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Topical Meeting
on ISTC Projects 1648.2, 2916, 2936, 3194

MINUTES

Podolsk-Moscow, Russia
NPO LUCH
July 3-5, 2007

Subject:	Topical Meeting on ISTC Projects 1648.2, 2916, 2936, 3194
Place:	NPO LUCH, Podolsk; KI, Moscow
Date:	July 3-5, 2007
Participants:	22 participants of 9 organizations from 3 countries:

Mr. D.Bottomley	EC, DG JRC / ITU, Karlsruhe
Mr. W. Erdmann	GRS, Garching
Mr. Ch. Journeau	CEA/DTN, Cadarache
Mr. A. Miassoedov	FZK, Karlsruhe
Mr. M. Steinbrück	FZK, Karlsruhe
Mr. J. Stuckert	FZK, Karlsruhe (**secretary**)
Mr. A. Borovoi	KI, Moscow
Mr. A. Boldyrev	IBRAE, Moscow
Mr. S. Gavrilov	IBRAE, Moscow
Mr. A. Goryachev	RIAR, Dimitrovgrad
Ms. L. Degtyareva	NPO LUCH, Podolsk
Mr. D. Ignatyev	NPO LUCH, Podolsk
Mr. A. Kisselev	IBRAE, DNS, Moscow
Mr. V. Konstantinov	NPO LUCH, Podolsk
Mr. V. Nalivaev	NPO LUCH, Podolsk
Mr. A. Palagin	JRC / ITU, Karlsruhe
Mr. N. Parshin	NPO LUCH, Podolsk
Mr. V. Semishkin	OKB GIDROPRESS, Podolsk
Mr. V. Shestak	IBRAE, Moscow
Mr. V. Strizhov	IBRAE, Moscow
Mr. M.Veshchunov	IBRAE, Moscow
Ms. T. Yudina	IBRAE, Moscow

Topic #1: #3194 PARAMETER Project meeting, 3rd July '07, Podolsk (Participating Institutions: LUCH, Gidropress, IBRAE)

The meeting was opened by the Deputy Director–General of Luch *V. Deniskin*, who welcomed the participants to Podolsk and Luch.

Luch is scientific production association and was established as a research and development and production facility for rare-earth metals for the nuclear industry. The construction engineering company OKB Gidropress is located also in Podolsk and is involved in research and design of VVER reactors, reactor components, and safety and control systems. Russia's Atomenergomash and France's Alstom have just signed (end June) documents to establish a joint venture to manufacture half-speed turbines for nuclear power plants in Russia and abroad. The Joint Venture will operate on the platform of the machine-building firm ZIO Podolsk.

Luch performed during the last years different investigations in the field of the Design Basis Accidents (DBA) and Beyond DB Accidents using the PARAMETER facility. The links with the QUENCH Project were also of great value both for model verification and for simulating severe accident measures. Particularly the effectiveness of top flooding was of importance during the last ISTC-PARAMETER project. The tests corresponded to the 19-rod fuel assembly of a VVER-1000 type. The rods are natural UO_2 with Zr-1%Nb cladding and with central electrical heating by W rods. Comparison with the QUENCH facility results was very useful to see the geometry effects of a VVER vs. western PWR bundle design.

The PARAMETER-SF1 test had been performed on 15.04.2006 and had involved a preheating and testing phase before the overheating to ~2000°C, and then the water-quenching of the bundle by top flooding. After the initial, rapid top cooling, there was cooling from the bottom and it was realized that there had been a by-pass. Furthermore, the extreme temperatures had also melted many of the central thermocouples. In the post-test examination the central hot zone was observed to have been extremely degraded and that there was a slug of corium that had formed a blockage and caused the by-pass. For this reason it was thought that the SF2 should not be so hot, therefore it was proposed to set the maximum temperature at 1500°C. This had been successful, and the thermocouple data was complete. The PARAMETER-SF2 was a combined top and bottom flooding test of the overheated fuel assembly compared with the SF1 top flooding only test.

The first presentation was the pretest calculations of the SF2 test by *T. Yudina*, IBRAE. Here one of the main points was the determination of the H_2 source term during the test. Other point was the status of structural elements. This was modelled using SOCRAT, ICARE/CATHARE and PARAM TG codes. The latter is a thermal/hydraulic code specific for the PARAMETER bundle. The modelling showed reasonable correspondence for the pre-oxidation phases and heating power, although the transition phase was difficult with considerable power losses to the thermal shroud (~50%). ICARE was different from SOCRAT for the heating losses. The hot zone was correctly estimated by the codes at 1200-1300 mm height - just at the top of the heated zone (heated up to 1275 mm). SOCRAT predicted the rapid top cooling but some reheating at 1250-1400 mm was predicted as the combined top & bottom cooling was performed. ICARE calculations predicted no fast cooling from the top flooding. With the bottom cooling there was only a gradual cooling down of the central bundle by the steam /water mixture.

The second presentation was the experimental results of the SF2 PARAMETER test by *V. Konstantinov*, LUCH. The test facility had 18 heated rods with a central unheated rod (natural UO_2 with Zr-1%Nb cladding and cylindrical /ID = 70 mm, thickness 2 mm/ Zr-1%Nb shroud; instrumentation with 47 bundle thermocouples and 4 shroud thermocouples). There had been major refurbishments to the facility. The middle section had been redesigned to make it more air-tight and avoid a by-pass as before. There were improved Ar lines to enable switching between upper and lower sections of the bundle. There had also been improved flooding facilities, with separate tanks for the upper and lower ends.

During the test they underwent the initial pre-oxidation to grow a thick oxide layer (3000 s at 1200°C), then they were ramped to achieve the target temperature of 1450°C at the hottest elevation (+1250 mm). Initially there was just top quenching, then after ~25 s bottom quenching was started. The topmost section 1400-1500 mm was rapidly cooled with the top cooling, but the next zones (the hottest: 1250-1300 mm) were not affected by the top cooling. The bottom cooling was effective for the lowest regions with 600 mm and below being cooled immediately. The next region (e.g. 800 -1000 mm) was cooled only after a long delay. However the hottest zone (1250-1300 mm) after complete quenching with the bottom cooling onset they then displayed substantial, if not complete reheating phases. This was presumed to be due to the further steam oxidation or even H_2 release from the metallic Zr-1% Nb cladding or shroud causing heating with oxidation. The system pressure rose 50 s later after initiation of the bottom reflood.

The H_2 measurements also showed that the production was in bursts, but mostly came in the later, combined bottom & top cooling stages. The H_2 total was 28 g - practically all from the combined cooling phase (this compares with 130 g of H_2 during SF1 - mostly from the bottom cooling).

The top flooding appears to have been blocked by a steam ‘pressure wave’ especially when the bottom flooding started. It was noted that this is a problem caused by the bundle's cylindrical geometry so this effect is unlikely to happen in a reactor. It was noted that the top flooding is certainly more destructive or more demanding on the materials’ mechanical properties but nevertheless it appears to be effective.

The third PARAMETER presentation was for the material investigation of the SF1 fuel assembly and the post test appearance of the SF2 bundle by *D. Ignatyev*, LUCH. SF1 had been severely degraded and its examination was performed in 5 zones from top to bottom. In the central zone (~650-800 mm height) there was the large lump of corium that probably acted as a blockage since a pressure rise was also observed during the experiment. The corium appeared to be a sub-oxidic material of Zr-U-O, with 3 zones: a light, a grey and a dark phase. A eutectic Zr-SS was noted along with a (U,Zr)O partial oxide. Parts of the corium also contained Fe, Cr, and Ni. An U,Zr intermetallic and α-Zr(O) were also observed as interaction products between the UO_2 fuel and the Zr cladding. The central part of the corium blockage had separate metallic and oxidic areas; this implied that these zones represent flows down to the blockage at different times. Above the corium lump there was a zone of strong interaction and oxidation (800 - 1000 mm), nevertheless the UO_2 pellets had been dissolved by the melted cladding. The melt had also penetrated down the cracks in the fuel pellets. At the top (1000-1200 mm) there was still considerable oxidation with through-wall cracking of the cladding. Moreover there were also a lot of small fragments that may have spalled from the oxidized shroud.

SF2 was in stark contrast and had no such melt or blockage. It displayed considerable oxidation but was still in reasonable condition with the shroud intact. The results also confirmed that it was remained leak-tight up to 1500°C.

The afternoon was spent *visiting the PARAMETER Facility* in LUCH.

It was also reported that a follow-on project had been submitted (PARAMETER-SF #3690; Leading Institution: Scientific Industrial Association “LUCH”, Project Manager *V. Nalivaev*) which was to perform a metallographic examination on the SF2 bundle and two more, top– only flooding quench tests from different temperatures. It was reported that it had received full financing at the recent Governing Board Meeting in June.

Topic #2: #1648.2 VVER QUENCH, 4th July '07, Podolsk, (Participating Institutions: RIAR, IBRAE)

This project meeting reported the progress since March 2007. It was expected to finish during the summer 2007, but they had requested a delay up to the end of the year (Dec. 2007) in order to properly finish certain analyses that had needed repeating.

The first presentation (*by A. Goryachev,* RIAR) gave a review of the latest project's results (3 single rod quench tests) after repair of the oven in the hot cell with quench initial temperatures of 1400°C (1 test) and 1600°C (2 tests). The burn-up is of the order of 65 GWd/tU and tests were performed without pre-oxidation (quenching with water after pre-conditioning phase in Ar). The samples all had embrittled cladding after the quenching.

On-line gamma spectroscopy was used to check the fission gas (^{85}Kr) release rate. On-line mass-spectrometer measurement was used to check the release of the stable Xe. Total releases of Xe were variable at 5 and 34 ml. The initial and residual content of Xe and ^{85}Kr was measured by dissolution of the fuel reference and annealed samples correspondingly. The ^{85}Kr on-line measured releases were variable at 4% to 31%. The ^{85}Kr release calculated by dissolution was between 2% and 42%. The fuel dissolution technique appears to work well for simulators tested at higher temperatures: (~1700°C), but for the simulator tested at lower temperature (~1400°C) the error exceeds the result.

Scanning of the rods showed that the losses of ^{137}Cs were mostly in the pellets of the upper, hotter regions of the 15 cm rod. The relative Cs releases were estimated based on the ^{154}Eu (1274 keV) activities in the fuel (it was assumed that this isotope was not released). The ^{137}Cs γ-scanning releases were 3 to 5%. Finally the release of ^{137}Cs was calculated on the base of fuel probes dissolution in HNO_3; the measured values are between 2% and 38%.

The Cs content in the quench water was also determined (and was 0.6 to 1.3% of ^{134}Cs & ^{137}Cs).

The second presentation (*by A. Goryachev*, RIAR) was the post-test examination of the bundle. The QUENCH-12 test was similar to the QUENCH-06 test except Q-06 had PWR geometry and cladding material Zircaloy-4 while Q-12 had VVER-1000 geometry and cladding material E110 (Zr1%Nb). However the H_2 production was very different. The thermocouple data had been used to estimate the oxide layer thickness and check for heterogeneity in the temperature field.

5 samples (elevations 554 mm, 654 mm, 754 mm, 854 mm and 954 mm) were sent from FZK to RIAR for post–test examination. Residual metal thicknesses were measured for all rods of these 5 cross-sections. For the simulator rods most of the oxide had spalled off at 554 mm height and many fragments in between the rods were noted. It was presumed that they had probably fallen from a higher elevation. At 654 mm height, the oxide spallation was also considerable but there were no fragments between the rods. The variation in the oxide layer thicknesses indicated that thermal gradients existed during test at each elevation. At 754 mm a smaller breakaway oxidation were observed. Very strong oxidation was evident at 854 mm. A large droplet of melt was attached to the oxidized cladding. There was much distortion of the rods. At 954 mm there was complete oxidation of the claddings with no residual metal remaining. In other areas the oxidized cladding had completely spalled away. Locally some melt was seen in the cladding–fuel gap and between rod simulators. Formed melt is completely oxidized

The solid rods were also examined to note the oxidation layering and revealed up to 200 µm loss of radius.

In conclusion it was seen that severe cracks were not oxidized so that the cracking occurred only during the quench. The oxidizing of the melt is thought to be the main contribution to the observed high hydrogen production.

The third presentation considered the modeling of the RIAR tests on fresh & irradiated fuel _by A. Boldyrev_ (IBRAE). The RIAR quench tests with the fresh fuel rod simulators were modelled with the SVECHA/QUENCH (S/Q) code. The simulation well reproduced: a) temperature evolution of the outer surface of the fuel rod cladding and the fuel pellet centre, b) mechanical state of the oxidized Zr-1%Nb cladding after the quenching. By contrast the prediction of the total hydrogen generation differs from the measured values. There are two possible reasons for this discrepancy:

1. Temperature measurement. In the most of the tests only the central TC was used. The reason was the measurement error of the surface TC during the quench phase. At the same time, the surface TC measurements during the other phases are very reliable and provide the real temperature profile and the oxidation kinetics before quenching.
2. Steam measurement. First, the mass flow rate of the steam is not measured and controlled during test and in some cases starvation instead of oxidation occurs. Second, inside the test rig an unknown amount of steam is always present due to water evaporation from the flooding tank.

Comparison of the obtained experimental results and S/Q code simulations of the non-irradiated VVER fuel rod simulators with FZK quench test data allows us to conclude that generally, under the same test conditions, the mechanical behaviour of Zr-1%Nb and Zry-4 cladding simulator is very similar. This fact leads us to the assumption that the reasons of the through-wall crack formation are the same for the both alloys. The main difference in the behaviour of these alloys under these conditions is the appearance of the breakaway oxidation in the VVER cladding in the range of 900 to 1000 °C. On the oxidized VVER cladding one can observe surfaces of three types: a) covered by lustrous black ZrO_2, b) covered by numerous white spots, c) surface with spalled oxide scale.

Further code developments allowed us to simulate the RIAR quenching tests with irradiated uranium fuel rods. First detailed simulations were performed for two irradiated rods and showed a good correlation with experimental results.

Cs release was also calculated by means of the MFPR code developed at IBRAE in collaboration with IRSN. The results were very encouraging: e.g. at 55 GWd/tU they got Cs 6.4 % release modeled against 4.6% experimentally obtained.

The fourth presentation (by _A. Palagin_) concerned the SVECHA/QUENCH modeling of Q- 06 & Q-12 tests. This involved the re-evaluation of the Q-06 thermocouple data and re-nodalisation to allow for the central rod being unheated (effective channel approach). The recalculation of the data using the corner rod - that was withdrawn early in the test - gave an improved correspondence with the experimental values. Also certain T/C's that passed through hot zones gave anomalous results with premature peaks. These were due to the cabling passing through zones hotter than the measurement zones so that there is either 'shunting' effects or excessive potentials (too high a reading). After discounting this, the remaining data gave much more comprehensible results and with the estimated central rod values gave better agreement. The comparison of the model with the Q-06 experimental data revealed an unexpected reheating at the top end of the bundle. This is thought to be due to the sudden dry-out of the rods with steam 'blast' and hence a temporary reheating. The H_2 production and its timing were also well-predicted. The experimental comparison with Q12 which had a VVER-1000 geometry shows that more hydrogen is formed in Q-06. (36 g total H_2 for Q-06 compared to 24 g total H_2 for Q-12). There are 20 heated rods from 21 in Q-06 but 18 from 31 in Q-12 along with the central unheated rods. This resulted in a substantial temperature difference in the radial temperature profile for the Q-12 that is not seen in Q-06. Thus the central rod in Q-12 could be at 1200°C at 950 mm while other rods at the same elevation would be 100-150 K cooler. At this temperature, breakaway oxidation is more likely and so there would be more intense oxidation and more friable oxide in Q-12. The rod withdrawn midway through the Q-12 test showed that breakaway oxidation had occurred at low and high elevations with greater oxide thickness than expected. By the end of Q-12, there were higher max. oxide thicknesses (& hence temperatures) at +950 mm than those modeled and also the reheating phenomena (steam 'blast' dry-out) were more evident in Q-12.

So Q-12 & Q-06 experiments showed differences: in Q-12, the H_2 production was a bit under-estimated - mainly during the quench phase. In comparison, Q-6 had greater oxide thickness with earlier and greater H_2 production than for Q-12. A model for the breakaway oxidation is suggested for SVECHA/QUENCH following the Q-12 observations.

Topic #3: #2396 REACTOR CORE DEGRADATION MODELLING, 4th July '07, Podolsk (Participating institution: IBRAE)

This project is due to finish in August 2007.

The first presentation was 'Model for corium melt oxidation and its application to slug (blockage) relocation' by *M. Veshchunov*, IBRAE. The main starting point for the model was the FZK small pot tests for the UO_2-Zr and ZrO_2 –Zr systems. The model correctly describes the following main observations of crucible tests for the Zr-ZrO_2 system: 1) erosion: ZrO_2 dissolution, 2) corrosion: subsequent growth of the oxide crust at the melt-wall interface, 3) Secondary dissolution of oxide crust in the late stage, 4) Growth of oxide layer on the free (non-oxidised) surface of melt, 5) Heavy precipitation of ceramic phase in sub-oxidised melt.

Here it is shown that oxygen can diffuse down a temperature gradient that will always exist at a boundary such as the crucible or the upper crust. This will continue until saturation and then complete precipitation. This final stage of heavy precipitation in sub-oxidized ceramic and oxide crust formation on free non-oxidized melt surface was then proved with the long term tests at FZK. Calculations showed that even slight gradients can be sufficient (ΔT=1-2K). Most important is the direction. The oxygen flows down a concentration gradient and so can withdraw oxygen from a hot side surface (that is inductively heated) to produce an oxide crust on a cold top surface of a crucible, even if there is already a top crust. This was then checked for the remaining pot tests and then extended to the FPT0 and FPT1 bundles and it was able to show that the melt could have been metallic on solidifying but oxidized almost completely (85% oxide). This would explain the high melting point measured for the solidified melt (higher than the maximum temperature measured in the bundle).This could also be modeled as movement across the pseudo-ternary phase diagram towards the UO_2-ZrO_2 tie-line.

The model was also developed to deal with molten pool (MP) oxidation in the lower head of a reactor vessel. Model predicts that in oxidizing atmosphere the oxide crust growth will take place on internal side of the walls accompanied with (possible) precipitation of ceramic phase in the melt bulk.
After relatively quick attainment of steady-state conditions with oxide crust thickness of several mm: 1) cooling of the MP through walls can be significantly suppressed; 2) the MP temperature will further increase; 3) possible corrosion (oxidation) and/or melting of a thin metal layer at the interface between the oxide crust and the wall can occur.

To verify these predictions the new 2-D MP oxidation model was applied to simulate melt oxidation behaviour in simplified crucible tests (the possibility of conducting such tests is under discussion with FZK and KI). Lacking detailed thermal hydraulic consideration, the main model predictions can only be extended qualitatively to large scales (eg. RPV). A quantitative solution of the problem can be sought by coupling the MP oxidation model with the thermal-hydraulic code CONV (ISTC Project proposal THOMAS)

The final aspect of the project was the development of the Slug Model for the SVECHA code to explain the molten pool relocation. The Zircaloy melting and fuel dissolution results in small rivulets of melt that trickle rapidly down the rods to freeze on a fuel rod assembly or other cool objects. They flow very rapidly and freeze together into a slug but the lower crust slowly breaks and releases rivulets through its lower crust. The rivulets flow down at ~50 mm/s, whereas the slug itself has an effective net velocity of 1-2 mm/s –sliding slowly down (and interacting with) the rods. Calculations estimate the critical thickness of the crust as ~100 μm.

The second presentation explained the following stage of introducing the slug model into the SVECHA/QUENCH model and validating this against the CORA tests (*V. Shestak*, IBRAE). The results of CORA W1 &W2 were modelled with the slug heat exchange model and a simplified geometry (use of a single rod of equivalent dimensions). They were able to model the movement of the slug formed in CORA very well with the thermocouple data (the TC data could be used to trace the flows of the hot molten slugs). The model was also able to predict the evolution of the composition, downward velocity and the pool height; it is now successfully integrated into the SVECHA package.

Topic #4: #2906 CHESS-1. Model for Nuclear Fuel Behaviour during the active phase of the Chernobyl Accident, 5th July '07, Moscow (Participating institutions: KI, IBRAE)

This represents the final meeting of the project as it is due to finish in August 2007 after 2.5 years. The meeting was held on the third day at the Kurchatov Institute, Moscow. The meeting was opened (*first presentation*) by *A. Borovoi, KI,* to present the main conclusions and also, in the second part, discuss the proposed follow-on project.

The second presentation was from *V. Strizhov, IBRAE* looking at the modelling of the Fuel-Containing Material (FCM) behaviour. The first aspect is the data uncertainties. The best example is the amount of material dropped on the reactor. Estimates are 16600 tonnes dropped into the reactor of which 5000 tonnes was in 2 days. Others were boundary conditions, degree of air cooling of debris and amount of graphite burning.

The so-called 'Pancake' model of the 'layers' of materials that made up the ensuing melt were: 1) Basemat concrete 2) the under-reactor structure, 3) structural steel & serpentine 4) Fuel containing masses (cladding & fuel), 5) sand and lead dropped into the reactor. There is an estimated 1200 tonnes of material in the FCM. The time of 10 days is considered to be the sufficient for the reactions to be completed. Times of graphite burning of 4, 7, & 10 days were used together with the data available for estimation of the corium penetration depth and decomposition depth to estimate the ablation rates. It was concluded that the final state was reached after 12-15 days. The erosion depths for the basemat concrete were typically 0.5 -0.7 m in the room 305/2 beneath the reactor and the temperature of the melt in its final position was thought to have been typically 1300 K down to 450 K in the concrete below with a cooling time of 20 h (although initial temperatures during the fire/oxidation/reactions would have been higher). The initial temperature would vary with the fuel content (black is ca. 4 wt% U and brown is ~8 wt%U and is essentially UO_2 fuel diluted with Serpentinite (impure Mg_2SiO_3 containing Fe and lesser amounts of Ca, Al)).

The criteria for the spreading & stopping for the model required definition. The spreading depends upon the kinematic viscosity (ν) which is very dependent on temperature (if T rises from 1000 K to 1050 K, ν drops by a factor 10 and the spreading time (30 m) drops from 12 h to 1.2 h; above 1373 K spreading time << cooling time). The model was set to stop the lava flow below 1100 K. It was noted that the base plate broke into 2 parts during the explosion and that one half fell down on to the steam distribution passages and blocked the lava from flowing in this direction. In conclusion the lava is thought to have flowed very rapidly in the early stages at 1273-1373 K and then cooled and interacted mostly when it was in place.

Decay heat was approx. 90% of the total heat with 10% coming from the chemical reactions (graphite burning & Zr oxidation). The concrete wall in room 305/2 had 0.93 m intact thickness. The axial erosion was 0.7 m as was the radial erosion, except at the corner where extreme ablation had occurred. This weakness at the corner was of interest to C. Journeau, CEA as this corresponded to findings in their VULCANO test bed for MCCI. The estimates from the model of the fuel materials distribution in the main rooms and steam passages beneath the reactor corresponded very well with the amounts based on expert evaluations. Fearing further lava penetration into the earth, they had constructed some time previously a water-cooled sub-foundation plate of concrete (30 x 30 x 2.5 m thick) under the reactor hall. Following this modelling it is now realised that this was an excessive measure.

In the large black lava mass in room 305/2 under the reactor they have noted at least 2 'hotspots'; these higher temperatures imply higher uranium contents. They have retrieved samples of up to 30 wt% U in the bores extracted from these lava masses. On one occasion a 60x increase in neutron activity of a hotspot was noted following very heavy rain in Chernobyl. This was thought to be due to water ingress of this high U concentration zone and it becoming critical temporarily.

The model can be used to calculate the amounts in the reactor rooms and the steam passages beneath. It can be checked against the known distribution but it can also locate possible fuel concentrations. The model and the database will be included in the final reports which will be an important contribution for the SIP (Shelter Implementation Plan) where it is intended to build a moveable construction that can go over the old sarcophagus and allow the sarcophagus and the reactor structure to be dismantled & the FCM safely removed for the gradual return of the land to a 'green site'. The Database includes around 6000 publications, papers, reports photos and other items. This database enables the lava generation model to be formulated and gives the starting point for any melt spreading. It also includes the model for the lava spreading. This data will also be useful to the Ukrainian institutes.

Topic #5: #3702 CHESS-2 – Long Term Behaviour of Corium after Accident (using the Data of the Chernobyl NPP Accident)

This is the follow-on project of the CHESS-1 project and will last 2.5 years. It will build on the model developed in CHESS-1 to estimate the distribution of the Fuel and FCM in the Reactor. Dust has been observed in the Shelter and it has feared (a suspicion picked up and widely reported by the media) that the lava could in the long term degrade into fine, even sub-micron dust which would easily permeate out of the Sarcophagus or even the new Shelter (SIP) to be constructed. *A. Borovoi* does not believe that this process will be so extreme since the U level is low (mainly between 5 & 10 wt%) so the self-irradiation in the long term (by α-decay) will be low. However it is very important to assess the speed of this process (particularly for the few 'hot spots') and the future risks.

The Project was presented by *S. Gavrilov*, IBRAE. The Project's objective is to examine the available samples and their properties and construct a model to predict their degradation with time. They will look at temperature, humidity / water immersion and self-irradiation effects in the Shelter. The data will include the testing of low & intermediate -level active waste glass under external weathering; the climate in the Shelter is more 'moderate' so that these are accelerated tests but will still give very useful information.

This will be done in parallel with the project carried out by the UIAR, Kiev and sponsored by the STCU (#4207: Long -Term Prognosis of Behaviour of the Fuel Dust in Chernobyl Shelter). The UIAR project will examine the degradation of the fuel and corium particles that were formed during the accident and lie on most surfaces in the reactor hall. This is the main source of dust compared to that formed by lava degradation. UIAR will analyse the existing data base and make further specific tests of selected samples before developing a model for its long-term degradation. UIAR have noted that the fuel dust/particles do not seem to spread as fast as expected. The researchers at UIAR, Kiev are well known to Dr. Borovoi and his team at the Kurchatov Institute and they look forward to an interesting & valuable project for both institutes.

The projects will have joint meetings for discussion of mutual progress and exchange information (such joint ISTC/SCTU projects have been carried out in the past). Particularly in the last stages the two models for the degradation of the lava and the fuel particles will be combined to create an overall model for the behaviour of all fuel-containing forms in the reactor hall in the time frame: 10 days to 100 years. This will help with the prediction of conditions in the Shelter especially for its final removal for disposal at some stage in the future under the Shelter Implementation Project (SIP).

Finally there was a request to have all the CHESS reports in electronic form for the next CEG-SAM meeting. A copy of the input deck of IBRAE's modelling would also be of great use to C. Journeau for the CEA to do some modelling of the Chernobyl lava flow with their codes.