Insights from the German Risk Study in the Light of the Reported Comments on NUREG 1150

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INTRODUCTION

It has been just over fifteen years since a pioneering team guided by N.C. Rasmussen undertook to estimate the risk to the public from accidents as a result of the operation of the first 100 NPPs in the United States. Since that time about 40 fullscope risk studies and additionally about 60 Level 1 studies have been carried out in many different countries. In some countries, e.g. in Sweden, all the NPPs under operation were analysed with the methods and tools of Probabilistic Safety Assessment (PSA)(Carlsson et.al., 1987). In other countries, e.g. the U.K. (GEGB 1982, W 1982) and the F.R.G. (GRS, 1979, 1989) a very detailed and comprehensive PSA was completed for a reference plant respectively.

For the customers and the analysts it was a learning process to move from generic studies, representative for a specific class of NPPs, to the plant specific studies, representative for the plant considered.

In addition to the PSAs released several reviews, comparisons of results and lessons learned from different studies were issued (e.g. Joksimovich 1984, Garrick 1987). These publications followed three main aims:

- to review and comment the study for further improvements;
- to extract insights and conclusions related to safety issues and research strategies for a given plant or others of similar design and construction principles;
- to expand the state of knowledge about PSA in the field.

If we assess the support and benefit from the analysis process itself and then from the review and comparison, the following conclusions are possible:

- the analysis process and the state of knowledge of the study team are of great benefit to the vendor and the utility for identification and treatment of plant-specific availability and safety issues;
- the review process of a given study increases their quality and the acceptance by third parties;
- the comparison and discussion of different PSAs expands the state of knowledge in the field, stimulates use and application and support decision making processes in reactor safety.

Certainly, the German Risk Study and the U.S. Reactor Risk Reference Document NUREG 1150 (US NRC, 1987) play an important role in reactor safety and therefore I would like to show in this paper some insights from these two documents and to draw some comparative comments.

INSIGHTS FROM THE GERMAN RISK STUDY

The reference plant for the study is the 1300 MWel PWR Biblis unit B. The plant has been in operation since 1976. The results of the Phase A of the Study have been published in 1979 (GRS, 1979, EPRI 1981).

The main objectives of Phase B (Heuser, 1988) are:

- completion of event tree analysis taking into account further initiating events (IE);
- improvement of the analysis on the basis of realistic assumptions;
- consideration of system improvements and modifications since the completion of Phase A;
- identification and assessment of accident management AM measures which are adequate to minimize the risk.

From a list of about 40 items identified during the Phase A analysis process the most important system modifications considered in Phase B with respect to the results are:

- the installation of a semi-automatic system for the cooldown in the case of small leak;
- automatic partial cooldown in the case of a loss of the main heat sink;
- position control of the different valves of the pressurizer relief system by means of various additional isolating signals;
- the improvements in the main steam relief station, i.e. installation of 15% safety valves which can be blocked, and quick-closing main steam isolation valves;
- the installation of a stand by grid connection;
- the possibility of switching back the 10 kV emergency busbars to the 10 kV busbars in the case of the failure of the emergency power diesel generators.

In addition to the IE dealt with in Phase A assessments have been performed for the following IEs:

- large and medium sized leaks in a main steam line;
- leaks in steam generator heating tubes;
- leaks in connecting lines (interfacing system LOCA).

The minimal requirements on the basis of best-estimate analysis were evaluated for the whole spectrum of leaks in the primary system. If the cooldown is started within 40 minutes after LOCA then 1-out-of-4 HPI and 1-out-of-4 LPI are sufficient for the whole leak spectrum. For details see also (Hörtner, 1987).

Regarding the event tree and fault tree analysis, the following overall insights have been obtained:

- best-estimate analysis were used to establish the minimum requirements;
- investigations have been made for plant internal fire and for earthquakes loads.
- system modifications have been considered; they have been reduced the core melt frequency;
- additional IEs, evaluated in Phase B, yield new contributors to the core melt frequency;
- the overall core melt frequency in the range of 7 x 10E-5/a assessed in Phase B is sightly lower than the mean value assessed in Phase A;
- unplanned AM measures were taken into account for prevention and mitigation of accident consequences. The core melt frequency were lowered by AM about one order of magnitude to a value of 7 x 10E-6/a.

As AM measures secondary and primary bleed and feed were considered. A successful primary bleed, is a prerequisite to prevent core melt under high pressure. For approximately 90% of event sequences, not coped with the design basis safety system, core melt can be prevented by AM measures.

COMMENTS ON REACTOR RISK REFERENCE DOCUMENT (NUREG 1150)

NUREG 1150 is based on PSA techniques to evaluate the risk from five specific nuclear power plants. The five plants were representative in that each uses a different one of the containment concept for NPPs built in the United States. Although the analysis broadly follows that used in modern PSAs. It includes new approaches to some aspects of nuclear safety. In particular, the analysis of containment response is accomplished through the use of large event trees. This requires the evaluation of a large number of different source terms. Other tools that have been introduced is the use of new codes like the health effect model MACCS.

The most important feature in NUREG 1150 is a major effort to generate an estimate of the uncertainty of the risk estimates (Kouts 1987). The uncertainty estimation was developed in some of the steps of the risk estimation process. The major contributors to the result were identified and a base case was done, based on these. A measure of uncertainty was then assigned to each of the major contributors either by statistical analysis or by use of expert judgement. Used was a modified Delphi method. The uncertainty was expressed as a discrete probability distribution. Less significant inputs were simply assigned unique the but best estimate values. The original calculation was then redone using the Monte Carlo method employing a particular stratified Latin Hypercube Sampling Scheme to propagate the assigned uncertainty through the calculation. This procedure was followed in the main steps of the PSA and an uncertainty representation was generated for the final result.

It is important to say that "uncertainties" as calculated and displayed in NUREG 1150 have two important components. The first is the actual uncertainty in scientific comprehension and predictability as perceived by each individual whose opinion was solicited and used. The second was a person-to-person variability in this perception, which can be stated otherwise thoroughly homogenized in the final uncertainty estimate.

As a comprehensive and summarized comment on NUREG 1150 one can say it is a product of a substantial advance in the art of PSA (e.g. Kastenberg 1988). It introduces and produces more up-to-date understanding of reactor safety. It has made a serious attempt to deal with the sources and the implications of important lack of knowledge in form of uncertainties. It makes use of advanced statistical techniques. The work includes several factors that raise questions as to the estimated uncertainties that are presented. These are in the areas of selection of experts, the degree to which opinion substituted for expert finding in polling of experts, the mode of display of results, the quality of some deterministic calculations in the post core melt phase and the quality assurance of PSA codes and their use in general.

There are some places in the study where there is disregard for technical rigor and/or the state of the art. This includes facets of both probabilistic and mechanistic analyses. A number of probabilistic analyses such as the treatment of common mode failures, human reliability assessment and system availability have serious deficiencies. Similarly, the currant knowledge with respect to in-vessel core melt progression, containment loading and fission product release, are not reflected in the PSAs. For example, recent considerations on direct heating were not considered and the use of parametric models for source terms were not demonstrated. A number of assumption, considerations and decisions were made, but were not successfully articulated in the document. The separation between "internal" and "external" events in a PSA is artificial and several PSAs have demonstrated the risk significance of these events. Therefore it is not understandable why these potential risk contributors have been excluded from the risk assessment.

All these questions apply not only to the uncertainty bands of the results, but also to any conclusions that might be reached on central values of distribution functions. These reduce the confidence of the numerical results of NUREG 1150. It is expected that these overall shortcomings will be overcome in the final version of NUREG 1150.

SOME COMPARATIVE COMMENTS

Although the German Risk Study and NUREG 1150 (including the five reference PSAs) regarding aim, structure and volume are different it offers in spite of that the possibility for comparative comments:

- the main procedures conducting the PSA are in both documents the same. The event scenarios are identified, structured and assessed on the basis of event tree and fault tree models;
- the used codes for the deterministic calculations of physical parameters in the course of event scenarios are different (e.g. ATHLET in GRS, TRAC in NUREG 1150);
- plant specific specialities e.g. train separation, grid connections, containment design, required specific modeling;
- the basic principles in the common mode and human factor analysis are comparable. Some specific models, adapted on the available row data, are different. (e.g. binomial distribution for common mode failures);
- the uncertainty analysis in the area of fault tree analysis is comparable. Uncertainty analysis through the whole process of frequency and consequence estimation differs widely. Perhaps, additional investigations for the German Risk Study will be settled for this task complex. Even if the judgment of uncertainties in models and parameters is "uncertain" then a formal method to judge or to calculate and to propagate uncertainties should be used in PSAs;
- the main results e.g. core melt frequencies lie in the same range in both documents. The importance of comparable IEs and event sequences is different and plant specific (e.g. station blackout);
- the display of results are slightly different. The German Risk Study do not use e.g. a misleading "box and whisker" display for core melt frequencies. It uses the traditional statistic display with mean value and confidence interval.

CONCLUSION

In concluding I would like to stress that the independent and diverse analysis process in these two documents has demonstrated once more a comparable overall safety standard in different plants (see Fig. 1).

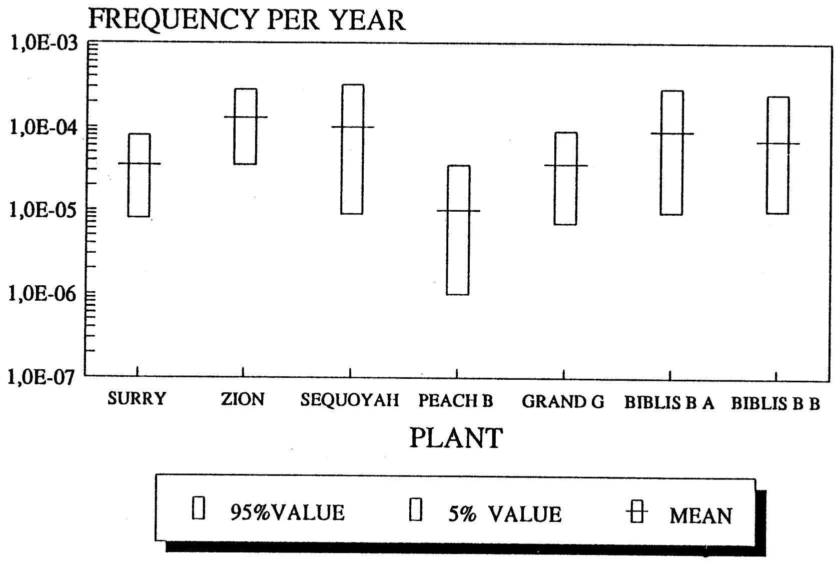

Fig. 1: Core damage frequencies assessed in different PSAs

Plant specific differences in risk contributors should be considered carefully by adequate operating regimes, periodic inspections and a preventive maintenance program focussed on the PSA insights. As PSAs become more plant specific, the detailed evaluation of operating experience for the different plants is therefore of increasing importance.

Both discussed documents are so-called snapshot PSAs, relevant for a given time window. Insights from plant performance programs indicate an increasing significance of aging resp. deteriorating processes on components. As a consequence a PSA should in future be evaluated time dependently with an update on the basis of actual system configurations and reliability figures. Averaging for the number of failures over a very long time window smoothed out such time dependent processes on components. This problem is a new challenge to the analysts. Such a PSA, characterized as a "risk monitor", should be a prerequisite for decision making with respect to relicensing or lifetime extension of older NPPs.

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