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1. Introduction

The Research Center Rossendorf Inc. that originates from the scientific traditions of the former Nuclear Research Center Rossendorf has been founded on January 1st, 1992. The Institute for Safety Research is one of the five scientific institutes of the Research Center Rossendorf Inc.

This institute is concerned with evaluating the design based safety and increasing the operational safety of technical systems which include serious sources of danger. It is further occupied with methods of mitigating the effects of incidents and accidents. For all these goals the institute does research work in the following fields

- modelling and simulation of thermofluid dynamics and neutron kinetics in cases of accidents,
- two-phase measuring techniques,
- safety-related analyses and characterizing of mechanical behaviours of material,
- measurements and calculations of radiation fields,
- process and plant diagnostics,
- development and application of methods of decision analysis.

At present mainly problems concerning the safety of the Soviet-type VVER reactors are worked on. These studies are focussed on the more recent types VVER-440/213 and VVER-1000, according to the recommendations given after the evaluation in 1990/91 by the German Scientific council.

Most of the methods the institute is concerned with are largely independent of the special objects they are applied to or can, at least, be adapted to other applications. For this reason and in regard of increasing safety requirements and safety consciousness the scope of our work will be continuously expanded to technical systems and facilities that do not belong to the field of nuclear technology but are also source of considerable risks. Within the field of industry chiefly facilities intended for manufacturing, storing, transporting and transloading toxic, combustible and explosive substances must be taken into consideration. In addition, safety research should of course also deal with the non-nuclear technology of power generation and the technical systems of waste disposal and distribution, e.g. waste deposits, combustion facilities and facilities for special waste. In accepting these tasks the institute has begun to work on studies of application of decision analysis with respect to the evaluation of abandoned polluted areas and the selection of remediation strategies.

The following principal results have been obtained in the above-mentioned fields of research:

The analyses of courses of nuclear accidents refer to the accident behaviour of VVER reactors. For this purpose the thermohydraulics code ATHLET of Gesellschaft für Reaktorsicherheit (GRS) and the Rossendorf neutron kinetics programme DYN3D were used. Within the framework of a project sponsored by BMFT the two codes are coupled to generate an advanced tool for VVER.

The interfaces that are of importance for coupling have been already defined in agreement with GRS. Moreover the programme for neutron kinetics will be equipped with a burn-up module, which is now nearly complete, and connected to modern data libraries. The verification of DYN3D for the three-dimensional modelling of core behaviour in reactivity-initiated accidents is continued. The programme has been tested by comparison with other codes and by recalculation of experiments to such an extent that it could be made accessible to research

institutes and supervisory boards in the Central and East European countries with the aid of the IAEA. It is already used by Czech, Slovakian and Bulgarian institutions.

Another project, which is supported by SMWK (Saxon Ministry of Science and Arts), deals with the application of the thermohydraulics programme ATHLET to VVER reactors. Pursuing this project our institute has successfully taken an active part in the benchmark test for the examination of the international standard problem 33.

As to two-phase measuring techniques, the development of needle-shaped conductivity probes is continued and has already led to an applicable two-point probe for steam content and steam velocity measurements. In further work the stability of the probes shall be improved. The probes are applied to large thermohydraulic experiments, for instance in PKL in Erlangen and in the integral facility ISB for VVER-1000 in Elektrogorsk (Russia). The most important actual task is the participation in the experiments concerning the specification of the IAEA standard problem SPE-4 at the facility PMK in Budapest. By applying these probes our institute has the access to the experimental data necessary for post test calculations.

For testing the probes a depressurized water loop has been constructed, which can be also used in ultrasonics experiments aiming at the identification of the two-phase flow. These investigations are performed within a pre-project sponsored by BMFT.

Characterizing the mechanical behaviour of irradiated material of VVER steels from the so-called Rheinsberg Irradiation Programme is substantially hindered by the missing operating licence for the radioactive test laboratory. The reconstruction of the appertaining preparation laboratory is well under way. The studies were focussed on methodical development to determination of the correlation between mechanical-technological and fracture mechanics parameters. An instrumented Charpy test for determining the impact energy and the dynamic fracture resistance toughness, and a bending test for measuring fracture resistance curves by the single-specimen-compliance method have been developed. The capability of these methods has been proved in national and international round robin tests. By each of these two experimental methods the unirradiated initial states of RPV steels have already been tested. Tests on the dependence of pressure vessel embrittlement on depth position demonstrated that the surface layer of the pressure vessel is more sensitive to irradiation than hitherto assumed. For further elucidation of the embrittlement mechanism analyses by Abnormal Small Angle X-Ray Scattering will be carried out at HASYLAB in 1993. If the assumption of a more sensitive surface layer is confirmed, this will necessarily influence the surveillance programmes.

The calculations and the measurements of radiation fields aim at

- a) determinations of neutron fluences of reactor pressure vessels and
- b) transport calculations for shielding of magnets and for particle distribution in the chamber of an intensive 14 MeV neutron source on the basis of a so-called mirror machine. These studies are intended for promoting fusion material research.

To ensure the determination of the fluences of all samples of the Rheinsberg Irradiation Programme, all theoretical and experimental requirements for neutron fluence determination in pressure vessels do already exist: the Monte-Carlo codes TRAMO and TRAWEL, a low level measuring laboratory for evaluating irradiated activation detectors, and codes for spectrum adjustment. These calculations and measurements represent the content of a BMFT project;

they are executed according to the plan. Fortunately calculations performed with different nuclei data and different numbers of energy groups have led to approximately identical results.

The work on the plasma neutron source is performed in close cooperation with Efremov Institute in St. Petersburg and Budker Institute in Novosibirsk. In agreement with the Russian partners the contributions from Rossendorf deal with

- the development of a Monte-Carlo-transport code for the distribution of fast ions in the plasma in order to get information about distribution and intensity of the neutron source,
- the adaptation of the programme EIRENE of KFA Jülich to the conditions existing in the mirror machine for estimating energy and mass balance of the plasma; but the modification of EIRENE, which has been written for Tokamaks, has proved very extensive; therefore the implementation of an own, adapted model has been started,
- the estimation of the neutron fluences in the magnets of the plasma source; first results indicate that the superconductive parts of the magnet system can be reliably shielded; the radiation exposure of the non-superconductive mirror magnet, however, may become problematic.

At present the application of transport calculation methods to the localization of the radiation doses in human tissue, which has been subjected to tumor therapy by light ions, is in preparation.

The field of process and plant diagnostics comprises the subjects "vibration modelling for VVER-440", "leak detection in geometrically complicated structures" and "technical conception of a centralized safety and information system for the operation of East European Nuclear Power Stations".

As to vibration modelling for VVER-440, which is partially sponsored by BMFT, in our institute a Finite-Elements model (FE model) has been almost completely worked out for an 1:10 test facility. Modelling the original facility could be started. In parallel analytical and experimental tests were made in order to find out in which way the influence of the co-vibrating fluid in the reactor pressure vessel can be taken into account in form of point or area load or as an additional attenuation in the FE-model. Theory describes the influence of the fluid with a precision better than 10 %. The completely instrumented 1:10 facility is used for vibration tests to adjust the FE-model. Analogous experiments will be carried out in Greifswald in autumn 1993 and afterwards in the nuclear power plant of Dukovany in Slovakia.

Studies on the nonlinear numerical simulation of control element vibrations during abnormal core barrel motions could be finished with success at the end of 1992.

For developing an acoustic method of leak detecting and locating in geometrically complicated structures all prerequisites concerning hardware and software could be realized. A test equipment for studying ultrasonics propagation in complicated topology has been mounted, and the first series of measurements applying simulated leaks in the region of the reactor vessel head has been performed in the nuclear power plant of Greifswald. To measure the leak sound in such parts acoustic emission sensors for structure-borne sound and

high frequency microphones for air-borne noise are used. Localization is performed by means of pattern recognition, in particular the suitability of associative tensors and neural networks is tested. The method of acoustic leak detection will be largely applicable to many problems arising in chemical industry.

In cooperation with an American firm a feasibility study concerning remote monitoring systems for nuclear power plants in East Europe has been started and concluded in 1992. These remote monitoring systems are designed as instruments for the national supervisory authorities and, in addition, as prerequisites for international early warning systems. By order of BMU these investigations are continued to realize a technical conception for a pilot project in the Ukrainian Republic. In this pilot project cooperate at German side: TÜV Rheinland and the Research Center Rossendorf, and at Ukrainian side: the Saporoshje NPP, the Ukrainian supervisory authority and an institute of the Academy of Sciences. In contrast to the German nuclear reactor remote monitoring system (KFÜ) the pilot project in addition to meteorological and radiological data will also include about 100 technological parameters to render possible the prevention of nuclear accidents by the designed system. The structure of this information system should provide the opportunity of an international early warning and protection of people in case of unacceptable radioactive releases. The last mentioned item is the special subject of a project sponsored by SMWK.

Investigations and applications of methods of decision analysis with respect to safety problems are performed by our institute in cooperation with VKTA (Nuclear Engineering and Analytics Rossendorf Inc.) since autumn 1992. This is mainly intended for supporting the Saxon Ministry of the Environment in the evaluation of waste deposits and, of course, also of sites radioactively contaminated by uranium mining. Therefore the expert system XUMA of KFK Karlsruhe has been implemented in Rossendorf. The goal of this work is the connection of methods of decision analysis with this deterministic expert system. In this way the user will be enabled to take explicitly into consideration the uncertainty and the partially hypothetical nature of the knowledge about the content of pollutants in the dump or in the ground and about the migration of pollutants in the hydrosphere and in the lithosphere, when he establishes plans for analyses, strategies, priorities of remediation, etc. Meanwhile the bases for decision analysis have been completed with respect to the goal mentioned above. Furthermore the institute is working on coupling physical codes for simulating propagation of pollutants, especially radionuclide propagation, in the air and in the ground to the expert system.

In future the decision analysis is to be made applicable to the aims of accident management and crisis management. Applying these procedures to purposes of energy strategies is in preparation.

Section 2 gives some results of the research activities performed in the fields mentioned and also in some other topics.

Integration of the subjects into national and international research programmes and research cooperations

All the above mentioned topics are either complementary or even exclusively funded by external sponsors. This is indispensable in regard of only 13 scientists allowed by the staff plan. From the 10 projects dealing with reactor safety 6 are financed by BMFT, 1 by BMU

and 3 by SMWK. The studies on decision analysis application to the evaluation of contaminated sites have been financed by SMWK. An overview and more detailed description is given in section 3.

Even from the beginning of 1992 both the embrittlement behaviour analysis of pressure vessel steels and the calculations concerning the plasma neutron source are essential parts of the WTZ (scientific and technical cooperation) with Russia. In agreement with the project coordinator for reactor safety research the verification of reactor accident codes for VVER will be also integrated into the contract of scientific and technical cooperation with the countries in Central and Eastern Europe. Especially the code DYN3D for neutron kinetics is already made available to institutions in Eastern Europe by the technical cooperation project "Safety Assessment of VVER-440-Reactors" of IAEA. For this purpose an IAEA workshop that deals with DYN3D will be given in Prague in January 1993 by staff members of the institute.

The embrittlement studies are not only part of the WTZ programme, they are also included in the work of the Working Group on Reactor Dosimetry of VVER-Reactors, the European Working Group on Reactor Dosimetry and the Coordinated Research Programme "Optimizing of RPV Surveillance Programmes and Their Analyses" of IAEA.

The institute is furthermore a member of the research association "AER" of the VVER-operating countries, which is occupied with the physics and the safety of these reactors. The great number of cooperations with institutes and utilities in Eastern Europe represents a characteristic feature of our institute. The most important partners are: Kurchatov Institute Moscow, ENIS Elektrogorsk (Russia), Scientific and Technical Center of the Ukrainian Nuclear Power Supervisory Board, Novosibirsk Nuclear Physics Institute, Atomic Energy Research Institute (KFKI) in Budapest and the Institute for Nuclear Research at Rez near Prague. The relations to these institutions are maintained by common workshops and guest working visits (in 1993 we will be prepared for visits of about 60 person-weeks). In November 1992 a series of workshops on safety research for VVER reactors has been started together with the Rez Nuclear Research Institute. This series will be continued during the next years.

The institute cooperates with a great number of universities, in particular with the Technical University of Dresden and the Technical College of Zittau/Görlitz, but in addition there are also intensive cooperations with other institutions in Germany, especially with GRS, Bundesanstalt für Materialprüfung, IKE Stuttgart, Institut für Kerntechnik und Zerstörungsfreie Prüfverfahren Hannover, the University of Leipzig, TÜV Rheinland, and the Nuclear Research Center Karlsruhe.

For investigations concerning the remote monitoring of Eastern European nuclear power plants the study performed in common with the American firm E-Systems and the final meeting of the German and the American partners were of great importance.

Technical equipment

The pressing need of equipment with modern computers could be satisfied to a large extent in 1992. In addition to the use of several local PC networks the institut also disposes of 3 workstation clusters for accident and decision analysis and vibration modelling.

Since large-scale experiments are carried out at external experimental facilities, our own set-ups are relatively simple. In this context a depressurized loop needed for two-phase flow

investigations and for the test of the measuring equipment is worth mentioning, and further an 1:10 model of a VVER-440 that can be used for vibration analysis and flow mixing experiments, and a structure model to study structure-borne sound propagation in complicated structures.

Due to the conditions described above the institute presents itself as a favourable location for future test facilities, for instance with respect to the safety of VVERs, but first of all in fields of technical safety out of nuclear reactor technology.

At present the laboratories intended for testing damaged materials are completed; the licence for the radioactive preparation laboratory is expected in the near future. This laboratory is equipped with a servo-hydraulic testing machine, a tension-compression tester and a resonance fatigue test machine. In this way it will be very suitable for tests with radioactive materials, but for turning it into an universal testing laboratory completing its equipment will be indispensable.

2. Scientific Contributions

Modelling of Fuel Rod Behaviour and Heat Transfer in the Code FLOCAL for Reactivity Accident Analysis of Nuclear Reactor Cores

U. Rohde

The module FLOCAL for thermal-hydraulic calculation of reactor cores during Reactivity Initiated Accidents (RIA) performs a part of the code DYN3D/M2 for three-dimensional nuclear reactor dynamics, developed in Research Center Rossendorf /1/.

FLOCAL consists of

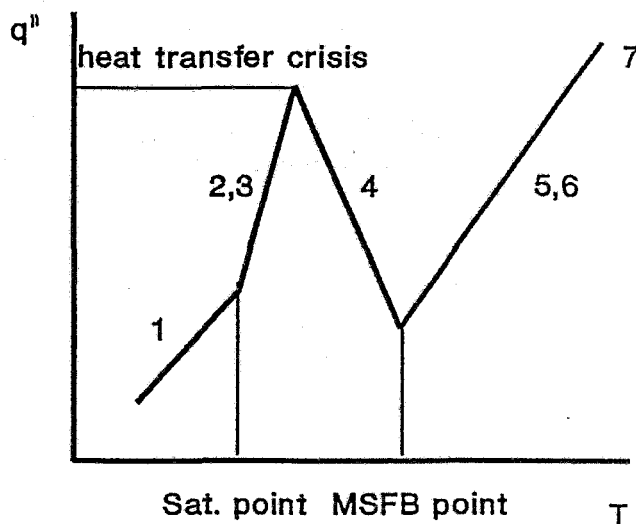
- a one- or two-phase coolant flow model on the basis of four differential balance equations for mass, energy and momentum,
- a heat transfer regime map from one-phase liquid up to superheated steam,
- a fuel rod model for the calculation of fuel and cladding temperatures and some parameters for fuel rod failure estimation.

In the present paper a detailed description of the fuel rod model used in FLOCAL is given. That includes the solution of the one-dimensional heat conduction equation, the estimation of heat transfer through the gas gap by conduction, radiation and fuel-cladding contact and a simple thermo-mechanical model. The thermo-mechanical model is based on the superposition of thermal, elastic and plastic deformations, while the cladding is described in thin shell approximation.

As parameters for diagnostic of possible fuel rod failure

- fuel enthalpy for each axial node of the rod,
 - cladding temperature and oxide layer thickness,
 - contact pressure in relation of the yield point
- are estimated.

The wall-to-coolant heat transfer model includes liquid convection, developed boiling, post-crisis heat transfer for inverted annular or dispersed flow and a transition region (Fig. 1). Heat transfer crisis is established by several criteria. Non-equilibrium corrections to Leidenfrost temperature and post-crisis heat transfer coefficients are considered.



- 1 one-phase liquid convection
- 2 boiling heat transfer
- 3 convection in annular flow
- 4 transition boiling
- 5,6 film boiling
(inverted annular or dispersed flow)
- 7 convection to superheated steam

Fig. 1
Heat transfer logic

For the validation of the model, calculations for RIA experiments from literature /2, 3/ were carried out (Fig. 2, 3). The results are significantly influenced by plastic deformation model, parameters of metal-water-reaction equation and non-equilibrium effects in heat transfer. A qualitatively good agreement with experimental results can be achieved. Due to lack of experimental material under power reactor conditions some sensitivity studies in this parameter region were carried out.

The results of the first validation activities demonstrate the applicability of the module FLOCAL to RIA analysis, but also the directions of further model improvements.

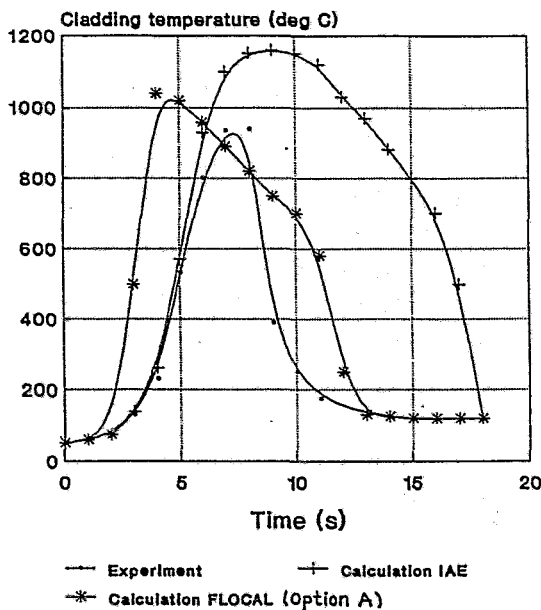


Fig. 2
 Results for a RIA experiment at the IGR facility, subcooling $\Delta T = 80$ K, $\Delta E = 360$ cal/g, $\tau = 300$ ms

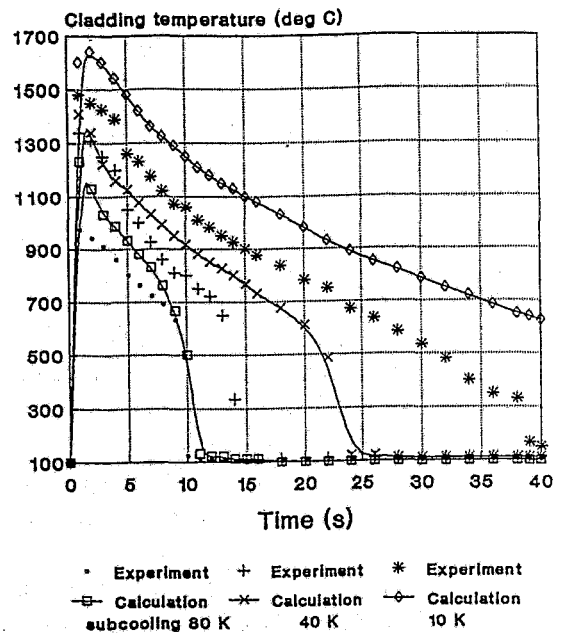


Fig. 3
 FLOCAL (Option B) calculations for NSRR RIA experiments with variation of fluid subcooling, $\Delta E = 190$ cal/g, $\tau = 7$ ms

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The Code DYN3D/M2 for the Calculation of Reactivity Initiated Transients in Light Water Reactors with Hexagonal Fuel Elements

U.Grundmann, U.Rohde

The code DYN3D/M2 is used for investigations of reactivity transients in cores of thermal power reactors with hexagonal fuel elements. The 3-dimensional neutron kinetics model HEXDYN3D of the code is based on a nodal expansion method for solving the two-group neutron diffusion equation. The thermo-hydraulic part FLOCAL consists of a two-phase flow model describing coolant behaviour and a fuel rod model. The fuel elements are simulated by separate coolant channels. Additional, some hot channels with power peaking factors belonging to chosen fuel elements can be considered. Several safety parameters as temperatures, DNBR and fuel enthalpy are evaluated. Macroscopic cross sections depending from the thermo-hydraulic parameters and boron concentration are input data of the code. The stationary state and transient behaviour can be analyzed.

Analyzing a static state, there exists some possibilities to make the reactor critical:

- Division of multiplication cross sections by K_{eff}
- Variation of boron acid concentration
- Variation of reactor power

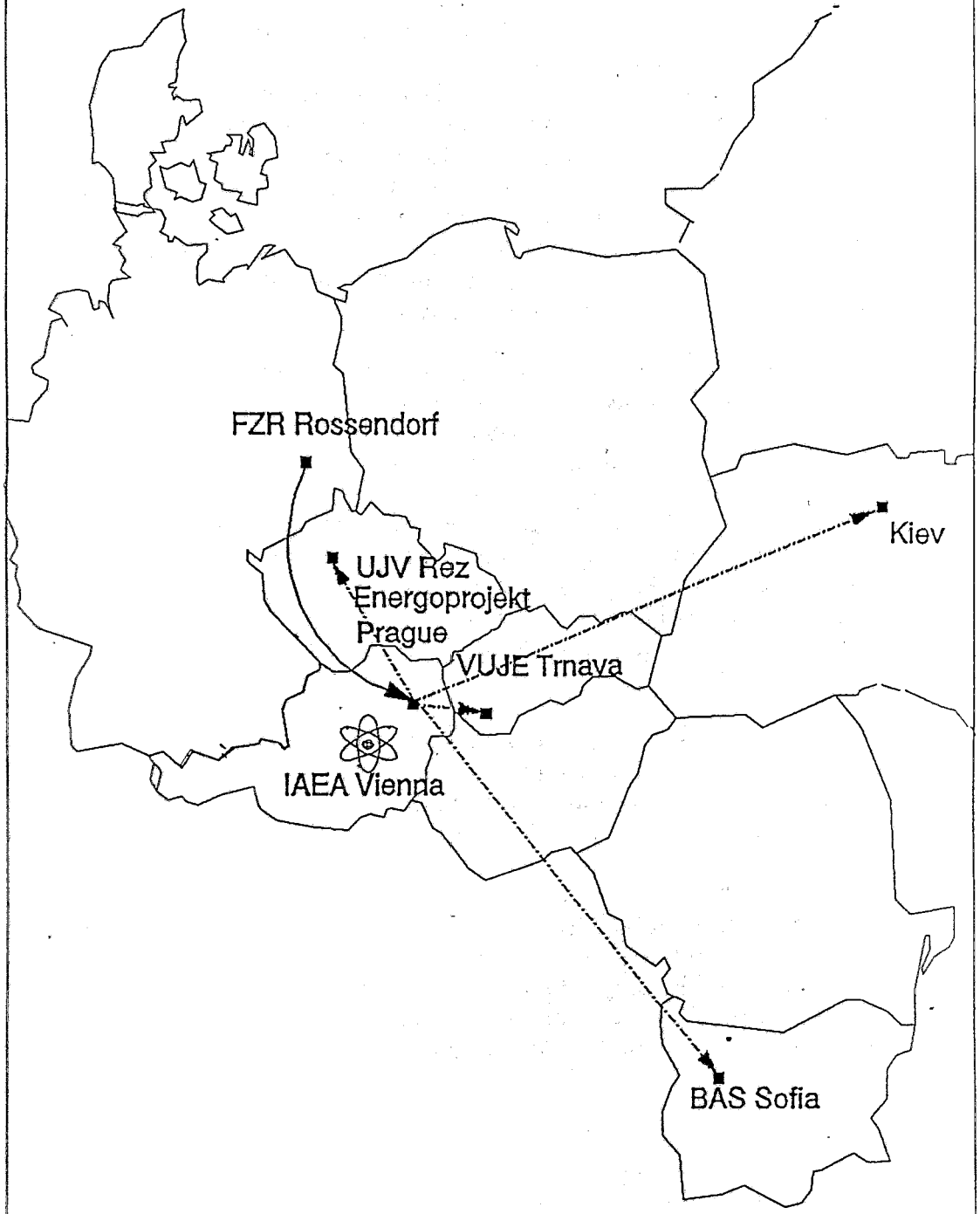
If a transient calculation should be carried out, the following perturbations can be treated:

- Movements of single control rods or control rod banks
- Variation of core coolant inlet temperature
- Variation of boron acid concentration
- Changes of core pressure drop or total mass flow rates
- Changes of pressure

The model and the basic equations of the code are described in Ref. /1/. The codes and its precursor codes HEXNOD23, HEXDYN3D, FLOCAL and DYN3D/M1 are validated by comparison with benchmarks, other codes and experiments (see /2, 3/). Some analyses of reactivity accidents in the reactor VVER-440 were considered by the help of DYN3D/M2 (see /4, 5/).

In the present report, the features of the code and the code structure are outlined. A detailed description of input data set, output data and post-processing of results is given for users. The code DYN3D/M2 is going to be included into the IAEA Technical Cooperation Project "Safety Assessment of VVER-440/V-213 Reactors". In the frame of this project, the code transfer to interested organizations in countries using VVER is supported by the IAEA (see Fig.).

DYN3D FOR SAFETY ANALYSES OF VVER REACTORS
Code Transfer to Interested Countries
Supported by the IAEA



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Paper presented at "1st Symposium of Atomic Energy Research", Rez near Prague (CSFR) 1991

Conductivity Probes for Two-Phase Flow Pattern Determination During Emergency Core Cooling (ECC) Injection Experiments at the COCO Facility (PHDR)

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1. Introduction

The scientific programme of nuclear safety investigations at the experimental power plant HDR in Karlstein includes thermal hydraulic emergency core coolant injection experiments. They were carried out at the COCO facility located within the containment of the HDR. COCO stands for COntact COndensation, that means, the study of the condensation phenomena in a mixture of saturated steam with sub-cooled water was one of the main goals of the tests.

In the case of the hot leg injection experiments, the COCO facility models the part of the main circulation loop of a KONVOI type NPP which connects the reactor outlet with the inlet of the steam generator. The coolant flow is directed to the reactor vessel against the steam flow from the core by a cylindrical half-shell (Hutze) welded to the bottom of the main circulation tube.

The report presents the results obtained by needle shaped conductivity probes developed in Rossendorf. The probes allow the detection of the state of the fluid (liquid or vapour) at the tip of the probe with a relatively high time resolution in the range of milliseconds.

2. The Measuring System

The sensitive element of the probe is a small ceramic tube with an electrically conducting tip which is in contact with the fluid. The supply with a small voltage (2 V (AC), 4 kHz) causes an electric current from the tip via the liquid toward the wall of the tube or vessel which is being interrupted when vapour (bubbles, plugs) covers the probe. In this way, water and steam can be distinguished from each other.

In the result of pre-tests the probes were improved to increase their mechanical stability. First of all the diameter of the ceramic was increased from 1 to 1.6 mm. In this way a satisfactory stability under the hard conditions within the COCO facility was reached.

The applied 8 probes were located in the injection region around the end of the injection channel (Hutze). The locations are shown at the following figures. The probes 1 and 2 were located at the same axial position but mounted in slightly different depths (10 and 30 mm). The probes 5 and 6 were mounted in the same way at the position accurately above the end

of the injection channel. The probes 3 and 4 were inserted from below and located 300 mm downstream from the injection orifice. The distance between probe 3 and 4 was 30 mm in order to measure the velocity by means of cross correlation.

The data acquisition system is based on electronic modules for digital data preprocessing. Each module is equipped with a micro processor and can treat the signals of two probes. The signals are digitalized with the help of two ADC. Digital controlled amplifiers allow the optimal use of the range of the ADC. The preprocessed data are transmitted to a central data acquisition computer (PC) through a serial interface (RS232c). The data rate is 57.6 kBaud. With the help of optically coupled insulators the modules and the central PC are completely potential disconnected from each other. In this way, the disturbance level is kept low. The modules have to be placed near the probes to keep the connection short, therefore they dispose of robust water sprite safe aluminium casings. The measuring system can be extended up to 16 probes by connecting 8 modules with one PC.

The measurement has to be started by pressing a button on the PC keyboard. Therefore an additional binary output has been organised, which provides the standard PCM tapes of the COCO facility with a synchronization signal.

3. First Results

To show the changes in the flow pattern caused by various steam velocities a series of test points with constant ECC injection flow rate of 14.958 kg/sec but varying steam flow rates from 1.121 kg/sec to 4.203 kg/sec was chosen (test points from E33.4271 to E33.4278 at a pressure of 2.5 MPa). During this series all important probes were available.

At the small steam flow rate the water flows out of the "Hutze" toward the reactor, forming a counter current flow with the steam coming out of the reactor and flowing toward the steam generator.

At the high steam flow rate, a perfect flow reverse can be observed. The water is directed toward the steam generator by the strong steam flow, so that the probes 1 and 4 (2 and 3 respectively) remain dry. There is a surplus of steam, so it can not be condensed completely. The flow pattern is now a droplet flow toward the steam generator, which is indicated by the probes 6, 7 and 8 (Fig. 1).

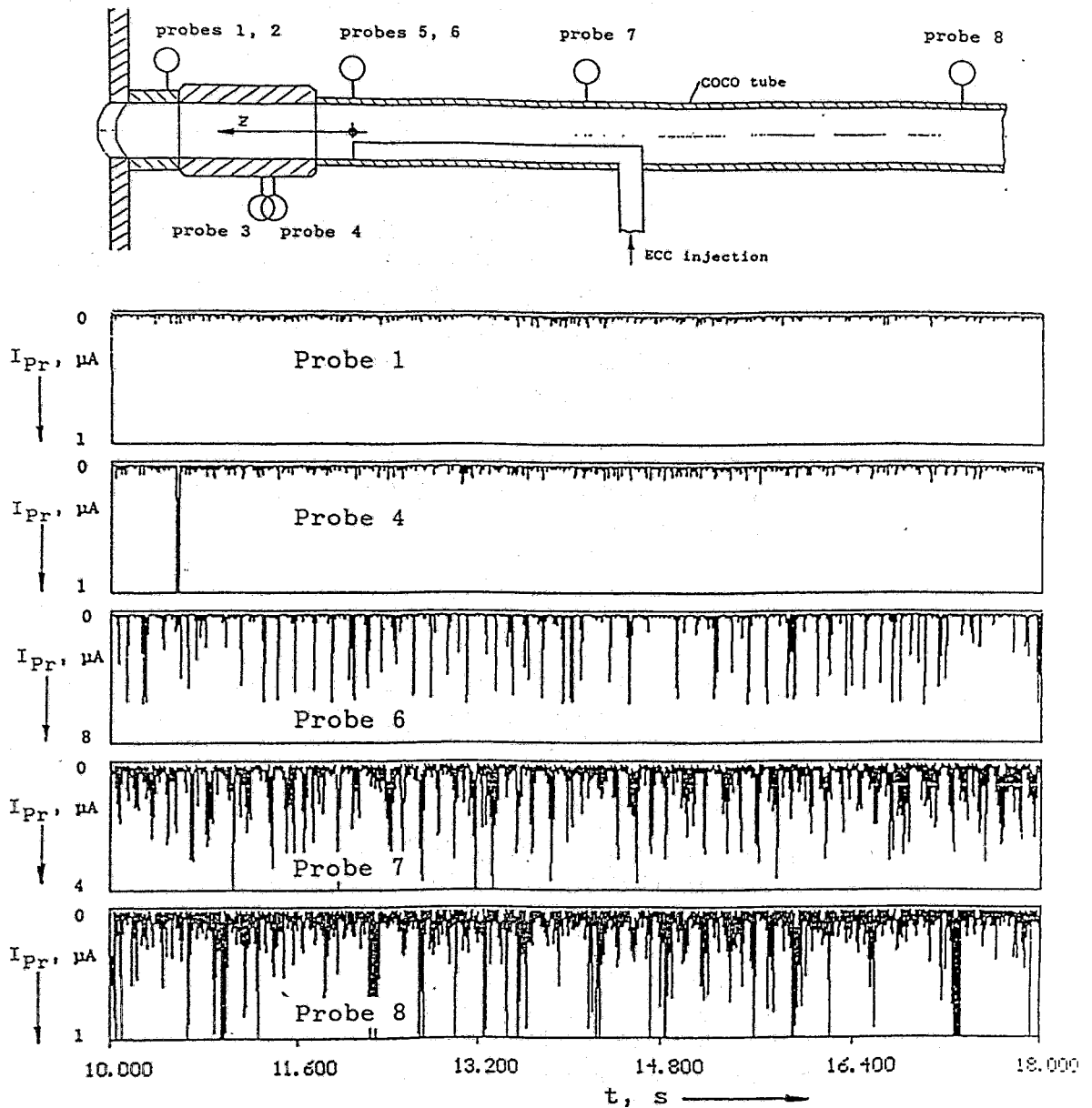


Fig. 1 Perfect flow reverse toward the steam generator
 Test point: E33.4271
 Steam flow rate = 4.203 kg/sec
 ECC injection flow rate = 14.985 kg/sec

Between the two extreme cases there is a transition state at average steam mass flows characterised by particular flow reverse with an intermitting of counter current flow and flow reverse (Fig. 2, test point E33.4274). The probe 4 shows alternating phases of steam and two phase flow. One part of the injected water is still flowing into the reactor, but another part is directed to the steam generator. At these values of flow ratio of steam to water injection,

the steam can be mainly condensed and large plugs are moving toward the steam generator, indicated by the probes 7 and 8. Taking into account that these probes are mounted at a depth of only 10 mm from above, it is obvious that the plugs fill the tube completely.

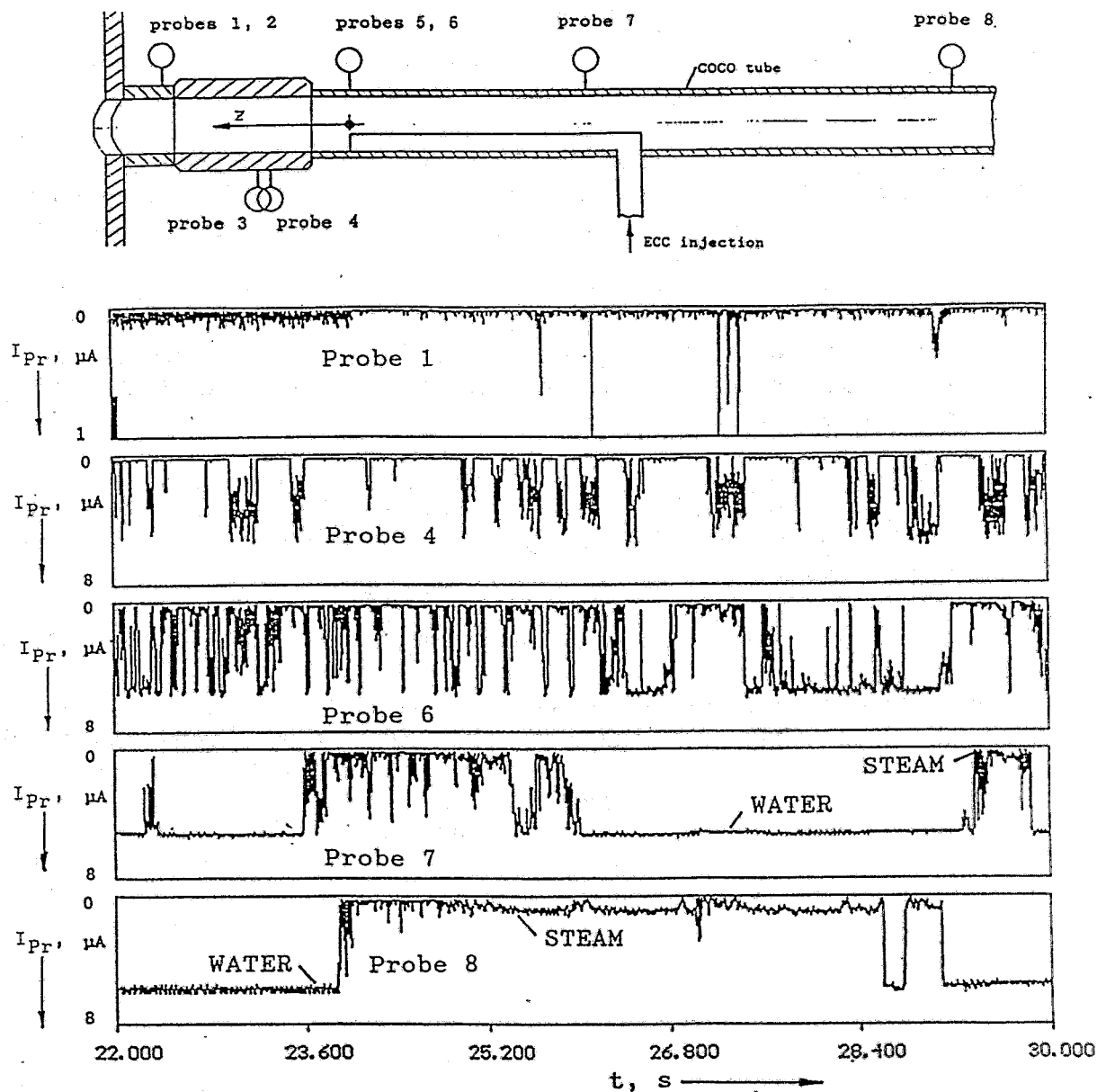


Fig. 2 Particular flow reverse with plugs and void collapses
 Test point: E33.4274
 Steam flow rate = 2.587 kg/sec
 ECC injection flow rate = 14.985 kg/sec

Calculation of Neutron Fluence in the Region of the Pressure Vessel for the History of Different Reactors Using the Monte-Carlo-Method

H.-U. Barz (FZR) and W. Bertram (IFE)

Embrittlement of pressure vessel material caused by neutron irradiation is a very important problem for VVER-440 reactors. For the estimation of the fracture risk highly reliable neutron fluence values are necessary. For this reason a special theoretical determination of space and energy dependent neutron fluences has been performed mainly on the basis of Monte-Carlo calculations. The described method allows the accurate calculation of neutron fluences near the reactor vessel in the height of the core region for all reactor histories and loading cycles in an effective manner.

For the needed 3D neutron field calculation it is useful to split the problem into two parts:

1. Calculation of the fission source distribution using diffusion approximation. These values can be supposed further as known, because they are prepared for the reloading cycles in each case.
2. Calculation of the fluences at the pressure vessel for known sources of fission neutrons. For these calculations transport methods are needed. For several reasons, for instance the description of geometrical details, the Monte-Carlo-method can be used successfully.

The calculation of the source distribution by standard methods results in source values for a certain grid of the core.

The idea was to calculate partial solutions G^i for each of those source elements with unit fission sources and to compute the neutron fluence at the pressure vessel by superposition of of these partial solutions taking into account the time-dependent real number of fission neutrons in the elements. Different sources are generally connected with different burn up and therefore different shielding properties of the fuel assemblies, but it can be shown that this shielding properties are approximately independent of the burn up in the relevant energy range above 0.1 MeV. The great advantage of this method is the possibility that the partial solution can be used for different reactors and different reloading cycles and have to be calculated only one time.

The calculations are performed for a 30° symmetry sector of the core (see Fig. 1). According to the burn up calculation codes we use 10 layers of 25cm for each fuel element as grid. For the outer assemblies a detailed source distribution (127 fuel rods) was taken into account. Within the given time intervalls of 40 days a linear slope was assumed for the total number of fission neutrons in each source element. Special variance reducing methods within the used Monte-Carlo-code TRAMO allow results with small errors also for great optical distances between the source and detector region. Further the accuracy is better than for a pure group calculation, because the elastic slowing down is calculated exactly by using the scattering law together with a special library for taking into account the anisotropy of elastic scattering in the cms sytem.

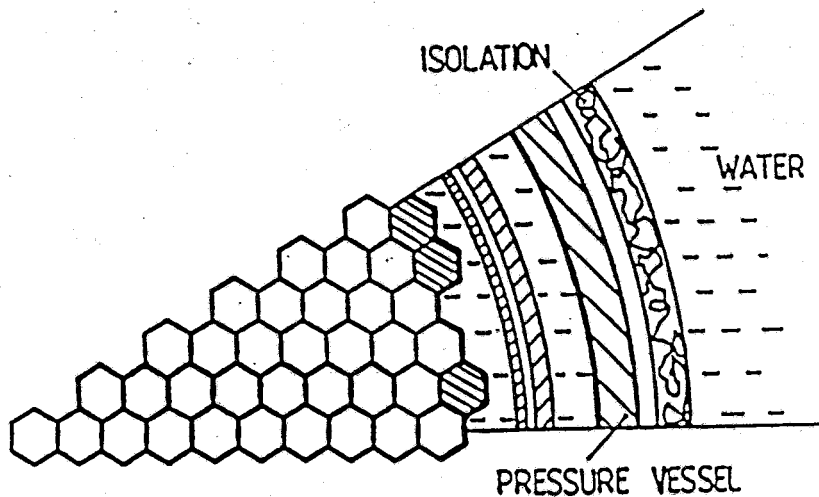


Fig. 1
 Top view for the VVER-440 symmetry sector with marked coreplaces for shielding assemblies

With the help of all these calculated functions G^i the time-dependent fluences were determined for all 4 reactors and all burn up cycles with and without shielding assemblies (see Fig. 1 the hatched positions) in the power plant Greifswald. As the partial solutions have to be calculated only for one reactor (only for the case with shielding assemblies we need a new set of partial solutions because of the change of shielding properties in this case) the used method is very effective. With the help of these functions together with a simple evaluation PC-code the results can be used also for planning the loading schemes according to safety requirements against brittle fracture.

In Fig. 2 one example of our investigations is given.

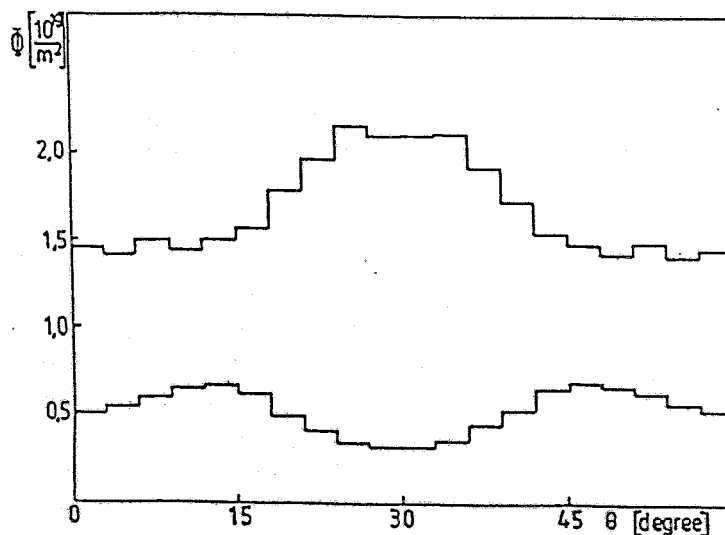


Fig. 2
 Comparison of loading cycles using dummy and normal assemblies (averaged fluxes per day)

There we can see the neutron fluxes for energies greater than 0.5 MeV in the central region of the core height on the inner surface of the reactor vessel for different angle intervalls in the symmetry sector. Compared is the case of normal loading of the reactor core with the case of inserting 3 shielding assemblies on the outer border of the core instead of fuel assemblies (see Fig. 1). As we can see by the use of shielding assemblies the flux at the position of the old maximum can be considerably decreased, but the effect at the position of the new maximum is much lower.

In Fig 3 the time dependent fluences according to the special history of a reactor are given at two critical points.

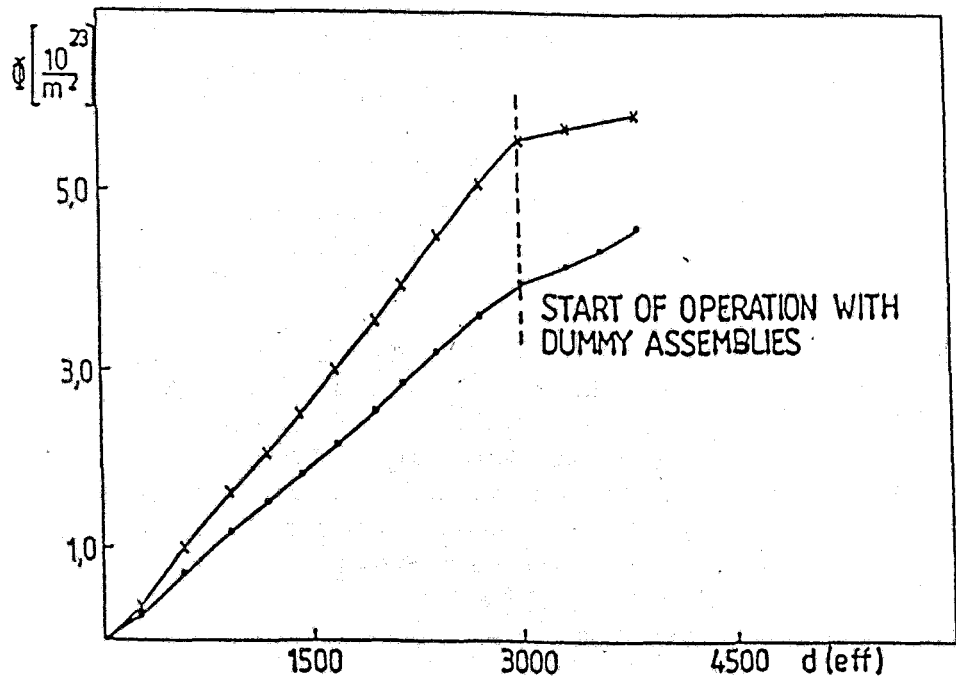


Fig. 3
Time dependence of fluence ($E > 0.5$ MeV) near the inner weld surface for two different angles intervals

We see that for the not so remote future the space point with the old maximal flux determines the possible operation time, but depending on the allowed fluences the new maximum can become more and more important. In this way you can investigate the different reactors to optimize the further reactor history taking into account not only the burn up but also the fluence of the pressure vessel without doing new transport calculations.

Parallel to the theoretical investigations measurements were performed on the outer pressure vessel surface. The comparisons were performed for three different core configurations (normal core, core with highly burned up fuel assemblies at the core border, core with shielding assemblies at the core border). In all cases the comparison shows the good agreement of the calculated values with the experimental results at least for the central region of the core.

Comparative Studies of High-Temperature Corrosion of ZrNb1 and Zircaloy-4

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1. Introduction

Zircaloy and ZrNb1 represent two different types of zirconium alloys which, for historical reasons, have been introduced as cladding materials of the fuel elements of water-cooled reactors and have proved equally reliable under reactor operating conditions.

In order to achieve a comprehensive assessment of the suitability of these alloys, it is also of importance to undertake a comparative evaluation of their high-temperature behaviour under boundary conditions being typical for loss-of-coolant accidents. The aim of the study is to discuss such a comparison for one aspect of accident behaviour, namely high-temperature corrosion in a steam atmosphere including its effect on ductility.

2. Experimental

The oxidation tests were carried out on cladding tubes made of ZrNb1 (former USSR) and Zircaloy-4 (Zry-4 of Sandvik).

Tubes of both alloys were used in the as-supplied state tube specimens of lengths of 20 mm for mass gain, 8 mm for compression tests and 5 mm for metallography investigations in length were cut off, degreased and pickled in aqueous HF/HNO₃ solution.

The specimens were in a tube furnace isothermally exposed to a flowing steam at a water supply rate of 45 ml/h in the temperature-time range from 900 - 1100 °C and 10 - 30 min. In addition, ZrNb1 specimens were oxidized at lower temperatures (700 - 850 °C) and for longer times (75 - 250 min) under otherwise identical experimental conditions.

After exposure to steam the specimens were visually examined and their mass gain determined.

A microstructural evaluation was carried out on the basis of metallographic sections either in a polished or wipe-polished state (etchant: HF-HNO₃ solution in H₂O₂). The thickness of the oxide layer and that of the O-stabilized α -layer were determined with the Epiquant (Carl Zeiss Jena). Radial microhardness curves were furthermore measured for each test point. The hydrogen content was determined by vacuum hot extraction and in some cases also microprobe and SEM fracture surface studies were carried out.

For determining cladding tube embrittlement after oxidation specimens of 8 mm in width were subjected to radial compression tests executed by means of a tension-compression tester (Zwick) using a crosshead rate of 1 mm/min.

The mass gain Δm was evaluated by assuming a parabolic growth law

$$\Delta m = K \sqrt{t} \quad (1)$$

An Arrhenius relation

$$K = A \exp(-Q/RT) \quad (2)$$

is assumed to be valid for the temperature dependence of the rate constant K : The constants A and Q were determined from the experimental values by linear regression. The oxide layer and α -layer thicknesses were evaluated analogously. Further details are given in [1].

3. Results

The relation

$$\Delta m/g \text{ cm}^{-2} = 0.4873 \sqrt{t/s} \exp(-10261 / T/K) \quad (3)$$

was determined for ZrNb1 as the kinetic equation of mass gain by oxidation in water steam. The temperature dependence of the rate constant K resulting from this relation is shown in Fig. 1. The plotted points are mean values of K determined with several specimens for at least two different exposure times. The curve after an equation by Leistikow et al. [2] published for Zry-4 is also plotted. Both alloys show comparable oxidation kinetics indicated by the mass gain although ZrNb1 has a slightly lower oxidation rate.

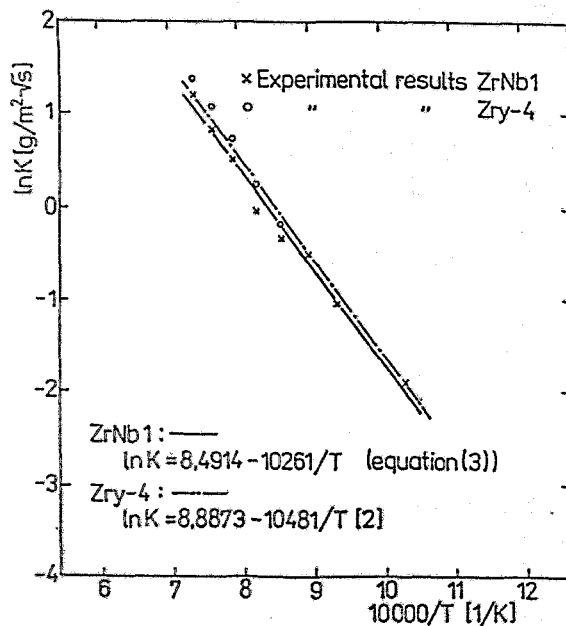


Fig. 1 Temperature dependence of the rate constants K for mass gain, comparison of experimental results for ZrNb1 and Zry-4

The basic microstructural constitution of ZrNb1 is also in agreement with one of Zry-4. After exposure in steam at sufficiently high temperatures it displays the well-known three-layer structure consisting of

- oxide scale,
- O-stabilized α -phase sub-surface layer and
- prior β -phase matrix-microstructure (α').

However, over these basic similarities notable differences can also be established:

- phenomenology of the oxide scales (types and homogeneity of oxide formation),
- microstructure of the O-stabilized α -layer,
- ratio between oxide scale and α -layer thickness,
- hydrogen uptake,

- microhardness profile,
- cladding tube embrittlement, measured after oxidation at RT.

From a safety-relevant aspect, above all the more intense hydrogen uptake, the more rapid cladding tube embrittlement and - with respect to waste management and final disposal - the poorer adherence of the oxide layer on the ZrNb1 cladding are of significance. Examples of the radial profiles of the microhardness are given in Fig. 2 and 3. The microhardness correlates with the oxygen concentration. The comparison shows that the oxygen uptake of ZrNb1 is higher but more uniformly distributed, whereas the radial profile of the microhardness is very irregular by Zircaloy.

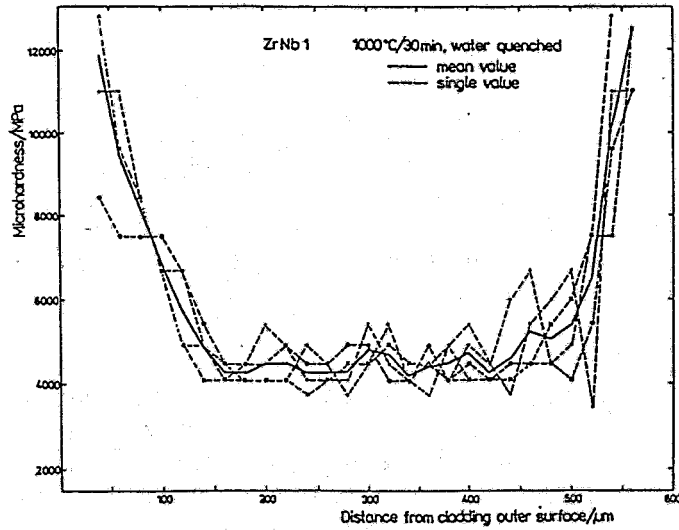


Fig. 2 Radial profile of microhardness, ZrNb1 after steam oxidation at 1000 °C, 30 min, quenched in water

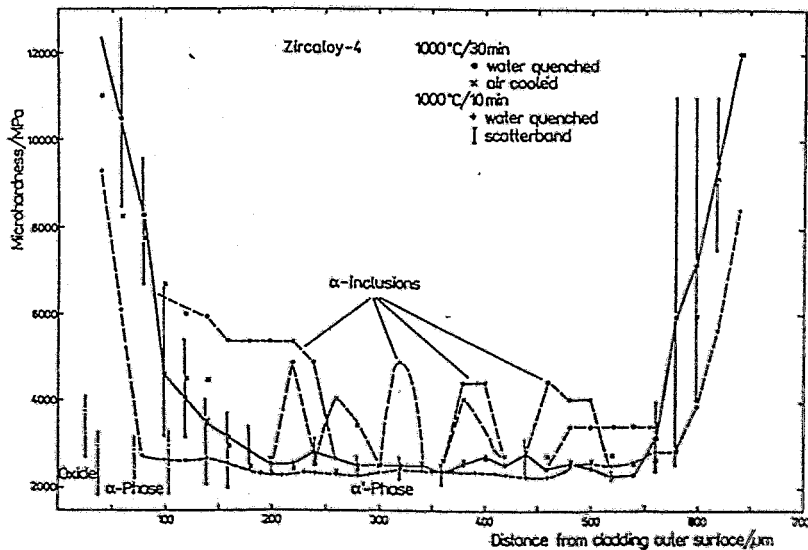


Fig. 3 Radial profile of microhardness, Zry-4 after steam oxidation at 1000 °C

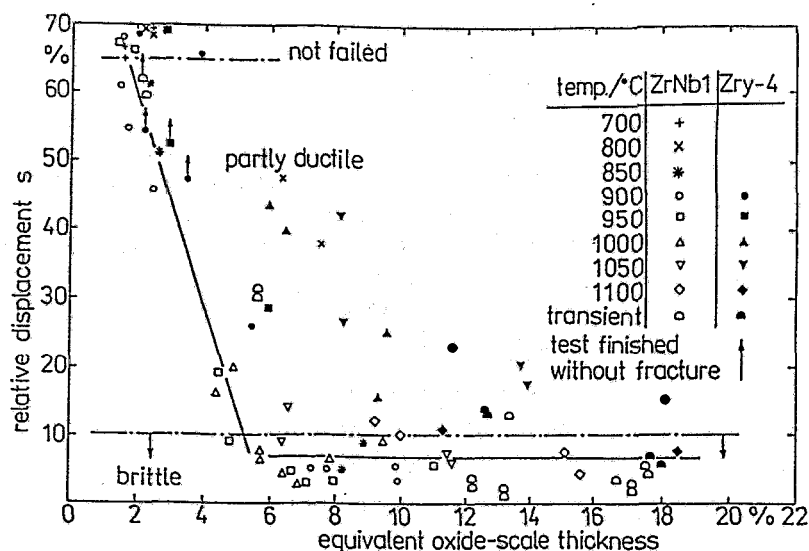


Fig. 4 Dependence of the relative deformation path on the relative equivalent oxide scale thickness from ring compression tests for a ZrNb1 and Zry-4 tube after steam oxidation

Consequently, ZrNb1 is considerably more susceptible to embrittlement than zircaloy, as demonstrated by Fig. 4. The relative deformation path is plotted as a function of the relative equivalent oxide layer thickness calculated from the mass gain of the respective specimen. Whereas in the case of Zry-4 the ductility drops gradually and total embrittlement is only reached at a relative equivalent oxide layer thickness of about 18 %, the ductility is drastically reduced for ZrNb1 with increasing oxide scale thickness. The given embrittlement limit is already reached at a relative equivalent oxide layer thickness of about 5 %.

The difference in ductility behaviour is paralleled to the scanning electron-microscopic fracture image. In the matrix region Zry-4 displays mixed fractures at higher oxide layer thicknesses. Areas of characteristic ductile fracture occur in addition to regions of plate-like transcrystalline brittle fracture. The fraction of areas with ductile fracture decreases with growing oxide scale thickness. Mixed fracture modes are not observed for ZrNb1, at most a tendency towards shallow dimple formation can be seen in the matrix region.

Validity of the 17 % criterion for ZrNb1 can no longer be taken for granted in view of these experimental findings. The differences between both alloys can be explained by means of differences in the thermodynamic conditions for equilibrium of the ternary systems Zr-O-Nb and Zr-O-Sn.

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Single Rod Burst Tests with ZrNb1 Cladding - Comparing Valuation with Zircaloy Cladding

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Introduction

An assessment of the safety of Russian VVER reactors requires, among other things, a detailed investigation of the material behaviour of ZrNb1 fuel element cladding under the conditions of loss-of-coolant accident. For this reason 115 single rod experiments were carried out under isobaric and temperature-transient conditions and within a range of parameters which are equivalent to the investigations of Zircaloy cladding. The tests were performed in the REBEKA single rod testing equipment of Nuclear Research Centre Karlsruhe /1/. Thus, the experiments were able to be carried out under the same testing condition as those applied to Zircaloy. This practice permits a well-founded comparison between the burst behaviour of Zircaloy and ZrNb1 cladding.

The testing conditions and the results are shown in detail in /2,3/.

Additionally, metallographic investigations were performed after testing. They are to help the interpretation of the results of burst tests. The following work explains some effects from the metallographic point of view.

Experimental

The microstructural study was carried out on the basis of metallographic sections either in a polished or wipe-polished state. In each case one section per point of the test matrix /3/ was made in and 20 mm above the axial middle plan of the burst position. The oxide layer formation and the morphology of the microstructure were estimated and the oxide scale thicknesses were measured. Besides radial and azimuthal microhardness curves were determined and scanning electron-microscopic investigations of fracture areas were executed.

In the comparison 14 Zircaloy specimen of REBEKA programmes of KfK were included and analysed in same manner.

Results

Fig. 1 and 2 show the burst temperature versus tangential stress for heating rate 1 K/s and 10 K/s. Additionally the curves after the deformation model REMOD for Zircaloy are plotted. It is evident that the burst behaviour of both alloys is similar for low or high burst stresses respective high or low burst temperatures. Clear differences are indicated for intermediate stresses and burst temperatures. ZrNb1 cladding tubes fail at lower burst temperatures within this range.

Furthermore the burst strains are clearly different as shown in Fig. 3. The burst strains of ZrNb1 are considerably lower and the minimum is shifted to lower temperatures.

Oxide layer structure and microstructure show qualitatively the same appearance as known from isothermal investigation without pressure /4/. The most important parameter is the burst temperature. For ZrNb1 the hexagonal α -phase occurs up to approximately 730 °C, dependent on the heating rate. Above these temperature changes of the microstructure are observed which indicate the α - β -transition. Pure α' - (prior β -) microstructure is present at temperatures above 870 °C. The corresponding temperatures run to 820 and 955 °C for Zircaloy.

The transition temperatures are clearly marked in the dependence of burst stress and strain on temperature. The temperature curve of the microhardness shows also the same form (Fig. 4). The influence of the heating rate results from the increase of the transition temperature with the heating rate. The strain minimum is placed within the (α + β)-range nearby the end of the phase transition. But the metallographic work does not explain the different levels of burst strains. Perhaps the lower burst strain of ZrNb1 is caused by the higher temperature dependence of the burst stress of ZrNb1 in the (α + β)-range. Thus, small azimuthal temperature differences will already produce prompt strain localization and rupture. That makes clear why there is no influence of azimuthal temperature distribution on burst strain, contrarily to Zircaloy.

The investigations show that despite of the fundamental likeness of the burst behaviour differences that cannot be neglected for computer codes modeling fuel element accident performance, exist between these two alloys. They cannot be abolished in a simple manner through a temperature scale modified in accordance with the different transition temperatures. Therefore the description of VVER fuel element accident performance by means of computer codes, which base on Zircaloy material data, is not recommendable.

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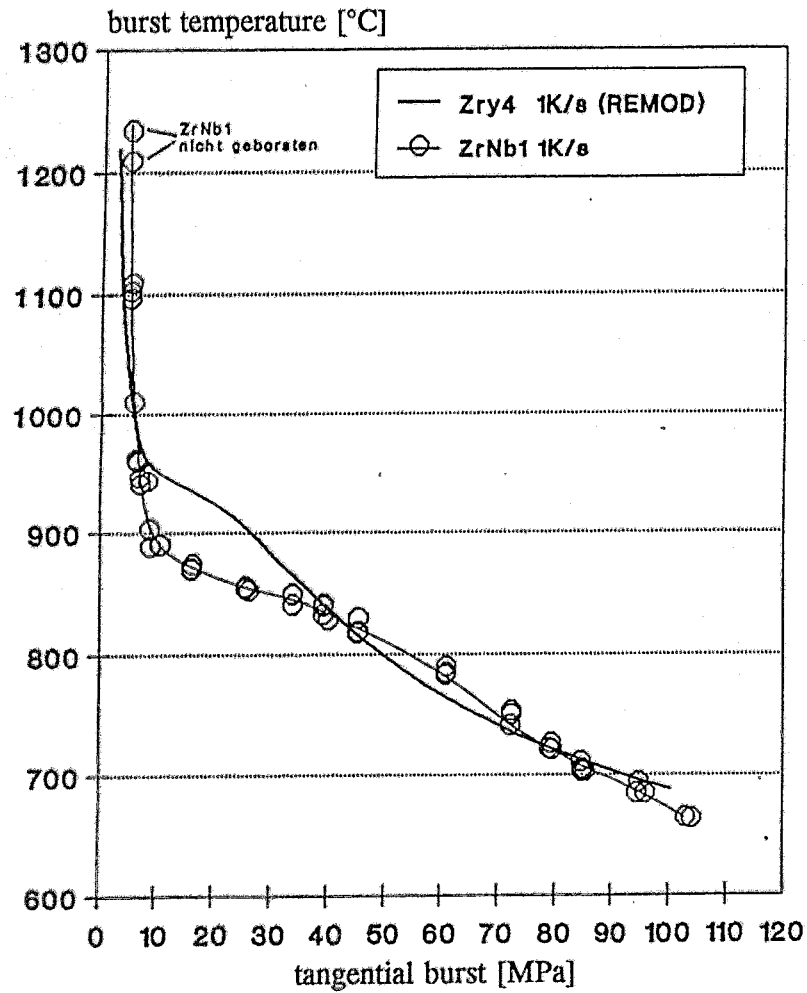


Fig. 1 Burst temperature versus tangential burst stress for ZrNb1 and Zircaloy 4
Heating rate: 1 K/s

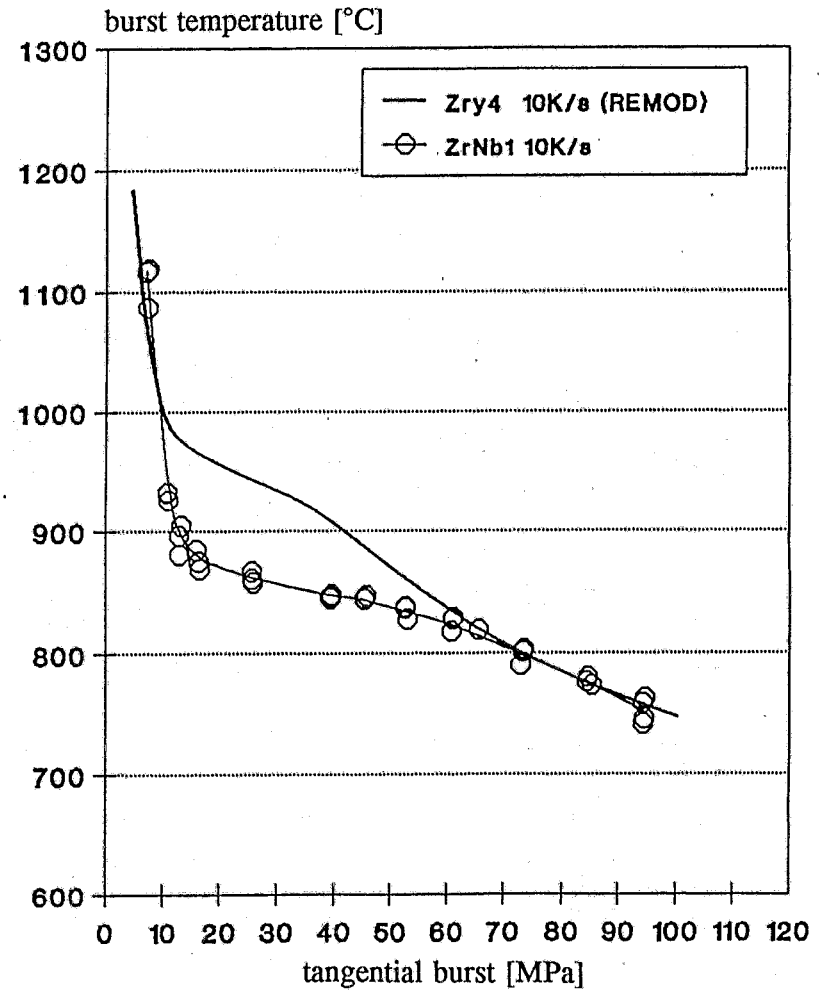


Fig. 2 Burst temperature versus tangential burst stress for ZrNb1 and Zircaloy-4
Heating rate: 10 K/s

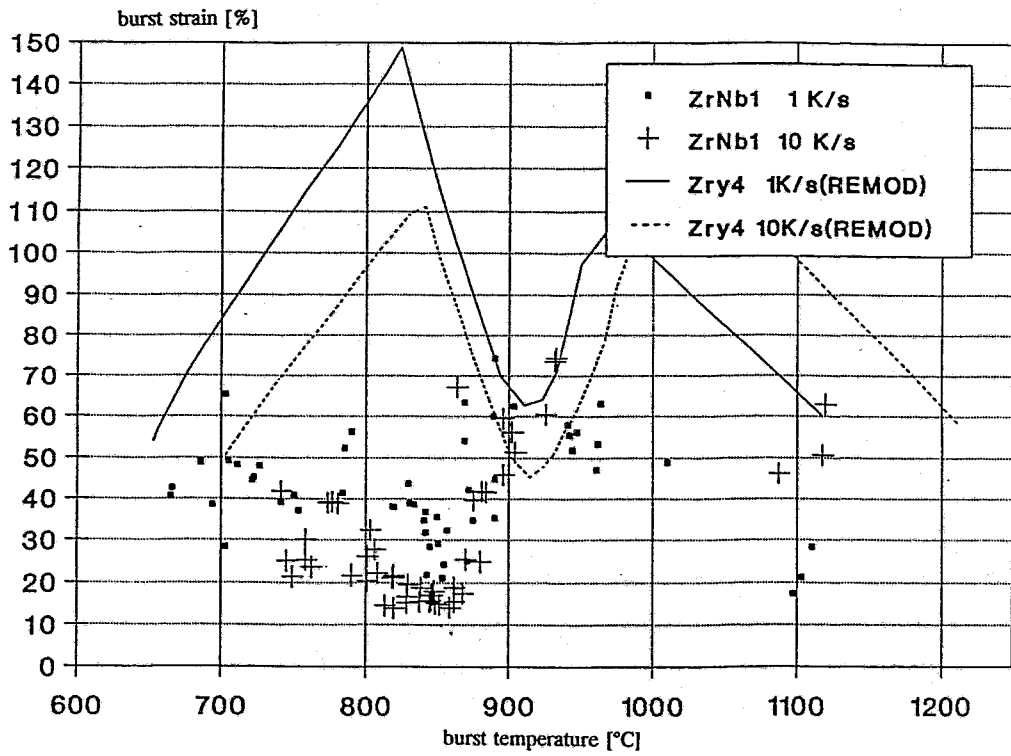


Fig. 3 Burst strain versus burst temperature for ZrNb1 and Zircaloy-4

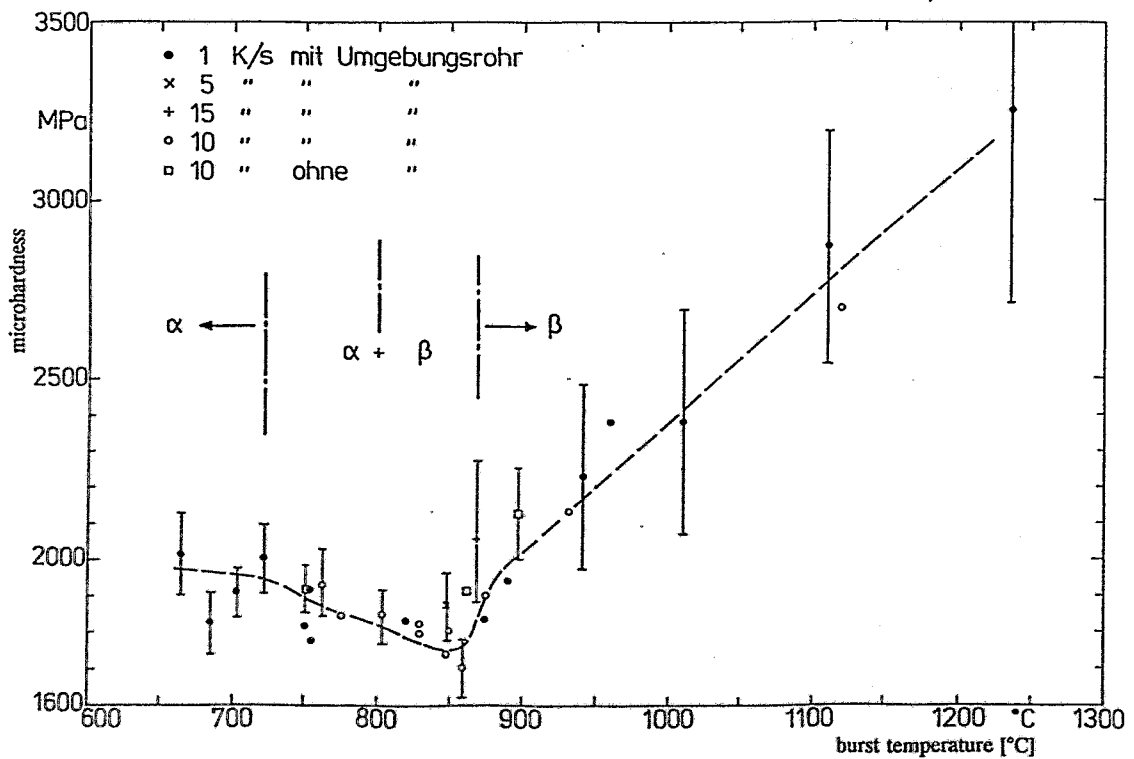


Fig. 4 Microhardness versus burst temperature for ZrNb1 cladding

Measurement of Dynamic Elastic-Plastic Fracture Toughness Parameters Using Various Methods

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The Charpy impact test has been widely used to determine the toughness properties of steels. The quantities measured in the test are the total fracture energy, the lateral expansion and the fracture appearance of the specimen. Additional information about the fracture processes, in general, is supplied by instrumented notched-bar impact test. It is possible to determine the dynamic fracture toughness for linear-elastic and elastic-plastic materials behaviour. A widely accepted method of elastic-plastic fracture mechanics is the J-integral approach. The critical J-integral J_{id} can be used as a toughness value at the initiation of crack growth. The main and most difficult problem in evaluating the critical J-integral J_{id} is to detect the exact crack initiation point on the load-displacement curve. The material used in this study is the heat-resistant steel 10CrMo9.10. From a 16 mm thick plate Charpy V-notch specimens with 20 % side-grooves and a fatigue crack of 3 mm were machined.

Two improved impact testing facilities are used for the dynamic fracture toughness evaluation of precracked Charpy V-notch specimens. The methods of single specimen acoustic emission (FZR) /1/ and crack mouth opening displacement testing (VTT Espoo) /2/ are assumed to indicate the initiation points of stable crack growth. Thus, the crack initiation points t_i and d_i can be defined on the load-time curve (Fig. 1) and the load-deflection curve (Fig. 2), respectively.

It was found that the crack initiation occurs in the most cases prior to the point of maximum load. After determining the crack initiation point on the load-time/load-deflection curve, the stored energy (E) needed for fracture initiation, the area below load-deflection curve from the start of test to the onset of crack propagation (cleavage J_{Fmaxdc} or ductile J_{id}/J_{Fmaxd}), can be calculated. The dynamic ductile fracture initiation toughness value (J_d) can then be derived by using the following formula /3/

$$J_{(t_i, Fmaxi, Fmaxe) d} = 2E / [B(W-a)] \quad (1)$$

Dynamic fracture toughness values of the precracked Charpy V-notch specimens with 20 % side-grooves obtained at both institutes are presented in Fig. 3.

At temperatures below about of -25 °C cleavage fracture initiation (J_{Fmaxdc}) was observed.

In the transition region at temperatures of about -25 to -10 °C some specimens revealed cleavage fracture after some amount of ductile tearing. In this case cleavage fracture occurred at or beyond the point of maximum load.

Above temperatures of approximately -10 °C the fracture mode was fully plastic.

In the face of the difficulties met in determining dynamic ductile initiation J-integrals both $J_{id}(T)$ -curves seem to agree quite well (Fig. 3).

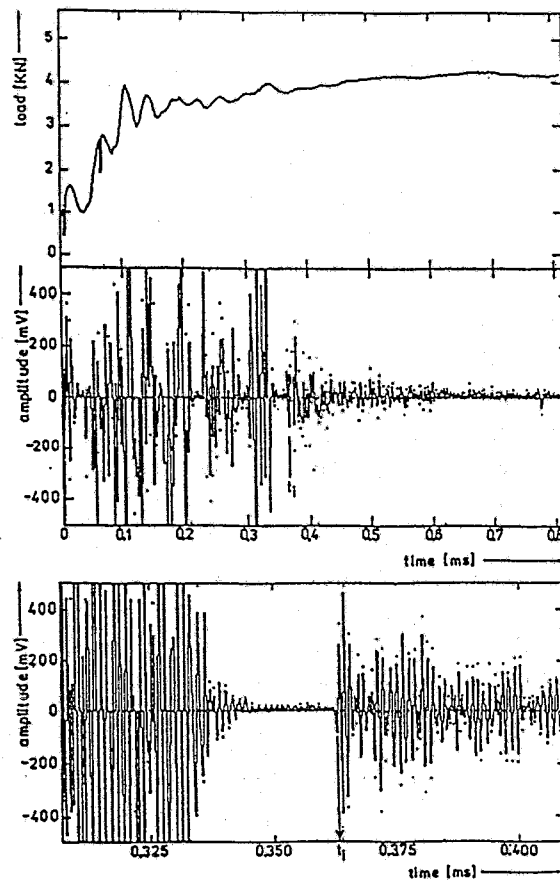


Fig. 1 Load and acoustic emission time curves of a specimen tested at a temperature of 50 °C in the ductile fracture range
 a. whole curves - time resolution 2.0 μ s
 b. the first part of the curves - time resolution 0.2 μ s

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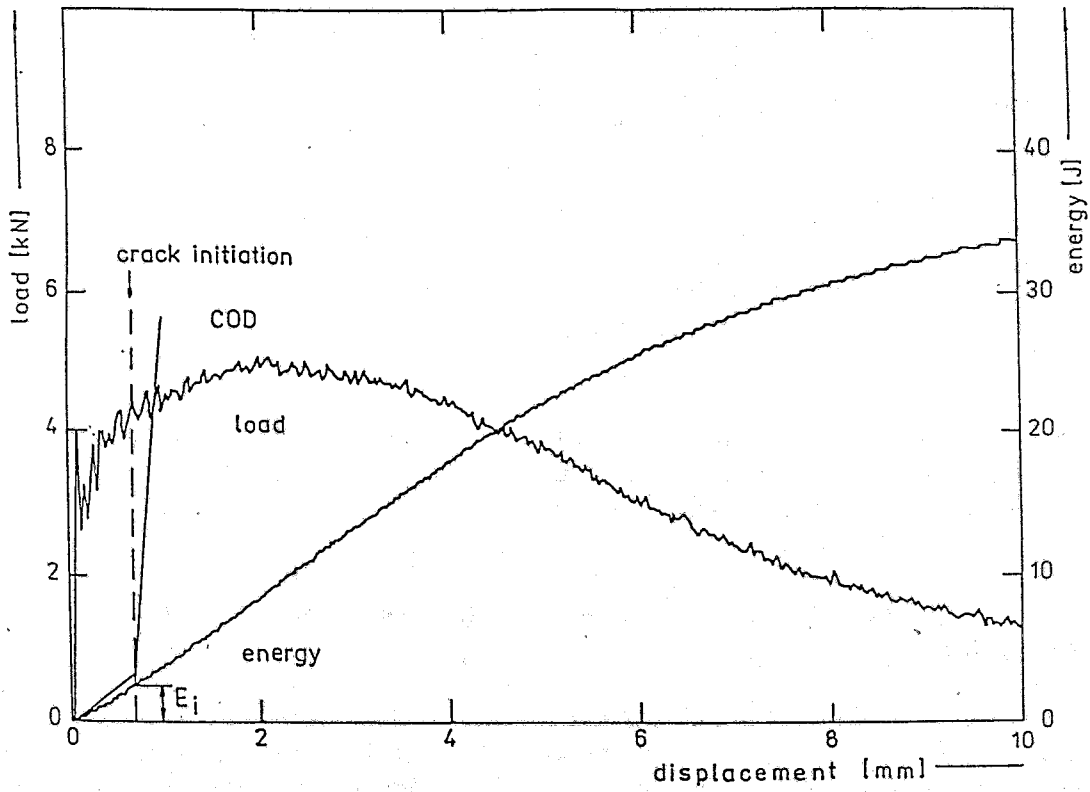


Fig. 2 Load, energy and COD displacement curves ($T = 100\text{ }^{\circ}\text{C}$; VTT)

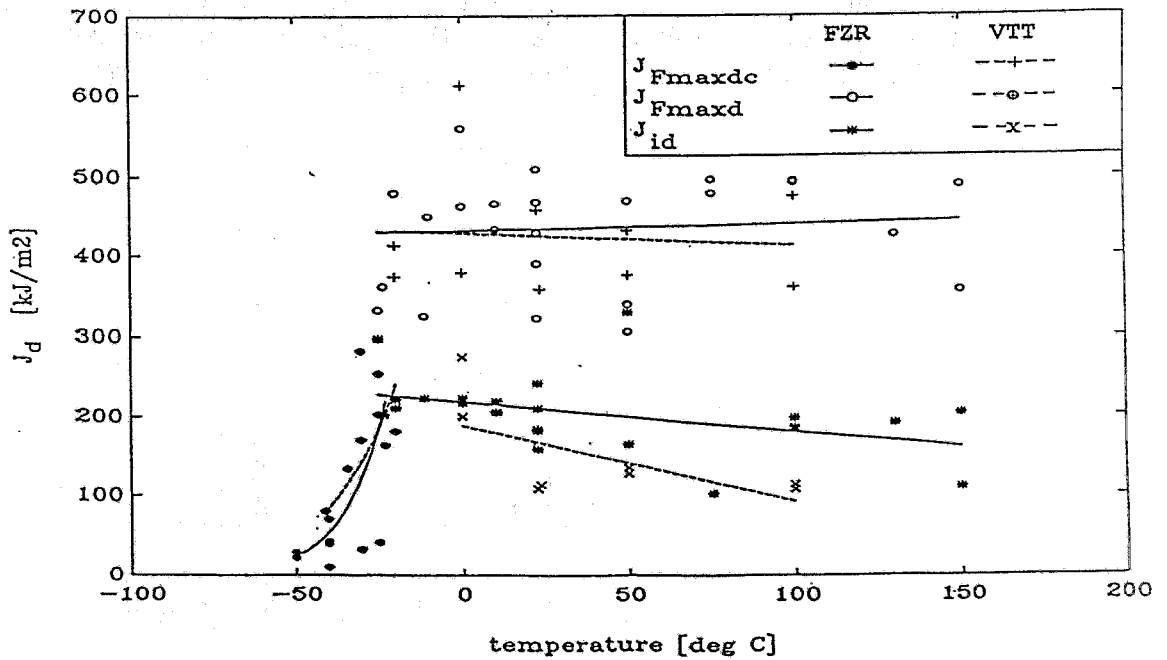


Fig. 3. Dynamic J-values at cleavage fracture (J_{Fmaxdc}), at the onset of ductile crack initiation (J_{id}) and at the maximum of the load-deflection curve (J_{Fmaxd})

Neutron and X-Ray Investigations on the Oxygen Bonding in $\text{YBa}_2\text{Cu}_3\text{O}_{7-x}$ Combined with Physico-Chemical Methods

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The oxygen content is a regulating value on structural and electrical properties of the $\text{YBa}_2\text{Cu}_3\text{O}_{7-x}$ superconductor /1,2/. The occupancy of the different oxygen lattice places at different annealing conditions in $\text{YBa}_2\text{Cu}_3\text{O}_{7-x}$ were determined by Rietveld refinement /3/ with neutron and X-ray data.

The $\text{YBa}_2\text{Cu}_3\text{O}_{7-x}$ pellets were prepared from chemical precipitation powder /4/ by a press and sinter process.

The cell parameters and the fractional atomic coordinates of the cations were refined with X-ray data. The investigation of these parameters was successful on the basis of the well-established model of an orthorhombic perovskite cell /5/. Because of the small atomic form factor of oxygen the refinements of fractional atomic coordinates and occupation numbers of the oxygen atoms gives high standard deviations for these parameters.

Therefore we use neutron diffraction for investigation of the location of oxygen in the lattice cell. With the results of the refinement on the X-ray data the analysis of the neutron data were started. The X-ray results for cell parameters and the fractional atomic coordinates of the cations were fixed. The results of the combined refinements are shown in Table 1.

	sample 1	sample 2	sample 3
temperature of oxygen installation		400 °C	500 °C
(7-x) from SE coulometry	6.304	6.762	6.846
(7-x) from Rietveld ref.	6.36	6.66	7.03
lattice cell	tetragonal	orthorhombic	orthorhombic
n_{O1} in %	100(2)	94(6)	98(3)
n_{O2} in %	100	100	100
n_{O3} in %	100	100	100
n_{O4} in %:	36(2)	94(6)	107(5)

Table 1

Oxygen formula index in $\text{YBa}_2\text{Cu}_3\text{O}_{7-x}$ and occupation of the oxygen lattice places

It could be shown, that the atoms of oxygen exchangeable by thermal treatment are O4, that is the oxygen in the O-Cu-O chains and O1, the oxygen atoms connecting the planes. The oxygen in the planes is fixed and can not be removed up to 935 °C. The strength binding O1 in the lattice is stronger than those of O4 atoms, what is demonstrated by the higher tempera-

ture of dissoziation of O1 atoms. At 935 °C only less than 10 % of the O1 positions are vacant.

In a thermoanalyzer the samples were heated up to 935 °C in oxygen and cooled down with a cooling rate of 6 K/min (pellet) and 25 K/min (powder) with holds at constant temperatures for 100 min (pellet) and 20 min (powder) In Table 2 it is shown, that the oxygen reception is already nearly finished in the cooling periods and the oxygen content does no rise at constant temperatures. That is also the fact if the cooling rate is increased from 6 to 25 K/min. That means, the diffusion from the surface into the solid material and the installation on the lattice positions is very quick. This is the case for the powder and to a some smaller extent also for the pellets up to a density less than 90 % of the theoretical density. At temperatures below 400 °C the installation rate of the oxygen becomes very slow /6/.

Powder:

T [°C]	ΔM [mgO]	$\frac{\Delta M}{\Delta t}$ [$\frac{mmol(O)}{min \cdot mol}$]
935	0.103	0.043
935-800	0.401	0.699
800	0.001	0.001
800-650	0.392	0.658
650	-0.029	-0.012
650-500	0.474	0.790
500	0.003	0.001
500-400	0.145	0.356
400	0.003	0.013
400-350	0.022	0.115
350	0.009	0.004

Pellet:

T [°C]	ΔM [mgO]	$\frac{\Delta M}{\Delta t}$ [$\frac{mmol(O)}{min \cdot mol}$]
935	0.036	0.015
935-800	0.447	0.720
800	0.003	0.001
800-650	0.331	0.515
650	0.047	0.020
650-500	0.297	0.498
500	0.020	0.008
500-400	0.107	0.264
400	0.033	0.014
400-350	0.015	0.080
350	0.009	0.009

Table 2

Temperature treatment and reaction rate of the oxygen of $YBa_2Cu_3O_{7-x}$ pellet and powder

The amount of the received oxygen is not exactly equal to the really installed oxygen at the lattice places. Especially in the temperature region below 400 °C adsorption and chemisorption processes are ascertained.

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Vibration Modelling - Investigation of Mechanical Accident Sequences at a VVER-440 Type Reactor

E. Altstadt and F.-P. Weiß

1. Introduction

In 1985 extremely large neutron noise with amplitudes 7 to 8 times higher than normal had been observed at unit 2 of the Greifswald NPP. It could be shown by means of spectral analysis methods that this anomaly being rather similar to the phenomenon occurred at the Palisades reactor in 1973 - was due to abnormal core barrel motion. The abnormal core barrel (CB) motion originated from a plastic deformation and fatigue of the hold down spring segments and the guide lugs (see Fig. 1). Amplitudes of up to 4 mm could be estimated from the external neutron noise signals.

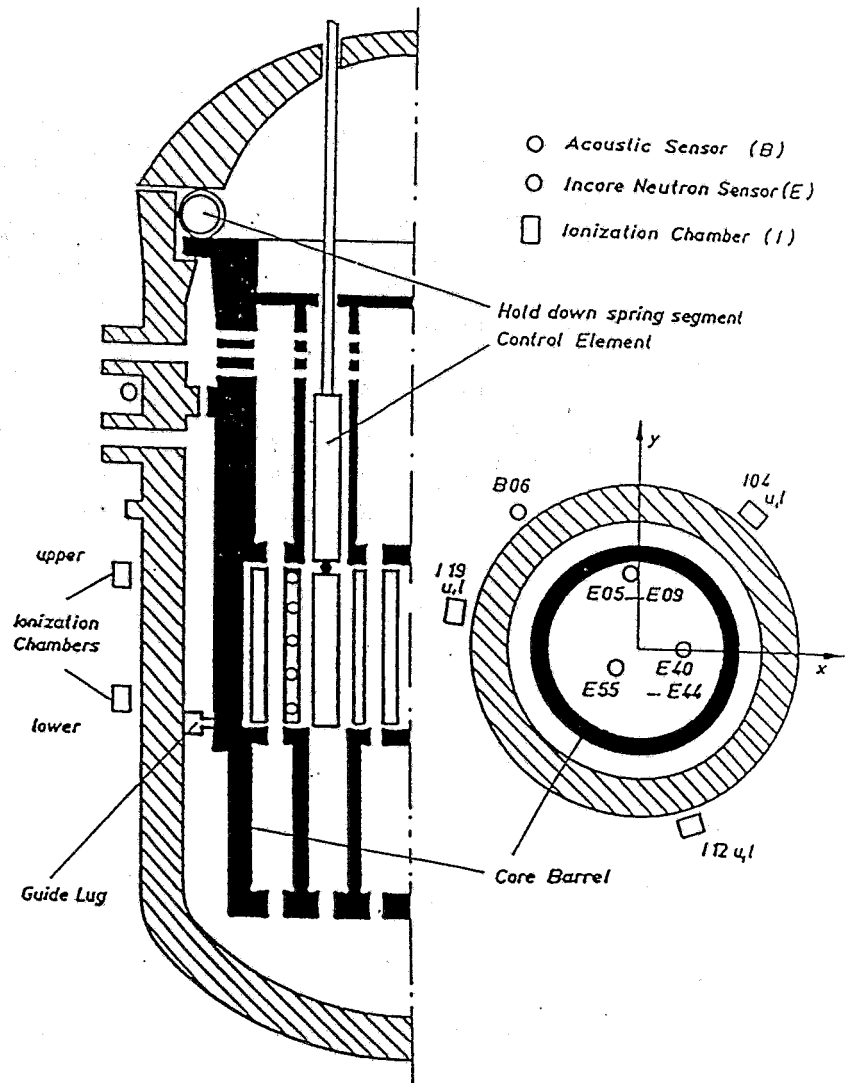


Fig. 1 Structural scheme and detector arrangement of VVER-440

2. Incore - Excore Measurements at VVER-440

Due to the large CB-displacement even the control elements (CE) were excited to vibrations by impacting the neighbouring fuel cassetts mainly in the joint region between fuel- and absorber part. A second path of excitation was realized via the cooling water acting as a spring-damper element in the gap between CE and fuel cassetts. Incore and excore neutron detectors (position see Fig. 1) were used to get information about CB- and CE-displacements. The varying width of the water gap in the downcomer modulates the neutron transmission and thereby generates external neutron flux fluctuations. Since the displacements are small compared with the mean width in the downcomer the flux fluctuations can be assumed proportional to the mechanical relative displacement between pressure vessel and CB. Analogously the incore neutron noise is taken proportional to the relative displacement between CB and CE-fuel part. The coherence and phase functions between incore and excore neutron noise obtained from time signals by using digital signal processing techniques exhibit some remarkable features, for example

- high coherence, sometimes 60-70 %, between incore and excore neutron noise,
- almost linear phase drifts ranging over 180°, 360° or even more,
- strong dependence of phase behaviour on the azimuthal string position and on the vertical detector position within a string.

These peculiarities were to be modelled experimentally and theoretically to draw conclusions about the character of the CE-vibrations and to get a founding for a surveillance procedure.

3. Experimental Set-Up

To get principal understanding about the described phenomena a rather simple set-up was constructed (Fig. 2). It essentially consists of a double pendulum and a pipe shaped channel surrounding the lower pendulum part. The channel is stiffly connected with an electrodynamic shaker acting as CB. The coupling between double pendulum and channel is realized by a spring-damper element in the hinge region and by impacting respectively. The displacements are measured with HALL probes. Signal 1 is the displacement of the channel (corresponding to the CB-motion)

and signals 2 and 3 are relative displacements between channel and pendulum hinge and between channel and pendulum bottom point (corresponding to incore neutron noise).

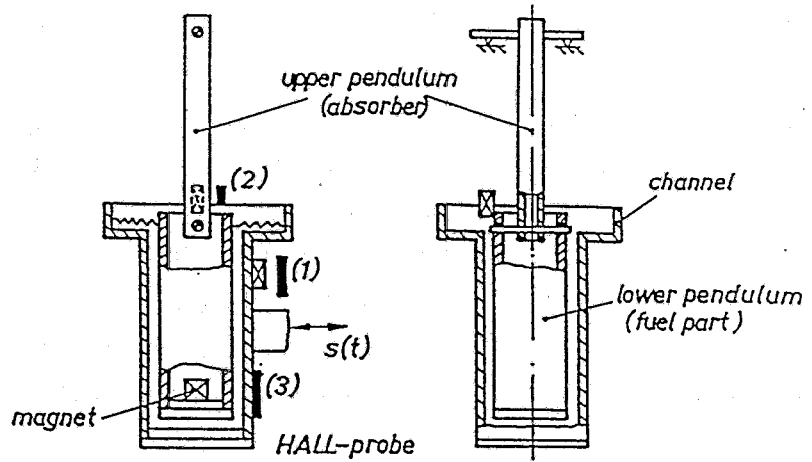


Fig. 2 Experimental set-up

This physical model can be operated in the linear mode (small excitation amplitudes, no impacts) and in the non-linear mode (impacts between channel and lower pendulum).

4. Numerical Simulation

The numerical simulation is based on a mechanical system (see Fig. 3) consisting of linear elements (inertia, dampers, springs) and non-linear elements (gaps, impact limits). This system can be described by a set of linear equations of motion

$$\underline{M}_i \ddot{\underline{q}} + \underline{B}_i \dot{\underline{q}} + \underline{C}_i \underline{q} = \underline{p}_i + \underline{d} \underline{0}_i s + \underline{d} \underline{1}_i \dot{s}$$

with index i representing various system states depending on which of the gaps is closed. Furthermore a set of evolution equations

$$\dot{w}_j + a_j w_j = 0 \quad (j = 1, \text{ngap})$$

is necessary to describe the eigenmotion $w_j(t)$ of the gap ends. The simulation is performed by integrating the equations of motion and the evolution equations with that parameter set i which corresponds to the instantaneous system state. This state is determined by supervising the geometric conditions (gap closed?) and the dynamic conditions (contact force between impact limit and pendulum). Thus the non-linear impact problem is treated by switching between different linear solutions. The adjusted time step width is determined in a preceding modal analysis for each system state. After simulating time series $q(t)$, which is possible for any detector position at the control element or at the physical model respectively, transfer functions, coherences, phases etc. can be computed.

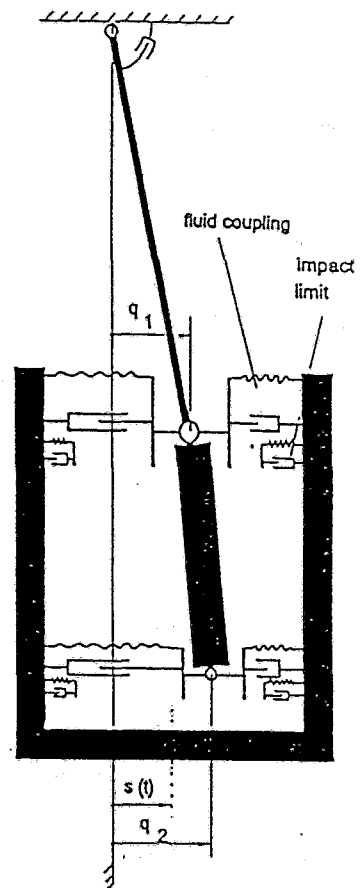
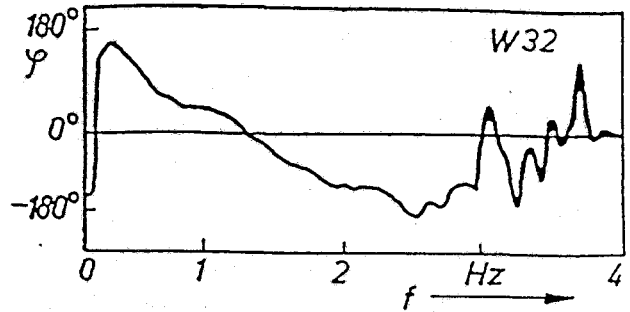
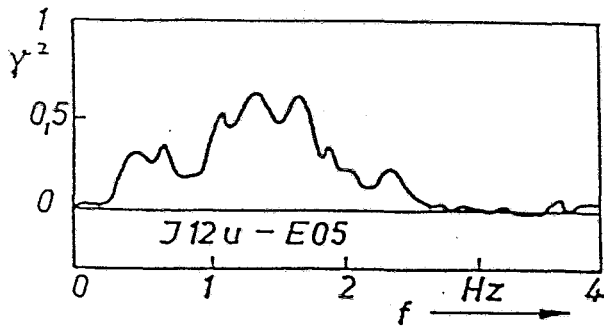


Fig. 3 Nonlinear Analytical Model Scheme

5. Results

There is a good quantitative agreement of the experimental results from the set-up and the numerical simulations as well for the linear case (no impacts) as for the nonlinear case (with impacts). The comparison of these results with the VVER-440 measurements provides at least qualitative agreement if a medium impact rate is adjusted (Fig. 4).

It could be shown that the linear phase curves are due to the impacts. The obtained results can be used to establish a sensitive detection procedure for CB-vibrations induced by CB-motion.



VVER-measurement

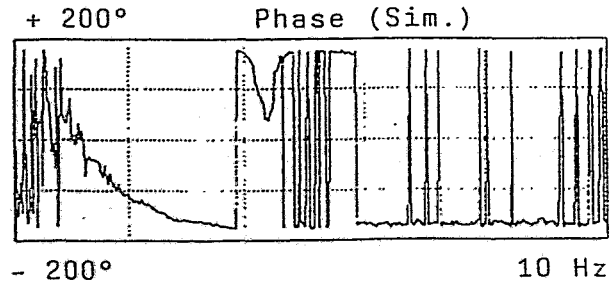
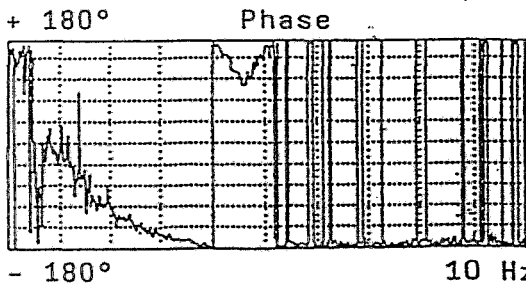
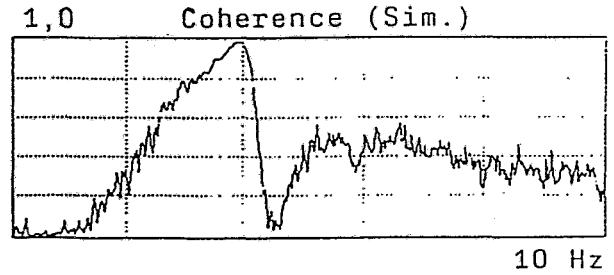
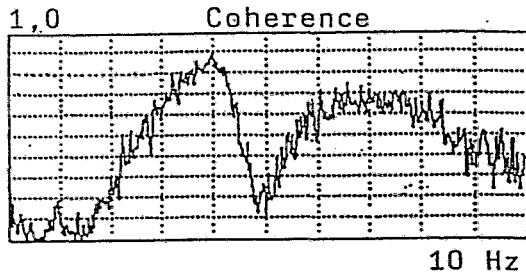


Fig. 4 Examples for coherence and phase relations between CB and CE motion

Analytical and Experimental Investigations for Modelling the Fluid-Structure-Interaction in Annular Gaps

G. Grunwald and E. Altstadt

In modelling the mechanical vibrations of pressurized water reactor internals the fluid-structure-interaction is to be taken into account. Especially at VVER-440 reactors there is a strong influence of the specific geometry. The intention of the presentation is to provide a solution of the continuity and the Navier-Stokes equations for the special case of a narrow annular gap geometry considering the fluid friction.

To obtain an analytical solution for these coupled 3D partial differential equations further assumptions and simplifications must be made:

- the width of the annular gap is small compared with the diameter,
- displacements of the mechanical structure are small compared with the gap width,
- the fluid flow velocity components are independent of the radius.

Keeping these assumptions in mind one can reduce the dimension of the continuity equation from 3D to 2D by averaging over the gap width. The 2D-velocity distribution in the gap then follows from an inhomogeneous potential equation containing the structural motion at the right hand side.

The pressure distribution follows from the adequate 2D-form of the Navier-Stokes equations with a simple approximation for the laminar or turbulent fluid friction.

Two elementary types of motion of the cylinder are considered: parallel displacement and pendular motion. By superimposing these elementary types even more general motions can be described.

A new coefficient α introduced for the velocity potential resulting from pendulum motion (respectively β for axial displacements) gives the possibility for a better consideration of the pressure boundary conditions.

The analytical results are compared with experimental ones from a cylindrical pendulum setup. The criteria of comparison are the eigenfrequency and the eigendamping of the pendulum in the resting and flowing fluid. There is a good agreement between analytical and experimental results (Fig. 1 to 4). Especially the strong influence of the chosen boundary conditions upon the pressure equations can be shown.

In practice the application could be meaningful for core barrel motion at LWRs in general and for flow induced vibrations of control elements at VVER-440 reactors.

The latest theoretical model considers an 8 degree of freedom structure consisting of 2 concentric cylinders coupled via the fluid in the annular gap.

An aim of these investigations is to create a general fluid-structure-element, which can be included into finite element vibration models of VVER-440 components.

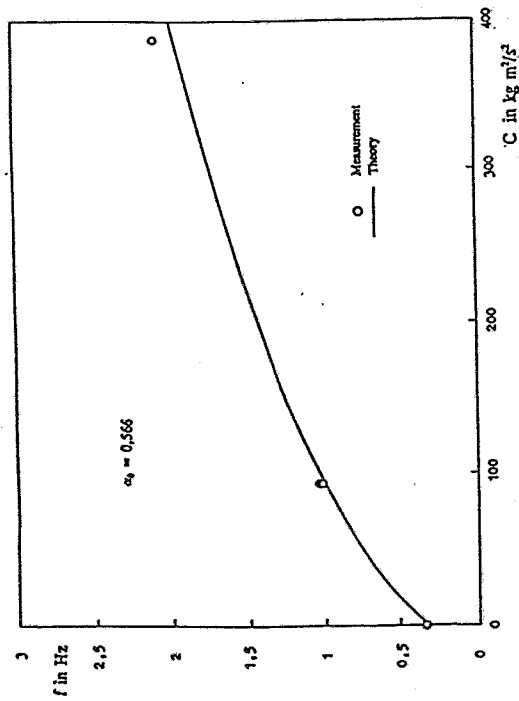


Fig. 1 Pendulum frequency in the resting fluid in dependence on C

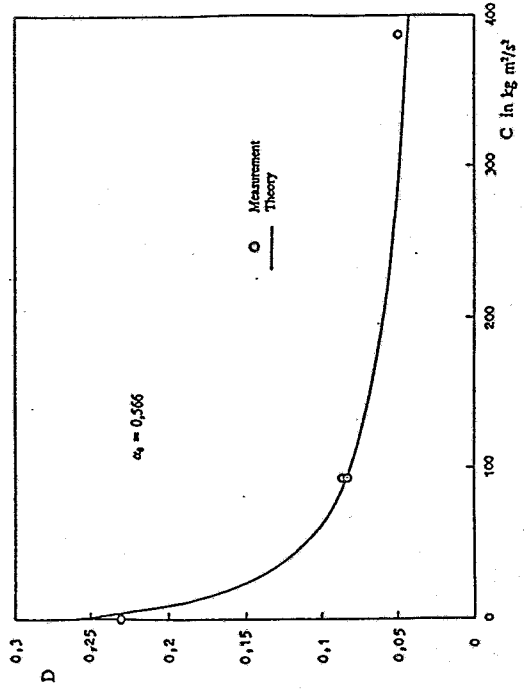


Fig. 2 Dimensionless Damping in the resting fluid in dependence on C

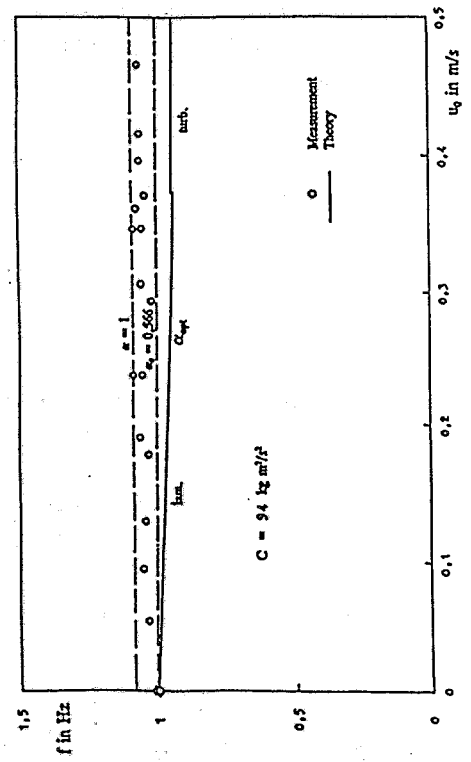


Fig. 3 Pendulum frequency in the flowing fluid in dependence on u_0

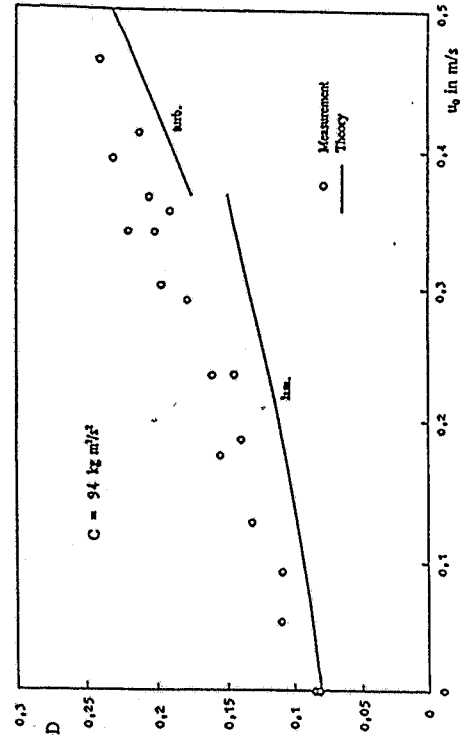


Fig. 4 Dimensionless Damping in the flowing fluid in dependence on u_0

Investigation of Cross-Flow Induced Tube-Bundle Vibrations in Heat Exchangers - Interpretation as Synergetic System

W. Schmitt and F.-P. Weiß

1. Introduction

During the commissioning of new heat exchangers in the primary circuit of the Rossendorf research reactor mechanical impacts that could be detected at the heat exchanger shells were observed. At flow rates higher than 80% of the nominal value the impacts occurred periodically. The intensity of impacts increased drastically with increasing the flow rate, while the frequency of impacts kept constant.

The source of the impacts could not be identified by acoustic burst signals detected at the heat exchanger shells. To reach a guarantee repair by the producer, extensive measurements and signal analysis were necessary.

2. Measuring Method

To measure tube-bundle vibrations directly, acceleration and displacement sensors were introduced into the tubes after removing the lower and upper calotte of the heat exchanger that was operated without the secondary circuit (Fig. 1). The structure-borne sound due to impacts was measured at the heat exchanger shell by a piezoelectric transducer. The sound level was derived from the highfrequent original signal by simple demodulation.

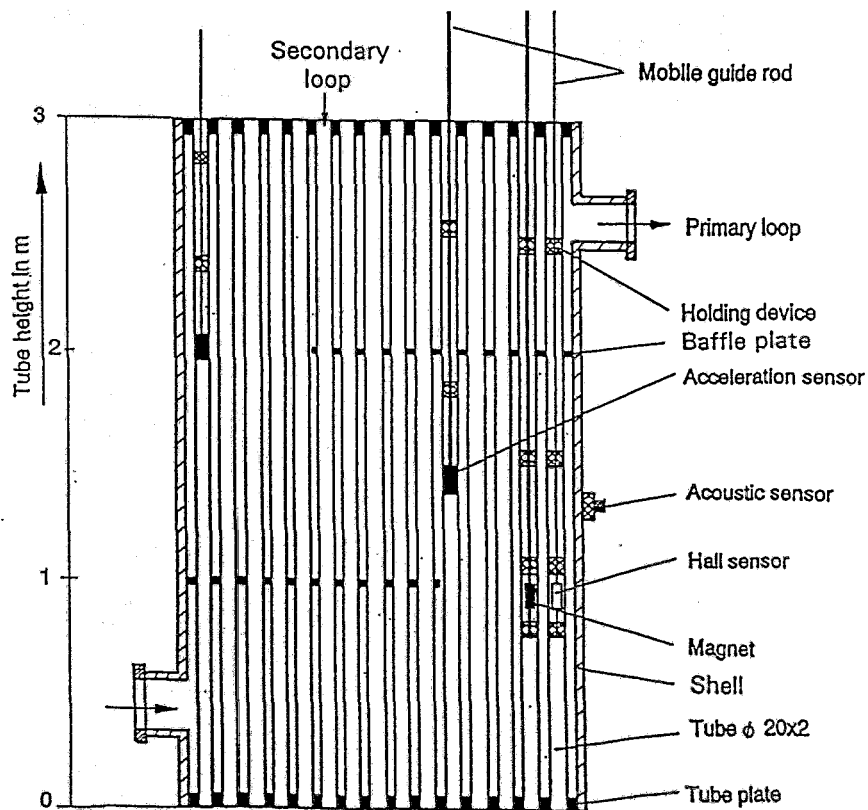


Fig. 1 Measuring equipments for tube vibrations and impact sound at the heat exchangers shell without secondary loop

3. Results

The coherence between the tube vibrations and the sound envelope of the impacts measured at the heat exchanger shell are shown in Fig. 2:

- The eigenfrequency of the tube vibration (14 Hz) is identical with the fundamental frequency of the impacts. The spectrum of the sound envelope (S40) also contains overharmonics, because the mapping of vibration into the envelope is a non-linear operation.
- The coherence γ^2 between tube vibration and sound envelope equals one at that frequency, what means there is the same reason for both effects.
- High coherence exists between vibrations of tubes in the same row and in neighbour rows. There is high coherence even if the tubes are far apart from each other, e.g. tube 1 and 34.
- Constant phase relations ϕ exist between tube vibrations, e.g. 180° between tube 1 and 34 in the 11th row (Fig. 2).

The correlations between impacts and tube vibrations as well as mutual correlations of tube displacements prove the *Collective Tube-Bundle Vibrations* hypothesis.

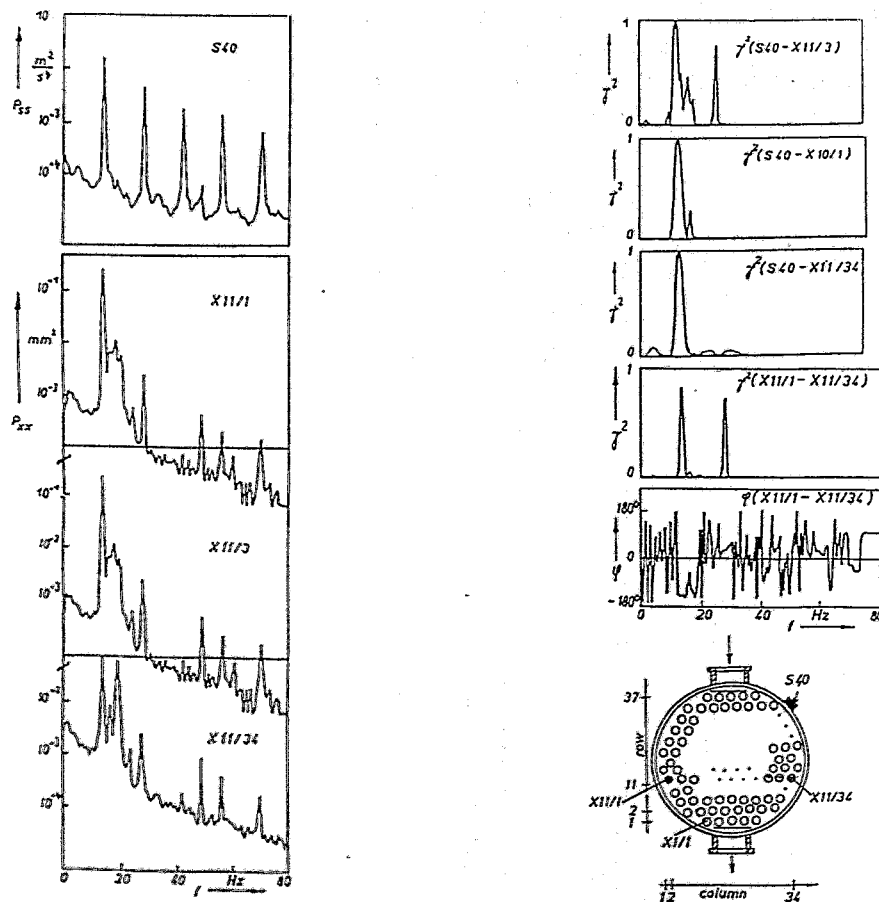


Fig. 2 APSs, coherence - and phase functions of the sound envelope S40 and different displacement signals X in the tube height of 1.5 m (mass flow rate = $1520 \text{ m}^3/\text{h}$)

4. The Origin of Collective Tube-Bundle Vibrations

Increasing the flow rate above a certain limit, the 14 Hz-tube-vibration amplitude rises drastically. This fact is interpreted as loss of stability that sets on at a flow rate of 1300 m³/h (80% of the nominal value) for this bundle construction (Fig. 3).

Unstable flow-induced tube vibrations are characterized by large amplitudes and by typical directions of the displacement with respect to the direction of the coolant velocity, for instance the wake galloping-vibrations are characterized by out-of-phase tube motions perpendicular to the flow direction and the jet-switching-vibrations by out-of-phase tube motions in coolant flow direction (see Fig. 3). These preferred directions of the motion are confirmed by both, opposite phase signals (Fig. 2) and by damage traces originating from mutual impacts of the tubes and from tubes impacting the baffles (Fig. 4 and Tab. 1).

In those bundle areas with a high fluid velocity and a free tube length of 2m (see Fig. 1) self-organization of unstable vibrations leads to *Collective Tube-Bundle Vibrations* at the eigenfrequency of the tubes.

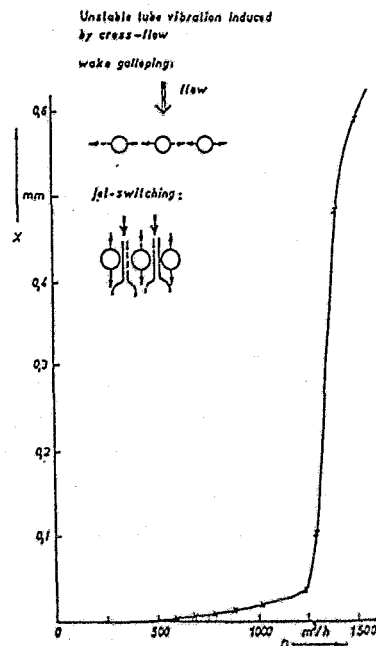


Fig. 3 Dependence of the 14 Hz - displacement signal on the mass flow rate (Hallsensor in tube height of 1.5 m)

Tab. 1 Positions of the damage marks at the tube 4/28

height [cm]	size [cm]	angle φ^{90} [°]	damage mark	cause
190	0.8	0 ± 75	rubbing marks	rubbing and impacting at the upper baffle
	0.1	180 ± 90	notch	
147	-3 ... +3	0 ± 30	hammer stroke	bending of the tube over the length of 3 m
		180 ± 30		
113	-9 ... +3	126	narrow tapering impact traces	bending of the tube over the partial length of 2 m
100	-9 ... +9	0		overharmonic of the 3 m-mode
83	-1,5 ... +1,5	0		

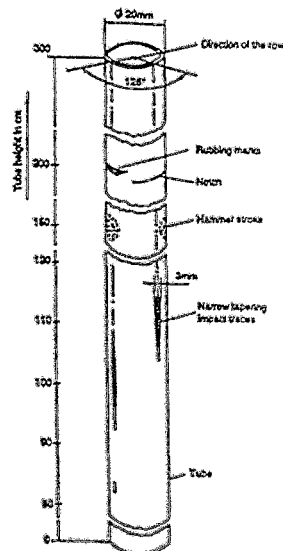
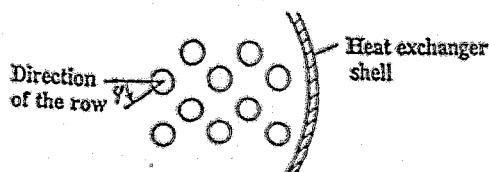


Fig. 4 Drawing of the damage marks at the tube 4/28 (Tab. 1)

5. Interpretation as Synergetic System

With a control parameter (fluid velocity) below the stability limit, the about 1000 microsystems (tubes) vibrate mutually uncorrelated (small amplitudes, arbitrary directions, see Fig. 5). This is a disordered system state characterized by chaos in the microsystems (Fig. 6a).

With the control parameter beyond the stability limit depending on the bundle geometry and on the tube eigenfrequency, unstable tube vibrations occur with large amplitudes at the eigenfrequency. Neighbouring tubes are slaved (locked in) through non-linear fluid-structure interaction (Fig. 5). The slaving principle is working, that means the unstable mode of the tubes turns into the order parameter of the macrosystem (tube-bundle).

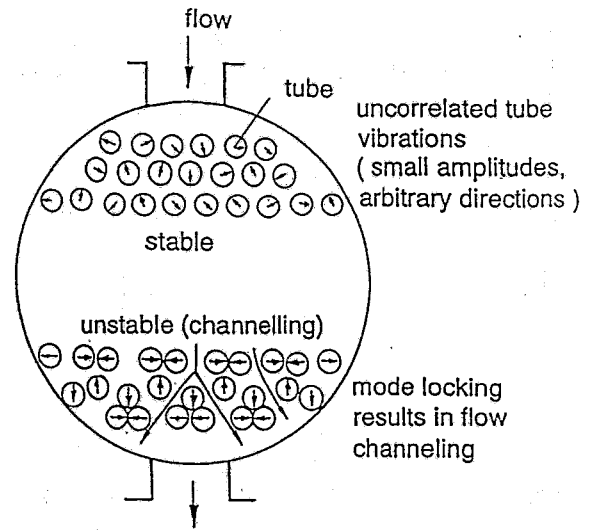


Fig. 5 Channelling in the tube-bundle after loss of stability of the individual tube vibration

Owing to non-linearities the order parameter equations describing the macrosystem show two stable modes that result in bifurcation (Fig. 6b):

- transversal vibration in the row (wake galloping)
- longitudinal vibration in the row (jet-switching).

Both the modes mutually stabilize each other by means of fluid-structure feedback. Through self-organization in the bundle the space-time order states of the macrosystem called *Collective Tube-Bundle Vibrations* are generated. Their features are

- * tube vibrations with large amplitudes and common eigenfrequency
- * impacts with the sequence of the eigenfrequency
- * damage marks and tube failures
- * channelling (Fig. 5): The coordinated displacements of the single tubes form "coolant channels," characterized by a minimum loss of the coolant pressure.

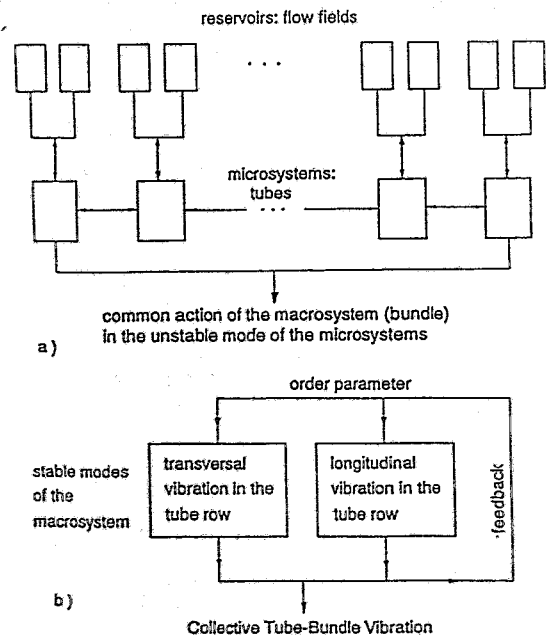


Fig. 6 Diagramm of the synergetic system: tube-bundle with cross-flow
a) microsystem (tubes)
b) macrosystem (tube-bundle)

6. Conclusion

The phenomenon of *Collective Tube-Bundle Vibrations* that can be detected through the impacts with the sequence of common eigenfrequency can only happen after the loss of stability of the individual tube vibrations. That means, the flow rate was too high for this type of heat exchanger. Owing to large vibration amplitudes tube failures could be caused. Based on the results of this investigation, the producer was forced to build new heat exchangers with an other tube-bundle construction at his own costs.

Systematic Analysis of Noise Signals in the Nuclear Reactor Noise Diagnostics of Abnormal Core Barrel Motion

P. Liewers, W. Schmitt, P. Schumann, F.-P. Weiß

In nuclear power plants the noise diagnostics has become a routine tool for monitoring disturbing processes. The routine application is optimal only if the disturbing process has been investigated in detail, and an effective evaluation procedure for the parameter estimation from the noisy signals has been explored. This detailed analysis of noisy signals or, in other words, the experimental provement of the process character and his consequences is briefly described for the abnormal core barrel motion process in a PWR of the Russian designed WWER-440 type reactor.

Usually the neutron noise measured in excore positions around the reactor is used for core barrel motion detection because of the effect of varying neutron transmission following by the varying water gap between the moving core barrel and the fixed pressure vessel. The sensitivety threshold of this method is in the order of 10^{-6} m.

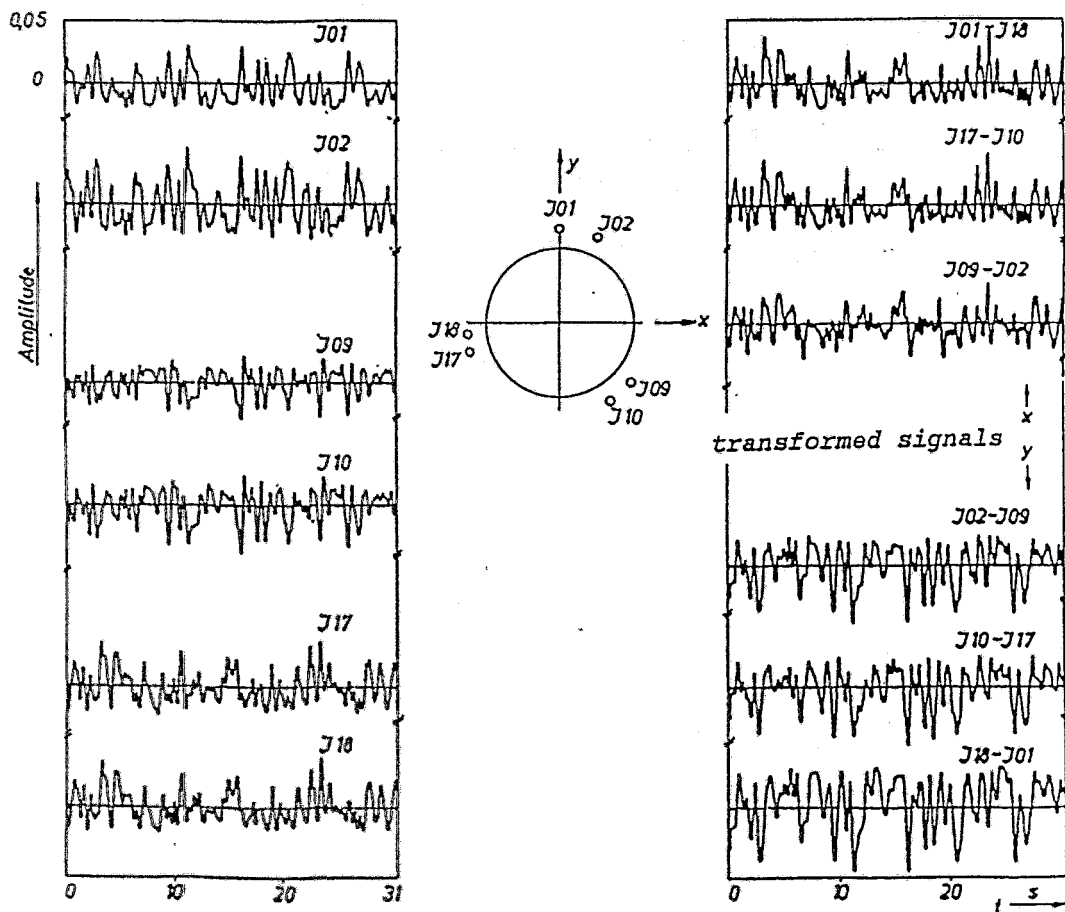


Fig. 1 Neutron noise signals measured by external ionization chambers (left) and after their coordinate transformation (right).

First, the displacement character of the excore neutron noise has to be demonstrated. To do this a simple coordinate transformation of the measured excore neutron noise is usable as shown in Figure 1. The resulting fluctuations, at right in Fig. 1, are in good agreement in the directions x and y respectively. The mean amplitude is nearly 2 mm, but maximum values reach 6 mm.

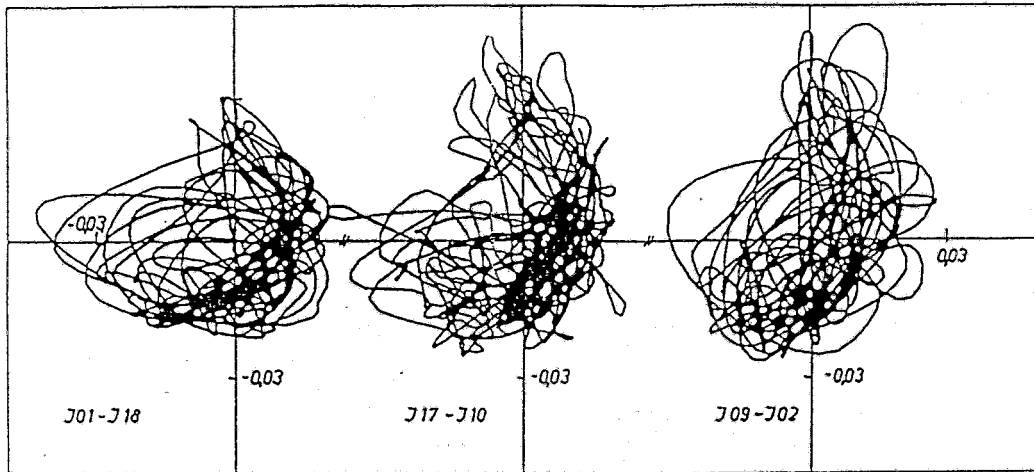


Fig. 2 Trajectories of the core centre using the transformed signals of Figure 1

The transformed signals can be used to represent the trajectory of the centre of gravity of the reactor core as given in Figure 2. It is surprising that, in most cases, the centre of gravity moves to the right and the movement seems to be limited at the right side. Furthermore, included in the trajectories there are small loops and turns which have been observed at other objects in conjunction with touch events. The reason of this turns is of diagnostic interest.

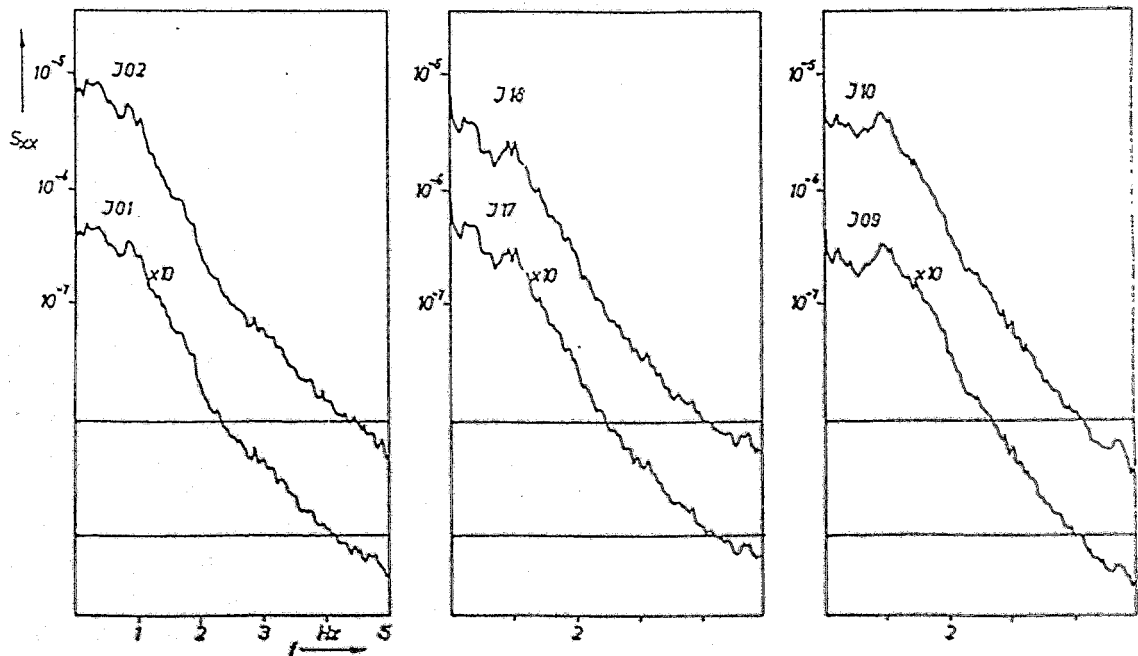


Fig. 3 Normalized auto power spectral densities of the excore neutron noise at different positions around the reactor

Second, the normalized auto and cross power spectral densities are estimated and show a significantly damped resonance structure at frequencies < 1.5 Hz. in Figure 3.

In this resonance region the coherence γ^2 (Figure 4) of excore neutron noise signals in nearly rectangular positions (e.g. I17-I10) are very small. Greater values are obtained at higher frequencies with all signal combinations.

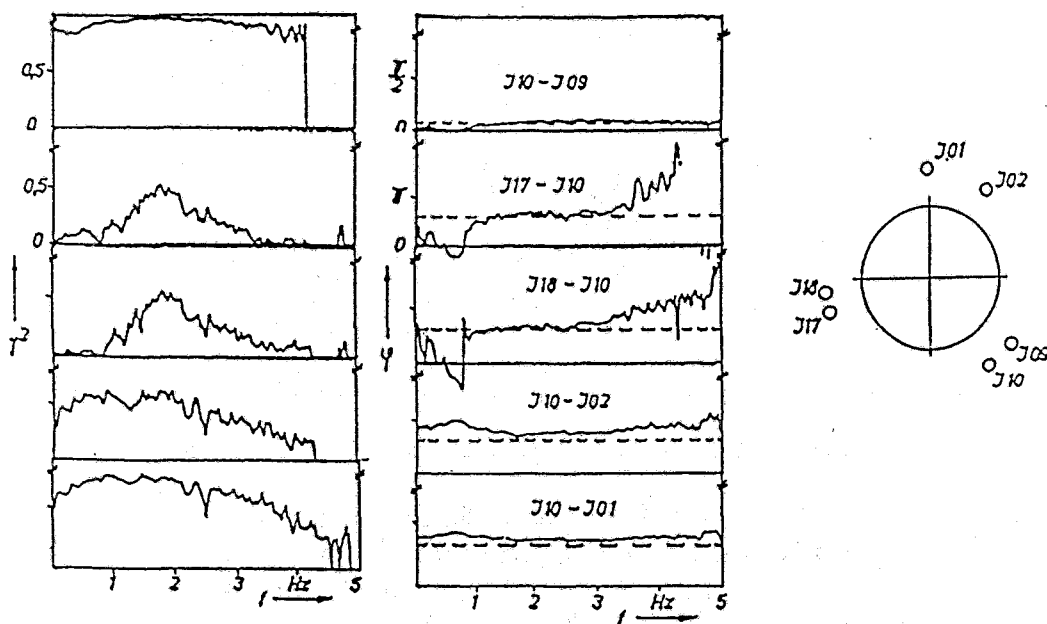


Fig. 4 Coherence and phase difference functions between the excore neutron noise at different positions

Interpreting the power spectra for every frequency point as an elliptical motion of the core barrel the reason of this different behaviour can be found by analysing the amplitudes of the elliptical axis a , b and the rotation direction: At lower frequencies greater amplitudes appear and the rotation direction varies. Only for frequencies > 1.5 Hz smaller amplitudes and an unique left-rotation occur. The reason especially for the change of the rotation direction of the large mass of the core barrel is of diagnostic interest.

Third, searching for touch events acceleration signals from different positions of the pressure vessel surface are used. Occuring small strokes of the body sound can be used to create a pulse signal by amplitude demodulation, usable for triggering the data acquisition procedure. Selecting all measured sweeps with the same elongation direction (provable by the displacement character of the excore neutron noise) the average functions of measured signals can be estimated as in Figure 5. The most important information from this averaged signals is the excitation behaviour in the acceleration signals (middle column in Fig. 5) from the pressure vessel flange B10 - B13, containing the eigenfrequency of the pressure vessel and the phase differences, as found during artificial excitations in the comissioning period. Of additional interest is the small negative pulse (left column in Fig. 5) from the pressure vessel bottom position signal B01. Therewith, touch events of the core barrel within the lower part of the pressure vessel are experimentally demonstrated.

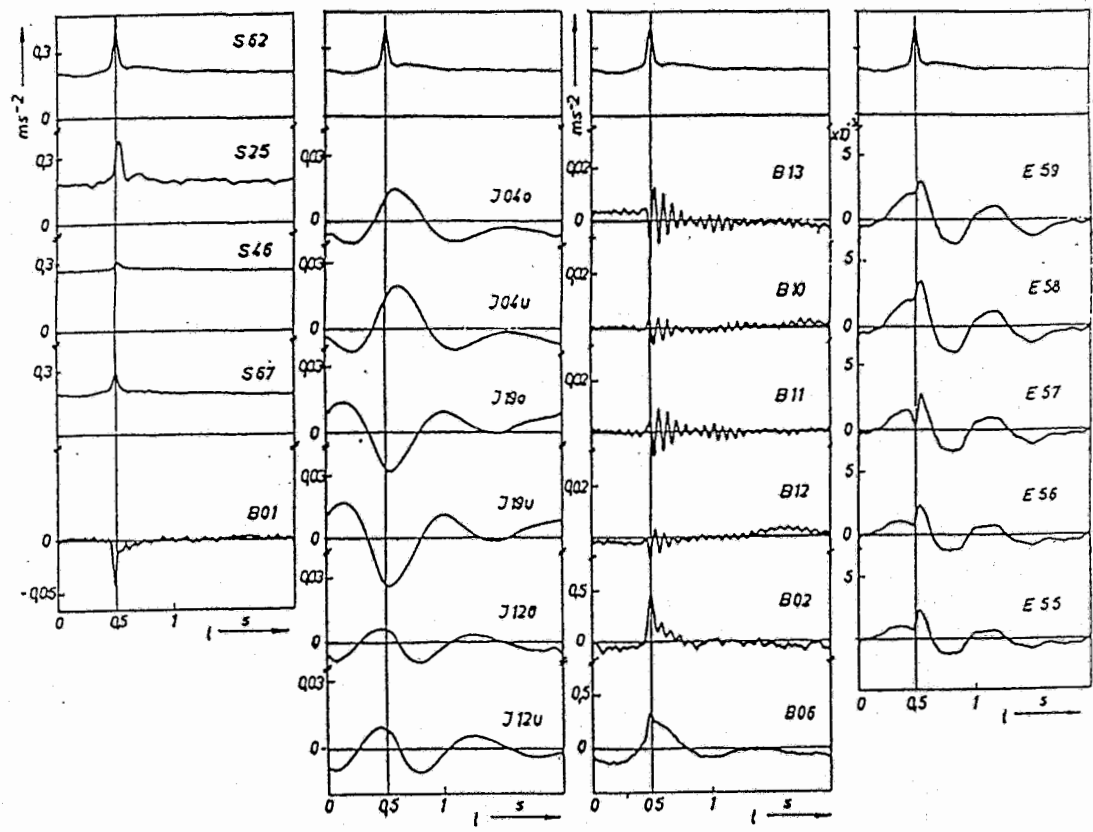


Fig. 5 Average functions of measured signals, correlated to the sound event of signal S62

Fourth, for practical purposes the positions of touches are of interest. This positions cannot be directly investigated by using sound propagation because the number of acceleration detectors at the pressure vessel surface is small and many following small touches can appear. Therefore the above-mentioned loops and turns in the trajectories (Figure 2) are analysed to estimate that elongation direction existing during the turn and to registrate their frequency distribution. The result in Figure 6 at the left shows the optimization procedure using only turns with a large elongation greater the mean elongation \bar{R} to reduce the contribution of randomly occurring loops. The mean event direction corresponds to the elongation direction in which the guide wedge number 1 can be touched. Two months later, in the second measurement at right in Fig. 6, turns occur in elongation directions in which the guide wedges 8 and 4 can be touched.

Relating to this result the following Table presents the findings during the inspection. The guide wedges fitting in the guide lugs normally have a thickness of 70 mm. By impacts they where reduced to smaller values.

Table: Thickness of the wedges in mm (March 1986, unit 2, Greifswald)

wedge-thickness	1	2	3	4	5	6	7	8
	60.8	62.6	64.2	64.4	64.0	62.4	61.6	61.2

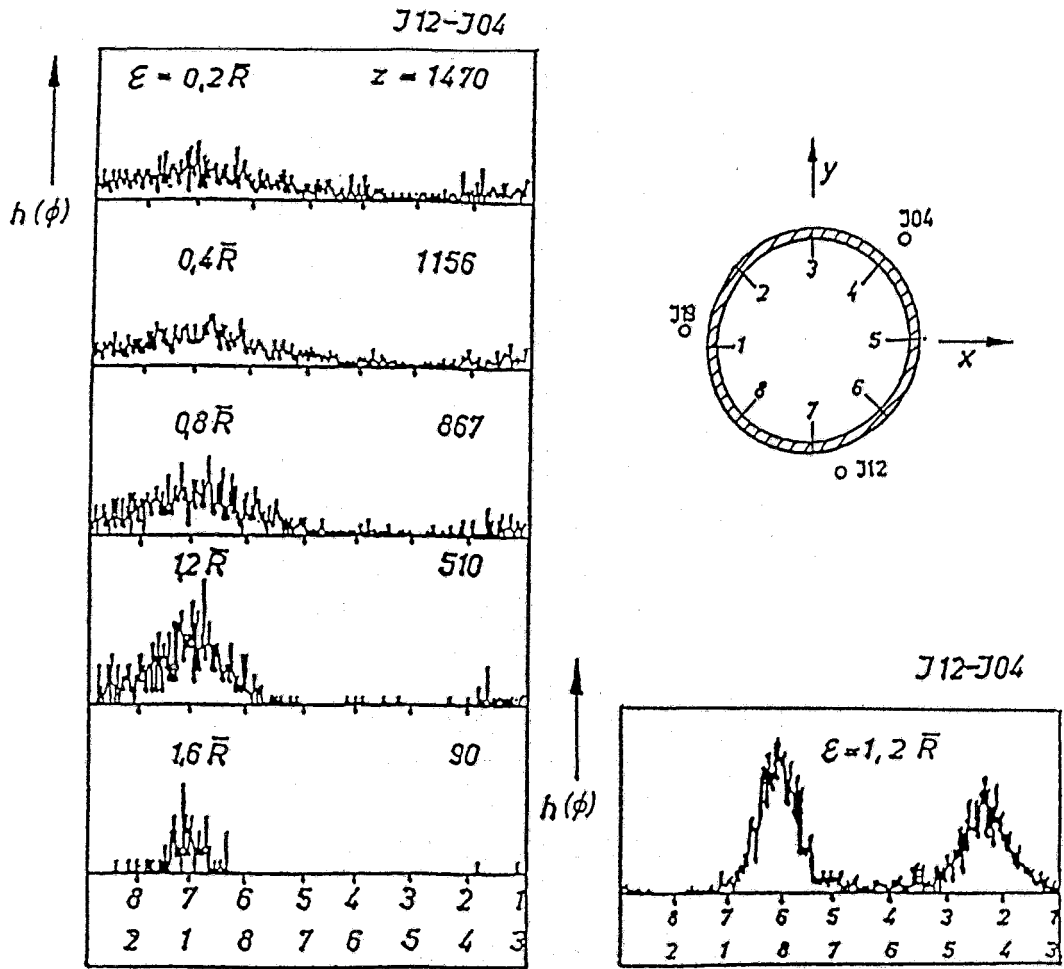


Fig. 6 Frequency distribution of the turns during the rotation of the core centre gives hints for the loaded guide wedges

Conclusions

For routine monitoring of the core barrel motion the following analysis procedures can be suggested in dependence on the investigated amplitudes :

- 1) Prove the significance of the displacement behaviour by coordinate transformation and investigate mean values, maximum values and turns.
- 2) Using the power spectral densities search for different behaviour in the coherence and phase functions in frequency ranges of large and small amplitudes for signal combinations at rectangular positions.
- 3) Search for body sound events and identify the excited eigenfrequency.
- 4) Use the turns for localization of touched construction elements.

Nuclear Accident Prevention by Remote Monitoring (NAPREM)

H. Carl, P. Schumann and F.-P. Weiß

It is imperative to ensure the safe operation of those Eastern European nuclear power plants requiring improvements and being well suited for improvements. User-independent remote monitoring of this nuclear power plants offers effective and advantageous means of contributing towards satisfying this objective. The NAPREM study examines the technical feasibility, required architecture possibility and basic functions of the system, as well as the magnitude of the costs associated with remote sensing of pressurized water reactors of the Russian-types WWER-440 and WWER-1000, respectively.

The remote monitoring is accomplished by means of downloading safety-relevant technological measurement and status information, accident instrumentation signals, additional reactor block system and component diagnostic measurement values, and dosimetric and meteorological values, supplemented by seismic, acoustic and fire warning signals. After being pre-processed, tested, and pre-evaluated, the downloaded information is transferred to a user-independent National Monitoring Center by an appropriate remote data transfer means. There this information is monitored and evaluated in terms of compliance with limits and conditions of safe operation, and stored for statistical analyses for the purposes of developing an unbiased knowledge base and providing meaningful data retrieval. Additional local air pollution calculations at different conditions directed to the power plant environment are carried out. The exact quantity of data transmitted to the National Monitoring Center depends strongly on the detailed objective: both a limited plant monitoring as well as a detailed monitoring of internal processes inside the core, primary and secondary circuits and auxiliary systems are possible purposes.

Furthermore, the NAPREM concept considers that selected information is forwarded, again by means of remote data transfer, to an independent International Monitoring Center. Besides collecting and evaluating statistical operational and environmental data, the tasks of this center consist of incident monitoring and/or radiological environment monitoring and/or mesoscale air pollution calculations at accidental conditions. The objectives of the center are incident support, determination of warning information and warning measures for crisis consultations, and crisis management for administrations and authorities, as well as forwarding of experience from statistical evaluations.

The suggested system architecture of NAPREM is shown in Fig. 1.

A side result of the study shows that, on one hand, a high quality of data transfer is necessary and possible by commercially available equipments, but that, on the other hand, the communication systems available in Eastern Europe do not satisfy these requirements. At this time, the task at hand can be accomplished only by means of a special remote data transfer system, using special land lines or satellite channels.

The study shows, above all, that remote monitoring of WWER reactors is technically and financially feasible. Compared to other necessary improvement measures, it is advantageous for both schedule and cost.

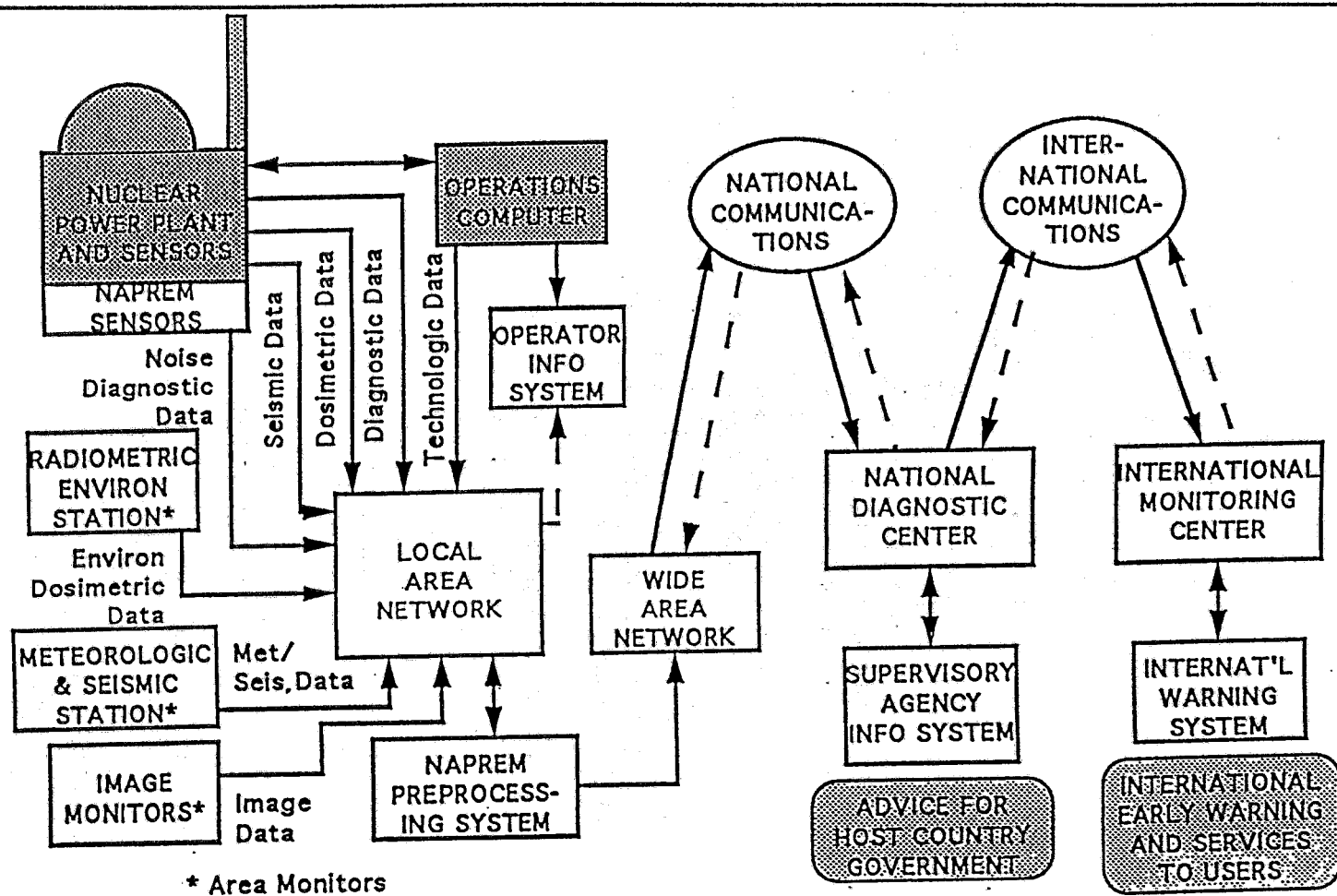


FIGURE 1

Suggested System Architecture for Remote Monitoring NAPREM

Neutronic Problems of a Compact 14 MeV Plasma Neutron Source

H. Kumpf, K. Noack and V.G. Krasnoperov

The development of proper materials for various components of future fusion reactors is a major issue of fusion technology. Materials for the first wall should withstand the neutron fluence of 14 MeV neutrons during the life span of the reactor ($10\text{-}30 \text{ MeV}\cdot\text{a}\cdot\text{m}^{-2}$) without too serious mechanical damage and get only weakly activated. Progress in the field is largely hampered by the lack of a sufficiently intense source of 14 MeV neutrons, the source strength of existing neutron generators missing the demand by five orders of magnitude. There exist several conceptions for neutron sources intended for fusion material research. The proposal of Budker Institute Novosibirsk and Efremov Institute St. Petersburg is based on a special kind of plasma mirror. Its main advantages are a proper DT-spectrum, high energy efficiency and sufficient irradiation volume. It is the only plasma neutron source with a technical predesign published /1/. In our paper /2/ some neutronic problems pertaining to this design are treated. Indeed, it is by no means obvious, that the sensitive parts of such a compact, high flux device can be adequately shielded. After all, space for shields is still more limited than in Tokamaks.

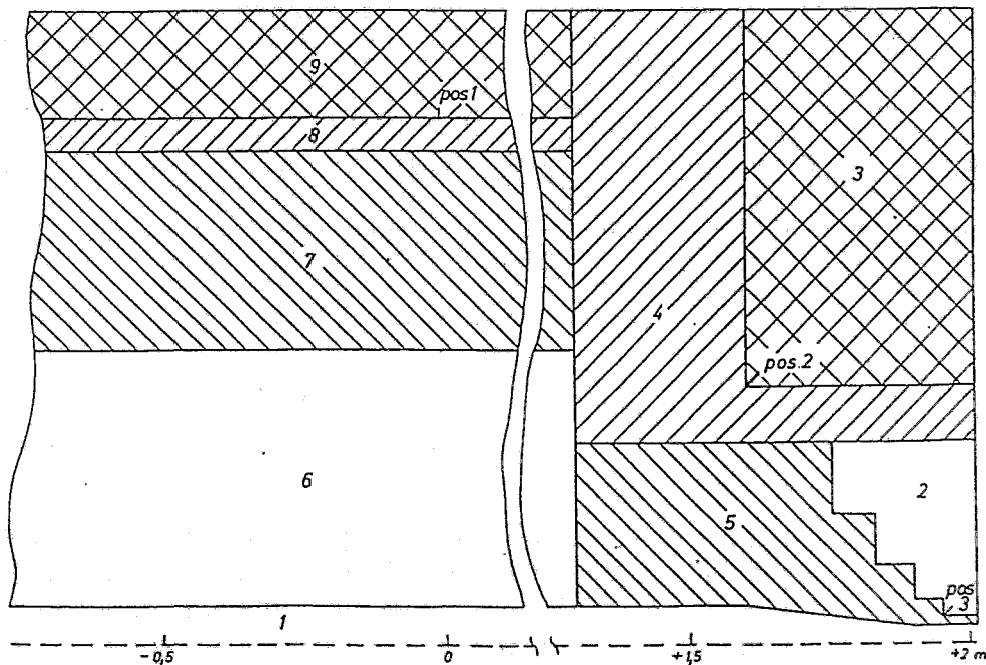


Fig. 1 Cylindric neutronic model of the Efremov design of a plasma neutron source (right half of the device). 1 Plasma 2 warm part of mirror magnet (Cu 0.8 natural density) 3 SC part of mirror magnet 4 structure (steel 0.5 natural density) 5 shield 85 vol % aqueous solution of H₃B₃O₃ enriched to 80 % in ¹⁰B 6 irradiation zone (steel 0.5 natural density) 7 shield, composition as 5 8 structure 9 SC coil. The source strength of 1 MW neutron 14 MeV is distributed in a cylinder 1.5 m long, 2.5 cm diameter centered at the origin.

The Figure demonstrates the right half of the neutronic model of the Efremov source predesign used as input to the Los Alamos Monte Carlo Transport Code MCNP. In spite of its idealizations it should retain the essential to neutronics features of the design. In anticipating the difficulties in housing the necessary shields within the magnets, tungsten has been selected as the principal component in zones 5 and 7. The other component is a 15 vol % aqueous solution of boron acid enriched in 10B. The considerable activation of tungsten will, of course, be a serious disadvantage, but, on the other hand, any hands-on maintenance can be avoided and remote handling, which already starts at the very beginning of operation, is planned. In the positions shown in Fig. 1 the most exposed parts of the magnet system are:

- Pos. 1, a ring on the bobbin of the superconducting main magnet located in the same plane as the center of the source,
- Pos. 2, the ring-shaped edge of the SC part of the hybrid mirror magnet,
- Pos. 3, the innermost ring-shaped edge of the warm insert of that magnet.

The results are listed in the following Table, where life spans of 10 full power years (FPY) were assumed for the SC coils and of 1 FPY for the warm insert. In parentheses the ratios of the calculated results to the limits of the corresponding quantities are noted. The charges applied to the SC part of the hybrid magnet exceed considerably the given limits - an inadequacy that is to be eliminated in a further stage of the design. The dimensions are expected to be tolerable.

	position 1	10 FPY	position 2	10 FPY	position 3	1 FPY
total dose in epoxy /rad/	3.0E9	(.60)	3.4E11	(68)	3.9E10	(2)
dpa in Cu	7.9E-4	(.13)	8.5E-2	(14)	1.3E-2	(.007)
n-fluence ($E_n > 0.1$ MeV) /cm ⁻² /	1.6E18	(.16)	2.3E20	(23)	1.4E19	

Besides questions of shielding two other neutronic problems related to the plasma neutron source are treated in paper /1/. Above all the potential of the plasma neutron source for thermal neutron research is evaluated. It is shown, that a thermal flux of about $5.E14$ n cm⁻²·s⁻¹ is obtainable with a D20 moderator. This corresponds to a good, but not to an excellent research reactor. Furthermore the question arises whether there exist reflectors for fast neutrons increasing the fast flux. The answer is negative.

References

- /1/ V.G. Krasnoperov, V.N. Odinzov, V.N. Skripunov and V.V. Philatov, Plasma Devices and Operations Vo. 3, No. 2, 1993
- /2/ H. Kumpf, K. Noack and V.G. Krasnoperov, Neutronic Problems of a Compact 14 MeV Plasma Neutron Source, FZR 93-11, April 1993

Improved Techniques of Analog and Digital Dynamic Compensation for Delayed Self-Powered Neutron Detectors

D. Hoppe and R. Maletti

Practical application of dynamic perturbation measurements for on-power determination of important parameters of nuclear reactors by means of delayed reacting neutron detectors is only possible, if a correction method is given to measure the time-dependent neutron flux behaviour without delay and with high accuracy.

An improved model of dynamic signal compensation is presented and illustrated by examples of analog and digital correction methods.

Knowing the transfer function of the neutron detector, it is possible to invert a dynamic (prompt jump response) system by transforming the output equation of the state equation system to the input.

An analog circuit corresponding this inverse detector kinetics was developed. This circuit, its structure in the case of rhodium self-powered neutron detectors is shown in Fig. 1, contains only one active electronic element, which is responsible for the compensation of detector dynamics, the signal amplification and the current-to-voltage transformation. The steady-state prompt fraction of the detector current is adjustable with the help of only one resistance. The circuit can be considered as the minimum of effort.

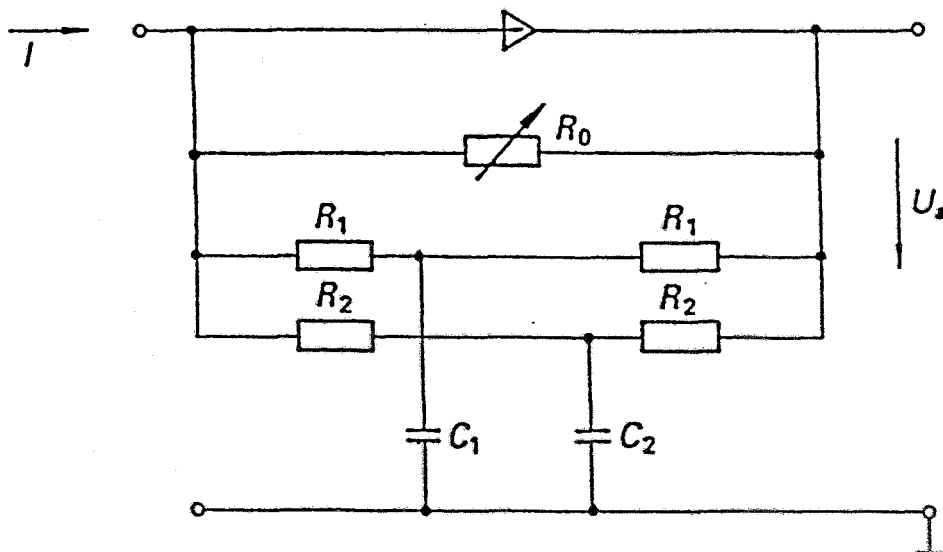


Fig. 1 Analog compensation for delayed RSPND

On the other hand a recursive digital algorithm of high computational speed and accuracy with regard to real-time processing was found. This discrete algorithm includes no differentiation of detector current and thus has a good noise gain.

The improved analog and digital dynamic compensation methods were developed and used in German and Hungarian nuclear power plants with pressurized water reactors of Soviet VVER type. By means of many rhodium self-powered neutron detectors and the named correction methods, the time- and space-dependent neutron flux behaviour during power changes or reactivity perturbations was followed to estimate reactivity coefficients like differential control rod worths or power coefficient. Furthermore, both the developed compensation principles and the reactor-dynamic perturbation method allow the estimation of the very important detector value of the steady-state prompt fraction of detector current by an experimental process analysis.

The German 1000-Roof-Photovoltaic-Programme: System Design and Energy Balance

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The German 1000-Roof-Photovoltaic-Programme was started in September 1990 in the Federal Republic of Germany supporting the installations of more than 2000 small grid-connected PV plants on the roofs of single- and two-family-houses.

The PV systems have to be operated for at least five years under the same technical conditions. During that time, the user has to participate in a standard measuring programme, in which the monthly results of three AC counters for the total PV energy, for the PV energy supplied to the grid and the energy drawn from the grid are registered. In the Federal States of Lower Saxony and Saxony, ISFH and FZR are supervising the PV systems and are collecting the data of the standard measuring programme.

- System design

In order to achieve a maximum energy output of the total PV system, the nominal power of the generator must be fitted to the nominal power of the inverter.

The best-matched generator inverter ratio can be determined from the inverter efficiencies and the frequency distribution of the yearly irradiance taking the typical power losses due to realistic operating conditions of the PV modules and due to DC components into account.

For the calculation of the maximum PV energy output, irradiation data (Hannover, 10-minute-means, 45° inclined and horizontal) are used to determine the nominal PV energy per kW_p for each irradiance class. The decrease of PV module efficiency at realistic operating conditions due to low irradiance, high module temperature and low incident angle was taken into account by various correction factors depending on irradiance.

The efficiencies of three different inverters in dependence on the input power were tested by TÜV Rheinland. The results of the calculations show, that the optimum matching ratio of inverter to PV peak power is 0.8 to 0.9 for the investigated systems.

- Results of performance of PV plants

For realistic calculations of the total PV energy output, all losses within the PV system have to be analyzed and determined.

Figure 1 presents the PV energy output per year of a 1 kW_p plant with specified energy losses. For a global irradiation of 1000 kWh/m²a, the expected PV energy output is calculated to be 637 kWh/year.

For 34 PV plants in Lower Saxony the specific system yields were calculated from the yearly means of PV energy output. The specific system yield (power), SSYP, is defined in kWh. Figure 2 shows a frequency distribution on the specific system yield marked for different inverters used in the installed PV systems.

In average, a SSYP of 788 kWh/a kW_p is gained by inverter 1 in comparison with 692 kWh/a kW_p by inverter 2.

Moreover, first results of measurements with solarimeters in Saxony are obtained. Totally running 13 months, three PV plants (5, 3.1 and 1.4 kW_p) are operating with a mean monthly performance ratio (total PV energy output/nominal PV energy) of 71.5 %. Considering seasonal corrections, the specific system yield will be in agreement with Figure 1.

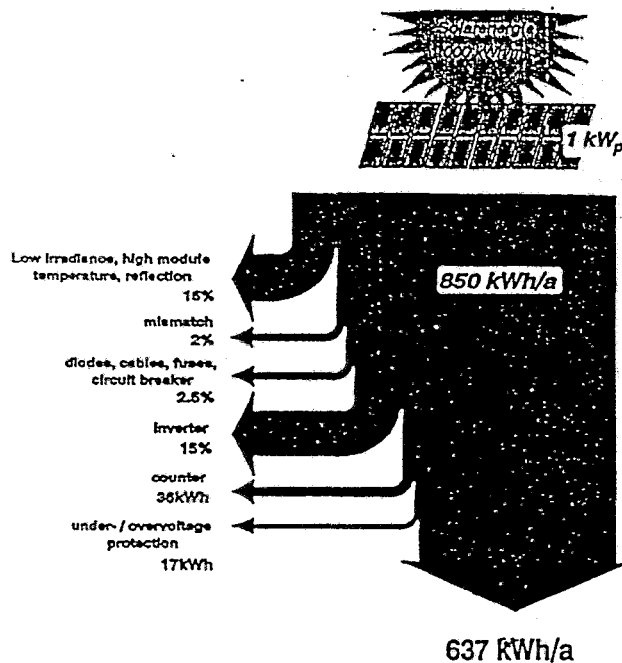


Fig. 1
PV energy output of a 1 kW_p PV plant per year

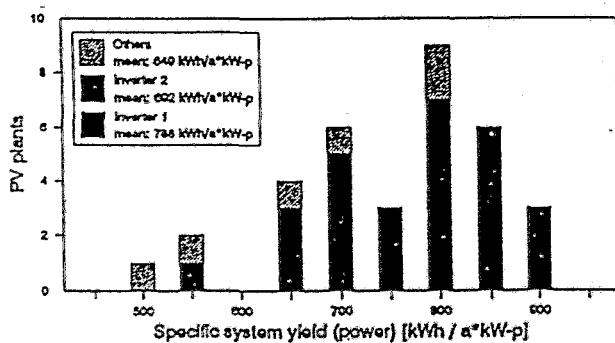


Fig. 2
Specific system yield, yearly means for 34 PV plants

Linear Stability of Marangoni-Hartmann Convection

K. Nitschke, A. Theß, G. Gerbeth

The paper presents a study of the influence of a homogeneous magnetic field on the surface tension driven instability (Marangoni instability) in an electrically conducting layer which is heated from below or cooled from above. Due to its practical relevance in crystal growth (cause of inhomogenities) the Marangoni instability has become interesting since a few years. It is known that the stability of flows in electrically conducting liquids can be dramatically changed and improved by application of magnetic fields. The paper provides a study of this effect on a simple model. The term "Hartmann" is used as a synonym for the magnetic field influence.

Mathematically, the problem can be posed as follows: We consider an infinitely extended layer of a liquid metal which is confined by a solid wall at the bottom and bounded by a free deformable surface at $z = d + \eta(x,t)$. A linear temperature distribution $T = T_1 - \beta z$ ($\beta = (T_1 - T_2)/d$, d : layer thickness) is maintained by heating from below and cooling from above, respectively. The surface tension is assumed to vary linearly with temperature. Under the neglect of buoyancy forces, which is admissible for very shallow layers under terrestrial conditions or for arbitrary layer thickness under microgravity conditions, the evolution of the system is conducted by a coupled set of nonlinear equations (Navier-Stokes-, heat conduction- and induction equation). This system is supplemented by appropriate boundary conditions. At the bottom we impose the non-slip condition, the continuity of the magnetic field and constant temperature. At the deformable surface the balance of normal and tangential stresses has to be taken into account. The stability of the basic state (quiescent liquid) is studied with respect to two-dimensional infinitesimal perturbations (normal modes) in the framework of linear stability theory. After scaling one gets a coupled system of linear differential equations which contains all relevant nondimensional parameters like Marangoni- ($d\sigma/dT\beta d^2 / (\rho_o \nu \kappa)$), Hartmann- ($Bd \sqrt{\sigma_{cl}/\rho_o \nu}$), Bond- ($\rho_o g d^2 / \sigma_o$), Capillary- ($\rho_o \nu \kappa / \sigma_o d$) and Prandtl number. ($\sigma, \rho_o, \nu, \kappa, \sigma_{cl}, B, g$ denote surface tension, density, kinematic viscosity, thermal diffusivity, electric conductivity, magnetic field strength and gravitational acceleration.) The problem is a complex interplay of thermocapillary, electromagnetic and surface wave phenomena and we make no attempt to completely explore the six-dimensional parameter space. We confine the computations to values typically for liquid metals and semiconductor melts. The obtained curves Ma ($a, Ha, Bo = 0.01, C = 10^{-5}, Pr = 0.02$) for several Hartmann numbers are shown in Fig. 1. The Marangoni instability sets in as stationary convection if the liquid layer is heated from below. Changing the heating direction and admitting surface deflections an onset of Marangoni instability in form of an oscillatory convection can be reached.

The suppressing influence of the magnetic field on the onset of both types of Marangoni instability is clearly indicated. It is caused due to the interaction of the magnetic field with the electric current density produced by a fluid flow. The appearing Lorentz force brakes the fluid motion by diminishing the net traction which the surface fluid flayer experiences. The modification of the spatial structure of the first unstable mode in the presence of a magnetic field for the case of stationary Marangoni-Hartmann instability is shown in Fig. 2. With increasing Hartmann number the distance between the roll cells becomes lower. The formation of a Hartmann layer with a layer thickness proportional to $1/Ha$ due to the dominance of electromagnetic over viscous forces causing suppression of bulk fluid motion can be recognized.

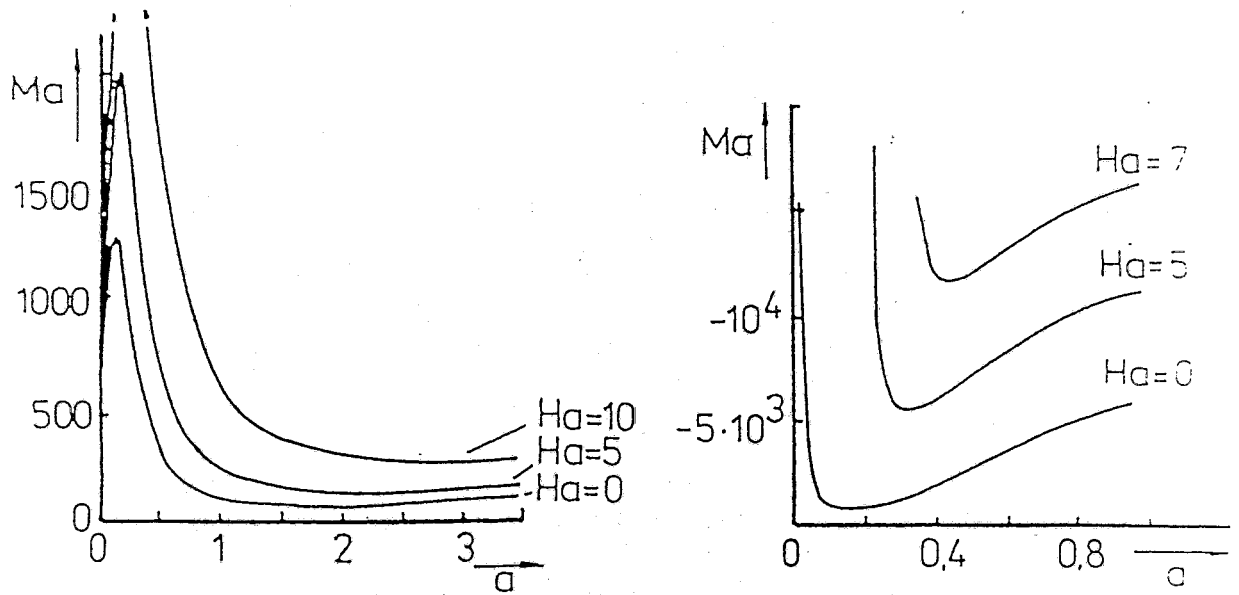


Fig. 1 Neutral curves for stationary (left) and oscillatory Marangoni instability for several Hartmann numbers

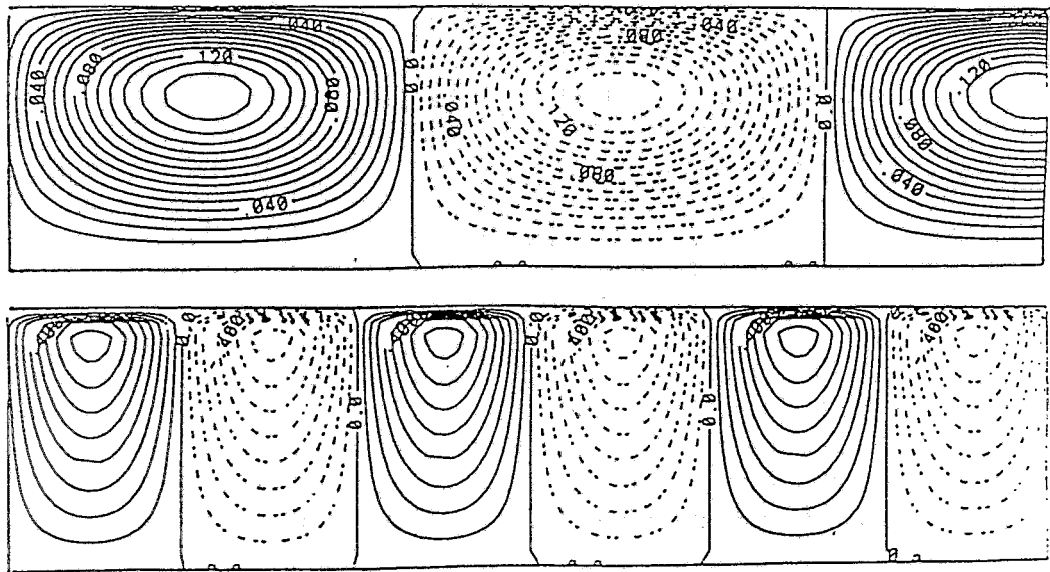


Fig. 2 Streamfunctions (stationary convection) for $Ha = 0$ (above) and $Ha = 30$ (below)

The paper performs a comprehensive study of a class of surface tension driven instabilities in an electrically conducting fluid exposed to an external magnetic field. It is shown that both types of Marangoni instabilities are suppressed. The results lay the foundation for an experimental test of the prediction about the delay of stationary and oscillatory Marangoni instability due to the action of a magnetic field.

Bubble Detection in Liquid Metals

F.R. Block, R. Dittmer (RWTH Aachen), G. Gerbeth (FZR)

Paper presented at the "International Conference on MHD Processes to Protection of Environment", Kiev, June 1992

The paper describes a first experimental test of a new electromagnetically based bubble detection method in an electrically conducting fluid. A reliable and continuous bubble detection method is important for any liquid metal heat transfer system, e.g. in Liquid Metal Fast Breeder Reactors or, in future, in a liquid metal cooled blanket of fusion reactors.

For such an early detection system electro-magnetic systems are thought to be eminently suitable. For this reason such a measuring system has been conceived. Initial tests have been carried out. The results are presented in the paper.

The experimental approach for the detection of inhomogeneities was developed at RWTH Aachen for an early slag detection in steel production. The method is based on the fact that a primary electromagnetic signal is changed by the motion of an electrically conducting fluid. This change is very sensitive to any inhomogeneities in the flow which have a different electrical conductivity compared to that of the fluid. The signals of suitably arranged secondary coils can be connected in order to separate the influence of the inhomogeneity. The system is sketched in Fig. 1. Typical results are shown in Fig. 2 for two different primary frequencies. In fact, we were able to detect each single bubble (which were of diameters in the range 0.5-3 mm).

Compared to usual, mainly acoustically based detection methods the following advantages of the electromagnetic approach are obvious:

- No time delay between bubble occurrence in the coil region and the signal.
- High sensitivity up to single bubble detection.
- Contactless method. No need to install sensors at hot walls, even a thermal isolation between the coils and the channel is possible.
- Each flowing bubble is detected whereas acoustic methods are able to detect acoustically active bubbles only.

The results of our one-day preexperiment without special preparation show the efficiency of such an electromagnetic detection method convincingly.

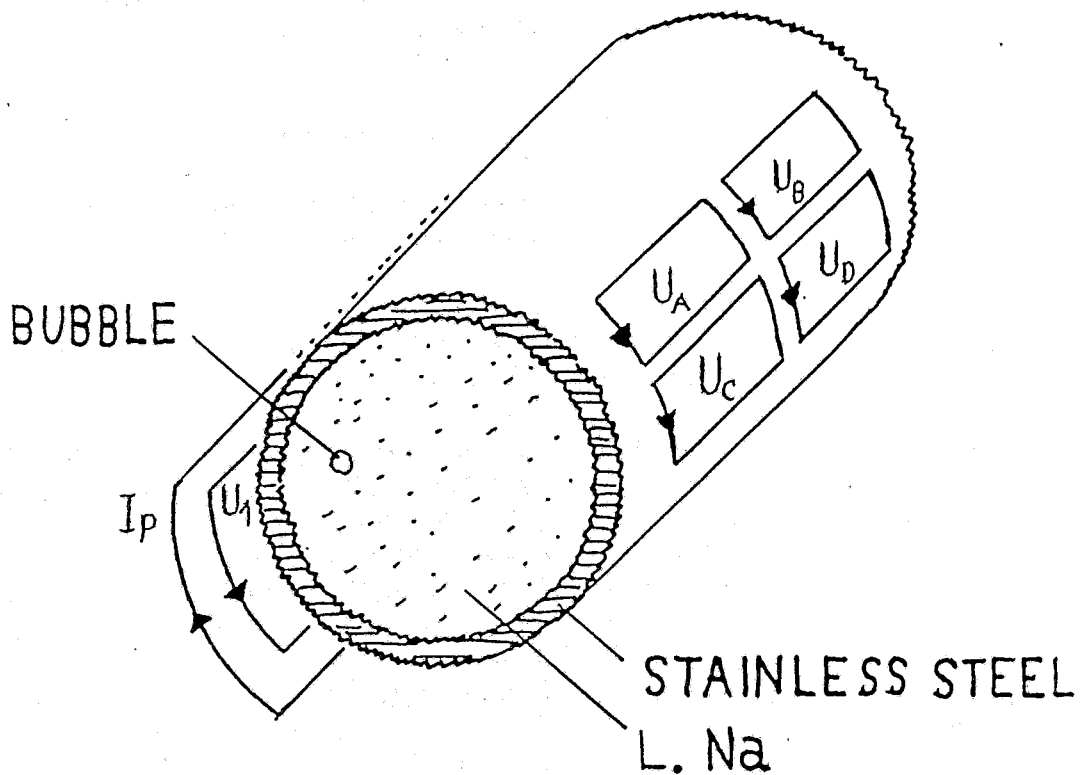


Fig. 1 Horizontal tube with a big secondary coil between a primary coil and the tube and four small secondary coils at the opposite side

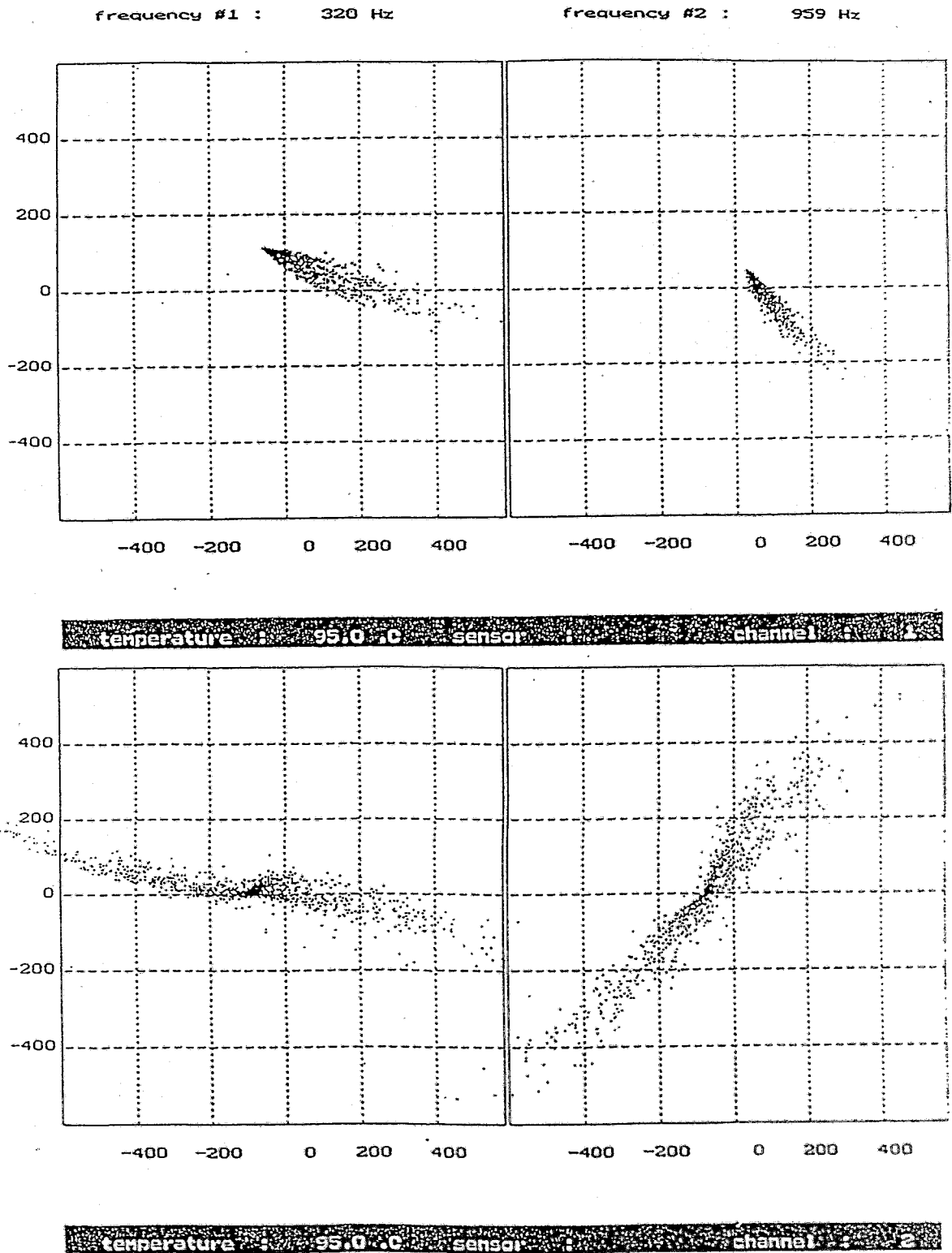


Fig. 2 Measured impedances with a vertical arrangement
 flow velocity: 1.0 m/s voidfraction: 0.45 %
 upper part: config. 1: $\Delta U = U_1 - (U_A + U_B + U_C + U_D)$
 lower part: config. 2: $\Delta U = (U_A + U_C) - (U_B + U_D)$

3. Survey of Projects

Project Title	Sponsor	Volume (manpower)	Project Leader	Project Period		
				1992	1993	1994
Photovoltaic Energy Supply for Dosimetric Measuring Systems	Zentralstelle Solartechnik Hilden		Dr. Rindelhardt	████████████████████		
Project Coordination and Realization of the 1000-Roof-Photovoltaic-Programme in Saxony	BMFT/SMWA 40 % / 60 %	2	Dr. Rindelhardt	████████████████████		
Numerical Simulation of MHD Flow Around a Circular Cylinder	DARA	1	Dr. Gerbeth	████████████████████		
Dependence of Mechanical Properties, Irradiation and Annealing Behaviour on the Through-the-thickness Position for Forging of 15Ch2MFA VVER RPV Steel	BMFT	1	Dr. Böhmert	████████████████████		
Investigation of Relations between Fracture Mechanical and Technological Parameters of Irradiated Reactor Pressure Vessel Steels	BMFT	1,5	Dr. Viehrig	████████████████████		
Analytical Modelling of Mechanical Vibration of VVER-440 Primary Circuit Components Using Finite Elements	BMFT	2	Dr. Altstadt	████████████████████		
Determination of the Neutron Fluence of Irradiated Specimens of VVER Reactor Pressure Vessel Materials	BMFT	2	Dr. Barz	████████████████████		

Project Title	Sponsor	Volume (manpower)	Project Leader	Project Period		
				1992	1993	1994
Further Development of a Three-Dimensional Core Model for VVER-Type Reactors and Its Coupling with the Code ATHLET for Accident Analysis	BMFT	2	Dr. Rohde			
Leak Detection at Complicated Three-dimensional Topologies	SMWK	2	Dr. Schmitt			
Collaboration with Institutions of Central and Eastern European Countries in the Field of Reactor Safety Research	SMWK		Dr. Weiß			
Remote Monitoring of Eastern European NPP	SMWK	2	Dr. Carl			
Feasibility Study on Ultrasonic Two-Phase Flow Measurements Based on Pattern Recognition Techniques	BMFT	2	Dr. Prasser			
Thermocapillary Migration of Bubbles and Drops	DFG	1	Dr. Gebeth			
Design of a Nuclear Power Plants Remote Monitoring System as a Tool for the State Regulatory Authority	BMU	4	Dr. Carl			
Assistance of the Saxon State Ministry of Economy and Labour in the Promotion of Technologies for Economical Use of Energy and for Application of Renewable Energy	SMWA	8	Dr. Riedelhardt			

Photovoltaic Energy Supply for Dosimetric Measuring Systems

U. Rindelhardt

Dosimetric measuring systems are widely used in the supervision of the environment of NPP's and nuclear research centres. The measuring points very often are laying in such a long distance to the control room that a connection to the mains would be very expensive. A advantageous alternative is the use of a battery driven measuring system with solar power supply. Two prototypes of such kind of systems with different types of sensors were designed and tested for the planned environment supervision of the FZR. The work was sponsored by Zentralstelle für Solartechnik Hilden.

The two used types of sensors differ in both measuring range and power input, they are controlled by microcontrollers and include integrated dataloggers.

The simulation program ISLAND was used for estimating the values of the components of the solar system.

System 1:

Probe: FHZ 601 A (Kugelfischer); energy range: 5 nSv/h ... 5 mSv/h; voltage: 12 V dc; input power: 1.3 W; module: MQ 36/53 D (AEG); accumulator (rechargeable battery): Solarakku voltage: 12 V; capacity: 140 Ah (Akku-Gesellschaft).

System 2:

Probe: RD-02 (Alnor); energy range: 0.01 μ Sv/h ... 10 Sv/h; voltage: 12 V dc; input power: 0.34 W; module: M 12 (Siemens); accumulator: KOBE HP 12-24; voltage: 12 V, capacity: 24 Ah.

A one-year test period will follow to check the designed systems, especially to make statements about dimensions of the components, the power supplying certainty etc. The following parameters are measured and registered: solar generated voltage and current, battery voltage and currents (balance), and load current. The measuring routine, the calculation of the 15-minutes averages and the storage of the values are realized by external dataloggers (R. Schuehle). After the reading of these values with a portable computer a suitable software allows to calculate the results of hours, days and months of several or of combination of parameters (e.g. power). The graphic presentation is also possible. The available arrangements were put into operation at the end of October 1992.

A solar power supply was also built for the two dataloggers. The photovoltaic modules and the accumulator are very great: Modules: 4 pieces of MQ 36/53 D in parallel; accumulator: 2 pieces of Solarakku 12 V, 140 Ah in parallel. This arrangement significantly improves the reliability of the system.

Project Coordination and Realization of the 1000-Roof-Photovoltaic-Programme in Saxony

U. Rindelhardt

The German 1000-Roof-Photovoltaic programme was started in 1990 supporting the installation of small grid coupled PV systems (peak power: 1 - 5 kW_p) on the roofs of single- and two-family houses. The main aims of the programme are:

- Demonstration of the usefulness of roofs for PV system installation (architectonical aspects included),
- Testing of the users behaviour concerning the adaption of the electrical consumption to the solar energy production,
- Qualification of craftsman in designing and mounting of PV systems (know how),
- Long time test of PV systems from different suppliers and with different technologies concerning the reliable energy production.

In Saxony, the Institute for Safety Research of the FZR manages the project by order of BMFT and the Saxon State Ministry of Economy and Labour. Totally 150 PV systems can be installed in Saxony.

According to the conditions in the new Federal States, the work was focussed on the qualification of craftsman and a broad information on photovoltaics on different ways. The result of these efforts were more than 1300 inquiries of prospective users of PV systems up to the end of the year 1992. At the same time 159 serious proposals were registered. The first PV system was commissioned in January 1992, up to the end of the year 42 systems were coupled to the grid.

The technical concepts of the realized PV systems can be characterized by the used module and inverter types. In 1992 the shares of the module suppliers were AEG/Dasa 60 %, Siemens 36 % and Helios 4 %. The shares of the inverter suppliers were Siemens 46 %, SMA 37 % and Solarkonzept 17 %. The mean power of the installed systems was 3,2 kW_p, i.e. more than the averaged power in Germany.

The yearly electric energy production could be estimated to nearly 700 kWh/kW_p from the first measuring results, with remarkable differencies between some PV systems. To find the origin of these differencies an own Saxon measuring programme was planned and realized with support of the regional energy suppliers. In the frame of this programme about 20 PV systems, well distributed over the Saxon region, have been equipped with solarimeters in the module plane. The monthly irradiance can be measured with integrator units and the performance ratio of the systems can than be calculated. The performance ratio is the most important parameter in describing the energy losses in grid coupled PV systems. This programme will be continued in the next years.

Numerical Simulation of MHD Flow Around a Circular Cylinder

G. Gerbeth

General Aim:

The flow of an electrically conducting liquid around a circular cylinder under existence of an external magnetic field is interesting both for fundamental questions in fluid-dynamics as well as for some practical aspects.

In this project, the stationary flow of the 2-dimensional problem and its stability are being calculated by means of a finite-difference method and a global stability analysis, respectively. The influence of the magnetic field will be investigated in a quantitative way to find a stability curve in the (Re, Ha) -parameter plane above that the arising of the Karman vortex street is suppressed. Two- and three-dimensional instabilities will be discussed. The results obtained by the numerical simulation will further be compared with results from an Galerkin approximation of the flow as well as with results for the stability of a Kolmogorov flow model.

Results:

In the first month of the project period the new scientist made himself familiar with the topic and some related literature. Acquisition of a IBM/RISC6000 workstation was started paid by DFG in the frame of the project.

Dependence of Mechanical Properties, Irradiation and Annealing Behaviour on the Through-the-thickness Position for Forging of 15Xh2MFA VVER-RPV Steel

J. Böhmert

General Aim:

The standards of irradiation embrittlement surveillance programmes for reactor pressure vessels refer to specimen position of at least one quarter of the overall wall thickness because this position is expected to be most susceptible to irradiation embrittlement. There are experimental results for VVER-440 RPV steel hinting a different dependence of the embrittlement on the depth position.

Therefore the project is aiming at a detailed investigation of connection between mechanical properties and the through-the-thickness position for 15Xh2MFA steel in distinct states. The research serves the further qualification of VVER safety assessment.

Results:

In the unirradiated state specimens from different thickness positions were investigated by means of metallography, X-ray diffraction and measurements of hardness, microhardness, and Charpy-V-impact energy.

The surface position shows a lower nil-ductility transition temperature and a higher hardness and strength than positions from larger depths. This dependence can be connected with different amounts of ferrite and differences both of carbide microstructure and defect structures.

Investigation of Relations Between Fracture Mechanical and Technological Parameters of Irradiated Reactor Pressure Vessel Steels

H.-W. Viehrig

General Aim:

The aim of this project is to provide correlations between values of mechanical-technological properties and fracture toughness properties of irradiated Soviet type VVER and western (ASTM) reactor pressure vessel (RPV) steels. Researches are focussed at the determination of mechanical-technological properties and fracture mechanical values, the evaluation of correlations between both groups of data and of the proof of confidence of correlations. A further objective of the project is to develop techniques that help to interpret data obtained from specimens of surveillance programmes. Necessary prerequisites are to be established enabling a loading analysis in a wide range of operating and accidental conditions.

The first main task is the determination of fracture mechanical values of unirradiated, irradiated and post-irradiation thermal annealed RPV-steels. For dynamic testing the instrumented impact test is used.

Results:

The second main task is evaluation of Correlations between mechanical-technological values (for example hardness, tensile yield strength, tensile strength and impact energy adsorbed) and fracture mechanics values should be proved and confirmed.

An instrumented impact pendulum with equipments for measuring the impact load, the deflection of the specimen and acoustic emission signal during loading was implemented and a special signal processing software was developed for this test. We took part in the DVM round robin test of instrumented impact testing. The single specimen compliance method is used to determine R-curves during quasistatic loading. These measurements are carried out with a servohydraulic test system MTS 810. The loading equipment and the control and signal processing software were developed in our department. Calibration experiments to determine the crack length were carried out.

The licensing procedure for handling irradiated material in a preparation and a material testing laboratory is under way at the Saxon Ministry of Environment.

Analytical Modelling of Mechanical Vibrations of VVER-440 Primary Circuit Components Using Finite Elements

E. Altstadt

General Aim:

The project is to contribute to the evaluation of the mechanical integrity of VVER-440 type reactors. For this purpose the mechanical vibration behaviour of primary circuit components is to be modelled considering the theory of fluid-structure-interaction. The generated computation models are adjusted by means of vibration measurements to ensure a satisfying description of the reality. Using the adjusted computation models it is to be clarified how hypothetic mechanical failures will influence the vibration behaviour of the construction and which signals are the most suitable for the early detection of these failures.

Results:

Finite Element Modelling of a 1:10 scaled VVER-440 set-up

In order to establish the general structure of the finite element model that means to select the suitable FE, to formulate the stiffness matrices and to introduce a realistic coupling between the degrees of freedom of different structural components theoretical modelling started with the simulation of the vibrations of an 1:10 scaled set-up of a VVER-440 which is available in Rossendorf.

Beyond the determination of mass- and inertia parameters, emphasis was put on the formulation of the stiffness matrix of the pressure vessel bearing on the foundation ring. Special attention was spent to the influence of the clamping bracket forces on the eigenfrequencies and eigenshapes of the reactor pressure vessel bending and shell modes. Thereby it turned out that the clamping bracket forces affect the lower eigenfrequencies only.

Further items of generic interest were the modelling of the core barrel upper and lower fixing by so called spring pipe segments between the core barrel flange and the pressure vessel head and by the 8 guide lugs respectively.

All eigenfrequencies and eigenmodes occurring up to 500 Hz were calculated.

Fluid Structure-Interaction

The vibration model should explicitly consider the interaction of the vibrating components with the flowing coolant.

Two different types of interaction must be taken into account, namely:

1. excitation of the components by fluid forces
2. reaction forces of the coolant to component displacements

To introduce the fluid-structure-interaction into the finite element modelling approximated solutions are derived for the interaction forces and for the damping from the coupled equations of motion of a certain component, equation of continuity and Navier-Stokes equation. After experimental or numerical verification these approximations can be used in the finite element code.

The interaction was analytically investigated for a double cylinder geometry with a water gap between the two cylinders. The influence of the boundary conditions at the inlet and at the outlet of the double cylinder upon the reaction forces and upon damping were investigated in detail.

The best boundary conditions for the velocity field are those producing the smallest pressure differences over the circumference at the inlet and outlet level of the annular gap.

Using eigenfrequencies and dampings of the fluid-structure-system as criteria for comparison a satisfying agreement between the experimental and the approximated analytical results was reached. In the laminar flow regime the deviations are less than 10 % and still less than 15 % in the turbulent range.

It could be shown that in the frame of the used approximation elementary structural motions like the discussed double cylinder pendulum can be superposed to describe more general types of motion.

Modelling of the original reactor

Also the FE-modelling of the original VVER-440 was started providing as the first result that all shell mode vibrations of the pressure vessel are beyond 42 Hz.

Determination of the Neutron Fluence of Irradiated Specimens of WWER Reactor Pressure Vessel Materials

H.-U. Barz

General Aim:

Within the period from 1984 to 1988 a comprehensive set of specimens of different reactor pressure vessel steels (RPV steel) for the Russian power reactor types WWER-440 and WWER-1000 (manufactured in Russia and Czechoslovakia) as well as other steel types (ASTM-steel made in Japan and modified Russian steel types molten under laboratory conditions in the GDR) have been irradiated in the WWER-70 reactor of the atomic power station Rheinsberg. Simultaneously the neutron fluences have been monitored at some positions of the steel specimens.

The main objective consists in the theoretical calculation of neutron fluences in the different irradiation channels with high precision for a reliable prediction of fluences at all positions of specimens and monitors. The calculated spectra shall also be used as an input for the spectrum adjustment procedure to determine a best estimate "experimental" reactor spectrum on the base of measured reaction rates.

Results:

The first step was the collection of all basic data of the Rheinsberg reactor including the different reactor histories (burn up, power density etc.) for the considered reactor periods. On the basis of these data the source distributions of fission neutrons for different fission isotopes were calculated by special produced software. For this calculation we needed the spectrum of the Rheinsberg reactor which we have calculated by an own Monte Carlo criticality code TRAMOC.

Furthermore the adjustment of the Monte Carlo code TRAMO was performed and a proper model for the Monte Carlo calculations established. To save calculation time for the great number of calculation variants the improvement of the statistical errors of the Monte Carlo method is very important. Therefore among other things "forced weights" in the frame of the "weight window" method have been calculated using the own computer code TRAWEI. The most important changes and extensions of the codes TRAMO and TRAWEI have been finished.

For the test of all evaluated methods the first calculations were performed with respect to the RH-2 and RH-7 experiments in the target channel T6 of the Rheinsberg reactor using the 26-group-data base ABBN-78.

To improve the reliability of the calculations different group cross section sets are used. A special code development was performed for the conversion and re-organisation of the multigroup data in order to adapt the input cross section data generated by the university of Stuttgart (IKE).

The planned gamma spectroscopic analysis of the fluence monitors could not be started because of the missing operating license but an interim solution to handle low radioactive materials has been obtained. For the analysis of low radioactive fluence monitors a "low level" facility was installed.

For the spectrum adjustment procedure needed codes and data libraries were adapted to the computer CONVEX 3220 and calculations were prepared for those experiments for which all required data are known. First test calculations were performed.

Further Development and Verification of a Three-dimensional Core Model for VVER-type Reactors and Its Coupling with the Code ATHLET for Accident Analysis

U.Rohde

General Aim:

The goal of the project is to contribute to the improvement of calculational methods for a realistic safety assessment of the Russian VVER-type nuclear reactors. Especially, the project includes the coupling of the reactor core model DYN3D/M2 with the thermo-hydraulic code ATHLET of the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS). The 3D code DYN3D/M2 has been developed by Research Center Rossendorf to simulate VVER core behavior during reactivity accidents and xenon transients. The project is a part of the general efforts to couple 3-dimensional core models for hexagonal and square fuel element geometries with the ATHLET code.

The coupling of the DYN3D core model to the ATHLET code is the main objective of the project. Additional goals are

- the development of a DYN3D option for burn-up calculations,
- investigations about the necessary extent of completion and modernization of the neutron physical data base of DYN3D,
- further verifications of the DYN3D code,
- operating test of the coupled code version DYN3D-ATHLET.

Results:

The basic conception for coupling the codes has been completed. To enable methodical comparisons, two different approaches were chosen, namely:

- A) the insertion of the neutron kinetics part of DYN3D into ATHLET as the module "neutron kinetics" in analogy to the available point kinetics and 1-dimensional models ("internal coupling"),
- B) replacement of the reactor core modelling in ATHLET by the complete DYN3D core model, which includes neutron kinetics, thermo-hydraulics and fuel rod description by means of user defined routines in the frame of General Control Simulation Model GCSM. In this approach, the coupling is realized only via boundary conditions at core inlet and outlet ("external coupling").

Details of the coupling strategies will be clarified in close co-operation with the GRS. For approach A, essential preparational work for the technical realization of the coupling was finished. The approach B has already been realized in a first version with strongly simplified point model of the core neglecting neutron kinetics.

The existing group data libraries, the corresponding interfaces with DYN3D and the code itself have been implemented at SUN workstations as a condition for the development of an burn-up option. Methodical investigations for the burn-up version were carried out.

Calculations of benchmark problems contribute to further verification of the DYN3D neutron kinetics. A benchmark problem formulated by the international association "Atomic Energy Research" (AER) has been solved.

Acoustic Leak Localization at Complicated Three-Dimensional Topologies

W. Schmitt

General Aim:

The project aims at the development of an acoustic leak detection and localization method being capable of enhancing the operational safety of pressurized plant with dangerous media and complicated geometrical structure. Such plants are chemical or nuclear reactors and vessels containing toxic, explosive or aggressive media.

Up to now worldwide there is no generally applicable acoustic method for leak localization and leak rate specification. It is the intention of the project to provide an acoustic method that can as well be applied to chemical and to nuclear plant.

Due to the complexity of sound propagation in real technical structures featured by a many-fold of modes of structure-borne sound and by reflections, the known acoustic localization methods based on attenuation or propagation time measurements will fail in general, linear pipe geometry or vessels without connection branches excepted.

Therefore, the problem is approached by using pattern recognition methods with local sound intensities and sound frequency spectra as feature vectors. To train such pattern recognition procedures, leak simulation experiments must be carried out at the original or at least at geometrically similar structures.

To reach satisfying localization accuracy and transferability to geometrically different plants, pattern recognition methods based on fuzzy logic, neural networks, and correlation matrices were selected.

Results:

Being able to perform fundamental experiments on sound propagation in complicated geometry, to test and calibrate sensors and artificial sources of acoustic leak noise, an acoustic set-up was constructed. Ultrasonic microphones and acoustic emission (AE)-sensors were tested at that facility. As a result of these investigations a measuring equipment shown in Fig. 1 was designed to be applied at the original plants. This design utilizes as well structure-borne as also air-borne sound for leak localization. Compressed air jet facilities and AE-sensors driven in the inverse mode were used to excite artificial leak noise.

Moreover, the permission was given by the regulatory body and by the former operator of the Greifswald NPP to carry out experiments at the original reactor pressure vessel head of Greifswald unit 6.

In this project the University Zittau/Görlitz (HTWS) and the Fraunhofer Institution for Acoustic Diagnostics and Quality Assurance (EADQ) Dresden collaborate with the Institute for Safety Research.

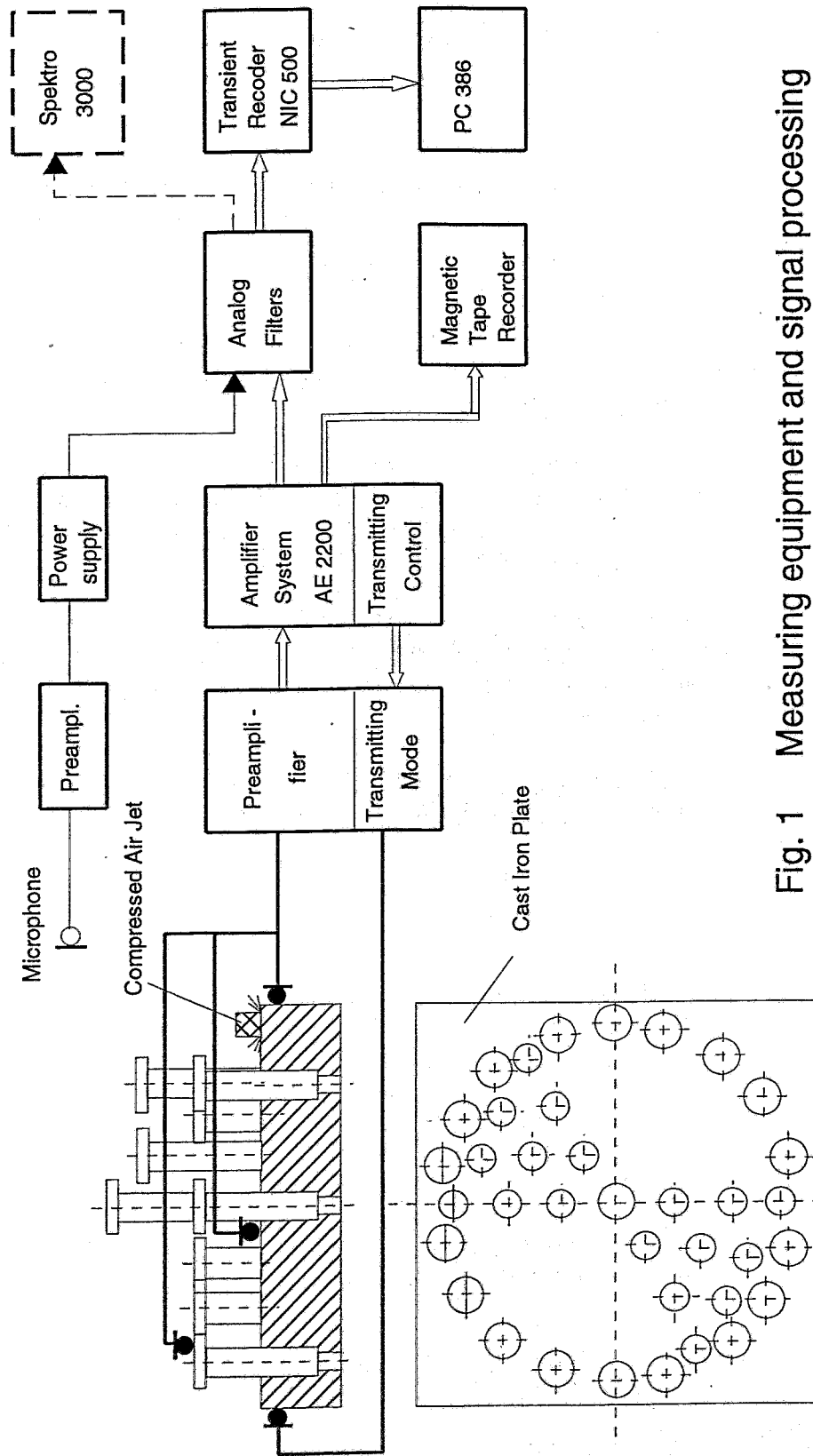


Fig. 1 Measuring equipment and signal processing system for leak simulation experiments

Collaboration with Institutions of Central and Eastern European Countries in the Field of Reactor Safety Research

F.-P. Weiß

General Aim:

The work of the Institute for Safety Research concerning the neutron kinetics and the thermohydraulics of accidental sequences at VVER-reactors, concerning the early recognition of mechanical damages and unwanted process states in the primary circuit of these reactors and also the research on material safety needs to be based on stable cooperations with scientific institutions and utilities in VVER operating countries of Central and Eastern Europe.

The project aims at the mobility of scientists from Eastern countries and is to render possible the stay of guest scientists at Rossendorf.

Results:

The project funded activities led to the following administrative frames

- * Agreement between the Kurchatov-Institute Moscow and the Research Center Rossendorf on numerical and experimental investigations of accident sequences in nuclear power plants with VVER-type reactors
- * Agreement between the All-Union Research Institute for Nuclear Power Plant Operation Electrogorsk (Russia) and the Research Center Rossendorf about experimental thermohydraulic investigations at the integral ISB facility for VVER-1000 reactors
- * Agreement between the AEKI Budapest and the Research Center Rossendorf on experimental and analytical researches in the field of VVER reactor safety
- * Settlement on material investigations using transmission electron microscopy and on the preparation of activated material specimens in the UJV Rez/Prague according to an order of the Research Center Rossendorf
- * Membership of the Research Center Rossendorf in the Eastern European Association of Atomic Energy Research (AER) for VVERs
- * Preparation of a collaboration with the Budker-Institute Novosibirsk on transport calculations for an intense 14 MeV neutron source for fusion material research
- * Settlements about the participation of the Research Center Rossendorf in vibration measurements at the Dukovany NPP (Czech Republic)
- * Agreement with the Saporoshje NPP on a common pilot project aiming at the creation of a NPP remote monitoring system in the Ukraine

Remote Monitoring of Eastern European Nuclear Power Plants

H. Carl

General Aim:

To cover their energy requirements Eastern European countries have to continue the contemporary nuclear power plant generation with certain degrees of safety deficit for an extended period of time.

The practical experience of the Tschernobyl reactor disaster results that an international information and early warning center is required during accidents in nuclear power plants because of the border-crossing transport of pollutants. The task of such a center should be the registration, evaluation and persecution of accident situations and of regional and border-crossing dangers resulting from the accident by the transport of the radioactive substances to protect effectively the population of the affected areas. For that purpose, a qualified selection of data which are transferred from the nuclear power plant to an national monitoring center is intended for supervising dangerous processes caused by the plant.

Within the bounds of the project the chances of success for the foundation, the spectrum of tasks and the costs expected of an international information and early warning center are investigated. Moreover the possibilities for the transfer of data from Eastern European nuclear power plants to the countries of European Community are discussed.

Results:

- The working capability of the project group was established, hardware conditons for the realization of the project are complete.
- A detailed project description was formulated and a first proposal for realization was elaborated.
- A first solution proposal for the structure of the remote monitoring system and for the tasks of its elements was acquired.
- A first contact with representatives of the International Atomic Energy Agency and the Ukrainian and Russian partner institutions was organized to present the project.

Feasibility Study on Pattern Recognition Based Methods to Determine the Structure and to Estimate Parameters of a Two-Phase Flow by Means of Active Ultrasonic Testing

H.-M. Prasser

General Aim:

Based on the state of art, the feasibility of an active ultrasonic method for the determination of the structure and the estimation of the parameters of a two-phase or two-component flow in steel tubes was studied. The state of art is characterised by several methods using pulse-echo techniques to identify single bubbles and plugs and Doppler measurements to determine the velocity. They are commonly feasible only in the case of low void fractions or simple flow structure. In order to minimise the hardware and to overcome the limitations of the existing approaches, i.e. to enable measurements at higher void fractions and complex flow situations, a combination of a continuous ultrasonic through-transmission with a pattern recognition method was investigated. An ultrasonic beam crossing the two-phase flow is modulated by the changing structure of the voids passing by and therefore the through-transmission signal must contain information about the parameters of the two-phase flow even if information about individual flow effects cannot be derived. It was supposed that a pattern recognition algorithm trained with signals obtained at known conditions is able to identify the set of the flow parameters (flow rates, void fraction etc.) in an unknown situation. The experimental tests were carried out at a vertical tube section of an air-water driven test loop with a diameter of 50 mm. Through-transmission signals were recorded at 41 different volume rate couples of water and air. The actual values of the volume rates were chosen to cover different flow pattern (bubble flow, plug flow, churn flow). Each measurement was repeated several times in order to test the pattern recognition method. The latter was trained by one of these realizations and tested by trying to identify the others.

Results:

The work on the feasibility study has been finished, the final report is under preparation. It was found that about 80-90 % (measuring time 10 sec) of the recorded signals were correctly classified even by a pattern recognition algorithm based on a comparatively easy binary pattern extraction method. In the case of the wrong classifications the majority of the detected parameter couples was located in the neighbourhood of correct points. If the measuring time is decreased the number of correct classifications is decreasing too. At a measuring time of for example 1 sec 55 % of the classifications are still correct, while 15 % are located in the immediate neighbourhood of the correct volume rate couples. It is expected that the accuracy can be increased by applying more sophisticated pattern recognition methods and by improvements of the hardware. For higher accuracy demands also the measuring time can be increased. On the other hand even measurements in transient flow situations seem to be possible. According to the positive results of the feasibility study the continuation of the work is planned.

Thermocapillary Migration of Bubbles and Drops

G. Gerbeth

General Aim:

From the literature it is known that up to the present no complete theoretical description of the time transient behaviour of the thermocapillary drop migration exists. The theoretical analysis of this problem is important for designing drop tower experiments, because there are only a few seconds for the development of the flow within and around a drop starting from the rest. Particularly analytical solution of subproblems are useful and necessary to obtain a better physical understanding of the flow development.

Results:

As a first approximation, the unsteady flow and the temperature field are calculated in the creeping flow limit, i.e. at low Reynolds and Marangoni numbers. In this case the non-linear terms in the equations of motion and energy transport are neglected. Therefore it is possible to find an analytical solution of the problem in the Laplace transform domain. The time dependence of the drop migration velocity was obtained by numerically transforming the results from the Laplace transform domain back to the time domain. In this way, an analytical description of the time transient behaviour of the thermocapillary drop migration up to the well-known final velocity given by Young, Goldstein and Block was obtained for the first time. The result of Young et al. for the final migration velocity contains two material parameters: the ratios of the dynamical viscosity and the heat conductivity. A set of five independent parameters is now necessary to describe completely the time transient behaviour in the creeping flow limit: the continuous phase Prandtl number and the ratios of density, dynamical viscosity, heat conductivity and thermal diffusivity, respectively. If the buoyancy is taken into account, the number of parameters increases to six.

To support the selection of appropriate substances for the drop experiments, a database software system containing a wide set of material data and its temperature dependencies was procured in December 1992. The numerical calculations were performed on a IBM RISC/6000 workstation.

Design of a Nuclear Power Plants Remote Monitoring System as a Tool for the State Regulatory Authority

H. Carl

General Aim:

It is imperative to ensure the safe operation of those Eastern European nuclear power plants requiring improvements and being well suited for improvements. User-independent remote monitoring of this nuclear power plants offers an effective and advantageous instrument of contributing towards satisfying this objective.

The remote monitoring is accomplished by means of downloading safety-relevant technological measurement and status information, accident instrumentation signals, additional reactor block system and component diagnostic measurement values, and dosimetric and meteorological values. After being preprocessed, tested, and pre-evaluated, the downloaded information is transferred to an user-independent National Monitoring Center by an appropriate remote data transfer means. There this information is monitored and evaluated in terms of compliance with limits and conditions of safe operation, and stored for statistical analyses for the purpose of developing an unbiased knowledge base and of providing meaningful data retrieval.

The project is aiming at the support of the construction of a technical system for improving the operational monitoring of Eastern European NNP's through the national regulatory authorities.

Research Centre Rossendorf Inc. and several Ukrainian partners have agreed to select unit Nr. 5 of the Saporoshje NPP with a Russian-type VVER-1000/320 reactor for the preparation of a pilot project.

Results:

- The organizational and technical base and the scientific capacity for the realization of the project were established.
- The Ukrainian attitudes and intentions to the project were analysed, and the first adjustment of the cooperation with the Saporoshje NPP, the Ukrainian State Regulatory Authority GosAtomNadsor Ukraina and the Institut of Nuclear Research (INR) of the Ukrainian Academy of Science in Kiev was carried out.
- The tasks to be treated by the Ukrainian and the German partners were discussed and defined.
- The task definition with regard to the safety functions of the system, application of nuclear laws, rules and recommendations was tuned, and the interfaces between the partners were determined.

Assistance of the Saxon State Ministry of Economy and Labour in the Promotion of Technologies for Economical Use of Energy and for Application of Renewable Energy

U. Rindelhardt

In the frame of a project in the Institut für Sicherheitsforschung a special group was established, that deals with the realization of the following promotion programmes of the Saxonian State Ministry of Economy and Labour:

- (1) Renewing of district heating systems
- (2) Modernizing of heating facilities in little enterprises
- (3) Rebuilding and modernizing of small water power stations
- (4) Rational use and application of renewable sources of energy
- (5) Installation of wind power stations.

By the help of these programmes the Ministry accelerates the reorganization of the energy supply structure in Saxony. These programmes will be an important contribution to fulfill requirements for reliable and priceworthy energy supply as well as for energy saving and environmental protection.

Especially the renewing and modernizing requirement of heating systems is considerable: Wastage at heat generation and distribution is enormous, measuring and regulating installations do not exist.

To secure the competitiveness of district heat and of little enterprises by means

- of the programme (1) in 1992 funds of 80.148 million DM were given for 120 projects of renewing and
- of the programme (2) funds of 8.184 million DM were given to support 491 projects of modernizing of heating systems.

In the programme (3) in 1992 funds of 2.155 million DM were divided among 19 projects of rebuilding and modernizing of small water power stations. Water power is the most important of renewable energy sources in Saxony.

By means of the programmes (4) and (5) 351 projects of rational use of energy and application of other renewable energy sources than water and wind power (funds of 1.809 million DM) and 2 projects of utilization of wind power (funds of 0.030 million DM) were supported.

The project group of energy funding in Forschungszentrum Rossendorf gave advices to the applicants of the programmes and prospective users of sophisticated energy facilities. Furthermore the group tested on the one hand, if the projects were worthy for promotion from technical and economical point of view and, on the other hand, if the funds were used in the planned and correct manner.

4. Lists of Publications, Conference Contributions, Seminary Lectures, and Research Reports

Publications

Aounallah, Y., Th. Bandurski, U. Grundmann u.a.
Bewertung der Methodik der Störfallanalysen im Sicherheitsbericht zum KKW
Beznau II
Bericht Nr. 7, Projekt STARS
Villingen / Schweiz, Paul Scherrer Institut, May 1992

Barz, H.-U., W. Bertram
Calculation of Neutron Fluence in the Region of the Pressure Vessel for the History of
Different Reactors by Using the Monte-Carlo-Method
Nucl. Engineering and Design 137(1992), 71

Block, F.R., R. Dittmer, G. Gerbeth
Bubble Detection in Liquid Metals
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Kiev / Ukraine, June 1992

Böhmert, J.
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Steam Atmosphere
Kerntechnik 57(1992), 55

Decker, B., U. Jahn, U. Rindelhardt, W. Vaaßen
The German 1000-Roof-Photovoltaic-Programme: System Design and Energy
Balance
Proc. 11th European Photovoltaic Conf.
Montreux / France, Oct. 1992

Dietze, K., G. Hüttel, E. Lehmann
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Specially Designed Neutronic Properties
Progress Report on Nuclear Data Research in the F.R. of Germany
NEANDC-Report, 1992

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Integral Test of FPND by Reactivity Measurements in Reactor Configurations with
Specially Designed Adjoint Spectra
Proc. Specialists Meeting on FPND, in Report JAERI
Tokai-Mura / Japan, May 1992

Häusler, R, J. Böhmert, F.J. Erbacher, L. Lübke, H. Schmidt, L. Wetzel
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Zircaloy-4-Hüllrohren
Proc. Jahrestagung Kerntechnik 1992
Karlsruhe, May 1992

Hoppe, D., R. Maletti
Improved Techniques of Analog and Digital Dynamic Compensation for Delayed Self-Powered Neutron Detectors
Nucl. Sci. Eng. 111 (1992), 433

Lenkey, G.B., Z. Major, H.-W. Viehrig
The Dynamic Calibration Problems in Instrumented Impact Testing
Proc. 9th Biennial European Conference on Fracture (ECF 9)
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Liewers, P., W. Schmitt, P. Schumann, F.-P. Weiß
Systematic Analysis of Noisy Signals in the Nuclear Reactor Noise Diagnosis of Abnormal Core Barrel Motion
Proc. 5th Symposium IMECO TC-10
Dresden, Sept. 1992

Nitschke, K., A. Thess, G. Gerbeth
Linear Stability of Marangoni-Hartmann Convection
Ed. H.J. Rath: Microgravity Fluid Dynamics
Berlin, Heidelberg, New York, Springer 1992, p. 285 - 296

Prasser, H.-M., L. Küllers, R. May
Conductivity Probes for Two-Phase Flow Pattern Determination During Emergency Core Cooling (ECC) Injection Experiments at the COCO Facility (PHDR)
Proc. 1st OECD (NEA) CSNI- Specialist Meeting on Instrumentation to Manage Severe Accidents
Cologne / Germany, March 1992

Rohde, U.
Modelling of Fuel Rod Behaviour and Heat Transfer in the Code FLOCAL for Reactivity Accident Analysis of Reactor Cores
Proceedings of the First Baltic Heat Transfer Conference (Gothenburg, Sweden, August 1991) sowie Transport Processes in Engineering 2: Recent Advances in Heat Transfer, Elsevier Publ., Amsterdam, 1992, p. 1214-1228

Schuster, G., M. Große
Neutronographische Untersuchungen zur Temperaturabhängigkeit der Besetzung der Sauerstoffpositionen in der $\text{YBa}_2\text{Cu}_{30}\text{O}_{(x-7)}$ -Elementarzelle
Proc. Jahrestagung der Deutschen Gesellschaft für Kristallwachstum und Kristallzüchtung
Dresden, March 1992

Schuster, G., M. Große, K. Teske, W. Anwand, K. Henkel, E.M. Nicht
Investigation on the Installation and the Bonding Behaviour of Oxygen Atoms in the $\text{YBa}_2\text{Cu}_{30}\text{O}_{(x-7)}$ Lattice
Proc. 4th European Conference on Solid State Chemistry
Dresden, Sept. 1992

Weiß, F.-P.
Safety Research at Eastern Germany's Rossendorf Center.
In: Nuclear Europe Worldscan, Journal of ENS, Topform'92 Czechoslovakia (1992) 9/10, p. 87

Lectures (Conference Contributions, Seminary Lectures)

Altstadt, E., F.-P. Weiß

Experimental Investigation and Numerical Simulation of Control Element Behaviour During Abnormal Core Barrel Motion at VVER-440 Type Reactors
IMORN-23 (Informal Meeting on Reactor Noise)
Nyköping / Schweden, June 1992

Block, F.R., R. Dittmer, G. Gerbeth

Bubble detection in liquid metals
Int. Conference on "MHD Processes to Protection of Environment"
Kiev / Ukraine, June 1992

Carl, H., P. Schumann, F.-P. Weiß

Overview of VVER-Reactor Design
NAPREM-Meeting (Nuclear Accident Prevention by Remote Monitoring)
Greenville / USA, May 1992

Carl, H.

NAPREM-Architecture and Tasks of its Elements
NAPREM-Meeting (Nuclear Accident Prevention by Remote Monitoring)
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11th European Photovoltaic Conf.
Montreux / France, Oct. 1992.

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Integral Test of FPND by Reactivity Measurements in Reactor Configurations with Specially Designed Adjoint Spectra
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Tokai-Mura / Japan, May 1992

Gerbeth, G., A. Alemany

Magnetohydrodynamic Flow Around a Circular Cylinder
IUTAM Symposium
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Krepper, E.

Nachrechnung eines Ablase-Experimentes mit den thermohydraulischen Störfallprogrammen ATHLET und RELAP
1. ATHLET-Anwederseminar
GRS Garching, May 1992

Kumpf, H.
Wirkungsweise und Anwendungsmöglichkeiten einer Plasmaneutronenquelle
Technische Universität Dresden, Sept. 1992

Kumpf, H., K. Noack
Neutronic Problems of a Compact 14 MeV Plasma Neutron Source
International Conference on Open Plasma Confinement Systems for Fusion
Novosibirsk, June 1993

Lenkey, G.B., Z. Major, H.-W. Viehrig
The Dynamic Calibration Problems in Instrumented Impact Testing
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Investigation of Cross-Flow Induced Tube Bundle Vibration in Heat Exchangers - Interpretation as Synergetic System
IMORN-23 (Informal Meeting on Reactor Noise)
Nyköping / Sweden, June 1992

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Bremen, Sept. 1992

Prasser, H.-M., L. Küllers, R. May
Conductivity Probes for Two-Phase Flow Pattern Determination During Emergency Core Cooling (ECC) Injection Experiments at the COCO Facility (PHDR)
1st OECD (NEA) CSNI- Specialist's Meeting on Instrumentation to Manage Severe Accidents
Cologne / Germany, March 1992

Schumann, P.
Spezielle Signalanalyseverfahren zur Schadensfrüherkennung und Diagnostik an WWER-Druckwasser-Reaktoren
Oberseminar im Kurt-Risch-Institut für Dynamik, Schall- und Meßtechnik der Technischen Universität Hannover
Hannover / Germany, Febr. 1992

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Monitoring Tasks and Selection of Operational and Diagnostic Signals for the NAPREM-System
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Schuster, G., M. Große
Neutronographische Untersuchungen zur Temperaturabhängigkeit der Besetzung der Sauerstoffpositionen in der $\text{YBa}_2\text{Cu}_3\text{O}_{(x-7)}$ -Elementarzelle
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Investigation on the Installation and the Bonding Behaviour of Oxygen Atoms in the $\text{YBa}_2\text{Cu}_3\text{O}_{(x-7)}$ Lattice
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5th Meeting of CRP-Phase III
Balatonfüred / Hungary, Sept. 1992

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Öffentlicher Vortrag im Rahmen des Berufungsverfahren auf die Direktorenstelle des Institutes für Sicherheitsforschung
Rossendorf, Nov. 1992

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Bergner, F.
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FZR-Report, FZR 92-07, May 1992

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Machbarkeitsstudie in 2 Bänden
Hrsg. Forschungszentrum Rossendorf e.V. und E-Systems Inc., Texas, USA
Oct. 1992

Grundmann, U., U. Rohde
The Code DYN3D/M2 for the Calculation of Reactivity Initiated Transients in Light Water Reactors with Hexagonal Fuel Elements - Code Manual and Input Data Description
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Solarthermie in Sachsen - PC-Datei SOSA

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Nr. 1.

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IBM 3090

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Nachrechnung eines Abblaseexperimentes mit den thermohydraulischen Störfallpro-
grammen ATHLET und RELAP

FZR-Bericht, FSS 08/92, May 1992

5. Lists of Institute Seminars and Workshops

Seminars

1. Dr. F.-P. Weiß
Vorstellung des Institutes für Sicherheitsforschung und dessen Abteilungen,
- Perspektiven - Trends
08 Jan. 1992
2. Dr. Kumpf
Eine kompakte Plasmaneutronenquelle für die Fusionsmaterialforschung
20 May 1992
3. Dr. Noack
Plasmaphysikalische Grundlagen einer Plasmaneutronenquelle
10 June 1992
4. Dr. F.-P. Weiß
Überblick über das Institut für Sicherheitsforschung
02 Sept. 1992
5. Dr. Gerbeth
Forschungsarbeiten zur Magnetohydrodynamik im FZR
26 Oct. 1992
6. Dr. Rindelhardt
Erneuerbare Energien - Projektarbeit im Auftrag des SMWA
09 Nov. 1992
7. Dr. Thibault
Der magnetohydrodynamische Schiffsantrieb
17 Nov 1992
8. Dr. Ivanov, Novosibirsk
INP's approach to developing a high flux 14 MeV neutron source for fusion technology
research
Dr. Krasnoperov, St. Petersburg
IN-1 neutron source design
17 Nov. 1992
9. Dr. Krasnoperov, St. Petersburg / Dr. Schreiner, Novosibirsk
1. Some engineering problems of IN-1 design
2. Der Stand der Arbeiten zum Wasserstoffprototyp einer Plasmaneutronenquelle
(mit Übersetzung aus dem Russischen)
19 Nov. 1992
10. Dr. Tsibulja, Obninsk
Einschätzung und Verarbeitung von Neutronen- und Gammadaten im FEI Obninsk
11 Dec. 1992

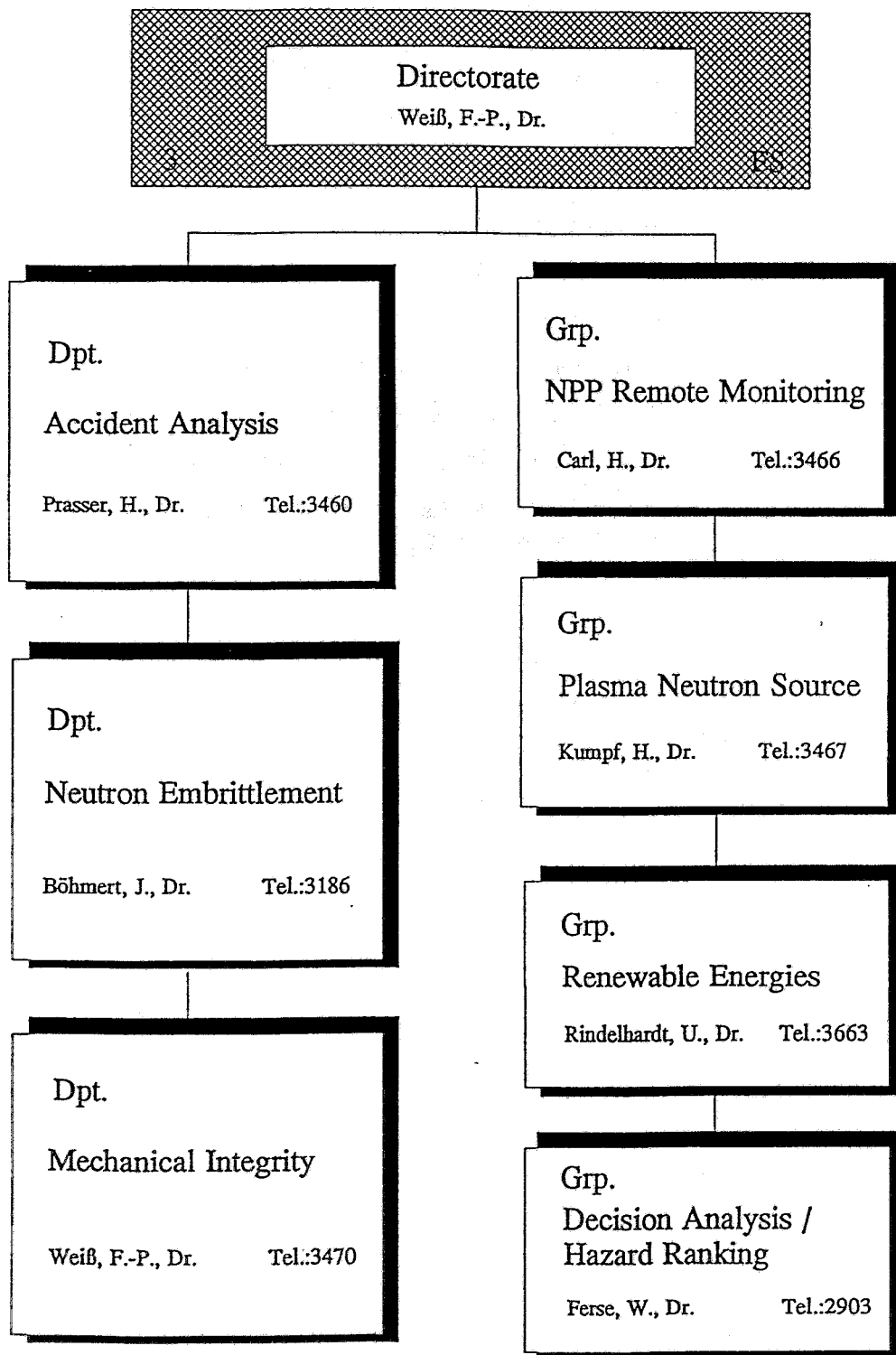
Workshops

1. Workshop on Safety Research for VVER Reactors, Bilateral Cooperation UJV Rez/FZR, 24-25 Nov. 1992
2. NAPREM-Workshop "Remote Monitoring", 24-25 Nov. 1992
3. NAPREM-Workshop "Remote Monitoring of Middle and East European NPP", 25-26 June 1992
4. NAPREM-Workshop "Remote Monitoring of Middle and East European NPP", 06-08 Oct. 1992

Visiting Scientists

Dr. S. von Krosigk,	Bundesministerium für Forschung und Technologie, Bonn
Prof. G. Wolf,	Forschungszentrum Jülich
Dr. J. Runkel,	Universität Hannover
Prof. D. Stegemann,	Universität Hannover
Dr. V. Aksenov,	JINR Dubna, Russia
Prof. Ullrich,	Fachhochschule Münster
Prof. Karwat,	TU München
Dr. Krasnoperov,	Efremov-Institut St. Petersburg, Russia
Dr. Schreiner, K.,	Budker Institut für Kernphysik Novosibirsk, Russia
Dr. Ivanov, A.,	Budker Institut für Kernphysik Novosibirsk, Russia
Artemschuk, V.V.	Chief of Technical and Documentation Energodar, NPP Saporoshe, Ukraine

6. Structure and Personnel



Personnel (List Updating: December 1992)

Scientific Staff	Technical Staff
<p>Altstadt, Eberhard Dr. Barz, Hansulrich Dr. Bergmann, Uwe Beyer, Matthias¹⁾ Böhmer, Bertram Böhmert, Jürgen Dr. Carl, Helmar Dr.¹⁾ Dietze, Klaus Dr.¹⁾ Enkelmann, Wolfgang Dr.¹⁾ Ferse, Wolfgang Dr. Galindo, Vladimir Dr.¹⁾ Gerbeth, Günter Dr. Große, Mirco¹⁾ Grundmann, Ulrich Dr.¹⁾ Grunwald, Gerhard Dr.¹⁾ Hensel, Frank¹⁾ Hermsdorf, Dietrich Dr.¹⁾ Hessel, Günter Höfer, Martina¹⁾ Krahl, Steffen¹⁾ Krepper, Eckhard Dr. Kumpf, Hermann Dr. Langer, Lutz¹⁾ Lotzmann, Roland¹⁾ Lucas, Dirk Dr.¹⁾ Maletti, Rainer Dr.¹⁾ Mittag, Siegfried Dr. Mutschke, Gerd¹⁾ Naehring, Friedrich Dr. Noack, Klaus Dr. Richter, Holger (Doktorand) Rindelhardt, Udo Dr. Rohde, Ulrich Dr. Scheffler, Michael¹⁾ Schütz, Peter¹⁾ Schmitt, Wilfried Dr. Schumann, Peter Dr.¹⁾ Seidel, André¹⁾ Teichmann, Günther¹⁾ Viehrig, Hans-Werner Weiß, Frank-Peter Dr. Werner, Matthias Dr.¹⁾ Zippe, Winfried Dr. Zschau, Jochen Dr.¹⁾</p>	<p>Baldauf, Dieter Behrens, Sieglinde Blumentritt, Thea Borchardt, Steffen Eckert, Sven Eichhorn, Christine Elert, Edith Fischer, Manfred¹⁾ Futterschneider, Hein Gebel, Margitta¹⁾ Heinze, Gerda Kaule, Christian Kunadt, Heiko Lang, Dorothea¹⁾ Leonhardt, Wolf-Dietrich Leuner, Bernd Leuschke, Grit¹⁾ Losinski, Claudia Matthes, Wilfried Otto, Gerlind Richter, Annett Richter, Henry Richter, Joachim Richter, Karl-Heinz Rott, Sonja Russig, Heiko Seidler, Christa Skorupa, Ulrich Stephan, Ingrid Dr. Tamme, Günter Tamme, Marko Webersinke, Wolfgang Weiß, Rainer Willkomm, Heike Wolrab, Günter Zimmermann, Wilfried</p>

¹⁾ financed by project sponsoring

7. Glossary

AEKI	-	Atomic Energy Research Institute Budapest
AER	-	Atomic Energy Research Association of the Eastern European Countries
BMFT	-	Bundesministerium für Forschung und Technologie
BMU	-	Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit
CB	-	Core barrel
CE	-	Control element
DARA	-	Deutsche Agentur für Raumfahrtangelegenheiten GmbH
DFG	-	Deutsche Forschungsgemeinschaft e. V.
Dpt.	-	Department
DVM	-	Deutscher Verband für Materialforschung und -prüfung e. V.
ENIS	-	All-Union Research Institute for Nuclear Power Plant Operation Electrogorsk (Russia)
FE	-	Finite Elements
FZR	-	Forschungszentrum Rossendorf e. V.
Grp.	-	Group
GRS	-	Gesellschaft für Anlagen- und Reaktorsicherheit
HASYLAB	-	Hamburger Synchrotronstrahlungslabor
IAEA	-	International Atomic Energy Agency
IFE	-	Institut für Energetik Leipzig
IINR	-	International Institute of Nuclear Research Dubna
IKE	-	Institut für Kernenergetik der TU Stuttgart
INR	-	Institute of Nuclear Research of the Ukrainian Academy of Science
ISFH	-	Institut für Solarenergieforschung Hannover
KFA Jülich	-	Kernforschungszentrum Jülich GmbH
KFK	-	Kernforschungszentrum Karlsruhe
KFÜ	-	Kernreaktorfernüberwachung
LWR	-	Light water reactor
MHD	-	Magnetohydrodynamics
MOEL	-	Mittel- und osteuropäische Länder (Central and Easteuropean countries)
NAPREM	-	Nuclear Accident Prevention by Remote Monitoring
NPP	-	Nuclear Power Plant
PV	-	Photovoltaic
RPV	-	Reactor pressure vessel
RWTH	-	Rheinisch-Westfälische Technische Hochschule Aachen
SEM	-	Scanning electron microscope
SMWA	-	Sächsisches Staatsministerium für Wirtschaft und Arbeit
SMWK	-	Sächsisches Staatsministerium für Wissenschaft und Kunst
TU	-	University of Technology
TÜV	-	Technischer Überwachungsverein e. V.
UJV	-	Nuclear Research Institute UJV Rez
VKTA	-	Verein für Kernverfahrenstechnik und Analytik Rossendorf e. V.
WTZ	-	Wissenschaftlich-Technische Zusammenarbeit (Science & Technology Cooperation)