German Risk Study – Influences of data base, minimal requirements and system modifications

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The reliability analyses for Phase B of the German Risk Study take into account an improved reliability data base, best-estimate minimal requirements for the relevant system functions and the design modifications, which have been carried out after completion of Phase A.

These points and their influence on the frequency of core melt accidents will be discussed in the following sections, emphasizing the reliability data.

1 RELIABILITY DATA

For the assessment of the frequency of core melt accidents the following kinds of reliability data have to be estimated:

- Expected frequencies of the initiating events (i.e. of the loss-of-coolant accidents and the transients),
- failure rates or failure probabilities for independent failures of the components in the safety systems,
- failure rates or failure probabilities for common-mode failures of these components and
- probabilities of human error.

Probabilities of human error are required with respect to maintenance measures taken during operation of the plant and, in particular, manual interventions which are necessary after the occurrence of an initiating event. Although German nuclear power plants have reached a high degree of automation, there are still various actions, depending on the initiating event in question, which have to be performed by the control room personnel. As only little information is available in this context which was obtained from operating experience, expert assessments have to be relied upon. The most important tool for these assessments is the "Human Reliability Handbook" (Swain and Guttmann 1983). To obtain relevant data, research programs have been launched aiming at a systematic evaluation of both, simulator experiments and the available operating experience.

For the estimation of the frequencies of initiating events and of the failure rates or failure probabilities of the components three general types of information from operating experience have to be distinguished:

(1) Observations in areas related to nuclear power plants (coal-fired power plants, etc.) or observations in other areas of

industry (chemical engineering, laboratories, etc.)

(2) Observations in other nuclear power plants

(3) Observations in the nuclear power plant under investigation.

The information of types (1) and (2) above is collectively referred to as so-called "generic" information. Type (3) constitutes "plant-specific" information.

The question arises which information is available and which type of data is to be used for the issues to be investigated.

If reliability analyses are to be prepared for a system concept, or if reliability predictions are to be made for a plant under construction, plant-specific data are not available. Thus, generic data have to be used in these cases.

If reliability assessments are needed for a class of a certain type of plants, as it was the case within the scope of the U.S. Reactor Safety Study (1975) or Phase A of the German Risk Study for a PWR plant of the type Biblis B (Gesellschaft für Reaktorsicherheit 1979), generic data have to be used as well.

In this context, it should always be attempted to use information of type (2), i.e. observations in nuclear power plants. This information is characterized by higher quality than equally detailed information of type (1). The reason is that for the information of type (2) the question of transferability to nuclear power plants does not arise.

In recent years, reliability analyses have increasingly been carried out for individual plants already in operation, often within the scope of probabilistic risk assessments. For this purpose, plant-specific data should be used which have been collected over an operating period that should be as long as possible in order to include effects such as changes of components, aging, maintenance modifications and improvements as a result of experience. The use of generic data for such analyses is only a makeshift. The way the plant-specific estimates and their uncertainties are determined, is described in the following. As a matter of principle, there are two ways which are dealt with in the following sections 1.1 and 1.2.

1.1 The determination of plant-specific data using generic data as pre-information

The generic information is used as a basis for "prior" probability density functions. By including the plant-specific information, one ultimately arrives at plant-specific "posterior" probability density functions. The fundamental tool for this purpose is Bayes' theorem (Box and Tiao 1973, DeFinetti 1974, Martz and Waller 1982, Apostolakis et al. 1982).

On the basis of the generic information obtained from Phase A of the German Risk Study, an attempt was made in Phase B to use a two-stage Bayesian approach to determine plant-specific data with pre-information.

1st Stage:

Inclusion of generic information derived from nuclear power plants

Prior distribution:

Data from Phase A of the German Risk Study (Gesellschaft für Reaktorsicherheit 1980). These data are predominantly based on non-nuclear operating experience (at least with respect to mechanical components). Thus, they constitute above all type (1) information.

Information used:
Observations from up to 8 Swedish nuclear power plants (Bento et al. 1985) and one U.S. nuclear power plant (Stringham and McClymont 1982), the German nuclear power plants of Stade and Biblis A. This is type (2) information.

Posterior distribution:
Generic distribution for nuclear power plants.

2nd Stage:
Inclusion of plant-specific information

Prior distribution:
The posterior distribution of the 1st stage.

Information used:
Observations in Biblis B, the reference plant of the German Risk Study.

Posterior distribution:
The plant-specific distribution with pre-information for Biblis B.

In Phase B of the German Risk Study, this approach was used for independent failures covering approx. 50 component failure modes. As an example, the failure to start of pumps is addressed here. The pumps are understood to include their circuit breakers (including fuses) for electric power supply and their actuation devices. The corresponding failure rates are greater than for most other components so that the pumps are more important for system reliabilities than many other components.

Figures 1 - 3 show the probability density distributions for the failure to start of service water pumps.

Figure 1: Generic distribution for the failure to start of pumps.
Figure 2: Generic distribution for the failure to start of service water pumps in nuclear power plants (without Biblis B) which constitutes a relatively broad distribution because of its consideration of various plants.
Figure 3: Plant-specific distribution with pre-information for Biblis B nuclear power plant. It is a relatively narrow distribution which is very well compatible with the broader generic distributions.

A similar result in terms of quality was obtained for about one half of the component failure modes investigated.

Figures 4, 5 and 6 show the probability density functions for the failure to start of residual heat removal pumps.

Here, the generic distribution and the plant-specific distribution are poorly compatible; the plant-specific distribution is clearly at higher failure rates than one would expect it to be on account of the generic distribution.

A qualitatively similar result was obtained for the other half of the component failure modes investigated. In more than three-fourths of these cases, the pre-information resulted in a shift to lower failure rates. Here, a number of questions arise and one has to bear in mind that as many as nine of a total of twelve available data sources concerning independent failures of components refer to observations made at foreign nuclear power plants:

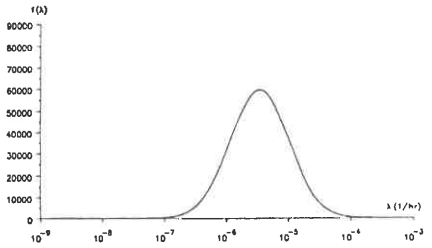

Figure 1. Generic probability density of the failure rate for the failure to start of pumps, taken from Phase A.

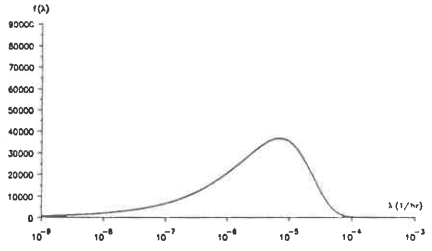

Figure 2. Generic probability density of the failure rate for the failure to start of service water pumps in nuclear power plants.

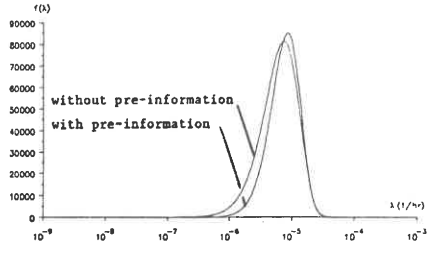

Figure 3. Plant-specific probability density of the failure rate for the failure to start of service water pumps in Biblis B.

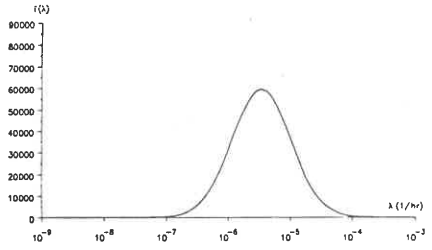

Figure 4. Generic probability density of the failure rate for the failure to start of pumps, taken from Phase A.

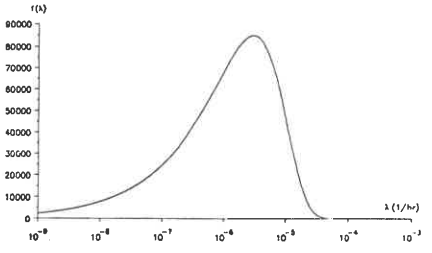

Figure 5. Generic probability density of the failure rate for the failure to start of residual heat removal pumps in nuclear power plants.

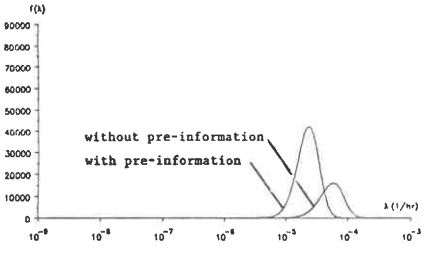

Figure 6. Plant-specific probability density of the failure rate for the failure to start of residual heat removal pumps in Biblis B.

- Have all failures which occurred at these nuclear power plants been documented and taken into account when the failure rates were determined? Or in other words: what about the quality of the realiability data?
- Have identical component boundaries been taken into consideration, or could it be that circuit breakers and actuation devices have not been included?
- Do the components differ as to design type, maintenance or mode of operation?
- Are there different intervals between the function requirements, and in particular, are in-service inspections not carried out every 4 weeks so that mistakes may have occurred when converting failure probabilities per demand - which are frequently quoted in foreign data sources - to failure rates?
- Has the evaluation only considered failures during power operation, or have failures during plant outage been included?

Because of the manifold discrepancies between the pre-information and a very detailed evaluation of plant-specific observations in the Biblis B reference plant, it was concluded, plant-specific data with pre-information should not be used unless and until a sufficient amount of observations at German nuclear power plants will have been evaluated and will be available as pre-information.

For the reason described before it was decided, as far as Phase B of the German Risk Study is concerned, to use plant-specific data without pre-information instead of plant-specific data with pre-information.

1.2 Determination of plant-specific data without pre-information

As in Section 1.1, Bayes' theorem is used for the determination of corresponding probability density functions although the approach is based on a so-called non-informative prior distribution.

In literature, the uniform distribution and the special Γ distribution with parameters ($A = \frac{1}{2}$, $B = 0$) are referred to as non-informative prior distributions (Box and Tiao 1973). The latter has the advantage of eliminating the question which otherwise often arises of whether the prior uniform distribution should be chosen for the failure rate or for the mean life-time. The Γ distribution has been used for Phase B of the German Risk Study.

The plant-specific data without pre-information as determined for the failure to start of the service cooling water pumps and the residual heat removal pumps are shown in Figures 3 and 6, resp., together with the plant-specific data with pre-information.

For the service cooling water pumps, the distribution obtained without pre-information is very similar to and only slightly broader than the distribution obtained with pre-information.

For the residual heat removal pumps, the distribution obtained without pre-information is not only broader (logarithmic scale of the abscissa) but also situated at higher failure rates than the distribution with pre-information. This is a clear indication of the fact that the available pre-information and the plant-specific distribution are not compatible.

Distributions without pre-information for the plant-specific failure rates or failure probabilities per demand, differentiated as to component types and operating conditions, were determined not only for pumps but also with respect to various other components.

The expected frequencies of initiating events were also based on plant-specific operating experience without pre-information to the extent possible. For a number of initiating events having a low occurrence probability, the entire German or the worldwide operating experience had to be used.

1.3 Failure rates and failure probabilities for common-mode failures

Plant-specific operating experience is not sufficient, for an assessment of common-mode failures (dependent failures). Here, it was necessary to utilize the documentation of common-mode failures which have occurred both, in the Federal Republic of Germany and worldwide. On the basis of this information distributions can be assessed for generally defined component groups. In this context, the applicability of common-mode experience to the Biblis B reference plant had to be carefully investigated.

2 MIMIMAL REQUIREMENTS

To determine the minimum requirements for system functions, best-estimate analyses were performed for loss-of-coolant accidents and transients. As far as "leaks in the reactor coolant system" are concerned, the minimum requirements can be summarized as follows (Hoertner et al. 1986):

If a cooldown with 100°C/hr is performed within 30 minutes, 1-out-of-4 High Pressure Injections and 1-out-of-4 Low Pressure Injections will be sufficient for emergency cooling covering the entire leak spectrum; but the High Pressure Injection is not required for all leak cross-sections. In the case of a small leak of a cross-section between 25 and 200 cm^2, a core meltdown cannot be prevented, if there is a total failure of the High Pressure System. In the case of a leak under 25 cm^2, no High Pressure Injection is needed if the cooldown is carried out within 30 minutes. Otherwise, 1 High Pressure Injection is required.

The Accumulator Injections, which are used for the rapid refilling of the reactor pressure vessel in the case of medium-sized and large breaks, are not needed for the prevention of a core damage.

3 SYSTEM MODIFICATIONS

Biblis B reference plant has seen a number of system modifications since the completion of the analyses carried out in Phase A of the German Risk Study (Hoertner et al. 1986). These system modifications are taken into account in the analyses of Phase B. Thus, the analyses of Phase B are based on the current state of system engineering at the reference plant. The following system modifications substantially influence the results:

- Installation of a semi-automatic system for the controlled cooldown with 100°C/hr in the case of small breaks,
- automatic partial cooldown of the plant in the case of a failure of the main heat sink,
- improvements in the main steam relief station,
- various additional isolation signals for the pressurizer relief system,
- possibility of reconnection to grid in the case of emergency

diesel failure,
- installation of a standby grid connection.

4 RESULTS FOR THE FREQUENCY OF CORE MELT ACCIDENTS

The influences of the points discussed in the paragraphs 1 - 3 on the frequency of core melt accidents allow the following statements:

- Although the detailed evaluation of operating experience for the estimation of the reliability data does result in an increase of contributions, the best-estimate minimal requirements and the system modifications carried out for the reference plant reduce the core melt frequency due to those initiating events which were dominant in Phase A of the German Risk Study.
- The detailed investigation of additional initiating events which had already been recognized as important during Phase A leads to additional contributions to the frequency of core melt accidents. Such initiating events are the main steam line break and the steam generator tube rupture (Gesellschaft für Reaktorsicherheit 1979).
- Altogether, the evaluated contributions to the frequency of core melt are lower than the values assessed in Phase A.

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