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CURRENT GERMAN INVESTIGATIONS OF FURTHER DEVELOPMENT OF PSA FOR NUCLEAR POWER PLANTS

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ABSTRACT

The supreme regulatory authority has established a comprehensive investigative programme which includes the development of methods and models to improve the PSA tool and to use its results in the licensing and supervising procedures for nuclear power plants. The main topics of this programme and first results are presented, in particular the comparison of methodology and principal results of recent international PSA studies.

1 INTRODUCTION

The prime responsibility for the safe power plant operation ist assigned to the operating organisations. Nevertheless, Federal Ministries and institutions in the Federal Republic of Germany performed investigative programmes in order to give input and direction to improved safety assurance. The Federal Ministry for the Environment, Nature Protection and Nuclear Safety (BMU) has established in its responsibility as the supreme regulatory authority a Nuclear Regulatory Research Programme. This programme is now mainly implemented at the Federal Office for Radiation Protection (BfS), an autonomous Superior Federal Authority within the portfolio of the BMU. This programme covers all areas of nuclear safety, in particular with respect to nuclear power plants.

One topic of this programme is the further development of PSA methodology and application. The increasing importance of PSA in the Federal Republic of Germany is based on the progress achieved by several detailed studies like the German Risk Study - Phase A and B [1, 2] - and the recent Probabilistic Safety Analysis for a BWR [3]. Backfitting measures, plant modifications and major changes of operational procedures have resulted from such studies. The experience with PSA finally led the supreme regulatory authority to the decision, that a PSA has to be performed for each nuclear power plant as part of the periodic safety reviews that in future have to be performed by the licensee any ten years of operation.

PSA constitutes an effective investigative tool to supplement but not at all to replace the traditional deterministic approach. The PSA not only grants a quantitative perspective on plant safety but also provides a more balanced and realistic picture. Both approaches are subject to assumptions and uncertainties. The PSA makes them more transparent and allows to assess their relative safety significance.

The most important results provided by a PSA are engineering insights, i.e. the identification of vulnerabilities to a high safety level of the plant and the assessment of alternative means for improvements. Such results usually are not undermined by the numerical uncertainties involved.

Further comprehensive research into methods and data gathering should be pursued in order to reduce the uncertainties in the results and extent the area of issues that be investigated systematically by PSA methodology.

PSAs should account for common cause failures as well as some human errors and should depict uncertainties. Additional development is needed to account for errors of commission and influences of the operating organization. Moreover, the scope of the basic level-1 studies is currently being extended to cover other operational states than full power operation. A special topic of the above mentioned investigative programme is the further development of PSA taking into account international experience.

2 FURTHER DEVELOPMENT OF PSA FOR NUCLEAR POWER PLANTS

For the further development of PSA in the Federal Republic of Germany, a major investigative programme has been defined by BfS and BMU which has started at the end of 1991 and the beginning of 1992. Main topics of this programme are:

- comparison of different international PSA studies with respect to methodology, models and results in order to gain insights for further investigations and developments in the Federal Republic of Germany,
- characterisations of the state of safety technology by using qualitative and quantitative PSA results,
- improvement of methods for special issues like human factor dependencies, low power and shutdown operation as well as uncertainties,
- development of a proposal for a periodical precursor report system taking into account international experiences,
- PSA for fire events including a proposal for a guideline for this specific topic within PSA,
- containment performance related to severe accidents,
- evaluation of existing computer codes based on explicit event sequences and fault trees of real plants comparing the results, calculation rules, input parameters and techniques to determine the cut-sets; development of a procedure for the qualification of PSA-codes,
- evaluation of existing PSA applications for passive and inherent safety features; resulting proposals shall be checked on the basis of examples,
- contribution to an integral risk management approach investigating different energy supplying systems,
- further investigations for providing a living PSA tool,
- improvements and further development of the PSA guideline based on experiences of the users and new scientific knowledge as well as
- a first approach to a guideline for the review of PSAs which are performed within the framework of periodic safety reassessments.

The approach to precursor evaluation and to PSA of fire events (both investigations are completed) shall be presented elsewhere (cf. [4], [5]). The comparison of different PSA studies which takes into account - among others - initiating events, plant status, event sequence analysis, and accident management procedures, is under progress, first results are discussed below in more detail.

3 COMPARISON OF INTERNATIONAL PSA STUDIES

Several PSA studies in order to evaluate the safety of nuclear power plants (PWR and BWR) have been completed in recent years, among others, in the Federal Republic of Germany, France, Japan, Sweden and the USA (cf. Table 1).

Table 1. Characteristics of recent PSAs for PWR and BWR.

PWR Plants	Surry	Sequojah	REP 900	REP 1300	Biblis B	Japanese PWR	Ringhals 3/4
Country	US	US	France	France	Germany	Japan	Sweden
MWe	790	1 150	900	1 300	1 300	1 100	915
Vendor	Westinghouse		Framatome		Siemens		Westing-house
Nr. of Loops	3	4	3	4	4	4	3
Containment Design	subatmos-pheric	ice condenser	large dry	large dry	large dry	large dry	large dry
Analysis Level	3	3	1	1	2	1	1
Plant Conditions	full power generation		full power generation, low power, shut down		full power generation	full power generation	full power generation
Types of Initiator	internal, some external	internal	internal	internal	internal, external	internal	internal
State of the Plant	August 1988	August 1988	January 1990	January 1989	August 1988	March 1988	January 1992

BWR Plants	Peach Bottom	Grand Gulf	Japanese 1100/MWe BWR	Forsmark 1/2	Gund-remmingen II
Country	US	US	Japan	Sweden	Germany
MWe	1 150	1 250	1 100		1 300
Vendor	GE	GE		Asea Atom	KWU
Containment Design	Mark I	Mark III	Mark II		Series 72
Analysis Level	3	1	1	1	1
Plant Conditions	full power generation		full power generation,	full power generation	full power generation
Types of Initiator	internal, some external	internal	internal	internal	internal, external
State of the Plant	August 1988	August 1988	September 1990		October 1992

The first step of a comparison of different international PSA studies is the comparison of the principal results of these studies and of the insights derived from these results, as well as the identification of generic problems common to all studies and of approaches to their solution. This includes the identification of commonalties and differences in data and methods, and the discussion of their impact on the results. A top down comparison with an identification of dominant sequences and system unavailabilities has been performed which provides insights of general interest for the use and improvement of PSAs.

Furthermore, distinction is made between accident initiating events originating within the plant (plant internal initiating events) and events originating outside of the plant (plant external initiating events). Finally, the analyses distinguish between initiating events in the state of power generation and during non full power operation of the plant.

First results and general observations of the ongoing investigations are described below.

3.1 Comparison of the PSA-studies for PWR

The design of the newer plants REP 1300, Biblis B and Japanese 1100 MWe PWR includes more redundancies and better spatial separation of redundant trains, and an increased use of automatic systems relative to the older plants Surry, Sequoyah and REP 900.

The core melt frequencies for the newer plants are about one order of magnitude lower than for the older plants, if compared on basis of power generation states (cf. Fig. 1). This significant reduction of core melt frequencies is accompanied by a substantial increase of the relative contribution of common cause failures and a corresponding reduction of the relative contribution of human errors of omission. However, the relative contribution due to operator errors of commission, when introduced in the PSA, increases for the newer plants.

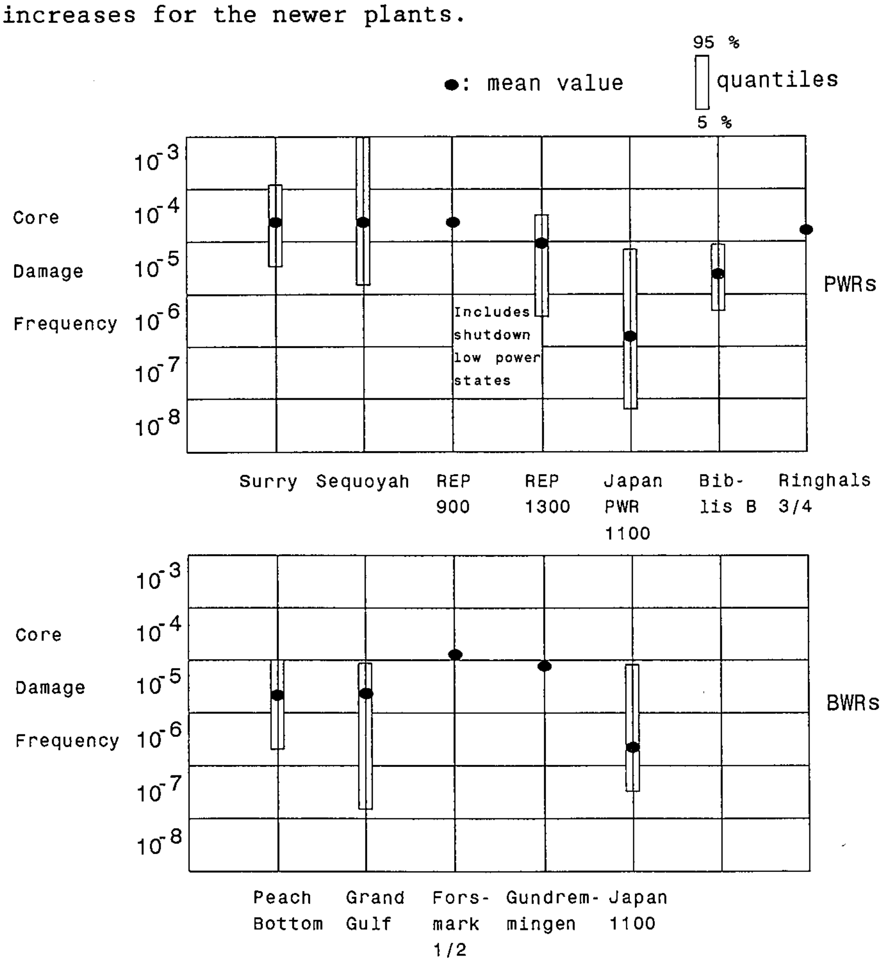

Fig. 1. Core damage frequencies resulting from internal events.

For example contribution of the inadvertent shut off of safety injection is higher in EPS 1300 than in EPS 900, due to the lower failure probability of the systems (cf. Table 2).

In general due to the high effectivity of preventive AM measures for transients, system unavailabilities for transients are significantly lower than for LOCAs, the exception being unavailabilities of the systems needed to cope with LOSP cases at the two US plants Surry and Sequoyah.

Aside from the Surry plant, contributions of SLOCA events to core damage frequencies are dominant. Between the Surry, REP 1300, Biblis B, Ringhals 3/4 and Japanese 1100 MWe PWR plants, system unavailabilities differ little for that case. The core damage frequency then mainly depends on the initiator frequency of the SLOCA, which may be an artefact of the analysis, determined by the choice of the data base and the number of reactor years considered in it. This conceptual problem may lead to sizeable uncertainties which do not show up in the quantified uncertainty distribution.

Table 2. Core damage: risk reduction importance measures* for important systems.

PWRs	Surry	Sequoy-ah	REP 900 Power Generation [3]	REP 1300 Power Generation [3]	Biblis B	Japanese PWR	Ringhals 3/4
System for Emergency Power Supply	65	25	1,3 (1,6)	1 (1)	-	3,8[1]	6,6
Auxiliary-Feed Water Supply	19	22	12 (11)	20 (10)	6[2]	6,1	15,6
HP-ECCS	8,5	42	3,7 (2,9)			82	21
LP-ECCS	10,2	18	35 (28)	10 (5)	87[1]	84	60
Secondary Side Cooldown x_t	-	-	2,3 (1,7)	-	1,5	0,1	19
Systems for the Control of SG-Tube Ruptures	4,6	2,8			1	4,5	11
Reactor Protection System	4,2	-	13 (9)	32 (16)			-
Secondary Side Bleed and Feed	-	-	-	-	10,3	-	-
Primary Side Bleed	8,8	4,6	1,7 (1,2)		8,6	10,1	9,0

[1] For Emergency DGs
[2] Includes emergency feed by cross-ties from Unit A
[3] The figures in parenthesis refer to core damage frequency for all states

$$* \ 100\% \times \left[1 - \frac{\text{total core damage frequency with system unavailability} = 0}{\text{total core damage frequency with nominal value of system unavailability}}\right]$$

3.2 Comparison of PSA-studies for BWR

As one can see from Fig. 1 the core melt frequencies differ little with the exception of the Japanese plant.

At the Gundremmingen, Forsmark 1/2 and Japanese plants the dominant cause for core damage is early loss of RPV injection (cf. Table 3).

At the two US-plants loss of electrical power is the dominant cause for core damage. Apart from this particular weakness the core damage frequencies of the US-plants would be very low, with nearly loss of RPV injection as dominant cause.

Table 3. Core damage: risk reduction importance measures* for important systems (* cf. Table 2).

BWRs	Peach Bottom	Grand Gulf	Forsmark 1/2	Gundremmingen	Japan 1100 MWe
Systems for Electric Power Supply	49	97.5	2	15	1.8
Decay Heat Removal	31	-	6.7	6.4	4
Feed Water Supply to RPV	52	92	83	90	54
Depressurization of RPV	6.4	5	76	32	65
Pressure Limitation System	-	-	-	4	-
Reactor Protection System	41	3	4	13	15

The contribution of loss of heat removal from the suppression pool is insignificant because of the long times and many alternatives being available for AM and recovery actions.

Loss of coolant accidents are generally insignificant. However, the calculation of LOCA contributions rests on the assumption that sump recirculation is not adversely affected by clogging of strainers at the suction lines. In view of the Barseback event in July 1992 some justification of this assumption may be required.

3.3 General insights from the PSA comparison

The largest impacts on the results and their uncertainties are consistently observed for human influence, common cause failures, initiator frequencies for LOCAs, in particular SLOCAs in PWRs.

This underlines the need for intensified efforts for improved quantification of these issues; this is particularly important for the SLOCA case where the computed numerical results are associated with the combined uncertainties of the initiator frequencies and the common cause failure data.

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