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Severe Accident Issue Closure Guidelines

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FOREWORD

This report was developed with the guidance of the NEI Severe Accident Working Group (SAWG), with input on seismic aspects from the NEI Seismic Issues Working Group (SIWG). It provides one approach for bringing closure to the severe accident issues for the population of existing plants on a plant-specific basis; other approaches can be formulated. The document is not intended to address the specific tasks involved in identifying candidate safety enhancements in response to any given severe accident insight. Rather, it provides a screening tool for addressing a given severe accident insight relative to the overall safety of the plant. Individual utilities may elect to adopt the closure guidance in Sections 2.0, 3.0 and 4.0 (IPE, IPEEE and containment Performance Improvement), but NEI does not intend that the NRC should require utilities to do so.

Section 5.0, on the other hand, contains implementing guidance relative to the formal industry position on severe accident management approved by the NEI Strategic Issues Advisory Committee on November 4, 1994. The formal industry position, binding on all utility members, is:

> *Each licensee will:*
>
> *Assess current capabilities to respond to severe accident conditions using Section 5 of NEI 91-04, Revision 1, "Severe Accident Issue Closure Guidelines."*
>
> *Implement appropriate improvements identified in the assessment, within the constraints of existing personnel and hardware, on a schedule to be determined by each licensee and communicated to the NRC, but in any event no later than December 31, 1998.*

The report espouses a graded approach in prioritizing those severe accident sequences warranting closer scrutiny and the subsequent identification of candidate safety enhancements to: (i) plant hardware, (ii) administrative controls, (iii) procedures, or (iv) accident management guidance.

Probabilistic risk assessments (PRAs) involve different levels of precision and substantial numerical uncertainty. Methods, assumptions, data, etc., can vary widely depending on the analyst. Focusing on a bottom line numerical threshold as a "go, no-go" index for determining the areas of plant design and operation that should be considered for enhancement would, in the industry's view, be inappropriate. That is not to say the numbers do not count. The NRC staff has requested that utility IPE submittals include a listing of the major sequences affecting core damage and public safety, and the corresponding numerical values.

The state of the art of internal and external event PRAs differ. Since publication of WASH-1400, less emphasis has been placed on the quantification of external events than on internal events. The relatively limited experience/databases for external phenomena provide greater uncertainty in the initiator frequencies as compared to internal initiators. This uncertainty, coupled with the inherent robustness of the plants against external events, has led NRC staff to allow more deterministic methods in evaluating the external events. Therefore, in the IPEEE, utilities should emphasize identifying candidate safety enhancements, rather than quantifying safety.

Subsequent to publishing NEI 91-04, Revision 0, the NRC staff issued NUREG-1488 (reference 14) which indicated a profound reduction in seismic hazard estimates. These results, in conjunction with other seismic risk assessment insights, prompted NEI to recommend (Reference 15) to industry and NRC staff that many “focused-scope” plants should instead perform “reduced-scope” studies. Reference 16 provides additional insights and bases for the industry recommendations. Nevertheless, the resolution processes outlined in Section 4.0 of this report remain unchanged, because each is dependent on the scope of the review performed.

Consideration was given to the Qualitative and Quantitative Health Objectives of the Commission's Safety Goal Policy Statement when developing these guidelines, within the precision of PRA methodologies and databases. However, this does not imply that NEI recommends that NRC's subordinate safety goal objectives be used to make plant-specific decisions. Rather, NEI agrees with the Commission that the safety goals provide criteria by which to evaluate the aggregate population of plants. The rather well recognized limitations of, and uncertainties in, current PRA techniques speak to the potential for pitfalls, if such a tool is applied too stringently when comparing quantitative results against quantitative goals as part of the decision-making process.

In summary, the guidance in this report is intended to help utilities facilitate judgements as the industry implements the Commission's Severe Accident Policy Statement and on the advisability of implementing candidate safety enhancements.

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1.0 INTRODUCTION

1.1 Background

For a number of years the nuclear power industry has been evaluating the implications of postulated severe accidents on the current generation of plants though consideration of severe accidents is beyond their design bases. This evaluation is comprised of a number of widely varied efforts ranging from severe accident research to the assessment of plant specific severe accident capabilities. As an outgrowth of the NRC's efforts in this area, a number of severe accident related rules have been enacted which extend significantly beyond the original licensing design basis of these plants. The rules have been primarily focused on specific severe accident related issues which were thought to be generic and potentially significant (e.g., ATWS, Station Blackout, Hydrogen Control, etc.). In 1985, the Commission issued its policy on Severe Reactor Accidents Regarding Future Designs and Existing Plants (50 FR 32138) which concluded that existing plants pose no undue risk to the public. However, based on NRC and industry experience in performing plant specific probabilistic risk assessments (PRA), it was recognized that systematic, limited scope evaluations of existing plants would be beneficial. Consequently, the NRC and Industry have been working toward the development and implementation of programs to address severe accidents on a plant-specific basis. In mid-1988, the NRC staff formulated a program plan for the integration and closure of severe accident issues (Reference 1). Four areas requiring licensee action are:

- Individual Plant Examination (IPE)

- Containment Performance Improvements (CPI)

- IPE of External Events (IPEEE)

- Accident Management

The industry fully supports the NRC objective to resolve and close severe accident issues. As a result, the industry has numerous generic and plant specific programs already underway to help address the relevant areas. However, because consideration of severe accidents is not a part of the design basis of the current generation of plants and many of the issues related to severe accidents are plant specific, it is desirable to define how each individual plant could deal with and use, within the regulatory environment, the plant-specific insights and knowledge gained through the implementation of the various programs.

1.2 Guiding Principle of Closure Guidelines

The guiding principle of this document can be stated as follows:

> These guidelines are intended to be a useful framework which utilities can objectively use toward: (1) closure of the major areas related to the severe accident issue, and (2) determining how to disposition and utilize the plant specific insights and knowledge gained through the studies performed. They should accommodate the safety and investment protection objectives of the utility.

The report contains a combination of quantitative and qualitative guidance. This is a reflection of the perceived usefulness that numbers provide in order to judge the relative value of the various IPE and IPEEE insights, while simultaneously recognizing the inherent uncertainties and varying precision in these numerical results. Attempting to limit the guidance to only a philosophical and qualitative discussion was deemed too nebulous. However, a strict focus on a single, bottom-line number, is just as inappropriate.

It is recognized that several aspects of the IPE process and of these guidelines involve making decisions in light of uncertainty and using judgement. Specifically, the estimated frequencies of core damage or large radioactive release have uncertainty in both the estimate of the mean and the distribution about the mean value, and the definition of core damage accident sequence groups is variable. Therefore, flexibility has been included in these closure guidelines to allow for these considerations in their application.

It is intended that the guidelines can facilitate closure if they are applied by the licensee with the above guiding principle clearly in mind. It is believed that most utility decisions relative to the consideration of candidate safety enhancements to address particular IPE and IPEEE insights can be made without performing a formal backfit analysis. The industry agrees with the NRC staff guidance provided in Generic Letter 88-20 that when disagreement exists between NRC and a licensee as to whether a particular candidate safety enhancement is worthwhile, the NRC is burdened to provide a plant-specific cost justification in accordance with 10 CFR § 50.109. This is not the preferred path of issue resolution, unless significant differences of view remain. Nonetheless, should a modification be indicated, it is believed that under most circumstances an intuitive, acceptable plant change can be identified without a formal backfit process.

1.3 Overview of this Document

This document provides guidelines for licensee use in the closure of the severe accident issues on a plant specific basis. This document provides an overview of proposed closure guidelines for each major severe accident issue. The guidelines are structured on an issue by issue basis with the closure process described for each: IPE Closure (Section 2), CPI Closure (Section 3), IPEEE Closure (Section 4) and Severe Accident Management Closure (Section 5). Where there is overlap or coordination between issues and closure processes that is also described. [NOTE: The acronym IPE is used in this document to mean IPE of Internal Events.] A table of definitions for key terms is provided in Appendix A.

1.4 Consideration of Plant Changes Initiated in Response to Resolution of Severe Accident Issues Relative to Future Plant Changes

In performing the assessments identified in this document, a licensee may decide to alter the plant design or operating practices in some fashion. The implementation of such changes is most likely in response to accident sequences that are beyond the current licensing basis of the plant. Some of these changes will be reported to the NRC in an IPE or IPEEE submittal report. Others may be enhancements to be implemented as part of an accident management program and will not be formally transmitted to the NRC for review. In either case, the licensee may make such changes without prior NRC approval if such changes satisfy the requirements of 10 CFR § 50.59, "Changes, Tests and Experiments."

A plant's current licensing basis is comprised of the information submitted to the NRC as part of the application for an operating license and technical specifications issued by the NRC staff and any subsequent amendments or licensee commitments. 10 CFR § 50.59 establishes an evaluation process and criteria (e.g., definition of an unreviewed safety question) by which one determines whether or not changes can be made to the facility or its procedures, relative to the licensing basis, as amended, without prior NRC approval. The intent of 10 CFR § 50.59 is to preserve the final safety analysis report (FSAR), as amended pursuant to 10 CFR § 50.71(e), and the technical specifications. These two documents encompass the limiting set of analyses, descriptions, commitments, programs, etc. that were either:
(1) originally evaluated by NRC staff and confirmed in a safety evaluation report to be acceptable in support of issuance of an operating license, or (2) subsequently evaluated by NRC staff to confirm that any proposed amendments to the license or the FSAR are acceptable. Therefore, while the proposed changes identified during the closure process outlined in this document are intended to address severe accident challenges, each must

also be evaluated prior to implementation with respect to whether it has any effect on the licensing basis.

However, licensees are not required to modify their criteria for making an unreviewed safety question determination to reflect severe accident considerations. That is to say, the unreviewed safety question determination required by the 10 CFR § 50.59 process should not be expanded to include consideration of the effect of any proposed plant change on a licensee's capability to respond to severe accidents beyond those described in the final safety analysis report, as amended (e.g., station blackout, ATWS, etc.). The term "accidents" referred to in 10 CFR § 50.59 are the anticipated operational transients and postulated design basis accidents that are analyzed to demonstrate that the plant can be operated without undue risk to the health and safety of the public. These accidents are typically found in Chapter 15 of the FSAR, although 10 CFR § 50.59 is also applicable to other events with which the plant was designed to cope and are described in the FSAR (e.g., turbine missiles or flooding) and to plant modifications and analysis added to the licensing basis and reflected in the updated FSAR. As noted in NSAC-125 (Reference 2), it must be emphasized that probabilistic risk assessment, such as an IPE, is just one of the tools for evaluating safety, but it is not a necessary component for addressing the requirements of 10 CFR § 50.59.

Still, as a matter of good practice, the licensee does have a long term obligation to maintain the basis for how and why the plant was modified as a result of the IPE, IPEEE and accident management enhancement process. Documentation should be sufficiently clear such that any future plant changes affecting the same hardware, procedures, etc., are made with full cognizance of what severe accident considerations prompted the earlier modifications. This will ensure that during future operation those aspects of a plant's design and operating configuration put in place to address severe accident insights are known. The method and level of configuration control used by a licensee and the interface with the NRC staff on future changes is a matter that is left to the discretion of each licensee.

2.0 INDIVIDUAL PLANT EXAMINATION (IPE) OF INTERNAL EVENTS CLOSURE

The assessment of plant specific severe accident capabilities being performed as part of the Individual Plant Examinations (IPE) of Internal Events is one of the cornerstones of utility severe accident efforts. In Generic Letter 88-20 (Reference 3), the NRC has several stated objectives for the performance of the IPE:

1) For utilities to develop an appreciation for severe accident behavior;

2) For utilities to understand the most likely severe accident sequences that could occur at their plants;

3) For utilities to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases; and

4) If necessary, for utilities to reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

One of the products of a plant specific IPE is the identification of important plant specific severe accident scenarios. Depending upon the format of the IPE analysis, scenarios take the form of event tree sequences comprised of combinations of function, system, or sub-system failures with an associated frequency of occurrence. These sequences can be used in the assessment of the applicability of various severe accident issues to the specific plant. The following section provides guidelines for the use of plant specific IPE results in achieving closure of the IPE process and for use of IPE results in closure of other severe accident issues.

The basic premise of these guidelines is that by evaluating IPE results in terms of groups of accident sequences and containment releases, licensees can assess the candidate safety enhancements related to severe accidents. Consistent with current PRA techniques, IPEs are likely to contain a large number of initiating events, and consequently, a large number of sequences. However, these can be consolidated into a relatively small number of categories. Alternatively, when appropriate, only the dominant sequences are of further interest and the subject of detailed evaluation. The appropriateness of plant changes is evaluated with a hierarchical structure based on the contribution to core damage serving as a first screen. In other words, a high ranking sequence value is an indicator of the possible need for further analysis. This analysis could be a more detailed evaluation replacing simplifying analytical assumptions with more realistic modelling or a further assessment of the accident sequence focusing on the magnitude and likelihood of a radiological release from containment.

Sequences having relatively high frequencies with correspondingly high significance to public safety warrant more serious consideration of potential plant design or operational changes. Conversely, sequences with low frequencies or public safety significance are unlikely to warrant costly changes, but may be more appropriately addressed through procedural or accident management guidance enhancements.

Implicit in the screening criteria provided in Appendix 2 of Generic Letter 88-20, the NRC identifies several aspects of IPE results which are of interest. These aspects are: core damage frequency, containment bypass frequency and release significance. The IPE closure process delineated below evaluates IPE results on each of those bases through three basic steps:

1) Categorization of IPE core damage sequences into groups of core damage events,

2) Screening of IPE core damage sequence group frequencies based upon the closure guidelines; and

3) Evaluation of significant core damage sequence groups.

The more detailed assessment suggested for comparatively high care damage frequency (CDF) sequences is in keeping with the objectives of the IPE, because it provides for an appreciation of the key containment failure modes, the impact of phenomena and plant features, the impact of operator actions, and how additional administrative controls, training, procedures or hardware modifications could enhance a utility's ability to prevent or mitigate specific sequences. This prioritization also allows for the segregating of, over a broad spectrum of credible accidents, specific insights associated with containment and containment mitigating systems in order to reduce or eliminate specific scenarios of concern.

It is important to note that the guidelines described herein may be superseded by requirements driven by existing regulations or corporate safety goals, if more conservative. The proposed approach assumes that the utility is in compliance with existing regulations and with its own internal corporate safety goals, when applicable.

2.1 Grouping of IPE Core Damage Sequences

The first step of the IPE closure process involves the evaluation and grouping of all IPE core damage sequences into core damage sequence groups. The intent is that utilities, if possible, utilize the binning scheme developed as part of their IPEs and not perform significant reorganization in utilizing this process. Each accident sequence group should be made up of an initiating event plus a set of plant faults. These faults could be functional faults or system faults, depending upon the approach utilized in defining IPE core damage sequences. The objective in grouping individual core damage sequences is to allow common component, system and operator faults to be evaluated together.

As defined, each sequence group definition should designate IPE core damage sequences which are mutually exclusive of all others; that is, an individual IPE sequence should fall under only one of these group definitions. Depending upon the approach used in their IPE, utilities may have an alternate grouping scheme already developed. Alternate schemes should include consideration of the following items:

- Each category should be based on similarities in the plant response and plant system failures required to cause core damage (i.e., based on initiator grouping and the systems or functions which were required to prevent core damage, but failed);

- Each category should be mutually exclusive of the others (i.e., the frequency of each IPE sequence should be counted in only one category); and

- The categories should include all explicitly quantified core damage sequences analyzed in the IPE

It is sometimes practical to group sequences with a common initiator as a separate sequence group even though the functional response may vary somewhat among accidents (e.g., Station Blackout). Past PRAs have generally been successful in grouping their results into 10 to 15 group definitions for the purposes of reporting and evaluating results.

An example set of functional sequence definitions is provided in Appendix B. These definitions are provided for example only as a means to illustrate the nature of core damage sequence groups. Other grouping schemes are certainly acceptable and expected.

2.2 Screening of Core Damage Sequences (Except Containment Bypass) Against Closure Guidelines

Once the IPE results have been categorized as desired in terms of accident sequences, they can be compared against the closure guidelines. The process used in this evaluation is described in Table 1 and portrayed in graphic form in Figure 1. The process involves the comparison of the total mean core damage frequency (CDF) for each accident sequence group against a set of guidelines to determine the plant candidate safety enhancements to be considered (e.g., design or operational change, administrative control staffing, training, emergency operating procedure change, or accident management change). Utilities should assess their sequence groups on the basis of absolute core damage frequency contribution (i.e., core damage frequency per year) and on the basis of relative contribution to the total IPE core damage frequency (i.e., percent of cumulative internal event CDF). That is, compare the values (CDF or percentage) for each sequence category against the table and enter the hierarchical process at the highest tier within the table.

Assessing CDF contribution on both an absolute or relative basis is provided in recognition of the varying accident sequence profiles that exist from plant to plant. For instance, some plants have the majority of cumulative CDF represented in only a few accident sequences. A dominant contributor, even on a relative basis, warrants scrutiny from the perspectives of: (1) eliminating or minimizing the potential for the initiator occurring, (2) enhancing the ability to take effective measures to prevent core damage, and (3) if necessary, enhancing the ability to take mitigative actions that either prolong the time to vessel or containment failure. This should be the case, whether the dominant sequences are in the range of 1E-4 per reactor year or 1E-5 per reactor year. On the other hand, no further corrective action need be pursued for those sequences with CDF values below the threshold of 1E-6 per reactor year (or 1E-7 for containment bypass events). Again, a word of caution is advisable. That is, the specific numerical values alone are not sufficient. Judgement should be applied when considering which sequences to pursue. For instance, utilities may wish to consider sequences below these cutoff thresholds if the "consequences" were relatively high.

Should the CDF value be relatively high (i.e., greater than 1E-5 per reactor year for most sequences and 1E-6 per reactor year for bypass sequences), one of the following options should be pursued:

(1) Assess the safety significance to the public (e.g., release categorization), or

(2) Remove simplifying assumptions from the original IPE analysis. For example, a more detailed or refined thermal hydraulic model may alter the understanding of what constitutes success criteria for a given sequence or sequence category.

It should be noted that the intent of the second item <u>is not</u> to lessen the significance of legitimate insights via application of further analytical techniques, but allow for more detailed evaluation, if appropriate. Such refined analyses may not have appeared to be warranted during an earlier stage of the IPE. Yet, analysts are cautioned not to be overly anxious in re-quantifying portions of the IPE.

Table 1 contains a hierarchical listing of potential licensee responses. The priority for consideration is from Response 1 through 3, in descending order, dependent upon the CDF contribution from the sequence(s) under review. The intent of such a hierarchy is, for the more dominant sequences, to focus attention on first considering candidate safety enhancements that eliminate the accident initiator (or at least significantly reduce its likelihood). These may be hardware fixes, administrative changes or restrictions to routine operating practices, or changes to normal operating procedures (e.g., system or general operating instructions). Administrative change is a rather broad category, encompassing changes or restrictions to routine operating practices, surveillance test intervals, maintenance call-up changes, restrictions or equipment out of service, etc. The latter response categories identify enhancements that address mitigating the consequences of the event once it has initiated, whether it be focused on preventing core melt (e.g., EOPs) or reactor vessel/containment failure (e.g., Emergency Plans or severe accident management guidance). New or enhanced training to more effectively address an item fits in any category.

The guidelines are based on the following:

- The highest level considers the Commission's Safety Goal Policy Statement and the NRC staff's subordinate core damage frequency objective of 1E-4 per reactor year for the industry.

- The graded nature of the responses for the guidelines consider the decrease in potential benefit which corresponds to lower core damage frequencies.

- The hierarchical levels represent approximate cost benefit thresholds for the most costly option evaluated (i.e., a CDF of 1E-4 per reactor year with a corresponding significant release potential has the benefit potentially sufficient to warrant consideration of a design change).

As each accident sequence group is evaluated and potential plant changes are considered (i.e., design, administrative, normal operating instruction or emergency operating procedure change), the core damage frequency impact should be calculated and the sequence should be re-evaluated against the guidelines (e.g., Table 1) to determine if the sequence should be addressed further. However, the basis for re-evaluation should be the same form of core damage criteria as originally applied: absolute or relative. It is not necessary to move from an absolute CDF comparison to a relative one, or vice versa. When the sequence is dispositioned by referring it to severe accident management guidance, re-evaluation is not required, and in fact, is discouraged. It is not intended to create an endless "do-loop" in the evaluation hierarchy.

2.3 Screening of Containment Bypass Sequences Against Closure Guidelines

Sequences involving unisolated breach of the primary system outside the primary containment can result in fission product bypass of the containment and present a potentially significant risk to the public. Therefore, particular attention should be paid to the safety significance of these sequences. In order to compensate for the potential increase in event severity presented by containment bypass sequences, the guidelines used for screening are an order of magnitude lower than for the core damage frequency evaluation. Such an approach is consistent with the NRC staff reporting criteria for bypass sequences already requested in Generic Letter 88-20, Appendix 2. The containment bypass evaluation process is shown in Table 2 and portrayed in graphic form in Figure 2.

2.4 Evaluation of Significant Core Damage Sequence Groups

It is recognized that the evaluation of core damage sequence groups exceeding the various closure guidelines requires engineering judgement. A number of factors contribute to this. First, not all core damage sequences have the same impact on public health and safety due to differences in containment response from sequence to sequence. As a result, decisions regarding plant changes can not be made solely on the basis of core damage frequency. The closure guidelines in Tables 1 and 2 are designed to be a screen to focus the closure process on likely areas of potential cost effective enhancement.

In addition, in performing an IPE some core damage frequencies may be artificially increased by conservative assumptions regarding uncertain plant response or phenomena. For example, the response of some reactor coolant pump (RCP) seals to loss of seal cooling is a generic issue under study by the industry and NRC. If a core damage sequence group included sequences resulting from a conservative treatment of RCP seal performance, then expensive plant modifications aimed at addressing such sequences may be premature, given the additional research and the NRC regulatory analysis underway. In such cases, it would be reasonable for licensees to defer consideration of costly plant modifications until the additional research validates the conservative assumptions utilized in the IPE. Similar considerations can be identified for sequences involving containment response to uncertain physical phenomena. Nevertheless, interim actions may be warranted in lieu of final safety enhancements awaiting resolution of the related issues.

In some cases, licensees may identify a number of cost effective safety enhancements to address a specific sequence group (or multiple groups). In these cases, it is not necessary for hardware changes to be selected over procedural or administrative changes. Rather, it is more important that sufficient explanation be given to support the plant change selected.

A final consideration in identifying and selecting a plant enhancement for implementation is that a change made to address one accident sequence group may have an impact on other groups. This impact can, in some cases, be a positive impact (i.e., also serve to reduce the safety significance of another group) or a negative impact (i.e., increase plant vulnerability to another group of sequences). Consequently, candidate safety enhancements should be considered in terms of their overall impact on public safety and not in isolation.

2.5 Closure of IPE Process

Closure of the IPE process for a plant is achieved when the utility has evaluated the results of the IPE against the closure guidelines as described above. Additionally, the results of the evaluation may be submitted to the NRC along with the schedule for any proposed safety enhancements. This evaluation involves the assessment and documentation of quantitative results using the processes shown in Figures 1 and 2 and the evaluation of qualitative insights. Qualitative insights and issues are to be evaluated based on an estimation of their potential impact on the total core damage frequency. These estimates should be rough order of magnitude type estimates and should be based on some semi-quantitative review of the IPE results. These estimates should then be used to assess the cost benefit of various candidate safety enhancements (e.g., design, training, operational, etc.).

Table 1

Primary IPE Core Damage Evaluation Process

Mean CDF Per Sequence Group (per reactor year)	Licensee Response
Greater than 1E-4 or greater than 50 percent of total CDF	1. Find a cost effective plant administrative, procedural or hardware modification with emphasis on eliminating or reducing the likelihood of the source of the accident sequence initiator. 2. If unable to satisfy above response, treat in EOPs or other plant procedure with emphasis on prevention of core damage. 3. If unable to satisfy above responses, ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.
1E-4 to 1E-5 or 20 percent to 50 percent of total CDF	1. Find a cost effective treatment in EOPs or other plant procedure or minor hardware change with emphasis on prevention of core damage. 2. If unable to satisfy above response, ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.
1E-5 to 1E-6	Ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.
Less than 1E-6	No specific action required.

Table 2

Primary IPE Containment Bypass Evaluation Process

Mean Containment Bypass Frequency (per reactor year)	Licensee Response
Greater than 1E-5 or greater than 20 percent of total CDF	1. Find a cost effective plant administrative, procedural or hardware modification with emphasis on eliminating or reducing the likelihood of the source of the accident sequence initiator. 2. If unable to satisfy above response, find cost effective treatment in EOPs or other plant procedure with emphasis on prevention of core damage. 3. If unable to satisfy above responses, ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.
1E-5 to 1E-6 or 5 to 20 percent of total CDF	1. Find a cost effective treatment in EOPs or other plant procedure or <u>minor</u> hardware change with emphasis on prevention of core damage. 2. If unable to satisfy above response, ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.
1E-6 to 1E-7	Ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.
Less than 1E-7	No specific action required.

Figure 1

IPE Core Damage Evaluation Process

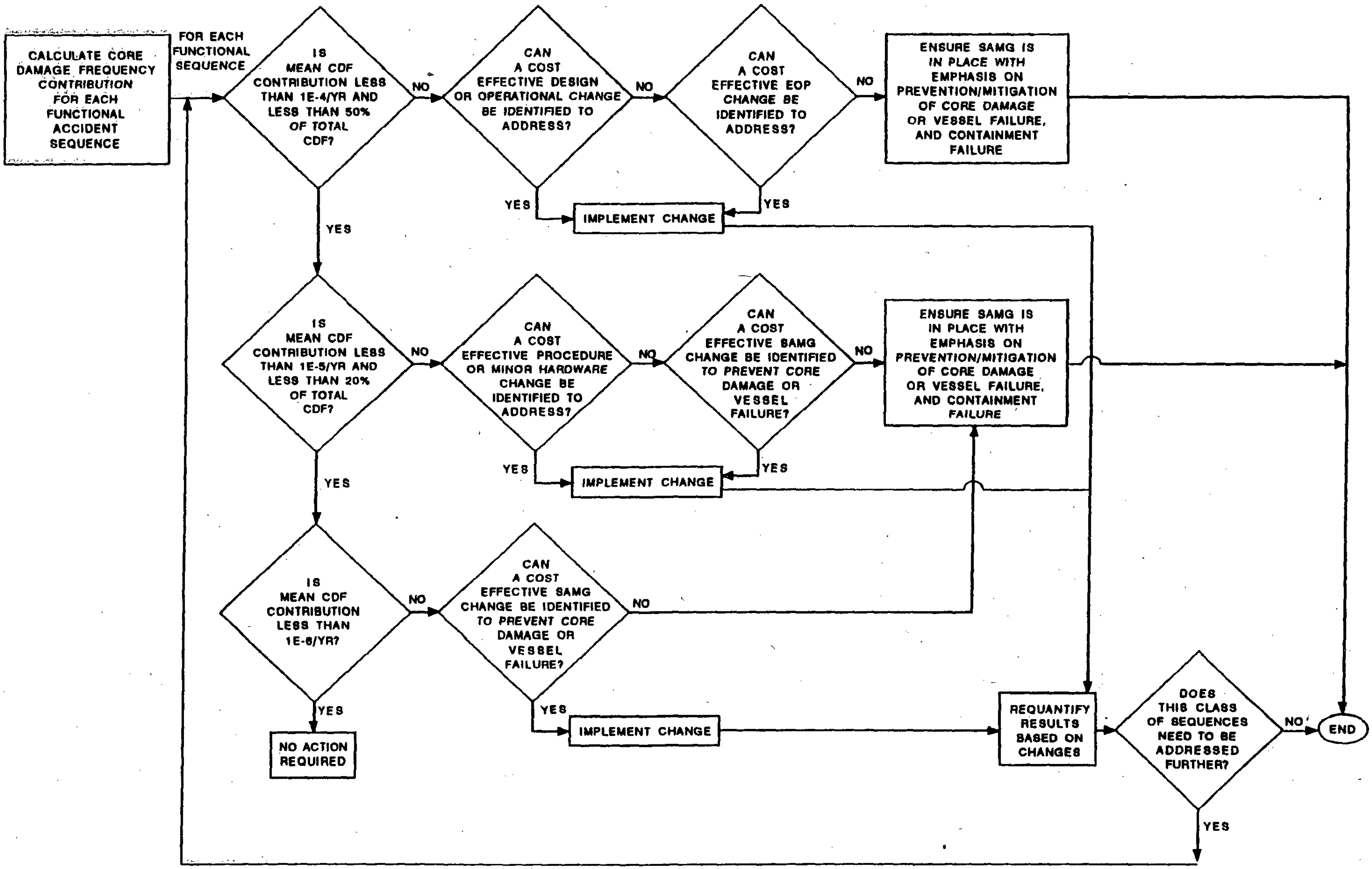
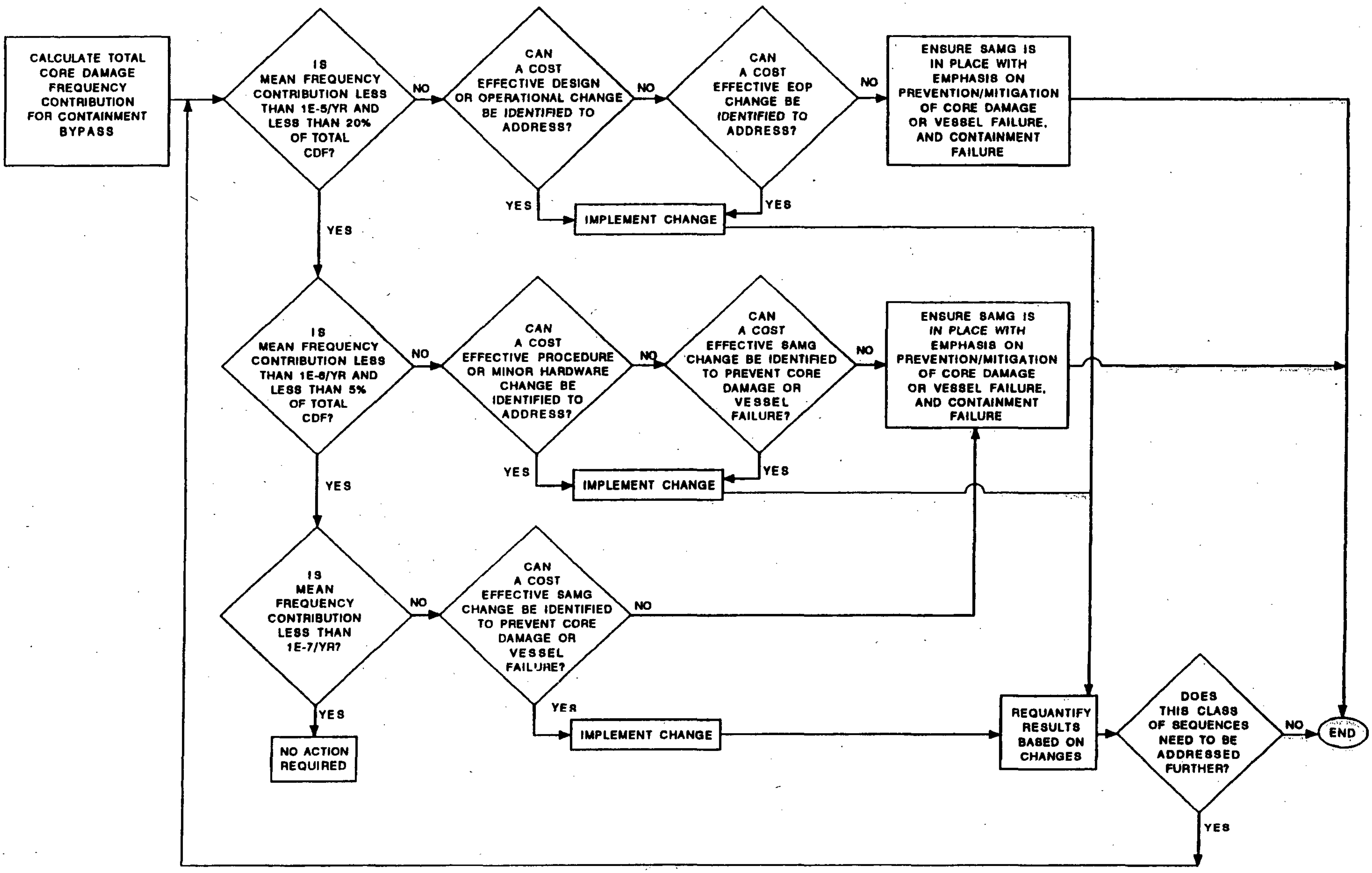


Figure 2

IPE Containment Bypass Evaluation Process



3.0 CONTAINMENT PERFORMANCE IMPROVEMENT (CPI) CLOSURE

In Supplements 1 and 3 to Generic Letter 88-20, the NRC has effectively subsumed the issue of containment performance improvement (CPI) into the IPE process for all plants except for the consideration of a hardened wetwell vent capability for General Electric BWR Mark I's. In Generic Letter 89-16, Mark I plants were requested to volunteer to install a hard piped containment vent to improve containment performance. Licensees of Mark I plants are proceeding to develop designs for such a vent system or are still working with the NRC to evaluate the cost effectiveness of a hard piped vent capability in their plant. Closure of the CPI issue for Mark I plants is to be accomplished through the resolution of the hard piped vent issue, either through installation or other resolution with the NRC, and through the evaluation of other Mark I containment performance improvements in the IPE. Other improvements to be evaluated include those identified by the NRC in Supplement 1 of Generic Letter 88-20.

The closure of the CPI issue is accomplished by the evaluation of IPE insights described in Section 2 and the evaluation of potential containment performance improvement issues identified by the NRC in Enclosure 2 of Supplement 1 to Generic Letter 88-20 (Mark I plants) and in Supplement 3 to Generic Letter 88-20 for other containment types.

4.0 IPE OF EXTERNAL EVENTS (IPEEE) CLOSURE

4.1 Overview

Another major part of the NRC's severe accident program for utilities is the IPE of External Events (IPEEE). The IPEEE involves the systematic evaluation of each plant's response to external event initiators that may potentially lead to severe accidents. The NRC staff, working with input from NEI, has spent considerable effort identifying events for consideration and establishing acceptable examination methods. The result of this effort is described in Supplement 4 to Generic Letter 88-20. This letter identifies the following events for evaluation:

- Seismic Events
- Internal Fires
- High Winds and Tornados
- External Floods
- Transportation and Nearby Facility Accidents
- Other Plant-Unique External Events

The NRC has identified acceptable examination methods for each of these events in Supplement 4 and the associated submittal guidance, NUREG-1407 (Reference 4). In general, the level of effort required for an individual plant will depend on the level of the hazard at that plant. For example, plants in areas of negligible seismic activity will perform a reduced scope seismic assessment. Conversely, plants located on the seismically-active California coast will perform more extensive seismic assessments.

The NRC states in Reference 4 that they will permit some external hazards to be evaluated with methods other than PRA. Therefore, the closure process for IPEEE allows for both probabilistic risk analyses and deterministic methods. For those utilizing PRA, the closure guidelines already established in terms of accident sequence groups for the internal events IPE (Section 2) should be used. However, there are several distinctions between typical

PRA treatment of external events versus internal events worth noting, including: (1) the larger uncertainties in characterizing the initiation and magnitude of the external hazards relative to internal events; (2) the lower likelihood of identifying practical severe accident management opportunities to address external events different from those already being developed to mitigate events once the core is damaged (and probably included as part of the IPE sequence assessment) and (3) the use of differently defined accident sequence groups.

In evaluating the effects that external events may have on plant structures, systems and components, a distinction should be made between damage and loss of function (or failure). For example, destruction of pipe insulation caused by impact on a nearby plant feature (spatial interaction) is considered damage, but it is not necessarily significant to the IPEEE analysis if the equipment remains capable of performing its intended function. It would be considered a failure only if the pipe is unable to perform the function pertinent to the IPEEE analysis (Reference 10). Considering internal fires as another example, extensive fire damage in a given location may be easily tolerated if the shutdown functions of concern are still provided.

Whether one performed a seismic PRA (SPRA) or a seismic margin assessment (SMA), or a fire PRA (FPRA) or the EPRI Fire-Induced Vulnerability Evaluation (FIVE) method, achieving the objectives of the IPEEE requires a thorough understanding of the unique aspects of each methodology relative to the identification of plant characteristics that dominate the external hazard contribution to plant risk. Especially when dispositioning the insights from the seismic or fire assessment, care must be given to identify candidate safety enhancements that either address, or obviate the item of concern. To do otherwise would result in plant hardware or procedural modifications that only achieve a reduction in the perceived, but not real, risk.

For instance, in a SPRA, the goal of achieving real risk reductions is achieved by plant enhancements which may reduce the likelihood of core damage or reduce public safety significance. Yet, interpretation of seismic PRAs are complicated by the fact that the uncertainties associated with seismic hazards are very large. Thus, before any enhancements are implemented, the dominant core damage sequences should be carefully examined to assure the calculations leading to those sequences are performed with the same relative degree of robustness. For example, if one believes a sequence is dominated because the fragility calculations for a component involved in that sequence were performed extremely conservatively, these calculations may be repeated using a refined methodology to see if the core damage sequence is still dominant. Only after assurance of the consistency of robustness in calculations and an evaluation of the uncertainties can one conclude which sequences are credible and what real risk reduction opportunities exist.

In a SMA, the same goal can be achieved by examining the High Confidence of Low Probability of Failure (HCLPF) values for the component outliers in the safe shutdown path. In the FIVE methodology, the focus should be on the fire ignition sources, combustible loading and fire protection systems of the fire areas not screened out during the evaluation process. Since the focus of the alternative methodologies is to ensure, with high reliability, a given safe shutdown path capable of maintaining a cooled and subcritical reactor, the limiting components on that safe shutdown path are of primary importance. Before decisions are made relative to selecting enhancements, the ability to address the potential loss of function(s) of the limiting components should be carefully examined.

For example, with results from an SMA, one would typically start with the component with the lowest HCLPF value and identify candidate safety enhancements to increase its ruggedness. However, it is important to keep a broad perspective. That is, an integrated assessment is also necessary to determine whether the HCLPFs for the other outlier components in the safe shutdown path can be appropriately, and cost-effectively, enhanced. It is not productive to increase one component HCLPF tremendously only to find that the next outlier component provides little opportunity for enhancement (i.e., little likelihood of finding cost effective means of increasing its HCLPF to that of the previous component). Nonetheless, the examination of the limiting HCLPF values that are lower than the target HCLPF values should be continued until a clear point is reached where cost effective enhancement is no longer feasible.

4.2 Seismic IPEEE Closure

This section presents guidelines on the appropriate use of seismic IPEEE review approaches in formulating decisions for closure of the seismic severe accident issue. These closure guidelines are substantially consistent with the guidelines and framework recommended for resolving severe accident issues for internal events, yet they also include the use of deterministic review procedures and they reflect important differences in NRC guidance for treatment of external versus internal severe accident initiators. The seismic IPEEE review approaches, which are used to support the closure guidelines, include:

(1) the reduced-scope assessment,
(2) the focused-scope seismic margin assessment (SMA),
(3) the full-scope SMA, and
(4) the SPRA.

Although the NRC's seismic margin methodology is not discussed further in this document, one should recognize that the closure guidance contained herein is equally valid for both the NRC and EPRI SMA approaches. Descriptions of review approaches and discussion of technical background for the recommended seismic IPEEE closure guidelines are presented in more detail in Reference 5. That document provides additional background on the implementation and use of the seismic IPEEE.

Procedures recommended by industry for performing reduced scope assessments, focused-scope SMA or full-scope SMA evaluations for seismic IPEEEs rely on use of the EPRI SMA methodology (Reference 6). In the EPRI SMA approach, success-path logic diagrams (SPLDs) are constructed to convey the various combinations of components or operator actions that lead to a long-term safe shutdown condition, given a seismic margin earthquake (SME) ground motion. An important part of this approach is to walkdown the plant (reviewing the safe shutdown systems and equipment) and to evaluate limiting elements using the SMA methods given in Reference 6.

In the reduced scope assessment, the limiting elements identified during the reviews are evaluated using the plant licensing basis (FSAR) or the Generic Implementation Procedure (GIP) guidelines that were developed as part of Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Plants. The closure guidelines for the reduced-scope plants are described in Section 4.2.1.

In the focused- and full-scope SMA, the outliers (i.e., the elements that do not pass the SMA screening at the SME) are evaluated to estimate the ground motion for which there is a "high confidence of low probability of failure" (HCLPF). To develop a seismic IPEEE closure approach for these plants that is consistent with the internal events closure guidelines, HCLPF capacity values for each success path must be related to major seismic core damage sequence group frequencies. The key element in establishing this relation is the determination of appropriate plant specific review level ground motions (RLGMs). A closure RLGM is the plant specific HCLPF capacity (for a success path with a given functional plant state) that must be met to satisfy a specified core damage frequency criterion associated with the particular major functional state or sequence group. The procedure for obtaining plant specific RLGMs is described in Reference 5; three RLGMs denoted RLGM-A, RLGM-B and RLGM-C, are determined corresponding to three different core damage frequency-based closure guidelines. RLGM results for 58 central and eastern United States plants are presented in Reference 7. The closure guidelines for the focused- and full-scope categories are given in Section 4.2.2.

In the SMA review, two alternative success paths are chosen for distinct functional SPLDs. Each alternate success path should involve substantially different components and different functional-sequence conditions. The motivation for developing two alternate success paths is to demonstrate redundancy; it is therefore important to evaluate each success path against the closure guidelines.

The success path can be used as a conservative surrogate to a functional accident sequence; hence, closure guidelines defined in terms of functional accident sequences may be applied to the success path sequences. Failure along any success path will be dominated by the component having the lowest HCLPF capacity. So, instead of evaluating the success path as a complete sequence, components on the success path are treated individually. If the HCLPF capacity of each component on a given success path exceeds the guidelines based on a RLGM, then the corresponding guidelines in terms of functional accident sequence frequency are likewise satisfied.

This understanding allows closure guidelines already established in terms of accident sequence groups for the internal events IPE evaluation to be used in terms of component HCLPF comparisons for focused- and full-scope seismic IPEEE evaluation, to achieve a substantially consistent development. If a SPRA is performed, the resulting core damage frequency is treated in terms of core damage sequence groups consistent with the closure guidelines discussed earlier for internal events (Section 2). The closure guidelines for SPRA evaluations are delineated in Section 4.2.3.

4.2.1 Seismic-IPEEE Closure Using Reduced-Scope Evaluation

If the seismic IPEEE is conducted using the reduced-scope evaluation, the closure process consists of the following steps:

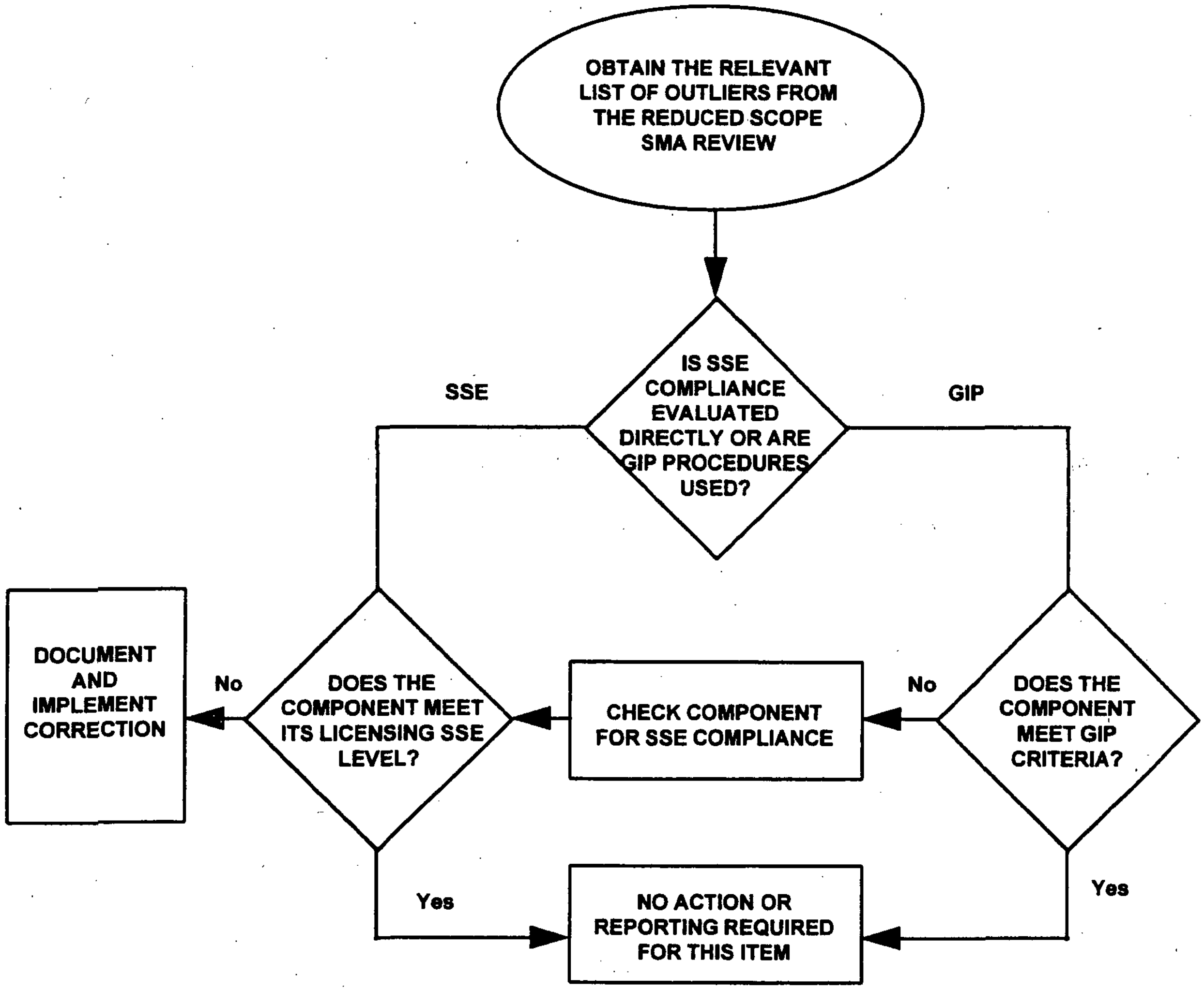
- Delineate preferred and alternate success paths and their major functional states;
- Develop a list of screened-in outliers using SMA screening tables; and
- Evaluate elements for compliance with licensing commitments (FSAR) or with the GIP guidelines based on earthquake experience verification.

If the FSAR commitment is satisfied for a particular component, then closure is reached with respect to that component. The GIP guidelines can be used in lieu of a direct FSAR-consistent evaluation. Figure 3 describes, in flowchart form, the framework for closure when a reduced-scope evaluation is conducted.

The seismic-IPEEE-related closure process is completed when every reduced-scope evaluation element has been evaluated, and actions have been determined, documented and scheduled for implementation.

Figure 3

Seismic-IPE Closure Process Recommended for Reduced-Scope Evaluation



4.2.2 Seismic-IPEEE Closure Using Focused-Scope or Full-Scope SMA Methodology

If the seismic IPEEE is conducted using focused- or full-scope SMA methodology, the closure evaluation process consists of the following steps:

- Delineate preferred and alternate success paths and their major functional states;
- Develop an initial list of screened-in elements using SMA screening tables;
- Calculate HCLPF capacities for initial screened-in elements using the median NUREG/CR-0098 (5% damped) spectrum as input and develop a list of screened-in remaining elements (i.e., those with a calculated HCLPF capacity less than NUREG/CR-0098 spectrum anchored at the RLE peak ground acceleration);
- Obtain RLGMs to be used for evaluation of remaining elements; and
- Evaluate remaining elements against closure guidelines by comparing component HCLPF values with RLGMs.
- Report results of evaluation to the NRC including the schedule for any proposed safety enhancements.

Prior to evaluating success path elements against closure guidelines, HCLPF capacities are computed for the appropriate set of components identified in the seismic IPEEE, using the 5% damped NUREG/CR-0098 median spectrum as input. Figure 4 describes the complete pre-closure process used to screen-in a list of remaining outliers from the initial list of SMA outliers.

Although the IPEEE is not intended to be a confirmation of the current licensing basis, it is incumbent upon the licensee to assess those conditions in which there is some question as to SSE compliance. Instances of noncompliance would be handled via the applicable plant procedures.

Figure 5 describes, in flowchart form, the framework for closure evaluation of core-damage success-path elements when a focused- or full-scope SMA is conducted. With the exception of the RLGM-based guidelines in the top row of triangular decision elements, this framework is identical to that for IPE core damage evaluation (Figure 1) in internal events closure. The RLGM-based guidelines are themselves developed to be consistent with the corresponding core damage frequency related guidelines in the internal events IPE closure evaluation.

For seismic containment sequences and related success-path elements (see Reference 5, Appendix D), the SPRA database suggests that substantial margin exists to ensure appropriate containment response. Figure 5 is, therefore, also applied for closure evaluation of success-path elements related to containment performance (Reference 6). Consistent with NRC guidance in Supplement 4 to Generic Letter 88-20 for seismic events (Reference 4), separate closure guidelines for evaluation of containment-related elements and for evaluation of core-damage elements is not required.

The seismic-IPEEE-related closure process is completed when focused- or full-scope SMA limiting elements have been evaluated, any safety enhancements have been scheduled for implementation, and the evaluation results documented, and as appropriate, reported to the NRC.

Figure 4

Seismic-IPE Pre-Closure Assessment for Evaluation of Outliers in Focused-Scope and Full-Scope SMAs

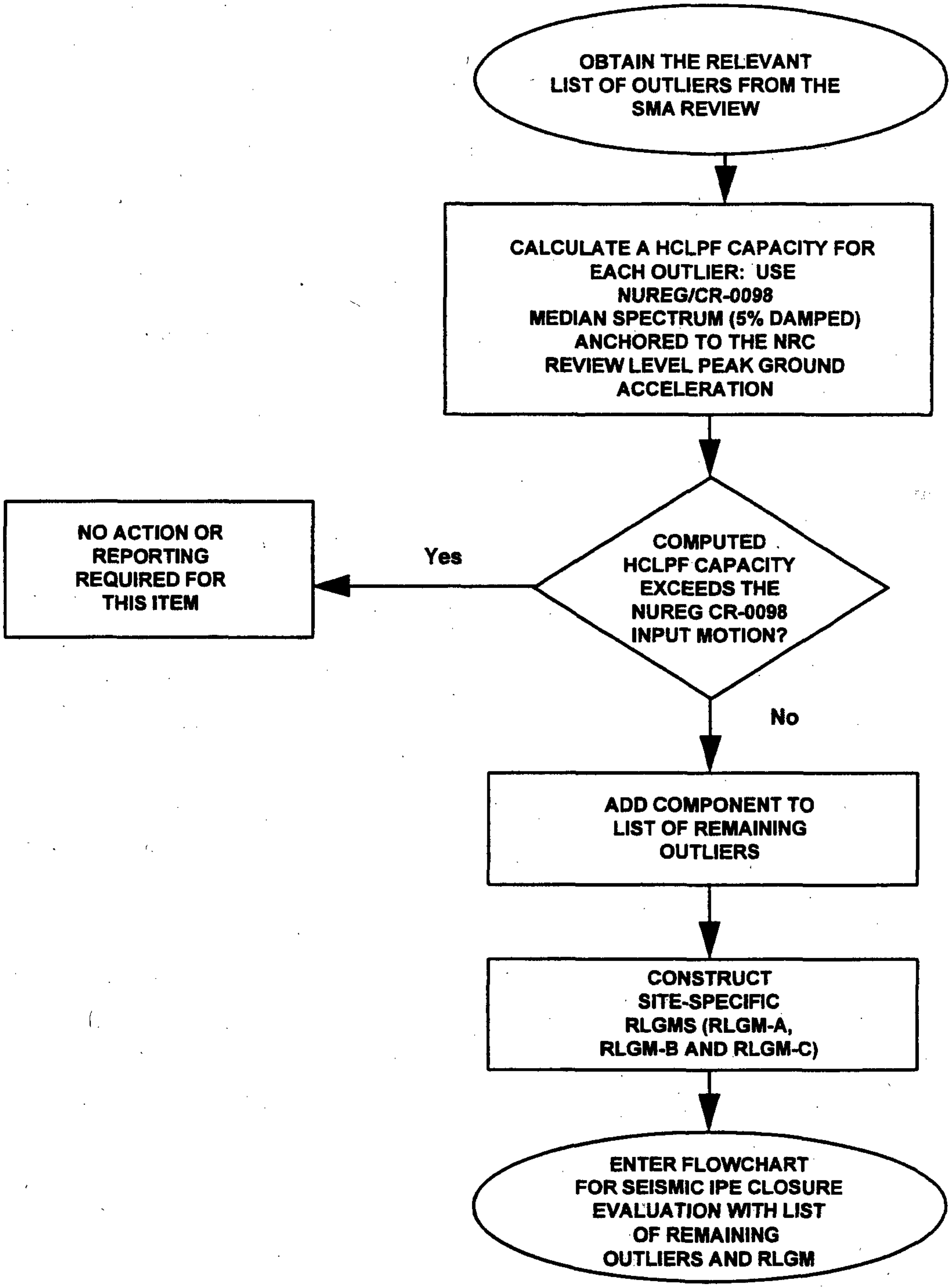
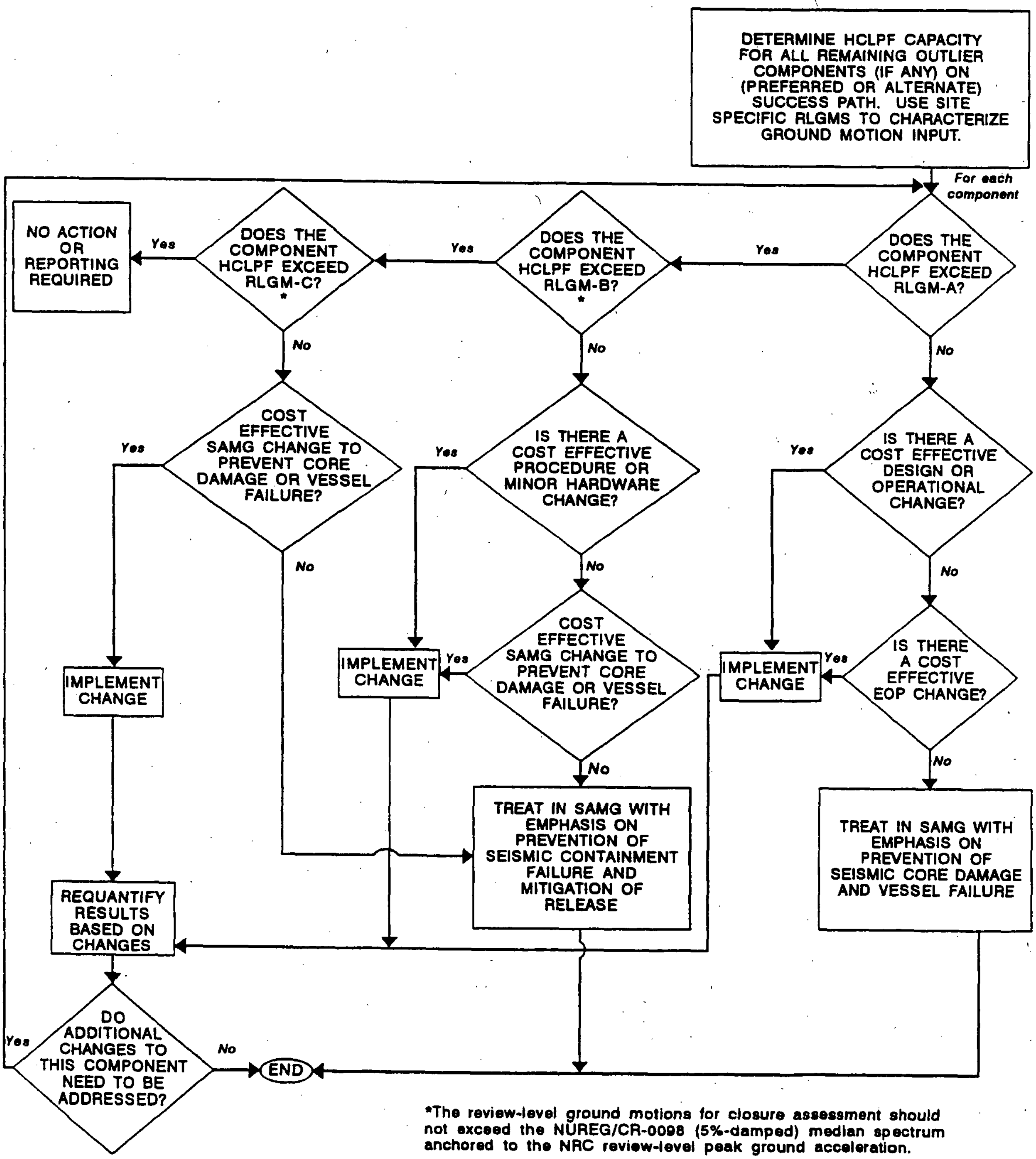


Figure 5

Closure Process Recommended for Seismic IPE Evalution: Focused-Scope and Full-Scope SMAs



*The review-level ground motions for closure assessment should not exceed the NUREG/CR-0098 (5%-damped) median spectrum anchored to the NRC review-level peak ground acceleration.

4.2.3 Seismic-IPEEE Closure Using Seismic PRA Methodology

If a SPRA is performed, the seismic core damage frequency will be treated either as a single core damage sequence group or as multiple sequence groups for evaluation against closure guidelines. If multiple seismic core damage groups are utilized, a similar philosophy to the IPE of internal events should be used in defining the groups. However, in the case of external events it is also reasonable to group based on the nature of the sequence induced by the external hazard (i.e., seismic induced LOCAs, seismic induced station blackout, etc.). Components important in seismic core damage sequences that may lead to containment bypass should also be included for consideration in the closure process. An example grouping scheme for SPRA sequences is provided in Appendix C. This grouping scheme is provided only for the purposes of demonstrating the philosophy of grouping, and other approaches are acceptable. Once the sequences have been grouped, the seismic IPEEE closure process involves comparing seismic core damage group frequencies and containment bypass frequencies to the closure guidelines in Tables 1 and 2, respectively. Thus, the closure process for SPRA implementation is similar to that for the IPE. The seismic-IPEEE-related closure process is completed when SPRA sequences have been evaluated, and any safety enhancements have been scheduled for implementation, and the evaluation results documented, and as appropriate, reported to the NRC.

4.3 Fire IPEEE Closure

The evaluation of severe accident fire vulnerabilities can be accomplished through the performance of a fire PRA (FPRA) or an alternative approach such as Fire Induced Vulnerability Evaluation (FIVE). If a FPRA is performed, then closure is accomplished similar to the SPRA closure process. That is, the fire core damage sequences are grouped, e.g., either on the basis of fire location or in multiple core damage sequence groups and compared to the closure guidelines in Tables 1 and 2. The core damage sequence grouping should be based on the nature of the sequence induced by the fire hazard.

(i.e., fire induced LOCAs, fire induced station blackout, etc.). An example grouping scheme for FPRA sequences is provided in Appendix C. This grouping is only for the purposes of demonstrating the philosophy of grouping, and other approaches are acceptable.

If a utility uses FIVE in performing the fire IPEEE, then the figure of merit of each of the fire compartments which were not considered to be insignificant (i.e., all fire compartments with a figure of merit of greater than 10^{-6}/year) should be compared to the closure guidelines in Table 1. These figures of merit may be multiplied by a fire non-suppression factor to yield a revised figure of merit to be compared to the closure guidelines. The value to be used as a fire non-suppression factor would depend on the individual accident sequences and the capabilities of the plant. The fire non-suppression factor accounts for manual actions such as by a fire brigade to suppress the fire before it has caused the loss of safety functions. This factor should be credited only if a numerical value was not previously credited by Section 6.3.6 of the FIVE methodology. This factor would be specific to each sequence and to each plant. In lieu of a plant specific number, an estimate of 0.4 can be used to calculate the figure of merit. The Fire Risk Scoping Study (Reference 8) has calculated values of 0.73 to 0.14. Another fire risk study referenced in that report calculated values of 0.4 to 0.04.

The resulting figure of merit is used to compare to the core damage frequencies in Table 1, although it should not be considered an estimate of core damage in contexts outside of this application.

4.4 Other External Hazard IPEEE Closure

The IPEEE generic letter and NUREG-1407 describe a series of analysis steps which can be undertaken to address each of the other external event initiators. This process involves the performance of increasing levels of detail, effort and resolution. If the licensee can demonstrate that the 1975 Standard Review Plan (SRP) criteria are met or potentially limiting plant characteristics are demonstrated to be insignificant, then the event can be considered to be fully addressed.

The first step of the process is to review plant specific hazard data and licensing bases to determine the current resolution of the event for the plant. The second step involves the identification of significant changes since issuance of an operating license to the facility with respect to (1) military/industrial facilities, (2) on-site storage or other activities involving hazardous materials, (3) transportation, or (4) developments that could affect the original design. Following this, the utility is expected to determine whether the plant/facilities meet current criteria (1975 NRC SRP criteria) by reviewing the compiled information and performing a plant walkdown to verify plant data. If the SRP criteria are met, then the event is not considered to present a potential threat which could be a vulnerability.

In the event the SRP criteria are not met, then the utility can assess whether the hazard frequency is sufficiently low. This is accomplished by calculating the frequency of the original design basis hazard combined with the conditional probability of core damage to determine if the frequency is less than 10^{-6}/year. If so, then the event can be excluded from further consideration. If not, then a bounding analysis can be performed to provide a conservative calculation showing either the hazard would not result in core damage or the core damage frequency is sufficiently low (i.e., less than 10^{-6}/year). If the bounding analysis is not adequate to eliminate the event, then a PRA can be performed.

The method of closure of this IPEEE element depends upon the approach used to resolve the hazard. If the SRP criteria are met, or the hazard frequency was found to be sufficiently low, or the bounding analysis found the frequency of core damage to be less than 10^{-6}/year, the event can be excluded from further consideration. If a PRA was performed for the hazard, then closure is accomplished by comparing the frequency of core damage sequence groups due to the hazard to the closure guidelines in Tables 1 and 2. As such, the core damage frequency of each hazard may be treated as a single core damage sequence group, or some other logical set of grouping similar to what was done for fire or seismic events. The process is complete once the results of the evaluation are documented, any safety enhancements scheduled, as may be appropriate, and reported to the NRC as part of IPEEE report.

5.0 SEVERE ACCIDENT MANAGEMENT CLOSURE

5.1 Scope of Severe Accident Management

Accident management consists of those actions taken during the course of an accident by the Emergency Response Organization (ERO); specifically plant operations, technical support and plant management staff, in order to:

- Prevent the accident from progressing to core damage;
- Terminate core damage progression once it begins;
- Maintain the capability of the containment as long as possible; and
- Minimize on-site and off-site releases and their effects.

The latter three actions constitute a subset of accident management referred to as severe accident management, or more specifically, severe accident mitigation. Post-TMI actions and IPE insights have already addressed most aspects of preventing core damage. The focus of the industry effort is to provide guidance where Emergency Operating Procedures (EOPs) are no longer effective, or revise EOPs if appropriate.

The goal of severe accident management is to enhance the capabilities of the ERO to mitigate severe accidents and prevent or minimize any off-site releases. The objective is to establish core cooling and ensure that any current or immediate threats to the fission product barriers are being managed. To accomplish this the ERO should make full use of <u>existing</u> plant capabilities, including standard and non-standard uses of plant systems and equipment.

Significant interaction among utility, INPO, EPRI, vendor Owners Groups, NRC, and other recognized experts has produced the foundation of actions and plant response from which plant-specific severe accident management guidance can be developed (see References 11, 12 and 13). These actions can be categorically divided into elements similar to those described by the NRC in SECYs 88-147 and 89-012 (References 3 and 9).

5.2 Severe Accident Management Closure Process

The severe accident management closure process for a given licensee is recommended to consist of the following steps (illustrated in Figure 6):

- Evaluate industry-developed bases and Owners Group severe accident management guidance (SAMG) along with the plant IPE, IPEEE and current capabilities, to develop severe accident management guidance for accidents found to be important in your plant as screened with the criteria provided in Section 2.0. Consider other generic and plant-specific information (e.g., NRC and industry studies, PSA results, etc.) as appropriate;

- Interface SAMG with the plant's Emergency Plan;

- Incorporate severe accident material into appropriate training programs; and

- Establish a means to consider and possibly adopt new severe accident information from licensee self assessments, applicable NRC generic communications, PRA studies, etc.

Because this is an industry initiative, there are no specific regulatory criteria. Rather, industry has defined its goals and objectives by its actions relative to severe accident management. These include, but are not limited to, performance and submittal of IPE and IPEEE, development of generic (Owners Group) SAMG, and numerous interactions at various levels among industry, NRC and vendor personnel. The following element descriptions provide a tool that may be used for focusing licensee efforts to enhance their capabilities.

5.3 Severe Accident Management Implementing Elements

5.3.1 Severe Accident Management Guidance/Strategies for Implementation

Guidance is to be provided for use by ERO personnel in assessing plant damage, planning and prioritizing response actions, and implementing strategies that delineate actions inside and outside the control room. Strategies and guidance will be interfaced with the utility EOPs and Emergency Plans.

The guidance should include: (1) an approach for evaluating plant conditions and challenges to plant safety functions; (2) operational and phenomenological conditions that may influence the decision to implement a strategy, and which will need to be assessed in the context of the actual event; and (3) a basis for prioritizing and selecting appropriate strategies, and approaches for evaluating the effectiveness of the selected actions.

The strategies should make maximum use of existing plant equipment and capabilities, including equipment and alignments that may not be part of the typical "safety-related" systems. Critical resources and procedures, if necessary, to implement strategies will be identified and reasonably available, but need not be prestaged. Rather, what is important is a clear delineation of the flow of information, identification of the decisions that have to be made, and some up front consideration of the viability of implementing the more significant strategies (e.g., not detailed procedures, but a small number of lists that include a description of system lineups, benefits and negative impacts, interlocks to be overridden, special equipment required, etc.).

5.3.2 Training in Severe Accidents

Severe accident training should be provided for ERO personnel commensurate with their responsibilities defined in the Emergency Plan. In particular, training is recommended for those specific personnel with the following severe accident assessment and mitigation responsibilities:

- evaluators responsible for assessing plant symptoms in order to determine the plant damage condition(s) of interest and potential strategies that may be utilized to mitigate an event

- decision makers in the ERO designated to assess and select the strategies to be implemented

- implementers responsible for performing those steps necessary to accomplish the objectives of the strategies (e.g., hands-on control of valves, breakers, controllers, and special equipment)

Existing training programs already address most of the tasks associated with strategy implementation by implementers (e.g., licensed and non-licensed operators, maintenance personnel, radiation protection specialists, etc.). Thus, it is expected that severe accident considerations should be a minor addition to the scope of their training, commensurate with the frequency, importance and difficulty of the potential tasks. The areas of emphasis and level of detail in the implementers training will be different than that provided to the evaluators or decision makers.

Suggested learning objectives and related training materials will be developed using a systematic approach to training and include training techniques proven successful with similar materials.

5.3.3 Computational Aids for Technical Support

ERO personnel should be provided computational aids, as appropriate, in estimating key plant parameters and plant response relative to accident management decisions. The aids should be easy to use and need not be computer based.

5.3.4 Information Needed to Respond to a Spectrum of Severe Accidents

Provide an awareness, and encourage use, of instrumentation that is reasonably expected to be available for assessing plant status. The availability and survivability of the information source and the ability of these sources to provide indication of sufficient accuracy for the intended use should be considered. Alternative, indirect means for providing necessary information should also be considered.

5.3.5 Delineation of Decision-Making Responsibilities

Ensure responsibilities for authorizing and implementing accident management strategies are delineated as part of the Emergency Plan. The ERO personnel task descriptions should be modified to specify responsibilities. Nonetheless, the decision-making process needs to be flexible enough to accommodate situations beyond the scope of currently recognized situations.

5.3.6 Utility Self-Evaluation

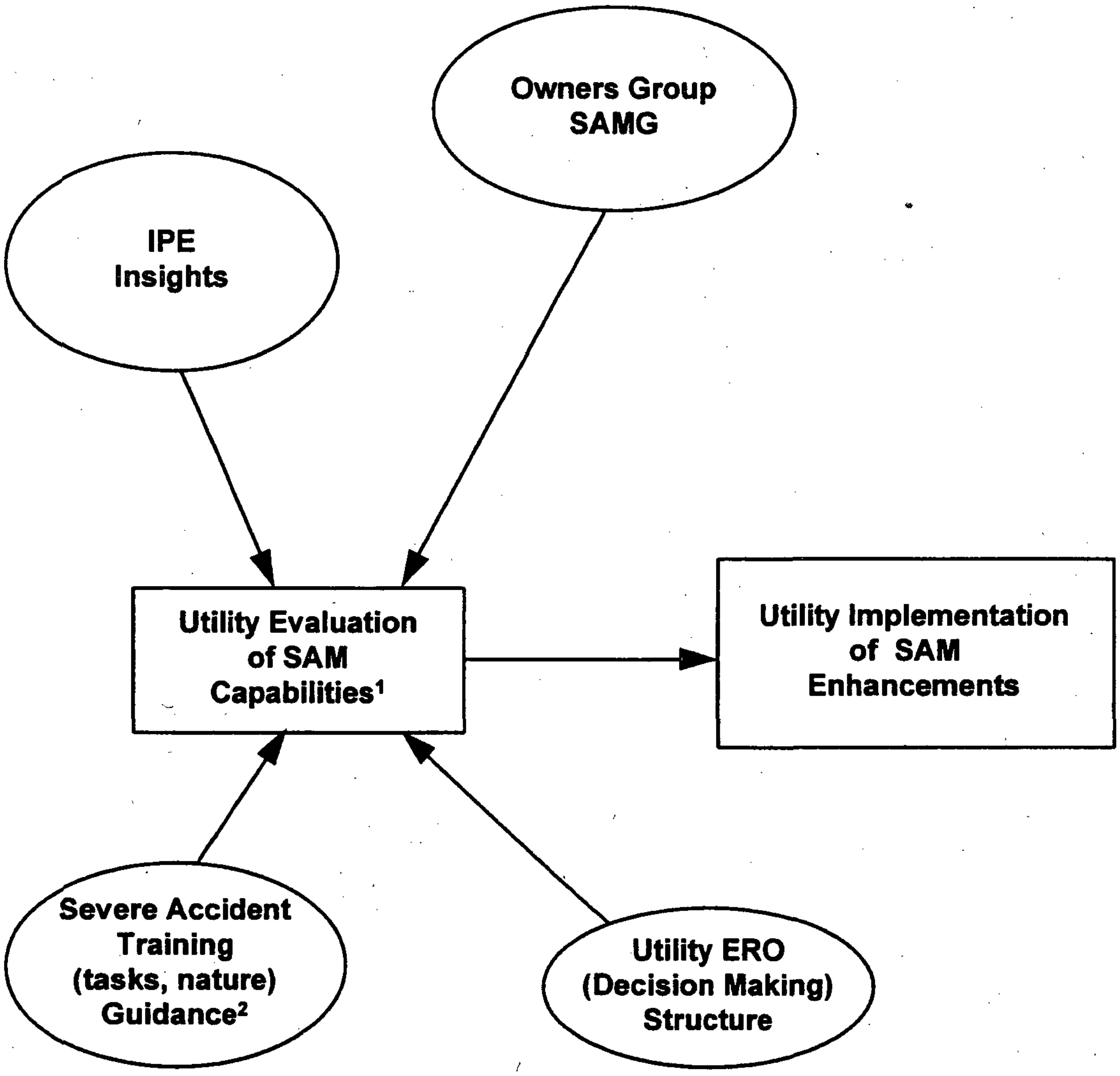
Self-evaluation of the licensee's severe accident response capability is recommended to ensure its feasibility and usefulness. Upon creation of the plant-specific SAMG, an initial evaluation should be performed to ensure the material has been integrated into the licensee's emergency response capability without adversely affecting emergency response.

Subsequently, periodic table-top and/or inter-facility mini-drills should be utilized to ensure that ERO personnel are familiar with the use of the SAMGs and with the interfaces and delineation of responsibilities between EROs during SAMGs use. The objective of the table-top and/or inter-facility mini-drills should be training, evaluating and improving the in-plant, severe accident management response capability. These activities should include exercising of preventive or mitigative measures as well as appropriate critiques immediately following the drill to capture lessons learned (e.g., assess performance and perform a technical assessment of any useful preventive or mitigative measures identified during drills).

There is no need for such mini-drills to be part of the graded Emergency Plan exercises; in any case, evaluations of severe accident strategy use should be separate from these formal exercises.

Figure 6

Severe Accident Management Closure Process



Key:
SAM = Severe Accident Management
SAMG = Severe Accident Management Guidance
ERO = Emergency Response Organization
IPE = Individual Plant Examination

1. Utilize NEI Report 91-04 Revision 1 (formerly NUMARC Report 91-04) section 2 screening criteria to assist in determining amount of effort warranted. NUMARC Report 92-01 offers insights as to appropriate attributes for given accident management elements.

2. Generic industry task analysis, learning objectives and activities and lesson plans will be available.

6.0 REFERENCES

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2) "Guidelines for 10 CFR § 50.59 Safety Evaluations," NSAC-125, Final Report, June 1989.

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

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8) J. A. Lambright, et. al., "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," NUREG/CR-5088, January, 1989.

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10) Paul Smith, "Earthquake Safety and Licensing Report," March 15, 1991, Vol. 3, No. 3.

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12) EPRI Report TR-101869, Severe Accident Management Guidance Technical Basis Report, December 1992.

13) NSSS Owners Group-Specific Accident Management Guidance Reports, to be published.

14) USNRC, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains, "NUREG-1488, October 1993.

15) NEI White Paper, "Justification for Reduction in IPEEE Program Based on Revised LLNL Seismic Hazard Results," forwarded to NRC (Dr. A. Thadani) by NEI letter, April 5, 1994.

16) NEI Letter, "Industry Comments on NRC Workshop and Draft Contractor Reports on Seismic Revisit of the IPEEE," forwarded to NRC (Dr. J. Murphy), November 18, 1994.

APPENDIX A
DEFINITIONS

DEFINITIONS

SEVERE ACCIDENTS are those that result in catastrophic fuel rod failure, core degradation and fission product release into the reactor vessel, containment or the environment.

DESIGN BASES are the information that identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state-of-the-art" practices for achieving functional goals or (2) requirements derived from analysis (based on calculations and/or experiments) of the effects of a postulated accident for which a structure, system or component must meet its functional goals. (10 CFR § 50.2)

BYPASS EVENTS are those events that involve the release of fission products through an unisolated breach of the primary system outside the primary containment. Events involving loss of failure of primary containment occurring after core damage are <u>not</u> considered bypass events.

SEVERE ACCIDENT MANAGEMENT GUIDANCE (SAMG) is the plant-specific guidance developed to assist the plant operating and technical staff in implementing strategies for the best use of the existing plant capabilities to diagnose, respond to, and recover from a severe accident.

COST-EFFECTIVE enhancement is one in which the cost of implementation of the enhancement under consideration is less than the expected value of the averted risk.

APPENDIX B
EXAMPLE FUNCTIONAL SEQUENCE DEFINITIONS
FOR IPE OF INTERNAL EVENTS

Table B-1

EXAMPLE BWR FUNCTIONAL ACCIDENT SEQUENCE DEFINITIONS

FUNCTIONAL ACCIDENT SEQUENCE	DEFINITION
IA	Accident Sequences Involving Loss of Coolant Inventory Makeup in Which the Reactor Pressure Remains High
IB	Accident Sequences Involving a Loss of All AC Power and Loss of Coolant Inventory Makeup (i.e., Station Blackout)
IC	Accident Sequences Involving a Loss of Coolant Inventory Makeup Induced by an ATWS Sequence
ID	Accident Sequences Involving a Loss of Coolant Inventory Makeup in Which Reactor Pressure Has Been Successfully Reduced
II	Accident Sequences Involving Loss of Containment Heat Removal Leading To Containment Failure and Subsequent Loss of Coolant Inventory Makeup
IIIA	Accident Sequences Involving Inadequate Coolant Makeup Initiated By RPV Rupture Where Containment Failure Has Not Occurred
IIIB	Accident Sequences Initiated or Resulting in Small or Medium LOCAs for Which the Reactor Cannot be Depressurized and Inadequate Coolant Inventory Makeup is Available
IIIC	Accident Sequences Initiated or Resulting in Medium or Large LOCAs for Which the Reactor is at Low Pressure and Inadequate Coolant Inventory Makeup is Available
IIID	Accident Sequences Which are Initiated by a LOCA or RPV Failure and for Which the Vapor Suppression System is Inadequate, Challenging Containment Integrity
IV	Accident Sequences Involving an ATWS Leading to Containment Failure Due to High Pressure and Subsequent Loss of Inventory Makeup
V	Unisolated LOCA Outside Containment Leading to Loss of Effective Coolant Inventory Makeup

Table B-2

EXAMPLE PWR FUNCTIONAL ACCIDENT SEQUENCE DEFINITIONS

FUNCTIONAL ACCIDENT SEQUENCE	DEFINITION
IA	Accident Sequences Involving Loss of Both Primary and Secondary Heat Removal in the Injection Phase
IB	Accident Sequences Involving Loss of Both Primary and Secondary Heat Removal in the Recirculation Phase
IIA	Accident Sequences Involving an Induced LOCA with Loss of Primary Coolant Makeup or Adequate Heat Removal in the Injection Phase
IIB	Accident Sequences Involving an Induced LOCA with Loss of Primary Coolant Makeup or Adequate Heat Removal in the Recirculation Phase
IIIA	Accident Sequences Initiated by a Small LOCA with Loss of Primary Coolant Makeup or Adequate Heat Removal in the Injection Phase
IIIB	Accident Sequences Initiated by a Small LOCA with Loss of Primary Coolant Makeup or Adequate Heat Removal in the Recirculation Phase
IIIC	Accident Sequences Initiated by a Medium or Large LOCA with Loss of Primary Coolant Makeup in the Injection Phase
IIID	Accident Sequences Initiated by a Medium or Large LOCA with Loss of Primary Coolant Makeup or Adequate Heat Removal in the Recirculation Phase
IV	Accident Sequences Involving Failure of Reactivity Control
VA	Systems LOCA Outside Containment with Loss of Effective Coolant Inventory Makeup
VB	Steam Generator Tube Rupture with Loss of Effective Coolant Inventory Makeup

APPENDIX C
EXAMPLE ACCIDENT SEQUENCE GROUPINGS FOR IPE OF EXTERNAL EVENTS

Table C-1

EXAMPLE SEISMIC ACCIDENT SEQUENCE GROUP DEFINITIONS

ACCIDENT SEQUENCE GROUP	DEFINITION
S1	Seismic Induced Accident Sequences Involving Non-Station Blackout Transients
S2	Seismic Induced Accident Sequences Involving Station Blackout Transients
S3	Seismic Induced Accident Sequences Involving Medium and Large LOCAs
S4	Seismic Induced Accident Sequences Involving Small LOCAs
S5	Seismic Induced Accident Sequences Involving ATWS Events
S6	Seismic Induced Accident Sequences Involving Containment Bypass

NOTE: Not all plants are expected to have measurable frequencies in all of these sequence groups.

Table C-2

EXAMPLE FIRE ACCIDENT SEQUENCE GROUP DEFINITIONS

ACCIDENT SEQUENCE GROUP	DEFINITION
F1	Fire Induced Accident Sequences Involving Non-Station Blackout Transients
F2	Fire Induced Accident Sequences Involving Station Blackout Transients
F3	Fire Induced Accident Sequences Involving LOCAs
F4	Fire Induced Accident Sequences Involving ATWS Events
F5	Fire Induced Accident Sequences Involving Containment Bypass

NOTE: Not all plants are expected to have measurable frequencies in all of these sequence groups.

APPENDIX D

NRC STAFF COMMENTS ON THE DRAFT NEI SEVERE ACCIDENT ISSUE CLOSURE GUIDELINES AND NEI RESPONSES

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

November 14, 1991

William H. Rasin, Vice President and Director
Technical Division
Nuclear Management and Resources Council
1776 Eye Street, NW
Suite 300
Washington, D.C. 20006-2496

Dear Mr. Rasin:

Responding to the request of your September 26, 1991 letter, we have reviewed NUMARC Report 91-04, "Severe Accident Issue Closure Guidelines," of September 1991. Our overall reaction is that the development of these guidelines was a worthwhile undertaking and that they should prove useful to utilities in closing severe accident issues. We also support your position that, in keeping with the philosophy of the Individual Plant Examination (IPE) process, a decision by a licensee to use these guidelines should be voluntary. Consistent with this philosophy, the NRC does not plan to enforce any specific set of guidelines at this time and would request that NUMARC ensure that its guidelines do not contain any statements that may lead licensees to believe that NRC has endorsed them.

We also recommend that NUMARC provide explicit guidance to utilities to give prompt attention to identified vulnerabilities. For example, if potential vulnerabilities are identified during the course of the study, it would be prudent to not await submittal of the IPE before notifying the NRC. Potential vulnerabilities should be evaluated in detail to determine if they are real and, if so, prompt attention should be given to the vulnerability prior to submittal of the IPE, as discussed in Generic Letter 88-20. Following this approach for vulnerabilities such as the internal flooding sequence identified during the Surry IPE would provide greater assurance that potential safety issues will be given appropriate attention as early in the review process as possible.

The enclosure provides our specific comments on the NUMARC guidelines. Our only major comment on these guidelines involves the closure guidance for the seismic Individual Plant Examination of External Events (IPEEE). The document refers to a "plant specific review level ground motion (RLGM)." Without knowing the details, we cannot provide specific comments. However, this approach may introduce unnecessary complications and controversy regarding the hazard curves used to establish the RLGM.

We appreciate the opportunity to comment on these guidelines and hope that they will prove useful to licensees in closing severe accident issues.

Sincerely,

Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosure: As stated

ENCLOSUR

COMMENTS ON NUMARC REPORT 91-04, "SEVERE ACCIDENT ISSUE CLOSURE GUIDELINES," DRAFT, SEPTEMBER 1991

1. Section 1.2, Guiding Principle of Closure Guidelines (p.2)

The NUMARC report should not include any statements which refer to the NRC's use of the guidelines for closing severe accidents. Delete the following: (1) "and NRC" and "and the safety objectives of the regulatory agency" from the paragraph 1, and (2) "and the regulator" from the paragraph 4.

2. Section 1.4, Consideration of Plant Changes... (p.3)

The discussion of the 10 CFR 50.59 safety evaluation process is confusing. We are not certain what NUMARC is proposing.

3. Table 1, Primary IPE Code Damage Evaluation Process (p.11)

The report does not provide a reason for distinguishing sequences with a probability above 10^{-4} occurrences per year from those below that value. The staff believes licensees should perform cost-effective modifications that might eliminate or reduce the likelihood of an accident sequence initiator for sequences in the range of 10^{-4} to 10^{-5} per reactor year. Also, the word "minor" is unnecessary and should be deleted. To be cost effective, changes for sequences with little risk would necessarily be minor.

4. Section 4.2, Seismic IPEEE Closure

The discussion on seismic review approaches (p.19) should be made consistent with the NRC's guidance document, NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events." This discussion should include statements identifying the two approaches: seismic margins methods and seismic probabilistic risk assessments (PRAs), and that both are acceptable to NRC. If a licensee chooses to use the seismic margins method in its IPEEE, the seismic conditions at the individual site will determine the amount of effort needed for the examination. The licensee should perform one of the three assessments: reduced scope, focused scope, or full scope, as specified in NUREG-1407.

The term "review level ground motions (RLGMs)" (p.20) has not been clearly defined, and we cannot determine how this term compares with the "review level earthquakes (RLEs)," which is used in the NUREG-1407. The NRC has not reviewed References 6 and 8 (pp. 20 & 31). Therefore, we cannot determine if the RLGMs specified in these documents are consistent with those specified in the NUREG-1407, and if the recommended resolutions are acceptable to the NRC. However, these appear to add unnecessary complication and will lead to controversy regarding which hazard curves are used to establish the RLGM. Our past experiences with PRAs and margins reviews have seldom required us to perform complicated analysis to determine if any corrective actions would be cost effective.

5. Section 4.2.2, Seismic IPEEE Closure Using Focused... (p.21)

   The closure evaluation process should provide discussions on using the existing spectra the plant used in the original seismic design, by scaling, in addition to the discussion of "calculate HCLPF capacities...using the median NUREG/CR-0098 spectrum." The NRC does not require licensees to generate new spectra to meet the NUREG/CR-0098 specifications for the seismic IPEEE stated in NUREG/CR-0098, "Seismic Review of Selected Nuclear Power Plants."

6. Section 4.2.3, Seismic IPEEE Closure Using Seismic PRA Methodology (p.24)

   NUMARC should also provide a discussion on how HCLPFs can be used in the evaluation process for closing severe accident issues. If the licensee does not provide HCLPFs, the NRC will calculate and use the HCLPFs in the process of resolving severe accident vulnerabilities resulting from seismic events.

7. Section 4.3, Fire IPEEE Closure (p.26)

   The fire non-suppression factor should not be arbitrarily set at 0.4. Each licensee should provide its own basis for the fire non-suppression factors that is used in the analysis.

8. Appendix C, Tables C-1 and C-2 (pp. C-1 & C-2)

   The seismic and fire accident sequences should have the group of loss-of-coolant accidents (LOCAs) divided into two groups, one for small LOCAs and the other for medium and large LOCAs.

NEI RESPONSE TO NRC COMMENTS

1. Section 1.2: The recommended changes have been made.

2. Section 1.4: The discussion has been expanded to clarify NEI's intent.

3. Table 1: Sequences having a probability greater than 1E-4 per reactor year are distinguished from those between 1E-4 to 1E-5 per reactor year, because of our desire to emphasize a hierarchy in the objectives to be emphasized with respect to consideration of plant administrative, procedural or hardware modification. In the former case, the objective is to first consider elimination or reduction of the likelihood of the initiator. If that is not feasible (e.g., cost-effective), then consider actions to prevent core damage, etc. In the latter case, the objective is to focus on prevention, then mitigation, because thumbrules indicate a change in CDF less than 1E-4 will most likely not justify expenditure of dollars large enough to eliminate accident initiators.

   NEI agrees with the NRC staff that cost-benefit analysis is an appropriate discriminator when considering modifications that might eliminate or reduce the likelihood of an accident sequence, including those above 1E-4 or in the range of 1E-4 to 1E-5 per reactor year. The adjective minor is added to the latter category, because those are the only types of changes that would be shown to be cost-effective.

4. Seismic IPEEE Closure: No changes have been made as a result of the NRC staff comments. However, the following clarifications or thoughts are offered on the NRC staff comments:

   a. First paragraph: The comment appears to reflect a lack of understanding of the purpose of the closure guideline. It is intended to pick up where the NRC

Generic Letter and guidance document leave off. That is, the staff guidance suggests how one might conduct the seismic review. The closure guideline provides guidance as to how to disposition any insights resulting from the review.

b. The Review Level Ground Motion approach referred to in the closure guidelines is used to allow each site to correlate the target CDF values contained in Tables 1 and 2 of the guidelines to the "equivalent" HCLPF values for its specific site. A copy of references 6 and 8 will be forwarded to NRC staff for information. As NRC staff is aware, NEI recommended to the industry only to use the EPRI hazard curves if performing a PRA. The controversy over the use of the Livermore and EPRI hazard curves is not affected by the RLGM calculational process.

5. Section 4.2.2: The statement implies a desire on the part of NRC staff to correlate the plant seismic design bases values to the calculated capacity derived during the IPEEE. We do not believe that is a valuable indication as to the seismic robustness of a given component. Instead, we prefer the risk-based approach discussed in the guidelines.

6. Section 4.2.3: As the NRC staff is aware, NEI has informed licensees that there is little value in transcribing PRA results into HCLPFs.

7. Section 4.3: For the application intended and uncertainties prevalent in the external PRA process, a factor of 0.4 is adequate. There seems to be little value in taking the effort to more precisely define this value on a plant-specific basis.

8. Appendix C: Table C-1 was revised to distinguish between small and medium/large break LOCAs. Table C-2 was not revised.

APPENDIX E

NRC STAFF COMMENTS ON THE DRAFT FORMAL INDUSTRY POSITION ON SEVERE ACCIDENT MANAGEMENT AND NEI RESPONSES

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

JUN 20 1994

Mr. William H. Rasin
Nuclear Energy Institute
1776 Eye Street, N.W., Suite 400
Washington, DC 20006-3708

Dear Mr. Rasin:

This is in response to your letter of May 10, 1994, in which you forwarded a revised draft of the "formal industry position" on severe accident management, and a separate enclosure addressing the NRC staff comments contained in a February 16, 1994, letter.

The staff has carefully considered the revised position, as well as the substantive industry effort and commitment behind the initiative, in judging whether the formal position provides an acceptable vehicle for obtaining the types of enhancements to accident management capabilities envisioned at the inception of the program. We conclude that the revised formal industry position can achieve the overall objectives established for the accident management program in SECY-89-012, and obviate the need for the NRC to issue a generic letter on this matter, provided the position is strengthened by adding a description of how the IPE and IPEEE insights and results are to be used in accident management implementation, and a statement that each licensee will perform a plant-specific assessment of their capabilities to respond to accidents found to be important in their plant. We consider the processes documented in NUMARC Report 92-01 or NUREG/CR-6009 to provide an acceptable basis for performing this systematic evaluation. However, alternative approaches may also result in effective implementation. Therefore, our acceptance of the industry position is not contingent upon the endorsement of any particular methodology. Whatever approach is used, licensees should maintain appropriate internal documentation of their evaluation.

Following receipt of the formal industry position, we would find it helpful if each utility could provide a schedule and estimated date of completion for severe accident management implementation. This information will allow us to plan for staff follow-up activities. We have not yet reached a decision regarding the exact nature of these confirmatory activities. However, we expect to perform audits of the plant-specific implementation, with an emphasis on the interface of SAMG with the Emergency Operating Procedures and Emergency Plan and the incorporation of severe accident material into personnel training. Our current view is that the staff's follow-up activities can most effectively be performed, at least in part, through our observation of the use of the SAMG during the biennial emergency exercises.

We are hopeful that industry will support the modified formal position, and are looking forward to working with your staff and industry on completion of the remaining activities regarding severe accident management.

Sincerely,

BASheron for

Ashok C. Thadani, Associate Director
for Inspection and Technical Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

cc: E. Fuller (EPRI)

NUCLEAR ENERGY INSTITUTE

June 24, 1994

Dr. Ashok Thadani, Associate Director
Inspection and Technical Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr. Thadani:

In response to your letter of June 20, 1994, we would like to formally invite senior NRC staff management to provide an NRC perspective on severe accident management (SAM) at the associated industry implementation workshop. It is to be held in Alexandria, Virginia on September 28-30, 1994. The NRC presentation would occur in the early morning with an industry/NRC panel session in the afternoon. As we have discussed before, we also encourage attendance of NRC regional personnel as well.

We view your letter as an endorsement of the industry approach and plan to proceed with completion of the generic industry documents and seeking approval of a formal industry position. However, we believe it is important that you understand how we plan to address the three suggestions offered in your letter.

First, per the request in your letter, once we have approval of the formal industry position, we will request that each utility provide an estimated date of completion for SAMG implementation to the NRC staff. We envision this to be the only licensee submittal that would occur on this topic.

Second, regarding strengthening the formal industry position description of how plant-specific IPE and IPEEE insights are to be used in SAM implementation, we agree with two points made: (1) adding IPEEE insights; and (2) noting that the assessment should be focused on "capabilities to respond to accidents found to be important in their plant." We plan to adjust Section 5.2, first bullet of the formal industry position to read: "Evaluate industry-developed bases and Owners Group severe accident management guidance (SAMG) along with the plant IPE, ***IPEEE*** and current capabilities, to develop SAMG ***for accidents found to be important in your plant.***"

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While noting NUMARC 92-01 or NUREG/CR-6009 as acceptable bases for performing the assessment, your letter also stated that alternative approaches may result in effective implementation. We would like to reiterate your recognition of alternative approaches, because we are confident that utilities have used a systematic process in the review and disposition of IPE insights, including documentation and assignment of some insights for further consideration in conjunction with the generic SAMG.

Finally, while we understand the need for NRC staff to conduct a few site visits to check on the end-result following implementation of the formal position, use of the term "audits" is an inappropriate characterization. Nonetheless, we believe ample time exists prior to the industry implementation target date, such that we and NRC staff can better define the specific process to be followed to confirm adequate implementation of the SAMG.

We look forward to NRC staff participation at the workshop in September.

Sincerely,

William H. Rasin
Vice President,
Technical/Regulatory Division

DJM/rs

c: M. Virgilio, NRC
F. Congel, NRC
B. Boger, NRC