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Department of Reactor Technology Annual Progress Report

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Risø National Laboratory, DK-4000 Roskilde, Denmark
April 1982

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DEPARTMENT OF REACTOR TECHNOLOGY
ANNUAL PROGRESS REPORT
1 January - 31 December 1981

Abstract. The general development of the Department of Reactor Technology at Risø during 1981 is presented, and the activities within the major subject fields are described in some detail. Lists of staff, publications, and computer programs are included.

INIS descriptors. HEAT TRANSFER, REACTOR PHYSICS, REACTOR TECHNOLOGY, RELIABILITY, RESEARCH PROGRAMMES, RISOE NATIONAL LABORATORY.

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1. DEVELOPMENT DURING 1981

1.1. The Department of Reactor Technology

The public and political support for energy forms alternative to nuclear energy is gradually changing the orientation of the research and development work at Risø.

During the past decade, the Department of Reactor Technology has initiated an energy systems analysis group, a testing station for small windmills, and theoretical work on heat storage and aquifers (Annual Progress Report, 1980, Risø-R-442). During the first half of 1981, the aquifer work was developed into oil and gas reservoir modelling. Studies of pressurized fluid bed combustion were initiated during the year.

A re-evaluation of the entire work programme of Risø began during the summer of 1981. For the Department of Reactor Technology the result was the following:

- The Section of Reactor Engineering decreased its effort on reactor technology, safety and licensing, and concentrated instead on the work of reliability and risk analyses. The name of the section was accordingly changed to "Systems and Reliability Analysis".

- The Section of Reactor Physics and Dynamics maintained its volume of work within the field of reactor technology, but a time limit was set within which the Section was expected to show benefits in terms of work for the government, commercial contracts, etc.

- The Section of Heat Transfer and Hydraulics substantially reduced its theoretical and experimental efforts on accident analysis, although work is still being maintained on theoretical and experimental emergency cooling, and small break codes. The effort on oil and gas reservoir modelling was increased significantly, and the project on pressurized fluid bed combustion strengthened.

As a result of the re-orientation and re-evaluation, the Department changed its name to the Department of Energy Technology on the first of January 1982.

The academic staff of the Department numbered 34 at the end of 1981, and approximately 20 of these have been working with nuclear technology during the year. Furthermore, 4 Ph.D. students and stipendiates were engaged in nuclear research.

The staff of the Department is listed on page 39.

1.2. System and Reliability Analysis

The main part of the work is concerned with developing methods for assessing the reliability of systems and components in nuclear power plants and other industrial installations. Furthermore, a core simulator is being developed, and a limited effort is being spent on monitoring the general trends within the nuclear field.

The competence regarding reliability analysis, which originates from the nuclear field, is used for safety and risk analysis of industrial installations. Accordingly, for the past several years the Section has been engaged in analyses of that kind. These include offshore installations, chlorine production facilities, and the Danish natural gas system. During 1981 the main project was an analysis of the security of supply for the natural gas system.

The Section is engaged in the 4-year Nordic co-operative effort: "Probabilistic Risk Analysis and Licensing", and furthermore, the Section participates in the reliability benchmark set up within the EEC. This project was started very recently and deals with the emergency feedwater system in a 1300-MWe French nuclear power station.

The core simulator program COSMA combines a module for calculating the reactor physics of the power history with a module for predicting fuel failures. This work is done in collaboration with the Section of Reactor Physics and Dynamics and with financial support from the Danish utilities. The analysis of local pin power was taken up in a Ph.D. study which has been completed. The COSMA program itself for linking together the different blocks in the core simulator (i.e. cross sections, 3-D calculation, local pin power, and fuel reliability prediction) was tested on a simplified PWR core.

Finally, one member of the staff is assigned to the OECD Halden experiment in Norway.

1.3. Reactor Physics and Dynamics

In steady-state reactor physics the core follow study of a BWR has been continued, while the corresponding work on a PWR has reached a point where comparisons between calculations and measured data from the start-up phase have been made with satisfactory results. A Ph.D. study concerned with the development of a method for core surveillance is concluded and the work reported.

In reactor dynamics further refinements of the three-dimensional model ANTI for a PWR core have been made, and work on a new fast three-dimensional model was continued as part of a Ph.D. study. The PWR plant model PWR-PLASIM is now fitted to the Ringhals Power Plant and a series of simulations of transients in the plant will soon begin.

The Section of Reactor Physics and Dynamics has co-operated with the Danish power utilities in several areas. Thus, the utilities have given financial support to the development of the fuel management program SOFIE as well as the core simulator mentioned in 2.3. In connection with a study on a Danish waste repository in rock salt, the Section has acted as consultant to the utilities. The Section carried out calculations of the radionuclide inventory of the waste and temperature distributions in the rock salt repository. This activity was concluded in 1981.

Because of the possible need for using uranium of a lower enrichment for the DR 3 research reactor at Risø, a series of reactor physics calculations have been carried out to predict the changed behaviour of the reactor if lower enriched fuel has to be used.

In previous years the Section has had some activity in studying core-melt accidents. This is now terminated following the re-evaluation of the nuclear activities at Risø.

In 1980 a project to predict environmental effects of energy production was formulated. Since then a small effort has been put into the project. It was hoped that this project could be carried out in co-operation between the Scandinavian countries. This has turned out to be impossible at present. However, during 1981 a possibility for financing the project via governmental funds has turned up, and thus the project will start early in 1982.

1.4. Heat Transfer and Hydraulics

In the past the Section was concerned mainly with studies relevant to nuclear reactors. During this year, however, a transition from nuclear to non-nuclear work has taken place. At the end of the year approximately one-third of the work has been nuclear, while the balance has been concerned with oil and gas reservoir engineering, and with fluidized bed combustion.

In the nuclear field the main effort has been on accident analysis. The work was concentrated on application at the NORCOOL II code for emergency core cooling experiments, and implementation of the American TRAC code for PWR blowdown and emergency core cooling calculation. The experimental work on accident analysis has dealt with an experiment concerning rewetting of parallel heated channels and a rewetting experiment under the indirect action program of the Commission of the European Communities.

The work load for the Temperature Calibration Laboratory has increased during the year because of additional orders received from private industry. A study concerning the possible use of solar collectors for domestic heating was concluded.

During the year the effort on developing oil and gas reservoir models was increased, and a project on fluidized bed combustion was initiated.

1.5. Danish Reactor No. 1

DR 1 is used for student training as well as a source for neutron radiography. In addition, the reactor has been opened to the public during normal Risø week-end tours and has been visited by more than six thousand people.

Forty-two students from various universities have carried out experiments at the reactor lasting two or more days. For the first time there was high school participation in reactor physics experiments, in this case by three classes.

In order to improve dimensional measurements of fuel rods, the signal-to-noise ratio of typical radiographs has been determined in collaboration with the Metallurgy Department. The measurements were carried out by equipment which digitized the radiographs. A report concerning the work was presented at the first world conference on neutron radiography in San Diego, USA, 7-10 December 1981. Two other reports concerning the work on neutron radiography were presented at the conference.

2. ACTIVITIES OF THE DEPARTMENT

2.1. Probabilistic Risk Analysis and Licensing

The use of probabilistic risk analysis methods in the licensing of nuclear power plants as well as in the regulation of other industries is gaining importance. In 1981 a Nordic co-operative project was initiated in order to study problems in this context. The work is sponsored partly through the Nordic Liaison Committe for Atomic Energy (NKA), and the participants in the work are groups from The Technical Research Centre of Finland, The Institute for Energy Technology, Norway, Studsvik Energy Technology, Sweden, and Risø National Laboratory.

The project is aimed at:

1) verification of risk analysis methods concerning the completeness of the models and accuracy of the quantitative predictions
2) improvements in the data bases for the reliability of components, and
3) presentation of guidelines for the application of probabilistic methods in licensing, including an evaluation of the benefits and limitations.

The Department's work on the project during 1981 has taken place mainly within two subject areas: data problems and the influence of test intervals and repair times.

A study has been performed on problems in collection, treatment, and organisation of data in order to be able to assess procedures in existing data bases. Concerning collection and organisation, the possibilities and limitations in the use of generic data and homogenization of these data have been studied. The use of Bayes' theorem to derive plant specific distributions for the failure rates of components on the basis of generic data has also been analyzed. This work will be continued as a separate task, also in connection with its use in a data handbook which is being prepared by the Swedish utilities and authorities based on information from the ATV data base. (ATV: Arbetsgrupp för Till-Tillgänglighet av Värmekraftverk).

As a contribution to the data handbook, an analysis has been carried out of pipe failures in light-water reactors during the period 1960-75. This work is continuing with an updated version in which the experience reported in the ATV data base will be included.

The influence of test intervals and repair times on the unavailability of systems has been studied by means of the Monte Carlo simulation code MOCARE. In order to do this, various modifications of the code have been implemented. The primary purpose of the work was to verify results obtained by means of approximate analytical calculation methods. Results obtained so far indicate that the approximate methods give satisfactory results within the regime where the assumptions made are valid. Calculations outside this regime are to be carried out in the future work in order to quantify possible deviations. The sample problem studied is a simple auxiliary feedwater system consisting of two 100% strings each with a pump, a motor operated valve, and a check valve. Figure 1 shows one interesting result ob-obtained from the MOCARE calculations. The distribution function for the unavailability differs substantially from the usual Normal Distribution.

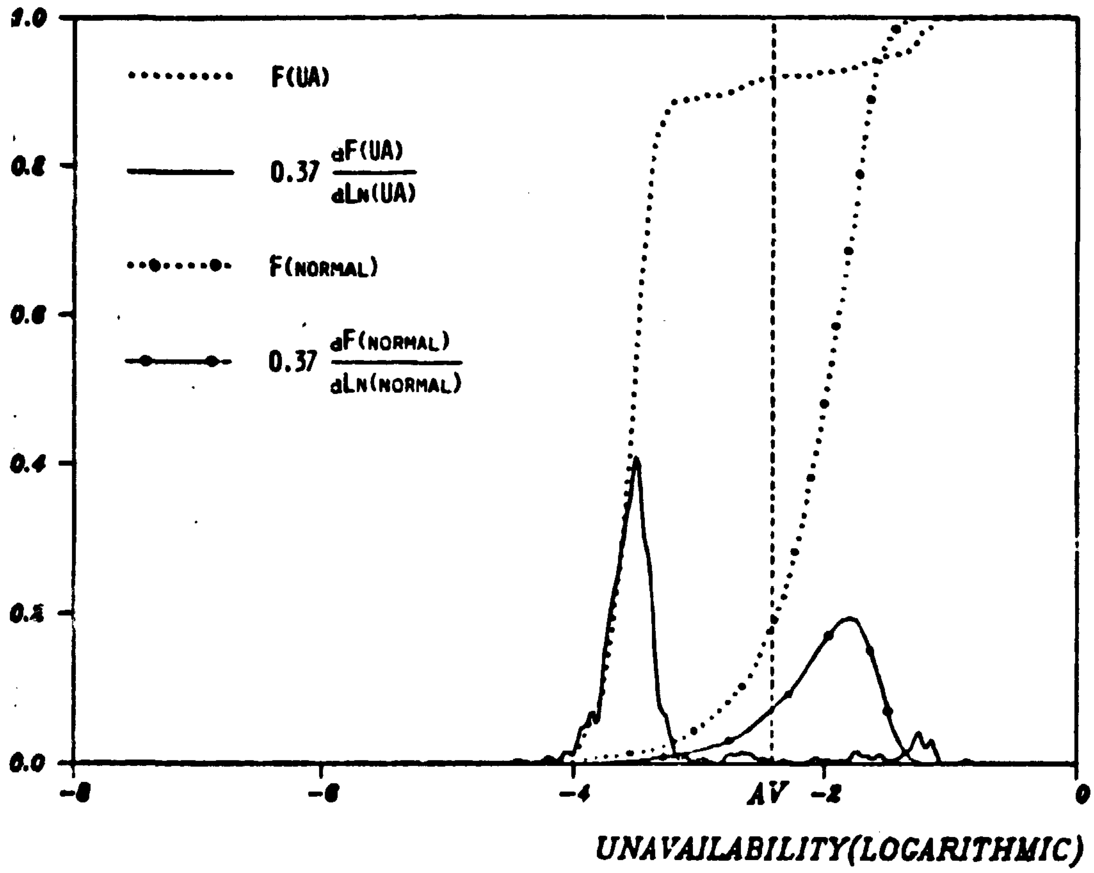


Figure 1. The distribution and probability density functions for the unavailability of an auxiliary feedwater system, simulated over periods of 1 year.

2.2. An Analysis of the Security of Supply of the Danish Natural Gas System. Phase II.

An analysis of the security of supply of the Danish natural gas system was performed in co-operation with the Electronics Department and the Energy Systems Group. The analysis comprises the following subtasks:

1) analysis of the gas treatment plant,
2) comparison and evaluation of two previous analyses of the belt crossings,
3) updating of the security of supply analysis of the transmission system,

4) collecting reliability data from natural gas systems in other European countries, and
5) analysis of selected medium and low pressure distribution systems.

The work in the Department of Reactor Technology concentrated upon the following two topics:

a. Analyses within the Danish natural gas system

The work on belt crossings consists of a comparison and evaluation of two previous investigations. The investigations comprise estimates of the average time between major pipe breakages of the underwater pipelines in Lillebælt and Storebælt. The average time between a major loss of supply was found to be about 400 years for the Storebælt-crossing and about 5000 years for the Lillebælt-crossing. The repair times would typically be in the order of one month. On this background it was recommended that some way of establishing an extra possibility for supplying Sjælland should be considered.

Reliability data from experience with natural gas systems in other European countries have been collected and prepared for use in the analysis of the Danish natural gas system.

An analysis of a medium pressure distribution system, which applies ring structures, has been performed. The main goal is to estimate the amount of natural gas not supplied per year because of various failures of the system, given the failure rates and repair times of all system components. The level of security of supply was found to be very high, mainly due to the use of ring structures in the systems.

b. Development of a Monte Carlo simulation model

A flexible Monte Carlo simulation model has been implemented. As input this model needs the topology of the system, as well as the consumer profile for each specified node of the system, the failure rate and repair time for all system components, the variation of temperature over the year, and a table describing the supply in each node for each possible failure of the system. The model takes into account the influence of dividing the customers into a non-interruptible and an interruptible part.

The program which computes the table of supply for each failure is developed at the Electronics Department on the basis of the Hardy and Cross method. This program has to be run in advance in order to create the table.

The computer code MOCARE generates failures of the system based upon estimates of failure rate and repair time for the components. By evaluating the performance of the system at each moment where system failure occurred, it is possible to calculate the average amount of natural gas lost per year. Furthermore, the distribution of this quantity is computed.

2.3. Core simulator

The core simulator is a complex intended for use in reactor core analysis. The system consists of four major blocks: cross-section generation, 3-D calculation, local pin power calculation, and fuel reliability prediction. A separate program COSMA (Core Simulator Manager) is used to gain access to the system and to facilitate the input preparation to the single blocks as well as couple the blocks via disc files.

The present status of the system is that a small PWR testcase has been used to demonstrate that the system is able to fulfill the intention laid down for the system. The blocks in the system are well-known reactor physical models: CDB, ANTI, NOTAM (see Appendix), and a fuel reliability model: FRP. The model for the local pin power is a preliminary one, which will be changed during 1982 to the model described later in this report. The small PWR testcase naturally gives considerable output. In Fig. 2 results are shown that are obtained with the preliminary version of the model which calculates the local pin power history. This pin power history is used later in the fuel reliability model for estimating fuel failures.

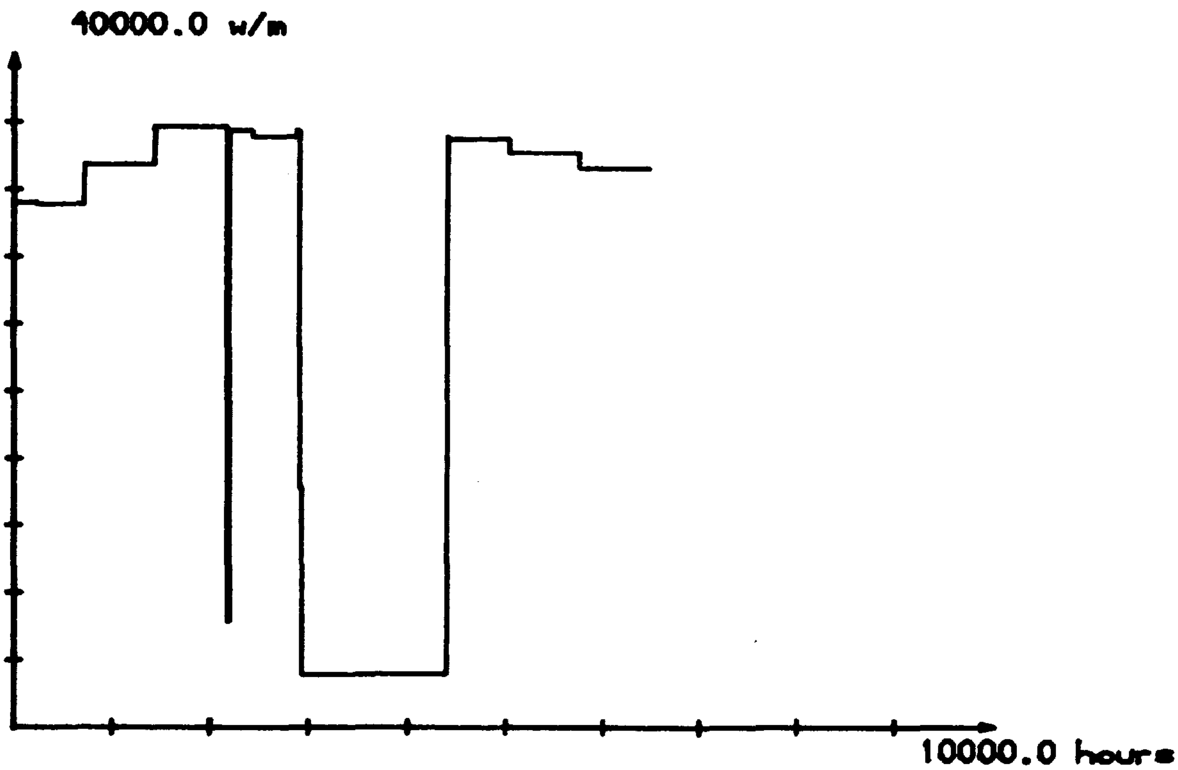


Fig. 2. Power history for a fuel pin as calculated by the core simulator.

2.4. Local Pin Power

A local pin power investigation, performed as part of a Ph.D. dissertation, was completed at the end of 1981, and a report including the results is under preparation.

The following problem has been examined: How to calculate local pin powers as an extension of coarse mesh nodal methods - or in other words: When the 3-D nodal coarse mesh calculation has been performed and the average nodal powers are known together with the values of fluxes and currents on the node boundaries, how the local pin powers can be determined based on this information.

The 3-D overall nodal calculation is based on homogenized cross sections for each node. In order to generate these homogenized cross sections one "fuel assembly calculation" has to be performed for each type of fuel assembly. These assembly calculations are made by assuming zero net current conditions on each assembly surface. The simplest procedure to determine local pin powers would be to use the power distribution found during the homogenization process and renormalize the average power to the actual nodal power from the nodal solution. Consequently, we assume that the power distribution in each assembly is unaffected by the power distribution in neighbouring assemblies. This method is called the normalization method. The normalization method has been examined on several 2-D pin power benchmarks and it has been shown that it can be used only on fuel boxes situated at places in the reactor where the global flux shapes are such that only a slight curvature is present across each assembly. When used on fuel boxes placed where the surroundings are very different, for instance near the reflector and near inserted control rods, the normalization method will give errors up to 30% in the local pin powers. In such cases the net surface currents have to be considered. The problem is that the ordinary way of calculating the homogenized parameters, the flux-weighted homogenization scheme, cannot preserve the nodal surface currents and the eigenvalue. Other homogenization methods have to be introduced. Some of them have been investigated and it has been shown that an "equivalence theory" (Koebke, 1978) will preserve both the eigenvalue and the nét surface currents.

Several methods of determining local pin powers have been examined based not only on the average nodal powers but also on the the fluxes and currents on the boundaries, and a new method named "superposition" has been developed. The basic principle of this new method is summarized as follows: During the assembly homogenization scheme several assembly calculations are performed with different boundary conditions. The solutions are called base-solutions. When the "actual" boundary fluxes and currents are known, the final solution is expanded in the base-solutions the coefficients of which are chosen by matching the "base-boundary" values to the "actual-boundary" values.

The main advantage of the superposition method as opposed to the other pin power methods is that the only assumption made in this method concerns the way the boundary parameters are approximated. The number of base-solutions required for each type of assembly depends on exactly how the local pin powers have to be known and how detailed the boundary currents and fluxes can be derived from the 3-D overall coarse mesh calculation.

REFERENCE

KOEBKE, K. (1978). A new approach to homogenization and group condensation. IAEA-TECDOC-231.

2.5. Iterative Techniques for Solution of Nodal Equations

When the neutron diffusion operator is approximated by nodal techniques, some of its sign-preserving properties are destroyed. This contrasts with the usual finite-difference formulation. Nevertheless, the iterative techniques employed to solve finite-difference equations apparently work quite satisfactorily in nodal programmes as well. However, it seems reasonable to investigate iterative techniques whose convergence can be proved. Therefore, the nodal equations used in the MIT program QUANDRY were solved by a number of alternative techniques. First, conjugate gradient methods were tried in combination with various forms of preconditioning. For small values of the mesh width the running time became too large compared with that of the original program. Next, Arnoldi's method was used and was found to work well for one-group two-dimensional problems, where there is a systematic decrease in running time. Numerical experiments are necessary to judge the performance of this method when applied to more complicated problems.

2.6. Three-dimensional PWR Calculations

The ANTI computer code is a three-dimensional program intended for overall PWR core calculations. In its static version it may be used in connection with burnup calculations, and in this mode of operation it forms part of the core simulator system described elsewhere in the present report. The chief aim of the program development, however, has been to obtain a capability for calculating transients other than that present in the loss-of-coolant accident to be used for PWR safety studies.

The main effort regarding ANTI during the past year has concentrated on testing the program. The calculated results have been successfully compared with those of other computer programs; one example of this is a benchmark calculation proposed by Finland for the meeting of the Nordic Reactor Physics Group held in November 1980. This calculation concerns a control-rod ejection accident in an imaginary reactor (Larsen, Nonbøl, and Thorlaksen, 1981). Comparisons with transient measurements made in an existing reactor have so far not been carried out.

The testing of the static part of ANTI by comparison with reactor measurements is in progress. The Swedish State Power Board (Statens Vattenfallsverk) has kindly supplied information about their Westinghouse PWR plant Ringhals 3 which went into operation in 1980. The data obtained from the Board for testing the ANTI results include:

1) Westinghouse calculations of control rod worths and moderator boron concentrations for various critical configurations,

2) burnup calculation from Vattenfall performed prior to reactor startup, and

3) measurements of the power distribution in the reactor core at power levels from 3% to 100% and with different control-rod patterns.

The preparation of ANTI input data for Ringhals-3 has now been completed and preliminary calculations show good agreement with the measurements. However, one problem in connection with ANTI arises with the use of the nodal method for the three-dimensional power distribution calculation. This method is very fast, but its results are highly dependent upon the so-called g-factors, i.e. four empirical constants which are chosen by comparison with more exact calculations. In order to remove the uncertainty involved in this fitting, the development of a new program has been initiated; its aim is to produce a fast, but accurate, three-dimensional coarse-mesh neutron kinetics code.

The program now under development is based on the Nodal Expansion Method (NEM),an improved nodal technique, where the inter-nodal coupling coefficients are determined by means of one-dimensional diffusion calculations assuming separability within the nodes of the fluxes in each spatial direction. The one-dimensional fluxes inside each node are expanded after polynomials, and the expansion coefficients are determined by weighting moments. A very strong coupling between 1-D and 3-D calculations is achieved and this provides good convergence properties for the method.

In the basic form of the method quadratic polynomials are used, but for improved accuracy higher-order expansion functions (third and fourth order) must be included. An important advantage of the NEM method is that even the basic variant converges to the exact diffusion equation solution when the mesh size is reduced. Another property of NEM is that the polynomial expansion coefficients need not be stored. Only the coarse-mesh variables, average fluxes, and partial currents must be kept in storage.

At the present stage only a simple (few expansion coefficients), static version has been programmed to test various acceleration techniques for the iteration scheme. A version with third- and f[illegible]th-order expansion functions is under development, and it is planned to continue with a version for dynamic calculation.

2.7. The Ringhals-3 Power Plant Model

The power plant model PWR-PLASIM has been modified and further developed for unit 3 at the Ringhals station. It has a Westinghouse PWR nuclear steam supply system with 3 primary loops and a Stal-Laval twin turbine system with a total capacity of 960 MW electrical. The work is carried out in co-operation with the Swedish State Power Board, which has given us all the data.

The structure of the model has been simplified from three to one primary loop and from two to one turbine system, using scaling factors at the boundaries. The model, therefore, includes the following units: reactor, one primary circuit with pump, pressurizer and steam generator, the steam load circuit with steam line, turbine and feedwater heaters, and all units with associated control circuits.

The model is one-dimensional for both the reactor and steam generator, but with calculations of the mean and hot channel in the reactor. The mean channel gives the temperature feedback and the hot channel the void feedback to the neutron kinetics, which is calculated by the one-dimensional diffusion equation with prompt jump approximation. Outside the core, the coolant is limited to single-phase flow only.

The steam generator model is designed to work at steam qualities of up to one, in order to be able to simulate dry-out by loss of feedwater.

The main purpose of the steam line model is to simulate the pressure and flow oscillations that arise at sudden load variations. The turbine model includes one HP- and one LP-turbine with one reheater and 6 feedwater heaters.

The working range is from zero to full power with the possibility for including high-power excursions. The model can be used for calculating transients for both normal operation and incidents such as turbine trip, loss of feedwater flow or feedwater heating, run down of primary pumps, or various valve fail-

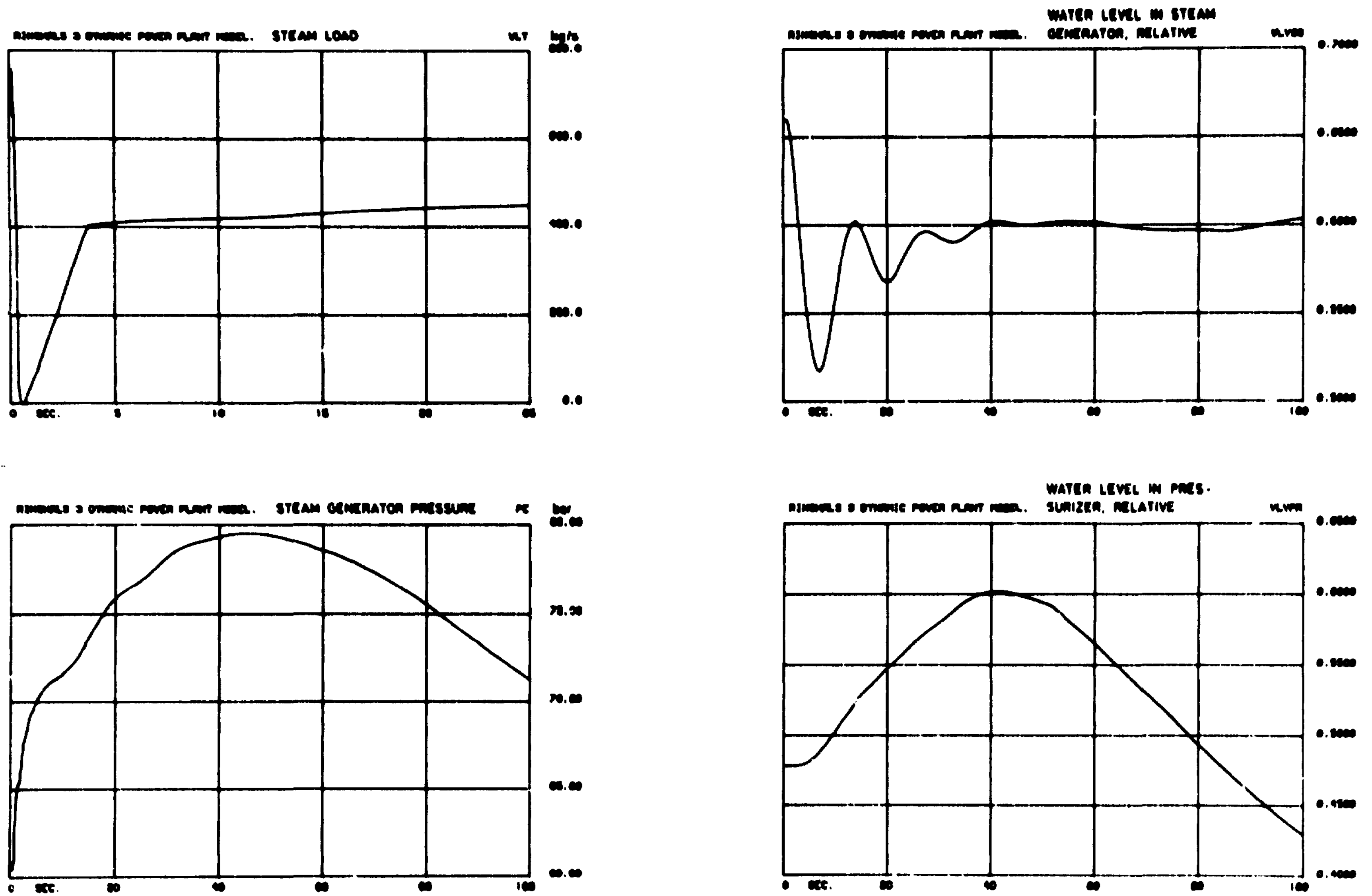
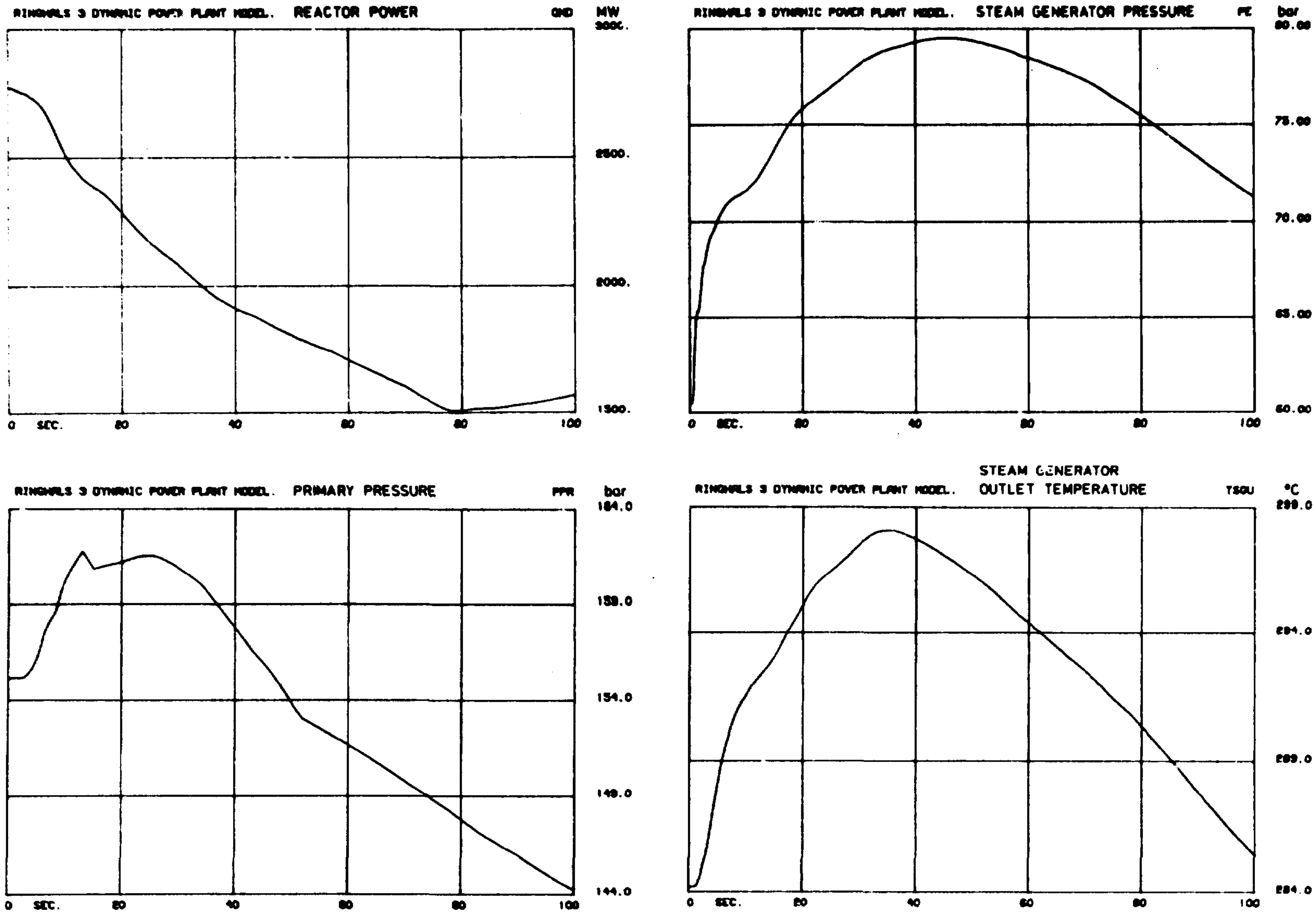


Fig. 3. Turbine trip transients



<u>Fig. 4</u>. Turbine trip transients

ures. The main limitations are that normal cooling in the reactor core and single-phase fluid flow in the primary circuit must be maintained.

An example of a transient calculation is given in Figs. 3 and 4 showing some main variables for a turbine trip without reactor scram which normally follows automatically. The results are only provisional as several data are uncertain or even estimated in cases where real data are not yet received. In any event, it is quite clear that the dump system and the control circuits can handle the situation in a safe manner even without introducing reactor scram.

2.8. DR 3 Calculations

In 1979 the Section of Reactor Physics and Dynamics was asked by the DR 3 (Danish reactor no. 3) operating staff to make a study of the consequences of reducing the enrichment of the reactor fuel from its present 93% to a much lower value, e.g. 20%. DR 3 is a 10-MW heavy water, tank-type research reactor.

Cell burnup calculations were made with the CCC program, which is a multigroup, transport theory code, capable of dealing with the annular DR 3 geometry, although it was originally developed for rod-type power reactors.

From the start the results were suspected of being somewhat erroneous due to the way, in which the U^{238} resonance absorption was treated. This was confirmed when these results were compared with others presented in an IAEA benchmark calculation on a problem very similar to the Danish study. A new, much more general approach to dealing with resonance absorption had to be developed and this lead to the adoption of the so-called subgroup method.

The new results now compared favourably with other contributors with regard to reactivity, flux spectra, etc., but still there were certain discrepancies, such as a plutonium build-up that was too slow and a significant fuel temperature coefficient in the 93% enrichment case. As the fuel temperature coefficient is almost solely due to Doppler broadening of the U^{238} resonances, it ought to be very small in the 93% case, where the U^{238} concentration is low.

A closer examination of the phenomenon showed that the effect came from the way in which the resonance data library was generated. Being originally intended for low-enriched power reactors, the U^{235} absorption and fission cross sections in the library contained an appreciable temperature variation due to the overlapping effect of the U^{238} resonances.

The obvious remedy was to generate more data libraries, each to be used for a certain enrichment range.

Thus, this international benchmark excercise has shown the problems encountered when a well-tested power reactor computer program system is applied to a radically different reactor, and it has led to significant improvements of the basic reactor physics program system.

2.9. Small Break Analysis

Two projects are carried out in a Nordic co-operative effort with the following participants:

- Institutt for Energiteknikk, Kjeller, Norway,
- Statens Tekniska Forskningscentral, Helsinki, Finland, and
- Studsvik Energiteknik AB, Studsvik, Sweden.

The aim of the first project - called SÄK-5 - is to provide computer codes for a small break loss-of-coolant accident (LOCA) analysis for light water reactors.

The aim of the second project - called SÄK-5 - is to establish a set of reliable heat transfer correlations for application in advanced computer codes for LOCA analysis.

Within the SAK-3 project three selected computer codes will be examined in theory as well as practice by means of a number of representative test cases. The work at Risø will be concentrated on the TRAC-PF code. (The first version, PF1, was received from Los Alamos National Laboratory in October).

So far, work has been carried out within the following areas:

1) Selection of test cases.
   Two test cases have been selected. The first is a 2.5% cold leg break experiment conducted in the LOFT reactor in Idaho (Test L3-6). The second is a 0.4% cold leg break experiment performed in the electrically heated LOBI test facility at the CEC Research Center in Ispra (Test SDSL-03).

2) Treatment of test cases.
   Procedures for treating each test case have been established together with a format to which the documentation should conform.

3) Theoretical code studies.
   The computer codes involved are to be studied theoretically in order to increase knowledge and pinpoint weak spots. This work has begun.

4) Preparation of input.
   The preparation of the TRAC-PF1 input deck for the LOFT L3-6 experiment is in progress.

Within the SAK-5 project the first step has been to conduct a literature survey concerning existing wall and interfacial heat transfer correlations relevant to the description of LWR LOCAs. The work at Risø has been concentrated on the transition boiling heat transfer regime, and three correlations have been chosen for further study.

2.10. The BWR Emergency Core Cooling Program NORCOOL-I

NORCOOL-I is a fixed-geometry thermo-hydraulic computer code simulating reflooding of a BWR. The code has been developed during the NORHAV co-operation.

Code modifications were introduced during the year to prevent calculational breakdowns and mass balance errors.

10 selected relevant experiments from the Swedish Göta reflooding experiment series have been simulated using NORCOOL-I. Fair agreement between calculation and experiment was obtained, but with the general trend that the code predicted the reflooding to be too fast at the beginning of the experiment and too slow at the end.

The final version of the code, version 812, was released to the NEA computer Library in October 1981. The updated documentation reports are finished, but are awaiting printing.

2.11. The Advanced BWR Emergency Core Cooling Program NORCOOL-II

NORCOOL-II is an advanced thermo-hydraulic computer code with input specified geometry for simulation of a LOCA in a BWR.

The code, as existing at the beginning of 1981, was a result of a joint Nordic work within the NORHAV cooperation, but the development in 1981 was a pure Danish effort. A number of physical models for water film and droplets were introduced into the code. The updated code was tested using, as a test case, the refilling of a 4-m downcomer connected to a lower plenum and with an open end at the top. It proved necessary to introduce a number of error corrections and modifications in the solution method and in the physical models. After this second update the initial 6 seconds of a slow refill (nominally 2.5 cm/second) was simulated successfully with a 0.25-second timestep consuming 80 minutes of computer time (B6700).

2.12. Emergency Core Cooling Experiments

In order to become acquainted with the situation after a LOCA a test facility has been constructed. The special purpose of the test facility is to investigate the parallel channel interaction during flooding.

During the construction period some burnout measurements were carried out. The measurements have been compared against the burnout code proposed by Becker. The deviation was less than 15%.

Some preliminary measurements of the natural circulation in the test facility were carried out.

2.13. Experimental Study of Rewetting and Quench Phenomena

An experimental programme has been started for the purpose of investigating the behaviour of electricity heated fuel pin simulators during reflooding with special emphasis given to the pin composition. The experiment includes directly heated pins, conventionally indirectly heated pins and advanced indirectly heated pins containing uranium dioxide as filler material.

The test facility has been erected and experiments with the first test section, a directly heated tubular section to be used as reference, were successfully started in October 1981. The programme will last 3 years in all and will be terminated at the end of 1983.

The programme is a part of the Commission of the European Communities "Indirect Action Programme on the Safety of Thermal Water Reactors" and is parttially financed by the CEC.

2.14. Consequences of a Large Accident

The dose consequences of a hypothetical large accident with a core melt-down in a nuclear power reactor were calculated. Three different fission product release categories were studied: The first was the BWR-2 release (of WASH 1400) where there is an early major rupture of the containment, and the release fraction is very high. The second takes into account the evidence of recent work (on PWR's) that the probability of an early major fracture is extremely small, and that the release of essential fission products (as I, Cs, Sr) is decreased by an order of magnitude with a late rupture. The noble gases like Xe and Kr are assumed to escape to a great extent in all three cases. The second release category is represented by the PWR-4 release. The third is established as a best estimate from empirical data, and it is called the BEED case. The experience from accidents at TMI and SL-1 is used in the estimate of the BEED release. The effects of physical and chemical processes in the water-steam atmosphere in the containment reduce the release fraction by at least one order of magnitude, and the third release category is two orders of magnitude lower than BWR-2 category.

The long-term dose consequences to the population are roughly proportional to the release fraction of the fission products (see Fig. 5). The noble gases have a significant influence on the bone marrow dose, and as the noble gas release is assumed almost constant in the three cases,the proportionality is not maintained for the bone marrow dose.

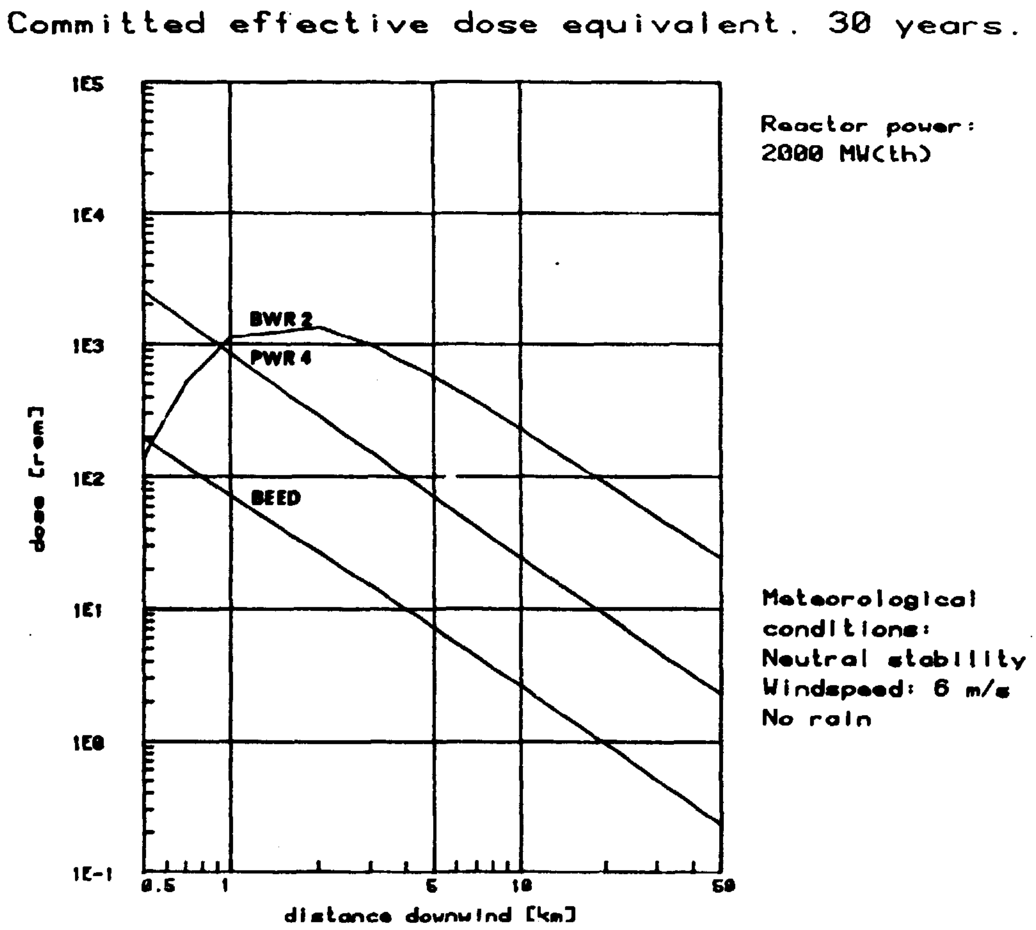


Fig. 5. Total committed effective dose equivalent to individuals from the BWR-2, PWR-4, and BEED releases. Doses from deposited activity are integrated over 30 years.

For the two large release categories, BWR-2 and PWR-4, the I-131 in the plume is dominant in the acute dose, while the long-term dose is mainly that due to gamma dose from deposited activity (Cs). For the BEED case the noble gases are the most significant radioactive fission products both for the long-term and acute doses.

One of the results is that the maximum relocation distance for the three release categories are:

BWR-2 release:	15	km	maximum	relocation	distance
PWR-4 - :	3.5	-	-	-	-
BEED - :	0.9	-	-	-	-

where the relocation criteria is that a reduction of 10 rem/month is obtained by evacuating people.

This work on a large accident was accomplished in a co-operative effort between the Health Physics and Reactor Technology Departments of Risø.

2.15. Oil and Gas Reservoir Models

During 1981, a project on oil and gas reservoir modelling has been initiated. The project has been undertaken jointly by Risø and the Laboratory of Energetics at the Technical University of Denmark in collaboration with other laboratories at the University and with the Geological Survey of Denmark, The Danish Energy Agency, and Danish Oil and Natural Gas, Ltd.

The reason for the project is the increased interest in and need for reservoir simulation in connection with the development of the oil and gas reservoirs in the Danish part of the North Sea.

The object is to implement, modify, and develop computer models for reservoir simulation. The present work has consisted of the implementation and study of existing two- and three-dimensional black-oil models, combined with more fundamental studies. These last mentioned have among other subjects been concerned with numerical and fluid mechanics problems in connection with the simulation of fractured reservoir rocks (crackpermeability).

The work may be seen as a natural development of the work on models reported earlier for the heat storage in aquifers and for geothermal energy, as the models have obvious similarities. The work is therefore also proceeding in close contact with the continuing studies on aquifer heat storage models.

2.16. Solar Heating of Buildings

The feasibility of solar heating of buildings by means of large focusing collectors has been investigated in a pre-project under contract with The Ministry of Energy. A preliminary design has been made of a dish-type, tracking collector, with a 5-m paraboloidal reflector. The collector has a rim angle of 110^{o} and a spherical absorber, and the concentration factor is slightly above 50. The reflector is designed as a light-weight construction with vacuum-shaped facets of metalized acrylic, mounted on an aluminium frame. Protection against wind and precipitation is supposed to be provided by a transparent dome surrounding the collector. A detailed mathematical model for the analysis of collector efficiency with emphasis on radiation intercept, as well as heat-removal factors has been developed. The annual collector yield at about 100°C would be about 700 kWh/m^2 as evaluated on the basis of the Danish standard reference year. However, measurements of the insolation (being part of the project) and correlation with sunshine hours seem to show that the direct radiation (annual average) is some 25% below that of the reference year. The objective is to investigate the feasibility of solar heating systems including seasonal storage. Some preliminary investigations of both individual and collective systems have been made. A report describing the work is in preparation.

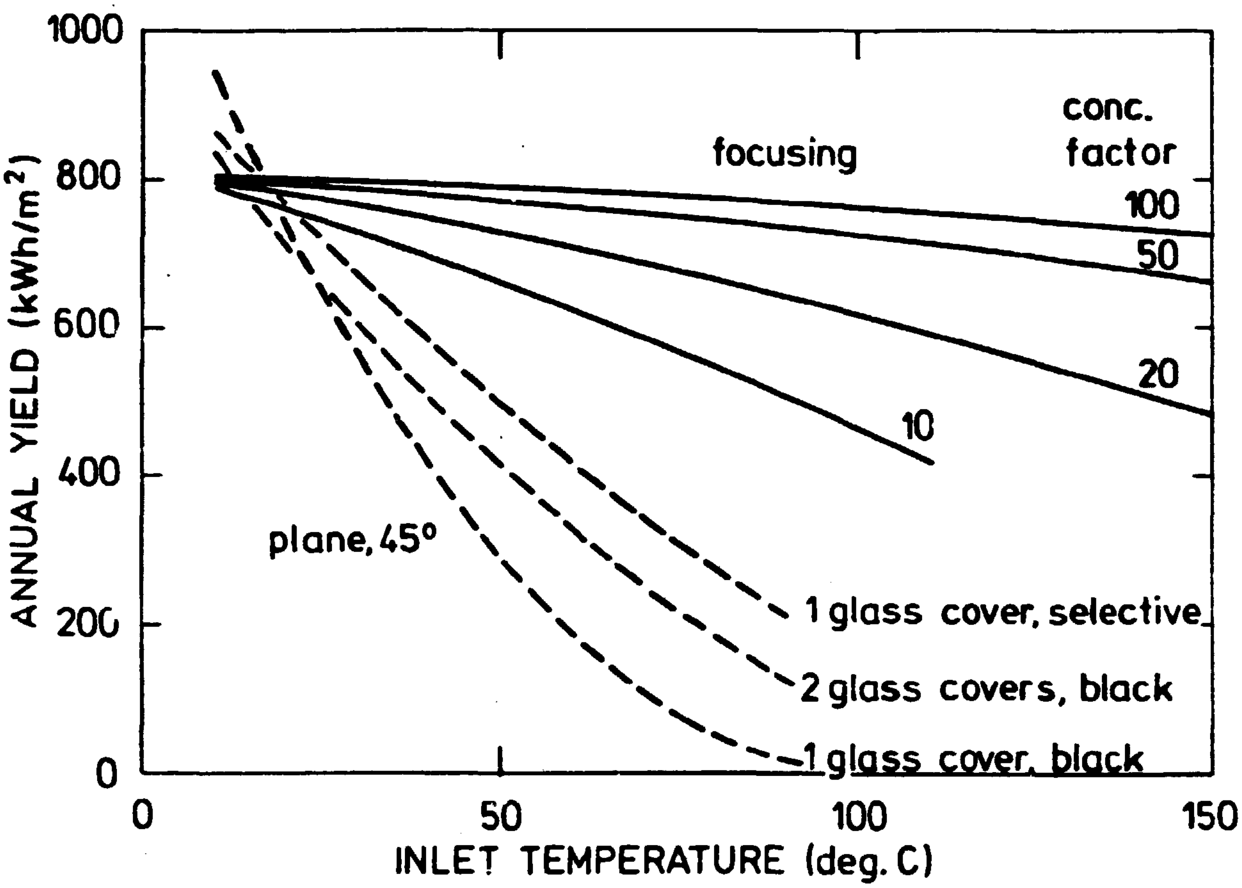


Fig. 6. Calculated solar collector yield versus inlet temperature for focusing collectors with various concentration factors (full line) compared with various types of plane collectors (dashed line).

2.17. The Temperature Calibration Laboratory

The Temperature Calibration Laboratory was authorized in 1978 by the Danish National Testing Board to carry out certified calibrations of temperature sensors in the range from -150°C to 1100°C, according to the International Practical Temperature Scale IPTS-68. The standard thermometers in the laboratory are traceable to the National Physical Laboratory, England. Late in 1981 the authorization of the Laboratory was renewed for another 3-year period.

In 1981, the Laboratory had performed 68 jobs for external customers and 9 for other Risø departments. In all, 210 thermometers ranging from liquid-in-glass models to advanced digital types and 5 thermostats have been calibrated during the year. The calibrations have been made in the temperature region from -100°C to 1100°C, which covers the main part of the range authorized.

2.18. Surface Temperatures of Radwaste Containers

For metallurgical reasons, the surface temperature of radwaste containers imbedded in salt is required.

The mathematical model applied is that of a linear heat source representing the decay heat of the waste. The line source is surrounded by a steel cylinder which in turn is encompassed by salt. Thus, the model is cylindrical with two regions of different material parameters. The line source varies with time, making the problem a dynamical one.

After a Laplace transformation of the heat conduction equation, the Green's functions for the two regions are pieced together to form the Green's function of the problem. This latter Green's function, in turn, is approximated by a sum of functions which are Laplace transforms of known functions. Finally, the surface temperature is found by convolution. The method can readily be generalised to other distances from the axis and to other simple geometries (point sources or planar sources), and the computations are made rapidly.

The surface temperature depends on the radius of the steel cylinder and on the properties of the salt. Typical results for an initial source strength of 300 W/m and a cylinder radius of 37.5 cm is that the maximum surface temperature is about 50°C above the initial temperature of the salt, and that excess surface temperatures of this order exist for a period on the order of 10 years. The maximum temperature itself is obtained after 6 to 8 years of deposition.

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APPENDIX

COMPUTER PROGRAMS

Description	Code name

Reliability Analysis

System Reliability — MOCARE

Calculates reliability characteristics for systems of any kind using Monte Carlo simulation with or without variance reduction. Particularly suited to systems with complex design and/or operation. Very flexible modelling based upon "subsystem models". User-friendly iteractive input code is available, and a graphical display of both component and system performance can be obtained. Computer time may be excessive in special cases, for instance, in the case of calculation of the probability density function for the unavailability of a system, if the magnitude of the failure rates varies by several decades.

Component Reliability — ANPEP

Calculates the probability of failure of a structure for stress-strength models used in probabilistic fracture mechanics. The parameters describing stress and strength are given by probability density functions. The calculations are based upon numerical integration in one or several dimensions. In the present version of the program the number of parameters are limited to six.

Reactor Physics and Dynamics

Group Cross Section Generation — SIGMA

SIGMA generates a 76-group neutron cross section master tape for reactor physics calculations from UKNDL. Resonance absorption and thermal scattering cross sections are treated separately in the program RESAB and NELKINSCM.

Resonance Absorption — RESAB

RESAB generates self-shielded, few-group neutron cross sections in the resonance region using collision probability calculations in several thousand groups.

Scattering Data — NELKINSCM

NELKINSCM generates multigroup neutron scattering cross sections in the thermal region. The program uses the NELKIN model for H and D and the Free Gas Model for other nuclides.

Pin and Cluster Cell Calculation — CCC

CCC is a 76-group, collision probability, pin cell calculation program. In addition,collision probability cluster cell calculations and burnup calculations in an arbitrary number of groups can be performed. The program has facilities for condensation and homogenisation of cross sections for further use in fuel element calculations.

Fuel Element Calculations for LWR — CDB

CDB performs burnup calculations for LWR fuel elements. Few-group cross sections to be used in overall three-dimensional flux distribution calculations by means of coarse-mesh methods can be generated. The program uses multi-group collision probability pin cell calculations with burnup, and XY-diffusion theory calculation for calculation of power distribution within the element.

Steady-State Neutron Diffusion Theory

The steady-state neutron diffusion equation for calculation of reactivity and flux-power distribution can be solved in two and three dimensions with different approximations.

TWODIM is a two-dimensional difference equation program, using center mesh points. (X,Y), (R,Z), and (R,θ) geometry can be used. Arbitrary number of groups. TWODIM

TVEDIM is a two-dimensional difference equation program, using corner mesh point. (X,Y), (R,Z), and (R,θ) geometry can be used. Arbitrary number of groups. TVEDIM

FEM uses the finite element approximation for two-dimensional calculations. Up to 5th order has been used. (X,Y) geometry and arbitrary number of groups. FEM

DC4 is a three-dimensional difference equation program using corner mesh points. (X,Y,Z) geometry and arbitrary number of groups. DC4

FEM3D uses the finite-element approximation for three-dimensional calculations. Up to 3rd order can be used. (X,Y,Z) geometry and arbitrary number of groups. FEM3D

SYNTRON is a three-dimensioanl single channel flux synthesis program. (X,Y,Z) geometry and arbitrary number of groups. SYNTRON

Three-dimensional Steady-state BWR Simulator — NOTAM

NOTAM is a three-dimensional steady-state BWR simulator. It combines a neutronic module based on FLARE-type nodal theory with a hydraulic module based on multichannel, one-dimensional flow. The program has facilities for burn-up calculations.

Three-dimensional PWR Simulator — ANTI

ANTI is a three-dimensional PWR simulator. It combines a neutronic module based on FLARE-type three-dimensional nodal theory with a hydraulic module based on subchannel flow. The program can perform both steady-state and transient calculations for a PWR core. In addition it has facilities for burnup calculations.

Three-dimensional BWR Transient Simulator — DANAID

DANAID is a three-dimensional BWR core transient simulator. It combines a neutronic module based on FLARE-type three-dimensional nodal theory and a hydraulic module based on multichannel one-dimensional flow.

Simulation System for Dynamic Processes — DYSIM

DYSIM is a standardized program system for simulation of dynamic processes which are described by a mixed set of differential and algebraic equations. The state variable concept is used with the possibility for shifting variables between state and algebraic variables. True delay simulation may be included. Fortran programming by subroutines which are bound to DYSIM. Standardized input with specification of initial conditions, pertubations, control parameters, and output tabulations.

<u>Simulation of PWR Plant Dynamics</u> PWR-PLASIM

Dynamic model of a PWR power plant for transient calculations at "normal conditions". The primary circuit consists of one loop with reactor, pressurizer, steam generator, and pump. One-dimensional calculation of nuclear power coupled to hydraulic equations, and one-dimensional calculation of heat transmission to the secondary side of the steam generator. Secondary side with one-dimensional steam line, turbine model, and feedwater system. Main control and protection systems are included.

<u>Simulation of BWR Plant Dynamics</u> BWR-PLASIM

Dynamic model of a BWR power plant for transient calculations of "normal conditions". One-dimensional calculation of nuclear power coupled to hydraulic equations. Forced recirculation in reactor with external pumps. The steam load system consists of a one-dimensional steam line, turbine model and feedwater system. Main control and protection systems are included.

<u>Simulation of PWR Plant Dynamics Including Detailed Turbine Model</u> BWR PLANT

Dynamic model for a BWR power plant for transient calculations. The reactor model consists of a one-dimensional, multigroup, finite element neutronic model and a one-dimensional model of hydraulics. The reactor model is coupled to a detailed turbine model consisting of a high-pressure turbine, a moisture separator, a reheater, a condenser, feedwater heaters, and extractions for district heating.

<u>Fuel Management</u> SOFIE

SOFIE is a program which can determine economically optimal refuelling strategies. It is a combination of a one-dimensional reactor physics model of a reactor core and a module for calculating fuel economics. Optimization is carried out by means of linear programming.

Reactor Thermo-Hydraulics and Accident Analysis

Reactor Steady-State Heat Transfer and Hydraulics — SDS

Subchannel core heat transfer and hydraulics. Best-estimate, steady state, flow, and enthalpy, BWR and PWR, non-equilibrium, drift-flux, boundary value solution technique, saturated steam only. Arbitrary number of subchannels.

PWR Blowdown — TINA

Calculation of flow rates, void fractions, and liquid and fuel rod temperatures in the PWR core. Best estimate, subchannel approach, non-equilibrium, drift flux. Arbitrary number of subchannels. Saturated steam only. Fuel rod damage is not considered.

BWR Top Spray Emergency Core Cooling — CORECOOL

Best estimate. Single loop geometry, spray cooling, one-dimensional non-equilibrium two-fluid plus falling films, detailed radiation heat transfer. Results: Two-phase hydraulic and thermal state, temperatures of fuel, cladding, and fuel element box.

BWR Emergency Core Cooling — NORCOOL-I

Best estimate. BWR-vessel geometry with single fuel element. Spray injection and normal bottom reflooding. One-dimensional numerics, saturated steam drift-flux in continous water, non-equilibrium two-fluid plus falling films in continuous steam regions, explicit two-phase level and quench front tracking. Detailed radiation heat transfer. Results: Two-phase hydraulic and thermal state, temperatures of fuel, cladding, and fuel element box. (Does not calculate blowdown).

PWR Blowdown and Emergency Core Cooling **TRAC-PF1**

Best estimate system code. Any system geometry can be modelled with pipes, tees, valves, pumps, etc. One-dimensional numerics, three-dimensional numerics optional for reactor vessel component. Non-equilibrium two-fluid hydraulics. Moving mesh in fuel rod heat conduction calculation doe quench front tracking. Point kinetics. Results: Two-phase hydraulic and thermal state, temperature of fuel, cladding, and heat structures. (Program developed by Los Alamos, USA).

PS-Containment LOCA Response **CONTAC-III**

Calculates pressure- and temperature response to LOCA for a pressure suppression-type containment. The code includes models of the reactor vessel and two containment volumes. The code assumes thermal equilibrium and is based on a quasi-stationary calculation.

Containment Response to Core Meltdown **MARCH**

Models a variety of physical processes consistent with the phenomena expected to be associated with meltdown accidents in light-water power reactors: heat-up and boil-off of water in reactor vessel, clad oxydation and slumping of the fuel, vessel melt-through, interaction of core debris with water and concrete and hydrogen burning. Results: Containment building response: temperature, pressure and inter-compartment flows, and release of fission products from fuel. (Program developed by Battelle Columbus, USA).

Fission Product Transport and Removal **CORRAL**

Fission product removal processes in the containment as leakage, deposition and spray are modelled. The containment thermal hydraulics and fission product source terms required for the program are provided by

MARCH. Results: Quantity of airborn fission products available for release to the environments at any time. (Program developed by Battelle Columbus, USA).

Aquifers and Reservoirs

Heat Storage in Aquifers - Linear Finite Element Model **PORFLOW**
x-y or r-z geometry. Calculates the transient pressure and temperature field during heat storage/withdrawal. Simplified representation of temperature field by a hot and a cold zone.

Heat Storage in Aquifers - Two-dimensional, Quadratic Isoparametric Finite Elements **D2AQ**
Detailed temperature and pressure field during injection/withdrawal in a porous medium. z-y and r-z geometry. (Program from Laboratory for Energetics, Technical University of Denmark).

Ground Water Pollution Dispersion **SWIP**
Three-dimensional, finite-difference method for transient calculation of pressure- and temperature distribution, and transfer of fluid and soluble matter through a porous matrix. x-y-z and r-z geometry. (Program from U.S. Geological Survey).

Petroleum Reservoir Simulation **DM**
Black Oil, unsteady-state reservoir simulator. Two space dimensions (X-Y plane), three components and three phases including water. The gas may be dissolved in the water phase. The code calculates the distributions of flow rate, pressure, and saturation for each phase as well as production rates and recovery factors. (Program developed by Comtech, USA).

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