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Seismic info

STATUS REPORT ON SEISMIC RISK ASSESSMENT

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INTRODUCTION

This paper is a status report on an evolving subject. I will briefly describe some seismic risk assessments which have been performed during the past four years in order to show the development of the subject. It is tempting to draw some broad inferences from these assessments. This temptation should be strongly resisted since the state-of-the-art is highly fluid and today's inference could be voided by tomorrow's study. Where I appear to draw a conclusion, I do so to illustrate the type of insight that one can obtain from a seismic risk assessment. The focus of this paper is on the means rather than the end.

Probabilistic risk assessment is broad in that it attempts to deal with a whole subject but, in another sense, it is quite shallow since many detailed technical issues are not studied. I will not be addressing the traditional seismic design methodology and its vast literature since it is outside the scope of this paper.

Probabilistic risk assessment is concerned with perspective and with allocation of resources. The principal engineering insights of the Reactor Safety Study were that public risk is dominated by accidents involving core melting and that the dominant accident sequences are initiated by transients and small LOCAs rather than by large LOCAs. In my opinion, it is most

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unfortunate that these valuable insights from the RSS have been obscured by controversy over the Executive Summary and comparisons of reactor risks to other societal hazards. The motivation for applying probability risk assessment techniques to seismic design is to gain similar insights. For example, how important is seismic risk vis-a-vis other potential accident sequences? Which are the critical components (e.g., piping, diesel-generators)? Which technologies contribute the greatest uncertainty (e.g., soil-structure interaction, load combinations)? Insights on such questions should guide the focus of regulatory attention and the priorities of a research program.

With respect to the first question on the relative importance of seismic risk, one needs a benchmark. As noted earlier, the RSS found that core melting accidents contributed over 90% of the public risk. In the event of core melting, there was a broad spectrum of public consequences ranging from almost zero to very large, depending upon the mode of containment failure, the prevailing weather, and the size of the local population. The most comprehensive gauge of public risk would be probability versus magnitude of the release to the environment since it would account for loss of containment integrity. For simplicity of expression, I will use the probability of core melting as the gauge. Excluding seismically-initiated events, the RSS estimated that probability to be about $5 * 10^{-5}$ or one chance in 20,000 per reactor-year. I would judge the associated error band to be multiplicative/divisive factors of about 12 which is somewhat larger than that stated in the RSS. Let me emphasize that I do not attach great importance to the precise probability. What is important is that it lies in the range $4 * 10^{-6}$ to $6 * 10^{-4}$/reactor-year and is not infinitesimal (e.g., 10^{-10}/r-y). In order to assure ourselves that specific accident sequences including seismically-initiated ones are not dominant contributors to public risk, we merely have to assure ourselves that each one is less than 10^{-6}/r-y or so which is not beyond the available technology in most instances.

Let us review the elements of the seismic calculation as illustrated in Figure 1. The starting point is the seismic input which will be some probabilistic statement of the free field seismic hazard perhaps a complementary

cumulative distribution function for peak horizontal ground acceleration. This stimulus is translated via transfer functions for the soil-structure interface and structural response to a set of seismic loads on components. These loads are combined with the normal operating loads plus any induced loads due to a seismically-initiated transient or LOCA. By comparing the combined loads to the appropriate fragility curves (conditional probability of failure versus load), one can estimate the probabilities of components' failure. Ideally, one should consider initiating events (viz. LOCA, transient) before the engineered safety features since the former may trigger an induced load which may affect the failure probability of the latter. Finally, these component failure probabilities are combined into a system model in order to estimate the probability of core melting. For the idealized models to be discussed in this paper, this model has been simplified to three elements by considering the component failure probability as a function of the peak ground acceleration; this simplification is illustrated by the large dashed box in Figure 1.

Seismic risk assessment is different from the traditional licensing calculations in several respects. For a risk assessment, one considers a spectrum of earthquake intensities and the Safe Shutdown Earthquake, per se, is not considered except as a benchmark to which components are designed. Likewise, one estimates the probability of component failure. The fact that a component meets code does not preclude its failure at below-code loads and equally exceeding code limits is not automatic failure. Finally, the emphasis is upon realistic calculations to predict risk rather than conservative calculations to protect the public.

I will review briefly four idealized models in order to show the development of thinking and then describe the seismic risk assessment by Pacific Gas and Electric for their Diablc Canyon Plant which is the most comprehensive study to date. Finally, I will mention the NRC-sponsored seismic research program at Lawrence Livermore Laboratory and offer some thoughts on the future use of probability risk assessment in reactor licensing and safety design.

SOME IDEALIZED MODELS

Reactor Safety Study

The Reactor Safety Study[1] adopted a relatively simple model to assess seismic risk as illustrated in Figure 2. The reactor was assumed to be located within the Eastern United States and designed for a Safe Shutdown Earthquake (SSE) of 0.2g. A complementary cumulative distribution function for peak ground acceleration suggested by Hsieh[2] was used; Figure 3 shows probability versus the ratio (r) of applied peak ground acceleration to the SSE value. As may be seen in Figure 3, the probability of exceeding the SSE intensity (r=1) was estimated to be about $6 * 10^{-4}$ or one chance in 1500 per year. The fragility curve for components was suggested by Newmark[3] in which factors such as damping, ductility, response spectrum, and soil-structure interaction were combined[4,5] into a single relationship between failure probability and the ratio (r). As shown in Figure 4, this model predicts component failure probability to be $1.5 * 10^{-3}$ for an SSE intensity. Based upon its experience with other accident initiators, the Reactor Safety Study assumed that core melting would result from the failure of two components in parallel as shown in Figure 2. Some allowance was made for common cause failures so the probability of system failure is $3 * 10^{-5}$ for an SSE intensity; the complete curve is shown in Figure 5. By combining the probabilities of an earthquake and of system failure for a range of ground accelerations from 0.2 to 1.0g (r=1 to 5), the probability of seismically-initiated core melting was estimated to be about 10^{-7} per reactor year.

Seismically-initiated accidents appeared to contribute only 1% to the total probability of core melting. Since they appeared to be non-dominant contributors, a more sophisticated model was not deemed necessary. The only other conclusion of note was that, for this simple model, the seismic risk was dominated by earthquakes whose peak intensities ranged from 0.5 to 1.0g or 2 to 5 times the SSE intensity.

Hsieh-Okrent Model

Since publication of the Reactor Safety Study, Hsieh and Okrent[6] have suggested that three different failure probability distributions should be used to represent different failure paths. Group A included containment collapse, scram failure, to which the above Newmark fragility curve was assigned. Groups B and C were assumed to be more susceptible to failure and were assigned higher failure probabilities as shown in Figure 4. Group B included foundation failure, loss of AC or DC power; the failure probability at SSE intensity was $3.7 * 10^{-2}$. The concern about Group C, which included multiple pipe failures, RHR system failure, was the multiplicity of items in series; the failure probability at SSE intensity was $1.1 * 10^{-2}$. As illustrated in Figure 2, Hsieh and Okrent assumed a system configuration of ten 'components' in series, the failure of any one of which would result in core melting. The probability of system failure depends upon the relative numbers of 'components' from Groups A, B, and C. The range of system failure probabilities is shown in Figure 5. The probability of seismically-initiated core melting was estimated by combining the probabilities of earthquake and of system failure and was found to be in the range of 10^{-4} per reactor-year.

By comparing the above probability to the RSS value for core melting, the Hsieh-Okrent model suggests that seismically-initiated accidents might be the dominant contributors to public risk. This work probably underlies in part Dr. Okrent's dissent in a recent ACRS letter in which he stated:[7]

> "I believe that the philosophy and criteria of Appendix A of 10 CFR 100, and their application by the NRC Staff in setting SSE values, should be reevaluated as part of an early overall reassessment of the current approach to seismic safety design. I believe that the estimates of the contribution of earthquakes to overall nuclear reactor safety risk, as given in the RSS are not without fault, and that seismic contribution to risk is underestimated in that study.
>
> I find the Applicant's estimate of the SSE frequency at the sites of greater than 10^{-4} per year to be unsatisfactorily large, ... I find the proposed SSE of 0.15g marginally acceptable and would prefer that a value of 0.2g be employed..."

The other inference from this model was that the seismic risk appears to be dominated by the range of 0.2 to 0.4g or 1 to 2 times the SSE intensity. There appears to be some double accounting in this model by simultaneously increasing the failure probability due to series systems and using these higher probabilities in a series system. Such faults reflect the limitations and grossness of the assumed system models.

Cornell-Newmark Model

Cornell and Newmark[(8)] have illustrated a slightly more sophisticated model. (Figure 2). Dr. Newmark revised his assessments of component fragility so that the failure probabilities are comparable to Group C of Hsieh and Okrent as shown in Figure 4. More importantly, a more realistic system model is adopted. For the PWR analyzed in the Reactor Safety Study, a dominant accident sequence was initiated by a transient (e.g., loss of offsite power) followed by unavailability of the auxiliary feedwater system (AFWS). Since an earthquake could well trigger a loss of offsite power, AFWS might be the critical system and its resistance to seismic events would determine the magnitude of seismically-initiated risk. A simple model for AFWS is illustrated in Figure 2; there are three parallel sources of water in series with a series/parallel set of three pumps and diesel-generators. The failure probability for this system is shown in Figure 5. By combining this distribution function with that for the earthquake intensity, the probability of seismically-initiated core melting was found to be in the range of $2 * 10^{-5}$/reactor-year.

This model suggests that the seismic contribution to risk, while greater than that stated in the RSS, is comparable to other accident sequences. Again, the dominant risk stems from an intensity range of 1 to 2 times SSE.

Acres Model

This model was published[9] between the draft and final RSS reports. Its format is very similar to the above models, and discussion would not add to this paper; the system failure probabilities derived by the authors are comparable to those of Hsieh and Okrent. However, the paper emphasizes sensitivity studies which are important. Based upon an admittedly idealized model, the authors conclude that uncertainties in the seismic input and in the fragility curves would dominate the uncertainty in seismic risk estimates.

Summary

The conclusion one can draw from these idealized models is that there is a substantial uncertainty in the relative contribution of seismically-initiated accidents to public risk. Although the Reactor Safety Study suggested that seismic risk was a small contributor, the results of Hsieh-Okrent and Cornell-Newmark imply, at a minimum, that the question merits further consideration. While the sensitivity to the assumed fragility curves was hardly surprising, the importance of the system model may not have been fully appreciated by the structural engineering community.

The Cornell-Newmark model probably represents the realistic limit in sophistication for such idealized models. In order to gain valid insights into the critical components and the dominant sources of uncertainty, it is clearly necessary to develop a more comprehensive system model and to consider the capacities of individual components more carefully. Such refinements were the principal developments in the Diablo Canyon study which is discussed below.

DIABLO CANYON STUDY

At the request of NRC, Pacific Gas and Electric prepared a seismic risk assessment for its Diablo Canyon Plant.[10] Since issuance of a Construction Permit, the Hosgri Fault was found near the site and NRC has required PG&E to make some major modifications to the plant for the purpose of raising its seismic qualification level. In Amendment 52, PG&E assessed the public risk

with/without these modifications and concluded that the public risk from seismically-initiated accidents was very small and that the decrement in risk as a result of the modifications would not be significant.

The probability of an earthquake as a function of its intensity at the site was assessed from detailed consideration of the local faulting and of the probable attenuation of seismic energy during its transmission to the site. The complementary cumulative distribution function for peak effective horizontal ground acceleration is shown in Figure 6. PG&E and its consultants believe that there is a physical limitation to the amount of shaking that ground can sustain and hence to the potential accelerations. In their opinion, the probabilities shown for accelerations in excess of 0.75g are unrealistic and are shown merely to indicate values used in the risk analysis. The original SSE intensity was 0.4g and its exceedance probability is about 10^{-4} per year.

The response of the plant was assessed by modifying the event/fault trees prepared by the Reactor Safety Study for the Surry plant. These modifications were necessary to adequately handle an external event with potential common cause failures and to reflect the Diablo Canyon design. For each component or structure, conditional failure probabilities were generated as a function of peak ground acceleration as illustrated in Figure 7. They were established from stress analyses, subjective estimates by cognizant engineers, and testing data. Two types of functions were used; the ramp function for components which are qualified by analysis, e.g., piping, structures, and the step function for equipment which are qualified by testing, e.g., mechanical, electrical. As illustrated in Figure 7, tails were added to both these failure distributions for low accelerations to account for the non-zero failure probability. The results were found to be fairly insensitive to shape of most of these tails for reasons which will become clear later. The preparation of event/fault trees and the generation of the failure probability distributions have been described in more detail elsewhere.[10, 11, 12]

For each discrete value of the peak ground acceleration (viz. 0.3g, 0.4g...1.7g), the corresponding failure probabilities for each component were introduced into the fault/event trees in order to calculate a conditional probability of core melting and release of radioactive material to the environment. These probabilities were combined with those for the earthquake in Figure 6 to obtain an absolute probability of core melting. By working with conditional failure probabilities in the event/fault trees, proper account is taken of dependencies between components due to the common cause nature of the earthquake. The probability of seismically-initiated core melting was estimated to be about $8.5 * 10^{-6}$ and $8.3 * 10^{-6}$ per reactor-year before and after the Hosgri modifications. PG&E extended this work to estimate the impact upon man by calculating, in a manner comparable to the RSS, radiation doses to various body organs for people located at the site boundary and within local communities.

There are several interesting observations to be drawn from this analysis:

- For what it is worth, seismically-initiated accidents appear to be significant but non-dominant contributors to public risk.

- The dominant intensity range for seismic risk is 0.9 to 1.4g or 2 to 4 times the original SSE intensity. With this perspective, it is unsurprising that the estimate of public risk was essentially unchanged by the Hosgri modifications which basically upgraded the seismic qualification criteria for certain equipment from 0.4 to 0.75g. The validity of this observation is questionable for several reasons. First, as emphasized by PG&E, the achievement of ground accelerations in excess of 0.75g is probably physically impossible. Second, the capacity of components to absorb high loads was probably underestimated since plastic yielding was

ignored. If the contribution to risk from this higher acceleration range was reduced, there would be comparable reduction in the total seismic risk estimate.

- The dominant accident sequences were a) loss of offsite power followed by unavailability of the auxiliary feed-water system and b) a small LOCA followed by unavailability of emergency coolant injection and containment spray and heat removal. A key component in each sequence was the turbine building which houses the diesel-generators and 4 kV switchgear; its failure would probably cause the loss of all AC power. Most components, (e.g., piping), in the engineered safety features were not a factor in the dominant accident sequences. For this reason, the risk results were insensitive to the selection of tails for the most of the failure probability curves.

When weighing the above observations, the limitations of the PG&E analysis should be recognized. First, the system model developed for the Surry plant was basically used. The error involved is probably small but it introduces uncertainty in the results. Second, the failure probability of a component was represented mathematically as a function of peak seismic acceleration. Potential induced loads due to a LOCA or transient could not be properly incorporated. Third, the analysis appears quite conservative in some respects. For example, the seismic stress in components, structures, etc. was assumed to be a linear function of ground acceleration with no allowance for ductility. Further, although the seismic input was characterized solely by peak ground acceleration (i.e., frequency content was not varied), the failure probabilitites for specific equipment were based upon the most adverse response spectrum for that equipment (i.e., an envelope approach). These limitations may not have affected the risk estimates since the critical component was the turbine building but they introduce uncertainty in the results. Despite these limitations, I believe that PG&E performed a very instructive analysis which broke

a lot of new ground. Many of the acknowledged limitations stemmed from time and budget constraints and from the fact that the study was part of an NRC licensing docket.

SUMMARY AND OBSERVATIONS

The foregoing discussion shows the progress in seismic risk assessment during the past four years. The objective of such studies is to gain insights into the relative importance of seismically-initiated accident sequences and to identify the more significant failure paths. Clearly, the state-of-the-art in seismic risk assessment has not reached the point where one has sufficient confidence in its inferences to modify regulatory criteria. For example, more sophisticated models will probably not support the conclusion that the high acceleration ranges (2 to 4 times SSE) dominate seismic risk. On the other hand, the conclusion from the Diablo Canyon study that AC power supply is the critical element as opposed to piping or other mechanical components will probably stand the test of time. This conclusion has also been reached in an unpublished risk assessment of another commercial LWR. In this study, seismically-initiated accidents were a dominant contributor and also stemmed from loss of AC power. [The auxiliary building was a common cause failure since all power lines including those from diesel-generators passed through it]. Such a conclusion, if valid, could have a significant impact upon current regulatory requirements. For example, if common cause failure of AC power supply dominates, the arguments over the correct methodology for combining seismic and induced loads may have marginal significance with respect to public risk. However, risk assessment methodology would be helpful in resolving this vexing issue. Similarly, a computer methods error in piping calculations led the Nuclear Regulatory Commission to shut down five plants since the piping in question did not meet current regulatory requirements. The rationale of this decision can be challenged on several grounds but, in addition, if the inference from the Diablo Canyon and other studies is valid, it would not appear to have been justified on the basis of public risk.

The costs of seismic design requirements have been estimated by an NRC-sponsored study to be about $10M per 0.1g increment assuming the seismic design requirements remain fixed and unchanged during design and construction.(13) Retrofitted changes, of course, substantially amplify such costs. Whether this is an effective application of funds and whether it is being spent on the right equipment could be strongly influenced by an improved risk perspective. The Seismic Safety Margin Research Program sponsored by NRC at Lawrence Livermore Laboratory is developing realistic seismic models which will be used in sensitivity studies to identify sources of conservatism in the Standard Review Plan seismic safety requirements. Such a systematic approach should also provide insights into the dominant accident sequences and critical components. I believe that this ambitious program has the potential for properly focussing industry/regulatory efforts upon the important seismic issues. I encourage industry support.

Much of the contentiousness between industry and NRC derives from a lack of concensus on which issues are important and what is an acceptable level of risk. The existing regulatory requirements appear to be in poor congruence with our best understanding of the risks to the public. It is questionable whether the current allocation of resources devoted to nuclear power plant safety is cost- or even safety-effective. In this respect, I am encouraged that NRC has initiated a value-impact analysis of the Standard Review Plan. Value is defined as public risk reduction. Unfortunately for Phase 1, impact has been narrowly defined in terms of NRC regulatory costs only. I would encourage NRC to adopt a broader definition to include the total costs imposed upon the utility and ultimately the public. I would hope that industry, through AIF, would cooperate by estimating realistic cost data for meeting specific regulatory criteria.

Another role of probabilistic risk assessment would be to facilitate communication between the parties since it structures and disciplines the thinking process. One positive fallout from the Three Mile Island accident

would be a renewed dialog between industry and NRC to utilize probabilistic risk assessment judiciously but effectively to focus upon the significant issues. In my opinion, both parties need to make some changes. NRC needs to demonstrate a willingness to unwind the ratchet on some issues having negligible contribution to public risk (e.g., turbine missiles). Industry needs to desist in submitting unrealistic risk assessments (e.g., probabilities of 10^{-10} per year or smaller). Such submittals support the skeptics within NRC who believe that all such analyses are 'number games'. Overall, I believe that it is timely to step back and comprehensively review all regulatory requirements for their contribution in mitigating public risk. Too much of the licensing process has degenerated into a stylized debate over the details of a calculation while its relevance to public risk is overlooked.

The above process would be aided by an agreement on a realistic goal. Unrealistic goals (e.g., less than 10^{-7}/year of exceeding 10 CFR 100 dose guidelines) encourage unrealistic analyses, a false sense of security, and open oneself up to criticism when there is a failure. While I agree with Hal Lewis that the precise number is unimportant, I think that a general concensus on a goal is very desirable. The experience of the Reactor Safety Study and elsewhere suggests that an overall core melting probability of less than 10^{-5} to 10^{-6} per reactor-year is an illusion since some unconsidered common cause event will void one's analysis. The proper emphasis should be on demonstrating that we have truly achieved this realistic goal. Probabilistic risk assessment can assist in improving reactor safety and in stabilizing the licensing process. It should supplement rather than supplant the current deterministic criteria. We need to learn to use this powerful tool in a way which provides benefits to both NRC and industry.

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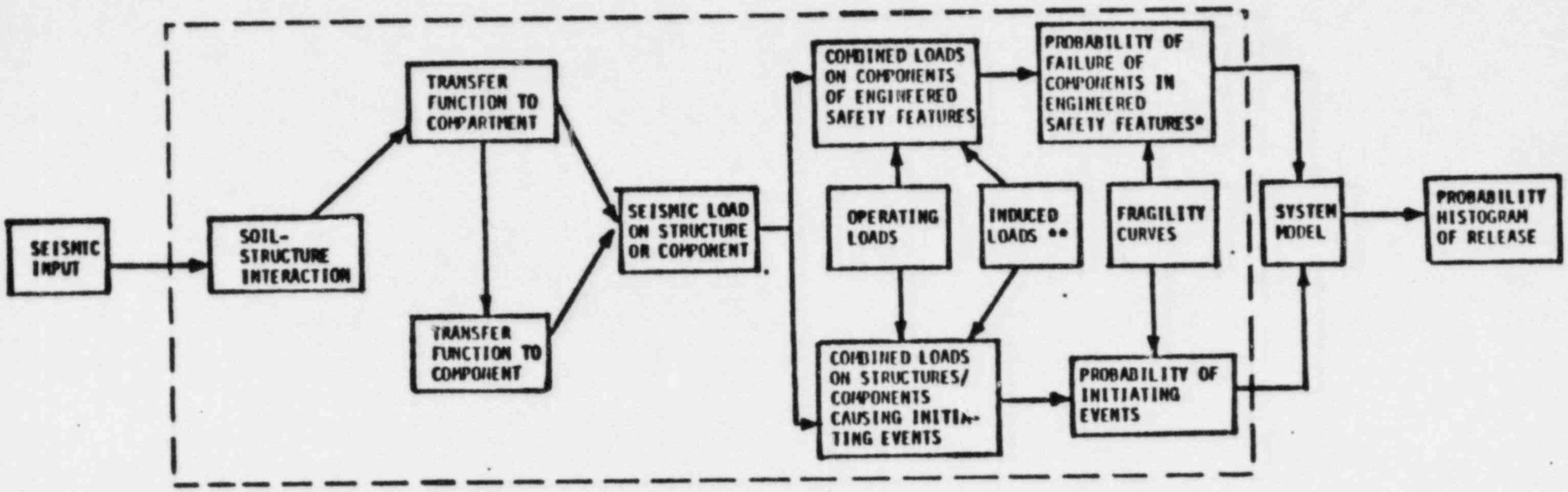

* INCLUDES CONTAINMENT
** FUNCTION OF INITIATING EVENTS

Figure 1 CONCEPTUAL MODEL FOR SEISMIC RISK ASSESSMENT

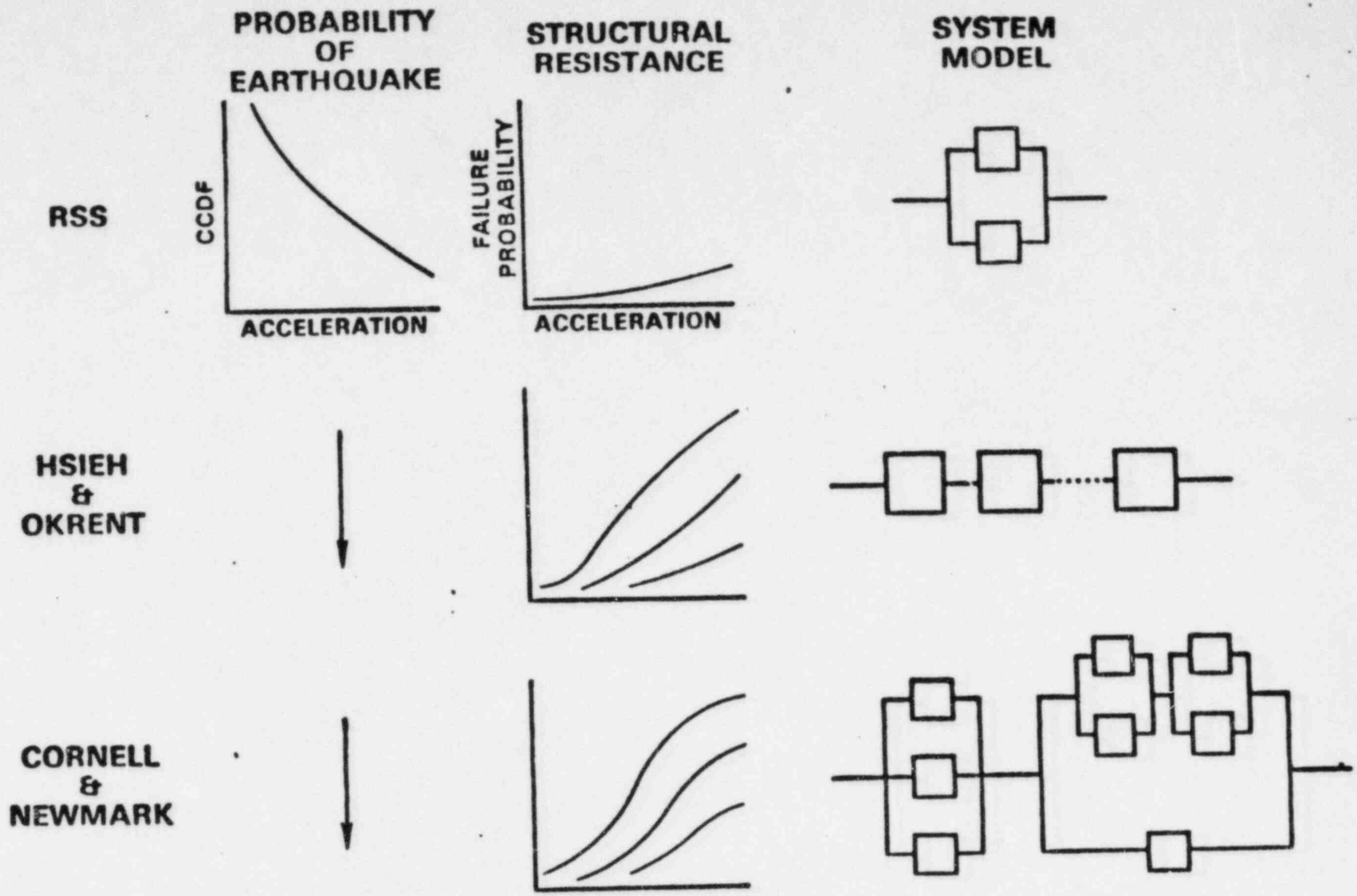

Figure 2 IDEALIZED MODELS FOR SEISMIC RISK ASSESSMENT

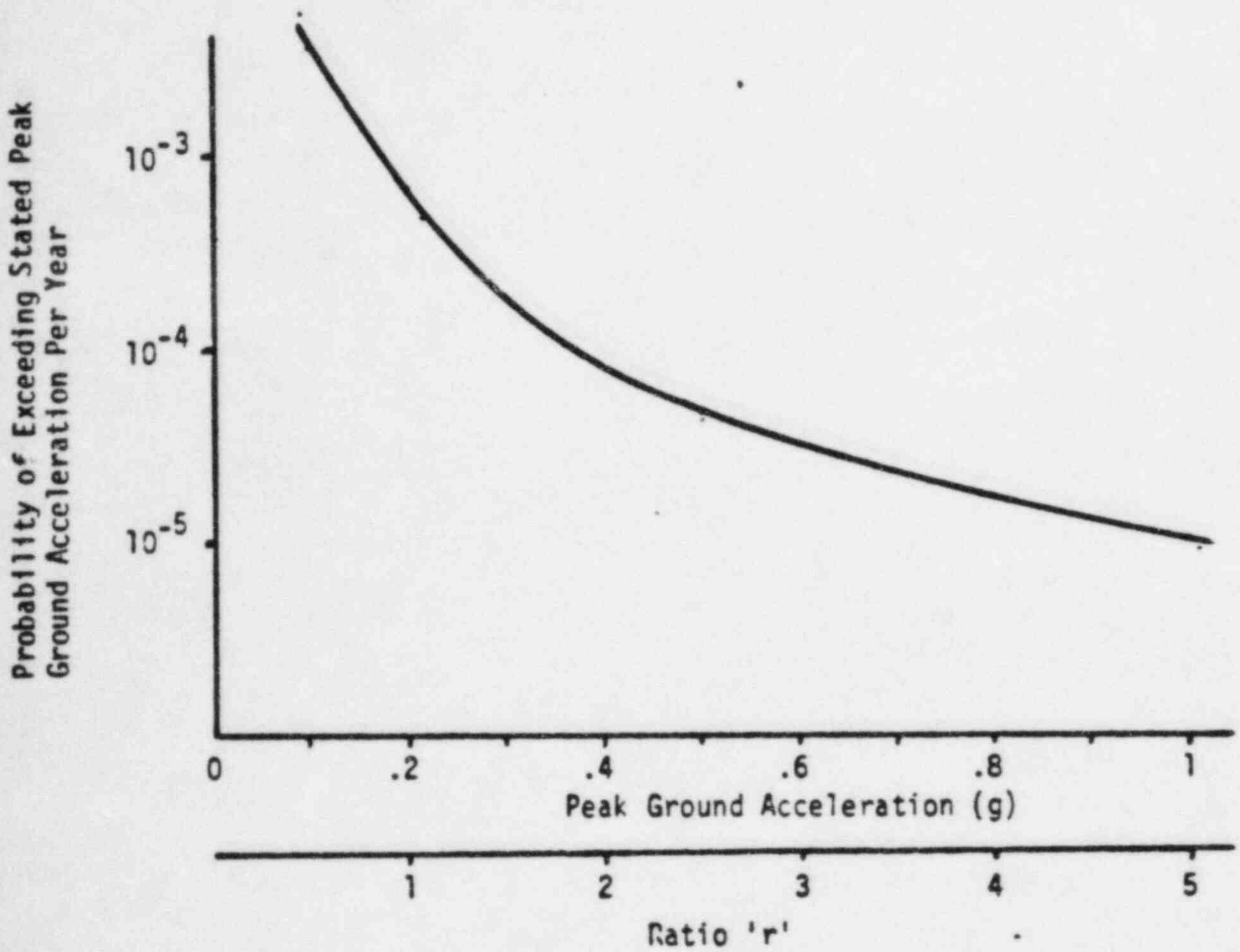

Figure 3 SEISMIC INPUT FOR TYPICAL EASTERN U.S. SITE (Hsieh Reference 2)