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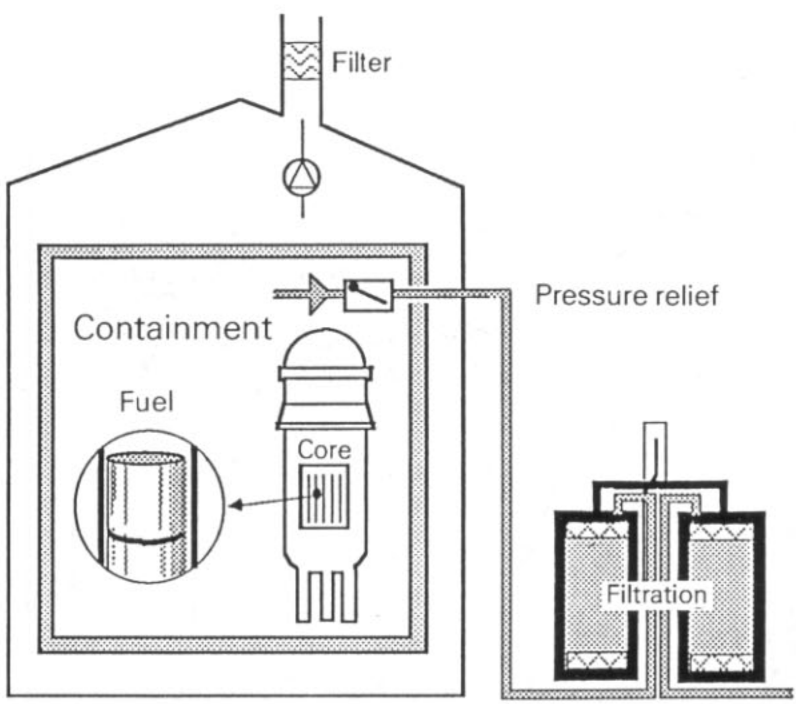
SEVERE ACCIDENT ANALYSIS

A Nordic Study of Codes



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SEVERE ACCIDENT ANALYSIS

Final Report of the Project NKA-AKTI-130

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SUMMARY

In order to analyse courses of events in severe accidents that theoretically might occur in nuclear power plants, computer codes of certain complexity are needed. The same is true when the consequences of mitigating actions are considered and also when probabilistic safety analyses are made.

Several steps should be included in codes used for calculation of accident progression. Steps to be considered include melting of the reactor core, melt-through of the pressure vessel, behaviour of corium in the containment as well as release and behaviour of radioactive matter. Timing of a severe accident is important. Particularly, time points are needed indicating start of core melt, pressure vessel failure, containment failure or start of venting. Thermal-hydraulic conditions in the primary circuit and in the containment must also be known for the purposes of design and analysis. Reliable methods are needed to analyse the behaviour of radioactive matter and factors influencing releases and source term.

Codes for integrated analysis have been developed mainly in the United States for probabilistic safety analyses and for plant specific evaluations. Two code systems have been widely used: MAAP 3.0 and the Source Term Code Package (STCP). STCP was developed for US Nuclear Regulatory Commission (NRC) related studies, and MAAP used in was developed for industry related studies. The MAAP code has been widely used in the Nordic countries in the design of mitigating measures against the effects of severe accidents. Plant specific versions of the MAAP code have also been developed for several reactor plants in the Nordic countries.

In the AKTI-130 project, these two codes have been studied and compared in regard of how different phenomena are taken into account in the codes. In order to obtain increased confidence in the predictions of these codes, benchmark calculations with both code systems were made for nuclear power plants of different types in

the Nordic countries. Thereby certain confidence could be achieved in that the codes performed within reason. This work is important because there are necessarily many uncertainties related to analysis models which are handling such a broad area of severe accident phenomenology. Also, validation and verification of code systems of this extent is a difficult task reguiring studies and large experiments.

In addition to the code comparisons sensitivity calculations were made with both codes in order to study the effects of different input data, and different phenomena that may influence the results. These analyses increased knowledge of important factors and showed to what extent the two codes are capable of handling different situations and phenomena.

The studies indicate that both codes usually give reasonable representations of possible progressions of severe accidents including core melt. The codes are suitable as a basis for safety assessment of nuclear reactors in the Nordic countries. However, due to uncertainties regarding some phenomena involved, the results must be evaluated with care. The two code systems proved to complement each other in the sense that they represent alternative modelling in certain respects, allowing corresponding uncertainties and sensitivities to be explored.

In the report detailed conclusions are drawn which suggest additional actions in certain topics, e.g. as concerns core melting and pressure vessel failure.

SAMMANFATTNING

Komplicerade datorkoder måste användas för att analysera förlopper under svåra haverier i kärnreaktorer. Det samma gäller vid analys av konsekvenslindrande åtgärder eller vid probabilistika säkerhetsanalyser. Koderna måste inkludera många steg, såsom reaktorhärdens smältning, genomsmältning av reaktortanken, smältans beteende i inneslutningen samt frigörelse och fortsatt beteende av de radioaktiva ämnen. Det är viktigt att tidsförloppet för postulerade svåra haverier analyseras; viktiga tidpunkter är exempelvis start av härdens smältning, genomsmältning av reaktortanken och eventuell avblåsning från inneslutningen orsakad av medveten ventilering eller brott på inneslutning. Också termohydrauliska parametrar avseende tillståndet i reaktorns primärsystem och i inneslutningen är intressanta ur konstruktions- och analyssynpunkt. Det är viktigt att de radioaktiva ämnenas beteende och de faktorer som påverkar ämnenas frigörelse och källterm analyseras på ett tillförlitligt sätt.

Integrerade analyskoder för probabilistiska analyser och för utvärdering av enskilda anläggningar har huvudsakligen utvecklats i USA. Det finns huvudsakligen två kodsystem som idag används, nämligen MAAP 3.0 och Source Term Code Package (STCP). STCP har utvecklats för US Nucler Regulatory Commission (NRC) anknutna studier och MAAP har utvecklats för industri-anknutna studier. MAAP-koden har i omfattande utsträckning använts i de Nordiska länderna, bl a i samband med utvecklingen av system för att lindra konsekvenserna av svåra haverier. Speciella versioner av MAAP-koden har utvecklats för reaktorer i Norden.

I AKTI-130-projektet har dessa koder studerats genom teoretiska jämförelser och studium av hur olika fenomen beaktats i koderna. För att öka tilltron till de båda kodernas prediktioner har såkallade benchmark-beräkningar utförts för utvalda reaktorer av olika typ i Norden. Genom detta arbete har man kunnits verifiera de speciella Nordiska versionerna av dessa koder och respektive beräkningar för reaktorerna. Denna typ av insats är viktig eftersom det finns osäkerheter i de beräkningsmodeller som behandlar ett så omfattande

spektrum av fenomen som ingår i förloppen vid svåra haverier. Validering och verifiering av kodsystem av denna omfattning är en svår uppgift som kräver stora experiment och andra studier.

Utöver kodjämförelseberäkingarna har många känslighetsberäkningar utförts med båda koderna för att studera inverkan från såväl skilda kodsystem, skilda kod- och anläggningsspecifika indata och skillnader i fenomen och modeller som andra känsligheter. Dessa analyser har ökat kunskapen om viktiga faktorer och demonstrerat kapaciteten hos de båda koderna med avseende på behandlingen av olika slags situationer och fenomen.

Baserat på dessa studier kan de konkluderats att de båda koderna allmänt sett ger en god representation av möjliga fortskridanden av postulerade svåra haverier innefattande smältning av härden och att koderna är lämpliga som grund för värderingen av de Nordiska reaktorernas säkerhet. På grund av vissa osäkerheter beträffande en del av de inblandade fenomenen måste emellertid resultaten från koderna utvärderas med omsorg. De två kodsystemen har visat sig komplettera varandra när viktiga parametrar och fenomen studerades och defienerades.

I rapporten dras många detaljerade slutsatser som ger anvisningar beträffande fortsatta arbeten inom ett antal områden.

CONTENTS

1 INTRODUCTION

1.1 Summary of the Work in the Project NKA-AKTI-130

The project AKTI-130 "benchmark and sensitivity analysis" was carried out during 1985-89 and it was sponsored by NKA, the Nordic Liaison Committee for Atomic Energy. The project handled severe accident phenomena and calculation models describing accident progression inside the reactor containment. One of the objectives of the AKTI-project was to get a common Nordic understanding about the use of the existing analysis methods in severe accident analysis. The specific objectives were as follows:

1) to increase understanding about the capabilities of the severe accident codes MAAP 3.0 and MARCH 3/Source Term Code Package by making
   - benchmark calculations
   - other comparisons between codes and analyses

2) to make sensitivity analyses to study the effects of different parameters, submodels and phenomena on the whole accident process and to identify important parameters, submodels and phenomena for further action.

The participants in the project AKTI-130 were Ilari Aro, Finnish Centre for Radiation and Nuclear Safety (AKTI-110 project group, project co-ordinator), Roland Blomquist, Studsvik Nuclear (Sweden), Esko Pekkarinen, Technical Research Centre of Finland (Finland), and Bjarne Schougaard, Elsam (Denmark). Uffe Steiner Jensen, Elsam participated in the initial phase of the project (1985) and Peter Fynbo, Risø, in the final phase of the project (from 1988).

The work to be performed in the project was defined in 1985 /1/. This definition was based on a short theoretical comparison of the existing codes and on the calculations performed so far in the Nordic countries. The Swedish RAMA study and the Danish Elsam study served as a good basis for the definition /2,3/. In 1986 analyses concerning Nordic BWRs were performed and the results are presented in the report AKTI-130(86)1 /4...9,18/. This work was continued in 1987 and 1988 by making calculations with the MARCH 3/TRAP-MELT codes for comparison with the MAAP 3.0 code /10,11,17/. In 1987 the main emphasis was put on the calculation of Nordic PWRs and the results are presented in the report AKTI-130(88)3 /12...19/. After the benchmark and sensitivity analyses for the Nordic BWR and PWR plants the work was directed to a theoretical comparison of the existing codes used in severe accident analysis in the Nordic countries. Three special seminars were arranged and results are presented in this report and in /20/. Besides NKA reports, results from the project were presented in international meetings in Brussels /9/, Sorrento /16/ and Avignon /17/.

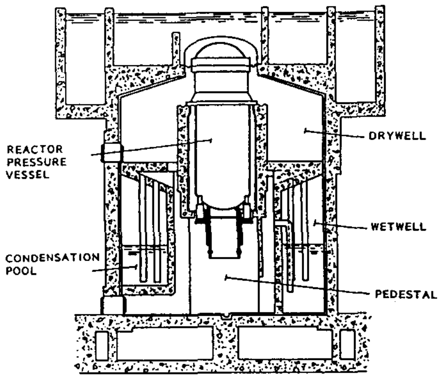
1.2 Codes and Accident Sequences Studied

The codes used in the severe accident analyses in the Nordic countries are MAAP 3.0 /21,22/ and the Source Term Code Package (STCP) which contains several separate codes like MARCH 3, TRAP-MELT, VANESA, NAUA, ICEDF and SPARC /23,24/. MAAP 3.0 was applied by STUDSVIK for the Forsmark 3 and Ringhals 2/3 plants and by VTT for the TVO and Loviisa plants. MARCH 3-(STCP mod 1.0) was applied by Elsam for the the Forsmark 3 and Loviisa plants. The computers used for the calculations were CDC in STUDSVIK, MicroVax II in VTT and SPERRY 1100 in Elsam.

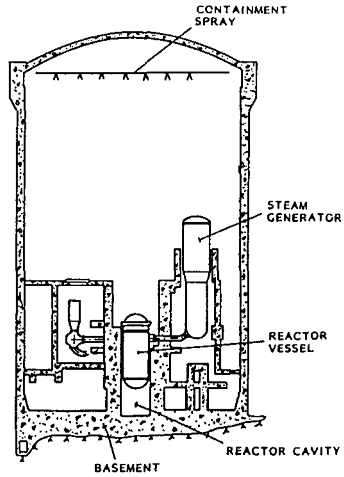
The nuclear power plants used as reference plants in the analyses were:

- Forsmark 3 (Asea-Atom 1100 MW BWR)
- TVO I/II (Asea-Atom 710 MW BWR)
- Ringhals 2 and 3 (Westinghouse 800 respective 915 MW PWR, dry containment) and
- Loviisa 1/2 (VVER-440, ice condenser containment).

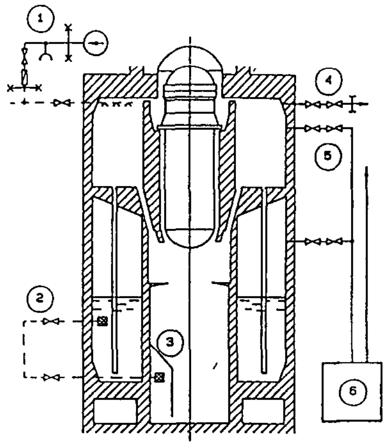
Containment types are presented in Figure 1. Some assumptions which do not conform with the present status of the structure and operation of the safety systems in these plants have been made because the actual designs were not known at the time of the analyses and because these features were considered for these power plants. The smark 3 and Oskarshamn 3. Containment configuration.



Ringhals 3 and 4. Containment configuration.

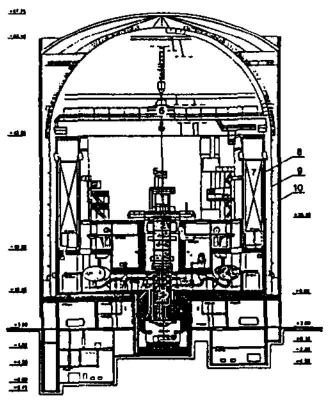


TVO I/II Systems for severe accident mitigation



1 Water filling of the containment
2 Flooding of the lower drywell
3 Shielding of penetrations in the lower drywell
4 Containment overpressurization protection
5 Containment venting
6 Filter

Loviisa Power Plant



Loviisa ice condenser containment

1. Reactor pressure vessel
2. Control rod drives
3. Main circulation pump
4. Steam generator
5. Refuelling machine
6. Polar crane
7. Ice condenser
8. Inner concrete wall of containment building
9. Steel containment
10. Outer concrete wall of containment building

Fig. 1. Nordic containment types used as reference in the AKTI-130 project.

assumptions in the analyses performed in this project were as follows:

- depressurization of the primary system (BWRs, Loviisa)
- flooding of the reactor cavity with water (BWRs, Loviisa)
- start of independent containment spray at 8 h after the initiation of the accident (Forsmark 3, Ringhals 2/3)
- venting from the containment.

In Sweden and Finland nuclear power plants are, or will be, furnished with containment venting lines with a special filter system. However, in this report the decontamination factors relating to this filter system have not been taken into account. Thus, readers should keep in mind when studying the calculated source term values that an additional decontamination factor of about 100-500 will decrease the release fractions.

Because of the above assumptions the results should not be treated as representative for the present power plants related to containment system behaviour and source term.

The accident sequences studied were "total loss of AC-power (TB, TMLB') and "small and intermediate LOCAs with total loss of AC-power (S_2B, S_1B)". In the selection of accident sequences Swedish experience in the design of filtered vented containment systems and Finnish safety guides (YVL guides) have been taken into account. The latest large PSA study, U.S. NUREG-1150 shows that a "total loss of AC power" and a "small LOCA with total loss of AC power" are representative initiating events for this kind of study.

A short description of these severe accident situations is as follows: TB (BWR), and TMLB' (PWR) mean an accident in which all electrical AC-power that is outer grid lines and reserve power (gas turbines and in-plant diesel generators) are inoperable. Only battery power and safety functions based on it operate properly until batteries are empty. The reactor and turbines are tripped and because of the loss of electricity main feedwater is lost (for Ringhals, also the steam driven auxiliary feed water and emergency core cooling pumps are assumed to fail). The water inventory of the primary system is boiled off via the safety valves to the containment. The reactor core is uncovered and melted. The core melt penetrates the bottom of the pressure vessel and drops into the reactor cavity. The melt is cooled down by the water in the cavity. Containment is pressurized slowly by steam to the design pressure of the containment when venting is started to protect containment integrity. S_2B/S_1B differ from TB and TMLB' in that there is a break in the primary system followed by an immediate loss of all AC-power.

1.3 Benchmark and Sensitivity Analyses Performed

Benchmark analyses were necessary to carry out because of the uncertainties in the analysis models. Experimental data for the validation of code systems modelling the complex phenomena involved in severe accidents are, however, limited. It was valuable in this situation to compare models and results for two code systems developed by different organizations. The two code systems compared were the MAAP 3.0 code and the STCP codes MARCH 3 and TRAPMELT. Code comparison was made by performing benchmark calculations for a Nordic BWR-plant (Forsmark 3) and a PWR plant furnished with ice condenser (Loviisa). Two accident sequences , namely a "total loss of AC-power" and a "LOCA and loss of AC-power" were studied.

Benchmark calculations were performed for Forsmark 3 in the cases of TB-sequence and S_2B-sequence (steam line LOCA, area 0.009 m^2) with the following assumptions: ADS and lower drywell flooding take place automatically at about 12 min; containment and drywell/wetwell-wall are leak-tight; and venting line (Ø 0.15 m) is situated in the wetwell and release pressure is 0.7 MPa. STUDSVIK made these calculations with the MAAP 3.0 code and Elsam with the MARCH 3 and TRAP-MELT codes.

Benchmark calculations were performed for the Loviisa PWR in the cases of a S_1B-sequence (hot leg LOCA, area 0.0143 m^2) and a TMLB'-sequence (pressurizer safety valve locked open when opened at the first time with a flow area of 16.4 cm^2) with the following assumptions: containment is leak-tight and there is no bypass area of ice condenser (in the MAAP TMLB' case 0.78 m^2 bypass was used);

the venting line (Ø 0.15 m) is in the upper part of the containment and the release pressure is 0.17 MPa. VTT made calculations with the MAAP 3.0 code and Elsam with the MARCH 3 code.

Sensitivity analyses were performed for both BWR and PWR plants for studying the most important models and parameters. The main items studied were: core melt progression and hydrogen production, thermal hydraulic conditions in the primary system and in the containment, time behaviour of the accident sequences,aerosol transport, source term and performance of mitigation systems. Decontamination in the filters was not taken into account.

In the sensitivity analyses for the BWR and PWR plants the following parameters were varied: accident sequence, parameters related to core melt progression and hydrogen production, heat transfer, aerosol behaviour, location and size of primary circuit break, containment leakage area, location of the venting line (BWR), starting time of venting, bypass of condensation systems and operator actions.

Results from the benchmark and sensitivity analyses are presented in chapters 4 and 5 and in /4,11,12,17/.

1.4 Theoretical Comparison of the Codes

Besides benchmark calculations the two codes were also compared theoretically by studying the modelling differences. A comparison of the models with the experimental results available from TMI-studies or latest tests in the U.S. SFD, LOFT- or German CORA-experiments has also been made in some detail especially relating to core melting and hydrogen production. Three special seminars were arranged for studying thermal hydraulics in the primary circuit and in the containment as well as aerosol behaviour, where representatives from AKTI-130-, RAMA- and VARA-projects compared the modelling assumptions of the two codes and their effects on the analysis results. Main headings were as follows: (I,1) core heatup and melting and hydrogen production; (I,2) corium behaviour in lower plenum and melt-through; (I,3) heat transfer inside primary system; (II,1) corium quenching in containment and pressure and temperature transients; (II,2) special phenomena like core-concrete interaction etc.; (II,3) engineered safety systems; (III) fission product release and aerosol behaviour /20/. Detailed comparisons have also been made when differences in the results of the benchmark calculations have been studied /4,12/. The main results from theoretical comparison have been presented in chapters 2 and 8.

2 DESCRIPTION AND THEORETICAL COMPARISION OF CODE SYSTEMS

2.1 General Description of MAAP 3.0

The Modular Accident Analysis Program (MAAP) is a computer code which simulates light water reactor system response to accident initiating events. The code has been developed in successive steps by Fauske & Associates, Inc (FAI), Chicago, USA under contract with IDCOR (Industrial Degraded Core Rulemaking Program). The first version of the code, MAAP 1.1, was released in 1983. Since then, the code has been continually improved. The MAAP calculations presented and discussed in this report have been performed with the version MAAP 3.0 which was released in 1986. The code description given below will concentrate on this version. A new version called MAAP 3.0B has been released in 1988.

Separate MAAP versions exist for BWR and PWR. The US version of MAAP 3.0B/PWR is also applicable for the Swedish PWRs R2, R3 and R4. However, due to significant design differences the US MAAP 3.0/PWR and BWR versions are not directly applicable to the Finnish reactors Loviisa 1/2 and TVO 1/2 and the Swedish reactors Barsebaeck 1/2, Forsmark 1/2/3, Oskarshamn 1/2/3 and Ringhals 1. Therefore, special versions of MAAP 3.0/PWR and BWR have been developed for Lo, TVO, B1/B2/O2, F1/F2, F3/O3, O1 and R1. The special versions have been developed by FAI with the exeption of the F1/F2 version, which has been modified from F3/O3 model by VTT.

As indicated by the program name, the MAAP code is structured in a modular format in which phenomenological models are treated in individual subroutines. This format makes it easy to incorporate improvements and new models into the code.

The code is organized on the basis of the physical regions of the reactor plant. The BWR code includes the following parts: primary system, drywell, pedestal cavity, wetwell and auxiliary building.

The PWR code includes the following parts: primary system, pressurizer, steam generators, cavity (Compartment C), upper containment (Compartment A), annular compartment (Compartment D), pressurizer relief tank (Quench Tank), lower compartment (Compartment B) and auxiliary building.

For each region, MAAP calculates the instantaneous rates of change of temperature, pressure, mass of steam, mass of UO_2 and other dynamic variables. This is done by calling the phenomenological subroutines which calculate the rates of the various physical processes. Important phenomena modeled in the subroutines are: coolant flow during LOCAs, coolant boil-off, core heat up, core recooling by flooding or spray, zirconium-water reaction, core melting, transfer of corium to lower plenum, water cooling of corium in containment, vessel melt-through, transfer of corium to containment, containment spray, containment cooling, quenching of corium in containment, corium-concrete interaction, containment pressure build-up, containment venting.

Beside this thermal-hydraulic part, the fission product region subroutines calculate simultaneously the mass rate of change for each of the six fission product groups, which are noble gases, CsI, CsOH, Te-group, Sr-group, Mo-group and structural materials. Revaporization of the fission products is included in the models.

An important feature of the code is the use of an array of "event codes" to characterize the instantaneous state of the reactor plant and to control problem execution. There are three types of event codes:

1) events calculated by the code
2) user specified external events
3) user specified operator actions.

The events are either true or false and tell the region subroutines which subroutines are on, which valves are open, if there is a hydrogen burn in any compartment etc.

The code is written in Fortran IV and has been successfully run on a number of different computers.

2.2 General Description of Source Term Code Package (STCP)

The STCP contains six separate codes such as MARCH 3, TRAP-MELT 3, VANESA, NAUA and SPARC or ICEDF depending on plant type (Fig 2). The STCP is made under the auspices of the United States Nuclear Regulatory Commission, USNRC, and all descriptions of models and code user manuals and experience in using the code package are publicly available. The version of the STCP used for calculations in the NKA-AKTI-130 project is the mod 1.0 of June 1986 /23/.

MARCH 3 calculates the overall thermal-hydraulic behaviour of the reactor and the containment. The following phenomena are described: heatup of the reactor coolant inventory and pressure rise; initial blowdown of coolant; generation and transportation of heat within the core; heatup of fuel following core uncovery including metal-water reactions; melting and slumping of fuel onto the lower core support structures and into the vessel bottom head; interaction of core debris with residual water in the vessel; interaction of core debris with reactor vessel bottom head and subsequent failure of head; interaction of core debris with the water in the reactor cavity; attack at the concrete basemat by the core and structural debris; relocation of the decay heat source as fission products are released from the fuel and transported to the containment; mass and energy additions to the containment associated with all the mentioned phenomena and their effects on containment temperature, pressure and steam condensation; effects of the burning of hydrogen and carbon monoxide on the containment pressure and temperature; leakage of gases into the environment.

MARCH 3 is primarily intended for addressing accidents leading to a complete core meltdown but it can also be used for calculating events involving only partial core degradation as well as for

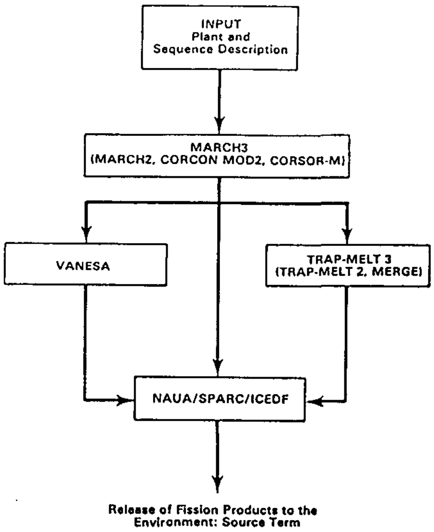


Fig. 2. The NRC's Source Term Code Package

MARCH CONTROL VOLUMES

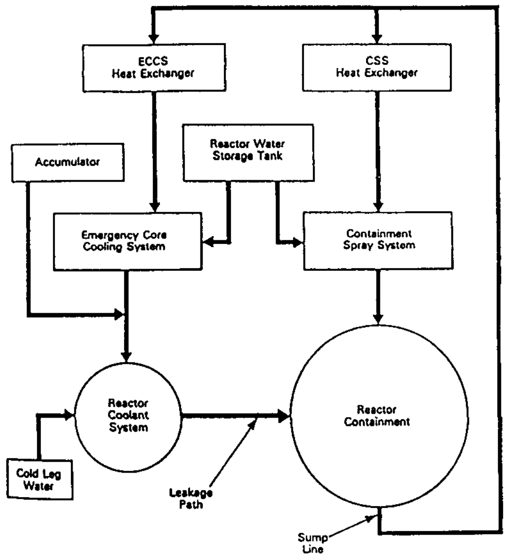


Fig. 3. Control volumes in the MARCH code for a typical PWR analysis

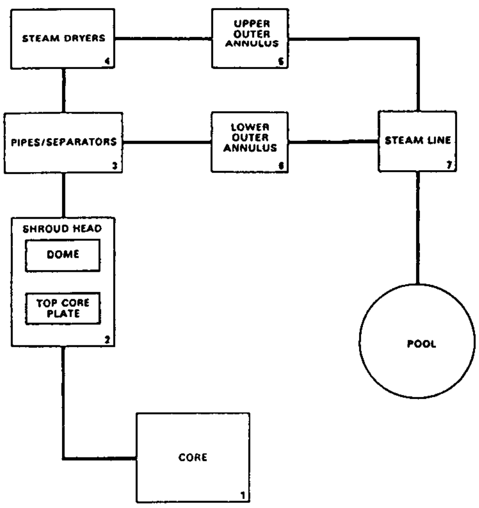


Fig. 4. Seven control volumes used in the TRAP-MELT code for analysis of the TB sequence in a BWR plant

MOLTEN CORE DEBRIS MODEL

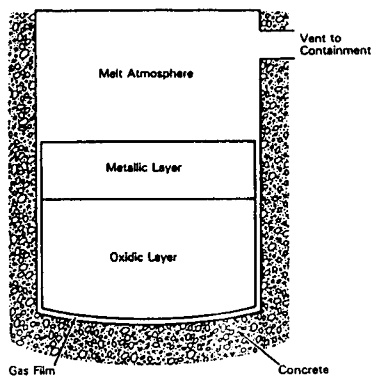


Fig. 5. Molten core debris in the reactor cavity as modeled by the CORCON code

/24/

assessing the minimum levels of engineered safety feature operabilities required to cope with various accident events. MARCH 3 is designed to cover the entire accident sequence, from the initiating accident event to the core concrete interaction, for a variety of accident initiators and including coverage of a wide variety of reactor systems designs, e.g. BWRs with MARK I, II and III design containments and PWRs with ice condensers or large dry containments or subatmospheric containment design.

In the programming of the code, the idea is to describe well understood phenomena to a level consistent with the needs. For phenomena which are not well understood, a number of user-specified options in the code may be selected to explore the effects of various modelling assumptions. Recommended default choices are provided when possible. A number of user-selected options are maintained to make the code capable of covering a wide range of reactor designs and accident sequences. In all cases mass and energy are conserved so that calculated sequences are self-consistent. There is no general bias in the code to produce "conservative" or "non-conservative" calculations; however, choices of models and parameters by the user can make a calculation conservative or non-conservative, although it is not often easy to judge whether a calculation is conservative or not.

The MARCH control volume scheme is shown in Figure 3. The single control volume used for the reactor coolant system is too simple for providing thermal-hydraulic data for the TRAP-MELT code which is used to calculate the transportation and retention of fission products. Therefore, there is a separate subcode MERGE in the TRAP-MELT to provide flow rates and temperatures in more detailed volumes of the primary system downstream of the core (Fig 4). The materials released from the core are divided into ten groups which are treated separately. It is assumed that iodine and cesium are in the form of CsI and CsOH and Te in elemental form. These three groups leave the core as vapors. Condensation and revaporization on walls and aerosol particles are described. The rest of the less volatile fission products and construction materials like Zircaloy, stainless steel and the control rods are treated as aerosols. Aerosols can deposit and agglomerate but they cannot evaporate. As a result of MARCH 3- and TRAP-MELT 3-calculations, fission product and aerosol release from the primary system to the containment is calculated. The TRAP-MELT calculation is stopped when a melt-through of the pressure vessel takes place.

The core-concrete interaction is described by a subcode of MARCH 3 called CORCON-Mod 2. CORCON calculates the rate of erosion of the concrete cavity, the temperature and composition of the melted layers (Fig 5) and the temperature, flow rate, and composition of the gases (CO_2, CO, H_2 and steam) which evolve from the concrete. Then, the VANESA code calculates the release of fission products from the melted core debris.

The behaviour of aerosols in containment volumes is calculated by the NAUA code. The code does not handle volatile species. This code version can handle condensing steam atmospheres and containment spray systems. When treating multi-compartment containments, NAUA calculations are performed sequentially for connected volumes. The code calculates e.g. the size distribution of airborne material as a function of time, the cumulative settled-out and plated-out quantities and the cumulative leaked mass. The codes SPARC and ICEDF calculate aerosol retention in the suppression pool of BWRs and in the ice condenser of some PWRs.

2.3 Primary System Thermal Hydraulic Models

Some of the most important discrepancies between MAAP and MARCH concern core heat transfer during core heat-up, core melt progression, zirconium-steam-reaction, core slump, gas circulation inside primary system after core slump, corium behaviour in lower plenum and melt-through of the reactor pressure vessel bottom head.

2.3.1 Physical Regions

In MAAP and MARCH the primary system is divided into the following physical regions:

MAAP

BWR: core (incl 2 heat sinks), shroud head (1), standpipes & separators (1), upper head (2), upper downcomer (1), lower downcomer (2), lower head (1), recirculation loop (1). (See Fig 6.)

PWR: core (incl 1 heat sink), upper plenum (3), dome (1), hot leg (1), pressurizer (1), hot leg tubes (1), cold leg tubes (1), intermediate leg (1), cold leg (1), downcomer (1). (See Fig 7.)

MARCH

BWR and PWR: core (incl 3 heat sinks), upper head and downcomer (1), lower head (1), dead water volume;
Additional heat sinks: steam generators (PWR) and piping. (See Fig 8.)

2.3.2 Water Losses

Under severe accident conditions, water is lost from the primary system in the following ways:

- LOCA break, water or steam
- through safety valves, steam only (at pressures above set points)
- through relief valves, steam only (automatic depressurization, etc).

Both codes model these three kinds of water losses.

2.3.3 Core Heat Balance

Both codes take decay heat and heat from metal-water reaction into account. For water covered parts of the core they both model water cooling by convection and boiling.

For the uncovered parts of the core, they both take into account convection heat transfer to steam and hydrogen and radial radiation heat transfer to adjacent core nodes and to the core barrel. In

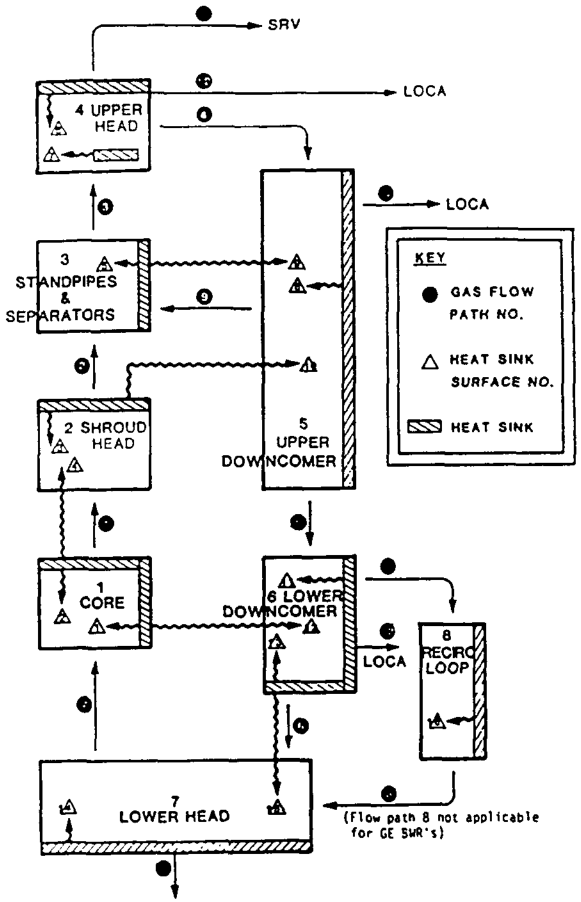


Fig. 6 BWR primary system nodalization.

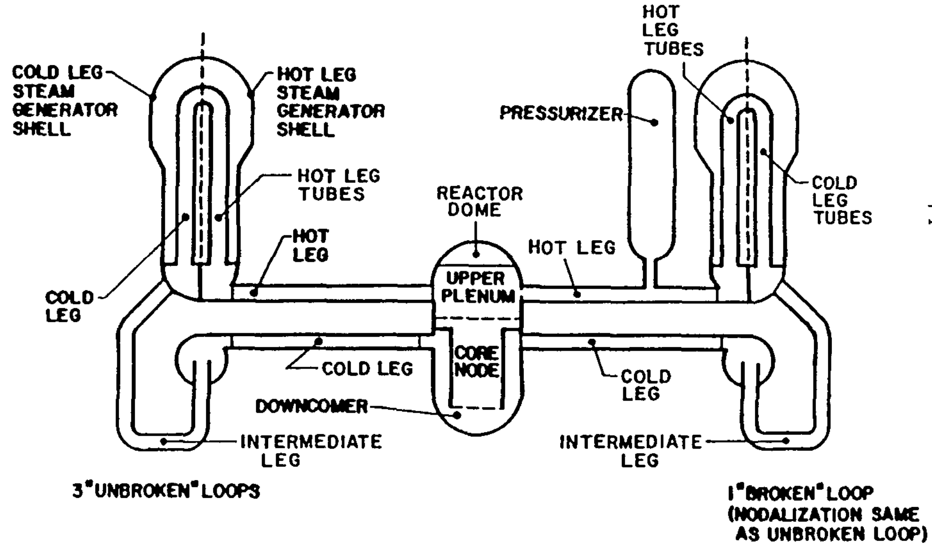


Fig. 7 Application of PWR primary system nodalization to a Westinghouse 4 loop design. /21/

addition, MARCH also takes into account radiation heat transfer to steam and hydrogen, axial conduction, radiation heat transfer from the top nodes to the structure above the core and radiation heat transfer from the nodes above the steam-water-mixture level downwards to the water. Additional models in MAAP take into account radial heat transfer from the core barrel to the vessel wall and further to the containment atmosphere. MAAP also models spray recooling of the dry core. Only unmelted parts of the core can be cooled by the spray. Water cooling of molten core material within the core volume is not modelled in MAAP.

The heat balance calculation in MARCH is made dependent on the choice of model for melting of the core and whether a detailed BWR core model is used or not.

2.3.4 Core Melting Progress

The MAAP model for core melting assumes a common melting temperature for fuel, cladding and fuel channel. The melting model does not consider the control rods. The core is divided into 50 nodes (10 axial layers and 5 radial columns). The division is such that all nodes have equal mass contents.

When one node starts to melt, the molten material is transferred to the node below as long as this node is not completely filled. The downflowing material transfers heat to the not melted material. The downflowing material may refreeze. Due to an internal energy generation, the mixture of unmelted and refrozen material then melts.When the lowest node in a column is completely melted, all molten material leaves the column. Material is not allowed to flow between columns. Unmelted material stays in the upper part of the column. This material melts continually and flows out of the column.

Performed calculations show that an outer ring of fuel will be left unmelted in the core. The radial heat losses to the core barrel keep its temperature below the melting point.

In the MARCH model the core is divided into up to 500 nodes (50 axial layers and 10 radial columns). Two different core models can

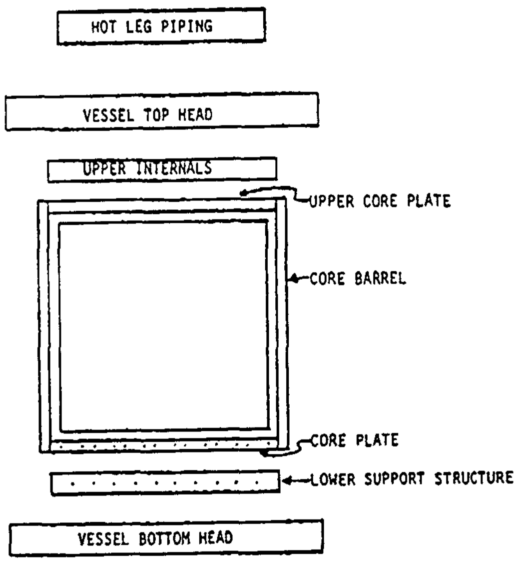


FIGURE 8. REPRESENTATION OF CORE REGION GEOMETRY USED IN MARCH

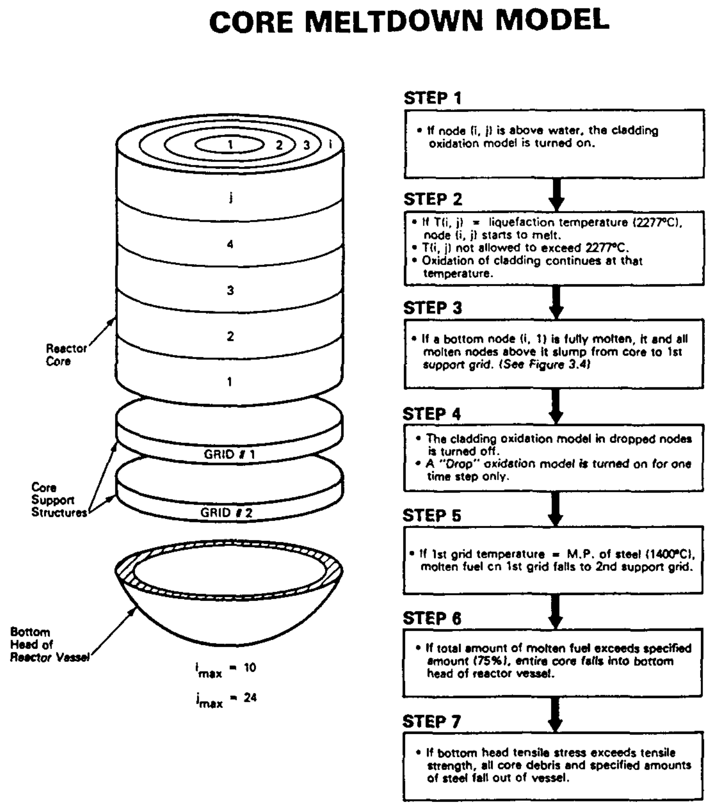


Figure 9. Meltdown model in the MARCH code as used in BMI-2104 and in the current *Source Term Code Package* analyses. /24/

be used. A simple model treats the canning material as a part of the fuel rod cladding. A detailed core model treats the fuel rods in the fuel channels, the flow boxes,and the control rods in the moderator channel separately. This detailed model was used in the MARCH calculations for Oskarshamn 3/Forsmark 3, Loviisa and TVO I/II.

The MARCH code has three models for core melting:

1) melt progression is downward
2) melt progression is upward
3) each node slumps when melted.

Another possibility is a gradually slumping model, which means that the core drops nodewise to the lower structures when the melted part of the core is larger than a certain input-condition. This combines models (1) and (3).

In the melting model (1) it is assumed that a molten region forms in the core and that it grows downward, so that the average temperature of the melted nodes is held at the melting temperature, which is input. The downward movement is modelled so that excess heat (if temperature is above the melting temperature) in the uppermost melted node in a radial region is distributed to the next downward node and so on. (See Fig 9.)

In this model radial zones are treated separately with no mixing between the regions. This model can be superposed with the gradually slumping model, which means that melted nodes drop to the lower structures.

2.3.5 Core Blocking and Zirconium-Water Reactions

MAAP models the flow of steam and gas through the core during core melting. MAAP also models blocking of core columns by molten material. The hydrodynamic balance between downflowing melt and upflowing gas is calculated and in this way it is decided whether blocking should take place or not.The blocking,which is irreversible, takes place column by column. The whole core can be blocked.

Upflowing steam and gas from lower plenum can bypass the core through a modelled channel.

As long as the overheated core is unmelted, MAAP models the hydrogen-producing chemical reaction between Zr and H_2O. The reaction is limited by the Zr and H_2O available. When a BWR-column is blocked by molten or refrozen material the reaction ceases due to steam starvation. (As mentioned above, this blocking is irreversible.) When the whole core is blocked there will be no further reaction according to the model. Possible further reaction after the central part of the core has dropped out is not modelled.

In MARCH, the zirconium-steam reaction is governed by minimum of a gaseous diffusion rate/steam flow or a solid state diffusion rate in the oxidized layer of the fuel.

Four different assumptions for the zirconium-water reaction can be used:

1) no zirconium-water reaction
2) no zirconium-water reaction in melted nodes
3) no zirconium-water reaction above the lowest melted node in a radial region (channel blockage)
4) zirconium-water reaction not stopped by node melting.

There is a choice between Urbanic-Heidrick, Catchcart and Baker-Just models for the zirconium-steam reaction. Only Catchart and Baker-Just models can be used together with the detailed core model.

2.3.6 Core Collapse

According to the MAAP modelling, only molten material leaves the core volume. Any unmelted fuel material stays in its position.

The MARCH code has three possibilities for initiating core slumping:

1) slumping occurs when the amount of melted core exceeds input fraction (default 0.75)

2) slumping occurs when the core barrel exceeds its melting temperature
3) slumping occurs when the structure below the core is heated above its melting temperature by the part of the core that has dropped down onto it.

When the core slumps, the whole core, both melted and unmelted nodes, falls into the bottom head.

2.3.7 Cooling of Corium in Lower Plenum

According to the MAAP results, the first transfer of melt from the core volume to the lower plenum will usually include a large amount of molten material. It will therefore form a molten pool. If there is water above, the heat flux between the melt and the water is calculated based on the critical heat flux correlation.

In MARCH, the behaviour of the melt after a core slump is governed by a choice between 6 models for the formation of a debris in the bottom head. I.e.:

1) March 1.1 model
2) particulate model
3) flat-plate model plus Berenson film boiling
4) Dhir-Catton debris bed
5) Lipinski debris bed
6) Ostensen-Lipinski debris bed.

An important parameter is the diameter of the particles if a debris bed is formed. This diameter governs the heat transfer surface of the melt and thereby the boil-off of water and the pressure build up.

The temperature of the water into which the melt falls is also an important parameter. If the water in the bottom head is at saturation temperature the slumping will give a blow of steam and hydrogen.

2.3.8 Metal-Water Reactions in Lower Plenum

The hydrogen-producing reaction between hot Zr and H_2O in the lower plenum is modelled in MAAP/PWR. The reaction is not considered in actual versions of MAAP3.0/BWR.

In MARCH it is assumed that the melt takes the form of spheres when it falls into the bottom head water. The diameter of the spheres is input and the total number of spheres is calculated from the total volume of the melt.

The spheres can be assumed to have a central core with up to two shells surrounding it and through input fuel, zirconium and zirconium-oxide is distributed between the core and the shells. When the temperature of the shells drops below 1366 K (2000 F) hydrogen production stops.

2.3.9 Temperature of Vessel Wall Close to the Core

The transient temperature of the pressure vessel wall outside the core is modelled in MAAP. The model includes heat transfer from the core through gas gaps and core barrel, heat transfer to the containment atmosphere through vessel wall insulation and heat received from deposited fission products.

There are no such models in MARCH.

2.3.10 Vessel Melt-Through

As stated in 2.3.7, according to the MAAP modelling, a pool of molten corium will usually form in the lower plenum. The MAAP/BWR code includes a model for the heat transfer between the melt and the pipes penetrating the lower head of the reactor vessel. The usual result is that the welds or the wall of the pipe will melt in a short time and there will be an open connection between the lower plenum and the containment.

MAAP/PWR has no such model. Instead, a user-specified time delay between the first drain of melt from the core and the melt-through of the vessel is used. The suggested time is about 60 s. This seems reasonable for vessels with lower head penetrations. But for Loviisa where there is only a solid steel wall in the lower head this time must be considerably longer.

MARCH can not model in detail the bottom head with all its penetrations. It is possible to specify either the actual thickness of the bottom head, an effective thickness considering the influence from the penetrations or the actual thickness of the penetrations. MARCH calculates the time for the bottom head failure taking into account stresses in the bottom head due to the weight of the corium and the pressure in the vessel.

2.3.11 Transfer of Core Material out of Vessel

The MAAP model for the discharge of molten material from the reactor vessel includes the effect of the static head, pressure difference between vessel inside pressure and containment pressure and the continuous ablation of the vessel material surrounding the opening. Only molten material will leave the reactor vessel.

MARCH models a gross head failure or a small hole in the bottom head with a fixed diameter.

2.3.12 Gas Flows and Temperatures inside the Reactor Vessel

When most water has left the reactor vessel there will be possibilities for thermally driven gas circulation through the main parts of the vessel. The melt away of the central part of the core will further decrease the resistance against this flow. Within the vessel, the following four phenomena interact: thermally driven gas circulation, temperature distribution, fission product behaviour and fission product heating.

The four phenomena and their interactions are modelled in MAAP. Besides the gas flows through the main parts of the primary system, the gas circulation model also includes the gas flows in one or two connections between the primary system and the containment (the hole in the failed bottom head and an optional LOCA break).

Important outputs from the models are the transient temperature distribution within the vessel and in the vessel walls and the transient distribution of the fission products.

In MARCH the flows of steam, hydrogen and fission product gases are upwards. The flows exchange heat with the lower structures, the fuel, the fuel channels and the control rods, upper structures and pressure vessel upper head. This flow stops when the reactor pressure vessel bottom head fails.

2.4 Containment System Thermal-hydraulic Models

2.4.1 Physical Regions

In MAAP and MARCH the containment is divided into the following physical regions:

MAAP

BWR MARK II: pedestal (lower drywell), drywell (upper drywell), wetwell. The concrete walls are modelled as 8 heat sinks. The internal equipment is modelled as 1 heat sink each in pedestal and drywell, 1 in wetwell water and 1 in wetwell gas. An auxiliary building model can optionally be connected to containment vent or break. (See Fig 10.)

PWR: cavity, ice condenser, ice condenser upper plenum, upper containment, annular compartment, lower compartment, quench tank. The concrete walls are modelled as 8 heat sinks. The internal equipment is modelled as 1 heat sink in upper and lower compartments. An auxiliary building model can optionally be connected to the containment vent or break. (See Fig 11.)

MARCH

PWR and BWR: Up to eight compartments can be modelled in MARCH, but the compartments are series connected except vacuum breakers and fan flows. Heat sinks are modelled as up to 15 slabs with a

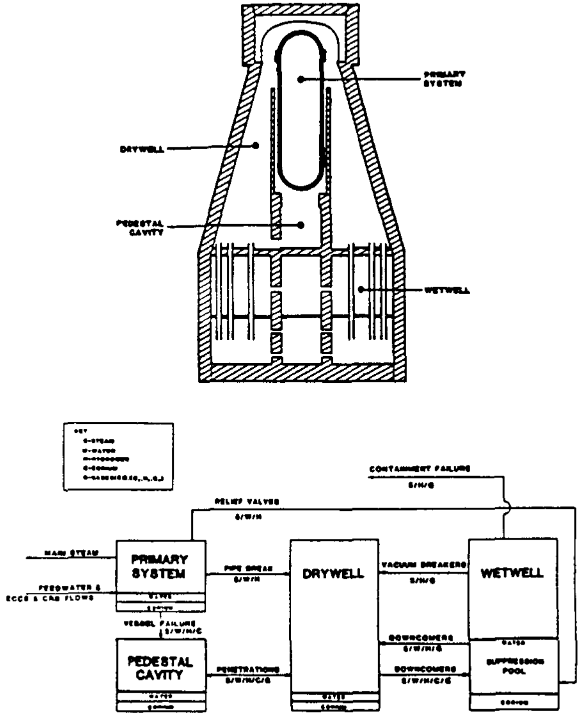


Fig. 10 : Mark II containment.

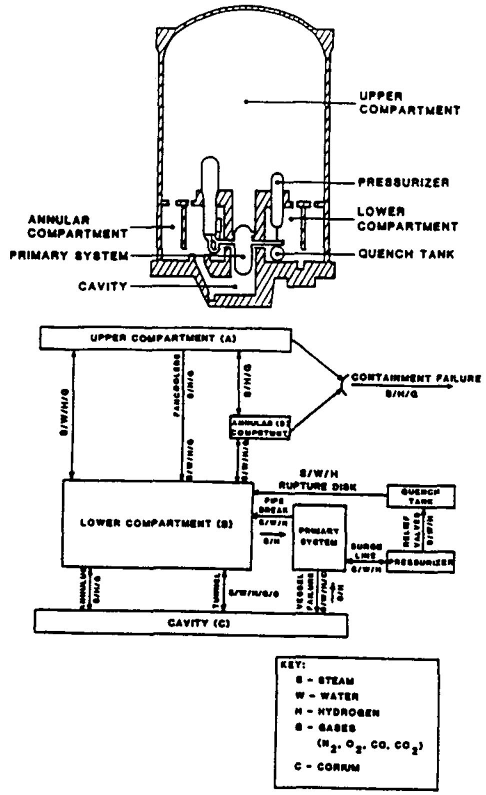


Fig. 11 large dry containment. /21/

total of 200 nodes. The slabs can contain max 5 different materials. The equipment in the compartments can only be modelled as slabs. (See Fig 3.)

2.4.2 Quenching and Long Term Cooling of Corium in Containment

The MAAP/BWR for MARK II includes a model which calculates steam production when the molten particles drop into sub-cooled water and are quenched. The particles are assumed to be cooled to a temperature corresponding to the saturation temperature of the water. The MAAP/PWR does not include this quenching model. Instead, the melt is assumed to be cooled according to the model for long term cooling described below.

If the progression of the accident has been such that the watercovered corium has been quenched, heat transfer from corium to water will be equal to residual heat. MAAP keeps a record of the fission products which are still in the corium.

If the progression of the accident has been such that the watercovered melt is not quenched, the heat flux between the melt and the water is calculated in MAAP based on a critical heat flux correlation. The melt is assumed to be evenly spread on the available area and the heat trans- fer area is equal to that area.

In MARCH, steam production in the reactor cavity is calculated. Debris bed material includes all the uraniumdioxide from the core, cladding, flow boxes and control rods, the steel structures below the core, the melted part of the bottom head and a user-specified fraction of the non-melted part of the bottom head and a user-specified part of the structures below the pressure vessel. This material form spheres, i.e. UO_2-Zr-ZrO_2 spheres and steel spheres the diameters of which are user-specified. The two kinds of spheres are supposed to have the same temperatures. The models are the same as are used in the reactor pressure vessel bottom head (see 2.3.7) for the debris bed.

In MARCH, the flooding of the reactor cavity can not be modelled during the transient but the mass of water in the cavity and the temperature of the water can be input as an initial condition.

2.4.3 Metal-Water Reactions in Containment

In MAAP/PWR, the hydrogen-producing reaction between hot Zr and H_2O in the cavity is modelled. MAAP/BWR does not consider any metal-water reaction in the containment (except for the reactions in connection with the corium-concrete interaction, see below).

Hydrogen production and the energy released in the metal-water-reaction in the reactor cavity is calculated in MARCH. The assumed components in the debris bed, the geometrical form and the temperature assumptions are described above in 2.4.2. The models are the same as are used in the reactor pressure vessel bottom head (see 2.3.7) for the debris bed. The available steam is first used to oxidize the UO_2-Zr-ZrO_2 particles and the rest is used to oxidize the steel spheres.

2.4.4 Corium-Concrete Interaction

The reaction between the not quenched corium and the concrete in the cavity of a PWR and in the wetwell/lower drywell of a BWR is modelled in MAAP. Modelling includes the heat transfer from debris to concrete and to containment or water pool above, heat transfer within the concrete, concrete decomposition and ablation, debris crusting and solidification, production of H_2, CO and CO_2 due to concrete ablation and chemical reactions between H_2O/CO_2 from the concrete and Zr and Fe in the debris. Downward and sideward heat transfer from debris to concrete is assumed equal.

The core-concrete-interaction part of MARCH, CORCON, will be used if the debris bed particles in the reactor cavity are not cooled or directly by setting a flag in the input. The concrete ablation rate (horizontal and vertical) is calculated and the amount of gases, i.e. CO, CO_2, H_2O and H_2 released by the decomposition of the concrete is calculated. CORCON treats both free and chemically bound water.

The released gases contributes to the pressure build-up in the containment. The reactor cavity can be a cylinder with a flat or a hemispherical bottom or it may have a freely chosen geometry.

CORCON models heavy oxide, metal, light oxide and water pool layers and chemical reactions between 25 different oxides, 6 metals, 21 different gases and some other chemical species.

2.4.5 Hydrogen and Carbonmonoxide Combustion, Steam Explosions

MAAP includes two models for hydrogen and carbonmonoxide combustion, a global burn model and an incomplete burn (igniter) model. Lean and upper flammability limits are modelled. Detonation is not modelled.

MAAP also includes a model for steam explosions in the compartment below the reactor vessel (according to the MAAP Manual may such explosions take place in this compartment. Since only a limited amount of materials can paticipate, no serious damage will occur to the containment).

In MARCH, the burning of hydrogen and carbonmonoxide can be calculated. The burning is started with igniters or for given contents of carbonmonoxide, hydrogen and oxygen.

In MARCH, a steam explosion which fails the pressure vessel and the containment can be initiated by a core slump.

2.5 Fission Product Release Models

2.5.1 MAAP Models

The release of materials from the core is governed by two fundamental physical mechanisms. The first determines the release rate from the fuel matrix. The MAAP 3.0 code has two options for this release mechanism: The steam oxidation model of Cubicciotti and the NUREG-0772 correlations. The model of Cubicciotti is based on the fact that the rate of UO_2 grain growth is high in a steam atmosphere. Subsequent grain boundary sweeping releases fission products. The

Cubicciotti model assumes that the rate of release is proportional to the rate of fuel sintering in steam. The NUREG-0772 correlations are temperature dependent exponential expressions fitted to observed release rates. The empirical data used for the fitting is from the ORNL and KfK fission product release tests.

Fission products in MAAP 3.0 are characterized by 6 groups:

1. Xe, Kr
2. CsI
3. TeO_2, TeH
4. Sr, SrOH, $Sr(OH)_2$, SrO
5. Ru
6. CsOH

Groups 1, 2, 3 and 6 compose the "volatile" fission products. Their release from the core is limited by their release rate from the fuel. If the Zr cladding is less than 90 % oxidized Te will chemically bind with the cladding and will not behave as a volatile fission product. The tellurides formed are then released in the core-concrete interaction phase of the accident.

The second mechanism limiting the release of fission products from the core is the ability of the flow to carry the material to the upper plenum. This mechanism limits the release of the nonvolatiles. The flow has the ability to carry at most the saturated vapor density of the fission products and structural materials and any entrained aerosols. The flow model for the nonvolatile fission products and structural materials is such as to allow for release or condensation within the core or aerosol formation at the core outlet.

2.5.2 MARCH 3/CORSOR Models

The fission products are initially distributed throughout the core with the same distribution as the power peaking factors and noding specified. For the release, the fission products are distributed into seven groups, represented by Xenon, Iodine, Cesium, Tellurium, Strontium, Ruthenium and Lanthanum. Each of these groups are further divided into metal and oxide phases. It is possible to choose between five fission product release models in MARCH 3. That is

1) small melt fission product release
2) reference WASH-1400 melt fission product release
3) large melt fission product release
4) use CORSOR model
5) use CORSOR-M model.

The three first possibilities correspond to small, best-estimate and large melt releases and these data sets are internally coded. The melt releases include the gap releases of fission products which have been released from the fuel pellets during normal operation and which occupy the free volume of the fuel pins. With the two last possibilities (CORSOR-M is the preferred choice), it is possible to get a more accurate and detailed release. 40 species altogether are tracked and among them cladding components (Zr, Sn), one structural component (Fe) and the UO_2-fuel and release fractions for Silver-Indium-Cadmium control rod material used in many PWRs.

The fission product release rates in CORSOR depend on the temperature and have the form A exp (BT), where A and B are constants fitted to experiments and T temperature. The only exception from this is Tellurium, which reacts with unoxidized Zircaloy in the cladding and therefore has been given an additional dependency. CORSOR has one expression of the form A exp (BT) for Tellurium early in the accident. Later when cladding oxidation reaches 70 %, that release rate is multiplied by 40, based on experimental observations, to account for the inability of oxidized zirconium being unable to retain Tellurium.

CORSOR-M has release rate expressions of the form C exp (-Q/RT) where C is a constant fitted to experiments, Q is the activation energy for the release process, R is the gas constant and T the temperature. In the CORSOR model for a release from the Ag-In-Cd control rods it is assumed that the rods fail at 1673 K, with an initial fractional release. The release proceeds as the temperature of the control rods rises and the release is total at 3073 K. In

CORSOR-M this model for a release from the Ag-In-Cd rods has been changed by multiplying the releases of Silver and Indium by 0.1, and the releases of Cadmium by 0.7. These changes give a better fit to experimental observations. The CORSOR-M alternative is the preferred option.

2.6 Fission Product Behaviour Models

2.6.1 Fission Product Behaviour Models

The fission product behaviour is treated in the STCP by the codes TRAP-MELT3 and NAUA. These codes are mechanistic aerosol transport codes. TRAP-MELT3 considers the reactor coolant system and NAUA considers the containment. Both codes represent the size distribution by a number of size classes and model deposition, agglomeration and condensation.

In contrast, the aerosol part of MAAP is based on correlations obtained by nondimensionalization and by calibration both with experiments and with calculations performed with mechanistic codes.

The aerosol of an LWR accident may be very dense. In the containment peak density may be higher than 20 g/m^3 and in the reactor pressure vessel peak density may be of the order of 100 g/m^3.

High number density makes agglomeration a dominating process and is necessary for the success of the aerosol calculation scheme of MAAP. In the high-density aerosols agglomeration will produce heavy particles which will sediment fast. Thus, the decay rate of a high-density aerosol is high.

An important characteristic of LWR containment aerosols is the presence of saturated or near-saturated steam. This has a number of consequences:

- owing to surface tension, vapour pressure on particle surface depends on particle radius (Kelvin effect). Therefore, particles below a critical size are dry and do not grow by steam condensation

- particles are nearly spherical, possibly because of surface tension. Therefore, shape factors can be set equal to 1 in calculations, which reduces uncertainty

- particles (above the critical size) become heavier owing to steam condensation. This generally enhances removal by sedimentation

- steam condensing on the walls gives rise to an additional removal mechanism, diffusiophoresis.

TRAP-MELT3 includes thermohydraulic calculations in a 1-dimensional model. A number of aerosol processes are considered:

- condensation of fission product vapours onto existing particles is described by a (too simple) diffusion model

- agglomeration of aerosol particles takes place by gravity, diffusion and turbulence

- deposition is modelled for gravity, diffusion, thermophoresis, and the effects of carrier gas flow and inertia.

TRAP-MELT3 treats also fission product vapour chemisorption on stainless steel. The model overpredicts chemisorption of CsOH at the Marviken experiments.

NAUA assumes a homogeneous distribution of the aerosol particles in the containment. The aerosol mechanisms considered are:

- condensation of steam onto aerosol particles
- agglomeration due to gravity and diffusion
- deposition due to gravity, diffusion and diffusiophoresis.

Steam condensation on particles is very sensitive to the thermal-hydraulic input (in agreement with experiment). Containment spray is modelled in STCP-NAUA.

MAAP applies empirical correlations for quantities such as suspended mass concentration, particle size, and time which are nondimensionalized by scaling. The correlations are derived from experiments and from model calculations with more detailed codes. The basis is the assumption that for a dense aerosol agglomeration will soon transform the initial size distribution into a size distribution characteristic of the suspended mass concentration. Different sets of correlations for the various mechanisms apply for the two cases, continuous source and aging aerosol.

2.6.2 Detailed Description of TRAPMELT

In the MARCH calculation made before the TRAP-MELT3 calculation two result-data-files are made as a start for the TRAP-MELT3 calculations /23/. These files are the MERGE-file and the TRAP-file which contain thermalhydraulic data and fission product release rates as a function of time.

The thermal-hydraulic data are as follows: max. and average core temperature, steam saturation temperature, exit core gas temperature, primary system pressure, exit core steam and hydrogen flows, MARCH time step length, fraction of core melted, saturation enthalpy and radiation from core upwards.

TRAP-MELT3 treats CsOH, CsI, Noble Gases, Te, Ba, Sr, Ru, La, Ce and structural material separately. The volatile species which are CsI, CsOH and Te can both exist as vapors, liquids and solids. I_2 can also be treated but not at same as CsI. The rest of the materials are at present treated either as gases (the noble gases) or as solids and no chemical interactions between any of these materials are considered. In the calculations made for AKTI-130, it was assumed that iodine was present as CsI and the rest of Cs as CsOH.

The code determines whether the volatile species appear as vapor or particles and it calculates the transport and retention of all the different species in the different parts of the system considering the chemisorption of vapors, vapor condensation and particle deposition on structure surfaces. At present, data on chemisorption is available in the code only for CsOH and Te.

Revaporation of particles from structure surfaces can be included for the volatile species but only for the core region (vol. 1). Chemisorption is considered an irreversible process.

The reactor coolant system should be divided into several so called control volumes for calculations with TRAP-MELT3. For the control volumes, the starting gas temperature, gas volume, volume length (in flow direction) and volume height are specified. For the associated structures, structure thickness, stucture heat, equivalent diameter, flow area and structure surface area are specified. (See Fig. 4)

Multiple choices can be made for chemisorption/no chemisorption, particle deposition/no deposition, fall back to previous volume/no fallback, vapor condensation/no condensation, coagulation/no coagulation and revaporization of settled particles/no revaporization.

Apart from a printed output, files are written for further calculations with NAUA/SPARC/ICEDF. The end time of TRAPMELT calculation is the time when the pressure vessel fails. Thus revaporization during the later phases of the accident is not taken into account. This is a clear difference compared to the MAAP calculations in which revaporization has an important role in source term formation.

2.6.3 Fission Product Transport in the MAAP 3.0 Code

Fission products, exluding noble gases, are modelled as aerosol particles in the MAAP code. The behaviour of aerosols is described with special correlations. Same correlations are used in primary system and containment conditions.

In the primary system, the aerosol correlations in MAAP have been compared with a mechanistic aerosol model called RAFT for the Loviisa power plant. According to RAFT, the chemical compounds used in MAAP seem to be sound. However, when RAFT and MAAP results are compared, the location of deposited material is different. According to RAFT, the location and amount of deposited material depends much on the thermal-hydraulic conditions while, according to MAAP results, it does not. Based on these results, it is recommended to compare the MAAP results with experimental measurements for aerosol deposition in primary system conditions. This is important because the revaporization process depends strongly on the location of deposited material.

MAAP correlations have been compared with the LACE (LWR Aerosol Containment Experiments) test results for aerosol behaviour in containment conditions. The results indicate that under these conditions the model used to describe the behaviour of hygroscopic aerosols in MAAP gives too rapid an aerosol removal by sedimentation. Thus, it is recommended to limit the maximum relative humidity used in this model to 99 % instead of 100 %.

When the MAAP results for Loviisa large and small break LOCAs were compared with the NAUA results, large differences were found in upper compartment airborne mass concentration. According to MAAP almost all airborne mass in the containment is removed after few hours. The reason for this is still uncertain, but the code developer (FAI) is looking after the problem.

It was also found that in the primary system MAAP has a lower limit for the airborne aerosol mass. According to MAAP, there is always airborne fission products in the primary system and in some accident sequences the source term is due only to the artificial airborne aerosol transport from the primary system to the environment.

3 VALIDATION AND VERIFICATION OF CODE SYSTEMS

In the following validation and verification of the code systems are presented. To clarify the terms the following definitios are used:

Validation: A stringent test of reasonableness by a quantitative comparison of code results with directly applicable experiments.

Verification: A test of reasonableness by quantitative comparison of code results with hand calculations, comparison of code results with results from other codes, line-by-line evaluation and comments on coding.

3.1 Validation and Verification of MAAP 3

Although the ABB-ATOM type BWR reactors and Loviisa have some outstanding differences in geometry and control logic as compared to commercial BWRs and PWRs in the USA, the MAAP phenomenology models in these models are in general the same. Therefore, the validation and verification program for the MAAP-models of American reactors is considered applicable to these MAAP-models for Nordic reactor types.

In addition to the U.S. validation and verification work, Nordic verification work was necessary to evaluate the applicability of different code versions on Nordic power plant types. In the NKA-AKTI-130 project the effect of geometry and control logic differences on general plant behaviour during severe accident sequences was tested. Also, the adequacy and applicability of the special models for Nordic reactors were evaluated. Benchmark calculations (with STCP codes) were performed. These calculations and special workshops concerning thermal-hydraulic and aerosol behaviour clarified the different modeling approaches for severe accident simulations. Sensitivity analyses were performed for studying important code input and model parameters. No stand-alone testing of the MAAP subroutines was performed. The evaluation of the MAAP fission product transport and aerosol modeling was performed in the NKA-AKTI-160 project.

Electric Power Institute (EPRI) in the United States sponsors the Independent Verification and Validation (V&V) of the MAAP 3.0 PWR and BWR codes. The contractor is the consultant firm JAYCOR from San Diego, California. JAYCOR defined verification as: establishment of a process to assure that Fortran coding is free from bugs and errors and represents the appropriate equations in the users manual. Validation was defined as: the checking of models against physical experiments or more detailed codes to confirm results. The MAAP subroutines were prioritized for V&V evaluation. Since MAAP models a large number of phenomenology the subroutines must be removed to model the small scale experiments available. This means that some coding changes were required to run the routines as stand alone models. The objectives of the program were to provide a systematic procedure for evaluating the importance of various subroutines. The purpose of the program was to provide a detailed verification process which offers assurance of the coding being a true reflection of the documented models (described in the user's manual) and that the numerical methods and logic are valid. In this effort the coding was translated into easily understandable and legible algebraic expressions and English text (a detailed line-by line review of the coding of important subroutines). The final report on the V&V effort will probably be issued in 1989. It was also seen important to provide a procedure for maintaining the verification documentation as coding is modified. Archived versions of the code are periodically released to MAAP-users. A new archived version of MAAP is considered as one which improves upon the current model.

Structured sensitivity analyses using the MAAP code were performed by Science Application International Corporation (SAIC). The project was sponsored by EPRI and reported by SAIC in 1987. In these analyses the fission products released to the environment were calculated for two risk dominant accident sequences: a small break LOCA with a failure of recirculation and sprays (S_2HF) at Sequoyah and an anticipated transient without scram (TC) at the Peach Bottom Plant. The sequoyah plant is a Westinghousetype pressurized water reactor with an ice condenser containment and Peach Bottom a General Electric boiling water reactor with a Mark I -type pressure suppression containment. The sensitivity evaluation considered three separate categories: 1) Time step and spatial nodalization sensitivities, 2) sequence description sensitivity variations primarily caused by differences in system behaviour and operator actions and 3) sensitivity variations in phenomenological parameters.

EPRI financed the benchmarking of MAAP against transients and thermal-hydraulic experiments and is planning to continue the work in co-operation with the MAAP Users' Group. MAAP has been compared to the TMI-accident, the LOFT-FP-2 test and the Browns Ferry incident. LOFT is used to benchmark Westinghouse PWRs and it was chosen instead of Semi-Scale because it is larger (volume/surface ratio much different than real plant in Semi-Scale facility). In general the test results from the large integral test facilities are planned to be used in the code benchmarking activity. There are plans to use data from the BETHSY-test facility to benchmark Combustion Engineering PWRs, data from the FIST-facility to benchmark General Electric BWRs and the MIST-facility data to benchmark Babcock&Wilcox Once-Through Steam Generator PWRs. In addition, the MAAP-code is used to simulate real plant transients. In this comparison, the calculational results are compared to well-documented plant transient data. MAAP, for example was used to simulate long term transients like the loss of residual heat removal event during the first refueling outage for Unit 2 of the Diablo Canyon nuclear power plant. In this case the MAAP code was viewed as complementary to RETRAN-code. A MAAP-code design review is performed by special review group. The work of the group is still in progress. The following description of important phenomena modeled in the MAAP code and the respective verification and validation of data sources is based on the list of references to model benchmarks used in the IDCOR program compiled for this MAAP design review.

Core heatup and clad oxidation

Severe Fuel Damage Tests (SFD) in the Power Burst Facility (PBF) have been simulated using the MAAP-code. Also experiment LP-FP-2 in the LOFT-facility has been calculated. The MAAP-code has been applied to TMI-2 accident simulation. In the IDCOR-project, more detailed core heatup codes for both BWRs and PWRs were developed. The MAAP-code results have been compared succesfully to the results from these codes.

Fisson product release

In addition to PBF-SFD, LOFT FP-2 and TMI-2, data from ORNL Fission Product Release Tests and German SASCHA experiments have been used in modeling.

Aerosol transport and deposition

As references to the aerosol transport and deposition models ABCOVE tests, ORNL (NSPP) tests, AI tests, Marviken tests, JAERI (Japan) tests, CEA (France) tests, CSE tests, EPRI tests, DEMONA tests, Gillispie and Langsworth experiments are listed.

Hydrogen combustion

The MAAP hydrogen combustion models are based on the information from Westinghouse flame temperature criteria tests, Whiteshell tests, EPRI tests, EPRI/FMC model, SNL VGES tests and EPRI Nevada tests.

Debris fragmentation

The choice of the MAAP debris fragmentation assumptions is based on SNL FITS tests, ISPRA tests, Higgin's experiments and ALCOA tests.

Debris dispersal and coolability

In some containment geometries the dispersal of corium may be significant. The parameters needed in the MAAP calculations are based on the ANL experiments. Debris coolability and important phenomena related to it are modeled in MAAP. The modeling assumptions are based on the 1st SNL steel-concrete experiment, U.K. experiments, EPRI experiments, ANL particle bed experiments, ANL CWTI tests, ANL SHOTDROP tests and SNL SWISS experiments.

Core-concrete interaction

Information used in core-concrete modeling is from the first SNL steel-conrete experiment, WECHSL analysis, SNL SWISS experiment and BETA experiments.

Primary system thermal hydraulics

The decay heat calculation is based on the ANSI standard. For MAAP thermal-hydraulics modelling assessment, the results from RELAP5 calculations for Seabrook Station have been utilized. The Brown Ferry fire transient and Oyster Creek loss of feedwater transient have been simulated using the MAAP code. Also the Semiscale MOD-2C SBLOCA test has been calculated using the MAAP-DOE version. The primary system natural circulation modeling is based on the Westinghouse experiments.

Containment natural circulation

Modeling of buoyantly driven flows through openings in partitions and flows in multicompartment enclosures is based on data from HEDL helium mixing experiments and FAI brine-water experiments.

Containment strain

The MAAP-code includes a simple containment strain failure model. In the development of this model, data from the SNL experiments and analysis were used.

3.2 Validation and Verification of Source Term Code Package

The Source Term Code Package, STCP has been extensively used and the single codes in the package have undergone continuous evaluation by several laboratories in the United States during the last ten years. Also, code validation and verification has been made in the United States, in the United Kingdom and other European countries. A lot of non-confidential references can be found in literature.

An STCP validation report will be written later by Batelle Columbus Laboratories, USA in connection with the documentation of the Source Term Code Package, Mod 2.0.

The Source Term Code Package has been used by Batelle Columbus for BWR Mark Design, BWR Mark I Design and PWRs with a subatmospheric containment, a large dry containment and with an ice condenser containment for several accident sequences. These calculations and their validity are discussed in the reports NUREG/CR-4624, 5062 and 4696.

As a verification and testing of the "reasonableness" of these calculations, hand-calculations were performed on the TB-sequence of the Peach Bottom BWR plant. The TB-sequence was chosen because it is a major contributor to the risk profile and because most of the Source Term Code Package is excercised by calculating this sequence. This is reported in NUREG/CR-4656. In the following, some of the major findings will be discussed.

The report states that hand calculations have some limitations when used for complex and coupled processes such as a core meltdown accident but anyhow, they are a strong indicator if anything in the code is wrong, e.g. if the basic conservation laws are not satisfied. The report lists some important phenomena which are not modeled in the Source Term Code Package and which may be of importance. These phenomena are as follows:

1) in-vessel circulation flow
2) direct heating (and fission product release) from core material ejected at high pressure
3) revolatilization of fission products from reactor coolant system surfaces
4) natural convection in multi-compartment containments
5) chemical interactions of fission product vapors with aerosols
6) changes in the chemical form of fission products in the reactor coolant system.

In general, the hand calculations show what they were intended to show. The STCP codes obey the basic conservation laws and produce reasonable results but the degree of agreement and significance of the comparison differ among the models evaluated.

The comparisons made with the MARCH 3 thermal-hydraulic models are typically in quite a good agreement. However, the MARCH 3 models are themselves typically simple, intuitive models based on the application of conservation laws. The hand calculations are in many cases very similar to the calculations occurring in the code. The value of the code is in that it enables a large number of processes to be analyzed with consistent interfaces and the coupling between the processes to be taken into account.

CORCON has a consistent theoretical framework but relies, like MARCH, on some intuitive assumptions about the physical behaviour (e.g. the separation of the melt into layers). By hand calculations it has been possible to show that as melt attacks concrete, energy is conserved. However, only experiments can demonstrate the validity of the modeling assumptions.

TRAPMELT analyses very complex processes. Simple hand calculations are not capable of adressing some aspects of the analysis such as interacting and competing mechanisms very well. Although some of the processes modeled by NAUA are very similar to those in TRAPMELT, hand calculations can be used more effectively in testing NAUA. In TRAPMELT the interaction between vapors and aerosols and the sensitivity to flow and temperature make it difficult to isolate phenomena. The NAUA calculations on the other hand, deal only with aerosol transport. Since the physics of aerosol behaviour have been well studied, it was possible to find closed form solutions or approximations to the basic aerosol equations which were valid for phases of the accident. As a result, the hand calculations for the NAUA code not only provide evidence that the results are reasonable, but also that they are correct.

The greatest discrepancies observed in this exercise were in comparisons with the VANESA code. The checking calculations went beyond the boundaries of "hand" calculations as a computer code was used to solve the chemical equilibrium problem for the mixture of species in the melt. But anyhow the degree of agreement was limited. Variations in the input data base showed that the vapor pressure of species and their related release from the melt are quite sensitive to the CORCON thermal-hydraulics and chemistry of the molten core. The processes involved are complex and intimately coupled. The release rate of fission products during core-concrete attack is already recognized as an area of major uncertainty. The comparison calculations in this study support this recognition.

As a sort of validation, the Three Mile Island accident gives some qualitative insight on core behaviour during the initial stages of a core meltdown accident, although the core naturally was not instrumented to measure all the core meltdown parameters of interest. E.g. the actual thermal response is unknown and much of the information needed to perform a definitive calculation of the heatup is not available. The most important information lacking is knowledge of amount of water added and removed from the vessel by the various ECC makeup and letdown systems in operation during the transient. Thus, the water level in the core is not well known.

Some knowledge about the accident is based on direct observations. The top-central 30 percent of the core is missing. About a half of the missing core is in the lower head and appears to have been previously molten. The composition of the previously molten material indicates maximum melting temperature of about 3000 K. Other parts of the missing core are rubblized and displaced downward. The rubble is supported on the ends of the remaining fuel rod stubs by a hard crust area about 2 inches thick. The amount of hydrogen generated in the transient is consistent with about 50 percent cladding reaction. The core was recovered with water in about 100 minutes after the initial uncovering. Core heatup was thus stopped.

The amount of core melting in the MARCH 3 calculation on Peach Bottom TB-sequence is comparable to the amount of core melting believed to have occurred in the TMI accident at two hours after the core uncover. The MARCH 3 meltdown model assumes the retention of all the molten material in the beginning. The TMI core shows crust formation below the molten region but also shows a substantial fraction of the core in the lower head. The nominal melting temperature of 2550 K used as default melting temperature in MARCH 3 is significantly lower than the maximum melting temperature of 3000 K inferred from the TMI data.

In MARCH 3, break out and slumping into the lower head begins as the molten region melts out the last remaining solid core material in the bottom central region of the core. TMI experience appears to indicate that the break out of a molten fuel region can occur even with the core partially covered. In the MARCH 3 calculation, the presence of the molten core in the lower head quickly fails the fuel element supports, and consequently, also the remaining non-molten core collapses into the lower head. The beginning of slumping into the water below the core generates steam which feeds the metal-water reaction. This metal-water reaction stops when the whole core collapses into the water and the molten material is cooled.

4 BENCHMARK CALCULATIONS PERFORMED

A code comparison was made by performing benchmark calculations for a Nordic BWR-plant (Forsmark 3) and a Nordic PWR plant furnished with an ice condenser (Loviisa). Two accident sequences, namely a "total loss of AC-power" and a "LOCA and loss of AC-power" were studied.

Some of the presented mitigation measures were taken into account in the analyses; these are depressurization of the primary system and flooding of the reactor cavity with water. Containment venting was assumed at about the design pressure instead of the assumption of a containment break.

The main items compared in the benchmark calculations were core melt behaviour, hydrogen production, thermal hydraulic conditions in the primary system and in the containment, time behaviour of the accident sequences and aerosol transport.

The idea in the benchmark calculations was to select similar basic assumptions and to analyse the differences in the results. These basic calculations were completed with sensitivity calculations by varying closely related parameters (see chapter 5).

4.1 Benchmark Calculations of a Nordic BWR Plant

The main results of the benchmark calculations for a BWR (Forsmark 3) are shown in Table I. The table includes times to certain events, masses of hydrogen produced, mass of fuel left in the core for a long time and presence of core-concrete interaction.

Only the MAAP code can model the reactor control system and calculate the time when the automatic depressurization system (ADS) starts working. In MARCH the time for this to happen is an input.

The MAAP and MARCH results show relatively big differences in the calculated times to vessel melt-through. The MARCH code predicts a faster core melt. The reason for this is the assumption that the

TABLE 1:
Comparison between results from MAAP3.0 and MARCH3 calculations for BWR (Forsmark 3) and PWR (Loviisa)*

	BWR Loss of AC power		BWR Small LOCA + Loss of AC power		PWR LOCA + Loss of AC power		PWR Loss of AC power**	
	MAAP3.0/ rev. 0	MARCH3	MAAP3.0/ rev. 0	MARCH3	MAAP3.0/ rev. 6	MARCH3	MAAP3.0/ rev. 6	MARCH3
Loss of AC power (s)	0	0	0	0	0	0	0	0
LOCA (s)	-	-	0	0	0	0	-	-
ADS (s)	710	710	740	740	-	-	-	-
Core uncovered (s)	830	790	920	610	1 660	880	22 900	20 200
Start of core melt (s)	3 300	3 900	2 900	2 480	2 770	3 240	34 300	27 700
Core slump (s)	5 890	4 600	6 470	3 660	4 390	4 050	37 000	28 600
Vessel melt-through (s)	5 950	4 660	6 530	3 710	10 300	9 160	37 100	44 700
Ice depleted in ice condenser (s)	-	-	-	-	46 000	18 500	64 100	65 200
Containment vent (s)	73 000	62 500	64 000	55 300	69 000	26 700	97 600	75 200
Hydrogen production:								
- in core (kg)	160	310	170	530	50	240	60	300
- in bottom head (kg)	-	20	-	15	-	10	0	10
- in the reactor		0						
cavity, Zr (kg)	-	-	-	-	-	20	0	20
Fe (kg)	-	-	-	-	-	380	0	350
Mass of fuel left in core region (kg)	36 000	0	38 000	0	0	0	0	0
Core-concrete interaction	No	No	No	No	Yes***	No	Yes***	No

* No bypass of condensation pool or ice condenser nor containment leakage are assumed. Venting is started at 0.7 MPa in BWR and 0.17 MPa in PWR via line of 0.15 m in diameter. ** Stuck open pressurizer safety valve after first opening is assumed. *** Erosion depth is about 0.001 m.

melting of the core does not block the flow of steam through the core. MARCH predicts that the fuel channels will melt before the fuel and thus channel blocking is very unlikely. Compared to the MAAP results, which are based on the assumption that the melt will block the steam flow, this leads to a more intense reaction between Zr and H_2O resulting in more heat from the reaction and also a higher generation of, H_2. In the MAAP calculations H_2-production was about 12 % of Zr in the fuel cladding reacted in the core while in the MARCH calculations the corresponding reaction rate was 23 % in the TB case and 39 % in the S_2B case. The higher H_2 production in the MARCH cases also shortens the time to containment vent.

Besides the above mentioned blocking, there are also other significant differences between the MAAP and MARCH modellings of the core overheating and melting process:

- The MAAP modelling of the core degradation is such that when the lowest node in one of the core columns melts completely, all the melted material in that column moves through the "core plate" which successively heats up, melts and drops into the lower plenum. During the continuation of the melting after the core plate failure, the melted material drops into the lower plenum. A similar assumption in the MARCH code has been made in the performed calculations instead of the assumption of a core slump when e.g. 75 % of the core has melted.

- The radial heat losses from the core are modelled both in MAAP and in MARCH. Due to the modelling of the radial heat losses from the core in the MAAP code, an outer ring of fuel will remain unmelted in most cases. The MARCH code models the heating and melting of the structure supporting the core. When this structure reaches its melting temperature the whole core is assumed to slump. At this time only the central part of the core has melted. A total of about 50 % of the core has melted before the core slumps.

In the calculations presented, it has been assumed for both MAAP and MARCH that the vessel melt-through takes place at one of the penetrations for the control rods. Both codes give a melt through of the pressure vessel one minute after the core has slumped. If it is assumed alternatively that the whole thickness of the lower head should be melted-through, the time to vessel failure will increase. The performed MARCH calculations have indicated that this assumption in combination with larger amounts of water in the lower plenum may sometimes delay the melt-through from one to several hours. The water will cool down the core debris and only after all water has boiled away the debris will reheat and fail the vessel.

In the MARCH results, the gas temperatures in the upper plenum are about 600 K but just before the core slumps when the hydrogen production is high the temperature rises about 100 K. Similar temperatures are calculated by the MAAP code at the same point. However, because in the MAAP code a part of the fuel is left in the core region the calculated mean gas temperature in the primary system increases gradually and reaches a maximum of 1100 K at the time of the containment failure in the TB case. Then it decreases slowly. For the S_2B case, the history of this temperature is similar but the max value is about 100 K lower.

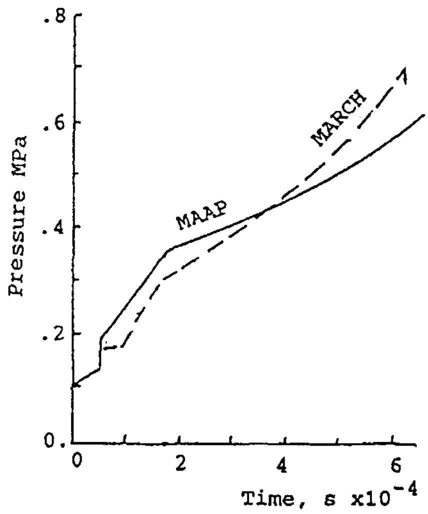


Fig 12 Forsmark 3.
Loss of AC power.
Drywell pressure.

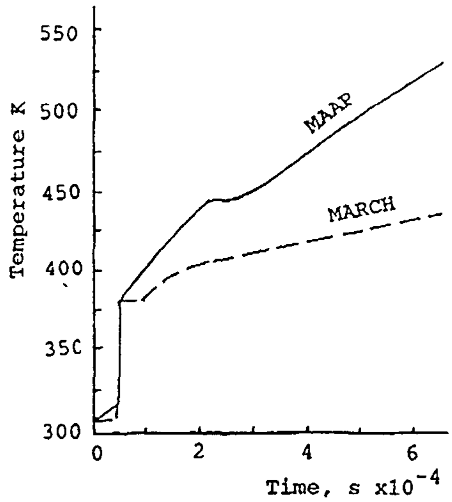


Fig 13 Forsmark 3.
Loss of AC power.
Drywell gas temp.

Figures 12 and 13 present comparisons of calculated pressure buildups and temperatures in the containment. The reason for the faster long-term pressure increase in the MARCH results is two-fold: a higher H_2 production and a higher steam production from the water in the lower drywell (in the MARCH cases the whole core is at this time located in this water). In the MARCH calculation it is assumed that the corium in the reactor cavity forms a debris bed with rather small particles (diameter 1.2 cm). It is also assumed, that the particles bear a homogenous distribution of Zr, ZrO_2 and UO_2. These assumptions are essential for the boiling rate of the water in the reactor cavity and, thus, for the time to containment venting. The predicted venting times are 18...20 h in the MAAP calculations and 15...17 h in the MARCH calculations showing a 3 h difference of time in the predictions. The reason for the higher drywell gas temperature in the MAAP cases is that the fuel remaining in the core dissipates its residual energy to the drywell atmosphere.

The TRAPMELT calculations were carried out in conjunction with the MARCH calculations to see the retention of fission products in the primary system. The materials released from the core are divided into ten groups treated separately. The primary system is divided into six volumes, that is core, mixing plenum, steam separators, steam dryers, upper dome and blow down pipe. The code calculates the transport and retention of species considering vapor sorption, vapor condensation and particle deposition on structure surfaces. Revaporization from surfaces and particles is also included. The release from the fuel is dependent on temperature and holding time at a certain temperature. As discussed above, only part of the core is melted before core slump, when the corium is quenched again and thus only part of the fission products are released from the fuel.

In the TB-sequence/S_2B-sequence the fractions of core inventory retained in fuel are 0.21/0.36 for CsI and CsOH and 0.90/0.93 for Te, respectively and the fractions deposited in the primary system are 0.53/0.19 for CsI and CsOH and 0.09/0.05 for Te. Thus 26 % of CsI and CsOH in the TB-sequence and 45 % of CsI and CsOH in the S_2B-sequence are released to the containment at the time of the pressure vessel melt-through which is the end time of the TRAPMELT calculation.

Also the MAAP code models the transport and retention of fission products. The materials released from the core are divided into six groups. The fission products are modelled to exist in four states: vapor, aerosol, deposited and contained in core or corium. The modelled modes of transfer between the states include revaporization.

In the MAAP calculations, the releases of CsI and CsOH from the primary system up to the time of vessel failure are 35 % for the TB-sequence and 45 % for the S_2B-sequence. However, at the later phase of the accident, MAAP predicts revaporization to take place from the primary system surfaces so that release fractions from the primary system to the containment are much higher than TRAPMELT predicts. In the MAAP calculations the Te has optionally been assumed to be bound up in the zircalloy in the core and not released.

4.2 Benchmark Calculations of a Nordic PWR Plant

The main results of the benchmark calculations for PWR (Loviisa 1) are shown in Table I. The table includes times to certain events and the masses of hydrogen produced and the presence of a core-concrete interaction. In the following the results of the S_1B-sequence are described.

In the MAAP3/Lo code, the specific control logic for Loviisa and the horizontal steam generators are described. There are no such modelling features in the MARCH code.

The MARCH code cannot model the LOCA blowdown precisely and to compare the results of the two codes, an approximate blowdown table from MAAP output was made for the first 750 s of the accident and used in MARCH. When MARCH calculates the blowdown from 750 s and onwards it overestimates the flow of steam out of the break in the hot leg and the time to core uncovery is shorter than MAAP calculates.

Compared to MAAP, MARCH models a rather fast melting of the core as about one third of the Zr is oxidized and this reaction contributes to the heat up of the core. The detailed core model is used in MARCH to take into consideration the assembly shroud and the bypass channel. In the MAAP calculation, hydrogen production in the core is much lower (about 6 % of Zr reacted) because the fuel channel blockage model stops the oxidation of Zr due to a lack of steam.

In the MARCH code, core slumping is initiated when 75 % of the core has melted and then the whole core, both melted and unmelted parts, falls to the bottom head together with the lower structures. Some further oxidation takes place when the core slumps to the bottom head where there is still some water left. In the MAAP code, the melted material from the core falls down onto the support plate the temperature of which starts to rise. After the melting temperature is reached, the melted core material can fall into the pressure vessel lower head. At this time there is about 60 % of fuel in the core and the rest of the fuel dropped into the lower head during and after core plate failure.

In the MARCH calculation, the gas temperatures in the upper plenum are much influenced by the hydrogen production and reach about 1000 K just before the core slumps. In the MAAP calculation, the average gas temperatures in the primary system before pressure vessel failure are in the range of 470-540 K and the maximum temperature in the upper plenum is 600 K.

The Loviisa pressure vessel has no bottom penetrations. The time and size of any failure caused by a contact with a corium is, therefore, uncertain. The MAAP code does not calculate the failure of the pressure vessel but the time and size of the failure are given as input. 60 s failure time of the pressure vessel has been suggested for input /22/. The MARCH code calculates the time from core slump to the failure of the pressure vessel, taking into account stresses in the bottom head due to the weight of the corium and the pressure in the vessel. The result of this MARCH calculation was selected as an input for the MAAP benchmark calculation for the S_1B-sequence.

In the MARCH calculations after the remaining water has boiled away the corium started to heat up the bottom head and it took about 1.5 hours after the core slump before the vessel failed. At that time the corium and the lower structures have a temperature of 2260 K. A debris bed is formed with homogeneous ZrO_2-Zr-UO_2-particles and Fe-particles. The debris bed particle size is determined through input and set to a 0.012 m diameter which is a uniform assumption with BWR-calculations. MARCH has models for the oxidation of both the Zr and Fe particles in the reactor cavity. The temperature of the particles is falling and after 300 seconds their temperature is so low that oxidation is stopped. About 8 % of the Fe is oxidized by giving 380 kg hydrogen. The partial pressure of the hydrogen reaches 0.2 bar. About two hours after the ice is depleted containment pressure reaches 0.17 MPa and the containment vent opens. The core-concrete interaction is not predicted by the MARCH code because the reactor cavity is flooded with water.

In the MAAP calculation, hydrogen is produced neither in the lower head nor in the reactor cavity although the temperature of the corium is high when the vessel fails. The reactor cavity is flooded with water above the core level at the time of the vessel melt-through and therefore, the core-concrete interaction is negligible with an erosion depth of 0.0005 m. Because the cavity is totally filled with water the calculation of the pressure transient may be a difficult task.

There is a major difference in the containment pressure evolution calculated by the two codes. In the MAAP calculation the increasing steam partial pressure after ice depletion causes the venting after 69000 s whereas, in the MARCH calculation, the venting pressure is reached much earlier, after 27000 s. There are two reasons, namely hydrogen production and the amount of concrete used as a heat sink. In the MARCH calculations, 50 % of the concrete amount and surface area used in the MAAP calculations was used.

5 PLANT SPECIFIC SENSITIVITY ANALYSES PERFORMED

Many sensitivity calculations have been performed for Nordic power plants by using the MAAP 3 and MARCH 3 codes. From these calculations experiences of the applicability of the codes have been gained and it has been also possible to compare the results from both codes with each other.

Different types of sensitivities have been studied. These are

- the use of different codes (MAAP and MARCH)
- parameters related to the description of different phenomena and models
- code specific parameters related to input data selection
- plant specific parameters related to system performance
- effects of different accident sequences
- effects of different reactor types.

The main items studied in the sensitivity calculations were core melt behaviour, thermal-hydraulic conditions in the primary system and in the containment, time behaviour of the accident and aerosol transport.

The idea in the sensitivity calculations was to vary one or a couple of closely related parameters at a time and study the effects on the accident behaviour. In the following chapters the summary of the most important parameter variations and results is presented. More detailed results appear in Tables 1...6 and in Figs 14...20.

5.1 Sensitivity Analyses for a Nordic BWR, Forsmark 3

5.1.1 MAAP 3.0 Calculations

Most of the sensitivity studies have been based on the accident sequence Loss of AC power (TB).In two cases the analyses have been based on the sequence Small LOCA, loss of AC power (S_2B). The influences from the following input parameters have been investigated:

a) TB, primary system heat losses
b) TB, steel masses in the primary system
c) TB, mass of inactive aerosol
d) TB, melting temperature of the core

TABLE 2 NKA-AKTI-130. FORSMARK 3/OSKARSHAMN 3. MAAP SENSITIVITY ANALYSIS. SUMMARY OT THERMAL/HYDRAULIC RESULTS

Independent containment spray not considered

Figures within parentheses refer to the corresponding base case

	Start of core melt, h	Core plate failure, h	Vessel failure, h	Containment vent, h	Total mass of H_2 produced, kg	Max drywell temperature,C	Mass of unmelted fuel, tons
TB, primary system heat losses reduced 30 %		1.64 (1.64)	1.67 (1.66)	27.3 (27.0)	111 (111)	290 (275)	32.5 (32.7)
TB, primary steel masses reduced 30 %	1.11 (1.11)	1.62 (1.64)	1.64 (1.66)	26.0 (27.0)	100 (111)	290 (275)	31.5 (32.7)
TB, core melting temp reduced from 2500 to 2123 K	.89 (1.11)	1.40 (1.64)	1.43 (1.66)	24.0 (27.0)	79 (111)	190 (275)	9.9 (32.7)
TB, core melting temp increased from 2500 to 3123 K	1.64 (1.11)	2.15 (1.64)	2.16 (1.66)	27.8 (27.0)	176 (111)	350 (275)	41.0 (32.7)
TB, cladding failure temp reduced from 1500 to 1100 K	1.06 (1.11)	1.62 (1.64)	1.64 (1.66)	27.0 (27.0)	97 (111)	285 (275)	32.5 (32.7)
TB, core injection/spray recovery at .6 hrs or earlier	- (1.04)	- (1.52)	- (1.54)				
TB, failure of automatic depressurization (ADS)	1.11 (1.11)	2.44 (1.64)	2.45 (1.66)	24.5 (27.0)	220 (111)	235 (275)	6.0 (32.7)
S_2B, steam line LOCA area incr from .009 to .2 m^2	.96 (.81)	1.78 (1.79)	1.81 (1.81)	16.5 (17.8)	204 (165)	303 (285)	35.7 (37.8)
S_2B, core melting temp reduced from 2500 to 2123 K	.75 (.81)	1.52 (1.79)	1.54 (1.81)	15.5 (17.8)	122 (165)	184 (285)	8.1 (37.8)

e) TB, failure temperature of the cladding

f) TB, tellurium bound up or not in the zircaloy

g) TB, spray or injection cooling of an over-heated or partly degraded core

h) TB, failure of the automatic depressurization

i) TB, time step

j) TB, method of integration

k) TB, revaporization, deposition area

l) TB, decontamination factors, particle diameters

m) S_2B, LOCA area

n) S_2B, core melting temperature.

The parameters a) to j) have been investigated within the Swedish RAMA II project and made available for NKA. The parameters k) to n) have been investigated within the NKA-AKTI-130 project. A summary of the numerical results of the sensitivity analysis is given in Tables 2 and 3.

According to the results, the most important input parameters seem to be d), g), h), m) and n).

The d), variation of the melting temperature of the core, showed that many of the output parameters are sensitive to this input. The mass of UO_2 remaining unmelted in the vessel, the temperature levels of the primary system, the drywell max temperature and the time history of the fission product distribution were all greatly affected.

The g), variation of the time for restoring of spray or injection cooling of an overheated or partly degraded core, has shown that it is not possible to quench the core if the core max temperature is about 1500 K or higher. However, the result does not mean that it is really impossible to quench the core during such conditions.(Compare the TMI-2 accident). It only means that MAAP has an uncertain and probably conservative modelling of the quenching and that calculations based on this modelling result in a failure to quench if the core is highly overheated.

The calculations for the h) case, the failure of automatic depressurization, show that this failure influences significantly the progression of the accident. The times to certain events were changed, the amount of fuel left unmelted decreased, hydrogen production increased, the release of fission products from the primary system was much slower, etc.

The variation of the LOCA area, case m), was performed for the accident sequence Steam line small LOCA. The result shows that hydrogen production increases somewhat with an increased LOCA area. In the case of the larger LOCA area, the long term primary system temperature is much lower due to the cooling by the thermally driven flow of gas from pedestal through the reactor vessel after the melt-through of the bottom head. Also the partition of CsI between the primary system, drywell and suppression pool is quite different. Due to the lower primary system temperature in the case with the larger LOCA area, no revaporization of fission products occurs in later phases.

The variation of the core melting temperature for the Steam line small LOCA sequence, case n),shows principally the same parameter sensibility as in the case d).

5.1.2 MARCH 3 Calculations

The sensitivity studies have mainly been based on the accident sequence Loss of AC power (TB).Regarding parameter c) below, an analysis has also been performed for the accident sequence Small LOCA, loss of AC power (S_2B). The sensitivity of accident progression has been investigated for the following input parameters:

a) core slump model
b) core blockage
c) debris particle diameter (in water in containment)
d) vessel failure (gross or local)
e) independent containment spray.

The parameters a) to c) have been investigated within the NKA-AKTI-130 project. The parameters d) to e) have been investigated within the Swedish RAMA III project and made available to NKA. A summary of the numerical results of the sensitivity analysis is given in Table 3.

The variation of the parameter a) above, the core slump model, showed a significant influence on the hydrogen production in the core (if no core blocking was assumed). If molten parts of the core are allowed to drop early into the water below the core this results in a steam flow through the core and to an increased Zr-H_2O reaction. An early start of the core slumping process will also increase the core melting rate (heat from the Zr-H_2O reaction) and shorten the time to a vessel melt-through.

As expected, the assumptions regarding core blocking, parameter b), influence the incore hydrogen production. If the molten core material is assumed to block the whole core, hydrogen production is reduced. Due to decreased reaction heat, also the core melting process and the vessel melt-through are delayed. The time to containment vent also increases.

For the interaction between debris and water in the containment, a particle debris bed model has been chosen. Variation of the particle diameter, c) above, has shown that this will influence the heat flow from the particles during quenching. For increasing the diameter, the quenching rate, steam production and pressure build up in the containment are delayed.

MARCH includes possibilities to model vessel failure either as a local melt-through at a penetration or as a gross failure of the whole bottom head (parameter d) above). According to calculated results, the two models give very different results regarding the time from a core slump to a vessel failure, 40 s or 2.9 h, respectively.Other important output parameters such as hydrogen production, max containment pressure, max drywell temperature, etc did not change significantly.

Forsmark 3 is equipped with an independent containment spray (parameter e) and a containment venting system connected to the drywell. According to MARCH calculations, the independent spray

TABLE 3 NKA-AKTI-130. FORSMARK 3/OSKARSHAMN 3. MARCH SENSITIVITY ANALYSIS. SUMMARY OT THERMAL/HYDRAULIC RESULTS

Figures within parentheses refer to the corresponding base case

	Start of core melt, h	Core plate failure, h	Vessel failure, h	Containment vent, h	Mass of H_2 produced, kg	Max drywell temperature,C	Mass of unmelted fuel, tons
TB, core slump delayed from 46 to 75 % of core melted[1]	1.08 (1.08)		2.09 (1.29)		195 (307) Core		0
TB, channel blockage assumed[1]	1.12 (1.08)		1.54 (1.29)		156 (307) Core		0
TB, chan. blockage + core slump delayed fr 41 to 75 %[1]	1.12 (1.08)		2.15 (1.29)		160 (307) Core		0
TB, debris particle diam incr fr .012 to 1.2 m (in cont)[1]				23.3 (17.4)			
S_2B, debris particle diam incr fr .012 to 1.2 m (in cont)[1]				21.8 (15.4)			
TB, local pres vessel failure instead of gross failure[2]	.73 (.73)	.92 (.92)	.93 (3.83)	-	680 (680) Tot	154 (152)	0
TB, failure of independent containment spray[2]	.73 (73.)	.92 (.92)	3.83 (3.83)	8.9 (-)	680 (680) Tot	155 (152)	0
TB, local pres vessel failure, failure of indep spray[2]	.73 (.73)	.92 (.92)	.93 (3.83)	8.3 (-)	680 (680) Tot	155 (152)	0

1) Independent containment spray not considered

2) Independent containment spray considered

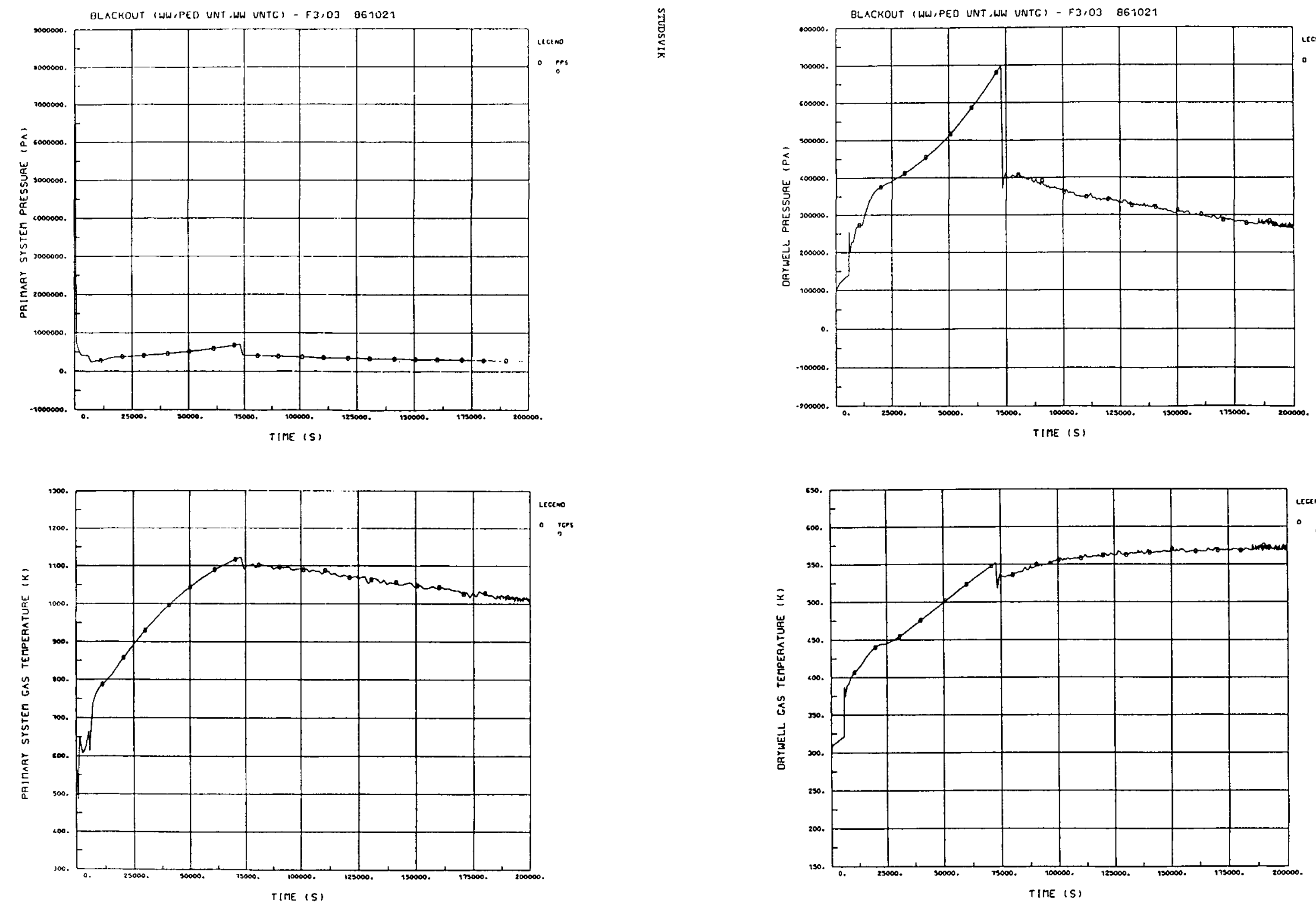
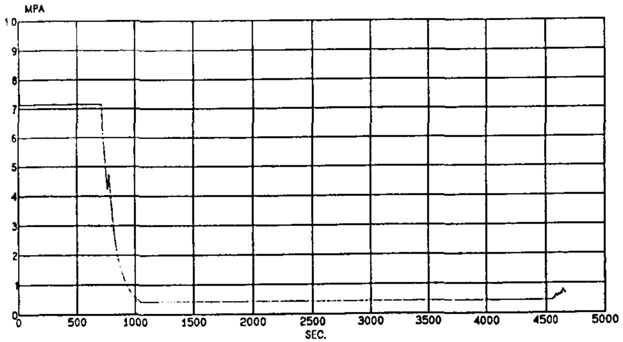


Fig. 14. MAAP 3 analysis for BWR TB sequence. (F3,O3). Some important parameters.

BWR3000 03/F3 - TB ACCIDENT (MARCH3) (186)
PRIMARY SYSTEM PRESSURE
BP6605



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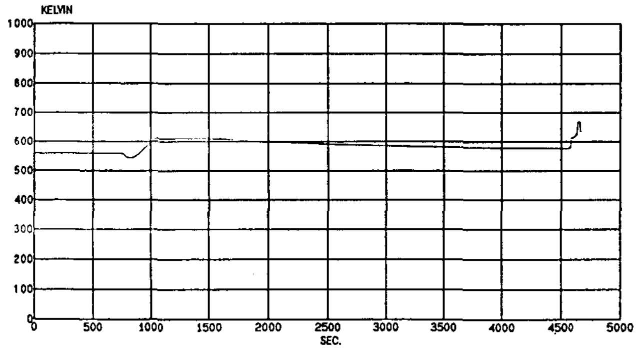
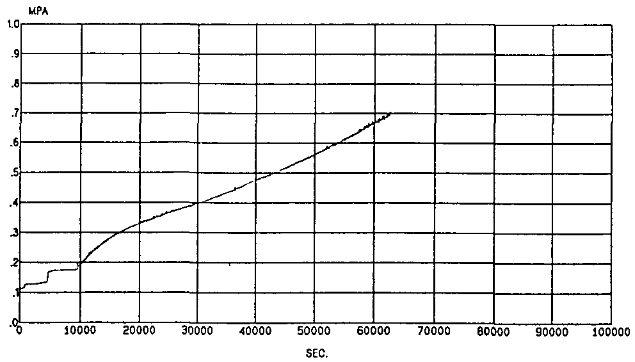
TEMPERATURE OF GASES IN PRESSURE VESSEL
BP2314



BWR3000 03/F3 - TB ACCIDENT (MARCH3) (186)
TOTAL PRESSURE IN DRYWELL
MP2314



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TEMPERATURE IN DRYWELL
MP2318

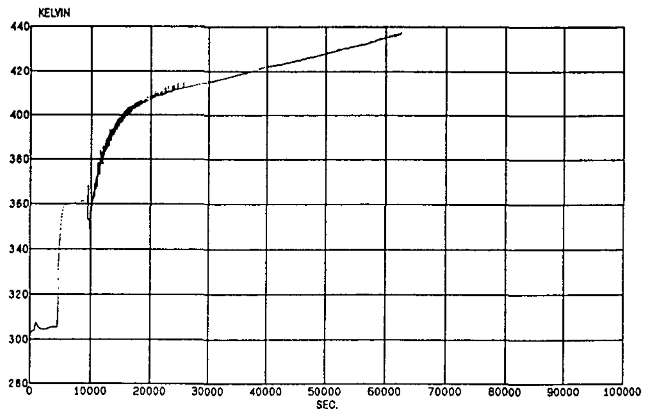


Fig. 15. MARCH 3 analysis for BWR TB sequence. (F3,O3). Some important parameters.

will in the TB sequence start before the venting pressure is reached. The spray will then reduce containment pressure and venting is not needed. As a variation of the calculation, a failure of the spray has been assumed. In this case, the containment vent will open and blow off to the vent filter. The max containment pressure and drywell max temperature will not be much different from the case with a spray. The steam coming from the boiling water in the lower drywell will keep the drywell temperature at an acceptable level.

5.2 Sensitivity Analyses for a Nordic BWR, TVO I/II

5.2.1 MAAP 3.0 Calculations

The studied accident sequences TB (total loss of AC-power) and S_2B (small steamline LOCA + loss of AC power) are rather similar. The main difference is that in the TB-case the loss of coolant is directed to the condensation pool and in the S_2B-case it is directed to the drywell gas phase. There is also a steam flow through the pressure vessel from the bottom head to the steam line break after the vessel melt-through when in the TB-case this flow is prohibited. In the LOCA -case the steam flow through the pressure vessel led to lower temperatures in the pressure vessel and thus revaporization of aerosols was smaller.

The effects of the following assumptions were studied:

a) TB, S_2B, containment break location and area
b) TB, pre-existing opening in the containment pressure boundary
c) TB, leakage between drywell and wetwell gas atmospheres
d) TB, flow from pressure vessel to suppression pool after vessel failure
e) TB, no manual depressurization of the primary system and no pedestal flooding
f) TB, S_2B, reactor building fission product retention.

The leakage area between drywell and wetwell atmosphere had a significant effect on the containment pressurization rate if the area was relatively large. In the TVO sensitivity analysis related

Table 4 . Results of the sensitivity analysis of the Finnish BWR with the MAAP 3.0 code. Important event times and other key results. Calculation time is 2.5 days.

Accident sequence**	Core uncovery s	ADS/ pedestal flood s	Vessel failure s	Start of venting h	CsI/CsOH release fraction°°	Hydrogen prod./ Mass of fuel left in core kg	Max.aver. gas temp. in prim. syst. °C	Peak DW temp pressure °C/bar
TB, wetwell venting								
1. ref. case A*°	2500	3600	8500	25	$4x10^{-3}$ / $5x10^{-3}$	90/5000	750	250/7
2. DW-WW leak 50 cm^2	"	"	"	11	$5x10^{-2}$ / $5x10^{-2}$	" /2000	700	220/7
TB, drywell venting								
3. DW-WW leak zero*	"	"	"	31	$5x10^{-2}$ / $5x10^{-2}$	" /5000	750	270/10
4. DW-WW leak zero	"	"	"	"	$7x10^{-3}$ / $1x10^{-2}$	" /7000	650	"
5. DW-WW leak 50 cm^2 *	"	"	"	12	$2x10^{-1}$ / $2x10^{-1}$	" /5000	800	220/7
TB, pre-existing opening in DW								
6. venting line	"	"	"	-	$8x10^{-2}$ / $8x10^{-2}$	" /5000	650	180/2
S_2B-sequence								
7. drywell venting	900	700/1800	5900	30	$2x10^{-2}$ / $5x10^{-2}$	" /7000	600	270/10

** In the analysis containment is assumed to be leak tight except in case 6. DW-WW-wall is leak tight except in cases 1, 2 and 5. Vent area is 0.018 m^2 except in case 1. Melt-through time of the pressure vessel is 60 s.
* Relief valves are locked closed after the vessel failure prohibiting cooling flow through the vessel
° Vetwell vent area is 0.01 m^2 and DW-WW leakage area is 5 cm^2
°° Release fraction to the venting line. Containment spray has not been taken into account

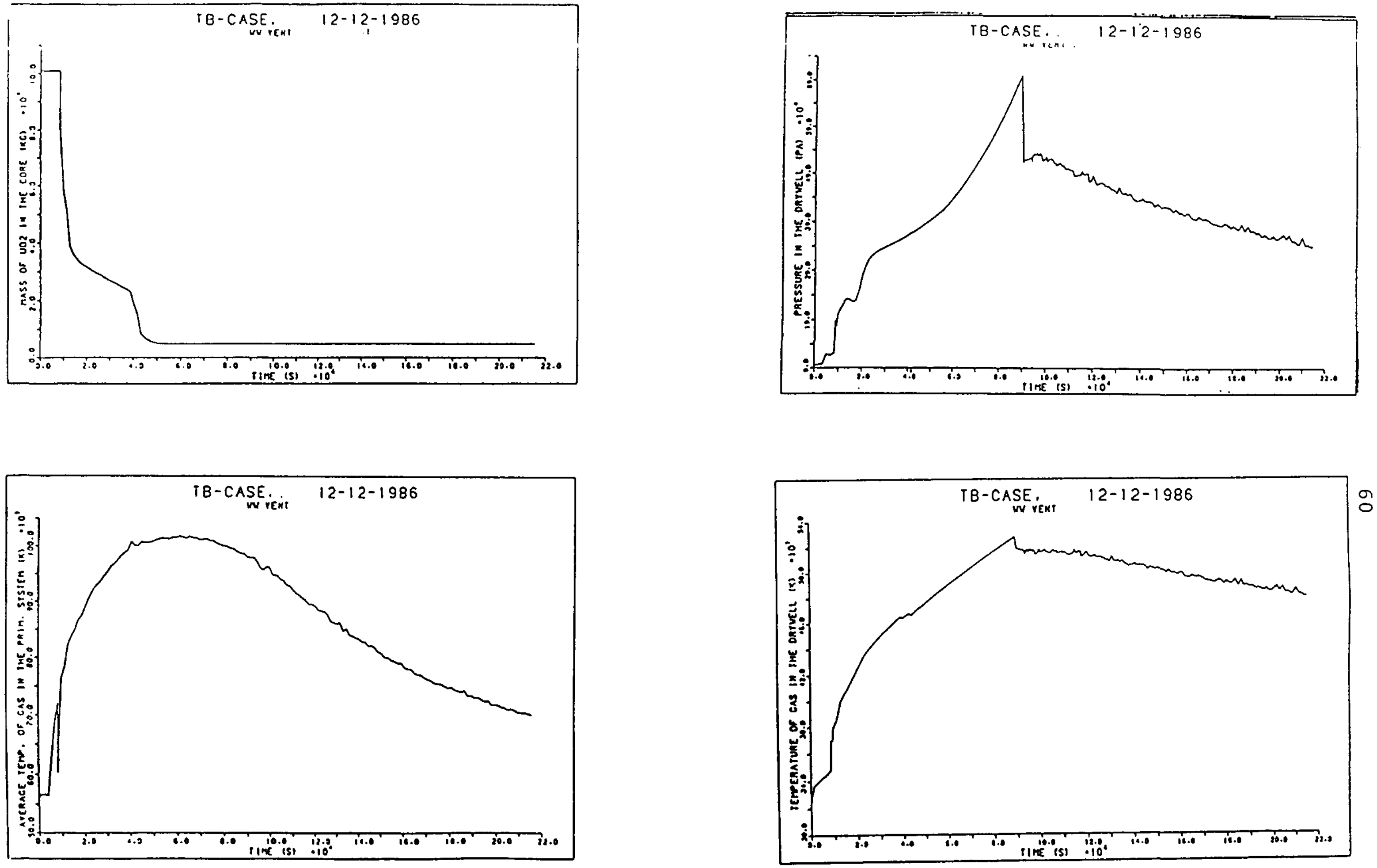
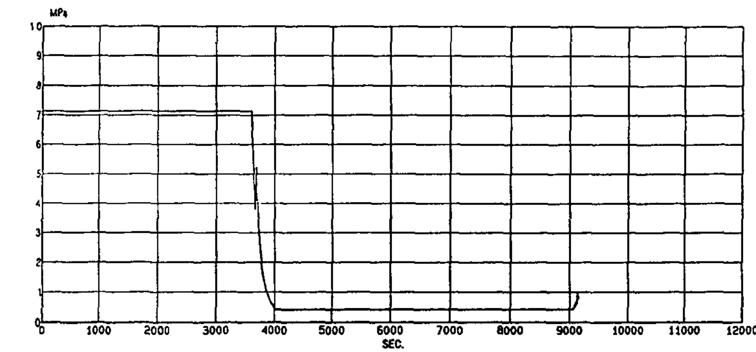
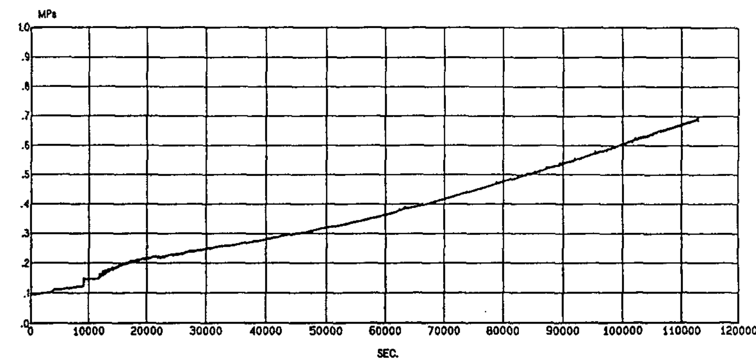


Fig. 16. MAAP 3 analysis for BWR TB sequence. (TVO). Some important parameters.

OLKILUOTO TB (MARCH3) (186)
PRIMARY SYSTEM PRESSURE
IBOIL0105

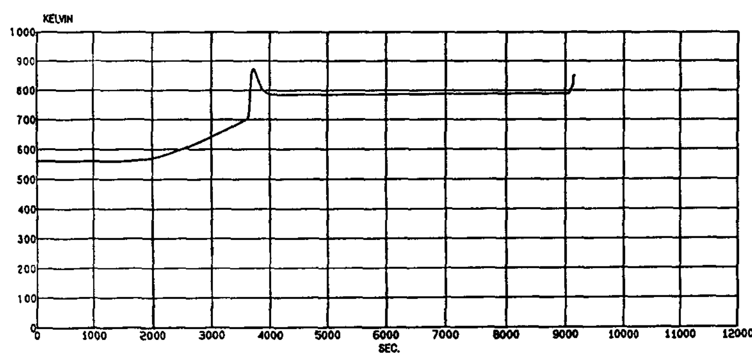


TVO TB (MARCH3) (186) COMP. 1 COMP. 2
TOTAL PRESSURE IN COMPARTMENT
IMACE010114



ELSAM 89.03.07

TEMPERATURE OF GAS IN UPPER PLENUM
IBOIL0114



ELSAM 89.03.07

TEMPERATURE IN COMPARTMENT
IMACE010518

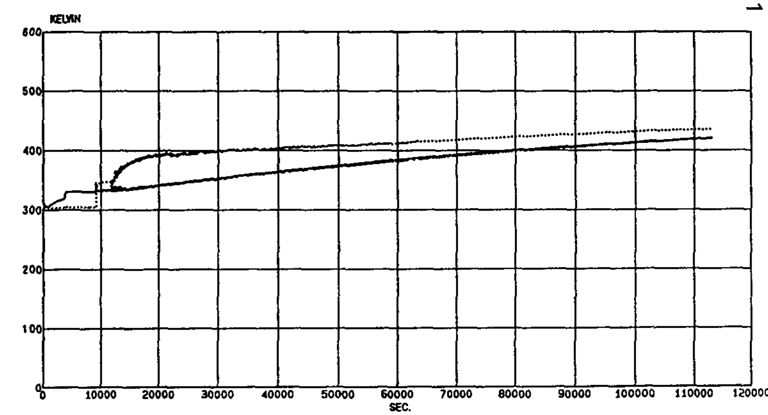


Fig. 17. MARCH 3 analysis for BWR TB sequence. (TVO I/II). Some important parameters.

to TB-sequence the leak was simulated by using the vacuum breakers in open position and by changing their flow area. The wetwell venting pressure (0.7 MPa) was reached 14 hours earlier if the leak area between drywell and wetwell gas atmosphere was changed from 5 cm^2 to 50 cm^2. The effect of leakage area on fission product release during venting can be seen in Table 4. If the DW-WW leakage is zero the release fraction of CsI is in the order of 10^{-6} compared to the release fraction of the order 10^{-3} in the design leakage case (5 cm^2) and to the release fraction of the order 10^{-2} in the large leakage case. It can be concluded that tightness of the wall seals, penetrations and valves is of great importance in order to keep the containment pressure suppression system intact.

The effect of containment vent or break location was studied in both TB- and S_2B-cases. In the performed sensitivity analysis the containment was vented from the wetwell at 0.7 MPa. Sensitivity of changing the vent or break location from the wetwell to the drywell was studied by assuming that the containment will break in drywell at 1 MPa. The break area was assumed to be the same as the wetwell vent area (diameter 0.15 m). The main differences were in the fission product releases from the containment. In the wetwell vent/break situation the release fraction of CsI was of the order of 10^{-6} in the base cases of TB- and S_2B-sequences compared to the release fraction of CsI of the order 10^{-2} in the drywell vent/break situation.

The effects of leakage area and starting time of the fission product releases from the containment were studied in the TB-case. In the base case it was assumed that containment barrier is leak tight until venting starts or a break occurs and thus the whole fission product release took place via vent or a break at a later phase of the accident. The effect of a leakage from the drywell at an early phase of the accident was studied by using two different pre-existing opening sizes. The smaller one represented 10 times the DBA-leakage and corresponded to an instrument pipeline and the larger one represented a 0.15 m open pipeline. In the former case the pressure and temperature behaviour and the time of venting were very similar to the TB base case. Also the CsI distribution between the primary system, condensation pool and drywell is very similar. However, this small leakage is able to transfer revaporized aerosols to the reactor building after a pressure vessel melt-through. In the case of a larger pre-existing opening, the maximum pressure and temperature in the drywell were lower (0.2 MPa, 180°C) except for a higher pressure spike at the time of the vessel melt-through. The CsI distribution differed from the base case. The main part of the release took place at the time of the vessel melt-through and at the later phase due to revaporization from the primary system surfaces.

Without operator actions (i.e. no manual primary system depressurization and pedestal flooding) the gas temperatures in the containment are high due to the production of hot non-condensable gases from the core-concrete interaction. In the case in which some steam flow through the relief valves to the suppression pool was allowed also after the vessel failure, smaller total releases due to pool water retention capability were calculated.

The modelling of the reactor building reduced releases to the environment although in the TVO-calculations the model was as simple as possible (one volume of 30000 m^3). The model also calculated temperatures too high in the gas atmosphere of the building.

5.2.2 MARCH 3 Calculations

In the sensitivity studies done with MARCH 3 for TVO I/II, the effects of the following assumptions were studied:

a) TB, core slump
b) TB, core channel blockage
c) TB, reactor pressure vessel bottom head thickness
d) TB, S_2B, debris bed particle size.

The slumping criterium for the reference calculation was the melting of the structures below the core as molten nodes in a core column were allowed to drop down onto the lower structures when the lowest node in a core column was melted. As a variation, a slumping criteria which slumps the core when 75 % of the core has melted, was used. The 75 % slumping criteria delayed core slump and pressure vessel failure by about one hour compared to the reference calculation.

No core channel blockage was assumed in the reference calculation. When core channel blockage was introduced (stop of hydrogen production in and above the lowest melted node in a core column) the amount of hydrogen produced in the core fell to about two thirds of the amount in the reference calculation.

The reactor pressure vessel bottom head has a lot of penetrations. The thickness of the bottom head in the reference calculation was set to the thickness of a control rod guide tube. This gave a melt-through of the bottom head in about half a minute after a core slump. When an "effective" thickness of 5 cm was used, the melt-through lasted one and a half hours and when the real thickness of the bottom head was used the melt-through lasted three hours.

The particle size of the debris bed which was formed in the reactor cavity after the melt through is essential for the boil-off rate of the water in the cavity and thus the pressure build-up in the containment. When one assumed 1.2 m particles instead of 1.2 cm particles the time to containment vent was two hours longer. A further variation of the particle size showed that 40 cm particles are coolable and 80 cm particles are not coolable.

For the S_2B sequence 1.2 m particles instead of 1.2 cm particles delayed containment venting by three hours.

5.3 Sensitivity Analyses for a Nordic PWR, Ringhals 2/3

5.3.1 MAAP Calculations

The sensitivity studies have been based on the accident sequence Loss of AC power (TMLB'). The sensitivity of the accident progression has been investigated for the following input parameters:

a) primary system heat losses
b) masses of the heat sinks in the containment
c) concrete composition
d) steel masses in the primary system
e) corium coolability in the cavity

TABLE 5 NKA-AKTI-130. RINGHALS 3. MAAP SENSITIVITY ANALYSIS. SUMMARY OT THERMAL/HYDRAULIC RESULTS

Accident sequence: Loss of AC power (TMLB')

Figures within parentheses refer to the corresponding base case

	Start of core melt, h	Vessel failure, h	Containment vent, h	Cavity dry, h	Mass of H_2 prod in core, kg	Mass of H_2 prod in containm, kg 4)	Mass of unmelted fuel, tons 5)
Primary system heat losses increased 100 % 1) 3)		2.56 (2.53)	5.4 (5.9)	7.5 (8.0)	247 (243)	382 (389)	17.0 (14.9)
Masses and areas of heat sinks in cont incr 100 % 1)		2.57 (2.58)	11.2 (5.2)	12.3 (6.1)	184 (191)	287 (433)	8.8 (9.3)
Primary system heat sinks reduced 50 % 1)		2.53 (2.58)	5.0 (5.2)	5.9 (6.1)	200 (191)	475 (433)	9.3 (9.3)
Heat flow corium to water in containment reduced 50 % 1)		2.58 (2.58)	5.2 (5.2)	6.1 (6.1)	188 (191)	410 (433)	9.3 (9.3)
Heat flow corium to water in containment reduced 90 % 1)		2.58 (2.58)	13.0 (5.2)	- (6.1)	188 (191)	501 (433)	9.3 (9.3)
Core injection recovery at 1.9 hrs or earlier 6)	- (1.94)	- (2.30)					
No core blocking, const Zr-H_2O reaction area in core 2) 6)	1.86 (1.94)	2.09 (2.30)	4.11 (7.61)	- (-)	563 (167)	- (-)	0 (22.5)

1) Independent containment spray not considered

2) Independent containment spray considered

3) MAAP 2.99

4) Up to 70000 s

5) Up to 30000 s

6) Ringhals 2

f) injection cooling of an overheated or partly degraded core

g) variation of hydrogen production (variation initiated by prescribing no blockage and unchanged reaction area).

The parameters a) to e) have been investigated within the Swedish RAMA II project and made available for NKA. The parameters f) to g) have been investigated within the NKA-AKTI-130 project. A summary of the numerical results of the sensitivity analysis is given in Table 5.

According to the results, the most important input parameters seem to be c), e), f) and g).

The variation of c), concrete composition, was effected by doubling the contents of Na_2O and K_2O. It was expected that this would increase aerosol production and the aerosol particle sizes during the debris-concrete reaction and would lead to an increased condensation of vaporized fission products on the aerosols. The expected results were reached. The releases of CsI and TeO_2 were significantly reduced.

The variation of e), the corium coolability in the cavity, included a reduction of the MAAP-modelled heat transfer coefficient between the corium and the water by 50 % and 90%. The reduction by 50 % resulted in a slower rate of quenching of the debris and an initially slower rate of containment pressurization. For longer times there were no significant changes in the progression of the accident. However, for the 90 % reduction, the debris was no longer coolable. The debris-concrete reaction including hydrogen production started soon. The pressurization of the containment was much slower, the time to 0.5 MPa increased for the base case from 5.2 hours to 13 hours. For time > 8 hours there were less airborne aerosols for the 90 % case. A part of the aerosols from the debris-concrete interaction were trapped in the water pool above. The 50 % reduction gave no significant change in the releases of fission products from the containment up to 70 000 s. The 90 % reduction gave a small increase of the Te release and a reduction of the Mo release up to 70 000 s. The releases of CsI and CsOH were unchanged.

In order to get a realistic picture of the progression of a transient including a delayed core injection (f) above), the assumed recovery of the water injection was initiated through an assumed recovery of the AC power. According to the analysis,it is not possible for the accident sequence Loss of AC power (TMLB') not possible to recool the core if the return of the AC power is delayed more than until about 6600 s. However, the real situation may be somewhat better than the MAAP calculations indicate. As mentioned earlier, according to the MAAP modelling it is not possible to quench a pool of molten corium within the reactor vessel. But the TMI accident has shown that this may in some cases be possible.

As mentioned above, the variation of the incore hydrogen production (g) above) was initiated by prescribing no blocking and unchanged reaction area during core melting. It was not possible to perform this variation by input changes. A few temporary code changes were necessary. According to the results, the variation made increased the part of reacted Zr in the core from 23 to 78 %. The hydrogen production and the core melt rate were increased. The mass of fuel in the core region which was left unmelted was reduced from 23 tons to 0 and the pressure increase in the containment accelerated.

5.4 Sensitivity Analyses for a Nordic PWR, Loviisa 1/2

5.4.1 MAAP 3.0 Calculations

Sensitivity studies are based on three types of accidents: S_2B, AB (primary system breaks and loss of AC power) and TMLB' (total loss of AC power). The variations of these sequences were divided into primary system break and containment studies.

Primary system break studies:

a) S_2B, AB, primary system break location and size
b) S_2B, AB, TMLB', pressure vessel failure delay.

Containment studies:

a) S_2B, TMLB', pre-existing opening in the containment pressure boundary

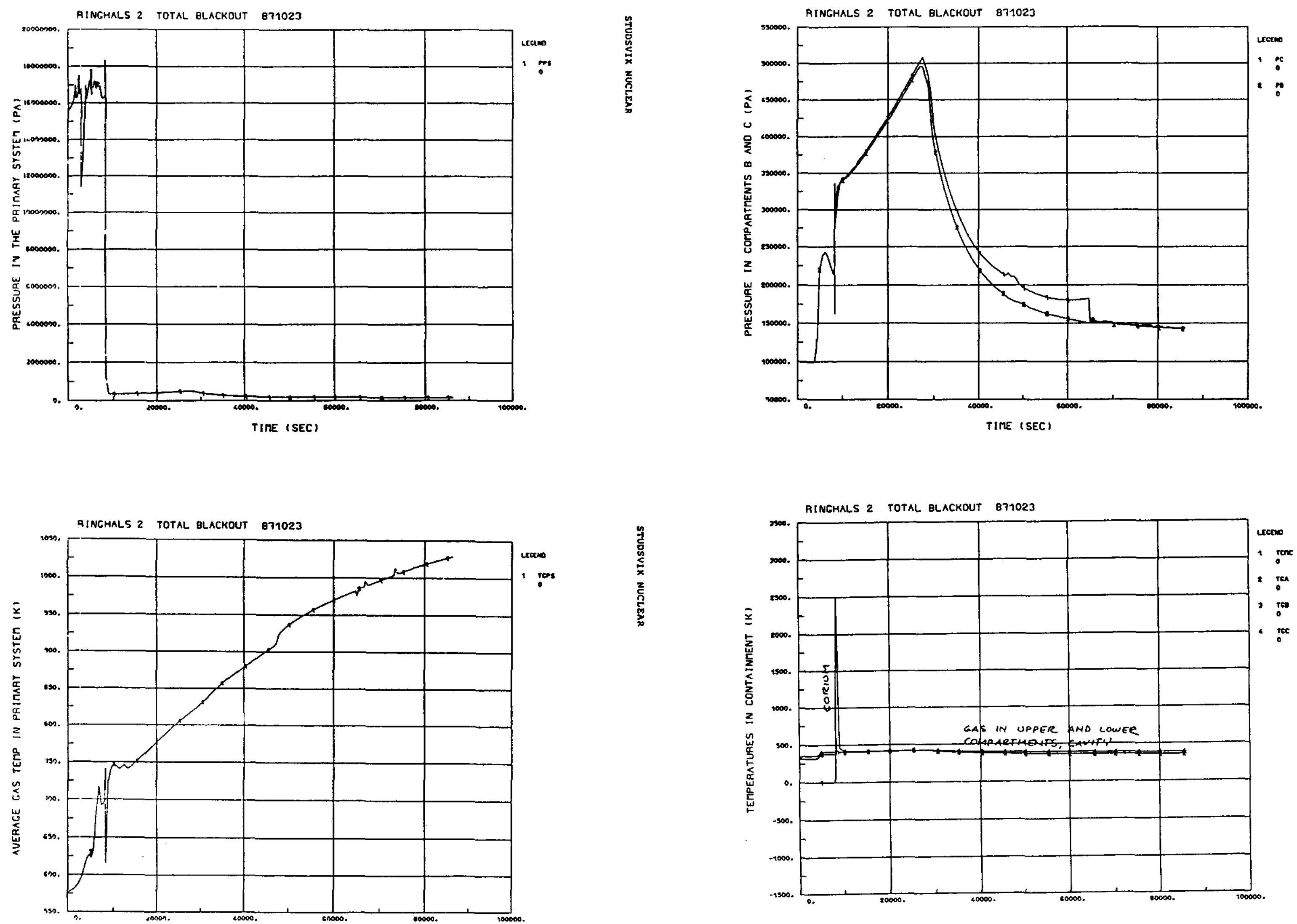


Fig. 18. MAAP 3 analysis for PWR TMLB´secuence. (Ringhals). Some important parameters.

b) S_2B, large bypass of ice condenser

c) S_2B, hydrogen behaviour (with different core melting model input assumptions).

In AB-sequences the break areas (in hot or cold leg) were relatively large. Most of the Loviisa comparison calculations for sensitivity studies were variations of TMLB'- and S_2B-sequences. The main difference between these accident scenarios was the occurence of the break in the hot leg (20 cm^2) in the latter sequence whereas in the TMLB'-sequence the primary system was depressurized by operator action when the steam generator secondary side water was depleted. There were considerable differences in the timing of important events when comparing these two sequences as can be seen in table 6. The coolant is lost much sooner in the S_2B sequence and consequently the core melt, ice depletion and containment venting occur earlier. There are also significant differences in the behaviour of the fission products in the primary system. For example, more Cs usually remained inside the primary system in the TMLB'-sequences than in the LOCA-cases.

Core melting and penetration of the pressure vessel in the Loviisa MAAP calculation deviates from the Ringhals-calculation because core modelling is different in MAAP 3.0/Lo -code. Because the pressure vessel in the Loviisa power plant has no bottom penetrations, sensitivity calculations with longer melt-through times (contact times) than 60 s of the pressure vessel were done. However the modelling of the melt-through process is the same as in the normal MAAP. There is no adequate heat transfer modeling for this particular delay situation in the MAAP code. Therefore, the temperatures in the primary system did not differ very much from the non-delayed cases. The MAAP 3.0/Lo code calculated a complete core melt (no corium left in the pressure vessel) in all cases. Due to the delay, corium temperature was high when the vessel failed. However, not much hydrogen was produced in the MAAP calculation neither in the primary system nor in the reactor cavity. The corium-water pool hydrogen generation model was called only when the initial quantity of corium dropped down so there was not much difference between the hydrogen production in the delayed pressure vessel failure cases compared to others.

In the Loviisa reactor core, each fuel assembly has a hexagonal shroud around it. Therefore the progression of core melting is assumed to be different from the normal PWR results calculated by the MAAP/PWR core heatup subroutine. The Loviisa core heatup model resembles more the MAAP/BWR heatup model. The changes in the core melting temperature, blockage and natural circulation in the core region changed the hydrogen production somewhat but in general the overall effect on the total of hydrogen produced was smaller than expected.

The recovery of AC power calculation showed that the present MAAP-model cannot handle the recooling process of the severely damaged core in a proper way.

In the primary coolant leakage area and break location sensitivity calculations in some cases natural circulation occured in the hot legs in some cases. In the Loviisa MAAP modeling it is estimated that during any sizable LOCA in the primary system the gas flow through the break will be sufficient to entrain the water in the pipe bends and prevent water blockages. This assumption allows natural circulation to occur in certain situations. The sensitivity calculations showed that if natural circulation occured in the hot legs more Cs remained in the primary system after vessel melt-through.

In general, the revaporization of fission products in the Loviisa MAAP calculation was small at the later phase of the calculated accidents. The containment venting pressure of 0.17 MPa is rather low and therefore, no significant flow effects occurred. The temperatures in the primary system predicted to be low and this led to a reduced revaporation when the pressure vessel failed. The pressure vessel failure delay did not much affect the fission product behaviour due to an inadequate heat transfer modeling from the corium at the lower plenum.

The decontamination effect of the ice condenser was studied by comparing two bypass sizes. Based on this comparison it can be concluded that the effect of a 2.5 m^2 bypass of the ice condenser compared to to the best estimate bypass area of 0.78 m^2 did not

Table 6. Results of the sensitivity analysis of the Finnish PWR with the MAAP 3.0/Lo code. Important event times and other key results. Calculation time is 2.5 days.

Accident sequence**)	Core uncovery h	Vessel failure h	Ice depletion h	Containm. venting h	CsI/CsOH release fraction	Hydrogen prod./mass of fuel left in core kg	Max. aver. gas temp. in prim. system °C	Peak cont.gas temp./pressure °C/bar
1	0.7	4	11	18	$2*10^{-4}$ / $3*10^{-4}$	50/0	280	130/1.7
2	0.8	4	11	19	$2*10^{-4}$ / $1*10^{-3}$	"	280	"
3	0.4	1	8	15	$2*10^{-6}$ / $5*10^{-5}$	"	260	"
4	6.4	10	18	27	$4*10^{-6}$ / $3*10^{-6}$	60/0	320	"
5	0.7	9	11	18	$5*10^{-4}$ / $5*10^{-5}$	50/0	280	"
6	6.4	14	18	27	$9*10^{-5}$ / $3*10^{-5}$	60/0	330	"
7	0.8	4	11	-	$8*10^{-3}$ / $8*10^{-3}$	50/0	280	130/1.1
8	6.4	14	18	-	$6*10^{-3}$ / $6*10^{-3}$	60/0	330	"

*) In the analyses the containment is assumed to be leak tight (except in cases 7 and 8) until (unfiltered) venting (diameter 150 mm) is started at containment design pressure (1.7 bar). In the cases 1-5 pressure vessel failure time is assumed to be 1 min after core slump.

**) Accident sequences are:

(1) base case, S_2B, 20 cm^2 hot leg LOCA
(2) S_2B, 20 cm^2 cold leg LOCA
(3) AB, 0.38 m^2 cold leg LOCA
(4) TMLB' and stuck open pressurizer safety valve (after first opening)
(5) base case but pressure vessel failure delayed about 5 h
(6) same as TMLB' sequence 4 but pressure vessel failure delayed about 4 h
(7) base case with pre-existing opening (vent line)
(8) TMLB' sequence 6 with pre-existing opening (vent line)

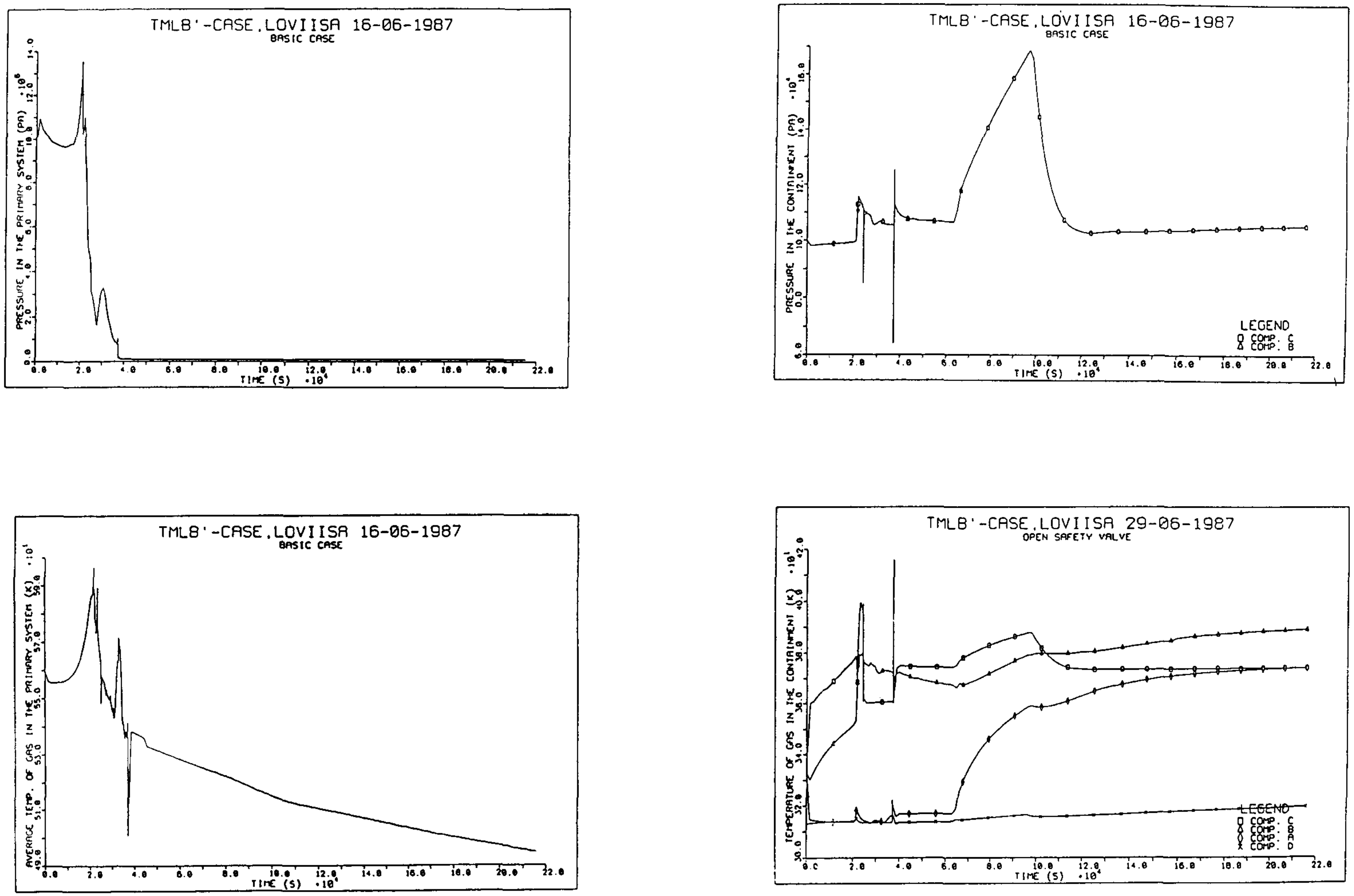
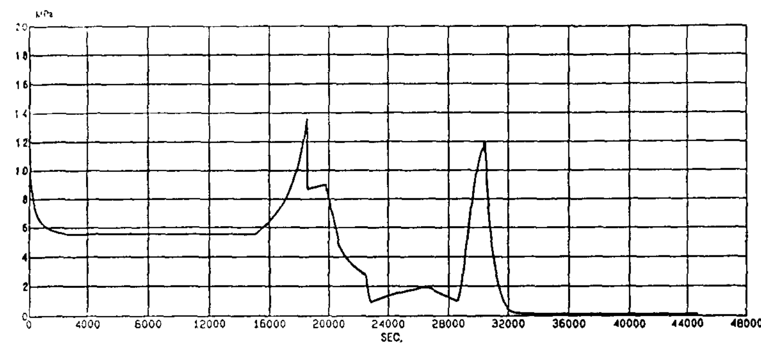
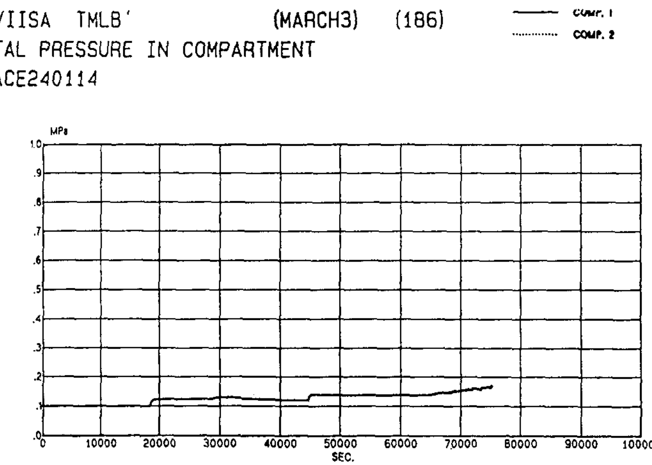


Fig. 19. MAAP 3 analysis for PWR TMLB´secuence. (Loviisa). Some important parameters.

LOVIISA TMLB` (MARCH3) (186)
PRIMARY SYSTEM PRESSURE
IBOIL2405

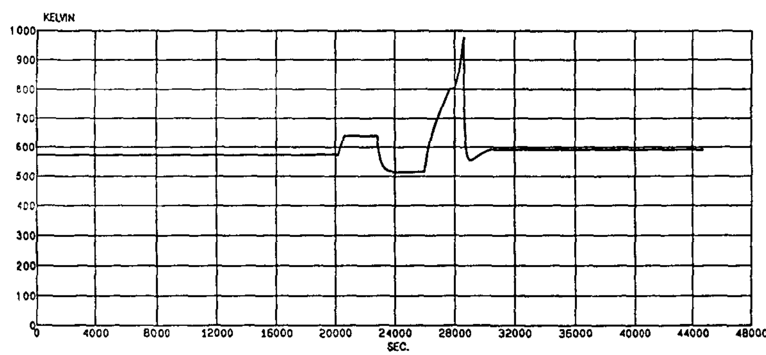


LOVIISA TMLB' (MARCH3) (186)
TOTAL PRESSURE IN COMPARTMENT
IMACE240114



ELSAM 88.10.30

TEMPERATURE OF GASES IN UPPER PLENUM
IBOIL2414



ELSAM 88.11.04

TEMPERATURE IN COMPARTMENT
IMACE240518

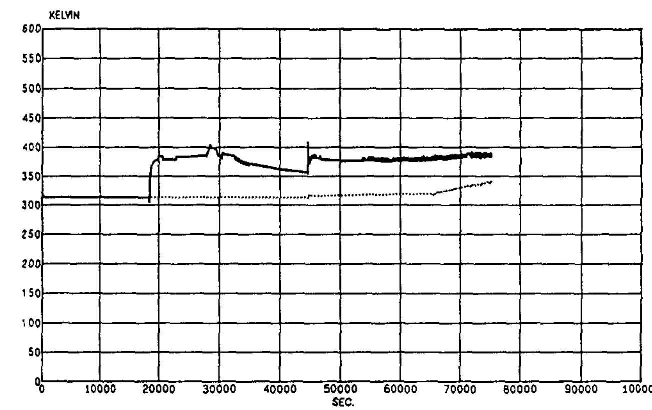


Fig. 20. MARCH 3 analysis for PWR TMLB´secuence. (Loviisa). Some important parameters.

much change either the thermal hydraulic response of the containment or the releases from the containment in the MAAP calculations. As expected, the pre-existing containment leakages caused the highest releases to the environment.

5.4.2 MARCH 3 Calculations

In the sensitivity calculations made for Loviisa with MARCH 3 (See Table 1), the effects of the following assumptions were studied:

a) TMLB', S_1B, core channel blockage
b) TMLB', S_1B, debris bed particle size.

In the reference calculations, no core channel blockage was assumed. With a core channel blockage, hydrogen production in the core fell from 236 kg to 194 kg in the LOCA sequence and from 300 kg to 235 kg in the TMLB' sequence.

When the particle size of the debris bed in the LOCA sequence was changed from 1.2 cm to 1.2 m containment venting was delayed by ten hours. The hydrogen produced in the reactor cavity reduced from 400 kg to 200 kg. The reduction in the hydrogen production was due to the 100 times smaller total surface of the particles and the differences in the debris bed particle temperature progress.

When the particle size of the debris bed in the TMLB' sequence was changed from 1.2 cm to 1.2 m containment venting was delayed by ten hours and the amount of the hydrogen produced in the reactor cavity reduced from 365 kg to 184 kg.

6 SPECIAL CONTAINMENT PHENOMENA

6.1 Core-Concrete Interaction

AKTI-130 made some preliminary calculations concerning the core-concrete interaction. Both MAAP 3.0 and MARCH 3 were used. In the analyses performed, the core-concrete interaction did not take place when there was enough water in the reactor cavity. If, however, the temperature of the corium was increased well over its melting temperature (~ 2500 K) some erosion of concrete was predicted by the MAAP 3.0 code /15/. E.g. at the temperature level of 4000 K an erosion depth of 0.16 m was calculated. The coolability of the corium was studied with the MARCH 3 code and it was found out that if the particle size of the corium was increased to 0.8...1.2 m in diameter, the corium was not coolable any more and the temperature of the corium was 3500-4000 K /13/.

In two cases the reactor cavity was assumed dry /4,13/. In these cases the erosion depth was about 2 m during the 55 hours calculated by the MAAP code and 0.6 m during the 11 hours calculated by the MARCH code. Different concrete types and reactors were used.

In the OECD/NEA/CSNI comparisons related to the core-concrete interaction are continuing. In /33,34/ the core-concrete interaction studies are described by making a code comparison for the conditions in large dry containments as well as benchmark calculations for the simulation of the SURC-4 experiment. Results from the CSNI work should be taken into account when the effects from the core-concrete interaction are considered.

6.2 Hydrogen Effects

The amount of hydrogen production was estimated by using both codes. The MAAP code predicted lower amounts of hydrogen produced in the core and no production in other places. The MARCH code predicted hydrogen production in the core, in the bottom of the pressure vessel and in the reactor cavity. Typically in the BWR and Loviisa cores where there are flow channels the differences in the production amounts were as follows: for the 1100 MWe BWR 5-6 % by MAAP and 15-24 % by MARCH, for the 700 MWe BWR 5-7 % by MAAP and 11-37 % by MARCH, for the VVER-440 6-7 % by MAAP and 30-37 % of the Zr total in the core by MARCH. In the MAAP calculations, a channel blockage model was used and in the MARCH calculations no blocking was assumed. For the 800 MWe PWR without flow channels, MAAP predicted a 23 % hydrogen production in the core.

An about 50 % oxidation of the cladding for the TMI-accident has been estimated. A part of this oxidation was probably caused by the reflood of the core. Additional experimental information is available from the SFD- and LOFT-experiments. From the above experiments and considerations it can be concluded that calculations based on the core blockage modeling tend to give too low hydrogen production rates.

Hydrogen production generates additional energy and therefore, with an increasing hydrogen production, core melting is somewhat faster and more complete, temperatures are higher in the pressure vessel affecting e.g. the revaporization of fission products. Production of noncondensible gases increases containment pressure.

The Nordic BWRs are inerted and therefore there is no danger of the hydrogen burning. In the PWRs, containments are larger and hydrogen concentrations are lower than in the BWRs. However, the possibility of local hydrogen burns must be taken into account and the effects of global hydrogen burns should be considered which is the case with the Nordic PWRs.

6.3 Temperature Effects

The performed MAAP analyses show that temperatures inside the pressure vessel and, in certain cases, also inside the containment, rise to high values and exceed design limits in some cases. High temperatures affect e.g. revaporization of radioactive matter, environmental conditions of components, leak tightness of the containment and integrity of components inside the pressure vessel.

For the primary system, the internal parts around the core will be intensely heated by thermal radiation from the overheated and melting core. In the MAAP analyses, the calculated temperatures of the lower core barrel (PWR) and moderator tank (BWR) for the accident sequences TB and S_2B are very high, in the order of 1800 K. This high temperature means that these structures may collapse and partly melt. The collapse and/or melting of these structures is not modelled in MAAP. The main reason for the high temperatures is that part of the fuel is left in the core region. In the MARCH analyses, all fuel slumps from the core and, therefore, temperatures in the pressure vessel are lower than in the MAAP calculations. According to the MARCH predictions, temperatures in the upper plenum are typically in the order of 800-1000 K.

Also the calculated temperatures of the inner side of the reactor vessel are high in the MAAP calculations, the max inside temperatures of the vessel are about 1100-1300 K. It should be investigated whether these high temperatures have any influence on the structural integrity of the vessel. In the Loviisa case, however, the pressure vessel is partly submerged into water and the MAAP 3.0/Lo code melts all fuel from the core into the bottom of the pressure vessel which has no bottom penetrations. Therefore temperature behaviour is different from the other cases studied.

In the BWR containments the maximum gas temperatures in the TB- and S_2B- sequences predicted by the MAAP code are the following: 520-580 K in the drywell and 410-430 K in the wetwell gasphase and pool. The main reason for the high drywell temperature is that part of the fuel is left in the core region. The typical design limits for containment temperatures are 420 K for the drywell and 360 K for the wetwell pool. Even higher temperatures are predicted for the containment if the lower drywell is not flooded. Then maximum drywell temperature is more than 770 K which probably leeds to a containment failure. For the PWR plant, the MAAP code mostly predicts gas temperatures of about 400-450 K in the upper and lower compartments and in the cavity when the reactor cavity is dry in the beginning of the accident. In this case, after a vessel melt-through, hot corium is collected in the cavity and cooled by water drained from the primary system. For the LOCA + Loss of AC power case, all water in the cavity boils away after about 25000 s. Corium temperature increases and after 8 hours, when it has reached about 1500 K, the corium is quenched by water from the independent containment spray.

As mentioned in chapter 7, the Nordic reactors are being equipped with an independent containment spray. This spray, which is not considered in most of the above calculations, will cool down the containment atmosphere. This spray and also the planned filling up of the containment will also help to keep the pressure vessel temperature down.

6.4 Fission Product Behaviour in the Containment

Fission product and aerosol behaviour have been studied in large scale experiments e.g. in the CSE facility, in the DEMONA experiments, in the Marviken experiments and in the LACE-facility /35,36/. Experimental results of aerosol removal from the gas phase are presented in Figs 21 and 22 to get an idea of the time behaviour of aerosol concentration under accident conditions. Experiments have been simulated by using different aerosol codes like NAUA /36/ for development and validation purposes. In the development of aerosol correlation of the MAAP code e.g. CSE experiments have been taken into account /21/.

Key features affecting aerosol behaviour and release from the containment are the assumption of a homogeneous concentration and principal removal processes such as gravitational settling and steam condensation.

Experimental results from the CSE tests show that the homogeneous mixing inside one volume is good but between different volumes poor. The typical percentage of gravitational settling in a CSE-facility was about 70 % /35/. Steam condensation has an effect when the ice condenser, sprays or outside cooling are applied.

Concerning late containment failures (e.g. 20 h), natural removal processes have decreased the aerosol concentration by the factor of $10^2 \ldots 10^3$. Therefore, processes which transfer deposited fission products or aerosols back to the gas phase are of major importance. Principal phenomena are revaporization from primary system surfaces, resuspension from containment sumps and chemical behaviour of fission products in the waterphase.

The removal rate of the aerosol particles from the containment atmosphere is significantly enhanced by a working spray. The effect of the spray dominates all passive deposition phenomena. An effective spray will remove most aerosols from the containment atmosphere in a couple of minutes.

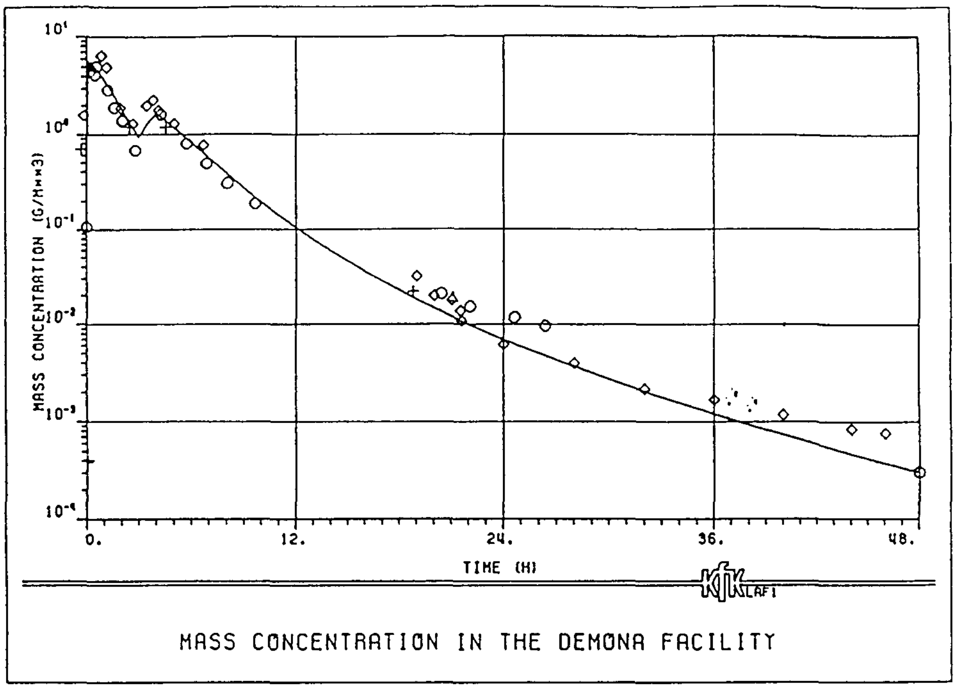


Fig 21. Comparison of Measured DEMONA Data to the Calculation Performed after the Experiment, Aerosol Generation Rate and Leak Rate fitted according to the Experimental Results

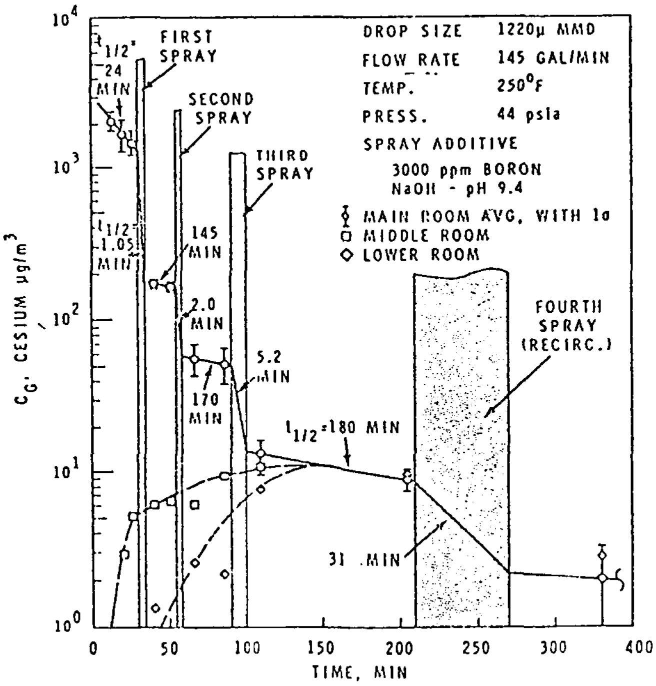


Fig 22. Removal of cesium from the containment atmosphere by spray and natural processes

Both the condensation pool and the ice condenser are effective in aerosol removal. The results from the BWR analyses performed show that 45-75 % of CsI ends up in the condensation pool in TB- and S_2B-sequences. In the PWR-analyses the ice condenser removes about 30 % of the aerosols (gravitational settling about 60 %) in the containment. Other interests from the source term point of view are the chemical behaviour of CsI in the pools and sumps and a resuspension of the fission products from these during the venting period or after the break of the containment.

Figure 23 shows the release fractions (CsI, CSOH, Te and structural materials) from the primary system to the containment calculated by the TRAPMELT code for the Forsmark 3 BWR. Figure 24 shows CsI distributions in the primary system and in the containment calculated by the MAAP code for the Forsmark 3- and TVO BWRs and for the Ringhals- and Loviisa PWRs. As it is presented the differences in the calculations of the MAAP and TRAPMELT codes up to the time point of pressure vessel failure are small but, later because of revaporization, MAAP calculates larger release fractions to the containment. Figure 24 shows that there are large differences in the calculations for different reactor systems.

The MAAP code assumes that all iodine is bound to cesium as cesium iodide. Any iodine present in a volatile form would not be as efficiently retained in the pool water, as predicted in the MAAP calculations. A small part of the iodine, possibly of the order of 1 %, can be expected to combine in volatile organic compunds. Oxidizing conditions in the pool water may cause increased volatility of the iodine. However, these phenomena have not been taken into account in the MAAP code. The MAAP analyses performed show that in the BWR cases, during venting when containment pressure has exceeded 0.5-0.7 MPa the energy which is stored in the pools begins to release by steaming the pool water. In the analyses the amount of off-boiled water during the following 24 hours corresponds to 20-25 % of the volume of the condensation pool.

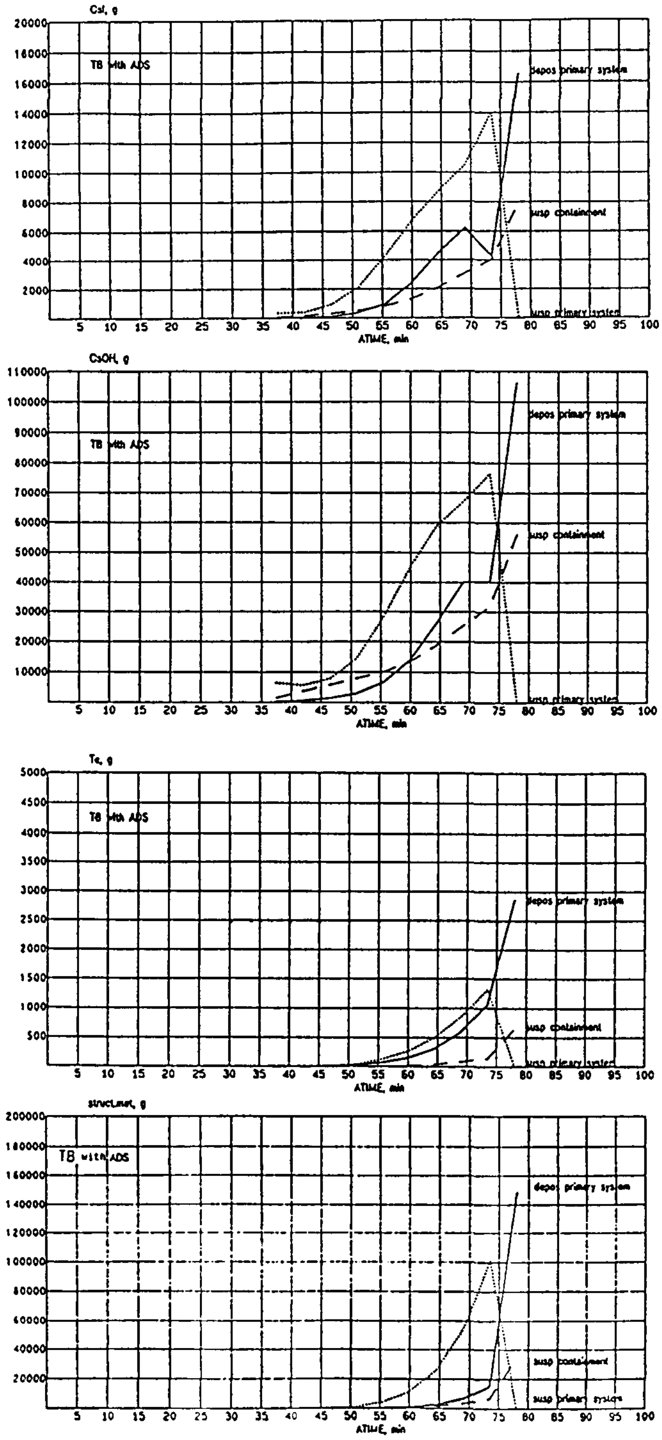
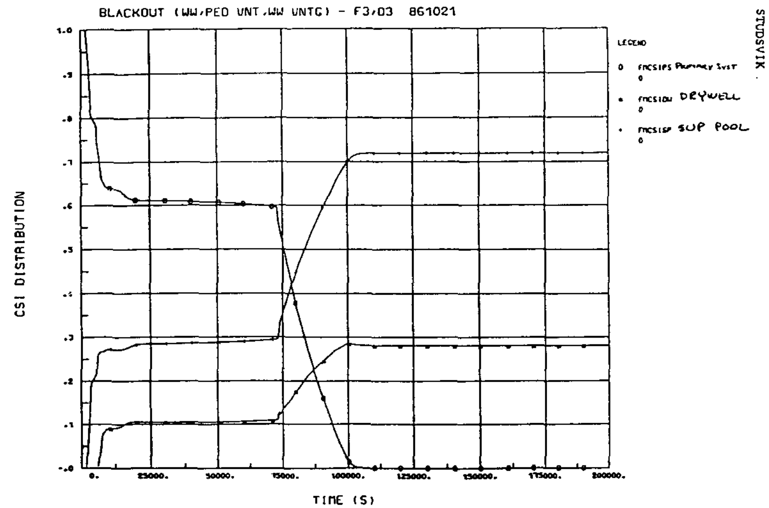
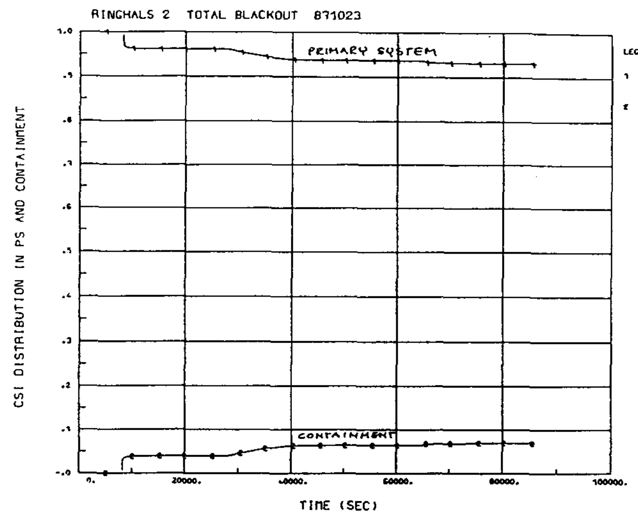
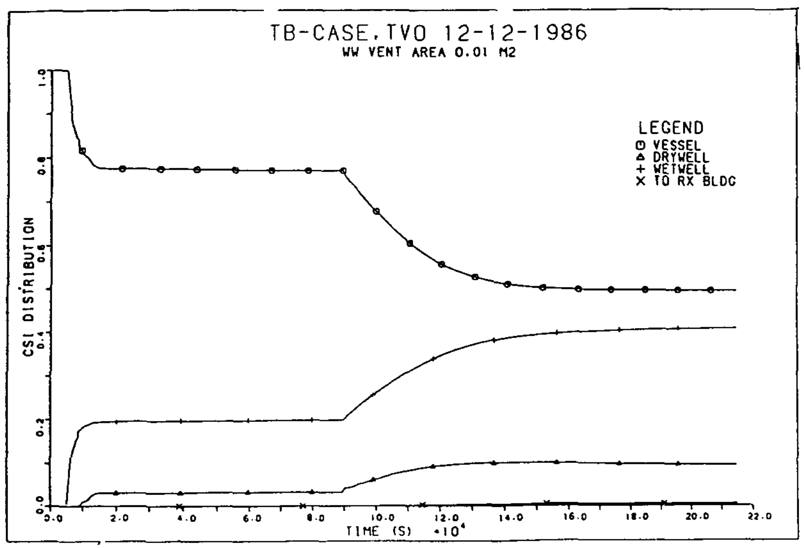


Fig. 23. TB accident with ADS, TRAPMELT results (with Revap), (F3,03)







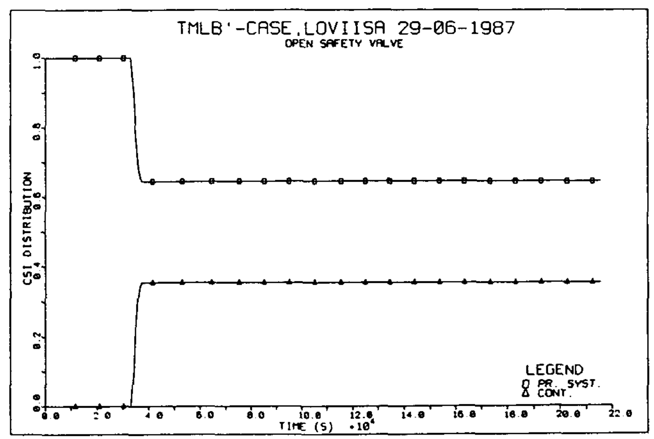


Fig.24. CsI distribution between primary system and containment in TB sequence in different plants.

7 ACCIDENT MANAGEMENT

7.1 Mitigation Systems

With respect to the mitigation systems, the Swedish reactors can be divided into four categories:

- the two Barsebäck BWRs, where the condensation pool covers the whole bottom area of the containment and the two containments are connected to the FILTRA gravel bed

- the Ringhals 1, Oskarshamn 1 and Oskarshamn 2 BWRs, where the condensation pool covers the whole bottom area of the containment and the containments are connected to venturi scrubber systems with water pools

- the three Forsmark and the Oskarshamn 3 BWRs where the compartment below the reactor vessel is dry and the containments are connected to venturi scrubber systems with water pools

- the three Ringhals PWRs with large dry containments and the containments connected to venturi scrubber systems with water pools.

The Barsebäck FILTRA system consists of a 10000 m^3 gravel bed connected to the containment of each of the two reactor units. Venting is feasible both manually through pipes connected to the upper drywell and automatically via a rupture disc and a vent pipe connected to the gas volume of the wetwell. The system has a large venting and filtration capacity. The design basis event sequence includes a loss of coolant, a simultaneous degradation of the pressure suppression function and a loss of AC power for 24 hours. The system is designed for a removal efficiency of 99.9 % of radioactive substances which can cause land contamination. The system is designed to function passively for 24 hours.

The mitigation measures for the BWRs Ringhals 1, Oskarshamn 1 and Oskarshamn 2 include systems for a large capacity unfiltered venting, a medium capacity filtered venting and an independent containment spray. The design basis event sequence for the large capacity venting includes a large LOCA and a degraded pressure suppression function. The spray cooling of the core is assumed to be available and there will not be any significant releases of radioactive material. For the medium capacity filtered venting, the design basis event sequence is a loss of all AC power (and steam driven feedwater system when installed). In addition, the Oskarshamn 1 has been modified in order to facilitate the drainage of the down flowing corium from the pedestal and to protect the pillars in this area from the hot corium.

The Oskarshamn 3 and the three Forsmark BWRs have also been fitted with double venting systems and independent containment spray systems as described above. In addition, a valve for the transfer of water from the wetwell to the lower drywell (pedestal) has been installed. The valve is actuated automatically or manually. Measures have also been taken in order to protect the pipe and cable penetrations in the lower drywell floor from the hot corium.

The containments of Ringhals 2, 3 and 4 PWRs have been equipped with systems for a filtered containment venting and an independent containment spray. The design basis event sequence is a loss of all AC power (and steam driven feedwater system).

The filters at Ringhals, Oskarshamn and Forsmark are all of the same design. The main part is a multi venturi scrubber with a pool for collecting iodine and particulate matter. The efficiency of this filter is essentially the same as the efficiency of the Barsebaeck FILTRA. The entire filter system is passive in the sense that no external supplies or operator activity is necessary for 30 hours after system activation.

For the independent containment spray, the water sources will be industrial water or sea water. Powering and control will be taken from independent and separated systems.

In Finland, the TVO I/II -BWR plants have similar mitigation systems as Swedish Forsmark-BWRs. Main differences are that at TVO German type venturi scrubbers are used and that manual actions are relied on instead of automation which is used in Sweden. In the Loviisa-PWR plants an outer, independent spray cooling system is planned for the cooling of the containment shell to avoid venting. However, filtered venting is also considered. Mitigation measures planned in Finland so far are presented in /37,38/. Technical solutions adopted in Sweden, France, Germany and Switzerland have also been presented in /39/ and the policies adopted in other countries are presented in /40/.

7.2 Operation of Mitigation Systems

The containment venting systems of all Swedish reactors open automatically at the prescribed pressures. Manual opening as well as closure is also possible. With the exception of Barsebäck, they will also close automatically during prescribed conditions.

The valve between the suppression pool and the lower drywell (pedestal) in the Oskarshamn 3 and Forsmark reactors opens automatically when there are indications of insufficient core cooling. The valve can also be operated by manual actions.

The independent containment spray system is brought into operation by prepared operator actions.According to FSAR, this spray will be operating within 8 hours after accident initiation. Besides its function of lowering the pressure and washing out the aerosols of the atmosphere, the spray will also be used for filling up the containment with water to a level equal to the normal top of the core. In order to keep down containment pressure, manual venting of the containment is foreseen during this operation.

If a core melt accident, including a vessel meltthrough, has occured, the long term goal is to reach a situation where the containment is partly waterfilled as described above and the pressure inside the containment is equal to the ambient pressure.

Besides the prepared operator actions connected to the mitigating systems, there are also a number of prepared actions for the early part of a severe accident in order to identify the status of the core, restore the core cooling, lower the primary system pressure, etc.

However, for the safety assessment of the mitigation measures, no credit is taken for operator action during the first hours in a severe accident. For Barsebäck this time is 24 hours. For the other reactors a more active accident management approach has been taken. Some well prepared operator action should be performed within 8 hours at least.

The above information reveals some differences between Barsebäck and the other reactors regarding equipment and credit for operator action. The explanation is the development of the licensing requirements and the increase of knowledge about severe accidents which has taken place between the establishment of the concepts.

In Finland in the TVO-BWR plant, the operation of all mitigation systems is based on manual actions. Thus the depressurization of the primary system and the filling of the lower drywell with water from the suppression pool are activated manually. According to the emergency operating procedures, these actions are performed between 30 and 60 min after the initiation of a severe accident. The isolation valves in the large capacity (unfiltered) venting line are shut manually to avoid a loss of containment situation via an unfiltered venting line in the case of a core melt accident. At a later phase of the accident the independent containment spray system and successively filtered venting system are taken into operation manually. The necessary actions will be defined in the emergency operating procedures. Symptom oriented emergency operating procedures for handling severe accidents have been developed by TVO. In the case of the Loviisa PWR, the design of mitigation systems is being developed and thus, the handling of the operation of these systems is premature.

8 CONCLUSIONS AND RECOMMENDATIONS

The following general conclusions can be drawn after the benchmark and sensitivity calculations performed and the MAAP/STCP workshops held:

8.1 General conclusions from code comparison

1. MAAP and MARCH give in general reasonable representations of the possible progression of severe accidents with a core melt in the case of Nordic nuclear reactors. The codes are accordingly suitable as a basis for the safety assessment of our reactors. However, due to uncertainties regarding some phenomena involved, the results from the codes must be interpreted with care.

2. The integrated thermal-hydraulic and aerosol code MAAP proved to be an efficient tool in the analysis of the progression of hypothetical severe accidents. An obvious advantage with the MAAP code is that all the phenomena involved in accident progression are evaluated in parallel and their interaction is thereby accounted for. However, much work remains to be done and is in progress in the verification and validation of the different MAAP models.

3. Compared to MAAP, the Source Term Code Package required much more work but offered better possibilities for some detailed physical and phenomenological studies.

4. The two code systems proved to complement each other in the sense that they represent alternative modelling in certain respects, allowing corresponding uncertainties and sensitivities to be explored.

5. For both codes, the preparation of the input and the evaluation of the results must be made carefully. A good knowledge of the physical phenomena involved and the models in the codes is necessary.

   Much effort is needed for obtaining a correct data set up for the codes. MAAP and MARCH have different definitions for many input variables and it is sometimes difficult to find out the precise definition of a variable. This may introduce some uncertainty into the calculated results and into the comparison of the results from the codes.

6. Any one of the codes may give slightly different results when executed on different computers. This problem of numerics needs further attention.

As a discussion of this problematic the following can be presented:

The computers have not the same accuracy. Depending on the coding of the algorhitms, this may give a considerable difference in the results.

An unprecise definition of the method of calculation may be interpreted in a different way by different compilers. Furthermore, different level of optimization may give differences in the calculation method.

In the United States attention has been paid to the problem of different results, specially for MAAP-PWR. One speaks of "numeric chaos". The problem is seriously looked upon and a remedy is attempted.

8.2 Conclusions Concerning Primary System Thermal-Hydraulic Models

1. A common observation is that MAAP and MARCH give a lower hydrogen production in the core than TMI-2 or the results of such experiments as LOFT FP-2, SFD and others. A possible explanation is that the codes, besides possible assumptions of blockage underestimate the supply of steam to the core (or the experiments overestimate the supply of steam during severe accidents).

2. The irreversible total blockage of the core assumed in MAAP can cause that the hydrogen production is underestimated. MARCH, but not MAAP, provides the option of assuming presence or absence of core blockage. If the

MARCH-option for a core-blockage is chosen, calculations for a BWR using the two codes, give approximately same results. However, calculations without this MARCH-option for Forsmark 3 on the accident-sequence total blackout shows that MARCH gives five times as high a hydrogen production as MAAP.

3. According to MAAP modelling, only molten material can move inside or out of the core. In MARCH, only molten material can move inside the core but both melted and unmelted material can move out of the core in certain input options.

4. It is important to know how much unmolten material stays in the core position for a long time. This influences the long time temperatures in the primary system and in the containment.

5. MAAP does not model the water-cooling of the melted material in the core. This means that the recovery sequences with core melting give unrealistic results.

6. For MAAP 3.0, only the PWR version considers hydrogen production in the lower plenum. The actual subroutine for PWR is only called during one timestep. As a consequence, the H_2 production may be underestimated.

7. MAAP's model for a melt-through of the reactor pressure vessel bottom head may not be valid for a pressure vessel without penetrations. For such reactors MAAP should be completed with a model which considers heat transfer between the melt and internal parts of the pressure vessel and the bottom head and which describes the melt-through. This influences temperatures in the primary system, time for melt-through, flows and revaporization.

8. MAAP's model for the cooling of the core-material in the lower plenum gives a moderate cooling and the result is often that the melt is not cooled and melts through the

vessel in a short time and then drains. It would be of great value here to adapt alternatively Lipinski's correlation for the heat exchange between the core material and the water to make sensitivity studies related to particulate debris. In MARCH, it is possible to use Lipinski's correlation etc.

9. It has not been modelled in the codes whether a melt lying in the bottom head can melt away parts of the control rod guide tubes and cause parts of the core to drop.

10. MAAP models heat exchange, gas circulation and heat-sinks well. This gives a good basis for the calculation of the transport, deposition and revaporation of fission products.

11. The nodalization of the core (50 nodes) used in MAAP as well as in MARCH is rather coarse and may influence the fission product release from the fuel and the hydrogen production. This has not been studied in the sensitivity analyses performed.

8.3 Conclusions Concerning Containment System Thermal Hydraulic Models

1. Many of the models for the containment in MAAP and MARCH are similar. However, differences are found, e.g. in the models concerning the cooling of the melt in water and in the models for the hydrogen producing chemical reactions between H_2O and metals. Furthermore, the possibilities for doing sensitivity studies by changing the input are different for the two codes.

2. MAAP and MARCH contain models which describe the heat transfer between hot and possibly melted core material and the overlying water. If the default-values are set for the parameters in the models, this gives a heat transfer which gives the cooling of the core material for the studied cases. As both models and parameter values contain a degree of uncertainty, the results of the codes do not give a

sufficient support of the coolability or noncoolability of the core material.

3. Experience from experiments shows that the core-melt is in most cases fragmented when it falls into water and is cooled. The size of the fragments is important for coolability.

4. There is a considerable number of steel structures below the BWR pressure vessel. These structures include control rod equipment, steel radiation shields, etc. The codes do not take into account how these structures possibly influence the melt flow from the pressure vessel to the lower drywell and whether the melt flow will split or melt a channel directly to the drywell floor.

5. A direct heat exchange between melt ejected at high pressure and the containment atmosphere (so called Direct Containment Heating) is not modeled in the used versions of MAAP and MARCH.

6. MAAP contains a simple model for steam-explosions in the pedestal. According to the MAAP-model, only very limited amount of material will participate and steam explosions therefore will cause no severe consequences to the integrity of the containment. Steam explosions will fragment core material, however, and may in this way influence the coolability of the core melt.

7. MAAP calculations for Loviisa overestimate the hydrogen burn for sequences with a dry cavity in regard of the availability of oxygen. Oxygen is sucked from other compartments in an unrealistic way. To improve MAAP's predictions concerning the burning of hydrogen, a better model for the distribution of the gases by time in the compartments is needed.

8. There are some shortcomings in the codes concerning "Engineered Safety Systems":

When assuming filling of the BWR-containment with water, MAAP diverges, when the wetwell is nearly full of water.

MARCH can not model the flow of water from the wetwell to the lower drywell (this must be done at the initiation of the sequence).

9. It is possible that the codes underestimate the hydrogen-production in the containment. Possible hydrogen-producing reactions which are neglected, are e.g. reactions between UO_2, Zn, Al, Fe and H_2O. (For BWR Al is found in considerable amounts in the equipment just under the pressure vessel).

8.4 Conclusions, Aerosol and Fission Product Behaviour

1. The end time of a TRAPMELT calculation is the time when the pressure vessel fails. Thus re-evaporation from the primary circuit surfaces during the later phases of the accident is not taken into account. In the MAAP calculations this re-evaporation from the primary circuit surfaces often controls the release of CsI and CsOH from the containment.

2. MAAP has no model for condensation of steam on non-hygroscopic particles, and the hygroscopicity model is very crude. It is suggested that it should not be used because, if applied, it overestimates the deposition significantly. MAAP applies a LMFBR value of the shape factor. For containment calculations an LWR value of 1 would be more appropriate.

3. MAAP has a suspected error in the calculation of the suppression pool decontamination, as steam is not considered as noncondensible for a boiling pool.

4. MAAP overestimates the decay heat of Cs for times longer than one or two hours. This may be important for re-evaporation and hygroscopicity.

5. The output from MAAP is in many cases insufficient to give a clear picture of aerosol behaviour.

6. MAAP has a difficulty in treating aerosols with different material densities (from different sources or when steam condenses on the particles). This adds uncertainty to the results.

7. The model of particle growth kinetics in TRAPMELT3 (by condensation or evaporation of fission products) is very crude. This adds uncertainty to the results.

8. TRAPMELT3 seems to be very sensitive to the computer. A version of the code that ran smoothly for a certain problem on one computer ran into all kinds of computational troubles on another for the same problem.

9. In MAAP and STCP it is assumed that Cs and I form CsI and CsOH. This modelling may be insufficient. This problem has been investigated by the NKA-AKTI-150 working group and the results are described in the final report of this group.

8.5 Recommendations for future work

1. It is important to continue Nordic cooperation in the field of nuclear reactor safety.

2. Nordic competence within the field should be maintained and further developed. The results from international research should be picked up and used in the continued work directed to evaluate the safety of the nuclear reactors both within and outside the Nordic countries.

3. In particular it is important to continue the benchmarking and evaluation of the improved versions of the codes MAAP and STCP.

4. The more advanced codes, CONTAIN, MELCOR, RELAP5/SCDAP etc should also be evaluated.

5. The experience from the implementation and running of the codes on different computer systems may be a valuable piece of experience to be exchanged.

6. The new and improved codes include new models for some of the phenomena involved. It is therefore important to include also sensitivity analyses in the continued work.

7. The results presented in this report consider only part of the new mitigation systems installed or being installed in the Nordic reactors. Application of the codes on such system evaluation should be an important part of the continued work.

8. An important part of the safety evaluation is the selection of accident sequences to be included in the analyses. It is recommended to include a study of these questions in the further cooperation work.

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