Fragility and Hazard Aspects of the Chinshan Seismic PRA

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ABSTRACT

This paper provides an overview of the methodology and the results of the fragility and hazard portions of the seismic Probabilistic Risk Assessment (PRA) and the containment overpressure fragilities at the Chinshan Nuclear Power Plant in Taiwan. The Chinshan seismic PRA involves the generation and combination of the seismic hazard at the site, the fragilities of equipment and structures and the dominant accident sequences leading to core damage. The seismic fragilities and the hazard study and the containment overpressure fragilities have been completed at this time, and the results of the risk analysis will be available at the time of the SMiRT conference. The seismic fragility and hazard results provide basic insights into the following:

- Dominant Contributors to Seismic Risk
- Seismic Hazard Level for the Northern Taiwan Region
- Fragility Levels for Chinshan Equipment and Structures

The fragilities for the postulated containment overpressure failure modes, along with their estimated leak areas and the results of the core damage analysis, are used to determine the effect of an accident on public health.

BACKGROUND

The Republic of China Atomic Energy Council (ROCAEC) is conducting a probabilistic risk assessment of the Chinshan nuclear power plant. The Chinshan plant is a 2 Unit 604 MWe BWR plant that went on-line in the 1978-79 time frame. The plant was designed to a 0.3 g peak ground acceleration Safe Shutdown Earthquake. Dynamic analysis was performed on all of the Category I structures and conservative floor response spectra were generated for all elevations. In general, the equipment and structures at Chinshan were designed for significant seismic loads and their corresponding seismic fragilities reflect a conservative design basis.

METHODS

Seismic Fragility Evaluation

The methodology for evaluating Chinshan seismic fragilities of structures and equipment is documented in (Hardy and Campbell 1983, Kennedy and Ravindra 1984, Ravindra and Kennedy 1983, Ravindra, et al 1987). Seismic fragility of a structure or equipment item is defined as the conditional probability of its failure at a given value of the seismic input or response parameter. Seismic fragilities are needed in a probabilistic risk analysis (PRA) to estimate the conditional probabilities of occurrence of initiating events (i.e., large LOCA, small LOCA, RPV rupture) and the conditional failure probabilities of different mitigating systems (e.g., safety injection system, residual heat removal system, and containment spray system).

The objective of fragility evaluation is to estimate the peak ground acceleration or spectral acceleration capacity of a given component. This capacity is defined as the peak ground or spectral acceleration value at which the seismic response of a given component located at a specified point in the structure exceeds the component's resistance capacity, resulting in its failure. Because there are many variables in the estimation of this acceleration capacity, component fragility is described by a family of fragility curves; a probability value is assigned to each curve to reflect the uncertainty in the fragility estimation (Figure 1).

For selected levels of the seismic hazard, soil-structure interaction (SSI) and structure response analyses are typically performed to estimate median level response for structural members and equipment support locations. These probabilistic response analyses are performed on a representative structure to provide quantitative guidance on the conservatism introduced in the design seismic analysis procedure. This is an essential aspect of the seismic PRA because of the site conditions and the conservatism introduced in design analysis procedures. An estimate of median response conditional on the occurrence of an earthquake of a specified peak ground acceleration is the end product. Such estimates more easily permit realistic evaluations of structure and equipment fragilities.

Seismic Hazard

The Chinshan site-specific ground response spectrum was developed for the ROCAEC by National Central University in Taiwan. The Chinshan site is located on a rock site. However, the rock is soft with a shear wave velocity of 1400 feet per second. A total of 162 strong motion accelerograms recorded on hard sites (hard soil and rock) in Taiwan area were used for developing the spectrum. Baseline and instrument corrections were applied to each of the 162 records. In addition to the site-specific spectrum, a set of uniform hazard spectra (UHS) was developed for the Chinshan site. Three different spectral acceleration attenuation equations (Kanai, Joyner and Boore, and Campbell) were considered in developing the Chinshan seismic hazard curves. These hazard curves were developed at discrete frequencies at 0.75, 1.0, 2.0, 5.0, 7.0, 10.0, 20.0, and 25.0 Hz. A typical Chinshan seismic hazard curve based on a spectral acceleration equation is given in Figure 2. From these seismic hazard curves, the Chinshan uniform hazard spectra with a specified annual probability of exceedance were developed. A comparison of Chinshan site-specific ground response spectrum with the 10,000 year return period uniform hazard spectrum is given in Figure 3. Both spectra are anchored to a 1.0g spectral acceleration at 10 Hz frequency.

Containment Internal Pressure Fragility Evaluation

The internal pressure fragility of the containment structure is defined as the conditional probability of its failure for a given level of internal pressure. This internal pressure can occur as a result of an accident involving core damage. The fragilities for the postulated containment structure failure modes along with their estimated leak paths and leak areas are computed to assist in defining the risks involved with any of the postulated overpressure scenarios.

Failure of the containment structure is defined as incipient leakage and release of contents into the atmosphere. A variety of potential failure modes exist. The fragility evaluation provides the distributions of containment internal pressure capacities as limited by these failure modes. The capacities are based upon actual ultimate strengths of the structural elements, taking into consideration as-built conditions and material properties. Containment structures are typically designed using stresses predicted by elastic analysis. Thus, original design basis loads are of limited value in the fragility evaluation since extrapolation out to failure levels is inappropriate. Consequently, estimation of the forces acting on the containment at internal pressures causing failure is performed using more appropriate methods during the fragility evaluation. To account for uncertainties in the element strength and analytical modeling the fragility of each containment failure mode is described by a family of fragility curves.

RESULTS

Containment Overpressure Fragilities

The primary containment system of the Chinshan Mark I BWR consists of the steel constructed lightbulb-shaped drywell vessel and the doughnut-shaped suppression chamber (torus). The drywell and the torus are connected by a set of eight vent pipes. Major penetrations on the drywell vessel are that due to the personnel air lock, equipment hatches, and the vent pipes. The design temperature and the internal pressure of the primary containment system are 340 degree Fahrenheit and 56 psig, respectively.

Nonlinear analyses were performed for the drywell and the torus under the high temperature and high internal pressure associated with a major accident. The critical areas of the Chinshan primary containment system were found to be located at the torus shell, the bolted flange connection of the drywell head, and the drywell shell at the knuckle region. The internal pressure capacities of these failure modes were determined to be 170 psi, 190 psi, and 195 psi, respectively. The uncertainty associated with these failure modes are about 0.2.

Median-Centered Seismic Response

A probabilistic soil-structure interaction (SSI) analysis following the SMACS methodology (Johnson et al 1981) was performed for the Chinshan combination structure. Variabilities considered in the analysis are variations in the earthquake excitation, uncertainties in the soil shear modulus and damping values as well as structure frequencies and dampings, and uncertainty in the soil-structure interaction analysis. The key elements in the SSI analysis are: specification of the free-field ground motion, determination of the foundation input motion, determination of the foundation impedances, calculation of the dynamic characteristics of the fixed base structure, and analysis of the coupled soil-structure system. An ensemble of thirty earthquake time histories matching the Chinshan mean and mean plus one standard deviation spectra were used as the input ground motion. As a result of this probabilistic soil-structure interaction analysis, the median centered structure response as well as the instructure response spectra were generated. The Chinshan component fragilities were developed based on these median response. The resulting median centered spectra were found to be much lower (often a factor of 2 or more) than the conservative spectra generated for the original plant design basis.

Chinshan Civil Structures Seismic Fragilities

The civil structures included in the Chinshan seismic fragility study are: the combination structure, the emergency pumphouse, the condensate storage tank, and the diesel fuel oil storage tank. The combination structure provides structurally integrated enclosures and shielding to accommodate the primary containment and reactor auxiliary systems (reactor section), control room area, radwaste area, and diesel generators. The primary containment system consists of the lightbulb-shaped drywell and the doughnut-shaped pressure suppression vessel. The two field erected storage tanks were found to have low fragilities among the Chinshan civil structure evaluated.

Chinshan Equipment Component Seismic Fragilities

Critical NSSS and balance-of-plant equipment necessary for mitigating core damage during and after a seismic event were identified. Two methods were used in assessing equipment component seismic fragilities related to overall risk:

1. Seismic margin screening criteria (Ravindra, et al, 1987)

2. Analytical evaluation

The seismic margin screening criteria separates components which have very high seismic capacities that do not warrant a detailed fragility analysis from components which will contribute to overall plant risk. Thus, additional effort is spent evaluating the contributors rather than the high capacity components which contribute little to plant risk.

The criteria used in screening out the Chinshan equipment components having high seismic capacities is a median ground acceleration capacity greater than 3 g's. This value is selected based on discussions with the system analysts whereby components with median ground acceleration capacities greater than 3 g's would have negligible contribution to the overall risk based on the results of both the Kuosheng and Maanshan PRA's. Well over half the equipment components evaluated for the Chinshan Nuclear Power Plant were screened out using the seismic margin screening criteria. The Chinshan equipment components screened out as having a high seismic capacity were primarily the mechanical equipment and selected electrical components which had no relay functionality concerns. The remaining components were analytically evaluated to determine their fragility curves.

Low Capacity Components

Components determined to have relatively low median ground acceleration capacities contributing significantly to overall risk to the Chinshan Nuclear Power Plant are listed in Table 1. The table identifies the critical failure modes for each component. As can be seen from Table 1, a large portion of the dominate contributors to plant risk are electrical components with relay chatter or functional failure modes.

CONCLUSIONS

The Chinshan plant is located at a high seismicity area with a design SSE level at 0.3g. The plant is relatively old and the equipment qualification information was scarce. More reliance was placed on generic and earthquake experience data in the process of screening out high capacity components. As a result, the project resources were concentrated on the dominant contributors to the plant risk. The dominate contributors continue to be electrical equipment relay chatter and functional failure modes as seen in past PRA's. Operator actions will be required to reduce this potential risk. It was demonstrated in the Chinshan seismic fragility study that the PRA methodology is quite useful even when applied to a plant which is located in a high seismic site and has very little seismic qualification documentation.

Majority of the Chinshan equipment and structures have been demonstrated to have very high seismic capacities and not to contribute appreciably to the plant risk.

The containment overpressure capacity was found to be governed by the membrane failure of the suppression chamber (torus) at a median internal pressure capacity of 170 psi which is about three times the design value. The uncertainty associated with this failure mode is about 0.2.

REFERENCES

Hardy, G.S. and R.D. Campbell, "Development of Fragility Description of Equipment for Seismic Risk Assessment of Nuclear Power Plants", 1983 ASME Pressure Vessel and Piping Conference, Portland, Oregon 83-PVP-13.

Kennedy, R.P. and M.K. Ravindra, " Seismic Fragilities for Nuclear Power Plant Risk Studies", Nuclear Engineering and Design 79, No.1 (May (I) 1984): 47-68.

Ravindra, M.K., and R.P. Kennedy, "Lessons Learned from Seismic PRA Studies", Paper M6/4, Proceedings of the Seventh Conference on Structural Mechanics in Reactor Technology, Chicago, Illinois, August 1983.

Ravindra, M.K., Hardy, G.S., Hashimoto, P.S., and Griffin, M.J. (1987). Seismic Margin Review of the Maine Yankee Atomic Power Station: Fragility Analysis, NUREG/CR-4826 Volume 3.

Johnson, J.J., Goudreau, G.L., Bumpus, S.E., Maslenikov, O.R., "SSMRP Phase I Final Report - SMACS -Seismic Methodology Analysis Chain with Statistics (project VIII)", Lawrence Livermore National Laboratory, Livermore, CA, UCRL-53021, Vol. 9, also NUREG/CR-2015, Vol. 9, September, 1981.

Table 1: Low Capacity Equipment Components

Equipment Item	Failure Mode
Reactor Vessel Stabilizer	Connection failure
Reactor Internal Structures, Engineered Safety Features	Shroud support leg failure
Fuel Rods	Deflection of rods causes failure to SCRAM
Control Room Ceiling	Failure of ceiling components falling & impacting sensitive relay cabinets below
ESWP MCC	Anchorage failure
Hydrogen Recombiner Power Supply Panel	Anchorage failure
Off-Site Power	Failure of yard components, (generic evaluation)
4.16 Kv Switchgear	Relay chatter
Power Center Transformers	
Units 1A & 2A	Unanchored core/coil assembly
Units 3A & 4A	Clamping assembly of the core/coil
480V Motor Control Centers	Contact chatter
Battery Chargers	Functional failure mode

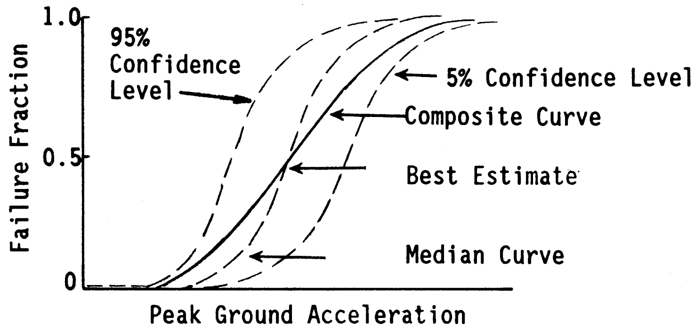

Figure 1: Typical Fragility Curve

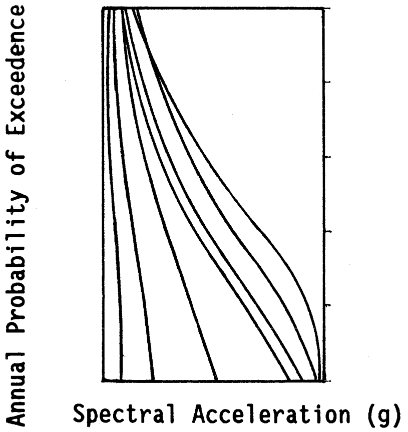

Figure 2: Seismic Hazard Calculation Based on Spectral Acceleration Formula

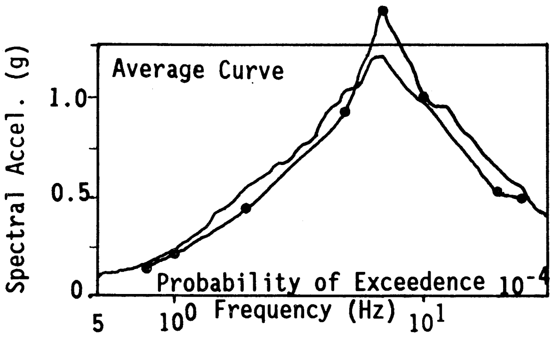

Figure 3: Comparison of Site Specific Spectrum and Uniform Hazard Spectra Both Anchored to 1.0g at 10 Hz Frequency.