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Pre-Test Calculational Support for the QUENCH-13 Experiment

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Abstract – *The QUENCH experimental programme at FZ Karlsruhe investigates phenomena associated with reflood of a degrading core under postulated severe accident conditions, in the early phase where the geometry is still mainly rod-like. The latest large-scale bundle test, QUENCH-13, is the first in this programme to include a silver/indium/cadmium (SIC) control rod of prototypic PWR design. The effects of the presence of the control rod on early-phase degradation and on reflood behaviour are examined under integral conditions, while the opportunity is taken to measure, in realistic geometry, release of SIC aerosols following control rod rupture. These materials can affect the chemistry of fission products in the reactor circuit, and hence the radioactive source term to the environment in the event of containment failure. In particular, the sharp release of cadmium on control rod failure, which can involve some tens of percent of the inventory, is ill-defined experimentally. Pre-test calculations were performed to determine the test boundary conditions, such as the electrical power history to the bundle, the coolant flow, and the reflood timing and rate. The aim was to stabilise the bundle at maximum temperature of 1250 K, then ramped at about 0.25 K/s to give the best chance to measure the control rod aerosol release under controlled conditions, then to reflood, without provoking an oxidation excursion, at maximum bundle temperature of 1800-1850 K. A further aim was to check thermal conditions in the offgas pipe, where the aerosol instrumentation was situated. The calculational support was organised through the Source Term area of the EU 6th Framework Network of Excellence SARNET, linking the experimental team at FZK with modellers at PSI, GRS and EdF. Following agreement of the target test conditions, the modelling teams used SCDAP-based codes, ATHLET-CD and MAAP4 respectively to help the definition of the test boundary conditions, and in the latter two cases to estimate the control rod aerosol release. The facility models used were benchmarked against data from previous QUENCH tests, while also the ATHLET-CD release modelling was checked against Phebus FP data. The experimental protocol took account of the recommendations from the pre-test studies. Benefit was gained in the cooperation through the use of independent codes by different organisations, in lending confidence to the test predictions, and in obtaining different perspectives on the test conduct. The experiment was successfully performed according to the agreed specification on 7 November 2007, and the results are to be analysed on a collaborative basis. Post-test calculations are planned following release of the definitive results.*

I. INTRODUCTION

An important accident management measure to terminate a severe accident in a light water reactor is to inject water to cool the uncovered degraded core. Analysis of the TMI-2 accident[1] and results of integral experiments[2] showed that before core cooling is established, this action may provoke enhanced oxidation, causing a sharp increase in temperature, hydrogen production and fission product release, which may threaten containment integrity and increase the probability of release to the environment.

The QUENCH programme[3] at Forschungszentrum Karlsruhe (FZK) investigates hydrogen generation, material behaviour, and bundle degradation during reflood. It provides experimental and analytical data to assist development and validation of models used in reactor accident analysis codes. Integral bundle experiments are supported by separate-effects tests (SET) and code analyses. The latest experiment, QUENCH-13, investigates the effects of the presence of a PWR control rod on early-phase bundle degradation and on reflood behaviour under integral conditions. The opportunity is also taken to measure, in a realistic geometry, release of silver/indium/cadmium aerosols following control rod rupture. Such data are required for modelling structural material release in postulated PWR severe accidents, as the Ag, In and Cd can react with radiologically important fission products such as iodine and affect their potential release to the environment[4].

Analytic support is provided cooperatively in the Source Term area of the EU 6th Framework Network of Excellence SARNET[4], by PSI (Switzerland), GRS (Germany) and EdF (France), as well as experimental support for aerosol measurements by PSI and AEKI (Hungary). This cooperative support extended that provided by PSI for the three previous tests[5].

The paper concentrates on how judicious application of code models, typically two or more independent codes, by different organisations, has enabled definition of the test protocol to promote the achievement of experimental objectives in a safe and reliable manner.

II. QUENCH FACILITY AND TESTS

The QUENCH programme at FZK started in 1996 as the successor of the CORA programme in which material interactions under the conditions of a hypothetical severe nuclear accident were investigated, with increased emphasis on quantifying hydrogen production during reflood. The main component of the QUENCH facility is the bundle, which comprises typically 21 fuel rod simulators about 2.5 m long, of which 20 are heated over a length of 1024 mm by 6 mm diameter tungsten heaters in the rod centres, surrounded by annular ZrO_2 pellets to simulate fuel. The geometry and most other bundle components (Zry-4 cladding, grid spacers) are prototypical for Western-type PWRs, except for QUENCH-12, that used VVER-typical materials and geometry. The central rod is unheated and is used for instrumentation or to simulate a control rod. The heated rods are filled with argon-krypton or helium at about 0.22 MPa to allow rod failure detection by the mass spectrometer. The pressure in the test section is around 0.2 MPa. Four Zircaloy corner rods are installed to improve the thermal hydraulic conditions and to mount additional thermocouples. Two of these rods can be withdrawn during the test to determine the axial oxidation profile at critical phases. The bundle is surrounded by a Zircaloy shroud, a 37 mm thick ZrO_2 fibre insulation, and a double-walled stainless steel cooling jacket. The shroud provides encasement of the bundle and simulates surrounding fuel rods in a real fuel element (Fig. 1). The whole set-up is enclosed in a steel containment.

The test bundle, shroud, and cooling jacket are extensively equipped with thermocouples at different elevations and orientations. The test section incorporates pressure gauges, flow meters, and a water level detector. Hydrogen and other gases are analyzed by a mass spectrometer at the off-gas pipe about 2.7 m behind the test section. A redundant hydrogen detection system, based on heat conductivity measurement of binary Ar-H_2 mixtures (CALDOS), provides data when no gases other than Ar, H_2 and steam are present.

A QUENCH experiment typically consists of the following phases: heat-up, pre-oxidation, transient, and quenching or cool-down. During heat-up the bundle reaches temperatures at which cladding oxidation begins at the upper elevations. The temperatures are then controlled at a roughly constant level to achieve the desired oxidation before a further excursion is initiated, usually by an increase in electrical heating. The excursion can result in maximum bundle temperatures of well above 2000 K and is accompanied by increased hydrogen generation. During most of the test, a flow of 3 g/s steam and 3 g/s Ar as carrier gas for H_2 measurement is typically maintained. During the last phase, water or saturated steam is injected at the bottom of the test section, and power is reduced to simulate decay heat, or turned off completely.

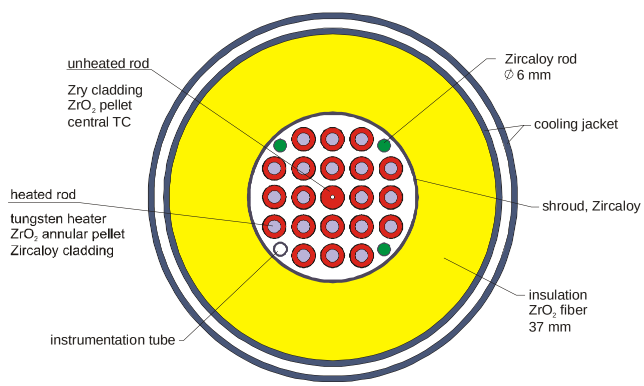

Fig. 1 : Cross-section of QUENCH test bundle of PWR type**.**

Up to the start of the present work, twelve bundle experiments had been performed, with varying degrees of pre-oxidation, mode of reflooding/cool-down, bundle/cladding type (PWR/VVER), presence or absence of boron carbide absorber material, and rates of flow and steam/gas composition through the bundle. Experience has shown that the thermal response of the bundle can be very difficult to control, particularly during transition phases of the tests such from heat-up to pre-oxidation and the reflooding/cool-down. Indeed, the challenges arise from the very reason that the tests are needed, namely to eliminate limitations in current knowledge of phenomena that pose safety concerns to nuclear plants.

In QUENCH-13, the single unheated fuel rod simulator in the centre of the normal 21-rod bundle was replaced by the PWR control rod. On-line particle counting equipment was installed in the offgas pipe to enable measurement of aerosol flow rates, while the provision of impactors enables data on the physical and chemical forms of the aerosols to be obtained.

III. PRE-TEST CALCULATIONAL SUPPORT

III.A. Planning of test conduct

Definition of the experiment involved intensive discussion within a specialist technical circle within the Source Term area of SARNET, involving FZK, EdF, GRS, IRSN and PSI. It was decided to use the QUENCH-06 sequence as the basis for the test conduct; this involved pre-conditioning the bundle at about 1473 K for about 4600 s to build up a maximum oxide layer thickness of about 210 µm, before ramping the bundle to about 1973 K and reflooding with room temperature water at about 40 g/s to terminate the test. In the present case, the plateau temperature was reduced to precondition the control rod prior to the final thermal transient within which its failure would occur and the absorber material would be released. This procedure helped to optimize the conditions for measurement of the control rod aerosols released. Experience from bundle experiments, e.g. Phebus FPT1 as summarised for example in an earlier review[7], indicates a rapid release of Cd following control rod rupture, of several tens of percent of the initial inventory in a few seconds, 'burst release', followed by release at much lower rates of the In and Ag components, e.g. a few percent over a few thousand seconds. The aerosol measurement strategy needed to cope with these very different conditions.

The planning analysis focussed on defining a suitable temperature during the pre-conditioning phase, a power ramp rate to provide an adequate time window after control rod failure during which to measure the aerosol transport, and a reflood initiation temperature to avoid an oxidation excursion and hence prevent damage to the bundle during quench. Off-gas pipe conditions were also investigated, to ensure temperatures would remain within the operating envelope of the aerosol measurement system. Variant studies were performed to evaluate the effect of different reflood criteria taking into account timing and temperature uncertainties, different Zircaloy/steam oxidation kinetics and uncertainties in the control rod failure temperature. A contingency action was identified in case the control rod failure was not apparent until a higher temperature than expected.

The determination of the QUENCH-13 protocol was based on numerous calculations with SCDAP/RELAP5[8], SCDAPSIM[9], ATHLET-CD and MAAP4, performed by PSI (SCDAP-based codes), GRS and EdF respectively. The SCDAP-based codes have been extensively and successfully used for defining the protocol of previous QUENCH tests, while ATHLET-CD and MAAP could estimate the control rod aerosol release rates, valuable in the test planning. The combined approach took advantage of the relative strengths of the codes. The final pre-test calculations were performed by PSI with SCDAP/RELAP5. The calculations performed by each organization are summarised in the following three sub-sections; an overall summary is given after.

III.B. PSI calculations

The PSI models for QUENCH-13 were developed from those used in analyses of previous QUENCH tests, which had been benchmarked against experimental results. The models use 16 SCDAP axial nodes for the bundle, of which 10 represent the middle tungsten heated section, 2 each for the neighbouring molybdenum conductors, and one each for the top and bottom copper electrodes. In the radial direction, separate SCDAP components represent the control rod, the inner heated ring of 8 rods, the outer heated ring of 12 rods, the 4 unheated corner rods, the shroud, the outer cooling jacket and the containment respectively. An example of the agreement obtained is shown in Fig. 2, for the steam pre-oxidation phase of PWR test QUENCH-10[10].

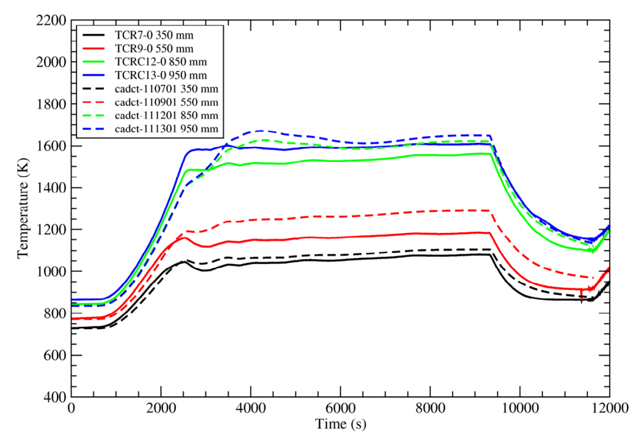

Fig. 2 : SCDAPSIM calculation of QUENCH-10 centre rod temperature evolution

A feature of the SCDAP codes is a mechanistic model for control rod failure, based on a kinetic treatment of the eutectic interaction between the stainless steel cladding and Zircaloy control rod guide tube, and on the Fe-Zr phase diagram. As part of preliminary studies for QUENCH-13, the implementation of this model was checked and the kinetic part found to be faulty, giving predicted failure at too low a temperature. The SCDAP codes used here were corrected using data from the original report[11] on which the model was based.

In the present work, particular attention was paid to the model of the offgas pipe, for which 30 nodes were used. Reliable simulations of the pipe wall and gas temperatures were needed to assist in the planning of the installation of the on-line aerosol measuring equipment. The model of this part of the facility was therefore improved taking better account of thermal capacities and heat transfer in this region of the facility, and benchmarked against results from QUENCH-12[12] for both bundle and offgas line, since this experiment had a similar offgas pipe configuration. This test also involved a power ramp and hold, this time to give a maximum bundle temperature of about 1473 K in the hold period, followed by a further ramp and reflood. Typical results for the offgas line are shown in Fig. 3. The model could therefore be used to predict temperature and fluid conditions where instrumentation was lacking.

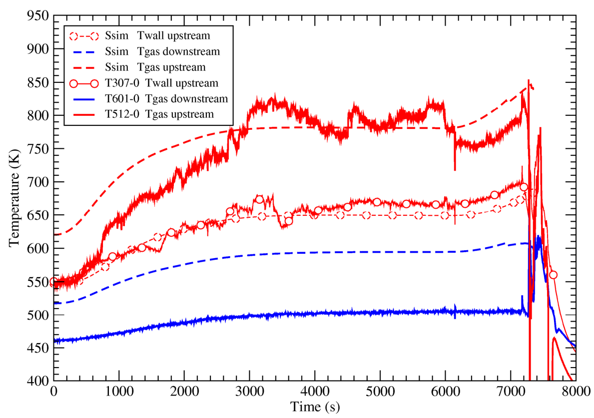

Fig. 3 : Comparison of QUENCH-12 offgas pipe temperatures with SCDAPSIM calculation

First, attention was paid to the pre-conditioning temperature. Initial calculations aimed at plateau temperatures of 1250 K and 1350 K, seeing that SCDAP predicted control rod failure temperatures of about 1430-1450 K for typical QUENCH test conditions. It was then decided by the calculational partnership to adopt a very conservative approach, avoiding any possibility of rod failure before the final ramp (which would prevent useful aerosol data from being obtained) by keeping the plateau temperature down to 1250 K, i.e. at the minimum stainless steel liquefaction temperature as defined in the SCDAP stainless steel / Zircaloy dissolution model on the basis of the Fe/Zr phase diagram, and all subsequent calculations were performed with this condition.

The next step was to define the power ramp rate that would lead to a long enough period for aerosol measurements from control rod failure through to the proposed quench temperature range of 1800-1900 K (precise temperature to be defined on the basis of sensitivity studies). The rates chosen were 0.05, 0.1 and 0.2 W/s/rod. The effect of these ramp rates on the control rod temperature at the axial position of maximum temperature, 950mm, is shown in Fig. 4.

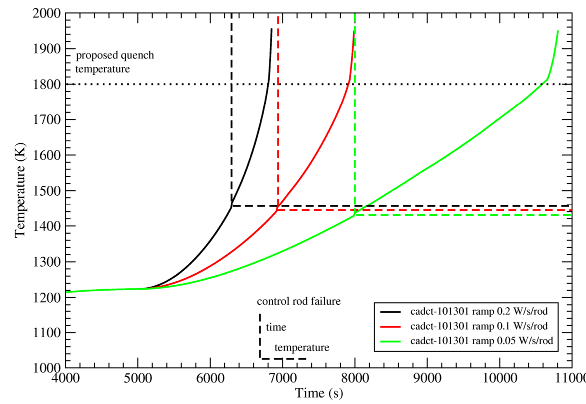

Fig. 4: SCDAPSIM calculation of effect of power ramp rate on control rod temperature at 950mm

On the basis of these results, it was decided that a rate of 0.075 W/s/rod was sufficient to give a long enough time window for the Ag and In aerosol release to be measured, in the range 1000-2000 s.

The step after was to define the conditions for test termination, to avoid the possibility of an oxidation excursion during the quench phase. Based on experience with QUENCH-06 and other tests, a temperature of 1873 K was chosen as the starting point for the power reduction and injection of reflood water (initially at a high rate to fill the lower plenum, then at 50 g/s). Sensitivity studies were performed on possible delays on reflood injection of up to about 350 s, and on uncertainties in the Zircaloy/steam oxidation rate at high temperatures. For the latter item, the evaluation of Schanz[13] was taken as the basis; this takes into account the well-known correlations of Urbanic-Heidrick (U/H) and Prater-Courtright (P/C), as well as proposing a new treatment that considers the range of data now available. Typical results are shown in Fig. 5. It is seen that there is a possibility of an excursion if there is a delay in reflood, with conservative assumptions regarding the oxidation kinetics (use of the correlations predicting the highest reaction rate). Therefore, it was decided to reduce the temperature on initiation of test termination to that of 1813 K on the control rod at 950 mm, the position of maximum temperature.

The *final conditions for the test* were thus agreed: electrical power of about 9 kW to stabilise the bundle at 1250 K for 5000 s preconditioning, power ramp at 0.075 W/s/rod to achieve a heat-up rate of about 0.25 K/s up to a maximum control rod temperature of 1813 K during which period control rod rupture would be expected to occur, then terminate the test by turning off the electrical power and reflooding the heated section with room temperature water at 50 g/s. The signature variables are illustrated in Fig. 6. The power in the heater rod is less than the gross power owing to the presence of external resistance in the power supply lines.

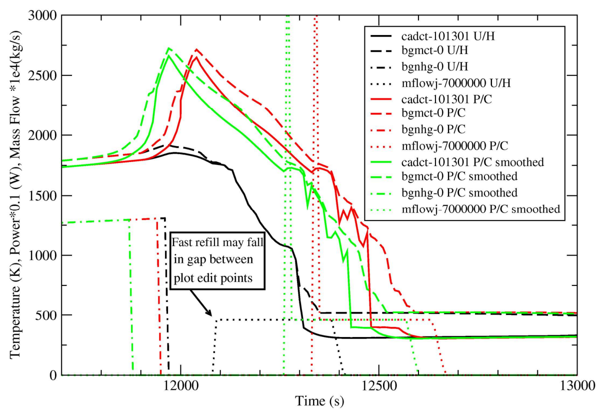

Fig. 5 : SCDAP/RELAP5 calculation of power trip at 1873K, delayed injection effect of oxidation kinetics

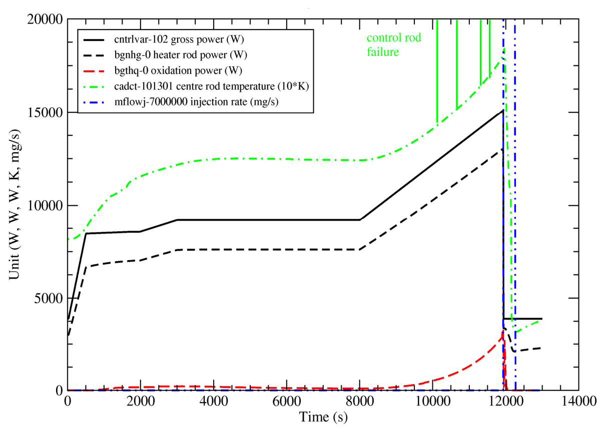

Fig. 6 : QUENCH-13 transient signature variables as calculated by SCDAP/RELAP5

The calculated progression of the quench front is shown in Fig. 7, while the fluid and wall temperatures in the offgas line are shown in Fig. 8, taken into account in the aerosol measurements.

Sensitivity studies with a revised version of the Cathcart-Pawel/Prater-Courtright-Schanz oxidation kinetics (transition in the P/C/S correlation ramped between 1800 and 1900 K, which increases the kinetics, here denoted 'P/C smoothed'), fast injection delayed by 5% and reduced to 90% nominal flow, power trip and reflood at 1863 K, and finally all of the above (bounding case), showed sufficient margin to ensure a smooth cooldown in the quench phase. These studies allow for realistic uncertainties in the termination procedures (the true maximum temperature might be not be at an instrumented position, for example).

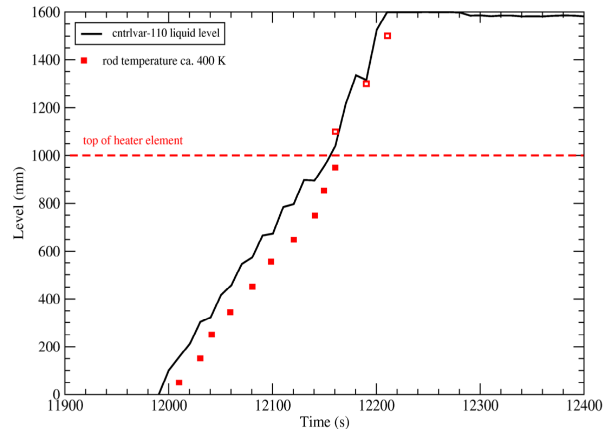

Fig. 7 : SCDAP/RELAP5 calculated bundle reflood quench progression

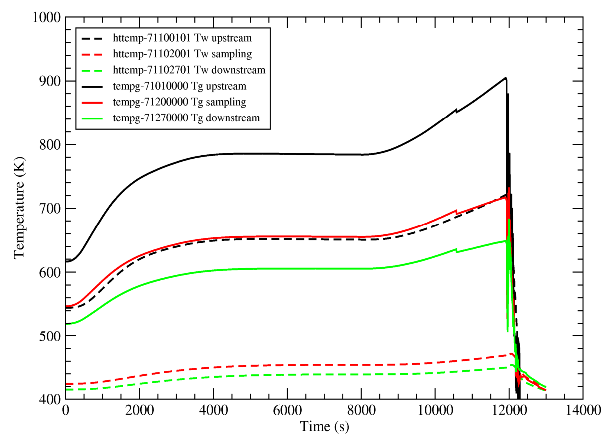

Fig. 8 : SCDAP/RELAP5 calculated wall and fluid temperatures in the offgas pipe

Finally, contingency calculations were performed on the possibility of control rod failure being later than expected. The planning calculations predicted failure between 1400 K and 1450 K, while companion separate-effects tests at FZ Karlsruhe[14] showed failure between 1500 K and 1550 K. To provide a good time window (more that 1000 s) for aerosol data collection, failure is wanted below 1600 K. A change of power ramp was considered if there was no clear indication of failure by 1600 K. SCDAPSIM cases examined the effect of holding the power constant for 1000 s at this time, or of then reducing it to 0.025 W/s/rod, as shown in Fig. 9. The former was more effective in achieving a good time window, and was therefore adopted as the contingency position.

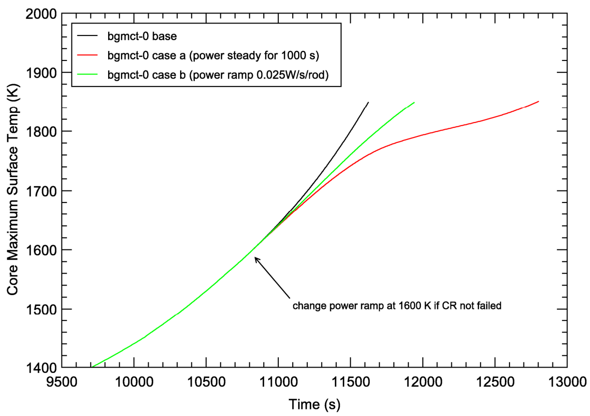

Fig. 9 : SCDAPSIM calculation of maximum bundle temperature for contingency cases on late control rod failure

III.C. GRS calculations

The GRS calculations for the specification of test QUENCH-13 were performed with the system code ATHLET-CD[15]. This code applies the detailed models of the thermal hydraulic code ATHLET in an efficient coupling with dedicated models for core degradation and fission product behaviour. It is being developed by GRS in cooperation with the Institut für Kernenergetik und Energiesysteme (IKE), University of Stuttgart.

The input data set was mainly based on the standard ones used for calculation of previous QUENCH experiments for code assessment[16]. It includes the bundle fluid channel, subdivided into 20 axial nodes (10 nodes within the heated length) and connected via cross-flow junctions with a bypass channel to allow flow deviation in the case of blockage formation due to melt accumulation. The rod bundle is simulated in the code module ECORE by two concentric rings, an inner ring containing the absorber rod and 8 heated rods, and an outer ring composed by 12 heated rods. In addition, the five grids, the shroud with its thermal insulation and the outer cooling jacket with the counter-current flows of argon (heated region) and water (upper region) have been simulated by standard fluid-dynamic objects and heat conduction objects.

This basic input data set for the QUENCH facility has been extended with modelling of the offgas pipe cooling, benchmarked against data from the previous QUENCH tests 08, 10 and 11, as well as with calculation of the AIC release. The AIC release model in ATHLET-CD is based on rate equations taking into account the partial pressure of the evaporating gases. It has been assessed against data from the Phebus FP experiments[17].

The first step was to evaluate the influence of the power ramp rate on the time period for aerosol measurements from control rod failure until the proposed quench temperature of 1800 K is reached. Ramp rates from 0.05 to 0.2 W/s/rod have been used. The calculated effect of these ramp rates on control rod temperatures at the axial position of maximum temperature is shown in Fig. 10.

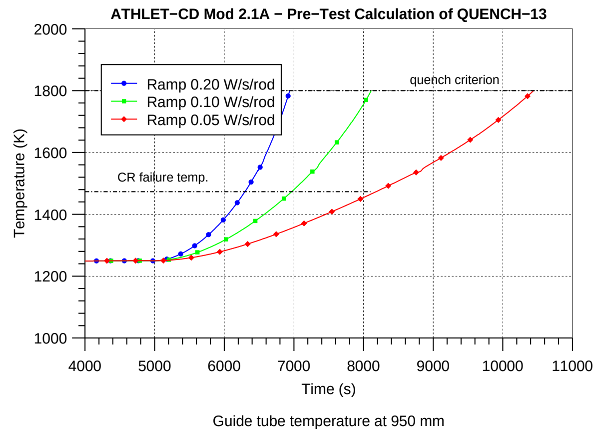

Fig. 10 : ATHLET-CD calculation of effect of power ramp rates on control rod temperature at 950 mm

The ATHLET-CD calculations confirmed the corresponding results obtained by PSI with SCDAPSIM, presented in the previous section. A power ramp rate of 0.075 W/s/rod would be sufficient to give the envisaged time window for AIC release in the range of 20 to 30 min.

These calculations were complemented by a series of sensitivity studies concerning the plateau temperatures during the pre-conditioning phase (1250 K - 1350 K), the temperature criterion for quench initiation (1800 K - 1900 K) as well as the influence of the assumed temperature for control rod failure for a power ramp of 0.075 W/s/rod. Calculations with plateaus of 1250 K and 1350 K show very similar results with respect to the time window for measurements as well as to the integral AIC releases, the lower temperature been finally chosen to preclude any possibility of rod failure before the final power ramp.

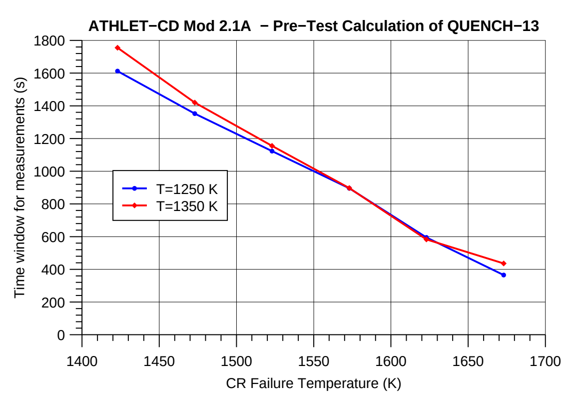

Fig. 11 : Influence of control rod failure temperature on time window for measurements

The influence of the control rod failure temperature on the time window for measurements is illustrated in Fig. 11. An adequate time window for data collection is provided for temperatures below 1550 K, the upper bound of the failure temperatures observed in the separate-effect tests performed at FZ Karlsruhe[14].

The ATHLET-CD calculations indicated a rather small release of Ag for the proposed temperature criterion for quench initiation (Fig. 12). In order to increase AIC release but at the same time to avoid the possibility of an oxidation excursion during the quench phase, it has been suggested to change the conditions for test termination, with a power reduction when the maximum bundle temperature (elevation 950 mm) reaches 1873 K and reflood initiation only after a temperature decrease of 100 K at this elevation.

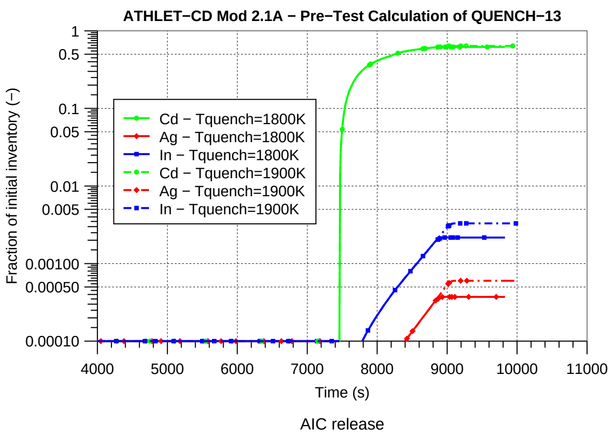

Fig. 12 : Calculated AIC release for different quench initiation criteria

With this proposed scenario additional calculations have been performed, using different correlations for the calculation of Zircaloy oxidation rates at high temperatures (Urbanic-Heidrick and Prater-Courtright) and taking into account possible delays on reflood injection up to 240 s. The results indicated that the possibility of an oxidation escalation, mainly at the shroud inner surface, cannot be excluded if the reflood initiation is delayed. Therefore it was agreed to reduce the temperature criterion for both power reduction and reflood initiation to 1813 K.

The work was concluded with a calculation on the basis of the finally agreed test conditions, as presented in the previous section. The results are very similar to those of PSI with SCDAP/RELAP5 (Fig. 6 to Fig. 8), increasing confidence that the test objectives would be met.

III.D. EdF calculations

The MAAP4 code is the reference tool for modelling reactor severe accidents at EDF. Due to its modular structure, it can be used to analyse tests such as the QUENCH experiments.

A QUENCH bundle model, already validated for previous benchmarks, was modified according to QUENCH-13 specifications. Specifically, the parameter file previously used for analysing QUENCH-07 (with a B_4C control rod in the centre of the bundle) was modified by replacing the B_4C control rod by a silver-indium-cadmium control rod.

The initial and boundary conditions files were adapted from the corresponding files for the similar previous experiment QUENCH-06, to simulate the anticipated QUENCH-13 chemical and thermodynamic conditions (steam and argon flows, water flow at reflooding, temperatures, etc.). Two options were considered for the plateau preconditioning temperature: 1250 or 1350 K. The power programme was fitted to obtain the desired temperature evolution, Fig. 13.

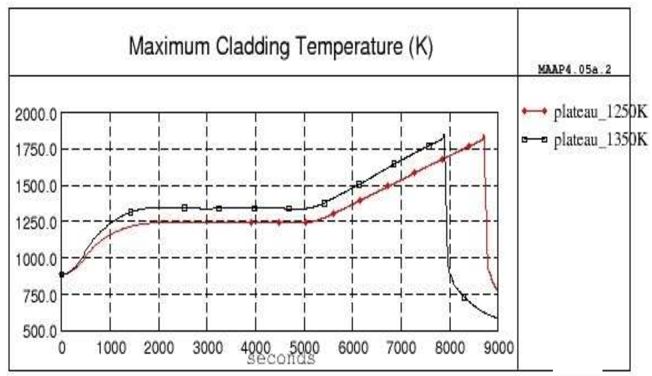

Fig. 13 : MAAP calculated temperature evolution

So, parameters of interest could be calculated, such as:

- hydrogen production, Fig. 14;
- silver, indium and cadmium released from the control rod, Fig. 15 as mass and Fig. 16 as percentage of the initial SIC rod mass.

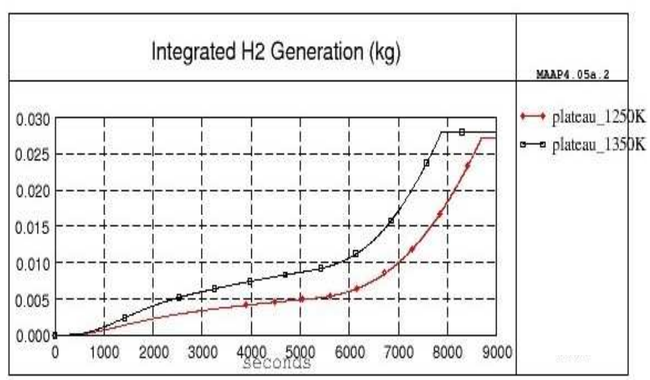

Fig. 14 : MAAP calculation of hydrogen production

It was observed that the two different preconditioning plateaus do not change the predicted SIC releases. The modelled releases correspond to the volatility of the different species, with cadmium almost completely released, indium about a half and silver release of a few percent. Thus these results are consistent with the results from previous experiments involving SIC control rods, such as Phebus FPT1[18].

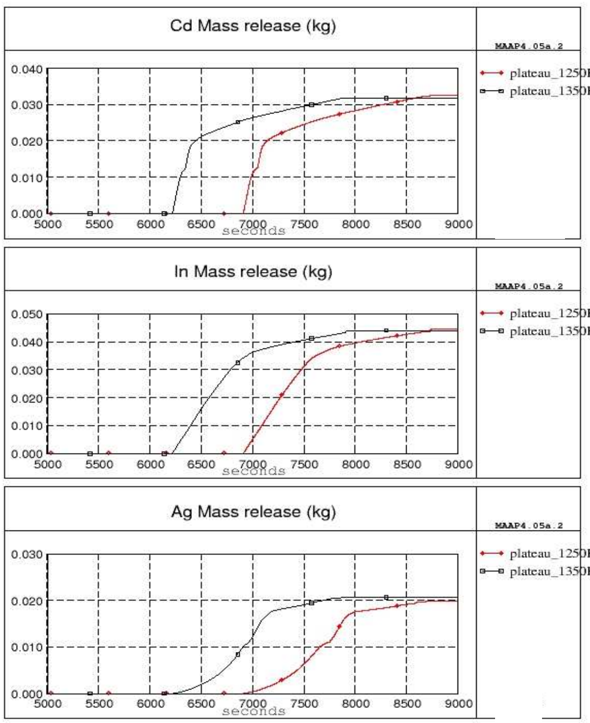

Fig. 15 : MAAP calculated SIC mass released

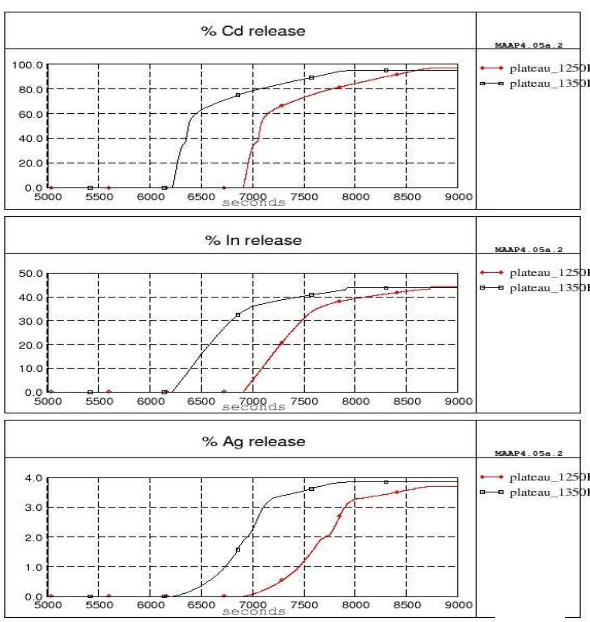

Fig. 16 : MAAP calculated percentage of initial SIC mass released

IV. DISCUSSION

The final test protocol agreed between the calculational and experimental teams is illustrated in Fig. 17. It was judged to be the best to achieve the joint experimental objectives of studying the effect of a PWR control rod on bundle heat-up and quench behaviour in mainly rod-like geometry, and of measuring the release in aerosol form of Ag, In and Cd following the control rod rupture, while allowing for uncertainties in test conduct, experimental conditions and predictions of Zircaloy/steam oxidation.

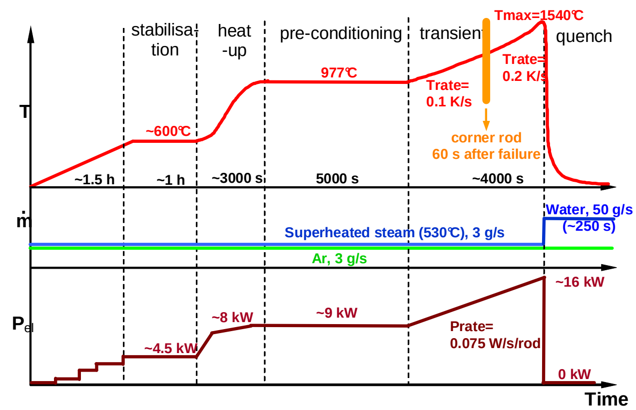

Fig. 17 : Final test protocol for QUENCH-13

The experiment was successfully performed on 7 November 2007, taking full account of the recommended protocol[19]. Post-test analysis is planned cooperatively amongst the partners after release of the definitive results.

V. CONCLUSIONS

The paper has demonstrated how the use of advanced severe accident analysis codes has enabled the definition and safe conduct of the QUENCH-13 experiment, taking into account the inevitable uncertainties in the test conduct and outcome. It provided a good test of severe accident code capabilities and so lent confidence regarding the use of these codes in reactor applications. The pre-test computational support is being followed by a programme of post-test interpretative analysis that will provide further insights into code capabilities, needs for model improvement, and resolution of remaining safety issues.

The strategy based on independent analyses, use of different codes and comparison with data from earlier tests minimised the potential impact of model limitations in individual codes, and provided additional confidence for defining test conditions. The effectiveness of analytical support depends critically on discussion amongst the users and with the experimental team, facilitated by the networking arrangements in place in SARNET.

Several of the questions to be answered in the present context are analogous to those arising in reactor analyses, for example in the definition of criteria for water injection as an accident management measure. The experience of performing analytical support for experiments of this kind provides a spin-off benefit to reactor application, helping to make the most effective use of the available tools in addressing plant safety issues.

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NOMENCLATURE

EdF	Electricité de France
EU	European Union
FZK	Forschungszentrum Karlsruhe
GRS	Gesellschaft für Anlagen und Reaktorsicherheit
IKE	Institut für Kernenergetik und Energiesysteme
P/C	Prater/Courtright
PSI	Paul Scherrer Institute
PWR	Pressurised Water Reactor
SET	Separate-Effects Tests
SIC	Silver/Indium/Cadmium
U/H	Urbanic/Heidrick
VVER	Vodo-Vodyanoi Energetichesky Reactor (PWR of Russian type)

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