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UNCERTAINTY ANALYSIS BECOMING COMMON PRACTICE IN SAFETY ASSESSMENT

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ABSTRACT

During the recent years an increasing interest in computational reactor safety analysis is to replace the conservative evaluation model calculations by best estimate calculations supplemented by uncertainty analysis of the code results. The evaluation of the margin to acceptance criteria, e.g. the maximum fuel rod clad temperature, should be based on the upper limit of the calculated uncertainty range. For example, due to power increase, licensing limits are approached. Therefore, regulators are looking closer on the way, how calculations are performed to meet these acceptance criteria. Methods have been developed and presented to quantify the uncertainty of computer code results. They are briefly presented in this paper.

The present overview considers the international situation of development of uncertainty evaluation of computer code results and their application in licensing. Best estimate analysis plus uncertainty evaluation is used in licensing up to now in approximately seven countries. Demonstrations of applying uncertainty methods have been performed in nine additional countries at least. Most organizations use statistical methods.

One statistical method is the GRS method proposing ordered statistics. Several demonstrations to apply the GRS method have been performed by GRS for courses of events in the nuclear steam supply system, calculating experiments as well as nuclear power plants. The method has also been applied for post test calculations of containment behavior, as well as severe accidents. One of the most important conclusions is that care must be exercised in determining ranges and probability distributions of the uncertain input parameters.

INTRODUCTION

The present overview considers the international situation of development of uncertainty evaluation of computer code results and their application in licensing.

Best estimate computer codes are used to calculate postulated loss of coolant accidents and transients in a realistic way and not in a conservative way. There is an increasing interest in computational reactor safety analysis to replace the conservative evaluation model calculations by best estimate calculations supplemented by a quantitative uncertainty analysis.

The first proposal to perform uncertainty analysis in licensing applications was initiated by the US Nuclear Regulatory Commission in the year 1989. The USA Code of Federal Regulations (CFR) 10 CFR 50.46, [1], for example, allows either to use a “best estimate” (BE) code plus identification and quantification of uncertainties, or the conservative option using conservative computer code models listed in Appendix K of the CFR. However, when using a best estimate computer code, it is required that uncertainties have to be identified and assessed so that the uncertainty in the calculated results can be estimated.

A high level of probability has to be applied that acceptance criteria would not be exceeded. That high level of probability is specified in the NRC Regulatory Guide 1.157 to 95% or more [2].

IAEA SAFETY GUIDES AND REPORT SERIES

The International Atomic Energy Agency (IAEA) Safety Guide “Safety Assessment and Verification for Nuclear Power Plants” NS-G-1.2, §4.90, issued in the year 2001, recommends that uncertainties should be statistically combined if a combination of BE computer code and realistic assumptions on initial and boundary conditions is used [3]. The calculated results shall not exceed acceptance criteria with a specified high probability. The high probability is not specified in that IAEA Guide.

The change from conservative codes to best estimate codes plus uncertainty analysis in the USA has shown significant margins to the regulatory acceptance criterion 1200 °C, for example. Consequently, many utilities applied for power up-rates.

Regulations in most of the other countries permit the use of best estimate codes in licensing without uncertainty analysis. That is a result of a survey performed by the Organization for Economic Co-operation and Development/ Committee on Safety of Nuclear Installations (OECD/CSNI) in the year 1996, [4]. Added requirements for conservative assumptions, e.g. initial and boundary conditions and loss of off-site power – if that is leading to unfavorable conditions - have to be assumed. In addition, unavailability of equipment, like single failure, needs to be considered in the safety analysis. Since no uncertainty analysis was required in these countries, the conservatism of calculation results is not quantified.

Such a procedure is acceptable according to the IAEA Safety Guide No. NS-G-1.2, §4.89, however, a "sufficient" evaluation of the uncertainties of the results should be performed [3]. What is meant by "sufficient" evaluation is not described in that Safety Guide.

Some reasons to apply best estimate codes plus uncertainty analysis are for example: Conservative code models may not always lead to conservative results and may show misleading sequences of events and unrealistic time-scales. Thermal-hydraulic system codes became more and more realistic codes based on comprehensive development and validation over the time. Consequently, inherent conservatisms of the code models were reduced to become more realistic.

An IAEA Safety Report Series No. 23: "Accident analysis for Nuclear Power Plants", issued in the year 2002, recommends sensitivity and uncertainty analysis if best estimate codes are used in licensing analysis, [5]. A comprehensive overview about uncertainty methods can be found in the IAEA Safety Report Series No. 52, „Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation", issued in 2008, [6].

A recently published IAEA Safety Guide SSG-2 "Deterministic Safety Analysis for Nuclear Power Plants" [7] provides harmonized guidance to designers, operators, regulators and providers of technical support on deterministic safety analysis for nuclear power plants. Three ways of analyzing anticipated operational occurrences and design basis accidents to demonstrate that the safety requirements are met, are currently used to support applications for licensing:

1. Use of conservative computer codes with conservative initial and boundary conditions (conservative analysis);
2. Use of best estimate computer codes combined with conservative initial and boundary conditions (combined analysis);
3. Use of best estimate computer codes with conservative and/or realistic input data but coupled with an evaluation of the uncertainties in the calculation results, with account taken of both the uncertainties in the input data and the uncertainties associated with the models in the best estimate computer code (best estimate analysis). The result, which reflects conservative choice but has a quantified level of uncertainty, is used in the safety evaluation.

The Safety Guide SSG-2 focuses on thermal-hydraulic and source term evaluation for operational states and accident conditions for nuclear reactors. The quality of the analysis of computer codes and their verification and validation are also described together with the relationship of deterministic safety analysis to engineering aspects of safety and to probabilistic safety analysis. The Safety Guide SSG-2 also addresses applications of deterministic safety analysis for the development and validation of emergency operating procedures and the determination of safety margins for modifications to nuclear power plants.

STATUS OF APPLICATIONS IN LICENSING AND SIGNIFICANT ACTIVITIES PRIOR TO USE IN LICENSING

A new focus is on best estimate plus uncertainty analysis due to many applications of power up-rates as well as optimized fuel strategies using higher enrichment to achieve higher burn-up values. Mainly due to power increase, licensing limits are approached. Therefore, regulators are looking closer on the way, how calculations are performed to meet these acceptance criteria.

Main uncertainty methods

Several methods exist to evaluate the uncertainty of computer code results:

1. Code Scaling, Applicability and Uncertainty (CSAU) evaluation method proposed by USNRC establishing a procedure for performing uncertainty analysis, consisting of 14 steps [8];
2. CSAU demonstration using a fitted "response surface" to calculate the uncertainty of a single valued result, like peak cladding temperature [8];

3. Ordered statistics methods for time dependent and single valued uncertainty bounds: First proposed method is the GRS-Method [9, 10], then followed the AREVA-Method [11], ASTRUM-Method of Westinghouse [12] and several more;
   the AREVA method has been licensed by USNRC in the year 2003 and the ASTRUM Method in 2004;

4. Direct use of differences between experimental data and calculation results – the Uncertainty Method of Accuracy Extrapolation (UMAE) and Code with the Capability of Internal Assessment of Uncertainty (CIAU) of University of Pisa [13].

The last method determines the output uncertainties by differences between calculated and measured values of different experiments investigating the same accident scenario. These approaches are:

1. University of Pisa methods (UMAEA and CIAU mentioned above).
2. Siemens method applied to evaluate model uncertainties in NPP Angra-2, Brazil LBLOCA licensing analysis.

Siemens applied to get their method licensed in USA. The USNRC, however, had a concern of compensating errors of the computer codes and asked for propagation from input uncertainties through output uncertainties. There is also a concern of scaling distortions of integral experiments and different time scales in the experiments compared with reactor scale and possible influence on the deviation between calculations and data. The derived uncertainties are also dependent on the selection of integral experiments. Additional uncertainties of plant conditions and fuel related parameters should be quantified either by additional bias or statistical evaluation.

Statistical uncertainty and sensitivity analysis provides statements on:

− Uncertainty range of code results that enables to determine the margin between the bound of uncertainty range closest to an acceptance criterion and the acceptance criterion
− Sensitivity measures about the influence of input parameters on calculation results, i.e. a ranking of importance which
    - allows a ranking of input parameters on output uncertainty as result of the analysis,
    - guides further code development,
    - prioritizes experimental investigations to obtain more detailed information.

The sensitivity or importance measures give useful information about those input parameters influencing the uncertainty of computer code results most. That information can be used to find out which ranges and distributions of input uncertainties should potentially be determined more accurately.

Applications

Best estimate analysis plus uncertainty evaluation is used in licensing up to now in the following countries: USA (ordered statistics), Netherlands (other statistics method), Brazil (output uncertainties and other statistics method as well as CIAU), Korea (ordered statistics), Lithuania (ordered statistics), Spain (ordered statistics), and Argentina (CIAU).

Significant activities for use in licensing are performed in these countries: Canada, Czech Republic, France, Germany, Hungary, Japan, Russia, Slovak Republic, and Ukraine.

International comparisons of uncertainty analyses

Mainly two international comparisons have been performed in the frame of OECD/ CSNI to compare applications of uncertainty methods:

1. Uncertainty Methods Study (UMS),
2. Best Estimate Methods – Uncertainty and Sensitivity Evaluation (BEMUSE).

One result of the BEMUSE program, e.g. the Zion application with regard to maximum peak clad temperature is shown in Figure 1.

The BEMUSE applications by different participants can be summarized as follows. Two uncertainty methods were applied:

1. Statistical method was applied by the majority of participants: 10 from 11 participants in the application of the LOFT experiment**,** and 2 from 14 in the Zion application [14].
2. Only University Pisa used their UMAE/ CIAU method by a compilation of numerous results of integral experiments.

The use of both methods was successfully mastered and the quality of base case calculation turned out to be essential for the uncertainty results.

Differences of results may come from different methods. When using statistical methods differences may be due to different input uncertainties, their ranges and distributions. Deviations between participants are already seen in the basic or reference calculations. It is claimed that a conservative method bounds all uncertainties by conservative assumptions. That may be right when conservative code models are used in addition to conservative initial and boundary conditions. In many countries, however, a best estimate code plus conservative initial and boundary conditions are accepted for conservative analysis in licensing. Differences in calculation results of best estimate and even conservative codes would also be seen, comparing results of different users of the same computer code due to different nodalisations and code options. That was observed in all

International Standard Problems where participants calculated the same experiment or reactor event.

Another important lesson learned to improve uncertainty analysis is the importance of training the users to apply uncertainty methods.

NUMBER OF CALCULATIONS FOR STATISTICAL METHODS TO MEET MORE THAN ONE REGULATORY LIMIT

Another very controversial international discussion took place about the number of calculations to be performed using ordered statistics methods [15-19]. That issue was mainly brought up when more than one regulatory acceptance criterion or limit has to be met. Wilks' formula gives the minimum number of calculation runs to be performed [10, 20]. Wald [21] and others [15-19] extended Wilks' formula for multi-dimensional joint/ simultaneous tolerance limits or intervals. However, it seems that a direct and satisfactory extension of the concept of tolerance limits for safety-relevant applications in nuclear safety is difficult, and even not necessary. A slightly modified concept has therefore been proposed by Krzykacz-Hausmann from GRS, introducing a **lower confidence limit** [22]. The lower confidence limit according to Clopper-Pearson [23] for the binomial parameter is now the unknown probability that a result is lower than a regulatory acceptance limit. Instead of direct joint tolerance limits for the outputs of interest, one considers the lower confidence limit for the probability of "complying with the safety limits for **all** outputs", i.e. "meeting the regulatory acceptance criteria". Basis is that both of the following statements are equivalent:

1. The Wilks' (probability a=95% and confidence b=95%) limit for the results is below the regulatory acceptance limit.
2. The **lower b=95% confidence limit** for the probability that the value of the result stays below the regulatory acceptance limit is greater or equal a=95%.

The regulatory acceptance limits are incorporated into the probabilistic statements. It turns out that (1) in the one-dimensional case, i.e. for a single output parameter, is this concept equivalent to the one-sided upper tolerance limit concept, and (2) the necessary number of model runs is also the same in the general case, i.e. independent of the number of outputs or criteria involved and of the type of interrelationships between these outputs or criteria. Therefore, the number of necessary model runs is the same as in the one-dimensional tolerance limit case, even if several output parameters are involved. In the one-dimensional case the lower 95%-confidence interval for the probability of "complying with the regulatory limit" corresponds to the two step procedure: (1) compute the tolerance limit as usual and (2) compare this tolerance limit with the given regulatory limit. In other words: The statement "there is a 95% confidence that the probability of "complying with the regulatory limit x_{reg} exceeds 95%" is equivalent to the statement "the computed 95%/ 95% tolerance limit x_{TL} lies below the regulatory limit x_{reg}". In the multi-dimensional case there is no such direct correspondence or equivalence.

The principal advantage of the confidence interval or limit (or "sign-test") approach seems to be that it can directly be used in the multi-dimensional case, i.e. multiple output or several output variables, too, while the multidimensional extensions of the tolerance limit approach suffer from (1) not being unique because the runs with the highest value for checking the first limit has to be eliminated for comparison with the next limit, and so on, and (2) require substantially increased calculation runs, (3) are in most cases not necessary since functions of several variables can be reduced to the one-dimensional case.

Much more influence on the uncertainty range of computational results has the specified input uncertainty ranges. Less important is the distribution of these input uncertainties. Therefore, high requirements are on the specification and the justification for these ranges. Investigations are underway to transform data measured in experiments and post test calculations into thermal-hydraulic model parameters with uncertainties. Care must be taken to select suitable experimental and analytical information to specify uncertainty distributions. The selection of suitable experiments is important for the UMAE/ CIAU method as well.

EXAMPLE OF APPLICATION TO SAFETY ANALYSIS OF A REACTOR PLANT

An uncertainty analysis was performed for a double ended cold leg offset shear break design basis accident of a German PWR of 1300 MW electric power using the GRS method [10]. Figure 2 shows at any point of time, at least 95% of the combined influence of all considered uncertainties on the calculated clad temperatures is below the presented uncertainty limit (one-sided tolerance limit), at a confidence level of at least 95%. A "conservative" calculation result is shown for comparison, applying the best estimate code ATHLET with default values of the models, and conservative values for the initial and boundary conditions reactor power, decay heat, gap width of fuel rods between fuel and clad, fuel pellet thermal conductivity, and temperature of accumulator water. All these conservative values were also included in the distributions of the input parameters for the uncertainty analysis. The maximum clad temperature does not bound the 95%/ 95% one-sided tolerance limit of the uncertainty analysis over the whole transient time.

The "conservative" calculation is representative for the use of best estimate computer codes plus conservative initial and boundary conditions. Such an analysis is accepted in the licensing procedure of several countries, but not in the USA. The uncertainty of code models is not taken into account by this approach. It is claimed that the conservative initial and boundary conditions bound all model uncertainties. That is obviously not the case for the whole transient in the present example.

An uncertainty analysis quantifies uncertain initial and boundary conditions as well as model uncertainties. The peak clad temperatures, however, are bounded due to cumulating conservative values of the highly sensitive parameters fuel gap width and fuel thermal conductivity. It is obvious that the results are dependent on the extent of conservatism implemented in the conservative calculations. Therefore, the US Code of Federal Regulation [1] requires that "uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated" when a best-estimate computer code is used for the analysis.

ONGOING RESEARCH

Ongoing research is listed here in order to complete activities going on in the area of uncertainty analysis in safety assessment. A status has been described for example within the 6th Framework Program NURESIM (Nuclear Reactor Simulation) of the European Union (EU) [24]:

- A Global Adjoint Sensitivity Analysis Procedure (GASAP) is proposed by Cacuci but not yet tested (status October 2009), and therefore not ready for application
- According to the authors of the NURESIM final report it "appears to be a promising avenue combining GASAP with global statistical methods … to achieve a unified methodology for performing efficiently and accurately, global sensitivity and uncertainty analyses for large-scale systems"
- Two main applicable approaches are discussed, i.e. the "propagation of code input uncertainty" and "propagation of code output errors". "These approaches are pursued by two reference methods ready for application, i.e.
  - the GRS method and
  - the CIAU."

Follow-on research work is proposed in the EU NURISP (Nuclear Reactor Integrated Simulation Project), Sub-Project 4 within the 7th Framework Program which started in 2009 and will end in 2011. It consists mainly in working on Adjoint Sensitivity methods to overcome their limitation to local derivatives, whereas an uncertainty analysis should cover a range of parameter uncertainty. The Adjoint Sensitivity Method calculates the local partial derivatives of the code output variable of interest with respect to each of the code input variables.

The current program consists of the following working steps:

- Development of local adjoint-functions-based deterministic and statistical model/ modules, including software modules based on stochastic finite element methods („Karhunen-Loeve expansion" and „polynomial chaos expansion").
- Development of adjoint-functions-based model/ modules for global sensitivity and uncertainty analysis.
- Development of software modules based on „RaFu (Random-Fuzzy)"-method for combining probabilities and possibilities.
- Development of new hybrid methods and software modules by combining adjoint and statistical sensitivity and uncertainty analysis methods.
- Development of formal procedures for "industry"-wide standards of model validation, sensitivity and uncertainty quantification.

More promising for practical applications is a concept developed by University of Pisa and GRS in a Co-ordinated Research Program of the IAEA. Uncertainties of results of various accident events would be determined for one computer code using the GRS method. These values are to be stored in tables, like in the CIAU method, and would be ready for application in accident analyses. Users of that combined GRS-CIAU method would not need to identify and to determine the input uncertainties with their ranges and distributions individually any more. Filling these tables would be a high amount of work, and international contributions in another future international program would be worthwhile.

Research on the GRS method is going on in GRS with regard to separate treatment of epistemic and aleatory uncertainties to avoid a separate two-stage treatment, and to develop a less time-consuming one-step approach. Epistemic uncertainty is lack of precise knowledge. Aleatory uncertainty comes from unpredictable random performance of the system and its components, like random failure of equipment, as well as from random values of plant parameters. This is especially important in the frame of new activities to combine probabilistic and deterministic approaches, e.g. in the frame of determining safety margins. Other GRS activities are with regard to improvements of sensitivity measures for groups of input uncertainties, for example to determine the influence of plant or experimental uncertainties and code model uncertainties. Another area is the determination of significance levels of sensitivity measures.

CONCLUSIONS

The safety demonstration method "uncertainty analysis" is becoming common practice world-wide, mainly based on ordered statistics. Basis for applications of statistical uncertainty evaluation methods is the development of the GRS-method.

Several activities are carried out on an international world-wide level, like in OECD and IAEA. Comparison of applications of existing uncertainty methods have been performed in the frame of OECD/ CSNI Programs. Differences were observed in the results of uncertainty analysis to the same task. These differences are sometimes reason for arguing about the suitability of the applied methods. Differences of results may come from different methods. When statistical methods are used, differences may be due to different input uncertainties, their ranges and distributions. However, differences are already seen in the basic or reference calculations.

When a conservative method is used, it is claimed that all uncertainties which are considered by an uncertainty analysis are bounded by conservative assumptions. That may only be right when conservative code models are used in addition to conservative initial and boundary conditions. In many countries, however, a best estimate code plus conservative initial and boundary conditions are accepted for conservative analysis in licensing. Differences in calculation results of best estimate and even conservative codes would also be seen, comparing results of different users of the same computer code due to different nodalisations and code options the user is selecting. That was observed in all International Standard Problems where applicants calculated the same experiment or a reactor event. The main reason is that the user of a computer code has a big influence in applying a code. A user effect can also be seen in applications of uncertainty methods.

Another international discussion took place about the number of calculations to be performed using ordered statistics methods. That issue was mainly brought up when more than one acceptance criterion has to be met. However, much more influence on the results is by the specification of uncertainty ranges of these input parameters. Therefore, high requirements are on their specification and the justification for these ranges.

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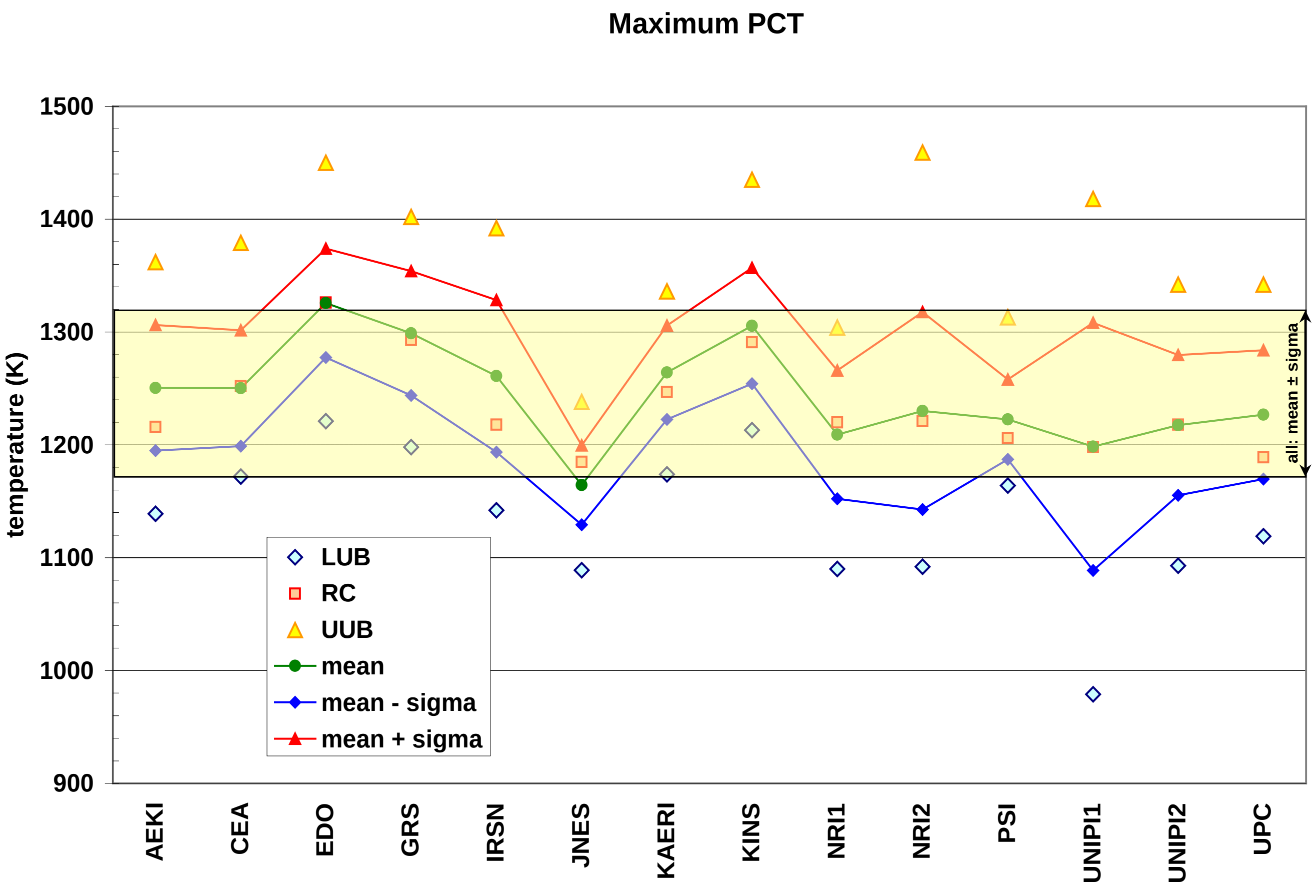

FIGURE 1: COMPARISON OF CALCULATED UNCERTAINTY RANGE FOR MAXIMUM PCT OF ZION NUCLEAR POWER PLANT (MEAN VALUE AND STANDARD DEVIATION) BY PARTICIPANTS OF THE OECD BEMUSE PROGRAM; LUB = LOWER UNCERTAINTY BOUND, UUB = UPPER UNCERTAINTY BOUND, RC = REFERENCE CALCULATION.

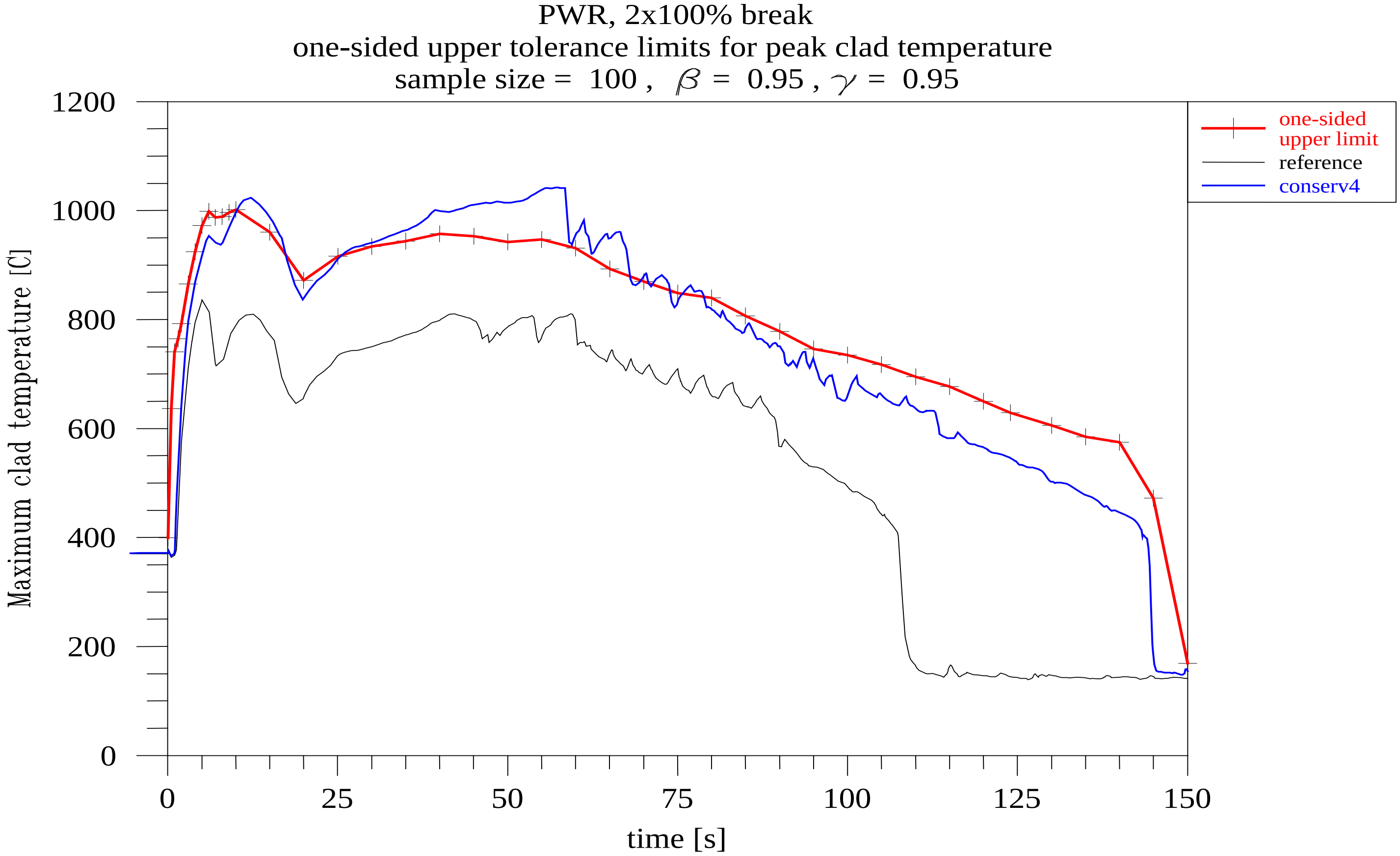

FIGURE 2: CALCULATED ONE-SIDED 95%/95% UNCERTAINTY LIMIT AND BEST ESTIMATE REFERENCE CALCULATION COMPARED WITH A "CONSERVATIVE" CALCULATION OF ROD CLAD TEMPERATURE FOR A REFERENCE REACTOR DURING A POSTULATED DOUBLE ENDED OFFSET SHEAR COLD LEG BREAK.