A COMPLEX STUDY ON THE RELIABILITY ASSESSMENT OF THE CONTAINMENT OF A PWR

Part I. — Magnitude and Probability of Internal Load Behaviour

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SUMMARY

For evaluation of the reliability of the safety enclosure in the case of accidents the time-dependent loads by internal pressure and temperature on the spheric steel containment and the correspondent probabilities had to be calculated.

Of the spectrum of possible accidents, e.g. large LOCA which leads to a maximum pressure of \( \geq 4.7 \) bar, working of all safety systems presumed, small LOCA or rupture of a primary steam pipe, only those have been selected which result in a considerable increase of internal pressure in the safety containment.

The pressure buildup in the steel containment depends roughly on the radioactive decay energy produced in the containment, on the performance of the safety systems operative after the accident and on the energy absorbed and transferred by the structural parts of the containment. For simplification the analysis of system behaviour was performed in separate steps. Analysis was started by evaluation of alternate possibilities of pressure buildup depending on the function of different safety systems. Then the time dependent changes of temperature and pressure in the containment were calculated as well as the probabilities of the occurrence of the different maximum pressures.

Technical data and accident event sequences describing the system analysed were taken from the PWR Biblis B, which at this time is typical for the PWR-line construction in the FRG.

In order to avoid event sequences leading to complicated physical phenomena such sequences were selected which allowed well-defined description of consequences as hydrogen production by reaction of water with the Zircalloy fuel cladding or pressure buildup by CO\(_2\) or steam generated from concrete getting in contact with the core-melt.

The computer code ZOCO VI was used to calculate pressure buildup for the different event sequences. This code calculates time dependence of pressure and temperature in a multiply segmented safety containment considering accumulation and condensation processes. Energy balances calculated by this code take into consideration the alternate influence of emergency and regular core cooling systems e.g. coolant supplied from pressure accumulators, flooding vessels or the regular core cooling system. In comparison to the energy release during and immediately after the accident the effects to be considered here are relatively small in magnitude. The statistical variance of the calculated pressures for certain cases considered was checked by sensitivity analysis.

The probability of certain event sequences leading to an accident has been calculated by multiplication of probabilities \( P_1 \times P_2 \times P_3 \times P_4 \).

For instance in WASH-1400 (Reactor Safety Study) the probability \( P_1 \) for a loss of coolant accident has been determined to be \( 10^{-4} \) for a "large LOCA" and \( 10^{-3} \) for a "small LOCA". \( P_2, P_3, P_4 \ldots \) are the probabilities of certain safety systems to be inoperative on request. The values have been determined by fault tree analysis of the different systems. The magnitude of probabilities calculated are between \( 10^{-4} \) per demand to \( 10^{-3} \) per demand. To get correct values for the release of radioactivity a large spectrum of accident sequences has to be analysed separately to determine probabilities of failure of the safety enclosure and to accomplish a correct description of the physical processes leading to release. Describing all accident sequences by a spectrum of calculated values for internal pressures only yields one value for the probability of containment rupture as a consequence of LOCA's but would be of no use for calculation of risk. Evaluation of the full spectrum of probability values for rupture of the safety enclosure by internal pressure spectrum was beyond the task of this work but was the aim of Part III of the research programme "Structural Reliability Assessment under Internal and External Loading Conditions" and is presented with the description of this programme.
1. **Introduction**

The primary coolant system and part of the secondary coolant system of a pressurized water reactor (PWR) can, in rare cases, fail as a result of a pipe severance; thus leading to the build-up of pressure and temperature loadings inside the steel hull of the containment (Fig.0, see Introduction Part).

Given these design characteristics, the task of the reliability assessment of the containment in general is to evaluate, in quantitative as well as in qualitative terms, all the possible alternative accident sequences following a pipe rupture (see also Part III). In order to understand and explain the reliability assessment qualitatively, system analysis technique is normally utilized. For the quantitative evaluations, the probabilities are assessed with the help of reliability analysis and the pressure behaviour is estimated with the aid of thermodynamic computer programs.

In this paper, the considerations of all possible sequences are limited to the loss of coolant accidents (LOCA) and the steam line break (SLB). For all cases only a rough estimation of the possible, but very improbable, core-melts with their complex physical processes in the course of accident sequences can be made. In this context, it is to be noticed that in those cases where core-melts could occur, the internal pressure of the containment will be influenced by the additional partial pressures due to zircon-water reaction, concrete-melt reaction and, in certain cases, due to sumpwater-melt reaction.

In this part of the project, the choice of some particular accident sequences (Fig.1, A1, A2,...) was not based on the view that the expected failure probabilities for individual cases were particularly high or particularly low, but rather on the point of view of the differences in consequences and the feasibility of the projects. On the other hand, it is obvious that, e.g., for risk studies, ultimately the considered cases and the rare cases with core-melts, especially in regard to the time behaviour, must be treated more accurately. This means, the questions handled here are only a partial contribution to the whole problem.

For both of the alternative accident sequences, LOCA and SLB, it is possible to treat first an integral of the loads working on the steel hull due to internal pressure, and, then to consider the local impact-loads due to pipe-whipping.

As the applicability of reliability predictions in risk analyses showed /1/, it was worthwhile to consider each of the loading conditions and the possible failure modes of the containment independently in view of the time and the amount of radioactive releases; and to evaluate individual probabilities separately. Thus, a partial contribution to the risk from an accident sequence based on consequence and frequency can be evaluated (Fig.1).

For other types of problems, e.g. concerning cost-risks, the question of overall reliability analysis of the containment can be of interest as well. For this, all the different accident sequences (A1 to Cn) and all the various pressure build-ups as a part of the pressure distribution can be taken into
considerations. Thus, the failure probabilities as well as the reliability predictions can be developed further.

2. **Accident Sequence Analyses**

In an accident sequence analysis using the event trees, those conditions are evaluated and given under which the initiating event over certain accident sequences leads to certain consequences /2/. Depending upon fulfillment/not fulfilment of certain demanded functions of the considered unit, the initiating event has different consequences. The event tree is mainly a qualitative logical representation of the sequences. It furthermore enables a quantitative evaluation. Here, the mean value \( P_{ij} \) for the frequency of each accident sequence within a certain time interval is of interest. To determine this, the following definitions must be known (General nomenclature):

\[
\begin{align*}
H(E_i) & \quad \text{mean value of the number of occurrence of the initiating event } E_i \text{ per unit time} \\
W(A_{ij}) & \quad \text{conditional probability for the consequences } A_{ij} \text{ on occurrence of the initiating event } E_i \\
S(A_{ij}) & \quad \text{magnitude of the corresponding damage consequence} \\
P_{ij} &= H(E_i) \cdot W(A_{ij}) \quad \text{mean value of the number of the occurrence of accident consequences } S(A_{ij}) \text{ per unit time}
\end{align*}
\]

For quantitative evaluation of the event trees, one should consider the following:

\[
\begin{align*}
\text{no (loss of function)} & \quad e \ldots \text{Input Event} \\
\text{yes (loss of function)} & \quad a_2 \quad w \ldots \text{Conditional probability}
\end{align*}
\]

\[
w(a_1) = w(e) \cdot w(\text{yes}) \\
w(a_2) = w(e) \cdot w(\text{no}) \\
w(\text{yes}) + w(\text{no}) = 1
\]

2.1 **Loss-of-Coolant Accident**

The primary system and a part of the secondary system are the only systems in the containment which in the case of a rupture can lead to the significant loadings of the steel hull due to internal pressure.

For evaluating all the possible pressure distributions, it is necessary in the first place to consider the loss-of-coolant accidents (LOCA). The LOCA's are normally categorized as follows:

- **Large LOCA**
  - rupture area: > 1000 cm\(^2\) (case A)
  - rupture area: 400-1000 cm\(^2\)
- **Medium LOCA**
  - rupture area: 80-400 cm\(^2\)
- **Small LOCA**
  - rupture area: < 80 cm\(^2\)

Such a classification is required, because for these alternative cases partially alternative safety systems are necessary.
- Accumulator Injection
- Low-Pressure Injection
- Emergency Cooling Systems
- Low-Pressure Recirculation
- Long-Term Residual Heat Removal

As pointed out in section 2, for the above mentioned systems in the case of large LOCA, the unavailability per demand as well as failure probabilities per time interval need to be calculated. For this purpose, it is necessary to define precisely the conditions which determine the function or disfunction of these systems. This can be done on the basis of the so-called functionability calculation using deterministic codes such as FLASH, RELAP, BRUCH /3/, in the framework of licensing procedures.

The conditions for the case A considered here are:

<table>
<thead>
<tr>
<th>cold-leg rupture area &gt; 1000 cm²</th>
<th>Minimal Requirements based on Licensing Procedures</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>HP-Injection</td>
</tr>
<tr>
<td></td>
<td>0</td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
</tbody>
</table>

From these system characteristics and the evaluations of the corresponding reliability figures (section 4), the event tree for the case of large LOCA can be constructed (Fig.2).

As it is known from the risk studies of PWR /1/,/4/, alternative steel hull failure modes can result from the accident sequences shown in Fig.2. That means, the failure mode due to overpressure is only one of the possibilities.

It is therefore convenient that for the various possible failure modes of the steel hull, separate failure probabilities be calculated, since each mode of failure is coupled with different time of release and different radiological consequences /1/,/4/.

The probability figures used in the event tree (Fig.2) are all Point Values. Since the frequency of initiating event H (E₁) and the unavailability of each system involve standard deviations (section 4), the mean value of the frequency of accident consequences P₁j would also have a standard deviation. In Ref. /1/, the following is given:

Large LOCA (A) \( H_A = 10^{-4} /a \) \( H_{A5} \% = 10^{-3} /a \)
\( H_{A95} \% = 10^{-5} /a \) type of deviation: lognorm

In the course of the development of this project, the standard deviation of \( P_{1j} \) must be computed from the standard deviation of H (E₁) and the standard deviations of W (A₁j). Qualitatively, however, it can be ascertained even now that the standard deviation in the frequency of the consequences of each accident as compared to the standard deviation in the accident consequences (i.e., overpressure; section 3) would be dominating. For provisional clarification, a sensitivity analysis concerning standard deviations is given in Part II.
2.2 Steam Line Break

Besides the integral loading of the steel hull due to internal pressure following a LOCA, the steel hull can be also locally constrained as a result of a steam line break. This load on the steel hull can be generated first directly by a steam blow from the fixed pipe and then indirectly from the unsupported pipe or from pipe missiles. This loading of the steel hull is primarily not relevant to the safety.

One is concerned with the question of safety only in two cases: First, when as a result of steam line break, a consequential failure of the steam generator-heat tube is possible and the probability for a local failure of the steel hull as a result of steam line break is large and; secondly, when the important components of the Auxiliary Feedwater System of the Steam Generator are failed by the steamy environment caused by a leakage in the steel hull. These interrelations can be qualitatively represented by a simplified event tree (Fig. 3).

As a contribution to the quantification of whole accident sequence, the purpose of this project is to evaluate the probability for a local failure of the steel hull as a consequence of a steam line break.

As a possible failure of the steam line, a double-ended rupture of the steam pipe before the last bending inside the steel hull is considered. In such a case, because of the elastic pipe connections (bellow) to the steel hull, the pipe would swing in the form of a circular pendulum. The driving force is the steam repulsion due to the moving pipe and the steam blow of the remaining steam line. The impact mass and the impact velocity of the moving pipe-end is given as:

\[ v = \omega \cdot l \]

\[ \omega = \frac{3 \cdot F}{m \cdot l} \]

\[ F = 1,2 \cdot 10^6 \text{ N} \]

\[ m = 730 \text{ kg} \]

\[ l = 6 \text{ m} \]

\[ v = v' \left( \frac{3}{2} \right) = 52 \text{ m/sec} \]

These are the input data for the dynamic stress analysis in Part II. For the further development of this project also the "steam blow" and "pipe missiles" would be considered.

3. Mechanistic Analysis

The task of mechanistic analysis is to explain quantitatively all the accidents—which the accident sequence analysis defines qualitatively—with their relevant physical phenomena. The emphasis is not on the further development of the physical knowledge of the occurred phenomena, rather on the various probabilities of these events. Effects such as \( \text{H}_2 \), \( \text{CO}_2 \) formation or possibly sumpwater-molten core reaction, etc., would be considered only as
far as their feasibility in this project and their relevancy to the results are felt.

For evaluating the reliability of steel hull, the physically relevant parameters are:
- The maximum pressure in the containment following LOCA or SLB, independent of time. (To make the results usable for the risk analysis, the time of maximum pressures would as well be determined.)
- The maximum temperature on the steel wall of the steel hull after LOCA or SLB.
- The impact velocity and the impact mass of the steam pipe or of the fragments of the broken pipe.

3.1 Calculation of the pressure in the containment

The calculation of the overpressure in the containment after a LOCA is being performed using the computer code ZOCO VI/2 of the IRA /5/, /6/. This model assumes a closed volume in thermodynamic equilibrium, into which water and steam mass flow is discharged. Plain walls with defined surfaces and thicknesses are considered representative for the internals, on which the condensation and heat transfer to the cold structures are simulated. This condensed water is collected together with the water of the blowdown phase in the sump.

The heat transfer to the environment occurs first via steel hull into the annulus and then from there through the secondary shielding to the outside. In the low pressure recirculation phase (after ~ 20 minutes), a further transfer of heat takes place through the Residual Heat Removal Systems. They take the warm sumpwater of the containment and feed it via the residual heat exchanger back to reactor pressure vessel. The injection rates into the containment are determined by the size of the leakages of the main coolant pipe and by the type and the number of failed systems of the ECC and residual heat exchanger. The steam fraction and the enthalpy of the mass-flow are calculated using:

- the computer code BRUCH /3/ during the Blowdown Phase (Fig.5),
- the computer code WAK /8/ during the Refilling Phase (Fig.6),
- analytical methods in the time-interval between the calculation-end of the code BRUCH and the calculation-begin of the code WAK,
- analytical methods during the low pressure recirculation phase (if vaporization occurs).

The accurate values of all the input data for the computer code ZOCO will not be given here again. However, to get a general impression, some main data values are shown:

- volume of the containment: 69,993 m³
- volume of the annulus: 11,200 m³
- heat transfer coefficient in the annulus considering heat radiation: 13 W/m² grad
heat transfer coefficient to the environment \(1 \text{ W/m}^2 \text{ grad}\)

The internals are simulated on the basis of following wall-parameters:

<table>
<thead>
<tr>
<th>wall</th>
<th>surface area (\text{m}^2)</th>
<th>wall thickness (\text{m})</th>
<th>number of layers</th>
</tr>
</thead>
<tbody>
<tr>
<td>crane and lattice floors</td>
<td>3539</td>
<td>0.009</td>
<td>3</td>
</tr>
<tr>
<td>accumulator</td>
<td>393</td>
<td>0.020</td>
<td>6</td>
</tr>
<tr>
<td>containment sump</td>
<td>361</td>
<td>2.40</td>
<td>16</td>
</tr>
<tr>
<td>walls and ceilings</td>
<td>27968</td>
<td>0.49</td>
<td>10</td>
</tr>
<tr>
<td>floor</td>
<td>2357</td>
<td>0.56</td>
<td>10</td>
</tr>
</tbody>
</table>

3.2 Probability Density Function of the Pressure

The pressure, which mounts in the containment during an accident, can not be described as an explicit function of the form \(p = p(x,y,...)\). It is calculated by a system of differential equations which are programmed in ZOCO. The probability density function of the pressure, therefore, cannot be calculated analytically; it must be approached by approximations.

For this purpose, the whole spectrum of the probable maximum pressures will be subdivided, depending on the required accuracy in different number of intervals; and then tried to variate the input parameter of the pressure calculations in such a way that in every pressure interval just one result of the pressure calculation should fall. In this way, it seems convenient to avoid the hardware systems which lead to the overpressure, and focus our attention to the effective physical parameters.

The effective physical parameters of the pressure, which will vary by accident sequence, are:

- energy input into the containment
- energy output from the containment
- mass flow rate into the containment
- leakage rate of the steel hull

With these considerations, Fig.4 was obtained as the first choice of the performed pressure calculations where the steel hull is always thought to be tight, since a leakage is taken as a failure of steel hull /7/.

The probabilities, however, are given only for the function or disfunction of the realistic technological systems (Fig.2). To evaluate the occurrence probabilities of the calculated pressures for all pressure cases (Fig.4), the configurations of function or disfunction of the technological systems are to be correlated and the probabilities be added according to the rules of probability calculations. One assumes also that all pressures can take place between the calculated figures, e.g., due to the partial function of groups of systems. Thus one can smoothen the gained histogram, which must fit the equation:

\[
\int_{0}^{P} \ dp = W(p)
\]
3.3 Results of the Pressure Calculations

For the case of A1 (Fig. 4, case 3, PWR design basis) the pressure calculations were performed. The results showed that the maximum value would reach after ~ 10 sec and would be about 3.9 kp/cm². The whole pressure distribution with respect to time can be seen in Fig. 7. The corresponding temperature curve in the steel hull wall is shown in Fig. 8. These values are the input values for the calculations in Part II and III. All the other necessary pressure calculations must still be performed. Also the pressure distribution for every case, 1 to 4 (Fig. 4), is to be evaluated. As a result of random load fluctuations, the different initial conditions for blowdown influence the pressure distribution rather strongly.

Furthermore, the standard deviations of the dominant parameters in ZOCO such as the condensation effects and the heat transfer through the steel hull wall, are also of importance and their influences are to be clarified by sensitivity analysis. Provisionally, the following assumed pressure distribution will be taken and applied in Part III.

mean value $\bar{E}(p) = 3.90$ kp/cm²
standard deviation $\sigma = \sqrt{\text{Var}(p)} = 0.2$ kp/cm²
variation coefficient $\nu_p = 0.05$

4. Reliability Analysis

As supplement to the deterministic concept of the "single failure criteria" /9/ for the design and construction of safety systems, the unavailability per demand or the failure probability in a time interval should be quantified only by probabilistic approaches. As generally known, this, in the framework of reliability analysis, is performed mostly by the fault tree methods /10/.

The quantitative evaluation of the extensive fault trees is accomplished exclusively by computer programs, many of which involve simulation procedures (Monte Carlo). In this reference case the computer code CRESSEX /11/ is employed.

4.1 Loss-of-Coolant Accident

Today, usual fault trees for the safety systems have extensively increased in the number of analysed coupled systems and in the number of the relevant components (about 1000 components, 600 logical couplings). It is now worthwhile to reconsider the existing fault tree analyses /12, /13/, and to examine these critically for the respective case of application. Since, in the framework of the German risk-study /4/, the fault tree analyses based on Biblis B power plant are being performed in IRA, it is reasonable to use the preliminary results of these fault trees provisionally. Moreover, the German study would analyse also the following rain systems which are of interest to us:
- emergency core cooling system
- component cooling system
- service water system
The analyses of these systems consider the respective logical connections of the reactor protection system, the energy supply, and the mechanical systems /12/. The calculations for the overall systems are performed /4/. In order to include the common mode failures, especially the failures having a common cause which occur due to the intermeshing of systems, in the analyses more-or-less completely, the system analyses were implemented in full detail up to the level of component and circuit diagrams /14/.

In the probabilistic analyses, the question of the type of distribution and its standard deviation for the estimated unavailabilities is one of the central problems.

Towards this end, a computer program STREUSL /15/ is developed in the IRA. Given the distribution and standard deviation for the input data - in the failure rates $\lambda$ - of the various components ($\lambda = 1/\text{MTBF}$ Mean Time Between Failure) this computer program would calculate the distribution and standard deviation in the relevant reliability figures. The first comprehensive test-calculations with this code showed that the system with respect to the frequency of individual component failures is very homogeneous. In such a case, the standard deviation in the results is not much different from the standard deviation in the failure rates. That means, the upper 95 % limit of the failure rate $\lambda$ is the medium value multiplied by 3; and the lower 5 % limit of $\lambda$ is the medium value divided by 3. Then it would correspond approximately to the circumstances of the unavailability of the overall system.

In addition, the effort is being made to find and use the actual distribution and standard deviation of the failure rates of the components and determine their effects on the results. Then, it will be possible to calculate the standard deviation of the mean value $P_{ij}$ of the number of accident consequences $S (A_{ij})$.

5. Conclusion and Future Aims

On the basis of the analyses performed up till now, it looks promising to evaluate the reliability of the steel hull by analysing the various accident sequences and the resulting pressure and load distributions.

In view if the immense calculation costs from the use of many data-wise coupled computer codes (BRUCH, WAK, ZOCO, CRESSEX, SAP) it is necessary to restrict the number of cases to be investigated. It is also required to find out the optimal conformity for the system resistance and the material behaviour in the analysed cases.

The current work will be extended to the calculation of further pressure cases and probabilities on the basis of the well-defined concept. A deeper understanding of the pressure distribution in respective accident sequences and of the deviations in frequencies of each accident sequence poses a major object of this research project. Subsequently, the small LOCA will also be given a detailed attention.
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MRR - i.e., Report of the Laboratory für Reaktorregelung und Anlagensicherung, Technische Universität München
Fig. 1: Schematic Diagram of Analysed Event Tree Cases

SLB .... Steam Line Break
SG ..... Steam Generator Heat Tube Break
CS ..... Containment Structure
AF ..... Auxiliary Feedwater System
RH ..... Residual Heat Removal System

Fig. 2: Event Tree for Large LOCA

Fig. 3: Condensed Event Tree for Steam Line Break

Fig. 4: Safety Injection Configuration
Fig. 5: Blowdown-Massflow to the Containment

Fig. 6: Energy Flow Rate to the Containment during Refill Phase

Fig. 7: Pressure Distribution in the Containment following a LOCA

Fig. 8: Temperature-Time Curve for the Containment following a LOCA