RSK STATEMENT

Safety aspects of the use of high burn-up fuel elements under reactivity accident conditions

1 Advisory request

At the 372nd meeting of the RSK on 27.05.2004, the BMU requested the RSK to prepare a statement on the permissible enthalpy increase for fuel rods with and without oxide layer spalling during a reactivity-initiated accident (RIA).

2 Course of discussions

Preliminary remark: *At its 320th meeting on 16.09.1998, the RSK made a statement on the use of fuel elements with high burn-ups [1]. This statement includes, among other things, recommendations directed to operators of nuclear power plants referring to the design of fuel elements. In particular, an in-depth review of the fuel element design by experiments is recommended. In its statement, the RSK requested manufacturer and plant operators, among other things,*

- *to determine burn-up-dependent RIA defect limits for UO_2 and MOX fuel elements on an experimental database by participation on international experiments (CABRI), and*
- *to obtain a report of experiences on international measurements on RIA defect limit for high burn-up of fuel elements*

and expressed its opinion that a transient fuel rod code for RIA conditions has to be developed in addition to the fuel rod design code of the manufacturer.

At the 122nd meeting on 08.12.1999, the RSK Committee on REACTOR OPERATION dealt with the use of fuel elements with high burn-ups.

For an in-depth preparation of the discussion of safety-related issues in connection with high burn-ups, it was proposed at the 330th meeting of the RSK on 04.05.2000 in agreement with the chairman of the RSK Committee on REACTOR OPERATION to establish a Working Group on HIGH BURN-UP in the RSK Committee on REACTOR OPERATION. This working group received the following mandate by the RSK:

It discusses safety-related questions in connection with the use of highly burnt fuel elements in light-water reactors. Its discussions serve to prepare the forming of opinion and statement in the RSK Committee on REACTOR OPERATION and in the RSK.

Regarding RIA, the Working Group on HIGH BURN-UP held discussions at the 2nd meeting on 18.01.2001, at its 3rd meeting on 22.03.2001, at its 8th meeting on 19.06.2002, at its 9th meeting on 03.09.2002, at its 10th meeting on 19.11.2002 and at its 11th meeting on 20.02.2003 [2-21] on

- the PIRT activities and international assessments of the phenomena associated with higher burn-ups,
- the report of the manufacturer SNP on the use of fuel elements with high burn-ups,
- the report of the manufacturer Westinghouse Atom on the experiences regarding fuel elements with high burn-up at BWRs,
- the new results and test programmes on the behaviour of highly burnt fuel elements during RIA,
- the results of an evaluation of literature on the effects of high burn-up in case of RIA, and
- the influence of the pulse width during RIA on fuel rod loads.

The available report was discussed by the Working Group on HIGH BURN-UP at its 12th and 13th meeting on 24.04.2003 and on 03.07.2003 and subsequently approved according to § 16 (1) of the articles of the Reactor Safety Commission of 22nd December 1998.

At its 158th meeting on 25.02.2004, the Working Group on HIGH BURN-UP informed the RSK Committee on REACTOR OPERATION, among others, about the above-mentioned report. The draft of the statement was adopted at the 161st meeting on 25.08.2004 by the RSK Committee on REACTOR OPERATION. The RSK discussed the draft and adopted the statement at the 379th meeting on 27.01.2005.

3 Assessment criteria applied

3.1 Requirements

Regarding the reactivity-initiated accidents on which the design for German PWR and BWR plants is based, namely, e. g., “fuel rod ejection” in PWRs and “rod drop” in BWRs, all requirements resulting from the safety criteria of the Federal Ministry of the Interior (BMI), the RSK guidelines for PWRs and the incident guidelines must be fulfilled for all burn-up and core conditions. According to the BMI safety criterion 3.1 on reactor design, the reactor core shall be designed and manufactured such that the compliance with the load limits defined for specified normal operation and incidents is ensured for the load of the fuel elements and the other safety-relevant plant components during the entire duration of their use. According to the RSK guidelines, Chapter 3.1 (10) and (11), the controllability of an accidental control rod withdrawal or ejection shall be verified. It shall be verified that the energy release due to ejection of the control rod with the largest reactivity value will definitely not cause any damage to the reactor core and the reactor cooling system. Further, the reactivity-initiated accidents are assumed to be controlled in the incident guidelines so that no radiological consideration is required for them. According to KTA 3101.2, Chapter 4.1 a), an incident leading to an increase of the power density in the core shall not result in impermissible loads of the fuel rods. In practice, compliance with these requirements is currently verified by demonstrating the maintenance of the fuel rod integrity for RIA.

3.2 Acceptance criteria

As acceptance criterion for the integrity of the fuel rod, the "radially averaged enthalpy increase" or the "radially averaged enthalpy" at the fuel rod position with maximum power is referred to. The permissible value shall be determined from the comparison with experimental database.

For a higher burn-up, Siemens/Framatome ANP and the power utilities base the design on a burn-up dependent defect limit of the fuel rods which is oriented towards the ROS curve proposed by EPRI. Here, the proposed defect limit for the fuel rod integrity for fuel rods with a very low burn-up decreases from 170 cal/g fuel enthalpy increase to 100 and for high burn-up fuel rods (> 50 MWd/kg) without or local oxide spalling to 60 cal/g for PWRs. For BWRs, limit curves defined by plant operator and the authorised expert organisation are used in the individual case which are conservatively oriented towards the EPRI ROS curve.

In its statement of 1998, the RSK agrees to the provisional use of the value of 100 cal/g for fuel rods with a burn-up > 50 MWd/kg and cladding tubes without oxide spalling, as proposed by Siemens (today: Framatome-ANP) and recommends at the same time to determine the defect limits with a reliable experimental database for the fuel rods used in German LWRs.

The authorised expert organisations currently base the quantitative analysis of RIAs on the limit curve proposed by the RSK for the maximum radially averaged fuel enthalpy increase for fuel rod burn-ups > 50 MWd/kg (100 cal/g). For fuel rods with cladding tubes where oxide spalling may occur, 60 cal/g are referred to as defect limit. Below a burn-up level of 50 MWd/kg, the burn-up dependent enthalpy limit follows the ROS curve proposed by EPRI. An exception from this ROS curve is made for much lower burn-up levels. The acceptable enthalpy limit is here set constant to 140 cal/g. For a burn-up of about 33 MWd/kg, this results in a value of 140 cal/g as upper defect limit for the maximum radially averaged fuel enthalpy increase. The values are both used for uranium and MOX fuel rods.

In addition, the authorised expert organisations referred and refer to the entire field of experimental results available at the time of the analysis for the assessment of subsequent cores. In this respect, it is examined to which extent the experimental results can be applied to the subsequent core under consideration of the respective plant and have to be considered with regard to the defect limits (individual case study). Until final validation of the limiting enthalpy curve, the limiting curves used by foreign institutions are also referred to in the consideration.

So, the TÜV Nord also checks, on the basis of new findings published in connection with RIA experiments with BWR fuel rods in Japan, whether BWR fuel rods with burn-ups > 55 MWd/kg show calculated enthalpy increases > 50 cal/g in order to find out, where appropriate by means of an individual case study, whether a lower defect limit is to be considered according to the Japanese results.

Regarding the definition of the value for the defect limit, the RSK arrives at the following results: The verification of the fuel rod integrity is also the verification that definitely no damage to the reactor core and the reactor cooling system will occur due to the reactivity-initiated accidents to be postulated. The necessary precaution against fuel dispersion into the coolant is then taken conservatively because it is conservatively ensured when applying the acceptance criteria for the fuel rod integrity that the margin against this failure limit is sufficiently large. Since with regard to the radiological consequences the safety analyses can be performed realistically, the acceptance criteria for the fuel rod integrity may also be specified realistically.

The RSK is of the opinion that in case that scaling laws (see Chapter 4.2 below) are fulfilled, those burn-up dependent values for the enthalpy increase are referred to as acceptance limits which clearly separate the fuel rod experimental data with defects from that with no defects. If, in the individual case, the values shall be extrapolated beyond the range of experience, the reliability of the extrapolation shall be substantiated taking into account the full scope of available experimental data by means of detailed theoretical considerations using, where appropriate, transient fuel rod codes. The latter shall also apply if in the individual case experimental data shall be used for the definition of the defect limit values for which the applicability conditions are only fulfilled to some extent.

3.3 Test scope

The criterion for the maintenance of fuel rod integrity during reactivity-initiated accidents is checked for each refuelling. The calculation of the maximum fuel enthalpy or the maximum enthalpy increase is carried out for representative core loadings with 3D core models. The compliance of the planned refuelling is to be demonstrated for relevant core parameters, such as the maximum reactivity addition due to the control element, the power density distribution, the kinetic parameters such as β_{eff} and the lifetime of the prompt neutrons as well as the reactivity coefficients.

4 Assessment

4.1 Relevant high burn-up effects during reactivity-initiated accidents

A detailed description of the changes of the fuel pellet, the gas gap and the cladding tube in case of high burn-ups was given in Chapter 3.3.1 of the report on safety aspects of the use of high burn-up fuel elements during specified normal operation, under loss-of-coolant-accident (LOCA) conditions and under reactivity-initiated accident conditions *(Sicherheitsaspekte des Einsatzes hochabgebrannter Brennelemente im bestimmungsgemäßen Betrieb, unter Kühlmittelverlust-(KMV-) sowie unter Reaktivitätsstörfall-(RIA-)Bedingungen*, appendix to the minutes of the 372nd RSK meeting on 27.05.2004).

During a reactivity-initiated accident, the load of the fuel rod is caused by a fast and high power pulse. This power pulse leads to thermal heat-up of the fuel pellets. The thermal expansion of the fuel leads to pellet-clad mechanical interaction (PCMI), which is considered as dominant load mode for the fuel rods. In the international scientific and technical literature, the contribution of the transient fission gas release from the fine-grained high burn-up structure of the pellet to the load of the cladding tube is discussed controversially [6]. The determination of the transient fission gas release from the high burn-up structure is subject of experimental investigations within the French SILENE Programme.

The behaviour of the fuel pellets during the fast power pulse determines the loads of the cladding tube. The behaviour of the cladding tube under these loads depends on its ductility and the mechanical strength and this again depends on the temperature and the condition of the cladding tube. Of importance for the cladding tube condition are thickness of the oxide layer, occurrence of oxide spalling, the mean hydrogen content, the local hydrogen distribution in the cladding tube including the hydride orientation in radial or tangential direction as well as local corrosion effects. The ductility of the cladding tube is decisively influenced by the hydrogen content. The hydrogen is generated by oxidation of the zirconium and is absorbed by the cladding tube to a certain degree which depends on the material type. When exceeding temperature-dependent limit concentrations, the hydrogen precipitates as hydrides. The deteriorating influence of hydride precipitation on the fuel rod cladding ductility depends on its concentration and orientation. Radially oriented hydrides lead to a larger reduction of the ductility compared to tangentially oriented hydrides. Oxide spalling may result in the formation of local cold spots and thus in the formation of local hydride accumulations which might cause a substantial reduction of the cladding tube ductility.

In PWRs, different types of fuel rod cladding alloys are used which clearly show a different corrosion behaviour. The degree of hydrogen absorbed by the cladding tube also depends on the material. This results, at identical residence times, in significant differences with regard to corrosion level and hydrogen content. The ZrNb materials increasingly used, e. g. M5, show lower corrosion and reduced hydrogen absorption compared to Zry-4 so that all in all a more favourable initial situation for the transient fuel rod behaviour is to be assumed.

For BWRs, oxide spalling is not to be expected due to the oxidation of the Zry-2 cladding tubes which is significantly lower compared to PWRs also during higher burn-ups. Thus, the hydrogen absorption of the cladding tube also remains significantly lower compared to PWRs. However, there are indications for BWR cladding tubes that, depending on the manufacturing method, radially oriented hydrides can precipitate in case of higher burn-ups to a larger extent.

For the fuel rod behaviour under reactivity accident conditions, the pulse width of peak power and the initial temperature of the coolant and the cladding tube are also of importance. Calculations showed that smaller pulse widths lead to the formation of higher stresses in the cladding tube with lower cladding tube temperatures at the same time. This results in a higher cladding tube load at comparable enthalpy increase. Lower cladding tube temperatures lead to a reduced ductility of the cladding tube and intensify the influence of the hydrogen on the ductility of the cladding tube. The influence of the cladding tube temperature is clearly demonstrated in the material tests. From this it follows that the lower initial temperatures in case of reactivity-initiated accidents in BWRs occurring under “zero load cold” conditions, intensifying the ductility reducing effect of the hydrogen compared with PWR conditions.

4.2 Conclusions

On the basis of its discussions on the behaviour of high-burn up fuel elements under reactivity accident conditions, the RSK gives answers to the following questions:

1. Which applicability conditions are to be considered for the experimental results regarding the verification of the derivation of the fuel rod defect limits?

2. Will the criteria included in the RSK statement [1] further be suitable?

3. Have the recommendations of the RSK statement of 1998 on RIA been implemented?

Regarding the first question concerning the applicability conditions of the experimental results in the verification analysis, the RSK arrives at the following result:

The available experimental results regarding the determination of the defect limit of the fuel rod cladding tubes should be referred to to the full extent. For the applicability of the experimental database, the main factors are: burn-up, the fuel type (UO_2 or MOX), the cladding tube condition regarding external corrosion, hydrogen content and hydride orientation, the gap width and the cladding tube material, as well as the experimental conditions regarding pulse width and initial temperature.

For the assessment of the fuel rod behaviour, the values for the radially averaged fuel enthalpy determined in the core calculations on the reactivity-initiated accident shall be compared with the experimental results and the defect limit derived from it. The calculations should consider, as is practised, conservative assumptions for the model parameters and the initial and boundary conditions.
The RSK recommends that the plant operators and authorised experts of the RSK submit a status report on the calculation methods used for the verification analyses for the determination of the enthalpy increase. These status reports should present the models used, the determination of the burn-up dependent effects and

the consideration of the uncertainties in the calculations.

Regarding the second question concerning the suitability of the assessment parameter “radially averaged enthalpy increase” and the criteria used so far in accordance with the RSK statement [1], the RSK states the following:

The available experimental results on fuel rod behaviour under reactivity accident conditions, i. e. the results from the early SPERT, PBF and NSRR experiments as well as recent results from CABRI and NSRR experimental programmes, are characterised by specification of the radially averaged enthalpy increase. The fuel rod defect limit is equivalent to the value reached at the time of the failure of the radially averaged total enthalpy of the fuel or the radially averaged enthalpy increase, respectively.

Until now, it has not been specified internationally at which values for the enthalpy increase or burn-up dependent enthalpy fuel rod defects cannot be excluded. Decisive is always the available experimental database supplemented, as the case may be, by evaluations with transient fuel rod codes.

In its statement on the burn-up dependent defect limits for reactivity-initiated events (RIAs) [1], the RSK agreed to the provisional use of the value of 100 cal/g proposed by Siemens for fuel rods with a burn-up > 50 MWd/kg (fuel rod segment) without oxide spalling. The results of recent experiments confirm this value. As, however, in two experiments (REP Na1, REP Na7) fuel dispersion occurred, in one case with a pulse width of 40 ms, the RSK proposes to reassess the safety margin for PWRs and recommends for RIAs a limit values of 80 cal/g at a fuel rod segment burn-up > 65 MWd/k for fuel rods without oxide spalling (with advanced, corrosion-resistant cladding tube materials, such as Duplex or M5) (see Figure 1).

For fuel rods where oxide spalling is to be postulated, a limit value of 60 cal/g shall be referred to for burn-ups > 50 MWd/kg as already practised (see Figure 1).

For BWRs, the same limit values for the fuel rod defect limit may be used as for PWRs, i. e. 80 cal/g for burn-ups > 65 MWd/kg and fuel rods without oxide spalling. For BWR RIAs with low initial power or from zero load cold, a case-by-case examination is required for fuel rod segment burn-ups > 50 MWd/kg with calculated enthalpy releases > 60 cal/g. Here, the applicability of the results of Japanese BWR RIA experiments shall be assessed regarding the corrosion condition of the cladding tubes used. This may lead, e. g., to a statement on the admissibility if the hydrogenation condition is better than in Japanese plants (cladding tube condition). An admissibility may also be derived from the consideration of the larger pulse width than in the Japanese experiments.
For MOX fuel, only three datasets from integral experiments are available for burn-ups between 47 and 65 MWd/kg. From the point of view of the RSK, it is not to be expected on the basis of these three experiments and under consideration of further results from MOX-specific single effect experiments that the differences regarding the transient fission gas release between UO_2 and MOX fuel for higher burn-ups, observed in experiments so far, become so large that they significantly increase the cladding tube loads. However, the current and future experimental and analytical studies on the transient fission gas release shall be watched within the course of the burn-up increases for MOX fuel elements accordingly.

From the point of view of the RSK, the above-mentioned limit curves for the maintenance of the fuel rod integrity may thus generally be used for UO_2 and MOX fuel because, according to current knowledge, the status of fuel rod corrosion and its load-bearing capacity are the main influencing factors for the fuel rod integrity. When limiting the cladding tube corrosion (no oxide spalling), an explicit burn-up limitation is not required; the permissible burn-ups result from the area of the experimental database and the above-mentioned applicability conditions.

For the derivation of this defect limit it was taken into account that for the core designs currently realised at German PWR and BWR plants the pulse width of the peak power is larger than 30 ms during a RIA.

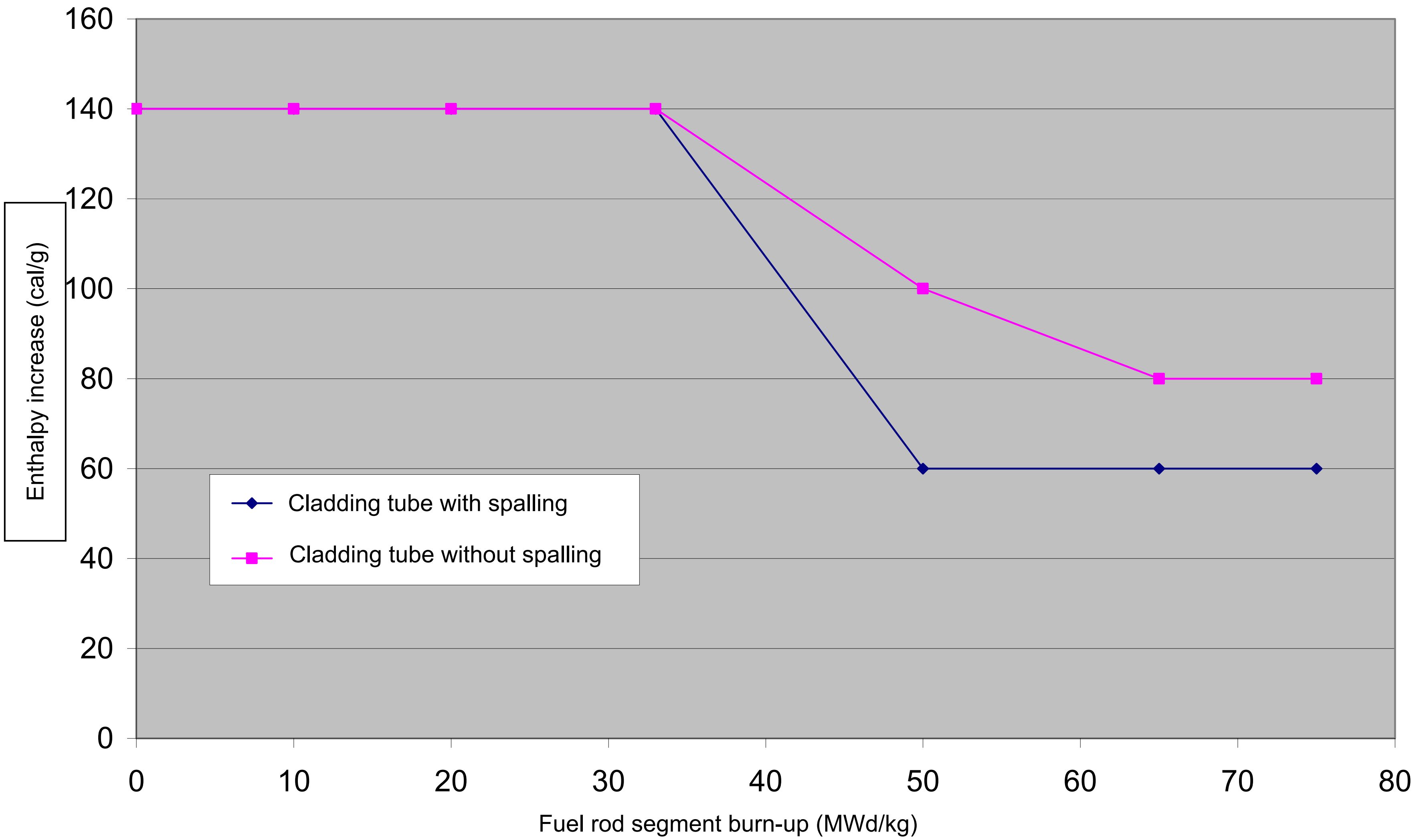
The limitation of the maximum uranium enrichment to 5 % results from the design requirements for the manufacture and transport of the fuel elements and also from the fuel element strategies generally applied today. Under these conditions, maximum fuel rod burn-ups of about 75 MWd/kg can be reached during reactor operation.

Regarding the third question concerning the implementation of the RSK recommendations on the RIA-related problems it can be stated that the recommendation on the expansion of the experimental database for high burn-ups has been complied with by the participation of the German plant operators in the CABRI waterloop project. For the experimental programme, the German plant operators also provide own fuel rods with high burn-ups. It is expected from the experimental programme that experimental results are obtained for the higher burn-ups planned for use in reactors and that the understanding for the relevant effects of cladding tube failure increases. The performance of the CABRI experimental programme should be accompanied as planned. However, the experiments intended to be performed within the framework of the CABRI programme will mainly provide experimental data that are representative for PWR conditions. The RSK therefore recommends taking efforts, e. g. within the framework of the international exchange of information of the RSK or the BMU, to also obtain access to the details of the BWR-related experiments performed in Japan.

In its statement of 1998, the RSK recommended the development of a transient fuel rod code for the fuel rod behaviour during RIAs. Internationally, the fuel rod codes FALCON, FRAPTRAN and SCANAIR have been further developed and applied for the post-calculations of test fuel rods under RIA conditions of the NSRR and CABRI programme. The issues of modelling of the physical processes in the fuel pellet, the interaction between pellet and cladding tube and the mechanical behaviour of the cladding tube under the loads under RIA conditions have not been satisfactorily solved in full for high burn-ups. However, the simulation of the transient fuel rod behaviour makes an important contribution to the determination of the relevant contributions to the cladding tube loads and supports the application of the results from test to reactor conditions. GRS adopted the SCANAIR code of IRSN to further develop the calculation methods for the transient fuel rod behaviour and to investigate to which extent the cladding tube failure can be calculated with improved models. The RSK should receive reports on the experiences with the use of the code and an assessment of the potential of the transient fuel rod code for the description of the fuel rod behaviour in case of high burn-ups under RIA conditions.

Against the background of the continuing international discussions [22, 23] and the current experimental test programmes, the RSK also recommends to resume the discussion on the RIA defect limit in about three years.

Figure 1: RIA: Permissible enthalpy increase for fuel rods with and without oxide spalling



List of abbreviations

EPRI	Electric Power Research Institute
KMV	Kühlmittelverlust
NRC	Nuclear Regulatory Commission
NSRR	Nuclear Safety Research Reactor
PBF	Power Burst Facility
PIRT	Phenomena Identification and Ranking Table
PCMI	Pellet Clad Mechanical Interaction
RIA	Reactivity Initiated Accident
ROS	Region of Success
SNP	Siemens Nuclear Power
SPERT	Special Power Excursion Reactor Test

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