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Advances in Operational Safety and Severe Accident Research

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VTT Automation, Finland

February 2002

Nordic Nuclear Safety Research (NKS)

organizes joint four-year research programs involving some 300 Nordic scientists and dozens of central authorities, nuclear facilities and other concerned organizations in five countries. The aim is to produce practical, easy-to-use reference material for decision makers and help achieve a better popular understanding of nuclear issues.

To that end the results of the sixth four-year NKS program (1998 - 2001) are herewith presented in a series of final reports comprising reactor safety, radioactive waste management, emergency preparedness, radioecology, and databases on nuclear threats in Nordic surroundings. Each report summarizes the main work, findings and conclusions of the six projects carried out during that period. The administrative support and coordination work is presented in a separate report. A special Summary Report, with a brief résumé of all projects, is also published. Additional copies of the reports on the individual projects as well as the administrative work and the Summary Report can be ordered free of charge from the NKS Secretariat.

The final reports - together with technical reports and other material from the 1998 - 2001 period - will be collected on a CD-ROM, also available free of charge from the NKS Secretariat.

During the last few years a growing interest has been noted among sister organizations in the three Baltic States, especially in the field of emergency preparedness, radiation protection and radioecology. This has widened the scope of our joint Nordic work and fed new influences and valuable competence into the NKS program. The Baltic participation is therefore gratefully acknowledged.

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Advances in Operational Safety and Severe Accident Research

Final Report of the
Nordic Nuclear Safety Research
Project SOS-2

Kaisa Simola

VTT Automation, Finland

February 2002

This is NKS

NKS (Nordic Nuclear Safety Research) is a scientific cooperation program in nuclear safety, radiation protection and emergency preparedness. It is a virtual organization, serving as an umbrella for joint Nordic initiatives and interests. Its purpose is to carry out cost-effective Nordic projects producing seminars, exercises, reports, manuals, recommendations, and other types of reference material. This material, often in electronic form on the official homepage www.nks.org or CD-ROMs, is to serve decision-makers and other concerned staff members at authorities, research establishments and enterprises in the nuclear field.

A total of six projects were carried out during the sixth four-year NKS program 1998 - 2001, covering reactor safety, radioactive waste, emergency preparedness, and radioecology. This included an interdisciplinary study on nuclear threats in Nordic surroundings. Only projects of particular interest to end-users and financing organizations have been considered, and the results are intended to be practical, useful and directly applicable. The main financing organizations are:

- The Danish Emergency Management Agency
- The Finnish Ministry for Trade and Industry
- The Icelandic Radiation Protection Institute
- The Norwegian Radiation Protection Authority
- The Swedish Nuclear Power Inspectorate and the Swedish Radiation Protection Authority

Additional financial support has been received from the following organizations:
In Finland: Fortum (formerly Imatran Voima, IVO); Teollisuuden Voima Oy (TVO)
In Sweden: Sydkraft AB; Vattenfall AB; Swedish Nuclear Fuel and Waste Management Co. (SKB); Nuclear Training and Safety Center (KSU)

To this should be added contributions in kind by all the organizations listed above and a large number of other dedicated organizations.

NKS expresses its sincere thanks to all financing and participating organizations, the project leaders, and all participants, all in all some 300 persons in five Nordic countries and the Baltic States, without which the NKS program and this report would not have been possible.

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Abstract

A project on reactor safety was carried out as a part of the NKS programme during 1999-2001. The objective of the project was to obtain a shared Nordic view of certain key safety issues related to the operating nuclear power plants in Finland and Sweden. The focus of the project was on selected central aspects of nuclear reactor safety that are of common interest for the Nordic nuclear authorities, utilities and research bodies.

The project consisted of three sub-projects. One of them concentrated on the problems related to risk-informed decision making, especially on the uncertainties and incompleteness of probabilistic safety assessments and their impact on the possibilities to use the PSA results in decision making. Another sub-project dealt with questions related to maintenance, such as human and organisational factors in maintenance and maintenance management. The focus of the third sub-project was on severe accidents. This sub-project concentrated on phenomenological studies of hydrogen combustion, formation of organic iodine, and core recriticality due to molten core coolant interaction in the lower head of reactor vessel. Moreover, the current status of severe accident research and management was reviewed.

Key words

Condition monitoring, decision criteria, HRA, human factors, hydrogen, iodine, maintenance, PSA, risk-informed decision making, reactor safety, recriticality, safety analysis, severe accidents, uncertainty

Summary

The project on reactor safety in the NKS programme focused on certain safety-related topics that were identified to be of common interest within the Nordic nuclear community, and that were not covered by other international research projects.

The project was realised in three sub-projects, each of them consisting of several tasks and research topics. These sub-projects were:

SOS-2.1 Safety development: The sub-project concentrated on the problems related to risk-informed decision making, especially on the uncertainties and incompleteness of probabilistic safety assessments (PSA) and their impact on the possibilities to use the PSA results in decision making.

SOS-2.2 Management of plant maintenance and renewal: One aim of this sub-project was to promote the analyses of human and organisational factors in maintenance. Another aim was to enhance understanding related to maintenance management.

SOS-2.3 Severe accidents: This sub-project concentrated phenomenological studies of hydrogen combustion, formation of organic iodine, and core recriticality due to molten core concrete interactions in the lower head of reactor vessel. Also the current status of research and management of severe accidents in Nordic countries was reviewed.

The main activities and findings of the sub-projects are shortly described in this summary.

Risk-informed decision making

Three studies related to uncertainties and incompletenesses of probabilistic safety assessments were conducted in sub-project SOS-2.1. The focus was on the comparison of two PSAs, communication of uncertainties, and study of active human errors and their role in PSAs. Risk-informed decision making was addressed in sub-project SOS-2.1 by arranging a Nordic seminar to review the status of risk-informed applications and by conducting some case studies.

A comparative study of two PSAs of nearly identical nuclear power units, but with significantly different results, was conducted. The aim was to identify, clarify and explain the differences between PSA-studies, and to give recommendations for the comparison of PSA-studies. The impact of assumptions and uncertainties on the results was evaluated. The study resulted in recommendations concerning the documentation of PSAs. A need for harmonisation of certain parts of the studies also arose.

A second study highlighted the need for structured analysis and presentation of uncertainties to facilitate the communication between different experts and authorities. The adoption of risk-informed decision making principles sets requirements for uncertainty analyses and their documentation, since it is important to be aware of the assumptions made in the analyses. To improve the communication between the various parties involved in the analyses of phenomena and the decision-makers, a study attempting to present a view upon uncertainty analysis of physical models was conducted. The emphasis was on the identification and documentation of various types of uncertainties and assumptions in the modelling of the phenomena.

A study on active human errors, also known as commission errors, was conducted. The study covered initially control room activities, maintenance, surveillance testing and outage management, i.e., no human activities in nuclear power plants were left outside its scope. The priority was laid on scanning the problem area and on forming interdisciplinary views about the principles of commission error analysis. According to the study, a significant number of events were due to human actions outside the control room, which should be reflected in the PSA models.

A review on decision criteria was done and the principles for evaluating the criteria were identified. The need for probabilistic decision rules or criteria has arisen, but there are many practical and theoretical problems in the application. The decision alternatives can not always be modelled in the same degree of detail. Thus, the decision criteria have to be selected or applied in a context sensitive way. The use of the criteria in the decision problem must be justified and evaluated.

A pilot-study was conducted to develop a safety classification proposal based on risks for selected equipment of a nuclear power plant. The application plant in this case was the Loviisa NPP, Unit 1. Comparisons to original safety classifications and technical specifications were made. The analyses showed a wide range of importances within all safety classes.

As the risk-informed in-service inspection applications have become increasingly attractive, the quantitative estimation of pipe break frequencies has become an interesting topic. It is recognised that pipe break frequencies calculated with fracture mechanistic models and those estimated from the operating experience may lead to quite different results. A comparative analysis of pipe failure probabilities due to stress corrosion cracking based on two alternative analysis methods was performed. The main reasons for the differences in the numerical results were analysed, and the applicability and restrictions of the approaches were discussed.

Quality and management of maintenance

Sub-project SOS-2.2 addressed the quality of maintenance work by considering the role of human errors in maintenance with respect to operability and safety. In Finland, systematic and in-depth analysis of operating experience of human errors related to maintenance was started during the previous NKS-programme and continued during this one.

The human common cause failure analyses at Finnish power plants show that the maintenance work order data is helpful in the identification and analyses of human failure events in relation to the maintenance activities. These events include failures in the verification of operability before the work is ready, and in the planning needed prior to the execution of the maintenance or modification work in the plant. A structured classification and analysis facilitate the identification of failed barriers and the error mechanisms that have penetrated the barriers and remained hidden in the system for longer periods. An appropriate classification, in-depth analysis and statistical treatment of maintenance-related errors also provide valuable information for the focusing of e.g. psychological studies on the most relevant aspects in maintenance and operability activities.

A review of the research needs in the area of human factors in maintenance in Sweden was done by interviewing both the authority, SKI, and the utilities. The interviews gave an overview of the completed, ongoing and planned research and development work related to human factor problems in maintenance. The needs for future research and development projects identified by the plant personnel and SKI were classified and summarised.

The management of maintenance was considered in SOS-2.2. Also a review on transformer failures and maintenance was included, since transformers are crucial equipment from the point of view of availability. Transformer explosions are also a risk. The report gives recommendations on condition monitoring of the transformer isolation and oil.

A discussion and working group on maintenance decisions was established within the NKS framework. The group consisted of representatives from power plants. Exchange of information was carried out in order to compare and identify good practices, especially to assure economically competitive electricity production without decreasing reactor safety. As an example, maintenance strategy classifications for ranking the maintenance items for better allocation of condition-based, preventive or corrective maintenance at various plants were discussed and clarified. The ways to measure the outcome and effectiveness of maintenance based on the analysis of experience data were compared.

A survey on the management of condition monitoring information was conducted by interviews at several Nordic power plants. Predictive maintenance strives to prevent the failure of the component by utilising condition monitoring and different information systems for maintenance steering. The interviews and plant visits show that the maintenance strategies are only rather slowly shifting towards condition-based maintenance despite extensive condition monitoring methods and equipment acquired for the plants.

Severe accidents

Severe accident research in SOS-2.3 consisted of a review of the current status of research and management of severe accidents in the Nordic countries. The phenomenological studies focused on hydrogen scenarios and formation of organic iodine. In addition, a study on recriticality of a BWR core after molten core concrete interactions in the lower head was conducted.

A scenario of hydrogen detonation in a BWR reactor building was investigated in order to evaluate the integrity of containment in case of detonation loads from the outside. During a severe accident, significant release of hydrogen into containment can occur and a hydrogen leak from the containment into the surrounding reactor-building rooms cannot be ruled out. The study consisted of analyses of detonations based on earlier calculations of hydrogen concentrations, and of structural calculations.

A study was conducted to investigate the most important accident sequences concerning hydrogen generation and containment pressure at hydrogen deflagration in Ringhals 3 PWR. The focus was on the analysis of sequences with reflooding of the damaged core, and detailed analyses of the hydrogen production and containment pressure were performed for the most important sequences.

The formation and behaviour of organic iodine was addressed by literature reviews and small scale experiments. The work aimed at creating an understanding of the underlying chemistry and performing small-scale experimental work. Two literature surveys were conducted within the project; one to review the methods of preventing a source term of methyl iodide during a core melt accident and another to gather valuable information on the behaviour of methyl iodide in the gas phase during a severe accident.

In the experimental studies, the dependence of the formation of organic iodine on the pH of the filter solution was verified. Experiments with painted surfaces simulated the formation of organic iodine in an accident situation in the Loviisa reactor. The third experiment investigated the possibility to trap iodine by silver nanoparticles. These particles absorbed efficiently elemental iodine, but the results with methyl iodide were not as good.

A study was conducted to determine the potential for recriticality of the degraded core of a BWR. In the analysed scenario, a large amount of melt enters the lower head resulting in a melt-water interaction. A steam explosion or a strong evaporation in the lower plenum may push a water slug into the downcomer and core regions, which may lead to a prompt power excursion, that in turn may fragment the fuel pins.

Concluding remarks

It can be stated that the need for interdisciplinary work seems to be increasing along with the growing use of risk-informed regulation and plant management. This sets growing demands on the understanding between various experts is, since in decision making situations the expertise of several parties has to be combined in a coherent way. The limitations of the PSA model have to be identified and evaluated in all applications where it is used as an aid for decision making.

Maintenance management has not traditionally been considered a reactor safety research issue. However, lately the importance of human and organisational factors in maintenance work has received growing attention, and further research needs were identified also within this project.

The deregulated electricity market has forced the utilities to identify cost savings, e.g. in maintenance actions. However, this should be achieved without compromising plant safety. Procedures, such as reliability-centred maintenance and risk-informed in-service inspections are aimed at optimising the maintenance by taking into account the reliability and risk analysis results. Experience shows that with such approaches it is possible to simultaneously increase safety and availability and reduce the maintenance costs. Furthermore, the maintenance strategies can be potentially improved by optimising inspection and testing and by increasing the use of condition monitoring information for maintenance steering.

Severe accident research has long traditions in the Nordic countries, with active international participation. The topics within the NKS have naturally focused on open questions related to the Nordic BWR plants, and within this programme period the research efforts were concentrated on hydrogen issues, behaviour of organic iodine and recriticality of a degraded core. These studies have increased the understanding of these phenomena and identified remaining work in these topics. In addition, other areas where continued research is needed have been identified and reported within this project.

This final report summarises the studies and highlights the needs for future research in a number of areas that have traditionally been subjects in the Nordic co-operation. However, the SOS-2 project covered only some specific topics of reactor safety, and there may be a need to modify the subjects within the forthcoming NKS programmes. As EU funding for nuclear reactor safety research is significantly decreasing, the importance of the Nordic co-operation within NKS is growing and the focus of the Nordic research should be a subject of continuous discussion.

Sammanfattning

Projektet avseende reaktorsäkerhet i NKS programmet var inriktat mot vissa säkerhetsrelaterade frågor som under förprojektfasen konstaterades vara av gemensamt intresse för parterna inom den nordiska kärnsäkerhetsforskningen, men som inte--ej täcks in av övriga internationella forskningsprojekt.

Projektet genomfördes i tre delprojekt som vart och ett omfattade flera uppgifter och forskningsområden. Dessa delprojekt var:

SOS-2.1 Säkerhetsutveckling: Delprojektet koncentrerades på problem relaterade till risk-informerat beslutsfattande, speciellt rörande osäkerhet och ofullständighet av sannolikhetsbaserade säkerhetsanalyser (PSA) och deras effekt på möjligheterna att använda resultat från PSA vid beslutsfattande.

SOS-2.2 Ledning av underhåll och förnyelse: Ett mål var att befrämja analyser av mänskliga och organisatoriska faktorer rörande underhåll. Ett annat syfte var att öka förståelsen rörande ledning av underhåll.

SOS-2.3 Svåra haverier: Detta delprojekt koncentrerades på att studera fenomen som förbränning av väte, bildande av organiskt jod, och återkriticitet av reaktorhärden på grund av växelverkan härdsmälta - betong vid undre delen av reaktortanken. Även nuläget för forskningen och bemästrande av svåra olyckor i nordiska länder undersöktes.

Huvudaktiviteterna och slutsatserna från delprojekten beskrivs kortfattat i sammanfattningen.

Risk-informerat beslutsfattande

Tre undersökningar rörande osäkerheter och ofullständigheter av sannolikhetsbaserade säkerhetsanalyser genomfördes inom delprojektet SOS-2.1. Utgångspunkten var jämförelse av två PSA-studier, kommunikation av osäkerheter, och undersökning av aktiva mänskliga fel och deras roll i PSA-studier. Riskinformerat beslutsfattande behandlades inom delprojektet SOS-2.1 genom att arrangera ett nordiskt seminarium för utredning av läget för riskinformerade tillämpningar och genom att göra några tillämpningssudier.

En jämförande studie genomfördes av två PSA rörande nästan identiska kärnkraftsblock, som uppvisade anmärkningsvärt olika resultat. Syftet var att identifiera, klargöra och förklara skillnaderna mellan PSA-studierna, samt ge rekommendationer för jämförelse av PSA-studier. Effekten av antagandena och osäkerheterna i resultaten utvärderades. Undersökningen resulterade i rekommendationer rörande dokumentation av PSA. Ett behov att harmonisera vissa delar av studierna uppenbarade sig också.

En annan studie underströk behovet att på ett strukturerat sätt analysera och presentera osäkerheter för att möjliggöra kommunikation mellan olika experter och myndigheter. När man använder riskinformerade principer för beslutsfattande ställs krav på osäkerhetsanalyser och dokumentation av dem, varför det är viktigt att känna till antagandena gjorda vid analyserna. För att förbättra kommunikationen mellan olika parter inblandade vid analyserna av fenomenen och beslutsfattarna, genomfördes en studie som syftade till att presentera en syn på osäkerhetsanalys av fenomenologiska modeller. Man betonade identifiering och dokumentation av olika typer av osäkerheter och antaganden vid modellering av fenomenen.

En studie genomfördes av aktiva mänskliga fel även kända som felaktiga handlingar. Studien omfattade redan från början kontrollrumsaktiviteter, underhåll, funktionsprovning och ledning av avställningar, d.v.s. inga mänskliga aktiviteter vid kärnkraftverkets drift lämnades utanför studiens omfattning. Man inriktade sig på att gå igenom problemområdet och bilda en tvärteknisk syn på principerna för en analys av aktiva felhandlanden. Enligt studien förekom ett anmärkningsvärt antal av mänskliga felhandlanden utanför kontrollrummet vilket borde återspeglas i PSA-modellerna.

En granskning av beslutskriterierna gjordes och principerna för att värdera kriterierna identifierades. Behovet av sannolikhetsbaserade beslutsregler eller -kriterier har framkommit, men det finns många praktiska och teoretiska problem vid tillämpningen. Beslutsalternativen kan inte alltid modelleras med samma detaljeringsgrad. Således måste beslutskriterierna väljas eller tillämpas med hänsyn till sammanhanget. Användningen av kriterierna vid beslutsfattande måste begrundas och utvärderas.

En pilotstudie genomfördes för utveckling av ett förslag till riskbaserad säkerhetsklassificering av utvald utrusting vid ett kärnkraftverk. I detta fall gjordes tillämpningen för Lovisaverket 1. Jämförelser gjordes med den ursprungliga säkerhetsklassificeringen och med säkerhetstekniska föreskrifter. Analyserna uppvisade en vid variationsbredd av viktigheter inom samtliga säkerhetsklasser.

När riskinformerade tillämpningar av oförstörande provning har blivit alltmer lockande, har kvantitativ uppskattning av rörbrottsfrekvenser blivit ett intressant ämne. Det har konstaterats att rörbrottsfrekvenser beräknade utifrån brottsmekaniska modeller eller utifrån drifterfarenheter kan leda till betydligt varierande resultat. En jämförande analys som byggde på de alternativa metoderna gjordes för rörskadesannolikheter av spänningskorrosionsprickor. Huvudorsakerna för skillnaderna i de numeriska resultaten analyserades, och tillämpbarheten och begränsningarna av förfaringssätten diskuterades.

Kvalitet och ledning av underhåll

Vid delprojektet SOS-2.2 behandlades kvalitén av underhållsarbetet genom att undersöka mänskliga fel i samband med underhåll och deras effekter för driftklarhet och säkerhet. En systematisk och ingående analys av drifterfarenheter av mänskliga fel i samband med underhåll påbörjades under det tidigare NKS-programmet och fortsatte under detta program i Finland.

Analyser av mångfaldiga mänskliga fel med gemensam orsak (HCCF) vid finska kraftverk visar att arbetsorderdata är nyttiga för identifiering och analys av mänskligt felhandlande i samband med underhållsaktiviteter. Dessa händelser innehåller brister vid verifiering av driftklarheten innan arbetet är färdigt, och vid planering som erfordras innan underhålls- eller ändringsarbetet påbörjas i anläggningen. En strukturerad klassificering och analys möjliggör identifiering av bristande barriärer och felmekanismer som penetrerat barriärerna samt blivit kvar i systemet under långa perioder. En ändamålsenlig klassificering, ingånde analys och statistisk behandling av underhållsrelaterade felhandlanden genererar också nyttig information för riktning av t.ex. psykologiska studier till de mest relevanta områdena av underhåll och driftklarhet.

En genomgång av forskningsbehov rörande mänskliga faktorer inom underhåll i Sverige gjordes genom att intervjua både myndigheten, SKI, och kraftbolagen. Intervjuerna resulterade i en sammanställning av genomförda, pågående och planerade forsknings- och utvecklingsarbeten relaterade till problem avseende mänskliga faktorn vid underhåll. Man klassificerade och sammanställde behovet av framtida forsknings- och utvecklingsprojekt som identifierats av personal från anläggningar och SKI.

Ledning av underhåll undersöktes inom SOS-2.2. Transformatorfel och -underhåll undersöktes också därför att transformatorer är kritiska ur tillgänglighetssynpunkt. Transformatorexplosioner är också en risk. Rapporten ger rekommendationer för konditionsövervakning av transformatorns isolering och olja.

En diskussions- och arbetsgrupp rörande underhållsbeslut bildades inom ramen av NKS-projektet. Gruppen bestod av representanter från kraftbolag. Informationsutbytet skedde för att jämföra och identifiera god praxis, speciellt för försäkra sig om ekonomiskt konkurrenskraftig elproduktion utan att reducera reaktorsäkerheten och genom att utveckla säkerheten selektivt. Man diskuterade och utredde till exempel underhållsstrategiska klassificeringar för prioritering av anläggningens underhållsobjekt för granskning och förbättrad allokering av tillståndsbaserade, förebyggande eller avhälpande underhållsstrategier inom olika kärnkraftverk. Dessutom jämförde man sätt att mäta effekten och effektiviteten av underhållet liksom underhållsoptimeringar utgående från analys av fel- och underhållsdata.

En undersökning av hanteringen av informationen från tillståndsövervakningen gjordes genom intervjueer vid olika nordiska kärnkraftverk. Med tillståndskontroll avser man att förhindra degradering av komponenten, och en efterföljande felhändelse, genom att utnyttja tillståndsövervakning och olika informationssystem för underhållsstyrning. Intervjuerna och anläggningsbesöken visar att underhållsstrategierna förandras endast rätt långsamt mot tillståndsbaserat underhåll på anläggningarna trots att man anskaffat omfattande tillståndsövervakningssystem och -instrument.

Svåra haverier

Forskningen av svåra haverier i SOS-2.3 omfattade en genomgång av nuläget för forskning och bemästrande av svåra haverier i nordiska länder. Studier av fenomen inriktade sig på vätgasscenarier och bildande av organiskt jod. Dessutom gjordes en studie av återkriticitet av BWR-härden efter växelverkan mellan härdsmälta och betong vid undre delen av reaktortanken.

Ett scenario med vätgasexplosion inom en BWR-reaktorbyggnad undersöktes för utvärdering av integriteten av reaktorinneslutningen vid explosionsbelastningar utifrån. Under ett svårt olycksfall kan det förekomma betydliga utsläpp av vätgas till inneslutningen och vätgasläckan till en omgivande reaktorbyggnad kan inte regleras bort. Studien omfattade analyser av explosioner utgående från tidigare beräkningar på vätgaskoncentrationer, och av strukturella beräkningar.

En studie genomfördes för undersökning av de viktigaste olycksfallssekvenserna rörande generering av vätgas och inneslutningstrycket vid vätgasdeflagration i Ringhals 3 PWR. Man koncentrerade sig på analys sekvenser med översvämning av en skadad härd, och detaljerade analyser av produktion av vätgas och inneslutningstrycket gjordes för de viktigaste sekvenserna.

Formation och funktion av organiskt jod behandlades genom litteraturundersökningar och småskaleprov. Arbetet syftade till att skapa förståelse av underliggande kemi samt att göra experimentellt småskalearbete. Två litteraturstudier gjordes inom projektet; en för att gå igenom metoder för att förhindra utsläpp av källtermen av metyljodid under härdsmältningsolyckan och den andra för insamling av värdefull information om hur metyljodid uppträder gasfasen under en svår olycka.

I de experimentella studierna verifierades inverkan av formationen av organiskt jod på pH-värdet av filterlösningen. Prov med målade ytor simulerade bildandet av organiskt jod i en olycksafallssituation vid Lovisa-reaktorn. Vid det tredje experimentet undersöktes möjligheten att fånga jod med nanopartiklar av silver. Dessa partiklar absorberade effektivt elementärt jod, men resultaten med metyljodid var inte lika goda.

En studie gjordes för att bestämma potentialen för återkriticitet av den skadade härden i en BWR. I det analyserade scenariet tränger en del av smältan in i den lägre delen av reaktortanken och resulterar i växelverkan mellan smälta och vatten.

En ångexplosion eller stark uppångning i lägre plenum kan skjuta en vattenmassa till utfallsdelen och härdregionen, vilket kan leda till en prompt effektexkursion, som i sin tur kan splittra bränslestavar.

Slutsatser

Det verkar finnas ett uttalat behov av ett ökat tvärtekniskt arbete, när riskinformerad reglementering och anläggningsledning ökar. Detta ställer ökande krav på förståelsen mellan olika experter eftersom man behöver kombinera expertisen från flera parter på ett integrerat sätt vid beslutssituationerna. PSA-modellens begränsningar måste identifieras och utvärderas vid samtliga tillämpningar, när de används som stöd för beslutsfattande.

Underhållsledning har traditionellt inte betraktats som en fråga för reaktorsäkerhetsforskning. Emellertid har vikten av mänskliga och organisatoriska frågor inom underhållet uppmärksammats i ökad omfattning, och ytterligare forskningsbehov identifierades inom detta projekt.

Avregleringen av elmarknaden har tvingat kraftbolagen att söka efter produktivitetsökningar och kostnadsbesparingar bl.a. vid underhåll. Emellertid måste konkurrenskraften uppnås utan att man kompromissar med anläggningssäkerheten. Planeringsmetoder, såsom erfarenhetsbaserat tillförlitlighetsstyrt underhåll och riskinformerad oforstörande provning, är avsedda att optimera underhållet genom att beakta resultat från risk-tillförlitlighetsanalyser omfattande systematiska analyser av fel- och underhållsdata.

Erfarenheten visar att det är möjligt att med sådana metoder samtidigt öka säkerheten och tillgängligheten och reducera underhållskostnaderna selektivt. Dessutom kan underhållsstrategier potentiellt förbättras genom optimering av inspektioner och provningar samt genom att öka användningen av information från tillståndsövervakningen för underhållsstyrning.

Forskningen om svåra haverier har långa traditioner i de nordiska länderna, med aktivt internationellt deltagande. Ämnena inom NKS har naturligtvis inriktat sig på öppna frågeställningar rörande nordiska BWR-block, och inom denna programperiod har forskningsinsatserna koncentrerat sig på vätgasproblem, uppträdande av organiskt jod och återkriticitet av en skadad reaktorhärd. Dessa studier har ökat förståelsen av dessa fenomen och identifierat återstående arbete inom dessa områden. Dessutom har man inom detta projekt identifierat och rapporterat övriga områden, där fortsatt forskning behövs.

Slutrapporten sammanfattar studierna och understryker behoven för framtida forskning inom ett antal områden som traditionellt varit föremål för nordiskt samarbete. Emellertid täckte projektet SOS-2 endast in några specifika teman av reaktorsäkerhet, och det kan vara nödvändigt att modifiera områdena inom framtida NKS-program. När EU-finansieringen för forskning om den nukleära säkerheten minskar betydligt, ökar behovet av det nordiska samarbetet inom NKS. Inriktningen av den nordiska forskningen borde vara ett ämne för en kontinuerlig diskussion.

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Introduction

The objective of the NKS/SOS-2 project has been to obtain a shared Nordic view of some key safety issues related to the operating nuclear power plants in Finland and Sweden. The focus of the project has been on selected central aspects of nuclear reactor safety that are of common interest for the Nordic nuclear authorities, utilities and research bodies.

The project succeeded RAK-1, strategies for reactor safety, and RAK-2, severe accidents, projects of the previous NKS-programme (1994-1997), and the recommendations of the final reports of these two projects (Andersson 1998, Lindholm et al. 1998) were considered in the planning of the SOS-2 project. This is reflected e.g. in the continuation of maintenance related studies, in addressing probabilistic safety analyses, and in the selection of the phenomenological issues for consideration in the area of severe accident research. Furthermore, the final report of the programme group (Magnússon 1998) summarising the project proposals from the research organisations and nuclear regulators, provided a basis for the planning of the project. The background for the planning and the resulting research plan are documented in (Bennerstedt 1999).

The decrease in the price of electricity due to the deregulation of the electricity market has forced the power producers to find ways to lower production costs. Saving potential has been identified e.g. in maintenance costs. On the other hand, risk-informed applications seem to provide help as the results of probabilistic safety analyses may reveal possibilities to re-direct / optimise inspections and maintenance by increasing safety and reducing costs. This makes the use of PSA results attractive for both the regulatory and industrial side. The applicability of PSA results is not straightforward, however, and assumptions and uncertainties related to the modelling significantly affect the analysis results. A better understanding of the applicability and its limitations is needed.

The human aspects are considered an important research area. The modelling in safety analyses is not an easy task. Increasing attention has been paid not only to the human errors in plant operation and accident situations but also to the daily plant maintenance actions.

Although the research topics on severe accidents have been well covered in international and European projects, some Nordic studies within the NKS-programme have been considered to add value to the international research. The main phenomenological issues included in the project are related to accident scenarios of the ABB BWRs.

The work was organised in three sub-projects:

SOS-2.1 Safety Development

SOS-2.2 Management of Plant Maintenance and Renewal
SOS-2.3 Severe Accidents

Sub-project SOS-2.1 concentrated on the development and application of probabilistic safety analyses. The tasks were related to specific areas of interests, such as uncertainties in the analyses, risk-informed safety management and the use and interpretation of PSA models and results.

Sub-project SOS-2.2 focused on questions related to maintenance of nuclear power plants. Quality assurance in maintenance was addressed by consideration of human aspects in maintenance actions and by identifying the research needs in this field. The other tasks were related to maintenance strategies and management of condition monitoring information.

Sub-project SOS-2.3 dealt with severe accidents with focus on specific questions related to hydrogen issues and the behaviour of organic iodine. These tasks were defined to have a particular Nordic interest and not to overlap with other studies in this area. Furthermore, the present state of severe accident research and its impact on accident management in the Nordic countries was reviewed. In the course of the project, a study on recriticality due to a hydrogen explosion in the lower head of the reactor was added to this sub-project.

A working group consisting of interested authority and utility representatives and researchers met twice a year within each sub-project to discuss and decide on the progress of the work. The work consisted of several smaller studies, in addition, seminars were organised for exchanging information and reviewing the current status of the research and practice. One characteristic of the SOS-2 project is the interdisciplinary nature of the work: the research and support personnel consisted of PSA experts, reliability engineers, specialists on fracture mechanics, structural analysts, maintenance managers, psychologists, chemists, and physicists.

Although the project dealt with power reactors only, some of the results may also be applicable to research of similar smaller reactors. Moreover, many of the studies related to the management of uncertainties, risk-informed decision making and maintenance issues can contribute to other non-nuclear industries, especially those with risk concerns.

Figure 1 shows the main topics considered in the project SOS-2 and the division into sub-projects.

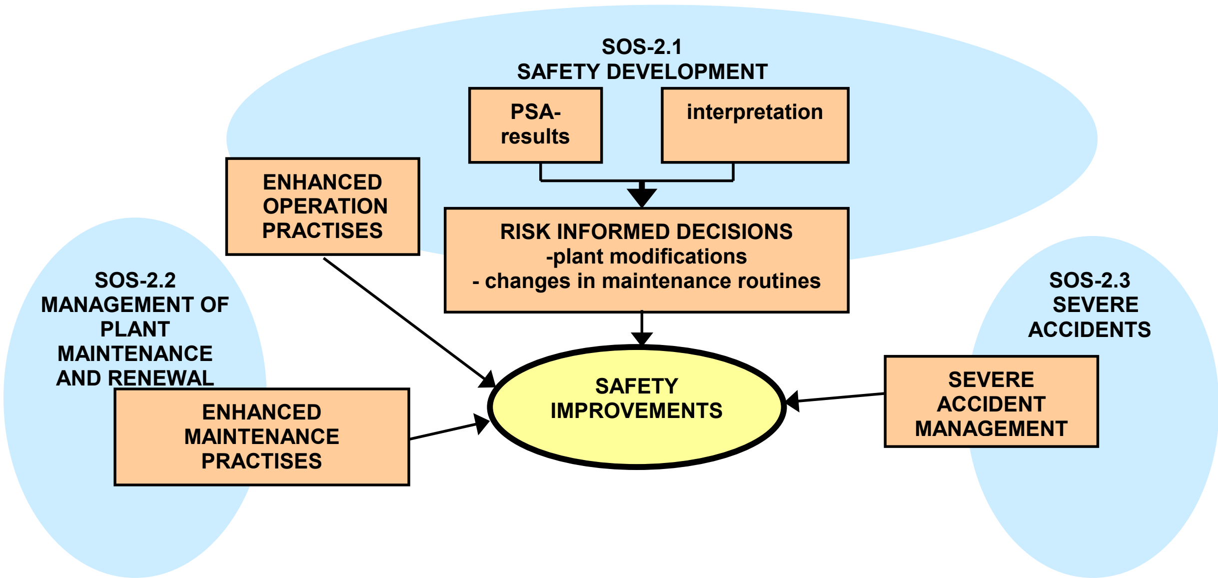


Figure 1. SOS-2 division in sub-projects.

Safety development (SOS-2.1)

The previous NKS research projects dealing with probabilistic safety assessments have concentrated on areas such as common cause failures, human errors, uncertainty and sensitivity analyses, optimisation of safety related technical specifications, living PSA and safety indicators, and integrated accident sequence analysis. Some of the above issues still require further research, and the increasing use of PSA results to support decision making by both the regulatory authorities and the utilities calls for continuous evaluation and improvement of the probabilistic analyses and their applications. Furthermore, a better understanding of the benefits and limitations of PSA is needed.

The general objective of the sub-project SOS-2.1, ”Safety development”, was to develop, promote and compare the use of novel risk-informed approaches in the daily work of Nordic nuclear safety authorities and power companies. On one hand, the problem areas of PSA were studied, and on the other hand, applications of PSA in spite of the problematic issues were considered. This sub-project was divided into two tasks, namely “Uncertainty in safety analyses” and “Risk-informed principles in safety management”.

Various uncertainty issues connected to PSA and deterministic safety analysis models were considered within the first task. The purpose was to give recommendations on how to use, interpret and communicate the results of safety analyses. The second task aimed at comparison of the Nordic activities and views on application of the PSA results in the regulation and operation of nuclear power plants, and at the development of specific PSA techniques for this purpose.

Uncertainty in safety analyses

Uncertainties and uncertainty analysis have been considered a relevant topic since WASH-1400 (1975). PSA uncertainties play an important role in the adaptation of risk-informed approaches. One has to know which aspects of decisions are affected by uncertainties and what their impact on the decisions is.

The work in this task aimed at a classification and characterisation of uncertain issues related to the risks of operating nuclear power plants. In this connection, recommendations on how to interpret, use and present the results of PSA analyses were given. The results of this work are also linked with the SOS-1 project, where the uncertainties of safety analyses and risk communication to the public were discussed.

The above aims were reached by addressing the issue of uncertainties with three studies. An evaluation of the impact of assumptions and uncertainties on the results of PSA was performed by comparing two Swedish PSA studies of nearly identical plants. The aim was to identify, clarify and explain the differences between PSA-studies and to give recommendations for the comparison of PSA-studies. Another study presented a view upon uncertainty analysis of physical models with an emphasis on the identification, communication and documentation of various types of uncertainties and assumptions in the modelling of the phenomena. The uncertainties related to human reliability analysis were addressed by studying the errors of commission, i.e. errors due to wrong actions.

Comparative study of two PSAs

Probabilistic safety assessment (PSA) identifies and quantifies risks by applying probability models thoroughly and consistently. It integrates many kinds of knowledge as well as results from technical analyses into a comprehensive probability model. Since the reactor safety study (WASH-1400 1975), the overall approach for performing PSA has been about the same in all PSAs.

Inevitably, PSA is based on many assumptions and modelling restrictions. Some are known and explicitly presented in the analysis, some are implicitly accepted and used in the modelling work. However, the many degrees of freedom in the analysis process and methods make the comparison of different PSA-studies difficult. Therefore, it is perhaps not fair to draw conclusions from two nuclear power plants based on PSA-studies alone.

There are, nevertheless, needs for comparing PSA-studies, because there is an intention to use PSA as a complement to deterministic safety analyses to support decision making in safety-related issues. For instance, acceptance criteria based on core damage frequency have been formulated in many countries. A comparative study supports also the harmonisation of PSAs.

Since we wish to apply PSA for safety management but we are also aware of the sensitive nature of the approach, the maturity of PSA methodology might be worth studying more deeply. One way to study such question is to compare PSA-studies. For that purpose, the recently issued PSAs for two nearly identical Swedish reactors, Forsmark 3 and Oskarshamn 3, are excellent material. The PSAs were made by two different power companies and analysis teams, and the analyses turned out quite different.

The comparative study (Holmber & Pulkkinen 2001) aimed at finding answers to questions such as:

- Why can two PSA-studies be so different?
- Which differences are the most important?
- Should methods, documentation or boundary conditions be harmonised?
- How should PSA be used?
- How should PSA be validated for applications?
- How should PSA-studies be compared?

The comparative study was carried out by going through the PSA-documentation, using the computer model and interviewing persons involved in the two PSA-projects. The review was divided in six parts corresponding to various areas of the PSA. The differences were identified for each analysis area, the impact of the differences on the study was evaluated and the reasons for the differences explained. Figure 2 illustrates the review process.

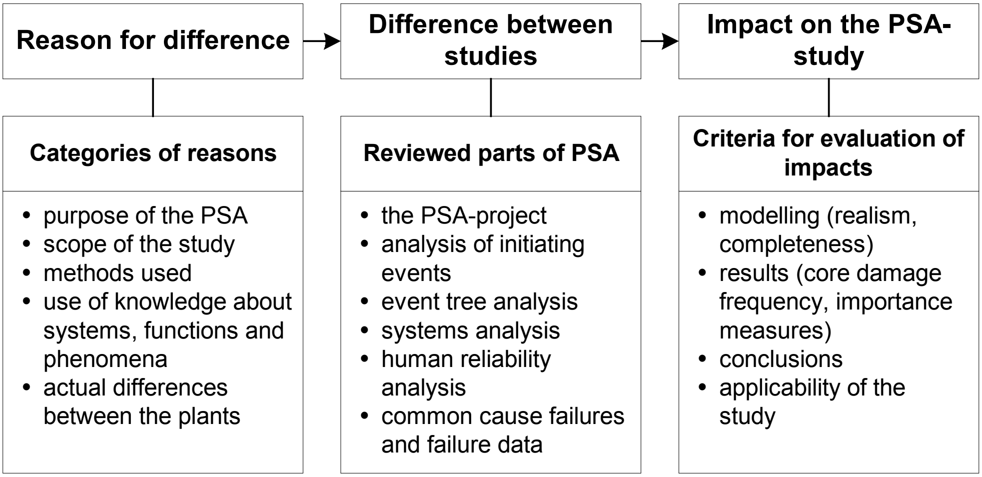


Figure 2. Analysis of the differences between two PSA-studies, reasons for the differences and effects of the differences on the PSA-studies.

Differences between the PSA-studies

Both PSA-projects have quite similar backgrounds and goals for the studies, namely to have up-to-date level 1 and 2 PSA studies covering all initiating events. However, one plant had the additional goal of a very detailed study similar to their other unit, and the plant used more than three times the man-power for the study than the other one. Furthermore, the plant aimed at a living PSA model. These differences in the goals for the PSA are reflected throughout the study, e.g., the choice of methods, scope and degree of detail of the documentation.

Initiating events are categorised differently in the two studies and this has a large effect on the end results. One of the studies has more initiating event categories and has applied more specific system categories than the other one. One study had larger frequencies for some initiating events than the other one, and there were also differences in the treatment of common cause initiators. This resulted in different risk importance values for the initiating events.

There are some differences in the definition of end states in the event tree analyses. This is due to the fact that one PSA is an integrated level 1 and 2 study whereas the other one has an independent level 1 study. Comparison of the final results is difficult because the definition of core damage is not identical. There are important differences in the system success criteria due to e.g. details of the model and definitions of controlled end states. Both analyses apply the ”small event tree – large fault tree” approach, but the degree of detail of the models is different.

A major difference in the system analyses is that one study applies failure mode and effects analyses (FMEA) to document the critical failure modes accounted in the fault tree models. The other one describes the modelling assumptions in free text. One PSA included more systems in the PSA than the other one. Comparison of the fault tree analyses for the main safety functions showed that the studies have in many case very different models. The system success criteria were different in several instances.

Although both PSAs apply a human reliability analysis (HRA) model based on performance shaping factors, these factors are not the same in the analyses. Another difference in the HRA models is that one PSA includes in some cases recoveries as a part of the decision or action by the operator whereas the other one does not take recoveries into account. Basically, this difference is due to different decomposition of human actions. In both PSAs human error events are assumed to be independent on each other, e.g. erroneous ground states. The human error events included in the analyses as well as their probabilities differ rather significantly.

The CCF analyses of both PSAs follow the CCF modelling principles of Risk Spectrum, the applied PSA computer code. The quantitative CCF estimates are calculated according to the recommendations of the Swedish SUPER-ASAR-project using the alpha-factor model and basically the same values for the CCF-model parameters. The differences are mainly due to the differences in the system models, which are more detailed in one PSA than in the other.

Both studies apply the same reliability data source, the Nordic Reliability Data Book, T-book (T-book 1996). However, the component reliability models differ. One PSA aimed at Living PSA model, and it applies time dependent component unavailability models with many failure modes and parameters of the unavailability model. Since data for every failure mode is not directly given in the T-book, various interpretations and judgements must be made. Due to conservative repair times, differences in plant specific data and these assumptions, the unavailabilities of the same type of components in the compared PSAs differ in some cases significantly.

Recommendations and conclusions

Based on the experience gained from the comparative study, recommendations concerning the performance of PSA, presentation and interpretation of the results, and the use of PSA were given.

The performance of PSA is directed in the regulatory requirements and in state-of-the-art PSA-guidelines. There is general understanding what should be included in a level 1 and 2 PSA, but there are also open items, e.g., related to area events and external events. These questions are linked to the overall requirements for safety analyses of a nuclear power plant, and they are not discussed in this context.

It is recommended that the analysis team and the future user of the study, i.e. the power plant, should carefully plan the end product in the beginning of the PSA project. Forsmark 3 and Oskarshamn 3 PSAs carry some good features but improvements can also be made. One important stage is the release of the study, and resources should be allocated for introducing the new/updated study for the plant personnel.

The recommendation regarding the work and analysis methods is to follow clearly defined references. This will facilitate the review process as well as reduce the need for documenting things. Naturally, further case by case development work by the plant itself may be needed, and that should be encouraged. However, the established definitions should be used.

An important question is the level of realism and details that should be required. The trend is to try to have as realistic PSAs as possible, but this kind of a goal has increased the complexity of PSA-studies. The increasing degree of details implies increased costs to perform the PSA, and more difficulties to review and validate the study as well as to understand the results. A recommendation is that the requirement concerning realism and details of a PSA-study should be elaborated and specified. A detailed study need not be more realistic than a less detailed study, e.g., because there may be no data for the estimation of probabilities for the various failure modes.

There are several areas in the PSA-methodology that should be harmonised. This would facilitate the review and comparison of the studies. Such areas are e.g. presentation of the results, presentation of the methods, the scope, the main limitations and assumptions, the definitions for end states (core damage categories), the definitions of initiating events and the definitions of common cause failures. Harmonisation should follow the experience from the use of the studies and the results from the research and development work.

It is recommended that the following items should always be a part of the presentation of the PSA results:

- Goal and purpose of the study
- Status of the study (revision number and date, history)
- Scope (initiating events, operational states, consequences, systems credited, human actions analysed)
- Main results (core damage frequencies per initiating events and per core damage category, safety margins i.e. conditional core damage probability given an initiating event, initiating event frequencies)
- Dominating minimal cut sets, sequences, initiating events, human errors
- Risk importance measures for basic events, systems and human errors
- Analysis of uncertainties and sensitivities
- Discussion (interpretation of the results)
- Conclusions and recommendations.

It is also recommended that an electronic documentation system that could facilitate the presentation of the results be developed.

Methods for validation of the PSA for different application areas should be developed. The development of various standards aims at this purpose, but a review based on a standard can provide only a general quality grade for a PSA. The real feasibility of a PSA can be evaluated only after experience from numerous applications.

Based on this comparison study, it can be said that in any application consultation with the expert of the specific study is needed. It is usually difficult for a person that has not participated in a PSA-project to understand all the features that should be accounted for in a PSA-application. The document and the model are seldom good enough and discussion with the analysts is needed. The analysts also need feedback from the users of the PSA.

Safety authorities and power utilities are interested in comparing and developing both PSA results and PSA applications. The findings of this study do not encourage direct comparison of quantitative PSA results only. Instead, one should concentrate on comparing the PSA-models, their assumptions and purpose, and on the safety arguments derived from the models. In addition, one should compare the insights deduced from the PSA at each level of model. This kind of in-depth comparison also helps identify the evidence behind the model and its relationship to the quantitative risk estimates.

Uncertainties of phenomenological models in PSA

When PSA is used in decision making, results from several sub-models made by several analysts have to be taken into account. The decision-maker must understand the uncertainty of each sub-model and the relationship between the models and their uncertainties. It is common that the decision-makers, who are in position to e.g. require risk-informed applications, have not usually participated in PSA themselves. They only use the results of PSA, and any unclarity of assumptions or PSA results appears to them in similar way: they do not usually know whether the uncertainty is due to phenomenological or modelling uncertainties, or due to incompleteness or boundary conditions of the model. Also the interface between submodels is one source of uncertainty.

PSA is often completed by a quantitative uncertainty analysis. Traditionally, this means the use of distributions for basic event probabilities instead of expectation values and applying Monte Carlo Simulation to propagate the uncertainties through the model. The end result of the simulation is, thus, the uncertainty distribution of the event probabilities or consequences of accident sequences. This kind of uncertainty study cannot be seen as comprehensive enough for many reasons. First, it only concentrates, by definition, on random uncertainties in the basic event data. Secondly, it does not treat modelling and identification related uncertainties. Further, as purely quantitative, it doesn't sufficiently document the causes of uncertainty and the evidence behind uncertain assumptions.

Since the views upon uncertainty differ among various participants of PSA, there is a need to improve the understanding of uncertainties, and to facilitate communication between PSA-analysts and "physicists" who analyse the phenomena. In other words, there is a need to establish more agreement between system analysts', reliability engineers' and physicists' views on uncertainty. The adoption of risk-informed decision making principles also sets requirements for uncertainty analyses and their documentation, as it is important to be aware of the assumptions made in the analyses.

Figure 3 illustrates the uncertainties of phenomenological PSA-models. In many cases, the physical phenomenon itself is not perfectly predictable, and it cannot be described with deterministic models. This kind of inherent uncertainty cannot be reduced by any means. The models try to catch the essential features of the phenomena, but they are always idealisations based on simplifying assumptions. The models are also uncertain due to insufficient data on the model parameters, which may vary from case to case. One important source of uncertainty or incompleteness is the goal or purpose of the model, which may give sufficiently good prediction for its original purpose, but may be misleading for other purposes. The predictions made by using the model may thus include uncertainties due to the phenomenon itself, and due to the restrictions of the model.

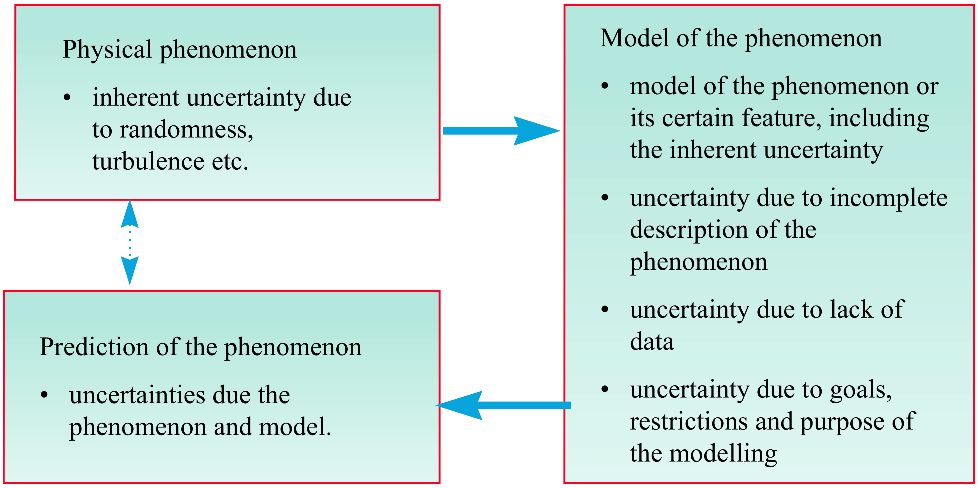


Figure 3. Relationships between uncertainties in a phenomenological PSA model.

In phenomenological models, all phenomena having impact on the systems behaviour may not be included in the model. This type of uncertainty is often referred to as *incompleteness*. It can be due to intentional decisions during the analysis planning: some things have been left out of the scope of the analysis on purpose. The reason for this kind of decisions is usually lack of resources. In the worst case, the incompleteness may be due to misunderstanding or lack of knowledge about the plant features. This kind of uncertainty can only be taken into account by independent review or analysis of the model. Incompleteness can be seen as a special form of *model uncertainty or model inadequacy.* Model uncertainties are often related to assumptions behind the model, level of detail, and scope or domain.

Parameter uncertainties refer to the unknown parameter values of valid models. This uncertainty is present as well in probability models (fault-trees, and failure time distributions) as in deterministic or probabilistic phenomenological models. Parameter uncertainty has been traditionally taken into account in uncertainty propagation of PSA models.

In addition to the classification of types of uncertainties to incompleteness, model and parameter uncertainties, a distinction in the nature of the *phenomenological uncertainty* may be made. One can speak about stochastic or *aleatory uncertainty* and knowledge or *epistemic uncertainty*. Aleatory uncertainty is sometimes called irreducible, since they cannot be made smaller without observing the real realisation of the uncertain process. Epistemic uncertainty can be decreased by obtaining additional information or by making experiments. There are differing opinions whether such distinction can be made, but often they may have useful implications for the practice of modelling, e.g. in decomposition.

A study aiming at presenting a view upon uncertainty analysis of phenomenological models with an emphasis on the identification and documentation of various types of uncertainties and assumptions in the modelling of the phenomena was conducted (Pulkkinen & Simola 2001, Pulkkinen et al. 2000). In an uncertainty analysis, it is essential to include and document all unclear issues, in order to obtain a maximal coverage of unresolved issues. This holds independently on their nature or type of the issues. The classification of uncertainties is needed in the decomposition of the problem and it helps in the identification of means for uncertainty reduction. Further, an enhanced documentation serves to evaluate the applicability of the results to various risk-informed applications. The need of a broad qualitative uncertainty analysis, forming a basis for determining requirements for quantitative uncertainty analyses, is emphasised. The study resulted in recommendations for uncertainty communication between e.g. PSA Level 1 and 2 experts, analysts and decision-makers.

A specific form was developed for the documentation of major uncertainties. The approach should help the documentation of assumptions and uncertainties in analyses of physical phenomena, and thus improve the uncertainty communication and the identification of possibilities to uncertainty reduction.

The approach consists of formats or uncertainty documentation tables, in which each phenomenon or issue is considered. First the phenomenon is described and its significance for PSA (or decision under consideration) is evaluated qualitatively. In this connection, the decomposition of phenomenon and related accident sequences are documented, and the decomposition is justified. The relationships to other issues and other models are discussed. Next, the models or computer tools used in the analysis, and reasons to use them are discussed. The theoretical basis and the degree of validation of the models and tools are described. Furthermore, the use and role of formal or informal expert judgement is explained, and the sensitivity and uncertainty analyses together with applied methodologies and main results made are presented.

In addition to the above mentioned general description each possible source of uncertainty is evaluated. In this connection both qualitative characterisation and, if possible, the impact of uncertainty to the final results is evaluated in a (semi)quantitative way. In some cases it may be advantageous to evaluate whether the analysis is based on conservative, optimistic or ”best estimate” assumptions. In order to direct additional analyses, the possibilities to reduce the uncertainty are presented.

In the documentation format, the sources of uncertainties to be covered include:
- the inherent and knowledge uncertainties related to the phenomenon under analysis (e.g. randomness, turbulence, material properties)
- model uncertainties, including those originating from the scope of the analysis, incompleteness

- uncertainties due to input data
- uncertainties due to boundary conditions applied in the model
- uncertainties selection of initial states for calculations (e.g. initiating events, assumptions on the amount of certain substances in the system, the results from another model)
- uncertainties due to computational or numerical properties of the model (nodalisation, time steps).

The approach and the uncertainty documentation forms were tested in a case study related to a BWR reactor building hydrogen combustion scenario that was analysed in the sub-project *Severe Accidents.* The scenario is described later in this final report. In the trial application of the forms, the analysts of the scenario felt that the approach is useful in structuring and decomposing the uncertainties related to the analysis. In this specific scenario, the expertise from several disciplines, were needed, which makes it even more important to have a comprehensive understanding of the major uncertainties. A structured summary with short description of applied modelling approach serves also in verifying that all major assumptions, limitations, and uncertainties have been described and their importance has been evaluated.

Commission errors

The study concentrated upon the unforeseen effects of human actions on processes and components of nuclear power plants (Pyy et al. 2001). Especially, the area of active human failures called errors of commission was studied. The classic definition for error of commission (EoC) is a somehow wrong human output i.e. selection error, error of sequence, time error (too early, too late) or qualitative error (too little, too much). This is often called the phenotype of error. For comparison, errors of omission (EoOs) mean omitting an entire task or steps in a task. Other definitions for these types of errors are listed, e.g., in Pyy (2000).

The need to complete PSAs with the analysis of errors of commission has been noticed world wide, and there is a lot of experience about significant nuclear events with considerable human contribution. In the previous NKS/RAK-1 project (Andersson 1998), methods for the human reliability analysis were developed with the concept of integrated sequence analysis (ISA). ISA formed the basis for this NKS/SOS-2 work by emphasising the need for broad analysis of man-machine systems as whole.

The objectives of the study were to perform a survey of active human errors, to summarise Nordic views on this topic, and to recommend items and approaches for further development work. The scope of the study covered control room activities, maintenance, surveillance testing and outage management, i.e., no human activities in nuclear power plants were initially left outside the scope. From the PSA point of view, the primary goal was to identify failure modes that are not included in the current PSA studies.

Developing methodology for the survey and mapping of commission errors

A methodology for the survey was developed as the first phase of the study. This included developing a classification method and an approach for information retrieval. The work was based on prior experience and the needs of the project so that a reasonable level of detail could be reached with optimal use of the resources.

Case histories were used to steer the development work so that a barrier model could be drafted for events including a significant human contribution. Figure 4 shows how causal factors lead to failed human activities, and result in consequences due to the fact that barriers fail. Finally, the progression of the event is stopped by an efficient barrier function, which may be physical, a result of engineered design or organisational. Logically, the failure of all barriers would lead to an accident.

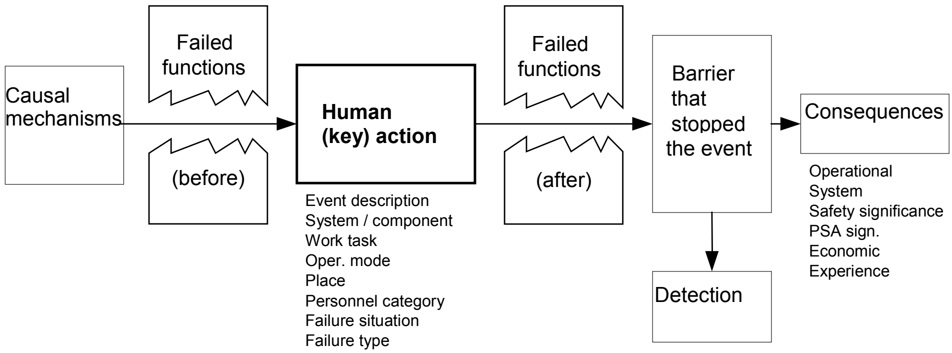


Figure 4. A schematic representation of the classification used in the study of commission errors and the location of various categories in a barrier model.

The mapping was based on NPP operating experience from the years 1997-1999. The main part of the material consisted of reports about scrams, disturbances, safety-related faults and other types of licensee event reports (LERs). Although the working group saw the need to analyse operator activities in detail, it was also established that the LER material would probably include many maintenance and testing activities. These information sources were completed by using interviews and other sources such as MTO-analyses.

The mapping of information was performed in three separate studies for Forsmark, Olkiluoto and Oskarshamn material. The classification principles of Figure 4 were transferred to columns of MS Excel tables. Each and every case has one row in the tables so that the classification could be followed.

Results

Human actions play a significant role in LERs, as shown in Table 1. Despite the importance of human actions, they are not generally analysed comprehensively unlike the technical features of the LERs. Because of this fact, human actions as causes and contributors to significant events may remain hidden in many reports. This means that a follow-up analysis is very difficult and requires extensive interviews. Because of their importance human actions deserve more attention in the analysis of operating experience.

Table 1**.** Human contribution in scram reports and in other LERs (Licensee Event Reports) through 1997-1999.

Report type	**Number of cases with human contribution (Percentage of all cases, %)**		
	Olkiluoto*	**Forsmark**	**Oskarshamn**
Scrams	7 (70 %)	1 (10 %)	15 (70 %)
Other LERs	14 (48 %)	29 (15 %)	136 (40 %)

* a more restrictive definition of LERs in Finland

Human reliability analyses have concentrated upon the omission of human actions prescribed in the procedures. However, this study shows that significant events include for the most part wrong actions, as illustrated in Table 2. Since events include many types of different human actions the study of the consequences of all reasonable human actions and their contexts is recommended rather than restriction of the analysis to a specific subset only.

Table 2**.** Share of errors of commission (EoC) compared to errors of omission (EoO) in scram and other LER events of the three different analyses.

Report type	**Number of cases with errors of commission (Percentage of cases with human contribution, %)**		
	Olkiluoto*	**Forsmark***	**Oskarshamn**
Scrams	3 (43 %)	1 (100 %)	8 (60 %)
Other LERs	17 (81 %)	12 (52 %)	74 (54 %)

* notice that especially for Olkiluoto and Forsmark some cases could not be classified - the percentage refers to the classified cases (EoO or EoC) only and the amounts may thus not be compared to Table 1. For Olkiluoto, 10 events not classified as LERs were calculated with.

Although the proportion of wrong human actions was high in the analysed material only few actions led to wrong system functions and disturbances. A significant number of events were due to human actions outside the control room. Although the control room is a focal point of operations and information exchange, maintenance, testing and operating actions take place all over the installation. This should be reflected in the HRA models.

Many important events resulted from deficiencies in work practice. Also competence, training programme, work organisation and administration had to do with many events. It is important to maintain competence in a modern NPP subject to many types of both technical and organisational changes. Also improved communication, questioning attitude to the situation and simple self-control tools (e.g. STARC, Stop-Think-Act-Review-Communicate would have possibly helped in many of the analysed events. These factors are safety culture related, which is a message for a more efficient safety management. Everybody's attitude plays a role in safety work.

The assessment of the risk significance of the faults and disturbances caused by human actions was difficult. Mostly, the analysed events did not play a significant role according to the PSAs. Only very few high Risk Achievement Factors (RAWs) due to the events were identified. Partly this was because of the shortcomings of the PSA models (e.g. no basic events exist for spurious system behaviour) and especially their HRA part. Also models for shutdown states were quite general except for Olkiluoto. However, one has to bear in mind that PSAs are intentionally based on simplified logical models.

No events aggravating the plant state during a disturbance (post IE) were identified in the material. To be able to define suitable ways to handle such events in PSA, the modelling paradigm should be discussed as a whole. Simulator exercises are recommended to support both the analysis of events and the corresponding PSA modelling.

In the study, an extended concept of active human failures was developed. Active human failure here means an event where individuals have affected technical systems in an unexpected way, which leads to other types of functional equipment consequences than unavailability of equipment only. Disturbances and spurious system actuations are examples of such consequences. Development of approaches for the analysis of active human failures and for integration of that analysis into PSA is a very demanding task. Another direction requiring, at least, equally great effort is the integration of the lessons learnt in the design process of new plants. No analysis or design principle should be based on humans acting only according to procedures, but on the goals and rational action alternatives that they are likely to have in real situations.

Risk-informed principles in safety management

One of the most recent PSA applications is risk-informed regulation. It aims at more rational safety management based on the results of plant-specific PSA. The safety authorities and power utilities both in Sweden and Finland have established programmes for implementing risk-informed decision making. This results in a need to compare and evaluate the approaches.

The work in this task aimed at reviewing the Nordic use of risk-informed principles, such as risk-based in-service inspections, risk monitoring and risk-informed evaluation of technical specifications. Further, some specific studies related to the use of PSA in decision making were conducted.

The status of the development of risk-informed principles and their adaptation in both the regulatory work and at the power plants in the Nordic countries was reviewed in a workshop on risk-informed safety management, arranged in 1999 (Pulkkinen & Simola 1999). The objectives of the workshop were

1) description of risk-informed regulation and management policies in the Nordic countries
2) comparison of the policies
3) identification of method development needs
4) recommendations for future work in the area

In addition to the review of the current Nordic status, the US experience was evaluated from a Nordic perspective (Hultqvist 2002). The US has developed and promoted the use of risk-informed approaches and thus lessons could be learned from their experience.

Some methodological issues of risk-informed approaches were considered in separate case studied. These were the probabilistic decision criteria, safety classification and pipe degradation probabilities for risk-informed in-service inspection applications. The case studies are described here in more detail.

Decision criteria for risk-informed decision making

The management of a NPP throughout its lifetime involves decisions related to design, re-design, commissioning, maintenance and operation of the system. Also external random events might affect the system in a way that requires decision making with respect to operability and accident management. Changes generally raise the question of risk significance, which has to be evaluated.

Risk significance is evaluated by using the PSA model and some probabilistic criteria, but there are many practical and theoretical problems in these applications. In some cases, the PSA application is central to the decision making process, in other cases the PSA can provide information that is complementary to the deterministic decision rules. The role of the PSA in the decision making has a direct impact on the selection of the decision criteria that give guidance in the evaluation of the risk significance of the considered change.

Different decision criteria utilised in the context of operability management were reviewed in the SOS-2 project along with principles of evaluating and applying various decision criteria, which were identified and discussed (Holmberg et al. 2001).

The use of PSA in decision making involves checking of the PSA scope and models, production of probabilistic results relevant to the decision under consideration, selection of probabilistic and other relevant decision criteria and implementation of the decision. PSA can be used for decision only if its scope includes failure modes, initiating events and phenomena relevant for the decision case. In order to be useful, these should be properly modelled in the PSA. In practice, the decision rules or criteria are defined on the basis of the PSA scope and results. Thus, different decision criteria are useful for different decision cases.

Some examples of application areas of PSA criteria are shown in Table 3. Typical decision criteria are importance measures, such as risk increase factor, risk decrease factor, and Fussel-Vesely importance measure. The applicability and feasibility of the criteria depends on the scope, purpose and modelling principles of the PSA used in the decision making. One cannot always be sure that the criteria lead to a good decision, and it is not clear how the criteria should be used.

Table 3. Examples of application areas of PSA criteria.

Examples of PSA applications	
Evaluation of risk significance	**Risk-based ranking**
LONG TERM • Analysis of technical specifications • Back-fit evaluations • Main Risk Contributors • Plant Change Assessments • IST /ISI • Compliance with safety objectives • Maintenance planning • Risk Follow-up of Licensee Events SHORT TERM • Analysis of safety margin after incident • Exemption from technical specifications	• Prioritisation of plant changes • Prioritisation of testing / inspection • Identification of risk significant systems, structures and components • Maintenance prioritisation

One has to consider the proper way to apply probabilistic rules together with deterministic ones, i.e. can the good values of probabilistic risk indices compensate the poorly fulfilled deterministic design principles? In the case of several criteria, one has to determine how the criteria are weighted with respect to each other.

The study illustrated the behaviour of various criteria (importance measures) with a biased PSA-model. The biases in the PSA-model in question were incompleteness regarding the modelled hazards, conservatism, biased failure data and incompleteness regarding the assessment of consequences. A biased model can naturally lead to the wrong decision. Some possible biased decisions are:

- An incomplete model can lead to acceptance of a system or process that actually has an unacceptably high accident frequency.
- A conservative model can point out measures for risk reduction that have minor importance.
- Biased failure data can distort the risk rank of basic events — the optimal risk reduction measures are missed.
- Incomplete assessment of consequences makes comparison of the expected usefulness of different decision options implicit — verification of full rationality of the decision is impossible.

Since there are always biases in PSA, it is recommended to pay attention to the identification and analysis of uncertainties. Conclusions made in the PSA should be validated or questioned using e.g. sensitivity studies.

It is not possible to set up general probabilistic criteria, since due to the limitations of the PSA model, all important issues are not modelled in the same way. In some cases the probability estimates are based on expert judgement, in other cases they lean on generic data and in some cases plant-specific data is used. The decision alternatives can not always be modelled in the same degree of detail. Thus, the decision criteria have to be selected or applied in a context sensitive way. The use of the criteria in the decision problem must be justified and evaluated. A set of principles for the evaluation was identified in the course of study. In order to make sure that the criteria are measurable in the case under consideration, the calculation of the numerical values for the criteria should be explained in detail, and the impact of uncertainties, the role of expert judgement and model incompletenesses should be evaluated.

Risk-informed safety classification of components

The systems, structures and components of nuclear power plants are grouped into several safety classes according to the safety regulations. In Finland, these are Safety Classes 1, 2, 3, 4 and Class EYT (classified non-nuclear). The items with the highest safety significance belong to Safety Class 1. The safety class of each system, structure and component has to be specified and it is determined by its safety significance.

The original safety classification of systems and components is based on deterministic safety assessment and has lately been a subject of criticism. It is recognised that the safety significance according to PSA may in some cases disagree with the current safety classification. In a risk-informed safety classification, the PSA results are taken into account in determining the safety importance of systems and components.

A pilot study of a risk-informed safety classification was carried out for the Loviisa NPP within the SOS-2.1 sub-project (Jänkälä 2002). The application systems and equipment were selected so that all the different safety classes and a wide range of safety importances were covered. PSA importance measures were quantified for the selected components and groups of components as needed. PSA importance measures were compared to the safety classes and to the safety importance as considered in the technical specifications, and the reasons for possible differences were studied. The PSA importance measures that are best applicable to different purposes in safety classification were selected. The risk estimates covered at least internal initiating events and the relevant part of external initiators.

Importance measures that are usually calculated in the PSAs today can be used for estimating the safety significance of a component or a system. Risk Achievement Worth or Risk Increase Factor (RIF) gives the risk increase when a component is taken out of use. If RIF is small then the safety class could be lowered. RIF can be used in considering the safety importance of a system or of redundant identical components. A problem with RIF is that in a redundant system it gives a higher importance for a more reliable component, which is not reasonable if a change to the safety class is considered. The Fussell-Vesely importance of an item i is the share of the probabilities of those minimal cut sets (MCS) that include i. FV gives the relative risk reduction when the probability of item i is decreased to zero. It gives also the relative risk increase when the probability of item i is doubled. It can be used when considering changing the safety class.

The effect of the safety class on the systems and components is substantial starting from the design, manufacturing and construction and their quality assurance and extending to technical specifications, inspections and tests during operation. The amount of documentation and costs linked to the equipment increases dramatically as the safety class is raised.

The effect of the safety class on the reliability of a component is not clear in all respects. The qualification of a component to the operating conditions has to be shown by tests for classified components, and therefore it is known that these components will operate in accident conditions as designed. In the sense of reliability and risk analysis it is known that the reliability parameters obtained in normal conditions for the qualified components can be applied. The situation is different for non-classified components, since their operation in harsh accident conditions is not guaranteed in a similar way. In many cases, where the environment is not different under accident conditions, the conventional components out of large batches and with extensive operating experiences may be more reliable than components of a small batch of qualified components.

Basically the same preventive maintenance actions are performed for both classified and non-classified components, but only specified and qualified spare parts can be used for classified components, and their testing is more comprehensive. This leads to the fact that the failure probability due to maintenance and ageing problems should be lower for classified components. Also, the possibility of common cause failures should be smaller for classified components.

However, no convincing empirical studies exist to demonstrate how much the failure probabilities of different safety class components differ. It can only be assumed that the failure rate of a higher class component is smaller than that of a lower class component. But it is not known if the difference is significant or negligible. This can be even vice versa in some cases.

If the safety class affects the test interval of a component the effect on the unavailability of the component is approximately known, because there exists evidence that most component failures tend to be more time-related than demand-related. Time-dependent behaviour dominates with long test intervals.

It is known that design, installation, maintenance and ageing problems contribute significantly to the common cause failure probabilities. Therefore it can be concluded at least for this pilot study that common cause failure rates are lower for higher class components. The test interval effect should be quite clear, too: the longer the test interval the higher the CCF unavailability. Testing schemes affect also the CCF probabilities but they do not depend on the safety class.

Considering the potential effect of a change in the safety classification of piping components one has to pose the question: If the safety classes of such components are changed into a lower class are the rupture frequencies increased due to lower quality requirements and less inspection and control? Piping failure data collection systems can hopefully help answer such questions in the future. The risk-informed in-service-inspection programmes are also promising.

A wide range of importances were found in this pilot study within all the different safety classes, indicating that a reclassification should be considered. These importance ranges of the different safety classes are overlapping so that non-classified components can have larger importances than Safety Class 1 components, among primary circuit components, primary coolant pump seal water, high pressure safety injection, component cooling and sea water systems.

As a conclusion it can be said that it is possible to change safety classes into both directions without endangering the safety or even by improving it. Such an exercise should be started from the most extreme importances, aiming at allocating the limited resources to points where they are most needed or useful, and not wasted in less important systems or components. It is also worth while to apply this when modifications and new systems are designed.

Pipe break frequencies for risk-informed applications

Estimates on pipe break frequencies are needed both in PSA models as initiating event estimates, and in risk-informed in-service inspection applications to evaluate the piping degradation potential. There is interest to update the pipe rupture frequencies used in PSAs, because they often originate from the reactor safety study WASH-1400 dating from nearly 30 years ago, and their applicability is sometimes criticised.

In the NKS/RAK-1 project (Andersson 1998), the LOCA frequency estimation was considered and a probabilistic fracture mechanics model for estimating IGSCC (inter-granular stress corrosion cracking) degradation was developed. It was concluded that both modelling and database approaches should be further developed and applied, and the co-operation between PSA experts and fracture mechanics experts should be promoted.

The risk-informed in-service inspection (RI-ISI) applications have become increasingly attractive due to the possibility to re-direct the inspection efforts in an optimal way, leading to both an increase in safety and a decrease in inspection costs. The basic idea is to reduce inspection activities in locations with low risks and identify the riskier locations where the inspection efforts should be concentrated. The main inputs for the decision making are the estimates of the consequences derived from the probabilistic safety assessments (PSA) and pipe rupture probabilities. The implementation of RI-ISI methodology calls for more accurate estimates of pipe rupture probabilities related to various degradation mechanisms in order to enhance the basis for the decision making.

A comparative analysis of pipe failure probabilities due to stress corrosion cracking based on two alternative analysis methods was performed in this SOS-2 project (Simola et al. 2001). It is recognised that pipe break frequencies calculated with fracture mechanistic models and those estimated from the operating experience may be quite different. There have been no studies where the numerical results for the same piping components were evaluated.

The aim of the study was to identify the advantages and weaknesses of the approaches and to give recommendations on the applicability and restrictions of the approaches. 28 pipe welds were selected in the main circulation system and the shutdown cooling system of Barsebäck 1 plant for the comparative study. The welds were selected to include significant variation in the loads. The leak and rupture probabilities for these welds were evaluated with probabilistic fracture mechanistic codes (Brickstad 2001). The corresponding values obtained in the BLAP-project could be found from ref. (Lydell 1999).

The two approaches have slightly different purposes, and thus the models and methods are selected in a different way. The methods do not address the same aspects of the phenomena under analysis, which leads to differences in the results and in their interpretation. Whereas the BLAP approach aims more at better LOCA-frequency estimates, based on extensive use of operating experience from nuclear power plants, the PIFRAP approach concentrates on explicit probabilistic modelling of failure mechanisms and the effect of inspections. Both approaches give estimates for the location or weld-specific pipe rupture probabilities. Thus, both approaches should support the applications of risk-informed in-service inspection.

One of the objectives of the BLAP method is to utilise the extensive international operating experience collected from nuclear power plants. Statistical analysis is a natural approach for reaching this objective. Statistical analysis is often a kind of black-box model, in which all aspects are not directly addressed. PIFRAP directs towards explicit modelling of crack growth, failure mechanism and in-service inspections.

An example of the results is presented in Figure 5. This example shows that the results obtained with the fracture mechanics code have more variation than those estimated from the statistical data. This is obvious, because the model uses more weld-specific data, such as stresses. Both approaches give the lowest rupture frequencies for welds B14 and B15. This is due to the material properties (316L-NG) that are accounted for in both approaches. In PIFRAP, it is judged that no SCC can occur, and consequently, the cut-off value 10^{-11} is given as the result. In BLAP, the "nuclear grade" material is accounted by a factor 20 for welds B14, B15 and B16. The highest rupture frequencies calculated by PIFRAP (welds B1, B2, B5, B6, W1 and W7) are explained by the presence of vibrations. The vibrations are not accounted for in the BLAP results. In the cases where SCC is the only degradation mechanism, the BLAP results are higher than those calculated by PIFRAP.

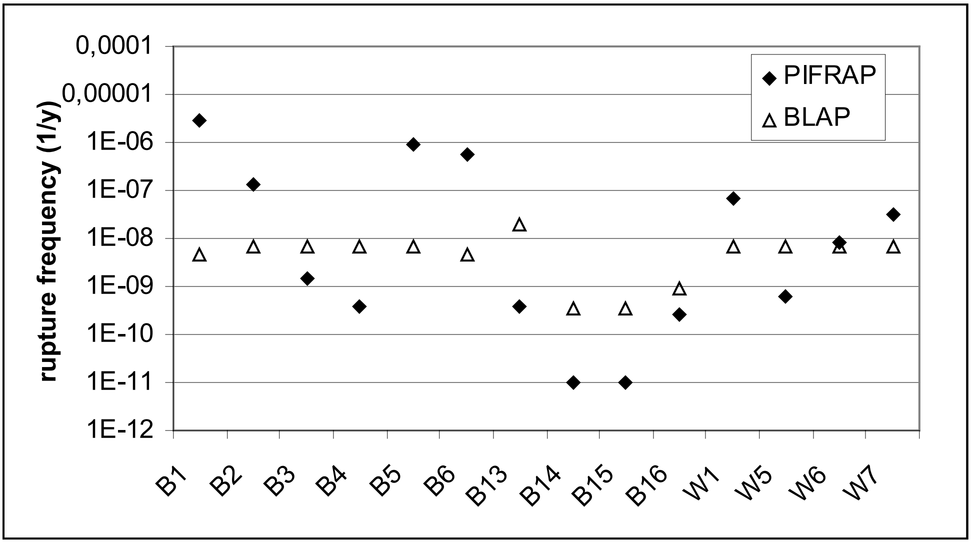


Figure 5. Calculated rupture frequencies for analysed DN 100 piping in system 313.

The results of the rupture frequencies obtained by the two alternate approaches are quite different, but one approach does not give systematically higher values than the other one. Some of the differences may be explained by different expert judgements made while applying the models. Some are related to the different possibilities to account for example for weld-specific information. In some cases, the results are not directly comparable because the considered degradation mechanisms are not the same. For instance, the effect of vibrations is accounted for in the PFM modelling whereas the statistical approach considered only the IGSCC.

The probabilistic fracture mechanistic approach requires a lot of weld-specific information about the stresses, and it may be argued that the approach may be too laborious for some applications. However, in RI-ISI applications, where the procedure requires quite detailed analysis of the piping system, the quantification of leak and break probabilities may provide additional support for decision making with relatively little extra effort. Despite the large uncertainties related to the quantification of these probabilities, a probabilistic fracture mechanistic approach in the selection of potential degradation locations can be considered as an appropriate decision support. The advantage of the PFM model is also its explicit treatment of inspection reliability, which enables sensitivity studies with different inspection policies. The principles of the statistical approach may be sufficiently accurate for the purpose of updating LOCA estimates.

Although both approaches include large uncertainties and may also require further development, they are important steps towards better quantification of pipe break frequencies. Further development of both PFM and statistical estimation approaches and especially discussions between PSA and material experts should be strongly encouraged. For example, it is recommended that the expertise of the material and structural engineer be included in the statistical approaches especially if the operating experience data is very scarce. On the other hand, the statistical evidence collected in the structured database may support the development of fracture mechanistic models.

SOS-2.1 conclusions and recommendations

Based on the previously described studies on uncertainties and on the interpretation and use of PSA results, some general conclusions and recommendations can be given.

It is in the interest of both the utilities and regulatory bodies to increase the use of PSA in decision making. Although PSA is a rather established methodology, its different qualitative and quantitative results are sensitive to assumptions, modelling choices and the purpose of PSA. This may not encourage the use of PSA in decision making. Straightforward or blind application without consulting PSA analysts and without understanding the limitations of the model may lead to biased decisions and ineffective use of resources. In risk-informed applications, the purposes for which the results will be used should be defined and the question of whether the degree of detail in the modelling corresponds to the needs of the decision case should be asked.

A PSA is a complex model for estimating probabilities for very rare events. There are many types of uncertainties, and some of them cannot be reduced. It is essential that the uncertainties are identified, and their importance is evaluated. When PSA results are used in decision making, the importance of the uncertainties should be evaluated from the angle of the decision case.

The risk-informed PSA applications require usually many types of expertise from differing disciplines. The experts' view upon PSA and its uncertainties and limitations may differ significantly. This emphasises the importance of documentation of analyses and communication between the analysts and decision-makers, as it was highlighted in the case studies. There is a need to harmonise the PSAs to some extent and to develop and adopt better documentation systems to facilitate the presentation of the results.

This subproject dealt basically with level 1 PSA applications. However, it is obvious that the situation is the same for level 2 PSA. Thus, it would be recommended to compare level 2 PSAs, as well as analyses of area events and low power PSAs. These comparisons would presumably raise other types of methodological questions than the comparison of basic PSAs.

Management of plant maintenance and renewal (SOS-2.2)

The significance of maintenance to the safe and economical operation of nuclear power plants has been identified in several studies. The maintenance issues include a variety of problems: the quality of maintenance depends on human and organisational factors; poor quality of maintenance and operability verification may result to latent failures preventing the operation of the safety systems; the ageing plants require well targeted maintenance strategies and properly defined modification programmes. In addition, the modernisation of the ageing units calls for a complicated transfer of the organisation and routines from the “operation culture” to a “construction culture” at the sites and authorities.

Sub-project SOS-2.2 focused on questions related to maintenance and renewal of nuclear power plants. In the previous NKS/RAK-1 project, maintenance-related issues were studied in several sub-projects, covering selected topics in maintenance strategies, modification and modernisation works, and the reliability of non-destructive testing. SOS-2.2 utilised the results of RAK-1 in further development of approaches for studying maintenance-related problems and strategies. The sub-project aimed at giving recommendations for improving safety and economy in maintenance work and management.

The objective of this sub-project was twofold. On the one hand it aimed at promoting better quality assurance of maintenance works by addressing questions related to human and organisational factors in maintenance. On the other hand, the sub-project focused on improvement of maintenance strategies and decision making.

Quality assurance of maintenance was addressed by analysing occurred errors related to maintenance, especially those meant to be identified before or during the plant start-up after maintenance outages. As a continuation to the human HCCF study started in RAK-1, studies and developments related to the Loviisa plant case were performed. A survey on the needs for human factors research related to maintenance in Sweden was also carried out.

The following activities took place within the other topic, focusing on decision making of maintenance and modifications. A discussion and working group of utilities was established to exchange and develop information on maintenance optimisation and maintenance strategy classifications. A review of the status of management of condition monitoring information for maintenance steering in the Nordic countries was done alongside a review of condition monitoring of transformers. The development of supportive condition displays for improving operative maintenance decisions was also partly supported by the sub-project.

Human aspects in maintenance quality

The focus in human reliability analysis relating to nuclear power plants has traditionally been on control room operator performance in disturbance conditions. In the area of maintenance activities, the emphasis has been on human reliability of non-destructive inspections. On the other hand, some studies and incidents have shown that errors related to maintenance that have taken place earlier in plant history may have an impact on the severity of a disturbance, for instance by disabling safety related equipment. Operational experience has shown that faults have passed inspections and functional tests after maintenance and modification activities (Laakso, Pyy & Reiman, 1998). The causes of these failures have often been complex event sequences involving human and organisational factors. Especially common cause and other dependent failures of safety systems may significantly contribute to the core damage risk.

In this project, one aim has been to promote the analyses of human factors related to maintenance. The topic was addressed in Finnish studies of human common cause failures (HCCFs) where operating experience on maintenance- related human errors are analysed in detail. The current activities and research needs in Sweden were identified in a survey where all Swedish plants and the authorities were interviewed. In addition, a Nordic preparation project on operability verification was conducted, but the activities were pursued in smaller national working groups.

Study of human common cause failures in relation to maintenance activities in Finland

Pilot studies for identification and analysis of human common cause failures (CCFs) in relation to the maintenance activities have been conducted for both Finnish nuclear power plants. The analysis of three years of experience at Olkiluoto covered 4400 fault repair work orders where 334 human errors were identified. The number of dependent human error events derived and analysed was 23. In a parallelt study of the Loviisa plant, the corresponding numbers were 14091, 231 and 39, respectively. The analysis results of the Olkiluoto plant study were summarised within the RAK-1 project and published in 1998.

A detailed classification model for human errors related to maintenance activities was developed and adopted for trial use within this SOS-2 project and as a part of the Loviisa study (Laakso 2000). The purpose of this new classification is to provide an enhanced basis for identification and statistical analyses of maintenance-related errors, and especially human CCFs, utilising the maintenance database.

After the preliminary verification of the dominating human error classes with the plant staff a new classification was confirmed for re-analysis purposes. The developed new classification model of human and quality errors related to maintenance was first tested and refined by identification and review of about 100 human errors with the Loviisa plant maintenance staff. The proposal for classification of direct causes of human errors is presented in Table 4.

Table 4. Classification of direct causes of human errors related to maintenance.

Errors of Omission

1. Restoration errors after work, such as omission of the realignment of process or instrument valves, breakers or limit settings.
   Omission of refilling of fluid or gas into lines or tanks.
2. Cables or electronic components not connected, settings/adjustments omitted or omission to install packings.
3. Foreign objects or impurities left behind inside the object of the work. Examples are dirt, garbage, tools, scaffolds or covering material.

Errors of Commission

Wrong order or direction,

4. Wrong order, such as cables or instrument pipelines connected crosswise.
5. Wrong direction, such as reversed or twisted installation of valve or another sub-component, or wrong positioning of valve.

Wrong selection,

6. Wrong place or object, such as cabling fixed on wrong connection, setting of wrong tripping conditions or draining of wrong pipeline.
   Item installed on wrong component place.
7. Wrong or mixed parts, materials, tools, fluids or chemicals selected for work.

Wrong settings/adjustments/calibrations,

8. Wrong settings of trip limits, limit switches, reference, indication or time delay values, or adjusting devices.
   Deficient alignment of shaft, stem/spindle or pipe. Wrong setting of pipe support.

Other quality problems,

9. Too little force, e.g. loose connections of bolts or cables.
10. Too much force, e.g. excessive tightening or greasing.
11. Damaging other equipment e.g. cabling, cable trays or small diameter piping by slugging/contacting. Possibly due to carelessness and narrow spaces for work or transport.
12. Other carelessness (if 1-11 are not applicable), e.g. worn tools, falling, dropping or intrusion of foreign material, deficient weld, solder joint or insulation. Unclear trips initiated during testing or maintenance, wrong timing.

In addition to the observed and direct error mechanisms also the underlying contributing factors are studied and classified for the dependent human errors according to another new classification. The root causes can be assigned to one of the following groups:

- Planning deficiencies: Incorrect, incomplete or unclear work planning, procedure, work order or operation order. Deficient definition of work scope.

- Design deficiency: Error or deficiency in design or documentation of modification, equipment, system, installation or computer program. Documentation not updated.

- Violation of procedure or order: Violation due to insufficient knowledge or poor information. Deviations from procedure or order due to gradual organisational learning of “bad habits”. Or conscious violation.

- Poor co-ordination, supervision or information transfer: Poor project co-ordination or supervision of subcontractors, poor information transfer due to organisational changes or boundaries. Or weaknesses in experience feedback or quality control.

- Insufficient knowledge: Deficient training or specialist knowledge.

A good quality of experience data is helpful in the identification and analysis of deficient operability verifications and near-misses. A structured classification and systematic analysis facilitate the identification of failed barriers and the error mechanisms that have penetrated them. One target of the study was to identify the dependent human error mechanisms and to search for causes of non-detection of the errors in the operative and organisational defensive barriers. This analysis was done in cooperation with plant maintenance and operability experts in order to capture the tacit knowledge and close the operating experience feedback loop more tightly. The dependent human errors were summarised in condensed maintenance event reports, which include a qualitative description of:

- multiple error and its failure consequence,

- originating erroneous or defective work task, and

- primary deficiency in operability verification, allowing the errors to remain latent in the system e.g. throughout the start-up testing programme after the plant maintenance outage or for a lengthy time period.

An example of a maintenance event report is presented in Table 5.

Table 5. An example of a maintenance event report.

Marking	Work order time and number	Title and description of event	Operating event marking
1HCCFYP12	1996-10-08	Deficient adjustment and testing of the actuators as implementing new motor operated blowdown valves in the pressurizing system	No
	238769D, 238769A 238769B 238769C	The gate valves 12YP12S038, 12YP12S039, and 11YP12S036, 11YP12S037 were not tight in hot state during power operation 1996-10-08. New motor operated gate valves (MOVs) had been installed during the preceding maintenance outage in September 1996. The MOVs had to be closed as a corrective action by manual operations from the switchgear during the power operation state at 1996-10-09. The common cause setting errors had passed through from the maintenance outage to the power operation period because the setting of the limit and torque switches of the MOVs in the cold state only was insufficient as start-up testing of the modification work.	

The review of the studied set of the multiple error events shows that plant modifications are a significant source of common cause failures. The review also shows that work planning in a complex planning environment of different requirements and objectives and calling for multifunctional plant knowledge is a very demanding task. The dependent human errors originating from modifications could be reduced by improved specification and coverage of the start-up testing programmes.

A reduction of the high number of technical modifications could also be achieved by utilisation of decision models and decision analysis sessions on proposed modifications. These provide a more systematic basis and clear documentation for selecting the best decision option under multi-criteria conditions (Laakso et al. 1999).

Improvements can also be realised by itemising and planning the installation inspection and functional testing phases in the work orders for significant maintenance items and modifications. For the most significant maintenance items and the technical modifications a review of the work planning documents should be introduced. The review is supposed to enhance the quality and understanding of these instruments for the performance and steering of the actual works.

In addition, the detailed classification and analysis of the maintenance-related errors provide good material for training of the maintenance personnel, and for checking how the identified multiple human errors are covered in the common cause failure modes and data in the PSA studies of today.

A detailed classification and analysis of maintenance related errors provide also valuable information for the focusing of e.g. psychological studies on the most relevant aspects in maintenance activities.

A survey on research needs in Sweden

The aims of the study (Salo & Svenson 2001) were to review research and development projects related to human factor issues in maintenance during the past few years in Swedish NPPs and SKI, and to identify future research needs. The personnel, representatives of maintenance and human factors, were interviewed about the projects that they consider important and illustrative of this kind of work at their plants. The people working in maintenance were more specifically asked in an open structured interview to speak about what they related to human factors and maintenance work. The plant personnel was asked to identify future research and development projects that they consider important considering human factors and maintenance.

All four Swedish NPPs and the Swedish Nuclear Power Inspectorate (SKI) participated in the study. An interview group was formed at each location. The composition of the interview groups varied from site to site depending on the availability of personnel. All persons included in the groups were professionals in the areas of maintenance and/or safety.

A semi-structured questionnaire was used for the interviews. The participants were asked to give answers related to the following main categories of questions:

- (I) previous research/development projects,
- (II) ongoing research/development projects,
- (III) planned research/development projects,
- (IV) research/development projects in the future related to problems requiring solutions,

(V) additional questions and background questions with more general characteristics.

The identified areas of interest and related project proposals are listed in Table 6.

A majority of the interviews on completed, ongoing and planned research/development projects gave interesting answers on how professionals at the plants and SKI view their work and progress on issues related to maintenance and human factors. The interview answers clearly revealed rather diversified viewpoints on this matter. A number of reasons may explain these differences. For example, specific or general differences between the sites (e.g., technological, organisational, cultural, etc.) result in different problems, different angles of approach and different solutions to the problems.

In the light of the results, it would be important to investigate which underlying processes and structures contribute to the differences in the views on human factors in maintenance and their relation to maintenance work and safety in practice. An organisational culture approach was considered appropriate for this research problem.

The interviews on the needs for future research and development projects gave interesting answers concerning which problem areas related to maintenance and human factor issues are considered important by the professionals at the plants and SKI. Because of the great diversity in the answers from the various sites, no single uniform priority problem area could be identified. Instead, a number of possible problem areas for future research and development were generated, each including one or more specific proposals for future projects, as shown in Table 6.

Table 6. Future project ideas categorised in 10 problem areas.

Problem area	Project proposal in brief
Reporting feedback	Maintenance event deviation reports Use of event reports as safety indicators Natural channels of communication
Instructions and recommendations	To improve work quality by means of improving instructions (various aspects) The role of instructions and motivation Practical recommendations for conditions for performing NDT in field conditions
Organisational	Methods of creating common platforms across sub-units to enhance communication, attitudes and safety Organisational issues such as interplay, transfer of learning in organisational change, human factors in maintenance Co-ordination between maintenance and operational personnel Shortage problems of specific maintenance personnel Leadership and safety
Competence and learning	Demands on competence, utilisation of informal knowledge Generation and competence shifts Learning in the maintenance process
Methodology	Creation of a common methodology for similar analyses Methodology for qualification of NDT personnel Safety criteria, baseline measurements to analyse effects of change Evaluations of maintenance intervals and verification of maintenance measures
Failures	Latent and hidden failures, occurrence and prevention
System specific	System 500 (control equipment) maintenance
Other MTO-related	Integration of MTO into maintenance management and work Risk evaluation for prevention of MTO events in maintenance Early planning of MTO into new activities Utilisation of MTO analyses in prevention of worker accidents Introduction of HF as a new perspective in design
Other maintenance related	Prevention of unnecessary maintenance and testing Plant modifications and maintenance
Other / general	Relation between economy and safety Change processes. Small things that make a difference International comparisons of maintenance to identify improvements Attitudes towards safety in maintenance work.

It is impossible to judge if a particular problem area is more important than another. Instead, the local circumstances at a plant must be taken into consideration in a prioritisation process. However, there is a number of specific project proposals that are considered important and possible to manage within the near future, some of them also suitable for a doctoral thesis project. They are:

- *Deviations in maintenance (event deviation reports)*. How to gain knowledge from prior errors, faults, failures, barriers, etc.

- *Attitudes towards safety in maintenance work*. Attitudes towards work and attitudes towards one's own personal role in safety. The role of attitudes in and dependency on the management of a plant.

- *The use of event reports as safety indicators.* Questionable bonus systems – for example, "the fewer the number of reported work injuries, the better". Better criteria for event reporting – usefulness. Feedback systems.

- *Latent and hidden failures*. How do they occur? How can they be prevented (instructions, event reports (feedback), cause analysis)? This is especially important during modifications. In such a context unintended modifications, e.g. routine replacement of specified components by similar components not specified, and possible consequences are of interest (for example, replacement of a computer hard drive with specified data by another with higher performance result in negative consequences to the system as a whole because of incompatibility). Studies on CCFs (common cause failures) are important here.

- *MTO in risk evaluation*. It is considered to be important to find out how MTO can be included in the evaluations of risks in the planning of maintenance work. This could, for example, include demand characteristics/specifications (what resources do we have, which ones do we need?) to ensure that proper measures have been taken towards the prevention of MTO-related events.

- *Early planning for MTO*. The earlier MTO is introduced in the activities (for example: inventions; functional testing; design; etc.), the better. It is important to find out the best way to introduce MTO as early as possible in the planning of various activities at the plant.

- *Development of methodology for qualification of NDT personnel with a view to general personnel qualifications*. The present praxis of qualifying personnel separately for every project and every task is ineffective and expensive.

- *Issue of practical recommendations*, concerning working environment and organisational conditions when performing NDT in the field.

- *A project on learning in the maintenance process*.

Maintenance management

Needs for changes in maintenance practices and planning may arise from several sources, e.g. learning from operating experience or development of novel condition monitoring and information techniques. However, operative and strategic decision making concerning maintenance and modifications is not always straightforward because of different, conflicting or even non-measurable objectives.

The aim of this task in SOS-2.2 was to enhance the use of plant information systems and the available expertise for improvement of maintenance strategies. Reliability Centred Maintenance (RCM) analyses utilising maintenance data bases and efficient data analysis tools, completed with structured decision models, were recently introduced and demonstrated at the Barsebäck plant in the NKS project RAK-1.4 (Laakso et al. 1999a). The RCM studies based on experience helped to pinpoint and justify correct maintenance and testing actions and optimal intervals for decreasing failures, repairs, wear and maintenance costs. As continuation to the pilot study, a project on experience-based RCM analyses, completed with cost comparisons of optional maintenance strategies, has pursued at the Barsebäck plant in co-operation with two other plants.

The identification of decision criteria, estimation of the benefits of maintenance actions and the use of PSA results are examples of open questions in developing maintenance strategies.

In order to exchange information and discuss good practices of maintenance management, a seminar on reliability-centered maintenance was arranged (Laakso & Simola 2000), and according to a decision by this seminar, a discussion and working group on maintenance decisions was established. This group consisting of power plant personnel involved in maintenance-related decision making and development met several times during the SOS-2 project. The scope of the discussion can be illustrated by a simplified model of target-steered maintenance management in Figure 6.

In addition to the issues of strategic maintenance decisions, this task also considered the decision support for operative maintenance actions. In such situations, information is needed about the degradation state of the systems or components. Condition monitoring aims at identifying an optimal timing and correct object for maintenance actions. Even if the condition of an equipment cannot be directly monitored, information on the degree of degradation can be produced indirectly from the process parameters. Intelligent display systems can utilise this data and provide better information on component degradation, which results to improved quality of operative maintenance decisions.

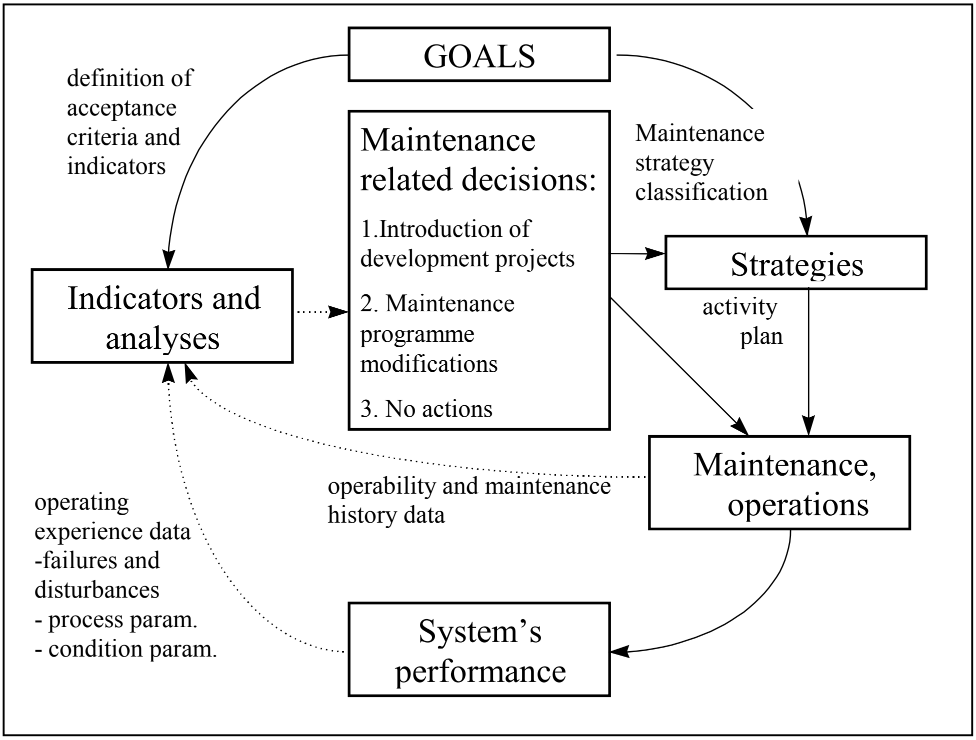


Figure 6. A model of target-steered maintenance management.

Maintenance strategy classifications

A survey on principles and examples of maintenance strategy classifications of equipment and systems was done in the SOS-2 project. The survey describes and compares the important decision objectives and criteria used at different plants in their maintenance planning, based on benchmarking of examples and ideas on maintenance strategy classifications from the different Nordic utilities and researchers.

The aim of a maintenance strategy classification is to rank the plant maintenance items for the allocation and planning of predictive, preventive or corrective maintenance actions and related resources to the correct equipment locations at the plant. For instance, a combination of the predictive and preventive maintenance can be directed to the highest maintenance class equipment that should function without interruptions through the power operation period. On the contrary, the lowest strategy class is selected if a functional failure of the equipment does not affect negatively the plant safety or availability and a planned corrective maintenance of the equipment leads to a lower maintenance cost than a preventive strategy.

Another aim of the maintenance strategy classifications is to help to prioritise the maintenance analysis work into the maintenance items, which are important from the maintenance optimisation point of view. The detailed and resource-demanding failure and maintenance data analysis is then directed to equipment exhibiting deviating maintenance indicators, e.g. a high number of failures or man-hours (Laakso & Strömberg 2001).

The structure and comparison of maintenance strategy classifications at some Nordic nuclear power plants is presented in Table 7. The classification is based on the plant-level effects of functionally critical failures at the equipment place level. As a reference, the selection strategies of the Maintenance Significant Items in the Reliability Centred Maintenance (RCM) approaches for the US aviation industries (MSG-3 1988) are also summarised.

The Olkiluoto plant utilises plant-level PSA and availability analyses to steer the selection of the maintenance strategy classes carried out by the maintenance supervisors. The Loviisa plant applies the plant life management programme to the equipment in maintenance strategy class 1. The plant has defined and prioritised clear operability and maintenance indicators and targets for the classes of different importance. The Barsebäck plant, in turn, has performed an experience-based, reliability-centred maintenance analysis (EBRCM) on four technical systems prioritised as important for maintenance programme review and optimisation (Laakso & Strömberg 2001). The maintenance programme optimisation could, however, be completed by an optimisation of the price of the resources paid for the importance of the equipment place.

The similarities and differences of the maintenance objectives and strategies defined for the various plants could be identified and discussed. Furthermore, the means for measuring of the results and the effectiveness of the maintenance, as well as the related maintenance optimisation work, were compared and documented for learning. Some gaps in the hierarchical maintenance management system were identified between the fundamental maintenance objectives (decision criteria) and the attributes (means objectives) as well as between the maintenance objectives and the indicators. Some differences are indirectly visible for the studied plants from the Table 7, especially concerning the attributes and indicators related to plant and maintenance economy and plant-level safety, which were estimated to require further benchmarking.

Recognisably, the present comprehensive maintenance strategy classifications of the plants are big steps towards ensuring and steering the achievement of operability with a view to plant safety and availability. The next step in the maintenance strategy development is to optimise the maintenance by enhancing actions of important plant objects and reducing maintenance costs on equipment of less maintenance importance. A selective and experience-based RCM analysis of the bulk of the failure and maintenance history data is recommended to support the optimisation.

Table 7. A structure of various maintenance strategy classifications.

Strategy	Objectives and criteria	Attributes	Indicators	Classes
Loviisa	Reactor safety Production availability Procurement or repair cost	LCO/AOT[1] Production loss Cost	Number of AOT repairs Loss of MWhs Number of repairs CM/PM[2] ratio Component unavailability	3, ranking based on additive value function
TVO	Reactor safety Plant availability Maintenance cost Operating experience	LCO/AOT Risk importance measures Production loss Repair cost Failure history	Loss of MWhs Nr of faults + trend Maintenance cost	4 , classes based on plant safety, availability or cost requirements
BKAB	Reactor safety Production effects Work environment	LCO/AOT Plant operational state Radiation zone	Nr of faults + trend PM cost + trend CM cost + trend	4, based on deviating indicators on high attribute scores.
VTT/ BKAB pilot study	Reactor safety Radiation protection Reliability Repair costs Preventive maint. costs	FSAR safety class Risk increase factor Radiation zone Nr of faults+trend CM manhours PM manhours	Nr of faults + trend PM man-hours CM man-hours	Ranking equipment importance based on use of additive value function (Criteria weighting, scoring attributes)
RCM/ MSI[3]	Operational safety, Operational capability, Significant support functions Maintenance cost	Potential failure modes, Occurred failure modes	Failure detection method, Failure characteristics	3, Preventive maintenance strategy, repair strategy or redesign

[1] LCO = Limiting Conditions for Operation, AOT = Allowed Outage Time
[2] CM = Corrective Maintenance, PM = Preventive Maintenance
[3] RCM = Reliability Centred Maintenance procedure, MSI = Maintenance Significant Items

Condition monitoring of transformers

Transformers are equipment having a high importance for the availability of power plants. Transformer failures are nearly always expensive, as an example the costs of an explosion in a transformer in Norway were estimated to reach NOK 30 million. Thus, a study on transformer condition monitoring possibilities, with the aim of early fault detection using the existing information, was judged to be of interest for the utilities.

The most important parts of the transformer are the active parts and the isolation material. The active parts consist of the iron core of metal sheets and the windings in either copper or aluminium. The isolation material consists of cellulose-paper, which acts both as isolation for the winding and as isolation between the windings. The transformer oil acts both as isolation material and as a cooling medium. Based on an interview with experts in the transformer business it was deduced that the most serious failures in transformers are explosions and fire. In case of local hot temperatures in the transformer, explosive gases will develop and this can lead to explosions and fires. Local hot temperatures can occur in case of a short cut due to degraded isolation material but also in case of lightning, perhaps combined with degraded isolation material. The focus in condition monitoring seems to be on the gas content of the oil and on the isolation material.

As the number of transformers is increasing and existing transformers are getting older, there is a need to know more about how to determine the condition of transformers. In this light, a Danish company ELTRA has performed a study of condition monitoring of transformers “ Tilstandsvurdering af transformere” (ELTRA 1998), which was reviewed within the SOS-2 project (Paulsen 2000a) and is briefly summarised here.

The report gives recommendations on condition monitoring of the isolation of the transformer windings based on tests on oil samples. The oil test is used to measure the quality of the oil and the content of gases in the oil. The timing of oil change is based on the results of this analysis. The content of gases in the oil is an indicator about the condition of the transformer.

It is recommended to use inhibited oil and to take oil samples regularly for the purpose of oil and gas tests. The interval for taking samples is 1-5 years depending on the transformer type. The following measurements are recommended for the samples: punch through voltage, water content, acidity, surface tension, and dissipation factor. If any of the preset reference values are exceeded a new sample should be taken for control of the measured values. The inhibitor content should be checked and, eventually, an oxidation test done. The decision alternatives are to change or regenerate the oil or plan the next test.

The content of dissolved gases should be measured from the oil samples taken routinely. If the content of one or more gases exceed the limit for the normal value for these gases, a sample should be submitted for a GC-analysis. If the content of one or more gases exceeds a value indicating with a certain probability that there is a failure in the transformer, some more tests and measurements should be performed for assessing the failure type and its location.

A list of on-line condition monitoring equipment, as well as literature references related to transformers can be found in the summary report.

Survey on management of condition monitoring information for maintenance steering

In a modern operating and maintenance organisation for nuclear and advanced power plants, condition monitoring plays an important role in the steering and optimising of the maintenance actions and work. Thereby, condition monitoring helps to control the economy and safety of the operation. Condition monitoring of equipment and systems can be used as a part of the systematic maintenance strategy. A systematic maintenance strategy is a prerequisite for an optimal selection and mixture of predictive maintenance, preventive maintenance, corrective maintenance, and functional testing. The development of condition monitoring methods and information technology together with the current more stringent economic and safety requirements for power plants steer the focus of the maintenance strategy more and more towards condition-based maintenance.

Selection and presentation of condition-sensitive information from the accumulating measurement, process and control data in process computer systems was addressed as a potential development area at several plants already in 1995 in the NKS/RAK-1.4 interviews. Large amounts of condition-related data were automatically collected into different computerised systems or stored into different individual computers. However, much of this information at the plants seemed not to be available for off-line and on-line condition diagnosis in any usable form, e.g. comparison of fingerprints, evaluation of trends, or combination of different measurement data for an integrated analysis and prevention of subsequent failure types.

Condition monitoring is often understood as equivalent with vibration measurements on rotating machines and use of other special equipment for measuring the condition of oil, and measurement of electric parameters only. However, condition monitoring can also use chemistry or process data for assessment of e.g. system performance or the condition of equipment such as piping, condensers, other heat exchangers or pumps. A better integration of condition monitoring and process measurement data for follow-up of deviations and degradations like fouling, wear-out and leakages in the main equipment and sensitive auxiliary items could steer the timing decisions of operative maintenance actions before functional failures, disturbances or damages occur. It could also help in replacing or relaxing disassembly inspections, preventive maintenance and periodic tests by condition monitoring.

One aim of the ongoing or future condition monitoring developments is to contribute in making all systems and means for condition monitoring more visible for the maintenance staff and operators. In this way correct maintenance actions can be planned and operative and correct timing decisions made in case of identified deviations from the normal condition (Laakso et al. 1999b).

Interviews were performed at three nuclear power plants about how condition monitoring information is collected, how the information is analysed and used for the supervision of the condition of equipment or plant systems and how condition-based maintenance is performed, planned and should be developed.

The questions addressed are presented in Table 8. The main conclusions from the interviews are summarised in the following.

Although various condition monitoring systems are in use at all plants, the linking of condition monitoring data to the maintenance information system has not been done. Some plants show a clear interest to develop the systems so that integrated use of the various information could be facilitated. For example, in the information system of one plant, the preventive maintenance, periodic testing, condition monitoring and NDT actions have been compiled to total preventive action programmes covering the equipment places.

The reduction of maintenance activities has in some cases been possible based on the condition monitoring measurement results, and all plants could give examples of extending the maintenance or overhaul interval of equipment. It was pointed out that the extension of the periodic testing intervals of safety-related equipment has to be accepted by the safety authority. These decisions can not be made by the plant alone.

None of the interviewed plants had a defined predictive maintenance strategy for implementation of condition monitoring as a part of the overall maintenance strategy. Although the development towards integrated use of condition monitoring information for maintenance steering is still rather slow, condition monitoring was seen as an important part of the maintenance strategy of ”class one” equipment, i.e. that with high significance for safety or availability. A combined strategy of condition monitoring and disassembly inspections is needed for the most important equipment, however, to reduce the risks of detecting surprising faults contributing to unplanned outage extension.

The aim of the predictive maintenance strategy is correct timing and direction of the maintenance actions, taking into account also production and resources. The maintenance needs are identified from condition measurements and other collected data. The maintenance decisions are made proactively based on utilisation of this data, including failure history and personnel expertise as information.

Table 8. Questionnaire for condition monitoring management.

Questions addressing the use of condition monitoring in NPPs	
Long-term maintenance decision making	*Short-term maintenance planning*
1. How is condition monitoring data linked to the maintenance information system? Is it analysed off-line together with other failure data?	4. How is the existing process and control data for process operability monitoring utilised for on-line condition monitoring of equipment? What good examples do you have at your plant?
2. How can periodic testing or inspection be reduced by implementation of condition monitoring? Do you have examples or ideas from your plant?	5. Do you utilise display systems for on-line condition monitoring of equipment by using existing process and control data (without installation of extra measurements)? How can the displays and diagnosis be improved?
3. How is the implementation of condition-based maintenance linked to the strategic maintenance decision making? What decision criteria or decision support methods are used?	6. Are decision models utilised for determining the timing of operational maintenance action when an incipient failure indication is observed? How reliable are such indications considered? What kind of decision principles on operative actions do you have at your plant, please give examples?

The use of the process and control data for process operability monitoring is very limited for purposes of on-line condition monitoring of equipment. One technical limitation is that access to the process computer information from the outside is prohibited due to security reasons.

Finally it is noted that a development and implementation of decision models for condition-based maintenance steering and operational decisions would promote the proactivity in maintenance management.

Advanced display systems for maintenance planning

Advanced displays could be used for maintenance steering by using the component operating points or system characteristics to indicate the condition. One aim is to make all the condition monitoring systems visible for the maintenance staff and for the operators, so actions can be taken in time in case of deviations from the normal condition (Paulsen 2001b).

The display design follows a strategy that is based on the following questions:

- What is the task of the display or system?
- What are the problems of the plant or system?
- How do the problems occur?
- How are the problems observed?
- How can the problems be presented for the user?

A condenser and vacuum system is considered as an example of development of a display system of condition monitoring information. The efficiency of a power plant is mainly dependent on the condenser pressure, and thus it is important to follow the functioning and condition of the condenser.

The strategy used on the condenser is presented in Table 9.

Table 9. Strategy used on condenser.

What is the purpose of the condenser?	Keep pressure low
What are the problems?	Temperature increase in the condenser Non-condensable gases in condenser
How do the problems occur?	Ejector problems Cooling problems Incoming air Fission gases
How are the problems observed?	Increase in condenser pressure
How should they be presented for the users?	Actual operating point and expected operating point relative to the steam saturation curve

The display developed for the condenser is shown in Figure 7.

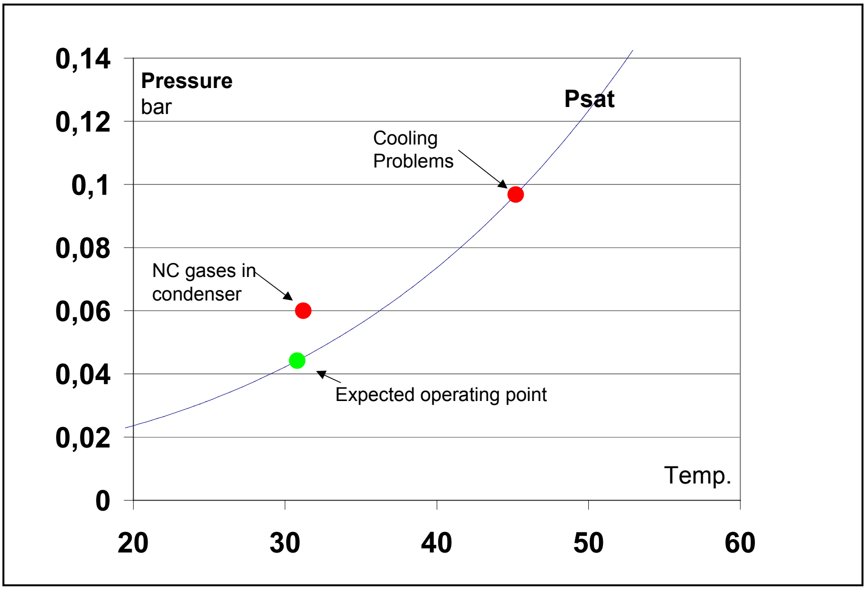


Figure 7. Condition monitoring display for the condenser.

The aim of the display is to show the deviations from the expected operating point of the condenser and the actual operating point. Furthermore, the display should be able to show which type of degradation or fault has occurred. Cooling problems and fouling on the heat transmission tube surfaces give an increase of the temperature in the condenser and thereby in the condenser pressure, but the operating point will still lie on the saturation curve. Non-condensable gases are expected to increase the pressure in such a way that the operating point will lie above the saturation line, but the gases will also give some increase in the condenser temperature due to a decrease in heat transmission. Fouling on the heat transmission surfaces is expected to increase the pressure in the condenser more slowly than the pressure increase due to failures in the ejector system or to an incoming flow of air. Tests on a Danish power plant where the ejector system was stopped for half an hour showed a fast increase in condenser pressure due to the increase of non-condensable gases.

The present display has importance for maintenance planning of the condenser and for optimisation of the thermal performance of the plant. The display is implemented for tests at the Forsmark 3 simulator at the Halden Reactor Project in Norway. The idea of using the component characteristics is also used for condition monitoring of other types of components in the ongoing project. The display tests will show if this type of presentation is well understood by the operators and if it sensitive enough to observe incipient failures. Replaying data from Barsebäck has yielded promising tests. The development of a strategy for design of this type of visualisation will continue in the future.

SOS-2.2 conclusions and recommendations

The overall objective of all Nordic nuclear power utilities is to assure the economically competitive electricity production of the plants in long-term. The target of the optimisation of maintenance activities is therefore to reduce the unnecessary maintenance costs and steering the correct maintenance actions and resources to the important equipment. This should be done without negative effects on the plant safety and by developing it selectively. The work within the sub-project SOS-2.2 has been conducted in close co-operation with the power plants. This applies both to the analyses and surveys on human factors in maintenance, and the studies related to maintenance management.

As emphasised in the human common cause failure studies in the previous NKS/RAK-1 project and in this SOS-2.2 sub-project, as well as within the study on commission errors in SOS-2.1, more attention should be paid to maintenance related human and organisational factors. A review of multiple error events shows that plant modifications are a significant source of common cause failures. These failures could be reduced by an improved specification and coverage of the start-up testing programmes and better planning of the installation inspection and functional testing phases in the work orders. The survey on research needs in Sweden identified several research needs in this area, and it was recommended that the regulatory body would have a responsible post for maintenance related questions.

The management and development of maintenance were addressed by creating a discussion and working group on maintenance decisions within the NKS-framework. The group exchanged information, identified differences in their maintenance management and discussed about good practices. An optimal allocation of predictive, preventive and corrective maintenance, functional testing and modifications should be achieved by a living maintenance programme in an ageing plant. The plant life management programmes could be completed by asset management principles, combining maintenance investment and risk analyses.

The current status of the use of condition monitoring information for steering of maintenance was identified in interviews at several nuclear power plants. It can be concluded that a systematic utilisation and analysis of failure, maintenance, process and condition monitoring data, and an integrated utilisation of the condition monitoring methods, could significantly help to increase the proactivity in maintenance. However, this development towards condition-based planning of maintenance and operative actions is quite slow, even if extensive condition monitoring methods and equipment have been acquired for the plants. An effective use of process information for condition monitoring would require better access to this information for maintenance personnel. The use the process and control data is also a prerequisite for the adoption of advanced displays for identification of deviations in component operating points or other incipient faults for condition monitoring of equipment.

Severe accidents (SOS-2.3)

In the previous NKS project on severe accidents, RAK-2, the main phenomena studied were related to in-vessel melt progression and core coolability (Lindholm et al. 1997). The final report of the RAK-2 project recommended the consideration of some selected phenomenological issues in the following Nordic research programme. In the planning phase of the SOS-2 project, hydrogen combustion in a BWR reactor building and PWR containment, and the behaviour of organic iodine were identified as the main topics to be studied. Moreover, there was an interest to review the current status of severe accident research and management in the Nordic countries. This topic was addressed both by arranging a seminar for exchanging information and writing a current report (Frid 2002). A recriticality study of a degraded BWR core due to steam explosion in the lower head was included in the project at a later phase of it.

Current status of the severe accident research and management

The status of severe accident research and accident management development in Sweden, Finland, Norway and Denmark was reviewed in this task. The emphasis was on severe accident phenomena and issues of special importance for the severe accident management strategies implemented in Sweden and Finland.

The main objective of the severe accident research has been to verify the protection provided by the accident mitigation measures and to reduce the uncertainties in risk-dominant accident phenomena. Another objective of the research has been to support the validation and improvements of accident management strategies and procedures as well as to contribute to the development of level 2 PSA, computerised operator aids for accident management and certain aspects of emergency preparedness.

The development of severe accident management has embraced improvements of accident mitigating procedures and strategies, further work on Computerised Accident Management Support (CAMS) system at Halden, as well as plant modifications including new instrumentation.

Severe accident management efforts in Sweden have mainly concentrated on improvements and development of accident management procedures. Examples of these activities are the development of knowledge-based handbooks for accident management strategies for the Emergency Control Centre at Forsmark and the implementation of Westinghouse Severe Accident Management Guidelines at the Ringhals PWRs.

In addition to the development of accident management procedures a number of plant modifications have been made or planned in Finland. Good examples from Olkiluoto are the containment pH control and upgrading of the containment sampling system, and the decision to protect the leak tightness of containments in case of ex-vessel steam explosion by strengthening the lower drywell personnel access lock. From Loviisa, one could mention the implementation of in- vessel melt retention by means of ex-vessel cooling and the installation of new motor-operated pressuriser relief valves at Loviisa.

Nordic severe accident research can be deemed comprehensive and addressing all important severe accident issues in practice. Co-operation between the Nordic countries plays an important role in this context. Both the in-vessel and the ex-vessel accident progression phenomena and issues are investigated. Even if there are differences between Sweden and Finland as to the scope and content of the research programmes, the focus of the research in both countries is on the same topics. These are in-vessel coolability, integrity of the reactor vessel lower head and core melt behaviour in the containment, in particular the issues of core debris coolability and steam explosions. Other important areas where uncertainties are significant, and where continued research is recommended, are thermal-hydraulic phenomena during reflooding of the partially degraded core, fission product chemistry, in particular formation of organic iodine, and hydrogen transport and combustion phenomena.

The remaining uncertainties in important severe accident phenomena have been identified and recommendations for future research have been given. It should be noted, that in addition to uncertainties connected to a particular severe accident phenomenon, there is uncertainty connected with the accident scenario itself. Thus, in many cases the uncertainty in predicting the containment performance is mainly related to the initial conditions resulting from uncertainties in in-vessel accident progression. Significant progress has been made in understanding melt behaviour and melt-structure interactions in the lower head of the reactor pressure vessel. However, the in-core accident progression, modes of melt relocation from the core region to the lower plenum and melt composition will influence the lower head phenomena to a great extent. As a consequence, bounding deterministic analysis, sometimes combined with probabilistic elements, is often used in assessing the risk significance of particular severe accident phenomena or issues.

Understanding of the above mentioned phenomena, as well as of the integrated behaviour of the plant during an accident, is necessary for the development of accident management strategies (organisational aspects and human performance also play an important role). It can be concluded that the results and insights from the Nordic as well as international severe accident research have been essential for the improvements and further development of accident management for the Nordic power reactors.

Adequate instrumentation is vital for successful accident management. Efforts in this area have been modest but there are differences in this respect between the plants.

The understanding of both in-vessel and ex-vessel accident progression and phenomena, as well as the ability to assess containment threats, to quantify uncertainties, and to interpret the results of experiments and computer code calculations have improved significantly. However, important phenomenological uncertainties still exist. This requires continued research in the following areas:

- Reflooding of the partly degraded core. Important issues here are thermo-hydraulic phenomena at very high temperatures, recriticality and hydrogen generation.

- Core material coolability in the lower plenum as well as timing and mode of reactor vessel failure, should it occur.

- Melt-water interactions in the containment, including melt fragmentation and coolability, and steam explosions. Physical properties of the melt appear to play important role in melt fragmentation and steam explosion phenomena.

- Hydrogen distribution and combustion in the containment and reactor building. Important issues here are turbulent combustion and deflagration-to-detonation transition (DDT) and the rate of hydrogen leakage from the containment to the reactor building.

- Fission product behaviour in the primary system and containment. Important issues here are organic iodine formation, pH control and fission product re-vaporisation.

In addition, it is recommended to continue research and development in the areas of:

- Instrumentation for severe accident management.

- Computerised diagnostic and predictive operator aids for plant status assessment and accident management.

Formation and behaviour of organic iodine

Iodine is probably the most important fission product released in a nuclear reactor accident. The main reason for this is that a significant share of iodine may exist in a volatile form. The isotope ^{131}I has a half-life of 8 days, short enough to have a high specific activity and long enough to persist for a significant time after the accident. The biological activity of iodine increases the health hazard, because iodine concentrates in the human thyroid gland. The formation and behaviour of organic iodine is an important issue to BWRs utilising filtered venting of the containment as a part of their severe accident management strategy. Organic (or elemental iodine) is not efficiently scrubbed by the filtering system, unlike the case with iodine in aerosols. Especially in the case of possible early release, the issue of organic iodine and elemental iodine formation and filter efficiency in retaining the iodine isotopes become more important as there is less time for emergency operations.

The data on possible formation of organic iodine through reactions of boron carbide with steam and iodine is scarce. Degradation of coating and cable materials would also increase the formation of organic and elemental iodine. A more accurate modelling approach of the iodine chemistry in the containment could lead to a significantly different containment source term than the design basis of the filtered venting system.

In this task, the work aimed mainly at creating an understanding of the underlying chemistry and performing small-scale experimental work. Two literature surveys were conducted within the project. In the first of these surveys, the purpose was to review the methods to prevent a source term of methyl iodide during a core melt accident (Karhu 2000). In the second study, the aim was to gather valuable information on the behaviour of methyl iodide on the gas phase during a severe accident (Karhu 2001). In the experimental studies, the formation of organic iodine in solutions with high pH and on painted surfaces were investigated. Additionally, possibilities to use silver nanoparticles for absorbing iodine were studied experimentally.

Methods to prevent a source term of methyl iodide

The reaction pathways and formation of methyl iodide in the containment are still not fully understood. This is due to the fact that there are several different chemicals present in the containment. Also, the equilibrium of the various possible reactions is difficult to estimate during a core melt. Computer simulations provide one tool for visualising the conditions and the behaviour of iodine species. However, actual experiments are required to test the efficiency of the current and new methods for preventing the formation of methyl iodide or for removing it.

The most widely studied methods to prevent a source term of methyl iodide in nuclear power plants include impregnated carbon filters and alkaline additives and sprays. It is indicated that some deficiencies may emerge in these methods.

Filters with a large surface area are common, even though high humidity and other reactive compounds decrease their efficiency. The impregnants are used to improve the properties of the filters. However, more stable and reactive impregnants should be found. Amine based compounds such as TEDA are typical impregnants. A variety of different amines exist, and they could offer unexplored opportunities. Studies of these compounds could clarify their adequacy for use in nuclear power plants.

Considerable effort is put in maintaining a high pH that is able to prevent the formation of elemental iodine. However, several difficulties are likely to evolve with very alkaline compounds. Strong alkalines are required to neutralise the acid compounds such as HCl and HNO_3 released during a high irradiation field. The buffering of sump water may be one solution.

A wide unexplored area is transition metals. The indications of catalytic surface reactions point out their potential ability to decompose methyl iodide. Further studies are required to be able to estimate the applicability of these metals during a core melt. One should mention that Pd is already studied for hydrogen recombination and is going to be taken into use at the Loviisa NPP.

Some other methods that are considered minor here, have been described. The decomposition of methyl iodide by reactions on electrodes seems to give good results. However, the electrodes are dependent on the energy supplied from outside, which can cause problems during an accident. Also, the reliability and expenses of the equipment may become an issue when planning and considering the new safety systems.

Two different photolytical reactors are introduced. Photolysis is known to decompose methyl iodide and it enhances the reaction on most surface catalytes. Thus, photolytical reactions are considered more as auxiliaries to other methods described here.

It seems that the research on the prevention of methyl iodide is a rather narrow area. The emphasis is on the filtrating systems and maintaining a high pH. Further experiments about the new possibilities introduced here are necessary to estimate the capacity of these compounds. They could provide a great improvement to the current methods.

Gas phase chemistry and removal of CH_3I

Iodine is released from the core most likely as CsI. Volatile elemental iodine is expected to form via radiolytic oxidation of iodide ions. This reaction is strongly dependent on pH and increases significantly in acidic conditions. Elemental iodine is able to undergo further reactions. Iodine is a free radical scavenger and reacts readily with alkyl radicals. This may lead to the formation of various organic iodides such as methyl iodide. Furthermore, methyl iodide may become a source term during a severe accident, since it can penetrate most filters currently in use.

Opinions about the formation of organic iodides vary somewhat. However, it is generally accepted that the homogeneous gas phase reactions are of minor importance. The formation proceeds most likely on the surfaces or in the aqueous phase. Radiation is the driving force in the formation of organic iodides. The formation rate may be decreased by competitive reactions of organic free radicals with other compounds such as nitrogen oxides. Methyl iodide may also be decomposed thermally at temperatures below 400 ^oC or at exposure to light. Volatile iodine compounds will partition between the gas and aqueous phase and surfaces. Methyl iodide is more volatile than for example elemental iodine. Thus, the formation of methyl iodide is likely to increase the total iodine concentration in the gas phase.

Since iodine is able to penetrate the current filters in use, more efficient and reactive impregnants and bed materials should be developed. Bed materials have a large impact on the capacity especially at high relative humidity. Water vapor condensates easily to the micropores of the filter and decreases the reactive surface. However, if the pores are too large, there are less active reaction sites and the reaction rate is decreased. In order to develop more stable filters especially for high relative humidity, new carriers should be found. The efficiency of the current filters could also be improved if one could control the pore size for example in zeolites during the production. However, the balance between the reaction rate and water condensation should be defined in order to produce an optimal porous bed material.

The mitigation rate may be increased by using impregnants. Transition metals such as silver and copper may also be used as impregnants along with the conventional compounds, such as TEDA. Non-carbenous bed materials such as zeolites are typically substrates for silver compounds. Silver is a relatively inert, thermally stable material. Iodine and methyl iodide are known to react well with silver. Figure 8 illustrates the efficiency of silver compounds in removing methyl iodide. According to the studies, the reaction of methyl iodide with a silver surface produces ethane. In fact, ethane was the only carbon compound that could be detected. Iodine was accumulated on the surface. It was released again, when the sample was heated to 600 ^{o}C.

Copper is also able to react with radio iodine compounds. The reaction produces copper iodide. Even though Cu is effective in removing inorganic iodides like I_2 and HI, organic iodides are able to pass through the copper column. Thus, organic iodides have to be converted to an inorganic form prior to the actual removal using CuO at 800 ^{o}C. The reaction of CH_3I and CuO produces I_2, which can be removed by the currently used filters.

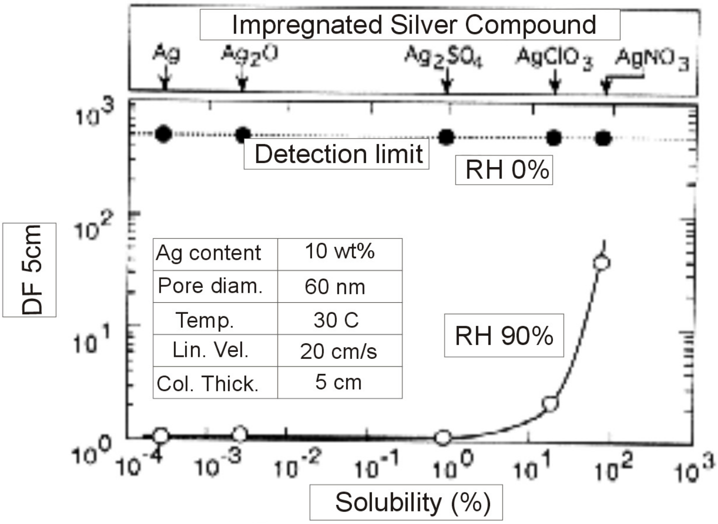


Figure 8. Efficiency of different silver compounds supported on alumina in removing methyl iodide, presented as decontamination factor vs. solubility (Funabashi et al. 1994).

Experimental work and simulation of pH behaviour

The small scale experimental work concentrated on the analysis of the formation of organic iodine both in solutions used in the filtered venting systems and on painted surfaces. The pH behaviour in the containment of Olkiluoto nuclear power plant was studied through simulations in order to evaluate the amount of alkaline chemical needed for pH control. Experiments to trap iodine by silver nanoparticles were also conducted.

Simulation of pH behaviour

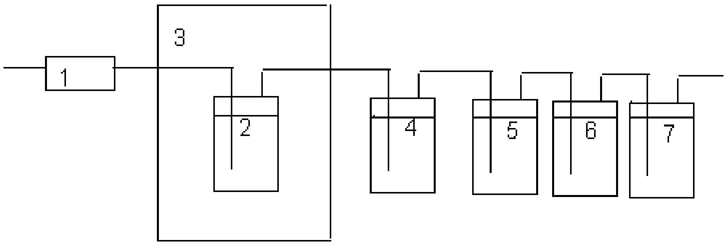
In severe accidents the insulator material of the cables situated below the pressure vessel release HCL and hydrocarbons in pyrolytic decomposition. The HCL decreases dramatically the pH of the wet well and hydrocarbons can react with iodine and form organic iodides. The pH behaviour of the water in the wet well of Olkiluoto BWR was calculated in order to define the amount of NaOH-solution needed to control the pH during a severe accident.

The thermal hydraulic conditions in the containment were calculated with MELCOR 1.8.3 and the results were used in the pH calculations by Chem Sheet. According to the calculations the pH of the wet well goes down rapidly and stays below 2 without any control. The radiolysis and acidic conditions change practically all iodine to I_2 and our bubbler tests showed that reactions of I_2 with organic material produce methyl iodide and other organic iodine compounds.

According to previous experiments, the reactions are much slower in neutral water and after increasing the pH to above 9, only a small part of the iodine is in the form of I_2 and forms organic compounds. The calculations show that 5 m^3 of 50 % NaOH-solution is needed to keep the pH at above 9 during a severe reactor accident.

Experiments on formation of organic iodine in solutions with high pH

The first experiments concentrated on the formation of organic iodine in solutions with high pH. These were made to study the organic iodine formation in scrubber filters. By bubbling a gas mixture (N_2 99 % and CH_4 1%) through a CsI-solution in a gamma radiation field the formation of organic iodides was as follows. A schematic diagram of the bubbler test arrangement is shown in Figure 9.



1. Flow meter
2. Reactor vessel
3. Irradiation facility
4. NaOH-solution (2g/l)

5, 6, 7 Ethanol

Figure 9. Schematic diagram of bubbler tests.

The results clearly verify that at pH 9 the formation is considerably slower than at pH 7. The solutions used in the filters were also studied. The previous filter solution contained 5g/l NaOH and 2 g/l $Na_2S_2O_3$ and formation of organic iodine was detected. After increasing the amount of $Na_2S_2O_3$ to 35g/l organic iodine formation was detected no more. The chlorine concentration of the solution had no noticeable effect.

Experiments on organic iodine production on painted surfaces

One source of organic iodine in a reactor accident is the reaction with painted surfaces. Experiments were performed to study this formation in waters having the same chemical composition as the sump water in the Loviisa reactors. Painted concrete blocks were irradiated in glass bottles. The block was partly in the water having an identical chemical composition to the sump water in the Loviisa reactors. The production of organic iodides was measured by conducting a slow flow of N_2 through the bottle and trapping the inorganic iodine into NaOH-solution and organic iodides into ethanol.

The results of the tests are presented in Tables 10 and 11.

Table 10. Volatile iodine formation on painted surfaces. *Experiment with air flow.

Dose kGy	% of iodine released as organic compound	% of iodine released as organic compound/kGy	% of iodine released as I_2	% of iodine released as I_2/kGy
14	0.012	0.00086	0.021	0.0015
57	0.023	0.00040	0.039	0.00068
88	0.03	0.00034	0.054	0.00061
104	0.11	0.00106	0.33	0.0032
27*	0.019	0.00070	0.17	0.0063

Table 11. Share of Iodine found on the painted surfaces after irradiation.

Dose kGy	% of iodine, wet surface washed with ethanol	% of iodine, wet surface washed with NaOH	% of iodine, dry surface washed with ethanol	% of iodine, dry surface washed with NaOH	% of iodine left on the surfaces (after washings)
14	0.53	2.3	0.3	0.4	17
57	0.12	0.8	0.3	0.2	18
88	0.83	6.2	0.5	0.2	28
104	0.73	2.7	0.7	1.2	23
27*	0.32	1.4	0.3	0.2	10

Experiments on Ag nanoparticles for iodine absorption

Since iodine and methyl iodide are known to react with silver, it was of interest to study experimentally the potential use of silver nanoparticles for arresting methyl-iodide. This study included the production of silver nanoparticles and testing of their capability to absorb both elemental iodine and methyl-iodide (Backman & Zilliacus 2002).

Silver nanoparticles were synthesised with a high number concentration using two different facilities. Both facilities were evaporation-condensation based systems, where silver was evaporated from a ceramic crucible into a tubular flow reactor. A regulated N_2 gas stream carried the silver vapor from the reactor. As the vapor cooled downstream of the reactor, it became supersaturated with nucleation and particle growth as a consequence. In the first system the cooling took place by heat conduction, whereas in the second system the cooling was accelerated by diluting the carrier gas with a cold gas stream. Quenching the vapor exiting the reactor enhances the nucleation of small particles and suppresses the formation of agglomerates and other particle growth mechanisms.

The size and shape of the produced particles were determined with a transmission electron microscope (TEM). The particle sizes in the experiments without dilution were above 10 nm in all experiments. In the system with dilution the particles were smaller, between 4 nm and 10 nm in size. The total particle number concentration was greater than 10^8 particles/cm^3 in all experiments. It could be seen from the TEM micrographs that the particles without dilution are spherical but partly agglomerated whereas with dilution they are non-agglomerated. The tests with methyl and elemental iodide were conducted with silver particles collected from the experiments without dilution, i.e. with the bigger particles.

The absorption of I_2 and CH_3I were tested by using an experimental set up previously developed for filter solutions. N_2-gas was lead through an acidic I-solution for the production of elemental iodine in a nitrogen stream. Methyl iodide was prepared in a similar way in a radiation field by using 0.1 % CH_4 in the nitrogen stream. In this case, the elemental iodine was washed from the gas stream by bubbling through NaOH-solution. Radioactive ^{131}I was used as a tracer in the experiments and the iodine absorption was calculated by measuring the activities in the filters and the trapping solutions after the filter. NaOH was used to trap elemental iodine and ethanol to trap methyl iodide. The tests show a strong absorption of elemental iodine in the filters and also absorption of methyl iodide. The absorption is strongly dependent on the I/Ag molar ratio. The results of the test are shown in Table 12.

Table 12. Results of the absorption tests with silver nanoparticles.

	Molar ratio	Absorption, %
I_2/Ag	2×10^{-3}	98.4
I_2/Ag	1×10^{-4}	>99.9
CH_3I/Ag	9×10^{-6}	12.6
CH_3I/Ag	4×10^{-7}	53

The tests show that it is possible to trap both forms of iodine by silver nanoparticles. However, the amount of silver nanoparticles needed to absorb the organic form of iodine is significantly higher than the amount needed to absorb elemental iodine.

Hydrogen issues

The BWR containment is normally inerted with nitrogen during operation and thus hydrogen combustion phenomena inside the containment are prevented and hydrogen detonation issues are not considered in severe accident management studies for BWRs. However, a hydrogen explosion in the reactor building due to leakage from the containment during a severe accident might jeopardise the leak-tightness of containment penetrations from the outside. This scenario was analysed in detail in SOS-2.3.

Another hydrogen issue considered in this project was the hydrogen deflagration scenario in the Ringhals 3 PWR containment. The most important accident sequences concerning hydrogen generation and containment pressure at hydrogen deflagration were investigated in this study.

Analysis of a hydrogen detonation in BWR reactor building

A significant release of hydrogen into a relatively small containment can occur in a severe accident, and a hydrogen leakage from the containment into the surrounding reactor-building rooms cannot be ruled out. Because the atmosphere in the reactor building consists of normal air, the ignition and combustion of hydrogen is possible. The safety concern is whether the hydrogen in the reactor building can detonate and jeopardise the containment integrity against the outside.

A special motivation for studying detonation phenomenology derives from the recent studies of hydrogen leakage from an overpressurised BWR containment and consequent accumulation of hydrogen into the reactor building rooms during severe reactor accidents. The atmosphere in the reactor building is normal air, making hydrogen combustion possible. The assumed accident scenario was a station black-out sequency with depressurisation of the reactor coolant system. Oxidised zirconium was assumed to lead to a hydrogen release of 1900 kg in the containment. Studies by Manninen and Huhtanen (1998) on hydrogen distribution in selected reactor building rooms in the Olkiluoto BWR suggest that the hydrogen accumulates near the ceilings of the rooms. Furthermore, the stratification tends to be rather stable and yield very high hydrogen concentrations.

The work in the SOS-2 project included a survey on detonation dynamics in hydrogen-air-steam mixtures, assessment of detonation pressure loads on room structures, and structural analyses of the integrity of the reactor building walls. An overview of physical mechanisms under detonation conditions in pre-mixed hydrogen-air-steam mixtures, and introduction of the basic laws and relationships applicable to first-order estimates of pressure loads connected to detonations are given in the report by Silde & Linholm (2000).

Detonation simulations

The detonation shock pressure loads were analysed for a selected Olkiluoto reactor building room B.60.80 in three different basic cases. In two of them, the leak area from the containment to the reactor building was 20 mm^2, and the detonations were assumed to initiate in two different moments of time, 13 000 s and 7500 s, corresponding to the detonable hydrogen mass of 1.43 and 3.15 kg in the reactor building, respectively. The third case considered a 2 mm^2 leak area (containment design leakage), in which the detonable hydrogen mass was 1.4 kg. The gas flow pattern in the reactor building had been calculated earlier. The FLUENT 3D-code had been used to obtain the hydrogen concentrations in the reactor building compartments. The distance from the explosion origin to the wall was assumed to be 2.0 m in all the basic cases.

The detonation pressure loads were at first estimated by computer using DETO, which was developed during the work at VTT. The code is based on the strong explosion theory incorporating oblique shock reflection relations. The DETO code is a 1-D program without any modelling of fluid mechanics. The basic simplification is that the total energy is instantly released in the explosion origin. The local explosion then induces spherical shock waves without a combustion zone propagating at certain velocity. The velocity of the shock wave is very sensitive to the distance from the explosion centre. Thus, the shock velocity and the shock pressure induced by a strong explosion may be very high if the distance of the shock wave is very small. On the other hand, the shock wave velocity and pressure decrease relatively strongly as the distance increases, because no propagation of the self-sustained combustion zone is modelled. This means that the strong explosion theory does not simulate a propagation of a real, self-sustained detonation wave properly, if the distance from the explosion origin is very long.

Since only the first reflection of the 1-D shock wave was considered, the results had large uncertainties. The real detonation processes are complicated interactions between thermodynamics, flow mechanics, and chemistry. Multiple shock wave reflections, collisions and focusing in three-dimensional geometry may results in local pressures in the detonation front higher than predicted by such approximate 1-D theories as are used e.g. in the DETO code. It was decided to use more detailed three-dimensional numerical analyses in order to assess these complicated interactions and their influences on the pressure loads under detonation conditions.

The three-dimensional detonation simulations were carried out with the DET3D code developed at Forschungszentrum Karlsruhe (FZK). The code uses the finite difference method based on three-dimensional Euler equations for a multi-component reacting gas. The approach of the code enables a more detailed assessment of detonation pressure loads in real 3-D geometry taking also into account multiple shock reflections and superposition of the shock waves. DET3D is mainly developed for modelling of gaseous detonations initiated by a direct ignition, and DDT phenomena are not treated. Prior to the simulations, the numerical parameters of DET3D were verified against the theoretical Chapman-Jouguet values in one-dimensional geometry. The general conclusion from the test calculations was that the DET3D results were quantitatively accurate enough over a wide range of hydrogen concentrations to justify the use of the DET3D code for the hydrogen detonation assessment in the Olkiluoto reactor building.

The initial conditions of the DET3D simulation were based on previous CFD-analyses carried out with the FLUENT code, as in the case of 1-D calculations. The influence of the ignition location and the numerical method (1. order versus 2. order) of the hydrodynamics solver of DET3D on the detonation loads were studied. The same computational grid having 778 320 cells with the cell size of 0.117 m was used in all simulation cases.

The DET3D simulation indicated that the highest pressure spikes occurred in the room corners due to reflections and superposition of the shock waves. The highest pressure maximum in all simulation cases was about 10.6 MPa. This value was obtained in the case of 20 mm^2 leakage in the upper corner of the room next to the containment wall. The highest pressure impulses on structures during the 150 ms simulation were about 30 - 35 kPa-s.

At the end stage of the 20 mm^2 leakage, the gas concentration gradient in the reactor building room was very steep in the upper part of the room, and a hydrogen inerted layer existed near the room ceiling. Under these conditions, only a relatively small amount of hydrogen was burned during the first detonation wave. Later propagation of a slow combustion front to the still hydrogen rich upper region was predicted to lead to flame acceleration and a second detonation, now at a different location from the first one. In all other detonation simulation cases, the gas mixture above the level of leakage was initially relatively homogeneous and no hydrogen inerted layer existed in the room. In these cases, practically all hydrogen was burned during the first and only detonation. Figure 10 shows the simulated detonation waves in the case where the leakage size is assumed to be 20 mm^2 , and the detonation occurs at 13 000 s from the start of the accident.

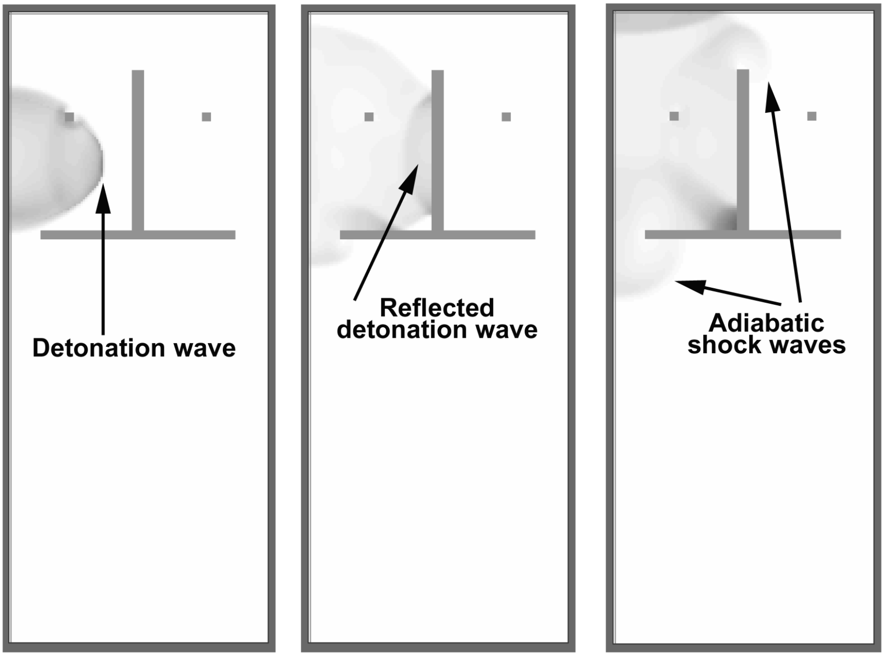


Figure 10. Illustration of pressure waves in room B.60.80 during the first detonation at three different moments of time: 2.0 ms (left), 4.0 ms (middle), and 6.0 ms (right) in one studied case.

Structural analyses

The structural integrity of a reinforced concrete wall in the reactor building under detonation conditions was analysed in order to see whether the containment integrity can be jeopardised by an external hydrogen detonation. The load carrying capacity of a reinforced concrete wall was studied. Structural integrity may be endangered due to slow pressurisation or dynamic impulse loads associated with local detonations. The static pressure following the passage of a shock front may be relatively high, whereby this static or slowly decreasing pressure after a detonation may damage the structure severely. The mitigating effects of the opening of a door on the pressure history and structural response were also studied.

Preliminary calculations were carried out using pressure transients calculated by a simple one-dimensional DETO-code, which is based on the strong explosion theory. The detonation was assumed to occur in the middle of the wall. The non-linear behaviour of the wall was studied under detonations corresponding to a detonable hydrogen mass of 0.5 kg and 1.428 kg. Structural analyses were carried out by ABAQUS/Explicit code, which is a finite element program based upon the implementation of an explicit central difference integration rule. This work is reported in Saarenheimo 2000 and Saarenheimo & al. 2001a.

According to three-dimensional detonation calculations by the DET3D code, the highest pressure spikes occurred in the corners of the room. Due to the fact that the detonations occurred in the corners, a three dimensional FE model was needed. This DET3D data, that would correspond more realistically to the detonation pressure loads, was transferred to the ABAQUS model as input data for the structural analyses. A transfer tool was developed to enable flexible data transfer between the DET3D and the ABAQUS codes (Silde & Pättikangas 2001).

Non-linear finite element analyses of the reinforced concrete structure were carried out by the ABAQUS/Explicit program. The reinforcement and its non-linear material behaviour and tensile cracking of the concrete were modelled. Reinforcement was defined as layers of uniformly spaced reinforcing bars in the shell elements. In these studies, the surrounding structures of the non-linearly modelled reinforced concrete wall were modelled using idealised boundary conditions. This work is reported in Saarenheimo & al 2001b.

Concrete cracking and yielding of the reinforcement were especially monitored during the numerical simulation. Elastic deformations in the reinforcement are recoverable and cracks in these areas will close after the pressure decrease. According to these studies, the structure may survive a peak detonation transient. A relatively slowly decreasing static pressure after a peak detonation may damage the structure more severely.

Hydrogen deflagration in a PWR containment

A study was conducted on hydrogen deflagration in the Ringhals 3 containment (Gustavsson & Möller 2001). The objective of the study was to investigate the most important accident sequences concerning hydrogen generation and containment pressure at hydrogen deflagration. The focus was on the analysis of sequences with reflooding of the damaged core.

In the earlier hydrogen studies, the calculations of the peak pressure in the containment at hydrogen deflagration have shown results close to the containment failure pressure of Ringhals 3 and 4. The probability of a containment failure in Ringhals 3 and 4 as a conservative assumption has been estimated to be slightly below $1x10^{-7}$ / reactor year. The previous investigations were carried out with a lower quantity of zirconium in the core than according to the inventory today. The difference is about 10-15%. These facts motivated further investigations of the hydrogen issue using the updated zirconium content in the core in the calculations.

A large number of accident sequences was condensed to a small number of cases at first. Six key accident sequences were selected, four of them are LOCA cases and two are transients. The computer code MAAP (Modular Accident Analysis Program) was used to analyse the accident sequences and to calculate the hydrogen production. MAAP gives the evolution of the accident and particularly the pressure in the containment and the production of hydrogen as a function of time.

Five of the sequences listed in the Ringhals 3 and 4 safety study were recalculated with the new version of MAAP and the results showed good agreement with the previous calculations. The difference regarding the timing of the events such as reactor vessel failure was expected and is a result of the changed modelling of the molten corium and steel vessel interaction. The amount of hydrogen produced in each case was significantly higher than expected due to several reasons, mostly to the greater total mass of Zirconium in the fuel elements in the present core.

Two sequences were chosen for detailed analysis of the hydrogen production and containment pressure. Both of these sequences were modified slightly to create an extremely severe case where a high amount of hydrogen was produced and allowed to exist in a high-pressure environment. The main aim with a detailed analysis of the simulated accident sequences was to determine the worst case in line of a hydrogen deflagration which might endanger the last protection barrier – the containment. In the considered sequences, a high concentration of hydrogen mass and moles is achieved in the containment. The high initial pressure in the containment before deflagration is also a strong factor endangering the containment.

Figure 11 shows an example of a case where deflagration is probable. In this accident sequence, the initiating event is a medium size LOCA. It is assumed that the PORVs will not open on signal. Further, it is assumed that the high-pressure injection system stops due to a component failure, and the core melts. This sequence has been shown not to meet the criteria for hydrogen deflagration. However, there is a risk that containment spray is started and if that occurs the deflagration criteria are met and a combustible gas mixture is achieved. In this case (Fig. 11), the containment spray is assumed to start at t= 50000 sec

The pressure peaks at deflagration of the hydrogen were calculated by the AICC-Adiabatic Isochoric Complete Combustion method. The results from these calculations are conservative for two reasons. Adiabatic combustion means that the heat losses to the structures in the containment are neglected. The combustion is also assumed to occur once and to burn up all the available hydrogen.

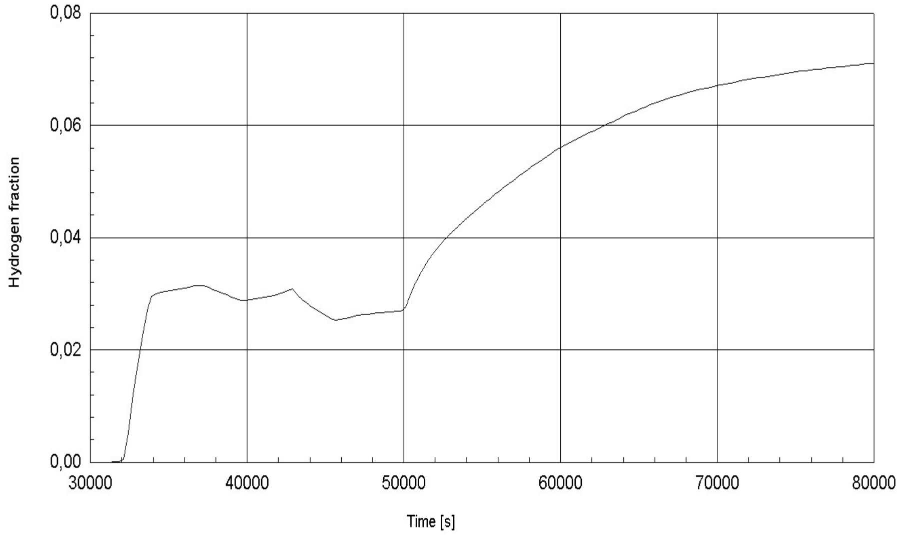


Figure 11. Mole fraction of hydrogen when containment spray is started at t=50000 s.

The AICC calculation for the above example gives a post deflagration temperature of 725 K, which correspond to a pressure peak of 0.53 MPa. The analysis shows that the containment is not threatened in this sequence. The estimated containment failure pressure of Ringhals 3 is 0.7 MPa.

In the worst accident scenarios that were analysed in detail the calculated peak pressure exceeded the containment failure pressure in one case. However, as mentioned before, the AICC calculations are conservative. There is also conservatism in the evaluation of the failure pressure of the containment, and this failure pressure has an impact on the probability of containment failure. A conservative estimate for the containment failure probability due to hydrogen deflagration is as low as $0.6x10^{-7}$ /year. Furthermore, a relatively small increase in the failure pressure of the containment would decrease that probability strongly.

Recriticality after steam explosion in the lower head

Steam explosions have been intensively studied during the last 20 years. The main focus has been on the α-mode containment failure. The α-mode failure sequence assumes that a large amount of core melt relocates to the lower plenum of the pressure vessel during a severe accident. An intense in-vessel steam explosion occurs which demolishes the upper head of the reactor pressure vessel and is able to produce a missile that reaches and penetrates the containment above. The probability of the α-mode failure has been recently judged extremely low. The possibility of smaller scale steam explosions in the lower head of the pressure vessel can not, however, be excluded. A steam explosion there can endanger the lower head integrity or push coolant into the damaged core.

The first material to melt in a BWR severe accident are the control rods. This is due to the low eutectic melting temperature for the mixture of boron carbide and stainless steel. The molten material is accumulated on top of the lower support plate of the core. Some of it resolidifies supporting an accumulating melt pool. The supporting layer eventually breaks, and melt pours into the lower plenum. Thermal energy from the melt is released in the coolant. If no steam explosion occurs, the relatively slow coolant heatup causes a level swell. A steam explosion may eject a coolant slug into the upper parts of the pressure vessel.

In-vessel melt behaviour and control rod melting in the Nordic BWRs have been studied in the earlier NKS/RAK-2 project. Core recriticality was a part of the NKS/RAK-2 and especially of the EU 4th framework "Severe Accident Recriticality Analysis" (SARA) project. The sequence studied in these projects was that an ECCS injection refloods the core from which neutron absorbing material has relocated. The effect of the ECCS injection rate was one of the variations investigated. It was concluded that the maximum energy deposited during the recriticality peak was in all cases below the fuel failure limit, and hence recriticality would not cause large scale fuel damage.

The present NKS/SOS-2.3 project continues the NKS/RAK-2 and SARA recriticality investigations. The difference is that coolant penetration into the core is assumed to occur due to a melt-coolant interaction in the lower plenum. No integral code is capable of describing all the necessary phases , and the work requires combining several separate simulations:

- accident progression and prediction of the control rod relocation,
- evaluation of the heat released from the melt to the coolant in the lower plenum, and the penetration rate of the coolant into the core,
- recriticality due to coolant entering the core from the lower plenum

This case has a more uncertain and different coolant injection rate compared to the previous ECCS cases. The lowest rates are related to the level swell and the highest to the steam explosions. The latter may be much higher than the highest reflooding rate studies in the RAK-2 and SARA projects.

Accident progression estimates from the NKS/RAK-2 project were considered applicable. The case selected was an Olkiluoto station blackout assuming depressurisation as specified. The main assumption concerning recriticality was that melt relocates into the lower plenum at a moment when the maximum amount of control materials B_4C, stainless steel and Zr from the canister walls has accumulated to the lower support plate but all of the fuel still remains intact. The assumption does not maximise the steam explosion but tries to maximise the recriticality event.

The case was calculated with MELCOR 1.8.4. Depressurisation to 5 bar was assumed as the accident management feature. The case assumed resulted in melt relocation at 4700 s from the initiation of the accident. At that time the water level is below the core, and 13 tons of steel and 1.1 tons of B_4C are in a molten state on top of the bottom plate. The molten steel and B_4C have a temperature of 1700 K, which is the MELCOR default stainless steel melt temperature. 14.1 tons of melt produce a 0.96 m high pool into the space between the fuel elements.

Two flow paths through the lower core support plate are possible:

- through the control rod hole, which has a crucifix form with a 60 cm^2 flow area
- through the fuel bundle inlet hole, which is spherical having a 78 cm^2 flow area

The flow paths influence the magnitude of melt-coolant interaction in the lower plenum. In case one the melt fills a control rod guide tube. The guide tube is initially full of water and also surrounded by water. If the water swells, it fills only the bypass. The melt mass is so small that a steam explosion is not very energetic. In case two, the fuel element wall is ablated and the molten metal flows into the lower plenum water. In this case, a level swell or steam explosion could be more energetic. The melt pouring velocity was calculated manually and also simulated with the FLUENT-4.5.2 –code. The melt flow through the 60 cm^2 flow area was estimated at 40 - 110 kg/s and through the 78 cm^2 flow area at 50 – 150 kg/s. The respective enthalpy flows as superheat compared to the coolant temperature are 40 – 100 MW and 50 – 150 MW.

It was concluded that the melt flow rate through a single passage is too small for initiating any significant level swell or energetic steam explosion. Therefore, a much more conservative scenario was postulated, in which all flow passages open at the same time. In addition to this, the melt was assumed to penetrate into the lower plenum directly. In this case, the duration of the pour is approximately 1.0 s. The level swell can be estimated based on the film boiling correlations. Due to the effect of the phase separation, only 3.0 m^3/s flow into the core.

Estimation of the coolant flow rate after an energetic steam explosion is more problematic. 14.1 tons of melt has 11 GJs of thermal energy. The conversion ratio of the thermal energy into mechanical energy in a steam explosion may be 0.1 –5.0 %. The mechanical energy accelerating the water column in the lower plenum would thus range from 11 to 550 MJ, respectively. When the liquid masses available in the lower plenum are about 32 tons, the estimated velocity range is from 26 to 185 m/s. Finally, the mass flow rates into the core are from 10 to 72 m^3/s if the entire coolant mass penetrates through the 500 fuel channels.

Recriticality analyses were done with the GENFLO code, which contains a 2-dimensional neutronics model allowing the calculation of prompt criticality. The GENFLO model originates from the RECRIT code that was used as the recriticality analysis tool in the RAK-2 project. GENFLO is extended with respect to other simulation applications and has been recently coupled with

- a 2D neutronics model,
- a fuel rod model, and
- a severe accident simulation model

The renewed GENFLO neutronic version was validated against reactivity transient data for Olkiluoto. The goal of the validations was to demonstrate the capability of the code to describe the power peak due to the void collapse properly and the effect of the additional reactivity from the control rods.

The GENFLO calculation assuming a level swell with a 3 m^3/s inflow resulted in 100 cal/g energy deposition into the fuel. The result is only slightly more alarming than that calculated with the highest ECCS reflooding rate in the RAK-2 and SARA projects. Recriticality has also been calculated for the 10 m^3/s flow rate estimated for the steam explosion. This case created an energy deposition as high as 150 cal/g into the fuel. The results with higher flow rates were judged unrealistic. Recriticality due to a steam explosion seems to be slightly more dangerous to the fuel than the cases with the highest ECCS reflooding rate analyses in the RAK-2 and SARA projects, but large scale fuel disintegration may be avoided. It must be borne in mind that the assumption for lower plenum relocation was very conservative. Further analyses are being performed to check these results.

SOS-2.3 conclusions and recommendations

The emphasis of the SOS-2.3 sub-project has been on phenomenological issues of importance for accident progression and accident management in Finnish and Swedish nuclear power plants. The studies on hydrogen combustion scenarios, formation and behaviour of organic iodine and re-criticality due to a steam explosion in the lower head have produced valuable information on these phenomena.

The detailed analysis of a hydrogen detonation in a BWR reactor building evaluated the possibility of such a scenario and the uncertainties related to the phenomena. The safety concern is whether the hydrogen in the reactor building could detonate and jeopardise the containment integrity from outside by damaging the pipe penetrations. The detonation loads were calculated with a three-dimensional model. These loads were transferred to structural analyses. The integrity of a reinforced concrete wall of the reactor building was studied, as one pipe support is located on this wall. According to the structural analyses, the static pressure after detonation could be of concern for the piping penetrations. More detailed analyses of the pressure decrease after detonation, taking into account the holes and cracks in the reactor building wall, would reduce the uncertainties related to the structural integrity.

The most important accident sequences concerning hydrogen generation and containment pressure at a hydrogen deflagration in Ringhals 3 PWR were investigated.

The focus was put on the analysis of sequences with reflooding of the damaged core. Detailed analyses of the hydrogen production and containment pressure were performed for the most important sequences. A conservative estimate for containment failure probability due to a hydrogen deflagration is 0.6 x 10 -7/year.

Small scale experiments on the behaviour of organic iodine aimed at studying both the formation of organic iodine and the possibilities to improve filters to become capable of trapping organic iodine compounds. The analyses showed that relatively large quantities of alkaline chemicals are needed to counterbalance the effect of acids released during the accidents, such as HCl. The analyses also indicated that such quantities could be introduced into the containment in a timely fashion. Trapping of iodine with silver nanoparticles was studied experimentally. As these experiments did not prove efficient retention capability of organic iodine, other potential absorption methods need further investigation.

Recriticality due to melt coolant interaction in the lower plenum was estimated for a BWR severe accident. Three separate analyses were combined to obtain the estimates: core heatup and control material melting, melt relocation rate into the lower plenum and extent of the melt-coolant interaction, and coolant reflood rate into the core and the resulting power generation. Preliminary results from the first and last part are available, but the part determining the extent of melt-coolant interaction needs to be refined. The preliminary analyses resulted in higher amounts of energy deposited into the fuel than in cases calculated for emergency core coolant system flooding. The results do not indicate global fuel damage, however.

According to the review of the current status of severe accident research and management, there are some important uncertainties related to both in-vessel and ex-vessel accident progression phenomena, which motivate continued research. Such are e.g. in-vessel and ex-vessel coolability and mode of reactor vessel failure. Other potential future research topics are SAM instrumentation and computerised operator aids for plant status assessment and accident management.

Concluding remarks

The NKS/SOS-2 project consisted of studies and reviews related to three main areas identified in the pre-project phase of the NKS programme period, 1998-2001. These areas were risk-informed decision making, maintenance quality and management-related issues, and severe accidents. These topics were selected on the basis of common interest in the Nordic countries. The definition of the work scope was also influenced by the research within the EU 5th framework programme in order not to repeat the previous studies.

The project was organised in three sub-projects, each of them consisting of several studies. The studies were partly funded by NKS and partly supported by an end-user, i.e. safety authority or utility. The contents, progress and results of the studies were discussed in sub-project working group meetings twice a year.

The focus of the sub-project SOS-2.1 was on the use of probabilistic safety assessments in decision making. The incompleteness and uncertainties of PSA models were considered from the point of view of risk-informed decision making, and recommendations for dealing with the uncertainties were given. In risk-informed regulation and safety management, the applicability of the PSA model should be judged case by case and its role in the decision process should depend on this judgement.

The sub-project SOS-2.2 concentrated on maintenance related issues. The safety importance of human and organisational factors in maintenance work has lately been recognised, and these issues were also promoted in the work of this sub-project. Furthermore, questions related to maintenance management and to the use of condition monitoring for predictive maintenance were addressed in close co-operation with nuclear power plants.

The sub-project SOS-2.3 was directed to some phenomenological studies related to severe reactor accident scenarios of the Nordic nuclear power plants. The research efforts were concentrated on hydrogen issues, behaviour of organic iodine and re-criticality of a degraded BWR core. The studies increased the understanding of these phenomena, identified remaining work in these topics and produced valuable insights into how severe accidents should be managed. In addition, other areas in need of continued research were identified.

The SOS-2 project has basically dealt with nuclear power plants, but many of the results can be applied outside the nuclear power industry. Some results related to severe accident research and management may be useful for accident analyses of research and similar smaller reactors, too. Furthermore, the results obtained in many of the studies related to management of uncertainties, risk-informed decision making and maintenance issues can be exploited in non-nuclear industries, especially in those with risk concerns.

In addition to the scientific and technical results produced in the project and summarised in this report, one important achievement of the Nordic nuclear safety research and the SOS-2 project is the networking of Nordic experts in this field. It can be stated that it is a unique forum of co-operation with the participation of both safety authorities and utilities together with research bodies in several countries.

Acknowledgements

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Acronyms

AICC	Adiabatic Isochoric Complete Combustion
AOT	Allowed outage time
APRI	Accident Phenomena of Risk Importance, Swedish severe accident research project
BKAB	Barsebäck Kraft AB
BWR	Boiling water reactor
CAMS	Computerised accident management system
CCF	Common cause failure
CFD	Computational fluid dynamics
CM	Corrective maintenance
DDT	Deflagration-to-detonation transition
ECCS	Emergency core cooling system
EoC	Error of commission
EoO	Error of omission
EU	European Union
FMEA	Failure mode and effects analysis
FSAR	Final safety analysis report
FZK	Forschnungszentrum Karlsruhe
HCCF	Human common cause failure
HRA	Human reliability analysis
IE	Initiating event
IFE	Institutt för Energiteknik, Institute for Energy Technology, Norway
IGSCC	Intergranular stress corrosion cracking
ISA	Integrated sequence analysis
KTH	Kungliga Tekniska Högskolan
KTM	Finnish Ministry of Trade and Industry
LCO	Limiting conditions for operation
LOCA	Loss of coolant accident
LER	Licensee event report
MAAP	Modular Accident Analysis Program
MTO	Människa-Teknik-Organisation, Man-Technology-Organisation
NDT	Non destructive testing
NKS	Nordic Nuclear Safety Research
NPP	Nuclear power plant
OKG	Oskarshamn Nuclear Power Plant
PFM	Probabilistic fracture mechanics
PM	Preventive maintenance
PSA	Probabilistic safety assessment
PWR	Pressurised water reactor
RAK	NKS projects in the field of reactor safety 1994-1997
RCM	Reliability centred maintenance
RI-ISI	Risk-informed in-service inspection
RO	Rapportervärd omständighet (event reports)

SARA	"Severe Accident Recriticality Analysis" EU-project
SKI	Statens kärnkraftsinspektion, Swedish Nuclear Power Inspectorate
STUK	Säteilyturvakeskus, Finnish Radiation and Nuclear Safety Authority
TVO	Teollisuuden Voima Oy
VTT	Valtion teknillinen tutkimuskeskus, Technical Research Centre of Finland

References

Andersson, K. 1998. Strategies for Reactor Safety, NKS(97)FR1, ISBN 87-7893-021-9.

Backman, U. & Zilliacus, R. 2002. Ag nanoparticles for iodine absorption. Working Report, VTT Processes. WR/SOS-2(02)3. 8 p.

Bennerstedt, T. (ed.) 1999. Planer for NKS-programmet 1998-2001, 1999-08-02, NKS-5, ISBN 87-7893-053-7.

Brickstad, B. 2001. Evaluation of pipe failure probabilities for Barsebäck unit 1. Report No. 10259800-1. WR/SOS-2(01)1.

ELTRA 1998. " Tilstandsvurdering af transformere".

Frid, W. (ed.) 2002. Severe Accident research and management in Nordic countries, a status report. NKS-XX.

Funabashi, K., Fukasawa, T. and Kikuchi, M. 1994. Investigation of Silver-Impregnated Alumina for Removal of Radioactive Methyl Iodide. *Nucl. Techn.*, Vol.109, pp. 366-372.

Gustavsson, V. & Möller, E. 2001. A study on hydrogen deflagration for selected severe accident sequences in Ringhals 3. NKS-46, ISBN 87-7893-099-5

Holmberg, J. & Pulkkinen, U. 2001. Experience from the comparison of two PSA-studies. NKS-36, ISBN 87-7893-087-1.

Holmberg, J., Rosqvist, T., Simola, K. & Pulkkinen, U. 2001. Decision criteria in PSA applications. NKS-44, ISBN 87-7893-097-9.

Hultqvist 2002. Application of risk based and risk informed decision in USA. FKA-report FT-2002-139. WR/SOS-2(02)2.

Jänkälä, K. 2002. A Risk Informed Safety Classification for a Nordic NPP. Fortum Nuclear Services Ltd, Report NUCL-700. NKS-XX. 30 p.

Karhu, A. 2000. Methods to Prevent the Source Term of Methyl lodide During a Core Melt Accident. NKS-13, ISBN 87-7893-063-4.

Karhu, A. 2001. Gas phase chemistry and removal of CH_3I during a severe accident. NKS-25, ISBN 87-7893-076-6.

Laakso, K. 2000. A cause classification model of human errors related to maintenance. Report number VTT/TAU B001. WR/SOS-2(00)1. 9p.

Laakso, K., Pyy, P. & Reiman, L. 1998. Human errors related to maintenance and modifications. Finnish Centre for Radiation and Nuclear Safety STUK, Helsinki. 42 p.STUK-YTO-TR 139.

Laakso, K. & Simola, K. (eds.) 2000. Reliability centered maintenance miniseminar, 14.10.1999. NKS-18, ISBN 87-7893-068-5.

Laakso, K., Simola,K., Dorrepaal, J., Skogberg, P. 1999a. Systematisk underhållsanalys med beslutsstöd. Metodik och verktyg för effektivisering av underhållsprogram, January 1999, NKS-7, ISBN 87-7893-055-3.

Laakso, K., Sirola, M. Holmberg, J. 1999b. Decision Modelling for Maintenance and Safety, In International Journal of Comadem (Condition Monitoring & Diagnostic Engineering Management 1999). Volume 2. Number 3. Pp.13-17.

Laakso, K. Strömberg, A. 2001. Systematic analysis of maintenance strategies of technical systems. Presented at BALTICA V International Conference on Condition and Life Management for Power Plants. Porvoo, Finland, June 6-8, 2001. Documented in VTT Symposium 211. Vol. 2. Pp. 535-545. Espoo 2001.

Lindholm I., Berg, Ø. & Nonbøl, E. 1998. Safety against Releases in Servere Accidents, NKS(97)FR2, ISBN 87-7893-022-7.

Lydell, B. 1999. Failure rates in Barsebäck-1 reactor coolant pressure boundary piping An application of a piping failure database. SKI Report 98:30.

Magnússon, S. 1998. NKS-programmet 1998-2001; Programgruppens slutrapport, Maj 1998, NKS(98)1, ISBN 87-7893-037-5.

Manninen M. and Huhtanen R. 1998. Hydrogen spreading in the Olkiluoto reactor building in a severe reactor accident, part 2. Research report ENE21/4/99. VTT Energy. Confidential.

MSG-3 1988. Airline/manufacture maintenance program development document. Maintenance Steering Group-3 Task Force. Air Transport Association of America. Rev. 31.3.1998 No 1.

Paulsen, J. 2000. Transformer study. Risø National Laboratory. Work Report WR/SOS-2(00)9.

Paulsen, J. 2000. Design of simple display systems for condition monitoring, using invariants and physical laws. Paper presented at CSEPC2000, Korea, November 2000.

Pulkkinen, U., Pyy, P. & Simola, K. 2000. How to enhance uncertainty analysis of PSA studies. In Cottam, M.P., Harvey, D.W., Pape, R.P. & Tait: Foresight and

Precaution, Proceedings of ESREL 2000 Conference, Edinburgh, Scotland, 15-17.5.2000. pp. 1097-1105.

Pulkkinen U. & Simola, K. (eds.) 1999. Proceeding of the NKS/SOS-2 Seminar on Risk Informed Principles, Bergendal 13.4-14.4. 1999, 1999-09-02, NKS-6, ISBN 87-7893-054-5.

Pulkkinen U. & Simola, K. 2001. Identification and communication of uncertainties of phenomenological models in PSA. NKS-45, ISBN 87-7893-098-7.

Pyy, P. 2000. Human reliability analysis methods for probabilistic safety assessment. VTT Automation, Espoo. 63 p. + app. 64 p. VTT Publications 422.

Pyy, P. Bento, J-P. & Flodin, Y. 2001. Analysis of errors of commission in the Nordic nuclear power plants based on plant operating experience. NKS-XX.

Saarenheimo, A. 2000. Reinforced concrete wall under hydrogen detonation. NKS-26 ISBN 87-7893-077-4.

Salo & Svenson 2001. Human factors in maintenance: development and research in Swedish nuclear power plants. NKS-48, ISBN 87-7893-101-0.

Silde A. & Lindholm, I. 2000. On Detonation Dynamics in Hydrogen-Air-Steam Mixtures: Theory and Application to Olkiluoto Reactor Building, February 2000, NKS-9 ISBN, 87-7893-058-8.

Silde, A. & Pättikangas, T. 2001. D2A version 1.0: interface between the DET3D and ABAQUS codes. Programme document, report ENE4-PD-3/01. VTT Energy.

Silde, A. & Redlinger, R. 2001. Three-dimensional simulation of hydrogen detonations in the Olkiluoto BWR reactor building. NKS-27 ISBN 87-7893-078-2.

Simola, K., Pulkkinen, U. Karjalainen-Roikonen, P. & Saarenheimo, A. 2002. Comparison of approaches for estimating pipe rupture frequencies for risk-informed in-service inspections. Paper to be presented in the European Conference on System Dependability and Safety (ESREL 2002), Lyon 19.-21.3.2002.

T-book 1996. Reliability Data of Components in Nordic Nuclear Power Plants. 4th edition. The TUD Office, Vattenfall Energisystem AB.

WASH-1400 1975. Reactor safety study, an assessment of accident risks in U.S. commercial nuclear power plants. Report WASH-1400 (NUREG-75/014), U.S. Nuclear Regulatory Commission.

Zilliacus, R., Koukkari, P., Karjunen, T. & Sjövall, H. 2001. Formation and behaviour of organic iodine. NKS-42, ISBN 87-7893-095-2.

Appendix 1 Participants in SOS-2 project

The following persons participated in the SOS-2 project either by contributing actively to the studies summarised in this report, taking part in the sub-project group meetings, or participating in the seminars as invited speakers.

Lennart Agrenius, Agrenius
Gisle Andresen, IFE
Jean-Pierre Bento, JPB-consulting
Ingvar Berglund, FKA
Jonas Bergman, OKG
Jan-Tomas Bergström, Relcon
Helen Bigün, SwedPower
Torbjørn Bjørlo, IFE
Björn Brickstad, DNV
Per Evenėus, SwedPower
Paolo Fantoni, IFE
Yngve Flodin, SwedPower
Lars Fredlund, Ringhals
Wiktor Frid, SKI
Ninos Garis, SKI
Lars Gunsell, SKI
Veine Gustavsson, SwedPower
Anders Hallman, SKI
Anders Henoch, Ringhals
Risto Himanen, TVO
Jan Holmberg, VTT Automation (at SwedPower until 1/2000)
Göran Hultqvist, FKA
Ingemar Ingemarson, BKAB
Karl-Fredrik Ingemarsson, FKA
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Kurt Pörn, Pörn Consulting
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Ari Silde, VTT Energy
Kaisa Simola, VTT Automation
Heikki Sjövall, TVO
Snorre Sklet, SINTEF
Pekka Skogberg, BKAB
Jari Snellman, Fortum Power & Heat (Loviisa)
Leif Spanier, Sydkraft Konsult
André Strömberg, BKAB
Ola Svenson, Stockholm's University
Knud L. Thomsen, Risø
Petteri Tiippana, STUK
Esa Unga, TVO
Jussi Vaurio, Fortum Power & Heat (Loviisa)
Reino Virolainen, STUK
Per-Olof Waessman, SwedPower
Tord Walderhaug, IFE
Dan Wilson, NUSAB
Zhilin,Yang, KTH
Mika Yli-Kauhaluoma, TVO
Riitta Zilliacus, VTT Chemical Technology
Tomas Öhlin, Westinghouse Atom
Knut Øien, SINTEF

In addition to the above mentioned contributors, a number of persons took part in the seminars and interviews during the project period.

Appendix 2 SOS-2 publications

NKS-reports

Pulkkinen U. & Simola, K. (eds.) 1999. Proceedings of the NKS/SOS-2 Seminar on Risk Informed Principles (13.-14.4.1999). NKS-6.

Silde A. & Lindholm, I. 2000. On Detonation Dynamics in Hydrogen-Air-Steam Mixtures: Theory and Application to Olkiluoto Reactor Building, February 2000, NKS-9.

Karhu, A. 2000. Methods to Prevent the Source Term of Methyl lodide During a Core Melt Accident. NKS-13.

Laakso, K. & Simola, K. (eds.) 2000. Reliability centered maintenance miniseminar. NKS-18

Karhu, A. 2001. Gas phase chemistry and removal of CH_3I during a severe accident. NKS-25.

Saarenheimo, A. 2000. Reinforced concrete wall under hydrogen detonation. NKS-26.

Silde, A. & Redlinger, R. 2001. Three-dimensional simulation of hydrogen detonations in the Olkiluoto BWR reactor building. NKS-27.

Holmberg, J. & Pulkkinen, U. 2001. Experience from the comparison of two PSA-studies. NKS-36.

Zilliacus, R., Koukkari, P., Karjunen, T. & Sjövall, H. 2001. Formation and behaviour of organic iodine. NKS-42.

Holmberg, J., Pulkkinen, U., Rosqvist, T. & Simola, K. 2001. Study on decision criteria for risk informed decision support. NKS-44.

Pulkkinen U. & Simola, K. 2001. Identification and communication of uncertainties of physical models in PSA. NKS-45.

Gustavsson, V. & Möller, E. 2001. A study on hydrogen deflagration for selected severe accident sequences in Ringhals 3. NKS-46.

Salo, I. & Svenson, O. 2001. Human factors in maintenance: Development and research in Swedish nuclear power plants. NKS-48.

Pyy, P. Bento, J-P. & Eveneus, P. Analysis of errors of commission in the Nordic nuclear power plants based on plant operating experience. NKS-XX

Journal and conference papers:

Laakso, K., Sirola, M. & Holmberg, J. 1999. Decision modelling for maintenance and safety. Int. Journal of COMADEM, 2(3).

Paulsen, J. 2000. Design of simple display systems for condition monitoring, using invariants and physical laws. Proceedings of CSEPC2000, Cognitive Systems Engineering in Process Control, Taaejon, Korea.

Pulkkinen, U., Pyy, P. & Simola, K. 2000. How to enhance uncertainty analysis of PSA studies. In Cottam, M.P., Harvey, D.W., Pape, R.P. & Tait: Foresight and Precaution, Proceedings of ESREL 2000 Conference, Edinburgh, Scotland, 15-17.5.2000. pp. 1097-1105.

Simola, K. Nordic research project on nuclear safety, SOS-2. In Kondo, S. & Furuta, K.: PSAM5 - Probabilistic Safety Assessment and Management, Proceedings of PSAM5 conference, Osaka, Japan, 27.11.-1.12.2000. Vol. 1. pp.139-144.

Laakso K. & Strömberg, A. 2001. Systematic analysis of maintenance strategies of technical systems. BALTICA V International Conference on Condition and Life Management for Power Plants. Porvoo, Finland, June 6-8, 2001. VTT Symposium 211. Vol. 2. Pp. 535-545. Espoo 2001.

Saarenheimo, A., Silde, A. & Lindholm, I. 2001. A reinforced concrete structure under detonation conditions. SMIRT 16 conference 12.-17.8.2001.

M. Manninen, A. Silde, I. Lindholm, R. Huhtanen and H. Sjövall. Simulation of Hydrogen Deflagration and Detonation in a BWR Reactor Building. Accepted to be published in Nuclear Engineering and Design.

Simola, K., Pulkkinen, U., Karjalainen-Roikonen, P. & Saarenheimo, A. 2002. Comparison of approaches for estimating pipe rupture frequencies for risk-informed in-service inspections. Paper to be presented in the European Conference on System Dependability and Safety (ESREL 2002), Lyon 19.-21.3.2002

Work reports

Holmberg, J. & Pulkkinen, U. Analysis of uncertainties in probabilistic safety assessment. A position paper for NKS/SOS-2 project. Report GES 99/010. WR/SOS-2(99)1.

Silde, A. 1999. Program interface for the DETO and ABAQUS codes. Research report ENE/43/99. WR/SOS-2(99)7.

Laakso, K. 2000. A cause classification model of human errors related to maintenance. Report VTT/TAU B001. WR/SOS-2(00)1.

Paulsen, J. 2000. Transformer study. Risø National Laboratory. WR/SOS-2(00)9.

Brickstad, B. 2001. Evaluation of pipe failure probabilities for Barsebäck unit 1. DNV-Report No. 10259800-1. WR/SOS-2(01)1.

Holmberg, J. 2001. Aktivitetsrapport av FoU-projektet NKS/SOS-2.2 - Utvärdering av driftklarhetsverifieringsprocessen. WR/SOS-2(01)2.

Silde, A. & Pättikangas, T. 2001. D2A version 1.0: interface between the DET3D and ABAQUS codes. Programme document ENE4-PD-3/01. WR/SOS-2(01)3.

Sairanen, R. 2001 Recriticality due to fuel coolant interaction in the RPV lower head: initial conditions. Project report ENE4-PR-6/01. WR/SOS-2(01)4.

Hultqvist 2002. Application of risk based and risk informed decision in USA. FKA-report FT-2002-139. WR/SOS-2(02)2.

Backman, U. & Zilliacus, R. 2002. Ag nanoparticles for iodine absorption. Working Report, VTT Processes. 8 p. WR/SOS-2(02)3.

Reports planned to be published as NKS-reports in 2002

Frid, W. (ed.) 2002. Severe accident research and management in Nordic countries, a status report.

Jänkälä, K. 2002. A Risk Informed Safety Classification for a Nordic NPP.

Laakso, K. 2002. Maintenance strategic classification of equipment.

Laakso, K. 2002. State of the Art Report on research results on human reliability of maintenance at the Finnish NPPs.

Laakso, K., Paulsen, J. & Rosqvist, T. 2002. Collection and management of condition monitoring information for maintenance planning.

Miettinen et al. 2002. Recriticality due to fuel coolant interaction in the RPV lower head.

Saarenheimo, A. Silde, A. & Calonius, K. 2002. A reinforced concrete structure and a pipe outlet under hydrogen detonation conditions

Simola, K., Pulkkinen, U., Karjalainen-Roikonen, P. & Saarenheimo, A. 2002. Comparative study of approaches to estimate pipe break frequencies

Title	Advances in operational safety and severe accident research. Final report of the Nordic Nuclear Safety Research Project SOS-2
Author(s)	Kaisa Simola
Affiliation(s)	VTT Automation, Finland
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Project	NKS/SOS-2
No. of pages	97
No. of tables	12
No. of illustrations	11
No. of references	41
Abstract	A project on reactor safety was carried out as a part of the NKS programme during 1999-2001. The objective of the project was to obtain a shared Nordic view of certain key safety issues related to the operating nuclear power plants in Finland and Sweden. The focus of the project was on selected central aspects of nuclear reactor safety that are of common interest for the Nordic nuclear authorities, utilities and research bodies. The project consisted of three sub-projects. One of them concentrated on the problems related to risk-informed decision making, especially on the uncertainties and incompleteness of probabilistic safety assessments and their impact on the possibilities to use the PSA results in decision making. Another sub-project dealt with questions related to maintenance, such as human and organisational factors in maintenance and maintenance management. The focus of the third sub-project was on severe accidents. This sub-project concentrated on phenomenological studies of hydrogen combustion, formation of organic iodine, and core recriticality due to molten core coolant interaction in the lower head of reactor vessel. Moreover, the current status of severe accident research and management was reviewed.
Key words	Condition monitoring, decision criteria, HRA, human factors, hydrogen, iodine, maintenance, PSA, risk-informed decision making, reactor safety, recriticality, safety analysis, severe accidents, uncertainty

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