Chimia 59 (2005) 950–956
© Schweizerische Chemische Gesellschaft
ISSN 0009–4293

Review of the Licensing Basis for RIA and LOCA Transients in Light of New Evidence on High Burnup Fuel Behavior

Martin A. Zimmermann* and Gerhard Bart

Abstract: The fuel of power reactors can be damaged by both rapid reactivity insertion accidents (RIA) and by Loss of Coolant Accidents (LOCA) due to the break of a coolant pipe. The consequences of such postulated accidents with a very low probability can be mitigated by dedicated safety systems. Yet, the possible consequences need to be assessed as part of the safety evaluation in the framework of the reactor licensing process as these transients form part of the so-called design basis. Most importantly, coolability of the reactor core must be demonstrated. The current paper reviews the important fuel related phenomena that may occur during such transients and their relation to the current licensing basis. Finally, possible changes in light of new experimental evidence from high burnup fuel behavior tests are discussed.

Keywords: High burnup fuel · Licensing basis · LOCA-transient · LWR · RIA-transient

1. Introduction

The need to protect the general public from release of radioactivity also in case of postulated nuclear power plant accidents was recognized from the very beginning: The design of nuclear reactors was guided by principles such as defense-in-depth, stipulating that more than one out of the three barriers 1) fuel cladding, 2) reactor pressure vessel, 3) containment must fail before any significant release of radioactivity can take place. In order to ensure the functionality of these barriers, the plant operation must remain within envelopes for process parameters that are determined for normal operation and several sets of plant transients. Besides operational disturbances that do not require the initiation of any protection systems in addition to the reactor shutdown system (SCRAM), three categories of design basis transients can be defined according to their postulated occurrence frequency (*e.g.* in Switzerland): Category 1 with postulated frequency between 10^{-1}/y and 10^{-2}/y, category 2 with postulated frequency between 10^{-2}/y and 10^{-4}/y and finally category 3 with postulated frequency between 10^{-4}/y and 10^{-6}/y.

It is generally accepted that both reactivity initiated accidents (RIA) and loss of coolant accidents (LOCA) fall into category 3, even though some studies suggest a frequency much below 10^{-6}/y for BWR RIA.

RIA and LOCA transients have been identified as so-called design basis transients that were employed to design emergency protection systems that form part of the light-water moderated nuclear power plants (LWR) built so far.

2. General Description of Fuel Behavior during RIA and LOCA

2.1. *Description of RIA Transients*

2.1.1. *Reactor System*

A Reactivity Initiated Accident (RIA) is defined by the insertion of positive reactivity into a critical reactor [1]. This can happen through different mechanisms: Removal of absorbing materials, *e.g.* the absorbing control rods or replacing the borated moderator in the reactor by a slug of non-borated moderator in the course of some accident scenarios. From a probabilistic point of view, the first scenario is more likely and in fact the fast removal of the control rods (Rod Ejection Accident, REA) serves as design basis transient for PWRs: After the postulated break of the control rod guide housing, the control rods are ejected from the reactor by the system pressure (~15 MPa) that acts against the containment pressure of 0.1 MPa. A very rapid reactivity insertion within a time period of ~0.1 s is the consequence of the control rod ejection. As a result, a power excursion (up to many times the nominal power) with a typical duration of a few 10 ms develops that is inherently mitigated by the negative fuel temperature reactivity feedback (Doppler effect) due to high fuel temperature. In order to maximally delay the Doppler-feedback, the reactor is assumed to be at hot zero power conditions: The reactor pressure stays at nominal value, and the fuel is assumed to be at minimal temperatures (close to the saturation temperature of the coolant, ~325 C).

In the case of BWRs, the respective scenario is more complicated because more than one single error is required to cause an RIA. However, BWR control elements have higher absorbing power (the ‘worth’ of a single BWR control element is larger than the ‘worth’ of a PWR control rod assembly not forgetting the intrinsically different neutronic characteristics of the two reactor types). Also, a BWR-RIA is typically assumed to occur during the start-up procedure with its many control rod move-

*Correspondence: M.A. Zimmermann
Labor für Reaktorphysik und Systemverhalten (LRS)
Forschungsbereich Nukleare Energie und Sicherheit (NES)
Paul Scherrer Institut
CH-5232 Villigen PSI
Tel.: +41 56 310 27 33
E-Mail: martin.zimmermann@psi.ch

ments; hence, the reactor is assumed to be at cold-zero power conditions with fuel temperatures slightly above the moderator temperature (between room temperature and ~80 °C).

2.1.2. *Fuel Behavior*

The rapid power excursion following the insertion of positive reactivity into the reactor causes an almost adiabatic heat-up of the fuel pins: As a consequence, the temperature profile – parabolic for fuel pins in normal operation – flattens out because the heat conduction process is slow in comparison to the duration of the power excursion.

Because the thermal expansion of the ceramic fuel pellet is larger than that for the metallic cladding, the gap between them will close early in the transient, and pellet cladding mechanical interaction (PCMI) starts. As a consequence, high stresses develop in the cladding and finally cause plastic deformations.

If the power pulse is large enough and consequently yields high fuel temperatures, sectors of the fuel pellet start to melt. This induces additional expansion due to the volume increase of melted fuel. The fuel pin can fail and molten fuel may disperse into the coolant. There, its enthalpy is liberated through rapid heat transfer facilitated by the large surface of the highly dispersed fuel, thereby potentially pressurizing the primary reactor system with the generated steam.

Alternatively, also non-melted particles (especially for high burnup fuel) may be dispersed into the coolant after the failure of the cladding. In this case, the size distribution of the particles is very important to determine how much energy stored in the fuel particles is rapidly transferred to the coolant. The effectiveness of the conversion from thermal to mechanical energy increases with increasing amount of energy deposited into the fuel by the rapid power excursion [2].

Transient spalling of the oxide layer on fuel cladding tubes leads to a significant local improvement of the heat transfer from the cladding to the coolant during the later phase of the RIA transient, with the result that departure of nucleate boiling (DNB) occurs and the cladding surface temperature rises accordingly [3]. This may lead to a late cladding failure.

2.1.3. *Safety Significance of RIA in Relation to Fuel Behavior*

The safety issue in relation to RIA transients concerns excessive pressure pulses generated as a consequence of the power excursion, possibly damaging or even disrupting the reactor pressure vessel. In order to prevent this scenario from happening, the fuel is not allowed to melt. It has been observed with many RIA-experiments – subjecting fresh and high burnup fuel to fast power excursions – that even non-melted fuel particles have been dispersed to the coolant [2] and their thermal energy has been partly transferred to mechanical energy (pressure pulses). As reactor cores are operated to continuously higher burnup levels, the fuel experiences longer duty periods with the potential of cladding embrittlement. Hence, the best way to withstand the challenges of postulated RIA-transients is to maintain a high level of cladding ductility combined with a minimal oxide thickness.

2.2. *Description of LOCA Transients*

2.2.1. *Reactor System*

The postulated break of a large pipe (Large Break LOCA) causes rapid depressurization and loss of the coolant that leaves the reactor core uncooled. As a consequence, the fuel heats up and if no emergency core cooling system is initiated, the fuel will be destroyed. Depending on the reactor type (BWR or PWR) as well as the size and the location of the break, different temperature transients evolve.

The LOCA serves as design basis transient that poses the highest challenges to the removal of the decay heat by the emergency cooling systems that inject coolant at different locations into the reactor and by means of several diverse injection systems. In order for this emergency cooling water to become effective, the reactor core must remain in a coolable geometry. The most important condition is the requirement that the fuel rods largely maintain their geometry. This means that the understanding of the fuel degradation mechanisms such as oxidation and embrittlement occurring during this design basis transient is crucial in order to determine proper acceptance limits.

2.2.2. *Fuel Behavior*

Since the reactor depressurizes as a consequence of the postulated break, the pressure outside of the fuel pin is strongly reduced, and the typically negative pressure difference across the cladding at the initiation of the accident reverses.

The cooling of the fuel pins degrades very rapidly. Correspondingly, the fuel temperature rises almost adiabatically, causing the radial temperature distribution to become flat. Fig. 1 shows a typical temperature history for a LOCA transient.

With the temperature increase, from initially 300–350 °C (normal operating temperature) to 500–600 °C, any radiation-induced cladding embrittlement due to lattice crystal defects is cured out, and precipitated hydrides in the metal phase become ductile and are dissolved (hydrides might deteriorate the mechanical cladding response later during the accident when reprecipitating under quenching conditions, see below): The cladding behaves in a ductile manner. But it loses its strength at the same time due to the temperature rise and it typically balloons between 800–900 °C because of the mentioned pressure reversal and finally may fracture in the ballooned region. With increasing temperature, the cladding oxidation (leading to ZrO_2) and the oxygen diffusion into the metal phase (leading to α-Zr(O)) rapidly increase and, as soon as the mentioned cladding balloon opens, the oxidation and oxygen diffusion into the metal phase also occur from the cladding inner surface due to the steam ingress. Therefore, at 800–900 °C the cladding reveals an outer and an inner oxide layer (ZrO_2), followed towards cladding mid-wall by an oxygen saturated metal layer (α-Zr(O)) and a central or mid-wall part of unchanged metal,

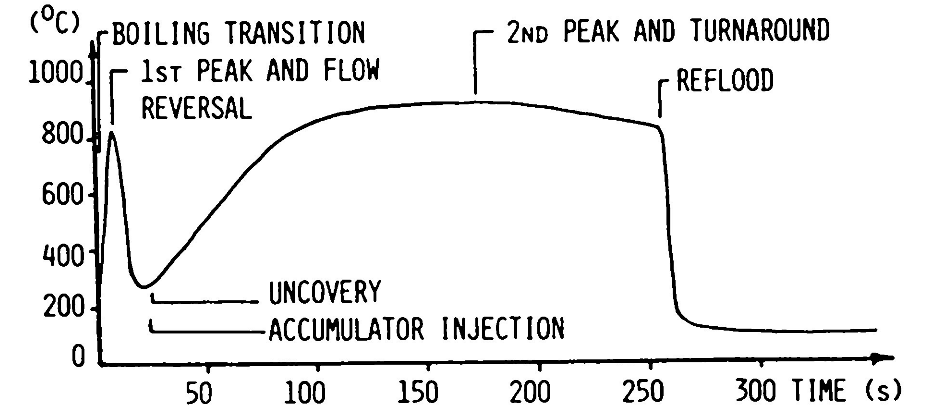

Fig. 1. Typical peak cladding temperature history of a large-break LOCA transient for a PWR, showing the first temperature peak after the boiling transition cause by flow stagnation, a reduction of the temperature after the flow reversal and a second increase after the core is uncovered, culminating in the second peak. After the turnaround, reflood sets in at a later time and the fuel is quenched. Taken from Parsons *et al.* [4].

CHIMIA 2005, 59, No. 12

more or less free of oxygen. When passing the temperature range of ~860 °C, the oxygen-free mid-wall part undergoes an allotropic transformation from the low temperature α-phase to the high temperature β-phase, while the adjacent oxygen-bearing metal regions (both towards the inner and the outer cladding wall) are hindered in their α→β transition as the oxygen content stabilizes the α-phase to significantly higher temperatures. A high temperature period follows – depending on the accident scenario – during which significantly more oxygen (high temperature oxidation) is picked up by the cladding. The high temperature phase is terminated by the activation of the emergency core cooling systems (ECCS): The injected cold water cools down the fuel slowly at the beginning and later in a rapid (quenching) process upon rewetting. At the end of the accident sequence the cladding wall shows an oxygen concentration profile as indicated in Fig. 2. We talked about the mid-wall 'oxygen-free' region which underwent the α→β phase transformation This region transforms back from β→α (the so-called prior β-phase) upon cooling. As the oxidized peripheral parts and the adjoining oxygen-saturated alpha-parts are brittle, the only load bearing part remains the prior β-phase with a low oxygen concentration (Fig. 2). Significant thermal stresses develop in the cladding material upon quenching and the cladding could fail, leading to a loss of geometry and core coolability if the cladding wall had been oxidized for too long at too high a temperature (leading to a too narrow prior β-phase) [6].

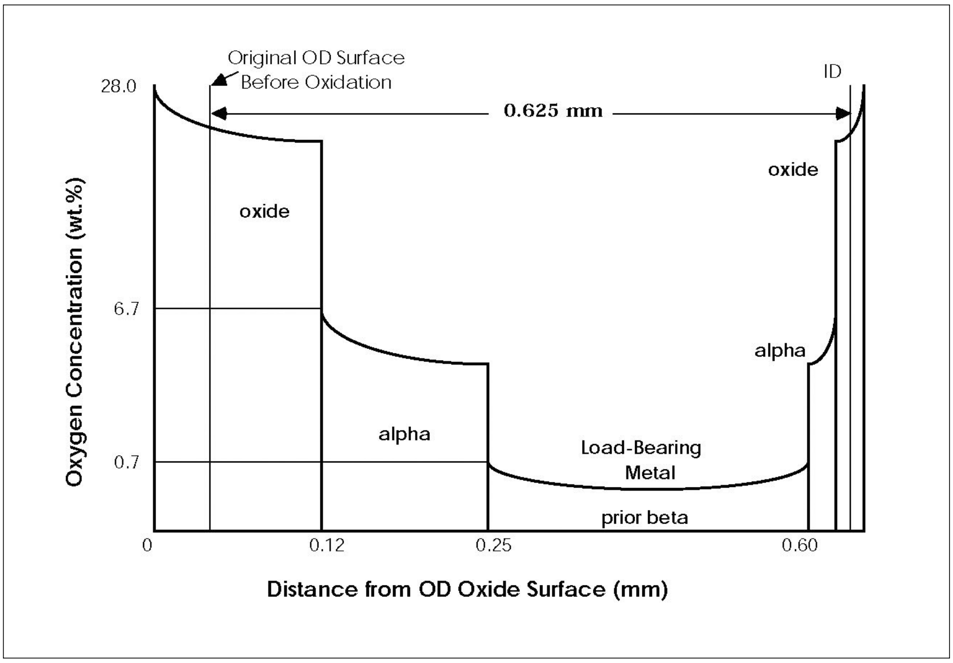

Fig. 2. Schematic illustration of oxygen distribution in oxide, stabilized α, and prior-β (transformed-β layers in Zircaloy cladding after oxidation near 1200 °C. Taken from Hache and Chung [5].

2.2.3. *Safety Significance of LOCA in Relation to Fuel Behavior*

The key safety concern in relation to LOCA resides with the capability of the fuel to maintain a coolable geometry during the LOCA transient and especially during the quench phase when the hot fuel is rapidly cooled down to low temperatures. Shattering of the cladding must be avoided. This also includes the capability to handle the damaged fuel assemblies after the assumed accident.

A second concern relates to the above-mentioned plastic deformation of cladding (ballooning) resulting possibly in the restriction of coolant flow in the sub-channels between rods if the ballooning develops in a co-planar manner.

3. Current Licensing Basis

For both the RIA- and the LOCA scenarios, acceptance limits have been defined by the regulators that ensure that the consequences to the public from such reactor transients remain below the legal limits in terms of doses.

3.1. *RIA*

Core coolability must be maintained. Therefore, excessive fuel heat-up needs to be avoided to keep the fuel pins intact. The US Nuclear Regulatory Commission (NRC) therefore adopted a limit on the (radially averaged) maximal fuel enthalpy, set to 280 cal/g, based on experiments for fuel of rather limited burn-up (see [7] for a discussion of the interpretation of the original experimental data).

For the purpose of evaluating the release of activity, cladding failure is assumed if departure of nucleate boiling (DNB) is detected for PWRs or if the secondary lower value of 170 cal/g for the fuel enthalpy is surpassed in the case of BWRs.

The regulations of most other countries follow this pattern, some of them assuming different values for the fuel enthalpy limit.

3.2. *LOCA*

Coolability of the core must be ensured during and after LOCA transients. After extensive hearings [5] the US NRC defined the following set of acceptance criteria in 1973, that later also became a model for many other countries [8]:

1) *Peak Cladding Temperature*: The calculated maximum fuel element cladding temperature shall not exceed 1204 °C.
2) *Maximum Cladding Oxidation*: The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
3) *Maximum Hydrogen Generation*: The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
4) *Coolable Geometry*: Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5) *Long-term Cooling*: After any calculated successful initial operation of the emergency core cooling system (ECCS), the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

This set of acceptance criteria was chosen based on a number of experimental facts: Through the exothermic oxidation of the cladding material, additional heat is generated that further aggravates the transient. As mentioned, the load-bearing cladding wall thins and becomes brittle by oxidation and oxygen diffusion, but also due to the reprecipitation of hydrides, which were dissolved at high temperature. The embrittlement is especially critical during fuel quenching by the injected emergency cooling water from 600–475 °C (minimum film boiling temperature) to the low temperatures around ~135 °C (saturation temperature at 0.2 MPa, the typical containment back-pressure after a LOCA accident), causing severe thermal stresses to the embrittled cladding. Furthermore, the injection of cold water into the overheated core also induces severe mechanical loads that are difficult to assess quantitatively.

With this background, the consensus was that maintaining a sufficient level of ductility best guarantees a well-performing cladding during the LOCA transient. The experimental evidence available at that time showed that limiting the peak clad temperature restricted the embrittlement to an acceptable value. The limit on the maximum allowable high-temperature oxidation (17% equivalent cladding reacted [ECR] calculated according to Baker and Just [9]) and the temperature limit of 1204 °C itself ensures a minimum thickness of the ductile (load-bearing) prior β-phase. The degree of ductility of the cladding was gauged by means of ring-compression tests [10]. In addition, criterion 4 stipulates that the (plastic) deformations of the fuel cladding need to remain within bounds in order to avoid the blockage of the space between the fuel rods, thereby inhibiting cooling. Based on many experimental findings available at that time, one main conclusion of Parsons *et al.* [4] in relation to PWR claddings (Zircaloy-4) states that co-planar blockage is not expected.

Criteria 3 and 5 relate to the system design, capable of mitigating the consequences of LOCA-transients and are not further pursued here.

It should be noted that an alternative approach has been taken in Japan where quench tests of unirradiated fuel formed the basis for the regulation. Uetsuka *et al.* [11] subjected fresh fuel samples to a simulated LOCA transient in a furnace and included ballooning and rupture, followed by high temperature two-side oxidation in steam. The experiments were terminated with fuel quenching by injecting cold water from the bottom of the test section. A boundary between failed and surviving samples was found at 15% ECR; a value which in Japan was taken as failure and coolability criterion.

4. New Challenges Due to High Burnup

The current licensing rules rely on experiments mainly derived from fuel with limited burnup, yet the burnup of both PWR and BWR cores has been increased for economic reasons and accordingly new cladding materials, less susceptible to oxidation, have been developed and are successfully used ([12]). In addition, the ^{239}Pu produced at the fuel pin periphery due to the accumulated neutron capture in ^{238}U yields peripheral peaking of the radial power profile. The fuel starts restructuring at local burnup levels between 60 and 70 GWd/t and a fine-grained high burnup structure [13] develops, beginning from the pellet surface. Therefore the need for a thorough review of the scientific basis of the licensing criteria is widely accepted.

4.1. *RIA*

Operating the cores to higher burnup leads to thicker oxide layers on the cladding surface, dependent on the cladding material (*e.g.* Zry-4, Zirlo or M5). Depending on the cladding variant and reactor operating conditions, hydrogen is picked up at a higher or lower rate and precipitates within the metal, preferably at the cooler outer clad periphery. It is important to note that not only the content of hydrogen alone but also the morphology of the hydrides, their orientation, and density determine the mechanical properties of the cladding in the case of RIA [14].

Since the degree of ductility of hydrided cladding varies in function of temperature, the susceptibility of cladding failure during RIA varies accordingly. This effect also introduces a strong dependency on the pulse width: For the short power pulses (half-width of a few ms) that are typical for many of the RIA tests, the cladding is strained (by PCMI) at comparably low temperatures; for excursions expected in power reactors with longer pulse widths (half-width >20 ms), the cladding will be strained at higher temperatures (and shows higher ductility) owing to the effect of transient heat conduction. Therefore, the experiments performed in the test facilities using short power pulses tend to be conservative in relation to cladding failure.

The accumulated fission gas can lead to grain boundary separation, and in combination with the fine-grained high burnup structure at the pellet periphery may lead to fuel fragmentation. The released fission gas pressurizes the fuel plenum and contributes to the mechanical load of the cladding, especially if a balloon develops [15].

To date, an important database of RIA experiments for high burnup fuel has been accumulated internationally. It includes ten RIA-tests with PWR UO_2-fuel and four tests with MOX-fuel in the French CABRI testing facility, covering burnup levels from 28 to 77 GWd/t and varying oxide thickness values of different modern cladding materials (Zircaloy-4, M5, and Zirlo). The pulse width was varied between ~10 and 60 ms. The coolant had a temperature of 280 °C. Another Japanese database from the NSRR reactor includes 26 tests with PWR fuel and 16 tests with BWR fuel, among them six tests using MOX fuel (in addition also fuels of experimental reactors were tested.). The coolant is stagnant water at a temperature between 20 and 85 °C. The pulse width is <10 ms. The cladding materials included in the test matrix were Zircaloy-4, Zirlo, MDA for PWRs and Zircaloy-2 for BWRs.

Investigations towards a thorough understanding of the phenomenology as well as simulations of these experiments using transient fuel behavior codes are pursued in many research and regulatory organizations.

New insights gained from the recent experiments include [16]:

- For high energy injections with correspondingly high fuel temperatures, the fuel thermal expansion is supported by fission gas induced (intra-granular) swelling, with the result of stronger PCMI. In some tests, ovalization of the fuel rods has been observed and was attributed to azimuthal heterogeneities developed during the reactor operation.
- The failures of specific UO_2-fuel pins in the CABRI test series are explained by a drastic loss of apparent ductility of the Zircaloy-4 cladding, linked to a high corrosion level with hydrogen absorption into the metal and in oxide spalling which lead to formation of large hydride concentrations or hydride lenses.
- Due to the formation of the high burnup structure at the pellet periphery, a significant amount of fission gas is stored in over-pressurized grain boundary bubbles. Upon heating, the grain boundary may separate and the gas is released, contributing to the mechanical loads of the cladding and possibly ejecting fuel fragments into the coolant.

It is generally agreed that the original burnup-independent fuel enthalpy acceptance limit for RIA must be revised; the new limit curve will include some dependence on burnup (or better on the thickness of the oxide layer). To date, a number of new limit curves have been proposed by several international regulatory bodies. Some curves simply seek to conservatively envelope the forbidden parameter space with the failed experiments, other limit curves have been developed using a sophisticated computational methodology (see *e.g.* [17]) in an attempt to extrapolate from test reactor- to power reactor conditions. As it appears, based on the limited experimental data available for MOX-fuel, separate acceptance limits need to be developed.

The magnitude and the impact of the transient fission gas behavior on clad loading during the whole transient remains a question awaiting a definitive answer. The same is true for the post-failure phenomena (fuel ejection, fuel-coolant interaction with finely fragmented solid fuel) [15] which need further studies. Also the RIA behavior of MOX-fuel must be further investigated.

4.2. *LOCA*

Modern advanced cladding materials have been developed and deployed in the reactors with a main goal of reducing the oxidation. Since the degree of mechanical degradation due to oxidation has a bearing on the fuel behavior during LOCA transients, a review of the technical basis of the existing LOCA criteria is called for.

During normal operation, loss of the load-bearing base metal thickness due to

CHIMIA 2005, 59, No. 12

oxidation and the amount of hydrogen pick-up are the primary high burnup phenomena that may affect cladding response during LOCA transients. The effective thickness and the chemistry of the prior β-phase layer following steam oxidation and quenching as well as the decreased fuel thermal permeability and the tightness of the fuel-cladding bond are important phenomena.

Consequently, the transient behavior of modern cladding during LOCA scenarios has become an important element of international research efforts. In this context, the adequacy of the testing procedures to determine the zero-ductility limit are reassessed as well.

The US NRC initiated an experimental research program with the Argonne National Laboratories (ANL) with the goal of investigating the LOCA behavior of high burnup BWR- and PWR fuel. The research includes fuel and cladding characterization, cladding high-temperature steam oxidation kinetics studies with different cladding materials (Zircaloy-2, Zircaloy-4, ZIRLO, and Zircaloy-1Nb alloys), LOCA integral experiments of fueled segments and ring-compression tests for post-quench ductility testing of such specimens as well as post-quench ductility testing of unirradiated zirconium-based cladding alloys. Supporting microstructural materials research to further elucidate the test findings is being performed as well.

Since this research program is ongoing and the new experimental evidence still waits for the definitive interpretation, only preliminary results are available yet.

Post-quench ductility data were acquired for a rather comprehensive test matix that includes [18]:

- as-fabricated 17×17 Zircaloy-4, ZIRLO and M5 oxidized to Cathcart-Pawel (CP) calculated ECR values ≤20% at 1000 °C, 1100 °C, and 1200 °C;
- prehydrided 17×17 Zircaloy-4 oxidized at 1200 °C to CP-ECR values of 7.5% and 10%;
- prehydrided 15×15 Zircaloy-4 oxidized at 1200 °C to CP-ECR values of 5% and 7.5%;
- high-burnup Zircaloy-4 (800 ppm wt. H) oxidized at ≈1200 °C to a measured ECR value of ≈5.5%;
- post-oxidation ductility results for high burnup Zircaloy-4 (550 ppm wt. H) oxidized at ≈1200 °C to measured ECR values of ≈5.5% and ≈7.5%.

The ductile-to-brittle transition of the cladding during cool-down was found dependent on the hydrogen concentration. Furthermore, the respective ECR values according to Cathcart-Pawel [19] were observed to also depend on the heating rate: The high burnup Zircaloy-4 results are found to be consistent with the results of prehydrided (un-irradiated) Zircaloy-4 cladding when the comparison is made in terms of measured ECR.

Some shortcomings of the application of the current licensing rules to high burnup fuel have been manifested [18]: The current LOCA acceptance criteria do not include the amount of hydrogen absorbed, whereas a dependence of the cladding embrittlement on hydrogen concentration has been observed. In addition, they do not differentiate between low ductility at 17% ECR accumulated at 1200 °C and ample ductility at 17% ECR accumulated at 1100 °C, again in contrast to the recent experimental findings. Lastly, the criteria do not give credit for some alloys that might embrittle at a lower ECR-value, but oxidize at a much slower rate.

Quench tests with prehydrided cladding have been performed in Japan [20], showing that the cladding loses some of its capability to withstand the loads from LOCA transients on hydriding: The rupture temperature and the circumferential burst strain reduce with increasing initial hydrogen concentration. For conditions with full axial constrain of the cladding, a reduction of the failure limit from 15% ECR to 10% ECR was observed while for partially constrained conditions similar reductions of the failure limit were found, but at higher values of ECR. Hence, the fully constrained conditions are limiting.

Which of the two basic approaches (US-NRC ring-compression tests or Japanese-JAERI quench tests) forms the best scientific basis to develop updated LOCA-acceptance criteria is currently under debate. It is clear though that further investigations are needed to advance the understanding of the failure mechanism during LOCA scenarios and develop operable criteria that include the important factors: oxidation time, oxygen-, niobium-, and hydrogen content. How the ductility of the irradiated (oxidized) cladding material is reliably measured is also under discussion.

Another concern separate from the issue of post-quench ductility of the fuel cladding is related to the LOCA-behavior of the high burnup fuel itself: It is hypothesized that fuel particles from the high burnup structure developing at the pellet periphery may axially relocate into the additional space of the cladding balloon, and the linear heat generation rate would then locally increase accordingly. This is of safety significance as the peak linear heat generation rate is one of the most important factors influencing the fuel response to LOCA transients. An international experimental research program was initiated at the Halden Test Reactor in Norway [21] in relation to such fuel relocation. The first in-pile LOCA experiment using a fuel sample pre-irradiated in a power reactor to a burnup of ~82 GWd/t was performed. After the coolant was evacuated from the test rig, the tested fuel rod was heated up with a gradient of ~5 K/s until the cladding surface temperature reached 800 °C. This temperature was kept for a duration of 300 s. Thereafter, a slow cooling was initiated in order to avoid any mechanical shocks (that would occur if the fuel would have been quenched) acting on the tested fuel rod and possibly causing a further reconfiguration of the (fragmented) fuel inside the cladding. The results of this interesting experiment are not yet available.

5. Analysis Tools

Advanced fuel behavior computer codes are used to fully exploit the experimental database. They are used for the design and even more for the interpretation of the experiments because only a limited set of measured parameters can be obtained from RIA tests due to the harsh environment characterized by the high levels of radiation and temperatures.

Basic features of such codes include the thermal-mechanical description of the fuel pin in typically axisymmetric geometry. First, the heat transfer problem is addressed, based on the solution of the transient heat conduction equation and a simple model for the coolant flow in the subchannel adjacent to the fuel rod that determines the heat transfer from the cladding surface to the coolant. It takes into account the thermal properties of the different materials that constitute the fuel pin, including special models for the gap between the fuel pellet and the cladding. Second, the mechanical deformation of the fuel pin in response to changes of the thermal conditions as well as to changing boundary conditions (*e.g.* reactor pressure) are determined. The mechanical models describe the interaction between the stack of fuel pellets and the cladding due to thermal expansion and consider the detailed shape of the pellets (dishing, champfer) in order to obtain an accurate estimate of the stress distribution. Special models describe the friction between the pellet and the cladding, the oxidation process of the cladding and the transport of the fission gas from the fuel matrix to the fuel gap. In some codes, the interaction between the behavior of the fission gas bubbles and the stress field is explicitly taken into account, thereby further increasing the computational difficulty due to the non-linear coupling between the different phenomena.

Examples of transient fuel behavior codes that are available to nuclear regulatory and research organizations are FALCON [22], FEMAXI [23], FRAPTRAN [24], and SCANAIR [25], the latter strongly applied in the framework of the French CABRI-program.

CHIMIA 2005, 59, No. 12

At the Paul Scherrer Institut, investigations in relation to the transient behavior of high burnup fuel focus on the application of FALCON for RIA and LOCA as well as SCANAIR for RIA with the perspective of providing technical support to the Swiss Federal Nuclear Safety Inspectorate (HSK). Aside from code validation, an interpretation of selected CABRI-RIA experiments is aimed at: Through the comparison of results obtained with FALCON (no model for transient fission gas release) and SCANAIR (with model for transient fission gas release), partial answers were sought on the issue of the impact of transient fission gas release on the cladding deformations (Fig. 3). A possible significant role of transient fission gas release was found but for high-energy injection transients [26][27].

6. Summary

For RIA-transients, an experimental data base covering a large parameter space exists for PWR UO_2-fuel mainly with Zircaloy-4 clad. For Western reactor applications, the experiments in the CABRI- and the NSRR-facilities are most relevant. For MOX fuel, only limited RIA data are available. In the framework of the international CABRI-Waterloop program, RIA testing of high burnup PWR fuel will continue with coolant conditions representative of power reactors. For BWR fuel, a reasonable set of experiments including high burnup fuel is available from the NSRR facility, and more experiments are planned for the near future in the framework of the international ALPS-program.

General consensus has been reached on the testing method, based on a sound understanding of the interplay of the key physical processes. Corresponding acceptance criteria are being developed by the different national nuclear regulators, and currently a set of fuel failure limit curves has been proposed and is implemented in some countries.

In the case of LOCA-transients, more progress of the on-going research programs is needed in order to arrive at new acceptance limits well based on new experimental facts obtained from the testing of high burnup fuels.

In relation to the fuel behavior codes, model updates might become necessary depending on the results from the RIA- and LOCA-experiments.

Acknowledgement

Antonino Romano and Hannu Wallin are acknowledged for providing Fig. 3. This work was partly supported by the Federal Office of Energy, represented by the Swiss Federal Nuclear Safety Inspectorate (HSK).

Received: September 8, 2005

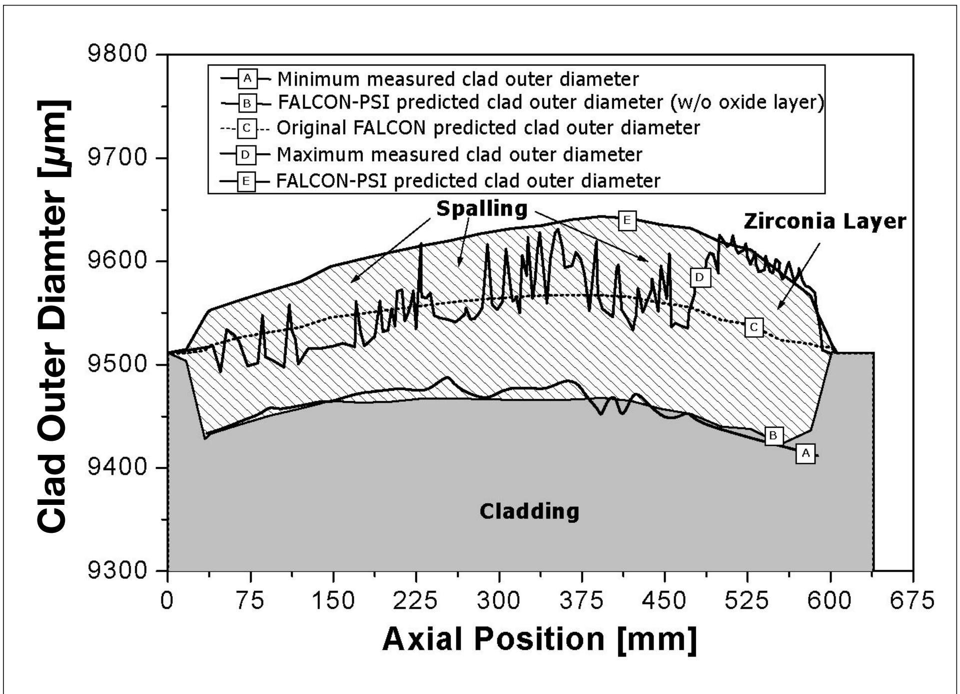

Fig. 3. Analysis of the CABRI RIA-experiment CIP0-1C using two versions of the transient fuel behavior code FALCON. It can be seen that the outer diameter of the Zirconia layer fits very well the measurements for the regions without spalling. The analysis using the FALCON-PSI version includes the effect of the stresses imposed by the oxide layer. By comparison with the results obtained with the original FALCON code, it is inferred that the mechanical effect of the cladding is small.

[1] G. Yadigaroglu, *Chimia* **2005**, *59*, 877
[2] T. Sugiyama, T. Fuketa, K. Ishijima, 7th International Conference on Nuclear Engineering, Tokio, Japan, April 19–23, **1999**, paper # 7070.
[3] J. Papin, M. Balourdet, F. Lemoine, F. Lamare, J. M. Frizonnet, F. Schmitz, 'French Studies on High-Burn up Fuel Transient Behavior Under RIA Conditions', *Nuclear Safety* **1996**, *37*, 289.
[4] P.D. Parsons, E.D. Hindle, C.A. Mann, 'The Deformation, Oxidation and Embrittlement of PWR Fuel Cladding in a Loss-of-Coolant Accident, State-of-the-Art Report', NEA, CSNI Report 129, December **1968**.
[5] G. Hache, H.M. Chung, 'The History of LOCA Embrittlement Criteria', in 'Phenomenon Identification and Ranking Tables (PIRTs) for Loss-of-Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Burnup Fuel', NUREG/CR-6744, December **2001**, Appendix I, I–97ff.
[6] S. Leistikow, G. Schanz, H. v. Berg, 'Kinetik und Morphologie der isothermen Dampf-Oxidation von Zircaloy 4 bei 700–1300 °C', KERNFORSCHUNGSZENTRUM KARLSRUHE, Institut für Material- und Festkörperforschung, Projekt Nukleare Sicherheit, KfK 2587 (**1978**).
[7] P.E. MacDonald, S.L. Seiffert, Z.R. Martinson, R.K. McCardell, D.E. Owen, S.K. Fukuda, 'Assessment of Light-Water-Reactor Fuel Damage During a Reactivity-Initiated Accident', *Nuclear Safety* **1980**, *21*, 582.
[8] 'Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors', U.S. Code of Federal Regulations, Title 10, Part 50, Section 46, Revision of Nov. 3, **1997**.
[9] L. Baker, L.C. Just, 'Studies of Metal-Water Reactions at High Temperatures – III. Experimental and Theoretical Studies of the Zirconium-Water Reaction', ANL-6548, May **1962**.
[10] D.O. Hobson, 'Ductile-brittle behavior of Zircaloy fuel cladding', Proc. ANS Topical Mtg. on Water Reactor Safety, Salt Lake City, 26 March, **1973**.
[11] H. Uetsuka, T. Furata, S. Kawasaki, 'Failure-Bearing capability of oxidized Zircaloy-4 cladding under simulated loss-of-coolant accident condition', *J. Nucl. Sci. Techn.* **1984**, *20*, 941.
[12] G. Bart, J. Bertsch, *Chimia* **2005**, *59*, 938.
[13] H. Matzke, M. Kinoshita, 'Polygonization and high burnup structure in nuclear fuels', *J. Nucl. Mater.* **1997**, *247*, 108.
[14] H.M. Chung, T.F. Kassner, 'Cladding metallurgy and fracture behavior during reactivity-initiated accidents at high burn up', *Nuclear Engineering and Design* **1998**, *186*, 411.
[15] T. Fuketa, H. Sasajima, T. Sugiyama, 'Behavior of High-Burn up PWR Fuels with Low-Tin Zircaloy-4 Cladding under Reactivity-Initiated-Accident Conditions', *Nuclear Technology* **2001**, *133*, 50.

[16] J. Papin, B. Cazalis, J.-M. Frizonnet, E. Federici, F. Lemoine, 'Synthesis of CABRI-RIA tests interpretation', Eurosafe **2003**, Paris, France, 25–26 November 2003.
[17] Topical Report on Reactivity Initiated Accident: Basis for RIA Fuel Rod Failure and Core Coolability Criteria, EPRI, Palo Alto, CA: **2002**. 1002865.
[18] M.C. Billone, 'LOCA Embrittlement Correlation', Argonne National Laboratory, Argonne, IL 60439, US NRC publication on ADAMS, ML051010265, April 8, **2005**.
[19] R.E. Pawel, J.V. Cathcart, J.J.Campbell, 'The oxidation of Zircaloy-4 at 900 and 1100 °C in high pressure steam', *J. Nucl. Mater.* **1979**, *82*, 129.
[20] F. Nagase, T. Fuketa, 'Effect of Pre-Hydriding on Thermal Shock Resistance of Zircaloy-4 Cladding under Simulated Loss-of-Coolant Accident Conditions', *J. Nucl. Sci. Technol.* **2004**, *41*, 723; F. Nagase, T. Fuketa, 'Behavior of Pre-hydrided Zircaloy-4 Cladding under Simulated LOCA Conditions', *J. Nucl. Sci. Technol.* **2005**, *42*, 209.
[21] E. Kolstad, W. Wiesenack, V. Grismanovs, 'LOCA testing at Halden: Second in-pile test in IFA-650.2', Halden, 2004 (IFE/HR/E-2004/027); Nuclear Safety Research Conference, Washington, DC, October 25–27, **2004**.
[22] 'FALCON MOD01: Fuel Analysis and Licensing Code – New, Volume 1: Theoretical and Numerical Bases', EPRI, Palo Alto, CA, and ANATECH Corp., San Diego, CA, **2004**. 1008109.
[23] M. Suzuki, H. Uetsuka, H. Saitou, 'Analysis of mechanical load on cladding induced by fuel swelling during power ramp in high burn-up rod by fuel performance code FEMAXI-6', *Nuclear Engineering and Design* **2004**, *229*, 1.
[24] M.E. Cunningham, C.E. Beyer, P.G. Medvedev, G.A. Berna, 'FRAPTRAN: A Computer Code for the Transient Analysis of Oxide Fuel Rods', **2001**, NUREG/CR-6739, Vol. 1 + 2, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001.
[25] E. Federici, F. Lamare, V. Bessiron, J. Papin, 'The SCANAIR code version 3.2: main features and status of qualification', IAEA TCM on fuel behavior under transient and LOCA conditions, Halden, Norway, 10–14 September **2001,** IAEA-TEC-DOC-1320, pp. 88–101.
[26] A. Romano, H. Wallin, M.A. Zimmermann, 'Modelling Reactivity Initiated Accident Experiments with FALCON and SCANAIR: A Comparison', PSI Scientific Report 2004, vol. 4 **(2005).**
[27] A. Romano, M.A. Zimmermann, 'Effect of the Power Pulse Shape on the Fuel Thermal-Mechanical Response to RIAs: a Study Using FALCON and SCANAIR', 2005 LWR Fuel Performance Meeting, Paper # 1034 (**2005**).