# EDF CURRENT AND FUTURE HOTLAB NEEDS

BENJAMIN THERACHE
*EDF Nuclear Engineering Division – Basic Design Department*
*12-14 avenue Dutriévoz 69628 Villeurbanne, France*

## ABSTRACT

The operation of nuclear power plants requires the evaluation of the processes leading to failure and the development of adequate basic design methodologies to fulfil the related safety issues. Fuel cycle optimisation is strongly connected to fuel behaviour assessment: pellets and cladding behaviour under irradiation, effects of increasing burn-up or power, development of advanced materials. There is also a need to increase the database on irradiated fuel rod behaviour on the long term for the spent fuel field. NPPs lifetime assessment involves surveillance programmes on the effects of irradiation on the behaviour of key components. Hot cells laboratories exams and material test reactors results are a source of crucial information regarding all these issues and as a major utility, EDF has to be concerned on these facilities improvement.
In general EDF needs well designed in-pile or out-of-pile separate effect tests that will provide relevant data used to validate and qualify the models of its fuel performance code. To satisfy these needs, EDF has developed its own capabilities, is referring to external facilities (hot cells laboratories and test reactors) and is involved in international R&D programmes. This paper presents EDF current and future needs regarding hot cells requirements.

## Hot laboratories exams as a support of operational needs

### Survey programmes

The first kind of exams on irradiated fuels are the pool-side measurements. On-site exams such as visual examinations, geometrical measurements, oxidation and wear measurements are performed in the framework of survey programmes to check that the fuel assemblies behave properly and to validate new fuel designs. These exams can also be ordered if some unexpected fuel behaviour occurred during the reactor cycle and need for specific examinations. In some specific circumstances the deteriorated assembly may be repaired on-site directly. Innovative non-destructive techniques are obviously welcome to better characterize the status of the fuel assemblies. As an example, rod pressure measurement could be assessed using an acoustic technique recently developed in the Montpellier University [1].

### Failure processes assessment

If needed a fuel rod can be extracted during fuel handling and shipped to external hot cells. The hot laboratories exams are the main source of information to explain a rod failure during nuclear power plant operations. The reasons for the failure are various: debris fretting, grid-to-rod fretting, primary hydriding, end-plug and weld defects. The failure process is often assessed with certainty only after hot cell exams. In this objective, hot cells are and will always be needed in the nuclear industry.

## Basic design studies and fuel performance code development

Basic design studies strongly rely on fuel performance code calculations. The reliability of the calculations is directly connected with the quality of the code validation process [2], [3]. In this framework, the needs for valuable hot laboratories exams are of the highest importance.

### Fuel performance code models development and qualification

EDF fuel performance code CYRANO3 is composed of algorithms solving thermal and mechanical equations and using empirical and analytical formulae to simulate the numerous physical, chemical, metallurgical and neutronic phenomena induced in the fuel rod under irradiation. The fundamental fuel physical models are implemented in the code. These models are established on separate parameter measurements (e.g. thermal diffusivity, annealing, sintering, gaseous diffusion…) performed on both fresh and irradiated fuel pellets and on cladding mechanical models identified by mechanical tests on as-fabricated and irradiated cladding tubes out and under irradiation for different fast neutrons flux levels, strain/stress loadings and temperature ranges. The chemical evolution of the material has strong consequences on the local physical properties. Therefore, all improvement in the localisation of species (EPMA, SIMS) and in the nano-scales observations and characterisation (SEM, TEM, high energy DRX) is welcome to improve the comprehension of the basic mechanisms.

After implementation into the code, the models follow a verification stage during which the results of each model are compared with some analytic solutions in order to detect any errors in the coding process. This step guarantees the reliability of the models in the specific conditions of the code environment. Then the models qualification phase ends with the verification of the numerical (convergence of the code, stability of the solving schemes) and phenomenological aspects.

## Validation process of the fuel performance code

The goal of the validation process is to demonstrate that the fuel performance code is able to predict properly the global fuel rod behaviour in industrial PWR operating conditions and that the code is then available to carry out safety studies.

The validation process is performed for each "macroscopic model" (corresponding to the main physical phenomena directly checked by rod safety rules) in which specific models are included, in particular:
The thermal models, guaranteeing that the temperatures (of the cladding, of the fuel and in the void volumes) are correctly predicted;
The mechanical models, confirming that the fuel stack and the cladding dimensional evolution are accurately simulated;
The fission gas release and internal pressure models;
The corrosion models.
In order to perform the validation process of each "macroscopic model", the entire experience feedback is used, checking that the relevant parameter is correctly simulated.

## The importance of a well-characterised experimental database

The origin of EDF extensive PWR rod behaviour feedback used to build the validation database is twofold. EDF fuel suppliers (AREVA and EFG) commercial rods irradiated during survey programmes in French and international PWR are used as well as rods irradiated in CEA experimental reactors (SILOE, OSIRIS) and within the framework of international programmes (Halden Reactor Project in Norway, Studsvik in Sweden, BR3 in Belgium,…).

<u>Database from experimental reactors</u>
Instrumented irradiations in experimental reactors provide EDF with great-value data for models and code validation. Thanks to the online recording of the macroscopic parameters (fuel temperature, internal pressure, cladding elongation,…) the evolution of the main physical parameters through irradiation can be assessed and the models of the code can be challenged as functions of burnup. However, as the final quality of the code will directly depend on the quality of the database, the code developers have to be very careful when using data from experimental reactors. The data which are kept for the validation process must be recent and well characterised. The traceability of these data as well as a very accurate evaluation of the experimental uncertainties are essential to include the results in the final code database.

The data from experimental reactors selected by EDF to enhance CYRANO3 database are coming from CEA experiments performed in SILOE (Grenoble) and OSIRIS (Saclay) reactors, from the Studsvik Reactor (Sweden), from the Halden Reactor Project (Norway) and from BR3 Reactor (Belgium). The data from these French and international R&D programmes have been crucial for the validation process of the thermal models of UO2, MOX and UO2-Gd2O3 fuels. In addition to the data that they have provided, some of these experimental tests have also been used to develop design methodologies. For instance the alternative rod pressure design methodology based on the "no gap re-opening" criterion is based on the results of the IFA-610 series performed in Halden [4].

<u>Database from industrial PWR rods</u>
Finally, it is necessary to qualify the code to simulate all kinds of fuel rods irradiated in EDF PWRs. As the irradiation conditions may differ from the experimental reactors to the industrial ones (lower fast flux in experimental reactors, different moderator temperature and pressure), it is necessary to include some extensive results from industrial PWR rods in the code validation database. Actually the rods irradiated in test assemblies in commercial reactors can provide useful data from pool-side measurements (rod length and outside diameter, corrosion thickness) and also from the exams (non destructive and destructive: rod and fuel stack length, cladding diameter, fission gas release, internal pressure, free volume, corrosion thickness, density) performed on rods extracted during fuel handling

and shipped to hot cells. In the future the development of methods allowing to perform standard hotcells measurements such as rod internal pressure directly on-site with non-destructive techniques is of great interest.

Historically, AREVA NP used to be the exclusive fuel supplier for EDF industrial reactors. That's why a large amount of experimental data on rods designed by AREVA NP is now available in CYRANO3 database. These data include rods with UO2, MOX and UO2-Gd2O3 pellets and Zircaloy-4 and M5 cladding. However, EDF has chosen to diversify its fuel suppliers and some WESTINGHOUSE fuel assemblies are now loaded in French PWRs. Consequently CYRANO3 database has been extended with data from industrial PWR rods designed by EFG (European Fuel Group). Some fuel products made by ENUSA, WESTINGHOUSE or ABB and irradiated in French and international reactors have been studied and included in EDF experience feedback. These rods have UO2 pellets and Zircaloy-4 or ZIRLO cladding. Given to EDF extensive experience feedback, the current industrial version of CYRANO3 is validated for the following burnup levels:

| Product | | Maximum Rod Burnup (GWd/tM) |
|---|---|---|
| Fuel | Cladding | |
| $UO_2$ / $UO_2$-$Gd_2O_3$ | Zircaloy-4 | 65 |
| $UO_2$ / $UO_2$-$Gd_2O_3$ | M5 | 70 |
| MOX | Zircaloy-4 / M5 | 60 |
| $UO_2$ | ZIRLO | 70 |

Table 1: BU levels of CYRANO3 code validation

## Class 2 transients, power plant manoeuvrability

As a consequence of the high percentage of electricity from nuclear power plants in the global EDF electricity production (86%), the plants have to follow the load of the electrical network, leading to a significant plant manoeuvrability need: frequency control, daily load follow, extended reduced power operations. In this particular context, French nuclear safety authority requirements impose that no fuel failure by PCI should occur during normal operation, as well as during incidental situations.

### Power ramp tests in MTR

The adequate methodology developed by EDF and AREVA to take into account the PCI class 2 risk during the operating phase [5] needs for power ramp tests performed in test material reactors and also mechanical tests on fresh and irradiated claddings. The power ramp tests are performed in experimental reactor such as OSIRIS at CEA or former R2 at Studsvik on pre-irradiated fuel segments. These experiments require test loops which can combine PWR irradiation conditions together with typical class 2 power transients. To be able to transpose these tests conditions to an industrial PWR, the ramp tests must be well-specified with low uncertainties, especially on linear power levels.

In case of rod failure during the power ramp test, post-test exams are unavoidable to assess the fracture localisation and mechanism. These exams also allow to detect any crack initiation process that has not led further to an open crack. Visual inspection, profilometries, eddy current defect determination and metallographies of the failed cladding are usually performed to confirm whether SCC-PCI is the failure cause or not. Furthermore ceramographies are performed so as to investigate the pellet crack pattern and the dishings filling after the ramp test, both data providing useful information in the perspective of understanding the pellet mechanical behaviour.

### Models development and qualification for PCI applications

So as to qualify the fuel performance code used within the methodology high temperature creep tests and high stresses hardening-relaxation tests have to be performed on both as-received and pre-irradiated cladding materials. These mechanical tests have to cover a wide range of conditions in terms of cladding temperature and hoop stress.

If some tests on irradiated fuel claddings are performed, the fuel performance code used for PCI studies uses pellets mechanical models based on fresh materials. Some improvements in the power ramps simulation can be obtained in the years to come by assessing the evolution of the mechanical properties of pellets under irradiation. That's why hot laboratories exams should be able to assess properties such as hardness, tensile behaviour, elasticity module or visco-plasticity on irradiated pellets [6], [7]. The estimation of pellets gaseous swelling under typical class 2 transients would be also of great interest regarding fuel rod PCI simulation improvement.

## Hot laboratories exams as the basis of safety studies

### Fuel behaviour during a LOCA transient

A LOCA transient scenario consists of a blowdown, a cladding heat-up in steam cooling conditions resulting in clad ballooning up to burst, a high temperature oxidation phase, and a quench. The different phases of this scenario as well as the post-quench phase are currently studied using mainly separate effects tests such as temperature ramp tests on cladding, HT oxidation furnaces, quench devices or ring compression tests. As it has been proven according to comparable tests carried on irradiated materials that pre-hydrided materials are good surrogates for irradiated claddings, these tests do not specifically need for hot cell capabilities.

However these tests do not cover all the phenomena occurring during a LOCA transient and complementary issues such as fuel relocation, oxygen diffusion through the cladding or source term evaluation need for experiments on irradiated fuel rods. Regarding fuel relocation process, LOCA tests on pre-irradiated segments like those performed in the Halden Reactor can provide experimental values of the filling ratio of the clad balloon for different burn-ups and different fuel microstructures. These data are then needed to calculate more precisely the cladding temperature in the LOCA safety studies. Besides the oxygen diffusion from the fuel-clad bounding through the cladding can only be estimated using hot cell examinations on a fuel rod after a LOCA transient. The rod fission gas behaviour during a LOCA transient has also to be investigated through hot cells experiments: the axial circulation of the gases within the fuel rod and the gas release (the source term) for different fuel types and burn-ups are key issues regarding the radiological consequences evaluation of a LOCA accident.

### Fuel behaviour during a RIA transient

RIA transients are sudden extra-reactivity increases in the core and may be characterised by the energy deposit within the fuel and the transient peak duration. Some RIA tests have been performed in CABRI, NSRR and BIGR reactors. These tests are used to assess the fuel behaviour during the transient and to develop models implemented in the codes used for RIA safety studies. The parameters of interest in RIA studies are the PCMI cladding behaviour (especially the impact of hydrogen content) and the pellets thermal behaviour (fuel swelling, fission gas release, influence of the rim structure). As these parameters depend on both burnup level and cladding material type, the need for performing adequate RIA tests is connected to the changes in the fuel management schemes (increasing burnup for instance) or to the use of new cladding materials. The qualification of new fuel products has also to be completed by specific mechanical tests on the cladding.

In both RIA and LOCA tests, the post-test hot cells expertise is also of the highest importance to understand the related mechanisms.

### Fuel behaviour during a severe accident

During a severe accident fuel assemblies are strongly deteriorated as the core initial geometry is challenged. Severe accidents sequences lead to fuel issues such as containment integrity and fission product release. Consequently hot cells experiments may be needed to ensure codes upgrading regarding fuel degradation mechanisms for both UO2 and MOX pellets. The investigation of air ingress impact on fuel degradation and fission gas release has also to be considered. Hot cells can then be used either to perform separate effects tests or to complete more integral in-pile experimental programmes to understand the different mechanisms and to develop and validate codes models.

## A database to lifetime plan assessment

Plant lifetime assessment has become a major industrial issue with the increase of the age of operating nuclear power plants. The behaviour under irradiation of nuclear power plants components is studied through material research programmes. Hot cells exams are used to generate long-term database on potential reactor vessel internals degradation mechanisms such as irradiation assisted stress corrosion cracking or irradiation enhanced stress relaxation. To build a significant database on these topics, EDF is referring to international R&D programmes (these questions are studied in the Halden Programme for instance) and has also its own analysis capabilities in the Chinon hot cells laboratories.

## Transport – Interim / short term disposal

Irradiated fuel behaviour during transportation can be assessed thanks to hot cells experiments. Due to a higher cladding temperature and to air environment, the mechanical stability of the fuel rod can be challenged. Hot cells exams allow to determine cladding creep laws under these specific conditions. The hydrogen distribution will also be a key parameter regarding cladding integrity preservation and need to be assessed on irradiated fuel rods. The pellets thermal behaviour (swelling, source term) needs also to be experimentally evaluated to be able to calculate the stress applied on the cladding. Hot cells will still be needed in the future to examine these different issues for new cladding materials. Some improvements can be obtained on the fuel assembly structure (grids, welds) behaviour evaluation too.

## GEN IV

Future needs on GEN IV fuels is so far a vast and difficult to anticipate topic. For sure hot cells exams will be strongly needed for the materials selection process. Furthermore the implementation of GEN IV technologies will imply new qualification procedures as other accidental situations may rise from these systems.

## Conclusion

Nuclear power plants operation requires hot cells examinations on fuel to support survey programmes, fuel expertise, basic design methodologies development, manoeuvrability optimisation and safety studies. Besides materials behaviour studied in hot cells are of the highest importance for lifetime assessment of nuclear components, transport and disposal. All these fuel issues should keep a significant part in the future.

Hot cells will also be needed for new fuel products development and qualification. For instance the qualification of high Pu content MOX pellets or advanced cladding materials needs for hot cells experiments.

Separate effects tests and integral tests performed in hot cells are providing some key elements for the next evolutions of the safety criteria. Current discussions on RIA and LOCA criteria revision may indeed involve new out-of-pile and in-pile experiments.
It can be concluded that there are still some major requirements on hot cells exams regarding all fuel issues. If the needs of GEN IV fuels development have to be anticipated, the requirements on the current fuel generation are still very important, as some significant operating margins still exist on fuel performance. If the effects of an increasing burnup have been heavily documented in the past years, some experiments will still be needed on the evaluation of the fuel behaviour under new more demanding boundary conditions, as increasing power.

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