Development of Probabilistic Seismic Failure Relationships of Nuclear Components for the SSMRP

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SUMMARY

The Seismic Safety Margins Research Program (SSMRP) is an NRC-funded multi-year program directed towards estimating the conservatism in the Standard Review Plan seismic safety requirements with the ultimate goal of developing improved seismic requirements. As part of this program, calculations of the seismic risk from a typical commercial nuclear reactor are being made. These calculations require a knowledge of the probability of failure (fragility) of all safety related components in the reactor system which actively participate in the hypothesized accident scenarios.

Component failure is defined as either loss of pressure boundary integrity or loss of operability. Failure (fragility) is characterized by a cumulative distribution function which describes the probability that failure has occurred given a value of loading. In the context of the SSMRP, loading may be local spectral acceleration, local zero period acceleration or internal force resultant such as moment, depending on the component and failure mode under consideration. Contrary to previous work, fragility is related to the appropriate local response, rather than being related directly to free-field peak acceleration.

The initial determination of the necessary fragility functions has been completed. As a first step, all components identified in the reactor fault tree analyses have been grouped into 37 generic categories, and fragility functions have been determined for each category. For example, all motor operated valves located on piping with diameters between 2-1/2 and 8 inches are placed into a single generic category, and similarly, all motor control centers have been placed into another generic category. All piping, tees, elbows, butt welds, and reducer sections have been placed into one generic category, and scaling factors (dependent on size, material and temperature) are utilized to relate the individual piping components to a single master fragility curve. The paper describes the 37 generic categories, and discusses the adequacy and limitations of such a categorization scheme.

Fragility functions for the 37 generic categories have been developed based on a combination of design analysis reports, experimental data and an extensive expert opinion survey. In this survey, questionnaires were sent to over 250 recognized specialists in the nuclear industry (representing nuclear power system vendors, utilities, testing laboratories, nuclear component manufacturers, architect-engineers, and consultants) which resulted in 147 detailed responses covering (to varying degrees) virtually all the 37 generic categories. The responses to the questionnaires identified various failure modes as well as the failure percentiles as a function of loading. Both our experience with the survey and the statistical methods used to combine multiple respondent data and to combine failure data for multiple failure modes are outlined in the paper.

The experimental data utilized in developing fragility curves were obtained from the results of component manufacturers qualification tests, independent testing lab failure data and data obtained from the U.S. Corps of Engineers SAFEGUARDS Subsystem Hardness Assurance Program. These data were critically examined for applicability and then statistically combined with the expert opinion survey data to produce the final fragility curves for the 37 generic component categories.

Finally, ongoing work (namely Phase II of the SSMRP) aimed at improving and further benchmarking the preliminary fragility functions obtained to date is described.

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1. Introduction

As part of the Seismic Safety Margins Research Program (SSMRP), calculations of the seismic risk from a typical commerical reactor are being made (See Smith [1]). These probabilistic risk calculations use the same event tree/fault tree methodology utilized in the earlier WASH-1400 study, Rasmusson [2], except that the primary concern of the SSMRP is earthquake induced loading and the possible compromise of component redundancy due to the pervasive nature of earthquake excitation.

As part of these risk calculations, event and fault trees were constructed which relate potential radioactive release to failure of essential components in the reactor Engineered Safety Features (eg., the Emergency Core Cooling System, the Auxiliary Feedwater System, etc.). Over 2300 basic seismically-induced component failures were identified on the event/fault trees. In order to perform risk calculations, it was necessary to develop probabilistic failure criteria for each of these 2300 components. This paper describes the set of probabilistic fragility (failure) relations which were developed for the initial (Phase I) SSMRP demonstration risk calculations.

2. Categorization of Components

The first step in the development of the necessary fragility relations was to reduce the problem to a manageable size. This was done by grouping the 2300 components into 37 generic categories shown in Table I. For each generic category, a single fragility relation was developed which was then used for all components in that category. The one exception to this is the generic category for piping. This category includes pipes of various sizes as well as elbows, miters, butt welds, and reinforced and unreinforced branches. Each of these piping system elements are related to a single master fragility curve through a scale factor which depends on size, material (carbon or stainless steel) and temperature. Over 100 different scale factor combinations were developed.

3. Data Sources

Actual experimental data on failure of components as a function of local base acceleration are scarce. The type of data most commonly available results from qualification tests in which the component is experimentally shown to function as designed for a prescribed acceleration spectrum input. While such data do provide a lower limit to the fragility level, it is difficult to extrapolate from this data to higher response levels. One notable exception to the lack of actual fragility data was the data obtained in the U.S. Army Corps of Engineers SAFEGUARDS program. This 11 year program, conducted as part of a missile-site hardening effort, included tests as both mechanical and electrical components. The items tested were "off-the-shelf" and were typical of components used in commercial reactors in the late 1960's, and the results are thus directly applicable to the ZION power plant. Sixty-four test programs involving shaker table tests of approximately 300 items were conducted. Excitation consisted of sine beat pulse tests, selected to fit a prescribed acceleration spectrum. Equipment function was monitored during the test. Thus these tests were truly tests of fragility with respect to both functional and structural failure. Typically components were tested to over 15 g peak acceleration. Out of the nearly 300 reports generated in the SAFEGUARDS program, 64 were found to be directly applicable to components needed in the SSMRP. In particular, these data were the only data available for electrical components, and thus all our electrical component fragilities are derived from this source.

A second source of information available was the generic design analyses performed by Westinghouse and various component manufacturers for components used in the Zion plant. In these analyses, the component was assumed to be excited by a base acceleration corresponding to a prescribed design spectrum. Then an analytical solution for the stresses or loads in the component was obtained. From these analytical solutions we obtained the acceleration at failure by extrapolating the stresses to our estimate of the ultimate stress capacity using a procedure due to Newmark [3]. In this procedure, the acceleration at failure is determined from the relation:

$$A_F = A_D\, F_S\, F_\mu \qquad (1)$$

where A_D = design peak acceleration
F_S = factor accounting for ultimate load capacity
F_μ = factor accounting for the inelastic energy absorption capability.

The factor accounting for the ultimate load capacity is computed from

$$F_S = \frac{\sigma_{ult} - \sigma_{dead}}{\sigma_{seismic}}$$

where σ_{ult} is the ultimate load or stress capacity, σ_{dead} is the static load due to weight, pressure, thermal, etc., and $\sigma_{seismic}$ is the peak load induced by the seismic excitation. Thus F_S scales up the design acceleration to the failure acceleration, assuming all loads (or stresses) are calculated by a linear elastic analysis, since the peak load (or stress) is proportional to peak acceleration.

Before failure occurs, however, a significant amount of inelastic deformation (and hence energy absorption) takes place. In this inelastic response range, the stress increases much more slowly than the peak acceleration. Hence, the actual acceleration at failure is much higher than that predicted by the product $A_D F_S$ alone. This additional acceleration capacity is accounted for by the ductility factor F_μ. This ductility factor was introduced by Newmark [4] and is a function of both the ductility of the component and the component damping. The ductility μ is usually estimated on the basis of engineering judgment and a knowledge of component construction details.

The statistical distribution of the acceleration at failure (the fragility relation) is obtained by assuming that the factors F_S and F_μ are lognormally distributed random variables. This choice of distributional form has been found to be appropriate in several studies (References [5] to [7]) and also results in considerable computational convenience. If M_S and M_μ denote the median values of F_S and F_μ, and if β_S and

β_μ denote the standard deviations of the natural logarithms of the variables F_S and F_μ, then by the multiplicative property of log-normal random variables, the median and log-standard-deviation of the acceleration at failure are given by, respectively,

$$M_A = A_D \, M_S \, M_\mu \qquad\qquad \beta_A = \sqrt{\beta_S^2 + \beta_\mu^2}$$

These two parameters completely define the distribution of acceleration at failure. Values of the uncertainty in the factors F_S and F_μ are estimated from data, analysis or engineering judgement depending on the component. Complete details for any particular component may be found in Campbell [8]. While this method of estimating fragility of components is not based directly on failure tests, it does allow an estimate of failure incorporating experimental determination of ultimate strength, weld and connector ductilities etc., and the choice of the uncertainty factors β_S and β_μ may be made so as to reflect our confidence (or lack thereof) in the analysis. This measure of confidence can then be propagated through the entire SSMRP calculational scheme, and its effect on the final prediction of radioactive release probability can be determined.

The final source of information on fragility of components was an expert opinion survey performed in the spring of 1980. In this survey, a carefully worded questionnaire was mailed to several hundred well-known specialists in the nuclear industry. These individuals were selected from the NSSS vendors, architect/engineering firms, consultants to the nuclear industry and from the ranks of colleges and universities. In each case, the individual was asked to respond only for those components for which he felt a high degree of expertise. For each component, the respondent was asked to provide:

a) the three lowest (weakest) failure modes
b) the appropriate response quantity for each mode (eg., peak acceleration, spectral acceleration at some frequency and damping or force resultant, etc.)
c) the response values at 10%, 50% and 90% probability of failure.
d) the primary source of his information (ie., experience, test data, etc.).

The expert opinion responses covered virtually every category of component needed for Phase I of the SSMRP, with 147 detailed responses being returned. Comparison of responses from different experts for the same component showed, in general, surprisingly good agreement. Inasmuch as the expert opinion responses were provided at three probability levels, it was necessary to develop a method of statistically combining them.

The procedure adopted was based on a combined least squares and nested analysis of variance approach as described in detail in George [9]. In this approach, each failure mode (for each component) is treated as independent, and a single fragility curve is developed for each mode based on the responses of all experts who identified that particular failure mode. The statistical model used was

$$A_{ijq} = A_q + T_j + E_{ijq} \qquad (2)$$

where i refers to the i^{th} expert, q denotes the fractile level (10%, 50% or 90%) and j denotes the group number. Based on our subjective evaluation of the expert opinion responses, we combined different expert's responses into a common group if we had reason to believe that these experts were all referring to the same type of component within the broad generic category being considered. The use of the nested analysis of variance procedure then allowed us to identify the total variance from

$$\sigma^2 = \hat{\sigma}^2 + \sigma_T^2 + \sigma_E^2$$

where $\hat{\sigma}^2$ = inherent uncertainty in each individual expert's fragility estimate

σ_T^2 = uncertainty resulting from the different groups of components within the generic category

σ_E^2 = uncertainty between experts whose data were combined in the same group.

By this procedure, we can identify whether or not the generic categories selected (as shown in Table 1) are too broad, for if σ_T^2 is the major contributor to the total variance, then this is an indication that the generic category should be further subdivided into two or more separate generic categories.

In the analysis of equation (2), a weighted least squares approach was used in estimating $\hat{\sigma}^2$. The weights were assigned as a product of two factors: a factor for presumed expertise of the specialist providing the opinion and a factor for source of his opinion. For the expertise factor, a value of 3 was given for any NSSS Vendor, Architect/Engineer or component manufacturer, a value of 2 for a military expert, and a value of 1 for all others. For the data source factor, a differentiation was made between pressure boundary failures and functional failures. For the former, values of 4, 3, or 1 were given for the source being test data, analysis or opinion, respectively. For the latter, the corresponding values assigned were 4, 2, or 1. This differentiation between pressure boundary failure and functional failure was made to reflect a lesser degree of confidence in analytical methods for predicting functional failure.

It is at this point that data from the other sources (the SAFEGUARDS fragility data and the generic component design analyses previously described) were incorporated. These additional data were treated as independent expert opinions, with weight factors assigned based on our subjective evaluation of the quality of the data.

The final step in the development of a single fragility curve for a given generic category was to combine the fragility estimates (obtained from equation 2) for each independent failure mode. This combination of modes was performed using the relation

$$F(r) = 1 - \Pi[1 - F_i(r)]$$

where F(r) is the single combined mode fragility curve and $F_i(r)$ are the fragility curves derived for the n failure modes identified for the generic category. This is the statistical union of failure modes and, in effect, produces an effective fragility curve which is nearly a lower bound.

Finally, it should be noted that, for Phase I of the SSMRP, the total uncertainty in the fragility relation was used for the radioactive release calculations. However, in

Phase II, it is planned to propagate inherent (non-reducible) and systematic (modeling) uncertainties separately. In equation 2, the variances σ_T^2 and σ_E^2 are a measure of the systematic uncertainty, and thus this statistical model will permit a differentiation between the two sources of uncertainty.

4. Results

To illustrate the results of the fragility development, a number of typical log-normal fragility relations are presented in Table 2. In each case, the value of the median response at failure M and a value of β, the standard deviation of the natural logarithm are given. The probability of the response A exceeding the strength is then given by

$$P_{failure} = \phi\left[\frac{\ln (A/M)}{\beta}\right]$$

where $\phi(\ )$ is the standard normal cumulative distribution function available in any statistics text.

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TABLE 1 Generic Categories for Components

MECHANICAL

- Reactor core assembly
- Reactor pressure vessel
- Pressurizer
- Steam generator
- Reactor coolant pump
- Piping
- Large vertical storage vessels with formed heads
- Large vertical storage tank - flat bottom
- Large horizontal vessels
- Small-medium vessels and heat exchangers
- Large vertical centrifugal pumps with motor drive
- Large vertical pumps
- Motor driven compressors
- Large motor operated valves
- Large relief and check valves
- Small valves

ELECTRICAL

- Horizontal motors
- Generators
- Battery racks
- Switchgear
- Dry transformers
- Control panels and racks
- Auxiliary relay cabinets
- Local instruments
- Motor control centers
- Computers
- Light fixtures
- Invertors
- Cable trays
- Circuit breakers
- Relays
- Ceramic insulators
- Communications equipment

MISCELLANEOUS

- Air handling units
- Instrument racks
- Duct work
- Hydraulic snubbers

Table 2 SELECTED FRAGILITY RELATIONS

Category	Parameter	M_A	β_A	Lowest Failure Mode
PWR Reactor Vessel	SA at 4 Hz and 5%	3.72	0.18	Nozzle Support Failure
Pressurizer	SA at 8 Hz and 5%	2.00	0.39	Anchor Bolt Failure, Support Failure
Steam Generator	SA at 8 Hz and 5%	4.00	0.27	Support leg buckling and failure
Coolant Pump	SA at 4Hz and 5%	2.64	0.34	Failure of connection to support leg
Diesel Generator	A	0.65	0.33	Control system malfunction, oiling system failure
Large Motor Operated Valve	A	6.30	0.60	Electrical failure, actuator binding
6-in. Pipe, Carbon Steel	M	2.44	0.38	Butt Weld Failure at 70°F
29-in Pipe, Stainless Steel	M	203.3	0.38	Straight Pipe (2.5 in thk) at 600°F
Motor Control Center	A	7.70	0.73	Circuit breaker trips
Cable Trays	SA at 4 Hz and 5%	2.23	0.39	Weld and support failures
Control Panels and Racks	SA at 12 Hz and 5%	3.66	0.83	Instrument failure
Ceramic Insulators	A	0.17	0.39	Breakage

*SA = Spectral Accel in g's; A = Peak Accel. in g's; M = Vector Moment in in-lbs x 10^6