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► **To cite this version:**

Iain Shepherd, Tim Haste, Naouma Kourti, Francesco Oriolo, Mario Leonardi, et al.. Investigation of core degradation (COBE). Nuclear Engineering and Design, 2001, 209 (1-3), 10.1016/S0029-5493(01)00393-4 . hal-01282019

HAL Id: hal-01282019

<https://hal.science/hal-01282019v1>

Submitted on 1 May 2024

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Investigation of core degradation (COBE)

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The COBE project started in February 1996 and finished at the end of January 1999. The main objective was to improve understanding of core degradation behaviour during severe accidents through the development of computer codes, the carrying out of experiments and the assessment of the computer codes' ability to reproduce experimental behaviour. A major effort was devoted to quenching behaviour and a substantial achievement of the project was the design and commissioning of a new facility for the simulation of quenching of intact fuel rods. Two tests, carefully scaled to represent realistic reactor conditions, were carried out in this facility and the hydrogen generated during the quenching process was measured using two independent measuring systems. The codes were able to reproduce the results in the first test, where little hydrogen was generated but not the second test, where the extra steam produced during quenching caused an invigorated Zircaloy oxidation and a substantial hydrogen generation. A number of smaller parametric experiments allowed detailed models to be developed for the absorption of hydrogen and the cracking of cladding during quenching. COBE also investigated other areas concerned with late-phase phenomena.

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There was no experimental activity – the work included code development and the analysis of experimental data available to the project partners – either from open literature or from other projects such as Phebus-FP. Substantial improvement was made in the codes’ ability to simulate heat transfer in debris beds and molten pools and increased understanding was reached of control rod material interactions, the swelling of irradiated fuel and the movement of molten material to the lower head. © 2001 Elsevier Science B.V. All rights reserved.

1. Introduction

The only way that the efficacy of mitigation measures for severe accidents can be assessed is through the use of simulation models. It is thus essential firstly that these models can treat the entire range of severe accidents and mitigation strategies and secondly that the results of these models are credible. The main objectives of this shared-cost project, which started in February 1996 and lasted for 3 years, were to:

- develop the capability of calculating the behaviour within a reactor vessel during a severe accident up to vessel melt-through so that the efficacy of accident management strategies can be assessed;
- understand how accurate the results of such calculations are.

2. Work programme

The project was divided into four main work packages:

- Development of modelling codes so that they can better represent reactor sequences.
- Execution of experiments that can be used for calibration and validation of such codes.
- Validation that these codes are able to calculate experiments. Some of the experiments analysed were carried out as part of this project. Others are the heritage of past programmes.
- Scaling studies to make sure that the experiments carried out in this project represent as well as possible phenomena expected to be encountered in reactors.

There were various problem areas tackled in the project but two major themes were:

- Quenching. A major part of the project was concerned with the design, construction and

commissioning of the QUENCH experimental facility and the carrying out of experiments in it that simulated the behaviour of an overheated core during reflood. This was accompanied by a considerable effort for planning and analysing the tests.

- The analysis of debris bed and molten pool behaviour and the subsequent movement of material towards the lower head. This part of the project did not directly involve experimental activity but experiments carried out as part of other projects were analysed.

3. Main achievements

3.1. Code development

The code development work considered quenching and late-phase phenomena.

3.1.1. Quenching

Five codes were used in the analysis of quenching. These were:

- SCDAP/RELAP5. No developments of the code were made for this project but it was used by many project partners, some of whom had to make private modifications to handle some features of the QUENCH facility. Large mass errors were encountered, when performing quench calculations for the bundle tests. It was only possible to circumvent this problem by artificially introducing a low argon flow (0.5 g s^{-1}) with the water. The problem was reported to the USNRC.
- KESS. The main aim of the KESS calculations was to validate KESS’s embrittlement model. In KESS, the cracking of ZrO_2 -layers and U–Zr–O crusts is estimated by analysing thermal stresses in the layers. If the effective stress exceeds a given limit, crack formation is as-

sumed. The resulting increase of the oxygen diffusion in the cladding is simulated by a parameter in the oxidation models. This leads simultaneously to an enhanced heat release. Enhanced oxidation occurs in the region of pure vapour flow as well as under film boiling conditions.

- ICARE2. Since the version of ICARE2 used in the analysis had no model to describe the quench phase – in fact it was not possible to model liquid water at all – the ICARE2 calculations were stopped just before the onset of the quench phase.
- ATHLET-CD is being developed alongside KESS. The ATHLET part describes coolant thermal hydraulics and the thermal response of intact core structures, while the core degradation (CD) part, based on selected KESS models, predicts damage progression up to the formation, relocation and solidification of ceramic melts.
- SVECHA/QUENCH (S/Q) was developed during this project for the detailed modelling of reflooding phenomena. So far, it can only handle single rod geometries. The code treats the main physical phenomena occurring during quenching of fuel rods:
 - cladding oxidation,
 - cladding mechanical deformation,
 - hydrogen uptake and release,
 - heat conduction inside the fuel rod,
 - heat and mass exchange in the surrounding two-phase media.

3.1.2. Other late-phase phenomena

The interest in developing better models of late phase phenomena was stimulated by the generally poor calculations of the late phases of the Phebus-FP experiments. Efforts have concentrated on:

- material movement to the lower head – using two specialised codes MECI and MARCUS;
- the modelling of debris bed and molten pool formation – mostly using the ICARE2 code;
- control rod effects – using KESS;
- degradation of irradiated fuel – using LAKU and MFPR.

The relocation of core melt arriving at the peripheral or lower boundaries of the core and

transforming into core debris resting on the lower head is presently considered in a parametrically by system codes, with a user-specified extent of fragmentation and cooldown. Neither retention nor coolability capabilities of core support assemblies nor thermal hydraulic aspects of core melt draining through these structures are considered in detail. Therefore, efforts to understand these phenomena better have concentrated on development of special purpose computational fluid dynamics tools. Application of the MECI and MARCUS codes to molten material flowing through a 2-D model of an upper core support assembly segment (see Fig. 1 for MARCUS calculation) has indicated that further efforts must be spent on solving numerical instability problems arising from high-speed melt flows with moving interfaces and their interaction with walls and obstacles. The physical and chemical interactions of molten material with intact structures were not considered, but the results are a good basis for future code developments.

Considerable progress has been made in ICARE2 on heat and mass transfer in debris beds and the subsequent transformation into molten pools. The 2-D approach now includes models for core debris oxidation and dissolution, debris bed collapse on formation of extended voids, radiative heat transfer through large cavities between the top of a debris bed and the upper plenum walls, melt agglomeration, pool formation and growth as well as crust formation on contact of cold surfaces. The checking of these models against experiments is described under ‘analysing other experiments’.

A PWR control rod model has been developed, providing information about the radial and axial distribution of absorber melt after control rod cladding failure. Based on the internal pressure from the filler gas and the cadmium vapour, absorber melt is assumed to be blown out through the cladding failure orifice according to a steady-state Bernoulli equation and the continuity equation.

Different approaches for modelling the degradation behaviour of irradiated fuel have been tested. These indicated that it is feasible to model the competing processes of release and swelling

but faster-running models will need to be introduced if these processes are to be implemented into system codes. The influence of fuel swelling on the degradation behaviour has not been investigated yet.

3.2. Scaling studies

3.2.1. Quenching

A literature review and a number of calculations were undertaken in order to determine the boundary conditions to be used in the experiments that would best represent real accident conditions.

Flooding rates were found to lie in the range between 0.025 and 0.4 m s^{-1} , depending on the break size which in turn determines the system pressure and on principal ECCS availability (accumulators, high- or low- pressure injection). The oxide thicknesses (10 – $250 \text{ }\mu\text{m}$) and average fuel temperatures are mainly determined by the timing of reflood after the core uncover.

The results of a parameter study using SCDAP/RELAP5 on a station blackout sequence for the EPR reactor show that a representative axial temperature increase before quenching is about 700 K

m^{-1} in the upper core, but it can be much higher later. When reflood is initiated, flooding velocities (velocity of the collapsed water level) vary between about 10 – 20 cm s^{-1} in the lower core and 0.2 – 1 cm s^{-1} in the upper part, because of the steep axial temperature gradient. The wide temperature range leads to a wide range of oxide layer thickness of about 50 – $300 \text{ }\mu\text{m}$. The pressure in the primary system of the reactor is about 10 bar . Some repressurisation is possible during reflood — a peak value of 30 – 40 bar was estimated.

For a Westinghouse four-loop design four accident scenarios were calculated using SCDAP/RELAP5 — a station blackout (TMLB') where no operator action is taken; a small break ($2''$ or 5 cm) LOCA where the HHSI pumps are initially unavailable but two are recovered later (at various times); a small break ($3''$ or 7.5 cm) LOCA where the HHSI pumps are initially unavailable but two are recovered later and a small break ($2''$ or 5 cm) LOCA where the HHSI pumps are initially unavailable but two are recovered later after the operator depressurises the vessel and the accumulator water boils down. In the TMLB' case, depressurisation occurs as a result of surge

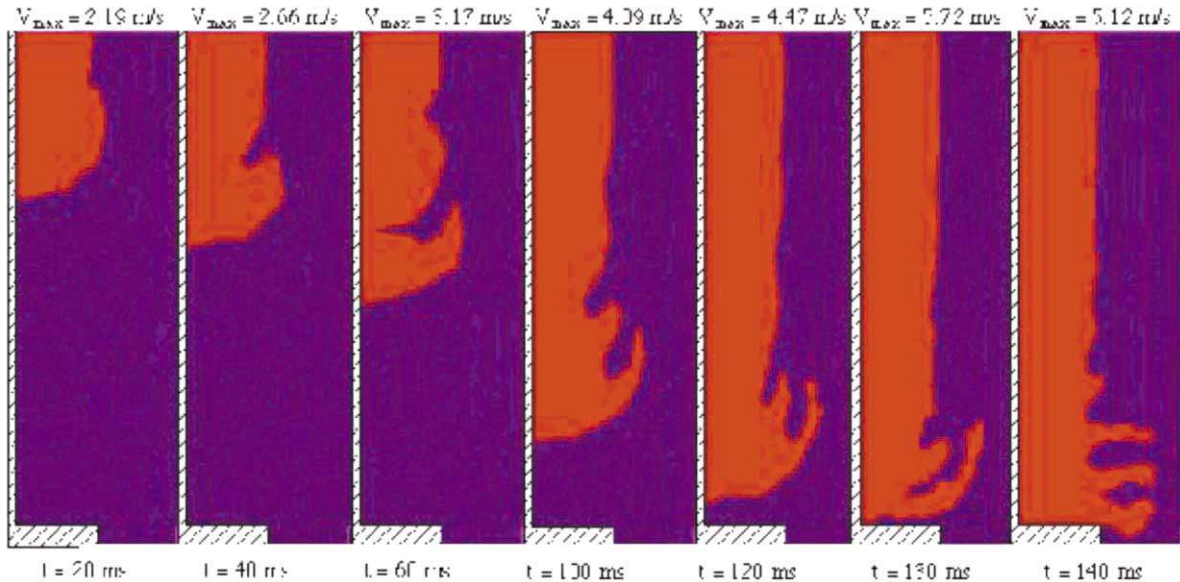


Fig. 1. MARCUS simulation of the flow of molten material to the lower head.

line failure, allowing accumulator water to be discharged and refill the core. In all cases, the average quench rate is 0.0065–0.023 m s⁻¹. The expected oxide thicknesses were consistent with previous analyses. The pressure at onset of quench was in the range 15–70 bar. Although some repressurisation occurred, this was insufficient to slow the quench rate significantly.

The main parameters for the bundle QUENCH tests: maximum temperature up to 2000 °C, maximum oxide thickness 300 µm, and quench rates in the range 0.005–0.010 m s⁻¹, were based on, and are consistent with, the plant calculations. The main parameter that it is not possible to match in the QUENCH facility is the system pressure, which is typically 2 bar in the facility but can be higher in real scenarios. Application of the results to the plant scenarios requires the use of validated thermalhydraulics models for the quenching process.

3.2.2. Air ingress

Air ingress has also been considered as part of the reactor studies carried out under the COBE project, though this is strictly more relevant to the sister OPSA project. The work considered scenarios for air ingress, the main ones being a case where vessel breach has occurred, or where the head is open during a shutdown. Calculations were performed for a specific scenario (a TMI-2 like scenario except with the pressure-operated relief valve (PORV) remaining open and vessel breach occurring). From this analysis, it was found that pressure variations in the vessel could cause air ingress through the PORV, but in this case vessel breach did not lead to a large chimney effect because of sealing of the pressuriser loop.

3.3. Design calculations

The rig used in the QUENCH experiments was designed, built and tested during the COBE project. A number of calculations were carried out to check the structural integrity of the design. The basic aim was to ensure firstly that under all possible operating conditions it would not overheat beyond the melting point of the materials used and secondly that vapour leaving the top of

the test-train would not condense into droplets and rain back onto the facility.

The design calculations, which examined a number of possibilities, were carried out using

- SCDAP/RELAP5/MOD3.1,
- the TAU finite element code,
- the computational fluid dynamics code CFX4,
- engineering calculations using the ‘effectiveness number of transfer units method’.

The final design functioned as planned – thus vindicating the design calculation work.

3.4. Pre-test calculations

For each QUENCH test carried out in the COBE, a series of SCDAP/RELAP5 pre-test calculations was performed to assist in the planning, and to ensure the desired conditions were achieved. These calculations have also given the experimental team useful information concerning the monitoring of the tests.

The base SCDAP/RELAP5 calculations were checked against both ICARE2 and KESS calculations. Once some care was taken to achieve the same boundary conditions with both codes the results were similar. Differences in oxide layer thicknesses were attributed to slight differences in temperature and confirmed the extreme sensitivity of the oxidation to the temperature.

3.5. Separate effect experiments

3.5.1. Single rod tests

A set of parametric tests was performed. Pre-oxidation of the Zircaloy tube specimen (0–300 µm ZrO₂ layer thickness) was at 1400 °C in an Ar/O₂ or an Ar/steam mixture. The specimen was then quenched from between 1000 and 1600 °C by water (30 or 90 °C) or by rapid steam cooling. The main results were:

- Formation of macroscopic through-wall cracks can be observed in steam-quenched experiments with zirconium dioxide layer thickness greater than 200 µm. Oxidation of crack surfaces increases with decrease of the specimen temperature at the onset of quenching.
- Contrary to the water-quenched experiments (pre-oxidised in an Ar/O₂ mixture) in the

steam-cooled experiments, it was not possible to determine the cooldown-related amount of hydrogen as a function of the crack surface oxidation.

- Deviation from the regular cooldown curve during the quench phase was observed in all experiments with quench temperatures less than 1600 °C, when the ZrO₂ layer thickness was larger than 200 µm. No temperature increase was observed quenching from 1600 °C even for large ZrO₂ thicknesses.
- In general, the results show that after reaching a certain oxide layer thickness (200 µm) and crack density, the amount of hydrogen absorbed in the metallic part of the Zircaloy tube increases considerably.
- In tests with 1600 °C temperature at onset of cooldown even for large oxide layer thickness (> 300 µm) the amount of absorbed hydrogen is still relatively low compared with similar specimens cooled from 1100, 1200 and 1400 °C. Only for very large oxide layer thickness (~ 380 µm) is the amount significant.
- The amount of absorbed hydrogen correlates not only with the extent of the pre-oxidation at the onset of cooldown, but also with the total crack length.
- The uptake of the hydrogen in Zircaloy and its additional release during oxidation of the metal and during the quench phase can influence the kinetics of hydrogen release and has therefore to be studied quantitatively in more detail.
- The possibility that the deviation from the regular cooldown curve could be triggered by exothermal hydrogen absorption needs to be investigated further.

3.5.2. Hydrogen absorption tests

A simultaneous thermal analysis (STA) apparatus coupled with a mass spectrometer was used to determine the hydrogen solubility in Zircaloy cladding tube segments. It allowed on-line recording of the interaction between Zircaloy and the hydrogen-containing atmosphere. A sophisticated gas supply system reproducibly provides any needed argon/hydrogen mixture. Conclusions were:

- Hydrogen solubility of Zircaloy-4 is well described by Sieverts' law up to temperatures of $T \approx 1500$ °C and up to hydrogen partial pressures of $p_{H_2} = 1$ bar.
- An intact oxide scale prevents hydrogen from being absorbed/released by/from the metal phase.
- The hydrogen solubility in Zircaloy-4 decreases with increasing oxygen content in the α - and β -phases.
- No differences in the hydrogen absorption behaviour between Zircaloy-4 and Zr1%Nb were found.
- During steam oxidation a limited amount of hydrogen is absorbed in situ.
- The different transient behaviour during absorption and desorption, which was experimentally observed, could be well described by a computer model.

3.6. The bundle *QUENCH* experiments

The most significant milestone in the project was the successful completion of a number of commissioning tests and two quenching tests in the bundle *QUENCH* facility. The bundle consists of 21 fuel rods approximately 2.5 m long. The experiments basically consisted of a heat-up phase, a pre-oxidation phase (in test *QUENCH-01* but not in *QUENCH-02*), a transient phase, and a quench phase. The main measurements were from thermocouples, two independent hydrogen measurement systems and post-test examination of the bundle. The main findings were:

- The oxide layer on the clad surface was enough to prevent a temperature escalation in *QUENCH-01*. In *QUENCH-02* with no additional pre-oxidation of the clad, the bundle experienced a temperature escalation with heating rates around 20 and 50 K s⁻¹ for the rods and shroud, respectively. The escalation started prior to flooding of the bundle.
- The maximum temperature during reflood in *QUENCH-01* was approximately 1870 K, that in *QUENCH-02* approximately 2500 K.
- The temperature escalation and enhanced steam generation due to flooding increase the hydrogen generation rate.

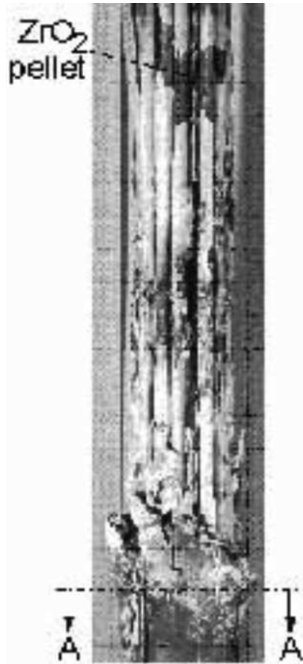


Fig. 2. Bundle after QUENCH-02.

- In QUENCH-01, where there was no temperature escalation, the bulk of hydrogen was produced in the pre-oxidation phase (30 g), while in QUENCH-02, where there was a temperature escalation, about 190 g of hydrogen was

generated during the whole test.

- The maximum hydrogen production rate was measured during QUENCH-02, i.e. 2.5 g s^{-1} as compared to 0.08 g s^{-1} in QUENCH-01 (Fig. 2).
- The ‘onset of cooling’ of the cladding (before rewetting) occurs simultaneously for all axial positions.
- Embrittlement of the rod cladding and the shroud is more severe for the QUENCH-02 test bundle than for the QUENCH-01 test bundle.
- Parts of the QUENCH-02 test bundle were molten and had formed localised blockage zones in the coolant channels (Fig. 3) whereas, the QUENCH-01 test bundle shows no sign of melting.

3.7. Analysing the QUENCH experiments

As indicated in Table 1, a number of codes were used for the analysis of the QUENCH tests. The main results of the analysis are explained in this section.

3.7.1. Single rod tests

For the single rod tests S/Q managed to reproduce the experimentally observed temperature curves under the full range of conditions using one unique set of input parameters. KESS can

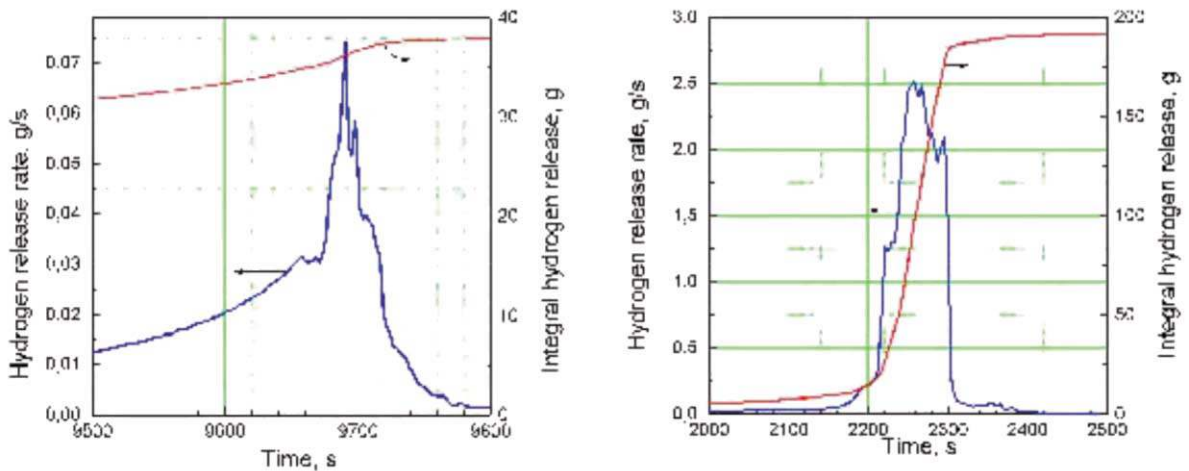


Fig. 3. Hydrogen generation during quenching in QUENCH-01 and -02.

Table 1
Codes used for analysis of QUENCH tests

Code	Single rod tests	Commissioning tests	QUENCH-01	QUENCH-02
S/Q	IBRAE			
KESS	IKE		IKE	
SCDAP/RELAP5		FZKA	AEA, FZKA, ENEA	AEA, FZKA, ENEA
ATHLET-CD			RUB	
ICARE2			ENEA/IPSN	

reproduce the temperatures but not the hydrogen production. Adjusting the cracking model to increase the hydrogen increased the temperatures as well due to the assumption of instantaneous crack formation, so time development of the cracking must be considered in further model development.

3.7.2. Bundle tests

- During the heat-up and pre-oxidation phase, there were no significant differences between the codes' behaviour. All codes managed to reproduce the behaviour well.
- ICARE2 could not calculate the quenching phase because it only models single phase flow — it had no model for liquid water.
- KESS, SCDAP/RELAP5 and ATHLET-CD gave reasonable results for QUENCH-01, where there was no escalation. Some work needs to be done to produce a more accurate model of the facility.
- SCDAP/RELAP5 (which was the only code used) underestimated the hydrogen produced in QUENCH-02 — even when the global shattering model was used.

3.8. Analysing other experiments

A number of calculations were run so as to check the ability of the codes to simulate phenomena likely to be found in a severe reactor accident. The calculations are summarised in Table 2.

3.8.1. Phebus calculations

The KESS code was used to evaluate the influence of AIC control rod on the overall CD using FPT-0 results. The damage due to AIC–Zr–Fe

melts on the cladding of surrounding rods was found to be limited ($\sim 6\%$ of the cladding was dissolved). Nevertheless, it is not possible to propose a final conclusion on the impact of control rods on CD due to limitations of current models, in particular regarding the dissolution of UO_2 by AIC–Zr–Fe melts, and due to too limited quantified data on the fuel rod degradation by these AIC melts, in particular in reducing conditions.

ICARE2 calculations of FPT-0 and FPT-1 using a similar methodology to prepare the input deck (same models, meshing, material properties ...) showed that a 'unique data deck approach' applied on different tests carried out in similar conditions could not give the same level of quality of calculation. The main reason is, that in spite of similarities, there are always experimental differences that can impact the results more than expected such as the burnup (fresh fuel in FPT-0 and 23 GWd/tU in FPT-1), the power ramp or some uncertainties on boundary conditions (nuclear power). Finally the input decks changes between FPT-0 and FPT-1 were justified by differences in initial or boundary conditions.

In some conditions, it is not possible to use the same models in calculating different experiments. For instance, when some models give good predictions of a set of experiments (Phebus-SFD) but must be changed to better fit an other set of experiments (Phebus FP tests), this could indicate that they are too simple to have a large range of application. This is specially true for the clad failure criteria for which a temperature limit of 2500/2600 K is recommended in Phebus FPT-0 and FPT-1 oxidising conditions instead of 2300/2400 K in Phebus-SFD reducing conditions.

More important is the identification in FPT-0 and FPT-1 of an unexpected early fuel relocation when the rod temperature is ~ 2550 K in FPT-0 and ~ 2450 K in FPT-1. This relocation which is probably caused by a partial liquefaction of fuel rods cannot be calculated by standard models of ICARE2. Indeed, there is no model to predict an ‘early loss of rod geometry’, when the rod temperature is below the UO_2 liquefaction range. The SCDAP/RELAP5/MOD3.1 degradation criterion considers a U–O–Zr mixture and can simulate the molten pool formation in FPT-1 but assumes that some of the cladding had remained unoxidised and this may not have been the case. The further behaviour of the molten pool could not be simulated by SCDAP/RELAP5 because of the inadequate model for heat transfer to surrounding structures. The two Phebus experiments showed an influence of burn-up. The comparative ICARE2 analysis of FPT-0 (fresh fuel) and FPT-1 (irradiated fuel) confirmed that burn-up facilitates fuel rod liquefaction. There is no model to take into account the clearly observed effect of burn-up on degradation.

3.8.2. Debris bed calculations (DC1, MP1 and FPT-4 tests)

ICARE2 V3mod0 calculations of ACRR-DC1 and MP1 tests enabled improvement of heat transfer model in porous debris beds composed of either UO_2 or PuO_2 and ZrO_2 particles with a diameter range of 0.09–5 mm. On the basis of these calculations, the Imura–Yagi correlation was selected as the most suitable to predict thermal behaviour in particulate debris beds. The improved ICARE2 V3mod0 code was used for pre-test analysis of Phebus FPT-4 which has a debris bed of UO_2 and ZrO_2 particles similar to MP1.

Table 2
Code validation (other than quench tests) in COBE project

Code	Phebus-FP	MP1, DC1	TMI-2	XR2-1
KESS	TUD		IKE	
SCDAP/RELAP5	ENEA		ENEA	
ICARE2	JRC, ENEA	ENEA	ENEA, IPSN	UPM ^a

^a Using some private modifications to ICARE2.

Although these good model-measurement comparisons are encouraging, extrapolation to a reactor case is hard. In the ACRR tests, the volume is small, melt progression is rapid and limited by ‘cold boundary conditions’. In these tests, it is difficult to assess the modelling of each phenomenon such as chemical interactions, capillary forces, solid–liquid heat transfer, crust formation and debris oxidation.

3.8.3. TMI-2

This accident is a reference for code developers to check the ability of new versions of codes to predict the progression of the CD up to the final corium-vessel interaction.

ICARE/CATHARE was used to calculate the accident up till the start of pump 2B, i.e., before the initiation of quenching. Underestimation of the core dry-out zone led to underestimate of the hydrogen production. This calculation was repeated with SCDAP/RELAP5 using conditions similar to ICARE/CATHARE and similar results were obtained. Raising the clad failure limit (2500 K instead of 2300 K) and making adjustments to the make-up flow rate enabled significant improvements of the hydrogen production (from ≈ 180 to ≈ 300 kg as expected in TMI-2) and on the CD – the water liquid level was significantly improved and more core liquefaction was obtained.

KESS calculations focused on the hydrogen production during the running of pump 2B – i.e. including quenching, indicating that most of the metallic Zr mass available for oxidation during quench is located above the bottom edge of this crust (relocated Zr-rich mixtures) and in mixtures in a 1 metre zone above this crust. Even with the quench embrittlement model activated, the hydrogen production during the quench phase was un-

derestimated showing the need for further model development.

4. Interactions with other projects

The subjects treated and the software tools used were similar to those in the OPSA and CIT projects. Many of the project partners were involved in all three projects. The project meetings were held together and many partners attended all three meetings – even those in which they were not directly involved.

5. Conclusions and benefits

The COBE project has always attracted a great deal of interest because it concentrates on areas such as hydrogen generation during quenching that interest those concerned with accident management.

And while, as we have seen, the development of reactor safety software in Europe remains under-critical and fragmented, the COBE project has contributed to an integrated European experimen-

tal effort. Tests in facilities such as Phebus, QUENCH or CODEX are now completely complementary. The QUENCH facility was designed and commissioned on the basis of calculations performed by a private company, two universities and a national research organisation. All the partners of COBE were given the opportunity of influencing test parameters. Both pre- and post-test calculations were cross-checked by different organisations using different computer codes. The lessons learnt from the QUENCH calculations, described in Section 3.7, were shared by all participants who were able to go on and calibrate their computer codes for better estimation of hydrogen production during quenching in reactors.

Some emphasis was given within the project to modelling late phase behaviour. Prompted by the forthcoming Phebus FPT-4 test better code models for the behaviour of molten pools were developed. Increased understanding of other processes was also reached including the influence of control rods, the degradation of BWRs, relocation to the lower head and the swelling of irradiated fuel. Because this understanding has not yet been fed into system code models, the impact on accident management procedures is hard to quantify.