Appendix A

TECHNICAL BASIS AND DESCRIPTION OF APPROACH FOR REVIEW METHOD SELECTION

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Appendix A

TECHNICAL BASIS AND DESCRIPTION OF APPROACH FOR REVIEW METHOD SELECTION

INTRODUCTION

This appendix describes the recommended approach and its basis for selecting an appropriate (i.e., effective and efficient) review method for performing the seismic IPE. This material supports the more-specific guidelines suggested in Section 2 for review-method selection. The approach is based on considerations of site seismic hazard results, of plant seismic design bases, and on consistent use of probabilistic measures. The selection of review method complements both the execution of specific recommended implementation procedures for seismic evaluation, the actual details of which are discussed in Section 3, and the use of decision criteria for severe accident policy closure, as discussed in Section 6.

The development in this appendix is consistent with the objectives of the IPE (or, more generally, objectives of the severe-accident policy (SAP) statement (1) of the Nuclear Regulatory Commission (NRC)); i.e., the purpose of seismic evaluation is to identify potential plant vulnerabilities to severe accidents that may be initiated by earthquakes.

SEISMIC REVIEW METHODS, HAZARD RESULTS, DESIGN BASES

Development of an approach for selecting an efficient seismic review method for severe-accident evaluation requires knowledge of available seismic review methods, characterizations of seismic hazard, and basic descriptions of plant seismic capacity. We provide below a brief review and description of each of these items.

Seismic Review Methods

The set of common choices available for plant-specific seismic evaluation -- seismic probabilistic risk assessment (SPRA) and seismic margin assessment (SMA) -- is limited, and each choice is generally costly. Out of need for a more cost-effective implementation of severe-accident evaluations, additional seismic review procedures have been designed; these include the *focused-scope* margins assessment and the *reduced-scope* assessment (see Section 3 and References (2, 3). Attributes of the four available alternative review methods are given below:

- Seismic Probabilistic Risk Assessment (3, 4): Broad systems modeling, detailed plant walkdown, fragility quantifications [including analyses for anchorage and (if warranted) relay chatter], and risk quantification.

- Full-Scope Seismic Margin Assessment (5): Limited systems modeling, detailed plant walkdown, anchorage evaluation to seismic margin earthquake, component screening (and identification of potentially vulnerable relays), HCLPF capacity quantifications.

- Focused-Scope Seismic Margin Assessment (Section 3): Limited systems modeling, detailed plant walkdown, anchorage evaluation to seismic margin earthquake, component screening, limited and conservative HCLPF capacity quantifications.

- Reduced-Scope Assessment (Section 3): Limited systems modeling, detailed plant walkdown, anchorage evaluation to licensing commitment, component screening, component evaluation with respect to the SSE.

The common denominator in each of these review methods is a thorough plant walkdown, the task that experts agree is the most important part of conducting an effective seismic evaluation for potential plant vulnerabilities.

If conducted with the same detail in plant walkdowns and component capacity calculations, the SPRA must involve somewhat more cost than the full-scope SMA, due to (among other things) the effort required to more-thoroughly describe plant systems and to quantify risk. SMA methodologies have been developed (5, 6, 7) with the explicit purpose of more efficiently determining plant resistance to ground motions that exceed the seismic design basis; hence, SMA may often be a preferred alternative to SPRA for conducting severe-accident evaluations.

On average, a focused-scope SMA will involve significantly lower cost than a full-scope SMA, due to selectivity (and conservative simplifications) in HCLPF calculations and more-limited relay chatter considerations. Similarly, on average, a reduced-scope assessment will involve significantly lower cost than a focused-scope SMA, due to elimination of HCLPF calculations. (The reduced scope assessment is evaluated to the licensing commitment, and is appropriate for plants where the seismic hazard is very low, such that the design basis can itself be considered a severe-accident level).

By design, this hierarchy of four review choices is interwoven with the development of procedures to select review methods appropriate for severe-accident implementation in seismic IPEs. In other words, an increased set of review

choices was developed (for seismic-IPE considerations) to accommodate the varying array of results found for seismic characterizations important to severe-accident analysis. This common development was designed to encourage efficiency and cost-effectiveness in the review process.

Roughly speaking, in the context of the seismic IPE, the reduced-scope assessment 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.

Seismic Hazard Results

Seismic hazard results (in particular, uniform hazard spectra) are also needed to make review choices for severe-accident evaluation. The NRC has expressed the position that both EPRI and LLNL hazard results should be considered in decision-making on seismic review-type selection (i.e., plant binning) (3). The proper intent is to find consistency in the two methods and resulting consistency in identification of potential vulnerabilities or dominant risk contributors. Consistency (among plants) in the two approaches, however, is (generally speaking) found only in the median hazard results. On statistical bases, one expects the median of a distribution to be one of the most stable among distribution parameters; hence, it is not surprising that the greatest [both relative (among plants) and absolute] consistency in the two approaches is found in the medians. The mean seismic hazard results for EPRI and LLNL methodologies, on the other hand, differ substantially in absolute and relative (among plants) comparisons. Because the LLNL mean results are generally substantially greater than the EPRI mean results, equal weighting of these means implies that decisions would be dominated by the LLNL results. The advantages of consistency and more equal decision weighting clearly suggest a preference for use of median seismic hazard results in developing systematic conclusions from the EPRI and LLNL hazard methodologies.

Plant Design Bases

In addition to having hazard information, it is important to also factor available plant-specific information into the seismic IPE review-development process. For

example, if one has a-priori information that suggests a plant has large seismic margin compared to another plant, then (given similar seismic threats) one would have a basis for implementing an alternate set of review procedures of a more limited nature (aimed mostly at confirming the a-priori expectation) for that plant. The individual utility is in the best position to make such judgments concerning the seismic capability of its plant, and may wish to factor its own experience and information-base into the seismic IPE decision process.

Without more detailed knowledge of a plant, however, the best characterization that is conveniently available is the plant design basis, or SSE, spectrum. For a number of reasons, the plant SSE will not correlate extremely well with plant seismic margin (as measured by the plant HCLPF capacity, for example). There are a number of reasons, however, to believe that the HCLPF capacity does at least have an underlying, although non-deterministic, relationship with SSE. The available HCLPF data supports this to some extent; for instance, a correlation coefficient of roughly 0.75 between HCLPF and SSE (in terms of peak ground acceleration) is implied by published SPRA and SMA results. (A summary of plant HCLPF data is provided in Reference (9). In obtaining our results here, we start with that data set, make a modification for one plant to account for more recent fragility data, and add results from recent studies for two additional plants).

Although we do not expect plant SSE to be a very good predictor of the HCLPF, it is valid to conclude that SSE may be a reasonable predictor of the HCLPF. More assuredly, SSE is at least a basis to bound the HCLPF. For instance, HCLPF data derived from past SMA and SPRA studies suggest an average factor of seismic margin (i.e., HCLPF/SSE) for plants of about 1.7, with values ranging from 1.2 to above 2.0. If the SSE sufficiently establishes a lower bound on plant HCLPF, and the lower bound qualifies as a severe-accident event (considering the level of seismic hazard), then there is no compelling need for detailed quantifications to determine the HCLPF precisely. For explicit consideration of severe-accident behavior, the seismic-IPE itself is undertaken (in any case) and is critical to identifying plant-specific weak-links (i.e., anchorage, bracing, interaction, etc., problems). The extra effort required to precisely quantify the HCLPF capacity, however, will likely not provide additional insights for improving plant safety, and thus may have limited merit in a severe-accident evaluation.

The currently available data appear to be adequate to support a rough bounding relationship of HCLPF with SSE. Consequently, factoring design basis into the review-method selection process is considered appropriate.

The SSE spectrum is only one aspect of the plant seismic design basis. For instance, load factors, load combinations, damping levels, etc., are all elements of design requirements which may be different from plant to plant. To the extent these factors can be included in characterizing the seismic capability of a plant, refinements in severe-accident review selection can be made. It is often impractical, however, to consider anything other than the SSE spectrum and perhaps the damping value used in the design analyses.

APPROACH FOR SELECTING SEISMIC REVIEW METHOD

Basis of Approach

Following is discussed details of the approach for selecting a review method (among the four available methods described earlier) for the seismic IPE. The approach checks for consistency (appropriateness) in design-basis hazard, i.e., probabilities of exceeding a plant's design-basis spectrum, as compared to design-basis hazard for other plants (including those designed to modern criteria).

In addressing severe-accident issues, the approach relies implicitly on well-founded conclusions derived from earthquake engineering experience. First, the seismic experience database (10) reveals that components in industrial facilities (of the type found in existing nuclear power plants) are generally seismically rugged. Second, the database also indicates that cases of poor seismic performance result primarily from basic inattention to anchorage and bracing details, and from situations where an opportunity for adverse physical interaction (e.g., valve impact) exists.

These observations suggest that the designs of existing plants are generally adequate to insure severe-accident resistance through functionality of safe shutdown components (structures and equipment) following a larger-than-design-basis earthquake. An exception may exist, however, if the perception of seismic hazard has changed (since design) to be markedly larger, for example due to the Charleston earthquake issue[1]. For such an exception, one may suspect that the threat of a design-dependent failure may be significant. Such potential design-dependent vulnerabilities, although rare, may need to be considered in the severe-accident review of existing plants that have seismic hazards that are

[1]Documentation describing the effects, on seismic hazard results, of the Charleston earthquake issue indicates that such cases are rare (11).

comparatively high relative to their designs.

Problems of anchorage, bracing, and physical interactions can be referred to as design-independent vulnerabilities, because they have the potential to exist in any plant, regardless of its design level. Such potential design-independent vulnerabilities should be considered in the severe-accident review of any plant, unless the seismic threat is negligibly low.

The NRC statement on severe-accident policy (1) declares that the existing population of U.S. plants poses no undue risk to public health and safety, and that the intent of severe-accident policy is not to re-evaluate the design bases of existing plants. For U.S. plants, therefore, the severe-accident concern should be properly focused on identifying potential vulnerabilities of the design-independent type.

Experience indicates that potential vulnerabilities, of either the design-independent or design-dependent type, are most effectively found in a thorough plant walkdown that follows recommendations for seismic margin reviews (5). No detailed calculations (for instance, of HCLPF capacities) need be conducted to find and resolve design-independent potential vulnerabilities, which are generally inexpensive to fix.

In addition, based on actions taken in light of results from past seismic studies, it is unlikely that implementation of modifications to a design-dependent condition will either be cost-effective or will achieve a substantial (absolute) reduction in plant or plant-population risk. These arguments further support the conclusion that analysis of design-dependent conditions is generally unwarranted except where the seismic hazard is comparatively high relative to the design basis.

Description of Approach

Differentiating plants based on design-basis hazard is, therefore, clearly a useful process for making decisions regarding severe-accident evaluations. The process of determining design-basis hazard (or, even more generally, the hazard associated with any spectrum) is relatively straightforward, as can be described by the following steps (see Figure A-1):

1. Obtain plant seismic design (SSE) spectra. For each plant, adjust the spectral ordinates appropriately to account for the damping used in the plant design analyses. For plants located

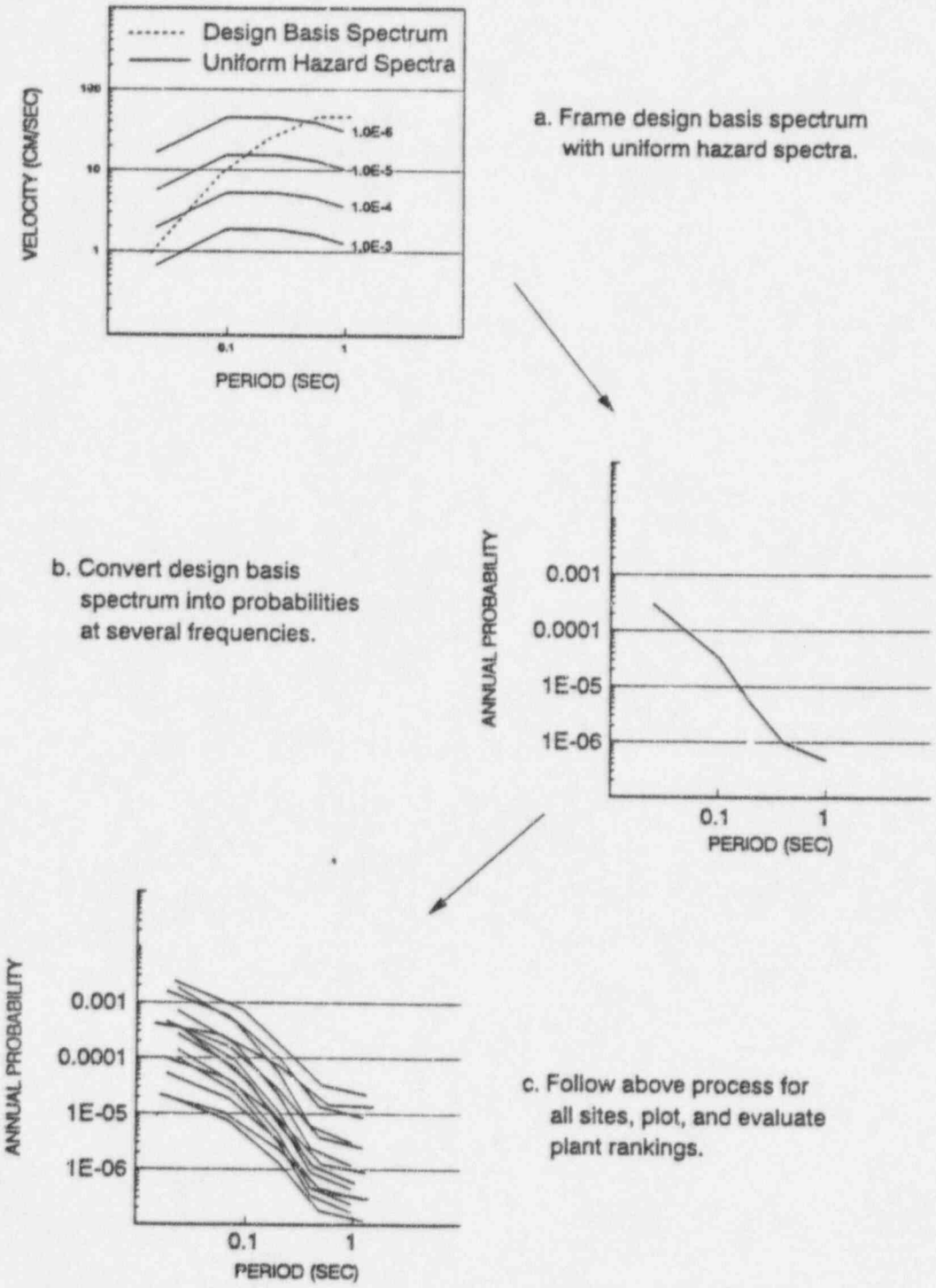

Figure A-1. Illustration of process to obtain design-basis hazard.

on both rock and soil, there may be two or more design spectra to consider. (Figure A-2 plots SSE spectra for the population of eastern U.S. nuclear power plants).

2. Obtain, for various exceedance levels, the (median[2]) plant uniform hazard spectra (5%-damped[3]), reflecting the different types of soil conditions at the power plant.

3. For each design-basis, frame (interpolate or graphically overlay) the spectrum within the array of UHS curves.

4. Select a given frequency of vibration and interpolate, between UHS for different exceedance frequencies, the value of hazard at the design-basis spectral ordinate. Repeat this interpolation procedure for several vibration frequencies, to construct a spectrum of design-basis probabilities of exceedance.

5. Perform the analysis for all design-basis spectra of interest.

Results of performing this process for eastern U.S. (EUS) nuclear power plants, using the EPRI median hazard results, are shown in Figure A-3. It is seen there that seismic design-basis hazards vary substantially both from plant to plant and from (vibration) frequency to frequency. (A similar observation is obtained if one examines design-basis hazard spectra derived using median results of the LLNL methodology).

Application of Approach

As discussed in Section 2, the design-basis hazard results guide the selection of review methods for seismic IPEs. The question each licensee must answer is: should its plant undergo a reduced-scope assessment, a focused-scope SMA, a full-scope SMA or SPRA?

In approaching this question, to simplify the description of design-basis hazard, a composite probability of design-basis exceedance is determined, considering

[2]Use of the median UHS is recommended here because it produces the greatest consistency in comparisons between EPRI and LLNL hazard results.

[3]Uniform hazard spectra are typically constructed for a 5% damping level. When making comparisons among plants, it is important to use a consistent reference damping level, regardless of the damping used in plant design analyses. In this manner, the reference damping level is associated with realistic seismic response, whereas the design damping level is associated with plant seismic capacity. A lower design damping (resulting in a higher capacity for a given design-spectrum shape), for instance, should properly produce a lower probability of exceedance (as measured by hazard results based on the reference response damping.)

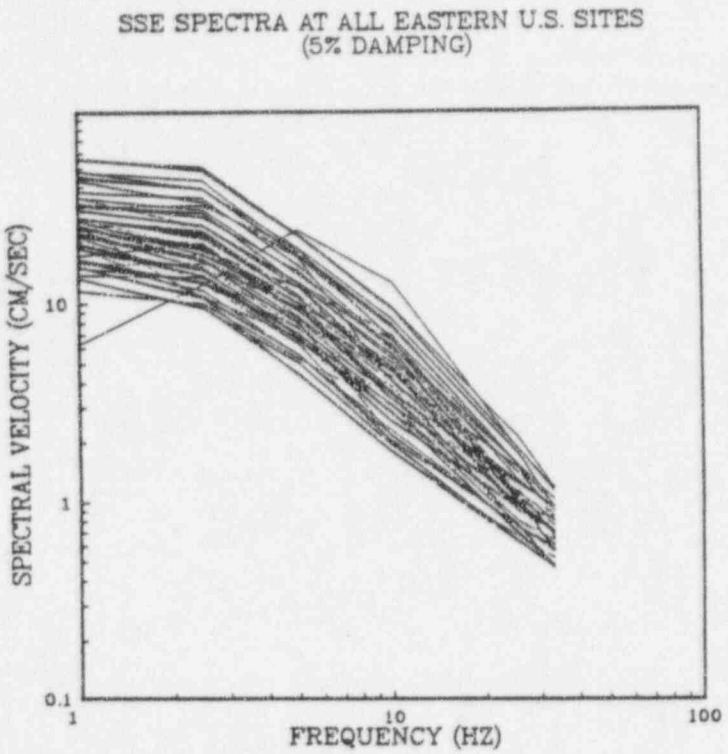

Figure A-2. Design spectra for the population of EUS nuclear power plants.

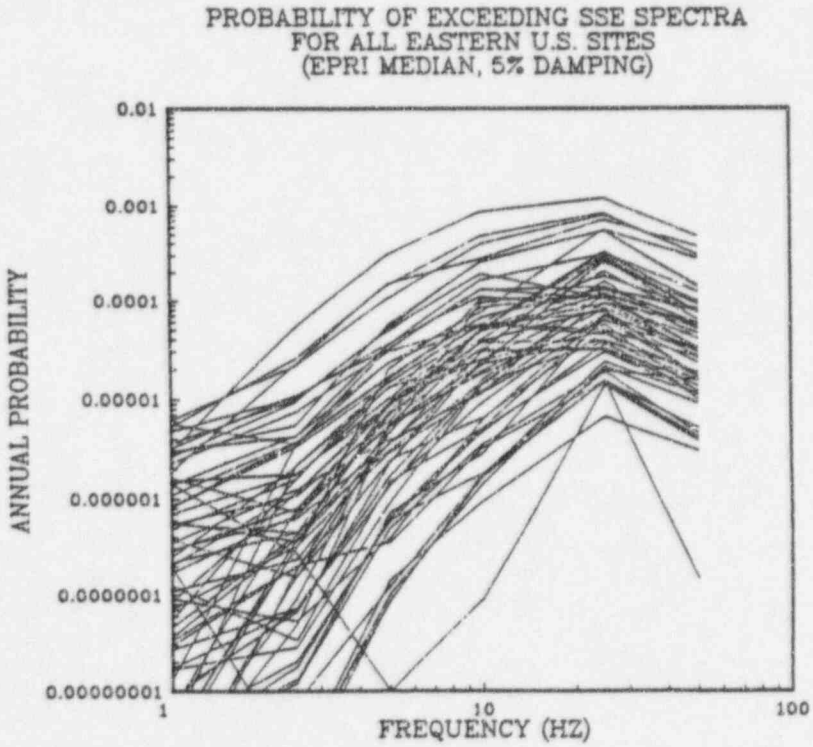

Figure A-3. Design-basis hazards for EUS nuclear power plants, based on EPRI median results. (For simplicity, 5\% damping is used for all SSE spectra).

results at multiple (vibration) frequencies. The method to use for obtaining this composite design-basis hazard measure is that recommended by the NRC (3), where 2/7 weight is given to the design-basis hazard at each of the three frequencies: 2.5 Hz, 5.0 Hz, and 10.0 Hz; and 1/7 weight is given to the design-basis hazard at the peak ground acceleration (PGA). By this process, one obtains a composite measure, (separately) from EPRI median and from LLNL median hazard results, for each EUS plant considered.

Next, these composite design-basis hazards are ordered, from highest to lowest, and the results plotted separately for EPRI and LLNL hazards analyses, as have been shown in Figures 2-1 and 2-2. Using these results of ordered design-basis hazard, one can differentiate plants based on the level of composite design-basis hazard, as discussed in Section 2.

Assisted by the information in Section 2 and, in consideration of NRC's final revisions to Supplement 4 of Generic Letter 88-20, licensees can select appropriate review methods for seismic IPEs of their plants. The decision on an appropriate review method will depend on the specific situation that exists for any given plant. For instance, some plant-specific considerations that must be factored into the seismic-IPE review selection process include:

- Whether or not an SMA or SPRA already exists for the plant;
- Whether or not seismic hazard results have been obtained for the plant site;
- Whether or not the plant must satisfy A-46 criteria; and
- To what level of detail, if any, the licensee intends to conduct cost-benefit analyses for potential modifications.

If a deterministic review method is selected, the licensee has the option to justify a lower review scope and/or lower review level than that recommended by the NRC, based on consideration of Section 2 or based on more site-specific information or evaluations. Alternatively, the licensee may wish to exercise its option to perform an SPRA review (e.g., if it intends to conduct explicit cost-benefit analyses to provide quantitative support for decisions), even though Section 2 of this document or the NRC may recommend a deterministic review. Such decisions are best made by licensees on a plant-by-plant basis. In formulating these decisions, the licensee may wish to keep in mind that the guidances in this document and in NUREG-1407 on review method selection provide general recommendations based on general analyses; hence, the licensee is given latitude

in selecting a review method, in consideration of its preferences and of more plant-specific information it may have or may develop.

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Appendix B

RECOMMENDED PROCEDURES TO ADDRESS HIGH-FREQUENCY GROUND MOTIONS IN SEISMIC MARGIN ASSESSMENT FOR SEVERE ACCIDENT POLICY RESOLUTION

by

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Section 1

INTRODUCTION

The purpose of this appendix is to provide recommendations for addressing high-frequency seismic motions in a Seismic Margin Assessment (SMA) for Severe Accident Policy resolution. A research program to address high-frequency seismic motions was performed by the Electric Power Research Institute (EPRI) to provide the basis for the recommendations (1). An analytical investigation was conducted, and a series of tests are planned by EPRI in the near future.

The recommendations for addressing high-frequency motions are made in the context that the SMA procedures (2) are acceptable to both industry and the NRC for seismic capacity reviews of nuclear power plants and that only minor changes are needed for the effects of high frequencies. It is recommended that the SMA screening guidelines for structures and equipment be keyed to response spectrum parameters, not to the peak ground acceleration (PGA). Also, it is recommended that the input ground response spectrum to be used in a SMA be modified in the high-frequency region to reflect the lack of damageability from motions in this range. Recommendations for electrical functionality failure modes (e.g., relay chatter) for relays and other electrical components, which may be sensitive to motions with frequencies greater than about 16 hz, are also provided.

It is important to recognize that the recommendations made here for structures and components reflect the ability of the supporting structural elements to resist seismic motions in a ductile fashion. Although some elements (e.g., welds) are brittle relative to low-frequency damaging seismic motions, they have some ductile capacity to resist the small displacements associated with high-frequency motions. Separate recommendations are also given for electrical functionality failure modes, which maybe acceleration sensitive and do not have ductile capability.

Section 2 of this appendix gives the background on the high-frequency seismic motion issue. In Section 3 recommendations are given for modifying the SMA procedures to include the effects of high-frequency seismic loads. Finally, Section 4 lists the references used in this appendix.

Section 2

BACKGROUND ON HIGH-FREQUENCY SEISMIC MOTIONS

The results of recent seismic hazard analyses conducted for the eastern U.S. (EUS) indicate that uniform hazard spectra (UHS) have relatively large high-frequency spectral acceleration content above approximately 10 Hz. This is in contrast to typical plant design or seismic margins-type input, such as the NRC Regulatory Guide (R.G.) 1.60 or NUREG/CR-0098 response spectra. However, these same UHS have significantly lower spectral acceleration content at the low-frequency end of the response spectrum (i.e., between about 1 and 10 Hz), which indicates to the seismic engineering community that these motions are less damaging than traditional design or evaluation-type input. Figure B-1 shows the relationship between the median NUREG/CR-0098 response spectrum anchored to 0.3 g peak ground acceleration (pga) and an example UHS, with a 10,000 year return period (at the 0.85 fractile), from a hazard analysis using the EPRI methodology for a EUS nuclear power plant site.

In the analysis and design of nuclear power plant structures it is widely felt that to be damaging, earthquakes must be rich in spectral content in the 1 to 4 Hz frequency range. Because most nuclear power plant structures and equipment have fundamental frequencies significantly above 1 Hz, strong amplified motions below 1 Hz caused by soft soil conditions, although potentially damaging to flexible structures such as high-rise buildings, do not affect typical nuclear power plant sites. Thus for stiff nuclear power plant structures at typical soil or rock sites it is necessary for energy to be present in the 1 to 4 Hz frequency range for earthquakes to be potentially damaging. It is believed that the damaging frequency range for equipment probably is slightly higher, but less than 10 Hz. A possible exception is anchorage of equipment at high ground motion levels, but for this case spectral displacement, not acceleration, is more important in defining damage potential. In Figure B-1 the NUREG/CR-0098 spectral shape has damage-potential characteristics but the example UHS does not.

It has also been observed that damage correlates well with elastic spectral acceleration in the 1 to 4 Hz range. Thus, it is likely that it is an "effective" frequency in this range as opposed to the fundamental frequency which is important

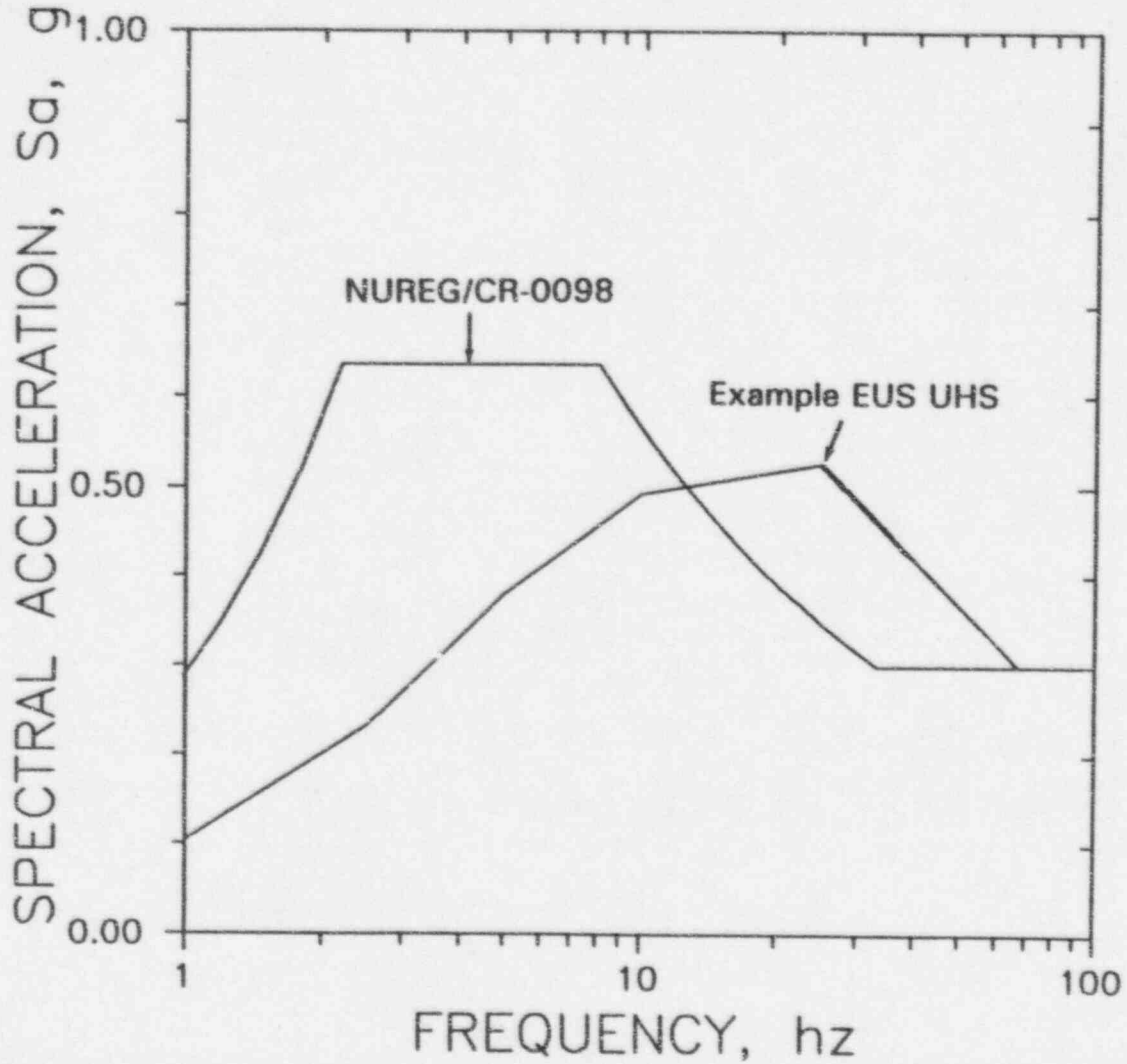

Figure B-1. Comparison of example uniform hazard response spectrum with NUREG/CR-0098 response spectrum (5 percent damped).

to significant damage (1). These results support the conclusion that it is the low-frequency content of earthquake records which causes damage.

In a recent detailed study of the Diablo Canyon Turbine Building, conducted for the Diablo Canyon seismic probabilistic risk assessment (PRA), 25 earthquake records were input to a nonlinear building model where the model material properties were varied in a probabilistically consistent manner (3). Several hundred deterministic analyses were conducted in this study, which spanned all realistic combinations of structural and ground motion parameters. It was found that for a ground motion to be damaging it must have broad frequency content. Specifically, high spectral accelerations were required at the elastic frequencies for those dynamic modes corresponding to the relatively weak building shear walls (i.e., approximately 8 Hz) to initiate substantial inelastic drift. In addition, high spectral acceleration content also was required in the approximately 2 Hz range in order to drive the shear walls to drift levels associated with the onset of severe distress (i.e., significant damage).

One example of a high-frequency earthquake which occurred near a nuclear facility was the Leroy earthquake (magnitude M_L 5.0) which took place January 31, 1986 about 11 miles from the Perry Nuclear Power Plant (4). At the time of the event Cleveland Electric Illuminating Company was within days of core load and receipt of their 5 percent power license for Perry. The staff observed no indications of damage to systems in operation at the time, which was confirmed by follow-up inspections. However, after the earthquake the analysis of the motion records indicated that the Operating Basis Earthquake (OBE) ground response spectrum had been exceeded at frequencies above 10 Hz, and the Safe Shutdown Earthquake (SSE) response spectrum had been exceeded above 15 Hz. The maximum acceleration of the ground motion at the foundation level was 0.18 g with a corresponding peak 5 percent damped spectral acceleration of about 0.8 g; however, the peak ground velocity was only 0.9 inches per second, and the maximum ground displacement was only 0.06 inches. The latter two values indicate low energy content in the 1 to 4 Hz frequency range; thus, it is not surprising that no damage occurred. In addition, the cumulative absolute velocity (CAV) value for this event was only 0.08 g-sec which is much lower than the OBE exceedance criterion limit of 0.30 g-sec (4).

Experience with high-frequency motions from other environments indicate that they are much less damaging than their low-frequency counterparts. In the development of a criterion for determining the exceedance of the OBE, earthquake experience

concerning the damaging characteristics of high-frequency motions was collected and documented (4). Based on a review of ground motions caused by conventional high explosive blasts, and their effect on structures, it was concluded that no damage will occur to engineered structures and equipment for short-duration ground motions with spectral accelerations below the envelope of the OBE response spectrum and a threshold cracking spectrum (established from blast data) as shown in Figure B-2. However, it has not been established whether these conclusions can be extrapolated to longer duration large-magnitude high-frequency events. The conclusions that damage is not caused by high-frequency motions is also supported by fragility data for ductile equipment, performance of equipment under industrial environment vibrations and code requirements for high-frequency loading.

Based on past experience, both from earthquakes and other physical phenomena which produce high-frequency motions, it is believed that high-frequency input, which is a predominant part of UHS, as shown in Figure B-1, will not be damaging to ductile structures and equipment. However, there is still a potential concern that these motions may be an issue for acceleration sensitive equipment and relays, contactors, motor starters and switches. For this category of components the effects of high-frequency motions should be addressed only if it is determined from a systems perspective that functional failure (e.g., relay chatter) during strong motion is detrimental or equally that the safety function affected can not be confidently recovered after an earthquake.

In order to evaluate the capacity of relays, high-frequency input at the floor level is required. The difficulty in analyzing acceleration-sensitive components for high-frequency input is in the development of realistic floor response spectra. The number of finite elements and mass nodes used in traditional dynamic analysis of nuclear power plant structures is generally not large enough to properly model the effects of high-frequency input. On one hand, using the current models, the high-frequency motions will be filtered out giving the incorrect impression that high-frequency response at the various floor levels does not exist. On the other hand, if models are constructed which properly pass the high-frequency motions through the structure, the analysis cost would be prohibitive. Based on past experience high-frequency motions do propagate through structures. The earthquake near the Perry plant demonstrated this to be true. However, there is evidence to suggest that high-frequency motions are not significantly amplified, which allows a simplified approach for addressing these motions as recommended in the following section.

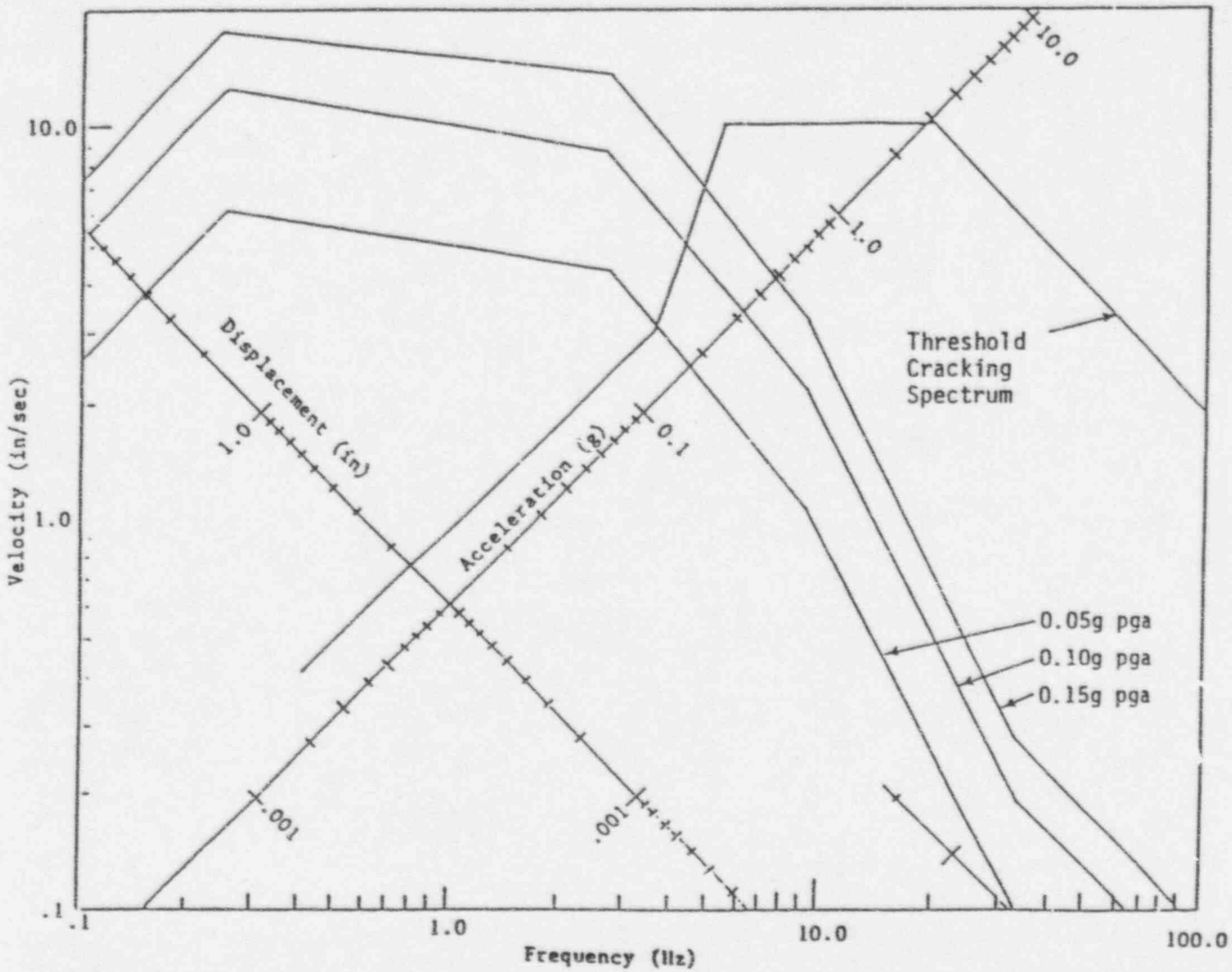

Figure B-2. Comparison of R.G. 1.60 ground response spectrum to the threshold cracking spectrum developed from ground motion due to blast (3).

It is believed that high-frequency ground motions which have dynamic characteristics consistent with the EUS UHS in the range of about 10,000 year return period will be much less damaging than low-frequency earthquakes with the same peak spectral acceleration. These latter ground motions traditionally have been the basis for seismic analysis and design (e.g., Olympia, Taft and El Centro No. 5).

Section 3

RECOMMENDED PROCEDURES

Conservative recommendations for incorporating the effects of high-frequency seismic motions in SMA are provided in this section. These recommendations reflect the current data and engineering understanding of high-frequency ground motions that might occur at nuclear power plant sites in the central and eastern United States.

References (2) and (5) describe the procedure for performing a seismic margin assessment and are incorporated here by reference. The following modifications for consideration of high-frequency seismic motions are recommended to the SMA procedures:

- The seismic capacity screening guidelines, which are keyed in References (2) and (5) to peak free-field ground acceleration values (i.e., 0.3 g and 0.5 g), are referenced to response spectral parameters: i.e., spectral acceleration (S_a) and spectral velocity (S_v).
- Input ground response spectra used in a SMA should be modified at high frequencies to reflect the lack of damageability to safety-related nuclear power plant components.
- For relays that are sensitive to input motions above 16 hz special considerations are recommended.

A discussion of the three recommended modifications and their supporting bases are given in the following subsections.

RESPONSE SPECTRUM PARAMETER LIMITS FOR SEISMIC CAPACITY SCREENING GUIDELINES

The seismic capacity screening guidelines used in SMA are keyed to peak free-field ground accelerations (2 and 5). Three categories are provided: when the PGA corresponds to less than 0.3 g, 0.3 g to 0.5 g and greater than 0.5 g. For each of the three categories, requirements are given for different structural and equipment classes. For example if the PGA is less than 0.3 g, concrete containments do not need to be evaluated, but masonry walls do. Detailed screening tables are given in Reference (2) which extend the tables originally developed in Reference (5).

Consistent with the level of conservatism provided in the screening guidelines in References (2) and (5), Figure B-3 shows the corresponding spectral acceleration and spectral velocity limits which should be used with the SMA screening guidelines for earthquakes with high-frequency motions. The lower curve which consists of a S_a limit of 0.8 g and a S_v limit of 20 in/sec corresponds to the 0.3 g PGA value. Similarly, the top curve which has a 1.2 g S_a limit and a S_v limit of 30 in/sec corresponds to the 0.5 g PGA value.

When making a comparison using the screening guideline limits, the amplified portion of the input ground response spectrum in the high-frequency region should be compared to the S_a limits. Note that the check in the high-frequency region only is made with the spectral acceleration limits, not with the PGA values. Since damage is ultimately related to strain, or equivalently displacement, the PGA check is not required.

The basis for the response spectrum limits shown in Figure B-3 has its roots in the original development of the SMA screening guidelines, where the first two authors of this appendix were principal contributors (5). At that time the Seismic Qualification Utility Group (SQUG) had formed in response to the NRC issue designated as: "Seismic Qualification of Equipment in Operating Plants," which is also referred to as Unresolved Safety Issue (USI) A-46.

As SQUG began evaluating the earthquake experience data this information also was available to the authors (referred to as the Expert Panel) of Reference (5). Ultimately, the Senior Seismic Review and Advisory Panel (SSRAP) endorsed the Reference Spectrum given in Reference (6) (which is controlled by S_a equal to 1.2 g) and the Bounding Spectrum (which is controlled by S_a equal to 0.8 g). The Bounding Spectrum, which is a factor of 1.5 below the Reference Spectrum, is currently being used by SQUG in the evaluation of safety-related equipment in existing nuclear power plants. Note that the 1.5 factor reflects conservatism in using the experience data base and for the possibility that floor response spectra in nuclear power plants might be amplified more than motions in the data base buildings (6).

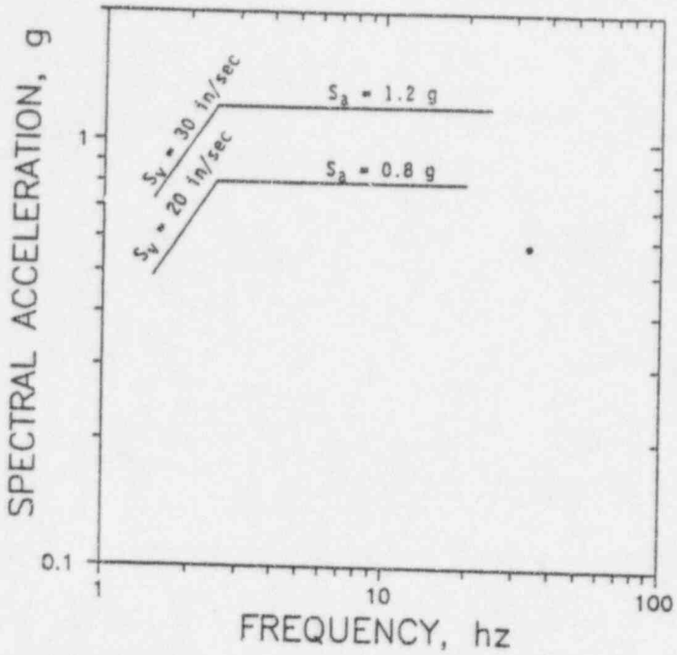

Figure B-3. Spectral parameter limits for SMA seismic capacity screening guidelines.

The equipment covered by the SMA screening guidelines essentially corresponds to the SQUG equipment. Structures are included in the SMA review which are not included in the A-46 program. However, nuclear power plant structures are rugged, and the guidance given in the SMA screening tables conservatively reflect their capacities. Moreover, a walkdown will be conducted in both SQUG and SMA reviews and there are important caveats and exclusions that must be followed in order to use the screening guidelines, both of which ensure conservative assessments.

The 1.2 g S_a limit used in the SMA screening guidelines is used to designate capacities for equipment in the 0.8 g S_a to 1.2 g S_a category and for the greater than 1.2 g S_a category. Although the SQUG guidelines do not include multiple categories, the SMA guidelines for the two ranges are consistent with the earthquake experience data base, which corresponds to the higher Reference Spectrum. Thus, the upper curve in Figure B-3 reflects the philosophy of the NRC Expert Panel and is appropriate for delineating the two higher SMA screening categories.

At the lower frequency end of the spectral capacity curves shown in Figure B-3 the spectral acceleration limits change to spectral velocity limits at about 2.5 hz,

which again correspond directly with the Bounding and Reference Spectra. The lower frequency limits of the spectra in Figure B-3 are about 1.5 hz, which is believed to be at about the practical frequency limit of nuclear power plant structures and equipment on soil sites.

The experience data base, upon which the SQUG program derives nuclear power plant equipment capacities, is tied to the earthquake characteristics at the data base sites. Thus, the Bounding (and Reference) Spectrum roll off at frequencies above 7.5 hz as shown by the spectral curve in Figure B-4. However, at the upper frequency end of the limit spectra shown in Figure B-3 the spectral acceleration values at frequencies higher than 7.5 hz are projected flat at the S_a limits of 0.8 g and 1.2 g corresponding to the two curves. The limitation on the Reference and Bounding Spectra is not the equipment capacities per se, but rather the frequency characteristics of the data base earthquakes, which are western North American events that may be less rich in high-frequency content.

Extending the spectral acceleration capacity limits at constant levels for frequencies beyond 7.5 hz is justified based on an understanding of the potential failure characteristics of structures and equipment at nuclear power plants. For critical components (i.e., safety-related components that have structural properties in the range that require seismic evaluation) the fundamental frequencies are generally less than about 10 hz with a few classes of equipment with fundamental frequencies up around 15 hz. Some classes of equipment have higher fundamental frequencies, but their elastic strength is also relatively high (e.g., check valves and well-anchored heat exchangers). Thus for most components the high-frequency earthquake motions are at frequencies above their fundamental frequencies.

As discussed in Reference (1) most components in nuclear power plants are very strong. If components have high fundamental frequencies (i.e., relatively stiff compared to their mass) they also have high strength, since stiffness and strength are correlated. In addition to being strong most high-frequency components in nuclear plants also have ample capacity to resist the small displacements associated with high-frequency motions. Examples include the casing on a vertical pump motor and the exterior shell on a small heat exchanger. These types of components are also ductile, and small displacements associate with high-frequency motions can be safely accommodated if the yield capacity is exceeded.

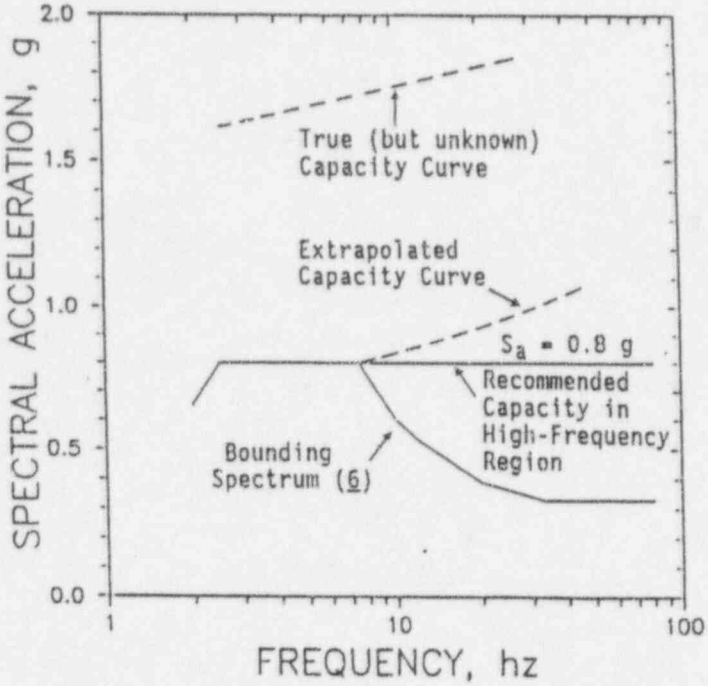

Figure B-4. Extension of SQUG bounding spectrum into high-frequency region.

In Reference (1) it is concluded that the potentially limiting case for high frequency components is welded equipment anchorage where the welds are sized exactly to met the seismic demand. Note that anchorage is not included in the SMA screening tables and must be considered separately in addition to the screening table decisions for the equipment per se. Thus, the extension of the spectral limits into the high-frequency region is for functionality and structural issues covered in the screening tables, not for equipment anchorage.

In general, the capacity of a component is either controlled by an elastic displacement limit (e.g., the gap between a fan and its housing) or a strain limit where the component goes into the inelastic range (e.g., bending of a steel support). At the top of Figure B-4 an idealized capacity curve is shown above the recommended capacity spectrum, which corresponds to the 0.8 g S_a value. It is assumed here that the fundamental resonance frequency is to the left of the left end of the true capacity curve. This capacity spectrum increases as the frequency of input increases. Figure B-5 shows example real fragility curves for a pressure control valve from Reference (7). Note that the capacity in Figure B-5 increases with frequency in a manner similar to the idealized curve at the top in Figure B-4.

The results of the high-frequency analytical study reported in Reference (1) demonstrate that the capacity beyond yield also increases at higher frequencies. In general, if components are designed to yield at a given earthquake level the ratio of the earthquake input at failure to the earthquake input at yield increases as the component frequency is increased. This is observed in detailed nonlinear time history analyses and also is explained by pseudo linear-elastic models which fit the time history results. The principal reason for this increased capacity is due to the shift in frequency from the elastic case to the effective frequency at failure (1). A greater shift occurs as the component frequency is increased, which leads to a greater margin from yield to failure.

It is expected that the capacity beyond 7.5 hz should increase as indicated in Figure B-4. For either an elastic or inelastic type of failure model the S_a capacity increases with frequency, and the assumption that the capacity is constant (i.e., flat as shown in Figure B-3) for frequencies above 7.5 hz is conservative.

There are some limitations to these arguments. The principal issue is when the fundamental frequency of the component under consideration occurs at a frequency higher than the upper bound frequency at which the capacity has been established from experience data. For this case there may be a dip in the capacity versus frequency curve in the vicinity of the fundamental frequency, and the shape of the capacity curve would be different than shown in Figure B-5. However, it is unlikely (anchorage aside) that high-frequency components exist that either do not have high yield capacities, or which do not have high displacement capability between yield and failure which can accommodate the small displacements associated with high-frequency earthquakes. Experience from other environments supports the conservative recommendations. Based on the work in References (1) to (4), strong evidence is found that high-frequency seismic motions will not be damaging to nuclear power plant components. Again these results emphasize that it is the seismic motions in the low frequency range (i.e., approximately 1 to 4 hz) that potentially can cause damage to nuclear power plant components.

INPUT RESPONSE SPECTRUM REDUCTION AT HIGH-FREQUENCIES

Reference (1) provides a procedure and technical basis for reducing ground-level response spectra at high frequencies for the purposes of analyzing components which have not been screened out and for equipment anchorage. For acceleration sensitive components such as relays the unmodified ground response spectra must be used in the evaluation. Consideration of electrical functionality failures is

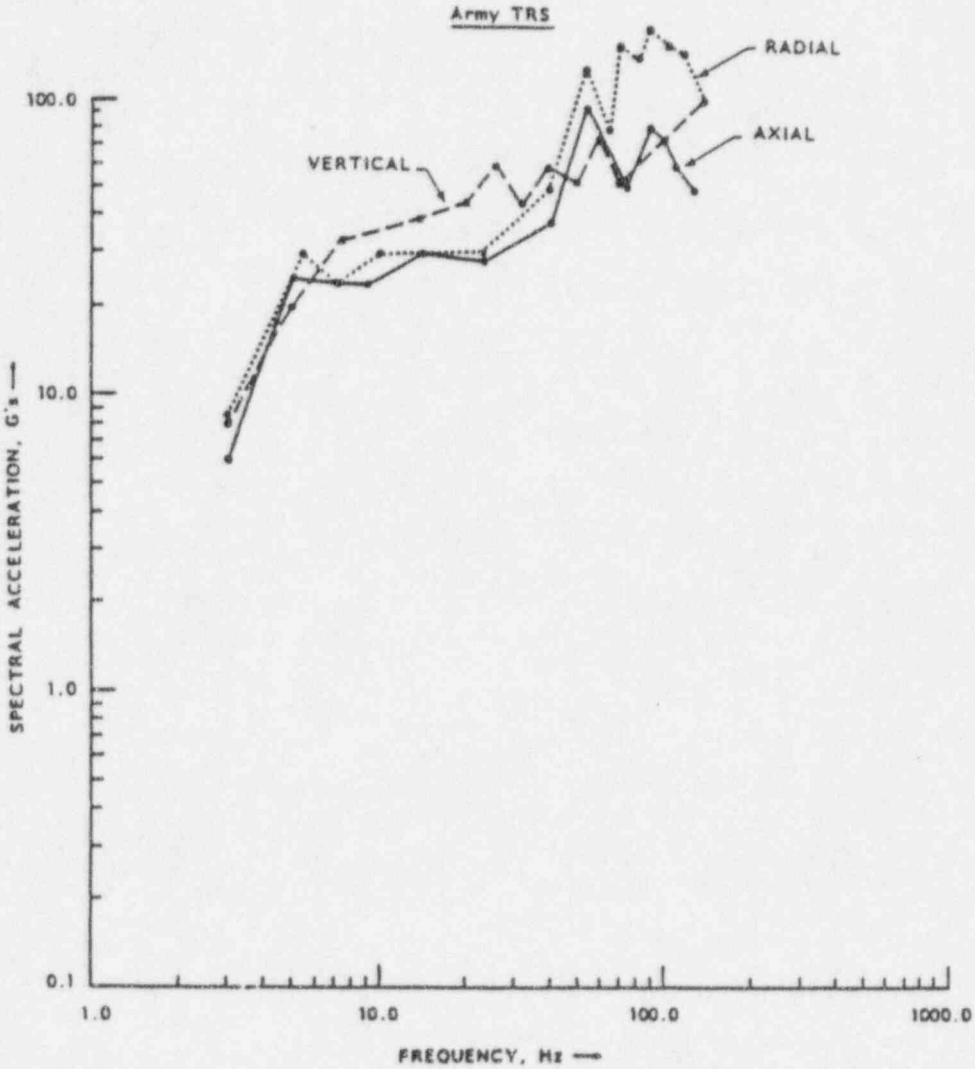

Figure B-5. 2 percent damped pressure control valve capacity spectra (Z).

discussed the last section of this appendix. The reductions for high-frequency should be applied in addition to the reductions recommended for the incoherence of ground motion recommended in Reference (2).

In resolving severe accident issues a probabilistically-based review level earthquake (RLE) is recommended to be used as discussed in Appendix E. The RLE for a EUS site generally will be rich in high-frequency content. Because of the increased capacity for ductile components as the component fundamental frequency increases the RLE can be reduced at high frequencies. This follows in order to be risk consistent with the demand-to-capacity ratios for low-frequency components, where the ultimate capacities are essentially at the yield level. It is recommended that the RLE ground response spectrum be modified to a review level ground motion (RLGM) response spectrum following the procedures given in Reference (1) as summarized below.

Note that in screening components the RLE should be compared to the screening spectral limits shown in Figure B-3. This avoids double counting the benefits of ductility that are used in the arguments for extending the 0.8 g and 1.2 g spectral limits into the high-frequency region. The same arguments are also used for decreasing the RLE, as presented in Reference (1). However, once the screening step is completed components which are not screened out should be analyzed using the modified RLGM spectrum.

Figure B-6 shows an example RLE for a EUS site and two RLGM spectra. The higher RLGM response spectrum should be used to generate in-structure response spectra for analysis of equipment up high in a building, and the lower RLGM can be used to analysis equipment which is supported at the ground level. The higher RLGM spectrum is required because of amplification of the ground motion at higher elevations.

It is recommended that spectral reduction factors used to reduce a RLE response spectrum to obtain a RLGM response spectrum for analyzing high-frequency components in a SMA be obtained using the simplified sliding model presented in Reference (1). It is recommended that the fraction of mass at the model base and the coefficient of friction both be set at zero. Figure B-7 shows the model for component sliding from Reference (1). The following guidance is given when applying these recommendations:

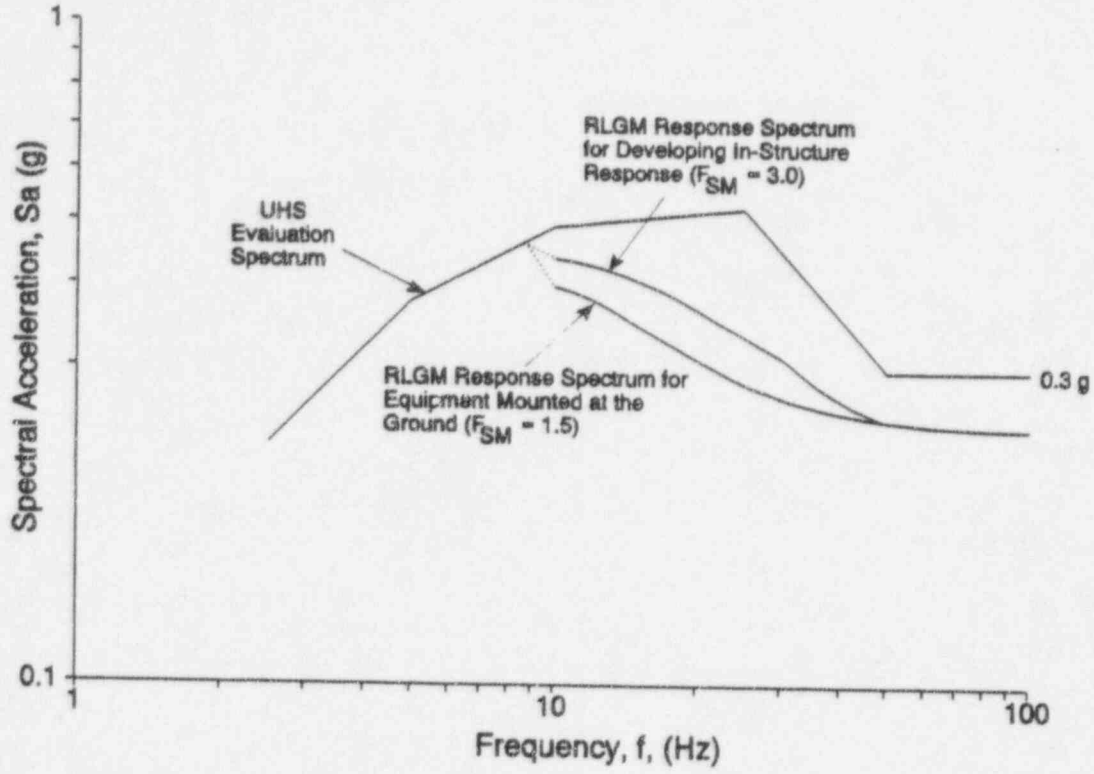

Figure B-6. Example Reduced Response Spectra for UHS Anchored to 0.3 g PGA.

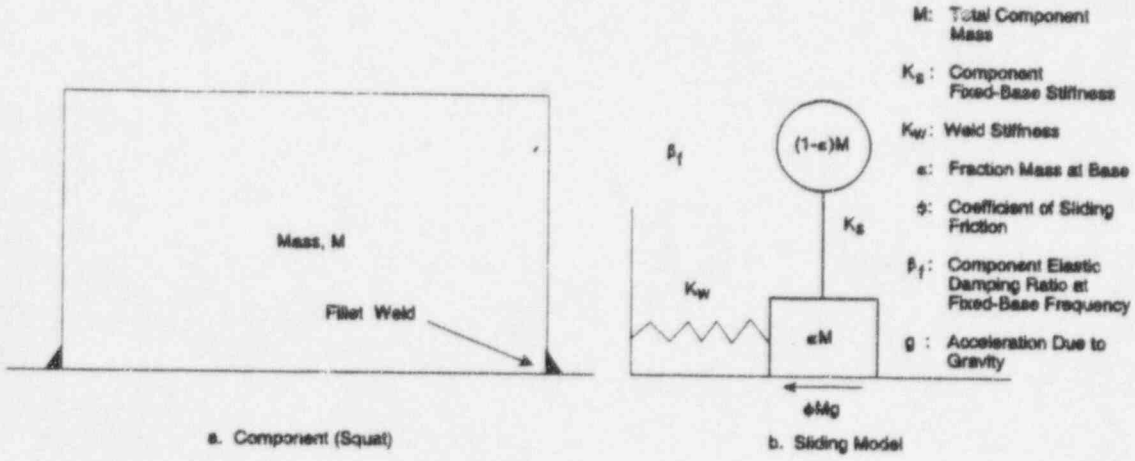

Figure B-7. Model for Component Sliding

1. The reduction should be performed for frequencies above 10 Hz, and the reduced response spectrum should be reconnected to the RLE evaluation spectrum at 8 Hz.

2. The reduced response spectrum should not be reduced below a response spectrum value equal to the peak evaluation spectral acceleration at 10 percent damping divided by 1.6 unless additional cases are considered as discussed in Reference (1).

3. The RLGM to be used to generate in-structure response spectra should be based on a safety factor, F_{SM}, of 3.0. This is equivalent to using a F_{SM} of 1.0, but based on a response spectrum equal to three times the RLE ground response spectrum. The resulting reduction factors are applied to the original ground response spectrum.

4. For equipment mounted at grade the evaluation ground response spectrum with a F_{SM} value of 1.5 can be used to determine the reduced response spectrum.

The safety factors given above include consideration of amplification of typical high-frequency EUS ground motions up in nuclear power plant structures as well as the safety margin between a HCLPF capacity and the median yield capacity.

A permissible anchorage distortion of 0.01 inch should be used to develop the reduction factors. This corresponds to the ultimate displacement capacity of a 3/16-inch weld. It is assumed in selecting a 3/16-inch weld that there may be 1/8-inch welds used to attach the sides of electrical cabinets to embedded floor plates. However, there are other sources of flexibility in this type of component that are equivalent to the models used in the high-frequency study which assumed 3/16-inch welds. Thus, the use of a 3/16-inch weld size also represents these cases. The authors believe that a permissible anchorage distortion of 0.01 inch is conservative and that all nuclear power plant components have a least the minimum amount of displacement capacity.

For plants that can justify greater permissible anchorage distortions, it may be possible to base the spectrum reduction factors on these larger permissible anchorage distortions. However, the capacity of electrical cabinets anchored with nominal welds in the plant must be considered in justifying larger permissible distortions.

The analyst should realize when performing a SMA evaluation using reduced response spectra as input that the portion of the reduced response spectrum above 8 Hz takes partial credit for ductile capacity (0.01 inch nonlinear distortion). In performing a SMA the inelastic energy absorption factor recommended in Reference

(2) also may be used, in general, for high frequency components since these factors are based on the characteristics of WUS ground motions. Note that Reference (2) recommends $F_\mu = 1.0$ for welds and other small distortion capability anchorages.

Figure B-6 shows example reduced ground response spectra for a UHS anchored to 0.3 g pga based on the sliding model following the above recommendations. The two spectra are based on F_{SM} values of 1.5 and 3.0 and would be used for an SMA evaluation for equipment at the ground and for developing in-structure response spectra, respectively.

The shape of the reduced response spectra in Figure B-6 above 10 hz are very similar to the type of ground response spectra used in the past to design and evaluate nuclear power plants (e.g., R.G. 1.60 and NUREG/CR-0098 response spectral shapes). Thus, the UHS ground response spectra currently being obtained for Eastern U.S. sites have similar characteristics to traditional design spectra when the UHS are modified to have consistent safety margins.

An intuitive justification for the recommended reductions is provided by examining displacement capacities of equipment anchorages, which are the critical structural links. Figures B-8 through B-12 show force-deflection curves for single-direction loading for fillet welds, wedge-type expansion anchors and cast-in-place concrete inserts. As can be seen from these figures welds have the minimum displacement capacity.

In Fig. B-8 the fillet weld data in the transverse direction (i.e., θ equal to 90°) has a minimum displacement capacity. However, compared to the longitudinal direction weld (i.e., θ equal to 0°), which has a high ductility limit but lower strength capacity, the transverse direction weld strength is approximately 40 percent stronger. This extra strength in the transverse direction offsets its lack of ductility. The recommended reduction factors are based on transverse loading for the smallest practical weld size, which is conservative.

Note that typical high-frequency motions are small. For example, a 1 g acceleration at 10 hz is only about a 0.1 inch displacement. Ductility capacity is relative, depending on the frequency of the input motion. For low-frequency damaging motions the displacements are large and the inelastic response of welds as shown in Figure B-8 does not provide ductile capacity. In contrast, at higher frequencies where the displacements are small, the ductility characteristics of

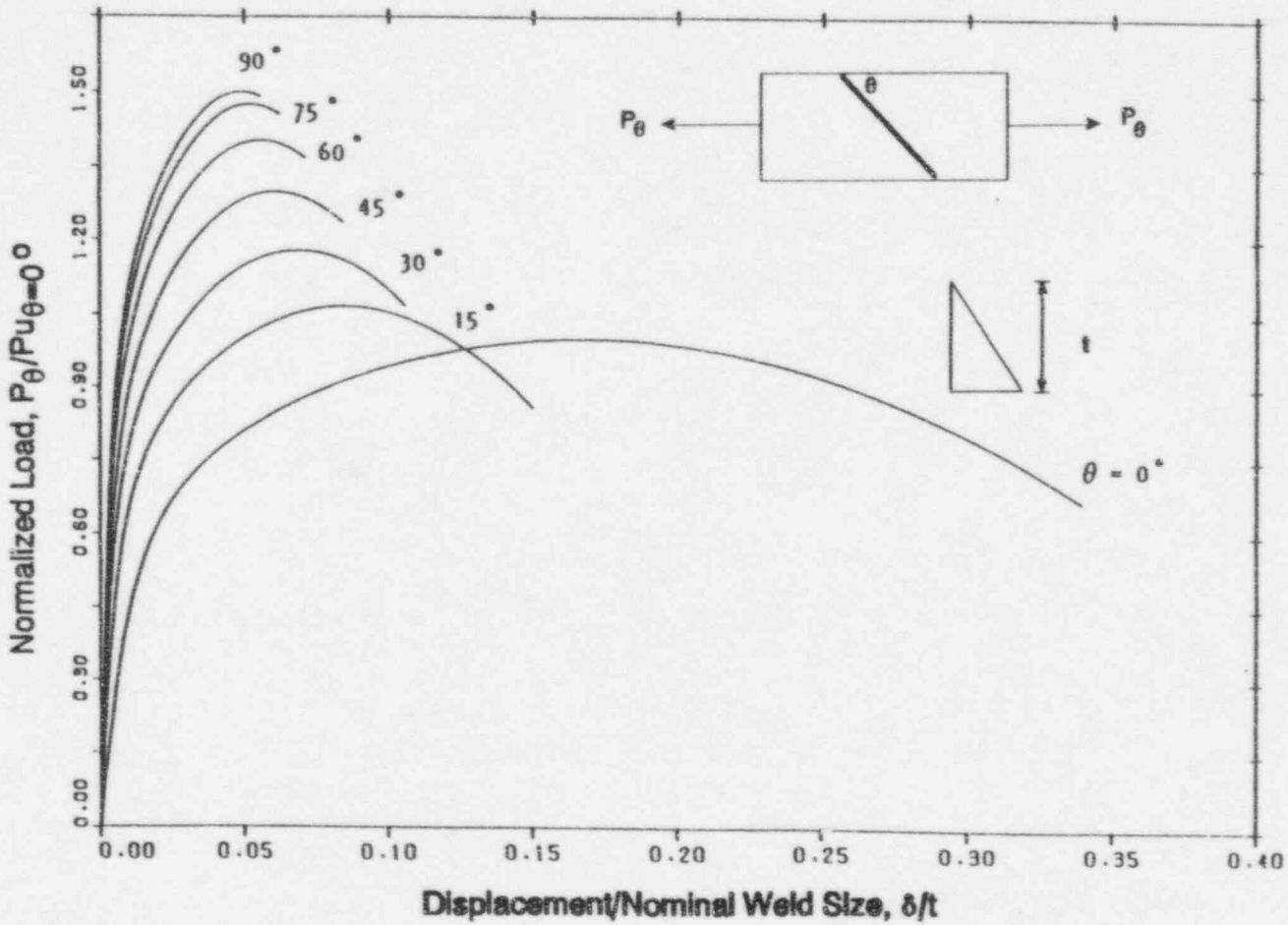

Figure B-8. Normalized Weld Curves (8)

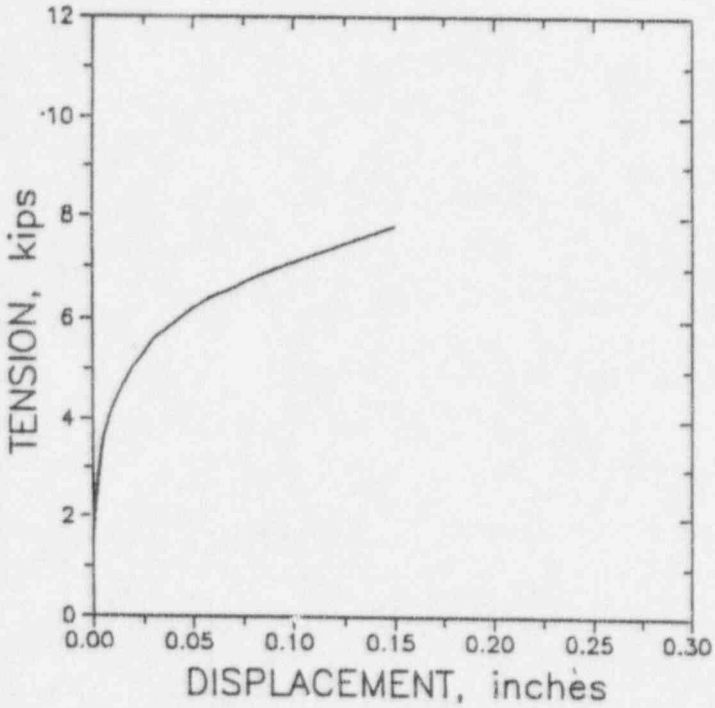

Figure B-9. Example force-displacement curve for 3/4 in. wedge type anchor bolt in tension, f'_c=3700 psi (concrete failure mode) (9).

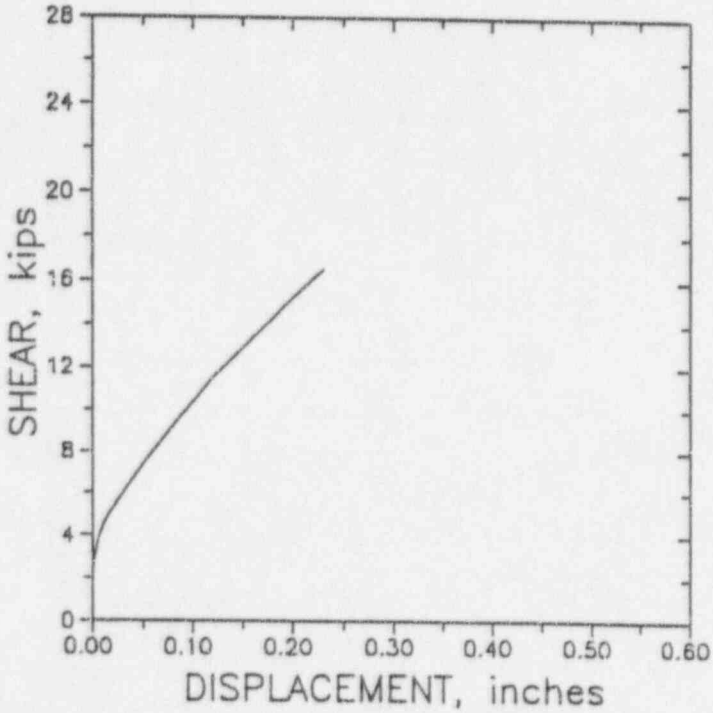

Figure B-10. Example force-displacement curve for 3/4 in. wedge type anchor bolt in shear, f'_c=3700 psi (shear failure through threads) (9).

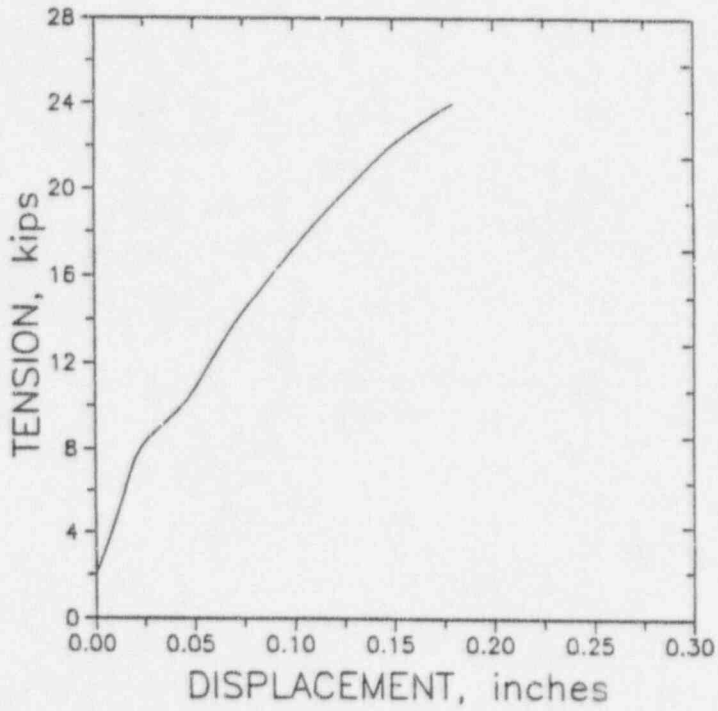

Figure B-11. Example force-displacement curve for 1 in. Richmond insert in tension, f'_c=2850 psi (10).

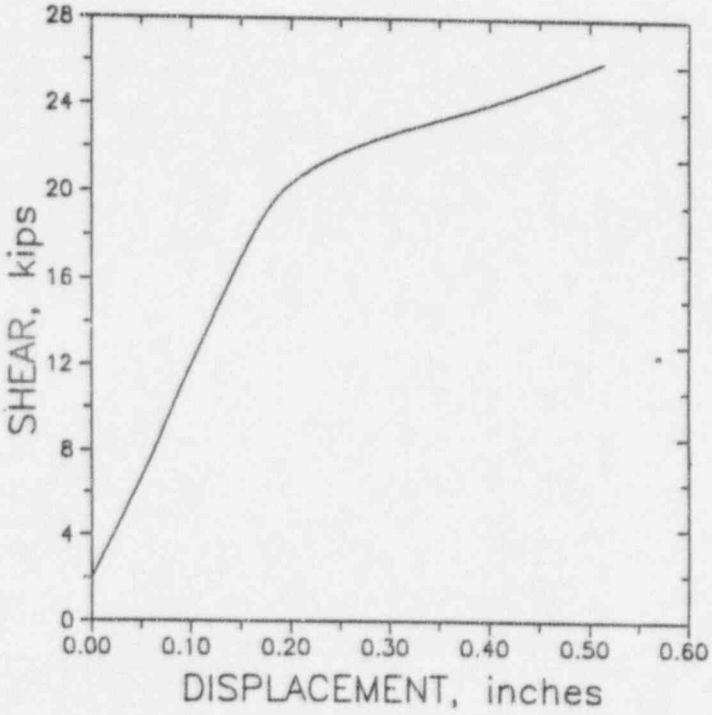

Figure B-12. Example force-displacement curve for 1 in. Richmond insert in shear, f'_c=2850 psi (10).

weld material contribute significantly to the total capacity and must be considered in determining component capacity. Note that in both the low and high-frequency cases the ultimate capacity is assumed to occur at the same displacement; thus, for high frequency components the inelastic energy absorbing factor is larger.

ELECTRICAL FUNCTIONALITY FAILURE MODES

For relays and other electrical components which are acceleration sensitive, the procedures given in Reference (2) for addressing functionality concerns are adequate when the relay frequency is less than about 16 hz because cabinet amplification factors can be confidently estimated in this range. Also, the peak of floor response spectra will occur below 10 Hz in typical nuclear power plant structures. This coupled with the use of a flat relay capacity spectrum which ranges up to at least 16 Hz ensures that the procedures in Reference (2) can be confidently applied. For these cases no new requirements are needed and the SMA procedures are adequate.

For relays and other electrical components which are acceleration sensitive, and are sensitive to frequencies in excess of 16 hz, alternate procedures are recommended. Strategies other than direct evaluation should be attempted before trying to qualify the relays per se, for the seismic floor motion. Recently a list of relays that are sensitive to high-frequency low-level seismic motions has been developed (11). These relays should be avoided if at all possible. Other high capacity relays are also identified in Reference (12). The recent relay functionality guidelines developed by EPRI can be used to determine whether chatter is acceptable (11). The following approaches are recommended:

- If relay chatter would cause an unacceptable change of state, determine whether the plant operators can recover the required plant operating condition in a relatively short period of time after the earthquake.
- If necessary, modify the electrical circuitry to bypass the dependance on the critical relay in question.
- Replace the relay in question with a rugged model which can be easily qualified (e.g., critical frequency less than 16 hz).

If these options are not feasible then a direct capacity check of the candidate relay will be required. For this case realistic floor response spectra based on the unmodified RLE ground response spectrum which span the frequencies of interest will have to be estimated. It is anticipated that simplified analytical procedures could be developed to modify in-structure response spectra which are

based on the RLGM response spectrum. This will avoid having to generated two sets of in-structure response spectra for equipment evaluation. For example, there is evidence for typical ground response spectrum shapes currently being developed for the EUS, which are rich in high-frequency energy and significantly reduced at frequencies below 10 hz, that the spectral amplifications at higher elevations in nuclear power plant structures are less than a factor of two relative to the ground (1).

Next, an in-cabinet capacity spectrum should be developed which includes the critical relay mounted in the cabinet. This step is necessary since at high frequencies (i.e., greater than about 16 hz) motion amplification factors from the floor to the mounting point of the relay device are highly variable and a single acceleration value that can be used in SMA can not be confidently determined. It is likely that a shake-table test will have to be performed to develop the required cabinet-level capacity spectrum. Using this capacity spectrum, the calculated floor response spectrum can be compared in the standard way to evaluate the relay.

SUMMARY

In summary, the capacity screening guidelines have been defined in terms of spectral acceleration and velocity limits (see Fig. B-3). In performing a seismic evaluation for ductile components the input ground response spectrum should be modified as demonstrated in Figure B-6 in order to reduce the portion of the response spectrum which is not potentially damaging to ductile components. The reductions for high-frequency should be applied in addition to the reductions recommended for the incoherence of ground motion recommended in Reference (2). For evaluation of relay chatter and other functionality failure modes the unaltered response spectrum should be used. Care should be exercised in developing floor response spectra for relay chatter evaluation in order to properly transmit the high-frequency motion to the floor level.

Section 4

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Appendix C

INTEGRATION OF SEISMIC IPE AND A-46 REVIEWS

by

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Robert P. Kennedy
Robert P. Kassawara

CONTENTS

Appendix C

INTEGRATION OF SMA AND A-46 REVIEWS

INTRODUCTION

Appendix C provides guidance for Individual Plant Examination for External Events (IPEEE) when a seismic review also is being conducted for NRC Unresolved Safety Issue (USI) A-46. In the following discussion it is assumed that the plant will be reviewed for A-46 using the Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment (GIP) (1). It is recommended that the IPEEE and A-46 reviews be conducted concurrently and that the review tasks be combined, whenever possible.

Background on the two review programs is given in the next section. This is followed by the procedures for conducting combined A-46 and IPEEE reviews. Finally, suggested strategies for combining tasks for the two reviews are given in the last section.

BACKGROUND ON IPEEE AND A-46 PROGRAMS

As discussed in Section 3 the A-46 and IPEEE reviews serve different purposes. The A-46 program is directed at assuring that the plant has been constructed properly and that the capacity of components are consistent with the plant design basis. In contrast, a SMA review is directed at the issue of seismic margin and seeks to answer the question of whether there is high confidence of a low probability of failure at an earthquake level which is higher than the plant design basis. In general, the seismic input and the component capacities are different for the two programs. Depending on both the relative seismic input and factors of safety for capacity for the two programs one review will control over the other. However, it is possible that for one type of element (e.g., anchor bolts) one review may control, but for another type (e.g., relay chatter) the other review may control. It can not be determined a priori which one will control in all cases for all components.

In general, it is expected that the HCLPF capacity using the requirements for the SMA review will be higher than the capacity using the requirements for an A-46

review. In other words, for the SSE design level the engineer should have very high confidence that the plant has a low probability of failure. This is true since the factors of safety to resist the plant design earthquake will be higher because of the underlying review philosophies. For example, the HCLPF capacity of flat-bottom tanks should be about a factor of 1.5 higher than the A-46 capacity as determined using the procedure in the GIP. By examining the plant at the SME level (which is higher than the SSE) the potential for a brittle failure can be investigated. This gives assurance that there is still adequate capacity if the SSE is exceeded. In fact, there is an assumption in the SMA methodology that the median capacity (i.e., the capacity at which there is a 50 percent chance of failure, or alternately, a 50 percent change of survival) of a component is at least twice the HCLPF capacity (2). This assures the engineer that there is no "cliff" just above the HCLPF capacity.

The objective of this appendix is to provide examples for the plant seismic capability engineers in order to encourage pursuing the differences between the requirements for the two reviews. This will provide a basis for determining which review controls for each component at the beginning of a IPEEE study to enable the analysts to perform only a single capacity calculation for each outlier that covers both reviews.

PROCEDURES FOR PERFORMING IPEEE AND A-46 REVIEWS CONCURRENTLY

There are three types of deterministic reviews in the IPEEE program, full-scope seismic margin assessment (SMA), focused-scope SMA and reduced-scope assessment. At the option of the plant owner 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 combined. Since it is acceptable to perform the reviews independently, the SRT can always use directly the corresponding requirements for each review.

The procedures for each type of IPEEE review are discussed in the following subsections.

Full- and Focused-Scope Seismic Margin Assessments

Section 3 gives general requirements for performing a full- or focused-scope SMA. Guidance is provided below for plant walkdown, seismic capability assessment and review documentation when a SMA is conducted for either of these two approaches at the same time as an A-46 seismic verification of nuclear power plant equipment is

performed.

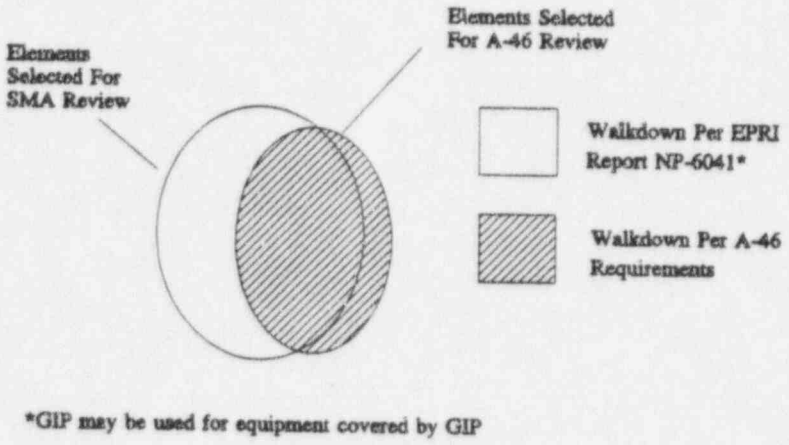
Walkdown, screening and determination of outliers. The components selected in the success paths will generally be different to some extent for IPEEE and A-46. For example, structures, piping and secondary equipment which could fail and affect success-path components are included in a SMA review but not in an A-46 review. In addition, the philosophy for providing redundancy for the primary success path is different for the two programs. This also leads to the selection of different components.

Figure C-1a shows schematically the overlap between the two sets of elements for plant walkdown for the two programs. It is recommended that elements which overlap the two programs (as well as equipment in the A-46 program) be walked down using the requirements in the GIP for the A-46 program. For elements which are common only to the A-46 review the GIP must be used. For elements which are only in the SMA review the requirements in EPRI report NP-6041 (3) can be used; however, the GIP may be used instead of EPRI report NP-6041 for all elements covered by the GIP if the bounding spectrum exceeds the seismic margin earthquake ground response spectrum and the GIP caveats are followed. This is true for the case of the median shape NUREG/CR-0098 spectrum anchored to 0.3 g.

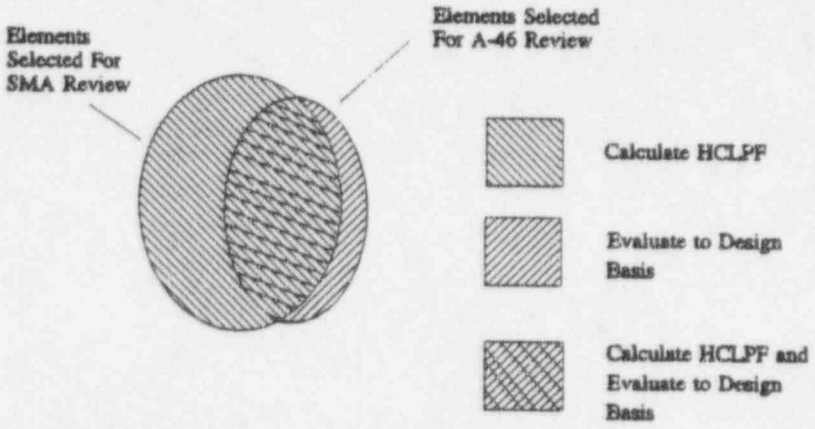
The walkdown requirements in the GIP are more demanding than the walkdown requirements for a SMA. There are some constraints which must be considered in the review when the GIP is used for the walkdown of a SMA component. The GIP requires that the following four criteria be met (1):

- The seismic capacity must be greater than the seismic demand, as defined by the earthquake experience data base and bounding spectrum, the generic seismic testing data base or equipment-specific seismic qualification data.
- Specific caveats are given in the GIP for use of the data as listed above. It is necessary to verify that the equipment meets those caveats.
- The equipment anchorage capacity, installation and stiffness must be adequate to withstand the seismic demand.
- Nearby equipment, systems and structures must not fail and prevent safety-related equipment from performing its function.

For elements common to both reviews, the seismic input is generally different. However, for functionality or structural integrity failure modes the GIP provides conservative screening procedures for equipment included in the SMA review.



a. Walkdown



b. Seismic Capability Work

Figure C-1. Combined Seismic Margin Assessment and A-46 Review Strategy

For cases where a component in the SMA review is not screened out when using the GIP requirements, the SRT can always utilize the evaluation procedures in EPRI report NP-6041. In most cases it will be more efficient to use the A-46 screening guidelines initially for all common equipment, since they are already being used for equipment in the A-46 program.

When evaluating equipment anchorage, the SRT should first investigate the relative differences between the seismic input as well as the assumed anchorage capacities for the two programs. Since the seismic input, as well as the factors of safety for anchorage capacity, is generally different for the two programs the anticipated demand-to-capacity ratios should be calculated for the two reviews. The requirements for the program with the largest ratio should be used initially for both programs. Again, if the anchorage capacity for a particular component is found to be inadequate using this approach then a specific calculation should be conducted for each program using the corresponding seismic input and acceptance criteria.

The benefits from this approach are more apparent in the walkdown process when component anchorage systems are similar to each other and generic values can be developed prior to the walkdown. However, even if most components have unique anchorage details it still will be easier to perform the anchorage capacity review for the most conservative criteria. This is true as long as it is generally found that the capacities are adequate. For cases where the anchorage is not adequate, a capacity check should be conducted using the criteria for each program.

The equipment anchorage installation and stiffness requirements for the A-46 program are more conservative than required for a SMA review. This follows from the general philosophies where it is assumed in the SMA program that the plant has been designed and constructed according to licensing requirements. In contrast, one aspect of the A-46 review is to verify systematically that equipment anchorage has been properly installed. Thus, the A-46 anchorage walkdown requirements are more rigorous. For equipment outside the scope of the A-46 program a torque test for the purpose of the IPEEE program is not required, unless the SRT suspects that a problem exists.

When walking down equipment in a plant the potential for adjacent equipment, systems or structures to fail, fall and impinge on safety-related equipment is reviewed as part of the A-46 program. This also is required when conducting a SMA walkdown. One additional consideration must be included for common components in

order to comply also with SMA requirements. The possibility of a component failing which could flood a safety-related component must also be considered.

When conducting a plant walkdown, screening and evaluation work sheets must be filled out for each element reviewed. Work sheets are provided in the GIP for A-46 walkdowns and in EPRI report NP-6041 for SMA walkdowns. These sheets should be used for elements which are included in one program. For equipment common to both programs (or for IPEEE equipment initially being screened using the GIP) the work sheets in the GIP should be used whenever possible. In addition, they should be supplemented by the additional work sheet shown in Figure C-2. The supplemental work sheet documents the required information beyond the data requested in the GIP which is required in the SMA review.

Calculation of HCLPF capacities and resolution of outliers. Equipment which is screened out during the plant walkdown is not considered further in the review. It is assumed that screened-out equipment which is common to both programs has a HCLPF greater than the seismic margin earthquake (SME) and also has a capacity which complies with the plant design basis. For elements which are not screened out during the walkdown, capacity calculations must be performed. For equipment in the A-46 review, capacities are computed using the GIP. For equipment in the SMA review HCLPF values are computed based on the requirements in EPRI report NP-6041. Thus, two capacity calculations are in general required for equipment common to both programs. Figure C-1b shows schematically the seismic capability requirements for elements in the two reviews.

As discussed in the last section in this appendix there are strategies where a common calculation can be made which provides an efficient procedure for obtaining capacities required by both programs. As discussed above for anchorage, and in Section 3 of this report, generic anchorage calculations should be performed prior to the plant walkdown. By calculating the expected demand-to-capacity ratios for the two programs the most conservative criteria can be used for equipment common to both reviews.

As stated above, when the capacity of a component is less than the demand the SRT has the option to perform the reviews independently using the requirements of each program separately. However, the purpose of combining calculations is to minimize the total effort.

Status Y N U

Sheet 1 of _____

Equip. ID No. _____ Equip. Class _____

RELAY WALKDOWN

1. Does spot check of essential relays indicate relays present and properly mounted? Y N U N/A

2. Are essential relays required to function during earthquake screened out? Y N U N/A

   If no, attach list of relays with locations in cabinet and general dimensions, thicknesses and details of mounting plates that support relays for later analysis.

3. No other relay concerns? Y N U N/A

Requirements for relays satisfied? Y N U

SYSTEMS INTERACTION EFFECTS

1. No potential sources could flood or spill onto cabinet? Y N U N/A

DESCRIBE POTENTIAL PROBLEMS INDICATED BY NO OR UNSATISFACTORY (Use additional sheets, if necessary)

IS EQUIPMENT FREE OF NEED FOR FURTHER INVESTIGATION, EXCLUDING RELAY CHATTER?
YES _____ NO ____

IS EQUIPMENT FREE OF NEED FOR FURTHER RELAY CHATTER INVESTIGATION? YES ___ NO ___

Evaluated by: _____ Date: _____

Evaluated by: _____ Date: _____

Figure C-2. Additional Screening and Evaluation Worksheet For SMA

Documentation of review. The review documentation should follow the requirements of each program (i.e., EPRI report NP-6041 and GIP). In general, two reports should be prepared, each which follows the documentation requirements of the applicable program.

Reduced-Scope Assessment

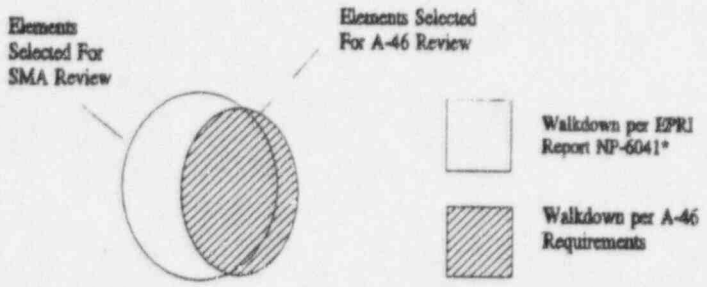
Section 3 gives general requirements for performing a reduced-scope review. Guidance is provided below for plant walkdown, seismic capability assessment and review documentation when a reduced-scope assessment is conducted at the same time an A-46 seismic verification of nuclear power plant equipment is being performed.

Walkdown, screening and determination of outliers. For the same reasons given above for the full- and focused-scope SMA reviews the elements selected for evaluation will be different for the A-46 and SMA programs. The walkdown requirements for a reduced-scope review should follow the guidance given in NP-6041. The Seismic Review Team (SRT) can use either the GIP or the screening tables in EPRI report NP-6041 to screen equipment in the IPEEE review. If a plant is conducting an A-46 review it is recommended that the screening requirements in the GIP be used for all components, whenever possible. Figure C-3 shows schematically that all components identified in both reviews should be reviewed using the requirements in the GIP.

Elements such as piping and structures are outside the scope of an A-46 review. For these elements the screening requirements in EPRI report NP-6041 should be used.

Calculation of element capacities and resolution of outliers. Equipment which is screened out during the plant walkdown is not considered further in the review. It is assumed that screened-out equipments have capacities which comply with the plant design basis. For elements which are not screened out during the plant walkdown, capacity calculations for the SSE input must be performed.

It is recommended that the acceptance criteria in the GIP be used to develop capacities for all components in a combined review, whenever possible. For structures, piping and other components which are not included in the GIP the requirements in the FSAR should be used.



*GIP may be used for equipment covered by GIP

Figure C-3. Combined Reduced Scope and A-46 Review Strategy

Documentation of review. The review documentation should follow the requirements of the GIP for the A-46 review. The basis should be documented for the selection of additional components identified for the reduced-scope review.

STRATEGIES FOR REVIEW EXECUTION

As discussed above it is suggested that the requirements for the IPEEE and A-46 reviews be considered at the beginning of a plant seismic investigation to develop short cuts to avoid doing the work twice. Some examples of how this can be done are provided in this section.

During the screening step it is suggested that the requirements in the GIP (1) be used in the review. This implies that the caveats in the GIP are satisfied. The principle requirement is that the SME ground response spectrum is enveloped in the frequency range of interest by the Bounding Spectrum in the GIP. For example, this will occur when a NUREG/CR-0098 median response spectrum shape is anchored to a 0.3g, or less, peak ground acceleration (pga). This was the argument made in the review of the Edwin I Hatch Nuclear Plant SMA for using the GIP in the plant walkdown (4). For SMEs higher than 0.3g it may be still possible to use the GIP at lower elevations. For this case it must be shown that the in-structure response spectra are enveloped by 1.5 times the Bounding Spectrum in the GIP.

For the evaluation of outliers it may be possible to perform one calculation for each component which serves for both the IPEEE and A-46 reviews. For example, in the Hatch SMA/A-46 combined review, expansion anchor bolts were evaluated using the input and capacities as required for the SMA as given in Reference (5). In the Hatch review, new in-structure response spectra were developed for the SMA where the SME was selected to be a median shape NUREG/CR-0098 response spectrum anchored to 0.3g pga. The SSE pga for Hatch is 0.15g, and it was assumed that the A-46 input could be obtained by scaling the SMA input by the ratio of the SSE pga to the SME pga. General procedures for scaling in-structure response spectra are given in Reference (3). In order to determine which review controls, both the seismic demand and the corresponding bolt capacities must be considered. At the time that the Hatch SMA was performed factors of safety for shell-type expansion anchors were 4 and 3 for the A-46 and SMA reviews, respectively.

Since the in-structure response spectra for both reviews are considered to be median centered, the A-46 spectra are obtained by using half the SMA spectra (because the A-46 pga is 0.15g and the SMA pga is 0.30g) times a factor of 1.25 to convert to approximately the one standard deviation level. Thus, the demand to capacity ratio, D/C for the two studies is proportional to the following factors:

$$\frac{D}{C} \propto \frac{\overset{\text{A-46}}{\dfrac{0.5 * 1.25}{1/4}}}{2.5} \qquad \frac{\overset{\text{SMA}}{\dfrac{1.0}{1/3}}}{3.0}$$

As seen in this example the D/C ratio is higher for the SMA review; thus, by evaluating shell-type expansion anchors for the SMA input and the capacities required in Reference (5) both reviews were covered.

Currently, the factor of safety in the GIP for the shear capacity of both shell-type and non-shell-type anchor bolts is 3.0, while the corresponding SMA value which is recommended in Revision 1 to Reference (5) is 2.0 (3). Even with these revised factors of safety the D/C ratio for the Hatch example SMA D/C ratio would be about 7 percent higher than the A-46 D/C ratio. Again, for this case performing the analysis for the SMA review would cover both reviews for bolts subjected to only shear forces.

Similar type generic demand to capacity calculations can be performed for other

elements (e.g., tension in bolts, welds and relay chatter) to determine which review controls. One component class where the calculations are generally involved is flat-bottom liquid storage tanks. Methodologies are given separately for the A-46 and SMA reviews to calculate the tank capacities (1, 3). In lieu of performing two analyses for each tank it is acceptable to use the procedure in Reference (3) for the SMA review to obtain a HCLPF capacity and to divide it by a factor of 1.5 to be used as the capacity for the A-46 review.

For all components it is conservative to calculate a HCLPF in-structure response spectrum capacity using the procedures in Reference (3) and divide the result by a factor of 1.875 to obtain an equivalent A-46 evaluation capacity, when realistic median-centered methods are used to generate the A-46 in-structure response spectra. Similarly, a factor of 1.50 is adequate when conservative design in-structure response spectra are used. If the equivalent A-46 capacity is greater than the A-46 input then the component satisfies the A-46 requirements. In general, the 1.875 factor is conservative (note above that a factor of 1.5 is adequate for flat-bottom tanks). If the A-46 capacity calculated in this manner is not greater than the A-46 input, then the analyst should use the procedures in the GIP directly since it is likely that a larger A-46 capacity will be found.

The factor of 1.875 above was obtained by examining the various requirements for calculating anchorage and equipment functionality capacities for both the SMA and A-46 programs (1,3). It was found that the largest reduction factor occurs for anchor bolt capacity for the case of pure shear. As stated above the factors of safety for anchor bolt shear capacity are 2.0 and 3.0, respectively for the SMA and A-46 requirements. However, to make the comparison compatible when using in-structure spectra generated with median-type methods, the A-46 input must be multiplied by a factor of 1.25. Thus, the conversion factor is 3/2 * 1.25, or 1.875. If conservative design-type methods are used in the A-46 evaluation, the 1.25 factor can be eliminated given a total factor of 1.50.

Note that it may be possible to develop more realistic A-46/SMA reduction factors by examining the specific factors of safety for each type of review. For example, for tension in expansion anchors the limiting factor of safety of 3.0 is applicable for both A-46 and SMA reviews for single bolts when hairline concrete cracks are considered unlikely. For the case when realistic median-centered methods are used dividing the HCLPF capacity by 1.25 (i.e., 1.25 * 3.0/3.0) to obtain the A-46 capacity is more realistic.

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Appendix D

CONTAINMENT PERFORMANCE REQUIREMENTS

by

David R. Buttemer

CONTENTS

Appendix D

CONTAINMENT PERFORMANCE REQUIREMENTS

INTRODUCTION

The current seismic margin assessment (SMA) methodology (Reference (1), developed in 1988 and revised in 1991) focuses on the verification of success paths necessary to prevent core damage and intentionally does not address containment functions/systems necessary to mitigate the consequences should core damage occur. Indeed, the success path logic diagrams developed in Reference (1) identify alternate means (e.g., systems) of achieving a stable long term safe shutdown (i.e., core damage prevention) following a seismic margin earthquake. In late 1988, the U.S. Nuclear Regulatory Commission (NRC) issued to all licensees generic letter (GL) 88-20, which addressed individual plant evaluation (IPE) requirements. This letter indicated that the IPE should follow NRC's "defense-in-depth" principle and should: (1) examine core damage prevention, and (2) conduct a containment performance analysis (following the general guidance for the "back-end" analysis given in Appendix 1 of GL 88-20). The GL 88-20 indicated that the initial IPE should address internal events only and that external events (including earthquakes) would "proceed separately on a later schedule." In late 1989 the NRC staff issued a draft letter extending the IPE process to include external events (i.e., the IPEEE); earthquakes were identified as a major class of external events. In 1990, the NRC issued draft documentation providing guidance for conducting an IPEEE and what is to be included in the licensees submittal. The purpose of the Appendix is to describe the proposed industry approach to reviewing containment performance for the seismic portion of the IPEEE process.

The system considerations, which lead to the selection of structures and equipment necessary for containment performance, are given in the next section. This is followed by guidance for performing a seismic capability assessment of the elements included in the selected containment success paths.

SYSTEMS CONSIDERATIONS

The methodology for identifying the systems required to assure a long term safe shutdown condition following a seismic margin earthquake (SME) with a safely

cooled core are described in Reference (1); by "long term", the evaluation required that a stable hot or cold shutdown condition be maintained for at least 72 hours.

The major systems related assumptions which were made were that the SME failed offsite power (and that no credit be taken for offsite power recovery), that a seismically induced small LOCA should be considered unless plant specific evaluation could rule it out, and that seismically induced electrical relay and contactor chatter needed to be evaluated. This Reference (1) methodology explicitly excluded consequence mitigation (read "containment") systems from the evaluation process. The relay chatter evaluation addresses the operability of those systems necessary to assure a long term safe shutdown condition, but also addresses the potential of relay chatter causing normally closed isolation valves that separate high and low pressure primary coolant lines to open. If such an event were to occur, it would cause overpressurization and could potentially fail the low pressure piping, resulting in what is referred to as an interfacing system LOCA (or the classical V-sequence in WASH-1400 parlance), which results in containment bypass and could cause core damage.

Before discussing the containment functions and systems that should be addressed in a seismic margin assessment, it is informative to consider what is typically done for an internal events containment evaluation. The "back-end" IPE analysis typically addresses a number of containment functions (including isolation, sprays, and long term heat removal), as well as the potential for containment bypass events. Ice condenser and Mark II/IIIs also address hydrogen control (e.g., ignitors) and recent NRC guidance to Mark I licensees (Supplement 1 to the GL 88-20, a result of the NRC Containment Performance Improvement Program) suggests enhancements to drywell spray and the ability to vent the torus. Clearly, the potential environment and demands on containment systems during a severe accident may exceed the original design basis. Superimposed on the containment systems issue are a large number of very complex phenomenological issues that currently have large uncertainties. These phenomena include in-vessel and ex-vessel fuel-coolant interactions, direct containment heating, hydrogen burning/detonation, core-concrete interactions with noncondensible gas production, basemat and/or liner melt-through, etc.

More recently (Supplement 3 to the GL 88-20, dated July 6, 1990, which announced the completion to NRC's Containment Performance Improvement Program), the NRC provided additional insights for PWR containments and BWR Mark II and Mark III

containments. Specifically, this letter addressed for BWR Mark II containments the pros and cons of suppression pool venting and the possibility that core debris may fail the downcomers or drain line and result in suppression pool bypass. For BWR Mark III and PWR ice condenser containment designs, the letter pointed out potential vulnerabilities of these plants wherein an extended station blackout could result in core damage and hydrogen accumulation because of unpowered ignitors and, should electric power be restored later, the ignitors could ignite detonable concentrations of hydrogen. The letter alludes to the use of an alternate AC option to the Station Blackout rule to ensure uninterrupted ignitor operation. The letter describes concern for local detonable hydrogen mixtures developing in large dry PWR containments or of a globally detonable hydrogen mixture developing in smaller subatmospheric PWR containments.

As described above, there are a large number of containment systems and containment phenomena that have an impact on containment performance. It can be generally stated that the original containment design criteria did not specifically address the full ramifications of a severe accident, which may result in vessel melt-through and significant amounts of molten core debris on the containment floor. Clearly, the complex phenomenological issues related to containment performance in severe accidents are beyond the scope of a SMA. It is recommended that the SMA demonstrate with a high degree of confidence that those containment related functions that are necessary to prevent <u>early</u> containment failure survive the SME (early means roughly the 12 hours following the seismic margin earthquake (SME)). For essentially all containment designs, this includes: (1) successful containment isolation, (2) that the SME does not fail the containment structural integrity (including penetrations and closures), and (3) the SME (or seismically induced relay chatter) does not result in containment bypass (this is in the current SMA methodology). For large, dry PWR containments with only successful isolation, containment failure would not be expected for several 10s of hours and containment spray and/or fan coolers are generally unimportant in the short term. The SMA already addresses long term heat removal for BWRs, but does not address drywell sprays (for Mark I) or hydrogen control (for uninerted Mark II, Mark III or ice condenser containments). These systems may be required to prevent early containment failure. Because there is a significant variation in containment designs, it is suggested that the IPE internal events containment evaluation team be consulted to determine which systems are required (not just desired) to assure early containment integrity following a SME.

The success path logic diagrams (SPLDs) describe in Reference (1) denote those systems which are necessary to provide a long term safe shutdown condition. These SPLDs should be extended to include early containment performance requirements following a SME and assuming that severe core damage has occurred. An example SPLD extension is shown in Figure D-1 for no plant in particular. Clearly, post SME containment isolation and containment integrity must be demonstrated along with assurance that relay chatter can not cause normally closed isolation valves to opening and induce an interfacing system LOCA.

Care must be exercised as to what active equipment or systems are required (not just desired) to prevent early containment failure. For example, in a large dry PWR containment, fan coolers may not be designed to operate in the harsh environment associated with a severe accident (and indeed, nonsafety class coolers generally trip on a safety signal) and the effectiveness of containment sprays in providing debris bed cooling depends on the sprayed water having access to the reactor cavity area. Another example would be the reactor building and standby gas treatment system in a BWR Mark I containment. Clearly, neither one of these has an impact on primary containment failure and, depending on plant specific design features, may not have much influence on consequence mitigation if containment failure occurs. For pressure suppression containment designs the evaluation must demonstrate that 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) survives the SME. The SMA team should solicit guidance from the IPE containment evaluation team as to what functions are required to prevent early containment failure.

SEISMIC CAPABILITY ASSESSMENT

As discussed in Section 2 of this report it is recommended that potential containment failure be reviewed for those plants which elect to perform a full- or focused-scope deterministic review. The identification of structures and equipment in the containment function success path should be based on the guidance given in this appendix. Thus, the structures and equipment selected in the success path will be the same for all full- or focused-scope plants.

In performing a deterministic review (i.e., full- or focused-scope SMA) all elements in the success path should be evaluated according to the requirements in Section 3 of this report, in the same manner as for elements required to prevent

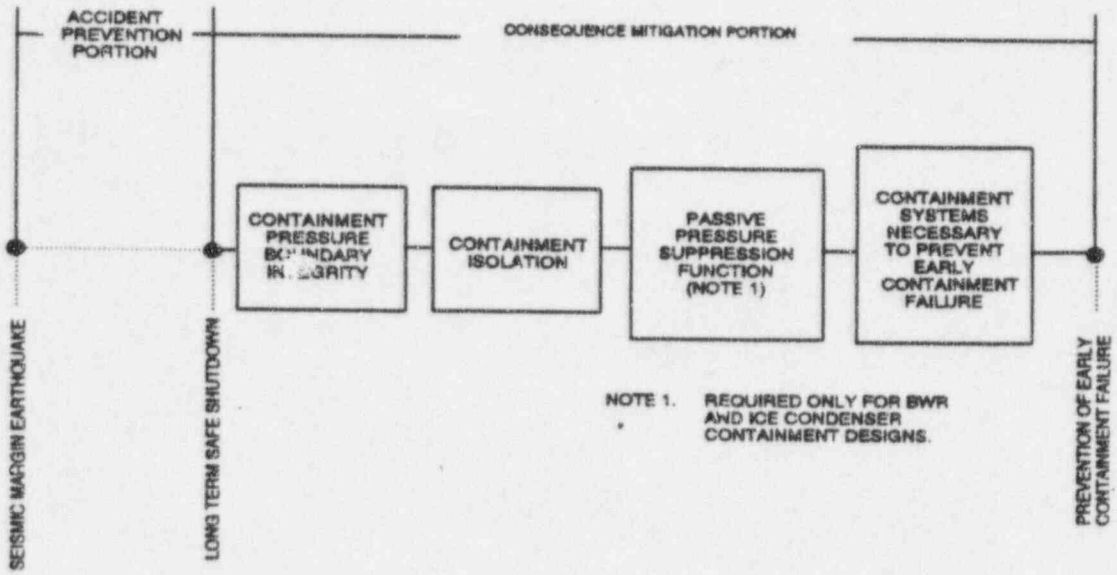

Figure D-1. Success Path Logic Diagram for Containment Performance Evaluation

core damage. Guidance for performing IPEEE in conjunction with an A-46 review, is given in Section 4.

The guidelines in EPRI report NP-6041 (1) are for seismic margin assessment to prevent a core damage accident. However, the same element types considered in that report also include structure and equipment types which will be selected in the containment success path. Requirements for both steel and concrete containments are provided in Section 6 of that report. In general, containment structures are rugged and a review of construction drawings and the design-basis calculations can be performed to screen out the containment structure or to identify features which require additional consideration. Penetrations need to be carefully reviewed to verify that relative motions between structures can be accommodated.

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Appendix E

DETERMINATION OF REVIEW-LEVEL GROUND MOTIONS FOR SEISMIC IPE CLOSURE

by

Robert T. Sewell
J. Carl Stepp

CONTENTS

Appendix E

DETERMINATION OF REVIEW-LEVEL GROUND MOTIONS FOR SEISMIC IPE CLOSURE

INTRODUCTION

The tasks of determining whether or not a screened-in component is a potential vulnerability, and of deciding how a potential vulnerability should be treated, fall within the scope of seismic-IPE closure procedures. As discussed in Section 6, the determination of a set of review-level ground motions (RLGMs) is a key element of the seismic-IPE closure guidelines and framework. The purpose of this appendix is to outline the technical procedures for the development of plant-specific RLGMs; the development is consistent both with the use of focused-/full-scope SMA methods and with risk-based closure criteria presented for the internal-events IPE (1). The RLGMs themselves can be found in Reference (2) for plants for which EPRI-SOG seismic hazard results have been computed.

Section 3 and Reference (3) provide guidelines for identifying SMA outliers and for determining component HCLPF capacities of these outliers. The median 5%-damped NUREG/CR-0098 (4) spectrum is recommended in Section 3 as the basis for seismic input in HCLPF calculations when determining remaining outliers. This characterization of seismic input, however, is known to be substantially conservative for the majority of sites located in the central and eastern U.S. Because closure criteria ultimately rely on assessment of cost-benefits (i.e., costs incurred versus benefits achieved), it is important to have (in contrast to the NUREG/CR-0098 spectrum) a realistic, site-specific characterization of the ground-motion input. The uniform hazard spectrum (UHS) provides this realistic input characterization and, when tied to the appropriate exceedance frequency, describes the plant HCLPF capacity required to satisfy a specified safety target measured in terms of core-damage (or plant-damage-state) frequency. The following discussion pertains primarily to selecting appropriate exceedance frequencies for anchoring ordinates of the UHS.

APPROACH FOR DETERMINING REVIEW-LEVEL GROUND MOTION

Basis of Approach

To determine a RLGM, it is necessary to establish an appropriate safety criterion.

Severe-accident-issue closure guidelines for the IPE (1) specify such safety criteria. These criteria are consistent with a total mean core-damage frequency of 1×10^{-4} (or less), which is based on current recommendations (5, 6) for implementation of NRC safety-goal policy (7). As presented in Section 6 of this document, industry's IPE closure framework employs a graded set of decision elements with decision criteria expressed in terms of percentages of total core-damage frequency for major functional accident sequence groups; the greater the percent contribution, the greater the scope in cost-benefit comparison. For instance, in the IPE core-damage evaluation process: if the contribution to total core-damage frequency of a major functional accident sequence is less than 5%, no action is required; if the contribution is between 5% and 20%, a cost-effective change to utility accident management guidelines (UAMGs) only is considered; if the contribution is between 20% and 50%, cost-effective procedural and minor hardware changes are considered, in addition to UAMG changes; otherwise, if the contribution is greater than or equal to 50%, cost-effective changes in design, normal and emergency operating procedures, and UAMGs are all considered. For a consistent development here, three RLGMs (for any given plant) are needed for seismic-IPE closure, corresponding to safety targets of 5.0×10^{-5} (50% contribution to the total core-damage-frequency safety target of 1.0×10^{-4}) 2.0×10^{-5} (20%), and 0.5×10^{-5} (5%). Each of these three plant-specific RLGMs can be obtained using the procedure outlined below. Reference (8) outlines, in greater detail, the analysis procedure and justification for evaluating the safety-target-consistent RLGMs. As denoted in Section 6 (Figure 6-3), the review-level ground motions for mean[1] core-damage frequency safety targets of 5.0×10^{-5}, 2.0×10^{-5} and 0.5×10^{-5}, are termed RLGM-A, RLGM-B and RLGM-C, respectively.

For review-motion determination, one asks what is the minimum plant seismic capacity which insures that the particular seismic core-damage-frequency criterion is met. This safety-based seismic capacity determines a review-level earthquake

[1]As an alternative to the use of a mean core-damage safety target, one may use a median measure that is consistent with the mean target. For determining a consistent median measure, results of seismic core-damage frequency distributions obtained in past risk studies can be used. NUREG-1150 (9), for instance, provides mean and median core-damage frequency results for two nuclear power plants; results are given separately for EPRI and LLNL seismic hazards. These results suggest that a mean-to-median ratio between 5.0 and 6.0 is appropriately representative of seismic core-damage-frequency distributions obtained from modern analyses. Such a factor would imply, for instance, that a median core-damage frequency target of about 4×10^{-6} is consistent with a 2×10^{-5} mean safety target.

(RLE) spectrum, used as a basis in obtaining a RLGM spectrum, for which a given plant may be reviewed in its seismic IPE. If the seismic IPE reveals that the plant seismic capacity equals or exceeds the review level, then the plant is shown to have adequate seismic resistance; consistent with safety objectives, the plant satisfies an acceptable (and conservative) safety criterion. For such a case, consideration can justifiably be focused away from major, expensive safety enhancements (such as design, hardware, and major procedural changes) without explicitly appealing to cost-benefit considerations. The seismic IPE closure process may reveal obvious cost-effective, simple upgrades (e.g., anchorage strengthening or installation) to improve seismic resistance; if a review level cannot be met after these obvious fixes, then cost-benefit analysis, consistent with NRC backfit policy (10), ascertains the extent of any further consideration of potential vulnerabilities. Using this approach for determination of the RLE spectrum and implementation of seismic-IPE closure guidelines is consistent with objectives of NRC severe-accident policy (11).

The general approach for obtaining a plant-specific review-level spectrum is outlined below:

- Select a target mean seismic core-damage frequency safety guideline (e.g., 2×10^{-5}).
- Obtain mean hazard curves, or obtain pseudo-mean curves by converting from median hazards[2]. Hazard curves should be obtained for spectral acceleration at appropriate (dynamic) response frequencies (i.e., 1, 2.5, 5 and 10 Hz).
- Assume a trial seismic capacity (i.e., in terms of seismic fragility curve parameters).
- Compute a mean seismic core-damage frequency corresponding to the trial capacity, given the seismic hazard.
- Iteratively adjust the seismic capacity to just meet the assumed target mean seismic core-damage frequency guideline.

[2]This document recommends the use of EPRI mean seismic hazard results. If, however, both EPRI and LLNL results are used, it would be appropriate to start with median hazard curves to achieve the greatest consistency among use of EPRI and LLNL analyses. Since safety comparisons are based on mean core-damage frequency, though, one would need to convert the median curves to equivalent or pseudo means. For this conversion, a representative variation (with ground-motion) of uncertainty in seismic hazard, characterized by the logarithmic standard deviation in hazard (at a given ground motion), β_H, can be used. The equivalent mean hazard is obtained by multiplying the median hazard by the factor $\exp(0.5\beta_H^2)$. A representative value of β_H, an average for ground motions between the typical plant HCLPF and median capacities, might be 1.0 or so.

- Find the mean seismic hazard (exceedance frequency) corresponding to the adjusted seismic capacity.

- Perform the above steps for 1 Hz, 2.5 Hz, 5 Hz and 10 Hz spectral acceleration hazard curves, and average the resulting hazard levels.

- Establish the RLE spectrum as the mean (or equivalent/pseudo mean) uniform hazard spectrum evaluated for the level of hazard determined above; alternatively, use the adjusted seismic capacities themselves, at 1 Hz, 2.5 Hz, 5 Hz and 10 Hz, to define the RLE spectrum.

No assumptions concerning plant median HCLPF capacity nor concerning resistances of plant structures and components are made in formulation of a RLE (i.e., target HCLPF) spectrum. Site-specific seismic hazard results are the key element required to implement the methodology. The RLE spectrum can, in fact, be simply thought of as an alternate description of the seismic hazard, i.e., it is a characterization of the seismic hazard that has direct significance to an acceptable plant core-damage risk.

Description of Approach

Specific details describing particular elements of methodology involved in assessing the RLE spectrum are discussed below:

Seismic hazard description. Available spectral acceleration seismic hazard curves for response frequencies ranging from 1 to 10 Hz are obtained. Ground motion frequencies in the range of 2 to 15 Hz, and in particular from 2 to 10 Hz, have been found in past seismic studies to have the most significant effect on critical plant structures and components. PGA, although used as a ground motion descriptor in past SPRAs, is a poor indicator of earthquake damageability to nuclear power plants, and is sensitive to high-frequency components of ground motion that are generally believed to be non-damaging. Determination of ground motions for seismic severe accident review should, therefore, be keyed to frequencies in the range of 2 to 10 Hz.

Seismic capacity description. The RLE analysis characterizes plant capacity by use of a family of seismic core-damage fragility curves. A core-damage fragility curve conveys, for all values of a given ground motion parameter (e.g., 10-Hz spectral acceleration), the likelihood or frequency of core damage, conditional upon the given ground motion parameter value. (A family of core-damage fragility curves is used to represent the uncertainty in the true core-damage fragility). A

convenient way to characterize the core-damage fragility family is by means of the familiar lognormal parameters $\breve{A}$, β_R, and β_U.

Examination of published, peer-reviewed PRAs reveals that values of β_R and β_U vary little from plant-to-plant. The results of plant-level values of β_R and β_U as derived from six PRA studies for eight reactor units, indicates plant-to-plant averages for β_R and β_U respectively, of 0.22 and 0.24. The plant-to-plant coefficients of variation in β_R and β_U are extremely small (0.14 and 0.16, respectively), indicating that the average values are sufficient as generic characterizations that may be used to describe the shape of plant-level fragility curve families.

In addition to parameters $\breve{A}$, β_R, and β_U, the HCLPF capacity can be obtained from the family of fragility curves. The core-damage HCLPF at a given frequency[3] f is given by the following expression (that is strictly valid only for lognormal plant fragilities):

$$HCLPF_f = \breve{A}_f \exp[-1.65(\beta_R + \beta_U)] \qquad \text{(E-1)}$$

When a HCLPF is determined in this manner and is used as an "anchor point" of a review earthquake spectrum, it is said to be a HCLPF reported at the 50% non-exceedance probability [denoted here as HCLPF(50%)]. HCLPF capacities are by convention, however, reported at an 84% nonexceedance probability. To convert from HCLPF(50%) to HCLPF(84%) we use the following expression (see Reference 12):

$$HCLPF(84\%) = HCLPF(50\%) \times e^{\beta_{pp}} \qquad \text{(E-2)}$$

where β_{pp} is the logarithmic standard deviation in spectral response due to peak-to-peak randomness. A typical value of β_{pp} has been estimated in past studies to be about 0.18. Assuming this value, HCLPF(84%) is obtained as:

$$HCLPF(84\%) = 1.2 \times HCLPF(50\%) \qquad \text{(E-3)}$$

[3]In the past, HCLPF capacities have typically been expressed in terms of a PGA value. Because of the drawbacks of PGA noted above, however, it is preferable to report a HCLPF capacity at some (response-significant) spectral frequency f. Consequently, in this study, when describing a HCLPF capacity (or core-damage fragility), it is necessary to specify a corresponding spectral frequency f used as the basis for its evaluation.

Mean seismic core-damage frequency assessment. Given the seismic hazard and fragility-shape descriptions, the primary effort in evaluating a RLE spectrum is conducting the many numerical determinations of mean seismic core-damage frequency. This numerical procedure is well documented in the seismic risk analysis literature (see, for instance, Reference 13).

In the determination of RLE values, a trial HCLPF is selected as the basis for scaling the fragility curve family. Once the trial HCLPF is selected, the fragility family is scaled by determining the corresponding value of $\check{A}$, as follows:

$$\check{A} = \text{HCLPF} \exp[1.65(\beta_R + \beta_U)] \tag{E-4}$$

where parameters β_R and β_U that govern fragility shape are assumed to have the generic values of 0.22 and 0.24, respectively, as noted above. These values of $\check{A}$, β_R and β_U provide a complete description of the fragility family. This fragility family is integrated with seismic hazard curves of spectral acceleration at 1 Hz, 2.5 Hz, 5 Hz and 10 Hz to obtain estimates of mean seismic core-damage frequency.

Iterative determination of RLE values. The frequency-dependent plant-level HCLPF is iteratively adjusted, resulting in an iterative scaling of the seismic fragility, and corresponding hazard-fragility integrations are performed until a mean seismic core-damage frequency equal to the target seismic core-damage safety guideline is obtained. The HCLPF value for which the target safety guideline is just met is the minimum required plant HCLPF or the RLE value. This HCLPF is reported at the 50% nonexceedance probability; to report the required HCLPF at a 84% nonexceedance probability, Eq. E-3 is used.

RLE values are determined in this iterative manner for spectral frequencies of 1 Hz, 2.5 Hz, 5 Hz and 10 Hz. Once generic values of β_R and β_U are assumed, the RLE values are dependent entirely on the site seismic hazard and the chosen level of the core-damage safety guideline. No assumptions of median plant capacity are made in the assessment of the RLEs.

Determination of RLE spectrum. To obtain a RLE spectrum, mean seismic hazards for 1 Hz, 2.5 Hz, 5 Hz and 10 Hz spectral accelerations are evaluated, correspondingly, at the frequency-dependent RLE values. These levels of mean seismic hazard are averaged, in recognition of the fact that seismic plant response is governed primarily by ground-motion input energy between 2 and 10 Hz

(i.e., critical plant structures and components are sensitive to motions in this frequency band).

This resulting average hazard value from the frequency-dependent RLE evaluations is used to determine the RLE spectrum. The plant-specific RLE spectrum is constructed as the mean UHS evaluated at this average level of hazard. Alternatively, the frequency-dependent RLE values ($HCLPF_f$) may themselves be used directly as ordinates of the RLE spectrum.

Determination of review-level ground motion. High-frequency ground-motion input has generally been shown to have little potential for damaging equipment in nuclear power plants. As discussed in Appendix B, the high-frequency portion of a RLE spectrum may be reduced, therefore, when determining the input motion for severe-accident evaluations. This reduction should occur before making comparisons, for instance, with seismic-IPE closure criteria. The specific procedure for reducing the high-frequency input is presented and discussed in Reference (14). This procedure makes use of spectral ordinates for different damping values, and hence, RLE spectra must be obtained for multiple damping levels. With this reduction of high frequency input, the RLE spectra for the various dampings are converted to a review-level ground-motion (RLGM) for use in severe-accident analysis.

When used as a basis for performing the seismic IPE, the RLGM insures that plants are evaluated for consistent safety levels. Being constructed as a target plant HCLPF spectrum, the RLGM serves as a natural basis in implementing a seismic margin assessment. The result of a SMA is an evaluation of a plant-level HCLPF capacity. If a SMA is conducted for a plant, and the plant is found to have a HCLPF at a level that meets or exceeds the RLGM, then the plant has met the safety criterion associated with the RLGM. The comprehensive walkdown performed in the SMA assures that any potential vulnerabilities that may compromise seismic resistance will be uncovered.

RLGM RESULTS FOR IMPLEMENTING SEISMIC IPE CLOSURE GUIDELINES

The above procedures have been implemented to obtain RLE spectra and RLGM spectra for 58 eastern U.S. plant sites, based on EPRI mean hazard curves. The resulting RLE spectra (for 5% damping) are assessed and presented in Reference (8) for RLE-A, RLE-B and RLE-C; corresponding RLGM spectra (i.e., after high-frequency reduction) are presented in Reference (2). To simplify their use by licensees, each RLGM has been enveloped, over the vibration frequencies of interest (i.e., 2

to 10 Hz), by the NUREG/CR-0098 5%-damped median spectrum, and the associated PGA values have been obtained, as indicated in Table E-1. Hence, for use in closure criteria, licensees can simply compare component HCLPF-PGA values (based on the NUREG/CR-0098 5%-damped median shape) assessed in their seismic IPE, against the PGA values of Table E-1. Alternatively, a new component HCLPF can be computed based on the RLGM spectral shape itself, as provided in Reference (2).

Table E-1

SAFETY GOAL BASED REVIEW LEVEL GROUND MOTION PGA VALUES
(BASED ON NUREG/CR-0098 SPECTRAL SHAPE); EPRI MEAN HAZARD INPUT

Plant No.	RLGM-PGA Values			Plant No.	RLGM-PGA Values		
	RLGM-A	RLGM-B	RLGM-C		RLGM-A	RLGM-B	RLGM-C
01	0.13	0.25	0.40	31	0.10	0.14	0.25
02	0.16	0.23	0.37	32	0.14	0.20	0.30
03	0.08	0.12	0.19	33	0.08	0.13	0.22
04	0.11	0.16	0.26	34	0.02	0.05	0.08
05	0.25	0.35	0.53	36	0.02	0.04	0.07
06	0.13	0.19	0.31	38	0.07	0.11	0.17
07	0.11	0.16	0.28	39	0.08	0.13	0.22
09	0.13	0.19	0.29	40	0.05	0.07	0.11
10	0.13	0.19	0.29	41	0.02	0.04	0.07
11	0.13	0.20	0.36	42	0.04	0.06	0.10
12	0.13	0.19	0.28	43	0.02	0.04	0.07
13	0.07	0.11	0.18	44	0.02	0.04	0.06
14	0.06	0.08	0.13	45	0.07	0.11	0.17
15	0.06	0.08	0.13	47	0.14	0.19	0.29
16	0.06	0.10	0.16	48	0.05	0.07	0.13
17	0.08	0.12	0.22	49	0.08	0.14	0.26
18	0.10	0.14	0.26	50	0.06	0.08	0.17
19	0.08	0.13	0.24	51	0.08	0.13	0.26
20	0.11	0.16	0.24	52	0.08	0.16	0.31
21	0.13	0.18	0.28	57	0.05	0.08	0.13
22	0.13	0.19	0.31	58	0.12	0.18	0.31
23	0.13	0.19	0.31	59	0.05	0.07	0.12
24	0.06	0.08	0.14	60	0.13	0.19	0.32
25	0.13	0.20	0.35	61	0.05	0.08	0.14
26	0.10	0.13	0.22	62	0.05	0.07	0.13
27	0.12	0.17	0.30	67	0.07	0.11	0.19
28	0.11	0.16	0.28	68	0.07	0.10	0.17
29	0.13	0.19	0.32	69	0.10	0.13	0.23
30	0.14	0.19	0.29	70	0.06	0.08	0.16

Although the use of EPRI mean seismic hazard curves is recommended in obtaining RLGM results, one may elect to consider both EPRI and LLNL seismic hazard curves

in implementation of the seismic IPE. In such a case, to obtain the most consistent result, one would use equivalent/pseudo-mean hazard curves and spectra derived from each (EPRI and LLNL) set of median hazard curves.

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