Gesellschaft für Reaktorsicherheit (GRS) mbH

PROPOSED CRITERIA FOR THE EVALUATION OF FUEL ELEMENT DAMAGE DUE TO LOSS OF COOLANT IN PRESSURIZED WATER REACTORS
(a discussion document)

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J. Keusenhoff

Schwertnergasse 1 D-5000 Cologne 1 . Telephone +49 (0)2 21 20 68-0 . Telex 8 881 807 grs d

Preamble

This report is a final report relating to work-item A 1.9 of resolution SR 271/2.
- Development of proposals for the formulation of criteria relating to damage measurement

There are further final reports relating to the following items of the work-plan:
- Preparation of a blockage model based on the coolant flow-rate

P. Pana, J.D. Schubert, A.K. Chakraborty
A blockage model for calculation of coolant flow hindrance due to ballooning of the cladding
GRS-A-1233, January 1985

- Follow-up of current research projects into fuel element (BE) behavior in order to check the criteria proposals developed

A. K. Chakraborty
Summary comparison of out-of-pile and in-pile bundle studies into the distortion characteristics of fuel rods in the event of malfunction causing coolant loss
GRS-A-945, March 1984

J. Keusenhoff
Bursting of zircaloy fuel rod cladding subjected to internal pressure loading
An evaluation of experimentally based calculation models GRS-A-1232

- Testing and modification of new program versions for application in the approval procedure
- Exchange of experience and feedback with developers of fuel rod behavior programs

No related report was agreed upon for these work-items - the results of the work are included in two dissertations:

J.G. Keusenhoff, J.D, Schubert, A-K. Chakraborty
A Model of Asymmetric Ballooning and Analyses of Ballooning Behavior of Single Rods with Probabilistic Methods Transactions of 7th Int. Conf- on SMIRT
Chicago, USA, August - 22-26, 1983

J-G- Keusenhoff, J-D, Schubert, A-K, Chakraborty
Development and Application of an Asymmetric Deformation Model to describe the Fuel Rod Behavior during LOCA, OECD-NEA-CSNI/IAEA Specialists' Meeting on Water Reactor Fuel Safety and Fission Product Release in Off-Normal and Accident Condition. 16-20 May 1983, Risǿ National Laboratory - Denmark

ABSTRACT

The effectiveness of the Emergency Core Cooling (ECC) criteria formulated in the guidelines of the RSK (RSC - Reactor Safety Committee) for Pressurized Water Reactor (PWR) plants was checked and new proposals are made. The temperature and the oxidation criteria can be withdrawn. For large ruptures not more than 10% of fuel rods should be expected to fail. For small ruptures a temperature-time relationship is given as a limit.

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INTRODUCTION

The efficacy of the emergency cooling system of a pressurized water reactor for the control of a coolant loss is handled in the RSK guidelines. In item 22.1.1 of the guidelines 5 individual requirements are mentioned for the core emergency cooling. In item 22.1.3 the assumptions for the verification of efficacy are specified in further detail.

On the basis of newer findings and experience of the approval procedure it is possible to define these assumptions and requirements or to redefine the emphases. This applies particularly in respect of the requirements for the capability of cooling the core, which can now be represented in more detail on the basis of further analytical and experimental knowledge with regard to fuel rod behavior. In anticipation of any further definition, under item 22.1.1 of the guidelines the addition is made that the specifications stated there are only used until replaced by a phased temperature-time function.

A relevant modification proposal is incorporated into this report. The relevance of the limiting values used to date is discussed on the basis of the origin of these requirements and of the current verification practice.

The proposals only relate to pressurized water reactors. For boiling water reactors corresponding similar requirements would be established, however in detail other formulations would apply. Since the result of examination of the verification of the core cooling capability is of particular interest, and as the fuel rod characteristics in pressurized and boiling water reactors are significantly different, for boiling water reactors, in which other malfunction transients are to be expected than those for pressurized water reactors with effectively different emergency cooling systems, the considerations are somewhat modified. For example the low system pressure of the Sm fuel rods gives rise to a different deformation characteristic; the fuel element packs in the boiling water reactor cause special interactions due to cooling duct restrictions (distribution of flow and heat transportation mechanisms)

2. THE CURRENTLY APPLICABLE EMERGENCY COOLNG CRITERIA

2.1 Origin and significance of the main requirements

The aim of the requirements for the nuclear legislation approval procedure relating to nuclear power stations is to safeguard the environment against harm that may occur during their construction and operation, as well as in the event of any malfunction. Such harm would mainly be caused by the release of radioactive materials. The approval procedure involves checking that the necessary precautions are taken in accordance with state-of-the-art scientific knowledge and good engineering practice. In this respect it is necessary to verify that operational Breaks are kept to a minimum and that any release due to malfunction is limited as far as possible and that no harm is to be expected to health or property in the environment.

There are several graduated barriers to protect the environment against the release of radioactive materials from nuclear power stations:

1. the fuel matrix

2. the fuel rod cladding
3. the pressure enclosure of the primary coolant
4. the reactor containment

In the event of a malfunction causing coolant loss the 3rd barrier will fail and, in many cases, the 2nd barrier may also break down.

Due to the temporary imbalance between heat generation and dissipation within the core, overheating can occur if the emergency cooling system can not keep the steep rise in core temperatures at a low level.

The greater part of radioactive material can only be released if the fuel is overheated, particularly if it melts then the fuel rod and core geometry would disintegrate. In order to prevent this happening certain limiting values must not be exceeded.

However, even at increased core temperatures well below the melting point of the fuel or the cladding material and also below the limiting values of the safety criteria, damage occurs to the cladding due to coupled thermal-mechanical loadings, as well as due to chemical reactions and changes in material properties. Such damage is undesirable in principle because

- increased release of radioactive material will occur, particularly of fission gases in the containment.
- fuel rod deformation can lead to partial or total blockage of the coolant ducts and thus affect cooling of the core.
- due to shattering, fragments of the cladding or fuel pellets can block the coolant ducts.

The emergency cooling system is the more effective the less damage and deformation of the fuel rod cladding there is, or the less fission products enter the containment atmosphere. The RSK guidelines achieve this primarily by means of limiting values for emissions, temperatures and the zirconium-water reaction. The requirements for the cooling capability of the reactor core, or limiting of the extent of damage to the fuel rod cladding, are on the other hand generally maintained.

Due to the relatively late introduction of the Federal Republic of Germany to the peaceful application of nuclear energy, the nuclear approval procedures applied in the USA are somewhat more advanced than those in Germany. At first the US Atomic Energy Commission (USAEC) was responsible for questions of approval in the USA, followed later, and to date, by the Nuclear Regulatory Commission (USNRC).

In June 1971 the "Interim Acceptance Criteria for Emergency Core Cooling Systems" was published in the USA, and then in January 1972 extensive hearings were held to clarify which changes should be made to the preliminary criteria. The resulting Acceptance Criteria 2 was published in December 1973.

The first version of the German RSK guidelines for pressurized water reactors was published in April 1974 and the latest version (the 3rd) in October 1981 /1/. Both are clearly based on the US-Acceptance Criteria. This applies primarily to the main requirements for the emergency cooling system stated under 22.1.1 (1), which are practically the same in both issues, including the details of the maximum extent of damage in /1/.

In the following these main requirements of the Interim Acceptance Criteria (IAC) /3/, the Final Acceptance Criteria (FAC) /2/ and the RSK guidelines /1/ are contrasted and explained:

1. Maximum cladding temperature

IAC (Interim Acceptance Criteria): The calculated maximum fuel element cladding temperature does not exceed 2300 °F....

FAC (Final Acceptance Criteria): The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.

RSK: The emergency cooling has to ensure that the calculated maximum fuel rod cladding temperature does not exceed 1200 °C.

The basis for this requirement is explained more fully in the Interim Acceptance Criteria /3/, the views of the commission /8/ and the approval authority /4, 7/ relating to the Acceptance Criteria. In accordance with the above the limiting temperature value, together with the following oxidation criteria, serve to prevent shattering of the cladding. This is intended to ensure that the cladding remains sufficiently intact in order to retain the UO_2 pellets within the fuel rod cladding and so to achieve an arrangement that is capable of cooling.

It was generally acknowledged that in the event of malfunction causing coolant loss the cladding can burst, however the view was that the burst cladding would remain in one piece and retain the pellets so long as the cladding is not severely oxidized. The limiting values for temperature and oxidation of the cladding are intended to ensure that, when the cladding is wetted again, sufficient residual ductility will be present in order to maintain the core in a form capable of being cooled in spite of the existing material stresses.

In the approval authority's report /4/ amongst the reasons stated for limiting temperature values, the aims are also stated of eliminating melting of the cladding and of limiting the energy released by means of the zirconium-water reaction. However it is subsequently made clear that these reasons do not form the basis for the 2200 °F limiting value, since a value of 2300 °F would also have sufficed for this purpose. This also applies when taking account of the pre-oxidation and the fuel element splitter. The overriding reasons for lowering of the limiting temperature value from 2300 °F to 2200 °F were experimental indications at Oak Ridge that the cladding ductility at higher temperatures is reduced more quickly than expected from previous observations.

The manufacturers were not in agreement with the limiting temperature value of 2200 °F. Stress calculations and experimental data were submitted to demonstrate that higher values were also permissible. The AEC held to the 2200 °F criterion because, in their opinion, the detailed knowledge in respect of material characteristics in the event of malfunction causing coolant loss was still incomplete, and because evaluation of the zircaloy ductility offered the best guarantee of the cladding remaining intact. Those protesting against this did admit that the 2200 °F criterion was sufficient to prevent any uncontrolled heating up due to the zirconium-water reaction, however it was denied that the temperature and oxidation limiting values would eliminate the possibility of shattering when the cladding was wetted again. However all newer experimental data /5/ refute this view.

It can therefore be determined that the temperature criterion is primarily a requirement to prevent the occurrence of shattering. In spite of this, in practice this became the decisive criterion for the design of the emergency cooling systems and for assessment of the efficacy of the emergency cooling. One principal reason for this may be that the numerical limiting values are certainly stated as decisive, whilst the more significant criterion for the capability of cooling only exists to date in qualitative form.

2. Maximum cladding oxidation

IAC: No details

FAC: The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation....
... If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculation time of rupture....

RSK: (The emergency cooling must guarantee that) ... the calculated depth of oxidation of the cladding shall nowhere exceed a value of 17% of the actual cladding wall-thickness.

According to the corresponding opinion of AEC, the manufacturers and opponents of nuclear energy, the requirement of 17% maximum oxidation layer thickness represents a necessary limiting value, because the extent of shattering of the zircaloy is dependent on the cladding oxidation. In the view of the AEC however, this was an insufficient criterion, because the ductility and stability of the oxidized zirconium are dependent on the ß-phase state. The temperature criterion should take this effect into account.

In the Acceptance Criteria the calculation specifications for verification of the oxidation criterion are more precisely represented than in the RSK guidelines. In particular, express reference is made to the fact that, after bursting of the cladding account has to be taken of two-sided oxidation without vapor limitation. Also calculation of the cladding thickness in the event of ballooning and bursting of the cladding is clarified in more detail. In Appendix K there is also a requirement that the zirconium-water reaction, after bursting, should be assumed to be 1.5 inch or more away from the bursting point, on the inside of the cladding. In practice, in the German approval procedure, the oxidation criterion is applied in the same way as in the USA. When applying the Baker-Just relationship /6/ this procedure is conservative if the time of bursting is not calculated too late.

3. Maximum hydrogen production

IAC: The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of cladding in the reactor.

FAC: The calculated total amount of hydrogen generated from the chemical reaction of cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated, if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding, surrounding the plenum volume, were to react.

RSK: (The emergency cooling must ensure that ...) ... for the zirconium-water reaction no more than 1% of the total zirconium contained within the cladding will react.

In this respect the Final Acceptance Criteria include a more specific wording than that of the RSK guidelines and the Interim Acceptance Criteria. Whilst initially the entire mass of the cladding in the reactor formed the basis for the calculations, subsequently the cladding parts were reduced by the fission gas plena, so that less hydrogen would suffice in order to exceed the 1% limiting value. The principal reason for the 1% limiting value is to restrict the hydrogen production from the zirconium-water reaction together with hydrogen formation due to radiolysis processes to less than the quantity necessary to form an ignitable mixture within the containment area /7/. Whilst the aforementioned 17% criterion is to be applied to the individual rod, for the 1% limiting value the complete core has to be considered. This means that the power distribution within the core will affect the results of analysis. However in /7/ it was already determined that, even in the event of a very long oxidation time, exceeding of the 1% limiting value would not be calculated so long as the maximum cladding temperature remained below 2300 °F. (E.g. at a temperature transient of the hot spot with temperatures of 2300 °F for a duration of approx. 200 s the total produced hydrogen quantity would remain below 1% and only in 10% of the core more than 1% of the cladding material would react.) The quantities of hydrogen arising from the zirconium-water reaction are relatively low at the malfunction transients under consideration. The question as to whether local increased, hazardous hydrogen concentrations could form, and to what extent the hydrogen released could affect the thermo-hydraulics of the primary circuit does however need to be verified separately.

4. Maximum extent of damage

FAC: No details

RSK: (The emergency core cooling must ensure that ...) ... due to cladding damage as mentioned in Chapter 2.2 (4) under No. 2 ("It is to be assumed that 10% of all fuel rods would fail ...) the stated release of fission products would not be exceeded.

Since in the assumptions mentioned in Chapter 2.2 (4)
No. 2 relating to the release of fission products is specified in relation to the inventory of a single fuel rod, this criterion represents a definitive restriction of the cladding damage to a maximum of 10%.

The determination of this maximum value has an entirely different qualitative significance to the determination of the maximum values in the first 3 criteria. Whilst for the temperature criterion in Chapter 22.1.3 of the guidelines the assumptions are formulated for the calculation, there are no corresponding details for the extent of damage. The assumptions for the temperatures tend to be conservative (unfavorable values), whilst the extent of damage can only represent the most probable value (best estimate value), because at the time of preparation of this criterion much higher limit estimates existed for the extent of damage.

A newer examination of the KKP2 plant confirms this. Whilst the probabilistic (most probable) value for the extent of damage is 1.9%, using more conservative limitation estimates this figure is around 22%. It should be mentioned in this respect that the most probable value for the existing thermo-hydraulics can be determined very accurately, whilst the upper limiting value is subject to larger fluctuations depending on the choice of the influencing parameters (degree of conservatism). The extent of damage increases exponentially with the maximum core power factor (peak RELEB), so that determination of the maximum extent of damage in fact becomes equivalent to a limitation of the maximum power factor.

5. Capability of core cooling

IAC: The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling and before the cladding is so embrittled as to fail during or after quenching.

FAC: Calculated changes in core geometry shall be such that the core remains amenable to cooling.

RSK: (The emergency core cooling must ensure that...) ... no change occurs in the geometry of the reactor core, which would hinder sufficient cooling of the reactor core.

This criterion is of the utmost importance both in principle and in practice. An incapability of cooling would have to lead to overheating and melting of the fuel rods and intolerable emissions as a result.

At the time of publishing of the Interim und Final Acceptance Criteria there was clearly a risk of incapability of cooling due to shattering or melting of the cladding. When establishing the criterion for the capability of cooling /8/ the possibility of blockage of the cooling ducts by blistering of the cladding was raised. Consequently the manufacturers considered this criterion to be unnecessary because it was already covered by the other criteria. The commission did however retain the criterion of capability of cooling due to its fundamental significance.

("Considering all of the required features of the evaluation models, we are inclined to agree that, for any situation we have been able to anticipate, this criterion should be superfluous. However, in view of the fundamental and historical importance of maintaining core cooling capability, we retain this criterion as a basic objective, in a more general form than it appeared in the Interim Acceptance Criteria. It is not controversial, although the extent of flow blockage resulting from cladding swelling is a matter of controversy" /8/).

On the basis of the findings in recent years, the risk of incapability of cooling the core due to blockage of the cooling ducts as a result of expansion of the cladding has gained in importance, whilst on the basis of extensive experimentation and theoretical studies into the shattering of the cladding, which can now be confidently estimated, the significance of the criterion for the prevention of this risk is meanwhile somewhat reduced.

In respect of the consideration of the incapability of cooling due to expansion of the cladding and blockage of the cooling ducts, the circumstances are much more complicated.

On the one hand the bursting expansion observed in experimentation is at a maximum at cladding temperatures of around 800 °C and a minimum at around 900 °C, but on the other hand, along with the maximum temperature value, the temperature-pressure-time relationship of the fuel rod is responsible for the expansion characteristics of the cladding during a malfunction causing coolant loss, and furthermore the extent of cooling duct blockage is not determined by a fictional (conservatively calculated) characteristic of a hot spot, but rather by the real characteristics and the spatial arrangement of many fuel rods, by which the parameters determining the expansion process (temperature, pressure, service life, material characteristics) can combine in an unfavorable manner. The limiting value of 1200 °C and the verification by means of conservative analysis of hot spots are not sufficient to ensure limitation of the extent of damage in the core; for this purpose definition of the capability of cooling criterion, as well as realistic (best estimate) calculations for the entire core area are necessary for verification.

By limitation of the number of cladding bursts to a figure of 10%, restriction of the number of cladding tubes exhibiting very high levels of expansion follows naturally, because by restricting the power, and therefore the rod temperature, the number of rods that either burst of expand excessively in the corresponding temperature range is also restricted.

A study of the KKP2 plant, /9/, indicates that for the maximum power factor and an expected amount of 1.9% of burst rods for a total of only 38 rods exhibiting expansion, over 35% have to be calculated for. If the spatial distribution of the highly expanded rods is ignored and a study by the KfK is followed, for an arrangement of 12 X 12 fuel rods, even with cooling ducts completely blocked, cooling capability is still maintained (prevention of pellet melting), then a core geometry capable of being cooled is maintained /lo, 11/. The cooling capability problems are also greatly relieved by limitation of the number of cladding bursts.

2.2 Verification in practice

Verification of compliance with the criteria or the efficacy of emergency cooling is carried out by the manufacturers and inspectors using extensive calculation programs, which analyze the various phases of a malfunction. However the programs can only represent the course of a malfunction with a certain range of distribution.

Due to a lack of sufficient, verified "Best-Estimate" programs, in the past the limitation estimate was performed by so-called conservative methods. This approach was supported in that, for the approval procedure to date the 1200 °C criterion was maintained for the hot spot determining the design of the emergency cooling system and that for many assumptions and stages of analysis, it can be relatively easily determined what would lead to an increase in the maximum cladding temperature and what is therefore conservative (e.g. high decay heat leads to higher temperatures). For other assumptions that cannot necessarily be stated (e.g. is a higher or lower pressure loss coefficient conservative?). In such a case parameter studies are generally performed, in order to identify the unfavorable malfunction process.

The conservative, deterministic procedure does however carry the risk that assumptions and procedures which have at some time proven to be conservative in nature may no longer be questioned or that for new problem areas or programs these are simply expanded. For example, it is not in every case that the assumptions that lead to the highest hot spot temperature also produce the greatest extent of damage.

If, during the course of analysis, several conservative assumptions are made, then due to aggregation of the conservative factors the results of analysis may vary greatly from the reality. For an individual rod, for example, this would mean that with every new, not accurately quantifiable effect, the conservatively calculated hot spot temperature would be increased and so the difference between the conservative and the "Best-Estimate" temperature would become ever greater. In the worst case queries may be raised by the results of conservative analysis that are not relevant in practice, whilst risk-related queries may become obscured.

In the case of analyses relating to the entire core area, conservative assumptions for each individual rod may give rise to unrealistic operating conditions and a misleading overall view. For example, assumptions relating to the maximum possible power factor for each rod may cause the overall power of the reactor to be greatly exaggerated in relation to the maximum operational capacity. The fuel rod characteristics calculated in this manner would give a completely false impression of the overall core situation and the extent of damage would probably be viewed as intolerable. A strictly deterministic, conservative procedure is therefore unsuitable, in particular in the case of analysis of the extent of damage.

These examples demonstrate that for individual rod analysis, at least supplementary "Best-Estimate" calculations should be included, whilst for analyses relating to the entire core area another rather probabilistic/statistic procedure should be opted for.

Due to the lack of clear standards, manufacturers and inspectors have tended to develop varying procedures. The manufacturer essentially attempts to verify compliance with the criteria from /1/. Because of the uncertainties in the calculation programs and input data described, in the past this has given rise to discussions between the manufacturers and the inspectors regarding the conservatism of the calculation assumptions and data. Alongside compliance with the RSK guidelines, GRS has therefore been involved for some time with description of the extent of damage in objective terms, as well as with identification of the distribution range and the uncertainties related to the results of analysis. By applying the probabilistic methods developed under /12, 13/, we are attempting to put the evaluation of emergency cooling systems onto a more solid basis. Using these methods it is an advantage that the correctness of the methods of analysis and assumptions can be directly checked against the reality (experiments, results from nuclear power plants) and that a result comprising various "Best-Estimate" calculations are significantly more similar to each other.

2.3 Relevance of the limiting values from the RSK guidelines

In the following, the relevance of the proposed limiting values as specified under 2.1, and in particular that of the 1200 °C criterion, is discussed. The predominant significance of the 1200 °C criterion has already been indicated. For all malfunctions causing coolant loss that were studied in the German approval procedure it has been apparent that the 17% oxidation depth and the 1% zirconium oxidation criterion are always fulfilled if the 1200 °C limiting value is met. Corresponding studies indicate that the 17% criterion can only be checked for oxidation times in excess of 300 s. Such times, at cladding temperatures in excess of approx. 900 °C, are not to be expected for large breaks under the approval-specific failure assumptions. The checks can therefore be restricted to smaller and medium sized breaks and can also be limited in those cases to malfunctions for which the temperatures in excess of 900 °C occur for longer than 300 s.

The criterion for limitation of the release of hydrogen from the zirconium-water reaction is, strictly speaking, already covered by the RSK guideline 24 /1/ and could therefore be deleted without replacing it. As already demonstrated in the example described under 2.1, it is also not to be expected that the 1% limiting value would be achieved at the temperature ranges typical for malfunction causing coolant loss and at maximum temperatures below 1200 °C. Furthermore, the greater part of hydrogen release is due to radiolysis and not to the zirconium-water reaction, whereby the radiolysis can only lead to higher hydrogen concentrations in the longer term. For the stated reasons this criterion can be deleted.

Of the criteria discussed up to now, those for the maximum temperature of the fuel rod cladding at the hot spot are clearly predominant. Verification of compliance with this criterion, which is reached by means of conservative, deterministic methods, does however allow only minor conclusions to be drawn with regard to the expected behavior of the heating rod and practically none regarding the overall condition of the core after a malfunction causing coolant loss. Significantly more information can be obtained from probability-related temperature analyses: In this respect all uncertainties and statistical variances of the effective parameters, physical effects and analytical methods are taken into account. The results obtained comprise probabilities with which certain temperature limiting values are exceeded, the expected values and the distribution density for individual results. When grouping these distribution densities with the spatial distribution of the power factors statements are obtained relating to the overall characteristics of the core, e.g. the probability that the core temperature at any rod will exceed 1200 °C. Eventually levels of confidence can be specified for all stated results.

In /12/ a probabilistic temperature analysis was first performed for a German Pressurized Water Reactor in the 1200 MW class, whereby a 2F rupture was assumed between the pump and the Reactor Pressure Vessel, along with the associated individual malfunctions and the repair case. For this case a probability of approx. 10^{-5} was determined that the hot spot temperature would exceed the 1200 °C limitation. Furthermore, by relating this to a spatial, conservatively selected power distribution it was verified that it was not to be expected that any rod within the entire core would achieve the limiting temperature of
1200 °C. Values of this order apply generally for newer systems of German design.

Probabilistic temperature analyses do however clarify that there is no 100% guarantee that the 1200 °C criterion would be met. In this way it can be calculated that, whilst meeting the specifications of the RSK guidelines for conservative hot spot analysis, that depending on the type of assumptions for which the guidelines give no details, maximum temperature values ranging between approx. 1000 °C and 1300 °C would be achieved. Therefore it would be possible for any applicant to verify the 1200 °C criterion in whilst meeting the RSK guidelines. For stricter standards for the parameters not specified in the criteria (e.g. the heat transfer coefficient within the gap) the same reactor temperatures can be calculated, which would differ by more than one hundred degrees. The relevance of such results would only become clear on application of a probabilistic method.

Deviations between the results of "conservative" temperature analyses by the inspectors and those of the manufacturers can be explained in that individual assumptions are designated as "sufficiently conservative" if they cover the actual characteristics perhaps not definitely, but with high probability and yield an overall result which is conservative in respect of the expected value. However, such an approach can not be regarded as conservative in the strictest sense as it includes probabilistic elements, albeit in a qualitative form. Furthermore, at least one "Best Estimate" calculation would be necessary to verify that the final result was conservative.

For "Best Estimate" calculations the question remains of the uncertainty of the results and the probability of exceeding critical values. This information can only be gained by probabilistic methods, which also employ Best Estimate calculations, however the effective parameters for the analyses are correspondingly uncertain and the distribution ranges can vary. Probabilistic statements relating to the results of analysis are compatible with risk analyses. A further advantage of a probabilistic procedure is that the reduction in distribution ranges and uncertainties have a direct effect on the results of probability-related analyses due to the progress in analytical and experimental know-how. For these reasons a probabilistic approach is clearly preferable.

Regarding the relevance of the 1200 °C criterion the following can be stated in summary:

1. The temperature criterion is intended to form part of an oxidation criterion. Its significance is overemphasized.

2. Compliance with this limiting value only allows conclusions to be drawn regarding the behavior of one rod, however not regarding the overall core.

3 The conservatively calculated behavior of such a rod does not correspond to the actual behavior.

4 The specifications for conservative hot spot analysis included in the RSK guidelines are not so detailed as to exclude the possibility of deviations of several hundred °C between the calculations for different spots in the case of extreme parameter combinations.

5. Therefore the conservative, deterministic verification of compliance with the temperature criterion can only be subjectively performed and is of little consequence for design of the emergency cooling system.

6. Verifications based on conservative, deterministic methods can not quantify the degree of safety as can probabilistic methods.

It is proposed that the temperature criterion be
re-formulated such that its objective is clarified as the oxidation criterion and that its subordinate significance is clarified in respect of the criterion for cooling capability.

3. CALCULATION OF THE EXTENT OF DAMAGE

3.1 The expansion characteristics of fuel rods subject to internal pressure loading

The mechanical characteristics of zircaloy cladding subject to internal pressure loading is described by empirically derived equations for rate of expansion, bursting stress and oxidation /15, 16, 17/. The determining parameters for the plastic deformation and for bursting of the cladding are the temperature of the cladding, the internal positive pressure and the residence time. Systematic studies of these parameters in /14/ demonstrated their effects and elaborated the determining interdependencies.

Bursting of the cladding is represented in diagrams of the bursting expansion over temperature for the effective parameters of pressure, oxidation and temperature gradient. The effect of temperature gradients is elaborated. The time-related expansion process was studied and a relative time axis was introduced. This simplifies the graphic representation.

Oxidation of the cladding leads to embrittlement and reduction of the bursting stress. The relationships taken from the literature, which apply for a constant temperature, were augmented for temperature transients.

The bursting expansion is studied for constant specified temperature gradients. It appears that the decisive factor is not the temperature gradient, but - alongside the internal positive pressure - the actual temperature. This demonstrates that bursting of the cladding is largely independent of the type of temperature control used.
Studies conducted at constant, specified cladding temperature were able to identify the holding time at a specified internal pressure. Thus the temperature, pressure and holding time form the residual determining factors for bursting expansion. A series of clear graphic diagrams were developed which represent bursting of the cladding and possible holding time of the cladding when subjected to internal pressure. In this form they can serve as the basis for subsequent criteria proposals, with which the extent of damage to pressurized water reactors with coolant loss due to larger and smaller breaks should be limited and the core cooling capability maintained. When comparing the bursting data to a previously developed bursting criterion /21/, a satisfactory consistency was ascertained.

The studies were completed by a model for the assessment of azimuthal temperature gradients. These are preferably generated with rapid transients, as they would occur in the case of larger breaks, and they lead to a reduction of the average bursting expansion compared with the local expansion at the hot spot where the rod bursts /19, 20/. The reduction factor is specified in dependence of the azimuthal temperature difference.

3.2 The probabilistic concept

The probabilistic method was already mentioned in the previous Chapters and is documented in /18/.
The most significant parameters have proven to be

(1) The fuel rod internal pressure
(2) The pellet diameter
(3) The determining parameters affecting the rate of expansion (stress exponent and activation energy)
(4) The after-heat

(5) The fuel rod power
(6) The condition of the coolant (pressure, temperature, vapor content)

(7) The determining factors for heat transfer from the cladding to the coolant (mass flow in the pressure discharge phase, duration of refill, heat transfer coefficient in the flooding phase)

The first 3 variables only involve the fuel rod, knowledge of its expected values, its dependencies and its range of spread or if its distribution function is in order and is verified by numerous calculations. The after-heat is a physical quantity of which the time-related dependence and inaccuracy is derived from the applicable regulations or standards. The power of the fuel rods is dependent on the operating duration, the operating mode and the set values of the reactor's power limiting system (peak-RELEB). In this respect either operational power distribution-related values can be inputted or, for approval calculations, fictional distributions that are limited in size, which also include transient, unfavorable operating modes and provisions for subsequent reactions.

The condition of the coolant and the heat transfer coefficient to the coolant is the result of prior thermo-hydraulic calculations, whereby the heat transfer is of far greater influence than the condition of the coolant. This heat transfer coefficient can only be determined with very great uncertainty; the effect on the maximum rod temperature achieved and the extent of damage are correspondingly large. In experiments, and even in well-defined standard trials, it is barely possible to separate the uncertainties in the results into their fuel rod-specific and thermo-hydraulic components. However, from experiments in which the cladding temperature was measured, it can be concluded that the fuel rod behavior at a specified cladding temperature can be determined with satisfactory accuracy, so that any inaccurate preliminary calculation made during other experiments is almost exclusively due to unsatisfactory preliminary calculation of the thermo-hydraulics.

For an individual quantity that is subject to statistical deviations, a differentiation has to be made between the upper and lower limiting value (maximum value and minimum value), which form the distribution range and the statistical mean value (expected value). If a multi-variable is involved as the result of a calculation or a similar selection procedure, then the most probable value is also designated as the Best Estimate value.

If a calculation is performed using the expected values of the parameters, then the result is designated as the Best Estimate value. If a probabilistic analysis is performed then the result is an expected value which does not have to agree with the Best Estimate value. It is precisely the dominant influence of the temperature on the expansion characteristic that causes the relatively few rods that achieve a higher temperature range according to the probabilistic analysis to determine the extent of fuel rod failure. The accuracy in temperature determination by thermo-hydraulic calculation is in the order of 100 K. However, for the fuel rod behavior within this temperature range the entire range exists, between imperceptible plastic expansion and bursting, involving more or less large bursting expansions.

In this case a simplified procedure is conceivable:
The thermo-hydraulic calculation is performed using a contingency added to the calculated temperatures. A Best Estimate calculation of the fuel rod behavior is then performed using this temperature. The contingency then has to be assessed such that the Best Estimate calculation produces just the expected value of a probabilistic analysis. This procedure is very simple but it is not without its problems:

- The contingency can be determined for a defined plant, but it is not guaranteed that this can be applied for other plants or for other assumed malfunctions.
- It is possible that the contingency is dependent on the phase of malfunction and on the temperature level itself.

A complete probabilistic analysis, even just using the main parameters, is preferable in any case. The difficulty in performing such a calculation would hardly exceed that of a thermo-hydraulic analysis.

The thermo-hydraulic boundary conditions must exist for the analysis; e.g. the heat transfer coefficient characteristics over time, as well as some details of the distribution range.

3.3 The effect of break size

In relation to the fuel rod behavior there are in principle no differences between large and small breaks; any differences are caused by the varying types of thermo-hydraulic processes:

- The temperature variations are slower, thus promoting a uniform temperature distribution across the extent of the cladding (smaller azimuthal temperature differences). If plastic expansion of the cladding occurs then concentric expansion will be predominant, which leads to greater expansion than the eccentric expansion in the case of larger breaks.

- Local variations in the cooling characteristics are smaller than for larger breaks. This can cause large areas within the core to expand uniformly, thus causing blockages.

In the case of smaller breaks there is therefore more risk of the cooling ducts blocking up, insofar as the corresponding temperatures are achieved, than for larger brakes.

4. PROPOSALS FOR NEW CRITERIA

In general the maintenance of core geometry capable of being cooled and re-wetted applies as the overriding requirement for all break sizes.

4.1 Large breaks

For large breaks (rupture cross-section greater than 0.5 F) with corresponding rapid pressure reduction in the primary circuit the RSK guidelines propose

- Less than 10% of all fuel rods can burst due to the effect of internal pressure

It must be clarified in this respect that this concerns Best Estimate values (expected values) which, in some cases of an unfavorable combination of input data, can certainly be exceeded. Determination of the extent of damage should generally be performed using probabilistic methods; for simplicity a Best Estimate calculation could also be performed, whereby the uncertainties in the thermo-hydraulic analyses have to be taken into account by means of a contingency.

As mentioned in the previous Chapters, the oxidation criteria are insignificant in the case of larger brakes. The temperature criterion would not have proved sufficiently meaningful in respect of the extent of damage. Since its function as oxidation criterion is insignificant, the temperature criterion can be omitted without substitution.

On the assumption of a larger brake and its characteristic in the event of malfunction, for compliance with the 10% criterion blockages over large areas of the cooling ducts causing loss of flow rate are not to be expected, so the core geometry remains capable of being cooled.

4.2 Small breaks

In the case of small breaks it is of paramount importance that blockages over large areas that can cause loss of flow rate are eliminated, so that cooling capability of the core is guaranteed. Since due to the long-term effect of a low temperature significant plastic deformation can occur within a short time-span, higher levels of expansion (e.g. 10%) should be eliminated from the outset by specifying a holding time-temperature curve. For calculations involving individual rods no level of expansion in excess of 10% should be permitted (expected value).

Such holding times are identified in /14/. The following Figure from /14/ plots the holding time against the temperature for various internal pressures. The holding times are calculated such that an expansion of just 10% is achieved. Thus for example, for an internal pressure of 60 bar the holding time is 2 s at 900 °C, 1.5 min at 800 °C, etc.

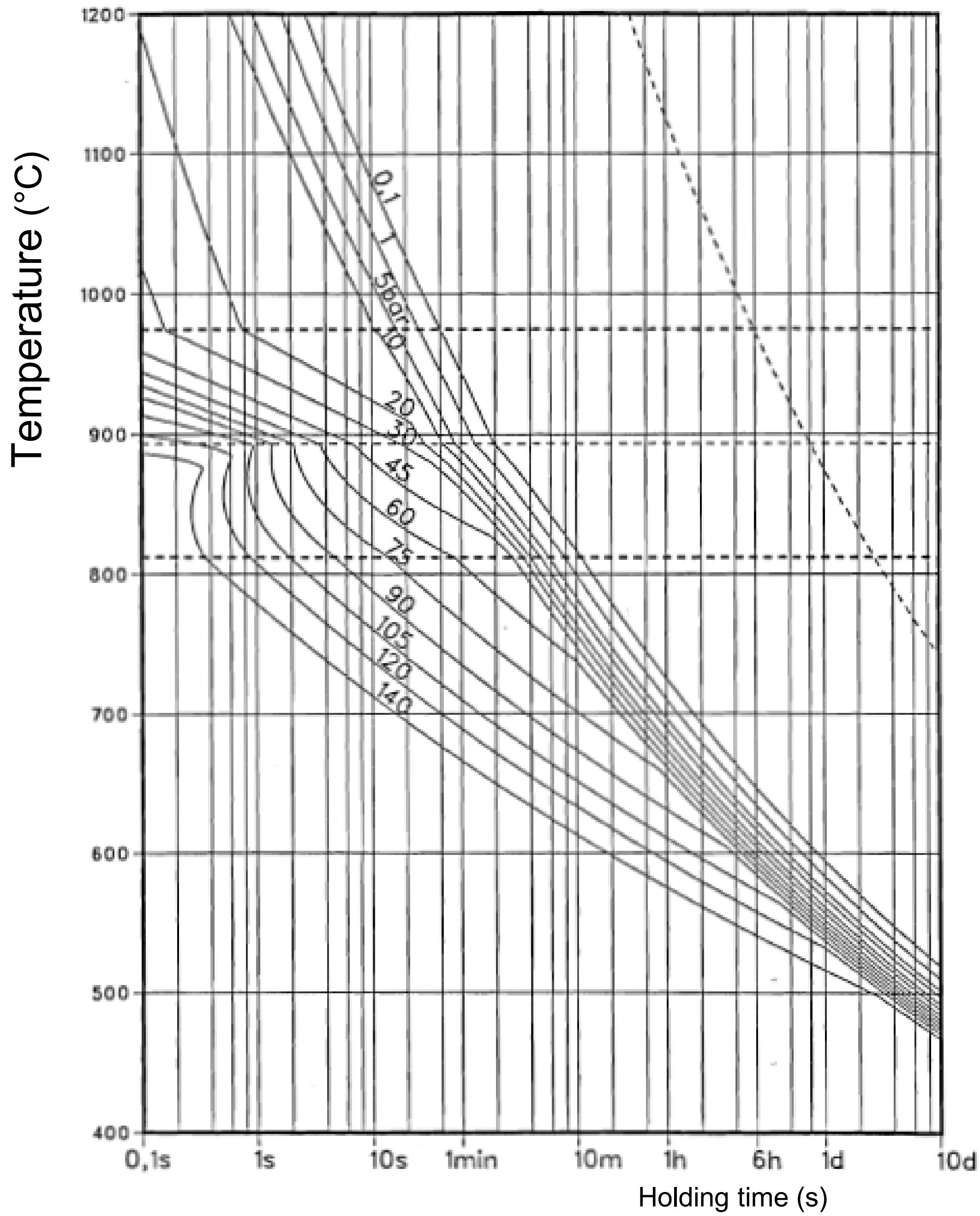
In the Figure the time is indicated (broken line) that is necessary for an oxide layer of 17% to accumulate. It then appears that a 10% plastic expansion will always be achieved before the limiting value for oxidation from /1/. Therefore the oxidation criterion can also be omitted in the case of smaller breaks.

5. REQUIREMENTS FOR COMPLIANCE WITH THE NEW CRITERIA

5.1 Experimental validation

Safety criteria should fulfill two minimum requirements:

1. They must quantitatively and clearly identify the limiting values to be complied with.
2. They must describe the methods and assumptions with which the verifications of compliance with the criteria are conducted.



Bursting criterion for constant temperature
Maximum expansion 10%

The limiting values for efficacy of the emergency core cooling system were specified in Chapter 4. The related methods and assumptions are discussed in the following.

For the assumptions relating to emergency cooling analyses specified in the RSK guidelines /1/ the requirement is included for "experimentally verified analytical confirmation". As a detailed definition of such verification, in accordance with the Acceptance Criteria it could be required that:

1. FOR THE ANALYSIS OF THE EFFICACY OF THE EMERGENCY COOLING, ONLY ANALYTICAL METHODS, CORRELATIONS, ASSUMPTIONS AND PROGRAMS SHALL BE APPLIED FOR WHICH THE VALIDITY OR APPLICABILITY HAS BEEN VERIFIED BY COMPARISON WITH EXPERIMENTS IN THE ENTIRE RANGE OF APPLICATIONS.

2. THE UNCERTAINTIES INVOLVED IN THE ANALYTICAL TOOLS AND METHODS SHALL BE QUANTIFIED.

Whilst the first requirement speaks for itself, the second is derived from the probabilistic criteria concept: only by ascertaining these uncertainties can quantitative limits be set for the validity of a statement.

5.2 Assumptions relating to probabilistic verification

In contrast to conservative, deterministic verification, for a Best Estimate criteria concept the assumptions have to be as close to the reality as possible, however at the same time they have to be performed using details relating to their variances and uncertainties. However, in order not to hinder the continuous improvement of the assumptions and analytical methods, the stipulation of licensed correlations, assumptions and approval models analogous to the current American approval practice must be avoided. The direct advantage of reduction of the variances and levels of uncertainty in probabilistic verifications can thus lead to a continuous improvement in the state of knowledge.

It is therefore recommended that in the RSK guidelines the specified assumptions and correlations are no longer listed, because the requirement for experimental verification in connection with the Best Estimate criteria concept only permits experimentally verified calculation assumptions in accordance with the current state of scientific and technical knowledge in any case. The assumptions specified under 22.1.3 of the RSK guidelines only cover a portion of the more important parameters for emergency cooling system analyses. Details for verification of the expansion and blockage characteristics are lacking.

5.3 Calculation and modeling assumptions in the RSK guidelines

Carrying over of the recommended calculation and modeling assumptions under 22.1.3 of the RSK guidelines into the new Best Estimate criteria concept is discussed in the following:

1. Break Flow:

   There are plenty of brake flow models currently in existence, which can be roughly subdivided into homogenous and two-phase models as well as models for the sub-cooled and saturated flow phase.

For the fuel rod and temperature characteristics, none of the models mentioned can be designated as being fundamentally conservative; although it is known that for mixture break flow in accordance with the Moody model higher break flows are calculated than when using the homogenous equilibrium model, and that of these two the latter generally represents experimental results better. The models in the sophisticated codes are not yet verified to such an extent that one can be identified as being preferred, although these do have a greater potential for accurate description of the reality because of their greater degree of detail.

2. Burnout delay

Of the many existing DNB correlations, depending on the test conditions during experimentation, sometimes one and sometimes another proves to be the most accurate. It is recommended to estimate the level of uncertainty of a correlation by means of comparison with as many suitable experiments as possible. If the DNB time is a parameter for a program then statistic evaluation of suitable experiments is also conceivable.

3. Heat transfer during break flow and before draining down
----
the reactor

Also in this case one should not fundamentally commit oneself to a correlation, but rather opt for the most suitable according to the malfunction parameters, possibly also as alternatives to each other. In the case of low flow rates "pool film boiling" correlations are to be recommended, whereby a differentiation should additionally be made according to the vapor content. Also re-wetting should not be discounted in respect of a probabilistic verification concept.

4. Heat transfer before start of flooding

In this case either representative experiments or verified models should be used as a basis. To date there is a lack of specifications for calculation of the refill phase duration, which has a decisive effect on the temperature rise. Effects such as vapor-water counter-flow within the annular space and at the upper core support grid; filling up and leveling of the water over the upper core support grid affects the transport of heat and the duration of the refill phase.

5. Break flow from the rupture point

The proportion of the emergency cooling water conveyed directly to the rupture point should be identified implicitly and time-related within the programs.

6. Heat transfer during refill

On this subject the guidelines include notes on heat transfer during flooding. The heat transport mechanisms above and below the wetting surface, along with their oscillatory progression, should be modeled in detail, and likewise all effects mentioned under item 4. Insofar as they still occur during the flooding phase. A separate description of the vapor and droplet phase is necessary in order to facilitate calculation of vapor superheating and water entrainment and to accurately determine the interactions

between the phases. Also water entrainment within the steam generator, as well as the thermo-hydraulic processes taking place there, should be taken into account.

7. Vapor blockage

This requirement is implicitly included under item 6.

8. Pump characteristics

These requirements can be directly adopted with the exception of replacing the phrase "by means of appropriate assumptions" with "by means of determining the level of uncertainty".

9. Residual water content

From all integral experiments (e.g. SEMISCALE, LOFT) it is clearly apparent that residual water content is to be expected on completion of pressure discharge. In the case of "Best Estimate" analyses this can be taken into account, however the level of uncertainty of the current method of analysis must also be incorporated into the calculations. Furthermore, the residual water can only be ascertained for programs that work with a water level model.

10. Flow-rate reduction

The time-dependent flow-rate characteristics within the hot and normal ducts are to be modeled separately. Furthermore, the repercussions of cooling duct blockage should be implicitly taken into account.

11. Back pressure in the reactor containment

In this respect the distribution of possible back pressures is to be taken as a basis.

12. Power distribution within the core

This item concerns the status of the core before the occurrence of a malfunction. Since, for statistical determination of the initial status years of data evaluation would be necessary, it is recommended that initially the least favorable power distribution be taken as a basis. In general this would be that which, for overall hot spot factors in excess of approx. 1.6, comprises more rods than are operationally possible. Since the probability of a certain power distribution would be approximately the same for all cases of malfunction causing coolant loss, the relationship of the risk factors of individual malfunction cases to each other would remain unaffected by the power distribution selected.

13. Pressure differentials within the RDB

- can be adopted –

14. Core decay power

    The uncertainty level involved with this has clearly reduced in recent years. The 36 limit from the experimental data is now below 1.1 times the ANS standard curve.

Discussion of the individual assumptions indicates that specification of the Best Estimate condition, or even of the level of uncertainty including its distribution function, is far more difficult than specifying a conservative limiting value. Furthermore, the list of parameters used is also dependent on the program concept. For example, for programs including calculation of the water level and separate program segments for pressure discharge and flooding, the matter of the residual water comes into question. On the other hand, for programs that use a "flow chart" this quantity is not calculated and is also not used.

Discussion of the individual assumptions also indicates that it is not useful to include a detailed list of parameters in the guidelines or criteria formulations.

5.4 Calculation and modeling assumptions relating to the fuel rod behavior

There are to date no detailed specifications for analysis of the fuel rod behavior. On the basis of the description of the current methods of analysis it should be indicated which procedure is appropriate for the current state of scientific and technical knowledge. For a probabilistic concept this can also be directly recommended as verification within the approval procedure.

Although for verification of the core cooling capability the integral calculation of the behavior of several fuel rods, along with their interaction with each other and with the coolant, would be desirable, it should be assumed that today's programs are models of individual rods, whereby coupling to the cooling duct hydraulics is done partially by means of inputs and partially implicitly. The fuel rod under consideration is subdivided both axially and radially into nodes, whereby special models exist for the areas of fuel, gap and cladding. These were initially concentric, however it soon appeared that eccentric expansion and heat transport models /18, 19, 20/ were necessary, because only using these reactor-representative experiments can a satisfactory analysis be made.

There are meanwhile plenty of models for the deformation of the cladding. Since all models in the core are based on similar approaches, and on the other hand verification of individual models is affected by the extreme temperature sensitivity and limited calculability or measuring accuracy of the temperature limits, there is no preference for any specific model.

The description of the heat transport and the pressure characteristics within the gap between the fuel and its cladding influences the release of the fuel, heating up of the cladding and the expansion process to a great extent. Both phenomena have to be analyzed in sufficient detail.

For blockages and the release of fission gases, not only the expansion model is of decisive importance, but also the bursting criterion. Most institutions use deterministic bursting criteria /17/; GRS also possesses a probabilistic criterion /21/ based on reactor-representative trials. Special models for the behavior of the rods in case of contact with each other have not to date been implemented in any known program, although bursting and expansion of the cladding can certainly be influenced by such. The temperature and pressure across the cladding have proved to be the most significant parameters for the bursting process.

The description of cladding oxidation is mostly in accordance with the Baker-Just relationship /6/, although this generally produces too high a rate of oxidation. For "Best Estimate" calculations newer relationships would be more suitable. It should be mentioned that for fuel rod behavior analyses an extensive database is necessary, which for example would include the material characteristics of the fuel and the cladding, the fission gases and inert gases and also water vapor tables.

For emphasis, the following requirements for the calculation and modeling assumptions relating to fuel rod behavior can be mentioned:

1. Transient hydraulic analysis

   For each axial section the hydraulic boundary conditions are to be specified or calculated; repercussions of the rod behavior on the cooling duct hydraulics, in particular any change in the flow-rate, flow velocity, vapor content and pressure losses, as well as possible superheating of the vapor phase are all to be taken into account.

2. Material characteristics

   The material data for fuel, zircaloy, zirconium oxide, inert and fission gases as well as water and water vapor on which the calculation is based are to be documented along with the statistical distribution of their occurrence.

3. Initial condition

   The influence of burnout, contact between the fuel and the cladding, cracks in the fuel pellets, redensification and relocation as well as stationary asymmetries already existing are to be taken into account. The size and effect of distribution ranges are to be quantified.

4. Nodalization

The fuel rod is to be subdivided both axially and radially so finely that
- the fission gas plena are separately covered
- in the most highly loaded area at least 2 axial segments occur between 2 spacers.

5. Heat transport equations

A transient, eccentric heat transport model is to be taken as a basis. The temperature dependency of material values is to be taken into account.

6. Heat transfer coefficient within the gap

The heat transfer coefficient within the gap is to be determined dependent on the time-temperature direction, and on the expansion.

7. Transient fission gas pressure

The transient temperature and expansion-dependent fission gas pressure is to be calculated.

8. Fuel deformation

Changes to the fuel due to axial and radial expansion, cracking, shifting, redensification and relocation are to be taken into account.

9. Cladding deformation

An eccentric deformation model is to be taken as the basis. The uncertainty and distribution levels of the parameters for the applied expansion laws, as well as their effect on the calculated expansion shall be quantified. The axial elongation and radial thinning of the cladding are to be taken into account. Verification of the expansion model shall be performed.

10. Bursting of the cladding

A bursting criterion based on testing representative of the reactor shall be taken as the basis and shall as a minimum take account of the effects of temperature, pressure and rate of heating up. Distribution ranges shall be quantified within the criterion itself or by means of details for the parameters of the laws of bursting.

11. Cladding oxidation

Calculation of the zirconium-water reaction is to be based on correlation that is verified by experimentation and is valid throughout the range of application.

12. Cooling duct blockages

The models on which the cooling duct blockages are based shall be defined and justified in detail.

13. Analyses relating to the core overall

Analyses relating to the entire core area are to be based on a tested power and burnout distribution. The effect of statistical distribution ranges on the behavior of rods of the same specification is to be taken into account. Any thermo-hydraulic repercussions are to be discussed.

The calculation and modeling assumptions listed here correspond to the current state of scientific and technical knowledge and represent a maximum requirement in the analytical differentiation, which can only be handled by complex program systems /22, 23/. However simpler programs /24/ can also achieve weighable results. Therefore these calculation and modeling assumptions should also not be stipulated in the guidelines. The general requirements of Chapter 5.1 are entirely sufficient.

6. SUMMARY

The significance of the emergency cooling criteria stipulated in the RSK guidelines were studied and proposals were made for new, improved criteria. The emphasis is on the maintenance of a geometry capable of being cooled. The temperature and oxidation criteria are no longer of significance. They can be omitted without replacement.

For large breaks the upper limit of damage should be stipulated as 10% - to be understood as an expected value. For smaller breaks a temperature-time function is to be specified.

The many calculation and modeling assumptions listed in the guidelines should be replaced by the overall requirements in accordance with experimental verification of the calculation program and according to quantification of the uncertainties. Determination of the extent of damage is based on statistical methods.

When using the newly formulated criteria proposed, a more realistic and therefore more meaningful assessment of the emergency cooling systems is facilitated. The safety margins are therefore quantifiable in relation to an effective emergency cooling system.

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