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DEPARTMENT OF REACTOR TECHNOLOGY
ANNUAL PROGRESS REPORT

1 January - 31 December 1977

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Abstract

The work of the Department of Reactor Technology within the following fields is described:

- Reactor Engineering
- Reactor Operation
- Structural Reliability
- System Reliability
- Reactor Physics
- Fuel Management
- Reactor Accident Analysis for LOCA and ECC
- Containment Analysis
- Experimental Heat Transfer
- Reactor Core Dynamics and Power Plant Simulators
- Experimental Activation Measurements and Neutron Radiography at the DR 1 Reactor
- Underground Storage of Gas
- Solar Heating and Underground Heat Storage
- Wind Power

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CONTENTS

1. INTRODUCTION

1.1 Organization

Organizational alterations were made in the Department of Reactor Technology during the year. To improve the utilization of theoretical and experimental knowledge and results in the field of heat transfer, the earlier theoretical and experimental sections were combined into one section of heat transfer and hydraulics. The Department now has the following five sections:

Reactor Engineering
Reactor Physics
Heat Transfer and Hydraulics
Dynamics
DR 1 Reactor

New section heads were appointed in the section of reactor physics and the section of heat transfer and hydraulics. A staff chart for the Department is shown on page 49.

1.2 Work

Studies of district heating were carried out at the beginning of 1977 and during the spring this work was extended to cover the modelling of total energy systems. A new "Energy Systems Group" was formed outside - but in close collaboration with - the Department, four of our scientific staff are now working in this group. During the autumn the group started work on the modelling of the (future) Danish energy system, and at the same time they assisted the authorities, e.g. by evaluating reports on energy systems.

Efforts concerning alternative energy have increased significantly during the year, and work on alternative energy and energy system analysis now occupies ~ 20% of the scientific staff - with a corresponding reduction of activities concerning nuclear technology. The following subjects are under study:

Underground heat storage for private houses.
Heat storage in aquifers.
Solar heating. Sun panels and the heating/storage system.
Underground storage of gas.

Windmills. Design of rotors of 40 m in diameter.
Plans for a testing station for small windmills were under way at the end of the year.

Some of this work is financed by the alternative energy research programme sponsored by the Ministry of Commerce.

The number of staff working on nuclear energy has been reduced, while new tasks are being channelled to the Department (e.g. from the Inspectorate of Nuclear Installations). Some of these tasks are:

Evaluation of transient calculations in PSARs.
Acceptance criteria for EEC.
Classification of systems and components.
Codes and standards for pressure retaining components.

Coupled with efforts to maintain a reasonable level of research and development in nuclear reactor technology, this work is quite a strain on the resources of the Department and reductions have been unavoidable in some fields, such as reactor physics and dynamics.

2. SECTION OF REACTOR ENGINEERING

2.0 Introduction

The main object of work in this section is to establish and maintain know-how concerning the design, construction, operation and safety of light water reactors.

This aim is chiefly pursued through much work on System Delineation. As a supplement, an investigation of Reactor Operation was started (described in 2.1 below). In addition more specific topics are dealt with, namely: Structural Reliability and System Reliability (see 2.2 and 2.3 below).

General knowledge of the design and construction of reactor systems is covered by the term System Delineation. Based on the information found in safety analysis reports, descriptions in Danish are produced to facilitate the assessment of the merits of various systems. At present, the Kraftwerk Union PWR and the Asea Atom BWR are being investigated, while the General Electric BWR/6 has been dealt with in reports on the following systems:

Primary pressure boundary
Residual heat removal system
Emergency core cooling system
Containment system
Service water system
Reactor core
Reactivity and power control system
Power conversion system (turbine)
Buildings and general lay-out

A report presents the design, physical lay-out, operational features and design criteria for the system in question, as well as its interconnections and functional relationship to other parts/systems of the power plant.

Studies of specific issues of concern to the safety of nuclear power plants were initiated partly on behalf of the Inspectorate of Nuclear Installations; they are mainly concerned with the regulatory framework. At present the following topics are being dealt with:

- Acceptance criteria for ECC calculations in Germany and USA,
- Classification of systems and components in different safety classes,
- Codes and standards for pressure-retaining components.

2.1 Reactor Operation

As a supplement to the work on system delineation, a study of reactor operation was taken up that covers subjects such as fuel management, control management and load following.

A special study was initiated of the restrictions imposed on operation in order to protect the fuel. The recommendations of various reactor vendors and fuel suppliers have been compared. The restrictions are imposed on control rod movements, on permissible power changes, and on load following. The restrictions imposed on BWRs seem to be more stringent than those imposed on PWRs. The restrictions recommended also deviate among different suppliers of the same type of reactor.

2.2 Structural Reliability

The purpose of this work is to develop methods for evaluating the reliability of structural components; in particular to develop computer codes based on probabilistic methods for evaluation of the reliability of primary components in light water reactors. Work was focused on the steel pressure vessel and the fuel element cladding.

Steel pressure vessel

Three steel blocks (i.e. drop-outs from a pressure vessel for a BWR) with dimensions T x W x L = 160 mm x 200 mm x 1000 mm were purchased from a European manufacturer. The blocks were welded together and cut into slices 25 mm thick. COD specimens will be manufactured and tested to evaluate the statistical variation of COD in the base material, weldings and heat affected zones. Furthermore, the variation in ultrasonic damping will be recorded and correlated with the COD measurements.

Preliminary studies were initiated of time-dependent phenomena apart from crack growth by fatigue. Main emphasis was laid on the influence of the hydrotest on failure probability.

A number of international contacts and collaboration projects have been established with manufacturers and research institutes in the FRG, Italy, USA and France. Furthermore, we participate actively in the CSNI task force on Problems of Rare Events in the Reliability Analysis of Nuclear Power Plants.

Reliability of fuel cladding

In nuclear reactors the fuel cladding yields the first protection against the release of radioactive products. The number of cladding failures must therefore be kept as low as possible. A computer program, FRP, was developed for the statistical analysis of fuel performance. The statistical methods are either Monte Carlo simulation or a Taylor approximation to the moments of the distributions.

The program utilizes a deterministic fuel performance code, FFRS, which has been verified on several irradiation experiments.

The failure mode considered to be the most important is stress corrosion. Based on available out-of-reactor stress corrosion experiments, a correlation for time to failure (time to crack penetration) was postulated. Under the assumption that the corrosive environment in irradiated fuel is comparable to that in iodine stress corrosion experiments, a cumulative damage index for stress corrosion was calculated.

Two examples of the use of FRP are:

- Safety related applications -

In order to estimate the consequences of some minor, but frequent, accidents leading to local or overall fuel ramps, FRP includes a simple core simulator. The simulator can give the power as a function of time for a number of axial segments in each fuel rod.

The consequences of the following situation were investigated for a BWR reactor:

At the end of the second cycle (for the fuel elements considered) a control rod is half inserted. After 3 months the control rod is withdrawn at full power. The overall power is assumed to be almost unchanged.

The fuel rods in the four elements surrounding the control rod are divided into 5 groups as shown in fig. 2.1. The pin power histories, the nodal failure probability, and the total number of failed rods assuming the node size to be 10 cm are given in table 2.1.

The node size of 10 cm corresponds approximately to the size of the test specimens in stress corrosion experiments.

- Comparison between designs -

As reliability is the logical basis for design comparisons, it is necessary to include probabilistic methods in the design comparison. For certain "reference power histories" both reliability and the influence of design and material parameters are calculated. These calculations show, besides the difference between the designs, that many of the tolerances specified for the fuel are unnecessarily low, and that they could be increased without affecting the reliability of the fuel.

Figure 2.2 shows a comparison between two standard BWR fuel designs: the design data and the reference power history are specified in table 2.2. The figure also illustrates the influence of the ramp rate (time to full power after changes in the power distribution).

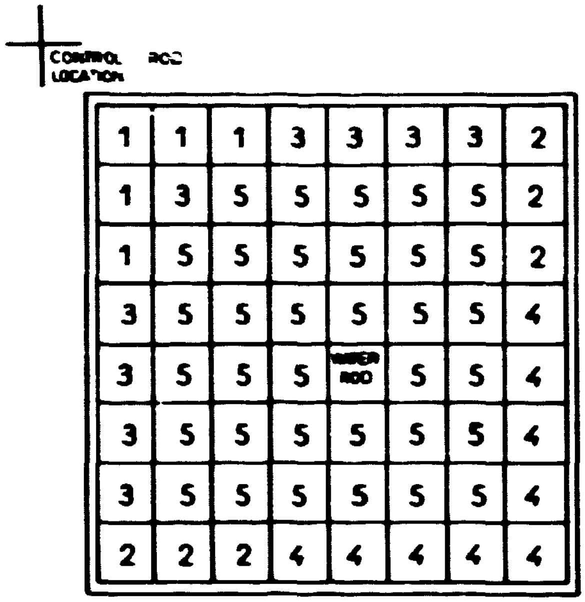


Fig. 2.1 Pin groups

Table 2.1

Data and consequences for the control rod withdrawal

Pin group	Heat load, w/cm 0-15400 h	-17630h	-17654h	P(failure) per node %	Failed pins 0-40 cm axial location	total
1	360	125	430	3.3	2.6	3.5
2	365	260	410	3.0	2.9	4.0
3	350	180	390	0.8	1.1	1.3
4	360	320	390	0.24	0.4	0.5
5	310	280	350	0.02	0	0

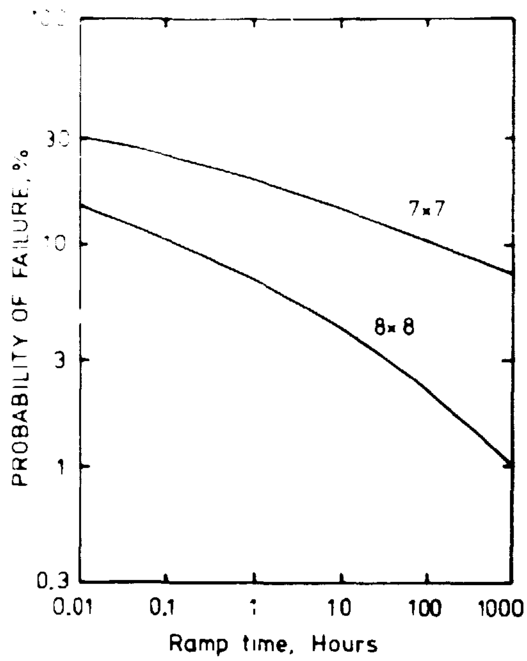


Fig. 2.2 Failure probability as a function of the ramp rate for two designs.

Table 2.2

Design data and reference power history for the fuel pins

Specification	BWR 8x8	BWR 7x7
inner cladding diameter (mm)	(10.8, 0.015)*	(12.42, 0.017)*
cladding thickness (mm)	(0.864, 0.022)*	(0.94, 0.024)*
diametral gap (mm)	(0.228, 0.023)*	(0.30, 0.03)*
density (% af theoretical)	(94.4, 0.66)*	(94.4, 0.66)*
grain size (μm)	(25, 5.0)*	(25, 5.0)*
densification	stable	stable
fill gas/pressure (atm)	1 He	1 He
heat load, 0-8000 h (W/cm)	440	556
heat load, 8000-16000 h (W/cm)	150	193
heat load after the ramp (W/cm)	550	707
fast flux at max. power (n/cm^2)	1.5×10^{14}	1.5×10^{14}
burn-up at the ramp, % FIMA	2.5	2.5
burn-up at the ramp, MWd/tUO_2	20000	20000

* (mean, standard deviation)

1.3 System Reliability

Work in the field of system reliability comprises: development of methods, a doctoral thesis project, and analysis of nuclear power plant incidents.

Development of methods

For calculating reliability characteristics for systems with a high degree of complexity in design or operation, the Monte Carlo method is preferable.

The computer program REDIS is based on Monte Carlo Technique. A new program, MOCARE, based on the REDIS code, has been developed in collaboration with the Electronics Department. The additional features are highly increased flexibility and possibilities for verification of the simulation modelling that directs the key processes of the program.

The increased flexibility of the MOCARE program is obtained by using keywords for the specification of input data types and by using subsystems for the specification of conditions for the occurrence of basic faults and system failures.

The subsystems can be specified by means of reliability diagrams or by means of fault trees, analyzed by the FAUNET program, which are developed by the Electronics Department. The cut sets or tie sets found by FAUNET can be used as input for the MOCARE program via the disc of the computer.

The simulation models can be controlled by a very useful facility, which will register special conditions, specified by subsystems or single faults.

The program can handle a series of different types of fault:

a) Faults with various probability density functions for the time to failure and the repair time.

b) Faults having a constant probability of failure per period of observation.

c) Consequential faults, occurring with a specified probability per event, that can be defined as the failure of a specified subsystem.

d) Faults, which can only occur under certain circumstances, that can be specified by means of subsystems.

The program has been extensively tested, and has proved to be very flexible and in good agreement with other methods of calculation.

It will be further developed in connection with the doctoral thesis project mentioned below.

Optimization of reliability techniques

This project was started in October 1977 and is part of a doctoral dissertation. It is carried out in collaboration with the Electronics Department.

The purpose of the project is to study different techniques for analysis of the reliability of structures and systems.

The work will include a study of different Monte Carlo methods and the use of various variance-reduction techniques such as importance sampling, stratified sampling, etc.

An optimization of Monte Carlo methods as well as of numerical methods will be carried out, and a comparison of their applicability to the analysis of the reliability of structures and systems will be performed.

Nuclear power plant incidents

Incidents in nuclear power plants are analyzed on the basis of the Nuclear Power Experience Documents (NPE). NPE compiles and reports on the operating experience of all large light-water nuclear power plants in the USA. LWRs located in other countries are also included, but not as much information is available on these plants. NPE concentrates on operating problems, equipment breakdowns, malfunctions, outages, etc.

A classification system has been set up. Each incident is registered on punchcards in order to facilitate an automatic analysis of the material by a computer. A total of 18 classification criteria are employed. Some of the classification criteria comprise information such as: Docket Number, Time of Commissioning, Time of Failure, etc., whereas others comprise information in the form of key-words: Primary Component, Cause, Fault Symptom, etc.

Work on a computer program to analyze the data has been initiated. It is the intention to analyze all reports from 1977 concerning BWRs and PWRs first, and then later to work backwards in time in order to cover as large a time interval as feasible.

3. SECTION OF REACTOR PHYSICS

3.0 Introduction

The main topics for the Reactor Physics Section were core-follow studies and studies of the economic optimization of fuel costs for power reactors.

The object of a core-follow study is to simulate the operation of the reactor in order to calculate the dependence of power distribution and reactivity on burn-up, control rod movements, etc. In this field some calculations were performed for a large modern BWR.

Although a program system covering the range from processing of fundamental cross section data to 3-dimensional core simulators already exists, the development of new methods is still necessary. Developments in two areas will be described in the following.

The first method is concerned with the treatment of burnable poison in reactors. A burnable poison is a neutron absorber introduced into the core to deminish the excess reactivity following loading of fresh fuel. By suitable arrangement, burn-up of poison and fuel compensate each other to give constant reactivity until the absorber has totally disappeared. The absorber may be placed in the core in separate structures or, as in the case studied here, mixed with the UO_2 fuel.

The second area of development concerns the setting up of fast methods to solve the three-dimensional diffusion equation. In order to obtain a fast numerical method to solve the diffusion equation, the reactor core has to be subdivided into rather few and thus fairly large regions. If sufficient accuracy is to be obtained, the so-called coarse mesh methods are used. Investigations of two such methods will be summarized in the following.

For fuel management, there is also a complete program system, SOFIE. In the period under review only minor modifications of the principal methods have been made. Nevertheless, valuable experience was gained by running the program system for sveral cases. An example will be given of the specific use of this program system.

This work is performed in cooperation with a utility.

3.1 BWR Calculations

In the previous year, tools for core-follow studies of a BWR were tested and evaluated. Data for the study of a 1500 MWth boiling water reactor and some TIP (Traversing Incore Probe) measurements were made available by a reactor vendor.

Cross sections were generated by means of the CCC-CDB program complex. Starting from a set of 76 group cross sections, fuel pin calculations are made by means of collision probability theory. The cross sections are condensed into a 10-group set in these calculations. The next step is a fuel box calculation. In this the 10-group set is used for collision probability calculations for the fuel pins, while the flux distribution for the fuel box is calculated by a 5-group diffusion calculation. The result is 2-group cross sections which are used for the 3-dimensional BWR-simulater NOTAM, in which the neutronics part is based on nodal theory.

The first three cases to be considered were three critical configurations that were established during the initial fuel loading of the reactor. In the first, the reactor was only partially loaded, while in the other two the core was complete but was kept just critical by different control rod settings. The results for k_{eff} were 1.004, 1.007, and 1.008. These results indicate that the cross section generation system works reasonably well.

The next step is to calculate a three-dimensional power distribution for different power levels such as 60%, 80%, and 100% of full power and at zero burn-up.

For the nodal theory calculations, the reactor core was subdivided into nodes with one node per fuel element horizontally and 25 vertical nodes, giving a total of ~ 12000 nodes. The computing time for the combined neutronics/hydraulics calculation for one case is approximately 3.5 hours on a Burroughs B 6700 computer (which is a factor of 10 slower than a CDC6600).

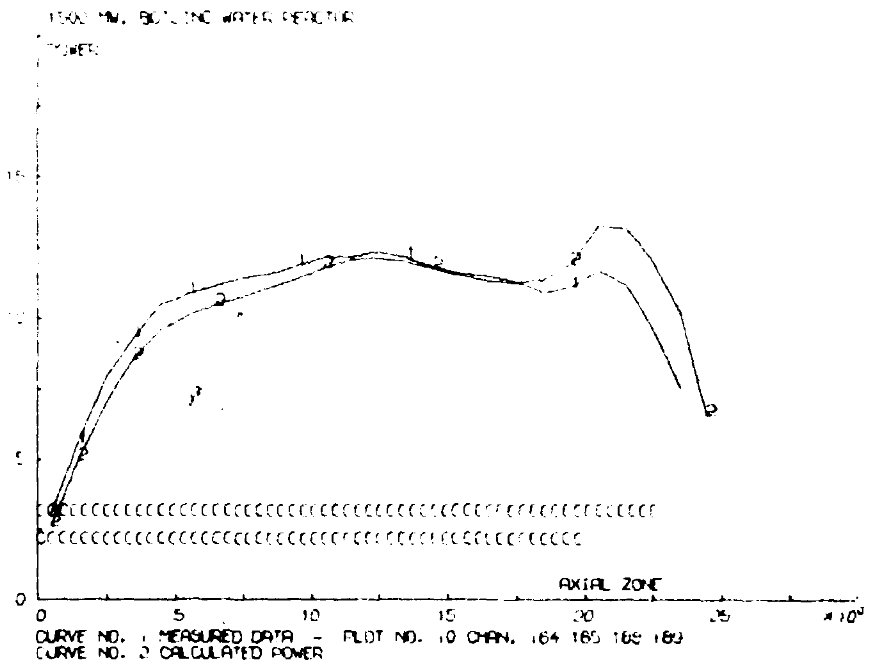

1500 MW. BOILING WATER REACTOR
POWER
AXIAL ZONE
CURVE NO. 1 MEASURED DATA - PLOT NO. 10 CHAN. 164 165 169 169
CURVE NO. 2 CALCULATED POWER

fig. 3.1

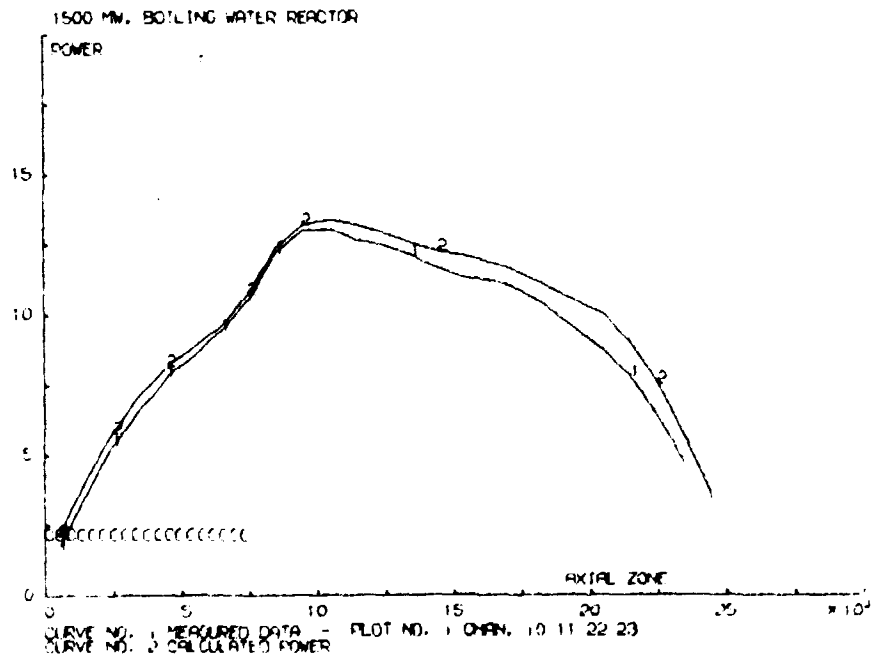

1500 MW. BOILING WATER REACTOR
POWER
AXIAL ZONE
CURVE NO. 1 MEASURED DATA - PLOT NO. 1 CHAN. 10 11 22 23
CURVE NO. 2 CALCULATED POWER

fig. 3.2

At this stage of the calculations studies were made of the influence of parameters such as coupling coefficients, albedos, etc., for the nodal model and of parameters for the hydraulic models.

As an example of one of the final results, comparisons are shown between measured and calculated TIP curves (figs. 3.1 and 3.2). The TIP detectors are placed at the intersection of the narrow water gaps between four fuel boxes. The position in the core is close to the center and near the edge. The calculated curves are taken as the average of the power of the four fuel boxes adjacent to the TIP detectors. The C's on the figures represent the insertion of the neighbouring control rods.

3.2 Burnable Absorber. The AFG-MONSU Program Complex

A program complex AFG-MONSU was designed to calculate the neutron flux in a cylindrical fuel pin cell into which axial heterogeneities are introduced, e.g., in the form of regularly spaced fuel pellets containing burnable absorber.

The theory is based on discrete integral transport theory (DIT) combined with Monte Carlo technique. A superposition principle makes it possible to let the neutrons in the Monte Carlo calculation start only on the surface of the burnable absorber surface. This method improves the efficiency of the Monte Carlo calculation, since most of the neutrons will "score"; i.e., be absorbed.

Zonal burn-up of the poison was introduced into the program, which was tested against experimental results.

As the Monte Carlo calculation is still rather time-consuming, the direct use of MONSU in routine calculations is out of the question, but the program was used to assess other more approximate solutions.

In the combined pin cell and cluster burn-up program CCC, the axial lumping was simulated by concentrating the poison inside an infinite cylinder of smaller radius than the actual radius of the poison pellets, and at the same time the poison concentration is such that the total number of poison atoms per unit length remained the same. As a first guess, a radius was chosen that gives the same poison surface as the spaced poison

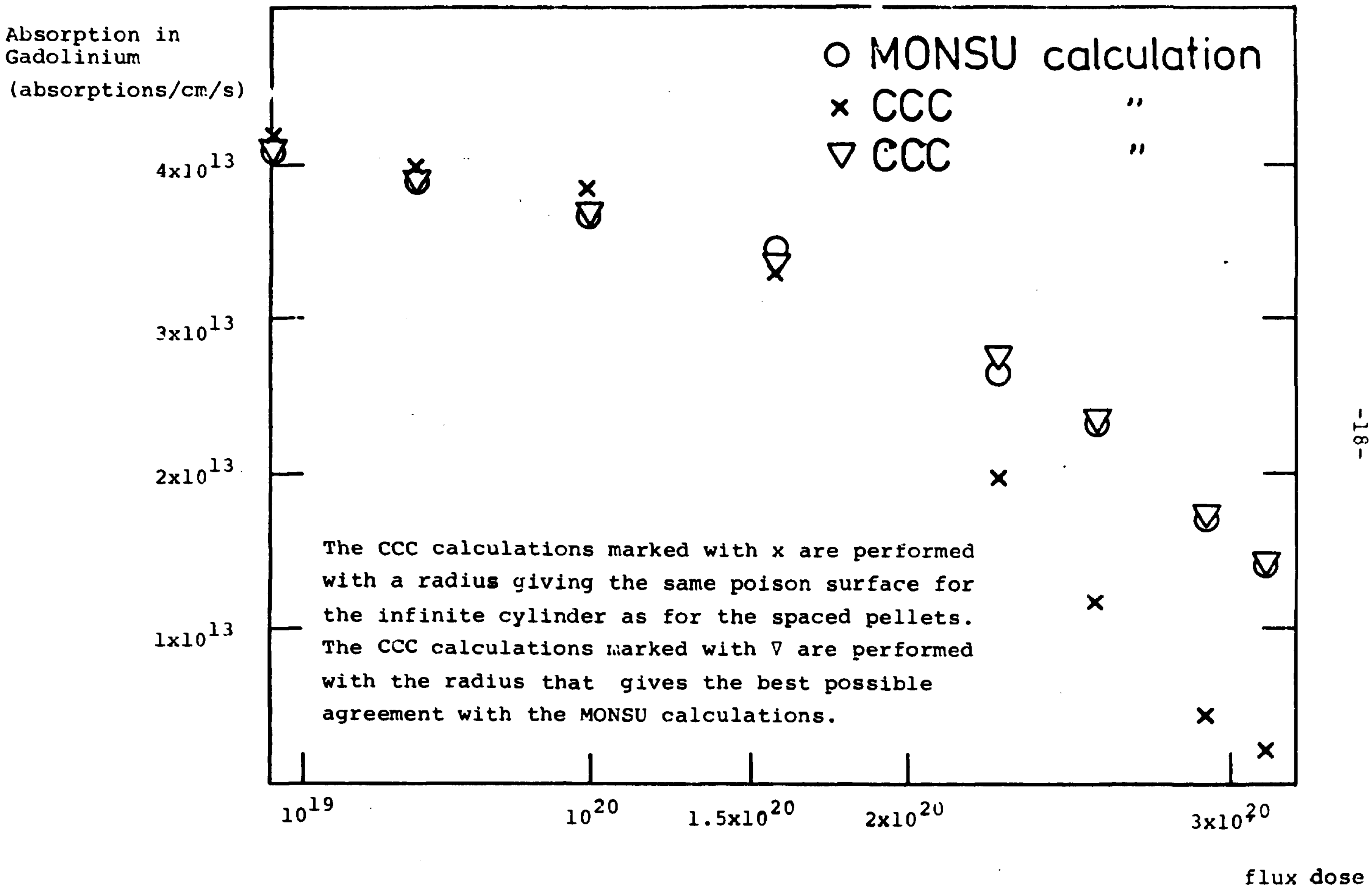


Fig. 3.3 The absorption in gadolinium vs. the flux dose.

pellets. In fig. 3.3 a comparison between a MONSU calculation and a CCC calculation performed according to this principle is given. The figure also shows a CCC calculation where the radius has been adjusted by trial and error to give as good agreement as possible between the MONSU and the CCC calculations. It is seen that excellent agreement can be obtained between the two types of calculation.

3.3 An interface Method for Solution of the Neutron Diffusion Equation

The calculation of the flux distribution for a whole reactor is usually done by means of few-group diffusion theory, and the reactor is typically subdivided into box-shaped elements inside each of which the group coefficients are space-independent. The diffusion equation may then be discretized by means of the finite-difference or finite-element method, perhaps supplemented by some separability assumption. Essentially these methods are concerned with determination of volume parameters, while attention in the newer response matrix methods is shifted towards the element interfaces, across which element interaction takes place. It is, in fact, possible to take the full step and have only interface-defined quantities as unknowns. Investigations of two such methods have begun.

In one of them, Green's theorem is used to transform the diffusion equation into a Fredholm integral equation of the second type. With suitable discretization it seems to be possible to obtain an accuracy comparable to that of the finite-element-method but with a smaller number of unknowns.

In the second approach, the interface flux distributions are, for each element, expanded after a set of functions having simple continuations inside the element, satisfying the diffusion equation here. Some care must be taken to ensure uniform convergence. Preliminary results are encouraging, but an analysis shows that further refinements of the method may prove to be necessary.

3.4 Fuel Management

A typical case that may be solved by the SOFIE fuel management program is the obtimization of the economy for several fuel cycles, for example 12 cycles. In order to do this, one has to specify the length and load factors for each cycle.

When the reactor is actually running, it may turn out that the load factors at the end of some cycles, e.g. 6 cycles, have been lower than expected, and thus lower than specified for the program. If the reactor is loaded according to the original loading scheme at the start of cycle 7, there would be more excess reactivity than necessary for the remaining cycles. In order to save some fuel elements, it is desirable to be able to re-optimize so that the original loading scheme is used for the first 6 cycles, while a new loading scheme is calculated for the following cycles.

SOFIE has a possibility of preserving the data describing the status of the fuel at the end of any cycle. The program may then be restarted at the required cycle, and a new load factor inserted for that cycle.

The economic optimization is based on the determination of the optimal number of fuel elements in a number of pre-defined MOCs. A MOC (Mutual Operating Condition) is a group of fuel elements having the same operating history. By selecting the number of elements in each MOC, the fuelling strategy for a specified number of cycles is defined. In order to obtain a good solution without having too many MOCs, these should be specified in a reasonable manner to represent the type of MOC that previous experience has shown will be used for the final solution.

Thus, in order to utilize the excess reactivity, new MOCs have to be defined. For example, if the original loading scheme uses a MOC where elements stay in the reactor for cycles 4, 5, 6, and 7. At the restart of the program, the burn-up and number of elements available for this MOC must be specified.

Some tests were made to illustrate this method. In the first case, SOFIE was restarted at cycle 7 without defining any new MOCs. In the second case, the elements which have been in the reactor for the previous three cycles, and which would have been discharged in the original loading scheme, remain in the reactor for another cycle. In the first case, the use of the fuel is not optimal, that is, there is still some excess reactivity left at the end of cycle 7, while in the second case the reactor is just critical at the end of cycle 7. This gives a saving of 7 fuel elements for cycle 7.

4. SECTION OF HEAT TRANSFER AND HYDRAULICS

4.0 Introduction

The main efforts of the section are directed towards obtaining an understanding of the thermodynamic and hydraulic phenomena pertaining to nuclear power reactors. This work includes both theoretical and experimental work as a basis for the development of computer models. The section's general knowledge in the fields of heat transfer and fluid dynamics has, however, also been applied in non-nuclear energy fields.

The main working areas are:

1. Reactor Accident Analysis
2. Participation in International Reactor Safety Experiments
3. Reactor-related Experiments
4. Non-nuclear Energy

4.1 Reactor Accident Analysis

The work has included the development of computer codes for the analysis of LOCAs and other severe transients, as well as basic studies in support of the code development. Most of this work has been undertaken within the framework of the NOR-HAV project in collaboration with institutions in the other Nordic countries and the USNRC (US Nuclear Regulatory Commission).

Furthermore, work in connection with the safety analysis of existing reactors and reactors under construction has been undertaken.

A. Basic studies

The study of the basic formulation of the model equations for two-phase flow has been continued. In particular, the RISQUE computer code, which integrates the six conservation equations of a two-fluid model, has been used to investigate various constitutive equations for the dynamic part of the interfacial momentum transfer. Some of the results were reported in a paper to the ANS Water Reactor Safety Meeting, July/August 1977.

It is planned to use the RISQUE code in connection with the Marviken tests on critical flow in large diameter pipes (MXIII-CFT).

B. Blowdown

TINA is a computer code for the calculation of the blowdown phase of a LOCA in a PWR. The program considers the core only, and the proper boundary conditions must be obtained either from experiments or from a separate code, which describes the entire system. The hydraulic model is based upon the subchannel approach, and the two-phase flow is described by a drift-flux model that permits thermodynamic non-equilibrium of the water phase.

During the year the code has been extended to include a more realistic determination of critical heat flux (CHF) and post-CHF heat transfer. The numerical technique has also been improved through the introduction of a new and faster method for the solution of the large block-tridiagonal system of linear equations.

TINA has been used to calculate a blowdown in Ringhals 3, a 2775 MW, three loop, Westinghouse PWR. The results show that the flow redistribution resulting from a radially uneven power distribution does not have a significant influence on the peak cladding temperature. This is in agreement with the conclusion reached by EG & G Idaho, Inc. (TREE-NUREG-1031, Feb. 1977).

Additional calculations are being carried out on the Ringhals 3 reactor to examine whether or not uneven boundary conditions can create three-dimensional effects in the core, which influence the peak cladding temperature to a significant extent. These calculations are made in cooperation with AB Atomenergi, Studsvik, where the boundary conditions required by TINA are produced by means of the RELAP code.

C. Spray cooling

CORECOOL, which is a further development of REMI/HEATCOOL, is a computer program for analysis of top-spray cooling transients in boiling water reactors. CORECOOL was developed in cooperation with the General Electric Company.

During the year CORECOOL has been verified against results from full-scale experiments made by General Electric at their test facility in San Jose and full-scale experiments made by

AB Atomenergi, Sweden, at their test facility in Studsvik, with excellent results. Figure 4.1 shows a typical comparison of the measured and the calculated maximum temperature in the bundle for a simulated BWR/ECCS experiment.

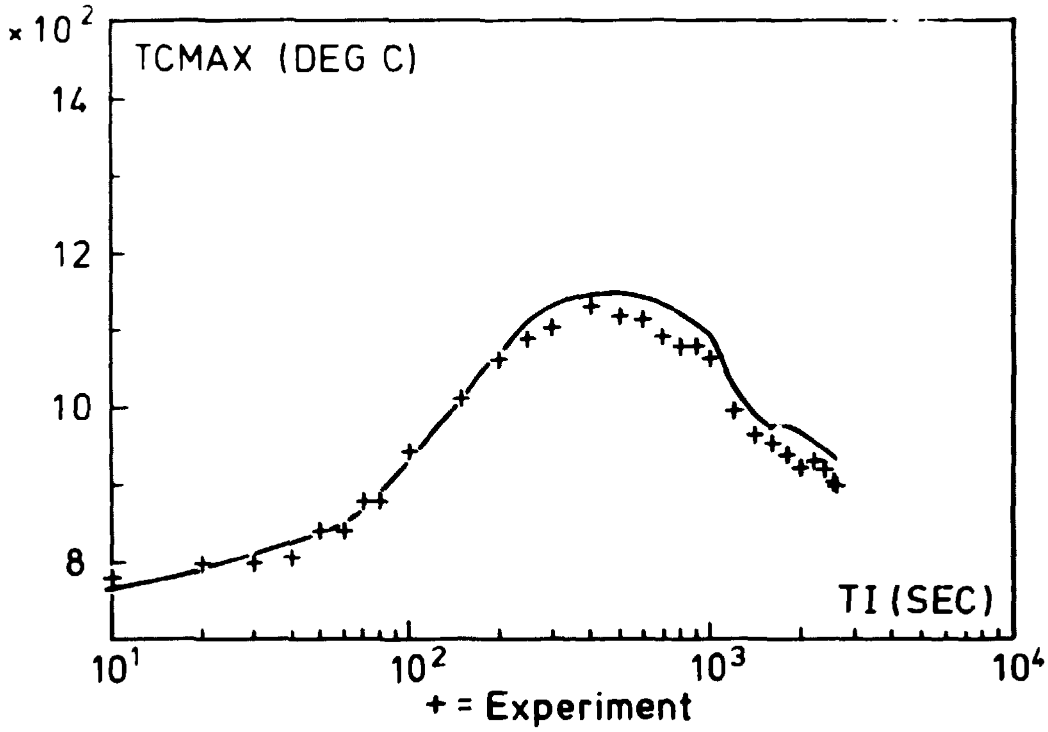


Fig. 4.1

D. Reflooding

The NORCOOL project, which is part of the NORHAV cooperation, consists of the development of computer models for reflooding calculations for boiling-water reactors (BWR).

NORCOOL-I, which was developed during the year, is based on a detailed description of the physical phenomena during reflooding, but it contains a rather simple description of the BWR geometry. NORCOOL-I consists of two basic models, a fuel rod model and a model for the two-phase flow. The fuel rod model is a heat conduction model. The two-phase flow model is based on a solution of the conservation equations for mass, momentum anu energy, and the equation of state. The flow regimes covered by NORCOOL are single phase liquid, bubbly flow, inverse annular flow, film flow and dispersed flow. Thermodynamic equilibrium is not assumed and the steam is allowed to be superheated and the water subcooled. The coupling between the fuel rod model and the two-phase flow model is taken into account through a number of physical models and correlations for the heat transfer, which include conduction, convection and thermal radiation. In particular, 1-2 dimensional axial conduction is included in the rewetting fronts.

The geometrical model in NORCOOL-I consists of one fuel element and a simplified representation of the primary system inside the vessel scaled down to one fuel element. The model for the fuel element is detailed, but the modelling of the surrounding reactor system is fairly simple.

In figure 4.2 a comparison is shown between the measured and calculated cladding temperatures for an electrically heated BWR fuel element during a reflooding transient.

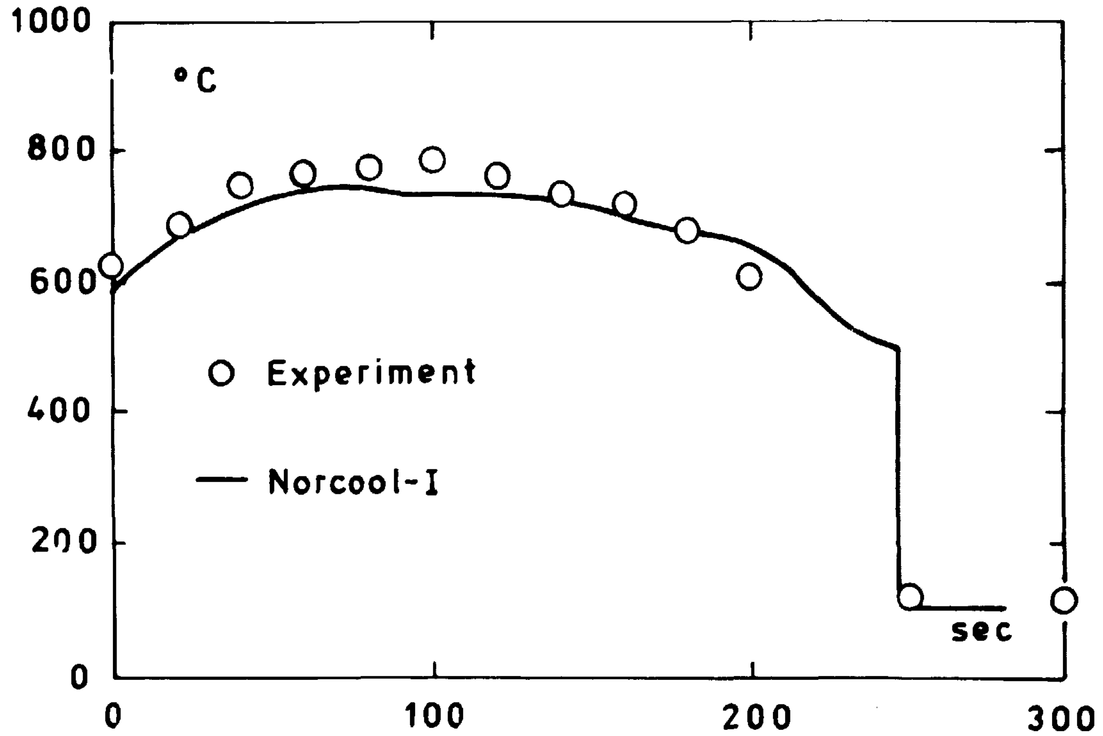


Fig. 4.2

NORCOOL-II, which is presently being developed, is a more advanced version of the reflooding model. The models for the involved physical phenomena are essentially the same as in NORCOOL-I, but NORCOOL-II has an improved numerical technique, and a detailed description of the geometry of the primary system of a BWR. This expecially includes the effect of several parallel fuel elements in the core.

E. Three-dimensional transient calculations for PWR

As part of the accident analysis programme it is planned to extend the TINA code by adding a neutron kinetics calculation to supply the time-dependent total power and power distribution. The resulting computer code will be used for the study of PWR transients where an immediate shutdown of the reactor cannot be assumed, so that the interaction between neutron physics and thermal-hydraulics becomes important. Such incidents could be small break Loss-of-Coolant Accidents, Anticipated Transients without Scram, Control Rod Ejection, etc.

For the neutron kinetics module, the neutron physics part of the ANDYCAP program will be used. It is three-dimensional, based on nodal theory, and was chosen because it is available and in the past has been used for this type of problem (for BWRs).

TINA was originally intended for blowdown calculations for large break Loss-of-Coolant Accidents, and it is therefore expected also to be capable of handling less violent transients. For a start, an attempt has been made to run a number of problems where the transient is initiated by increasing power, simulating a part of a PWR core where one fuel element is represented as a subchannel; apparently this caused no trouble for the code.

The programming of the combination of TINA with neutron physics is presently in progress.

F. Safety analysis

The possible consequences for Danish territory of hypothetical severe accidents with core melt-down in the Swedish Barsebäck BWR are still being discussed. Further work in this connection is under way and will be reported in 1978.

A minor study in connection with the safety analysis of the Swedish Ringhals III reactor was performed for the Swedish authorities (SKI).

G. Advanced electrically heated rods

A project concerning the development of a new type of electrically heated rod for reactor safety experiments was proposed to EEC Working Group No. 2. The idea is to develop and do experiments with an electrically heated rod that more closely simulates the transient thermal behaviour of a real fuel rod. The design is to be based on technology available at Risø, including the production of annular UO_2 pellets.

4.2 Participation in International Reactor Safety Experiments

A. The Marviken containment response tests, MXII-CRT, which constitute the second series of full-scale containment experiments at Marviken, Sweden, were terminated. This was an international cooperation project with the primary objective of obtaining experimental results on containment pressure oscillations.

B. TECPO (Theoretical Efforts on Containment Pressure Oscillations) is a joint Nordic project associated with the MXII-CRT project. The theoretical investigations were supported by blowdown experiments in a simplified small scale model of the Marviken pressure suppression containment. The final reporting of the project was practically completed by the end of the year.

C. The Marviken critical flow tests, MXIII-CFT

In this third series of tests in the Marviken facility critical flow rates in large diameter pipes are being investigated. The nozzle diameter will be between 0.2 and 0.5 m. The stagnation pressures will range up to 5 MPa and the liquid subcooling will be up to 30 K.

The experiments are carried out as an international project with participation from Denmark, Finland, France, the Netherlands, Norway, Sweden and the USA. One member of the staff of the Department is stationed in Sweden, while another is on the Technical Review and Advisory Committee (TRACE)- The preparations for the tests have been slightly ahead of schedule and the first shake-down test was successfully run in December.

4.3 Experiments

The theoretical work was supported by experiments performed by the experimental group (SEHT).

A. High pressure water loop
B. Annular steam-water flow in tubes and annuli
C. Inverted annular film boiling during the refloodi.ng phase

A. High pressure water loop

The loop was in regular service throughout the year. Late in 1976 the remaining problems concerning noise from the thyristor-regulated power supply were solved, and the loop was put into service in January 1977.

The loop has run 4 days a week, Monday-Thursday from 0700 to 2000, while Fridays are used for modifications, repair and calibration to test equipment.

A minor incident with the main pump caused a two-week shut-down in March 1977. In close co-operation with the pump manufacturer, a minor modification was made by the SEHT staff.

Four different test sections have been used: $10^{\varnothing}$ tube 9 m long, $20^{\varnothing}$ tube 9 m long, and $26^{\varnothing}/17^{\varnothing}$ annulus with two different lengths 3.5 and 9 m.

B. Annular steam-water flow in tubes and annuli

During 1977 more than 250 film flow experiments were carried out in the high pressure loop.

The experiments were performed with steam-water at 30, 50, 70, and 90 bar under both adiabatic and diabatic conditions. More than 200 measurements of the film flow rate were carried out together with measurements of pressure gradients, film thicknesses, wave frequences and velocities, and burnout heat fluxes.

The adiabatic experiments were carried out under conditions that allow the data to be regarded as equilibrium data. The equilibrium measurements in annuli showed an asymmetric film flow condition, where the tube film carried considerably more liquid per unit perimeter than the rod film. In diabatic experiments it was shown that the diabatic film flow is independent of the subcooling, as long as there is no steam at the inlet. It was demonstrated that burnout takes place at the axial position where the film flow vanishes. The wave measurements gave an indication of proportionality between the wavelength of the roll waves and the film thickness.

On the basis of the experimental data, a film flow model for annular flow in tubes and annuli was set up. It was shown that the velocity profile in the film could be described adequately well by Prandtl's turbulent two-layer model. The velocity distribution in the gas core was described by the turbulent, logarithmic profile for completely rough walls. A general film roughness correlation between the roughness and the film thickness was derived. By the introduction of a new entrainment parameter, a general entrainment correlation was shown to be valid for both air-water at low pressure and steam-water at 30-90 bar.

The capability of the model is demonstrated by comparisons shown in figs. 4.3 - 4.6.

In fig. 4.3 measurements of film flow rates in the tubular test section with an inner diameter of 10 mm are shown together with the calculations. Here the film flow rate, in per cent of the total flow rate, is shown versus the outlet steam quality with the mass flux as parameter. A similar comparison between film flow measurements in the long annular test section and predictions is shown in fig. 4.4.

Experimental values of the frictional pressure gradient are compared to the calculations in fig. 4.5, and examples of predictions of burnout are shown in fig. 4.6. Here the experimental variation of the steam quality at burnout with the system pressure is compared with the theoretical variation.

C. Inverted annular film boiling during the reflooding phase

To describe the heat transfer during an emergency core cooling after a Loss-of-Coolant-Accident (LOCA) in a light water reactor, it is necessary to analyze the heat transfer on vertical surfaces under annular film boiling.

Experimental and theoretical work in this field concerning inverted film-flow has been started. The experimental part consists of the development of various methods of void measurement in two-phase flows using γ-rays or x-rays and hot-film or hot-wire constant temperature anemometry. For preliminary experiments, test sections consisting of glass tubes containing a two-phase flow of nitrogen will be used. Later experiments will be based on heated steel tubes using water as a flow medium.

A better understanding of the heat transfer taking place in the reactor core during the reflooding phase involving inverted film boiling would lead to a better prediction of how fast it is possible to rewet the reactor core after a postulated accident.

D. Temperature calibration laboratory

For several years the section has calibrated its own temperature sensors (thermocouples and resistance thermometers). Excellent temperature calibration equipment has been collected over the years, and this is now installed in a special laboratory with temperature control.

The laboratory has available:

Temperature fixed point cells:

0°C	Melting ice
100°C	Boiling water
444.7°C	Boiling sulphur

Thermostats:

-40 - +50°C	(Ethanol)
0 - 80°C	(Water)
50 - 300°C	(Oil)

200 - 650°C (Molten salt)

500 - 1100°C (Electrically heated furnace)

Resistance measurements:

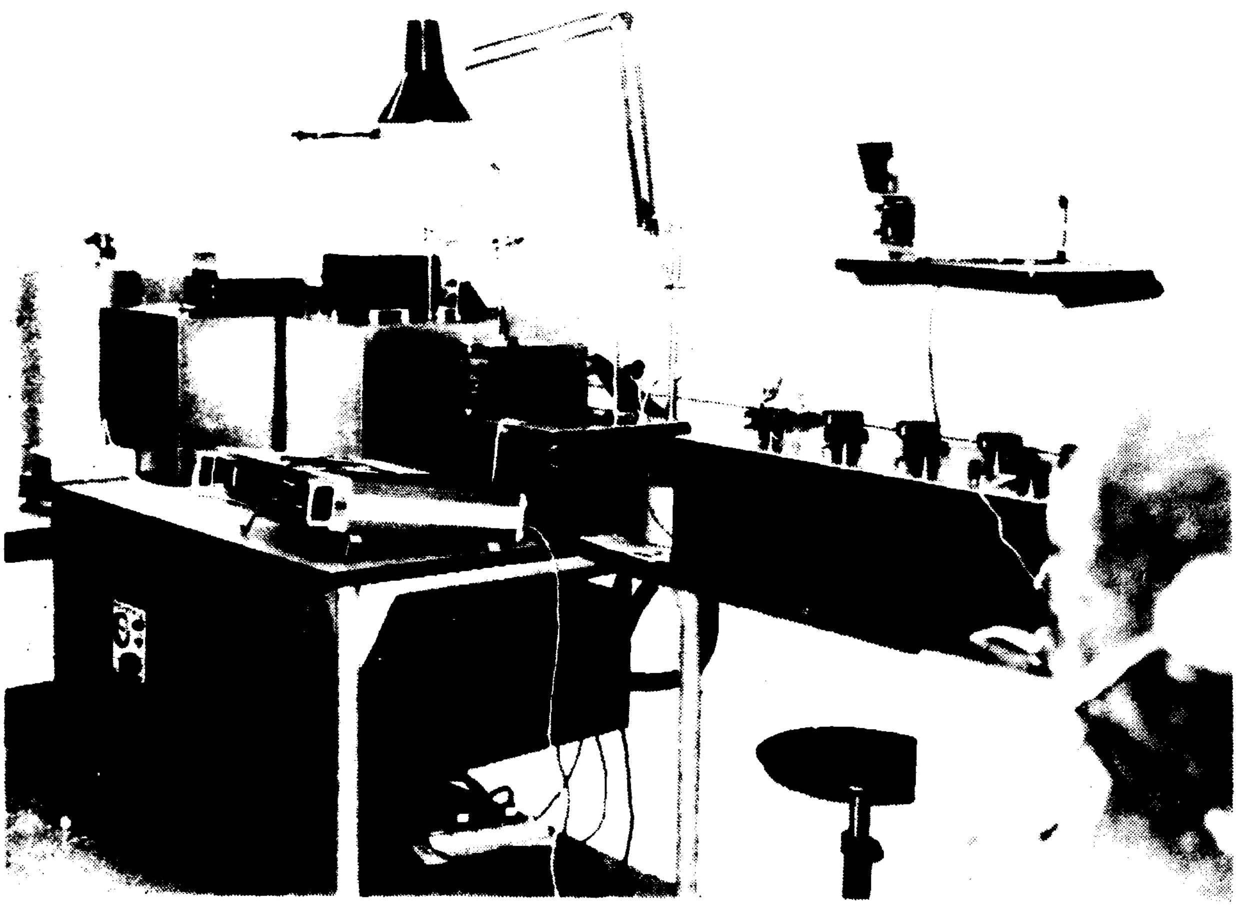
Müller bridge: 0-111Ω ± 0.0001Ω

Voltage measurements:

Dieselhorst Potentiometer: 0-111mV ± 1μV

Two platinum resistance thermometers and a Pt/PtRh thermocouple have been recalibrated and certified at Statens Provningsanstalt, Borås, Sweden, and the electronic equipment has been certified at Elektronikcentralen and SAS Normallaboratorium with traceability to NPL, England, and PTB, Germany.

An application for a temperature calibration authorization was forwarded to the Danish National Testing Board in August 1977. The authorization is expected in the spring of 1978.



Temperature Calibration Laboratory

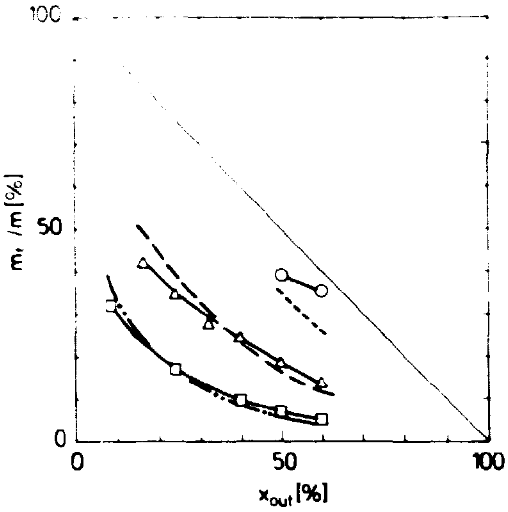


Fig. 4.3. Comparison of theoretical and experimental film flow rates in percent of total flow rate (m_f/m) as function of outlet steam quality (x_{out}) with mass flux (G) as parameter. Tubular test section 10$^\varnothing$ mm. Adiabatic. System pressure 90 bar.

G(kg/m^{2}s)	Exp. value	Cal. value
500	o	– – – –
1000	Δ	—— —— ——
2000	□	—— · · ——

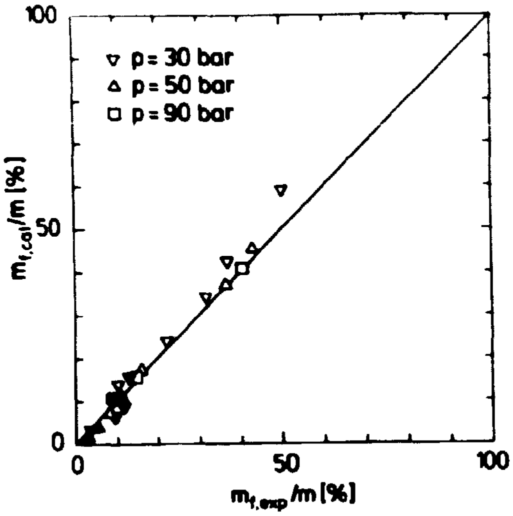


Fig. 4.4. Comparison of theoretical and experimental film flow rates in percent of total flow rate (m_f/m). Annular test section 17$^\varnothing$/26$^\varnothing$ mm. Adiabatic. System pressure (P) 30, 50 and 90 bar. Solid symbols apply to the rod film, open symbols apply to the tube film.

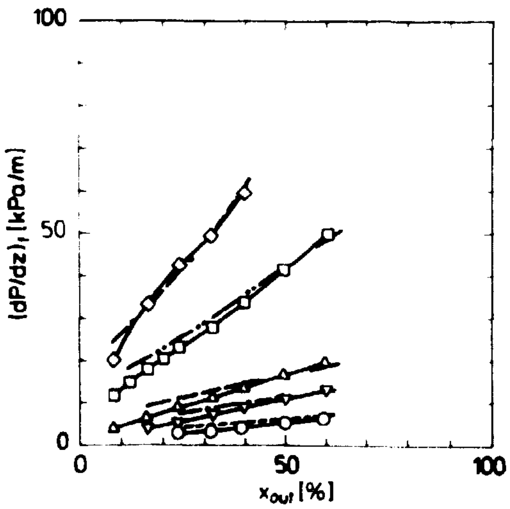


Fig. 4.5. Comparison of theoretical and experimental frictional pressure gradients ($(dp/dz)_f$) as function of outlet steam quality (x_{out}) with mass flux (G) as parameter. Tubular test section 10$^\varnothing$ mm. Adiabatic. System pressure 70 bar.

G(kg/m^{2}s)	Exp. value	Cal. value
500	o	– – – –
750	▽	—— · ——
1000	Δ	—— —— ——
2000	□	—— · · ——
3000	◇	—— —— ·

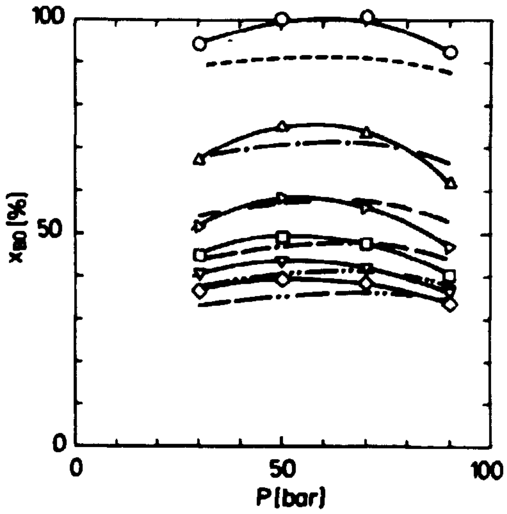


Fig. 4.6. Comparison of theoretical and experimental burnout steam quality (x_{BO}) as function of system pressure (P) with mass flux (G) as parameter. Tubular test section 10$^\varnothing$ mm. Diabatic. Heated length 4 m.

G(kg/m^{2}s)	Exp. value	Cal. value
500	o	– – – –
1000	Δ	—— · ——
1500	▷	—— —— ——
2000	□	—— —— ·
2500	▽	—— · · ——
3000	◇	—— —— · ·

4.4 Non-nuclear Energy

A. Solar heating and long-term heat storage

Various methods of solar heating of buildings were studied. For the purpose of evaluating and comparing the performance and economy of different systems, the American SOLSYS code was imported and equipped with Danish weather data (the reference year). A special type of trickle plate collector has been proposed and pretests are in progress.

In order to study seasonal low-temperature heat storage in soil, a computer code CONDUC was completed. With this code the performance of soil heat reservoirs of various sizes and shapes has been simulated, using an anticipated pattern of heat flow to and from the reservoir. This pattern was based on the estimated heat consumption of private houses provided with flat plate solar collectors. The reservoir temperature ranged from 30°C to 90°C.

The results indicate that 65% to 70% of the stored heat may be regained from a top-insulated 2,500 m^3 reservoir (a hemisphere with radius 10.5 m). This would thus supply around 40% of the heat required for the winter season (Nov.-March) of 8-10 houses with pre-oil-crisis standard insulation. Ground water flow was disregarded in the calculations.

Large-scale heat storage in aquifers is studied in a joint project in collaboration with Risø's Engineering Department, laboratories of the Technical University of Denmark, and with the Geological Survey of Denmark.

In aquifer heat storage the heat capacity of sand, gravel and water is exploited, and the method should be suitable for seasonal storage of district heating water from combined power plants.

A project proposal for an aquifer pilot plant of 10^5 m^3 with a capacity of 2700 Gcal has been prepared. The section will contribute with computer modelling of aquifer performance. The project is expected to start in the spring of 1978 financed by the Ministry of Commerce.

B. Wind power

The section has taken part in the official programme for the development of wind power in Denmark. Three of Risø's departments collaborate as consultants for the structural design of rotor blades for the windmills. In August 1977 the preliminary studies for the rotor design were reported. Since then detailed design work has been done on the rotor blades. The two experimental windmills, each with a diameter of 40 m and a power of 600 kW, will be erected in Jutland in the spring of 1979.

5. SECTION OF DYNAMICS

5.0 Introduction

Following upon a reduction in the number of Staff, activities in this Section have had to be limited.

The main area of work is the development of models for the dynamics of nuclear power plants. For BWR plants, both one-dimensional models for the whole plant and a three-dimensional model for the reactor core have been developed earlier. Improvements were made on these models and test calculations performed. Concerning PWR plants, no work was done this year on the one-dimensional model mentioned in the previous annual report, but the development of a three-dimensional core model was taken up in cooperation with the Section of Heat Transfer and Hydraulics. Furthermore, a program for static calculations of parameters and variables for a steam turbine at different power levels was improved, so that it can now be used for superheated steam.

Efforts were concentrated on two subjects: Calculations for an underground natural gas storage facility, and a review of models used for transient calculations in General Electric's preliminary safety analysis report GESSAR.

5.1 Calculations for a Natural Gas Storage Facility

In connection with the preliminary design of an underground storage for natural gas, the thermodynamic properties essential for the compression-expansion process were calculated on the basis of procedures developed at the Technical University. Some of these properties are utilized in a model of the storage facility and the high pressure transmission lines.

The model is used for calculation of the storage requirements and associated pressure and temperature variations in the storage at different load conditions. The model covers both the storage and the transmission line, and the load is determined by assumptions for the gas consumption and the production strategy.

The storage model takes into account temperature variations caused by the compression - expansion process both in the storage cavern and in the equipment at ground level, as well as heat exchange with the surrounding rock salt. Furthermore, calculations of the compression and cooling power during loading and the heating power during unloading are performed.

The work is carried out as part of a commercial project.

5.2 Review of GESSAR transients for the Danish Inspectorate of Nuclear Installations

Models and procedures for calculation of transients for the safety analysis in chapter 15 of GESSAR were reviewed as far as possible from GESSAR itself and from model descriptions in references. (Loss of coolant accidents were not included). It was found that the majority of transients was calculated by means of the Linford model that utilizes point kinetic calculations and a two-node model for the core hydraulics. The general conclusion was that the model and the verification is insufficient, and that more detailed documentation and calculations would be desirable in connection with Danish construction permits.

6. THE DR 1 REACTOR

6.0 Introduction

On 15th August 1977 this reactor reached the age of 20 years. DR 1 is a homogeneous solution-type reactor fabricated by Atomics International. The maximum effect is 2 kW and no fresh fuel has been added since the start-up in 1957. Samples taken from the core solution indicate that there is little corrosion of the core container.

During the year more than 50 students from various universities have operated and carried out experiments at the reactor.

6.1 Neutron Radiography

The reactor is mainly used as a source for neutron radiography of uranium fuel pins. Figure 6.1 shows a non-typical picture with some hydride spots.

The content of liquid freon in thin capillary tubes was determined by neutron radiography using gadolinium foils (fig. 6.2). This work was done for Danish industry.

In close co-operation with the Metallurgy Department, experimental work on cellulose nitrate plastic was carried out in an attempt to improve the imaging pictures by varying the irradiation and the etching times.

6.2 Neutron Metrology

The need for measuring the activities of thermal and fast flux monitors seems to be decreasing. For this reason the equipment used for these measurements, the ionization chamber, germanium detector and beta-gamma coincidence counter determination, will be taken out of service, but maintained in working order.

More than 100 aluminium wires containing a small amount of cobalt were measured by a scintillation counter in order to calculate the neutron flux for the Si-rig in DR 3.

6.3 Pile Oscillator

The equipment is now in routine use. It is particularly suited for determination of neutron-absorbing agents in samples. It was used to determine the boron-equivalent in zircaloy, and the boron content in some metal glasses.

6.4 Mössbauer Effect

Through an agreement with the H.C. Ørsted Institute, Roskilde University Center, and Risø, the velocity spectrometer was presented to Roskilde University Center.

Fig. 6.1 Uranium pin with spots ot zirconium hydride.

Fig. 6.2 Freon in a capillary tube.

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1. April 1978

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