German Risk Study of PWR's

P. Kafka
Gesellschaft für Reaktorsicherheit mbH, Forschungsgelände, D-8046 Garching, Germany

Abstract

In this paper, first the status of German Risk Study is presented briefly. Specific reference is made to the investigations in Phase B of the study and related programs. Significant elements involved in the risk assessment for NPPs, mainly in the field of system and structural reliability analyses are mentioned. In particular, important outcomes and limiting facts in the process of a Probabilistic Risk Assessment (PRA) to evaluate the safety standard and above all the influence of individual components or subsystems on core melt frequency are discussed.

1. STATUS OF THE GERMAN RISK STUDY

The investigations on Phase B of the German Risk Study were offered by the BMFT in 1981 /1/. The aim of this procedure was to get contributions from and cooperation of the most experienced institutes and by these means to obtain a wide technical and scientific base for the investigations.

The most significant goals of the Phase B are:
- To complete the investigations carried out in Phase A /2/; particularly those concerning the initiating events of the LOCAs and Transients;
- To consider the further development and improvements of models applied to the safety research in WASH-1400 and in the Phase A; and
- To enhance the certainty of the frequencies for radioactive release and the accident consequences

The plant specific technical investigations will be performed on the reference plant Biblis B, as done in the Phase A. System modifications performed meanwhile in the reference plant will be taken into account. Furthermore differences in the results for recent plants will be roughly assessed. The main contributions of the GRS for the Phase B is the system analysis, the event tree- and reliability analysis. Since these topics are particularly related to the structural reliability and, therefore, of main concern to the SMiRT-7, they will be described here. Topics such as the treatment of core melt, the radioactive release, and the calculation of accident consequences, are not of great relevance to this conference and therefore will not be considered.

In Phase B in the field of the Loss-of-Coolant accidents, very small leakages (cross section $< 2\ cm^2$), and failures of steam generator tubes will be analysed and the studies concerning interfacing systems LOCAs will be done in greater details.

As to the Transients, a deeper analysis of the spectrum of the likely-events, that means, transients with the frequency of occurrence of $\gtrsim 10^{-2}$ per year and plant will be improved, mainly on the basis of operational experiences. From the unlikely-events, mainly the pipe breaks in the secondary system will be analysed.

Fig. 1 provides perspective on the nuclear steam supply system (NSSS) of the plant Biblis B. The NSSS is a 4-line system. In each case two steam lines from the steam generators to the main steam isolation valve are paired. Every line is equipped with a safety valve. Moreover, the NSSS contains two relief valves for cooling down of the secondary ciruit, that are connected via a common header to all the steam lines. The figure shows the most important break positions which are to be investigated:
- B1: Break within the safety containment
- B2: Break between the safety containment and the main steam isolation valve
- B3: Break behind the main steam isolation valve.

Of particular importance are the breaks occurring before the main steam isolation valves. In this case, a not closeable leak is present in the main steam supply system. Unfavourable, than a break in a main feedwater pipe, one must therefore assume that for the corresponding line or steam generator also the emergency feedwater is not available. For the residual heat removal, therefore, one steam generator is already failed.

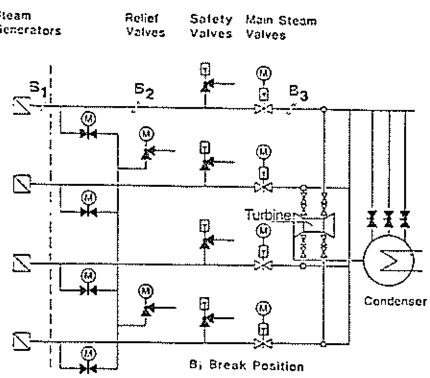

Fig. 1: Schematic flow sheet for the NSSS, Biblis B

Up to now, those event sequences have been studied whereby a special reactor protection signal is initiated to recognize the leakages in steam supply system, because the $\Delta p/\Delta t$ -value in a main steam line is greater max.

It should be mentioned that the problem of the consecutive breaks caused by pipe whipping is of particular significance. In the analyses carried out up to now, in the case of large breaks, conservatively, the direct consecutive break of the neighbouring line has been assumed. An important role plays also the main steam line breaks in the containment. In such a case, components inside the containment may be affected by high pressure and temperature due to an accident. This could lead especially to a failure of power generated relief valves on the pressurizer. Thus, additional contributions to the LOCA sequences by a small leak on the pressurizer must be taken into account.

Replacing the values used in the Phase A for the ruptures in the main coolant circuit for the occurrence frequency a main steam line break a core melt frequency of $2 \cdot 10^{-5}$ per year and per plant can be derived from the total number of investigated steam supply pipe ruptures. These results are still provisional. If this contribution cannot be improved through more precise evidences, one yields a significant increment to the core melt frequency. To value this partial contribution, one must also keep in mind that altogether in the Phase A assessed occurrence frequencies of core melt will be reduced considerably when taking into account the technical improvements in the systems and the present best-estimate calculations. Further preliminary results of the investigations in the Phase B are published in /3/.

2. IMPORTANT ELEMENTS IN THE ANALYSIS

The important elements of the plant's technical investigations with regard to the Probabilitistic Risk Assessment (PRA) will be discussed on the basis of Fig. 2.

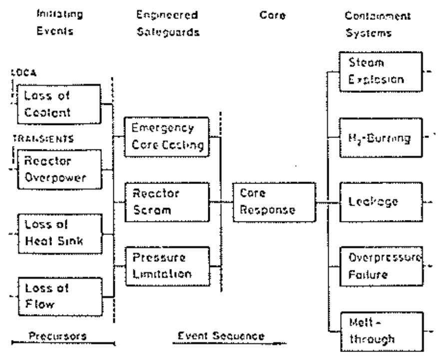

Fig. 2: Outline of a Risk Study

The analysis of the frequencies of initiating events encompasses questions concerning system reliability as well as structural reliability. In the case of a Loss-of-Coolant accident, the frequency of a leak in the reactor coolant pipe system is of main concern. The relevant quantitative estimates and also operational experience have shown that a LOCA is caused perhaps more frequently by faulty opening or not re-closing of isolating valves in the primary system rather than by failures of pipes of the primary system itself. In connection with primary pipe breaks, steam-line breaks and steam generator tube failures the structural reliability analysis for assessing the contribution to the core melt frequency is of particular importance. The operational experience with steam generators and the locally limited material dammages found thereby, draw attention to the same point.

Regarding the relevant spectrum of initiating events in the field of likely Transients, the evaluation of operational experience is of primary concern. Additionally, analytical investigations should be performed to increase the completeness of the initiating events.

The analysis of engineered safeguards is carried out generally using fault trees applying the reliability figures for individual components. These values are mostly assessed directly from the operating experience. With the probabilistic models employed presently, it is possible to determine the expected values, based on inaccurate input data, for the reliability figures, as well as determine the relevant confidence intervals. It can be verified with examples that analytical derived reliability figures for partial systems are sufficiently consistent with the values taken from the corresponding operational experience. The consideration of human interferences and of Common-Mode-Failures are still significant elements in this area. Almost all the PRA investigations performed up to now show their dominant influence.

The important elements in the treatment of the core melt will not be considered further in this paper.

The containment, an important barrier for the radioactivity in the case of a core melt, contains a number of various types of analysis elements. The possibility of the containment damage as a result of core melt requires, the estimation of event sequence as well as the occurrence frequency of the relevant physical processes. To assess the structural reliability for the containment, is also a main task in this analysis.

For the reference plant of the German Risk Study, the containment failure due to pressure overload following a core melt is signifcant. Compared to Phase A, in the Phase B a more comprehensive treatment of the interaction between melt and concrete, leads to a later increase of the pressure load as well as to a later containment failure due to pressure overload (after about 3-4 days). However, event sequences are possible, that are connected with the type of pressure relief from the primary circuit and Hydrogen combustion, and lead to pressure peak values slightly below the design pressures of the containment. Here, it appears to be a verification of the deterministic statement that no failure of the containment occurs with the elements of the probabilistic structural mechanics.

3. USEFUL OUTCOME FROM THE ANALYSIS

PRA constitute an useful tool in safety evaluation /4/, /5/, /6/. In so far, supported from many examples, nuclear installations on a higher safety standard are indirectly the outcome of such studies /2/, /7/.

The probabilistic reliability analyses, as the main element in PRAs /8/, permit a better balanced judgement of the safety relevant design than only reliability statements on the basis of deterministic criteria. This type of analysis allowed more or less an optimation of the whole systems with respect to the different failure possibilities, and the combinations of them. The reliability analysis permits in here process to judge much better influence of individual components on the unavailability of the system and so far the identification of the weak points in the whole system.

Experiences show, that design and operational problems preferabely arise at interfaces between different type of systems and where different engineering science interact. Utilizing reliability analysis it can also be determined in which situations manual actions are important for the control of accident sequence. This has also been shown by the German Risk Study. In the well-known case it was necessary to shut-down the plant with a gradient of 100° C per hour by manual actions. These manual actions have been automized in the recent plants; thus, for example, in the PWR plant, Grafenrheinfeld, an improvement of relevant error probability of about $1.3 \cdot 10^{-3}$ has been achieved.

On the other hand, the sequence of automatic actions has to be oriented according to predetermined courses of accident. This requires the investigation, if accident sequences may occur, which cause the automatic device to react in a wrong way. Also such examples are the outcome of the German Risk Study. In the case of an ATWS the main-load feedwater control valves for the steam generators will be closed as soon as a reactor scram is triggered, even if the scram should fail mechanically. Investigations of the operational control system have shown that in these cases also the low-load feedwater control valves would be closed. Thus, main feedwater flow would be blocked totally and the termination of the ATWS become more difficult.

4. LIMITING FACTS FOR GENERAL APPLICATIONS

Interpretations of PRAs certainly have to take into account that considerable uncertainties are attached to their results. However, considerable variations of the estimated frequencies of core melt also indicate that plant-specific design characteristics are more important than assumed in the past. This view is also supported by a comparison of WASH-1400 and the German Risk Study. Both studies determine dominant risk contributions from very specific accident sequences resulting from design characteristics of the reference plants. For both reference plants small leaks dominantly contribute to core-melt frequency. In Surry 1, the main cause for this was a possible random failure of the emergency core cooling system. For Biblis B, on the other hand, it was a possible failure of the secundary heat removal system caused by operator error. Dominant contributions to the frequency of the release categories leeding to severe release came from interfacing system LOCA in Surry 1. For Biblis B, such sequences were less significant.

A current question is whether the results of PRAs are too optimistic compared to the results of to the "Precursor Study" /9/. In this study approximately 19,000 events in US-plants have been screened. 169 of this have been selected for a detailed analysis. Finally, three events contributed about 84 % to the result of $4.5 \cdot 10^{-3}$ per year. The TMI-accident alone contributes 50 % to it, and the Browns Ferry fire about 20 %. In this context, it seems to be more realistic to consider the 1500 years of LWR operation all over the world and not only the 433 in the United states. Then the expected frequency of severe core dammage estimated from TMI-accident would be reduced by a factor of 3.5, that means to about $7 \cdot 10^{-4}$ per year.

Experience has shown that PRAs, in spite of the uncertainties in the estimated figures, significantly contribute to the improvement of safety evaluations and to an increase of the safety of the plants. This is particularly true for the technical part of the investigations, particularly for the event sequence and reliability analyses. Of course, also in this field there remain limiting facts, which are only partially resolved. Several of them will be stressed in the following in view of the experience gained in the German Risk Study.

- Data Base for Random Component Failures

Considerable influence on the variations in the results of the analyses comes from the uncertainty of the data base for the reliability analyses. Such data describe component failure rates and the influence of maintenance work on component reliability. Also in the contex of the German risk study it has been attempted to broaden the data basis for independent component failures /10/.

- Common-Mode-Failures

Besides independent failures of individual components, simultaneous failures of redundant components due to common cause have to be considered. By a generic term such failures are called "common-mode-failures". Studies performed in the United Staates exhibit unrecognized functional dependencies between redundant components as the most important cause for common-mode-failures /11/. Such functional dependencies can be largely avoided by functional and spatial separation of redundant safety systems. Remaining functional dependencies can be considered and evaluated through sufficiently detailed reliability analyses.

Further causes fo common-mode-failures could be errors in planning, manufacturing, operating, or maintenance. Although various measures are taken against the occurence of such failures, these cannot be positively excluded. Since common-mode-failures occur very rarely, and if they have occurred, measures against a repetition are taken, the quantification of probability of occurence of a common-mode-failure is difficult. In the German Risk Study it has been attempted, to derive such probabilities from operating experience in German nuclear power plants.

- Human Errors

Operating experiences and PRAs show, that human errors may contribute significantly to the risk of nuclear power plants /12/.

It is particularly difficult to evaluate the reliability of human influence in the control of disturbances and accidents in a quantitative way. The importance of human influence for the termination of transients can be seen from the operational experience /13/.

- Completeness

Discussions about risk analyses often raise the question, whether all important influences and all accident sequences significant for the risk have been considered. Principally, there is no possibility to demonstrate the completeness of the analysis in an inductive way. Therefore, one has to rely on systematic analyses and on experience and knowledge of the analysis. In order to judge the completeness of analysis it is important to continuously compare theoretical investigations and practical experience from operation in order check whether important accident sequences have been overlooked and whether new results require amendments of the analysis.

Generally speaking, the completeness problem, the problem of evaluating common-mode-failures and human error influences are responsible for some peoples view that reliability analyses and PRA are more an art than a science. Indeed, engineering judgement and intuition of the analysts play an important role in these analyses. However, we can satisfactory recognize that all this limiting facts are worldwide more or less wellknown and a great number of experts works on their solution.

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