SKI Report 96:40

Status and Use of PSA in Sweden

Michael Knochenhauer

May 1996

ISSN 1104-1374
ISRN SKI-R--96/40--SE

SKI

STATENS KÄRNKRAFTINSPEKTION
Swedish Nuclear Power Inspectorate

SKI Report 96:40

Status and Use of PSA in Sweden

Michael Knochenhauer

Logistica Consulting AB,
Domkyrkoesplanaden 5B, S-722 13 Västerås, Sweden

May 1996

SKI Order Number 95235

NORSTEDTS TRYCKERI AB
Stockholm 1996

Table of contents

Summary

The performance and use of PSA:s in Sweden goes back about two decades. During all of this time, the field of PSA has been developing intensively, both internationally and within Sweden. The latest years have been characterised by an increased use of PSA models and results, and by major extensions of existing PSA models.

The aim of this document is to describe PSA in Sweden with respect to development, scope and maturity, as well as to the contents of the analyses and the use of results. PSA activities will be described from the point of view of both the authorities and the utilities.

The report gives an overview of the development within the area of PSA in Sweden, both its history and current trends. The aim has been to include a reasonable amount of detail, both on the methods and results in PSA:s performed and on the numerous supporting research programs dealing with various aspects of PSA.

Sammanfattning

Probabilistiska säkerhetsanalyser (PSA) har genomförts och använts i Sverige i omkring två decennier. Under hela denna tid har området varit under intensiv utveckling, både i Sverige och internationellt. De senaste åren kännetecknas av en ökande användning både av PSA-modeller och av resultat från analyserna. Parallellt med detta har PSA-modellerna utökats avsevärt.

Syftet med detta dokument är att beskriva PSA i Sverige med avseende på utveckling, omfattning och mognad, samt att beskriva resultatanvändning och analysernas innehåll. Beskrivningen är gjord både ur myndighetens och kraftbolagens synpunkt.

Rapporten ger en överblick över PSA:s utveckling i Sverige historiskt, och beskriver aktuella trender och utvecklingsområden. Ambitionen har varit att inkludera en rimlig mängd detaljinformation, både vad gäller metoder och resultat och med avseende på de många pågående och avslutade forskningsprogram som rör olika aspekter av PSA.

Introduction

There are in all twelve nuclear power plants in Sweden, generating a total of about 10 GWe per year, i.e. somewhat less than half the total Swedish electricity production. Nine of the plants are boiling water reactors (BWR) of ABB Atom design, and three are Westinghouse pressurised water reactors (PWR). The first plant to be taken into operation was Oskarshamn 1, a 462 MWe BWR which entered operation in 1972; the newest plants are the Forsmark 3 and Oskarshamn 3 1200 MWe BWR:s entering operation in 1985. In all, the BWR plants represent four reactor generations, and the PWR plants two. Attachment A gives some characteristics of the plant generations, with emphasis on the basic safety features of the plants.

Probabilistic Safety Analyses (PSA) of Swedish nuclear power plants have been performed since the middle seventies, and probabilistic analyses have been increasingly used during the eighties and nineties. During the latest decade, there has also been a marked trend in PSA work away from infrequent major efforts resulting in general purpose analyses, towards frequent limited analyses with specific purposes.

> For the benefit of those readers who are unfamiliar with the concept of probabilistic safety assessments of nuclear power plants, Attachment B gives a short overview of the structure and contents of a PSA.

A characteristic feature of Swedish PSA activities, is the iterative manner in which they have evolved. Thus, the pace, direction and contents of PSA development has been guided mainly by conclusions from the performance and use of previous analyses. In parallel with the performance of the PSA:s, continuous and rather extensive research projects have been launched in order to address specific problems encountered.

PSA activities have also been both influenced and, at least to some extent, limited by the level of experience and competence within the Swedish PSA community. This competence has gradually changed bias from "learning how best to perform a PSA" to "learning how best to make use of a PSA".

The steadily increasing maturity and the continuous activities within the field of PSA, has lead to a growing acceptance of PSA as a tool for supporting decision-making in safety related matters.

These positive introductory statements do not preclude the existence of grey areas in performed PSA:s and in Swedish PSA activities in general - there are a number of areas where the development or extension of existing analyses is highly desirable.

Development of PSA in Sweden

During the design, licensing and commissioning of the Swedish nuclear power plants, safety related activities largely relied on deterministic analyses. As the assessment and comparison of risks associated with the operation of nuclear power plants became increasingly important, probabilistic analysis were identified as a suitable tool for safety evaluation.

A schematic overview of PSA activities in Sweden can be based on five development phases, as described in table 1. The table will be used as a starting point for an overview of the development of PSA in Sweden.

Table 1 *Overview of Swedish PSA activities*

Phase	Keywords	Activities
1974 - 1980 Early activities	• Risks from close location of nuclear power plants • Comparison with WASH-1400 • Beginning of systematic use of PSA • Focus on accident mitigation	• Urban Siting Report (Närförläggningsutredningen) • Government Energy Commission (Energikommissionen) • Reactor Safety Investigation (Reaktorsäkerhetsutredningen)
1980 - 1985 The first round of periodic safety reviews (ASAR 80)	• Performance of basic analyses • Development of analysis tools • Discussion of analysis format	• PSA level 1 • Initiation of the ASAR 80 programme • NKA/SÄK Nordic Research Program
1985 - 1990 The Post ASAR 80 period	• Data collection and evaluation • Optimisation of Technical Secifications • CCF models • Containment integrity	• Updates of initial level 1 PSA:s • Conclusion of the ASAR 80 programme • SUPER-ASAR - A comparative review of Swedish PSA:s • NKA/RAS Nordic Research Program • Development of efficient fault tree analysis tools
1990 - 1995 The second round of periodic safety reviews (ASAR 90)	• Completeness of existing PSA models • Modelling of CCI:s (Common Cause Initiators) • Living PSA	• PSA level 2 • PSA for shutdown period • Initiation of the ASAR 90 programme • NKS/SIK Nordic Research Program • APRI - Research on accident phenomena
1995 - 2005 The Post ASAR 90 period	• External events • Utilisation of PSA results • Time-dependent analyses • Safety indicators • Design review • Living PSA • Quality assurance of PSA:s	• PSA for external events • Conclusion of the ASAR 90 programme • NKS/RAK Nordic Research Program • Increased level of detail of LOCA analyses • Modelling of signals and power supply

PSA Activities During the Seventies

In parallel with the Reactor Safety Study (WASH-1400), the Swedish Government initiated an analysis, which resulted in the publishing of the Urban Siting Report in 1974[1]. In principle, the analysis used the same methods as WASH-1400, but on a lower level of detail. It modelled accident sequences for a reactor located in the vicinity of Stockholm, and aimed at estimating the risks involved with such a location.

In 1977, the SKI and the Government Energy Commission initiated a number of analyses which aimed at using methods and assumptions from the WASH-1400 analysis in comparative analyses of two Swedish nuclear power plants. Two analyses were made for Barsebäck 2, and one for Forsmark 3. All three analyses rely heavily on WASH-1400 and present results that lie within the uncertainty margins of the results obtained by WASH-1400. In the case of Forsmark 3, the possible safety enhancement from a number of design changes was demonstrated.

After the Three Mile Island accident (TMI), the Reactor Safety Investigation (RSU)[2] was initiated in 1979, with the aim of

- considering if there was reason to change the general assessment of the level of safety in the production of electrical energy in nuclear power plants, and
- proposing possible safety enhancing measures in Swedish nuclear power plants and indicating the need of research concerning such measures.

The conclusions of the RSU showed that there was no reason to change the conclusions from previous assessments of the level of safety in nuclear power plants. However, it was stressed that both previous risk assessments and the TMI accident indicate the need for considerably increased requirements on safety activities in connection with nuclear power. These requirements should apply to all parts of nuclear activities, from the design of the plants and of their safety systems, through the activities of the supervising authorities, to the day-to-day safety work in connection with the operation and maintenance of the nuclear power plants.

The RSU also stressed the need to properly evaluate experiences from disturbances and incidents occurring during plant operation and outages in order to prevent accidents. Among the preventive measures mentioned, was the utilisation of probabilistic methods in safety review of the nuclear power plants. It was therefore recommended to perform PSA:s for all Swedish plants.

It was also stressed, that there is always a risk for future accidents. Therefore, the RSU recommended that more attention should be given to measures aimed at mitigating the consequences from such accidents. This recommendation was later to result in considerable research efforts in connection with accident mitigating systems, and ultimately resulted in the design and installation of filtered venting systems in all Swedish nuclear power plants. This programme was executed and implemented mainly during the second half of the eighties.

PSA Activities During the Eighties

PSA activities during the eighties were dominated by the performance of the first round of periodic safety reviews (ASAR 80), which included an internal event level 1 PSA:s for all Swedish nuclear power plants.

The performance of the PSA:s resulted in a number of spin-off projects, mainly concerning:

- Comparative review of PSA:s
- Treatment of special issues
  (e.g. common cause failures and human reliability)
- Use of PSA for safety related applications
  (e.g. development of Technical Specifications)
- Increase of realism
  (by making systematic use of experience data)

ASAR 80, the First Round of Periodic Safety Reviews

As previously stated, a number of probabilistic safety analyses had already been performed at the time of the general RSU recommendation on performing PSA:s. These analyses had resulted in the following attitude towards PSA:

- PSA was seen to be a promising tool as a supplement to the traditional (deterministic) safety analysis.
- PSA had proved to make possible a mapping of the risk picture of a plant. This mapping was found to provide an excellent basis for further decisions on safety enhancing measures or for evaluating and prioritising proposed modifications.
- The performance of PSA:s provided a useful basis for the systematic evaluation of disturbances and incidents.
- Due to inherent uncertainties in data and accident models, PSA was found to be less suited for deciding if a certain technical activity is acceptable from the point of view of risk or in relation to other activities.
- PSA could be used for training plant personnel to manage various accident situations and to develop emergency operating procedures.

In 1981, following the recommendations of the RSU, the Swedish Parliament ruled, that every nuclear power plant should be made subject to at least three complete safety reviews during its useful life. Reports were to be submitted every 8-10 years to the Government by SKI. These reports were to be compiled on the basis of analyses carried out by the utilities. The intention was, that the depth of the reviews should correspond to the Final Safety Analysis Reports (FSAR) required for the original licensing of the plants. For this reason, the acronym ASAR (As-operated Safety Analysis Report) was chosen for these periodical safety review. The principal emphasis of the ASAR was to be varied from safety review to safety review.

In 1982, the SKI issued guidelines for the first round of ASAR:s, stressing the following purposes:

- The most important purpose of the ASAR work is the periodical performance of a comprehensive systematic auditing of the safety status of each plant. Thus, the ASAR work will help both the utilities and the SKI to supplement the usual focus on the next operating year with a view of plant safety in the longer perspective.
- The ASAR work should promote systematic documentation and transfer of experience.
- The ASAR work should include a systematic review and evaluation of measures required in order to maintain and improve safety in a 3-5 year perspective.

A typical list of contents of an ASAR 80 report, would include the following items:
1. Organisation and quality assurance
2. Operating experience
3. Quality control - operation and maintenance
4. PSA level 1
5. Training and personnel
6. Safety improvements implemented during the reporting period
7. Planned and ongoing safety improvements

Thus, during the eighties, level 1 PSA:s were performed for all Swedish nuclear power plants, starting in 1980 with Ringhals 1 and Ringhals 2, and ending in the beginning of the nineties with Ringhals 3 and 4. Resources spent on the initial PSA:s were in the order of 5-10 man-years.

The SKI guidelines did not contain any detailed descriptions on how to perform the ASAR. This applied also to the ASAR requirements concerning the performance of a PSA, i.e. no specific recommendations were given on the choice of methods or on the layout, contents and level of detail of the analysis. This has resulted in considerable differences between the analyses performed. The differences reduced the comparability of the PSA:s, but also contributed to a rapid early development of PSA, by encouraging the development and testing of alternative methods, and by improving the possibilities to detect problem areas.

Some common features of this first round of PSA:s, is that they were limited to internal events (transients and LOCA:s (loss of coolant accidents)) and that they constitute level 1 PSA:s, i.e. they estimate the frequency of core damage, and identify the dominant contributors to this frequency. Furthermore, due to initial constraints in the capacity of fault tree analysis codes, the level of detail in the fault tree models for support systems was limited.

For a number of plants, analyses of some external events, mainly internal fires, were initiated immediately upon the completion of the level 1 PSA.

Research Projects

The performance of the first round of PSA:s generated a multitude of experiences and resulted in the identification of some major problem areas, such as:

- Treatment of common cause failures (CCF)
- Treatment of human interaction
- Treatment of uncertainties
- Component failure data
- Data on frequencies of initiating events
- Computer codes for fault tree analysis
- Consistency and general comparability of the PSA:s performed

A number of research projects and development programs were initiated in order to address identified problems related to methodology and reliability data. This section gives an outline of some of the most important projects. Further details on the contents and conclusions from some of the projects are given in Attachment C.

PRA Uses and Techniques (NKA/SÄK-1)[3]

The project was the first systematic review of the possibilities and limitations of PSA techniques. The project was a Nordic four-year effort (1981-84), aiming at evaluating and comparing the methods and computer codes that were available at the time.

Within the project, Benchmark exercises were performed for the quantification of the system model of a PWR high pressure injection system, and for the modelling of a BWR loss of feedwater transient. In the Benchmark exercises, a variety of data sources and computer codes were used and compared, and the modelling was made using different modelling methods (cause-consequence diagrams vs. event trees, reliability block diagrams vs. fault trees).

The most important outcomes from the project were connected with

- further development of analysis codes,
- methods for identification of potential CCFs,
- methods for modelling CCF in four train systems, and
- statistical methods for treatment of field data.

The project was of great value to the ongoing Nordic PSA work and in the development of the Swedish Reliability Data Handbook (T Book). The results and conclusions from the SÄK-1 project were also used as a basis for continued analysis, especially in the areas of CCF analysis and component failure data analysis.

Optimisation of Technical Specifications by Use of Probabilistic Methods (NKA/RAS-450)[4]

The NKA/RAS-450 project aimed at providing a framework for the analysis of issues related to the evaluation and optimisation of Technical Specifications. The project was a Nordic five-year effort (1985-89) dealing primarily with the:

- optimisation of Limiting Conditions of Operation (LCO), including Allowed Outage Times (AOT) of components,
- optimisation of Surveillance Test Intervals (STI), including analysis of test strategies,
- planning and evaluation of preventive maintenance during power operation,
- analysis of testing, and
- analysis of failure data.

The project is described in more detail in Attachment C.

Dependencies, Human Interaction and Uncertainties in Probabilistic Safety Assessment (NKA/RAS-470)[5]
Three areas were investigated in a five year Nordic programme (1985-89):
- dependencies with special emphasis on common cause failures,
- human interaction, and
- uncertainty aspects.

The approach was based on comparative analyses in the form of Benchmark analyses, reference studies and retrospective reviews. Weak points in available PSA:s were identified, and recommendations were made aimed at improving the consistency of the PSA:s. The sensitivity of PSA results to basic assumptions was demonstrated and the sensitivity to data assignment and choice of methods for analysis of selected topics was investigated. The outcome of the project was an important input to the SUPER-ASAR project.

The project is described in more detail in Attachment C.

The SUPER-ASAR Comparative Review of Completed PSA:s[6]
In 1986, PSA:s had been performed on eight out of the twelve Swedish nuclear power plants. The review of these analyses, carried out by SKI, indicated significant differences in scope, degree of detail, coverage, etc. Application of a broad spectrum of methods and assumptions have had a decisive impact on PSA results, which made a thorough comparison complicated. This was the background to the SUPER-ASAR programme launched by SKI in 1986.

The main objectives of this project were
- to survey and compare the results of Swedish PSA:s with due consideration for the differences in assumptions, modelling and completeness,
- to facilitate the use of completed PSA:s in the decision-making process, and
- to establish priorities for research projects within the area of PSA.

The project was carried out in two phases. During the first phase, the qualitative features of the studies were reviewed, including qualitative methods and data selection. In the second phase, a quantitative analysis was made of the discrepancies identified in the first phase.

The project is described in more detail in Attachment C.

The SKI Review of the Safety Status of Swedish Nuclear Power Plants
In 1989, the Swedish Government requested a separate study of the safety situation at the twelve nuclear power plants in Sweden[7]. The request was related to the decision that was to be taken in 1990 regarding which two units were to shut off by 1995-96. Therefore, the study was partly aimed at determining if the plants could be ranked with respect to safety.

The study, carried out during the second half of 1989, covered various aspects of safety, e.g. operational experience, significant events, lifetime and ageing of components, OSART assessments, competence and quality assurance. As part of the study, all previously performed PSA:s, were again reviewed.

The report concluded, that the PSA:s had not been intended for comparison between different plants. It was also noted, in accordance with the results from the recently concluded SUPER-ASAR project, that the PSA:s exhibited a lack of consistency with regard to choice of models, coverage, level of detail, etc. At the time no level 2 analyses existed and analyses of external events were only being initiated, which resulted in an inherently incomplete risk picture. There was also a need for further refinement of models for human interaction and common cause failures.

Thus, the differences in the calculated total core damage frequencies in the PSA:s were found to be dominated by the uncertainties inherent in the PSA methodology and the data bases used, as well as by incompleteness. Therefore, no conclusions in terms of safety ranking could be made.

PSA Activities During the Nineties

The ASAR 80 programme and parallel activities resulted in basic PSA:s being performed for all Swedish nuclear power plants. It has also resulted in a rapid development of methods, data bases, and areas of application of PSA:s, as well as of computer tools for modelling and quantification of PSA:s.

However, the existing PSA:s were still limited in coverage by basically including only internal events and being restricted to level 1 consequences. Therefore, the ASAR 90 programme has placed increased stress on providing an integrated risk picture, suitable for the living PSA approach that had started to evolve during the late eighties. The PSA extensions required in ASAR 90 involve:

- Operating modes, i.e. inclusion of both power operation and shutdown periods
- Initiating events, i.e. inclusion of both internal and external events
- Consequences, i.e. both level 1 (core damage) and level 2 (radioactive releases)

As part of the extension of the scope of existing PSA:s, more detail was also added to the event tree and fault tree models concerning e.g.:

- Common Cause Initiators (CCI)
  Previous analyses have shown that CCI:s may give dominant contributions to

the total core damage frequency. Therefore, the identification and modelling of relevant CCI:s is crucial in order to obtain a relevant risk profile.

- LOCA categories
  Current LOCA models are limited in their ability to model LOCA consequences in detail, including dynamic effects from LOCA. In parallel with activities aimed at improving the basis for assigning LOCA frequencies, LOCA models were further developed.
- Modelling of electrical power supply and signals
  These systems must be modelled in sufficient detail in order for analyses of CCI, internal fire and flooding to give relevant results.

ASAR 90, the Second Round of Periodic Safety Reviews

Guidelines for the second round of ASAR Reports, ASAR 90, were issued by SKI in 1991. Compared to the previous ASAR, these guidelines place more stress on the analysis of organisational aspects of safety, of experiences gained in the operation and maintenance of the plants, and of aspects related to the increasing age of the plants.

For the PSA:s, the guidelines call for a substantial increase in scope:

- Findings from the SUPER-ASAR project are to be implemented
- The coverage of the analysis is to be extended (CCI, external events)
- More operating modes shall be covered (shutdown period)
- A level 2 PSA shall be performed (analysis of containment performance and radioactive releases)

The first plant to submit an ASAR 90 was Oskarshamn 1 (1993). The entire programme will be completed within the next five years. Table 2 shows the current status of the ASAR programme.

Table 2 *Status of the ASAR programme*

Plant	ASAR 80 Report	ASAR 90 Report
Barsebäck 1/2	1985	1995
Forsmark 1/2	1991	1998-00 p
Forsmark 3	-	1996 p
Oskarshamn 1	1982	1993
Oskarshamn 2	1987	1996-97 p
Oskarshamn 3	-	1996 p
Ringhals 1	1984	1994
Ringhals 2	1983	1994
Ringhals 3/4	1991	1999-01 p

p = planned

Research Projects

In parallel with the update and extension of plant PSA:s that are to be performed as part of the ongoing ASAR 90 programme, analyses and research projects will be carried out within a number of areas.

As previously, many of the areas aim at increasing the level of realism in the existing PSA:s by making systematic use of the operating experience gained. However, while the focus previously had been mainly on Swedish and Nordic experiences, some of the present projects aim at making use of world-wide experiences.

Safety Evaluation by Living PSA (NKS/SIK-1)[8]

The project was a four year Nordic effort (1990-94) dealing with definition and demonstration of the use of living PSA (LPSA) for safety evaluations and for the identification of improvements in operational safety.

Routines and procedures of how to utilise LPSA were demonstrated in case studies. The demonstrations include applications such as planning of surveillance tests and test schemes, maintenance planning, optimisation of limiting conditions of operation and risk control of exemptions from Technical Specifications.

The project is described in more detail in Attachment C.

Accident Phenomena of Risk Importance (APRI)[9,10]

The project is performed in co-operation between SKI, the Swedish utilities and TVO (Finland). It was initiated in 1992, and it's first phase was finished in 1995.

The aim of the project was

- to provide a basis for evaluation of phenomena occurring in connection with severe accidents, and participation in probabilistic analyses of these phenomena,
- to support experiments aimed at validating and developing MAAP and other analysis tools, and
- to develop the knowledge basis required for the further development of accident management methods.

At present, the second part of the APRI project is being started. It will be performed during the period 1996-98.

The project is described in more detail in Attachment C, which also gives an outline of the plans for the second part.

Methods for the Analysis of External Events[11]

External events are to be analysed as part of the ASAR 90 programme. In order to support these activities, SKI and the utilities have decided to develop a common basis for the performance of these analyses. Thus, a three year project was initiated in 1994 with the aim of evaluating the state of the art within the field of external events and to propose an analysis approach (excluding seismic events, which are handled in a separate project). During the first year of the project, the following areas were covered:

- classification of external events,

- selection of relevant external events,
- estimation of frequencies of rare external events,
- performance of internal fire analysis,
- performance of internal flooding analysis, and
- mapping of room dependencies of safety components.

The work concerning *classification of external events* has resulted in a general model for the identification and grouping of various types of initiating events, and in a system for naming initiating events. The results will be used as a basis for coming extensions of the Swedish I Book, the data handbook on frequencies of initiating events.

The work concerning *selection of relevant external events* represents an overview of available methods for screening of initiating events. The recommended approach is iterative. A complete identification of potentially relevant initiating events is crucial. Thereafter, simplified frequency and consequence estimates are made in order to screen out non-critical initiators. Increasingly sophisticated methods are used for the screening of the remaining initiators, in order to arrive at a final set of relevant initiators, that will be included in the PSA.

To date, Swedish PSA:s have mainly included initiators whose frequency has been decided using experience data (transients) or by application of generic approaches (LOCA:s). The inclusion of rare external event initiators has highlighted problems concerning the *estimation of frequencies of rare external events*. A review has been made of available methods for frequency estimation.

A review and evaluation of some available methods for the *performance of internal fire and internal flooding analysis* has been made. The work was partly based on a literature review, and mainly includes Swedish and U.S. fire and flooding analyses. The description of results describes the methods studied and summarises general conclusions within specific areas of analysis involved:

- determination of occurrence frequency,
- detection and mitigation,
- propagation,
- impact on safety components, and
- operator interaction.

The *mapping of room dependencies of safety components*[12] aims to provide guidance on the required and sufficient level of detail in PSA component and system models with respect to

- the selection of relevant systems and subsystems,
- the mapping and modelling of the dependencies of individual components, and
- the identification and modelling of relevant failure modes.

A general approach for the identification of potentially important areas where dependencies between systems and components may be strongly affected by external events has been suggested, and is illustrated in figure 1.

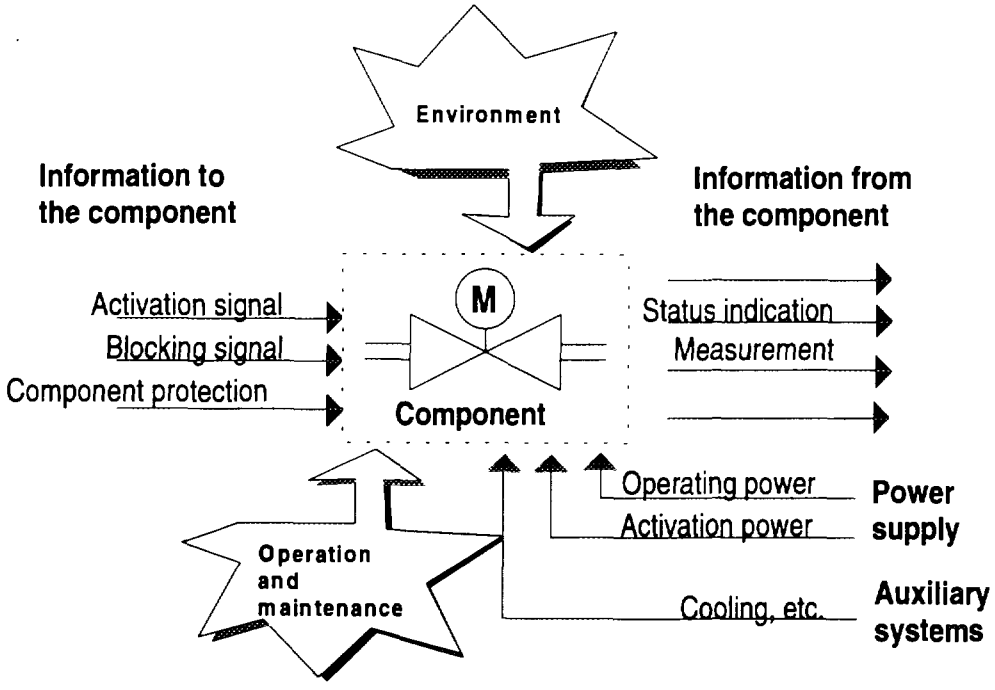


Figure 1 *Overview of component interactions affected by external events*

The second project year (1995-96) will concentrate on the determination of fire initiation frequencies, modelling of fire sequences including consideration of fire protection, and identification of relevant component failure modes as well as methods for modelling of relevant room dependencies.

Strategy for Reactor Safety (NKA/RAK-1)[13]
A new Nordic four year programme was launched in 1994. The NKA/RAK-1 programme includes the following subprojects:

- Investigation and evaluation of the safety work
- Initiating events - Estimation of pipe rupture frequencies
- Integrated sequence analysis with focus on human reliability
- Maintenance strategies and ageing

The general objective of the programme is to investigate how a sufficient level of safety can be achieved in practical work and what strategies and methods should be used.

The subproject *Investigation and evaluation of the safety work* deals with the interaction between safety objectives/requirements for the operation of nuclear power plants and requirements regarding the design and operation of the plants. Thus, it addresses the questions of

- whether the safety work is adapted to its purpose, or if there are gaps in potentially critical areas, and
- whether the safety work is effective, or if some areas are excessively emphasised.

The subproject *Initiating events - Estimation of pipe rupture frequencies* is an effort to evaluate the LOCA frequencies used in present PSA:s. Knowledge gained since the WASH-1400 analysis will be considered, including both frequency of occurrence aspects and other aspects influencing the initiator severity, e.g. leak before break. The work will include the development of a probabilistic model for pipe rupturing initiated by IGSCC (Intergranular stress corrosion cracking), and the application of this model to the detailed piping model developed within the Oskarshamn 1 PSA.

The subproject *Integrated sequence analysis with focus on human reliability* implies an analysis with participation from different disciplines, such as PSA, thermohydraulics and human factors. It aims at developing more integrated and dynamic methods for the analysis of event sequences. The project has been initiated with an overview of the current status of methods used or being developed within the Nordic countries.

The subproject *Maintenance strategies and ageing* represents a broad effort to improve maintenance practices. The project includes the evaluation of methods for surveying and interpreting maintenance indicators, or of maintenance introduced common cause failures. A survey of maintenance strategies and development needs will be performed, addressing ageing problems, condition monitoring, maintenance indicators and decision criteria for maintenance and replacement of components.

Reliability of High Energy Pipework[14,15]

In the first generation of Swedish PSA:s, LOCA categories and LOCA frequencies were taken directly from the WASH-1400 report. In principle, this is still the case, despite modifications aimed at increasing the realism of the LOCA models. Thus, some PSA:s have divided the basic LOCA categories (large, medium-sized and small) into sub-classes in order to better represent the consequences from pipe breaks above and below the core level. For the latest version of the Oskarshamn 1 PSA, an extremely detailed subdivision has been made in order also to take adequate account of the dynamic effects of pipe breaks in various locations. Thus, while the qualitative modelling of LOCA:s has become increasingly plant specific, the quantitative basis is still generic (WASH-1400).

In order to try to resolve this problem, a research project addressing reliability of high energy pipework was initiated by SKI in late 1994. The primary objective of the project is to develop a data base on the world-wide operating experience with piping and piping components, including both nuclear and non-nuclear experience. The data base will include failure information together with the known or assumed root cause of failures. The detailed analysis of the failure information is anticipated to result in a new LOCA classification scheme. The ultimate objective is to prepare an updated basis for generation of plant specific piping leak and rupture failure rates for input to the I Book.

The project includes four work phases:

1. World-wide piping failure data base
2. Piping failure rate estimation
3. Piping reliability analysis
4. Application of piping reliability analysis procedure

The first work phase will be concluded during 1995. A review of the available operating experience indicates that leaks or ruptures are more prevalent in tees and elbows than in straights. Only to a degree is piping reliability determined by inherent ageing factors. Piping reliability is controlled through sound design and construction practices and through effective in-service inspections. Human factors tend to significantly affect ultimate piping reliability.

Ageing Analysis

As previously stated, ageing is one of the issues covered by the ongoing Nordic programme "Strategy for Reactor Safety" (NKA/RAK-1), which will be concluded by 1998. Previous work within the field includes both qualitative and quantitative aspects of ageing.

A qualitative analysis has been performed[16], including discussions of vulnerability of components to ageing, causes of ageing problems, maintenance experiences, and the possibility to detect ageing problems in failure reporting systems. The analysis was largely based on interviews of maintenance personnel at one of the oldest Swedish plants, Ringhals 1.

Quantitative analysis of ageing includes a project aimed at developing a trend models for ageing analysis[17]. A number of problems with the use of recorded failures were encountered, including the assessment of repair quality (the choice between "as good as new" and "as bad as old"), the difficulty to keep track of exchanges of components or of major parts of components, and the treatment of non-critical failures or of evolving failures eliminated by preventive maintenance.

ICDE/ International Common Cause Failure Data Exchange[18]

Since the early eighties, a number of Swedish and Nordic projects have been initiated with the aim of gaining a better understanding of common cause failures (CCF). The projects have dealt both with the exploration of root causes of CCF, and with the development of models, especially for systems with high or ultra high levels of redundancy (e.g. reactor shutdown systems or pressure relief systems). The analyses performed, have resulted in a reasonably consistent basis for the qualitative and quantitative treatment of CCF events in Swedish PSA:s. Examples of analyses are:

- CCF method development in Nordic projects (NKA/SÄK-1, NKA RAS-470),
- CCF Benchmark studies in the SUPER-ASAR project,
- CCF analysis for relief valves,
- CCF analysis for ABB Atom BWR shutdown system, and
- CCF analysis for diesel generators.

A common experience from the analyses performed to date, is that the interpretation of data is usually complicated, and that CCF data analysis is time-consuming and always dependent on a very limited basis of experience data.

For this reason, in 1994 SKI took the initiative to start the International CCF Data Exchange project (ICDE). Currently, Sweden, Germany, Switzerland, the Netherlands and the USA are participating. In all of these countries, analyses of CCF events have previously been performed on a national basis. The aim of the ICDE project is to make

common use of the national experiences gained and to co-ordinate future reporting and evaluation of CCF events.

The objectives of the project are to provide a framework for co-operation on:
- collection and analysis of CCF events,
- generation of an efficient experience feedback on CCF phenomena and on defences against CCF, and
- quantification of CCF data.

The ongoing initiating work consists of agreeing upon common formats for data analysis and for failure data processing and presentation. Thus a decision will be made on the key components to include in the exchange, and on the associated component boundaries. Thereafter, an initial retroactive review will be made of national experiences and a common database compiled. In the future, yearly updates of the common database will be made based on compiled reports from each participating country.

M Book - Experience Feedback of Modifications and Backfittings[19]
In 1994 a project was initiated, with the aim of improving international feedback of experiences gained from modifications and backfits. Within the project, modifications and backfits implemented in Swedish, German and U.S. nuclear power plants are listed and evaluated.

Summary reports of implemented backfits have been compiled on a national basis. In Sweden this has been made based on reviews of PSA:s and of plant operating history as well as in dedicated summary reports for all Swedish plants[20][21][22][23]. The national summary reports have been used as the basis for a common report listing the backfits and making comparisons for various areas of BWR/PWR plant design, such as emergency core cooling, electrical power supply and residual heat removal. As far as possible the effectiveness of the safety improvements, in terms either of the reduction of the core damage frequency or the increase in safety system availability, is also discussed.

The Seismic Safety Project[24]
None of the Swedish PSA:s performed to date includes an analysis of seismic initiators. The reason for this lies mainly in the low seismic activity in Scandinavia. The two newest reactors, Forsmark 3 and Oskarshamn 3 have been analysed and designed to resist specified earthquakes. For older reactors no such analyses and designs were made. Generally, their design was considered to be robust enough to withstand earthquakes of a magnitude that could reasonably be taken into account.

However, the increasingly detailed and integrated view of operational risks that has evolved during the last two decades, has increased the demands on providing a means to assess seismic risks for Swedish nuclear power plants in a more realistic manner, and to present risks in a way that can be applied to the operation and modification of the plants.

Therefore, since 1986, the SKI and the utilities have co-sponsored a number of projects within the Seismic Safety programme. The aim of the programme has been to develop methods for calculating the ground response to be used in the safety analysis of nuclear power plants in Sweden. The programme also included a survey of geological and seismological conditions in the regions around the power plants. Results have been presented separately for the Barsebäck and Ringhals sites and generally for sites situated on bedrock (Forsmark and Oskarshamn).

In addition, assessments are made of seismic responses and capacities of safety related structures and components in Swedish nuclear power plants[2526]. The assessments are based on the seismic load input resulting from the Seismic Safety project.

Treatment of Modelling Issues

Through the years, a common industry standard for the performance of PSA has evolved. The methods and techniques applied usually correspond to the current state of the art of PSA internationally. They will, therefore, be described only very briefly. Instead, particular features of PSA in Sweden, where directed activities have resulted in an improved capability of the PSA models to correctly describe the risk profile of nuclear power plants, or where they have resulted in improving the basis for interpretation of PSA results will be described in more detail.

Background - Characteristics in the Design of Swedish Nuclear Power Plants

As a background to the description, a number of characteristic features of plant design are described in brief.

The design of safety systems in Swedish nuclear power plants has been based on a number of fundamental safety principles. Important examples are:

- The *single failure criterion* states, that safety systems should be designed in such a way, that for each analysed initiating event, any postulated single failure within the safety systems shall not jeopardise their function.
- The *30 minute rule* states, that safety functions in BWR:s shall be automated to such an extent, that no operator intervention shall be required during the first 30 minutes after each analysed initiating event. With some exceptions, the rule is also applicable to Swedish PWR:s.
- Systems for *filtered venting of the containment*. Following the conclusions from the RSU (the Reactor Safety Investigation 1979), all Swedish nuclear power plants have been gradually equipped with filtered venting systems, designed to retain at least 99.9% of the radioactivity released in connection with a core melt, excluding noble gases. These systems will significantly reduce the frequency of large radioactive releases following a core melt (by a factor of 10-50).

These criteria have lead to a highly automated safety system design with at least 2x100% or 3x50% capacity in active components. In the latest generations of BWR plants the single failure criterion has been further modified to the "n-2" criterion, meaning that safety systems have at least 4x50% capacity. This has made possible increased flexibility in performing preventive maintenance (PM), as some of the PM in standby safety systems can be performed during power operation.

Initiating Events

Initially, initiating events were defined independently within each PSA, based on IEEE and IAEA lists. As part of the SUPER-ASAR project, a common classification of initiating events was developed for all BWR and all PWR. This classification was also required in order to make possible a common approach towards the analysis of transient data, and ultimately resulted in the development of the I Book, presenting frequencies of initiating events.

In the first versions of Swedish PSA:s, initiating events were restricted to the basic set of internal events (transients and LOCA:s) included in the I Book (Initiating Event Data Book, described in paragraph 3.7.2), and listed in the table in section 3.7.2. Transient frequencies were calculated based on a review of plant operating history, while LOCA frequencies were basically derived from WASH-1400. An advantage with basing initiating event frequencies in all PSA:s on the I Book, has been the uniformity in classification and interpretation of events, which has improved the comparability of the analyses.

Through the years, the amount of initiating events modelled has been increased in order to represent plant response in a more realistic way. Presently, the initiating events in most PSA:s have been or are planned to be expanded in order to take better account of the effects that the initiators have on the safety system. This applies mainly to LOCA:s, common cause initiators (CCI) and external events.

Event Tree Analysis

The first generation of PSA:s included plant specific event tree analysis, based on either the Final Safety Analysis Report (FSAR) system success criteria (which are sometimes conservative) or on realistic thermal-hydraulic calculations. As a result, boundary conditions of the plant PSA:s differed in degree of realism.

In the SUPER-ASAR project, critical differences in the risk profile of the plants were identified. Based on these identified differences, recommendations were made on common boundary conditions concerning e.g.

- crediting of safety systems,
- system success criteria for various initiating events, and
- classification and quantification of LOCA events.

Today, the PSA aims at providing as realistic results as possible, whilst avoiding undue conservatism. In many cases, realistic success criteria, based on plant specific thermal-hydraulic calculations have replaced FSAR success criteria.

The event tree modelling is made using the same software as for fault tree modelling, Risk Spectrum. This has resulted in a uniform way of drawing and presenting event trees.

Fault Tree Analysis

Swedish PSA:s are constructed based on small event trees, involving the success or failure of main safety functions. Consequently, fault trees are large, involving all secondary safety functions and all auxiliary functions as well as most of the human interaction. All system dependencies are modelled in detail in the fault trees.

The layout and format of fault trees are based on common principles agreed upon during the ASAR 80 programme. Generic fault trees have been developed for principal components in safety systems (centrifugal pumps, motor-operated valves etc.). To the

extent possible, component failure data are assigned based on the T Book (component failure data in Swedish nuclear power plants). As within other areas of PSA, this has resulted in a reasonably consistent approach towards fault tree modelling and quantification. The development of the commonly used analysis tool, Risk Spectrum, has supported this standardisation.

A specific feature of fault tree models as modelled in the Risk Spectrum code, is the possibility to assign a large number of attributes to the component basic events. This feature has been used in the modelling and quantification of external events, by assigning to each component attributes associated with the component location, cable routing etc. Figure 2 shows an example of a fault tree in Risk Spectrum format.

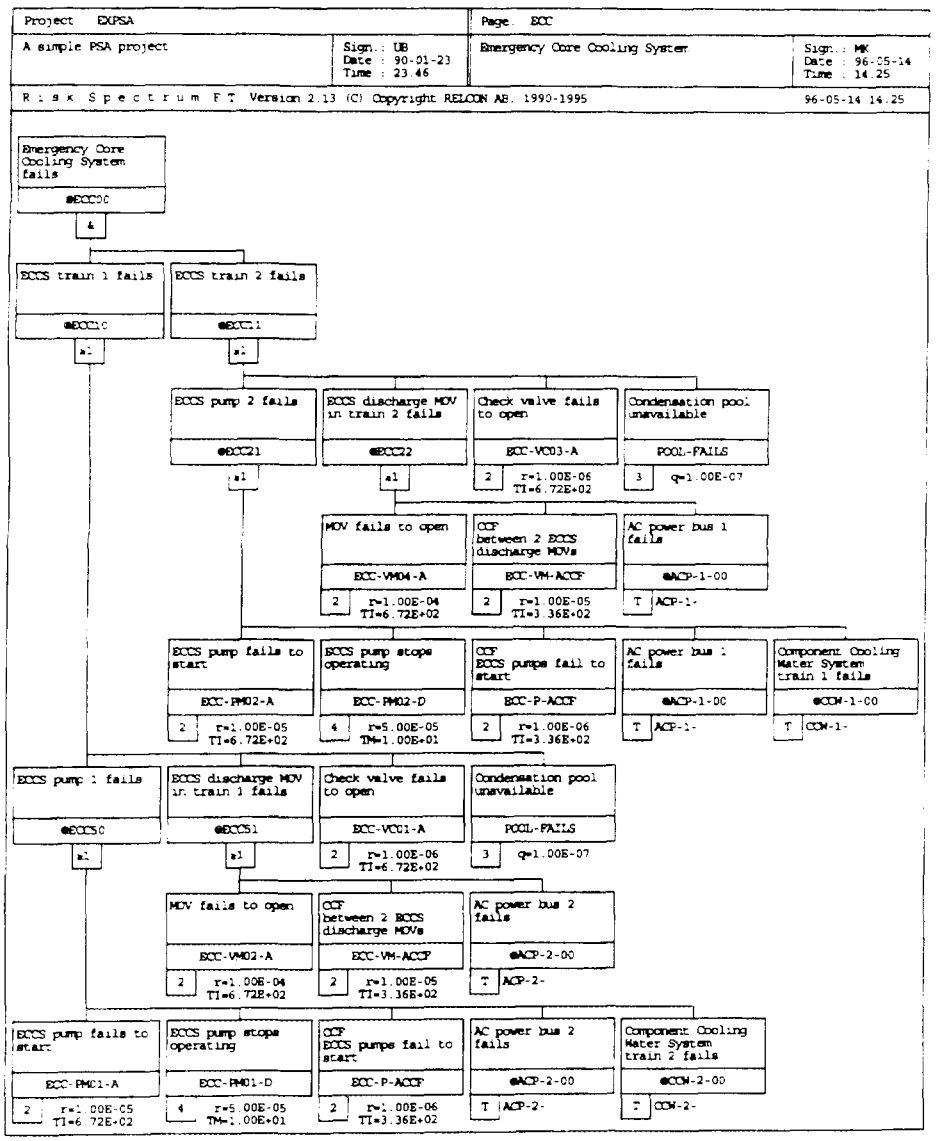


Figure 2 *Sample fault tree*

Dependent Failures

The safety systems in Swedish nuclear power plants are characterised by substantial redundancy and diversification in safety critical functions. The resulting risk profile is usually strongly dominated by dependent failures, i.e. failures that will affect a number of components or functions simultaneously, resulting in the loss of more than one system sub. For this reason, all PSA:s performed have included a very thorough identification and modelling of dependencies. This applies both to functional dependencies (dependence on common components, functions or support systems) and to CCFs. CCFs are simultaneous failures of redundant components of the same type due to a common cause.

Functional dependencies are usually modelled explicitly in the fault tree models, while the modelling of CCFs is made with parametric models. In earlier PSA:s, the Beta factor method was used. For newer plants, modelling of CCF, was based on the Multiple Greek Letter Method (MGL). Later, in-depth evaluations and performance of Benchmark exercises, including work performed within the SUPER-ASAR and the NKA/RAS-470 projects, has resulted in the recommendation of the Alpha-factor Method. The method is presently being introduced in all PSA:s.

Systems with ultra high levels of redundancy (defined as including more than four parallel components or trains), such as the shutdown system and the depressurisation system, pose additional problems and have turned out to be unsuited for modelling with parametric models like the Alpha factor method. For these systems, detailed analyses of failure information have been performed (a number of actual or potential CCF events were recorded), and alternative modelling methods, the binomial probability model or the common load model, suggested[27]. The detailed data analyses involved in these projects also resulted in a number of qualitative observations regarding e.g. the characteristics of CCFs with respect to failure mechanisms, failure location and recommended preventive maintenance practices.

In the analysis of external events, special emphasis must be placed on identifying mechanisms and interactions that could create new and previously unknown dependencies between redundant components or system subs. The following are some important areas where dependencies between systems and components may be strongly affected by external events:

- supply of power for control, activation and component power supply,
- component activation or blocking logic,
- auxiliary systems,
- operating environment, and
- shared manual interactions.

Therefore, for external events, increased stress must be placed on the correct modelling of component functional dependencies. The required level of detail in this modelling is usually considerably higher than for internal events. For this reason, considerable additional analysis effort must be put into adapting system models from internal events analysis to the requirements in an external events analysis.

Human Reliability Analysis

With few exceptions, human reliability analysis was rather superficially treated in the first generation of Swedish PSA:s. Typically, human reliability was not considered separately and not addressed in an integrated manner, but applied independently within the various subtasks of the PSA, e.g. systems analysis, event tree analysis, analysis of initiating events and common cause failure analysis. As a result, most early PSA:s suffered from a lack of consistency in the human reliability analysis (HRA) methodology applied and from a considerable incompleteness in the HRA modelling. In spite of this, the early PSA:s were successful in identifying a number of critical human interactions.

Within the SUPER-ASAR project, a review of human reliability analysis in early Swedish PSA:s was performed[28], leading to the following general conclusions:

- Human interactions have a relatively strong impact on PSA results.
- Analyses of human interactions in Swedish PSA:s are usually rather superficial. This is partly justified by the "30 minute rule", reducing the need of operator actions in BWR:s within 30 minutes of an initiating event. Operator actions within 30 minutes are either conservatively credited or not credited at all.
- The principal human interactions involved in the risk dominant sequences include for BWR:s are:
    - manual depressurisation of the reactor vessel after transients with loss of main feedwater and auxiliary feedwater, an
    - back-flushing of strainers in the emergency core cooling system and containment cooling spray system after a large or medium sized LOCA.
- The principal human interactions involved in the risk dominant sequences include for PWR:s are:
    - failure to depressurise and failure to switch to high-head recirculation after a small LOCA (some other operator actions are almost as important).

In recent PSA updates, as well as in ongoing and planned PSA updates, the treatment of human reliability has received considerable attention. A recent review[29], concludes the following (using the definitions from IAEA:s guidelines for conducting HRA):

- Type A human errors (introduced during test or maintenance) are generally modelled explicitly in the fault trees, using unavailability figures derived from experience data.
- Type B human errors (human interactions as initiating events) are excluded from all Swedish PSA:s. To some extent, they may be implicitly considered in the initiating event frequencies.
- Type C human errors (operator actions during the course of an accident sequence) include errors of omission and recovery actions. They have, to date,

not included errors of commission, i.e. human interactions that aggravate the situation.

For type C human errors, the analysis methodology is generally similar to that suggested in the SHARP method[30], i.e. an approach that divides each operator action into a number of distinct phases: observation - diagnosis - decision - action - recovery. For each of these phases, probabilities are assigned based on the tables of Swain's handbook[31], plant specific data, performance shaping factors, or time-reliability correlation.

Data

Already at an early stage in Swedish PSA development, the evaluation of operating experience with the aim of creating common data bases was stressed. The aim has been both to make possible efficient feedback of operating experience, and to increase the realism in PSA models and results.

Unlike some of the other PSA development work, data analysis must be performed as a continuous effort, resulting in periodical updates of previously established data bases.

Thus, a number of common projects have been initiated and maintained. As early as 1975, a data collection system, was developed jointly by the Swedish utilities, resulting in the Scandinavian Thermal Power Reliability Data System (ATV). All Swedish plants (and the Finnish TVO plants) report all corrective maintenance actions performed on all active components in safety related systems.

As a result, the ATV data base has developed into one of the largest and best co-ordinated component failure data bases within the nuclear field world-wide. It has provided a basis both for general-purpose data books and for advanced in-depth analyses of specific issues.

Data analysis has been performed within five main areas, as illustrated in table 3. A short description is given for each of the data books; the C Book and M Book have been described previously (section 2.3.2) among current research projects.

Table 3 *Overview of Swedish data analysis programmes*

	T Book	I Book	C Book	M Book	Incident catalogue
Appli-cation	Reliability data of active components in stand-by safety systems	Frequencies of initiating events	Common cause failure data	Experience feedback of backfitting	Trend analysis of plant LER:s (Licensee Event Reports)
First issued	Version 1, 1982	Version 1, 1990	1991	1989	1995
Latest version	Version 4, 1994	Version 2, 1995	1995, ICDE, International Common Cause Data exchange	1995, Backfitting project	1995

T Book - Component Reliability Data[3233]

A data handbook (the "T Book"), covering failure rates in the ATV system as well as licensee event reports was compiled and presented for the first time in 1982. Both generic data and plant-specific data are presented. The handbook is updated at regular intervals; the latest edition was issued in 1994.

Initially, average failure rates of the components were presented. In later editions, data on time dependent availability have been provided for components in stand-by safety systems. This makes possible the performance of time dependent analyses based on plant specific data. The parameters presented are q_0 (time-independent failure probability), λ_s (stand-by failure rate), and λ_d (runtime failure rate). The following are examples of component groups included in the T book:

- Centrifugal pumps
- Reciprocating pumps
- Pneumatic isolation valves
- Check valves
- Motor operated control valves
- Safety valves
- Diesel generators
- Gas turbines

Figure 3 gives an example of a table from the T book.

I Book -Frequencies of Initiating Events[34]

In the SUPER-ASAR project, agreement was reached on the definition and classification of initiating events (transients and LOCA:s). This formed the basis for a data handbook for initiating events (the "I-book"). Plant specific transient frequencies are calculated based on analysis of licensee event reports and plant shutdown reports. The resulting frequencies are analysed statistically to indicate trends and uncertainty bonds. The handbook is updated at regular intervals; the latest edition was issued in 1994.

Table 4 lists the initiating events covered by the I book. The basis for the categorisation of BWR initiating events is shown in a categorisation tree, figure 4.

LOCA data are mainly based on WASH-1400 data, but also includes some plant specific considerations, e.g. length of piping and division above/below core level.

Data for transients are plant specific and are based entirely on operating data. The initiator frequencies are presented as mean values with confidence bounds. Additional information is provided, such as a trend analysis of occurred events and predictions.

Figure 5 gives an example of a table from the I book.

Centrifugalpump, horisontell

Flöde: 40-60 kg/s
Tryckuppsättning: 0.5 - 0.7 MPa
Driftläge: Driftsatt
Antal komponenter: 30
Antal fel: 26
Drift-/standbytid: 1.497E6

Felmod: **Obefogat stopp**

Felintensitet:	(λ_d, 10^{-6}/h)				Effektiv medelreptid
Anläggning	5%	50%	95%	medelv.	(h)
Barsebäck 1	1.0	14.2	42.2	16.9	12
Barsebäck 2	0.4	7.8	29.0	10.4	-
Forsmark 1	5.3	19.7	50.2	22.7	25
Forsmark 2	0.8	15.3	54.9	19.9.	9
Forsmark 3	0.5	9.4	33.8	12.3	-
Oskarshamn 1	0.2	6.3	25.6	8.8	-
Oskarshamn 2	6.8	20.0	45.6	22.3	6
Oskarshamn 3	0.5	9.3	33.3	12.1	-
Ringhals 1	5.0	23.4	62.5	27.2	10
Ringhals 2	4.7	18.0	45.1	20.6	3
Ringhals 3	2.1	22.8	85.7	30.8	6
Ringhals 4	0.6	10.2	36.6	13.3	-
TVO 1	1.0	12.3	38.3	15.0	1
TVO 2	1.0	12.3	38.8	15.1	8
Generisk	1.0	13.9	56.5	18.7	9

Figure 3 *Example of table from the T Book (centrifugal pump)*

Table 4 *Initiating events covered by the I Book*

	BWR		PWR	
Transients	Ts	Unplanned shutdown	T1	Reactor coolant system pressure barrier affected
	Tp	Planned shutdown		
	Tf	Loss of main feedwater	T2	Reactor coolant system pressure barrier not affected and system not required
	Tt	Loss of turbine condenser		
	Ttf	Loss of main feedwater and turbine condenser	T3A	Total loss of main feedwater
	Te	Loss of external grid	T3B	Temporary loss of main feedwater
			T3C	Loss of salt water system
			TSI	Internal steam line break
			TSY	External steam line break
			T4	Loss of external power supply
			T5	Steam generator tube break
			T6	Transient after reactor shutdown
LOCA:s	At	Large LOCA above core level	A	Large LOCA
	Ab	Large LOCA below core level	S1	Medium LOCA
	S1t	Medium LOCA above core level	S2	Small LOCA
	S1b	Medium LOCA below core level	V	Interfacing LOCA
	S2	Small LOCA		

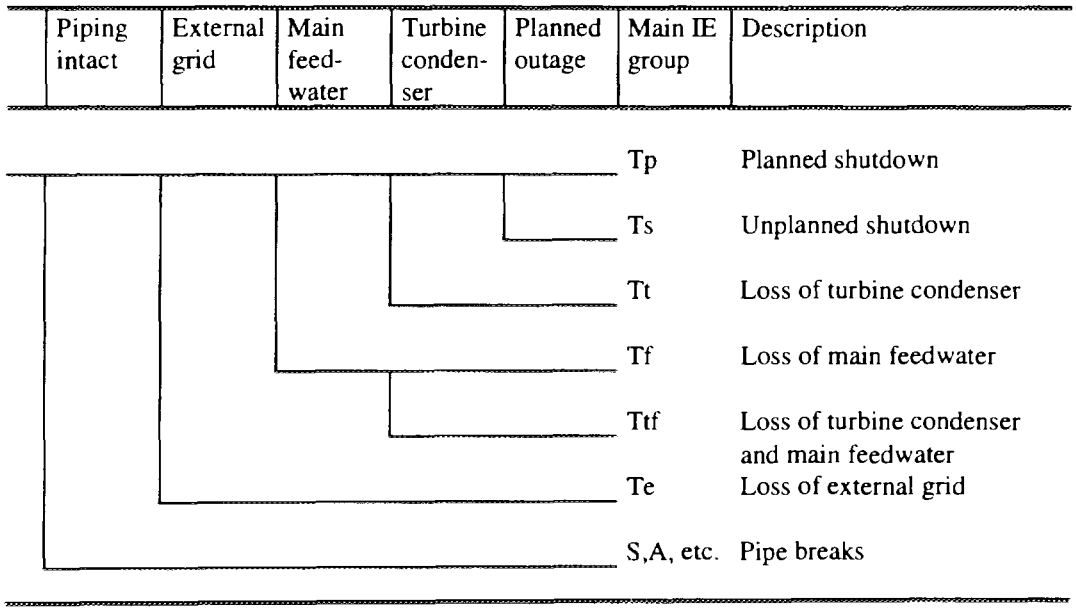
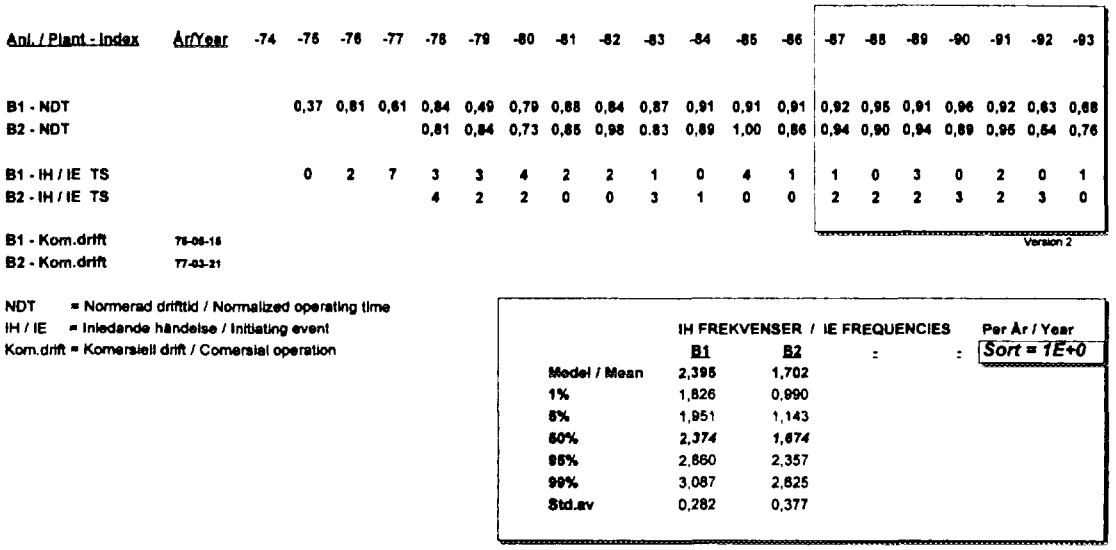


Figure 4 *Categorisation tree for BWR initiating events*



Ani. / Plant - Index	År/Year	-74	-75	-76	-77	-78	-79	-80	-81	-82	-83	-84	-85	-86	-87	-88	-89	-90	-91	-92	-93
B1 - NDT			0,37	0,81	0,61	0,84	0,49	0,79	0,88	0,84	0,87	0,91	0,91	0,91	0,92	0,95	0,91	0,96	0,92	0,83	0,88
B2 - NDT						0,81	0,84	0,73	0,85	0,98	0,83	0,89	1,00	0,86	0,94	0,90	0,94	0,89	0,95	0,84	0,76
B1 - IH / IE TS			0	2	7	3	3	4	2	2	1	0	4	1	1	0	3	0	2	0	1
B2 - IH / IE TS						4	2	2	0	0	3	1	0	0	2	2	2	3	2	3	0
B1 - Kom.drift	75-06-15																				
B2 - Kom.drift	77-03-21																				

Version 2

NDT = Normerad drifttid / Normalized operating time
IH / IE = Inledande händelse / Initiating event
Kom.drift = Komersiell drift / Comersial operation

IH FREKVENSER / IE FREQUENCIES	B1	B2	:	:	Per År / Year Sort = 1E+0
Medel / Mean	2,395	1,702			
1%	1,826	0,990			
5%	1,951	1,143			
50%	2,374	1,674			
95%	2,860	2,357			
99%	3,087	2,625			
Std.av	0,282	0,377			

Figure 5 *Example of table from the I Book (Unplanned shutdowns in Barsebäck 1 and 2)*

The STAGBAS Incident Catalogue[35]

The STAGBAS2 database is a database information system at the SKI for experience feedback based on reported safety related occurrences and reactor trip reports for Swedish nuclear power plants. All event reports from the start of operation of the plants until today are recorded. The database is continually updated, and currently contains about 6000 event reports.

STAGBAS2 makes it possible to identify patterns and trends in the information. Standardised output formats have been developed, and are used in periodically presented Incident Catalogues for all nuclear power plants. Thus, yearly rates and short term and long term time trends are presented for events within a number of areas, e.g. reactivity control or residual heat removal systems. Figure 6 shows an example of a trend curve.

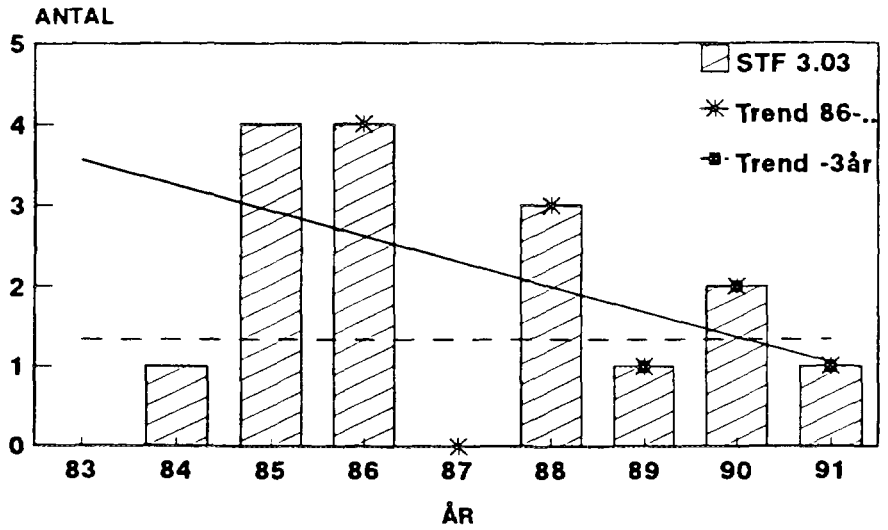


Figure 6 *Example of table from the STAGBAS2 incident catalogue (events concerning reactivity control in the Forsmark 1 plant)*

External Events

Deterministic analyses of a large number of potentially safety significant external events were included in the FSAR:s of the nuclear power plants; this applies to e.g. aircraft crash, extreme weather conditions, external flooding etc.
However, external events were not initially included in PSA:s, and are still mainly covered on a scoping level. Limited analyses of internal fires were performed for several plants in the mid-eighties, and a common research programme on seismic safety was initiated in 1986. The ongoing periodic safety review (ASAR 90) includes a requirement on probabilistic analysis of relevant external events. Thus, detailed analyses are being performed or planned for all plants, with emphasis on the analysis of internal fires, internal flooding and internal steam releases.

From the scoping analyses of internal fires performed previously, it has become evident that many of the simplifying assumptions commonly applied are not acceptable, due to the rather arbitrary introduction of a mixture of conservative and non-conservative effects on the results of the risk assessment. Some examples of simplifications with a major (but unknown) impact on the calculated risk level are:

- components in an area affected by an external event are often generally assumed to be inoperable (conservative),
- mitigating systems (e.g. fire detection and fire fighting) are often not credited (conservative),
- component dependencies that are especially vulnerable to external events are often not modelled in sufficient detail, e.g. power supply and exchange of signals with process (non-conservative), and
- failure modes of safety components and auxiliary systems are usually not adapted to the external event (non-conservative).

As a result, the ongoing activities also aim at considerably increasing the level of realism in the modelling of external events. This will require a major increase of the level of detail of the fault tree models for electrical power supply and signals. Furthermore, a three year project was initiated by the SKI and the utilities in 1994 with the aim of evaluating the state of the art within the field of external events and to propose an analysis approach; the project is described in paragraph 2.3.2.

Level 2 PSA

Limited probabilistic analyses of containment integrity in connection with various initiating events were performed for all Swedish nuclear power plants around 1985. These analyses were not integrated into the existing level 1 PSA:s, but used as a basis for the design and verification of the filtered venting systems developed at the time.

However, a full scope level 2 PSA is required to be submitted as part of the ongoing second round of periodic safety reviews (ASAR 90). Lately, level 2 analyses have been developed for the Ringhals 2 and Barsebäck 1 and 2 plants.

The analysis methodology used in the level 2 analyses performed to date is largely based on NUREG-1150[36], and is briefly described in the PSA overview in Attachment B.

Accident phenomena have been analysed in the APRI project, described in chapter 2.3.2 and in Attachment C. The project was initiated in 1992, and is performed in co-operation between SKI, the Swedish utilities and TVO (Finland).

Development of Computer Tools

PSA Modelling and Quantification - Risk Spectrum[37]

During the ASAR 80 programme, a number of fault tree analysis codes were being developed in parallel, and utilised in different PSA:s. This promoted an intensive development of the capacity and capability of the codes. During the late eighties, it was agreed between the utilities and SKI, to use one of the original codes as a basis for a standard code package for development, analysis and storage of event trees and fault trees, Risk Spectrum.

Risk Spectrum is a PC programme, originally developed for nuclear power plant PSA modelling and quantification, but also used in other applications. Since the end of the eighties, the programme is the analysis tools for all Swedish PSA work, both at the utilities and at the SKI, who have also sponsored its development. Thus it has been tailored to meet the modelling and capacity requirements of a full-scale PSA. It is continually being developed to meet new requirements arising from the development of existing PSA:s.

The basic principle of the program is to handle the entire information of a PSA model within one single project data base, i.e.:

- event trees,
- fault trees,
- basic events,
- boundary conditions (house events),
- reliability parameters,
- attributes, and
- analysis results (together with corresponding quantification specification).

Fault trees and event trees are built up in a graphical environment, using a set of simple commands. The same graphical environment is used in later editing or review of the project.

Risk Spectrum supports the following analysis options:

- Minimal cut set determination for fault tree top events
- Minimal cut set determination for event tree sequences
- Uncertainty analysis
- Importance analysis (Fussel-Vesely, Risk Achievement Worth and Risk Reduction Worth)
- Sensitivity analysis
- Time dependent reliability

Analysis of PSA Results - SKIRES[38]
Lately, the SKI has sponsored the development of SKIRES, a post-processor for quantification of results from Risk Spectrum. SKIRES makes possible advanced post-processing of the result files from a project, and has been used by SKI in several PSA reviews. Results can be restructured, modified, and presented graphically.

The following are the main features of SKIRES:

- Standardised numerical and graphical presentation of results from different PSA:s
- Advanced handling of lists of minimal cutsets on different levels
  - system
  - component
  - sequences
  - consequence
  - safety function
- Graphical and numerical presentation for new grouping of results of
  - Importance measures
  - Sensitivity
- Grouping of results for a number of different consequences into common multi-consequence results

Authority Requirements

In all safety related work, a clear distinction is made between the roles of the nuclear utilities and the nuclear safety authorities. The utilities have full and undivided responsibility for the safe operation of the plants. The safety authorities are responsible for supervising and controlling the activities of the utilities and for promoting safety development.

This way of handling safety matters is commonly named "the Swedish Model" and implies that the utilities are expected independently to identify, formulate and propose the necessary measures to achieve and maintain an acceptable level of safety. These measures are then implemented after having been reviewed and accepted by the SKI. There is only a limited amount of detailed regulation. The Swedish Model is believed to have had a positive impact, resulting in an open atmosphere and promoting dialogue and co-operation both between utilities and between utilities and authorities.

In the area of PSA, the SKI guidelines associated with the ASAR programmes have specified the approximate contents and scope of the analyses to be performed along with a preliminary timetable. The detailed contents and timetable have later been decided in discussions with the utilities.

Only Oskarshamn 3 had the performance of a PSA as a licensing requirement. For all other plants, requirements on the performance of a PSA came as part of the ASAR 80 programme, which included a general requirement on the utilities to perform basic level 1 PSA:s for all nuclear power plants. The ongoing ASAR 90 programme calls for an extension of existing PSA:s to level 2, and also calls for an extended scope by requiring the inclusion of relevant external events as well as of the shutdown period. In parallel, the requirements on updating of the PSA:s have been increased; PSA:s are now expected to be updated on a yearly basis.

There are no strict numeric requirements on PSA results, e.g. in the form of numerical safety targets. Instead, a relative approach has been adopted. The policy of the SKI has been that detected safety degradations shall be corrected in such a way that the plant is brought back to the level of safety it was believed to have before the degradation occurred or was detected.

Without using this as a formal acceptance criterion, the SKI sees the requirements of INSAG 8 as the minimum level of safety to be achieved. The industry has established numerical targets that are used as a basis for identifying areas of potential safety concern. These safety targets are generally stricter than the INSAG 8 requirements by at least one order of magnitude, but correspond to European targets for modern designs. Thus, industry guidelines set the safety goal for core damage at 10^{-5}/year, and for major radioactive releases at 10^{-7}/year.

Recently, risk-based regulation has been applied in some cases. In two of these, probabilistic criteria were applied to the handling of steam generator cracks, and to probabilistically based optimisation of Technical Specification rules.

Lately, PSA has also been used actively in conjunction with the upgrading of the Oskarshamn 1 plant, which was not built according to modern licensing requirements. For such plants, a two-step approach has been adopted:

- The plant is required to fulfil its original licensing requirements.
- All deviations from modern licensing requirements are evaluated using PSA.

The approach is in accordance with the IAEA/INSAG guidelines which are currently being developed.

Documentation and Quality Assurance

Organisation of PSA Work

At the utilities, the PSA work is typically managed by a PSA department, consisting of 3-5 persons. This department may be located at the nuclear power plant or at the company head office. It handles much of the maintenance of the PSA:s, performs limited specific analyses and does the planning and specification for major revisions or extensions of the analyses. The actual PSA analysis work is usually made by specialised consultants working in close co-operation with the utilities and receiving support with e.g. systems analyses and evaluation of operating experience.

At the SKI, PSA is one of the main issues handled by the Department of Plant Safety Assessment (RA). The department consists of approximately ten persons. It has the responsibility for maintaining and developing the authority PSA competence, for following the international development within the field, for reviewing plant PSA:s, and for performing limited evaluations making use of submitted PSA models. A major task is also to compile and analyse common experience data on e.g. failures of safety components or occurrence of operational disturbances. Furthermore, the department takes an active part in Swedish PSA development in general by initiating, co-ordinating and participating in a large number of PSA-related projects.

Handling of PSA Documentation

The PSA documentation includes all the texts, computer models and data bases that are needed in order to understand and reproduce all the models and results presented in the PSA.

The initial PSA:s were published as one-time issues, providing a snap-shot picture of the plant safety status at a certain time. The documentation was not suited for subsequent developments in operational experience or in plant design.

The iterative manner in which PSA models and results are currently being used, and the fact that a steadily increasing part of the PSA models, documentation, and results will be stored as computer files has made necessary a more strict handling of the PSA documentation. This mainly includes procedures for maintaining the PSA documentation in view of plant development and development in data bases.

Therefore, most PSA:s are today handled as plant technical documentation, following the associated quality assurance procedures. In principle, this implies that the PSA should be held continuously up to date with changes in the plant, both regarding its models and its results. In practice, such a procedure would be next to impossible to introduce. Instead, an approach is chosen, where the models are kept up to date with important plant changes with as little delay as possible, while regular (e.g. yearly) updates of the written documentation and of the complete presentation of results are made.

PSA Review

During the performance or update of a PSA, the discussions between the utilities and the SKI mainly concern the scope, the timetable and problems encountered. The results and conclusions from the PSA are presented after the completion of the analysis, e.g. as part of the ASAR report.

After the updated PSA has been submitted, SKI will perform a review of the models and results. The review is basically performed as a limited version of the IAEA procedures for IPERS reviews.

The outcome of the review is summarised relative to the following evaluation criteria:

- Assessment of Credibility
  - Quality of documentation (intelligibility, completeness, use of references, etc.)
  - Degree of coverage (systems, initiating events, phenomena)
  - Supporting analyses (success criteria, phenomena, available time for actions etc.)
- Assessment of Usefulness
  - In future use at utility (relative to analysis aims)
  - In future use at the SKI
- Previous Reviews
  - Handling of comments from previous reviews (SUPER-ASAR, IPERS, ASAR 80)
- Assessment of Effects from Limitations and Boundary Conditions
  - Effects from known or identified limitations, simplifications, conservatisms, non-conservatisms, etc.
  - Identification of boundary conditions that have decisive impact on the numerical results of the analysis
- Assessment of Effects from Missing Analyses (if any)
  - Level 2 PSA
  - Shutdown period
  - External events

Development Work and Research Projects

By tradition, much of the research and development activities are performed jointly by SKI and the Swedish utilities. In some cases, Finnish utilities (and occasionally authorities) also participate.

Research activities have typically been of two types, either limited analyses aimed at solving a specific problem, or broader investigating analyses aimed at developing and trying out areas and methods for future PSA activities.

Problem oriented research activities are usually, but not always, of a smaller volume and run over a relatively short period, up to about one year. They aim at finding a solution to a specific problem. Often they are prompted by or related to important current safety issues. Examples of such activities are projects concerning the development of methods for modelling CCFs in systems with ultra high levels of redundancy (e.g. reactor shutdown systems or pressure relief systems) and a number of projects aimed at identifying functional dependencies between circuit breakers.

Research projects of the investigating type tend to be of greater volume, and run over a longer period of time, typically three to five years. They aim at paving the way for extensions of the models, scope or use of existing PSA:s. Usually these long-term projects are preceded by rather extensive reviews of the opinions of utilities and authorities on the perceived problem areas.

SKI has an important role in initiating research work and in providing a framework for major research activities. In many cases, SKI will also have a better overview of generic issues and of the international status than the utilities. SKI spends a total of approximately 6-7 million SEK per year on research projects. In most cases, research projects are funded jointly by SKI and the utilities. The participation of the utilities is important in order to improve the understanding and use of project results.

During the last 10 years, a number of research projects, performed on a national or Nordic basis, have had the common aim of improving the capability of current Nordic PSA:s to serve as tools for risk evaluation and decision support. A number of these (and other) projects have been shortly described in paragraphs 2.2.2 and 2.3.2, and a few are also described in more detail in Attachment C.

Results and Conclusions from PSA

Overview of Results from Swedish PSA:s

A short overview of results from Swedish PSA:s is presented, including a description of analysis status and a presentation of results from level 1 PSA:s.

Analysis Status

Table 5 summarises the scope and analysis status for all Swedish nuclear power plants. For BWR:s, the table is organised according to the reactor generation (1-4). As can be seen from the table, the ongoing updates, that are mostly part of the ASAR 90 periodic safety review, generally concern the following topics:

- Inclusion of detailed CCI analysis
- Extension of analyses of internal events to level 2
- Inclusion of analyses of external events
- Inclusion of an analysis of shutdown and refuelling

Table 5 *Scope and analysis status of Swedish PSA:s*

		Oskars hamn 1 BWR	**Ring-hals 1 BWR**	**Barse-bäck 1/2 BWR**	**Oskars hamn 2 BWR**	**Fors-mark 1/2 BWR**	**Fors-mark 3 BWR**	**Oskars hamn 3 BWR**	**Ring-hals 2 PWR**	**Ring-hals 3/4 PWR**
BWR Generation number		**1**		**2**		**3**	**4**			
Internal events	**Transients**	L1 95 *L2 p98*	L1 93	**L2** 95	L1 92 *L2 p97*	L1 88	**L2** 96	L1 93 *L2 p97*	**L2** 95	L1 92
	LOCA	L1 95 *L2 p98*	**L2** 93	**L2** 95	L1 92 *L2 p97*	L1 88	**L2** 96	L1 93 *L2 p97*	**L2** 95	L1 92
	CCI	L1 95 *L2 p98*	L1 83 scoping	*L1 p97*	*L2 p97*	L1 88 scoping	L1 85 scoping	*L2 p97*	L1 83 scoping	*L1 p*
External events	**Internal fires**	L1 95 *L2 p98*	L1 93	L1 91	*L2 p97*	*L1 p96*	*L1 p96 scoping*	*L2 p97*	L1 94	L1 95
	Internal flooding	L1 95 *L2 p98*	L1 93	*L1 p97*	*L2 p97*	*L1 p*	*L1 p96 scoping*	*L2 p97*	L1 94	-
	Seismic	A common SKI/utility project is currently being performed and will be used as a basis for future seismic PSA:s								
	Other external events	A variety of external events are analysed in the FSAR (final safety analysis report)								
Revision	**Shutdown and refuelling**	*L1 p98*	L1 95	**L2** 95	*L1 p98*	*L1 p*	*L1 p96/97*	*L1 p98*	L1 93	L1 90

Notes: L1 95 = Level 1 analysis presented in 1995 *L1 p97= Level 1 analysis planned by 1997*
L2 95= Level 2 PSA presented in 1995

Results from Level 1 PSA:s

Comparison of results from different PSA:s should be made with great care. This is due to the possibility of significant differences in design and operation, analysis aim, scope,

level of detail and other boundary conditions. The analysis will also make use of different data and models, which may considerably influence both qualitative and quantitative results. This has been experienced e.g. when comparing the results from PSA:s of twin plants.

As can be seen from the above summary of analysis scope and status, there is at present a considerable variety in the coverage and level of detail of the PSA:s. Direct comparison of numerical results for different plants is not possible. Therefore, the results presented in this paragraph cannot be used as a basis for ranking the different plants with respect to safety, but should rather be seen as indicators of a safety level and as a basis for identifying risk drivers.

Table 6 gives a summary of the numerical results of Swedish level 1 PSA:s. In order not unduly to complicate this overview, it has been chosen to present a simplified picture, which in some cases has made necessary reclassification and reinterpretation of the basic results presented in the various PSA:s.

Table 6 *Summary of results from Swedish level 1 PSA:s*

	Oskars hamn 1 BWR	**Ring-hals 1 BWR**	**Barse-bäck 1/2 BWR**	**Oskars hamn 2 BWR**	**Fors-mark 1/2 BWR**	**Fors-mark 3 BWR**	**Oskars hamn 3 BWR**	**Ring-hals 2 PWR**	**Ring-hals 3/4 PWR**
BWR Generation number	**1**		**2**		**3**	**4**			
Core melt frequency per year	$2.1*10^{-5}$	$3*10^{-6}$	$4*10^{-6}$	$4*10^{-6}$	$8.5*10^{-6}$	$7*10^{-6}$	$3*10^{-6}$	$2.4*10^{-5}$	$2.1*10^{-5}$
CM due to ATWS	23 %	22 %	25 %	23 %	10 %	2 %	< 1 %	-	-
CM due to loss of make-up water supply	72 %	78 %	50 %	73 %	> 80 %	89 %	37 %	-	-
CM due to loss of residual heat removal	5 %	< 1 %	25 %	4 %	< 10 %	9 %	63 %	-	-
CM after LOCA (including reactor vessel rupture)	17 %	80 %	25 %	66 %	8 %	5 %	7 %	63 %	63 %
CM after external LOCA	5 %	6 %	25 %	< 1 %	13 %	< 1 %	< 1 %	-	-
CM after transient (including spurious isolation signals)	18 %	14 %	50 %	34 %	79 %	95 %	92 %	37 %	37 %
CM after CCI	Quantitative identification	60 %	Not quantified		Quantitative identification		Not quanti-fied	Quantitative identification	Not quanti-fied

Figures 7, 8 and 9 show the same results graphically. Some general comments can be given to the results:

- The PSA models for three of the BWR plants (B1 and 2, O2 and R1) are still largely based on the original PSA, which was presented as part of the first periodic safety review, ASAR 80, in the mid-eighties. They are characterised by limited scope and

level of detail and can, therefore, be expected to underestimate some risk contributions.

- Only the Oskarshamn 1 PSA includes a complete CCI analysis, making use of a detailed modelling of signals and electrical power supply. As can be seen from the summarising table, the resulting risk contribution is substantial. In most cases, and especially for the older plant generations, models of a similar level of detail can be expected to result in the identification of significant risk contributors.
- Due to significant differences in boundary conditions, the summary does not include risks from external events. Such events can be expected to yield significant risk contributions, especially for earlier plant generations with lower levels of redundancy in safety systems and with a less pronounced area separation.
- For BWR generation 1, the risk is dominated by CCI and LOCA initiators. Among the safety functions, ATWS and loss of make-up water dominate.
- For BWR generation 2, the risk is more evenly spread among transients and LOCA initiators. Future CCI analyses will probably identify significant risk contributions. Among the safety functions, ATWS and loss of make-up water dominate.
- For BWR generation 3 and 4, the risk from transient initiators dominate, while risks from LOCA and ATWS initiators are small.
  Among the safety functions, loss of make-up water dominates.

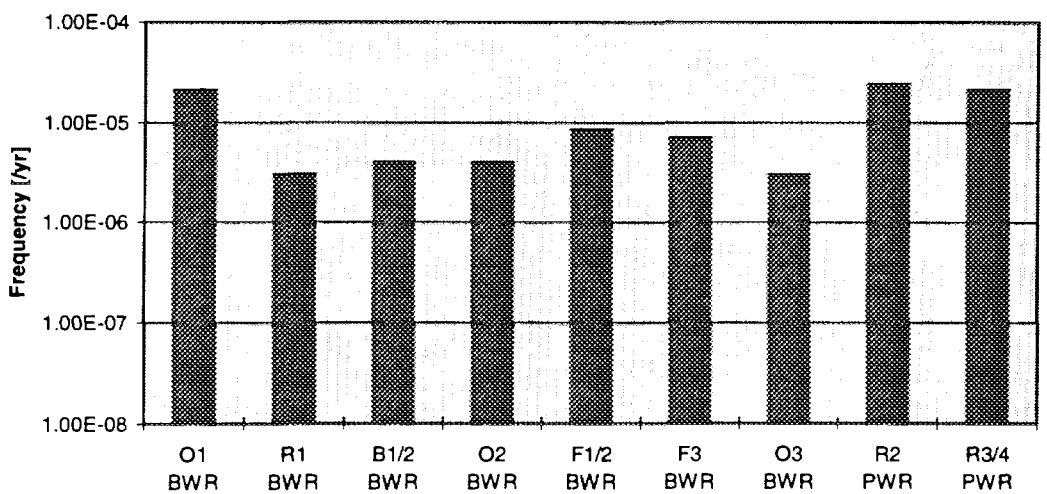


Figure 7 *Total core melt frequency, internal events*

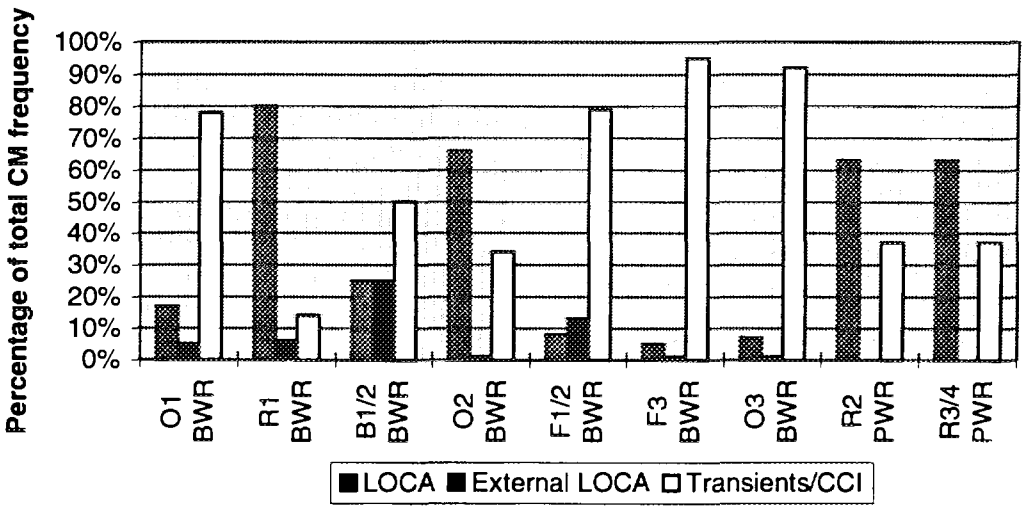


Figure 8 *Percentage of total core melt frequency by initiator categories*

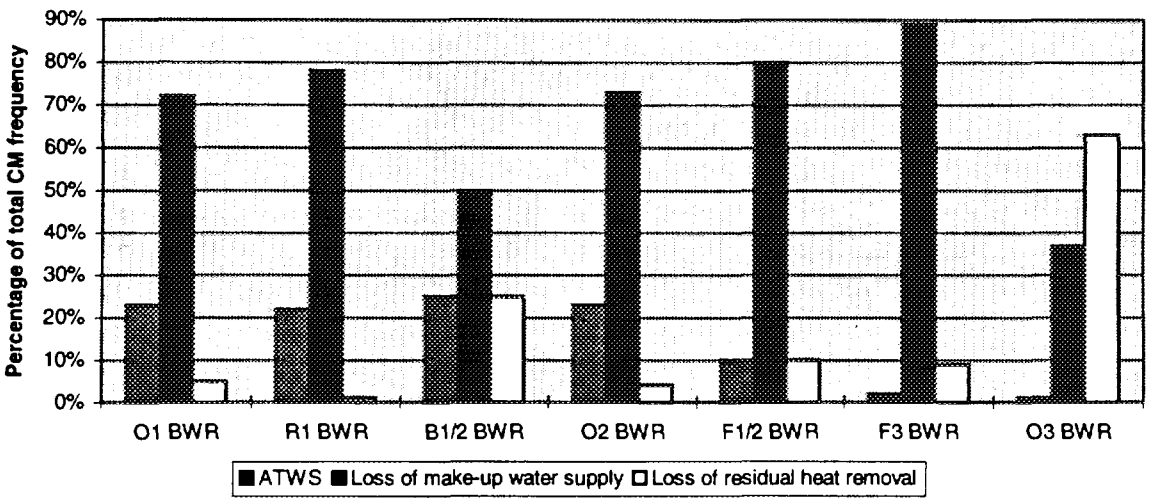


Figure 9 *Percentage of total core melt frequency by safety functions (BWR plants)*

Use of PSA Models and Results

In the seventies and early eighties, the aim in using PSA results was to obtain a risk profile of the plant, making it possible to identify risk driving events, components and procedures. Both PSA models and results were produced for one-time use. As a consequence, although the initial PSA:s were quite successful in fulfilling their expressed aims, and resulted in the introduction of many changes in designs and procedures, they turned out not to be a suitable basis for continued or extended analyses.

In later years, the development and use of PSA has become more interactive and aimed at producing general-purpose models and results suited for extension and post-processing. Basic results are still presented as part of a PSA document, but the analysis model is actively and continuously used, e.g. for the evaluation of design changes.

In recent years, both authorities and utilities have shown an increased interest in using probabilistic methods as a means to generate decision support in questions regarding the safety or reliability of nuclear power plants. This is due to the steadily increasing level of competence in Swedish PSA activities in general, to the increasing quality and usefulness of the plant-specific PSA models, and to an increasing acceptance and understanding of PSA among SKI and utility decision makers.

Typical SKI uses of plant PSA models and results include the compilation of a yearly overview of the safety status at the plants and the use of PSA results for feasibility checks in connection with applications for temporary exemptions from technical specification rules and plant modifications. Thus, the SKI has a pronounced need for being able to understand the background of PSA results and the impact from limitations and boundary conditions. The SKI also needs to be able to make quick estimates based on a PSA model.

This will result in partly different requirements on the PSA. One of the specific SKI requirements is the need for co-ordination of the PSA work at the various utilities. This requirement has played an important role in the co-ordination of the PSA work with respect to data for initiating events and components, to the choice of analysis tools and to the structure, format and failure coding of fault trees and event trees.

When using PSA models and results as a basis for decision-making, the challenge lies in succeeding in making use of the unique possibilities offered by the PSA methodology to produce a compact, structured and balanced picture of plant safety, but at the same time not to disregard or become unduly susceptible to the many sources of uncertainty that are to a large extent an integral part of probabilistic techniques.

There are a number of common pitfalls in the use of PSA results. One of the most important is the risk for an undiscriminating acceptance of the simplified and easily accessible, but at the same time limited risk picture, that is conveyed by the purely numerical results of the analyses. PSA analysis results must always be seen in context in order for a fruitful interpretation to be possible. This context consists of a careful assessment of the impact and interpretation of limitations, uncertainties and boundary conditions of the analysis.

For the above reasons, one of the prerequisites for a continued increase in the availability and usefulness of PSA methods, is the development of comprehensive and easily accessible tools for retrievement and interpretation of PSA results. Such tools should highlight both the possibilities and the limitations in PSA methods and in the use of PSA results. These considerations have been a driving force in the specification of tools for PSA modelling and analysis, such as Risk Spectrum and SKIRES, both of which are described in paragraph 3.10.

PSA Based Safety Improvements

As previously stated, the aim in performing PSA:s in Sweden has been primarily to

- identify dominant contributors to the total core melt frequency,
- generate a basis for identifying and ranking safety enhancing measures,
- decide the total frequency of core damage (or large radioactive releases), and
- generate models and results that can be used in the continued safety work at the plant.

In many cases, PSA:s have contributed to the identification of system weaknesses, critical components, insufficient redundancy, lacking operating procedures or emergency operating procedures. In these cases, an increased safety level has been achieved by redesign or procedural changes. In many cases, findings from PSA have confirmed the existence of previously known problems, and made it possible to assess them in the context of a total risk perspective as well as to rank them according to their risk importance.

Other areas where PSA models and results have been used are:

- risk management,
- optimisation of Technical Specifications,
- licensing support,
- training, and
- general development of PSA technique.

As an important spin-off, the performance of a PSA, especially the development of detailed system models, will also result in a thorough system review. This has often resulted in valuable qualitative insights. In many cases, the review has helped to identify previously unknown weaknesses or inconsistencies in system design, especially in the detailed design of electrical system or control systems.

Table 7 shows examples of a number of areas where PSA has contributed to identifying or selecting safety improvement measures. A systematic presentation and evaluation of modifications and backfittings is made in the M Book project, described in paragraph 2.3.2.

Table 7 *Examples of safety improvement measures partly or entirely based on PSA*

Area	Examples of safety improvement measures
Protection against Common Cause Failures	• Elimination of functional dependencies • Improvement of separation • Modification of maintenance or test strategy • Elimination of common cause initiators (CCI)
Other Design Changes	• Added redundancy • Improvement of functional reliability of specific components or systems • Modification of actuation logic of protection systems
Improvement of Operator Support	• Modification of emergency operating procedures • Improved feedback of critical process parameters • Additions to existing emergency operating procedures
Improvement of Maintenance and Testing	• Introduction of preventive maintenance during power operation • Optimisation of test intervals of components in stand-by safety systems

Conclusions from Swedish PSA Activities

General

There are a number of features of PSA in Sweden, that have contributed to what is commonly considered a rather successful development within the field of PSA:

- The Swedish Model puts a high degree of responsibility on the utility, but imposes very few detailed regulations on utility activities. This has promoted an open atmosphere and co-operation both between the utilities and the SKI and among the utilities.
- A homogenous reactor population and few utilities. All BWR:s come from one vendor (ABB Atom) and, therefore, have many features in common. The same applies to the three (Westinghouse) PWR:s.
- High levels of ambition from the outset has led to the development of efficient analysis tools. This has put few limitations on models and made expansion possible. Thus, new analyses have been gradually performed on the basis of existing ones.
- The ASAR programme has provided frames for taking the "next step" in developing the analyses, and has assured that the utilities keep pace with each other.
- A large number of major research projects have been jointly funded and performed by SKI and the utilities.
- An efficient way of defining and carrying through research projects has lead to the identification and elimination of many problems encountered, which has extended the applicability of PSA.

- An iterative approach has been adopted in research and development work. This means that the development has kept pace with the build-up of competence, which has helped to minimise the amount of blind alleys explored.
- A very successful system for collection and processing plant specific experience data regarding transients and component failures has lead to in increased credibility in the results produced.
- It has not been tried to use PSA as a tool for ranking reactors with respect to safety. Instead, there has been a stress on achieving safety improvements by concentrating on qualitative findings, i.e. by identifying and correcting flaws in the design, weak spots and trying to promote an even safety level.

Tools and Models

- In order to make possible the use of fault tree models for new applications, and in order to model functional dependencies in sufficient detail, the level of detail in the fault trees must be rather high. This has been accentuated in recent detailed analyses of external events and of CCI, especially for systems belonging to electrical power supply and signals.
- Efficient, flexible and user-friendly computer tools are a prerequisite for fruitful PSA activities. This applies to tools for modelling and analysis of fault trees and event trees as well as to tools for post-processing of PSA results.
- Analysis of experience data, both for component failures and transients, is an important area. It has proved to be indispensable when it comes to increasing the level of realism in the PSA models, and in allowing the introduction of a number of new areas of application of PSA (e.g. related to Technical Specifications).

Trends in PSA Activities

- Generally, PSA is expected to be used more and more actively in the future, both by the utilities and by the SKI. The ongoing or planned modernisation programmes for the plants will be important areas of application. Targets and strategies will be based on a complete picture with both deterministic and probabilistic criteria, utilising IAEA procedures[39].
- QA aspects on the performance of PSA:s as well as on the handling of PSA documentation and models has become increasingly important. This is due both to the complexity and volume of modern PSA models, and to requirements following from an increased interaction between PSA and plant operation and development.
- At the SKI, it has been discussed to establish a programme for risk based inspection, that would be used in order to guide the performance and contents of SKI inspection activities, and for prioritisation in the treatment of upcoming safety issues. To date, this is still in a planning stage. The inspection programme would be initiated by performing preparatory work on each plant, partly based on a systematic post-processing of PSA results, i.e. by preparation of lists of risk dominant components, human interactions and dependencies. This information

would be supplemented and updated on a regular (yearly) basis by evaluation and consideration of recent operating experiences and safety issues.

Attachment A - Characteristics of Swedish Nuclear Power Plants

Plants	Start of operation	Reactor Supplier	Effect [MWe]	Capacity of main safety functions: Low pressure injection	High pressure injection	Residual heat removal	Separation between subs	Auxiliary power	Accident mitigation
BWR Generation 1		ABB Atom		2 x 100 %	2 x 100 %	2 x 100 %	Total electrical separation Limited space separation	2 diesel generators	Filtered venting
Oskarshamn 1	1971		462						
Ringhals 1	1974		825						
BWR Generation 2		ABB Atom		2 x 100 %	2 x 100 %	2 x 100 %	Total electrical separation Limited space separation	2 diesel generators	Filtered venting
Barsebäck 1	1975		615						
Barsebäck 2	1977		615						
Oskarshamn 2	1974		630						
BWR Generation 3		ABB Atom		4 x 50 %	4 x 50 %	4 x 50 %	Total electrical and space separation	4 diesel generators	Filtered venting
Forsmark 1	1980		1006						
Forsmark 2	1981		1006						
BWR Generation 4		ABB Atom		4 x 50 %	4 x 50 %	4 x 50 %	Total electrical and space separation	4 diesel generators	Filtered venting
Forsmark 3	1985		1200						
Oskarshamn 3	1985		1205						
PWR Generation 1		Westinghouse		Common to residual heat removal (4x50% containment spray pumps)	3 x 50% (charging pumps)	2 x 100% (1x100% + 2x50% auxiliary feedwater)	Total electrical separation. Total space separation with some exceptions.	4 diesel generators	Filtered venting
Ringhals 2	1974		914						
PWR Generation 2		Westinghouse		Common to residual heat removal (4x50% containment spray pumps)	3 x 50% (charging pumps)	2 x 100% (1x100% + 2x50% auxiliary feedwater)	Total electrical separation. Total space separation with some exceptions.	4 diesel generators	Filtered venting
Ringhals 3	1980		960						
Ringhals 4	1982		960						

Attachment B - A Short Introduction to PSA

Safety analyses can be deterministic or probabilistic. Deterministic analyses are typically used when evaluating the basis for a given design, e.g. to determine if it can withstand certain accidents or to calculate system capacity requirements under different operating conditions.

Probabilistic analyses get their boundary conditions from deterministic analyses, e.g. as success criteria for a safety system. This is used as a basis for generating additional information, that cannot be derived from the deterministic analysis. Thus, probabilistic analyses make it possible both to assess the severity of a situation and to identify and prioritised possible improvements to a situation.

In nuclear applications, the aim of probabilistic analyses is usually to assess risks and to prioritise safety enhancing measures. For this reason, these analyses are usually referred to as PRA (probabilistic risk assessment) or PSA (probabilistic safety assessment). During recent years, PSA has become the accepted designation in Sweden.

The classification of types of PSA is based on the scope of the analyses, which in turn is decided by three parameters:

- risk measure,
- operating modes covered, and
- types of initiators analysed.

The first parameter in the classification of PSA analyses is the *risk measure* chosen. In this context it is customary to refer to PSA of levels 1, 2 or 3:

- PSA level 1
  Frequency of core damage
- PSA level 2
  Frequency of radioactive releases outside the containment
- PSA level 3
  Frequency of consequences of radioactive releases

To date, no level 3 PSA:s have been performed in Sweden. A PSA of level 1 and level 2 for the same plant will lead to partly different conclusions because the risk measures are different - usually the sequences dominating the core damage frequency are not the same that dominate the frequency of radioactive releases.

The second parameter in the classification of PSA:s are the *operating modes* covered by the analysis. Basically there are two operating modes:

- Power operation
- Shutdown, including cold shutdown, hot shutdown, close-down and start-up of operation

The third and last parameter in the classification of PSA:s are the *types of initiators* analysed. It is customary to differ between internal and external events. The difference between these categories can be described in the following, somewhat simplified, manner:

- Internal events are disturbances or accidents within the process, e.g.
  - Transients

- Pipe breaks (LOCA)
- Common cause initiators (CCI, i.e. failures that will cause a transient and simultaneously degrade a safety system)

- External events are disturbances from outside the process, e.g.
  - Internal fires (fire within the plant)
  - Internal flooding (flooding within the plant)
  - Seismic events (earthquake)
  - Other external events (Aircraft crash, extreme weather conditions, etc.)

The completeness of a PSA is decided by the number of operating modes and types of initiators included, while the choice of risk measure is normally determined by the purpose of the analysis.

A general and simplified description of the methodology used in a PSA can be based on the following basic parts, as illustrated in figure B1:

- Initiating Event,
- Event Tree,
- Consequence, and
- Fault Tree.

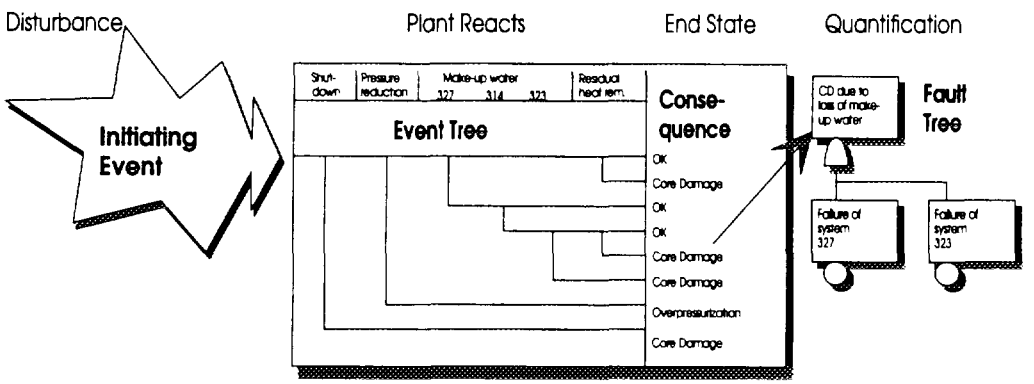


Figure B1 *Overview of PSA methodology*

The four basic parts are connected in the following way:

- Initiating Events
  Accident or disturbance (internal or external event) that results in the automatic or manual shut-down of the plant, and consequently will require safety systems to operate.

- Event Trees
  Event trees are a means to model the response of the plant to a given initiating event. In the event trees, alternative ways to fulfil the safety functions are analysed. The aim of the analysis is to identify all possible event sequences that can follow a disturbance. In a level 2 PSA, event trees are used also to model the containment response in connection with an accident. These event trees are called containment event trees (CET).

- Consequence
  The consequence expresses the end state of a given event sequence. The end state may be either a more or less stable condition, or some kind of a core damage or radioactive

release.
In a level 2 PSA, end state are called plant damage states (PDS).

- Fault Trees
  Fault trees are a means to model and quantify system functions. The design of the system and it's various failure modes are expressed in a quantifiable model by use of logical operators. In the fault tree quantification, the frequencies of the sequences that have been found to result in core damage are calculated.

A stable end state (OK in the event tree in figure B1) usually requires the correct operation of four safety functions:

- reactor shutdown,
- depressurisation of reactor vessel,
- make-up water supply, and
- residual heat removal.

The functional requirements on the safety functions (success criteria) can vary considerably between different event sequences, depending both on the nature of the initiating event and on accident progress.

After the conclusion of the fault tree quantification an analysis is made of the result obtained. This analysis often includes the performance of uncertainty and sensitivity analyses.

LEVEL 2 PSA

The analysis methodology used in Swedish level 2 analyses performed to date is largely based on NUREG-1150, and is briefly described below (from Barsebäck 1 and 2 PSA). As an illustration of the methodology used, an overview of the Barsebäck 1 and 2 PSA level 2 analysis is shown in figure B2.

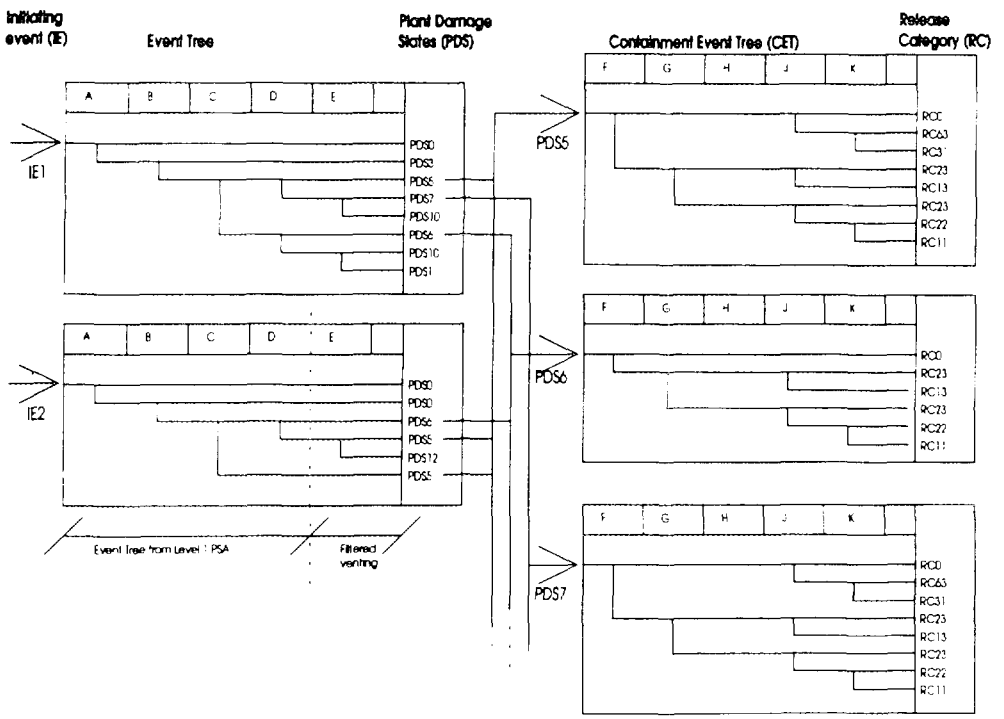


Figure B2 *Overview of methodology used in Barsebäck 1 and 2 level 2 PSA*

The event trees from the level 1 analysis have been expanded by including manual and automatic activation of the filtered venting systems. The possible plant states at the end of the sequences are described by a total of twelve plant damage states (PDS). The PDS:s are characterised by approximately ten parameters describing e.g.:

- the cooling and sprinkling of the containment,
- the pressure in the reactor vessel and the containment,
- the activation of the filtered venting system, and
- the status of the containment (intact or not).

The status of these parameters decides the continued accident development and influences the frequency and magnitude of a release.

Each PDS is analysed in a separate Containment Event Tree (CET). The CET describes the continued development of an initiated core damage by taking into consideration physical phenomena and mitigating systems. In order to characterise the radioactive releases resulting from the various accident sequences, 18 release categories (RC) have been defined, defining the starting time of a release (three categories: early, medium, late) and the amounts of iodine and cesium released (six categories).

In order to make possible the categorisation of sequences into release categories, a number of characteristic sequences have been analysed with the accident analysis programme MAAP 3.0B version 6. Based on the outcome of these analyses, all sequences were categorised.

The following physical phenomena were considered:

- hydrogen deflagration/explosion in reactor vessel and containment,
- steam explosion in reactor vessel and containment,
- direct Containment Heating (DCH),
- loss of containment integrity due to slow heating of containment walls,
- loss of containment integrity due to slow pressure increase, and
- global vessel melt-through.

These phenomena are included as events or fault trees in the CET:s. The probabilities of the various phenomena are dependent on the PDS, and have been decided using expert judgement techniques.

Attachment C - Description of Major Research Projects

The contents and results from the following completed projects are described in some detail:

- NKA/RAS-450 - Optimisation of Technical Specifications by Use of Probabilistic Methods
- NKA/RAS-470 - Dependencies, Human Interaction and Uncertainties in Probabilistic Safety Assessment
- SUPER-ASAR - A Comparative Review of Completed PSA:s
- NKS/SIK-1 - Safety Evaluation by Living PSA
- APRI - Accident Phenomena of Risk Importance

NKA/RAS-450 - Optimisation of Technical Specifications by Use of Probabilistic Methods

Technical Specifications (TS) for Nordic nuclear power plants were originally defined based on both deterministic analyses prepared for the final safety analysis report (FSAR) of the nuclear power plants and on engineering judgement.

Through time, extensive operating and design experience has been accumulated, and a number of problems had appeared, requiring modification of TS rules. Therefore, it was considered important both by the Nordic utilities and by the authorities to evaluate the problem further. The ultimate goal was to improve plant safety and enhance the efficiency and flexibility of plant operation, maintenance and testing.

By the time the project was launched, it was obvious that probabilistic methods could be utilised in analysing and comparing the risk effects of alternative requirements in the TS rules.

Thus, the NKA/RAS-450 project aimed at providing a framework for the analysis of issues related to the evaluation and optimisation of Technical Specifications. The project was a Nordic five-year effort (1985-89) dealing primarily with:

- optimisation of Limiting Conditions of Operation (LCO), including Allowed Outage Times (AOT) of components,
- optimisation of Surveillance Test Intervals (STI), including analysis of test strategies,
- planning and evaluation of preventive maintenance during power operation,
- analysis of testing, and
- analysis of failure data.

The main decision situations concerning TS are, whether it is possible either to justify and allow proposed permanent TS modifications or temporary exemptions from TS rules.

It was demonstrated that a guide for prompt decision making in specific failure and maintenance situations during plant operation, i.e. temporary exemptions from TS, can be provided by precalculated risk importance measures presented as part of the PSA results. The risk increase factor is a useful measure for the estimation of the safety significance of a component fault or of the temporary isolation of equipment due to maintenance.

Resulting from the review of methods, method development and proposals for criteria within the project, the following TS evaluation cases were deemed feasible:

- Making risk-based comparisons of alternative plant operating principles during failure situations, and identifying the associated operating modes that result in minimum risk.
- Evaluating temporary risk increases caused by unavailable equipment, e.g. due to preventive maintenance during power operation in stand-by safety systems.
- Analysis of coverage and efficiency of testing and quantification of the effects of alternative test schemes for redundant equipment.

A number of case studies were performed as part of the project, and have produced useful results within a number of areas:

- Reconsideration of plant shutdown requirements in situations with multiple failures.
- Justification of modified rules for preventive maintenance during power operation in high-redundant stand-by safety systems.

- Improvement of the efficiency of surveillance test procedures for stand-by equipment.

The project also gave some general recommendations on actions to be taken in order to increase the applicability of existing PSA:s to TS evaluations and other types of Living PSA analyses:

- Include the Living PSA aspect in the planning stage of the PSA (or of a planned PSA revision). This will make it possible to decide early on which systems to include, suitable levels of details of system fault tree models, etc.
- Use time dependent component models instead of mean unavailability.
- Model test schemes of systems including redundant components or subsystems explicitly, as this may have major influence on the total system reliability.
- Calculate and present risk importance measures on component, subsystem, system, and safety function level.
- Perform and present systematic sensitivity analyses, covering important modelling aspects, simplifications, and assumptions.
- Perform and present statistical uncertainty analyses for system and safety function unavailability as well as for plant level results.

NKA/RAS-470 - Dependencies, Human Interaction and Uncertainties in Probabilistic Safety Assessment

Three areas were investigated in a five year Nordic programme (1985-89):

- dependencies with special emphasis on common cause failures,
- human interaction, and
- uncertainty aspects.

The approach was based on comparative analyses in the form of Benchmark analyses, reference studies and retrospective reviews. Weak points in available PSA:s were identified, and recommendations were made aiming at improving the consistency of the PSA:s. The sensitivity of PSA results to basic assumptions was demonstrated and the sensitivity to data assignment and choice of methods for analysis of selected topics was investigated.

The following summarises the approaches chosen and the findings from the Benchmark exercises and reference studies that were carried out within the project:

Common cause failure data Benchmark exercise

Motor operated valves (MOV) of Swedish BWR:s were selected for this study.

- The exercise demonstrated that available failure reports may provide a sufficient basis for identification of potential CCFs.
- It was observed that estimates of CCF contributions vary considerably. This was due to differences in the treatment of data rather than to the choice of a particular estimation method.
- The use of parametric CCF models represents a suitable approach when good quality single failure data are available.
- It was confirmed that the alpha-factor method gives a more correct representation of uncertainties than the multiple Greek letter (MGL) method.
- The main weaknesses of the current state of CCF analysis concern the limited understanding of relevant failure mechanisms and possibilities of defensive measures against CCF as well as the treatment and quantification of CCF in systems with ultra high redundancy level (e.g. shutdown systems or pressure relief valves).

Reference study on human interaction

The study concerned manual depressurisation at the Forsmark 3 plant. The PSA identified this as a major risk contributor in accident sequences involving loss of feedwater.

- Human interactions have a relatively strong impact on PSA results.
- Contrary to the treatment of dependencies, analyses of human interactions in Swedish PSA:s are usually rather superficial. This is partly justified by the "30 minute rule", reducing the need of operator actions in BWR:s within 30 minutes of an initiating event.
- The principal human interactions involved in the risk dominant sequences include for BWR:s are:
  . manual depressurisation of the reactor vessel after transients with loss of main feedwater and auxiliary feedwater, and
  . back-flushing of strainers in the emergency core cooling system and containment cooling spray system after a large or medium sized LOCA.
- The principal human interactions involved in the risk dominant sequences include for PWR:s are:

. failure to depressurise and failure to switch to high-head recirculation after a small LOCA (a number of other operator actions are almost as important).

Reference study on uncertainty and sensitivity analysis

The study concerned the risk dominant accident sequence described above. The selected sequence is dominated by CCF contributions for MOV:s (previously studied in the CCF data Benchmark exercise) and by operator failure to initiate manual depressurisation (previously studied in the reference study on human interaction).

- The statistical uncertainties associated with the estimated frequency of the analysed accident sequence are large, i.e. the uncertainty interval covers at least two decades.
- Computer codes for Monte Carlo analysis of uncertainty propagation (MONTEC, MOCARE, SAMPLE and SPASM) were compared and appear to be in good agreement.

SUPER-ASAR - A Comparative Review of Completed PSA:s

In 1986, PSA:s had been performed for eight out of twelve Swedish nuclear power plants. The review of these analyses, carried out by SKI, clearly indicated significant differences in scope, degree of detail, coverage, etc. Application of a broad spectrum of methods and assumptions had had a decisive impact on PSA results, which complicated adequate comparison. This was the background to the SUPER-ASAR programme launched by SKI in 1986.

The main objectives of this project were

- to survey and compare the results of Swedish PSA:s with due concern to differences in assumptions, modelling and completeness,
- to facilitate the use of completed PSA:s in the process of decision-making, and
- to establish priorities for research projects within the area of PSA.

The project was carried out in two phases. During the first phase, the qualitative features of the studies were reviewed, including qualitative methods and choice of data. The following elements of the analyses were made the subject of a detailed comparison:

- modelling of initiating events,
- modelling of accident sequences,
- system modelling,
- treatment of reliability data,
- treatment of dependencies, and
- treatment of human interactions.

The qualitative phase confirmed that direct comparison of the quantitative results of Swedish PSA:s is not meaningful. It was, however, concluded that an unbiased use of the PSA:s, e.g. in the context of decision-making, would be facilitated if quantitative evaluations of identified qualitative differences in models and assumptions were available.

This was the basis for the second phase of the programme, where a quantitative analysis was made of the discrepancies identified in the first phase. These discrepancies were divided into two categories:

1. Trivial, i.e. items that are not subject to controversy and that may be easily handled as a part of the revision of the PSA:s.
2. Unresolved problems, i.e. items which call for more detailed supplementary studies and/or research work.

Some of the identifies problem areas, which were judged to be important for the quantitative results of the PSA, and for which further in-depth studies were required are:

- Internal pipe breaks
  Basis for plant-specific assignment of frequencies, treatment of the "leak before break" issue, proper categorisation
- External pipe break
  As above, scope and categorisation not always clear
- Transients
  More consequent and consistent frequency estimation using the common categorisation models generated in phase 1 of the programme

- Definition of core damage
  Clarification of assumed interaction between system crediting and containment behaviour
- Consistency in system modelling and crediting
  Applies to e.g. feedwater system and pressure relief system
- Reliability data
  Updating of PSA:s and revision of available data sources
- CCF models and data
  More consistent use of quantification methods and parameters, collection of CCF data based on Swedish and Finnish experience
- Physical interaction dependencies
  Extended scope, in particular dynamic effects in connection with LOCA:s
- Common Cause Initiators
  Need for supplementary analyses
- Back-flush operation
  Need for this safety function after LOCA, associated operator actions
- Human interactions
  Need for more consistent modelling including supplementary situation-specific studies, recovery actions and errors of commission

The ultimate aim of the second phase of SUPER-ASAR was to generate plant-specific reference PSA:s. These were to be based on the available PSA:s, with findings from the SUPER-ASAR programme concerning items of central importance integrated into the models. This would provide a basis for performing consistent and more comparable PSA:s by retaining the plant-specific features of the PSA:s in models and data and applying similar basic assumptions and a similar approach to the identified problem areas.

NKS/SIK-1 - Safety Evaluation by Living PSA

The project was a four year Nordic project (1990-94) dealing with procedures and applications for the planning of operational activities in analysis of operating experience. It defined and demonstrated the use of living PSA (LPSA) for safety evaluations and for identification of improvements in operational safety.

Routines and procedures of how to utilise LPSA were demonstrated in case studies. The demonstrations include applications such as planning of surveillance tests and test schemes, maintenance planning, optimisation of limiting conditions of operation and risk control of exemptions from Technical Specifications.

Often, "living PSA" simply means that a PSA is kept up to date with plant changes. In the NKS/SIK-1 project, it is attempted to give a much wider definition to the concept. Thus, LPSA implies making use of the dynamic properties of a PSA to assess, monitor and follow up plant risk. The modification of static component and system models into dynamic ones is the main effort to be carried out in the development of a basic LPSA model.

In table C1, the key features of the LPSA approach as covered in the NKS/SIK-1 project are summarised.

In an LPSA model, all observations, such as maintenance and repair should be included and easily updated in order to reflect changes in component configuration. This requires an extensive and flexible component model. The basic model used in the project is an extension of the component model used in the Nordic Reliability Data Book, including unavailability due to test and repair, and with the possibility to model periodic testing. The model should account for the fact that some failures cannot be detected in tests, but will only manifest themselves at a real demand. A general model, covering all combinations of time-dependent and time independent failure modes, detectability with respect to both modes, etc. is difficult to create, and even more difficult to apply.

Table C1 *Living PSA as studied in the NKS/SIK-1 project*

	Long term risk planning	**Risk planning of operational activities**	**Risk analysis of operating experience**
Approach	Risk assessment	Risk monitoring	Risk follow-up
Result	• Identification of risk contributors. • Comparison of alternative designs and procedures	• Test and maintenance planning. • Evaluation of Technical Specifications • Operational decision making	• Analysis of operating experience. • Feedback of operational risk experience. • Verification of PSA models.
Risk measure	Nominal risk. Baseline risk	Instantaneous risk	Retrospective risk Probabilistic indicators
Objective	To continue the risk assessment process started with the basic PSA by extending and improving the basic model and data to provide a general risk evaluation tool for analysing the safety effects of changes in plant design and procedures.	To support the operational management by providing means for searching optimal operational, maintenance and testing strategies from the safety point of view. The results shall provide support for risk decision making in the short term or in the planning mode.	To provide a general risk evaluation tool for analysing the safety effects of incidents and plant status changes. The analyses are used to identify possible high risk situations, rank the occurred events from the safety point of view and get feedback from operational events for the identification of risk contributors.

In a number of demonstration case studies, an LPSA model for the Oskarshamn 2 BWR has been developed and evaluated. Time dependent component models were introduced and updated with data from the Nordic Reliability Data Book. Variations of the plant risk level over one operating year with the existing test scheme was calculated and compared to the modified schemes. By making systematic use of time dependent component data and risk importance measures, the number of tests could be reduced by 43% without increasing the average risk.

APRI - Accident Phenomena of Risk Importance
The project is performed in co-operation between SKI, the Swedish utilities and TVO (Finland). It was initiated in 1992 and it's first phase was finished in 1995.

The aim of the project were
- to provide a basis for evaluation of phenomena occurring in connection with severe accidents, and participation in probabilistic analyses of these phenomena,
- to support experiments aimed at validating and developing MAAP and other analysis tools, and
- to develop the knowledge basis required for the further development of accident management methods.

The APRI project was divided into four parts:

- Development of methods and phenomenology for use in PSA:s.
- Phenomenology in connection with reflooding of overheated reactor core.
- Fragmentisation and coolability of the core melt outside the reactor vessel but within the containment.
- Participation in the NRC research programme on severe accidents.

For BWR:s, the following risk-dominant phenomena have been identified:
- core melt coolability in containment,
- steam explosion after reactor vessel melt-through, and
- direct containment heating.

For PWR:s, the following risk-dominant phenomena have been identified:
- hydrogen deflagration,
- direct containment heating,
- steam explosion in reactor vessel or containment,
- melt-through of containment floor,
- thermal stress on electrical penetrations, and
- reactor vessel rupture with missiles.

The analysis of phenomena within the reactor vessel, was focused on the prerequisites for various melt-through cases. Modelling with the MAAP and APRIL codes were critically reviewed.

The APRI analyses of phenomena within the reactor vessel have resulted in a need to partly revise conclusions from previous projects, but also largely confirmed the established picture of the accident progression. Thus, as long as there is some water left in the bottom of the reactor vessel, the risk for an early local melt-through of the vessel seems to be lower than previously thought. Calculations show, that a damaged core can still be cooled in the core region if the vessel is supplied with water within one to two hours after the start of the accident. There is a risk for recriticality, but its consequences are deemed to be manageable.

Within the sub-project on fragmentisation and coolability of the core melt, the interaction of core melt and water was studied. The subject is crucial for Swedish BWR:s, especially for the ones having an annular-shaped condensation pool. In these BWR:s, the space under the reactor vessel will be water filled as part of the accident management procedures, in order to protect electrical penetrations on the containment floor. The aim of the analyses performed

was to investigate if the containment integrity is challenged by a non-coolable melt on the containment floor or by steam explosions. The APRI analyses of containment phenomena have shown that the phenomenological uncertainties are larger than previously known.

The APRI project also aimed at developing methods and methodology for the performance of level 2 PSA:s, and was partly co-ordinated with the performance of the first generation of complete Swedish level 2 PSA:s. A methods group within the project was created to discuss issues resulting from the performance of level 2 analyses, and to propose ways of handling the various phenomena within a level 2 PSA.

Presently, the second part of the APRI project is being started. It will be performed during the period 1996-98, and is planned to include the following main parts:

- CSARP (Co-operative Severe Accident Research Program)
  Participation in CSARP, i.e. the NRC research program on severe accidents including theoretical and experimental investigations as well as development and validation of computer programs for severe accident analysis.
- ACE/ACEX (Advances Containment Experiments/ ACE Extensions)
  Participation in remaining ACE experiments on coolability of the core melt.
  Participation in remaining ACEX, including model development for ACE experiments.
- Research program at KTH (Royal Institute of Technology, Stockholm) concerning the behaviour of the core melt in the reactor vessel, including both experiments and theoretical calculations.
- PHOEBUS
  PHOEBUS is an EU program, with the participation of some non-EU countries (e.g. Japan and the USA). The program includes large scale experiments on severe accidents. The experiments focus on the behaviour of fuel elements during melting as well as on the release, transportation and separation of radioactive products in the primary system and containment.
- Development of a methods handbook for level 2 PSA
- Experiments concerning melt-through of vessel bottom wall-entrances in ABB Atom BWR, including validation of computer programs.
- Generic investigation on coolability of core melt in reactor vessel

References

[1]Närförläggningsutredningen, Närförläggning av kärnkraftverk. SOU 1974:56, Industridepartementet, Stockholm 1974, ISBN 91-38-01579-X.

[2]Reaktorsäkerhetsutredningen (RSU) (Final report from the Reactor Safety Investigation, in Swedish), SOU 1979.

[3]Dinsmore, S. (editor), PRA Uses and Techniques - A Nordic Perspective. Summary Report of the NKA Project SÄK-1, NKA Report, June 1985, ISBN 87-503-5539-2.

[4]Laakso, K. (editor), Optimization of Technical Specifications by Use of Probabilistic Methods - A Nordic Perspective. Final Report of the NKA Project RAS-450, NKA Report Nord 1990:33, May 1990, ISBN 87-7303-422-3.

[5]Hirschberg, S. (editor), Dependencies, Human Interactions and Uncertainties in Probabilistic Safety Assessment. Final Report of the NKA Project RAS-470, NKA Report Nord 1990:57, April 1990, ISBN 87-7303-445-1.

[6]Projekt SUPER-ASAR (in Swedish), SKI Technical Report 90:3, 90:4.

[7]Säkerhets- och strålskyddsläget vid de svenska kärnkraftverken 1994-95 (Status regarding safety and radiation protection at Swedish nuclear power plants 1994-95, in Swedish). SKI Technical Report 95:63

[8]Johansson, G. and Holmberg, J (editors), Safety Evaluation by Living PSA - Procedures and Applications for Planning of Operational Activities and Analysis of Operating Experience. SKI Technical Report 94:2, NKS/SIK-1(93)16, January 1994, ISSN 1104-1374.

[9]APRI Slutrapport från fas 1

[10]Frid, W., Description of the APRI project (in Swedish), Nucleus - Forskningsnytt från statens kärnkraftinspektion, 11/95 pp 36-37, June 1995.

[11]Angner, A. (editor), External Events - Report from the first project year, ES Konsult Report 14/94, May 1995.

[12]Knochenhauer, M., Mapping of Component Dependencies for the Analysis of External Events. PSAM III - Probabilistic Safety Analysis and Management, Crete, Greece, June 1996.

[13]Andersson, K., NKS/RAK-1: Strategy for Reactor Safety, Project Status in April 1995. OECD/NEA meeting on "Generic Study on Effects of Ageing on Components of Selected Safety Related Systems in NPPs", Stockholm, April 6-7, 1995.

[14]Lydell, B. O. Y., Reliability of High Energy Pipework - Industrial Pipework Failures, Failure Causes and Risk Management. 1st Status Report: Survey of Pipe Reliability Analysis Considerations. RSA Technologies Report RSA-R-94-33, SKI Ref. No. 14.2-940477, December 1994.

[15]Nyman, R. et al., Reliability of High Energy Pipework - Industrial Pipework Failures, Failure Causes and Risk Management. A Summary of Project Status. SKI Report SKI/RA-010/95 (RSA-R-95-10a, SKI Ref. No. 14.2-940477), April 1995.

[16]Kecheng Shen and Nyman, R., Inventory and Statistical Treatment of Ageing Phenomena in Swedish Nuclear Power Plant Unit. SKI/RA Report 012/94.

[17]Pörn, K., Pilot Study - Further Development of a Trend Model for Ageing Analysis. SKI Report 94:31, April 1995.

[18]Johansson, G., Werner, W., International Common Cause Failure Data Exchange (ICDE) - Project Description, October 1995.

[19]Werner, W., Compilation of Selected Modifications and Backfits in Swedish, German and US PWR and BWR Plants, SKI Report 95:25.

[20]Karlsson, L., Karnik, P, Lidh, B., Modification and Backfitting in Ringhals Nuclear Power Plant in Safety Related Systems. SKI Technical Report 95:3, December 1994, ISSN 1104-1374.

[21]Karlsson, L., Karnik, P, Lidh, B., Modification and Backfitting in Forsmark Nuclear Power Plant Unit 1 and 2 in Safety Related Systems. SKI Technical Report 95:8, January 1995, ISSN 1104-1374.

[22] Modification and Backfitting at the Oskarshamn Nuclear Power Plant Unit 2 in Safety Related Systems. SKI Technical Report 95:23.

[23] Modification and Backfitting at the Barsebäck Nuclear Power Plant Unit 1 and 2 in Safety Related Systems. SKI Technical Report 95:24.

[24]Seismic Safety - Generell projektbeskrivning

[25]Project Seismic Safety - Characterization of Seismic Ground Motion for Probabilistic Safety Analyses of Nuclear Facilities in Sweden, (Summary Report). SKI Technical Report 92:3, April 1992.

[26]Engelbrektsson, A., Project Seismic Safety - Characterization of Seismic Ground Motion for Probabilistic Safety Analyses of Nuclear Facilities in Sweden. SMIRT 89, Structural Mechanics in Reactor Technology, Anaheim, California, August 1989.

[27]Mankamo, T., Björe, S. and Olsson, L., CCF Analysis of High Redundancy Systems - Safety/Relief Valve Data Analysis and Reference BWR Application - Main Report. SKI Technical Report 91:6, December 1992.

[28]Bengtz, M. and Hirschberg, S., Retrospective Analysis of Human Interaction in Swedish Probabilsitic Safety Studies. SUPER-ASAR, Final Report from Phase 1, Chapter 9, Human Interaction, July 1987.
[29]Holmgren, P., HRA in Swedish PSAs, RELCON Report, 1994-08-20.
[30]Systematic Human Action Reliability Procedure (SHARP). EPRI NP-3583, Project 2170-3. Interim report, June 1984.
[31]Swain, A.D., Guttman, H.E., Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications. Final report. NUREG/CR-1278, USNRC, August 1983.
[32] T-boken, Tillförlitlighetsdata för komponenter i nordiska kraftreaktorer, version 4, 1995. ISBN 91-7186-303-6 (in Swedish)
[33] T-book, Reliability data for components in Nordic nuclear power plants, version 3, 1992. ISBN 91-7186-294-3 (in English)
[34]I-boken, Initiating Events at Nordic Nuclear Power Plants, version 2, 1994. SKI Report 94:12, ISSN 1104-1374, ISRN SKI--94/1--SE.
[35]Nyman, R. and Angner, A., STAGBAS2-DB and the Production of the Incident Catalogue and Trend Catalogue. ESREL'93 - European Reliability Conference, Munich, Germany, May 10-12, 1993.
[36]NUREG 1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants.
[37]Risk Spectrum, User's Manual, RELCON Teknik AB, 1994
[38]Nyman, R., SKIRES, SKI Resultathantering av svenska PSA-studier, October 1994.
[39]Safety Evaluation of Operating Nuclear Power Plants Built to Earlier Standards, IAEA reports INSAG 8 and CB5.

Postadress/Postal address
SKI
S-106 58 STOCKHOLM

Telefon/Telephone
Nat 08-698 84 00
Int +46 8 698 84 00

Telefax
Nat 08-661 90 86
Int +46 8 661 90 86

Telex
11961 SWEATOM S