Unclassified **NEA/CSNI/R(2000)23**

Organisation de Coopération et de Développement Economiques
Organisation for Economic Co-operation and Development

OLIS : 08-Jan-2001
Dist. : 23-Jan-2001

English text only

NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA/CSNI/R(2000)23
Unclassified

CSNI/NEA RASPLAV SEMINAR 2000

Summary and Conclusions

14-15 November, 2000, Munich, Germany

English text only

98930

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 27 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.

CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus in different projects and International Standard Problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups and organisation of conferences and specialist meeting.

The greater part of CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

In implementing its programme, CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

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CSNI/NEA RASPLAV Seminar 2000

14.- 15. November 2000, Munich, Germany

Summary and Conclusions

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1 INTRODUCTION

1.1 Sponsorship

The workshop, held on 14th-15th November 2000 at the International Meeting Centre of Science in Munich, Germany, was sponsored by the Committee on the safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA). It was organised in collaboration with Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS).

1.2 Background and Objectives

The objective of the RASPLAV Project was to provide data on the behaviour of molten core materials on the RPV lower head under severe accident conditions, and to assess the possible physicochemical interactions between molten corium and the vessel wall. Data were also obtained to confirm heat transfer modelling for a large convective corium pool within the lower head. The project consisted of the following components:

- To provide data from large-scale integral experiments on the behaviour and interactions of prototypic core-melt materials within the lower head;
- To perform small-scale corium experiments to measure the thermophysical properties (density, electrical and thermal conductivity, viscosity, etc.) required for performing and interpreting the integral (large-scale) tests;
- To determine the uncertainties introduced by using non-prototypic conditions and materials by means of the small-scale corium experiments;
- To carry out the molten-salt experiments with the following objectives:
    - To study heat transfer processes in the melt;
    - To justify the choice of procedures for large-scale experiments, such as the heating method;

- To develop an understanding of relevant phenomena, such as crust formation, and non-eutectic materials behaviour;

- To develop computer tools and models for analysis of results from the large-scale integral tests and the supporting small-scale experiments.

During the first phase of the RASPLAV Project (1994-97) the large-scale experiments demonstrated clearly that behaviour of corium melts differed from that of simulant materials. Under certain conditions, the corium would separate into two layers that were enriched in zirconium or in uranium.

The second phase of the RASPLAV Project started in July 1997 and concentrated on exploring the physical and chemical phenomena occurring in a convective molten pool. The effect of different corium compositions, the potential for and the effects of material stratification and the influence of various boundary conditions were investigated. The work involved a combination of integral and separate effect tests including molten-salt tests to investigate non-eutectic mixtures and the effects of stratification, extension of the material property database to allow interpretation and modelling of the experimental data.

The CSNI decided to hold a seminar where the major outcome of the RASPLAV Project could be presented and discussed also in the context of other experienced activities on Severe Accidents. The objectives of the seminar are:

- to review the experimental results of the RASPLAV Project
- to exchange information on complementary research
- to discuss the progress made on understanding severe accident progression
- to discuss the applicability to nuclear power plants and use of the results

1.3 Organisation of the Seminar

The Seminar was intended to provide an in-depth review of the RASPLAV Project in terms of the technical capabilities, results and analyses produced during the project execution. The application of the results and their significance for power plant applications were addressed. Relevant results of the complementary research carried out at various laboratories were also presented.

The seminar consisted of five sessions organised as follows:

- Opening and overview
- Experimental results
- Theoretical Analyses
- Application and complementary research
- Conclusion

The agenda with the actual timing and presenters is attached (Att. 1).

An Organising Committee (OC) has been nominated to prepare the technical content of the seminar, organise the seminar and prepare the summary record.

For the meeting 80 persons were registered. It was attended by 70 participants from OECD (2), from 14 OECD countries and one non-OECD country: Germany (21), France (12), Russia (10), Finland and Czech Republic (5 each), USA (3), Italy, Japan, Sweden Switzerland (2 each), Belgium, Canada, Hungary, and Korea (1 each).

The list of registered delegates is attached (Att. 2).

Two welcomes and 22 presentations were given. The abstracts of the presentations are attached (Att 3). The papers of these presentations as well as the "Application Report" were distributed at the end of the seminar on CD-ROM. They are available for the members of the RASPLAV and MASCA project by http://www.nea.fr/download/rasplav/. User names and passwords are provided by the NEA secretariat.

2 SESSION SUMMARIES

2.1 Opening and Overview

After organisatorical matters the seminar was opened by **Frescura** as representative of OECD/NEA. He welcomed the participants and gave a short review over the historical background of the project. The second welcome was given by **Birkhofer** as representative of GRS. He reflected on the situation of the two universities in Munich, which provided the localities for the seminar in the International Meeting Center of Science, and the cooperation between the research centers in Western countries and Russia.

After these two welcomes three presentations illustrated how the project fits into the common effort to close the issues and what the major outcomes have been.

Sehgal discussed the remaining issues and concerns with different importance to severe accident safety. In the following a prioritization is given, based on the rationale that the highest priority should be given to those research areas which represent a high safety risk and for which insufficient knowledge-base has been acquired so far (CSARP meeting 2000). The listing is subdivided for issues to the current plants and to the future plants.

As first priority issues for current plants have been selected:

- Ex-vessel debris/melt coolability is essential for timely stabilization and termination of a postulated accident and for assuring the public that this is the case. The current research programs have not reached that goal. New innovative ideas may be needed for assuring ex-vessel melt/debris coolability. Development of a model for melt coolability will require experiments with prototypic materials and simulant materials at different scales.
- Ex-Vessel Steam Explosions can lead to early containment failure for some BWRs and possible leakage in the containments of some PWRs. There is a connection between ex-vessel steam explosions and coolability. Lack of the former can provide the credible accident management option of establishing a pool of water under the vessel and forming a

coolable particulate debris bed. Recent data have shown that oxidic corium may be resistant to triggering and propogation of steam explosions. This may be the key to the resolution of the steam explosion issue. Thus, a fundamental understanding of these observations is essential, in particular with ex-vessel conditions of low pressure and high subcooling.

- The Basemat Failure issue is important for those plants where the access of water to the ex-vessel melt/debris is not available. Basemat failure may imply contamination of ground water supplies and spread of radioactivity to the environment. The technical issue is that of the prediction of the basemat failure time due to the long term multidimensional erosion of the concrete basemat.

As second priority issues for current plants have been selected:

- The mode of lower head failure is needed for specification of the initial conditions of melt discharge for containment loadings, in particular for the ex-vessel steam explosion analyses. The timing is important for the feasibility of SAM measures to prevent vessel failure.
- Core Quenching refers to the accident management actions of water delivery to the vessel for (1) flooding a damaged but not yet relocated core (ii) flooding the lower head when an oxidic melt pool, covered by a metallic layer, is present. The former action may produce more hydrogen and the latter could produce stratified steam explosions.
- The Iodine Chemistry is investigated in the PHEBUS FP Project. Data indicate formation of organic Iodine, which may require additional systems for removal. Its presence can increase the environmental release in case of leaky containments and filtered-vent releases.
- Instrumentation and diagnostics to identify for the operator the progression of the accident, will facilitate proper accident management actions.

As third priority issues for current plants have been selected:

- Steam Generator Tube Failure is of concern for the high pressure scenarios and for aged steam generators. The accident management actions are to flood the secondary side and reduce the primary pressure. Evaluation of this bypass sequence should be completed with the effects of the accident management measures.
- Regarding the Hydrogen Mixing and Combustion much work has been performed. Further evaluation and validation of the calculations for mixing and distribution of hydrogen in the containment compartments, coupled with containment thermal hydraulics, is needed.
- A serious effort is required in updating the existing codes to incorporate new knowledge gained and to validate them against new data obtained.

As first priority issues for future plants have been selected:

- Much work has already been performed for In-Vessel Melt Retention (IVMR). The remaining questions are on (i) the effects of melt stratification on vessel wall thermal loading (ii) the composition of the metal layer and its effect on the focussing of the heat flux, (iii) the reliability of the gap cooling mechanism and (iv) the plant maximum power level that can be reliably certified for IVMR. Some of these questions need additional experiments, while others need evaluation type research.

- Core Melt Spreading and Retention in a Ex-Vessel Core Catcher is the severe accident management scheme, which has been employed for the EPR. The main uncertainty is in the process of retention in a crucible for mixing sacrificial material in the corium melt and its subsequent failure. It is necessary to generate high flow rates in order to assure spreading over the whole surface area of the core retention device. The other uncertainty is in the long term cooling of the spread melt. The remaining research is of evaluative type.
- Core Melt Retention in an External Vessel is the concept of having the core melt of a large power LWR discharge from the vessel into a much larger diameter steel vessel housed in the containment below the reactor vessel. This external vessel may be lined by a ceramic material and is cooled by water. This concept has been promoted by Westinghouse-Atom for future BWR design. Evaluation of this and similar concepts is needed.
- Innovative Ex-Vessel Melt/Debris Coolability Concepts are new concepts for stabilizing the core melt in the containment. A prominent concept is that of adding water to the melt layer from the bottom. This concept appears to help in cooling and quenching even relatively deep layers of melt. For this concept only evaluation work is needed. Another concept is of employing downcomers which have been shown to increase the dry out heat flux in particulate debris beds. Further experimental and analytical research is needed for this concept. Other innovative in-vessel and ex-vessel melt stabilization and coolability concepts have been advanced. Some of these may need extensive research investigations.

Asmolov gave a brief overview over the RASPLAV Project. He stated that this project provided important data on the thermal hydraulic behaviour of simulant and corium materials, corium chemistry and corium properties in a wide temperature range. He summarized the main findings as follows:

- The main feature of the sub-oxidised corium obtained from the different scale corium tests was the occurrence of the Zr-based liquid ZrO_x in the$(UZr)O_{2-x}$ solid matrix in the temperature range between solidus and liquidus.
- The amount of liquid decreased with the increase of zirconium oxidation degree. The behaviour of sub-oxidised corium (two-phase system) in the temperature range between solidus and liquidus determined significantly the possible stratification phenomenon due to competitive processes of uranium dioxide dissolution by molten zirconium and macro mass transfer processes. The fixing of the upper layer by high temperature zirconium carbides prevented complete dissolution of the upper layer.
- The later experiments and analysis showed that the stratification phenomenon was featured only for corium containing some amount of carbon. Special series of small scale experiments performed with carbon free coria did not reveal any stratification into two liquids regardless to the oxidation degree and did not confirm the boundaries of the miscibility gap (MG). Those experimental series allowed to conclude that corium compositions of practical interest representing the mixtures of uranium dioxide and zirconium do not fall in the miscibility gap if the zirconium oxidation degree exceeds 10%. Stratification of the sub-oxidised corium containing 0.3 – 0.4 wt.% of carbon can be

explained by the chemical reactions between zirconium and carbon, the movement of ZrC, $Zr(CO)_x$ and ZrO_x agglomerates due to the gravity, buoyancy force or their carry-over due to gas lift.

- Tests performed with completely oxidised coria did not featured any peculiarities in the thermochemical behaviour. The thermal hydraulic behaviour of the corium melt pool seems to be similar to the behaviour of simulant fluids. This is true for the completely oxidised C-100 corium and coria which do not contain any additives which potentially may induce the stratification as it was observed for carbon containing melts.
- Development of the melting process was followed by the convection heat transfer which was observed in all tests. This conclusion was proven by the heat flux distribution through the test wall and by the shape of the interface boundary between solid and liquid phases.
- No transition area corresponding to the temperature range between solidus and liquidus was found at the interface boundary between the solid and liquid phases of the corium when steady state cooling was applied. Special studies of heat and mass transfer processes performed with a binary non eutectic salt mixture confirm this conclusion. It was found that the solidification and remelting processes were governed by the diffusion of one of the components of the melt. A similar phenomenon was also observed in the corium mixtures when insignificant (1.5 times) change of U/Zr ratio along the C-100 ingots height was found in the post test examination. Correct predictions of heat and mass transfer require consideration of diffusion process. Integral heat transfer in case of steady state regime correlate well with the results of tests with eutectic mixtures.
- Molten iron interacted with corium metallic zirconium in both liquid and solid phases forming a number of easily melting eutectics. The oxide part of corium and iron melt slightly interacted (up to 5 at.% of iron) and formed practically immiscible phases. Small amounts of metallic material additions (Fe, Mo, W, Ta) in completely (C-100) and partially (C-32) oxidised coria behaved very similarly. They formed metallic alloys with different element ratios and absorbed partially Zr from corium and small amount of oxygen as well. Small amounts of La_2O_3 were distributed very similar to the uranium distribution regardless of zirconium oxidation degree.
- The simulation model developed for the heat transfer processes in the fluid was validated against different tests including experiments with water and salt. The main phenomena such as melting, temperature behaviour, heat flux distribution were simulated reasonably in case of one component liquid.

He mentioned that some important issues are still open. Among those are:

- The composition and temperature conditions which may lead to stratification of the molten pool;
- Partitioning of fission products (FP) and decay heat associated with the stratification.

These issues may change the heat flux imposed on the cooled reactor vessel wall and influence the in-vessel retention concept. The follow-up MASCA project will provide additional data concerning these issues and thereby expand the knowledge of corium behaviour and interactions obtained during the RASPLAV Project.

Strizhov presented a paper dealing with the overview of analytical activities in the frames of RASPLAV Project. During the feasibility studies several issues such as choice of heating method and geometry have been studied using analytical methods. That required the use of complex CFD codes accounting for the magnetic field behavior. The choice of slice geometry and side wall heating method required the similarity of heat transfer with the volumetrically heated pool. Such a conclusion was later justified by the special series of salt tests. Complexity of the processes and unique character of each test required extensive analysis for successful conduction of the test. A three-dimensional CFD code has been developed to simulate the complex heat transfer processes in the RASPLAV facility. Codes have been extensively validated and successfully used during the project. Furthermore the codes were improved for the simulation of physico-chemical phenomena. Thus from the beginning of the project analytical studies played an important role for decision making. The balance between theoretical and experimental activities determined significantly the success of the RASPLAV Project. Experience gained is significantly used in the development of codes for IVMR problem and core-catcher design.

2.2 Experimental Results

This session contained 8 papers, four out of these were on the experimental facilities and the results obtained in the RASPLAV Projects and the other four described research results obtained at other institutions on experiments concerned with subjects which complemented the work performed in the RASPLAV Project. A short discussion followed at the end of the papers in this session.

The first paper was presented by **Kiselev** describing the construction of the facilities and their technical characteristics employed during the RASPLAV Project work from 1994 to 2000. The facilities range from STF, the small tungsten facility in which 1 to 2 kg of corium can be heated to ~ 3100 °C to the RASPLAV-AW-200 facility in which the large scale experiments employing 200 kg of corium were performed. He described the details of the facilities.

Merzlyakov presented the paper on the physical property measurements for the high temperature corium melts. In general, this work provided the data base of physical properties for the various corium compositions used in the RASPLAV experiments, viz. C-22, C-32, C-100 and also for C-50. These are first of a kind measurements. The data measured were for kinematic viscosity, surface tension, thermal conductivity, liquidus-solidus temperatures and density. Data were also measured for thermal conductivity of the fluoride salts employed in the RASPLAV-Salt test program.

Degaltsev provided the presentation on the material studies. He presented many results and explained each one carefully. Much work on material studies was performed, which employed reasonably accurate instrumentation. He showed details of his post-test examination work that provided most of the information that was obtained from the RASPLAV tests. He also described that the RASPLAV AW-200-1 and AW-200-1 tests showed stratification due to the presence of ~0.3w% carbon and when carbon was removed, the stratification was not observed. However, molten Zr separated downwards in some tests.

The fourth paper was presented by **Strizhov** on the results obtained with the salt tests. The initial purpose of the salt tests was to show that the side-wall heating did not differ greatly from the direct electric heating (DEH) in terms of the heat flux experienced by the vessel walls. The measurements showed that indeed this was the case. The other tests investigated the crust formation and its effect on the heat transfer. It was found that in the thermal hydraulic steady state the interface temperature was the liquidus temperature of the remaining liquid after the segregation. A series of tests was performed in which crust was formed, melted and refrozen. Analyses of the tests were performed with CONV-2D and good agreement was suggested.

Bechta presented his paper on corium-steel interaction performed using the RASPLAV-2 facility. In the facility, a small corium melt pot was provided in a cold crucible and specimen such as a steel rod was inserted into the melt pot. Post test calculations were performed using the FEA.PL code to examine the several parameters involved, such as heat flux at interface. Replying to one question, the observation was made, that ablation of iron probes was accelerated when the probe surface temperature exceeded 1100 °C

Powers introduced his model to predict the partitioning of fission products (FPs) among core debris phases based on chemical reactions for a wide range (16) of FP elements. The developed model is a frame ready for further upgrade and currently provides only trend results. Example calculations at 2600 K, as functions of oxygen partial pressure were presented. He commented that for future quantitative estimation the model provided should be validated against the experimental data taking the oxygen partial pressure as one of the controlling parameters. Replying to a question, he suggested that the model should be applicable also to steam environment.

Seiler summarized results of four types of French experiments performed in support of in-vessel melt retention (IVMR). First was ISABEL-miscibility gap experiment to identify liquid/solid interface conditions and to demonstrate the miscibility gap in O-U-Zr system. Second was ISABEL-Vessel experiment to clarify physico-chemical interaction between metallic corium and vessel steel using electron beam heating at a heat flux of ~1 MW/m2. Third was SULTAN experiments for external coolability of vessel under natural convection two-phase flows. Fourth was BALI experiment to simulate corium pool convection in lower plenum and to study heat transfer characteristics including the focusing effect

The final presentation was made by **Theerthan** on the SIMECO experiments to study the effect of melt pool stratification on vessel wall thermal loads. He indicated the experimental results that high heat flux on the vessel wall was observed when stratification occurred compared to non-stratified cases. The data further indicated a Richardson number ~ 5 as critical value for stability of the interface between layers. A comment was given by Prof. Sehgal that the test results obtained were compared to CFD calculations performed separately for the stratification and focusing of the maximum thermal loading even though Ra number in the CFD calculations is lower.

A short **discussion** was held after these eight presentations. An estimate of critical thickness for focusing effect was asked by Trambauer. The prediction would be difficult as the condition depends strongly on the scenario. It is difficult to be determined from thermal-hydraulic approach only. The focusing effect could be weakened by 2-D effects. It was also pointed out that composition of stratified upper layer is important, since any mixing of steel with oxidic materials will increase its liquidus temperature and increase the radiative heat transfer. Sehgal declared this question an open issue.

2.3 Theoretical Analyses

There were four papers in Session 3. The first one, given by **Aksenova**, discussed the development, validation and application of the CONV codes at IBRAE in support of the RASPLAV experimental program. The second paper, presented by **Strizhov**, discussed the major results achieved during the six years of the OECD RASPLAV Projects and highlighted the continuation of work within the OECD MASCA Project. The third paper was presented by **Seiler** and **Froment** regarding the interpretation and application of RASPLAV results at CEA. **Müller** discussed in the fourth paper the salt test analysis efforts with a CFD method at GRS.

The technical value of this session was in the validation the analytical tools using results from the RASPLAV salt and corium tests, and in the interpretation of various observed phenomena such as crust behaviour, pool separation, and pool stratification.

The salt test results were utilized in all four papers. The first paper showed results of comparison between side-wall heating and direct heating. Comparison of cases with crust versus cases without crust as well as comparison of eutectic and non-eutectic salts were discussed in the second paper. The third paper pointed out the important result that the liquidus temperature of the residual liquid, and the absence of the mushy zone, can be really applied as the boundary temperature for the oxidic pool when performing heat transfer calculations under thermal hydraulic steady state. One consequence is that, for non-eutectic materials, the final steady state temperature distribution in the melt depends on test procedure (demonstrated by the comparison of the RASPLAV salt tests and the SIMECO salt tests), since the chemical composition of the crust and liquid pool becomes different, and thus also the pool liquidus temperature is different. The fourth paper pointed out the importance of fundamental heat transfer physics considerations and importance of the obtained RASPLAV salt test results for the code analysis validation.

The corium tests were the object of the macroscopic mass and heat transfer analyses (both pre-test and post-test calculations) applying the CONV codes. The CONV codes have proven to be an adequate tool for the planning and control during the RASPLAV corium tests. The development of CONV codes includes the extension of applicability of the code to higher Rayleigh numbers, implementation of physico-chemical models and arbitrary geometry.

Strizhov and Froment (second and third papers) also addressed the problem of corium stratification. The corium separation in two layers observed in the tests performed with C-22 corium containing carbon, is not due to a miscibility gap, since carbon free corium with oxidation degree beyond 20% is outside the U-Zr-O miscibility gap. Below the liquidus temperature, there is partial separation of the metallic liquid through gaps, cracks and porosities as observed in carbon free C-22 or C-32 corium tests. However, in case of carbon free corium, this phenomenon seems not to be sufficient to promote the transfer of the metal to the surface of the oxidic material. Two driving forces have been proposed, density differences and CO-gas bubbles entrainment.

Froment showed, by means of calculations, that the microsegregation (U/Zr distribution in the oxidic phase) is linked to solidification processes of the oxidic phase after power shut-down.

2.4 Application and Complementary Work

Seven papers were presented in this session. The first (presented by **Fichot** and **Zvonarev**), third (given by **Kawabe**) and fourth paper (presented by **Nakamura**) dealt with validation of models in the codes ICAREII, COSMO and CAMP against the RASPLAV corium and salt experiments.

Such experiments have been found to be extremely valuable to model developers. These validations have also shown good agreement with experimental results.

Corium material properties obtained in the RASPLAV program have added valuable information to material properties data base and were used by the investigators of the first two presentations. In the second presentation, given by **Kymäläinen**, the impact of knowledge coming from RASPLAV for IVR in Loviisa plant was discussed.

The second part of this session was devoted to phenomena with application to in-vessel cooling and vessel failure when not cooled.

Kim presented the SONATA program and discussed formation of gap between aluminum oxide/iron crust and vessel wall, permitting water ingression into the gap, therefore, cooling the vessel. In this 1/8-scaled test set-up formation of gap and cooling was demonstrated. It should be mentioned that the feasibility of gap cooling has not been demonstrated for reactor size lower head. The last two papers discussed experimental findings on creep rupture behavior when the vessel is not cooled. **Theerthan** presented data on the multiaxial creep behavior of French reactor pressure vessel steel in experiments of the FOREVER program, which maintained melt pool convection inside a ~25 bar pressurized 1/10th scale vessel. Maximum temperatures of ~1000°C were maintained and vessel failure was achieved. Future tests will attempt gap cooling. The last presentation, given by **Chu**, about the OECD-Lower Head Failure Project, clearly indicated the need for additional data on creep properties for vessel steel and additional experiments on vessel creep failure (e.g., with penetrations and non-symmetrical).

In the subsequent discussion the importance of boron, which behaves like carbon, on the molten pool behaviour was emphasized by Schwarz. It was questioned if Boron effects will be studied in MASCA Project.

2.5 Final Discussion

The summaries of sessions two, three and four were presented by the respective chairmen.

In connection with the summary of the third session, **Hache** re-iterated the physico-chemical behaviour regarding macro stratification (layer separation). During the discussion, he proposed (on the basis of recent GEMINI calculations including carbon in the database) that solid zirconium oxycarbides could form at rather low temperature. These zirconium oxycarbides ($ZrC_{1-x}O_{y<x}$) do not melt when the temperature increases and should not be dissolved by the oxidic phase. Thus, these oxycarbides could separate by density effects after the melting of the oxidic phase. This explanation supposes that thermodynamic equilibrium is achieved over the whole

melting process, which is different from the previous explanations. Boron is likely to behave like carbon and there is more boron than carbon in reactors. The effect of steel on stratification will be studied in the MASCA program. A further comparison with test results in terms of volumes of phases, compositions and temperatures is recommended.

Before the general discussion, **Tuomisto** presented the major recommendations of the report "Application of the OECD RASPLAV Project Results to Evaluations at Prototypic Accident Conditions", which represents the knowledge of 1998:

Concerning the applicability of the RASPLAV results to severe accident modelling, it is recommended that two main areas of further work be pursued:

- completion of the RASPLAV Project and related experiments to solve physicochemical aspects of convective corium pool behaviour; and
- continued refinement of the models and codes as the results of future experimental work become available.

Also the issues to be resolved by further experiments have been identified in this application report. Some of these have been resolved already, some of them still remain:

- The effect of melt stratification on thermal loading of the vessel walls must be quantified before the more complex issues of corium thermochemistry as a function of composition can be addressed.
- It is recommended that thermodynamic analyses of the small-scale and medium-scale corium tests are performed for real corium compositions to establish
  - whether melt stratification is caused by liquid immiscibility and
  - whether thermodynamic equilibrium was achieved in these experiments.
- It would also be important to establish whether prototypic corium compositions are likely to fall within the miscibility gap before launching a large program on corium thermochemistry. This would involve examination of hypothetical melt relocation scenarios. The task is very challenging because of the large variety of reactor types and accident progression scenarios.
- It is recommended that further tests be performed with non-eutectic salt mixtures in the recently modified molten-salt test facility.
- The molten-salt test facility could also be used for experiments with a stratified molten pool containing a thin metallic surface layer.
- The physical property measurements should be continued and their relevance to severe accident phenomena established. For successful modelling of reactor accidents, the following material properties should be known for corium as a function of chemical composition: heat capacity, thermal conductivity, viscosity, density, emissivity and surface tension.
- The critical properties governing prolonged in-vessel retention of molten corium are probably density and thermal conductivity. The densities of different corium compositions may significantly influence the extent of stratification, and thus, the local heat flux distribution.

- The corium surface tension could also play an important role in controlling relocation behaviour, and thus the likelihood of steam explosions.

Furthermore the long term code developmental needs are discussed in the application report:

- Additional code development work is still needed to address the full range of likely accident scenarios. The objectives of this work are twofold:
  - the applicability of computational fluid dynamics (CFD) codes should be extended to
  - the reactor conditions and
  - analysis methods should be further developed to simulate late phase core degradation
  - and the transient heat-up process in the lower head.

- The code development recommended for verifying the in-vessel melt retention concept can be divided into four main activities:
  - melt pool thermal hydraulics,
  - corium chemical behaviour and interactions,
  - mechanical response of the reactor vessel, and
  - upper crust stability.

- To simulate corium convection, the applicability range of the codes needs to be extended from the current Rayleigh number limit of $\leq 10^{14}$ up to values of 10^{16} - 10^{17}. In turn, this will require development and assessment of a turbulence model. At present, there is no generally accepted turbulence model for studying heat transfer from the molten corium pool to the vessel wall.

- Corium thermochemical models cannot be developed further, before the controlling phenomena in the RASPLAV experiments have been identified. As a first step, the existing codes (based on thermal convection, and validated for Rayleigh numbers of $\leq 10^{14}$) could be used to simulate heat transfer in a melt pool with an assumed level of stratification.

- The ultimate goal for code development can be set as to predict correctly corium thermochemical behaviour, such as crust formation, uranium and fission-product partitioning between the different layers and phases, and intermetallic exothermic reactions between zirconium and steel.

- It is important to understand the mechanical behaviour and failure mode of the RPV lower head, i.e. failure time and location as well as the crack rupture size, and to improve the modelling of vessel creep rupture. The effect of stress redistribution is currently being examined in the EC-FOREVER and the OECD Lower Head Failure programmes.

- Realistic melt relocation predictions for the late-phase core degradation process are necessary for decreasing uncertainties and for defining conditions within the lower plenum. The predictions should account for the internal vessel structures and hence, they should be plant-specific.

Asmolov re-iterated in the **final discussion** some points that had emerged during the seminar. He fully agreed with the observation, that for practical compositions of steel free corium, the miscibility gap is not an issue, even if there might be uncertainties of the composition. He also pointed out that sub-oxidized carbon containing corium leads to stratification. This process is initiated by the separation of metallic liquid from the ceramic components $(U,Zr)O_{2-x}$. The MASCA program will focus on conditions under which this may occur.

Seiler and Hache pointed out that the effect of iron is important and should be addressed in future tests. Seiler also explained that debris porosity may be important for the formation of molten pool and stratified metal layers.

Powers noted the importance of in-vessel melt retention in the US as related to advanced reactors and the need to answer regulatory questions on some aspects. He stated that the RASPLAV Project has produced relevant material data and good heat transfer correlations in well defined conditions. However, in actual reactor conditions there are materials slumping from the core region into the molten pool in the lower plenum, thus disturbing the convection. He also stressed the importance of the presence of steel as this is relevant for reactors and has important effects. Finally he referred to the need of incorporating experimental severe accident findings into the accident management codes, and that doing this effectively represents a challenge in itself.

Birchley said that efforts should concentrate on key technical questions that are essential for safety. He said that for accident management measures, transients effects that have durations of hours are important. He added that in-vessel water addition can produce pressure and thermal stress. As a last point, he agreed that core materials dropping into the vessel lower plenum may affect stratification to a substantial degree.

Tuomisto informed that the Finish TVO has filed an application for the fifth nuclear power unit, involving different concepts. Some of them consider in-vessel retention strategies for which RASPLAV and MASCA data will be very relevant.

Finally Strizhov informed on the rapid development taking place in the analytical work he is carrying out in support of, and as a result of the RASPLAV experimental work. He also emphasised that this analytical effort is being done such that the code will have direct applicability to reactor cases.

APPENDIX 1

AGENDA

RASPLAV Seminar 2000

Program Review Meeting of OECD RASPLAV Project
14th - 15th November, 2000
Munich, Germany

First day *Tuesday:* *14th November, 2000*

08:30		Registration
Session 1:	**Opening and Overview**	Chairmen: Trambauer &Vitanza
09:00		Opening and Organisation / Trambauer & Dumm
09:10	*Frescura*	Welcome by OECD
09:20	*Birkhofer*	Welcome by GRS
09:30	*Sehgal*	Remaining Issues in Severe Accident Safety Research
10:00	*Asmolov*	RASPLAV Project Major Activities and Results
10:30	*Bolshov*	Theoretical Studies and Activities
11:00		Coffee Break
Session 2:	**Experimental Results**	Chairmen: Sehgal & Nakamura
11:30	*Kisselev*	RASPLAV Facilities Description
12:00	*Merzlyakov*	Physical Properties Measurements of High Temperature Melts
12:30		Lunch Break
13:30	*Degaltsev*	Material Studies
14:30	*Strizhov*	The Results and Analysis of the RASPLAV Salt Test
15:10		Coffee Break
15:40	*Bechta*	Experimental Studies of Ceramic Corium Melt Interaction with Steel
16:10	*Powers*	Partitioning of Uranium and Fission Products among Condensed Core Debris Phases

16:30	*Seiler*	Experimental Results obtained in France in Support of In-Vessel Retention
17:00	*Theerthan*	Experiments on the Effects of Melt Pool Stratification on Vessel Thermal Loads in the SIMECO Facility
17:30		Discussion
18:00		Adjourn

Second day Wednesday: 15th November, 2000

Session 3: Theoretical Analyses Chairmen: Hache & Tuomisto

08:30	*Aksenova*	Development and Application of the CONV Codes
09:00	*Strizhov*	Major Outcomes of the RASPLAV Project
09:30	*Seiler & Froment*	Interpretation and Use of RASPLAV Results in Support of In-Vessel Retention
10:10	*Müller*	Post Test Analyses of RASPLAV Salt Test Series 3 and 4
10:30		Discussion
10:50		Coffee Break

Session 4: Application, Complementary Work Chairmen: Behbahani & Dienstbier

11:20	*Fichot & Zvonarev*	Use of RASPLAV Results in IPSN Severe Accident Research Program
11:40	*Kymäläinen*	Confirming the In-Vessel Retention for the Loviisa Plant
12:00	*Kawabe*	Analysis of Turbulent Natural Convection Heat Transfer in a Lower Plenum Cooling Using COSMO
12:20	*Nakamura*	Validation of Camp Code for Thermofluid-dynamics of molten Debris in Lower Plenum
12:50		Lunch Break
13:50	*Kim, S.B.*	Major Results of In-Vessel Corium Cooling in the SONATA-IV Experimental program
14:20	*Theerthan*	Coupled Melt Pool Convection and Vessel Creep Failure: The EC-FOREVER Program
14:50	*Chu, T.Y.*	Mechanical Behaviour of Reactor Vessel Lower Head During Late Phase of Reactor Accidents

15:20	Discussion
15:40	Coffee Break
Session 5: Conclusion	Chairmen: Kollath & Asmolov
16:00	Summary of Session 2
16:10	Summary of Session 3
16:20	Summary of Session 4
16:30	General Discussion
16:50	Concluding Remarks
17:00	Adjourn

APPENDIX 2

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APPENDIX 3

ABSTRACTS

1 SESSION 1: OVERVIEW

1.1 Remaining Issues for Severe Accident Research

B.R. Sehgal

Severe Accident research, active worldwide, has pursued the resolution of the myriad safety issues that the postulated severe accident scenarios have posed. The objectives of the research have been to obtain sufficient knowledge-base to (1) assess the risks posed from the various phases of a severe accident scenario and (2) devise accident management measures and assess their efficiency.

The research efforts have made remarkable advances: much knowledge-base has been accumulated in the short span of 20 years, and many important issues as e.g., in-vessel steam explosion, direct containment heating, Mark 1 liner attack have been resolved. The severe accident phenomenology, however, is very complex and sufficient advances have not been made in resolution of some of the previously identified severe accident issues. Other factors have also come into play, recently, e.g. the focus on maintaining the integrity of the vessel, and the requirement by some regulatory authorities to consider the impact of severe accidents in the design and operation of new plants.

This paper will describe the progress of the severe accident research with a brief description of the advances made and those expected from the on-going research efforts. It will describe and prioritize the remaining issues, and concerns, for which additional research results are needed. Finally, research efforts needed to resolve these issues will be described.

1.2 RASPLAV Project Major Activities and Results

V. Asmolov

The RASPLAV Project objectives were realised through a combination of several major activities that included:

- Development of technology for conduction of the high temperature corium tests including design, fabrication and assembly of large-scale facility capable to heat up and sustain molten corium at relevant temperatures;

- Performance confirmatory large-scale experiments to investigate the behaviour of the melt of the core prototypical materials in the reactor vessel low head as well as to determine principal differences in the behaviour of the melt as compared with simulating liquids;

- Performance of supporting corium experiments and analysis to conduct the integral test and develop the methodology describing the phenomena of interest, among the objectives are also measurements of material properties;

- Performance of salt tests to support the development of facility and facilitate the analysis and interpretation of corium tests;

- Development of the computer tools for the pre- and post-test analysis of experiments to produce a consistent analysis, interpretation and understanding of the results

- General outcomes are presented in the paper

1.3 Theoretical Studies and Activities

L. Bolshov and V. Strizhov

Methodology of severe accident analysis includes several important issues which can not be considered as independent. It is difficult to overestimate the importance of computer simulations of the severe accident phenomena, which provide finally the overall knowledge on the accident progression, fission products release and risk for population. Those codes accumulate the results of experimental efforts and allow extrapolating the results to the reactor scale. Large-scale tests with prototypic materials and conditions serve for confirmation of major findings of those studies.

Analytical activities in different countries are concentrated around the development of codes for analysis of severe accident phenomena. However, the level of modeling in the codes is usually much below the level of knowledge. Approach based on the physical simulation of phenomena is being developed at the IBRAE for a long time. Such a development is possible only if the work in code development is well correlated with the experimental efforts. SVECHA code package accumulated contemporary knowledge on the core degradation phenomena and was extended to the simulation of the quench phenomena. This code package was extensively used for FZK quench tests and allowed to predict important phenomena observed in the tests.

Balance between theoretical and experimental activities determined the success of the RASPLAV Project. Complexity of the processes and unique character of each test require extensive analysis for successful conduction of the test. Developed codes use a computational approach, based on two- or three-dimensional fluid dynamics, to simulate the complex heat transfer processes occurring between the major components of the RASPLAV facility.

Summarizing experience gained in the area of severe accident phenomena modeling one may say that the objectives in code development activities for simulation severe accident phenomena fall in the area of realistic modeling which obviously need a best estimate approach to simulation. Russian National program in code development emphasize the development of realistic codes of the best estimate level. The program contains several parts: development of the user environment; development of models and codes; analysis of experimental database and determination of experimental needs for code validation. Status and plans in the code development activities is presented.

2 SESSION 2: EXPERIMENTAL RESULTS

2.1 RASPLAV Facility Description

V. Asmolov, E. Dyakov, S. Abalin, I. Isaev, K. Pechalin, E. Samarin, A. Surenkov, I. Khazanovich, N. Kiselev

RASPLAV Project was aimed to reproduce accidental conditions in controlled experimental cell to simulate molten corium pool into reactor vessel lower head.

The severe accident parameters required to research and development an experimental technique, diagnostic systems, structural materials and ways of their protection against dangerous interactions.

Within RASPLAV Project extensive set of different scale facilities with unique features has been constructed to investigate corium behaviour in sufficient large melt pool (about 200 kg) at the temperature up to 3000 ^{o}C.

The most important features of the facilities RASPLAV AW200, TULPAN, KORPOS, TF, STF, TIGEL and RASPLAV SALT are described in the paper.

2.2 Properties Measurements of High Temperature Melts

V. Asmolov, S.S. Abalin, A.V. Merzlyakov

The data on physical properties of melts including uranium dioxide, zirconium dioxide and metallic zirconium is needed to describe the behavior of molten corium in case of unfavorable progress of a serve accident in nuclear reactor.

The various physical properties of corium (C-22, C-32, C-50 and C-100) were measured within RASPLAV Project. Corium of different composition was investigated at temperatures up to 2800^{0}C.

The review of experimental data on viscosity, density, electrical conductivity, thermal conductivity and surface tension of corium is presented in the report.

Several molten salt compositions were investigated as model melts. The results of physical properties measurement of molten salts are also presented.

2.3 Material Studies

V. Vlasov, Yu. Degaltsev, V. Zagryazkin, Yu. Utkin

The paper main objective is to present the basic results obtained from the experiments performed within the framework of the RASPLAV Project aimed to study the behaviour of the reactor core real materials in the lower head in the course of accidents beyond the design basis.

The following three experimental methods were employed to investigate the behaviour of materials under accident temperatures:

- Investigation of corium ingots using different methods of material studies;
- Analysis of samples taken from the melt;
- Performance and analysis of supporting small-scale experiments on individual problems.

The most realistic core compositions based on U, Zr, O were chosen to perform the experiments: the U/Zr ratio 1.6 - 1.2 (the area of fuel elements for the VVER and PWR), Zr oxidation degree ranging from 10 up to 100%. In a series of tests, minor additions of the structure materials C, FeO and simulators of fission products Nb, Mo, La_2O_3 were introduced into the corium.

The paper presents the following results:

- The macro patterns (the volume and shape) of the liquid corium were restored in the large- and middle-scale experiments. The obtained results were employed to verify thermal-hydraulic calculation models. The temperature distribution over the areas of the ES side walls was restored from the degree of Wa-Ta and Ta-C interaction to compare that with the calculation results.
- The separation of the liquid and solid phases in the suboxidised corium within the temperature interval T_{liq}-T_{sol} was experimentally revealed. This may play a definite part in the problem of the corium interaction with the vessel lower head as well as of the melt possible stratification.
- A stable stratification of the suboxidised corium with an admixture of carbon (~ 0.3 mass%) into two liquid layers, the upper one enriched with Zr and the lower one enriched with an oxide phase based on U, was experimentally revealed.
- In the suboxidised corium specially purified from carbon (< 0.01 mass%), the stratification of the liquid corium was detected neither in small-scale experiments nor in the AW-200-4 test.
- Some behaviour aspects of minor additions and admixtures in the liquid corium were investigated as well as the distribution of those in ingots.
- The character of the suboxidised corium interaction with solid and liquid steel was studied in some experiments (AW-2.5, AW-200-4, T-6). The interaction takes place with the formation of eutectic compounds.

2.4 The Results and Analysis of the RASPLAV Salt Tests

S. Abalin, I. Gnidoi, V. Semenov, V. Strizhov, A. Surenkov

Complimentary to the RASPLAV corium tests a set of experiments with binary molten salt mixtures has been performed. The objectives of molten-salt experiments were twofold. The first was to investigate the effects of non-prototypic conditions in the corium experiments, such as the use of side-wall heating instead of volumetric heating, and the existence of non-isothermal conditions at the upper pool boundary. The second objective was to extend data base with simulant materials that exhibit modeling of additional phenomena such as crust formation (eutectic mixture), and a large solidus-liquidus temperature difference (non-eutectic mixture).

The RASPLAV salt facility was designed as a slice model of the reactor vessel similar to the approach for corium tests. The facility operated with the molten salts at the temperatures of about $750^{o}C$. Several series of tests have been conducted in an intermediate range of Rayleigh numbers from $9.8 \cdot 10^{11}$ to $3.5 \cdot 10^{13}$ with the variation of Prandtl ranged from 4 to 40.

The tests with eutectic binary salt mixtures have been performed with both sidewall and volumetric heating to study the differences in melt behaviour caused by sidewall heating. A eutectic salt mixture ($8NaF$-$92NaBF_4$) was used to investigate the effect of different heating methods and of crust formation on heat transfer across the test-section wall, simulating the lower vessel head. The heat transfer results obtained using the two heating methods were found similar within measurement errors.

Moreover, three series of tests have also been performed with a non-eutectic salt mixture ($25NaF$-$75NaBF_4$). These were the first tests to investigate the effects of large difference between solidus and liquidus temperatures on heat transfer and have provided valuable data for modeling heat transfer and crust formation from complex mixture.

A methodology was developed to describe the conditions between the solid layer and the liquid**,** based on a thermochemical solidification model. It was shown that such a methodology allowed to simulate crust formation phenomenon from non-eutectic mixture. The model predicts the interfacial temperature between the liquid and solid phases. It was found that the diffusion of high temperature NaF component governed the approaching thermodynamic equilibrium. Post-test analysis performed with the specially developed quasi-steady-state crust formation model showed that such a model can be applied to the analysis and interpretation of experimental data.

2.5 Experimental Studies of Oxidic Corium Melt Interaction with Steel

S.V. Bechta, V.B. Khabensky, S.A. Vitol, E.V. Krushinov, V.S. Granovsky, S.V. Kovtunova, V.V. Martinov, D.B. Lopukh, Yu.B. Petrov, A.Yu. Petchenkov, I.V. Kulagin, V.V. Gusarov

The experimental studies of corium melt interaction with vessel steel 15X2NMFA are presented. The experiments are carried out on the "RASPLAV-2" test facility with different corium compositions having low, medium and high liquidus temperatures in air and inert atmosphere. In this way different interaction patterns are simulated. In tests the melt mass ranges between 1,5 – 2 kg, the melt surface temperature - between 1400-2600 °C.

The presented experimental studies enable to determine certain specific features of oxidic melt – vessel steel interaction.

The paper contains experimental matrix, test facility schematics, experimental procedures including tests with samples submerged into the melt or placed on the bottom. The test and posttest analysis data along with numeric modeling of interaction thermal characteristics enable to put forward the interaction mechanisms and specify effects relevant for the corium retention inside the vessel.

The first-priority tasks for further studies are identified on the basis of current experimental results.

2.6 Thermochemistry of Core Debris: Partitioning of Uranium and fission Products Among Condensed Core Debris Phases

D.A. Powers

Partitioning of fission products between the metallic and oxidic phases can affect heat generation and natural convection in core debris retained within the reactor vessel. A simplified model of fission product partitioning between these phases of core debris is described. The model treats the metallic phase as iron that forms subregular solutions with uranium, zirconium, and elemental fission products. The oxide phase is treated as ionic and is composed of U^{6+}, U^{4+}, U^{2+}, Zr^{4+}, Zr°, O^{2-}, and cations of the oxidized fission products. Example calculations of the partitioning of Ag, As, Ba, Ce, La, Mo, Nb, Nd, Pd, Rh, Ru, Sb, Sn, Sr, Te, and Y at 2600 K as functions of the oxygen partial pressure are presented.

2.7 Experimental Results obtained in France in Support of In Vessel Retention

J.M. Seiler, J.M. Bonnet, K Froment, S. Goldstein, F. Barbier

Different items concerning In Vessel Retention have been experimentally investigated in France. The different experiments will be shortly presented and the main results will be discussed.

The interface conditions of corium melt submitted to residual power dissipation has been investigated. A dedicated PHYTHER experiment allowed to verify the conclusions of the theoretical analysis: in Thermalhydraulic Steady State, the mushy zone disappears and the interface temperature between the liquid phase and the solid phase is equal to the liquidus temperature corresponding to the composition of the residual liquid. This conclusion has also been supported by the results from RASPLAV SALT (KI) and from SIMECO (RIT) experiments.

The thermodynamic behaviour of corium melts containing U, O, Zr, and Fe has also been investigated and some experiments have been performed in order to investigate the extend of the miscibility gap in the U, O, Zr, Fe system and to precise the orientation of the tie-lines (which determines the densities of the liquid phases).

An experiment has been performed in ISABEL, with representative heat fluxes of ~1MW/m², in order to investigate the potential for steel dissolution by Zirconium melts. No consequence of exothermic reaction was observed. Furthermore, the experiments permitted to state that In and Sn do not migrate from the melt into the steel; thus, migration processes from the corium do not affect the mechanical properties of the cold part of the steel.

The thermalhydraulic of corium pools has been investigated in the BALI simulant-material (water) experiment. The scale was the reactor scale. Heat flux distributions have been measured for different pool sizes, residual power level and variable viscosity. The power distribution, which has been found is coherent with the results obtained in other large scale experiments (COPO, ACOPO). However, the absolute heat exchange coefficients are larger than those derived from the ACOPO experiment.

The problem of the focusing effect was especially investigated in an experiment simulating the heat transfer in a stratified metal layer. Results are summarised.

The SULTAN experiments have been performed in order to investigate the potential of external cooling under natural circulation with boiling under inclined surfaces. The aim of this experiment was to establish a data bank (in terms of two-phase flow characteristics and critical heat flux) which can be used for code validation and also be applied in the design of other core-catchers.

2.8 Experiments on the effects of melt pool stratification on vessel thermal loads in the SIMECO Facility

B.R. Sehgal, S.A. Theerthan, G. Kolb and A.A. Gubaidullin

Experiments were performed on convection in a two layer stably-stratified pool, in the SIMECO facility: a semi-circular slice vessel, with internal heat generation in one or both layers. The objective was to study the effect of stratification on the heat transfer to the boundaries of the pool. Effects of miscibility or immiscibilty of the layers and the density difference between the layers were investigated. The stratification with miscible fluids was established using salt water and pure water and that for immiscible fluids was established using pure water and paraffin oil. These experiments were replicated in a numerical simulation, albeit, at lower Rayleigh numbers.

When only the lower layer was heated, for the two miscible layers, the layers mixed completely. The mixing time was directly proportional to the initial density difference between the layers and the height of the upper layer; and inversely proportional to the heat input. The maximum heat fluxes and the maximum temperature in the heterogeneous pool were substantially larger than those in the homogeneous pool. These differences were even larger when the layer was immiscible. The maximum values, when the pool is heterogeneous, always occur just below the interface. The Qup/Qdn increased by a factor of four as soon as the heterogeneous pool became homogeneous as the layers completely mixed. It appears that the additional boundary layers on either side of the interface reduce the flow of heat to the upper boundary, thereby increasing the heat flux to the vessel wall.

When both miscible layers are heated, the two layers mixed completely only when the density difference was less than 5 %. The maximum values of heat flux on the vessel wall and its temperature are, again, higher when the pool is heterogeneous than when the pool is homogeneous as the layers mix. The results from numerical simulation, with variation of the properties for the upper layer, show that the heat flux split depends on the ratio of the conductivities of the two layers.

3 SESSION 3: THEORETICAL ANALYSES

3.1 Development and Application of the CONV Codes

A.E. Aksenova, V.V. Chudanov, V.A. Pervichko, V.N. Semenov, V.F. Strizhov

An important part of the RASPLAV Project was in a code development area. Before the start of the Project the modelling capabilities were limited by the two dimensional approach. The specific features of the RASPLAV-AW facility which utilised the side wall heating technique required the use of a computational software, based on three-dimensional fluid dynamics methods, to simulate the complex heat transfer processes between the major components of the facility (graphite-plate heaters, tungsten protector, thermal insulation and externally cooled test wall). The software, satisfying those requirement, was developed. Application of these codes to the RASPLAV-AW facility design provided valuable information that was used in preparing and performing the experiments. The codes successfully predicted the qualitative behaviour of the melt, and allowed optimisation of the test procedures.
The applicability of the codes was proved by the extensive validation against the results of water and molten-salt tests to ensure that the phenomena of interest are modelled reasonably. More specifically effects of side wall heating were studied using the results of the salt tests. Moreover, simulation of crust formed from eutectic mixtures allowed to validate models for both heating methods.

This paper presents the latest results of numerical simulation of AW-200 experiments, being accomplished at the RASPLAV facility. Moreover, the validation results of CONV codes versus salt experiments are given.

However, the applicability of the codes was validated by Rayleigh numbers about 10^{13}, while for reactor case expected Rayleigh number may exceed 10^{16}. Besides, the CONV2D and CONV3D codes have not incorporated models describing thermochemical phenomena which determined the stratification of the melt pool in large-scale RASPLAV tests. In this case the main problems are linked to adequate simulation of high Rayleigh number flows and accounting for a multicomponent nature of corium mixture.

Therefore further development of codes is carried out in a direction of taking into account of the following features:

- large eddy simulation under high Rayleigh numbers;
- corium stratification on two layers: metalic and oxidic components;
- behaviour of the multi component mixture in the temperature range between solidus and liquidus.

Paper presents also the verification results of newly implemented models.

3.2 Major outcomes of the RASPLAV Project

V. Strizhov, V. Asmolov

The basic objective of the RASPLAV Project was to provide data on the behaviour of molten core materials on the RPV lower head under severe accident conditions, and to assess the possible physicochemical interactions between molten corium and the vessel wall. Those objectives have been met in a series of large scale tests with prototypic materials, and in a number of small and medium scale tests phenomena of interest have been studied.

Previous studies related to the in vessel retention problem provided a vast amount of information about the thermal hydraulic behavior of the simulant liquids up to the prototypic Rayleigh numbers. Salt tests investigated additional effects such as crust formation and non-isothermal upper boundary conditions. Special tests have been conducted to study heat transfer in fluids with large temperature difference between solidus and liquidus. The model developed for interpretation of the non-eutectic salt tests predicts that the solidification front becomes planar under steady-state conditions, hence, convection within the melt pool should not be affected by formation of a transition zone.

To confirm that heat transfer in a corium melt pool is of the same nature, it was necessary to develop and validate the three-dimensional fluid dynamic code. Analysis of experiments allowed to identify appearance of different phenomena and to make conclusions about the heat transfer mechanisms.

In first two large scale tests, corium stratified into two layers, from which the upper layer was enriched with zirconium and the lower layer was enriched with uranium. The primary hypothesis was that thermodynamics of complex multi-component U-O-Zr system is responsible for such stratification. However, later experiments and analysis showed that stratification was observed only when corium contained some amount of carbon. The possible reasons for such stratification are discussed. It is believed that the formation of liquid phases in the temperature range between solidus and liquidus and effects of macro mass transfer processes such as gravity determined separation of different phases in course of heating. In case of carbon containing corium, gas bubbles formed at high temperatures caused the upward movement of the liquid phase. The fixing of the upper layer by high temperature zirconium carbides prevented complete disappearance of the layer.

Further studies will increase the knowledge of the thermochemical behavior of the corium melt and clarify the mechanisms for stratification and separation.

3.3 Interpretation and use of RASPLAV Results in Support of In Vessel Retention

J.M. Seiler, J.M. Bonnet, K. Froment, S. Goldstein

Among all theoretical efforts performed in France on the subject of In-Vessel retention, the paper will focus on the main contributions from the RASPLAV programme: 1) on the Interface Conditions in Thermalhydraulic Steady State, 2) the interpretation of the RASPLAV non-eutectic SALT experiments and 3) on the analysis of the origin of the stratification observed in RASPLAV.

The theoretical analysis has been performed on the basis of the knowledge developed in Metallurgy and Physicochemistry. From this investigation it was clear, since a long time, that the mushy zone disappears when the solidification (or dissolution) rate tends towards zero, which is the case in thermalhydraulic steady state. New models have been developed and the conclusion was that the interface temperature between the liquid phase and the solid phase is equal to the liquidus temperature corresponding to the composition of the residual liquid. The composition of the residual liquid depends on the composition of the solid which, in turn, depends on the transient which leads to the final steady state. Thus, the non-eutectic material behaves, solely front the point of view of heat transfer and in thermalhydraulic steady state with internal heat generation, like a pure material with the difference that the interface temperature may vary as a function of the solidification (or dissolution) processes which took place before reaching the final steady state conditions. Interface conditions for stratified layers with different compositions can be derived from this approach.

One main consequence is that the heat transfer results obtained from (pure) simulant material experiments are pertinent for reactor applications.

The interpretation of the non-eutectic RASPLAV SALT experiments confirmed the preceding conclusions. The solid crust was predicted to be enriched in NaF which has been experimentally confirmed. If plane front solidification models are used, with a partition coefficient equal to zero, then the final conditions under steady state do not depend on assumptions concerning the diffusion kinetics in the liquid phase.

If the material is molten in-place, then the final interface temperature is equal to the liquidus temperature corresponding to the initial composition of the mixture (confirmed by the interpretation of SIMECO).

We concluded from the analysis of the stratification observed in the RASPLAV real material experiments that the miscibility gap may probably not be the main reason for this effect. A potential explanation is thought to be linked to a density driven phenomenon due to the early melting of the metallic phase within the oxidic material, and where carbon plays a role

Other results from the RASPLAV programme (heat transfer correlations, …) do not contradict the results obtained from other programmes.

3.4 Post Test Analyses of RASPLAV Salt Test Series 3 and 4

C. Müller

In-vessel melt retention by internal gap cooling or alternatively by external cooling is investigated in the frame of the German nuclear safety research program. The main points of interest in the evaluation of experiments are

- Determination of the peaking factor of the heat flux
- Verification of correlations for the local distribution of the heat flux from the melt with a special interest in the upward/downward distribution
- Validation of the current approach of interpreting the thermohydraulics of the melt by a Boussinesq approximation with isothermal boundaries.
- Investigation of gap formation

In this paper the Salt Test series 3-4 of the RASPLAV program are evaluated in view of the above topics. First, it is shown by a rigid mathematical analysis that a local Nusselt number may be derived which is different from the usual Nusselt number. Correlation for this new Nusselt number are derived from the measured data and discussed. The direct evaluation of the measured temperature fields and heat fluxes leads to two new parameters, the overheating and the pushup factor. Both parameter show a significant dependence from the volumetric heating.

Since in the Salt Tests only temperature and heat fluxes were measured the fluid flow is investigated by calculations with a Boussinesq code. Questions of fluid patterns, pattern formation, the nature of the secondary flow and its interaction with heat transfer are discussed. The main point of the analysis is the question if and how a secondary flow can be determined from the measured data.

The vertical temperature distribution is analyzed for hints of a presumed gap formation between crust and vessel wall but no indications could be found.

In the final section the question is investigated if contradictions to the hypothesis of the Boussinesq approximation can be found in the measured data and how the findings from these small scale tests with simulant fluids can be applied to reactor conditions.

4 SESSION 4: APPLICATION AND COMPLEMENTARY WORK

4.1 The Use of RASPLAV Results in IPSN Severe Accident Research Program

F. Fichot, V. Kobzar, Yu. Zvonarev, P. Bousquet-Mélou

The results of RASPLAV experimental program were used by IPSN to improve the relevance and quality of prediction of severe accident codes during the late phase of an accident. Three research topics have considered selected RASPLAV data as useful information for development or validation :

- The IPSN material data-base for severe accident codes and the thermochemical code GEMINI have included (or will include) material properties measurement tests and phase equilibrium small scale tests.
- The detailed modeling of multi-component mixture solidification (Ph.D. thesis) will be applied to RASLAV SALT configuration (7^{th} series) and compared to experimental results.
- The validation matrix of IPSN code ICARE/CATHARE will include one large scale test (AW-200-4). Comparison with experimental data will be useful to improve the modeling of molten pool and debris bed behavior.

The main activity with RASPLAV results, up to now, has been ICARE2 code validation. The current version of ICARE/CATHARE code is designed to calculate a wide variety of accidents that may occur in pressurized water reactors. It covers the whole sequence of accident events (except jet fragmentation) up to reactor vessel failure. The behavior of structures in the active as well as in the upper and lower plenum of the reactor vessel is managed by ICARE2 code. The last version of ICARE2 (V3mod1.0) allows to simulate phenomena taking place in the lower plenum at later stages of SFD accidents.

RASPLAV AW-200-4 test was chosen for validation of new ICARE2 models. The test simulates the behavior of molten and solid corium in the reactor lower plenum together with reactor vessel behaviour.

In the present work an attempt was made to simulate all the sequence of events of RASPLAV AW-200-4 test by ICARE2 code. The part of test facility which was modeled includes the corium briquettes and the test-wall. Detailed spatial discretization was used in the calculation. The nodalization scheme included 471 nodes. Typical size of spatial meshes was ~2 cm. Experimental data on material properties were used to tune ICARE2 models.

Performed calculations showed that ICARE2 successfully predicts the most important phenomena taking place in RASPLAV AW-200-4 test including corium heat-up and melting, creation and expansion of molten pool, slump of molten corium, creation of cavern, melt solidification and cool-down. Recommendations on code improvements and modifications were developed to provide fruitful future work on code validation using RASPLAV data base.

4.2 Confirming the In-Vessel Retention for The Loviisa Plant

H. Tuomisto, O. Kymäläinen

Extensive studies on in-vessel melt retention were carried out for Fortum´s Loviisa NPP in the early 1990's. As the Finnish regulations require demonstration of the coolability of molten core, the final report on the IVR studies was submitted to the Finnish Nuclear Safety Authority in 1994. After receiving the regulatory approval of the approch, Fortum has still participated actively in experimental research programmes. Fortum has among other things participated together with other Finnish nuclear organisations in the OECD RASPLAV Project and carried out confirmatory experimenting with the COPO-II facilities. The confirmatory research has helped in reducing several residual uncertainties in the original IVR case. The only major, potentially adverse effect discovered after the 1994 report was the stratification of the oxidic corium found in the RASPLAV experiments. However, in the case Loviisa, with the low power density, the margins to failure are wide enough not to endanger the overall conclusions of the IVR study even assuming a completely insulating layer on top of the oxidic pool.

4.3 Analysis of Turbulent Natural Convection Heat Transfer in a Lower Plenum during External Cooling using the COSMO Code

R. Kawabe, H. Nagasaka, N. Noguchi

The concept of in-vessel retention of molten corium during a severe accident has been studied extensively as an attractive accident management measure. One of the key parameters in maintaining the integrity of lower head is the heat transfer rate from corium to the vessel lower head wall. A computer program COSMO has been developed for the aim of understanding the natural circulation in corium pool and heat transfer under high Rayleigh numbers expected in severe accidents. In the program, mass, momentum and energy conservation equations are integrated by SIMPLEST Method to simulate the thermal energy transport with the fluid flow induced by volumetric heat generation. The k-epsilon turbulence model is incorporated and turbulent Prandtl number is calculated by a correlation based on temperature stratification model. COSMO is validated by the comparison with BALI experiments, where test section consisted of a full-scale slice of lower plenum and water was used as simulant fluid. Calculated Nusselt numbers by COSMO are in good agreement with those of the experiments where Rayleigh number range up to 7×10^{16}. Calculated results by COSMO for natural circulation in PWR severe accident condition are also presented focused on heat transfer distribution at corium pool boundary.

4.4 Validation of Camp Code for Thermo-Fluiddynamics of Molten Debris in Lower Plenum

Y. Maruyama, K. Moriyama, H. Nakamura, K. Hashimoto, M. Hirano, K. Nakajima

An analytical code, CAMP, is being developed at Japan Atomic Energy Research Institute (JAERI) for thermo-fluiddynamics of a molten debris in the lower plenum of the reactor pressure vessel. Validation of CAMP code is in progress by analyzing natural convection experiments with various Ra up to 10^{17}. A low Reynolds number (Re) type k-? two-equation model for turbulent flow and a two-equation model for turbulent heat transfer were added to CAMP code. Analyses were performed for an experiment at University of California at Los Angeles (UCLA) on natural convection of a volumetrically heated fluid in a hemispherical vessel and a RASPLAV-Salt experiment with a side wall heating sliced vessel. A distribution of local heat transfer coefficients along the vessel wall was well predicted by CAMP code for the UCLA experiment. On the other hand, CAMP code failed to quantitatively reproduce a profile of local Nusselt number for the RASPLAV-Salt experiment. Models were also incorporated into CAMP code for the formation of a narrow gap between solidified debris and vessel wall and the water penetration into the gap. In-vessel debris coolability experiments performed at JAERI were analyzed with CAMP code, where a molten aluminum oxide (Al_2O_3) was poured into a water-filled hemispherical vessel. A thermal response of the vessel wall was reproduced using the both models. A coolable depth of the molten debris by the gap cooling was separately evaluated in a reactor scale.

4.5 Major Results of In-vessel Corium Cooling in the SONATA-IV Experimental program

Sang-baik Kim, Kyoung-ho Kang, Jong-hwan Kim, Rae-jun Park, and Hee-dong Kim

A potential for creating gap resistance between the debris bed and the reactor vessel wall and consequent effective cooling mechanism through the gap was considered to support the survival of the TMI-2 vessel in the course of the severe accident. A research program, SONATA-IV (Simulation of Naturally Arrested Thermal Attack In Vessel), has been launched to investigate this inherent cooling nature of degraded core inside the lower head vessel at the KAERI (Korea Atomic Energy Research Institute). As the proof-of-principle tests of the SONATA-IV program, the LAVA (Lower-plenum Arrested Vessel Attack) and CHFG (Critical Heat Flux in Gap) experiments are in progress for the systematic investigation of this proposed in-vessel cooling mechanism since the early 1997. The principal objectives of the LAVA and CHFG experiments are to corroborate a proof of gap formation between the debris and the lower head vessel wall and to evaluate the effect of the gap on the cooling characteristics of lower head vessel wall, respectively.

The LAVA experiments were conducted in the hemispherical test vessel simulated with a 1/8 linear scale mock-up of the reactor vessel lower plenum using Al_2O_3/Fe thermite melt as a corium simulant. A series of tests have been performed to assess the effects of the melt compositions and the coolant conditions on the melt relocation and cooling process in the lower head vessel. No evidence of the lower plenum failure has been found in the 11 tests performed so far, even though a small amount of deformations was experienced due to the internal pressure load to the highly heated vessel wall. The debris was made of the continuous layer of the solidified debris (cake) with a small amount of the debris particles rested on the top. The results of the gap thickness measurement using the ultrasonic pulse echo method address that a gap formed at the interface between the debris crust and the lower head vessel wall with the order of several mm thick. Temperature histories measured at the outer surface of the lower head vessel imply that the enhanced cooling capacity through the gap was highly distinguished in this study. It could be inferred from the experimental results that water penetration into a narrow hemispherical gap together with the steam ventilation via the pores inside the melt layer are the key role on the heat removal through the gap.

The CFHG tests were performed to measure the critical power in the hemispherical gap using distilled water and Freon R-113 by varying experimental parameters of a system pressure from 0.1 to 1.0 MPa and a gap thickness of 0.5, 1.0, 2.0, 5.0, and 10.0 mm. Temperature measurements over the heater surface showed that even if local dryout occurred there existed a quasi-steady state. When the heater power was large enough, the dryout region expanded by itself without an increase in heater power, finally which leaded to a global dryout. The measured values of the critical power are lower than the predictions made by empirical CHF correlations applicable to flat plate gaps and vertical annulus. Based on the visual observations and the comparison with the values converted from the CCFL (counter-current flow limit) data, it is assumed that a CCFL brings about local dryout and finally, global dryout in hemispherical narrow gaps.

4.6 Coupled Melt Pool Convection and Vessel Creep Failure: The FOREVER Program

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The FOREVER (**F**ailure **O**f **RE**actor **VE**ssel **R**etention) program is concerned with the phenomena of melt-vessel interactions during a postulated severe accident in a light water reactor. The objectives of the FOREVER program are to obtain data and develop validated models on (i) the melt coolability process inside the vessel, in the presence of water (in particular on the efficacy of the postulated gap cooling to preclude vessel failure) and (ii) the lower head failure due to the creep process in the absence of water inside or outside of the lower head.

Integral experiments were performed in a 1/10 scaled carbon steel vessels of 0.4 m diameter, 15 mm thickness and 60 mm height. Up to 20 litres of binary oxide melt (30wt%CaO - 70wt%B2O3) was poured into the vessel and maintained at about 1300 °C by a specially designed electrical heater operating at about 40 kW. The melt pool undergoes natural con- vection as it would in the prototypic scenario and the vessel wall temperatures vary from ~ 600 to 1000 °C azimuthally. The pressure inside the vessel was maintained at about 2.6 MPa. The main diagnostics were several types of thermocouples and linear position transducers (LPT) which measure the displacement of the vessel due to its creep deformation. The first three experiments, described here, are focussed on vessel creep and failure.

The FOREVER/CI test was performed with the German steel (15Mo3) lower head, at about 25 bars of internal pressure and at the input power level of 22 kW. The maximum vessel wall temperature was about 800 °C. A sizable database was obtained for the creep deformation rates over a period of 24 hours and the maximum creep strain obtained was about 5 %. The second test FOREVER/C2 employed a French reactor vessel steel (16MND5) lower head, and a power level of 40 kW. The maximum vessel wall temperature was measured to be about 1000 °C and the maximum creep strain obtained was about 10 %. The test ended without the failure of the lower head as the heater failed due to its unconvering from the melt as the melt level receded due to vessel expansion.

The third test, the EC-FOREVER-I, incorporated higher initial melt level to avoid heater failure. Although, the heater power level was maintained same, at about 40 kW, the internal pressure was increased to 28 bars in order to obtain the lower head failure. The lower head failure was achieved, albeit, at a lower than expected creep strain of only 6%. The failure site was located just above the welding joint between the hemispherical part and the cylindrical part. The cause for this mode of failure is being investigated.

A coupled thermal-structural analysis of these FOREVER tests was performed with the ANSYS Multiphysics code. An improvised creep model was incorporated into this code which avoids the use of a single creep law for the entire lower head. Instead, a three dimensional array was developed where the creep strain is evaluated according to the actual total strain, temperature and equivalent stress for each element. The material damage is evaluated considering the creep and the prompt plastic deformations. The calculated results for the creep strain and vessel failure time are in good agreement with those from the experiments.

4.7 Mechanical Behavior of Reactor Vessel Lower Head During Late Phase of Reactor Accidents

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The lower head of the reactor pressure vessel (RPV) can be subjected to significant thermal and pressure loads in the event of a core meltdown accident. The mechanical behavior of the reactor vessel lower head is of importance both in severe accident assessment and the assessment of accident mitigation strategy. For severe accident assessment the failure of the lower head defines the initial conditions for all ex-vessel events, and in accident mitigation the knowledge of mechanical behavior of the reactor vessel defines the possible operational envelope for accident mitigation.

A review is made of current understanding of the mechanical behavior of RPV lower head under accident conditions. While discussion draws heavily on the NRC/SNL LHF (lower head failure) and OECD OLHF experiments, collaborative observations from the RIT FOREVER experiments, the IPSN RUPTHER experiments, the JAERI WIND experiments and the experiments from the German Risk Study are also discussed. The latter two investigations are for reactor piping but the phenomenology is similar to that governs the behavior of reactor vessel lower heads. Some of the key observations are:

- Failure size was typically smaller than the heated region;
- Localized heating led to localized failure;
- Failure typically initiated near locations of high membrane stress (due to variations in wall thickness or temperature or both);
- The membrane stress at failure initiation was typically 40% to 60% of yield. Therefore, failure was due to creep; and
- The experiments were highly repeatable and, therefore, amenable to modeling.

The LHF experiments were performed with relative small through-the-wall temperature diferential (ΔTw~20K). However, recent analyses indicated that stress redistribution due to large ΔTw could be quite important in vessel failure. This observation and the need to obtained data for low RCS pressures (2-5 MPa) in support of accident management assessment led to the current OECD program on lower head failure.

A general review will be made of the current state of modeling of the lower head failure experiments by international partners, and the information needed to support the development of a predictive model of lower head failure. The paper ends with a qualitative discussion of the application of the research to real reactor pressure vessels.