Nuclear Safety

Safety Evaluation by Living Probabilistic Safety Assessment and Safety Indicators

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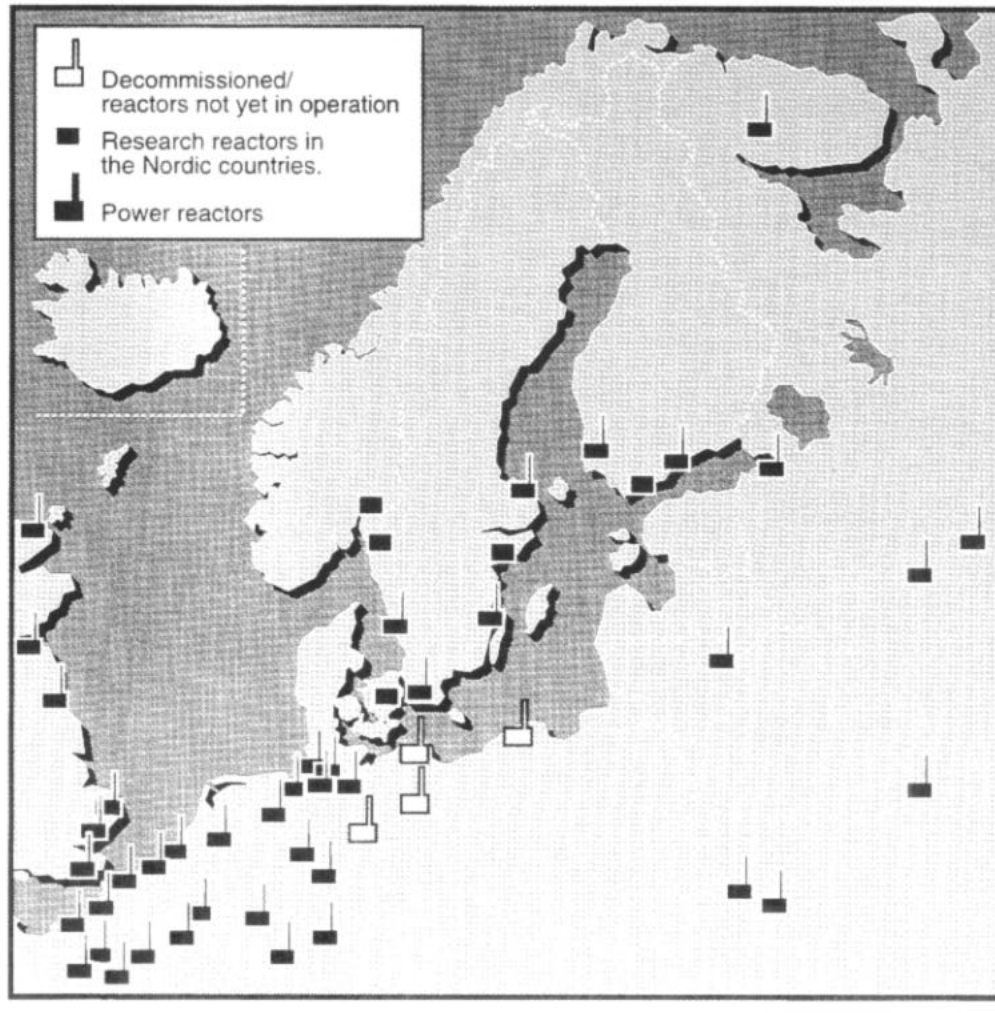

Safety Evaluation by Living Probabilistic Safety Assessment and Safety Indicators

Safety Evaluation by Living Probabilistic Safety Assessment and Safety Indicators

Final Report of the Nordic Nuclear Safety Research Project SIK-1

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The Nordic Council of Ministers
was established in 1971. It submits proposals on co-operation between the governments of the five Nordic countries to the Nordic Council, implements the Council's recommendations and reports on results, while directing the work carried out in the targeted areas. The Prime Ministers of the five Nordic countries assume overall responsibility for the co-operation measures, which are co-ordinated by the ministers for co-operation and the Nordic Co-operation Committee. The composition of the Council of Ministers varies, depending on the nature of the issue to be treated.

The Nordic Council
was formed in 1952 to promote co-operation between the parliaments and governments of Denmark, Iceland, Norway and Sweden. Finland joined in 1955. At the sessions held by the Council, representatives from the Faroe Islands and Greenland form part of the Danish delegation, while Åland is represented on the Finnish delegation. The Council consists of 87 elected members - all of whom are members of parliament. The Nordic Council takes initiatives, acts in a consultative capacity and monitors co-operation measures. The Council operates via its institutions: the Plenary Assembly, the Presidium, and standing committees.

ABSTRACT

A continuous monitoring and follow-up of the risks involved in the operation of a nuclear power plant is an important part of the operational safety management. In living probabilistic safety assessment (PSA), the plant specific PSA is applied in daily safety work to support solving of short-term problems, and maintaining as well as enhancing safety in the long term. A quicker and more problem oriented feedback of operating experience can be achieved by well defined safety indicators, highlighting important trends and possible recurrence of operational problems at the plant. Living PSA and safety indicators should be used in combination to effectively support decision making on safety related issues. The Nordic NKS/SIK-1 project also showed how a more systematic and clear basis for such decisions can be formulated by the method of decision analysis.

Keywords: Safety analysis, safety management, decision making, operations research, nuclear power plants, performance indicators, probabilistic safety analysis.

TIIVISTELMÄ

Pohjoismaisessa NKS/SIK-1-projektissa on tutkittu menetelmiä, joilla voidaan edistää ydinvoimalaitoksen käyttöturvallisuuden hallintaa. Todennäköisyyspohjaisesta turvallisuusanalyysistä (PSA) on kehitettävissä työkalu, jonka avulla tilannekohtaiseen riskiin vaikuttavat tekijät voidaan nopeasti arvioida ja tätä tietoa voidaan käyttää hyväksi päätettäessä toimenpiteistä. Elävä PSA antaa entistä paremmat mahdollisuudet kunnossapidon, korjausten, koestusten ja muiden käyttötoimenpiteiden turvallisuussuunnitteluun. Turvallisuusindikaattoreiden avulla voidaan käyttökokemuksia hyödyntää nopeammin ja käyttää valikoidummin hyväksi. Koska elävällä PSA:lla ja turvallisuusindikaattoreilla on samat tavoitteet ja informaatiolähteet, niitä pitäisi käyttää tehokkaasti toisiaan täydentäen päätöksenteon tukena. Projektissa on myös osoitettu, kuinka turvallisuuteen liittyvän päätöksenteko-ongelman voi jäsennellä ja selkeyttää päätösanalyysimenetelmän avulla.

SAMMANFATTNING

Att fortlöpande övervaka eller följa upp säkerheten och risken under drift av ett kärnkraftverk är en viktig del av säkerhetsstyrning. Utveckling och demonstration av metoder för detta är den huvudsakliga målsättningen med det nordiska projektet NKS/SIK-1. I en levande sannolikhetsbaserad säkerhetsanalys (PSA) används en anläggningsspecifik PSA i det dagliga säkerhetsarbetet för att öka medvetandet om de risker som är involverade i olika driftskeden och underhållsåtgärder. En snabb och mera fokuserad återföring av säkerhetsrelaterade drifterfarenheter kan erhållas med väl definierade säkerhetsindikatorer. Indikatorerna ger information om verkliga trender eller problem om de möjligen upprepas i en anläggning. Arbetet med levande PSA och säkerhetsindikatorer har samma målsättning, och använder samma information. De kan användas var och en för sig eller tillsammans som underlag och stöd i beslutssituationer som berör säkerheten. Projektet har även demonstrerat metodik för strukturering och viktning av alternativa lösningar vid beslutsfattande som berör säkerheten.

EXECUTIVE SUMMARY

During more than ten years, probabilistic safety analyses (PSA) have been performed for each of the 16 nuclear power units in Finland and Sweden. As a result of such analyses, significant risk contributors have been systematically identified, and corresponding weak points in design and operating procedures have been corrected. This project has further developed probabilistic methods with the aim of applying them for management of safety also during operation.

Living PSA is a safety management system for daily use. It requires availability of PSA made specifically for the plant in question, and an information support system. In the living use of PSA, the PSA model must be able to express the momentary risk as a consequence of the actual conditions of equipment that is safety related in the plant. The PSA model and programme should thus be developed to become more dynamic. A living PSA programme involves steps to keep the PSA model updated, to evaluate the safety implications of current or planned conditions of the safety related systems, and to direct changes in the operation, maintenance and testing of these systems. The main aims of living PSA are:

- long-term safety planning,
- short-term safety planning of operational activities,
- risk analysis of operating experience.

A successful living PSA programme requires the full dedication of the involved plant personnel. It is up to the plant management to take the leadership in the use of the methods and tools offered so that they lead to continuous improvement of the safety management.

The safety of a nuclear power unit can be managed more effectively by enhancing the use of suitably chosen indicators of safety performance and monitoring their trends from time to time. Incident rates can for example serve as valuable indicators of the recurrence of operational problems at a unit. Such indicators make the experience feedback faster so that early signals of impending problems can be detected before severe incidents or accidents occur. Once an indicator system has been set up, it can benefit from the extensive data regarding maintenance and incidents, that are now becoming available in computerized information systems. In the project, indicators that are specific for individual units have been defined. They are chosen so that they reflect degrading as well as improving trends in the performance of safety related systems. These findings would otherwise be hidden in the large amount of incident data submitted to the regulatory body or in the maintenance data stored in the maintenance data bases at the

plants. After an in-depth analysis of causes, the actual problem areas can be identified in order to consider measures to improve the operational safety in individual units. Successful measures and good practices can also be identified. They can subsequently be transferred for use at similar units and in other organizations.

A safety indicator system has been designed to assist utilities and authorities in the selection and use of safety indicators for specific plants. The individual indicators represent the actual defence lines of the defence-in-depth strategy to assure the confinement of the radioactive material within the plant. The system will thus steer the reporting of experience and safety evaluation to cover the actual safety and physical barriers that are essential for nuclear safety. The safety-relevance of the various unit-specific indicators, or the incidents occurred, can be evaluated by using the plant-specific PSA to express their significance for the overall risk. Other uses of safety indicators which are also considered include work management in the organization. An early warning system with use of indicators has been developed by a utility participating in the project. A part of the system has been implemented in routine use and another part is in trial use. The indicator system can provide a comprehensive view of the units' safety status and has lead to particular consideration of trends requiring attention.

Many factors and objectives must be considered prior to complex decisions related to safety. This applies e.g. to technical modifications in a plant or to a utility's application for an exemption from the technical specifications for safe operation of its plant. In the project, it is shown how decision analysis can be used to formulate objectives for complex decisions and to rank different options. Decision analysis helps structuring and weighting the decision criteria and choosing from the options. The raw data for the decision analysis may consist of results of PSA, experience feedback by safety indicators and other relevant technical as well as operational information. A structured framework for risk decision making has been evaluated for an actual case – a temporary exemption from technical specifications.

In the project, it has been demonstrated that the use of living PSA, safety indicators and decision analysis has an important potential in safety management and in decision making regarding risk levels. Further practical implementation is needed, however. The use of these methods could be promoted by training of the staff concerned. The project recommends for consideration that living PSA as well as safety and maintenance indicators would become parts of the plant technical documentation and information systems. An improved selectivity of the experience feedback saves resources and puts more emphasis on the analysis of safety related operating experience.

CONTENTS

1 INTRODUCTION

1.1 Objectives and outline

The NKS/SIK-1 project was performed within the Nordic research program on reactor safety [1]. It is part of the Nordic nuclear safety research (NKS) program for the period 1990–93. The project was carried out in co-operation with Nordic nuclear power utilities, authorities, research institutes and consultants. In order to collect different experiences and views to be used in the planning of the project, a survey was made among the Nordic utilities, regulatory authorities, research institutes and consultants [NKS/SIK-1(90)13]. An international survey was also made to give more perspective to related developments [NKS/SIK-1(90)10].

The main objective of the project [NKS/SIK-1(90)8] is to define and demonstrate the practical use of:

- living probabilistic safety assessment (PSA), and
- safety indicators,

for safety evaluation and identification of possible improvements in operational safety.

It was pointed out that the state of a nuclear power unit, especially its operational safety and availability, can change:

- through sudden events, e.g. plant transients or pipe breaks, or
- gradually, e.g. through degrading or improving trends of incident rates.

The conceptual idea of using living PSA and safety indicators is shown in Figure 1.

This concept would supplement the use of the operational safety rules in the Technical Specifications by providing improved means for continuous monitoring of the risk level, and for early identification of degradation in safety performance of an operating nuclear power unit.

1.2 Plant and system types studied

The nuclear power units in the Nordic countries have served as objects for case studies in this project. The total number of the nuclear power units in Sweden and Finland is sixteen. The boiling water reactor (BWR) units in Sweden are Barsebäck 1 and 2, Forsmark 1, 2 and 3, Oskarshamn 1, 2 and 3 and Ringhals 1. TVO I and II are the BWR units in Finland. The pressur-

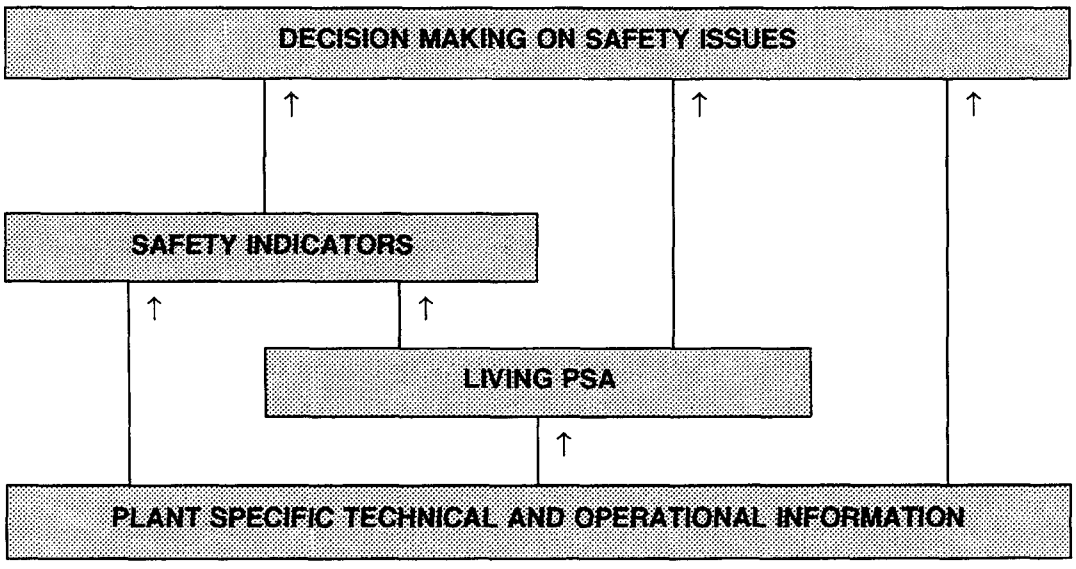


Figure 1. Conceptual idea of the use of living PSA and safety indicators.

ized water reactor (PWR) units in the Nordic countries are Ringhals 2, 3 and 4 in Sweden and Loviisa 1 and 2 in Finland. All Swedish and Finnish nuclear power plants have to some degree been involved in the case studies of this project.

The BWR units can be divided into four generations which correspond to stages in the reactor technology. It is characteristic for the BWR units of the generations 3 (Forsmark 1, 2 and TVO I, II) and 4 (Forsmark 3 and Oskarshamn 3) that the active safety functions are divided into four redundant subsystems. The subsystems are separated physically from each other and each subsystem has a separate electrical bus. This design makes it possible to justify power operation during a limited time with one subsystem unavailable due to planned maintenance actions. The principles of redundancy of the safety systems are not applied as strictly in their designs of the older BWR generations and in the PWR units. For instance, the emergency core cooling systems of the second generation BWRs consist of two redundant subsystems.

Accidents such as core damage can originate from initiating events in combination with failures of safety systems. Thus the process functions and the steam and electricity generating parts of the nuclear power units are also covered in the case studies of the project.

1.3 Introduction to report chapters

This summary report contains the following items:

- In chapter 2, the status, developments and applications required to continue the process towards the living use of PSA are described. Emphasis is put on risk analysis of operating experience.
- In chapter 3, the development and use of a set of safety indicators for operational safety monitoring and effective experience feedback is described.
- In chapter 4, the insights gained in testing of decision analysis to support risk decision making are described.
- The report ends with conclusions and remarks in chapter 5 and with selected abbreviations and terms in Appendix 1.

2 LIVING PSA

2.1 Introduction

By probabilistic safety assessment (PSA) nuclear power plants are assessed with respect to the likelihood of accidents. PSA provides a structured and logical procedure for the identification of credible accident sequences and for the assessment of their corresponding likelihood. To increase the availability of PSA for the operational safety management, the model as well as the whole PSA programme should be developed to a more dynamic tool. The process, to update the PSA model to represent the current or planned configuration and to use the model to evaluate and suggest the changes in the configuration, is called *living* PSA programme [2].

The Finnish and Swedish nuclear power companies have completed the first phase of wide range plant-specific level 1 PSA studies. These analyses have been directed to studies of the internal initiating events, including fires and floods, and accident sequences leading to reactor core damage. The scope of the basic level 1 studies is being expanded to cover other operational states than power operation. The utilities are also continuing with level 2 PSA that concentrates on the analysis of radioactive release terms starting from a core damage.

A natural continuation is the living use of the present level 1 models of the plants. Persons involved in the PSA activities at the utilities and at the regulatory bodies are convinced about the usefulness of these activities. The following PSA application areas got most support by the persons interviewed in the survey [NKS/SIK-1(90)13]:

- comparison of alternative design and procedure changes,
- maintenance planning,
- optimization of the Technical Specifications and evaluation of risks in case of exemptions from the Technical Specifications, and
- planning of surveillance tests and their schemes.

The most important methodological problem areas when carrying out PSA are how to control the incompleteness and conservatism in the models. A general opinion is that the status of present models is not sufficient. The many conservative assumptions reduce the usefulness and acceptance of the results from PSA. The development of proper computer codes is perhaps the easiest problem in the realization of living PSA.

In an international perspective, the number of living PSA applications is limited, and practical experience concerning the use of PSA as an operation-

al tool has not yet accumulated to the point where a general framework for design and structure has been established [NKS/SIK-1(90)10]. The applications have common denominators in their efforts to quantify risk levels according to projected or assumed plant conditions, but in actual use the aims may be quite different. The emphasis is on research efforts in which the applicability of the PSA technique is tested in a reduced scale.

2.2 Living PSA concept

Living PSA is a safety management system for daily uses and it is based on a plant-specific PSA and information support system. The concept is presented in Figure 2.

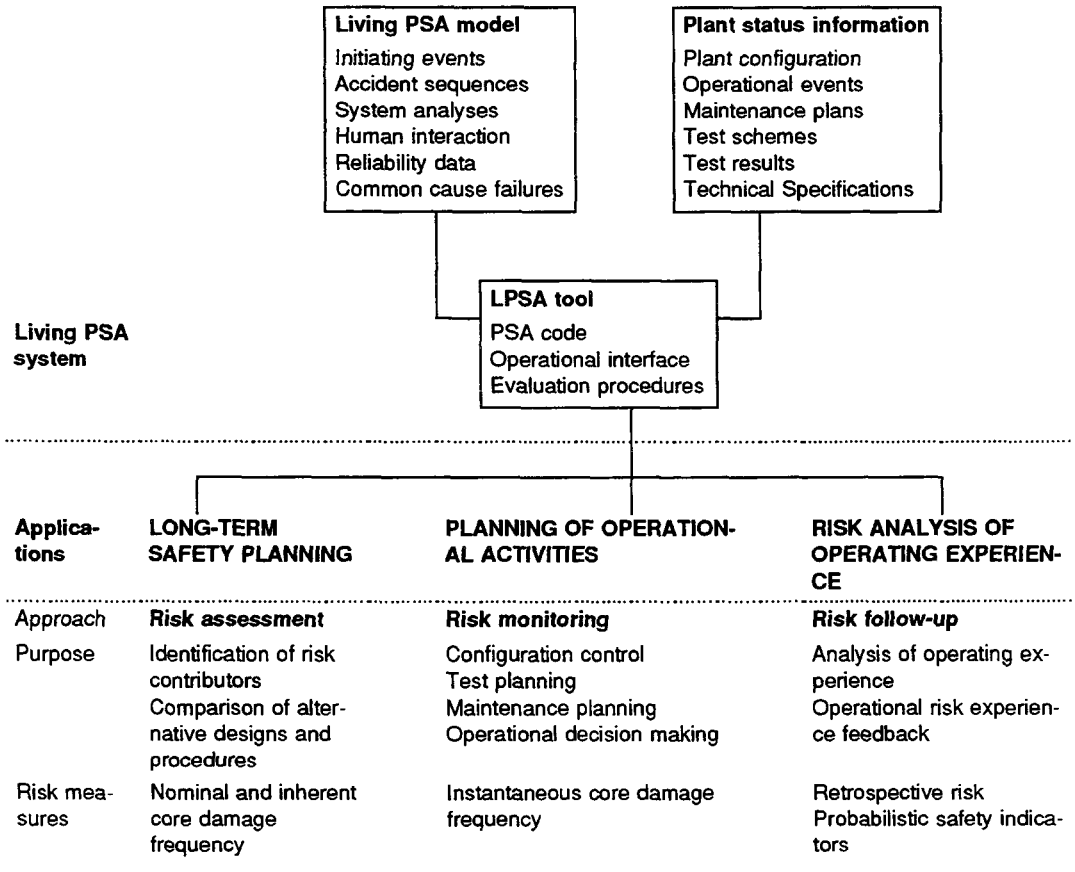


Figure 2. The living PSA concept.

The main application areas of living PSA are [NKS/SIK-1(91)38]:

- long-term safety planning,
- risk planning of operational activities and
- risk analysis of operating experience.

Long-term safety planning continues the risk assessment process started with the basic PSA by expanding and improving the basic models and data to provide a general risk evaluation tool for analyzing the safety effects of changes in plant design and procedures.

Risk planning of operational activities provides means for searching optimal operational, maintenance and testing strategies from a risk point of view. The results support risk decision making in the short term or in a planning mode. Furthermore, the operational limits and conditions given by Technical Specifications can be analyzed with the purpose to balance the requirements with respect to operational flexibility and plant economy. The effect of test intervals and possible staggering of tests of redundant equipment can be evaluated from the risk point of view by a dynamic and time-dependent plant model.

By risk analysis of operating experience, the safety effects of occurred incidents and plant status changes are evaluated. The occurred events can be ranked from safety point of view to get feedback from operational events for the identification of risk contributors.

To use PSA in the proposed applications, the PSA model should be able to express the risk given a specific condition of safety systems, at a given time [NKS/SIK-1(91)38]. This requires the development of time-dependent basic event models. The basic events are related to *evident* (observable) or *hidden* (latent) unavailabilities of components. Evident unavailabilities are known with certainty by the observer. Other unavailabilities remain hidden. It should be easy to update the model when evident unavailabilities such as maintenance or repair of components change the condition of the systems. One problem is to develop realistic common cause failure (CCF) models in this context.

The evaluations with the PSA model can be divided into three approaches depending on the time perspective: risk assessment, risk monitoring and risk follow-up (see Figure 2). The main quantitative results of the risk assessment are the nominal core damage frequency representing an average, long-term plant configuration and inherent core damage frequency representing the lowest theoretically achievable core damage frequency given by the plant design. The instantaneous core damage frequency is considered in the risk monitoring and in the risk follow-up approaches.

The PSA analyst has the responsibility to apply the methods and choose the most relevant risk measures and give them an understandable interpretation. The applicability of the risk measures has been tested in case studies to find a practical set of quantities for the presentation of the results. While some measures do not seem to be applicable for core damage frequency level

evaluations, they might be descriptive for evaluations on system and component level, and vice versa.

2.3 Case studies

Parts of the living PSA concept have been tried out through case studies aimed to identify problem areas for model and method development and to demonstrate the uses of the results. By risk monitoring applications, safer and more economical operational strategies have been studied for the arrangements of periodic tests and preventive maintenance during the power operation cycle. Different operational alternatives were analyzed in the case of failures in safety systems to determine risk optimal allowed down times. Risk follow-up analyses of real operating experiences have been performed to evaluate the significance of the occurred events for safety. The analyses have enabled to identify potential opportunities for future risk reductions.

2.3.1 Practical results

The living PSA model developed for Oskarshamn 2 has been applied for several cases [NKS/SIK-1(92)27]. A surveillance test series could be planned from the risk point of view in which the same risk level can be maintained although 43 % less tests are performed [NKS/SIK-1(93)26]. By shortening the interval for tests decreasing the risk mostly (and prolonging others) and by optimal timing or staggering of the tests, an improved the test scheme was generated. The risk follow-up of one year led to recommendations to always test the gas turbine after maintenance and to always test the redundant gas turbine when one gas turbine has been detected unavailable. If these rules would have been applied, the risk increase due to observed events during the examined year would have been only 30 % of the risk increase actually obtained. A criterion based on comparison of shutdown risk to risk in continued operation would suggest many times longer allowed down times than according to present Technical Specifications except for gas turbines. In the analysis of operational alternatives when a gas turbine failure occurs, allowed down times could be prolonged further, a factor 3, by testing a redundant gas turbine. The relationship between PSA results and safety indicators has been presented by risk importance ranking of the indicators and observed trends [NKS/SIK-1(92)49].

The risk follow-up application performed for Forsmark unit 1 provided results that did not correspond with the operators' perception of the severity of the occurred events [NKS/SIK-1(91)30]. Through the discussions that followed, the actions and events that occurred were better understood and the risk awareness increased. Direct feedback to verify and revise PSA was also obtained. The ranking of the severity of events was facilitated and it

could be improved. Risk increase due to preventive maintenance was 40 %. The retrospective curve of the relative core damage frequency due to system unavailabilities is shown in Figure 3. The nominal core damage frequency is used as a reference level.

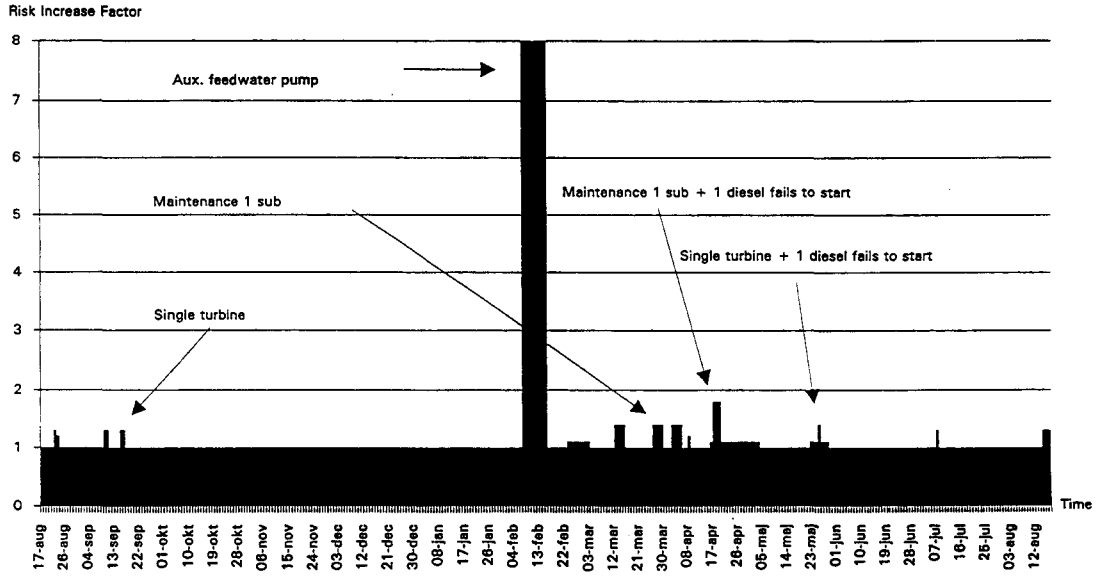


Figure 3. Risk follow-up during power operation at Forsmark 1.

In addition, the occurred initiating events were evaluated. An interfacing system loss of coolant accident (LOCA) gave a risk dose of nine normal operating years.

The risk follow-up with TVO I/II PSA showed the high risk significance of preventive maintenance activities [NKS/SIK-1(91)27]. Subsequently, the maintenance strategy has been changed so that trains in the high pressure and low pressure injection systems would not be simultaneously unavailable. In the analysis of a pressure relief transient at TVO, the progression of the common cause failure phenomenon was modelled by a new approach [NKS/SIK-1(92)35]. Time dependent aspects on testing and failure dependence were shown and changes in test procedures could be suggested accordingly. In the analysis of an external pipe break at TVO, the evolution of an operating procedure error to an incident was analyzed by a qualitative root cause analysis [NKS/SIK-1(93)17]. This method complements the analysis of pipe break frequencies that constitutes one of the large uncertainties in PSA.

In the shutdown risk analysis for TVO I/II, allowed down times for single and multiple failures in a 4x100 % redundant residual heat removal system were examined [NKS/SIK-1(91)4]. The allowed down times for single and double failures were justified. For triple or quadruple failure, a 3 day

allowed down time was suggested instead of cold shutdown within 24 hours. The operational risk at continued operation with a triple or quadruple failure was considerably lower than for the reactor shutdown alternative.

Time-dependent common cause failure models have been developed to compare operational strategies when failures in a diesel generator are detected [NKS/SIK-1(92)13]. It is preferred to complete the repair before carrying out the test of a redundant component.

2.3.2 Limitations

Even though extensive work has been performed to improve the fault tree and event tree models to make them more complete and to reduce conservatism, there are still many remaining deficiencies and uncertainties in PSAs.

Incompleteness/conservatism. The incompleteness problem has to do with missing elements (component failure modes, initiating events and plant functions) in the model, i.e. the entire risk is not covered by the model. In many cases, PSA models are made with conservatism built into the model and data. The reason is usually to simplify the model while at the same time making estimates on the conservative side. Conservatism is acceptable in situations where the main purpose of the calculations is to verify a certain absolute risk level. In many living PSA applications, however, the conservatism may lead to wrong relative importance measures and wrong decisions, and thereby in the end leading to non-conservative actions. The solution to this problem can only be found in a long-term improvement of the plant model. Experience from the development of basic PSAs demonstrates how the models are gradually improved. The next step is to use the PSA for risk analysis of operating experience, risk follow-up, on a long-term basis. This activity will provide experience regarding the capability of the model and provide modelling feedback that will gradually reduce the impact of these problems.

Validation of results. A qualitative validation will always be necessary due to the limitations in the quantitative results caused by incompleteness and conservatism.

Common cause failures (CCF). Time-dependent system unavailabilities and common cause failures are not fully modelled in conventional PSAs. The stand-by system unavailabilities are dependent on test arrangements. The problem is to avoid conservatism and to allow non-symmetric test arrangements as well as to treat events in which one or more redundant components are unavailable. A time-dependent CCF model, analogous to the single failure time-dependent model, can be created by taking into

account the dependence on the test time points, as possible points, where latent faults can be detected and removed, or new faults can be introduced [NKS/SIK-1(92)13]. Further experience, from data analysis, will show if the models suggested in this project are valid. It is now assumed in PSAs that common cause failure probabilities have the same type of dependence of the time between tests as single failure probabilities.

Testing and test effectiveness. The failure data presented in the Nordic reliability data book [3] are based on failure experience generated from surveillance testing. This implies that the failure data do not account for possible limitations of the testing in truly representing the failure characteristics under real working conditions of the components. In practice, a simplified assumption is made that the test conditions are equivalent to the demand requirements – the test is perfect. More profound data analyses are needed to implement the suggested component model that takes test effectiveness into account.

Practical time constraints. There may be too little time to carry out time-dependent risk monitoring and risk follow-up evaluations with the whole model, instead of which simplified or shortened calculations are made. One way to reduce the calculation time is to use precalculated minimal cut set list, and only update basic event probabilities. The changes among evident conditions of the components may bias the result strongly. Many computer programs used in PSA apply only nominal (static) basic event probabilities. A simplified approach for time-dependent evaluations can be applied in this case [NKS/SIK-1(91)38]. The time limit is also one motivation for the use of integrated uncertainty analysis [NKS/SIK-1(91)23], because the time-consuming Monte Carlo simulation is avoided.

Integrated uncertainty analysis. Integrated uncertainty analysis results in a core damage frequency distribution that is directly applicable in decision analysis. Specific problem areas, like uncertainties in CCF-data and state-of-knowledge dependence, are easier to handle, because multi-dimensional uncertainty distributions over probabilities of multiple basic events are integrated on the basic event level. However, the approach requires code developments to calculate the total (unconditional) component failure probability for the basic events [NKS/SIK-1(91)23].

2.4 Development of safety management

PSA should be better integrated with other safety management methods. Operational or design alternatives can be compared in a more understandable way, and a more effective support can be gained to react on gradual or sudden changes in the operational safety status of the plant.

2.4.1 Staffing and cost benefit of living PSA

On an average 4—10 persons are directly working with PSA activities at each of the Nordic power utilities to establish and maintain the basic PSAs. The living PSA activities, to monitor and follow-up the risk, will require a staff of this size on a long term basis. The living PSA activities have a closer relationship to plant operation and maintenance than the basic PSA applications, which to a much larger extent are directed towards plant safety management, designers and authorities. The costs and benefits of living PSA activities are summarized in Table 1.

Table 1. Costs and benefits of living PSA.

Task	Cost	Benefit
Basic PSA	5—10 man-years	Increased risk awareness
Expansions of PSA	1—3 man-years per expansion (6—9 man-years, the level of ambition differs due to plant generation)	Completeness improved
Maintenance of PSA model	0.5—1 man-year per year and unit	Updated models
Living PSA applications	0.5—1 man-year per year and unit	Increased flexibility in testing, maintenance and limiting conditions for operation
		Improved experience feedback
Totally	Upper estimate ≈ 20[1)] man-years plus 2 man-years/year.	"If you think safety is expensive try an accident"

1) In Sweden: The initial investment, the basic PSA, of 10 man-years was required as a part of the first round of periodic safety review, ASAR-80. The expansions, ≈10 man-years, are required as a part of the second periodic safety review, ASAR-90. In Finland: The situation is approximately the same, the work is not carried out within the framework of a periodic safety review.

The valuation of the benefits must be made by the plant personnel in deciding whether or not to implement these activities in the daily safety management.

2.4.2 Implementation and use of living PSA

The development of routines and procedures for living PSA includes transfer of PSA-related information within the organizations. Living PSA application will always require specialists to operate and maintain the model. A better operational interface will allow a more effective use and a

broader spectrum of users to carry out the applications. The plant personnel must be involved and appreciate the benefits of working according to this procedure. The plant organization will in the end decide for itself to what extent these methods are to be used in the safety management of the plant.

Based on the work and the demonstrations carried out it is recommended that a living PSA programme is considered on a plant specific basis. The implementation can preferably be divided into two steps:

1) Prepare procedures, models, and data to carry out:

   - Evaluation of surveillance test intervals: Following this application also configuration control and risk planning will be possible.
   - Evaluation of allowed down times: Following this application also risk planning of maintenance will be possible.
   - Analysis of operating experience by risk follow-up, generation of severity ranking of incidents and use of probabilistic safety indicators.

2) Prepare probabilistic criteria and procedures for risk decision making, e.g. exemptions from limiting conditions for operation in Technical Specifications.

2.4.3 Decision making

A proper use of living PSA applications requires that decision making criteria are established. Probability based criteria are not sufficient in complex decision making situations. They might, however, give guidance or first indication about the acceptability of the decision option. A distinction can be made whether the criteria are used in an absolute or relative manner. An absolute criterion or measure may be needed for regulatory purposes but a relative criterion may be sufficient in plant-specific applications.

2.4.4 Regulatory aspects

Regulatory and inspection activities relate to all of the above mentioned applications of PSA. It is important to review, justify and approve the requirements in the Technical Specifications. Inspection guidance can be obtained by risk assessment using results such as dominant risk contributors.

2.4.5 Broadening the use

Besides the recommended implementation programme, it may be worth considering the following aspects in the use of living PSA [NKS/SIK-1(91)33].

On-line operational activities. Today simplified PSA models are applied in control of down times of systems. In the on-line risk monitoring, the instantaneous core damage frequency is evaluated based on the information about the plant condition. The aim is to support decision making of the operators in the short-term. It will still take some time before a full scale plant model can be used on-line, meaning that it can give nearly continuous assessment of plant condition and risk level.

Integration with other information systems. At present the concept of operator support system based on a full scale PSA is not well developed although there is a number of proposals outlining the structure and functioning of such a system.

Expert system techniques. The PSA model can be looked upon as a knowledge base which is organized according to principles suited for probabilistic assessment. Expert systems techniques can be applied to assist the user in the analysis of the situation and the selection of appropriate actions to bring the plant to a safer state as effectively as possible.

Living PSA as a training tool. A practical introduction of PSA for plant staff that has little or very little experience with PSA should be considered. It enhances the general level of understanding of the capabilities of probabilistic thinking and risk awareness as compared to deterministic safety assessments.

2.5 Conclusions

The early and fast identification of discrepancies and deficiencies in plant design and operation is considered essential for safety. The design aspects on plant safety are handled to a large extent by the basic PSA. By living PSA, the safety aspects on operational, maintenance or testing practices can be evaluated, and modified, and more flexibility in operation and maintenance can be justified. By analyzing occurred incidents, the significance of the events for the safety can be better recognized and the operating experience feedback for proposals of safety-related corrective actions is improved.

3 SAFETY INDICATORS

3.1 Introduction

3.1.1 Standardization of the international performance indicators

The utilities, safety authorities and international organizations have used indicators of various kinds to monitor the performance of nuclear power units. The indicators have often been indefinite or they have had unclear calculation rules. Identically or nearly identically named indicators may have had different definitions and the figures they have yielded have been compared although they, in fact, have not been comparable.

The World Association of Nuclear Operators (WANO) was formed by the international nuclear power community after the Chernobyl accident. The worldwide standardization of the nuclear power plant performance indicators was seen by WANO as one of the important aims. The need for uniform international performance indicator definitions was based both on the demonstrated benefits gained from sharing of operating experience information, and also the recognition that comparisons of plant performance are inevitable. Confusions and inappropriate conclusions or actions due to inconsistent data are expected to be avoided by using consistent and uniform performance indicator definitions.

In 1990, after an international development effort, WANO established for international use a set of 10 performance indicators in the areas of nuclear power plant safety, reliability, efficiency and personnel safety [4]. The standardized performance indicators are presented in Table 2. These indicators are considered to be applicable to nuclear power units of a variety of designs and operational practices.

3.1.2 Need for plant-specific safety indicators

Concerns have been raised if the extent of safety emphasis in the WANO indicator set is sufficient. Although the international performance indicators are generally considered to have quite a positive correlation with safety, they were not believed to have any preventive function at individual plants [NKS/SIK-1(90)13]. Further development and implementation of more detailed and plant-specific indicators is considered useful among nuclear plant operators as well as regulators [5]. Such indicators would be used more closely related to operational practices, programs, and work management of individual plants. Undesired events might thus be better prevented by using more detailed plant-specific indicators for monitoring of safety significant activities and identification of deficiencies at the plant.

Table 2. The WANO performance indicators.

1. Unit Capability Factor
2. Unplanned Capability Loss Factor
3. Unplanned Automatic Scrams per 7000 Hours Critical
4. Safety System Performance
5. Thermal Performance
6. Fuel Reliability
7. Collective Radiation Exposure
8. Volume of Low-Level Solid Radioactive Waste
9. Chemistry Index
10. Lost-Time Accident Rate

One practical screening problem at the plant is the large amount of information received every day. A more problem oriented feedback of experience can be obtained by the use of plant-specific safety indicators, which provides information about the actual trends and recurrence of operational problems at the plant.

3.2 Definition of safety indicators

3.2.1 Purpose of safety indicators

The basic purpose of safety indicators is to help in identifying the early signals of deteriorating performance and thereby to provide means for early warning of impending problems before a serious incident or accident occurs. A warning from safety indicators should initiate further investigation of the causes of the symptoms in order to ensure that corrective actions will be effectively directed.

In addition to their preventive function, the indicators can be used also for:

- definition and tuning of goals or targets and comparing the actual performances with them,
- follow-up of effectiveness of corrective actions and changes after their implementation, and
- identification of good performances in specific areas for transfer of good solutions to similar units or organizations.

3.2.2 Indicator types

The indicators are based on different types of experience data. Information can be obtained directly from the events occurred at the plant, e.g. the number of plant transients, or failure rates in safety related systems and equipment. Such event-based indicators are here called *direct* indicators.

However, only a few safety significant events occur in the plants during operation periods. The management will not have enough feedback to react on and the possible benefit provided by the direct indicators can be limited. Therefore it has become necessary to find such measurable features of plant performance which should provide an advance warning of deteriorating performance before the direct outcome indicators are affected.

One way to approach this problem is to find and develop indirect indicators which can measure the performance of the functional units within the plant organization, such as operation, maintenance, training, and engineering support. The theoretical basis of such indicators implies that certain characteristics of management and organizational behaviours are associated with changes in the likelihood of plant accidents or incidents [6]. If these organizational behaviours can be identified, and also measured by indicators, indicators should lead or signal future physical performance indicators. The value in identifying the *indirect* indicators is the possibility for them to be anticipatory indicators of potential problems. Indirect indicator type is often called predictive or programmatic [7]. An indirect indicator can utilize physical parameters, too. E.g. the Chemistry Index gives a warning on the concentration of important impurities in BWR reactor water before significant corrosion damage is likely. A relation between the indirect and direct indicators is illustrated in Figure 4.

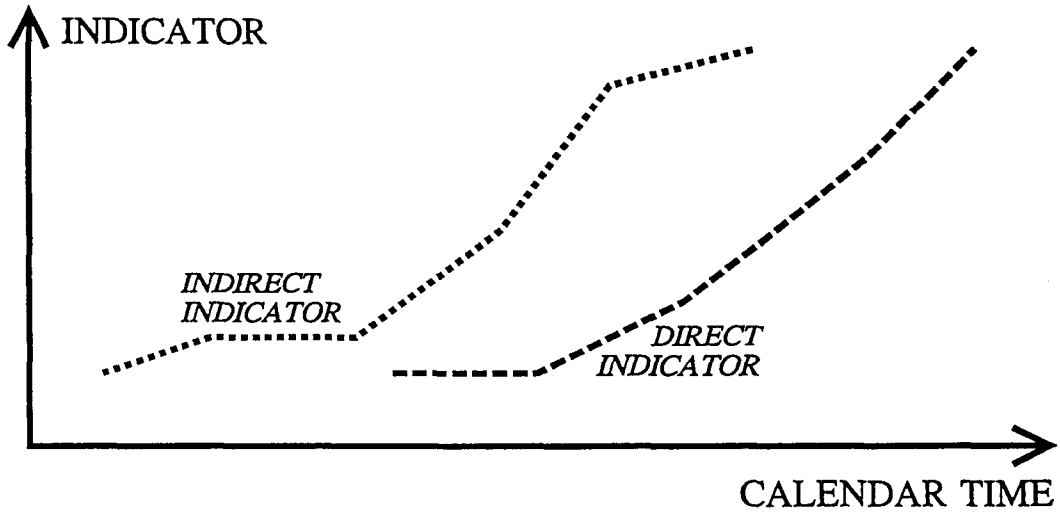


Figure 4. Hypothetical relation between indirect and direct indicator.

3.2.3 Safety indicator system

The development of a system of safety indicators was one of the aims of the project. The defence lines of the strategy of defence-in-depth [8], completed with PSA logic model and structure, were singled out as a suitable framework for identification and structuring of the performance areas related to safety. Once these areas have been identified the safety indicators can be defined. The indicator system shall not cover only the physical barriers providing the hardware means of achieving the nuclear safety. The safety barriers, known as levels of protection in defence-in-depth, shall also be covered to assure the integrity of the physical barriers. Both the international performance indicators and the *plant-specific indicators* can be located into this system in a hierarchical way.

We defined briefly the following performance areas of the plant and its organization related to operational safety [NKS/SIK-1(93)2], NKS/SIK-1(93)23]:

0) *Safety Management:* Managing the performance promoting activities and the day to day operations and works of the plant.
1) *Control of Operation:* Operating the plant under control for generation of electricity and detecting as well as responding to anomalous conditions and incidents.
2) *Safety Functions:* Maintaining the operability of safety systems.
3) *Physical Barriers:* Maintaining by hardware means the continued integrity of the physical barriers.

Any strict boundaries cannot be drawn between the performance areas defined above. Accordingly some overlapping of the performance areas of operational safety cannot be avoided, either. The objective of Safety Management is to foster a good safety culture [9]. This means striving for excellence of plant performance and strengthening the three other performance areas: Control of Operation, Safety Functions and Physical Barriers. The safety systems are provided to prevent incidents from developing into accidents. The physical barriers, providing the confinement of the radioactive material at successive locations, are the fuel matrix, fuel rod cladding, primary coolant boundary and containment.

3.2.4 Indicator specifications

About one hundred safety indicators used by utilities, authorities and international organizations have been collected and refined, and then described in the form of specification. A specification of an indicator comprises the following items: the indicator's name, performance area, respon-

sibility, purpose, definition, data needed, calculation, use and remarks [NKS/SIK-1(93)2].

The indicators have been related to the four performance areas presented above. This division and the specifications enable the utilities and authorities to check the coverage of their indicator sets from the operational safety point of view. A set of operational safety indicators is given as an example in Table 3.

Table 3. Example safety indicators.

Safety Management	**Control of Operation**	**Safety Functions**	**Physical Barriers**
Recurrent fault modes	Transient index	Safety system performance	Tightness index
Maintenance ambition index	Mean time between repairs of components	Common cause failures	Crack index
Safety issues backlog	Unplanned capability loss factor	Length of component unplanned outage	Fuel reliability index

3.3 Application of indicators

3.3.1 Evaluation of indicators

The indications of the absolute level as well as trends of the safety performance of a nuclear power unit are the objective of the safety indicators. Therefore indicators can be used to evaluate safety performance in basically two ways [10]:

- to evaluate the level of performance,
- to evaluate the trend of performance.

Level of performance: The indicator value is compared to some reference value to determine if the indicator value is significantly deviating from the reference value. One way is to use the absolute acceptable performance target as a reference value to determine whether the indicator value is unacceptably high or low compared to the target. Alternatively, the indicator value can be compared to the past performance.

Trend in performance: A trend analysis is performed to determine whether there are any significantly increasing or decreasing trends in the indicator values. Various nonparametric statistical tests, and estimation of appropriate stochastic models, for example, the non-homogeneous Poisson process using

ENHPP code, provide suitable approaches and methods for practical trend analyses [11], [NKS/SIK-1(93)36].

The indicator values should be able to be displayed both numerically and graphically. The different graphical presentations enable the user to visualize the potential trends and levels of indicators.

3.3.2 A utility defined indicator set

The production costs, production, nuclear safety and environmental protection are steered according to the management philosophy of Vattenfall AB by overall goals and policies. A development project has been performed at Vattenfall in close relation to the Nordic research project on Safety Evaluation [NKS/SIK-1(92(6)]. The objective of the Vattenfall project was to suggest, test and commission an indicator set. The aim of the indicator set was, as accurately as practically possible, to indicate the safety levels and their development in different nuclear power units of the utility.

The defence-in-depth principle was selected by the utility as a framework for definition of a set of safety indicators covering the areas mainly related to operational safety. The Figure 5 gives a concrete and understandable view of this framework for the practical safety work.

It is recommended that one or more indicators should be defined for every functional area to cover the different aspects related to operational safety. The several safety barriers (i.e. "levels of protection") to be monitored mainly by indirect indicators are shown in the lower functional areas of Figure 5. The safety barriers of defence-in-depth presuppose also the functional area of safety systems which consists of hardware to be monitored by direct indicators.

The indicators will form a part of an "early warning system" on the nuclear power plant safety and the related quality of activities. The titles of the indicator system are structured in Table 4 according to their potential relation to operational safety. A division into direct and indirect indicators is also made.

The nine indicators "Unplanned Automatic Scrams", "Transient Index", "Fuel Reliability", "Chemistry Index", "Tightness Index", "Licensee Event Report Significance Index", "Recurrent Failure Index", "Unplanned Capability Loss Factor" and "Safety System Performance" were accepted after the development project for routine use. The six indicators, Unplanned Automatic Scrams, Transient Index, Fuel Reliability, Tightness Index, Licensee Event Report Significance Index and Safety System Performance, are now used as a communication tool between the utility management and the nu-

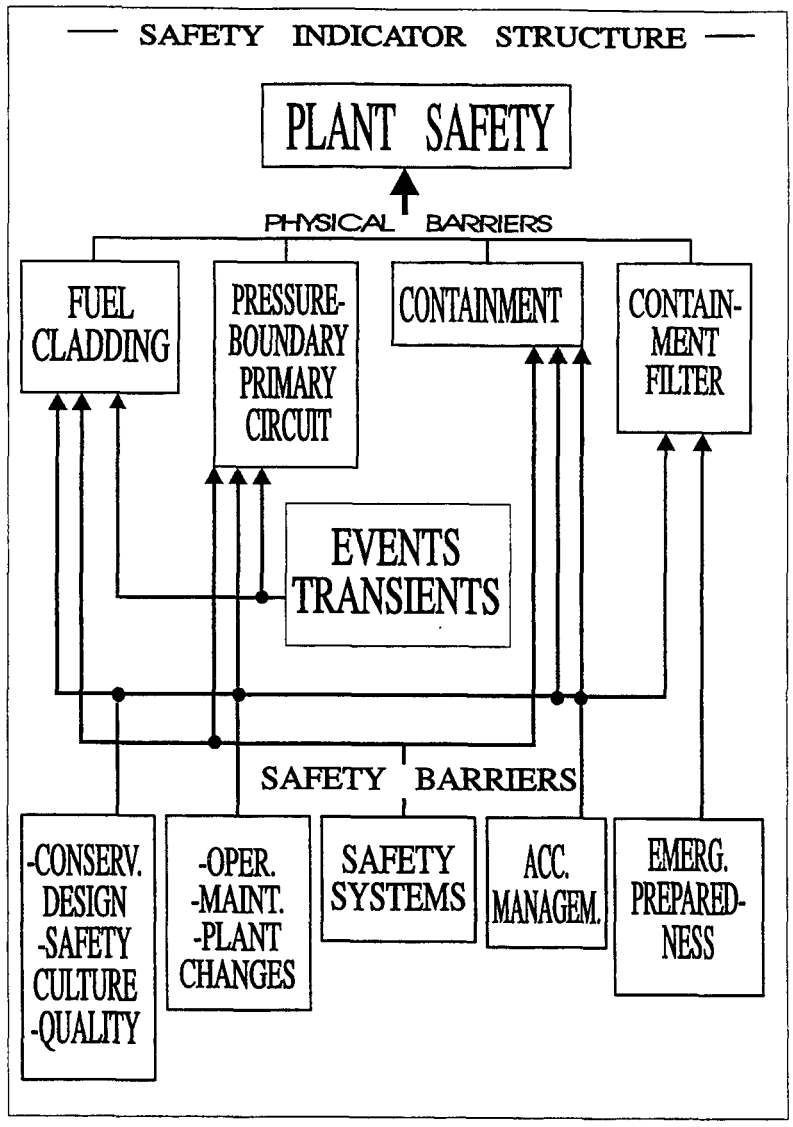


Figure 5. A Vattenfall framework for defining a set of indicators.

clear power units. An analysis and evaluation of these indicators on the different nuclear power units is reported in the 4-months reports to the utility management and plants. These indicators are also used by the utility's central controller function to support the management with safety evaluation and their trends. The remaining indicators of the indicator system are either used at the plants or have been defined and tested for possible later uses. Brief descriptions of some used or tested indicators are given as follows.

TRANSIENT INDEX: This indicator is related to the lifetime of the primary coolant boundary and the quality of the activities of the operation organization. The number of thermal transients during the past year is divided by the remaining annual average number of transients. The number of remaining transients is the difference between the transient budget and the number of occurred transients. The ratio is calculated for all transient

Table 4. An example of location of proposed indicators into different functional areas.

	Title	Defined by	Type
E	EVENTS		
E.1	Unplanned automatic scrams	WANO	D
E.2	Transient index	Vattenfall	D/I
P	PHYSICAL BARRIERS		
P2	Fuel cladding		
P2.1	Fuel reliability	WANO	D
P3	Primary circuit pressure boundary		
P3.1	Chemistry index	WANO	D/I
P3.2	Crack index	Vattenfall	D
P4	Containment		
P4.1	Tightness index	Vattenfall	D
F	SAFETY BARRIERS		
F1	Safety culture, quality		
F1.1	QA index	Vattenfall	I
F1.2	Exemption index	Vattenfall	I
F1.3	Licensee event report significance index	Vattenfall	D
F1.4	Recurrent failure index	Vattenfall	D/I
F2	Operation, maintenance, changes		
F2.1	Maintenance quality index	Vattenfall	I
F2.2	Maintenance ambition index	Vattenfall	I
F2.3	Work order management index	Vattenfall	I
F2.4	Unplanned capability loss factor	WANO	D
F3	Safety and protection systems		
F3.1	Safety system performance	WANO	D
F3.2	Valve failure index	Vattenfall	D

Abbreviations: D = Direct indicator, I = Indirect indicator

types defined in the technical specifications. The highest ratio is the value of the index. The evaluation of the index has initiated modifications in procedures contributing to larger margins against the remaining transient budget. This indicator will replace Unplanned Automatic Scrams as a communication tool at management and controller function level.

TIGHTNESS INDEX: The leakages of the containment isolation valves prior to their maintenance are annually followed up. This indicator is related to the integrity of the containment and the quality of the activities of the maintenance organization. Two indicators have been tested:

- percentage of isolation valves (D>50mm) exhibiting a leakage over the acceptance value in the tightness test, and

- the sum of the measured leakages of all isolation valves divided by the accepted leakage according to release calculations.

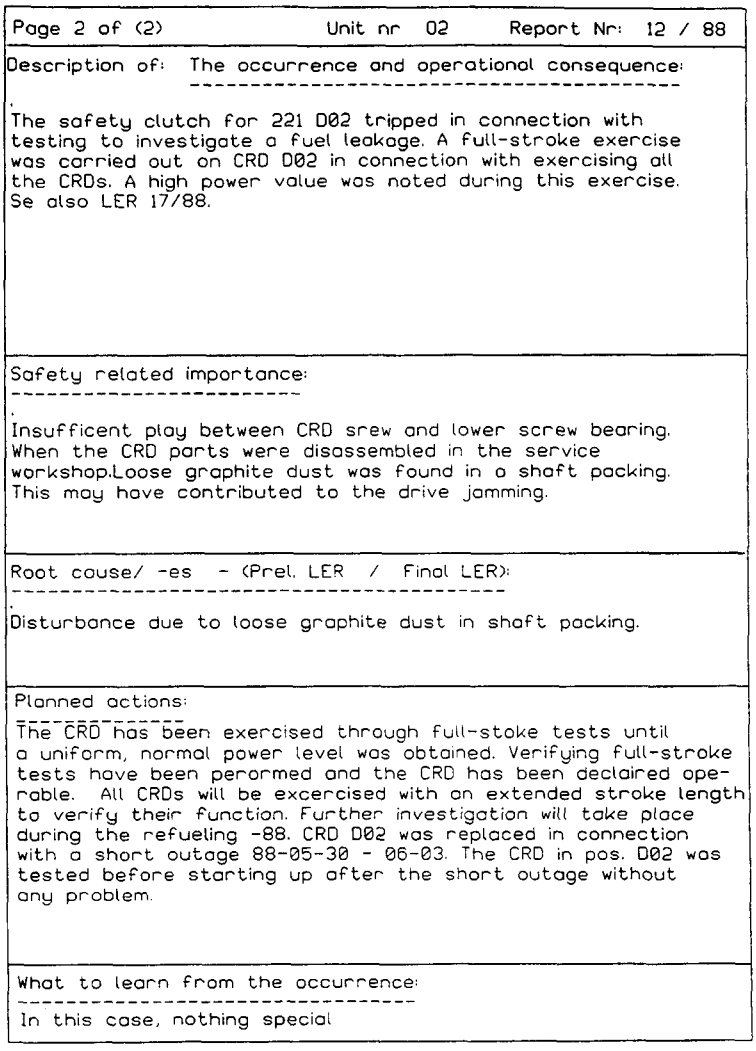
MAINTENANCE AMBITION INDEX: The work orders in the maintenance information system are divided into corrective (CM) and preventive maintenance (PM). Work orders on design modifications are excluded. The ratio CM/PM is the indicator. Very large deviations can be noticed between different plants. It is in some cases important to monitor whether the CM portion is increasing with time so much that the plant is later on overwhelmed by work.

In addition, the nuclear power units use locally other detailed indicators to reveal trends of the causes of scrams, safety related occurrences and component faults for learning from experience and ranking of corrective actions including maintenance development.

3.3.3 Regulatory body uses of trend analysis of incident data

The Swedish Nuclear Power Inspectorate manages a computerized operating experience database, STAGBAS II [NKS/SIK-1(91)20]. The main objective of STAGBAS is to be a tool for classifying, searching, sorting, and analyzing incident information in order to reveal potential problem areas that otherwise would be hidden in a large amount of incident reports. The data base was reconstructed in 1991. At the end of 1993 it included descriptions and standardized classifications of about 5200 safety related occurrences and 960 reactor scrams. A classification part is always attached as front page to a qualitative incident description which is shown as an example in Fig. 6.

The definition and application of safety indicators is based on systematic analysis of these plant-specific operating experiences reported by the Swedish nuclear power plants to the regulatory body. Search patterns on safety related occurrences were generated in STAGBAS. The annual number of incidents originating from different safety functions, systems and components is visualized by statistical diagrams in the so called "Incident Catalogues". The screening of potential observation areas, also called *indicator candidates*, for further studies is based on the answers of the following questions:



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Description of: The occurrence and operational consequence:

The safety clutch for 221 D02 tripped in connection with testing to investigate a fuel leakage. A full-stroke exercise was carried out on CRD D02 in connection with exercising all the CRDs. A high power value was noted during this exercise. Se also LER 17/88.

Safety related importance:

Insufficent play between CRD srew and lower screw bearing. When the CRD parts were disassembled in the service workshop.Loose graphite dust was found in a shaft packing. This may have contributed to the drive jamming.

Root cause/ -es - (Prel. LER / Final LER):

Disturbance due to loose graphite dust in shaft packing.

Planned actions:

The CRD has been exercised through full-stoke tests until a uniform, normal power level was obtained. Verifying full-stroke tests have been perormed and the CRD has been declaired operable. All CRDs will be excercised with an extended stroke length to verify their function. Further investigation will take place during the refueling -88. CRD D02 was replaced in connection with a short outage 88-05-30 - 06-03. The CRD in pos. D02 was tested before starting up after the short outage without any problem.

What to learn from the occurrence:

In this case, nothing special

Figure 6. A report on a safety related occurrence.

- Large number of "similar incidents" in all nuclear power units?
- Significant differences in number of "similar incidents" between similar units or compared to generic averages?
- Large number of "similar incidents" in an individual unit?
- Strongly increasing trend of "similar incidents" in individual units?
- Strongly decreasing trend of "similar incidents" in individual units?

An example diagram presenting a candidate of a unit-specific indicator in the incident catalogue is shown in Figure 7. The safety-relevance of various candidates for unit-specific indicators has also been evaluated by plant-specific PSA to express their significance to the overall risk [NKS/SIK-1(92)49]. In the continuation the underlying failure causes of the potential observation areas are studied in detail order to identify the actual problem areas and confirm their trends [NKS/SIK-1(92)44]. Then the unit-specific indicators, for monitoring of the actual problem areas during several years for individual or several units, can be determined. The results of the studies

show that e.g. the following potential observation areas are common for several units:

- reactivity control; hydraulic scram systems or control rod drives,
- fire protection systems,
- electric power supply, AC and DC buses, battery backed network,
- isolation valves, and
- core spray system and containment vessel spray system.

Potential indicator candidates could however not be identified for the refuelling outage from the database, apparently due to insufficient reporting practices on the occurrences related to e.g. fuel handling and heavy lifts during the shutdown and refuelling state.

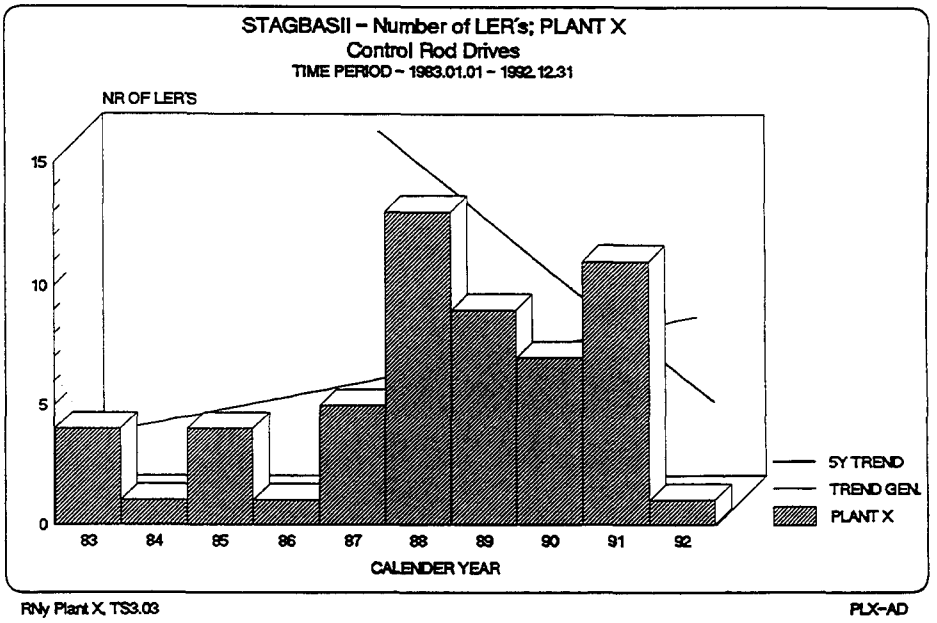

Figure 7. Monitoring of the annual number of safety related occurrences due to faults in control rod drives in one unit.

When the recent incident rates or trends within the same observation areas vary largely between individual units, conclusions may be drawn regarding design features or operational practices to be preferred [12]. The regulatory body promotes activities to develop systematic experience feedback and safety indicator systems at each unit [NKS/SIK-1(92)44].

3.3.4 Maintenance related case studies

Monitoring of safety system unavailability, Loviisa nuclear power plant: A study on maintaining the availability of the Loviisa 2 emergency diesel generator system (DG) was done by Technical Research Centre of Finland in co-operation with Imatran Voima Oy [NKS/SIK-1(92)5]. The Safety System Performance indicator of WANO, among others, was calculated [13]. This indicator for important safety systems is defined as the unavailabilities, due to all causes, of the components in the system during the time period, divided by the number of the trains in the system. However the unavailabilities for the emergency AC power indicator are being recorded only when the emergency DG is unavailable to produce emergency power. The trend of the indicator is shown in Figure 8.

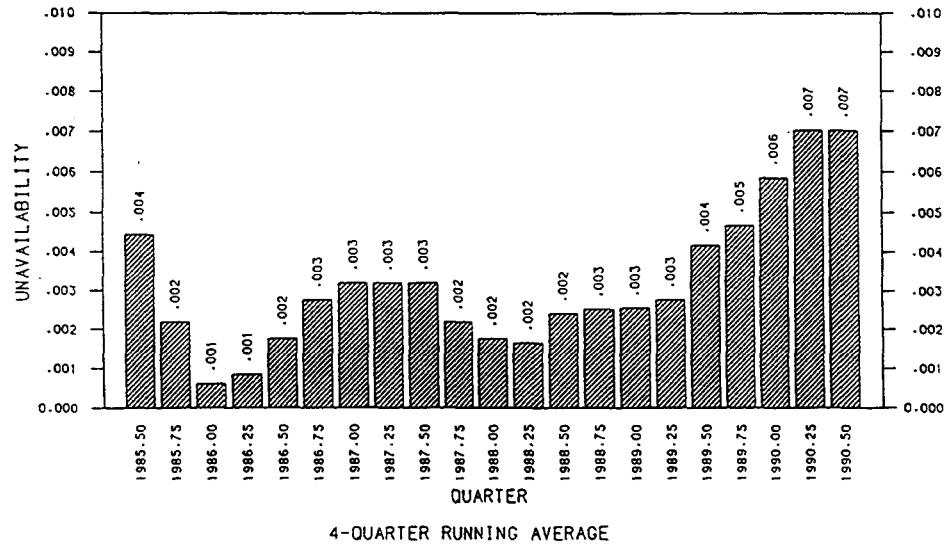

Figure 8. The average unavailability of the four diesel generator trains.

The general trend in the unavailabilities of the DG trains increased since 1989. It was, at least partially, explained by the improved reporting practices of operating experience after the installation of the Loviisa power plant information system (LOTI). No functionally critical failures, preventing a DG directly from operating during a test or a demand, were identified. This indicates that the surveillance program has been effective for the DG system. The unavailabilities originated from corrective maintenance actions of functionally non-critical faults. The study demonstrated the feasibility and usefulness to combine various indicators to the LOTI system for evaluation of the maintenance strategies.

Maintenance indicators, Barsebäck nuclear power plant: A pilot project was carried out at the Barsebäck nuclear power plant on development of indicators at the lowest level of the structure of the safety indicators. The "condition monitoring indicators" can be used for maintenance planning

purposes. These indicators should give information about ageing and degradation of components. Pilot studies on the feed water pump components and main steam isolation valves were done by Risö National Laboratory and Sydkraft [NKS/SIK-1(92)18]. The indicators should utilize the data already existing at the plant and the data should be stored for a proper period in the computers.

The information extracted was failure data from the Swedish TUD data base and the local maintenance planning system and the process data. From the process data it was identified deviations from the initial conditions that are an indication of change, either due to wear or other degradation, although the functional performance of the equipment fulfilled well the requirements of the process. Examples of such deviations could be found by monitoring e.g. the increases in the positions of the control devices of the hydraulic couplings of feed water pumps or the pressure drops over heat exchangers.

Another important part of the project was to present the indicators in an easy way by displays showing both the function and the condition of components as a function of time. This kind of information helps the maintenance planners to detect earlier developing faults, and thus plan better the maintenance actions, in order to avoid functional failures or damage. The maintenance indicator study continues on other safety significant systems concerning "condition monitoring indicators" and their suitable presentation and implementation in the maintenance information system of the plant.

3.4 Conclusions

3.4.1 Specific data needs in development and use of indicators

The development and use of safety indicators benefits from the availability and coverage of the maintenance, incident, safety and quality related data bases. Some of the most important incompletenesses encountered in analysis of incident and maintenance data concern:

- identification and description of root causes and corrective actions,
- definition and reporting of unavailability times of equipment, and
- descriptions in plain language and identification of involved equipment in incident reports.

3.4.2 Indicator system

It is emphasized that the indicators should be used as a group [14]. Focusing on a single indicator, or a narrow set of indicators, can be counterproductive to both safety and long-term performance improvement [13]. The

integrated use of indicators with detailed information on various functional areas enables the user to effectively and quickly track and identify the causes of deviations in the indicators [15].

3.4.3 Implementation into information systems

An effective system for utilization of indicators requires a well organized and motivated organization. All involved parties should have an access to compilations of indicators and their underlying data base.

In daily practice, the user-friendliness of the computerized information systems, supporting the management of plant's day to day operations, plays a central role in the use of the indicators. Examples of information systems are the production management and maintenance information systems. The indicator output from the information systems is proposed to be received on the terminals or PCs connected to the computer network at the plant and possibly at the headquarters.

3.4.4 Improvement of operational safety

The safety indicators form one part of an overall system of performance monitoring and evaluation to assist plant management and personnel in detecting and correcting poor performance. Improving and good performance can be identified, too. Thus the indicators should be used in combination with other assessment tools, and not as the sole basis for decision making.

The use of indicators has lead to identification of safety issues that would otherwise be hidden in the large amount of reports and data available. After an in-depth analysis of causes, the actual problems can be identified in order to consider measures to improve the operational safety in individual units including transfer of good practices between similar units and organizations.

3.4.5 Improvement of effectiveness of experience feedback

The use of a set of well tested and accepted safety indicators has the potential to make the feedback of experience faster and more selective for continuous safety evaluation and development. We recommend this investment for consideration because it will direct the emphasis more on problem oriented and safety-related operating experience and its analysis. The improved selectivity will in long-term save resources in the experience feedback, too.

4 RISK DECISION MAKING

4.1 Introduction

The operational safety management of a nuclear power plant involves efforts made by the utility to minimize the operational risks so that the safety objectives of the utility and society are fulfilled. The management of safety of the plant is based on the assessment of the risks involved, and on the subsequent decisions taken by the designers, operators, regulators and politicians. Many efforts can be considered risk decision making problems, and decision analysis can support solving them. The main phases of the decision analytic process are:

1) the structuring of the problem,
2) the construction of the preference model, and
3) sensitivity studies.

4.2 Benchmark study

A pilot benchmark study was made on a decision making case of an exemption from Technical Specifications [NKS/SIK-1(92)17]. In the benchmark study, VTT [NKS/SIK-1(92)8] and Studsvik [NKS/SIK-1(91)29] simulated the decision making of the safety authorities by applying decision models as an aid.

The benchmark case was the following. At a Swedish boiling water reactor, in the main feedwater system, one of the inner isolation valves of the containment gave an indication of failure to close in a periodical valve closing test. According to the Technical Specifications, either the unit must be shut down or the power must be reduced to 65 %, while one pipe line is kept closed by the outer isolation valve. The experience of indication failures in other valve indications of the same type made the utility to suspect that the failure was in the indication, and not in the operation of the check valve. To avoid the requirement stated by Technical Specifications, the utility applied for an exemption from the rules in order to continue the power operation during the remaining seven weeks to the next annual refuelling outage.

The task of the case study was to simulate the decision making situation from the safety authority side. The decision analysis was made simultaneously and independently by VTT and Studsvik. Both teams were allowed to choose the decision analytic approach freely.

The approach used by VTT is an example of how the decision analysis could be carried out when there is time to discuss the objectives, attributes, value assessments, etc. A decision model was constructed for the prioritization of the decision options. The model construction involved three phases so that the details of the model were gradually increased. First, the decision options were identified, and an objectives hierarchy was formed (see Figure 9). Then the analytic hierarchy process was applied to weight the decision options with respect to the criteria and subcriteria used [16]. Finally, the achievement of objectives was measured quantitatively by means of a multi-attribute value function [17] so that attributes such as increase in core damage frequency were defined for each criterion.

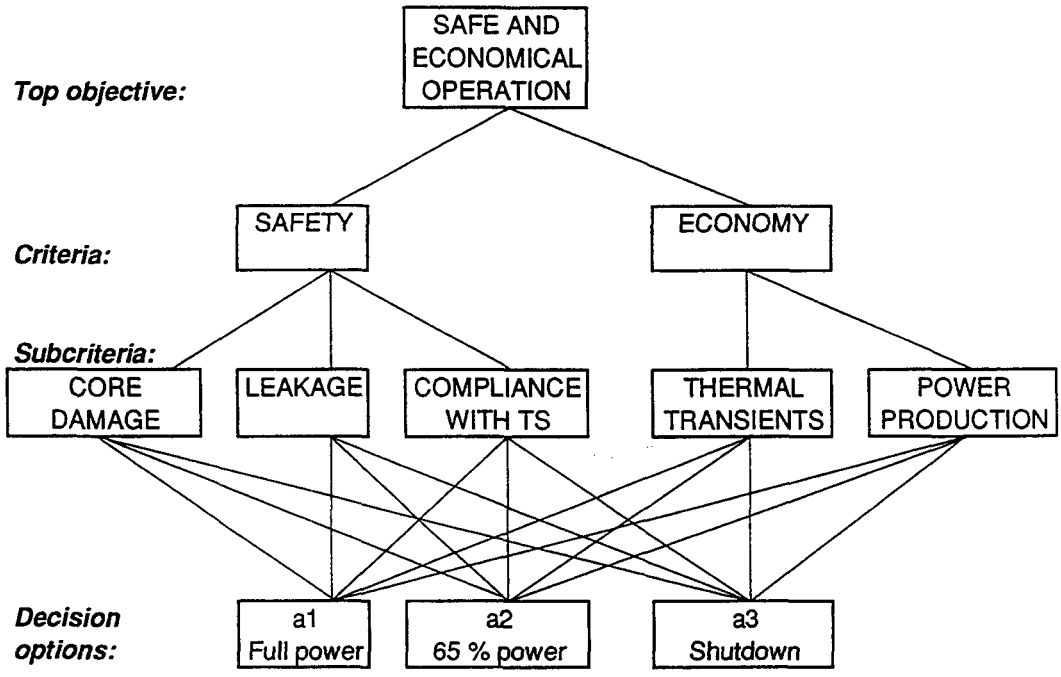

Figure 9. An example objectives hierarchy used in the benchmark study.

The preference order according to the VTT decision model was:

1. to continue the operation at full power (decision "100 %"),
2. to shut down the reactor, and inspect the check valve ("0 %"),
3. to continue the operation at the reduced power level 65 % and to have the outer isolation valve closed ("65 %").

The three decision models of VTT gave slightly different arguments for the preference orders. The attribute of leakage probability had a strong influence on the order. Without the leakage considerations, there would be little doubt on the superiority of the decision option 1.

In the Studsvik approach, the analysis was focused on the safety functions, rather than on the increased risk for core damage. The decision situation was analysed both from the authority and the utility point of view. In both cases, a single-attribute utility function was applied [18].

The Studsvik decision model in the authority case suggests that, with applying a reasonable probability criterion for the operability of the check valve (R_0=0.99), the authority should reject the application for continued operation of the check valve, i.e., the whole plant or the feedwater train should be closed. The utility's decision situation was analysed taking into account the economic risks. The analysis of the utility's decision situation suggests the same preference as VTT above. However, a seemingly small difference between the expected utilities of options "100 %" and "65 %" may be considered insignificant.

The benchmark study illustrated the importance of reliable indications of the positions of valves. In addition to the positional indication of the check valve being judged as unreliable, both teams also concluded that the probability of the check valve being faulty was, in fact, also high.

4.3 Decision analysis procedure

A decision analytic procedure has been outlined [NKS/SIK-1(92)17]. The practical application depends on the analysts and decision makers involved. In a team work, a procedure is needed for carrying out various phases of the problem solving. When the use of the decision models becomes a routine, many of the steps can be copied from earlier decision analyses. The basic conditions required by the decision model, however, should be kept in mind and verified. There should be a computerized decision support system in the management of the analysis process. The procedure is shown in Figure 10.

4.4 Practical needs for decision analysis

Decisions concerning operation of a nuclear power plant can be divided into long-term and short-term decision situations. Long-term decisions are typically related to changes in the plant design, in operating procedures, in maintenance programs and in the Technical Specifications. Formal analysis could provide a method to manage this type of problem solving process. Various activities required for the problem solution can be organized according to the scheme of decision analysis.

The short-term problems belong to situations where time for decision making ranges from one hour to a few days. Such situations are the incident

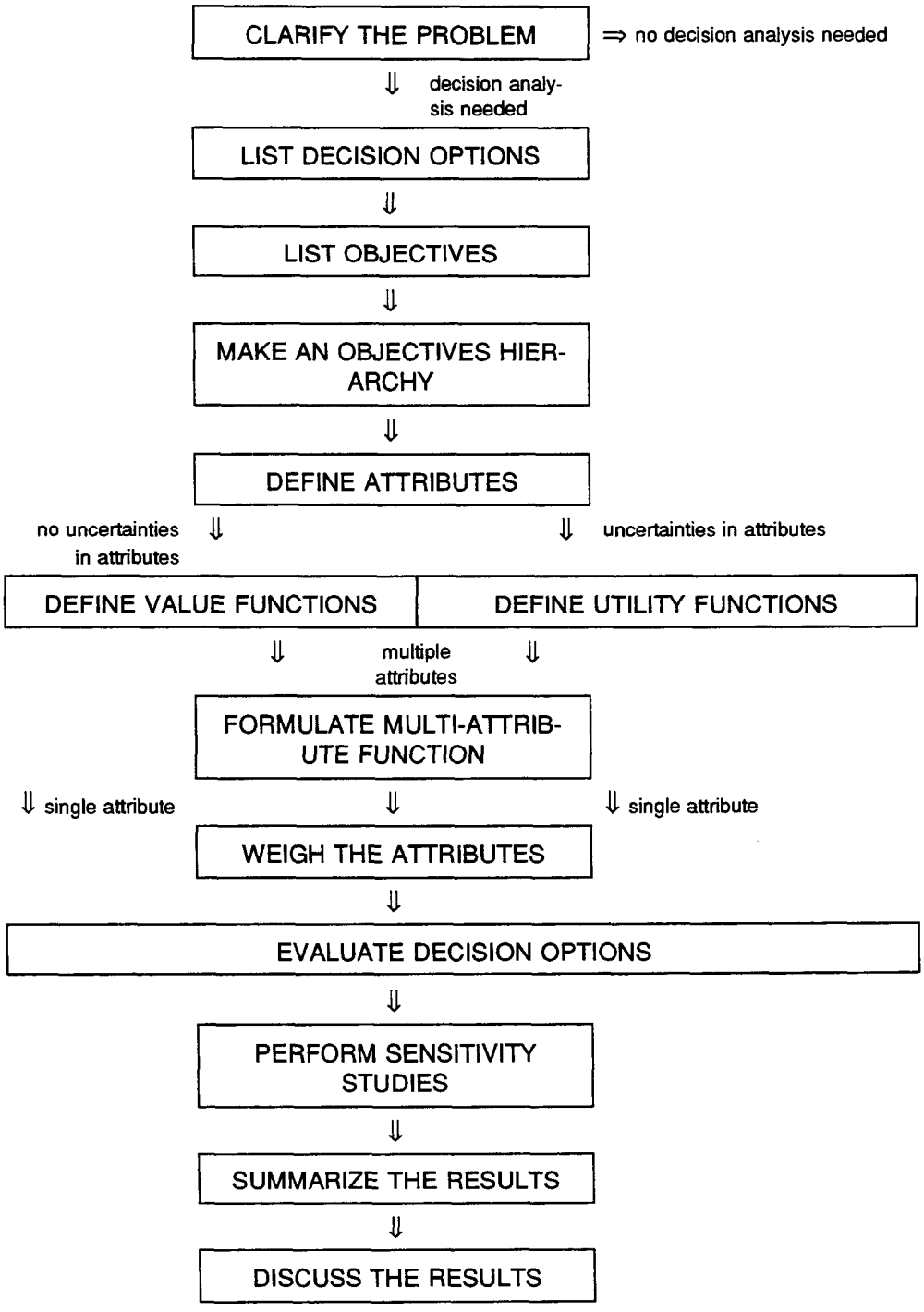

Figure 10. A scheme for a decision analysis.

as well as accident management, and planning of daily operational and maintenance tasks. In short-term problems, the major part of the decision analysis has to be prepared in advance. Probabilistic safety criteria, e.g., maximum allowed instantaneous core damage frequency or increase in the

core damage frequency, form the first hand rules for making risk-based judgements on the decision options.

4.5 Conclusions

The benefit of a decision analysis is that facts and decision options of the problem will be identified and they will be listed for an open discussion. Decision supporting calculations can be developed to a preference model by assessing a hierarchy of objectives and by introducing attributes to express quantitatively the level of achievement of objectives. This emphasizes the quantifiable aspects of the problem with the expense of qualitative aspects such as compliance of the option with the operating procedures or regulatory guides.

A good decision requires sound knowledge and experience of the object, carefully collected and rigorously analysed data and risk-taking attitude of the individuals involved. In connection with PSA, it is realized that a decision under uncertainty should not be based solely on probabilities, particularly, concerning the incompleteness of PSA. The probabilities generated by PSA should be used together with results of the other analyses and direct engineering judgements to support a decision.

A systematic decision analysis is useful if the choice of the decision is not immediately clear. A decision model allows e.g. to perform sensitivity studies which stimulates the decision maker to analyse the problem more closely. The definition of the decision making criteria could be reconsidered, if the decision options are very equal. The establishment of the decision making criteria is, in fact, the main part of the decision analysis.

5 CONCLUSIONS AND REMARKS

5.1 Conclusions on living PSA

What are the purposes and benefits of living PSA?

The objective of living PSA is to bring the use of the plant-specific PSA model out to the daily safety work. This allows feedback and interpretation of operational risk experience and increase of risk awareness of the users of the results. In the project, it has been demonstrated that safety planning of operational activities and risk analysis of operating experience provide means for searching optimal operational, maintenance and testing actions and schedules with regard to their impact on safety. An increased flexibility and effectiveness in operation, maintenance and testing can also be justified in specific cases.

What are the practical results and findings from the case studies?

The case studies showed how a surveillance test series could be planned from the risk point of view in which the same risk level can be maintained although less tests are performed. By shortening the interval for risk efficient tests (and prolonging others) and by optimal staggering of the tests, an improved test scheme was generated.

The allowed down times based on risk planning by living PSA appeared in many cases to be many times longer than according to present Technical Specifications. In certain situations, allowed down times of failed components could be prolonged further by testing a redundant component.

The risk follow-up studies performed for several units showed the applicability and usefulness of the approach. New operational recommendations to enhance safety were discovered, and the results did not always correspond with the operators' opinion of the severity. Direct feedback to verify and revise the PSA models was also obtained.

What are the requirements of a living PSA model?

The PSA model must be able to express the instantaneous alignment and conditions of the safety systems. These conditions change when components are maintained or repaired, or when faults are detected in surveillance tests or when actual demands on safety systems occur via process disturbances. This requires a development of time-dependent basic event models. The basic events are related to evident or hidden conditions of components. One

problem, in this context, is to develop realistic common cause failure models.

The present PSA models and codes do not consistently support time-dependent risk evaluations. It will still take some time before a full scale plant model can be used on-line, meaning that it can give nearly continuous assessment of the plant status and risk level. On the other hand, in many applications for safety planning of operational and maintenance activities, it is sufficient to refine the existing system models to time-dependent models to obtain relevant results.

What are the main gaps in the implementation of living PSA at the plants for safety management and which measures are recommended?

Plant personnel should be better informed about the potential applications of living PSA. Information about applications would offer a practical introduction to PSA for plant staff that has little or very little experience with PSA. It enhances the general level of understanding of the capabilities of probabilistic safety and how it can be used in relation to deterministic safety.

Plant safety management is proposed to consider the use of living PSA for off-line monitoring and risk follow-up of events at the sites. This can be performed by reliability and safety engineers who are more easily trained to use PSA models and to apply results from living PSA. The living PSA applications will require about 0.5—1 man-year per year and unit.

The project proposes for consideration an integration of the living PSA system with other information systems including plant technical documentation to support information retrieval. This would enable interested plant personnel to have access to and to utilize the information stored in PSA.

5.2 Conclusions on safety indicators

What are the purposes and benefits of indicators?

An early warning system for indication of declining safety with the use of indicators has been developed and tried out by a utility participating in the project. An indicator set included in this system is used for steering of safety management of several nuclear power units. The indicators are used for identification of trends and amounts of similar or recurrent operational problems and for comparison of their values with reference values obtained from feedback. The indicators are especially used for early identification of degrading developments but they are also suitable for identification of improvements in performance of an operating nuclear power unit.

The indicators should be used as a group because focusing on a single indicator, or a narrow set of indicators, can be counterproductive to both safety and long-term performance improvement. The project prefers an indicator system covering the fundamental safety principle of defence-in-depth completed with the PSA logic. The use of a well tested and accepted set of safety indicators has the potential to make the feedback of operating experience faster and more problem oriented. The detailed analysis of operating experience can then be directed to the actual problem areas.

How to use trend and pattern analysis to improve the effectiveness and safety relevance of feedback of operating experience?

An incident data base on safety-related occurrences has been upgraded by an authority and used as a tool to reveal potential problem areas in different units. The identification of unit-specific indicators from operating experience generates relevant questions on needs for actual improvements in different observation areas that would otherwise be hidden in the large amount of incident reports and classified data available. The safety-relevance of these candidates for unit-specific indicators were evaluated by using plant-specific PSA to express their significance on overall risk. We recommend for consideration this kind of experience feedback because by the selectivity it will in the long-term save resources in experience feedback and direct the emphasis more on operating experience relevant to safety problems.

What are the benefits of using indicators related to system and equipment performance and maintenance?

It was shown how the availability and condition trends of equipment could be monitored and the maintenance planning enhanced by means of indicators utilizing data from maintenance information systems and selected process data indicating deviations in equipment condition. The feasibility and usefulness of combining various maintenance indicators to plant information systems was subsequently demonstrated. In addition, the presentation of the condition of the components as a function of time helps the maintenance personnel to earlier detect developing faults. It thus supports the planning of maintenance actions to avoid functional failures or damage.

What are the main gaps in the implementation of an effective system for utilization of indicators at the plants?

An effective indicator system requires a well organized and user friendly information system and motivated staff. All involved parties should have an easy access to the indicator compilation process and the underlying data base. That involves the experience feedback loop from event reporting up

to and including the actions at the unit. This would be facilitated by testing of an indicator set with regard to availability of data, calculation and usefulness at the plants. The verified indicator sets could then be included in the plant information systems and could also be connected to the head-quarters.

5.3 Conclusions on decision analysis

What are the purposes and needs of decision analysis?

The use of decision analysis supports the identification of decisive facts and decision options. It enhances the open discussion on disputed issues. Decision analysis helps in structuring and weighting the decision criteria and choosing from options. By a systematic approach, the consistency and transparency of decision making can be improved. A good decision requires sound knowledge and experience of the object, carefully collected and rigorously analysed data and formulation of the risk-taking attitude of the individuals involved.

A decision analysis can be especially useful when the problem is not clear, and the best solution is not obvious. This applies often to e.g. modifications of technical solutions or technical specifications for safe operation in a plant or to temporary exemptions from the technical specifications.

How to use PSA in decision making on safety related issues?

In connection with PSA, it is realized that a decision under uncertainty should not be based solely on probabilities, particularly when the unwanted event is a rare one and its probability of occurrence is estimated by means of different kinds of approximations. PSA, at least level 1 PSA, can only support partly the decision making process. A proper use of the living PSA applications requires that decision making criteria are established. Probabilistic criteria only are not sufficient in complex decision making situations. The probabilities generated by PSA should be used together with results of other analyses and engineering judgement to support a decision when using a systematic decision analysis process.

5.4 Future plans

Development work is needed for the practical implementation of ideas demonstrated and proposed in this project. This includes development of the living PSA interface applicable for non-experts of PSA and definition of procedures for utilization of safety indicators. These safety management methods set new demands on current decision making routines, so that decision making processes should be studied as well as decision making

procedures should be developed for various situations. Methodologically, some areas still remain to be improved in PSA, such as treatment of uncertainties, common cause failures, human reliability, external events, analysis of other than full power operation states, level 2 PSA etc.

In the future, ageing questions should be considered more systematically. Living PSA, safety indicators and analysis of operating experience are important parts of the plant life management. In addition, the utilities will pay more attention to planning and updating of preventive maintenance, including condition monitoring, to become more effective. A balance between a safe, economic and effective maintenance program cannot be found, justified and verified without combined use of systematic analysis of operating experience and probabilistic methods. In addition, the utilities will continuously consider the possibility to modernize and to apply new technological solutions such as digital control and automation systems in the old plants. Consequently, methods to evaluate the reliability of the new techniques must be developed as an input for the decision making.

A better cooperation between the research of human factors, PSA, safety and maintenance indicators as well as decision analysis is needed for the improvement of the function of the organisations. It has been noticed in the analyses of organisational factors that a functional orientation and conceptual skills increase the personnel's ability to understand and to control complex problems. Therefore, the methods developed and proposed in this project play an important role in the development of new organisational concepts, management methods and the personnel's skills.

5.5 Concluding remarks

In the project, it has been demonstrated that the use of living PSA, safety indicators and decision analysis have an important potential in safety management and in decision making regarding risk levels. Further practical implementation is needed, however. The use of these methods could be promoted by training of the staff concerned. Systematic safety and risk awareness would then become more clearly focused at the plants and authorities. The project recommends for consideration that living PSA as well as safety and maintenance indicators would become parts of the plant technical documentation and information systems. An improved selectivity of the experience feedback saves resources and puts more emphasis on the analysis of safety related operating experience.

NKS/SIK-1 PUBLICATIONS AND REPORTS

NKS/SIK-1(90)5. Mankamo, T., Pörn., K. & Holmberg, J. Uses of risk important measures. Espoo 1991, Technical Research Centre of Finland. VTT Research Notes 1245. 36 p. + app. 8 p.

NKS/SIK-1(90)6. Holmberg, J., Laakso, K., Lehtinen, E. & Pulkkinen, U. Ongoing activities of Technical Research Centre of Finland for nuclear power plants operational safety assessment and management. Proceedings of the 2nd TÜV-Workshop on Living-PSA-Application, Hamburg, May 7-8, 1990. Report VTT/SÄH 14/90, RISKI(90)1. 19 p. + app. 1 p.

NKS/SIK-1(90)7. Laakso, K., Engqvist, A., Knochenhauer, M., Kosonen, M., Liwång, B., Mankamo, T. & Pörn, K. Optimization of technical specifications by use of probabilistic methods. A nordic perspective. Proceedings of the IAEA Technical Committee Meeting on the Use of Probabilistic Safety Analysis to Evaluate Nuclear Power Plant's Technical Specifications, Vienna, June 18-22, 1990. Research report VTT/SÄH 20/90. 15 p.

NKS/SIK-1(90)8. Laakso, K., Johanson, G., Björe, S., Virolainen, R. & Gunsell, L. Safety evaluation by use of living PSA and safety indicators. Work plan 1990—1993. Espoo 1990. 21 p.

NKS/SIK(90)10. Holmberg, J., Laakso, K., Lehtinen, E., Johanson, G. & Björe, S. International survey of living PSA and safety indicators. Espoo 1992, Technical Research Centre of Finland, VTT Research Notes 1326. Report RISKI(90)2. 51 p. + app.

NKS/SIK-1(90)11. Mankamo, T. & Kosonen, M. Operational decision alternatives in failure situations of standby safety systems. Proceedings of the IAEA Technical Committee Meeting on the Use of Probabilistic Safety Assessment to Evaluate Nuclear Power Plant's Technical Specifications, Vienna, June 18-22, 1990. IAEA-TECDOC-599. Report RISKI(91)15. 20 p.

NKS/SIK-1(90)12. Laakso, K., Engqvist, A., Knochenhauer, M., Kosonen, M., Liwång, B., Mankamo, T. & Pörn, K. Optimizing tech specs by probabilistic methods - a nordic perspective. Nuclear europe worldscan, 1990. No 5/6, p. 20.

NKS/SIK-1(90)13. Holmberg, J., Laakso, K., Lehtinen, E., Johanson, G. & Björe, S. Nordic survey on safety evaluation by use of living PSA and safety indicators (NKS/SIK-1). Stockholm 1991, Statens Kärnkraftinspektion. SKI technical report 91:3, 28 p. + app. 16 p.

NKS/SIK-1(90)23. Lehtinen, E. & Saarelainen, P. Monitoring of safety system unavailability and maintenance performance using LOTI information system and operational safety indicators. A trial study for diesel generators of Loviisa 2 plant. In: Proc. of the IAEA Technical Committee Meeting on Exchange of Experience in Managing Nuclear Power Plant Safety Using Numerical Indicators, Vienna 26-28, November 1990. Report RISKI(90)10. 11 p. + app. 3 p.

NKS/SIK-1(91)4. Mankamo, T. & Kosonen, M. Continued plant operation versus shutdown in failure situations of standby safety systems - application of risk analysis methods for the evaluation and balancing of allowed outage times for the residual heat removal systems at TVO I/II plant. Espoo 1992, Avaplan Oy. Report RISKI(91)16. 100 p.

NKS/SIK-1(91)6. Johanson, G. Survey of time dependences in LPSA models. Stockholm, 1991, Statens Kärnkraftinspektion. 14 p.

NKS/SIK-1(91)7. Johanson, G., Gunsell, L., Laakso, K. & Hellström, P. Safety evaluation by use of living PSA and safety indicators. Current status and future development of models and tools within the Nordic project "Safety Evaluation, NKS/SIK-1". In: Use of probabilistic safety assessment for operational safety, Proc. of an international symporium, Vienna, 3-7 June, 1991. Vienna, 1992, International Atomic Energy Agency, Pp. 659—676.

NKS/SIK-1(91)8. Holmberg, J. & Himanen, R. Uncertainty study in probabilistic risk Assessment for TVO 1/2 nuclear power plant. In: Proc. of the OECD/BMU-workshop on special issues of level 1 PSA, Cologne, May 27—29, 1991. Gesellschaft für Reaktorsicherheit. Pp. 148—160.

NKS/SIK-1(91)20. Nyman, R. & Angner, A. Stagbas II — an incident reporting database for safety related occurrences and reactor trips in Swedish nuclear power plants. In: Malmén Y. & Rouhiainen V. (ed.) Reliability and safety of processes and manufacturing systems. Proc. of the SRE-Symposium 1991, Tampere, October 1—3, 1991. Pp. 101—110.

NKS/SIK-1(91)23. Pörn, K. & Shen, K. Integrated uncertainty analysis in PSA. Nyköping 1991, Studsvik Ecosafe Ab. Report STUDSVIK/NS-91/71. 37 p.

NKS/SIK-1(91)27. Holmberg, J., Pulkkinen, U., Laakso, K. & Mankamo, T. The risk follow-up by PSA – report of the Finnish pilot study. Espoo 1992, Technical Research Centre of Finland. Report VTT/SÄH 14/91, RISKI(91)4. 18 p. + app. 2 p.

NKS/SIK-1(91)29. Pörn, K. & Shen, K. Decision making under uncertainty – A pilot study on exemption from technical specification. Nyköping 1992, Studsvik Ecosafe Ab. STUDSVIK/NS-91/90. 30 p. + app. 15 p.

NKS/SIK-1(91)30. Erhardsson, U.-K. & Flodin, Y. Momentan risknivå. Fud-projekt inom levande PSA. (Instantaneous risk level). Vällingby 1991, Vattenfall Ab. 20 p. (In Swedish)

NKS/SIK-1(91)33. Stokke, E. Operational interface for LPSA. Halden 1993, IFE Halden. (draft)

NKS/SIK-1(91)35. Holmberg, J., Pulkkinen, U. & Mankamo, T. Risk follow-up by PSA. In: Proc. of the IAEA Technical Committee Meeting on Guidelines on Probabilistic Safety Assessment (PSA) Requirements for Use in Safety Management, Stockholm 16–20 September 1991. Report VTT/SÄH 16/91, RISKI(91)9. 13 p. + app. 2 p.

NKS/SIK-1(91)36. Angner, A. & Nyman, R. STAGBAS-Incident katalog. Sundbyberg 1990, Relcon Ab. Report RELCON 10/90, 80 p.

NKS/SIK-1(91)38. Holmberg, J., Johanson, G. & Niemelä, I. Risk measures in living probabilistic safety assessment. Espoo 1993, Technical Research Centre of Finland. VTT publications 146, 59 p. + app. 8 p.

NKS/SIK-1(91)40. Holmberg, J. A limited survey on the ASP methodology. Espoo 1991, Technical Research Centre of Finland. Report VTT/SÄH 18/91, RISKI(91)40. 11 p.

NKS/SIK-1(91)47. Paulsen, J.L. & Liisberg, C. Expert systems and plant conditions. Proc. of the World Congress of Expert Systems, 16–19 December 1991. 15 p.

NKS/SIK-1(91)48. Mankamo, T. Timedependent models/risk monitor exercise. TVO I/II SRV CCF Qauntifications. Espoo 1993, Avaplan Oy. Work notes. 25 p. (draft)

NKS/SIK-1(92)2. Holmberg, J. & Johanson, G. Definition of a concept for safety evaluation by use of living PSA – the Nordic project "safety evaluation, NKS/SIK-1". In: Petersen, K. & Rasmussen, B. (ed.) Safety and reliability '92. Proc. of the European safety and reliability conference '92 (ESRC '92), Copenhagen, June 10–12, 1992. London 1992, Elsevier. Pp. 995–1006.

NKS/SIK-1(92)3. Mankamo, T. Extended common load model. A tool for dependent failure modelling in highly redundant structures. Espoo 1990, Avaplan Oy. Publication manuscript. 26 p.

NKS/SIK-1(92)5. Lehtinen, E. & Saarelainen, P. Monitoring of a safety system's unavailability and maintenance performance using LOTI information system and operational safety indicators. In: Petersen, K. & Rasmussen, B. (ed.) Safety and reliability '92. Proc. of the European safety and reliability conference '92 (ESRC '92), Copenhagen, June 10–12, 1992. London 1992, Elsevier. Pp. 261–272.

NKS/SIK-1(92)6. Laakso, K., Lehtinen, E., Erikson, H., Rollenhagen, C., Nyman, R., Angner, A. Development of operational safety indicators for use in experience feedback - a part of the Nordic NKS/SIK-1 project. In: Petersen, K. & Rasmussen, B. (ed.) Safety and reliability '92. Proc. of the European safety and reliability conference '92 (ESRC '92), Copenhagen, June 10–12, 1992. London 1992, Elsevier. Pp. 261–272.

NKS/SIK-1(92)7. Holmberg, J., Johanson, G. & Sandstedt, J. The generation of probabilistic safety indicators from the risk follow-up results. In: Balfanz, H.-P. (ed.) Proc. of the 3rd workshop on living-PSA-application, Hamburg, May 11–12, 1992. TÜV-Norddeutschland, Hamburg, 1992. 15 p.

NKS/SIK-1(92)8. Holmberg, J. & Pulkkinen, U. Decision analysis on an exemption from the technical specification. Espoo 1992, Technical Research Centre of Finland. Report VTT/SÄH 4/92, RISKI(92)1. 19 p. + app. 16 p.

NKS/SIK-1(92)10. Techdoc on risk-based application of nuclear power plant. Technical specification improvements. Working material of the IAEA consultants' meeting in Vienna, 21 August - 4 September, 1992. Reproduced by the IAEA, Vienna 1992. Report IAEA-J4-CS53/92. 44 p.

NKS/SIK-1(92)11. Erikson, H. Trend analysis of LER:s and reactor trips at the Forsmark nuclear power plant. Sundbyberg 1992, Vattenfall. Report PT-29/92. 4 p. + app. 4 p.

NKS/SIK-1(92)12. Pörn, K. & Shen, K. On the integrated uncertainty analysis in probabilistic safety assessment. Nyköping 1992, Studsvik Ab. 13 p.

NKS/SIK-1(92)13. Mankamo, T. A time-dependent model of dependent failures — Application to a pairwise symmetric structure of four components. Espoo 1993, Avaplan Oy. 31 p. (draft)

NKS/SIK-1(92)17. Holmberg, J., Pulkkinen, U., Pörn, K. & Shen, K. Risk decision making in operational safety management - experience from the Nordic benchmark study. Nyköping 1993, Studsvik EcoSafe. Report STUDSVIK/ES-93/37, RISKI(93)1. 19 p. + app. 6 p.

NKS/SIK-1(92)18. Paulsen, J.L. & Clementz, M. Pilot study on maintenance indicators. Copenhagen 1992, Risö National Laboratory. Report RAG-02732-90-33. 17 p.

NKS/SIK-1(92)19. Paulsen, J.L. Plant condition and maintenance indicators. Proc. of the Euro-maintenance 92, Lissabon, 1-3 June, 1992. Report Jr.No.02732-90-31. 9 p.

NKS/SIK-1(92)20. Johanson, G. & Holmberg, J. The use of living PSA in safety management, a procedure developed in the Nordic project "Safety Evaluation, NKS/SIK-1". In: Proc. of the Probabilistic Safety Assessment International Topical Meeting in Florida, January 27-29, 1993. 11 p.

NKS/SIK-1(92)22. Sandstedt, J. & Berg, U. Living PSA applications for a Swedish BWR with the aid of Risk Spectrum. In: Balfanz, H.-P. (ed.) Proc. of the 3rd workshop on living-PSA-application, Hamburg, May 11-12,1992. Hamburg 1992, TÜV-Norddeutschland. 25 p.

NKS/SIK-1(92)27. Sandstedt, J. Demonstration case studies on living PSA. Stockholm 1993, Swedish Nuclear Power Inspectorate. SKI Technical Report 93:33. 25 p.

NKS/SIK-1(92)28. Nyman, R. & Angner, A. Stagbas 2 — Incidentkatalog Oskarshamn 2. Stockholm 1992, SKI. SKI/RA report - 028/91, technical report 92:4, 38 p.

NKS/SIK-1(92)30. Jönsson, J. & Sandin, P. Underhållsindikatorer. Maintenance indicators. Sydkraft report (Draft). Malmö 1992. 36 p. (In Swedish)

NKS/SIK-1(92)32. Simola, K. Trial application of multiple correspondence analysis to nuclear power plant incident data. Espoo 1992, Technical Research Centre of Finland. Report RISKI(92)11. 7 p. + app. 13 p.

NKS/SIK-1(92)34. Pyy, P., Laakso, K., Hänninen, S. & Simola, K. Maintenance and safety - experience from some finnish studies. In: Proc. of the 3rd ESReDA seminar on equipment and ageing and maintenance in Chamonix, October 14—15, 1992. Report RISKI(92)14. 13 p.

NKS/SIK-1(92)35. Mankamo, T. A pressure relief transient with pilot valve function affected by a latent CCF mechanism. Espoo 1994, Avaplan Oy. 34 p. + app.

NKS/SIK-1(92)39. Johanson, G., Holmberg, J. & Sandstedt, J. Living PSA application for a Swedish BWR. In: Kafka, P. & Wolf, J. (ed.) safety and reliability assessment - an integral approach proc. of the European Safety and Reliability Conference, München, May 10-12, 1993. Elsevier, Amsterdam. Pp. 455-465.

NKS/SIK-1(92)40. Nyman, R. & Angner, A. Stagbas 2 — Incidentkatalog Forsmark 1. Stockholm 1992, SKI. Ski/RA report - 24/91, SKI technical report 92:9. 41 p. (In Swedish)

NKS/SIK-1(92)41. Nyman, R. & Angner, A. Stagbas 2 — Incidentkatalog Forsmark 3. Stockholm 1992, SKI. Ski/RA report - 26/91, SKI technical report 92:11. 37 p. (In Swedish)

NKS/SIK-1(92)42. Nyman, R. & Angner, A. Stagbas 2 — Incidentkatalog Oskarshamn 1. Stockholm 1992, SKI. Ski/RA report - 27/91, SKI technical report 92:05. 44 p. (In Swedish)

NKS/SIK-1(92)43. Nyman, R. & Angner, A. Stagbas 2 — Incidentkatalog Oskarshamn 3. Stockholm 1992, SKI. Ski/RA report - 29/91, SKI technical report 92:06. 47 p. (In Swedish)

NKS/SIK-1(92)44. Nyman, R. & Angner, A. The Stagbas database and the production of the incident catalogue and trend catalogue. In: Kafka, P. & Wolf, J. (ed.) safety and reliability assessment — an integral approach proc. of the European Safety and Reliability Conference, München, May 10-12, 1993. Elsevier, Amsterdam. Pp. 121—133.

NKS/SIK-1(92)49. Sandstedt, J. Betydelseanalys och känlighetsanalys av O2-indikatorer FAS 1 och 2 (Importance and sensitivity analysis of O2-indicators phase 1 and 2). Stockholm 1993, SKI. Ski/RA report - 4/93. (In Swedish)

NKS/SIK-1(93)2. Lehtinen, E. A concept of safety indicator system for nuclear power plants. Espoo 1994, Technical Research Centre of Finland. (Draft)

NKS/SIK-1(93)3. Holmberg, J., Lehtinen, E. & Laakso, K. Safety management supported by living probabilistic safety assessment (PSA) operational safety indicators. Presented in Workshop on Integrated Risk Management for Large Industrial Complexes and Energy Production Systems, Moscow, February 22–26, 1993. Espoo 1993, Technical Research Centre of Finland. Report RISKI(93)2. 16 p.

NKS/SIK-1(93)4. Mankamo, T. A risk-based approach to AOTs. Espoo 1994, Avaplan Oy. 28 p + app. (draft)

NKS/SIK-1(93)5. Holmberg, J. Operating experience feedback in probabilistic safety assessment. Espoo 1993, Helsinki University of Technology. Licentiate thesis. 99 p.

NKS/SIK-1(93)7. Nyman, R. & Angner, A. BWR generation 2. Barsebäck 1, Barsebäck 2, Oskarshamn 2. SKI/RA report - 1/93, SKI technical report 93:01. 48 p. (In Swedish)

NKS/SIK-1(93)10. Paulsen, J.L. Concepts and design of MMI systems for process plants. Roskilde 1993, Risö National Laboratory. Report Risö-1-688(EN). 15 p.

NKS/SIK-1(93)11. Nyman, R. & Angner, A. Stagbas 2 — Incidentkatalog Ringhals 1. Stockholm 1993, SKI. Ski/RA report - 30/91, SKI technical report 93:16. 57 p. (In Swedish)

NKS/SIK-1(93)12. Nyman, R. & Angner, A. Stagbas 2 — Incidentkatalog PWR annläggningar. Stockholm 1993, SKI. Ski/RA report - 2/93, SKI technical report 93:10.

NKS/SIK-1(93)13. Lehtinen, E., Holmberg, J., Laakso, K & Hoffström, A. Use of living PSA and safety indicators to support operational safety management. Presented in the IFAC symposium Safeprocess '94, Espoo, June 13-15, 1994. Report RISKI(93)11. 6 p.

NKS/SIK-1(93)16. Johanson, G. & Holmberg, J. (eds.) Safety evaluation by living PSA — Procedures and applications for living probabilistic safety assessment in risk planning of operational activities and risk analysis of operating experience. To be published in SKI Technical Report Series, Stockholm 1994. 105 p.

NKS/SIK-1(93)17. Holmberg, J. & Pyy, P. Analysis of an external pipe break. Espoo 1993, Technical Research Centre of Finland. Report RISKI(93)11. 12 p. + app. (draft)

NKS/SIK-1(93)20. Nyman, R. & Angner, A. BWR generation 3 Forsmark 1, Forsmark 2 och BWR generellt. Stockholm 1993, SKI. Ski/RA report 3/93, SKI technical report. 66 p. + app. 7 p. (In Swedish)

NKS/SIK-1(93)21. Holmberg, J., Pulkkinen, U., Pörn, K. & Shen, K. Risk decision making in operational safety management — experience from the Nordic benchmark study. Espoo 1993, Technical Research Centre of Finland. 16 p. (to appear in Risk Analysis)

NKS/SIK-1(93)23. Lehtinen, E.A., Holmberg, J.-E. and Laakso, K.J. Safety indicators — A tool for self-assessment of nuclear power plant's operational safety. Presented in SRA-Europe 4th conference, Rome, October 18–20, 1993. 1993. 7 p.

NKS/SIK-1(93)26. Sandstedt, J. Förstudie. Analys av föreskrivna testinterval Oskarshamn 2 (Analysis of prescribed test intervals Oskarshamn 2). Sundbyberg 1993, Relcon Ab. Report Relcon -12/93. 32 p. (In Swedish)

NKS/SIK-1(93)30. Johanson, G., Berglund, L., Holmberg, J. and Karlsson, C. Regulatory decision making: Test of qualitative and quantitative decision criteria for improvements in Technical Specifications. In Proc. of IAEA Technical Committee Meeting on "Procedures for use of PSA for optimizing nuclear power plant operational limits and conditions", Barcelona, September 20–23, 1993, (IAEA-J4-TC-855). Report RISKI(93)17. 12 p.

NKS/SIK-1(93)31. Knochenhauer, M. and Johanson, G. Derivation of time dependent component unavailability models and application to Nordic PSA:s. To be presented in PSAM II conference, San Diego, March 20–24, 1994. 1993. 6 p.

NKS/SIK-1(93)32. Simola, K. Application of correspondence analysis to nuclear power plant incident data. Presented at SRE Seminarium, Malmö, November, 1993. 9 p.

NKS/SIK-1(93)34. Nyman, R. (ed.) NKS/SIK-1 OSI seminarium på SKI 1993-12-07. Seminariekompendium. (Safety indicator seminar). Report SKI/RA-001/94. Stockholm 1994. 150 p.

NKS/SIK-1(93)36. Lehtinen, E., Pulkkinen, U., Pörn, K. & Simola, K. Statistical trend analysis methods for temporal phenomena. Espoo 1994, VTT Automation. (draft)

REFERENCES

1. Pershagen, B. (ed.) Nordic studies in reactor safety. Final report of the Nordic Research Programme SIK. Roskilde, 1994, NKS. Report TemaNord 1994:544. 114 p.
2. Bonaca, M.V. (ed.) Living probabilistic safety assessment for nuclear power plant management. Paris 1992, OECD/Nuclear Energy Agency, OECD Publications. 81 p.
3. T-book. Reliability data of components in Nordic nuclear power plants. 3rd ed. Prepared by the ATV Office and Studsvik AB. Vällingby 1992, The ATV Office, Vattenfall AB. 235 p.
4. WANO\IG19.1. WANO performance indicator programme implementation guideline. WANO. 6 April 1990. 4 p. + app.
5. IAEA-TECDOC-600. Numerical indicators of nuclear power plant safety performance. IAEA. Vienna, 1991. 70 p.
6. Marcus, A.A. et al. Organization and safety in nuclear power plants. USNRC. NUREG/CR-5437. May 1990. 165 p. + app.
7. Olson, J. et al. Development of programmatic performance indicators. NUREG/CR-5241. October 1988. 178 p.
8. IAEA Safety Series No. 75-INSAG-3. Basic safety principles for nuclear power plants. A report by the international nuclear safety advisory group. IAEA. Vienna, 1988. 74 p.
9. IAEA safety series No. 75-INSAG-4. Safety culture. A report by the international nuclear safety advisory group. IAEA.Vienna, 1991. 31 p.
10. Anon. IAEA-J.4. CS 55. Development of risk-based safety indicators. System unavailability indicators. Report of a consultant's meeting organized by the International Atomic Energy Agency and held in Vienna, 3-7 July, 1989. Vienna 1989. Working material reproduced by the IAEA (limited distribution). 83 p.
11. Huovinen, T. Estimation of some stochastic models used in reliability engineering. VTT research reports 598. Espoo 1989. 144 p.
12. Laakso, K. A systematic feedback of plant disturbance experience. Thesis. Helsinki University of Technology. 1984. 150 p.
13. UNIPEDE. Detailed descriptions of international nuclear power plant performance indicators. June 1991. 119 p.
14. Wahlström, B., Laakso, K., Lehtinen, E. Feedback of experience for avoiding a low probability disaster. IAEA-SM-302/23. Vienna 1989. pp. 251 - 262.
15. Lehtinen, E., Himanen, R., Viitasaari, O. Experiences from the developing of performance indicators for the Finnish nuclear power plants. Transactions of the international conference on availability improvements in nuclear power plants. Madrid. April 10-14, 1989. Sociedad Nuclear Espanola. pp. 22-23.
16. Saaty, T. The analytic hierarchy process. New York 1980, McGraw-Hill. 287 p.
17. Fishburn, P.C. Decision and value theory. New York 1964, John Wiley. 451 p.
18. von Neumann, J. & Morgenstern, O. Theory of games and economic behaviour, 2nd ed. Princeton, New Jersey 1947, Princeton University Press. 641 p.

ABBREVIATIONS

AHP	analytic hierarchy process
AOT	allowed outage time (down time) of safety related equipment
ASAR	As operated Safety Analysis Report program, Sweden
BWR	boiling water reactor
CCF	common cause failure
CM	corrective maintenance
DG	diesel generator
IAEA	International Atomic Energy Agency
IFE	Institutt for energiteknikk, Norway
INES	The International Nuclear Event Scale
IVO	Imatran Voima Oy, Finland
LCO	limiting conditions for operation
LOCA	loss of coolant accident
LOTI	Loviisan voimalaitoksen tietojärjestelmä, Loviisa power plant information system, Finland
NKS	Nordic nuclear safety research
PM	preventive maintenance
PSA	probabilistic safety assessment
PWR	pressurized water reactor
SIK	Nordic research program on reactor safety, 1990–93
SKI	Statens Kärnkraftinspektion, Swedish Nuclear Power Inspectorate
STAGBAS	störningsanalys grupp databas, database of the disturbance analysis group, Sweden
STI	surveillance testing interval
STUK	Säteilyturvakeskus, The Finnish Centre for Radiation and Nuclear Safety
TS	Technical Specifications
TUD	Tillförlitlighet, Underhåll och Drift, reliability data base, Sweden
TVO	Teollisuuden Voima Oy, Industrial power company, Finland
VTT	Valtion teknillinen tutkimuskeskus, Technical Research Centre of Finland
WANO	World Association of Nuclear Operators

TERMS

Allowed down time. This stipulates the maximum allowed down time (allowed outage time, AOT) for an equipment in a safety system. The unit must usually be placed to a safer operational state, if the operability of the faulty equipment is not reached within its AOT.

Accident. The safety significance of events is classified on the International Nuclear Event Scale (INES) into 7 levels. Events are considered in terms of three criteria on off-site impact, on-site impact and defence-in-depth degradation. The upper levels (4–7) of events are termed accidents. Accidents relate only to nuclear or radiological safety. Partial or severe damages of the reactor core, acute health effects to workers or public releases of the order of prescribed limits or more are termed as accidents.

Basic event. A reliability analysis can be carried out down to a component failure mode or human error level where sufficiently reliable experience data can be obtained. The occurrences, included in a reliability model, at the most detailed level are called basic events. Basic events are related to *evident* or *hidden* unavailabilities of components. Examples of evident unavailabilities are maintenances and repairs of the components. The related basic events are true or false. *Hidden* unavailabilities may be detected only in tests or demand situations. Related basic events may be modelled by time-dependent unavailability models.

Common cause failure. Common cause failures (CCF) are failure causes or mechanisms which result in multiple failures in redundant components. CCF basic events are usually added in the PSA model to cover residual, not explicitly identified dependences between redundant components.

Defence-in-depth principle. The fundamental safety strategy of defence-in-depth is implemented to compensate for human and mechanical failures. This concept is centered on several levels of protection to assure the integrity of the physical barriers preventing the release of radioactive material to the environment. The integrity of physical barriers of the current reactor plants in the Nordic countries is maintained by the application of the defence-in-depth principle in all stages of reactor design, construction and operation. Principal emphasis is placed on averting the damage to the reactor, and to the physical barriers themselves, by application of levels of protection as safety barriers such as control of operation, safety functions and safety management.

Incident. The lower levels (1—3) of events are termed incidents according to INES. The defence-in-depth considerations classify incidents as levels 3 through 1. The deviations from the limiting conditions for operation are examples of level 1 incidents, also termed anomalies. The most events reported to the safety authorities are classified as below the scale, i.e. level 0. The licensee event reports on unavailabilities in safety systems and reactor trip reports fall mostly on the 0-level of the INES scale. (see also Accident)

Inherent core damage frequency. The inherent core damage frequency according to PSA represents the plant configuration where no component is unavailable due to maintenance or repair events, and all standby components have been recently tested without any failure indications. It represents the "lowest theoretically achievable" core damage frequency with current design to be used as a reference level.

Initiating event. An initiating event or initiator is a disturbance in the normal (power) operation which requires actions by the plant protection and safety systems. Initiating events are divided into several categories depending on the required plant responses. Main categories are loss of coolant accidents (LOCA) of various leakage sizes and process transients such as a loss of the main feedwater system, and external events such as loss of off-site power.

Instantaneous core damage frequency. The instantaneous core damage frequency according to PSA corresponds to using a PSA model with basic events modelled based on knowledge about conditions of systems and components of the plant. The component or system concerned is presented in the model by evident events (true or false) and by hidden events (time-dependent unavailability model). If the evident unavailability caused by maintenance and repair is excluded, then an instantaneous baseline core damage frequency is obtained.

Limiting conditions for operation. The limiting conditions (LCO) for operation are rules to be followed in order to maintain the plant operation within the bounds of safety analysis. The LCOs specify requirements on the number of subsystems operable at different operational states and the allowed down times for equipment. These operational rules shall assure that safety systems are either ready for use or functioning on real demands, i.e. plant transients and accidents. The action statements require the plant to be brought into a safer operational state, often cold shutdown, if faulty equipment cannot be restored with its allowed down time (AOT).

Minimal cut set. A cut set is a combination basic events, e.g. component failures, leading to the unwanted state considered, in PSA e.g. core damage. This cut set is called minimal cut set, if the intended system function can be achieved by elimination of a single basic event only.

Nominal core damage frequency. The nominal core damage frequency according to PSA obtained by the use of nominal or time-average failure probabilities for component and system failures as well as for operator errors and by the use of nominal initiating event frequencies. If the evident unavailabilities caused by maintenance and repair are excluded, then a (nominal) baseline core damage frequency is obtained. The nominal core damage frequency is used in long term risk planning.

Performance indicator. Performance indicators are measurable parameters providing a quantitative means of monitoring the effectiveness of the spectrum of activities at commercial nuclear plants that can influence safety.

Physical barrier. The technical means of achieving reactor safety is to confine the radioactive material by multiple physical barriers. The four physical barriers are the fuel matrix, the fuel cladding, the boundary of the primary coolant system and the containment.

Probabilistic safety indicator. The results of a risk follow-up by PSA provide probabilistic safety indicators from the operating experience. Examples of probabilistic safety indicators are

average core damage frequency during an observed period, risk doses and number of incidents exceeding a probability or frequency criterion.

Risk. The combination of the probability and the consequence of a specified hazardous event such as core damage.

Risk assessment. Risk assessment is the basic evaluation approach with a risk model. The aim of risk assessment is to calculate the nominal core damage frequency of the plant and related risk measures. The results can be used to the identification of risk contributors and to long-term risk planning.

Risk dose. Risk dose is the retrospectively calculated core damage probability of an incident or the core damage probability over the examined operating period.

Risk follow-up. The aim of risk follow-up is to calculate the risk doses and related risk measures based on the evaluation of operating experience.

Risk measures. Risk measures are means to present the results of various applications of PSA in a form of information, which is suitable for making conclusions. The basic risk measures are nominal, inherent and instantaneous core damage frequency. Generated risk measures such as risk importance measures are used in applications.

Risk monitoring. Risk monitoring has a short-term or an on-line evaluation perspective. The aim is to calculate the instantaneous core damage frequency of current or currently planned plant configuration.

Safety barrier. The safety barriers are levels of protection and defences to assure the integrity of the physical barriers. (see also Defence-in-depth principle, Physical barrier)

Safety culture. Safety culture is that assembly of characteristics and attitudes in organization and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance. A good safety culture is dependent on the orientation and commitment mechanisms of the personnel as well as the management to the daily work and developmental activities.

Safety function. The three basic safety functions of a nuclear power unit are controlling the power, cooling the fuel and confining the radioactive material. In probabilistic safety assessments of the Nordic BWRs, the cooling of fuel is divided into the short-term cooling water supply and long-term residual heat removal due to the different time frames available for recovery prior to the occurrence of core damage. The successes and failures of safety functions are represented in the event trees of the PSAs.

Safety indicator. Safety indicator is an observable characteristics of an operational nuclear power unit, presumed to bear a positive correlation with the safety of the reactor. The safety indicators have been selected, among other means, for the purpose of supervision of safety. The safety indicators can be related to defence lines according to defence-in-depth such as physical barriers and safety functions.

Safety related occurrence. The utility reports the safety related occurrences to the safety authority according to the administrative rules prescribed in the technical specifications. The similar reports are termed licensee event reports in USA. A Swedish safety related occurrence (RO) report is prepared and checked subsequently at the plant prior to the reporting to the safety authority. The RO reports include descriptions and classifications of failures involving unavailability of safety-related equipment and deviations from the rules in the technical specifications.

Technical Specifications. The technical specifications (TS) are safety rules, approved by the regulatory authority, stipulating the limits and conditions for safe operation of a nuclear power unit.

Utility function. Utility function is a representation of the decision maker's preferences over uncertain outcomes. Let x^* be the most preferrable outcome and x^0 the least preferred one. If the decision maker considers a sure outcome x equally preferrable as an lottery where x^* has the probability p and x^0 the probability 1-p then the utility $u(x)$ is p.

Value function. Value function is a representation of the decision maker's preferences over an attribute like money. If and only if an attribute level x is preferred to y then $v(x)>v(y)$ where v is the value function from attribute levels to real numbers. A multi-attribute value function is representation from a multi-dimensional attribute space to real numbers.

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There are 16 nuclear power plants in operation in the Nordic countries (in Finland and Sweden), and various techniques are available to assure a high level of safety during their operation. In a Nordic project, advanced methods have been studied, they help to inform the plant personnel about the actual state of safety and about the implications on safety that may result from changes introduced during operation. As an example, living PSA can be used for safety planning of maintenance activities and risk analysis of operating experience. Safety indicators that may reveal whether the safety is actually improving or degrading can be identified, so that corrective measures can be taken.

The Nordic Committee for Nuclear Safety Research - NKS
organizes pluriannual joint research programmes. The aim is to achieve a better understanding in the Nordic countries of the factors influencing the safety of nuclear installations. The programme also permits involvement in new developments in nuclear safety, radiation protection, and emergency provisions. The three first programmes, from 1977 to 1989, were partly financed by the Nordic Council of Ministers.

The 1990 - 93 Programme
comprises four areas:

* Emergency preparedness (The BER-Programme)
* Waste and decommissioning (The KAN-Programme)
* Radioecology (The RAD-Programme)
* Reactor safety (The SIK-Programme)

The programme is managed - and financed - by a consortium comprising the Danish Emergency Management Agency, the Finnish Ministry of Trade and Industry, Iceland's National Institute of Radiation Protection, the Norwegian Radiation Protection Authority, and the Swedish Nuclear Power Inspectorate. Additional financing is offered by the IVO and TVO power companies, Finland, as well as by the following Swedish organizations: KSU, OKG, SKN, SRV, Vattenfall, Sydkraft, SKB.

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