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BEPU Approach in Licensing Framework, including 3D NK Applications

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1. Introduction

During the recent years, a world-wide renewed interest in the exploitation of nuclear energy for electricity production is seen among both the Western and the new industrializing Countries (e.g., China and India). As a result, 61 reactors are now under construction and more than 100 units are planned for the incoming decade. Such impressive development is totally based on Light and Heavy Water Reactor (LWR & HWR) technologies [1], on designs that are an evolution of the robust and reliable Nuclear Power Plants (NPP) designed and built during the seventies-eighties of the last century. At that time, the need to guarantee an high safety level on one side and on the other the limited computational capabilities and the scarce knowledge of some phenomena, drove the main nuclear safety authorities to establish extremely conservative rules. Nowadays, after that tremendous progress has been made into the computational power availability, models accuracy and the knowledge of relevant phenomena, there is the need to go toward more realistic safety analyses and to relax some levels of conservativeness without compromising the always elevated safety level of the nuclear industry.

The aim of this Chapter is to give an overview of the current trends in the licensing frameworks for a NPP. International best-practices are presented and discussed and sample applications derived from works of the San Piero a Grado Nuclear Research Group of the University of Pisa (GRNSPG/UNIP) on existing industrial facilities are also reported.

2. The licensing framework and the best international practices

Three internationally recognized fundamental safety objectives [2] constitute the basis from which the requirements for minimizing the risks associated to NPPs shall be derived. A general nuclear safety objective is stated as “to protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defenses against radiological hazards”. Two complementary safety objectives deal, respectively, with radiation protection and technical aspects.

The established terms for the technical safety objective are: “to take all reasonably measures to prevent accidents in nuclear installations and to mitigate their consequences should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any

radiological consequences would be minor and below prescribed limits, and to ensure that likelihood of accidents with serious radiological consequences is extremely low".

In a recent publication of the International Atomic Energy Agency (IAEA) [3] the safety objectives have been rephrased into one safety objective and ten safety principles.

To demonstrate that all applicable safety requirements are fulfilled by the design and the operation of a nuclear power plant, a systematic evaluation must be conducted throughout the lifetime of the installation. According to well established safety practices [4], this systematic assessment should follow two complementary paths (or twofold strategy): a comprehensive safety analysis of the plant, and a thorough evaluation of engineering factors embedded in the design and operation of the installation.

The comprehensive safety analysis shall address the dynamic response of the plant to a sufficiently broad spectrum of faults and accidents scenarios to demonstrate that states that could result in high radiation doses or radioactive releases are of very low probability of occurrence, and plant states with significant probability of occurrence have only minor or no potential radiological consequences. For performing plant safety analysis, methods of both deterministic and probabilistic analysis shall be applied [4].

The twofold safety assessment strategy shall be consolidated and documented in the so called Safety Analysis Report (SAR) which, according to recognized safety standard [5], must support the regulatory decision making process within the plant licensing framework. The requirements on SAR format and contents are dependent on country's regulatory regime, although some consolidated practices have been widely followed [6], [7].

The achievement of a high level of safety should be demonstrated primarily in a deterministic way. The deterministic approach typically considers a limited number of events for which conservative rules for system availability and parameter values are often applied.

Recently, with the development of code capabilities supported by experimental data basis, best estimate (BE) methods have also been applied within the design basis spectrum of events. In such situation, however, for licensing applications there is a need to address uncertainties in the calculations.

For the deterministic BE analysis this kind of uncertainty (epistemic) results among others from imperfect knowledge of the physical phenomena or of values of code model parameters. Instead, aleatory uncertainty, resulting from inherent randomness or stochastic variability, is by its very nature the subject of Probabilistic Safety Analysis (PSA) type of analysis.

Any BE analysis considering uncertainties (in short Best Estimate Plus Uncertainty, BEPU) which is applied in the field of licensing should be consistent with [8] and [9]. More details on the international framework on BEPU analysis in licensing please could be found in [10] and [11].

3. The BEPU approach

GRNSPG/UNIPi developed a procedure for the consistent application of BEPU in deterministic safety analysis (Fig. 1 shows a simplified flowchart), which has been applied in the licensing process of Atucha 2 NPP. The approach adheres to common practices of categorizing postulated initiating events (PIE) according to their frequency, and establishing more stringent acceptance criteria for more likely events. The following subsection provides an explanation on the terms, while the subsection after introduces a procedure for BEPU application.

The procedure follows well accepted design philosophy for NPPs, which recognizes the principle that plant states which could result in high radiation doses or radioactive releases

are of very low probability of occurrence, and plant states with significant probability of occurrence have only minor or no radiological consequences.

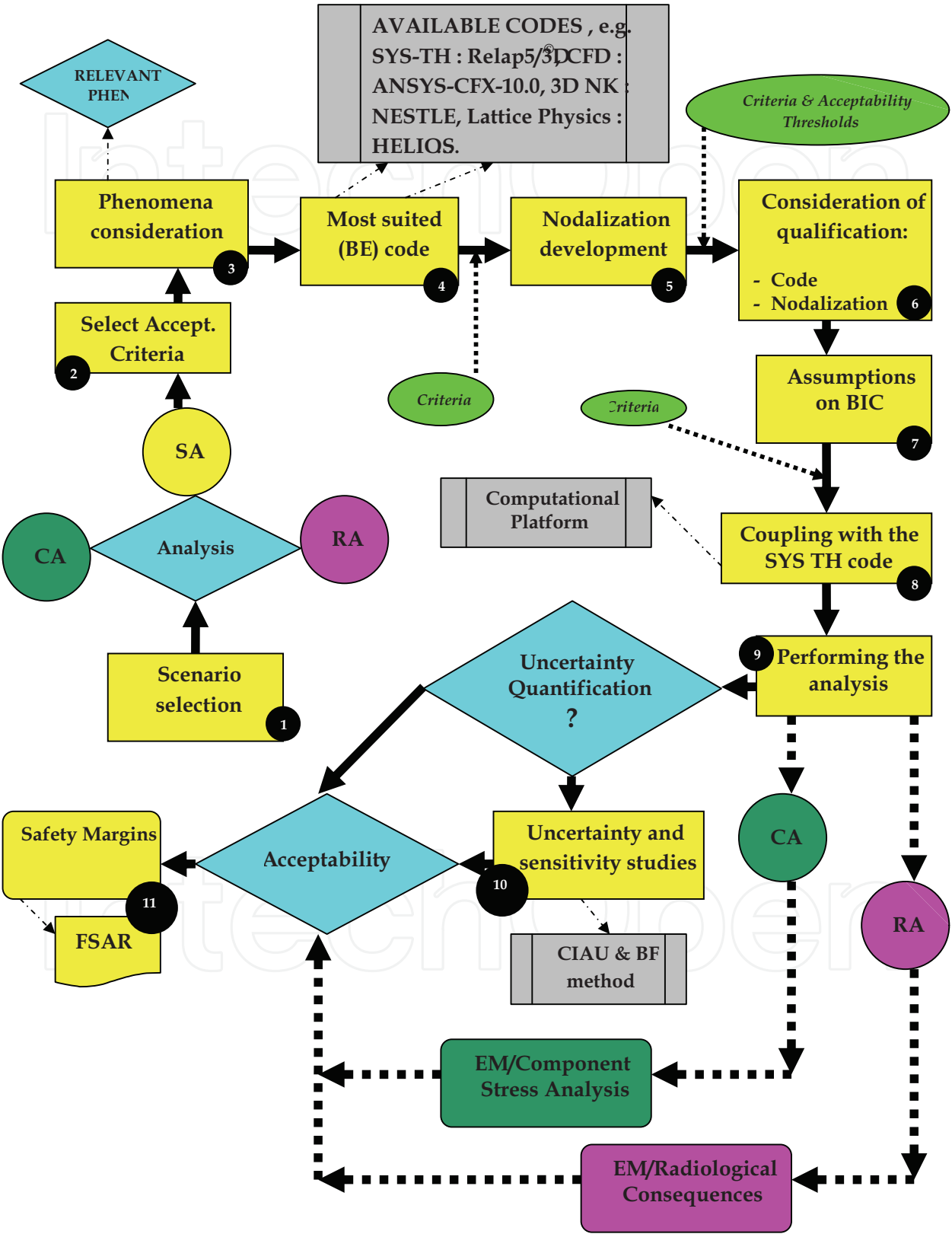


Fig. 1. General scheme of applying BEPU in accident analysis

3.1 The preliminary categorizing of events and acceptance criteria

Before the scenario selection for the analysis, a preliminary categorization of the events leading to such scenario and the acceptance criteria for the evaluation of the safety margins have to be fixed.

The event sequences postulated in the design of the plant are analyzed to demonstrate that in operational states, in and following a design basis accident and, to the extent practicable, on the occurrence of some selected accident conditions that are beyond the design basis accidents, the following three fundamental safety functions are performed:

- Safe shutdown and long term subcriticality
- Residual heat removal
- Limitation of radioactive releases.

PIE are grouped according to their anticipated probability of occurrence in anticipated operational occurrences (AOO), design basis accidents (DBA), beyond design basis accidents (BDBA) and severe accident (SA) (e.g., see Tab. 1). In the specific case of the Atucha-2 Final Safety Analysis Report (FSAR), an intermediate category or the selected beyond design basis accident (SBDBA) was introduced to address specific scenarios beyond design basis, including Double-Ended Guillotine Break (DEGB) Loss-of-Coolant-Accident (LOCA) and Anticipated Transient Without Scram (ATWS). Also in the Atucha-2 FSAR, the proposed BEPU analysis scheme was applied only to AOO, DBA and SBDA, while the remaining ones (BDBA and SA) were treated through the probabilistic safety analysis.

Range of frequency (1/reactor-year)	Characteri stic	Category	Common Terminology	Safety Consequences
10 ⁻² to 1 (Expected in life of the plant)	Expected	AOO	Anticipated transients, transients, frequent faults, incidents of moderated frequency	No additional fuel damage
10 ⁻⁴ to 10 ⁻² (Chance greater than 1% over the life of the plant)	Possible	DBA	Infrequent incidents, infrequent faults, limiting faults, emergency conditions	No radiological impact at all or no radiological impact outside exclusion area
10 ⁻⁶ to 10 ⁻⁴ (Chance less than 1% over the life of the plant)	Unlikely	BDBA	Faulted conditions	Radiological consequences outside the exclusion area but within limits
<10 ⁻⁶ (Very unlikely to occur)	Remote	SA	Faulted conditions	Emergency response needed

Table 1. Category of event based on its expected frequency of occurrence

Acceptance criteria are applied in the deterministic safety analysis, following some rules and methods which have been developed to introduce conservatisms in plant safety evaluations.

Acceptance criteria are directly or indirectly related to the barriers against releases of radioactive material. Current adopted values and rules for using acceptance criteria have been developed considering some decoupling techniques that cover the range from plant processes to environmental impact. The decoupling should ensure that if, for example, the fuel safety criteria are fulfilled during the accident, then the radiological releases are limited and acceptable provided that the criteria for the two other barriers are also fulfilled.

Safety criteria are mostly derived from the radiological reference values by applying several decoupling actions where, for some cases a phenomenon is substituted to the primitive one (decoupling phenomena), for some others, more restrictive values are imposed in order to be sure that the original requirement is satisfied (decoupling parameter). At each step, conservatisms are introduced that can be considered as margins for safety. Frequent events should have minor consequences and events that may result in severe consequences should be of very low probability. In this sense, the risk across the spectrum of AOO and DBA should be approximately constant. Acceptance criteria are derived for each category of event based on its expected frequency of occurrence, as shown in the Tab. 1 above.

3.2 Grouping the events

Generally, all selected scenarios are grouped in a classical families of events (around ten different families) where each family covers events with similar phenomena.

For the FSAR Chapter 15 analyses, and for each category of events, the results of the analyses are assessed in terms of the fulfillment of safety functions which are graded according to the expected frequencies of occurrences for the correspondent PIE.

To keep a consistently flat risk profile over the entire spectrum of AOO and DBA, the more frequent the event is, the less tolerable its consequences are. In this sense, acceptance criteria are selected for different event categories, for safety parameters as fuel and cladding temperatures, departure from nucleate boiling ratio (DNBR), primary circuit pressure, containment pressures, and total effective dose equivalent (TEDE).

3.3 Selection of the evaluation models & phenomena consideration

The BEPU approach takes credit of the concept of evaluation model (EM, see below), and comprising three separate possible modules depending on the application purposes:

- for the performance of safety system countermeasures (EM/SA);
- for the evaluation of radiological consequences (EM/RA);
- for the review of components structural design loadings (EM/CA).

A fundamental step in performing safety analysis is the selection of the EM. With EM is generally intended the calculation framework for evaluating the behavior of the reactor system during Chapter 15 analyzed events. EM could include one or more codes, analytical models or also calculation procedure and all other information for use in the target application.

To start analyzing typical events scenarios for the chapter 15 of an FSAR, EMs rely mostly on system thermal-hydraulic (SYS TH) codes (as for EM/SA) to solve the transport of fluid mass, momentum and energy throughout the reactor coolant systems. The extent and complexity of the physical modes needed to simulated plant behavior are strongly dependent of the reactor design and of the transient itself.

For some scenarios, or regarding some analysis purposes, the SYS TH code may, for example, be complemented by (or coupled with) a three-dimensional neutron kinetics code

or the reference model may need an expansion to include a detailed simulation of controls and limitation systems which play a relevant role for determining the plant response.

For the scope of the proposed approach for accident analyses, the complexity of the evaluation model may range from a simplified qualitative evaluation (EM/QA) to a complete combination of the three possible modules (EM/SA + EM/RA + EM/CA).

The two main aspects which have been considered for developing the evaluation model with the ability of adequately predict plant response to postulated initiating events are intrinsic plant features and event-related phenomena characteristics.

For the two modules EM/SA and EM/CA, the first set of requirements for the evaluation model is imposed by the design characteristics of the nuclear power plant, its systems and components. Requirements on the capability of simulating automatic systems are of particularly importance for AOO, in which control and limitation systems play a key role on the dynamic response of the plant. For evaluation of radiological consequences, the EM/RA module has demanded additional appropriate site-related features to be built in.

The third set of requirements is derived from the expected evolution of the main plant process variables and the associated physical phenomena. For the proposed approach, this is performed through the process of identifying the Phenomenological Windows (PhW) and the Relevant Thermal-hydraulic Aspects (RTA). The relevant timeframe for the event is divided into well defined intervals when the behavior of relevant safety parameters is representative of the physical phenomena.

3.4 Selection of boundary and initial conditions

Additionally to the computers codes and the selection of modeling options, the established procedures for treating the input and output information are also recognized as comprising key parts of the evaluation model. The adopted procedures to select initial and boundary conditions (BIC), which follows the original design safety philosophy, are of particular importance for supporting the regulatory acceptability of the results provided by the EM. As the foreseen use of an EM is for licensing purposes, it is necessary to evaluate the suitability of conservative assumptions or to adopt BE approaches with the quantification of uncertainties.

Suitability of conservatism should be understood as addressing the issue of “how conservative is conservative enough”. Alternatively, when a BE approach is adopted, then realistic assumptions will be input to BE models, conducting to realistic estimates for plant behavior. In these cases, licensing applications demand the quantification of uncertainties in the calculated results to ensure that safety margins are still available. For the scenarios where the conservative assumptions may provide enough safety margins, it can be derived by the analysts a criterion to determine the need for uncertainty calculations.

Typically, SBDBA and some DBA can involve quantification of uncertainties.

3.5 Selection of qualified tools

For the adequate simulation of the identified phenomena (step “3” of Fig. 1), computational tools have to be selected from those which have previous qualification using an appropriate experimental data base. Satisfactory qualification targets provide basis for acceptability of the postulated application (see later). As referenced in Fig. 1 (step “4”), “computational tools” expression comprises:

- BE computer codes

- qualified detailed nodalizations for the adopted codes
- established computational methods for uncertainty quantification
- computational platform for coupling and interfacing inputs and outputs of the selected codes

In the GRNSPG/UNIPI approach, with the full scope of application of BEPU quantification, a pre-requisite is the availability or the support of the most advanced-qualified computational tools available on the market.

Generally, for most event scenarios, the single purpose evaluation model EM/SA may be necessary and sufficient to be developed. In this sense, the availability and the application of qualified system thermal-hydraulic code and reliable uncertainty methodology (UM) is the minimum requirement.

Additionally, depending on the specific event scenario and on the purpose of the analysis, it is necessary the availability of calculation methods that are not embedded in the SYS TH code, as, e.g. for burst temperature, burst strain and flow blockage calculations. This may imply an evaluation model EM/CA composed by a fuel rod thermal-mechanical computer code. Another example is in transients where strong asymmetric neutron flux changes happen; this could require the adoption of a three-dimensional (3D) neutron kinetics (NK) code (see later).

3.6 Nodalization development & qualification

Nodalizations (i.e., codes input decks and plant schematization) should be developed according to predefined qualitative and quantitative acceptance criteria. Different methods and procedures are available, depending on the analyst choices and code types. Nevertheless, in a licensing process, it is fundamental to demonstrate to the Safety Authority the efficiency and the reliability of such qualification procedures. At GRNSPG/UNIPI, a suited set of criteria is proposed for SYS TH codes qualifications according to references [13] to [16].

A major issue in the use of mathematical models is constituted by the model capability to reproduce the plant or facility behaviour under steady-state and transient conditions. These aspects constitute two main checks for which acceptability criteria have to be defined and satisfied during the nodalization-qualification process. The first of them is related to the geometrical fidelity of the nodalization of the reference plant; the second one is related to the capability of the code nodalization to reproduce the expected transient scenario. A simplified scheme of a procedure for the qualification of a TH nodalization is depicted in Fig. 2. In the following, it has been assumed that the code has fulfilled the validation and qualification process and a “frozen” version of the code has been made available to the final user.

3.7 Couplings, including the Neutronics

As anticipated in Chapter 1.3.5, SYS TH codes have a leading role in the licensing analyses because of their capabilities to simulate with sufficient level of details the thermal-hydraulic phenomena of primary and secondary circuits of a LWR/HWR. In some specific transient analyses (that could belong to every category of the Tab. 1), it could be requested with a high level of detail the simulation of non-TH phenomena. In such cases, other BE codes have to be employed and coupled with robust and qualified procedures with the SYS-TH code [18]. A typical example is when a detailed core power reconstruction is needed, both during

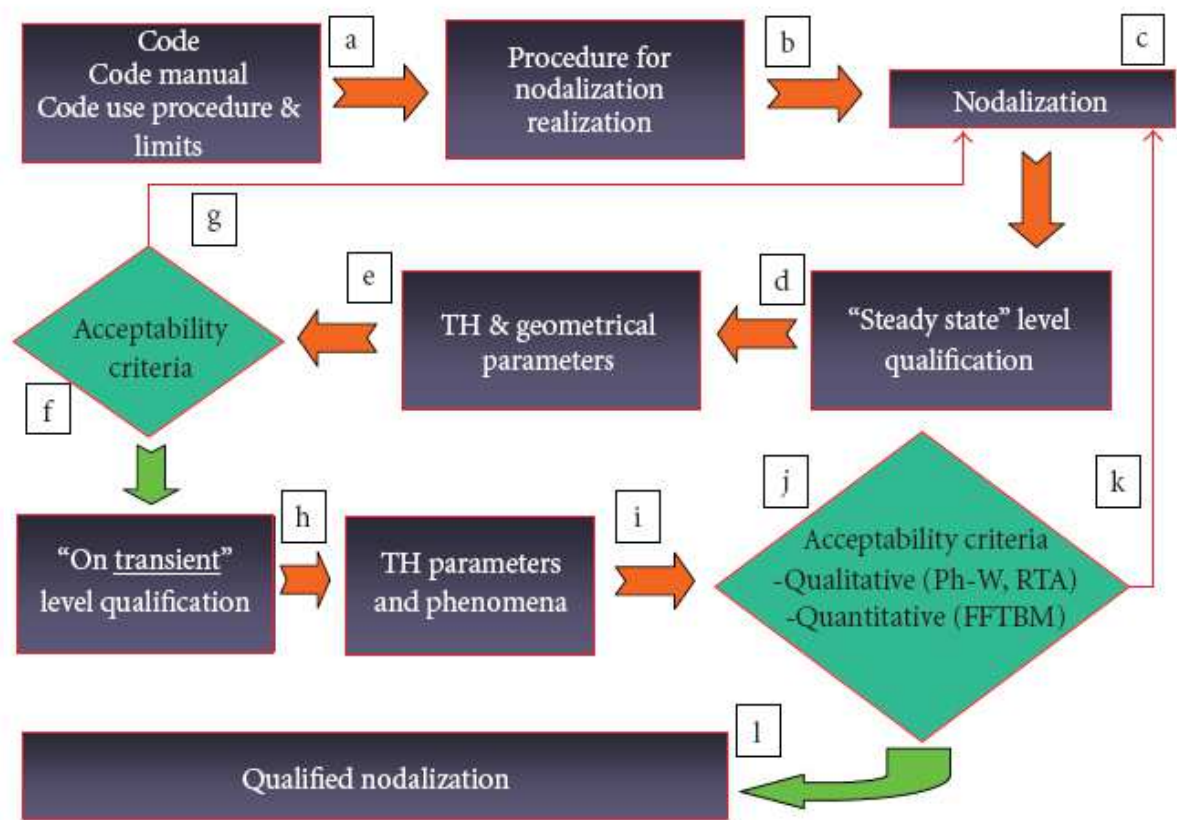


Fig. 2. Flow sheet of nodalization qualification procedure [17]

steady-state and transients analyses. A chain of Neutronic codes has to be employed, [starting from the Evaluated Nuclear Data Files processing, passing through the neutron transport simulations (both in deterministic and stochastic ways), the few group homogenized cross section libraries calculations and ending in the 3D core-wide NK simulations (eventually coupled with a SYS TH code). An example of this sophisticated chain of BE codes is given in Fig. 3, and it was applied by the GRNSPG/UNIP team for the licensing calculation of the Atucha-2 HWR, Argentina. In this case, needs of detailed neutronic analyses were due to the evaluation of the safety margins during a SBDDBA (the DEGB LBLOCA). Because the positive void reactivity coefficient of the reactor, that transient was, at the same time, also a Reactivity Initiated Accident (RIA). The peculiar characteristics of the reactor (e.g., oblique Control Rods and a second emergency scram system injecting boron solution in the full pressure moderator tank) requested the use of advanced Monte Carlo neutron transport simulations too.

Needs for coupling different codes with SYS TH codes, could be necessary also in other technological fields, e.g. for the Pressurized Thermal Shock analyses (coupling Structural Mechanics, Computational Fluid Dynamics and TH codes), the Fuel Pin Mechanics analyses (coupling of Fuel Pin Mechanics, Neutronics and SYS TH codes) or the Containment analyses (coupling of Containment and SYS TH codes).

Quality assurance of the coupling process as well as procedures for codes qualification has to be declared and demonstrated to the National Safety Authority. Example of procedure and validation campaigns of codes of different technological areas can be found in [19], [20], [21].

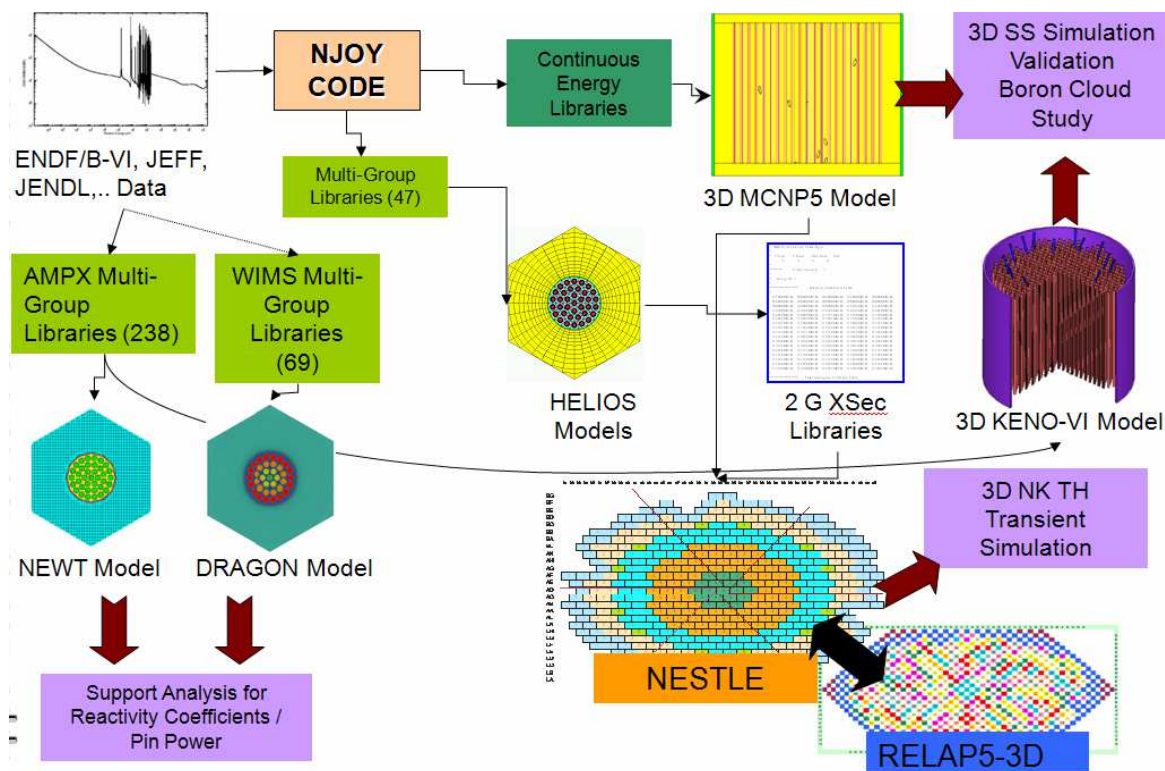


Fig. 3. Example of BE Neutronic Chain of Code use for a set of Licensing Calculations

3.8 Criteria for the application of uncertainty analyses

In many cases, BE calculations on plant behaviour demonstrate a performance with no significant challenges to the applicable safety limits to such extent that, even adding the maximum expected uncertainty, acceptance criteria are fulfilled. For this reason, the GRNSPG/UNIPi proposed BEPU approach derived a non-safety related criterion to decide upon the need for performing uncertainty calculation. Whenever the safety parameter, as calculated by the EM, comes within an established range or distance from the limit value, the uncertainty in the calculated results is quantified (BEPU application, see step “10” of Fig. 1). The development of the non-safety related criterion, implies to establish the “range or distance from the limit value” for the plant safety parameters. The GRNSPG/UNIPi general adopted formula (except for DNBR) can be represented by:

$$Par_{CALC}^{MAX} * \left(1 + U_{PAR}^{MAX} \Big|_{PS}\right) \geq Cr_{Cat}^{Par}$$

where:

- Par_{CALC}^{MAX} is the maximum value of the calculated parameter for the transient-event under consideration;
- $U_{PAR}^{MAX} \Big|_{PS}$ is the maximum uncertainty value affecting the best estimate prediction of the parameter. The maximum uncertainty value is derived from the CIAU database considering the phase-space (PS) characterizing each event-category (AOO, DBA non-LOCA, etc...). Therefore, the “range or distance from the limit value” is given by $Par_{CALC}^{MAX} * U_{PAR}^{MAX} \Big|_{PS}$. In some case, just the maximum value of the uncertainty for the whole phase space is considered (U_{PAR}^{MAX});

- Cr_{Cat}^{Par} is the limit value for the selected parameter and event-category (AOO, DBA non-LOCA, etc...) below of which no uncertainty analysis shall be performed. In some case, no distinction between event category is done (Cr^{Par}). Further information for each specific six non-safety related criteria can be found in [11].

4. The uncertainty quantifications

For licensing applications, evaluation of uncertainty constitutes a necessary complement of BE calculations, which are performed to understand accident scenarios in water-cooled nuclear reactors. The needs come from the imperfection of computational tools, on the one side, and from the interest of using such a tool to get more precise evaluation of safety margins. Several uncertainties methods were developed since the development of the code scaling, applicability and uncertainty (CSAU) evaluation methodology [22] by the U.S. Nuclear Regulatory Commission, e.g. see [23]. Hereafter, a brief description of the key features of the GRNSPG/UNIPI CIAU methodology for the uncertainty is given, together with some sample applications to different LWR.

4.1 A method for the uncertainty quantification: the CIAU method

The UMAE (Uncertainty Method based on the Accuracy Extrapolation) is the prototype method for the consideration of “the propagation of code output errors” approach for uncertainty evaluation. As described in section 3, the method focuses not on the evaluation of individual parameter uncertainties but on the propagation of errors from a suitable database calculating the final uncertainty by extrapolating the accuracy from relevant integral experiments to full scale NPP.

Considering integral test facilities of reference water cooled reactor, and qualified computer codes based on advanced models, the method relies on code capability, qualified by application to facilities of increasing scale. Direct data extrapolation from small scale experiments to reactor scale is difficult due to the imperfect scaling criteria adopted in the design of each scaled down facility. So, only the accuracy (i.e. the difference between measured and calculated quantities) is extrapolated. Experimental and calculated data in differently scaled facilities are used to demonstrate that physical phenomena and code predictive capabilities of important phenomena do not change when increasing the dimensions of the facilities.

Other basic assumptions are that phenomena and transient scenarios in larger scale facilities are close enough to plant conditions. The influence of user and nodalization upon the output uncertainty is minimized in the methodology. However, user and nodalization inadequacies affect the comparison between measured and calculated trends; the error due to this is considered in the extrapolation process and gives a contribution to the overall uncertainty.

The method utilizes a database from similar tests and counterpart tests performed in integral test facilities that are representative of plant conditions. The quantification of code accuracy is carried out by using a procedure based on the Fast Fourier Transform characterizing the discrepancies between code calculations and experimental data in the frequency domain, and defining figures of merit for the accuracy of each calculation. Different requirements have to be fulfilled in order to extrapolate the accuracy [24].

Calculations of both Integral Test Facility experiments and NPP transients are used to attain uncertainty from accuracy. Nodalizations are set up and qualified against experimental data

by an iterative procedure, requiring that a reasonable level of accuracy is satisfied. Similar criteria are adopted in developing plant nodalization and in performing plant transient calculations. The demonstration of the similarity of the phenomena exhibited in test facilities and in plant calculations, accounting for scaling laws considerations, leads to the Analytical Simulation Model (ASM) or Evaluation Model, with a qualified nodalization of the NPP that The flow diagram of UMAE is given in Fig. 4. The bases of the methods and the conditions to be fulfilled for its application, including the use of the FFTBM can be found in references [25] and [26 to 29].

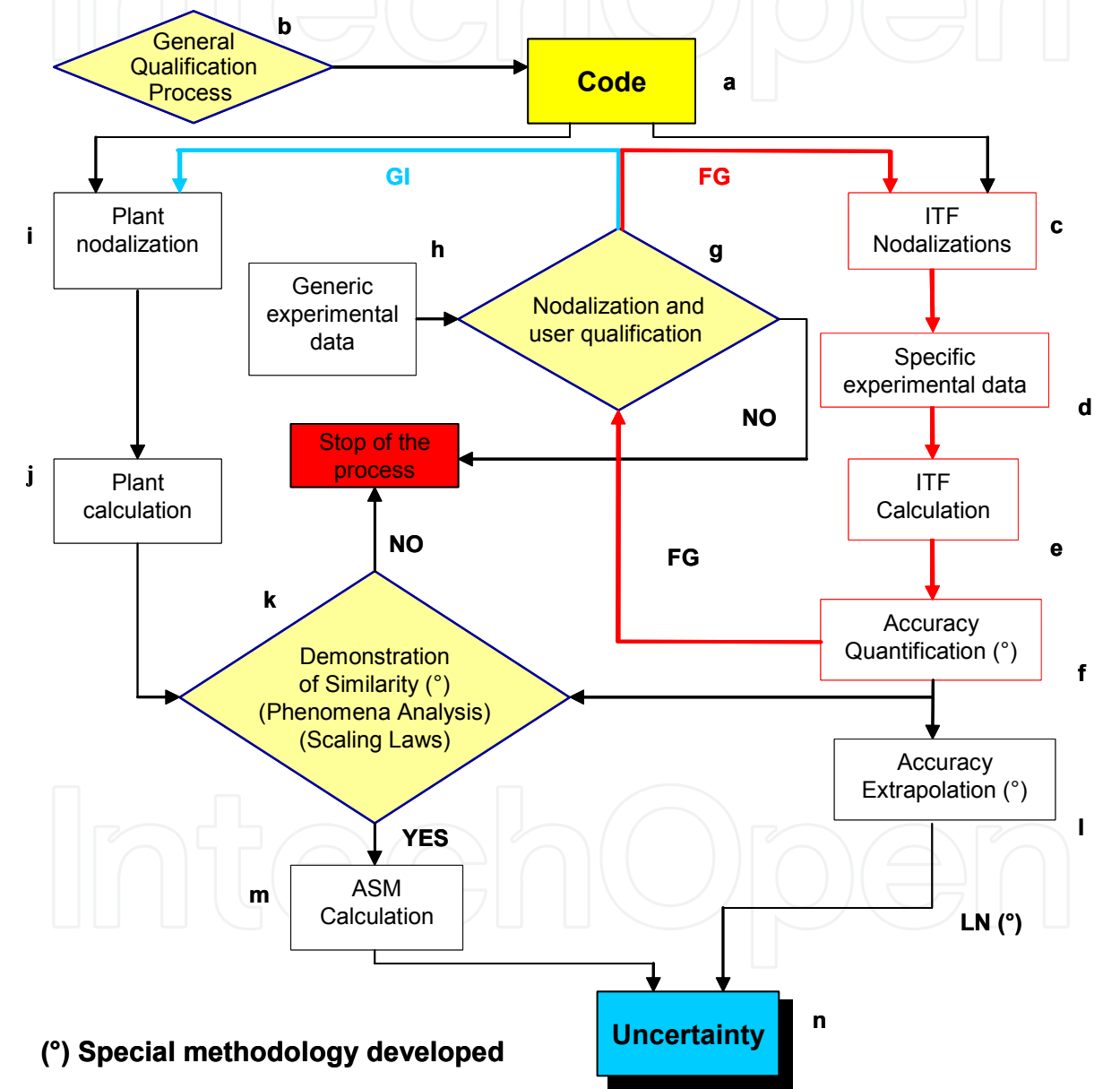


Fig. 4.UMAEE flow diagram (also adopted within the process of application of CIAU). All of the uncertainty evaluation methods are affected by two main limitations:

- The resources needed for their application may be very demanding, ranging to up to several man-years;
- The achieved results may be strongly method/user dependent.

The last item should be considered together with the code-user effect, widely studied in the past, e.g. reference [28], and may threaten the usefulness or the practical applicability of the results achieved by an uncertainty method. Therefore, the Internal Assessment of Uncertainty (IAU) was requested as the follow-up of an international conference jointly organized by OECD and U.S. NRC and held in Annapolis in 1996 [31]. The CIAU method [32] has been developed with the objective of reducing the above limitations.

The basic idea of the CIAU can be summarized in two parts, as per Fig. 5:

- Consideration of plant status: each status is characterized by the value of six relevant quantities (i.e. a hypercube) and by the value of the time since the transient start.
- Association of an uncertainty to each plant status.

In the case of a PWR the six quantities are: 1) the upper plenum pressure, 2) the primary loop mass inventory, 3) the steam generator pressure, 4) the cladding surface temperature at 2/3 of core active length, 5) the core power, and 6) the steam generator down-comer collapsed liquid level.

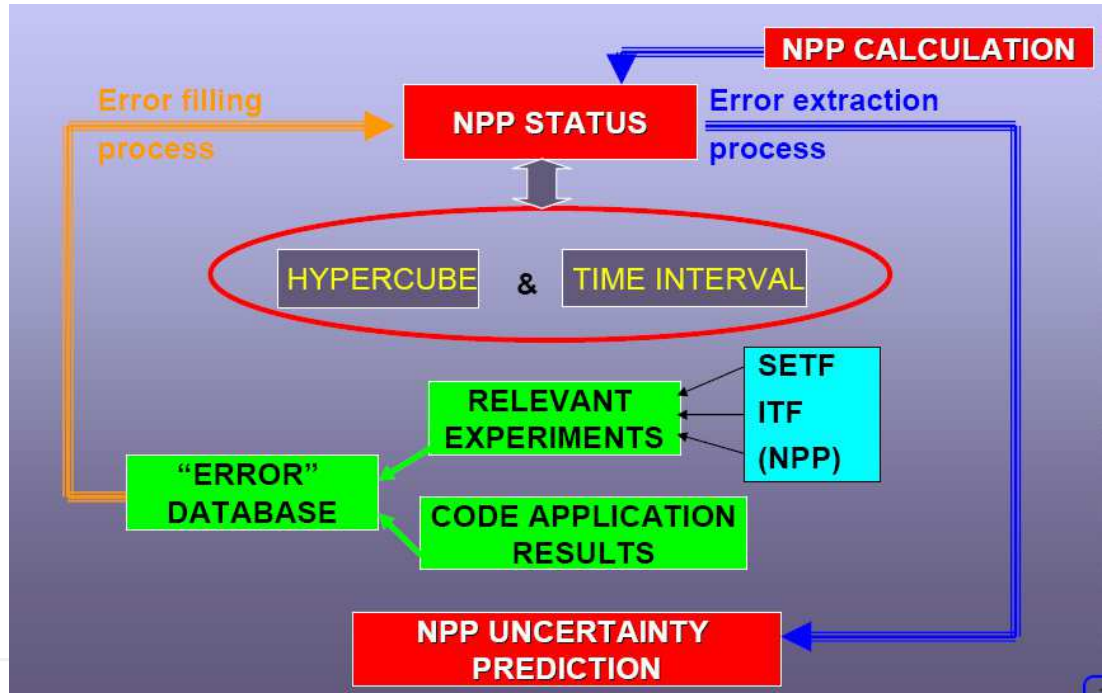


Fig. 5. Outline of the basic idea of the CIAU method.

A hypercube and a time interval characterize a unique plant status to the aim of uncertainty evaluation. All plant statuses are characterized by a matrix of hypercubes and by a vector of time intervals. Let us define Y as a generic thermal-hydraulic code output plotted versus time. Each point of the curve is affected by a quantity uncertainty (U_q) and by a time uncertainty (U_t). Owing to the uncertainty, each point may take any value within the rectangle identified by the quantity and the time uncertainty. The value of uncertainty, corresponding to each edge of the rectangle, can be defined in probabilistic terms. This satisfies the requirement of a 95% probability level to be acceptable to the NRC staff [33] for comparison of best estimate predictions of postulated transients to the licensing limits in 10 CFR (Code of Federal Regulation) Part 50.

The idea at the basis of CIAU can be made more specific as follows: the uncertainty in code prediction is the same for each plant status. A Quantity Uncertainty Matrix (QUM) and a

Time Uncertainty Vector (TUV) can be set up including values of U_q and U_t derived by an uncertainty methodology, Fig. 6.

At the moment the UMAE constitutes the ‘engine’ for the rotation of the CIAU shaft. The QAM and TAV, respectively Quantity Accuracy Matrix and Time Accuracy Vector in Fig. 6 are derived from an UMAE like process and are the precursor of QUM and TUV. However, within the CIAU framework, any uncertainty method can be used to derive directly QUM and TUV.

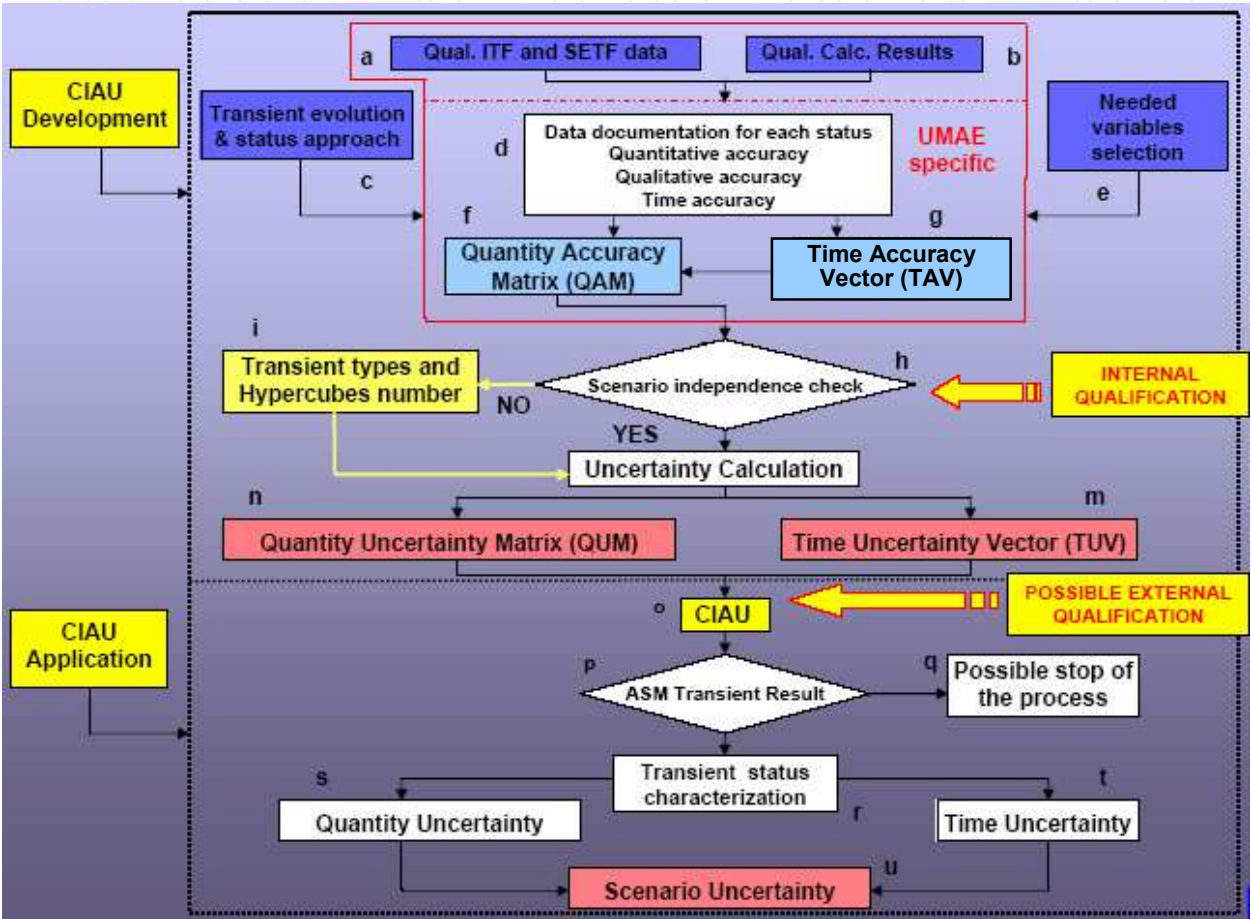


Fig. 6. Flow diagram at the basis of the CIAU methodology

4.2 The CIAU qualification

The UMAE and the CIAU have been used within the OECD/NEA/CSNI international projects UMS [23] and BEMUSE [34, 35 and 36]. The application of CIAU within the licensing process of Angra-2 PWR (a regulatory audit calculation, see later) [37 to 39] and the evaluation of safety of the Kozloduy-3 VVER-440 [40] can be recalled. Sensitivity studies and comparison with results from different uncertainty methods confirmed the qualification level of CIAU in those frameworks. Finally, different internationally available publications have been issued in relation to what is called external qualification process of CIAU, e.g. reference [41].

4.3 Sample applications of CIAU

Application of CIAU method to licensing analyses was performed by GRNSPG/UNIPI in various occasions, and last to the under-construction Atucha-2 NPP. Results of this work could not be included in this Chapter because at the time of writing the review process from the Safety Authority was still ongoing. Hereafter, there are shortly reported hereafter some other examples of CIAU applications produced in the last years.

4.3.1 Uncertainty Analysis of the LBLOCA-DBA of the Angra-2 PWR NPP

Angra-2 is a 4 loop 3765 MWth PWR designed by Siemens KWU. The NPP is owned and operated by the ETN utility in Brazil. The NPP design was ready in the '80s, while the operation start occurred in the year 2000 following about ten-year stop of the construction. The innovation proposed to the licensing process by the applicant consists in the use of a BE tool and methodology to demonstrate the compliance of the NPP safety performance with applicable acceptance criteria set forth in the Brazilian nuclear rule.

The CIAU application was employed for performing an 'independent' BEPU analysis of the LBLOCA-DBA of the NPP. The analysis was classified as 'independent' in the sense that it was carried out by computational tools (code and uncertainty method) different from those utilized by the applicant utility. The main results are summarized in Fig. 7 and 8, where the PCT and the related uncertainty bands obtained through the CIAU and through the computational tools adopted by the applicant, are given. The following comments apply:

- Continuous uncertainty bands have been obtained by CIAU related to rod surface temperature (Fig. 7), pressure and mass inventory in primary system. Only point values for PCT are considered in Fig. 8;
- The CIAU (and the applicant) analysis has been carried out as BE analysis: however, current rules for such analysis might not be free of undue conservatism and the use of peak factors for linear power is the most visible example;
- The conservatism included in the reference input deck constitutes the main reason for getting the 'PCT licensing' from the CIAU application above the acceptability limit of 1200 °C;
- The amplitude of the uncertainty bands is quite similar between the CIAU and the applicant. Discrepancies in the evaluation of the 'PCT licensing' outcome from the way of considering the 'center' of the uncertainty bands. In the case of CIAU, the 'center' of the uncertainty bands is represented by the phenomenological result for PCT obtained by the reference calculation (1100 °C in Fig. 7). In the case of applicant the 'center' of the uncertainty bands is a statistical value obtained from a process where the reference calculation has no role (796 °C in Fig. 8);
- The reference BE PCT calculated by the applicant (result on the left of Fig. 8) plus the calculated uncertainty is lower than the allowed licensing limit of 1473 K;
- The reference BE PCT calculated by CIAU (result on the right of Fig. 8) is higher than the PCT 'proposed' by the applicant and the upper limit for the rod surface temperature even overpasses the allowed licensing limit of 1473 K thus triggering licensing issues;
- Based on the results at the previous point, new evidences from experimental data have been made available by the applicant. This allowed to repeat the BE reference calculation (both for the CIAU and the applicant). The new reference BE PCT calculated by CIAU is lower than the previous (about 200 °C) and close to the new reference PCT calculated by the applicant ('base case' in Fig. 7) and it is shown that the new CIAU upper limit for the rod surface temperature is lower than the allowed licensing limit of 1473 K.

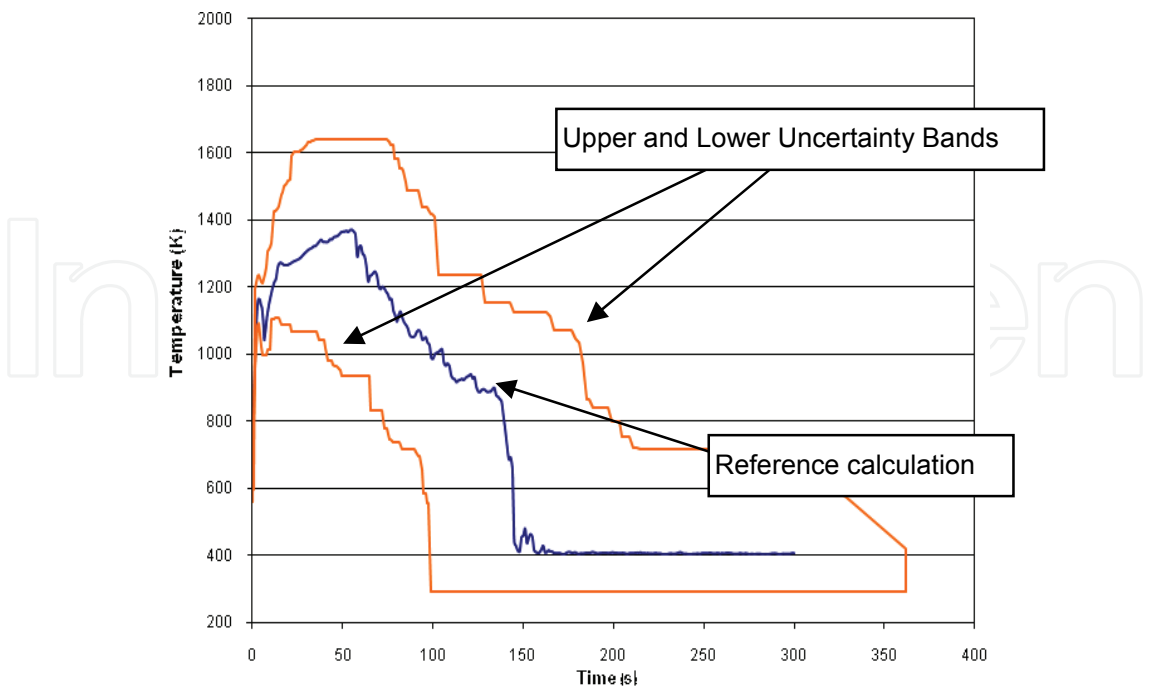


Fig. 7. Uncertainty bands for realistic hot rod

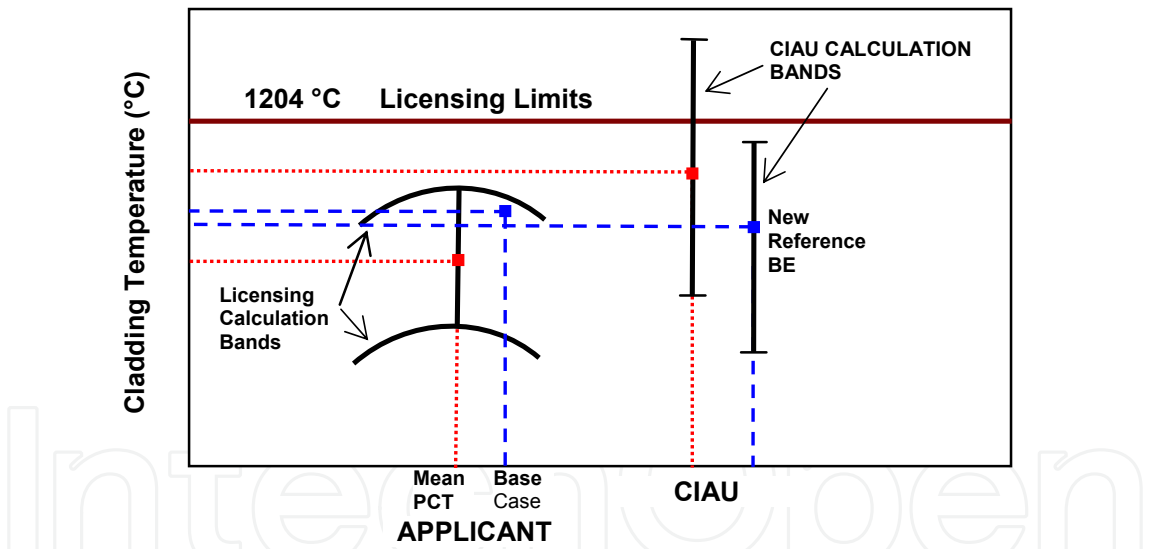


Fig. 8. Final result from the CIAU study and comparison with results of the applicant.

4.3.2 BE and uncertainty evaluation of LBLOCA 500 mm for Kozloduy-3

The analysis of the ‘LBLOCA 500 mm’ (DEGB in CL) transient [40] was carried out by adopting the Relap5 code. The specific purposes of the analysis included the assessment of the results and the execution of an independent safety analysis supported by uncertainty evaluation. A BE transient prediction of the ‘LBLOCA 500 mm’ was performed. Evaluation of the uncertainty was performed by CIAU for the RPV upper plenum pressure, the mass inventory in primary system and the hot rod cladding temperature. Only the last parameter is shown in Fig. 9 together with the uncertainty bands. The most relevant result is the

demonstration that the PCT in the concerned hot rod is below the licensing limit. In the same Fig. 9, bounding results (PCT and time of quenching) from two conservative calculations (i.e. obtained by a BE code utilizing conservative input assumptions) are given: one is the conservative calculation ('driven' conservatism in Fig. 9) performed by the applicant, the other is the conservative calculation performed by GRNSPG/UNIPI ('rigorous' conservatism in Fig. 9). The following can be noted:

- The 'driven' conservative calculation has been performed by the applicant using a set of values for the selected conservative input parameters different respect to the values adopted in a previous analysis and accepted by the regulatory body;
- The 'driven' conservative calculation is not "conservative" and does not bind entirely the BEPU upper bound. This implies that code uncertainties are not properly accounted for by the adopted conservative input parameter values;
- The 'rigorous' conservative calculation performed by GRNSPG/UNIPI is correctly conservative (i.e. it use the same set of values for the selected conservative input parameters previously licensed), but its conservatism is such to cause PCT above the licensing limit;
- The comparison between the conservative PCT obtained by GRNSPG/UNIPI and the CIAU upper band of the BEPU calculation shows the importance of using a full BE approach with a suitable evaluation of the uncertainty.

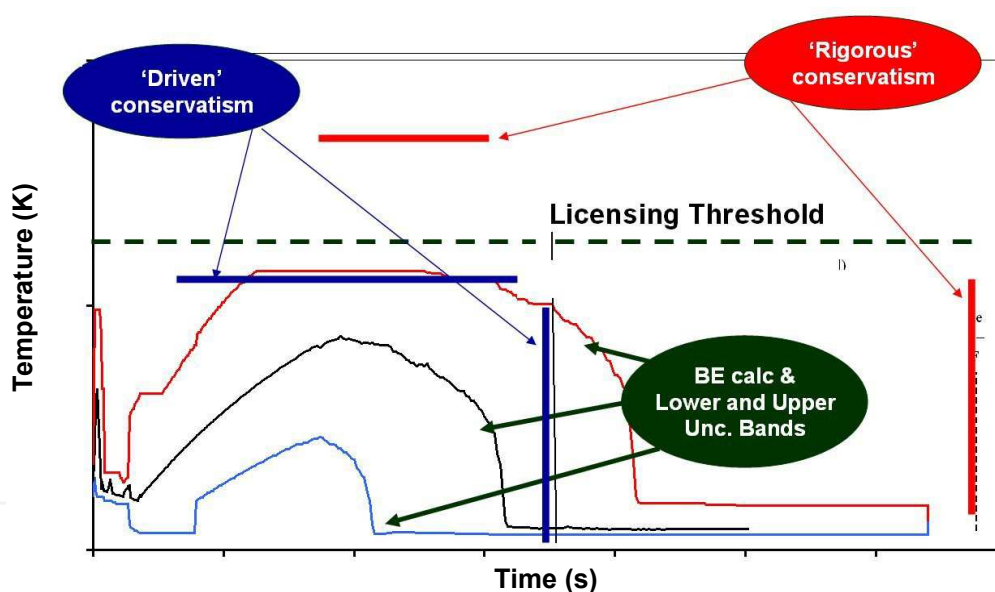


Fig. 9. Surface temperature at the PCT location in hot rod and uncertainty bands derived by CIAU application

4.3.2 The BEMUSE project

The last selected CIAU application constitutes a qualification study that at the same time allows a comparison with results of different uncertainty methods. Within the OECD (Organization for Economic Cooperation and Development) framework, two main activities related with the uncertainty evaluation have been performed (actually the second one is still in progress): the UMS [23] and the BEMUSE, [35], respectively. The objective of the BEMUSE project was to predict the LBLOCA performance of the LOFT experimental nuclear reactor

(i.e. test L2-5). The process included two steps: the derivation of a reference calculation, involving a detailed comparison between experimental and calculated data, and the derivation of uncertainty bands enveloping the reference calculation. The success of the application consisted in demonstrating that the uncertainty bands envelope the experimental data. Ten international groups participated to the activity. A sample result from the BEMUSE project is outlined in Fig. 10.

The application of the CIAU was performed by the GRNSPG/UNIPI (dotted vertical line in Fig. 10) while all other participants used uncertainty methods based on the propagation of the input errors supplemented by the use of the Wilks formula. The consistency between the CIAU results and the experimental data can be observed as well as the spread of results obtained by the other uncertainty methods based on the propagation of the input errors.

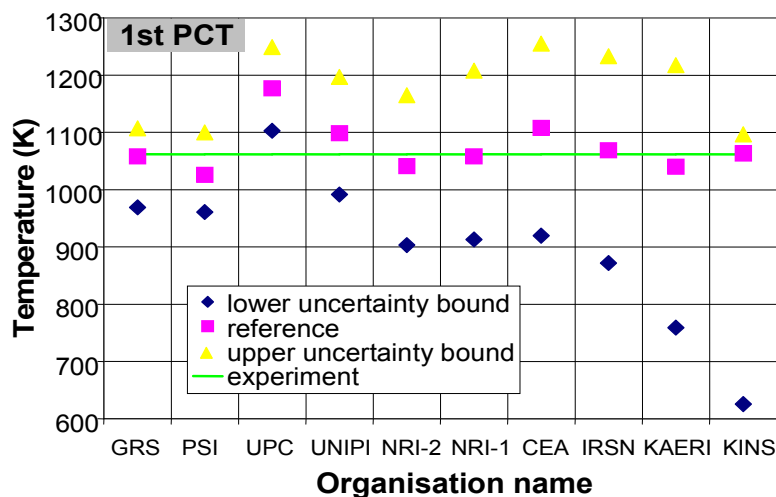


Fig. 10. Uncertainty bounds from each participant ranked by increasing band width from left to right related to the 1st PCT' of the LOFT experiment L2-5.

5. Conclusions

Massive and, at the same time, safe and reliable exploitation of nuclear energy constitutes a great challenge for the today world. Tremendous progresses were obtained during the last 30 years in the field of nuclear reactor simulation thanks to an extended knowledge of several physical phenomena, to the code validation & assessment campaigns and to the impetuous advance of (cheap) computer technology.

This chapter briefly presented the state-of-the-art in the licensing process for nuclear reactors, focusing on the potentialities and the advantages of the so-called BEPU approach. References to works and specific examples of the GRNSPG/UNIPI applications of such methodology to being-built or in operation nuclear power plants were also reported for giving a clearer idea of this sophisticated industrial process.

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