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Department of Reactor Technology
Annual Progress Report
1 January - 31 December 1979

Risø National Laboratory, DK-4000 Roskilde, Denmark
September 1980

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DEPARTMENT OF REACTOR TECHNOLOGY
ANNUAL PROGRESS REPORT
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Abstract. The activities of the Department of Reactor Technology at Risø during 1979 are described. The work is presented in five chapters: Reactor Engineering, Reactor Physics and Dynamics, Heat Transfer and Hydraulics, The DR 1 Reactor, and Non-Nuclear Activities. A list of the staff and of publications is included.

INIS-descriptors. FLUID MECHANICS, FUEL MANAGEMENT, HEAT TRANSFER, REACTOR PHYSICS, REACTOR TECHNOLOGY, RESEARCH PROGRAMS, RISOE NATIONAL LABORATORY.

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1. INTRODUCTION

The Three Mile Island accident took place in the spring of 1979, and the Department of Reactor Technology took part in the review of this accident on request from government authorities. The basic research and development work within reactor technology was not changed much by the accident; the reason for this was that work on related subject fields was already being pursued, and new effects called for more manpower and money. Proposals for studies relevant to TMI and to large accidents were made in Nordic and CEC committees, cevering subjects like: small break codes, qualitative transient analysis, analysis of licensee events reports, filtering effects in houses, and cleaning of soil and houses. Only the first of these subjects received support, however, and Nordic co-operation is now foreseen from the beginning of 1981.

The research and development work within reactor technology showed steady progress during the year although the limits of the resources are being more strongly felt. A contract with Danish utilities on the fuel management code system was signed. In the Nordic LOCA-ECC analysis work (NORHAV), the BWR EEC code, NORCOOL I, was completed, and a good step forward was taken on the advanced, multi-channel code NORCOOL II. NORCOOL I is to go into the American WRAP packet. In the thermo-hydraulic field of accident analysis, Risø is also using the American TRAC (-P1A) code in the pre-prediction exercise for the ISPRA's blowdown experiments (LOBI).

Under contract with ISPRA an analysis of American "Licensee Events Reports" was made. This work was part of the efforts of the Department within the field of reliability and safety assessment, where the major application at the moment is non-nuclear. Two of the tasks here were: a safety assessment for Danish oil and gas production platforms in the North Sea, and a risk analysis for a chlorine production plant in Copenhagen.

During the year quite a number of jobs emerged, not directly using the methods developed in nuclear R & D, but related to the experience or methodology in the Department. Amongst these were calculations of the temperature distribution and brine migration around a waste repository.

The efforts in non-nuclear, alternative energy R & D increased during the year. The testing station for small windmills was completed and testing of six windmills began. The theoretical, thermo-hydraulic work on aquifers and geothermal reservoirs reached a substantial level.

Finally, the staff of the Department took part in the work of governmental committees on energy and energy research planning.

2. REACTOR ENGINEERING

The work of the Reactor Engineering Section is concentrated on developing methods for assessing the reliability of systems and components in nuclear power plants. Furthermore, a core simulator is being developed and nuclear power plant incidents are analysed as part of a general study of conditions and limits for operating reactors (see 2.1 and 2.2).

The Reactor Engineering Section has been involved in safety assessment of nuclear power plants for several years. This work combined with methods and tools developed within the reliability fields, has proved very valuable in connection with safety and risk analysis of industrial, non-nuclear installations. Accordingly, the Section is heavily engaged in projects of this kind and has participated in an analysis of offshore oil and gas platforms, (see 2.3) and a chlorine production facility (see 2.4).

Research and development within the area of reliability was concentrated on the reliability of fuel elements (see 2.5) and optimization of reliability techniques (see 2.6).

The work within the CSNI Group of Experts on the Reliability of Mechanical Components and Structures was terminated with a final report to the CSNI. The general conclusion was that available data (in the form of event reports) were not sufficiently detailed to enable an identification and quantification of rare events such as failure to scram due to mechanical failure in the control rod drives or inadvertently opening of a safety/relief valve. Finally the CSNI group recommended that CSNI should sponsor a structural reliability benchmark exercise.

2.1. Core Performance Evaluation. The Core Simulator

Operational restrictions are imposed on light water reactors in order to avoid fuel failures, or at least diminish the number of such failures. As these types of restrictions necessarily result in reduced power production from the reactors, they are undesirable from an economical point of view. Knowledge of the local power ramps and their consequences for the fuel is required in order to reduce the level of restrictions consistent with safety requirements.

A comprehensive system for the calculation of the failure probability for the individual fuel rods throughout the reactor core in light water reactors is being developed. The calculational system is set up as a modular one. The modules to be included are:

1. A 3D-nodal neutronic/hydraulic module for the calculation of the 3D power distribution (based on the codes ANTI and NOTAM

2. A fuel box module for the calculation of homogenized cross-sections of the individual fuel boxes (The code CDB). mentioned later).

3. A module for the calculation of the power history of each individual fuel pin.
4. A fuel reliability module for the calculation of the failure probability of the individual fuel rods (The code FRP).

At present, work is performed on the individual modules to be included in the system. This first part of the project is carried out under a collaboration agreement with a Danish utility.

In connection with the core simulator project a study was begun in July 1979 as a part of a Ph.D dissertation on the determination of the power histories for the individual fuel pins. As a first approximation, the local pin powers can be estimated on the basis of the fuel box calculations performed in order to generate the box-average cross sections for the 3D-nodal calculations. However, it is intended that a more sophisticated method be developed utilizing the boundary conditions obtained from the 3D-nodal calculations.

2.2. Nuclear Power Plant Incidents

In the legislation of most countries regarding nuclear power plants it is specified which types of incidents have to be reported to the responsible authorities. The first reporting of an incident is normally performed by means of some sort of standard data entry sheet. Typically, several incidents are reported per reactor each year. An example of such a reporting system is the American "Licensee Event Reports" (LERs). A computer-based data file was established in 1973. This data bank provides a centralized source of information that may be used for quantitative assessments of off-normal events at nuclear power plants. In 1978, as an example, 2400 incidents were reported. In addition to the official reporting system, a commercial compilation of operating experience is found in the US, called The Nuclear Power Experience Documents (NPE).

On the basis of the above-mentioned Nuclear Power Experience Documents a feasibility study was undertaken with the purpose of analysing what can be learned from a systematic computer analysis of the compiled information. The ultimate goal is to provide feedback from the experience of system component performance in a form faciliting detailed statistical analysis; this could benefit suppliers, utilities, and authorities. In the present study 772 incidents reported for PWRs in 1977 were analysed.

Another study concerning the analysis of Licensing Event Reports was carried out under contract with ISPRA. A US Licensing Event Report contains coded information as well as two descriptive items: event description and cause description. In particular, this study was dealing with the possibility of analysing, by means of a computer, "as is" information found in the descriptive items. This study was partially successful as satisfactory results were obtained with regard to retrieving LERs concerning component or equipment failure. On the other hand, when human errors are involved, the present system is inadequate. This is due to the typical implicit rather than explicit description of personnel errors.

In general, it can be concluded on the basis of the studies carried out that the compilation of LERs comprises a comprehensive and useful source of information, and valuable lessons can be learned from analyses of this information. However, it would be useful if there were international standardization of the reporting schemes, and much more work has still to be done on analytical methodology.

2.3. Safety and Risk Analysis - Oil and Gas Production Platforms

The work on the oil and gas production facilities in the Danish part of the North Sea was accelerated during the period. The oil production facilities at the GORM-field received preliminary approval by the authorities, while the documentation for the gas production facilities at the CORA/BENT-field was delivered.

The Danish Energy Agency (DEA) requested a risk analysis for the GORM-facility as part of the safety documentation. This analysis was carried out by Det norske Veritas (DnV) and Risø. The GORM-field is situated approximately 200 km west of the Danish coast at a water depth of 40 meters. The complex consists of two well-head platforms and a production platform containing processing equipment, utilities, living quarters and a helideck.

In the analysis, hazards to the platforms and its crew were established based on experience from other offshore installations. A set of safety and acceptance criteria's were suggested concerning

- Health and safety of the crew
- Environment
- Major economical losses

The list of hazards analysed were

- Blowouts
- Fire and explosions
- Earthquake
- Ship collision
- Dropped objects and helicopter crash.

These hazards were imposed on the platforms, and their ability to withstand them was assessed. Where necessary, recommendations were put forward.

As a consultant to DEA the adequacy and acceptability of a similar risk analysis for the CORA/BENT-field were assessed. Recommendations for additional analyses and design limits were put forward to the authorities.

2.4. Risk Assessment of a Chlorine Production Plant

A risk assessment was performed of the chlorine production plant at "Danish Soyacake Factory" in Copenhagen on a contract for the municipal environmental authorities (NIELSEN et al., 1980). Chlorine is produced by electrolysis of salt from Danish mines, and the total annual production of chlorine is about 20000 tons, which covers the entire domestic consumption. The risk assessment, comprised a general analysis, a detailed analysis, and a list of possible measures against accidents.

The general analysis showed that the chlorine storage and connected piping comprise the plant areas in which the maximum escape of chlorine can take place. The chlorine storage consists of two basement rooms with four tanks, each containing 20 tons, and four other tanks, each containing 27 tons. A large break in a storage tank can cause an escape of several tons during a few seconds. Breaks in the connected piping can cause leakages of more than 3 tons of chlorine.

The detailed analysis comprises an evaluation of possible causes of breaks in the chlorine storage tanks and the connected piping in addition to an assessment of the corresponding probabilities. Amongst the possible causes for breaks studied in detail was corrosion of the equipment. Moist chlorine is extremely corrosive, and may reach penetration velocities of several millimeters per month in the construction material in question, whereas dry chlorine just creates a thin surface layer of $FeCl_3$.

The study indicated that few investigations have been performed concerning the corrosive properties of moist chlorine, but useful investigations had been performed at the Danish Soyacake Factory and the Technical University of Denmark. The investigations of the possible ways of ingress of moisture showed much inherent safety in the system. In fact, it is considered very unlikely that moisture should enter the system without being detected either by instruments or by relatively harmless malfunctions of certain components.

Extensive measures have been taken in the design and construction as well as in the operation of the chlorine storage tanks in order to reduce the risk of leakages and breaks. The tanks are designed with no pipe penetrations of the shroud and end pieces, all penetrations being in the manhole cover. In addition, the tanks were manufactured from a steel which is ductile even at - 40°C, and they are designed corresponding to an extra safety factor of approximately 1.7 beyond what was required according to the codes and regulations. It is estimated that the additional safety factor will reduce the probability of failure by one or two orders of magnitude. In connection with the manufacturing, a series of analyses of the steel was performed and the welds were controlled by means of x-rays and ultrasonics. Following the installation, the tanks were pressure tested and visually inspected. In addition, each storage tank is inspected visually every third year on a routine basis.

No statistical data are available which could serve as a basis for a more exact assessment of the probability of failure of the chlorine storage tanks in question.

Until now chlorine storage tanks of modern design have accumulated approximately 300,000 tank years in all and only one failure - a minor one - has occured. This corresponds to a "probability" of a minor failure of $3 \cdot 10^{-6}$ per tank per year, and a "probability" of a serious failure of maximum $3 \cdot 10^{-6}$ per tank per year.

The quality assurance measures taken during the manufacturing of the chlorine tanks at "Danish Soyacake Factory" are at least as comprehensive as those of the tanks, covered by the above statistics.

A comprehensive analysis was performed of the possibilities of leakages from the storage tanks through the piping system. The probabilities were all estimated to be very low; however, in some cases, for instance, regarding operational errors, it was not possible to assess the probabilities numerically.

In case of a large break in one of the above-mentioned pipes, chlorine concentrations above 600 ppm will probably arise at the fence. Chlorine concentrations above 1000 ppm are lethal within a few seconds. However, chlorine possesses the attractive property that its smell can be detected and it thereby give a warning down to approximately 0.1 ppm - far below the level, where human life is endangered.

The environmental authorities in Copenhagen themselves are analysing the effects upon the environment from leakages in the chlorine production plant, and in this task they gain support in the assessment of the initial dispersion, which is presented in NIELSEN et al. (1980).

In this reference a list is also included comprising possible measures for the prevention and limitation of accidents. One of these is an installation of automatic excess flow valves in all pipes connected to the chlorine storage tanks, and these valves are already being installed. No economic evaluation of these measures has been performed and the list of measures does not imply that any decision has been made concerning which measures should be carried out.

REFERENCE

NIELSEN, D.S. et al. (1980). Risikovurdering af Dansk Soja-kagefabriks kloralkalianlæg (Risø, April 1980) 114 pp.

corrosion, creep rupture, overstress, and overstrain. Furthermore, the correlation between the power shocks and the failure probability was considered.

The analysis showed that the stress corrosion failure criteria, based on out-of-reactor experiments performed on irradiated zircaloy with iodine present, provides reasonable agreement between calculated and observed failures. This is illustrated in the following table, where the out-of-reactor failure limits X_L are compared to the failure limit found by regression analysis of a large number of irradiation experiments.

Table: Results of the regression analysis of 39 ramp experiments.

	X_L* out-of-reactor	X_L* in-reactor	ρ coefficient of correlation	ρ^2
Stress corrosion	(225,18)MPa	(220,38)MPa	0.86	0.76
Creep rupture	(2.5,0.5)%	(0.32,0.08)%	0.65	0.42
Overstrain	(0.21,0.04)%	(0.1,0.02)%	0.94	0.88
Overstress	(500,25)MPa	(234,20)MPa	0.95	0.90
Overpower	-	(161,27)w/cm	0.89	0.79

*(a,b) = (mean value, standard deviation)

2.6. Optimization of Reliability Techniques

Importance sampling can be used as a variance reduction technique in Monte Carlo simulation of system reliability. One of the problems is to find a weighting function which is best in the sense that it gives the greatest variance reduction.

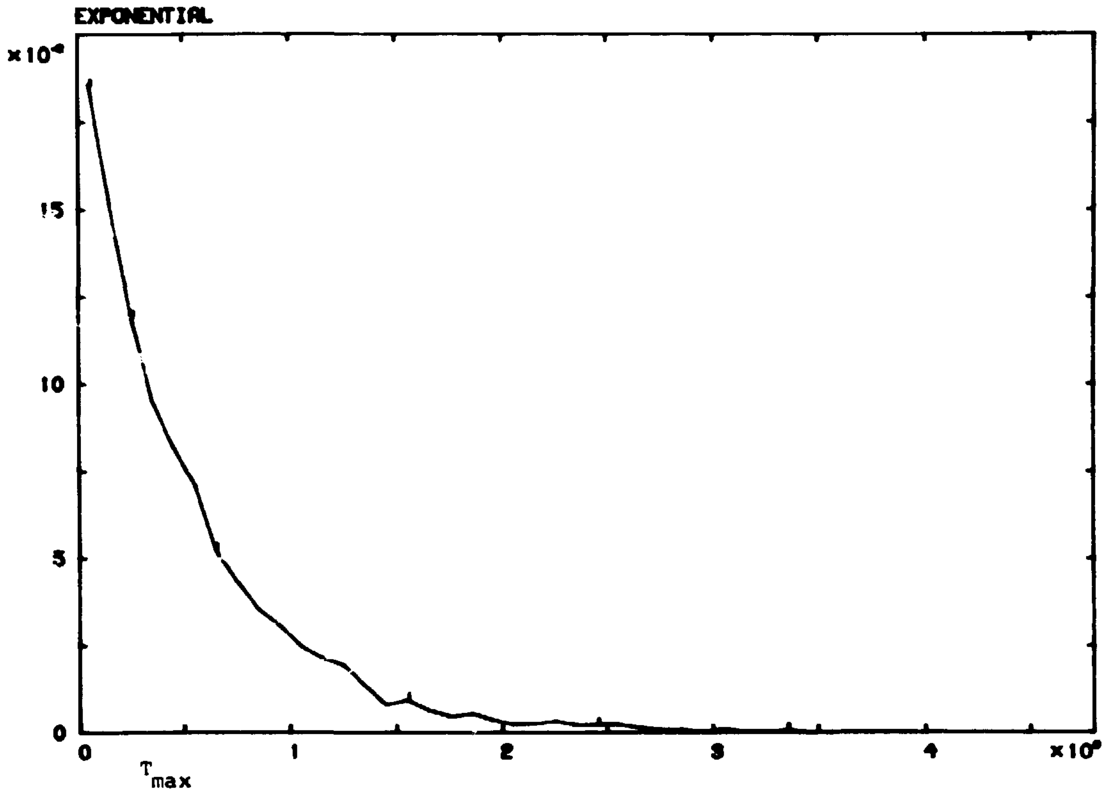

Fig. 2. Exponential density distribution function for the time-to-failure for a component.

$$= \int_{x_o}^{T_{max}} af(x)dx + \int_{T_{max}}^{\infty} bf(x)dx$$

$$= a-(a-b)e^{-\lambda(T_{max} - x_o)} .$$

Now

$$a - (a-b)e^{-\lambda(T_{max} - x_o)} = 1 \text{ equals}$$

$$b = a + \frac{1-a}{e^{-\lambda(T_{max} - x_o)}}$$

If you find a value of a, such that

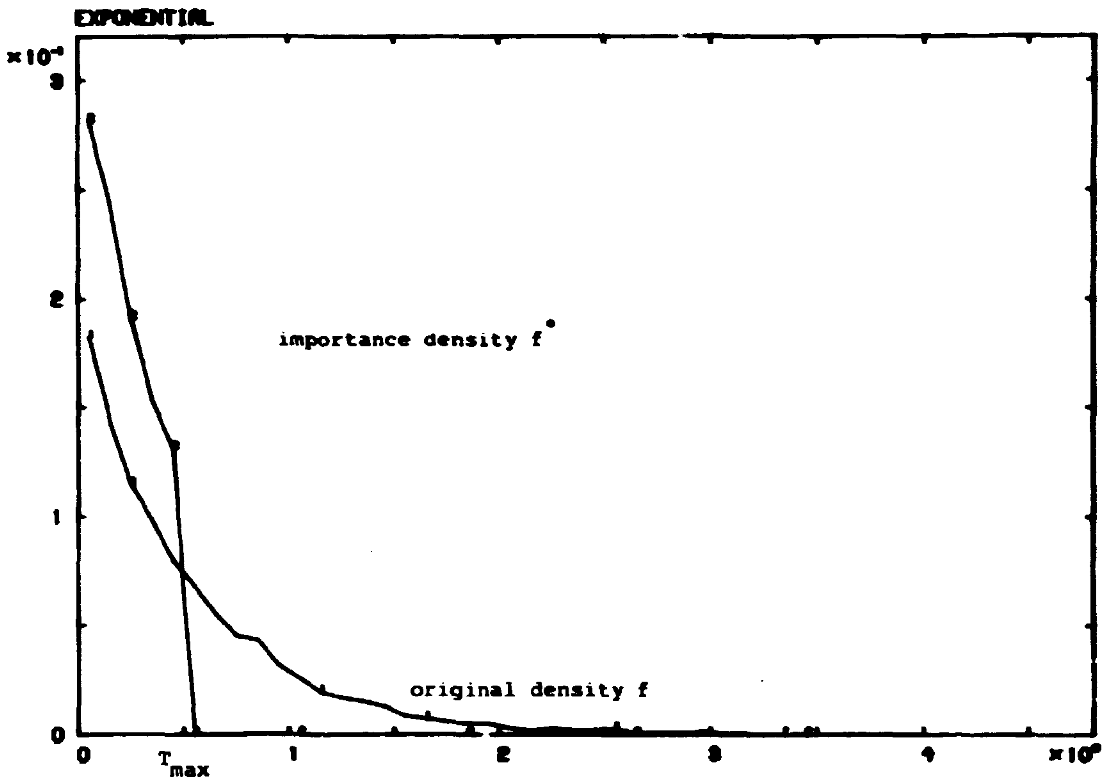

Fig. 3. Weighted density distribution function for the time-to-failure for a component by application of importance sampling.

$$\frac{1-a}{e^{-\lambda(T_{max} - x_o)}} = -a, \text{ then } b = 0.$$

It is shown that it is not allowed to change a sampling function in a way such that values from the tail will occur with probability zero. The value of a must be chosen, so that the term representing the tail,

$$\int_{T_{max}}^{\infty} bf(x)dx = b\cdot e^{-\lambda(T_{max} - x_o)} \quad \text{is positive.}$$

A test has shown that with a choice of a, which satisfies that at least 10 of the values are chosen outside T_{max}, an optimum is reached. This gives the true result when the total number of simulations N_{TOT} is between 1000 and 50,000.

Summarizing one calculates

$$a = \frac{1}{1-e^{-\lambda(T_{max} - x_o)}} \cdot \frac{N_{TOT} - 10}{N_{TOT}}$$

and

$$b = a + \frac{1-a}{e^{-\lambda(T_{max} - x_o)}} .$$

In the implementation a value x is generated from the computer random generator RANDOM. If

$$0 \leq x \leq \frac{N_{TOT} - 10}{N_{TOT}} \text{, then}$$

the term f (x) = af(x) is used; otherwise, the term f (x) = bf(x). This can be done for each component succesively. If the value just found is t_o, then the lifetime for the next component has to be between t_o and T_{max}, and so on.

The reduction in computer time for a simple system with use of the above given technique is a factor 10 to 200 depending on the system and the time of observation T_{max}.

3. REACTOR PHYSICS AND DYNAMICS

The main subject areas in the Section of Reactor Physics and Dynamics are steady-state reactor physics, reactor dynamics, and fuel management. The last topic includes the economy of the fuel cycle. However, three essential circumstances have influenced the work of the Section and have created work to some extent outside reactor physics and dynamics.

Firstly, the TMI accident has changed the basic work only gradually. However, the work involved in reviewing the accident for government authorities and for the public has been considerable.

Secondly, the Danish utilities have been required to define and evaluate a storage concept for highly radioactive nuclear fuel waste. The work was initiated by the Danish government and the results will form part of a basis for a decision on the installation of nuclear energy in Denmark. Risø National Laboratory has acted as consultant to utilities, and the Reactor Physics Section has been involved in this work. The tasks were partly within reactor physics but also problems concerning temperature distributions and brine migration in rock salt were treated. These last topics were taken up by the Section of Reactor Physics and Dynamics since the method which could be used were similar to those used in reactor physics (3.5).

Thirdly, the section has taken part in the work on calculating the long-term consequences of a hypothetical large reactor accident at the Barsebäck nuclear power station. Results from this study will also form a part of the material on which a Danish decision on nuclear power will be based and the study shall be completed in 1981. The Section is mainly involved in assessing the amount of radioactive material released, if a large accident were to occurs, and the probabilities for accidents. Both assessments will be based mainly on already existing and internationally available material. Furthermore, a program for calculating the processes during a core melt, CHEMLOG, has been implemented, and the conclusions of this study will also be used. This work is carried out in cooperation with the Section of Heat Transfer.

Within the area of steady-state reactor physics most of the standard tools for reactor calculations are available. From UKND library a master tape with cross sections in a 76-group structure is created. Resonance cross sections are treated in the RESAB program system. On the basis on the 76-group cross sections a transport theory fuel pin or cluster cell calculation is carried out in the CCC program. Results from the CCC calculations may be few group microscopic (typically 10 groups) and few group macroscopic (typically 5 groups) cross sections which can be used in the fuel element burn-up program CDB. Three-dimensional power distribution calculations may be performed by a series of different available methods. For short computing times, either a nodal theory model NOTAM or a flux synthesis model SYNTRON can be used. For accurate reference calculations the finite difference equation program DC4 or the finite element FEM are used.

The main effort in the past years has been to define methods and to use and verify the existing methods for different problems. Since a coarse mesh (nodal) method is mostly used at the moment at Risø for three-dimensional calculations, the work on an improved coarse mesh method has been continued (3.1).

The work on the core follow-up study for a BWR has been continued and has reached a point where the burn-up calculations for a complete core can be performed. Another major area where reactor physical programs have been applied is the evaluation of a waste storage concept as mentioned above and reported in 3.4. Finally, the work on generating data for three-dimensional dynamics calculations for a PWR has been initiated. As a new topic, the surveillance of the power distribution in a reactor core during operation has been taken up (3.2).

In reactor dynamics the two main areas are three-dimensional dynamic models for the reactor core, and integral plant models. A program, ANDYCAP, for analysing transients in three dimensions for a BWR-core has been in use for several years. The work on a similar program which will make three-dimensional dynamics calculations for a PWR-core including hydraulics was started two years ago within the NORHAV project. The work is now well under way and a version of the program, ANTI, is operable (3.3).

Finally, the work on a plant model for a BWR-station has been brought as far as data available has allowed, and reported (CHRISTENSEN, 1979). As a result of the TMI accident, an already existing plant model for a PWR-station will be updated. The accident showed the need for such models to be able to handle more extreme transients, for example the dry out of steam generators as a result of feedwater failure. The work is in progress.

In fuel management the use and development of the SOFIE program is continued. SOFIE includes a simplified reactor physical model of an LWR and an economic model of the fuel cycle. By means of linear programming the refuelling strategy for several cycles can be optimized to give the best economy. An agreement on co-operation with the Danish utilities on the use and further development of SOFIE has been reached (3.4).

3.1. Interface Methods for Solving the Neutron Diffusion Equation

An interface method results from casting the neutron diffusion equation in a form where only interface-defined quantities enter. As an example, a two group, two-dimensional, nodal program has been developed in which the flux distribution inside a node is expanded in local solutions to the diffusion equation, while the equations to be solved express current continuity at interfaces. The unknowns are corner flux values so that, asymptotically, we have one unknown per group per node. For each mode we have to specify how the total buckling is split into directional bucklings. Calculations on a two-dimensional version of the benchmark problem of Micheelsen and Larsen (Argonne Code Center, 1977) showed a pointwise flux error of 8% (relative to maximum flux value) with a mesh size of 20 cm (10 cm close to the core boundary) when (for each mode) the two directional bucklings were chosen to be equal. By letting the program optimize the buckling split (effectively doubling the number of unknowns) the flux error decreased to 3%. However, the numerical method for estimating buckling splits is not satisfactory, and higher-order methods might prove to be competitive.

3.2. Core Power Surveillance

Light water reactors are provided with detectors, by means of which the power distribution can be measured. However, the number of detectors is limited for economical and practical reasons, and the power distribution can be determined only in a limited number of points (~100) in the reactor. A computational procedure has been set up which calculates the three-dimensional power distribution on the basis of the detector readings. The procedure has been established for boiling water reactors because of access to data for this type of reactor. Nodal theory has been chosen for the procedure partly because of the limited storage requirement and partly because of the modest computing time. Taking the boxes of the reactor in cells of four boxes, approximately one quarter of the cells are equipped with detectors. The procedure starts by calculating the readings that would be measured in the cells without instrumentation, if they were instrumented. These pseudo-readings are calculated at each detector level. The calculation is performed for the horizontal plane at that level in two dimensions and comprises outer iterations, where the eigenvalue is adjusted, and inner ones, where the flux values are determined. Axial leakage is neglected. In the inner iterations the fluxes of the instrumented cells are replaced by auxiliary variables, and the equations are solved for these. The cell powers are calculated, and the auxiliary variables are normalized, so that calculated and measured values agree.

Having determined readings, pseudo or real, for all cells in the reactor, the axial power distribution of each cell is calculated using these readings. The power distribution for a cell is calculated using the detector readings of that cell alone. Each cell consisting of four boxes with no, one, or two inserted control rods, is homogenized. A one-channel calculation is performed in which the detector readings are used to modify the boundary conditions in the horizontal direction. Initially, reflecting boundary conditions are used. The power distribution corresponding to reflecting boundary conditions will produce detector readings different from the real ones. The deviation between real and calculated readings is a measure of the leakage in the horizontal direction. Therefore, the boundary conditions are altered iteratively according to this deviation so as to produce a power distribution giving the same values for both measured and calculated readings. The fuel box power is calculated using the axial power distribution of the cell to which it belongs. The box power is calculated by multiplying the cell power by predetermined correlation factors. These factors are determined by a two-dimensional box calculation on four boxes. During operation a cell is either uncontrolled or has one or two control rods inserted in it. The box calculation is made for each of these configurations at void and exposure points characteristic of the reactor. The method for calculating pseudo-detector readings has been applied to a test example, displaying a deviation below 4%. The method for calculating the axial cell power distribution has been compared to a three-dimensional calculation with much heavier computer costs which also uses nodal theory. The deviation was below 6% with a maximum above the tips of the control rods. The calculation procedure is summarized in the flow chart of Fig. 4.

3.3. Three-Dimensional PWR Dynamics

The ANTI computer program is being developed for three-dimensional-coupled neutronics/thermal-hydraulics transient calculations for the PWR core. It combines the three-dimensional nodal theory neutron kinetic part of the BWR program ANDYCAP with the transient subchannel hydraulic program TINA. The program is intended mainly for transients where the spatial distribution of power and coolant flow in the core is important, especially in case of a local power increase. It is hoped that ANTI, when completed, will be a useful tool for analyzing transients ranging from normal and abnormal operational transients to postulated accident conditions.

A working version of the program has now come into existence, but quite a lot of improvements are still needed, such as the introduction of various options and optimizations to reduce the demand for computer time and fast memory space. Test calculations to evaluate the capabilities of the program have begun.

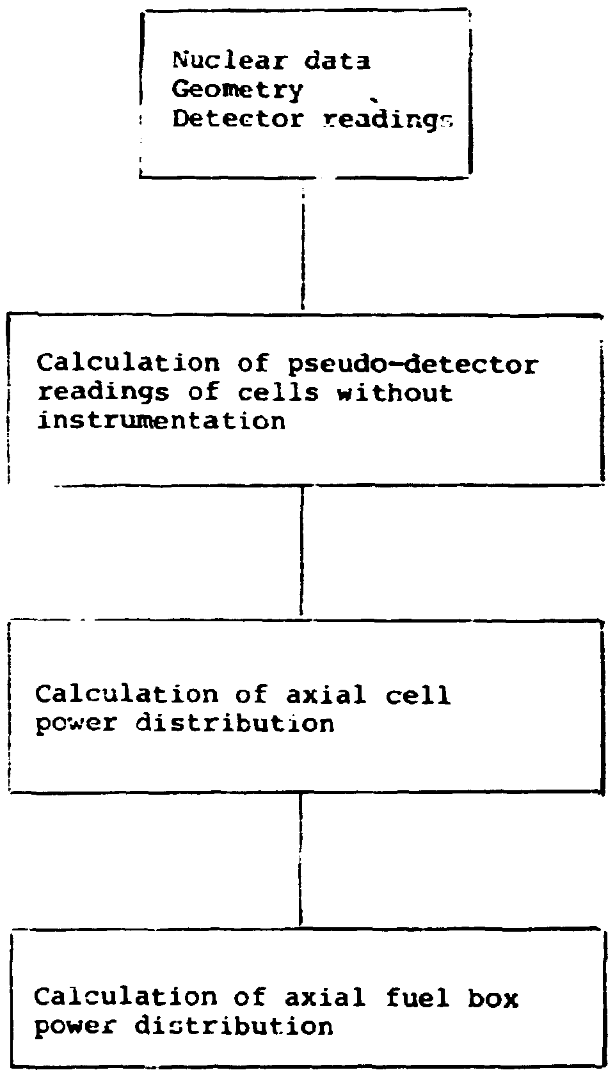

Fig. 4. Calculation procedure for 3D power distribution.

Figs. 5 and 6 show results calculated by ANTI. The example chosen is the calculation of a control rod ejection accident. To give an impression of the influence of the crossflow on this particular transient, the test case was repeated without crossflow between subchannels. In order to keep the computer costs at a reasonable level the example is only a small reactor core from which 1/8 is represented by means of 112 nodes in the neutronics part and 4 subchannels in the hydraulics. A control rod is removed from the central fuel element at a constant speed which takes the rod from a fully inserted position at time zero to fully out at 0.3 s.

The example is intended as nothing more than an initial test of the program, and the results of the calculation should not be expected to give information about the severity of a control rod ejection accident, since the neutron cross sections and the hydraulics input data were chosen more or less by chance. The calculation is therefore not representative of any existing reactor, but serves the purpose of demonstrating that the ANTI program is working and that the coupling between the hydraulics and the neutronics functions in the way expected.

The total reactor power, shown in Fig. 5, increases rapidly at the beginning of the transient as a result of the reactivity insertion. After approximately 0.2 s, the fuel temperature increase causes the reactor core to become subcritical and the power to drop. From about 1 s into the transient the power stabilizes at a power level of about twice the initial power, corresponding to a new critical condition of the core without the ejected control rod. The power histories are nearly identical for the two cases with and without crossflow.

The maximum void fraction in the core is shown as a function of time in Fig. 6. The feedback from the water density has only a limited influence on the course of this transient, since the voiding of the core occurs after the power peak. For the case without crossflow the void content in the hot channel goes up to 30 per cent, whereas the influence from adjacent, cooler channels brings the maximum void down to 15 per cent in the crossflow case. In any event, the void fractions never get very

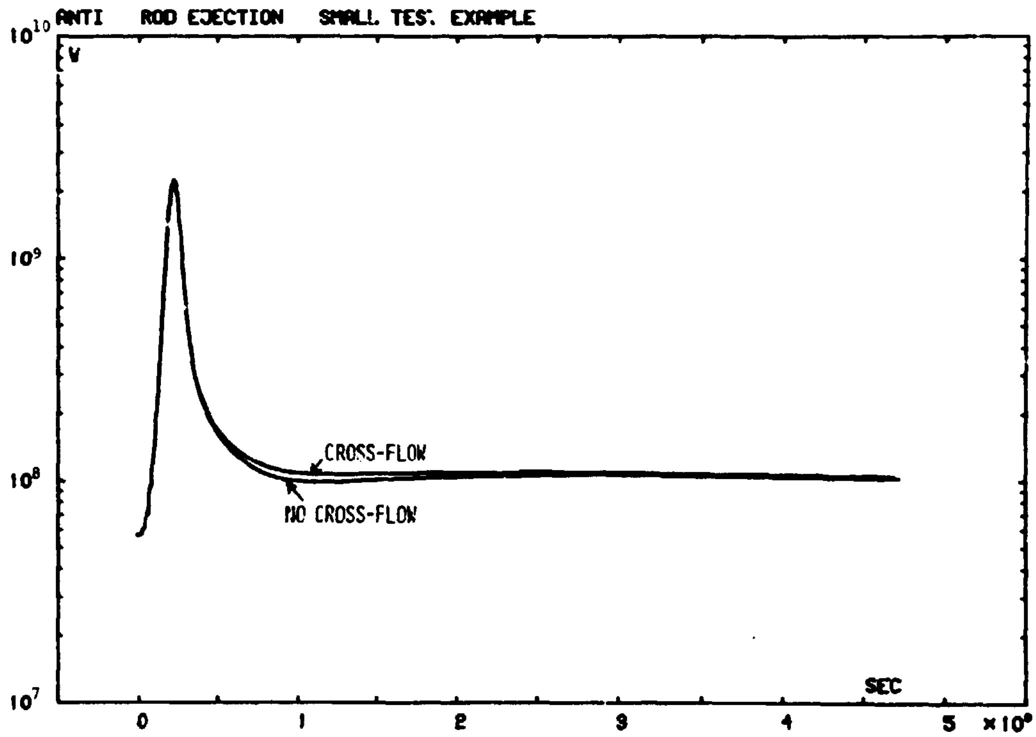

Fig. 5. Total reactor power as function of time during rod ejection transient.

high, and no burnout is predicted. The transient is therefore not a very dramatic one; one reason for this is that the start condition chosen for the reactor is hot full power. A control rod ejection from zero or low power would result in a much higher power peak.

The test calculation is described in more detail in LARSEN (1980).

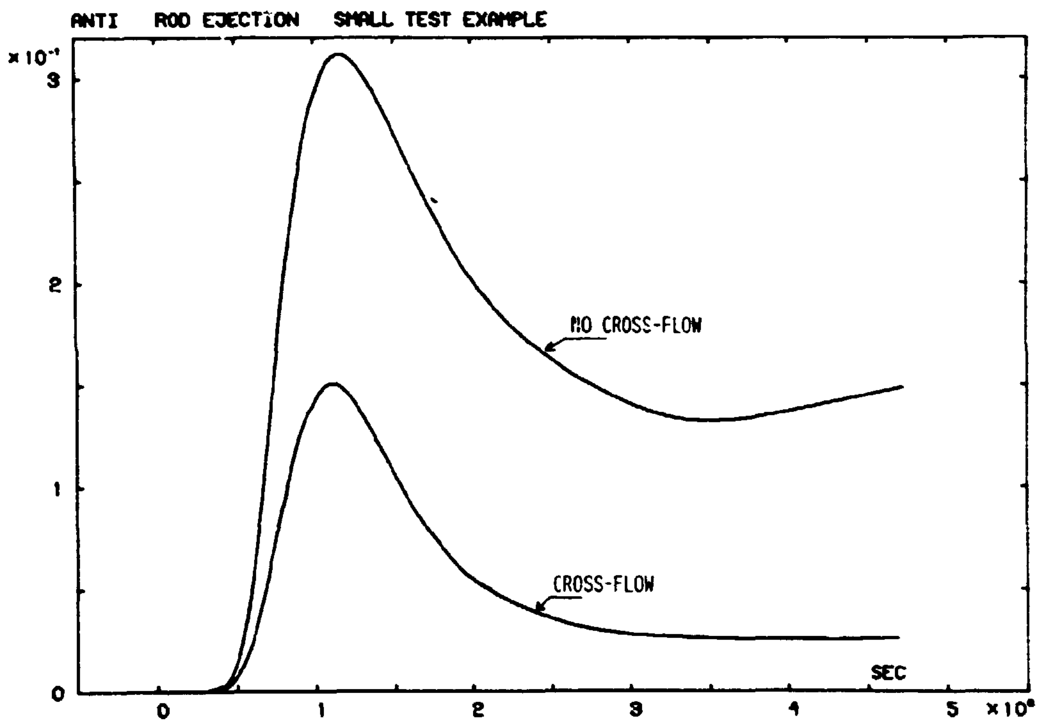

Fig. 6. Maximum void fraction as function of time during rod ejection transient.

3.4. Fuel Management

The SOFIE fuel management program is a computer program which minimize fuel cycle costs for a nuclear reactor. The reactor physics of the reactor core as well as the fuel shuffling is treated by use of a one-dimensional reactor model (concentric regions). The optimal operating strategy using this model is decided by using linear programming with the total fuel cycle costs as the objective function. During the year only minor modifications to the program have been introduced, mostly corrections to the code when errors have been detected. At the start of the year an agreement between the Danish utilities and Risø has been reach, concerning the use and the future development of the program.

As a qualification of the program it has been used by the utilities to determine the fuel cycle for a BWR reference case. The

SOFIE program was able, after som adjustment of the input parameters, to give the same fuel cycle as found in the reference case. Even the burnups, cycle-by-cycle, for each of the fuel batches were in good agreement.

3.5. Storage of Highly Radioactive Waste

In order to demonstrate the feasibility of safe storage of highly radioactive waste in Danish geological formations, the Danish power utilities have launched a project investigating possible designs of a waste repository in rock salt. As part of the project the inventory of nuclides and heat generation of waste was calculated. In addition calculations of temperature distributions and temperature gradient-induced brine migration in rock salt have been carried out.

3.5.1. Nuclide Inventory of Radioactive Waste

To be able to perform the required calculation some development had to be done on already existing programs, essentially to be able to calculate the heat generation from actinides.

Thus the program CCC (HØJERUP, 1976a) has been extended with routines for calculating fission product concentration (the procedure FISPRO (HØJERUP, 1976b), actinide concentrations (MORTENSEN, 1977) decay heat from fission product (HENNINGSEN, 1976) and decay heat from actinides. The calculations were made for a PWR with a power density of 34.4 MWth/T U and 3.1% enrichment. The power was held constant until the burnup was 33 MWd/kg U and then changed to zero. Various cases with reprocessing of the spent fuel after 3 or 10 years cooling time were examined.

Figs. 7 and 8 show examples of results obtained.

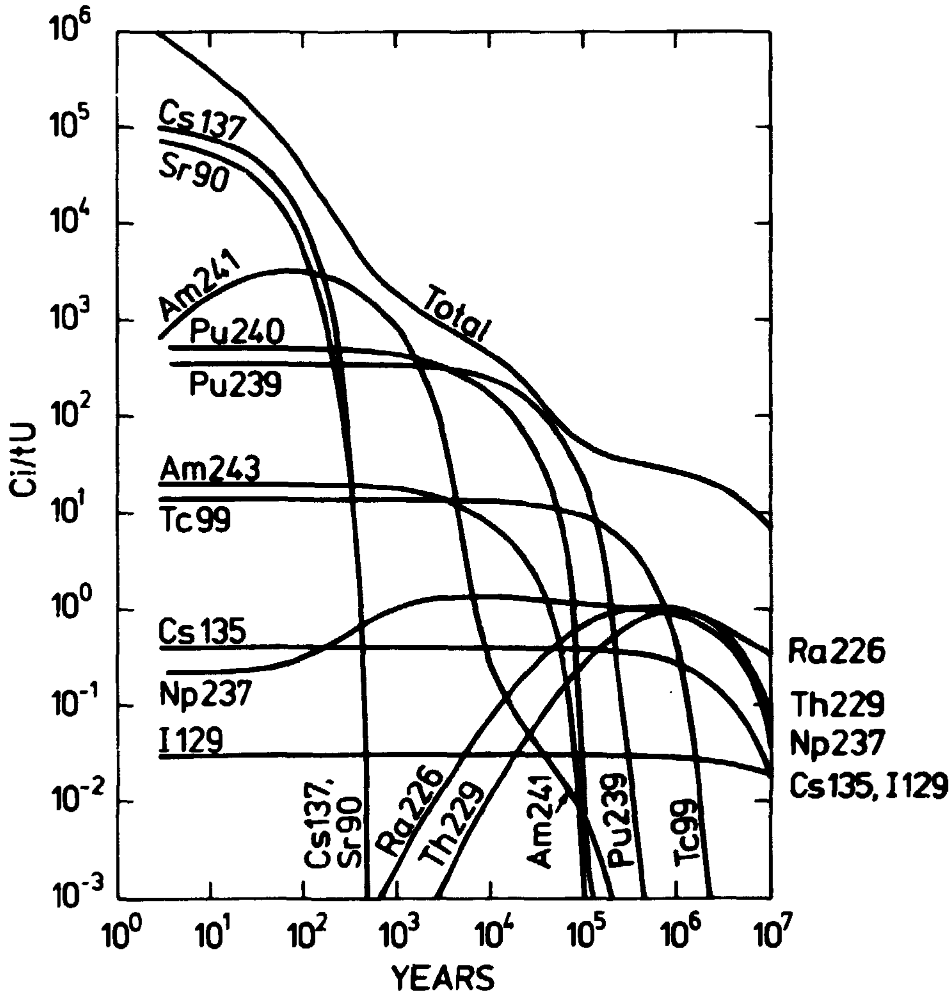

Fig. 7. Activities from PWR-fuel, 33 MWd/kg U.

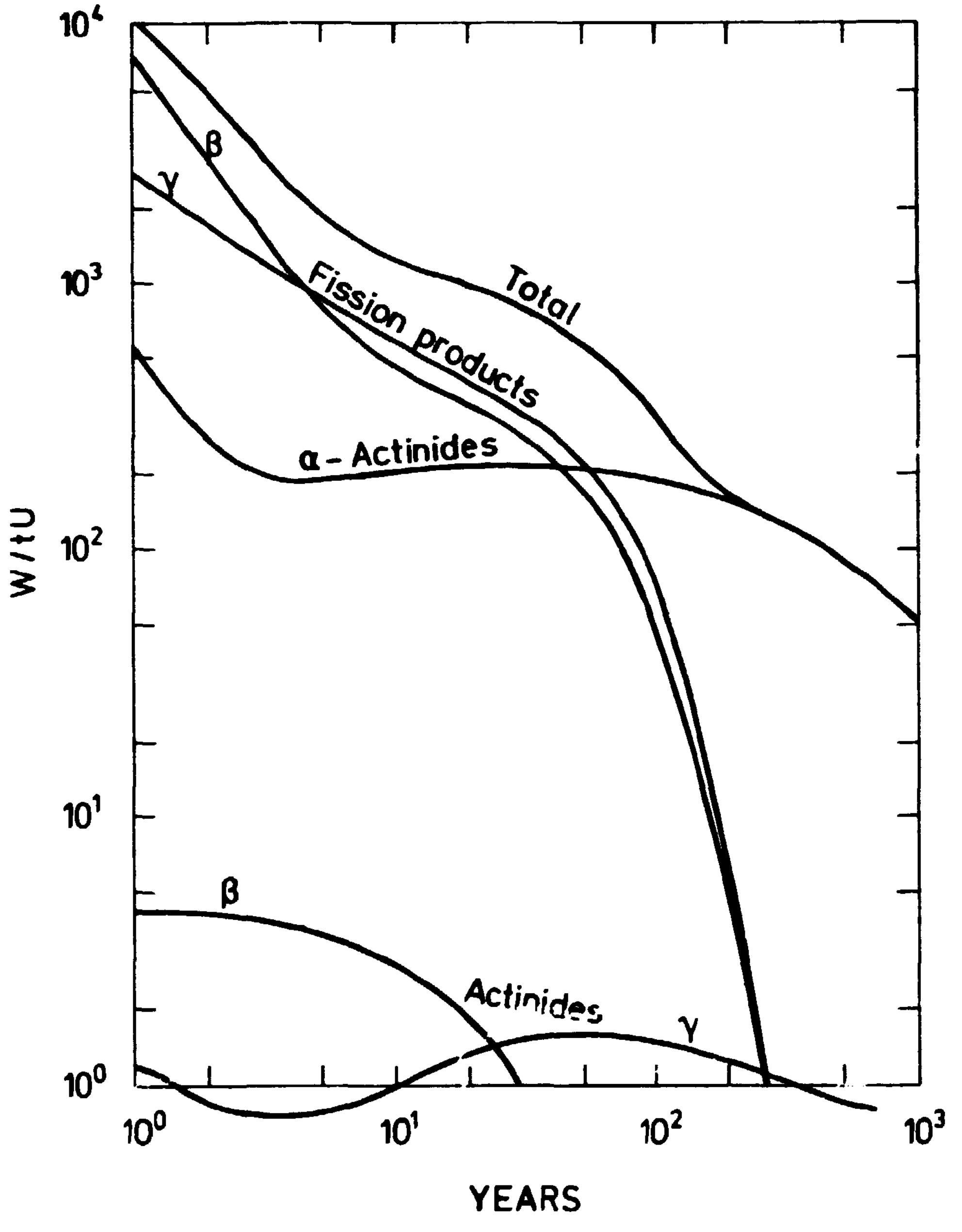

Fig. 8. Heat production from radioactive decays in spent fuel from PWR, 33 MWd/kg U.

3.5.2. Temperature Distribution

In the designs considered for a repository, the waste is either stored in an array of rather short holes or in isolated deep holes (2-3 km). The waste may therefore be simulated by suitable line sources. The time dependent temperature field around such systems is well suited for solution by Green functions, whereby short computing times can be achieved.

Algorithms based on the Green functions of the heat conduction equation in solids have been developed with respect to several different geometrical configurations of finite and infinite, line-shaped heat sources. A comparison of results from this type of algorithm has been made with those from programs based on finite element methods. Since Green functions cannot take temperature dependence of the thermal constants into consideration, whereas finite element programs can, particular care has been taken in assessing the errors committed by using fixed thermal constants. In Figs. 9 and 10 a comparison is made of some plots of the temperature field around a finite line source as calculated with Green functions in the BMCH-program and with the FEM-programs, YAC 2D and ADINAT. As is evident, the error can be kept low by suitable choice of the thermal constants.

3.5.3. Brine Migration

Brine migration has been treated in pure cylindrical geometry around an infinite, time-dependent line source calculating the temperature and the temperature gradient from Green functions. By inversion of the time variable in the differential equation for the brine displacement and obtaining a solution by standard numerical integration, the program finds the distance traveled at any chosen time following the insertion of the heat source by a brine inclusion, which just crosses the boundary of the deposition hole. As all brine inclusions inside this distance must have passed into the hole, the total brine inflow is easily calculated when the density of inclusions in the salt is known.

Simple approximations can be made that lend themselves easily to hand calculations, when a fixed, temperature-independent, ratio exists between the speed of the brine inclusion and the temperature gradient. A comparison with such hand calculations has provided a good check on the more general program.

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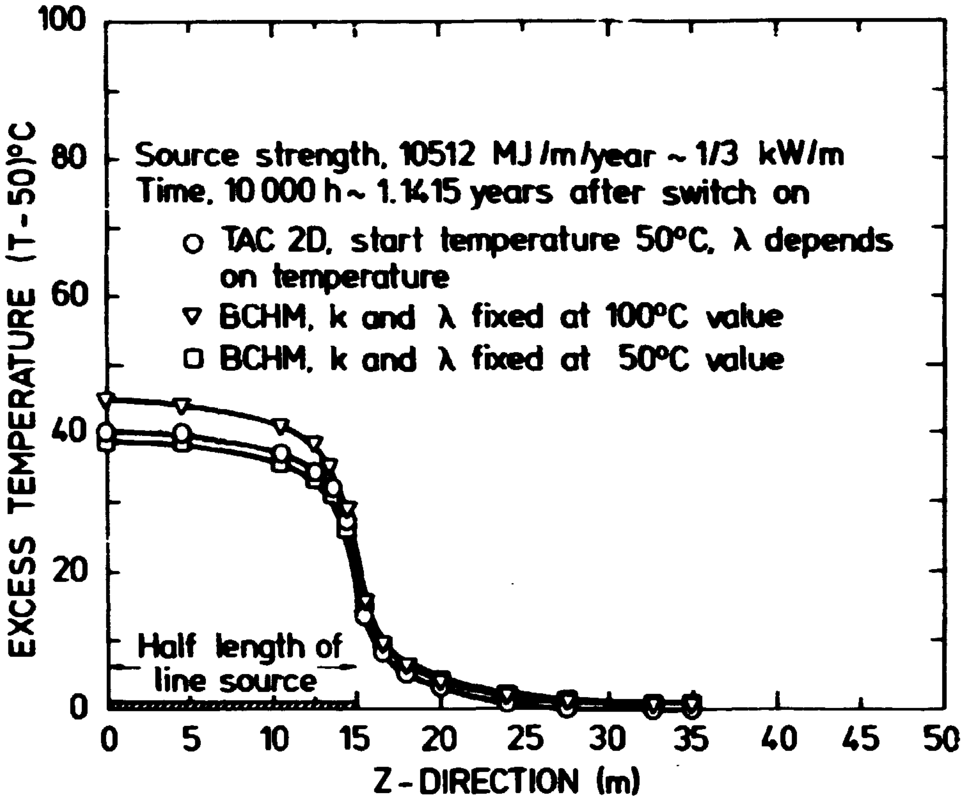

Fig. 9. Axial temperature distribution in 0.35 distance from 30-m long line source.

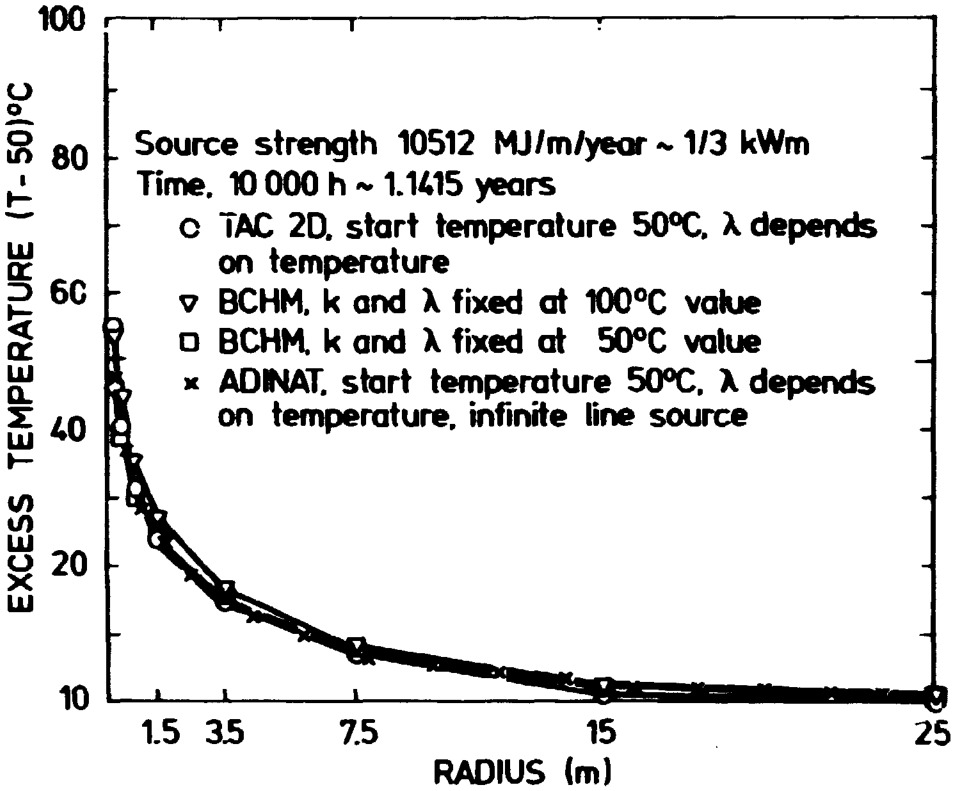

Fig. 10. Radial temperature distribution in symmetric midplane around 30-m long line source.

4. HEAT TRANSFER AND HYDRAULICS

The aim of this field is the understanding the thermo-hydraulic phenomena relevant to nuclear power reactors. To a large extent both the experimental and the theoretical work is concerned with the development of computer models. Furthermore, the basic thermo-hydraulic knowledge has been used in the non-nuclear field, see Section 6.

The main areas in the nuclear field have been:

1. Reactor accident analysis
2. Reactor-related experiments.

4.1. Reactor Accident Analysis

The development and verification of computer codes for LOCA-analysis constitute the major part of this work. Most of the work has been done within the NORHAV agreement in cooperation with laboratories from the other Nordic countries and the US NRC. Three Nordic guest scientiests have participated in this work at Risø.

4.1.1. Transfer of TRAC to Risø

The Transient Reactor Analysis Code, TRAC, version P1A, has been transferred to Risø from Los Alamos Scientific Laboratory, Los Alamos, New Mexico, USA. TRAC is a best-estimate, non-equilibrium, multidimensional, thermohydraulic, steam-water (two-phase), systems analysis, computer code developed specifically to analyze loss-of-coolant accidents in light water reactors. The code employs a three-dimensional (r, θ, z), two-fluid hydrodynamic treatment in the vessel, and a one-dimensional drift-flux treatment in the rest of the system components.

TRAC-P1A has been implemented on Risø's Burroughs B6700 computer and on CDC's Cyber 175 installation in Stockholm.

The code is being used for the LOBI PREX calculations.

4.1.2. Participation in LOBI PREX

The LOBI (Loop Blowdown Investigations) integral test facility at the Joint Research Centre, Ispra, Italy, has been built for the study of the influence of PWR primary loops on blowdown. The LOBI project activities are performed within the framework of a contract between the German Ministry of Research and Technology (BMFT) and the Commission of the European Communities.

The objective of the LOBI pre-prediction exercise (LOBI PREX) is to perform blind predictions of the most important measured results from the first LOBI system blowdown experiment. This exercise has been asked for by the US NRC and agreed upon by the

BMFT-Bonn. Apart from contractors from both parties, several institutions from member countries of the European Communities are participating.

Risø participates in the LOBI pre-prediction exercise using TRAC as the sole institution to do so in addition to Los Alamos Scientific Laboratory.

4.1.3. NORCOOL-I

NORCOOL-I is a BWR reflood code, and it is going in as part of the American Water Reactor Analysis Program packet (WRAP). NORCOOL-I uses a fixed geometry representation of the reactor core consisting of upper and lower plenum, one downcomer-, one bypass-, and one fuel-channel. Water and steam are treated separately, thus allowing unequal velocities and temperatures of the two phases. The code was originally thought to start from zero flows and pressure equilibrium between the reactor vessel and the surrounding dry well, but when used in the WRAP packet the input is generated from the output of a RELAP blowdown calculation and it does not meet the above-mentioned requirements.

During the year the code has been updated to accept the kind of input it will be given in the WRAP system.

To secure a stable and physical run on different test cases quite a few changes in both physical and numerical models have been carried out. To improve the performance of the code the radiation model has been extended to include non-uniform reflection. Further, the automatic timestep control has the possibility of letting the last calculation cycle be recalculated with a smaller timestep, thus ensuring that steep gradients are studied with small timesteps at the very start of the gradient.

By the end of the year, instabilities still caused breakdowns in the calculation of certain test cases while others ran well and with fair results.

4.1.4. NORCOOL-II

NORCOOL-II is an advanced emergency core cooling code for simulation of the emergency cooling of the hot core of a nuclear reactor. The code uses non-equilibrium three-field hydraulics (steam, water film, and water droplets) and a one-dimensional network as geometric representation. In particular, the network allows separate simulation of a number of parallel fuel channels in a reactor core. The development of NORCOOL-II took a major step forward in 1979.

As the NORCOOL-II work is part of the NORHAV agreement, the job was performed jointly by Finnish, Swedish, and Danish researchers, the Finnish and Swedish being stationed at Risø. The work proceeded along three lines: 1) code development, and physical models for 2) force interactions and for 3) heat transfer.

The code development passed through an interim single-tube version to end with a network version, the network consisting of tubes, each subdivided in nodes linked together by coupling nodes. Although this network version formally allowed for droplets, the droplet regime was not entered, as implementation of droplet generation models was postponed until 1980. Also, the internal wall heat transfer was modelled by a preliminary primitive wall heat transfer model.

The major part of the development was performed by making the code run a number of test cases: a) a tube blowdown, i.e. the pressure and velocity evolution after the sudden opening of a tube, which initially contains a pressurized steam/water mixture, b) a counter current flow case, i.e. water falling down in a tube against steam flowing up, and c) simulation of a stationary boiling experiment with subcooled inlet.

In parallel with the above-mentioned code development, the basic programming of a heat component module for a proper representation of the internal wall heat transfer was accomplished.

The work with physical models consisted in a revision of an existing interim package of physical models. Regarding the force interaction models, an adequate formulation for the "added mass force" was established while the interaction force associated with void gradients was found negligible. The heat transfer work was concentrated on the subcooled boiling and inverse annular film boiling regimes.

The rather short subcooled boiling regime may not be very important to emergency core cooling in itself, but through steam production the regime may be decisive for the quenching velocity of the hot fuel rods. A subcooled boiling model, also covering the heat transfer for single-phase liquid and for nucleate boiling, was developed. For the inverse annular flow regime, which may exist above the quench front, an existing model was generalized to include as well, the effect of liquid subcooling. With these new models, the heat transfer package for NORCOOL-II seems to be complete. However, the above-mentioned comparative run by NORCOOL-II of a stationary boiling experiment showed that at high liquid subcooling the heat transfer into the subcooled liquid is exaggerated at the expense of steam generation.

4.2. Experiments

The theoretical work was supported by the following experiments:

1. Inverse annular flow experiments
2. Temperature calibrations.

4.2.1. Inverse Annular Film Boiling Experiments

Inverse annular film boiling may occur during emergency core cooling of a nuclear reactor. The present experiments are part of a PhD-study that aims at a better understanding of this boiling regime.

Using an experimental facility, which was constructed during the first part of the PhD study, a systematic series of experiments with inverse annular film boiling in liquid nitrogen was performed. For different flow rates and nitrogen subcoolings, the heat transfer as well as the axial void profile were measured, the latter by γ-absorption equipment also developed during the study.

The experimental data were analyzed by using the thermohydraulic two-fluid code RISQUE, equipped with nitrogen material data routines. Excellent agreement between data and RISQUE predictions was obtained when heat transfer correlations relevant to the existing flow regime were used in the code.

A typical plot of the measured and calculated void fraction versus length is shown in Fig. 11.

Along with the nitrogen experiments, a test facility for similar experiments with steam-water was constructed; the experiments are expected to start in 1980.

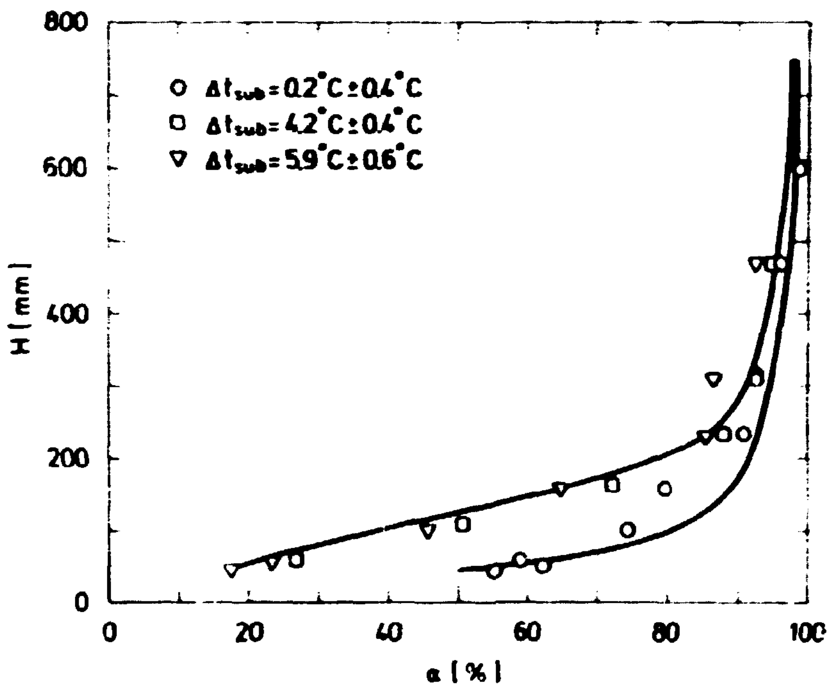

Fig. 11. Measured and calculated axial void profile. $G = 50\ kg/m^2s$.

4.2.2. Temperature Calibration Laboratory

In 1978 the Temperature Calibration Laboratory was authorized by the Danish National Testing Board to carry out certified calibrations of temperature sensors in the temperature range from -150°C to 1100°C according to the International Practical Temperature Scale IPTS-68.

The temperature sensors are compared with standard thermometers in a series of thermostats, where the desired temperatures are established.

Temp. range °C		type of thermostat	liquid
-150	0	cryostat	-
- 35	+20	stirred liquid bath	ethanol
+ 5	+90	- - -	water
+ 80	+250	- - -	oil
+200	+550	- - -	molten salt
+500	+1100	electr. furnace	-

Furthermore, three temperature fixed point cells are available, 0°C (melting ice), 100°C (boiling water), 444.7°C (boiling sulphur).

The temperature measurements are traceable to National Physical Laboratory (NPL), England. Our laboratory standard thermometers, a 25Ω platinum resistance thermometer, and a Pt/PtRh thermocouple, are calibrated and certified at NPL once a year.

In 1979, the Laboratory has performed 38 jobs for external customers and 14 jobs for other departments of Risø comprising in all about 200 temperature measuring devices. The calibrations have been made in the temperature range from -100°C to 1100°C thus covering the main part of the authorized range.

5. THE DR 1 REACTOR

The reactor is used mainly for training purposes and as a neutron source for neutron radiography.

During the year approximately 50 students from various universities have carried out experiments at the reactor. The reactor itself is in good working condition, but two new recorders for the neutron channels have been ordered for the control console.

New safety documentation for the DR 1 reactor has been written for the Danish authorities.

5.1. Neutron Radiography

Several irradiated fuel pins have been radiographed. Fig. 12 shows a radiograph of a BPI (Beam Purity Indicator), and some IQI (Image Quality Indicators). These indicators have been suggested by ASTM committee on neutron radiography as suitable control standards. Fig. 13 shows some calibration fuel pins, fabricated at Ispra, and sent to Risø for comparative neutron radiography.

5.2. Reactivity Measurements

When a test sample is introduced into the reactor the position of the fine control rod has to be changed in order to keep the reactor critical. This change is relatively large for DR 1 because of the small dimensions of the reactor.

The critical position can now be determined to an accuracy of 0.01 mm using a new position indicator on the control console.

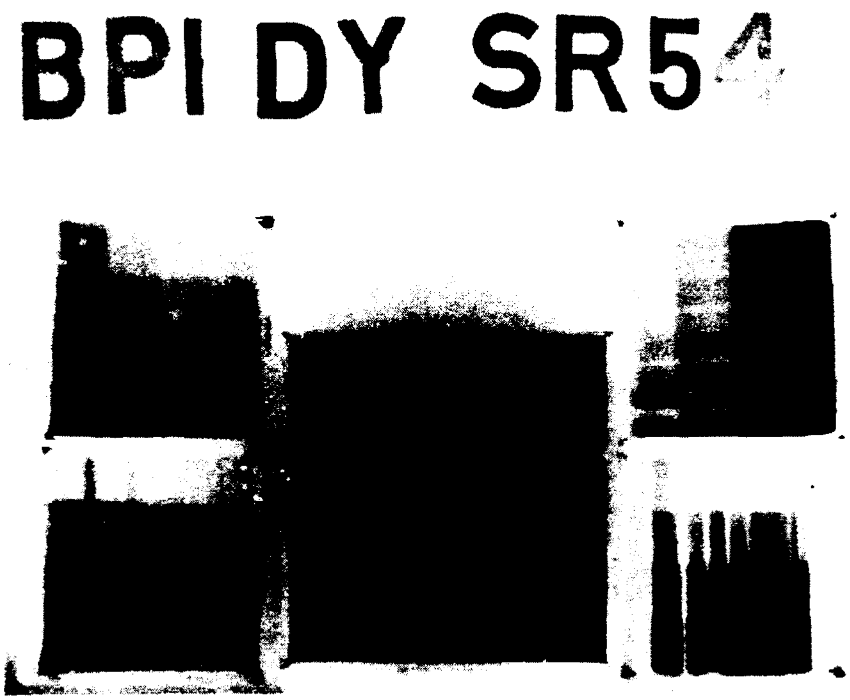

Fig. 12. Beam purity indicators using a Dysprosium-foil on a SR-54 film. The indicators are employed for estimating the influence of collimation, epithermal neutrons, and gamma rays.

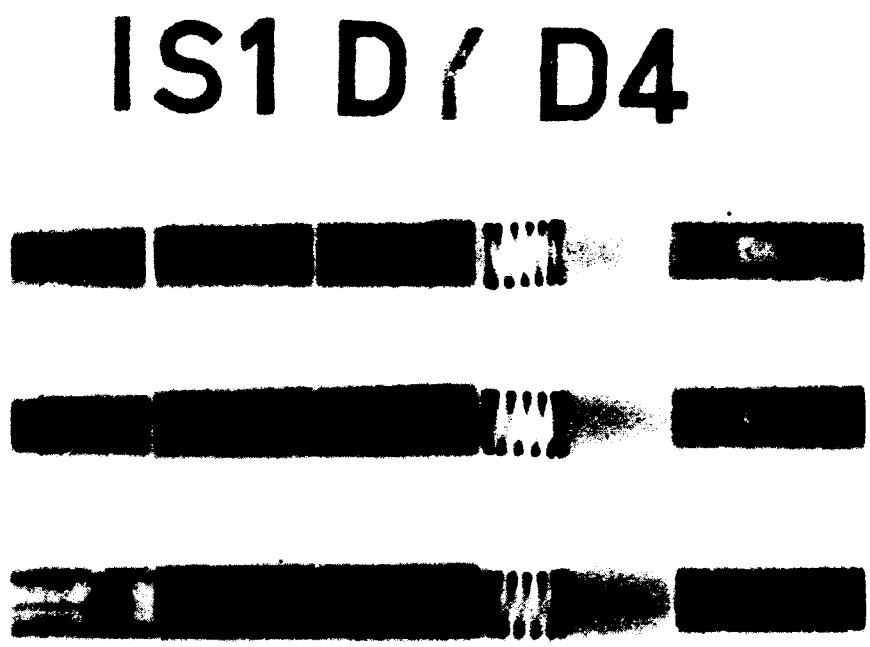

Fig. 13. Uranium pins manufactured at Ispra with built-in faults. The picture has been made by using a Dysprosium-foil on a D·4 X-ray film.

5.3. Neutron Metrology

The cobalt foils which determine the thermal neutron flux to the silicium-rigs in DR 3 are measured relative to a cobalt standard source using a 3"x3" NaJ detector. During the year several hundreds of foils have been measured.

6. NON-NUCLEAR ACTIVITIES

6.1. The Test Plant for Small Windmills

The test plant for small windmills was initiated at Risø in June 1978 as a part of the programme for energy research and development of the Ministry of Commerce (later the programme was transferred to the new Ministry of Energy).

The main activity of the project is to establish the test plant and perform the testing of windmills. The plant was established during 1978 and the beginning of 1979 with seven test platforms. By November 1979 six windmills had been erected, and the data sampling and handling system was completed and tested. Fig. 14 shows the test plant.

Fig. 14. The test plant for small windmills.

The tests and measurements of the first windmills: a 30-kW Riisager windmill, a 15-kW gyro-mill from Dansk Vindkraft and a 10-kW multi-vaned mill from Wind Power, were conducted towards the end of the year, with reports being issued in the beginning of 1980.

The staff of the test plant offers consulting assistance to windmill designers. A law was passed in the summer of 1979 which gave the Government funds to subsidize small windmills. This necessitated an approval of the windmills to be subsidized, and the staff of the test plant was responsible for this work

Furthermore, an economic analysis was carried out for a small windmill producing electricity under Danish economic conditions. The analysis was made for society as well as for the individual consumer.

6.2. Design of Rotor Blades for Large Windmills

Two large 630-kW windmills are built as a part of the Danish programme for energy research. This programme is conducted by DEFU, an organization of electrical utilities, for the Ministry of Energy, and Risø designed the blades for the rotor.

The design and manufacture of these blades was completed during 1979, and the rotors were mounted on the two windmills in the early autumn for mill A, and in December for mill B. Windmill A was connected to the electric grid, but developed electric and hydraulic problems which gave rise to an intermittent experimental programme.

6.3. Solar Heating of Buildings

Solar heating systems for buildings are studied theoretically supplemented by small-scale experiments.

A number of trickle collectors (Fig. 15) with non-volatile liquids as heat carrier have been run continuously for nearly a full year of a long term test of the stability of five selected liquids, several of which seem suitable for the purpose.As expected, the thermal performance shows an efficiency about the same as for collectors with closed channels.

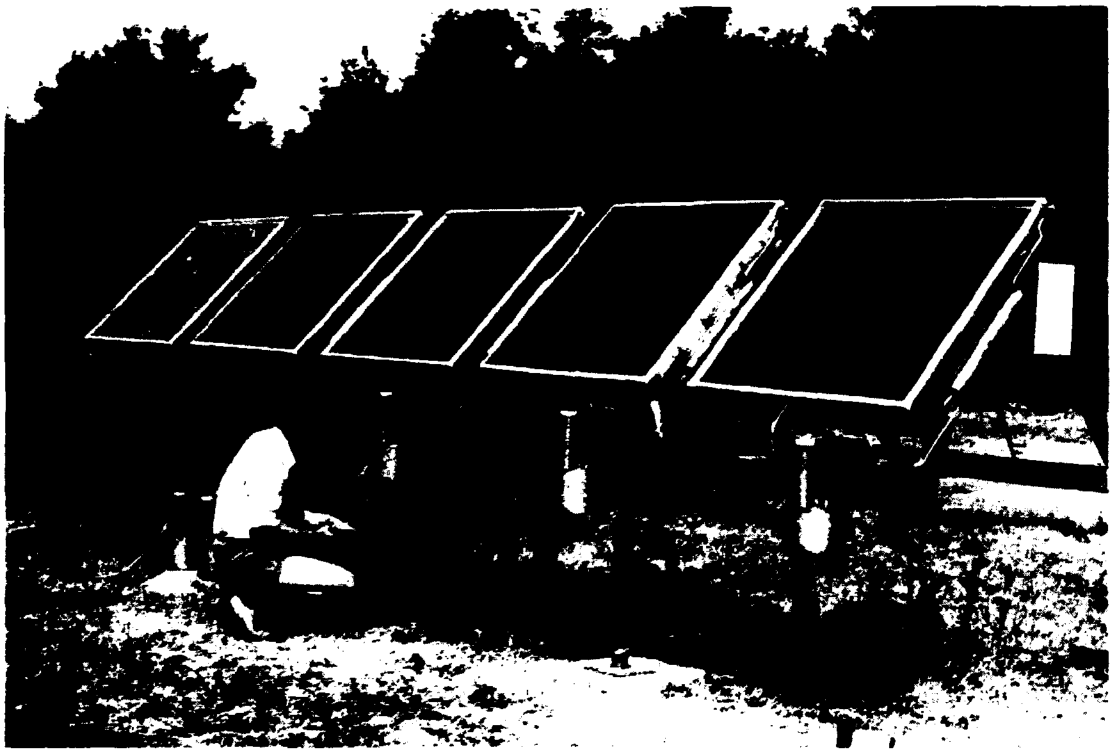

Fig. 15. Trickle collectors under construction.

The possibility is being investigated of using concentrating collectors for solar heating systems under Danish climatic conditions. Preliminary studies indicate that the direct radiation received by a focusing collector following the sun will amount to approximately 1200 $kWh/m^2 \cdot year$, roughly corresponding to the total radiation received by a fixed south-oriented flatplate collector with a 45^o slope.

The collector would have to be placed under a transparent dome or roof, in order to allow a light-weight, low-cost construction. The focusing collector would have a lower efficiency than a flatplate collector at low temperatures due to higher optical losses. The efficiency would, however, be superior at higher temperatures (55 - 60% at 100°C) due to small thermal losses. In connection with seasonal storage, this might reduce the storage costs sufficiently to make focusing collector systems competitive.

Further investigations concerning focusing collectors and heating systems will be undertaken, while existing data for direct as well as total insolation will be supplemented by additional measurements.

6.4. Seasonal Heat Storage in Aquifers

In collaboration with the Technical University of Denmark and the Geological Survey of Denmark, Risø participates in a project on large-scale seasonal heat storage in aquifers. The project includes geological exploration, construction, and test run of a 100000 m^3 pilot plant, as well as development of mathematical models.

In close cooperation with the Laboratory of Energetics at the Technical University, the Department is responsible for the development of the numerical simulation models.

At the present time two two-dimensional (or axisymmetrical three-dimensional) models have been developed: A simplified model PORFLOW and the more detailed D2AQ. It is planned to develop a fully three-dimensional model on the basis of these two. Due to the similarity in the heat and mass transfer in aquifer heat storages and in geothermal reservoirs, the models are able to simulate conditions in a geothermal reservoir as well (6.5.). In both models the finite element method has been applied for the numerical solution of the governing equation.

In Fig. 16, examples of simulation with D2AQ are shown. The hot water is injected at the central well to the left on the figures, and the cold water is extracted from an outer relief well on the right. It appears that the thermal front tilts due to the buoyancy of the hot injected water, and that the thermal front width increases during the injection.

Simulations of this kind have been used to select the optimal well configuration for the pilot plant.

To optimize the operation of the pilot plant it is planned to use the models to make prognoses during the test runs on the basis of the experimental measurements recorded.

6.5. Geothermal Energy

During the year the Department has been involved in four projects concerning modelling of geothermal reservoirs. All projects were carried out on a contract basis for Dansk Olie og Naturgas A/S in collaboration with other institutions and consulting engineers.

Two of the projects were carried out at Aars in connection with the first geothermal drilling in Denmark. The Department's contribution was, in this case, to compare the applied analytical reservoir model with the numerical models PORFLOW and D2AQ (cf. 6.4) to investigate if the implicit assumption in the analytical model could be accepted. An example of such a comparison is shown in Fig. 17.

The third project, planned for completion in September 1980, is carried out in cooperation with the Geological Survey of Denmark and the University at Aarhus. The purpose is to evaluate the fraction of the geothermal resources in Denmark which can be exploited from an energy point of view.

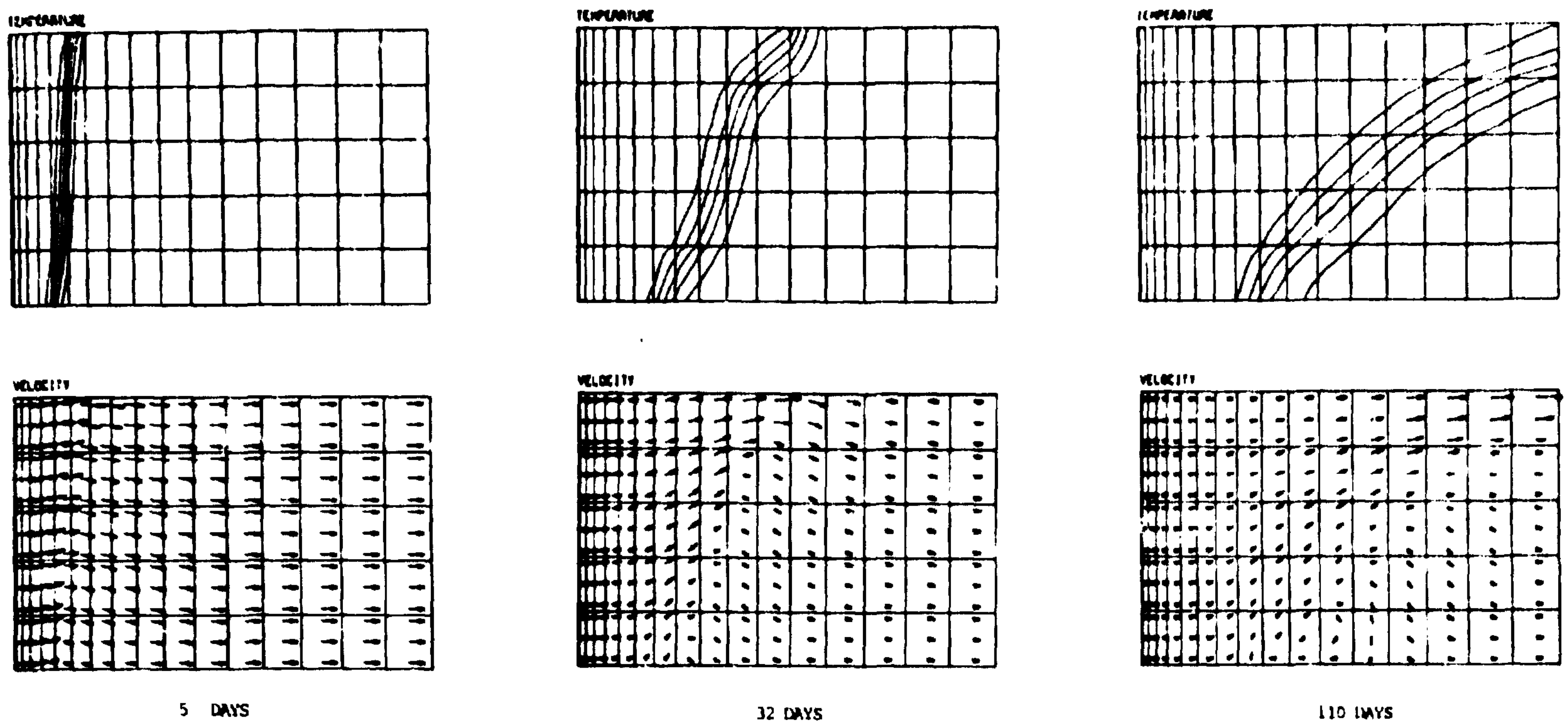

Fig. 16. Calculated temperature distribution and flow field after 5, 32, and 110 days of injection. The hot water is injected at the central well to the left of the figures and the cold water is extracted at an outer well to the right. The temperature distribution is represented by isothermals 10, 30, 50, 70, and 90 per cent of the difference between the injection and the initial temperature. The flow field is represented by velocity vectors multiplied by the radial position.

The following input data have been applied: Injection rate: 6 l/sec. Injection temperature: 100 °C. Initial temperature: 10 °C. Aquifer thickness: 30 m. Distance between central well and outer well: 45 m. Permeability: 5 Darcy.

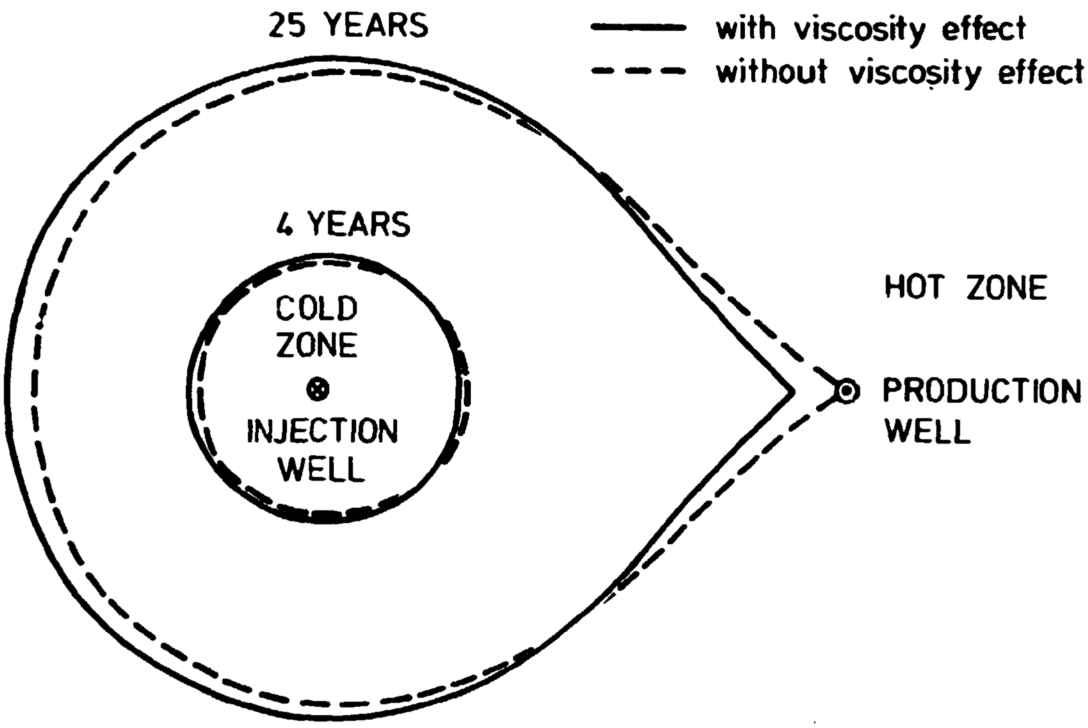

Fig. 17. PORFLOW calculation of the horizontal movement of the thermal front between the initial hot water and the injected cold water in a geothermal doublet. Two different calculations are shown to illustrate the influence of the viscosity difference between the hot and the cold water. In the analytical model this viscosity difference is neglected. It appears that the analytical model estimates a shorter lifetime (i.e. a shorter time until the cold injection water reaches the production well) than the numerical model, where the viscosity effect is taken into account.

The following input data have been applied:

Initial temperature:	85 °C
Injection temperature:	15 °C
Injection and production rate:	230 m^3/h
Distance between injection and production well:	1150 m

Finally, the Department has participated in a joint project with the purpose of pointing out the 5-10 economically most suitable locations in Denmark for exploitation of geothermal energy. This project was initiated December 1979 and finished June 1980.

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