Note:
This is a translation of the RSK statement entitled
“Stellungnahme zu noch offenen sicherheitstechnischen Fragen im Hinblick auf Verformungen von Brennelementen in deutschen Druckwasserreaktoren (DWR) einschließlich einer Bewertung der statistischen KMV-Analyse”
In case of discrepancies between the English translation and the German original, the original shall prevail.

RSK statement
(515th meeting of the Reactor Safety Commission (RSK) on 17 June 2020)

Statement on pending safety issues with regard to fuel assembly deformation in German pressurised water reactors (PWRs) including an assessment of the statistical LOCA analysis

CONTENTS

1 Background

The increase in fuel assembly (FA) deformations observed in German PWR plants since 2000 has already been the subject of deliberation of the Reactor Safety Commission (RSK). In its statement of 18 March 2015 [2], the RSK provided a general review and a safety assessment of all relevant phenomena that had become known in this respect in recent years going beyond earlier considerations on individual aspects of fuel assembly behaviour and fuel assembly deformation. Here, particular attention was paid to questions of safety demonstration arising from fuel assembly deformation. The RSK noted that the operators and the manufacturer have taken a number of measures to reduce fuel assembly deformations which, according to recent data from the German PWR plants, already have improved the situation and that further improvements can be expected. Regardless of this, the RSK came to the conclusion that fuel assembly deformation can be relevant for safety. The impacts of the deformations on the safety analyses for the design and operation of the reactor core are therefore to be considered. Furthermore, the handling of severely deformed fuel assemblies requires special precautionary measures to prevent mechanically caused damage to fuel assemblies. Based on the results of the earlier deliberations, the RSK issued recommendations which are to ensure that

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

Moreover, the RSK had published the statement on “PWR neutron flux fluctuations” of 11 April 2013 [1] that addresses another aspect of reactor core behaviour which is also related to the mechanical design of the fuel assemblies.

Based on these two RSK statements, GRS and Physikerbüro Bremen (PhB) drew up a list of questions on behalf of the BMU with the objective to supplement or concretise the RSK recommendations. BMU forwarded the list to the supervisory authorities of the *Länder* with the request for answers.

An evaluation of the feedback [3] on the questions showed that, with regard to thermohydraulics and neutron flux fluctuations, the results were partly not comprehensible and open questions remained. Regarding mechanical design, testing and monitoring as well as the handling of fuel assemblies, the RSK recommendations were considered as largely implemented or partly still being processed.

Based on this evaluation, the RSK was requested by the BMU to submit a written statement (advisory request [16] of 10 January 2016) on to which extent, from the point of view of the RSK, there are still open safety issues with regard to the above-mentioned topics. At its 487th meeting on 11/12 October 2016, the RSK requested the Committee on Plant and Systems Engineering (AST= Anlagen- und Systemtechnik) to prepare the requested statement. In this context, the effectiveness of the measures taken to reduce FA deformation should be considered. For the purpose described, the AST established a working group on FA deformation.

2 Course of consultations

In a letter dated 14 December 2016, the AST Committee had submitted questions to the operators on, among other things, fuel assembly deformation. The questions were concretised in a supplementary e-mail dated 18 January 2017. The operators responded by sending the report [4] to the RSK Secretariat by e-mail on 18 September 2017. At its first meeting on 23 November 2017, the FA deformation working group discussed the scope of work and identified open issues. The group filed a list of questions that was sent to the operators by letter dated 2 January 2018 and asked VGB for more detailed explanations on this topic. At the second meeting of the working group on 14 February 2018, the questions were answered by the FA manufacturer in a telephone conference. In addition, the working group heard a report by GRS on pending issues regarding the influence of fuel assembly deformation on the stress and strength analysis of control rod guide tubes.

At the third meeting of the working group on 27 March 2018, topics of consultations were the answers to the questions of GRS, GRS reports on the statistical analysis of loss of coolant accidents (LOCA), and on the influence of fuel assembly deformations on power distribution. In addition, the working group dealt with a recommendation of the French GPR on fuel assembly deformation.

At the 127th meeting of the AST Committee, GPR statements of 1/2 April 2015 on the principles of the safety review in the framework of the fourth 10-yearly safety review of the 900 MW reactors (Avis relatif aux orientations du réexamen de sûreté associé aux quatrièmes visites décennales des réacteurs du palier 900 MWe, 01./02.04.2015) and of 15 June 2017 on fuel rod criteria (Avis et recommandations relatif aux critères de tenue du combustible des réacteurs à eau sous pression, 15.06.2017) were discussed. Thereupon, the FA deformation working group was asked to also deal with the statements of GPR on the influence of fuel assembly deformation and to take into account available background information.

The topics of deliberations at the fourth meeting of the working group on 24 May 2018 included among others the review of the statements of GPR and background information provided by IRSN. As a result it turned out that these reports did not contain any unknown aspects despite an approach differing in its details. At its fifth meeting on 16 August 2018, the working group heard a report of the manufacturer on the results of LOCA analyses taking into account fuel assembly deformations. In preparation for the sixth meeting of the working group on 14 November 2018, the operators were sent a further list of questions by e-mail dated 28 August 2018, which the working group had prepared following its fifth meeting and which were answered by Framatome at the sixth meeting of the working group.

In parallel with the hearings, the working group had started to draft a statement. At its seventh and eighth meeting on 17 January 2019 and on 6 March 2019, the working group continued its deliberations on this issue. At its 140th meeting on 11 September 2019, the AST Committee entered into the discussion on the statement prepared by the working group and continued its deliberations on the subject at its 141st, 142nd and 143rd meeting on 17 October 2019, on 5 December 2019 and, following a technical discussion between members of the working group and representatives of Framatome on 18 February 2020 in Erlangen, on 19 February 2020. The statement was revised by an editorial group on the basis of comments received in a telephone conference on 24 April 2020 and following the 144th meeting of the Committee on 14 May 2020. Subsequent to this meeting, the statement was adopted by circulation procedure as draft for the RSK. The RSK discussed the statement at its 515th meeting on 17 June 2020 and adopted it by circulation procedure on 7 July 2020.

3 Answers to safety-related questions of the RSK (RSK recommendations of 18 March 2015)

A summary of the RSK recommendations of 18 March 2015 referred to in this chapter is given in Annex 1.

The following assessment by the RSK presumes that the FA deformations having occurred in former cores referred to by the operators for their investigations will also cover deformations to be expected in future.

3.1 Safety demonstration concept of the operators

According to the RSK recommendations, complementing analyses are required showing the influence of deformation on the minimum DNB ratio ($DNBR_{min}$)[1] throughout the core as well as on the hot rod and extent of damage analysis for LOCAs.

It is the objective of the safety demonstration concept presented by the operators in [4] to demonstrate that the necessary damage precautions have been ensured hitherto and that for all future cores, the impacts of FA deformation realistically to be expected on compliance with the DNB and LOCA acceptance targets are covered by the conservative deterministic analysis methods applied in practice. Thus, recommendations 1 to 3 of the RSK statement are implicitly also regarded as fulfilled by the operators.

For the respective safety analyses, typical FA deformations were used as observed in cores which, on the one hand, were mapped well and, on the other hand, showed large deformations. Based on the explicit consideration of these real cores and the correspondingly calculated FA deformations during power operation (see Chapter 3.2.1), according to the operators opinion, the comparison of the current results with the earlier obtained margins also allows assessing the safety of other cores with large FA deformations. However, there is no sufficient information from measurements available for the actual water gap distribution in the core.

[1] DNBR (DNB ratio) is the ratio of the critical to actual heat flux. $DNBR_{min}$ is the lowest value of the ratio of critical to actual heat flux during normal operation (source KTA 3101.1).

Since the deformations have been reduced again in recent years by appropriate measures, these typical FA deformations in connection with statistically determined water gap distributions for the LOCA analyses are also regarded in [4] as generically covering for the future.

In [4], the following approach was chosen:

Based on conservative deterministic calculations for the above-mentioned typical reactor cores with “straight” FAs, the changes of the parameters relevant for the safety demonstration (power density distribution, DNB) were ascertained for changes of the gap widths between the FA rows due to FA deformation.

With regard to compliance with the minimum permissible DNB ratio ($DNBR_0$)[2], the change in the minimum DNB ratio resulting from the deformation in the typical cores was compared with the margins existing in the previous calculations.

A statistical approach was applied to prove compliance with the LOCA criteria, which also includes a statistical treatment of the gap widths assumed in the core. Insofar as the effects caused by the calculated gap width distributions are covered by the existing margins in the conservative deterministic LOCA analyses for FAs without deformation, according to the operators opinion, no further constraints result for the core design compared to the previous designs. The principle of such a kind of safety demonstration is also mentioned as a possible statistical approach in the RSK statement on requirements for LOCA analyses of 2015 [7].

3.2 Implementation of RSK recommendations 1 to 3

Recommendations 1 to 3 of the RSK statement of 18 March 2015 [2] deal with the influence of changed water gaps between the fuel assemblies due to FA deformation on the power density distribution and on the minimum DNB ratio in the reactor core.

3.2.1 Identification of FA deformations during power operation

Facts and circumstances

According to the report of the operators [4] and the answers of the manufacturer [8], the in-core gap widths are identified as follows:

The FA deformation in the hot core is calculated iteratively by coupling a mechanical and a hydraulic model in the non-linear KWUSTOSS 3D finite element computer code. This code was originally developed at the manufacturer for static and dynamic analyses for the verification of fuel assembly integrity during normal operation and in case of accidents. Later it was extended for the prediction of FA deformation.

[2] $DNBR_0$ is the minimum permissible DNBR during normal operation. It is specified such that when it is maintained during normal operation – in conjunction with other design requirements – the fulfillment of the safety-related requirements at levels of defence 1 through 4a can be verified.

The mechanical FA model represents the main properties (stiffness, deformation mode) of the real FA (including the guide tubes, spacer grids, fuel rods). Due to the higher operating temperature of the core compared to the temperature during straightness measurement in the spent fuel pool, thermal expansions (influencing the hold-down force) occur and the material properties change. This is taken into account in the mechanical model.

The calculations are performed separately for the BOC (beginning of cycle) and EOC (end of cycle). For the identification of the in-core FA deformation in the whole core, the ex-core deformations measured on free-standing FAs are transferred to the detailed mechanical model as initial deformation in a first step. The individual FA models are then arranged as FA rows according to the core loading pattern to form the full-core model.

Mechanical contact of the spacer grids of neighbouring FAs and with the core baffle is possible. This occurs when neighbouring FAs are deformed to different degrees and then come into contact. Particularily, this occurs at BOC and, according to the operators, has the effect, e.g., that strongly deformed FAs are "straightened" in the core. Since the spacer grids are modelled with realistic stiffness, changes in the spacer grid outer dimensions due to contact between neighbouring FAs are also taken into account. However, these changes are smaller than 0.1 mm. Compliance with the geometric boundary conditions (core baffle) in case of contact provokes elastic FA deformations.

In the reactor core, the FAs are subjected to the following forces in addition to the mechanical loads resulting from the contact with the neighbouring FAs or with the core baffle:

- axial flow forces on fuel rods,
- axial flow forces on spacer grids,
- hold-down force on the FA header, and
- lateral flow forces.

KWUSTOSS includes a module to consider these forces in the modelling of the fluid-structure interaction between the fuel assembly and the coolant flow in the core.

When the coolant flows through the core, deformed FAs create additional altered cross flows. These in turn generate lateral forces primarily on the spacer grids and additional elastic FA deformations. The lateral forces are calculated for each FA row using a hydraulic model.

The influence of the coolant flow on deformed fuel assemblies has been investigated experimentally by Framatome. The calculation model was validated against the experiments and can reproduce the experimental results [17].

The calculation of the equilibrium FA deformations under operating conditions is carried out iteratively: alternately, the lateral forces are calculated with the hydraulic model and the elastic deformations with the mechanical model until equilibrium is reached. This converged equilibrium is the basis for the identification of the deformation during operation. Further information on the method, examples for validation and applications are included in [5].

Regarding the question of the working group on how and with which results the identified core-wide FA deformations under operating conditions were validated on the basis of the deformations measured in cold condition, the manufacturer referred to [17] and stated that best-estimate calculations were performed whose results were validated against the measured FA deformations.

KWUSTOSS is also used to predict FA deformations. For this purpose, a forecast of the in-core EOC state is performed starting from the BOC state. This forecast allows a prediction of the deformation state of the FAs ex-core after their use. This option thus allows a direct comparison with the data of the straightness measurements so that a further verification of the overall system is available. According to the manufacturer [17], the comparison of the observed deformations with the predicted deformations after use showed very good agreement with deviations in the range of about ± 1 mm, which corresponds to the uncertainty in the straightness measurement.

Furthermore, the manufacturer stated that the root causes of fuel assembly deformation are understood, which is also reflected in the meanwhile successfully reduced deformations by introducing fuel assemblies with higher transverse stiffness. In addition, it has been possible to reduce again the deformation amplitudes of individual pre-deformed fuel assemblies by inserting them in largely straightened cores.

The change from cold to hot condition of the fuel assemblies generates comparatively low stresses, so that with regard to fuel assembly deformation, from the point of view of the manufacturer, there is no reason to assume an undetected reversible deformation. Other effects that lead to a different deformation in hot condition and which are cancelled again when predicting the deformation state of the FAs ex-core are not recognisable from the manufacturer's point of view or cannot be derived from the modelled effects.

The nominal water gaps between the FAs and the predicted in-core deformations of the FAs are used to calculate the corresponding water gaps during operation. The water gap is defined as the distance between two neighbouring spacers which is limited by the total gap width in the core row. Gap reductions are limited by the contact with the neighbouring FA. The axial profile of the water gap is characterised by typical bending patterns (S-shape, C-shape, position of the “belly”) and the maximum water gap size.

The manufacturer also commented on whether the power distribution has an impact on the gap width.

Accordingly, there are deformation patterns that are typical for individual reactors which have proven to be stable over several cycles (e.g. with regard to preferred directions or bending shapes). The formation of these patterns is mainly associated with reloading of the same FA design and the typical repositioning of the FAs during refuelling. Together with the laterally acting forces due to the cross flows, largely stable longer-term deformation patterns are formed in the core. These effects are considered by the manufacturer to be more significant for the gap width than possible influences of the fuel assembly power and can also explain the observed weak correlation between power and fuel assembly deformation.

In connection with RSK recommendation 5 that for control assemblies with a recognisable trend towards increased drop times more extensive measurements should be performed, the operators note that different bending shapes may lead to the same drop times. The radii of curvature and the amplitudes that occur with the individual bending shapes were decisive. The properties of the control assembly also played a role. A clear assignment of the drop time to fuel assembly deformation was not possible in a simple way.

Assessment

From the RSK's point of view, decreasing FA deformations in recent years confirm that an understanding of the essential causes of FA deformations has been achieved.

Given this fact, from the RSK's point of view, the approach to calculate the in-core deformations by a prediction of the expected EOC ex-core deformations and a comparison with actual EOC ex-core measurements is also basically suitable for a validation of the model. However, the calculated EOC values are compared again with the ex-core values measured in cold condition. The RSK sees no indication that relevant effects are not considered in the calculation of the transition from cold to hot condition and the calculation backwards to cold condition.

Using indicators such as the drop times of the control rods, it can qualitatively be concluded that the actual in-core deformations under operating conditions are smaller than those having occurred in cases of increased drop times. However, according to the operators, a quantitative statement on actual or maximum FA deformations cannot be derived from these indicators at present.

As regards the question to what extent the uncertainties relevant for the calculation of the FA deformations under operating conditions were quantitively assessed and whether sensitivity analyses had been performed, the manufacturer stated that the uncertainties of the individual parameters were not explicitly taken into account. According to [4], the water-gap distributions resulting from the FA deformations are to be understood as best-estimate water-gap distributions; uncertainties of the calculated water-gap distributions can be taken into account in the subsequent analysis steps by treating these distributions statistically.

Against the background discussed above, that in fact there is an uncertainty regarding in-core deformation of FAs during operation, that the model calculations on deformation however were additionally validated experimentally in cold condition and that even the derived predictions on deformations to be expected agreed with the straightness measurements within the limits of measuring accuracy, the RSK considers it sufficient to implicitly take into account the uncertainty in the calculation chain by the statistical treatment of water gap distributions (i.e. of different gap types and associated gap widths as well as their spatial distribution in the core).

3.2.2 Power density distribution and DNB analysis

Facts and circumstances

Report [4] presents further methodological developments for evaluating the impacts of changed gap widths on the local power density, on the one hand, based on CASCADE 3D code (AREVA) calculations, on the other hand, based on CASMO5/SIMULATE5 code (Studsvik/PreussenElektra) simulations. The report also presents the results of the application of these approaches to different reactor cores. Extensive ex-core straightness measurements were performed for cycles which showed deviations with regard to control assembly drop times, handling difficulties, FA damage or strong deformation measured in the previous cycle (deviations of the centre lines between straight and deformed FAs). Concretely, these are the KBR cycles 25 to 27 with maximum ex-core measured deformations of approx. 21 mm (calculated in-core 14 mm) and the GKN II cycle 28 with a maximum ex-core deformation of approx. 15 mm (calculated in-core 9 mm). For the KBR cores, the analyses with CASMO5/SIMULATE5 provides increases in the maximum linear heat generation rate of 2 W/cm, 6 W/cm and 34 W/cm (see Table 2 in [4]) for cycles 25 to 27, and 30 W/cm for the GKN II core, calculated with CASCADE 3D (see Table 1 in [4]).

The DNB ratios calculated for these reactor cores with the respective FA deformations are at maximum reduced by 2 to 3% (DNB points) for the KBR cycles 25, 26 and 27 and of 10 DNB points for the GKN II core (see [4], Table 2 and p. 24), respectively. According to the explanations of the operators in [4], unfavourable deformation conditions were quantified with the cores investigated which can be regarded as covering current and future cores with constant or reduced levels of deformation.

Table 2 of the VGB report [4] mentions typical $DNBR_{min}$ values for plants with 16x16 FAs in the range of 2.3 to 2.5[3]. Thus, changes in the range of 10 DNB points were clearly covered by the conservative assumptions applied in the analysis practice (according to [10], in particular this is due to the neglection of feedback effects between neutronics and thermohydraulics) so that there is no need to specifically calculate the impact of FA deformations on the DNB ratio for each cycle.

Assessment

Consideration of uncertainties of the calculation methods to evaluate the effects on local power densities or DNB conditions:

In the course of the deliberations on the consideration of the uncertainties in the analysis results, the working group in particular dealt with the uncertainties of the codes applied for the calculation of power densities (CASCADE-3D or CASMO5/SIMULATE5). In this respect, the operators refer to results of comparative calculations using "reference codes" (CASMO3/APOLLO-A or MCNP). According to these results, the highest under- and overestimations of the local power density for high-power rods and for an additional gap of 1 cm range symmetrically between 10% to 15% with a standard deviation of 5% (in the case of an arrangement of

[3] $DNBR_{min}$ ist he minimum DNB ratio during operation.

MOX/UO2 fuel assemblies, there was an overestimation of 27%). From the point of view of the operators, these uncertainties do not have to be considered separately but can be taken into account within the framework of the overall analysis procedure.

Altogether, the operators regard the "exemplary" applications of the methods to individual cores as "sufficiently realistic". However, no analyses have been performed so far which would allow a quantitative assessment of the uncertainties associated with the analyses presented in [4] regarding the impacts of changed water gap widths on the local DNB conditions.

In view of the simplifications, approximations or best-estimate approaches in the DNB analyses (e.g. with the determination of the various possible water gap distributions in the reactor core and the resulting local power density), the RSK is of the opinion that the uncertainties in the DNB behaviour resulting from the possible water gap distributions should be quantified. As addressed in RSK recommendation 3 [2] [4], sensitivity analyses should be conducted for this purpose (see the recommendation in the following section).

Covering character of the analyses carried out:

By means of the analyses presented in [4] for exemplary cores which, on the one hand, were measured well and, on the other hand, exhibited large deformations, it is to be shown generically that the acceptance criteria as defined in the regulations were or will be met with margin when taking into account FA deformations that occurred in the past or are to be expected in the future and that cycle-specific analyses on the impacts of FA deformations are therefore not necessary.

Against the background that

- with a reduction of the DNBR of e.g. 0.1 due to changed gap widths, as calculated for the exemplary GKN II core according to [4], in fact there are still considerable DNB margins. However, no sensitivity analyses have been presented so far which could be used to quantify the uncertainties due to changed gap widths, and

- such sensitivity analyses are regarded necessary by the RSK to confirm the covering and generic character of the presented exemplary DNB analyses,

the RSK recommends the following:

The uncertainties to be considered with the evaluation of the DNB behaviour due to changed water gaps in the reactor core should be quantified by means of suitable sensitivity analyses. As regards the method applied, the RSK holds the view that these analyses can be performed generically and analogously to the statistical LOCA analyses for the reactor cores (or cycles) and water gap distributions selected in [4]. The results (e.g. minimum

[4] "*To verify the existing exemplary findings on the influence of FA deformations on the core-wide minimum DNB ratio $DNBR_{min}$, additional sensitivity analyses should be performed.*"

DNBR for a probability quantile of at least 95% and a statistical confidence of at least 95%) are to be assessed in relation to the existing DNB margins for the leading transients. (**Recommendation)**

Core monitoring:

RSK recommendation 1 requires that a relevant increase of the maximum local power density must be considered when deriving the actual maximum power density from the measured values.

In this respect, the operators state [4] that for compliance with the design limits, the consistent interaction of the elements of core monitoring and safety analyses is necessary and that according to the nuclear rules and regulations, the maximum permissible power density for core monitoring is the power density at which the safety analyses can be successfully performed under consideration of all relevant uncertainties (here: in particular effects from non-nominal water gaps). The analyses presented in [4] show that the proof can be successfully provided if non-nominal water gaps are taken into account.

Firstly, the RSK states that due to non-nominal water gaps, power density changes may occur locally which may not be registered by the core instrumentation if these changes are not close enough to detector positions. Since, however, it was demonstrated that the accident analyses yield margins with respect to the acceptance criteria for the power density that are larger than the possible effect of non-nominal water gaps, it is not necessary from the point of view of the RSK to penalise the measured values of power density monitoring such that the effects from non-nominal water gaps are covered. The task to limit the power density such that initial conditions are maintained for accidents to be considered (protective limitation) is fulfilled with such an approach by a covering accident analysis. The RSK has therefore no objections to the methods applied by the operators.

3.2.3 LOCA analysis (using statistical methods)

3.2.3.1 LOCA analysis without consideration of changed gap widths

To prove compliance with the LOCA criteria, the operators use a statistical method, which has been dealt with and assessed in the recommendation adopted at the 143rd AST meeting [11]. For an understanding of the interrelationships, the main facts and circumstances that led to this recommendation are given below.

Nuclear rules and regulations

The "Safety Requirements for Nuclear Power Plants" [6] stipulate that the initial conditions to be applied in the case of a safety analysis using statistical methods shall be determined by means of realistic parameter values taking into account their uncertainty ranges. The uncertainties associated with the respective analysis result are to be quantified in their entirety and need to be taken into account. For this purpose, the parameters (initial and boundary conditions as well as model parameters) and models shall be identified which significantly influence the uncertainties of the results, and the ranges of uncertainty that exist according to current knowledge shall be quantified, together with the parameter distributions. When assessing the overall uncertainty with statistical methods, the one-sided tolerance limit in the direction of the acceptance criterion shall be calculated for a probability of at least 95% and a statistical confidence level of at least 95% to demonstrate compliance with the acceptance criterion.

With regard to the demonstration of safety for loss-of-coolant accidents, the RSK statement of the 475th meeting on 15 April 2015 [7] specifies as overriding standard for accident analysis to demonstrate the fulfilment of the acceptance criteria with high confidence. If the analyses use statistical methods, the statement provided must apply to the entirety of the fuel rods. Therefore, in case of statistical treatment of uncertainties, the criterion must be applied such that with a probability of at least 95% and a statistical confidence level of 95% not more than one fuel rod exceeds the acceptance criterion. From the point of view of the RSK, a method can be used which provides a statement about the entirety of the fuel rods by analysing a sufficiently large number of the unfavourable real fuel rods in the core. For this purpose, a preselection can be made from the total number of fuel rods in the reactor core or from the total number of calculation cases, e.g. based on results of calculations for the fuel rod state before LOCA or based on engineering judgement in order to limit the number of required thermal-hydraulic calculations to a practicable extent.

The RSK statement of the 475th meeting specifies that for determining the probability distributions, full load conditions are to be postulated conservatively. In addition to cycle pre-planning and operating experience, the operating modes expected during the cycle due to setpoints of limitations and, if applicable, other measures and provisions in place (e.g. administrative provisions for load cycles) which ensure compliance with the values and distributions used in the analyses are also to be considered for the integral power and the maximum local power density. The influence of fuel assembly deformations on the maximum power density are to be assessed additionally and taken into account where applicable.

Facts and circumstances

The VGB described the implementation of the requirements of the "Safety Requirements for Nuclear Power Plants" [6] and of the "RSK statement of the 475th meeting" [7] for LOCA analyses using statistical methods in [4].

The statistical LOCA hot rod analysis is an essential element of safety demonstration. Based on a core loading for one cycle under consideration of the preceding cycles, the power distribution is calculated as function of time and burn-ups using a core simulator. A stationary fuel rod code determines the fuel rod condition at LOCA initiation for all verification rods, the "hot" rods, i.e. the rods in the highest-power fuel assemblies and the rods in fuel assemblies in the surrounding channel and in the residual core. Afterwards, in the case of the statistical hot rod analysis, the accident analysis and the hot rod analysis are then performed for the LOCA (2A break) using a thermohydraulic system code and a transient fuel rod code.

When determining probability density functions for the consideration of uncertainties regarding power and power density, values of up to 106% are considered for the total reactor power and values of up to the second activation level of the reactor power limitation system are considered for the power density in accordance with the RSK statement [7].

With regard to the uncertainties related to the power density, the operators found a 1-sigma value of 3.5% for the power density distribution in the most unfavourable case (MOX fuel), which is conservatively applied to cover all fuel rods. To ensure that the statistical distribution reaches up to the value given by the limitation systems, the uncertainty limits of the power density distribution are defined with $\pm 3\sigma$ if 3σ includes the second activation level (LOCA limit) or with $n\sigma$ ($n > 3$) to reach the second activation level.

The statistical criterion for the maximum cladding temperature "at most one fuel rod exceeds the acceptance criterion with a probability of at least 95% at a statistical confidence of at least 95%" requires nine randomly determined parameter sets per fuel rod and thus about 500,000 (pre-Konvoi, Konvoi) calculations have to be performed for the entire core [4].

To limit the number of detailed thermohydraulic analyses with S-RELAP, the maximum cladding temperature (maximum of the cladding temperatures at the first or second peak) during LOCA (2A break) is approximated for all roughly 500,000 calculations by means of a correlation based on a multiple linear regression. This plant- and accident-specific correlation for the maximum cladding temperature (peak cladding temperature – PCT) is derived from the results of detailed variation calculations (using a core of KKP-2 as an example) that take into account the essential input and model parameters for the thermohydraulics (TH) code (fuel rod conditions, in particular linear heat generation rates and stored energies, reactor parameters before LOCA and model parameters of the TH code S-RELAP) which are subject to uncertainties. The standard deviation between the PCTs determined with this correlation and the PCTs from the S-RELAP analyses is approx. 15 to 20 K in the high PCT range.

The results of the correlation for the maximum cladding temperature for the approx. 500,000 calculation cases are ranked in descending order of the maximum cladding temperature. Afterwards, detailed TH calculations are performed with S-RELAP beginning with the fuel rods exhibiting the highest cladding tube temperatures according to the prediction based on the correlation. After processing a certain number of S-RELAP calculations and having reached a sufficient margin between the maximum cladding temperature of the last calculation run and the maximum cladding temperature obtained up to then, no further S-RELAP calculations are conducted[9].

Regarding the RSK opinion that also full load conditions with a pronounced maximum of the linear heat generation rate in the upper half of the core (power density distribution with peak in the upper core region) are to be considered in the probability distributions, operators and manufacturer argue having chosen a conservative approach in their analyses by multiplying the typical axial power density distribution occurring during operation (approximately trapezoidal, except for the initial core) at different times in the cycle (BOC, MOC, EOC) over the entire axial height by a factor which was statistically selected (random sampling) from the normalised distribution of the maximum power density, which is statistically derived from the operating data and extended to the LOCA activation value. With this approach, a redistribution to an axial power density distribution with peak in the upper core region, which cannot be derived from operating experience, does not occur. Nevertheless, possible high local power densities in the upper half of the core are statistically covered thereby. Operators and manufacturer justify this approach, among other things, by stating that the selected variation method for the local linear heat generation rate leads to conservative results with regard to the cladding temperature.

Regarding the most unfavourable point in time of the cycle to be considered in statistical LOCA analyses according to the SiAnf, the uncertainty analyses of operators and manufacturer show that no significant differences of the maximum PCT was recognised as function of time.

Assessment

Altogether, from the point of view of the RSK, it results from [11] and the questions discussed therein that the concept of the operators [4] for a statistical approach to demonstrate compliance of the maximum cladding temperature with the acceptance criteria for LOCA is comprehensible and basically corresponds to the RSK specifications in [7]. However, a statistical LOCA analysis in which the point in time during the cycle has also been statistically varied is to be evaluated in each case to identify whether there is a point in time leading to significantly unfavourable maximum cladding temperatures. In such a case, a dedicated statistical LOCA analysis should be performed for this/these point(s) in time of the cycle, otherwise statistical consideration of the point in time of the cycle is sufficient. It can be seen from the available statistical analyses that no relevant difference in the distribution of the maximum cladding temperatures results from the three investigated points in time of the cycle (BOC, MOC, EOC), [15].

The results reveal that, compared with the conservative deterministic LOCA analysis, the statistical LOCA analysis provides significantly lower maximum cladding temperatures (868 °C compared to 1,005 °C [9]) and that thus safety demonstrations using the conservative deterministic LOCA analysis are thus covering.

3.2.3.2 LOCA analysis considering changed gap widths

Regarding statistical LOCA analysis, this statement deals with the changes resulting from changed gap widths due to FA deformation, as requested in the RSK statement of the 475th meeting [7].

Facts and circumstances

In order to take fuel assembly deformations into account with statistical LOCA analysis, deformation measurements of 2012 and 2013 were selected for cycle 25 in KBR and cycle 28 in GKN II [4], which allow a reliable classification according to gap types and a statistical evaluation of the gap widths and which, according to the operators, cover the expected future conditions with regard to the largest measured gap widths. The deformation data for BOC and EOC were prepared such that they can be used as a basis for the generation of statistically varied water gaps in the entire core (in-core, hot condition). By considering these two plants, cores with S-shape deformations as well as cores with C-shape deformations are recorded in the database. After the classification of all water gaps (into ten gap types), a statistical evaluation is performed for each gap type. Due to the geometrical boundary conditions of the outer FAs at the core baffle, according to [4], there is an increased probability for large water gaps at the core edge. It was presented at the technical meeting [12] that in the applied statistical selection procedure (hereinafter simply referred to as the random sampling method), in particular the following measures were taken to relatively increase the frequency of large water gaps in the core:

- the random sampling method is only applied to gap types that cause potentially large water gaps in the core, and

- the random sampling method is not applied separately to outer (at the core edge) and inner gaps (this increases the probability of larger inner gaps).

The gap between two fresh fuel assemblies is the nominal water gap. Based on the gap distributions determined as mentioned above, random gap distributions are derived by means of random sampling. In this process, only the five gap types of the ten classified gap types are considered which create sufficiently large gaps (e.g. deformation of two fuel assemblies in opposite directions). The random sampling works as follows:

- sampling a gap position in the row/column under consideration; if this position has already been sampled once, then sampling is repeated,

- sampling the gap type,

- sampling the gap width according to gap type and gap class (inner or outer gap),

- checking whether the compatibility condition (total residual gap > 0 mm) is fulfilled, and

- checking whether the total residual gap of the row/column at any spacer level is less than a threshold value (9 mm). If the value falls below the threshold value, random sampling for this row/column is completed.

Finally, the total residual gap of each row and column and for each spacer position is evenly distributed over the gap positions which were not sampled.

With this random sampling method, BOC, MOC and EOC states are generated for 300 water gap distributions. In a further step, the calculation of the power density distribution is carried out exemplarily for a KKP-2 core using the AREVA code CASCADE-3D. The 95% / 95% value of the maximum linear heat generation rate with the 300 sampled water gap distributions amounts to 463 W/cm for the KKP-2 core at BOC. The mean value for the maximum linear heat generation rate at BOC amounts to 436.4 W/cm (see table in [13]).

For the water gap distribution in cycle 25 of KBR derived from the measured water gap distribution a maximum linear heat generation rate of 448.2 W/cm was calculated at BOC and of 439.4 W/cm in cycle 28 of GKN II [4].

The VGB report [4] concludes that the comparison of the 95% / 95% values derived by statistical methods with the also determined maximum values using the "measured water gap distributions" proves the conservatism of the applied statistical approach and that, thus, the method of random sampling of the water gaps is suitable to adequately model their impact on the power density distribution for the statistical LOCA hot rod analysis.

Assessment

From the point of view of the RSK and with regard to the above-described method applied by the operators to evaluate the impact of changed gap widths on the power density for the statistical LOCA hot rod analysis, it needs to be assessed whether this method is suitable to adequately evaluate the impact of changed gap widths.

Here, the RSK comes to the following conclusions:

- The 300 water gap distributions generated on the basis of the deformation measurements of 2012 and 2013 used for cycle 25 in KBR and cycle 28 in GKN II, for BOC and EOC states each, as well as for 300 interpolated MOC states, provide a good coverage of the frequencies of the individual gap types and, above all, of the maximum gap widths in the cores of KBR and GKN II.

- The application of these water gap distributions to a KKP-2 core (cycle 22) results in distributions of the maximum linear heat generation rate which, with their mean values, are close to the maximum values calculated for the water gap distributions in the KBR and GKN II plants and, with their peak value, come close to the LOCA activation level of the peak-RELEB.

- The selection of the deformation measurements of 2012 and 2013 used for the application of the method to cycle 25 in KBR and cycle 28 in GKN II is appropriate. Although these measurements do not necessarily cover all maximum gap widths measured in this period, they show changed gap widths having occurred in two cores and provide a sufficiently representative basis from the point of view of the RSK.

- In the opinion of the RSK, the number of 300 core variations generated to assess effects of changed gap widths is large enough to adequately reveal the ranges of variation with regard to a statistical LOCA hot rod analysis.

In summary, the RSK comes to the conclusion that the analyses presented by the operators have shown that in view of the conservatism connected with the conservative deterministic analyses, also non-nominal water gaps due to FA deformation do not challenge the safety-related information of a conservative deterministic analysis. The RSK assessment provides that the deformations of the cores referred to by the operators in their analyses remain covering in future.

3.3 Implementation of RSK recommendation 4

Recommendation 4 of the RSK statement of 18 March 2015 [2] refers to the effects of FA deformations on the stress and stability analysis of the fuel assemblies.

Impact on stress and strength analyses

FA deformations have an impact on specified normal operation as well as on the control of design basis accidents, external hazards and emergency conditions. Here, the impact of possible fuel assembly deformations on parameters to be considered within the framework of the FA structural design can be limited to the stress and stability analysis.[2]

The main objective of the stress analysis is to prove that inadmissible plastic deformations are prevented (stress criterion). The stability analysis is to show that guide tube buckling does not occur (stability criterion). In the previous design reports for the fuel assembly structure, fuel assembly deformations with amplitudes as determined during operation of German PWR plants were not considered, neither for specified normal operation nor for accidents and external hazards. It is the objective of recommendation 4 to remedy this deficit in the previous safety analyses by showing that the design margins cover the possible effects of fuel assembly deformations on the stress and stability analysis.

Facts and circumstances

The VGB Report [4] provides the results of new AREVA analyses with regard to recommendation 4. The investigations for normal operation were performed for HTP 16x16 and 18x18 fuel assemblies without measures to improve stiffness. The 20 mm C-shape deformation and 15 mm S-shape deformation were considered and their influence on the strength analyses was quantitatively evaluated. According to [4], during normal operation, the additional bending stress in the guide tube remains below 30 MPa and the remaining margin to the design limit for plastic deformations of the guide tube is about 40 MPa. This corresponds to a safety factor of 1.6.

The behaviour during accidents was analysed for the current 18x18 fuel assembly design HTP and HTP-I. These accident analyses take into account the operating experience at plants with 18x18 FA design with 20 mm C-shape deformation and 10 mm S-shape deformation. According to [4], the analyses yield a remaining design margin of more than 30 % for the sum of membrane and bending stresses. The calculations were carried out with conservative boundary conditions: A free-standing fuel assembly without lateral support by neighbours was considered, so that an increase of the axial force cannot be absorbed by the neighbouring fuel assemblies but only by higher deformation. Furthermore, the admissible stresses are derived from the properties of unirradiated material; the strength of irradiated material is approx. twice as high. In addition, it is to be taken into account that the strength analysis was carried out for a static load using tensile tests while strength is higher for short-term loads. Moreover, in general, the stress level during normal operation is lower for 16x16 fuel assemblies than for 18x18 fuel assemblies [14].

Furthermore, AREVA provided the proof that the buckling stress limit is not exceeded and that the guide tubes do not buckle. In addition, a stability analysis of a deformed fuel assembly was presented. This analysis shows that an additionally applied load on the fuel assembly, which is subjected to the operating loads, generates only small additional deflections and additional stresses. With regard to stability analysis, this means that the fuel assembly has sufficient stiffness also in the deformed state and can take up additional loads.

Assessment

Ex-core S-shape deformations in the order of 15 mm have been observed in the past operating experience of plants with 16x16 FAs. Referring to various conservatisms included in the accident analyses performed, from the point of view of the RSK, the operators and manufacturer demonstrated comprehensibly that for 16x16 fuel assemblies, the stress limits are met in the event of an accident also with S-shape deformations of 15 mm. This also applies to normal operation. Likewise, the analyses conducted with regard to the stability of the fuel assemblies are suitable to show that the fuel assembly has sufficient stiffness also in the deformed state and can take up additional loads.

In summary, the RSK holds the view that it was shown plausibly that the design margins cover the possible effects of the FA deformations on the stress and stability analysis.

3.4 Implementation of RSK recommendations 5 to 8

Recommendations 5 to 8 from the RSK statement of 18 March 2015 [2] concern the procedure in the case of increased control rod drop times, the handling of deformed fuel assemblies during plant operation and the assessment of the design features of fuel assemblies with regard to possible FA deformation.

In their report [3], GRS and PHB come to the following results regarding RSK recommendations 5 to 8:

Recommendation 5:

Although no trend towards higher drop times could be identified in some plants, drop time measurements with recording of way-distance oscillograms, freedom of movement tests, and random FA straightness measurements as well as additional event-related measurements (e.g. drop time measurements) are performed. The measures laid down in the plant regulations (e.g. in the operating manual, testing manual) are sufficient to reliably detect a trend towards higher drop times.

At KKI-2 in 2011, a control assembly (CA) reached the lower end position only with a considerable delay. This circumstance is accommodated by extensive measures. In KKI-2, for example, way-distance oscillograms are recorded and analysed periodically before shutdown/start-up, independent of the measured drop times. Fuel assembly / control assembly (FA/CA) configurations with the highest drop times are additionally subjected to further inspections during the maintenance and refuelling outage (optical findings, straightness measurement using a special measuring equipment). Corresponding specifications can be found in the testing manual. In addition, FA/CA lifting and set-down processes are monitored by measurements of the load and friction behaviour and there are specifications for positioning deformed FAs in the subsequent cycle.

The KBR plant has also been significantly affected by FA deformations. In the past, extensive measures were taken to cope with this issue. In particular, the approach proposed in RSK recommendation 5 has been implemented. Scope and contents of FA inspections, friction force measurements of CA/FA, drop time measurements etc. are specified in the written regulations of KBR.

Recommendation 6:

In all PWR plants, random straightness measurements are carried out during the maintenance and refuelling outage, irrespective of whether or not indications of relevant deformations exist. The random measurements cover 5 to 15 % of the FAs used in most of the plants. This scope is sufficient if it includes potentially particularly affected FAs (FA types, residence times, pre-deformation). The inspection programme is co-ordinated with the authority and the authorised expert prior to refuelling. If “anomalies” are detected, an extension of the inspections is stipulated. Some plants refer to a corresponding constraint in the operating licence. In the past, 100 % straightness measurements were performed at KKP-2, KKI-2 and KBR. For the other plants, the above-mentioned general stipulation allows to agree on an individual procedure for each “anomaly”.

Recommendation 7:

With regard to management and monitoring of the handling of deformed FAs, [3] specifies suitable measurement methods (e.g. CA drop time measurements, CA friction force measurements, recording of runtime data by means of way-distance oscilloscope, load-displacement recordings) and criteria (MAX and MIN load values defined on the basis of empirical values of loads occurring during fuel assembly handling) which can be used to assess whether there is an increased risk of FA damage. Suitable special handling devices as well as other aids (e.g. dummy FAs, underwater cameras which are e.g. attached to a “submarine”, FA positioning system) and procedures (e.g. laterally offset lifting: a FA which is freely standing on three sides is lifted by 150 mm by means of the FA loading machine and then moved sideways out to prevent contact with neighbouring FAs) are available and used to minimise the risk of FA damage when handling deformed FAs.

Recommendation 8:

Supplementary qualitative assessment parameters (e.g. stiffness of the entire FA and FA structure as well as the creep behaviour of the FA structure) were identified in accordance with RSK recommendation 8 and corresponding design modifications have been implemented in the plants. The design modifications were introduced and assessed pursuant to common practice regarding the introduction of new FA designs or significant design modifications. For this purpose, an AREVA tool was used to estimate the impacts of the design modifications on the straightness of the FAs. As a methodology for the derivation of information on the deformation potential of the FAs in the reactor core, the use of lead assemblies and their analysis and measurement during refuelling is mentioned as an example.

Overall, it is shown that appropriate supplementary parameters for the assessment of the design features with regard to FA deformations have been identified. Within the design, the selected construction was assessed with regard to deformation under consideration of these parameters.

Assessment of implementation of RSK recommendation 5 to 8

Based on the above statements given in report [3], the RSK comes to the conclusion that RSK recommendations 5 to 8 have been implemented.

3.5 Implementation of RSK recommendation 9

In its statement of 18 March 2015 [2], the RSK had referred to the statement on neutron flux fluctuations in PWRs of 2013 [1]. Reference was made to more recent investigations to clarify the causes of neutron flux fluctuations according to which there could be a correlation between the magnitude of the amplitudes of the neutron flux fluctuations and the level of fuel assembly deformation. Thereupon, in [2], the RSK had recommended to continue the clarification of causes 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.

As requested in recommendation 9, root cause analyses on neutron flux fluctuations have been intensified in recent years. To date, no results are available which would allow a full explanation of the spatial correlation of different neutron flux signals and of the fluctuation amplitudes.

The more recent model approach of collective flow-induced FA vibrations is, according to the current state of knowledge, able to exactly model the phase relationships between the neutron flux signals and to qualitatively describe the spatial distribution of the fluctuation amplitudes.

The RSK welcomes that the operators continue supporting the relevant research projects by providing measurement data.

4 Summary and recommendation

In summary, the RSK concludes that, with the exception of the recommendation given below, the recommendations of RSK statement [2] were taken into account and the analyses performed by the operators and the results achieved are plausible. However, the following **recommendation** remains to be complied with:

The uncertainties to be considered in the calculation of the DNB behaviour due to changed water gaps in the reactor core should be evaluated by means of suitable sensitivity analyses. As regards the method applied, the RSK holds the view that these analyses can be performed generically and analogously to the statistical LOCA analyses on the basis of the reactor cores (or cycles) and water gap distributions selected in [4]. The results (e.g. minimum DNBR with a probability of at least 95% and a statistical confidence of at least 95%) are to be assessed in relation to the existing DNB margins for the leading transients.

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A1 Annex 1: RSK recommendations on “Fuel assembly deformation in German pressurised water reactors” of 18 March 2015 [2]

Recommendation 1:
The influence of FA deformations on the maximum power density in the reactor core is to be evaluated. If 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.

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.

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.

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

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 way-distance curves and/or way-distance oscillograms during the cycle. Corresponding specifications should be included in the operating procedures.

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.

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.

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.

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.