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## NEW LOCA AND RIA FUEL SAFETY CRITERIA IN FRANCE

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**ABSTRACT:** *Based on technical assessment of IRSN, the French Advisory Committee for nuclear reactors (GPR) recommended in 2010 in the frame of the French LOCA rulemaking to examine the validity of the safety requirements related to the two modes of fuel degradation during LOCA (ductile mode and brittle mode) in the light of state-of-the-art of the last thirty years. In particular, a strength-based approach including an additional axial loading to be applied to the fuel rod during the quench phase was accepted. Then, the second GPR meeting on this topic held in 2014 focused on redefining an acceptable safety limit concerning the brittle mode based on the LOCA semi-integral tests developed by JAEA. Especially, IRSN assessed the new ECR criterion proposed by EDF, which is now expressed as a function of in-reactor hydrogen pick-up combined with the historical 1204°C criterion. It was also the opportunity of concluding on the physical phenomena to be taken into account in a new methodology to analyze LOCA. The safety assessment of this method has been made by IRSN in 2015.*

*During RIA, the fuel rods are subjected to a mechanical loading due to thermal expansion of the pellets and increase of internal pressure. The first phenomenon can lead to PCMI clad failure and the second one to clad ballooning up to burst. Based on interpretation of specific integral tests, EDF proposed to revise and complete existing RIA fuel acceptance criteria regarding PCMI-induced clad failure: they are expressed by enthalpy variation and pulse width; their limits depend on hydrogen content. Moreover, EDF developed an approach which aimed at demonstrating that the control rod ejection accident initiated at zero power is the most limiting. Lastly, EDF has developed an approach based on the actual number of failed fuel rods instead of the number of fuel rods experiencing DNB. IRSN reviewed EDF's proposals and presented its conclusions to the GPR in June 2017.*

**KEYWORDS:** *Fuel safety criteria in France, LOCA, RIA, REA.*

### I. Background

The reactor safety depends on three essential safety functions which are reactivity control, core cooling and fission products containment. Fuel is involved in each of these three safety functions:

- Reactivity control is ensured notably by the motion of rod cluster control assemblies requiring not to exceed a limited deformation of the fuel assembly structure,
- Fuel cooling requires not to exceed a limited deformation of the geometry of fuel clads array,
- Fission product containment is primarily ensured by the first barrier integrity.

A main issue in the safety demonstration for French PWRs is to respect the objectives related to the behavior of the containment barriers, which depend on Plan Condition Category (PCC). For instance, for postulated accidents (PCC-4) such as Rod Ejection Accident (REA) and Loss-of-coolant Accident (LOCA), the objectives associated with the first barrier are the following:

- The number of damaged fuel rods must be limited,
- The geometrical structure of the core must not be damaged so as to ensure an adequate core coolability.

For each category and type of transients, these objectives are then expressed as requirements associated to the limitative physical phenomena occurring during the transient. Afterwards, the requirements are supported by fuel safety criteria that are limits on computable metrics representative of the relevant physical phenomena.

The French acceptance criteria related to the first barrier (and notably their bounding values) came in the 1970s from Westinghouse's license. At that time state-of-the-art and computing capacities led to establish decoupling criteria enabling to implement simplified and robust approaches to analyze the more complex and severe accidental conditions. Moreover, to maintain the core coolability, requirements are based on either fuel rod cladding integrity or the absence of fuel dispersal in the primary coolant during the transient. Indeed, such requirements avoid especially studying the impact of hot or melt fuel-water interaction on core cooling, the impact of fuel deposit on the vessel bottom on criticality or the impact on the Emergency core cooling injection and recirculation strainers of fuel dispersal in the primary coolant.

Since the French nuclear program was initiated, both operating experience, experiments carried out by operators and research institutes as well as international R&D programs aimed at improving the calculation method allowing continuous development of the knowledge and collecting experimental results, especially in RIA (Reactivity-initiated accident) and LOCA conditions. Moreover, new types of alloy cladding characterized by enhanced performances, such as M5, ZIRLO and Optimized ZIRLO<sup>1</sup> (e.g. cladding corrosion in normal operating conditions), have been introduced in French PWR. In this context, the fuel safety criteria especially those addressing LOCA and RIA have been reviewed from 2011 to 2017: in the French regulatory framework, new fuel safety criteria are suggested by the French operator EDF on request of the French Nuclear Safety Authority (ASN) and submitted to it for approval. The safety assessment of EDF's proposals is made by Institute of Radiological Protection and Nuclear Safety (IRSN).

In this way, the paper describes the evolutions of LOCA and RIA fuel safety criteria, pending questions in the field of safety studies and ongoing programs. The new regulation is assumed to be applied for EDF's French PWR (except for EPR) and for claddings loaded in these reactors (except for Zircaloy-4).

## **II. LOCA fuel safety criteria**

### **II.A. Introduction**

A LOCA caused by a postulated coolant pipe break on the primary loop leads to a sudden depressurization and a loss of water inventory of the primary circuit. The fast pressure drop leads to large horizontal and vertical hydraulic loads on the internal structures of the reactor vessel and on the fuel assemblies. Nevertheless, these components must retain their geometries to ensure the shutdown of the reactor by control rod cluster drop and to maintain core coolability. The loss of water at the break may cause a partial or total uncovering of the core which can lead to damage fuel rods up to the burst of a number of them. The progression and consequences of a LOCA transient in terms of hydraulics and fuel behavior are directly related to the location and the size of the postulated break on the reactor coolant system.

During a LOCA, the fuel can be damaged according to two modes, as followed.

During the core uncovering, the clad temperature increases up to values about 750-800°C. From a thermal mechanical viewpoint, due to the pressure drop in the primary system, fuel rods are subjected to internal pressure higher than the external one. Thus, under the effect of tensile stresses and high temperatures reached during the transient, clads are able to balloon up to burst for some of them, which may lead to fuel dispersal in the primary coolant. Some recent R&D programs showed an accumulation of fuel fragments in the volume of the ballooned and burst fuel rods. The fuel relocation can significantly modify the local heat generated in the fuel rods and tends to increase locally their temperature. Moreover, for the highly irradiated fuel, a dissemination of fuel particles outside the fuel rod was observed after its burst. This phenomenon is called fuel dispersal Refs. 1, 2, 3, 4. Moreover, the clad ballooning phenomenon leads to blocked geometry, which can under a number of conditions reduce the heat exchange surfaces between the fuel rods and the coolant and redistribute coolant flows, jeopardizing core coolability Refs. 5, 6, 7. The so-called ductile mode deals with phenomena of fuel rods ballooning and burst associated with fuel relocation.

During the reflooding, clad temperature reaches about 800-900°C, cladding oxidation reaction in steam phase (with or without pressure depending on the break size) speeds up and the transient oxide growth becomes significant Refs. 7, 8. For

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<sup>1</sup> M5 is the reference alloy designed by AREVA while ZIRLO and Optimized ZIRLO are Westinghouse's alloys (the historical Zircaloy-4 cladding is no longer loaded in EDF's reactors since early 2017).

the ballooned and burst fuel rods, a significant amount of hydrogen produced during transient oxidation is absorbed at the inner side of the fuel rods cladding Ref. 7. This is the phenomenon of transient secondary hydriding Refs. 9, 10. The so-called brittle mode deals with the high temperature cladding oxidation in steam environment including transient hydrogen pick-up which is the leading physical phenomenon susceptible to alter the fuel rods strength.

Finally, under the effect of the thermal shock following the quench, the application of stresses on embrittled fuel rods may lead to their failure, jeopardizing core coolability. The thermal stresses due to the quench may be added with additional mechanical loads. The origins of such loads are discussed in detail in the paper Ref. 11.

Besides, the additional mechanical loads occurring during the long term of the LOCA transient, such as seismic forces, may have an effect on the fuel behavior.

### II.B. Before the French rulemaking

In France, since the start of the PWR program, break sizes up to the double-ended guillotine break (2A) have been postulated to analyse some of the consequences of a LOCA transient such as core coolability, resistance of the reactor containment and radiological consequences. For other consequences, like the mechanical resistance of the reactor vessel internal structures and fuel assemblies, only limited break sizes considering pipe whip restraints have been considered.

At the start of the French PWR nuclear program, the fuel acceptance criteria established by the AEC (now the U.S. NRC) in the 1970s were adopted in France, notably the clad oxidation rate (which must remain lower than 17%) and the peak cladding temperature or PCT (which must remain lower than 1204°C i.e. 2200 F) criteria Ref. 12. It is important to underline that for the ductile mode the requirement of maintaining a coolable geometry had not been formally expressed in a decoupling criterion and contrary to RIA (see §III.C), there is no formal requirement on fuel dispersal in the primary coolant.

In 1994, EDF chose to calculate Equivalent cladding reacted (ECR) taking into account the in-service corrosion and the transient corrosion. In 1999, the original requirement of a residual cladding ductility at the end of transient, based on Ring Compression Tests (RCT), changed into a requirement of fuel rod quenching resistance without additional load, based on leakage tests carried out on cladding sections oxidized at high temperature and having undergone a thermal shock quench Ref. 7. This change did not call the original criteria in PCT and ECR into question.

To sum up, Table 1 recalls the objectives, requirements and safety criteria related to fuel behavior, respectively during LOCA transients before the French rulemaking. These criteria were applied for all fuel assembly designs loaded in the French PWRs.

TABLE 1. Objectives, requirements and fuel safety criteria related to LOCA transients before the rulemaking.

OBJECTIVES	REQUIREMENTS	FUEL SAFETY CRITERIA
The geometrical structure of the core is not damaged so as to ensure an adequate core coolability.	No clad failure during the quench due to cladding embrittlement by high temperature oxidation (based on French analytical quench tests without mechanical load in DEZIROX facility)	Mean clad temperature <1204°C Oxided clad layer thickness (ECR <sup>2</sup> )<17% of the clad thickness
	Preservation of a coolable geometry	No specific quantified criterion
	No fuel dispersal in the primary coolant	∅
Criterion linked with radiological consequences assessment : failed fuel rods rate = 33% (value came from EURATOM recommendation Ref. 13)		

### II.C. About the French rulemaking and the new regulation

Compared to the situation forty years ago the discharge burn-up of the fuel rods has increased notably. It has led to increased oxide thickness and higher hydrogen uptake in the fuel cladding, which influence its behavior under LOCA conditions. Since then, the behavior of fuel in LOCA conditions has been the subject of research and development programs.

<sup>2</sup> ECR: Equivalent Cladding Reacted.

The vast majority of these programs were focused on situations representative of large break LOCA scenarios. Thus, recent research findings have identified new phenomena under LOCA conditions with increasing burn-up, such as in particular the embrittlement mechanisms of fuel rods due to oxygen and hydrogen pickup Ref. 9 and the Fuel Fragmentation, Relocation and Dispersal (FFRD) Refs. 1, 2, 3. Results of numerous experiments are also used to develop and improve the predictive models of software simulations of LOCA transients. These new physical phenomena related to fuel highlighted by R&D studies conducted during the last decade have to be taken into account in the frame of the safety analyses. Moreover, the operating conditions of the French plants have changed (notably by stretch-out operation conditions) and new cladding materials have been introduced.

Given these evolutions, ASN decided to review the LOCA safety demonstration concerning core coolability encompassing the definition of the LOCA reference transients, the LOCA analysis method and the physical phenomena to be taken into account and LOCA safety requirements associated with safety limits to be verified.

The objectives of Refs. 14, 15 were to discuss about the evolution of LOCA reference transients and the development of a new method to verify core coolability. In particular, EDF takes now into account the same break sizes limited by pipe whip restraints for both thermal-hydraulic and mechanical analysis. Regarding the thermal-hydraulic analysis, it is important to emphasize that such an evolution of LOCA reference transients leads to focus on a better modeling of the physical phenomena for Intermediate Breaks (IB) LOCA conditions rather than focus only on the 2A break.

Based on IRSN's technical assessment, in 2010, the French Advisory committee for nuclear reactors (GPR) examined the safety requirements relating to the two modes of fuel degradation in the light of state-of-the-art since the last thirty years. In particular, a strength-based approach including an additional axial loading to be applied to the rod during the quench phase was accepted Ref. 16. Then, a second GPR meeting held in 2014 focused on redefining an acceptable safety limit concerning the brittle mode based on integral LOCA tests developed by JAEA. It was also the opportunity of concluding on the physical phenomena to be taken into account in the safety demonstration Ref. 17. In 2015 and 2016, the new LOCA analysis method proposed by EDF using CATHARE 2 method system code and a statistical approach was reviewed by IRSN. This method so-called CathSBI will be first applied for the fourth 10-yearly safety review of EDF's 900 MWe nuclear reactors starting in 2017 Ref. 18.

The main conclusions of the rulemaking are summarized below.

Regarding the ductile mode, the issue of an acceptable demonstration of the core coolability was widely discussed during technical meetings between EDF and IRSN. IRSN summarizes its technical positions, as the result of detailed reviews of the existing technical basis Refs. 5, 6, as follows.

Concerning the formation of a flow blockage, even if axial and azimuthal temperature heterogeneities are expected to limit local cladding deformations, the contacts between neighbouring fuel rods would tend to homogenize the temperature in a plane section favouring the extension of the deformation in the axial direction. In addition, axial power profiles on neighbour rods will likely induce peak deformations at approximately similar axial locations, which could favour a significant coplanar blockage. Related to the coolability of a partially blocked zone, the detailed review of the experiments performed in the 1980s had allowed identifying, in separate tests series, the main parameters that influence the cooling process in the blocked zone; but the results of these experiments do not allow quantifying a coolability limit.

With respect to the approach based on a 1D calculation with CATHARE code Ref. 19, the core coolability is verified by a calculation of the cladding temperature during the transient, which has to be less than 1204°C. Nevertheless IRSN underlined that the current EDF's approach did not allow taking into account the effects of contacts between neighbouring rods. The calculated temperature rises in such blocked regions are thus underestimated and may exceed the PCT limit. As required Ref. 17 in 2014, the new safety demonstration proposed by EDF takes into account the negative effects of ballooning, burst and contacts between fuel rods.

Also, EDF had to model the impact of fuel relocation phenomenon in calculations performed by the CATHARE software to verify core coolability. Moreover, the risk of fuel dispersal in the primary coolant during LOCA was not established with current core loadings and fuel assembly burn-up limit (52 GWd/tU) in EDF's reactors. Indeed, the current non-dispersal threshold in terms on local burn-up is based on Halden and Studsvik LOCA tests and is established at 61 GWd/tU. But more research and detailed analyses are required, in particular for MOX fuel which is characterized by a specific microstructure.

In comparison with the current deterministic method, the fuel behavior modeling in the new EDF's method is improved and more accurate by taking into account more physical phenomena that were so far either not properly modelled or not taken into account in the models Ref. 18:

- Clad ballooning and burst: Modeling is underway by EDF to improve the rupture criterion and to cover the IB LOCA heating rates,
- Coplanar clad strain: This parameter is subjected to a statistical approach based on a range of variations,
- Blockage of fuel channels hydraulics: The important enhancement of the new method consists in taking into account the balloon length. This parameter is subjected to a statistical approach based on a range of variation,
- Thermal exchanges between fuel rods and primary coolant and the reduction of the exchange surface due to the contacts between the rods: The new EDF's approach is a notable improvement,
- Transient clad oxidation: The pressure effect due to IB LOCA transient on the clad high temperature oxidation kinetic is taken into account,
- Possible accumulation of fuel fragments in the ballooned section of the fuel rods (fuel relocation phenomenon): Various models are developed to take into account relocation consequences on clad temperature. Some parameters of these models are subjected to a statistical approach based on a range of variation.

IRSN's assessment was focused on the validity of the statistical approach proposed by EDF and on the fuel modeling qualification. IRSN's review identified a lack of justification for:

- Some uncertainties regarding for instance the balloon length, the parameters linked with the fuel relocation modeling (fuel fragments size, packing ratio in the ballooned region),
- Some qualification elements, for instance the modeling of the thermal conductivity of fuel fragments into the ballooning volume.

To conclude, the new EDF's CathSBI methodology includes noteworthy enhancements, but EDF still has to improve its methodology in order to justify the previous points.

Regarding the brittle mode, ASN asked EDF to review the way of defining the original limits (17% ECR and 1204°C PCT). The main motivation was to integrate into the LOCA limits definition several physical phenomena that were not represented in the historical RCT approach such as transient secondary hydriding, axial loading during quench, wall thinning and hydrogen taken during normal operation.

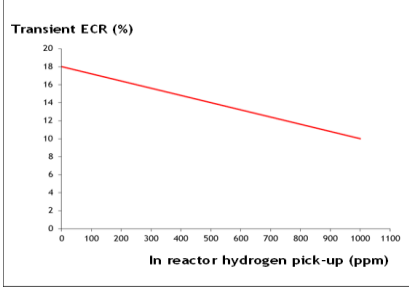
During the GPR meeting in 2010, the safety principle of a strength-based approach, based on the LOCA semi-integral tests developed by JAEA Refs. 10, 20, 21, including an additional axial loading to be applied to the rod during the quench phase was accepted. A new French transient ECR criterion, expressed as a function of in-reactor hydrogen pick-up and combined with the historical 1204°C peak cladding temperature criterion, was proposed by EDF. The revised LOCA limits were accepted in 2014 Ref. 17. Nevertheless, in order to confirm the validity of the new transient ECR criterion compared to JAEA tests representativeness and sufficiency, EDF shall provide complementary elements based on experimental results. Details of the proposed approach and the new LOCA limit can be found in Ref. 11. In addition, in 2014, it was stated that EDF shall demonstrate that an earthquake occurring during the phase of long-term cooling after LOCA does not prevent the core cooling. Indeed, the occurrence of this external hazard cannot be excluded in this phase.

#### **II.D. Conclusion and pending questions**

To conclude about the new LOCA methodology to study the core coolability, IRSN considered in particular that (see Table 2):

- Regarding the brittle mode, the new transient ECR criterion, expressed as a function of in-reactor hydrogen pick-up ([H]) and combined with the historical 1204°C peak cladding temperature criterion was acceptable. These criteria aim at checking the new strength-based requirement including an additional axial loading to be applied to the fuel rod during the quench,
- About the ductile mode, the new fuel behavior modeling was improved and more accurate by taking into account more physical phenomena. Nevertheless, EDF shall improve its current LOCA method to justify various modeling parameters. Moreover, fuel dispersal in the primary coolant is not a safety concern with current core loadings and fuel assembly burn-up limit (52 GWd/tU) in French reactors,
- EDF shall demonstrate that an earthquake occurring during the phase of long-term cooling after LOCA does not prevent the core coolability.

TABLE 2. Objectives, requirements and fuel safety criteria related to LOCA transients following the rulemaking.

OBJECTIVES	REQUIREMENTS	FUEL SAFETY CRITERIA
<p style="writing-mode: vertical-rl; transform: rotate(180deg);">The geometrical structure of the core is not damaged so as to ensure an adequate core coolability.</p>	<p>No clad failure during the quench including an additional axial loading during the quench (based on Japanese integral LOCA tests achieved 530 N – unirradiated, pre-hydrated and irradiated specimen)</p>	<p>Peak cladding temperature at the hot spot &lt;math&gt;&lt;1204^{\circ}\text{C}&lt;/math&gt;  <b>Transient ECR=f([H]):</b></p> 
	<p>Preservation of a coolable geometry</p>	<p>No specific quantified criterion but <b>improved fuel behavior modeling</b> by taking into account more physical phenomena</p>
	<p>No fuel dispersal in the primary coolant</p>	<p>Fuel dispersal is <b>not a safety concern</b> in France with current core loadings and FA BU limit (52 GWd/tU).</p>
<p>Criterion linked with radiological consequences assessment : failed fuel rods rate=<b>33%</b> Ref. 13</p>		

PCT and ECR calculated without taking into account all the fuel physical phenomena may be underestimated: the first results obtained by this new method show a significant sensitivity of the cladding temperature to the input parameters, appearing from 750 to 800°C. This sensitivity is linked to the fuel rod phenomena (ballooning, burst, blockage, contact between rods and fuel relocation) activation, when the difference between cladding internal and external pressures is high enough and for cladding temperature levels above 800°C: their cumulative effects lead in studies with CathSBI method to a significant increase of the cladding temperature (350 to 500°C). This negative effect is partially compensated by the use of a statistical method instead of a deterministic one and by the use of new thermal hydraulics models for the vessel. However, the validation of this statistical method and of new thermal hydraulics models still raises open issues.

Given the high sensitivity to input parameters for LOCA cases reaching cladding temperature levels above 800°C with high enough difference between cladding internal and external pressures, it is difficult to conclude with a high level of confidence that the peak cladding temperature meets the safety criterion (1204°C). This issue of PCT range with high sensitivity, must be addressed by the methodology of EDF under development in order to ensure the robustness of LOCA safety analyses. As the CathSBI method will be first applied for the fourth 10-yearly safety review of EDF's 900 MWe nuclear reactors starting in 2017, technical exchanges between EDF and IRSN are under way.

Finally, regarding on-going LOCA investigations and research programs, IRSN considers namely that:

- Fuel relocation modeling needs to be improved and clad behavior modeling needs to be adapted to take into account claddings loaded in French NPP's and the specific characteristics of IB LOCA transients,
- Some physical phenomena which may enhance cladding embrittlement during high temperature oxidation, such as the internal pre-oxide dissolution during LOCA need to be more analysed,
- Additional experiments have to be carried out to obtain data on MOX fragmentation and dispersal behavior during LOCA conditions and define more precisely the non-dispersal limit in terms on burn-up based threshold. Moreover, investigations are needed to conclude on the limiting test conditions of FFRD dedicated experiments,
- The question of the post-quench tests relevancy to justify a well post-LOCA clad behavior after a seismic event needs to be asked.

### **III. RIA fuel safety criteria**

#### **III.A. Introduction**

A Reactivity-initiated accident (RIA) is caused by a control Rod ejection accident (REA), which is defined as the mechanical failure of a Rod cluster control assembly (RCCA) drive mechanism casing, located on top of the reactor pressure vessel. The RCCA is ejected vertically from the reactor core due to the high coolant pressure. Such a RIA is characterized by a very rapid increase of reactivity and power in some rods of the reactor.

First, the rapid increase in power leads to nearly adiabatic heating of fuel pellets, which induces thermal expansion of the fuel pellets and potentially fission-gas-induced fuel swelling. At this early stage of the transient, the cladding material remains at a fairly low temperature (about 350°C for French PWR), and the thrust imposed by the expanding fuel pellets may lead to stress and strain on the cladding and potentially to failure depending on fuel enthalpy increase and the level of clad embrittlement due to precipitated hydrides (high burn-up fuel rods): this first fuel rod damage is so-called PCMI (Pellet Clad Mechanical Interaction) failure Ref. 22.

At a later stage of the transient, the increase of cladding temperature leads to the Departure from Nucleate Boiling (DNB) and boiling crisis occurrence. In such a case, if the cladding is ductile enough to survive through the stage of the PCMI loading and if fuel enthalpy continues to increase, the cladding can reach a high temperature (>700°C) for several seconds, until the rewetting takes place. During this period, the internal pressure of the fuel rod due to fission gas release can be higher than the external one, which may lead to clad ballooning up to burst (medium burnup fuel rods): this second fuel rod damage is so-called DNB failure Ref. 22.

On the contrary, in the last stage of the transient, if the internal pressure of the rod is low for fresh fuel or low burn-up fuel rods, another failure mode may occur during the rewetting of the clad, due to the application of thermal stresses on overheated fuel clad which is embrittled by high temperature oxidation during the film boiling.

Moreover, if the clad reaches very high temperature in the post-DNB phase (up to clad melting) or if the energy deposited in the fuel is significantly high, clad and fuel pellet melting may occur. Post-test rods showed that the cladding wall near the failure became thin with melting. This type of clad failure refers to fresh or very low burn-up fuel. Indeed, since the fissile content becomes low at a high burnup, the possibility of the pellet melting in high burnup fuels is very weak even taking into account a reduction of the melting point due to burn-up Ref. 22.

#### **III.B. Before the French rulemaking**

At the start of the operation of French NPP's, to avoid the loss of coolable geometry and the generation of coolant pressure pulses, empirical safety criteria for REA ensuring no hot/molten fuel dispersal in the primary coolant were established on the basis of RIA full-scale SPERT-CDC<sup>3</sup> tests carried out at zero-power on fresh and low irradiated UO<sub>2</sub> fuel<sup>4</sup>. Notably, the value of 200 cal/g maximum fuel enthalpy came from Westinghouse's extrapolation of fuel behavior. This criterion was applicable for mean fuel assembly burn-up inferior to 33 GWd/tU.

The maximum peak cladding temperature of 1482°C (2700°F) was used to demonstrate core coolability. This limit was taken from fuel failure boundary for LOCA conditions. The rationale for retaining a higher temperature limit for non-LOCA transients, such as RIA, was that film boiling occurs briefly during those transients, so that the fuel rods could withstand without suffering serious damage Ref. 23.

Moreover, the progressive increase of fuel assembly discharge burn-up led ASN to ask EDF to demonstrate that the previous RIA acceptance criteria are still applicable. Thus, some full-scale tests in the French CABRI<sup>5</sup> test reactor and also in the Japanese NSRR test reactor using high burn-up fuel rods led to fuel dispersal in the primary coolant for fuel enthalpy far below 200 cal/g Ref. 24. These tests had therefore clearly showed that the old RIA criteria were no longer relevant. Based on

<sup>3</sup> SPERT-CDC: Special power excursion reactor – Capsule driver core (USA). The coolant medium is stagnant water, the coolant temperature is 20°C, the coolant pressure is 1 bar and the pulse width is between 13 to 31 ms.

<sup>4</sup> The maximum UO<sub>2</sub> fuel rods burn-up in the integral RIA SPERT tests is 32 GWd/tU.

<sup>5</sup> CABRI: Name of an experimental French reactor where a major research program was jointly carried out by IRSN and EDF – 14 experiments performed in 1993-2002 with sodium coolant loop (CABRI REP-Na): the temperature reached in the fuel rod remained limited due to the use of sodium in the CABRI loop and only the first phase of a RIA transient, the PCMI, can be characterized. That, thereby, prevented from full-scale investigation of the second phase of the transient, the DNB (and post-DNB) – The coolant medium is flowing sodium, the coolant temperature is 280°C, the coolant pressure is 5 bar and the pulse width is between 9 to 75 ms.



the results of the full-scale tests, EDF has established an empirical “safety domain” defined by four parameters (oxide thickness, enthalpy variation, pulse width, clad temperature) which intends to preclude fuel rod failure due to PCMI and ballooning during boiling crisis for high mean fuel assembly burn-up (>47 GWd/tU) Ref. 25. IRSN considered this proposal as acceptable in 2011.

Besides, the number of fuel rod failures must be calculated so that the radiological doses to the public can be estimated. A requirement was defined to limit the number of rods affected by departure of nucleate boiling for this purpose. The conservative assumption is that all fuel rods entering in boiling crisis are assumed to be failed. In addition to that, fuel melting is considered as a cladding failure: all fuel rods that experience fuel melting are assumed to be failed for radiological dose calculation. Nevertheless, a limited amount of fuel melting is acceptable, less than 10% of pellet volume.

Table 3 summarises the objectives, requirements and safety criteria related to fuel behavior, respectively during REA before the French rulemaking. Up to now, these criteria were applied for all the fuel rod designs loaded in the French PWRs up to the fuel assembly burn-up limit (52 GWd/tU) and, for Zircaloy-4 cladding, as long as the oxidation thickness is less than 108 µm.

TABLE 3. Objectives, requirements and fuel safety criteria related to REA before the rulemaking.

OBJECTIVES	REQUIREMENTS	FUEL SAFETY CRITERIA
The geometrical structure of the core is not damaged so as to ensure an adequate core coolability.	No fuel dispersal in the primary coolant linked with cladding and fuel melting	<u>Mean fuel assembly BU&lt;33 GWd/tU:</u> Maximum fuel enthalpy<200 cal/g
	No fuel rod failure due to PCMI and ballooning during boiling crisis	<u>Mean fuel assembly BU&gt;47 GWd/tU:</u> - Oxide thickness<108 µm - Enthalpy variation<57 cal/g - Pulse width>30 ms - Clad temperature<700°C
	No clad failure during the rewetting due to cladding embrittlement by high temperature oxidation	Maximum clad temperature <1482°C
The number of damaged fuel rods must be limited. ⇒ Objective linked with the radiological assessment	Limited number of rods affected by departure of nucleate boiling	Number of rods exceeding the CHFR <sup>6</sup> <10%
	Limited fuel melting	Fuel melting amount<10% of pellet volume

### III.B. About the French rulemaking and the new regulation

As shown by Table 3, it can be underlined that no requirement and fuel safety criterion associated with fuel dispersal in the primary coolant further to PCMI-induced failure and burst during boiling crisis had been defined for mean fuel assembly burn-up between 33 and 47 GWd/tU. Moreover, the SPERT, CABRI and NSRR tests were carried out at hot-zero power while French safety demonstrations require REA studies for all possible initial power level.

Based on IRSN’s technical assessment, the GPR meeting in June 2017 about the French rulemaking on fuel safety acceptance criteria related to PCC-1, PCC-2, PCC-3 and PCC-4 (except for LOCA, see §II.C) led to examine the sufficiency and validity of requirements and fuel safety acceptance criteria relating to all modes of fuel degradation in the light of state-of-the-art, especially for REA transients. The main conclusions of IRSN’s assessment about EDF’s proposals for REA are summarized below.

<sup>6</sup> CHFR: Critical Heat Flux Ratio.

With regard to the 200 cal/g of maximum fuel enthalpy criterion, EDF confirmed its validity on the basis of the CABRI REP-Na9 test carried out on MOX fuel with a low clad corrosion and a fuel rod burn-up of 28 GWd/tU. The tested fuel rod was not failed for a maximum fuel enthalpy of 200 cal/g.

EDF has revised existing fuel acceptance criteria and completed the safety demonstration for fuel assembly burn-up higher than 33 GWd/tU. The new acceptance criteria, expressed by enthalpy rise and pulse width, aim at precluding PCMI fuel clad failure. Their limits depend on cladding corrosion performances, more specifically on hydrogen content which is of interest to cope with PCMI loading. More precisely, EDF's approach to define these new REA acceptance criteria depends on the fuel rods types:

- For UO<sub>2</sub> fuel rods with ZIRLO, Optimized ZIRLO and M5 claddings, the approach is based on the interpretation with for instance SCANAIR code Ref. 26 of some full-scale RIA tests carried out in CABRI and NSRR reactors and associated with PCMI issue. But, the threshold values of enthalpy rise and pulse width are different for M5 cladding than for ZIRLO and Optimized ZIRLO alloys due to their cladding corrosion performances. Regarding M5 cladding, IRSN considered acceptable the 150 cal/g of enthalpy variation criterion. But, concerning the limit of enthalpy rise for ZIRLO and Optimized ZIRLO claddings, it has been identified by IRSN that no uncertainty about experimental data has been taken into account to calculate with SCANAIR code the enthalpy rise of the restrictive test, CABRI CIP0-1. It is important to underline that this test covers hydrogen pick-up of ZIRLO, Optimized ZIRLO claddings loaded in EDF's reactors,
- For MOX fuel rods with M5 cladding, EDF's approach is different from this for UO<sub>2</sub> fuel rods based on the application with SCANAIR code of specific RIA test related to ballooning and burst to PCMI behavior. IRSN considered that the approach was complicated and unsupported. Also, EDF planned to define fuel safety criteria for MOX fuel rods with M5 cladding based on an analysis of specific integral RIA tests devoted to MOX fuel.

For REA initiated at non-zero power levels, EDF developed an approach which aimed at demonstrating that the REA transient initiated at zero power is the most limiting compared to transients initiated at higher power levels. IRSN estimated that EDF's approach, based on the comparison of thermo mechanical parameters calculated with SCANAIR code for the PCMI fuel clad behavior, was acceptable. However, EDF shall apply this approach for each plant series.

Moreover, EDF proposed an approach to demonstrate the absence of fuel dispersal in the primary coolant after clads ballooning and burst during boiling crisis. This approach is based on the comparison between the restrictive PCMI criterion and results of various full-scale tests associated with ballooning and burst (IGR, BIGR, NSRR, PBF - Refs. 27, 28, 29). As compared with available experimental database, no fuel dispersal was noticed up to French fuel rods burn-up discharge limit (57 GWd/tU) and up to enthalpy rise limited to 150 cal/g.

Regarding the maximum temperature criterion of 1482°C, even though EDF planned to complete this current criterion to take into account the duration of cladding oxidation at high temperature by defining ECR criterion expressed as a function of maximum clad temperature (based on DNB tests carried out notably in PBF reactor Ref. 30), REA is not a relevant accident because of the short time of boiling crisis (less than 10 s).

Moreover, like the NRC's requirement, a limited amount of fuel melting is acceptable provided it is restricted to the fuel centerline region and is less than 10% of pellet volume Ref. 31. Indeed, during a REA transient, due to effects of edge peaked power and lower solidus temperature, fuel rods may undergo fuel melting in the pellet periphery. Thus, fuel melting outside the centerline region is precluded to avoid molten fuel coolant interaction. Therefore, for REA safety studies, EDF shall demonstrate that this requirement is satisfied based on appropriate analysis rules.

### **III.C. Conclusion and pending questions**

Table 4 summarizes the objectives, requirements and safety criteria related to fuel behavior, respectively during REA following the French rulemaking.

TABLE 4. Objectives, requirements and fuel safety criteria related to REA following the rulemaking.

OBJECTIVES	REQUIREMENTS	FUEL SAFETY CRITERIA
<b>The geometrical structure of the core is not damaged so as to ensure an adequate core coolability.</b>	No fuel dispersal in the primary coolant linked with cladding and fuel melting	<u>Mean fuel assembly burn-up</u> <33 GWd/tU: Maximum fuel enthalpy <200 cal/g
	No fuel rod PCMI-induced failure	<b>REA TRANSIENTS INITIATED AT ZERO POWER LEVEL</b> <u>Mean fuel assembly burn-up</u> >33 GWd/tU: - <i>UO<sub>2</sub>/ZIRLO ou Optimized ZIRLO:</i> { $\Delta H$ definition in progress ; $L_{1/2}$ >30 ms} for [H]<1000 ppm - <i>UO<sub>2</sub>/M5:</i> { $\Delta H$ <150 cal /g ; $L_{1/2}$ definition in progress} for [H]<160 ppm, - <i>MOX/M5:</i> $\Delta H$ definition in progress
		<b>REA TRANSIENTS INITIATED AT NON-ZERO POWER LEVEL</b> EDF's demonstration that the REA transient initiated at <b>zero power</b> is the most penalized transient as <b>compared</b> to REAs initiated at <b>higher power</b> levels
	No fuel dispersal in the primary coolant after ballooning and burst during boiling crisis	EDF's approach based on the <b>comparison</b> between the <b>restrictive PCMI criterion</b> ( $\Delta H$ <150 cal/g) and <b>results of integral tests</b> (BIGR) up to French fuel rods burn-up limits (57 GWd/tU)
	No clad failure during the rewetting due to cladding embrittlement by high temperature oxidation	Maximum clad temperature <1482°C (REA transient duration<19 s)
	Limited fuel melting restricted to the fuel centreline region	Centreline fuel melting amount<10% of pellet volume* * : also used as input data to estimate radiological consequences
The number of damaged fuel rods must be limited. ➔ Objective linked with the radiological consequences assessment	Limited number of rods affected by departure of nucleate boiling	Number of rods exceeding the CHFR<10%

To conclude about REA safety studies, the application of new requirements and fuel safety criteria is planned for the fourth 10-yearly safety review of EDF's 900 MWe nuclear reactors and the second 10-yearly safety review of EDF's 1450 MWe nuclear reactors both starting in 2017. The French rulemaking allowed reviewing satisfactorily the RIA acceptance criteria even if the definition of some fuel safety criteria are still under discussion, especially the definition of fuel safety criteria for MOX fuel rods with M5 to preclude PCMI clad failure.

Finally, regarding on-going RIA investigations and research programs, IRSN considers namely that Cabri International Project (CIP<sup>7</sup>) tests planned in the CABRI-water loop facility may be used to confirm the absence of fuel dispersal in the primary coolant after clad ballooning and burst during boiling crisis for high fuel burn-up and will improve knowledge on the MOX fuel behavior, during in particular post-DNB phase.

<sup>7</sup> CABRI CIP: Tests with water coolant loop plan to start in 2018.

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