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# Degraded Core Quench: Summary of Progress 1996-1999

## Executive Summary

*T.J. Haste,*
*Institute for Systems, Informatics and Safety,*
*European Commission, Joint Research Centre, Ispra, Italy*

*K. Trambauer*
*Gesellschaft für Anlagen und Reaktorsicherheit (GRS) mbh*
*Garching, Germany*

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Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

**NUCLEAR ENERGY AGENCY**

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 27 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

## COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.

CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus in different projects and International Standard Problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups and organisation of conferences and specialist meeting.

The greater part of CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

In implementing its programme, CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

# DEGRADED CORE QUENCH: SUMMARY OF PROGRESS 1996-1999

T. J. Haste [a] and K. Trambauer [b]

[a] Institute for Systems, Informatics and Safety, European Commission, Joint Research Centre, Ispra, Italy

[b] Gesellschaft für Anlagen und Reaktorsicherheit (GRS) mbh Garching, Germany

## EXECUTIVE SUMMARY

A status report on experiments and modelling relating to quench of degraded cores was issued by CSNI in August 1996, following the publication of the In-Vessel Core Degradation Code Validation Matrix. In response to a request by PWG2 through the TG-DCC, a review of progress since then to June 1999 has been performed. The scope is broadly the same as before, restricted to mainly rod-like geometries and not considering pure debris bed configurations. The scope has been increased slightly to include a VVER bundle quench experiment, CODEX-3, which falls within the parameter range of the Western bundle experiments performed to date. The same format has been adopted as before, with the experimental results for bundle and separate-effects tests being summarised in separate tables, updated from the earlier report. Most of the European activity (in particular that concerning FZKA experiments/modelling and KESS development) has been in the framework of the EC 4th Framework shared cost action “Core Degradation” (COBE project), work for which concluded at the end of January 1999.

The earlier status report noted that quenching a severely damaged core could cause renewed oxidation of the Zircaloy cladding, giving reheating of the rods, a sharp increase of hydrogen production and enhanced release of fission products, but the detailed mechanisms were unknown. Experiments since then have continued to investigate reflooding of mainly rod-like core configurations, by means of electrically heated out-of-reactor experiments. Bundle experiments have addressed integral effects, while single rod experiments and tests on hydrogen absorption and release by the cladding have sought to identify the mechanisms in detail.

This review shows further evolutionary progress made in understanding the phenomena of fuel rod quench under severe accident conditions. The successful performance of commissioning and four main tests in the new bundle QUENCH facility at FZ Karlsruhe has provided valuable new data, supplemented by the VVER test CODEX-3 at AEKI Budapest. Temperature excursions and excess hydrogen production were only observed for quench from high temperature (2300 K) with a non pre-oxidised bundle (2 relevant tests); for quench from lower temperatures (1750-1870 K) and with pre-oxidation (50-500 µm oxide) smooth cooling with no significant excess hydrogen production was observed (3 relevant tests). When cooling a non pre-oxidised bundle from 1870 K rapidly by steam, no significant excursion was observed (1 test). These new lower temperature bundle tests have usefully extended the parameter range down from that previously covered (quench temperature 2150K and above, no pre-oxidation, temperature excursions/excess hydrogen production always observed), and have shown that there are conditions for quench from high temperature where excess temperatures and hydrogen production do not necessarily occur.

The main requirement for further bundle quench tests would appear to be generally towards experiments at higher temperatures where excursions are probable (hence where there is a possible threat to the containment), where control material and/or ballooning is present (more prototypic, look towards low

pressure transients), and towards one at least with VVER materials where there is a likelihood of an excursion (no data under these conditions are now available). The effect of system pressure is another parameter not yet studied systematically. In all these cases, there should be a sound basis in terms of assessed plant transient conditions in setting the boundary conditions for any new bundle experiments.

Turning to separate-effects tests, extension of the FZKA single rod series has again widened the scope of the database, to cover for example the effect of the cooling medium (cold steam vs. reflood water), in the temperature range 1273-1873 K (the current limit of the facility). The effects of oxide cracking and oxidation of newly exposed metallic surfaces have been clearly demonstrated, also the role of hydrogen absorption and release, particularly at 1673 K and below. The database on hydrogen absorption/release has itself been extended to these higher temperatures. The relative lack of cracking and excess generation of hydrogen at 1873 K compares with similar results obtained in the bundle tests performed with pre-oxidised clad at similar temperatures. No separate-effects data are however available for temperatures above 1873 K, which is where most of the excess hydrogen generation which poses a potential threat to the containment occurs; this is an incentive to extend the range upwards if at all possible. French quench experiments from temperatures of 1573 K and below have addressed the issues of the effects of irradiated cladding, and also hydrogen absorption, showing some beneficial effects of irradiation.

Concerning hydrogen absorption, this has been clearly observed in the FZKA and in earlier Hungarian experiments, and has been postulated as the cause of low temperature excursions seen in QUENCH-03 and in the earlier Phebus C3 test. A model for this process validated to 1723 K has been developed. However it is not clear that the conditions in the experiments which lead to such excursions are likely to occur in-reactor, so therefore it would be prudent to establish whether this is indeed the case, and whether there is an associated safety concern, before committing substantial effort to new experiments and further modelling development in this field. The amounts absorbed in situ (5-10% of the total) appear small and within the error margins allowed on the hydrogen source term to the containment. However, use of the recently developed absorption/release model should allow this uncertainty to be reduced or eliminated.

Concerning modelling, the most detailed work has been performed by IBRAE who have extended the SVECHA code to include mechanistic models calibrated largely on the FZKA single rod data, and assessed the code against data from the same series. Development of KESS to include oxide cracking models is being continued, again using the FZKA data, and significant progress has been made. ICARE/CATHARE V1 was released in mid-1999, and testing of a shattering model based on that in SCDAP/RELAP5 is under way. Development is also under way for reflood/shattering models in MELCOR, while further assessment of the existing models in SCDAP/RELAP5 itself against the new FZKA bundle data is now in progress. The work so far shows an improvement in capability in that where there is a temperature increase and excess hydrogen production on quench (possible safety problem for the containment), this can be calculated, but to a lesser extent than that observed (therefore optimistic). Where there is no excursion, this is correctly simulated.

This review has confined itself to the terms of reference of the original status report, thus not considering debris quench and other late phase matters. However it is noted that the experiments performed in the last two years have filled in gaps in knowledge at lower temperatures where the core would be in a mainly rod-like state. The LOFT LP-FP-2 experiment remains the only one (excluding TMI-2) performed with reflood from high temperatures where significant debris bed formation had occurred. Therefore, the possibility of conducting separate-effects experiments on quenching debris beds should be considered. Finally, the updating of the Degraded Core Code Validation Matrix which is now in progress should consider the new quench experiments, particularly with a view to including a QUENCH bundle test in the list of preferred experiments, when qualified, detailed data become freely available.

# 1. INTRODUCTION

A status report on experiments and modelling relating to quench of degraded cores was issued by CSNI in August 1996 [1], following the publication of the In-Vessel Core Degradation Code Validation Matrix [2]. In response to a request by PWG2 through the TG-DCC, a review of progress from the issue of the original status report to June 1999 has been performed. The scope is broadly the same as before, restricted to mainly rod-like geometries and not considering pure debris bed configurations. The scope has been increased slightly to include a VVER bundle quench experiment, CODEX-3, which falls within the parameter range of the Western bundle experiments performed to date. The same format has been adopted as before, with the experimental results for bundle and separate-effects tests being summarised in separate tables, updated from [1]. Most of the European activity, in particular that concerning Forschungszentrum Karlsruhe (FZKA) experiments/modelling and European code development, has been in the framework of the EC 4th Framework shared cost action "Core Degradation" (COBE project) [3], which ran from February 1996 to January 1999. This was summarised in the FISA-99 meeting scheduled for November-December 1999. Some conclusions from the COBE project are quoted here.

The status report [1] noted that quenching a severely damaged core could cause renewed oxidation of the Zircaloy cladding, giving reheating of the rods, a sharp increase of hydrogen production and enhanced release of fission products, but the detailed mechanisms were unknown. Experiments since then have continued to investigate reflooding of mainly rod-like core configurations, by means of electrically heated out-of-reactor experiments. Bundle experiments have addressed integral effects, while single rod experiments have sought to identify the mechanisms in detail. These have included detailed measurements of hydrogen absorption and release by Zircaloy at high temperatures. This knowledge is beginning to be fed into improved modelling, firstly in detailed codes, then in system codes. Progress is summarised in turn below.

To introduce the new material, a new initial section has been added summarising the effects of hydrogen in the primary circuit. The combustion of hydrogen in the containment, giving a pressure spike which could threaten the containment's integrity, has been reviewed elsewhere, for example [4], and a state-of-the-art report on containment issues (thermal hydraulics and hydrogen distribution) has recently been completed by PWG4 [5]. These latter issues are therefore not considered further here.

# 2. HYDROGEN BEHAVIOUR IN THE REACTOR COOLANT SYSTEM

The hydrogen, generated mainly by cladding oxidation and reaction of steam with boron carbide absorber material (if present) in the hot core region, flows upwards through the core, the reactor pressure vessel (RPV) and into the primary circuit.

If along the way it passes cooler core regions there is a possibility of absorption by unoxidised cladding material. The solubility of hydrogen in metallic zirconium depends on the cladding temperature and the hydrogen partial pressure in the fluid. It increases with decreasing temperature and increasing pressure. On the other hand oxide layers on the cladding (as formed during normal operation, by waterside corrosion) strongly inhibit diffusion into the metal, and the diffusion coefficients decrease with decreasing temperature. This yields a potential maximum of hydrogen absorption at temperatures of about 1200 K. As a consequence, some hydrogen can be stored in colder core regions (unless inhibited by protective oxide layers) and released later if the core heat-up reaches these regions. Some would be accumulated in any case as a result of waterside corrosion under normal operating conditions.

The hydrogen released from the core follows the general flow path towards the pressure sink, which is on the one hand the pressure suppression system or the leak or break to the containment or to the steam generator, and on the other hand the heat sinks in the steam generators. Hot hydrogen-rich flow has a lower density than cooler, steam-rich fluid which results in enhanced natural convection. Furthermore hydrogen has a higher heat conductivity than steam. Both effects, enhanced natural convection and high heat conductivity, lead to increased heat transfer to the heat sinks (cold pipes?). The convection is limited by friction loss and possibly by counter-current flow in the hot leg. Note that increased natural convection by this mechanism is likely to bring more steam into the hot part of the core.

Due to the fact that convection exists at all locations in the reactor cooling system (RCS) at every time, hydrogen diffusion in the gas phase might be neglected (note however that diffusion in the gas phase may limit the rate of reaction locally near rapidly oxidising metal surfaces). Therefore and as long as steam does not condense, any kind of de-mixing of the steam hydrogen bulk is not possible. But at locations where steam condenses (through the presence of a sufficient heat sink), hydrogen can be accumulated and subsequently the steam partial pressure, the steam saturation temperature and the condensation rate could be reduced.

Hydrogen accumulated in the steam generator might reduce or even suppress further flow to the heat sink, which results in reduced heat transfer to the secondary coolant system and subsequently to higher system pressure and/or increased leak or break flow.

In computer codes hydrogen is treated as other noncondensable gases. They are in mechanical and thermal equilibrium with the gas or steam phase. In general the main effect on heat and mass transfer is considered by the use of the steam partial pressure for the condensation with the assumption that the noncondensable gases are perfect gases and the mixing is perfect too. Also the effect of hydrogen on the transport properties of the gas phase is generally considered by use of simplified equations. Some codes consider the boundary layer effect on mass transfer, which is significant for laminar flow conditions.

# 3. QUENCH EXPERIMENTS

The sections below provide brief details of the experiments performed in the review period. Detailed numerical information is provided in Table 1 for the bundle experiments and in Table 2 for the separate-effects experiments. Table 1 has been extended to include the lengths of the rods that are heated.

It should be noted that the Validation Matrix [2] included quench tests in its Category 1 list (most preferred), namely CORA-13 and LOFT LP-FP-2 (as well as TMI-2), along with CORA-12 and CORA-17 in its Category 2 list. Their attributes may be readily compared with the newer tests by inspecting Table 1.

## 3.1 FZ Karlsruhe

### 3.1.1 Bundle experiments

A major part of the COBE project has been concerned with the performance of two bundle quench tests in the newly-constructed 21-rod electrically heated QUENCH facility at Forschungszentrum Karlsruhe (FZKA). The facility is designed to investigate the hydrogen source term that results from water injection into a light water reactor (LWR) core, using Zircaloy-clad rods containing zirconia pellets. There are strong similarities with the CORA tests, with the exception of the use of zirconia rather than urania pellets, but with instrumentation substantially improved to characterise better the temperature and offgas conditions during the reflood phase.

The choice of the test conditions took full account of likely plant conditions on the onset of reflood as well as the reflood rate itself, for cases where a mainly rod-like geometry is maintained. New SCDAP/RELAP5 calculations for a Westinghouse 4-loop PWR (TMLB’, SBLOCA transients) and for the proposed European Pressurised Water Reactor (LOCA and Loss of Offsite Power transients) were supplemented by an extensive literature review (BWR-4 and PWR plant; LBLOCA, SBLOCA and TMLB' transients). Specifically, the plant conditions relate to the maximum temperature at the onset of reflood, the axial and radial temperature gradients, and the oxide thicknesses (up to a few hundred µm). The latter depend for example on the prior temperature history (such as heatup rate) and the timing of the reflood after core uncovery, with delayed reflood tending to more oxidation. The waterside corrosion during normal operation (up to about 30 µm for modern PWR cladding) forms only a small part of the oxidation at this stage, most would have normally occurred in the reactor transient before the main reflood commences. In the QUENCH facility, the desired oxidation state at reflood was produced by using a "preoxidation" phase where necessary, calculated to give the required maximum oxide thickness; this could be checked by withdrawing a small diameter solid test specimen at temperature before reflood commenced. The most important parameter that cannot be simulated in the facility in the system pressure, which can be up to 70 bar at the time of reflood, whereas the rig operates essentially at atmospheric pressure.

The two main quench tests 01 [6] in the COBE programme (preoxidised to a maximum 300 µm oxide thickness, quenched from 1870 K) and 02 (not pre-oxidised, quenched from 2270 K) followed an extensive series of commissioning tests [7] in bundle 00 which was terminated by a quench test from an estimated temperature of 1750 K with an estimated 500µm pre-oxidation. Following the conclusion of COBE, two more quench tests were performed; test 03 which was similar to 02 but with a higher heat-up rate and with reflood starting at a higher temperature, and test 04 in which a non-pre-oxidised bundle was cooled by cold steam injection after the temperature excursion had started.

In summary, the results from the four main tests were as follows:

- QUENCH-01: The pre-oxidised bundle, heated up at 0.5 K/s, appeared to quench steadily during reflood (started at ~1870 K) with no evidence of any temperature excursion or excess hydrogen production (8-9 g associated with the transient plus reflood phase compared with 26-30 g in the pre-oxidation phase). The bundle was severely oxidised in the top third with large cracks in the embrittled clad, but the shroud was intact and undeformed. The maximum hydrogen concentration in the Zircaloy was 5 at%;
- QUENCH-02: The non pre-oxidised bundle, heated up at 0.45 K/s rising to 0.9 K/s, showed strong temperature excursions and substantial excess hydrogen production in the reflood phase (started at 1870-1970 K), where most (170 g) of the 190 g total hydrogen was detected. There was severe oxidation and embrittlement of the rod claddings with localised melt formation and relocation in the top quarter of the bundle, and some damage to the shroud. The maximum hydrogen concentration in the Zircaloy was 15 at%;
- QUENCH-03: The non pre-oxidised bundle, heated up at 0.6 K/s rising to 1.3 K/s, and with the reflood phase starting at 2070 K, showed similar temperature, melt and relocation behaviour to QUENCH-02. The damage to the shroud was more severe than in the previous test, and there was again strong hydrogen production in the quench phase. However there were disagreements amongst the instrumentation concerning the amount produced. The most reliable value for the total amount of hydrogen produced is 142 g, of which 123 g was detected in the quench phase;
- QUENCH-04: This test was also perfomed without pre-oxidation. The bundle was heated at 0.5 K/s rising to 1.5 K/s, and a temperature excursion started at 1560K. The steam cooling phase started at 2100K, this led to immediate rapid cooling of most of the bundle. At two elevations (two thermocouple positions) the excursion slowed, then continued to 2300-2350 K before the rapid cooling supervened. There was little hydrogen generated in the cooling phase, about 1 g out of a total of 12 g measured overall, this was consistent with the very limited continuation of the excursion following the rapid steam injection.

A notable feature of the experiments was the occurrence of temperature excursions starting in the unheated region at the top of the shroud, from temperatures of 750-800°C, which is more than 300 K lower than excursion temperatures associated with runaway oxidation by steam. FZKA have postulated that these excursions are driven by the exothermic hydriding reaction of Zircaloy in the shroud. This would suppose that the oxide layer which normally protects against hydrogen uptake is either absent (possibly absorbed into the metal under very steam-starved conditions as the excursion proceeds), or is defective in some way which allows hydrogen to diffuse in readily. This point has not yet been resolved. CEA/IPSN report that similar low temperature excursions have been seen in the Phebus C3 experiment, but no detailed data are yet readily available.

The heavily oxidised commissioning bundle showed no excursion when quenched from about 1770 K, but data here are sparser since much of the instrumentation had by then ceased to function (the aim of the commissioning experiments was mainly to test the facility, not to generate quench data per se). However, the overall behaviour appeared similar to that in QUENCH-01.

Post-test examination of the bundles has been performed for the commissioning test [7] and for QUENCH-01 [6], and is under way for the two later tests. Detailed post-test calculations using SCDAP/RELAP5 in particular have been performed within the framework of COBE, and accounts of these are included in the project's final report, which at the time of writing was not openly available (some general remarks are included in the section on the codes SCDAP/RELAP5, ICARE/CATHARE and KESS below). While the main focus of the analysis has been on material behaviour, the experiments also provide valuable information on thermal hydraulic performance under beyond design basis conditions, an area previously identified [1] as one where data are sparse.

Plans are being drawn up for possible new experiments during the next three years. These include further tests using steam cooling, faster flooding, effect of PWR control rods, effect of boron carbide absorber material, use of VVER-specific materials etc. Up to now, no quench experiment with VVER-specific cladding, where reflood follows the onset of an oxidation excursion, has yet been performed, so the performance of such a test would fill a gap in knowledge. At the time of writing FZKA had been invited to submit one of the new proposed tests as a candidate for an International Standard Problem.

### 3.1.2 Single rod tests and hydrogen uptake measurements

An extensive series of single rod tests is in progress at FZKA [8],[9] using short samples of Zircaloy tubing quenched over a wide range of pre-oxidation and cooling conditions. The initial series employed water quench (more prototypic of a core in general) while the second used steam cooling (thermal hydraulic conditions easier to analyse, and representative of the top of a reactor core). These series used argon/oxygen for the pre-oxidation phase, which does not give the hydrogen absorption by the Zircaloy metal that arises in steam oxidation. When the importance of the latter was fully realised, the second series was repeated with pre-oxidation in steam, and a fourth series will repeat the first similarly. While the radiative heat losses are high so that temperature excursions are not observed, deviations from monotonic cooldown behaviour and hydrogen production can be observed which can be correlated with the post-test condition of the specimens. The single rod experiments are supplemented by measurements of hydrogen absorption kinetics and saturation values for Zircaloy at high temperatures where no data were previously available. No difference in the behaviour of Zircaloy and Zr1%Nb in this respect were observed (see [10] for Hungarian data on the binary alloy).

Some main findings from the experiments are:

- the mechanical behaviour of the oxidised cladding depends on the quench temperature, quench medium and the extent of pre-oxidation;
- through-wall crack formation is observed, leading to oxidation of the remaining metallic parts of the cladding tube, increasing hydrogen generation;

- crack formation and oxidation of the crack surfaces is more pronounced at lower specimen temperatures before quenching;
  - for the series with pre-oxidation in steam, temperature increase during quench was seen for quench temperatures less than 1873 K and oxide layers more than 200 µm, but was never seen at 1873 K even for large oxide thicknesses;
- the cracking of the oxidised cladding can be explained by thermal shock, and phase change in the oxide layer;
- an intact oxide film acts as an effective barrier against hydrogen absorption from the coolant;
- during oxidation in steam, a limited amount of hydrogen (5-10%) is absorbed in-situ, the measured hydrogen concentration in the metallic part of the cladding amounts to <3wt% after the test;
- hydrogen absorption correlates with the extent of pre-oxidation at the onset of cooldown, and on the total crack length.

The protective effect of an intact oxide layer against hydrogen absorption at these higher temperatures is consistent with that long-established under normal operating conditions [11], [12].

The experimenters also judged that deviations from the regular cooldown curves could be triggered by the exothermal hydrogen absorption process (as postulated for the low temperature excursions in QUENCH-03), and that this needed further investigation. Hydrogen absorption and release were important and needed to be treated in mechanistic models.

These data have been analysed in detail by IBRAE and used in the development and validation of the SVECHA/QUENCH code (see section 4.5). Further analysis has been performed by IKE Stuttgart using KESS (see section 4.3).

### 3.2 AEKI Budapest

Experimental investigation of the interaction of a high temperature VVER bundle with cooling water under severe accident conditions has been carried out in the CODEX facility at AEKI, Budapest [13]. The test section included a seven-rod hexagonal fuel rod bundle, which was surrounded by a shroud of Zr alloy. The six peripheral rods were electrically heated using tungsten bars. The bundle was constructed of VVER materials: the cladding was made of Zr1%Nb alloy, real $UO_2$ pellets were used and the rods were fixed by stainless steel spacers.

The CODEX-3 experiment was performed in two phases:

- During the first phase the initial preheating period was followed by a power increase, which resulted in temperature increase. When 1473 K was reached the bundle was quenched with water. This quenching process did not result in temperature escalation, but led to cooling down of the bundle. Later examination of the rods showed that there was no damage, but some oxidation had taken place, with the estimated oxide layer thickness in the highest temperature region was approximately 50 µm;
- The second phase started with the heating up of the facility with hot argon and some electric power on the rods. After a stabilisation period at 773 K the electric power was increased at 2 W/s and steam was added to the coolant. Above 1473 K the steam oxidation accelerated the temperature increase. At 1773 K cladding temperature the power was switched off and the bundle quenched with water from the bottom. Slight temperature escalation was observed, however the maximum temperature was not higher than 1973 K.

The upper part of the bundle was strongly oxidised, the cladding had become brittle and the bundle damaged. The typical oxide layer thickness in the upper part was between 100-200 µm. Fragmentation and mechanical break-up as results of thermal shocks and embrittlement were observed, but no signs of cladding or pellet melting. The lower part of the bundle remained intact and the lower temperatures resulted only in a few micron thick oxide layer.

### 3.3 CEA/EdF

EdF and CEA/IPSN have collaborated in extensive series of small-scale oxidation and quench experiments which have addressed amongst other things the effects of oxidation, irradiation and hydrogen charging on quench behaviour at high temperature. While the experiments were primarily aimed at the design basis accident, the results are of interest here since the temperature range (to 1573K) extends into the lower end of that relevant to severe accidents, and provides unique data on the performance of irradiated clad. In addition, the data on the effects of hydrogen absorption complements those obtained at FZKA. Conclusions quoted by Cauvin and Grandjean at the 1997 Quench Workshop at Karlsruhe include:

- irradiated Zircaloy exhibits increased oxidation kinetics, with the enhancement increasing with temperature;
- at a similar oxidation rate, irradiated samples exhibit a slightly better resistance to mechanical failure during quench than unirradiated samples;
- hydrogen charging during irradiation seems to explain this behaviour;
- the kinetics of the quench are important.

## 4. CODE MODELS

### 4.1 SCDAP/RELAP5

The previous status report [1] noted that SCDAP/RELAP5/MOD3.1 included “local” and “global” bounding correlation-based oxide models to treat cracking and spalling of oxide during reflood. The current, recently-released version MOD3.2 [14] contains these models along with error corrections and thermal hydraulic improvements embodied in stand-alone RELAP5/MOD3.2. The MOD3.2 Development Assessment report in [14] describes assessment of the new code against data from the reflood experiments PBF SFD-ST, CORA-13 and CORA-17. Assessment of MOD3.2 has been performed against data from the FZKA QUENCH bundle experiments 01 and 02, in the concluding stages of the EC 4th Framework shared cost action “Core Degradation”. FZKA have extended the electrical heater rod model so that the code can be used for analysis of these experiments; a version of MOD3.1 was extensively used by FZKA and AEA Technology in design studies for the facility. FZKA have also developed a new transition boiling model for the RELAP5 part of the code.

The new work has shown that with suitable choices of input parameters the code can adequately simulate the heat-up phase (temperatures, hydrogen production) in QUENCH-01 and QUENCH-02, and also the quench phase in the first of these tests, where no additional heat-up and hydrogen production attributable to quench-induced oxide shattering phenomena were seen. In the quench phase of QUENCH-02, a temperature peak (lower than observed) and limited extra hydrogen production were calculated with the best-estimate local shattering model. This represents an improvement in capability. With the global shattering model invoked and oxidation of Zr-rich melt treated, the excess temperatures and extra hydrogen production could be calculated at reasonable rates until about half the excess had been calculated, however at that stage numerical failure supervened.

## 4.2 ICARE/CATHARE

The ICARE/CATHARE V1 version, based on the coupling of the ICARE2 V3mod0 and CATHARE2 V1.3L codes, was released in mid-1999. This version allows calculation of complete reactor accident sequences, including behaviour of the primary and secondary circuit description and degradation inside the vessel, up to vessel failure.

The main improvements of the new core module, V3mod0, compared to the previous one, V2mod2, are the following: a two-phase thermal hydraulics model, mechanistic transition between the early and the late degradation phase, and description of phenomena occurring in the lower head. Moreover, reflooding and quenching models have been included. Concerning reflooding, the CATHARE2 constitutive laws are used. Thanks to the standard 2D ICARE2 model for heat conduction, no special treatment has to be made for the thermal calculation. As for quenching, a shattering model similar to that in SCDAP/RELAP5 has been introduced as a first attempt. For both aspects, validation is underway on the LOFT LP-FP-2, CORA-13 and QUENCH-01 tests. Calculations for the QUENCH bundle tests show good agreement up to the start of the reflood phase, and further results are awaited.

## 4.3 ATHLET-CD/KESS

ATHLET-CD uses the same Quench Front Tracking Model as ATHLET for intact fuel rods. This model is sufficient to describe adequately the thermal-hydraulics during dry-out and quench (LOFT LP-FP-2, CORA tests) in the early phase of core degradation but it does not model the additional hydrogen generation due to crack formation. Such models are under development in the KESS code.

In KESS the cracking of $ZrO_2$ layers and U-Zr-O crusts are being modelled by taking into account the thermal stresses in the layers. If the effective stress exceeds a given stress limit, crack formation is assumed. The resulting increase of oxygen diffusion in the cladding or in the crust respectively is simulated by a parameter in the oxidation models. This leads simultaneously to enhanced heat release. Enhanced oxidation occurs in the region of pure vapour flow as well as under film boiling conditions. Thus, a correlation for inverted annular flow is used to take into account the heat transfer to the surrounding fluid. Post-test calculations of the FZKA single-rod quench experiments [8] show relatively good agreement with the measured temperatures. However, the hydrogen production was significantly underestimated by the calculation. A first analysis of the QUENCH-01 experiment [6] shows that by the use of suitable parameters the measured temperatures as well the hydrogen production can be matched by the code. It can be concluded that the interplay of the processes (crack formation, oxidation and heat transfer to the fluid) is crucial for an adequate simulation. More detailed modelling of these processes will be done in future using the extended data base derived from the FZKA quench tests. A basic model was planned to be implemented in ATHLET-CD at the end of 1999.

## 4.4 MELCOR

The previous status report [1] noted that MELCOR version 1.8.3 contained no model for oxide cracking and spalling on quench, etc., and that the code authors recognised this lack. This position remains for the current version 1.8.4 [15], and model development is now underway, in two related tasks. One task calls for including quench-related effects whenever the code user causes the reactor to be reflooded (bottom). The reflooding will produce additional steam which will oxidize hot exposed cladding, but the phenomena associated with quenching are not modelled. The task is therefore to add criteria for enhanced oxidation for cases where oxide shell cracking is to be expected, using QUENCH data as a guide, and to include a fuel fracturing provision as well so that under appropriate conditions the fuel rods will collapse to debris when quenched. The treatment for this is likely to be simplified and criteria-based. Another task that is very closely related is a core top spray model, primarily for BWRs. This model is to include counter-current flooding limitations in addition to the quench-specific oxidation phenomena. Work on both tasks is in progress.

### 4.5 SVECHA/QUENCH

The computer code SVECHA/QUENCH (S/Q) [9],[16],[17] is under development in the Nuclear Safety Institute (IBRAE) of the Russian Academy of Sciences for the detailed modelling of reflood phenomena. The code considers the main physical phenomena occurring during quenching of fuel rods. Zirconium oxidation leads to heat release and hydrogen generation and affects the mechanical properties of the cladding. High cooling rates in the course of quenching sustain non-equilibrium conditions at the interface boundaries of the cladding layers and lead to the appearance of a temperature gradient across the cladding. For the adequate description of oxidation kinetics under these conditions the *oxidation module* based on partial derivative equations for the multi-layer oxygen diffusion problem was developed. The oxidation module includes as a part the *hydrogen absorption module* which describes hydrogen interactions with the cladding in the steam/hydrogen gas mixtures, self-consistently with the cladding oxidation kinetics. The hydrogen absorption model was originally developed on the basis of Hungarian results [10], on the absorption by Zr1%Nb during steam oxidation from 900 to 1200°C

The influence of the beta-alpha phase transformation of Zircaloy and tetragonal-to-monoclinic phase transformation of the oxide during quenching as well as the additional stresses generated by temperature gradients on the stress state of the oxidized cladding, crack formation and spalling are considered by the *mechanical deformation* module.

The description of the heat exchange process requires simultaneous solution of two problems: (i) heat conduction problem inside the solid body (fuel rod) and (ii) heat convection problem in the surrounding two-phase water-steam media. For the solution of the first problem in the axially non-uniform multi-layer cylindrical structure (fuel pellets/gap/cladding) the *heat conduction* module was developed. This module is based on a 2-D finite difference numerical scheme with adaptive grid. The *thermal hydraulic* module developed for the description of the heat convection process considers different boiling regimes and accounts for the non-stationary motion of water-steam regions.

The heat exchange in the core determines the temperature of the rod surface and thus the oxidation kinetics, hydrogen generation and mechanical deformations of the cladding. The heat released due to Zircaloy oxidation considerably affects the heat exchange, especially at high temperatures. The mechanical behaviour of cladding determines cladding rupture that provides direct access of oxygen to fresh (non-oxidized) metal surfaces and thus intensification of oxidation and hydrogen production rates.

These detailed models have been incorporated into SVECHA/QUENCH in a tightly-coupled way which reflects the mutual feedback amongst the physical processes. The code was developed and extensively assessed in close collaboration with FZKA experimentalists on the basis of their single rod quench tests and hydrogen absorption/release measurements. The comparison of the calculated results with the experimental data (23 tests with water quenching and 6 tests with cooling by steam) shows satisfactory agreement.

## 5. CONCLUSIONS

This review shows further evolutionary progress made in understanding the phenomena of fuel rod quench under severe accident conditions. The successful performance of commissioning and four main tests in the new bundle QUENCH facility at FZ Karlsruhe has provided valuable new data, supplemented by the VVER test CODEX-3 at AEKI Budapest. Temperature excursions and excess hydrogen production were only observed for quench from high temperature (2300 K) with a non pre-oxidised bundle (2 relevant tests); for quench from lower temperatures (1750-1870 K) and with pre-oxidation (50-500 µm oxide) smooth cooling with no significant excess hydrogen production was observed (3 relevant tests). When cooling a non pre-oxidised bundle from 1870 K rapidly by steam, no significant excursion was observed (1 test). These new lower temperature bundle tests have usefully extended the parameter range down from that previously covered (quench temperature 2150K and above, no pre-oxidation, temperature excursions/excess hydrogen production always observed), and have shown that there are conditions for quench from high temperature where excess temperatures and hydrogen production do not necessarily occur. Some initial analysis of the QUENCH bundle experiments has been performed, particularly with SCDAP/RELAP5, and substantial efforts are being continued.

The main requirement for further bundle quench tests would appear to be generally towards experiments at higher temperatures where excursions are probable (hence where there is a possible threat to the containment), where control material and/or ballooning is present (more prototypic, look towards low pressure transients), and towards one at least with VVER materials where there is a likelihood of an excursion (no data under these conditions are now available). The effect of system pressure is another parameter not yet studied systematically. In all these cases, there should be a sound basis in terms of assessed plant transient conditions in setting the boundary conditions for the experiments.

Turning to separate-effects tests, extension of the FZKA single rod series has again widened the scope of the database, to cover for example the effect of the cooling medium (cold steam vs. reflood water), in the temperature range 1273-1873 K (the current limit of the facility). The effects of oxide cracking and oxidation of newly exposed metallic surfaces have been clearly demonstrated, also the role of hydrogen absorption and release, particularly at 1673 K and below. The database on hydrogen absorption/release has itself been extended to these higher temperatures. The relative lack of cracking and excess generation of hydrogen at 1873 K compares with similar results obtained in the bundle tests performed with pre-oxidised clad at similar temperatures. No separate-effects data are however available for temperatures above 1873 K, which is where most of the excess hydrogen generation which poses a potential threat to the containment occurs; this is an incentive to extend the range upwards if at all possible. French quench experiments from temperatures of 1573 K and below have addressed the issues of the effects of irradiated cladding, and also hydrogen absorption, showing some beneficial effects of irradiation.

Concerning hydrogen absorption and release, this has been clearly observed in the FZKA and in earlier Hungarian experiments, and has been postulated as the cause of low temperature excursions seen, for example, in QUENCH-03 and in the earlier Phebus C3 test. A model for this process validated to 1723 K has been developed. However it is not clear that the conditions in the experiments which lead to such excursions are likely to occur in-reactor, so therefore it would be prudent to establish whether this is indeed the case, and whether there is an associated safety concern, before committing substantial effort to new experiments and further modelling development in this field. The amounts absorbed in situ during oxidation (5-10% of the total) appear small and within the error margins allowed on the hydrogen source term to the containment. However, use of the recently developed absorption/release model should allow this uncertainty to be reduced or eliminated. The hydrogen concentration in the metallic part of the clad, measured after the test, amounts to <3wt% of the hydrogen produced during the test.

Concerning modelling, the most detailed work has been performed by IBRAE who have extended the SVECHA code to include mechanistic models calibrated largely on the FZKA single rod data, and assessed the code against data from the same series. Development of KESS to include oxide cracking models is being continued, again using the FZKA data, and significant progress has been made. ICARE/CATHARE V1 was released in mid-1999, and testing of a shattering model based on that in SCDAP/RELAP5 is under way. Development is also under way for reflood/shattering models in MELCOR, while further assessment of the existing models in SCDAP/RELAP5 itself against the new FZKA bundle data is now in progress. The work so far shows an improvement in capability in that where there is a temperature increase and excess hydrogen production on quench (possible safety problem for the containment), this can be calculated, but to a lesser extent than that observed (therefore optimistic). Where there is no excursion, this is correctly simulated.

This review has confined itself to the terms of reference of the original status report, thus not considering debris quench and other late phase matters. However it is noted that the experiments performed in the last two years have filled in gaps in knowledge at lower temperatures where the core would be in a mainly rod-like state. The LOFT LP-FP-2 experiment remains the only one (excluding TMI-2) performed with reflood from high temperatures where significant debris bed formation had occurred. Therefore, the possibility of conducting separate-effects experiments on quenching debris beds should be considered. Finally, the updating of the Degraded Core Code Validation Matrix now in progress should consider the new quench experiments, particularly with a view to including a QUENCH bundle test in the list of preferred experiments, when qualified, detailed data become freely available.

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# 7. REFERENCES

[1] Haste T.J. et al., "Degraded Core Quench: A Status Report", NEA/CSNI/R(96)14, OCDE/GD(97)5, August 1996.
[2] Haste T.J. et al., "In-Vessel Core Degradation Validation Matrix", NEA/CSNI/R(95)21, OCDE/GD(96)14, January 1996.
[3] Shepherd I. et al., "Investigation of Core Degradation (COBE) - Final Report", EUR 18982 EN, August 1999.
[4] IAEA/CEC, "Hydrogen in Water-Cooled Nuclear Power Reactors", EUR 14037 EN, October 1991.
[5] CSNI, "State-of-the-Art Report (SOAR) on Containment Thermal-Hydraulics and Hydrogen Distribution", NEA/CSNI/R(99)16, June 1999.
[6] Hofmann P., Miassoedov A., Leiling W., Piel D., Schmidt L., Sepold L. and Steinbrück M., "QUENCH-01 Experimental and Calculational Results", FZK 6100, November 1998.
[7] Hofmann P., Miassoedov A., Leiling W., Piel D., Schmidt L., Sepold L. and Steinbrück M., "Results of the QUENCH Commissioning Tests", FZK 6099, August 1998.
[8] Hofmann P. and Noack V., "Physico-Chemical Behavior of Zircaloy Fuel Rod Cladding Tubes During LWR Severe Accident Reflood. Part I: experimental results of single rod quench experiments", FZKA 5846, May 1997.
[9] Hofmann P., Miassoedov A., Steinbock L., Steinbrück M., Veshchunov M.S., Berdyshev A.V., Boldyrev A.V., Palagin A.V. and Shestak V.E., "Quench Behaviour of Zircaloy Fuel Rod Cladding Tubes: Small Scale Experiments and Modelling of the Quench Phenomena", FZKA 6208, March 1999.
[10] Frecska J., Konczos G., Maróti L. and Matus L., "Oxidation and Hydrating of Zr1%Nb Alloys by Steam", KFKI-1995-17/G, 1995.
[11] IAEA, "Corrosion of Zirconium Alloys in Nuclear Power Plants", IAEA-TECDOC-684, International Atomic Energy Association, Vienna, January 1993.
[12] IAEA, "Waterside Corrosion of Zirconium Alloys in Nuclear Power Plants", IAEA-TECDOC-996, International Atomic Energy Association, Vienna, January 1998.
[13] Hózer Z., Maróti L. and Windberg P., "Quenching of High Temperature VVER Bundle", accepted for the NURETH-9 Conference, October 1999.
[14] SCDAP/RELAP5 Development Team, "SCDAP/RELAP5/MOD3.2 Code Manual", NUREG/CR-6150 Revision 1 (5 volumes), July 1998.
[15] Gauntt R. et al., "MELCOR Computer Code Manuals, version 1.8.4", NUREG/CR-6119, Volumes 1 and 2, Revision 1, May 1998.
[16] Veshchunov M.S., Berdyshev A.V., Boldyrev A.V., Matweev L.V., Palagin A.V. and Shestak V.E., "Physico-Chemical Behavior of Zircaloy Fuel Rod Cladding Tubes During LWR Severe Accident Reflood. Part II: Modelling of quench phenomena", FZKA 5846, May 1997.
[17] Berdyshev A.V., Boldyrev A.V., Palagin A.V., Shestak V.E. and Veshchunov M.S., "Development of a Rod Quenching Model", NSI-SARR-57-97, IBRAE, Moscow, July 1997.

| Test | Number of Rods (Fuel/ Abs) | Spacer Grids (no.) | Heating Method/ Length (m) | Fuel Irrad-iation (GWD/ tU) | Fluid | Pressure (MPa) (System/ Rod) | Heat -up Rate (K/s) | Maxi-mum Tempe-rature (K) | Transient Duration (s) * | Test Termin-ation | Special Condi-tion | Date of Test |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| CODEX-3/1 | 7 | SS(3) | elect-rical/0.6 | none | Ar, stm | 0.21/0.22 | 0.6 | 1450 | 730/ 0/0/0 | Q(wtr) | N | 28Nov96 |
| CODEX-3/2 | 7 | SS(3) | elect-rical/0.6 | none | Ar, stm | 0.21/0.22 | 0.6 | 1900 | 1250/ 350/ 0/0 | Q(wtr) | pre-ox 50µm oxide | 29Jan97 |
| CORA-12 | 23/ 2(AIC) | Zry(3) | elect-rical/1.0 | none | Ar, stm | 0.2/0.3 | 1.0 | 2300 | 1500/ 1150/ 600/0 | Q(wtr) | bypass | 09Jun88 |
| CORA-17 | 18/ blade | Zry(3) | elect-rical/1.0 | none | Ar, stm | 0.2/0.5 | 1.0 | 2300 | 1400/ 1000/ 750/0 | Q(wtr) | bypass | 29Jun89 |

**Table 1 : Bundle Reflood Tests - Main Experimental Conditions**

Key

AIC : silver/indium/cadmium; $B_4C$ : boron carbide; Incl : Inconel; Zry : Zircaloy; SS : stainless steel
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 $B_4C$-loaded rodlets)
* : 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

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| Test | Number of Rods (Fuel/ Abs) | Spacer Grids (no.) | Heating Method/ Length (m) | Fuel Irrad-iation (GWD/ tU) | Fluid | Pressure (MPa) (System/ Rod) | Heat -up Rate (K/s) | Maxi-mum Tempe-rature (K) | Transient Duration (s) * | Test Termin-ation | Special Condi-tion | Date of Test |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| CORA-13 | 23/ 2(AIC) | Incl(1) Zry(2) | elect-rical/1.0 | none | Ar, stm | 0.2/0.4 | 1.0 | 2500 | 1400/ 1000/ 800/0 | Q(wtr) | bypass | 15Nov90 |
| LOFT LP-FP-2 | 100/ 11(AIC) + 10(G) | Incl(5) | decay heat/ 1.67 | 0.448 | stm, wtr | 1.40/2.41 | 2.2 | 3100 (esti-mated) | 650/400/ 300/5 | Q(wtr) | bypass | 03Jul85 |
| PHEBUS B9R-1 | 21 | Incl(2) | fission/ 0.8 | none | stm, He | 2.0 to 0.5/ 0.7 | 0.2 | 1800 | 7400/ 2800/ 0/0 | S(He) | (b) | 01Apr88 |

**Table 1 : Bundle Reflood Tests - Main Experimental Conditions**

Key

AIC : silver/indium/cadmium  $B_4C$ : boron carbide  Incl : Inconel  Zry : Zircaloy  SS : stainless steel
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 $B_4C$-loaded rodlets)
* : 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

| Test | Number of Rods (Fuel/ Abs) | Spacer Grids (no.) | Heating Method/ Length (m) | Fuel Irrad-iation (GWD/ tU) | Fluid | Pressure (MPa) (System/ Rod) | Heat -up Rate (K/s) | Maxi-mum Tempe-rature (K) | Transient Duration (s) * | Test Termin-ation | Special Condi-tion | Date of Test |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| PHEBUS B9R-2 | 21 | Incl(2) | fission/ 0.8 | none | He, stm | 0.5/0.7 | <0.1 to 1000K (a) | 2150 | 3200/800/ 50/0 | R(stm) | (b) | 14Apr88 |



**Table 1 : Bundle Reflood Tests - Main Experimental Conditions**

Key

| | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| AIC | : silver/indium/cadmium | $B_4C$ | : boron carbide | Incl | : Inconel | Zry | : Zircaloy | SS | : stainless steel |
| 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 $B_4C$-loaded rodlets)

* : 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

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| Test | Number of Rods (Fuel/ Abs) | Spacer Grids (no.) | Heating Method/ Length (m) | Fuel Irrad-iation (GWD/ tU) | Fluid | Pressure (MPa) (System/ Rod) | Heat -up Rate (K/s) | Maxi-mum Tempe-rature (K) | Transient Duration (s) * | Test Termin-ation | Special Condi-tion | Date of Test |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| PBF-SFD ST | 32 | Incl(3) | fission/ 0.91, 1.0 | trace | wtr | 7/>7.5 | 0.1-0.15 | >2673 | 7300/4100/ 500/? | Q(wtr) | N | 28Oct82 |
| QUENCH-00 | 21 | Incl(1) Zry(3) | elect-rical/1.0 | ($ZrO_2$ pellets) | Ar, stm | 0.2/0.22 | 1.0 | ≤1750 | 18784/ 265/0/0 | Q(wtr) | pre-ox ~500µm oxide | 9-16Oct97 |
| QUENCH-01 | 21 | Incl(1) Zry(4) | elect-rical/1.0 | ($ZrO_2$ pellets) | Ar, stm | 0.2/0.22 | 0.5 | ≤1870 | 4260/ 535/0/0 | Q(wtr) | pre-ox ~300µm oxide | 26Feb98 |
| QUENCH-02 | 21 | Incl(1) Zry(4) | elect-rical/1.0 | ($ZrO_2$ pellets) | Ar, stm | 0.2/0.22 | 0.45-0.9 | 2470 | 1470/260/ 120/0 | Q(wtr) | without pre-ox | 07Jul98 |

**Table 1 : Bundle Reflood Tests - Main Experimental Conditions**

Key

AIC : silver/indium/cadmium  $B_4C$ : boron carbide  Incl : Inconel  Zry : Zircaloy  SS : stainless steel
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 $B_4C$-loaded rodlets)
* : 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

| Test | Number of Rods (Fuel/ Abs) | Spacer Grids (no.) | Heating Method/ Length (m) | Fuel Irrad-iation (GWD/ tU) | Fluid | Pressure (MPa) (System/ Rod) | Heat -up Rate (K/s) | Maxi-mum Tempe-rature (K) | Transient Duration (s) * | Test Termin-ation | Special Condi-tion | Date of Test |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| QUENCH-03 | 21 | Incl(1) Zry(4) | elect-rical/1.0 | ($ZrO_2$ pellets) | Ar, stm | 0.2/0.22 | 0.6-1.3 | 2470 | 1200/600/ 150/0 | Q(wtr) | without pre-ox | 21Jan99 |
| QUENCH-04 | 21 | Incl(1) Zry(4) | elect-rical/1.0 | ($ZrO_2$ pellets) | Ar, stm | 0.2/0.22 | 0.5-1.5 | 2350 | 1200/400/ 15/0 | R(stm) | without pre-ox | 30Jun99 |

**Table 1 : Bundle Reflood Tests - Main Experimental Conditions**

Key

AIC : silver/indium/cadmium   $B_4C$ : boron carbide   Incl : Inconel   Zry : Zircaloy   SS : stainless steel
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 $B_4C$-loaded rodlets)
* : 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

| Institute | Facility/ Programme Name | Geometry | Specimen Length (mm) | Heating Method | Pressure (MPa) | Atmos-phere | Quench Temperature Range (K) | Reflood Rate (mm/s) or Steam Flow (g/s) | Special Condition | Date |
|---|---|---|---|---|---|---|---|---|---|---|
| CEA/IPSN + EdF | TAGCIS, TAGCIR, CODAZIR, HYDRAZIR | single rod, no pellets | 70 to 100 | electric, inductive | 0.1 | steam | 1323 - 1573 | Specimen immersion in quench tank | irradiated clad up to >60GWd/tU, pre-corrosion, $H_2$ charging | mid-1991/98 |
| FZK | QUENCH series 1 | single rod, $ZrO_2$ pellets | 150 | electric, inductive | 0.1 | Ar+25%$O_2$/ water quench | 1273/1473 1673/1873 | 15 mm/s reflood rate | pre-ox in Ar/$O_2$ <20/~100/ ~200/~300µm | 1995/97 |
| FZKA | QUENCH series 2 | single rod, $ZrO_2$ pellets | 150 | electric, inductive | 0.1 | Ar+25%$O_2$/s team cooling | 1473/1673/ 1873 | 1.5-2 g/s steam flow | pre-ox in Ar/$O_2$ <30/~100/ ~200/~300µm | 1997 |
| FZKA | QUENCH series 3 | single rod, $ZrO_2$ pellets | 150 | electric, inductive | 0.1 | Ar+steam/ steam cooling | 1373/1473/ 1673/1873 | 1 g/s steam flow | pre-ox in Ar/steam ~100 - ~350µm stepwise ~30µm | 1998 |
| FZKA | QUENCH series 4 | single rod, $ZrO_2$ pellets | 150 | electric, inductive | 0.1 | Ar+steam/ steam cooling | 1373/1473/ 1673/1873 | 1 g/s steam flow | pre-ox in Ar/steam to 150 and 250µm, Zr1%Nb | Planned for 2000 |
| FZKA | QUENCH series 5 | single rod, $ZrO_2$ pellets | 150 | electric, inductive | 0.1 | Ar+steam/ water quench | 1373/1473/ 1673/1873 | | pre-ox in Ar/steam up to 300µm, + effect of quench water subcooling, Zr1%Nb and Zry4 | Planned for 2000 |

**Table 2: Separate Effects Tests - Reflood**



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