DEVELOPMAENT OF METHODOLOGY FOR THE ANALYSIS OF FUEL BEHAVIOR IN LIGHT WATER REACTOR IN DESIGN BASIS ACCIDENTS.

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The report attempts to analyze the current experience of the safety fuel for light-water reactors (LWRs) under design-basis accident conditions in terms of its compliance with international requirements for licensing nuclear power plants.

The components of fuel behavior analysis methodology in design basis accidents in LWRs were considered, such as classification of design basis accidents, phenomenology of fuel behavior in design basis accidents, system of fuel safety criteria and their experimental support, applicability of used computer codes and input data for computational analysis of the fuel behavior in accidents, way of accounting for the uncertainty of calculation models and the input data.

A brief history of the development of probabilistic safety analysis methodology for nuclear power plants abroad is considered.

The examples of a conservative approach to safety analysis of VVER fuel and probabilistic approach to safety analysis of fuel TVS-K are performed.

Actual problems in development of the methodology of analyzing the behavior of VVER fuel at the design basis accident conditions consist, according to the authors opinion, in following:

1) Development of a common methodology for analyzing the behavior of VVER fuel in the design basis accidents, implementing a realistic approach to the analysis of uncertainty - in the future it is necessary for the licensing of operating VVER fuel abroad
2) Experimental and analytical support to the methodology:
   - experimental studies to identify and study the characteristics of the key uncertainties of computational models of fuel and the cladding,
   - development of computational models of key events in codes
   - validation code on the basis of integral experiments.

The objectives of the analysis of fuel behavior in design basis accidents

Analysis of fuel behavior (fuel rods, fuel assemblies) is the necessary element of NPP safety confirmation.

The objectives of the analysis of fuel behavior under accident conditions is the determination of degree of damage of fuel system including

- ensure the preservation of cooling geometry of core
- ensure of post-accident unloading of assembly
- assessment of radiation effects of accidents

The degree of damage of fuel system under design accident conditions is measured by system of fuel safety criteria, implementation of these criteria is the necessary condition to save core geometry, i.e. to prevent melting and fragmentation.

For analysis of radiation effects of accidents the conservative estimation of amount of depressurized fuel rods is determined.

Methodology components of the design accident fuel behavior analysis

Methodology for the analysis of fuel behavior in design basis accidents includes the following components

1) A list of design accidents to be analyzed, expert determination of the selection of different accident types (LOCA, non-LOCA, RIA) depending on the impact on fuel,
2) Phenomenology of fuel behavior in design accidents, expert option of key phenomenon, impacting on the criterion fuel characteristics,
3) Regulation criteria for fuel in design-based accidents:
   - list of criteria
   - experimental confirmation of criteria
4) Qualification of computer codes (neutron, thermo-hydraulic and thermo-mechanical), using in frame of this methodology:
   - determination and confirmation of uncertainty characteristics of calculation models for key phenomenon
   - code validation for compliance field of application with design-basis fuel characteristics
5) Qualification of input data
   - determination and substantiation of the uncertainty characteristics for technological and design fuel parameters,
   - determination and substantiation of the uncertainty characteristics for base-irradiation conditions,
   - determination and substantiation of the uncertainty characteristics for design-basis accidents conditions.
6) Interface interactions between codes in joint calculation
7) Selection and determination of calculational method for fuel criteria in design-based accidents:
   - Conservative methodology - it requires justification of conservative sets of code parameters and input data for determination of conservative estimation of safety fuel criteria for each type of design-based accident
   - Realistic methodology with uncertainty analysis – it requires the calculational algorithm providing needed amount of computed runs using random vectors of model parameters as input data and subsequent statistical processing of the results.

Brief history of the development of NPP safety methodology abroad

The nuclear industry can have a significant impact on the lives and health of people and the environment, and therefore it is strictly controlled by the regulation authorities and the special normative documents. Various types of accidents at nuclear power plants (NPPs) are studied to substantiate their safety from the 50s of the XX century. At the time, computers did not exist, and the analyses of reactor safety were based on the experiments, thermal-hydraulic models and engineering evaluations.

More systematic thermal-hydraulic studies and experiments were conducted in the 60s, at that date the specific phenomena like two-phase critical flow, critical heat flux, fuel rod depressurization, etc. were considered. New findings from those researches were used in reactor safety and licensing documents.

Wide spread of the computer calculations for nuclear reactor safety analyses was started in the '70s. The accident analysis included primitive numerical codes and results of lately called integral-system experiments. The nuclear regulatory point-of-view was well established by the USNRC (US AEC at the time) in 1971 [1]. This triggered a wide variety of researches aimed at the evaluation of safety margins focusing on the estimation of the maximum temperature on the surface of fuel rods following Large Break Loss of Coolant Accident (LB-LOCA). In 1974, the Code of Federal Regulation 10 CFR-50.46, Appendix K [2] formulated criteria to limit the maximum temperature and the degree of oxidation of fuel rod by a conservative approach when the safety analysis of technological systems is realized with such values of system parameters and characteristics which is known that lead to the most unfavorable results.

Robust versions of the thermohydraulic codes became available in the 80s. Soon the importance of code verification and validation [3] at the final stage of licensing in the area of security became apparent. To do this, CSNI (Committee on the Safety of Nuclear Installations of OECD/NEA) established databases to solve the problems of scaling (i.e., the applicability of the models and the results of specific experiments to the full-scale reactor), carried out international activities to establish the best ways of verification and validation, to study so-called effect of the user (i.e. the individual scientist conducting the settlement), the expert opinion. Appendix K of the document 10 CFR 50.46 has also been used in 80s for the purpose of licensing.

In the early 90s it became clear that the development of methods of uncertainty estimation in the results of calculations of thermal hydraulic system codes is necessary. The impetus for the development of methods of estimation uncertainty US NRC was the decision in 1989 to use realistic models for the estimation of uncertainties in the licensing of emergency core cooling systems for reactor core under accident conditions. The first versions of these methods have been proposed organizations US NRC [4], GRS [5] and the University of Pisa. [6] US NRC issued a regulatory guide (Regulatory Guide (RG) 1.157) [7], where it was planned to use thermal-hydraulic system codes with conservative models. These models have been used in phenomenological areas that did not have enough information. Requirements of document RG 1.157 determined the guiding principles of the best estimate of code during licensing. Document 10 CFR 50.46 has opened the possibility of using codes of "best-estimate" with the uncertainty for LOCA analysis under licensing. The reduction BEPU (best estimate plus uncertainty) began to be used.

Using BEPU approaches the licensing began in the 00's. The impetus for this was some of the key events:

1) US NRC issued a document RG 1.203 [8] opens the possibility of using the BEPU approach in licensing,

2) CSNI started and completed a six-year project BEMUSE [9] devoted to demonstrate the applicability of the methods of estimation uncertainty for accidents LBLOCA. The objective was achieved, but too large difference between the results of calculations of program participants require careful interpretation.

Currently BEPU methods are used, and continue to develop.

CSAU methodology

The aim of the CSAU methodology is to investigate the uncertainty of safety related output parameters, for example PCT (Peak Cladding Temperature). CSAU methodology consists of three stages

1) evaluation of the code's applicability to expansion of the phenomenon with a small object to the scale of nuclear power plant (scale effect)
2) evaluation of the code's applicability to a selected plant scenario.
3) Evaluation of uncertainty of basic results of calculations

Experts identify all the relevant phenomena. Following this step, the most important phenomena are identified and are listed as 'highly ranked' phenomena, based on an examination of experimental data and code predictions of the scenario under investigation. In the resulting phenomena identification and ranking table (PIRT), ranking is accomplished by expert judgment. All necessary calculations are performed using an optimized nodalization to capture the important physical phenomena. This nodalization represents a compromise between accuracy and cost, based on experience obtained by analysing separate effects tests (SETs) and integral experiments. No particular method or criteria are prescribed to accomplish this task.

Only parameters important for the highly ranked phenomena are selected for consideration as uncertain input parameters. The selection is based on the expert opinion of their influence on the output parameters. Information from the manufacture of nuclear power plant components as well as from experiments and previous calculations was used to define the mean value and probability distribution or standard deviation of uncertain parameters Uniform and normal distributions were used in the two applications performed to date. Output uncertainty is the result of the propagation of input uncertainties through a number of code calculations.

Input parameter uncertainty can be either due to its stochastic nature (i.e. code independent) or due to imprecise knowledge of the parameter values. No statistical method for uncertainty evaluation has been formally proposed in CSAU. A response surface approach [10] has been used in the applications performed to date. The response surface fits the code predictions obtained for selected parameters, and is used instead of the original computer code. Such an approach then entails the use of a limited number of uncertain parameters in order to reduce the number of code runs and the cost of analysis.

GRS method

The GRS method [11] developed within the CSAU approach has some other important features in addition to those mentioned above:

a) The uncertainty space of input parameters (defined by their uncertainty ranges) is sampled at random according to the combined probability distribution of the uncertain parameters, and code calculations are performed by sampled sets of parameters.

b) The number of code calculations is determined by the requirement to estimate a tolerance and confidence interval for the quantity of interest (such as the PCT). Following a proposal by GRS, Wilks' formula [12], [13] is used to determine the number of calculations required to obtain the uncertainty bands.

c) Statistical evaluations are performed to determine the sensitivities of input parameter uncertainties on the uncertainties of key results (parameter importance analysis).

The minimum number n of code runs to be performed is given by Wilks' formula [12] for one output key parameter for one-sided and two-sided statistical tolerance intervals

$1 - \alpha^n \geq \beta$ and $1 - \alpha^n - n\,(1 - \alpha)\,\alpha^{n-1} \geq \beta$ respectively,

where α – fractile (probability of not exceeding the maximum permissible value), β – confidence level, i.e. probability that probability that output parameter will be in the tolerance interval is more than α. Quantile and confidence level (β / α) is called the reliability level of this estimate.

The maximum value of output parameter from n calculations is the upper limit of tolerance interval, which is compared with the criterion.

The values of the input parameters are generated by Monte-Carlo method according to the selected distribution densities and the dependencies between them. The required number of calculations necessary to provide (β/α) reliability level for one criterion parameter is shown in Table 1. When licensing one-sided tolerance intervals are used to estimate the upper limits of criterion parameters.

Table 1

The required number of calculations for (β / α) reliability level

	One -sided statistical tolerance intervals			Two -sided statistical tolerance intervals		
β/α	0.90	**0.95**	0.99	0.90	**0.95**	0.99
0.90	22	45	230	38	77	388
0.95	29	**59**	299	46	**93**	473
0.99	44	90	459	64	130	662

Experts identify significant uncertainties to be considered in the analysis, including the modelling uncertainties and the related parameters, and identify and quantify dependencies between uncertain parameters. Probability density functions (PDFs) are used to quantify the state of knowledge of uncertain parameters for the specific scenario. Uncertainties of code model parameters are obtained based on validation experience. The scaling effect has to be quantified as a model uncertainty.

Sensitivity analyzes the output parameter to the input parameters allow you to:

−rank the input parameters on the contribution to the total output uncertainty of the outcome;

−help determine the future direction of development of the code;

−help identify the most important experimental studies that should be spent for more information.

This information allows to understand uncertainty of what input parameters (density and ranges) should be determined more precisely. Sensitivity measures by using regression or correlation techniques from the sets of input parameters and from the corresponding output values allow ranking of the uncertain input parameters in relation to their contribution to output uncertainty. The ranking of parameters is therefore a result of the analysis, not of prior expert judgement.

There are similar statistical methods, using Wilks formula for determining the required number of calculations: the method of AREVA [14], ASTRUM - method of Westinghouse [15], KREM in Korea, and some others. Method AREVA licensed USNRC in 2003, the method ASTRUM in 2004.

The use of a safety analysis methodology for licensing fuel loads of NPP is preceded by documentation and licensing of this methodology in the regulatory body of the country, operating this NPP.

The current state of methodology development for the analysis of the fuel behavior under design-basis accidents conditions in Russia

VVER fuel

To this day deterministic conservative approach was used for justification of safe behavior of fuel in VVER design basis accidents.

First and only time in the Russian practice methodology of design and safety analyzes was systematically set out in the framework of the contract of JSC "TVEL" with CEZ a.s. for the purpose of licensing TVSA-T fuel for Temelin NPP in the regulatory body of the Czech Republic.

In the work took part: NRC "Kurchatov Institute", OKBM, VNIINM. Codes which were used: neutron codes of the Kurchatov Institute, thermal-hydraulic codes TIGER-1, RELAP5/mod3.2, thermomechanical codes START-3, RAPTA-5 [16 - 19].

The methodology for developing the project TVSA-T for the Temelin NPP has been documented in a set of reports with descriptions of the design basis, the development of methods of neutron-physical design of the core, the thermal-hydraulic design of TVSA-T fuel rod, thermo-mechanical design, analysis of accidents such as LOCA, the analysis of accidents such as non-LOCA and RIA, the analysis of the behavior of fuel elements in the design-basis accidents.

The methodology has been considered, analyzed by experts of the customer, rethought in light of their comments and suggestions, and ultimately approved by the regulatory authority of the Czech Republic. The most difficult questions were related to uncertainty estimates of material properties, simulation models, input data and conservative estimates of criteria justification of the fuel in the considered variants of design basis accidents.

Deterministic conservative approach allows the implementation of fuel safety criteria for reactor plant with VVER with a large enough margin to their limits, which can be seen in the study of fuel TVSA-T, the main results of which are shown in Table 2.

Table 2

Criterion characteristics of fuel rods of VVER-1000 in design basis accidents

Parameter / Type of accident	Maximum fuel temperature, ^{o}C		Peak fuel enthalpy, cal/g		PCT, ^{o}C	ECR, %		ECR, %	Failed fuel rods, %
	Fuel rod	Gd Fuel Rod	≤ 50 MWd/kgU	> 50 MWd/kgU		DA	NO +DA		
Large Break LOCA	1876	1904	-	-	1059	1.7	3.06	0.17	12.05
Control rod ejection at nominal reactor power level	2164	2026	107.5	100.5	517	0	3.06	0	0
Control rod ejection at minimuml reactor power level	2066	1623	120,4	94,9	715,2	0,03	3,06	-	0
Criterion	2790	2360	230	165	1200	18		1	No criteria

Fuel TVS-K

Methods of safety analyzes for the licensing of fuel ***TVS-K*** were developed within the projects of JSC "TVEL", aimed at promoting fuel produced in Russia to market PWR.

In the works took part: FEI, OKBM, VNIINM, IPM.

Used codes: RELAP/SCDAPSIM/MOD3.4 [20], START-3, RAPTA-5.2 [21], the code CaPpaPI [22].

Interface data files from the operator were used (in particular, for LOCA - basic boundary conditions and power distribution in the core).

The fundamental difference between this method of safety analysis and method of the project TVSA-T for Temelin NPP is that here they used a realistic approach to the assessment of uncertainties criterion characteristics.

The need to use a realistic approach to the uncertainty analysis consists in the following reasons:

1) realistic approach of uncertainty analysis is adopted by regulatory authorities of countries operating PWR,

2) conservative approach does not satisfy the fuel safety criteria because of higher power density in PWR vs. VVER.

Developing statistical methodology includes the following elements:

1) Identification of important events and relevant key parameters for each of the codes START-RELAP-RAPTA.

2) Quantifying the uncertainties of individual models and key parameters (ranges of changing and probability density function).

3) Generation of random sets of input parameters with uncertainty (random vectors) by Monte-Carlo method in accordance with a given density distribution.

4) Performing the required number of calculations using the random vectors of input data;

5) Processing of the results: from the results of N calculations maximum values of criteria parameters (PCT, ECR, CWO, ...) are selected, these values are upper bounds for a given probability α level of confidence β.

The block-diagram of the method of calculation of fuel behavior in LOCA is shown in Figure 1.

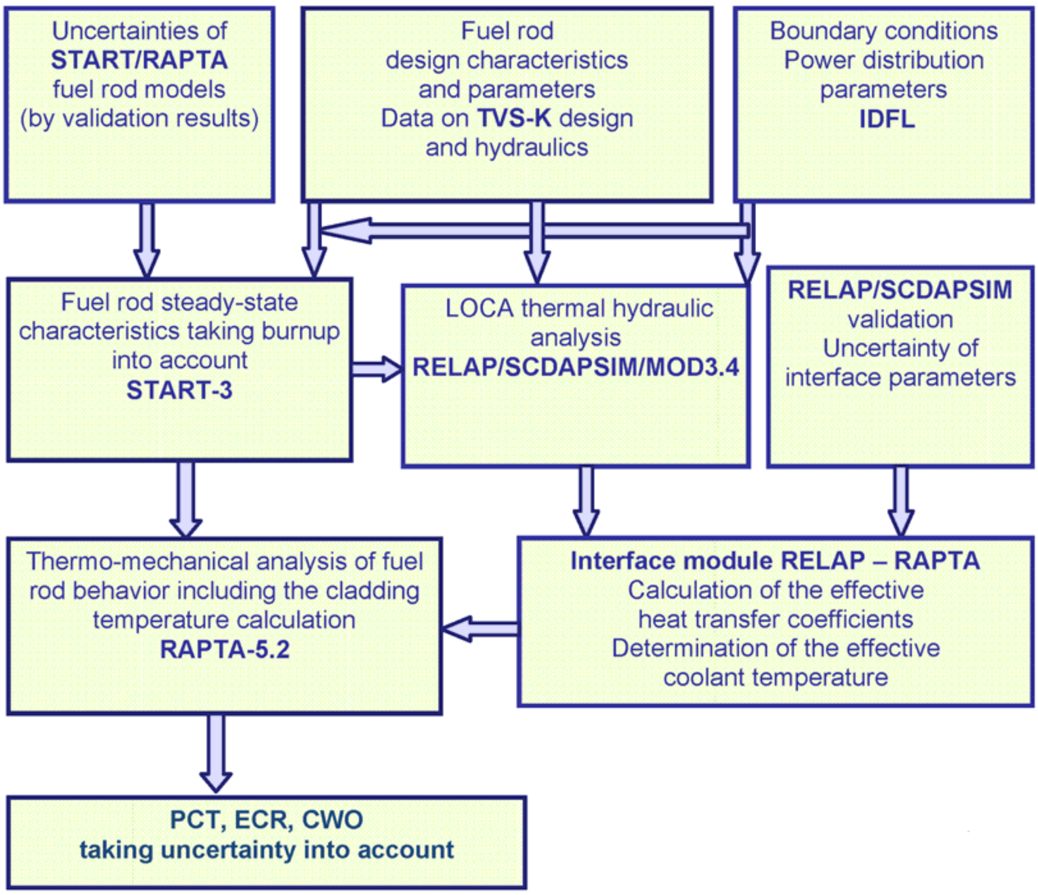

Figure 1 – Block-diagram of the method of calculation of fuel behavior in LOCA

Identification of key parameters and models of fuel rod TVS-K for the LOCA analyzes was performed using tables PIRT (NUREG/CR-6744 Phenomenon Identification and Ranking Tables (PIRTs) for Loss-of-Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Burnup Fuel). As a result, the main parameters of the fuel rod design and technological characteristics, model parameters for NO and LOCA conditions, parameters of NO and LOCA loading conditions were determined (see Table 3). Sensitivity analyzes show that the greatest impact on the value of PCT has the deviation of a key parameter of the thermal conductivity of the fuel.

At this stage of methodology development the preliminary distribution functions of the probability densities for key parameters were selected, it will be further estimated more accurately.

The calculations of LB LOCA sample problem using boundary conditions for PWR with fuel burnup of 0, 30, 50 MWd/kgU. The results are shown in Table 4 and illustrated for variant with burnup of 50 MWd/kgU in Figures 2 - 4. The accordance of the safety criteria with certain margins for PCT and ECR, similar to margins obtained for VVER-1000 is shown in Table 2.

Table 3

List of fuel rod key parameters for LOCA analysis

Type of parameter	№	Parameter
Design and technological	1	Outer radius of fuel rod cladding
	2	Inner radius of fuel rod cladding
	3	Fuel pellet radius
	4	Fuel density
	5	Fuel resintering
	6	Initial internal pressure
	7	Gas plenum length
	8	Fuel open porosity
Model	9	Fuel thermal conductivity
	10	Fuel thermal expansion
	11	Thermal conductivity of zirconium oxide
	12	Thermal conductivity of the fuel rod cladding material
	13	Thermal conductivity of fuel rod internal gas
	14	Fission Gas Release from fuel
	15	Cladding corrosion
	16	Fuel swelling
	17	Cladding creep
	18	Cladding thermal expansion
	19	Cladding high-temperature oxidation
	20	E110 yield stress
	21	Cladding burst criterion
Operational, NO	22	Pre-accident enveloping power history
Loading in LOCA	23	Power distribution parameters: FΔH, FQ
	24	Heat transfer from cladding

Table 4

The results of calculations of criteria parameters of fuel rod in the example of LB LOCA accident

№	Burnup, MW*day/kgU	Criteria characteristics	Base value	Upper limit to the level of reliability (95/95)
1	0	Maximum temperature of fuel, °C	1669.8	1833.6
2	30		1729.5	1925.9
3	50		1893.3	2027.9
4	0	Maximum temperature of cladding (PCT),°C	964.3	1028.4
5	30		966.2	1023.1
6	50		1110.2	1136.5
7	0	Equivalent cladding reacted (ECR),%	0.5402	0.9249
8	30		1.8413	1.9725
9	50		2.3362	2.4812
10	0	Core-wide oxidation (CWO), %	0.2178	0.2945
11	30		0.0395	0.0603
12	50		0.1296	0.1691

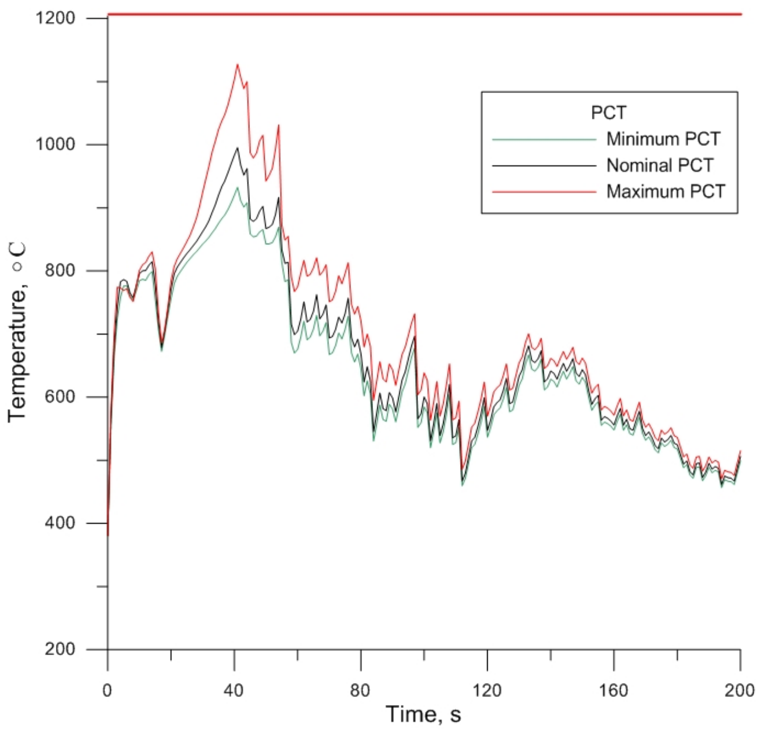

Figure 2 – Cladding temperature during the accident in the calculations with implementation of max, nom, min PCT at a burnup of 50 MW *day/kg U

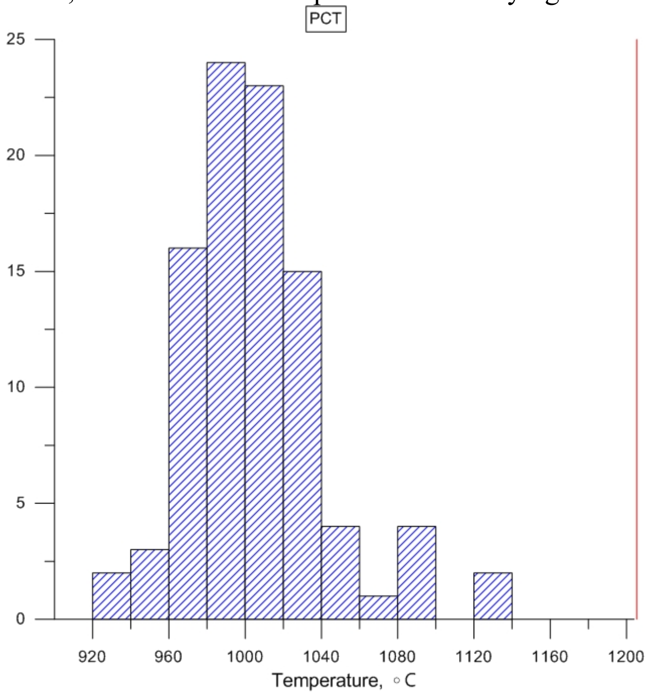

Figure 3 – Histogram of 93 calculations PCT at a burnup of 50 MW*day/kg U

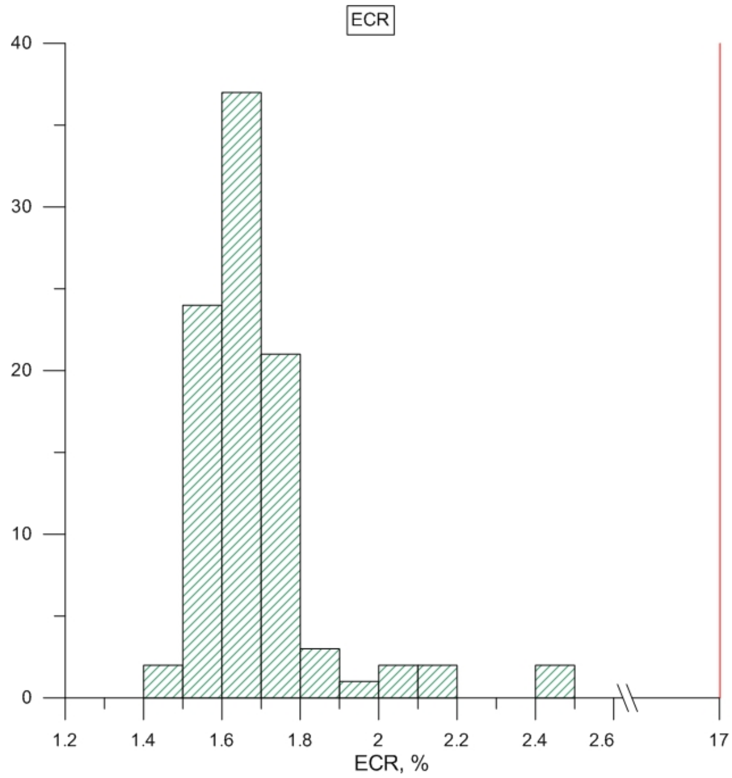

Figure 4 – Histogram of 93 calculations ECR
at a burnup of 50 MW*day/kg U

Actual problems of developing of methodology for analyzing the behavior of VVER fuel in the design basis accidents

In summary, there are following actual tasks for development of analyzing the behavior of the fuel in the design-basis accidents:

1) Development of a methodology for analyzing the behavior of VVER fuel in the design-basis accidents, realizing a realistic approach to the analysis of uncertainties - in the future it is necessary for licensing the operation of reactor plant with VVER fuel abroad;

2) Experimental and analytical support to the methodology:

- experimental studies to identify and study the characteristics of the key uncertainties of computational models of fuel and cladding,
- development of computational models of key events in fuel rod codes,
- code validation on the basis of integral experiments.

List of acronyms and designations

AREVA	French company engaged in the development and production of equipment for nuclear power
ASTRUM	Automatic Statistical TReatment of Uncertainty
BEMUSE	Best Estimate Method – Uncertainty and Sensitivity Evaluation
BEPU	Best Estimate Plus Uncertainty
CIAU	Capability of Internal Assessment of Uncertainty
CSAU	Code Scalling Applicablity and Uncertainty
CSNI	Committee on the Safety of Nuclear Installation
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
KREM	KEPRI (Korea Electric Power Institute) Realistic Evaluation Model
NPP	Nuclear Power Plant
LOCA	Loss of Coolant Accident
OECD	Organization for Economic Co-operation and Development
PIRT	Phenomena Identification and Ranking Table
PCT	Peak Cladding Temperature
RIA	Reactivity Initiated Accident
RG	Regulatory Guide
SAR	Safety Analysis Report
TVS-K	PWR fuel assembly designed by JSC TVEL
US NRC	United States Nuclear Regulatory Commission
V&V	Verification & Validation
UMAE	Uncertainty Methodology based on Accuracy Extrapolation

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