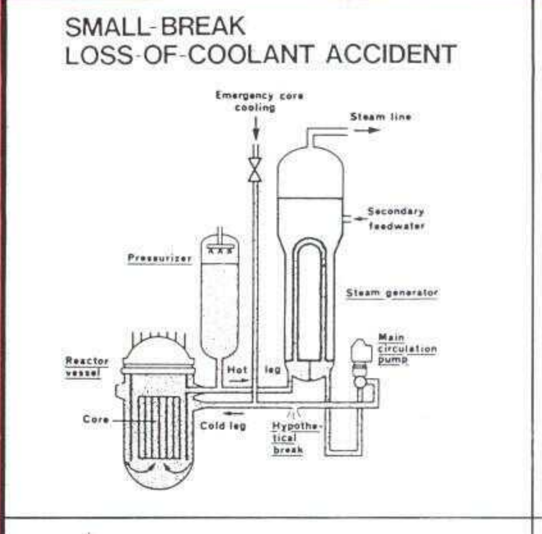
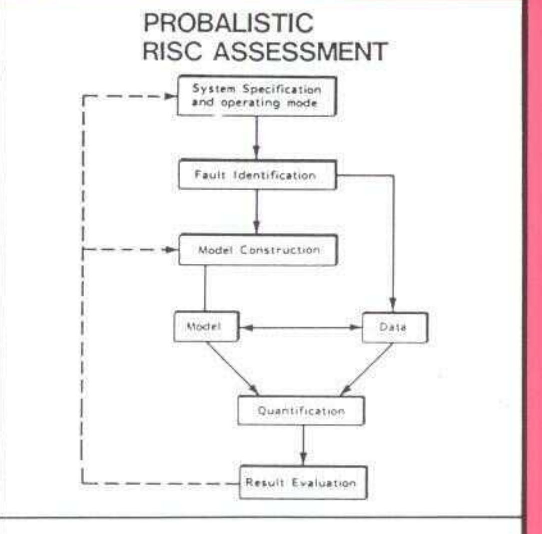
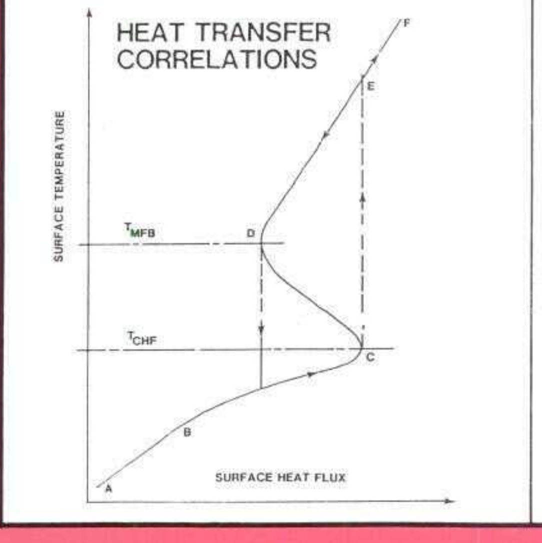
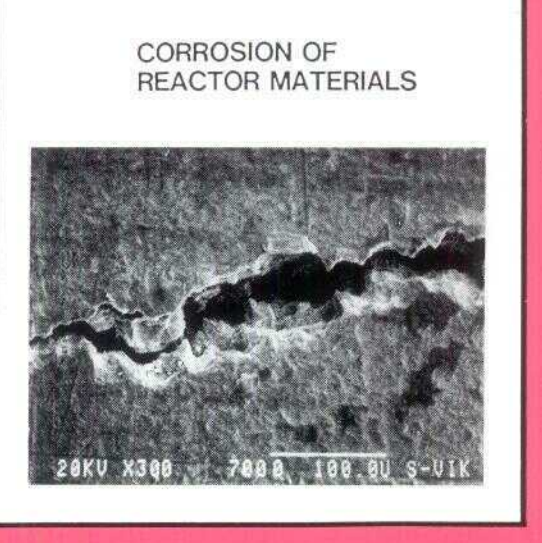
JUNE 1986

NORDIC REACTOR SAFETY RESEARCH 1981-85









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Nordic
liaison committee for
atomic energy

NORDIC
REACTOR SAFETY RESEARCH
1981-85

SUMMARY REPORT OF THE NKA/SÄK PROGRAMME

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JUNE 1986

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ABSTRACT

National resources in Denmark, Finland, Norway, and Finland were put together with Nordic funds in the four-year research programme 1981-85 on selected areas of nuclear safety. The outcome of the programme, edited in four separate reports, is summarized, and important findings are listed in the areas of probabilistic risk assessment (PRA), loss-of-coolant accidents with small breaks, heat-transfer correlations, and corrosion in the nuclear industry.

INIS DESCRIPTORS

COMPUTER CODES; CORROSION; HEAT TRANSFER; LOSS OF COOLANT; NUCLEAR INDUSTRY; NUCLEAR POWER PLANTS; PROBABILITY; REACTOR SAFETY; RISK ASSESSMENT.

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CONTENTS

1. INTRODUCTION

Safe and reliable operation of nuclear reactors has been a major issue in the public debate in the Nordic countries, and reactor safety is for that reason a major object of research.

The Nordic Liaison Committee for Atomic Energy created in 1980 a common Nordic reactor safety research programme in fields that were considered vital in the evaluation of nuclear reactor safety, and where common knowledge, common methods, and a consensus of opinions in the Nordic countries were considered essential.

Probabilistic risk assessment (PRA) has been developed and used to a large extent in reactor safety studies, notably in the Rasmussen report (WASH 1400), but the technique has been questioned on a number of points as for example: data, calculation methods, common cause failures, completeness and verification. For that reason a joint Nordic project was organized in 1980 to apply the somewhat limited national resources efficiently in the study of quantitative reliability analysis techniques.

A loss of coolant from the reactor core is an event which can lead to a serious accident. Specialists in the Nordic countries had since 1973 worked together on thermo-hydraulic computer codes for the analysis of large breaks. However, the Three Mile Island accident in 1979 brought to the attention the fact that a small break Loss Of Coolant Accident can be as dangerous as a large break LOCA, that the probability of small breaks is larger than of large breaks, and thus the small breaks give a larger contribution to the overall risk. The huge and complex codes developed in the USA (RELAP and TRAC) and in the earlier Nordic cooperation (e.g. NORCOOL) were made with the purpose of simulating a break over a short time span (seconds, minutes) while the analysis of small breaks should describe hours or days. A Nordic project for small break LOCA analysis was taken up with the aim of selecting one or more computer codes for this analysis. The project was mainly based on the existing large computer codes, which were evaluated for their applicability to small break analysis.

In parallel with the study of the overall applicability of the thermo-hydraulic codes, a detailed study of the heat transfer from the fuel rod surface to the coolant was initiated. The temperature of the fuel rod is dependent on the heat transfer mechanisms between rod surface and coolant, and these are described by heat transfer correlations. In the Nordic project the heat transfer correlations used for the different heat transfer regions corresponding to different heat fluxes were studied. This investigation was initially seen as a minor project but as difficulties were recognised in the basic physical description, the project effort was increased in order to achieve the original objectives.

Corrosion of reactor materials exposed to the water in the systems has given problems. These problems have so far mainly affected operation and repair. Corrosion in the primary system and in outer systems exposed to sea water was taken up as a Nordic project with the aim of collecting data, enhancing the understanding of the phenomena, and thus of improving the performance of the materials.

2. PROBABILISTIC RISK ASSESSMENT (PRA)

Probabilistic risk assessment (PRA) techniques are increasingly being used at industrial plants to identify specific areas where safety and reliability can be improved in a cost effective manner. In the Nordic countries, PRA techniques are currently employed as a tool during the evaluation of the safety of chemical plants, off-shore platforms, nuclear power plants, and other complex industrial systems. In the nuclear field these techniques are used extensively for system evaluation in Finland and in the Swedish As-operated Safety Analysis Reports (ASAR's) which are periodic reviews of the operating nuclear power plants.

With PRA methodology it is possible to establish a comprehensive framework which leads to a well documented analysis of a plant and its functions. In order to take full advantage of PRA, techniques must be available which are systematic enough for general use and easily documented, yet sufficiently accurate to resolve the issues in question.

In the Nordic project some established PRA techniques were compared while others were further developed. Two "Benchmark" studies, where the same object was analysed independently by three study groups, were performed to compare different basic data compilation techniques, computer codes, and system modelling methods.

The Benchmark 1 study was concerned with a high pressure injection system typical of PWR plants. Each group independently compiled the required reliability parameters for a common system model using data handbooks and other generic sources. The data sets showed significant variations due primarily to the initial data source. The groups then used each set of data in turn in their own computer code, and the results were found to be quite consistent for each data set. Thus, the result is more dependent on the input data than on the computer programmes used.

In the Benchmark 2 study three plant response models for a loss of feedwater transient in a BWR plant were independently developed and then compared. The emphasis here was on the comparison between different modelling methods such as cause-consequence diagrams compared with event trees, and reliability block diagrams compared with fault trees. This study indicated that the choice of an appropriate method depends, to a large extent, on the complexity of the system to be analysed and on the objectives of the analysis. Furthermore, the need for careful review and close contact with persons intimately familiar with the system, such as plant operators, was found to be at least as important as the choice of techniques.

Systematic search methods - both computerized and manual - were developed to identify potential common cause failures. These methods were also verified in the context of the Swedish ASAR studies. Intensive work was done to improve the dependent failure models necessary to quantify the identified dependencies. The new models were required in order to consistently take into account the high level of redundancy typical for new nuclear power plants in the Nordic countries.

Statistical methods were developed for the treatment of field data collected from the power plants. Particular emphasis was placed on the estimation of the uncertainty in the calculated parameters. The methods are adapted for use in Nordic PRA studies and were used during the compilation of the second version of the Swedish Reliability Data Handbook (T-boken).

The available PRA techniques are useful tools which can help the licensing authorities evaluate the safety of nuclear power plants. The numerical results from such evaluations provide a means for referencing existing functions to a common scale allowing comparisons which otherwise would be impossible. When used with proper regard for the current limitations, these results can also be used to identify weak points and help the licensing authorities evaluate proposed changes and choose between alternatives.

3. COMPUTER CODES FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS

The purpose of the project was to evaluate computer codes for small-break LOCA analysis. Four computer codes were studied: three codes developed in the USA, mainly for large-break LOCAs, namely TRAC/PF1 (from Los Alamos Scientific Laboratory), RELAP5/MOD1, and RELAP5/MOD2 (from Idaho National Engineering Laboratory); in addition the simpler Finnish code, SMABRE, specially developed for small-break LOCAs, was used.

The codes were studied by applying them to experimental cases. The code simulations could then be compared with the experimental results and the deviations analysed. The test cases include small-break experiments performed in the Loss-of-Fluid-Test (LOFT) facility at the Idaho National Engineering Laboratory (USA), in the Loop Blowdown Investigation (LOBI) test facility at the EEC Joint Research Centre Ispra (Italy), and in the (Swedish type) BWR integrated test facility FIX-II at Studsvik Energiteknik AB (Sweden).

The LOFT test cases comprise two experiments with a 2.5% break. In one experiment the circulation pump remained running (L3-6) while in the other, otherwise similar test, it was stopped (L3-5). The LOBI test cases comprise a 0.4% break experiment, SD-SL-03, in the original LOBI facility, instrumented for large break LOCA's, as well as a 1% break experiment, A2-81, in the modified LOBI facility instrumented for small breaks. The A2-81 was used as an international standard problem, ISP18, and especially valuable for code evaluation, as the experimental results were unknown to the participants during their computer calculations.

A single FIX-II experiment was included: no.3031, with a 48% break, which is an "intermediate" break size rather than a "small" one. However, it represents the critical break size for Swedish type BWR's, i.e. the largest break size not leading to core dryout before a considerable decay of core power has taken place.

The theoretical study of the codes (review of manuals and code sources) showed that most of the steam/water flow effects occuring in a small break LOCA are modelled reasonably well, both regarding the purely mechanical effects and representation of the system. One exception is the modelling of the special fluid dynamic effects related to stratified flow, which was found to be either inadequate or missing.

This finding was confirmed by the practical part of the code study - the comparison with experimental test cases. The integral properties such as system pressure and temperature were predicted with reasonable accuracy (see fig.1). In the case where a running circulation pump ensured a nearly homogeneous steam/water mixture (LOFT L3-6), the prediction of local flow properties such as flow velocity and void fraction (steam volume fraction) was also satisfactory.

In all test cases no single code was found to be superior to the others regarding agreement with the experimental results. However, steam/water separation and the stratified flow effects, observed in the test cases with the circulation pumps stopped, were poorly predicted by all the codes studied. Thus, for example, the codes were not able to simulate properly the emptying of a horizontal pipe, connected to the reactor vessel, when the water level in the reactor vessel dropped below the pipe entrance. Large temperature differences were observed in the experiment between steam and water in the horizontal pipes, and this effect was not predicted by the codes.

Furthermore, it was observed that under conditions simulating boiling water reactors (the FIX-II test case) the RELAP5/MOD1 code at least may not be able to predict rewetting of fuel rods by the droplet spray cooling effect.

As no significant differences in prediction accuracy was found between the codes, the choice of a code for small-break analysis should depend on user-friendliness, speed and degree of detail. The SMABRE code has a simplified description of the physics and is fast, and for that reason it is superior for extensive parametric studies. The two RELAP5 codes have more

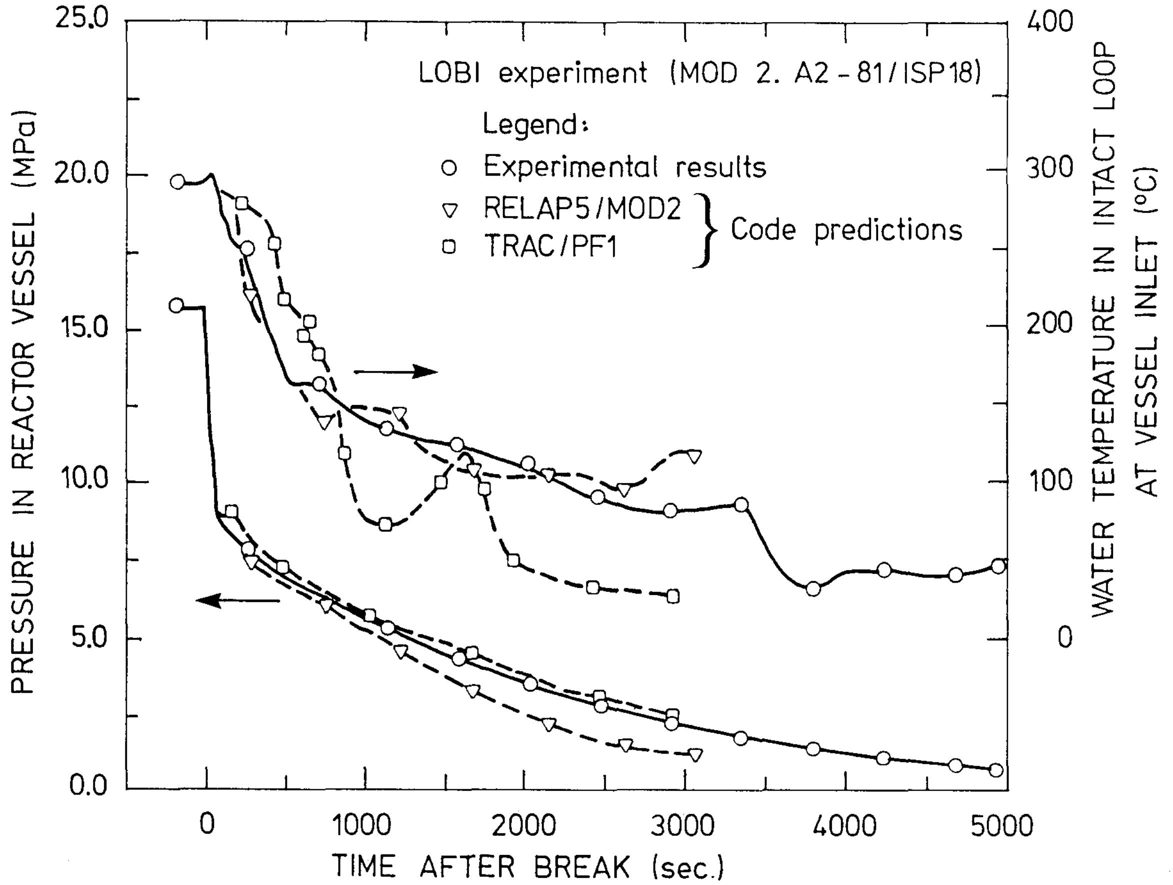

Fig 1: Comparison of experimental results and code predictions for a small-break LOCA.

flexible representation of the reactor system, and the RELAP5/MOD2 is faster than TRAC/PF1. The latter code should preferably be used in cases with significant multi-dimensional flow.

Finally, it should be noted that the test cases studied, all with functioning emergency core cooling, suggest that small break LOCA's of this type imply little danger for severe core melt-down accidents, when the emergency core cooling is provided in time.

4. HEAT TRANSFER CORRELATIONS IN NUCLEAR REACTOR SAFETY CALCULATIONS

An important part of nuclear power reactor safety analysis is to show that the temperatures on the surface of the fuel cladding during operational transients and postulated accidents such as LOCAs, do not exceed values that would damage the fuel cladding, when the engineered safety features of the reactor are activated and working. These critical temperature values are set by the safety authorities.

In order to calculate realistic temperature differences between the cladding surface and the coolant, reliable heat transfer correlations are needed for the different heat transfer regions occuring during a transient. Correlations in this sense are mathematical expressions, partly derived from physical principles but primarily empirical in nature. The heat transfer correlations used in the best-estimate LOCA computer programmes TRAC (USA), RELAP5 (USA) and NORA (Norway) were studied in this project.

Fig. 2 shows the surface temperature of a fuel rod as a function of the heat flux. A critical heat flux is reached at point C, and this critical heat flux (CHF) is characterized by the loss of contact between the surface of the cladding and the liquid coolant due to the formation of an isolating layer of vapour.

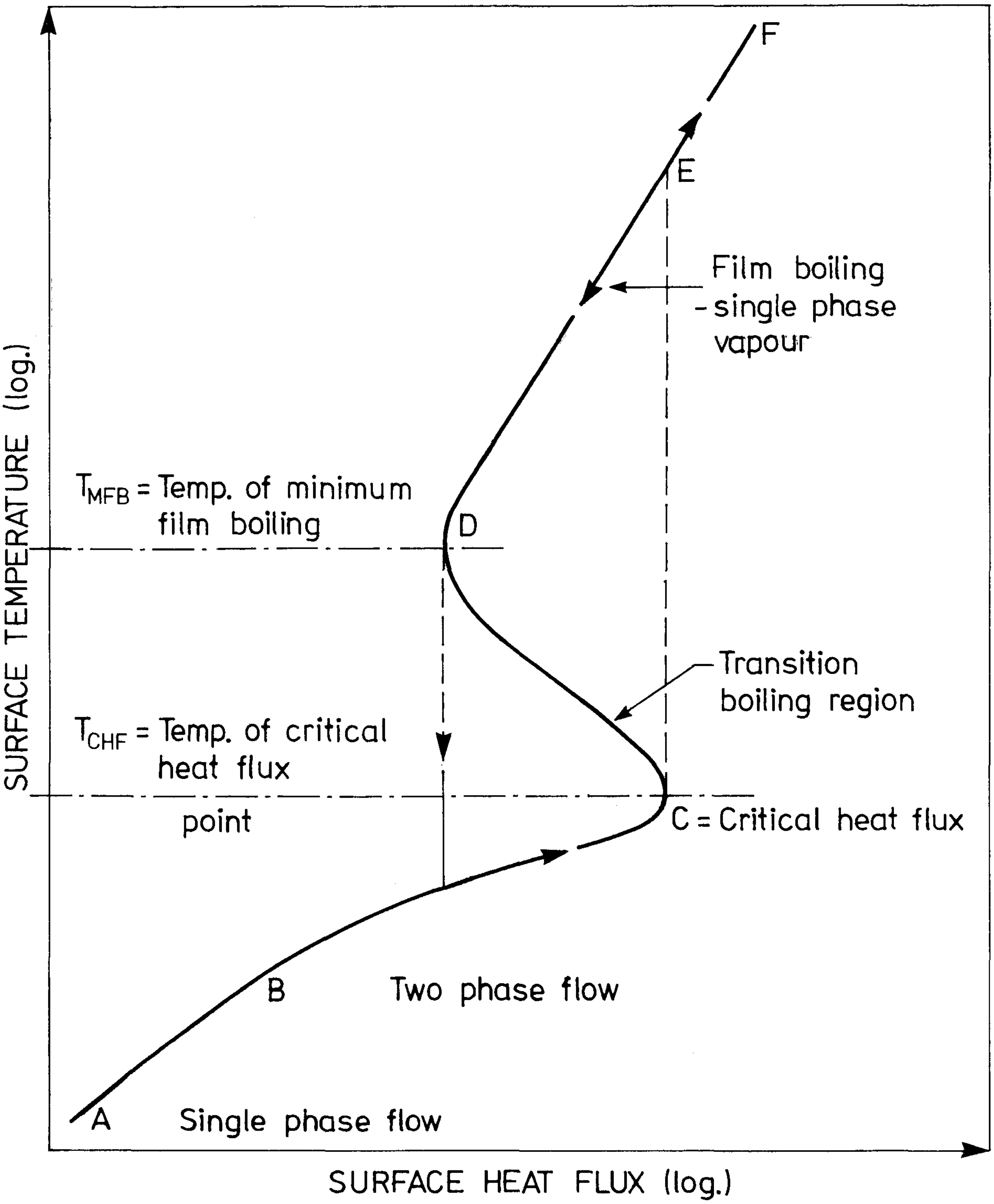

Fig. 2: Boiling curve (Heat Flux Controlled)

The examination of heat transfer correlations for the regions before the critical heat flux confirmed that they are sufficiently reliable. The heat transfer here is very efficient and will not be a limiting factor for the cladding temperature during an operational transient or a postulated accident.

The correlations in the computer programmes for the prediction of the value and the locus of the critical heat flux were found not to be sufficiently accurate, especially in the prediction of the locus of CHF. An examination with a programme, especially developed for that purpose, was made using two correlations from the large computer programmes, and two older correlations. The result showed that the two older correlations could predict the experimental data from 6 full-scale electric heated rod bundles (from 36 rods to 81 rods in a bundle) much better.

The examination of heat transfer correlations for regions after the critical heat flux (post-CHF) indicated, that they may predict measured surface temperatures with sufficient accuracy, if the following two physical phenomena were taken into account:

1. thermodynamic non-equilibrium
2. droplet cooling

These two phenomena were, however, not described correctly in the LOCA computer programmes.

Thermodynamic non-equilibrium is characterized by a vapour temperature higher than the liquid temperature as the heat will be transfered primarily to the vapour and from the vapour to the liquid-droplets. The vapour will be superheated and this non-equilibrium state will primarily be governed by mass transfer between droplets i.e. the vapourization. Taking this superheating into account an essentially better agreement between experimental data and calculated surface temperature could be attained.

The examination of thermodynamic non-equilibrium in the transition boiling region (between C and D in fig. 2) revealed furthermore that droplet cooling is a very important mode of heat transfer for this region. The heat transfer in the transition boiling region is larger than in the film boiling region (the region between D and E in fig. 2), because droplets, which hit the surface of the cladding, may wet the surface, and the cooling is enhanced by the direct evaporation of the droplets.

The two phenomena mentioned are strongly interacting in the coolant channel where there is heat transfer between wall, vapour and droplets.

The conclusion of the examinations is, that the importance of the droplets for the heat transfer in the post-CHF regions makes it necessary to improve modelling of the physical phenomena in the computer programmes.

The heat transfer phenomena and correlations discussed here for nuclear reactors, are of great interest also for other industrial plants, for instance in the analysis of steam generator behaviour, fluidized bed combustion, chemical processes etc.

5. CORROSION IN THE NUCLEAR INDUSTRY

Corrosion in nuclear power plants is a well known problem both in the Nordic countries and worldwide. Corrosion occurs in many different forms, such as general corrosion, crevice corrosion, pitting, stress corrosion cracking, and erosion corrosion.

Corrosion damage in the operating Nordic nuclear power plant varies in both form and extent. The effects on plant availability have so far been small, and all corrosion effects have been recognized before they could constitute a significant threat to plant safety. It should, however, be recalled that corrosion in most cases does not produce a uniform attack on the material. Thus, in principle, intergranular stress corrosion cracking and crevice corrosion could cause serious and rapid deterioration in a short time.

A compilation of corrosion experiences in the Nordic countries and worldwide was made. The most serious problem in boiling water reactors is intergranular stress corrosion cracking in stainless steel piping. In pressurized water reactors serious corrosion has occurred in steam generators, due to a combined effect of construction practice, material characteristics, and operating conditions.

The compilations show that so far the corrosion experience in the Nordic countries is considerably more favourable than in the rest of the world.

Corrosion in seawater of varying salinity, typical for the Nordic countries, was one of the topics of investigation. Experience from nuclear power plants as well as from conventional power plants has been compiled. The investigation deals with pumps, heat exchangers, valves and pipes. It turns out that a proper choice of material, different for each type of component, can decrease the corrosion in the seawater systems and therefore increase plant availability and safety. Aluminium bronzes are commonly used materials in seawater systems. A study made on dealloying of such materials shows that heat treatment of cast aluminium bronzes increases the dealloying resistance and also improves mechanical properties.

Corrosion of stainless steel in salt water can be evaluated by means of accelerated tests. A critical review has been made of tests available to study pitting and crevice corrosion. It turns out that international standards have frequently been modified at individual laboratories in order to improve the testing procedure. Although this yields better results, it makes it difficult to compare results from one laboratory to another.

Nickelbase alloys are sensitive to intergranular stress corrosion cracking in the environments prevailing in both boiling water reactors and pressurized water reactors. The influence of heat treatment and microstructure on stress corrosion cracking of nickelbase alloys was therefore studied. The results show that the corrosion susceptibility is affected by heat treatment and microstructure. There are, however, still many questions to solve and more work is requested in this area.

A number of alternatives to the materials employed today in power plants, both nuclear and conventional, are available. The condenser tubes in Swedish power plants are now manufactured from titanium. Other materials under development for power plants include a number of stainless steels (e.g. the new Swedish steel 254 SMO with increased contents of molybdenum and the titanium stabilized steel Monit). They are now being tested in power plant environments and in seawater.

Generally speaking, a number of methods are available to control corrosion in nuclear power plants, ranging from a proper choice of material to using adequate water chemistry. The results obtained show that the corrosion experience from the nuclear industry is valuable in conventional industry, and vice versa.

Apart from developing and testing new materials, current experimental work is concentrated on the understanding of microstructure and on corrosion protection through controlled water chemistry. It is one conclusion from this work that different Nordic laboratories have unique knowledge and competence which can be combined for mutual benefit through joint action.

6. CONCLUSION

The Nordic Co-operation on reactor safety research programmes has been efficient in pooling national resources, and in creating a common knowledge and consensus on the vital subject fields delt with.

The work has given the following results:

- The calculation methods in Probabilistic Risk Assessment are well developed but they require careful selection of data and close collaboration with the plant operator.
- Systematic search methods have been developed for identification of common cause failures.
- Statistical methods for treatment of raw component reliability data have been developed.
- Computer programmes for small-break LOCA are available and are adequate for cases where circulation pumps are functioning and stratified flow is avoided.
- The expertise to run the big complex computer codes is available in the Nordic countries.
- Proposals are made for the detailed modelling of heat transfer at high heat fluxes where there is a partial dry out of the fuel rod surface under LOCA.
- Corrosion statistics have been collected, and proposals concerning choice of materials and water chemistry are made.

The methods and knowledge obtained through this work concerning probabilistic risk assessment, heat transfer and thermohydraulics, and corrosion of materials in a water environment, are valuable both in the evaluation and design of nuclear plants as well as non-nuclear plants.

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Annex

Nordic Reactor Safety Research. 1981-85

Expenditures in thousands of Norwegian kr.

	1981	1982	1983	1984	1985	Total	Contributions Nordic	Contributions National
Probabilistic Risk Assessment	1747	2728	2637	2359	120	9591	3433	6158
Small-Break LOCA Codes	1619	1952	1649	1807	740	7767	3108	4659
Heat Transfer Correlations	537	1030	1277	1672	478	4934	1854	3080
Corrosion of Reactor Materials	0	440	730	830	130	2130	1003	1127
Travel expenses etc.	180	40	54	80		354	354	0
						24776	9752	15024

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