EXECUTIVE SUMMARY

Background and scope of the Workshop:

Severe Accident Management Guidelines (SAMGs) increase focus on containment integrity after some progression in the course of a severe accident. This change in priorities is made according to criteria that vary depending on reactor type and specific procedures. Non challenging the priority given to containment integrity, trying to cool the degrading fuel and/or the corium within the Reactor Pressure Vessel (RPV) is a way to slowdown or stop the progression of an accident. This may also delay or avoid the Reactor Pressure Vessel (RPV) rupture that may subsequently endanger the containment integrity by dynamic loads (Direct Containment Heating DCH, ex-vessel Steam Explosion SE) and/or static loads (Corium-Concrete Interaction CCI).

This issue called "In-vessel Coolability" has been identified as most important by both the CSNI/WGAMA (see WGAMA work-plan for Severe Accidents NEA/SEN/SIN/AMA/2008(3)) and EC-SARNET (see Severe Accident Research Priorities final report SARNET-SARP-D96).

Once a water source has been recovered, different accident management strategies can be used: send water into the core and/or cool the RPV externally. It should be noticed that, depending on the amount of water available, these strategies might conflict with other uses of water such as for instance activating spray systems in the containment.

Concerning the first strategy, sending water into a degrading core is not straightforward as:

- The efficiency of reflooding for significantly delaying or stopping core degradation is not demonstrated for all situations;
- It may result in high hydrogen production rates that may threaten the containment integrity by dynamic loadings (hydrogen combustion);
- It may also result in a pressure peak that may endanger the containment integrity by DCH if the RPV vessel has been previously weakened by corium slumps;

Given these adverse considerations, some Severe Accident Management Guidelines consider cautions in how and when to send water in the core. In addition, reflooding models used for their evaluation suffer from a lack of validation that makes it difficult to assess the relevance of different accident management strategies.

Trying to cool the RPV externally to assure in-vessel retention is also not straightforward as:

- This accident management measure was not taken into account in the original design of existing reactors;
- The probability of success strongly depends on the reactor detailed specific features such as the reactor pit geometry, flooding rate, the type of heat insulation, the connections to the dome…and it generally decreases with the reactor power;
- If the external cooling turns out to be non efficient, occurrence of an ex-vessel steam explosion cannot be ruled out in case of lower head failure and this is still considered as a non resolved issue.

As for in-vessel reflooding, the models used for evaluation of accident management measures suffer from a lack of validation.

Given this background, the objectives of the workshop were:

- To exchange information on different Severe Accident Management strategies used or contemplated for the in-vessel coolability issue;
- To review recent, ongoing and planned experimental programmes on reflooding;
- To review models used for reflooding in severe accident calculation tools, either simplified or sophisticated;
- To exchange information on the treatment of reflooding in different safety studies such as Probabilistic Safety Assessment;
- To provide recommendations for future work, as necessary.

According to these objectives, the workshop was organised in four technical sessions, followed by a final one devoted to the presentations of sessions' chairs conclusions and a general discussion:

- General studies;
- Experimental work;
- Phenomenological and modelling work;
- Specific reactor studies.

Twenty-two papers were presented in the technical sessions, authors being members of research organisations, industry and technical safety organisations. The workshop enjoyed the participation of 66 people coming from Belgium, Bulgaria, Canada, Czech Republic, Finland, France, Germany, Hungary, Italy, Korea, Slovak Republic, Spain, Sweden, Switzerland, United Kingdom, United States and OECD/NEA Secretariat.

Short summary of sessions

Four papers were presented in the first sessions (general studies):

- One paper by the *Karlsruhe Institute of Technology* (KIT, formerly FZK) synthesising existing knowledge on degraded core reflood and identifying the main influential parameters;
- Two papers by the *Institut de Radioprotection et de Sûreté Nucléaire* (IRSN) and the *Gesellschaft für Anlagen und Reaktor Sicherheit* (GRS) on results and main lessons learnt from PSA level 2 studies for French and German reactors;
- One paper by the *Commissariat à l'Energie Atomique* (CEA) presenting a new tool to be used by the French utility EDF for PSA level 2 studies.

Five papers were presented in the second session (experimental work):

- One paper by KIT on QUENCH experiments dealing with reflooding of bundles;
- Two papers by the *University of Stuttgart* (IKE) and the IRSN on debris bed coolability experiments (respectively DEBRIS and PEARL);
- One paper by KIT on molten pools coolability (LIVE experiments);
- One paper by CEA on Reactor Pressure Vessel external cooling (CNU experiments).

Eight papers were presented in the fourth session (phenomenological and modelling work):

- One paper by CEA giving an overview of melt dynamics and treating the strong coupling between material property effects and thermal-hydraulics;
- Three papers by IRSN and IKE on the modelling of reflooding for a severely damaged core including debris cooling;
- Two papers by the *Ruhr University of Bochum* (RUB) and IRSN on the simulation of two QUENCH experiments conducted under conditions adverse to quenching;
- One paper by GRS on the simulation of TMI-2 accident by the ATHLET-CD code;
- One paper on the results of the OECD benchmark exercise on an alternative TMI-2 scenario, the authors being the participants to the benchmark.

Five papers were presented in the fourth session (specific reactor studies):

- Two papers by *Inzinierska Vypoctova Spolocnoast* (IVS) and *Paks Nuclear Power Plant* on Reactor Pressure Vessel external cooling for VVER-440/213 showing good prospects;
- One paper by *AMEC* and *British Energy* about the optimal use of water after core degradation has started (case study for Sizewell B NPP);
- Two papers by the *Royal Institute of Technology* (RIT) and *AREVA NP* about Reactor Pressure Vessel external cooling for BWRs.

General conclusions

As a result of discussions, there is a general agreement on the importance of the In-vessel coolability issue as described above in the “background” section.
The possibility of stopping and/or delaying the progression of a core melt accident by the use of a recovered water source or by taking benefit of specific engineered systems is taken into account in a number of PSA studies.

The likelihood to stop the progression of a core melt-down accident by water injection is generally considered as high in the early phase of core degradation and depends on reactor specific features, nevertheless even in later sequences, e.g. during the relocation in the lower head, cooling still can be achieved but depends on reactor specific features and the accident scenario..

Ongoing, starting and planned experimental programmes address the coolability issues in the different relevant configurations, i.e. reflooding of bundles, debris beds, molten pools and Reactor Pressure Vessel external cooling.

There is still a difficulty with present models to predict reliably if reflooding during the early core degradation would or not trigger a cladding oxidation runaway. Whether this is due to deficiencies in thermal-hydraulics description or problems for taking into account the oxidation of melts is a matter of discussion.

The code developments are promisingly directed towards a more mechanistic approach using a porous medium modelling able to treat the different configurations of a degraded core. Secondly, the models to describe adequately the relocation of parts of the molten core to the lower head and the debris bed formation still need further development and qualification. Their validation is expected against the results of ongoing experimental programmes.

The transposition of results to the reactor scale where multi-dimensional effects are expected needs to be evaluated, all the more as larger scale experiments are probably not feasible.

Another way to cope with the uncertainties is to implement specific engineered features and/or management procedures to act on influential parameters such as an increase of the available water flow rate. Specific examples were given during the workshop such as: (i) the good prospects for external RPV cooling for VVER-440/213 reactors; (ii) the use of spray found efficient for Sizewell B PWR; (iii) the potential of Control Rods Guide Tubes flow to cool molten pools in BWRs.

More generally, it was concluded that the present efforts to solve the In-vessel Coolability issue are well-oriented.

Recommendations

The recommendations are directed towards two goals:

- Monitor the progress made;
- Make a status in a few years from now.

Monitoring should be continued both in the frame of OECD/CSNI through the WGAMA and by EC-SARNET through its Work package n° 5. As the feedback experience might not been shared enough, a specific recommendation is given:

R1 Feedback experience from the analysis of safety cases of Nuclear Power Plants having, planning and/or contemplating the implementation of specific engineered features to solve the in-vessel coolability issue would be of great benefit. CSNI should encourage the members of its working groups to report on this item.

It is expected that ongoing experimental programmes and analytical efforts will help making progress in the coming years. For that purpose, two recommendations are made:

R2 A state-of-the-art report on the in-vessel coolability issue should be prepared as foreseen in the WGAMA work-plan. The time frame for this action should be proposed by the WGAMA according to its monitoring of progress made and in accord with EC-SARNET.

R3 The WGAMA should discuss, together with EC-SARNET, the suggestion of organising a follow-up workshop on in-vessel coolability.

Annex

SESSION SUMMARIES

Session 1

General Studies

Session Chairperson:

B. Clément (IRSN)

Session summary

The objectives of this first session of the workshop were to exchange information on different Severe Accident Management strategies used or contemplated as well as on the treatment of reflooding in different safety studies, with an emphasis on Probabilistic Safety Assessment (PSA) studies. Four papers were presented and discussed:

- In the first one by KIT the available experimental database on core reflood was analysed so as to derive useful information on the probability of success of core reflood actions;
- The second one described the IRSN experience in level 2 PSA studies and highlighted the benefit that could be gained from a better understanding of corium cooling phenomenology;
- GRS presented in the third one a status of lessons learnt from level 2 PSA for German PWRs and BWRs;
- In the fourth one, CEA presented the LEONAR tool developed by CEA on behalf of EDF for level 2 PSA in order to evaluate the probability of vessel failure and basemat melt-through.

Summaries of the presentations

W. Hering, Ch. Homann and W. Tromm (KIT), "Status of experimental and analytical investigations on degraded core reflood".

Thorough investigations were performed at Karlsruhe taking into account the available database including all public available data of experiments and including analytical work. A list of global parameters influencing degraded core reflood was established. Then the investigations were summarised in a preliminary "reflood map" addressing the issues of accident termination and released hydrogen fraction as a function of reflood mass flow rate in g/s*rod and of the damage status of the core. This map can be considered as a tool to summarise existing knowledge and identify areas for efficient future work. From these analyses, it is expected that the performance of the core reflood procedures can be extended beyond DBA (Design Basis Accidents) conditions, if a minimum water flow rate is available. A figure of about 1g/s*rod is given in the paper. The points that are not yet correctly addressed were pointed out. This is the case for example of the influence of fuel type (MOX or high burn-up) on fragmentation during quenching. The issue of transposition to reactor scale where multi-dimensional effects are expected to become important was also raised. Finally, the need to transfer rather detailed research findings to existing power plants through numerical simulation tools was highlighted, ASTEC V2 being considered as a first step to this goal.

E. Raimond, C. Caroli, R. Meignen, N. Rahni and B. Laurent (IRSN), "Importance of the in and ex-vessel corium coolability in case of severe accident for the French PWRs; Some views from L2 PSA and perspectives".

A particular attention was given in those studies to the application of Severe Accident Management Guidelines. The circumstances of water reflooding were identified and consequences analysed for flooding at the beginning of core degradation, after corium relocation in the RPV lower head or ex-vessel. These analyses took into consideration adverse effects (hydrogen production rate, pressure increase, DCH…). For early reflooding, it was concluded that the quality of modelling was not sufficient to derive firm conclusions so that R&D wok is needed. For reflooding after corium relocation in the RPV lower head, it was concluded that coolability is possible but not sure and that we should be cautious when estimating the time of RPV rupture because of the uncertainties; estimations being however possible in PSA studies. Adverse effects of reflooding were also analysed. The conditional probability of containment failure due to high hydrogen production rate was determined; it is quite affected by uncertainties as illustrated by the differences between best-estimate and conservative approaches. This tends to confirm the interest of a prudent approach in sending water into the core though this is still a matter of discussion. The risks of loss of containment integrity by DCH or overpressure were also assessed. Again, large differences were found between best-estimate and conservative cases. During the discussion that followed the presentation, some participants expressed the opinion that the conclusions about adverse effects were too pessimistic.

H. Löffler (GRS), "The issue of in-vessel coolability from the PSA level 2 point of view".

It presented the results of GRS level 2 PSA studies for two German PWRs and two German BWRs, addressing the probabilities for cooling in the core region and in the RPV lower head, the preconditions for successful in-vessel cooling and accident sequences with particularly high and low potential for successful in-vessel cooling. Examples of the influence of specific design details were also given. Two different aspects were considered: (i) the probability of reflooding systems to be operated in due time and (ii) the degree of core damage for which fuel/corium could be successfully retained inside the RPV. The probability for fuel/corium retention in the original core region was estimated for PWRs as a function of core degradation at reflooding initiation: if more than 40% of fuel from a large PWR core has melted, retention appears not to be possible. For BWRs, the probability of retention in the original core region depends much on the issue of blockage formation in the lower part of fuel elements: a very high probability of blockage formation was identified for sequences with wet core. The retention in RPV lower head for PWRs seems possible if core damage (i. e. loss of original fuel geometry) at flooding does not exceed 70%. For BWRs, it depends on specific design details of penetrations in the RPV lower head.

B. Tourniaire, B. Spindler, G. Ratel, J.M. Seiler, B. Iooss, M. Marquès, F. Gaudier (CEA) and G. Greffier (EDF), "The LEONAR code: a new tool for PSA level2 analyses."

LEONAR is developed by the CEA on behalf of the French utility EDF for PSA level 2 applications, the main goal being to evaluate the probabilities of RPV melt-through and basemat penetration in case of a core meltdown accident. It is intended to serve as a tool for analysing the impact of Severe Accident Management measures, including their potential failure, and to identify the main parameters that could influence the corium coolability during the late phases of a core meltdown accident. The LEONAR analyses focus on the likelihood of RPV failure and then basemat penetration, sequence of events that can open a release path for radioactive material to the environment. As input data, LEONAR uses a set of MAAP results for core degradation. Then it treats the core-melt evolution after flooding, the corium relocation in the RPV lower head in dry and wet conditions, the corium evolution in the RPV, the vessel failure, the corium relocation in the reactor pit, the molten corium concrete interaction and the possible corium spreading in neighbouring compartments. An external vessel cooling is also simulated. The influence of user-defined uncertainties on input parameters is treated statistically. In order to achieve these goals, the code must be fast-running and therefore the physical models are simplified.

GENERAL DISCUSSION AND CONCLUSIONS ON SESSION 1

From the discussions during the session, the following conclusions were derived:

- The possibility of stopping or delaying the progression of a core melt accident by the use of a recovered water source or taking benefit of specific engineered systems is taken into account in a number of PSA studies;

- Important parameters have been identified either through a thorough review of existing experimental and analytical data or from the results of PSA studies, most important being the core damage state at the initiation of reflooding and the available water flow rate;

- The uncertainty on the likelihood to stop the progression of a core melt-down accident by water injection is generally considered as high and depends on reactor specific features;

- A reduction of uncertainties is deemed important for existing reactors as core melt-down accidents were not considered as design-basis accidents in the original design;

- This need calls for a sustained R&D effort, both on experimental and analytical point of views – these two aspects were discussed in sessions 2 and 3 of the workshop.

Session 2

Experimental work

Session Chairperson:

W.Tromm (KIT)

Session summary

The objectives of this session were to review the experimental programmes on in-vessel degraded core coolability. Five papers were presented and discussed

The first paper by KIT presented the QUENCH programme on reflooding of bundles at high temperature. The major outcomes were highlighted and future activities briefly presented.

Two papers by IKE and IRSN presented experimental programmes on the coolability of debris beds with a special emphasis on multidimensional effects.

A paper by KIT addressed the melt behaviour in the RPV lower head.

Finally, a paper by CEA presented a facility for external vessel cooling experiments.

SUMMARIES OF THE PRESENTATIONS

M. Steinbrück, M. Große, L. Sepold, J. Stuckert (KIT), "Lessons learnt from the QUENCH programme at FZK"

The QUENCH program has delivered many advances with respect to high-temperature bundle behavior and the reflood process. Reflood progression and oxidation of the cladding tubes under highly transient conditions and the corresponding hydrogen source term under various boundary conditions were the major topics. Temperature escalation was seen during reflood only in tests where substantial quantities of zirconium bearing melt were exposed to the flowing steam. This is typically the case when temperatures exceeded the melting point of the α-Zr(O), but is also observed when presence of other materials caused melt to form at lower temperatures or when steam starvation had led to erosion of the protective and confining layer of ZrO_2. Another crucial phenomenon is steam starvation which causes thinning and degradation of the protective oxide scale and thus increases the probability for temperature excursions during reflood. As a third topic, advanced cladding materials have been extensively investigated. Although strong differences in the oxidation behavior were found, only a limited effect on bundle degradation and coolability was observed provided the oxide scale had not been weakened by breakaway. Generally, a nuclear reactor core seems to be coolable when the core is still intact and no or only local melt formation has already taken place. This is a realistic boundary condition up to 2200 K provided that the reflood water flow rate is >2 g/s rod, no strong eutectic melt formation occurred, and extended steam starvation phases before reflood could be avoided. Future experimental activities in the QUENCH program will be devoted, on the one hand, to loss of coolant accident (LOCA) scenarios, i.e. design basis accidents in the context of higher burnup and use of advanced cladding materials, and, on the other hand, to the investigation of formation and coolability of debris beds and molten pools in the core. The corresponding sub-programs are named QUENCH-LOCA and QUENCH-DEBRIS.

M. Rashid, R. Kulenovic, E. Laurien (IKE), "Experimental investigation of multidimensional cooling effects on the coolability of a debris bed"

IKE studies are currently related to experimental investigations on coolability of debris beds with volumetrically heated particle beds in a downcomer configuration (vertical tube mounted in bed centre) at different system pressures (1, 3 and 5 bar). Steady-state boiling as well as transient dryout experiments with a polydispersed bed content (mixture of steel balls of 2, 3 and 6 mm diameter) under top- and bottom-flooding as well as lateral flow conditions were carried out in order to study the influence of the downcomer on the hydraulic and thermal behavior (pressure drop, dryout heat flux) of the bed. For top-flooding (downcomer closed), the pressure gradients along the bed height are generally smaller than the corresponding hydrostatic pressure gradient due to interfacial friction of liquid and vapor in the counter-current flow. For bottom-flooding (downcomer open) CHF increases significantly due to the co-current flow regime. A natural circulation driven cooling is established which alleviates the counter current flooding limit substantially and hence increases coolability. For lateral flow condition (perforated downcomer) the pressure drop behaviour is distinctly enhanced compared to bottom-flooding flow condition but the observed dryout heat flux is more or less the same. Therefore, further investigations with perforated downcomer are needed to understand this behavior. Comparing the dryout heat fluxes for different system pressures at top- and bottom-flooding flow situations it has to be pointed out that with increasing system pressure a strong increase of dryout heat fluxes can be observed which improves distinctly the bed's coolability.

N. Stenne, M. Pradier, J. Olivieri, S. Eymery, F. Fichot, P. March, J. Fleurot (IRSN), "Multi-dimensional reflooding experiments: the PEARL programme"

To address the damaged core coolability issue, an experimental program devoted to study reflooding of debris beds has been launched at IRSN. A step-wise approach has been adopted, with feasibility and scoping reflooding tests in PRELUDE facility and more complex representative tests in the larger PEARL facility to address the multi-dimensional thermal-hydraulic processes involved in reflooding situations. PRELUDE results obtained so far show that the chosen technology is able to deposit a sufficient power density to achieve temperatures of 1000°C with induction heating. The reflood experiments will be continued on a homogenous debris bed from 400°C up to 1000°C to qualify the measurement of the steam flow rate generated during reflooding and the measurement of pressure inside the debris bed. The work on the PEARL facility design will be completed in order to start PEARL experiments second-half of 2010.

A. Miassoedov, T. Cron, J. Folt, X. Gaus-Liu, A. Palagin, S. Schmidt-Stiefel, T. Wenz (KIT), "LIVE experiments on melt behavior in the RPV lower head"

Within the LIVE experimental program several large-scale tests have been performed so far with water and with non-eutectic melts (KNO_3-$NaNO_3$) as simulant fluids. Besides the transient behavior, for which the LIVE tests provide unique data concerning temperature evolution and crust growth, the experiments address other important phenomena, such as the top/sideways splitting and local distribution of heat flux, and the influence of solidification on the thermal-hydraulics of the pool, i.e. the possible existence of a mushy region and its impact on the heat transfer. In the post-test analysis the crust thickness profiles along the vessel wall, the crust composition and the morphology are determined. One of the outcomes is that melt pouring near the vessel wall at the beginning of the test results in considerable asymmetric heat flux distribution even during the steady state. The experimental results are being used for the assessment of correlations and development and validation of mechanistic models for the description of molten pool behavior. These calculations are complemented by analyses with the CFD code CONV developed at IBRAE. The analysis showed that the LIVE experiments can be quite accurately predicted by the CFD simulation. Future analyses with the improved CONV code (e.g. implementation of phase diagrams) will be applied to high-temperature LIVE tests with molten salts and binary oxidic melts thus providing valuable support for understanding and improvement of modelling in severe accident codes.

E. Verloo, S. Szarzynski (CEA), "External vessel cooling experimentation: the CNU facilities

The CNU-M2 facility is a unique experimental set-up to investigate core retention capabilities by ex-vessel cooling. It is a large scale section representing the RPV, dedicated to study two-phase flow with steam production around a heated RPV geometry. An experimental program is on-going to determine the ability of cooling circuits to ensure efficient ex-vessel cooling. The influence of numerous parameters, both thermal-hydraulic and geometric, will be studied under different heat fluxes (values and profiles). The analysis of the first results is ongoing.
Among other experimental possibilities experiments tests on the influence of surface conditions (roughness) could be performed in future.

GENERAL DISCUSSION AND CONCLUSIONS ON SESSION 2

The QUENCH experiments have demonstrated that a cooling flow rate higher than 2 g/s/rod is sufficient to enable cooling can be guaranteed in a rod-like geometry. The focus for the future in the severe accident research will be on the investigation of the cooling options of debris beds. QUENCH DEBRIS experiments will investigate the formation of debris beds, the size and the structure of the debris.

The DEBRIS and the PEARL programme will look at the coolability issues, especially focusing on 3-d effects and effects of different particles sizes in different core regions.
Further experimental investigations with LIVE will focus on 3D effects in the lower head and the effect on coolability with the presence of water in the lower head during melt pour from the upper core region, which will be simulated in the LIVE facility using oxidic melts in future.

To investigate different effects on CHF with the CNU facility experiments with the addition of boric acid in the cooling water and the addition of impurities resulting from the failure of the insulation would be in principle possible. Very small gaps are as well an important issue for the VVER 440 PWR and could be investigated with CNU.

Session 3

Phenomenological and modelling work

Session Chairpersons:

J. Birchley (PSI)

W. Hering (KIT)

Session summary

The objectives of this session were to review the status of knowledge and modelling of in-vessel degraded core coolability. Eight papers were presented and discussed:

The first paper (CEA) discussed the phenomenology of in-vessel melt progression and identified a most credible range of melt relocation scenarios.

Two papers (RUB-LEE and IRSN) presented model validation studies for reflood cooling of the core while still in essentially rod-like geometry, using QUENCH data.

Three papers (IRSN (2) and IKE Stuttgart) reviewed the knowledge of debris bed characteristics, coolability and modelling.

One paper (GRS) presented a simulation of the TMI-2 sequence up to and including melt relocation and cooling.

The final paper (GRS) summarised the recent TMI-2 alternative scenario CSNI benchmark.

Thus the opening paper set the stage for discussion of modelling of specific areas, while the final two papers provided assessment of applicability to plant SA sequences.

SUMMARIES OF THE PRESENTATIONS

J.M. Seiler, B. Tourniaire (CEA), "Phenomenological analysis of in-vessel melt progression." Considerations of the physical processes suggests that a major melt relocation involving all or a large fraction of the core is improbable; instead the melting and relocation are thought more likely to take place over a period of time and in a series of stages, during which time less than half of the core would be molten in the lower head. There are realistic prospects for cooling the melt/debris in such a situation, whereas the cooling of the limiting case of a 100% melt relocation of a large reactor core would be problematic.

F. Fichot, N. Chikhi, O. Coindreau, J. Fleurot (IRSN), "Status of knowledge and modelling to simulate the reflooding of a severely damaged core."
The paper provided a short survey of limiting processes – frictional resistance to water penetration, dryout and countercurrent flow – and showed how the magnitudes affect quench progression and coolability. One of the key parameter is the representative particle size. The model developed for ICARE/CATHARE was assessed against selected test data, revealing some remaining deficiency, motivating the need for new data. The PEARL experimental program is designed to meet this need.

O. Coindreau, F. Fichot, N. Chikhi, J. Fleurot (IRSN), "Characteristics of the geometry of a severely damaged core and the debris bed expected to form during reflooding."
This study was motivated by the considerations discussed in the previous paper. The paper reviewed the sources of data on size distribution of fuel fragments, concentrating on the influence of burnup and thermal cycling with the aim of estimating the debris characteristic as a function of irradiation history. Data indicate an increase in fragmentation as burnup increases to moderate levels, while Halden data indicate much finer fragmentation at very high burnup. As yet lacking is a criterion for transition to this degree of fragmentation. The review continued with discussion of debris formation due to destruction of the cladding, as might occur during reflood at thigh temperature and significant cladding oxidation.

M. Buck, M. Bürger, S. Rahman, G. Pohlner (IKE), "Validation of the MEWA model for quenching of a severely damaged reactor core."
Following on from the previous discussions, the paper presented validation of recently developed models in the MEWA module of ATHLET-CD. MEWA provides a multi-field (solid, melt, two-phase fluid) 2-D cylindrical representation for debris/melt behaviour and reflooding. Comparisons were presented with data from experiments at Brookhaven, KTH and IKE Stuttgart covering a range of debris characteristics, as well as examples of debris quenching in a reactor. Results highlight the importance of multi-dimensional paths for water penetration and affirm the greater effectiveness of downcomers to facilitate bottom ingress and faster quenching.

P. Kruse, M. Koch (RUB), "Simulation of the fuel rod bundle test QUENCH-03 using the system codes ASTEC and ATHLET-CD."
This was the first of two papers on the subject of QUENCH simulation, and comprised a comparison analysis of QUENCH-03 simulations using different oxidation models (Prater-Courtright, Cathcart-Pawel and Urbanic-Heidrick) in conjunction with the both ASTEC and ATHLET-CD codes. A feature of QUENCH-03 was the very adverse conditions (high temperature, low pre-oxidation) at which reflood was initiated, which resulted in a major oxidation excursion. Both codes successfully simulated the experiment up to the initiated of reflood, but there was wide variation in the calculated oxidation during reflood, where only ASTEC/Prater-Courtright reproduced the observed excursion. This stark contrast highlights the difficulty in reliably a predicting reflood excursion.

N. Chikhi, F. Fichot, O. Coindreau, J. Fleurot (IRSN), "Validation of an Improved Reflooding Model in ICARE-CATHARE code: QUENCH-11 Test Calculation."
This second QUENCH simulation paper included a description of the new reflood model which represents the heat transfer in three regions near the quench front (quenched, transition and dispersed flow) and the axial conduction from the unquenched to quenched zone. The thermal-hydraulic model was aimed primarily at design basis LOCA sequences, and in the present code is coupled to the degradation and oxidation models. The model successfully reproduced the hydrogen generation and heat-up of the cladding in the upper part of the bundle during reflood. Since steam starvation was a factor in QUENCH-11, the analysis was extended to address the impact of delayed or reduced injection on the likelihood of a reflood excursion. Results indicated a cliff-edge effect at a temperatures above 1800 K and injection rates below 1 g/s/rod.

K. Trambauer, S. Weber, H. Austregesilo (GRS), "Analysis of the TMI-2 accident with the code ATHLET-CD."
The paper provided a strong demonstration of the capabilities of the code ATHLET-CD for whole plant simulation, where the first three phases before core slump into the lower plenum during TMI-2 were successfully reproduced. The calculated pressure history after pump trip, during the pump restart and until core slump was in good agreement with the measured data, while the calculated hydrogen generation before the pump restart is in accordance with the value deduced from TMI-2 data. Contrary to estimates based on the system behaviour, only a relatively small increase of hydrogen production was calculated during the quench phase. This observation is consistent with the ATHLET-CD simulation of QUENCH-03. Underestimation of debris bed and melt pool formation was attributed to the lack of a model for embrittlement and relocation of solid fuel fragments and to the assumption of a complete release of volatile fission products from the melt. The lack of knowledge in these areas has long been recognised. Further model extensions regarding the quenching of degraded core material and the fracture and relocation of solid fuel rods are necessary to further improve the simulation.

S. Weber, H. Austregesilo (GRS), F. Fichot, O. Marchand (IRSN), G. Bandini (ENEA), M. Barnák, P. Matejovič (IVS), S. Paci (UPI), K.Y. Suh (SNU), M. Buck (IKE), L. Humphries (SNL), "A benchmark exercise on an alternative TMI-2 accident scenario."
This and the previous paper on TMI-2 simulation were in many ways complementary, although the aims of the studies were separate. This was the first CSNI exercise on TMI-2 since 1990. A modified sequence was defined to minimize uncertainties arising from the thermal-hydraulic calculation and to

focus on uncertainties in the core degradation calculation. The severe accident characteristics were similar to the actual sequence. Eight organisations participated, using seven different code versions: ASTEC V1.3 (2), ATHLET-CD Mod2.1, ATHLET-CD/MEWA, ICARE/CATHARE V2.1, MELCOR 1.8.5, MELCOR 1.8.6 and MAAP4.03. Thus nearly all of the widely used codes were employed, with a wide range of models and modelling approaches. Although there is still a lack in knowledge of some physical processes, all the codes calculated the specified scenario with only minor tuning of parameters or optimization of input decks. Uncertainties remained concerning some key variables such as hydrogen generation, but the simulations were significantly more consistent than in previous benchmarks, thus demonstrating robustness of the current codes and evidence of significant progress in the last 20 years.
In view of the regulatory importance of uncertainty estimation, it was suggested that systematic uncertainty analysis be included in any future extension of this benchmark, using the techniques investigated in the BEMUSE programme as a basis. It could also be extended to study the influence of SAM actions and how they can be simulated by current codes.

GENERAL DISCUSSION AND CONCLUSIONS ON SESSION 3

From the discussions during the session, the following conclusions were derived:

The efficiency of reflooding is not demonstrated for all situations, with high uncertainty on the likelihood to stop the degradation by water injection.

During early stage degradation, about 1g/s/rod for was judged the minimum flow rate for successful quenching. Difficulties remain in predicting if injection would trigger an oxidation excursion.

The need to reduce uncertainties calls for a sustained and coordinated R&D effort, both of an experimental and analytical nature. Ongoing or planned programmes are addressing coolability of different core configurations.

A review of experimental data and analytical work on degraded core reflood resulted in a preliminary reflood map identifying the main parameters influencing in-core coolability.

Code developments are seeking a more mechanistic approach applicable to different configurations; validation is expected against data from ongoing experiments. However, transposition of results to reactor scale needs to be evaluated since larger scale experiments are probably not feasible.

Major progress has been achieved in the last ten or so years. Techniques to quantify and manage uncertainties have been developed and exercised. Benchmark studies will continue to play a key role.

Session 4

Specific reactor safety studies on in-vessel coolability

Session Chairperson:

H. Löffler (GRS)

Session summary

The objective of this session of the workshop was to provide information on plant specific studies on the issue of in-vessel retention (IVR). Five papers were presented and discussed:

- In the first and second one a concept for ex-vessel cooling in VVER-440/V213 reactors and its analysis and evaluation has been presented. It is to be expected that this concept will soon be implemented in the Hungarian Paks plant.
- The main objective of the third study was to support the SAM strategy for the optimum use of recovered water supplies in the period following the onset of core degradation but prior to RPV failure for the Sizewell B PWR.
- In the fourth study a method for coupled simulations of the flow inside the control rod guide tubes at the bottom of a BWR and the melt pool heat transfer has been developed and applied to a prototypical Swedish BWR ABB-Atom vessel design,
- In the fifth paper a dedicated thermodynamics code has been developed to assess whether the ex-vessel cooling can ensure RPV wall integrity. It has been applied to the KERENA™ advanced boiling water reactor concept.

SUMMARIES OF THE PRESENTATIONS

Peter Matejovic, Miroslav Barnak, Milan Bachraty (IVS Trnava, Slovakia), Robert Berky (IBOK, Slovakia), "Assessment of In-vessel Corium Retention for VVER-440/V213" and József Elter, Éva Tóth (Paks NPP, Hungary), Peter Matejovič (IVS Trnava Ltd., Slovakia), "Proposal of In-vessel Corium Retention Concept for Paks NPP"
Recent activities devoted to IVR concept via ERVC for standard VVER-440/V213 reactors were (are) performed in the frame of 5th, 6th and 7th FW EU Programme as well as within national programmes performed in the countries operating this type of reactors. In total there is 12 VVER-440/V213 reactors operated in Central European Countries. There is serious interest for the adoption IVR concept for these plants and co-operation on this field is going on between the plant operators as well as between technical support organisations. However, the VVER-440/V213 reactors were designed and built according to earlier standards and the possibility of core melting was not accounted for in the design stage of these low-powered reactors. Thus, the possibility of additional implementation of technical modifications, which are necessary for the creation of an ERVC loop, is rather limited. However, this shortcoming is compensated with significantly lower maximum heat fluxes than expected for higher-powered reactors. An IVR concept with simple ECVR loop based only on minor modifications of existing plant technology was proposed for Paks NPP. The coolability efficiency in such ERVC loop geometry was confirmed by RELAP 5 analysis. Further research should be focused on confirmation of low maximum heat flux values. Here the outputs from SARNET and results of ASTEC V2 analysis should be valuable. The coolability limits in given ERVC geometry should be confirmed experimentally on Hungarian CERES facility (Cooling Effectiveness on Reactor External Surface), which is just under construction and the first experiments are planned to be performed during next year. The analyses supported the assumption that the proposed solution is effective in preserving RPV integrity in the case of severe accident. This solution after having licence from regulatory body is planned to be implemented in 2011 on the 1st Unit in Paks NPP.

K. L. Peers, E. Grindon (AMEC, United Kingdom), P. Lightfoot (British Energy Generation, United Kingdom), "A Case Study to Support the Sizewell B SAM Strategy - Evaluation of the Optimum Use of Water after Core Degradation Has Started"

This paper reports the results of an early and innovative study of the feasibility of in-vessel retention (IVR) for the Sizewell B PWR, where the containment includes a modified floor drainage arrangement. This ensures that under accident conditions, even without the operation of engineered safeguards, the reactor cavity is flooded to a depth of 1 to 3 metres. The main objective of the study was to support the SAM strategy for the optimum use of recovered water supplies in the period following the onset of core degradation but prior to RPV failure. The accident scenario adopted was based on a station blackout with loss of emergency core cooling system (ECCS) and all other safeguards. The study considered four scenarios with recovered water sources at various times and considered issues such as water accessibility, prevention of RPV failure and reduction of radiological releases to the environment. One of the significant developments in MAAP4.0.3 was the introduction of a model for the external cooling of the RPV. However, this model for heat transfer from the lower head to its surroundings was considered too simplistic to support the study and this led to the implementation of a number of modelling improvements.

Francesco Cadinu, Tran Chi Thanh, Pavel Kudinov (Nuclear Power Safety, Royal Institute of Technology, Stockholm, Sweden), "Analysis of In-Vessel Coolability and Retention with Control Rod Guide Tube Cooling in Boiling Water Reactors"

In normal operation of a Boiling Water Reactor (BWR) there is a purge water flow through the Control Rod Guide Tubes (CRGTs). The goal of this paper is to develop appropriate simulation tools and study the feasibility and the efficiency of the CRGT flow for the cooling of core melt material in the lower head of a BWR as a Severe Accident Management measure. It was found that a strong feedback exists between the heat transfer inside the CRGTs and the phase change dynamics in the solidifying metal layer of a stratified heterogeneous configuration of core materials, relocated to the lower head. In order to capture this feedback, a method for coupled simulations of the flow inside the CRGTs and the melt pool heat transfer has been developed. The calculations are performed for a prototypical Swedish BWR ABB-Atom vessel design, considering a representative 3D slice of the lower plenum, containing six CRGTs and a section of the vessel wall. The coupled simulations demonstrate the effect of the CRGT mass flow rate on the melt pool cooling. In order to insure its integrity, the CRGT wall temperature, during a transient, should never exceed the creep limit. The transient interplay between CRGT flow regime and heat transfer, metal layer solidification and latent heat release rates are discussed in detail.

Patrick Levi, Manfred Fischer (AREVA NP GmbH, Paul-Gossen-Straße 100, 91052 Erlangen, Germany), "In- and Ex-Vessel Flooding as Part of the Severe Accident Strategy in the KERENA™ Reactor"

Despite its low core damage frequency, the KERENA™ advanced boiling water reactor is equipped with dedicated severe accident mitigation systems. The provided combination of passive in-vessel and ex-vessel flooding will cool the core inside the RPV and at the same time ensure thermal stabilization of the RPV wall. In the unlikely event that the passive in-vessel flooding cannot be actuated or fails, the core will melt and relocate into the lower head of the RPV. In this case, as a further line of defence, decay heat removal can be achieved through the RPV wall into the water in the cavity. To assess whether the ex-vessel cooling can ensure RPV wall integrity a dedicated thermodynamics code has been developed which considers heat transfer from the molten corium pool into the RPV wall and the resulting wall ablation. The stratification behaviour of the oxidic and metallic phase of the molten pool is examined. According to the analyses, in the case of a light metallic phase on top, high heat fluxes caused by a thin metallic layer are practically eliminated due to the large masses of the molten steel internals. The analyses also show that the worsening of this focusing effect due to transient states can practically be excluded.

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GENERAL DISCUSSION AND CONCLUSIONS ON SESSION 4

The session was devoted to “specific reactor safety studies on in-vessel coolability“. The summary of presentations is reported above. The following statements are an attempt of the session chairman to put the different aspects of plant specific studies into perspective:

1. Phenomenological assessment of general issues such as core degradation process, heat fluxes from the corium into the RPV wall, or the critical heat flux from the RPV outside to a surrounding water pool, related code development: These issues are in principle covered by research and development which is the subject of sessions 2 and 3. It is the task of the plant specific analyses to correctly apply the latest status of knowledge in these fields.

2. Assessment of reactor type specific conditions which affect IVR. Prominent examples for these issues are the accessibility of water to the RPV bottom, the availability of flow paths or of a final heat sink. Since these aspects aredesign specific, analysis will have to be performed on an individual basis. Unexpected problems may be encountered, such as flow oscillations, or possibly unsymmetrical distortions of components during the accident. Excellent examples of related analysis efforts have been reported in the contributions to this session 4.

3. Assessment of the probability that necessary preconditions for successful IVR on part of systems or human actions: A particular difficulty of this aspect is that the plant and the staff are in a totally disturbed condition – otherwise the core would not have melted. Only few such analyses exist or are reported publicly. Such information is necessary to provide a complete picture of IVR

4. PSA level 2 would be the ultimate means for assessing the benefits of IVR, taking into account all the issues mentioned above. An interesting (however not probabilistic) contribution is the presentation for the Sizewell B plant where different options for the optimal use of water which has become available have been compared.

Appendix 1

Workshop Programme

MONDAY OCTOBER 12TH, 2009

WELCOME AND OPENING ADDRESS

14:00 B. Clément (IRSN), A. Amri (OECD-NEA)

SESSION 1: GENERAL STUDIES CHAIR B. CLÉMENT (IRSN)

14:30 W. Hering, Ch. Homann, W. Tromm (FZK)
Status of experimental and analytical investigations on degraded core reflood
15:10 E. Raimond, C. Caroli, R. Meignen (IRSN)
Importance of in and ex-vessel corium coolability in case of severe accident for the French PWRs. Some views from L2 PSA and perspectives
15:50 Coffee Break
16:10 H. Löffler (GRS)
The issue of in-vessel coolability from the PSA level 2 point of view
16:50 B Tourniaire, B. Spindler, G. Ratel, J.M. Seiler, B. Looss, M. Marques, F. Gaudier (CEA),G. Greffier (EDF)
The LEONAR code : a new tool for PSA level 2 analyses
17:30 Adjourn

Tuesday October 13th, 2009

SESSION2: EXPERIMENTAL WORK CHAIR: W. TROMM (FZK)

8:45	M. Steinbrück, M. Große, L, Sepold, J. Stuckert (FZK) Lessons learned from the QUENCH programme at FZK
9:25	M. Rashid, R. Kulenovic, E. Laurien (IKE) Experimental investigation of multidimensional cooling effects on the coolability of a debris bed
10:05	N. Stenne, M. Pradier, J. Olivieri, S. Eymery, F. Fichot, P. March, J. Fleurot (IRSN) Multi-dimensional reflooding experiments: the PEARL programme
10:45	Coffee Break
11:05	A. Miassoedov, T. Cron, J. Folt, X. Gaus-Liu, A. Palagin, S. Schmidt-Stiefel, T. Wenz (FZK) LIVE experiments on melt behaviour in the RPV lower head
11:45	E. Verloo, S. Szarzynski (CEA) External vessel cooling experimentation: the CNU facilities
12:25	Lunch

SESSION 3: PHENOMENOLOGICAL AND MODELLING WORK
CHAIR: J. BIRCHLEY (PSI)

13:30	J.M. Seiler, B. Tourniaire (CEA) Phenomenological analysis of in-vessel melt progression
14:05	F. Fichot, N. Chikhi, O. Coindreau, J. Fleurot (IRSN) Status of knowledge and modelling to simulate the reflooding of a severely damaged core
14:40	O. Coindreau, F. Fichot, N. Chikhi, J. Fleurot (IRSN) Characteristics of the geometry of a severely damaged core and the debris bed expected to form during reflooding
15:15	Coffee Break
15:35	M. Buck, M. Bürger, S. Rahman, G. Pohlner (IKE) Validation of the MEWA model for quenching of a severely damaged reactor core
16:10	P. Kruse, M. Koch (RUB) Simulation of the fuel rod bundle test QUENCH-03 using the system codes ASTEC and ATHLET-CD
16:45	N. Chikhi, F. Fichot, O. Coindreau, J. Fleurot (IRSN) Validation of an Improved Reflooding Model in ICARE-CATHARE code: QUENCH 11 Test Calculation
17:20	Adjourn

Wednesday October 14th, 2009

SESSION 3: PHENOMENOLOGICAL AND MODELLING WORK (CONTINUED)

08:45 K. Trambauer, S. Weber, H. Austregesilo (GRS)
Analysis of the TMI-2 accident with the code ATHLET-CD

9:20 S. Weber, H. Austregesilo (GRS), F. Fichot, O. Marchand (IRSN), G. Bandini (ENEA), M. Barnák, P. Matejovič (IVS), S. Paci (UPI), K.Y. Suh (SNU), M. Buck (IKE), L. Humphries (SNL)
A benchmark exercise on an alternative TMI-2 accident scenario

9:55 Coffee Break

SESSION 4: SPECIFIC REACTOR STUDIES CHAIR: H. LÖFFLER (GRS)

10:05 P. Matejovic, M. Barnak (IVS), R. Berky (IBOK)
Assessment of in-vessel corium retention concept for VVER-440/V213 reactors

10:40 J. Elter, E. Toth (PAKS NPP), P. Matejovic (IVS)
Proposal of in-vessel corium retention concept for PAKS NPP

11:15 K. L. Peers, E. Grindon (AMEC), P. Lightfoot (British Energy)
A case study to support the Sizewell B SAM strategy – evaluation of the optimum use of water after core degradation has started

11:50 F. Cadinu, C.T. Tran, P. Kudinov (RIT)
Analysis of CRGT cooling efficiency for in-vessel coolability of a stratified melt pool

12:25 P. Levi, M. Fischer (AREVA NP)
Assessment of in-vessel melt retention in an advanced boiling water reactor design

13:00 Lunch

GENERAL DISCUSSION AND CONCLUSIONS

14:30 Conclusions by Sessions' chairs and discussion

15:45 Adjourn

Appendix 2

List of Participants

BELGIUM
Mr. Tim DE VITS
Civil Engineer
Tractebel Engineering
Arianelann 7
1200 Sint-Lambrechts- Woluwe
Tel: +32 474 12 28 55
Fax:+32
Eml: tim.devits@gdfsuez.com

BULGARIA
Mrs. Boryana PAVLOVA ATANASOVA
Nuclear Engineer NPP Safety Analyses
INRNE-BAS
1784 Sofia
72 Tzarigradsko Chaussee Boulevard
Tel: +359 2 979 5583
Fax: +359 2 975 3619
Eml: b_atanasova@inrne.bas.bg

CANADA
Dr. P. Mani MATHEW
Section Head, Severe Accidents
Atomic Energy of Canada Ltd
Chalk River Laboratories
Chalk River, Ontario
K0J 1J0
Tel: +1 613 584 8811 Ext 44867
Fax: +1 613
Eml: mathewm@aecl.ca

CZECH REPUBLIC
Dr. Miroslav KOTOUC
Researcher
Nuclear Research Institute Rez plc
130, Husinec-Rez
250 68
Tel: +420 266 172 116
Fax: +420 266 143 570
Eml: ktc@ujv.cz

FINLAND
Mr. Risto SAIRANEN
Section Head
Radiation and Nuclear Safety Authority (STUK)
P.O. Box 14
FI-00881 Helsinki
Tel: +358 9 75988 701
Fax:+358 9 75988 382
Eml: Risto.Sairanen@stuk.fi

FRANCE
Mr. Garo AZARIAN
Severe Accident R&D Manager
AREVA NP SAS
Tour AREVA
Tel: +33 1 34 96 77 22
Fax: +33 1 34 96 76 12
Eml: garo.azarian@areva.com

1 Place Miller
92084 Paris-la-Defense Cedex 16

Dr. Ahmed BENTAIB
Engineer
IRSN /DSR
Centre d'Etudes Nucléaires de Fontenay-aux-Roses
77-83 Avenue du Général de Gaulle
B.P. 17, F92236 Fontenay-aux-Roses Cedex
Tel: +33 1 58 35 98 54
Fax: +33 1 46 57 22 74
Eml:ahmed.bentaib@irsn.fr

Dr. Nourdine CHIKHI
Engineer-Researcher
Major Accident Prevention Division – IRSN
Centre d'étude de Cadarache
IRSN/DPAM/SEMCA/LESAM, BP3, Bat 700
13115 Saint Paul-lez-Durance
Tel: +33 4 42 19 97 79
Fax: +33 4 42 19 91 65
Eml: nourdine.chikhi@irsn.fr

Mr. Bernard CLEMENT
IRSN/DPAM/SEMIC - Bt 702
Centre de Cadarache
BP3
13115 Saint-Paul-lez Durance, Cedex
Tel: +33 4 42 19 94 70
Fax: +33 4 42 19 91 67
Eml: bernard.clement@irsn.fr

Ms. Olivia COINDREAU
Research Engineer
IRSN
IRSN/DPAM/SEMCA/LESAM, Bat 700
13115 St Paul-lez-Durance
Tel: +33 4 42 19 92 63
Fax: +33 4 42 19 91 65
Eml:olivia.coindreau@irsn.fr

Dr. Florian FICHOT
IRSN/DPAM/SEMCA/LESAG
CEA Cadarache - Bat. 700
B.P. 3
F-13115 St Paul-lez-Durance Cedex
Tel: +33 4 42 19 95 19
Fax: +33 4 42 19 91 66
Eml: florian.fichot@irsn.fr

Mrs. Joelle FLEUROT
Head of Laboratory, IRSN
Severe Accident Prevention Division
IRSN/DPAM/SEMC Batiment 700, BP3
13108 Saint Paul-lez-Durance, Cedex
Tel: +33 4 42 19 95 21
Fax: +33 4 42 19 91 65
Eml: joelle.fleurot@irsn.fr

Mr. Patrice GIORDANO
Head of Service
IRSN/DPAM/SEREM
BP3
F-13115 Saint Paul-lez-Durance
Tel: +33 4 42 19 95 01
Fax: +33 4 42 19 91 60
Eml: patrice.giordano@irsn.fr

Mr. Gilles GREFFIER
Senior Engineer
Electricité de France - EDF SEPTEN
12-14 Avenue Dutriévoz
Tel: +33 4 72 82 74 02
Fax: +33 4 72 82 73 56
Eml: gilles.greffier@edf.fr

69628 Villeurbanne, Cedex

Mr. Olivier GRÉGOIRE Chargé d'affaires codes de calcul STXN (Service Mixte des Chaufferie Nucléaires) 25 rue Leblanc 75015 Paris	Tel: +33 1 64 50 27 43 Fax: +33 1 64 50 13 07 Eml: olivier.gregoire@cea.fr
Mr. Emmanuel RAIMOND Level 2 PSA Project Manager IRSN/DSR/SAGR/BEPAG B.P. No. 17 92262 Fontenay-aux-Roses	Tel: +33 1 58 35 78 70 Fax: +33 1 58 35 85 59 Eml: emmanuel.raimond@irsn.fr
Mr. Gilles RATEL Engineer DEN/DTN/SE2T/LPTM CEA Grenoble 17 Rue des Martyrs 38054 Grenoble Cedex	Tel: +33 4 38 78 52 57 Fax: +33 4 38 78 52 51 Eml: gilles.ratel@cea.fr
Dr. Jean-Marie SEILER Director of Research DTN/SE2T Bat. 1005 CEA - Grenoble 17 rue des Martyrs 38054 Grenoble Cedex 9	Tel: +33 4 38 78 30 23 Fax: +33 4 38 78 52 51 Eml: jean-marie.seiler@cea.fr
Mr. Bertrand SPINDLER CEA/DEN/DTN/SE2T/LPTM CEA - Grenoble 17 rue des Martyrs 38054 Grenoble Cedex	Tel: +33 4 38 78 46 87 Fax: +33 4 38 78 52 51 Eml: bertrand.spindler@cea.fr
Mrs. Nathalie STENNE Project Leader IRSN/DPAM IRSN/DPAM/SEREM Bâtiment 327 Centre de Cararache 13115 Saint-Paul-lez-Durance	Tel: +33 4 42 19 96 97 Fax: Eml:nathalie.stenne@irsn.fr
Mr. Eric VERLOO` CEA Cadarache (DTN/STRI/LHC)	Tel: +33 4 42 25 43 47 (Sec. 3847) Fax: +33 4 42 25 45 75 Eml: eric.verloo@cea.fr
Ms. Magali ZABIÉGO CEA - DTN/STRI/LMA, Bâtiment 708 13108 Saint-Paul-les-Durance	Tel: +33 4 42 25 28 37 Fax: +33 4 42 25 77 88 Eml: magali.zabiego@cea.fr

GERMANY

Dr. Henrique AUSTREGESILO	Tel: +49 89 3200 4443

Scientist - Reactor Safety Research Division
Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH
Forschungsginstitute, D-85748 Garching
Fax: +49 89 3200 4599
Eml: henrique.austregesilo@grs.de

Mr. Michael BUCK
Universität Stuttgart
Institut für Kernenergetik
und Energiesysteme (IKE)
Pfaffenwaldring 31
D-70569 Stuttgart
Tel: +49 711 685 62124
Fax: +49 711 685 62010
Eml: buck@ike.uni-stuttgart.de

Mr. Manfred FISCHER
Team Leader
AREVA NP GmbH
Paul-Gossen-Str. 100
D-91052 Erlangen
Tel: +49 9131 189 25 77
Fax:+49 9131 189 75 07
Eml: manfred.fischer@areva.com

Mr. Micro GROSSE
Research Scientist
Forschungszentrum Karlsruhe (FZK)
PO Box 3640
D-76021 Karlsruhe
Tel: +49 7247 82 3884
Fax: +49 7247 82 4567
Eml: Micro.Grosse@IMF.FZK.de

Dr. Wolfgang HERING
Forschungszentrum Karlsruhe (FZK)
Institut für Reaktorsicherheit
Anlagendynamik und Reaktorsicherheit
Postfach 3640
D-76021 Karlsruhe
Tel: +49 7247 822556
Fax: +49 7247 823718
Eml: wolfgang.hering@irs.fzk.de

Mr. Mathias HOFFMAN
Research Scientist-Reactor Simulation and Safety
Ruhr-University Bochum-Energy Systems
and Energy Economics
Ruhr-Universität, Bochum
Universitätsstr. 150, D-44801 Bochum
Tel: +49 234 322 6376
Fax: +49 234 321 4158
Eml: hoffman@lee.rub.de

Mr. Ch. HOMANN
Forschungszentrum Karlsruhe (FZK)
Tel:
Fax:
Eml:

Mr. Philipp KRUSE
Research Scientist
Ruhr-University Bochum-Energy Systems
and Energy Economics
Universitätsstraβe, 150
D-44801 Bochum
Tel: +49 234 32 28231
Fax: +49 234 32 14158
Eml: Kruse@lee.rub.de

Dr. Patrick LEVI
Project Engineer
Tel: +49 9131 1893778
Fax: +49 9131 1897507

AREVA NP GmbH
Paul-Gossen-Str. 100
D-91052 Erlangen
Eml: patrick.levi@areva.com

Mr. Horst LÖFFLER
Project Leader
Gesselschaft für Anlagen-und Reaktorsicherheit (GRS) mbH
Schwertnergasse 1, D-50667 Köln
Tel: +49 221 2068 690
Fax: +49 221 2068 10690
Eml: horst.loeffler@grs.de

Mr. Wolfgang LUTHER
Research Engineer
Gesselschaft für Anlagen-und Reaktorsicherheit (GRS) mbH
Forschungsgelände
D-85748 Garching
Tel: +49 89 32004 433
Fax: +49 89 32004 599
Eml: wolfgang.luther@grs.de

Dr. Alexei MIASSOEDOV
Forschungszentrum Karlsruhe (FZK)
Institut für Kern-und-EnergieTechnik
P.O. Box 4640
D-76021 Karlsruhe
Tel: +49 7247 82 2253
Fax: +49 7247 82 4837
Eml: alexei.miassoedov@iket.fzk.de

Mr. Georg POHLNER
Scientific Assistant
Institute of Nuclear Technology and Energy Syst. (IKE)
Pfaffenwaldring 31
D-70569 Stuttgart
Tel: +49 711 685 62730
Fax: +49 711 685 62008
Eml: pohlner@ike.uni-stuttgart.de

Mr. Saidur RAHMAN
PhD Student
Institute of Nuclear Technology and Energy Syst. (IKE)
Pfaffenwaldring 31
D-70569 Stuttgart
Tel: +49 711 685 69672
Fax: +49 711 685 62008
Eml: saidur.rahman@ike.uni-stuttgart.de

Dr. Felix Philipp SASSEN
Safety Analyst (Probabilistic Safety Analysis)
Westinghouse Electric Germany GmbH
Dudenstraβe 44
68167 Mannheim
Tel: +49 621 388 2105
Fax: +49 621 388 2104
Eml: SassenFP@westinghouse.com

Dr. Martin STEINBRÜCK
Head of Department
Karlsruhe Institute of Technology (KIT, former FZK)
Postfach 3640
D-76021 Karlsruhe
Tel: +49 7247 82 2517
Fax: +49 7247 82 4567
Eml: steinbrueck@imf.fzk.de

Dr. Juri STUCKERT
Senior Research Scientist
Forschungszentrum Karlsruhe (FZK)
Hermann-von-Helmholtz-Platz 1
Tel: +49 7247 82 2558
Fax: +49 7247 82 3956
Eml: juri.stuckert@imf.fzk.de

D-76344 Eggenstein-Leopoldshafen

Dr. Wolfgang TIETSCH
Westinghouse Electric Germany GmbH
Postfach 10 05 63
D-68140 Mannheim
Dudenstraâe 44
D-68167 Mannheim
Tel: +49 621 388 2120
Fax: +49 621 388 2104
Eml: tietsch@westinghouse.com

Dr.-Ing. Klaus TRAMBAUER
Gesellschaft für Anlagen- und Reaktorsicherheit
(GRS)mbHEml:
Forschungsinstitute
85748 Garching
Tel: +49 89 32 00 44 36
Fax: +49 89 32 00 45 99
klaus.trambauer@grs.de

Dr. Th. Walter TROMM
Deputy Head
Forschungszentrum Karlsruhe GmbH
Programm Nukleare
Sicherheitsforschung (Nuklear)
Postfach 3640,76021 Karlsruhe
Tel: +49 7247 82 5509
Fax: +49 7247 82 5508
Eml: walter.tromm@nuklear.fzk.de

Dr. Peter VOLKHOLZ
Section Manager
AREVA NP GmbH
Paul-Gossen Strasse 100
PO Box 1109
D-91052 Erlangen
Tel: +49 9131 1892736
Fax: +49 9131 1897507
Eml: peter.volkholz@areva.com

Dr. Sebastian WEBER
Scientist, Reactor Safety Research Division
Gesselschaft für Anlagen-und Reaktorsicherheit
(GRS)mbH
Forschungsinstitute
D-85748 Garching
Tel: +49 89 32004 438
Fax: +49 89 32004 599
Eml: sebastian.weber@grs.de

HUNGARY

Dr. József ELTER
Head Nuclear Department
Paks Nuclear Power Plant Ltd
POB 71
8803/10
H-7031 Paks
Tel: +36 20 982 1151
Fax: +36 75 506 733
Eml: elter@npp.hu

Mrs. Éva Laki TÓTH
Group leader
Paks Nuclear Power Plant Ltd
POB 71
8803/10
H-7031 Paks
Tel: +36 20 912 7856
Fax: +36 75 506 733
Eml: tothnel@npp.hu

ITALY

Dr. Felice DE ROSA
Responsible Severe Accident and Accident Managem.
Ente per le Nuove Tecnologie
l'Energia e l'Ambiente (ENEA)
Via Martiri di Monte Sole, 4
I-40129 Bologna
Tel: +39 51 6098715
Fax: +39 51 6098279
Eml: felice.derosa@enea.it

Dr. Flavio PAROZZI
Power Generation Systems Department
Nuclear and Industrial Plants Safety – Head
ENEA - Ricerca sul Sistema Elettrico SpA
via Rubattino 54
20134 Milano
Tel: +39 2 3992 4775
Fax: +39 2 3992 5626
Eml: flavio.parozzi@erse-web.it

KOREA (REPUBLIC OF)

Dr. Jung-Jae LEE
Senior Researcher
Risk Assessment Department
Korea Institute of Nuclear Safety (KINS)
Guseong-dong 19, Yuseong-gu
Daejeon 305-338
Tel: +82 42 868 0787
Fax: +82 42 868 0252
Eml: jjlee@kins.re.kr

SLOVAK REPUBLIC

Dr. Miroslav BARNAK
Senior Researcher, Deputy Director
IVS Hviezdoslavova 12
P.O. Box 141
917 01 Trnava
Tel: +421 33 5503203
Fax: +421 33 5503204
Eml: ivstt@nextra.sk

Mr. Juraj JANCOVIC
Researcher
VUJE a.s.
Okruzná 5
918 64 Trnava
Tel: +421 33 5991339
Fax: +421 33 5991708
Eml: jancovic@vuje.sk

Dr. Peter MATEJOVIC
Director of IVS
IVS Hviezdoslavova 12
P.O. Box 141
917 01 Trnava
Tel: +421 33 5503203
Fax: +421 33 5503204
Eml: ivstt@nextra.sk

SWEDEN

Mr. Francesco CADINU
PhD Student
Royal Institute of Technology (KTH)
AlbaNova University Center
KTH Nuclear Power Safety
Roslagstullsbacken 21, SE-10691 Stockholm
Tel: +46 8 5537 8825
Fax: +46 8 5537 8830
Eml: francesco@safety.sci.kth.se

Prof. Bal Raj SEHGAL
Emeritus Professor - Royal Institute of Technology
Nuclear Power Safety, KTH
AlbaNova University Center
Roslagstullsbacken 21, SE-10691 Stockholm
Tel: +46 8 5537 8820
Fax: +46 8 5537 8830
Eml: sehgal@safety.sci.kth.se

Mr. Chi-Thanh TRAN
Researcher
Royal Institute of Technology (KTH)
Roslagstullsbacken 21, D5
SE-10691 Stockholm
Tel: +46 8 5537 8826
Fax: +46 8 5537 8830
Eml: thanh@safety.sci.kth.se

Dr. Walter VILLANUEVA
Researcher
Nuclear Power Safety Division
Royal Institute of Technology
KTH-AlbaNova Roslagstullsbacken 21, D5
Stockholm SE-10691
Tel: +46 8 5537 8826
Fax: +46 8 5537 8830
Eml: walterv@kth.se

SWITZERLAND

Dr. Jonathan BIRCHLEY
Senior Scientist
Labor für Thermohydaulik
Paul Scherrer Institute
OVGA/312
Bachstrasse CH-5232 Villigen
Tel: +41 56 310 27 24
Fax: +41 56 310 21 99
Eml: jonathan.birchley@hotmail.com

Dr. Olivier ZUCHUAT
Head of PSA level 2
BKW FMB Energie AG
Kernkraftwerk Mühleberg (KKM)
CH-3203 Mühleberg
Tel: +41 31 754 71 11
Fax: +41 31 754 71 20
Eml: olivier.zuchuat@bkw-fmb.ch

UNITED KINGDOM

Mrs. Elizabeth GRINDON
Senior Safety Consultant
AMEC
Booths Park, Chelford Road
Knutsford, Cheshire
WA16 8QZ
Tel: +44 1565 684828
Fax: +44 1565 684870
Eml: liz.grindon@amec.com

Mr. Peter LIGHTFOOT
Safety Case Development Specialist
British Energy Generation, Engineering Division
Barnett Way
Barnwood,
Gloucester GL4 3RS
Tel: +44 1452 654803
Fax: +44 1452 652298
Eml: pete.lightfoot@british-energy.com

Mrs. Karen PEERS
Senior Safety Consultant
Tel: +44 1565 684800
Fax: +44 1565 684870

AMEC
Booths Park, Chelford Road
Knutsford, Cheshire
WA16 8QZ
Eml: karen.peers@amec.com

UNITED STATES OF AMERICA

Dr. Randall O. GAUNTT
Manager, Reactor Modelling and Analysis Department
Sandia National Laboratories
1515 Eubank SW
Albuquerque
New Mexico 87185-0748
Tel: +1 505 263 6849
Fax: +1 505 844 2829
Eml: rogaunt@sandia.gov

International Organisations

OECD Nuclear Energy Agency, Issy-les-Moulineaux

Mr. Abdallah AMRI
OECD-NEA / Nuclear Safety Division
Le Seine St-Germain
12 bd des Iles
F-92130 Issy-les-Moulineaux
Tel: +33 1 45 24 10 54
Fax: +33 1 45 24 11 29
Eml: abdallah.amri@oecd.org