Note:
This is a translation of the RSK statement entitled
“Verformungen von Brennelementen in deutschen Druckwasserreaktoren (DWR)”
In case of discrepancies between the English translation and the German original, the original shall prevail.

RSK statement
(474th meeting of the Reactor Safety Commission (RSK) on 18 March 2015)

Fuel assembly deformation in German pressurised water reactors (PWRs)

1 Background

The reactor core consists of the fuel assemblies (FAs) together with the other core components. The fuel assembly of a pressurised water reactor is composed of the fuel assembly structure (bottom nozzle, top nozzle, spacer grids, guide tubes) and of a bundle of fuel rods which contain the nuclear fuel pellets. The fuel rod cladding is part of the multi-level barrier concept for the retention of radioactive material (fundamental safety function “confinement of radioactive material”). In addition to the fuel rods, the fuel assemblies contain a number of control rod guide tubes. These guide tubes serve to guide the control rods in such a way that during operation and during the events to be postulated, they can move or drop into the core to reduce the power or to shut the reactor down in a short time. The FA structure has to ensure that the control rod assemblies can be fully inserted if required (fundamental safety function “reactivity control”).

Damages and deformations (e.g. bending, twisting) of fuel assemblies as well as their possible impacts on neutron physics and thermal-hydraulics of the reactor core (fundamental safety functions “heat removal” and “reactivity control”) may have a detrimental impact on compliance with the fundamental safety functions of a reactor. Therefore, they need to be assessed with regard to possible negative effects on one of the mentioned fundamental safety functions. If necessary, corrective measures are to be taken to ensure that the fundamental safety functions are fulfilled.

Beginning from about the year 2000, there has been an increase of permanent fuel assembly deformations during reactor operation in German PWR plants. These deformations first led to FA handling problems and, in individual cases, also to increased drop times or failure to reach the lower end position at control rod assembly drop. Therefore, from 2009 on, the development of fuel rod deformations has been regularly monitored by the RSK Committee on Reactor Operation. With [1], the BMU requested the RSK to give its opinion on safety-related issues associated with FA deformation and other phenomena (e.g. increasing neutron flux fluctuations and damage to core components).

This statement is limited to phenomena where a causal link with fuel assembly deformation is seen or cannot be excluded. The other damages – mentioned in [1] – to core components such as fuel assembly centring pins, hold-down springs or flow restrictor assemblies and the related quality assurance issues are dealt with in a separate statement. In case the deliberations on fuel assembly deformation gave indications that fuel assembly deformations cannot be excluded as a factor contributing to core component damage, this is commented correspondingly in this statement. The questions asked in [1] will be answered after completion of the separate statement.

Summarised result

The increase of FA deformations observed in German PWR plants since 2000 has already been subject of several deliberations within the RSK. With this statement, the RSK delivers, as requested, an overall consideration of all relevant phenomena that have become known in this respect in recent years including their safety assessment going beyond the previous discussions on individual aspects of fuel assembly behaviour and fuel assembly deformation. Here, particular attention is paid to the issue of demonstrating safety in the context of fuel assembly deformation. Another focus is set on the identification of suitable measures to ensure that fuel assembly deformations are limited to a degree which is not relevant in terms of safety.

The RSK notes that the operators and manufacturers have already taken a number of measures to reduce fuel assembly bending which, according to more recent data from the German PWR plants [12], resulted in a relative improvement of the situation and let expect further improvement.

Regardless of this, the RSK comes to the conclusion that fuel assembly deformations can be of safety significance. Therefore, it is a priority to take precautions that deformations are limited in future by appropriate construction, material selection and operating conditions.

Nevertheless, deformations cannot be excluded completely. Therefore, the resulting impacts on the safety analyses for the design and operation of the reactor core are to be considered. This concerns the fundamental safety functions “heat removal” and “reactivity control”. Furthermore, the handling of more severely deformed fuel assemblies requires special precautionary measures to prevent mechanically caused damage to fuel rods (fundamental safety function “confinement of radioactive material”).

Based on the result of the deliberations, the RSK issued recommendations which shall ensure that

- the probability of occurrence of inadmissible FA deformation is reduced,
- safety analyses are performed under consideration of given fuel assembly deformations, and
- measures to be taken if control rod assemblies do not move freely and for handling of deformed fuel assemblies are specified in the operating procedures.

In about one year, the RSK expects a report of the operators on whether and in which way the recommendations were implemented.

2 Course of consultations

The members of the ad hoc working group on fuel assemblies (RSK AG-BE) were appointed at the 452nd meeting of the RSK on 18 October 2012; the assignment of the working group to the RSK was determined at the 455th meeting of the RSK on 21 February 2013.

The working group heard reports

- of AREVA on fundamentals of the mechanical design of PWR fuel assemblies,
- of AREVA on the current state of knowledge in root cause analyses and in modelling of fuel assembly deformation,
- of AREVA on the influence of fuel assembly deformations on the DNB (departure from nucleate boiling) behaviour,
- of the KTA working committee on the revision of safety standard KTA 3101.3 “Design of Reactor Cores of Pressurized Water and Boiling Water Reactors; Part 3: Mechanical and Thermal Design”,
- of VGB on the impacts of fuel assembly deformations on the operating behaviour of fuel assemblies and control rod assemblies in 2012,
- of TÜV SÜD Energietechnik on the involvement of the authorised expert into mechanical FA design and quality assurance,
- of GRS on a literature study on operating experience with fuel assembly deformations,
- of Westinghouse on experiences with fuel assembly deformations,
- of the Emsland nuclear power plant (KKE) on the status of fuel assembly bending at KKE

and visited test rigs at AREVA.

In addition, the working group held in-depth technical discussions with AREVA and Westinghouse on mechanical and physical fuel assembly design. The RSK adopted the statement at its 474th meeting on 18 March 2015.

3 Evaluation criteria

The general safety requirements and the safety-related and radiological acceptance targets and acceptance criteria for the neutron-physical and thermal-hydraulic core design and the thermomechanical FA design are derived from the Safety Requirements for Nuclear Power Plants (SiAnf) [3] and the interpretations I-1 on the SiAnf [4].

Under 3.2 (2), the SiAnf require that the reactor core, the relevant equipment for the monitoring, control and limitation of reactor power and for reactor shutdown shall be designed, manufactured and maintained in such a condition that in combination with the cooling systems for the reactor core, the respective design limits of levels of defence 1 to 4a are not exceeded.

Interpretation I-1 [4] requires under 2 (1) that any events at levels of defence 1 to 4a, internal or external hazards as well as emergencies must not cause deformations of the fuel rods, the fuel assembly structure or the control rod assemblies that would compromise the ability for mechanical shutdown (in case of a large leak inside the containment of a PWR, the ability for permanent shutdown must not be impacted). Furthermore, 2 (2) in [4] stipulates that within the framework of the mechanical design of the reactor core, design limits shall be defined for the conditions of the specified normal operation such that if they are adhered to, no defects need to be assumed of the fuel rods, the fuel assembly structure or the control rod assemblies including the associated structural parts.

More specific requirements regarding the neutron-physical and thermal-hydraulic core design are given in the safety standards KTA 3101.1 [5] and 3101.2 [6]. Under 3.1 (5), KTA 3101.1 requires, among others, that model uncertainties as well the operational variations and uncertainties of the input parameters used in the safety analyses shall be taken into account. Section 5.1 of KTA 3101.2 requires that the power density shall be limited such that the safety-related boundary conditions of the mechanical design of the fuel rods are met during specified normal operation. In addition, the initial values of the power density for events of abnormal operation and for design basis accidents which have been verified as permissible by safety-related analyses must be met during normal operation.

A KTA draft safety standard is available (draft standard 3101.3 “Design of Reactor Cores of Pressurized and Boiling Water Reactors Part 3; Mechanical and Thermal Design” [7]) which specifically defines the requirements for the thermomechanical design of the fuel assemblies.

Further KTA safety standards (KTA 3103, KTA 3201.2, KTA 3204, KTA 3905), and international standards such as

- ANSI / ANS-57.5 – Light Water Reactors Fuel Assembly Mechanical Design Evaluation [2],
- NRC Regulations, Title 10, Code of Federal Regulations (10 CFR), and
- Nureg-0800 – US NRC Standard Review Plan (Section 4.2 – Fuel System Design)

are used for orientation in practice to the extent applicable.

In addition, VdTÜV decision 153 [8] (authorised experts' activities in the examination of the thermomechanical design, construction, manufacturing and operation of core components) is referred to, which contains information on the concrete procedure for the safety demonstration of subsequent cycles, for design review, for manufacturing surveillance, etc..

Plant-specific requirements for the core and fuel assembly design are laid down in the safety-related frame conditions (“Sicherheitstechnische Rahmenbedingungen”), which, at German nuclear power plants, are usually part of the operating licence. The safety-related frame conditions specify safety-relevant design parameters with regard to the neutron-physical and thermal-hydraulic core design and the thermomechanical FA design and substantiate these parameters with quantitative criteria. The demonstration of compliance with these criteria is provided in advance for each fuel cycle.

The safety-related frame conditions do not provide any criteria regarding permissible deformations of fuel assemblies and no requirements with regard to the consideration of fuel assembly deformations in the neutron-physical and thermal-hydraulic analyses of the core design.

From the above-mentioned requirements on core and fuel assembly design, the RSK derives the following criteria for the safety assessment:

Fuel assembly deformations

(1) must not inadmissibly impair control rod assembly movement under any operating and accident conditions with regard to the drop times,
(2) must not lead to changes in the power density distribution in the reactor core that challenge the validity of safety analyses on core design and event control; if necessary, relevant influences shall be considered in the safety analyses,
(3) must not systematically lead to damage to individual fuel assembly components (such as corner fuel rods, spacer grids), to the entire fuel assembly or the adjacent components (such as fuel assembly centering pins) during operation so that their functional capability is no longer ensured, and
(4) must not lead to a deterioration of the integrity of the fuel assemblies during handling and storage of the fuel assemblies.

4 Current situation, causes of FA deformations and measures taken

4.1 Current situation

From around 2000, anomalies have been observed during insertion of fuel assemblies into German nuclear power plants with PWRs which indicate increasing deformation of the FAs. Both the extent of deformation and the frequency of occurrence increased over the years. Towards the end of the decade, more reportable events related to fuel assembly deformation occurred. The individual power plants were affected to different degrees.

In one case (KKI-2, reportable event ME 08/058) it was found that a control rod assembly had not reached the lower end position during an event-related power reduction. In 2010 and 2011, three control rod assemblies (KBR, reportable event ME 02/2011) did not meet the specified drop times for reaching the lower end position in another PWR plant. However, the delays affected the area from the entry into the damper, which is not sensitive regarding the effectiveness of the shutdown reactivity. In some other cases, increased drop times were observed which, however, were still within the permissible limits.

Furthermore, fretting corrosion of different degrees of damage was observed at spacer grid corners in several power plants ranging from slight contact marks up to severed spacer corners, in some cases with damage to the corner fuel rod located behind.

Fuel assembly deformations were measured outside the reactor core in several power plants. Fuel assembly deflections of up to approx. 25 mm (with C-shaped bending) were observed and different bending patterns (C-shape, S-shape and superposition of the two lateral dimensions) were identified.

According to the operator, the KKE plant – one of three identical KONVOI plants – has so far hardly been affected by FA deformations and damage to core components. The degree of FA deformations is clearly lower, the rod drop times have not shown any anomalies, the loading and unloading times of the reactor core (indication of handling difficulties due to deformations) are in the usual range and neutron flux fluctuations are less pronounced than at other comparable plants.

4.2 Causes

Fuel assembly deformations result from creep deformations during reactor operation under the influence of external loads on the fuel assembly, depending on its load transfer behaviour.

External loads are the axial and lateral flow forces determined by the design, buoyancy, gravity and loads resulting from the axial growth of the guide tubes, the lateral interaction forces due to contact with neighbouring fuel assemblies as well as the hold-down force exerted by the hold-down springs. The flow-induced lateral forces are caused by the global flow distribution depending on the plant design (differences e.g. due to core support stool or sieve drum in the lower plenum) and local effects, namely pressure drop differences between the cooling channels due to different FA types and deformation. The external forces given by the plant design can only be changed within narrow limits, e.g. by varying the fuel assembly hold-down force.

The load transfer behaviour is determined by the structural design and the stiffness of the fuel assembly as well as the strength and the creep behaviour of the materials used. The transfer of forces is effected both via the FA structure, i.e. the skeleton consisting of spacer grids and guide tubes, and via the fuel rods clamped in the spacer grids by springs.

The forces acting on the fuel assembly result in an initially elastic lateral deflection of the assembly. The lower the stiffness of the fuel assembly, the more pronounced is this deflection for the same external load. Essential parameters for the stiffness of the FA structure (of the fuel assembly skeleton) are the area moment of inertia of the FA cross-section, the number and the strength (modulus of elasticity) of the guide tubes and the stiffness of the connections between spacer grid and guide tubes.

The stiffness of the entire fuel assembly is also determined by the supporting effect of the fuel rods. Since the stiffness of the fuel rod support depends, in addition to the structural design of the clamping, primarily on the relaxation of the spacer springs, the stiffness-supporting effect by the fuel rods decreases with the relaxation of the spacer springs over the fuel assembly residence time. Above all, the use of zirconium based materials as spring material leads, compared to Inconel materials, to a significant reduction of the stiffness-supporting effect by the fuel rods already after a relatively short fuel assembly residence time.

During reactor operation, the elastic fuel rod deformation having arisen at a respective point in time is largely converted by creep of the guide tubes into a permanent deformation measurable after use in the reactor. The creep strength of the guide tube material is decisive for the extent of this creep deformation. FA deformations can occur in both lateral directions with possibly different bending modes (C-shape, S-shape, W-shape). The latter have not yet been observed in German reactors.

The maximum possible deformation of the individual fuel assemblies is limited by the maximum available gap size between the fuel assemblies accumulated over the fuel assembly row (the nominal value of the water gap between two FAs is approx. 1.6 mm for Zircaloy spacer grids in hot condition). If a fuel assembly row would be formed exclusively from fresh fuel assemblies, this maximum cumulative gap size corresponds to the sum of the nominal gaps. For the plants currently still in operation, this results in values of up to 26 mm. However, due to the lateral spacer growth during the residence time, the real resulting gap is always smaller than the nominal gap. Compared to the past situation, the use of spacer materials optimised for low growth, as they have been preferably deployed in recent years, has resulted in decreased gap reduction due to spacer growth.

4.3 Measures taken so far

In recent years, the operators have introduced the following modification measures at the different plants to a varying extent to limit deformations and to take deformations into account in loading planning and FA handling:

- Reduction of stresses as the driving force of creep deformation:
    - reduction of FA hold-down forces, e.g. assemblies with optimised eight hold-down springs (multi-step hold-down spring) or reduction from eight to four hold-down springs,
    - use of thicker-walled guide tubes, e.g. in AREVA HTP FAs, Westinghouse (WSE) FAs, and
    - reduction of the hydraulic resistance of the fuel assemblies.

- Increase of creep resistance:
    - low creep materials for guide tubes and spacer grids, e.g. in HTP FAs Q12, Westinghouse Low-Tin-ZIRLO or the use of steel guide tubes.

- Increase of the lateral FA stiffness (stiffer fuel assembly skeletons reduce elastic deformation and thus the creep deformation and thereby the overall deformation of the FA) by:
    - thicker-walled guide tubes,
    - stiffer connection between spacer grid and guide tube, and
    - improved fuel rod support in the spacer grid, such as monometallic spacer grids made of more creep-resistant material, e.g. Q12 and Low-Tin-ZIRLO.

- FA-loading patterns:
    - consideration of the measured FA deformations,
    - pre-calculation of the deformation for consideration in the loading strategy,
    - use of an increased number of fuel assemblies with increased lateral stiffness,
    - no insertion of fuel assemblies with strong deformations at control rod assembly positions,
    - checkerboard arrangement of fuel assemblies with strong deformations in the core centre or core edge,
    - rows with different fuel assembly types, and
    - rotating of the fuel assemblies or shuffling into the diagonal quadrant.

- Measures to prevent handling damage during loading and unloading:
    - optimisation of the sequence of operations taking into account the FA deformations identified,
    - recording of the load measurement during handling with the loading machine,
    - use of aids (dummy FAs, FA positioning system “FUPSY”) for loading,
    - staggered FA positioning (HELIX), and

· reduction of spacer grid corner fretting by changing the spacer corner contours.

5 Effects of fuel assembly deformations and their assessment

Deformations of fuel assemblies during operation can generally not be avoided and are tolerable as long as safety-related functions are not impaired and the fundamental safety functions are not compromised.

5.1 Effects of FA deformations on the core design and operational monitoring of the reactor core

Due to the resulting change in gap width between neighbouring fuel assemblies, fuel assembly deformations have an influence on

- the local fuel rod power resulting from changed moderation, and
- the heat transfer from the fuel rod to the coolant due to changed mass flows and coolant enthalpy in the sub-channels near the gap.

These influences need to be assessed regarding their impact on the verification of safety.

5.1.1 Influences of fuel assembly deformations on power density distribution

AREVA reported about analyses with the 2D program CASMO, used for the nuclear design of FAs, regarding the influence of gap widening on the local power density of the neighbouring fuel rod rows [9]. These analyses show noticeable influences on the first (outermost) and the second fuel rod row (from the outside). The power peaking factors calculated by AREVA are presented in Table 1 below.

Table 1: Power peaking factors as a function of water gap widening

Additional water gap / Power peaking	5 mm	10 mm	15 mm
First fuel rod row	1.06	1.15	1.23
Second fuel rod row	1.02	1.05	1.08

This power increase decays in the course of the cycle due to the increased burn-up as a result of the initially increased fission rate. According to AREVA [9], an initial power increase of 15% would be reduced to less than 10% at constant water gap after about 75 days.

AREVA carried out analyses with 300 different randomly sampled, core-wide gap distributions for a pre-KONVOI plant [9][1]. These analyses indicate (slides 20 and 21 in [9]) that gap widening can increase

- both the maximum local power density in the monitoring area of individual detectors for the measurement of the local power density (**L**eistungs**v**erteilungs**d**etektoren – LVDs), and
- the core-wide maximum of the local power density.

Currently, the RSK does not have sufficient information to quantify the respective effects.

Monitoring of increases in the local power density due to FA deformations

The monitoring concept for the reactor core is based on the aeroball measurement system (Kugelmesssystem) and the LVDs for the measurement of the power distribution [15]. The core monitoring system could detect the influences of FA deformations if the aeroball measuring positions were located at the positions affected by the changes in local power density. However, the spatially distributed aeroball measurement positions are generally not located at FA positions with the highest power density and, in the fuel assemblies themselves, not at the fuel rod positions near the gap. They are rather located in the third fuel rod row seen from outside the FA. There, the effects of the gap changes on the linear heat generation rate of the fuel rods have already diminished to such an extent that they cannot be detected by the available measuring equipment.

Thus, local power density increases in FA edge rods due to gap widening cannot be detected by the existing power density monitoring program and, due to the lack of knowledge of the real gap width distribution, cannot be calculated either.

This leads to the following recommendation:

> **Recommendation 1:**
> The influence of FA deformations on the maximum power density in the reactor core is to be evaluated. In case, a relevant increase of the maximum local power density cannot be excluded, this effect must be considered in the precalculation of the next cycle and in the derivation of the actual maximum power density from the measured values. Based on this, specifications are to be developed by which compliance with the maximum permissible local power density can be ensured during operation over the entire cycle.

5.1.2 Effects of local power density changes on safety demonstration

The safety analyses are based on power densities that correspond to nominal gap widths. For FA edge rods, higher local linear heat generation rates can occur than for nominal gap sizes depending on the water gap

[1] The 300 different core-wide gap width distributions were calculated on the basis of existing measurement data of bending patterns of fuel assemblies using a mechanistic model developed by AREVA for the prediction of FA deformations under consideration of the total gap width available in the core applying a statistical approach.

distribution. With regard to safety analyses, this leads to the following implications for the core design requirements in [3], [4], [5] and [6] mentioned in Section 3:

- Fuel rod design, normal specified operation: since higher local linear heat generation rates of FA edge rods can also exist for a longer time, the power histories of the affected fuel rods change. This can influence the following design parameters:
  - level of defence 1: internal fuel rod pressure, plastic cladding tube expansion, cladding tube corrosion and H_2 uptake,
  - level of defence 2: transient tangential strain and fuel centre temperature.

- LOCA hot rod analysis: In case the core-wide maximum of the local power density increases due to FA deformations, its monitoring with the PEAK-RELEB system would be afflicted with an additional uncertainty factor, which is possibly not covered by the uncertainty margin applied in the hot rod analysis[2]. Within the framework of the cycle-specific safety analyses based on the maximum local power densities applied for individual burn-up classes, possibly existing effects on the calculation of the following quantities are currently not explicitly taken into account: axial transformation function and the rod-wise power density distribution to determine the "conservative limit distribution" of the power density.

- LOCA extent of damage analysis: Within the framework of cycle-specific analyses of the strain-burst behaviour of the individual fuel rods of a core load, any influences of FA deformations on the ascertainment of the "conservative limit distribution" and the operating history of the fuel rods are currently not explicitly taken into account.

This leads to the following recommendation:

Recommendation 2:
The possible influences of FA deformations on the power density distribution of the reactor core should be assessed with regard to their significance for the safety demonstration on the

- fuel rod design, and
- hot rod and damage extent analysis for LOCAs.

Here, the spatial distribution of possible power density changes can be taken into account.

If it cannot be demonstrated that the influences are insignificant or that the analyses contain sufficient margins to also cover, in addition to other uncertainties in the analysis, the effects of FA deformations, these influences are also to be taken into account.

[2] To the RSK's knowledge, an uncertainty factor for monitoring the maximum linear heat generation rate of 12% of the LOCA response value (475 W/cm) is applied for pre-Konvoi plants within the scope of LOCA safety analyses. In contrast, only 8% is applied for demonstrations of compliance with the DNB limit ratio. For Konvoi plants, the uncertainty factor presumed for LOCA safety analyses is > 13.5% [10].

5.1.3 Effects of FA deformations on the thermal-hydraulic properties of the reactor core

To quantify the effects of FA deformations on the thermal-hydraulic core design, in particular the DNB ratio, AREVA carried out core-wide sub-channel analyses using the COBRA-FLX code [11]. The analyses were part of a steady-state full-core calculation for a pre-Konvoi plant with a representative core loading. The bending patterns of the fuel assemblies in the core were calculated starting from measurements at free-standing fuel assemblies in the fuel pool. These measurements were converted to the conditions in the core at BOC using a model developed at AREVA for predicting the bending behaviour.

The analyses showed only a marginal influence on the core-wide minimum DNB ratio $DNBR_{min}$, although considerable decreases in DNBR were calculated in some cases for the FA edge channels close to the gaps. Two effects were responsible for this:

- Decrease of the DNBR by up to 10% for fuel assemblies that were not leading in terms of the DNBR.

- Decrease of the DNBR at edge channels for fuel assemblies leading in DNBR did occur, but was not large enough to lower the minimum DNBR value of the fuel assembly. This is due to the fact that the edge channels show significantly higher DNBR values. To become leading the gap changes would first need to reduce the DNBR to the level in the FA interior. However, based on the information provided in [11] it cannot be assessed whether the bending pattern of the fuel assemblies has led to a significantly enlarged edge gap at a fuel assembly that is leading with regard to DNBR. Thus, it cannot conclusively be assessed at present whether the resulting influence on the DNBR is covered by the results in [11].

According to AREVA's statement, which was confirmed by Westinghouse and TÜV SÜD Energietechnik during the consultations of the RSK AG-BE, the minimum DNBR at nominal gap geometry in the current core loadings in the German plants occurs in the FA interior near the control rod guide tubes. As the RSK AG-BE currently has no exhaustive knowledge on whether a reduction of the DNB ratio at the outer fuel rods resulting from changed gaps can lead to values below the minimum DNBR inside the FA on which the demonstration of safety and monitoring are based, supplementary sensitivity analyses should be performed.

This leads to the following recommendation:

> **Recommendation 3:**
> To verify the available exemplary findings on the influence of FA deformations on the core-wide minimum DNB ratio $DNBR_{min}$, additional sensitivity analyses should be performed. Here, the influences of enlarged gaps should be evaluated selectively for fuel assemblies which are leading with regard to DNBR. For this purpose, FA-internal distributions of the DNBR values that are typical for German plants should be used.
>
> If influences on $DNBR_{min}$ cannot be excluded, this should be considered accordingly within the framework of the safety demonstration concept.

5.2 Effects of FA deformations on the design of the fuel assembly structure

Normal specified operation

The effects of possible fuel assembly deformations on the parameters to be considered in the FA structural design can be limited to the stress and stability analysis.

FA deformations lead to changed load conditions in the guide tubes, which cause a local increase of bending stresses.

With regard to the buckling load (stability analysis), increasing pre-deformation makes the system developing from the ideal buckling case to a load case with initial bending. As a result, the force that can be transferred by a free-standing FA under axial load (buckling force) decreases with increasing pre-deformation. However, under the geometry conditions in the core (limited lateral space), there is only limited deformation of the guide tubes and thus a limited change of the force that can be transferred compared to the case without FA pre-deformation. Nevertheless, there is a reduction of the reserves with regard to buckling safety.

Depending on the slenderness ratio of the system and the degree of FA deformation, either the stress load (including the bending part) or the buckling load can become leading.

With the knowledge available to the RSK, it cannot be excluded that the observed FA deformations could have a relevant influence on the loads that can be transferred by the fuel assembly or the control rod guide tubes.

This leads to the following recommendation:

> **Recommendation 4:**
> It must be shown that the design margins cover the possible effects of the FA deformations on the stress and stability analysis.

Accidents, external hazards and emergencies

The objective of the safety analyses of the FA structural design is to show that no deformations occur that would endanger the unhindered drop of control rod assemblies. For this purpose, it must be shown that

- under vertical loads, the axial compressive stress in the guide tubes does not lead to buckling of the guide tubes, and
- under horizontal loads, no or only slight lateral permanent deformation occurs in the spacer grids.

The “loss-of-coolant accident” is the relevant case for the vertical loading; for the horizontal loading, it is the “design basis earthquake” [3].

Vertical loads:

In the event of an accident, FA deformations affect the level of vertical loading in the same way as during normal specified operation. Therefore, the same conclusions and recommendation apply as stated in the section on normal specified operation.

Horizontal loads:

The relevant horizontal loads result from horizontal pounding forces between the spacer grids as well as between the spacer grids and the core baffle for the load case design basis earthquake.

With regard to the horizontal loads it can be noted that different gap distributions arise within the core baffle in the available mounting space during the earthquake so that the initial distribution of the gaps between the fuel assemblies has no safety-relevant influence on the mechanical stresses in the fuel assemblies.

5.3 Effects of FA deformations on control rod assembly drop

Increased FA deformations can lead to an increase in the frictional forces between the control rods and the guide tubes and consequently to an increase in the control rod assembly drop times. Especially in the shock absorber, i.e. the lower section of the guide tube with reduced inner diameter, the frictional forces can increase such that the control rod assembly gets stuck and does not reach the lower end position.

By default, the proof of compliance with the specified drop times is provided by assessing and testing the construction, supplemented by operational validation as part of recurrent testing:

- assessment of the constructive design of the guide tubes and their interaction with the control rods,
- ex-core testing of the chosen design, and
- operational validation by means of recurrent tests specified in the operating or test manuals of the power plants.

At its 194th meeting, the RSK Committee on Reactor operation had suggested that the drop times of the control rod assemblies should be measured at the beginning and at the end of the cycle, as far as this is not yet practised in all German plants. The operating experience gained from such measurements shows that even slight changes

[3] In the load case earthquake, both horizontal and vertical loads occur. Separate treatment of the individual spatial directions is only permissible without restrictions in the case of linear structural behaviour. During an earthquake, however, there is a non-linear behaviour due to the impact loads occurring as a result of the contact between some spacer grids and – if permissible for the specific plant – stability failure of spacer grids. Nevertheless, to the RSK's knowledge, it is common practice to treat the horizontal and vertical impacts separately. The background to this is the consideration that the decisive stresses are absorbed by different components depending on the direction: in vertical direction by the control rod guide tubes, in horizontal direction by the spacer grid strips.

in the drop times can be detected and that they can be an indication of incipient impairment of the free movement of control rod assemblies due to FA deformations. Measurements at reactor cores with strong FA deformations have also shown that the control rod assembly drop times can change over the cycle; in such cases, supplementary drop time measurements are therefore necessary within a running cycle.

In the case of larger anomalies, path-distance curves of the rod drop and/or runtime oscillograms were recorded in some cases.

The RSK gives the following recommendation to substantiate the evidence regarding delayed control rod assembly drop times and to improve the prognosis for the affected or the following cycle:

> **Recommendation 5:**
> Based on the routine drop time measurements at the beginning and end of the cycle, further measurements should be carried out for control assemblies with a recognisable trend towards higher drop times, such as additional drop time measurements with recording of path-distance curves and/or runtime oscillograms during the cycle. Corresponding specifications should be included in the operating procedures.

In addition to the control rod assembly drop time measurements specified in the operating or test manuals, straightness measurements were carried out on a sample of unloaded fuel assemblies in several power plants where increased FA deformation was detected or suspected. In case of relevant findings, the sample had been extended to up to 100% of the fuel assemblies. The RSK considers the performance of such measurements and the systematic evaluation of the resulting database as a suitable means to identify trends and influencing factors in the deformation behaviour of the fuel assemblies as well as to assess the effectiveness of corrective measures and, if necessary, to take measures in case of more severely deformed FAs. This leads to the following recommendation.

> **Recommendation 6:**
> In case of indications of relevant deformations, e.g. if FAs do not move freely during unloading, random straightness measurements should be carried out within the framework of the FA inspection programme during refuelling. If the random measurement provides evidence of a relevant number of increased FA deformations, these measurements should be extended to 100% of the FAs of the affected FA types of the previous cycle.

5.4 Effects of FA deformations on FA handling

Deformations of the FAs can lead to handling difficulties when loading and unloading the reactor core. In addition to the increased time required for such handling – a merely operational aspect – there is also an increased risk of damage to the fuel assemblies (spacer damage, cladding tube damage). Therefore, the operating instructions for handling fuel assemblies should provide instructions for handling such FAs and for monitoring the handling processes.

Recommendation 7:
The operating instructions for the refuelling process should provide criteria under which conditions special measures with regard to handling and monitoring during handling of deformed fuel assemblies are to be taken. For cases where there is an increased risk of fuel assembly damage, suitable handling devices should be used that minimise the risk of FA damage when handling deformed fuel assemblies.

5.5 Assessment parameters for the structural design

During the deliberations, it became apparent that the stiffness of the FAs (FA skeleton and fuel rods) and the creep behaviour of the materials used are decisive parameters for the extent of FA deformations. However, currently there are no specified quantitative parameters for assessing the structural design. There are also no quantitative criteria for the permissible deformations of fuel assemblies.

The manufacturers consider the definition of such assessment parameters difficult due to the complexity of the influencing factors. Nevertheless, the RSK deems the definition of such assessment parameters a necessary measure to minimise the risk of undesired FA deformations or their consequences in the future.

Recommendation 8:
For the assessment of the design features of the fuel assemblies with regard to FA deformations, supplementary assessment parameters for relevant design features such as the stiffness of the FA and FA structure, and the creep behaviour of the FA structure are to be considered in the design and the selected construction is to be assessed with regard to deformation under consideration of these parameters within the framework of the design and before use in the reactor core.

5.6 Fretting of the spacer grid corners

In recent years, fuel assembly damage due to spacer fretting corrosion at diagonally opposite spacer grid corners (corner fretting) has been observed to an increasing extent in several power plants. This concerns both the number of damages and the degree of damage, which occurred in the form of slight surface abrasion to partial or complete severance of the spacer grid corners up to damage of the corner fuel rod with fission product release due to contact with the corner of the neighbouring fuel assembly.

Corner contact of diagonally neighbouring spacers can only occur if the fuel assemblies have shifted at a spacer grid level due to FA deformation as shown in Figure 1.

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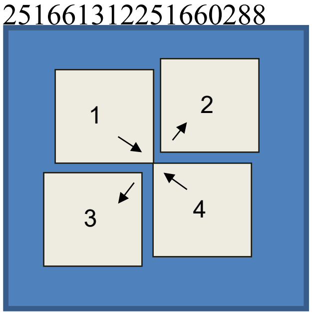

Assumption: moving apart of FA 2 and 3

Possible: corner contact between FA 1 and 4

Figure 1: Disturbed positioning of the FAs and increased risk of corner fretting

The extent of fretting that may occur depends on the magnitude of displacement and the structural design of the spacer grid corner. Furthermore, a relative movement between diagonally neighbouring spacers, as always given by the FA vibrations, is required. The additional conditions under which fretting occurs that induces increased abrasion have not yet been clarified.

Problems with corner fretting are not exclusively observed in the plants with the largest fuel assembly deformations. However, a certain degree of FA deformation is always required to cause corner fretting.

The prediction of the extent of possible corner damage and the effective prevention include the ascertainment of FA deformation as a prerequisite. However, beyond that there are further influences which would have to be quantified. Yet, the knowledge is still incomplete in this respect. Nevertheless, the different influencing factors that contribute to increased FA deformation can be regarded– with some uncertainty – to be relevant also here and can contribute to reducing the risk of damage.

Accordingly, several measures have been introduced by the power plant operators to prevent corner fretting which aim at improving the main influencing factors of FA deformation.

The effectiveness of the individual measures is difficult to assess due to the interplay of several factors, especially since it takes several refuellings before sufficient effectiveness of the implemented measures can be confirmed. Nevertheless, a trend can currently be observed indicating that these measures reduce the extent of corner damage.

The operational inspections currently performed, where necessary with an extended scope of the inspection in the case of findings, are considered suitable for reliably detecting any FA damage and for initiating corresponding measures for the FAs concerned (e.g. corner replacement, fuel rod replacement).

Repeated inspections of pre-damaged but further deployed FAs have shown that existing pre-damage usually does not expand in the subsequent cycle. Due to the shifting of the FAs to other core positions with other neighbouring FAs in the subsequent cycle, a different global deformation pattern results in the core with the consequence that other corner surfaces are pounding than before shifting. Applying appropriate core loadings, continued operation of FAs with pre-damaged spacer grid corners can be considered in case safe fuel rod support is still ensured despite the existing damage and spacer corner catching is not to be expected during handling.

6 Other aspects

Neutron flux fluctuations

The RSK dealt with the phenomenon of increased neutron flux fluctuations at PWR plants and published a statement with recommendations in 2013 [13]. In the course of the hearings on the causes and effects of FA deformations – in particular on the experiences at the Emsland nuclear power plant with comparatively low FA bowing – indications were found that there might be a connection between the magnitude of the neutron flux fluctuations and the degree of fuel assembly bending. In their annual status report, at the 232nd meeting of the RSK Committee on Reactor Operation, the operators reported on the development and clarification of cause of the neutron flux fluctuations.

It was reported that a temporal correlation between the fuel assembly types used and their stiffness on the one hand and the increase or decrease of the neutron flux fluctuations on the other hand could be observed. This would indicate that the strength of the transported reactivity perturbations in the core is a function of the fuel assembly stiffness. Nevertheless, there is still no clarity about the mechanical/physical processes in the core which lead to the neutron flux fluctuations.

> **Recommendation 9:**
> The RSK recommends continuing the clarification of causes (of neutron flux fluctuations) within the framework of research projects, particularly with regard to the extent and causes of influences that may lead to a displacement of the fuel assemblies relative to the coolant plumes.

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