Nordisk kernesikkerhedsforskning
Norrænar kjarnöryggisrannsóknir
Pohjoismainen ydinturvallisuustutkimus
Nordisk kjernesikkerhetsforskning
Nordisk kärnsäkerhetsforskning
Nordic nuclear safety research

NKS-71
ISBN 87-7893-127-4

Severe Accident Research and Management in Nordic Countries

A Status Report

Wiktor Frid (ed.)
Swedish Nuclear Power Inspectorate, SKI

January 2002

Abstract

The report describes the status of severe accident research and accident management development in Finland, Sweden, Norway and Denmark. The emphasis is on severe accident phenomena and issues of special importance for the severe accident management strategies implemented in Sweden and in Finland. The main objective of the 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 has been to support 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.

Severe accident research addresses both the in-vessel and the ex-vessel accident progression phenomena and issues. Even though there are differences between Sweden and Finland as to the scope and content of the research programs, the focus of the research in both countries is on 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. Notwithstanding that our understanding of these issues has significantly improved, and that experimental data base has been largely expanded, there are still important uncertainties which motivate continued research. Other important areas are thermal-hydraulic phenomena during reflooding of an overheated partially degraded core, fission product chemistry, in particular formation of organic iodine, and hydrogen transport and combustion phenomena.

The development of severe accident management has embraced, among other things, improvements of accident mitigating procedures and strategies, further work at IFE Halden on Computerised Accident Management Support (CAMS) system, as well as plant modifications, including new instrumentation. Recent efforts in Sweden in this area have been mainly concentrated on further development of accident management strategies and aids for source term predictions whereas in Finland in addition to further development of accident management strategies some important plant modifications have been carried out.

Key words

Severe accident research, severe accident management

NKS-71
ISBN 87-7893-127-4

Pitney Bowes Management Services Denmark A/S, 2002

The report can be obtained from
NKS Secretariat
P.O. Box 30
DK – 4000 Roskilde, Denmark

Phone +45 4677 4045
Fax +45 4677 4046
www.nks.org
e-mail nks@catscience.dk

Severe Accident Research and Management in Nordic Countries

A Status Report

Edited by

W. Frid

Swedish Nuclear Power Inspectorate

January 2002

Contributors to this report

Paolo Fantoni
Institute for Energy Technology (IFE), Halden, Norway

Wiktor Frid
Swedish Nuclear Power Inspectorate (SKI), Sweden

Veine Gustavsson
SwedPower AB, Sweden

Ilona Lindholm
VTT Energy, Finland

Petra Lundström
Fortum Nuclear Services Ltd, Finland

Bal Raj Sehgal
Royal Institute of Technology (KTH), Stockholm, Sweden

Heikki Sjövall
Teollisuuden Voima Oy (TVO), Finland

Knud Ladekarl Thomsen
Risø National Laboratory, Denmark

Tord P. Walderhaug
Institute for Energy Technology (IFE), Halden, Norway

Acknowledgements

This report was financially supported by the Nordic Nuclear Safety Research (NKS) and the Swedish Nuclear Power Inspectorate (SKI). The in-kind contributions were provided by Halden Reactor Project in Norway, Risø National Laboratory in Denmark, and VTT Energy, Teollisuuden Voima Oy (TVO) and Fortum Nuclear Sevices Ltd in Finland.

Helpful comments of Dr. Kaisa Simola, VTT, and Mr. Timo Karjunen, STUK, and contribution and comments concerning PSA studies provided by Mr. Risto Himanen, TVO, are gratefully acknowledged.

CONTENTS

ABSTRACT **2**

CONTRIBUTORS TO THIS REPORT **3**

ACKNOWLEDGEMENTS **4**

1. INTRODUCTION **6**

1.1 BACKGROUND 6
1.2 OBJECTIVES, SCOPE AND ORGANISATION OF SEVERE ACCIDENT RESEARCH IN NORDIC COUNTRIES 7
1.3 OBJECTIVES AND SCOPE OF THE REPORT 9

2. THE ROLE OF SEVERE ACCIDENT RESEARCH IN REACTOR SAFETY **9**

2.1 SEVERE ACCIDENT MANAGEMENT 10
2.2 EMERGENCY PREPAREDNESS 12
2.3 PSA LEVEL 2 STUDIES 12
2.4 LICENSING DOCUMENTATION 14

3. SEVERE ACCIDENT RESEARCH IN NORDIC COUNTRIES **15**

3.1 AN OVERVIEW OF BWR AND PWR SEVERE ACCIDENT PROGRESSION AND PHENOMENA 16
3.1.1 In-vessel accident progression *17*
3.1.2 Ex-vessel accident progression *22*
3.1.3 Hydrogen combustion *24*
3.1.4 In-vessel and ex-vessel steam explosions *25*
3.1.5 Molten corium-concrete interactions *26*
3.1.6 Melt debris stabilization and coolability *27*
3.1.7 Fission product release and transport during ex-vessel accident progression *28*
3.1.8 Filtered-vent performance *29*
3.2 CONCLUSIONS AND RECOMMENDATIONS OF THE SWEDISH APRI3 PROJECT 29
3.3 UNCERTAINTIES AND RECOMMENDATIONS - KTHS PERSPECTIVE 30
3.4 CONCLUSIONS AND OPEN RESEARCH ITEMS - VTTS PERSPECTIVE 31
3.5 OECD HALDEN REACTOR PROJECT 36
3.6 RISØ NATIONAL LABORATORY 36

4. SEVERE ACCIDENT MANAGEMENT DEVELOPMENT **36**

4.1 OLKILUOTO 1 AND 2 37
4.2 LOVIISA NPP 39
4.3 SWEDISH BWRS 40
4.4 SWEDISH PWRS 41
4.5 HALDEN REACTOR 42

5. CONCLUSIONS AND RECOMMENDATIONS **44**

REFERENCES **46**

APPENDIX A: COMPUTERISED ACCIDENT MANAGEMENT SUPPORT **50**

A.1 PURPOSE AND SCOPE 50
A.2 METHODOLOGY 50
A.3 APPLICABILITY OF COMPUTERISED PROCEDURES TO SAMG 52

APPENDIX B: LOWER HEAD COOLABILITY MODEL FOR TMI-LIKE ACCIDENT **53**

B.1 REVIEW OF TMI-2 LOWER HEAD PHENOMENA 53
B.2 TOP CRUST SLAB MODEL 54
B.3 LOWER CRUST SPHERICAL MODEL 55
B.4 RECOMMENDATIONS 59

1 Introduction

1.1 Background

In the aftermath of the TMI-2 accident, the Swedish and the Finnish regulatory bodies required measures to achieve reasonable capability to manage severe accidents and to limit radioactive releases to the environment in such accidents, especially of nuclides causing long-term ground contamination. These requirements were based on, or confirmed by, governmental decisions in both countries. In the bill to the Swedish Parliament in 1980/81, the government proposed guidelines for the nuclear safety work within the frame of the Swedish nuclear power program. In 1986, the Swedish government required that the severe accident mitigation program is implemented at all nuclear power plants by the end of 1988 (Högberg, 1988). In Finland, the Council of State made in 1991 a decision concerning the safety requirements for Finnish nuclear power plants (Vuorinen et al, 1993).

The basic requirements and guidelines are similar in Sweden and in Finland. However, their actual implementation, with regard to accident mitigation systems, strategies and procedures, is specific to particular plant design. In this respect, the Nordic power reactors can be grouped into four classes: the first generation BWRs with external recirculation pumps and the second generation BWRs with internal recirculation pumps, the Swedish PWRs and the two Loviisa VVER-440 units in Finland. Obviously, as a consequence of differences between plant designs, the severe accident issues which are important for safety assessments and accident management may differ between various reactors.

Central to the fulfilment of regulatory requirements - in addition to prevention of core damage, which in accordance with the defence-in-depth principle should have the first priority - is the integrity of the reactor containment building. Therefore, in case of an accident with core damage there should be prepared emergency procedures aiming at protecting the containment and reaching as soon as possible a stable final state with the damaged core properly cooled and covered with water. The containments of the above-mentioned two generations of Nordic BWRs differ in significant way from the containment accident progression viewpoint, in particular the core melt coolability issue. In the first generation BWRs, the condensation pool covers the whole bottom area of the containment. In the second generation BWRs, the condensation pool is annular and therefore the compartment below the reactor vessel, so called lower drywell, is dry. In order to protect the containment floor and containment penetrations against core debris attack, which could result in early containment failure, the lower drywell compartment will be flooded using suppression pool water, on indication of core melt accident. Protection of the BWRs containments and the Swedish PWRs containments against late overpressurization is achieved by filtered venting.

With respect to BWRs and Swedish PWRs it was clear, considering large uncertainties in the melt progression, that strategies based upon retaining the core in the vessel could not fulfil the requirement of robustness. Therefore, the chosen strategy was to strengthen the resistance against uncontrolled containment failure. In the case of the Loviisa power plant, melt retention in the reactor vessel by means of external cooling of the vessel was considered as a possible accident mitigation measure. These considerations have to a great extent determined the objectives and the content of research programs in Sweden and in Finland as well as the work with further development of accident management strategies.

1.2 Objectives, scope and organisation of severe accident research in Nordic countries

It was realised from the start of the work with the severe accident mitigation, that there were essential uncertainties in the severe accident scenarios and phenomenology. Therefore, the solutions had to be robust both in the sense that a variety of scenarios and phenomenological uncertainties would be covered and that new information would not make essential changes of the solutions necessary.

Following implementation of the severe accident mitigation measures, the main objective of the severe accident research in Sweden and in Finland has been to continuously verify the protection provided by the measures and to further reduce the uncertainties in risk dominant phenomena. Another objective of the research has been to support validation and improvements of accident management strategies and procedures as well as to support further development of level 2 PSA (**P**robabilistic **S**afety **A**nalysis) and certain aspects of emergency preparedness.

The focus of the research has been on the in-vessel and the ex-vessel accident progression issues of importance for accident management. An example of such in-vessel issue is reflooding of an overheated partly degraded BWR core. Reflooding studies were carried out for Swedish and Finnish BWRs using SCDAP/RELAP5, MELCOR and MAAP4 computer codes (Lindholm et al, 1995). In BWRs it is necessary to consider a possibility that the partly control rod free core may become critical during the reflooding phase. Recriticality during reflooding has been investigated using RECRIT, SIMULATE-3K and APROS computer codes (Höjerup et al, 1997, Frid et al, 1999).

Another important in-vessel issue is melt coolability and melt-structure interactions in the lower plenum of the reactor pressure vessel, including an assessment of the likelihood and the mode of reactor vessel failure. Here, recent findings indicating that the probability of retaining core debris in the reactor vessel may be larger than earlier anticipated have played a significant role. For Lovissa VVER-440 reactors, research on the in-vessel melt retention through the ex-vessel cooling of the reactor lower head had high priority.

The important ex-vessel issues that have been addressed are melt fragmentation and coolability in the containment, steam-explosions and containment structural response as well as direct containment heating and hydrogen threats.

Much of the Swedish severe accident research has been carried out in co-operation between SKI and the Swedish nuclear industry, lately in the framework of a series of the APRI projects (**A**ccident **P**henomena of **R**isk **I**mportance). The main objective with these joint research projects has been to establish a common knowledge base. These co-operative efforts also have a historical background. The research programs which supported the technical basis for the implemented measures at all nuclear power plants in Sweden were the FILTRA and RAMA, RAMA II and RAMA III (**R**eactor **A**ccident **M**itigation **A**nalysis) projects. The RAMA projects were followed by the HAFOS project and then by the APRI, APRI2 and APRI3 (**A**ccident **P**henomena of **R**isk **I**mportance) projects. The major part of the Swedish severe accident research is performed at the Royal Institute of Technology (KTH) in Stockholm at the Department of Nuclear Reactor Safety.

The latest APRI project, APRI4, started in 1999 and will continue through 2002. The following issues are addressed in the project:

- experimental and analytical investigations of melt-water-structure interactions, both in-vessel (in the lower plenum) and ex-vessel

- chemistry in the primary system and the containment with focus on elimination of iodine in the BWR containments

- accident management, in order to assess the potential for improvements of accident management strategies and their further validation

- containment threats due to ex-vessel steam explosions

- core debris coolability in the containment

Participation in the international projects PHEBUS, USNRC Co-operative Severe Accident Research and EPRI ACE/ACEX is also a part of the APRI4 project.

In Finland, the major part of the severe accident research is conducted at the Technical Research Centre of Finland (VTT). The 1999 - 2002 Finnish Research Programme on Nuclear Reactor Safety addresses the severe accidents issues through the MOSES project (**MO**delling and **S**imulant **E**xperiments of **S**evere Accident Phenomena). The main goals of the MOSES project are to investigate pressure vessel lower head mechanical response under severe accident conditions, to obtain and assess experimental information on debris bed heat transfer and coolability, to examine the chemistry of fission product iodine, in the frame of the Nordic Nuclear Safety (NKS) programs and European Union (EU) research programs, to develop a computer-based training tool for severe accident mitigation using the APROS process simulator, and to evaluate the BWR containment loading during hydrogen combustion in the reactor building.

The MOSES project is a continuation of previous Finnish severe accident research projects. During 1990-1994 the thermal-hydraulic and severe accident studies were performed within the VARA II project. The next research project was the VAHTI project (1994-1996) with the overall objective to assist in the validation, implementation and reduction of uncertainties in the severe accident consequence estimates. The ROIMA (**R**eactor **A**ccidents **P**henomena and **M**odelling) project, conducted during 1997 - 1999, was a continuation of the VAHTI project.

Severe accident research in Denmark and Norway has been mainly organised in the framework of the NKS projects and, in the case of Denmark, European Union reactor safety research. Risø National Laboratory in Denmark has participated in research on, among others, aerosol phenomenology, hydrogen issues and recriticality during reflooding of a degraded BWR core. Halden Reactor Project in Norway has been developing Computerised Accident Management Support (CAMS) system and conducting experiments on fuel behaviour under abnormal operating conditions.

The severe accident research conducted in the framework of the NKS programs has been an important component of Nordic research activities. One should also mention participation in the work organised by the OECD/CSNI. Moreover, since the adherence to the EU, Sweden and Finland also take active part in the research activities carried out under the responsibility of the EU.

1.3 Objectives and scope of the report

The objectives of this report have been to present the current status of severe accident research and accident management development in Sweden, Finland, Denmark and Norway as well as to identify unresolved severe accident safety issues which require continued research. The emphasis is on severe accident phenomena and issues of special importance for the severe accident management strategies implemented at Swedish and Finnish nuclear power plants. Another aim has been to illustrate mutual interdependence of severe accident research and accident management development. Insights from well adapted severe accident research are required for validation and development of accident management, including relevant aspects of accident phenomenology, mitigating strategies and procedures as well as operator aids. And vice versa, the accident management issues often determine the direction and the content of the research.

In Chapter 2, the role which severe accident research plays in various areas of reactor safety assessment, such as accident management, level 2 PSA and emergency preparedness, is briefly described. Severe accident research in Nordic countries is presented in Chapter 3. This chapter contains an overview of the status and progress of the severe accident phenomenological research which has special importance to the severe accident management strategies adopted for the Nordic BWRs and PWRs, as well as gives an account of remaining uncertainties and recommendations for future research. Chapter 4 contains a brief overview of severe accident management development for Nordic reactors. Finally, conclusions and recommendations for future research are given in Chapter 5.

2 The role of severe accident research in reactor safety

Since several decades, large research programmes have been performed with the aim to achieve an increased knowledge about severe accident phenomena and issues. Many important safety issues have been resolved but there still remain areas where the uncertainties are large. As mentioned in the introductory chapter, the results of severe accident research are used in many different ways; thus, the knowledge about severe accident phenomena is used within the following areas:

- Severe Accident Management (SAM)
- Emergency Preparedness
- PSA level 2 studies
- Final Safety Analysis Report (FSAR) and other licensing documentation

Information about severe accident phenomena achieved from the research programmes is in most cases not directly applied as a basis for conclusions about full-scale reactors. Instead, it is used to set up models. Those are then implemented in computer codes intended for predictive calculations on power reactors. To a large extent, the information used in the areas mentioned above is obtained from such code applications.

An accident management programme comprises strategies and pre-planned actions to prevent core damage, to stop an accident in progress, to restore stable conditions and to maintain containment integrity to prevent radioactive releases to the environment. Hence, knowledge about severe accident phenomena is an important basis for development of severe accident management strategies.

Emergency preparedness contains many different aspects such as planning, organisation, education and exercises. However, in the development of emergency preparedness there is also a need for information about severe accidents. In this context, the most important issues are assessment of plant status, the time between the beginning of an accident and releases to the environment and also the magnitude and composition of the releases.

PSA level 2 studies include assessments of the probability of releases to the environment for accident sequences where core damage has been identified in the PSA level 1 studies. In order to perform such assessments, the knowledge about severe accident phenomena is required.

FSAR and other licensing documentation contain sections about severe accidents. In the past, the knowledge about severe accidents and code calculations (mainly using the MAAP code) were the basis for the implementation of the mitigation programme including the construction of filtered venting systems.

The next four sections of this chapter focus on the following aspects of severe accident phenomena related to the areas mentioned above:

- identification of accident phenomena of importance for risk assessment
- need of knowledge for these applications

Relevant severe accident phenomena are discussed very briefly since the aim here is to illustrate the role which severe accident research plays in the various areas of safety assessments. A more detailed presentation and discussion of severe accident phenomena from the physical point of view is given in Chapter 3. Most of the practical applications of severe accident research presented in this chapter are for the Nordic BWR nuclear power plants.

2.1 Severe accident management

In this section, the accident phenomena of importance for accident management are identified and the need of information about these phenomena in order to establish severe accident management strategies, is discussed. The development of accident management for Nordic reactors is described in Chapter 4.

The first priority in severe accident management is to stop fuel degradation and to keep the damaged core cooled in the reactor vessel. To achieve this goal, the core will be flooded with water, assuming the cooling water is available and that the injection systems are operable. The present strategy is very straightforward and does not take into account certain accident phenomena which could occur during reflooding of a severely damaged core.

Both in BWRs and PWRs reflooding can cause steam explosions. Moreover, in a BWR, at a stage when the control rods have melted and relocated from the core but most of the fuel is intact, reflooding can lead to recriticality. This issue has been investigated in the EU SARA-project in the fourth EU framework programme (Frid et al, 1999). Knowledge about recriticallity phenomena is valuable for the staff in the control room.

If the damaged core cannot be cooled and reactor vessel failure is impossible to avoid, measures are taken to protect the last barrier to the environment, i. e. the containment. One such a measure is to prevent reactor vessel failure at high primary pressure. Especially in BWRs, the systems for pressure reduction in the primary system before an anticipated reactor vessel failure are very reliable. Also in PWRs, the probability of pressure reduction in this case is high.

During a severe accident, large amount of hydrogen can be produced as a result of metal-water interaction. If the oxygen concentration is above 5% and the steam content is below 55% in the containment, a hydrogen deflagration can occur. Hydrogen combustion in Nordic BWRs is very unlikely since BWRs containments are filled with nitrogen during plant operation. The situation is different for PWRs, where the containment is filled with air during normal operation. For example, in the severe accident management procedures at Ringhals PWRs, the BERG (**B**eyond **E**mergency **R**esponse **G**uidelines) instructions contain restrictions related to the use of the containment spray during a severe accident when the hydrogen concentration exceeds 10% measured in dry air (Gustavsson et al, 1994). Condensation of steam at higher hydrogen concentration could start a deflagration resulting in a high pressure spike, which under certain initial conditions could threaten the containment integrity.

In BWRs, on the other hand, the containment is inerted during normal operation. However, during short periods at start-up and shutdown, the containment is filled with air. In this case a hydrogen burn is possible if the steam concentration is sufficiently low. In this scenario, the integrity of the containment might be threatened. The geometry in BWRs is more complex than that of PWRs. Therefore, the probability of turbulent hydrogen burn and DDT (Deflagration to Detonation Transition) is higher in BWR than in PWR.

In order to assess the threat against the containment integrity from hydrogen phenomena a basic knowledge about the chain of events from hydrogen production to deflagation or DDT is relevant. For PWRs a realistic estimation of the total hydrogen production (both in-vessel and ex-vessel) is important for risk assessment. However, for BWRs the amount of oxygen is usually a limiting factor. Therefore, estimation of the amount of hydrogen is not as important as in PWR cases.

In order to cool the core debris ex-vessel in BWRs containments of the second generation, the area below the reactor vessel is filled with water before anticipated reactor vessel failure. An advantage with this arrangement is that the probability of basemat failure is strongly reduced compared to the case where the melt falls down on a dry concrete floor. However, a disadvantage is that the contact between core debris and water might result in a steam explosion. Therefore, as a basis for selection of strategy for ex-vessel cooling of the core debris, knowledge about phenomena related to coolability as well as steam explosions is required.

SAM strategies have also a potential to mitigate the consequences during a severe accident. Examples of measures to minimise radiological consequences are use of containment spray, filtered venting system and chemical additives to spray and sump water. To evaluate the effect of alternative strategies, a knowledge base about thermal-hydraulic containment phenomena and fission product behaviour is needed.

2.2 Emergency Preparedness

Emergency preparedness is a complex pattern, where the utilities and the local and central authorities are involved. In Sweden and in Finland, the utility has the responsibility for actions on-site, while the local authority is responsible for the protection of people and environment off-site. The central authorities, Swedish Radiation Protection Institute and Swedish Nuclear Power Inspectorate in Sweden and Radiation and Nuclear Safety Authority in Finland, are mainly responsible for giving the local authority advice in radiological and reactor safety matters.

The bases for decisions on countermeasures to mitigate the consequences of a severe accident are judgements on the following issues related to the scenario:

- timing of core degradation and reactor vessel failure
- fission product release from the fuel
- fission product phenomena in the containment
- release of radioactivity to the environment

Timing of the accident sequence is important because it determines how long time is available before decisions concerning countermeasures have to be taken. Releases to the environment depend on the source term in the containment and possible release routes from the containment to the environment.

2.3 PSA Level 2 Studies

The main steps of a PSA level 2 study are the following:

1. The sequences in the PSA level 1 are grouped in a number of PDS (Plant Damage States). These are used as initial states for the level 2 PSA.
2. A containment event tree is constructed for each PDS.
3. A number of release categories are defined.
4. The contribution to the total release frequency from different release categories is calculated.

Consequently, the results of a PSA level 2 study comprise the following two kinds of information:

- frequencies of releases to the environment for accident sequences grouped in a number of categories, as well as single accident sequences
- magnitude, timing and composition of the release to the environment for each of these categories

Below, there are three examples of PSA level 2 studies, one for PWR and two for BWR, illustrating which severe accident phenomena were considered in the analyses.

The latest PSA level 2 study for Ringhals 2 PWR, which was a part of recurrent (every tenth year) safety review, was performed in 1994 (Gustavsson et al, 1994).

The following severe accident phenomena were included in the study because they may cause containment failure and activity releases to the environment:

- hydrogen deflagration/detonation
- Direct Containment Heating (DCH)
- steam explosion in the reactor vessel
- steam explosion in the containment
- global reactor vessel failure
- containment floor melt-through
- thermal attack on containment penetrations

The issues mentioned above are included in the event trees and in the fault trees in the study. For each of the phenomena, the contribution to the probability of leakage of activity from the containment was estimated based on research results.

The listed phenomena are different with respect to timing. The first five are highly energetic events, which under conservative assumptions may cause early containment failure and large releases to the environment. However, the probability for such events is small. The largest contribution to early containment failure in PSA Level 2 study for Ringhals 2 comes from hydrogen deflagration/detonation.

The last two phenomena on the above list, i. e. containment floor melt-through and thermal attack on containment penetrations, can cause leakage from the containment later in the accident. The releases are usually smaller in these cases compared to early containment failure.

A PSA level 2 study has also been performed for Ringhals 1, an ABB BWR (Andersson et al, 1996). The severe accident phenomena considered in this study are the following:

- hydrogen deflagration/detonation
- Direct Containment Heating (DCH)
- steam explosion in the reactor vessel
- steam explosion in the containment
- global reactor vessel failure
- containment floor melt-through
- recriticality

These are the same phenomena as for Ringhals 2 except that recriticallity replaces thermal attack on the pentrations. Even if the phenomena addressed are identical, they are usually very different in BWRs and in PWRs.

If the BWR reactor core is reflooded when the control rods have melted down and relocated from the core, but the main part of the fuel is intact, the probability for recriticality is high. As a result a power level significantly higher than the residual power, i. e. what the filtered venting system is designed for, may be produced. Consequently, the pressure in the containment increases and the containment failure pressure can be exceeded unless the power is not reduced. Therefore, recriticality gives a contribution to containment failure in PSA level 2 analysis.

A level 2 PSA study has been performed in connection with the Olkiluoto 1 and 2 modernization project during the years 1994-97. Most important parts of the model have been updated in 2000. The severe accident issue and phenomena considered in this study are the following:

- non-inerted containment during start-up and shut-down phase
- hydrogen deflagration/detonation
- total pressure in the containment (steam and non-condensable gases)
- in-vessel steam explosions
- Direct Containment Heating (DCH)
- recriticality
- melt-structure interactions in non-flooded containment
- ex-vessel steam explosions
- local and global reactor vessel failure
- debris coolability in lower drywell
- core-concrete interaction and basemat penetration

The Olkiluoto PSA clearly showed that probability of large early releases given core damage is dominated by energetic phenomena. The numeric value of 30 % is typical for BWR plants with pressure suppression containment. The fraction of sequences that can be handled with the SAM filter is only 1 % of all core damage sequences. In about 10 % of the core damage sequences the SAM measures partly limit the release below large early release. The rest of about 60 % of the core damage sequences includes the sequences where the vessel failure is prevented by restoring core cooling and containment venting is prevented by restoring containment cooling. In addition, the Olkiluoto results showed that the manual operations, like depressurization of the reactor vessel and flooding of the lower drywell, as well as the restart of the core cooling and containment spray systems, form strong dependencies between level 1 and level 2 PSA.

Finally, it should be noted that several severe accident sequences give significant contributions to the frequency of releases to the environment, but are not related to what normally is included in severe accident phenomena. Such scenarios are for instance the containment bypass sequences. For example, in PWRs an important release route is via damaged steam generator tubes.

2.4 Licensing documentation

FSAR and other licensing documentation contain sections about severe accidents. After implementation of the severe accident mitigation programme, the documentation concerning accident management as well as FSAR was updated.

In FSAR, the design of the mitigating systems, e.g. the filtered venting system and the redundant water supply to the containment spray is described. The design analyses are to a large extent based on results from the MAAP code. In order to validate MAAP, comparisons with experiments and mechanistic codes have been performed. Thus, the use of results from severe accident research in this context is twofold:

- results from experiments can be used directly for validation of codes
- experimental results are used to build models, which are then incorporated into advanced so called mechanistic computer codes, such as for instance SCDAP/RELAP5

The mechanistic codes can then be used for validation of less accurate so called integral codes, such as MAAP and MELCOR

The FSAR severe accident analyses for Olkiluoto 1 and 2 were updated during the modernization project. The analyses performed on severe accidents were divided into the following categories (computer codes used in calculations are indicated in parentheses):

- containment structural capability (ANSYS, SOLVIA)

- conditions inside the containment (MELCOR, MAAP)

- conditions at the plant (hand calculations)

- source terms (MELCOR, MAAP)

Specific codes were used to calculate phenomena like vessel failure (PASULA) and steam explosions (SAPHIRA-PREMIX, SAPHIRA-FCI, PM-ALPHA, ESPROSE.m). Assessment of melt coolability is mainly based on experiments.

There are also certain specific issues, which are addressed in FSAR for PWRs, namely the following:

- ex-vessel debris coolability in the containment
- hydrogen deflagration/detonation
- reactor vessel failure at high primary pressure
- steam explosions
- recriticallity of core debris in the containment

3 Severe accident research in Nordic countries

In this chapter, the status and progress of severe accident research, including uncertainties and recommendations for continued work, is reviewed. The first section contains a general and relatively detailed presentation and discussion of severe accident phenomena of importance for Nordic reactors. Two particular research projects are described in some detail in Appendices, namely the development of Computerised Accident Management Support (CAMS) system by OECD Halden Reactor Project in Norway (Appendix A) and the development of Lower Head Coolability Model for TMI-like Accident by Risø National Laboratory in Denmark (Appendix B).

It should be mentioned here that Sweden, Finland and Denmark have participated in many severe accident research projects in the 1994-1998 Forth Framework Programme of European Union Research in Reactor Safety. These projects are (FISA 99):

- Molten Fuel Coolant Interactions (MFCI)
- Core Melt Pressure Vessel Interactions During a LWR Severe Accident (MVI)
- Severe Accident Recriticality Analyses (SARA)
- Reactor Vessel Integrity in Severe Accidents (REVISA)
- The Development and Demonstration of Integrated Models for the Evaluation of Severe Accident Management Strategies (SAMEM)

- Severe Accident Management Implementation and Expertise in the European Union (SAMIME)
- Benchmark Exercise on Expert Judgement Techniques in PSA Level 2 (BEEJT)
- A Data Set for Level 2 Probabilistic Safety Analysis Studies (PSAL2)
- Corium Spreading and Coolability (CSC)
- Melt Stratification for In-vessel and Ex-vessel Events (STRATIEX)
- Fission Product Vapour/Aerososl Chemistry in the Primary Circuit (CHEM)
- Iodine Chemistry (IC)
- Organic Iodine Chemistry (OIC)
- Aerosol Physics in Containment (APC)
- A Risk-Based Evaluation of the Impact of Key Uncertainties on the Predictions of Severe Accident Source Terms (STU)

Nordic countries also participate in a number of research projects in the ongoing Fifth Framework Programme of European Union Research in Reactor Safety.

3.1 An overview of BWR and PWR severe accident progression and phenomena

A severe accident by definition involves melting of the core and release of radioactivity. Clearly, the phenomena involved in a core-melt accident are extremely complicated, since the main characteristics of the accident scenario are the interactions of the core melt with structures, and water, and the release, transport and deposition of the fission product carrying vapours and aerosols. The interactions of core melt may lead to: (1) ablation of structures, (2) steam explosions, (3) concrete melting and gas generation, and (4) dispersion of heat-generating melt (debris). These phenomena involve the disciplines of thermal hydraulics, high temperature chemistry, high temperature material interactions, aerosol physics, among others. Predictions of the consequences of a severe accident have to be based on experimentation and models whose veracity may be limited by the scale at which the information about the phenomenology is derived. Scaling considerations become very important since large scale experiments with prototypic melts are very expensive and difficult to perform.

Another aspect about severe accident consequences should be mentioned. The LWR safety systems for the design base accidents have an acceptance criterion: the peak-clad temperature has to be maintained below 1200 ^{0}C, while employing conservative methods of analyses. No such criterion exists for severe accidents, which would focus the research adequately. Recently, the core damage frequency (CDF) < 10^{-4} to 10^{-6} and the conditional probability of containment failure < 0.1, are becoming criteria for severe accidents. This, however, is a probabilistic criterion and is subject to some interpretation. The CDF criterion also is not used as a design basis, but as a design goal. In the same vein, the research accomplishments are harder to evaluate, since there is no specific measure.

As mentioned above, it became clear quite early, and confirmed by the WASH-1400 (USNRC, 1975) and NUREG-1150 (USNRC, 1987) studies, that the containment had a central role in protecting the public against the consequences of a severe accident. Thus, the focus of the severe accident research became the evaluation of the survivability of the containment for the various severe accident scenarios. More recently, the focus has shifted a little, due to the accident mitigation perspective, from the survivability of the containment to that of the survivability of the vessel. Vessel external flooding has been adopted in the AP-600 design (Theofanous et al, 1995), and has been back-fitted in the containment of the Loviisa power plant in Finland (Kymalainen et al, 1997).

In this section, the progress of the severe accident phenomenological research will be described. The focus is placed on the phenomena of severe accident progression both in-vessel and ex-vessel. Special attention is paid to steam explosion, debris coolability, hydrogen release and combustion, fission product release and transport, and filtered-vent performance, which have special importance to the severe accident management (SAM) scheme adopted for the Nordic BWRs and the Swedish PWRs.

3.1.1 In-vessel accident progression

It is perhaps instructive to delineate the time scales involved in the various phases of the in-vessel accident progression. The core boil-off and the initial heat-up process are relatively lengthy (2-3 hours), before significant core damage takes place. Accident termination during this time is relatively straightforward, if operator is able to add water to the reactor vessel.

Clad melting, fuel melting, core blockage and core melt pool formation are relatively shorter duration processes (1/2 to 1 hour), during which access of water to some of the blockages and debris beds formed may become limited. The interaction of the core melt with the lower head water and structure, and the failure of lower head may be relatively longer duration (3 hours) processes if the melt quenches and re-heats. Alternatively, if melt cooling/quenching does not occur, the lower head may fail relatively fast (minutes). The character of the melt discharged to containment is different in the two scenarios.

Early phase of in-vessel accident progression

A severe accident in a PWR starts with core uncovery initiated by loss of reactor coolant inventory and failure of some of the reactor safety systems. The in-vessel progression of the accident, from that point on, is determined by thermal-hydraulics and material interactions. If accident management actions are not successful, the rise in core temperatures due to undercooling leads to exothermic Zircaloy oxidation transient which delivers heat to clad and fuel at a very large rate (~10 times the decay energy rate), and a large amount of hydrogen is produced and released to the containment. Core temperatures rise at the rate of 1 to 10K/sec and melting starts with the structural and control rod materials and progresses in turn to clad, fuel eutectic, and fuel. Substantial loss of geometry takes place, and a melt pool may be formed within the original core boundary as happened in the TMI-2 reactor. Eventually, the molten core material may be discharged, as a jet, to the lower plenum as occurred in TMI-2. Alternatively, the core slumps and eventually attacks, thermally and mechanically, the core support structure. Failure of the support plate or core barrel brings the corium (molten fuel-structure mixture) to the lower head. This ends the early phase of the in-vessel accident progression.

During the early phase of in-vessel accident progression the parameters of interest to the containment integrity are:

- the magnitude and rate of hydrogen generation
- the elapsed time before the onset of core melting
- the temperature levels of the reactor coolant system (RCS)

Information about hydrogen generated (and released to containment) is required for its management and for establishing that detonations or transitions to detonation will not occur. Information about the elapsed time before onset of core melting provides the time window, available to the operator, for terminating the accident without core damage or fission product release. During core-heat-up, a considerable fraction of energy generated may be transferred to the RCS by natural circulation of the steam generated, which may become hot enough to induce local failures. This could change the risk-dominant high pressure accident scenario, thus, accurate prediction of RCS temperature levels is essential in determining the consequences of some of accident scenarios.

Much research has been performed for the early phase of the in-vessel melt progression. A representative experimental research program is CORA (Hagen et al, 1997) in which several bundles representing PWR and BWR fuel arrangements were heated electrically and observations on fuel degradation were obtained. Previously, experiments were performed with the PBF (McDonald et al, 1983) and LOFT (Carboneau et al, 1989) reactor facilities and, currently, PHEBUS (Livolant et al, 1996) experimental program is directed towards in-vessel melt progression, and fission product release, transport and revolatilization.

Clearly, the above research programs have produced results which have reduced uncertainty. The state of knowledge with respect to the PWR in-vessel core melt progression confirms the picture conveyed by TMI-2. It is believed that a melt pool will form in the original core volume and will drain along the side of the core into the lower plenum to commence the loading on the lower head.

The state of knowledge regarding BWR in-vessel melt progression, in particular for the higher probability depressurized dry core scenario, is relatively confused. Core wide blockage formation could occur, similar to that for a PWR; however, there is not enough data, or analysis to delineate the conditions, under which it could occur or not occur. Thus, it is conceivable that the BWR in-core melt progression may terminate with failure of the core support plate.

The effects of accident management actions, e.g. water addition to a hot core, have been considered recently. It was found in the CORA tests (Hagen et al, 1997) that this increases the core damage and the hydrogen generation, due to the increase in Zircaloy oxidation by the steam produced. A new facility QUENCH (Sepold et al, 1999) was constructed with European funding to further investigate the increase in hydrogen generation as a function of the clad surface conditions. It was found that if a reasonably thick (~ 300 μm) oxide layer is present on the clad surface, the release of additional hydrogen during the quench process is not large. The converse is true if there is no oxide layer present on the clad surface. It is expected that the clad surface which has undergone some oxidation, prior to the accident management action of bringing water to the hot core, will be covered by a relatively thick (~ 500 μm) oxide layer. The oxide layer in the QUENCH experiment suffered some cracks, which allowed some hydrogen generation. The fresh clad tested produced much hydrogen and damage to the fuel bundle resulted due to the exothermic energy generated. In all cases fuel bundle quenched eventually.

An in-vessel issue related to the BWR accident management is that of addition of unborated cold water to the partially damaged core in which the control rods may have melted and the boron-carbide accumulated on the core support plate. Investigations on the reactivity effects of this scenario have been pursued in an EU Project (Frid et al, 1999). The Doppler effect and the void feed back mitigate the core damage in most of studied cases. However, for very high reflooding rates the analyses have indicated a possibility of local core damage. A prudent change in the emergency procedures maybe to add boron separately, as it is prescribed for the anticipated transient without scram (ATWS) event.

Late phase of in-vessel accident progression

Accurate description of the late phase of the in-vessel severe accident scenarios has assumed greater importance lately, since it has become evident that the assumptions made in its modelling determine the composition, amount and the rate of corium discharged to the containment, to which the containment loadings are directly related. In particular, if the projected loadings are severe enough to fail a containment soon after the vessel failure, e.g., due to direct containment heating or hydrogen detonation, the "source term" consequences of a severe accident can be very severe indeed.

The late phase of in-vessel accident progression did not receive as much attention before, except for some specific evaluations e.g. that of the AP-600 in-vessel melt retention (Theofanous et al, 1995). Recently more generic investigations have been pursued in a recently concluded EU Project in which the following questions were addressed (Sehgal et al, 1997a, Sehgal et al, 1997b, Sehgal et al, 1999b):

- Can the lower head fail immediately, in spite of the presence of water, due to the attack of a melt jet released from the core?

- Can the melt debris be cooled by the water in the lower head to preclude vessel failure?

- If the water can not be supplied, can the melt be retained within the lower head by cooling the external surface with water?

- In the absence of water, inside and outside of the lower head, how long will it take to fail the lower head by melting and creep processes?

- What is the mode and location of lower head failure and is it affected by the presence of the penetrations in the lower head?

and finally

- What is the rate of enlargement of a local lower-head-failure-site caused by the flow of melt through it?

The melt jet discharged from the core during its interactions with the lower head water would fragment and could generate a steam explosion. The questions relevant to that process are:

- What is the fraction of the melt jet that fragments in water?

- Can the steam explosion cause the failure of the lower head?

It is recognised that there is a relatively broad consensus that an in-vessel steam explosion will not cause containment failure, however, there is no consensus that a steam explosion can not cause lower head failure, particularly at the location of a penetration.

The investigation performed (Theofanous et al, 1995) for establishing the feasibility of the in-vessel melt retention for the AP-600 and those performed in the EU projects, Melt Vessel Interactions (MVI) and Molten Fuel Coolant Interactions (MFCI) have provided quite well-validated responses to some of the issues raised above. These are:

- It appears (Sehgal et al, 1997b) that the immediate failure of the lower head due to the impingement of a melt jet dropped from the core is physically unreasonable. Only in the case of a long-running thin melt jet attacking the lower head wall without water, there could be an ablative failure. This, however, is a physically unreasonable occurrence.

- The FARO experiments (Magallon et al, 1997) have shown that between 40 and 60% of the melt jet would fragment, and the remainder could form a cake of very low porosity at the bottom of the debris bed. The long-term coolability of such a bed has not been established.

- Much work performed recently (Sehgal et al, 1998a) and ongoing in the RASPLAV Project (Asmolov, 1998) has clarified the limitations on the power level of a reactor which would be amenable to melt retention in lower head by the cooling of the vessel from outside. It appears that the plants with electrical power generation level beyond 1000 MWe may not have sufficient margin. Recent results from RASPLAV have added the uncertainty of melt pool stratification, whose effect on the margins has not been clarified so far.

- Many experiments performed in the KROTOS facility (Huhtiniemi et al, 1999) with UO_2-ZrO_2 melt jets, and one very recently in the FARO facility, have failed to produce strongly-propagating steam explosions. On the contrary, spontaneous explosions have been observed when Al_2O_3 melt jets are employed. It appears that the explosivity and efficiency of a steam explosion with UO_2-ZrO_2 melt interacting with saturated or subcooled water is extremely low.

- The ablation of the vessel failure site was measured and scaling analysis developed (Sehgal et al, 1997b). It was found that a crust layer persists, reducing the heat transfer from the melt stream to the vessel wall. The most probable hole size, after ablation by the melt in a prototypic scenario, may be in the range of 15 to 20 cms. These are much lower estimates than those derived earlier.

- Considerable experimentation (Sehgal et al, 1998a, Sehgal et al, 1998b) and analyses (Sehgal et al, 1999a) have indicated that global vessel failure is highly unlikely for both PWRs and BWRs. The most probable mode of failure for the vessel is the creep of the lower head and the likely location of failure would be around a penetration. For the scenarios in which melt pool convection is established in the lower head, the likely location of failure is near the upper elevations of the hemispherical head, where the temperatures are the highest.

The results described above have been obtained in the last 5-7 years and the technology developed provides a relatively good basis for the description of the processes occurring in the late phase of the in-vessel melt progression. More work is needed, in particular, to:

- understand the reasons for the low explosivity of UO_2-ZrO_2 melt (this is also necessary for the evaluation of the consequences of ex-vessel steam explosions)

- to explore the coolability, in vessel, by either gap cooling (for melt pool) or water ingression (for a debris bed)

- to determine the fragility of lower head against dynamic loads

- to obtain confirmatory results on the timing, mode and location of the lower head failure for the commonly-used pressure vessel steels (It has been observed that the creep deformation laws for the various pressure vessels steels are very different from each other.)

Fission product release and transport during in-vessel accident progression

The "source term", i.e., the magnitude, the chemical and the physical form of the fission product source distribution in the containment atmosphere received great attention right after the TMI-2 accident and currently the PHEBUS Project is providing confirmatory data on this subject. During the in-vessel accident progression phase, the parameters of interest are:

- the fraction of the core fission product inventory released

- the fission product chemical species

- the fraction of released fission products deposited on the reactor coolant system (RCS) surfaces

- the revaporization of the fission products from the RCS surfaces

The research work pursued made great progress and provided good estimates for the parameters above. It was found that, in general, 70 to 80% of the volatile fission products inventory is vaporized from the core, except for tellurium, some fraction of which is retained by the unoxidized Zirconium in the core and is released as Zr oxidizes. The fission product vapors change into aerosols as they cool down in the cooler parts of the RCS and aerosol physics determines the fission product deposition on the RCS surfaces. A substantial fraction of the fission products released from the core will deposit in the primary system before exit from the break location to the containment. The deposited fission products, thus, are not immediately available as the source term; however, as the temperatures in the RCS increase due to the continued decay heat generation by fission products, the revaporization of the deposited volatile fission products occurs and in time much of the deposited volatile fission products will leave the RCS and enter the containment. Early on, the importance of the re-vaporization process was not fully realized, however it has become quite clear that re-vaporization plays a significant role in determining the fission product "source term" for the cases of late containment failure.

The total release of relatively low volatile fission products, e.g., oxides and hydroxides of Ba, Sr, Ru, Ce etc., during the early phase of in-vessel accident progression, is of the order of a few percent of the inventory at most. The Molybdenum is an exception since its release is significant. However, the release estimate is based on very uncertain knowledge about the chemistry of Molybdenum.

During the late phase of the in-vessel accident progression, the vessel lower head may be full of a convecting high temperature melt pool, which may contribute to a release of the non-volatile fission products. The in-vessel melt retention accident management scheme results in the high temperature melt pool residing in the lower head for hours or days. There are very little data on the release of the less-volatile fission products from a high temperature melt pool. The melt pool upper surface will have a crust. The efficiency of the crust in stopping the fission products is not known. Such information will be needed for estimation of the source term if the in-vessel accident management scheme is adopted, for new or existing plants.

The chemical character of the fission products released is an important element in the estimation of the source term. The research work conducted after the TMI-2 accident identified the compounds formed by the various fission products during their release in the core and also during their transport in the RCS. The dominant species for Iodine and Cs releases were found to be CsI and CsOH, which are extremely soluble in the water present in the containment and the sump. The recent PHEBUS tests (Ktorza et al, 1999) have found that a few percent of the total Iodine release may be in the form of Iodine gas, and that silver Iodide may be formed. The small amount of the gaseous iodine, released from the core, was found to diminish rapidly during its stay in the containment.

3.1.2 Ex-vessel accident progression

The ex-vessel accident progression is basically the interaction of the products of the in-vessel accident progression, namely fission products, hydrogen and corium melt with the contents of the containment. The pressure and temperature loadings exercised during these interactions on the containment structure may cause failure of the containment which should be prevented. Thus, the study of the ex-vessel accident progression is primarily that of the containment loadings, and of the evaluation of the probability of its failure. In this respect two time zones can be defined, namely "early" and "late" for the failure of the containment. This distinction results from the observations on the radioactive aerosol source in the containment, which diminishes, exponentially with time, due to its deposition on the containment floor and surfaces, and its dissolution in water. It has been observed that with steam in the containment atmosphere 99.9% of the aerosol in the containment atmosphere are removed in 4-6 hours. Thus, the time span of interest for the early failure of containment is 4-6 hours and for the late failure of containment more than 4-6 hours. It should be obvious that the greater public hazard is posed by the early failure of the containment.

Early failure of containment

After a prolonged review of the severe accident scenarios, initially by the Containment Loads Working Group, formed by the USNRC and later by the expert panel working with the Sandia National Laboratories on the NUREG-1150 report (USNRC, 1987), the following major challenges, which may lead to an early failure of LWR containments, were identified:

- direct containment heating as a result of melt discharge at high pressure from a vessel breach in a PWR

- melt attack on the liner of the BWR Mark I containment (this case is not relevant for Nordic BWR plants)

- hydrogen detonation

- in-vessel and ex-vessel steam explosion

Each of these challenges, in turn, became a severe accident issue and led to several years of concentrated research. Some of these issues are resolved, or close to resolution, while others still are far from resolution. By resolution, we mean that a technical consensus is reached on either the adequacy of the existing containment systems to meet the challenge posed with a very high degree of confidence or that a technical consensus is reached on the necessary measures (accident management and/or backfit), which would impart that character to the existing containment systems.

Late failure of containment

The time span of interest is beyond 4 hours after the initial release of radioactivity in the containment. In this time span, if the melt is discharged into the containment, it is essential that a heat transport system is established within the containment, i.e., the containment heat removal systems, e.g., fan coolers in PWRs and suppression pool coolers in BWRs are functioning. Otherwise, the slow pressurization resulting from either the prolonged heat addition to the containment atmosphere, or the generation of steam from melt (debris bed) cooling, or the non-condensable gases generated from the molten corium-concrete interaction (MCCI) can reach pressure levels at which the containment may fail or leak excessively. This may occur after several hours (more than 4), or a few days, depending upon the water availability, the type of concrete and the pressure-bearing capacity of the containment. Another potential radioactivity pathway to the environment can result from the containment basemat penetration when the melt can not be cooled and it keeps attacking the basemat. This may occur after a day, or after many days, depending upon the heat removal from the melt debris, the type of concrete, and the thickness of the basemat.

The outstanding safety issues, identified for this time span are:

- melt spreading
- melt (debris) coolability
- concrete ablation rate
- non-condensable gas generation rate
- stabilization and termination of accident
- performance of venting (filter) systems

3.1.3 Hydrogen combustion

The hydrogen combustion loads on the containment were the first to be addressed by the USNRC, since the hydrogen combustion event in TMI-2 triggered a heightened awareness of these loads. The hydrogen rule requires management of hydrogen concentration in the containment resulting from the oxidation of up to 75% of Zirconium clad. This has already been incorporated in the ice condenser, BWR Mark III and BWR Mark II and I plants. The BWR Mark I and II plant containments are inerted, as all Swedish and Finnish BWRs, while the ice condensers and BWR Mark III plant have been fitted with igniters. The large volume of PWR containments were judged, in general, to be immune, since in most cases the hydrogen concentration did not reach high enough to produce combustion-induced pressure loads, which would threaten containment integrity. The hydrogen combustion loads issue for these plants relates to either high local concentration, or the transition to detonation, which can occur for special geometries (ducts, accelerating flow regions etc.) at relatively low (<10%) hydrogen concentrations.

Hydrogen mixing research has been performed at several laboratories and several large experiments have been performed (Takumi et al, 1993, Wolf et al, 1993). The overall conclusion derived from these experiments and from analytic studies is that hydrogen mixing is quite efficient and local non-homogenities do not persist for long periods, except when they are coincident with thermal stratification effects. Recently some very large-scale CFD calculations have been performed for several accident events in the complex geometry of an actual containment. These calculations do indicate some local concentrations of hydrogen greater than the average. Such complex analyses have been employed to determine the preferred locations for hydrogen recombiners; the hydrogen control option that has been implemented at a number of power plants in Europa. There has been extensive proprietary research, and testing, on the hydrogen recombiners to determine their performance in different environments that a containment may be subjected to during the course of a severe accident.

The current focus of hydrogen combustion research is on the issue of transition to detonation and for what geometrical conditions and hydrogen concentrations this phenomenon can occur. Experiments were performed at BNL (Ciccarelli et al, 1993) and are currently being performed at the RUT facility near Moscow, Russia. The main difficulty is in scaling the experimental results obtained to the prototypic geometries in containment, which could be prone to such transitions. Very recent work (Dorofeev et al, 1999) has indicated that flame acceleration and fast combustion (leading to detonation) can occur under favorable conditions, at sufficiently large scale, for only strong mixtures. Such mixtures have a value of expansion ratio greater than a critical value, which is a function of the Zeldovich and Lewis numbers. Measurements performed so far have already provided some estimates of the critical values, inspite of the uncertainties. More measurements are scheduled to cover the influencing parameters for which the data are lacking.

3.1.4 In-vessel and ex-vessel steam explosions

The steam explosion loads on the containment were first considered in the WASH-1400 report and, because of the assumptions made about the nature of this event at that time, the failure of containment (due to in-vessel steam explosion generated missiles) contributed a substantial fraction of the probability for early containment failure. The work on steam explosions (Theofanous et al, 1987) since that time, led to more realistic estimates of the probability of containment failure due to in-vessel steam explosions. A steam explosion review group (SERG) established in 1995 (SERG2, 1995), deliberated on the phenomenology of the steam explosion and provided expert estimates on the probability of the containment failure as a result of an in-vessel steam explosion. Although there were some differences of opinion, the vast majority of the experts concluded that the conditional probability (i.e., if there is a core melt) is less than 0.001, i.e., the containment failure is physically unreasonable.

Recent tests in the BERDA program at Forschungszentrum Karlsruhe (FZK) in Germany have shown that for a scaled upper vessel head subjected to impact loads, simulating those from a very strong steam explosion, the head and the bolts survived.

Much experimental and analysis-development work is in progress, presently, on in-vessel steam explosions. Experiments have been performed with several kilogram quantities of heated particles and molten materials. Elaborate three-field analysis codes MC3D (Berthoud et al, 1997), IVA (Kolev, 1999), ESPROSE.m (Theofanous et al, 1996a) and PM-ALPHA (Theofanous et al, 1996b) have been developed. Some of the insights gained are: (1) steam explosion probability is much reduced due to the extensive water-depletion that occurs around the fragmented particles of a jet in the premixture, and (2) super-critical steam explosions, however, can not be excluded.

Ex-vessel steam explosion loads on PWR and BWR containments are also an issue, since (1) in some PWRs, water discharged from the reactor primary system accumulates in the reactor cavity under the vessel and (2) in some BWRs, a deep water pool is established under the vessel, prior to vessel failure, as in accident management strategy employed in the Swedish and the Finnish BWRs. The ex-vessel water is generally highly subcooled and the extensive voiding, that develops in the premixture in a saturated pool, may not occur in the subcooled pool. Additionally, it has been found that the median particle size, obtained during the break-up process, may be much smaller for the subcooled water than for the saturated water. Contrary to these effects, which may argue, on heuristic grounds, for a larger probability of a steam explosion, there are the effects of cooling and solidification which argue for a reduction in the probability of a steam explosion.

The corium melt may be a complex mixture of metals and oxides, however, predominantly it is a mixture of UO_2-ZrO_2-Zr, whose phase diagram, in general, shows a liquidus curve and a solidus curve, which are apart from each other by at most 200 to 300 K. For the UO_2-ZrO_2 mixture the difference between the liquidus and the solidus curve is only 50 to 75K. As the corium mixture solidifies its properties change radically. In particular, the viscosity, which is infinite in the limit of solidus, changes radically. The process of break up of a corium melt jet during its interaction with water results in many corium melt droplets of complex shape undergoing solidification from the exterior surface to the interior of the droplets. The changes occurring in the physical properties of the droplets affect the potential for the participation of the droplets in the steam explosion process. For example, it has been found that a thin high viscosity layer on the surface of a spherical droplet will greatly impede its subsequent fragmentation by a pressure wave, or shear forces.

The most remarkable experimental observations derived from the experimental program employing prototypic corium melt (UO_2-ZrO_2) in the FARO (Magallon et al, 1999) and UO_2-ZrO_2 and Al_2O_3 in the KROTOS (Huhtiniemi et al, 1999) facilities at Ispra, Italy are:

- UO_2-ZrO_2 melt jets dropped in subcooled and saturated water at low pressure do not generate spontaneous steam explosions,

- strongly-triggered UO_2-ZrO_2 melt jets in subcooled and saturated water at low pressure may develop a propagating event, however, of very low efficiency (< 0.15%),

- Al_2O_3 melt jets (serving as a stimulant for the corium fuel) generally experience spontaneous strong steam explosions when dropped in low pressure subcooled water, and

- Al_2O_3 melt jets dropped in saturated water at low pressure, in general, have to be triggered to experience strong steam explosions.

These significant observations point to the important role that the melt physical properties may be playing in the steam explosion process. Much research on this aspect is being pursued in Europe under the auspices of the European Commission. Some physical mechanisms have been identified. Nevertheless, it appears that the prototypic corium mixtures may not be as explosive (very low efficiency and /or explosivity) as previously sometimes assumed to be.

3.1.5 Molten corium-concrete interactions

In a dry containment, the melt discharged from the vessel, after the short-time-spreading process, will attack the basemat concrete. The concrete ablation (melting accompanied by gas generation) occurs at much lower temperature than the melt temperature, resulting in substantial erosion of the basemat. The ablation process can continue, indefinitely, if a crust is formed on the melt upper surface, practically eliminating the heat loss from the melt upper surface. The rate of ablation in this limit would be governed by the melt heat generation rate and the ablation enthalpy of the concrete employed in the basemat. Thus, basemat melt-through can be envisioned. Concurrently, the gas generated during the concrete ablation process keeps pressurising the containment and containment failure can be envisioned.

Molten corium-concrete interactions (MCCI) research has been conducted over many years. A substantial body of experimental data have been accumulated from quite expensive programs e.g. SURC, BETA and ACE, where experiments were performed with heated corium and iron melts. Analysis development culminated in the codes CORCON (Cole et al, 1984) and WECHSEL (Reimann et al, 1990), which have employed 2-D and 1-D analysis with primarily empirical heat transfer correlations. These codes have also represented the major chemical reactions taking place during the interactions.

The experience in validating these codes has been, basically, not as satisfying as one would like. The codes predict the measured ablation rate and total ablation within 30%. The same is true for the prediction of the combustible (H_2, CO) and non-combustible (CO_2, steam) gas generation rates. There are several uncertainties in the choice of parameters and there is the fear that some phenomena are not being modelled or are modelled incorrectly.

One phenomenon which has been recently identified (Froment et al, 1999) is that of melt segregation, which may have a greater contribution in the late phase of concrete ablation than in the early phase. This phenomenon may lead to higher concentration of Uranium oxide near the bottom of the melt pool resulting in non-uniform heat generation in the pool. Inclusion of the melt segregation modelling in the overall MCCI process has led to prediction of pool temperatures which were close to those measured in the ACE tests employing prototypic melt compositions. Complete influence of the melt segregation phenomenon on the consequences of the MCCI process has yet to be determined.

3.1.6 Melt debris stabilization and coolability

Melt coolability is perhaps the most vexing issue impacting severe accident containment performance in the long term. As mentioned earlier, melt coolability is essential to prevent both the basemat melt-through and the continued containment pressurization, thereby, to stabilize and to terminate the accident, without the fear of radioactivity release from the containment.

Provision of deep (or shallow) water pools under the vessel may not assure long term coolability/quenchability of the melt discharged from the vessel. Interaction of the melt jet may lead to very small particles (in the event of a steam explosion), which may be difficult to cool in the form of a debris bed of low porosity. Incomplete fragmentation will lead to a melt layer on the concrete basemat under a particulate debris layer and a water layer.

Coolability of a melt pool interacting with a concrete basemat by a water overlayer has been under intense investigation in the MACE Project (Sehgal et al, 1992), sponsored by an international consortium and managed by EPRI. The experimental work has been performed at Argonne National Laboratory (ANL) in USA. Three experiments were performed in which melt pools of 30 cm x 30 cm x 15 cm depth, 50 cm x 50 cm x 25 cm depth and 120 cm x 120 cm x 20 cm depth were generated on top of concrete base-mats and water added on top. The melt material contained Uranium oxide, Zirconium oxide, Zirconium and some concrete products. The decay heat generation in the melt was simulated through electrical heating. It was found that for these three tests, the effect of the sidewall dominated the phenomena, since an insulating crust was formed, which attached itself to the sidewalls. The crust prevented intimate melt-water contact and the heat transfer rate slowly decreased from approximately 2 to 0.1 MW/m^2, which is less than the decay heat input to the melt.

Three modes of heat removal from the melt pool have been identified. These are: (1) the initial melt-water contact, (2) the conduction through the crust, and (3) melt eruptions into water, when the heat generated in the melt is greater than that removed by conduction through the crust. In the large test (120 x 120 x 20 cm), it appears that significant water ingression occurred since after the test the crust (or cooled melt) was 10 cm thick, i.e., about half the melt was cooled. Continued concrete ablation leads to the separation of the melt pool from the suspended crust, and the conduction heat transfer decreases substantially.

The results of 50 x 50 x 25 cm integral melt coolability test with siliceous concrete were approximately the same as for the earlier tests. Presently, no definite experimental proof of melt pool coolability with a water overlayer can be offered. However, it appears that crust can not be maintained as a solid body for spans of several meters found in prototypic-geometry containments.

Melt coolability has been investigated at FZK in the COMET facility (Alsmeyer et al, 1998) employing water entry at the bottom of the melt pool. This new approach works since it has been found that the injected water creates sufficient porosity in the melt pool to cool the melt in a relatively short time. Several experiments have been performed at different scales with Al_2O_3 and iron melt pools to prove the concept. The concept has been directed towards the design of a core catcher for a new containment design at FZK. The core catcher top face is made of some tens of millimetres of sacrificial concrete, under which nozzles are embedded in the basemat. These nozzles open when the concrete is ablated and inject water from the bottom into the melt pool. The COMET concept has been optimized through many experiments. No steam explosions have been experienced. It appears that addition of the sacrificial concrete in the Al_2O_3-iron melt considerably reduces the explosivity of the melt.

Presently, the physical mechanism that creates porosity in the melt with water injection from below is not known.

3.1.7 Fission product release and transport during ex-vessel accident progression

The fission products and the core materials released during the core heat up process arrive into the containment, from the break, as aerosols. Their transport in the containment is governed by aerosol physics, which determines the fission product concentration in the containment atmosphere as a function of time. As mentioned earlier, if there is steam atmosphere in the containment (as it should be for a severe accident), the fission product aerosol concentration in the containment atmosphere decreases exponentially with time, largely due to the process of aerosol particle size growth (due to steam condensation), agglomeration and sedimentation. Another aerosol deposition process is that of Stefan flow carrying aerosols to the walls of the containment where the steam is condensing. As mentioned earlier, typically, fission product concentration in the containment atmosphere can decrease by a factor of 10^4 in about four hours.

The release of fission products during the ex-vessel accident progression can occur during the MCCI due to the gas sparging and the high temperatures in the melt. The releases of interest are those of the less-volatile fission products, e.g. Ba, Sr, Ce, Ru and Mo, since the volatile fission products have already been released.

The ACE experiments provided systematic data on the release of the above-mentioned fission products. In general, it was found that the releases were much smaller than what were previously calculated. The measured values for releases were less than 1% of the inventory for all of the less-volatile fission products. Recently an analysis of the ACE experiments points out that these releases occurred after all of the Zr contained in the melt had been oxidized. If such was not the case, the fission product releases could be larger. Thus, some uncertainly has been created with respect to the implications of data obtained in the ACE tests.

3.1.8 Filtered-vent performance

All Swedish and Finnish BWRs and Swedish PWRs have been fitted with filtered vents. The design of the filtered vent has been based on the tests performed during the LACE Project supported by a consortium of international organizations and managed by EPRI. Full-scale prototypic filters were employed and the decontamination factors (DF) measured were very large (10^3-10^5). In general filtered vents provide a relatively safe way of relieving the pressure in containment.

3.2 Conclusions and recommendations of the Swedish APRI3 project

The recently completed APRI3 project had as the main goals (APRI3, 1999): (1) to elucidate the possibility of cooling and retaining the core melt in the reactor vessel, (2) to further develop methods for probabilistic assessment of severe accident phenomena, and (3) to follow-up and evaluate international severe accident research.

In one of the APRI3 sub-projects, Risk Assessment of Severe Accident Phenomena (RAF), an overview and assessment of severe accident phenomena of importance for PSA level 2 studies were performed. In the first phase of the RAF project, the risk dominant phenomena have been identified. The definition of a risk dominant severe accident phenomena is that it can cause a significant release from the containment with a frequency higher that 10^{-7} per reactor year. In the second phase of the RAF project, a more detailed investigation and assessment of these phenomena have been conducted. The results of the RAF project have shown that the following phenomena dominate the risk at the Swedish BWRs and PWRs and at the Finnish BWRs:

- hydrogen combustion
- steam explosions in BWRs
- melt-through of containment basemate due to non-coolable core debris configuration
- global failure of reactor pressure vessel

In BWRs, the possibility of hydrogen combustion when the containment is de-inerted should be minimised through accident management strategies. APRI3 recommendation was to continue studies of hydrogen risks in BWRs and PWRs in order to reduce remaining uncertainties. With regard to steam explosions, it was concluded that our understanding of this phenomenon has improved significantly and that it should be possible to make the "final" assessment of this issue. It has been proposed to carry out such an assessment. Another recommendation of the APRI3 project was to continue studies of in- and ex-vessel melt coolability. As described earlier in this report, the mode of vessel failure is strongly connected with the late-phase in-vessel melt progression, especially melt coolability in the lower plenum. An important insight has been reached namely that it seems possible to cool the core melt inside the reactor vessel, the fact that could greatly affect the accident management strategies. It was also concluded that the Direct Containment Heating phenomena is not a risk dominant containment threat for the Swedish PWRs. It was also concluded that there are indications that accident management strategies should be updated as a result of new insights into severe accidents progression and phenomena.

The above recommendations have been considered in formulating the objectives of the APRI4 project. Only hydrogen issue is not explicitly included in APRI4 but it is addressed in the ongoing NKS/SOS-2.3 project.

3.3 Uncertainties and recommendations - KTHs perspective

Most of the severe accident research in Sweden, both experimental and analytical, is conducted at the Department of Nuclear Reactor Safety at KTH. Considering our current knowledge about severe accident phenomenology, as described in Section 3.1, as well as severe accident strategies implemented at Nordic reactors, the following uncertainties and recommendations for the future research have been identified by KTH:

In-vessel accident progression

The main uncertainties connected with the in-vessel accident progression are as follows:

- The effects of the melt pool stratification (observed in the RASPLAV experiments)
- The creep behaviour of vessel, timing and modes of its failure with and without penetrations
- The effectiveness of the gap cooling and the external cooling
- The survivability of the lower head in the event of an in-vessel steam explosion

In order to reduce uncertainties in the above phenomena it is recommended to pursue experimental investigations in the FOREVER and the SEMICO test facilities at KTH. Experiments should address melt pool convection, gap cooling and rupture of the lower head as well as the effects of stratification and the metal layer on the thermal loads on the lower head wall during melt pool convection. Analytical studies are proposed to address the lower head dynamic loading due to in-vessel steam explosion and feasibility of melt retention in the reactor vessel.

Ex-vessel accident progression

The main uncertainties here are melt dispersion, jet formation, steam explosions and debris coolability. It is recommended to quantify the dispersion process under various accidental conditions through experiments in the FOREVER facility. With respect to steam explosions, the main uncertainties are connected with the effect of melt properties on melt fragmentation and on steam explosion, the explosivity of the real corium, the efficiency of steam explosions with real corium, the dynamic loadings of a steam explosion on the PWR reactor cavity and on containment as well as on the BWR drywell support wall and containment. In order to reduce the above uncertainties it is recommended to perform experiments on jet and drop fragmentation as a function of melt properties. In addition, it is recommended to evaluate data from the FARO and KROTOS experiments as well as to develop models for effects limiting steam explosion.

Uncertainties connected with debris coolability concern the effect of the multi-dimensionality (including stratification, three-dimensionality and backfits such as downcomers) on the improvement of the coolability of the debris bed, the interactions of the coolant and the melt pool as well as the impact on the long-term coolability. It is recommended to perform experiments on: (1) the coolability of the debris bed with consideration of different configurations with top flooding, (2) the mechanism of the quenchability and coolability of the melt pool by bottom flooding, and (3) the long-term coolability of a melt pool by top flooding with backfits, e.g. downcomers. An evaluation of long-term coolability for the

Swedish and the Finnish BWRs and for the Swedish PWRs for dominant accident scenarios is also recommended.

3.4 Conclusions and open research items - VTTs perspective

The recent severe accident research at VTT has focused on in-vessel melt progression and core coolability issues and on aerosol behaviour. A large part of the performed work was connected to the Nordic NKS/RAK-2 project and different EU-projects (MVI, REVISA and SARA as well as several aerosol projects).

Coolability in the core region

The issue of core reflooding has been studied extensively with numerical methods in the past VAHTI and ROIMA projects as parts of Nordic research programmes. The analyses have addressed overall melt progression and evaluation of different time frames for melt movements in the core region, lower plenum and out of the pressure vessel. Presently it is considered that little could be added to the performed work in this area with the numerical studies. The further development in the area is needing experiments on high temperature reflood heat transfer and on material behaviour (including properties and chemistry) under high temperatures and large thermal gradients rather than further assessment of computer code models. Furthermore, the operability of water injection systems under various reflooding conditions would need to be assessed critically. The behaviour of core lower support structure in core melt accidents needs more analyses. Most of the data is based on bundle-tests or TMI-2 accident. All available numerical models leave the definition of core support plate failure criteria to the user by selection of a critical material temperature for failure. The flow path and material geometry are usually not considered. Particularly, the situation in a BWR core should be further addressed.

Recriticality during reflooding

The work with reflooding was followed by the evaluation of recriticality issues. The work was performed as part of VAHTI- and ROIMA-projects as connected to Nordic collaboration and EU SARA-project efforts. APROS code was applied to recriticality analyses and a new code RECRIT was developed in collaboration with Risø National Laboratory. Recently, RECRIT thermal-hydraulic model has been validated against several reflooding experiments and both validation and code reference manuals have been published.

The investigations showed that the fission power peak related to recriticality is strongly dependent on the coolant injection rate, becoming higher with increasing water injection rate. In general, the energy deposited in the fuel during the first prompt power peak exceeds the limits recently observed for irradiated fuel. In view of thermomechanical loads related to reflooding of hot fuel with cold water, it is likely that some fuel fragmentation takes place, if reflooding occurs during a favourable time period for recriticality. However, the analyses also suggested that the recriticality would be a very local phenomenon, which limits the amount of fuel that may fragment.

After completion and documentation of the RECRIT code, separate studies on the applicability of in-core instrumentation to detect recriticality have been performed. The results of these assessments suggest that it would be possible to detect recriticality in the core with the existing in-core instrumentation during a severe accident. There is another scenario which could lead to recriticality in a partly degraded core. It is a situation, where molten control rod material undergoes a strong fragmentation in the lower head water pool. This in turn causes rapid vaporisation in the pool, which may push a slug of unborated water into the core. The fuel response to the water slug movement is here of interest. This accident scenario is investigated in the current NKS/SOS-2.3 project.

Core debris behaviour in the lower plenum

Debris coolability and failure of lower head has been assessed numerically. Coolability studies have been performed with MELCOR/BH model. PASULA code has been applied to assess the failure of lower head penetrations due to thermal attack by core debris. The results of this investigation show that the lower head in Olkiluoto reactor would fail in about one minute if debris does not fragment in the lower head water pool. The failure is delayed by an hour if significant melt fragmentation and temporary quenching takes place in the lower head water pool. The developed and applied PASULA model for heat transfer in granular debris bed needed validation. The data for validation was obtained from particle bed heat transfer experiments that were performed in the MOSES-project in 1999-2000.

The VTT dry particle bed test rig involved an instrumented and insulated test furnace containing a particle bed made of alumina balls. The vertically aligned particle bed had a diameter of 100 mm and was 200 mm high. The heating of the particles was realised with a spiral resistance heater at the top of the bed. The particle bed was placed in a thin ceramic tube surrounded by 100 mm thick ceramic wool-type of insulator. Temperatures were measured at 9 locations in the particle bed and at 12 locations in the insulator. The spherical, uniform alumina particles used in the bed were about 6.7 mm in diameter. Five successful testing runs were completed. The power was increased stepwise in each test after reaching a steady state in heat transfer with each power level. At the maximum heating power applied (418 W), the maximum achieved temperature before failure of the heating element was 1270 ^{o}C at a distance of about 10 mm from the surface of the heating element.

The dry bed heat transfer experiments were calculated with the PASULA code and specially developed BEDEXP code. The calculation efforts were complex due to the fact that the material property data provided by the manufacturer were not accurate enough for scientific applications. Furthermore, the aluminium oxide particles had some internal porosity that changes the material properties in comparison to the solid Al_2O_3 properties. BEDEXP analyses supported the assessment of correct material properties in the PASULA calculations. The analyses of VTT dry particle bed experiments suggest, that the numerical model developed for particle effective heat transfer coefficient in the PASULA code is capable of predicting heat transfer phenomena in a dry particle bed with good accuracy, once the solid particle material and cover gas properties are known.

A 3D-FEM-model has been developed in PASULA code system for assessments of lower head failure by creep rupture. The model handles small and large deformations taking into account the temperature dependent material properties. FEM-model has been applied to calculation of hemispherical FOREVER- creep tests and RUPTHER tube tests. Furthermore, analyses of creep behaviour of Loviisa NPP lower head under outside cooling conditions have been performed. The current creep rupture studies at VTT Energy focus on post-test analyses of 1/5th scale creep rupture tests performed at Sandia National Laboratory for the

OECD/OHLF-project. Further work on validation of creep rupture models in 3D-geometry is considered important and is included in the current MOSES-project.

One of the major uncertainties that was observed during MAAP and MELCOR calculations for late phase melt progression, was the criteria and modelling of core support plate failure. Both codes have simple, parametric models controlled by user dials for prediction of support plate failure. These models thus control also the pattern and history of the core debris relocations into the lower head. Another major uncertainty in the lower head studies is the downward migration of debris in the instrument tube channel, including the possible refreezing and blockage formation in the channel.

Coolability of debris in the lower head by in-vessel reflooding has not been a high priority issue, since current SAM strategies at Loviisa and Olkiluoto do not take credit of this possibility. For example, gap cooling has not been a specific research topic at VTT. Moreover, the failure mechanism of the BWR vessel and the debris discharge mode are considered more important uncertainties. At Loviisa plant the SAM strategy relies on in-vessel melt retention by ex-vessel cooling.

High pressure melt ejection (HPME) in BWRs

The topic of HPME in the Finnish BWRs was addressed in the ROIMA-project as part of NKS/RAK-2 contribution. The performed numerical analyses suggest that the pedestal and the drywell will experience a pressure spike of up to 0.8 MPa during the first minute of the HPME. Even more damaging to the containment penetrations may be the high gas temperatures in the containment, as the predicted drywell gas temperatures were 800-1000 K. High gas temperatures are caused by release of highly superheated steam and by oxidation of metals in the discharged debris. Particularly MELCOR/BH model predicts that the debris that is discharged through the failed instrument penetrations would have a very high metal content. In these studies the possible fuel-coolant interaction issues were not addressed.

However, the high pressure melt ejection issue is not considered important for Finnish NPPs. For the Olkiluoto plant, the reliable reactor vessel depressurisation system practically out-rules high pressure scenarios. For the Loviisa plant, the pressure vessel integrity will be maintained through outside cooling. Further studies on high pressure scenarios are therefore not included in current MOSES-project.

Ex-vessel coolability

Ex-vessel coolability is the key uncertainty that has remained despite extensive international research. The coolability of debris is particularly important in case of Olkiluoto plant. MACE experiments have not proved that the core melt would be coolable by pouring water on top, though some elements of coolability have been identified in the experiments.

For the Finnish BWRs, the ex-vessel coolability issue is different from the conditions of the MACE experiments. In Olkiluoto NPP the core melt will pass through a deep water pool and may experience significant fragmentation. The amount of fragmentation is still considered an open issue, though meaningful experiments have been carried out on the topic (e.g. FARO). The coolability of particle debris bed has not been addressed sufficiently in the past, in particular the effects of particle size distribution on coolability. Earlier DCC-experiments at Sandia National Laboratories suggest that layering of very small particles (size typical to that found after steam explosions) on top of coarser particles delays or even inhibits water access deeper into the particle bed.

The key goals of the second half of the MOSES-project are the performance of the particle bed dryout heat flux experiments. A literature study was performed on core melt fragmentation to aid in defining a representative particle size distribution for the simulant MOSES2-experiments. The available experimental data on fragmentation suggests that the formed particle bed would have a high porosity (40-60 %) and the particle bed would have an average particle size of 3.5 mm. With a simple assessment of the amount of core debris that may undergo energetic interaction with coolant forming finer particles, it was concluded that 10 cm additional finer particle layer might settle on top of the coarser bulk debris. The first experiment will be performed with a particle bed with a representative particle size distribution, when steam explosions are not taken into account. The second experiment is planned to have an additional 10 cm layer on top of the base bed with finer particles. This will address the effect of an order of magnitude smaller particles, formed in an energetic interaction, on dryout heat flux. A calculation tool will be developed and applied for numerical post-test analyses.

A literature study was also performed on previously performed particle bed dryout heat flux measurements. The existing data base on dryout heat fluxes covers a wide range of particle sizes and both homogeneous and stratified bed configurations. On the basis of simple dryout heat flux calculations, a bed with average particle size of 3.5 mm would be coolable by top flooding.

Hydrogen in BWR reactor building

The Finnish BWR containments are inerted with nitrogen during normal operation and consequently the hydrogen issue has not been considered as a major problem in severe accidents. However, BWR core contains a large amount of Zircaloy and the volume of the containment is small. Containment leakage can not be excluded considering thermal and pressure loads the containment is exposed to during a severe accident. Hydrogen leaking from the containment is discharged into reactor building, where the atmosphere is normal air. A specific question rises namely if external hydrogen combustion could pose a threat to containment integrity.

Hydrogen accumulation and distribution in the Olkiluoto reactor building during a station blackout situation has been simulated with the FLUENT 3-D code. Stratification of hydrogen to the upper parts of the rooms was observed in all cases. The low density gas rises up close to the containment wall and is collected near the ceiling. Mixing with the ambient gas was insignificant. The performed calculations indicate that nominal or small leak from a BWR containment can lead to highly flammable gas mixtures directly outside the containment. The hydrogen concentrations exceeded the traditional flame acceleration and even detonation limits.

The investigations were continued with combustion and flame acceleration studies with the FLUENT code and with rough 1-D calculations to assess pressure loads related to spherically propagating detonation. The flame acceleration studies revealed that supersonic combustion mode could not be excluded. On the other hand, published results from the experimental detonation research suggest, that a direct ignition of a detonation in an unconfined geometry is difficult.

The simple detonation load assessments were carried out in the MOSES-project as part of Nordic nuclear research project NKS/SOS-2.3. The detonable hydrogen mass was assumed to be 1.5-3 kg, and this caused a pressure spike of 13-39 MPa, with corresponding impulses being 2-9 kPa-s. These estimates, however, do not take into account the gradual energy release during propagation of the combustion. Neither do they account for the multiple, 3-D reflections and focusing of shock waves in corners. A more detailed analysis taking the detonation dynamics accurately into account has been performed with 3-D code DET3D developed at Forschungszentrum Karlsruhe. This work is underway in MOSES-project as part of Nordic nuclear research programme.

The hydrogen issue in a BWR reactor building is a new research topic, which has not been sufficiently studied to date.

Aerosol behaviour

The understanding of fission product behaviour in severe accidents is important for source term assessment and accident mitigation measures. For example, in SAM the operator needs to know the effect of different actions on the behaviour and release of fission products.

Fission product behaviour has been studied in different national and international projects. The projects have dealt with fission product vapour/aerosol chemistry in the primary system, aerosol physics in the containment, revaporization of fission product samples from experiments in the PHEBUS facility as well as assessment of a model for revaporization.

VTTs results on aerosol behaviour in containment (experiments in AHMED and VICTORIA test facilities) together with results from other facilities complete our understanding on soluble and non-soluble aerosol behaviour at known thermal-hydraulic conditions. Furthermore, a lot of knowledge has been gained in the area of fission product revaporization. Main uncertainties in this field exist in fission product behaviour in the reactor coolant system. The chemical forms of fission products transported through the reactor coolant system are important for iodine partition between gaseous and aerosol phase and retention and late phase revaporization of the fission products.

Further studies are carried out of revaporization phenomena by active follow-up and participation in the PHEBUS-programme and by investigating organic iodine formation and methods to retain it in the containment water pools or in the filter system.

The behaviour of organic iodine is studied at VTT in connection with the Nordic SOS-2.3 project. The study comprises both experimental and analytical studies and is co-ordinated by VTT Chemical Technology. The contribution of MOSES-project to studies of organic iodine was to gather available information on the methods to prevent a source term of methyl iodide during a severe accident. The most widely studied methods for nuclear power plant applications include the impregnant carbon filters and alkaline additives and sprays. The formation of elemental iodine, that could react further producing organic iodides, is minimal in alkaline solutions.

The behaviour of organic iodine is one of the key uncertainties during a severe accident in the Finnish nuclear power plants and the resolution of the problem requires further work.

3.5 OECD Halden Reactor Project

OECD Halden Reactor Project (HRP) at the Institute for Energy Technology (IFE) in Halden has been conducting research activities directed at the development of Computerised Accident Management Support (CAMS) system, to be used during normal operation and during accidents. Development of CAMS has been in the past supported by NKS. There are also other reactor safety related research activities at Halden which belong to the Fuel and Material Programme and the Man-machine Programme. An example is research into the fuel response to transients, such as reactivity initiated accidents. Halden Reactor Project has recently developed a new man-machine laboratory. The laboratory will primarily provide services to support the conduction of advanced human-factors experiments and also function as a test-bed for Computerised Operator Support Systems, to which CAMS belongs.

As a result of extensive use of computer-based solutions for safety-related and safety-critical functions, the HRP is conducting research to establish methods to improve software quality and software reliability in safety critical applications. This work comprises both improvement of the software development process as well as improved software testing strategies.

The recent development of CAMS is presented in Appendix A.

3.6 Risø National Laboratory

Recent severe accident research activities at Risø National Laboratory have comprised studies of recriticality phenomenon, in the EU-project SARA (Frid et al, 1999), and core melt coolability in the lower head of reactor pressure vessel. An important experience from the TMI-2 accident was that the reactor pressure vessel survived the relocation of nearly 20 tons of highly oxidized, molten core material into the lower plenum. This indicated that there may exist some unexpected, and still unexplained, inherent cooling mechanism. To shed some light on this issue, a lower head coolability code has been under development at Risø. Some details of this model are presented in Appendix B.

4 Severe accident management development

In parallel with the severe accident research, there have been significant improvements and further development of accident management strategies and measures, including emergency operating procedures, plant modifications, new instrumentation and operator aids. Increased knowledge about severe accident phenomena has played an important role in this context. Whereas the major part of the research has been conducted in the framework of national research programs, often in co-operation between nuclear authorities and industry, the development of accident management has been carried out by the utilities. In this chapter, a summary of these developments is presented. As mentioned earlier, accident management strategies are very plant specific. This and the fact that risk significance of several severe accident phenomena is still subject to different interpretations explain the differences in the approaches to the accident management issues.

4.1 Olkiluoto 1 and 2

The provisions for severe accident management were installed in Olkiluoto 1 and 2 during the SAM project which was finished in 1989. The measures implemented were:

- containment overpressure protection
- containment filtered venting
- lower drywell flooding from wetwell
- containment penetration shielding in lower drywell
- containment water filling from external source
- containment instrumentation for severe accident control
- Emergency Operating Procedure for severe accidents

Accident management activities at the Olkiluoto plant comprises both development of accident management procedures and additional plant modifications. They were initiated mainly during the Olkiluoto 1 and 2 modernization project. Some hardware changes have been implemented, others are planned. The necessary analyses are often carried out in close co-operation with VTT and other research institutions.

Emergency Operating Procedures for Severe Accidents

Emergency Operating Procedures for Severe Accidents were modified in order to take into account plant modifications and to enhance severe accident management. The containment filtered venting system rupture disk line from upper drywell will no more be closed in the beginning of the accident. This is a precaution for a possible rapid pressurisation of the containment if the generation of non-condensable gases is large. The previously manual depressurisation of the primary system in severe accidents without LOCA inside the containment has been replaced by automatic actuation of depressurisation system.

Containment filtered venting system - impact of chlorine in the filter

In a severe accident, a large amount of chlorine could possibly be released from the synthetic rubbers, used as insulation material of the electrical cables, due to irradiation and heating. In order to maintain the iodine retention capability, the sodium thiosulfate concentration of the filter was increased. The iodine retention capability and stability of the solution have been experimentally verified by TVO and VTT.

Containment pH

A large amount of chlorine, which can be converted to HCl in the containment, will reduce the pH of the water pools and wet surfaces. This may lead to significant amount of elemental as well as of organic iodine. Another source of organic iodine may be reactions between boron carbide in control rods, steam and iodine in the degrading core.

TVO has investigated the possibilities to enhance the retention of iodine by containment pH control system. The solution used would be 50 % NaOH, which is already normally used by the water treatment plant. A new NaOH tank will be installed. The required NaOH volume was analysed by VTT Chemical Technology with a ChemSheet model. The required volume is according to the calculations about 5 m^3. The solution is gravity driven into raw water storage tank near fire water outlet nozzles, from where the solution is delivered into the containment during containment water filling.

Lower drywell will be flooded from wetwell prior to the NaOH supply and the lower drywell water pool pH will be kept above 7.

Organic iodide

VTT Chemical Technology investigates possibilities to improve retention of organic iodide. The purpose is to find means to improve the existing filter so that they are capable of trapping the organic iodine compounds and of preventing iodine to form organic compounds. Possible means are the oxidation of elemental iodine by modifying the chemical composition of the filter or by using catalytic oxidation.

Energetic ex-vessel fuel coolant interactions

TVO has investigated the response of concrete structures in the containment to energetic fuel coolant interactions, i.e. steam explosions and concluded that they would withstand large steam explosion loads. The further studies deal with impact of possible steam explosions on the pipe penetrations and personnel access hatch in the lower drywell. The key issue is maintaining the containment leaktightness in severe accidents. TVO has decided to strengthen the lower drywell personnel access lock.

Containment gas sampling system

TVOs experts have come to the conclusion that modifying the gas sampling system for severe accident conditions would not significantly improve the possibilities to determine the conditions in the containment gas space. The amount of non-condensable gases can be calculated from the pressure and temperature measurements with the Containment Monitoring System. The amount of noble gases and iodine can be assessed using the Room Radiation Monitoring System detectors in the primary containment. The uncertainties in determining the amount of iodine in gas form, elemental iodine and organic iodide, in the containment would be large even with a sampling system. However, the discussions with the Radiation and Nuclear Safety Authority regarding requirements for modifications of the gas sampling system for severe accident conditions are not yet finished. The modification requirements concern the ability to measure iodine and noble gases in the containment as well as the concentrations of hydrogen and oxygen.

Diaphragm floor seal

TVO investigates how the diaphragm floor seal would behave in severe accidents. The leaktightness of the seal is important in order to maintain the pressure suppression function of the containment as long as possible.

Reliability of isolation valves

An additional second isolation valve was installed in 1998 on the nitrogen system piping lines from lower drywell to reactor building in order to ensure isolation function in severe accidents. The piping part inside the lower drywell may be damaged because of contact with core debris.

Hydrogen combustion phenomena in reactor building

During the severe accident, hydrogen gas leaking from the containment might lead to combustible hydrogen concentration in the reactor building compartments. TVO is investigating possible hydrogen combustion loads in the reactor building including hydrogen burns and detonations (see also section 3.4).

The concern is that containment penetrations might be damaged due to hydrogen combustion phenomena outside containment which could lead to a large leak.

Primary system depressurization in severe accidents

To secure depressurisation of the reactor primary system in severe accident situations and to prevent a new pressurization of the reactor, two valves of the relief system have been modified. It is now possible to keep the valves open with the help of a nitrogen supply or a water supply from outside of the containment. The modification was finished in 1999.

Recriticality

The SIRM detectors will be drawn in the beginning of the accident half a meter below the active core to detect possible recriticality. Analyses were performed in 1999 to determine how to relate the reading of the SIRM monitors to the actual reactor power.

4.2 Loviisa NPP

The Loviisa nuclear power plant is a two-unit VVER-440 with ice condenser containments owned and operated by Fortum Power and Heat Oy.

The overall severe accident management approach was structured around the idea of demonstrating in-vessel melt retention, hydrogen risk management and reliable long-term containment cooling. The effective management of severe accidents at Loviisa also requires depressurisation of the primary circuit to ensure low pressure during the long-term contact of the molten corium pool with the reactor vessel. Another important part of the SAM strategy is to reliably prevent containment bypass sequences, other sequences with an impaired containment function, and reactivity initiated core damage sequences.

New motor-operated relief valves with sufficient capacity have been installed on the pressurisers in 1996. Depressurisation is to be initiated from the first indications of superheated temperatures at core exit thermocouples.

Glow plug igniters were installed into the Loviisa containments in the early 80's. An extensive research program was conducted in IVO (now Fortum) during the first half of the 90's to study the reliability and adequacy of the igniter system. The work included the experimental program with the VICTORIA facility and associated development of calculational models. A new hydrogen management strategy was approved by the Finnish safety authority in 1997. The new scheme consists of three main components: ensuring efficient mixing of the containment atmosphere through forcing open ice condenser doors as a severe accident management measure, removal of hydrogen through passive catalytic recombination, and removal of hydrogen through deliberate ignition in the steam generator compartment.

In-vessel retention (IVR) of molten core constitutes the cornerstone of the Loviisa accident management approach. The Loviisa containment cavity will be passively flooded in the vast majority of severe accident sequences. The lower head insulation and neutron shield blocks should be lowered during the accident as a severe accident management measure. The approach to IVR was approved by STUK in 1995, and the plant modifications at unit 1 were carried out in the year 2000. The IVR strategy was supported by the Loviisa-specific COPO experiments in Helsinki and ULPU experiments at the University of California, Santa Barbara.

In the absence of corium-concrete interactions, there is no production of non-condensible gases, except hydrogen, which is to be removed as a part of the hydrogen management scheme. Thus, stabilization of containment pressurization can be achieved by steam condensation on the containment walls. External spray systems were constructed on the Loviisa containments in the years 1990-91. The design calculations were verified with almost full-scale and real condition experiments in the German HDR-containment.

New instrumentation qualified for severe accident conditions is also a part of the overall severe accident management strategy. New operator procedures (SAM guidelines and handbook) are also being developed concurrently with the implementation of the hardware changes.

Fortum has also been developing its capabilities in analysis of aerosol transport. Aerosol experiments have been carried out in the VICTORIA facility. At the moment, experiments for studies of aerosol behaviour in horizontal steam generators are being carried out in the HORIZON facility. Some of the aerosol experiments have been performed within EU Fourth and Fifth Framework shared cost action research projects.

4.3 Swedish BWRs

Activities related to severe accident mitigation have been primarily related to improvements and developments of emergency operating strategies and procedures, simple operator aids for emergency situations, such as core damage assessment methods, as well as development of methods for assessment of plant status and source term in the acute phase of an accident.

At Forsmark nuclear power plant, the Emergency Operating Procedures (EOPs) for shift supervisor have been improved by making them less detailed and more as an overview procedure (Löwenhielm et al, 1997). This has been achieved by a reduction of operative measures and number of controls. At a later stage of the accident when plant functions are severely degraded these control room EOPs are no longer the main support for the operators. At this stage the accident management is primarily knowledge based and the measures taken are guided by the knowledge of the emergency team in the Emergency Control Centre (ECC).

The main tools at this stage are the Technical Support Document (TSD) at the reactor unit and the Handbook for Severe Accidents Management (THAL) at Emergency Control Centre. TSD is under development. The main purpose of THAL is to be a support to the technical staff in the ECC in case of a severe accident. In THAL, the most important results and conclusions from severe accident analyses performed for the plants have been summarised with reference to the actual documents. It also contains strategies for handling various accident situations and points out important factors relevant to accident management. An important aspect, which is reflected in the THAL, is that certain short-term actions are of importance for accident management in the long-term perspective. The long-term aspect of accident management were studied in a separate project with the objective to increase the knowledge about a severe accident in a long-term perspective, up to five years after the accident (Gustavsson, 1991).

It is important to update the handbooks on a regular basis to implement new knowledge. Changes in severe accident management procedures could also be reasons for upgrading these documents.

Barsebäck Kraft and the Swedish Nuclear Power Inspectorate have jointly developed a manual procedure for assessment of plant status and potential source term in the acute phase of an accident to form the basis for decisions of protective actions (Lindvall et al, 1999). An important advantage of the method is that it is visible, possible to backtrack and reconsider as well as easy to communicate. The assessment cycle is 30 minutes or less. The use of pre-calculated scenarios based on PSA level 2 studies gives a probabilistic aspect and an expectation on sequence of events. Assessment of core damage, releases from the fuel, status of containment penetration as well as mechanisms for fission product retention in the containment and reactor building is performed.

4.4 Swedish PWRs

New Severe Accident Managenent Guidelines (SAMG) are being developed and implemented at Ringhals PWRs through adaptation of Westinghouse generic SAMG package (Henoch, 2000). SAMG is bases on so called Reference Decision Making Process. These new guidelines will replace the BERG (Beyond Emergency Response Guidelines) instructions. SAMG will provide structured guidelines for:

- diagnostic of plant status
- prioritisation of accident management measures
- evaluation of alternative measures
- verification of conducted measures

There are many differences between SAMG and BERG. One is that SAMG is symptom based while BERG is mainly event based. BERG is used in parallel with ERG (Emergency Response Guidelines) until melt-through of the reactor vessel has been diagnosed. In contrast, there is a clear transition from ERG to SAMG. The objective of ERG is to prevent core damage while the objective of SAMG is to mitigate releases and protect the barriers. The flexible structure of SAMG makes it possible to consider plant specific features and the results of level 2 PSA. Other important features of SAMG are that the availability of instruments is considered, that there is no need for diagnosis of vessel melt-through and that uncertainties in the physical processes are considered. The SAMG will also provide support to the control room during the time period before Technical Support Centre is operative.

A practical method for assessing the extent of core damage have been developed by Vattenfall and adapted for Ringhals NPP.

4.5 Halden reactor

The HBWR is a boiling heavy water research reactor with natural circulation and a maximum heat removal capacity of 25 MW. The operating temperature is 240 °C, corresponding to a pressure of 3.36 MPa. The core consists of 500 – 600 kg UO_2 in about 110 fuel assemblies, out of which 30 – 40 are instrumented test fuel assemblies.

The reactor is situated within a rock cavern and the containment consists of the surrounding rock body and an air lock in the entrance tunnel. The rock covering the main reactor hall is 50 m thick, while at the air lock the thickness is 30 m. The air lock doors are designed to withstand an overpressure of 0.3 MPa. Ground water, which penetrates into the reactor hall at an average rate of 25 m^3/day, is collected at various places and pumped to delay tanks outside the containment, where it is monitored and disposed off to sewage.

The reference severe accident scenario at HBWR assumes a sudden and complete rupture of the reactor vessel bottom pipe, with a resulting drainage of the reactor tank in 45 seconds. Without emergency core cooling it is estimated that half of the fuel will melt and collect at the bottom of the vessel, together with already molten silver-cadmium alloy from the control rods, and melted stainless steel from tubes and control stations. Calculations have shown that melt-through of the reactor vessel will not occur, mainly due to the large steel body in the bottom of the vessel.

Activities related to severe accident management at HBWR have been connected to the emergency operating procedures, with several exercises, and to the accident safety systems of the reactor.

Recently the radiological condition in the containment, after the accident situation has been brought to a final stable state, has been studied. It has been shown that with necessary shielding of the filters of the emergency purification system, re-entrance of the reactor hall is possible shortly after a stable state has been reached. The shielding of the filters where accomplished in the outage in May / June 2000.

The probability that the emergency core cooling system will function properly in case of a loss of coolant accident is high. If the system functions as intended, the core components will remain in place after an accident since melting will not occur. Currently the design of the system is studied with the aim of increasing its reliability further.

Calculations show that recriticality of the collapsed core is not possible without moderator. There is, however, a possibility of recriticality under certain circumstances when the core melt is mixed with water. The same applies to the remaining not collapsed fuel. In prevailing accident mitigation procedures recriticality is avoided by avoiding re-entry of water into the reactor tank. This has to rely on transfer of water out of the containment with one of four high reliable pumps, since ground water penetrates into the containment. At present the possibility of boron injection into the water is being studied, which would prevent recriticality even if the reactor vessel is re-flooded as a result of pump operation failure.

5 Conclusions and recommendations

In this report we have presented the status of severe accident research and accident management activities in Nordic countries. The emphasis has been on severe accident phenomena and issues of special importance for the severe accident management strategies implemented in Sweden and in Finland. Remaining uncertainties in important severe accident phenomena have been identified and recommendations for future research have been given. It should be noted here, that in addition to uncertainties connected with particular severe accident phenomena, there is an uncertainty connected with the accident scenario itself, which has not been addressed explicitly in the report. Thus, in many cases the uncertainty in predicting containment performance is strongly related to the initial conditions resulting from uncertainties in the in-vessel accident progression. This in-vessel accident progression uncertainty is larger for BWRs than for PWRs. As described in the report, significant progress has been made in our understanding of the in-vessel accident progression, including the phenomena of melt-structure interactions in the lower head of the reactor pressure vessel.

It can be concluded, that the Nordic severe accident research is comprehensive and in practice addressing all important severe accident issues. Co-operation between Nordic countries is playing an important role in this context. Both the in-vessel and the ex-vessel accident progression phenomena and issues are investigated. Even though there are differences between Sweden and Finland as to the scope and content of the research programs, the focus of the research in both countries is on in-vessel coolability, integrity of reactor vessel lower head and core melt behaviour in the containment, in particular the issues of core debris coolability and steam explosions. Other important issues are thermal-hydraulic phenomena during reflooding of an overheated partially degraded core, including recriticality issue, fission product chemistry, in particular formation of organic iodine, and hydrogen transport and combustion phenomena.

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 reactors. Recent efforts in Sweden in this area have been mainly concentrated on improvements and further development of accident management procedures. In Finland, in addition to further development of accident management procedures some plant modifications have recently been carried out. As examples of these activities one can mention the development of guidelines and handbooks for severe accident management and emergency situations, the development of Computerised Accident Management Support (CAMS), the implementation of pH control in the containments of Olkiluoto NPP, the decision by TVO to protect the leaktightness of the containment in case of ex-vessel steam explosions by strengthening the lower drywell personnel access lock, and the implementation of in-vessel retention by ex-vessel cooling at Loviisa NPP.

Adequate instrumentation is vital for successful accident management. We notice that development efforts in this area have been modest but there are differences in this respect between the plants.

Notwithstanding that our understanding of accident progression and phenomena, both in-vessel and ex-vessel, has improved significantly and, thereby, our ability to assess containment treats, to quantify uncertainties, and to interpret the results of experiments and computer code calculations, there still exist important phenomenological uncertainties. This motivates 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.

- Coolability of the core material in the lower plenum as well as timing and mode of reactor vessel failure.

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

- Hydrogen distribution and combustion in 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 the containment. Important issues here are organic iodine formation, pH control and fission product revaporization.

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.

References

Alsmeyer, H., Spencer, B., Tromm, W., 1998, The COMET-concept for cooling of ex-vessel corium melts. Proc. of ICONE-6, May 10-15, San Diego, USA .

Andersson, S-O., Gustavsson, V., Hallman, A., Hellström, P., Jung, G., Strümpel, H., Wendt , U-K., 1996, Ringhals 1- PSA NIVÅ 2. Vattenfall Rapport GE 34/96.

APRI3, 1999, APRI3 - Accident Phenomena of Risk Importance. SKI Rapport 99:34.

Asmolov, V.V., 1998, Latest findings of RASPLAV Project. Proc. OECD/CSNI Workshop on In-Vessel Core Debris Retention and Coolability.

Berthoud, G., Brayer, C., 1997, First vapor explosion calculations performed with the MC3D code. Proc. CSNI Specialists Meeting on FCIs, Tokai, Japan.

Carboneau, M.L., Berta, V.T., Modro, M.S., 1989, Experiment, analysis and summary report for OECD LOFT Project Fission Product Experiment LP-FP-2. OECD LOFT-T-3806.

Ciccarelli, G. et al., 1993, High temperature hydrogen-air steam detonation experiments in the BNL small scale development apparatus. Water Reactor Safety Information Meeting, Washington D.C..

Cole, R.K. et al., 1984, CORCON-Mod2: A computer program for analysis of molten core-concrete interactions. NUREG/CR-3920, SAND84-1246.

Dorofeev, S. et al., 1999, Flame acceleration limits for nuclear safety applications. CSARP Meeting, Albuquerque, New Mexico, USA.

FISA 99, 1999, Proceedings of FISA 99 - EU Research in Reactor Safety. Luxembourg, 29 November - 1 December 1999.

Frid, W., Höjerup, F., Lindholm, I., Miettinen, J., Nilsson, N., Puska, E. K., Sjövall, H., 1999, Severe Accident Recriticality Analyses. SKI Report No 99:32, EC INV-SARA(99)-D016.

Froment, K., Seiler, J.M., 1999, On the importance of a strong coupling between physicochemistry and thermalhydraulics for modelling late phases of severe accidents in LWRs. Proceedings of NURETH-9.

Gustavsson, V., 1991, FRIPP: a project concerning long-term effects of a severe accident. OECD Specialist meeting on severe accident management programme development, Rome, Italy.

Gustavsson, V., Hallman, A., Jung, G., Henriksson, M., Hellström, P., 1994, Ringhals 2 nivå 2- säkerhetsstudie. Vattenfall Rapport PT-45/94.

Hagen, S., Hofmann, P., Noack, V., Schanz, G., Schumacher, G., Sepold, L., 1997, The CORA-program: Out-of-pile experiments on severe fuel damage. Proceedings of the Fifth International Topical Meeting on Nuclear Thermal-Hydraulics, Operations and Safety, Beijing, China.

Henoch, A., 2000, Haverihantering på kärnkraftverk: SAMG i Ringhals (in Swedish). Beredskap & Svåra haverier seminarium, Tammsvik, Sweden.

Huhtiniemi, I, Magallon, D., 1999, Insight into steam explosions with corium melts in KROTOS. Proceedings of NURETH-9.

Högberg, L., 1988, The Swedish program on severe accident management and release mitigation. Int. Symp. On Severe Accidents in Nuclear Power Plants, Sorrento, Italy.

Höjerup, F., Miettinen, J., Nilsson, L., Puska, E. K., Sjövall, H., Anttila, M., Lindholm, I., 1997, On recriticality during reflooding of a degraded Boiling Water Reactor core. Nordic Reactor Safety Research, NKS/RAK-2(97)TR-A3.

Kolev, N.I., 1999, Verification of IVA5 computer code for melt-water interaction analysis. Proceedings of NURETH-9.

Ktorza, C., et al., 1999, An overview of the PHEBUS FPT1 results concerning the fission product release, transport and containment behavior. CSARP Meeting, Albuquerque, New Mexico, USA.

Kymalainen, O., et al., 1997, In-vessel retention of corium at the Loviisa Plant. Nuclear Engineering and Design, Vol.169, pp.109-130.

Lindholm, I., Nilsson, L., Pekkarinen, E., Sjövall, H., 1995, Coolability of degraded core under reflooding conditions in Nordic Boiling Water Reactors. Nordic Reactor Safety Research, NKS/RAK-2(95)TR-A1.

Lindvall, C-G., Calmtorp, Ch., 1999, Emergency Preparedness Improvements at Barseback NPP. ANS 7th Topical Meeting on Emergency Preparedness and Response, 13-17 September, Santa Fe, New Mexico, USA.

Livolant, M., Schwarz, M., von der Hardt, P., 1996, The PHEBUS FP Program. Proceedings of the FISA-95 Meeting "EU Research on Severe Accidents. EUR 16896 EN, pp.27-47.

Löwenhielm, G., Jansson, B., Gustavsson, V., 1997, New ideas about procedures and handbooks for severe accident management at the Forsmark Nuclear Power Plants. The fifth International Topical Meeting on Nuclear Thermal Hydraulics, Operations and Safety (NUTHOS-5) China.

Magallon, D., et al., 1997, The FARO programme recent results and synthesis. CSARP Meeting, Bethesda, Maryland, USA.

Magallon, D., et al., 1999, Corium melt quenching tests at low pressure and subcooled water in FARO. Proceedings of NURETH-9.

McDonald, P.E., Buescher, B.J., Hobbins, R.R., McCardell, R.K., Gruen, G.E., 1983, PBF severe fuel damage program: Results and comparison to analysis. Proceedings of the International Meeting on Light Water Reactor Accident Evaluation, Cambridge, Massachusetts, paper 1.7.

Reimann, M., et al., 1990, The WECHSEL-Mod2 Code: A computer program for the interaction of a core melt with concrete including the long term behaviour. Model description and User's Manual. KfK Report KfK-4477.

Sehgal, B.R. et al., 1992, MACE project overview. Proceedings of the OECD Meeting on Core Debris Concrete Interaction, Karlsruhe, Germany.

Sehgal, B.R. et al., 1997a, Core Melt Pressure Vessel Interactions During a Light Water Reactor Severe Accident (MVI Project). Proceeding of FISA-97 Meeting of EU Research on Severe Accidents, Luxembourg.

Sehgal, B.R., et al., 1997b, Experiments and analyses of melt jet impingement during severe accidents. Proc. NUTHOS-5, Beijin, China.

Sehgal, B.R., Bui, V.A., Dinh, T.N. and Nourgaliev, R.R., 1998a, Heat transfer processes in reactor vessel lower plenum during late phase of in-vessel core melt progression. J. Advances in Nuclear Science and Technology, Plenum Publ. Corp, Vol.26.

Sehgal, B.R., Nourgaliev, R.R., Dinh, T.N., and Karbojian, A.K., 1998b, Integral experiments on in-vessel coolability and vessel creep: Results and analysis of the FOREVER-C1 test. Proceedings of the Workshop on "Severe Accident Research in Japan, “SARJ-98", Japan.

Sehgal, B.R., Nourgaliev, R.R., Dinh, T.N., 1999a, Characterization of heat transfer processes in a melt pool convection and vessel-creep experiment. Proceedings of NURETH-9, San Francisco.

Sehgal, B.R. et al., 1999b, Core Melt Pressure Vessel Interactions During a Light Water Reactor Severe Accident (MVI Project). Proceeding of FISA-99 Meeting of EU Research on Severe Accidents.

Sepold, L., et al., 1999, Reflooding experiments with LWR-type fuel rod simulators in the QUENCH facility. Proceedings of NURETH-9, San Francisco, USA.

SERG2, 1995, A reassessment of the potential for an Alpha-Mode containment failure and a review of the current understanding of broader fuel-coolant interaction (FCI) issues. NUREG-1529.

Takumi, K. et al., 1993, Results of recent NUPEC hydrogen related tests. Water Reactor Safety Information Meeting, Washington D.C., USA.

Theofanous, T. G. et al., 1987, An assessment of steam-explosion-induced containment failure. Parts I-IV. Nuclear Science and Engineering, 97, 259-326.

Theofanous, T.G. et al., 1995, In-vessel coolability and retention of a core melt. DOE/ID-10460.

Theofanous, T.G., Yuen, W.W., Freeman, K., and Chen, X., 1996a, Propagation of steam explosions: ESPROSE.m verification studies. DOE/ID-10503.

Theofanous, T.G., Yuen, W.W. and Angilini, S., 1996b, Premixing of steam explosions: PM-ALPHA verification studies. DOE/ID-10504.

USNRC, 1975, Reactor safety study, An assessment of accident risks in U.S. commercial nuclear power plants. USAEC Report WASH-1400.

USNRC, 1987, Reactor risk reference document, USNRC Report NUREG-1150.

Vuorinen, A., Reiman, L., Haule, K., Okkonen, T., 1993, Policy and implementation of severe accident management in Finland. Symposium on Nuclear Safety, Tokyo, Japan.

Wolf, L., et al., 1993, Hydrogen mixing experiments in the HDR containment under severe accident conditions. Water Reactor Safety Information Meeting, Washington D.C., USA.

Appendix A: Computerised Accident Management Support

A.1 Purpose and Scope

CAMS (Computerised Accident Management Support) is a system developed to provide support in accident states and normal plant states. This support is offered in identification of the plant state, in assessment of the future development of the accident, and in planning accident mitigation strategies.

The MAAP code has been shown to be a powerful tool for accident analysis as it is demonstrated by its world-wide use. The main purpose of the present work is the integration of the MAAP code, in its most advanced version MAAP4, into the CAMS system for severe accident analysis.

The methodology that has been developed for the integration of the MAAP code into the CAMS system is described in this document. As a result, two new modules have been developed, for both the BWR and PWR case, for integration in the CAMS system. A parameter processing structure has been developed to implement those theoretical models. The parameter processing structure makes use of fuzzy logic mechanisms.

The methodology developed, the theoretical models and the parameter processing structure are described below.

A.2 Methodology

CAMS is a tool developed to give support in plant state identification, accident evolution and selection of mitigation strategies. It is comprised of three main modules: tracking simulator, predictive simulator, and state identification modules.

The tracking simulator module gives an estimation of the values that are not directly measured, calculates the initial values that are needed for the predictive simulator, and gives support in the validation of the signals by calculating values of certain parameters. The predictive simulator module predicts the evolution of the state of the plant, being faster than the real process. Finally, the plant state identification module gives information about the state of the plant, the state of the systems (their availability), and the state of the critical functions (heat sinks, core cooling, reactivity control, and containment integrity).

The MAAP4 code is able to carry out predictive and tracking functions, since the code is built with some degrees of freedom to adjust the phenomena to the diagnostic conditions.

To be able to use the MAAP4 code in the CAMS system it is necessary to modify the general structure of the system in order to include two new modules, *the Diagnosis Module* and the *Fitting Module,* instead of the original *Tracking Simulator* and *State Identification* modules of CAMS. The new structure is shown in Figure 1.

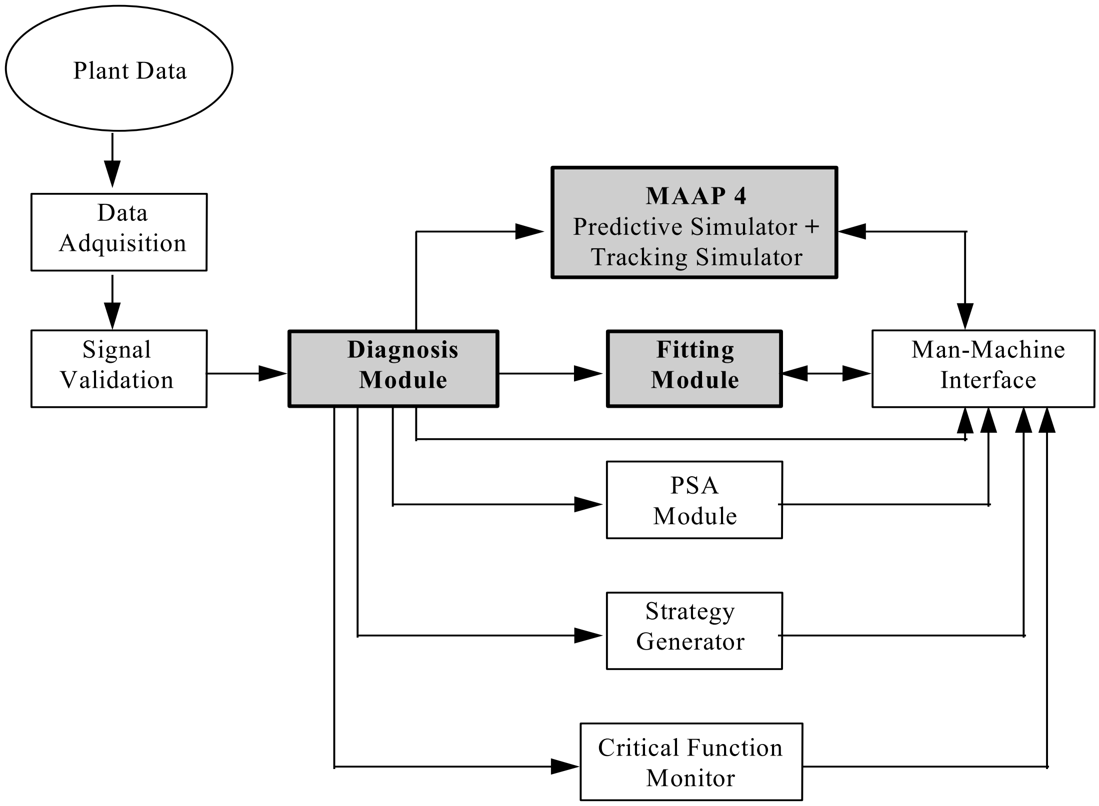

Figure 1. New structure for the use of MAAP4 in CAMS.

The **Diagnosis Module** receives the plant data from the process computer and processes them allowing the user to identify the instantaneous status and conditions of the plant during the accident. In the future it is possible that a signal validation module developed for severe accident conditions will be integrated in the CAMS system. Therefore this module has equivalent functions to those of the State Identification and the Tracking Simulator modules of CAMS, although without on-line fitting capacity.

The **Fitting Module** compares the plant state obtained from MAAP4 calculation to the new plant state processed through the Diagnosis Module and a set of additional rules, allowing the adjustment of the simulated scenario to the observed plant state from diagnosis module in a semiautomatic way.

One purpose of the Tracking Simulator is on-line feeding of the plant data (measured and calculated) to the Predictive Simulator Module, as input for predictive calculations. Since on-line initialisation is complex using the MAAP4 code, a new case with the new boundary conditions, determined from the new diagnosis, is run from the initial time. Thus, at least in this first approach the initialisation is done in a semiautomatic way. This is due to the complexity of the phenomena and to the way they are modelled in MAAP, which makes automatic adjustments difficult to realise. It is also due to the difficulty in obtaining some variables needed to initialise the calculations even by indirect methods.

To be able to make a diagnosis of the initiating event of the accident sequence, it could be necessary to dispose of historical data in order to give information to the Diagnosis Module about the transient start.

A.3 Applicability of computerised procedures to SAMG

The objective of this study was to verify the feasibility of the application of a structured approach in procedure automation to a sample of SAMG procedures, with the final goal to assess the possibility to automate the SAMG procedure management in the framework of the CAMS project.

This study has been carried out using, as input, 2 examples of SAMGs used at Cofrentes NPP, furnished by Iberdrola, and, as a computerised tool, DIAM, developed by Ansaldo DNU.
In this feasibility study the possibility to utilise structured approaches to represent and use SAMGs has been verified. The two guidelines, object of the study, have been formally restructured and reformatted.

The complete computerised SAMGs are now available on a CD-ROM.

Appendix B: Lower Head Coolability Model for TMI-like Accident

An important experience from the TMI-2 accident was that the reactor pressure vessel survived the relocation of nearly 20 tons of highly oxidised, molten core material into the lower plenum. This indicated the existence of some unexpected inherent cooling mechanism, which still has not been adequately explained twenty years after the accident.

Current accident management strategies consider external cooling of the reactor vessel but for many existing reactors this is not a viable option. Therefore further research to develop a dependable in-vessel coolability model for TMI-like accidents is motivated by the need to predict the thermal and mechanical loads of the vessel under varying conditions.

A lower head coolability code is under development at Risø National Laboratory. This code is based on the hypothesis that the crust formed by cooling of the melt at the top and at the melt contact with the wall is subject to thermal cracking. Water penetrating into the fractures against the escaping steam enhances the cracking thus allowing further penetration and cooling both at the top and along the wall. Before further discussion of this model we review some important findings from the TMI-2 Vessel Investigation Project [Ref. 1], which are useful for assessment of the model.

B.1 Review of TMI-2 Lower Head Phenomena

The relocation of molten (~ 3100K) debris from the core region occurred under water at a pressure of about 10 MPa. The relocation path was via the upper and lower core support assembly and the duration was about two minutes. There was no significant jet fragmentation due to the small fall heights.

The debris temperature at the arrival on the lower head is uncertain. The surprisingly undamaged state of some instrument nozzles buried in debris led to the proposal of a highly speculative “relocation scenario”, which is disregarded here, however. In the test examples below, the initial debris temperature in the lower head is simply assumed to be 2800 K and uniform. The debris condition is like wet sand, i.e. solid oxide grains with molten phases in the grain boundaries. The decay heat rate was 130 W/kg at the relocation time.

Metallographic examinations of lower head wall samples showed that somewhat displaced from the centre, an approximately elliptical area (~ 1m by 0.8m) know as the “hot spot” reached an inside wall temperature of about 1100°C, and the nozzles here were melted off almost flush with the bottom. After some 30 minutes the hot spot experienced a rapid cooldown, probably by quenching with water. Outside the hot spot, the wall showed no sign of exceeding the ferrite-to-austenite transition temperature (727°C). Internal debris temperatures remained high for several days.

The only model pretending to explain the TMI lower head coolability is the MAAP4 gap formation model developed by Henry and Dube [Ref. 2]. However, the postulated gap formation due to vessel creep expansion is rather hypothetical and the model is inconsistent as discussed previously [Ref. 3].

B.2 Top Crust Slab Model

During defueling of the lower plenum the upper part of the core material was found in the form of loose debris, while the lower part was the so-called "hard debris", a monolithic layer with cracks and fissures in its surface. In the model, the loose debris formation starts at the water cooled surface by crust formation and subsequent cracking due to differential thermal contraction. After a short period with surface boiling the water starts penetrating into the fractures in counter-current with the escaping steam. The wetting front is propagated downwards at a rate that is controlled by the heat flux from the dry debris and the flow limitation in the wet zone. The distributed heat generation in the wet zone also contributes to the total flux. The bulk permeability is treated as a constant based on a simplifying fracture model, assuming that the cracking front moves ahead of the water front and that the wetted debris immediately cools down to the water temperature at the water front.

The cracked debris permeability is estimated on the basis of Poiseuille's law in combination with simple fracture mechanics using incomplete data. The analysis seems to show that the fracture front reaches into the ductile zone at about 1200 – 1500 K. Setting the average matrix block size to 3 cm, typical of the rubble, the estimated absolute permeability becomes $\kappa \sim 0.15 \cdot 10^{-9}$ m^2. The relative permeabilities for water and steam are taken as third order power functions of the saturation, and the capillary pressure is considered negligible.

The Darcy equations together with the mass and energy conservation equations for the wet zone coupled with the heat conduction equation for the dry zone with moving boundary are solved numerically. The heat conduction equation is coded in general non-linear form but so far the code has only been tested with constant coefficients (k = 3W/mK, c_p = 445 kJ/kg K, ρ = 7800 kg/m^3).

The test case presented in Fig. 1 shows the propagation of the temperature profile from the surface into the debris. The unrealistically high temperature rise is due to the constant coefficients used so far. A zero-flux boundary condition is imposed at the depth 0.30 m, which is approximately halfway to the bottom. In this case the temperature profile approaches a steady state with the wetting front arrested at about 0.12 m. In cases with higher permeability, e.g. $0.4 \cdot 10^{-9}$ m^2, the water penetration continues to the bottom, if the zero-flux boundary condition is removed (i.e. semi-infinite medium). Thus the TMI observations confirm the former value as more realistic.

B.3 Lower crust spherical model

In the spherical model, the lower crust formation and cracking starts by cooling against the wall in the dry state. The contact resistance is represented by a heat transfer coefficient (here 10 kW/m^2 K). Water starts penetrating at the periphery (polar angle 42° from the bottom) and gradually expands the fractured zone, while the wetting front progresses downwards. In this model the flow becomes segregated with water flowing down along the wall, while the steam counter-flow occurs in the hotter parts of the fractures. Consequently, the relative permeabilities are linear functions of the saturation (i.e. water fraction averaged over the flow cross section). The Darcy equations and the mass and energy balance equations are solved in 1-D angular formulation. Obviously, the steam becomes superheated, but so far, only data for saturated steam have been used in the test calculations.

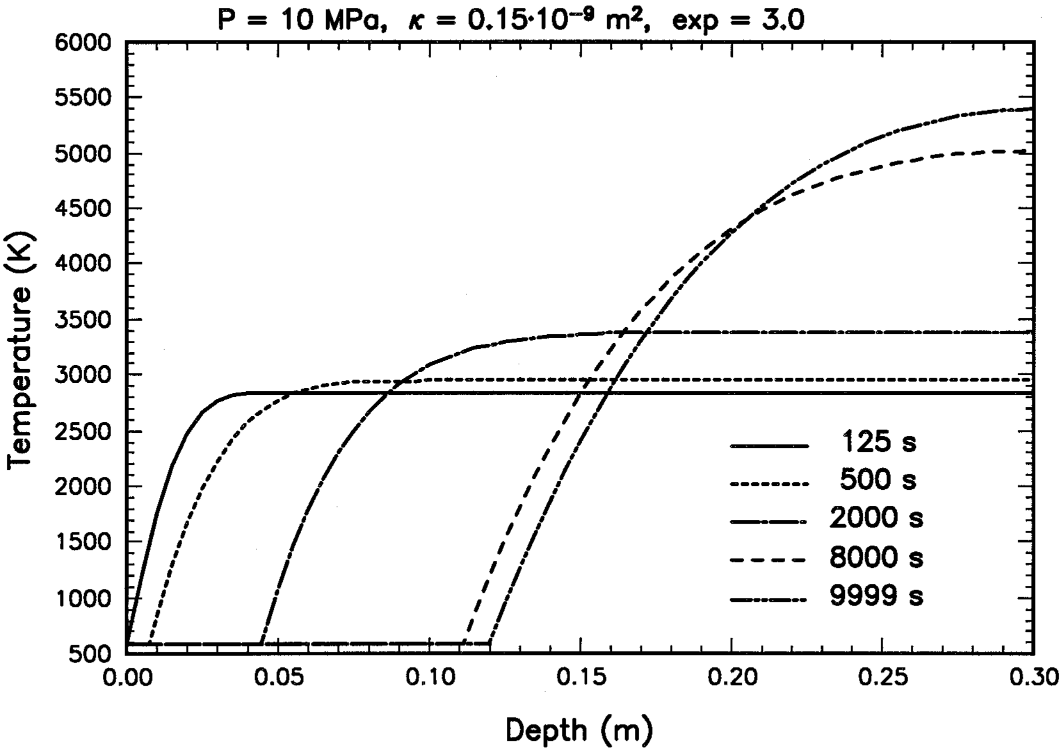

Figure 1. Temperature profiles during water percolation into the cracking top crust.

The heat conduction model is a quasi two-dimensional approximation, neglecting the conduction in the angular direction. Thus, in fact, an individual 1-D radial solution is obtained for each discrete value of the polar angle. Before wetting, the wall is heated up by the debris, and all the temperature profiles are identical. After wetting, when both the wall and debris are cooled, the individual temperature profiles are different depending on the wetting times. The moving boundary condition for the debris is taken as constant temperature equal to that of the water at the steam/water interface. This condition may be superseded by a limitation of the total heat flux from wall and debris, thus avoiding a singularity at the wetting front. The present maximum heat flux is set to 800 kW/m^2.

With the constant coefficients used so far, the temperature profiles do not propagate very deep into the debris, where the heat flux tends to zero by itself. The problem at the upper edge, where the debris thickness in the radial direction decreases to zero, may be neglected without serious consequences for the results. At a later stage, using larger conductivity above the liquidus temperature, it is planned to introduce a zero-flux boundary condition on an internal spherical surface dividing the debris in two approximately equal parts.

The permeability required to obtain a sufficiently efficient penetration rate is pretty large and in fact passing into the transition regime to turbulent flow. The latter is ignored so far, however. The test case presented in Figs. 2-5 is calculated with a tenfold higher permeability than in the slab case. In the simple model the bulk permeability is proportional to the square of the matrix block size, thus implying that the fracture spacing would have to be about 10 cm. This requires further analysis, considering that, unlike the upper crust, the lower crust cannot contract freely but is constrained by wall friction and "anchoring" by the nozzles. Hence, the fractures are pulled more apart during cooling. Furthermore, several other mechanisms must be considered. First, a gap probably arised at the upper edge in connection with contraction of the top crust (the unfractured zone). Second, warping of the crust at the wall interface may have created additional flow paths. Third, water trapped in wall crevices at the relocation time [Ref. 2] may have evaporated and enhanced the debris porosity near the wall.

It is seen from Fig. 2 that the water penetration rate decreases as the water front approaches the bottom, despite the downward displacement of the theoretical flow restriction point. The reason is that the gravity vector tends to be perpendicular to the flow direction. This unanticipated effect implies that the angular heat flux, especially in the wall, becomes important in the final phase and should be incorporated in the model. With a complete 2-D model, the maximum wall temperature would probably become somewhat lower and occur earlier than that of Fig. 3. Despite the possible improvements of the heat conduction model, the calculated wall temperatures (Figs. 3-5) seem to be in fair agreement with the TMI-2 VIP observations. The fact that the hot spot occurred off the centre may be attributed to clogging of the fractures by molten control rod material that happened to be accumulated here. As evidence, silver and cadmium that had relocated to the bottom prior to the corium relocation were found in the hot spot wall cladding cracks.

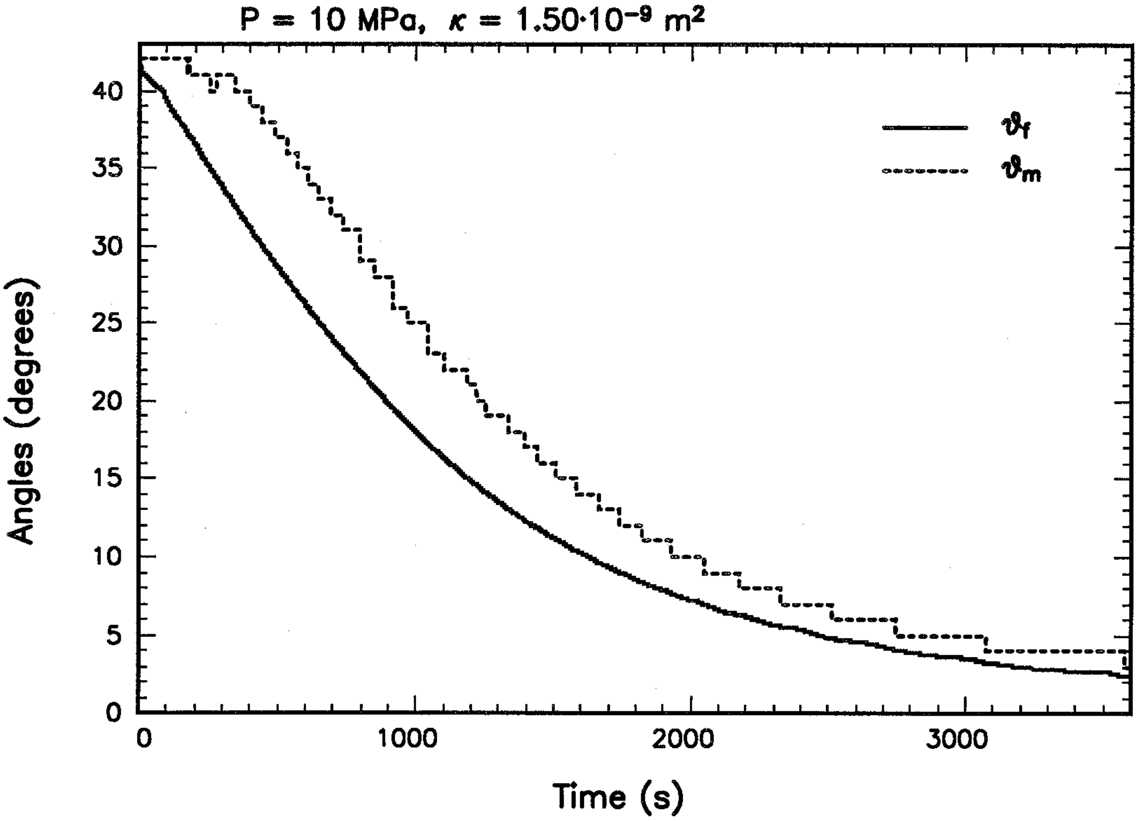

Figure 2. Progression of water front angle along the lower head (full line). The dashed curve represents the progression of a theoretical flow restriction point.

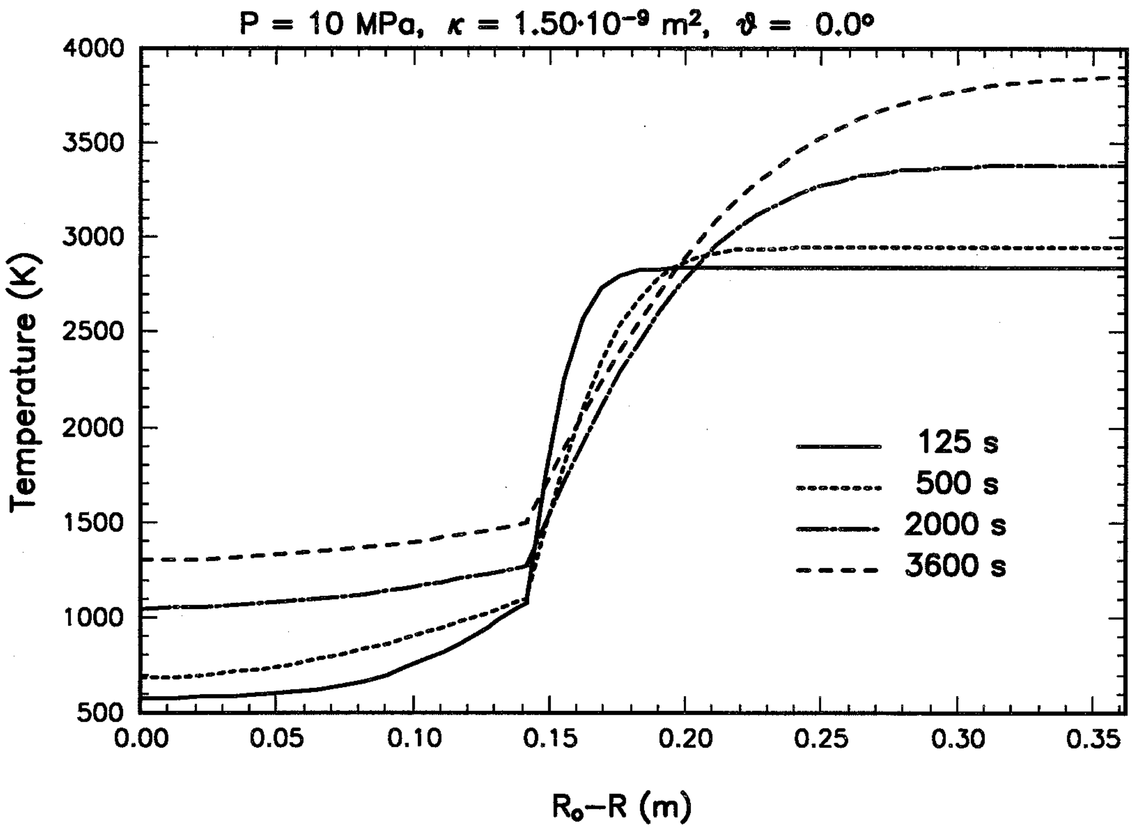

Figure 3. Temperature profiles through wall (left) and debris (right) at the vessel centre.

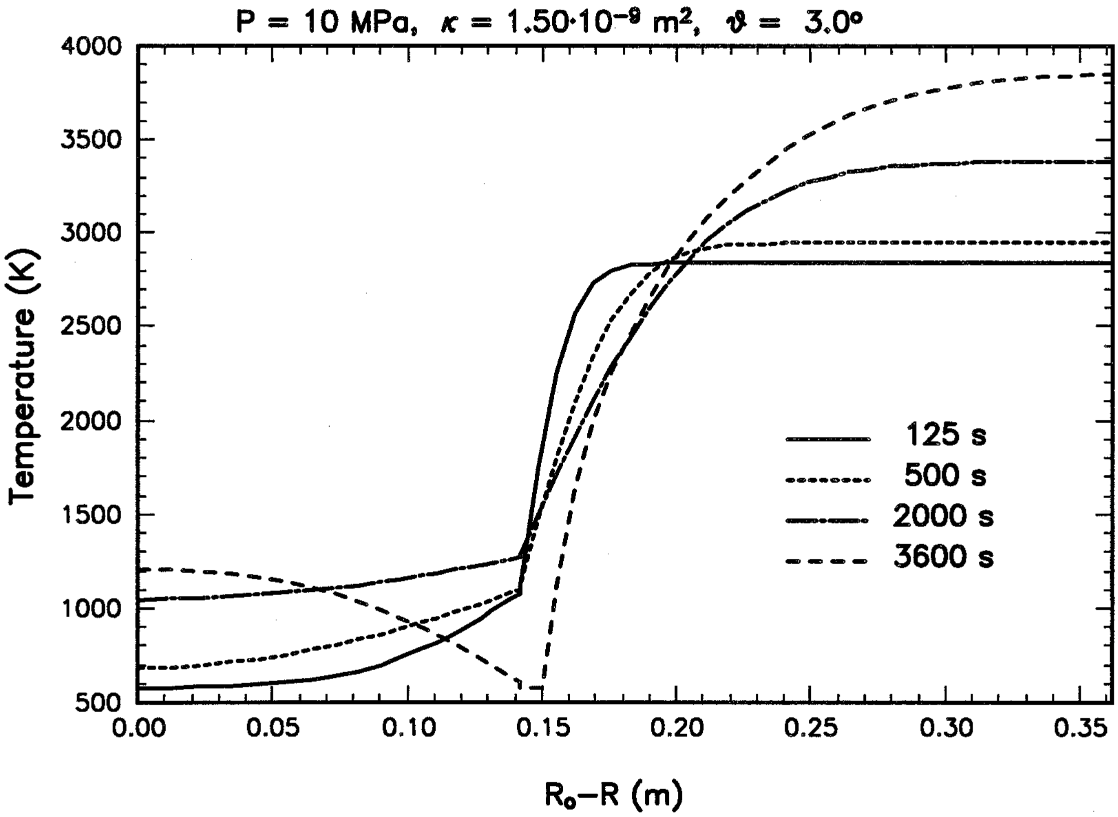

Figure 4. Temperature profiles 3° from the centre, wetted shortly before termination.

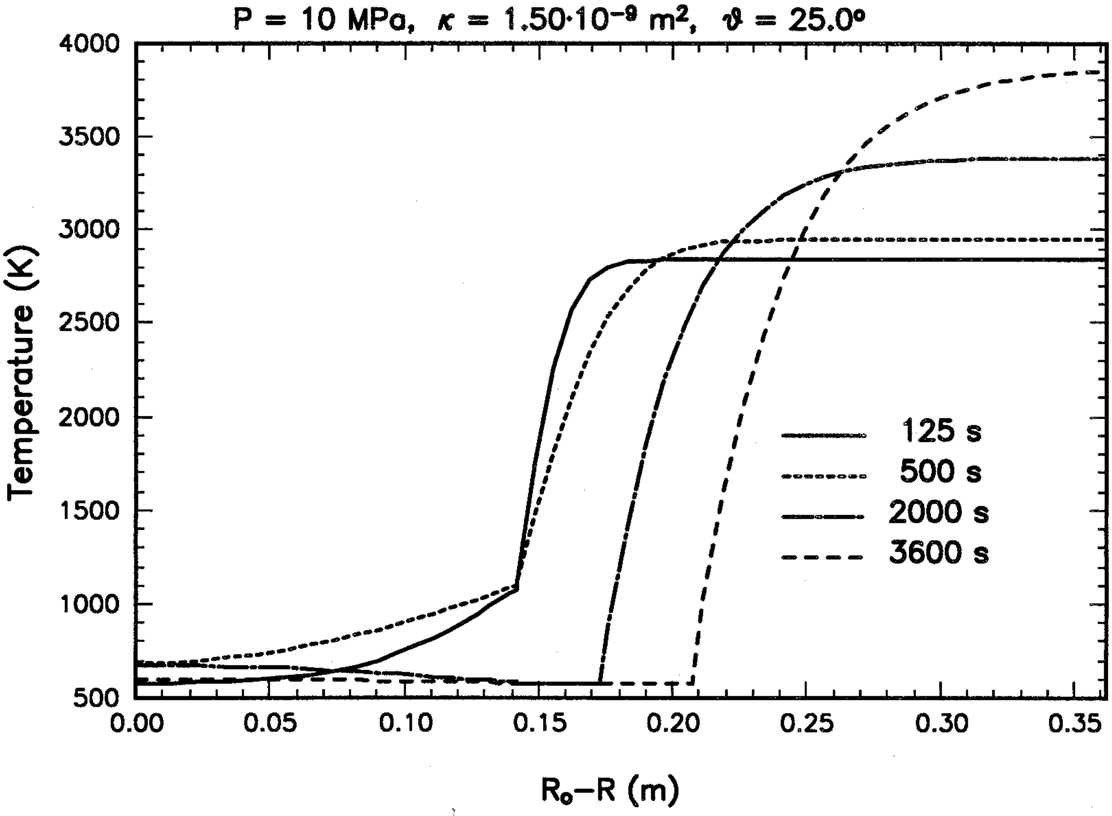

Figure 5. Temperature profiles at 25° from the centre, wetted after 11 minutes, approximately.

B.4 Recommendations

It is recommended to complete the spherical model with angular flux terms and an internal zero-flux boundary condition. Further, the effect of steam superheating should be investigated. The next step is to test both models with temperature dependent debris and wall properties, as given by Stickler et al. [Ref. 4], supplemented with an increased molten pool conductivity at high temperature.

Some of the interesting questions which are proposed to be investigated with the resulting model are how the temperature history depends on:

- high and low system pressure
- mass of debris in the lower plenum
- thermal conductivity of the debris (dependent on metallic fraction)
- vessel wall thickness

The development of a unified model of the entire lower head and debris should not be considered before the possibilities of the two separate models are exhausted. The permeability will always be an uncertain parameter that furthermore is temperature dependent. A unified model should encompass at least a two-dimensional two-phase flow model with variable steam temperature and distributed heat transfer from the debris.

References

1. WOLF, J.R. et al.: *TMI-2 Vessel Investigation Project Integration Report. NUREG/CR-6197, TMI V(93)EG10, EGG-2734, March 1994.*

2. HENRY, R.E. and DUBE, D.A.: *Water in the RPV: A Mechanism for Cooling Debris in the RPV Lower Head. Proceedings of the Specialist Meeting on "Selected Severe Accident Management Strategies", SKI Report 95:34, NEA/CSNI/R(95)3, Stockholm, Sweden, 1994.*

3. LINDHOLM, I., HEDBERG, K., THOMSEN, K., and IKONEN, K.: *On Core Debris Behaviour in the Pressure Vessel Lower Head of Nordic Boiling Water Reactors. NKS/RAK-2(97) TR-A4. October 1997.*

4. STICKLER, L.A., REMPE, J.L., CHAVEZ, S.A., THINNES, G.L., SNOW, S.D., and WITT, R.J.: *Calculations to Estimate the Margin to Failure in the TMI-2 Vessel. NUREG/CR-6196, March 1994.*

Bibliographic Data Sheet NKS-71

Title	Severe Accident Research and Management in Nordic Countries - A Status Report
Author(s)	Wiktor Frid (ed.)
Affiliation(s)	Swedish Nuclear Power Inspectorate, SKI
ISBN	87-7893-127-4
Date	January 2002
Project	NKS/SOS-2.3
No. of pages	59
No. of illustrations	0 + 6 (app.)
No. of references	48 + 4 (app.)

Abstract

The report describes the status of severe accident research and accident management development in Finland, Sweden, Norway and Denmark. The emphasis is on severe accident phenomena and issues of special importance for the severe accident management strategies implemented in Sweden and in Finland. The main objective of the 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 has been to support 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.

Severe accident research addresses both the in-vessel and the ex-vessel accident progression phenomena and issues. Even though there are differences between Sweden and Finland as to the scope and content of the research programs, the focus of the research in both countries is on 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. Notwithstanding that our understanding of these issues has significantly improved, and that experimental data base has been largely expanded, there are still important uncertainties which motivate continued research. Other important areas are thermal-hydraulic phenomena during reflooding of an overheated partially degraded core, fission product chemistry, in particular formation of organic iodine, and hydrogen transport and combustion phenomena.

The development of severe accident management has embraced, among other things, improvements of accident mitigating procedures and strategies, further work at IFE Halden on Computerised Accident Management Support (CAMS) system, as well as plant modifications, including new instrumentation. Recent efforts in Sweden in this area have been mainly concentrated on further development of accident management strategies and aids for source term predictions whereas in Finland in addition to further development of accident management strategies some important plant modifications have been carried out.

Key words — Severe accident research, severe accident management

Available on request from the NKS Secretariat, P.O.Box 30, DK-4000 Roskilde, Denmark.
Phone (+45) 4677 4045, fax (+45) 4677 4046, e-mail nks@catscience.dk, www.nks.org.