Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH

Annual Report 2008

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The illustrations contained in the Annual Report 2008 are listed in the following. In the main section, they have been reduced in size and are clickable (plus sign) (⊕) for enlargement. A mouseclick on the symbol (←) in the annex will lead you to the reference in the text of the PDF.

Lothar Hahn
Technical and Scientific Director

Hans J. Steinhauer
Commercial and Legal Director

Dear reader,

When you flip open our Annual Report 2008, you may do this for various reasons: Maybe you have known GRS for many years and would like to inform yourself about the development of our company and the main topics of our activities in 2008. If you have consulted previous Annual Reports of GRS for this purpose, you will have noticed the new format of the one in front of you: From this edition on we will make our Annual Report available on CD-ROM. The full text search for terms, bigger graphics in a better resolution and the inclusion of movies or animations are only some of the advantages of this medium which we would like to offer you in the future. Perhaps you still do not know GRS and would like to gain a first impression of our company by reading our Annual Report. In this case we would like to call your attention to the abstracts at the beginning of each chapter. With these abstracts you can get a general idea of our objectives and the contents of our fields of activity. The reason for your reading this Annual Report notwithstanding, we would like to provide comprehensive information about GRS. Therefore, it is our intention to answer, among others, the following questions:

What have been the major topics of GRS's technical activities?

GRS's central fields of activity are reactor safety, repository safety as well as radiation protection. As far as these areas are concerned, GRS is the expert organisation of the Federal Government and advises it on all safety-relevant problems. GRS works on both reactor safety and repository research and in the field of analysing operational practice. When making safety-related assessments, the findings gained from both fields of work are combined. In doing so, the provision of interdisciplinary knowledge, advanced analysis methods and qualified data to assess the safety of technical facilities represents a particular challenge.

GRS's reactor safety analyses refer to concrete evaluative problems and form the technical basis for regulatory supervision and licensing. To that end, GRS evaluates national and international operating experience, but also carries out analyses on current safety-related issues on its own. These analyses are related to the behaviour of the plant or its technical systems during power plant operation or to actual or theoretically assumed, safety-relevant incidents.

In this report, we will introduce GRS's activities relating to e.g. the evaluation of recent findings on the integrity of pressurised components, the development of simulation tools for the analysis of boron dilution accidents, and the monitoring of international developments on the defence-in-depth concept to control voltage transients. Subject of the project on the last-mentioned topic was the analysis of the accident at the Swedish Forsmark 1 plant in 2006. Not only does this project show the importance of the analysis of such accidents for the improvement of the safety of other plants, but also it makes clear that we attach great importance to international co-operation – in this case within the working group of the Nuclear Energy Agency of OECD.

GRS's reactor safety research plays a key role in the further development of the state-of-the-art in science and technology in Germany. With the development of analysis methods – in particular by providing validated calculation codes for the simulation of transients and incident and accident sequences – we make important contributions to the solution of current and future safety-relevant problems. GRS's research activities refer to, among other things, reactor physics, the thermal-hydraulics in the cooling circuit, the reliability of reactor components, and potential accident sequences in the so-called containment system of a nuclear power plant.

In this Annual Report, we will present, for example, a project in the field of reactor physics during which GRS experts have dealt with the calculation of the power distribution in the cores of power reactors – these are reactors used for power generation. Knowing this power distribution is an essential prerequisite for the safety analysis of such reactors. For the calculation thereof, so-called nuclear basic data are used, the validity of which are usually verified by means of experiments in which conditions different from those in power reactors prevail. In the project presented, GRS has analysed which impacts - if any - this has on the accuracy of the power calculations.

In the context of another project, you will learn how GRS develops methods with which the probability of leaks and breaks of pressurised reactor components – of pipelines, for example – can be determined. These methods represent a valuable supplement to the so-called probabilistic safety analyses (PSA) which have to be conducted for each nuclear power plant as part of the periodic safety reviews.

GRS's repository safety research is application-oriented research on the long-term safety case of repositories for radioactive waste. One integral element of the »safety case« decisive for this demonstration is the development of coupled thermal-hydraulic-mechanical-chemical models and visualisation models with which the complex processes in a repository can be analysed and clearly represented.

One of the projects carried out in this context will be introduced in this Annual Report. In this project, GRS experts examined, by comparing with the results of experiments, how accurately selected calculation models can simulate the transport of certain pollutants through geological strata. In another project, GRS examined the stability of so-called bentonites, materials which can be used in repositories to create barriers to seal the vaults, for example.

Scientific and technical problems gain more and more importance, also in the area of radiation and environmental protection. Here, GRS's activities range from the interim storage and final disposal of spent fuel assemblies or other radioactive waste, the decommissioning of nuclear reactors and the transport of radioactive substances to the radiological emergency management and radioecology. In addition to the respective research activities, GRS also supports supervising and licensing authorities in these areas by way of providing expert analyses and scientific consulting.

One example of the manifold issues GRS is dealing with in the field of radiation and environmental protection will be given in the article on the »Modelling of the atmospheric dispersion in case of accidents«: In this project, calculation codes were optimised with which the dispersion of radioactive substances in the air after an accident-induced release can be simulated. The development work presented here made it possible to link these codes with a new, more powerful weather forecast system of the German National Meteorological Service (*Deutscher Wetterdienst* - DWD) and to thus improve the accuracy of the dispersion calculations.

How important are international activities for GRS?

In our Annual Report 2006/2007, we had already announced that GRS was striving for an even stronger commitment at international level, in particular in respect of supporting foreign authorities as well as the European network of expert organisations. This resolution we have put into practice:

Our long-standing co-operation with the Argentine licensing authority Autoridad Regulatoria Nuclear (ARN), for example, has received a fresh impetus due to the resumption of the work on the Atucha II plant. Furthermore, GRS successfully participated in an invitation to tender by the British Health and Safety Executive (HSE) and will play a key role in the implementation of the so-called Generic Design Assessment (GDA). Overall, the reporting period was characterised by an increasing international demand.

The growing international interest in expert services in the field of nuclear safety is also reflected in the first expansion of the European Technical Safety Organisations Network (ETSON) in 2008. GRS had founded the network together with its French and Belgian partners IRSN and Bel V in 2006. The Finnish and Czech expert organisations VTT and UJV have also been members of the network since the autumn of 2008. In view of the expressions of interest at hand we assume that the number of network partners will keep increasing in the years to come.

How does GRS promote innovations and ideas?

With the Future Lab, GRS installed a new instrument to develop, co-ordinate and create ideas and visions in 2008. In the Future Lab, younger staff members - in a team with experienced colleagues - develop ideas for new products, but also for internal processes, work methods or optimised further training and qualification. After the ideas have been evaluated, the teams prepare concepts for the implementation of selected ideas.

One of the first results of the Future Lab was the use of the GRS intranet for improving the employee suggestion system. On a newly created intranet page, the so-called pool of ideas (»*Ideenpool*«), staff members can not only make suggestions, but also discuss them interactively with others.

What is GRS doing in the field of training and junior staff development?

Like all organisations, authorities and companies working in the field of nuclear safety, GRS also faces the challenge of having to compensate the retirement of experienced experts despite the lack of specifically qualified junior staff. To ensure that we win the experts of tomorrow for our work today and that we can train them correspondingly, we increased our activities relating to junior staff development in 2008.

Thus, GRS arranged, together with its ETSON partners, the first ETSON Summer School at the Garching location in August 2008. During this Summer School, experienced experts of GRS, IRSN and Bel V trained numerous young staff members of our ETSON partners and imparted special knowledge as well as recent developments in the area of reactor safety assessment. In the future, the Summer School will take place annually – in 2009 the venue will be Cadarache in France.

To spark the students' and pupils' interest in natural and engineering sciences in general and in the work on topics relating to nuclear safety in particular, we laid the foundations for an even more intensive co-operation with universities and schools last year: By the end of 2008, GRS could already record a considerable increase in the co-operation with technically oriented universities, among them RWTH Aachen University and TU Dresden. Furthermore, we have launched an initiative for the intensive co-operation with selected grammar schools at all locations. The co-operation with further universities and grammar schools is under way.

How have communication and corporate culture within GRS developed?

In recent years, GRS has hired many new colleagues to compensate the retirement of experienced experts. This posed new challenges to the entire GRS coporate culture and, above all, to the communication within GRS. We tackled these challenges and invested a lot of work in the improvement of the internal communication structures in 2008, too. And the result of our efforts is quite impressive.

The GRS intranet – our »GRS Portal« – has been developing into the central platform for the internal exchange of information. The information in our so-called »Yellow Pages« provided by our colleagues makes it easier for our new staff mem-

bers to quickly find their way in their new circle of colleagues. In addition, the GRS Portal has established itself as a valuable support in specialised work: Project portals as well as team and competence pages facilitate the access to technical information or project data and make it easier to find the right expert for one's own technical questions.

The communication forum, a regular professional exchange with colleagues which had been installed in Cologne in 2007 already, was set up at our Brunswick location, too, last year. Furthermore, we have created the GRS Staff Dialogue. Here, our employees regularly get together with our general management to openly discuss all issues relating to our work and the company GRS.

In addition to this internal exchange, GRS also attaches great importance to external communication. As a non-profit organisation, the activities of which are essentially financed by public funds, and as expert organisation of the Federal Government, we have the ambition to be a source of substantiated information on issues of nuclear safety for both the expert community and the general public. Therefore, we have been working on continuously improving our offer of information. At the beginning of 2008, this endeavour was reflected in the relaunch of our website which can be accessed via www.grs.de. The visits to our site which have since continuously increased confirm that with our efforts we have adopted the right approach.

You see, a lot has happened at GRS! We hope that this overview has made you somewhat curious and we trust that you will have an informative read.

Lothar Hahn   Hans J. Steinhauer

Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH
Cologne, 30 September 2008

2. Organisation and economic development

Veit Watermeyer

The GRS is an independent, scientific non-profit expert organisation working in the field of nuclear safety, radiation protection and waste management and is committed to the general public. It is GRS's function to maintain and further develop skills, to scientifically assess situations on the basis of the knowledge nationally and internationally available in the fields of nuclear safety, radiation protection and waste management and according to the state-of-the-art in science and technology and to thus to contribute to its further development. In the mentioned fields of expertise, GRS constitutes an internationally acknowledged centre of competence.

Organisation

Group structure

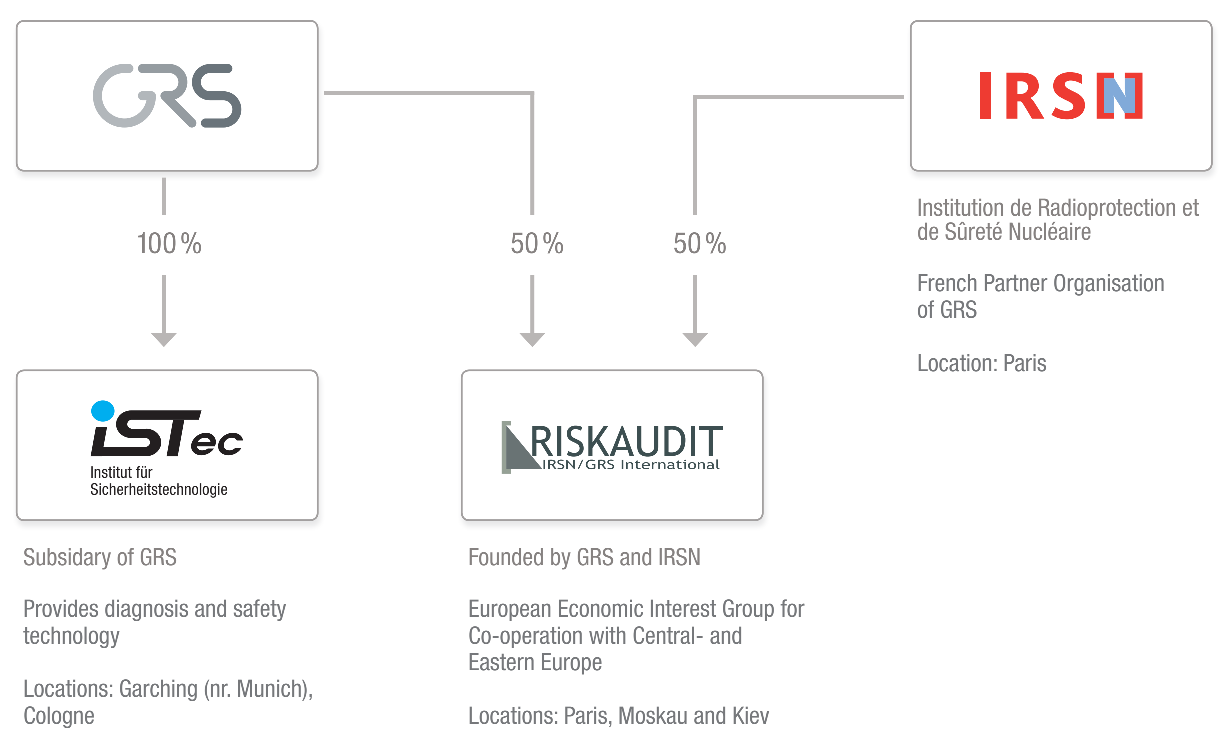
Pursuant to § 267 (3) of the German Commercial Code (Handelsgesetzbuch – HGB), GRS is a large corporation. As a wholly owned subsidiary of GRS, the Institute for Safety Technology (ISTec) GmbH (*Institut für Sicherheitstechnologie GmbH* – ISTec) is included in GRS's consolidated financial statement.

There is no obligation to make a capital contribution for the European Economic Interest Grouping (EEIG) RISKAUDIT which was founded together with our French partner organisation Institut de Radioprotection et de Sûreté Nucléaire (IRSN).

Subsidiaries and interest

Institute for Safety Technology (ISTec) GmbH. The Institute for Safety Technology (ISTec) GmbH is a subsidiary of GRS. Its headquarters is in Garching. ISTec ranks among the leading providers of diagnostic and safety technology. It pools the experience of many decades in research and development and the implementation and testing of advanced safety technologies. ISTec offers advisory and inspection services on the introduction of new technologies, comprehensive service related to the operation and use as well as its own integral technical solutions which comprise both systems for damage diagnosis and DP-supported monitoring systems.

GRS and its subsidaries ISTec, RISKAUDIT, plus its partner IRSN



RISKAUDIT IRSN/GRS International. The European Economic Interest Grouping RISKAUDIT was jointly founded by GRS and its French partner organisation IRSN; its headquarters is in Paris. RISKAUDIT is co-ordinator of safety-oriented projects in Eastern Europe of the EU and the European Bank for Reconstruction and Development (EBRD) and is representative in the Technical Safety Organisation Group (TSOG) established by the European Commission. For the co-operation of GRS and IRSN with Eastern Europe, RISKAUDIT operates joint offices in Moscow and Kiev.

2 Organisation and economic development CONTENT

Organisation of GRS

General Management

L. Hahn — H. J. Steinhauer

MANAGEMENT STAFF

Scientific Strategy	Public Relations	IT-Management	Knowledge Management
E. Kersting	S. Dokter	H. R. Seel	Dr. D. Beraha

Reactor Safety Research V. Teschendorff	Reactor Safety Analyses H. Liemersdorf	Final Repository Safety Research T. Rothfuchs	Radiation and Environmental Protection Dr. G. Pretzsch	Projects and Int. Programmes U. Erven	Central Services V. Watermeyer	Project Mgmt. Agency/Authority Support R. Zipper
Barrier Effectiveness Dr. M. Sonnenkalb	**Plant Engineering** Dr. R. Stück	**Safety Analyses** Dr. J. Mönig	**Nuclear Fuel** Dr. B. Gmal	**Project Management** Dr. U. Holzhauer	**Finance** V. Watermeyer	**Research Management** R. Zipper
Cooling Circuit Dr. H. Glaeser	**Plant Reliability** C. Verstegen	**Process Analyses** Dr. H.-J. Herbert	**Radiation Protection** H. Thielen	**International Programmes** Dr. H. Teske	**Human Resources and Legal Affairs** M. Fillbrandt	**Project Administration** H.-U. Felder
Core Behaviour Dr. A. Pautz	**Plant Behaviour** W. Pointner		**Final Storage** Dr. K. Fischer-Appelt	**Interdisciplinary Projects** Dr. M. Mertins	**Communication** S. Dokter	
					Local Administration G. Diepolder J. Hanrieder S. Krämer	

Moscow Technical Office*)	Kiev Technical Office*)
K. Shastin	M. Chouha

*) jointly with IRSN/RISKAUDIT

As at: December 2008

2 Organisation and economic development CONTENT

Company locations of GRS

Cologne

The offices in Cologne are the headquarters of GRS and also of the general management. All GRS divisions except the Final Repository Safety Research Division are represented there. The technical emphasis is on reactor safety analyses as well as radiation and environmental protection. In addition, the Projects Division and the Central Services Division as well as the Project Management Agency Reactor Safety Research are directed from Cologne.

Garching

The Reactor Safety Research Division is the biggest field of work and is managed in Garching near Munich. Here, among other things, GRS develops and verifies programmes and methods with which incidents and accidents in nuclear power plants can be simulated. Further fields of activities are the reactor safety analyses, radiation and environmental protection and project management. The offices of this business unit are located around the research institutes on the TU Munich campus and in immediate vicinity of the research reactor FRM-2.

Berlin

The key activities of the employees working at the offices in Berlin are related to international activities, in particular for Central and Eastern Europe. Here, experts of different disciplines work in close co-operation with foreign nuclear authorities and their expert organisations with the aim to improve the safety of nuclear facilities worldwide. In this context, the technical offices in Moscow and Kiev which are operated by GRS and IRSN together with their joint subsidiary RISKAUDIT play an important role. The Radiation and Environmental Protection Division as well as the International Programmes Department are managed from Berlin.

Braunschweig

At the Braunschweig offices, Final Repository Safety Research is the main field of activity. Here, the employees develop methods and procedures which are required for building a long-term safety case for repositories for hazardous waste in geological formations. The division is divided into the two departments Safety Analyses and Process Analyses and also has a geo-scientific laboratory at its disposal.

Shareholders and executive bodies of GRS

Shareholders	Executive bodies
Federal Republic of Germany (46 %)	Meeting of shareholders
Free State of Bavaria (4 %)	Supervisory board (12 members)
Land of North Rhine-Westphalia (4 %)	Chairman: Parlamentary State Secretary Michael Müller
Technical Inspection Organisations (TÜV) and Germanischer Lloyd (together 46 %)	Vice-chairman: Prof. Dr.-Ing. Bruno O. Braun
	Managing Directors: Dipl.-Phys. Lothar Hahn, Hans J. Steinhauer

Customers

As expert organisation of the Federal Government, GRS makes technical and scientific expertise available to the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (*Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit* – BMU) as well as to the Federal Office for Radiation Protection (*Bundesamt für Strahlenschutz* – BfS) at all times and provides qualified staff and adequate technical equipment corresponding to the long-term requirements. It conducts research of its own in the fields of work relevant for the ministry.

By order of the Federal Ministry of Economics and Technology (*Bundesministerium für Wirtschaft und Technologie* – BMWi), GRS supervises the reactor and fundamental safety research and carries out own research and development activities. By further developing the state-of-the-art in science and technology, GRS contributes to an improved analyses and assessment of the safety and risks of technical facilities and processes. It thus ensures its independent expertise and explanatory force.

GRS also works on behalf of other national and international organisations which intend to thus make use of GRS's knowledge, methods and experience in order to assume their responsibility for safety and environmental protection.

For the work carried out within the scope of own research activities, GRS receives real and non-taxable subsidies.

Staff

GRS employs 419 persons approx. 300 of which are technical and scientific employees coming from the following fields of specialty: physics, mechanical engineering, process engineering, structural engineering, geotechnics, electrical engineering, nuclear technology, meteorology, chemistry, geochemistry, biology, mathematics and computer science as well as law and business administration.

Business development

Overall statement on the economic situation

The year 2008 was again a very successful year for GRS. In 2008, too, the integral capacity of the company was reached at the beginning of the year already and could be increased, due to a growing number of orders, to 121 % until the end of the year. The higher budget appropriations of the BMWi for reactor safety and repository research have led to an outstanding workload in the respective research areas in 2008 as well. With the BMU, a gross contract volume of €24.4 million could be achieved.

General conditions

General economic environment. The economic environment has considerably worsened because of the financial crisis. The Ifo Business Climate Index which is considered the most important indicator of the German economy dropped to 82.6 points in December 2008 and has been continuing this downward trend. According to preliminary figures by the Federal Statistical Office, the German gross domestic product increased by only 1.3 % in 2008. Nevertheless, GRS was not directly affected by this negative development.

Political and legal environment. In view of the changed basic conditions regarding the peaceful use of nuclear energy, the complexity of the problems as well as the value change in the working environment and in society, GRS still faces the challenge to maintain and expand its technical competence by taking adequate human resources measures.

The agreement between the Federal Government and the power supply companies of 14 June 2000 on the use of nuclear energy expressly specifies that GRS's independence and qualification remain assured. In the coalition agreement of 2005 it is stipulated that the safe operation of nuclear power plants is of highest priority and that the research on the safe operation of nuclear power plants shall be continued and expanded and that the solution for a safe final disposal of radioactive waste shall be tackled in an efficient and goal-oriented manner.

Financial position and results of operations

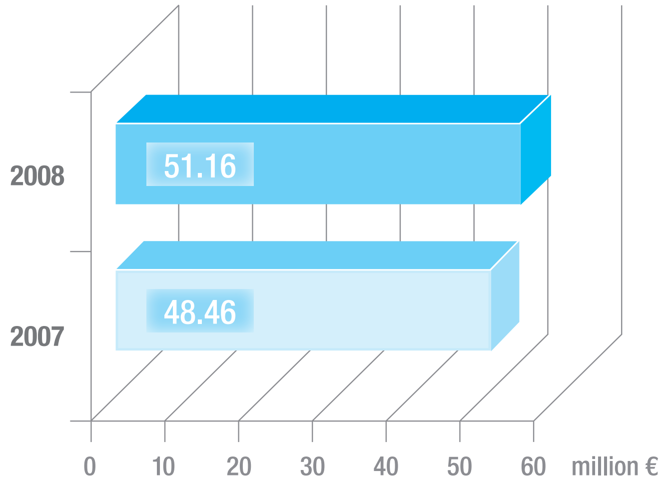
Net assets and equity. As of balance sheet date, GRS has a sound assets and capital structure. In the reporting year, GRS's balance sheet total increased by €1.31 million or 3.1 % to €43.07 million. GRS's fixed assets increased by €529,000 or 8.1 %.

The receivables from affiliated companies concern ISTec and receivables from companies with which GRS is linked by virtue of a participating interest concern RISKAUDIT. The actuarial reserve for the AHV company pension scheme amounts to €13,877 thousand (previous year: €14,265 thousand) and is included in the other assets. This is a long-term actuarial reserve.

Due to the annual surplus 2008, the equity has increased by €2.27 million and now amounts to €14.75 million. The equity ratio has increased by 4.4 % to 34.3 % (previous year: 29.9 %) because of the increased equity.

The long-term borrowed capital shows an increase of €68 thousand or 0.4 % to €16.73 million. This primarily results from the increase in pension provisions. As in the previous year, the reduction of liabilities to credit institutions by €87 thousand had the opposite effect.

As far as short- and medium-term borrowed capital is concerned, which dropped in the reporting



1 EARNING SITUATION
Turnover and subsidies

period by €1,032 thousand or 8.2 %, a reduction by €978 thousand can be recorded with received advance payments on orders.

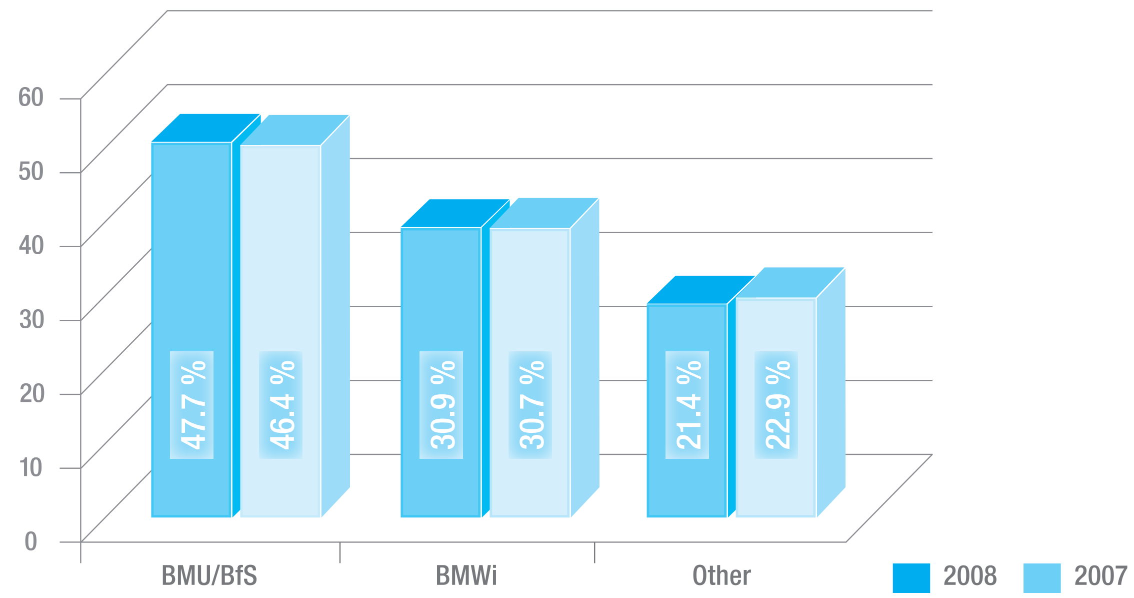
The fixed assets are fully covered by equity. The liabilities to credit institutions amount to a small extent of only 3.2 % of the balance sheet total (previous year: 3.8 %). The inventories reported in the balance sheet are covered by advance payment.

Financial position. During the course of a year, GRS's liquidity situation is decisively determined by the rhythm of accounting and payment receipt, respectively. With almost steadily accruing operating expenses, the instalments for public orders are made for the respective quarter in the middle of the quarter as specified in the contracts. Thus, the highest demands in liquidity which lead to short-term higher borrowing occur shortly prior to the middle of the quarter. Liquid funds are available at the end of the quarter and in particular at the end of the year to cover the expenses until there is an accrual of new liquid funds.

On average, the liquid position of the company was good during the year. Only in November, a short-term external financing was required. The line of credit granted by our principal bank entirely covered this demand for credit.

Cash and cash equivalents decreased by €1,055 thousand to €8,623 thousand. In the individual financial statement, the cash and cash equivalents reduced by €1,088 thousand to €8,502 thousand. The reduction of the cash and cash equivalents in the individual and consolidated financial statement is caused in particular by the outflow of funds due to investment activities.

Earnings situation. GRS managed in the preceding financial year to increase the volume of turnover and the volume of subsidy from €48.46 million to €51.16 million (see Fig. 1 »EARNING SITUATION«).

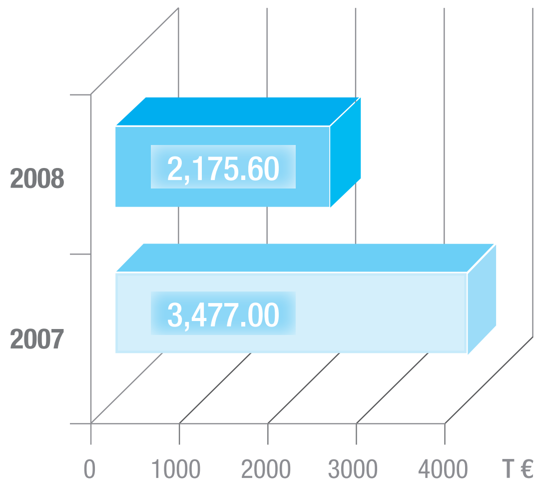


2 EARNING SITUATION
Distribution of turnover from R&D by customers

The revenues from expert R+D activities increased in the reporting period by approx. €2.6 million to €45.0 million (previous year: €42.4 million) and can be represented according to customers as follows: Fig. 2 »EARNING SITUATION«.

The total operating result (revenues and subsidies plus changes in inventory) increased from €49.3 million in 2007 to about €50.3 million in 2008. GRS's revenues of €51.2 million include subsidies amounting to €27.2 million. The total of working hours underlying the staff output increased in the financial year 2008 by 15,076 hours to 439,485 hours (+ 3.55 %).

Development of cost. The cost of materials dropped by €299 thousand to €4.23 million, in particular because of the decrease in external services qualifying for being further set off and purchased goods. The increase in personnel costs of about €1,451 thousand basically results from the increase in contributions (€553 thousand) to the Supplementary Pensions Agency for Federal and Länder Employees (*Versorgungsanstalt des Bundes und der Länder* – VBL) and the increase in wages. Compared to the previous year, the sum of depreciation/amortisation remained stable at €1.1 million and increased by only €56 thousand. The other operating costs increased by €839 thousand



3 OPERATING RESULT
Result of normal business operations

Income	2008 T€	2007 T€	Diff. T€
Income from lending of financial assets	4	17	–13
other interest and similar income	136	84	+52
∑	140	101	+39
Expenditures	**T€**	**T€**	**T€**
Interest and other expenditures	222	289	+67
Balance (expenditures)	-82	-188	+106

4 FINANCIAL RESULT
Development and composition of the financial result

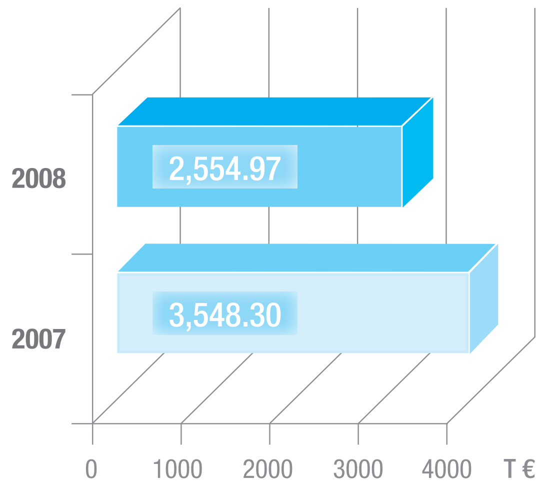
or 6.8 % to €13.22 million. This increase basically results from increased renovation costs of €523 thousand.

Operating result. With €2.18 million, the result of normal business operations is approx. €1.30 million below the previous year's level of €3.48 million. The annual surplus after tax amounts to €2.27 million (pervious year: €3.31 million). This reduction in earnings is the result of, among other things, the increased VBL contributions (+ €553 thousand), the increased qualification measures due to new hire (+ €358 thousand) and the increased maintenance and modernisation costs (+ €523 thousand) (see Fig. 3 »OPERATING RESULT«).

Financial result. The financial result increased by 56.4 % from -€188 thousand to -€82 thousand. The financial result includes the following items: Fig. 4 shows the different entries of the financial result.

Group result

With €390 thousand (previous year: €407 thousand), the operating result of GRS' subsidiary ISTec is positive as in previous years and could thus be more or less maintained. ISTec shows an overall annual surplus of €281 thousand (previous year: €233 thousand). The increase by €48 thou-



5 GROUP PROFIT
Development of group profit

sand results from an improved financial result as well as lower tax charges.

The revenues of the group's profit and loss account are decisively determined by the GRS turnover. For GRS and its subsidiary ISTec, the group turnover in 2008 is €56.2 million compared to €53.3 million in 2007. This corresponds to an increase of €2.7 million or 5.1 %.

The profit and loss account shows result of the group's ordinary activitiesnormal business operations of €2.6 million (previous year: €3.9 million). Profit tax of €13 thousand and other taxes of €2 thousand reduce this result only marginally. Overall, the surplus amounts to €2.6 million (previous year: €3.5 million).

Overall, the financial position and results of operation of the GRS group can be rated as very good for 2008 as both ISTec and GRS could achieve very good results due to high capacity (see Fig. 5 »GROUP PROFIT«).

3. Reactor safety research

Victor Teschendorff

GRS plays a key role in the further development of the state-of-the-art in science and technology in Germany which is vital for reactor safety. With the development of its own methods, especially by providing validated calculation programmes for the simulation of transients and accident and incident sequences, GRS makes important contributions to the solution of current and future safety-relevant problems. GRS is integrated in the German Alliance for Competence in Nuclear Technology, is a reliable partner in work-sharing European networks and adopts within the scope of the European Technical Safety Organisation Network (ETSON) an active role in the cross-sectional field of safety in the European technical platform on nuclear energy (Sustainable Nuclear Energy Technology Platform – SNE-TP). Some topics relating to reactor safety research which GRS has intensively dealt with and where special progress was made will be described in the following.

Reactor physics

Besides the safety-related assessment of the design of the reactor core and the description of the dynamic core behaviour under transient and accident conditions, the characterisation of the core load in respect of nuclide and activity inventories as well as decay heat generation belongs to the most important tasks of reactor physics. Furthermore, reactor physics provides important initial and boundary conditions for subsequent analyses. Examples for this are the source terms for the calculation of the release of fission products in the containment, the neutron fluences for embrittlement analyses of the reactor pressure vessel and the power histories for the thermal-mechanical fuel rod analysis. The overall task can only be accomplished with a consistent nuclear calculation chain which contains a balanced relation between effort and required accuracy of the individual codes integrated therein.

Work of GRS. As on the one hand, the computer performance available has considerably improved and, on the other hand, the required accuracy has further increased, GRS continuously refines its tools. In doing so, all modules are not covered by own developments, but also foreign codes and software subject to licence are included.

Code system »KENOREST« GRS-development

Multi-group cross-sections and high-resolution transport methods. Great potential for improvement is expected from the use of multi-group cross-sections and high-resolution transport methods (rod-by-rod representation of the reactor core) which GRS is currently developing. The reason for this development is that this method enables a geometrically detailed resolution of the reactor core. It must be recognised, however, that the simplified diffusion procedures still serve well in the core design. In addition, the maximum achievable accuracy of the calculations is not determined by the application of high-quality transport methods alone, but also by the uncertainties in the experimental basic data base and the thermal-hydraulic system codes.

Stochastic transport methods. Stochastic transport methods – also called Monte Carlo procedure – are always used where stationary reference solutions with high-grade geometric and energetic resolution are required. However, these methods require a calculation time which is at least one to two orders of magnitude higher than deterministic approaches and are basically unsuitable for transient and incident analyses. For this reason they are increasingly used for the confirmatory analysis of critical experiments as well as for the fuel element design and burn-up analysis.

Code system KENOREST. One example for the use of stochastic transport methods is the code system KENOREST developed by GRS. This code can be considered as reference code for the determination of nuclide and activity inventories of fuel elements. Recalculations of experiments have shown certain deviations between experiment and calculation for different important leader nuclides which were thoroughly analysed and assessed as to their relevance.

Coupled calculation systems indispensible for the assessment of innovative reactor concepts. The coupled calculation systems for the comprehensive incident simulation and the further development thereof by implementing multi-group transport method are of particular importance. With this, an improved validity is achieved for existing light water reactors at first. Coupled calculations are, however, indispensable when it comes to the assessment of some innovative reactor concepts which cannot be treated with traditional calculation methods. This applies, for example, to high-temperature reactors as are presently discussed within the scope of generation-IV concepts. GRS intends to participate with its nuclear calculation chain in the safety assessment of such reactor concepts. GRS's current calculation codes and those being under way constitute a sound basis for this and will be further developed in the future with this objective in mind.

Thermal-hydraulics in the cooling circuit

The thermal-hydraulics code ATHLET. The thermal-hydraulics code ATHLET (Analysis of the Thermal-Hydraulics of Leaks and Transients) developed by GRS is used to simulate the entire range of loss of coolant accidents in light water reactors. ATHLET is presently being used by more than 40

organisations, both in Germany and abroad, for the safety demonstration in supervisory and licensing procedures. The systematic validation and the continuous further development of ATHLET are the essential focal points of GRS's reactor safety research in the field of thermal-hydraulics.

Within the scope of the validation of ATHLET, among others, the German integral experiment PKL-III F4.1 was recalculated during which a systematic analysis of the boron dilution during reflux condenser operation in dependence on the inventory of the primary coolant was conducted at constant pressure and constant power. The main result is the good agreement between calculation and experiment in respect of the mass flows in the loops and the conditions at the beginning of the reflux condenser operation. The delayed boron dilution due to an intermittent flow through individual, short U-tubes of the steam generator which was observed in the experiment was not calculated. From this it follows that, among other things, a simulation of the steam generator tubes by three groups of different lengths does not suffice to correctly simulate these processes.

The findings gained from the validation of the programme and the feedback from the extensive and manifold use thereof in an ever-expanding field of application do not only serve quality assurance but are also included in the formulation of objectives for the further development of ATHLET. So, the simulation of transients and accident sequences under increased accuracy requirements shall be improved and the applicability of ATHLET to an expanded range of accidents shall be achieved. To this expanded range belong, among other things, the low-power and shutdown operation, accident management, boron dilution accidents in case of certain leaks in the pressurised water reactor as well as power increase and high burn-up.

Presently, one focus in the further development of ATHLET is the calculation of multi-dimensional flow processes. This expansion of the scope of simulation shall be achieved by means of the multi-dimensional module FLUBOX 2D/3D on the one hand, and by coupling ATHLET with the calculation programme CFX on the other hand.

The module FLUBOX 2D/3D. With FLUBOX 2D/3D, GRS is developing a module with which the multi-dimensional one- and two-phase flow shall be simulated without having to use the conventional technique with parallel channels and cross connections in the reactor pressure vessel. The fields of application for this programme extension are loss of coolant accidents and reactivity-initiated accidents. The basis of module FLUBOX 2D/3D is the balance equations for mass, momentum and energy which are extended by the transport equations for the phase boundaries and the turbulence. The module gets implicitly coupled with the system programme ATHLET. This extension enables a two-dimensional calculation for the annulus and a three-dimensional calculation for the rest of the reactor pressure vessel.

Difference between the CFD code CFX and the FLUBOX module. The coupling of the CFD calculation programme CFX with ATHLET serves the detailed calculation of multi-dimensional flows in individual components. In contrast to FLUBOX, with module CFX the flow is finely resolved. The higher resolution is required for those calculations of the reactor coolant system for which detailed simulations within individual plant components are necessary. As the calculation times with the CFX programme are very long, CFX can only be used for selected plant components. So far, GRS has already worked on the development of a concept for the exchange of data between the two calculation programmes.

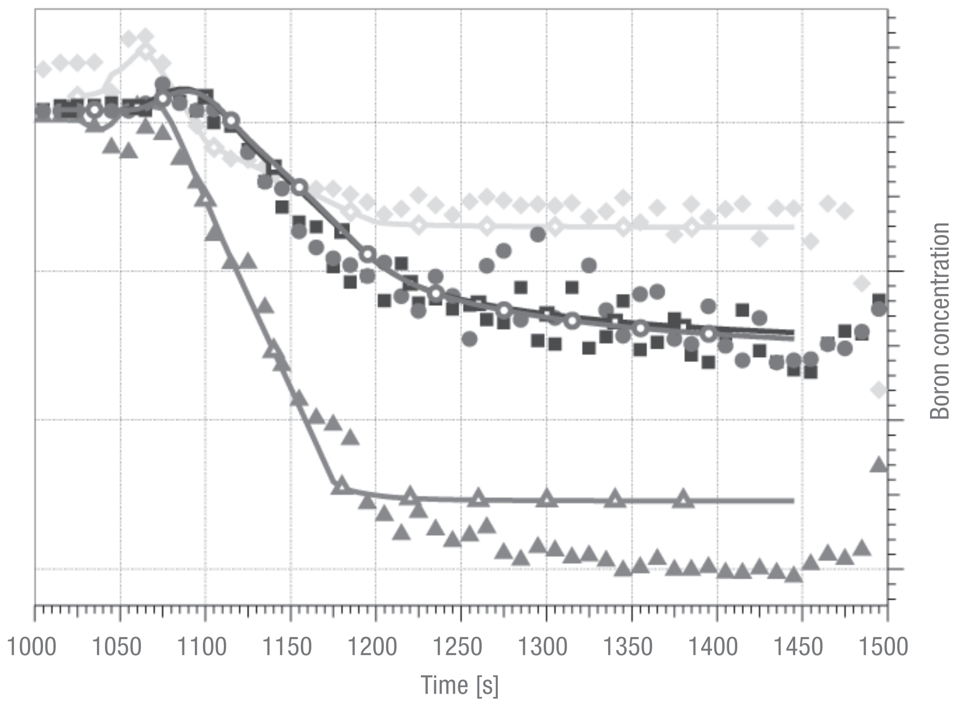
Simulation code »FLUBOX 2D/3D« developed by GRS

CALCULATION RESULT

Fig. 6
Confirmatory calculation of the TRAM-C1 experiment at the UPTF test facility.

With the help of the GRS code FLUBOX, the boron concentration occurrring at different levels of the annulus of the test facility were calculated.

The results of this simulation (represented by the curves) show good agreement with the real values (represented by the individual symbols) measured in the experiment.



Collaboration. The co-operation in the field of CFD calculation programmes in Germany is organised in the CFD research network which is co-ordinated by GRS. In addition, GRS engages in the European simulation platform NURESIM (Nuclear Reactor Simulation) and participates to this end in the project NURISP (Nuclear Reactor Integrated Simulation Project) in the 7th framework programme for research of the EU.

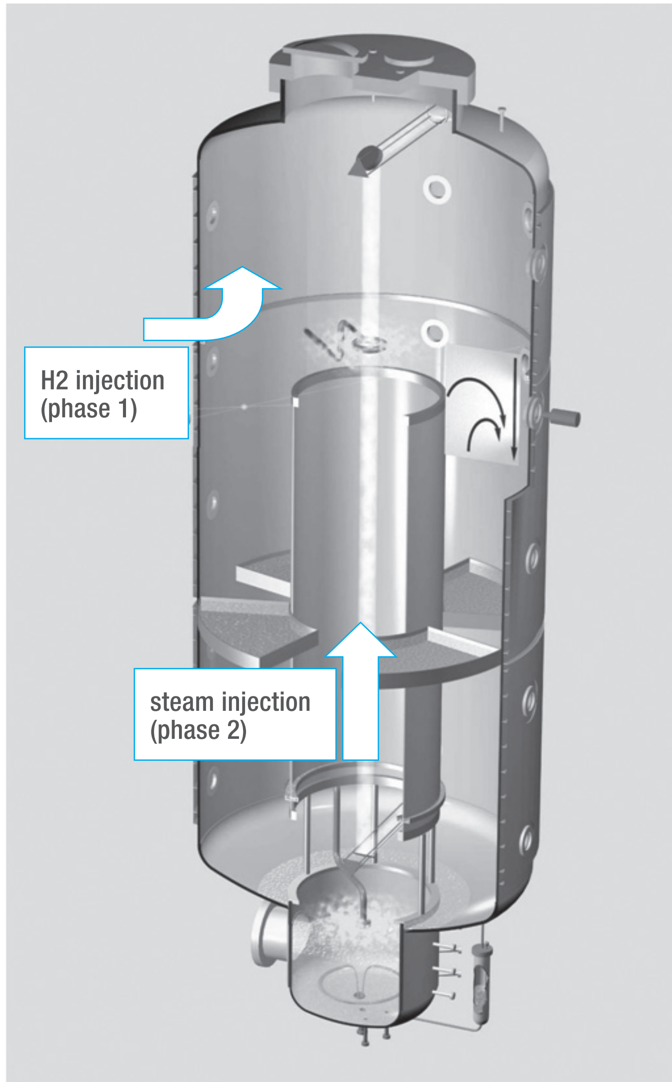
Accident and incident sequences in the containment system

In the safety concept of German nuclear power plants, the containment plays a key role: On the one hand, it constitutes the last barrier against a release of radioactivity into the environment of a nuclear power plant in case of a beyond-design-basis accident; on the other hand, the containment and the surrounding reactor building protect the plant against external events. Reactor safety research takes this double function into account and analyses both phenomena inside the containment and its behaviour in case of loads from the inside and outside.

Work of GRS. GRS's activities in this area comprise the development and validation of calculation programmes for the analytical simulation of accidents and incidents for the containment system (COCOSYS) and – in co-operation with our French partner IRSN – for the overall plant (ASTEC) as well as the exemplary application thereof for probabilistic safety analyses. Furthermore, three-dimensional analysis tools (CFX) are increasingly analysed in respect of their applicability to problems relevant for the containment. These programmes can be used for a wide range of currently operated and future nuclear power

TEST VESSEL OF THE THAI FACILITY

Fig. 7
Experiments at the THAI facility in Eschborn: Injection locations for hydrogen and steam for the HM-2 hydrogen distribution experiment.



plants. This includes the German pressurised and boiling water reactors, plants of the Russian model VVER-440 and VVER-1000 as well as plants of the III. Generation like the European pressurised water reactor EPR. These plants are characterised by, among other things, novel passive safety systems the simulation of which poses new challenges to the analysis tools.

COCOSYS and ASTEC. Special progress was made in the further development and validation of the calculation programmes COCOSYS and ASTEC. In this context, the focus was on the completion of the programme systems, the continuous adaptation thereof to the new findings in research and the implementation of the experience feedback gained from the internal and external use. The attendance and evaluation of national and international reactor safety experiments is essential for the validation of the analysis tools. For both calculation programmes, so-called validation matrices were developed which systematically record the available experiments and show the status of the inclusion of the experiments into the validation.

The experiments in the THAI facility in Eschborn represent one main focus of the validation (see Fig. 7 »TEST VESSEL OF THE THAI FACILITY«). The results of a blind recalculation of the hydrogen distribution test HM-2 conducted by GRS and international partners at this test facility have exemplified the quality of COCOSYS which was achieved through the long-time use and further development. The continuation of the German THAI experiment programme and its expansion to OECD/NEA countries is important for the development and validation of the programme as the tests conducted there decisively contribute to the closing of existing gaps in knowledge.

Currently, an uncertainty and sensitivity analysis of the ATHLET results on a main steam line break of a PWR is being carried out. This takes place within the scope of an international testing of the analysis method suggested by OECD/CSNI for the assessment of safety margins. The method connects deterministic and probabilistic approaches. Uncertainty analyses are an essential part within the scope of this method. The reference reactor is, again, Zion DWR in the U.S.A, a Westinghouse type with four circuits. The plant is not in operation anymore. The potential occurrence of critical boiling conditions and the capacity of the safety systems for pressure relief and feed on the primary side and feed after a power increase of 10 % shall be examined. Furthermore, the determination of uncertain input parameters shall be ensured.

Component behaviour and structural reliability

Work of GRS. Objective of GRS's activities relating to the main research »component behaviour and structural reliability« is it to further develop and validate analysis methods for the integrity assessment of safety-relevant components (reactor building, containment, pipelines, pressure vessel and cladding tubes) under load due to postulated accidents or incidents. In doing so, the influence of uncertainties in the calculation models, the load data, the material data and boundary conditions are examined in respect of safety-related assessments, too. Within the scope of the work on the validation of structure-mechanical analysis methods, primarily third-party programmes like ADINA (Automatic Dynamic Incremental Nonlinear Analysis) and AUTODYN (simulation programme for highly dynamic processes) are used. The further development of the analysis methods is currently focused on the GRS programme PROST (probabilistic structure calculation). The programme serves the calculation of the leak and break probability of damaged pipes and vessels. The activities are safety-relevant in view of taking into account the ageing of components, the interfaces to thermal-hydraulic and system-related problems as well as the contributions to probabilistic safety analyses.

Collaboration. GRS is partner in the European network NULIFE (Nuclear Plant Life Prediction) where a virtual communication platform with scientific and technical information about the »lifetime assessment of components and structures« shall be established. The NULIFE consortium consists of 37 European industrial and research organisations from 16 countries. German representatives are AREVA-NP, EON Kernkraft, FZ Dresden, FH-IWM, GRS, MPA Stuttgart and Siempelkamp. Within the scope of NULIFE, expert groups were set up which will deal with the topics material, integrity/lifetime and safety/risk. ∎

3.1 Calculation method for the determination of leak probabilities for complex structural geometries and load conditions

Dr. Jürgen Sievers

Yan Wang

When it comes to assessing the safety of pressure vessels, piping, containments and other passive structures in nuclear installations, the focus in Germany has so far been on deterministic methods. In other technical fields such as civil engineering, structural steelwork and offshore constructions, the rules and standards in Germany and in other countries increasingly provide for quantitative specifications of the reliability of structures in the form of leak and break probabilities. At international level, a similar trend can also be observed for the further development of nuclear regulations. Thus the US and Sweden, but also other countries, have for many years pursued the approach of establishing regulatory acceptance targets for the reliability of relevant components on the basis of quantitative risk analyses. Such an approach is called »risk-based«, »risk-informed« or »risk-oriented«. In the area of non-destructive in-service inspections, »risk-based« strategies were developed to choose both the locations to be tested as well as the chronological order and extent of the tests in correspondence with a risk-based ranking of the locations. In doing so, probabilistic models to determine the structural reliability of passive components will be increasingly used.

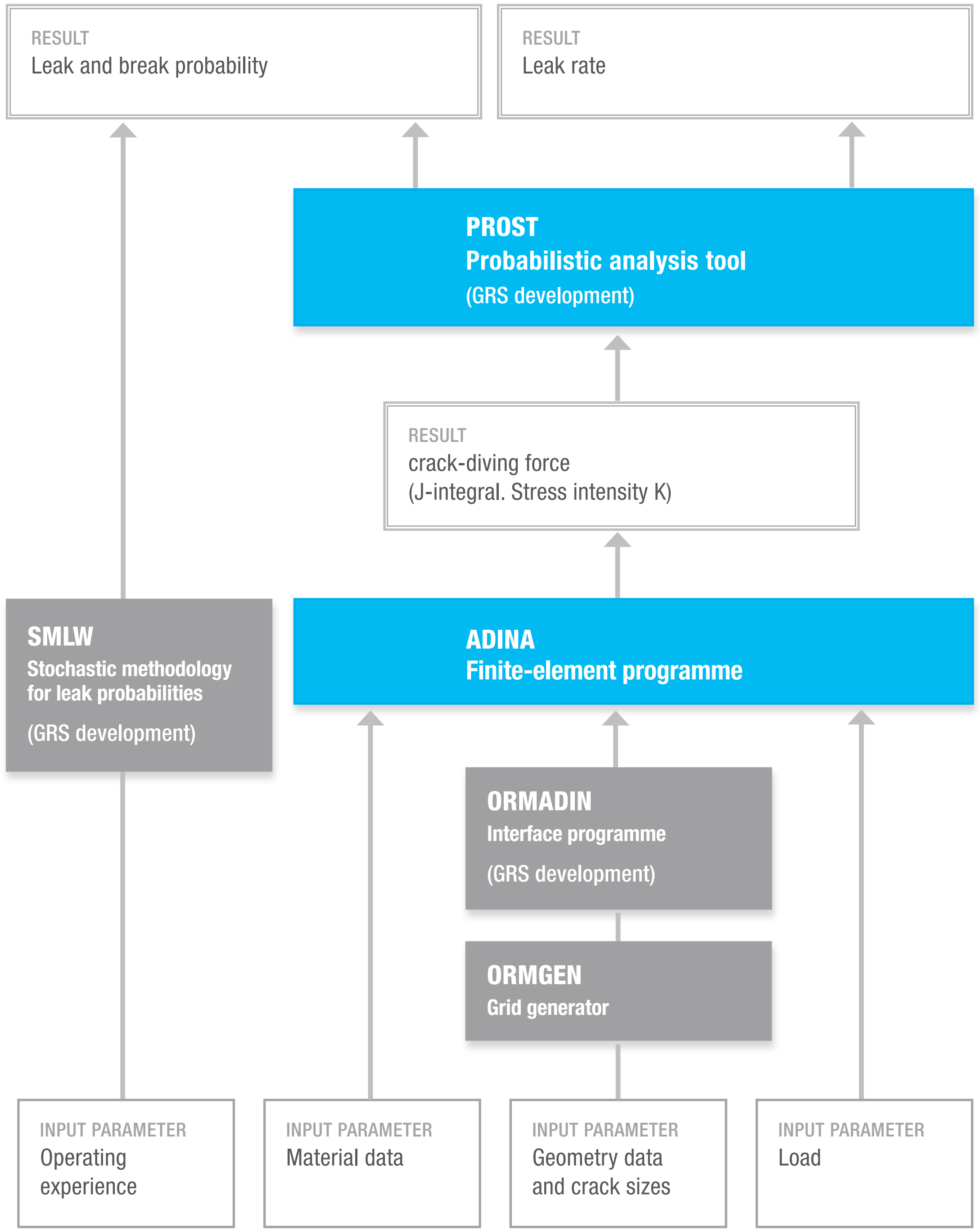
3.1 Calculation method for the determination of leak probabilities for complex structural geometries and load conditions
CONTENT

Further development of the probabilistic analysis method

Deterministic analyses. Integrity assessments and safety analyses for passive components in German nuclear power plants usually show high safety margins in respect of leak- or break-induced failures of the component. These safety analyses are based on deterministic »best estimate« analyses. In doing so, the loads in the component structure calculated for specified load scenarios are juxtaposed with stress and strain criteria as well as fracture-mechanical criteria. The methods of the demonstration procedure as well as the safety coefficients for pressurised components in German nuclear power plants are specified in the nuclear safety standards of the Nuclear Safety Standards Commission (*Kerntechnischer Ausschuss* – KTA).

Quantitative determination of structural reliability. Evaluations of operating experience gained in German nuclear power plants do not provide indications of any damage with significant influence on the reliability of the structures concerned. Due to the dense monitoring network for pressurised components, occasional limited damage was predominantly detected at an early stage and appropriate measures were taken. Due to the small number of comparable components and the low number of leak events, the database of German operating experience allows a direct quantification of the reliability of the structures up to a range of only approx. 10^{-1} to 10^{-2} per reactor year. An extension of the database with operating experience from other countries would extend the quantification range to approx. 10^{-4} per reactor year. For probabilistic safety analyses (PSA), GRS has developed and tested a method to determine leak and break frequencies in piping. This method is based on the evaluation of German operating experience and statistical procedures to take account of uncertainties in relevant calculation parameters in the form of distribution functions (see Fig. 8 »OVERVIEW: CALCULATION METHODOLOGY«). For statements on the leak frequency in piping for which only a few or no leak events at all exist, i.e. to assess the structural reliability of passive components in the low-value range, structural reliability models (SRM) based on probabilistic fracture mechanics are available. Within the scope of probabilistic fracture mechanics, the structural reliability is described by the probability that the size of a crack will exceed a certain critical value. In doing so, it is assumed that the crack size at the beginning can change in the further course of operation due to planned or incidental load conditions resulting from different damage mechanisms.

Probabilistic structural calculation – PROST. To be able to better quantify the impacts of cracks on the structural reliability of components, GRS is developing the probabilistic analysis tool PROST (PRObabilistic STructural calculation). Initially, the development of PROST is limited to the safety-relevant damage mechanism »Fatigue in cylindrical structures with crack-like damage«, i.e. to piping and pressure vessels. For this field of application, PROST allows at this stage already the cal-



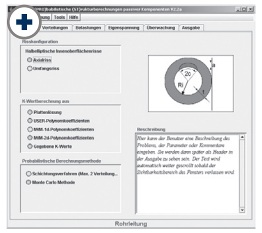
OVERVIEW: CALCULATION METHODOLOGY

Fig. 8
Calculation methodology for the determination of leak probabilities for complex structure geometries and load conditions

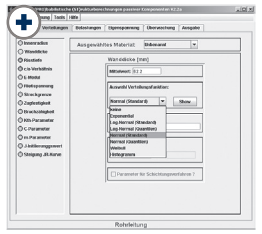
PROST
(Probabilistic analysis tool)

USER INTERFACE

Fig. 9–11
Example of input masks of the PROST code developed by GRS



9 FLAG »CONTROL«
Input: geometry data



10 FLAG »DISTRIBUTIONS«
Input: material data

11 FLAG »LOADS«
Input: load data

culation of leak and break probabilities for different piping geometries, load assumptions and crack distributions (see Fig. 9–11 »USER INTERFACE«). The assumed initial fault distributions are derived, for example, from analyses on the fault development in welds during manufacture or from indications. The distribution functions used for the calculations can normally be verified with the experience gained from manufacture only in areas of relatively small faults.

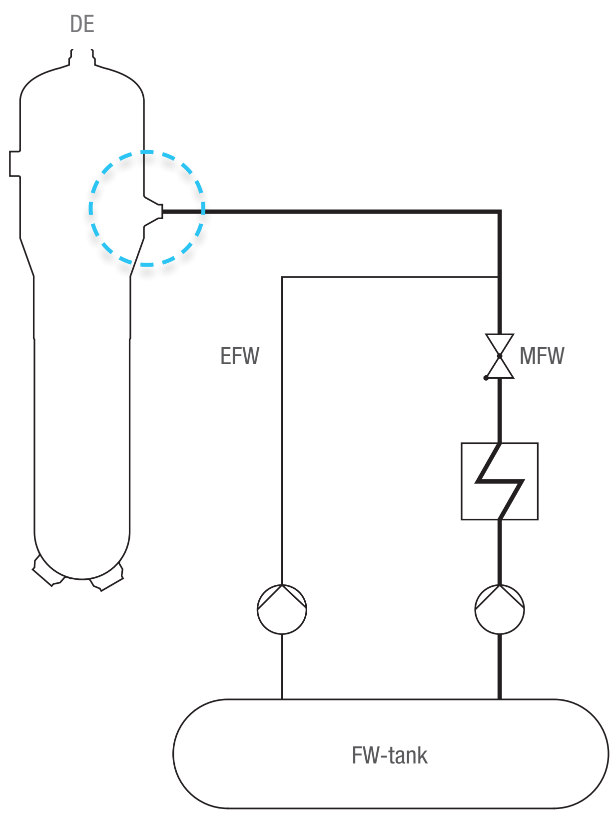
The finite-elements code ADINA. PROST contains analytical approaches to calculate the crack-driving force (stress intensity K) with a limited field of application. For complex boundary conditions regarding geometry and load, a deterministic fracture-mechanical analysis method based on the finite-element (FE) code ADINA (which has been coupled with PROST) is available. In fracture-mechanical FE models, a mesh refinement in the crack front area is required. To this end, the crack mesh generator ORMGEN is applied with which FE meshes can be generated for cylindrical structures with a crack by means of only few control commands. The GRS code ORMADIN was developed to transform the data generated by ORMGEN into the input format of the latest ADINA version. Via an interface, the crack-driving forces calculated with ADINA for different crack geometries can be made available for the probabilistic calculation in PROST (see Fig. 8 »OVERVIEW: CALCULATION METHODOLOGY«).

Validation of the probabilistic analysis method

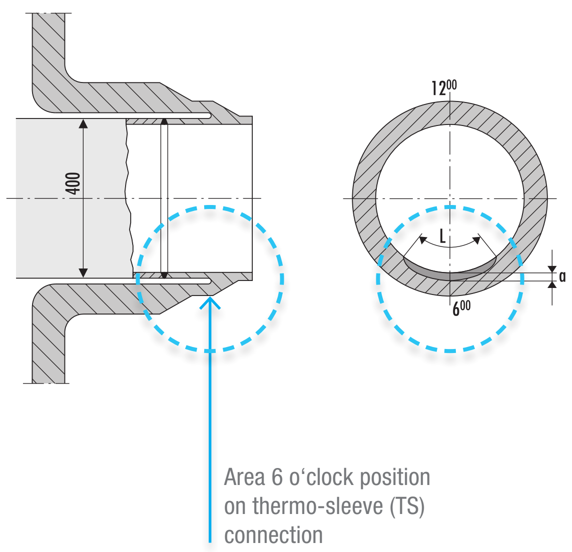
Damage mechanism »thermal fatigue«. First qualification steps for the developed code features were carried using the example of the damage mechanism »thermal fatigue« in a pipe section of the volume control system of a pressurised water reactor. A report on this can be found in the GRS Annual Report 2002/2003. Within the scope of the EU project NURBIM (Nuclear Risk Based Inspection Methodology for Passive Components), the precision of the analytical methods to determine the structural reliability of passive components were examined and compared at international level. To that end, a benchmark on the damage mechanism »fatigue« was conducted. The distributions of the partial circumferential cracks located on the inner surface in the form of manufacturing faults in weld joints of austenic straight pipes which in the further course of the operating time will be exposed to cyclical loads were examined. The influence of the individual parameters on the leak probability at the beginning of operating time and after 40 years of operation was analysed by varying the input parameters which were assumed to be distributed or set. Variations of the initial crack depth distribution, especially of the deep cracks, the crack growth constants and the maximum load have quantitatively the greatest impact on the leak probability after 40 years. The calculations taking into account the option »in-service inspections« yielded the result that for the damage mechanism »fatigue«, all tests performed after a first inspection have no remarkable impact on the leak probability anymore. The results GRS obtained with PROST show good agreement with the results of other current calculation codes to determine the structural reliability of pipes. For most reference analyses, the differences in the calculated leakage probabilities in the range between 10^{-4} and 10^{-8} (per reactor year) are less than one order of magnitude.

3.1 Calculation method for the determination of leak probabilities for complex structural geometries and load conditions
CONTENT

12 SYSTEM SECTION
of a pressurised water reactor



13 CRACK WOICATIONS
in a feedwater nozzle



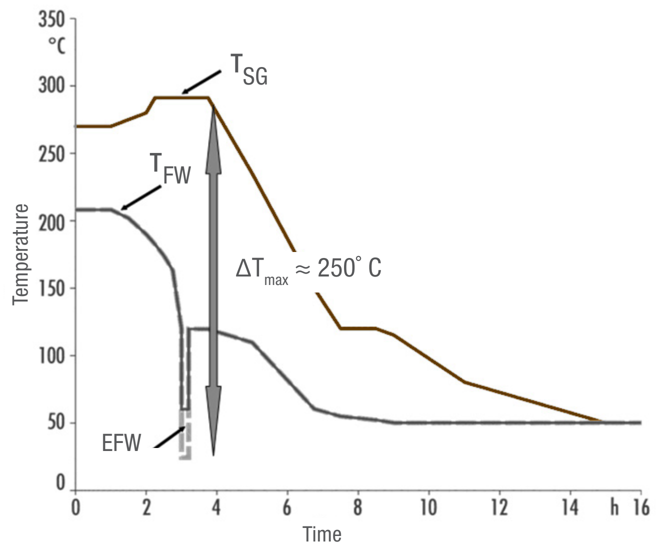
PROBLEM

Fig. 12–14
Crack indications in a feedwater line nozzle and loading due to thermal stratificatin

Calculations on the feed-water connection under loads due to thermal stratification

Exemplary application of the probabilistic analysis method. As an exemplary application of the probabilistic analysis method for complex structural geometries and load conditions, examinations on the leak probability in a feedwater connection under loads of internal pressure and thermal stratification were performed with ADINA and PROST.

14 LOADS
due to thermal stratification



SG	=	steam generator
MFW	=	main feedwater
EFW	=	emergency feedwater
FW	=	feedwater
TS	=	thermo-sleeve

Analysis results. Within the scope of in-service inspections, crack indications were detected on the inner surface of the feedwater connection of a pressurised water reactor. The crack indications were located in the base material at the beginning of the fillet to where the thermo-sleeves are fitted. One crack indication was located nearly symmetrically to the six o'clock-position in an angle bracket of approx. 80° C. If during cooldown feedwater is pushed from the pipe sections not preheated into the still hot steam generator, cyclical loads in the form of temperature differences of up to 250° C between upper and lower part of the feedwater pipe can occur at an operating pressure of 6.6 MPa. Changes in respect of starting the emergency feedwater supply can reduce the loads (see Fig. 12–14 »PROBLEM«).

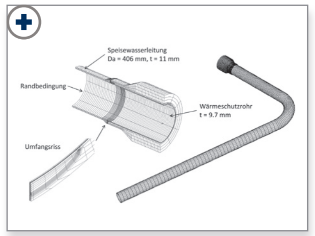
3.1 Calculation method for the determination of leak probabilities for complex structural geometries and load conditions

CONTENT

ADINA (Finite-Element-Programme)

ANALYSIS MODEL

Fig. 15
Feedwater nozzle under load from thermal stratification

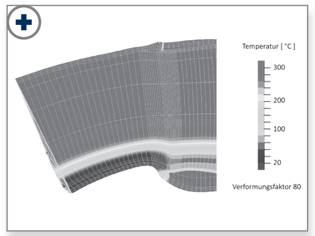


15 SHORT FE MODEL
with crack and long FE model

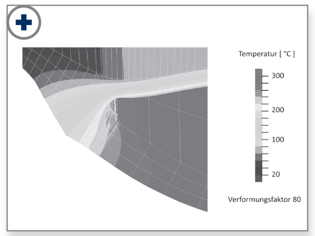
CALCULATION RESULT

Fig. 16–18
Deformation and temperature distribution of the feedwater nozzle under operating pressure and thermal stratification (ΔT = 250°)

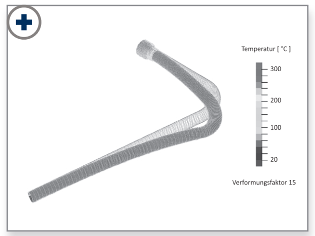
Fig. 19
Crack-driving forces for various crack geometries in the feedwater nozzle under thermomechanical loading, calculated with ADINA



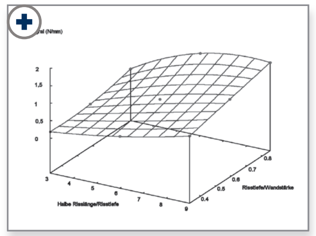
16 SHORT FE MODEL



17 SHORT FE MODEL
Detail



18 LONG FE MODEL
Deformation



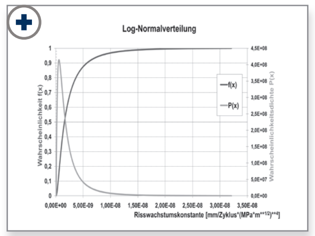
19 CRACK-DRIVING FORCE
(J-integral)

PROST (Probabilistic analysis tool)

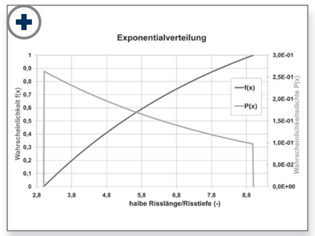
CALCULATION RESULT

Fig. 20–21
Distribution function for the crack-length-to-crack-depth ratio as well as for the crack growth constant C

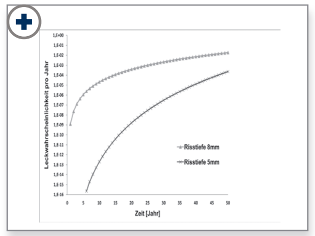
Fig. 22
Leak probability for the feedwater nozzle under loading due to thermal stratification as a function of operating time



20 DISTRIBUTION FUNCTION
for the crack growth constant



21 DISTRIBUTION FUNCTION
for the crack-length-to-crack-depth ratio



22 LEAK PROBABILITY
Crack depths
a = 5 and 8 mm, resp.

Short and long FE model. The stress due to the temperature differences in the area of the location of indication, i.e. the stress distribution starting from the inner surface of the feedwater connection through the wall at the six o'clock-position, was analysed with ADINA. The calculation results of the short FE model of the feedwater connection are highly dependent on the boundary conditions at the cut free cross-section surface of the feed-water pipe (see Fig. 15 »ANALYSIS MODEL«). The comparison with the results of the long FE model which reaches to the next fix point and is assumed to be the »best-estimate« model has shown that the boundary condition, in which the end cross-section of the feed-water pipe at a rotatable level is limited, yields the best results (see Fig. 16–18 »CALCULATION RESULT«). Crack-driving forces under thermal-mechanical load were calculated with ADINA for different crack geometries. The results constitute the bases for interpolations within the scope of the calculations with PROST (see Fig. 19 »CALCULATION RESULT«). As far as knowledge uncertainties regarding crack, material and geometry data as well as load parameters are concerned, corresponding distribution functions can be taken into consideration (see Fig. 21–23 »CALCULATION RESULT«).

The calculation of leak probabilities. With PROST, leak probabilities were calculated for two set cracks with crack depths oriented on the actual indications, with an internal pressure and thermal stratification with 160 cycles per year as well as distribution functions for the crack growth constant C and the relation between crack length and crack depth being assumed as load. The results have shown that with the described probabilistic analysis method, quantitative statements can be made on the leak probability of a detected or postulated crack as a function of operating time in the range of very small values (< 10^{-7}) up to high values in the range of (> 10^{-2}) (see Fig. 22 »CALCULATION RESULT«). Within the scope of an uncertainty and sensitivity analysis, the essential influencing variables on the leak probability can be identified.

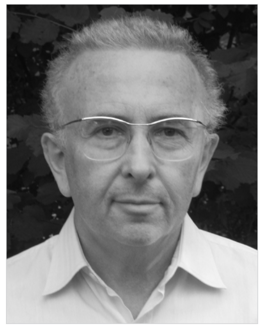
Summary

Cracks initiating on surfaces. Experience has shown that cracks in pressurised components in nuclear power plants which can impair the further operation of these components predominantly start at the surface. The decisive factors for this are often locally unfavourable load and media conditions. Manufacturing faults which have remained in the volume of the base material are normally less significant. To improve the knowledge about the impacts of abnormal load and media conditions, the instrumentation in the nuclear power plants has been considerably extended.

Structural reliability codes as valuable supplement. With the described probabilistic analysis method, leak and break probabilities can be determined quantitatively for certain damage mechanisms. When determining the leak probability dependent on the position, sections of the piping systems can be differentiated as to their failure relevance. For the probabilistic assessment of the indications, leak and break probabilities can be determined for assumed crack geometries. Furthermore, it is possible to quantitatively identify trends concerning the change of influencing factors. Limitations regarding the availability within the scope of probabilistic safety analyses (PSA) are seen in particular in respect of the precision of absolute leak and break probabilities as the results partially depend to a high degree on the uncertainties for relevant input parameters like crack geometry, expected loads as well as certain parameters for the characterisation of the damage mechanisms. Overall, structure reliability codes are a valuable tool to supplement the methods which have so far been applied within the scope of PSAs to assess leak and break probabilities. ■

3.2 Uncertainty and sensitivity analysis of results of the post test calculation of a LSTF experiment with the code ATHLET

Dr. Henrique Austregesilo



Bernard Krzykacz-Hausmann

Tomasz Skorek

One of the main goals of GRS's reactor safety research is the development of codes to calculate thermal-hydraulic processes. For the validation of such codes, experiments performed at test facilities are simulated. The comparison of the test results with the results of the calculation allows deductions on the code accuracy. Within the scope of the validation of the thermal-hydraulic code ATHLET, a post test calculation of the experiment SB-PV-09 carried out at the LSTF test facility has been performed. This calculation was complemented by an uncertainty and sensitivity analysis of the obtained results. The main goal was to investigate the influence of a combined variation of the input data upon the simulation of the main phenomena observed experimentally. In this analysis multiple sensitivity coefficients have been derived, allowing a comparison between the influence due to modelling uncertainties and those due to experimentally induced uncertainties.

The Experiment SB-PV-09

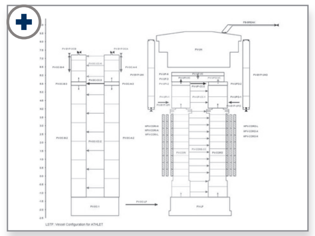
ROSA-V/LSTF test facility. The Japanese ROSA-V/LSTF test facility is the volumetrically scaled (1/48) model of a pressurised water reactor with four loops and a thermal power of 3,423 MW. The facility is designed for a full system pressure of 16 MPa. The four reactor loops of the reference plant are combined into 2 double loops. The horizontal legs of the loops are scaled by means of the Froude number in order that two phase flow regime transitions can be reproduced in a reactor-typical manner. The heights are scaled on a 1/1 scale to allow a realistic simulation of natural circulation. The power of the electrically heated core of the LSTF is 10 MW. This allows simulation of the reactor heating with the scaled-down decay heat of the real plant.

Test SB-PV-09 was conducted in November 2005 within the OECD/NEA project »Rig of Safety Assessment« (ROSA).

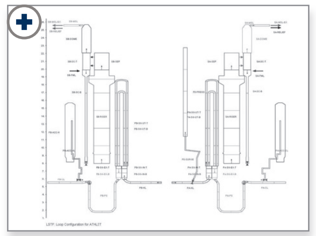
ATHLET (Thermal hydraulics code)

NODALISATIONS

Fig. 23–24
ATHLET model of the LSTF facility



23 ATHLET
Nodalisation of the LSTF pressure vessel



24 ATHLET
Nodalisation of the primary and secondary system

Goals of test SB-PV-09. The main goals of this test were the analysis of the thermal-hydraulic phenomena in the reactor coolant system in case of a postulated break of a control rod drive mechanism penetration nozzle, the evaluation of the impacts of symptom-oriented accident management measures on the coolability of the reactor core as well as the provision of experimental data for the validation of advanced numerical codes.

For this purpose, a small break at the pressure vessel upper head (Davis-Besse scenario) was simulated, under the assumption of a total failure of the high pressure injection system. A secondary side depressurization was foreseen as an accident management (AM) action. The leak size selected for the test setup corresponds to a 1.9 % break in the cold leg of the reference reactor.

The test. The test was initiated by opening the break valve. The break flow rates depended strongly on the water level in the upper head. The coolant in the upper plenum flowed via the control rod guide tubes into the upper head until the water level in the upper plenum sank below the penetration holes in the lower part of the control rod guide tubes. The relatively large break area led to a quick pressure drop in the reactor coolant system. Primary pressure dropped below secondary pressure at approx. 800 s, simultaneously with the beginning of core uncovery.

With the temperatures increasing in the upper core region, the limit (623 K) for the activation of the planned accident management measure was reached at approx. 1,090 s after opening of the break. The secondary side depressurization was initiated by manually opening the relief valves. However, this measure was not effective since the primary pressure was lower than the secondary pressure in this phase. The cladding tube temperatures kept increasing until the limit value for the response of the LSTF core protection system was reached, which caused an automatic reduction of the core power by approx. 75 % at t = 1,200 s.

After approx. 1,300 s, the actuation pressure of the accumulator was reached. The subsequent steam condensation in the cold legs led to the clearance of the pump suction lines which initiated the refilling of the reactor core. Approx. 1,400 s after break, the major part of the core was reflooded. In the further course of the transient, primary pressure dropped again until it reached the activation pressure of the low-pressure injection system at 2,900 s. The test was finished with the closing of the break valve at t = 3,265 s.

Reference calculation

ATHLET input dataset. The basis for the modelling of the LSTF facility was the ATHLET input dataset which was used for the post test analysis of test SB-CL-18 in the frame of the international standard problem ISP 26 (see Fig. 23–24 »NODALISATIONS«). The main modifications which were carried out in the test facility for the current phase of the ROSA test programme were taken into account.

In addition to the actual initial and boundary conditions of test SB-PV-09, the original dataset was changed in the following areas:

- Simulation of the break unit in the upper part of the upper head,
- Application of the swell level model in the upper head,
- Modelling of the new pressurizer and the surge line,
- Revision of the input regarding the accumulator connection lines,
- Simulation of the nitrogen injection after the empting of the accumulator.

Overall, ATHLET could simulate satisfactorily the main phenomena observed experimentally. The influence of the water level in the upper head on the break mass flow was reproduced correctly by the code. Both the calculated time point of initiating the accident management measure and the beginning of the accumulator injection are in good agreement with the experimental values

Simulation results. The multi-dimensional flow processes in the upper part of the reactor pressure vessel (RPV) which were observed in the test could be satisfactorily simulated by modelling parallel channels in the core and in the upper plenum (including the control rod guide tubes). An exception is the late draining of the upper plenum after uncovering the openings in the lower area of the control rod guide tubes which was calculated by the code. This led to a late beginning of core uncovery in comparison with the test.

The deviation in core uncovery may be caused by the modelling of the core bypasses. Parameter

3.2 Uncertainty and sensitivity analysis of results of the post test calculation of a LSTF experiment with the code ATHLET CONTENT

studies conducted along with these calculations show that the value assumed for the form pressure loss of the spray nozzles between the downcomer and the upper head can have a great impact on the calculated moment at the beginning of core uncovery. In addition, the calculated time point of the beginning of core uncovery could only be achieved by way of assuming an additional bypass path between the upper part of the downcomer and the upper plenum. This indicates that either the bypass flow through the spray nozzles was considerably higher than the specified value or that there was an unforeseen leakage between downcomer and upper plenum.

Generally, ATHLET could reproduce the experimentally determined coolant distribution in the coolant loops well and in the RPV satisfactorily, which confirms the applicability of the models for determination of the interphase friction for different geometries and flow conditions.

Uncertainty and sensitivityanalysis

Method to determine the uncertainty of numerical code results. The calculation of test SB-PV-09 was supplemented by an uncertainty and sensitivity analysis to investigate the influence of a combined variation of the uncertain input parameters on the calculated results. For this analysis, the method introduced by GRS to determine the uncertainty of numerical code results was used. This method is based on the simultaneous variation of uncertain input parameters of the calculation model together with a statistical evaluation of the code results. All potentially important input uncertainties can be considered in the analysis. The number of calculations to be performed does not depend on the number of parameters but on the desired probability level and the confidence level of the statistical tolerance limits which are used in the uncertainty statement of the results.

In addition, this method also allows the determination of sensitivity coefficients to quantify the influence of the individual uncertain parameters on the scatter range of the calculation results. From this, a ranking order of the input uncertainties according to their respective relative contribution to the uncertainty of the result ensues. Further developments within the scope of the GRS method also allow for the determination of multiple sensitivity indices which quantify the influence of the uncertainty of an entire parameter group on the uncertainty of the calculation result. With the aid of these multiple sensitivity indices, it is possible to estimate which portion of the uncertainty of results stem from the uncertainties in the experimental setup and which portion stems from modelling uncertainties.

Identification of the uncertain parameters and the probabilistic quantification of their uncertainty. The identification of the uncertain parameters and the probabilistic quantification of their uncertainty are essential for the uncertainty and sensitivity analyses. For the present study, a total of 50 potentially important uncertain parameters were identified and quantified. Among them, there are 40 parameters which describe the uncertainties of the physical modelling and the numerical simulation as well as another 10 parameters which refer to the uncertainties of the test facility and the experiment conducted.

The model uncertainties include:

- 4 parameters for the determination of the critical discharge flow,
- 20 parameters to describe the uncertainties in the momentum equations,
- 9 parameters for the heat transfer from fluid to structures,
- 4 parameters for the two-phase heat and mass exchange through evaporation and condensation,
- 1 parameter for the axial nodalization in the bundle area,
- 2 parameters to describe the form pressure losses in the facility.

The ten test-specific uncertain parameters consider the uncertainties regarding the core bypasses, the bundle power, and leakage through the venting pipe at the RPV upper head as well as the accuracy of the temperature measurements which are used for initiating the accident management measure and triggering the core protection system.

Generation of ATHLET datasets. On the basis of the identified and quantified uncertain parameters and by means of the code system SUSA, 208 ATHLET datasets with the combined variation of the input parameters were generated. The values were determined by simple random selection from the determined distributions (Monte Carlo simulation). Comparing to the usually applied minimum number of $N^{min} = 93$ calculations required for the 95%/95% tolerance limits, the limits calculated on the basis of 208 calculations are markedly less conservative. Furthermore, the accuracy of the derived sensitivity coefficients increases that way.

Uncertainty and sensitivity analysis results. Overall, eight scalar individual values and 15 time-dependent output quantities were chosen for the uncertainty and sensitivity analysis. Significant findings gained from this analysis will be exemplarily represented here by the water level in the active core region.

The double-sided (95%/95%) tolerance limits of the calculated water level in the core region, which were determined on the basis of the variation of all 50 parameters, are shown in Fig. 25 »TOLERANCE LIMITS«. During the core uncovery phase (between 800 and 1,300 s), the corresponding measured values are close to the lower tolerance limit of the calculation results. An additional analysis based only on the variation of the 10 experimental uncertain parameters shows that these parameters alone have less influence on the uncertainty range of the calculation results (see Fig. 26 »TOLERANCE LIMITS«).

The time history of the water level in the bundle can be divided into three different phases:
1. an initial phase of approx. 800 s with steam generation in the core and the formation of a swell level which is high enough to ensure the cooling of the heating rods;
2. the phase of core uncovery and heating of the fuel rods, and
3. the phase of core reflooding with the beginning of the accumulator injection at approx. 1,300 s.

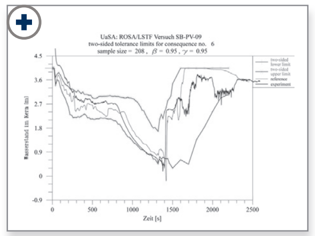
In the initial phase, parameter 17 (calculation of the interphase friction in bundle geometries) and parameter 40 (number of the axial nodes in the heated core area) contribute the most to the uncertainty of the calculated water level (see Fig. 27 »SENSITIVITY MEASURES«). They are significant for the calculation of the void profile in the core region.

ATHLET (Thermal hydraulics code)

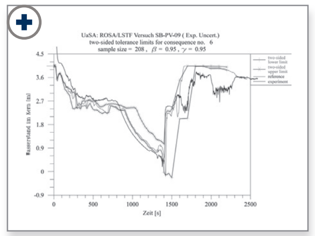
TOLERANCE LIMITS

Fig. 25
Double-sided tolerance limits of the calculated water level in the core with consideration of all 50 uncertain parameters

Fig 26
Double-sided tolerance limits of the calculated water level in the core with consideration of only the 10 experimental uncertain parameters



25 TOLERANCE LIMITS
All 50 uncertain parameters



26 TOLERANCE LIMITS
Only experimental uncertain parameters

However, during the phase essential for the entire test procedure, i.e. the core uncovery and heating phase, parameter 41 (leakage between downcomer and upper plenum) is the most influential parameter: The bigger its value, the lower the water level in the core.

In the re-filling phase, parameter 3 (contraction factor for the steam discharge flow out of the leak) and parameter 41 (now with a positive sign) have the greatest impact on the calculation results. Increasing values of these parameters tend to cause a lower primary pressure and thus a fast re-filling of the core.

The multiple sensitivity measures show (see **Fig. 28** »SENSITIVITY MEASURES«), that the model uncertainties (group 1) are the main contributor to the uncertainties of the calculated water level in the core. Nevertheless, the experimental uncertainties (group 2), particularly the leakage between downcomer and upper plenum, are significant in the important phase of core uncovery.

Summary

The results of the uncertainty and sensitivity analysis endorse the code capability to reproduce adequately the main experimental outcomes. The corresponding measured values lay mostly within the tolerance limits of the calculated results.

The main contributions to the global uncertainty of calculated results are due to the modelling uncertainties, especially to the modelling of the critical discharge flow rates. On the other hand, it has been shown that the experimental uncertainties, particularly the unknown bypass flow between the downcomer and the upper plenum, are crucial for the determination of the coolant distribution within the reactor vessel and consequently for the correct simulation of the important phase of core depletion and heat-up of the fuel rods.

The application of methods of uncertainty and sensitivity analysis to validation calculations is a valuable complement to the assessment and validation process of a computer code. ∎

3.2 Uncertainty and sensitivity analysis of results of the post test calculation of a LSTF experiment with the code ATHLET

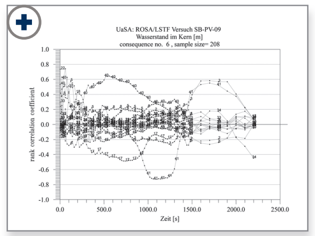
CONTENT

ATHLET (Thermal hydraulics code)

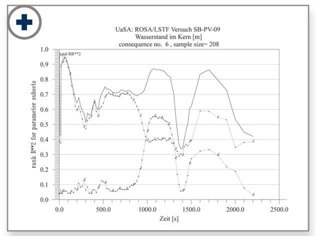
SENSITIVITY MEASURES

Fig. 27
Sensitivity measures of all uncertain parameters for the calculated water level in the core

Fig. 28
Multiple sensitivity measures of the group (1) of the modelled and the group (2) of experimental parameters for the calculated water level in the core

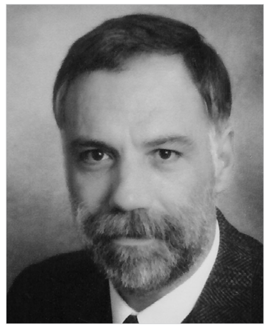


27 SENSITIVITY MEASURES
All 50 uncertain parameters



28 SENSITIVITY MEASURES
Groups of modelled and of experimental parameters

3.3
Influence of nuclear data evaluations on full scale reactor core calculations



Dr. Winfried Zwermann

Nuclear data are mainly validated by the recalculation of critical experiments. Since most of these systems are of compact size and at room temperature, they are not necessarily representative for power reactors in operating conditions. GRS applied different nuclear data evaluations based on JEFF, ENDF/B, and JENDL with Monte Carlo calculations for large UO2/MOX PWR and VVER reactor core configurations. The agreement between the resulting multiplication factors is reasonable, but it is found that the influence of the choice of the nuclear data basis on the radial power distributions can be significant, leading to differences of more than 10 % in fuel assembly powers for the most unfavorable cases. Influences of this type can normally not be observed in calculations for compact critical assemblies, which is demonstrated by comparing calculated pin power distributions for a critical experiment.

Status of nuclear data evaluations

Updating and improvement of existing data libraries. Evaluated nuclear data are continuously improved. During the last years, the European library was updated from JEF-2.2 to JEFF-3.1, the American library from ENDF/B-VI to ENDF/B-VII, and the Japanese library from JENDL-3.2 to JENDL-3.3/AC-2008, with the aim to increase the agreement of calculated and measured results for integral systems. Along with this, growing attention is paid to uncertainty and sensitivity studies concerning the nuclear data evaluations, accompanied by improvements in the covariance data files describing the uncertainties of nuclear cross section data, and the calculation methods using these covariance data.

Validation of nuclear data by means of integral experiments. Evaluated nuclear data are continuously improved. During the last years, the European library was updated from JEF-2.2 to JEFF-3.1, the American library from ENDF/B-VI to ENDF/B-VII, and the Japanese library from JENDL-3.2 to JENDL-3.3/AC-2008, with the aim to increase the agreement of calculated and measured results for integral systems. Along with this, growing attention is paid to uncertainty and sensitivity studies concerning the nuclear data evaluations, accompanied by improvements in the covariance data files describing the uncertainties of nuclear cross section data, and the calculation methods using these covariance data.

Validation of nuclear data by means of integral experiments. For the validation of the nuclear data libraries, a large number of integral experiments with all kinds of fissile and moderator materials and a large range of spectral conditions are used. A collection of descriptions of such experiments is found in the »International Handbook of Evaluated Criticality Safety Benchmark Experiments«. The vast majority of these validation calculations refer to multiplication factors, although other measured quantities like fission rate distributions and reactivity coefficients are increasingly considered; corresponding experiments are described in the »International Handbook of Evaluated Reactor Physics Benchmark Experiments«. Almost all of the systems considered are compact assemblies, mainly at room temperature. Likewise, uncertainty and sensitivity investigations based on covariance data, as performed, e.g., with the TSUNAMI code from the SCALE code package, primarily consider the multiplication factors of critical assemblies. Such compact critical systems at low temperatures are not necessarily representative for power reactors at operating conditions.

Critical assembly and full core calculations

Analyses of differences in multiplication factors and power distributions. GRS investigated the differences in the multiplication factors and power distributions of a critical assembly and of problems representative for large reactor cores arising from the use of different evaluated nuclear data libraries. For this purpose, the MCNP-5 code was used since the Monte Carlo method with continuous energy data currently provides the highest level of accuracy for neutron transport calculations, without significant restrictions in the geometrical modeling, and without preceding spectral calculations for cross section preparation. Therefore it is best suited to disclose nuclear data influences.

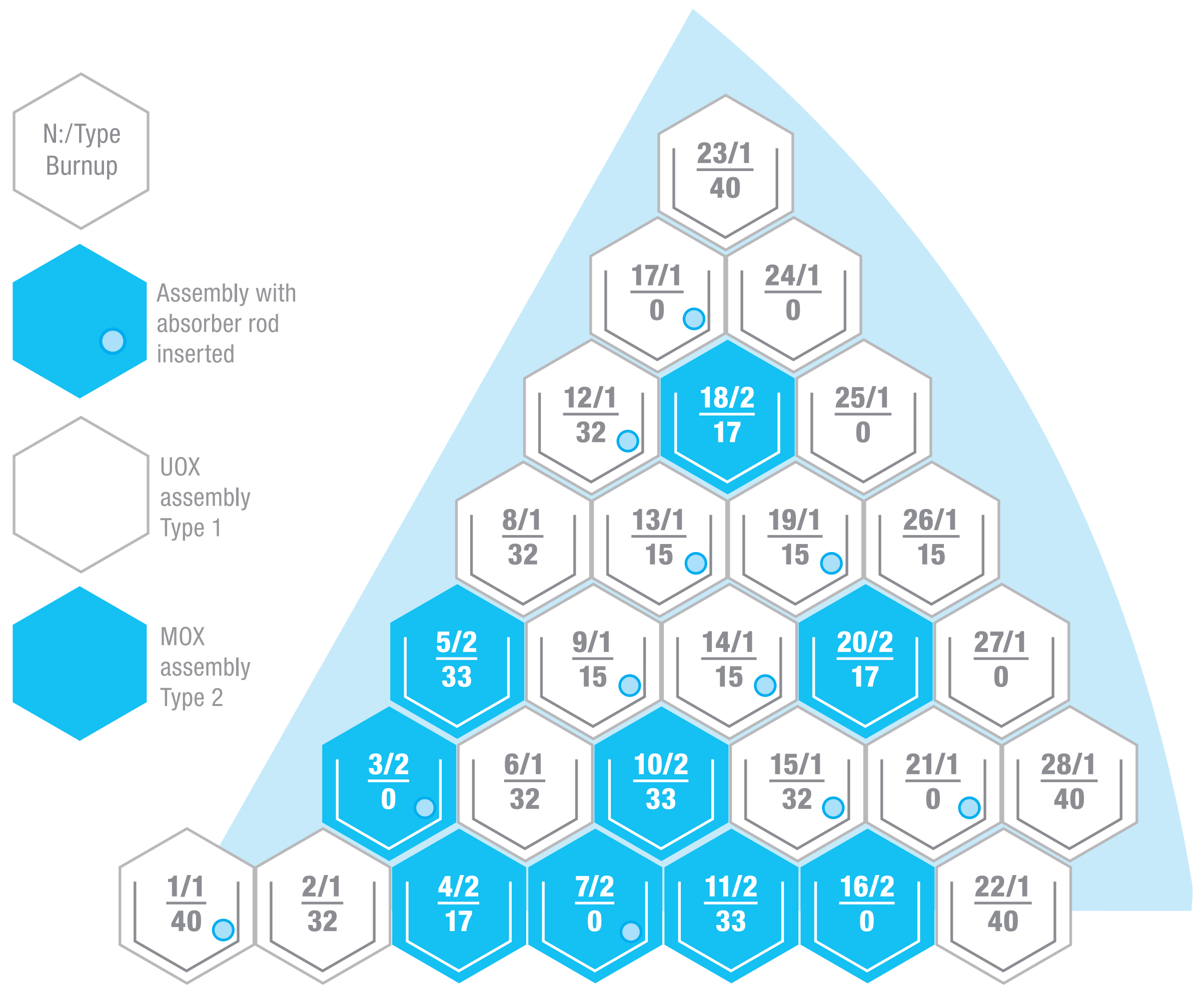
A critical experiment and two full core benchmarks. Calculations with various nuclear data evaluations are performed for a critical benchmark experiment and two full core benchmark arrangements recently dealt with in the framework of OECD/NEA international calculation benchmarks, namely the »Benchmark on the VENUS Plutonium Recycling Experiments – Configuration 7«, and the steady states of the »PWR MOX/UO_2 Core Transient Benchmark« and the »VVER-1000 MOX Core Computational Benchmark«. The VENUS-7 cores are cold square lattices containing approximately 900 fuel pins each. The configuration 7/1 is approximately cylindrical with a radius of 22 cm; it consists of an inner MOX and an outer UO_2 radial zone, and is moderated and reflected by light water. The active height of the fuel pins is 50 cm. The full core benchmarks describe mixed cores with a MOX loading of approximately 30 %. The Westinghouse type PWR core consists of 193 fuel assemblies with 17x17 pin cells, and the VVER core of 163 hexagonal fuel assemblies with 331 pin cells, with various UO_2 and MOX fuel types at several burn-up states up to 40 GWd/t HM. The fresh fuel assemblies contain burnable absorbers. For both core designs, the fresh MOX fuel contains a high percentage of ^{239}Pu in the isotopic composition (see Fig. 29–30 »FULL-SCALE CORE CONFIGURATIONS«).

Applied data libraries. With the aim to cover the influence of a broad spectrum of modern eval-

U 4.2% (CR-D) 35.0	U 4.2% 0.15	U 4.2% (CR-A) 22.5	U 4.5% 0.15	U 4.5% (CR-SD) 37.5	M 4.3% 17.5	U 4.5% (CR-C) 0.15	U 4.2% 32.5
U 4.2% 0.15	U 4.2% 17.5	U 4.5% 32.5	M 4.0% 22.5	U 4.2% 0.15	U 4.2% (CR-SB) 32.5	M 4.0% 0.15	U 4.5% 17.5
U 4.2% (CR-A) 22.5	U 4.5% 32.5	U 4.2% (CR-C) 22.5	U 4.2% 0.15	U 4.2% 22.5	M 4.3% 17.5	U 4.5% (CR-B) 0.15	M 4.3% 35.0
U 4.5% 0.15	M 4.0% 22.5	U 4.2% 0.15	M 4.0% 37.5	U 4.2% 0.15	U 4.5% (CR-SC) 20.0	M 4.3% 0.15	U 4.5% 20.0
U 4.5% (CR-SD) 37.5	U 4.2% 0.15	U 4.2% 22.5	U 4.2% 0.15	U 4.2% (CR-D) 37.5	U 4.5% 0.15	U 4.2% (CR-SA) 17.5	
M 4.3% 17.5	U 4.2% (CR-SB) 32.5	M 4.3% 17.5	U 4.5% (CR-SC) 20.0	U 4.5% 0.15	M 4.3% 0.15	U 4.5% 32.5	
U 4.5% (CR-C) 0.15	M 4.0% 0.15	U 4.5% (CR-B) 0.15	M 4.3% 0.15	U 4.2% (CR-SA) 17.5	U 4.5% 32.5	Assembly Type CR Position Burnup [GWd/t]	
U 4.2% 32.5	U 4.5% 17.5	M 4.3% 0.15	U 4.5% 20.0			UOX assembly MOX assembly	

FULL-SCALE CORE CONFIGURATIONS

Fig. 29
Core design for the »PWR MOX/UO2 Core Transient Benchmark«, with MOX fuel assemblies shown in blue

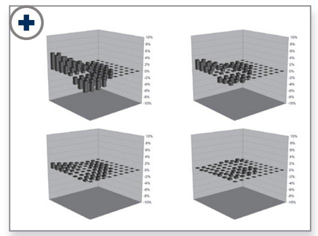


FULL-SCALE CORE CONFIGURATIONS

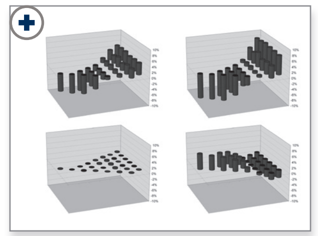
Fig. 30
Core design for the »VVER-1000 MOX Core Computational Benchmark«, with MOX fuel assemblies shown in blue and control rod positions represented by dots

uated nuclear data libraries, JEF/JEFF, ENDF/B and JENDL data were used for the investigations. The comparison comprises JEF-2.2 and JEFF-3.1, ENDF/B-VII.0, JENDL-3.2 and JENDL3.3/AC-2008. For the VENUS-7 assembly, the spread of the multiplication factors obtained with these data is approximately 400 pcm. The corresponding resulting pin power distributions are practically identical with differences overall smaller than 1 %.

Reasonable consistency of multiplication factors. For the comparisons of full core results obtained with different nuclear data, the hot zero power uncontrolled state was chosen for the PWR and the hot zero power controlled state for the VVER, because for these states, the differences in the calculated power distributions turned out to be highest. Although a completely controlled core with a multiplication factor higher than 1.0, as it is the case for the VVER state, is not typical for a reactor in operation, it has some relevance, e.g., for accident situations in case of a boron dilution transient. The largest differences in the multiplication factors obtained using these data are approximately 400 - 500 pcm; this seems reasonable regarding the number of different nuclear data evaluations used and is in the same order of magnitude as the differences obtained for critical assembly calculations.



31 RESULTS
Uncontrolled condition of the »PWR MOX/UO_2 Core Transient Benchmark«



32 RESULTS
Controlled condition of the »VVER-1000 MOX Core Computational Benchmark«

COMPARISON: POWER DISTRIBUTIONS

Fig. 31–32
Relative differences of radial fuel assembly power distributions, calculated with JEF-2.2 (top left), JENDL-3.2 (top right), ENDF/B-VII.0 (bottom left) and JENDL-3.3/AC-2008 (bottom right) in comparison with results obtained from the JEFF 3.1 data

Substantial influence of the applied nuclear data bases on radial assembly power distributions. Substantial differences are observed, however, for the radial power distributions. For the uncontrolled state of the PWR core, a tilt in the ratio of the distributions obtained with the different nuclear data libraries from the core centre to the periphery is found, with a maximum value of approximately 5 %. When comparing the results obtained with all nuclear data evaluations used, there are larger differences to JEFF-3.1 with the older evaluations JEF-2.2 and JENDL-3.2 as compared to the most recent evaluations ENDF/B-VII.0 and JENDL-3.3/AC-2008. One reason is the fact that in the newer data, significant changes were made in the ^{235}U capture and fission data; e.g., the ^{235}U capture resonance integral was increased by 5 – 6 % as compared to the older evaluations. Looking at the core load map, one recognizes that in the PWR core the MOX fuel assemblies tend to be located closer to the periphery of the PWR core, leading to a slight shift of the ^{235}U distribution to the core centre. However, there is also some smaller influence from the ^{239}Pu and ^{238}U data. Even larger differences in the power distributions are observed for the controlled, boron-free state of the VVER 1000 core; however, the centre to periphery tilt of the JEF-2.2 to JEFF-3.1 ratio is reversed as compared to the PWR case. A reason for this behavior is that in the VVER core, the UO_2 assemblies are on the average located closer to the core periphery. In the state with inserted absorber rods, most of the UO_2 assemblies located away from the core periphery are controlled, thus shifting the efficient ^{235}U distribution even closer to the periphery (see Fig. 31–32 »COMPARISON: POWER DISTRIBUTIONS«).

Summary and Outlook

Multiplication factors and power distributions with different nuclear data. GRS applied the Monte Carlo code MCNP-5 with continuous energy nuclear data based on JEF-2.2, JEFF-3.1, ENDF/B-VII.0, JENDL-3.2, and JENDL-3.3/AC-2008 for estimating the influence of differences in the evaluations on the multiplication factors and power distributions of large power reactors in operating conditions. The calculations were performed for two two-dimensional UO_2/MOX full core configurations used for calculation benchmarks performed under OECD/NEA auspices.

Results for the multiplication factors. Whereas the resulting multiplication factors were found to be in reasonable agreement (400 - 500 pcm differences), a significant effect was observed for the radial power distributions in certain cases. This effect is specific for mixed cores with UO_2 and MOX with a high amount of fissile plutonium in the isotopic composition and shows up only due to the large size of the cores.

Results for the assembly powers. The differences between the assembly powers obtained with the different evaluations reached 5 % for an uncontrolled, and more than 10 % for a controlled boron-free state. An important contribution to the discrepancies for these cases comes from differences in the ^{235}U cross sections in the different evaluations. Influences of this type can normally not be observed in calculations for compact critical assemblies, which primarily serve as the validation basis for nuclear data evaluations. For instance, practically no difference in the pin power distributions calculated with MCNP using JEF-2.2 and JEFF-3.1 data was observed for the mixed UO_2/MOX critical assemblies VENUS-7.

Conclusions and further development. This suggests that benchmark problems representative for large power reactor cores or even measurements from operation should be considered as much as possible by the nuclear data evaluation groups in the validation process of the libraries. Furthermore, it would be advantageous to have analysis tools available to investigate the influence of nuclear data uncertainties on local quantities like power distributions, with the aim to routinely accompany reactor core calculations by uncertainty studies. To this end, GRS is currently developing extensions to their sampling based uncertainty and sensitivity software package SUSA to handle nuclear covariance data. ∎

4. Reactor safety analyses

Heinz Liemersdorf

In the area of reactor safety analyses, GRS is concerned with tasks the subjects of which are – unlike the research on reactor safety basics as described in chapter 3 – primarily the assessments of specific operational events. These analyses serve the scientific expert consulting of supervisory and licensing authorities. In addition, they continuously improve the GRS experts' state of knowledge by ensuring that up-to-date findings on influencing factors relevant for the safety level of German nuclear power plants are always taken into account.

Reactor safety analyses

In doing so, the superordinate goal is it to maintain this level and to improve it as much as possible by considering the further developments of the state-of-the-art in science and technology. To that end, GRS primarily evaluates national and international operating experiences. In addition, GRS also carries out its own analyses on current safety-related issues regarding the behaviour of the plant or its technical systems during power plant operation or during actually occurring or theoretically assumed safety-relevant incidents. For these examinations, analytical tools such as the analysis simulator or recognised engineering methods are available. This reflects the special relationship between reactor safety research and the reactor safety analyses represented here: On the one hand, we also use methods during our application-oriented analyses which result from our own research activities and which have led to an improved state of the art in science and technology; on the other hand, these analyses often contain application-oriented research activities and provide important information on necessary research developments. This concurrence of research, application and assessment is one of the essential reasons for GRS's special competence in the field of reactor safety (see »OVERVIEW«).

Reactor safety analyses – technical basis for state supervision and licensing

The results of these inspections, which are normally carried out on behalf of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety or the supervisory and licensing authorities of the federal states, are documented differently. There are in particular Information Notices as well as expert opinions, comments and generic reports.

Information notices

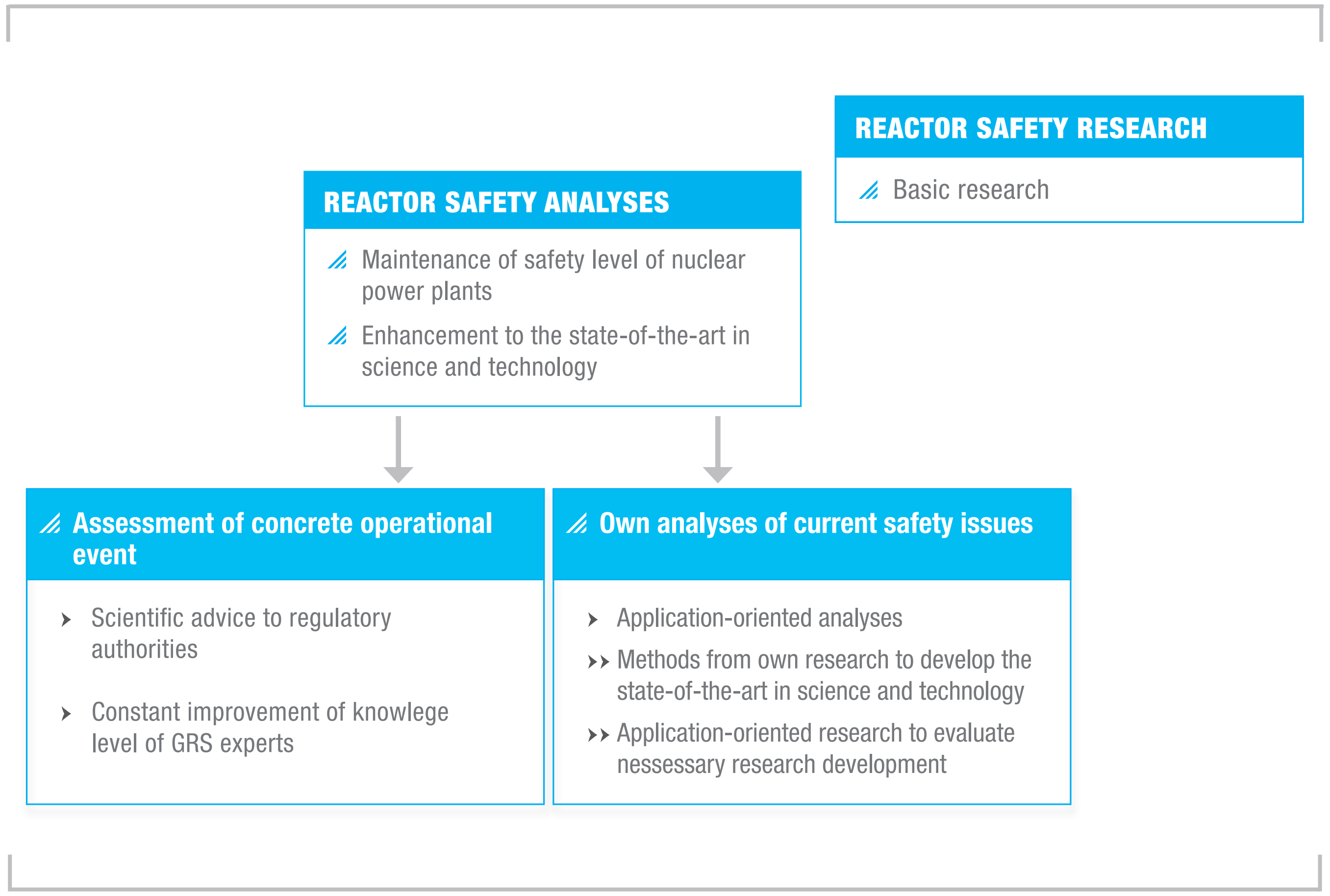
By order of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety, GRS informs the German nuclear power plant operators and the supervisory authorities of the federal states as well as other organisations incorporated into the use of backflow of experience with so-called Information Notices about current, particularly important safety-related findings. These findings are primarily based on the evaluation of operating experience and on recent research results. The Information Notices comprise statements on the applicability of the findings to other plants, on the safety-related significance as well as recommendations for necessary safety-related measures. Approx. 10 to 15 Information Notices are produced within one year. The Information Notices by GRS usually lead to considerable plant-specific inspections of the German nuclear power plants and – depending on the result – to manifold technical and administrative improvement measures. In the following, the contents of several Information Notices of the reporting period will be exemplified:

Examples information notices

Example 1: In the last GRS Annual Report already, a report was given on the event »Fire of a generator transformer in the KKW Krümmel« of the year 2007. Due to the findings gained from in-depth inspections, Information Notices have now been produced which refer to individual aspects of the event. One aspect was related to the input of fire gases into the control room. The result of our inspections was that similar plant situations could lead to comparable event sequences in other German nuclear power plants as well. In the Information Notice, recommendations were made in respect of measures to avoid such event sequences and to improve the venting concept in other nuclear power plants. Further aspects dealt with in the Information Notice were the short circuit in the generator transformer, the failure of the operational feed of the reactor pressure vessel and a faulty data storage: Here, GRS recommended, amongst other things, the application of modern test and monitoring methods for the assessment of transformers which are filled with oil, an improvement of the connection of feedwater pumps and more reliable data processing in the event of the simultaneous presence of a large number of signals (so-called plethora of signals).

Example 2: In another Information Notice, the significance of unintended impacts on safety devices due to electromagnetic interference was pointed out. In the underlying case, arc welding activities in the area near an emergency diesel generator had caused electromagnetic interference. This interference led to the unavailability of this generator during power operation of the plant. Our recommendations aimed at maintenance measures being carried out during power operation of nuclear power plants only when a disturbance of safety devices through electromagnetic interference can be reliably avoided.

The competence of GRS in the field of reactor safety
Interplay of research, application and assessment



Example 3: One Information Notice was related to crack indications in casings of safety-relevant valves made of stainless steel which were found during reviews. The development of these crack indications was caused by stress corrosion cracking due to chloride ingress. This is a damage mechanism common for these materials which was observed at the affected locations for the first time, though. GRS recommended to not only review comparable valves in particular, but to also include other safety-relevant components made of stainless steel in an inspection programme the focus of which is the identification of this specific kind of damage. In case of indications, appropriate corrective actions must be taken as the barrier »reactor coolant pressure boundary« is affected.

Example 4: Another Information Notice which should be emphasised dealt with crack indications at steam generator tubes. The remarkable thing is that the indications had not been detected until a new test method was applied. With the previously applied test methods, indications of the type detected, i.e. extent-related indications were difficult to identify. GRS recommended to test the steam generator tubes in other plants with the new test method as well. The tests ought to be focused on those parts of the steam generator tubes which are particularly susceptible to developing cracks.

Expert opinions, comments and generic reports

The expert opinions, comments and generic reports elaborated by GRS were mainly related to safety-relevant assessments of events at home and abroad. Key activities were the deepened technological assessment of national and international operational experiences on behalf of the BMU. Furthermore, generic examinations extending beyond individual incidents or plants were carried out and national and international findings on the nuclear safety of nuclear power plants and research reactors were collected, the data were edited and scientifically analysed. Some of the key aspects of these activities will be described in the following:

Examples
Expert opinions, comments and generic reports

Example 1: In 2008, there were several events in France and Belgium which – in particular due to the coverage in the media – gave reason for GRS to provide expert opinions or information. With the exception of one event, fuel cycle facilities and one facility for the production of radioactive medical materials were concerned where radioactive materials were released into the environment of the plants. Although these events did not directly pertain to reactor safety, examinations were carried out in respect of safety-relevant findings which could be applied to nuclear power plants.

During another event in Unit 4 of the French nuclear power plant Tricastin on 23 July 2008, activation products of metal materials (Co-58) of the reactor coolant system were released from a pipe into the reactor building with the reactor fully unloaded. The pipeline was made of plastic pipes which were used for the evacuation/flushing of the reactor coolant system. Leakages at the pipe joints led to a release of aerosol into the reactor building. At the time of release, extensive maintenance activities were being carried out. This caused a contamination of operating personnel which was, however, well below the annual permitted value of 20 mSv. The event was classified as INES Level 0 because it had no consequences for the personnel or the environment.

Example 2: A coolant leakage in the nuclear power plant Krsko in Slovenia in early June 2008 also aroused increased public interest. Due to the inappropriate reaction on the part of the Slovenian supervisory authority, this event was reported via the ECURIE system (European Community Urgent Radiological Information Exchange) and had therefore caused concern at first. In fact, however, the safety-related significance of the event during which a leakage of reactor coolant into the containment occurred at a temperature probe valve was low. No safety systems were challenged. A few days after the defective valve seal had been repaired the plant was restarted again.

Example 3: In an additional comment by GRS, the failure of so-called flywheel generators of the coolant recirculation pumps in both a Swedish and a Finnish boiling water reactor were assessed in respect of their applicability to German plants. In both nuclear power plants a failure of these flywheel generators, which are supposed to ensure a prolonged coast down of the coolant circulation pumps in case of electrical problems, occurred. In both events no damage was caused to the fuel elements. There are no such flywheel generators used in German nuclear power plants. However, the electric problems which occurred during the events are relevant for the operating experience of German nuclear power plants, too, and are thus considered further within the scope of an in-depth examination.

Example 4: Regarding special expert assessments carried out by GRS, examinations regarding the controllability of a loss-of-coolant accident in case of an input of insulating material into the sump of the containment have to be mentioned again for this reporting period. In recent years, GRS has repeatedly pointed in technical comments to necessary developments regarding safety-related requirements. From the perspective of GRS, no closed »robust« evidence for the controllability of a loss-of-coolant accident with subsequent release of fibrous insulation material has been provided so far. Therefore, GRS developed and introduced a concept in 2008 which from our point of view meets the demands connected with this objective.

One of the essential requirements for the controllability of such a loss-of-coolant accident is the call for a nearly insulating-material-free core. The possible impacts of a core being entirely blocked cannot be limited in a sufficiently safe manner, neither by measures of accident control nor by accident management measures. This results in the requirement that insulating material and other materials, if applicable, must be safely retained at the sump strainer at the latest. From GRS's point of view, a noticeable penetration, in particular by fibrous insulating material, through the sump strainer must be avoided by all means.

In experiments abroad, significant impacts of chemical reactive materials on the pressure loss via the sump strainers was detected during the residual-heat removal phase of such an accident. From our point of view, no tests with materials to be considered in German PWR facilities have yet been carried out systematically or to the required extent. A significant impact of chemical reactive materials on the pressure loss via the sump strainers and the core must be ruled out. The verification must take the entire period of a possible residual-heat removal phase up to the point where the fuel elements are unloaded from the core into account. The period to be examined ought to be at least several months.

Meanwhile, the operators of the German nuclear power plants held out the prospect of modifying the German PWR facilities currently in operation. In doing so, the diameter of the strainer openings is to be markedly reduced in order that the possible input of fibrous material into the core is safely limited to an uncritical value.

Preparing the decision-making bases regulatory for inspections and assessments

In connection with applications by three operators for transferring residual electricity volumes from newer to older nuclear power plants, GRS - together with one further expert organisation specified by the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety - was commissioned with a comparative safety review. This order deals with the selective examination of individual assessment items jointly specified by the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety, GRS, and other participating organisations. Assessment factor is the current state of the art in science and technology. In accordance with the order, particularly the plant-specific safety margins are included in the assessment in addition to the degree to which the regulatory safety requirements have been met. In the reporting period, GRS conducted examinations on the safety-related comparison of the Brunsbüttel and Krümmel NPPs. In accordance with the order, individual reports on 12 assessment items were prepared in co-operation with the Ökoinstitut e.V. The procedural method basically corresponded to the one developed during earlier safety-related comparisons of the Biblis-A and Emsland as well as the Neckarwestheim 1 and 2 nuclear power plants. ∎

4.1
Evaluation of recent findings on the integrity of pressurised components in nuclear power plants

Dr. Frank Michel

Hans Reck

For more than 30 years, GRS has been concerned in the area of reactor safety analyses with issues regarding the integrity of pressurised components used in nuclear power plants. One essential tool for this is the generic evaluation of operating experience. Within the scope of current activities, GRS examined by order of the BMU if and to which extent service-induced flaws affecting pressurised components can be detected in time by means of in-service inspections. To that end, the relevant state-of-the-art in science and technology on non-destructive in-service inspections of pressurised components was identified and a generic evaluation of the respective national and international operating experience was carried out. Relevant characteristics of selected individual events were analysed in-depth in respect of flaw characteristics and safety relevance. From the results of these activities, conclusions can be drawn for the concepts of non-destructive in-service inspections of German plants. In addition to the findings on flaw detection, two safety-relevant key aspects could be determined which are connected to environmentally-assisted cracking and for each of which a package of measures was recommended which GRS considers adequate.

Findings on flaw detection

Reasons for problems in connection with fault detection. Due to service-induced degradation mechanisms, problems can develop in pressurised components. To review the condition of pressurised components, non-destructive in-service inspections are conducted at random. It cannot be totally ruled out, however, that service-induced flaws are not detected within the scope of these inspections. As the evaluations carried out by GRS have shown, problems with flaw detection can be connected to the following aspects in particular:

- shortcomings relating to geometry and surface condition of the component,
- inadequate documentation of previous inspections,
- faults made by the inspector,
- faults during the evaluation of the inspection,
- shortcomings of the test procedure applied or of the test technology.

Guidance for correct fault detection. To avoid potential problems during flaw detection, the German nuclear regulations and the technical standards to be consulted contain a set of guidelines. Due to the aforementioned potential problems during flaw detection, it cannot be completely ruled out that even despite adherence to these guidelines, flaws or the growth thereof are not detected within the scope of non-destructive in-service inspections. As a basic principle, the exact boundary conditions – in particular in comparison with previous tests – should be challenged prior to test performance in order to prevent potential problems. To identify potential problems with the detection of flaws as soon as possible and to largely eliminate them, it is necessary from GRS's point of view to consistently answer the questions mentioned below when planning the tests:

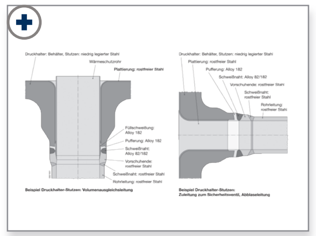
- Can falsifying influences caused by geometry and surface condition be ruled out during the test?
- Which human errors can occur during the evaluation and documentation of the tests and how can they be avoided?
- How is the informational value of the specified test procedures and technologies to be assessed against the background of technical progress?

Findings on flaws in bi-metallic welds

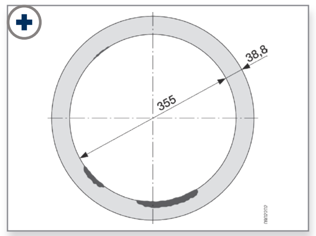
Structure of bi-metallic welds. Different types of steel, e.g. components made of ferritic steel and pipes made of austenitic steel, are connected to each other by means of bi-metallic welds. The bi-metallic welds are composed in a way that a buffer, e.g. made of a nickel alloy or of austenic steel, is welded on the base material made of ferritic steel. The weld metal of the girth weld consists of either austenic steel or a nickel alloy. Fig. 33 shows, as an example of such a connection, a schematic representation of the pressuriser nozzles in the American Wolf Creek nuclear power plant.

Based on operating experience in plants abroad, flaws in bi-metallic welds were identified which, due to an active degradation mechanism, show the potential for through-wall cracks. In addition to axial flaws running perpendicular to the progress of weld, flaws running parallel to the circumferential welds occurred.

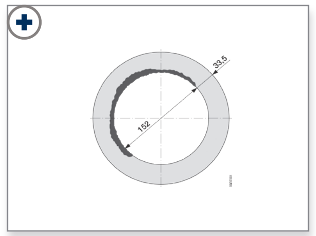
4.1 Evaluation of recent findings on the integrity of pressurised components in nuclear power plants
CONTENT



33 STRUCTURE OF BI-METALLIC WELDS IN THE NOZZLES



34 CRACK INDICATIONS IN THE SURGE NOZZLES



35 CRACK INDICATIONS IN THE RELIEF NOZZLES

CRACK INDICATIONS IN BI-METALLIC WELDS

Fig. 33-35
Structure of the bi-metallic welds and crack indications in the pressuriser nozzles of the US Wolf Creek nuclear power plant

Fault detection at the US Wolf Creek nuclear power plant. If flaws running in circumferential direction are not detected in time, then generally breaks must be taken into account as well. Therefore, such flaws must be identified early and with high certainty. Our evaluations have shown that the flaws which became known so far were not all detected within the scope of scheduled in-service inspections. In autumn 2006, for example, non-destructive tests were carried out on several pressuriser nozzles in the American Wolf Creek nuclear power plant as a preparation of a so-called overlay weld on the outside surface of the bi-metallic welds. The tests showed indications in three bi-metallic welds (see Fig. 34 and 35 »CRACK INDICATIONS IN BI-METALLIC WELDS«). These flaws had not been detected within the scope of a previous, scheduled in-service inspection because the test technology applied then was inappropriate for the identification of this type of flaw. The flaws were caused by intergranular stress corrosion cracking. The sensitivity of nickel alloys in contact with hot water to this particular degradation mechanism has been known for a long time. Residual stresses which are primarily caused by repair welding also play an essential role.

Inspection according to KTA Safety Standards in Germany. In German plants, the integrity of bi-metallic welds is ensured by means of in-service inspections in the form of non-destructive tests which are carried out according to the applicable KTA standards. Due to the above-mentioned degradation in foreign plants, the applicability to German plants was reviewed and basically confirmed since – despite partially different designs – this kind of welding material is also used in safety-relevant bi-metallic weldings in German plants. Thereupon, the operators developed and carried out an extensive supplementary test programme for the identification of flaws in bi-metallic welds. So far, no flaws due to intergranular stress corrosion have been detected.

Standards for inspecting bi-metallic welds. As far as the performance of in-service inspections of bi-metallic welds is concerned, the following two aspects have to be considered:

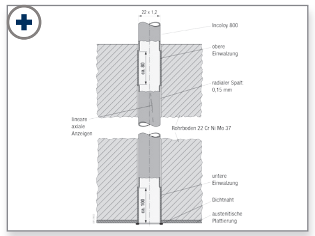
According to KTA 3201.4, ultrasonic examinations of the external surface are specified for the identification of flaws on the internal surface. With this test method, however, flaws caused by intergranular stress corrosion may not be detected: In particular at the early stage of growth, such cracks do not reflect the sonic waves due to the then still small size of the crack opening. In addition to that, possible difficulties in testing may occur as a result of the dendritic crystallisation in the weld metal. Basically, tests of internal surfaces, e.g. by means of eddy current, would be an adequate means to identify flaws. This is opposed, however, by the lack of accessibility for such tests.

Research and development have not produced sufficient information on incubation time and crack growth rates which would allow for reliable conclusions on the occurrence and progress of intergranular stress corrosion of nickel alloys. It can therefore not be ruled out that, because of corresponding incubation periods, degradation will not occur until after long-time operation. Also, it cannot be ruled out that in the period between the scheduled test cycles an initial crack with a subsequent fast crack growth will start to develop.

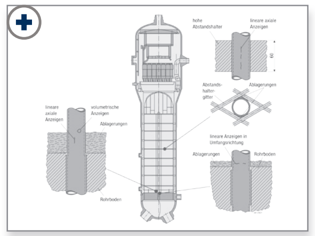
Optimisation of test methods, scope and cycle. The aspects described require a consistent concept for non-destructive in-service inspections. As far as the optimisation of test methods, test scope and test cycle are concerned, GRS considers the following starting points:

a) Test methods. Non-destructive test methods or technologies which are carried out from the external surfaces should be assessed in respect of the certain identification of flaws. This applies in particular to their initial stage of development. To that end, in particular the detected flaws should be analysed and tested by means of diverse test methods. Furthermore, the application of non-destructive test methods which are conducted from the internal surfaces should be examined.

b) Test scope and test cycle. Both test scope and test cycle should be plant-specifically adjusted in correspondence with the design and manufacturing boundary conditions as well as the respective findings on the degradation mechanism. In doing so, the early identification of potential crackings by means of tests in representative locations has to be ensured. The knowledge about incubation periods and crack growth in case of intergranular stress corrosion in bi-metallic welds still have to be enhanced, taking into account recent information resulting from research and development as well as operating experience. In correspondence with the latest findings, all bi-metallic welds with weld metal that is susceptible to corrosion in contact with the medium should be given particular attention with a representative test scope.



36 DEFECTS IN THE ROLLER-EXPANDED PORTION OF THE TUBE SHEET



37 DEFECTS ABOVE THE TUBE SHEET

DEFECTS IN STEAM GENERATOR TUBES

Fig. 36–37
Defects in steam generators in German 2nd-generation PWR nuclear power plants

Findings on fast-growing flaws in steam generator tubes

Operation experience. The steam generator tubes made of the material Alloy 800 which are used in all German nuclear power plants with pressurised water reactors have proved themselves so far. To date, no breaks of steam generator tubes have occurred in German plants yet. Local degradation led to leakages in only seven steam generator tubes. The leakage volume was relatively low and in only two cases - which occurred in the 80s - they amounted to more than 3 litres per hour. In Germany, the number of times where steam generator tubes had to be plugged due to flaws is very low compared to plants abroad. The operating experience which has been good so far does not rule out, however, that in the course of operation individual, unnoticed, fast-growing flaws can occur which in the end can cause steam generator tube leakages. Such flaws were detected for the first time in 2005 and after that within the scope of non-destructive tests (see Fig. 36 and 37 »DEFECTS IN STEAM GENERATOR TUBES«). In a total of two German 2nd generation nuclear power plants. It is assumed that the degradation was caused by intergranular stress corrosion.

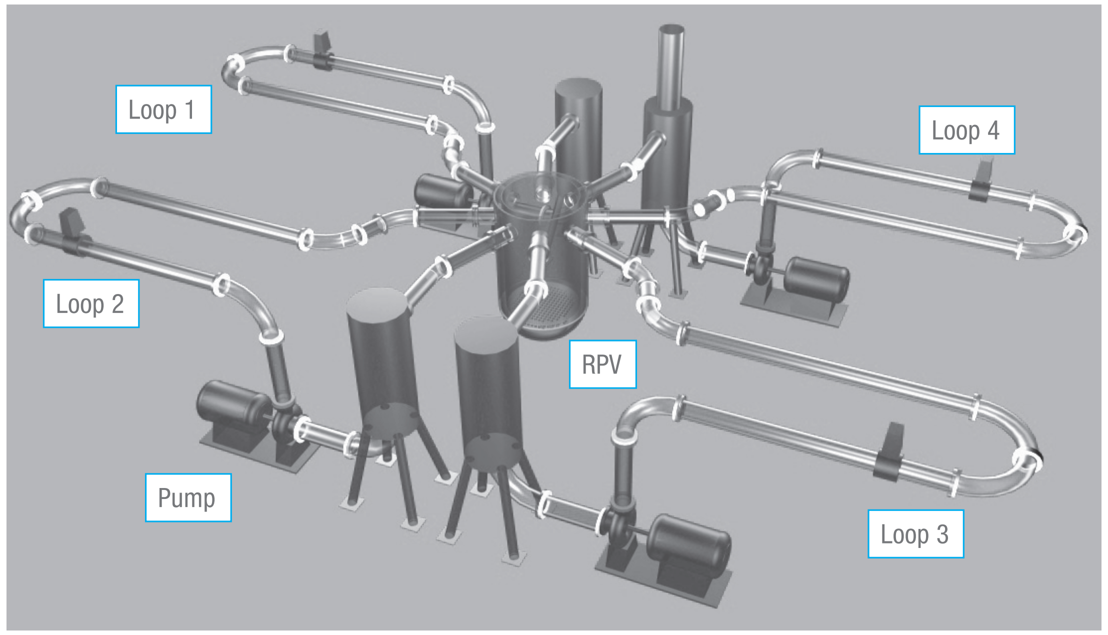
Therefore, an overall concept is required to ensure the integrity of steam generator tubes in German plants. GRS sees a set of starting points for such a concept. They pertain to the enhancement of knowledge about the degradation mechanism and the implementation of different measures to avoid degradation, the identification of locations susceptible to corrosion, analysis of flaws that have occurred, and the optimisation of the test concept as well as the applied test procedures and technologies. These approaches should be pursued in correspondence with the respective plant-specific conditions. ■

4.2
Test and validation of tools for the simulation of boron dilution accidents

Joachim Herb

Dr. Wolfgang Horche

Boric acid is used as soluble neutron absorber in the reactor coolant system of pressurised water reactors. If the boron concentration within the core drops below a critical value during an accident, recriticality occurs with an increase of reactor power. In this case, core damage cannot be ruled out if prompt criticality occurs due to a rapid and heavy drop of the boron concentration. Essential factors in case of boron dilution accidents are the accumulation and volume of low-borated coolant, the transport and mixing of the low-borated coolant with the borated coolant on their way to the core as well as the core behaviour in respect of increased power and heat removal. To be able to better predict the transport and mixing, numerous tests were carried out in the past, amongst others using the ROCOM test facility at the Research Centre Dresden-Rossendorf (FZD). By order of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) and the Federal Ministry of Economics and Technology (BMWi), GRS examined the applicability of different simulation tools regarding this issue. To that end, the results of the simulations were compared with test data. It could be seen that the simulation tools describe the mixing phenomena qualitatively well. Statements on the accurateness of the simulation predictions could be derived from the quantitative comparison of the results. These can serve as bases of decision when deciding in the future on whether tests may be substituted by simulations.



OVERVIEW: ROCOM TEST FACILITY

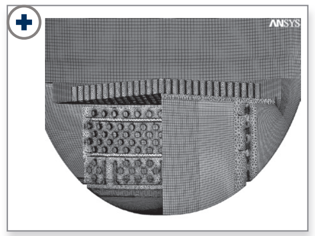
Fig. 38
The ROCOM test facility at the Dresden-Rossendorf research centre
(Source: Höhne et. al., Nuclear Engineering and Design 236 (2006) 1309-1325)

Interruption of natural circulation. If an accident involving a small leakage from the reactor coolant system of a pressurised water reactor (PWR) occurs, a failure of the safety systems can lead to an interruption of single-phase and two-phase natural circulation. In reflux condenser mode, the decay heat is then discharged from the reactor core to the steam generators. In doing so, a demineralised water plug may develop in the cold leg of the reactor coolant system. This plug contains little or no solute boric acid. When natural circulation sets in again, the demineralised water plug is transported into the reactor pressure vessel (RPV). If the demineralised water plug is not sufficiently mixed with the rest of the borated coolant, recriticality of the reactor core may occur.

Analysis of mixing under different boundary conditions. To analyse the mixing of the coolant on its way into the reactor pressure vessel and there in the area of the downcomer and the lower plenum, a test facility was erected at the Research Centre Dresden-Rossendorf (see Fig. 38 »OVERVIEW: ROCOM TEST FACILITY«), which represents the reactor coolant system of a modern German pressurised water reactor on a linear scale of 1:5 (with the volume on a scale of 1:125). A multitude of tests were conducted there during which the mixing under different boundary conditions was analysed. So the influence of density differences of borated and non-borated coolant was modelled by using different fluids. As another important influencing factor, the acceleration of the coolant dur-

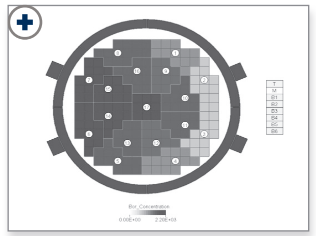
POSITION OF THE MEASURING LEVELS

Fig. 39
Position of the measuring levels in the ROCOM test facility



CFD-SIMULATION: COMPUTATIONAL GRID

Fig. 40
Computational grid of the CFD simulations in the area of the downcomer and the lower plenum



MULTICHANNEL REPRESENTATION

Fig. 41
Multichannel representation of the 17 core channels and 8 downcomer channels; the two sections through which the demineralised water plug is transported into the RPV are located on the right

ing restart of single-phase natural circulation was examined. The mixing was measured by means of a tracer fluid, which was added to the coolant and the concentration of which was determined in the test. These measurements were carried out at different measuring planes: at the inlet nozzle of the RPV, at two levels in the area of the downcomer, and in the inlet ports of the reactor core (see Fig. 39 »POSITION OF THE MEASURING LEVELS«).

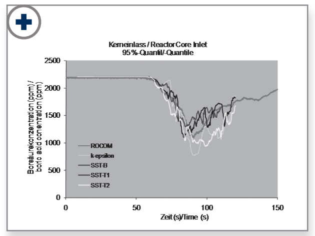
Comparison of the simulation tools. For the comparison of the simulation tools and the experiments, a test was chosen during which the restart of natural circulation was simulated in two legs. In the other two legs, there was a constant flow rate in accordance with natural circulation conditions.

The simulations were carried out by means of the commercial Computational Fluid Dynamics (CFD) code package ANSYS CFX and the GRS code ATHLET.

CFX-Modell

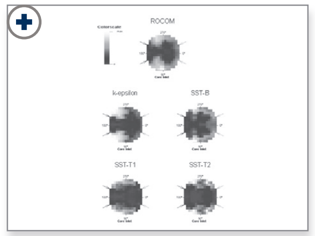
ANSYS CFX. GRS had already carried out CFD simulations of the (scaled) ROCOM test facility in the past. As on the one hand, the further development of computer capacities allowed for bigger computation grids and on the other hand, the impact of the scaling on the results was of interest, a computational grid on the scale of 1:1 of a modern German PWR was prepared for the simulation introduced here. The computational grid consisted of approx. 6.9 million computational cells (see Fig. 40 »CFD-SIMULATION: COMPUTATIONAL GRID«). A typical simulation of the time history of a boron dilution accident thus took approx. 2,500 CPU hours.

To ensure the applicability of the experiments or simulations of a scaled model to the real plant, the dimensions and boundary conditions of the test facility were chosen such that the various dimensionless parameters such as Reynolds number,



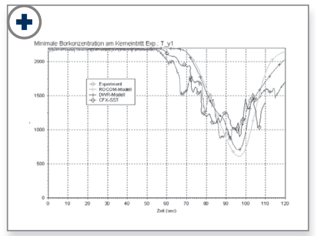
TIME HISTORY

Fig. 42
Time history of the 95 % quantile. The graph shows the time histories of the experiment (ROCOM) and of the different simulations.
Different turbulence models (k-ε and SST) were used, and different boric acid density models were applied for the SST model
(SST-B: boric-acid concentration, SST-T1/T2: temperature).



SPATIAL DISTRIBUTION

Fig. 43
Spatial distribution of the maximum decrease (yellow) in boric acid concentration in the area of the core inlet.
The red arrows indicate the position of the two pipe sections through which the demineralised-water plug is transported into the RPV.



MEASURANDS AND RESULTS

Fig. 44
Minimum boric acid concentration at the core inlet.
The graph shows the measured values of the experiment as well as the results obtained with the PWR model, the ROCOM model and the CFX code.

Strouhal number and Froude number showed comparable values. So when transferring speeds and density differences from the real plant to the scaled model, they were reduced to one fifth of the value in the real plant. In the CFD simulations, the further boundary conditions were selected according to the conditions which in case of an accident with small leakage were to be expected in the real plant (approx. 160 °C coolant temperature, 15 bar primary system pressure). To limit the required calculation time, the symmetry of geometry and boundary conditions were taken into account, and only two legs of the reactor coolant system as well as half of the RPV were simulated as a computational grid.

Impact of turbulence model on simulation results. The choice of the turbulence model can have a substantial impact on the results of a CFD simulation. Therefore, simulations with two of the models implemented in ANSYS CFX were carried out: one with the k-ε model, with which turbulent kinetic energy and its dissipation are determined, and the other one with the SST model. This is an extension of the k-ε model, in particular for areas close to a wall.

Two different calculation methods. To take the impact of the density of the boric acid and the accompanying buoyancy effect into account, two different simulation approaches were tested. One initial approach was the calculation of the local density of the cooling medium on the basis of the simulated boric acid concentration. The second calculation method was based on the use of different coolant temperatures to simulate the different densities.

ATHLET models

Two models were developed on the basis of ATHLET:

- One model of a modern German PWR (scale 1:1) based on a plant-specific GRS analysis simulator.
- One model of the ROCOM test facility (scale 1:5) by means of reports and drawings at hand.

The division of the 193 fuel elements into 17 core channels was identical for both models. In Fig 41 »MULTICHANNEL REPRESENTATION« the 8 downcomer channels are shown as well. Approx. 90 s after the restart of natural circulation, the set colours show a boron concentration of 760 ppm at the inlet of core channels 2 and 3.

Boric acid concentration. The concentration of boric acid inside the individual cooling channels of the reactor core is a decisive factor for the safety-related assessment of a boron dilution accident. Due to the partial mixing of the demineralised water plug with the borated coolant on the way through the cold leg of the reactor coolant system and within the RPV, different time histories of the boric acid concentrations were observed in the individual cooling channels.

Results of the analyses

CFD simulations. To be able to compare the results of the CFD simulations with the test, a statistical evaluation of the boric acid concentration in the overall 193 cooling channels was carried out. First, the concentrations inside the individual cooling channels were determined for each time step of the simulation. Then, different statistical parameters such as the arithmetic mean, median, the 95 % quantile and maximum value were determined. Finally, the time histories of these parameters for the simulation and the experiment were compared. In doing so, the 95 % quantile is of particular interest: it indicates the boric acid concentration which is exceeded in 95 % of the cooling channels and which is not reached in 5 % of the cooling channels. This parameter can be used as an indicator for the occurrence of recriticality because for this, the boric acid concentration must drop below a critical value in more than one cooling channel.

Comparison of experiment and simulation. The comparison of the different simulation models with the experiment Fig. 42 »TIME HISTORY« has shown that depending on the model assumptions used for the 95 % quantile of the boric acid concentration, the simulations reflect the experimental results with a maximum deviation of approx. 10 % to 30 %. With this, a noticeable impact of the turbulence modelling could be detected. The simulation using the k-ε model showed significantly higher deviations than the simulation with the SST model. In one simulation (SST-T2), the difference in density between the demineralised water plug and the normally borated coolant was varied. This also caused a stronger deviation between experiment and simulation. For the two simulations (SST-B, SST-T1) with the SST turbulence model and boundary conditions in accordance with the experiment, a deviation of approx. 15 % was detected for the minimum of the boric acid concentration.

A comparison of the spatial distribution of the boric acid concentration (see Fig. 43 »SPATIAL DISTRIBUTION«) showed that the CFD simulations reflect the behaviour qualitatively correctly. Thus, the area of the reactor core where the boric acid concentration declines can be identified.

ATHLET models. The minimum boron concentrations at the core inlet which were determined with both ATHLET models in comparison to the CFD analyses and the experiment are represented in Fig. 44 »MEASURANDS AND RESULTS«. It can be stated that there is relatively good agreement.

In respect of the area of the reactor core where the strongest decline in boric acid concentration occurs, slight deviations were observed. The experimentally determined minimum boron concentration at the inlet to the core channels 4 and 5 amounts to approx. 770 ppm. Compared to that, the minimum boron concentration calculated with the ROCOM model at the inlet to core channels 2 and 3 was approx. 600 ppm; with the original model, an approx. 700 ppm were calculated. In the case of the 5 experiments simulated with ATHLET, with increasing difference in density the minimum boron concentration moved with a left-hand twist to the core channels 6 and 7, which are located opposite to the injection port (see Fig. 41 »MULTICHANNEL REPRESENTATION«).

The short calculating time of less than 5 minutes allows a multitude of projections and confirmatory analyses. The ATHLET models are thus a conservative and calculation-time-efficient method suitable for initial assessments.

Best-Practice Guidelines (BPG) for CFD simulations. Although computer capacities have continuously grown in the past, they are still a limiting factor for the operational capability of CFD simulations. So it was not possible in this project to fulfil all requirements of the »Best-Practice Guidelines« (BPG) for CFD simulations. This applied in particular to the grid resolution, the increase of which would have led to even longer calculation times. Therefore, no evidence could be provided that the results of the calculations were independent of the grid resolution. A refining of these analyses is required in the future when extended computer capacities are available. Another problem at which a closer look should be taken is the impact that the modelling of the density differences between boric acid and demineralised water has on the simulation result.

Summary

More precise predictions though supplementary experiments. CFD simulation tools are available to answer the questions which arise in connection with boron dilution accidents (concentration of boric acid, distribution inside the reactor core). With their help, qualitatively good results could be achieved. The quantitative comparison between experiment and simulation showed – depending on the modelling approach – deviations of up to approx. 30 % for the boric acid concentration. If a more precise forecast is required, the CFD simulations would have to be supplemented by appropriate tests. As computer capacities keep being development, it can be expected that this uncertainty range will be further reduced in the future. ■

4.3 International developments regarding the defence-in-depth concept

Robert Grinzinger

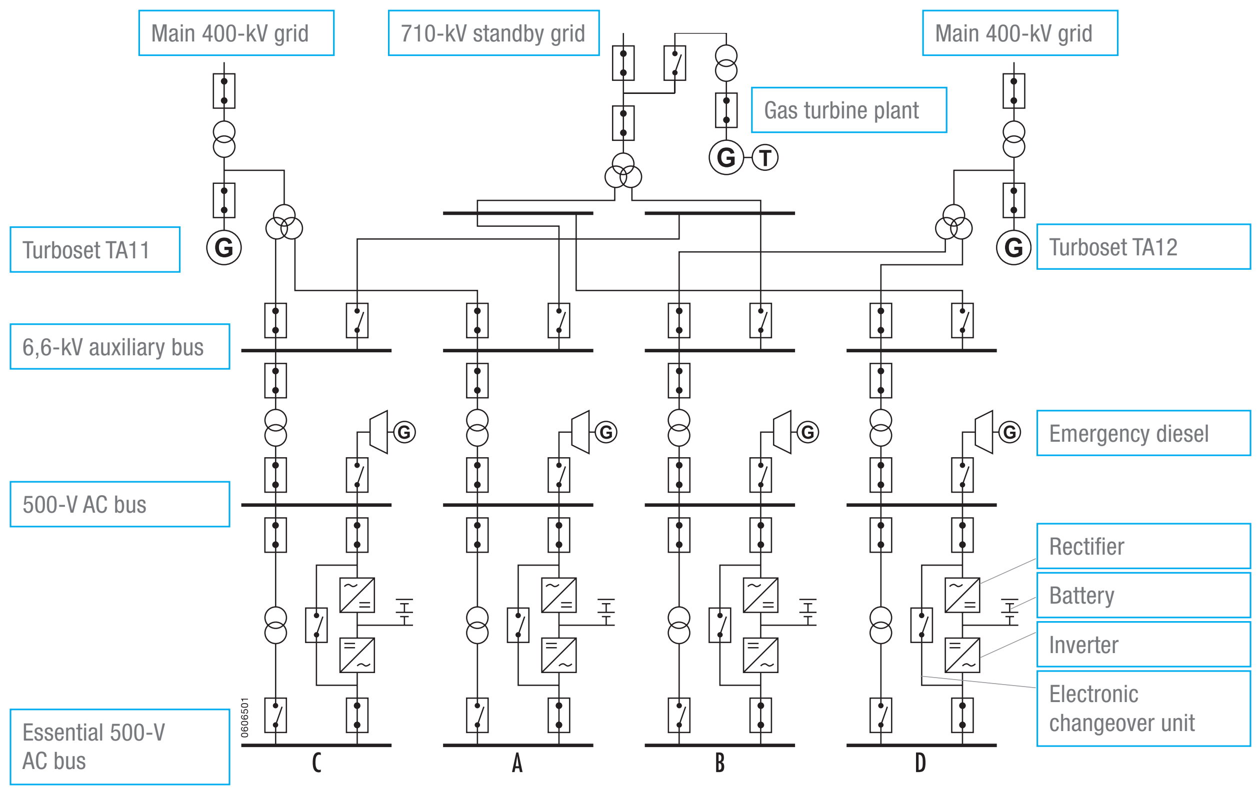
An event at the Forsmark-1 nuclear power plant in Sweden in July 2006 caused the further development of the defence-in-depth concept regarding the electrical power supply of nuclear power plants. At international level, the Committee on the Safety of Nuclear Installations (CSNI) of the Nuclear Energy Agency of the OECD (OECD/NEA) founded a working group, the essential results of which will be described in the following after a short description of the event and its significance.

Description of the event

Plant details. Forsmark-1 is a pressurised water reactor with an electrical output of 1,011 MW. The plant was erected by ABBATOM and has been in commercial operation since the end of 1980.

Fig. 45 »CIRCUIT DIAGRAM« shows a simplified electrical circuit diagram of Forsmark-1. The station service system and the emergency power system each have four redundant trains. Normally, the electrical power supply is provided by the two turbine generator sets, i.e. the 400 kV main grid or the 70 kV standby grid. In addition, the gas turbine plant offers another possible power supplye.

Event sequence. The event was caused by a short circuit in the 400 kV main grid outside the premises of the plant. This led to a loss of the main grid supply and caused a so-called load dump to station service supply. The plant was disconnected from the main grid by opening the corresponding circuit breakers. After the plant had been disconnected, the power supply was at first provided by the two turbine generator sets. Due to disturbances, the generator turbine sets failed after a short time and the power supply was switched to the 70 kV standby grid. This changeover was delayed due to a technical problem and caused the challenge of the emergency power criteria. This resulted in the 6.6 kV station service bus being disconnected from the 500 V AC bus and the start-up of the emergency diesels. In two of the four redundant trains of the emergency power system, the emergency diesels were not connected, which led to the failure of two of the four trains of the emergency power system.



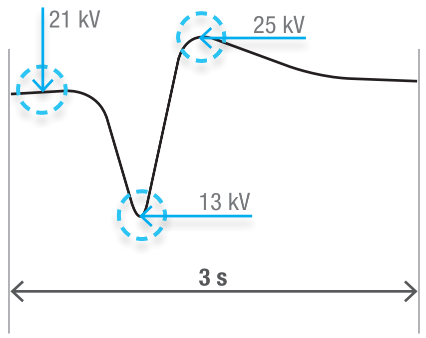
CIRCUIT DIAGRAM

Fig. 45
Simplified electrical circuit diagram of the Forsmark-1 boiling water reactor (BWR)

The plant could be stabilised with the installations available. After approx. half an hour, the shift personnel managed to restore the entire electrical power supply.

Cause and initiator of the event. The failure of the two emergency diesels was caused by the voltage transients which occurred during the event. Fig. 46 »VOLTAGE SHAPE« shows the voltage shape at the generator terminals during the first seconds of the event. Starting at 21 kV during undisturbed operation, the voltage dropped due to the short circuit in the main grid to approx. 13 kV. After disconnection from the main grid supply, i.e. during load dump to station service supply, the voltage increased within a few 100 ms to 25 kV before it normalised again in the further course of the event. This voltage transient propagated via the station service system into the emergency power system where it caused the protective tripping of the rectifiers and the DC/AC converters in two redundant trains. The DC/AC converters supply the secured 500 V AC bus which, in turn, supplies those consumers which are required for the connection of the emergency diesel units.

4.3 International developments regarding the defence-in-depth concept CONTENT



VOLTAGE SHAPE

Fig. 46
Voltage shape during the first seconds of the event on the generator terminals

Thus, the failure of the 500 V AC bus in two of the four redundant trains caused the failure of the two emergency diesels.

The protective tripping of the rectifier and the DC/AC converters can be ascribed to the systematic faulty setting of the protective trip values. The voltage transient at the rectifier input caused an increase of the output voltage of the rectifier and the input voltage at the DC/AC converter, respectively. This resulted in the protective trip value being reached for both the rectifier and the DC/AC converter (rectifier voltage HIGH; DC/AC converter voltage HIGH). Since the distance between the two protective trip values was too small, a selective trip was not provided.

The defence-in-depth concept in electrical power supply

Target and requirements of the concept. The target of the so-called defence-in-depth concept is to stave off the failure of one protective measure on one level by means of protective measures on the next level. This concept is supplemented among other things by requirements for high quality and fault control measures on the individual levels. In respect of the electrical power supply of nuclear power plants, this results in the following requirements:

- The operating materials used are characterised by a high degree of robustness (e.g. against electrical transients and electromagnetic interference).
- The operability is verified by means of periodic tests and inspections.
- The personnel are well-trained and adequately equipped.
- The safety systems of the electrical power supply are automated, redundant and highly reliably.
- The design takes faults such as short-term voltage losses into account.

To avoid impermissible safety-relevant impacts of grid transients, a robust grid connection is required as the first measure. To avoid a loss of offsite power during a loss of the main grid supply, the measures load dump to station service supply and switching to stand-by grid are available; and for the control of the loss of offsite power, the emergency diesel units are available. Should – in a beyond-design situation – the emergency diesel units fail as well, the plant is in the state of so-called »station blackout«. To avoid impermissible safety-related impacts during such a scenario as well, the battery-supported DC power supply and the third grid connection are available.

Generic significance of the event

Safety-related consequences. The generic significance of the event at Forsmark in July 2006 consists in the fact that a short circuit outside the power plant caused the failure of two redundant trains of the emergency power system due to a systematic faulty setting within the plant. In the run-up to the event, further deviations from the specified condition (disturbances at the turbosets, delayed switching to stand-by grid) led to a failure of several measures to avoid the loss of offsite power.

Due to the generic significance of this event, GRS wrote an information notice. In it, GRS recommends, among other things, to ensure that fault-induced voltage transients will not lead to impermissible impacts on safety-relevant electrical equipment. In addition, the German Association of Large Power Plant Operators (*Verband der Großkraftwerksbetreiber* - VGB) set up the »Forsmark« working group. This working group examines to which extent the electric energy supply of nuclear power plants can be optimised.

DIDELSYS task group

Defence in Depth of Electrical Systems and Grid Interaction (DIDELSYS). At international level, the CSNI of OECD/NEA established the »Defence in Depth of Electrical Systems and Grid Interaction« (DIDELSYS) task group in which a representative of GRS also participated. The function of the task group was to study two key issues in depth and prepare a corresponding report. On the one hand, this concerned the robustness of safety-relevant electrical systems. To that end, information was to be provided on the state of the art in science and technology, taking into account recent technologies and experiences gained from modernisation measures. On the other hand, regarding the interface between power plant and grid system, measures to improve communication and co-ordination between grid operators and network supervision, the nuclear authorities and power plant operators were to be pointed out.

Task Group composition. Taking part in the task group were representatives from the EU as wells as from organisations (names given in brackets) of the following nine countries: Belgium (Nuclear Safety Support Services), Finland (STUK), France (IRSN), Germany (GRS), EU (Joint Research Centre), Japan (Japan Nuclear Energy Safety Organization), Sweden (Evergreen Safety & Reliability Technologies and SSM), Switzerland (ENSI), Great Britain (Magnox Electric) and U.S.A. (NRC).

Proposals for optimisation by the DIDELSYS Task Group

The results of the task group DIDELSYS are contained in a report which meanwhile has been accepted by CSNI for publication. The essential proposals for optimisation mentioned in the report will be described in the following. It has to be considered that the proposals for optimisation are generic results worked out by an international group. It is understood that the implementation of these results requires the consideration of country- and plant-specific situations.

The proposals for optimisation elaborated by the DIDELSYS task group can be structured into three topics:

1. Interface power plant/network operators. The proposals for optimisation regarding the interface between power plant and network operators is based on the report SOER 99-1 by the World Association of Nuclear Operators (WANO) and its addendum of the year 2004. According to this, power plant and network operators are to make binding agreements on the communication and co-ordination of scheduled activities. Furthermore, inspection and maintenance activities relevant for the power plant are to be jointly planned and co-ordinated. In case of any problems, an early information exchange between power plant operator and network operator is to take place. Assuming the particular safety-related significance of the electrical power supply of power plants for long-term residual-heat removal, the network operator's procedures are to prioritise the grid connection of nuclear power plants, i.e. a disconnection from the grid is to be avoided. When restarting the power grid after it has been disconnected, nuclear power plants are to be given priority.

2. Robustness of the electrical systems in a nuclear power plant. To ensure the robustness of the electrical systems in a nuclear power plant it is recommended to identify possible transients between nominal voltage and lightning impulse voltage. In doing so, unfavourable combinations of failures such as load dump to station service supply and a simultaneous fault in the generator excitation are to be considered as well. The task group is of the opinion that the examination should focus on the voltage range between nominal voltage and lightning impulse voltage. For this voltage range, the task group detected a deficit in the rules and regulations as the relevant standards lack any requirements for this voltage range. Additionally, several events in this voltage range are known from operating experience. On the basis of the transients identified, the robustness of the electrical controls used is to be examined. In doing so, particular attention is to be paid to the controls with semiconductor technology which were installed in the course of modernisation measures. These can be, for example, systems for the uninterruptible power supply, rectifiers and chargers as well as voltage supply of instrumentation and control cabinets which are based on semiconductor technology.

3. Control of electrical power supply problems. In this regard, it was recommended to verify the existing procedures and technical measures. In addition to that, the impacts of assumed failures of essential emergency power buses is to be examined. It is be analysed, for example, whether the displays in the control room provide sufficient information to the shift personnel and whether there are any false trips in reactor protection. Besides, it is to be examined to what extent a diverse energy source can be used to cool the core. This could be, for example, a diesel-driven pump or a fast-starting gas turbine.

Summary

Reappraisal of the defence-in-depth concept in respect of the electric power supply. The Forsmark event induced a cross-national re-evaluation of the defence-in-depth concept in respect of the electric power supply of nuclear power plants. This leads currently to a further development of the state-of-the-art in science and technology with the focus being in particular on transient processes between nominal and lightning impulse voltages. New challenges are, amongst others, the robustness of electrical controls with semiconductor technology and the verification of detection methods. This development is another example of how findings gained from the evaluation of operating experience are used to further improve plant safety. ∎

5. Final repository safety research

Tilmann Rothfuchs

In its field of work regarding final repository safety research, GRS's essential task is application-oriented research and development (R+D) to build a long-term safety case for final repositories for radioactive waste and underground waste disposal sites for hazardous chemical waste in geological formations. In addition to that, GRS offers potential clients from both the public sector and the industry to use the experimental and safety analysis methods and tools which have been developed within the scope of its research activities so far in other areas of environmental research. This includes activities regarding the long-term safety of CO_2 removal into suitable geological formations, the extraction of geothermal energy, and the safety of chemicals as well as for the development of an international strategy for the safe disposal of mercury.

Final repository safety research today

Current status. After more than 30 years of R&D on final disposal in rock salt formations in Germany, the fundamentals for entering the licensing procedure are given. By order of the BMWi, GRS - together with the *»Ökoinstitut«* - has given a comprehensive representation of the respective findings in a report on the final disposal of heat-generating radioactive waste (*»Endlagerung wärmeentwickelnder radioaktiver Abfälle in Deutschland«*); the report can be downloaded from the GRS website.

Current issues. Problems which require further R+D have largely been identified. These apply in particular to the further development or adjustment of safety analysis methods and tools to the international state-of-the-art in science and technology as well as to improving the understanding of the processes occurring in repository systems. One essential element is here the further development of models which allow for a more in-depth analysis of the complex interactions between thermal, hydraulic, mechanical and chemical processes in a potential repository. In addition to this, softwares with which these process sequences can be clearly visualised for both interested laypersons and scientific researchers are to be developed. The related GRS activities are as far as possible integrated in international projects. This integration increases the transparency of the national approach and represents an indispensable element for its scientific substantiation.

Safety case. Internationally it is understood that the long-term safety of a repository system has to be demonstrated by means of a long-term safety case (»*Langzeitsicherheitsnachweis*«); in this respect the relevant safety requirements (»Geological Disposal of Radioactive Waste«) of the International Atomic Energy Agency (IAEA) are the central reference source. In a draft relating to relevant safety requirements, GRS formulated concrete suggestions for the development of a long-term safety case (cf. B. Baltes, »*Sicherheitsanforderungen an die Endlagerung hochradioaktiver Abfälle in tiefen geologischen Formationen*«, atw, February 2008). The draft is currently being discussed among German experts and was also subject of the proceedings of the Nuclear Waste Management Commission (»*Entsorgungskommission* – ESK«).

The methodology to build a long-term safety case provides for a continuous further development of the planning and safety assessments. This objective basically also applies to a one-step licensing procedure for repositories as it is provided for by German law with the plan approval procedure. Irrespective of the discussion about details, experts therefore unanimously agree that accompanying R+D activities have to be carried out even beyond the date of plan approval, i.e. also during both the construction and the operating phase until the safe long-term closure of the repository. Relevant requirements arise from both the user's point of view and from the regulatory perspective.

Final Repository Safety Research Division

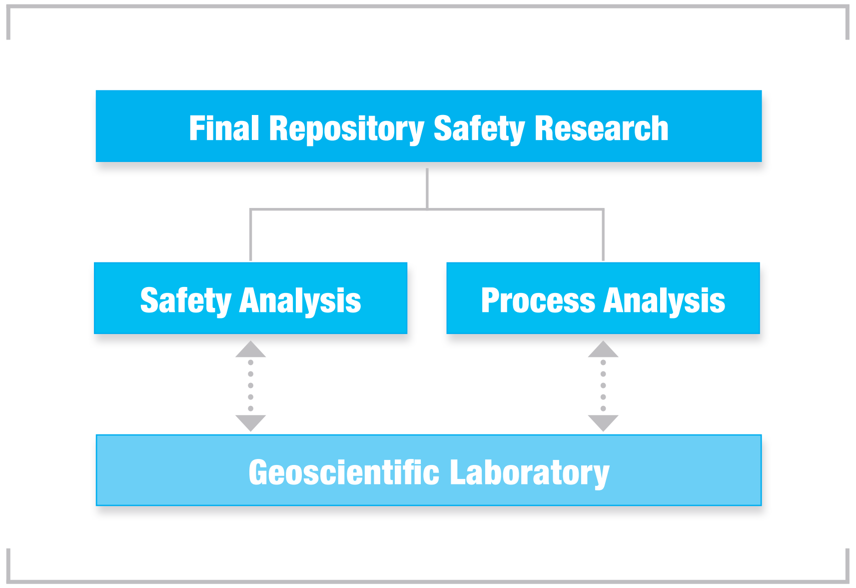
Competencies, organisational structure and research programme.

It is GRS's foremost intention to meet the above-mentioned requirements at the highest scientific level. To further expand the competencies in this field of work, GRS restructured its Final Repository Safety Research Division at the beginning of 2008. The special competencies, the new structure and the research programme of the Final Repository Safety Research Division will be described in the following.

Competencies of GRS. The assessment of concepts and site options, the scientific deduction and substantiation of criteria, the development and continuous improvement of safety analysis tools and methods and the realisation of complex long-term safety analyses for repositories or underground waste disposal sites pose interdisciplinary challenges. They require the co-operation of experimentally, analytically, theoretically and system-analytically working scientists. With its Final Repository Safety Research Division, GRS provides the technical knowledge of physicists, mathematicians, engineers, chemists, geologists, geophysicists and geochemists that is required for the solution of such interdisciplinary tasks. GRS's own lab capacities make this division largely independent of external assistance, generate additional expertise and allow the early processing of experimental, theoretical problems. The connection of these possibilities and the long-term experience from national and international R+D projects represent – also when compared at the international level – a unique competence.

Structure of the division. For many years, the Final Repository Safety Research Division was subdivided into three departments: Long-term

Organisation chart
Final Repository Safety Research Division



Safety Analysis, Geochemistry and Geotechnics. A detailed understanding of the decisive processes and its interactions in particular which is based on the results of experimental and theoretical examinations, requires, however, – as shown – interdisciplinary co-operation. It also requires the direct exchange of experimentally working experts who deal with the acquisition of basic data necessary for such understanding by means of lab and field tests. To achieve ideal basic conditions for a co-operation of this kind at the organisational level as well, the Final Repository Safety Research Division was subdivided in the two departments »Safety Analyses« and »Process Analyses«. These two departments can, where necessary for their activities, resort to the experimental data of the combined geoscientific lab which is organised as a service unit.

Safety Analyses Department. In the Safety Analyses Department the safety analysis methods and own calculation codes are developed and applied. Experiences gained from international practice and the continuously developing demands on long-term safety cases are incorporated in the developments. Within the scope of R+D projects, simplified model approaches for partial systems (near field, geosphere – see also the article on »Transport modelling with non-linear sorption and stochastic flow« – and biosphere) of the repository system are developed and improved for the different host rock formations to describe the complex coupled processes. Then they are transferred to program algorithms. These codes are then used for integrated system analyses and long-term safety analyses, respectively.

Process Analyses Department. In the Process Analyses Department, the thermal, hydraulic, mechanical and chemical (THMC) processes in repository systems and their complex interdependencies are explained and described in models. The development of such coupled process models was facilitated by the merging of the Geochemistry and Geotechnics departments. In addition to the expertise on geochemical and geotechnical problems, specialised knowledge in important fields such as theoretical chemistry, thermo-dynamics of high saline solutions, experimental analysis, and geophysics and groundwater hydraulics were merged. The findings gained from the development and application of process models are transferred in an abstract form to the Safety Analyses Department and used for integrated safety analyses.

Geoscientific laboratory. The experimental activities in the laboratory and in underground laboratories represent an indispensable part of final repository safety research. They serve for the verification and substantiation of the process models used for the safety analyses. Therefore, it is a special advantage of the Final Repository Safety Research Division that the theoretical and experimental activities can be carried out on the basis of our in-house network. One example for such network-based activities can be found in the article »On the short- and long-term behaviour of bentonites as technical barrier material in final repositories for radioactive waste« (»*Zur Kurz- und Langzeitstabilität von Betoniten als technische Barrierematerialien für Endlager für radioaktive Abfälle*«) in this Annual Report. Due to the longstanding participation in international projects in underground laboratories abroad, the division also has great expertise in applied research in underground laboratories. This is particularly helpful for the characterisation of host rocks, the optimisation and analysis of possible fields of application of sealing and backfill materials, the long-term behaviour of geomaterials and structural elements in repositories, and the further development and application of geophysical and geotechnical measuring methods.

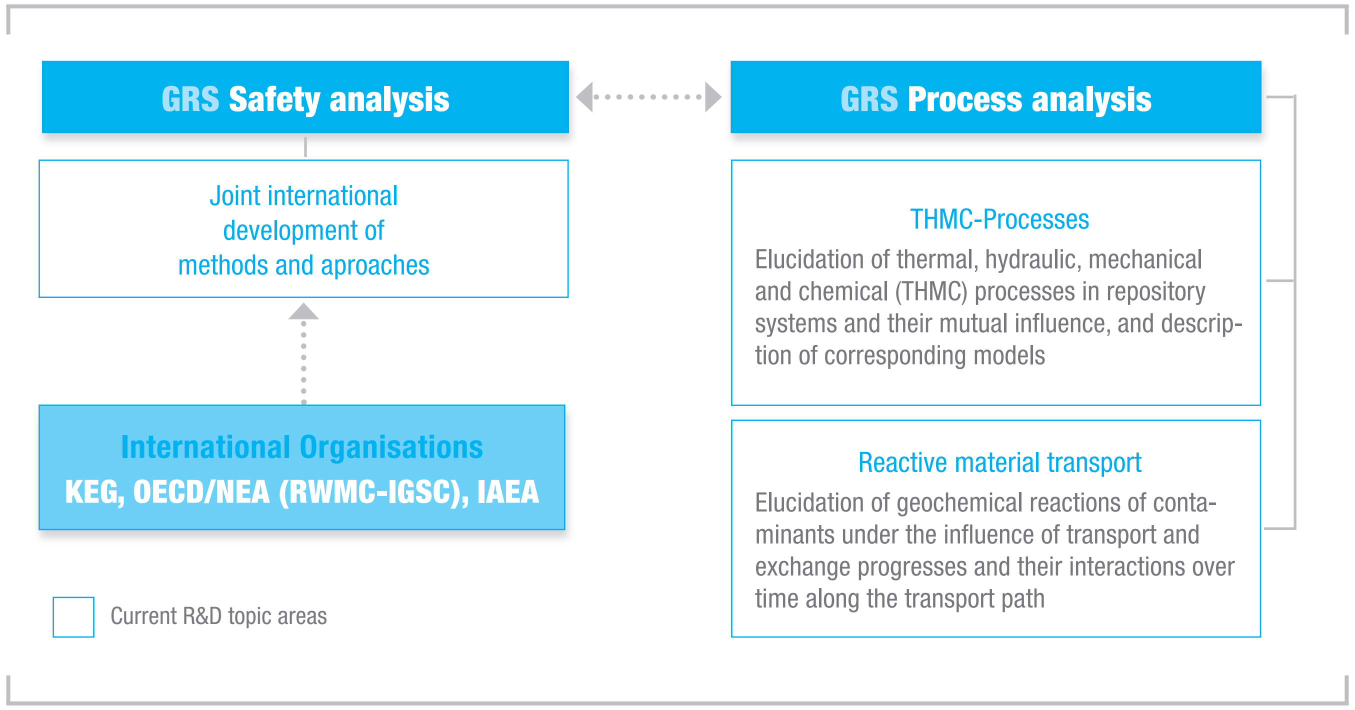
At the turn of the year 2008/2009, i.e. after the first 12 months with the division's new structure, an internal evaluation in respect of its efficiency was carried out. In doing so, it became clear that the optimisation aimed for with the restructuring had been partially realised already. Due to the integration of the former Geotechnics and Geochemistry Departments in the Process Analyses Department, the synergies regarding experimental-analytical and both theoretical and system-analytical-oriented activities became apparent after only one year through the increasing number of integrated projects. The merged geoscientific laboratory as central service unit for the two new departments resulted in the intended increase in efficiency.

Research programme of the division.

Research into the options salt and clay. The current research programme of the Final Repository Safety Research Division is geared towards the already identifiable requirements for building future long-term safety cases as well as towards the focal research issues mentioned in BMWi's R&D programme 2007-2010. In correspondence with the Federal Government's goal to include in addition to salt formations clay formations as alternative host rock in basic research, the respective R&D activities have been intensified since the beginning of this decade. Today, they represent one main focus of the GRS R&D programme.

Key future R&D projects. Based on its current R&D projects for the BMWi, the co-operation in international joint research projects of the European Commission as well as the co-operation in international committees of OECD/NEA and IAEA, GRS has identified the topics shown in Fig. »POTENTIAL FOCUSES OF FUTURE RESEARCH AND DEVELOPMENT« as potential focal points of future R&D projects. This assessment serves as basis for the agreement of expedient projects in regular expert discussions with potential customers of GRS.

Potential focuses of future research and development



Bundling competence through co-operation. During the projects carried out recently the increased bundling of individual research activities under central research topics has proved extraordinarily productive. A large part of the new projects will therefore be carried out in joint research projects with other research institutions. With this, GRS pursues the objective to better crosslink with the expert world and to thus improve the direct access to the latest scientifically relevant individual results. This allows GRS to strengthen its competencies and to keep fulfilling its task to further develop and substantiate safety analysis methods and tools at a high level – also in view of international developments. ∎

5.1

Transport modelling with non-linear sorption and stochastic flow

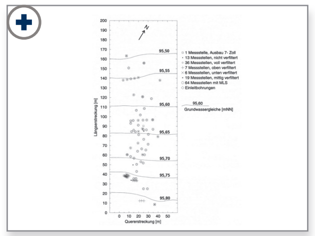
The code packages d³f (distributed density-driven flow) and r³t (radionuclides, reaction, retardation, and transport) were developed between 1994 and 2003 within the scope of projects supported by the Federal Ministry for Education and Research (»*Bundesministerium für Bildung und Forschung*« – BMBF) and the Federal Ministry of Economics and Technology (»*Bundesministerium für Wirtschaft und Technologie*« – BMWi), respectively. They were developed with the aim to allow the modelling of density flow and transport of pollutants for three-dimensional, hydrogeologically complex areas for long periods of time. The BMWi-supported project on the modelling of a large-scale transport of pollutants (»*Modellierung des großräumigen Schadstofftransportes*«) included among other things the modelling of transport with non-linear sorption and stochastic flow, which will be introduced in this article.

Dr. Eckhard Fein

Anke Schneider

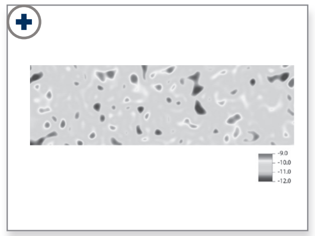
The code packages d³f und r³t

Function and scope of application. The code packages d³f and r³t were developed under the auspices of GRS at six and four universities, respectively. The codes have the ability to calculate flow and transport through porous media for two- and three-dimensional model areas. This requires that the porous medium is fluid-saturated, that there is a confined aquifer system and that both fluid and medium are incompressible. The hydrogeology may show strong heterogeneities and anisotropies. The model area may contain both sources and sinks of the fluid and the pollutant. Density and viscosity of the fluid can be functions of the salt concentration. Besides advection, diffusion and dispersion, the transport modelling also considers the equilibrium sorption and the kinetically controlled sorption, using either linear or non-linear isotherms. Additionally, precipitation can be taken into account as well.



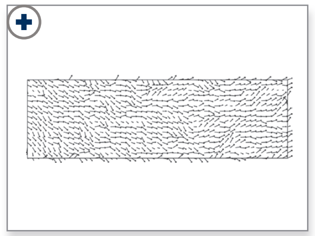
MEASURING POINTS

Fig. 47
The Krauthausen test site - water table contours and well locations



LOGARITHMICAL REPRESENTATION

Fig. 48
Permeability distribution



d^3f-SIMULATIONS

Fig. 49
Associated Darcy velocities

Numerical methods. The flow and transport equations are solved on basis the of the UG (»Unstructured Grids« by Goethe University Frankfurt) software. Triangular and tetrahedral grids as well as square and hexahedral grids can be used. The discretisation is based on the finite volume method. There is a choice between different upwind schemes. The systems of equations are solved with a multiple grid algorithm combined with a BiCGStab method. Grid adaptation and time step sizes are controlled by means of a posteriori error estimators. The codes can be used on LINUX PCs, workstations, clusters as well as on massively parallel processors. Both have graphic pre- and post-processors and grid generators.

Within the scope of this project, different application cases were calculated to verify the correct modelling of individual processes. In the example described in the following, primarily the stochastic modelling of permeability in a flow model and the non-linear sorption in the transport model are tested.

Krauthausen test site

Test area for ground water flow and contaminant transport. The Krauthausen test site is located in North Rhine-Westphalia, Germany, approx. 7 km south-east of the Jülich Research Centre. It was established in 1993, and several experiments on groundwater flow and the transport of pollutants have been carried out there since.

The size of the test area is 200 m x 70 m. A geological profile was created on the basis of the results of four drillings reaching down between 5 and 20 m. The confining base of the top aquifer is located at a depth of 9 to 10 m and consists of a clay layer which is several decimetres thick. It is covered by flood sediments and consists mainly of gravelly and sandy sediments. The groundwater regionally flows to the north-west with a hydraulic gradient of 0.2 %. The average annual precipitation amount is 690 mm a^{-1}.

To examine the influence of aquifer heterogeneity on the transport of pollutants, 74 wells with different instrumentations are operated (see Fig. 47 »MEASURING POINTS«). These include 52 wells to determine the spatial and temporal development of the pollutant concentration, 11 wells to identify the groundwater levels, 28 wells to determine the water velocities, 10 injection wells and one pumping well.

Groundwater flow

From May until August 2001, the flow velocity was measured in 21 wells at 361 positions. The measured velocities were evaluated by means of variogram analysis. An exponential model could be fitted to the experimental variogram. In geostatistical analyses the permeabilities are described by mean, variance and correlation length.

Two-dimensional flow modelling

Stochastic modelling. The two-dimensional model is 45 m wide and 150 m long. Both permeability and correlation were assumed to be isotropic. The mean value of permeability was set to $7.05 \cdot 10^{-11}$ m², variance to $1.81 \cdot 10^{-20}$ m⁴, correlation length to 3.43 m and porosity to 0.26. A second modelling with the permeability mean value $1.15 \cdot 10^{-10}$ m² and with all other parameters remaining unchanged was carried out, too. The permeability distributions were generated with the random number generator included in d^3f (see Fig. 48 »LOGARITHMICAL REPRESENTATION«). The calculations were carried out with Dirichlet boundary conditions for the pressure on a rectangular grid with approx. 200,000 knots.

Variation of mean permeability. The result of the d^3f simulations (see Fig. 49 »d^3f-SIMULATIONS«) was a mean Darcy velocity of $9.2 \cdot 10^{-7}$ m s^{-1} and thus a theoretical field velocity of $3.54 \cdot 10^{-6}$ m s^{-1}. This does not correspond to the velocity with which the centre of a tracer (bromide) moves. The variation of the mean permeability allowed the determination of a Darcy velocity of $1.91 \cdot 10^{-6}$ m s^{-1} which corresponded better to the propagation velocity of the bromide.

source	location			uranine	lithium	bromide
	x [m]	y [m]	z [m] (3d only)	mass [mol] time [s]	mass [mol] time [s]	mass [mol] time [s]
1	14.09	33.68	6 – 7	0.89 18 000	886.5 18 000	440.5 38 880
2	16.00	33.51	6 – 7	0.89 18 000	886.5 18 000	440.5 38 880
3	18.01	33.34	6 – 7	0.89 18 000	886.5 18 000	440.5 38 880

CO-ORDINATES

Sources and tracer masses

tracer	isotherm	a [m³ kg⁻¹]	p [-]
uranine	Freundlich	$4.77 \cdot 10^{-5}$	0.81
lithium	Freundlich	$1.02 \cdot 10^{-4}$	0.61
bromide	Henry	0.0	1.0
rock density [kg m-3]		2600	

SORPTION PARAMETERS

Two-dimensional transport modelling

PLUMES

Fig. 50–52
Uranine, lithium and bromide plumes after 85, 154, 365 and 449 days

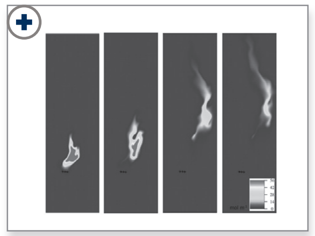
Transport of uranine, lithium and bromide

Contaminant transport experiments. In recent years, intensive examinations of the subsoil in Krauthausen have been performed by means of studying the transport of uranin, LiCl and NaBr, and the results have been documented. At a depth of between 6 and 7 m, the sorbing materials uranine ($C_{20}H_{20}Na_2O_5$) and lithium (as LiCl) were injected into three different wells. Overall, 2.67 mol uranin and 2,659.5 mol LiCl were injected over a period of 5 hours. Approx. 8 months later, 1,321.5 mol of the conservative tracer bromide were injected into the same well over a period of 10.8 hours.

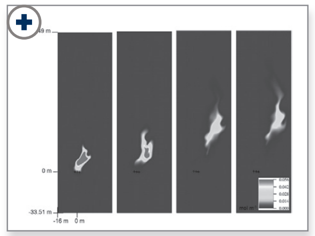
For at least 449 days, the plumes were verified at the different measuring points at different depths and their concentration was determined. The recovery was 50 % or less. The experimenters assumed that the pollutants get lost through the thin clay layer underneath the aquifer which was considered impermeable.

5.1 Transport modelling with non-linear sorption and stochastic flow

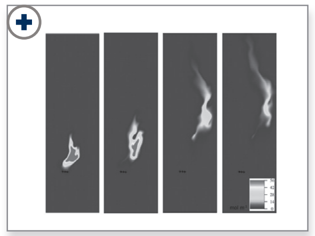
CONTENT



50 URANINE



51 LITHIUM



52 BROMIDE

Two-dimensional transport modelling

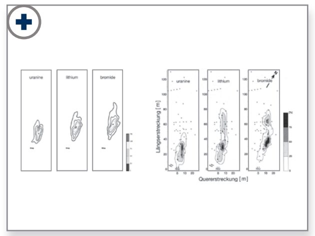
Transport modelling with non-linear sorption. The retention of uranine and lithium was modelled by means of non-linear sorption isotherms (Freundlich) whereas bromide does not get sorbed.

In Fig. 50-52 »PLUMES«, the plumes of uranine, lithium and bromide are represented after 85 d, 154 d, 365 d, and 449 d. It has to be considered that the pictures of the pollutants have different scales.

It is easy to see that the retention is strongest for uranine, whereas bromide is transported the farthest. This behaviour shows good agreement with the experiment.

Fig. 53 »PLUMES« shows the concentrations of uranine, lithium and bromide calculated with the higher permeability and those vertically averaged, respectively, both measured at the time of 85 d. In both figures, the concentrations are standardised to the highest vertically averaged concentration.

The comparison between calculated and measured concentrations shows that d³f and r³t provide realistic results.



PLUMES

Fig. 53
Uranine, lithium and bromide plumes after 85 days
Comparison of the calculated and the measured (on the right, according to Englert 2000) concentrations

Summary and outlook

In this project, stochastic flow modelling was coupled with a subsequent transport modelling with non-linear sorption. Both modellings were conducted both two- and three-dimensionally. In both cases, the results were consistent in comparison to the experimental results. In principle, the stochastic modellings should be repeated sufficiently often with different realisations of the permeability field. Reliable results can then be achieved by means of suitable averaging. Here, only one realisation was examined as an example.

The calculation codes are currently being developed further in respect of the modelling of fissures, heat transfer, and open groundwater surface. ∎

5.2

On the short- and long-term behaviour of bentonites as technical barriers material in final repositories for radioactive wastes

Dr. Horst-Jürgen Herbert

The objective of the final disposal of highly radioactive wastes in deep geological formations is it to isolate the pollutants from the biosphere for long periods of time. This objective can be achieved by means of a multi-barrier concept which combines geological and technical barriers. As potential materials for technical barriers, so-called bentonites are being discussed. These are expandable, plastic clay rocks with high sealing potential. Especially in salt formations, but also in other geological host rocks, saline solutions may occur which react with the bentonites and alter their properties. The interactions with the solutions alter the mineralogical composition and thus the swelling and sealing properties of the bentonites.

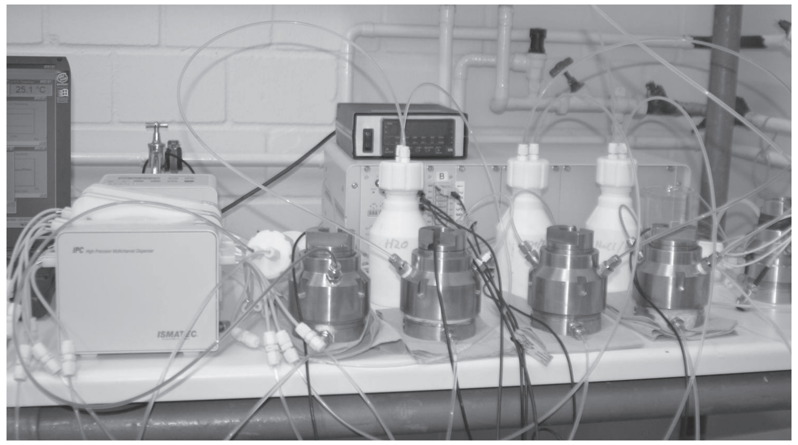
Within the scope of the EU project NF-PRO, GRS conducted a study with the aim to collect data concerning the relevant interactions between bentonites and saline solutions in a wide ionic strength and pH range and the impacts thereof on the short- and long-term behaviour of the technical barrier. The study was carried out in co-operation with Prof. J. Kasbohm (Greifswald University).

Scientific background

Working hypotheses. Herbert et al. (Long-term behaviour of the Wyoming bentonite MX-80 in high saline solutions, Applied Clay Science 26, 2004) reported on interactions between the Wyoming bentonite MX-80 with saline solutions. For the new GRS study introduced here, this resulted in the following working hypotheses which were to be verified:

1. Saline solutions of varying ionic strength influence the swelling pressure of compacted bentonites to different degrees.
2. When bentonites react with saline solutions, both the mineralogical and the chemical properties of the montmorillonite, an expandable smectitic clay mineral which makes up 70-90 % of the bentonite composition, are modified in a way that its interlayer charge is reduced which causes the swelling pressure to decrease and the permeability of the compacted bentonite to increase.
3. The end member of the mineralogical modification of the montmorillonite could be kaolinite or pyrophyllite which in the long run leads to the reduction of swelling capacity and swelling pressure and thus of the sealing effect, too.

Evaluation of research literature. In the relevant scientific literature, information can be found which indicates that these hypotheses are correct; however, some opposing opinions have been expressed as well. According to the so-called DLVO theory, the electrolyte concentration and the layer charge of the clay minerals influence the bentonites' swelling pressure. Savage (The Effects of High Salinity Groundwater on the Performance of Clay Barriers, SKI Report 2005 : 54) and Laird (Influence of layer charge on swelling of smectites, Applied Clay Science 34, 2006) described the interrelations between the layer charge of smectites and the swelling pressure of betonites. According



SWELLING PRESSURE MEASURING CELLS

Fig. 54
With data collection in the GRS laboratory at Braunschweig

to these authors, the swelling capacity decreases with increasing charge density since a higher charge leads to the integration of more higher-valent cationes into the interlayers. At first, this seems to be contradictory to the second working hypothesis. Furthermore - and just like Herbert et al. - the authors found, however, that with the increasing salinity of the solution the swelling capacity decreases. Laird described several different processes which control the swelling of smectites saturated with alkali and alkaline earth metal cations: a) crystalline swelling, b) double-layer swelling, c) internal volume swelling, and d) swelling due to the Brownian motion. A direct impact of the layer charge on the swelling can only be expected for the crystalline swelling. The crystalline swelling degree decreases with an increasing layer charge. Laird further described the appearance and disappearance of so-called quasi-crystals as a dynamic process in an aqueous smectite suspension. With the layer charge increasing, the quasi-crystals become bigger and more stable. Pusch et al. (Evolution of clay buffer under repository conditions. Reprints of the contributions to the workshop on long-term performance of smectitic clays, 2007) also developed a concept which describes those parameters which influence the swelling pressure. Pusch's definition »interlamellar pressure« or »disjoining pressure« can be compared to Laird's »crystalline swelling«; and Pusch's »osmotic pressure« corresponds with Laird's »double-layer swelling«.

5.2 On the short- and long-term behaviour of bentonites as technical barriers material in final repositories for radioactive wastes CONTENT

Experimental programme

Verifying hypotheses in experiments. To verify the above-mentioned hypotheses, a three-year experimental programme was set up. MX-80 bentonite was exposed to seven saline solutions of different compositions, ionic strengths and pH values and with two different solid to solution ratios (tests with compacted bentonite with little solution in the pore space and batch tests with non-compacted bentonite with a high excess of solution).

The solutions were as follows:

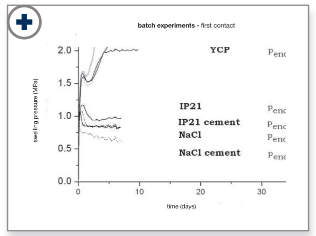
1. Fissure water out of the granite of the Swedish undergrouwnd laboratory Äspö,
2. Pore water out of the opalinus clay of the Swiss underground laboratory in Mont Terri,
3. a pure saturated NaCl solution,
4. a carnallite-saturated, Mg-rich IP21 solution as can be found in salt formations in northern Germany,
5. **and 6.** corrosion solutions which are produced when salt concrete (a special type of concrete with salt as aggregate which is used in salt mines) reacts with NaCl and IP21 solution, and
7. a low saline, but very alkaline young Portland cement pore water (YPC).

For comparison, the tests were also carried out with pure water as the knowledge about interactions of bentonites with water is the greatest. Samples were analysed after reaction times of seven days, one, two and three years. The composition of the solution, the mineralogical composition of the bentonite, the chemical changes of the montmorillonites and the swelling pressure of the bentonite were measured. Fig. 54 »SWELLING PRESSURE MEASURING CELLS« shows the setup of swelling pressure measuring cells and the data collection in the laboratory.

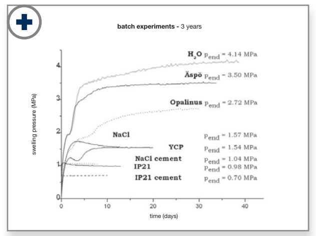
Results

Swelling capacity of the montmorillonities. Overall, both in the compacted and in the non-compacted bentonite samples, the swelling capacity of the montmorillonites persisted for all solutions they were in contact with over the entire observation period of three years. This was demonstrated through the expansion of the montmorillonite interlayers by means of ethylene-glycol. At the beginning and at the end of the experiments, the maximum interlayer distance at glycol saturation was reached and consistently amounted to 17 Å. Nevertheless, each of the eight solutions caused distinct mineralogical and chemical alterations of the montmorillonites albeit to different extents.

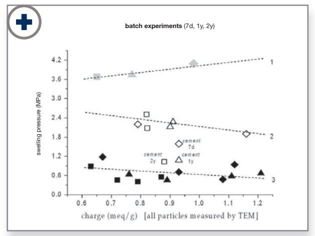
Changed swelling presure. These alterations are also reflected by the changes of the swelling pressures. As expected, the measured swelling pressures were highest for water and declined with increasing salinity (and increasing pH value) of the solutions (see Fig. 55 »SWELLING PRESSURE«). Although the drop in swelling pressure of compacted bentonite in contact with water compared to the swelling pressure with low-saline solutions is clear, it is still moderate compared to the strong decline during reaction with high saline solutions. In respect of the swelling pressures, the Young Portland cement pore water (YCP) takes a middle position: It does indeed show low salinity, but in turn, it also shows a high pH value. This observation, too, confirmed the working hypotheses described at the beginning. A surprising observation was, however,



55 SWELLING PRESSURE
of original, untreated MX-80 bentonite immediately after first exposure



56 SWELLING PRESSURE
of MX-80 bentonite after a three-year reaction time



57 INTERRELATION OF SWELLING PRESSURE AND CHARGE OF THE MONTMORILLONITE
in contact with:
1 – water
2 – solutions of medium ionic strength
3 – solutions of high ionic strength
after
◆ – 7 days, ■ – 1 year, ▲ – 2 years

EXPERIMENTAL RESULTS

Fig. 55–57
Swelling pressures with different solutions
Batch experiments with excess solution

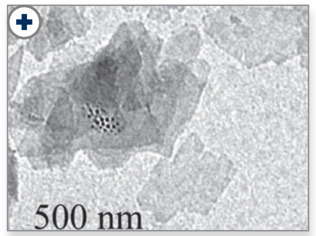
that with progressing reaction time the swelling pressures did not drop but, on the contrary, increased (see Fig. 56 and 57). This led – for a short time at least – to an improved sealing effect of the bentonite in contact with the examined solutions.

Partial dissolution of clay particles. At the same time, a partial dissolution of clay particles was observed which caused the Al content of the solution to increase. In the montmorillonite particles which were maintained, an increasing Si excess and a reduction of the total charge and the interlayer charge were observed with progressing reaction time. This is due to the exchange of Mg and Al in the octahedron layers. In addition, it was observed that the alterations of the montmorillonites in the compacted bentonite samples proceed faster than in the batch experiments with excess solution. One possible explanation for this is the acidity of the interlayer water which is higher compared to the pore water. Yariv & Michaelian (Surface acidity of clay minerals. Industrial examples– *Angewandte Geowissenschaften* 1, 181-190, 1997) discovered that the amount of dissociated water in the interlayers of the montmorillonite is 107 times higher than in the pore space of the bentonite because the dielectric constant of the interlayer water is lower than the one of the free water in the pore space.

Formation of new clay minerals. The expected kaolinisation/pyrophyllitisation (working hypothesis 3) and the Si excess in the modified montmorillonites were actually observed in several samples at an incipient stage (see Fig. 58 »TEM-PICTURE«).

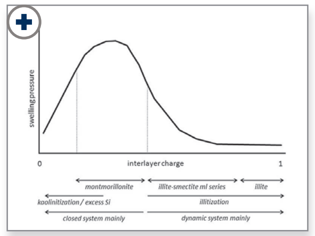
Confirmation and explanation of the hypotheses. The evaluation of these own results and of the data collected from literature which supposedly contradict the working hypotheses led to new findings all of which do not only confirm these observations, but also consistently explain them. Increasing interlayer charges lead – as Savage and Laird reported – to declining swelling pressures. As far as we can see, this is on account of the transformation of montmorillonite into illite, a clay mineral which

is not expandable anymore. The illitisation is a widespread process in open, natural systems with exchange of solutions. This process, however, does not take place under test and thus respository conditions. In our tests under repository conditions, i.e. in a closed system, in all samples we detected no increase of the interlayer charge but a significant decline thereof. Such a sequence cannot lead to illitisation. In literature, however, an illitisation has so far been generally assumed for repositories, too. A decline in charge leads – as observed – to a kaolinisation/pyrophyllitisation. As the clay minerals kaolinite and pyrophyllite are – similar to illite – not expandable anymore either, the apparent contradiction to the literature data is resolved. Both an increase and a decline of the charge can cause a loss of the swelling capacity. Fig. 59 »SWELLING PRESSURE/INTERLAYER CHARGE« schematically represents the interrelation between the mineralogical alterations of the montmorillonites and the change of the interlayer charge and swelling pressure of the bentonite in case of reactions with solutions in open and closed systems.



TEM PICTURE

Fig. 58
Newly-formed hypidiomorphic kaolinite crystals in a bentonite sample after a 2-year reaction time with a Mg-rich IP21 solution



SWELLING PRESSURE/INTERLAYER CHARGE

Fig. 59
Schematic interrelation between montmorillonite transformation, interlayer charge and swelling pressure in open and closed systems

Kaolinite and pyrophyllite with the interlayer charge of zero/half cell (left) and illite with the interlayer charge one/half cell (right) are end members of the montmorillonite transformation with different developments due to different boundary conditions.

Summary and conclusions

Significant differences between closed repository systems and open geological systems. The three working hypotheses established at the beginning of the R+D project were confirmed by new tests. Apparent contradictions with data from the literature could be satisfyingly clarified. The new understanding which has developed explains both the short- and the long-term behaviour of bentonites and makes clear that there are significant differences between a closed repository system and an open geological system. In a final repository in salt formations which can be considered practically closed and where no fast removal of Al can be assumed, no illitisation of expandable smectites, but a kaolinisation/pyrophyllitisation can be expected. In the long run, however, both directions of transformation lead to a decreasing swelling capacity of the bentonites, even though a short-term swelling pressure increase can be observed. This means that in the long term a permeability increase of the technical barrier bentonite is to be assumed.

The question which thus has to be answered in the context of bentonite barrier stability is not if, but how long the bentonite will maintain its sealing effect under final repository/underground waste disposal site conditions. In the long-term safety analysis it thus has to be clarified which degree of swelling capacity is required and over which period of time a certain permeability has to be maintained. ∎

6. Radiation and environmental protection

Dr. Gunter Pretzsch

Scientific and technical issues relating to radiation and environmental protection as well as regarding the supply and disposal of nuclear facilities are becoming more and more important. This applies in particular to the interim storage and final disposal of spent fuel elements and radioactive waste as well as the decommissioning of nuclear reactors. As central expert and research organisation, GRS performs research activities in these fields and prepares expert analyses and assessments. The GRS Radiation and Environmental Protection Division provides the expertise of scientists and engineers of different fields which is required to handle these interdisciplinary problems.

Key working field nuclear fuel. In the field of nuclear fuel, GRS is dealing with issues relating to the nuclear safety of nuclear facilities and mass balances. Focal points of the activities on nuclear safety are criticality, burn-up and decay of fissile materials, radiative transfer and activation of shielding, nuclear process engineering as well as the analysis of operating experience and accidents. The activities on mass balances include, for example, the monitoring and documentation of nuclear fuel and waste flows as well as activities regarding the records of proper waste management in the fuel cycle.

Key working field radiation protection. GRS deals with aspects of radiation protection primarily in the context of nuclear power plants in operation and during their decommissioning, and upon release of radioactive substances. Further issues are examined with regard to radiological emergency protection, radioecology in the surrounding area of nuclear facilities and contaminated sites and areas. Analyses on potential radiological consequences after accident-induced releases including the modelling of the atmospheric dispersion of radioactive substances and questions pertaining to the transport safety also play an important role.

Focal activities
Radiation and Environmental Protection Division

NUCLEAR FUEL	RADIATION PROTECTION	FINAL DISPOSAL
Criticality during transport, storage and handling of nuclear fuels Burnup and decay of nuclear fuels Shielding and radiation transport Nuclear process engineering Evaluation of operating experience and events Tracing and documentation of nuclear fuel and waste flows Waste management verifications	Radiation protection during nuclear power plant operation and decommissioning Clearance of radioactive substances Radiation emergency preparedness Analyses of potential radiological consequences following accidental releases Modelling of the atmospheric disperion of radioactive substances Radioecology studies in the vicinity of nuclear installations Studies into transport safety	Analyses and technical support of regulatory authorities regarding the safety-related assessment of repository concepts (incl. repository operation) and on waste characterisation and conditioning Development of concepts and methods for the selection, characterisation and long-term safety assessment of repository systems in different types of host rock Expert assessment within the framwork of licencing procedures Development of safety criteria and guidelines and promotion of international co-operation in the field of final disposal, support of licensing authorities in Eastern Europe

Key working fields waste and final disposal. The field of waste and final disposal comprises projects on disposal and repository concepts as well as the integral safety-related assessment of all waste management steps, from the waste formation to waste management and final disposal. In this connection, the GRS specialists deal with the characterisation of radioactive waste and the treatment and conditioning thereof as well as with the assessment of the selection, characterisation and long-term safety of repository concepts and repository sites.

Customers. The activities pertaining to the above-mentioned topics are carried out mainly on behalf of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (»*Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit*« – BMU) and the Federal Office for Radiation Protection (»*Bundesamt für Strahlenschutz*« – BfS). In addition, GRS also works by order of authorities of the European Commission. In many projects, GRS works in close co-operation with national and international institutions. Some of the current projects of GRS's Radiation and Environmental Protection Division will be introduced in the following.

Tasks and projects of the Radiation and Environmental Protection Division

Supporting the BMU in its participation in the WENRA work group on interim storage and decommissioning. In the project regarding the implementation of international rules and regulations (»*Umsetzung internationaler Regelwerke*«), GRS supports the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) in fulfilling its obligations arising from co-signing the policy statement of the Western European Nuclear Regulators Association (WENRA) of September 2005. With the Working Group on Waste and Decommissioning (WGWD), a work group of representatives of the authorities of the signatory states, WENRA aims to align the safety levels for the interim storage of irradiated fuel elements and radioactive waste and for the decommissioning of nuclear facilities. To this end, a catalogue of safety requirements (Safety Reference Level) was prepared for the interim storage and the decommissioning, respectively, which serves as basis for the verification and harmonisation of the national requirements. In the first phase of this process, the national safety requirements and the implementation thereof were examined and assessed (benchmarking) by means of the WENRA reference level. In the second phase, the member states are requested to prepare plans to complement their national safety requirements in those areas where the WENRA reference levels are not adequately covered.

In the years 2007 and 2008, GRS itself co-ordinated and prepared the essential part of the self-assessment with regard to the practice of interim storage at selected plants and the relevant regulations. The results of these activities were submitted to the WGWD for reviewing. The benchmarking was concluded at the end of May 2009. GRS is currently preparing a final report for WGWD on behalf of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety which describes the assessment procedure and gives an overview of the results of all participating countries. Furthermore, the activities of the next phase, the plan of action to align the national rules and regulations, were started. The Öko-Institut Darmstadt is involved in the activities as GRS subcontractor.

Besides, GRS supports the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety also in analysing the implementation of the WENRA reference level for the safety during the decommissioning of nuclear facilities in the German regulations. These activities are part of an own comprehensive project to address questions relating to the decommissioning of nuclear facilities and with which GRS provides the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety with its technical and scientific expertise. In addition to performing own studies, GRS took part in expert discussions within Germany in 2007 and 2008 and evaluated the results thereof. GRS worked out suggestions for the further development of the current reference level and introduced them in different WGWD meetings. Also in the future, GRS will continue these activities and support the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety in the further development of the safety during decommissioning pursuant to the WENRA objectives.

Safety-relevant aspects of the long-term interim storage of irradiated fuel elements. Due to the operation of nuclear power plants, in Germany highly radioactive materials in the form of spent fuel elements and vitrified waste in so-called HAW vitrified waste containers have accumulated during reprocessing which will have to be stored in a repository. The dry interim storage of spent fuel elements from nuclear power plants and of vitrified high active waste from reprocessing plants is an integral part of the disposal concept for radioactive waste in Germany. The licenses issued for the interim storage are limited to 40 years. In a project, GRS examines with the participation of the Öko-Institut Darmstadt and in consideration of the international state-of-the-art in science and technology the safety-related aspects of the dry interim storage which are subject to time-dependent changes and which are therefore potentially relevant for the interim storage safety. In a systematic examination, the potential long-term and ageing effects on safety-relevant components and systems shall be described and their impact on interim storage safety shall be discussed. These components and systems are also, besides the stored fuel elements and HAW vitrified waste canisters, the containers used and the handling and repository equipment such as cranes and storage buildings. In addition, the safety of operational procedures and measures to gain experience and know-how are considered as well. Finally, the existing concept for the long-term compliance with the safety requirements (e.g. in-service inspections) are analysed and called into question. In this context, GRS will also work out a suggestion regarding the extent and time interval of a periodic safety review for fuel element interim storages.

Safety of radioisotope-based energy sources for aerospace missions. In the project »European Space Nuclear Safety Framework« (ENSaF), GRS participated in the development of a concept for future licensing procedures relating to the use of radionuclide-based energy sources (Nuclear Power Sources, NPS) in aerospace missions of the European Space Agency (ESA). In the U.S.A. and in the Russian Federation, licensing procedures for missions using NPS have already been established. For the independent performance of missions using NPS from the Guyana Space Center in Kourou, however, ESA needs a licensing procedure at the European level. A concept for such a procedure was prepared by order of ESA and with the participation of representatives of the aerospace industry, potential providers of corresponding energy sources as well as expert organisations in the field of nuclear safety from four European countries. In this concept, practicable approaches for the safety assessment and licensing with international participation at the European level identified and the bases for the derivation of radiological safety objectives considering the specific risks of aerospace missions using NSP are pointed out.

Comparison of the decommissioning options »immediate dismantling« and »safe enclosure« using the example of two TRIGA research reactors. In a study conducted together with the German Cancer Research Center (*Deutsches Krebsforschungszentrum* – DKFZ), GRS retrospectively analysed the choice of decommissioning options for two research reactors of the TRIGA type. Aim of the study was it to examine the decommissioning option – immediate dismantling or dismantling after a 20-year-phase of safe enclosure – respectively chosen by operator DKFZ for both concrete decommissioning projects, to analyse them with regard to potential advantages and disadvantages and to draw general conclusions from this. To that end, first and foremost practical experiences gained from the mentioned decommissioning options were evaluated. Within the scope of the study, the information on the two decommissioning projects were prepared and evaluated according to a catalogue of criteria developed for this purpose. So the availability of the required personnel and their radiation exposure were examined as well as the amount of radioactive waste arising from the two options. The study comes to the conclusion that both immediate dismantling and dismantling after a phase of safe enclosure were a suitable and appropriate choice for both concrete decommissioning projects. In particular in case of the assessment for the safe enclosure, particularities of the research reactor and the operator's special personnel situation have a positive impact. However, the results of the study also show that it is not easy to make general recommendations on the selection of a decommissioning option: Which decommissioning option is the most advantageous for a specific facility can only be determined after evaluating the actual framework conditions. In this respect, the generally discussed abstract, option-specific advantages and disadvantages require critical examination.

Consultation of the BMU in the procedure on the decommissioning of Morsleben repository. Being the operator of the repository for radioactive waste Morsleben (*Endlager für radioaktive Abfälle Morsleben* – ERAM), the Federal Office for Radiation Protection (»*Bundesamt für Strahlenschutz*« – BfS) applied for a licensing procedure for the decommissioning of ERAM at the Ministry of Environment in Saxony-Anhalt (*Umwelt-ministerium des Landes Sachsen-Anhalt* – MU) in 1992. As part of this procedure, the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) supervises compliance with both legal and technical specifications. Subject of the technical supervision is to review the technical correctness and the expediency of the actions of the supervised authority. BMU commissioned GRS to exercise the technical supervision with the support of the BMU. Objective of this project is it to inform and consult the BMU continuously and in a timely manner on plant-specific, safety-related and procedural aspects of the procedure.

To this end, GRS views all application documents submitted by the applicant, reviews their consistency and evaluates them in correspondence with the mentioned aspects. Likewise, the statements by the MLU as well as the comments by experts of the licensing authority are evaluated in view of their safety-related and procedural appropriateness. In addition, GRS follows the process of the licensing procedure by participating in expert discussions and co-ordination meetings of the parties involved in the procedure and the consulted experts. In the phase of public participation, GRS will support the BMU in dealing with issues pertaining to the readiness of the documents for public display during public participation as well as in the associated public procedure.

In addition to the preparation of these tasks accompanying the procedure, GRS supports the BMU in respect of the backfilling of individual old mines in the central part of ERAM which was approved under mining law. Aim of the early backfilling measure is it to avoid a long-term degradation of the geomechanical situation by inserting more than half a million cubic meters of stowing material which is capable of flow. GRS professionally prepares the information regarding the corresponding technical planning and examines if these early measures will have consequences for the decommissioning of ERAM.

On the robustness of the operation of a repository for heat-generating radioactive waste. In an own research project, GRS deals with the question how robust the performance of the operation of a final repository for heat-generating radioactive waste can be. In connection with the operation of repositories, the term 'robustness' implies above all,

- the use of preferably proven and tested technology,
- a conceptual minimisation of the radiation exposure of the operating personnel,
- the reduction of potential accidents,
- high operational reliability, and
- the minimisation of secondary waste.

When assessing the robustness of a specific repository and storage concept according to the above-mentioned aspects, it also has to be considered which consequences will result from the implementation of this concept for the handling and co-ordination steps preceding the final disposal.

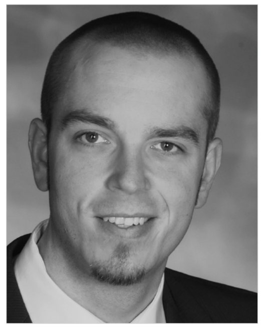
In this project, GRS examines the currently pursued repository concepts of drift emplacement and borehole disposal for the host rocks of rock salt and clay formations. For both repository concepts, the state-of-the-art in science and technology will be determined first. In doing so, some open questions have been identified in respect of the borehole disposal so far which concern the operational loads on the canisters and the nuclear safety of the emplacement into boreholes which are down to 300 m deep. The activities on the POLLUX reference concept as it was developed in the 90s were the starting point for the examinations of the drift emplacement concept. The examinations performed so far indicate already that a balancing of the emplacement concepts drift emplacement and borehole disposal are possible under the above-mentioned aspects of operational robustness. An in-depth analysis of the handling and co-ordination steps which are specified by the choice of repository and emplacement concept still has to be conducted for both concepts.

Additional questions will be raised and examined with this project. This includes the question whether CASTOR containers, which are currently approved as transport and interim storage containers, are suitable for final disposal. Besides conceptual differences, CASTOR and POLLUX containers show comparableness which suggests corresponding examinations. The question regarding the suitability of the very heavy CASTOR containers for final disposal leads to the question of alternatives to the shaft transport. Currently, the only alternative to make the subsurface mine accessible for the transport of such containers is a ramp from the top edge of the site as is intended for the repositories planned in the Scandinavian countries. Within the scope of the project, the possibilities and limitations of the shaft and ramp transport will be studied and examined in respect of their feasibility in a German repository. ■

6.1
Studies on the activation and decay storage of large components from the dismantling of nuclear power plants


Ulrich Hesse

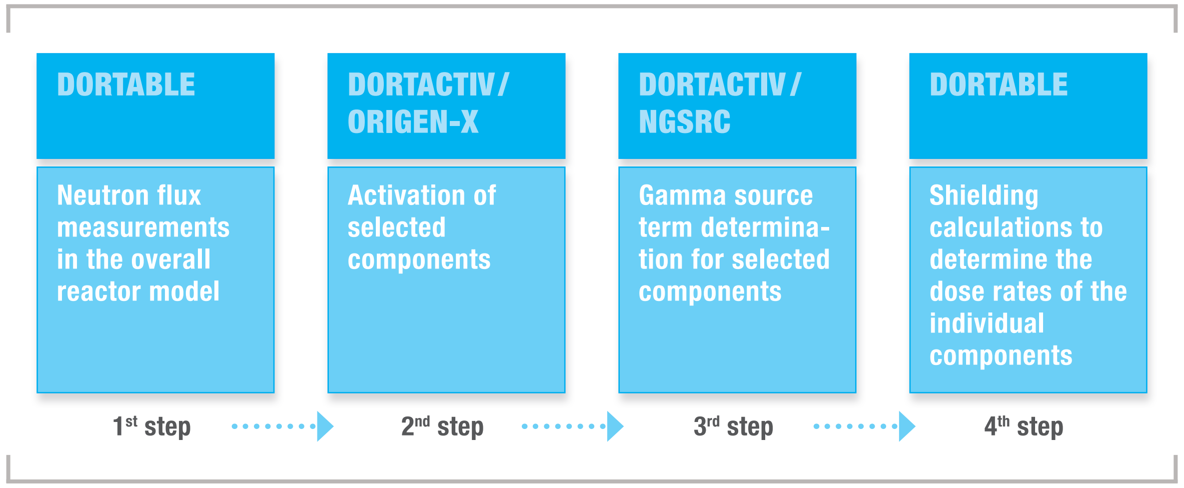

Klemens Hummelsheim


Markus Wagner

In Germany, the so-called decay storage of large components from the dismantling of nuclear facilities has been practiced at the Greifswald site for several years. The utilisation of the radioactive decay and the associated reduction of radioactivity shall reduce the amount of radioactive materials to be disposed and the radiation exposure of the personnel during dismantling activities. To assess the feasibility of these goals, precise information about the relevant radionuclides is required. This can be calculated by using modern calculation methods to determine the neutron flux and the resulting activity. A prerequisite for this is precise knowledge of the material compositions of the components and the radiation history thereof. GRS and Brenk Systemplanung examine by order of the BMU the decay storage to determine, for example, the required and optimal duration of decay storage and the thereby possible reduction of the radiation exposure of operational personnel and of the volume of waste.

Developing calculation methods. In its project on decay storage, GRS is developing a calculation method with which the neutron flux, the activation of components resulting from that – of a reactor pressure vessel (RPV) after 32 years of operation, for example – and the surface dose rate resulting from the activation can be calculated. With this procedure, also the reduction of the surface dose rate in dependence of the duration of decay storage can be determined.

DORTACTIV-2008. Flow chart



DORTACTIV CODE PACKAGE

Fig. 60
An integrated code system was developed from individual internationally renowned codes

Calculation methodology

The calculation system consists of internationally acknowledged individual programmes and libraries which are combined into one closed programme system (see Fig. 60 »DORTACTIV CODE PACKAGE«) by means of newly-developed connecting and interpretation modules

To calculate the quantities relevant for the assessment, a four-level procedure is required which was implemented with the development of the DORTACTIV programme package:

1st step Calculation of the neutron flux in the complete axially symmetrical reactor model in two-dimensional geometry (RZ geometry).

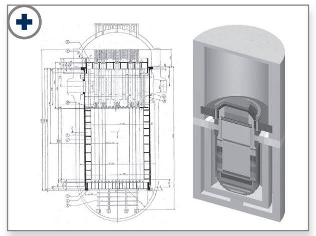
2nd step Activation calculation for each location or for the number of positions attributed to a large component over the lifetime of the reactor.

3rd step Calculation of the gamma ray spectra at each of the selected locations at each time point of the decay storage.

4st step Shielding calculations to determine the dose rate in the vicinity of the large components at different decay storage times.

Validation calculation methods. The calculation methods must be validated by means of recalculations on measured values. A first test was successfully carried out by comparing the calculation results with experimental data of irradiated parts of fuel element structures. Data available from the reactors in Greifswald in the form of dose rates of the components offer additional possibilities of validation.

While the assessment of the decay storage is carried out with the model of a typical pressurised water reactor (PWR), the validation with the Greifswald data is performed according to a geometry model of the Greifswald reactor.



PWR REACTOR PRESSURE VESSEL

Fig. 61
Cross-section and implementation in the calculation model for DORTACTIV
Graphical representation is by means of the KENO3D module

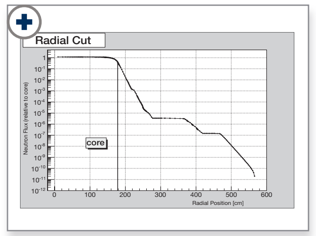
The model of the RPV of a PWR comprises all components which are necessary for the calculation of the neutron flux and for the later activation. The biological shield and an exemplary reactor building are considered in the model to correctly describe the reflection and moderation of neutrons toward the RPV (see Fig. 61 »PWR REACTOR PRESSURE VESSEL«).

First results regarding the neutron flux calculation (level 1)

Applying difference transport librarys. For the neutron flux calculations, a Konvoi-type PWR by KWU with an output of 1,300 MW was chosen. This plant type, i.e. the same configuration, is used in *Isar* 2, *Neckarwestheim*-2 and *Emsland*. The data on the geometry originate primarily from the safety report for this reactor type. The calculations were performed with five different transport libraries of up to 13 to 175 neutron groups. All libraries yield similar neutron flux distributions and remarkable deviations occur only in case of large distances from the core and correspondingly big attenuation factors. The dispersions are due to different library developments but not due to uncertainties of the used codes. The aim is to clarify and to use these dispersions for a variation range of the activation processes.

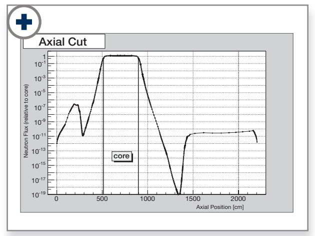
Neutron flux in radial direction. The attenuation of the neutron flux in radial direction from the middle of the core via core barrel, RPV and biological shield to the wall of the idealised reactor building is shown in Fig. 62 »NEUTRON FLUXES - RADIAL«. The flux level is shown in relation to the flux in the core which is indicated by a vertical, red line. As expected, the flux decreases by 8 orders of magnitude from the middle of the core toward the outside to the reactor wall.

Neutron flux in axial direction. In axial direction toward the closure head as well as toward the bottom dome plate, the neutron flux decreases considerably, but unexpectedly increases again by several orders of magnitude (see Fig. 63 »NEUTRON FLUXES - AXIAL«). This behaviour can be observed for the use of all transport libraries due to the streaming of the neutrons between RPV and concrete walls. The axial attenuation of the neutron flux, starting from the upper platform to the roof of the idealised reactor building is shown in Fig. 63. Again, the flux level is shown in relation to the flux inside the core and is indicated by vertical red lines.



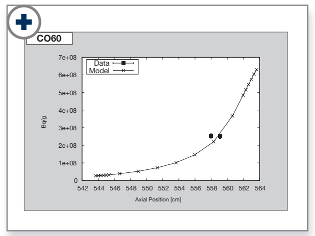
NEUTRON FLUXES - RADIAL

Fig. 62
Distribution of the attenuation factors (total flux) from the middle of the core and radially toward the outside to the biological shield
(library KORLIB with 83 groups)



NEUTRON FLUXES - AXIAL

Fig. 63
Distribution of the attenuation factors (total flux) from the reactor building floor to the ceiling
(library KORLIB with 83 groups)



ACTIVATION SQUEEZE-LOCK NUT

Fig. 64
Measured and calculated Co60-activation by comparison

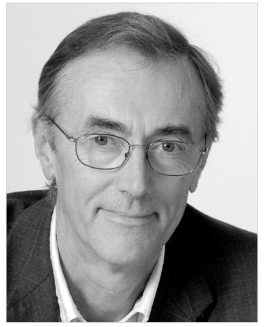
The surprising axial increase by neutron streaming could not be captured with the previously common axial and radial one-dimensional procedures. The future activation results regarding the bottom and closure head of the RPV will show whether this axial increase leads to so high neutron activations that clearance levels are exceeded.

First results of the activation calculations (level 2)

Comparison with experiment. As a first test of the calculation model, the activation of a cap screw and a squeeze-lock nut of the control rod guide thimble of a fuel element were calculated. As basis of comparison the measured data on concentrations of 11 radioactive isotopes were available. As an example of the comparison, Fig. 64 »ACTIVATION SQUEEZE-LOCK NUT« shows the activation of the squeeze-lock nut calculated for the isotope Cobalt 60 which reflects the measured data very well. Also the measured data of other isotopes are mostly reproduced very well. The results of the calculations were published at the conference GLOBAL 2009.

Before long, the first calculations on the dose rate will be performed. The results of these calculations will then be compared with the corresponding data measured at Greifswald. ∎

6.2
Analyses on the nuclear criticality safety of a repository for spent nuclear fuels



Dr. Bernhard Gmal

Dr. Robert Kilger

Both nationally and internationally, long-term safety analyses for final repositories for spent fuel elements in deep geological formations also include the nuclear criticality safety over very long periods. In this context, one essential challenge is posed by the fact that the future development of a geological repository und thus the boundary conditions for criticality analyses can not be forecasted and set with certainty.

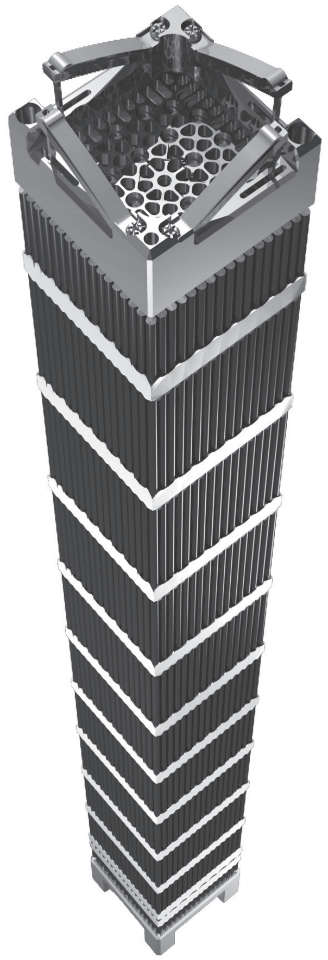
GRS has been conducting extensive interdisciplinary analyses on this issue for some time now. In a current project, GRS works on the further development of a method which allows making statements with probabilistic means on the probability of the occurrence of events which in the post-operational phase of a repository may lead to a criticality. Including the realistic, inhomogeneous burn-up profiles in the probabilistic analyses is an additional aspect of the further development.

Demonstration of subcriticality

Uncertainty of geological developments in the post-operational phase of a repository. Within the scope of long-term safety analyses for a repository into which spent fuel elements shall be disposed, the demonstration of subcriticality in the post-operational phase is to be verified. In doing so, especially possible developments (scenarios) which could lead to criticality must be considered. One example for such a scenario is the ingress of solutions to the stored waste packages. As far as such developments can not be ruled out they must be taken into account in the form of adequate conservative assumptions. If subcriticality cannot be demonstrated on this basis, another procedure must be chosen.

Probabilistic analyses as an alternative to deterministic safety analyses. In this case, the probabilistic analysis can be an alternative. From the current point of view, it may be considered »highly unlikely« that in a final repository for disused nuclear fuels with a residual portion of fissile material of a maximum of 2 weight per cent an uncontrolled nuclear chain reaction may occur, but it cannot be ruled out from the beginning. The reason for this lies in the fact that overall such a final repository contains a sufficient amount of fissile nuclides to form critical masses. With 7.038 x 10^8 years, the half-life of the fissile Uranium isotope ^{235}U reaches far beyond the considered period so that the decay of the fissile material during the post-operational phase cannot be taken as credit. Conclusive evidence for a period of approx. 1 million years can be produced according to the common criticality safety rules only if certain geological events which may cause the ingress of solutions, for example, can be ruled out or rated as so unlikely that they must not be considered any further.

Geological evolution scenarios. With this, also the question of a possible migration and selective deposition of fissile nuclides in the near-field of a vault is posed. The heterogeneity of axial burn-up distributions broadens the range of geometric setups in degradation scenarios which have to be considered in the criticality safety analysis for a repository for spent fuel elements in the post-operational phase of a repository. According to the present state of knowledge, a wider range of such possible scenarios which could also lead to criticality in the end does not seem impossible. If and to which extent such scenarios are actually possible, can only be determined by means of geological and geochemical analyses. In connection with this, the probability of the occurrence of such scenarios has to be assessed, too. To that end, a probabilistic approach is suitable.



FUEL ASSEMBLY

Fig. 65
View of a fuel assembly for a pressuised water reactor (PWR). Fuel assemblies consist i.a. of a bundle of fuel rods that contain the fuel in gas-tight cladding tubes

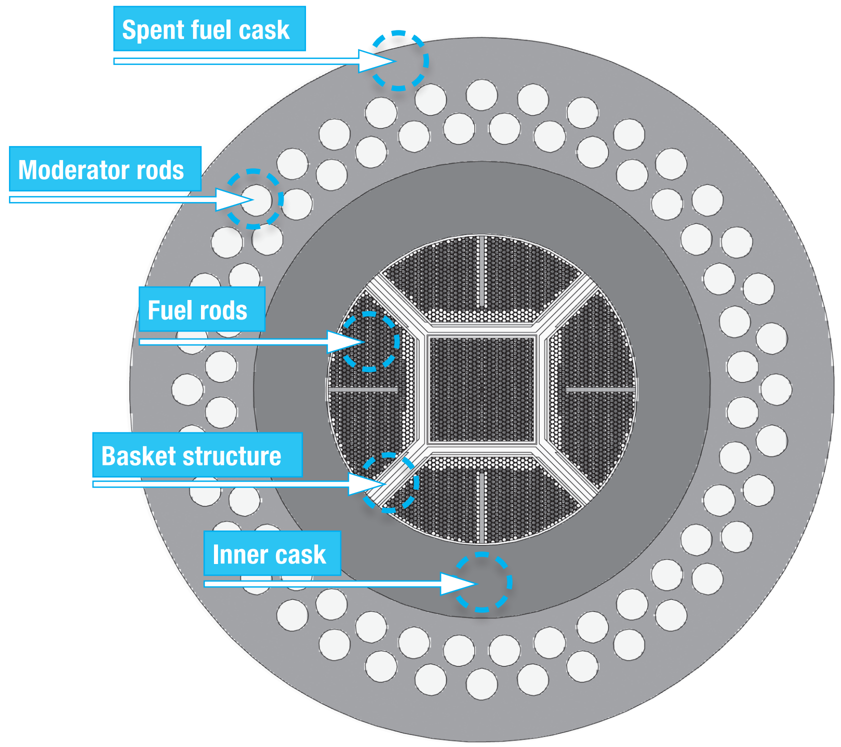
6.2 Analyses on the nuclear criticality safety of a repository for spent nuclear fuels
CONTENT

POLLUX CASK

Fig. 66
Above
Graphic representation of the model »Pollux« cask on the basis of which criticality calculations are performed with the MCNP5 code

Below
Cask of the »Pollux« type. This cask was designed for the final disposal of spent fuel assemblies

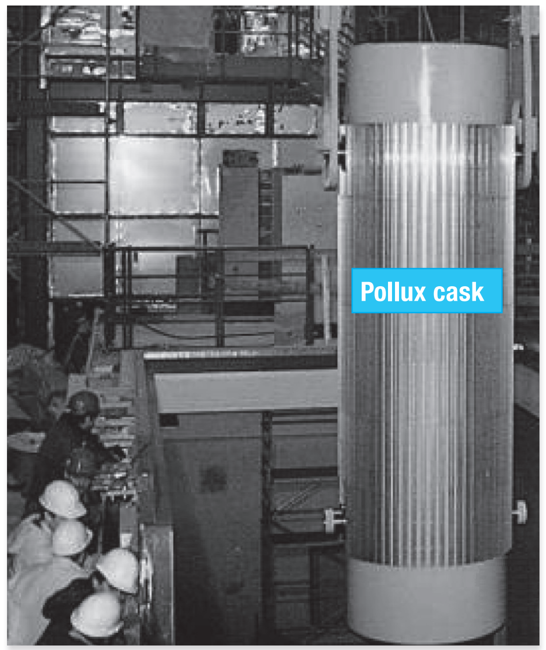
The fuel assemblies are segmented and the individual fuel rods are emplaced in the cask packed into canisters



Physical boundary conditions

Criticality safety analysis. The boundary conditions which can lead to an (over) critical system of fissile materials in a geological repository can be determined as part of a criticality safety analysis. The values relevant for this are, for example, the nuclide conditions of fission nuclides and absorbers, macroscopic concentrations, the overall mass and the geometric distribution of the fissile material as well as the portion of water in a mixture of fission nuclides.

Consideration of burn-up behaviour. Examinations regarding the consideration of burn-ups in the criticality analyses were in the current project aimed at the influence of the axial distribution of the real burn-up of fuel elements which causes an axially varying distribution of the residual fissile material, the fission products and actinide metal in the fuel rods. When considering actinide metals and fission products in the criticality calculations, in case of burn-ups from approx. 15 GWd/tSM, the axial burn-up profile leads to a higher k-value



compared to the averaged burn-up (so-called »positive final effect«). As an analysis of the fission rate distribution along the fuel elements has shown, the upper end of the rod of approx. 85 cm of the active zone determines the reactivity. This effect is important primarily for the loaded and flooded container with an intact structure of the arrangement of the fuel elements and must be considered in the container design (see Fig. 65 »FUEL ASSEMBLY«). Thus, subcriticality is ensured as long as the structure and arrangement of the fuel elements/fuel rods in the container is maintained (see Fig. 66 »PULLOX CASK«). If the fuel element structure dissolves and the chemical composition of the fuel inside the container changes, it has to be differentiated between scenarios which cause an even mixing of the nuclide distribution and scenarios during which the burn-up-induced final effect can even be intensified – such distributions are generally possible.

In this regard, GRS has calculated overcritical k-values of up to 1.04 for generic, very conservative systems. So far, these considerations are only of hypothetical character and must be substantiated further. Within the scope of this study, it has so far not been investigated in detail whether and, if applicable, how likely such a distribution due to geochemical processes is; this will also be subject matter of future research activities.

Analyses and developing methods at GRS

First analysis following the example of probabilistic safety analyses (PSA). To develop a suitable procedure for the assessment of the criticality risk, GRS conducted first analyses of relevant scenarios following the example of probabilistic safety analyses (PSA) in earlier investigations already. In doing so, at first simplified model scenarios were assumed and initial calculation examples were developed on the basis of simply estimated input parameters and their uncertainties.

Introduction of a time-depending model. In the project introduced here, these activities were continued and specified. So a time-depending model was introduced and individual input parameters were quantified on the basis of calculations or experimentally determined values. Due to the long forecast period, this procedure requires the use of assumptions and expert estimates; however, the use of PSA methods offers, among other things, the possibility to better justify the exclusion of certain scenario developments within the scope of the demonstration. In addition, important parameters of individual scenarios which have a particularly strong influence on the probability of the occurrence of a criticality excursion can be identified and, as a consequence, be analysed.

Site-independant generic application examples. The currently conducted analyses on the probability of the occurrence of criticality in the post-operational phase served primarily the further development of a method for the application to the system of containers with fuel elements in a geological repository (»*Behälter mit Brennelementen in einem geologischen Endlager*«). In parallel, the examples of use regarding the uncertainties of input parameters and their time dependence in particular which had been analysed earlier were further specified and – as far as possible – quantified with substantiated data. In this case, these are always conditional probabilities in the case of which the initiating event »ingress of water to the containers« was assumed as a given without further consideration. The time histories of these probabilities show that the greatest values for the probability of occurrence arise within approx. 10,000 years after ingress of water to the examined generic repository. In connection with the PSA, also corresponding uncertainty and sensitivity analyses were carried out and values for the statistical upper 95 %/95 %-tolerance limit were determined.

Calcuations relating to the disposal of research reactor fuel with initial high enrichment. For a final disposal of fuel from research reactors with a high initial enrichment of up to 93 weight per cent ^{235}U in uranium, special conditioning measures will be required to obtain a sufficiently low value for the probability of criticality in the post-operational phase within the scope of calculated demonstration. In view of the high ^{235}U enrichment of the fuel elements of research reactors, suitable concepts to ensure long-term subcriticality must be developed for a final disposal in crystalline rock or clay formations. To this end, a mass limitation or a backfilling with depleted uranium in a suitable chemical form come into consideration. In case of a final disposal in rock salt, a demonstration of subcriticality is possible if only one fuel element per container is assumed and an accumulation of several container inventories seems sufficiently unlikely.

All calculations and parameter studies performed within this scope are of generic character and refer neither to a certain final repository site nor to a specific host rock. They serve the further development of the methods and the provision of data on criticality to enable an appropriate handling of the issue of the criticality safety of nuclear fuels within the scope of long-term safety analyses.

GRS code for the calulation of the possible consequenses of criticality in a repository under development. In addition to the above-mentioned analyses, GRS is developing a computer programme with which the time history, energy release and the resulting radionuclides as consequences of a hypothetically assumed criticality under geological conditions can be calculated. With this, clues regarding a possible impairment of the barrier effectiveness and the extent thereof can be gathered by means of a critical excursion.

First results and outlook

Valitity of deductions. The activities have shown that the PSA methods developed by GRS can be adapted to the questions of long-term safety in a geological final repository. On the other hand, the so-far examined examples of use show that the validity of the statement essentially depends on few input parameters which bear great uncertainties. One essential function of the further activities is therefore to quantify the so far still inexact parameters as far as possible and, in particular, to consider substantiated probability values for the initiating events.

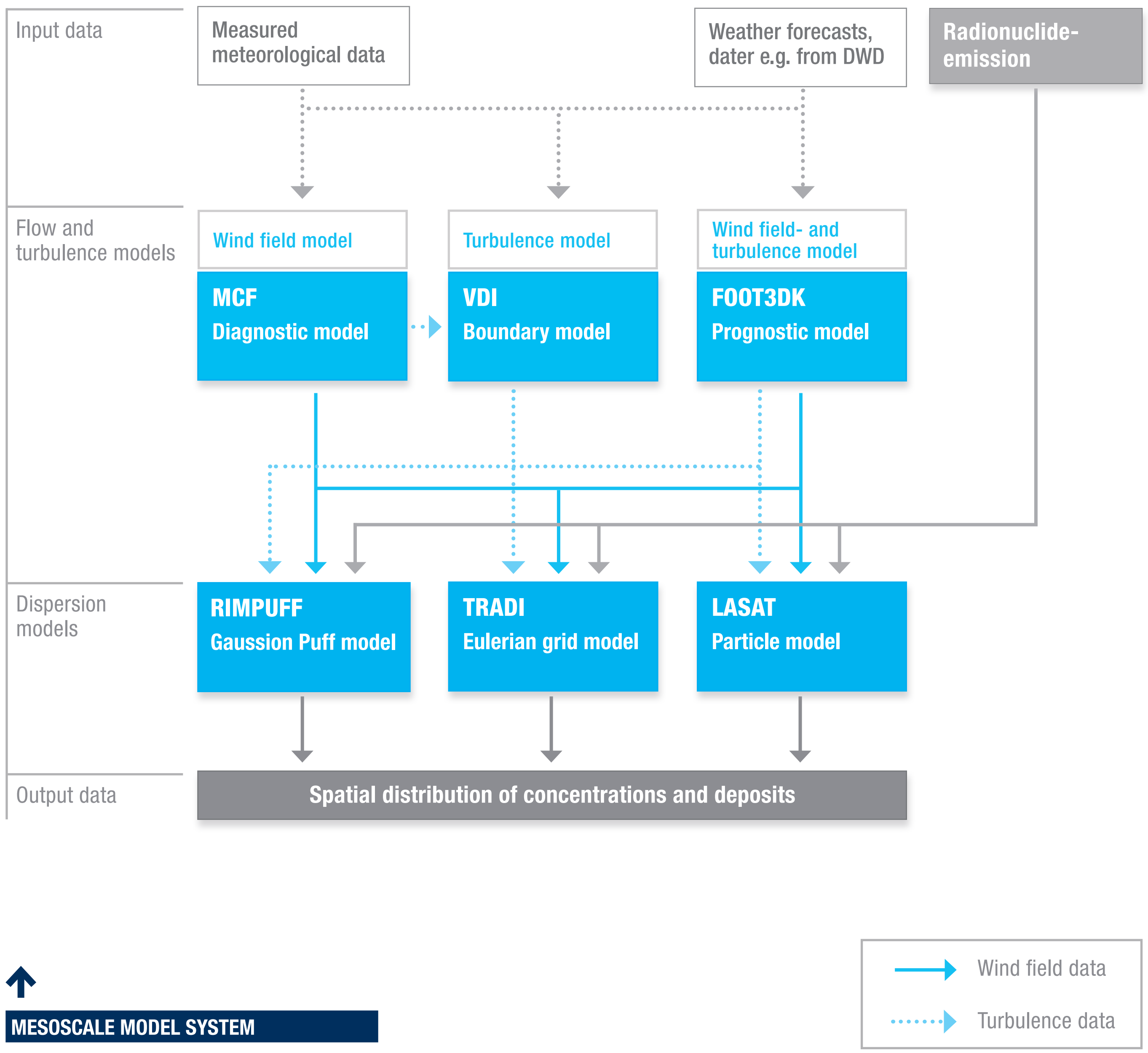
Code validation. Another challenge in respect of the criticality safety analyses for a repository for spent fuel elements is the validation of the calculation method. On the one hand, this concerns the very long periods to be considered and on the other hand, the sufficient validation in experiments with repository-relevant nuclides – here, above all, ^{35}CL if a final repository in salt is taken into consideration. ∎

6.3
Modelling of the atmospheric dispersion in case of accidents

Dr. Reinhard Martens

After radioactive materials have been released into the atmosphere due to an accident in a nuclear facility or a transportation accident, a prediction as soon and as precise as possible on the atmospheric dispersion of the released radionuclides is indispensable for the planning of disaster response measures. Also in connection with so-called »accident management« measures in nuclear power plants during events and accidents, dispersion forecasts are helpful to identify the moments, for example, where a deliberately initiated release from the plant (e.g. pressure relief of the containment) leads to the lowest possible radiation exposure of the environment.

In the 90s already, GRS developed two model chains to diagnose and predict the dispersion and deposition of airborne radioactive substances for a distance of up to 30 km. This so-called mesoscale model system was successfully used for the recalculation of meteorological measurement campaigns and of dispersion experiments. Both GRS model chains could be initiated with the numerical weather forecast data of the until 1999 operational *Deutschlandmodell* (DM) of the *Deutscher Wetterdienst* (DWD). That way it was possible to realistically determine the dispersion and deposition of radioactive materials for the forecast period of the weather forecast model which captured several days. After commissioning a new, improved forecast model at DWD, both model chains were updated, optimised and validated within the scope of a research project promoted by the Federal Ministry of Economics.



MESOSCALE MODEL SYSTEM

Fig. 67
Possible links between individual components of the model system

6.3 Modelling of the atmospheric dispersion in casa of accidents CONTENT

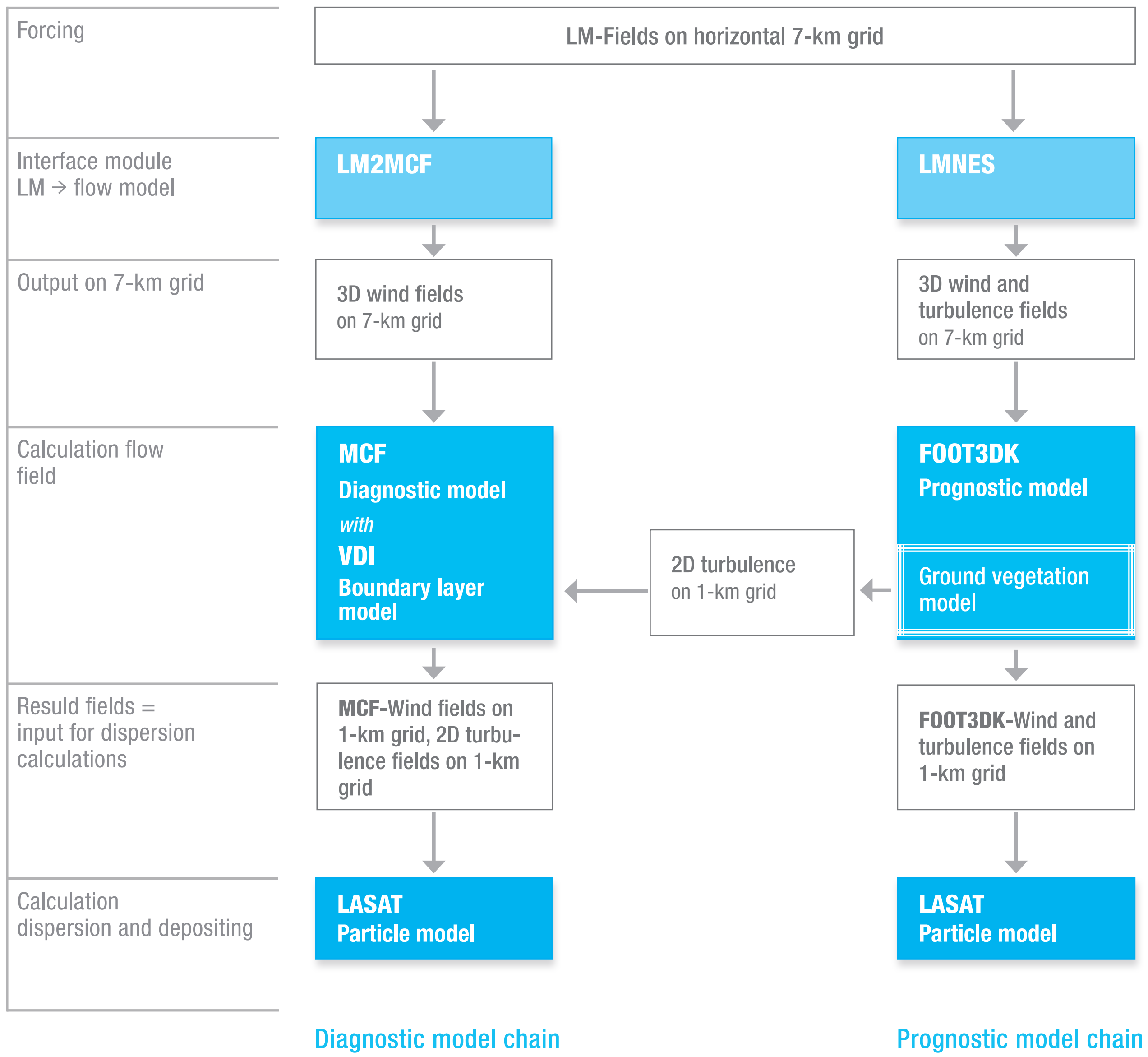
GRS's mesoscale model system

Components of the model system. GRS's mesoscale model system consists of a diagnostic and a prognostic model chain. As meteorological input data for the model system, either measured data – of the meteorological instrumentation of a nuclear power plant, for example – or the result fields of numerical weather forecast models can be used. The model system comprises different flow models to describe the wind field as well as several subsequent alternative dispersion models with which, along the transport path, the dilution of noxious pollutants due to turbulent motions in the atmosphere and the deposition of the materials can be simulated (see Fig. 67 »MESOSCALE MODEL SYSTEM«).

Diagnostic model chain. When calculating the flow field with a simple diagnostic model, three-dimensional wind fields are diagnosed based on the wind data which are already known (i.e. measured or given) for some positions within the calculation area. If initialisation takes place with wind data from a weather forecast model, also wind field forecasts are possible with diagnostic models. The diagnostic GRS model chain works with the mass-consistent flow model MCF (Mass Consistent Flow). The calculated wind fields fulfil the continuity equation (conservation of mass). However, this model type does not render turbulence fields; these are provided by the turbulence model of VDI guideline 3783, Part 8.

Prognostic flow model. As an alternative to the diagnostic model chain, the flow field can also be calculated with a physically more complex prognostic model which, besides the conservation of mass, also takes the conservation of momentum and of energy into account. Furthermore, the two model types differ in respect of considering the lower limitation of the calculation area which can be represented by ground, expanses of water, natural cover and development, for example, and which is very heterogeneous depending on the land use. So, the thermal influence of heterogeneous surfaces on the flow and turbulence at ground-level which is included in prognostic flow models as lower boundary condition is not considered in diagnostic models. Thus, thermally induced flow patterns can only be simulated in a realistic manner with prognostic models. In addition to the medium wind field, this model type also yields turbulence values (e.g. diffusion coefficients) and allows the calculation of the future development (forecast) of the wind and turbulence fields. To calculate the flow field, GRS's prognostic model chain uses the non-hydrostatic model FOOT3DK (Flow Over Orographically Structured Terrain, 3-dimensional version of the University of Cologne).

Models to predict dispersion. For the dispersion calculations subsequent to the flow field calculations also, different dispersion models are available such as the simple Gaussian puff model RIMPUFF, the Euler model TRADI, or the particle model LASAT® in the GRS version. GRS prefers to use the two model chains FOOT3DK→LASAT and MCF→LASAT for dispersion calculations. Despite the model-induced weaknesses of diagnostic flow models compared to the physically more complex prognostic models, the use of diagnostic model chains is often preferred for fast dispersion forecasts, for example in decision support systems, since due to the long calculation time and demand in resources, the use of prognostic models still does not seem practicable.



OPTIMISATION OF MODEL SYSTEM

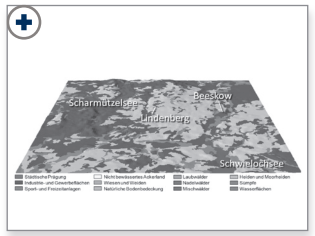
Fig. 68
Coupling of the two model chains to improve the accuracy of diagnostically calculated wind fields

Adaptation and optimisation of the model system

New boundary conditions due to improved weather forcast models. With the commissioning of newer, improved weather forecast models at DWD, the boundary conditions for the overriding initialisation of the two model chains have changed significantly: The new DWD model (Lokal-Modell, LM) has other model physics and a more refined, spatial and temporal resolution than the DM which has been used for the initialisation of the two model chains so far. GRS has adapted the interface modules with which the two model chains are linked to the Lokal-Modell (see Fig. 68 »OPTIMISATION OF THE MODEL SYSTEM«).

Optimisation of thermeral influence calculations. Within the scope of the adaptation and optimisation activities, it was also verified if the results achieved with the prognostic model chain in respect of the thermal influence of heterogeneous surfaces on the flow and turbulence at ground level can be used to improve the accuracy of the wind fields calculated with the diagnostic model. As a standard, the diagnostic model chain only includes turbulence information which is derived only from the LM initialisation on the 7-km-grid. The thermal influence of heterogeneous surfaces with a more refined spatial resolution (e.g. 1-km-grid) can not be taken into account with the MCF simulations used here. In the prognostic model chain, however, surface structures of such a fine resolution are captured with the high-resolution soil-vegetation module integrated in FOOT3DK. By using the boundary layer parameters thus gained on the 1-km grid to initialise the MCF, repercussions on the wind field at ground level can also be taken into account with MCF. One example for such repercussions is the local, thermally-induced flows at ground level which emerge due to intensive solar radiation on a summer's day. The effectiveness of this procedure could be demonstrated with the example of corresponding diurnal data gained during the meteorological measurement campaign LITFASS2003 (Lindenberg Inhomogeneous Terrain - Fluxes between Atmosphere and Surface: A long-term study).

Validating the calculations. The validation of the activities to adapt and optimise the two model chains was effected on the basis of the data available for LITFASS2003 and the calculated fields of LM. The investigation area of this campaign, which is located approx. 60 km south-east of Berlin and is about 40x40 km², is characterised by outstanding data availability for an investigation period in the summer. The area shows weak orographical structures and pronounced heterogeneity in respect of land use with changes of forest areas, agriculturally used farming and grazing areas as well as individual lakes and rural settlements (see Fig. 69 »LAND USE LITFASS2003«). LITFASS2003 comprises both weather conditions in the early summer with weak winds and pronounced weather conditions with strong west winds with low day temperatures; thus it provides a wide range of different situations in respect of intensity and variation range of thermally-induced turbulent flows.



LAND USE LITFASS2003

Fig. 69
Land use in the LITFASS2003 area
The pronounced heterogeneity of the surface of the 30x30 km² investigation area is clearly visible

Summary

Successful updating of the mesoscale model system. The mesoscale model system used by GRS for the diagnosis and forecast of the dispersion and deposition of airborne radioactive substances in structured terrain was successfully updated and adapted to the current structure of the numerical weather forecast models of DWD. With the investigations, the forecast capability of both model chains and the high level of accuracy of the calculated distributions of the airborne concentration and of the deposition after radionuclide release were not only maintained but also improved. Due to the procedure which was developed for the diagnostic model chain to consider the thermal influence of the surface inhomogeneity, future uses of diagnose-based model systems in integrated decision support systems of the entire environmental field (e.g. in RODOS/RESY) can be expected. ■

6.4 EU-Project MICADO

Dr. Guido Bracke

Many European states intend direct geological disposal of high active spent fuel elements into repositories. After decay storage direct, disposal means that the fuel elements will be stored in special containers and moved to the repository without further treatment. When making a safety-related assessment of this form of final disposal, it has to be considered that in case of a failure of the containers, the fuel elements could theoretically get in contact with water occurring in the host rock of the repository. In such a case the radioactive material might dissolve in the water and the thus generated solution could reach the groundwater located in the upper layers of the earth in the long term. Therefore, methods and theoretical models to describe the potential dissolution processes of fuel elements have been developed at the international level for several decades. Data which were gained in experiments form the basis for these developments. The EU-sponsored project MICADO (Model uncertainty for the mechanism of dissolution of spent fuel in a nuclear waste repository) studies in international cooperation if there are sufficiently reliable models to assess the corrosion resistance of spent fuel elements. Together with partners from France (IRSN) and Belgium (Bel V), GRS has been participating in the MICADO research activities since 2006.

Project MICADO

Co-operation partners. More than 20 organisations from seven countries participate in MICADO. Within the scope of MICADO, experts from the fields of electrochemistry, geochemistry and radiochemistry evaluate the different approaches to predict long-term processes on spent fuel elements and containers in respect of their applicability in safety analyses for repositories. The comparison of the approaches is of particular interest from both the operators' and the authorities' point of view.

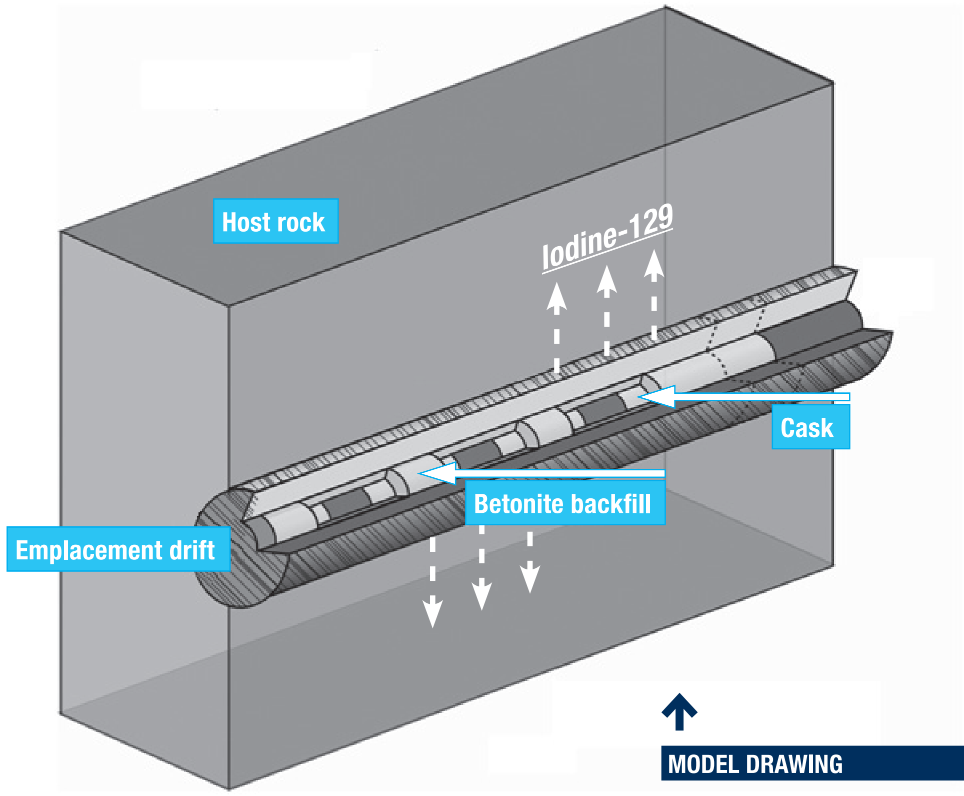
Tasks and objective. Key task is the assessment of the uncertainties in the experimental data base and in the underlying models. An extrapolation of the empirical data which were measured within few years to periods of hundreds of thousands of years is a challenge. Also the mechanistic models which transfer experimental observations to long periods show uncertainties. Therefore, MICADO is primarily aimed at assessing the quality of the experimental data and approaches by comparing the different approaches and underlying hypotheses. In this context, two forms of uncertainties are examined: Uncertainties which result from experimental data and uncertainties which result from different predictions of the models. To this end, the institutions involved carry out detailed analyses of fuels and modellings to improve the experimental data base on the one hand, and to reduce the modelling uncertainties on the other hand. In addition, the MICADO project allows the participating organisations to exchange their findings on the existing approaches and methods in the analysis of the long-term corrosion behaviour of containers and fuel elements. Finally, MICADO shall also contribute to the identification of issues relating to future research and point out which of the existing uncertainties can be further reduced.

Contributions by GRS and its partners

Studies into the release of the radionuclide iodine-129. In co-operation with its partners Bel V (Belgium) and IRSN (France), GRS examines the uncertainties of the simulated release of the radionuclide iodine-129 on basis of the French repository concept. The concept provides for the emplacement of disposal containers with spent fuel elements in drifts and sealing the latter with bentonite against ingress of groundwater (see Fig. 70 »MODEL DRAWING«).

Codes MELODIE and SUSA

The modelling of radionuclide release and transport according to the French concept was carried out with the IRSN program MELODIE. GRS contributed its expertise with the GRS probabilistic tool SUSA for the analysis of uncertainties of the modellings.

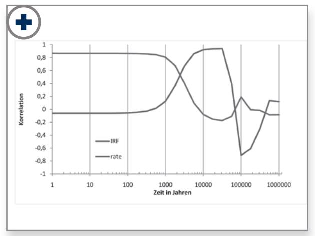


MODEL DRAWING

Fig. 70
Schematic representation of an emplacement drift with release (arrows) of the iodine 129 from the casks into the surrounding host rock

MELODIE. The simplified modelling of the release of the radionuclide Iodine-129 with the programme MELODIE considers as input parameter the inventory, a so-called »instant release fraction« (IRF), as well as a long-term dissolution rate of the fuel element and the associated uncertainties.

SUSA. The program package SUSA analyses the impact and interaction of parameter uncertainties on the uncertainty of the modelling results. With the additional sensitivity analysis the uncertainties of the parameters can be ranked according to their influence on the uncertainty of the modelling result.



MODELLING RESULT

Fig. 71
Correlation of the release of Iodine-129 from the host rock with the instantaneously released fraction (IRF) and the dissolution rate, respectively

Methodology. The distribution functions of the estimated uncertainties were assigned to these parameters. The probabilistic input values for the parameters were determined with the GRS program package SUSA. The modelling results for the released activity of Iodine-129 in Becquerel (Bq) obtained with the programme MELODIE using these values were evaluated. This allows to identify the link between the uncertainty of the modelling results and the uncertainty of the input parameters (see Fig. 71 »MODELLING RESULT«).

Results

The results of the analyses show that the activity release of Iodine-129 into the host rock within the first thousand years is correlated highly (close to 1) with the instant release fraction (IRF). The dissolution rate (rate) is irrelevant in this period. The dissolution rate of the fuel element is decisive for the activity release from approx. 2,000 years. Both influences are becoming meaningless for the modelling result form approx. 30,000 years.

Assessment of the results and outlook. The probabilistic sensitivity analysis with SUSA was applied as an example to a modelling of radionuclide release and transport with MELODIE with three parameters. The actual complexity of the analysis is highly dependent on the model properties and on the number of uncertain parameters included in the analysis. From GRS's point of view, it could be successfully demonstrated within the scope of MICADO that the used model calculations are accessible for a probabilistic analysis.

An in-depth use of the sensitivity analysis taking more parameters into account could demonstrate unexpected dependencies on the one hand, and the relevance of parameters for modelling results on the other hand. From the regulatory perspective the uncertainties and interrelationships related to the release of radionuclides within a safety case can be identified. ■

6.5 The regulators perspective on research and development on the long-term safety of repositories for radioactive wastes

Dr. Martin Navarro

Regulatory authorities as well as the technical expert organisations who are commissioned by these authorities do not only create the regulatory framework for repository implementation and the development of safety cases but also undertake independent research and development activities in this field of expertise. By doing so, the regulator develops and maintains the expertise which is necessary to prepare rules and guidelines, to assess licensing applications, to identify further needs for research and development, and to make regulatory decisions on a scientific and technical basis.

In a series of research and development projects on behalf of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU), GRS has conducted various research and development activities on the assessment of post-closure safety cases. In the present contribution, some key issues of these activities will be described using the example of project SR 2548 »*Verfolgung und Bewertung der Fortentwicklung des Standes von Wissenschaft und Technik beim Nachweis der Langzeitsicherheit von Endlagern*«.

An international field of research

Global co-operation. The development and assessment of safety cases covers a wide range of issues which can only be tackled within an appropriate timeframe by way of international co-operation. From the regulatory point of view, the international perspective is of particular importance in order to capture and develop the state-of-the-art in science and technology and to finally apply it on the national level as a basis for regulatory assessments.

International projects and working groups. The collaboration with international projects and working groups therefore was an essential part of project SR 2548, which was completed in November 2008 after a duration of three years. The project included the involvement of GRS experts in working groups and projects of the IAEA, the OECD/NEA (AMIGO, EBS, and P&T) and the EU (RedImpact, PAMINA), as well as a co-operation with the French partner organisation IRSN. In addition, international developments were monitored by means of detailed analyses of selected safety reports and regulations. For example, GRS has examined to what extent the recommendations of the Nuclear Energy Agency (NEA) of

OECD and the IAEA on the general content of a safety case have been incorporated in national regulations and safety reports of the countries France, Sweden and Switzerland. Based on this work, needs for the further development of the German safety requirements were concluded.

Working group on scenario development. With regard to the heterogeneity of country-specific approaches towards the development and assessment of safety cases, and with regard to the heterogeneity of viewpoints among German organisations there is a need for harmonisation or at least for a mutual understanding in terms of terminologies, strategies, concepts, methods, requirements, and expectations. With the publication of a position paper on the issue of human intrusion into a repository, the working group on scenario development (*Szenarienentwicklung*), which was co-ordinated in project SR 2548 and has been organised and chaired by GRS since 1997, has reached an important milestone towards harmonisation within Germany.

Code development

TOUGH2 und MARNIE. One sub-task of project SR 2548 was the development, testing and application of numerical codes for the simulation and assessment of processes in repository systems. Amongst other safety assessment tools, GRS uses the flow and transport codes TOUGH2 and the in-house development MARNIE. The code TOUGH2 allows detailed process analyses for the multi-phase flow in porous media whereas the code MARNIE is primarily suitable for system-level models, probabilistic analyses and modelling of complex drift networks. Both codes are highly flexible and can be adapted to specific problems. Thus, they are well-suited for regulatory considerations of alternative, conceptual or physical models within licensing procedures. For both codes, post-processing tools were developed in project SR 2548 which allow for the required flexibility.

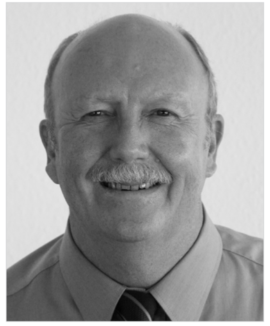
Integration of safety-relevant processes. To enhance the capability of simulating thermal-hydro-mechanical interactions and gas migration processes in clay formations, several modifications of the code TOUGH2 were developed. This includes a coupling with the geo-mechanical code FLAC3D for the simulation of three-dimensional, coupled thermal-hydro-mechanical interactions. In project SR 2548 and in the EU project PAMINA, the mechanism of micro-crack dilation, which may significantly affect the migration of gas in argillaceous host rock and the pressure evolution inside the repository, was implemented into the code TOUGH2.

Code qualification. Code qualification was an essential part of project SR 2548 and included participation in the international benchmark exercise »couplex gaz«. This benchmark exercise demonstrated the high capability of the TOUGH2 version used by GRS to simulate the migration of gas in the near field and far field of a repository in clayey host rock.

Relation to other projects

Project SR 2548 closely interacts with other GRS projects. While project SR 2548 investigated general, non site-specific aspects, the joint project VerSi focuses on safety analyses for well specified (generic) repository sites. The international collaboration in project SR 2548 on safety case issues was supported and supplemented by the participation of GRS experts in working groups of the OECD/NEA and the IAEA within the framework of other projects. In return, the project SR 2548 provided input to these work groups. The numerical codes developed the project SR 2548 were applied within the scope of expert activities for the Asse site. ∎

7. Projects and international programmes



Ulrich Erven

One of GRS's core tasks is it to provide interdisciplinary knowledge, advanced analysis methods and qualified data to assess and improve the safety of technical facilities and to thus further develop the protection of humans and the environment against the dangers and risks of such facilities. The scientific problems connected with that are complex and can often only be solved by means of interdisciplinary co-operation. Experts of different scientific disciplines jointly work on projects and combine the results of their analyses and assessments into integral safety statements. To effectively organise this co-operation, a comprehensive expert organisation is required. At GRS, this role is assumed by the Projects and International Programmes Division.

Tasks of the Projects and International Programmes Division. In this regard, the division's tasks encompass the co-ordination and control of work programmes, the management of resources and quality assurance. In addition, the division is also responsible for the comprehensive revision of scientific-technical issues in cross-section projects – on the further development of safety requirements, for example, or the representation of the state-of-the-art in science and technology.

International tasks. The tasks with international component include the project management for international projects as well as the co-ordination of GRS's international activities. Here, the further extension of the European Technical Safety Organisation Network (ETSON) which has been extended to five parameters within the reporting period is to be highlighted in particular.

European Technical Safety Organisation Network (ETSON). Besides GRS, the current members of ETSON are the organisations IRSN (France), Bel V (Belgium), VTT (Finland) und UJV (Czech Republic). Aim of ETSON is it to promote and further develop the scientific and technical co-operation between the European TSOs in the field of nuclear safety. This objective shall be achieved by, among other things, the systematic exchange of R+D results as well as of experiences gained from operating nuclear facilities. In 2008, the implementation of a »European Operating Experience

Feedback Systems« was also field of activity – in addition to the increased co-operation in work groups (»Safety Assessment Guide«, »Identification of Research Needs« and »Knowledge Management«). As TSO representative, ETSON also intensively participates in the »Sustainable Nuclear Energy Technology Platform« (SNE-TP) of the EU and is represented there in essential decision-making and working bodies.

Rising demand for nuclear safety expertise. The globally increasing demand for nuclear safety expertise is gaining more and more importance for the TSOs as they intensively follow the new safety-related developments: The processing of international projects also contributes to the expansion of expertise which is requested by national authorities in view of the assessment of new reactor concepts. Comparable, positive effects also ensue in respect of the support of authorities within the scope of supervisory procedures. GRS takes this development into account and acquires – both alone and in co-operation with the ETSON partners – EU projects and submits applications in answer to invitations to tender by foreign authorities. To successfully launch international projects on the basis of public invitations to tender, GRS is developing acquisition concepts which shall ensure an effective co-ordination of the contributions by GRS's different fields of expertise as well as the project partners. Our success proves that efficient tools and processes have been worked out. Due to the optimisation of the internal structure of the Projects and International Programmes Division and the establishment of a resource management for experts to be deployed internationally, this development will be further promoted.

Examples of international programmes. Within the scope of GRS's international activities, great importance is attached to the programme regarding the reactor safety in Eastern Europe (»*Reaktorsicherheit Osteuropa*«) which is currently being geared towards a new direction. The same applies to the programme on physical protection and disposal of nuclear material (»*Physischer Schutz und Entsorgung von Nuklearmaterial*«) which is being conducted as part of the global partnership of the G8 countries for facilities in Russia. In-depth information on both programmes will be presented in the following in a separate technical contribution.

The challenge of maintaining competence. In addition to the activities in connection with interdisciplinary projects and international projects, the Projects and International Programmes Division also deals with issues relating to education and training to maintain competencies. With well-trained, experienced GRS experts retiring, expert knowledge gets lost which has to be substituted. Authorities, operators and other expert organisations, too, face similar developments. To maintain and further develop its competencies, GRS has developed a modular training and qualification concept which has been implemented for some time now. Moreover, GRS has developed a one-year trainee programme during the reporting period which, with its combination of elaborate internal and external trainings and training-on-the-job, imparts broad knowledge in the field of reactor safety. The trainee programme was started for the first time on 1st January, 2009. The Projects and International Programmes Division is responsible for the development, organisation and evaluation of the programme. In addition, the division's officers intensively get involved with its implementation. ■

7.1 Status report on knowledge management

In 2008 again, GRS further extended its information and knowledge management. In doing so, the activities were focused on the further development of the GRS intranet portal (the »GRS Portal«). The number of documents made available in the portal, of team sites for communication and co-operation in teams and the current information have increased considerably. Furthermore, GRS has set up additional portals which can be accessed by other organisations. These external portals have proved as effective support for the communication and co-operation of several organisations. The present contribution provides an overview of the most important innovations relating to the GRS information and knowledge management.

Dr. David Beraha

Dr. Peter Puhr-Westerheide

Internal GRS project portals

Internal information and knowledge management. At the beginning of each GRS project, an individual website (the so-called »project portal«) is created where the information and documents (especially the so-called »project files«) pertaining to the project are made available. Until 2008, project portals had been set up on a separate MS project server. The change to a new version of the SharePoint Server in 2008 enabled the integration of the project portals into the GRS portal. This led to considerable advantages for the users: The project list now corresponds to the SharePoint standard with the known and appreciated extensions of the grouping, filter and sorting possibilities. Also the project sites with the associated libraries are now created in a SharePoint-compliant manner.

Data synchronisation. The synchronisation of the master data between the project portals and the GRS company software SAP was developed as an additional function. With this function, it became possible to extract the master data of a project from SAP and to represent them in the respective project portal. That way, the double storage of master data is being avoided.

Advantages. The further development of the portal solutions led to an increased efficiency of the administrative handling of the projects. So, for the projects GRS is working on by order of the Federal Ministry for the Environment and Nature Conservation and Nuclear Safety (BMU), the essential parts of the central files are replicated on the BMU intranet server on a daily basis. With this innovation, the previously required forwarding of altered project documents is not necessary anymore.

External GRS portals

External information and knowledge management. GRS has observed an increasing demand in platforms to support the co-operation with other organisations. Such platforms are provided by GRS in the form of portals which are made available to authorised external users via the Internet. The primary purpose of the portals consists in the facilitation of the exchange of information on certain topics or projects with ordering parties and project partners. A central access management for external users on the basis of the »Active Directory (AD)« has proved its worth for all portals on the GRS extranet. The provision and maintenance of the portal tools are performed by Solutions for Research (SfR).

IRRS mission by the IAEA. By the end of 2008, GRS had already set up 15 such portals the major part of which was established by order of the BMU. In this context, the portal for the German IRRS mission of IAEA is to be pointed out in particular. Due to the facilitation in the form of a fast and reliable exchange of documents and organisational information, the use of this portal has significantly contributed to the smooth course of action of the IRRS mission.

BMU »Reactor safety portal«. The »Reactor safety portal« (RS-Portal) of the BMU Division for Reactor Safety, which was developed by GRS within the scope of a project and which was integrated into the ordering party's intranet is still supported with solutions and suggestions. After the first version of the RS-Portal had been finalised, BearingPoint appraised it order of the Federal Office of Administration, Federal Office for Information Technology (*»Bundesverwaltungsamt, Stelle für Informationstechnik«* – BIT). In 2008, the appraisal was the basis for the BMU's decision to implement an overall intranet portal with the Microsoft tool SharePoint analogously to the RS-Portal developed by GRS.

Information officers

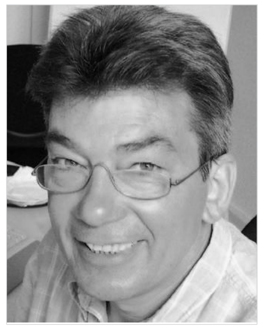
Function of the information officers. In the reporting period, GRS has further developed its internal information structures by introducing information officers. The information officers were appointed by the individual GRS divisions and are assigned different tasks. They support the work units responsible for the knowledge management and the communication within the company by providing up-to-date, substantiated information relating to the content of the websites to be created and of general parts of the GRS portal. In addition, they coordinate the creation of the so-called »knowledge pages« (»*Wissensseiten*«) of the GRS portal. The knowledge pages are prepared and maintained directly by the divisions themselves. They concisely describe the respective field of expertise and can thus also be used as basis for the technical information on both the Internet and intranet.

Outlook

The methods and tools of GRS's information and knowledge management have become well-established by now. The support of the users of this infrastructure as well as the development of team pages or portals for internal and external purposes will remain the focus of our activities. Furthermore, the »Information Brokering«, i.e. the provision of technical contents and portals for external ordering parties, will become increasingly important for GRS. Finally, new knowledge management methods which develop in connection with the evolution of the Internet towards social and semantic networks will be continuously reviewed as to their possible use in the company network. ■

7.2 International programmes


Edmund Kersting


Dr. Hartmut Melchior


Dr. Hartmuth Teske

Nuclear safety, the reliable waste management and the safe storage of nuclear material pose global challenges. These challenges can only be tackled by way of safety partnerships and joint efforts which go beyond national interests. It is imperative to continuously improve the already achieved safety standard, to safely take older facilities out of operation and dismantle them in an environmentally compatible way. Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) therefore advocates international co-operations. Within the scope of manifold bilateral co-operational relationships as well as by participating in multilateral organisations and committees such as the International Atomic Energy Agency (IAEA), the Nuclear Energy Agency (NEA), the Organisation for Economic Cooperation and Development (OECD), G8 and the EU, GRS actively participates by further developing scientific findings and methods.

Objectives of international Co-operation

As a scientific and technical centre of competence for nuclear safety and security, GRS is the leading expert organisation of the Federal Government. At both the national and the international level, GRS is a much sought after partner for scientific and technical co-operations, the definition of safety standards, safety assessments, risk reduction measures and for the strengthening of independent nuclear authorities and expert organisations.

Objective of the international co-operation is

- to analyse the international development of nuclear technology in an expert manner and to further develop the state-of-the-art in science and technology,
- to broaden our own knowledge base,
- to preserve and further extend our expertise, and
- to use the international division of labour to solve important generic, safety-related problems by bundling the resources.

International co-operation relating to the safety of Western-type reactor plants

European Technical Safety Organisations Network (ETSON). Together with its French partner Institut de Radioprotection et de Sûreté Nucléaire (IRSN) and their joint subsidiary RISKAUDIT, GRS is the core of an efficient scientific and technical expert organisation in the fields of nuclear safety, security and waste management in Europe. On 29 May 2006, IRSN and GRS founded, together with the Belgian expert organisation Bel V, the European Technical Safety Organisation Network (ETSON). In November 2008, the Finnish Technical Research Centre (VTT) and the Czech Nuclear Research Institute (UJV) joined the network. The objectives of ETSON are:

- to establish a suitable forum for the exchange of safety assessments, R&D results, and of experiences and corresponding technical and scientific expert opinions in the field of nuclear safety,
- to contribute to the harmonisation of nuclear safety practices,
- to jointly initiate and conduct research programmes on nuclear safety, and
- to further extend a European scientific and technical network in the field of nuclear safety.

Brasilien: Comissão Nacional de Energia Nuclear (CNEN). GRS supported the Comissão Nacional de Energia Nuclear (CNEN) in Brazil with the analysis of superordinate safety-related issues regarding the pressurised water reactor Angra-2. In addition to the transferability of German operating experience to the plant Angra-2, methods for reviewing a probabilistic safety analysis (PSA) as part of the periodic safety review (PSR) were of interest here. GRS's review experience shall be exchanged intensively during the stay of a visiting scientist. In addition to the international standards, in particular the German principles and methods in the form of the PSA guideline and the corresponding methodological manuals are used here. Thus, it is ensured that more recent safety-related findings from Germany are explained to the CNEN and are incorporated in the supervision of Angra-2.

Argentinien: Autoridad Regulatoria Nuclear (ARN). Another South American partner is the Argentinean licensing and supervisory authority (ARN). By order of ARN, GRS reviews selected chapters of the »Preliminary Safety Analysis Report« for the Atucha II plant. This includes, for example, the PSA Level 1, the leak-before-break concept as well as the core and thermal-hydraulic design of the plant. In this context, studies were carried out both at GRS and in Argentina which also served the purpose of knowledge transfer as well as the provision of the most recent technical and scientific findings.

Dutch licensing and supervisory authority (KFD). By order of the Dutch licensing and supervisory authority (KFD), GRS reported on the operating experience with German nuclear power plants and assessed the relevance thereof for the Dutch Borssele nuclear power plant. Furthermore, studies in connection with the application of the IAEA safety requirements to the Borssele nuclear power plant were carried out incorporating the experience gained from reviewing the German rules and regulations.

Generic Safety Issues. More recent findings in the field of generic safety-relevant issues which have arisen in other countries are continuously incorporated into the work on the GRS database »Generic Safety Issues« (*Generische Sicherheitsfragen* – GeSi). The GeSi database serves for the early information of the supervisory authority on new generic safety issues to be able to react to new safety-relevant developments by regulatory research or by adapting the rules and regulations. It is the objective to use the database as central element of a knowledge and information management system at the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) and GRS.

Global Nuclear Safety and Security Network (GNSSN). Currently, a »Global Nuclear Safety and Security Network« (GNSSN) is being established in co-operation with IAEA. GRS supports the BMU with this in order to bundle the globally available information and to benefit from the resulting synergy effects.

International co-operation on nuclear safety in Central and Eastern Europe

Objectives of co-operation. One of the main concerns of the bi- and multilateral co-operations is to promote the close co-operation with local expert organisations and to provide a high technical and scientific state of knowledge for the respective safety authorities, in particular in Russia and Ukraine. That way, their professional competence in relation to industry, manufacturers and operators shall be strengthened and joint safety analyses shall be performed on a trusting basis. The technical offices of GRS/IRSN/RISKAUDIT in Moscow and Kiev effectively support the co-operation with both countries.

Co-operation with BNRA. Together with IRSN and by order of the Bulgarian licensing and supervisory authority (BNRA), GRS has performed an assessment of the preliminary Interim Safety Analysis Report (ISAR) for the nuclear power plant in construction at Belene (Bulgaria) since the end of 2008. Here, a third generation nuclear power plant with Russian VVER-1000 reactor of the V-466B-type is concerned. Intermediate results have been presented to the BNRA already. The assessment shall be concluded in 2009. Parallel to that, GRS performs in-depth safety analyses for this new reactor generation. To that end, models and datasets for the simulation programmes ATHLET and COCOSYS developed by GRS were set up and tested in co-operation with the experts of Atomenergoproject Moscow and the Experimental Design Office (OKB) »Gidropress«. The models are used for independent analyses of selected accidents and the results are handed over to the Bulgarian authority. In-depth analyses using GRS analysis methods are under way. They shall deal with the assessment of the load-carrying capacity of the double-shell containment in case of different load effects.

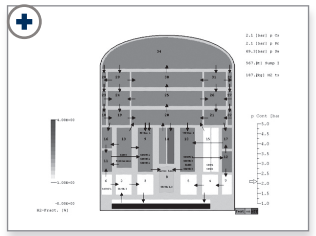
Rostekhnadzor and State Nuclear Regulatory Committee of the Ukraine (SNRCU). Since the beginning of the 90s, GRS has been working in close co-operation with the Russian and Ukrainian safety authorities Rostekhnadzor and State Nuclear Regulatory Committee of the Ukraine (SNRCU) as well as their scientific centres on improving reactor safety and the handling of the environmental impacts of the Chernobyl accident. Subject areas of the activities with the SNRCU and the State Scientific and Technical Center of Nuclear and Radiation Safety (SSTC NRS) are joint scientific and technical analyses on the issues thermal-hydraulics, PSA and fire protection analyses and a knowledge transfer within the scope of workshops and meetings. In this context, the co-operation with the Ukrainian authority at the Chernobyl site plays a major role.

With the **Scientific and Engineering Center for Nuclear and Radiation Safety (SEC NRS)** of the Russian licensing authority Rostekhnadzor, too, GRS has been working on the basis of multi-year co-operation programmes. The duration of the programmes is currently set from 2008 until 2010. Focal points are the analysis of transients and accidents in VVER-type pressurised water reactors and graphite-moderated RBMK-type pressure tube boiling water reactors. Furthermore, the co-operation focuses on the joint development of analysis simulators for different plant types, the evaluation of operating experiences and the analysis of realised modernisation programmes.

Use of simulation codes. The studies on phenomena in the containment/confinement using the COCOSYS code by GRS and the integral code ASTEC (see Fig. 72 »ATLAS GRAPHIC«) carried out jointly with the Eastern-European countries, represent an extensive field of activities.

EU-East programmes and multilateral projects on the nuclear safety in Eastern Europe

Co-operation with TSOs. GRS's bilateral activities for the improvement of nuclear safety have been supplemented by an extensive multilateral co-operation with IRSN and other Western Technical Safety Organisations (TSOs) as a part of Phare and Tacis projects. In addition, GRS assists the BMU in its activities of the Nuclear Safety Account (NSA), the Chernobyl Shelter Fund (CSF) and the International Decommissioning Support Funds of the Ignalina nuclear power plants (IIDSF). Further programmes of the European Bank for Reconstruction and Development (EBRD) are run together with Bohunice (BIDSF) and Kozloduy (KIDSF).



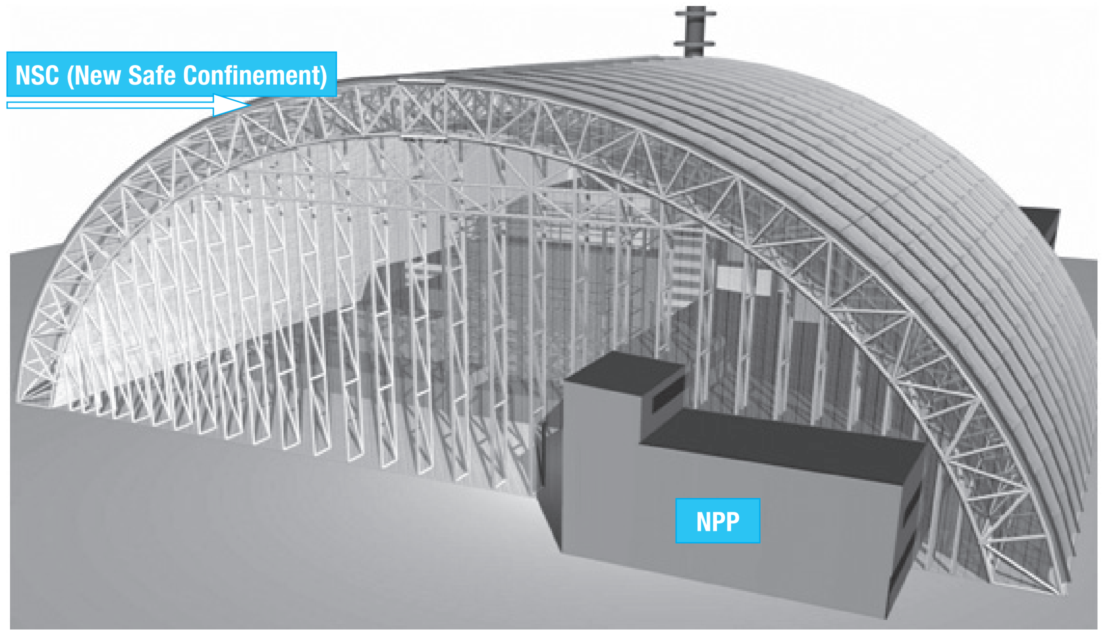
ATLAS GRAPHIC

Fig. 72
Hydrogen distribution in the containment of a VVER-1000 approx. 6 hours after the occurrence of an accident with core degradation
(result of an ASTEC calculation)

Phare and Tacis programmes. GRS is currently participating in approx. 30 projects of the EU Phare/Tacis programmes and EBRD projects. Focal point is the provision of technical and scientific findings:

- for the authorities during the decommissioning of nuclear plants,
- for the safe handling of nuclear fuel and radioactive waste,
- when extending the nuclear rules and regulations,
- during international safety assessments,
- in the licensing of upgrading measures,
- for the modern authority organisation, and
- in quality management.

Such activities are continued within the scope of the EU Instrument for the Nuclear Safety Co-operation (INSC).



MODEL CONFINEMENT

Fig. 73
Chernobyl Nuclear Power Plant, Unit 4:
The new safe confinement (NSC)

Chernobyl Unit 4: Shelter Implementation Plan (SIP)

New Safe Confinement (NSC). Together with Scientech (USA), RISKAUDIT supports the Ukrainian authority as a »licensing consultant« (LC) in the licensing process for the stabilisation of the existing sarcophagus of the destroyed Unit 4 of the Chernobyl nuclear power plant. Furthermore, the co-operation partners contribute to the construction of the New Safe Confinement (NSC). By order of the Ukrainian licensing authority SNRCU, experts of GRS, IRSN and Scientech have also assessed, together with the Ukrainian experts, the licensing documents for the implementation of the Shelter Implementation Plan (SIP).

The measures to stabilise the sarcophagus which also included repair work on the roof have been concluded in the meantime. The final report on the stabilisation measures is currently being revised. This report also contains the analysis of the achieved safety level of the entire construction of the sarcophagus.

Conceptual Design Safety Document (CDSD). Discussions on the principles of the NSC design took up a lot of space and were summarised in the safety document Conceptual Design Safety Document (CDSD). The discussions took place during meetings in a joint working group with representatives of the licensing authorities, the LC, Chernobyl nuclear power plant, and the Project Management Unit, and partly with Novarka as contractor for the NSC. During these meetings, important decisions on the details of the NSC design were taken. The licensing authorities agreed to the CDSD under certain conditions. Thus, the risks relevant for the license could be successfully reduced. The previous planning is based on the assumption that the construction work will be concluded and the plant be placed into operation in 2012 (see Fig. 73 »MODEL CONFINEMENT«).

Technical and scientific co-operation in code development

Analysis methods and computer codes for Russian-design reactors. In the technical and scientific co-operation with Russia regarding reactor safety research, which is promoted by the Federal Ministry of Economics and Technology (*Bundesministerium für Wirtschaft und Technologie* – BMWi) and ROSATOM, the focus is on the adaptation, further development and validation of Western analysis methods and calculation programmes for Russian-type reactors. Jointly, advanced methods for the safety assessment of VVER and RBMK reactors are developed further and used in an exemplary way. In addition to that, GRS increasingly co-operates with staff members of the expert organisations of these countries in international research projects.

ATHLET/BIPR-VVER code system. Within this framework, GRS is currently developing, in co-operation with the Kurchatov Institute, the coupled thermal-hydraulic and neutron physical programme system ATHLET/BIRP-VVER. For the further validation of the coupled programme system, the Russian partners provided measured data for a commissioning in the Kalinin-3 nuclear power plant. On the basis of the reprocessed data, a complete specification for another international OECD/NEA benchmark for the validation of coupled codes was developed. Furthermore, the ATHLET/BIPR-VVER model for the Kalinin-3 nuclear power plant was further developed. In doing so, the focus is on the modelling of the neutron flux detectors (Self-Powered Neutron Detectors – SPND) and the implementation of time-delay constants for such measured data where time-delay elements have to be considered. Presently, the results for the fuel element temperature sensors are being integrated into the ATLET/BIPR model.

Graphic code generator for the Russian Compressible Mixture Solver (CMS). Within the scope of the further development of the analysis simulator GeRuS for VVER-1000/W-320, a graphic code generator was developed for the Russian programme module Compressible Mixture Solver (CMS). The code generator supports the developers with the generation of thermal-hydraulic calculation models for any system. CMS is being used in the analysis simulator GeRuS for the simulation of the secondary system (see Fig. 74 »GRAPHIC USER INTERFACE«).

Adaptation of GRS codes for foreign partners. In addition, the Federal Ministry of Economics and Technology (BMWi) promotes the technical and scientific co-operation with Bulgaria, Slovakia, the Czech Republic, Ukraine and Hungary. In this context, GRS supports the foreign partners with the use and adaptation of the GRS calculation programme. So within the scope of the bilateral co-operation with ENPRO Consult (Bulgaria), for example, the analysis simulator GeRuS for VVER-1000/W-320 which had been developed together with Russian experts was adapted to the Kozloduy 5 and 6 nuclear power plant and verified and validated by means of comparative calculations (see Fig. 75 »OVERVIEW IMAGE«).

The annually performed seminar on the development, validation and use of GRS calculation programmes is the most important forum of information and experience exchange between the Central and Eastern European users and developers of the GRS simulation programmes ATHLET, ATHLET-CD, ASTEC, ATLAS, COCOSYS, and SUSA.

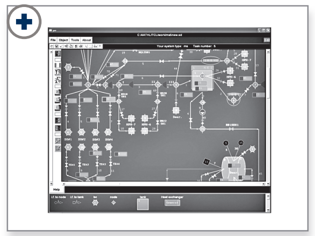
7.2 International programmes CONTENT

Provision of information and exchange of experience

Co-ordination and implementation of the BMU programme for Eastern Europe. Being an expert organisation for nuclear safety, GRS is particularly active in the co-ordination and implementation of the BMU programme for Eastern Europe. Manifold activities regarding the technical and scientific analysis of the current safety in Eastern European nuclear power plants were continued and used to provide expert assistance to the BMU and the Federal Office for Radiation Protection (*Bundesamt für Strahlenschutz* – BfS) as well as for the in-depth safety-related examination of the NPP models VVER-1000, VVER-440 and RBMK. The results are continuously being incorporated into the product line manuals and country dossiers which are used by GRS and the BMU.

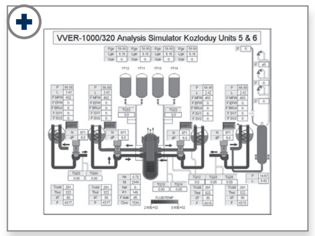
Step by step, available technical documents on the safety status and practice with special emphasis on Eastern Europe were systematically analysed, recorded and archived. Today, the database DOKU OST which was developed in 1990 comprises more than 37,000 references with diverse information on the reactor safety and safety practice in Eastern Europe. Full-text indexed electronic attachments enable a target-oriented search. In addition to that, approx. 3,000 documents with nuclear regulations, guidelines and laws of different Eastern European countries are available to selected external institutions via a special database DOCU EAST REG (Technical Documentation – Eastern European Regulations). Special tools allow the easy creation of pyramids and lists on the rules and regulations – sorted according to countries or structured according to nuclear situations.

With the databases which meanwhile also include an extensive CD and DVD collection, an effective work-sharing provision of information on nuclear safety, security, waste management and environmental protection has become possible on an international scale by way of communication and knowledge networks. These are also increasingly used for knowledge management functions and documentations on the GRS intranet portal. This also includes new procedures such as the provision of information via the Wiki function implemented in Microsoft SharePoint.



GRAPHIC USER INTERFACE

Fig. 74
CMS code generator with a fragment of the secondary system model of the GeRuS analysis simulator



OVERVIEW IMAGE

Fig. 75
Synopsis of the analysis simulator for Kozloduy 5 and 6

GRS's expert participation in international committees

Western European Nuclear Regulators Association (WENRA) und OECD/CNRA. GRS ensures the necessary technical and scientific expertise within the scope of the Western European Nuclear Regulators Association (WENRA) and OECD/CNRA activities on harmonising safety requirements. GRS's staff members are in charge of the technical preparation and follow-up of meetings and attend the meetings in accordance with the BMU as experts.

Work for the Regulatory Assistance Management Group (RAMG) and Instrument for Nuclear Safety Cooperation (INSC). The BMU also uses GRS's technical and scientific competence in the context of activities of the EU committees Regulatory Assistance Management Group (RAMG) and Instrument for Nuclear Safety Cooperation (INSC). This also includes the expert assistance during meetings and conferences, the commenting of EU programmes as well as the assessment of the resulting projects. With information platforms on a GRS information server (see Fig.76 »CO-OPERATION PORTAL«), the foundations for a contemporary provision and administration of documents – also for international participants – were laid for a series of workshops and for the work in committees.

G8 Nuclear Safety and Security Working Group (G8-NSSG). The G8 Nuclear Safety and Security Working Group (G8-NSSG) co-ordinates effective contributions to the improvement of the nuclear safety on an international scale. GRS actively participates in this working group. Furthermore, GRS provided professional expertise during the preparation and follow-up of the NSSG meetings and during the technical preparation of the topic »Nuclear Safety« for the G8 summit in Toyako (Japan). ■

CO-OPERATION PORTAL

Fig. 76
RAMG English
Joint work portal for the »Regulatory Assistance Management Group« with joint administration of the consultation documents on the info server of BMU and GRS »info.grs.de«

7.3 Development of the new »safety criteria for nuclear power plants«

By order of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit – BMU) and together with other subcontractors, GRS has been working on a project for the preparation of sub-legal rules and regulations on the safety-related assessment of German nuclear power plants since September 2003. According to the specifications made by the BMU, the new regulations shall describe the state-of-the-art in science and technology and contain the basic safety requirements of the previous BMI safety criteria and the previous RSK guidelines. In addition, the relevant international regulations and the practical experience gained from the application of the existing German nuclear regulations were to be considered during the preparation of the new rules and regulations, too. Drafts of the new regulations were discussed with stakeholders within the scope of a workshop and by using the internet. The final version of the Safety Criteria for Nuclear Power Plants (Revision D) was published in April 2009 and is available on the Internet (http://regelwerk.grs.de) and as appendix to this Annual Report.

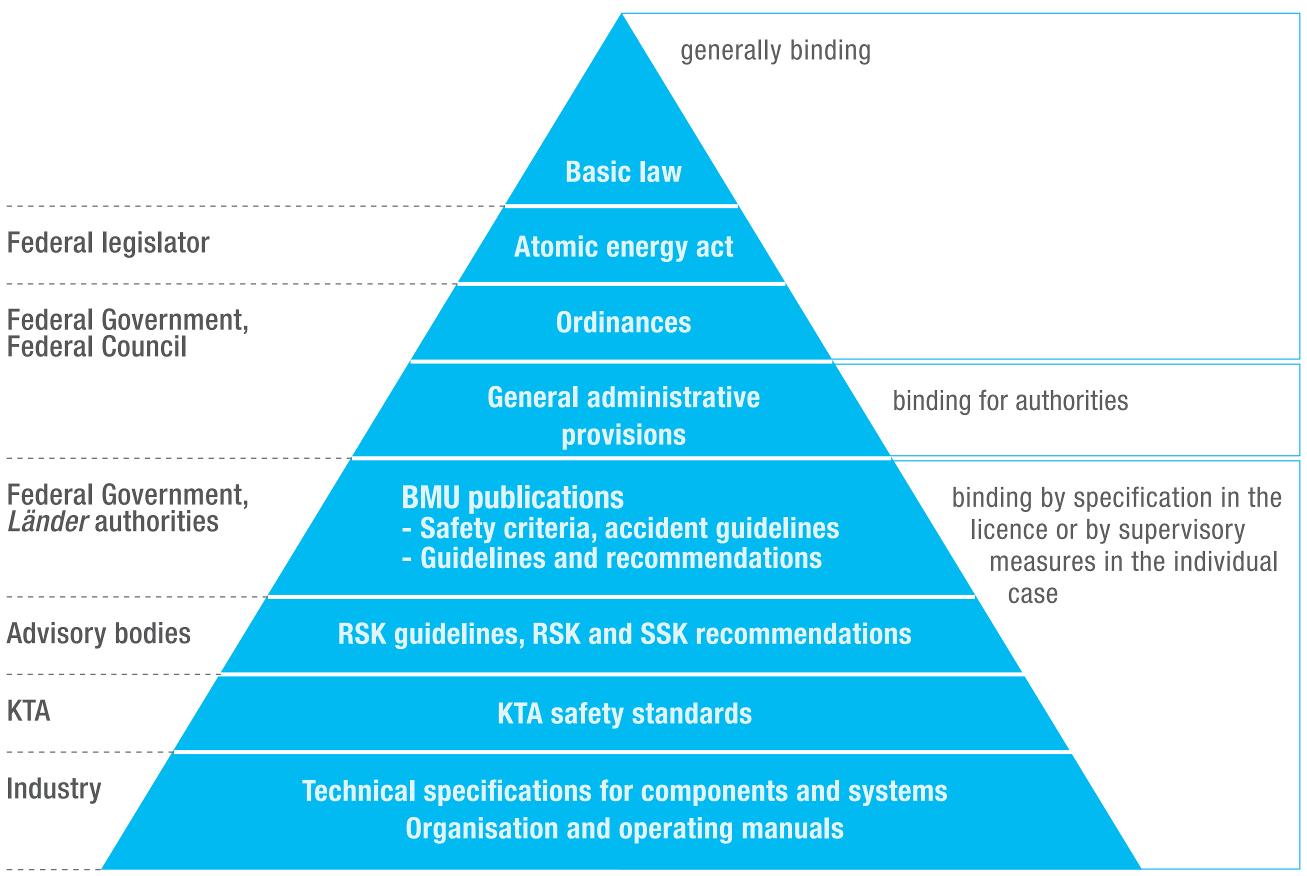
Dr. Manfred Mertins

Objectives

Basis for the preparation of the Safety Criteria for Nuclear Power Plants were the objectives the BMU had defined for the modernisation of the German nuclear regulations.

IAEA and WENRA recommendations. An essential and – in view of the field of application of the nuclear regulations – imperative requirement was the fact that the new regulations had to be in accordance with the state-of-the-art in science and technology. To this end, it had to be ensured, among other things, that the new regulations comply with the relevant recommendations of the IAEA (here in particular the Safety Requirements NS-R-1, NS-R-2, NS-R-3 and the Safety Standard GS-R-3) and the Western European Nuclear Regulator's Association (WENRA). In view of the pooling of individual regulations strived for with this revision, also the basic safety requirements of the previous BMI safety criteria and the previous RSK guidelines should be included in the new regulations.

Regulatory pyramid of the nuclear regulations



Methodological specifications. Additional fundamental requirements concerned methodological specifications. So the new regulations were to be based on a consistently deterministic safety concept and in doing so, an integrative man-technology-organisation (MTO) concept was to be implemented. The individual safety requirements were to be allocated universally to the four safety levels (»Levels of Defence«) of the defence-in-depth concept and the barrier concept.

Closing regulatory gaps. Finally, previously existing gaps were to be closed when formulating the new regulations. This applies to the consideration of all operating conditions, i.e. also of the conditions during low-power and shutdown operation. An essential expansion was also required with the inclusion of comprehensive safety requirements for boiling water reactors.

7.3 Development of the new »safety criteria for nuclear power plants« CONTENT

Development and discussion

State-of-the-art in science and technology in the field of nuclear safety. According to the specifications, the activities relating to the modernisation of the nuclear regulations were focused on determining the state of the art in science and technology in the field of nuclear safety. To this end, GRS and its subcontractors used, in addition to the existing regulations, the experience gained from the operation of nuclear power plants and evaluated the results of safety reviews and safety analyses and scientific studies.

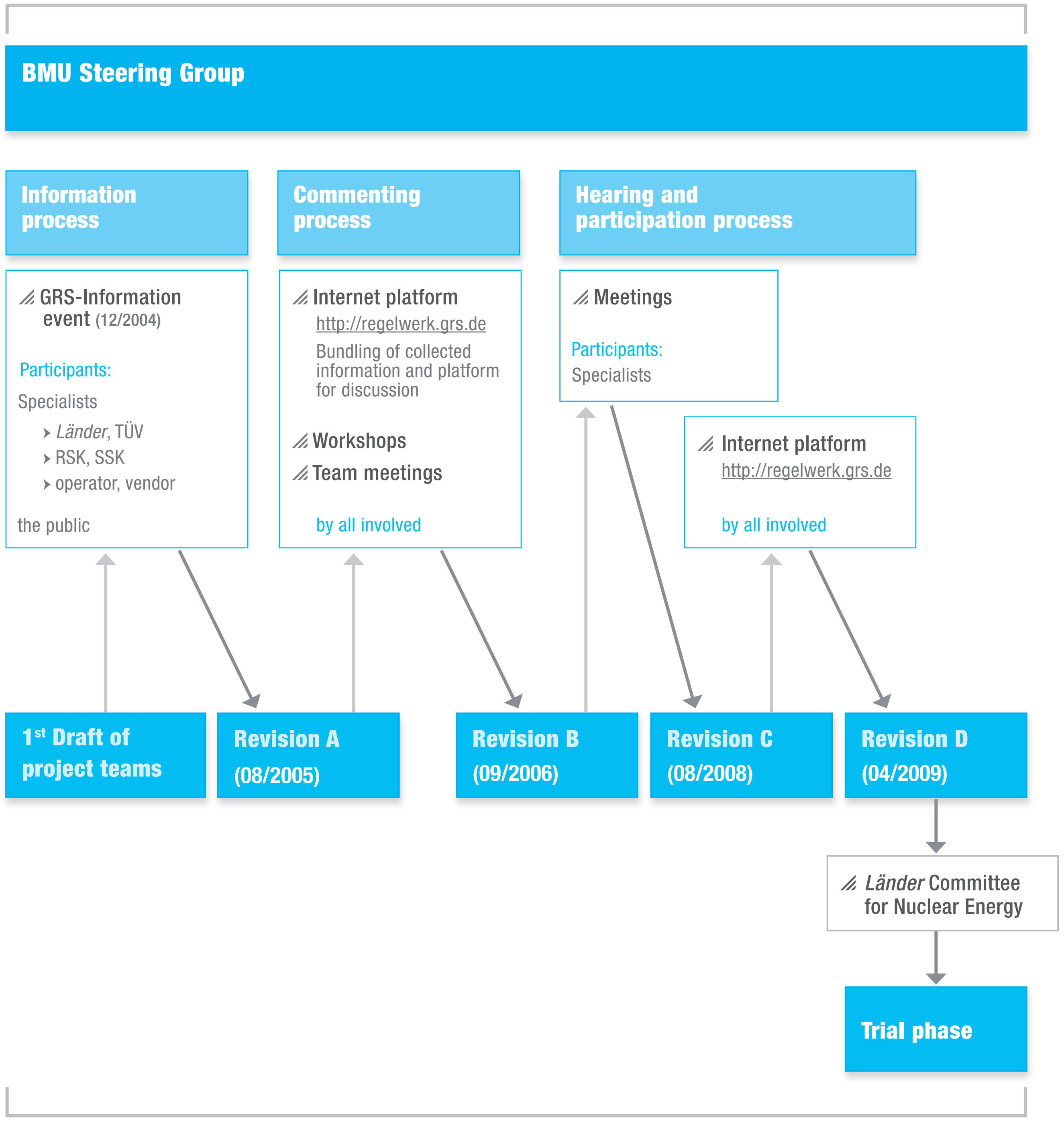
Alignment with international nuclear regulations. In addition, the findings resulting from the alignment with international nuclear regulations were included. To this end, among other things, the current IAEA requirements and the WENRA reference levels were assessed according to safety-relevant priorities, compared systematically with the current German regulations, commented and, if necessary, combined with a recommendation for the regulations to be developed. In the course of the preparation of the safety criteria, the BMU conducted comprehensive discussions and participation processes on the Safety Criteria for Nuclear Power Plants. Included in these were in particular the Reactor Safety Commission (*Reaktorsicherheitskommission* – RSK), the supervisory and licensing authorities of the Länder, the technical inspection agencies (*Technische Überwachungsvereine* – TÜV) as well as the operators and manufacturers of nuclear power plants. To facilitate the inclusion of the stakeholders, the individual revisions of the safety criteria were made available on the Internet for making comments.

Main topics. As regards content, the comments can be assigned to a series of main topics. Besides the technical realisation of the defence-in-depth concept with regard to the requirement that the individual safety levels shall be largely independent, the comments especially concerned the following areas:

- Application of the Safety Criteria for Nuclear Power Plants to older plants,
- the role of PSAs in safety assessments,
- criteria for the beyond-design-basis area,
- criteria for digital instrumentation and control, and
- safety assessment methods.

Overall, approx. 8,750 comments had been made until the Safety Criteria for Nuclear Power Plants were finalised; the comments were evaluated by GRS and its subcontractors and, where appropriate, considered in the revision.

Commenting and participation processes



7.3 Development of the new »safety criteria for nuclear power plants« CONTENT

The new safety criteria

Modular structure of the new Safety Criteria. The new Safety Criteria for Nuclear Power Plants were published in April 2009. After the implementation of the objectives described above, the safety criteria in their current version reflect the state-of-the-art in science and technology on 292 pages. For the outline of the safety criteria, a modular structure was chosen which is based on the essential focal points.

New content. Substantial changes as regards content compared to the previous regulations result from, among other things, the implementation of the required expansions, e.g. in respect of boiling water reactors, the application of software-based instrumentation and control and the demonstration methods to be applied, but also in respect of the consistent integration of the 4th safety level and the safety management requirements.

The safety criteria are available for download on the Internet under http://regelwerk.grs.de. On this website, there are also synopses available in which the inclusion of the previous regulations (especially the BMI safety criteria and the RSK guidelines) and the consideration of the recommendations by IAEA and WENRA can be followed. ∎

New safety criteria for nuclear power plants

Module 1	Fundamental safety criteria
Module 2	Criteria for the design and operation of the reactor core
Module 3	Events to be considered for pressurised and boiling water reactors
Module 4	Criteria for the design of the reactor coolant pressure boundary, the pressure retaining walls of the external systems and the containment system
Module 5	Criteria for instrumentation and control and accident instrumentation
Module 6	Criteria for safety demonstration and documentation
Module 7	Safety criteria for nuclear power plants: criteria for accident management
Module 8	Criteria for safety management
Module 9	Criteria for radiation protection
Module 10	Criteria for the design and safe operation of plant structures, systems and components
Module 11	Criteria for the handling and storage of the fuel elements
Module 12	Criteria for electric power supply

7.4
Contributions of physical plant protection to ensuring the safety of nuclear facilities

Helmut Meyer

Basically, services in the field of physical protection are ordered by Federal or *Länder* authorities, usually the Federal Minister for the Environment, Nature Conservation and Nuclear Safety (BMU) or the responsible supervisory and licensing authorities of the *Länder*.

Work for the BMU
Major activities

Revision of rules and regulations: The main focus was on the review of the Design Basis Threat as well as the further specification of new means and tools used by perpetrators, the utilisation thereof as well as resulting requirements and measures in the field of physical protection as a consequence of the events of 11 September 2001.

The revision of the German rules and regulations was at first focused on the Design Basis Threat with the aim to create them unequivocally and consistently while at the same time maintaining the contents. This editorial revision was expedited by a team of the work group »Physical Protection« led by the BMU; it was concluded in November 2008 and distributed with an additional explanatory text by order of the BMU to all participating *Länder* and police authorities. The items identified during the editorial revision for the planned further revision as regards content were compiled and listed.

First drafts and discussion papers for the so-called »graded approach« (of Design Basis Threat) were prepared by GRS and presented in the work group »Physical Protection«.

Participation in meetings of work groups/seminars of the International Atomic Energy Agency IAEA. Aspects from the German point of view were to be contributed by order of the BMU to continue with the Nuclear Security Plan (since March 2002).

The co-operation with IAEA was realised by participating in seminars and providing support

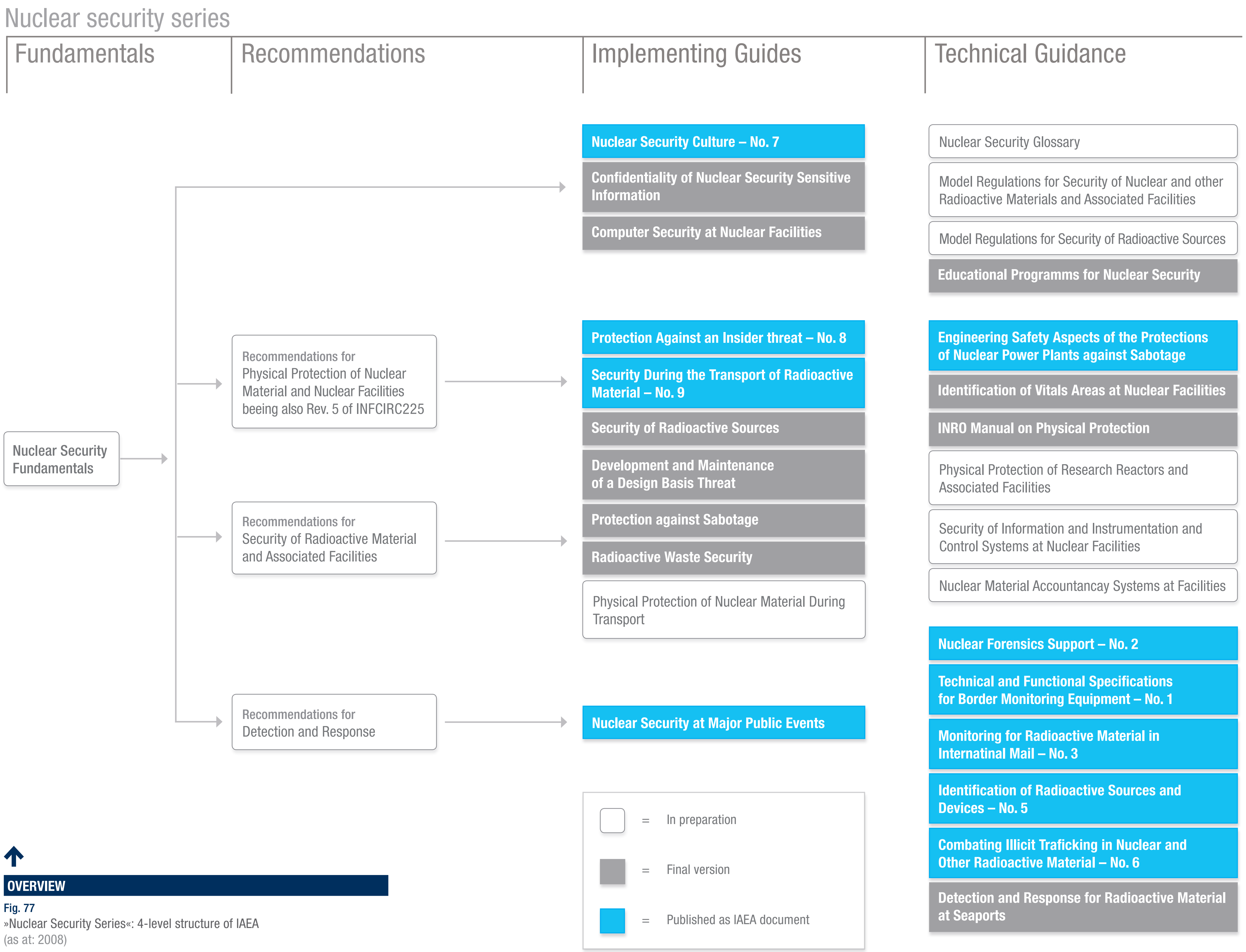
7.4 Contributions of physical plant protection to ensuring the safety of nuclear facilities CONTENT

in the implementation of seminars of IAEA abroad (Nuclear Security Culture; Design Basis Threat etc.). Furthermore, GRS took part in the meetings of IAEA work groups. The events predominantly took place in Vienna. The revision of the INFCIRC 225/Rev. 4 to Rev. 5 (Information Circular) which is being decisively influenced by GRS representatives plays a special role. The same applies to the Nuclear Security Fundamentals and the Recommendations for the Security of Radioactive Material and Associated Facilities being important parts of the 4-level-structure of the Nuclear Security Series. The IAEA strives to create a hierachic structure for its fundamentals, recommendations, implementing guides and technical guides (see Fig. 77 »OVERVIEW«).

Work for the *Länder*
Major activities

Assessment of the deterministic physical protection analysis. Activities by order of the responsible supervisory and licensing authorities of the *Länder* still focus on the assessment of the deterministic physical protection analysis carried out by the operator and the modifications resulting from them. The deterministic physical protection analyses of Lower Saxon nuclear power plants were assessed. The implementation of the modifications will be carried out over a longer period.

The deterministic physical protection analysis of a plant in Baden-Württemberg was not assessed by GRS; however, GRS is in charge of the assessment and an accompanying expert assessment of the modifications. GRS is currently reviewing the deterministic physical protection analyses of two other plants in Baden-Württemberg. The results will presumably be available at the end of 2009/beginning of 2010. The resulting modification applications, which will not be of insubstantial extent, will then be processed by the Physical Protection unit.



OVERVIEW

Fig. 77
»Nuclear Security Series«: 4-level structure of IAEA
(as at: 2008)

8.
Human resources and legal affairs

Asse II mine – Tenth Act Amending the Atomic Energy Act

Alexander Baginski

Andrea Küppers

The German Atomic Energy Act (AtG) represents the legal basis for the use of nuclear energy and the protection against its hazards. Since it first came into force in 1960, the AtG has frequently been adapted to societal, political and scientific changes. Then 10th and most recent amendment concerned the legal hold of the Asse salt mine in which nuclear waste has been emplaced for research purpurses. The Human Resources and Legal Affairs Department advises the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) in several research projects and in the field of nuclear legislation. Main focus of the project on legal issues regarding the decommissioning of nuclear facilities (»*Rechtsfragen zur Stilllegung kerntechnischer Einrichtungen*«) was the legal counselling of the BMU on the amendment to the Atomic Energy Act (»*Atomgesetz*« – AtG) within the scope of the operator change at the Asse II mine.

Need for the amendment to the Atomic Energy Act

Initial situation. Between 1967 and 1978, approx. 125,000 packages with low-active waste and approx. 1,300 packages with medium-active waste were stored in the Asse II mine (Asse) with the aim to ultimately dispose of the waste for research purposes. Originally, the Institute for Underground Disposal (»*Institut für Tieflagerung*« – IfT) of the Company for Radiation and Environmental Research (»*Gesellschaft für Strahlen- und Umweltforschung*« – GSF) was the operator of Asse. In 2008, GSF became the Helmholtz-Zentrum München German Research Centre for Environmental Health (»*Helmholtz-Zentrum München Deutsches Forschungszentrum für Gesundheit und Umwelt*« – HMGU). So far, the Federal Ministry for Education and Research (»*Bundesministerium für Bildung und Forschung*« – BMBF) – as sponsor of GSF and HMGU, respectively – has been responsible for the operation and closure of Asse.

Legal situation. When adding the provisions on the disposal of radioactive waste to the Atomic Energy Act (Fourth Act Amending the Atomic Energy Act of 30 August 1976, in effect since 5 September 1976), the legislator dispensed with a transitional provision for Asse. For that reason, §§ 9a and 9b of the Atomic Energy Act (»*Atomgesetz*« – AtG) which regulate the disposal of radioactive waste as well as plan approval procedures for repositories were not to be applied to Asse. Therefore, Asse should be closed pursuant to mining law.

With the status report submitted by the Ministry for the Environment and Climate Protection of Lower Saxony (»*Niedersächsisches Ministerium für Umwelt und Klimaschutz*«) on 2 September 1976, however, the conclusion was drawn that the procedure which had been applied so far and which was based on the mining law was no suitable basis for the safe decommissioning of Asse.

Content of the amendment to the Atomic Energy Act

Decision to amend the AtG. As a consequence of the status report, the Federal Cabinet adopted on 5 November 2008 key elements for the transition of the responsibility for the decommissioning of Asse to the Federal Office for Radiation Protection (»*Bundesamt für Strahlenschutz*« – BfS). In addition, a resolution was passed to amend the Atomic Energy Act. The thereupon opened legislative procedure was concluded with a resolution of the Bundestag on 17 March 2009. The provisions of the Tenth Act Amending the Atomic Energy Act which concern Asse have now been in effect since 25 March 2009.

With this legislative procedure, among other things, § 57b was added to the Atomic Energy Act and § 23 of the Atomic Energy Act was extended by the BfS's responsibility for Asse.

Plan approval procedure for decomissioing the Asse mine. In § 57b of the Atomic Energy Act it is specified that the provisions applying to the federal facilities pursuant to § 9a (3) of the Atomic Energy Act shall in the future also apply to the operation and decommissioning of Asse. Pursuant to § 57b (1) of the Atomic Energy Act, a plan approval procedure pursuant to § 9b of the Atomic Energy Act is required only for the decommissioning, but not for the operation of Asse in stand-by condition. From the legislator's point of view, a plan approval procedure on the operation in stand-by condition would delay the entire decommissioning procedure considerably. For safety reasons, however, this was indefensible. Therefore, § 57b (1) of the Atomic Energy Act provides for the facility to be decommissioned without delay. The granting of licenses to further store radioactive waste is unlawful until the plan approval notice for the decommissioning of Asse II is enacted (§ 57b (2) of the Atomic Energy Act).

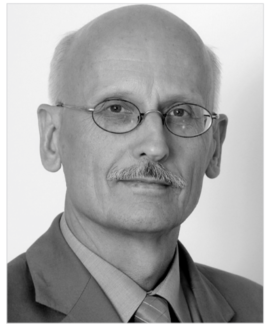
Asse II mine

Licence to handle radioactive material. Pursuant to § 57b (1) of the Atomic Energy Act, however, the handling of radioactive material including nuclear fuel requires a license pursuant to the Atomic Energy Act and the radiation protection law until the plan approval notice for the decommissioning of Asse has become effective. Apart from that, until the effective date of the plan approval notice on the decommissioning of Asse, the facility is being operated on the basis of the existing orders and licenses as far as these are not replaced or completed by licenses still to be issued. The facility is being supervised by the BfS itself within the scope of self-surveillance.

Tasks of the Human Resources and Legal Affairs Department. The Human Resources and Legal Affairs Department supported the BMU in the preparation of the legal provisions of the amendment to the Atomic Energy Act as well as within the scope of the legislative procedure. Along with the amendment to the Atomic Energy Act and the associated individual legal questions, the Human Resources and Legal Affairs Department also provided numerous expert opinions. In addition, with the legal support by the Human Resources and Legal Affairs Department, many legal inquiries regarding the amendment to the Atomic Energy Act by both the parliament and the general public were answered.

9. Project management agency/ authority support

Reinhard Zipper



Hans-Ulrich Felder

Since 1978, GRS has been the Project Management Agency for Reactor Safety Research (PT/R) of several federal ministries. Being an independent research and expert organisation, it uses ist competence in this function to put the ministries' funding measure into practice, both administratively and as regards content. For the »reactor safety research« funding concept of the Federal Ministry of Economics and Technology (BMWi), GRS has been entilled to half in trust federal funds and act as project management agency for the ministry's own projects. International co-operation at bilateral and multilateral level are a further focus of BMWi support. For the Federal Ministry of Education and Research (BMBF), GRS supports the research into nuclear safety and waste management as subcontractor for the Jülich project management agency.

Project management and project supervision

Reactor safety research of the BMWi

Trustee for federal funds. Since January 1998, GRS has been acting as an authorised project management agency, i.e. GRS has been entitled to hold in trust federal funds (reactor safety research of the Federal Ministry of Economics and Technology – BMWi). Within this scope, the GRS Project Management Agency/Authority Support Division (PT/B) takes care of all functions relating to the project sponsorship for reactor safety research of the BMWi taking the specifications made by the ministry into account. So GRS:

- participates in the update of sponsorship objectives and contents,
- takes sponsorship decisions in own responsibility, and
- continuously supervises the approved projects both technically and administratively and finally assesses these under technical and administrative aspects.

The so-called in-house projects of the ministry, especially all research projects GRS carries out by order of the BMWi, are excluded from project management. As far as these are concerned, the BMWi alone decides on their sponsoring; the PT/B Division provides technical support to BMWi as a project advisor.

Initiative on the maintenance of competence in nuclear technology (KEK). The preservation of the competence on safety-related problems in nuclear technology is of great importance in Germany. Therefore, the BMWi initiative on the maintenance of competence in nuclear technology (*Kompetenzerhalt in der Kerntechnik* – KEK) has been continued in the reporting period. With this initiative, young scientists are given the chance to further qualify by participating in projects relating to project-sponsored reactor safety research.

Taking into account the recommendations of the respectively responsible project committees and the budget available, 46 projects have been sponsored to date since the introduction of the KEK initiative in 1996. Overall, 25 scientist have earned a doctorate within the scope of this initiative so far.

Sponsorship volume of €17 million. In 2008, the PT/B Division oversaw about 100 projects with a sponsorship volume of about €17 million on behalf of the BMWi. The PT/B Division prepared these projects in technical discussions with German and foreign research institutions, surveyed the compliance with the approval conditions, took sponsorship decisions as a part of the project management, monitored and documented the orderly execution and assessed the results with respect to whether the technical objectives were achieved.

Consultation of independent project committees. The project management agency reactor safety research consults independent project committees, which are composed of leading experts for German reactor safety research, for specialist advice. The recommendations of the committees are an essential criterion for the Project Management Agency's sponsorship decisions.

Nuclear safety and waste management research of the BMBF

Support programme »*Grundlagenforschung Energie 2020+*«. As of June 2008, the PT/B Division of GRS has taken over the support of projects related to the subject area of nuclear safety and waste management research. This subject area is part of the support programme (»*Grundlagenforschung Energie* 2020+«) with which the Federal Ministry for Education and Research (*Bundesministerium für Bildung und Forschung* – BMBF) contributes to a long-term secure energy supply. The support measures basically serve the maintaining of competence and the promotion of young scientists and are divided into the fields of reactor safety, characterisation and treatment of radioactive waste, and radiology.

In 2008, the PT/B Division has dealt with ongoing projects, project proposals and joint project outlines of 16 research networks comprising approx. 60 single projects. The research projects of the first networks had already commenced when they were taken over by the PT/B Division, while the majority of the networks were still in the preparatory stages of drafting the outline and proposal of the project. Outlines and proposals were assessed by external experts. The overall sponsorship volume currently amounts to approx. €10 million per annum.

As far as the nuclear safety and waste management research is concerned, GRS acts as subcontractor of the Project Management Agency Jülich.

Support of the BMWi in the international co-operation

The international co-operation of the BMWi in the field of reactor safety research is accomplished on the basis of bilateral governmental or departmental treaties, individual agreements or because of the membership of the Federal Republic of Germany in multinational organisations.

Multinational co-operation

OECD-NEA

Committee on the Safety of Nuclear Installations (CSNI). One cornerstone of the international co-operation of the BMWi in the field of nuclear safety research is the multinational co-operation under the umbrella of the Nuclear Energy Agency of the Organisation for Economic Cooperation and Development (OECD-NEA). Especially the Committee on the Safety of Nuclear Installations (CSNI) provides a forum for scientific exchange on issues relating to the safety of nuclear facilities. The CSNI discusses the results and decides on work programmes of its working groups staffed by technical experts of the Member States. In special technical groups and during CSNI meetings, the German interests were represented, in close coordination with the BMWi, by one staff member of the PT/B Division.

OECD research projects. To investigate safety issues requiring a considerable experimental effort, the OECD provides a forum for joint research projects. The German participation in such international research projects supplements the national research activities of the BMWi in the field of project-sponsored reactor safety research and has a share in maintaining safety-related competence in Germany as well as in maintaining unique experimental facilities worldwide. By order of the BMWi, representatives of the PT/B Division participate in the technical and contractual design of these projects and they supervise the contractual execution by taking part in the respective supervisory committees, the management boards. In the reporting period, 11 OECD projects were being executed (see list).

European Union (EU)

Sustainable Nuclear Energy Technology Platform (SNE-TP). In September 2007, the »Sustainable Nuclear Energy Technology Platform« (SNE-TP) was founded within the scope of the 7th framework programme for research Euratom of the European Union. This platform is aimed at the sustainable development of nuclear energy generation in Europe by way of co-ordinated research and development activities. The head of the PT/B

Overview OECD projects 2008

Project	Laboratory	Test Facility	Research Area
CABRI-WLP (1999–2008)	France, IRSN	CABRI – Forschungs-reaktor	Fuel behaviour
PKL (2004–2006)	Germany, AREVA-NP	PKL	Thermal hydraulics
SCIP (2004–2009)	Sweden, Studsvik	Halden reactor	Fuel behaviour
ROSA-LSTF (2005–2009)	Japan, JAEA	ROSA LSTF	Thermal hydraulics
HALDEN (2006–2008)	Norway, Institutt for Energiteknikk	Halden reactor/ MTO Lab	Fuel behaviour/ man-machine interface
PRISME (2006–2010)	France, IRSN	DIVA	Fire analysis
MCCI-2 (2005–2009)	USA, Argonne National Laboratory	Melt Concrete TF	Melt-concrete interaction
SETH-2 (2007–2010)	Switzerland/France, PSI/CEA	PANDA/MISTRA	Thermal hy-draulics reactor/ containment
THAI (2007–2009)	Germany, Becker Technologies	THAI	Hydrogen-/fission product behaviour in the containment
BIP (2007–2010)	Canada, AECL	RTF	Fission product behaviour in the containment
SERENA (2007–2011)	France/Korea, CEA/KAERI	KROTOS/TROI	Steam explosion

Division is the appointed German representative in the Mirror Group which shall interlock the national research programmes with those of SNE-TP. During the SNE-TP General Assembly in November 2008, many German participants criticised the lack of a tangible German position in respect of fundamental issues. PT/B was therefore asked to initiate a corresponding co-ordination within the scope of the Alliance for Competence in Nuclear Technology. Thereupon, a small work group which shall prepare first suggestions for the positions to be jointly supported was established.

Consultative Committee for the Research and Training Programs in the Field of Nuclear Energy (CCE-Fission). The BMWi provides the German delegate of the Consultative Committee for the Research and Training Programme in the Field of Nuclear Energy (Fission), (CCE-Fission), the Advisory Committee of the European Commission on Programme Management of the Euratom research programmes on nuclear (fission) energy. On behalf of the BMWi, the PT/B Division prepares the meetings of the Advisory Committee on Programme Management for the German delegation as regards content and makes recommendations on the issues to be dealt with.

By taking part in the Advisory Committee on CCE-Fission Programme Management and in all important expert groups, GRS participates in the definition of project targets and contents, thus, for example, in the preparation of the annual work programmes of the 7th Euratom programme. Thus, and by participating in selected research projects, GRS contributes to the expansion of the international state of the art in science and technology.

National contact point (NKS). In addition, the PT/B Division is the national contact point (*Nationale Kontaktstelle* – NKS) for the »Nuclear Technology and Reactor Safety« scheme and informs and advises interested scientific institutions on current calls for tender of the European Commission.

9 Project management agency/ authority support

CONTENT

Bilateral co-operation

Administrative and technical support. The practical implementation of governmental and departmental agreements on bilateral co-operation in the field of nuclear safety research and development come under the responsibility of the BMWi. Here, the PT/B Division provides both administrative and technical support. The validities of individual contracts are monitored and upcoming prolongations/renewals are suggested and prepared. To update the subject terms of the agreements, interesting topics as well as concrete activities are coordinated with the foreign partners and German research institutions.

In the following, the scientific and technical co-operation (*Wissenschaftlich-technische Zusammenarbeit* – WTZ) will be exemplified by the co-operation with the Russian Federation and France.

Scientific and technical co-operation with the Russian Federation

Expert group of BMWi and State Atomic Energy Corporation (Rosatom). The co-operation with the Russian Federation is particularly intensive in the field of reactor safety and repository research. Every two years, co-ordination meetings in the form of conferences of the jointly co-ordinating expert group of the State Atomic Energy Corporation (Rosatom) and the BMWi alternately take place in Germany and in the Russian Federation. On the German part, the PT/B Division prepares the content and the organisation of the meetings on behalf of the BMWi.

In the meeting on 24th/25th May 2007, decisions were taken on 20 joint research projects on the safety of nuclear power plants and other nuclear facilities as well as on 14 research projects on the final disposal of radioactive waste the progress of which have been monitored and prepared by PT/B for the next meeting in the first half-year of 2009. It was agreed upon that the joint expert group will meet again in the spring of 2009 in Moscow.

Scientific and technical co-operation with France

Intensification of the co-operation with CEA. The continuity of the co-operation between the BMWi and the French Commissariat à l'Energie Atomique (CEA) was impaired by the breakup of the Institut de Radioprotection et de Sûreté Nucléaire (IRSN) which is in charge of the co-operation. In 2008 and by order of the BMWi, the PT/B Division therefore took steps to intensify this very important co-operation.

At a meeting of PT/B and CEA in Cologne, both organisations named contact persons for important areas of expertise. It is the objective to regularly conduct co-ordinator meetings between the BMWi and CEA during which decisions shall be taken on concrete joint activities.

Support of the BMWi in special tasks

Special tasks carried out by GRS within the scope of supporting the BMWi are in the first place activities relating to

- the spreading of the results of reactor safety research,
- ad hoc assistance during the response to inquiries from parliament, citizens or the press, and
- the participation in the superordinate coordination of matters concerning the Alliance for Competence in Nuclear Technology.

Participation in the Alliance for Competence in Nuclear Technology

Definition of individual research topics. One of the first tasks of the Alliance for Competence in Nuclear Technology after its foundation in March 2000 was it to specify the content of the topics relating to reactor safety research in Germany for the period between 2002 and 2006 which had been specified in 2000 by the Evaluation Commission of the BMWi. In its function as project management agency of the BMWi and being a permanent participant in meetings of the Alliance for Competence in Nuclear Technology, the PT/B Division has taken the lead in adopting this task. The results were published in the report »Issues of Nuclear Safety and Repository Research in Germany 2002-2006; Reactor Safety Research« in July 2003.

The PT/B Division has now taken the lead in updating this report for the period 2007-2011. The final version of this report of December 2007 is available both in German and in English and can be downloaded from the PT/B page of the GRS website. With this report, a guideline for the future technical co-operation of the German research institutions on reactor safety research within the scope of the Alliance for Competence in Nuclear Technology is at hand. In addition to that, it provides information on the focal points of reactor safety research at the time of its preparation and the development expected for the forecast period.

Database for progress reports

Searching by topics and research institutes. In addition to the usual spreading of progress and final reports, the existing GRS data-base (www.grs-fbw.de) has been further developed in the reporting period. Its user-friendly selection tools now render optimised inquiries relating to topics and/or certain research institutions possible. Also, the revised system-related workflow allows a faster publication of the reports.

10. Communication

Sven Dokter

Horst May

GRS is an independent and scientifically substantiated source of information for the media and the public when it comes to the topics nuclear safety and disposal. In this context, GRS's press relations, website, events and information material represent important and frequently used interfaces. To be available as a competent contact person in public discussions is part of GRS's self-image as non-profit research and expert organisation. In 2008, the focus was not only on the external communication but also on the further development of the communication within the company. The expansion of the intranet portal »GRS Intern« and many other activities contributed to the new design of the internal exchange and information processes and to the enhancements of a joint corporate culture.

External communication

GRS as information source for the public. Both in the public and in the media, there is a pronounced interest in information relating to nuclear energy. This is true in the context of the fundamental debate within society in respect of its use, but even more in case of events occurring in nuclear facilities. GRS is perceived by the media as a neutral and competent information provider and consulted on a regular basis when an independent, scientifically sound expert opinion on an issue is required.

Last year, press relations, website, information material and events were important channels when it came to providing information to third parties, coming into contact with interested parties and customers and to exchange information and experiences at an expert level.

Press relations

Press enquiries. In 2008, again, GRS was asked by journalists for information and expert explanations on incidents at home and abroad. The public interest was mainly focused on, among other things, the accident in the Krsko nuclear power plant in Slovenia, the incidents in the French Tricastin nuclear power plant, transports and transportation containers for spent fuel assemblies as well as the professional training of engineers in Germany and the next generation of scientists in the field of nuclear technology.

10 Communication CONTENT

Event in the Slovenian Krsko nuclear power plant. The incident in the Slovenian Krsko nuclear power plant attracted particular attention all over Europe as the ECURIE system (European Community Urgent Radiological Information Exchange) was activated for the first time since it had been established 21 years ago. All 25 EU Member States as well as Switzerland and Croatia are linked to the system.

On 4 June 2008, a loss of coolant occurred due to a leak in an instrument line. The event was classified as Level 0 on the International Nuclear Event Scale INES. The activation of ECURIE would have been unnecessary. Due to the Europe-wide alarm release, the incident led to numerous enquiries by national printed and online publications.

Incident at the French plant Tricastin. At the company SOCATRI, on the premises of the nuclear facilities in Tricastin in Southern France, a tank with a uranium-bearing liquid spilled in a treatment station for uranium-bearing waste water in the night from 7 to 8 July 2008. The liquid poured into an associated retention pond which, due to ongoing maintenance work, however, was not leak-proof at the time. Thus, approx. 6.25 m^3 of the solution spilled on the ground. Part of it seeped into the soil, another part got into the creeks Gaffière and Lauzon via the rain water drainage systems. The supervising authority Autorité de Sûreté Nucléaire (ASN) classified the incident as Level 1 on the International Nuclear Event Scale INES. The incident caused great public response. And GRS was asked many times by journalists to give an expert assessment.

Internet

Relaunch 2008. In 2008, the GRS homepage was completely revised with regard to its structure, design and features. Aim of the changes was it to streamline the website and to thus provide the visitors of the site with a more clearly arranged navigation. In the course of these alterations, layout and design were adapted to GRS's corporate design.

The editing and administration of the new web contents is handled on the basis of a Content Management System (CMS) which was written especially for GRS.

In 2008, nearly 350,000 people visited the GRS website. In many cases, due to the huge amount of information it contains, the website was also referred to for replying to press inquiries. Generally accessible GRS reports are increasingly published on the Internet as downloads. The availability of these reports to the public has thus been improved and, at the same time, printing and mailing costs have been reduced.

Information material

GRS publications. GRS publishes its own information material to meet the information requirements of the public. Interested members of the public, politicians, students as well as university lecturers and scientific staff belong to the typical clientele of these publications. To facilitate selection, the list of publications was updated again in 2008 and made available for download from the GRS website. In addition, a number of GRS reports comprised in the list of publications can also be downloaded from there.

EUROSAFE Forum – The ETSON Partners BEL V, GRS, IRSN, UJV and VTT

(From left to right) Jacques Repussard (IRSN), current ETSON chairman, is pleased with the new partners' signatures on the Memorandum of Understanding, as are Aleš John (UJV) and Seppo Vuori (VTT), Hans J. Steinhauer and Lothar Hahn (both GRS), and Benoît de Boeck (Bel V)

GRS events

International scientific exchange of experience. GRS takes part in a continuous process of exchanging experience with other experts and maintains a network with relevant expert organisations worldwide. This exchange of experience is implemented by, among other things, GRS staff participating in international committees and taking part in meetings, workshops and conferences hosted by other organisations.

GRS on its part organises numerous own scientific events and invites experts from Germany and abroad to take part. The most important events in 2008 were the international EUROSAFE Forum in Paris and the GRS expert forum in Cologne.

EUROSAFE Forum 2008 in Paris

10th anniversary. An anniversary could be celebrated on 3rd/4th November in the Cité Internationale Universitaire in Paris: The 10th EUROSAFE Forum for nuclear safety. Approximately 400 guests from 27 countries took part. The general topic of the forum was »The role of Technical Safety Organisations (TSOs) in the context of increasing demand for safety expertise«.

Expansion of ETSON. On the occasion of the forum, two additional TSOs joined the »European Technical Safety Organisation Network« (ETSON): VTT from Finland und UJV from the Czech Republic. Thus, within two years after its foundation by Bel V (formerly AVN, Belgium), GRS and IRSN, the network could be expanded to five TSOs. The network is still open to other European TSOs.