipment could damage safety-related equipment per Reference [17].

e. *Seismic-Induced Flood Evaluation*

This issue was addressed as part of the seismic actuation and degradation of fire protection system equipment as discussed above.

2.1.13 Treatment of USI A-45

The submittal simply notes that, as there are no outliers with respect to the design basis identified during the SMA, then all components and structures related to the SSEL are adequate for decay heat removal.

2.1.14 Treatment of GI-131

GI-131 is relevant to WCGS since the plant is a Westinghouse PWR, and it employs a movable flux mapping system. The seismic IPEEE submittal notes that the SRT checked the hold down assembly in order to verify the seismic adequacy of the system. The SRT concluded that the drive and transfer assemblies were adequately secured to the frame, and not a credible interaction source during a seismic event. In particular, the SRT noted that the wide flange beams could be bolted to the frame of the mapping system, thereby increasing the lateral stability of the entire assembly.

2.1.15 Other Safety Issues

a. *Eastern U.S. Seismicity Issue*

The submittal notes that this issue was resolved for the WCGS using the LLNL hazard curves, and that the IPEEE submittal itself satisfies the requirements for this issue.

b. *USI A-17*

Regarding USI A-17, "System Interactions in Nuclear Power Plants," the submittal states that closure for external events is provided by the seismic capability walkdown, which addresses systems interactions, including spatial, fire, and flooding interactions.

c. *USI A-40*

Concerning USI A-40, "Seismic Design Criteria," the submittal states that the refueling water storage tank (RWST) is the only above ground flat bottomed tank on the SSEL, and it was re-analyzed in 1989. Through the walkdown and a review of this analysis, the SRT judged that the RWST had a HCLPF of 0.2g PGA.

2.1.16 Peer Review Process

The WCGS seismic IPEEE received an independent peer review from Dave Moore, who was with NUS Corporation at the time of his review. Aspects reviewed included development of the SSEL, screening decisions and judgments, backup documentation, final report and results. The independent peer review concluded that there were no significant deficiencies in the seismic IPEEE.

2.1.17 Summary Evaluation of Key Insights

The plant HCLPF capacity has been determined to be equal to the 0.2g seismic design basis of the plant. There are no outliers at this level. This finding is not surprising, given the high design basis spectra and the recent vintage of WCGS.

The submittal states that if further SMA analyses were performed on four of the five items described in Section 2.1.8 above (i.e., LSELS/ESFAS cabinets, screens/strainers, RWST, and 60-cell battery racks), then the plant HCLPF capacity would be equal to or exceed the RLE (i.e., NUREG/CR-0098 median, 5%-damped, rock spectrum anchored to a PGA value of 0.3g). The lower capacity of the turbine building was not addressed. The turbine building contains a power feed cable from the switchyard to one of the ESF transformers, which is routed along the west wall through an electrical cabinet.

2.2 Fire

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

2.2.1 Overview and Relevance of the Fire IPEEE Process

a. *Method Selected for Fire IPEEE*

The WCNOC performed a comprehensive, 3 stage screening analysis using FIVE [17] guidance and data. As part of the screening process, fire damage modeling was performed, using the FIVE screening methodology to assess the potential of fire damage to cables owing to ceiling jet effects, and COMPBRN IIIe to assess the potential of fire damage from plume, thermal radiation and hot gas layer effects. A probabilistic risk assessment was performed for the eight unscreened compartments. A core damage frequency (CDF) estimate is provided as the sum of the CDFs of the unscreened compartments. A FIVE Phase-III verification walkdown was conducted as one step of the fire analysis. Walkdowns were also used to provide information for the discussion of the Fire Risk Scoping Study (FRSS) issues.

b. Key Assumptions Used in Performing Fire IPEEE

Assumptions are found throughout the submittal. The assumptions and conditions that had a significant effect on the results of the fire analysis are:

1. Thermo-Lag fire barrier wrap was not credited in the study.

2. Although essentially all safety-related cables are IEEE 383 qualified, the fire damage criterion used in fire damage calculations assumed they were not. However, self-ignited cable fires were ruled out.

3. During the screening analysis, if fire damage was calculated to occur in overhead cables, then all cables in the compartment were assessed as damaged. During the PRA of unscreened compartments, fire suppression failure was a pre-condition for failure of all cables in a compartment. If suppression was successful, cables within a damage radius established by judgment, based on "generic" COMPBRN runs, were assumed to fail.

4. The fire protection system is designed and installed per proper codes and standards. Therefore, it is effective against fires, unless it is unavailable.

5. A fire in an electrical cabinet results in failure of all functions in that cabinet.

6. Hot shorts, if they can occur in a compartment, were assigned a probability of unity, with the exception of a sustained hot short that could open the PORVs and simultaneously prevent block valve closure. This latter hot short event was assigned a probability of 0.07.

c. Status of Appendix R Modifications

WCGS is an Appendix R plant. The major sources of plant information for the IPEEE study were the internal event probabilistic safety assessment (PSA); an electronic cable and raceway database, called SETROUTE; and the Appendix R study found in the WCGS updated final safety analysis report (UFSAR).

d. New or Existing PRA

The IPEEE is a new fire study employing FIVE methodology, COMPBRN, and probabilistic risk analysis.

2.2.2 Review of Plant Information and Walkdown

a. Walkdown Team Composition

Many walkdowns were performed to obtain and verify plant information, to investigate spatial interactions, and to address the FRSS issues. The submittal does not describe the composition of the walkdown teams except to say that they were composed of the fire analysts. Judging by the detailed knowledge of the plant evidenced in the submittal, the walkdowns appear to have been thorough.

b. *Significant Walkdown Findings*

The scope of the walkdowns included a variety of considerations, including location of cabinets, cable trays, ignition sources (fixed and transient), drains, detection and suppression equipment, in-cabinet details, and the FRSS issues. The walkdowns resulted in subdivision of a few Appendix R fire areas. This subdivision was based either on FIVE criteria or on the area being large with no intervening combustibles between redundant sets of cables. The walkdown teams used standardized forms to record their observations.

c. *Significant Plant Features*

The following list describes some features of the WCGS, which are relevant to the fire analysis:

1. The auxiliary building houses engineered safety features equipment and nuclear steam supply system (NSSS) auxiliary equipment.

2. The control building includes the control room, cable spreading rooms, and other control and power distribution systems.

3. The turbine building includes a power feed cable from the switchyard to one of the ESF transformers, which is routed along the west wall through an electrical cabinet.

4. Cables are IEEE 383 qualified.

2.2.3 Fire-Induced Initiating Events

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

Reactor trip was assumed to occur for areas or compartments that contain components related to continuance of operation of the primary (reactor) side or the secondary (turbine side) of the unit. A reasonable set of criteria were used to define these components. In addition, all failures that could occur as a result of a fire in each compartment were reviewed to determine the appropriate initiating events. The following initiating events were found to result from fires, in addition to reactor trip: loss of station power, loss of 125 V DC bus, loss of component cooling water, loss of service water, transient without main feedwater, and main steam line break (owing to failure of atmospheric relief valves in the open position).

b. *Were the Initiating Events Analyzed Properly?*

Conditional core damage probabilities were calculated using event trees and fault trees, corresponding to the selected initiating event for a compartment, from the internal event PSA model. Both loss of reactor coolant pump seal cooling and inadvertent opening of pressurizer PORVs were included in the plant logic models.

2.2.4 Screening of Fire Zones

a. *Was Proper Screening Methodology Employed?*

Screening was performed in multiple stages using the protocol prescribed in the FIVE methodology. Starting with 133 Appendix R derived fire areas and compartment, qualitative (Phase I) screening removed 21 areas on the basis that neither equipment related to safe shutdown nor reactor trip are present. Phase II (Steps 1 and 2) quantitative screening was based on comparison of compartment screening core damage frequencies with a significance threshold of 10^{-6}/ry. The screening core damage frequency for each compartment is the product of the fire ignition frequency and the conditional core damage probability, assuming all cables and equipment in the room are failed. The fire ignition frequency was derived using ignition source data sheets and the data from FIVE methodology. Eighty-two compartments were screened out during this step.

Phase II (Step 3) screening removed another 23 compartments from further consideration. This step adds an additional multiplicative factor to Phase II (Step 2) screening. This factor is called the probability of critical combustible loading (PCCL). It is the conservatively determined likelihood that a fire from any of the ignition sources (fixed and transient) could damage critical targets, especially cables. The WCGS used a reasonable combination of FIVE methodology coupled with COMPBRN IIIe in an interesting approach. COMPBRN IIIe calculations were used to determine the minimum size room for which a hot gas layer would not damage cables, and the minimum distance between target and source for damage owing to radiation and plume effects. Damage owing to ceiling jet effects was estimated using the technique in the FIVE Fire Screening Methodology. If damage to cables might occur, as estimated by application of these methods, then the conditional core damage probability was modified to account for the source plus other equipment damage. At the end of the screening analyses, eight compartments were left for detailed probabilistic analysis.

The probability of failure of suppression was included as a factor (0.02) to reduce the estimated CDF in three compartments, two radiation area access rooms and the turbine building general area, without an estimate of the relative timing of damage versus suppression. These compartments would not have been screened out without this factor of 50 reduction in core damage frequency. Justification for crediting fire suppression in the radiation access rooms was based on the existence of code compliant fire detection and suppression systems. In these rooms, ignition sources are located below a noncombustible suspended ceiling, and all SSE cables are located in trays or conduit above this ceiling. Fire detection and automatic water suppression is located both below the ceiling and above the cables. In the turbine area, the only SSE related item is a power feed cable from the switchyard to one of the ESF transformers, which is routed along the west wall through an electrical cabinet. The submittal argues that there is sufficient space between this cable and all ignition sources, except transient combustibles and the cabinet, to take credit for the code compliant automatic suppression. Suppression was not credited, however, when analyzing the cabinet as an ignition source. The CDF of the turbine area, when considering all fire scenarios that could affect the ESF power cable calculated in consonance with FIVE guidance, was less than the 10^{-6}/yr screening criterion.

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

The control room was not included in the screening process and, therefore, was not prematurely screened out. The cable spreading room was included in the screening process and was not screened out. Detailed probabilistic risk assessments were performed on both of these rooms.

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

Sound logic and methods were used to screen out fire zones and areas.

2.2.5 Fire Hazard Analysis

The data and methodology, which is provided in Reference [17], has been used to estimate fire frequencies for individual compartments and fire areas. Ignition source weighting factors have been applied to apportion the overall fire frequency to the specific fire zones. As is typical, a great deal of judgment was necessary to associate as-built plant compartments with the plant locations in Table 1.2 of Reference [17]. The compartment derived frequencies appear reasonable, and the sum of all the fire frequencies of all compartments, as estimated in this study, is conservative relative to the industry average plant-wide fire frequency. Plant-specific fire occurrence data has not been used.

2.2.6 Fire Growth and Propagation

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

Cross-zone fire propagation has been considered in two ways: (1) a fire compartment interaction analysis (FCIA) using FIVE criteria [17] and (2) a multiple compartment fire analysis. As part of the FCIA, screening of boundaries used such deterministic factors as presence of fire-rated barriers, low combustible loading, and fire detection and suppression systems. The FCIA concluded that the boundaries between Appendix R defined fire areas will not allow fire propagation between areas. The fire barriers, doors and penetrations are also subjected to a periodic surveillance program.

A multiple compartment fire analysis, which quantitatively assesses the likelihood of damaging equipment in adjacent compartments, was performed if two criteria were met. The first criterion is that a hot gas layer capable of damaging cables can form in one of the compartments. The second criterion is that the adjacent compartments contain redundant equipment. The analysis considered a hot gas layer as the mechanism for propagating the effects of fire. A screening CDF calculation was performed on the compartment pairs that met these criteria, using a screening barrier failure probability of 0.1. If the screening value was greater than 10^{-6} /ry, then the barrier failure probability was adjusted. A few compartments required this adjustment. The adjusted barrier failure "probabilities" were taken from NUREG/CR-4840, which provides barrier failure "rates" in units of per reactor year. The WCNOC used these numbers to screen out the remaining compartment pairs with a CDF less than 10^{-6}/ry. This screening was achieved without including either manual or automatic suppression.

Note, that the barrier screening method requires a unitless barrier failure probability. Use of the barrier failure rates incorrectly gives units of (reactor year)$^{-2}$ in the CDF equation used by the licensee (Reference [1], page 4-39).

b. Assumptions Associated with Detection and Suppression

As described in Section 2.2.4a, automatic suppression was credited for three compartments, two radiation area access rooms and the turbine building general area, without an estimate of the relative timing of damage versus suppression. That is, the time-dependent likelihood of detection and suppression before damage was not calculated. Based on conformance to National Fire Protection Association (NFPA) 12A, 13 and 15 standards, it was assumed for these compartments that suppression would always put out any fire in the compartment unless a mechanical or electrical failure of the suppression system occurred. The failure of suppression is associated only with the suppression system unavailability as provided in Reference [17]. Manual suppression was not credited in the screening analyses.

Discussion of detection and suppression used during the PRA of unscreened compartments is presented in Section 2.2.8.

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

Suppression induced damage is not treated in the quantitative portion of the fire analysis. The submittal notes that this issue was treated as part of the Appendix R fire hazard analyses. In general, drip-proof safety-related pump motors are used, and wet-pipe sprinklers are not used in rooms that contain safety-related electric motor driven pumps or safety-related electrical equipment rooms. Adequate drainage is provided where needed. See Section 2.2.12 of this TER for further discussion of this issue.

d. Computer Code Used, if Applicable

A computerized database, SETROUTE, was used for cable raceway information. COMPBRN IIIe was used for fire damage calculations. A PRA code (probably NUPRA, although the name was not mentioned in the submittal) was used for conditional core damage frequency calculations.

Results of COMPBRN IIIe runs for typical plant configurations, with fixed and transient combustibles, were presented in the submittal. No configurations are ventilated. The inputs (e.g., heat release rates and duration), as well as the results, look reasonable. The submittal (page 2-6 of Reference [1]) summarized the results as follows:

- Damage to targets in the immediate vicinity of the source occurs quickly such that automatic suppression is not effective
- Damage is not sustained by targets outside the immediate vicinity before the fire self extinguishes

This is typical of findings derived by a conservative use of COMPBRN or the FIVE Fire Screening Methodology. One minor shortcoming of the runs is that a liquid pool transient combustible fire (e.g., with oil or cleaning fluid) was not included. The submittal states that the most appropriate transient combustible is Class A/B mixed combustibles, such as paper, oily rags, plastic bottles, etc. Fixed combustible fire runs, using liquid spills, were included.

2.2.7 Evaluation of Component Fragilities and Failure Modes

a. Definition of Fire-Induced Failures

Failures were taken to be any failure mode that failed the desired function of equipment owing to overheating. Other effects of fire (e.g., smoke and soot) on equipment were not included.

b. Method Used to Determine Component Capacities

Component capacities to resist overheating were taken from various sources including Sandia oven test data, Reference [17], and COMPBRN IIIe.

c. *Generic Fragilities*

Generic fragilities were used. The study assumed the failure temperature criterion for uncoated PE/PVC cable (523K). Failure was also assumed if the incident heat flux exceeded 5700 W/m^2. Fire within a cabinet was assumed to fail all equipment within the cabinet. Failure of electrical components (e.g., instrumentation through electric motors) was taken as 339K, and failure of relays and switches was taken to be 433K. Both the cable pilot ignition and spontaneous ignition temperatures were taken to be 773K.

d. *Plant-Specific Fragilities*

Plant-specific fragilities were not mentioned in the submittal.

e. *Technique Used to Treat Operator Recovery Actions*

Operator actions, modeled in the fault trees, were reviewed to determine if (1) they take place outside of the control room, and (2) they would be affected by the specific fire under consideration. Two human actions met these criteria: operator fails to establish alternate cooling, and operator fails to provide cooling to switchboard rooms. These were given an HEP of unity. The study also mentions that the HEP used for providing plant operation from the alternate shutdown panel is 0.02, as calculated using NUREG/CR-1278. All other operator actions used the internal event PSA model values. Actions that are considered recovery actions in the internal event model were all assigned a human error probability of unity for the fire IPEEE study.

2.2.8 Fire Detection and Suppression

Treatment of fire detection and suppression during the screening analyses was previously discussed. This section discusses their treatment during the detailed PRA of unscreened rooms. Halon is the automatic suppression system used in the unscreened rooms (in those rooms that have automatic suppression).

The extent of damage of fire was estimated by COMPBRN. Within the area of damage estimated by COMPBRN, all equipment was assumed to fail. Outside of this area, equipment was assumed to fail if suppression failed. If a compartment has the capability of automatic suppression, then it was assumed to be effective unless it failed. In effect, therefore, the non-suppression probability, used in the analysis, is the unavailability of a halon system (0.05).

For cabinet fires, two separate methods appear to have been used; one in the control room and one elsewhere. In the control room, it appears that fire damage was simply assumed to not spread from cabinets. However, control room abandonment owing to smoke is modeled as occurring with a probability of 3×10^{-3}. This is the probability of non-suppression of cabinet fires in the control room from Nuclear Safety Analysis Center (NSAC) 181, based on a 15 minute time for smoke to cause abandonment.

Elsewhere, the probability of propagation of fire damage from cabinets was treated in a probabilistic manner as follows. As a first step, cabinet fires were assumed to propagate from the cabinet. If the CDF was calculated to be less than 10^{-7}/ry, nothing further was done for that scenario. Otherwise, a refinement was developed based on a review of the EPRI fire database [11]. The review allowed the licensee to make judgments about the probability of fire propagation from various kinds of cabinets. The estimates are as follows:

Louvered, open, or vented cabinets	propagation probability = 1
Sealed cabinet	propagation probability = 0.69 (which is the complement of the cabinet self-extinguishment probability as gleaned from EPRI cabinet fire data [11])
Sealed cabinet in heavily traveled area	propagation probability = 0.15 (which is the complement of the cabinet self-extinguishment probability plus part of the manual extinguishment probability as gleaned form EPRI cabinet fire data [11])

For a fire scenario in which propagation out of cabinets does occur, COMPBRN IIIe was not specifically used for each source and compartment to determine the probability of cable damage or the probability of suppression (see page 4-47 of Reference [1]). Instead, for most cabinets, all cables and equipment within a 20-foot radius of the cabinet were assumed to fail. (In some cases, a 10-foot radius was used.) The submittal states that these assumptions are consistent with the general finding of "generic" COMPBRN IIIe runs mentioned in Section 2.2.6d, above. If automatic suppression is available in the room, detection is assumed to occur immediately, and the non-suppression probability is simply the Reference [17] suppression system unavailability. Cables outside the 20-foot (or 10-foot) radius were assumed to not fail if suppression is successful. Failure of suppression is assumed to result in failure of all equipment in the compartment.

2.2.9 Analysis of Plant Systems and Sequences

a. Key Assumptions Including Success Criteria and Associated Bases

The success criteria are directly taken from the IPE analysis and have not been modified for the fire analysis. Another key assumption is that the conditional core damage probabilities, calculated from the system event tree and fault tree models, were not screened for non-realizable combinations of failure modes. Therefore, the study could have implicitly included spurious actuation as well as failure to actuate, for the same component, when determining the CDF. For example, a valve could fail to close or fail closed in different branches of the system fault trees.

b. Event Trees (Functional or Systemic)

The event tree models of plant functions and systems were generally described in the text of the submittal, but not depicted graphically. The following event tree logic models were used: transient without main feedwater, including response to a stuck open PORV; loss of offsite power; loss of 125 V DC buses; loss of reactor coolant pump seal cooling; and response to a small break LOCA, including refill of the RWST.

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

No dependency matrix has been provided in the submittal.

d. Plant-Unique System Dependencies

The submittal does not present any unique system dependencies.

e. Most Significant Human Actions

The submittal does not provide information to determine the most significant human actions.

2.2.10 Fire Scenarios and Core Damage Frequency Evaluation

Following the screening analysis, eight compartments remained for detailed PRA. The eight unscreened compartments are as follows (a synopsis of the fixed fire sources in each room is included in parentheses): control room (cabinets), Train A and B switchgear rooms (cabinets and transformers), a portion of the auxiliary building general area at elevation 2000 (cabinets), a portion of the auxiliary building general area at elevation 2026 (CCW pumps and cabinets), south electrical penetration room (motor control centers), north electrical penetration room (motor control centers), and reactor trip switchgear room (MG sets). All the rooms contain safety-related cable.

Specific fire scenarios were developed for these eight compartments. Including the control room, about 70 scenarios were evaluated. The core damage frequency presented in the submittal is 7.5×10^{-6}/ry, which is the sum of the frequencies of these scenarios. The following table summarizes the distribution among compartments.

Compartment	**Fraction of Total CDF (%)**
ESF Switchgear Room, North	34
ESF Switchgear Room, South	28
Control Area	19
Electrical Penetration Room, North	5
Electrical Penetration Room, South	3

Because the core damage frequencies of the other 125 areas were not included, this cannot be considered the fire core damage frequency for the plant. However, presentation of the core damage frequency for only the unscreened compartments is a typical practice.

Hot shorts were found to be significant for four unscreened fire areas: two ESF Switchgear rooms and two electrical penetration rooms. In these areas, sustained hot shorts, which could open the pressurizer PORVs and simultaneously disable closure of the block valves, were assigned a conditional probability of 0.07. This value is based on information in NUREG/CR-2258. The study also conservatively assumed that hot shorts in motor control centers in ESF cabinets would be sustained and would occur (with a probability of unity) given a fire in that cabinet. However, hot shorts were assumed incapable of causing motion of motor-operated valves (MOVs) for fires in power and control cables that occur between the MCC (outside of the cabinet) and the valve motor.

Scrutability of Core Damage Frequency Evaluation

The chain of computations from fire ignition frequency through calculation of the CDF, included development of source-by-source scenarios, selection of initiating events, assessment of damage propagation and suppression, failure of operation from the remote shutdown panel, and calculation of conditional core damage probabilities. The text was supplemented by simple fire event trees, which helped explain the logic

used in calculating core damage frequency. Assumptions were stated and sensitivity studies were performed in important areas. The contributing scenarios to core damage frequency of the unscreened rooms were explained.
The CDF was estimated by multiplication of the source fire frequency with the conditional core damage probability using the appropriate initiating event models from the WCGS IPE study. Both transient and fixed sources were included in the determination of the set of equipment failed by each postulated fire. The IPE model was modified by postulating fire induced failures and hot short failures, with consideration of suppression.

As described in Section 2.2.8, suppression was considered in the PRA of unscreened rooms by postulating an area of assured damage and an area of damage conditional on suppression system failure. The latter was typically all equipment in the compartment. The area of assured damage was generally estimated from the results of "generic" COMPBRN runs. The CDF was evaluated source by source in the unscreened compartments. A typical equation for CDF, for a source, was composed of two terms. One was the product of the source fire frequency, the conditional core damage probability (CCDP) associated with damaged equipment in the assured damage area, and the suppression probability (which was taken to be 0.95 for halon systems). The other was the product of the source fire frequency, the CCDP associated with postulating damage to all equipment in a compartment, and the non-suppression probability (which is 0.05 for halon systems).

The above was gleaned from the licensee's submittal with a great deal of difficulty, given that Section 4 of the licensee's submittal [1] was unclear.

2.2.11 Analysis of Containment Performance

a. Significant Containment Performance Insights

The submittal includes a plant-specific discussion of containment features to determine whether more than one train of equipment could be damaged by a single fire. A conclusion was reached that this is not a concern. Therefore, fires in the containment were not quantitatively addressed.

The containment performance review included discussions of both inside-containment fires and outside-containment fires that may affect containment functions. Containment integrity, bypass, heat removal and isolation were addressed. Containment fires were found to be of little significance, because (1) penetrations and hatches do not rely on active systems, (2) high to low pressure boundaries are protected by at least two non-fire susceptible valves, (3) the reactor coolant pump (RCP) oil collection design is not fire susceptible, (4) redundant cables have adequate separation, (5) no more than two of four containment fan coolers can be affected by a single fire, and (6) no more than one (of two or more) containment isolation valves in a line can be impacted by a single fire.

The submittal states that the impact of fire on containment performance is considered acceptable.

b. Plant-Unique Phenomenology Considered

The submittal did not address plant-unique phenomenology related to fire scenarios.

2.2.12 Treatment of Fire Risk Scoping Study Issues

a. Assumptions Used to Address Fire Risk Scoping Study Issues

All of the Sandia fire risk scoping study issues have been addressed in a manner consistent with the outline and guidance of Reference [17]. The licensee has presented a detailed discussion pertaining to each issue, as summarized below.

b. Significant Findings

1. Seismic-fire interactions have been addressed by examinations of: (a) the potential for a fire event resulting from an earthquake; (b) the potential for inadvertent seismic actuation of the fire suppression system and resulting effects on safety equipment; and (c) the potential for seismically induced failure of the fire protection system with respect to the effects on the SSEL. A fire walkdown of the plant has been undertaken for these examinations, as well as being included in the seismic walkdown. Hydrogen lines, lube oil equipment, fuel oil tanks and the RCP oil system have been identified as potential fire sources. For all cases, the relevant components were found seismically rugged.

   Inadvertent actuation of suppression systems was addressed in some detail. It was found that inadvertent actuation could not lead to safety equipment damage that was not already considered in the fire analysis. Halon suppression systems or drip-proof motors are used in those areas of the plant where the potential for safety-related equipment failure from water spray may exist.

   The discussion about seismic degradation of fire protection systems concentrated on installation of suppression system piping through safety-related areas. It was found that this piping is adequately supported against the SSE.

   Further discussion of these issues are found in Section 2.1.12 of this TER.

2. A detailed discussion has been provided related to fire barriers. Specific inspection procedures are instituted to verify the integrity of penetration seals, fire doors, and fire barriers. Fire dampers are inspected at least every 18 months.

3. Several procedures are implemented for fire reporting, fire brigade requirements, and brigade training and drills. The drills are reviewed and critiqued by observers.

4. Regarding fire impacts on operator effectiveness in carrying out required tasks associated with fires, the submittal noted that self-contained breathing apparatus (SCBA) equipment is available in the control room and at other locations, and emergency lighting is installed. Two procedures are implemented at WCGS for potential control room fires, one guides operator actions if the control room is not threatened, and one guides operator actions if it is. The submittal states that, per FIVE guidance, the non-thermal effects of the products of combustion are considered insignificant. However, fire brigade training includes encountering toxic and corrosive materials because of a fire. The control room fire PRA included the effect of smoke when considering the probability of abandoning the control room.

5. Control system interaction is addressed via the use of a remote shutdown panel and the capability to isolate Train B operation and control from the remote shutdown panel. Necessary control and monitoring functions are independent of the control room.

2.2.13 USI A-45 Issue

a. Methods of Removing Decay Heat

The IPEEE fire analysis uses logic models developed for the IPE, which include the entire array of heat removal capabilities of the plant. The systems most relied upon for decay heat removal are main feedwater, auxiliary feedwater and RHR.

b. Ability of the Plant to Feed and Bleed

The IPE model, used in the IPEEE, includes the provision for feed and bleed cooling. The submittal describes the methods used to remove decay heat for each fire induced initiating event.

c. Credit Taken for Feed and Bleed

Credit has been taken for feed and bleed capability.

d. Presence of Thermo-Lag

The WCGS uses Thermo-Lag fire wrap, but the fire damage calculations were performed as if the cables were not wrapped.

2.3 HFO Events

The WCGS IPEEE submittal reports no unduly significant sequences (i.e. vulnerabilities) with respect to HFO events. The submittal indicates that implementation of the screening approach described in Supplement 4 to Generic Letter 88-20, and in the guidance of NUREG-1407, is the basis for the conclusion that the plant is not vulnerable to HFO events.

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

1. High winds, external floods, and transportation and nearby facility accidents have been considered for analysis.

2. For each of the hazards identified in Step (1), plant modifications and modifications within 5 miles of the plant, since the time the operating license (OL) was issued, have been identified.

3. If changes have occurred, the current state of the hazard was evaluated for conformance to the updated SRP requirements (NUREG-0800).

4. Beginning with the list in NUREG/CR-2300 and using the suggested method, therein, a review of other potential plant hazards confirmed that no plant unique events were excluded from this evaluation.

5. The resulting HFO event analysis has been documented in the IPEEE submittal report and several WCGS internal documents.

The review of changes since the time of OL issuance has revealed that there are no new conditions that would significantly affect the plant design basis. The submittal states that the following types of changes were reviewed:

- Military and industrial facilities located within 5 miles of the site,
- On-site storage or other activities involving hazardous materials,
- Transportation characteristics near the plant site, and
- Modifications that might affect the plant response to floods, high winds and tornadoes.

Because WCGS was designed and constructed in compliance with 1975 SRP criteria, and because all reviews and walkdowns confirmed continued compliance with these criteria, no further analyses (e.g., bounding analyses) were performed.

2.3.1 High Winds and Tornadoes

2.3.1.1 General Methodology

The high wind and tornado review was based on the Final Safety Analysis Report (FSAR) of the plant. A walkdown was performed to verify or identify changes since the OL.

2.3.1.2 Plant-Specific Hazard Data and Licensing Basis

The design wind velocity for all Category-I structures is 100 mph, at 30-feet above ground level, for a 100-year recurrence interval. The design-basis tornado has a maximum wind velocity of 360 miles per hour at a cyclone radius of 150 feet, and a maximum 3 pounds per square inch atmospheric pressure drop reached at a rate of 2 pounds per square inches per second. The design basis missile spectrum conforms to the SRP requirements. Non-Category I structures (e.g., turbine building) are not designed to fully withstand these tornado loads, but are designed such that their demise does not adversely affect the Category I structures.

2.3.1.3 Significant Changes Since Issuance of the Operating License

The walkdown reviewed the power block and vicinity for vulnerabilities not identified in the design basis. None were found.

2.3.1.4 Significant Findings and Plant Unique Features

The evaluation, performed to compare the SRP requirements with the current state of the plant, confirmed that the WCGS conforms to the SRP criteria with respect to high winds and tornadoes.

2.3.2 External Flooding

2.3.2.1 General Methodology

The external flood review was based on the FSAR of the plant. A walkdown was performed to verify or identify changes since the OL.

2.3.2.2 Plant-Specific Hazard Data and Licensing Basis

The WCGS is located next to a cooling lake, fed via upstream dams, and drained via downstream dams. The plant is protected by dikes, drainage, and site grading. The probable maximum flood (PMF) still water level includes consideration of coincident wind generated waves, failure of upstream dams, and failure of downstream dams (for loss of water supply).

2.3.2.3 Significant Changes Since Issuance of the Operating License

The walkdown and evaluation concluded that there were no significant changes since the OL.

2.3.2.4 Significant Findings and Plant Unique Features

The walkdown and evaluation confirmed that the WCGS conforms to the SRP criteria with respect to floods. The new probable maximum precipitation (PMP) estimates were previously assessed in response to Generic Letter 89-22. The evaluation demonstrated that the revised PMP estimates would not result in on-site flood levels or roof ponding that would adversely affect safe operation.

2.3.3 Transportation and Nearby Facility Accidents

2.3.3.1 Methodology

The review of transportation and nearby facility accidents was based on the plant's FSAR and subsequent evaluations of hazards identified during the walkdown. The evaluation was performed by comparison of the WCGS design to the SRP requirements using the current hazard profile for off-site and on-site hazards. A walkdown was performed to verify or identify changes since the OL, both on-site and within five miles of the plant.

The evaluation addressed: (1) off-site and on-site explosions associated with water, road and rail accidents, as well as nearby stationary facilities; (2) off-site flammable vapor clouds; (3) aircraft crashes; (4) on-site and off-site toxic chemicals; (5) off-site fires; (6) water traffic on nearby waterways; and (7) on-site liquid spills.

2.3.3.2 Plant-Specific Hazard Data and Licensing Basis

The design basis of all transportation and nearby facility hazards conformed to the SRP requirements.

2.3.3.3 Significant Changes Since Issuance of the Operating License

The walkdown identified significant nearby industrial, transportation and military facilities within a five mile radius of the plant, and significant material storage hazards within a five mile radius of the plant. The walkdown also helped gather sufficient information to assess the impact of these hazards on safe operation of the WCGS. The probability of aircraft crashes at the site was updated to reflect current frequency of over-flights, take-offs, and landings.

2.3.3.4 Significant Findings and Plant Unique Features

The evaluation confirmed that the WCGS design conforms to the SRP criteria.

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

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

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

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

The ability to operate from the alternate shutdown panel is discussed in Section 4.6.7.3.1 and in Item 5 of Section 4.8.2 of the IPEEE submittal [1]. The treatment of spurious actuation by hot shorts is discussed in the licensee response to fire RAI number 1 [15] and Sections 4.0.2, 4.1.3.2.2 and 4.3.5 of the submittal.

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

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

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

Reference [18] identifies possible reduction of manual fire-fighting effectiveness and causing misdirected suppression efforts as the central issue in GSI-148.

The licensee's response to the issues regarding non-thermal effects of a fire and the ability of operators to function follows FIVE guidance, and is addressed in Section 4.8.2 of the submittal.

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

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

Common Cause Failures (CCFs) Related to Human Errors

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

All human errors in the IPEEE inherent to the use of the IPE model in the fire IPEEE are discussed in Section 4.3.5 of the submittal. Some human errors were adjusted to an HEP of unity to account for the location of a fire with respect to the human action. Human actions with respect to the SME success paths were described in Reference [15], in the response to seismic RAI number 1.

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

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

The seismic walkdown conducted for the WCGS SMA explicitly included spatial interactions as described in Section 3.2.1, and shown throughout Section 3.5, of the IPEEE submittal. The submittal discussed the effect of non-Seismic Category I structures on Seismic Category I structures and non-safety equipment proximity to safety equipment. Section 4.6.7.3.1 and Item 5 of Section 4.8.2 of the submittal address the issue of control room and alternate shutdown panel interactions.

Heat/Smoke/Water Propagation Effects from Fires

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

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

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

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

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

Random failures are treated by using the IPE model as discussed in Section 4.3.5 of the IPEEE submittal. Relevant information regarding heat, smoke and water effects are found in the discussion regarding FRSS issues (Section 4.8.2). Fire propagation to redundant trains is discussed relative to the inter-compartment fire analysis found in Sections 4.1.1.4 and 4.3.7. The design criteria of the plant mitigate against failure of multiple, redundant trains (see submittal Section 3.2.8).

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

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

The effects of fire suppression system actuation on safety-related equipment were addressed as part of the FRSS issues, and the treatment is described in Section 4.8.2 of the IPEEE submittal.

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

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

This issue was addressed with respect to safety-related equipment as part of the FRSS issues, and supplemented by the FRSS walkdown, as summarized in Sections 4.2.2.2 and 4.8.2 of the IPEEE submittal.

Seismically Induced Spatial and Functional Interactions

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

A focus of the seismic capability walkdown was on spatial interactions as discussed in Section 3.2.1, and shown throughout Section 3.5, of the IPEEE submittal.

Seismically Induced Fires

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

This was addressed as part of the FRSS issues, and supplemented by the FRSS walkdown, as discussed in Section 4.8.2 of the IPEEE submittal.

Seismically Induced Fire Suppression System Actuation

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

This was addressed as part of the FRSS issues, and supplemented by the FRSS walkdown, as discussed in Section 4.8.2 of the IPEEE submittal.

Seismically Induced Flooding

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

This was addressed as part of the FRSS issues, and supplemented by the FRSS walkdown, as discussed in Section 4.8.2 of the IPEEE submittal.

Seismically Induced Relay Chatter

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

- remain functional (i.e., without occurrence of contact chattering);

- be seismically qualified; or
- be chatter acceptable.

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

The licensee performed a relay chatter evaluation as part of the seismic IPEEE. Refer to Section 3.3.3.2 of the IPEEE submittal for a discussion of the evaluation.

Evaluation of Earthquake Magnitudes Greater than the Safe Shutdown Earthquake

Description of the Issue [18]: The concern of this issue is that adequate margin may not have been included in the design of some safety-related equipment. As part of the IPEEE, all licensees are expected to identify potential seismic vulnerabilities or assess the seismic capacities of their plants either by performing seismic PRAs or seismic margins assessments (SMAs). The licensee's evaluation for potential vulnerabilities (or unusually low plant seismic capacity) due to seismic events should address this issue.
The SMA performed by the licensee included a walkdown that used EPRI seismic margin methodology screening criteria appropriate for a 0.3g RLE.

Effects of Hydrogen Line Ruptures

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

The submittal addresses seismic robustness of hydrogen line ruptures as part of the FRSS issues in Section 4.8.2 of the IPEEE submittal.

3 OVERALL EVALUATION AND CONCLUSIONS

3.1 Seismic

The licensee performed a satisfactory reduced scope seismic IPEEE assessment with some features, notably the walkdown and a relay chatter evaluation, that are consistent with a focused scope assessment.

The walkdown was comprehensive and well documented, and the success paths were derived in a logical fashion, although without the benefit of a separate systems and element walkdown, as suggested in EPRI NP-6041. The walkdown was conducted against a 0.3g RLE screening criterion per a focused scope plant in order to demonstrate the seismic margin of the plant. Anchorage calculations were performed to help support screening judgments.

Strengths and Weaknesses

Based on this submittal-only review, the following items have been identified as the primary strengths of the seismic IPEEE for WCGS:

(1) The walkdown was performed against an RLE of 0.3g consistent with the requirements of a focused scope evaluation.

(2) The submittal is clear and well structured, particularly with respect to the walkdown approach and findings.

(3) The comparison of design basis demand and in-structure response spectra to those of the RLE was well done.

(4) The walkdown was comprehensive, including not only equipment and structures associated with the SSEL, but also additional equipment that might be useful in responding to an earthquake-induced plant challenge.

(5) The success paths were derived in a logical fashion from the IPE LOCA event tree, and the SSEL was developed using a variety of sources. It was further supplemented by the performance of an accident simulation using an operating crew.

(6) Screening judgments were documented with respect to the criteria of EPRI-NP 6041.

(7) A relay chatter evaluation was performed although not required for a reduced scope plant.

(8) The study has made use of plant and contractor personnel who received training in seismic evaluation methods

No significant weakness have been identified in the seismic IPEEE; however, the following observations should be noted:

- The submittal can be improved by providing a more detailed description of the guidance used by the SRT, the procedure it used, and the basis for screening out the emergency service water (ESW) pump anchorages in the ESW pump house.

- Noting that the turbine building contains a power feed cable from the switchyard to one of the ESF transformers, which is routed along the west wall through an electrical cabinet, the submittal can be improved by providing the location of anchorages of safe shutdown equipment that is either in the turbine building or depends on control or motive power from components located in the turbine building.

This review concludes that the licensee's seismic IPEEE is capable of identifying severe accident vulnerabilities consistent with a reduced scope effort; however, no vulnerabilities were identified.

3.2 Fire

The licensee has performed an extensive and detailed fire screening study followed by a fire probabilistic risk assessment. The licensee appears to be well aware of available methods, their uses, and their limitations. Assumptions, sensitivity studies and uncertainties are well presented. Accepted methods have been used, and they have generally been applied using recognized data. The licensee demonstrated an understanding of fire propagation/damage methods.

The fire PRA of unscreened compartments was performed conservatively, but with a less rigorous approach than the screening study. Fire scenarios based on individual fixed and transient combustible fire sources were developed for each compartment.

Cabinet fires were treated by assuming an area of certain damage around a cabinet, surrounded by an area of damage conditioned on the failure of suppression. This is significant because cabinets are the dominant ignition sources of the unscreened compartments. Fires that propagate from cabinets are assumed to damage all equipment and cables within a 10 to 20 foot radius, and automatic suppression is assumed to prevent damage beyond this radius, unless the system is unavailable.

It can be said that the licensee's fire IPEEE process is capable of identifying severe accident vulnerabilities, and none were found.

The following items are considered to be the primary strengths and weaknesses of the submittal. Note, the weaknesses identified are associated with lack of methodological rigor or inadequately supported assumptions; however, they do not invalidate the stated results of the study.

Strengths

(1) The fire screening analysis, using the FIVE methodology, was generally thorough, even employing COMPBRN IIIe to assess the potential extent of damage from fire ignition sources. Conclusions and results reached by the screening analysis are generally reasonable.

(2) Detailed discussion of fire related phenomena, methods, and assumptions demonstrated competent practice in the screening analysis.

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

(4) The PRA of unscreened compartments was, in general, conservatively performed, although it was performed with somewhat less rigor than the screening analysis.

(5) The conclusions are reasonable, and are within the range of results expected for an Appendix R compliant PWR with IEEE-rated cables.

Weaknesses

(1) The treatment of propagation of fire from cabinets was inconsistent. Propagation of fire from cabinets into the control room was assumed not to occur, whereas propagation from cabinets into other rooms was assumed possible.

(2) Calculations to determine the relative timing of damage and extinguishment were not performed before taking credit for the presence of automatic suppression systems. Damage radii around cabinets were assumed in the WCGS fire PRA. This damage radius assumption may be thought of as equivalent to an implicit time delay before automatic suppression. This implicit time delay was not quantified.

(3) The selection of transient combustibles did not include a flammable liquid spill.

(4) The submittal was difficult to follow, and generally unclear.

3.3 HFO Events

The HFO evaluation implemented the progressive screening method of NUREG-1407. The performance of verification walkdowns was appropriate. After identification of changes made since the OL, an evaluation to determine compliance with the current SRP criteria was performed. The screening methodology appeared to be followed correctly, and no significant weaknesses were noted during this review.

4 IPEEE INSIGHTS, IMPROVEMENTS, AND COMMITMENTS

4.1 Seismic

Using a focused scope walkdown, the licensee screened out all but five of the 740 components and structures considered in the seismic IPEEE as follows:

- RWST, based on judgment after review of the design basis
- Turbine building, based on judgment after review of design basis
- Four 60-cell batteries and racks, because of spacing between the batteries and rails
- Twelve LSELS/ESFAS cabinets, because they may not be securely fastened together at the RLE
- Strainers and screens, because of lack of data and guidance in EPRI NP-6041

These items were judged to have a capacity capable of withstanding the design basis earthquake. No vulnerabilities were identified.

Four areas were noted in which the actual field installation did not conform to the seismic design basis. These areas are:

- A transformer inside an inverter was not bolted to the frame on one side,

- Structural members or fire protection material was in close proximity to electrical cabinets and motor control centers in one instance,

- A victaulic coupling on a drain line in the diesel generator building was in close proximity to a motor control center, and

- Loose and missing bolts, and shims missing, on chillers.

These issues were identified on work requests for corrective action. In addition, several housekeeping items with respect to trash barrels and cabinets near safety-related equipment were identified on a Plant Improvement Request.

The submittal concludes that WCGS equipment and structures are "seismically very rugged" because of a conservative design, and the plant "has minimal vulnerability to a seismic event." In [19], the licensee further argues that because the WCGS design basis ground spectra are very close to the site specific 0.3g SME ground spectra and floor response spectra follow the results of the ground spectra, then the "seismic response spectra to be used for a site specific SME (0.3g) evaluation would be very similar to the seismic response spectra used for the WCGS design basis SSE." The licensee continues to argue that there is "considerable design margin with respect to seismic events at WCGS". The licensee concludes as follows: "As a result of this considerable design margin, evaluation of a 0.3g SME at Wolf Creek would be nearly identical to evaluating the current design basis earthquake at WCGS.

4.2 Fire

Overall, the licensee has concluded "that the WCGS is not vulnerable to fire." The total fire CDF of "unscreened" scenarios is assessed as 7.5×10^{-6}/ry, which is 15% of the internal event CDF. This frequency is within the range of fire-induced CDFs obtained for other nuclear power plants. This frequency is not to be confused with the total plant fire CDF, because approximately 125 screened-out fire areas are not included in this CDF estimate. The dominant contributors to the fire CDF are the ESF switchgear rooms and control area.

The submittal states that no plant modifications are required for adequate defense against fires. The licensee states that the study reaffirms the value of divisional separation, maintenance of fire boundaries and good housekeeping. The licensee has used the Nuclear Energy Institute's (NEI's) severe accident closure guidelines (NEI-91-04) to evaluate the need for plant improvements. The submittal states that no changes are considered necessary for the control room. It also states that any future recommendations regarding the two switchgear rooms (C-9 and C-10) and hot short conditions "will be formally provided to the appropriate WCGS groups."

4.3 HFO Events

The HFO evaluation confirmed that the plant continues to conform to the SRP criteria. Accordingly, no vulnerabilities were reported.

5 IPEEE EVALUATION AND DATA SUMMARY SHEETS

Completed data entry sheets for the WCGS IPEEE are provided in Tables 5.1 to 5.3. These tables have been completed in accordance with the descriptions in Reference [14]. Table 5.1 lists the overall external event results. Table 5.2 summarizes general seismic data pertaining to the reduced-scope seismic evaluation. Table 5.3 provides the PWR Seismic Success Paths table, which gives a description of the success paths developed for the focused-scope seismic evaluation. The IPEEE submittal does not provide sufficient information regarding core damage sequences and system failures pertaining to the fire event analysis; hence, no data summary tables are provided pertaining to the fire evaluation. In addition, no PRA or bounding analysis was performed as part of the WCGS HFO events IPEEE; hence, no data summary tables are provided pertaining to evaluation of these external initiators.

Table 5.1
External Events Results

Plant Name: WCGS

Event	Screening	CDF	Plant HCLPF(g)	Notes
External Fire	O			
External Flooding	O			
Extreme Winds	O			
Internal Fire	S	7.5×10^{-6}/ry		
Nearby Facility Accidents	O			
Seismic Activity	S		At least 0.2g	
Transportation Accidents	O			
Others	O			

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

Table 5.2
SMM Seismic Fragility

Plant Name: WCGS

Review Level Earthquake (g): 0.3g (Focused-Scope Evaluation)

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

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

Component	HCLPF (g)	Seismic Sequence Description	Seismic Success Path Description
LSELS/ESFAS Cabinets (not bolted together)	0.2	S3 with LOOP	Injection with either closed-loop cooling via steam generators and RHR or feed and bleed cooling with recirculation from sump.
Battery Racks (spacing between batteries and rack frame)	0.2	S3 with LOOP	Injection with either closed-loop cooling via steam generators and RHR or feed and bleed cooling with recirculation from sump.
Strainer/Screens (lack of data for assignment of higher capacity)	0.2	S3 with LOOP	Injection with either closed-loop cooling via steam generators and RHR or feed and bleed cooling with recirculation from sump.
Turbine Building	0.2	S3 with LOOP	Injection with either closed-loop cooling via steam generators and RHR or feed and bleed cooling with recirculation from sump.
RWST	0.2	S3 with LOOP	Injection with either closed-loop cooling via steam generators and RHR or feed and bleed cooling with recirculation from sump.

Table 5.3
PWR Seismic Success Paths

Plant Name: WCGS

CHALLENGE	STRATEGY	SUCCESS PATH	RX		PRIMARY INTEGRITY					PRIMARY INVENTORY-INJECTION						PRIMARY INVENTORY-RECIRC					SECONDARY INTEGRITY					SECONDARY INVENTORY							CONTAINMENT									NOTES
			RPS	BI	PPORV	PSRV	PAD1	PAD2	RCPS	CHP1	HP1	LPI	ACC	AI1	AI2	CHPR	HPR	LPR	AR1	AR2	SGS	SGA	TT	MSIV	TB	SG	MFW	NISP	AFW	AM1	AM2	AM3	CS1	CS2	FC1	FC2	IC	CI1	CI2	IGN	RF	
		P	X							X	X								X	X		X				X			X						X			X				Preferred Path
		A	X		X					X	X							X		X															X			X				Alternate Path

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

Acronym of Support Systems: AC, ACBU1, ACBU2, ACBU3, AUXC2, AUXC3, AUXC4, CCW, DC, EAC, EDC, ESAS1, ESAS2, ESW, HVAC1, HVAC2, HVAC3, IA, NIT, OA3, 0A4, SA, STM, SW2, SW3, SW4, VAC

1,2,3...How many needed to operate   H = Human action required   T = Must be throttled/controlled
For Core Damage Prevention Challenges, show only hardware whose failure is modeled as contributing to core damage.

6 REFERENCES

1. "Wolf Creek Generating Electric Station Individual Plant Examination of External Events (IPEEE)," Wolf Creek Nuclear Operating Corporation, June 27, 1995.

2. "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," U. S. Nuclear Regulatory Commission, Generic Letter 88-20, Supplement 4, June 28, 1991.

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

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

5. "Probabilistic Seismic Hazard Evaluation at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Issue," Electric Power Research Institute, EPRI NP-6395-D, April 1989.

6. "Seismic Hazard Characterization of 69 Nuclear Power Plant Sites East of the Rocky Mountains," U.S. Nuclear Regulatory Commission, NUREG/CR-5250, January 1989.

7. "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," U.S. Nuclear Regulatory Commission, NUREG-1488, October 1993.

8. Nuclear Energy Institute, letter from W. Rasin to A Thadani (NRC), "NE White Paper, 'Justification for Reduction in IPEEE Program based on Revised LLNL Seismic Hazard Results,'" April 5, 1994.

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

10. E-mail message from Nilesh Chokshi to Alan Rubin, Brad Hardin, and John Ridgely, "Wolf Creek Paragraph," U.S. Nuclear Regulatory Commission, November 13, 1999.

11. "Fire Events Database for U.S. Nuclear Power Plants," Electric Power Research Institute, NSAC-178L

12. R.T. Sewell, et al., "Individual Plant Examination for External Events: Review Guidance," Energy Research, Inc., ERI/NRC 94-501, Draft, May 1994.

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

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

15. "Response to Request for Additional Information," Attachment to Letter from O. L. Maynard of WCNOC to USNRC, WM 97-0142, December 8, 1997.

16. "Enhancing the NRC and EPRI Seismic Margin Review Methodologies to Analyze the Importance of Non-Seismic Failure, Human Errors, Opportunities for Recovery, and Large Radiological Releases," U. S. Nuclear Regulatory Commission, NUREG/CR-5679, August 1990.

17. "Fire Induced Vulnerability Evaluation (FIVE)," Electric Power Research Institute, TR-100370, Revision 1, September 29, 1993.

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

19. Wolf Creek Nuclear Operating Corporation, letter from Richard A. Muench to U.S. Nuclear Regulatory Commission, "Docket Np. 50-482: Response to Request for Additional Information Regarding Wolf Creek Individual Plant Examination of External Events Submittal", June 11, 1999.