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|  | EUROPEAN COMMISSIONDIRECTORATE-GENERAL ‘RESEARCH’ | INTERNATIONALSCIENCE ANDTECHNOLOGYCENTRE |  |  |

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## NON PROLIFERATION THROUGH SCIENCE AND CO-OPERATION

**CONTACT EXPERT GROUP**

**on**

**SEVERE ACCIDENT MANAGEMENT**

**(CEG-SAM)**

**MINUTES OF THE 18th MEETING**

**(shortened version)**

Meeting organized by

**A.P. Alexandrov Research Institute of Technology, RIT-NITI**

**Sosnovy Bor, Russian Federation**

**September 28-30, 2010**

Meeting Location: Conference room of the hotel “Ambassador”

St.Petersburg, Rimsky-Korsakow Prospekt 5-7

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| Dissemination level: REPU: publicRE: restricted to EC and a group specified by the CEG-SAM membersCO: confidential, only for EC and CEG-SAM members |

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Final minutes, March 14, 2011 CEG-SAM / M-18

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| Subject: 18th Meeting of the ISTC/STCU “Contact Expert Group on Severe Accident Management” (CEG-SAM)Place: Conference room of the hotel “Ambassador”, St.Petersburg, RussiaDate: September 28-30, 2010Participants: 37 participants of 24 organizations from 11 countries: Mr. G.Azarian AREVA NP 8AS, Paris Mr. D.Bottomley EC, JRC - ITU, Karlsruhe Mr. B.Clement IRSN, Cadarache Mr. S.Güntay PSI, Villigen Mr. L.E.Herranz CIEMAT, Madrid Mr. Z.Hozer AEKI, Budapest Mr. P.Hofmann Consultant, Karlsruhe (**secretary**) Mr. M.Hugon EC, DG-RTD / J.2, Brussels (**chairman**) Mr. Ch.Journeau CEA/DTN, Cadarache Mr. M.Krause AECL, Chalk River, Canada Mr. H.Muscher KIT, Karlsruhe Mr. F.Oriolo University, Pisa Mr. A.Palagin KIT, Karlsruhe Mr. G.Pretzsch GRS, Berlin Mr. A.Schumm EdF, Clamart Mr. M.Sonnenkalb GRS, Köln Mr. J.Stuckert KIT, Karlsruhe Mr. W.Tromm KIT, Karlsruhe Mr. S.Weber GRS, Munich Mr. V.Baklanov IAE NNC RK, Kurchatov City Mr. S.Bechta RIT-NITI, Sosnovy Bor Mr. A.Blikov VNIIEF, Sarov Mr. V.Chudanov IBRAE, Moscow Mr. D.Ignatiev LUCH, Podolsk Mr. A.Kisselev IBRAE, Moscow Ms. V. Kornyeyeva NSC KIPT, Kharkov Mr. A.Kondrashenko VNIIEF, Sarov Ms. I.Kushtym NSC KIPT, Kharkov Mr. V.Loktionov MPEI, Moscow Mr. V.Nalivaev CSSC CCSNE, Moscow Mr. S.Pantyushin EDO “GIDROPRESS”, Podolsk Mr. V.Pazhetnov EDO “GIDRPRESS”, Podolsk Mr. V.Semishkin EDO “GIDROPRESS”, Podolsk Mr. V.Stepanenko STCU, Kharkov **(co-chairman**) Mr. M.Veshchunov IBRAE-RAS, Moscow Ms. T.Yudina IBRAE, Moscow Mr. V.Zhdanov IAE NNC RK, Kurchatov-CityDistribution list: (Shortened version of the minutes) Mr. J.Sanders DG-RTD / D3 Mr. O.Quintana Trias DG-RTD / J Mr. L. Bochereau DG-RTD / J.1 Mr. A. Zurita DG-RTD / J.1 Mr. S.Webster DG-RTD / J.2 Mr. P.Manolatos DG-RTD / J.2 Mr. G.Van Goethem DG-RTD / J.2 Mr. D. Haas DG-JRC / 2 Mr. J. P. Joulia DG-AIDCO/A.4 Mr. P. Servais DG-AIDCO/A.4 Intranet of Unit J.2 Mr. L.Tocheny ISTC, Moscow Mr. S.Vorobiev ISTC, Moscow Mr. W.Gudowski ISTC, Moscow Mr. A.Gozal ISTC, Moscow Mr. V. Stepanenko STCU, Kyiv EU CEG-SAM membersContact person: Mr. M. Hugon Tel.: +32 2 296 5719 – DG-RTD / J.2 |

Revised final agenda of the meeting see Annex 1, list of participants see Annex 2.

The A.P. Alexandrov Research Institute of Technology, RIT-NITI, at Sosnovy Bor organized the 18th CEG-SAM meeting in St.Petersburg, on September 28-30, 2010. The meeting location was the conference room of the hotel “Ambassador” in St.Petersburg, Russian Federation.

The CEG-SAM meeting is divided into restricted and extended sessions. The restricted sessions are to discuss internal matters and the status of current ISTC/STCU projects. The extended sessions are dedicated to presentations of the progress of on-going ISTC/STCU projects and of new or revised ISTC/STCU proposals by scientists from the Russian Federation, the Republic of Kazakhstan, and the Ukraine.

**Topic #1:** Welcome of the CEG-SAM members and opening remarks

The chairman M. Hugon (EC) and the co-chairman V. Stepanenko (STCU) welcomed the CEG-SAM members as well as the Russian, Kazakh, and Ukrainian participants of the meeting. He expressed his thanks to S. Bechta (RIT-NITI) and his team who organized and hosted the meeting in St. Petersburg. The co-chairman L. Tocheny (ISTC) was not able to attend the meeting.

S.Bechta welcomed all the participants and informed that NITI failed to obtain the permission for the CEG-SAM foreign experts to visit the KMS large-scale containment facility at NITI and the experimental installations of the RASPLAV platform located at the FSUE “RosRao” Leningrad Branch. However, the visit of the two VVER NPPs (AES-2006) under construction will be conducted as planned. He mentioned that the President of Russia signed on 11th of August 2010 instruction # 534 prescribing that Russia will step out of ISTC. This will become effective six months after the signed notifications will be sent to other ISTC parties.

M.Hugon mentioned that the ISTC secretariat will remain in operation at least until mid June 2011. On-going ISTC projects should be completed as planned latest by the end of 2011. A discussion on future types of collaboration was conducted. The STCU secretariat is not influenced by the ISTC termination.

**Restricted session**

**Topic #2:** Adoption of the agenda of the 18th CEG-SAM meeting in St.Petersburg

The agenda was accepted without changes.

**Topic #3:** Approval of the minutes of the previous 17th CEG-SAM meeting in Madrid, Spain, March 29-31, 2010

The secretary already took into account in the revised minutes the comments that he received from the participants. The revised minutes were then approved by the CEG-SAM members without any additional changes at the 18th CEG-SAM meeting in St.Petersburg, September 28, 2010.

The title of Topic #28 of the minutes has to be changed into STCU #5243 “Interaction studies of improved VVER structural materials at severe accident conditions”.

**Topic #4:** Discussion of the “Action List” of the 17th CEG-SAM meeting in Madrid

**Topic #5**: Reports by the secretariats

Because Russia intends to leave ISTC soon, M. Hugon wondered whether the CEG-SAM could be instrumental to propose new research topics on SAM to ROSATOM.

On the other side V.Stepanenko (STCU) assured that STCU has no intention to close the STCU activities. There exist many research and development programs between the countries issuing from the former Soviet Union. The Ukraine would like to continue the co-operation between STCU and ISTC. Some of the ISTC projects may be shifted to STCU. The co-operation could also be extended to other organizations. In future STCU could co-operate with Euratom on a fifty: fifty (financial) basis.

**Topic #6**: Preliminary discussion of updated and/or new ISTC/STCU project proposals

As all remaining ISTC project proposals were either withdrawn or accepted without funding, there are still two STCU project proposals, which are of potential interest to the CEG-SAM: STCU project proposal #5244 on “Nuclear fuel interaction products with structural materials under heavy nuclear – radiation accidents” and STCU project proposal #5243 on “Interaction studies of improved VVER structural materials at severe accident conditions”.

# Extended session

**Topic #7**: Welcome of the participants by the host of the meeting, general remarks

S.Bechta (RIT-NITI) welcomed the participants of the 18th CEG-SAM meeting and wished them interesting discussions and information exchanges . Concerning the changes in the planned technical tour to Sosnovy Bor see Topic #1.

**Topic #8**: Welcome of the Russian, Kazakh and Ukrainian colleagues; approval of the shortened minutes of the 17th CEG-SAM meeting in Madrid; adoption of the agenda of the 18th CEG-SAM meeting in St.Petersburg

M.Hugon opened the extended session of the meeting and welcomed the Russian, Kazakh and Ukrainian participants and expressed his thanks to S.Bechta (RIT-NITI) for organizing and hosting the 18th CEG-SAM meeting.

The shortened minutes of the 17th CEG-SAM meeting were distributed to the Russian, Kazakh and Ukrainian participants by the secretary. The obtained comments had been considered in the revised shortened minutes. This version of the minutes was accepted at the 18th CEG-SAM meeting without any additional changes.

**Topic #9**: Status of the official ISTC CEG-SAM webpage

The ISTC CEG-SAM webpage is hosted by GRS (Garching, Germany), is now fully operational (<http://cegsam.grs.de>) and is updated steadily. In the new structure of the webpage, all documents (project proposals, advice notes, work plans, progress reports, and joint publications) are collected under the ISTC project number. There will be a unique user name and password for each user and different read/write permissions for the different users. The Russian, Kazakh and Ukrainian project managers will be exclusively responsible for updating the project documentation/deliverables and the upload of presentations from project progress meetings.

The CEG-SAM members will have full access to the agendas, list of participants and minutes (restricted and open sessions), and all other stored documents (presentations). There will be special access rights for non-European members for the project(s) in which they participate.

In future a link between the CEG-SAM and SARNET web pages should be established (action 18/4).

**Topic #10**: Interaction between SARNET and CEG-SAM activities and future of the CEG-SAM

The document entitled "Interaction between SARNET and CEG-SAM activities" approved by CEG-SAM on 28 February 2005 and by SARNET Governing Board on 18 March 2005 was updated. The revised document was approved by the SARNET2 General Assembly on 10 May 2010. It was endorsed by the CEG-SAM with minor modifications during its meeting on 28 September 2010. M. Hugon will send the final version to the SARNET2 coordinator.

M.Hugon presented a short overview on the status and future of the CEG-SAM. The group is

very successful since its launching in April 2002. There exists an excellent interaction with SARNET2. 8 ISTC projects were funded and are completed. 6 ISTC projects are funded and still running. 1 STCU project is funded and ongoing.

The ISTC/STCU funding from EC will be about 7.4M€ in 2010. The basic annual costs for running ISTC and STCU secretariats are about 6M€; therefore, the funding for proposed projects in 2010 is limited to about 1.5M€.

The future of the CEG-SAM could be as follows: 1) continue its previous tasks with ISTC/STCU projects and interaction with SARNET2 hopefully until the end of 2011; 2) include in CEG-SAM the ERCOSAM-SAMARA coordinated project funded by Euratom and ROSATOM; 3) continue interaction with STCU beyond 2011.

The CEG-SAM could help in the definition and preparation of future coordinated research proposals on severe accident management to be jointly executed by Russian scientists (funded by ROSATOM) and European scientists (funded by Euratom). Some of them could participate in the meetings of the Euratom-ROSATOM Working Group, which monitors the progress of these coordinated research projects.

**Topic #11**: Update on SARNET2

B.Clément (IRSN) presented the SARNET-2 (**S**evere **A**ccident **R**esearch **NET**work of excellence) update. SARNET-2 started on April 1, 2009; altogether 21 countries with 42 organizations are participating in the programme that will last 4 years. The total effort is about 10M€ per year with about 1.5M€ per year of EC funding. The main objectives of SARNET-2 are to tackle the fragmentation that exists between the different R&D organisations, notably in defining common research programmes and developing computer tools; in particular the continuation of ASTEC assessment and its extension to cover BWRs and CANDU reactors.

B.Clement described briefly the work on Severe Accident Research Priorities within SARNET-2. Six issues remain open with high priority, four issues with medium priority, and five issues remain open with low priority and could be closed after finalizing the related research activities.

The 6 issues with **high priority** are research on 1) core coolability during reflood and debris cooling in the lower head; 2) ex-vessel melt pool configuration during MCCI and ex-vessel corium coolability by top flooding; 3) corium melt relocation into water and ex-vessel fuel coolant interaction; 4) hydrogen mixing and combustion in the containment; flame acceleration; 5) the impact of oxidising conditions on source term (Ru oxidising conditions or air ingress for HBU and MOX fuel elements); 6) iodine chemistry in the RCS and in the containment.

The tasks of SARNET-2 will be executed by 8 work-packages on management, spreading of excellence (courses, conferences), information systems, ASTEC, corium and debris coolability, MCCI, steam explosion and hydrogen combustion in containment and oxidising impact on source term.

Up to now the interaction between CEG-SAM and SARNET-2 works well and the SARNET-2 recommendations were considered in the final work programmes of the various ISTC/STCU project proposals. The results of ISTC/STCU projects are used by foreign collaborators in the framework of SARNET-2. The interaction between SARNET-2 and CEG-SAM brings mutual benefits and further assures a critical mass of expertise for ISTC/STCU proposals addressing specific issues in the SAM area. The objective of the interaction is the resolution of still-pending questions that are important for reactor safety, and the knowledge transfer for safety application.

B.Clement described the progress in the last 6 months: April 2010: release of the SARNET-2 newsletter N°2. May 10, 2010: 1st General Assembly (gathering one representative from each partner + M.Hugon, EC Corresponding Officer) in Bologna (Italy). May 11-12, 2010: 4th ERMSAR conference in Bologna (Italy), hosted by ENEA*.* June 2010: report on detailed description of 2nd period work (April 2010 – Nov. 2011, 18 months long). Continuation of the PRELUDE tests (IRSN), preparing the future PEARL tests on debris bed reflooding. Conduct of the second VULCANO separate-effect CCI test. June 8, 2010: 1st meeting of TWG Gen.II-III in SNETP frame where TSOs underlined the interest to enhance SARNET future role on R&D priorities. End of June 2010: release of ASTEC V2.0 Rev1.

The next main milestones of SARNET-2 are joint workshops, conferences, and education courses. Early oct.2010: SARNET-2 newsletter N°3. Oct.11-15, 2010: 4th ASTEC Users’ Club in GRS/Cologne. Nov.17, 2010: 1st meeting of the Severe Accident Priorities Group. Autumn 2010: technical progress meetings of WP6, WP7 and WP8. Early 2011: planned publication of the SARNET Book on SA (currently under review). January 2011: 1st Education & Training course on Generation II and III NPP SA phenomenology, jointly organised by Univ. of Pisa and CEA in Pisa (Italy). Before end of 2010: 1st meeting of Advisory Committee, composed of end-users (mostly out of SARNET), to be organized. 5th ERMSAR conference will take place in autumn 2011 (host and location to be defined soon).

**Topic #12:** R&D research priorities from SARNET-2 project (SARP: “Severe Accident Research Priorities”)

M.Sonnenkalb (GRS) mentioned that the work program on severe accident priorities has not yet started, therefore, no update is available. The meeting is organized by GRS; it will take place in Cologne on November 17, 2010. A request from SARNET-2 project leader was made to invite the SARNET-2 work package leaders to participate in the SARP activity. GRS will follow this request.

**On-going project presentations, updated and new project proposals**

**Topic #13**: Final results of the ISTC project #3831 on “Experiments at large-scale installation for heating and retention of Corium

A.Kondrashenko (VNIIEF) described the various performed activities in order to conduct the experiment on the corium-concrete behaviour. The main objectives have been: 1) To develop the technology of pyrotechnic compound and briquettes production. 2) To develop the pyrotechnic technology of corium briquettes. 3) To develop the conditions for dumping of pyrotechnic briquettes into corium. 4) To develop the measurement system of melt and concrete temperatures. 5) To conduct the medium-scale experiment for study of corium behaviour and melt interaction with concrete; and 6) To analyze and interpret the results of the experiments.

The procedure of test conduct of the medium-scale MCCI experiments has been as follows: To ignite and create the initial molten pool of 50kg of PTC (pyrotechnic components) were put into the concrete tank. The initial PTC volume was about 55 litre. The experiment started with the ignition of inserted PTC by means of an electric discharge. Four propane-oxygen burners started operating simultaneously together with the igniters. The temperature of the gaseous burning products above the corium surface was about 2000 C. These products create a thermal isolation “cap” above corium. After ignition of the inserted PTC inside the tank, combustion took place for about 30s. About 40s after the ignition the dumping of briquettes at time intervals of 5 to 30s was started. The last two briquettes were dumped into corium with time intervals of 10 and 30s, respectively. The average PTC drop rate was ~7s. Total time from ignition till the last briquette was dumped was about 520s. The average energy release inside the melt during the briquette dumping was about 280kW that equals the combustion of 100g/s of PTC. The velocity of the briquettes while striking the corium surface was ~8m/s. The briquettes fully penetrated the corium surface and fell to the centre of the melt, where the energy release took place and provided good melt mixing. It’s necessary to mention that the briquettes didn’t fall exactly in the centre of the melt but slightly off-centre. During the experiment (while the briquettes were dumped) no solid crust formation was noticed. The medium-scale MCCI experiment was recorded by 2 video cameras. Camera 1 was located at the same height as the concrete tank at the distance of ~8m. Camera 2 made the recording by viewing a mirror that was located at 3.5m above the concrete tank (the camera was located 1m horizontally away from the mirror. During the experiment the temperature of concrete and corium were experimentally measured. The thermal fluxes into the concrete will be calculated. The concrete ablation rate was experimentally measured after the experiment.

The following conclusions can be drawn: The middle-scale experiment with a characteristic corium volume of about 34 litre (tablet volume ≈25 litre), corium mass ≈130 kg (tablet mass ≈105 kg), corium temperature ~2500-3000°C, heat fluxes towards walls and bottom of the concrete tank of ~100-200 kw/m2 and melt retention time of ~15 min was achieved. The axial ablation rate of the concrete crucible (initial diameter: 45cm) varied between 2-3cm, the radial ablation rate was about 2.5cm.

The pyrotechnic technology applied in the medium-scale experiment to produce molten corium and its retention in a concrete crucible can also be used to carry out the large-scale experiment with the following characteristics: corium volume ~100-150 l, corium mass ~1000-1200kg, corium temperature ~2500-3000°C, heat fluxes towards walls and a bottom of a tank of ~100-150 kw/m2, corium internal energy release time: 1-2 hours.

The CEG-SAM expressed its main interest in the conduct of MCCI tests with UO2. It has still to be clarified if VNIIEF can perform the tests with UO2. Export control problems must be solved to continue with the planned fuel-containing MCCI tests.

**Topic #14**: Report on the SARNET meeting on VNIIEF experimental capabilities and discussion on future collaboration

 Ch.Journeau (CEA) reported on a meeting with the SARNET WP6 to inform the members on the information collected during a medium-scale pre-test for a later planned large-scale MCCI test performed by VNIIEF in Sarov (Russia). The final objective of the project, launched under the auspices of the Contact Expert Group on Severe Accident Management (CEG-SAM) is to perform large-scale (1t) 2D MCCI (Molten Core Concrete Interaction) experiments with corium-steel mixtures. VNIIEF has proposed an original technique in which the melt is generated by a thermite reaction (exothermal chemical reaction) and the decay heat is simulated by the regular insertion of thermite briquettes into the melt, so without any electrical heating. The use of this technique may give the possibility to detect system effects induced by other conventional heating techniques (inductive heating, direct Joule heating).

Due to the originality of this technique and the large cost of a large-scale test (estimated to 980 k$ in 2008), a 9-month project (from 1st May 2009 to 31st Jan 2010 of 98k$ or 10%) was launched by ISTC with EU funding to develop the technique and perform three feasibility tests of a medium-scale up to 100kg of corium melt. This feasibility tests used a ZrO2-Fe melt to avoid export control problems (as would be the case if it had contained UO2) for this feasibility phase.

A meeting was organized on 26-27 January in Nizhny Novgorod with VNIIEF. The handouts and the report were sent in April 2010 to the European collaborators (after security check and VNIIEF management approval). On May 7, VNIIEF informed us that due to bureaucratic delays these documents were delayed till mid-June. Eventually, the reports have been distributed to Ch. Journeau during this CEG-SAM Meeting (September 2010). The meeting was nevertheless maintained so that WP6 partners could be informed on this test and present their opinion.

It has been decided that CEA and KIT would analyze the data when they are available and compare the experimental results with code calculations. Some questions must be answered: What is the difference between a transient test in which all the thermite is ignited at once and this type of test? In case, little difference is found, would it be the same with a longer test with a larger mass? Are the experimental heat fluxes and ablation rates coherent and are they prototypic? If not, what could be done to propose an interesting test procedure with this technique?

There have also been some general remarks: The initial and boundary conditions must be well described. This includes: Measurements providing precise data on the thermal conditions at the upper surface with the gas burners. The determination of the effectively released energy by a PTC briquette. Is it theoretical or was it measured? The introduction of some concrete materials in the briquette could lower the pool temperature and the crust thickness. This has been recommended by several participants. Focus has been put on the fact that 1 t may be too small (already studied, albeit in 1D) and 1 hour too short to reach steady-state. Further details on the thermite fabrication (powder size, safety aspects of the briquette pressing) have also been requested.

For the future, it has been decided to wait for a thorough analysis of the results (not before the November 2010 SARNET MCCI meeting) and if needed a second feasibility test before a joint Euratom-ROSATOM project on these issues could be seriously discussed. The reporting procedures for VNIIEF with UO2-containing melts must also be clarified.

**Topic #15**: Progress report on the ISTC project #3592 "Corium Melt Interaction with Reactor Vessel Steel” (METCOR-P)

S.Bechta (RIT-NITI) described the objectives of METCOR-P project: Qualification and quantification of physico-chemical phenomena of corium melt interactions with reactor vessel steel with particular interest to interaction characteristics i) at vertically-positioned interfaces, ii) peculiarities of interaction with European vessel steel, and iii) corium melt oxidation effects.

The results of the conducted tests MCP-5, -6, -7 and -8 were presented. The МСР-5 test was performed in order to compare the rate and depth of corrosion of the European and Russian vessel steels at the interaction with molten sub-oxidized corium. The melt composition, steel specimen surface temperature and molten pool condition were found to be most similar to those in the MC9 test with Russian steel, that is, Сn≈30%; Ts≈1450ºC, and absence of a surface crust. The main results are as follows: Qualitatively, the processes that characterize corrosion kinetics are similar to those related to Russian steel, namely, prolonged incubation period and the corrosion rate correlation (accurate to within a coefficient). The maximum depth of corrosion is determined by the temperature at the final boundary between the interaction zone and steel specimen. This temperature value corresponds to that determined in tests with Russian steel.

The МСР-6 test was carried out in order to determine the rate of metallic melt oxidation in steam. The melt composition was (mass %): U-21; Zr-14; SS-65.The melt temperature amounted to 1400ºС. Preliminary results: The rate of oxidation is controlled by the surface oxidic crust. The time profile of oxidation rate is non-monotonous and is determined by cracking of the crust.

The МСР-7 test was performed for determining the rate of molten sub-oxidized corium oxidation in steam. The melt composition was С-30; (U/Zr)at =1.2. The melt temperature was 2350ºC and the surface crust external temperature amounted to 2080ºC. Preliminary results: The surface crust substantially reduces the rate of melt oxidation. Here the appearance of cracks also determines a non-monotonous time profile of the oxidation rate.

The purpose of the МСР-8 test is to investigate vessel steel corrosion during the interaction with the U-Zr-Fe metallic melt at the vertical position of the interaction interface. Current status: A pretest aimed at fine tuning of the vertical specimen cooling system and of the corrosion rate ultrasonic measuring system has been performed. The test facility has been completely prepared for conducting the test.

**Topic #16**: Progress report on the ISTC project #3813 “Phase relations in corium systems” (PRECOS)”

S.Bechta (RIT-NITI) described the objectives of the PRECOS project**. The** subject of the project is the experimental investigation of phase diagrams of oxidic and metallic-oxidic corium systems that form as the result of core meltdown and interactions of melt with construction and structural materials of the reactor core, concrete shaft, and core catcher.

The following systems will be studied in PRECOS: 1) Binary and ternary oxidic systems (CaO-UO2, CaO-FeO, SiO2-UO2, UO2-FeO-SiO2, UO2-FeO-CaO, ZrO2-FeO-SiO2, and ZrO2-FeO-CaO) that contain components of concrete and sacrificial materials, i.e., of importance for modeling the interaction of corium with materials of the concrete shaft and core catcher. The SiO2–containing systems should be specially mentioned, as their high viscosity and low conductivity make their experimental investigation problematic. Still, they are very important for modeling the ex-vessel corium behaviour for a series of power reactors, including such modern ones as EPR. 2) Metallic-oxidic systems U-Zr-Fe-O with different concentrations of components, especially in the miscibility gap. 3) Multi-component mixtures representing prototypic ex-vessel corium.

In the last 6 months experiments in the UO2-CaO, UO2-FeO-CaO, UO2-FeO-SiO2 systems have been conducted. Post-test analysis of samples is in progress. The laser pulse heating (LPH) setup of the Joint Institute of High Temperatures and Russian Academy of Sciences (IVT RAN) has been additionally equipped with 300 mW diode laser for specimen lighting. The LPH setup has been used for the verification experiments on previously studied Zr-O and ZrO2-FeO systems. Construction work necessary for getting a license on uranium handling has been completed. The license is expected to be issued in December 2010; after that the U-Zr-Fe-O studies will be started.

The experimental studies in the UO2-CaO system have been completed. In the system CaO-UO2 the liquidus temperatures at high CaO content and the melting point of CaO have been determined.

The tests performed in the system UO2-FeO-SiO2 confirm the existence of a miscibility gap in a fairly narrow region adjacent to the diagram corner on the SiO2 side. The ternary eutectics have been determined within a particular triangle of the UO2 - Fe2SiO4 - SiO2 ternary system. The PRS13 test has been performed for determining composition of the second ternary eutectics within a particular triangle of the UO2 - Fe2SiO4 - FeO ternary system. The ingot analysis is underway.

The PRS 14 test results in the system UO2-FeO-CaO are: The visual polythermal analysis in the cold crucible (VPA IMCC) is used to determine the liquidus temperature of the composition, mass%: 19.3±1.0 UO2 + 66.6±3.3 FeO + 14.1±0.7 CaO – Tliq= 1242±20°C. An EDX method is used to determine the ternary eutectic composition, mass%: 10.4±0.1 UO2 + 61.9±0.5 FeO + 27.7±0.5 CaO.

**Topic #17:** Progress report on the ISTC project #3876 on “Thermo-hydraulics of U-Zr-O molten pool under oxidising conditions in multi-scale approach (THOMAS)”; part #1

M.Veshchunov and V.Chudanov (IBRAE-RAS) described the objectives and work plan of the project THOMAS and its status. Non-destructive and destructive post-test examinations of bundles in various tests showed the formation of molten pools of different scales at various stages of core degradation. Small local pools were observed at different elevations in bundles in the early stage of core degradation in CORA and QUENCH tests. Results of the PHEBUS -FP tests confirmed that a significant part of the fuel bundle was liquefied and that the amount of fuel damage was close to TMI-2 with an extended molten pool located in a central zone of the bundle underneath a cavity. In the late stage of a severe accident, the formed melt can relocate into the lower head of the reactor pressure vessel and form a large molten pool interacting with cooled walls.

The 2-D stand-alone code developed to simulate (in the simplified geometry of the tests) simultaneous UO2 fuel dissolution, U-Zr-O corium melt oxidation accompanied with the bulk ceramic precipitates formation and oxidation of the steel wall of a vessel in contact with corium, was transformed in the previous stage of the Project into a 1-D corium melt – steel oxidation module, in order to describe local interactions at the corium-steel interface (in the geometry of the pressure vessel). This allowed the simplified description of the heat and mass exchange between the U-Zr-O corium melt and peripheral crusts in the code to be replaced (in future) with a detailed thermo-hydraulic approach of the CONV code.

The work for further development and testing against experimental data of the 1-D oxidation numerical module that simulates evolution of the solid phase layers ((Zr,U)O2-x crust, FeO corrosion layer and steel), temperature distributions in the layers and U, Zr, O molar fluxes into the melt, was continued. Further improvement of interface program unit for coupling of the melt – steel oxidation 1-D module with the corium melt 2-D thermo-hydraulic code was continued.

The preparatory work with the code CONV 2D continued. This includes 1) sources for implementation of melt oxidation model, and also 2) interface (input files) for insert of physicochemical melt oxidation model for modelling of thermal hydraulic behaviour of U-Zr-O melt under oxidizing conditions for small and medium scale experiments in code CONV 2D.

Designing the interface program unit for the representation of a minimum parameter set for melt – steel oxidation 1-D module was continued. The given set includes an additional orthogonal grid, on which the quasi one-dimensional melt oxidation model will be solved. The grid will be connected by the boundary conditions with a base calculated grid of CONV code. Besides the parameter set includes characteristics of materials, participating during oxidation, which with the help of the interface program will be transformed to a set of input files for CONV code. These will take into account the chemical structure (molar composition), physical properties (thermal capacity, thermal conduction, density, viscosity) and interfacial temperatures of steel and melt to allow a data exchange with the melt oxidation model.

Testing the modernized (parallel) version of CONV code was continued on such tests as T-junction thermal mixing test. A good agreement for the finest grids up to 40 million nodes was obtained.

**Topic #18:** Final results and conclusions of the ISTC project # K-1265 “Study of the processes of corium-melt retention in the reactor pressure vessel” (INVECOR)

V.Baklanov (IAE NNC RK) described the results and conclusions of the INVECOR experiments which were performed together with V.Zhdanov who left IAE NNC RK in August this year. The objective of the in-vessel corium retention experiments (INVECOR), is the improvement of the safety assessment of LWR corium in-vessel retention (IVR) and the modelling of the thermal and physico-chemical processes of the prototypical corium pool and its retention in the water-cooled RPV lower head. The project started on May 1, 2006 and was finished on April 30, 2010, and consisted of 4 tasks.

Task 1: Development of technologies to produce protective coatings on the internal surface of the graphite crucibles and on the outer surface of the graphite nozzles of the plasmatrons.

Task 2: Modeling of the corium pool in the RPV model and definition of the efficiency of heating, distribution of thermal fluxes and temperatures on an internal surface of the RPV model; calculation and optimization of the external cooling system of the RPV model; calculation of the temperature and deformed condition of the RPV model (pre-test and post-test calculations).

Task 3: Conduct of four large-scale experiments with maintenance of energy release in the corium pool that was contained in the RPV model. In the tests oxidic corium (2x), С-30 and oxidic-metallic corium (С-30+10 wt.% of steel) melts were used.

Task 4: Conduct of post-test examinations of the solidified corium melt and RPV steel samples by XRD; optical metallography and specific element analysis of the specimens.

The performance of the main task (Task #3) has been the conduct of integral large-scale experiments. In the tests up to 60kg of corium melt С-30 was discharged from the electric melting furnace from a height of 1,7m into the RPV model with the plasmatrons in place for the simulation of the decay heat. The duration of the experiment on corium retention in the vessel has lasted 1 to 2 hours. The specific capacity of energy release into the corium varied between 4 and 8 W/cm3. The maximum temperature of a RPV model wall was 1300°C that was reached due to thermal insulation of the external surface of RPV model and by regulation of the cooling water flow rate. The addition of steel in the corium was carried out by simulating the vessel inner surface plating by stainless steel (in the experiment INVECOR-2) and dumping of steel sheet on the surface of the corium pool (in the experiment INVECOR-3). In two experiments (INVECOR-1/3 and INVECOR-2) up to 10kg of oxidic corium C-90 was previously put into the vessel model. In the fourth experiment (INVECOR-1/4) efficiency of thermal insulation on the external surface of the vessel model was increased and an optional heat shield on the corium was installed.

Under Task #4 post-test examinations of samples of corium and vessel steel, including cutting the corium ingot and the vessel model, sampling, X-ray phase analysis (XRD); optical metallography, elemental analysis, and compilation of experimental results were performed.

During the post-test studies it was found that solidified corium was in the form of a continuous ingot, and in the form of small fragments, located on the top of the ingot. A slight erosion of the inner surface of the steel wall of the vessel model was detected.

Study of phase composition of hardened corium showed that the composition of the lower crust of the corium adjacent to the wall of the model of the hull and the composition of the upper layer of the fragments is almost identical and consistent with the rapid cooling of the core melt (quench-effect). This suggests that part of the fragmented corium formed as a result of the primary contact with the melt jet on the cold steel surface.

Weight fragmented corium may depend on the mass ratio of corium / steel. It should be kept in mind that in the experiment INVECOR-1.4 the highest weight of the fragments was obtained compared with previous experiments. Studies have shown that the particles of the lower layer will look like fragments of the upper crust on the corium ingot. Consequently, it could be formed as a result of cracking of the upper crust of the corium with increasing pressure in the closed pores in the ingot, as well as a result of cracking of the top of the ingot during thermal expansion in the later stages of pouring.

The hypothesis of the formation of a layer of fragments as a result of primary contact of a stream of the melt with pressure vessel model can be checked up in the course of further experiments in INVECOR where the thickness of a wall will be reduced; that will lower essentially a mass thermal capacity model of the body and, accordingly, to lead to its faster warming up at the expense of heat exchange with corium.

Reduction of a thickness of a in these proposed experiments will lower circumferential heat flux along a wall to the top flange and hence higher thermal fluxes through the wall in a radial direction.

In addition, an experiment without a thermal protection on an external surface (ie provide heat removal from corium mainly in a radial direction in cooling water) is offered.

On the basis of the performed experiments the following conclusions can be drawn:

-Four large-scale integral experiments have been conducted by modeling of core displacement in the lower head of RPV with subsequent imitation of decay heat in corium retained in a pressure vessel.

-The formation of fragmented debris bed can be connected with splashing of corium melt jet during its dropping into the test section and with large heat absorption from corium to relatively cold vessel wall.

-It is reasonable to research the condition of fragmented debris bed formation by decrease of RPV model mass (decreasing the RPV model wall thickness and/or pre-loading of once molten corium in RPV model). Estimation of fragmented debris bed fraction depending on molten core dropped mass has been made.

-The increase of radial heat flux through RPV model wall could be done by removal of thermal insulation from the outer surface of RPV model.

A proposal for a coordinated project funded by Euratom and Kazakhstan will be prepared as soon as possible if these ideas are supported by foreign collaborators (CEG-SAM members).

*Note from the Secretariat:* M. Hugon checked that there is a cooperation agreement between Euratom and the Republic of Kazakhstan in the field of nuclear safety which is still in force. This agreement covers in particular research on reactor safety, radiation protection and nuclear waste management."

The final report of the ISTC project #K-1265 has been placed on the CEG-SAM web-page.

**Topic #19:** Status of the ISTC project #3635 on “Scale experimental investigation of the thermal and structural integrity of the VVER pressure vessel Lower Head in severe accidents”

Loktionov (MPEI) presented the status of the project. The overall objective of this project is the experimental and numerical study of VVER-440 lower head (LH) reactor vessel models under thermal and overpressure loadings corresponding to realistic SA scenarios. The different tasks are 1) the manufacturing of the VVER LH reactor vessel scale models (scale 1:5), 2) the conduct of the scale experiments with VVER vessel models at high temperatures as well as 3) separate-effect tests on the creep behaviour of the VVER vessel steel and 4) numerical pre- and post-test analyses of the scale experiments.

The expected results will be experimental data on the creep behaviour, heat-up and failure of the VVER-440 vessel material. The data will be used for verification of thermo-mechanical codes that are used in safety assessment and in SA management strategy for NPPs.

The project efforts are focused on the following tasks:

Task 1: Pre-test simulations will be carried out by means of the numerical codes to determine the behaviour of the vessel models during the scale experiments for the chosen SA scenarios.

Task 2: Development and manufacturing of the experimental test facility and supporting systems for the VVER-440 scale vessel models testing (a geometrical scale ~1:5). The material and thermal treatment of the vessel steel have to correspond to the same conditions as for a regular VVER vessel.

Task 3: Examination of the VVER-440 vessel behaviour under SA conditions by experimental and numerical investigations (thermal and structural analyses). The mathematical treatment and analysis of scale experiments will be done with the domestic code ATM-VVR and with commercial codes MSC-Marc, MELCOR, RELAP/SCDAP for validation of the physical models implemented in these codes.

Task 4: Determination of the creep characteristics and mechanical properties of the vessel steel 15Kh2NMFA in the temperature range from 700 to 1200°C and times up to 50 hours. The execution of short-term tensile and material creep tests with samples from the VVER vessel steel have been conducted to obtain refined data for its mechanical characteristics.

Regarding Task 4 the creep experiments in vacuum and argon from 700 to 1200°C up to 50 hours at constant load have been carried out and finished. The obtained results are shown in dependence of time-to-failure and the applied stress. The elongation of the specimens considerably exceeds 100%. In some cases there were two or three necks in the deformed part of the specimen. At present the analysis and mathematical treatment of test results are in progress which will be presented in the final report.

The use of argon as protective medium leads to a considerable reduction (about 2 times and more) of the time-to-failure of the specimen. It probably depends on oxidation of the surface of the specimens due to partial ingress of air into the test section. The formation of an oxide film on the outer surface of the test specimens and its destruction under deformation leads to earlier failure of the specimens in comparison to tests conducted under vacuum.

One of the problems during the scale tests has been the oxidation of the outer vessel surface. The reason for the separate-effects tests in air was questioned.

Prolongation of the project without additional funding has been requested and was confirmed for up to one year.

Both in-house and external codes have been applied to check the test conditions proposed for the RPV model and show wide range of results. These were circulated to the collaborators and the feedback is that they need to monitor the test carefully and decide the final change in conditions during testing to ensure that the final creep failure will occur under constant conditions. The test is due to be done by the end of the year.

**Topic #20**: Technical tour to NITI at Sosnovy Bor; Visit of the VVER NPPs under construction

The Head of the Project Management Department G.V. Victorov welcomed the CEG-SAM group members at Leningrad NPP-2 at Sosnovy Bor (about 100km away from St.Petersburg).

Since the initially planned visit of the KMS large-scale containment facility was cancelled a presentation on KMS was given instead by the Head of the KMS Test Facility V. Zasukha.

The KMS large-scale test facility was constructed at NITI site and designed to model accident processes in VVER reactor plants and provide experimental data for safety analysis of both existing and future NPPs. The KMS phase #1 is at the completion stage. This is a containment model of 2000 m3 volume intended for experimentally simulating heat and mass transfers of steam-gas mixtures and aerosols inside containment (total height: 28m; diameter: 12m; max. pressure: 0.5MPa; max. temperature: 150°C). The KMS phase #2 is planned to incorporate a reactor model (1:27 scale) to be used for analysing a number of events including primary and secondary LOCA. The KMS program will include preparation and conduction of experiments, analysis of experiment data. Currently the KMS is using for modelling of containment passive cooling systems of new VVER designs.

It was discussed that (i) there is a lack of measurement points and (ii) the instrumentation of the KMS facility should be improved to carry out meaningful experiments.

Before the visit of the construction side of the VVER NPPs (AES-2006 design) a detailed presentation was given by V.V. Bezlepkin, the Director of the Science Department at SPbAEP. AES-2006 is abbreviation of evolutionary NPP design developed on the basis of the VVER-1000 Russian design with an overall operation life of 480 reactor-years. The NPP AES-2006 is a model project of Russian nuclear power plant of new generation "3+" with improved technical-economic parameters. The goal is to attain enhanced safety and efficiency with optimized capital investments. The capacity of the reactor is 1150MW (with possibility of raising up to 1200MW), the capacity factor is 92%. The period between refuelling is up to 24 months. It should operate at a thermal efficiency of almost 35%. The AES-2006 design life time will be about 60 years.

In terms of safety, the project complies with all national scientific-technical criteria and IAEA recommendations. The key peculiarity of the project is use of passive safety systems in addition to traditional active safety systems. The reactor can withstand a strong earthquake, tsunamis, hurricanes, crashing planes. The key improvements are a double containment with a liner, a core catcher (meltdown probability 5.8x10-7 /year), passive heat removal system, and a hydrogen removal system. The new AES-2006 pressurised water reactor (PWR) design is scheduled for commissioning in October 2013 and the second one a year later, respectively. These first AES-2006 units are expected to be built at a cost of some $3.0-3.7 billion per pair.

Conclusions: The AES-2006 design is the result of an evolution of the NPP designs with VVER reactors. This design conforms to all current Russian and International safety standards and to the IAEA requirements. The benefits of this evolutionary approach are reduction in the design time and costs, and will be a reference for later engineering approaches and enhanced project attractiveness. The constructed units #1 and #2 of Leningrad NPP-2 in Russia are leading units of this design. LNPP-2 unit #1 will be the first unit in Russia equipped with a core catcher.

**Topic #21:** Progress of the STCU project #4207 “Long-term prognosis of the behaviour of the fuel dust in the Chernobyl Shelter”

V.Stepanenko (STCU) presented the project progress instead of V.Protsak (UIAR) who was not able to attend the meeting. The Chernobyl shelter of the RBMK-1000 Chernobyl NPP unit 4 is a source of radioactive particles that formed during the accident (now present inside the construction in the form of dust) and in the subsequent period due to physical-chemical destruction of the fuel-containing material (FCM). In view of the planned transformation of the “Shelter” into an ecologically safe system, the presence of the fuel dust in the shelter (about 30000kg) will become a serious problem. In spite of the numerous data on the characteristics, composition and localization of the fuel dust in the shelter, the mechanisms of its formation and, especially, the prognosis of its further physical/chemical transformation are still not clear.

V.Stepanenko described the physical-chemical characteristics of the Chernobyl fuel particles and mechanisms of their formation. The project studies will be focused on the fuel particles and main types of the fuel-containing materials (FCM) in the shelter, as well as on the mechanisms governing their destruction. Experimental data show that the FCM destruction in the present time occurs due to both internal and external influences. It results in the FCM transformation into the highly-mobile and highly-radioactive dust. Therefore, it is very important to carry out experimental and theoretical studies within the framework of the project, which will enable the formulation of a model of the long-term behaviour (50-100 years) of the fuel dust under the shelter conditions. The model must describe both the transformation of the existing fuel particles and the processes of their formation from the main types of the FCM.

The project consists of 5 stages: Stage #1: Mechanisms of formation and classification of the Chernobyl FP, updating the DB. Stage #2: Characteristics and behaviour of RA and water in Shelter. Stage #3: Experimental study of the FP destruction. Stage #4: Creation of the model of FP transformation. Stage #5: Long-term prognosis of the fuel dust behavior in Shelter.

The following results have been obtained:

Stage #1: At present the obtained information is processed and stored in the specially developed Database. The main fields of the DB are: the premises code, characteristics of the premises, presence and types of the FCM in the premises, the FCM parameters, presence of the fuel dust according to the observations, presence of the water flow through the premises and its parameters, presence of the air flow, the models of destruction of the present types of the FCM.

Stage #2: During this stage the water samples for analysis of uranium concentration and specific activity of the radio nuclides (90Sr, 137Cs, 238Pu, 239+240Pu, 241Am, 244Cm) in aggregations of the LRW were collected at the lower marks of Shelter, namely in the rooms 001/3 VSRW, bubbler pool and in the south-east part of the block B. The radionuclide activities in the water samples collected were measured. The 90Sr activity was determined by means of β-radiometry, 137Cs – γ-spectrometry, U – spectrophotometer, TUE – radiochemical extraction and α-spectrometry. Uncertainty of all measurements did not exceed 10-15 %. The number of water samples and air samples exceed 100.

Stage #3: During the stage we have continued the experimental works for determination of the dynamics of the radionuclide leaching from the LFCM samples in the simulated Shelter waters. The experiments are in progress now. Also during the stage the samples of the three LFCM types (brown, polychrome and black ceramics) were prepared for study of 90Sr, 137Cs, 238Pu, 239+240Pu and 241Am leaching in the distilled water.

Stage #4: During this stage we started the works on the model of the fuel particles transformation in the Shelter conditions. The model will describe the FP destruction, change of their dispersal composition and the radionuclide leaching rates. We have analyzed the information on the physical-chemical properties of the Shelter environment and the chemical stability of uranium dioxide and its higher oxides. Also we reviewed the results of studies of the fuel particles destruction both in natural conditions and in the simulated media. The preliminary analysis show that the fuel particles dissolution kinetics is satisfactory described by a first order of differential equation. Since the dissolution rate of the particles will increase with time because of the increase of their surface/mass ratio, it is planned to apply either the concept of the dissolution rate normalized to the particle surface area (g/cm2/day) or the concept of linear dissolution rate (μm/yr).

**Topic #22:** Status of STCU Project Proposal #5244 on “Research of objects - nuclear fuel interaction products with structural materials under heavy nuclear-radiation accidents

V.Stepanenko (STCU) presented the project proposal progress instead of V.Krasnov (ISP NPP NAS Ukraine) who was not able to attend the meeting. When erecting a “New Safe Confinement” (NSC) for the ChNPP, temperature growth and moisture decrease can entail in some areas with a high probability a self-sustaining chain reaction. In this connection the concept of the “Shelter” Object should be revised in terms of criticality suppression in critical mass risk areas.

The objective of the project proposal will be to obtain information on neutron-physical processes occurring in clusters under impact of external factors. To improve existing data and to obtain new data on the structure and content of FCM produced as result of fuel/melt interaction with sub-reactor plate concrete and melting of reactor materials: graphite, fuel channels, metal and materials of the sand/gravel filling.

What will be new? The existing data bank obtained from the GRS-IPSN -KI studies and annexes or any other relevant reports, will be improved, in particular: refinement of nuclear fuel balance in sub-reactor room 305/2 will be performed; more accurate information assessment based on the structure, composition of the FCM mass as well as the corium's neutron physical and physicochemical characteristics will be made; a more realistic scenario of cluster production with high fuel content during the accident's active stage will be proposed.

The importance of the project proposal will be: Improved estimates for nuclear hazardous cluster that would increase safety and reliability during construction of New Safe Confinement (NSC). In connection with this, Concept of nuclear safety of “Shelter” Object should be revised in terms of criticality suppression in critmass risk areas that will be the objective of this Project.

The Project Proposal #5244 will cover researches and estimate-experimental works aimed at definition of structure, content, neutron-physical and physicochemical characteristics of corium in hidden LFCM clusters, as well as development of measures to control sub-criticality of such clusters.

**Topic #23:** Status of STCU Project Proposal #5243 on “Interaction studies of improved VVER structural materials at severe accident conditions”

V.Kornyeyeva (NFC KIPT) presented the revised STCU project proposal #5243. The operation of water-moderated reactors (VVER) does not exclude a possibility of beyond design-basis accidents with a core materials meltdown. Currently, activities are underway to extend the operation life time and reliability of reactor cores through application of new materials and structural improvements. Knowledge of how the use of new materials and structures will affect core melt formation during beyond design-basis accidents is a sine qua non of activities to increase NPP safety in general.

To identify solutions for the issues resulting from core melt formation in VVER type reactors, “Nuclear Fuel Cycle” Science and Technology Establishment of National Science Centre “Kharkov Institute of Physics and Technology” (NFC STE NSC KIPT) propose to carry out scientific and material testing examinations within the frame of the STCU Project Proposal #5243.

The project proposal is designed to: obtain data on interaction parameters of the materials in VVER improved core structural components; obtain data on melt formation, first and foremost of fuel and neutron absorbers with the structural materials; identify phase composition of the resulting melts in the solid state; determine physical characteristics of viscosity and fluidity of melts from materials used in standard and upgraded designs of VVER components.

The main tasks of the project proposal are: based on the analysis of the data available in the literature to identify a probable scenario for heavy accident development at NPP; upgrade and prepare process equipment and instrumentation for carrying out research; upgrade design and manufacturing of sample fuel and absorber rods for research; upgrade methodologies to conduct experiments and study the structure and composition of power reactor core materials before and after their interaction (in the solid state and after melting); study the effects of fuel and absorber rod structural features (close contact, presence of oxides on the surface) on the nature of the beginning of their materials' interaction; obtain data on the temperature parameters of the beginning of melt formation as a function of material state; obtain melts of standard VVER fuel and absorber rods, namely combination of UO2 + Gd2O3 materials with the alloy Zr1%Nb (E110) and stainless steel with boron carbide; study processes of melt formation for new combinations of absorber materials B4C, Dy2O3•TiO2, Hf and interaction of these components with the melt of fuel materials; identify phase composition of melts thus formed; identify melt viscosity and fluidity parameters depending on their phase composition.

**Topic #24:** The main results and conclusions of the VVER fuel assembly tests under severe accident conditions in the large-scale PARAMETER test facility (ISTC project #3690

V.Nalivaev (CJSC CCSNE) presented the final results on the ISTC Project # 3690 on “The behaviour of fuel rod assemblies under severe accident top quenching conditions in the PARAMETER-SF test series”. The project was executed jointly by three organizations: FSUE SRI SIA “Luch”, IBRAE RAS, OKB “GIDROPRESS” with participation of the leading specialists from JSC “VNIINM”, RRC “Kurchatov Institute”, A.I. Leipunsky SRC RF - IPPE and methodical support by foreign collaborators (KIT, GRS, JRC-ITU, PSI, EdF, CEA, AEKI, and IRSN).

Within the framework of the Project two experiments of PARAMETER-SF series (SF3 and SF4) were conducted with 19-rod fuel rod assembly simulators (18 heated rods and 1 unheated rod) of a VVER-1000. The simulators were manufactured with the standard reactor structural materials (UO2 pellets and Zr+1%Nb cladding tubes) similar to the PARAMETER-SF2 experiment (ISTC project #3194).

The PARAMETER-SF3 experiment was conducted under the following test conditions. Coolant flow rates: argon 2g/s (670K) and steam 3.5g/s (770K). The pre-oxidation of the bundle was carried out at cladding temperatures of about 1470K for 4000s. Then the bundle was heated up with 0.2 - 0.3 K/s to a maximum bundle temperature of 1870K. At this temperature, top flooding of the bundle with water (40 g/s) was initiated. The test parameters for the bundle experiments (heat-up rate, steam flow rate, extent of pre-oxidation of the cladding, maximum cladding temperature before quenching, flooding rates) were fixed on the basis of SVECHA code predictions by IBRAE. The cladding temperatures of the bundle were presented as function of time for different bundle elevations during the pre-oxidation and transient heat-up and quench stages. The total mass of hydrogen generated during the test was about 34g. Post-test destructive examinations of the fuel bundle have been performed to determine the extent of cladding and shroud oxidation.

The PARAMETER-SF4 experiment was conducted in an air environment with bottom quenching under the following test conditions. Coolant flow rates: argon 2g/s (670K), steam 3.5g/s (770K) and air 0.5g/s. The pre-oxidation of the bundle was carried out at cladding temperatures of about 1470K for about 6000s. Then the bundle power was decreased to reduce the temperature to about 1200K and switch the flow from steam to air before it was heated up to a maximum bundle temperature of about 2000K. At this temperature bottom flooding of the bundle with water (80 g/s) was initiated. The bundle cool-down took approximately 1000s. The test parameters for the bundle experiments (heat-up rate, steam flow rate, extent of pre-oxidation of the cladding, maximum cladding temperature before quenching, flooding rates) were again fixed on the basis of different code predictions. The measured cladding temperatures of the bundle were presented as function of time for different bundle elevations during the pre-oxidation and transient heat-up and quench stages. The total mass of hydrogen generated during the test was 110g maximum. The amount of hydrogen measured during the pre-oxidation of the bundle did not exceed 21g. Post-test destructive examinations of the fuel bundle were performed to determine the extent of cladding and shroud oxidation for comparison with code predictions.

Conclusions: The PARAMETER-SF3 and -SF4 experiments have provided valuable information on the physico-chemical behaviour of structural components of a 19-rod model VVER-1000 fuel assembly (FA) superheated up to (1600-1750°C) during bottom and combined top and bottom quenching. The tests demonstrated the temperature behaviour of model VVER-1000 FA under conditions of air ingress and the subsequent bottom quenching. Investigations of the degree of oxidation of structural components in a model VVER-1000 FA and phase changes in the FA cladding by posttest researches resulted in the determination the oxygen consumption and hydrogen generation.

The results of the tests will improve understanding of the physico-chemical processes that occur during quenching of the superheated core during severe accident in LWR NPP's, and will be valuable for verification of computer codes.

**Topic #25**: Results of PARAMETER-SF4 numerical modeling

T.Yudina (IBRAE RAN) presented results of numerical modeling of the PARAMETER-SF4 test performed at IBRAE. A comparison analysis (both code-to code and code-to-data) of pre-test calculations performed with integral codes to justify the proposed PARAMETER-SF4 test scenario was presented. The following codes have been applied in reactor calculations of severe accident: SOCRAT (IBRAE RAN), ICARE/CATHARE (RRC KI), RELAP/SCDAPSIM (OKB “GIDROPRESS”), ATHLET-CD (GRS), SCDAP/RELAP/IRS (PSI), and MAAP4 (EdF). By conducting the test in compliance with the developed scenario, at least up to the air ingress phase, the quality of the test modeling could be estimated from these blind pre-test calculations. Most codes predicted the same hydrogen mass about 21 – 22,5g close to measured one (21g). Differences in hydrogen mass calculated with ATHLET-CD and ICARE/CATHARE codes result from temperature deviations from designated temperature (1200ºС) at pre-oxidation phase. At the air ingress phase, pronounced shifting down of the hot spot was observed and before flooding onset it was detected at the elevation of 500 mm. The hottest zone location calculated with ICARE/CATHARE (550 mm), ATHLET-CD (500 mm) and SCDAP/RELAP/IRS (500 mm) is in good agreement with this. The calculated predictions are different from the experimental observations at the flooding phase and is perhaps due to differences in the assumed boundary conditions. Post-test analysis with exact experimental boundary conditions at the flood stage is needed to interpret the bundle behaviour.

PARAMETER-SF4 posttest calculations performed at RRC KI with the ICARE/CATHARE code indicate slight melting at the 450-550mm bundle elevation but intact geometry is kept. The application of the SOCRAT code is restricted at post-test analysis phase due to unavailability of air oxidation model (modified steam oxidation model was used). Post-test analysis with SOCRAT code indicates that high system pressure and reduction of water injection in the bundle at flood phase might be caused by design features of the PARAMETER test facility rather than processes occurring in the test section.

Conclusions: SF4 pretest calculations are in good agreement with measured data except in the flooding phase. Hot spot location before the onset of flooding was predicted over a rather extended zone 500-800 mm. Most codes predicted the elevation of 500-550 mm to be the hottest zone before flooding is initiated. It agrees with measured data (500 mm). Slight melting at the elevation of 400 mm but leaving the geometry intact during the flooding phase could be expected from pre-test calculations. Calculated (pre-test) and measured temperature histories at the flooding stage are different. Post-test calculations are needed to interpret the obtained experimental results. SF4 posttest analysis with ICARE/CATHARE code indicates slight melting at 450-550mm bundle elevations but intact geometry is maintained.

**Topic #26**: Post test analysis of PARAMETER experiments

S.Guentay (PSI) presented results of post-test calculations on PARAMETER experiments. To date, four experiments have been performed in the PARAMETER test series (i. e. SF-1 top injection, SF-2 top and bottom injection and SF-3 top injection). PARAMETER SF-4 is the first experiment with an air ingress phase.

A limited set of post- test calculations of the PARAMETER SF-4 were performed using local versions of SCDAP/RELAP and SCDAP/SIM to support development and benchmarking. The Cathcart Pawel – Urbanic Heidrick (CP-UH) and Sokolov oxidation models were compared with the experimental results. The principal aim of this analysis was to study the temperature behaviour during air ingress at a nominal 0.5 g/s. The oxygen starvation front was another important issue in this study.

The pre-oxidation transient was in fair agreement with the experiment. Some discrepancies between post-test calculation and data during the air and the reflood phase were observed. These discrepancies could be explained by the limitation of the code to represent accurately air ingress. Improvements in the code are currently taking place in order to solve this issue as well as extensions to SCDAP/SIM code for the implementation of oxidation models for Zircaloy cladding in steam and air.

**Topic #27**: PARAMETER-SF4 post-test calculation with MAAP 4.07

Air is a highly oxidizing environment that can potentially lead to an enhanced fission product release that is why it is important to model correctly the kinetics of cladding oxidation by air. PARAMETER SF4 test is designed to study the air ingress issue in severe accident. It was used to validate air oxidation models available in MAAP4.07 that have been implemented from the literature

A.Schumm (EdF) carried out a simulation calculation of the test PARAMETER-SF4 with a customized version of MAAP 4.07, including specific extensions for Zr oxidation in air and material properties of the heater element materials. The simulation covers the time from t=1500s (start of second ramp) to t=16000s (just before bottom reflood). The usual Cathcart/Urbanic correlation was used for steam pre-oxidation; the air oxidation was described by a NUREG correlation. It was taken from the “*Review of the technical issues of air ingression during severe reactor accidents*”. In this report, Powers established three correlations for different temperature regimes. For this modelling, the bundle is composed of four channels: the central rod for the first channel, 6 rods for the second one, 12 rods for the third one and 12 peripheral rods for the last channel.

For the validation of MAAP 4.07 against PARAMETER-SF4, the pre-oxidation phase under steam atmosphere is considered to check approximately the rod temperature and hydrogen production. The initial conditions before air ingress are thus well estimated. During air ingress, the hot point seems to have moved in a lower part of the bundle, in comparison with the steam phase. This is maybe due to an early melting of the bundle and its relocation in a lower part. Thermocouples failure during this air phase renders a precise analysis difficult. This hot point relocation is not reproduced with MAAP4.07, in which the hot point is still at about 1150 mm height. Furthermore, for the post-breakaway regime, no acceleration in the temperature escalation in the overall bundle is modelled, which is due to the use of parabolic laws. Oxygen starvation is predicted 200 s earlier in the modelling than in the experiment, which is a quite good approximation. These different results underline the need for improvements, especially in the modelling of breakaway and post-breakaway regime.

**Topic #28:** Next CEG-SAM meetings, March and September 2011; other matters

The 19th CEG-SAM meeting will take place in Pisa, March 2011 (in the 11th week starting March 14). F.Oriolo (University of Pisa) kindly offered to host the meeting.

V.Stepanenko (USTC) kindly offered to host the 20th CEG-SAM meeting in Kharkov, Ukraine, in the 41th week, 2011 (starting October 10).

M.Hugon thanked once more S.Bechta (RIT-NITI) for the organization of the 18th CEG-SAM meeting in St.Petersburg. He also expressed his thanks to all speakers and participants for their engagement at the meeting.

**Restricted session** (continued)

**Topic #29:** SARNET and CEG-SAM comments on ISTC & STCU proposals

M.Hugon presented once more a short overview on the current status of the various ISTC and STCU projects.

**Topic #30:** Detailed discussion of presented ISTC and STCU project proposals and preparations of specific CEG-SAM advices

A general discussion took place on future funding models for ISTC projects since the funding will be very limited. One possibility could be co-funding of projects by several sources, for example, Russian partner(s) institutions, EU institutions, and ISTC. Projects without ISTC financial support should cause fewer problems with export control measures, intellectual property rights and non proliferation.

Another possibility is the submission of coordinated Euratom-ROSATOM proposals (such as ERCOSAM-SAMARA) to nuclear fission calls. In these cooperative actions, Euratom will fund solely the EU partners, while ROSATOM will support the efforts of Russian partners. In each cooperative action, the Russian contribution is expected to be equivalent to that of Euratom.

**Topic #31:** Discussion of various actions

**Topic #32:** Other matters; Final remarks

The chairman M.Hugon thanked once more NITI (Sosnovy Bor) for hosting the meeting and for all its related excellent efforts and he thanked also the participants for their efficient work and contributions and wished them a safe journey back home.

**M. Hugon** (chairman) **P.Hofmann** (secretary)

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**Annexes:**

1. Updated final agenda of the 18th CEG-SAM meeting
2. List of participants at the 18th CEG-SAM meeting