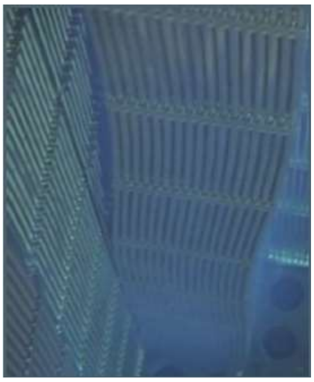
FUEL ASSEMBLY BOWING AND SAFETY DEMONSTRATION



The purpose of the report is to:

(1) review the fuel assembly bowing issue and its consequences on the safety demonstration,

(2) compare rules-making and practices based on feedback of France and Belgium (ETSON countries mainly concerned by the issue)

(3) provide some key messages based on TSO's analyses.

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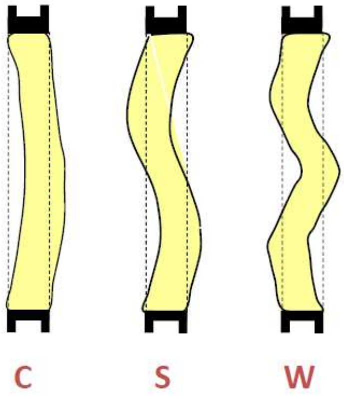
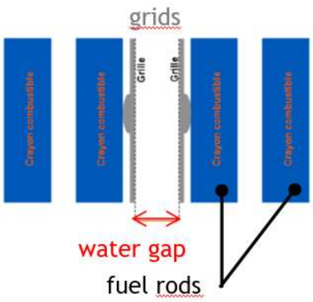
1 Background

Nowadays, an important issue is the current state of Nuclear Power Plants (NPP) cores considering that many reactors around the world are affected by Fuel Assembly (FA) bowing. The FA bowing phenomenon finds its root cause in different physical and technical origins. The FA deforms most commonly in the first bending mode (C-shaped bowing), but bowing in higher bending modes is also observed, such as S-shaped and W-shaped bowings [1][2].





Consequently, due to the collective FA bowing in core, an inter-assemblies' water gaps distribution is formed with gap sizes ranging from zero to a size much larger than the nominal one (about 2 mm for French Pressurized Water Reactors – PWR).

The FA bowing phenomenon has been detected mainly by Incomplete Rod cluster Inserts (IRI) or high Rod Cluster Control Assembly (RCCA) drop times during reactor shutdown. Moreover, excessive out-of-core FA bowing has been measured with specific devices during refueling outages. This is mainly because the bowing occurs gradually, showing that it has occurred when the consequence of it become evident. Thus, there are not early indicators that FA bowing may be underway. Current operational practices do not enable to foresee the occurrence of FA bowing at an early stage and the lack of "early warnings" is part of the reason for which FA bowing has become relevant issue that still remains to be addressed.

Some examples of relevant FA bowing events are given below according to the feedback coming from some ETSON countries.

1.1 Relevant events of fuel assembly bowing from some ETSON countries

In France[1], although FA bowing phenomenon took place since the 2000s, unexpected and noticeable increases of IRI and high RCCA drop times were observed since 2010 for Pressurized Water Reactors (PWR) using 14 feet FA. Especially in 2012, measurements found out significant FA bowing (cycle “n”). This specific situation led ASN to request that EDF carry out additional RCCA drop times measurements[2] to those planned at the beginning and at the end of the cycle, to better follow their evolution during the next cycle (cycle “n+1”). Finally, the noteworthy increased RCCA drop times and the IRI for five RCCA (Figure 1) led EDF to shut down the reactor three months before the date initially planned in 2014. Moreover, some compensatory measures related to the maintain of the core subcriticality in the event of automatic reactor shutdown were taken into account. During the cycle “n+2” in 2015, a reinforced monitoring of FA showed once again IRI. To prevent from RCCA drop anomalies during the cycle, EDF proceeded to an unprecedented operation in the middle of irradiation cycle for replacing two highly bowed FA by new skeletons.

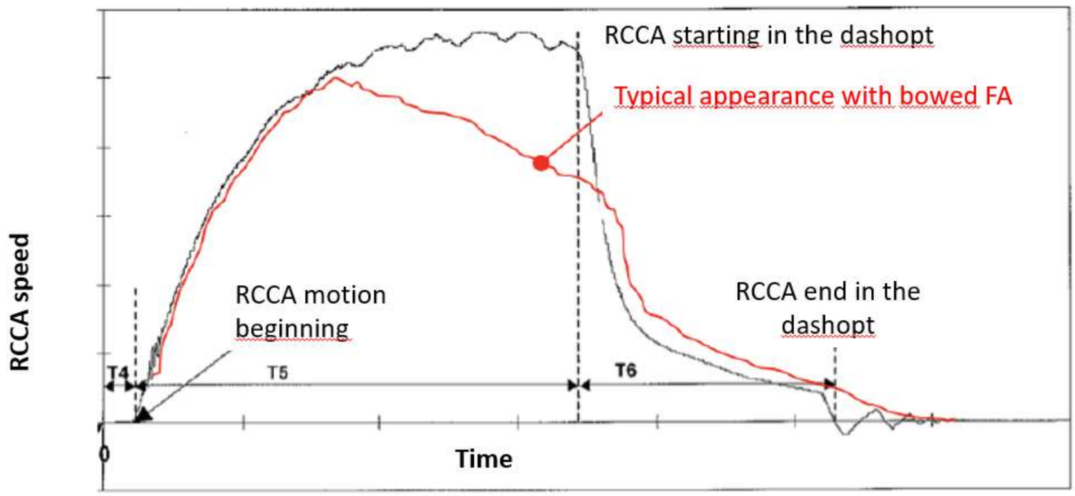


Figure 1 : Example of higher RCCA drop times in case of a bowed FA for a French 1300 MWe NPP

[1] The French operator EDF manages NPP including 56 PWR, currently in operating, put into service between 1977 and 1999. Composed of four standardized plant series, 900 (12 feet), 1300 (14 feet), 1450 MWe (14 feet) and 1650 Mwe (14 feet - the French EPR operating is planned in 2023), EDF’s fleet uses the same type of fuel assembly consisting of 17x17 rods array.

[2] French regulation requires EDF to perform RCCA drop times tests at the beginning and at the end of each cycle.

In Belgium, the Doel 4 and Tihange 3 units faced control rod sticking problems in 1996. These units are three loops, 1000 MWe, Westinghouse PWR. The reactor core of both units has 17x17 14 feet FA and 52 RCCA. They started commercial operation in 1985. At the Doel 4 unit, in June 1996, during the hot drop test, five rods experienced incomplete insertion, stopping at 17 to 71 steps above the rod bottom, one sticking above the dashpot region and another one in the dashpot. Measurement of the deflections highlighted that all concerned fuel assemblies were found to be bowed in one plane and that most of them showed an S-shape distortion.

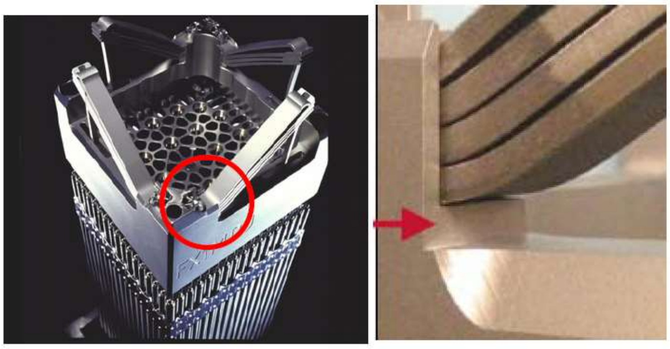
In August 1996, control rod drop tests were performed in Tihange 3. The tests evidenced an increase in the dashpot penetration time for two positions. In September 1996, drop tests were again performed on a limited number of rods (8), and increase of the drop time was evidenced in 5 positions. Finally, during a reactor trip in October 1996, two rods stopped at the dashpot level. During the two additional rod drop tests performed thereafter, three and five rods respectively experienced incomplete insertion.

As consequences, this phenomenon leads to operating safety issues insofar as FA bowing potentially interferes, directly or indirectly, especially with the core reactivity control safety objective: the complete RCCA insertion and the RCCA drop time are important assumptions in the safety studies. Moreover, in the frame of handling issues during refuelling outages such as FA hanging, FA bowing leads to grids-to-grids damage due to several numbers of inter-assemblies contact points. The W-shaped bowing is the least desirable mode because it presents an increased risk for control rod anomalies and grids damage due to the decreased bending radius and the potentially higher number of inter-assemblies contact points.

Since the occurrence of FA bowing, operators and fuel designers have taken several measures to counteract this safety issue. Some examples of curative and compensatory measures related to FA bowing are given below according to the feedback coming from France and Belgium (ETSON countries mainly concerned by the issue).

1.2 Curative and compensatory measures related to fuel assembly bowing

In France, on the one hand, at the core loading pattern stage, EDF has attempted to account for FA bowing, placing the bowed FA to prevent from further FA bowing propagation or to promote a reduction of FA deformations[3]. On the other hand, FA with new characteristics evolutions have been designed by the fuel suppliers Framatome [3] and Westinghouse to strengthen mechanical FA behaviour. More precisely, guide-thimble tubes using an increase of the thickness from the inside with or without more creep strain resistant material (in quaternary Zirconium alloy) have been deployed for increasing the FA lateral stiffness. Besides, for one of the fuel suppliers, modification of the number of FA hold-down springs (from 5 to 4 and – example in the figure opposite) have been considered in order to decrease the force transmitted to the structure from the FA hold-down system during operating. Since 2013, these FA design changes have been deployed in 1300 MWe and 1450 MWe reactors.



In Belgium, two major measures were taken at Doel 4 and Tihange 3 to address the observed FA bowing issue in 1996. A first measure was related to a modification of the FA design, aiming at improving its lateral stiffness. A second measure consisted in increasing the primary flow. Steam generators issues at both units had indeed resulted in a reduction of the primary flow, and the analysis had shown that this could contribute to the occurrence of FA bowing.

If FA deformations have been reduced in many instances due to the specific core loading plan stage or the introduction of new FA designs, FA bowing continues to be observed, which is not satisfactory. So, visual inspections and measurement techniques are carried out in order to monitor and evaluate the FA deformation for some “control” NPP chosen by the operator. The measurements of FA

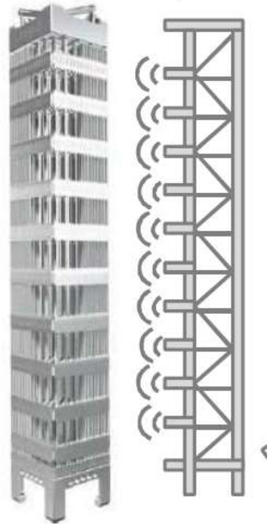
[3] Especially, for some reactors characterized by excessive FA bowing, French regulation requires not to load under a RCCA the FA for which the “gravity index” is higher than 15 %. The “gravity index” corresponds to the curve integral along the FA entire length representing the level of FA deformation. This parameter increases when a C-shaped bowing goes to S and W-shaped bowings.

deformation directly in core are not possible, but they could be done during refueling outages with specific devices. Some examples of devices for FA bowing measurements are given below according to the feedback coming from France.

1.3 Devices measuring fuel assembly bowing

In France, since the 2000s, a specific measurement tool so-called DAMAC has been developed in order to estimate FA bowing based on ultrasonic measurements of the deflections (or lateral deformations) at the level of the grids. These measurements are carried out in the spent fuel pool during the refueling outages. In the 2000s, the "control" NPP were chosen with respect to RCCA kinetics anomalies due to the excessive FA bowing. Since, the EDF's strategy evolves considering the introduction of improved FA designs by following the expected beneficial effects for the core. Today, about ten reactors (900 MWe, 1300 MWe and 1450 MWe) are subject to DAMAC measurements during the refueling outages.

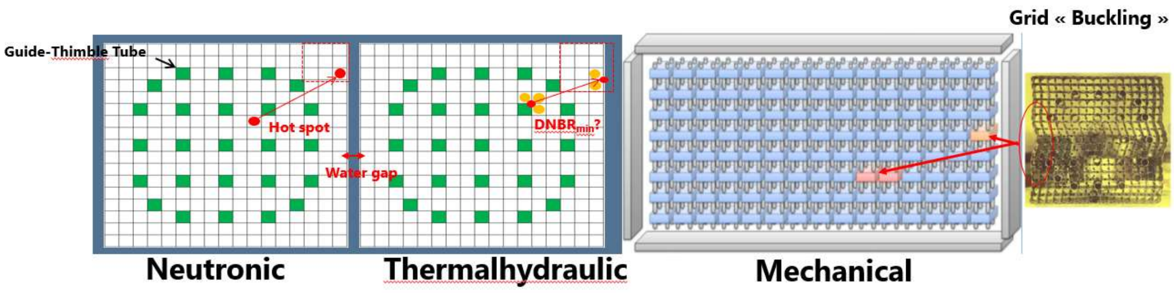
The DAMAC system is composed of transducers fixed on a submerged holder in the fuel pool



In Belgium, the problem of FA bowing seems to have been resolved quite soon after its occurrence in 1996, no problem occurred since then. Consequently, no specific device for measuring FA bowing is used now.

Consequently, FA bowing leads to different effects that need to be considered in the safety demonstration [2]:

- **neutronic effect**: increased water gaps may lead to a power distribution modification with an increase in local power density due to a more effective moderation (hot point shift),
- **thermo-hydraulic effect**: increased water gaps may cause different coolant flow distributions harming the heat transfer from the fuel rods by modifying the Departure from Nucleate Boiling Ratio (DNBR),
- **mechanical effect**: smaller water gaps than the nominal value is likely to increase the maximum impact force on FA grids due to lateral loads from seismic event and Loss Of Coolant Accident (LOCA) leading to FA grids buckling.



In France, a specific rulemaking related to FA bowing took place as described below. In Belgium, there is no specific rulemaking related to FA bowing, except that successful drop time tests for control rods must be performed regularly.

1.4 Rulemaking related to fuel assembly bowing

In France, ASN requested EDF to take the FA bowing in the safety demonstration in 2016 in the frame of generic guidelines for the 900 MWe NPP review and was addressed from a methodological point of view in the frame of an advisory committee for reactors (so-called GPR in French) related to fuel criteria hold in 2017 (the first application case being a 1300 MWe fuel management[4]). Then, the neutronic and thermohydraulic effects were applied for the safety studies related to the fourth 900 MWe NPP safety review in 2019 and the mechanical effect were assessed additionally in 2020.

In the following paragraphs, before giving more details about how FA bowing effects are considered in the safety demonstration, the FA bowing phenomenology and the determination of inter-assemblies' water gaps distributions are described. Such data can be used as input data for the evaluation of neutronic, thermal-hydraulic and mechanical effects of FA bowing.

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[4] A so-called "fuel management" is the fuel operating mode, specific to one or more reactors of the same design (standardized plant series), characterized in particular by:

- the nature of the fuel and its initial content of fissile material and, if appropriate, of neutron poisons (especially gadolinium),
- the maximum burn-up expected of the fuel when it is removed from the reactor, characterizing the quantity of energy extracted per ton of material (expressed in GWd/t),
- the number of the new fuel assemblies reloaded in each cycle when the reactor is shut down to renew the fuel (generally 1/4 or 1/3 of the total assemblies),
- the position of the fuel assemblies in the core, in particular the positions of the new fuel assemblages and the most irradiated,
- the nature and the position of the neutron absorber rods.

2 Fuel assembly bowing phenomenology

In-core FA bowing is considered to be a complex process with a large number of influencing mechanisms with interaction between them. The investigations pointed out several phenomena and parameters contributing to FA bowing, especially [1][4][5][6]:

- FA hold-down compressive force (permanent force applied to FA during operating): the higher hold-down force can contribute to the occurrence of FA bowing by means of the structural softening effect of normal compressive stresses in thin-walled structures. This effect could reduce the effective lateral stiffness and increase the lateral creep deformation rate which is one of the causes of the permanent FA deformation,
- FA design (lateral stiffness): the lateral stiffness of FA is a key parameter responsible to the elastic deformation and therefore to the stresses in the FA structure. Hence, the higher the elastic deformation is under an external load, the higher will be the creep deformation rate. Thus, the susceptibility of the FA to lateral deformation can be reduced by increasing its stiffness,
- irradiation and temperature induced FA creep deformation: during operating, the fast neutron flux irradiation causes the length increase of guide-thimbles tubes leading to an axial growth of the FA. This structural irradiation-induced growth is the root cause for increasing hold-down forces during operating. When the mechanical stresses in the structure (for example in guide-thimbles tubes) are sufficient, creep deformation may occur which is an important contributor to FA bowing. Besides, during operating, the irradiation induced-relaxation of spacer grid springs contribute to the reduction of lateral FA stiffness, and it can therefore be regarded as a bowing-enhancing mechanism. Moreover, the material temperature is considered as an important parameter for creep deformation under mechanical stresses. Thermal loads induced by thermal gradients might also drive permanent FA deformations,
- lateral hydraulic forces: due to the inhomogeneous coolant inlet and outlet flow distribution at the lower and upper core plates, crossflow is induced

in the core because of lateral pressure gradients. Then, the crossflow generates hydraulic loads on the single FA with the creation of bending moments on FA structures. These hydraulic loads may hence induce permanent FA deformations because of creep,

- interaction with neighboring FAs or core shroud: when the inter-assemblies' gap is closed between two neighboring FA (inter-assemblies contact), these latter are coupled mechanically in the lateral translational Degree Of Freedom (DOF). In practice, most FA in the core are getting in contact to each other during operating, creating a coupled non-linear mechanical system with a multitude of DOF. At the end, this coupling effect could either create collective bowing patterns in FA row (bowing propagation from deformed FA to undeformed ones) or sets a limit to the lateral deformation,
- fuel management: the fuel management can be considered as contributing factor to the FA bowing as much as it could influence fast neutron flux distribution, on in-core power distribution and on mechanical interaction between FA as result of core loading pattern.

The coupling effect of the multitude of influencing mechanisms is still not fully understood. TSO positions regarding FA bowing phenomenology are given below according to the feedback coming from France and Belgium.

In France, in the frame of a GPR related to the French fuel behaviour feedback from 2010 to 2019 and hold in July 2022, it was concluded that the FA bowing therefore results from a complex phenomenology whose main contributing and enhancing factors are identified but for which the individual contributions remain, in the current state-of-the-art, difficult to deconvolve and quantify given their concomitance and, for some, their interdependence. Indeed, using simulation tools could make it possible to deconvolute the various physical phenomena at the origin of the FA bowing. Fuel suppliers are developing numerical simulations tools that go as far as mechanical-thermal-hydraulic coupling in order to assess each individual effect. Unfortunately, the current tools are not yet able to quantify the impact of the various parameters. However, they make it possible to qualitatively assess the gains made by design changes. To understand and improve the state-of-the art of FA bowing phenomenology, IRSN considers important to develop numerical simulations tools based on appropriate data.

In Belgium, the FA bowing is considered to be solved for the Belgian units, but international operating experience is continued to be followed on this topic.

3 Inter-assemblies water gaps distribution

The FA deformation measurements carried out in the fuel pool are not representative of the current and realistic in-core FA bowing since the FA in the fuel pool aren’t subjected to the various loads and interactions which occur in the core during operating, for instance the physical constraints imposed on FA by the upper and lower core plates as well as neighbouring FA or core shroud. That is why, models have been developed to estimate the in-core FAs bowing and the resulting inter-assemblies water gaps distribution.

In Belgium, since the problem is considered to be solved for Belgian units, no detailed model has been developed.

In France, a mechanical model based on a finite element model, which doesn’t consider the hydraulic forces was developed by EDF to estimate the in-core FA bowing and the resulting water gaps distribution. This mechanical model takes as input data:

- out-of-core FA deformations measured by the DAMAC tool,
- hold-down springs compressive force after the reactor vessel closure,
- irradiation induced grid growth,
- flow induced lift force,
- assembly-to-assembly and assembly-to-core shroud forces,
- geometrical and mechanical FA characteristics.

In practice, inter-assemblies water gaps distribution is calculated for the Beginning Of Cycle (BOC) and the End Of Cycle (EOC). The DAMAC measurements are used for EOC calculation while the DAMAC measurements from the previous cycle are considered for the BOC. It is important to mention that this type of calculations does not predict explicitly the evolution of FA deformation resulting from irradiation. However, considering the cycle depletion, the time-effects of irradiation and also temperature on FA mechanical properties (such as the irradiation induced relaxation of the grids-to-fuel rods connection mechanical characteristics and the temperature evolution of the grids springs material Young's modulus) are taken to consideration.

Finally, for each FA position in the core, the inter-assemblies water gaps are estimated at different spacer grid levels. Up to now, the largest majority of water gaps remains less than 5 mm and the magnitude of inter-assemblies' water gaps varies from 0 mm (closed gap or contact) to more than 10 to 15 mm. This conclusion is valid for all French reactors, even for weakly deformed cores whether or not they exhibit high RCCA drop times. The IRI or RCCA drop times difficulties observed since 2010 are probably due to such excessive water gaps for strongly deformed cores. Examples of water gaps distribution in a French 1300 MWe NPP are illustrated in Figure 2. Moreover, the water gaps distribution doesn't seem to be very sensitive to the core loading pattern.

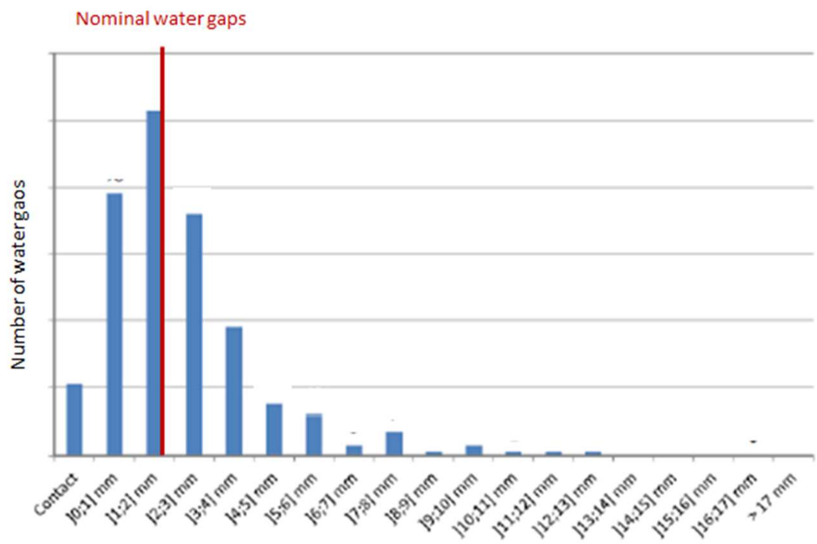

Figure 2 : Example of inter-assemblies' water gaps distribution in a French NPP

Although it is theoretically possible to develop a method to estimate in-core FA bowing based on out-of-reactor measurements, there is no experimental means to validate such a technique. That is why, results comparisons with another code developed by one French supplier were conducted. Moreover, to have confidence in the ability of the EDF's code to correctly evaluate the in-core water gaps distribution, sensitivity studies, notably related to the hydraulic lift forces non-modelling and the irradiation induced evolution of the mechanical characteristics of grids-to-rods connection (mainly due to spring relaxation, clad creep down and grids growth), were carried out, which do not call into question the conclusions of the calculations.

To conclude, in the frame of the GPR related to the fuel safety criteria in 2017, IRSN considered that the calculated in-core water gaps distributions with the EDF's method are at the state-of-the-art and can be used as input data for the evaluation of neutronic, thermal-hydraulic and mechanical effects of FA bowing. Nevertheless, R&D related to the model development is still needed.

4 Neutronic effect

A power distribution modification is expected when increasing the inter-assemblies water gaps which can shift the hot spot to a FA peripheral fuel rod and increase locally the linear power density since water gaps widening increases locally the moderator-fuel factor (local neutron moderation).

In France, EDF developed a new methodology for quantifying the neutronic effect of FA bowing considering a new component of the total uncertainty related to the fuel rod linear power density evaluation. The estimation process is divided in three steps:

- evaluation of the effect of water gaps widening on the fuel rod linear power density distribution, especially based on neutronic calculations using a Monte-Carlo code for different 3x3 FA patterns,
- evaluation of the total uncertainty to apply to each fuel rod linear power density (FA uncertainty map), combining the current uncertainties (fuel rod bow penalty during operating, neutronic calculations uncertainty, manufacturing tolerance penalty…) and the so-called FA bowing uncertainty,

- evaluation of safety margins regarding the safety criteria to be verified related to clad or fuel temperature and the linear power density for Plant Condition Categories (PCC) 1, 2, 3 and 4.

Regarding the first step, the FA bowing neutronic effect can be characterized on one hand at the macroscopic scale (e.g. core) and one the other hand at the microscopic scale (e.g. fuel rods):

- at the macroscopic scale, the impact of FA bowing can be covered by the AZimuthal Power Imbalance (AZPI) penalty and the neutronic calculations uncertainty,
- at the microscopic scale, as the effect of water gaps modification was not yet estimated, it was estimated by Monte-Carlo neutronic calculations for different 3x3 FA pattern performed at the BOC and EOC. Thus, the Monte Carlo calculations performed for Uranium OXide fuel (UOX) showed for example that with a water gap widening of about 5 mm, the linear power density increase can reach more than 10 % for FA corner.

IRSN's assessment was based on counter-calculations with its own code in order to verify the validity of EDF's hypothesis and corroborated the EDF's evaluations. However, it is worth noticing that the presence of different types of fuel[5] could have an influence on the linear power density distribution throughout the FA. This effect was considered by EDF as interface effect between FA, therefore, different kinds of assembly interfaces (UOX/UOX, UOX/MOX, MOX/MOX) were analyzed. The simulation results highlighted UOX/UOX and MOX/UOX interfaces respectively as the limiting case for UOX and MOX assembly. Thus, for a given water gap widening, the outer fuel rods row of UOX assembly (respectively MOX assembly) can contain the hot spot only at the UOX/UOX interface (respectively MOX/UOX interface).

Regarding the second step, the contribution of FA bowing to the total uncertainty was estimated considering especially the worst interfaces (UOX/UOX, UOX/MOX) for each FA type of the different 3x3 FA patterns and based on Monte Carlo procedure. For each type of FA, the maximum effect is observed for outer fuel

[5] Currently in France, the fuel management of some 900 MWe NPP includes UOX and Mixed OXide (MOX) fuels.

rods row. Compared to the current situation with nominal inter-assembly gap size[6], the total uncertainty could reach about 10 % for at MOX FA outer and about 15 % for MOX FA corner. IRSN considered acceptable to consider the FA bowing uncertainty in the total fuel rod linear power density evaluation uncertainty.

Regarding the third step, EDF considered the fuel rod linear power density increase due to FA bowing in the safety demonstration according to two different approaches:

- the first method considers the maximum increase of fuel rod linear power density for determining a conservative safety margin,
- the second method uses the uncertainty FA map directly to estimate the fuel rod linear power density rods-by-rods.

IRSN considered acceptable these approaches whose application depends on safety study analyze (0D *versus* 3D).

5 Thermalhydraulic effect

Any evolution in the inter-assemblies' water gaps sizes modifies the power distribution for the outer fuel rods row, but changes also the thermohydraulic channel flow area and therefore the flow velocity. With a nominal water gaps distribution, the minimal DNBR[7] is usually obtained inside the hot FA because the peripheral fuel rods are less powerful. Thus, with non-nominal water gaps distribution, two situations related to boiling crisis risk can be observed for FA peripheral thermohydraulic channels:

- the increase of local power due to inter-assemblies' water gaps widening (neutronic effect) is more significant than the benefit of increased thermohydraulic channel flow area and flow velocity, thereby resulting in a noticeable decrease in the calculated DNBR,

[6] With the nominal water gap, in the frame of the fourth 900 MWe NNP safety review, the current total uncertainty factor applied to each fuel rod linear power is about 8.8 %.

[7] DNBR is defined as the ratio between Critical Heat Flux (CHF) and local heat flux.

- conversely, the decrease of local power due to inter-assemblies' water gaps closing offsets the reduction of the thermohydraulic channel flow area and flow velocity, which increase the calculated DNBR.

In France, EDF determined DNBR penalties for PCC 1 and 2 for which boiling crisis risk must be avoided. For PCC 3 et 4, EDF must demonstrate that the percentage of the fuel rods assumed to enter in DNB is limited up to 5 % for PCC 3 and 10 % for PCC 4.

EDF suggested a methodology to evaluate FA bowing effect on the boiling crisis risk analysis for peripheral thermohydraulic channels. This methodology is divided into three steps:

- step 1: calculations of DNBR using hypothesis and input data available for the current safety demonstration. The obtained DNBR values are considered as reference values,
- step 2: neutronic and thermalhydraulic analyses for several FA patterns with water gaps widening (like those used for the neutronic study) to determine the case with higher increase of fuel rod linear power density,
- step 3: sensitive analyses to the parameters which play a significant role in the evaluation of DNBR for FA peripheral thermohydraulic channels like diffusion coefficient, pressure drop coefficient, hydraulic diameter and FA mixing vanes performance. The minimal values of DNBR are compared to the reference values obtained at step 1.

Up to now, CHF is evaluated by the mean of correlations which are determined from experimental results. It is worth mentioning that the CHF values depend on the location of thermohydraulic channels (peripheral between several FA or inner of the FA). The difference arises from the presence or absence of FA mixing vane and different thermal hydraulic conditions between inner or outer fuel rods rows (edge and corner rods). Accordingly, the worst CHF value should be obtained for peripheral thermohydraulic channels (without mixing vanes). Therefore, although the boiling crisis risk is suspected inner thermohydraulic channels (with nominal water gaps distribution), this feared phenomenon could be observed in peripheral thermohydraulic channels considering the effects of FA bowing. However, the current modelling codes are acceptable only for inner thermohydraulic channels and the local thermohydraulic conditions are different in peripheral thermohydraulic channels. Consequently, modelling codes must be validated in peripheral thermohydraulic channels. Besides, the current CHF correlations are based on experimental data carried out considering mixing vanes and geometry representative of inside thermohydraulic channels (with FA mixing vanes). That is why, in the frame of the fourth 900 MWe safety review, ASN asked EDF in 2021 to evaluate the reliability and acceptability of the current CHF correlations for peripheral thermohydraulic channels (without FA mixing vane) to ensure the acceptability of the EDF's methodology related to the presence of non-nominal inter-assemblies' water gaps.

To conclude, the evaluation of thermalhydraulic effect of FA bowing in the French safety demonstration is still on going.

6 Mechanical effect

Various thermomechanical loads encountered during the different PCC need to be considered during the FA design stage to evaluate the overall FA performance (top and bottom nozzles, guide-thimbles tubes, grids, …). Thus, FA stresses and deformations must be evaluated. Especially, during postulated accidents (PCC 3 and 4), the fuel safety requirements are based on FA structural damage to ensure control rod insertion capability and core coolability. However, the in-core water gaps distribution could have significant mechanical effect, especially small water gaps are likely to increase maximum impact force on FA grids due to lateral loads from seismic event and LOCA leading to FA grids buckling.

In France, the transients considered for the FA structural design include seismic events (SSE – Safe Shutdown Earthquake) and postulated pipe breaks in the reactor coolant system (LOCA). When an earthquake occurs, FA undergo loads caused by the motion of the reactor vessel. In addition, when a LOCA occurs, FA undergo loads induced by the pressure and hydraulic flow fluctuations in the reactor vessel. In fact, during SSE and LOCA events (so-called accidental conditions), the main loads applied to FA come from lateral motions between the FA and between FA and the core baffle plates. These lateral motions produce impact loads especially at grid locations leading to potential grid deformations. Accordingly, for the safety demonstration, these impact loads have to be estimated taken in-core water gaps distribution into account. This is the reason why ASN asked EDF to consider FA bowing in the recent frame of the 10-yearly safety reviews of EDF's 900 MWe units. To deal with this request, EDF developed a new methodology considering FA bowing. Before presenting the new EDF's methodology, the current safety demonstration concerning FA structural design (without FA bowing) in France is summarized.

The limitative mechanical study (lower safety margin) concerns the evaluation of spacer grids behaviour at the end of life under lateral loads for PCC 3 and 4 to ensure control rod insertion capability. Currently, this requirement is met using a decoupling criterion based on non-buckling of the spacer grids under accidental conditions. Thus, the most limiting grid load calculated would not exceed the non-buckling threshold. This threshold corresponds to the lower 95 % confidence limit on experimental results based on dynamic impact tests. For estimating grid loads, the so-called CASAC code (FRAMATOME computer code) is used to perform time-history analysis (due to SSE and LOCA) with dynamic models of one assembly row in the core, considering core plates movements and reactor coolant effects (Fluid-Structure Interaction – FSI) (Figure 3) [7].

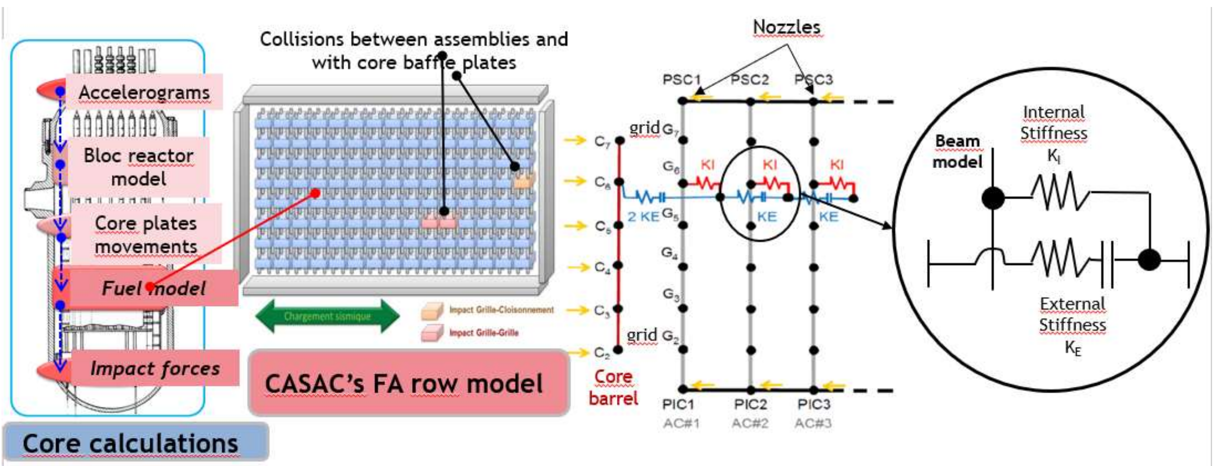

Figure 3 : CASAC code and the dynamic model of FA row in a core to calculate grids loads

In practice, the computer code calculates the maximum impact forces, between FA or between FA and core baffle plates, in the assembly row up to the maximum number of FA in a row loaded in the core (15 for 3-loop NPPs, 17 for 4-loop NPPs). It is worth noting that, in the current safety demonstration, EDF considered a nominal and homogeneous water gap in FA row in the core. In order to represent impacts at grid locations, the momentum transfers from or to the FA and through the FA are respectively represented by internal grid stiffness and external grid stiffness. Grid internal stiffness is an equivalent parameter reflecting the local rod bundle flexibility which is determined by cyclic compressive tests [8]. Besides, the grid external stiffness represents a grid-specific parameter determined by dynamic impact tests. For the safety analyses, seismic and LOCA calculations are performed separately, then the maximum loads for each event are combined in a quadratic manner. If the resulting impact load is lower than the non-buckling threshold, the safety demonstration is met (meaning that no permanent deformation of spacer grids is observed). This evaluation is done at BOL and EOL. Up to now, the EOL condition is the limiting one regarding the safety analysis.

As mentioned previously, in the frame of 10-yearly safety reviews of EDF's 900 MWe units, EDF had to consider FA bowing in the safety demonstration. Consequently, some calculations of impact grids forces have been performed for the 900 MWe units with CASAC's row model considering water gap distributions. For SSE event, the maximum impact force remained lower than the non-buckling threshold ensuring no permanent grid deformation. However, regarding LOCA conditions considering the new LOCA safety reference (with new defined pipe break sizes and including stretch out conditions) and taking the water gap distributions into account, the CASAC code predicted a maximum impact force higher than the non-buckling threshold, leading to permanent grid deformations.

Consequently, EDF had to develop a new CASAC's row model able to simulate inelastic behaviour of spacer grids during SSE and LOCA events. To understand inelastic behaviour of spacer grids, many results from various test programs related to the buckling response of PWR FA grids (irradiated and non-irradiated grids) were analyzed. Beyond the non-buckling threshold, the distortion of the grid showed permanent grid deformations in two directions (Figure 4) on the one hand following load direction (so-called “permanent grid deformation”) and on the other hand normal to the load direction (so-called “permanent thimble guide netwok offset”).

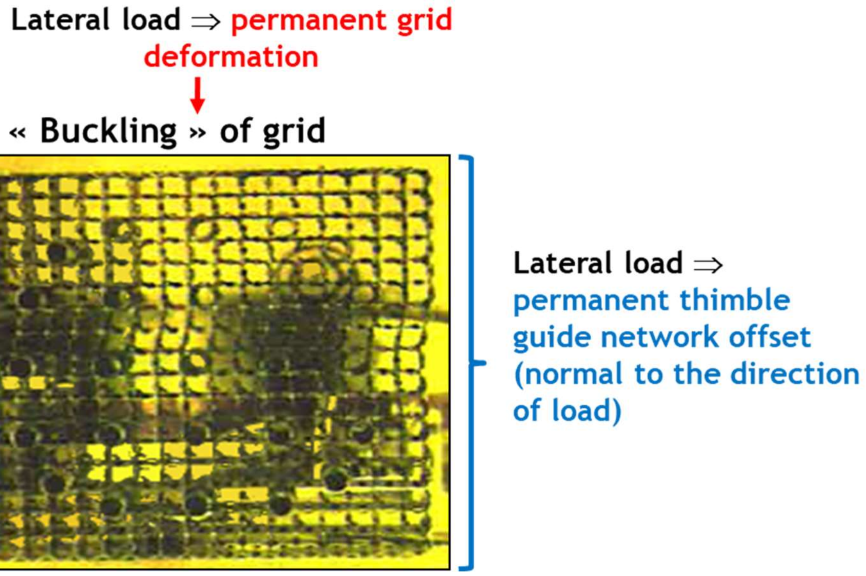

Figure 4 : Typical grid distortion from cycle compression test

Thus, EDF considered these experimental results to implement his new CASAC’s row model taking a non-linear FA grids behaviour and heterogeneous water gaps distribution on the various FA rows in a core into account. Indeed, the heterogeneous water gaps distribution in EDF’s rows model seems to influence damping coefficient [9], coupling mass or added mass (FSI) and subsequently on impact loads.

Moreover, with the different permanent grid deformations, EDF had to justify the respect of safety requirements:

- to ensure control rod insertion capability, EDF compared the permanent grid deformations to those obtained during the Japanese seismic tests NUPEC and JNES, which are full-scale shaking table tests [1][10]. The purpose of these program tests was to prove seismic reliability of PWR reactor core which included the confirmation of control rod insertion function during the time of earthquakes (Figure 5 showing the delay ratio of the PWR control rod insertion as function of FA grid displacement). Based on these results, EDF demonstrated that RCCA can be inserted during accidental conditions within a specified limit time with a sufficient safety margin,

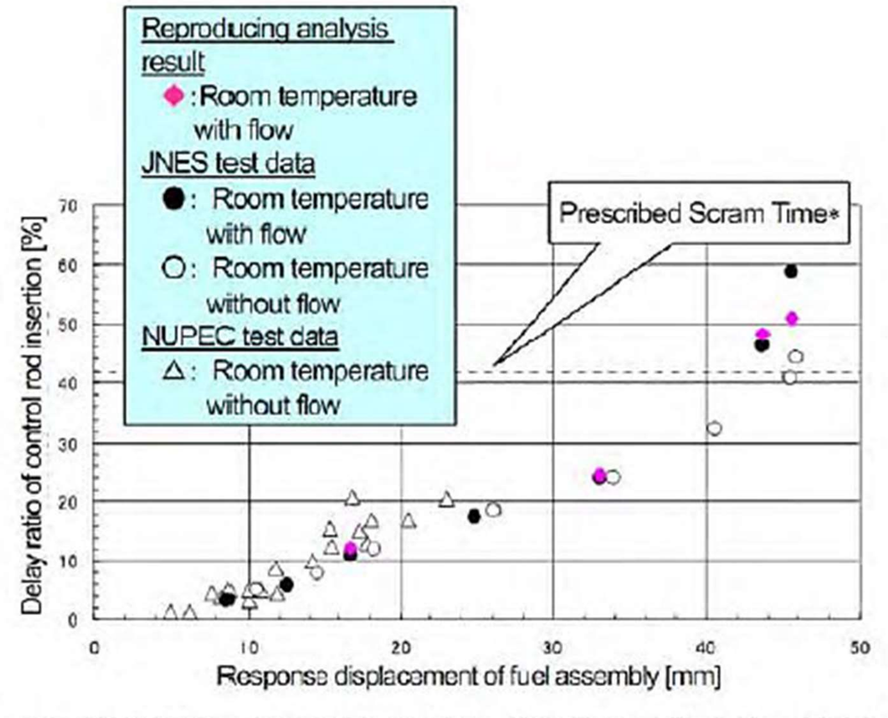

Figure 5 : Delay ratio of PWR control rod insertion during Japanese seismic tests [10]

- to ensure core coolability, the increase of pressure drop due to deformed grid cross sectional areas were estimated. Indeed, the flow in the deformed grid-cells decreases with reduced water gaps. Finally, the magnitude of pressure drop increase did not cast doubt on the LOCA safety demonstration especially regarding cladding temperature.

Besides, it’s worth noting that the LOCA and seismic loads are no longer conventionally combined in the square root manner for the safety demonstration, because of the non-linear response of FA grids beyond the non-buckling threshold.

From IRSN’s point of view, the new EDF’s methodology is complex and displays some lacks especially regarding:

- the characterization of the non-linear spacer grids behaviour and the validation of CASAC non-linear row model. Indeed, cyclic compressive tests carried out on single spacer grid (without considering the effect of interaction between FA rows on the post-buckling grids behaviour) are not enough representative compared with the reactor core conditions,

- the LOCA and seismic effects combination. Indeed, to ensure the French safety reference, the LOCA and seismic effects must be combined in an appropriate and justified manner to estimate grid deformations when the buckling threshold is exceeded.

From IRSN's point of view, it will be hard to achieve a justified safety demonstration considering the grids buckling at short time: ensuring the validity and the robustness of the safety studies is still a challenge for EDF. That is why, it would be more appropriate to aim at a safety demonstration based on the respect of the non-buckling threshold by improving the representativity of the cyclic compressive and dynamic impact tests and introducing in reactor new robust FA designs against grids buckling. Thus, in the frame of the fourth 900 MWe safety review, ASN asked EDF in 2021 to carry out more representative dynamic impact tests to better characterize the mechanical behaviour of spacer grids under accidental conditions considering FA bowing and to suggest a new methodology to combine LOCA and seismic effects. Recently, new elements have been provided by EDF and technical exchanges between EDF and IRSN are underway.

7 Synthesis

The FA bowing results from a complex phenomenology whose main factors are identified but for which the individual contributions remain, in the current state-of-the-art, difficult to deconvolve and quantify given their concomitance and, for some, their interdependence. This phenomenon cannot be considered completely well understood, especially due to the coupling effect of the multitude of influencing mechanisms. This phenomenon was highlighted through operating issues such as IRI or RCCA drop times anomalies during reactor shutdown or grids-to-grids damage during handling procedure.

Nevertheless, operators and fuel vendors have tried and continue to counteract the FA bowing issues mainly on the one hand by improving the FA design (especially by increasing FA stiffness and using more creep resistant material) and on the other hand by optimizing in-core FA loading patterns (based on reactor operating experience related to FA bowing measurements). Such practices should lead to absorb safety issues for cores presenting excessive levels of FA bowing.

However, from a nuclear safety point of view, FA bowing consequences in terms of neutronic, thermohydraulic and mechanical effects have to be assessed.

The following table provides the key messages from ETSON regarding the FA bowing and its consequences.

Country	Main FA bowing events	FA bowing phenomenology	Inter-assemblies water gaps distribution	Neutronic effect	Thermohydraulic effect	Mechanical effect
France	FA bowing occurred since the 2000s. Besides unexpected and noticeable increases of IRI and high RCCA drop times were found out since 2010 for some PWR using 14 feet FA. Water gaps distributions are similar for all French NPP (even for weakly deformed or excessive deformed cores).	To understand and improve the knowledge regarding to FA bowing phenomenology, numerical simulations based on appropriate data must be developed.	R&D related to the EDF's model calculating the in-core water gaps distributions is still needed. However, the EDF's method is at the state-of-the-art and can be used as input data for the evaluation of neutronic, thermal-hydraulic and mechanical effects of FA bowing.	FA bowing uncertainty is a new component in the total uncertainty related to the fuel rod linear power density in a core. EDF's method is considered acceptable.	To ensure the acceptability of the EDF's methodology, the reliability and acceptability of the current CHF correlations for peripheral thermohydraulic channels (without FA mixing vane) need to be justified. IRSN's assessment of the thermohydraulic effect is still on going.	It would be more appropriate to aim at a safety demonstration based on the respect of the non-buckling threshold by improving the representativity tests and introducing new robust FA design against grids buckling. IRSN's assessment of the mechanical effect is still on going.
Belgium	FA bowing occurred in 1996 at Tihange 3 and Doel 4. The problem was however resolved, and no FA bowing issues occurred on Belgian plants since then.	Since the FA bowing issue is considered to be solved in Belgium, no significant efforts have been devoted to this topic.				

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ABBREVIATIONS

AZPI	AZimuthal Power Imbalance
BOC	Beginning Of Cycle
CHF	Critical Heat Flux
DNBR	Departure from Nucleate Boiling Ratio
DOF	Degree Of Freedom
EOC	End Of Cycle
FA	Fuel Assembly
FSI	Fluid-Structure Interaction
IRI	Incomplete Rod cluster Inserts
LOCA	Loss Of Coolant Accident
MOX	Mixed OXide
NPP	Nuclear Power Plants
PCC	Plant Condition Categories
PWR	Pressurized Water Reactors
RCCA	Rod Cluster Control Assembly
SSE	Safe Shutdown Earthquake
TSO	Technical support organization
UOX	Uranium OXide