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Research Article

Regulatory Scenario for the Acceptance of Uncertainty Analysis Methodologies for the LB-LOCA and the Brazilian Approach

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The task of regulatory body staff reviewing and assessing a realistic large break loss-of-coolant accident evaluation model is discussed, facing the actual regulatory licensing environment related to the acceptance of the analysis of emergency core cooling system performance. Especially, focus is directed to the question of how to fulfill the requirement of quantifying the uncertainty in the calculated results when they are compared to the acceptance criteria for this system. As it is recognized that the regulation governing the loss-of-coolant accident analyses was originally developed by the United States Nuclear Regulatory Commission, a description of its evolution is presented. When using a realistic evaluation model to analyze the loss-of-coolant accident, different approaches have been used in the licensing arena. The Brazilian regulatory body has concluded that, in the current environment, the independent regulatory calculation is recognized as a relevant support for the staff decision within the licensing framework of a realistic analysis.

1. INTRODUCTION

The objective of this paper is to discuss the regulatory licensing environment related to the acceptance of the analysis of emergency core cooling system (ECCS) performance in light water reactors when using a realistic or best-estimate evaluation model. The focus is directed to the question of how to meet the requirement of quantifying the uncertainty in the calculated results when they are compared to the acceptance criteria for this system.

It also included the experience of the Brazilian nuclear regulatory body (CNEN) reviewing and assessing the Angra 2 nuclear power plant (NPP) large-break loss-of-coolant accident (LB-LOCA) analysis, submitted for licensing with a realistic evaluation methodology.

2. REGULATING THE USE OF BE + U

The United States Nuclear Regulatory Commission (US-NRC) emergency core cooling systems acceptance criteria, issued in 1974 [1], is recognized as a highly conservative approach due to limitations in knowledge at that time. This relevant aspect was identified and dealt with by the nuclear community through a huge effort in the reactor-safety research area. For additional details, see [2–6].

In 1983, based on experimental programs results, the ability of advanced computer codes to predict the behavior during an LOCA was demonstrated, and the conservatism in Appendix K could be quantitatively estimated. Because of this, through the release of SECY-83-472 [7], the NRC adopted an interim approach for evaluation models retaining the features of Appendix K which were recognized as requirements but allowing the use of best estimate methods in models and correlations. Even still conservative, this approach was the first step on licensing decision making based on realistic calculations.

On September 16, 1988, the NRC amended the requirements of 10 CFR 50.46 [8] reflecting the improved understanding of the thermal-hydraulic phenomena occurring during the loss-of-coolant accidents, obtained by the results of extensive research programs sponsored by the NRC and the nuclear industry. In Brazil, CNEN adopted this revision

which allows, as an option, the use of realistic evaluation models to calculate the performance of the emergency core cooling system. In such cases, the LOCA analysis will fulfill the requirement of identifying and evaluating the uncertainty in the analysis methods and inputs, and this uncertainty must be considered when comparing the calculated results with the acceptance criteria so that there is a high probability that the criteria will not be exceeded.

This revision of 10 CFR 50.46 allows licensees or applicants to use either the conservative evaluation model defined in Appendix K, with its conservative analysis methods, or a realistic evaluation model (best-estimate plus uncertainty analysis methods). The Regulatory Guide 1.157 [9] describes acceptable models, correlations, data, model evaluation procedures, and methods for meeting the specific requirements for a realistic calculation of ECCS performance during a LOCA.

Despite of that, there is still a lack of an established set of specific regulatory requirements and guidance applied to the acceptance of the uncertainty calculation related to the results of a realistic evaluation model used to analyze the LOCA. On January 11, 2001, the Advisory Committee on Reactor Safeguard (ACRS) of USNRC addressed the question of how the perceived weaknesses of the thermal-hydraulic codes may affect the regulatory role, and already emphasized in a Letter Report [10], "We perceive a need for the staff to be more specific about what are acceptable methods of deriving and expressing the uncertainties in codes and how these methods are to be used in the regulatory context".

More recently, NRC has issued section 15.0.2 of the Standard Review Plan [11] describing the review process and acceptance criteria for analytical models and computer codes used to analyze the accident and transient behavior, including methods to estimate the uncertainty in best-estimate LOCA calculation. Additionally, guidance to the industry was issued, set forth in Regulatory Guide 1.203 [12]. Despite of that, as it has been pointed out by ACRS in its January 11, 2001 Letter Report related to Regulatory Guide 1.157, these new regulatory guidance documents remain very qualitative and leaves considerable latitude in interpretation.

In parallel, NRC has been conducted research, together with industry, related to the acceptance criteria for ECCS. As an example, it should be mentioned that the ongoing development of a performance-based option for the embrittlement criteria in 50.46(b) [13–15], and also the proposed rule for a voluntary alternative to 10 CFR 50.46, related to the definition of LOCA break sizes [16].

In the United States, the first NRC approved best-estimate LOCA methodology was the Westinghouse methodology [17], patterned after the Code Scaling, Applicability, and Uncertainty evaluation methodology (CSAU), and uses response surfaces to estimate PCT uncertainty distribution with the 95th percentile PCT determined from a Monte Carlo sampling and accepted as the licensing basis PCT. In 1999, it was extended to other plants design (AP600 and 2-loops plants with upper plenum injection). By 2000, 14 plants in the United States had Westinghouse BELOCA methodology as a licensing basis and it was also used for Ringhals unit 2 in Sweden [18].

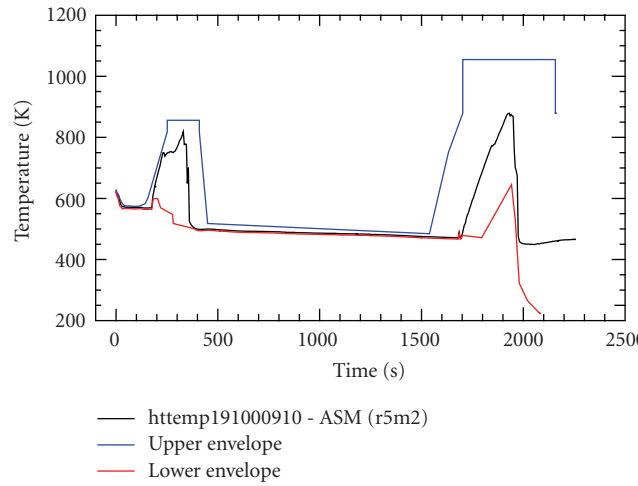

Figure 1: JONATER application of UMAE to Angra 1 NPP SB-LOCA: uncertainty bands for the PCT.

Framatome ANP has submitted its realistic LB-LOCA methodology and got NRC approval in 2003 [19]. It follows CSAU approach but was the first to use a nonparametric order statistic method, eliminating the need for response surfaces. By 2006, there were seven completed realistic LB-LOCA analyses with 3-loop and 4-loop Westinghouse and Combustion Engineering pressurized water reactors [20].

By 2004, Westinghouse updated its methodology to use nonparametric order statistic, and an advanced statistical treatment of uncertainty method (ASTRUM) [21] was approved for licensing. In US, by 2006, 24 plants are licensed or analyzed with Westinghouse 1996 and 1999 BELOCA methodologies and 10 plants are analyzed or being analyzed with ASTRUM [18].

It is worthwhile to mention the ongoing issue at the regulatory arena with the use of nonparametric order statistics methodology to demonstrate that the criteria in 10 CFR 50.46(b) are satisfied. The number of ECCS performance-evaluation model runs accepted to demonstrate a probability that the criteria will not be exceeded is different in two similar realistic LB-LOCA methodologies approved by the NRC [19, 21]. Such difference is due to the approach used to demonstrate the simultaneous satisfaction of the first three criteria in 10 CFR 50.46(b), peak cladding temperature, peak local oxidation, and corewide oxidation. There are still undergoing discussions on this philosophical issue [22–26].

In Germany, the use of best-estimate codes is allowed, in combination with conservative initial and boundary conditions, and efforts are being conducted to include uncertainty evaluation in the regulation with a revision in the German nuclear regulation. There is also a recommendation of the Reactor Safety Commission to perform LOCA licensing analysis [27].

In Canada, the Canadian Nuclear Safety Commission recently conducted a research program that resulted in a guide for safety assessment and applications of best-estimate analysis and uncertainty methodology [28].

3. BRAZILIAN REGULATORY EXPERIENCE

Angra 2 NPP is a 4-loop Siemens design 1300 MWe pressurized water reactor that started commercial operation in 2001. The best-estimate LOCA approach was formally adopted by the utility Eletronuclear (ETN) in 1994. By 1998, when the realistic LB-LOCA analysis was submitted, based on CSAU methodology to evaluate the uncertainty, there were only few applications of realistic evaluation models in the licensing arena.

Aiming at performing a consistent safety review and assessment of this analysis, the Brazilian regulatory body trained its staff and relied upon two international consultants, the German institute GRS (Gesellschaft fur Anlagen und Reaktorsicherheit) and the University of Pisa.

The cooperation with many international institutions involved in accident-analysis research provided a relevant technical background for the regulatory staff. In the same time, the national thermal-hydraulic journey (JONATER), coordinated by CNEN, has promoted the integration of seven institutions (regulatory body, research institutes, and utility) of the Brazilian nuclear sector. One result of JONATER was the use of an uncertainty methodology applied as an exercise for Angra 1 NPP, a Westinghouse 630 MWe 2-loop pressurized water reactor. The uncertainty bands were estimated with the UMAE [29] method for the results of the small-break LOCA simulated with the Relap5Mod2 code, as it is shown in Figure 1 [30].

UMAE is an uncertainty methodology based on accuracy extrapolation resulting from a comparison between code results and relevant experimental data obtained in experimental facilities. These facilities were simulated, for the chosen transient scenario, with an established nodalization that will be the basis for the nodalization adopted in the plant calculation. The extrapolated accuracy is superimposed directly to the results of the plant calculation. Uncertainty bands are constituted by a set of "punctual" error bands in the x-y plane (where x is the time t and y is Y_C quantity). Each value Y_C at a time t can be characterized by an error DY_C in the "y" direction and by an error Dt in the "x" direction. The total uncertainty is the superimposition of these two errors.

As the estimation of Angra 1 small-break LOCA uncertainty bands was an exercise for the application of an uncertainty methodology, for the accuracy calculation, only the large scale test facility (LSTF) database was considered (experimental and Relap5/Mod2 results for the SB-CL-21 test). It is important to mention that the accuracy should be obtained from more tests to avoid some poor accuracy that eventually can result for some specific parameter. For instance, code simulation of the LSTF experiment yielded a result for the heater rod temperature and time of its occurrence far from the verified experimental value. Therefore, the lower uncertainty band at the end of the transient for the peak cladding temperature shows no physical results due to the limited number of experimental data used.

The Angra 2 LB-LOCA analysis presented in the final safety-analysis report was reviewed by CNEN staff taking into account the two independent reviews performed by the international consultants. As a result, a preliminary safety-evaluation report (SER) requested additional information (RAI), with a total of 27 questions to the applicant, each one is classified according to their significance to safety [31].

Table 1 lists the main steps in the review and assessment process of Angra 2 NPP LB-LOCA analysis.

The Siemens uncertainty methodology applied to Angra 2 followed, essentially, the CSAU approach (Phenomena Identification Ranking Table, code capabilities for accident scenario) and used Monte Carlo calculations with response surface. The treatment of the uncertainties is performed separately from three basic categories: code uncertainties (statistical quantification of difference between calculated and measured PCT), plant parameters uncertainties (statistical variations), and fuel parameters uncertainties (statistical variations). Some additional parameters related to uncertainties have been required to be run at combined worst-case conditions. These parameters are break area and location, axial core power distribution, worst-case single failure and repair assumption, loss of offsite power, and reactor kinetics.

This uncertainty analysis is such that the 95% probability PCT was generated by using Monte Carlo to combine uncertainties from the three sources. The two other criteria (maximum cladding oxidation and hydrogen generation) were calculated considering conservative assumptions.

The number of data points, used to determine code accuracy through the quantification of the differences between calculated and measured results for LOFT and CCTF experiments, was one example of RAI from the preliminary SER. It was further required from the applicant to verify the implications of considering additional relevant experimental data into code integral uncertainties. Additionally, the applicant presented code uncertainty quantification with more experimental data.

After the issuance of the preliminary SER, the importance of an independent regulatory calculation was recognized. Together with CNEN staff, the University of Pisa performed independent calculation [32, 33]. Based on its conclusions, three requests for additional information were issued to the applicant, mainly related to plant modelling, which has to be consistent with those used for the validation calculations.

As future applications, the Brazilian regulatory body has already been informed by the utility ETN of its intention to uprate 6% the Angra 2 power together with a change in the fuel design, replacing it to a high thermal performance fuel with M5 fuel cladding. This will require the reanalysis of the LB-LOCA with uncertainty quantification.

Furthermore, for Angra 1 NPP steam-generators replacement, the utility will submit a realistic evaluation model for the LB-LOCA, using the Westinghouse methodology that encompasses the WCOBRA/TRAC code with the ASTRUM methodology for uncertainty calculation. Additionally, the power will be uprated 5% and a new fuel design will be used (16 next-generation fuel, developed jointly by Westinghouse, Korea Nuclear Fuel (KNFC), and Indústrias Nucleares do Brasil (INB)).

Table 1: Angra 2 NPP LB-LOCA Review.

Activity	Date
Submission of FSAR with the LOCA analysis	1998
CNEN's Preliminary Safety Assessment	1998
GRS Expert Mission-FSAR LBLOCA Analysis and Uncertainty Method revision	1998
IAEA Expert Mission/Pisa Uni-FSAR LBLOCA Analysis and Uncertainty Method revision	1999
Issuance of CNEN Preliminary SER with 27 RAI	1999
GRS Expert Follow-up Mission	1999
Answers to the request of additional information	1999
Licensing meeting: Utility, Siemens, CNEN, GRS and Pisa University	1999
Submission of Generic Thermal Mechanics Analysis of Fuel Failures after a LOCA	1999
Submission of Angra 2 Specific Thermal Mechanics Analysis of Fuel Failures after a LOCA	1999
Emission of CNEN SER about ECCS Technical Specification	2000
Pisa University Technical Consultancy to CNEN-Regulatory audit analysis	2001
Issuance of CNEN SER: additional RAI on core nodalization and uncertainty quantification	2001
Independent LB-LOCA calculation used to check the request of a temporary 6% power uprate	2002

4. REGULATORY INDEPENDENT ANGRA 2 LB-LOCA ANALYSIS DESCRIPTION

The independent calculation included the LB-LOCA calculation with Relap5/Mod3.2.2 Gamma code and the uncertainty evaluation with the CIAU method (code with capability of internal assessment of uncertainty) [34].

In this application, the CIAU method used UMAE methodology for uncertainty quantification that is based upon propagation of code output error and does not rely on statistics. The inaccuracies are obtained by experimental calculation comparison and are extrapolated to get uncertainty. The database for accuracy extrapolation was derived from 32 experimental transients that were calculated by Pisa University with Relap5/Mod3.2.2 Gamma code.

The independent LB-LOCA calculation activities were planned with the objective to consider the steps presented in a best-estimate analysis: a qualified nodalization development (steady-state level and on-transient level), transient reference-case calculation, uncertainty evaluation, and comparison between the results obtained in the sensitivity studies and in the uncertainty analysis.

A "fictitious" 3D nodalization of the reactor pressure vessel was adopted considering the experience in the analysis of the upper plenum test facility experiments [35]. Two main nodalizations were established at the beginning of the studies, characterized by:

(i) nonuniform upper plenum behavior, pursuing the nodalization strategy of the utility ETN in the FSAR analysis, top-down flow allowed only in the determined breakthrough channels [36];

(ii) uniform upper plenum behavior with top-down flow allowed in all channels except in the hot assembly, with the worst conditions for core cooling inside the hydraulic hot assembly, by "hydraulically separating" the hot fuel assembly from the average core region.

After defining a reference calculation and performing the sensitivity study, the reference-case nodalization chosen was the one without cross-flow simulation between the hot fuel assembly and the rest of the core (denominated tr12), that might bring undue conservatism in the results. The one considering this cross flow (denominated a2n04x) could be the reference case if experimental data was available to establish the flow energy-loss coefficients. Therefore, for the a2n04x run, these coefficients were established through engineering judgment without an experimental basis. The use of S-RELAP5 code in the Angra 2 FSAR LB-LOCA analysis considers implicit this cross flow through the full two-dimensional treatment added to the hydrodynamic field equations.

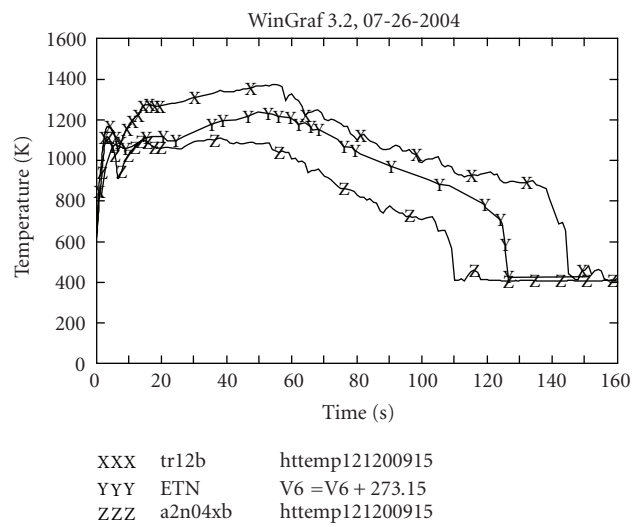

Figure 2: Cladding temperature of the hot rod.

Figure 2 shows a comparison of the reference calculation result to FSAR result for the peak cladding temperature (PCT) for the "base case". In the FSAR analysis, this "base case" is defined in the adopted ETN methodology as the nominal condition for the uncertainty analysis. This uncertainty analysis is such that the 95% probability PCT was

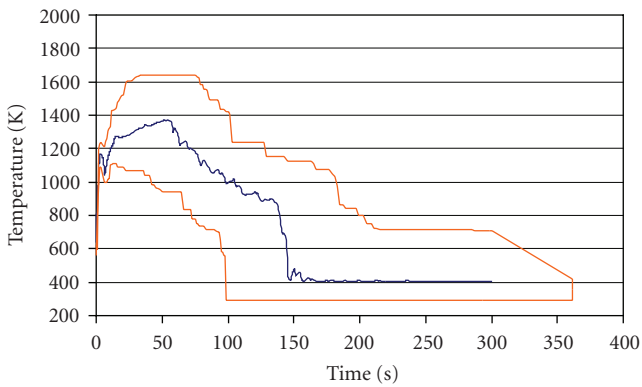

Figure 3: Uncertainty bands for rod surface temperature at 2/3 of the core height-CIAU result.

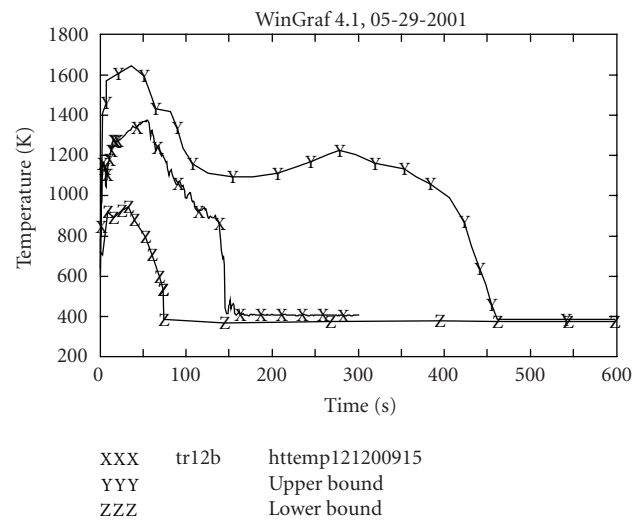

Figure 4: Angra 2 NPP LB-LOCA sensitivity study: upper and lower bounds from the rod surface temperature. Envelope uncertainty evaluation.

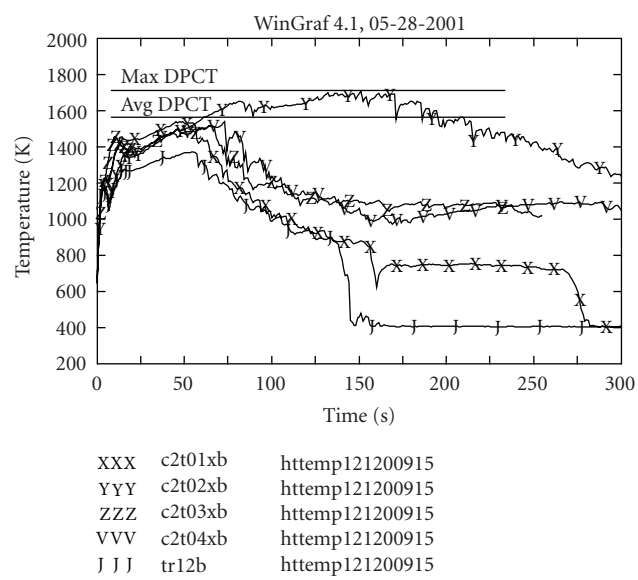

Figure 5: Angra 2 NPP LB-LOCA sensitivity study, achievement of a deterministic value for ΔPCT. Labels XXX through VVV representing code runs based on combination of three variations of input parameters.

generated by using Monte Carlo to combine uncertainties from the three sources. The "base case" is the reference case for the determination of the calculation-design matrix used to generate data for fitting the response surfaces. Also, the "base case" is the reference case where the effects of the plant uncertainties are determined.

The comparison of the PCT from the "base case" and the "reference calculation" indicates a discrepancy, with a higher value observed in independent calculation result. In the case of "reference calculation", it is shown that the removal of conservatism of assuming no cross flow to the hot channel substantially lowers the reported value. This outcome confirms the importance of assessing, by using experimental data, the cross flow to the hot channel if this is considered.

In the independent regulatory calculation, automatic uncertainty bands for primary-system pressure, mass inventory, and rod surface temperature at 2/3 of the core active height are generated by the CIAU method and constitute the results of the application. Figure 3 shows the result for PCT.

The number of experiments, which were used to derive code uncertainty from CIAU, is limited. Therefore, a sensitivity study has been performed to confirm the results obtained from this methodology. Additional objective was to confirm that the impact of an assigned input parameter upon the results is dependent on the nodalization.

A comprehensive-sensitivity study has been carried out including two series of calculations. Starting from the two main nodalizations, single parameters are varied in each code run. Six groups of input parameters are distinguished: "fuel", "nodalization", "loop hydraulics", "PSA and ECCS", "neutronics", and "others". The number of performed runs was 112.

Thefirstseries aims at confirming the influence of selected input parameters upon the LB-LOCA predicted scenario, and showing the importance of nodalization upon the same prediction when an assigned input parameter is varied. Code runs with single change of input parameters and with realistic variation ranges were used for the envelope uncertainty evaluation. Examples of input parameters varied, at one time, in the code run: fuel (gap thickness, UO2 conductivity, gap conductance), loop hydraulics (critical flow model, spacer grid modelling, reactor pressure-vessel bypass flow), nodalization (upper-plenum pressure drop, counter current-flow limitation in the core), PSA and ECCS (loss of offsite power delay, components actuation), and neutronics (moderator coefficient, decay power). The result is shown in Figure 4 where the envelope of all the considered calculations is reported.

The second series aims at determining boundary values for PCT. Three input parameters, chosen among those considered in the first series of calculations, are selected and varied simultaneously in each run. Examples of chosen

parameters are UO2 conductivity, break-discharge coefficient, ECCS components actuation, decay power, and gap conductance. The ranges of variations are maximized. These code runs are adopted for the deterministic evaluation of the uncertainty (see Figure 5).

The parameter ΔPCT is defined as the difference between the PCT of the reference calculation and the PCT obtained from the generic sensitivity run. The dispersion of results for ΔPCT obtained from the first series of code runs provides an overall picture of the influence of nodalization upon predictions, confirming the importance of the nodalization upon the predicted scenario.

The following valuable results were obtained.

(i) The upper and lower uncertainty bands from the envelope uncertainty evaluation in Figure 4 can be compared with the CIAU uncertainty bands in Figure 3. Therefore, the uncertainty results obtained by CIAU are supported by the outcome of the sensitivity study.

(ii) The uncertainty ranges predicted by CIAU, resulting from the sensitivity study and the ones reported in the FSAR, are comparable.

The adopted noding scheme, that is, the nodalization, has been found as the critical issue of the study. The nodalization features affect the prediction of the safety relevant parameters, the interpretation of the performed "sensitivity" runs, and the use of the outcomes from the uncertainty method. Namely, the application of a 1D designed assessed code, having at the basis a fictitious 3D model of the vessel, requires a number of engineering choices. These choices have been proven to impact noticeably the results, and must be adequately supported by a suitable experimental evidence.

Results from a best-estimate code prediction are largely affected by the nodalization features. Therefore, the full demonstration of the nodalization quality at the "steady state" and at the "on-transient" level is needed to derive meaningful conclusions about the safety performance of the concerned NPP. Considering Angra 2 features, basically, the hot leg injection, a decisive importance is revealed by the upper plenum and core outlet modeling.

5. CONCLUSIONS

As described in the previous sections, when using a realistic evaluation model to analyze the LOCA, different approaches have been used in the licensing arena to demonstrate the fulfillment of the ECCS acceptance criteria.

Besides the different approaches, the regulators are aware of the development in the uncertainty methodologies and, therefore, further actions should be required even after a methodology has been accepted.

The Brazilian regulatory body is monitoring these activities and it has concluded that, in the current environment, the independent regulatory calculation is recognized once again as a relevant support for the staff decision within the licensing framework of a realistic LB-LOCA analysis.

In the case of Angra 2 LB-LOCA, the independent calculation complemented, on a quantitative basis, the task of reviewing and assessing, and allowed to check the completeness and consistency of the submitted accident analysis. The use of an uncertainty methodology (CIAU) that has a different approach compared to the designer approach (Siemens) contributed to the understanding of the validity limits of the results submitted by the licensee within the FSAR. Conclusions are provided in relation to the acceptability of the actual safety margins of the Angra 2 NPP.

In the case of Angra 1 LB-LOCA reanalysis for the steam-generators replacement, to be submitted with Westinghouse methodology, the ASTRUM methodology uses a nonparametric order-statistics methodology to demonstrate that the criteria in 10 CFR 50.46(b) are satisfied.

The different approaches observed in the nuclear-power plants in Brazil increase the staff effort to deal with the licensing process. For a small size regulatory body, this diversity of methods, to demonstrate the fulfillment of the ECCS acceptance criteria, indicates a challenge to be faced with technical support organizations providing worldwide recognized experts in the use of best-estimate tools to contribute in the review and assessment process.

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