Competence Pool for Nuclear Technology

Topics of Nuclear Safety and Repository Research in Germany

2007 - 2011

Reactor Safety Research

November 2007 /

Revised Version dated December 14, 2007

Contents

1 Introduction

The report by the task force (Evaluation Commission) appointed by the German Ministry for Economics and Technology dated January 21, 2000 entitled “Nuclear Safety and Repository Research in Germany“ was technically detailed with the report by the Competence Pool for Nuclear Technology “Topics of Reactor Safety and Repository Research; Reactor Safety Research, 2002 - 2006“ for personnel and objective planning activities. This presented a quantitative orientation aid for chronological development and for maintaining safety expertise in the German state funded research institutes, as well as a guideline for issues to be processed on an urgent basis. After five years, some considerable progress had been achieved in the research areas identified as priority in the Evaluation Commission Report. This report was therefore – building on the expanded knowledge status – continued for a further five years period, i.e. from 2007 through to 2011.

2 Evaluation Commission

In recent decades, the funding provided by the federal government for reactor safety research has contributed significantly to the fact that German reactors are among the safest in the world. This outcome was achieved through a close collaboration between research centres and institutes, utilities, universities and German industry, as well as a close technical partnership with institutes abroad.

In view of the political goal anchored in the German Nuclear Energy Act/AkT 02/, i.e. to put an organised end to the use of nuclear energy and the significantly reduced funding for reactor safety and repository research at that time, the Federal Minister for Economics and Technology (BMWi) deemed it necessary to have this area examined by an Evaluation Commission.

The Evaluation Commission was appointed by way of a letter from the BMWi dated September 24, 1999. The stated objectives included, although were not limited to:

- Determining the priorities in nuclear safety and repository research in Germany against the background of scarce funding;
- Appraisal of medium-term manpower levels and shared labour collaborations between the institutes involved in nuclear safety and repository research, especially the Federal Institute for Geosciences and Raw Materials (BGR), Forschungszentrum Jülich GmbH (FZJ), Forschungszentrum Karlsruhe GmbH (FZK), Forschungszentrum Rossendorf[1] (FZR) and Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH;
- Consideration of medium-term financial planning;
- A focus on maintaining research efficiency and safeguarding expertise in the fields of reactor safety and repository storage.

The Evaluation Commission's tasks also included making recommendations for a more intense collaboration between the research institutes, one of the aims being to establish a so-called competence pool.

Appointed as members of the Evaluation Commission under the chairmanship of the BMWi were leading representatives of the research institutes BGR, FZJ, FZK, FZR, GRS, the staff directors of the technical project management agencies for reactor safety research (PT R) and water technology and disposal (PT WT+E), as well as representatives of the Federal Ministry for Finance (BMF) (part time), the Federal Ministry for the Environment, Nature Conservation and Reactor Safety (BMU) and the Federal Ministry for Education and Research (BMBF).

In three meetings held on October 27, 1999, November 26, 1999 and January 21, 2000 respectively, the Evaluation Commission devised and unanimously approved the following recommendations. The Evaluation Commission believed at the time that its report resulted in four key principles:

- A comprehensive overview of the institutes and activities in the field of reactor safety and repository research in Germany being funded by the BMWi and BMBF;
- A suitable basis for the competence pool to be established;
- A pinpointing of urgent activities and perspectives and also

[1] Now Forschungszentrum Dresden-Rossendorf, FZD

- References to other important activities in the fields of reactor safety and repository research in national and international collaborations.

Against this background, the Evaluation Commission made the following recommendations (quotations):

1. *Technical and shared employment collaboration in Germany in reactor safety and repository research, the aim being to further increase efficiency should be emphasized. The competence pool should make an important contribution through content-based coordination of objectives.*

2. *The priority focus is to be the urgent objectives defined in the Evaluation Report/EVK 00 (Part A, Para. 5.1)/.*

3. *The further key activities defined in the Evaluation Report/EVK 00 (Part A, Para. 5.2)/ are to be carried out as safety and temporal necessities develop and also financial resources are available.*

4. *In addition to the research activities at BGR, FZJ, FZK, FZR and GRS, considerable weight is also attached to the activities of other research institutes in Germany. Care should be taken to ensure that research activities on reactor safety and repository storage are effectively supported in the universities, not least with a view to sustaining scientific expertise (promoting young researchers).*

5. *It must be ensured that Germany continues its effective involvement in important international activities and projects for sustaining and advancing reactor safety and repository research. This applies to both the collaboration with partners in the west and also to those with central and eastern European countries.*

6. *Given the drastic funding cuts seen in recent years, which look set to continue, the possibility of a further slump in both project and institutional state funding must be reliably ruled out in order to prevent further personnel regression and hence loss of expertise in this area. Reactor safety and repository research must be allocated sufficient resources to allow the government to fulfil its legal duties.*

3 Competence Pool for Nuclear Technology

On March 16, 2000, the Competence Pool for Nuclear Technology (KVKT) convened for its inaugural meeting. Its permanent members are presently BGR, FZD, FZJ, FZK, GRS and MPA Uni Stuttgart. Regular participants are IAEA, ISAR Munich, ITU, PT R, PT WT+E and RWE Power. Regular guests are BMBF, BMU and BMWi. Since then coordinated meetings have been held bi-annually. Goals and the content of reactor safety and repository research are agreed between the institutes involved, collaboration potentials are discussed and resolutions are passed on realisation of same. A further key objective of the sessions is for the research institutes and the Federal Ministries holding technical responsibility, the BMBF, BMU and BMWi to agree on funding policy.

3.1 Objective

One of the recommendations made by the BMWi Evaluation Commission in its final report was a coordination of the subjects of German reactor safety research under the competence pool for nuclear technology.

The competence pool for nuclear technology asked the project management agency for reactor safety research to take on this task in collaboration with the leading research institutes in reactor safety research.

For the period up to 2006, the topics were to be appropriately detailed and the scientific and technical personnel quantified. This technical continuation of the Evaluation Commission Report is to serve as an orientation aid for German research institutes over the next few years. The results were presented in the form of a report entitled “Topics of Nuclear Safety and Repository Research in Germany 2002 – 2006; Reactor Safety Research“, July 2003.

At the end of the period under review at the time, the progress made was now to be appraised and, building on this progress, the key issues for reactor safety research were to be updated. This update, which was in turn progressed by PT R in close collaboration with the leading research institutes in this area, is now available in this report for the period from 2007 – 2011.

3.2 Procedure

The chapters in the report were revised from the perspective of the PT R, compiled to form a working basis and a task force was then reappointed for each of the six areas of reactor safety research stated in the report

- Determination of stress limits of materials and/or components and non-destructive material description,
- Thermo-hydraulics with transients and leakage failures, reactor physics and nuclear fuel rod behaviour, procedures for core destruction in the reactor pressure vessel,
- Core melt in the containment, steam explosion, hydrogen distribution and combustion and countermeasures, fission product behaviour in the containment,
- Methods development for probabilistic safety analyses, for control technology and diagnostics and also for assessing the human factor,
- knowledge transfer for the safety analysis of eastern reactors,
- Innovative concepts: including HTR, minimisation and avoidance of plutonium.

The task forces have been established such that the most important institutes with expertise in the special fields were represented. The goal of the task forces was to update the technical content of future activities on reactor safety research for the period from 2007 – 2011, as well as the forecasts for the deployment of scientific and technical personnel.

To that end, the special fields of reactor safety research were grouped into subject areas and these in turn into individual topics. The individual topics were then assigned to the research institutes working in the area concerned. **Table 1** shows this grouping for the example of - Determination of stress limits of materials and/or components and non-destructive material description.

The institutes involved in the individual topics are devising

- Forecasts for future deployment for the period from 2007 – 2011. A distinction had to be drawn here between crafts (for assessing experimental work) and scientists.
- Technical justifications for the forecasts concerned, based on what we know today.

The research institutes involved agreed that the research topics justified here must not be misunderstood as acquisition endeavours by the institutes. Their sole aim is to create the basis for an efficient collaboration in the future, making optimised use of the resources available.

The chapters below reflect the current status of knowledge and, on this basis, the topics and/or outstanding issues to be addressed as priority. All chapters have a unified structure:

- Statement by the Evaluation Commission on the area in question
- Subject areas, individual topics, research institutes (Table)
- Manpower forecast for the subject areas (Graphic)
- Technical description and justification for the activities in the individual topics.

Assignment of Technical Topics to Research Institutions (Example)

Chap.	Topic Areas	Specific Topic	Research Institution
4.1	**Determination of material respectively components load limits and non-destructive material description**		
4.1.1	Simulation of components loads	**Pressure boundary and main steam-feedwater system of PWR and BWR**	FZD, GRS, IWM, MPA-S
		Fluid-structure interaction in piping systems and reactor pressure vessel internals	FZD, GRS, IPM, IWM, MPA-S
		Reactor pressure vessel during to core-melt	FZD, GRS, MPA-S
		Leak tightness of containment structures of reinforced concrete	GRS, MPA-KA, MPA-S, TU-DD
		Integrity of construction structures in case of external events	GRS, MPA-KA, MPA-S, TU-DD
		Steel containment at dynamic load	GRS, MPA-S, Uni-KA, TU-DD
		Structure reliability of pressure boundaries	GRS, IWM, MPA-S, TU-DD
		Structure reliability of containment structures	GRS, MPA-S, TU-DD
4.1.2	Material behaviour	**Ageing due to cycling loads**	IWM, MPA-KA, MPA-S, TU-DD
....			
4.1.5	Non-destructive material description and material testing	**Improvement of the validity of non-destructive test methods**	BAM, IZFP, MPA-S, TU-Berlin
		Further development and verification of the methods for non-destructive testing on austenitic structures	BAM, IZFP, MPA-S, TU-Berlin
		Application of non-destructive testing for the description of material behaviour	BAM, IZFP, MPA-S, TU-Berlin

Table I: Assignment of Technical Topics to Research Institutions (Example)

3.3 Conclusions

The Evaluation Commission Report was the essential basis for the continuative research work forecast here. According to what was known at the time, the Evaluation Commission omitted to consider several subject areas due to adequate covering. Since these topics are also excluded from the updated report presented here, it is not possible to comment on the level of expertise available for these areas. A more detailed, technical structure promotes transparency and allows selective, quantitative statements to be made on future activities and efficient coordination of same, highlighting the key areas of expertise among the individual topics. Although the research subjects deemed by the Evaluation Commission as priority did not notably change, the focus within some of the subject blocks has shifted significantly. These shifts occurred as more knowledge was gained and are described in the chapters below. It must be remembered that the data is plagued by uncertainty, both under the frameworks of national funding and also under the aspect of imprecisely quantifiable funding for research activities for reactor safety in the 7th Framework Programme of the EU. This should not, however, cast doubt on the overall statement from the findings obtained. The collaboration between all participants that has intensified over the last five years with a view to achieving the common goal must be emphasised in this regard. This should be seen as a clear indication that all participants intend to continue the trusting collaboration into the future.

The existing data can be used to predict trends for future deployment in nuclear research. This forecasted deployment gives rise to the following clear trends and conclusions as compared to the figures published by the Evaluation Commission from 1998 and 2002 for the period from 2007 – 2011.

- The drastic cuts in public funding from 2002 to 2005 for research in reactor safety caused a more notable decline in deployment than that forecast for the period from 2002 to 2006. If in 2002, a moderate drop in the financed man years per year was still being assumed, this figure would still in fact be under-run by almost 25 %. Given that since 2006, funding levels have been growing under the new federal government, we can now once again expect to see higher levels of deployment in reactor safety research. The forecasts for 2011, however, are veering on the side of caution. This is possibly due to the relatively tight planning window for project-funded research.
- In the 2006 forecast, the two traditional research fields – cooling circuit and containment behaviour in the event of malfunctions within and beyond design limits - take up around half the research manpower resources. These positions could be expanded

even more in line with the true figures. This is down to work on cooling circuit behaviour, during which the transition from earlier simulation programs based on one-dimensional “Lumped Parameter” methods to multi-dimensional CFD codes demands considerable input. The use of CFD codes is also gaining in importance for applying to issues regarding the behaviour of the containment, albeit still at a significantly lower, absolute level. According to the forecast through to 2011, it is predicted that manpower requirements for research work in the field of containment behaviour will again outrank that of cooling circuit behaviour in terms of importance. The deployment forecasts, however, also show that these two research fields will decline pro-rata as a whole and together, tie up around half of existing manpower.

- The characterisation of the material in reactor components that are exposed to high levels of neutron radiation is extremely important to the safety assessment. This is done by inspecting surveillance specimens of irradiated reactor material in destructive tests. To preserve this specimen material, which has limited availability and to accurately determine the material state of components, material-saving and non-destructive test procedures are gaining importance for research in component safety, including destructive test methods. For this subject area, the forecast value for 2006, which is based on 2002, was a quota of approx. 17.5 % of the total manpower for reactor safety research. This quota was around 20 % in 2007 and if recent forecasts are correct, will grow to some 30 % in 2011, non-destructive testing showing the greatest increase.

- Forecasts and development per se in the field of innovative concepts are proving to be relatively stable. While this topic is pretty much forgotten when it comes to project-funded reactor safety research, it is principally the research centres, financed by industrial contracts and European research programmes, that are able to stabilise manpower here.

- A gradual decline is being seen in the field of methodology development for probabilistic safety analyses, man-machine interfaces and human behaviour. In the comparison between the forecast for 2006 and manpower deployment actually ascertained in 2007, this decline is still considered moderate. Given a forecasted quota in 2006 of approx. 10 %, the quota of total manpower resources ascertained for 2007 stands at about 8.5 %. However, 2011 is expecting to see a relative downturn to approx. 5.5 %, which in absolute terms is equivalent to a drop of just 1.8 man years per year.

To recap, it can be stated that this report represents a guideline for the future collaboration between German research institutes in reactor safety research under the competence pool for nuclear technology. The ongoing success of its practical implementation will depend on the willingness to continue the collaboration and the commitment of the research institutes.

The research institutes are willing. Further increases in research funding through the federal budget, especially in the field of nuclear safety, and a changing international environment are expected to demonstrate a perceptible rejuvenation of research activity and maintaining important expertise in the field of nuclear technology.

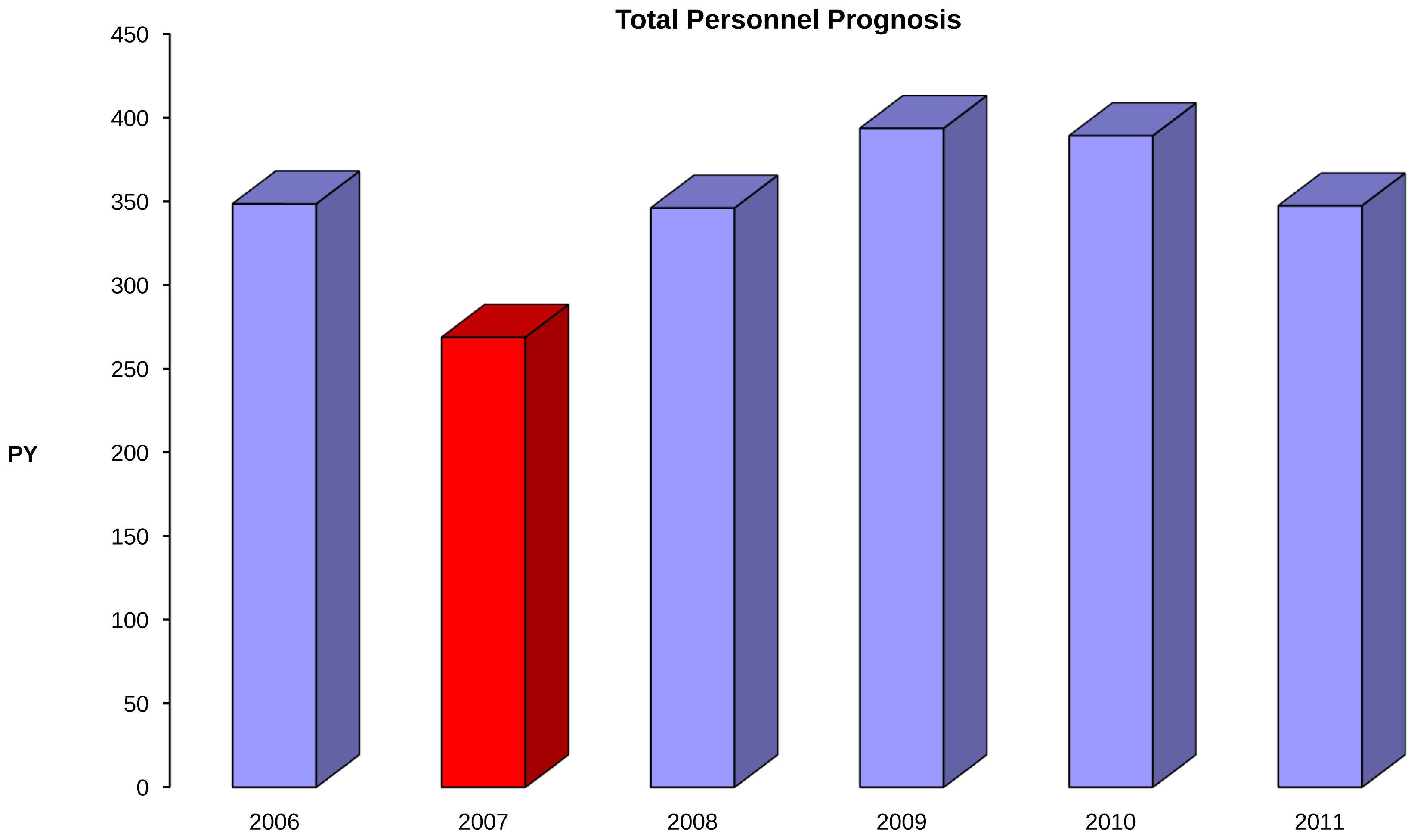

Fig. 1: Personnel Development, Scientists and Techniciens (in Person Years; blue: prognosis, red: actual value)

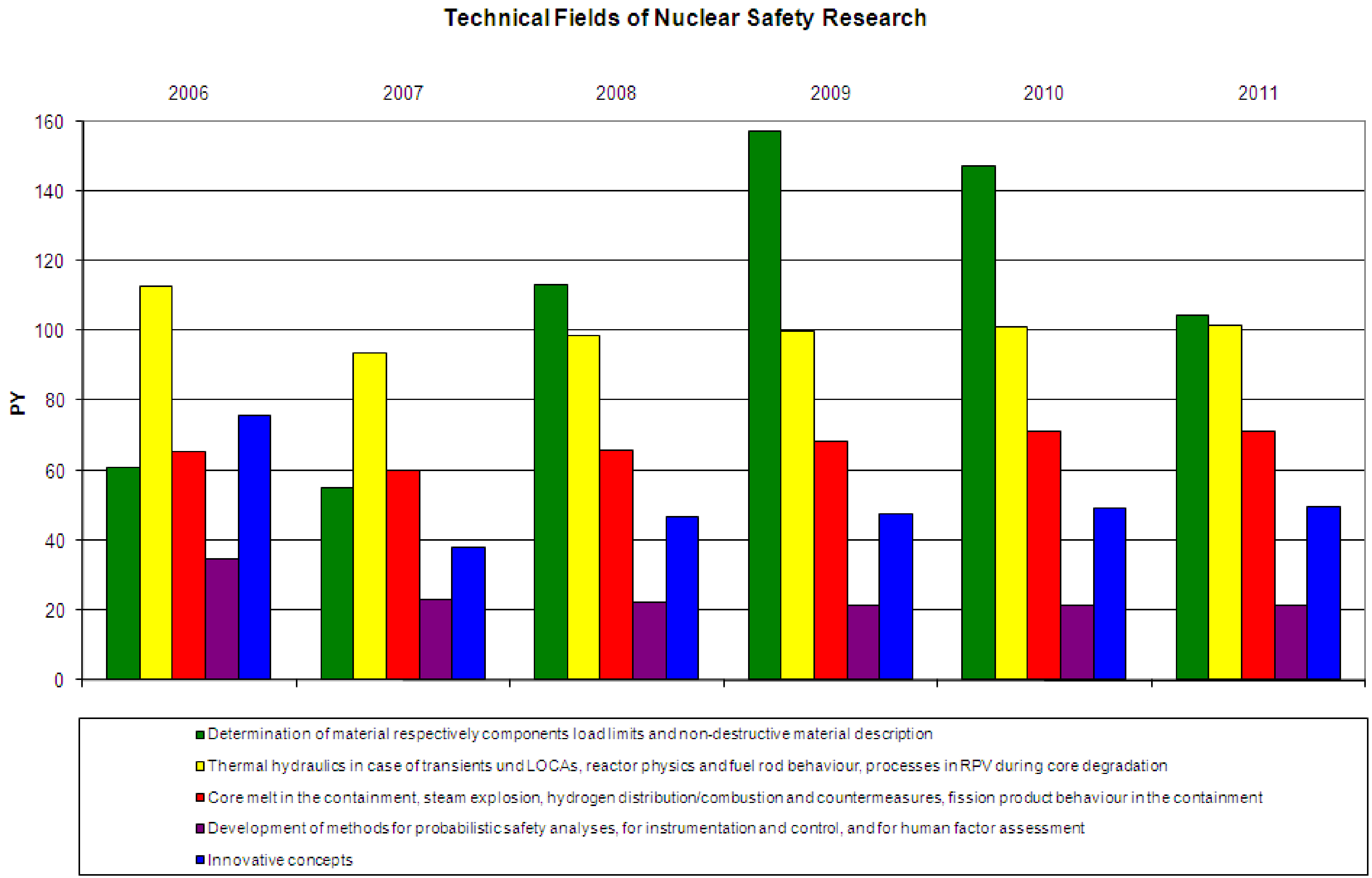

Fig. 2: Personnel Development, Scientists and Techniciens (in Person Years; 2007: actual value)

Technical Fields of Nuclear Safety Research

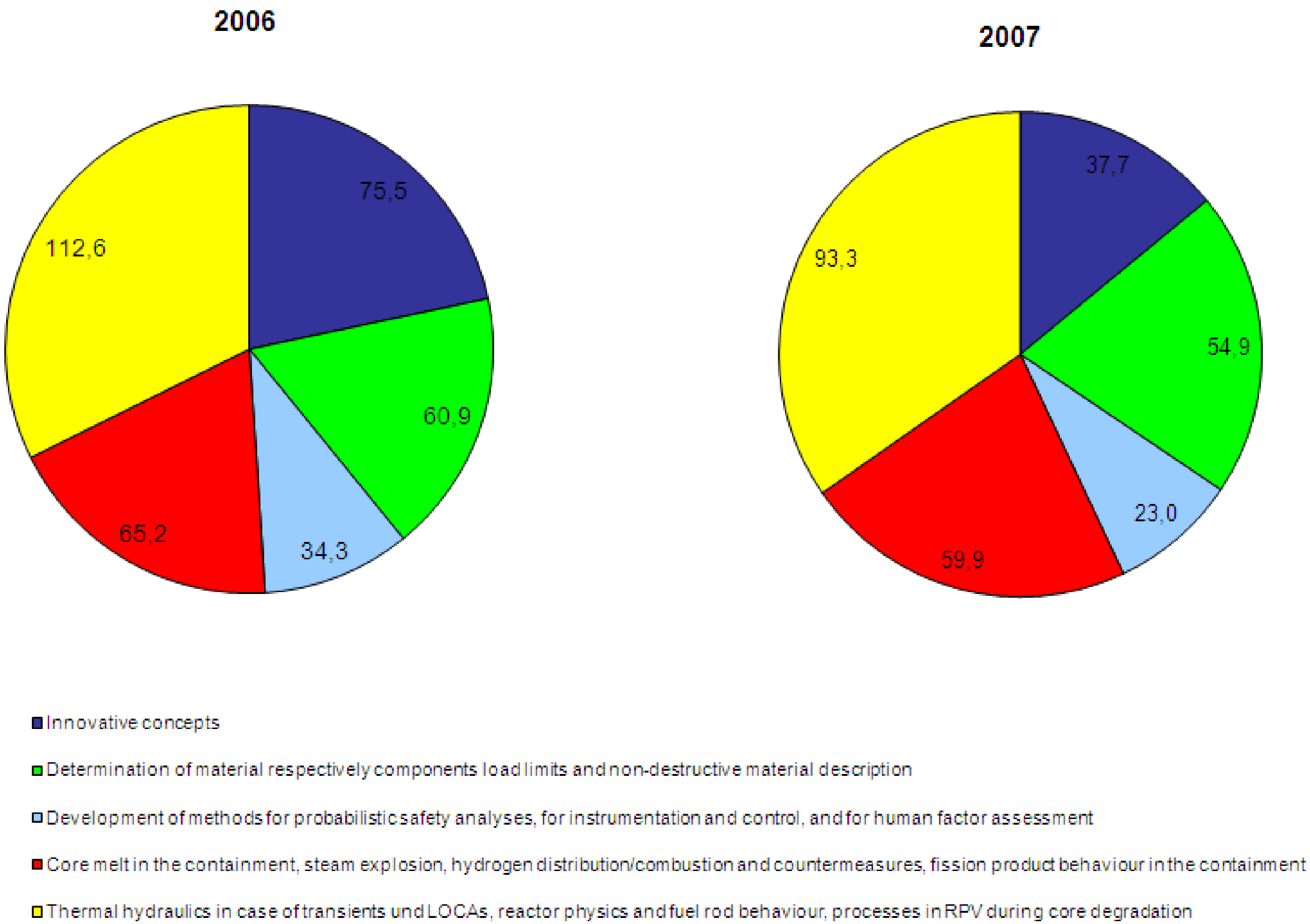

Fig. 3a: Issue-related Personnel Prognosis, Scientists and Techniciens (in Person Years; Prognosis 2006 and Actual Value 2007)

Technical Fields of Nuclear Safety Research

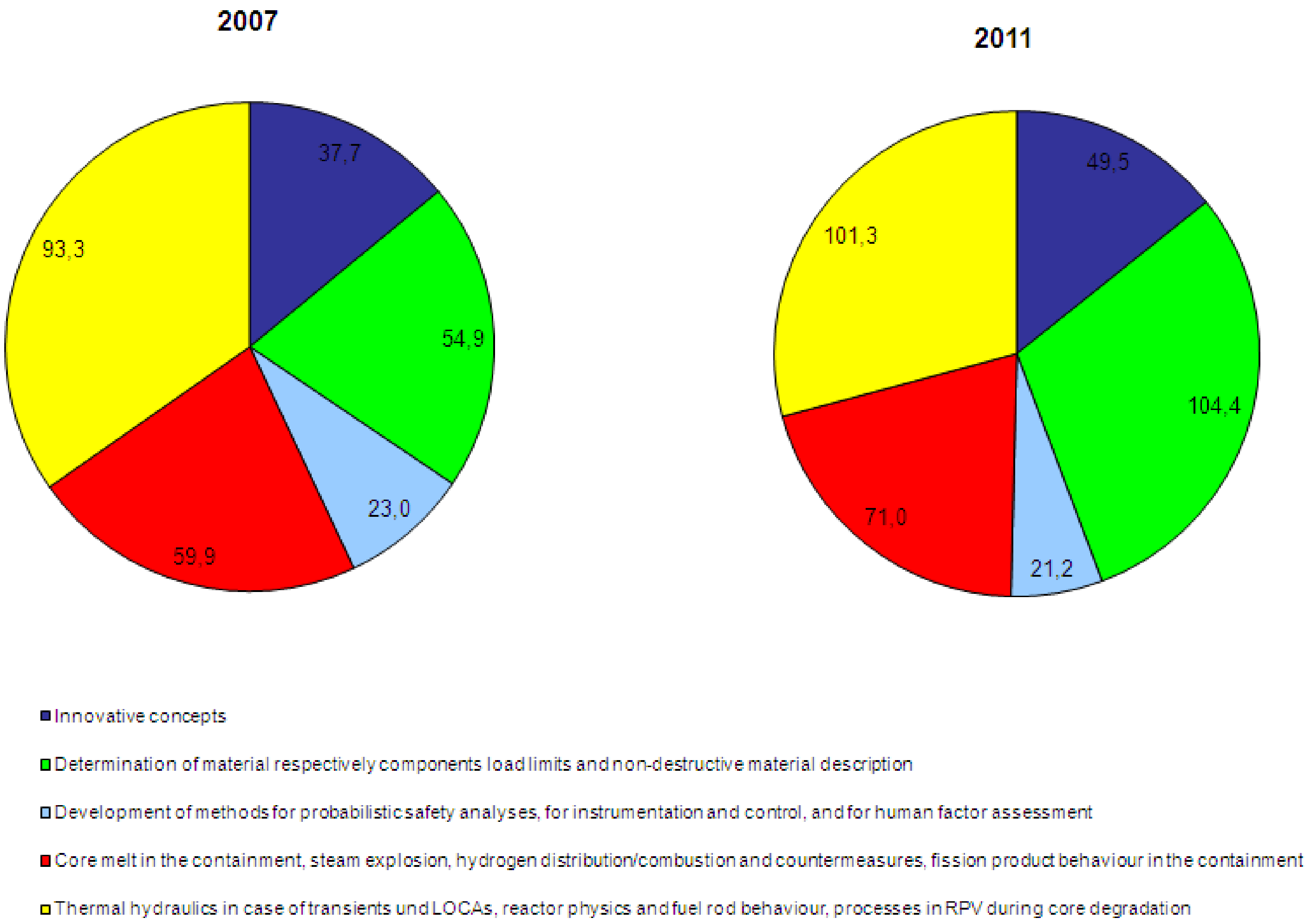

Fig. 3b: Issue-related Personnel Prognosis, Scientists and Techniciens (in Person Years; Actual Value 2007 and Prognosis 2011)

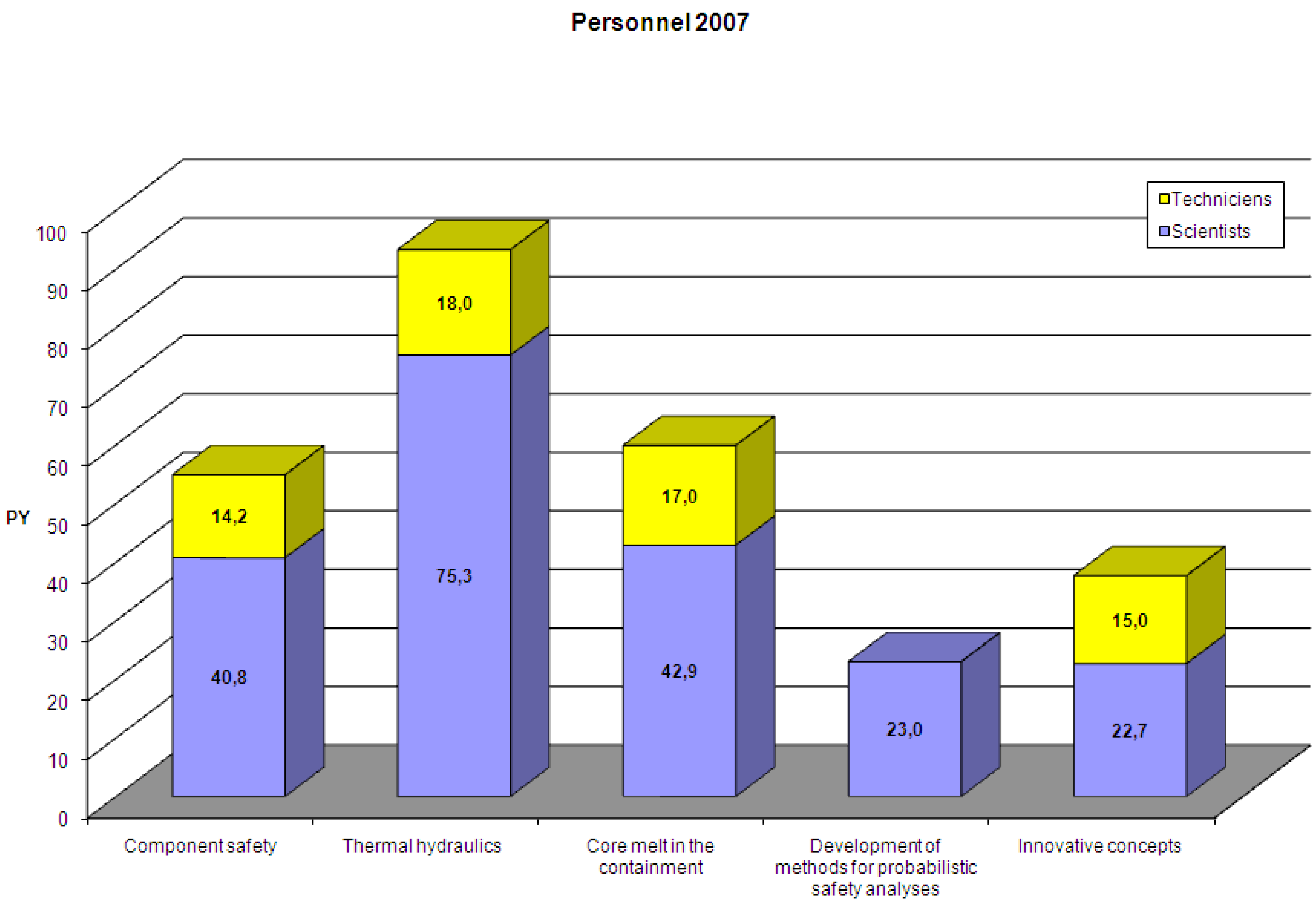

Fig. 4: Personnel 2007, split into Scientists and Techniciens (in Person Years)

4 Technical fields of reactor safety research 2007 - 2011

4.1 Determination of stress limits of materials and/or components and non-destructive material description

Statement by the Evaluation Commission:

“The problems of component and material aging and the resultant lower safety margins of components and functions become more and more important with continued plant operation”.

Table 1 : Subject areas, individual topics, research institutes

Fig 5 : Manpower forecast for the subject areas

Technical Field „ Determination of Material respectively Components Load Limits and Non-destructive Material Description “

Chap.	Topic Areas	Specific Topic	Research Institution
4.1	**Determination of material respectively components load limits and non-destructive material description**		
4.1.1	Simulation of components loads	**Pressure boundary and main steam-feedwater system of PWR and BWR**	FZD, GRS, IWM, MPA-S
		Fluid-structure interaction in piping systems and reactor pressure vessel internals	FZD, FZK, GRS, IPM, IWM, MPA-S
		Reactor pressure vessel during to core-melt	FZD, GRS, MPA-S
		Leak tightness integrity of containment structures of reinforced concrete	FZK, GRS, MPA-KA, MPA-S, TU-DD
		Integrity of construction structures in case of external events	GRS, MPA-KA, MPA-S, TU-DD
		Steel containment at dynamic load	FZK, GRS, MPA-S, Uni-KA, TU-DD
		Structure reliability of pressure boundary	FZK, GRS, IWM, MPA-S, TU-DD
		Structure reliability of containment structures	FZK, GRS, MPA-S, TU-DD
4.1.2	Material behaviour	**Ageing due to cycling loads**	IWM, MPA-KA, MPA-S, TU-DD
		Aging due to irradiation	AREVA NP, FZD, FZK, IWM, MPA-KA
		Ageing due to corrosion	AREVA NP, FZK, IWM, MPA-KA, MPA-S, TU-DD
		Material laws for high temperatures and transient loads	FZD, FZK, IWM, MPA-S, TU-DD
		Technological influences on the material behaviour (internal stresses)	AREVA NP, GRS, IWM, MPA-S, TU-DD

Chap.	Topic Areas	Specific Topic	Research Institution
4.1.3	Fracture mechanics	**Validity of the crack initiation value as material property**	AREVA NP, IWM, MPA-S
		Verification of the applicability of crack toughness curves, determined by laboratory tests, to components	AREVA NP, IWM, MPA-S
		Verification of the applicability of fracture ductility, determined by laboratory tests, to components	AREVA NP, IWM, MPA-S
		Further development of methods for the description of unstable crack propagation	AREVA NP, FZD, GRS, IWM, MPA-S
		Application of the crack arrest concept in nil ductility transition	AREVA NP, GRS, IWM, MPA-S
		Quantification of the chronological loss of integrity of bi-metallic welds with cracks	AREVA NP, GRS, IWM, MPA-S
4.1.4	Simulation of processes at the nano-, micro- und mesostructure level fort he description of material properties	**Further development of simulation tools**	FZD, FZK, MPA-S, TU-DD
		Verification of the models	FZD, FZK, MPA-S, TU-DD
4.1.5	Non-destructive material description and material testing	**Improvement of the validity of non-destructive test methods**	BAM, IZFP, MPA-S, TU-Berlin
		Further development and verification of the methods for non-destructive testing on austenitic structures	BAM, IZFP, MPA-S, TU-Berlin
		Application of non-destructive testing for the description of material behaviour	BAM, IZFP, MPA-S, TU-Berlin

Table 1: Main Research Topics of the Research Institutions in the Technical Field „Determination of material respectively components load limits and non-destructive material description“

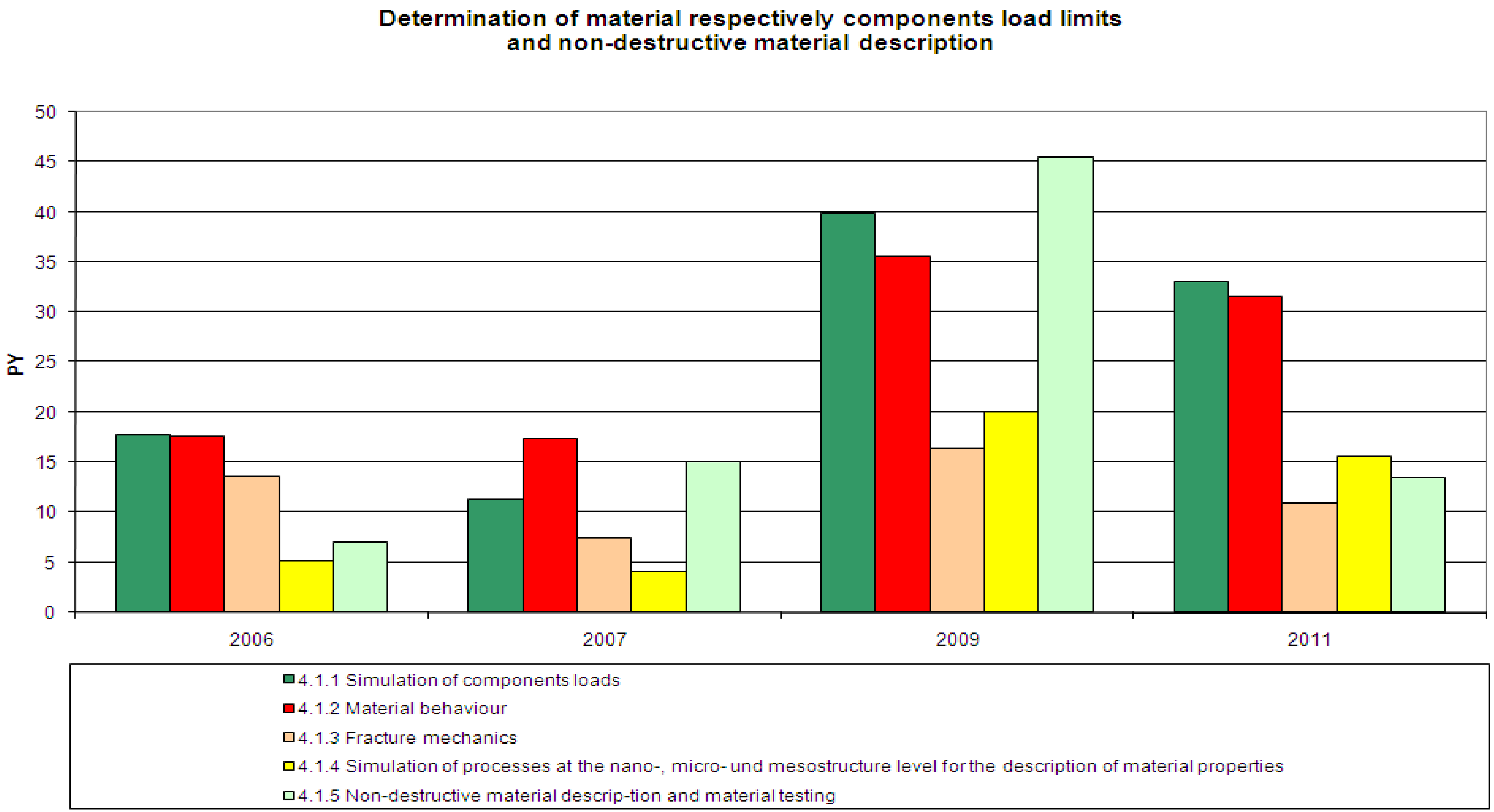

Fig. 5: Personnel Prognosis for Topic Areas in the Field of "Determination of material respectively components load limits…"

4.1.1 Simulation of component stresses

Pressure boundary and main feed water system of PWR and BWR

Recent years have seen advancement in the methodologies for simulating mechanical system behaviour (i.e. consideration of the interactions between components) and generic, quasi-static and dynamic analyses on various issues were conducted.

Component safety is usually established by determining the stress in components through an analysis on isolated structural model unhinged from the group. Only approximate account can be taken in the individual models of the stresses from adjacent components, such as forces, bending and torsional moments and deformation admitted through pipelines. A method of analysis that is able to simulate the reciprocal structural and mechanical effects of components in a PWR cooling circuit loop has been developed, tested and proven. It is therefore possible to describe with a high degree of accuracy definitive phenomena with regard to the system's safety and bearing capacity reserves under both operating and accidental loads.

In future, it will be necessary to develop or rather advance the analysis model for simulating mechanical system behaviour (i.e. allowing for the interaction of the components) for PWR and BWR. Generic, quasi-static and dynamic calculations on various issues (e.g. component behaviour in core melt scenarios, in the event of pipe leakage or breakage, under earthquake loads, impact loads caused by postulated air plane crash, water hammer) ought also be conducted. It is also important for advancements in response spectra methods to be included in this process.

A tool for evaluating user results and/or analysis methods based on reference solutions has been created.

Fluid-structure interaction in piping systems and reactor pressure vessel internals

The result of the thermal and mechanical load states in pipe systems and reactor pressure vessel internals anticipated under postulated accidents is twofold - local, high stresses on the one hand and large global deformations on the other. The mapping of entire systems, including fluid-structure interaction enables both local and global load states, e.g. by pressure waves in pipes (water hammer), pressure waves caused by pipe leakage or break to be correctly recorded.

The stress analysis of pipe systems and reactor pressure vessel internals requires calculation programs to be coupled structurally, mechanically and thermo-hydraulically. The influence of radiation exposure in the near-core area on non-linear material behaviour must be estimated and considered for reactor pressure vessel internals.

Given the complexity of the coupled calculation methodology, there remains a need for extensive research.

Reactor pressure vessels with core melt down

With an implied core melt, the reactor pressure vessel is exposed to a beyond-design load, which can jeopardise the integrity of the pressure vessel structure. In this case, detailed knowledge of material behaviour coupled with realistic stress is required to calculate the deformation and failure behaviour of the reactor pressure vessel. The material description must be improved with the aid of findings from selectively analysing the deformation process at high temperatures and stresses, as well as the temperature and time-dependent behaviour of the performance values to evaluate material degradation using heating tests in the temperature range as a whole.

Activities to date have involved analysing the behaviour of reactor construction steels at beyond-design temperatures of up to 1,200 °C. The results deliver a comprehensive description of material behaviour across the entire temperature range relevant for accidented. The developed material models and established degradation and failure behaviour of the analysed materials enable more precise accident analyses for the melt-structure interaction, especially for RPV integrity considerations.

Pre and post-test calculations on various experiments were conducted for a full validation of the analyses. The current status of knowledge for simulating vessel failure as part of mid-scale in-vessel retention experiments can be summarised as follows:

The temperature distributions in the vessel wall, as well as the position and time of failure can be adequately predicted. There are two reasons behind failure uncertainties. One is the spread of material properties (creep behaviour in particular) and the other is a strong temperature dependence of the visco-plastic deformation. Even a temperature difference of just a few Kelvin can result in widely varying creep speeds. The material degradation calculation produced a close consistency with metallographic post test evaluation analyses on specimens from the destroyed vessels.

Static uncertainty and sensitivity analyses are to be conducted in order to assess the models' reliability. This will enable the effects of inadequately established input parameters on the results (failure time, failure mode) to be quantified.

The modelling of thermo-chemical interaction between corium and the reactor pressure vessel wall and the thermo-mechanical consequences of flooding the reactor pit are top priority for further activities. The coupling between structural mechanics and thermo-fluid dynamics should be continued in order to consider the influence of wall erosion on thermal flows. Further, there is still no methodology for determining the chronological and local sequence of the reactor pressure vessel failure process, especially for determining outflow openings.

Some of the activities must be coordinated with those in the research focus of “Processes in core destruction and melt retention” in the reactor pressure vessel.

Integrity and tightness of containment structures from reinforced concrete

In order to determining the effects and/or manageability of serious accidents, the methodologies for furnishing proof of safety on the integrity and tightness of containment structures in existing plants require further development. When it comes to large-scale technical experiments, the finite element-based analysis methodology employed to determine the integrity of containment structures, as well as the leakage rate used to determine the integrity of pre-stressed reinforced concrete structures are to be further developed. Furthermore, the accuracy of the analysis methods for simulating the deformation behaviour of concrete structures must be demonstrated taking into account pre-stress and reinforcement. In the event of a serious accident, the increased pressure and elevated temperature in the containment can cause its walls to develop cracks. Should these cracks penetrate the wall; the likely result will be an outflow of radioactive substances. Hence, on the structural mechanics side, the criteria for a leakage rate calculation must be improved and included in the thermo-hydraulic model development for the leakage rate determination.

In some areas, the analysis method provided has been validated with the aid of test results. The analysis models used are not yet able to satisfactorily simulate the crack formation phase. An improvement in the analysis method calls for both the use of special joining elements, to be developed from suitable experimental analyses, for the contact between steel and concrete, as well as of stochastically distributed material values for concrete. The effect and the integrity of fixing elements under such stresses can also be incorporated into the evaluations.

Integrity of design structures under external loads

Events in recent years have given cause to re-examine the safety of nuclear power plants vis-à-vis selective external impact. This requires the provision of calculation methods that enable, for example, the integrity of design structures, the effects of vibrations caused by the impact of large masses and the integrity of design structures under detonative loads to be determined. These analysis methods, which are used not only to calculate the effects of defined impact stresses and transport processes, but also the interaction between the two processes (mechanical structure and thermo-hydraulic processes), are to be validated by means of suitable tests.

As part of the analyses previously conducted, development work began on a calculation tool for assessing the consequences of targeted attacks on nuclear power plants, as well as the effects of damage limitation measures. This analysis tool can be used to arithmetically illustrate event sequences from the targeted attack to any spread of radioactive damage in air and water. Work also began on refining the methods for determining load/time functions under the impact of deformable structures with or without fluid content and for simulating impact tests, including fluid spread and damage of affected structures.

These methods and tools must now be completed and validated by further research.

Steel containments under dynamic loads

Analysing the integrity behaviour of the steel safety vessel in the event of design stresses being exceeded also requires a consideration of postulated H_2 combustions with deflagrable burn-up or detonative implementation of the hydrogen gas. If H_2 detonation is assumed, the formation of accelerated fragments, the effect of which on the integrity of the containment is to be estimated, must be anticipated. No related detailed analyses have been conducted to date.

The considered goal of future activities is the provision and trialling by example of a mechanical structure analysis method for determining the limit capacity of a steel containment under dynamic loads following impact by flying fragments, which can be accelerated by H_2 combustion. The load assumptions are to be made in light of the core research activity “Core melts in the containment” (Chapter 4.3).

Structural reliability of pressure boundaries

The use of probabilistic methods for determining the structural reliability of components as part of risk-based maintenance is an internationally pursued trend, which is set to become an established one within the next few years. Research work has recently been conducted in this sector in Germany, the aim being to provide and trial probabilistic analysis methods for determining the leakage and break probability of pipes, in order to assess and where necessary improve the efficiency of probabilistic calculation bases from safety perspectives.

The developed prototype of the probabilistic analysis tool PROST (probabilistic structure calculation) enables leakage and break probabilities for various pipe geometries, load assumptions and crack distribution to be calculated, taking into account uncertain facts regarding distribution functions for the relevant calculation parameters (e.g. material data). The analysis method expanded by probabilistic approaches allows, in conjunction with the deterministic mechanical structure analysis chain, an assessment of the structural reliability of the pipe components.

The advancement of the program PRAISE (Piping Reliability Analysis Including Seismic Events) provides another analysis tool that can be used to conduct comparison analyses with the tool PROST and verify the results. In principle, the analysis tools currently available will allow quantitative leakage and break probabilities for certain damaging mechanisms to be calculated.

In ongoing research work, the methodology is being expanded to include vessels.

Probabilistic analysis methods are to be enhanced for optimising maintenance activities and adapting same in line with the resultant specific requirements.

Reliability of containment structures

Mechanical structure analysis methods are employed to determine the deformation and failure behaviour of reinforced concrete walls as part of the safety analysis of containments in the event of serious accidents. Knowledge gaps still exist here, some down to the complexity of the issue, some due to the fact that the relevant test results were attained from simple specimens and used for the calculation procedures.

As part of ongoing analyses, the deterministic methods for proving the safety (integrity and tightness) of containments made from pre-stressed concrete are being expanded by probabilistic approaches. This will enable the effects and/or manageability of accidents to be demonstrated with the quantitative assessments required for probabilistic safety analyses (PSA).

Sensitivity analyses are to be employed to determine, allowing for the usual material scatter, the variables that can have a dominant and/or negligible effect on the integrity and tightness of containment structures.

4.1.2 Material behaviour

Ageing (long-term behaviour)

As nuclear plants in operation increase with age, any potential material degradation must be detected at an early stage in order to prevent the number of damage events from increasing. Fatigue and stress corrosion in particular call for further activities for determining and quantifying essential variables so that remedial measures suitable for damage precautions can be defined.

Analyses on the ageing of reactor pressure vessel materials are incorporated chiefly into the activities of the EU, the OECD/NEA, the IAEA and the scientific and technical cooperation with eastern countries. The advanced, material-based degradation models verified by selected tests allow the effect of this degradation mechanism on lifespan to be more accurately predicted, especially under real load conditions.

Ageing under cyclical loads

Plastic deformations can occur in power station components under superimposed mechanical and thermo-cyclical load. If the load in the material exceeds a material-based limit, a fatigue analysis must be conducted to establish whether advancing plastic deformation (ratcheting) occurs. Real material-describing correlations (material laws/material models) are required to furnish this proof.

The conducted analyses were used to develop an analysis tool for assessing the safety of components under complex component loads (cyclical thermal secondary stresses and primary stresses), which permits, by employing a new numerical procedure, the reliable simulation of deformation and failure behaviour of components under monotonous and cyclical, inelastic load. Special emphasis is placed on characterising the ratcheting effect on component behaviour. An appropriate material model was used to simulate the material behaviour.

Given the low number of verification experiments for the analysed materials, the simple geometry of most of the analysed specimens and the high number of inadequately quantified variables, additional investigations are required to verify the employed material laws. The effect of the multi-axle stress status in particular must be safeguarded both by experiment and also with regard to the verification procedure.

High-frequency load change on the inside of components is not detectable from the outside. The resultant stresses cannot yet be described as adequately verified. Additional research is required.

Earlier results on the fatigue behaviour (low cycle fatigue) of austenitic CrNi steels in simulated oxygen-containing high temperature water demonstrate that the fatigue life of the stabilised austenite steels 1.4541 and 1.4550 employed in German nuclear power stations can be described using the prediction methods developed in the USA (Regulatory Guide 1.207) and Japan (JSME) essentially on the basis of non-stabilised austenitic CrNi steels. The design criteria to KTA 3201.2 and/or ASME II are not infringed by the results of the tests conducted under the influence of media. Together with the fatigue tests on the ferrous vessel and pipeline steel 22NiMoCr3-7 in oxygen-containing high-temperature water (PWR), the extent to which the safety margins of the fatigue curves currently are confirmed will be demonstrated. Other variables such as test frequency, plastic deformation and multi-axles must in future be investigated more closely so that the corresponding database can be expanded.

Ageing under irradiation

The change in material properties of the reactor pressure vessel as a consequence of the irradiation load in the near-core area is a safety-critical phenomenon. The reduction in fracture toughness and/or the shift in brittle ductile transition temperature of reactor pressure vessel steels through irradiation are complex phenomena, which are especially important when defining the operating time of the nuclear plants.

The research activities conducted in recent years have looked at the development and verification of a model for describing the physical material processes of irradiation-induced material brittleness, the focus being on the VVER load conditions. Methods for better assessing the safety of irradiated VVER reactor pressure vessel materials have been devised and the physical understanding of the irradiation damage mechanisms in steel consolidated. The fact that the irradiation-induced material damage caused by heat treatment can be largely healed was proven. The results of these analyses also show that once thermally healed, the material exhibits fewer irradiation-induced defects when irradiated again. The fracture toughness of VVER pressure vessel steels as a function of irradiation parameters can be predicted using the obtained findings. These findings form the basis for a realistic and material-related determination of the fatigue status of reactor pressure vessels.

In order to understand the physical mechanisms of irradiation-induced ageing, the atomic simulation tools must be further developed and verified (see Section 4.1.4).

As part of ongoing research work, an extensive database for fracture mechanic performance values for pre-irradiated specimen materials of German PWR construction lines has been created. This can be used to evaluate the irradiation-induced ageing of German reactor pressure vessel steels in a practical area of neutron fluence.

The outstanding issues in the key area of ageing/irradiation relate to how the austenitic materials making up the reactor internals and pressure vessel plating behave under the influence of gamma irradiation. Both the phenomenonological characterisation of the ageing process and its fundamental mechanisms must be clarified.

Ageing under corrosion

In recent years, the corrosion behaviour of ferrous reactor pressure vessel and pipeline materials in simulated oxygen-containing high-temperature water has been analysed and the mechanical fracture behaviour (crack growth behaviour) of such components determined under long-term load. The knowledge gathered on the causes of crack formation and the mechanisms that then unfold can be used to determine potential risk areas and define repeated suspection intervals. The risk corrosion mechanism findings allow components of the pressure boundary to be realistically assessed in consideration of corrosion formation.

An investigation program was initiated on the influence of dynamic strain ageing on the sensitivity of non-alloy and low-alloy steels against risk corrosion in oxygen-containing high-temperature water. It was established that sensitivity to dynamic strain ageing (DSA), as well as sulphur content and material strength, is to be considered a further essential material parameter for assessing risk corrosion sensitivity. High sensitivity of non-alloy and low-alloy steels to DSA promote both crack initiation and crack growth in oxygen-containing high-temperature water. Now DSA's role in crack initiation has been adequately clarified for practical purposes, future work on the influence of DSA on crack growth should be more precisely quantified. The reasons behind the continuous crack growth occurring at a significant rate in highly sensitive materials located in oxygen-containing high-temperature water under constant mechanical load, even under average stress intensity factors, should in particular be examined. These findings will then be used to assess the integrity of relevant components.

The deformation and permeation behaviour of ferrous steels in the presence of both hydrogen produced by corrosion and also oxygen-containing high-temperature water was able to be determined using the example of reactor pressure vessel material (20MnMoNi5-5). The established dependency of the hydrogen volume on the oxygen content of the high-temperature water allows conclusions to be drawn on the significance of operation-relevant oxygen content in the high-temperature water of nuclear power plants with regard to hydrogen formation and permeation. The findings obtained can be used to detect the effect of the medium on the material‘s mechanical performance values.

The tenacity behaviour of reactor pressure vessel steels (A533Bcl.1, A 508 cl.3 and 15Kh2MFA) under the effect of hydrogen and neutron irradiation was determined on non-irradiated and irradiated specimens, as well as specimens pre-loaded with hydrogen and specimens loaded with hydrogen in-situ at room temperature and at 250 °C. A brittleness effect under hydrogen was observed only in the case of in-situ hydrogen charging and at room temperature (RT). A stronger brittleness tendency at RT is expected only in specimens irradiated at low temperatures, specimens with vacancy defects and those with a high Cu and P content. At 250 °C, the specimens did not exhibit hydrogen-induced brittleness. Accordingly, sensitivity to hydrogen brittleness is determined from the chemical composition of the reactor pressure vessel steel, fluence, irradiation temperature and the nature of the irradiation defects caused. No further work is therefore required in this area.

Material laws for high temperatures and transient stresses

To protect component behaviour against beyond-design loads and/or to determine the consequences of same, material behaviour at high temperatures must also be considered in the context of time-dependent deformation (creep).

Research is required on the reciprocal effect between creep and fatigue in connection with the experimental determination of material performance values and translation of same into material laws and/or integrated material models. Such a determination can then be used to demonstrate the deformation behaviour and failure of components under realistic conditions. Special load situations, which overlap with those of malfunction-free operation, must in particular be considered and analysed. Should a beyond-limit load situation, characterised by high local temperatures and rapid load transients, occur, the existing material models must be modified in line with the associated deformation and damage mechanisms. Experience gained in non-nuclear technology shows that time-dependent deformation and failure processes are not to be treated analytically and must always be associated with the material condition. The analysis of high-temperature behaviour must therefore also include the effect of the operation-related change in the material condition. However, there are no findings in this area to date.

Technological influences on material behaviour; internal stresses

The manufacture and finishing of materials bring about technological influences on the behaviour of such materials. These influences affect both the specific mechanical technological fracture properties, as well as the mechanical physical properties. This also applies in particular to creep fatigue behaviour, which is governed by a special melt-specific scatter. Ongoing research work involves determining the technological influences on material behaviour and material performance values. The findings permit material damage, such as crack formation in risk zones, to be assessed at an early stage. The purpose is to maintain the existing high level of expertise.

The latest introduced and valid body of rules and regulations to be takes account of the technological influences caused by the manufacturing process through appropriate, commonly applied empirical safety margins. If these introduced, tested and established procedures are superseded by more modern methods, the technical influences must also be re-evaluated.

Residual stresses can occur as a consequence of merging two (or several) materials (with different properties). In power plants, these materials are generally joined by welding, a basic distinction being drawn here between build-up welding (e.g. plating) and girth welding (seams). The degree of residual stress (residual stresses after the welding process per se) depends on a host of parameters: thermal conditions before and during welding, seam preparation, welding process and parameters, post-treatment (thermal, mechanical). Since a knowledge of residual stresses plays a key role in assessing component lifespan, it is important to establish such knowledge by applying experimental and numerical methods. It has previously been possible to both bolster experimental methods for determining surface residual stress (e.g. x-ray methods) and also improve methods for determining vertical distribution (ring core procedure, bore hole procedure, neutron defraction). Particular progress has been made with numerical simulation methods, since the highly effective and ever-increasing computer capacities and two-dimensional models also enable three-dimensional calculations under the necessary and complex material laws. The stresses and deformations in the overall structure can thus be illustrated and very fast and effective parameter analyses can be performed by employing, for example, measures for reducing post-welding residual voltages using thermal or mechanical secondary treatment. The numerical analysis of welded joints in ferrous steels and combined welded joints under significantly more complex framework conditions and variables induced by phase conversion and other physical phenomena is to date incomplete. Further research work is required in this area.

As well as this general knowledge status on the formation and distribution of residual stresses during and after welding, another important issue is repair measures. Selective experimental and numerical analyses have recently been employed to start tackling this issue. Repairs to welded joints are important both as nuclear plants get older, and also during the construction of new plants. The first studies, e.g. repair welds on very small penetrations through to repairing a reactor pressure vessel defect, have already begun with a view to being able to understand the influence of these activities on safety assessments. It is especially important in this regard that evaluations allow for complex loads and the effect of media.

This will require, over the next few years, the development of progressive, multi-dimensional numerical weld simulations and the associated experimental test equipment. The primary goal is to support and optimise the qualification of repair measures, to improve integrity assessments allowing for operating conditions and, most importantly, to be able to quantify the influence of repair measures on crack formation and growth, so that safe, reliable plant operation can be guaranteed.

4.1.3 Fracture mechanics

Validity of the crack initiation value as a material parameter

As outcome of intense research work in recent years it is verified today that the multi-dimensionality of the stress state must be included in the fracture mechanical assessment of failure behaviour of specimens and components, especially where loads exceed crack initiation.

The influence of the initial crack length (short cracks) in fracture mechanic specimens to the fracture toughness value level at crack initiation has been studied in the area of upper-shelf toughness. The results of these studies showed no evidence of the short crack effect influencing the fracture mechanic value at beginning of ductile crack growth of a technically relevant amount.

In the brittle fracture and brittle-ductile transition area, however, an influence on fracture toughness and the reference nil ductility transition temperature thus maintained was seen for short cracks in specimens (crack conditions $a/W < 0.2$). This so-called 'loss of constraint' effect can play a key role in component assessment. It has been possible to quantify the positive effect on non-irradiated and irradiated basic materials and weld metal of German original reactor pressure vessel materials.

Introducing the J-integral has made it possible in principle to derive and experimentally determine physical crack initiation values for the overall ductile toughness range. These values can be determined on small specimens and hence on irradiation-capable specimens too. They also provide the basis for mechanical analyses under PTS conditions, whereby the possibility of ductile crack initiation cannot be excluded. The transferability of the fracture toughness values determined on small specimens to real components has yet to be validated. Furthermore, supplementary studies for the validity limits of the J-integral are required with regard to the absolute specimen and crack size, as well as material plasticity.

Verification of the transferability of crack resistance curves determined on lab specimens to components

Experimental results on large specimens have demonstrated that thick-walled components such as those used in nuclear power plants can fail following limited plastic deformation due to the multi-axiality of the stress state. The quantitative influence of the state of the multi-axiality, especially on the fracture mode, has not yet been adequately verified. This requires numerical analyses of component-like specimens to determine the development of the state of multi-axiality under increasing load and identify the start of failure.

Verification of the transferability of fracture toughness determined on lab specimens to components

With the aid of experiments on specimens with surface defects and multi-axial loaded stressed specimens the influence of multi-axiality on fracture toughness caused by specimens and crack geometry and also by the load has been qualified for materials of German DWR construction lines. Appropriate micro-mechanical models for describing the failure of these specimens have been developed and verified. These models can be used to describe multi-axiality. The aim of ongoing activities is to use these models to qualitatively desoibe the crack behaviour in reactor pressure vessels and hence apply the transferability of values determined from lab specimens to components such as reactor pressure vessels under realistic load conditions as e.g. a thermal shock transient.

Advancements in the methodologies used to describe unstable crack growth

New fracture mechanic assessment methods, which use unstable fracture instead of crack initiation, the ‘master curve’ concept being one example, have been developed in recent years and some have even been incorporated into bodies of rules and regulations.

While the previous brittle facture validation as defined in the KTA regulation is based on a purely deterministic concept, the basis of the recently introduced master curve concept is probabilistic. The two concepts show common and different aspects. The main difference is that use of the master curve concept is also approved for toughness transition area. The aim of completed and ongoing studies is to compare the two concepts and examine how the new might be applied to German plants. This concept enables plant-specific fracture toughness values to be determined on a small amount of irradiated material, e.g. from irradiation programs, and the accuracy of the fracture mechanics assessment to be improved.

The verification of the master curve concept on non-irradiated and irradiated original reactor pressure vessel materials has demonstrated that this concept and the application instruction to use the reference temperature T_0 as RT_{T0} have been confirmed.

Previous activities have been able to confirm the master curve concept for the reactor pressure vessel material under consideration (22NiMoCr3-7) and highlight the individual stages in the failure sequence in the toughness transition area. The transferability of the mechanical damage parameters to other specimen geometries has been verified in the area of low toughness.

The aim of ongoing activities is to create a database by determining fracture mechanical values using pre-irradiated specimens from materials of the German DWR construction line in order to quantify technological manufacturing influences. The obtained results are to be analysed and then illustrated using the master curve concept, one of the aims being to show the effects of the master curve T_0 concept on the safety assessment of German reactor pressure vessels with specimens in non-irradiated and irradiation state, another being to compare the two concepts, and finally to clarify the issue of suitable specimen shapes and sizes for the T_0 determination.

In order to apply the master curve concept to prove the safety of German nuclear power plants, the theoretical approaches have been worked up and the foundations for using the master curve concept created. The mechanical state around the crack front that causes the cleavage fracture has been analysed by simulation of experiments. The key basic assumptions of the concept, such as the functional form of the fracture toughness temperature curve, the crack front length correction and the statistical fundamentals of the concept have been confirmed.

Further application-related studies are required due to the “saturation” of the crack front length correction. This issue is raised by the weakest link theory, according to which “very long” cracks inevitably result in failure, even under “very small” loads. For practical applications, however, a limit for the maximum crack front length to be applied is being discussed so that no unrealistically small loads are enforced for “long” cracks. Appropriate studies in this area are required prior to a general application in safety-relevant issues.

Last but not least, there is also a lack of evidence that the concept is suitable for describing the toughness behaviour of irradiated reactor pressure vessel steels in German plants under dynamic load and for linking the concept with legislative procedure for irradiation embrittlement prognosis. Ultimately, the concept must be prepared for integration into irradiation monitoring programs.

A manifold of international activities (e.g. as part of the IAEA and EU DG RTD) are either underway or in preparation and are to be competently supported from a German perspective too.

Application of the crack arrest concept in the toughness transition area

Proof of an adequate safety margin against brittle fracture of reactor pressure vessels plays an important role in the operational safety of a nuclear power plant. The procedure for verifying the safety of reactor pressure vessels to be applied in Germany is set out in KTA 3201.2. It is based on the RT_{NDT}-concept developed in the USA in the 1970’s, for which a lower (deterministic) K_{Ic}-T-curve for crack initiation and/or a K_{IR}-T-curve for crack arrest has been created from measurements. This curve will be adjusted for the material under consideration using the reference temperature RT_{NDT} determined specifically for this material. German legislation also permits the use of measured K_{Ic}-fracture toughness values to create a K_{Ic},T-curve, though fails to specify the manner in which such a curve is to be deduced.

The determination of fracture mechanic values using pre-irradiated specimens on original materials of the German DWR construction lines has enabled the creation of an extensive database, for which a comparative assessment was made with the aid of the master curve concept and the RT_{NDT}-concept. Values for both crack initiation and crack arrest were determined. The results have been communicated to the specialists both nationally and internationally in appropriate publications. Further, the obtained experimental results were assessed and concepts devised, the aim of which is to develop an application instruction for using T_0 in safety-related assessments for crack initiation and arrest. Hence, it was demonstrated that all obtained results were able to confirm the application instruction for the reference temperature in the ASME Code Cases, namely replacing RT_{NDT} with RT_{T0}. It was also possible to demonstrate the linear correlation between the shift in reference temperature and irradiation (ΔT_{41} measured on charpy specimens and ΔT_0 measured in the fracture toughness test) that is already known from measurements in American databases. Previous crack arrest tests also demonstrate the existence of a linear correlation between crack arrest energy from the Charpy test and the crack arrest toughness K_{Ia} determined from fracture mechanic specimens. These assessments are an important initial step towards being able to employ and incorporate existing measurement results from irradiation programs into the new concepts.

Given previous research activities, there is from a German perspective a justified concern to remain actively involved in the international technical and scientific discussion. The ideal approach is to use rely on several fracture mechanic specimens irradiated more than two decades ago. These both cover the German plants of several generations of manufacturers and permit an upper-bound coverage with regard to fluences from plants of older generations. This creates the prospect of a better than now, safeguarded conceptual validation of the two procedures on identical, representative material states. Proof would also be furnished that the findings on the master curve and crack arrest concept obtained at both national and international level are also valid for irradiated German reactor pressure vessel steels and that the concept for proving the safety of reactor pressure vessels in German pressure water reactors in the covering fluence area and under relevant irradiation conditions can be applied. This calls for an adequately wide and representative database. A significant contribution is being made, which will enable progress in proving the safety of German nuclear power plants and maintain link to international procedures.

Quantification of the failure sequence of crack-affected dissimilar metal welds

The failure sequences per se, taking into account the structure state and actual material values, need be recorded and defined in order to provide an accurate description of material behaviour in dissimilar weld seams. Previous research activities have looked at the cause of defects and residual stresses in dissimilar weld joints. The global load deformation behaviour of dissimilar metal welds in pipelines can be recording using the developed and results-qualified analysis methods. Other research work has focussed on developing a proven and verified calculation and analysis procedure for assessing the failure behaviour of dissimilar metal welds with cracks. The fact that manufacturing-related residual stresses can significantly increase fracture mechanical loads as a result of operational loads was demonstrated in the process. Previous analyses have failed to explicitly consider the area of the heat-affected zone with various structures (coarse grain and fine grain). Future analyses will aim to systematically observe this area and theoretically demonstrate structure development in the welding process with regard to fracture mechanic loads. This ought also to consolidate the link into existing micro-mechanical calculation procedures.

A generally applicable and reliable assessment concept comprising all load scenarios and material states is crucial to assessing component safety. This is the task of future research activities.

4.1.4 Simulating processes in the nano, micro and meso structure range for describing material properties

The development of defects during manufacture or repair, as well as material properties can be described by recording processes in the nano and micro-structure range. Good progress has been made in assessing the irradiation influence on the reactor pressure vessel material. The expansion of theories, for example to describe diffusion processes when creating a dissimilar weld metal, both improve the physical understanding of the technological processes and allow the component safety and reliability to be assessed.

Further development of simulation tools

Material simulation for describing ageing phenomena holds key significance. As operating time increases, micro-structural changes can occur due to operational loads. The extent of these changes, as well as their influence on material properties, needs to be researched in advance with regard to changes occurring at a later date. The aim of future work is to integrate nano simulation so that the properties of complex material systems can be better understood. For the medium term, such activities will create the basis for an objective, and qualitative evaluation of the potential variation in safety margins as a consequence of operational loads.

The atomistic simulation for characterising the precipitation state and the chemical composition of the precipitations from previous activities were analysed. The development of various calculation methods has enabled molecular-dynamic motion and the adhesion of the dislocation to precipitations to be simulated. The obtained findings currently allow only nano and meso-scale damage and failure mechanisms of precipitation-hardened steels to be assessed, as, for example, they can occur in the form of 15NiCuMoNb5 (WB 36) under LWR (light water reactor) operating conditions. The results can be used for example to derive information on the ageing of steels under long-term thermal operating load.

These analyses are to be continued and expanded to include the relevant materials.

Model verification

It is vital that these model developments are linked with experimental results. The mechanism of the irradiation-induced defect formation in reactor materials is still not understood, despite the expansive activities of recent years. Neither existing experimental results nor the simulation tools currently available (e.g. molecular dynamics, Monte Carlo, reaction rate theory) permit the role, either of certain alloy elements (Mn, Ni) or polluting elements (Cu, P), in defect formation to be clearly described. The precise chemical composition of the observed defects is not clearly understood either. This concerns the contents of Cu, Ni, Mn, Fe, P and vacancies, as well as their spatial configuration. Simulation tools are not yet able to describe the atomistic processes in irradiated steels; their capability extends only to simplified systems (e.g. Fe-Cu, Fe-Ni). The aim of future activities, then, lies in advancing simulation tools to enable more complex systems to be modelled. Experimental analyses (e.g. SANS (Small Angle Neutron Scatter), hardness, tensile tests) on fitted model alloys (Fe-Cu-Ni-Mn-P in various combinations) with differing fluences are crucial to this process.

4.1.5 Non-destructive material description and/or testing

Non-destructive testing (NDT) plays a key role in determining a component’s quality during manufacture and operation. Various damage events reveal unanswered questions, relating not only to the search routine itself, but also and especially in determining the size of material imperfections.

Progress with qualifying inspection systems has also been made. A pilot study on the ENIQ procedure, for example, has been successfully concluded. It has yet to be implemented in practise. We also need to catch up with other European states that already have qualification and validation centres and test block pools in place. This also relates to issues regarding ‘Probability Of Detection’ (POD), according to which due consideration must also be given to the human factor influence, as well as the technical conditions. Further research is also required in this area.

Improvement in the reliability of non-destructive test methods

The closer a real material state under service loads approaches its design lifetime, the more important is a non-destructive inspection which is capable of delivering reliable quantitative information.

A recent research project has looked at the development and description of the technical resources required for a quantitative, load-dependent non-destructive test on pressurised components. The results from completed activities can be used to devise a suitable overall concept for detecting cracks in pipeline components. Integrated into such a concept are the methods for both non-destructive material testing and fracture mechanics. They can be used for nuclear power plant components in both manufacture and operation.

Based on deterministic treatment, the methodology is currently becoming probabilistic. The stochastic scatter in material values and load assumptions occurring in practise are taken into consideration, as well as the lack of definition in the information delivered by NDT. Important in this regard is the probability for verifying a certain error type, magnitude and orientation (POD), which also depends on the used NDT methods, the test system and the potential error effect of the operator. The method can be used for nuclear power plant components in both manufacture and operation. As far as ultrasonic testing is concerned, the synthetic aperture focusing technique (SAFT) has proved to be largest theoretically justifiable reconstruction algorithm for determining relevant material defect geometries under practical conditions. Use of the developed inspection system as a transportable unit for defect analysis has been successfully trialled on an application-specific basis in other industrial sectors, the steel industry for example for reducing false alarm rates in automated testing and in fossil and water power stations at ISI.

Advanced development and verification of the methods for the non-destructive testing of austenitic microstructures

Non-destructive testing is still inadequately reliable for testing austenitic cladding, austenitic weld seams and dissimilar metal welds. A specific aim of non-destructive testing is therefore to clarify these deficits.

The advanced methods of ultrasonic test methods for thick-walled austenitic weld seams have been qualified and verified on a real weld seam with a realistic material defect. The developed computer simulation methods enable sound propagation through austenitic welded structures to be calculated. Cracks right through the weld seam can be reconstructed with an optimised transducer arrangement. The improved test methods can be used for the non-destructive testing of welded claddings in pressure vessels at surface-breaking defects, defects in the cladding and sub-clad cracks.

Thanks to the advanced development of ultrasonic testing and imaging techniques for dissimilar metal welds, it has been possible to transfer the results from ultrasonic tests on austenitic weld seams to ultrasonic imaging. Hence, due consideration can now be taken not only of the cladding and/or dissimilar metal weld seam structures, but also their current external and internal geometries. The developed defect reconstruction codes have been validated by measurements. Advanced developments in evaluation methods for ultrasonic testing, also enable anisotropies and inhomogenities in a dissimilar metal weld to be considered, hence improving the measuring result. In addition, the recently developed InA-SAFT (inhomogenous ansiotrope) is able to accurately reproduce the defect pattern and the defect location. This has consequences for the fracture toughness defect evaluation in terms of whether, for example, a surface-breaking defect with certain geometries is assumed to be into the heat-affected zone of a weld seam or into the basic material. The aim of a recently registered project is to benefit from optimised, flexible multiple angle insonification using the phased array method. On the one hand this enables a much more accurate prediction of the true defect orientation and on the other, the specially pursued variant of the spatially sampled phased array transducer, by integrating SAFT algorithms, to suppress disturbing patterns and hence improve evidence quality. This variant allows in particular improved inspection sensitivity in the near-field of the test head. The project mentioned will continue to develop an algorithm, which, through an intelligent teach-in function, will initially determine individual dendrite structures, even in austenitic and dissimilar weld seams. This information will then be used for defect reconstruction.

Significant, practically-relevant advances with regard to the non-destructive testing of austenitic weld seams are being achieved by the progressive manufacturing technology used when the replacement of pipeline systems was performed. The internal and external grinding (flush with the sheet metal surface), as well as the lower seam volume, has significantly improved defect detectability. Nonetheless, the dendrite weld quality of austenitic and dissimilar metal welds continues to have a negative effect on ultrasonic ability for detection. A further improvement of ultrasonic verifiability using manufacturing procedures could bring about a modified weld quality (fine graining). This is the task of future research activities.

Application of non-destructive tests for describing material behaviour

Non-destructive testing has provably generated the first successes in predicting mechanical-technological characteristics by measuring magnitude of special quantities which are sensitively influenced by microstructure parameters and their modifications – for instance by ageing effects - and also by load-induced and residual stresses. The activities aim to pursue the microstructure changes caused by neutron irradiation and fatigue also with a view to characterising material behaviour in terms of thermal ageing (problem with Cu-precipitating steels). As far as the material influence of reactor pressure vessel steels (basic material and weld quality) from neutrons is concerned, a feasibility study examined the potential of new approaches based on micro-magnetic methods and magnetostrictive ultrasonic excitation, and also reliably proved the microstructure changes caused by significantly lower fluences over lifespan in Germany compared to same abroad. The two approaches proved to be supportable and will be optimised as part of a follow-up project.

A new test concept, referred to as fractal dimension, has been developed and verified on the reactor material 15 NiCuMoNb 5 (WB 36) for the non-destructive characterisation of material damage caused by fatigue. The developed methods can be used for the early detection of fatigue-related component damage. The achieved results suggest that analysing the fractional dimension is in principle capable of detecting fatigue-related material damage at the pre-crack stage.

It was possible to demonstrate, by optimising electromagnetic test methods and by minimising interference susceptibility, that non-destructive testing can be used to prove the material changes caused by copper precipitation in high-temperature steel 15 NiCuMoNb 5 (WB 36). These findings form the basis for a realistic and material-related determination of the fatigue status of pressurised components. The early detection and quantification of strengthening and brittleness using the suggested NDT approach enables ageing processes in plants to be evidenced and quantified at an early stage.

As part of a feasibility study, the NDT method was optimised and trialled by comparison on irradiated specimens (reactor pressure vessel, weld quality) in order to prove neutron brittleness in pressure vessel steels. It was demonstrated that the electromagnetic test variables enable the microstructure changes caused by neutron irradiation to be characterised and the reference variables to be quantitatively determined. There is a further need for research in the electromagnetic characterisation of fatigue behaviour compared to brittleness behaviour when thermal and mechanical material loads are superimposed.

There is also a need for additional research in verifying the transferability of the results to multiple-axial loads.

4.2 Thermal hydraulics during transients and loss-of-coolant accidents, reactor physics and control rod behavior, processes during core degradation in the reactor pressure vessel

Statement of the Evaluation Commission:

“The realistic description of processes in the reactor core and in the cooling circuits during incidents and accidents is of essential importance to the safety assessment and the further improvement of precautionary measures. New demands results from the progressing development of systems engineering and the operating procedures, as well as from the increasing scope of simulations.”

Table 2 : Topic areas, specific topics, research institutions

Fig. 6 : Personnel prognosis for topic areas

Fig. 7 : Personnel prognosis for the technical field

Technical Field „Thermo hydraulics in case of Transients und LOCAs, Reactor Physics and Fuel Rod Behaviour, Processes in RPV during Core Degradation“

Chap.	Topic Areas	Specific Topic	Research Institution
4.2	**Thermo hydraulics in case of transients und LOCAs, reactor physics and fuel rod behaviour, processes in RPV during core degradation**		
4.2.1	Further Development and validation of system- and CFD-codes	**One- and multi-dimensional description of single-phase flows**	FZD, FZK, GRS
		One- and multi-dimensional description of two-phase flows	FZD, FZK, GRS, IPM
		One- and multi-dimensional description of single- and two-phase flows with particle load	FZD, FZK, IPM
		Dynamic flow regime and condensation processes	FZD, FZK, GRS, IPM
		Heat transfer at high temperatures and under the influence of inert gas	FZK, GRS
		Effectiveness of accident management measures	FZK, GRS
4.2.2	Reactor physics for LWR	**Development of methods for the generation of nuclear cross-sections**	FZD, FZK, GRS
		Development of time-dependent neutron transport calculation codes for transient analyses of reactor accidents	FZD, FZK, GRS
		Physics of the reactor core in case of increased burn-up and use of MOX fuel and advanced nuclear fuel	FZD, FZK, GRS
		Use of advanced nuclear fuel	FZD, FZK, GRS
		Development of methods for on-line core monitoring	IPM
		Further development of Monte Carlo Methods (i. a. in reactor dosimetry)	FZD, FZK

Chap.	Topic Areas	Specific Topic	Research Institution
4.2.3	Special aspects of reactor physics	**Further development and verification of the coupling of neutron kinetics and thermal-hydraulic computer codes**	FZD, FZK, GRS
		Reactivity-initiated accidents	FZD, FZK, GRS
		Study on recriticality in case of core degradation	FZK
4.2.4	Fuel rod (-assembly) behaviour	**Further development of analytical tools for the assessment of fuel rod behaviour**	FZK
		Operationally realistic transients	FZK
		LOCA	FZK, GRS
		CABRI experiments on LWR fuel rods under accident conditions	FZK, GRS
4.2.5	Processes during core degradation and retention of melt	**Core degradation in the early phase**	FZK, GRS, RUB
		Core degradation in the late phase	FZK, GRS, IKE
		Coolability of the degraded core	FZK, GRS, IKE
		Relocation to the lower plenum and interaction with residual water: Retention of melt in RPV	FZK, GRS, IKE
		Coolability of debris in the lower plenum, formation of a melt pool	FZK, IKE
		Melt pool in the lower plenum of the RPV, external cooling and loss of integrity of the RPV	FZK, GRS, IKE
		Interaction of melt with the coolant (steam explosion)	FZK, IKE
4.2.6	Development of methods for the assessment of the reliability of computer results also with regard to the applicability to real plants		GRS, IPM

Table 2: Main Research Topics of the Research Institutions in the Technical Field „ Thermo hydraulics in case of Transients und LOCAs, Reactor Physics and Fuel Rod Behaviour, Processes in RPV during Core Degradation “

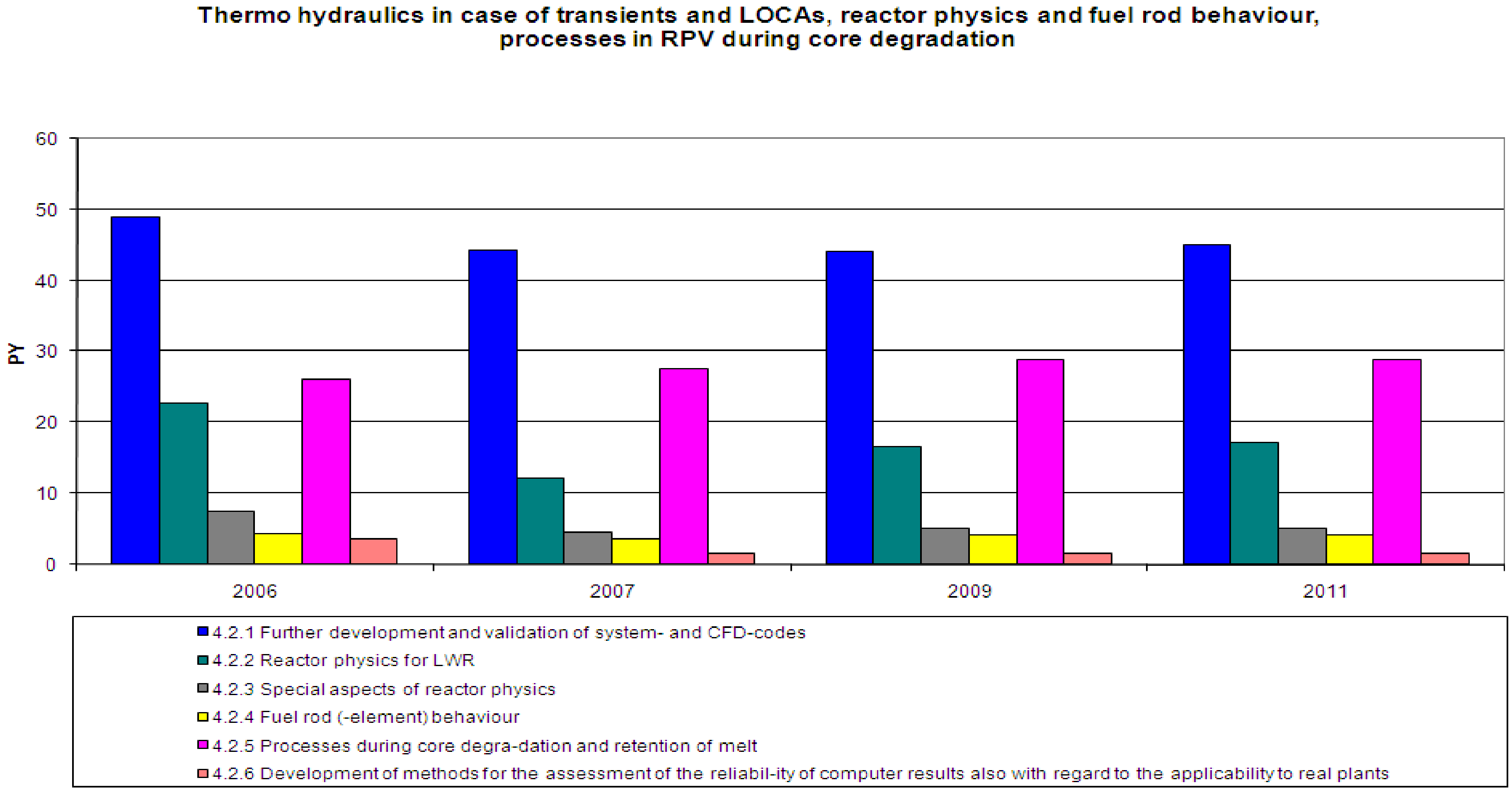

Fig. 6: Personnel Prognosis for Topic Areas in the Field of "Thermo hydraulics in case of transient and LOCAs, reactor physics …"

4.2.1 Advance development and validation of system and CFD codes

Efficient safety evaluation hinges on highly reliable computer programs. A precise analytical simulation is required, especially for quantifying safety margins in operating nuclear power plants, e.g. with efficiency increases and new core loading strategies. The uncertainties still in evidence today are primarily down to the inadequacy of the single-dimensional computer programs for reproducing multi-dimensional flow processes. To simulate three-dimensional complex mixing processes, for example, there is an increasing need to use high-resolution CFD codes in addition to and/or in combination with system codes.

ATHLET, the thermo-hydraulic system code developed in Germany in the foreseeable future forms an important basis for evaluating the safety of reactors currently in operation. It is also being used with increased frequency in combination with codes for core melts and fission product behaviour (ATHLET-CD), neutron kinetics (QUABOX/ CUBBOX, DYN3D), with improved multi-dimensional models for calculating multi-phase flow processes (FLUBOX 2D/3D) and containment behaviour (COCOSYS). It is also the most important process model in the analysis simulator ATLAS. Key activities of WTZ with eastern partners are also based on this code.

ATHLET should continue to be advanced. The consistent implementation of new findings, especially from the validation and reflow from the user experience, as well as an ongoing quality assurance procedure must also be guaranteed for the future.

Single and multi-dimensional description of single-phase flows

The focus of simulating single-phase flows is mixing processes, which occur in PWR in connection with cold water feed and dilution transients. This can be validated by existing analyses on various test plants. Other data sources are measurements in nuclear power plants both in Germany and abroad. These data are to be used for validation activities on system and CFD codes.

In the case of transients with extremely large power offsets, mixing processes in the reactor core itself are relevant. A detailed reproduction of the reactor core in CFD programs is not a feasible short term aim. Hence, experimental and theoretical examinations are currently being conducted to establish the extent to which a flow treatment of the core approaching the porous body is practical and possible.

Single and multi-dimensional description of two-phase flows

In safety analyses of transients and accident, the simulation of two-phase flows has to satisfy high requirements. There is a considerable development need, especially for multi-dimensional two-phase flow processes. The development of two-phase models for CFD codes is designed for the long-term. However, in order - especially with ATHLET- to enable multi-dimensional simulations that cover the entire two-phase range for processes in the reactor pressure vessel sooner rather than later, the FLUBOX module associated with ATHLET has been advanced and validated on the basis of the split-coefficient matrix method. FLUBOX must be advanced and updated until a complete CFD solution has been created. However, CFD codes have to date reached an advanced development status only for single-phase multi-dimensional flows. One of the main aims, therefore, is to develop two-phase multi-dimensional models for CFD codes. This task can be solved only as part of optimally matched experimental and analytical projects. The activities created over several years to be carried out jointly by several research institutes has begun.

Dynamic flow regimes and condensation processes

The dynamic transition between the flow regimes during an accident has been considered in recent years using improved models in computer programs. However, there is a need to continue experimental work and theoretical developments for describing gas content distribution over the pipe cross-section, for calculating and developing bubble size distribution along the pipe axis taking into account coalescence and fractioning rates, as well as condensation phenomena.

The modelling of dynamic flow image development has been advanced and extensively verified for both system codes and CFD codes. The objective of ongoing developments is to calculate heat and map exchange rates between phases with various gas contents.

Expansions for disperse flows, for the water carry-over area and also for horizontal flows with transient stratification processes are still required. The well-instrumented experiments to be conducted as part of the CFD research project are expected to deliver important information in this regard.

So far, models for the momentum exchange between phases, as well as bubble population for simulating adiabatic two-phase flows with disperse gas phase have been developed and validated by experimental data. The behaviour of free surfaces has to date been examined for the two-phase flow without phase transition. In future, the two-phase flow with phase transition will be analysed to determine the intensity of the phase transition and also to establish the expansion and topology of the interphase boundary layer and the local temperature gradients.

Single and multi-dimensional description of single and two-phase flows with particle pollution

Theoretical and experimental work has been carried out to clarify individual effects and phenomena, the aim being to model and simulate the behaviour of particle-entrained flows comprising cooling water and mineral wool fragments (particle ingress, sedimentation, re-suspension, pressure build-up at preventer mechanisms). The application in principle of CFD models has been proven through the fundamental consistency between flow fields determined by measurement and model calculations.

The focus of future work is twofold – to link the part models and the application to real plants and, based on the results of using digital image processing, to include several particle classes in the CFD simulation. Given the complexity and vagueness of parameters and phenomena, soft-computing methods are also used to interpret the measurements.

Heat transfer at high temperatures and inert gas influence

A challenge of code development remains the modelling of heat transfer and the advanced development of boiling models. The influence of non-condensable gases on heat transfer and the influence of these gases on condensation have been modelled in a rudimentary fashion, although need to be expanded to higher pressures and/or temperatures.

Effectiveness of accident management measures

The German PWR-based test programs UPTF-TRAM and PKL-III D/E have created the experimental principles for detecting and modelling thermo-hydraulic phenomena in connection with procedures for ensuring emergency protection inside plants. These findings are to be continually implemented in the development of 3D codes, as well as in further improvements to system codes and in the final validation process.

4.2.2 Reactor physics for light water reactors

Methods development for the provision of nuclear cross sections

Both high-quality computer programs and data on the nuclear cross sections themselves are essential for dealing with safety-relevant issues of reactor physics. Analyses of complex arrangements such as heterogeneously loaded reactor cores or partially destroyed cores pose specific requirements. From globally available data, the latest and most reliable nuclear cross sections are amalgamated and prepared by complex mathematical procedures in such a way that they can be used as basic data for computer programs for criticality and burn-up calculations etc.

The latest knowledge on nuclear cross sections is summarised in the European library Joint Evaluated Fission and Fusion File JEFF-3.1 and the American response document Evaluated Nuclear Data File ENDF/B-VII. A broad validation basis for LWR uranium and MOX fuels exists in this regard. A closer examination of nuclear data is required for the transition to new fuel concepts.

The data on nuclear cross sections are being continuously improved by measurements and perpetual evaluation. Information on the uncertainty of data is also being provided in the new data libraries using co-variance matrices. The preparation of the latest cross section data for the used, advanced computer programs remains an important topic for the future.

Development of time-dependent neutron transport programs for transient analysis of reactivity initiated accidents

The aim is to evaluate the inaccuracies of the diffusion process for analysing reactivity transients and eliminating these inaccuracies using neutron transport programs. The ultimate goal is to qualify 3D transport methods for core calculations and integrate transport effects into existing computer programs.

Models that consider the heterogeneity effects emerging as a result of new load strategies, operating modes or accident scenarios have been developed. These models are to be expanded through the explicit use of time-dependent neutron transport equations to make them suitable for analysing fast transients such as reactivity initiated accidents. Approximations to the transport equation can be considered to reduce calculation times.

Since the neutron transport equation approach with several energy groups is a more accurate calculation method than the neutron diffusion equation with few energy groups, the quality of the simpler and faster methods can be examined.

Physics of the reactor core under elevated burn-up, mixed oxide fuel (MOX) and advanced nuclear fuel

New nuclear fuels are being discussed for the LWR with a view to reducing the existing volumes of plutonium and minor actinides. The design of suitable fuel assemblies and their burn-up behaviour, as well as the reactivity coefficients of the reactor cores are to be examined for this purpose.

The established reactor physics programs have been advanced with a view to further increasing MOX use, as well as burn-up and enrichment due to the continuously growing heterogeneity in the reactor core inevitably caused as a result. Higher burn-ups also call for the development and implementation of improved burn-up models to take adequate account of historical effects, e.g. through directly solving the nuclide balance equations in the core calculation code.

The aim of future research work remains to determine and evaluate the physical safety aspects of reactors brought about by the increased use of MOX and higher burn-up with new fuel assembly designs in existing and advanced thermal reactors.

Use of advanced nuclear fuels

New fuels are being analysed with a view to reducing the amounts of plutonium (Pu) and minor actinides (MA). The use of IMF (inert matrix fuel) is being discussed in this context. IMF is free of fertile material U-238, which means that this nuclear fuel generates no plutonium during operation. As well as examining the Pu and MA amounts generated in at least partially with IMF charged reactors, the altered dynamic properties are also to be analysed. This concerns both the reactivity coefficient as well as the thermal transport from the fuel into the coolant.

Methods development for online core monitoring

To improve the preventive level of the defence-in-depth concept within control technology, a core instrumentation-supported auxiliary reactor core model is to be developed for calculating all safety-relevant variables in the reactor core, including those that are not directly measurable. Control technology is able react in a regulated pattern according to the situation and, likewise, the response values for the core protection can approximate those of the process itself. The problem, then, is twofold. One, the reactor core model has to be adequately fast as well as accurate and two, proof that the software is suitable for use in safety process control technology has to be furnished. It is also important to develop adaptive methods that enable the auxiliary core model to be continuously modified in line with actual plant developments.

Advance development of Monte-Carlo methods i. e. in reactor dosimetry

The Monte Carlo method is used in reactor physics to analyse complex correlations between the criticality calculation, the burn-up calculation and the fluence calculation and to simulate in detail the sequences on which they are based. The long computing times for numerous applications pose a problem here. Efforts are being made to cut computing times both by accelerating convergence through variance reduction and self-optimised Monte-Carlo procedures and also benefiting from the advantages of multi-computer technology.

Another trend is to couple deterministic neutron transport codes using Monte-Carlo programs in order to analyse burn-up and/or screening problems for LWR.

4.2.3 Special aspects of reactor physics

Advance development and verification in coupling neutron-kinetic and thermo-hydraulic programs

The reactor core accommodates nuclear phenomena on one side and thermo-hydraulic phenomena on the other, in close interaction at each location and at each point in time. Linking the progress made in computer programs for reactor physics with those for plant behaviour so as to allow the two phenomena to be calculated simultaneously was therefore a logical step. During this process, various coupling strategies were developed and the corresponding coding systems created. These are being validated at international level for PWR, BWR and VVER, by OECD benchmarks for example. The new development aims to complement the accident calculations with coupled codes through systematic uncertainty and sensitivity analyses.

Reactivity initiated accidents

Reactivity initiated accidents, by virtue of the dynamic, rapid time lapse and the associated potential risks, are worthy of special attention in the research geared towards the prevention and management of accidents. Typical initiated accidents include the ejection of control rods, boron dilution transients, ATWS (Anticipated Transients Without Scram), subcooling transients due to leaks in the main steam system and the neutron-kinetic thermo-hydraulic instability with BWR. To enable simulation of this accident class also, thermo-hydraulic system codes and nuclear kinetic codes have been coupled together and validated in initial benchmark tests.

Analysis for recriticality during core degradation

As far as we understand reactor safety today, the manageability of core melt accidents is also considered. The common denominator for most of these considerations is the exclusive thermo-hydraulic approach, which takes into consideration the melt’s cooling efficiency and its interaction with the reactor pressure vessel. It is assumed that the possibility of nuclear chain reactions during core destruction can be excluded. This assumption must be verified, since the absorber and control rods can melt and flow off while the fuel rod grid is still intact during the heat-up phase. Recriticality might occur if a partially destroyed core is flooded with partially melted control rods. Until now, Monte-Carlo programs have been used to analyse parameter-type criticality conditions for reactor cores with varying degrees of destruction, varying fragment size, varying material composition and various moderation ratios. The methodology was qualified by recalculating a large number of critical experiments under differing neutron-physical conditions and in international benchmark exercises. Since this could not be concluded, the systematic analysis of the recriticality of core states occurring during core destruction must be continued in the future.

4.2.4 Fuel rod assembly behaviour

Operational transients

German PWR and BWR applications are making increased use of fuel assemblies which are set apart by extended service life in the core, higher burn-up and altered fuel composition (MOX gadolinium use) and by altered cladding tube alloys. This use is resulting in changed material behaviour in the fuel rods. For example, hydrogen deposits in the cladding tube (corrosion) caused during operation as a consequence of the fuel rod being used for prolonged periods bring about a significant change in the cladding tube material properties. An assessment of the jacket tube strength with operational transients presupposes that appropriate account is taken of these material properties in the analysis tools for the purposes of furnishing proof (e.g. THAM method). These analysis tools therefore require further development.

Loss of coolant accident

Task forces in the OECD NEA are discussing the extent to which the existing practise of setting boundary values for loss of coolant (LOCA) and reactivity (RIA) is still permitted in light of the changing material properties. For example, new crack propagation models are being discussed in the context of corroded fuel rods that exhibit different dependencies to those already known. Given the involvement of such corrosion mechanisms, fracture criteria were to be redefined and hence also affected the determination of core damage scope in the event of loss of coolant accident. What’s more, the more recent LOCA analyses on high burn-up fuel rods in the HALDEN reactor demonstrate the possibility of substantial axial sagging of fuels in a cladding tube expanded by internal pressure. This newly developed high burn-up phenomena has far-reaching effects on the cooling efficiency of the reactor core under LOCA conditions and affect future methodologies used for furnishing proof of LOCA.

Further research activities therefore aim to validate the calculation methods for determining fuel rod behaviour with high burn-up under LOCA and RIA conditions, the idea being to consolidate the new findings and to validate fuel rod behaviour using current, internationally available experimental investigations. New findings are anticipated, especially from the international research projects CABRI (rapid reactivity transients), HALDEN (operating transients and irradiation experiments with corrosion) and SCIP (integrity of corroded cladding pipes under accident strain) being conducted under the auspices of the OECD NEA.

4.2.5 Processes with core degradation and melt retention

Computer program systems for simulating accident processes in water-cooled nuclear power stations are being developed worldwide, to be able to evaluate their sequences and damage-limiting or damage-reducing emergency measures, and also quantify existing safety margins. Recent years have seen a boost in both national and international activities on the experimental examination and modelling to demonstrate what happens when melt flows out of a failed reactor core into the bottom plenum of the reactor pressure vessel. Depending on the outflow conditions, the result can be either a particle bed or a melt pool. The analyses aim primarily to assess cooling efficiency and retention potentials and/or predict the possible failure of a reactor pressure vessel. Contrary to the sequence in the early core destruction phase, i.e. until the core melt starts to form, the phenomena occurring in the late core destruction phase have not yet been adequately investigated and modelled in computer programs.

Core destruction – early phase

According to general opinion, the processes in the early phase of core destruction, i.e. until the first blockades are formed, are adequately understood and modelled. Uncertainties remain with regard to the processes occurring during a reflood in this phase, especially in terms of increased hydrogen production occurring through oxidation. The cooling by flooding on the one hand and the thermal release through oxidation on the other determine the degree of oxidation.

Experimental work is required to better quantify hydrogen generation during the reflooding of an overheated core.

Core destruction – late phase

As the core melt propagates, the melt is displaced and blockades are formed by re-solidified material in even cooler areas, the geometric structure of the reactor core changes for form a complex shape. This alters the accessibility for the steam flows and, in the event of a flood, for the water influx.

From this core destruction phase onwards, then, the overriding question is whether and if so up to what degree of damage the melt can be cooled and retained in the core if the water supply is activated, or after how long and in which manner the melt starts to flow out of the core. The manner in which the melt flows out of the core dictates the cooling potential or the periods of retention in the bottom plenum of the reactor pressure vessel. The question of how much molten mass can accumulate in the core and how this ultimately flows out is elementary to the primary issues of cooling efficiency and retention. The lateral spread of the melt compared to downwards displacement plays a pivotal role here.

Models have been developed for simulating progressive core damage and core destruction. Further analyses are required, especially for modelling the melt, melt flow and re-solidification in increasingly complex structures.

The formation of melt in the core hinges on the formation of an adequately strong crust. The existence and thickness of a crust around a pool is largely determined by thermal flows from the pool and heat transport. A simplified, empirically adapted model that flexibly calculates thermal flows to edge has been developed. Uncertainties remain regarding the outflow of the melt, especially with regard to how the opening size and jet formation develop after it passes through and joins onto the grid plate.

Existing models are able to calculate scenario-dependent melt accumulation in the core, as well as outflow processes. There is, however, a need to verify and further develop the simplified pool model and the description of outflow and failure, e.g. due to the lower grid plate, the hydrogen formed by the reaction with metallic parts, the fission product release from the melt and related transport processes in the fluid.

Cooling efficiency of the destroyed core

Whether or not cooling efficiency can be restored during a water supply or how such a supply influences further development has to be determined at every phase of the accident process. This also creates potentials for retaining the melt in the core.

To be able to investigate cooling efficiency during core destruction, i.e. in states with molten content, appropriate models have been developed and compared against both other models and experimental results.

Displacement to the lower plenum and interaction with residual water: Melt retention in the reactor pressure vessel

The later phases of accidents involving core destruction are characterised by the behaviour of the melt in the upper plenum and the integrity of the reactor pressure vessel bottom. According to earlier model calculations, the melt outflow from a melt pool in the core occurs primarily on the upper edge of the pool and hence within a limited flow rate. The melt can be directed laterally and downwards or spill over and spread over the entire grid plate. The current models work on the assumption that the melt pours into the residual water in the lower plenum as a single stream or as multiple streams. Further experiments are being planned in Germany in this field. Fragmentation and debris bed formation issues must also be addressed.

Steam peaks in the mixing phase and the increased oxidation of metallic elements and associated hydrogen formation brought about by fragmentation are also important safety aspects.

Cooling efficiency of debris in the lower plenum, formation of a melt pool

Once debris has formed on the reactor pressure vessel bottom, which comprises compact and molten parts, its cooling efficiency becomes a key issue with regard to both delays through to a melt pool formation and also the cooling potentials, depending on the displaced molten mass and the restored water supply. Its potential prior to the formation of a large melt pool must be explored so that the opportunities resulting from a water supply can also be assessed in such a late accident phase.

In order to realistically record the flow conditions for complex, even heterogeneous areas and partially solidified debris configurations, the lateral water supply in lower debris areas must be described using at least a 2D model. The 2-fluid flow model must comprise both continuous current and counter-current.

The correlations for calculating thermal transition and friction pressure loss, however, are compounded by uncertainties and therefore need to be verified. The cooling potential of a steam flow through dry zones in the debris if water is able to penetrate these zones must also be quantified.

Numerous experiments have created an extensive database for developing and validating models for cooling debris and also for forming a melt pool in the lower plenum.

Melt pool in the lower plenum of the reactor pressure vessel, external cooling and reactor pressure vessel failure

Larger molten masses in the lower plenum can cause the reactor pressure vessel to fail through thermal-mechanical load. This can be delayed or even avoided by effective external cooling. Potential cooling effects through thin gaps between closed or porous crusts and the reactor pressure wall can contribute to a reduction in the thermal load.

In general, the development to form the melt pool, including the melt pool behaviour is modelled in simplified terms. The modelling for both the melt pool formation and for the thermal transition between melt and reactor pressure vessel has yet to be substantiated by experiments.

Due to the heterogeneous melt states, the behaviour of the melt pool and the thermal flow distribution at the reactor pressure wall is key aspect of this problem. Chemical reactions can cause melt point reductions in the reactor pressure vessel wall and also miscibility gaps, which in turn lead to phase separation and layer formation. In light of the findings from the OECD and ISTC projects in this field, corresponding models must continue to be developed.

Melt - coolant interaction (steam explosion)

The potential result of large quantities of core melt being mixed with water during a core melt accident is a steam explosion that can jeopardise the integrity of the containment. To be able to reliably verify and eliminate this possibility, realistic upper loads for steam explosions (pressure-time-flows, explosion energies) can be determined. Particular importance is attached to the inherent limiting of the participating melt and water masses through severe though non-explosive vaporisation during the pre-mixing phase that must precede the explosion itself. This effect is being closely analysed through both experiment and theory. Models are now available for describing the melt-cooling interaction.

The remaining uncertainties regarding the pre-mixing of melt and coolant, as well as the accompanying rapid steam formation, which can have considerable influence on the subsequent explosion event, need to be further analysed by experiments in the OECD project SERENA Phase 2.

4.2.6 Methods development for assessing the reliability of research results, including with regard to their transferability to real plants

The SUSA (system for uncertainty and sensitivity analysis) method for quantitatively evaluating the reliability of computer results has been extensively developed. This method has been successfully employed in several thermo-hydraulic accident analyses. A computer program processes parameters whose real value is not precisely known or which are in principle distributed stochastically. Today’s knowledge of these imprecisely understood parameters can be expressed by distribution. If these distributions are taken into consideration, distributions for the computer results are also obtained.

The parameters with the greatest influence on the end result of the calculation can therefore also be determined. This will in turn provide information on the need for future research work to dispel recognised uncertainties. The distributions of the computer results can be used to evaluate the clearance from the specified values for safety-critical parameters.

Increasing use is being made of this method for processes in the containment, in three-dimensional thermo-hydraulic analyses and also in the fields of fuel rod behaviour and reactor physics. The experience feedback can be used to determine the work required for advancement and/or optimisation of the methods.

4.3 Core melt in the containment, steam explosion, hydrogen distribution and combustion and countermeasures, fission product behaviour in the containment

Statement by the Evaluation Commission:

"The integrity of the containment, the final barrier against the release of radioactive substances into the atmosphere must also be evaluated for highly unlikely accident event sequences. A realistic assessment requires a more in-depth current knowledge of accident sequences, as well as of the efficiency and reliability of measures for preventing inadmissible containment loads.

Table 3 : Subject areas, individual topics, research institutions

Fig 7 : Manpower forecast for the subject areas

Technical Field „Core Melt in the Containment, Steam Explosion, Hydrogen Distribution/Combustion and Countermeasures, Fission Product Behaviour in the Containment“

Chap.	Topic Areas	Specific Topic	Research Institution
4.3	**Core melt in the containment, steam explosion, hydrogen distribution/combustion and countermeasures, fission product behaviour in the containment**		
4.3.1	Computer codes for simulation (system and integral codes)	**COCOSYS**	GRS
		ASTEC	FZK, GRS
4.3.2	Thermal hydraulics in the containment	**Short-lived phenomena**	FZK, GRS
		Formation and resolution of a stratified atmosphere	Becker, FZK, GRS, RUB
4.3.3	Oil and cable fires		Becker, GRS, IBMB
4.3.4	Hydrogen combustion	**Experiments and models**	Becker, FZK, GRS
		Loads and load transfer on containment structures	FZK, (GRS)
		Countermeasures, in paricular catalyst recombination	Becker, FZJ, FZK
4.3.5	Use of CFD-Modelling		FZK, GRS
4.3.6	Core melt behaviour	**Release into containment**	FZK, GRS
		Direct Containment Heating (DCH)	FZK, GRS
		Cooling and retention of core melt	FZJ, FZK, IKE
		Core melt/coolant interactions (steam explosion)	FZK, IKE
		Molten core/concrete interactions	FZK, GRS

Chap.	Topic Areas	Specific Topic	Research Institution
4.3.7	Fission product and aerosol behaviour	**Impact of thermal hydraulics on fission product and aerosol behaviour**	GRS, RUB
		Release of fission products and aerosols into the containment	GRS
		Impact of transient events on fission product and aerosol behaviour	Becker, GRS
		Release of fission products and aerosols from the sump	Becker, GRS, RUB
		Release of fission products and aerosols from a molten core/concrete mixture	FZK, GRS
4.3.7	Fission product and aerosol behaviour	**Transport and deposition behaviour of fission products and aerosols**	Becker, GRS, RUB
		Chemical and transport behaviour of iodine (and further chemical elements)	Becker, GRS
		Impact of spray systems on the fission product and aerosol behaviour, in particular on iodine	Becker, GRS, RUB
4.3.8	Phenomena respectively processes specifically relevant to BWR		Becker, GRS, IKE
4.3.9	Development of methods for the assessment of the reliability of computer results also with regard to the applicability to real plants		FZK, GRS

Table 3: Main Research Topics of the Research Institutions in the Technical Field „Core melt in the containment, steam explosion, hydrogen distribution/combustion and countermeasures, fission product behaviour in the containment”

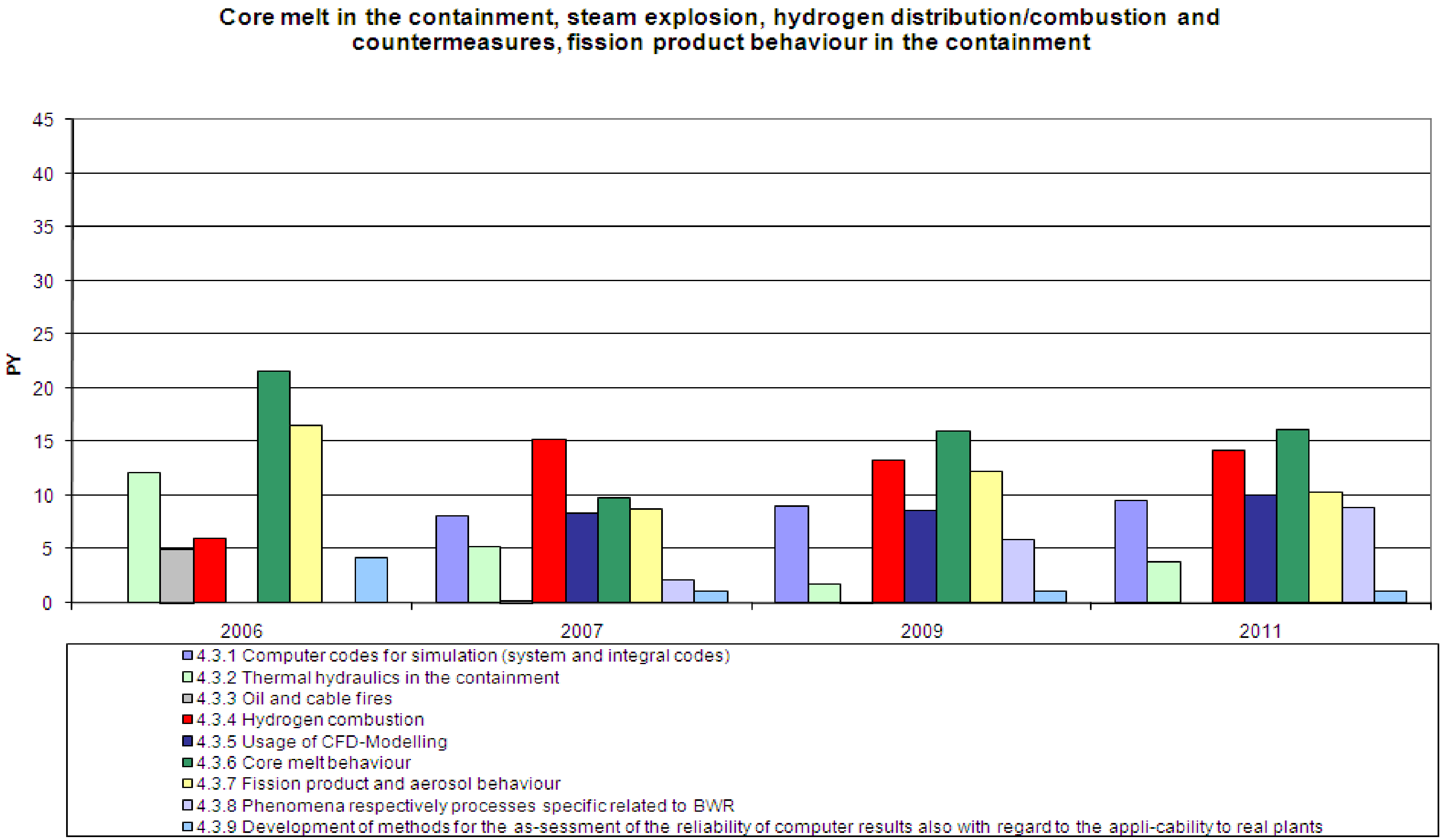

Fig. 7: Personnel Prognosis for Topic Areas in the Field of "Core melt in the containment, steam explosion, ..."

In the event of accidents in light water reactors where radioactive materials is released from the primary coolant circuit the containment represents the final barrier against the release of radioactivity into the atmosphere. If radioactivity is released into the atmosphere, the radionuclide source term is largely determined by the behaviour of radionuclides and aerosols within the containment. A highly detailed knowledge of the processes occurring in the containment is therefore required for assessing the safety of nuclear power plants and evaluating and defining “accident management” measures. Accident-related processes in the containment are highly complex. A reliable description therefore hinges on an understanding of all essential phenomena and their interactions.

Risk assessments also involve consideration for highly unlikely accident sequences and releases of core melt into the containment and a quantification of their effects. The load bearing limits of the containment and the potential loss of its functionality must also be determined. The source term for the release of radioactivity into the environment must be determined for the event of the containment losing its retention function.

To explain the phenomena involved in the accident-related processes occurring in containment, numerous experimental research activities have already been carried out at lab and pilot plant level. These were then used as a basis for the development and validation of more complex simulation programs. These programs can be used to assess the safety of nuclear power plants in the event of serious accidents, evaluate “accident management” measures and furnish safety proof for selected accident scenarios.

As well as lab and individual effect tests, large-scale experimental projects for analysing integral processes have also and are still being conducted in Germany.

Despite extensive knowledge regarding safety-critical phenomena and the modelling of same in simulation programs, there are still gaps that limit the accuracy of quantifying residual risk and which therefore have to be the object of further research.

4.3.1 Simulation programs (system and integral codes)

COCOSYS

The Containment Code System (COCOSYS) has been under development since 1994. COCOSYS is based primarily on mechanistic models and is used to simulate all key processes and states during serious accidents in the containment of light water reactors, where design basis accidents can also be extensively simulated. A further aspect is the close consideration of the interactions between the various phenomena, e.g. thermo-hydraulics, hydrogen combustion, aerosol and nuclide behaviour, as well as melt behaviour.

Numerous calculations and comparisons under national and international benchmarks show that COCOSYS has a high development status overall (also in the international comparison), although some modules and/or models still harbour significant deficits. The program structure needs to be revised and individual models need to be harmonised.

In terms of modelling and programming technology, the modules for simulating thermo-hydraulic and aerosol behaviour are the most advanced. The transport and behaviour of aerosols and fission products in the atmosphere are the most widely understood and analysed; although the modelling of iodine behaviour is at a more advanced stage than that of international competition, there is since a need for further experimental and analytical development and validation. The modelling of melt behaviour requires more intensive advanced development in terms of both programming and the models themselves.

In recent years, the expansion and/or development of the COCOSYS model has focused on the spray systems, expansions in simulating combustion processes, new developments in melt behaviour, improving couplings, eliminating inconsistencies and expanding the area of application. Additionally, a module that enables processes that occur as core melts are released from the reactor pressure vessels to be simulated has been incorporated into COCOSYS. In addition to the development work already complete, extensive validation and application calculations have been performed to highlight the simulation quality of the codes and weaknesses in program development that need to be remedied.

ASTEC

The integral code ASTEC, which was developed in a German/French collaboration, is partially based on simplified models and correlations. The purpose of this code is to simulate all processes occurring during incidents and accidents (reactor cooling circuit and containments), from the triggering event through to the fission product release from the containments into the environment. The first “full version of ASTEC (ASTEC V1) has been available since 2002. This also allows thermo-hydraulic simulation in the phase prior to core uncovery. The version released in December 2006 is able to simulate the key phenomena occurring during breakdowns and accidents (apart from steam explosion and fire).

Recent, important development activities have focussed primarily on increasing numerical robustness, improved modelling and on releasing volatile fission products, simulating the reflood of an intact or slightly destroyed core (i.e. without disbanding the fuel rod geometry) and also on installing a new melt-concrete interaction module.

A key development goal of further activities is the ongoing expansion to include boiling water reactors. This requires BWR-typical components to be mapped in ASTEC. The required model expansions must be specified and implemented.

4.3.2 Thermo-hydraulics in the containment

Short-term phenomena

Short-term phenomena are the flow processes in the containment after a major break in a coolant line within the first 100 seconds or thereabouts. They are characterised by huge uncertainties surrounding the further distribution of liquid water, steam and non-condensable gases from the break area into its environment. These phenomena relating to the design of a plant are especially important for determining the pressure differentials between adjacent areas.

Formation and disbanding of atmospheric layering

Atmospheric layerings can form in the containment through different temperature levels or through gases with varying density (e.g. hydrogen influx). Knowledge of the formation and disbanding of atmospheric layerings is required to both simulate potential hydrogen combustion and also fission product and aerosol transport in the containment.

The international standard problem ISP-47 has shown that the mapping of the processes involved in the formation and disbanding of atmospheric layerings with the used calculation codes has been only partially adequate and that the associated phenomena are still not fully understood. There are plans for further experiments that should achieve a better understanding of the processes involved in the formation and disbanding of light gas layerings and provide further measured data.

4.3.3 Oil and cable fires

Due to the fire loads present in a nuclear power plant (oil and cable in particular) and potential ignition sources, fires are also to be expected in nuclear power plants. Oil and cable fires can impair safety-critical systems and components. Reliable, qualified simulation codes are required for evaluating and improving fire safety measures, for investigating and assessing the effects of fire on systems and components and for assessing the fire-dependent influence on the release of radioactive substances.

3D simulation programs that are able to devise proposals for improved fire-fighting concepts have been developed for optimising fire safety in German nuclear power plants. However, benchmark calculations have shown that none of the fire models currently available can, with sufficient accuracy, simulate fire-dependent heat and smoke spread or the resultant effects on safety-critical objects.

The OECD-PRISME project involves conducting experiments that examine in particular smoke and heat spread. The derived results are to be used as a basis for both code validation work and also to achieve model improvements.

4.3.4 Hydrogen combustion

Experiments and modelling

Combustion of the hydrogen released during serious accidents and the associated extremely rapid pressure rises can call into question the retention ability of the containment, especially if this results in detonation. The focus must therefore be on implementing so-called DDT (Deflagration Detonation Transition) criteria. Lumped parameters and 3-dimensional CFD combustion models must therefore be advanced and validated with a view to influencing the combustion speed through the spatial relationships in the containment (chamber branches, merging of flame fronts, open jet ignition etc.).

The application field of CFD codes for rapid, turbulent combustion (e.g. COM3D) is hence being expanded by models for extinguishing processes, convective thermal loss, porosity and the transition from laminar to turbulent flame. There is also still a need to validate thin mixtures and H_2 gradients.

In this context, analytical and differently scaled experimental tests must be provided for a realistic assessment of the risk potential of hydrogen-steam-air mixtures. The tests must range from the transition processes through to the fast turbulent deflagration and finally to detonation. The criteria for flame acceleration and detonation transition must also be determined for mixtures with concentration gradients. Further experiments on water distribution, recombination behaviour and hydrogen deflagration are required.

Load and load absorption containment by structures

Vertical flames in particular pose a risk of thermal load, while the main problem with rapid combustion is the mechanical loads that can cause containment structures to fail. Important in this context as far as failure mechanisms and failure criteria are concerned are the differences in the containment structures, which depending on reactor type can be from steel or reinforced concrete with or without pre-stress (see chapter 4.1.1).

In future, thermo-hydraulic and mechanical structure analysis models must be linked in order to determine loads and assess the integrity of containment structures.

Counter measures, specifically catalytic recombinators

In line with an RSK recommendation, most PWR containments have been retrofitted with catalytic recombinators. In experimental tests and to accompany the development and validation of models, calculation tools that have been used to proof the efficiency of catalytic converters under various breakdown conditions have been provided.

Safety-critical assessments hinge on the effects of catalytic recombinators on other status variables in the containment, e.g. the release of gaseous iodine from aerosol particles, changes in the aerosol variable spectrum, convection conditions and atmospheric humidity being determined. The possibility of mixture at an overheated catalytic converter igniting causing immense hydrogen releases must also be considered. Given the complexity of the processes, experimental tests and model developments are required to clarify the sequences involved in reaction kinetics, temperature distribution, material and thermal transport and fluid dynamics. Experimental and analytical investigations are currently also being conducted.

In the medium term, other tests on the formation of volatile iodine from aerosol particles such as CsI in the recombinator must be conducted in order to reliably quantify their contribution to the fission product source term.

4.3.5 Use of CFD modelling

The “lumped parameter“ (LP) codes do not include the momentum equation. The simulation of local speed fields and so-called jet flows are therefore limited. CFD code limitations are the complexity (and hence prolonged computing times) of creating an extensive grid network and deficits in the modelling of two-phase flows and turbulence.

To overcome the fundamental limitations in LP codes and benefit from the advantages of CFD codes, the two modelling types have now been linked for the first time. This process revealed pronounced numerical oscillations, which are caused by using different thermo-physical property routines. Nonetheless, in order to benefit from the advantages of the CFD programs, the current preference is for separate CFD calculations that enable the obtained results to be used as input variables for LP codes.

The first experiences with CFD codes relate to calculations on the gas mixture, H_2 combustion in various spatial arrangements, kerosene fires, circulation flows in the sump of reactor buildings during the recooling operation and the thermo-hydraulics of a Russian-designed “confinement”. These calculations show that although CFD codes undoubtedly have a high potential for the spatially detailed analysis of processes, there is still a need for selective advanced development and considerable validation work, in order to deliver reliable information on reactor safety.

Work on the selective advanced development of CFD codes is required for application to the safety enclosure.

4.3.6 Core melt behaviour

Entry into the containment

In terms of melt entry into the containment, a distinction must be drawn between increased reactor pressure vessel internal pressure and (the objective of AM measures) pressure relief prior to reactor pressure vessel failure. While under elevated internal pressure in the reactor pressure vessel, ejected melt can cause local damage to fittings and especially to the containment shell itself (DCH: Direct Containment Heating), and also triggers an uncontrolled, initial distribution of the melt in the reactor pit and in the containment, the outflow of melt from a failed reactor pressure vessel once the pressure of the primary circuit has been relieved can be expected to bring about more moderate and even displacement processes. DCH is described in the sub-chapter below, while this section focuses on illustrating in more detail the entry of unpressurised melt.

Inadequate fragmentation and insufficient water trap and/or supply and the resultant failed cooling process causes a delay in the melting of the core material and in the melt-concrete interaction. The key safety issues then become the fusing of the concrete base with regard to duration and failure areas and also the pressure build-up caused by the gas released from the concrete-melt interaction and the release of fission product into the atmosphere of the containment (see section on core melt – concrete interaction). The aim is to evaluate possible accident management measures and their potentials, especially the recoverability of a cooling process through water and hence the retention of the melt in the containment.

Direct Containment Heating (DCH)

The processes associated with melt entry into the containment have a significant effect on the accident sequence that follows. The nature of reactor pressure vessel failure, the geometry of the reactor pit and, most importantly the pressure at which the core melt enters the containment are key factors in how the accident progresses develop. Dependent mainly on pressure, the core melt distributes itself inside the containment and interacts with the surrounding atmosphere. All phenomena that are essential to a sound assessment of risk and retention potentials must be investigated in detail.

For certain existing reactor geometries, the various aspects of melt dispersion and direct containment heating (DCH) at moderate pressure and with various failure types have been and are still being examined by both experiment and analysis. Safety and countermeasures can only be proposed once the potential risks caused by the dispersed melt have been analysed in detail.

Core melt cooling and retention

Limited cooling and retention options have already been applied in accident management (AM) measures on a design basis. Heat is generated as soon as concrete starts to melt. Flooding with water from above as an AM measure will bring about considerable cooling effects from crust formation interaction and the release of gas from the melt/concrete interaction. The most likely mechanism is a repeated, eruptive ejection of melt caused by pressure build-up under the crust (“volcano effect”) or a continuous process of entrainment of melt into the gas flow and outflows through openings in the crust. In both processes, coolable debris is expected to form on the crust. Further mechanisms in the discussion include water penetration through cracks in the crust and repeated break-up of formed crusts with subsequent water flow to the liquid melt once the melt level has dropped due to concrete erosion. These effects and cooling effects were the object of the MACE experiments at ANL, where they are now being further analysed in continuous experiments as part of the OECD-MCCI-2 project. However, these experiments with prototypical material have not yet been able to substantiate adequate cooling. Generally speaking, the explained mechanisms remain questionable, at least in terms of their sustained effect and efficiency for overall cooling. Adequate cooling was, however, achieved for limited melt masses.

If structural measures are applied, melt can be retained and cooled in the containment provided that a certain melt layer thickness is not exceeded. The fundamental differences between the target concepts is whether a rapid cooling and hardening of the melt is achieved or whether is it safely contained and adequately cooled at the edge, fusing thus being prevented.

Extensive experimental and analytical work is currently being carried out in Germany on this subject. International experimental tests focus on the long-term behaviour of core melts in the concrete base and also on the effects of the counter-measures taken to achieve cooling and rapid hardening.

Models and codes for debris formation in a water pool through fragmentation of melt streams and for debris cooling are available for reactor pressure vessel processes.

Core melt - coolant interactions (steam explosion)

A key aspect of comprehensive safety analyses for nuclear power plants is the safety-critical assessment of the interaction between the molten core material and the water inside the reactor pressure vessel (in-vessel steam explosion) and/or the interaction between the core melt flowing out from the reactor pressure vessel and the water in the containment reactor pit (ex-vessel steam explosion) that occur during an accident involving the outflow of core melt. Such a steam explosion can significantly intensify the course of an accident process. The main issue is whether such a steam explosion can be severe enough to cause considerable damage to essential structures or even the containment itself. The destruction of surrounding walls could jeopardise the safety objective of cooling in the deep water pool.

Overall, the calculations on reactor scenarios carried out in the first phase of the OECD project SERENA and as part of SARNET indicate severe restrictions in the explosion strength. These restrictions are derived from the scenarios themselves (limited mass flow into the water), the limited coarse fragmentation (e.g. only partial outbreak of thicker melt flows), the self-adjusting high steam content, especially with relatively small melt drips from the fragmentation of corium-melt flows, the restrictions in the fine fragmentation in surges during the explosion phase and the weakening of such surges under 2D/3D spread and reflexion at free boundary layers.

To clarify the safety-critical, significant and outstanding issues on void formation in pre-mixtures, on the distribution of melt drips in the mixture and the cooling of same, specific experimental analyses are to be conducted in the planned OECD project SERENA-2. The models are to be examined on the basis of experimental results.

Core melt - concrete interaction

Should a reactor pressure vessel fail during the course of an accident, core melt enters into the reactor caverns and reacts with the concrete structures inside. Today's improved knowledge of solidification processes in multi-component melts enables the long-term behaviour of the melt to be described in more precise terms. Although extensive experimental and analytical work has been carried out on this subject, further analyses are required to allow the medium and long-term thermo-chemical loads occurring on the containment during reactor accidents involving core melt spread in the containment to be quantified more accurately and hence feasible accident sequences to be realistically assessed.

The work involved in the OECD-MCCI project has selectively expanded the findings on the complex issue of ex-vessel melt behaviour, especially on crust formation, on the melt's cooling potential and on the concrete-melt interaction. Large-scale 2D experiments on concrete erosion (silica-based concrete and limestone concrete) caused by melt, however, indicate significantly different erosion profiles. To clarify these phenomena, appropriate long-term tests on 2D-erosion are envisaged in the MCCI follow-up project (OECD-MCCI-2).

The legalities of the interaction between combined oxide/metal melt and concrete are being examined as part of ongoing national lab tests. Further lab tests include experiments with reheated, prototypical melts in two-dimensional geometry.

Code development work aims, building on an evaluation of current and future experimental programs, to reduce the uncertainties associated with computation results.

4.3.7 Fission product and aerosol behaviour

As assessment of the radiological source term for potential releases from the containment (leaks, intentional pressure release or vessel failure) requires knowledge of fission product behaviour and fission product distribution in the containment and the chronological sequence of same. Considerable uncertainties remain in some areas.

Especially noteworthy in Germany is the multi-purpose ThAl test facility, in which experiments on pilot plant scale, e.g. on fission product behaviour in the containment, on iodine transport behaviour, on aerosols and hydrogen and on thermo-hydraulics have been carried out under typical accident conditions and which are being continued as part of an OECD project.

Influence of thermo-hydraulics on fission product and aerosol behaviour

The behaviour of fission products in the containment and hence also the extent of potential radioactive release into the atmosphere of a nuclear power station during serious accidents is strongly influenced by the thermo-hydraulic boundary conditions prevailing in the containment. Extensive experimental and analytical work has and is still being carried out on this subject in both national and international circles. Other activities are required as a priority for the advanced development and validation of CFD codes and also to validate system codes.

Fission product and aerosol entry into the containment

During accidents in light water reactors involving damage to fuel rod claddings and release of fission products from the fuel rods, radioactive substances are emitted into the primary circuit which can leak into the containment. The entry of fission products/aerosols into the containment largely determines subsequent accident progression. Suitable models must be validated using selected experimental results, e.g. from the PHEBUS-FP program.

Influence of transient events on fission product and aerosol behaviour

The small amount of experimental information on a host of phenomena such as High Pressure Melt Ejection (HPME), Direct Containment Heating (DCH), hydrogen combustion and dry resuspension must be used in future for an analytical description and/or has already been used in part (for DCH, see also chapter 4.3.7 ”Core melt behaviour“).

Small-scale experiments have been employed, for example, to selectively analyse the resuspension behaviour of previously deposited aerosols through transient flow pulses. A database of key parameters influencing resuspension has been set up, time-resolved particle size distributions and particle concentrations in the gas phase determined and the dependency of the aerosol carrier affinity and the maximum upstream flow speed for the relative overall particle erosion empirically demonstrated. Larger-scale tests on aerosol resuspension as a consequence of hydrogen deflagration have been conducted in the ThAI facility. Considerable volumes of the deposited aerosol material were released in the process. Further larger-scale analyses on dry and wet resuspension are required.

Fission product and aerosol release from a sump

The fission products dissolved or suspended in water traps can be discharged into the containment through droplet entrainment during the boiling process. Especially if the integrity of the containment is maintained over several days and the aerosol concentration in the safety vessel atmosphere is reduced by several orders of magnitude, even the smallest droplet entrainment can severely affect activity concentration in the containment.

Analytical model approaches for the release of fission products previously dissolved in liquids as a function of gas volumes admitted into the sump and the bubble spectrum existing on the surface of the liquid have been devised, for example, in connection with the development of the RECOM code. Due to the inadequate scaling of previous experimental tests in this field, however, the scope of these model approaches is still limited. Current experiments on fission product release from a boiling sump have recently been conducted as part of the ThAI tests. The established entrainment of a large number of tiny droplets demonstrates that this effect can contribute significantly to the air-carried fission product inventory, especially in the late phase of an accident. Further analyses are required.

The retention effects that are highly capable and that can significantly influence fission product distribution in the containment must also be considered in determining the release of fission products from water traps. The models developed for this purpose require further experimental validation. More precise measured values obtained by using current aerosol measuring technology must be provided to dispel these uncertainties.

Fission product and aerosol release from a core melt – concrete mixture

The release of gases, aerosols and fission products from the core melt is largely influenced by other melt behaviour, the interaction between concrete and melt in particular. Further research into the fission product and aerosol release from a core melt – concrete mixture is required.

Transport and sediment behaviour of fission products and aerosols

Detailed knowledge of the transport and sediment behaviour of aerosols and fission products is required to determine the chronological progression of fission product distribution inside the containment. A host of experiments and analytical work has and is still being carried on this subject. As part of the ThAI experiments, key data on iodine transport behaviour in single and multi-room geometries has been established, including on the exchange of gaseous iodine between sump and vessel atmosphere and also on iodine transport to walls with wall condensation.

Further experiments and analyses are required on the removal of aerosols deposited under iodine through the draining wall condensate film.

Another area in need of research is the agglomeration of finely-dispersed aerosols on a core melt aerosol. Finely-dispersed aerosols occur, e.g. through entrainment from a boiling sump or under an iodine/ozone reaction (gas-to-particle conversion) and can, unless agglomerated on a coarsely-dispersed aerosol (core melt aerosol), remain carried in the air for prolonged periods and hence contribute to the source term.

Chemical and transport behaviour of iodine (and other elements)

The significance of iodine for severe accidents is largely down to its high volatility, its chemical reactivity and its radiotoxicity. The iodine aerosols released in the containment deposit predominantly on the surfaces and are washed into the sump under condensing conditions. However, chemical reactions that also occur under the influence of the strong irradiation field, gaseous iodine is released from the sump and also from the walls into the containment atmosphere over the longer term. Some of this iodine is then partially re-adsorbed at the surfaces. Iodine transport processes between sump, containment atmosphere and structural surface play a key role in determining volatile iodine and hence for the source term.

Considerable progress has been made in recent years in the analysis of iodine sedimentation and the resuspension of steel surfaces, and also in the exchange of gaseous iodine between sump and atmosphere in single and multi-space geometries.

Further research is required to quantify the following phenomena:

- Iodine/ozone reaction,
- Formation of high and low volatile organically bound iodine (organic iodine) through the pyrolysis or burning of oils, cables and other synthetic materials.
- Interaction between iodine and colour-coated surfaces.
- I2-mass transfer under varying convection,
- I2-sedimentation of aerosols, associated with the conversion of a gaseous iodine species into a aerosol-like iodine species with different further behaviour,
- Formation of I2 from aerosol-like CsI in recombinators,
- Removal of I2-deposits through draining condensate and
- Iodine behaviour under complex accident conditions, taking into account key critical parameters influencing iodine behaviour in tests with integral character.

A selective expansion with regard to central chemical reactions for Cs, Rn and Te is a further medium-term requirement. Other fundamental activities are also required for “BWR-specific iodine chemistry (see chapter 4.3.8).

Influence of spray systems on fission product and aerosol behaviour, iodine in particular

The spray systems often installed in foreign nuclear power stations are supposed to reduce the pressure and temperature loads occurring under accident conditions and lessen the source term released into the atmosphere by removing aerosols and fission products. To be able to simulate these processes, spray models have been developed and implemented in the German Containment code COCOSYS. Earlier spray models take no account of either the entrainment of atmospheres through the spray drops (atmosphere entrainment), which contributes significantly to the containment blending, or the potential generation of aerosol-like water droplets from the spray jet (mist). Experimental validations and model expansions on the thermo-hydraulic influence of the spray models, as well as experimental data on the removal of aerosol particles or iodine from the atmosphere using spray systems are required.

4.3.8 Phenomena and/or processes with specific BWR relevance

Specific work on containment in boiling water reactors has related primarily to their pressure release systems. Nonetheless, existing documentation provides only isolated details on the mixing processes in the water zone of the BWR condensation chamber, no overall depiction is yet available. Neither is there a functional and mechanistic model for calculating these processes as part of integral accident simulations.

Overall, the BWR-specific aspects of the following phenomena and processes must be examined by both experiment and analysis:

- Influence of B_4C absorber material on melt composition, behaviour and iodine chemistry,
- melt composition,
- reactor pressure vessel failure modes,
- effects of pit geometry and the potential presence of water,
- ”pool scrubbing“ (iodine retention in the condensation chamber),
- cable fire.

For the sake of completeness, it should be mentioned at this juncture that the elevated simulation capability of the German/French integral code ASTEC for boiling water reactors requires model expansions and/or modifications, especially in the reactor cooling circuit and in the reactor pressure vessel, and that containment models must also be tested and toughened.

4.3.9 Methods development for evaluating the reliability of research results, including with regard to their transferability to real plants

The SUSA method (see chapter 4.2.6) was developed in Germany for the purpose of quantifying uncertainties. It has been selectively applied for thermo-hydraulics in reactor cooling circuits. This is now to be further expanded to include the simulation of phenomena and processes in the containment.

Further analyses are required in this area. In particular, fundamental rules for examining the extrapolation capability of individual models are to be devised and then applied (User Guidelines). This applies likewise for “lumped parameter” and CFD models.

4.4 Methods development for probabilistic safety analyses, for control technology and diagnostics and also for assessing the human factor,

Statement by the Evaluation Commission:

“To improve tools used for identifying weaknesses in plant design and process control, probabilistic methods are to be advanced and existing evaluation uncertainties reduced”.

Table 4 : Subject areas, individual topics, research institutions

Fig 8 : Manpower forecast for the subject areas

Technical Field „Development of Methods for Probabilistic Safety Analyses, for Instrumentation and Control, and for Human Factor Assessment“

Chap.	Topic Areas	Specific Topic	Research Institution
4.4	**Development of methods for probabilistic safety analyses, for instrumentation and control, and for human factor assessment**		
4.4.1	Development of methods for probabilistic safety analyses		GRS
4.4.2	PSA on level 1 and level 2	**CCF models**	GRS, IPM
		Fire	GRS, IBMB
		Reliability on active components under accident conditions	GRS
		Extended modelling of uncertainties	GRS, IBMB, IPM
		Dynamic PSA	
4.4.3	PSA on level 3		GRS
4.4.4	Further development of methods for specific PSA applications (e. g. internal and external events, electric auxiliary power system)		GRS
4.4.5	Development of methods for instrumentation	**Safety assessment of computer based systems (software und hardware)**	GRS, ISTec
		Characteristics of reliability of computer based systems	GRS, ISTec

Chap.	Topic Areas	Specific Topic	Research Institution
4.4.6	Human behaviour	**Probabilistic approach (assessment of human reliability)**	GRS, IPM
		Systematic approach (organisation, structures, implicit standards)	GRS
		Quantitative assessment of the contribution of organisations and safety management on safety and reliability	GRS
4.4.7	Technical systems to support human performance	**Task distribution man / machine (degree of automation)**	GRS, ISTec
		Prognostic tools (focal points: test control room, simulators)	GRS, TU-MGD
		Further development on diagnostic methods	IPM, ISTec

Table 4: Main Research Topics of the Research Institutions in the Technical Field „ Development of methods for probabilistic safety analyses, for instrumentation and control, and for human factor assessment “

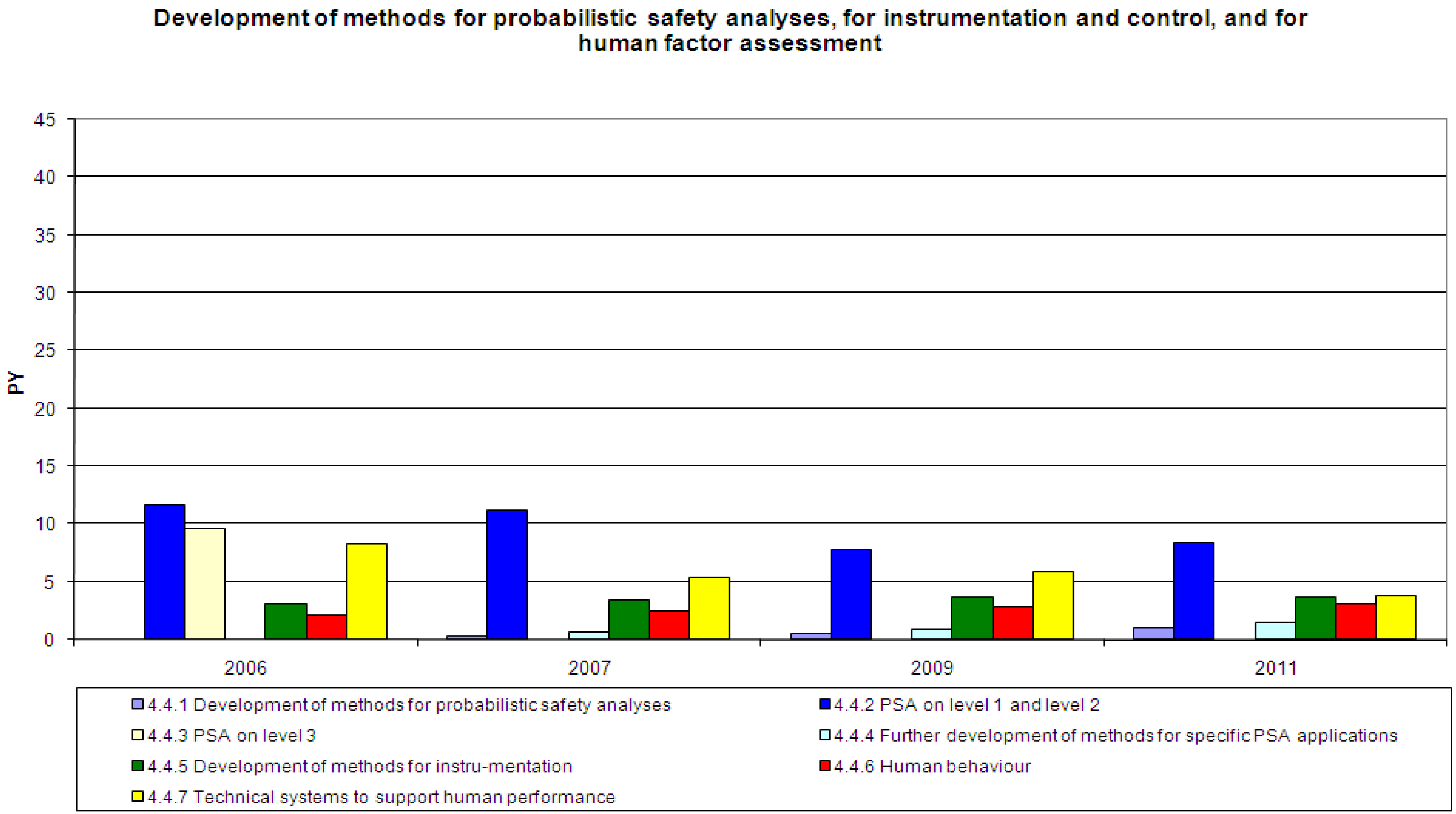

Fig. 8: Personnel Prognosis for Topic Areas in the Field of "Development of methods for probabilistic safety analysis, ..."

4.4.1 Methods development for probabilistic safety analyses

In a PSA, all key data on plant design, operating modes, service experience, component and system reliability, human actions, as well as cross-plant safety effects are analysed and amalgamated to enable an overall plant assessment. Plant behaviour during accidents is modelled in the PSA based on the knowledge of the accident analysis with regard to core damage (stage 1), radiological release into the containment (stage 2) or external damage / exposure (stage 3). A PSA enables the equilibrium of existing safety technology to be evaluated, potential weaknesses identified, potentials for remedying same highlighted and the efficiency of emergency measures assessed.

The aim of research activities is to further develop the methodical principles and tools for conducting a PSA and to quantify their reliability. Other aspects such as personnel operations, common cause failures, far-reaching internal and external effects (fire inside the plant, flooding earthquakes, high-water, plane crashes etc.) or Accident Management (AM) are also to be included, more recent technical developments (digital control technology etc.) or the failure of passive components or system functions to be taken into account and uncertainty and sensitivity analyses of key parameters to be conducted. As far as a higher efficiency of probabilistic safety analyses is concerned, user-friendly system interfaces are to be developed using the required ancillary programs for PSA methods.

4.4.2 PSA level 1 and level 2

So far in Germany, level 1 PSAs have predominantly been prepared. In the “BWR Safety Study“ (1993) and in the “Konvoi-PSA“ for GKN-II (2001), the processes within the plant after a core melt were also analysed (level 2 PSA). These two studies also analysed level 1 accidents that can occur when the plant is shut down, whilst most PSAs have looked only at the accidents that occur during service mode.

The tried and established methods for level 1 and 2 PSAs are able to provide reliable results and are described in detail in the report entitled “Methods for probabilistic safety analyses for nuclear power plants” (2006). There are, however, other methodical problems which can limit the completeness of the PSA and seriously impair the reliability of the results. Starting from the achieved standard, it is both wise and necessary to continue developing and rounding out the methods in order to further improve the reliability of the PSA. Knowledge-based and soft computing methods are also to be applied.

Further R&D work is required especially in the areas below:

CCF model

The simultaneous failure of redundant systems with highly-reliable components is dominated by “common cause failure” of redundant components. Due to suitable counter measures being taken, common cause failures are rare. This means that the probability of their occurrence cannot be fully based on operational experience. Hence, the development and verification of suitable models for evaluating common cause failures as part of a PSA are especially important. Methods to ensure the consistent evaluation of the transferability of observed common cause failure events to certain component groups must in particular be developed. Identified framework conditions of common cause failure events must be systematically recorded and weighted according to categorised attributes (e.g. prompt detection of CCF events) for the numerical evaluation.

Fire

Advancements are currently being made in this area, especially analyses of fire effects on system technology. Integrating fire-related component failures into existing error trees can be extremely difficult. Analyses are therefore currently being conducted to establish how the various fire-related failure potentials and failure sequences on safety-critical control technology components (interruption, signal change, surge insertion) and their various consequences can be incorporated into the PSA. Detailed analyses are currently being conducted in the OECD project PRISME for this purpose. One of the aims of experimental activities is to draw conclusions regarding damage to control components through flue gases and also fire propagation over linked areas through air ducts. The simultaneous presence of faulty signals in two redundancies of data logging with the potential consequence of faulty reactor protection actions, which is to be considered in the rare case of fire spread to two redundancy areas, continues to call for improved analysis methods. Methods for evaluating the relevance of fires in shut down mode must also be developed.

The codes used to calculate hydrogen combustion processes can also be used in part to calculate fires and extinguishing processes.

Reliability of active components and accident conditions[2]

Data on the reliability of active components under accident conditions are needed most of all for the probabilistic assessment of internal plant emergency measures and for assessing system availability at level 2 PSA. Since such data can be derived directly from operational experience to only a very limited extent, model developments – and in critical cases experimental analyses too – are also required in this area. A special method for evaluating the reliability of active components under accident conditions has, the function of which can favourably influence the accident sequence and dilute the accident consequences, has been developed over the last five years. The successful development of methods for evaluating the reliability of passive system functions, based for example on natural convection, and the application of the obtained results to a PSA should also be mentioned. Connectivity is seen in activities for improving and expanding these analysis methods for comparing active and passive systems.

Expanded modelling of uncertainties

The program system SUSA is currently being successfully used in numerous countries to conduct safety analyses. The conventional uncertainty analysis conducted using SUSA determines the influence of knowledge status uncertainties in parameters, model assumptions, phenomena and, in the application of numerical solution algorithms, on the result of computational models. For information on probabilistic safety analyses, however, a distinction must be drawn between uncertainty due to stochastic variability (aleatoric) and knowledge status uncertainties (epistemic).

Work is currently being carried out to analyse the effect of model and parameter uncertainties of an accident simulation on the result uncertainty of the PSA. To further improve reliability, methods for analysing the influence of uncertainties on the PSA results themselves and for precluding error sources must also be developed. A concept has been developed

[2] Reliability of passive components, see Chapter 4.1.1

for the approximate analysis of epistemic uncertainties of a probabilistic dynamic calculation in combination with a stochastic module. Methods for conducting such uncertainty and sensitivity analyses for dynamic PSA are to be developed on the basis of this concept. Methods for ensuring consistent and full consideration of epistemic uncertainties in reliability variable distributions are also required.

Furthermore, knowledge-based processes for information processing (Fuzzy Set Theory, artificial neuronal networks etc.) hold potential for resolving the vagueness of PSAs.

Dynamic PSA

For several years now, the focus has been on advancements in the methods for conducting dynamic probabilistic safety analyses which consider an incident's chronological development. The end of 2001 saw the development of the analysis tool MCDET (**M**onte **C**arlo **D**ynamic **E**vent **T**ree), which enables full consideration of the interaction between the dynamics of an incident's sequence and the stochastic influences as part of a PSA and delivers results in the form of dynamic incident trees that develop along a time axis. A crew module has also been developed. This models human actions as a dynamic process and, in combination with the MCDET methodology, integrally simulates the complex interactions between human actions, physical process variables, system components and stochastic events in a chronological sequence. This enables a more realistic and precise modelling of accident sequences and system behaviour than conventional PSA methods. Over the next few years, the MCDET method is to be advanced, validated and integrated into the deterministic computing code ATHLET.

4.4.3 Level 3 PSA

Level 3 of a PSA examines the damage caused by an accident-related release of radionuclides. Such analyses were conducted in the "German Risk Study of Nuclear Power Plants, Phase A" (1979) and in the "Risk-oriented Study on the SNR-300" (1982). No further research activities are planned for level 3.

4.4.4 Advancement of methods for special PSA applications (e.g. internal and external impacts, electrical auxiliary system)

As a result of the work on reconstructing electrical and instrumentation and control systems, as well as the "Forsmark Event" 6/2006, numerous simulation calculations for establishing the reliability of electrical installations and for determining current and voltage transients and load distribution have been carried out. There are no adequately verified simulation programs either for static or dynamic processes in the auxiliary system.

Process, control and electrical models need to be linked together to calculate transients in the electrical auxiliary system. This applies for both accident behaviour and for modification management. Further work must therefore include simulation code development and validation for the safety evaluation of modification measures in electrical auxiliary systems, taking into account accident scenarios. This could improve the quality and reproducibility of PSAs for initiating incidents in electrical auxiliary systems.

The fact that the frequency of damage states is very low increases the relative significance of very rare incidents, which have to date not been included in a PSA or included only in the form of estimates.

With regard to consideration of the rare incidents, which currently limit the reliability of PSAs, the key methodological problems can be summarised as follows:

- The spectrum of the analysed "initiating incidents" contains gaps, especially with respect to accidents that can be caused by overarching effects (such as fire, earthquakes, plane crashes, extreme weather, flooding, lightning strikes, and external voltage transients). It has not yet been possible to reliably assess the extent to which such events contribute to the risk of accident. Furthermore, PSA level 2 has so far considered only accidents that occur in service mode.
- In assessing the reliability of safety-critical systems, methodological gaps that must be closed or at least reduced still exist in a number of areas. In order to close these gaps, work is required on the probabilistic evaluation of the reliability of software-based control technology, on consideration for knowledge-based personnel actions in PSAs and on the probabilistic evaluation of organisational influences and also the effects of safety management on the reliability of personnel actions.
- To consider the effects of boron dilution and core melt phenomena in methods for simulating breakdown/accident sequences, conclusive experimental investigations

were conducted, e.g. in the OECD projects PKL-2, PKL-3 and MCCI-2. The results will be used as a basis for developing advanced evaluation methods.

Activities in the following subject areas are currently planned:

- Methods development for the probabilistic evaluation of the reliability of digital instrumentation and control technology, see chapter 4.4.5,
- Evaluation of operation and organisation influences on the reliability of components and human actions, see chapter 4.4.7,
- Advancements in methods for determining the frequency of large-scale radionuclide releases, even from highly unlikely triggering events caused by extreme high-water and weather situations and the failure of vessels with high energy content.
- Advancements in methods for determining leakage and break frequencies of pressurised components,
- Advancements in methods for evaluating the relevance of fire and methodical analyses on the relevance of overarching, external effects in shut-down conditions, see chapter 4.4.2,
- Advancements in the methods for the probabilistic evaluation of transients caused by surges or external voltage entries,
- Modification of methods for accident-related plane crashes to reflect state-of-the art of science and technology.

As well as accidents in full power condition, the PSA should also include part-power operation and power-up and power-down processes, as well as shut-down plant states.

4.4.5 Methods development I & C for control technology

Safety assessment of computer-based systems (software and hardware)

The use of computer-based control systems in nuclear power plants enables extensive information preparation, as well as improved self-monitoring and diagnostic functions and can thus help improve reactor safety.

We are currently seeing increased use of such systems in safety control technology, even in the highest safety category. Methods and tools for qualifying modern hardware/software systems must be adapted to keep pace with the fast technical progress in computer-based control technology systems and new issues for the safety evaluation of such systems, e.g. computer network technologies including security aspects, object-oriented programming paradigms, use of ASICs and FPGAs etc. must, in principle, be addressed. Fundamental methodical research and development work is required in this regard. In recent years, improved processes and methods for a more realistic safety evaluation have been provided by computer-supported control technology. A systematic and complete set of requirements for the development and safety proof of computer-supported control technology is being devised. By integrating various software solutions for furnishing proof, an integral, overarching proof procedure (prototype of a proof tool) has been devised.

Methods have also been developed that allow integrated tool environments for nuclear safety applications to be evaluated and qualified in accordance with the rules for state-of-the-art nuclear science and technology. Integrated tool environments that come under consideration for creating safety control technology functions in nuclear power plants have also been analysed, qualification methods developed and their application demonstrated through examples. Since the tools of integrated tool environments do not meet all the requirements for checking and verifying safety-relevant software, the potentials creating by using external tools have been included.

Work is currently being carried out on the development of tried and trusted methods that deliver standardised reliability and/or availability evaluations of increasingly complex control systems.

To develop error diagnosis and isolation procedures on the monitoring of safety-critical sensors, measuring chains and (gen.) instrumentation, besides conventional signal processing methods (e.g. frequency analysis, spectral analysis and statistical distribution analyses), research and development work on using so-called “soft computing” methods is currently being carried out, the aims being to

- increase the level of safety,
- provide early warning signals.

To allow a systematic analysis, error models of sensors and measuring chains are being developed using characteristic-based status monitoring. This will provide the basis for monitoring instrumentation status with regard to ageing problems.

Further activities are required to develop tried and trusted methods that deliver standardised reliability and/or availability evaluations of increasingly complex control systems.

Reliability indicators of computer-based systems

Current state-of-the-art technology involves furnishing proof of the reliability of software-based systems using qualitative methods, since the modelling of software failure mechanisms as a pre-requisite for determining quantitative reliability performance values is still successful only in exceptional cases.

There are also key differences between the architecture of software-based control technology and that of conventional control technology, which brings about new requirements for probabilistic reliability evaluation. The first solutions for quantifying software reliability are expected to be derived from the platform activities on determining the complexity of software-based control systems. Further variables on all system components are required for a PSA. Activities are therefore required on work for modelling the process of the occurrence and rectification of software errors and on defining limit values (barriers) for the implied failure of computer-based systems. As part of the OECD Halden Reactor project, research field "Man-Technology-Organisation (MTO), technologies have been developed for improving the reliability and safety of computer-supported digital control technology systems. Methods for the specification, testing, safety examination, risk analyses and for the qualification of hardware and software have been developed. Future activities will focus on:

- Requirement management for highly-reliable systems in the context of modernisation measures,
- Qualitative reliability evaluation of modern, highly-reliable management and control systems, including change management and the use of open-source software,
- Qualitative evaluation of error tolerance and error propagation based on the source code and formalised notation,
- Quantitative reliability evaluation of software systems and combination of same with qualitative evaluations.

Methods for the probabilistic reliability evaluation of software-based control technology are also required.

4.4.6 Human behaviour

Probabilistic approach (assessment of human reliability)

The relatively coarse methods of “Human Reliability Analysis” must be further developed for use in PSA for nuclear plants in order to reduce the uncertainty bands in existence today. It is necessary in particular to further validate the existing models and data that still come largely from the non-nuclear field, using experience garnered from the nuclear field. In addition to the methods available to date, which deal primarily with “control-based” actions, methods for evaluating “knowledge-based” actions, including execution errors, must also be developed and trialled. These activities ought to help reduce the limitations associated with executing a PSA and increase the reliability of results of probabilistic safety analyses.

A method has already been developed that enables knowledge-based human errors by operators with a detrimental effect on incident sequences to be analysed and quantitatively evaluated. This method records the key types of misjudgement and wrong decisions that can affect reliable actions, especially under stress. Methods must be devised for analysing and evaluating knowledge-based operator actions that can end incident sequences or reduce their effects.

As part of the OECD Halden Reactor project, research field “Man-Technology-Organisation” (MTO), human efficiencies and limits in monitoring and controlling complex systems have been analysed. The results from various test series for identifying human error and its likelihood of occurrence are applicable for probabilistic safety analyses. These activities are being continued.

Systematic approach (organisation, structures, implicit standards)

The incidents on Three Miles Island and in Chernobyl have contributed significantly to highlighting the attitudes and value systems of humans for safety and reliability of nuclear installations. Since then, the influence of safety-related attitudes and value systems on incident sequences (not only in nuclear technology) has been discussed under the buzzword “safety culture”. The IAEO defines safety culture as a part of the entire organisational culture of a nuclear power plant with origins that coincide with the theoretical approaches that describe above all behaviour, values, standards, attitudes and fundamental assumptions.

Whilst progress has now been made in the theoretical explanation of the concept, there is still a lack of practicable instruments for evaluating the quality of both the safety culture and suitable measures for its selective and sustained introduction and support. A first step for closing this gap was achieved with the development of a screening process, which, as part of an operator's self-assessment, provides an initial overview of the weaknesses and strengths of the safety culture in an organisational unit. It has also been demonstrated that implicit behaviour standards rank among the most important predictors of safety-oriented behaviour. However, methods for an extensive analysis of the safety culture status in nuclear plants that focus on safety-related conclusions of deep-set and concealed values, behaviour standards and fundamental assumptions have yet to be developed. The key issue is that consideration is taken of both operators and also the target group of high management, which plays an important role as decision-maker and promoter of an organisational culture.

A similar situation applies for the conceptualisation and design of institutional forms of safety management systems. This includes all management activities for planning, organisation, leadership and control, the aim of which is to achieve and maintain a high level of safety and a strong safety culture.

The age structure of nuclear plant personnel will result in the loss of high-qualified skilled workers in the foreseeable future. This poses the question of how the implied knowledge they have acquired over decades can be captured and safeguarded for their successors. Suitable knowledge management structures must be developed and institutionally implemented.

Selective, practical fundamental research is required in the stated subjects.

Quantitative evaluation of the contribution made by organisation and safety management to safety and reliability

Another aim of organisation and safety management is to maintain and enhance the reliability of human actions and plant safety at a high level. To date, however, it has been difficult to consider and demonstrate in a PSA, the quantitative contribution made by organisation and safety management. Quantitative factors include in particular the monitoring of activities by a second person. Other organisation and safety management factors must, by contrast, first be identified, precisely defined and quantified before they can be included in a PSA.

The following issues are important for definitive advancements and for promoting the maturity of the PSA:

- The development of a method for considering the influence of organisational factors,
- Determination of qualitative and quantitative data for influencing organisational factors and/or safety management on the reliability of personnel actions,
- Checking the applicability, the practical reference and the specific procedure using a case study which is to be based on a PSA conducted at the GRS for a German plant.

4.4.7 Technical systems for human support

Man-machine task distribution (degree of automation)

Various reportable incidents in German nuclear power plants caused in part by software-based systems and instruments (e.g. rod retraction blockage, crash of a fuel assembly cartridge) show that accidents and/or failure of software-based technology are frequently associated with human operation. To better address the problems posed by the use of this technology, research and development work on man and machine relationships in automated processes must be closely analysed.

Forecasting instruments (focus: test station, simulators)

The use of simulators to analyse emergency protection measures inside plants and also for accident diagnosis and prognosis is currently subject to restrictions with regard to

- the simulation and automation of procedures,
- the speed of simulation,
- the models used by the simulators.

Simulation instruments must therefore be advanced to enable a realistic review of the interplay between personnel actions and plant behaviour (e.g. procedures, AM measures). These instruments play a key role as the basis of operator and emergency support systems and improved reliability analyses (level 2 PSA).

Advancements should aim to further improve the reliability of analyses and expand the application spectrum. Deterministic codes are also being advanced in order to enable a simulation parallel to plant operation through an online adaptation of best estimate simulations in line with time-dependent measured data in the reactor system. The results of the simulations can be used to improve plant status information.

Advancement of diagnostics methods

Processes for the rapid and intelligent information preparation of multi-sensory variables are able, through the early detection of initiating malfunctions, enhance the safety of existing nuclear power plants under the current framework conditions in three areas:

- Mechanical ageing of components of the pressurised enclosure resulting from cyclical loads, irradiation and in some cases corrosion. Oscillation and structure-borne noise diagnostics have become extremely important for monitoring.

- Long-term loads in the core structure components, fuel elements in particular, alter significantly through increased burn-up, the use of mixed oxide fuels and modified material usage. Procedures for neutron flow and noise diagnosis, which are to be modified accordingly, are available for monitoring.

- Permanent loads from operating measuring chains cause defects on sensors, signal lines and electronic components, the frequency of which increases with operation. Measuring chain diagnostics are able to detect early failure indications. The transmission behaviour of measuring channels during operation can be determined and assessed using processes in the time, frequency and statistics range.

Further activities should focus on:

- Adaptation of internationally established procedures for specific objectives,

- Association of algorithms for enhancing reliability,

- Assignment and use of fast evaluation procedures for standard diagnoses.

This can result in the detection of generic weaknesses throughout the plant.

4.5 Knowledge transfer for the safety analysis of eastern reactors

Statement by the Evaluation Commission:

"The increase in safety of nuclear power stations in a Soviet design is one of the most urgent tasks to be managed in collaboration with central and eastern European countries. The ageing of reactor pressure vessels caused by neutron embrittlement is especially important in this regard. Support from the west and Germany in particular is vital due to this country's outstanding plant expertise.

Table 5 : Subject areas, individual topics, research institutions

Technical Field „Know-how Transfer regarding Safety Assessment of Eastern Reactors“

Chap.	Topic Areas	Specific Topic	Research Institution
4.5	Know-how transfer regarding safety assessment of Eastern reactors		
4.5.1	Code adaption and development	ATHLET	GRS
		ATHLET-CD	GRS
		COCOSYS, ASTEC	GRS
		Coupling of ATHLET with 3D neutron kinetic codes	FZD, GRS
		Reactor physics / Core data	FZD
		Development of interactive accident simulators	GRS
4.5.2	RPV ageing by neutron embrittlement	Exchange of test specimens (investigation of specimens from Greifswald)	FZD
		Reactor dosimetry	FZD

Table 5: Main Research Topics of the Research Institutions in the Technical Field „Know-how transfer regarding safety assessment of Eastern reactors “

The manpower prognosis for the transfer of knowledge to Eastern Europe relates primarily to the collaborations with Russia and the central and eastern European countries.

Experience with and results from German reactor safety research have been employed for many years as part of a long-term science-technology collaboration with Russia, the Ukraine and countries in central and eastern Europe. The results from this collaboration form the basis for safety analyses on VVER and RBMK reactors and for the information on the safety of these plants derived from this basis. Over the past five years, the collaboration has developed from the initial single-sided transfer of German expertise to the eastern partner countries into a research partnership with benefits for both sides. The collaboration is to be continued, the aim being to further contribute to increasing the safety of VVER and RMBK reactor plants.

As part of WTZ projects, criteria and combustion models for describing the combustion behaviour of hydrogen during potentially severe accidents breakdown have been developed and applied to VVER reactors. The criteria are being integrated into the 3D program GASFLOW. Corrosion analyses on steels in liquid lead or lead-bismuth have been continued and handling technologies and processes for surface treatment developed in order to reduce the susceptability to corrosion. The activities also included conducting and analysing experiments on oxidation kinetics and on the degradation behaviour of B_4C control rods. The activities for determination of the H_2 source term and for distribution and combustion of H_2 in the reactor building are well advanced. They must, however, like the analyses on the corrosion resistance of steel in contact with lead-bismuth be continued together with Russian research institutes.

4.5.1 Code adaption and development

The previous collaboration focussed on adapting and validating in more detail the computer programs developed in Germany to the design-specific conditions in Russian-designed reactors and also conducing generic accident analyses. During the period from 2002 to 2006, the partner countries were given, under the terms of the collaboration, the latest versions of the simulation and/or analysis programs advanced in Germany for evaluating nuclear safety and experts from the collaborating research institutes received support for integration and application. The experience garnered from applying the codes to VVER and RMBK plants was used to advance and validate the German programs. These activities are to be continued with the aim of making the code improvements achieved as part of German reactor safety research accessible to the partner countries, broadening the validation scope of the codes, especially for VVER, expanding the integration with codes and plant models of the partner countries and using the experience feedback from code application in Germany.

ATHLET and CFD-Codes

The thermo-hydraulic system code ATHLET continues to form the basis of joint further development and validation activities on accidents analyses in the cooling circuit of VVER and RBMK reactors. Particular importance is attributed to the best-estimate code ATHLET as a reference code for Russia’s proprietary development KORSAR. The German methodology for quantitatively determining the reliability of computer results is being increasingly adopted and applied by the partner countries; a considerable amount of experience has been fed back during the last five years. Work by Russian experts on approximating three-dimensional problem analyses through extremely fine local discretisation in the modelling of reactor pressure vessels with ATHLET is worth mentioning. It has enabled the enormous robustness of the numerical processes in the ATHLET code to be demonstrated. However, it also became clear that genuine 3D simulations using CFD codes to evaluate spatial distribution and mixing processes in large volumes and over the long term are still required. Validation calculations were carried out using CFD codes based on typical VVER mixing experiments and as part of the Phase II OECD Benchmark V1000CT.

German contributions are made in the form of reference calculations using ATHLET and involvement in developing and conducting counterpart tests on the test plants ISB and PSB. Instrumentation is provided for these tests. From 2002 – 2004, the work also received support in the form of the EU Project “**Imp**roved **A**ccident **M**anagement for VVERs“. To further optimise the experience exchange, an Internet-based user forum, which is also to serve as a model for similar forums regarding alternative code applications, has been set up for ATHLET.

ATHLET-CD

The focus remains on developing methods for simulating melt convection in the lower plenum and for calculating the load on and damage to the reactor pressure vessel. The intention is to transfer the developed methods to VVER conditions. The Russian side is to provide in particular material data from VVER and RBMK reactors at a given time.

To be able to make full use of ATHLET-CD to simulate accidents involving core destruction in VVER reactors, further model adaptions, especially on thermal transfer and heat irradiation, on material-specific melt-down behaviour and for melt cooling in the lower plenum and also on damage to the reactor pressure vessel, are required. There are still no specific plans in these fields for joint activities under the collaboration with Russia and the Ukraine or with central and European countries.

COCOSYS, ASTEC

Various central and eastern European countries use the COCOSYS program system for simulating accident sequences in the containment and/or confinement of VVER reactors. The DRASYS module for simulating dynamic condensation processes has already been modified to suit the special conditions prevailing in the pressure reduction systems of VVER-440 and RBMK reactors and partially validated. These activities are to be continued using new experimental results. Coupled versions of the most recent ATHLET/COCOSYS codes have been transferred to central and eastern European (MOE) countries and used for the initial integrated accident analyses.
Partners in Russia and in the CEEC have been incorporated into the pan-European work on validating the German/French integral code ASTEC for source term evaluation in core melt accidents, which is being supported by the EU in the 6th research framework program. This work is to be continued and will contribute to the universal applicability of the code.

Coupling ATHLET with 3D neutron kinetic codes

A more precise analysis of various reactivity initiated accident requires cooling circuit thermo-hydraulics to be linked to 3D neutron kinetics. ATHLET is also linked to various kinetics programs for the hexagonal fuel element geometry of VVER reactors (e.g. DYN3D, BIPR8). Useful experience feedback is expected from the generic application of recently linked versions, especially the validation of ATHLET/BIPR8 using measured transients in Russian VVER 1000 plants.

In parallel, the code complex DYN3D/RELAP is being validated with operational transients and by comparison calculations using DYN3D/ATHLET. Analyses are also being conducted on implied accident scenarios on VVER under load with MOX and/or CEOMET (ceramicmetal) fuel. The assumptions for the thermo-hydraulic framework conditions of these transient analyses are to be supported by mixing tests and by calculation of boron concentration profiles with dilution transients.

ATHLET is linked to the 3D kinetics program QUABOX/CUBBOX for the purposes of analysing reactivity initiated accident in RBMK reactors with quadratic fuel assembly geometry. This combination is being used successfully for safety evaluations, in some cases with EU support.

Retrofit measures and also new fuel assembly designs and new control rod constructions with RMBK will continue to require model adaptations to enable safety-critical issues such as reactivity behaviour and efficiency of the second shut-down system to be analysed.

This involves expanding and adapting the cross-section library by core loading with new fuel assembly with increased enrichment, the modelling of new rod constructions and the plant-specific adaptation of the complex reactor control and protection system and the modelling of the control rod cooling circuit. The used models must be verified through comparison with plant measurements and the results obtained from Russian codes.

Reactor physics / core data

To generate cross sections in VVER-1000 fuel elements with increased enrichment, the burn-up program KENOREST is to be expanded to include VVER conditions. Comparison calculations on equivalent transients with different cross section libraries to date have delivered extremely differing results. The reasons behind these differences must be explained.

Cross section libraries for MOX and CERMET fuel elements have been generated and connected to the code DYN3D. DYN3D is then used to calculate core designs for various MOX loading.

Development of interactive accident simulators

Analysis simulators with extensive, plant-specific modelling in control and system technology, graphic user interface and interactive control have also become established for VVER reactors. The simulator for the VVER-1000/W-320 has been developed on the basis of the simulation software ATLAS. Work has also started on developing an analysis simulator for a VVER-440/W230 plant, which is required in particular for examining the emergency protection measures inside plants. These activities are to be continued and concluded.

4.5.2 Ageing of reactor pressure vessels through neutron embrittlement

Evaluation of material specimens from the reactor pressure vessels in the Greifswald nuclear power plant

The recovery and analysis of specimens from the reactor pressure vessels in Greifswald plants is extremely important for assessing irradiation-induced material embrittlement of VVER in Eastern Europe. Data for irradiation load and operation history has already been compiled and evaluated in a WTZ project.

This work is to be continued using high-resolution calculations of neutron fluence at the material sampling locations and also analyses with retrospective neutron dosimetry. The material analyses are primarily geared towards determining embrittlement break transition temperature.

Reactor dosimetry

The calculation methods for determining neutron and gamma irradiation loads of the reactor pressure vessel at VVER-1000 have been modified for fluence methods. This enabled Monte-Carlo calculations to be used to assess neutron fluences accumulated over the entire operation period, even for critical positions of the construction.

4.6 Innovative concepts

Statement by the Evaluation Commission:

“Continued use should be made of the expertise available within Germany with a view to further enhancing nuclear safety standards“.

Table 6 : Subject areas, individual topics, research institutions

Fig 9 : Manpower forecast for the subject areas

Technical Field „Innovative Concepts“

Chap.	Topic Areas	Specific Topic	Research Institution
4.6	Innovative concepts		
4.6.1	Innovative international reactor concepts	Significance for strategies and safety	FZK, RUB
		Generation of electricity and thermal process energy	FZD, FZJ, FZK, RUB
		Developments on the six GIF concepts	FZD, FZK, RUB
4.6.2	Avoidance and reduction of long term radioactive waste	Extended fuel utilization in present reactors	AREVA NP, FZJ, FZK
		Transmutation	FZD, FZJ, FZK, RUB

Table 6: Main Research Topics of the Research Institutions in the Technical Field „Innovative Concepts“

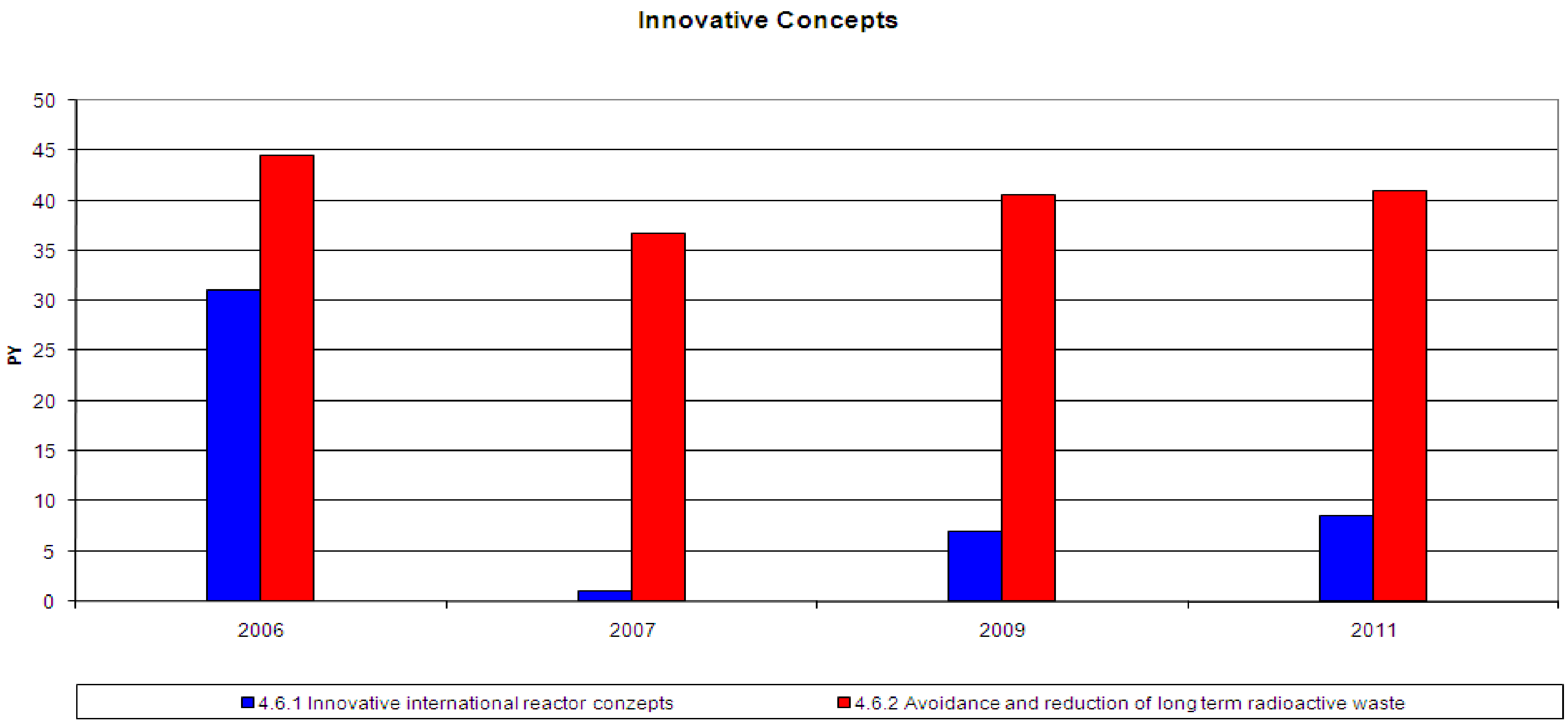

Fig. 9: Personnel Prognosis for Topic Areas in the Field of "Innovative Concepts"

A host of research institutes worldwide are working on innovative safety concepts. Verification is required as to whether these concepts contain elements that can be used to improve the safety of the reactors in operation in Germany. The transmutation of persistent radionuclides could also contribute towards improving the safety of the radioactive waste repositories.

The German state-funded activities on these issues comprise only the involvement in the advanced development of safety requirements and safety technologies. Incorporated into these activities are new findings from operational experience and from ongoing research work, as well as the development of basic technologies for minimising radioactive waste.

German nuclear research and development also have considerable expertise in these areas. This is being contributed to international networks (IAEA, OECD, EU), with a view to improving the future safety of nuclear plants. Proof of safety for nuclear plants must always be furnished in accordance with the international status of science and technology.

Scientists in German research institutes are to continue their involvement in international developments in order to verify their applicability for advanced developments of safety technology German plants and to contribute the existing knowledge into the international activities.

For junior scientists, activities on innovative concepts are especially interesting because they comprise scientifically challenging subjects, are conducted in international partnerships and are integrated into international projects. These activities thus contribute to the urgent need to sustain and advance the safety expertise for nuclear plants in Germany.

4.6.1 Innovative international reactor concepts

Strategic and importance to safety technology

International efforts are being ramped up as part of the Generation IV activities in the GNEP initiative in order to develop new nuclear reactor concepts. The aims are to improve safety, cost-efficiency and achieve non-proliferation. New safety systems such as passive safety components are also to be developed, tested and introduced. Some of these developments can also be applied to existing plants. Furthermore, an involvement in these developments contributes towards sustaining and advancing the expertise of German researchers.

The same also applies for the fuel cycle, including the repository system.

Generation of power and process heat

The activities on high-temperature reactors aim to maintain expertise with regard to the observation, evaluation and influence on the safety philosophy for modular high-temperature reactor projects currently being conducted in other countries. To that end, research is required into the safety-related effects of using nuclear process heat for hydrogen generation or producing synthetic hydrocarbons.

Developments on the six GIF concepts

Works on alternative concepts with innovative reactors was incorporated into the 5th and 6th EU Framework Program and IAEA activities and are the object of activities of the Generation IV International Forum (GIF). The following concepts are being considered:

- Gas Cooled Fast Reactor (GFR)
- Sodium Cooled Fast Reactor (SFR)
- Lead Cooled Fast Reactor (LFR)
- Very High Temperature Reactor (VHTR)
- Super Critical Water Reactor (SCWR)

- Molten Salt Reactor (MSR)

Research work is also emerging in the following areas:

- Reactor physics:
  microscopic and macroscopic cross sections, reactivity coefficients, new fuel concepts, flow behaviour for ballast fuels

- Thermo-fluid dynamics:
  Property changes in the cooling circuit, transport of corrosion and erosion products, consideration of heat sources in the coolant

- Material research:
  Corrosion, erosion, high-temperature properties, stability in the radiation field, test procedures.

4.6.2 Avoidance and reduction of persistent radioactive waste

Analyses have been conducted as part of international partnerships in order to minimise radioactive waste caused by the operation of nuclear reactors through the use of special fuels.

In the nuclear reactors operated today, plutonium can be decomposed by using thorium-based or inert matrix-based plutonium mixed oxide fuels significantly faster than is currently possible in conventional uranium-based reactor cores. Safety-critical analyses for pressurized water reactors are being conducted in this regard. These activities aim to increase the conversion rate of plutonium while retaining or even improving safety-critical characteristics of the reactor. Alternative fuel strategies, such as the combustion of minor actinides in light water reactors for reducing persistent radio-toxicity, now ought to be examined in detail in international partnerships with regard to their conversion efficiency and also their effects on the fuel cycle.

Improved fuel utilization in current reactors

A key element of waste minimisation is the increase in final burn-up. Increased burn-up caused by higher enrichment reduces the amount of actinides produced relative to the energy generated, especially thermally fissile plutonium since this already contributes significantly to energy generation with uranium fuel elements. In the light water reactors currently in use, this measure is essentially limited by the production, transport and handling concessions to an enrichment of 5% of U-235.

In this respect, current fuel elements exhibit a high development status and permit safe operation of the elements, even with high thermal utilisation and up to high burn-offs. The further optimisation of fuel element design and core loading aims to improve fuel utilisation.

Transmutation

The goals for the activities on transmutation include an evaluation of various concepts with regard to technological feasibility, safety-critical characteristics and achievable transmutation rates. Scenario analyses of the fuel cycle and physical neutron analyses to evaluate strategies for reducing radio-toxicity of radioactive waste through transmutation are being conducted and safety-critical issues on the transient and accident behaviour of transmutation plants are being examined in detail.

Concepts that demonstrate a potential alternative for the transmutation of plutonium and persistent actinides are to be pursued in international projects within the European Union and the International Science and Technology Centre (ISTC).

5 International collaboration

German research work conducted as part of international collaboration is included in the technical chapters of this report. The joint research work with scientists from Central and Eastern Europe and Russia under bilateral agreements of the Federal Government and/or German research institutes on the technical scientific partnership and as part of the ISTC program.

In recent years, both the German and international research environment was highly diluted in terms of available test plants. In light of the concern that a dearth of important test facilities could lead to deficits among the scientific fundamentals for the safety evaluation of nuclear power plants, German involvement in research projects of OECD NEA is paramount.

Common to all OECD projects is the interest of all participating countries in finding a solution to an equally recognised safety-critical issue and the efficient, joint use of the internationally available experimental resources by jointly financing the essential activities. Experts from participating member countries are in detailed discussions on the planning of these projects, are selectively influencing their execution and interpreting the results with regard to the safety-critical significance in the OECD task forces concerned.

Usually, OECD projects are agreed by international contract in a standardised form. This agreement contains the technical proposal for the working program, the financial obligations of the participating countries and the rules for executing the project and accessing the generated results. Up to approx. 50% of the project costs are usually carried by the country in which the test facility is being operated. The remaining costs are split between the participating OECD countries, the amounts being based on the corresponding gross national product quotas. In most projects, Germany’s financial participation takes the form of research activities sponsored within the project-funded reactor safety research of the BMWi.

Germany participates in all ongoing projects of OECD NEA and plans to continue doing so for all upcoming projects.

On behalf of and with the technical backing of OECD/CSNI (Organisation for Economic Co-operation and Development / Committee on the Safety of Nuclear Installations), so-called International Standard Problems (ISP) have been worked on for several years. An ISP is a direct comparison of analytically (calculated) and experimentally (measured or balanced) data. The experiments on which the ISP is based relate to various areas of reactor safety research. As far as test set-up and execution and also instrumentation are concerned, they usually reflect the latest international status of science and technology.

Computer programs that are employed in research and also by the technical experts working on behalf of regulatory authorities and/or the industry for the safety assessment, and in the design of nuclear power plants ought to be used to determine analytical data. The calculations are to be “best estimate” and preferably with no knowledge of the test results.

International Standard Problems are therefore suitable for validation (comparison between calculation result and experiment) and verification (comparison between calculation results achieved with different models) and also for a more in-depth discussion on the phenomenological understanding and the implementation in relevant models and also for identifying existing deficits. For these reasons, many earlier ISPs still play a crucial part in the basic validation of leading computer programs.

In the past, Germany has contributed significantly to IPSs, be that as a result of its orientation, i.e. providing experiments and technical implementation and documentation of comparative activities or through the participation of several German institutes. The need to retain this fundamental stance on ISPs in the future is emphasised especially in light of the now severely restricted experimental capabilities in Germany.

Two programs are being developed as part of Co-ordinated Research Programs (CRPs) of IAEO. The first program focuses on R&D activities for proving the technological feasibility of the transmutation of persistent radioactive waste and its effects on reducing radio-toxicity. Methods for describing the dynamics of transmutation systems are being developed and the future research requirement defined. The second program looks at the safety issues and methods development for transmuting plutonium in a Russian BN-600 reactor.

5.1 OECD projects completed between 2002 and 2006

HALDEN

Participants: 13 countries
Executor agency: Institute for Energy Technology, Kjeller and HALDEN, Norway
Duration: 01.01.2003 – 31.12.2005
Research topics: Fuel element research, man-machine interaction

MASCA (Material Scaling)

Participants: 17 countries
Executor agency: Russian Research Centre, Kurchatov Institute, Moscow
Duration: 01.07.2000 – 30.06.2003

Research topic: Phenomenological analyses on the behaviour of molten corium in reactor pressure vessels during severe accidents with differing material compositions and degrees of oxidation, as well as the ability of water and molten metal to penetrate covering crusts. Determination of material properties of corium and its components in the high temperature region.

MASCA II (Material Scaling)

Participants: 15 countries
Executor agency: Russian Research Centre, Kurchatov Institute, Moscow
Duration: 01.07.2003 – 30.06.2006

Research topic: Phenomenological analyses on the behaviour of molten corium in reactor pressure vessels during severe accidents with differing material compositions and degrees of oxidation, as well as the ability of water and molten metal to penetrate covering crusts. Determination of material properties of corium and its components in the high temperature region.

SETH

Participants: 15 countries
Executor agencies: Framatome ANP (Germany) for PKL, Paul Scherrer Institute (Switzerland) for PANDA
Duration: 01.04.2001 – 30.09.2006

Research topic: Experimental investigations in tests plants PKL (Germany) and PANDA (Switzerland) on the primary cooling cycle and on the containment for the prevention or management of accidents.

MCCI (Melt Coolability – Concrete Interaction)

Participants: 13 countries
Executor agency: Argonne National Laboratory on behalf of US NRC
Duration: 01.01.2002 – 31.12.2005

Research topic: Experimental investigations on the mechanisms for the cooling efficiency of melt in the containment and for the long-term 2D melt-concrete interaction.

5.2 Current and pending OECD projects

CABRI-WL

Participants: currently 13 countries
Executor agency: Institut de Radioprotection et de Sûreté Nucléaire (IRSN)
Duration: 2001 - 2008

Research topic: Conducting tests in the CABRI research reactor for determining the behaviour of high burn-up and MOX fuel rods under reactor accident conditions.

HALDEN

Participants: currently 18 countries
Executor agency: Institute for Energy Technology, Kjeller and HALDEN, Norway
Duration: 01.01.2006 – 31.12.2008

Research topics: The project goals are geared towards sustaining the safety of existing reactors and improving their operating behaviour. The focal points are: Man-machine interaction and fuel and material research as well as fuel irradiation activities.

MCCI-2

Participants: currently 13 countries
Executor agency: Argonne National Laboratory on behalf of US NRC
Duration: 01.04.2006 - 30.06.2009 (extended to 31.12.2009)

Research topic: Experimental investigations on the mechanisms for the cooling efficiency of melt in the containment and for the long-term 2D melt-concrete interaction, in order to provide important data for code developments.

PKL

Participants: 14 countries
Executor agency: AREVA NP GmbH, Erlangen, Germany
Duration: 01.06.2007 - 31.12.2009

Research topic: Experimental analyses on boron in the event of accidents with small leakage in the primary circuit and failure of the residual heat removal in the ¾ loop operation with oth closed and open primary circuit.

PSB-VVER

Participants: currently 7 countries
Executor agency: Electrogorsk Research and Engineering Center (EREC),
Russian Federation
Duration: 01.02.2003 - 30.06.2007

Research topic: Experimental investigations at the PSB test facility as a data basis for computer programs on the thermo-hydraulic plant behaviour of VVER-1000 reactors in the event of loss of coolant accidents and heat removal from the reactor core through natural circulation.

SETH-2

Participants: currently 9 countries
Executor agencies: Commissariat à l'Energie Atomique (CEA), France
Paul Scherrer Institute (PSI), Switzerland
Duration: 01.03.2007 - 31.12.2010

Research topic: Experimental investigations in the test facilitiesMISTRA (France) and PANDA (Switzerland) for the further development and validation of computer programs for modelling thermo-hydraulic processes and also the mixing behaviour of the atmosphere in containments in the event of severe accidents.

SCIP

Participants: currently 11 countries
Executor agency: Studsvik, Sweden
Duration: 01.07.2004 - 30.06.2009

Research topic: Determination of fuel rod cladding behaviour under load conditions of power transients as a consequence of plant states under normal operation and accidents. This involves analysis of various designs and productions of cladding pipes as well as their performance history.

ROSA / LSTF

Participants: currently 11 countries
Executor agency: Japan Atomic Energy Agency (JAEA), Japan
Duration: 01.04.2005 - 31.03.2009

Research topic: ROSA/LSTF is the world's larges integral test plant and has two symmetrical loops. The fundamental aim of the activities is to broaden existing knowledge of thermal-hydraulic safety aspects as a data basis for modern, multi-dimensional computing codes.

OECD-THAI

Participants: Expected 10 countries

Executor agency: Becker Technologies GmbH, Germany
Duration: 01.07.2007 - 30.06.2010

Research topic: Experimental investigations at the THAI test facility on accident-related hydrogen and fission product behaviour in the containment, in order to provide important data for code development and validation.

SERENA

Participants: Expected 10 countries
Executor agencies: Commissariat à l'Energie Atomique (CEA), France
Korea Atomic Energy Research Institute (KAERI), Rep. Korea
Duration: 01.07.2007 - 30.06.2011

Research topic: Mechanisms of the melt-coolant interaction and its influence on the energy release with steam explosions outside of the reactor pressure vessel at test facilities KROTOS (France) and TROI (Rep. Korea).

BIP

Participants: Expected 15 countries

Executor agenciy: Atomic Energy of Canada Limited (AECL), Canada
Duration: 01.07.2007 - 30.06.2010

Research topic: Conducting research at the RTF research facility, in order to improve the modelling of the iodine behaviour in the containment following severe accidents.

Abbreviations

ADS	Accelerator Driven System
AM	Accident Management
AMES	Ageing Materials Evaluation and Studies
ARTIST	Aerosol Trapping in Steam Generator
ASTAR	Advanced 3D Two-Phase Flow Simulation Tool for Application to Reactor Safety
ASTEC	Accident Source Term Evaluation Code
ATHLET	Analysis of the thermo-hydraulics of leaks and transients
ATHLET-CD	Analysis of the thermo-hydraulics of leaks and transients – core degradation
ATWS	Anticipated Transient Without Scram
BAM	Federal Institute for Material Research and Testing, Berlin
BGR	Federal Institute for Geosciences and Raw Materials
BMBF	Federal Ministry for Education and Research
BMC	Battelle Modell Containment
BMF	Federal Ministry for Finance
BMU	Federal Ministry for the Environment, Natural Protection and Reactor Safety
BMWA	Federal Ministry of Economic Affairs and Employment
BMWi	Federal Ministry for Economic Affairs and Technology
BWR	Boiling water reactor
CABRI	French Test Reactor of the CEA
CAPRA	Consommation Accrue de Plutonium dans les réacteurs Rapides
CATHARE	Französischer Thermohydraulikcode (CEA)
CEA	Commissariat à l'Énergie Atomique, French Atomic Energy Authority
CEEC	Central and Eastern European Countries
CFD	Computational Fluid Dynamics
COBRA	Model for thermo-hydraulics within a fuel element

COCOSYS	Containment-Code-System
COMET	Experiments for analysing core melt accidents
CORESA	Corium on Refractory and Sacrificial Materials
CSNI	Committee on the Safety of Nuclear Installations
CUBBOX	Approximation by Cubic Local Polynomials
DCH	Direct Containment Heating
DEMONA	Demonstration of nuclear aerosol behaviour
DSA	Dynamic Strain Ageing
DWD	German Weather Service
DWR	Compressed water reactor
DYN3D	3-dimensional reactor dynamic program
EOL	End of Life
FLUBOX	2D/3D Fluid model for describing dual-phase flows
FZD	Dresden-Rossendorf Research Centre
FZJ	Jülich Research Centre
FZK	Karlsruhe Research Centre
FZR	Rossendorf Research Centre
GASFLOW	3 D CFD-Code
GIF	Generation IV International Forum
GNEP	Global Nuclear Energy Partnership
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH
GVA	Jointly caused failures
HTR	High temperature reactor
IAEA	International Atomic Energy Agency
IBMB	Institute for construction materials, solid structures and fire safety, Braunschweig
I&C	Instrumentation and Control (Control Technology)
ICARE	French core melt system code

IKE	Institute for nuclear energy and energy systems at the University of Stuttgart
IPM	Institute for Process Technology, Process Automation and Measuring Systems, Zittau/Görlitz Technical College
IPSN (IRSN)	Institute de Radioprotection et de Sûreté Nucléaire (France)
ISB	Russian Thermo-hydraulic plant, minor leaks for VVER-440
ISP	International Standard Problem
ISTC	International Science & Technology Center
ISTec	Institut für Sicherheitstechnologie (ISTec) GmbH
IWM	Fraunhofer Institute for the Mechanics of Materials
IZFP	Fraunhofer Institute for Non-destructive Testing
KAEVER	Core melt aerosol behaviour
KATS	Karlsruher Thermit melts (experiments on the spread and cooling of core melts)
KESS	Core melt system code
KTA	Committee for Nuclear Technology
KV	Competence Pool
LAVA	Core melt spread program
LOCA	Loss of Coolant Accident
LP	Lumped Parameter (computing codes)
LWR	Light water reactor
MARCH	Thermo-hydraulic module of the Source Term Code Package (American Code)
MASCA	Material Scaling
MC3D	Model for the melt - coolant interaction (CEA Grenoble)
MEGAPIE	Mega watt Pilot Experiment (Pilot Spallationstarget)
MELCOR	American Integral Code For Analysing Serious Breakdowns (NRC)
MJ	Man years
MOX	Mixed-oxide core fuel (uranium/plutonium; or thorium/plutonium)
MPA-KA	Materials Testing and Research Institute of the University of Karlsruhe
MPA-S	State Materials Testing Institute of the University of Stuttgart

MSR	Molten Salt Reactor
NEA	Nuclear Energy Agency (OECD)
OECD	Organisation for Economic Co-operation and Development
PANBOX	Neutron-kinetic computer program (Framatome ANP)
PANDA	Passive reheat dissipation and pressure reduction test plant (Switzerland)
PHEBUS	Plant for nuclear analyses, Cadarache
PKL	Primary circuit loop (thermo-hydraulic 4-loop test plant)
PDO	Probability of Detection
PSA	Probabilistic safety analysis
PSB	Russian Thermo-hydraulic plant for VVER-1000
PT R	Project sponsor for reactor safety research (GRS)
PT WTE	Project sponsor for water technology and disposal
PTS	Pressurized Thermo Shock
QUABOX	Approximation by Quadratic Local Polynomials
QUENCH	Experiments for quenching rod bundles
RALOC	Radiolysis and Locale Concentration (Containment-Code)
RBMK	Graphite-moderated pressure pipe reactors in Soviet design
RDB	Reactor pressure vessel
RECOM	Code for the Simulation of Radionuclide Resuspension at Bubbling Water Pool Surfaces
RELAP	American breakdown analysis computer program
RIA	Reactivity Initiated Accident
ROCOM	Test plant for mixing tests
RODOS/RESY	Realtime Online Decision Support System
RUB	Ruhr University of Bochum
SCDAP	Severe Core Damage Analysis Package, Americal Core Melt System Code (NRC)
SUSA	System for uncertainty and sensitivity analysis
ThAI	Thermo-hydraulics, aerosol, iodine (test facility)

TOPFLOW	Transient Two Phase Flow Test Facility
TU-DD	Technical University of Dresden
TUM	Technical University of Munich
TU-MGD	Technical University of Magdeburg
Uni-KA	University of Karlsruhe
UPTF	Upper Plenum Test Facility
UPTF-TRAM	UPTF-Transient and Accident Management
US NRC	Nuclear Regulatory Commission, Washington DC (USA)
VANAM	Tests of the decomposition behaviour of nuclear aerosols in a multi-area containment geometry
VVER	Compressed water reactor in Soviet design
WTZ	Scientific Technical Cooperation (with Central and Eastern European States)

Reference documents

/EVK 00/	Nuclear Safety and Repository Research in Germany, Report by the Task Force appointed by the Federal Ministry for Economic Affairs and Technology (BMWi) (Evaluation Commission); January 21, 2000
/AtG 02/	Law on the peaceful use of nuclear energy and protection against the risks posed by same (Atomic Energy Law - AtG) dated December 23, 1959, New Version dated July 15, 1985, last revision by law of April 22, 2002 (BGBl. I 2002, No. 26)

**Gesellschaft für Anlagen-
und Reaktorsicherheit
(GRS) mbH**

Project management agency
of the BMWi
for reactor safety research PT R

Schwertnergasse 1
50667 Köln
Phone (02 21) 20 68 - 0
Fax (02 21) 20 68 - 6 29

www.grs.de/ptr/