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Date: September 15, 1992
Refer to: A1:92-330

Ms. Christiana H. Lui
US Nuclear Regulatory Commission
Mail Stop NL/S-372
Washington, DC 20555

Dear Christiana:

This letter is the formal version of the one I faxed you in July. Milestones for FY93 and some for FY94 have been added to those previously agreed to for FY92. They include Item F for Task 2 and all those for Tasks 3 and 4. The abbreviation SA/UA refers to sensitivity analysis and uncertainty analysis.

Task 1: Familiarization with MACCS and making it operational on LANL SUN computers.

A. Become familiar with MACCS and run sample problems on LANL Sun computers. 11/91
B. Under the direction of NRC and SNL, define a standard problem to be used for MACCS testing of SA/UA procedures. 6/92.
C. Implement and test the standard problem on LANL computers. 7/92

Task 2: Evaluation of Existing Techniques.

A. Produce a short, critical review of SA/UA to serve as a starting point for method evaluation and development. 11/91
B. Perform literature review of current methods. 3/92
C. Produce a description of existing SA/UA methodology, including taxonomy of models, which synthesizes current practices and is expected to point out the need for development of new and improved methods. 6/92.
D. Implement current methods, as necessary, for comparison purposes with new methods using MACCS and other models, to be determined. 7/92
E. Generate data to be used for comparison with and evaluation of new methods. 9/92
F. Provide a letter report. The report will collect previously written documents, present any additional important results and provide a short summary of the evaluation. 12/92

Task 3: Development and Demonstration of Improved Uncertainty and Sensitivity Techniques.

A. Formulate an initial classification of models relative to the problem of screening and sub-model assessment. 12/92
B. Summarize alternative statistical measures and techniques that might be used in the screening and validation steps of the prioritization process. 3/93

C. Perform preliminary evaluation of alternative screening and validation methods for uncertainty analysis. 7/93
D. Provide a letter report. The report will collect previously written documents, present any additional important results and provide a short summary of the research effort. 12/93

Task 4: Development and Demonstration of Uncertainty and Sensitivity Techniques for Code Submodels.

A. Make recommendations for assessing uncertainty due to submodels. 10/93

Task 7: Technical Support.

A. Provide, in a letter report, comments on the SNL methodology for parameter and model prioritization. 12/91
B. Provide any additional comments and observations following the January 8, 1992 meeting. 1/92

Sincerely,

Michael D. McKay

cy: LaRon L. Smith, N-DO, MS E561
Mark A. Cunningham, NRC
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

June 28, 1991

To All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities

SUBJECT: INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR SEVERE ACCIDENT VULNERABILITIES - 10CFR 50.54(f) (Generic Letter No. 88-20, Supplement 4)

1. Summary

In the Commission policy statement on severe accidents in nuclear power plants issued on August 8, 1985 (Ref. 1), the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on any regulatory requirements for these plants. However, the Commission recognizes, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents which could be fixed with low-cost improvements. As a key part of the implementation of the policy statement, the staff issued Generic Letter 88-20 (Ref. 2) on Nov. 23, 1988, requesting that each licensee conduct an individual plant examination (IPE) for internally initiated events only.

Current risk assessments indicate that the risk from external events could be a significant contributor to core damage in some instances. The staff, however, delayed the issuance of the request for a systematic individual plant examination for severe accidents initiated by external events (IPEEE) to allow the staff to carry out additional work to (1) identify which external hazards need to be evaluated, (2) identify acceptable examination methods and develop procedural guidance, (3) coordinate with other ongoing external event programs, and (4) conduct a workshop to explain the IPEEE process and to obtain comments and questions on the draft generic letter supplement and associated guidance document. The staff has completed this work and has revised this supplement and the guidance document (Ref. 3) and is now requesting that each licensee perform an individual plant examination of external events to identify vulnerabilities, if any, to severe accidents and report the results together with any licensee-determined improvements and corrective actions to the Commission.

The general purpose of the IPEEE is similar to that of the internal event IPE--that is, for each licensee (1) to develop an

appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at its plant under full power operating conditions, (3) to gain a qualitative understanding of the overall likelihood of core damage and radioactive material release, and (4) if necessary, to reduce the overall likelihood of core damage and radioactive material releases by modifying hardware and procedures that would help prevent or mitigate severe accidents. It must be emphasized that for the IPEEE the key outcome is the knowledge and appropriate improvements resulting from such an examination which can be conducted using any of the approaches discussed below or an alternate approach, if acceptable to the NRC. Besides the completion of the IPEEE, closure of severe accident concerns involves the completion of the internal event IPE, including applicable items resulting from the Containment Performance Improvement (CPI) Program, and future NRC and industry efforts in the areas of accident management. Additional discussion is provided in SECY-88-147 (Ref. 4) on the interrelationships among these three areas and the role they play in closure of severe accident issues for operating plants.

Therefore, consistent with the Commission's Severe Accident Policy Statement and pursuant to 10 CFR 50.54(f), licensees are requested to perform an IPEEE for plant-specific severe accident vulnerabilities initiated by external events and to submit the results to the NRC. NUREG-1407, which is enclosed, provides additional guidance for the performance and submittal of the IPEEE. (It is not the intent of NUREG-1407 to go beyond the information request contained in this generic letter supplement.)

2. Examination Process

The examination process for the IPEEE, in general, is similar to that for the internal event IPE (Ref. 2). Basically, the event/fault trees from the internal event IPE can be extended for external event PRAs, or used to identify important equipment for other acceptable evaluation methods, for instance, the seismic margin methodology. As in the internal event IPE:

(1) The quality and extent of the results derived from an IPEEE will depend on the vigor with which the licensee applies the method of examination and on the licensee's commitment to the intent of the IPEEE.

(2) The maximum benefit from the IPEEE would be realized if the licensee's staff were involved in all aspects of the examination; that involvement would facilitate integration of the knowledge gained from the examination into operating procedures, training programs, and appropriate hardware changes.

Therefore, each licensee is requested to use its staff to the maximum extent possible in conducting the IPEEE, by participating in the analysis and technical review, and by validating both the process and its results through a peer review by individuals who are not associated with the initial evaluation.

3. Identification of External Hazards

The external events to be considered, consistent with past PRAs, are those events whose cause is external to all systems associated with normal and emergency operation situations. A comprehensive list of external events can be found in NUREG/CR-2300, "PRA Procedures Guide" (Ref. 5). Some external events listed may not pose a significant threat of a severe accident. Some external events may have been considered at the design stage and have sufficiently low contribution to core damage frequency or plant risk. Some events may have been or will be reviewed under ongoing programs; for instance under IPE, the significance of lightning and severe cold weather conditions that could cause loss of offsite power will be assessed. Also, internal floods have been included in the internal event IPE request (Ref. 2). Based on staff's evaluation of References 6 through 8, the staff recommends that only five events be included in the IPEEE. However, licensees should confirm that no plant-unique external events known to the licensee with the potential to initiate severe accidents are excluded from the IPEEE. For example, volcanic activities should be assessed as part of the IPEEE process at plant sites in the vicinity of active volcanoes, and lightning effects should be assessed as part of the IPEEE process at those sites where, based on past operating experience, lightning strikes may fail equipment in addition to causing partial or complete loss of offsite power, (i.e., affecting safety-related instrumentation and control systems). The five external events requested to be assessed include:

1. Seismic Events
2. Internal Fires
3. High Winds and Tornadoes
4. External Floods
5. Transportation and Nearby Facility Accidents

A detailed discussion regarding the evaluation of external hazards can be found in NUREG-1407 and References 6 through 8.

4. Examination Methods

The NRC has identified the following approaches (details are provided in NUREG-1407) as being acceptable for the examination requested by this letter. However, the NRC recognizes that other methods capable of identifying plant-specific vulnerabilities to severe accidents due to external events may exist. The staff will review any systematic examination methods proposed to

determine their acceptability for IPEEE. A brief discussion of the staff identified approaches is provided below:

4.1 Seismic Events. A seismic IPEEE can be accomplished by performing a seismic probabilistic risk assessment (PRA) with enhancements or by using one of two seismic margin methods with enhancements.

The seismic PRA should be at least a Level 1 plus a containment performance analysis that uses current methods and plant information. Containment performance analysis guidance is provided in Appendix 2. The containment performance analysis should concentrate on identifying seismically induced vulnerabilities and sequences different from those obtained from the IPE. The staff considers the procedures described in NUREG/CR-2300 (Ref. 5), NUREG/CR-2815 (Ref. 9), and NUREG/CR-4840 (Ref. 10) to be adequate for the seismic IPEEE, provided the enhancements discussed in Appendix 1 of this generic letter are also included. The staff prefers that licensees use both mean (arithmetic) hazard curves (Refs. 11 and 12) developed by the Lawrence Livermore National Laboratory (LLNL) and the Electric Power Research Institute (EPRI), if available, in performing the PRA, since this will help to focus on the delineation of dominant sequences rather than on the bottom line numbers. If a licensee chooses to perform only one analysis, then the higher of the two mean (arithmetic) hazard estimates should be used.

Two seismic margins methods (SMMs) with enhancements, one developed by NRC and the other developed by EPRI, can also be used for the seismic IPEEE. However, the SMMs in their current form are not suitable for plant sites located in areas of high seismicity. For the remaining sites, a graded review approach (full scope, focused scope, and reduced scope) is defined (see NUREG-1407). The lists of review level earthquakes (RLEs) and review scope defined by the staff for all U.S. sites, and for use in SMMs, are presented in Appendix 3. The RLE does not represent a safety adequacy criterion or a threshold of vulnerability for the individual plant. The RLE is intended as a reporting criterion if the plant capacity is lower than the specific RLE. Detailed descriptions of the seismic margins methods can be found in NUREG/CR-4334 (Ref. 13), NUREG/CR-4482 (Ref. 14), NUREG/CR-5076 (Ref. 15), and EPRI NP-6041 (Ref. 16). The requested enhancements are discussed in NUREG-1407 and summarized in Appendix 1 to this generic letter.

4.2 Internal Fires. Fire initiated events can be treated by performing a Level 1 fire PRA as described in NUREG/CR-2300 or a simplified fire PRA as described in NUREG/CR-4840 (Ref. 10). The COMPBRN code can be used to model fire

propagation, provided that the shortcomings identified in Ref. 17 are addressed. When the licensee assesses the effectiveness of manual fire fighting, it should use plant-specific data from fire brigade training to determine the response time of the fire fighters. The effectiveness of fire barriers should be assessed, and the use of separation in determining fire zones critically examined. The walkdown procedures should be specifically tailored to assess the remaining issues identified in the Fire Risk Scoping Study (Ref. 17): (1) seismic/fire interactions, (2) effects of fire suppressants on safety equipment, and (3) control system interactions for severe accident vulnerabilities. Containment performance (Appendix 2) should be assessed to determine if vulnerabilities stemming from sequences that involve containment failure modes distinctly different from those obtained in the internal event analyses are predicted.

An alternative fire vulnerability evaluation (FIVE) method is under review by the staff at this time, and may become a viable option for the treatment of fire in the IPEEE.

4.3 High Winds, Floods, and Transportation and Nearby Facility Accidents. A screening type approach as shown in Figure 1 can be used to evaluate the impact of high winds, external floods, and transportation and nearby facility accidents. The steps shown in Figure 1 represent a series of analyses in increasing level of detail, effort, and resolution. The licensee should first determine if the 1975 Standard Review Plan (SRP) criteria (Ref. 18) are met. If the plant does not meet the 1975 SRP criteria, the licensee should examine further using the recommended optional steps. However, licensee may choose to bypass one or more of the optional steps, provided that vulnerabilities are either identified or proved to be insignificant. Again, the containment performance should be assessed to determine if vulnerabilities and sequences different from those obtained from the internal event analyses are predicted.

The application of the above approaches involves considerable judgment with regards to the requested scope and depth of the study, level of analytical sophistication, and level of effort to be expended. This judgment depends on how important the external initiators are likely to be compared with internal initiators, and a perceived need for accurately characterizing plant capacity or core damage frequency. The detailed guidelines presented in NUREG-1407 do not preclude use of this type of judgment. Consistent with engineering practice, expert opinions, simplified scoping studies, and bounding analyses (which should be documented), are expected to be used, as appropriate, in forming these judgments. At sites that have multiple units, some utilities may wish to reduce their review scope after completing the initial IPEEE plant evaluation. The licensee should discuss

any proposed reduction in the scope of the IPEEE with the NRC on a case-by-case basis.

5. Coordination with Other External Event Programs

Three programs, i.e., (1) the external event portion of USI A-45, (2) GI-131, and (3) the Eastern U.S. Seismicity Issue (formerly called the Charleston Earthquake Issue), are subsumed in the IPEEE. A brief discussion of these programs is provided below:

- USI A-45, "Shutdown Decay Heat Removal Requirements": USI A-45 had the objective of determining whether the decay heat removal function at operating plants is adequate and if cost-effective improvement can be identified. A part of the USI A-45 activities consists of assessing the adequacy of the decay heat removal system (DHR) to deal with external events initiators. This aspect of the DHR issue should be specifically addressed in the review of the IPEEE. The external event insights obtained from the USI A-45 study on five plants are presented in GL 88-20 (Ref. 2).

- GI 131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants": GI 131 (Ref. 19) deals with the seismically induced failure of the flux mapping transfer cart that would lead indirectly to the rupture of instrumentation tubes at the seal table. This could lead to core damage if loss of coolant through the ruptured instrumentation tubes is combined with unavailability of other mitigating systems. This scenario is applicable only to Westinghouse plants. Affected plants should explore the potential for this scenario and achieve a resolution of this concern through the IPEEE.

- The Eastern U.S. Seismicity (The Charleston Earthquake) Issue: As a result of work carried out by the NRC, LLNL, and EPRI to resolve the Charleston Earthquake Issue, probabilistic seismic hazard estimates (Refs. 11 & 12) exist for all nuclear power plant sites east of the Rocky Mountains. These estimates can be used directly by any licensee opting to satisfy the seismic IPEEE by means of a seismic PRA. The NRC/LLNL and EPRI work in this area also played a key role in determining the review level earthquakes to be used in the seismic margin option. The IPEEE will provide a resolution of the Eastern U.S. Seismicity issue without the need for utilities to perform any additional work.

Other external event programs listed below are either resolved or nearing completion. Their plant-specific implementation may require a plant-specific examination, which should be coordinated with the IPEEE to minimize unnecessary duplication of examination and review efforts.

- USI A-17, "System Interactions in Nuclear Power Plants," USI A-40, "Seismic Design Criteria, A Short-Term Program," and USI A-46 "Verification of Seismic Adequacy of Equipment in Operating Plants,": The scope of USI A-46 has been expanded to contain the seismic spatial system interaction of USI A-17 and the seismic capability of safety tanks of USI A-40 (NUREG-1407). The USI A-46 review is required on approximately 70 operating plants, which constitute a subset of all the nuclear power plants that are expected to perform an IPEEE. USI A-46 should be coordinated with the IPEEE so that the objectives of both activities may be accomplished with a single walkdown effort. (Both A-46 plants and non-A-46 plants will address spatial interactions within the IPEEE program through the seismic walkdown, which is guided by the EPRI methodology.)

- NUREG/CR-5088, "Fire Risk Scoping Study" and GI 57, "Effects of Fire Protection System Actuation on Safety-Related Equipment": The licensee should address the fire issues identified in the Fire Risk Scoping Study (Ref. 17) as discussed in Section 4.2 in NUREG-1407. However, it should be noted that additional research related to GI 57 is being performed in parallel with the IPEEE to obtain more rigorous and realistic estimates of risk; this research may identify other potential vulnerabilities. A specifically tailored walkdown for potential fire vulnerabilities should enable the licensee to collect information related to GI 57. Licensees may propose corrective measures that could resolve some or all of the GI 57 concerns.

If, during its IPEEE, a licensee (1) discovers a potential vulnerability that is topically associated with any other USI or GI and proposes measures to dispose of the specific safety issue, or (2) concludes that no vulnerability exists at its plant that is topically associated with any USI or GI, the staff will consider the USI or GI resolved for a plant upon review and acceptance of the results from the IPEEE. The licensee's IPEEE submittal should specifically identify which USIs or GIs it is proposing to resolve.

6. Severe Accident Sequence Selection

In performing an IPEEE using a PRA, it is essential to screen for potentially important severe accident sequences. The screening criteria that should be used to determine which of the potentially important sequences that lead to core damage or unusually poor containment performance, should be reported to the NRC with your IPEEE results, are listed in Appendix 3 of this generic letter.

If a seismic margin method is used in the IPEEE, the licensee should report all functional sequences and success paths considered in the analysis and their associated high confidence-low probability of failure (HCLPFs) values. In addition, the licensee should report all HCLPFs related to containment and containment systems performance. A HCLPF value lower than the specified review level earthquake (RLE) does not necessarily represent a plant vulnerability. The licensee should assess the significance of HCLPF values lower than the RLE and take any actions that are deemed appropriate.

NUREG-1407 describes the documentation needed for the accident sequence selection and the intended disposition of these sequences. A summary is provided in Appendix 4.

7. Use of IPEEE Results

Licensee

It is expected that the licensee will move expeditiously to correct any vulnerabilities that it determines warrant correction. Information on changes initiated by the licensee should be documented in accordance with the requirements of 10 CFR 50.59 and 10 CFR 50.90. Changes should also be reported in the IPEEE submittal (including reference to any previous submittal under 10 CFR 50.59 or 10 CFR 50.90) in response to this letter.

NRC

The NRC will evaluate licensee IPEEE submittals and will serve as a clearing house to disseminate all important IPEEE findings. These evaluations are intended to obtain reasonable assurance that the licensee has adequately analyzed the plant design and operations to discover instances of particular vulnerability to core damage or unusually poor containment performance given a core damage accident. Further, the NRC will assess whether the conclusions the licensee draws from the IPEEE regarding changes to the plant systems or components are adequate. The consideration will include both quantitative measures and nonquantitative judgment. The NRC consideration may lead to one of the following assessments:

1. If NRC consideration of all pertinent and relevant factors indicates that the plant design or operation does not meet the facility's current licensing basis, then appropriate actions will be required consistent with the Commission's rules and regulations.

2. If NRC consideration indicates that plant design or operation could be enhanced by substantial additional protection beyond NRC regulations, appropriate enhancement

will be recommended and supported with backfit analysis in accordance with 10 CFR 50.109.

3. If NRC consideration indicates that the plant design and operation meet NRC regulations and that further safety improvements are not substantial or are not cost effective, enhancements would not be required.

8. Accident Management

Licensees need not develop an accident management plan as an integrated part of the IPEEE. Licensees should plan to incorporate the results of the IPEEE and other relevant information into their accident management plans at a future date. Nevertheless, the IPEEE process may identify operator or other plant personnel actions that can substantially reduce the risk from severe accidents at the plant and that the licensee believes should be immediately implemented in the form of emergency operating procedures or similar formal guidance. The staff encourages each licensee to not defer implementing such actions until a more structured and comprehensive accident management program is developed on a longer schedule, but rather to implement such actions within the constraints of 10 CFR 50.59. These actions can be integrated later into the plant's accident management program.

9. Documentation of Examination Results

The IPEEE should be documented in a traceable manner to provide the basis for the findings. This can be dealt with most efficiently by a two-tier approach. The first tier consists of the results of the examination, which will be reported to the NRC. The second tier is the documentation of the examination itself, which should be retained by the licensee for the duration of the license. A summary of the documentation format and content is provided in Appendix 4 of this generic letter.

10. Licensee Response

Licensees are requested to submit within 180 days from the issuance date of this generic letter a response which describe their proposed programs for completing the IPEEEs. The proposal should:

1. Identify the methods and approach selected for performing the IPEEE,
2. Describe the method to be used if it has not been previously submitted for staff review (the description may be by reference), and
3. Identify the milestones and schedule for performing the IPEEE, and submitting the results to the NRC.

Meetings with NRC during the examinations will be scheduled as needed to discuss subjects raised by licensees and to provide necessary clarifications.

Licensees are requested to submit the IPEEE results within three years from the issuance date of this generic letter (Supplement 4 to Generic Letter 88-20). The NRC encourages those plants that have not yet undergone any systematic examination for severe accidents to promptly initiate the examination.

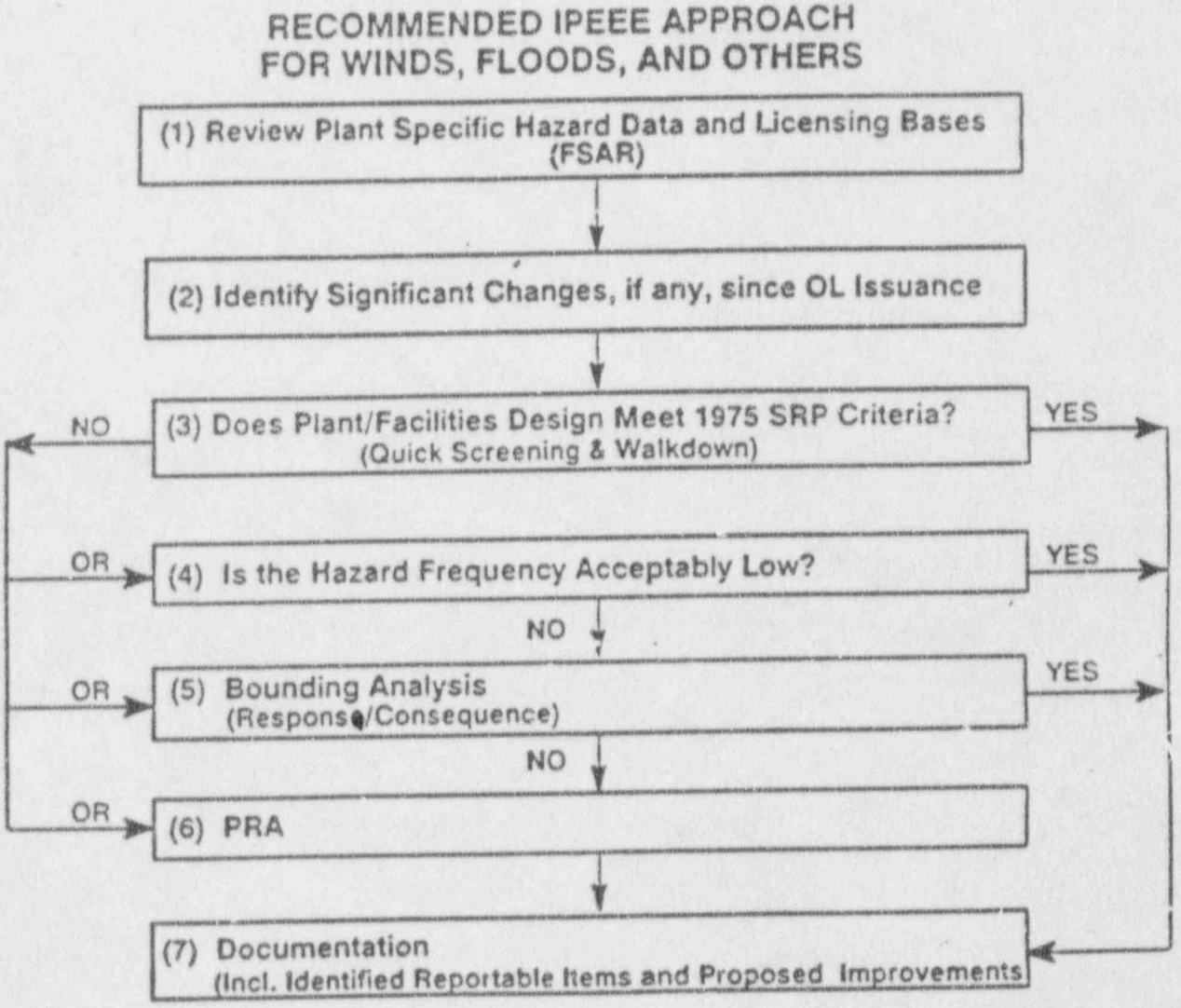
11. Regulatory Basis

This letter is issued pursuant to Section 182a of the Atomic Energy Act and 10 CFR 50.54(f). A 10 CFR 50.54(f) analysis is provided in the Appendix 5. Accordingly, all responses should be under oath or affirmation. This request for information is covered by the Office of Management and Budget under an Interim Clearance No. 3150-0011, which expires on June 30, 1991. The estimated average burden would not exceed 6 person-years per licensee response (Appendix 5) over a 3-year period, including assessing the request, searching data sources, gathering and analyzing the data, and preparing the IPEEE reports. A value/impact analysis for the implementation of the IPEEE is provided in the attachment to Appendix D of NUREG-1407. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management, Room 3208, New Executive Office Building, Washington, DC 20503.

James G. Partlow, Associate
Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Appendices 1 through 6
2. NUREG-1407

Figure 1


RECOMMENDED IPEEE APPROACH
FOR WINDS, FLOODS, AND OTHERS
(1) Review Plant Specific Hazard Data and Licensing Bases (FSAR)
(2) Identify Significant Changes, if any, since OL Issuance
(3) Does Plant/Facilities Design Meet 1975 SRP Criteria? (Quick Screening & Walkdown)
NO
YES
OR
(4) Is the Hazard Frequency Acceptably Low?
YES
NO
OR
(5) Bounding Analysis (Response/Consequence)
YES
NO
OR
(6) PRA
(7) Documentation (Incl. Identified Reportable Items and Proposed Improvements

APPENDIX 1
SUMMARY OF SEISMIC IPEEE METHODOLOGY ENHANCEMENTS

The following guidelines provide some specifics that are needed in a PRA, in a supplement to an existing PRA, or in the seismic margins method for an IPEEE submittal. A detailed discussion of these enhancements is presented in NUREG-1407.

New PRA:

Perform a plant walkdown following the procedures described in the EPRI seismic margin report (Ref. 16).

Perform an assessment of relay chatter effects in accordance with scope and procedure described in NUREG-1407.

Perform soil analysis, if needed, using procedures described in NUREG-1407.

Calculate the high confidence of low probability of failure (HCLPF) values for components, sequences, and the plant (optional).

Existing PRA:

Include the enhancements noted above for new PRA and add the following if not considered previously:

Perform sensitivity studies to determine if the use of LLNL or EPRI mean hazard estimates would affect the delineation and ranking of sequences.

Perform a supplementary analysis of nonseismic failures and human actions.

Perform containment performance assessment.

NRC SMM:

Perform an assessment of relay chatter effects in accordance with scope and procedures described in NUREG-1407.

Perform soil analysis, if needed, using procedures described in NUREG-1407.

Perform an analysis of nonseismic failures and human actions using procedures described in NUREG-1407.

Perform a walkdown and prepare its documentation in accordance with EPRI's recommendations (Ref. 16).

Evaluate containment and containment system performance.

EPRI SMM:

Select an alternative path so that it involves to the maximum extent possible systems, piping runs, and components that are different from the preferred success path.

Perform an analysis of nonseismic failures and human actions using procedures described in NUREG-1407.

Evaluate containment and containment systems performance.

APPENDIX 2
CONTAINMENT PERFORMANCE

The protection of public safety from any hazard of nuclear power plants has been fostered by applying the "defense-in-depth" principle, which relies on a set of independent barriers to fission product release to the environment. The containment and its supporting systems comprise one of these barriers.

The evaluation of the containment performance for external events should be directed toward a systematic examination of whether there are sequences that involve containment failure modes distinctly different from those found in the IPE internal events evaluation or contribute significantly to the likelihood of functional failure of the containment (i.e., loss of containment barrier independent of core melt). It should recognize the role of mitigating systems, and should ultimately result in the development of accident management procedures that could both prevent and mitigate the consequences of the severe accidents. The most efficient way to accomplish this is to use the information developed for the IPEEE to:

1. Identify mechanisms that could lead to containment bypass,
2. Identify mechanisms that could cause failure of the containment to isolate, and
3. Determine the availability and performance of the containment systems under the external hazard to see if they are different from those evaluated under the internal event evaluation.

Additional guidance on the containment performance associated with external events can be found in NUREG-1407.

Licensees are expected to evaluate the insights learned from CPI programs as discussed in References 20 & 21 and determine their applicability to external events.

APPENDIX 3
CRITERIA FOR REPORTING IMPORTANT SEVERE ACCIDENT SEQUENCES

The licensee should use the reporting criteria described in Generic Letter 88-20 for PRA analysis to determine which potentially important functional sequences and functional failures that might lead to core damage or unusually poor containment performance should be reported to the NRC in the IPEEE submittal. The licensee should use the reporting criteria described in NUREG-1335 (Ref. 22) to report systemic sequences to the NRC. These criteria do not represent a threshold for vulnerability.

If a seismic margin method is used in the IPEEE, the licensee should report in accordance with NUREG-1407 all functional sequences and success paths considered in the analysis and their HCLPFs. The review level earthquakes (RLEs) for all applicable U.S. sites are presented in Tables 3.1 and 3.2. In addition, the licensee should report all HCLPFs related to containment and containment systems performance. A HCLPF value lower than the specified review level earthquake (RLE) does not necessarily represent a plant vulnerability. The licensee should assess the significance of HCLPF values lower than RLE and take any necessary actions and make other improvements that are deemed appropriate by the licensee.

TABLE 3.1

REVIEW LEVEL EARTHQUAKE - PLANT SITES EAST OF THE ROCKY MOUNTAINS

Reduced Scope

Big Rock Point	Duane Arnold*	South Texas	Turkey Pt.
Comanche Peak	Grand Gulf	St. Lucie	Waterford
Crystal River	River Bend		

0.3g Focused Scope

Arkansas #2	Dresden	Limerick	Quad Cities
Beaver Valley	Farley	McGuire	Salem
Bellefonte	Fermi	Millstone	Shoreham
Braidwood	Fitzpatrick	Monticello	Summer*
Browns Ferry	Fort Calhoun	Nine Mile Pt.	Surry
Brunswick	Ginna	North Anna*	Susquehanna
Byron	Haddam Neck	Oyster Creek	Three Mile Is.
Callaway	Harris	Palisades	Vermont Yankee
Calvert Cliffs	Hatch	Peach Bottom	Vogtle
Catawba*	Hope Creek	Perry	Watts Bar
Clinton	Kewaunee	Point Beach	Wolf Creek
Cook	LaSalle	Prairie Island	Zion
Cooper			
Davis-Besse			

0.3g Full Scope

Arkansas #1	Maine Yankee	Robinson	Yankee Rowe
Indian Point	Oconee*	Sequoyah	

Committed to Perform a Seismic PRA**

Pilgrim	Seabrook

NOTES:

* Special attention to shallow soil conditions is appropriate for these locations (see NUREG-1407, Section 3.2.2 and Appendix A).

** Relay chatter evaluation should be similar to a full-scope review.

TABLE 3.2

REVIEW LEVEL EARTHQUAKE - WESTERN UNITED STATES PLANT SITES

0.5g*

Trojan
Washington Nuclear
Rancho Seco
Palo Verde

Seismic Margin Methods Do Not Apply To the Following Sites:

Diablo Canyon
San Onofre

NOTES:

* Indicates a Western United States site whose default bin is 0.5g unless the licensee can demonstrate that the site hazard is similar to those sites east of the Rocky Mountains that are found in the 0.3g bin.

Changes in the review level earthquake from 0.5g to 0.3g should be approved prior to doing significant analysis.

APPENDIX 4
DOCUMENTATION

This appendix provides the guidelines for documentation and reporting format and content for the IPEEE submittal. The major parts of this appendix are the guidelines for seismic analysis (Section 4.2), internal fire analysis (Section 4.3), other analyses (Section 4.4). Licensees are requested to submit their IPEEE reports using the standard table of contents given in Table C.1 of NUREG-1407 or provide a cross reference. This will facilitate review by the NRC and promote consistency among various submittal. The contents of the elements of this table are discussed further below.

The level of detail needed in the documentation should be sufficient to enable the NRC to understand and determine the validity of key input data and calculation models used, to assess the sensitivity of the results to all key aspects of the analysis, and to audit any calculation. All important assumptions should be reported. It is not necessary to submit all the documentation needed for such an NRC review. Relevant documentation should be cited in the IPEEE submittal, and be available in easily retrievable form. The guideline for judging the adequacy of retained documentation is that independent expert analysts should be able to reproduce any portion of the results of the calculations in a straight forward, unambiguous manner. To the extent possible, the retained documentation should be organized along the lines identified in the areas of review. Any information that is comparable to that provided under the IPE for internal events can be incorporated by reference.

4.1 General

4.1.1 Conformance with Generic Letter and Supporting Material

Certification should be provided that an IPEEE has been completed and documented as requested. The certification should also identify the measures taken to ensure the technical adequacy of the IPEEE and the validation of results.

4.1.2 General Methodology

An overview description of the methodology employed in the IPEEE for each external event examined should be provided.

4.1.3 Information Assembly

Reporting guidelines include:

1. Plant layout and containment building information not contained in the Final Safety Analysis Report (FSAR).

2. A concise description of plant documentation used in the IPEEE, (e.g., the FSAR; system descriptions, procedures, and licensee event reports); and a concise discussion of the process used to confirm that the IPEEE represents the as-built, as-operated plant. The intent of such a confirmation is not to propose new design reverification efforts on the part of the licensees but to account for the impact of previous plant modifications or modifications conducted within the IPEEE framework.

3. A description of the coordination activities of the IPEEE teams among the external events (e.g., for seismically induced fires).

4.1.4 Submittal of Vulnerability Definition and Potential Plant Improvements

The licensee should provide a discussion on how a vulnerability is defined for each external event evaluated. The licensee should list any improvements (including equipment changes as well as changes in maintenance, operating and emergency procedures, surveillance, staffing, and training programs) that have been selected for implementation based on the IPEEE (a schedule for implementation should be provided) or that have already been implemented. A discussion of anticipated benefits, in terms of averted potential risk or increased plant seismic capacity, as well as drawbacks to any improvements should be provided. Those improvements that have been taken credit for in the analysis and have not yet been implemented at the plant, should be specifically highlighted in the submittal.

4.1.5 IPEEE Team and Peer Review

The basis for requesting the involvement of the licensee's staff in the IPEEE review is the belief that the maximum benefit from the performance of an IPEEE would be realized if the licensee's staff were involved in all aspects of the examination and that involvement would facilitate integration of the knowledge gained from the examination into operating procedures and training programs. Thus, the submittal should describe licensee staff participation and the extent to which the licensee was involved in all aspects of the program.

The submittal should also contain a description of the peer review performed, the same type of review as requested for the internal event IPE, the results of the review team's evaluation, and a list of the review team members.

4.2 Seismic Events

Section 4.2.1 describes guidelines for submittal of information by licensees who choose the seismic PRA for the seismic IPEEE,

whereas section 4.2.2 describes information guidelines for licensees who choose the seismic margin method for the seismic IPEEE. The submittal should be presented in conformance with the table of contents provided in Table C.1 of NUREG-1407.

4.2.1 Seismic PRA Methodology

The following information on the seismic IPEEE should be documented and submitted to the NRC:

1. A description of the methodology and key assumptions used in performing the seismic IPEEE.

2. The hazard curve(s) (or table of hazard values) used and the associated spectral shape used in the analysis. Also, if an upper bound cutoff to ground motion of less than 1.5g peak ground acceleration is assumed, the results of sensitivity studies to determine whether the cutoff affected the overall results and delineation and ranking of seismic sequences.

3. A summary of the walkdown findings and a concise description of the walkdown team and the procedures used.

4. All functional/systemic seismic event trees as well as data (including origin and method of analysis). Address to what extent the recommended enhancements have been incorporated in the IPEEE. A description of how nonseismic failures, human actions, dependencies, relay chatter, soil liquefaction, and seismically induced floods/fires are accounted for. Also, a list of important nonseismic failures with a rationale for the assumed failure rate given a seismic event.

5. A description of dominant functional/systemic sequences leading to core damage along with their frequencies and percentage contribution to overall seismic core damage frequencies (for both LLNL and EPRI hazard curves if used). Sequence selection criteria are provided in GL 88-20 and NUREG-1335. If either hazard curve causes a sequence to meet these criteria, that sequence should be included. The description of the sequences should include a discussion of specific assumptions and human recovery actions.

6. The estimated core damage frequency (for both the LLNL and EPRI hazard curves, if used) and plant damage state, the timing of the core damage, including a qualitative discussion of uncertainties and how they might affect the final results, and contributions of different ground motions to core damage frequencies.

7. Any seismically induced containment failures and other containment performance insights. Particularly, vulnerabilities found in the systems/functions which will lead to early containment failure that might result in high consequences. This includes: isolation, bypass, containment integrity and systems (e.g., igniters) required to prevent early failure. The computed fragilities of containment components, systems, and functions as applicable should be provided. The licensee may submit computed HCLPFs associated with containment (Optional).

8. A table of fragilities, both generic and plant-specific, used for screening as well as in the quantification. The estimated fragilities for the plant, dominant sequences, and dominant components should be reported. (Optional: The estimated HCLPF for the plant, dominant sequences, and components with and without nonseismic failures and human actions may be submitted by the licensee.)

9. Documentation with regard to other seismic issues addressed by the submittal, the basis and assumptions used to address these issues, and a discussion of the findings and conclusions. Evaluation results and potential improvements associated with the decay heat removal function and movable in-core flux mapping system (for Westinghouse plants) should be specifically highlighted.

10. A discussion of nonseismic failures and human actions that are significant contributors, or have impacts on results.

11. When an existing PRA is used to address the seismic IPEEE, the licensee should describe sensitivity studies related to the use of the initial hazard curves, supplemental plant walkdown results and subsequent evaluations, and relay-chatter evaluations. The licensee should examine items 1 through 10 above to fill in those items missed in the existing seismic PRA (See NUREG-1407 3.1.2).

4.2.2 Seismic Margins Methodology

The following information on the seismic IPEEE should be documented and submitted to the NRC for a full-scope and a focused-scope SMM review:

1. A description of the methodology and a list of important assumptions, including their basis, used in performing the seismic IPEEE. Address the extent to which the following were taken into account: nonseismic failures, human actions, dependencies, relay chatter, soil liquefaction, and seismically induced floods/fires. Also, a list of important nonseismic failures with a rationale for the assumed failure rate given a seismic event.

2. A summary of the walkdown results and a concise description of the walkdown team and procedures used.

3. All functional/systemic seismic event trees data (including origin and method of analysis) when NRC SMM is used.

4. A description of the most important sequences and important minimal cutsets (for both seismic and nonseismic failures) leading to core damage (NRC method) or a description of the success paths and procedures used for their selection and of each component in the controlling success path (EPRI method).

5. Any seismically induced containment failures and other containment performance insights. Particularly, vulnerabilities found in the systems/functions which will lead to early containment failure and high consequences. This includes: isolation, bypass, containment integrity and systems (e.g., igniters) required to prevent early failure. Also, computed fragilities (if used) and HCLPFs of containment components, systems, and functions as applicable.

6. A table of fragilities (if used) and HCLPFs, both generic and plant-specific, used for screening as well as in the quantification. The estimated fragilities (if used) and HCLPFs for the plant, dominant sequences, and dominant components should be reported.

7. Documentation with regard to other seismic issues addressed by the submittal, the basis and assumptions used to address these issues, and a discussion of the findings and conclusions. Evaluation results and potential improvements associated with the decay heat removal function and movable in-core flux mapping system (for Westinghouse plants) should be specifically highlighted.

8. For NRC method provide a discussion of nonseismic failures and human actions that are significant contributors, or have impacts on results.

The following information should be documented and submitted to the NRC for a reduced-scope SMM review:

1. A description of the procedures used to identify systems and components for the walkdown in performing the seismic IPEEE.
2. A summary of the walkdown findings and a concise description of the walkdown team and procedures used.

3. A discussion and the results of any specific component capacity evaluations performed, the methods used, and assumptions.

4. Documentation with regard to other seismic issues addressed by the submittal, the basis and assumptions used to address these issues, and a discussion of the findings and conclusions. Evaluation results and potential improvements associated with the decay heat removal function and movable in-core flux mapping system (for Westinghouse plants) should be specifically highlighted.

4.3 Internal Fires

The following information on the internal fires IPEEE should be documented and submitted to the NRC:

1. A description of the methodology and key assumptions used in performing the fire IPEEE and a discussion of the status of Appendix R modifications.

2. A summary of the walkdown findings and a concise description of the walkdown team and the procedures used. This should include a description of the efforts to ensure that cable routing used in the analysis represents as-built information and a description of the treatment of any existing dependence between remote shutdown and control room circuitry.

3. A discussion of the criteria used to identify critical fire areas and a list of critical areas, including (a) single areas in which equipment failures represent a serious erosion of safety margin, and (b) same as (a), but for double or multiple areas sharing common barriers, penetration seals, HVAC ducting, etc.

4. A discussion of the criteria used for fire size and duration and the treatment of cross-zone fire spread and associated major assumptions.

5. A discussion of the fire initiation data base, including the plant-specific data base used. Describe the data handling method, including major assumptions, the role of expert judgment, and the identification and evaluation of sources of data uncertainties. A discussion of each case where the plant-specific data used is less conservative than the data base used in the approved fire vulnerability methodologies.

6. A discussion of the treatment of fire growth and spread, the spread of hot gases and smoke, and the analysis of detection and suppression and their associated assumptions,

including the treatment of suppression-induced damage to equipment.

7. A discussion of fire damage modeling, including the definition of fire-induced failures related to fire barriers and control systems and fire-induced damage to cabinets. A discussion of how human intervention is treated and how fire-induced and non-fire-induced failures are combined. Identify recovery actions and types of fire mitigating actions taken credit for in these sequences.

8. Discuss the treatment of detection and suppression including fire fighting procedures, fire brigade training and adequacy of existing fire brigade equipment, and treatment of access routes versus existing barriers.

9. All functional/systemic event trees associated with fire initiated sequences.

10. A description of dominant functional/systemic sequences leading to core damage along with their frequencies and percentage contribution to overall fire core damage frequencies. Sequence selection criteria are provided in GL 88-20 and NUREG-1335. The description of the sequences should include a discussion of specific assumptions and human recovery action.

11. The estimated core damage frequency, the timing of the associated core damage, a list of analytical assumptions including their bases, and the sources of uncertainties.

12. Any fire induced containment failures identified as being different than those identified in the internal events analysis and other containment performance insights.

13. Documentation with regard to fire risk scoping study issues addressed by the submittal, the basis and assumptions used to address these issues, and a discussion of the findings and conclusions. Evaluation results and potential improvements associated with the decay heat removal function should be specifically highlighted.

14. When an existing PRA is used to address the fire IPEEE, the licensee should describe sensitivity studies related to the use of the initial hazard supplemental plant walkdown results and subsequent evaluations. The licensee should examine the above list to fill in those items missed in the existing fire PRA.

4.4 <u>High Winds, Floods, and Others</u>

The following information on the high winds, floods, and others portion of the IPEEE should be documented and submitted to the NRC:

1. A description of the methodologies used in the examination.

2. Information on plant-specific hazard data and licensing bases.

3. Identified significant changes not reported per 10CFR 50.71(e) (See NUREG-1407 5.2.2), if any, since OL issuance with respect to high winds, floods, and other external events.

4. Results of plant/facility design review to determine their robustness in relation to NRC's current criteria.

5. Results of the assessment of the hazard frequency and the associated conditional core damage frequency if step 4 of Figure 1 is used.

6. Results of the bounding analysis if step 5 of Figure 1 is used.

7. All functional event trees, including origin and method of analysis (PRA only).

8. A description of each functional sequence selected, including discussion of specific assumptions and human recovery action (PRA only).

9. The estimated core damage frequency, the timing of the associated core damage, a list of analytical assumptions including their bases, and the sources of uncertainties, if applicable (PRA only).

10. A certification that the licensee knows of no other plant-unique external event that poses any significant threat of severe accident within the context of the screening approach for "High Winds, Floods, and Others."

APPENDIX 5
10CFR50.54(f) ANALYSIS
FOR INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)

10CFR50.54(f) requires that "... the NRC must prepare the reason or reasons for each information request prior to issuance to ensure that the burden to be imposed on respondents is justified in view of the potential safety significance of the issue to be addressed in the requested information." Further, Revision 4 of the Charter of the Committee to Review Generic Requirements (CRGR), dated April 1989 specifies that, at a minimum, such an evaluation shall include:

a. A problem statement that describes the need for the information in terms of potential safety benefit,

b. The licensee actions required and the cost to develop a response to the information request, and

c. An anticipated schedule for NRC use of the information.

The staff's 10CFR50.54(f) evaluation of the information request addressing the above elements follows:

a. A problem statement that describes the need for the information in terms of potential safety benefit.

In the Commission policy statement on severe accidents in nuclear power plants issued August 8, 1985 (50FR 32138), the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on any regulatory requirements for these plants. However, the Commission recognizes, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements. As a key part of the implementation of the policy statement, the staff issued Generic Letter 88-20 on Nov. 23, 1988, requesting that each licensee conduct an individual plant examination (IPE) for internally initiated events only. An analysis prepared to justify the burden associated with the internal event IPE (Ref. 23) is also generally applicable to the external event IPE request. This current analysis provides additional justification to support the extension of the IPE to include external events.

Current risk assessments Refs. 6-8, 13, and 24-29 indicate that the risk from external events could be a significant

contributor to core damage in some instances. Most recently, the NUREG-1150 (Ref. 30) study showed that the contribution to severe accidents initiated by internal fires and seismic events was comparable to or greater than that initiated by internal events. Examples of the severe accident sequences initiated by external events can be found in References 6-8, 13, and 23-29. Typically, these sequences involved external event initiated transients and small-break loss-of-coolant accidents and were frequently related to lack of redundancy, separation, and physical protection in safety trains for internal fires, floods, and seismic events. These results suggest likely areas for cost-effective improvements from plant-specific analyses that focus properly on external events (e.g., the plant support systems where there is less redundancy, less separation and independence between trains, poorer overall general arrangement of equipment from a safety viewpoint, and much more system sharing as compared to the higher level systems). Actual examples of cost-effective improvements that have been found and made are modification of structural design to improve the capability of the control room to withstand seismic events at Indian Point; changes to the turbine building, control room, turbine building equipment, and procedural modifications to reduce plant vulnerability to internal floods at Oconee; and enlargement of drainage divertment around the plant to withstand the effects of external flood and installation of a dedicated independent safe shutdown system and construction of a separate safe shutdown system building to improve plant capability to withstand seismic events, tornadoes, external floods, and fires at Yankee Rowe. In addition, deficient equipment anchorages have been identified and corrected in many plants as a result of walkdowns like those specified for performance in the IPEEE.

The staff delayed the issuance of the request for a systematic examination of external events to allow the staff to carry out additional work to (1) identify which external hazards need an examination, (2) identify acceptable examination methods and develop procedural guidance, and (3) coordinate with other ongoing external event programs. In December 1987, NRC created the External Events Steering Group (EESG) to coordinate the effort to address these issues. The EESG established three subcommittees (Seismic; Fires; and High Winds, Floods, and Others). The staff has completed this work and is now requesting that each licensee perform an individual plant examination of external events (IPEEE) to identify plant-specific vulnerabilities, if any, to severe accidents and report the results to the Commission. Experience with plant specific PRAs since the issuance of the Policy

Statement has continued to confirm that analyses of this type often reveal plant-specific vulnerabilities that can be and typically are corrected in a cost effective manner see the value/impact analysis presented in the Attachment to Appendix D of NUREG-1407. Because severe accidents dominate nuclear power plant risks, the Commission intends to take all reasonable steps to reduce the chances of occurrence of a severe accident and to mitigate the consequences of such an accident should one occur.

b. The licensee actions required and the cost to develop a response to the information request.

All licensees would be requested to perform an IPEEE on their plants for plant-specific vulnerabilities to severe accidents and report this information to the NRC. The licensees would also report to the NRC proposed modifications, if any, and indicate how the insights and lessons learned from the examination have been incorporated into plant operation. The licensees may perform the IPEEE using methods described in the Generic Letter or using other methods that the licensees may propose provided NRC review has shown that such proposed methods are effective and applicable.

We estimate that the cost of these systematic examinations will vary depending on specific site conditions, but, on the average, will cost no more than $1M or a maximum of about 6 person-years for the examination. However, we feel that, for most licensees, the scope will be less than that and the cost will also be less (see cost estimates presented in the Appendix D to NUREG-1407). Also, for those licensees who have already performed external event PRAs or seismic margin analyses, the actual cost of updating and submitting the analyses would be significantly less. We conclude that the burden to be imposed on respondents is justified in view of the potential safety significance of ensuring that vulnerabilities that may affect nuclear plant safety are properly identified and corrected.

c. An anticipated schedule for the NRC use of the information.

We expect that most of the IPEEEs will be submitted in mid 1994 and that staff review of the results to ensure that the intent of the Commission's Severe Accident Policy Statement is met will be completed by mid 1995.

APPENDIX 6
REFERENCES

1. U.S. Nuclear Regulatory Commission (USNRC), "Policy Statement on Severe Accidents," Federal Register, Vol. 50, 32138, August 8, 1985.

2. USNRC Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities--10CFR 50.54(f)," November 23, 1988.

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

4. USNRC SECY 88-147, "Integration Plans for Closure of Severe Accident Issues," May 25, 1988.

5. USNRC NUREG/CR-2300, "PRA Procedures Guide," Amarican Nuclear Society and Institute of Electrical and Electronic Engineers, January 1983.

6. USNRC NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States," Lawrence Livermore National Laboratory, December 1987.

7. USNRC NUREG/CR-5042, Supplement 1, "Evaluation of External Hazards to Nuclear Power Plants in the United States, Seismic Hazards" Lawrence Livermore National Laboratory, April 1988.

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

9. USNRC NUREG/CR-2815, "Probabilistic Safety Assessment Procedures Guide," Brookhaven National Laboratory, August 1985.

10. USNRC NUREG/CR-4840, "Recommended Procedures for the Simplified External Event Risk Analyses for NUREG-1150," Sandia National Laboratory, September 1989.

11. USNRC NUREG/CR-5250, "Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains," Lawrence Livermore National Laboratory, January 1989.

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

13. USNRC NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," Lawrence Livermore National Laboratory, August 1985.

14. USNRC NUREG/CR-4482, "Recommendations to the Nuclear Regulatory Commission on Trial Guidelines for Seismic Margins Reviews of Nuclear Power Plants," Lawrence Livermore National Laboratory, March 1986.

15. USNRC NUREG/CR-5076, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants: The Importance of BWR Plant Systems and Functions to Seismic Margins," Lawrence Livermore National Laboratory, May 1988.

16. Electric Power Research Institute, NP-6041, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," October 1988.

17. USNRC NUREG/CR-5088, "Fire Risk Scoping Study," Sandia National Laboratory, January. 1989.

18. USNRC NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants--LWR Edition," December 1975.

19. USNRC Memorandum from E. Beckjord to R. Houston, dated July 31, 1989, Subject: Generic Issue 131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants" (available in NRC Public Document Room).

20. USNRC Generic Letter 88-20, Supplement 1, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities--10 CFR 50.54(f)," August 29, 1989.

21. USNRC Generic Letter 88-20, Supplement 3, "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities," June 1990.

22. USNRC NUREG-1335, "Individual Plant Examination: Submittal Guidance," Final Report, August 1989.

23. USNRC Memorandum from B. Sheron to T. Speis, dated December 1, 1988, Subject: Staff Evaluation in Support of 10CFR 50.54(f) Generic Letter 88-20 Requiring Individual Plant Examination (available in NRC Public Document Room).

24. USNRC NUREG/CR-4458, "Shutdown Decay Heat Removal Analysis of a Westinghouse 2-loop Pressurized Water Reactor," Sandia National Laboratory, March 1987.

25. USNRC NUREG/CR-4713, "Shutdown Decay Heat Removal Analysis of a Babcock and Wilcox Pressurized Water Reactor," Sandia National Laboratory, March 1987.

26. USNRC NUREG/CR-4762, "Shutdown Decay Heat Removal Analysis of a Westinghouse 3-loop Pressurized Water Reactor," Sandia National Laboratory, March 1987.

27. USNRC NUREG/CR-4767, "Shutdown Decay Heat Removal Analysis of a General Electric BWR4/Mark I," Sandia National Laboratory, March 1987.

28. USNRC NUREG/CR-4710, "Shutdown Decay Heat Removal Analysis of a Combustion Engineering Pressurized Water Reactor," Sandia National Laboratory, March 1987.

29. USNRC NUREG/CR-4448, "Shutdown Decay Heat Removal Analysis of a General Electric BWR3/ Mark I," Sandia National Laboratory, March 1987.

30. USNRC NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Sandia National Laboratory, December 1990.