DEGRADED CORE QUENCH: A STATUS REPORT

August 1996

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Paris

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SUMMARY
An important accident management measure for controlling severe accident transients in light water reactors (LWRs) is the injection of water to cool the degrading core. This report for the CSNI summarises the status of experimental and theoretical knowledge of fuel behaviour aspects concerning the quench of degraded cores, where the fuel rods are in a mainly rod-like geometry at the time of quench; consideration of quench of late phase core configurations such as debris beds is outside its scope. Initially the review addresses thermal hydraulic aspects of reflood and quenching in non-degraded reactor cores, this is relevant because the models developed here are often extrapolated into the severe accident regime. The review then considers the phenomena exhibited in the bundle experiments LOFT LP-FP-2, PBF SFD-ST, and the CORA tests 12, 13 and 17. Following this, the extent to which the experimental knowledge has been translated into computer models is considered by reviewing the status of the treatment of quench in four major code systems; SCDAP/RELAP5, ICARE/CATHARE, ATHLET-CD/KESS and MELCOR.

The occurrence of renewed heat-up, increased hydrogen production, additional material relocation and (for irradiated fuel) increased fission product release during core quench has been firmly established through the performance of the integral bundle experiments. The phenomena are important regarding accident management; the additional hydrogen might threaten the containment, and the increased fission product release increases the source term. Also, the increasing fuel temperatures, being counter-intuitive, might confuse the operators into taking inappropriate action.

The mechanisms governing these effects are not well understood, and separate-effects experiments are underway and planned at Forschungszentrum Karlsruhe (FZK) to provide a database suitable for mechanistic model development. The experimental programme is currently expected to complete in 1998. In the meantime, preliminary models have recently been installed in the major mechanistic severe accident codes, based on the data available now. These models are mainly empirical, and typically predict up to half the excess hydrogen produced during quench, this being non-conservative from a safety case perspective. The development however represents a significant advance, for there were no models for the excess hydrogen production at the time of the CSNI State-of-the Art Review in 1991.

This review notes the importance of the fission product aspect of degraded core quench, however detailed consideration of this item is outside the scope of the report. It is therefore recommended that this aspect, and those concerning quench of late phase core configurations (for example those experienced in TMI-2), be separately reviewed by suitable expert groups, to complete the coverage of the present report. The effect of extrapolating design-basis thermal hydraulic models to severe accident conditions also needs further consideration. Given a satisfactory outcome of these additional reviews, completion of the existing and planned experimental and code development programmes should enable closure of the outstanding related safety issues concerning the determination of the hydrogen source term.
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1. INTRODUCTION

An important accident management measure for containing severe accident transients in light water reactors (LWRs) is the injection of water to cool the degrading core. This issue was considered in the Committee on the Safety of Nuclear Installations (CSNI) State-of-the Art Report (SOAR) [1]; it was concluded that the phenomena in quenching degraded fuel bundles by water injection were poorly understood and modelled by severe accident codes. It was established that the quenching could cause renewed oxidation of the Zircaloy fuel rod cladding, giving reheating of the rods, a sharp increase in hydrogen production and the release of additional fission products, but the detailed mechanisms were unknown. The additional hydrogen might threaten the containment, and the increased fission product release increases the source term. The increasing fuel temperatures, being counter-intuitive, might confuse the operators into taking inappropriate action.

Evidence for these effects has been obtained from analysis of the TMI-2 accident [2], and in the bundle experiments LOFT LP-FP-2 [3], PBF SFD-ST [4], and in the CORA tests 12, 13 and 17 [5], [6]. Analysis of CORA-13 in the framework of CSNI International Standard Problem (ISP) 31 [7] showed that none of the codes used at that time (1991/92) was able to predict the renewed heatup and excess hydrogen production in the reflood phase of that PWR test. Additional evidence concerning the behaviour of degraded cores on rapid cooling came from the PHEBUS B9R test [8], where oxidised and degraded rods were cooled in cold steam. Since then, a new programme [9] of separate-effects tests has been started at Forschungszentrum Karlsruhe (FZK) in which single rod oxidised Zircaloy tube specimens have been quenched with water, further tests are planned with rapid cooling induced by high-velocity steam, simulating conditions above the quench front where the water has been completely evaporated.

This report summarises the status of experimental and theoretical knowledge concerning the quench of degraded cores, where the fuel rods are in a mainly rod-like geometry at the time of quench. Consideration of quench in late-phase core configurations, e.g. debris beds and melt pools, of the effect of quench on fission product release, and of the purely thermal hydraulic aspects of beyond design-basis quench, is considered to be outside the scope of the review.

The accumulation of know-how on analytical and experimental aspects of the reflooding and quenching in the thermal hydraulic design basis area provides some basis for modelling reflooding and quenching of damaged core. This is presented in Section 2 of this report. Reflooding of a hot damaged core following the start of a severe accident can lead to significant increases in the heating, melting, and oxidation of the core prior to termination of the accident. These effects have been observed in fuel bundle heating and melting experiments terminated by the addition of water and will be reviewed in Section 3 of this report. The extent to which the experimental knowledge has been translated into computer models is considered by reviewing in Section 4 the status of the treatment of quench in four major code systems; SCDAP/RELAP5 [10], ICARE/CATHARE [11], ATHLET-CD/KESS [12], [13] and MELCOR [14]. Following a discussion, the conclusions and recommendations are presented.

2. QUENCH AND REFLOOD OF A NON-DAMAGED REACTOR CORE- THERMAL HYDRAULIC ASPECTS

2.1 Description of the Phenomena

Reflooding and quenching are important phenomena for the design basis of Light Water Reactors under large break loss of coolant accident (LOCA) conditions. Consequently,
quenching and quench front progression have been interesting physical phenomena studied intensively over the last two decades. Indeed, it is usually postulated that the core is uncovered and overheats due to decay heat from the fission products and the energy stored prior to LOCA during the so-called blowdown phase of a large break LOCA. In PWRs, the core is recovered, i.e. the overheated rods are quenched and adequate heat transfer is reestablished, by reflooding from below. Emergency core cooling (ECC) injection at the top of the core, or combined top and bottom injection are, however, also practised in certain types of PWR. In BWRs, the hot fuel rods may be quenched by a spray that forms a liquid film flowing down and cooling the rods as well as the walls of the “channels” containing the rod bundles and also the water rods in the rod bundles, together with bottom reflooding, due to injection of emergency core coolant. Emergency coolant injection by bottom reflooding creates a rising quenching front and upper head injection or core spray creates a falling film sputtering front. Many experiments in the 70’s and 80’s [16] were done for understanding and visualization of the reflooding front movement. Visual demonstrations with an overheated, red-coloured rod enclosed in a glass tube demonstrated clearly that quenching to the saturated temperature takes place in a distance of a few millimetres.

Water is forced into the core by the emergency core cooling system (ECCS) via bottom flooding (PWRs and some BWRs) or top spray (most BWRs) or by combined bottom and top flooding (some PWRs) in order to stop overheating of the fuel rods and reestablish cooling. Otherwise, cladding oxidation (Zircaloy-water chemical reaction) or clad melting and the consequent release of radioactive fission products can occur. In general, based on best-estimate or conservative assumptions during design-basis accidents, the boundary and initial conditions for reflooding tests can be established during the design-basis accidents. The variation of some of the main parameters can be summarized as [16]: system pressure 0.1 - 1.0 MPa, flooding velocities 1.5 - 30 cm/s (including natural reflood velocities), mass fluxes 7 - 300 kg/m²s, heater rod peak power 0.7 - 3 kW/m. The description of the thermohydraulic phenomena discussed below refers to design basis accidents with maximum clad temperatures up to about 1200 K. Effects like enhanced cool-down by turbulent steam flow above the two-phase region due to radiative heat transfer from clad to steam, which has a significant impact at temperatures above 1500 K, are therefore not taken into account.

Reflooding refers to a particular mode of post-burnout cooling of a hot channel by refilling it with coolant. Quenching or rewetting of the hot surface occurs during reflooding. Quenching refers to the transition from a mode of heat transfer characterized by total or almost total absence of liquid contacts with the wall to one where the wall is essentially wetted by the liquid. The heat transfer coefficient increases dramatically following quenching.

As a result of the high temperatures attained by the clad before the emergency coolant arrives, water does not initially wet the hot clad surface. Rewetting or quenching of hot surfaces occurs when the coolant re-establishes contact with the dry surface. The surface temperature corresponding to the achievement of liquid-solid contact is the rewetting or quenching temperature. The temperatures of the fuel pellets and fuel clad are reduced by heat conduction and convection to the coolant. As the coolant rises in a hot channel or in the overheated nuclear fuel rod bundle, complex heat transfer and two-phase flow phenomena take place and also the succession of regimes moves gradually up the rod bundle channels. The hot surfaces along the channels experience in turn free- or forced-convection cooling by steam, dispersed-flow film boiling, inverted-annular film boiling, transition boiling, nucleate boiling, and finally single-phase convection to the liquid. In the boiling curve, quench front propagation means a movement from the film boiling heat transfer mode through the transition boiling to the nucleate boiling regime. In a quenching front, the axial temperature gradient of the cladding may be so large that a significant part of the heat in the unwetted
part is removed by axial conduction in the cladding. For a fast front movement this gradient may be quite different between the inner and outer cladding surfaces. Other mechanisms may also support conduction-controlled quenching or be main contributors to the cooldown close to the saturation temperature. These are transition boiling caused by the collapse of the liquid-vapour interface in the inverted annular film boiling regime and droplet impingement through the steam layer in the transverse direction towards the wall.

During reflood multi-dimensional flow patterns occur in the core and upper plenum due to: flow rates and flow regimes such that gravitational forces are of the same order as inertial forces; non-uniformities in core power distribution and differences in resistance to flow through the intact and broken hot legs tend to promote multi-dimensional effects. As the flow passes through the upper plenum it must flow around several structural elements and additionally the flow behaviour of a collected pool in the upper plenum may be highly three-dimensional.

Another important phenomenon, observed in the PWR integral system large break LOCA experiments, is core wide cooling of the fuel cladding and quench during the blowdown phase. This phenomenon is important regarding heatup of the core, because it removes a large part of the stored energy from the fuel during the early phases of the transient. The blowdown-phase quench is caused by the hydraulic response of the primary system during the transition from subcooled to saturated choked flow at the break and by the operational characteristics of the primary system coolant pumps. The rapid cooling of the cladding is primarily a result of low-quality, high upward core flow (with an estimated velocity of about 2 m/s) at 7 MPa pressure range. This high mass flux rate may be enough for cooling the cladding on the whole core length without a specific quenching front, at a velocity of 1.0 to 1.5 m/s. In [17], it has been identified that there are no mechanistic models available for the sudden quenching during the blowdown phase. Safety codes currently apply an empirical heat transfer coefficient, based on transition to nucleate-boiling, as soon as low quality, high velocity, upward core flow is predicted to occur. As indicated in the same reference, also no directly applicable separate-effects type experimental data base exists for this phenomenon for temperatures above 1200 K.

The conduction-controlled quenching mechanisms for a rising reflooding front and a falling film front are quite similar. In the wetted region the heat is transferred, in both cases, by nucleate boiling and later by convection. In the unwetted part, the heat is removed by convection to vapour or to droplet-vapour mixture or by inverted annular flow film boiling during bottom reflooding. Precursory cooling effects support the quench front movement.

2.2 Present State of Knowledge - Predictive Capabilities

One possibility for calculating the quench front propagation velocity is using an analytical solution for the two-dimensional heat conduction problem in the cladding. In the mathematical model typically two fitting parameters are used, the quench temperature (Leidenfrost temperature, minimum film boiling temperature) and effective heat transfer coefficient (critical heat flux heat transfer coefficient) in the wetted region as user input. A better solution could be including the flow rate, void fraction, subcooling and pressure dependencies into fitting parameters, rather than just using constant coefficients. In principle these parameters could be identified with a heat transfer coefficient for critical heat flux (CHF) and a minimum film boiling (Leidenfrost) temperature. Experience during many years has shown, however, that the best fitting parameters in analytical correlations are more or less arbitrary. Very little work has been carried out in determining the maximum heat flux during rewetting. The reaction time of thermocouples is too slow for this purpose.
In advanced system codes quench front propagation is calculated by solving the two-dimensional (axial and radial) heat conduction in the fuel by a special fine mesh moving together with the quench front and dividing the normal calculational mesh into subsections. In this fine mesh the heat transfer modes are solved in detail from the boiling curve for all elements of the mesh, but the thermal hydraulic parameters are interpolated from the coarse mesh data. The axial conduction calculation requires that in the quenching zone the fine mesh size is a few millimetres. The quench front location is defined as a point where the critical heat flux temperature is reached. This temperature may be calculated as an intersection of the heat flux during nucleate boiling and critical heat flux. The advantage of the moving mesh method is the possibility of using the available heat transfer correlations; also for the fine mesh no special fitting is needed. The disadvantage is the very short mesh size needed in the rewetting zone.

The state of the art in modelling reflooding situations, mainly with the two-fluid system analysis codes is in some detail reviewed in reference [18]. Further details on the state of art are also provided with the 2D/3D Programme which studied multi-dimensional thermal hydraulics in a PWR core and primary system during the end-of-blowdown and post-blowdown phases of a large break LOCA and during selected small-break LOCA transients [19,20].

2.3 Measurement Ability and Experimental Data Base

The quenching front progression can be followed by cladding temperature measurements on different axial levels in the fuel rod. Thermocouples both on the inner and outer side of the cladding have been demonstrated to provide useful data. The question is, how well do the temperatures represent intact cladding temperatures in the surroundings? Due to the time constants of the thermocouples, there is only a limited possibility for measuring the rapid cooling characteristics in the precursory phase and during the final rewetting. The filler material of the fuel rod simulator has also a significant effect on the rewetting characteristics. In early test facilities steel tubes were used as rod simulators and the tubes were directly heated electrically by the Joule effect. Later indirectly heated rods were fabricated, where the cladding is filled e.g. by a compressed magnesium oxide powder, and a heater wire is positioned in the middle of the filler. These rods simulate better the heat capacity effect, but they cannot take into account the effect of the gas gap (in particular, the thermal resistance).

In addition to the surface temperature measurements, it is essential that the two-phase flow parameters are measured sufficiently accurately. The minimum measurements include:

- system pressure measurements
- pressure difference measurements for the core water inventory
- water inventory measurement in the upper plenum
- inlet flow measurement both for the net flow and for oscillations.

For a more detailed analysis, useful measurements include:

- steam temperature measurement above the swell level
- droplet size distribution measurement
- entrained water weighting measurement
- cladding temperature measurements in different circumferential positions.
2.4 Relevance to Nuclear Reactor Safety

The rewetting characteristics of the overheated core after a large LOCA was one of the most interesting research topics in the 70's and still has a significant influence on acceptance criteria in licensing and probabilistic safety analysis (PSA). The main interest is related to the maximum temperature in the core, but this turn-over temperature is determined by the liquid dispersed flow well before quenching. Depending on the amount of water available, the cooldown takes place earlier or later.

The large temperature gradient in the cladding gives rise to a mechanical stress on the cladding and it may affect fuel damage and radioactivity leakages. The rapid temperature drop is also associated with strong steam generation and this may have an effect on system characteristics including:

- liquid entrainment rate
- counter current flow limitation in the upper tie plate
- steam binding in the steam generator
- multi-dimensional flow distribution in the core.

3. QUENCH EXPERIMENTS- HIGH TEMPERATURE PHYSICO-CHEMICAL MATERIAL BEHAVIOUR

This section provides a review of the integral and separate-effects bundle experiments relevant to the reflooding of a damaged reactor core during a severe accident. The main features of the integral and separate-effects experiments are presented in Tables 1 and 2 respectively.

3.1 INEL Experiments

Two Idaho National Engineering Laboratory (INEL) experiments, OECD LOFT LP-FP-2 and PBF SFD-ST, contribute to the general database on the quenching of hot, damaged bundles. The LP-FP-2 experiment was the last, and final, experiment conducted in the Loss of Fluid Test (LOFT) facility at the INEL. This experiment, simulating a V-sequence LOCA scenario with a break in the Low Pressure Injection System (LPIS) line, was unique in a number of ways. The experiment was performed in the large scale (1/50 volume of a full-sized PWR) LOFT facility which was designed to represent the major component and system response of a commercial PWR. The experiment utilized a representative core with nine reduced length, 1.67 m, PWR assemblies. The central assembly, isolated from the remainder of the core by an insulated shroud, consisted of 11 Ag-In-Cd control rods, 100 fuel rods, and 10 instrumented guide tubes. LP-FP-2 is the only experiment with a combination of decay heating, severe fuel damage, and bundle quenching [15]. PBF SFD-ST was the first bundle heating and melting experiment conducted after the accident at TMI-2. It used a 0.91 m long bundle of 32 fuel rods. SFD-ST is the only experiment with combined fission heating, severe fuel damage, and bundle quenching with water.

The LP-FP-2 experiment subjected the LOFT core to a rapid blow down transient, with the core heating up to a maximum temperature of ~2400 K prior to reflood. The initial heating rate in the central assembly was ~1 K/s with an onset to rapid oxidation at a temperature near 1500 K. The peak temperatures in the central assembly remained above 2100 K for ~250 s prior to reflood. The SFD-ST experiment was a boildown transient with the bundle maximum temperatures estimated to be between 2670 K and 3100 K prior to reflood. The initial heating rate in the bundle was ~0.5 K/s. There was no apparent transition to rapid oxidation until peak bundle
temperatures exceeded 2100 K, with the slow initial heating rate intended to completely oxidize the bundle prior to reaching a temperature where the Zircaloy cladding would melt. However, as peak temperatures near 2100 K were reached, the oxidation heat generation, in combination with an inadvertent reduction in bundle liquid level, resulted in the rapid heating and melting of the bundle.

In both experiments, the reflood phase resulted in dramatic increases in hydrogen production, fission product release, and bundle damage. In the case of LP-FP-2, it is estimated that more than 75% of the total hydrogen was produced and a majority of the fission products was released during reflood and quenching of the core. Peak temperatures also increased from 2400 K to above 2800 K, the fuel and oxidized cladding melted, a substantial flow blockage was formed, and significant damage to the upper core plate occurred during this phase. In the case of SFD-ST, it was not possible to determine whether bundle temperatures increased during reflood since the bundle had already reached peak temperatures near the melting point of the fuel. However, it is estimated that over 50% of the total hydrogen was produced during reflood [15].

3.2 CEA/IPSN Experiments

Severe Fuel Damage experiments in light water reactors have been performed in the in-reactor PHEBUS SFD programme to study the early phase of core degradation (T < 2800 K, no significant loss of rod geometry) using fresh fuel [8], [21], [22]. The current PHEBUS FP programme is aimed at studying, regarding the core behaviour aspect, the core degradation and FP release for irradiated fuel in later phases of core degradation including large-scale fuel liquefaction and loss of rod geometry [23], [24], [25].

The impact of the quenching phase using liquid water was not directly investigated in the PHEBUS SFD tests but the test B9R enabled study of the continued oxidation of rods previously degraded by a transient characterised by significant oxidation followed by a cooling phase. Additional data were also obtained on the oxidation and fuel cladding fragmentation of oxygen-embrittled cladding induced by a rapid final cooldown simulated by a high steam flow rate injection [8], [26], [27].

The main characteristics and results of the PHEBUS SFD B9R test are:

- Due to an operating incident this test was carried out in two independent parts named B9R1 and B9R2.

- The B9R1 part was characterised by steam-rich conditions and a continuous cladding heat-up (maximum temperature of 1800 K) ended by a final cooling phase (1 to 2 K/s). This B9R1 transient was similar to the first B9 test of the SFD programme which gave valuable and precise information on the final clad oxidation and ZrO₂ layer cracking.

- The second part named B9R2 involved five phases:

  (a) an initial heating phase in pure helium up to 1000 C,

  (b) a first oxidation phase in steam-rich conditions (maximum temperature about 1900 K) characterised by an unexpected high oxidation escalation in the upper bundle zone (20 to 30 K/s),

  (c) an intermediate cooling phase (2 K/s to 0.3 K/s) resulting from both a reactor
power shutdown and a high steam flow rate injection,

(d) a second oxidation phase characterised by a new high oxidation escalation of the bundle mid-zone (10 K/s), and

(e) a final rapid cooling phase using high steam and helium gas flows.

During the B9R2 transient, the two oxidation phases were characterised by sharp temperature escalations and H₂ productions higher than those which would have been expected in similar conditions with intact oxidised cladding. Cracking of the existing ZrO₂ layers during the two cooling phases at the end of B9R1 and during B9R2 (reactor shutdown) are the cause of the unexpected clad oxidation enhancement. These two thermal shocks induced a loss of the protective effect of the cladding ZrO₂ layers. The same phenomenon is suspected to occur in the oxidised and hot cladding zone located above the quench front during the reflooding phase.

During the final high cooling phase of B9R2 with steam and helium injection no significant clad oxidation was observed except locally and with a limited extent in the mid-zone. This was due to the quite total oxidation of the cladding in the mid and upper zones of the bundle resulting from the previous oxidation periods (B9R1 and the two oxidation phases of B9R2). In the lower bundle zone the clad temperature was insufficient to cause a temperature excursion at the onset of the cooling.

The final cooling phase had a large mechanical impact on the fuel rod embrittlement. In particular a significant shattering of the totally oxidised cladding was observed in the mid and upper zones of the bundle. This clad disappearance induced locally a total loss of rod geometry characterised by a cavity overlying a debris bed of UO₂ pellets and fragments. The mid-zone of the bundle is characterised by intact rods surrounded by small particles of ZrO₂ of a sandy appearance, coming from the upper zone.

Destructive post-test examinations indicated also considerable deformations of the cladding (up to 70 %) with "flower shapes" and clad failures indicative of double-sided oxidation.

The current PHEBUS FP test matrix does not include quench conditions. Nevertheless, it is not excluded to obtain data on the impact of rapid cooling on rod degradation and FP release. In the first FPT0 test, a sharp clad oxidation escalation was observed in upper and mid-bundle zones (10 K/s) followed just after the total Zr consumption by a rapid cooling due to the radial heat losses through the shroud. This thermal shock could have induced a clad failure enabling an early and progressive loss of rod geometry. This process is one of the possible causes of the central rod disappearance which has been recently analysed [27].

3.3 FZK Experiments

3.3.1 Quench Experiments in the CORA Facility

Three of the nineteen CORA tests were performed as quench tests: CORA-12 and CORA-13 used PWR-typical bundles with Ag-In-Cd absorber and CORA-17 used a BWR-representative bundle with channel box walls and B₄C absorber [28]. Test CORA-13 was used as the basis for CSNI ISP-31 [7].

The test performance before quenching was the same as in the normal CORA tests; the power
increase was chosen to obtain an initial temperature rise of 1 K/s. A steam flow of 6 g/s was passed through the PWR-type bundles and of 2 g/s through the BWR-type bundle. The quench process was simulated by raising a water-filled cylinder over the heated bundle with a velocity of 10 mm/s.

The difference in the test procedure between the two PWR tests was the shorter time between power shutdown and quench initiation for CORA-13, resulting in a higher temperature of the bundle at start of quenching. The BWR test CORA-17 was similar to test CORA-12 in respect of the delay time between the power shutdown and the start of quenching.

All three tests showed during the quench phase a temporary temperature increase correlated with a peak in the hydrogen production. In test CORA-12, with a delay of 300 s between power shutdown and start of quenching, resulting in a cooldown of more than 100 K, a delay of about 50 s was registered between start of quenching and the initiation of the increase of temperature and hydrogen production. The water level at this time had already reached the elevation of about 200 mm. The temperature increase had started in the upper part of the bundle. In test CORA-13, with a start of quenching 40 s before shutdown of the power, temperature and hydrogen production increase started immediately after start of quenching. The temperature and hydrogen generation increase was higher in CORA-13 than in CORA-12.

The BWR test CORA-17 resulted immediately after quenching in a modest temperature increase for 20 s and changed then into a steep increase resulting in the highest temperature and hydrogen peaks measured in the three tests [28]. CORA-17 also showed a temperature increase in the lower part of the bundle, in contrast with CORA-12 and CORA-13 where temperature increases were only registered in the upper half of the bundles.

The common source for the temperature increase and hydrogen production in PWR and BWR tests is the exothermic Zircaloy/steam reaction. For 1 g of zirconium oxidised, 6.7 kJ of heat is released, and for 1 mole of zirconium oxidised, two moles of hydrogen are produced. The reaction is exponentially dependent on the temperature, following an Arrhenius relationship. As a result of the thermal shock during the quench process, the steam gains access to the hot metallic Zircaloy underneath the overlying oxide layer, however the mechanisms governing the breakup of the protective ZrO2 are not yet understood.

The earlier starting and stronger reaction in the BWR test can be interpreted as being due to the additional influence of the boron carbide absorber. This material has an exothermic reaction rate three times larger than that of Zircaloy and produces 4 to 8 times more hydrogen. Probably the hot remaining columns of B4C (seen in the lower half of the corresponding non-quench test CORA-16) react early in the quench process with the increased upcoming steam. The bundle temperature, raised by this reaction, increases the reaction rate (exponential dependency) of the remaining metallic Zircaloy. Due to the larger amount of Zircaloy in the BWR bundle (channel box walls) and the smaller steam input during the heat up phase (2 g/s instead of 6 g/s), it is probable that more metallic Zircaloy survived oxidation during the heatup phase.

3.3.2 Separate-Effects Tests

The Forschungszentrum Karlsruhe (FZK) is performing an experimental programme to investigate the mechanisms of quench-induced oxidation of Zircaloy. A small-scale test rig was designed and built in which it is possible to quench short Zircaloy tube specimens by water and steam. The objective of the tests is to investigate the generation of new metallic surfaces by crack formation and fragmentation of the cladding. It is assumed that the newly formed surfaces result in enhanced oxidation and hydrogen generation. The Zircaloy test specimen to be used in most of
the tests is a cylindrical sample of standard Siemens/KWU Zircaloy-4 fuel rod cladding with a length of 150 mm, either empty or filled with ZrO₂ pellets. The specimen is suspended by a thin Zircaloy capillary tube inside a quartz tube of 40 mm internal diameter. Heating is provided by an induction coil around the section of quartz tube enclosing the specimen. The QUENCH rig allows quenching by water or rapid cool-down by steam. The quench water is contained in a quartz cylinder which is moved up inside the outer quartz tube at a predetermined rate. The heating of the specimen is terminated as soon as the quench cylinder reaches the lower end of the specimen. The rapid cool down by steam simulates the cooling of the material by turbulent steam flow in the upper part of a core during reflood. Steam flow rates up to 2 g/s have been used.

The temperature of the Zircaloy tube is measured by three thermocouples attached to its outer surface. The hydrogen content of the outlet gas (rate and integral value) is determined by a mass spectrometer.

Up to now the focus has been on tests related to quenching by water. The main parameter investigated was the extent of pre-oxidation and its influence on generation of new metallic surfaces. The specimens were heated and pre-oxidized in an atmosphere of argon and oxygen before quenching. Pre-test oxide thicknesses up to 300 μm and initial quench temperatures up to 1600 °C were used. Large heat losses to the surroundings prevented the onset of a temperature escalation as observed in bundle experiments such as CORA. The total amounts of generated hydrogen were therefore very small.

The mechanical behaviour of the cladding tube depends on the extent of pre-oxidation and on the specimen temperature before quench. Specimens with an oxide scale of 100 μm thickness show local spalling of the ZrO₂ layer in experiments with water quenching only. The local spalling is more pronounced at high temperatures prior to reflood (> 1400 °C). However, metallographic investigations show that no subsequent oxidation occurred on the newly-formed surfaces. Macroscopic cracks, penetrating both oxide and metal, formed at specimens with oxide layers of 300 μm ZrO₂ in both water quenched and steam cooled experiments. The macroscopic cracks can be observed at all investigated temperatures, but they are more pronounced at lower temperatures prior to quench (1000 °C, 1200 °C). Metallographic examinations show the formation of oxide layers on the surface of the cracks. The influence of the cracks can also be found in the hydrogen measurement. Tests with specimens with a pre-formed ZrO₂ layer of 300 μm generated as much or even more hydrogen than tests with a ZrO₂ layer of 100 μm. The cracks are generated during the tetragonal-to-monoclinic transformation in zirconia. This transformation is connected with a rapid change of thermal strain, creating stresses in the material.

Three tests have been performed to investigate qualitatively the influence of hydrogen uptake by Zircaloy tube specimens cooling from 1500 °C. The output of hydrogen from a determined input has been measured without any specimen, with an unoxidised Zircaloy tube specimen and with a pre-oxidized specimen. A part of the hydrogen was absorbed by the Zircaloy tube specimens; more was absorbed by the unoxidised specimen than by the preoxidised specimen. The absorption of hydrogen by zirconium is a well-established phenomenon [29], [30]; its relative importance regarding other relevant unmodelled phenomena requires further investigation.

Hydrogen absorption by metallic Zircaloy during pre-oxidation in steam should alter the hydrogen release in the quench phase. It is therefore planned to perform the pre-oxidation in steam instead of an argon-oxygen mixture. Further experiments will also be focused on the investigation of critical oxide layer thicknesses and critical cool down rates resulting in cracking of the tube specimen.
The separate-effects tests carried out so far show that some of the typical boundary conditions during quenching of a reactor core can only be simulated by large-scale bundle tests. For this reason FZK will build a large-scale bundle test facility (QUENCH Facility) to address the issue of atypical boundary conditions.

4. CODE MODELS

This section provides brief summaries of the quench models in the detailed mechanistic codes SCDAP/RELAP5, ICARE/CATHARE and ATHLET-CD with KESS, along with the faster-running integrated system code MELCOR. Emphasis is placed on the material behaviour aspects (treatment of oxide shell cracking and enhanced hydrogen production).

4.1 SCDAP/RELAP5

SCDAP/RELAP5/MOD3.1 is the current production version of SCDAP/RELAP5 (developed at Idaho National Engineering Laboratories, USA) and is the first version which incorporates many of the model improvements recommended by the USNRC formal Peer Review Committee [31]. The thermal hydraulic model uses a one-dimensional, non-homogeneous and non-equilibrium thermodynamic two-fluid approach. The core degradation module contains two general sets of models developed to treat the accelerated heating and melting associated with reflood and quenching of a reactor core. These were developed as a result of the experimentally observed behaviour in the LOFT and CORA quenching experiments. The first set of models treat the cracking/spalling of the oxidized portion of the Zircaloy cladding. The second set of models treat the oxidation of (a) the liquefied U-Zr-O as it relocates along the surface of the fuel rods and (b) the solidified crust of debris.

The initial version of SCDAP/RELAP5/MOD3.1, which was released in 1994, contained two bounding correlational-based models to treat the cracking/spalling of the oxidized fuel rod cladding. One model, a local cracking model, cracked the protective oxide layer at a given elevation when oxidation-embrittled Zircaloy cladding cooled rapidly to a temperature between 1150 to 1560 K; the derivation of this model made use of the design-basis embrittlement experiments of Chung and Kassner [32]. The other model, a global cracking model, cracked the protective oxide layer over the full height of the fuel rod cladding as soon as water was added. In both cases, enhanced oxidation and heating then occurred due to the oxidation of the hot Zircaloy layer exposed by the cracking of the protective oxide. Mass diffusion of steam to the surface of the cladding was the oxidation-rate limiting process once the oxide was cracked. The flooding rates and heat transfer during the reflood process were described using standard RELAP5 thermal hydraulic constitutive models.

After a systematic assessment of these local and global oxide cracking models, a model was implemented into SCDAP/RELAP5/MOD3.1 to calculate the relocation and oxidation of liquefied U-Zr-O that slumps. During reflood, liquefied U-Zr-O is formed during the initial increase in temperature resulting from the oxidation of the hot zircaloy layer exposed by oxide cracking. The model of liquefied U-Zr-O relocation and oxidation, based on the observation of U-Zr-O rivulet and droplet flows in the German CORA experiments, describes the motion of droplets of liquefied U-Zr-O and the resulting oxidation of those drops. Subsequent assessment of this new model indicated a significant improvement in SCDAP/RELAP5 predictions of the response of the experimental bundles during reflood. Where versions of the code without this model had predicted little, if any, increase in bundle oxidation and heating during reflood. With this model SCDAP/RELAP5/MOD3.1 indicated a significant increase in the oxidation, bundle
temperature, and progression of core damage during reflood. Details of the results are provided in the code's Developmental Assessment report, volume 5 of [10], where analyses of CORA-13, LOFT LP-FP-2 and PBF-ST are described.

4.2 ICARE/CATHARE

The ICARE/CATHARE code is being developed at the Institut de Protection et de Sûreté Nucléaire (IPSN). The stand-alone ICARE2 V2 code which is developed at Cadarache in the framework of the PHEBUS programme, is aimed at calculating the behaviour of the core degradation during a severe accident [33], [34].

The current stand-alone ICARE2 V2 Mod2 version deals with the core degradation behaviour in a gaseous atmosphere (no two-phase flow model). Regarding the behaviour of rods under quench or rapid cooling conditions, there is no mechanistic model able to predict both the clad cracking with a partial or total loss of the protective effect of the clad ZrO2 layers and the resulting enhancement of clad oxidation.

A simple parametric model is available based on the existing oxidation model [35] (parabolic correlations for intact cladding). This model was derived from observations of the PHEBUS B9R test and can be summarised as follows.

- A user-specified criterion must be given to impose the loss of the protective effect at the external ZrO2 layer (cracking of the ZrO2 and of the underlying α-Zr(O) layer).

- The Zr oxidation is calculated using the usual parabolic correlations but the steam-Zr contact surface is supposed increased by a user-specified factor in order to take into account both clad deformations and cracking of the underlying Zr layer. This is based upon the observation of cracking through both the ZrO2 layer and the internal α-Zr(O) layer.

The parametric model enabled a correct calculation of the two temperature escalations observed in B9R2 (see section 2.2) when the Zr-steam contact surface was increased by 270%.

A more mechanistic clad embrittlement model is under development based on the calculation of local thermal stresses and associated cracks. This model should enable the reduction of the protective effect of the ZrO2 layer based on the calculated length of the cracks. This model will be improved and validated on future FZK quench tests. Whatever the shattering model, in case of renewed melt formation inside the cladding, relocation can occur. During all these processes of melting, dissolution and relocation, the Zr-rich material continues to be oxidised.

The future ICARE/CATHARE version able to calculate the two-phase flow behaviour and the core voiding and degradation up to the vessel failure is foreseen in mid-1997. This version should not involve a mechanistic damaged core reflooding model (thermal hydraulic aspect). The rising quench front model will be developed after 1997 taking into account the multi-dimensional thermal hydraulic model of CATHARE and the above-mentioned shattering models.

4.3 ATHLET-CD/KESS

The ATHLET-CD code [12], [36], which is developed by GRS Garching, uses a five-equation model (two energy equations) for thermo-fluid dynamics, taking into account thermodynamic non-equilibrium between liquid and vapour; a six-equation formulation is under development. The presence of non-condensibles (nitrogen, oxygen and hydrogen) in thermal and mechanical
equilibrium with the steam phase is considered. The quench front modelling has recently been updated [37], with good agreement with data reported for design-basis temperatures, however hydrogen production in high-temperature quench is severely underpredicted [38].

Core degradation is modelled using modules from the KESS code [13], developed at IKE Stuttgart. KESS can also be run independently, using a drift-flux thermal hydraulic model. The current development version of KESS [39, 40] includes two models for modelling the oxide embrittlement occurring on reflood: (1) a basic model in which cracking is triggered using a criterion depending on oxide layer thickness and cooling rate; and (2) a detailed model in which the thermal stress distribution in the cladding is calculated in a cylindrical geometry. Oxidation of cladding and U-Zr-O crusts is modelled. In the detailed model, shattering of oxidised film is assumed if the effective thermal stress exceeds the engineering stress; or an increase of the oxidation rate is assumed due to cracking of the ZrO$_2$ layer and/or crust using a correlation depending on the effective thermal stress. Testing of the models in KESS is confined so far to simulation of CORA-13. Further development is planned on modelling crack formation in the oxidic films (ZrO$_2$ layer, etc.). The models are to be implemented in ATHLET-CD, which provides a more realistic representation of the thermal hydraulic conditions. Further model improvements and validation using data from FZK experiments are also now underway.

4.4 MELCOR

MELCOR is a system-level integrated code developed at Sandia National Laboratory (SNL), US, for calculating the behaviour of light water reactor systems under severe accident conditions, from initiating event through to the source term. The thermal hydraulic modelling in MELCOR uses a control volume/flow path method which can account for two-phase water and an arbitrary number of non-condensable gases. Each volume can contain two independent fields, pool and atmosphere, and is characterised by a single pressure; the temperatures of the fields can be independent. The USNRC Peer Review [41] of version 1.8.1 noted limitations in applying the well-mixed control volume approach, citing reflood as a problem area.

SNL has performed two MELCOR studies relating to reflood experiments. Gross et al. [42] used MELCOR 1.8.1 to generate blind calculations for the CORA-13 experiment (ISP-31), and Kmeteyk [43] used the same code to perform an exhaustive analysis of the LOFT integral experiment LP-FP-2.

Concerning the material behaviour aspects of quench, the current code version 1.8.3 contains no models for oxide cracking and spallation on quench, etc., and the code authors recognise the need for improvements here [14]. Metallic oxidation is modelled using standard parabolic rate equations, and the current code version introduced diffusion limitation of oxygen transport to the metal surface, using a mass transfer analogy in a way similar to SCDAP/RELAP5/MOD3.1.

5. DISCUSSION

The above two sections have summarised the experiments relevant to in-vessel degraded core quench and the status of the modelling in four major representative code systems. The integral experiments are characterised by renewed oxidation and heat-up of regions of partially-oxidised Zircaloy cladding ahead of the rising quench front, accompanied by increased hydrogen production and renewed melt formation/relocation, before the renewed heat-up is terminated by the continuing quench process and/or by the complete consumption of the remaining metallic material. The PHEBUS B9R experiment shows that the thermal shock induced by a rapid cooling rate can cause the formation of an oxidic debris bed (ZrO$_2$ and UO$_2$ fragments). This loss of the
rod-like geometry will enhance the fuel melting and relocation processes. Where irradiated fuel is used, there is a sharp increase in fission product release. The renewed oxidation is associated with cracking and/or spalling of existing protective ZrO₂ oxide shells ahead of the quench front, leading to access of steam to newly exposed metallic surfaces (provided that the material is not heat fully oxidised) and a consequent increase in oxidation rate. Where steam-starved conditions existed prior to quench, relief of the steam starvation by boiling of the reflood water will also be a factor in increasing oxidation, though this may be countered by increased cooling brought about by the increased coolant flow. The single BWR experiment performed, CORA-17, shows that oxidation of boron carbide may also be significant in increasing the secondary heat-up on quench.

Spalling of ZrO₂ shells leads to the formation of oxidic debris beds composed of oxide shards, possibly accompanied by B₄C powder and fragments in BWR cases.

The integral tests do not provide sufficient information on the detailed physical processes governing quench-induced oxide shattering to enable the formulation of mechanistic models for the major code systems. A start has been made on the understanding of the processes at FZK Karlsruhe through the conduct of a series of separate-effects water quench tests on partially oxidised Zircaloy cladding. Cracking of the oxide shells has been observed, but no renewed heat-up has been observed, probably the large heat losses in the facility are a significant factor here. A second series is taking place using quench with a cold steam flow; the simpler thermal hydraulic conditions may make the results easier to interpret.

The heat transfer conditions expected during reflood in-reactor are not represented well in single rod experiments, and in recognition of this a series of bundle experiments is now scheduled to take place at FZK Karlsruhe as part of the European Union 4th Framework Programme, 1996-1998 [44]. These tests will employ 21-rod electrically-heated bundles in a circular shroud, in a configuration similar to that used in the PHBUS SFD and FP experiments. Preliminary design studies have been carried out, and special models for the heater elements to be used are being developed at FZK for SCDAP/RELAP5. The combination of results from the FZK single rod and bundle experiments should provide a sufficient database for the production of the required mechanistic models.

Given the importance of reflood as an accident management measure, preliminary models of degraded core quench have been introduced into the major mechanistic code systems such as those reviewed here. This development has been limited so far by the lack of suitable data, so the modelling is of a preliminary nature, based on the data that are available now. The codes now use mainly empirical- or correlational-based approaches to the oxide cracking; when shattering is signalled the effective oxide thickness which controls the reaction rate is set to zero. In the ICARE2 and KESS codes, a more mechanistic thermal stress-based approach is being introduced.

Limitations of oxidation, by lack of steam availability and by diffusion in the gas phase are modelled in most codes; the diffusive limitation modelling is a relatively new development and is commonly dealt with by a heat/mass transfer analogy. There are no models available for the possible boron carbide shattering effect observed in CORA-17, nor for the hydrogen absorption noted in the FZK single rod tests and elsewhere. In general, the hydrogen production calculated by previously released versions of the codes was less than that observed, typically by a factor of two (as for CORA-13) thus being optimistic from a safety case point of view; however the situation is being improved by the introduction of models for debris oxidation, for example for ICARE2 [21] and SCDAP/RELAP5 [45].

While detailed consideration of thermal hydraulic issues is outside the scope of this report, a brief discussion of those raised by ISP-31 (CORA-13) is nevertheless relevant here. A conclusion of the ISP-31 final report [7] was that the codes predicted too high a rate of quench, by a factor of up to two. During quench of degrading cores by water injection, modelling the rate of cooling (by
heat transfer to a highly non-equilibrium mixture of water, steam and non-condensibles is outside the range of validated correlations (developed for design-basis conditions), giving over-prediction of core cooling, thus this can again give non-conservative effects on the calculated renewed hydrogen generation and threat to the containment possible in such circumstances. The full benefit of development of mechanistic models for the mechanical effects of quench (e.g. thermal shock) will not be realised if the thermal hydraulic and heat transfer boundary conditions are not reliably calculated. Furthermore, since injection of water as an accident management measure may take place after at least part of the core has formed a debris bed, quench phenomena in such core configurations should also be reviewed; TMI-2 data are available here, including for the release of hydrogen. Also, in view of the importance of increased fission product release during reflood to the source term, it is recommended that this aspect receives specialist review.

6. CONCLUSIONS AND RECOMMENDATIONS

The occurrence of renewed heat-up, increased hydrogen production, additional material relocation, debris bed formation and (for irradiated fuel) increased fission product release during core quench has been firmly established through the performance of integral bundle experiments. The phenomena are important regarding accident management; the additional hydrogen might threaten the containment, and the increased fission product release increases the source term. Also, the increasing fuel temperatures, being counter-intuitive, might confuse the operators into taking inappropriate action.

The mechanisms governing these effects are not well understood, and separate-effects experiments are underway and planned at FZK Karlsruhe to provide a database suitable for mechanistic model development. The experimental programme is currently expected to complete in 1998. In the meantime, preliminary models have recently been installed in the major mechanistic severe accident code systems, based on the data available now. These models are empirical and/or correlation-based, and typically predict up to half the excess hydrogen produced during quench, this being non-conservative from a safety case perspective. Nevertheless, the development represents a significant advance, for no models for the excess hydrogen production were present at all at the time of the CSNI SOAR in 1991. Improvement and validation of these models, taking account of the thermal hydraulic aspect of quench as recognised below, are necessary.

This review notes the importance of the fission product aspects of degraded core quench, however detailed consideration of this item, and of quench in late phase (debris bed) core configurations, is outside the scope of the report. It is therefore recommended that this aspect be separately reviewed by suitable expert groups, to complete the coverage of the present report. The effect of extrapolating design-basis thermal hydraulic models to severe accident conditions also needs further consideration. Given a satisfactory outcome of these additional reviews, completion of the existing and planned experimental and code development programmes should enable closure of the outstanding related safety issues concerning the determination of the hydrogen source term.

7. ACKNOWLEDGEMENTS

The authors are grateful for comments from K Trambauer of GRS Garching and K Müller of IKE Stuttgart on the ATHLET-CD and KESS codes, and to K Bergeron of Sandia National Laboratories for material relating to MELCOR. Technical review of the document was provided by J N Lillington of AEA Technology, Winfrith. The AEA Technology contribution to this report was funded by the UK Health and Safety Executive.
REFERENCES


42. Gross R J et al., "MELCOR 1.8.1 Calculations of ISP31: The CORA-13 Experiment", 25


<table>
<thead>
<tr>
<th>Test</th>
<th>Number of Rods (Fuel/Abs)</th>
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<th>Heating Method</th>
<th>Fuel Irradiation (GWD/tU)</th>
<th>Fluid</th>
<th>Pressure (MPa) (System/Rod)</th>
<th>Initial Heat-up (K/s)</th>
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**Table 1: Bundle Reflood Tests - Main Experimental Conditions**

**Key**
- AIC: silver/indium/cadmium
- B\(_4\)C: boron carbide
- Incl: Inconel
- Zry: Zircaloy
- Ar: argon
- stm: steam
- wtr: water
- G: Guide
- Q: Quench
- S: Slow
- R: Rapid
- Y: Yes
- N: No
- blade: BWR control blade simulator consisting of Zircaloy channel box walls and a control blade simulator (stainless steel) plus typically 9 B\(_4\)C-loaded pellets
- *: Transient Duration is total time spent over 1100/1500/2100/2800K respectively, up to when there is no further significant change in core state (here taken as 2100K on final cooldown)
- (a): During pure helium phase
- (b): Cladding failure imposed when rod plug temperature reached 1120K
<table>
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<tr>
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Table 1: Bundle Reflood Tests - Main Experimental Conditions

Key:

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