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ENSURING LONG-TERM NUCLEAR FUEL SAFETY

- Stakes associated with nuclear fuel
- Consequences on safety
- Pending safety issues
- Fuel behaviour in reactor core
- Fuel cycle back end

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TO OUR READERS

Lothar Hahn and Jacques Repussard

As nuclear power programmes tend to resume in several countries after a two-decade stand-by situation and 'new' countries envisage including nuclear power in their energy mix, the question of fissile material availability at an attractive price is increasingly becoming a matter of concern for utilities who compete to expand their business on open, deregulated markets.

For many years, the characteristics of nuclear fuels and their in-core management by the operators have kept changing with a view to getting as much energy as possible out of the fuel assemblies. Moreover, new generations of reactors are either under construction, such as the EPR, or under feasibility study, such as the Generation-IV reactors, to optimise further the use of nuclear fuels. In this context, European Technical Safety Organisations (TSOs) are involved in supporting the development of safety organisations and honing corresponding competences.

These significant developments compound to challenge the TSOs' capability to establish, as a result of their research and assessment activities, that nuclear facilities are – and will remain – operated with satisfactory safety margins at each step of the nuclear fuel cycle.

The present issue of the EUROSAFE Tribune addresses successively the economic stakes associated with nuclear fuels as well as the safety issues linked to a closed fuel cycle, the different operating conditions and corresponding safety priorities, the achievements in the research and assessment of current and future fuels. It also explains some still pending questions, such as the behaviour of fuel assemblies in the reactor core, the relevance of the present safety criteria in the context of new developments and, last but not least, the validation of nuclide inventory calculations of spent fuel for transport, reprocessing or final disposal.

We hope the articles that follow will help you form your own opinions on these issues and wish you pleasant reading. ●

Lothar Hahn and Jacques Repussard

STAKES ASSOCIATED WITH NUCLEAR FUEL

The challenges of sustainable nuclear power

Rauno Rintamaa (VTT) I Giovanni Bruna (IRSN)

It is widely recognised that the operation of the world's largest Generation-II (Gen-II) NPP fleet beyond its originally intended lifetime could contribute to the competitiveness and secure supply of electricity in the EU as well as to achieving the Member States' commitments to reduce their greenhouse gas emissions. To take further advantage of this favourable situation whilst enhancing further the safety of nuclear facilities, considerable R&D has to be performed jointly by the different stakeholders in order to make nuclear energy sustainable over the long run.

Fuel for FBR BN-600

In 2008, about 440 NPPs were connected to the grid across the world, with a total capacity of 372 GWe, accounting for 16% of the total electricity production. Light water reactors represent about 90% of the fleet in operation with 300 pressurised water reactors and 100 boiling water reactors. About 1/3 of the world's fleet is located in the EU, where nuclear power accounts for nearly 31% of electricity generation and represents a non-emission of almost 900 million tonnes of carbon dioxide each year.

According to the EC, *"there are economic benefits in maintaining and developing the technological lead of the EU"* in the field of nuclear fission. The Council endorsed this communication in March 2007, which also committed the EU to meet ambitious objectives by 2020, i.e. a 20% reduction in greenhouse gas emissions (compared to 1990), 20% renewable energies in the energy mix, and a 20% reduction in energy consumption through energy savings.

A new generation of fission reactors for increased sustainability

Nuclear fission is cited together with other low carbon technologies such as renewable energies and carbon capture and storage (CCS) technology as one of the contributors to meet the 2020 challenges. By maintaining *"competitiveness in fission technologies, together with long-term waste management solutions"*, fission energy will continue to be the leading low carbon energy technology in Europe. Projections published in the World Energy Technology Outlook

(WETO) report indicate that by 2030, nuclear energy will continue to produce more than half of the electricity generated by non-fossil fuel-based technologies. Beyond the 2020 objectives, the *Strategic Energy Technology* (SET) plan also identifies fission energy as a contributor to the 2050 objectives of a low-carbon energy mix, relying on a new generation of reactors and associated fuel cycles. This objective is to be achieved by acting now to *"complete the preparations for the demonstration of a new generation of fission reactors for increased sustainability"*.

Taking up the safety challenges of future nuclear technology: European TSOs join SNETP

Alongside the improvement of system, structure and component (SSC) design, the upgrading of the man-system interface and the simplification of the reactor systems, advanced fuel and NPP availability rate is mentioned in the SET Plan as one of the 2020 objectives set to the operating experience feedback (OEF) of the first Generation-III (Gen-III) reactors, such as the European Pressurised Water Reactor (EPR) and of future R&D.

From a Technical Safety Organisation (TSO) perspective, preparing for the corresponding technological evolution whilst ensuring the safety of the NPP fleet in operation, in a context of lifetime extension, represents significant challenges, notably in the field of nuclear fuels, with a view to assessing the options aimed to optimise the use of fissile material and minimise the volumes and activity of generated radioactive waste. Taking up such challenges lead several European TSOs to join the Sustainable Nuclear Energy Technology Platform (SNETP), a group of 70 European stakeholders from industry, research and academia, technical safety organisations, non-governmental organisations and national representatives set up in 2007 to promote research, development and demonstration of the nuclear fission technologies necessary to achieve the SET plan goals, i.e.:

- *For the year 2020*: maintain competitiveness in fission technology, and provide long-term waste management solutions;
- *For the year 2050*: complete the demonstration of Generation-IV (Gen-IV) fission reactors with increased sustainability, and enlarge nuclear fission applications beyond electricity production.

Two conditions for a sustainable fuel cycle

Drawing upon the results of many studies carried out worldwide and particularly in the EU, there is a clear consensus today that sustainable nuclear power generation relies upon a sustainable fuel cycle that requires progress in two domains:

- *The optimisation of the use of fissile material* through the design of reactor cores with high conversion ratio and of corresponding high burnup fuels as well as the recycling of plutonium and reprocessed uranium;
- *The minimisation of the volumes and activity of generated nuclear waste* through the development of partitioning and transmutation of transuranic elements present in the spent fuel as well as through the enhancement of the containment properties of the waste matrix or waste container.

Optimising the use of fissile material: "The waste of today is the fuel of tomorrow"

According to the SNETP's Strategic Research Agenda[1], current reactors *"use less than 1% of the uranium (U) available in nature. With such a low efficiency, the economically extractable*

[1] See the SNETP Strategic Research Agenda, May 2009. To be downloaded at: www.snetp.eu

Fuel pellet inspection

uranium resources worldwide will be sufficient for only about 100 years depending inter alia on the nuclear power growth rate in the next decades. In order to get a long-term sustainability with nuclear energy from fission, new technological solutions improving the usage of this natural resource by up to 100 times are being developed."

➜ Reactor cores with high conversion ratio

The conversion ratio is the ratio between the total amount of artificial fissile material created inside the reactor core and the total amount of fissile isotopes "consumed". Part of the created artificial fissile isotopes, which are not burned in the reactor core, remains in the spent fuel. The recycling of this part can further contribute to saving the natural fissile isotopes.

The SNETP's Strategic Research Agenda (SRA) underlines that one of the most efficient routes to reduce natural uranium consumption is to increase the conversion ratio of present and future reactors and to recycle fissile material: *"Fast nuclear reactors can be designed to reach a conversion ratio equal or even greater than one, in such a way that no more natural fissile isotope is needed to sustain nuclear energy since the reactors generate more fissile isotopes than they consume to produce energy. These reactors, called "breeders", need to be fed only with fertile isotopes (^{238}U or even ^{232}Th), which are available in huge amounts. Therefore, it must be underlined that "breeder" reactors, in practice fast neutron reactors (FNRs), are the only solution that can lead to the long-term sustainable development of nuclear energy with regard to the "optimum use of natural resources".*

However, the industrial deployment of such technology remains a remote prospect and the use of natural resources can be enhanced earlier by increasing the conversion ratio of LWRs, and by improving their fuel design and the associated back end of the nuclear fuel cycle. Therefore, short-term R&D should concentrate on advanced Gen-III LWRs with high conversion ratios through neutronic, thermal-hydraulic, mechanic, safety, instrumentation and control, and economic assessments. In the medium term, core and component tests on experimental loops and critical mock-ups towards the design of an experimental high conversion ratio reactor should be addressed.

➜ High burnup fuels

If extended irradiation in the reactor core allows extracting more energy per unit of fuel, higher fuel burnups do not lead, *per se*, to a reduction of natural uranium consumption, as this requires higher fissile enrichment. Nevertheless, a higher burnup allows the improvement of other parameters of the in-core fuel management, such as an increase of the reload fraction of the core, resulting in a net reduction of natural uranium consumption. According to the SRA, *"R&D in the short term should concentrate on: feasibility studies of fuels able to reach very high burnups (100 GWd/tHM) for LWR and possibly high temperature reactors (HTR). In the medium term; irradiation and qualifications tests of these fuels must be performed."*

➜ Recycling of plutonium and reprocessed uranium

In some European countries, spent fuel reprocessing and plutonium recycling in LWRs are implemented, at industrial scale, in the form of mixed oxide (MOX) fuel. However, the use of reprocessed material is limited to a single recycling process, with a 12% plutonium concentration and only a fraction of the core being loaded with MOX.

The SRA suggests that: *"R&D in the short term concentrate on studies per-*

taining to 100% MOX cores, to plutonium multi-recycling for LWR and to 100% plutonium cores for HTR". In parallel, studies should be performed on: *"scenarios of nuclear materials management issues at the European level and the evolution of nuclear reactor fleet, including uranium and plutonium availability in the case of delayed deployment of fast neutron reactors."*

Though not a R&D priority, thorium could become an attractive option in the long term, and a minimum level of basic studies on this cycle should be maintained at European level.

Radioactive waste minimisation: relieving the burden on future generations

Radioactive waste is commonly divided into three categories – low-level waste (LLW), intermediate-level waste (ILW) and high-level waste (HLW) – depending on its activity and content of long-lived radionuclides. Specific packaging and deep underground geological disposal are considered as appropriate barriers to contain the radioactivity of long-lived waste and to avoid any hazard to the population or the biosphere.

➜ Minimisation of the volumes and activity of generated nuclear waste through the development of partitioning and transmutation (P&T) of transuranic elements present in the spent fuel

The SNETP considers that there are two complementary ways to minimise the volume and/or mass as well as the radiotoxic inventory of radioactive waste.

A first way is to reduce the amounts of radionuclides produced by nuclear reactors. For fission products, the production is directly proportional to the electricity generation, so that the only way to reduce their quantity is to increase the electrical efficiency of nuclear power reactors. Moreover, there are several means to decrease the production of the different actinides (americium, neptunium, curium…) including the choice of reactor types or even the choice of a fuel cycle.

A second way is, according to the SRA, to pursue R&D on advanced reprocessing of LWR fuels, i.e. partitioning and transmutation *"with a view to separating ("partitioning") from the spent fuel the transuranic elements (plutonium and minor actinides), which are responsible for the highest heat loads and radioactive inventory in the long term. The next step is to burn or "transmute" these minor actinides using special fast neutron reactors (FNR) or subcritical accelerator-driven systems (ADS) loaded with homogeneous fuels with high minor actinide content"*.

➜ Enhancement of the containment properties of the waste matrix or waste container

The R&D, technology development and implementation pertaining to radioactive waste conditioning and disposal are the topics of a dedicated technological platform referred to as Implementing Geological Disposal of Radioactive Waste (IGDTP)[2]. The membership of this platform, launched in Brussels on November 12, 2009, includes organisations from different EU Member States in charge of radioactive waste management.

Priority actions to be undertaken

From a TSO perspective, a sustainable nuclear fuel cycle is an ambitious goal that implies a considerable research effort over the short, medium and long term in order to be able to assess in due time the safety of the future reactor and fuel designs and operations. The corresponding topics are addressed in the next pages of this issue of the EUROSAFE Tribune. ■

[2] See the IGDTP website at: www.igdtp.eu

CONSEQUENCES ON SAFETY

Consequences of fuel behaviour on nuclear safety

Nadine Hollasky (Bel V) I Jean-Pierre Van Dorsselaere (IRSN)

The safety issues associated with the main types of fuel used in Generation II or III PWRs are changing, as illustrated by the respective situations in Belgium and France. The safety criteria may also have to be adjusted to changes in the fuel design. The general characteristics of each type of fuel used in Belgium and France are reviewed below, along with the different fuel management practices in both countries. A summary of the general safety approach is then provided, as well as the safety issues for the different plant category conditions that cover the design basis domain. Finally, the fuel behaviour in the case of severe (beyond design basis) accidents is addressed.

Nuclear fuel characteristics in Belgian and French PWRs

The 7 Belgian and 58 French PWRs use either ^{235}U enriched uranium oxide fuel (UO_2) or UO_2-PuO_2 mixed oxide (MOX) fuel. The main characteristics of the cores of Belgian and French NPPs are summarised in the Tables 1 and 2, respectively.

In-core fuel management

National safety regulators authorise the operation of the NPPs within specified limits in terms of power level, maximum enrichment, maximum average assembly discharge burnup and, in Belgium, maximum irradiation time. These limits are of course correlated between them.

Over the past 20 years, EDF has been adapting its nuclear power production plan in France to power market needs and to ensure fuel cycle consistency. This resulted in different changes in fuel management strategies. At the moment, the enrichment

Unit	Assembly Type	Active length (ft)	Cycle length (months)	Fuel Type	Gadolinium Number of rods / (Concentration)
Doel 1	14x14	8	12	EU (4.5%)	No
Doel 2	14x14	8	12	EU (4.5%)	No
Tihange 1	15x15	12	18	EU (4.35%)	8-12-16 rods (8-10%)
Doel 3	17x17	12	12	EU (4.15% MOX)	12 rods (8%)
Tihange 2	17x17	12	18	EU (4.6%)	8-12-20 rods (10%)
Doel 4	17x17	14	16→18	EU (4.25%)	8-12-16 rods (8%)
Tihange 3	17x17	14	18	EU (4.35%)	8-12-16 rods (8%)

Table 1 Main characteristics of Belgian PWR cores (EU = Enriched Uranium)

Unit class (number of units)	Assembly Type	Active length (ft)	Cycle length (months)	Fuel Type	Gadolinium
900 MWe (6)	17x17	12	18	EU (4.2%)	12 rods (8%) x 28 assemblies
900 MWe (28)	17x17	12	12	EU (3,7% or MOX)	No
1300 MWe (20)	17x17	14	18	EU (4%)	12 rods (8%) x 24 assemblies
1450 MWe (4)	17x17	14	17	EU (4%)	12 rods (8%) x 40 assemblies

Table 2 Main characteristics of French PWR cores (EU = Enriched Uranium)

of fresh reloads varies between 3.7% and 4.2% (see Table 2), depending on reactor types, and the maximum average assembly discharge burnup for UO_2 fuel allowed since 1999 is 52 GWd/tU for each assembly. A new fuel management has recently been allowed in France for 1300 MWe reactors, increasing these limits to a 4.5% enrichment and to 62 GWd/tU for UO_2 fuel with the M5 clad material, and with cycle lengths of up to 400 days. It is to be noted that fuel cycle plants are able to deal with enrichments up to at least 4.95%, which could enable the French utility to further increase burnup rates or cycle lengths if it fits with its economic strategy and if the safety of such fuel reloads can be established.

In Belgium, the maximum enrichment for UO_2 fuel authorised by law is 5%, this value coming from the fabrication and the reprocessing facilities. Today, the maximum enrichment for the fresh reloads amounts to 4.6%. The maximum allowed burnup is 55 GWd/t for UO_2 fuel and 50 GWd/t for MOX fuel.

The cycles can be either annual (12 months) or extended (18 months) in both countries. For long cycles (18 months), burnable poisons (gadolinium) have been introduced.

As regards reloading patterns, as many irradiated assemblies as possible are placed in the periphery of the core in order to limit vessel fluence, to improve the neutron economy and also to increase slightly the cycle length thanks to low neutronic leak patterns.

For the MOX fuel, the recycle rate is 23% (i.e. 23% MOX assemblies in the core) in Belgium and up to 30% in France. Therefore the MOX assemblies are put at the periphery of the core for their last cycle in order to ensure the good thermal-mechanical behaviour of the rods.

A safety approach divided into plant category conditions

The primary function of a fuel rod is to generate and to transfer heat to the reactor coolant. In this process both radioactive and stable fission products (FP) are produced in the fuel. To avoid or to limit the FP release in the reactor environment, three barriers are set up between fuel and environment, the first being the fuel clad. The structural integrity of the fuel rod must be maintained.

The safety demonstration is based on the analysis of initiating events that are classified into "plant category conditions" or "classes" (see box on p. 10). Reference transients, incidents and accidents are categorised according to the estimated frequencies of the classes of initiating events they cover. For each class, a list of initiating events and criteria are specified.

Four classes to cover the design basis domain

- **Class-1**: Normal plant operation.
- **Class-2**: Transients with moderate frequency.
- **Class-3:** Incidents potentially resulting from low frequency of initiating events, typically between 10^{-2} and 10^{-4} per reactor and per year (examples: uncontrolled single control-rod withdrawal, small break loss of coolant accident (LOCA), steam generator tube rupture).
- **Class-4**: Hypothetical accidents, with frequency of initiating events typically between 10^{-4} and 10^{-6} per reactor and per year (e.g. rod ejection accident, intermediate or large break LOCAs, main steam line break).

Safety issues for class-1 and -2 conditions: maintaining fuel rod integrity

The purpose of the fuel rod thermal-mechanical design is to ensure its integrity throughout the projected lifetime of the fuel. This is basically achieved by designing the fuel rod to satisfy a variety of conservative design criteria during both class-1 and class-2 operations. For each design criterion, the performance of the limiting fuel rod, with appropriate allowance for uncertainties, must not exceed the design limit. Fuel rod design criteria are related to, *inter alia*, fuel temperature, rod internal pressure, clad stress, strain and fatigue, growth, corrosion, clad temperatures. Analyses are performed by using a qualified thermal-mechanical performance code and are also based on the interpretation of tests (see for example the Halden programme on page 24).

The characteristics of the MOX fuel rod being the same as those of the UO_2 fuel rod, the same design criteria may be taken into account, i.e. mainly the temperature at the centre of the pellet, the clad temperature, the internal pressure, the stress level, the strain variation during a transient and the clad damage. Nevertheless, the evolution of irradiation-induced phenomena, such as spring relaxation and growth, is more important, the flux being faster.

All the criteria must be verified for the most penalising scenarios to provide for sufficient mechanical rod integrity under normal operating conditions as well as class-2 incidental conditions. Particular attention must be given to the *pellet-cladding mechanical interaction* (PCMI) phenomenon. The approach to prevent PCMI problems, and particularly to cover unknown phenomena in this process, is similar in Belgium and in France, even if some details may differ.

Two main criteria must be complied with to provide for fuel rod integrity at operating modes associated with class 1 and class 2 events:

- The maximum fuel temperature shall be less than the melting temperature of UO_2, thereby preventing expansion during the phase change which might rupture the fuel clad;
- The criterion on the minimum departure from nucleate boiling ratio (DNBR) must be met in order to prevent excessive clad temperature due to the degradation of heat transfer from fuel to coolant.

The melting temperature of fresh UO_2 is a given value. To preclude centre melting, a lower centreline fuel temperature limit is calculated, corresponding to a given linear heat generation rate. This fuel melting temperature reduction takes into account e.g. the decrease of the melting temperature with burnup, the fabrication tolerances, the model uncertainties, etc. An additional safety margin has to be applied on this temperature value.

Safety issues for class-3 and -4 conditions: the pivotal role of the reactor vessel's leaktightness

For class-3 accidents, the safety principles require that fuel rods retain their core-wide integrity except for localised and limited failure so that the amount of radioactivity released to the environment remains very low.

For class-4 accidents, the safety principles require that the geometry of the core remains amenable to cooling and that the containment retains its function, in order for radioactive releases to remain as low as reasonably achievable.

The behaviour and performances of the fuel are a central issue in rod cluster ejection accidents and in intermediate or large break LOCAs, which are both class-4 accidents. The safety requirements concerning the fuel behaviour are defined to ensure that the geometry of the core remains coolable and that decay heat can be safely extracted without challenging the containment.

During a rod ejection, the reactivity insertion leads to a quick power excursion with well-localised power disturbance. In order to ensure the coolability of the core during such an accident, designers usually decide to demonstrate, among other safety requirements, that:

- Only a limited number of fuel rods are submitted to boiling crisis,
- only a limited percentage of the fuel pellet is molten at the core hot spot,
- there is no clad embrittlement,
- there is no dispersion of fuel in the coolant.

In order to facilitate analysing every possible situation, decoupling criteria have been defined, usually by selecting umbrella values on the basis of experimental evidence. For instance, the safety analyses must establish that the deposited enthalpy remains below the limit that would cause dispersion of fuel in the coolant. Following American SPERT experiments, a limit of 225 cal/g for fresh fuel and 200 cal/g for low burnup fuel (< 33 GWd/tU) has been defined in France, the same limits being also applied in Belgium. However, as shown by more recent experiments such as CABRI tests, clad rupture can occur at lower enthalpy levels for highly irradiated fuel. These new facts persuaded most countries operating nuclear fleets across the world of the need for new criteria applicable to rod ejection accidents.

In the case of the large-break LOCA (also a class-4 accident with consequences highly dependent on fuel performance), several safety requirements and corresponding decoupling criteria are defined to which the *Engineered Safety Features* – including the emergency core cooling system (ECCS) – of LWRs must be designed. Criteria in use in many countries today are still based on the *10 CFR 50.46* issued by the NRC in 1973 that specifies, among others, that the "calculated maximum fuel element clad temperature shall not exceed 2200 °F (1204 °C)" and that the "calculated total oxidation of the clad shall nowhere exceed 0.17 times the total clad thickness before oxidation". Since 1973, research programmes have led to knew findings not taken into account in the original *10 CFR 50.46* criteria, for instance that clad ductility decreases as burnup and hydrogen pickup increase. How these new findings might be taken into account in future rulemaking is still under discussion in several countries.

Safety issues for severe accidents: limiting fission product release

Less frequent than class-4 accidents, but with more potential consequences, severe accidents are considered as "beyond design basis accidents" for Generation-II reactors, whereas they are considered in the design of the EPR. For such accidents, the unavailability of several plant safety systems is assumed; the fuel rods may lose their mechanical integrity and the core may no longer be cooled down. The main safety issues related to fuel behaviour concern its loss of mechanical integrity that will impact on the progression of core degradation, and


BWR fuel element

Centrifuges

the subsequent FP release, an essential parameter for the source term in the containment.

The loss of the original core geometry can occur gradually over a period of minutes to hours, covering a wide range of temperatures from 1000 °K to 3000 °K. Above the melting temperature of the Zircaloy clad (2000 °K), the UO_2 pellets may be dissolved by molten Zircaloy due to the formation of a thick protective oxide layer (ZrO_2) on the outer surface of the clad. It would impact on FP release: enhanced release due to the dissolution of the crystalline matrix and the migration of related gas bubbles through the bulk melt phase; or trap of large gas bubbles in the liquid phase, producing a foaming-like structure in which bubbles are stable. If the fuel has a sufficient level of burnup and the primary pressure is low enough, it can swell, causing additional reductions in the flow area, as initial porosity of the fuel increases.

The chemical interactions induce the liquefaction of UO_2 and ZrO_2 at about 1000 °K below their melting points and the formation of U-O-Zr mixtures. After failure of the ZrO_2 layer above 2300 °K, the ceramic melts will relocate to cooler regions of the core until they freeze, resulting in the formation of large blockages. These blockages can then trap molten materials that form subsequently at higher levels in the core. Bundle experiments with pre-irradiated fuel such as PHEBUS FP experiments in oxidising conditions showed fuel collapse temperatures around 2500-2600 °K quickly followed by a molten pool formation.

If water is injected in the vessel for severe accident management, the core materials that have absorbed a sufficient amount of oxygen to become brittle will fragment, in particular the fuel pellets. At temperatures below 1500 °K, the fragmentation of fuel rod materials has been relatively well characterised due to the research on clad embrittlement under DBA conditions. Above 1500 °K, the TMI-2 accident showed the formation of a debris bed at the time of reflood. The loss of integrity of fuel rods may also enhance the release of the less volatile FP. Many uncertainties on the possibility to efficiently cool the core remain due to the complex and heterogeneous state of the core: molten melts that have refrozen (with possible cracking), solid debris bed, and relatively intact fuel pellets.

From separate-effect tests on chemical interactions through to out-of-pile tests with electrical heating of bundles (such as CORA at FZK) and in-pile tests with nuclear heating of a bundle (such as LOFT-FP at INL/INEL or PHEBUS FP at IRSN), many experiments have studied since the 80's the UO_2 bundle degradation phenomena, including steam or water reflooding. New experiments are now focused on the reflooding of debris beds.

The effect of fuel burnup was studied in some of these experiments, but with little data above 33 GWd/tU. Thus uncertainties remain, for instance on the real impact of fuel foaming, occurring at high burnup, on core degradation and on FP release.

Very few experiments were conducted on the degradation of MOX fuel rods. The characteristics that would have some influence are the heterogeneity of PuO_2 distribution in the pellet, the PuO_2 chemical response to steam or hydrogen atmospheres and interactions with Zirconium. Some separate-effect tests in the VERCORS facility (operated by CEA in France) showed a larger release of volatile FPs at low temperatures and a lower relocation temperature than for UO_2. Further experiments are planned to answer pending issues. ■

CURRENT ISSUES

Fuel safety limits: experimental results and pending questions

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Considerable experimental effort has been made over the past decade to produce experimental data in support of the definition of fuel safety limits for a variety of fuel designs and considering the effect of burnup. In particular, tests have been performed in specialised facilities (see box p. 14) to address the fuel safety limits at conditions representative of design-basis accidents, i.e. reactivity-initiated accidents (RIA) and loss-of-coolant accidents (LOCA). In addition to assessing the effect of burnup, these tests were primarily focused on the safety performance of different cladding types, especially for PWR fuels.

Specialised test reactors have been – and are being – used to characterise the fuel response to power transient conditions representative of potential RIA, which are postulated to potentially occur in power reactors. The main objective of these tests was – and is – to assess the fuel failure limits (and possibly also the limits for fuel dispersal), as well as to promote a better understanding of burnup effects on fuel behaviour in RIA conditions.

Fuel failure limits in RIA conditions

Based on the experimental evidence available, the RIA failures can be broadly divided in two categories:

- *Brittle failures*, typically occurring in the low *enthalpy* range, i.e. 55-85 cal/g, which are basically caused by pellet-cladding mechanical interaction (PCMI) due to fuel thermal expansion. The driving force in this case is the enthalpy change during the RIA transient.
- *Ductile failures*, which are basically related to fuel and cladding heat-up and to mechanisms consequent to such heat-up. Ductile failures occurred at fuel enthalpy of 115 to 120 cal/g in the NSRR and CABRI experiments, and at ~150 cal/g or higher in IGR and BIGR tests (see box on p. 14). Ductile failures are mainly determined by fuel and cladding heat-up and should therefore be dependent on *fuel enthalpy level* (and not enthalpy increment as for brittle failures) and *rod internal pressure*.

Burnup effects on fuel behaviour in RIA conditions

- *Burnup effect*: Data at low and intermediate burnups show an enhancement of strain with burnup up to 50 to 60 GWd/t mainly due to gap closure. This increased PCMI is conducive to a reduced RIA failure threshold. However, this burnup effect tends to saturate in the very high burnup range, as data show no evidence of additional cladding strain enhancement as burnup progresses beyond 55 to 60 GWd/t.

Reactivity Initiated Accidents (RIA) tests: the main facilities

The Japanese Nuclear Safety Research Reactor (NSRR) test reactor, which has so far produced RIA fuel data mostly at cold coolant conditions (approximately 20 ºC). Its test capability was recently upgraded to include testing at hot conditions (~290 °C). The NSRR tests have been performed with both PWR and BWR fuel, with different types of cladding (Zry-4, MDA, NDA, ZIRLO, M5 and Zry-2) covering burnup up to 77 GWd/t and 69 GWd/t for PWR and BWR fuels, respectively. A limited number of MOX tests were performed with MIMAS/MOX and SBR/MOX up to a burnup of 59 GWd/t. The NSRR is by far the facility where the largest amount of RIA test data for both PWR and BWR fuels was produced.

The French CABRI reactor where testing was focused on PWR fuel including MOX fuel by using its sodium loop. CABRI has, in the past two decades, been the only source of modern RIA data obtained at hot coolant conditions, i.e. 280 ºC. CABRI experiments have been carried out with different pulse width, typically in the range of 9 to 75 ms. The fuel had Zry-4 cladding in most cases. The oxide thickness ranged from 20 to 80 µm, while the maximum burnup was 77 GWd/t for UO_2 fuel and 62 GWd/t for MOX fuel. The first test under a water-cooling condition in CABRI should be performed with a new ‘water loop’ by the end of 2010.

The Russian Impulse Graphite Reactor (IGR) and Large Impulse Graphite Reactor (referred to using the Russian acronym BIGR), which are the main source of RIA experimental data for VVER fuel. The pulse width was very large in IGR (~800 ms) and very short in BIGR (~2.5 ms). Due to limitations in the test instrumentation, the enthalpy at failure could not be determined as it was done in the NSRR or CABRI case. Thus, IGR and BIGR fuel failures data are reported in terms of maximum achieved fuel enthalpy (and not enthalpy at failure as for NSRR and CABRI).

- *Corrosion effect*: Cladding corrosion has been recognised to have an important impact on the failure limit. This is primarily due to the hydrogen accumulated in the cladding metal during the corrosion process, which makes the cladding more brittle. NSRR tests conducted at cold and hot conditions indicate that corrosion effects tend to be increasingly acute at cold conditions, since the cladding ductility diminishes at lower temperature. Corrosion and burnup act in synergy with each other, in the sense where burnup effects become stronger when corrosion is high and corrosion effects become more important when burnup is high. Thus, expressing the RIA failure limit only in terms of burnup or only in terms of corrosion oxide thickness represents, in the authors’ opinion, an oversimplification.
- *Behaviour of PWR advanced cladding*: CABRI and NSRR data show that the greatest benefit of advanced PWR cladding alloys would be represented by their capability to limit corrosion, hence avoid brittleness. The importance of such alloys is expected to be greater for transients at cold conditions, for which brittleness is an issue, than for hot transients, for which cladding can retain ductility regardless of corrosion unless it was spalled. The data so far available indicate that, in order to be effective, advanced alloys should not oxidise beyond a certain limit, provisionally set at 45µm. If corrosion becomes excessive the advantage would disappear, as the recent NSRR tests have pointed out. More data, however, are needed to confirm the improvements that are anticipated for advanced cladding.
- *Behaviour of MOX fuel*: The experimental evidence and analyses of CABRI and NSRR tests show that the failures of MOX fuel can be well predicted by the same failure threshold model used for UO_2 fuel, leading to the conclusion that a different treatment of MOX fuel as compared with UO_2 fuel is not needed, at least for what concerns failure threshold. This however does not exclude that other aspects could be different, for instance in relation to the post-failure behaviour of the two types of fuel.

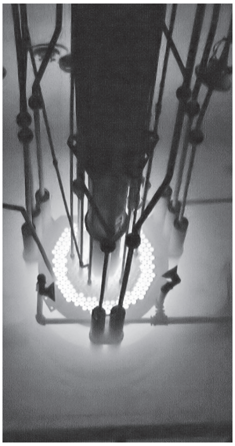
Ballooning, oxidation and corrosion: fuel response to LOCA conditions

As a design-basis accident, LOCA is an accident that the plant design must account and accommodate for in terms of ensuring that a core ‘coolable’ configuration is maintained. During a LOCA transient, the cladding may reach temperatures in the range of 600 to 1200 °C depending on the plant and core design and on the fuel operating conditions. Important phenomena occur in the fuel as the heat-up progresses:

- *Cladding ballooning and burst* normally occur in a 600 to 800 °C cladding temperature range. A number of experimental studies have been conducted in the past to verify that a ballooned fuel assembly remains coolable in spite of the reduced rod-to-rod spacing. The conclusions of these studies, which were conducted decades ago and which evidenced that ballooned assemblies remain coolable, have not been challenged in subsequent times. Hence, the ballooning as such does not constitute a limitation in the current LOCA safety assessment. This apparently remains valid regardless of the cladding type, i.e. Zry-2, Zry-4 as well as new alloys used in PWRs, such as Zirlo and M5.
- *High temperature cladding oxidation*, which occurs when the cladding is exposed to high temperature and steam environment, as it happens in a LOCA, is the major concern from the fuel safety perspective, regardless of the fuel design. In the high temperature range (>800 °C), the cladding oxidation kinetics increases dramatically with temperature, typically doubling for a temperature increase of ~20 °C. One should also consider that the zirconium-water reaction is exothermic and has a heat release of 580 kJ/mol, which enhances the cladding heat-up. The kinetics of high temperature cladding oxidation is described by Arrhenius-type correlation. Experiments performed at Argonne National Laboratory (ANL) in the US State Illinois, in Japan Atomic Energy Agency (JAEA) facilities and other laboratories have demonstrated that there is no need to apply different oxidation correlations for e.g. Zirlo or M5 alloys as compared to Zry-4.
- *Effect of high burnup corrosion on high temperature oxidation*: The oxidation rate (at constant temperature) is not constant with time but decreases as time elapses, since the proportion of reacted metal progresses with the square root of time. In other words, as the oxidation progresses, there appears to be a protective effect induced by the oxide layer created in the oxidation process itself. Hence the question arises as to whether this protective effect is also effective for cladding that has been subjected to corrosion during normal-power irradiation, as it would be the case for high burnup cladding. The experimental evidence on this particular point is scarce. From the application viewpoint it thus appears that high burnup fuel with a certain corrosion layer should be treated as fresh (non-corroded cladding) for the purpose of computing the oxidation kinetics in a LOCA, unless further experimental evidence is brought forward on this particular point.

Zero-ductility and quenching: the basis of LOCA safety criteria

- *LOCA safety criterion based on zero-ductility*: It is known that the zircalloy cladding becomes brittle when subjected to high temperature oxidation in a steam environment due to the hydrogen that remains entrained in the metal itself during the high temperature oxidation process, in addition to thinning of the metal and an increase of oxygen concentration in it. Consistent with the above, the current LOCA safety limits are in many countries based on ductility tests. At ANL, which is the most important source of data for high burnup LOCA ductility assessments, tests are performed by



Cherenkov flash during a pulse irradiation in the NSRR first subjecting tube test specimens to two-side steam oxidation at various temperature levels and for varying time durations. Two-side oxidation provides maximum cladding exposure to steam and is therefore a conservative test configuration. Upon completion of the high temperature oxidation phase, the specimen residual ductility is determined by a ring compression test. The major conclusions of these tests are that:

- The so-called 17% oxidation criterion, i.e. 17% equivalent cladding reacted (ECR), applies to fresh (zero burnup) cladding regardless whether it is Zry-2, Zr-4 Zirlo or M5;
- The oxidation limit decreases with burnup due to cladding corrosion and consequent hydrogen uptake. It decreases in fact from 17% ECR at zero burnup to 0 ECR when the hydrogen content at the beginning of the transient is 700 to 800 ppm, regardless of the cladding type (standard or advanced zirconium alloy);
- The LOCA embrittlement criterion is based on non-deformed cladding specimens, as a database of high-temperature oxidation and ring compression tests of ballooned cladding is nearly non-existent. Hence, in the use of the zero-ductility criterion and depending on assumptions on the way to apply it, situations may arise where requirements may go beyond the existing knowledge, an aspect of the needs to be addressed and resolved in a satisfactory manner.
- *LOCA safety criterion based on integral quench tests*: The experimental basis for the LOCA criteria in Japan is constituted by JAEA quench tests conducted on fuel rod simulators made of cladding tube segments de-fuelled and filled with alumina pellets. The cladding tubes are subjected to heat-up, ballooning and burst, high temperature steam oxidation and then quenching. Different types of PWR cladding and one BWR cladding were tested. The fuel burnup in the region corresponding to the cladding test segment ranged between 66 and 76 GWd/t. The corrosion layer at power reactor discharge was low for the M5 cladding (6-7 µm), while it varied from 32 to 79 µm for the other PWR cladding segments and from 24 to 30 µm for the two BWR segments. Correspondingly, the hydrogen content in the cladding prior to the LOCA test was ~ 70 ppm for the M5 cladding and in the range ~200-850 ppm in the other cases. The ECR calculated with the Baker-Just's oxidation rate was close to or higher than 17% in all cases. The conclusion of the JAEA testing so far is that cladding fracturing does not occur regardless of the initial hydrogen content unless a very high ECR is achieved. In other words, there would be no need to modify the current 17% (or 15%) ECR LOCA limit, as this would remain valid also for high burnup fuel, if the JAEA methodology is adopted.

Different test methods, different conclusions

- *Status on LOCA limit*: The outcome of the JAEA testing is quite a different conclusion than the one reached with the zero-ductility approach (ANL tests), which would imply a strong decrease of the ECR limit with initial hydrogen content. The difference resides in the test method. The ductility testing is on one side far more conservative, but, in spite of this, question marks remain as to how it should be applied, notably in the ballooned region. The JAEA method constitutes the best attempt to simulate a LOCA transient as realistically as possible. This different position is in fact a major consideration regarding the methodology to be adopted in LOCA tests, an issue that the nuclear community needs to address and resolve *in principle*.

- *Fuel relocation*: Integral in-pile LOCA tests conducted in the Halden research reactor, in Norway, have shown that substantial fuel relocation and fuel dispersal may occur in high burnup fuel. The major concern here is that the fuel may relocate towards the ballooned region and exit the fuel rod from the burst opening. Considering that the main aim in LOCA is to maintain a coolable geometry and hence retain the fuel inside the cladding, fuel relocation and dispersal represents a challenge that has not been considered earlier. Further testing on this particular phenomenon is necessary in order to understand the conditions under which it can occur. Type of fuel, burnup level, rod pressure and distance between plenum and burst are parameters that might need to be further investigated in the future.

Pending questions for the coming years

As to the future, the following observations can be made:

- Continued experimentations are needed to confirm several of the above observations, in particular regarding the effect of corrosion and (consequent) hydrogen pick-up, including the behaviour of different cladding alloys;
- As to the fuel type, more data are needed to confirm the behaviour of MOX fuel as compared with UO_2 fuel. Additive UO_2 fuel testing on selected additive compositions will also be required for the licensing of such fuel types;
- NSRR testing at high temperature will be very important for assessing the coolant temperature effect. Return to operation of the Cabri reactor will be essential for understanding dry-out effects and post-failure behaviour;

Visual inspection during skeleton assembly

- In order to understand the margin between fuel failure and loss of coolability, some fuel dispersal tests will be needed in both Cabri and NSRR;
- Finally, looking at the future options, more testing will be needed to qualify more advanced water reactor fuel types, i.e. fuel containing innovative claddings of new pellet composition and/or design, such as for instance Uranium nitride pellets. ■

HIGH BURNUP FUELS: RESEARCH

Nuclear fuel burnup: raising the bar

Kevin Hesketh (NNL) I Britta Helmersson (Westinghouse) I Alexey Dolgov (TVEL)

Historically, LWR fuel burnups have consistently tended to increase with time as fuel designs, materials and fuel management schemes have advanced. Mean LWR fuel burnups are now in the region of 50 GWd/t, with some plants already capable of achieving mean fuel burnups of 60 GWd/t and the likelihood is that this trend will continue to very high burnups of 75 GWd/t and possibly more, conceivably as high as 100 GWd/t. This will pose some significant technological challenges which this article reflects on.

In 2006, OECD-NEA published a study on very high burnups in LWRs that was based on the findings of an Expert Group. This study attempted to project from the then current LWR high burnup experience, when mean assembly burnups of 50 GWd/t were being achieved, to very high mean burnups up to 100 GWd/t. This upper range was used in the Expert Group study to ensure that it would encompass all possibilities and although this article focuses on mean burnups up to ~75 GWd/t, it is important not to rule out higher burnups (100 GWd/t or more) being reached at a later stage. Many LWRs are already capable of achieving mean discharge burnups of 60 GWd/t, within current constraints, although specific licence conditions vary from plant to plant. For example, the Russian designed VVER-1000 plants (see box on p. 20) are capable of achieving a mean discharge burnup close to 60 GWd/t (with a licensed peak rod burnup of 72 GWd/t). Only relatively modest extensions of current operational practices would be needed to attain mean burnups in the region of 75 GWd/t.

Economic benefits of high burnups in LWRs

The average fuel cycle cost shows a decreasing trend up to at least 55-60 GWd/t, where it reaches a minimum at a burnup that varies depending on the specific scenario. If burnups increase beyond 50 GWd/t, there will be diminishing economic returns and the possibility of a slight fuel cycle cost penalty beyond the optimum burnup, because of increasing uranium ore and enrichment requirements needed to support the high initial U-235 enrichment. Nevertheless, given that fuel cycle costs represent only a small fraction of overall generating costs, there is still the potential for operational benefits at very high burnups beyond the fuel cycle cost optimum.

For example, very high burnups may allow extended refuelling cycle lengths and benefit the utility by increasing the overall load factor achievable. Alternatively, a utility might benefit from the lower spent fuel masses, which can extend the effective capacity of the spent fuel ponds and/or interim storage facilities. On the other hand the necessity to increase the hold-up time of spent fuel in the cooling ponds and taking into account special measures while transporting spent fuel may result in higher costs. In both cases, the potential operational benefits can outweigh the modest fuel cycle cost penalty. Increase of fuel burnup gives utilities an opportunity to buy less fuel assemblies to generate the required electricity and hence the chance to save on fuel fabrication and transportation. The fact that the burnup trend shows no sign of slowing down suggests that utilities are well aware of these considerations and the guess is that the eventual stagnation of the burnup trend is still a long way off.

High burnups: the technological challenges for UO_2 fuels

➜ Higher initial ^{235}U enrichments

The initial enrichments for LWRs are currently in the region of 4.5% and even up to 4.8% for VVER-1000. While PWRs have essentially only one enrichment across the entire assembly, BWRs have several enrichment zones. A mean enrichment for a BWR therefore necessarily implies a higher peak enrichment which is already approaching the 5.0% fabrication limit. To achieve a mean discharge burnup of 75 GWd/t will require initial enrichments between 6.0 and 6.5%, depending on the details of the fuel management scheme. Manufacturing fuel of this enrichment may demand design and operational modifications and possibly the admission of different licensing approaches to criticality safety. The first and most pressing technological challenge is therefore to re-licence enrichment plants as well as fuel fabrication plants for higher enrichments. Transport packages for fresh fuel may also need to be re-licensed to accept the higher enrichments.

➜ New technologies in fuel manufacturing process

Realising very high burnups will also place considerable demands on fuel rod design and performance and materials behaviour. Fuel behaviour aspects particularly affected include fission gas release (FGR), cladding creep and corrosion, reactivity insertion accident (RIA) response, high burnup structure evolution, pellet-clad interaction (PCI) and fuel rod growth. In the past 30 years, LWR fuel assembly designs and materials have advanced considerably to the extent that mean burnups of 50 GWd/t are achievable. The main effort was made to improve thermo-mechanical behaviour of fuel assemblies in the core. Modified zirconium alloys with better corrosion resistance and mechanical properties as well as new technical solutions are used in fuel design to get a robust fuel for the whole lifetime. It is likely that achieving very high burnups up to 75 GWd/t will require long research and development lead times. Moreover, producing of fuel pellets with specified structure or fuel assembly components made of new, corrosion-resistant and low-creep zirconium alloys are major challenges for the industry.

➜ Enhanced nuclear reactor core design and performance

The higher initial enrichment increases the excess reactivity that needs to be controlled at beginning-of-cycle conditions and at the same time increases the differential between the reactivity of fresh assemblies and assemblies that are in their last irradiation cycle. These facts will both demand increased use of burnable poisons and possibly innovative core loading strategies to ensure that operating margins remain within acceptable limits. Reactivity feedback coefficients, control rod worths, soluble boron worths (for PWRs) and

Radiation measurement

Burnup trends: the Kalinin-1 NPP example

For an individual LWR plant, a typical burnup trend plot will show mean discharge burnups increasing in steps, resulting from periodic decisions to increase the initial enrichment of the fuel and possibly to change the core refuelling fraction. The burn-up steps are then followed by operation at a constant plateau as the core reaches a new equilibrium. Such a trend is well illustrated by the recent discharge burnup history for the Kalinin-1 NPP, which is a VVER-1000. This shows the discharge burnup increasing in two steps to 55.8 GWd/t in 2010. The small reductions in burnup in 2007 and 2009 are caused by the time lag effects of non-equilibrium operation.

Kalinin-1 bumup trend	Year					
	2005	2006	2007	2008	2009	2010
Mean discharge bumup, GWd/t	45.0	49.2	47.7	54.4	53.6	55.8

For an individual plant such as Kalinin-1, burnup trends can be complex, but the average discharge burnups over the entire world fleet of both PWRs and BWRs has shown a continuing gradual increase over many years.

shutdown margins will all be affected, with possible implications for normal operation and accident behaviour.

➜ Nuclear fuel cycle back-end

The decay heat and neutron outputs of irradiated fuel are both very sensitive to the discharge burnup. Immediately after discharge the decay heat is dominated by short-lived fission products and is not particularly sensitive to burnup. However, the fall-off in decay heat after a few days' cooling is notably slower at high burnups, so that at any given cooling time high burnup fuel assembly will have a stronger decay heat source than a low burnup assembly. Similarly, the neutron source increases with burnup, roughly doubling between 50 and 75 GWd/t. The capacity of irradiated fuel transport and interim storage containers may be limited by decay heat and neutron source considerations. This will have an impact on spent fuel management operations, specifically, the length of time that irradiated fuel needs to be retained in the cooling ponds before transfer to interim storage and on the loading of interim storage or transport flasks.

The elevated decay heat and neutron source terms also have important implications for the management of spent fuel (i.e. reprocessing or conditioning/disposal) and for geological disposal of either the high-level vitrified waste after reprocessing or the spent fuel. The neutron source term will impact on radiological protection in a reprocessing plant or a MOX fuel conditioning plant and the subsequent handling of highly active materials. The decay heat term may be the limiting factor determining the linear storage capacity of spent fuel in the geological repository. The feasibility of recycling high burnup fuel needs to be evaluated because it will contain elevated amounts of the ^{232}U, which is radiologically important and ^{236}U that has an important effect as a neutron absorber. When measured in terms of environmental detriment per TWh of electrical output, the LLFP source term is virtually independent of burnup since, while the discharge inventory of long-lived fission products (LLFPs) increases with burnup, the fuel throughput reduces in inverse proportion.

The technological challenges for mixed oxide (MOX) fuels

The same challenges will apply as for UO_2 fuels and some may be more difficult to address. There are also unique issues for MOX fuels, such as the need to ensure the total plutonium content is not so high as to cause the local void coefficient to be significantly positive. On the other hand, the economic incentive for very high MOX burnups is much more emphatic, because the cost of MOX fuel procurement does not depend to any significant extent on the initial plutonium content and this might provide the necessary incentive to tackle the technical hurdles.

The second-generation fuel TVSA for VVER-1000

Long-term prospects

The next generation of LWRs being built are still likely to be operational as far ahead as 2080. Over this time, considerable advances in fuel technology and understanding of fuel behaviour are anticipated. On such extended timescales, it is likely that new Generation-IV fast reactor systems will have matured, so that LWR fuel developments may proceed in parallel with fast reactor fuel development. It is possible that there will be commonalities between both fuels, particularly on advanced fuel and cladding materials and the computer codes used to predict fuel and materials behaviour. Experience from operating fast reactors in Russia, France, Japan, USA and UK has already demonstrated that very high burnups are achievable in fast reactors: Up to 11.3% heavy atom (ha) – i.e. uranium and transuranic elements – burnup has been achieved for BN-600 discharge batch, equivalent to roughly 110 GWd/t in thermal reactors, with research efforts being made to achieve burnups of 20% ha. The need to develop structural materials with adequate properties is a key aspect of very high burnup fuels for both LWR and fast reactors.

A number of technological challenges will need to be addressed if very high burnups are to be realised in LWRs. The lead times for developing new fuel designs and especially new materials are such that very high burnups will only be achieved following a continuation of the evolutionary progress that has been seen in the past. There is little doubt that the fuel vendors have the technological expertise needed to achieve this and it is virtually certain that further evolution towards very high burnups will continue for some time, with no suggestion yet of any limiting factor preventing further progress. ■

BEHAVIOUR IN REACTOR CORE

Nuclear fuel development: are present safety criteria relevant?

Georges Hache (IRSN) I Wolfgang Wiesenack (HRP)

The development of high-burnup fuels aimed at improving NPP operation poses questions regarding the relevance of the present safety criteria. To provide answers, TSOs and regulators draw upon the experience feedback from past accidents and present experiments performed in test reactors to assess the behaviour of high-burnup fuels in accidental conditions. In a second phase, the present article focuses on the licensing procedure in the Czech Republic, at Temelín NPP, where a Russian designed VVER-1000/320 is supplied with Westinghouse Electric Corp. fuel (see p. 26).

The relevance of the present safety criteria regarding the new fuel development is reviewed hereunder taking two examples of design basis accidents (DBAs), i.e. reactivity-initiated accidents (RIAs) and loss-of-coolant accidents (LOCAs).

Reactivity-initiated accidents

A RIA is a nuclear reactor accident that involves an unwanted increase in fission rate and reactor power. Regulatory bodies identified some RIA scenarios as design basis accidents: in particular, the control rod cluster ejection accident in PWRs and the control rod blade drop accident in BWRs.

The immediate consequence of a RIA is a fast rise in fuel power and temperature. The power excursion may lead to failure of the nuclear fuel rods and release of radioactive material into the primary reactor coolant. In severe cases, the fuel rods may be shattered and large parts of the fuel pellet inventory dispersed into the coolant. The expulsion of hot fuel into water has the potential to cause rapid steam generation and pressure pulses, which could damage nearby fuel assemblies, other core components, and possibly also the reactor pressure vessel. To prevent those potential consequences, safety criteria are usually set up to limit the energy injection into the fuel.

After the Chernobyl accident, in the early 1990s, experimental programmes, in the form of pulse irradiation tests, were initiated in France, Japan and Russia to study the behaviour of high burnup fuel under RIA. These tests show that cladding failure and fuel ejection occur at much lower fuel enthalpies for high burnup than for fresh fuel rods, when the cladding alloy is susceptible to waterside corrosion and associated hydrogen pickup during normal operation. Moreover, failures of high burnup fuel rods usually occur at an early stage of the power surge, when the cladding temperature is low. This is attributed to the combined effects of clad tube embrittlement and aggravated pellet-clad mechanical interaction (PCMI) in high-burnup fuel rods. It is also clear that the burnup dependent state of the rod, and in particular the de-

gree of cladding waterside radiation-accelerated corrosion, is very important for the survivability of high burnup fuel rods.
Many countries have adopted provisional modified criteria that take into account the results of the aforementioned pulse irradiation tests. However, more research is needed to define new criteria, as:

- The pulse irradiation tests performed in France in a sodium loop in the CABRI reactor did not test the high temperature failure mode. A water loop is being implemented in the CABRI reactor;
- The pulse irradiation tests performed in Japan with low temperature capsules are representative for cold zero power conditions of BWRs, but too conservative for hot zero power conditions of PWRs. Tests with high temperature capsules are being performed.

Loss-of-coolant accidents

A LOCA is a nuclear reactor accident that involves a break or valve opening on one of the coolant pipes that is not isolated from the reactor vessel.
In order to mitigate the consequences of this break, it is necessary to design the emergency core cooling systems (ECCS) so that the fuel is cooled efficiently during all phases of the accident. This requirement naturally leads to a criterion that the fuel must maintain its coolable geometry throughout the whole LOCA sequence and that the structural integrity of the fuel rods is maintained.
During the event, the cladding heats up to temperatures over 1000 °C. Zirconium metal is oxidised and oxygen dissolves in the metal and embrittles it. Therefore, there must be a limit on the oxidation, and associated calculation methods, since the load bearing metal layer may be too thin or may contain too much oxygen to ensure structural integrity of the fuel during and after the quench phase of the LOCA. When the fuel rods heat up during the LOCA and the external pressure is lost, the rod internal pressure is large enough to cause plastic deformation of the cladding, which leads to ballooning and burst. The ballooning can potentially be detrimental to cooling of the fuel assemblies, and the burst of a rod also leads to cladding oxidation from the inside. In addition, the cladding picks up a significant amount of hydrogen that exacerbates cladding embrittlement.

LOCA criteria and calculation methods

In recent years there has been considerable testing of the quench resistance and post-quench ductility of high burnup, or surrogate prehydrided, current cladding alloys in Japan, the U.S., France and Russia. The existing criteria were based largely on test results for unirradiated or moderately irradiated fuel. Hence, an extension of the experimental database to higher fuel burnup was needed. Results showed a strong effect of hydrogen concentration, when the cladding alloy is susceptible to waterside corrosion and associated hydrogen pickup during normal operation. Many countries have thus reduced their transient oxidation criteria by the amount of the corrosion obtained during operation. The US NRC has initiated a rulemaking process to revise its criteria. France will define its route to new criteria in early 2010. At this stage, more research is needed to study the toughness of the balloon zones affected by hydrogen pickup during internal oxidation following burst.
Regarding the LOCA calculation methods, the review of older test results has shown that important phenomena have been omitted, especially the fuel fragments relocation in the balloon at the burst time. Sev-

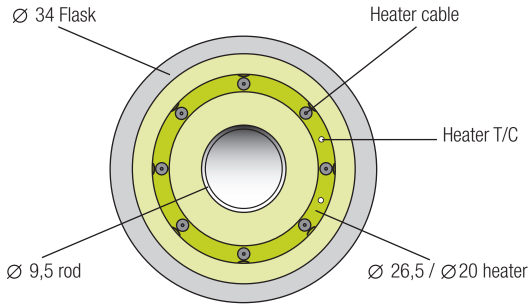


Figure 1
Cross section of fuel pin, heater and pressure tube used in HRP LOCA studies

eral countries, including US NRC and France, have postponed any modification of these methods until obtaining the quantitative results from the OECD Halden Reactor Project (HRP), carried out in Norway.

Some observations from the OECD Halden Reactor Project test series

The HRP has implemented a series of tests on issues related to fuel behaviour under LOCA conditions, with a focus on integral in-reactor effects that are different from those obtained in out-of-reactor set-ups. Of major interest for the investigations are the interaction of bonded fuel and cladding, the behaviour of fragmented fuel around the ballooning area, and the axial gas communication in high burnup rods as affected by gap closure and fuel-clad bonding.

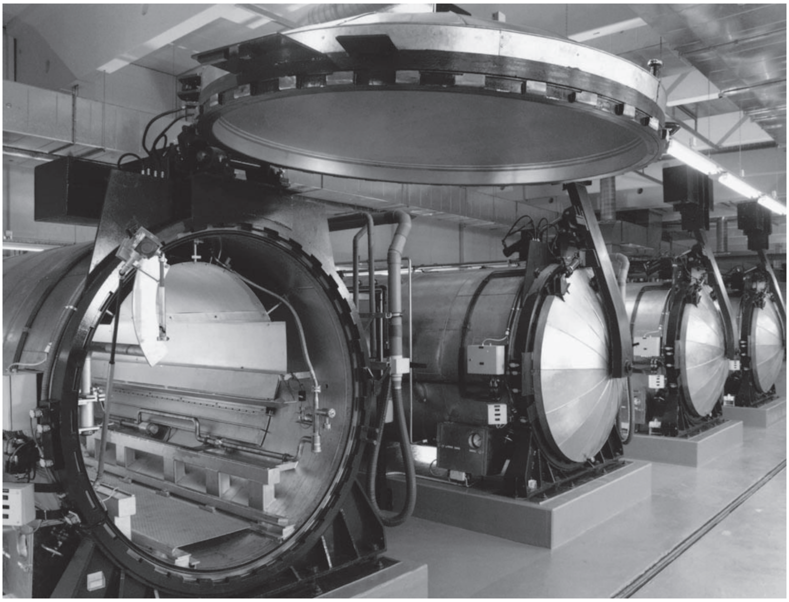
A primary objective of the test series is to observe the overall in-reactor fuel behaviour under expected as well as bounding conditions. An essential part is to assess the extent and effect of fuel relocation into the ballooning, the filling ratio of the ballooning spot with fragments, the heat/temperature distribution within the balloon area, and the effect of locally increased temperature on cladding behaviour and properties.

A secondary objective is to investigate local oxidation and transient hydriding, in particular in the transition zone between ballooned and geometrically unaffected parts of the fuel. As of 2009, several tests with fuels in the burnup range 35 to 92 GWd/t have been executed. All fuel segments had a length of about 50cm. Bounding conditions are addressed by aiming at a peak clad temperature (PCT) of about 1100 °C in some cases, while other tests with a PCT of 800-850 °C investigate the behaviour at conditions expected to be actually reached by high burnup fuel. An overview is shown in Table 1.

Test number 4 (92 GWd/t) has attracted particular attention since strong fuel fragmentation and expulsion out of the ballooned and burst rod occurred. As the complex interactions between material behaviour, fuel state and temperature distribu-

Test #	Fuel type	Burn-up (GWd/t)	PCT, °C
1-2	Commissioning runs	0	600-1150
3	PWR, Zry-4 duplex	82	800
4	PWR, Zry-4 duplex	92	800
5	PWR, Zry-4 duplex	83	1100
6	VVER, E110	56	850
7	BWR, LK3/L	35	1100
8	Commissioning run	0	800-1100
9	PWR, Zry-4 duplex	92	1100

Table 1 Overview of LOCA tests executed in the Halden reactor



Autoclaves

tion have to be taken into account in LOCA safety analysis codes, there are several implications from these tests as well as from code benchmarks:

- Fuel can fragment, accumulate in the ballooning volume and create a potentially hot spot with increased cladding oxidation and embrittlement;
- However, the observed filling ratio of 40-50% is lower than previously assumed;
- The ability to generate sufficient gas for driving the ballooning and for causing axial fuel relocation may be affected by plugs of fuel at the colder ends of a full-length rod;
- Codes for LOCA fuel behaviour predictions, while doing an overall reasonable job, have room for improvements regarding heat transfer calculations and models that consider the effect of axial gas communication.

Conclusion

Test results show that present safety criteria and calculation methods have to be revised for high burnup cladding, especially when the cladding alloy is susceptible to waterside corrosion during normal operation. In several countries, regulators have implemented provisional measures. On their side, fuel manufacturers have developed new alloys less prone to waterside corrosion than Zircaloy-4. With a view to finalising new criteria and calculation methods, the OECD-NEA Working Group on Fuel Safety will continue to use the Halden tests for model development and benchmarking. An important question remaining to be answered is the burnup dependence of fuel fragmentation and relocation. The test scheduled for April 2010 in the Halden Reactor will utilise PWR/Zry-4 fuel of about 62 GWd/t burnup and is expected to shed some light on this issue. ■

New fuel design licensing procedure in the Czech Republic

Alexander Miasnikov (SONS)

The Atomic Act entrusts execution of the state administration and supervision in the peaceful utilisation of nuclear energy as well as ionising radiation to the State Office for Nuclear Safety (SONS) of the Czech Republic, and establishes activities, such as new fuel designs, for which a licence issued by SONS is mandatory. Several new procedures and approaches were developed in the process of licensing the fuel for the Temelín plant.

Basic Requirements: the virtue of a flexible legislation

Basic Requirements are considered to be valid for all design changes. In addition to Czech codes and standards, SONS required all deliverables to meet the national codes and standards of the country of origin. Another obligation is the demonstration of the design's compatibility, reliability and safety-related influence.

Safety assurance for fuel safety related items has to be demonstrated by submitting complete documentation dealing with the design's compatibility with other components and parts, taking into account existing (original) materials, moderator (water chemistry), especially from the standpoint of:

- thermal hydraulic properties: vibration, hydraulic resistance, critical heat flux correlation, fuel rod bowing, effect of spacing grids, pressure losses,
- mechanic properties: rigidity, cyclic fatigue, wear, cladding abrasion, deformation by external forces (load during LOCA and seismic events), kinetics of control assemblies drop,
- chemical properties: corrosion, hydriding,
- neutronic-physical properties: peaking factors, influence of different enrichment, water-uranium ratio, etc.; shutdown reactivity margin; stability; maximum speed of the reactivity insertion, both calculated and experimental (especially for non-active tests).

A particular situation arose from the integration of technical equipment from different countries, designed and manufactured at different times, for the completion of the Temelín NPP. Whereas the plant was originally a Russian designed VVER-1000/320, the reactor fuel as well as I&C supply was awarded to Westinghouse Electric Corp. (WEC) who assisted ČEZ (the utility) in issuing preliminary as well as final safety analysis reports (SAR). Upon assessing the SAR amendment and related safety documentation, SONS paid particular attention to the impact of design changes on the original Temelín design.

Since the Czech legislation allows adopting any set of criteria or limits aimed at fulfilling general public health and safety requirements as well as general design requirements, the decision was made to apply the US NRC Licensing Review Process to the SAR parts concerning fuel (and I&C) as a basis. The difficulties arising from the fact that SONS staff was not familiar with the US NRC

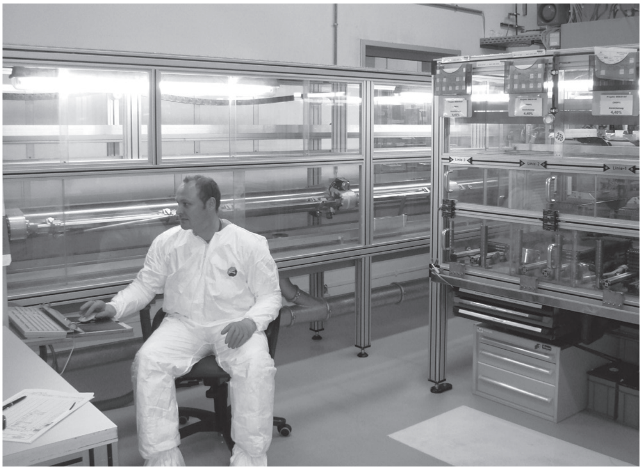
Licencing Review Process was overcome by introducing the Program for Transfer of US Licensing Methodology to the Czech Republic.

Independent expert committees to appraise computer codes

The use of validated computer codes increases the efficiency of the licensing process and reduces the pressure on both the regulator and licensee. To draw upon computer codes of good quality to assess the NPP characteristics, Technical Appraisal Committees were formed corresponding to different professional areas. Acting as independent experts, the Committee members carry out the appraisal of a given computer code and suggest that the evaluated code be incorporated into the set of evaluated codes. This procedure, classified as good practice by an IAEA international regulatory review team (IRRT) mission, is subject to reconsideration after three years. With this purpose, SONS has formed, according to specific areas, seven expert commissions tasked with validating the computer codes used in the safety assessments.

The usefulness of an 'issues & questions database' to support licensing activities

During the licensing review process, it was found out that a good filing system was needed to effectively manage the licensing process. It was therefore decided to computerise the paper files and to create an 'issues & questions database' as a tool to support the supervision of the Temelín NPP licensing process, at least for all areas related to WEC supplies. Records in the database are organised following predominantly the structure of the NUREG 0800 Standard Review Plan, consistently with the SAR content, which comply with the US NRC Reg. Guide 1.70. In addition, cross-reference tables of all



Control of fuel rod production

requirements from different sources – Czech legislative requirements, US legislation, procedure and regulations guides, etc. – were elaborated to establish that no issue was omitted. As a result, legal bases for requirements for the relevant areas are adequately covered, demonstrating that the database serves the purpose for which it was created, especially:

- registering the technical issues to be answered in the utility's documentation during the licensing process;
- storing the results of reviews that have been carried out as bases for the safety evaluation report;
- registering requests for additional information;
- registering the utility's (or vendor's) response;
- monitoring on-line the progress of the licensing process.

As the database proved itself as a very useful tool, it is applied to other fuel design changes. ■

Spent fuel management: nuclide inventory calculation and burnup credit application

Bernhard Gmal and Robert Kilger (GRS) I Kåre Axell (SSM)

The currently possible options in spent fuel management are either the long-term storage for time scales in the range of 40 up to 100 years, the separation of re-usable materials as uranium and plutonium, or the disposal in a geological repository with or without an option of retrievability. Whatever option is considered, good knowledge of the nuclide inventory of the spent fuel is essential to ensure its safe management, just as it is for safeguard purposes.

Gaining detailed knowledge of the nuclide inventory: a prerequisite for the safe management of spent fuel

The main issues related to the safe management of spent fuel are the safe enclosure of the radioactive inventory, the shielding of the γ- and neutron radiation, the sub-criticality under normal and accidental conditions, and the removal of the decay heat from the spent fuel to avoid any failure of the cladding.

Performing safety analyses on these issues requires a detailed knowledge of the nuclide inventory, e.g. for the design of transport and storage canisters, the reprocessing or disposal of spent nuclear fuel. In the case of burnup consideration for criticality analysis (burnup credit), which in fact reduces the intrinsic conservative safety margin with respect to the "fresh fuel assumption", the detailed knowledge of the concentration of fissile nuclides, higher actinides and certain fission products is necessary, considering even the axial distribution across the length of the fuel element.

Appropriate calculation methods and computer codes have been developed and improved over the years.

Experimental data from post-irradiation examinations (PIE) of spent fuel samples have been collected and prepared for recalculation with a view to validating these codes. In addition, international benchmarks for code comparison are being performed. The OECD NEA's[1] Committee on the Safety of Nuclear Installations (CSNI) supports many activities in this field.

Quantifying uncertainties in highly sophisticated codes

For safety analysis purposes, a detailed knowledge of the isotopic inventory of the spent fuel is indispensable:

- *for the calculation of the neutron multiplication factor* e.g. of an arrangement of spent fuel assemblies in a storage pond or inside a transportation cask, if burnup credit is

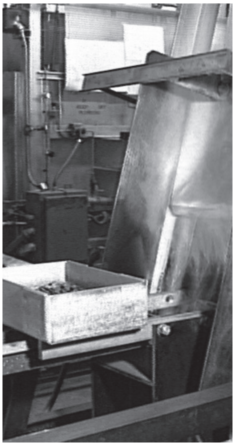
[1] Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (OECD NEA)

applied, the mass and concentration of the fissile nuclides ^{235}U, ^{239}Pu and ^{241}Pu as well as of the crucial neutron absorbing nuclides as e.g. ^{240}Pu or ^{149}Sm have to be well known;
- *for the calculation of decay heat*, where the knowledge of the fission products ^{134}Cs, ^{137}Cs+^{137m}Ba and others in the short term, ^{137}Cs and ^{90}Sr and their decay daughters in the mid term, and ^{244}Cm, ^{241}Am, ^{238}Pu and others in the long term is of primary interest;
- *for the determination of radiation source terms* in shielding applications in the range of up to one hundred years after discharge of the spent fuel, where nuclides like ^{144}Pr, ^{106}Rh, ^{134}Cs and others are the dominant contributors;
- *for the reprocessing of spent fuel*, where the spent fuel is dissolved in nitric acid;
- *for disposal*, as knowledge on long-lived fission and activation products such as ^{14}C, ^{36}Cl, ^{99}Tc and ^{135}Cs is required;
- *for the radiochemical burnup determination of spent fuel samples*, where other nuclides like ^{148}Nd play an important part.

One can clearly figure out that in any of the fields of safety analyses pertaining to spent fuel, the proper calculation of the masses of a vast amount of different nuclides is a primary task, performed by the use of depletion calculation code systems. Nowadays, burnup dependent 2D or 3D flux calculation tools coupled to rod-wisely applied point depletion codes being used to model a whole fuel assembly are state of the art. By this means, radial structures in modern fuel assembly designs, such as guide tubes, fixed or removable absorber (control) rods, water channels, or fuel rods with varying initial enrichment, can be accounted for properly. If calculation technologies have reached a very high level of performance and accuracy, it is nevertheless mandatory, in a licensing procedure validation of the calculation tools, especially in the field of criticality safety, to determine and quantify uncertainties and biases in the code system.

Validation of inventory calculation codes: need of experimental data

Such validation is performed by recalculation of a sufficient number of measured inventories taken from spent fuel specimens. These samples are taken from irradiated fuel rods and then are dissolved and analysed both radio-chemically and by spectroscopy. Up to the year 2000, the number of freely available PIE data was low and mainly focused on actinides. Another drawback was the fact that many samples were of rather low initial enrichment and burnup. Sometimes they were also not fully suitable for an optimum depletion code validation due to lack of important irradiation data or an inadequate choice of sample position within the assembly.
Since then, a number of dedicated measurement programs in different countries have been performed: samples of different initial enrichments and higher burnups, also including MOX fuel samples, have been analysed with the focus on actinides and a high number of selected fission products of high interest to various applications. However due to the participation of various partners from the industry, these data are proprietary and so far unavailable to the public. Ideally, every nuclide to be included in the safety analysis is also included in any PIE sample whose inventory has been recalculated, but also this is not always the case.
Another important issue is a comprehensive knowledge of the irradiation conditions of the respective samples within the core: e.g. local power, tem-



Sintering furnace

perature and neutron flux conditions. Every parameter influencing the neutron spectrum also impacts on the residual reactivity of the irradiated fuel after discharge. Main effects originate from the fuel temperature, coolant temperature and density, boron concentration, control rod movement, but also the vicinity of e.g. the guide tubes, gadolinium-bearing rods, or neighbouring MOX assemblies. Thus, for recalculation, the highest quality PIE sample is useless without the corresponding irradiation data. However, operation data and often also certain assembly details are property of the operator or the fuel vendor, and therefore also not always freely available.

Taking burnup credit into account for spent fuel management

In the classic approach for criticality safety analyses of spent nuclear fuel, the irradiated fuel assemblies are assumed as being fresh. This approach called "fresh fuel assumption" is applied straightforward, based on the initial fuel composition with full initial enrichment and without regard for decrease of fissile material and build-up of neutron-absorbing actinides and fission products. No burnup calculation is needed and the criticality calculations in general are based on very few well-known nuclides. Moreover, an *intrinsic safety margin* is included but not determined explicitly.

With the improvement of calculation tools, the intention to *reduce this overly conservative safety margin* arose, various economical and safety considerations giving incentives to take into account the net reactivity reduction of nuclear fuel due to irradiation in the reactor core, the so-called *burnup credit*. Fission and absorption processes during irradiation lead to a net reduction of fissile nuclides, as well as the generation of neutron absorbing nuclides, both actinides and fission products. Most prominent are e. g. ^{240}Pu or ^{149}Sm. Besides, other fissile nuclides than ^{235}U, mainly ^{239}Pu and ^{241}Pu are also generated, in turn giving positive contributions to the overall reactivity which is nevertheless being reduced.

In the criticality safety analysis, nuclides with no or insignificant reducing effect on the reactivity can be neglected. Conversely, every nuclide increasing the reactivity has to be regarded. Nuclides with significant reduction of the reactivity have to be stable or at least long-lived enough to be present in a chargeable amount during the licensing period. The NEA working group on burnup credit has focused on a set of 13 actinides and 15 fission products as being of main interest. The absorbing nuclides within the set of actinides mentioned above cover about 80% of the overall reactivity reduction of all absorbers, mainly due to the vast amount of ^{240}Pu in the spent fuel. The 15 fission products mentioned, on their part, cover about 75% of the overall fission product contribution. From the economic point of view, studies have shown that these fission products should be considered in burnup credit for shipping casks to get a significant positive effect. In the United States, e.g., the financial benefits from the application of burnup credit is potentially estimated to amount to up to US$ 600 million due to a reduced number of casks, shipments and disposal storage capacity.

Another important issue in the application of burnup credit is the fact that the distribution of burnup over the fuel assembly is inhomogeneous: it varies both horizontally and vertically, the latter being well known as axial burnup profile. This inhomogeneity impacts on the multiplication factor of the spent fuel bearing system and thus has to be accounted for properly. For burnups higher than about 15 GWd/tHM, the assumption of a uniform, homogeneous average burnup along the axis is conducive to non-conservative results for the calculated multiplication factor k-eff. This effect becomes more and more pronounced with increasing average burnup.

Boundary conditions: the limit to endless complexity

To avoid an excessive number of burnup calculations for each single discharged fuel assembly, boundary conditions can be defined for a reactor system to get a unique conservative boundary composition and profile to cover each spent fuel assembly in a fuel management system (see Fig. 1). While mainly being trivial for most of the other irradiation parameters, the definition of boundary conditions concerning the axial burnup profile is a sophisticated, challenging task which can be carried out by means of different approaches ranging from various statistical means, straightforward but nevertheless extensive calculation of k-eff for thousands of measured profiles to determine the most reactive, or specification of a lowest allowed minimum burnup for the top 50 cm of the assembly. Common sense is that this bounding profile is dependent on the fuel assembly design and, due to unique operation strategies, in principle also on each single reactor.
As a perspective, with the worldwide tendency of increasing initial enrichment, and in the frame of different advanced reactor and fuel cycle concepts, burnup credit is very likely to be of growing interest to countries and companies operating nuclear power plants and fuel management systems.

Nuclide inventory measurements: a major contribution to spent fuel safeguard

Apart from the safety concerns, safeguard aspects require consideration. Ideally, the same measurements could be used for both purposes, but that is not always the case. When fuel is placed in difficult-to-access storage, the IAEA requires that verification on a partial defect level (part of assembly) be performed. During operation, only the verification on gross defect level is carried out, i.e. may the fuel assembly be there or not. At the moment, requirements state that verification should be achieved using the "best available method" and a systematic test should discover a 50% partial defect with a 90% probability. Furthermore, the method should be neither too intrusive on the normal operations nor require too much inspection effort.
The best method currently available would be a tomography measurement of each fuel assembly that would establish the presence (or not) of every fuel rod. If such verification were performed in conjunction with

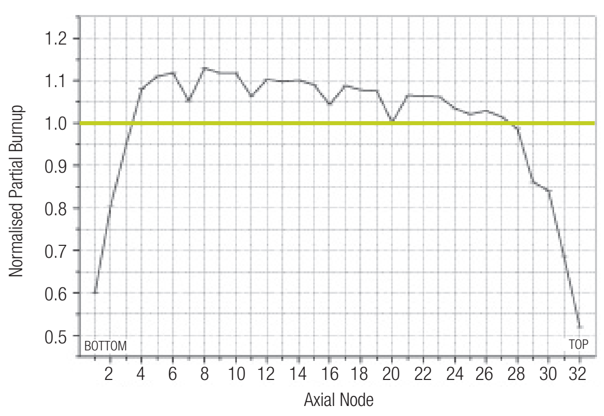
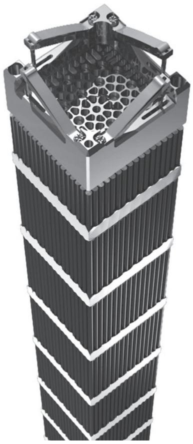


Figure 1 Example for a typical axial burnup profile for a PWR



PWR fuel element

fuel handling, intrusiveness would be minimised. Promising tests are being carried out in Finland and Sweden this year.

One question that arises is what to do with anomalies, e.g. if records of fuel pin removal are lost? This could happen several years after the removal took place, in some storage ponds even 40 years or more. In order to avoid such occurrences, a less intrusive and simpler method would be preferable, using the Digital Cherenkov Viewing Device (DCVD). Located at the railing over 10 m away from the fuel, the detector records the Cherenkov radiation that is produced in the water surrounding the fuel rods, at longer cooling times primarily from Caesium. Developed as a gross defect tester, the instrument is capable of seeing radiation from spent fuel with a burnup of only 10 GWd/t U that has been cooled over 40 years.

Further studies have concentrated on the use of the DCVD as a partial defect tester: a number of PWR fuel elements have been measured and their relative intensities can be found in Fig. 2. Most of the fuels have roughly the same burnup, 44-48 GWd/t U, except for the last fuel (5A0) that has a burnup of 23 GWd/t U. The range in cooling time is from 1.3 years to 14.3 years. A major advantage of the DCVD is to allow establishing readily the fuel parameters. In conclusion, the DCVD is a candidate for scanning fuel in order to verify the operator's declaration. This can be done readily at any time prior to encapsulation/transfer to not readily accessible storage without any fuel movement or fuel handling. ■

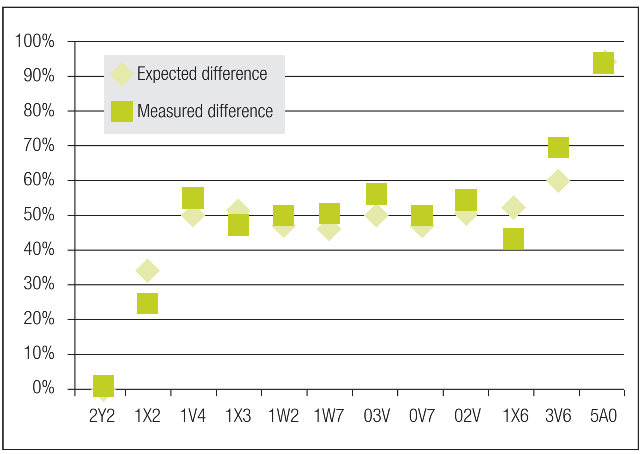


Figure 2 Expected and mesaured intensities relative to PWR assembly 2Y2

VENUES & WEBSITES

Upcoming meetings on nuclear fuel

23-27 November 2009, Saclay, France

International seminar on thermal hydraulics of light water reactors
Organised by the European Nuclear Education Network Association (ENEN)
Tel: +33 (0)1 69 08 97 57
E-mail: sec.enen@cea.fr

1-4 December 2009, Vienna, Austria

Technical meeting on safety issues related to the use of high-burn up fuel and to the long residence time of fuel in the reactor
Organised by the IAEA

7-11 December 2009, Kyoto, Japan

FR09' – Fast reactors and related fuel cycles: challenges and opportunities
Organised by the IAEA
Tel: +43 (0)1 2600 21311
E-mail: official.mail@iaea.org

21-25 March 2010, Marrakech, Morocco

RRFM 2010 : 14th international topical meeting on research reactor fuel management
Organised by ENS and CNESTEN
Kirsten Epskamp (ENS)
Tel: +32 (0) 2 505 30 54
E-mail: rrfm2010@euronuclear.org

30 May-3 June 2010, Barcelona, Spain

ENC 2010 - European Nuclear Conference
Kirsten Epskamp (ENS)
Tel: +32 (0) 2 505 30 54
E-mail: rrfm2010@euronuclear.org

26-29 September 2010, Orlando, USA

Top Fuel 2010, LWR fuel performance meeting
Sponsored by ANS, ENS, AESJ and KNS
www.ans.org/goto/fuel10
E-mail: fuel@ans.org

A few links for reading more about nuclear fuel safety

- SNETP strategic research agenda, May 2009
www.snetp.eu

- Implementing geological disposal of radioactive waste. Vision document of the IGD-TP
www.igdtp.eu

- Quantitative studies to detect partial defects in spent nuclear fuel using the digital Cerenkov viewing device
by D.A. Parcey, J.D. Chen, A.F. Gerwing, R. Kosierb, M. Larsson, K. Axell, J. Dahlberg, B. Lindberg, E. Sundkvist
Presented at the ESARDA 2009 Conference in Vilnius, Lithuania, 2009 May 26-28
www.channelsystems.ca/documents/PartialDefect09Jul8.pdf

- Safety of the nuclear fuel cycle
by B. Kaufer and D. Ross
www.oecdnea.org/html/pub/newsletter/2005/23-1-safety.pdf

- Mixed-oxide (MOX) fuel performance benchmark
by L.J. Ott, Oak Ridge National Laboratory, USA
www.nea.fr/html/science/reports/2009/6291-MOX.pdf

- Nuclear fuel behaviour in loss-of-coolant accident (LOCA) conditions – State-of-the-art report
www.nea.fr/html/nsd/reports/2009/nea6846_LOCA.pdf

- Nuclear fuel cycle transition scenario studies – Status report
www.nea.fr/html/science/reports/2009/nea6194_transition_scenario_studies.pdf

- Advanced fuel cycle initiative
Argonne National Laboratory
www.ne.anl.gov/research/afc/index.html

- A Network of Excellence Federating European Research on Core Meltdown Reactor Accidents
www.sar-net.org

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