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Industry Approach to Seismic Severe Accident Policy Implementation

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REPORT SUMMARY

Industry Approach to Seismic Severe Accident Policy Implementation

This report provides utilities with industry recommended guidelines for cost-effective seismic evaluation of nuclear power plants in response to NRC Generic Letter 88-20. Guidance is provided on application of seismic probabilistic risk assessment and seismic margin methods for full-, focused-, and reduced-scope evaluations. It provides strategies for coordinating these evaluations with similar reviews needed for resolution of Unresolved Safety Issue (USI) A-46.

INTEREST CATEGORIES

Nuclear seismic risk, design, and qualification
Nuclear component reliability
Risk analysis, management, and assessment

KEYWORDS

Earthquakes
Seismic effects
Seismic qualification
Electrical equipment
Mechanical equipment
Equipment anchorage

BACKGROUND NRC issued a severe accident policy for existing plants in August 1985. It describes the basis for NRC to use in resolving issues related to the potential for severe reactor accidents due to internal and external initiators. Individual Plant Examination for External Events (IPEEE), which includes earthquakes, is addressed in Supplement 4 of Generic Letter 88-20 and NUREG 1407. In these documents, NRC accepts two methods for evaluating plants for seismic events, seismic probabilistic risk assessment, and seismic margin assessment (SMA). NRC further classifies each U.S. nuclear plant into one of three categories for application of the SMA method: full-, focused-, or reduced-scope, indicating the level of review effort to be performed.

OBJECTIVE To provide the nuclear power industry with guidance for performing IPEEEs.

APPROACH The authors developed the basis for the guidelines through interactions with both NRC and utility industry representatives during the formulation of the Generic Letter. The interaction was coordinated by the Nuclear Utility Management and Resources Council. The iterative approach employed by the team allowed for resolution of issues before the issuance of the Generic Letter. The team developed cost-effective evaluation approaches based on seismic hazard relative to the plant design basis using results from EPRI report NP-6395-D. They also used the lessons learned from the three trial plant evaluations already performed by industry (Catawba, EPRI report NP-6359, and E. I. Hatch, report NP-7217) and NRC (Maine Yankee, NUREG/CR-4334). The guidelines direct emphasis and resources to those areas that have been shown to be the most cost-effective.

RESULTS The key features of the report are guidance on the use of the two methods acceptable to NRC for seismic IPEEE, including criteria for choice of methods, effective application to IPEEE, strategies for coordinating and combining the reviews with those to be performed to address USI A-46, and closure procedures and criteria for addressing potential vulnerabilities. The application of the margins method includes details on the level of effort appropriate for plants in each of the three NRC categories and provides procedures for limited evaluations of containment systems and components, an area not included in the original scope of seismic margins review in EPRI report NP-6041 or NUREG/CR-4334.

EPRI PERSPECTIVE The development of the guidelines in this report had a major influence on the requirements of the NRC Generic Letter in two areas. The first is the grouping of plants into the full-, focused-, and reduced-scope categories. The categorization is based not on absolute seismic hazard, as was originally envisioned, but on hazard relative to the design basis (safe shutdown earthquake) which allows accounting for the inherent margin in the plant's design basis in assigning a review category. This resulted in most plants being assigned to the focused-scope rather than the full-scope category. The second area was relay evaluation. Both of the EPRI trial plant evaluations (Catawba and E. I. Hatch) clearly showed that extensive relay review is a major cost element in the IPEEE and yields little or no increase in plant seismic safety. A simple check for certain predefined, seismically sensitive relays was determined to be much more cost-effective. The Generic Letter provides that focused-scope plants may limit the relay review to just such an approach within the constraints imposed by USI A-46. The interaction with NRC on these issues allowed common understanding with the industry. As a result, these methods are, with some minor exceptions, compatible with the Generic Letter.

PROJECT

RP2722-23

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ABSTRACT

The Nuclear Regulatory Commission (NRC) issued a severe reactor accident policy for existing plants on August 8, 1985 which describes the formal basis by which the NRC intends to resolve issues related to potential severe reactor accidents. Examination of plant-specific vulnerabilities due to seismic and other externally initiated events was considered on a later schedule and is addressed in Supplement 4 of the NRC Generic Letter No. 88-20 and a NRC guidance document, NUREG-1407, issued in June 1991. This report was prepared to provide a coherent and effective approach for seismic severe accident review which meets the intent of Generic Letter No. 88-20, Supplement 4.

The recommendations in this report provide guidance on plant review types and review implementations which is consistent with the "limited-scope" intent of systematic evaluations as described in the NRC's Severe Accident Policy Statement. In addition, to assist in implementing cost-effective modifications that reduce vulnerabilities, this report also presents specific guidelines for identification and treatment of vulnerabilities that may be used as a basis for defining closure of earthquake-related severe-accident issues.

In line with the severe-accident policy statement, the approach proposed in this report for treatment of seismic issues focuses on the objectives of completing high-quality systematic plant evaluations, effectively identifying plant-specific vulnerabilities, and implementing improvements that are cost-effective in mitigating the risk impact of the vulnerabilities. The intent is to achieve an optimum seismic-IPE program, where severe-accident policy concerns are completely satisfied, yet industry-wide effort is not wasted on identifying potential modifications that are not cost-beneficial. The elements of the proposed approach are:

- Development of guidance on the type of systematic seismic evaluation to perform.
- Delineating effective review procedures for deterministic systematic evaluations.
- Delineating a scope of seismic review for probabilistic systematic evaluations.

- Development of procedures for effective integration of the deterministic seismic IPE and other unresolved seismic issues.

- Development of procedures and closure criteria to delineate potential vulnerabilities and resolve their treatment within cost-benefit guidelines.

This report provides procedural instructions and guidance to support resolution of earthquake-related severe accident issues. More detailed background and technical justifications for the methods are documented elsewhere, and are referenced throughout this report as appropriate.

ACKNOWLEDGEMENTS

The preparation of this report has benefitted from extensive reviews and interactions with members of the Nuclear Management and Resources Council, Seismic Issues Working Group (SIWG). We want in particular to recognize the value of extensive discussions with Dr. Orhan Gurbuz.

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Section 1

INTRODUCTION

BACKGROUND ON SEVERE-ACCIDENT ISSUES FOR SEISMIC EVENTS

On August 8, 1985, the Nuclear Regulatory Commission (NRC) issued the document *Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants*, published in 50 FR 32138 (1). That statement describes the formal policy to be followed by the NRC to resolve issues related to potential severe reactor accidents; key highlights of the Commission's statement are noted as follows:

- Based on currently available information, the Commission concludes that existing nuclear power plants pose no undue risk to public health and safety;
- Based on NRC and industry experience with plant-specific PRAs (Probabilistic Risk Assessments), however, systematic plant examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents for which safety improvements may be justified;
- Each existing plant should, therefore, perform a systematic examination to identify any plant-specific vulnerabilities, and report the results to the Commission.

In response to a request by the Commissioners, the NRC Staff developed an implementation program for integrated closure of severe accident issues. This integration plan (2), consists of the following six major elements or sub-programs:

1. Individual Plant Examination (IPE) Program. A systematic examination of existing plants for severe accident vulnerabilities.
2. Containment Performance Improvements (CPI) Program. Development of generic containment performance improvements with respect to severe accidents to be implemented, if necessary, for the major containment types.
3. Improved Plant Operations Program. Development of improved NRC and utility programs for plant operations.
4. Severe Accident Research Program. Investigation of a variety of topics related to severe accident phenomena and progression.

5. External Events Program. Research to identify external events requiring severe accident examination, and development of procedures to conduct Individual Plant Examinations for External Events (IPEEEs).

6. Accident Management Program. Utility development and implementation of plant-specific, severe accident management plans.

Internal Events

NRC and industry programs addressing the above elements of the severe accident policy (SAP) integration plan are well underway. To commence with execution of the IPE program, on November 23, 1988 the NRC Staff issued a Generic Letter No. 88-20 (3) to licensees of existing plants, requesting them to perform an IPE for severe accident vulnerabilities that may be uncovered due to internally initiated events, and to report the results to the Commission.

The specific objectives of the IPE are, for each utility in charge of operating an existing plant, to (2):

- "Develop an overall appreciation of severe accident behavior."
- "Understand the most likely severe accident sequences that could occur at its plant."
- "Gain a more quantitative understanding of the overall probability of core damage and fission product releases."
- "Reduce the overall probability of core damage and fission product releases, if necessary, by appropriate modifications to procedures and hardware that would help prevent or mitigate severe accidents. (It is expected that achievement of these goals will ensure that the severe core damage and large radioactive release probabilities for U.S. nuclear power plants are generally consistent with the Commission's safety goal policy)."

A Level 1 PRA (4) (including containment-performance considerations) has been identified as an appropriate and recommended procedure by which internal-event IPEs may be conducted. A number of utilities are thus conducting or planning to conduct a Level 1 PRA for their plant(s).

Seismic Events

As noted in Generic Letter No. 88-20, examination of plant-specific vulnerabilities due to externally initiated events (e.g., earthquakes, internal fires, external floods and tornadoes) would be expected, but could proceed

separately and on a later schedule. This time lag was introduced to allow the NRC and industry to develop procedures to identify those external events requiring examination; to develop simplified, systematic examination procedures; and to integrate ongoing NRC programs [for example, unresolved safety issues (USI) A-45, A-46 and the Seismic Margins program] dealing with external events with the IPE program, to ensure efficient, non-redundant allocation of industry efforts in these programs. Utilities were encouraged to retain documentation and plant-specific data (e.g., as derived from plant walkdowns) from the internal-events IPE to facilitate the later conduct of their IPEEEs.

On November 5, 1989, the NRC Staff issued draft Supplement 4 to Generic Letter No. 88-20 concerning IPEEEs (5), following in March 1990 with a draft guidance for conducting the IPEEE (6). Earthquakes were identified as a major class of events requiring consideration. Another Draft Supplement 4 of the generic letter (7) and a revised draft of the guidance document (8) were issued in July 1990. The final NRC Generic Letter and NUREG-1407 were issued in June 1991 (9 and 10).

As outlined in the NRC documents, the specific objectives of the IPEEE are, for each utility in charge of operating an existing plant, to (2):

- "Develop an appreciation of severe accident behavior."
- "Understand the most likely severe accident sequences that could occur at the licensee's plant."
- "Gain a qualitative understanding of core damage and fission product releases."
- "If necessary, to reduce the potential of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents."

It is significant that the objectives of the IPEEE de-emphasize quantitative assessment of risks and use of safety goals. This de-emphasis clearly encourages the use of examination procedures other than PRA for conducting IPEEEs. For instance, the Seismic Margin Assessment (SMA) procedures and a reduced-scope implementation (Section 3) of SMA-type screening procedures have been identified by the NRC Staff as being appropriate for conducting the Individual Plant Examination for Seismic Events (seismic IPE).

It is generally recognized that the SMA-type walkdown procedures (11) result in thorough and efficient identification of "outliers" (or "weak-links") when

directed by a well-qualified Seismic Review Team (SRT). In fact, most knowledgeable engineers believe that such a well-focused, well-directed, thorough plant walkdown is the single-most important aspect of plant examination for identifying severe-accident vulnerabilities. For these reasons, application of PRA methods as an alternative to SMA-type procedures in conducting the Seismic IPE have been endorsed by the NRC Staff (9, 10) only if certain enhancements are undertaken; among these enhancements is the requirement that SMA-type walkdown procedures be implemented in the PRA walkdown.

Both the severe accident policy statement and the generic letter encouraged the nuclear utilities, through NUMARC (Nuclear Management And Resources Council), to propose a methodology for the IPEEE that meets the intent of severe accident policy (i.e., to efficiently find and correct, as justified, plant-specific vulnerabilities to external hazards). The development of such methodology for earthquake-related issues, as presented in this report, has relied on the understanding and appropriate use of both seismic hazard results and information on the seismic capacity of nuclear power plant structures and equipment.

BACKGROUND ON SEISMIC HAZARD AND REVIEW METHOD SELECTION

During the past 12 years extensive effort has been devoted to developing probabilistic seismic hazard procedures as a tool to assess the low-probabilities of exceeding seismic design bases for nuclear power plants. This work originally focused on early vintage plants which had design bases derived deterministically prior to implementation of the current siting guidelines contained in 10 CFR 100, Appendix A (12). More recently, the scope of this effort has been expanded to address issues of large earthquakes as well, in particular the so-called Charleston earthquake issue.

The Charleston Issue, briefly stated, is the hypothesis that large earthquakes may occur in the eastern United States (EUS) at locations where supporting tectonic conditions exist, even though such events have not been observed historically. The possibility that earthquakes of magnitude similar to that of the 1886 Charleston, South Carolina event may occur in regions throughout the EUS was formally raised in a U.S. Geological Survey (USGS) letter (13) which recommended to the NRC that "probabilistic evaluations of seismic hazard should be made for individual sites in the eastern seaboard to establish the seismic engineering parameters for critical facilities."

Responding to NRC concerns surrounding the Charleston Issue and its impact (if

any) on seismic safety, the nuclear utilities, working through the Electric Power Research Institute (EPRI), developed a methodology for seismic hazard assessment that specifically considered the Charleston Issue (see 14 and 15 for further background). The Seismicity Owners Group (SOG), an assembly of 42 nuclear utilities in the central and eastern United States was formed to finance, oversee, and advise the development and use of this methodology.

The EPRI-SOG methodology, consists of systematic procedures to determine the probabilistic seismic hazard at any site. The methodology accepts multiple input interpretations by earth scientists, and uncertainties resulting from these alternative interpretations are quantified by use of logic trees and propagated through the hazard results (16). Seismic sources were developed by six Earth Science Teams specifically to model the possible locations at which severe earthquakes might occur in the EUS, and to estimate the probabilities associated with those occurrences. In development of these interpretations, the six Earth Science Teams considered all proposed hypotheses on earthquake causes and characteristics in the EUS, and weighted those based on available data and evidence.

Using this state-of-the-art, EPRI-SOG approach, seismic hazard results (i.e., estimates of the probabilities of exceedance of ground-motion amplitudes) have been obtained for 58 nuclear power plant sites in the EUS (15). Uniform hazard (i.e., equal exceedance probability) spectra, obtained for each of the 58 sites, provide a complete description of the site-specific ground motion threat, including the effect of local soil conditions. These uniform hazard spectra results specifically consider the possibility of severe earthquakes and their influence on site ground motion characteristics; the EPRI-SOG uniform hazard spectra are, therefore, directly applicable to the present development of procedures for resolution of seismic severe-accident issues.

A separate hazard methodology has also been developed under the sponsorship of the NRC by Lawrence Livermore National Laboratories (LLNL). The LLNL-NRC methodology considers multiple expert opinion and accounts for the Charleston Earthquake Issue; it has been applied to obtain hazard estimates for 69 EUS nuclear power plant sites (17). Development of the LLNL-NRC methodology had proceeded on a somewhat earlier schedule than the EPRI-SOG program, but both programs produced hazard results for EUS plants at about the same time.

The NRC has expressed the position that both EPRI-SOG and LLNL-NRC hazard results

should be considered in decision making on seismic issues (18); in particular, for developing seismic IPE guidance, the NRC has required (informally) that both hazard results be used in grouping (binning) plants for selection of seismic IPE review levels. This report develops a seismic IPE approach that meets this NRC requirement. The primary objective has been to search for consistency among the two hazard results on a relative, plant-to-plant basis for guiding the process of grouping plants for similar review.

BACKGROUND ON SEISMIC CAPACITY OF NUCLEAR POWER PLANT STRUCTURES AND EQUIPMENT

In the past 12 years there have been at least three major programs that have addressed the issue of nuclear power plant seismic margin. The first program was Seismic Probabilistic Risk Assessment (SPRA) which has been performed for over 20 plants. Considerable resources have been expended on these analyses which have demonstrated that nuclear power plants are generally rugged for earthquakes. A few weak components have been identified; however, very few modifications have been required for seismic upgrading of nuclear power plants (19). The modifications that have been made were primarily the result of observations and findings made during the plant walkdowns.

A second program is currently addressing capacity of equipment in older nuclear power plants. In response to NRC unresolved safety issue (USI) A-46, the nuclear industry formed the Seismic Qualification Utility Group (SQUG) to address the seismic capability of equipment in these older plants. The purpose of the SQUG program is to demonstrate that older plants have adequate seismic ruggedness for safety equipment not seismically designed to current criteria (i.e., post-1973) to withstand the plant design Safe Shutdown Earthquake (SSE). The Seismic Qualification Utility Group has gathered an extensive data base on earthquake experience in fossil fuel power plants and heavy industrial facilities. This data base has been reviewed by both the NRC and a five-member Senior Seismic Review and Advisory Panel (SSRAP), who were jointly selected by the NRC and SQUG (20). With certain caveats and exclusions SSRAP has concluded that 20 classes of equipment in nuclear power plants are at least as rugged as similar equipment in the data base plants. Currently, detailed seismic walkdowns of the older plants are planned to address the adequacy of seismic anchorage, confirm compliance of the equipment with the SSRAP exclusions and caveats, and to look for certain seismic spatial-systems-interaction concerns (21).

In the third program, a Seismic Margin Assessment (SMA) was developed to address plant capability to reach safe shutdown for seismic motions beyond the SSE design

level (22). In response to this need a methodology was developed to determine whether high confidence of a low probability of failure (HCLPF) exists for a specified ground motion input. If a plant HCLPF does not exist for this level, the methodology provides a procedure for determining at what level a HCLPF can be stated. It is believed that SMA is more cost effective than SPRA, easier to use by the practitioners in the nuclear industry, and results in a better understanding of the plant's seismic performance.

While SPRA provides risk estimates, the results are highly uncertain. Seismic Margin Assessment does not provide risk values, but the HCLPF capacities are more certain since they are in the range of experience of most seismic engineers. The NRC methodology (22) and the procedures developed by EPRI (11) both provide alternate approaches to SMA for assessing seismic margin of nuclear power plants and to identify outliers, if any.

Based on the findings from these three programs there is a broad consensus that most safety-related equipment and structures necessary for shutdown and for containment of radioactive materials generally are inherently rugged. This consensus is reflected in the following documents which have broad authorship:

- American Society of Civil Engineers, Uncertainty and Conservatism in the Seismic Analysis and Design of Nuclear Facilities, 1986 (23).
- Expert Panel on the Quantification of Seismic Margin, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," August 1985 (22).
- Senior Seismic Review and Advisory Panel (SSRAP), "Use of Seismic Experience and Test Data to Show Ruggedness of Equipment in Nuclear Power Plants," February 28, 1991 (20).

The consensus among experienced practitioners is that equipment and structures at nuclear power plants have substantial seismic ruggedness. However, certain weaker elements have occasionally been found. Therefore, the principal seismic concerns are not with the vast majority of safety-related equipment and structures in operating plants, but with the potential for a limited number of seismically weaker elements (e.g., inadequate equipment anchorage) to affect plant safety. It is also believed that the plant walkdown phase of a seismic review is the most important step in finding these problems and any other potential vulnerabilities to earthquakes.

OUTLINE OF PROPOSED APPROACH

NRC's Severe Accident Policy requires that systematic evaluations be performed with the purpose of finding and correcting (within the guidelines of NRC Backfit Policy) severe-accident vulnerabilities. A fundamental objective of the Severe-Accident Policy is to verify the widely held belief that plants pose no "undue risk" and that "all reasonable steps are taken to reduce the chances of occurrence of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident should one occur." The Individual Plant Examination (IPE) program was developed by the NRC Staff to address particular facets of severe accident issues.

In its Severe Accident Policy Statement the NRC clarifies the level of effort the IPEs should involve: "licensees of each operating reactor will be expected to perform a limited-scope, accident safety analysis designed to discover instances (i.e., outliers) of particular vulnerability to core melt or to unusually poor containment performance, given core-melt accidents." Hence, a limited-scope plant investigation that makes effective use of insights gained through past detailed investigations, aimed at effectively identifying cost-effective mitigation of plant-specific vulnerabilities, is the course intended by the NRC Commissioners for implementation of IPEs in severe accident policy resolution.

The recommendations in this report provide guidance on plant review types and review implementation which is consistent with the "limited-scope" intent of systematic evaluations as described in the NRC's Severe Accident Policy Statement. In addition, to assist in implementing cost-effective modifications that reduce vulnerabilities, this report also presents specific guidelines for identification and treatment of vulnerabilities, that may be used as a basis for defining closure of earthquake-related severe-accident issues.

In line with the severe-accident policy statement, the approach proposed in this report for treatment of seismic issues focuses on the objectives of completing high-quality systematic plant evaluations, effectively identifying plant-specific vulnerabilities, and implementing improvements that are cost-effective in mitigating the risk impact of the vulnerabilities. The intent is to achieve an optimum seismic-IPE program, where severe-accident policy concerns are completely satisfied, yet little effort industry-wide is wasted on identifying potential modifications that are not cost-beneficial. The elements of the proposed approach, and the overall organization of this report, are outlined below:

- Development of preliminary guidance on the type of systematic seismic evaluation to perform (Section 2).
- Delineating effective review procedures for deterministic systematic evaluations (Section 3).
- Delineating a scope of seismic review for probabilistic systematic evaluations (Section 4).
- Development of procedures for effective integration of the deterministic seismic IPE and other unresolved seismic issues (Section 5).
- Development of procedures and closure criteria to delineate potential vulnerabilities and resolve their treatment within cost-benefit guidelines (Section 6).

This report provides procedural instructions and guidance to support resolution of earthquake-related severe accident issues More-detailed background and technical justifications for the methods are documented elsewhere, and are referenced throughout this report as appropriate. In particular, Reference (24) describes bases for using seismic hazard results in rational decisionmaking related to the treatment of seismic issues, and Reference (25) describes bases for the treatment of high-frequency seismic ground-motion effects in nuclear power plants.

Section 2

SEISMIC REVIEW-METHOD DETERMINATION

BACKGROUND AND OUTLINE OF APPROACH

This section provides guidance to establish an efficient and effective basis for selecting the type of plant systematic evaluation to perform for the seismic IPE and for determining the ground-motion level at which the seismic IPE should be conducted (see Reference (26) for background). Plants are differentiated and grouped based on the combined use of seismic hazard results and seismic design bases, in the manner discussed in Appendix A. Delineation of binning categories is ambiguous in light of imperfect information on plant hazards and on the unclear relationship between plant margin and seismic design. Before deciding on a systematic evaluation approach to implement therefore, a utility is advised to factor its own knowledge of plant design, maintenance, and backfit history into the process of deciding how to proceed with the seismic IPE.

Based on judgment consistent with results of past studies, the binning proposed in this section results in a very efficient (industry-wide) seismic-IPE program that will identify cost-effective modifications which best enhance plant seismic safety. The desirability of conducting the IPE program in an efficient manner has been stressed by the NRC in the severe accident policy statement (1). Key points from the severe-accident policy statement in this régard pertain to the intent and desired implications of the seismic IPE, i.e., that: (1) systematic evaluations should be of limited scope; (2) they should efficiently reveal low-cost modifications of the type found in past PRAs; (3) they should serve as a basis for verifying conclusions developed from more intensive studies (e.g., NUREG 1150 27); and (4) only those modifications found to be justified within the cost-benefit criteria of the NRC backfit policy should be implemented.

In line with these points, the recommended approach emphasizes the use of well-focused, systematic approaches for the efficient identification of outliers. Procedures and criteria discussed in Section 6 provide guidance on closure of the seismic IPE, and can be used by utilities regardless of which systematic evaluation procedure (reduced-scope assessment, focused-scope SMA, full-scope SMA,

or PRA) they decide to implement. The format of these resolution criteria help to ensure that the major set of cost-effective and safety-effective modifications will be found under whatever format a utility chooses to undertake for its seismic IPE.

As developed in Section 3 and discussed in Appendix A, four systematic evaluation procedures are available for conducting the seismic IPE: (1) the reduced-scope assessment, (2) the focused-scope SMA, (3) the full-scope SMA, and (4) the seismic PRA (SPRA). The first three of these methods are deterministic implementations, and the last is probabilistic. Roughly speaking, in the context of the seismic IPE, the reduced-scope SMA is considered appropriate where the seismic hazard is low; the full-scope SMA is considered appropriate where the seismic hazard is comparatively high relative to the design basis; the SPRA, an alternative to the full-scope SMA, is considered appropriate in special situations where risk results may be anticipated to facilitate decisionmaking; and the focused-scope SMA is considered appropriate for the remaining bulk of plants that have comparatively moderate seismic hazard relative to design basis.

PLANT SEISMIC IPE GROUPS

Figure 2-1 shows results of ordered design-basis hazards (i.e., probabilities of exceeding plant design-motion levels) for eastern U.S. nuclear power plants, based on EPRI median hazard results. Figure 2-2 presents similar data based on LLNL median hazards. Using the results in these figures, one can differentiate plants based on the level of composite design-basis hazard. For instance, in each of the two plots, one can clearly distinguish the following two groups: (1) a small group of comparatively high design-basis hazard plants, and (2) a large group (comprising the bulk of the population) of plants with comparatively moderate-to-low design-basis hazard. The six plants with the highest design-basis hazard from the EPRI median results are the same six plants with the highest design-basis hazard from the LLNL median results. This consistency among EPRI and LLNL results lends confidence to immediately differentiating two groups of plants: a small group of plants with comparatively high design-basis hazard, and the remaining population of plants.

It is appropriate, however, to differentiate a third group of plants (as mentioned above) on the basis of having negligibly low seismic hazard, regardless of design basis. For instance, certain areas in the deep south and in the northern-midwest portions of the eastern U.S. (EUS) are known to be substantially quiet tectonically; because the seismic threat is so low, plants in these regions would

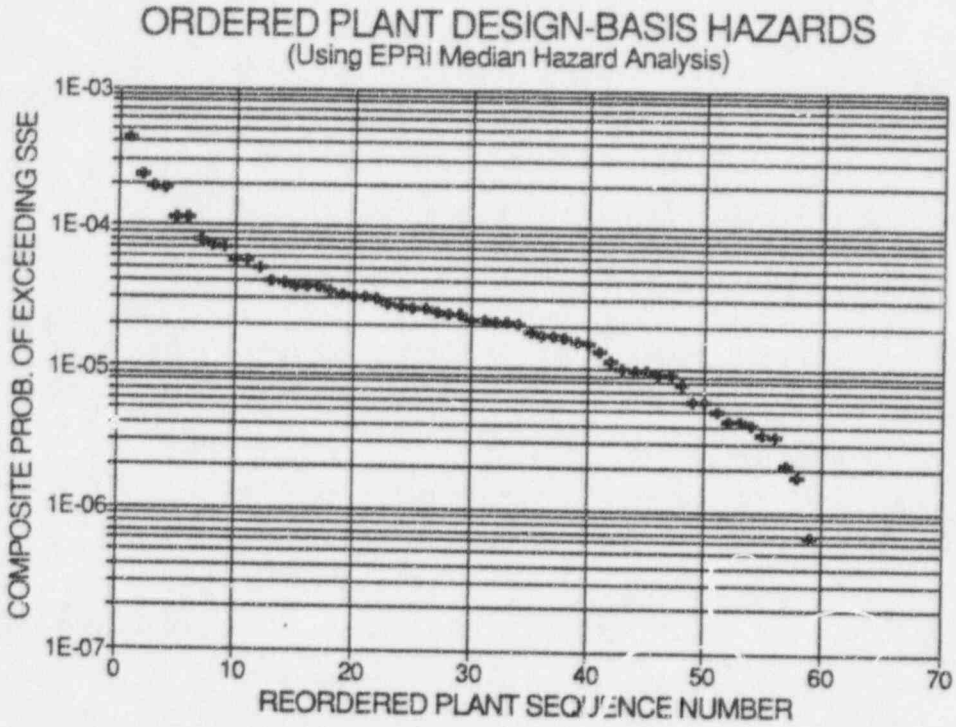


Figure 2-1. Ordered, composite design-basis hazards for EUS nuclear power plants, based on EPRI median results.

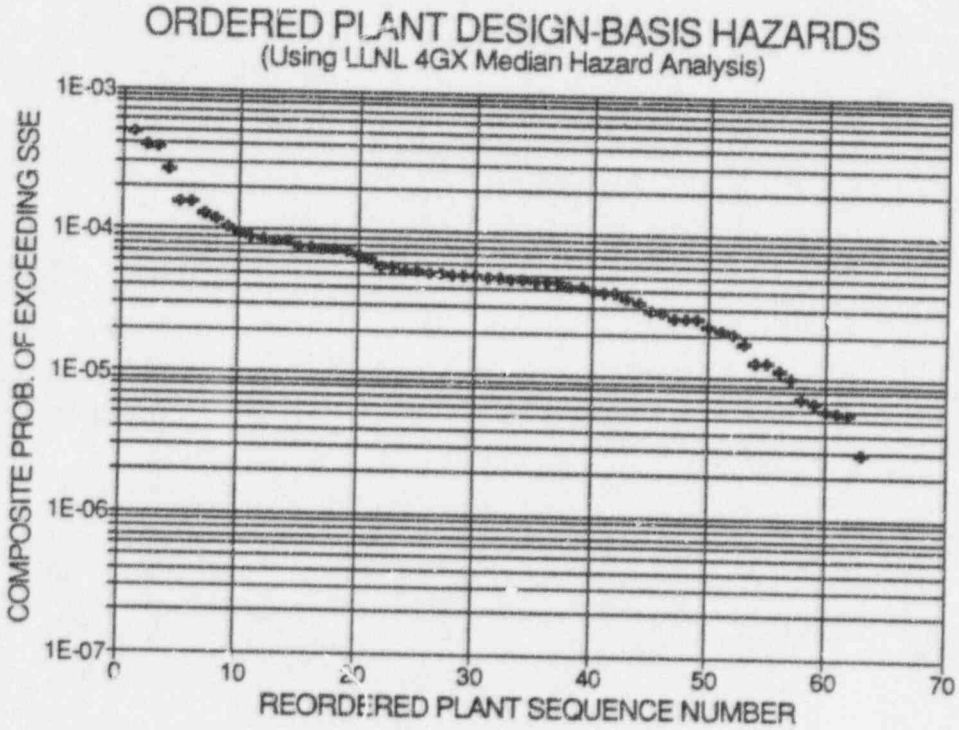


Figure 2-2. Ordered, composite design-basis hazards for EUS nuclear power plants, based on LLNL median results.

not warrant the same rigor in severe-accident evaluation as would plants in other areas of the U.S. The NRC has performed a plant ranking analysis (8), considering LLNL and EPRI hazard results, that identified such low-hazard plants. The analysis is based on a clustering methodology (developed by LLNL) applied to site hazard (as opposed to design-basis hazard) results. In their analysis, the NRC identified ten plants as belonging to a low-hazard group, where a reduced-scope severe-accident evaluation is appropriate. (Because these ten plants belong to a distinct group, they are not included in the plots of Figures 2-1 and 2-2.)

Based on Figures 2-1 and 2-2, and on NRC's identification of low-hazard plants, therefore, the following groups of plants are delineated:

- A small number of plants for which the design-hazard (i.e., probability of exceeding the SSE) is high compared to other plants (full-scope SMA category).
- Ten plants for which the site hazard is negligibly low (reduced-scope assessment category).
- The remaining (bulk of) the plant population, where the design-hazard is comparatively moderate (focused-scope SMA category).

The use of design-basis hazard and these three groupings is intended to provide guidance, not fixed rules or criteria, to plant licensees on the type of seismic IPE review they may wish to perform. For its own internal purposes, a licensee may wish to implement a more detailed review than would be recommended here. Morever, a licensee may use its discretion and judgment (based on its extensive experience and understanding of its plant) in deciding that a less detailed study than that identified by the procedure described here may be more suitable for implementation at its plant. In some instances, it may be useful for a licensee to undertake additional study to assist in its decision. In addition to the guidelines discussed here, the NRC also identifies, in its final revision to Supplement 4 of Generic Letter 88-20, the specific plants which it deems appropriate as belonging to each of these plant groups. Licensees are encouraged to also review that final document to assist in selecting seismic-IPE review types appropriate for their plants.

For the ten plants that have negligibly low seismic hazard, the need for any seismic severe-accident evaluation at all is questionable; for such plants, a reduced-scope set of procedures that emphasize a thorough plant walkdown has been developed (28). For the ten EUS plants identified above as falling in a

low-hazard group, therefore, a reduced-scope assessment is recommended for implementation of seismic IPEs.

A focused-scope SMA will find all design-independent potential vulnerabilities and also those design-dependent potential vulnerabilities that may compromise plant resistance to severe seismic accidents, the same potential design-dependent and design-independent vulnerabilities as would be found in a full-scope SMA or SPRA. The focused-scope SMA, therefore, is considered an appropriate procedure for severe-accident review of any EUS plant. For some plants where the seismic hazard or the probability of exceeding the design basis (i.e., SSE spectrum) is markedly larger than the major remaining plant population, it may be of benefit to licensees and to the NRC to have very realistic capacity calculations for certain components to assist in decisionmaking and in seismic-IPE resolution. In addition, in cases of higher seismic hazard, it may be prudent to perform (as discussed in greater detail in Section 3) a relay chatter evaluation, with the specific purpose of identifying and replacing known vulnerable relays and contacts. For these higher hazard plants, implementing the set of full-scope (vis-à-vis focused-scope) SMA procedures is considered potentially useful, although not clearly warranted.

For the set of EUS plants having comparatively high design-basis hazard, therefore, a full-scope SMA is recommended for implementation of seismic IPEs. (An SPRA would, however, generally be recommended for those plants where an SPRA has been performed previously). For the remainder of EUS plants that do not belong to either the reduced-scope or full-scope SMA class, a focused-scope SMA is recommended.

In all cases for EUS plants, SMA screening against the 0.3g PGA (0.8g spectral acceleration) screening tables is considered sufficient to identify severe-accident outliers. For some Western U.S. (WUS) plants, the severity of the seismic environment might lead one to conclude that a full-scope SMA at 0.5g PGA (1.2g spectral acceleration) or an SPRA may be appropriate or required for severe-accident review. Without seismic hazard results for these plants, a seismic IPE recommendation cannot be made. However, given the generally higher seismic design levels for WUS plants and given the geographic diversity of the WUS seismic threat, it is reasonable to suggest, based on the approaches in this document, that some WUS plants may be differentiated into a classification where less than full SPRA or full-scope SMA procedures are justifiably appropriate.

This approach for review-method selection of plant seismic-IPEs focuses on the clear merit of the following objectives: (1) emphasizing the value of performing a thorough plant walkdown, an accomplishment that will identify potential safety enhancements that past experience indicates are important or cost-beneficial to fix; (2) de-emphasizing the tasks of performing calculations and producing a paper product, items that clearly will not enhance plant safety; (3) de-emphasizing the task of performing evaluations (e.g., of relays) that past experience says will have a very low probability of identifying or achieving any significant, cost-effective safety enhancement; and (4) helping to insure that the work of available qualified technical professionals will be focused in such a manner that encourages a high-quality end product and the development of optimal safety-enhancement solutions. Each licensee is encouraged to use the guidelines presented in this section, factored with plant-specific information, to decide on the review method most appropriate for seismic-IPE implementation at its plant.

This process of determining seismic review method, by itself, does not assess earthquake review levels. The review method, consistent with conventional SMA procedures, may specify a standard (default) review-level ground motion. For full-scope and focused-scope SMAs, the review motion is taken as a NUREG/CR-0098 median spectrum (29) anchored to a PGA of 0.3g (9). For the reduced-scope assessment, because the seismic hazard is negligibly low, the severe-accident review earthquake is taken as the SSE spectrum (or licensing commitment) itself. Further guidelines on seismic-IPE implementation procedures and on ground-motion input are provided in the next section.

Section 3

DETERMINISTIC SEISMIC REVIEW IMPLEMENTATION

The requirements for deterministic seismic review are given in this section and are provided for the following three types of review:

- Full-Scope SMA
- Focused-Scope SMA
- Reduced-Scope Assessment

Depending on the review bin selected for a plant (i.e., see Section 2) one of the three types will be performed.

Seismic margins assessment (SMA) determines whether a plant has a high confidence of a low probability of failure (HCLPF) capacity for core damage and release from containment, for a selected review level earthquake (RLE). For these cases it is stated that the plant HCLPF equals or exceeds the RLE. When it is found that the HCLPF is less than the RLE, then the actual HCLPF for the plant is calculated.

The SMA is a walkdown-based seismic assessment (similar in approach to the A-46 seismic review). It focuses on structures and equipment in two independent safe shutdown paths and is based largely on available seismic experience, test and fragility data. In this approach, experienced engineers perform screening walkdowns to assess the capacity of identified safe shutdown piping, equipment and structures to withstand a review level earthquake which is well above the plant's licensing basis Safe Shutdown Earthquake (SSE). The SMA review level earthquakes for U.S. plants have been specified by the NRC and are given in Supplement 4 to NRC Generic Letter 88-20. For most U.S. plants, the review level is 0.3g. Four western U.S. plants in high seismic hazard areas have a review level of 0.5g; ten plants in very low seismic hazard areas in the South, Southwest and Northern Midwest will have a review level earthquake equal to their design basis SSE.

Equipment and structures which do not pass the deterministic screening criteria at the review level are identified as outliers and are subsequently evaluated in more detail to determine the earthquake level at which there is a "high confidence of a

low probability of failure" (HCLPF). The lowest of these HCLPF levels sets the HCLPF for the plant. These outliers are evaluated on a judgmental cost/benefit basis to determine the merits of various corrective actions to raise the plant's HCLPF level. Relay chatter is addressed in a somewhat simplified manner based on the A-46 approach.

Two approaches for performing an SMA are recognized by the NRC Generic Letter; the EPRI method, and the NRC method.

The EPRI SMA methodology has been used in two trial plant reviews, is documented in Reference (11), and the original version has been accepted by the NRC. The NRC method is documented in NUREG/CR-4334 and has been used in one trial plant. The HCLPF is obtained for core damage and is based on either event and fault trees (NRC method) or on components in a deterministically selected success path (EPRI method). For the latter approach, the plant HCLPF is the lowest HCLPF of all components in the success path selected for review.

In the EPRI methodology, the HCLPF is calculated using deterministic procedures that are generally familiar to the design engineer. The factors of safety are liberalized in SMA compared to plant design procedures to reflect the philosophy that a HCLPF has approximately a 95% confidence of about a 5% probability of failure at the Seismic Margin Earthquake (SME) level. A HCLPF can be determined in either a deterministic or probabilistic manner. The probabilistic approach is based on the same procedure as used in fragility analysis in a SPRA.

The procedures for performing a SMA are very similar to a A-46 review. Selection of review components, plant walkdown, screening, and evaluation of outliers are common steps to both methods. Both approaches rely heavily on engineering judgement by experienced, trained, seismic engineers and seek to eliminate unnecessary work by focusing efforts on the most significant concerns.

IPEEE plants which the NRC has designated as 0.3g RLE plants will either perform a full-scope review that follows NP-6041, Reference (11), or a focused-scope review which requires less effort. Plants selected for a reduced-scope review follow only selected portions of the SMA.

Table 3-1 summarizes the three types of reviews. The recommended requirements for the three types are given below. The discussion parallels the summary which is given in Table 3-1.

Table 3-1

SUMMARY OF RECOMMENDATIONS FOR DETERMINISTIC PLANT SEISMIC REVIEW[1]

Type of Review	Success[2] Path Elements	Containment	Relays	Soil Failure	Screening Criteria	Outliers	Input	Documentation
Full-Scope SMA	NP-6041 and Containment Systems[3]	Yes[3]	Yes[4]	Yes[6]	NP-6041[8]	HCLPF Calculations	NUREG-0098	NP-6041 (CH 8)
Focused-Scope SMA	NP-6041 and Containment Systems[3]	Yes[3]	Yes/ No[5]	Yes[7]	NP-6041[8]	HCLPF[9] Calculations	NUREG-0098	NP-6041 (CH 8)
Reduced-Scope	NP-6041	No	No	No	NP-6041[8]	FSAR (or GIP for A-46)	SSE	Concise

[1] Utilities may propose procedures different than those shown in this table
[2] Elements identified should be consistent with type of review and IPEEE enhancements to be included.
[3] Functions required for containment integrity, isolation, prevention of bypass.
[4] Search for low seismic ruggedness relays only.
[5] Review for low seismic ruggedness relays. For A-46 plants, if low seismic ruggedness relays are found, expand to IPEEE review.
[6] Review potential soil failure modes (i.e., instability, settlement, and liquefaction).
[7] Review based on existing soils analyses, soils test reports, and design and construction records only.
[8] A-46 GIP may also be used to screen components.
[9] Order outliers, perform bounding calculations for lowest fragility components only.

It is important to note for all deterministic reviews that a plant walkdown will be conducted by a Seismic Review Team (SRT). The steps in the walkdown will be the same in all cases. Experience in past seismic SPRAs and SMAs indicates that significant seismic concerns have been identified in the walkdown process. The only exception might be the refueling water storage tank (RWST) which was modified as result of the SMA conducted for the Maine Yankee Nuclear Power Plant. However, flat-bottom tanks are a special concern which must always be considered in any seismic capability assessment. It is the widely-held opinion of many seismic capability engineers that the plant walkdown is the most important step in a seismic review, which will address the same structures and equipment independent of the type of review. It is anticipated that most, if not all, seismic performance issues will be discovered during the walkdown.

IDENTIFICATION OF SUCCESS PATH ELEMENTS

For all three types of deterministic review, procedures for identifying structures and equipment to be reviewed are the same and are based on the recommendations in EPRI Report NP-6041 (11). Depending on the system functions suggested for review type as discussed below, primary and alternate success paths for achieving shutdown should be selected as outlined in EPRI Report NP-6041. The approach for identification of systems and components needed to prevent early containment failure are discussed in Appendix D of this report. Note that the steps and investigations leading to the selection of structures and components are essentially identical and independent of which type of deterministic review is performed.

Identification of low seismic ruggedness relays in the systems analysis is discussed below.

CONTAINMENT REVIEW

The EPRI Seismic Margin Assessment approach includes evaluation of systems and components that are included in the "success paths" that will assure safe shutdown of the plant and maintain it in a safe condition for 72 hours. The EPRI SMA methodology does not include consequence mitigation (i.e., "containment") systems in the scope of evaluations. Considering the "defense-in-depth" principle, the draft Generic Letter 88-20, Supplement 4 also requests evaluation of containment performance.

It is recommend that the full- and focused-scope SMA reviews be limited to

evaluation of only those functions that are necessary to insure containment isolation and to prevent containment bypass and early containment failure. For essentially all containment designs, this includes: (1) successful containment isolation; (2) maintaining containment structural integrity (including penetrations and closures); and (3) prevention of containment bypass. For pressure suppression containment designs, the equipment necessary for the pressure suppression function (e.g., the suppression pool and the vent system for BWRs or the ice buckets, ice chamber and inlet/outlet "doors" for ice condenser designs) would be included in the scope of seismic reviews. In addition, drywell sprays (for Mark I) or hydrogen control (for uninerted Mark II, Mark III or ice condenser containments) may be required to prevent early containment failure. Because there is a significant variation in containment designs, input and guidance should be obtained from the IPE internal events containment evaluation team to determine if any other systems are required to prevent early containment failure. The success path logic diagrams (SPLDs) described in EPRI Report NP-6041 (11) denote those systems which are necessary to provide a long term safe shutdown condition. The SPLDs should be extended to include containment functions necessary to prevent early containment failure assuming that severe core damage has occurred.

Evaluation of systems and equipment whose functionality is required to prevent long term containment failure is not considered necessary because previous PRAs indicate that risk to the public due to severe accident sequences involving failure of long term containment integrity is low. This implies, for example, that for large, dry containments, review of the spray systems and fan coolers should be excluded from the scope of the IPEEE. Moreover, physical examination of some of this equipment may not be practical due to access requirements for entering radiation areas.

RELAY EVALUATION

The SMA conducted for the Hatch Nuclear Plant clearly demonstrated that relay review is not cost effective (30). Relay evaluations at three other plants have shown that the only relays that were found to be not sufficiently rugged were those on the low seismic ruggedness relay list or those eliminated by operator actions. Furthermore, relay chatter has not been a significant issue at non-nuclear facilities and industrial sites that were subject to ground motions on the order of 0.3 g. On this basis, the scope of relay chatter review should be limited as follows:

Full-Scope SMA

It is recommended that plants that perform a full-scope SMA only conduct a review for low-ruggedness relays. However, plants performing an A-46 review should conduct a relay review according to the procedures in the GIP (21) for all relays within the scope of the A-46 review. Investigation of relays outside the A-46 review but within the scope of the IPEEE review should address only the issue of low-ruggedness relays.

Focused-Scope SMA

A full relay review will be performed for A-46 plants in any case. For non A-46 plants a search for low-ruggedness relays as described in the GIP will be conducted. For A-46 plants further review of relays outside the scope of A-46 is generally not required. This is justified because there will be a review for the relays included in the A-46 scope of work and extending the list to include the IPEEE scope is unlikely to be cost-effective. However, if a low seismic ruggedness relay is discovered during the A-46 program, then the scope of plant relay investigations for that plant should be expanded to check for low-ruggedness relays outside the A-46 scope but within the IPEEE and to assess the systems implications of their chatter.

Reduced-Scope SMA

No relay chatter review is needed for the plants in this category since it is unlikely that any cost-effective seismic risk reduction opportunities will be found.

The relay chatter review for IPE of seismic events should consist of the following steps:

- Identify relays that are part of the selected success paths which are considered to have low seismic ruggedness (e.g., based on Reference (11))
- Determine if the consequence of contact chatter for the identified low seismic ruggedness relays is unacceptable.
- Ascertain whether systems affected by susceptible relays can be reset in a timely manner after ground motion has ceased, and well before their function is needed.

Relays which fail to pass the ruggedness, consequence or recovery tests are considered to be outliers. These relays should be replaced, or a more detailed analysis should be conducted to determine their disposition.

SOIL FAILURE INVESTIGATION

Soil failure has not been found to be a significant issue in past SMAs and SPRAs. However, for some plants on relatively soft-soil sites, the margin for soil failures may not be as high as it is for typical nuclear power plant structures and equipment. As a matter of prudence, it is recommended in a full-scope review that potential soil failure modes (i.e., instability, settlement and liquefaction) be reviewed as required in EPRI Report NP-6041. It is anticipated that existing soil test data will be adequate. A review of plant site conditions, using state-of-the-art approaches, will quickly determine whether soil failure is a significant issue. For plants in the focused-scope SMA category, a review based on existing soils analyses, soils test reports, and design and construction records is considered adequate. A review of soil failure should not be required for plants in the reduced-scope bin.

SCREENING CRITERIA

The SRT must perform the walkdown themselves and take complete responsibility for all elements screened out. At the conclusion of the walkdown concise documentation will be prepared which records the basis for screening out each component and which is signed by all members of the SRT.

There are two basic parts to the inspection of each component. First, the structural integrity and/or functionality of a component (exclusive of the anchorage) is considered. The guidance given in the EPRI Report NP-6041, Tables 2-3 for structures and 2-4 for equipment can be used to screen components (11). For the full- and focused-scope review the first column criteria in these tables, corresponding to 0.8 g spectral acceleration (i.e., which replaces the 0.3 g peak ground acceleration limit, as discussed in Appendix B) should be used. Note that plants in the Western U.S. (i.e., with a review level earthquake of 0.5 g pga) would use the 0.8 g to 1.2 g S_a column.

For older plants which are also being reviewed for the A-46 program, as well as other plants, the owner utility can use the walkdown and screening requirements in the GIP in addition to the SMA screening tables as discussed below. Note that the Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment (GIP) (21) procedures are generally more conservative than the SMA screening tables. However, there may be some cost savings to using the GIP for plants already performing A-46 reviews (e.g., see Appendix C). If the provisions of the GIP are used, all caveats given in that report must be followed for the

components screened by that approach (e.g., the bounding ground response spectrum in the GIP must exceed the review level earthquake response spectrum).

The second part of the walkdown inspection is concerned with anchorage adequacy. The screening tables given in the EPRI Report NP-6041 are primarily for the capacities of elements. Anchorage must be considered in addition to the guidance given in the screening tables, which generally only considers structure and component integrity and functionality. It should be noted that anchorage for all cabinets which contain relays included in the selected success paths must be reviewed, even though the relays per se are not evaluated. Guidance for anchorage review is given in EPRI Report NP-6041.

Ultimately, the SRT must certify that the anchorage capacity exceeds the seismic demand based on the input as defined above. It is recommended that prior to the plant walkdown the SRT review the construction drawings and specifications, and develop generic capacities for as many of the anchorage configurations as practical. Hopefully, existing design calculations can be used as a direct means for determining anchorage capacities. If design computations are not immediately available then specific calculations will have to be performed for each group of similar components, or for individual components, if they are each significantly different. By becoming calibrated for the plant-specific anchorage details and seismic input the SRT can more efficiently screen anchorage during their walkdown. The product of the plant walkdown by the SRT is concise documentation listing components which were not screened out in the walkdown (i.e., outliers) and the basis for screening out all other components identified in the success paths.

EVALUATION OF OUTLIERS

For both full- and focused-scope SMA reviews, HCLPFs should be determined for elements not screened out during the walkdown. A principal difference between full- and focused-scope reviews is in the number of components for which HCLPFs should be calculated or estimated. For elements in the focused-scope review, it is recommended that judgement be used to rank the capacities of the outlier structures and equipment from the lowest to the highest. HCLPF capacities should be calculated as necessary (only for approximately the lowest one-third of the ranked components). The remaining components should be assigned a conservative HCLPF based on the highest calculated HCLPFs. This will reduce the analysis for plants in the focused-scope bin.

For some components where it may be difficult to calculate a HCLPF for the "as-is"

condition (e.g., batteries with no spacers or unanchored equipment), the HCLPF can be computed for the modified upgraded configuration. The input is the same as used in the screening analysis discussed below. Guidance for calculating HCLPF values is given in EPRI report NP-6041 (11).

For plants in the reduced-scope bin, which are also in the A-46 program, the outliers should be evaluated for the requirements in the GIP. For elements outside the scope of the A-46 review (e.g., structures and piping), the requirements of the plant FSAR should be used in the evaluation. For plants in the reduced-scope bin which are not being evaluated for the A-46 program, the requirements of the plant FSAR should be used in the evaluation for all elements in the success paths. All elements which do not meet the acceptance criteria should be addressed by using the normal plant procedures to resolve safety issues.

Structures and components which are screened out based on the plant walkdown and review of drawings and specifications are not considered further. By intention, the SMA philosophy dictates that the SRT has high confidence of a low probability of failure (HCLPF) for these elements. Outliers identified in this process are reviewed further to determine their HCLPF values. The term "outlier" does not imply that a plant procedure and/or physical modification is required.

It is recommended that when simple modifications will increase its margin, a component be strengthened. For potentially expensive changes, more realistic detailed calculations or component testing can be used to justify that the outliers are not truly safety-related deficiencies relative to the plant design basis. Also, alternate equipment or procedural changes should be considered.

SEISMIC INPUT

For full- and focused-scope SMA reviews it is recommended that the ground response spectra be the NUREG/CR-0098 median curve anchored to the review level peak ground acceleration for the plant (i.e., 0.3 g or 0.5 g). The guidance provided in EPRI Report NP-6041 should be used to develop in-structure response spectra for the evaluation, when necessary. The calculated spectra should be median centered. As discussed in Section 6 site-specific response spectra may be used in closure evaluations. For plants performing a focused-scope review, the use of simplified scaling procedures is encouraged in order to reduce the analysis cost. However, this may lead to development of conservative HCLPFs in these cases.

The input for the reduced-scope review should be the in-structure response spectra

developed for the SSE ground response spectrum. New floor spectra can be developed which reflect state-of-the-art, soil-structure interaction models and building response analyses. In this case, however, to be consistent with the conservatism in the design input, mean plus one standard deviation level in-structure response spectra should be developed.

REVIEW DOCUMENTATION

The documentation of the IPEEE for the full- and focused-scope SMA reviews should follow the guidance outlined in Chapter 8 of EPRI Report NP-6041. A list of HCLPF values are to be provided for elements not screened out or found to have HCLPF values less than the review level earthquake. However, in a focused-scope review HCLPFs are calculated for only about one third of the screened-in elements. Therefore, HCLPFs for the remaining elements can be conservatively assigned based on the highest calculated HCLPFs. Note that in both the full- and focused-scope reviews HCLPFs for some equipment may be estimated by conservative comparison.

At the conclusion of the review the SMA will be documented as required in EPRI report NP-6041. A list of all structures and equipment identified in the systems analysis is divided into two groups. The first group contains all elements which were either screened out during the walkdown or have calculated HCLPF values equal to or greater than the review level earthquake. The SRT will document the basis for their decisions. The second group will consist of a list of elements where the calculated HCLPF values, which are less than the review level earthquake will be provided.

The report for the reduced-scope review should be concise and should include the systems and elements identified in the success paths, the procedure for the walkdown and findings and the resolution of all outliers.

Section 4

SCOPE OF SEISMIC REVIEW USING SPRA APPROACH

Draft Generic Letter 88-20, Supplement 4, identifies seismic PRA (SPRA) methods as being acceptable for use in conducting the IPE of seismic events. Seismic probabilistic risk assessment produces a mean core damage frequency (CDF) and ranking of accident sequences, systems, and structures/equipment that are potentially significant contributors to core damage risk due to seismic events. A SPRA should be a Level 1 PRA with a partial Level II analysis to address containment performance. Seismic probabilistic risk assessments may be performed using the procedures described in NUREG/CR-2300 (31), NUREG/CR-2815 (32) or NUREG/CR-4840 (33).

In performing a SPRA for IPEEE, a mean seismic hazard curve, event and fault trees, and structure and equipment mean fragility curves are combined. Hazard curves, based on the studies performed at EPRI and Lawrence Livermore National Laboratory (LLNL), have been developed for most Eastern U. S. nuclear power plant sites. It is the industry's position that the EPRI curves are more realistic and should be the ones used. The NRC has consistently indicated that site-specific seismic hazard curves developed by both LLNL for the NRC, and by EPRI for the Seismicity Owner's Group (SOG) (or the higher of the two) should be used in the SPRA. However, it is the industry position that use of both sets of curves, or the highest of the two, is not needed or justified since the dominant sequences will be the same, and since only relative CDF values are important. The absolute values are to be deemphasized due to the large uncertainties.

For Western U. S. nuclear power plant sites, hazard curves applicable to each site will have to be obtained directly by the plant owners. These site hazard results present ground motion parameters, such as peak ground acceleration (PGA) and response spectra, for different annual probabilities of exceedance.

The SPRA approach results in identification of seismic-induced accident sequences, systems and specific equipment/structures which are significant contributors to the calculated risk. Potential chatter of relays and other contact devices as a result of earthquake excitation is not normally covered in the SPRA methodology,

but has been addressed in recent SPRA evolutions, and is required for IPEEE. Plant walkdowns similar in scope to SMA are required for SPRA. The information to be gathered during the walkdown is essentially the same as in a SMA review. Seismic probabilistic risk assessments conducted in the past may require new walkdowns to bring them up to the same level of thoroughness as currently requested by the NRC. The EPRI screening criteria in EPRI NP-6041 can be used to screen elements in a SPRA. It is assumed that screened out elements are not significant contributors to core damage risk. This can be verified easily by bounding analysis.

Elements which are not screened out are analyzed to develop fragility curves, which are characterized by median capacities and logarithmic standard deviations for variability.

A detailed guidance document for performing the SPRA that is comparable to the GIP for an A-46 review or NP-6041 for a SMA review does not now exist. However, References (31), (32), and (33) provide procedures and general information that can be used to assist in performing a SPRA. The systems analysis for SPRA involves developing event and fault trees that logically relate the plant structures and equipment which are significant to preventing core damage. Non-seismic failures, human errors, and dependencies between elements are considered. For each structure and component, fragility curves are developed that typically have been expressed in terms of a lognormal model with a median capacity and a logarithmic standard deviation for uncertainty and randomness. Fragility curves relate probability of failure or malfunction of a specific type of equipment to the intensity of loading (e.g., level of shaking). For IPEEE, only mean fragility curves are required; hence, variability can be combined into a single parameter for each element, which simplifies analysis.

Next, all the fragility curves for the elements in the event and fault trees are combined in a probabilistic manner to produce a mean core damage fragility curve. Intermediate output, including mean fragility curves for each of the accident sequences (or combination of sequences making up a particular plant damage state) that contribute to the probability of core melt can be obtained. Finally, the mean core damage fragility curve for each plant is integrated with the mean site hazard curve to obtain the mean frequency of core damage due to seismic events. The mean frequency of core damage is normally expressed as an annual probability.

The procedures in SPRA can be used to obtain a mean containment release frequency.

However, the NRC has only requested that vulnerabilities be identified for containment-related systems/functions that could lead to early containment failure and result in high consequences. Recommendations for performing a SPRA are summarized below.

PLANT WALKDOWN AND DOCUMENTATION

As part of performing a SPRA, a plant walkdown should be performed which is consistent with the procedures and guidelines used in a walkdown conducted for a SMA review (11). The documentation of the walkdown and PRA should be consistent in detail with the requirements for a SMA review.

USE OF SEISMIC HAZARD RESULTS

In a SPRA, the emphasis is placed on relative ranking of dominant accident sequences that contribute to overall risk; bottom line numbers should be deemphasized because of the large uncertainties. Since the EPRI and LLNL seismic hazard curves result in only minor variations, if any, in sequence ranking, use of only one set of hazard curves is adequate. Thus, it is recommended that the hazard results presented in the EPRI Report NP-6395-D (15) be used in performing the SPRA.

At sites where only LLNL hazard results are available, use of those results in the SPRA may be appropriate. However, licensees may wish to perform computations using the EPRI seismic hazard procedures in order to develop new seismic hazard data.

FRAGILITY CALCULATIONS

Mean fragility curves are considered adequate for use in a SPRA rather than a family of curves. Fragility curves should reflect the data obtained during the walkdowns. Potential for soil failure (e.g., slope stability, settlement and liquefaction) should be considered similar in scope to the requirements for SMA given in Section 3.

RELAY CHATTER

Relay chatter evaluation has not been generally addressed in past SPRAs; although, some recent SPRAs have addressed the relay chatter in detail. As noted in Section 3, evaluations performed as part of a SMA have been shown not to be cost effective. The USI A-46 program has identified the most common low seismic ruggedness relays; and the relay evaluations performed so far have shown that all

suspect relays are either on this low seismic ruggedness list or have been screened out by operator actions or systems considerations. Therefore, consideration of relays in a SPRA should parallel the effort for a deterministic review. For a plant at a full-scope site low seismic ruggedness relays can be explicitly considered in the SPRA in the fault trees. Although it appears that a SMA-type review for relays could be performed in lieu of embedding the relay chatter investigation in the SPRA. At a focused-scope site only a search for low seismic ruggedness relays should be conducted similar to a full-scope PRA. Finally, if a SPRA is conducted for a plant at a reduced-scope site no relay review is necessary.

HCLPF CALCULATIONS

Although the final Generic Letter 88-20, Supplement 4 makes calculation of HCLPF values for components, sequences, and the plant optional, SPRA without HCLPF values fully satisfies the IPEEE objectives stated in the Generic Letter 88-20, Supplement 4. Therefore, it is recommended that HCLPF values not be provided.

Section 5

INTEGRATION OF IPEEE AND A-46 REVIEWS

CONSIDERATIONS IN SELECTING THE IPEEE APPROACH

There are a number of considerations in selecting the best method of complying with the IPEEE requirements at a given A-46 site. These include the utilities overall objectives for use of the IPEEE results, cost, availability of prior seismic review work and data, availability of trained, experienced seismic engineers, and compatibility/overlap with the A-46 seismic review program. These factors are discussed briefly below in the form of advantages and disadvantages of the two approaches.

Seismic Margins Assessment

Advantages:

- The SMA is based largely on the same approach as an A-46 review and is proceduralized in a manner similar to the Generic Implementation Procedure (GIP). As a result, the SMA can be more easily integrated into the A-46 walkdown by utility engineers so as to minimize duplication of effort. The feasibility of this integration was demonstrated to the NRC's satisfaction at Plant Hatch.

- Equipment and structures to be reviewed in the SMA will normally be based on deterministic selection of safe shutdown paths. Significant potential exists for selecting shutdown paths such that the majority of equipment reviewed is the same equipment as that reviewed for resolution of A-46. Also, the option exists to select paths with higher equipment HCLPFs, so as to raise the plant HCLPF value.

- If components are screened out or have calculated HCLPFs that exceed the review level earthquake (RLE) then the components are adequate and do not need to be considered further. If all of the components exceed the RLE then the closure procedures (i.e., Section 6) do not have to be used.

- A combined SMA/A-46 review has been successfully completed (30).

Disadvantages:

- The Seismic Margins Assessment does not provide numerical risk

estimates as does the SPRA; thus, making cost/benefit analyses more difficult. Also, SMA does not provide insight into system dependencies and reliability.

- Seismic Margin Assessment was developed primarily for low-hazard sites (i.e., less than or equal to 0.3g pga). The methodology is not as applicable to higher-hazard sites.

- SMA has limited applicability for addressing safety issues in the future.

- Closure guidelines for SMA have been developed using the EPRI hazard curves which are generally lower than the LLNL curves.

Seismic PRA

Advantages:

- Method is compatible with the PRA which is being performed by the IPE for evaluation of internal accident initiators.

- May be economically attractive if fault trees developed in IPE can be used for equipment selection, although costs associated with relay evaluation and supplemental plant walkdowns need to be considered. Although it is expected that costs for relay review should be similar for both SMA and SPRA.

- Results in numerical risk assessment values which are relatable to safety goals and which can be used in cost/benefit analyses.

- SPRA is amenable to use on future safety issues; although, depending on extent of modification or evaluation, "pruned" trees may require extensive reconstruction.

Disadvantages:

- The results of the SPRA for most plants are strongly dependent on the seismic ground motion hazard estimates developed by EPRI and the NRC. These hazard estimates are significantly different (the NRC/LLNL estimates are higher) and are a source of ongoing debate and controversy.

- Beyond the plant walkdown and relay evaluations, there is less potential for integrating the A-46 seismic review with a SPRA.

- To date no attempt has been made to date to integrate the two reviews.

- No proceduralized document exists for SPRA, as does the GIP for A-46, and NP-6041 for SMA. Many utilities will have PRA-technical skill developed from the IPE; however, if this is not the case, utility seismic capability engineers cannot be practically trained to perform a PRA in the same time frame as with A-46 or SMA. Also, there are few in-house practitioners of fragility analysis; in most cases, an outside contractor will be needed.

- The SPRA has a potentially greater scope of equipment if the event/fault trees are not carefully "pruned".

On the basis of preliminary data and estimates, it appears that on balance the SMA approach is more compatible with A-46 than SPRA and, when relay evaluation and the need for plant walkdowns are considered, it appears the SMA would be comparable in total cost to A-46 if done separately. If the A-46 and SMA or SPRA reviews are integrated (e.g., an effort is made to select the same equipment and a single walkdown is performed for both the A-46 and SMA or SPRA), the savings will be substantial. It is expected that in general the SMA approach will be somewhat more cost effective (5 to 20%) than the SPRA, although plant-specific factors (such as availability of existing SPRA work, experience personnel, etc.) could increase the PRA cost.

Differences Between A-46, SMA, and SPRA Reviews

The intent of both the SMA and SPRA review programs is to identify seismic outliers which can be improved in a practical and cost effective manner with a significant safety benefit.

The principal differences between A-46/SQUG, SMA, and SPRA reviews are summarized below. Table 5-1 summarizes the technical differences, and Table 5-2 summarizes some of the managerial issues.

SMA Reviews - General. The A-46 and SMA review approaches are similar in that they are walkdown-based and make use of experienced, trained engineers who are expected to use screening techniques and engineering judgment to assess seismic capacities and identify seismic outliers. A key difference is that the SMA review is made for a higher ground motion level than the plant licensing basis. Items which do not pass the review level earthquake in the SMA are identified as outliers for further detailed review, but there is no commitment, nor necessarily a need, to implement corrective action. The SMA is less prescriptive and requires less documentation than the SQUG methodology, but after the initial screening walkdown, the SMA requires more extensive evaluation of outliers to establish the capacity of equipment and structures with a high confidence of a low probability of failure (HCLPF). In addition, the scope of equipment and structures to be evaluated for the SMA is broader than the A-46 review.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Scope of Plants Covered		
An A-46 review applies to plants designed for seismic equipment qualification requirements prior to IEEE 344-75.	Seismic IPE applies to all nuclear power plants. For SMA, plants are divided for analysis into three groups according to relative seismic hazard (i.e., full-scope, focused-scope, and reduced scope reviews).	Seismic IPE applies to all nuclear power plants. Site-specific hazard curves will be used for SPRA reviews.
Seismic Input		
Review based on licensing-basis SSE ground response spectrum. In-structure response spectra may be licensing-basis spectra or new spectra developed and used per GIP.	For focused- and full-scope reviews, median NUREG/CR-0098 response spectrum shape anchored to 0.3 g, or 0.5 g for plants in the western U.S. is used. Development of new in-structure response spectra, including effects of SSI, is encouraged. SMA seismic input is higher than A-46 ground motion. For Reduced-Scope reviews, SSE input based on GIP or licensing basis is used.	EPRI site-specific hazard curves for peak ground acceleration and response spectra should be used. If LLNL hazard curves are used, the absolute risk values will be significantly higher than using EPRI curves. The intent of IPEEE is met by relative results whether the EPRI or LLNL curves are used. The NRC GL 88-02 requests the use of both curves, but does allow use of one set, if the higher set is used. This issue requires resolution if SPRA is used. Ground motion input higher than both A-46 and SMA reviews is considered in SPRA, although the ground motions are weighted by their probability of occurrence.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Scope of Review		
Review limited to safe-shutdown equipment, including active electrical and mechanical equipment, tanks, heat exchangers, and cable trays and conduit raceways.	Review includes same types of safe shutdown equipment as for A-46 review, as well as passive mechanical and NSSS equipment, piping, containment, and structures. In addition, a small break LOCA is postulated to occur and soil failure modes are considered. Potential for earthquake-induced flooding/fire is also considered, as well as nonseismic failures and human actions.	Event trees and fault trees are developed for the seismic PRA using the IPE internal event/fault trees. Structures and elements whose failure could impact and fail safety-related elements are added to the trees. Generic letter requests evaluation of nonseismic failures and human actions.
Selection of Equipment		
Safe shutdown equipment may be in single safe shutdown path with redundancy for all active components, or in two separate safe shutdown paths. Single active failures are assumed.	Two separate and independent shutdown paths are selected using EPRI NP-6041 approach. Elements whose failure could lead to core damage are considered initially in NRC NUREG/CR-4334 approach. (Fault trees are "pruned" based on systems and fragility considerations.) Overlap with A-46 equipment list can be achieved by coordinating reviews.	Elements whose failure could lead to core damage are considered. (Fault trees are "pruned" based on systems and fragility considerations.) Overlap with A-46 equipment list can be achieved by coordinating reviews.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Required Experience and Training of Engineers		
A team of trained experienced seismic capability and systems engineers will perform A-46 reviews. A training course is required as part of SQUG program.	SMA will be performed by trained experienced seismic capability and systems engineers. An add-on training course will be given to cover areas of different between A-46 and SMA reviews.	SPRA will be performed by experienced systems and seismic capability engineers. PRA skills from IPE will be transferable, but fragility analyses will most likely require an outside contractor. The add-on course to A-46 training will address only general differences between A-46 and SPRA, and the supplemental walkdown and relay evaluation requirements for SPRA. Detailed SPRA methodology (e.g., event/fault tree methods, fragility analysis) will not be taught.
Screening Requirements		
Equipment functionality is investigated based on seismic experience or test data. Equipment anchorage is analyzed for each component.	The screening criteria in the GIP are recommended for equipment common to both A-46 and SMA programs (see EPRI Implementation Report). Requirements in EPRI NP-6041 are used for elements common only to the SMA structures/equipment list. Caveats and guidance are provided in SMA screening criteria tables for three ranges of seismic input.	The screening criteria given in EPRI NP-6041 can be used to screen elements in a SPRA. It is assumed that screened out elements are not significant contributors to core damage. This assumption can be verified easily by bounding analysis.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Walkdown Procedure		
Walkdown procedures are documented in the GIP. Principal elements of the walkdown are: • Seismic capacity versus seismic demand; • Caveats based on earthquake experience and generic testing data base; • Anchorage adequacy; and • Seismic-spacial interaction with nearby equipment, systems, and structures.	Walkdown using the GIP is recommended for equipment common to both programs, with a supplemental walkdown sheet for SMA requirements. EPRI NP-6041 is used for elements not common to both program (i.e., piping, structures, passive mechanical and NSSS equipment, and containment). Elements not screened out are identified as outliers for further review. Potential for earthquake-induced flooding is considered in SMA walkdown.	Walkdown procedures for SPRA will follow the requirements for SMA in EPRI NP-6041. The level of required detail and effort is the same as for SMA per NP-6041. Field data required for both SMA and SPRA reviews is essentially the same.
Evaluation of Outliers		
Evaluation of outliers follows requirements in GIP and procedures in plant Licensing basis.. The details for resolving outliers is beyond the scope of the GIP. It is the responsibility of the utility to resolve outliers using their existing engineering procedures as they would resolve any other seismic concern.	For focused- and full-scope reviews evaluation of outliers follows EPRI NP-6041. Factors of safety for development of HCLPFs are generally more realistic than for A-46 review. For Reduced-scope review evaluation follows GIP and/or FSAR requirements. Performing A-46 and SMA calculations at same time will minimize cost.	For elements not screened out during walkdown fragility parameter values (i.e., median capacities and combined logarithmic standard deviations) are calculated. Performing A-46 and SPRA calculations at same time will minimize cost.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Relay Review		
Detailed review per GIP is required of those relays needed for successful operation of safe shutdown equipment and support systems, and those relays whose inadvertent actuation due to chatter could result in an unacceptable event. Low ruggedness relays which have demonstrated low resistance to impact and whose chatter could prevent successful operation of safe shutdown systems or cause an unacceptable event are identified for corrective action.	It is recommended that plants that perform a full-scope SMA only conduct a review for low-ruggedness relays. However, plants performing an A-46 review should conduct a relay review according to the procedures in the GIP for all relays within the scope of the A-46 review. Investigation of relays outside the A-46 review but within the scope of the IPEEE review should address only the issue of low-ruggedness relays. For A-46 plants in reduced-scope category, the GIP relay evaluation scope is performed. If low seismic-ruggedness relays (bad actors) are found, perform bad actor review outside A-46 but within IPEEE. For non-A-46 sites in reduced-scope category, only location and evaluation of bad actors is requested. Relays are not investigated for plants in low seismic hazard areas which perform reduced-scope reviews, except for A-46 plants, which will require relay evaluation per the GIP.	The scope of relay chatter review should be consistent with the site's SMA review level. An SMA type review could be performed in lieu of embedding relay chatter investigation into the SPRA event/fault trees.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Containment Review		
Containment review is not required in A-46 review.	For focused- and full-scope SMAs, review for containment integrity, isolation, and prevention of bypass will be conducted. Scope is based on internal events IPE PRA. Note that with the SME equal to 0.3 g pga, the review of concrete and certain steel containments is minimal. Containment review is not conducted in reduced-scope review.	Containment evaluation is required consistent with SMA-type review. The SPRA for IPEEE relies substantially on IPE PRA; i.e., overlay IPEEE seismic hazard onto IPE PRA and report sequences which are negatively impacted.
Quality Assurance		
A-46 review includes some construction checks (e.g., equipment anchorage, cable tray/conduit raceways, interaction hazards, and mounting of internal elements).	SMA assumes that plant is constructed according to design. Also, for older plants it is assumed that either A-46 review has been conducted, or is being conducted concurrently.	SPRA assumes that plant is constructed according to design. Also, for older plants it is assumed that either A-46 review has been conducted, or is being conducted concurrently

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Products		
Specific plant review information is produced per GIP and includes: • Description of safe shutdown path(s); • Lists of equipment on composite, seismic review, and relay review SSELs; • Description of SSE used in A-46 program; • Qualification of review personnel; • Results of the screening verification and walkdown for equipment; and • Summary of main steps in plant operating procedures to bring plant to safe-shutdown condition.	The results of a SMA using the procedure in NP-6041 include: • General plant description and seismic design basis; • Seismic margin earthquake and development of demand on elements; • Seismic margin evaluation (approach, screening criteria, systems description, review team, walkdown, and shutdown path selection); • Assessment of elements not screened out (structures, equipment, and soils); • HCLPFs for components that are less than RLE; and • Results of evaluation and insights gained.	The results of a SPRA include the following: • Description of methodology and key assumptions; • Hazard curve(s); • Walkdown team, procedures, and findings; • Systems information (event/fault trees, nonseismic failures, and dependencies); • List of fragility parameter values; • Mean core damage frequency and ranking of accident sequences, systems, and structures/equipment which are significant contributors); • Seismic-induced containment failures and other containment performance insights; and • Results of evaluation and insights gained.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Documentation Requirements		
Requirements for documentation are given in the GIP, including SSEL Relay Evaluation and Seismic Evaluation Reports. A final completion letter is required.	Requirements for documentation are given in EPRI NP-6041, which includes general plant description, plant seismic design basis, development of SME demand, SME evaluation, assessment of elements not screened out, and summary and conclusions. Less documentation is required than for A-46 review.	The amount of documentation for a SPRA will be similar to SMA.
Closure Process		
Elements not meeting A-46 criteria (i.e., GIP) are corrected; or if safety benefit is not sufficient, are reported as not warranting corrective action. Back-fit provisions of 10 CFR 50.109 apply for these cases.	HCLPF results below SME can be resolved using NUMARC Severe Accident Issue Closure Guidelines (34), which would include cost/benefit of raising plant HCLPF.	Numerical results are relatable to CDF goals and are resolved using NUMARC Severe Accident Issue Closure Guidelines, which would include cost/benefit analysis of proposed risk reduction opportunities. Use of EPRI vs. LLNL hazard curves would need resolution.

Table 5-2

COMPARISON OF MANAGERIAL ISSUES BETWEEN SMA AND SPRA REVIEWS

SMA	SPRA
Regulatory Acceptability	
The EPRI NP-6041 methodology has been used in two trial reviews, and the NRC NUREG/CR-4334 methodology has been used in one trial review. The results of all three reviews have been accepted by the NRC.	SPRAs have been conducted for over 30 plants. Seven SPRAs have been submitted to NRC as nonlicensing analyses, and one submitted as a licensing condition. All have received favorable evaluations. There is greater potential for argument (compared to SMA) if absolute results are stressed. However, NRC states relative results will be emphasized.
Compatibility With IPE	
SMA is not directly compatible with IPE risk results; however, a risk goal-based closure procedure is given in NUMARC Severe Accident Issue Closure Guidelines when SMA is performed (34).	SPRAs are compatible with PRAs conducted for internal event and other external event accident initiators (e.g., fire and wind). Closure procedures are given in NUMARC Severe Accident Issue Closure Guidelines (34).
Compliance With IPEEE Requirements	
The SMA is an acceptable methodology for IPEEE, although NRC Generic Letter 88-02 does not explicitly state that SMA conforms with the general requirements: • Find risk from seismic events; • Develop an appreciation of severe accident behavior; • Understand the most likely core melt sequence and gain a qualitative understanding of core melt probability; and • Reduce probabilities.	SPRA complies directly with requests in NRC's Draft Supplement 4 to Generic Letter 88-20.

Table 5-2

COMPARISON OF MANAGERIAL ISSUES BETWEEN SMA AND SPRA REVIEWS

SMA	SPRA
Ease of Use by Utilities	
Procedures are well documented in EPRI NP-6041. Add-on training course will provide explanation of differences with A-46 review so work can be performed by utility personnel.	Procedures for performing SPRA are documented in NUREGs CR-2300, 2815, and 4840, and past SPRA submittal to NRC. PRA technical skills from IPE will transfer to SPRA at many plants. However, supporting fragility analysis skills are limited to few practitioners. It is likely that outside contractors will be used to perform fragility analyses. As a minimum, utility personnel will participate in walkdown and relay evaluation. Add-on training courses will address these aspects.
Relative Cost	
Cost is minimized if walkdowns for A-46 and SMA are conducted simultaneously. Costs can be further decreased if the A-46 and SMA evaluation of outliers is performed at same time by same engineers.	Cost is minimized if walkdowns for A-46 and SPRA are conducted simultaneously. Costs can be further decreased if the A-46 and SPRA evaluation of outliers is performed at same time by same engineers. Cost for SPRA likely will be greater (i.e., 5 to 20%) compared to SMA. Additional systems work is required (see Compliance with IPEEE Requirements above). Also, calculation of logarithmic standard deviations will require some extra work. In general, calculation of HCLPFs for SMA and median capacities for SPRA are comparable. However, in some cases calculation of median capacities will require some added effort.

Table 5-2

COMPARISON OF MANAGERIAL ISSUES BETWEEN SMA AND SPRA REVIEWS

SMA	SPRA
Ease of Use by Utilities	
Procedures are well documented in EPRI NP-6041. Add-on training course will provide explanation of differences with A-46 review so work can be performed by utility personnel.	Procedures for performing SPRA are documented in NUREGs CR-2300, 2815, and 4840, and past SPRA submittal to NRC. PRA technical skills from IPE will transfer to SPRA at many plants. However, supporting fragility analysis skills are limited to few practitioners. It is likely that outside contractors will be used to perform fragility analyses. As a minimum, utility personnel will participate in walkdown and relay evaluation. Add-on training courses will address these aspects.
Relative Cost	
Cost is minimized if walkdowns for A-46 and SMA are conducted simultaneously. Costs can be further decreased if the A-46 and SMA evaluation of outliers is performed at same time by same engineers.	Cost is minimized if walkdowns for A-46 and SPRA are conducted simultaneously. Costs can be further decreased if the A-46 and SPRA evaluation of outliers is performed at same time by same engineers. Cost for SPRA likely will be greater (i.e., 5 to 20%) compared to SMA. Additional systems work is required (see Compliance with IPEEE Requirements above). Also, calculation of logarithmic standard deviations will require some extra work. In general, calculation of HCLPFs for SMA and median capacities for SPRA are comparable. However, in some cases calculation of median capacities will require some added effort.

SPRA Reviews - General. The A-46 and SPRA review approaches are similar mainly in the areas of walkdown screening and evaluation of relays. The NRC Generic Letter requests that the walkdown and relay evaluations be done in accordance with SMA procedures; therefore, either method will result in the same review for these issues. The key difference is that a probabilistic approach requires initial consideration of all equipment which could lead to core damage; thus, a broader scope of equipment must be reviewed for SPRA than for A-46 and possibly for SMA. Also, site-specific hazard curves are typically at a higher level ground motion than either the A-46 or SMA review. SPRA considers the probability of earthquake, which may offset the effect of the higher magnitude, i.e., larger earthquakes have lower probability of occurrence. The bulk of the SPRA evaluation will need to be performed by experienced systems and seismic capability engineers with PRA expertise who can perform fragility analysis. While many utilities will have PRA skills developed from IPE, the seismic capability engineers will most likely be outside contractors, especially for the fragility analysis. Utility personnel will participate, as a minimum, in at least the walkdown and relay evaluation.

Scope of Plants. Unlike A-46, seismic IPEEE applies to all nuclear power plants. For SMA, plants are divided into three groups according to relative seismic hazard: full-, focused-, and reduced-scope. SPRA applies to all plants, but there is no grouping by relative seismic hazard, except for the requirements for relay chatter, containment and soil failure evaluation.

Seismic Review Level. The A-46 review is based on the plant's licensing basis SSE ground motion spectra and corrective action is required for those items which do not meet the A-46 criteria at this level, unless the utility invokes the backfit provisions of 10 CFR 50.109.

For focused- and full-scope reviews, the SMA review level is higher than the SSE (0.3g PGA together with a NUREG/CR-0098 median-centered spectra for most plants). For reduced-scope reviews, SSE input, based on licensing basis, is used.

For the SPRA review, site-specific hazard curves for peak ground acceleration and response spectra are used. Ground motion covers the range of accelerations from zero to maximum physical values which are higher for SPRA than in either A-46 or SMA. However, in a PRA the accelerations are weighted by their probability of occurrence.

Governing Criteria . The A-46 and SMA reviews are directed primarily at assuring

safe shutdown of the plant following an earthquake. The A-46 review is generally limited to safe shutdown equipment; the SMA includes piping, containment and structures as well. In both cases, the safe shutdown equipment is that equipment needed to achieve and maintain safe shutdown for 72 hours following an earthquake. In the A-46 review, concurrent LOCAs are not postulated to occur; in the SMA, a small break LOCA (SBLOCA) is postulated to occur. The primary impact of these differences is the addition of SBLOCA mitigation systems (i.e., high pressure make-up capability) to the safe shutdown equipment to be reviewed.

Scope of Review

The safe shutdown equipment included in the A-46 review consists of active electrical and mechanical equipment, tanks and heat exchangers needed for safe shutdown and cable tray and conduit raceways. The SMA scope includes these components and the following additional areas that are on the safe shutdown paths:

- Nuclear steam supply system components
- Containment systems (those which affect early containment failure)
- Piping
- Category 1 civil structures
- Soil failure mechanisms

In the case of the safe shutdown equipment selection, the A-46 and SMA rules are somewhat different. Both programs require redundant safe shutdown equipment and the systems necessary to support the primary safe shutdown systems. However, in the A-46 program the safe shutdown equipment may be in a single safe shutdown path with redundancy for all active components, or alternatively, in two separate safe shutdown paths. The SMA requires that the safe shutdown equipment be based on two separate and independent paths. In addition, the A-46 rules require assumption of single active failures, while the SMA methodology does not. However, in the SMA method, paths are chosen based on a screening criterion applied to nonseismic failures (e.g., battery depletion, PORV failure) and human actions (e.g., delays or failures in performing specified actions). These differences could lead to significantly different safe shutdown equipment lists. However, initial selection of the safe shutdown paths with both sets of rules in mind can result in significant overlap of the required active safe shutdown and support equipment, as was demonstrated in the Plant Hatch A-46/SMA review (30).

For SPRA, structures, components and systems whose failure could lead to core damage are considered initially. Event trees and fault trees are developed for the SPRA using the IPE internal event/fault trees. Structures, components and systems whose failure due to a seismic event could impact and fail safety-related elements are added to the trees. Nonseismic failures and human actions are to be included, unless shown to be insignificant, as in the SMA approach. Fault trees are "pruned" based on systems and fragility considerations. Overlap with the A-46 equipment list can also be accomplished by coordinating reviews. The extent of overlap which can be achieved is believed to be significant, but little direct experience is available for performing this conditional review.

Review Methodology. Some key differences in review methodology between the A-46 and IPEEE are as follows:

- The SMA and SPRA approaches assume the plant is constructed according to design; the A-46 review includes some installation checks.

- The factors of safety for equipment anchorages are generally more conservative in the A-46 program, i.e., the A-46 program uses a factor of safety of 3, the SMA uses factors of safety ranging from 2 to over 3 in some cases. SPRA doesn't explicitly use factors of safety. The variability in median capacity reflects a range of factors of safety.

- The methods for evaluation of ground mounted storage tanks are more realistic in the IPEEE review compared to the more conservative A-46 criteria, e.g., water hold-down forces are included in SMA but not A-46 evaluation.

- An extensive equipment experience data base is utilized in the screening of outliers in the SMA. This process results in estimates of equipment capacity which are intended to provide approximately 95% confidence that the probability of failure does not exceed about 5% at the HCLPF capacity level. SPRA develops median capacities and logarithmic standard deviations for variability. The A-46 or SMA capacity review is pass/fail while in SPRA there is flexibility for a graded response with the possibility of no action being recommended.

- The A-46 and IPEEE seismic reviews both include evaluation of seismic spatial interactions. For IPEEE the potential for earthquake-induced pipe ruptures and, if necessary, effects of possible flooding, are also reviewed.

- Review methodologies are included in the SMA for soil structure interaction, soil failure evaluation, piping assessment, containment systems review and evaluation of civil structures, consistent with the required level of review. SPRA includes similar added scope as SMA.

- The scope of containment reviews for both SMA and SPRA are based on the IPE internal events. However, the IPEEE SPRA relies substantially on this IPE evaluation, with the overlaying of the IPE PRA with the IPEEE seismic hazard, and reporting of those sequences which are negatively impacted.

Relay Evaluation. The A-46 methodology requires a detailed review of those relays which are necessary for functioning of safe shutdown equipment and support systems and those relays whose inadvertent actuation due to chatter could result in an unacceptable event. This relay evaluation process is defined in the GIP and includes a two-pronged approach involving review of the effect of relay chatter on system function and seismic adequacy review of those relays whose function is essential. As part of this process, essential relays which have low fragility and/or have demonstrated low resistance to impacts are separately identified for consideration of corrective action.

In the SMA review approach, the extent of relay evaluation is dependent on the earthquake review level. Those plants in the full-scope category will require a detailed review of low seismic ruggedness relays. Plants which are identified as low seismic plants will not require a relay seismic evaluation. The remainder of the plants, consisting of the large majority of plants in the U.S., are in the focused-scope category. For A-46 plants in the focused scope category, the GIP relay evaluation scope and procedure is used. If low seismic-ruggedness relays (bad-actor list) are discovered during the A-46 review, the relay reviews should be expanded outside the scope of A-46 but within the scope of IPEEE. For non-A-46 plants in the focused-scope category, the Generic Letter requests only the location and evaluation of low seismic-ruggedness relays (bad-actor list).

For an SPRA, the Generic Letter requests the scope of relay chatter review be consistent with the site SMA review level as described above. The same arguments as used in A-46 and SMA reviews can be used to screen relays from the analysis (i.e., operator recovery and/or high seismic capacity). Fragility curves are developed for relays which may be significant contributors to core damage. Both the potential for structural failure and operator failure to reset are considered as in an SMA review. An SMA type review could be performed in lieu of embedding relay chatter investigation into the SPRA fault trees.

Documentation and Quality Assurance Requirements. The A-46 review, as prescribed in the GIP (21) and required by the NRC's Safety Evaluation Reports, requires significantly more documentation and QA coverage than does the SMA or SPRA.

Specifically, the significant inspections, reviews and checks required by the A-46 process are included on checklists and on summary verification sheets which formally document the acceptability of each component on the safe shutdown equipment list. Personnel qualifications and training requirements are formalized. In addition, any corrective actions taken in the A-46 review which result in changes to plant licensing bases, plant procedures or plant hardware are required to be accomplished in accordance with the plant's formal 10 CFR 50 Appendix B QA program. The SMA or SPRA, on the other hand, are performed pursuant to an NRC request for information, are not related to the plant's licensing basis and accordingly, are not safety-related review programs. Any plant procedure or hardware modifications which result from the SMA or SPRA could be subject to each plant's QA program, independent of the reasons for such changes, although this would be at the discretion of the licensee.

Regulatory Acceptability. Two trial reviews using EPRI SMA methodology and one using NRC SMA methodology have been performed, and the results have been acceptable. Over 30 SPRAs have been performed--eight of which have been submitted to the NRC; all eight received favorable evaluations. There is greater potential for arguments with the SPRA if absolute results are stressed, but Supplement 4 to NRC Generic Letter 88-20 states that relative results will be emphasized.

Ease of Use by Utility. GIP methodology for A-46 evaluation is extensively proceduralized. SMA procedures are well documented in NP-6041. Training for SMA similar to that for A-46 will be provided to all interested utilities and should allow utility personnel to perform SMA reviews. Utilities may have some in-house PRA skills from the IPE; however, the fragility analyses will almost certainly need to be performed by an outside contractor. Training of utility engineers in SPRA would not be practical in the necessary time frame, although the walkdown and relay evaluation in the SMA training program would be applicable. As a minimum utility personnel will be participating in the walkdown and relay evaluation.

Future Use. SQUG/A-46 methodology can be used for new and replacement parts. SPRA can be used as a "living PRA" to address future changes and other safety issues although "pruned fault trees" may have to be reconstructed to consider certain specific systems and equipment. SMA is a single time point method; however, it would be relatively straightforward to apply the method when future plant modifications are planned.

Relative Cost. Cost can be minimized by performing walkdowns and outlier

evaluations for A-46 and either SPRA or SMA simultaneously. Overall cost for SPRA will likely be about 5% to 20% higher than SMA, based on additional systems work and potentially a longer list of equipment to be reviewed, although plant-specific factors (e.g., existing SPRA work, experience of personnel) could reduce the SMA cost. Calculation of HCLPF for SMA and median capacities for SPRA are generally comparable.

RECOMMENDATIONS

It is recommended that IPEEE and A-46 reviews be conducted concurrently and that the review tasks be combined, whenever possible. At the utility's option, the IPEEE and A-46 reviews can be conducted independently. However, this would require duplication of some work, which can be avoided if the reviews are coordinated. As a minimum the studies should be conducted at the same time, or the A-46 review performed first. This is necessary since the IPEEE review assumes that quality-related issues have been investigated and resolved. Since it is acceptable to perform the reviews independently, the SRT can always directly use the corresponding criteria for each review. Guidance for integration of IPEEE deterministic and A-46 reviews is given in Appendix C.

Combined Deterministic Reviews

Most of the equipment and components included in the A-46 and IPEEE deterministic programs should be the same; however, there may be some differences. For example, the philosophy for providing redundancy for the primary success path is different for the two programs. This may lead to the selection of different components.

For plants performing combined reviews, the walkdown requirements should follow the guidance given in the GIP (21). The GIP procedure will be generally more conservative for the 0.3 g pga RLE plants compared to the requirements given in EPRI Report NP-6041 for a SMA review, for those components also included in the IPEEE review. However, this approach should result in a cost-effective review. The SRT can always revert to the requirements in Reference (11) for components in the IPEEE assessment, if the requirements conflict with the A-46 review. For components outside the scope of an A-46 review, EPRI Report NP-6041 should be used for screening.

Equipment which is screened out during the plant walkdown is not considered further in the review. For the full-scope SMA and to a limited extent, the focused-scope SMA, HCLPF values should be determined for outliers. For the

components common to both programs, an evaluation to the design basis must be performed to satisfy the commitments in the A-46 program. For the reduced-scope assessment, components outside of the GIP should be analyzed based on the FSAR commitments. It is recommended that common calculations be performed which cover both programs, wherever possible, in order to minimize the work.

In documenting the evaluations, separate reports should be prepared for the IPEEE and A-46 programs, consistent with the reporting guidance of each program.

Section 6

DESCRIPTION OF CLOSURE PROCESS

BACKGROUND ON CLOSURE OF SEVERE-ACCIDENT ISSUES FOR SEISMIC EVENTS

This section describes guidelines on the appropriate use of seismic-IPE review approaches to assist licensees in formulating effective decisions for achieving closure on seismic severe-accident issues. These closure guidelines are substantially consistent with the guidelines and framework recommended by industry for resolving severe-accident issues for internal events (34), yet they emphasize 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 closure guidelines described in this section are included as part of an overall industry document for severe-accident-issues closure [see (34)]. That overall document describes how industry's severe-accident closure process and framework satisfy all six elements (see Section 1 of this report) of the NRC's integrated closure plan (2). The IPE plays a central role in the integration plan and in industry's closure process.

Probabilistic risk assessment, level-1 in scope (with enhancements for containment evaluation), is the format recommended by industry and the NRC for performing the internal-events IPE. The internal-events closure guidelines, therefore, are consistent with Level-1 PRA results; i.e., core-damage-frequency criteria and major accident-sequence groups are the basic elements of these guidelines. In contrast, as discussed in Sections 3 and 4, the recommended formats for performing the seismic IPE include: (1) the reduced-scope assessment, (2) the focused-scope SMA, (3) the full-scope SMA, and (4) the SPRA. The closure guidelines for seismic events must, therefore, support the use of deterministic (SMA) approaches in addition to the probabilistic (PRA) approach. In the EPRI SMA approach, success-path logic diagrams (SPLDs) are constructed to convey the various combinations of component or operator actions that lead to a long-term safe-shutdown condition, given a seismic margin earthquake (SME). To make consistent use of industry's internal-events closure guidelines, results in the SMA format associated with particular SPLDs must be related to major core-damage-sequence group frequencies.

OVERVIEW OF CLOSURE APPROACHES

The critical action in the seismic IPE, regardless of the specific implementation approach taken, is to walkdown the plant (reviewing the safe shutdown systems and equipment) and to evaluate elements identified as outliers (for instance, using the SMA screening tables in EPRI Report NP-6041, Rev. 1) or identified as dominant risk contributors.

In the reduced-scope assessment, the outliers identified during the review 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 the next subsection.

For the focused- and full-scope SMA plants, outliers (i.e., the elements that do not pass the SMA screening tables at the SME) are evaluated to estimate their HCLPF capacities. As mentioned above, to develop a seismic-IPE closure approach for these plants that is consistent with the internal-events closure guidelines, target HCLPF capacity values for each success path can 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 target HCLPF capacity (for a success path with a given functional plant state) that will 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 Appendix E [also see Reference (24)]. Three RLGMS, denoted RLGM-A, RLGM-B and RLGM-C, are determined corresponding to three different core damage frequency based closure criteria. RLGM results for 58 central and eastern United States plants are presented in Reference (35). (Table E-1 presents RLGM-PGA values obtained by scaling the 5%-damped NUREG/CR-0098 median spectral shape to just envelope the site-specific RLGM spectrum over the vibration frequency range of 2 to 10 Hz. These PGA values may be used as conservative surrogates to the RLGM spectra, and provide a simpler basis for comparison with HCLPF capacity results). The closure guidelines for the focused- and full-scope review categories are given in the subsection following the reduced-scope closure guidelines.

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

therefore important to evaluate each of the two success paths against the closure guidelines.

The success path can be used as a conservative surrogate to a functional accident sequence; nence, 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 demonstrably 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 IPE 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 PRA-based closure guidelines for internal events (34). The closure guidelines for SPRA review are presented following those for the SMA approaches.

CLOSURE GUIDELINES FOR REDUCED-SCOPE IMPLEMENTATION

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

- Delineate two alternate success paths and define their major functional states (Section 3);
- Develop a list of screened-in outliers using the SMA screening tables (Section 3); and
- Evaluate components for compliance with licensing commitments (FSAR) or with the GIP guidelines based on earthquake experience qualification (Figure 6-1).

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 for plants covered under the A-46 program (Appendix C). Figure 6-1 describes, in flowchart form, the framework for closure evaluation when a reduced-scope assessment is conducted.

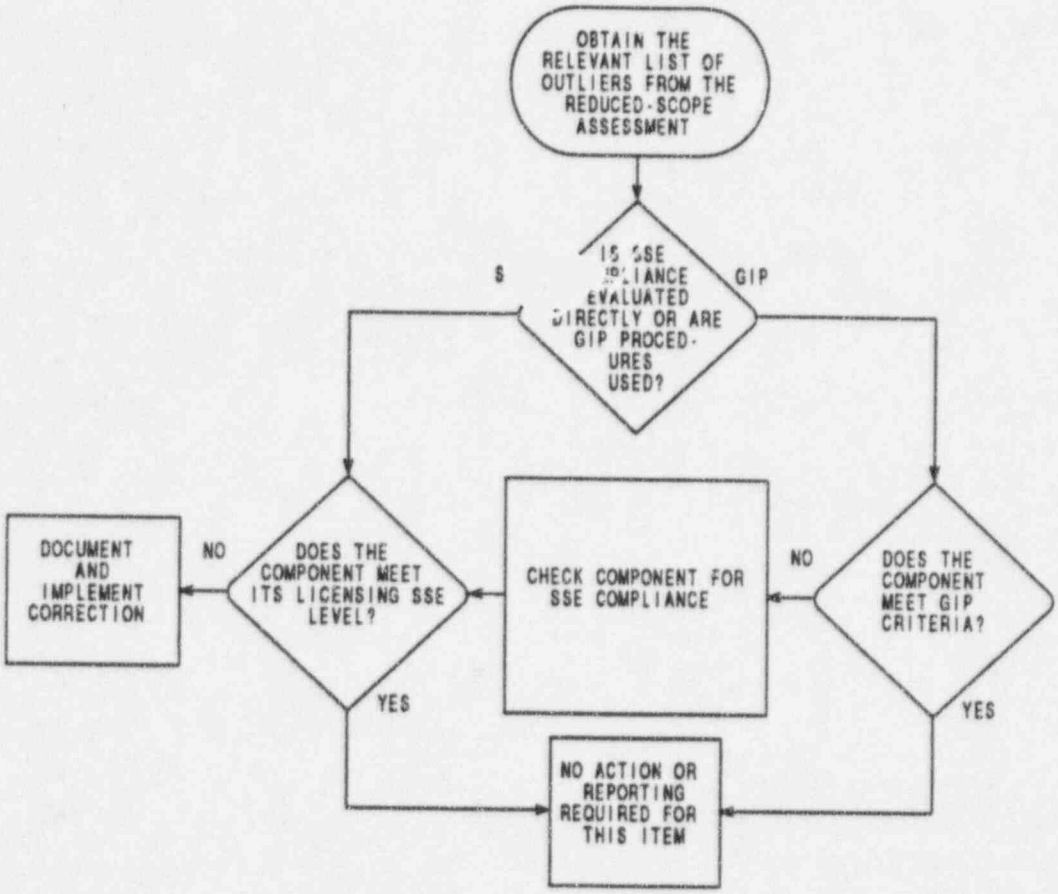


Figure 6-1. Closure process recommended for reduced-scope seismic-IPE evaluation.

The seismic-IPE-related closure process is completed when every reduced-scope SMA outlier has been evaluated, and all appropriate actions (if any) have been determined, documented, and scheduled for implementation.

CLOSURE GUIDELINES FOR FOCUSED-SCOPE AND FULL-SCOPE IMPLEMENTATIONS

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

- Delineate two alternate success paths and define their major functional states (Section 3);
- Develop a list of screened-in outliers using the SMA screening tables (Section 3);
- Calculate HCLPF capacities for outliers using the NUREG/CR-0098 (5%-damped) median spectral shape to characterize ground-motion input, and develop a list of screened-in remaining outliers (Section 3);

- Obtain RLGM spectra [Reference (35)] or RLGM-PGA values (Table E-1) to be used for evaluation of remaining outliers; and
- Evaluate remaining outliers against closure guidelines by comparing component HCLPFs with RLGMs.

Prior to evaluating success-path elements against RLGM-based closure criteria, HCLPF capacities are computed for the appropriate set of components identified in the seismic IPE, using the 5%-damped NUREG/CR-0098 median spectrum as input, as outlined in Section 3. A check for compliance with the SSE licensing commitment is performed when required (see Figure 6-2), similar to that in the reduced-scope SMA evaluation. Using the guidelines or results specified in Appendix E, three separate RLGMs for any plant (and given damping and soil type) are obtained for use in seismic-IPE closure; these three motions are determined for core-damage-frequency safety targets of 5.0×10^{-5}, 2.0×10^{-5}, and 5.0×10^{-6} per year.

Figure 6-2 describes the complete pre-closure process used to screen-in a list of remaining outliers from the initial list of SMA outliers. Although not specifically called out in the NRC staff guidance, the licensee should be cognizant of the status of the screened-in elements relative to the licensed SSE level. 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 6-3 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 criteria in the top row of (triangular-shaped) decision elements of Figure 6-3, this framework is identical to that for IPE core-damage evaluation [see Figure 1, Reference (34)] in internal-events closure. The RLGM-based criteria are themselves developed to be consistent with the corresponding core-damage-frequency related criteria in the internal-events IPE core-damage closure evaluation.

Figure D-1 shows a general extension of success-path elements associated with containment performance. For seismic containment sequences and related success-path elements (see Appendix D), the SPRA database suggests that substantial margin exists to prevent large early release and large containment bypass (24). Figure 6-3 is, therefore, also applied for closure evaluation of success-path elements needed to prevent large-early containment release and large

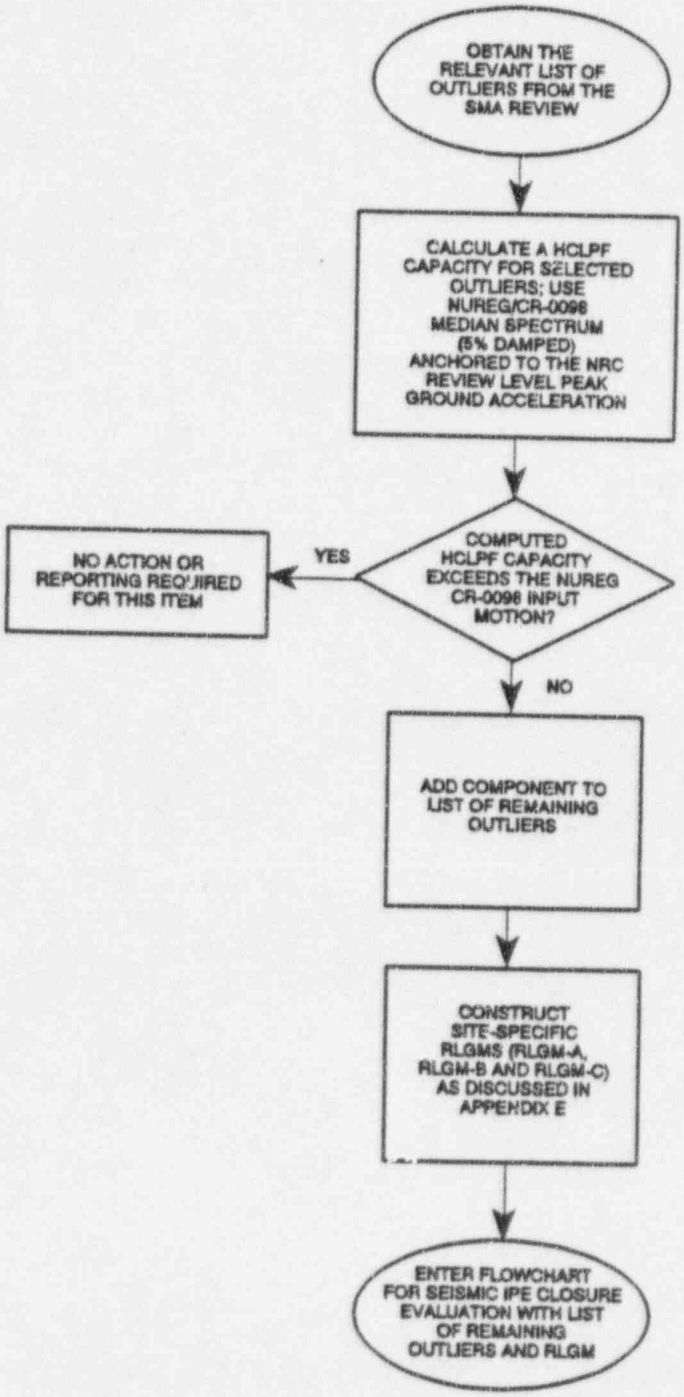
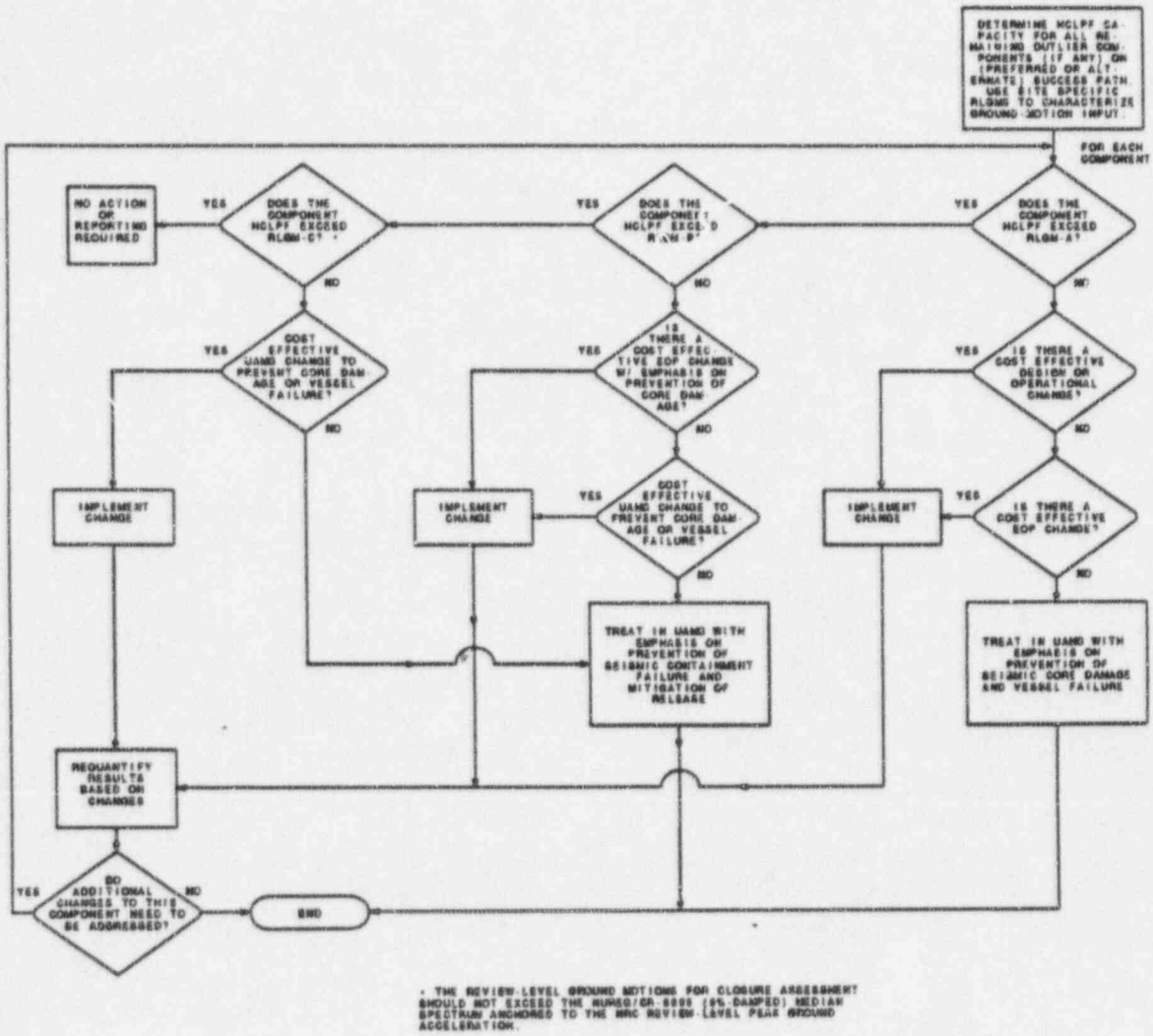


Figure 6-2. Pre-closure assessment for evaluation of outliers in focused-scope and full-scope SMA seismic IPEs.



UAMG: Utility Accident Management Guidelines
EOP: Emergency Operating Procedures

Figure 6-3. Closure process recommended for seismic-IPE core-damage evaluation: focused-scope and full-scope SMA.

containment bypass. Consistent with NRC guidance in Supplement 4 to Generic Letter 88-20 for seismic events [Reference (9)], separate closure guidelines for evaluation of containment-related elements and for evaluation of core-damage elements is not required.

The seismic-IPE-related closure process is completed when all components in the appropriate set of focused- or full-scope remaining SMA outliers have been evaluated and appropriate corrective actions (if any) have been determined, documented, and scheduled for implementation.

CLOSURE GUIDELINES FOR SPRA IMPLEMENTATION

If a SPRA is performed, the seismic core-damage frequency can be treated either as a single core-damage sequence group or as a multiple number of major sequence groups, for evaluation against closure guidelines. If multiple seismic core-damage sequence groups are utilized, a grouping philosophy similar to that intended for the internal-events IPE [see Reference (34)] should be taken in defining seismic accident-sequence groups. In the case of any external event, however, an appropriate set of accident-sequence groups can be simply obtained based on the nature of the sequence induced by the external hazard (e.g., seismic-induced LOCAs, seismic-induced station blackout, etc.). Components important in seismic core-damage sequences that may lead to containment bypass or release should also be considered in the closure process. An example grouping scheme for SPRA sequences is provided in Appendix B of Reference (34).

Once the SPRA sequences have been grouped, the seismic-IPE closure process involves comparing seismic core-damage group frequencies and containment-related sequence frequencies to the internal-events IPE closure guidelines in Tables 1A and 2A, respectively, of Reference (34). Thus, the closure process for SPRA implementation is similar to that for the internal-events IPE implementation.

This seismic-IPE closure process assumes compliance with the plant's licensing commitment. The closure process is completed when the SPRA sequences have been identified and evaluated, and all appropriate corrective actions (if any) have been determined, documented, and scheduled for implementation.

Section 7

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