in-active Safety Standard (Safety standard no longer included in the reaffirmation process acc. sec. 5.2 of the procedural statutes)

SAFETY STANDARDS

of the

Nuclear Safety Standards Commission (KTA)

KTA 3102.5

Reactor Core Design for High-Temperature Gas-Cooled Reactors
Part 5: Systematic and Statistical Errors in the Thermohydraulic Core Design of the Pebble Bed Reactor

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Comment:

In these English translations of KTA-Safety Standards the words shall, should and may are used with the following meanings:

shall indicates a mandatory requirement,

should indicates a requirement[1] to which exceptions are allowed. However, the exceptions shall be substantiated during the licensing procedure,

may indicates a permission and is, thus, neither a requirement (with or without exceptions) nor a recommendation: recommendations are worded as such, e.g., "it is recommended that ....".

The word combinations basically shall/shall basically are used in the case of mandatory requirements to which specific exceptions (and only those!) are permitted. These exceptions - other than in the case of should - are specified in the text of the safety standard.

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1 Please note that in the case of IAEA NUSS standards and ANSI standards, the word “should” indicates a mere recommendation.

KTA 3201.5

Reactor Core Design for High-Temperature Gas-Cooled Reactors
Part 5: Systematic and Statistical Errors in the Thermohydraulic Core Design of the Pebble Bed Reactor

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PLEASE NOTE:

Only the original German version of this safety standard represents the joint resolution of the 50-member Nuclear Safety standards Commission (Kerntechnischer Ausschuss, KTA). The German version was made public in Bundesanzeiger No. 44a on March 4,1988. Copies may be ordered through the Carl Heymanns Verlag KG, Gereonstr. 18-32, D-5000 Köln 1.

Nuclear Safety Standards Commission (KTA)

Federal Republic of Germany

Preliminary Remarks

(1) The safety standard series KTA 3102 "Design of the Reactor Core of Gas-Cooled High-Temperature Reactors" comprises the whole field of core design. With respect to the field of thermodynamics and fluid dynamics of a gas-cooled high-temperature reactor, the following areas are dealt with:

Part 1: Calculation of the Material Properties of Helium (Safety Standard 6/78)

Part 2: Heat Transport in Spherical Fuel Elements (Safety Standard 6/83)

Part 3: Loss of Pressure through Friction in Pebble Bed Cores (Safety Standard 11/84)

Part 4: Thermohydraulic Analytical Model for Stationary and Quasi-Stationary Conditions in Pebble Bed Cores (Safety Standard 11/84)

Part 5: Systematic and Statistical Errors in the Thermohydraulic Core Design of the Pebble-Bed Reactor

(2) The present Part 5 deals with the systematic and statistical errors in calculating the nuclear fuel and cooling gas temperatures which are required to demonstrate compliance with specified limit values for the thermohydraulic core design.

(3) On the basis of the mathematical thermohydraulic model, a failure analysis is used to ensure that the component and system related limit values are not exceeded in any of the operational conditions of the nuclear core.

1 Scope

(1) This safety standard applies to the pebble bed core of gas-cooled high temperature reactors. It deals with the requirements regarding the error analysis of the calculated gas and fuel element temperatures for specified normal operation and for incidents.

(2) This safety standard does not deal with the specification of numerical values for the individual types of errors nor for the component and system related limit values.

2 Definitions

(1) Nominal values of the thermohydraulic core design
Nominal values of the thermohydraulic core design are the result of the thermohydraulic calculations without consideration of systematic and statistical errors.

(2) Statistical errors of the thermohydraulic core design
Statistical errors of the thermohydraulic core design are deviations from the nominal values which are caused by inaccuracies of the empirical correlations and by inaccuracies of or variations in the input data for the calculations.

(3) Global statistical errors of the thermohydraulic core design
Global statistical errors of the thermohydraulic core design are deviations from the nominal values caused by inaccuracies of the input data and empirical correlations which are effective in all regions of the reactor core.

(4) Local statistical errors of the thermohydraulic core design
Local statistical errors of the thermohydraulic core design are deviations from the nominal values caused by statistically distributed, only locally effective variations of input data.

(5) Systematic errors of the thermohydraulic core design
Systematic errors of the thermohydraulic core design are established or assumed deviations from the nominal values which cannot be dealt with on a statistical basis.

3 Limit Values for the Thermohydraulic Core Design

Note:

In accordance with the Safety Criteria for Nuclear Power Plants (made public by the Federal Minister of the Interior on Oct. 21, 1977), limit values are such values of operational parameters of plant components, systems or therein contained media where, if they are not exceeded, a failure of safety related equipment can be precluded with a reasonable margin of safety.

(1) In order to maintain the effectiveness of the activity retention barriers in the fuel element, limit values shall be specified for the following parameters:

a) maximum temperature of the coated fuel particles in the fuel element,

b) maximum temperature of the fuel element surfaces,

c) maximum temperature differences within the fuel element.

(2) Since the effectiveness of the activity retention barriers in the fuel elements depends on the temperature and the individual exposition time, it is allowed to specify graduated limit values separately for different time intervals.

(3) For the protection of the components and systems in series with the reactor core, limit values shall be specified for the gas temperatures at the core outlet.

(4) In order to demonstrate that the limit values are adhered to, the systematic and statistical errors of the core design shall be determined and documented.

4 Errors in the Thermohydraulic Core Design

4.1 Systematic Errors

4.1.1 Model Simplifications

(1) The results of simplified cylinder symmetric calculations shall be corrected for the three-dimensional effects of

a) non-central charging cone,

b) non-central fuel element discharge pipes,

c) bulges in the reflector cylinder,

d) inserted absorber rods,

e) azimuthal asymmetries of coolant flow at the core inlet and core outlet,

f) azimuthal profile of the power density distribution,

by corrective supplements to the nominal values of the gas and fuel element temperatures.

(2) If the temperature within the fuel elements is calculated with a homogenized power density distribution then the temperature difference between the fuel particles and the surrounding fuel element zone shall be accounted for by corrective supplements.

4.1.2 Numerical Procedures

The effects on the temperature distributions of the spatial and temporal discretizations required for solving the thermohydraulic equations shall be determined and documented.

4.1.3 Void Distribution in the Spherical Fuel Element Pile

If the calculations of gas flow and temperature distributions inside the fuel element pile are simplified by assuming a uniform void distribution then the effect of the systematic changes of the void distribution at the radial boundaries and in the vicinity of absorber rods shall be determined and documented.

4.2 Statistical Errors

4.2.1 Global Statistical Errors

4.2.1.1 Neutron Physical Core Design

The Influence on the calculated temperature distributions of inaccuracies in the power density distribution inside the fuel element pile resulting from the neutron physical core design shall be determined and recorded as global statistical errors. Here, the axial and radial power density distribution profiles shall be normalized such that the overall power of the fuel element pile is maintained.

4.2.1.2 Empirical Correlations

The influence on the calculated temperature distributions of inaccuracies due to

a) the equation for calculating the material properties of helium in accordance with KTA 3102.1,

b) the equation for calculating the heat transfer coefficient inside the fuel element pile in accordance with KTA 3102.2,

c) the equation for calculating the pressure loss coefficient of the fuel element pile due to friction in accordance with KTA 3102.3,

d) the equation for calculating the effective thermal conductivity of the fuel element pile,

e) the equation for calculating the dependence of temperature distribution and neutron flux on the thermal conductivity inside the spherical fuel elements,

f) the equation for calculating the volume related specific heat of the spherical fuel elements,

g) the average void distribution of the pebble bed core

shall be determined and documented as global statistical errors.

4.2.1.3 Boundary Conditions

The influence on the calculated temperature distribution of inaccuracies due to

a) the specified operating parameters: gas throughput, gas inlet temperature, pressure and overall power,

b) the heat transfer at all boundaries of the pebble bed core,

c) the inwardly and outwardly directed coolant gas flows in the reactor core at the radial reflector and at inserted absorber rods,

d) the pressure loss coefficients of radially graduated coolant gas bore holes in the bottom reflector

shall be determined and documented as global statistical errors.

Note:

Global statistical errors can be determined by applying varied input data to the thermohydraulic calculation model.

When superposing independent global statistical errors, their frequency distribution may be taken into consideration.

4.2.2 Local Statistical Errors

The influence on the temperature distribution of variations of the fuel element characteristics due to manufacturing, i.e.,

a) thickness of the outer shell free of fuel,

b) content of heavy metals,

c) heat conductivity,

and of local variations of the mixture proportion of spherical fuel elements of different power shall be determined and documented as local statistical errors.

Note:

Local statistical errors of the fuel element temperatures can be directly determined from the calculated local nominal values.

When superposing independent local statistical errors, their frequency distribution may be taken into consideration.

Appendix A

Regulations Referred to in this Safety Standard

Regulations referred to in this safety standard are valid only in the versions cited below. Regulations which are referred to within these regulations are valid only in the version that was valid when the latter regulations were established or issued.

KTA 3102.1		Reactor Core Design for High-Temperature Gas-Cooled Reactors
	(6/78)	Part 1: Calculation of the Material Properties of Helium
	(6/83)	Part 2: Heat Transport in Spherical Fuel Element Piles
	(3/81)	Part 3: Loss of Pressure through Friction in Spherical Fuel Element Piles