BEST ESTIMATE PLUS UNCERTAINTY APPROACH IN NUCLEAR REACTOR SAFETY AND LICENSING
Brief History and Elements after Licensing

by

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The best estimate plus uncertainty is, at the same time, an approach, a procedure and a framework in nuclear thermal-hydraulics and nuclear reactor safety and licensing. The motivation at the basis of the best estimate plus uncertainty is the lack of knowledge in the areas of single and, mainly, two-phase transient thermal-hydraulics. In other terms and introducing some simplifications, the insufficient knowledge of turbulence imposes the design of roadmaps for the application of imperfect (thermal-hydraulic) models to the evaluation of design features and of safety for complex technological installations or systems like the nuclear power plants and, more specifically, the water cooled nuclear reactors. Furthermore, the legal counterpart of nuclear reactor safety, or the licensing, is concerned: therefore the best estimate plus uncertainty must account for rules and regulations derived from the fundamental radioprotection principle which imposes the minimization of the impact of radiations upon humans and the environment under any circumstance. In the present paper, the key elements of the approach are identified and characterized. These shall be seen as the support for a consistent application of thermal-hydraulics to the design and safety of water-cooled nuclear reactors.

Key words: best estimate, uncertainty evaluation, best estimate plus uncertainty, licensing process, validation, scaling

INTRODUCTION

Details of physics, like turbulence, which are essential for the simulation of liquid-steam mixture transient performance are known with approximation; indeed, within a nuclear reactor, thermal-hydraulics best estimate (BE) models are built, numerical calculations are performed and errors in calculation results, or Uncertainty (U), are expected. The words best estimate plus uncertainty, *i. e.*, BEPU, since their origin constituted the attribute of a calculation performed by a system thermal-hydraulic code. Three items allow a pedagogic definition of BEPU:

– Best estimate *vs*. conservative. The word conservative has been always used in engineering calculations to mask a lack of knowledge, or, in some cases to achieve resistance capabilities and strengths of a system higher than what strictly needed by the design. The words *best estimate* do not imply the removal of the conservatism; rather they impose the minimum conservatism consistent with the actual technological knowledge.
– Capabilities and limitations of system codes. Thermal-hydraulic system codes constitute the reference computational tools for the application of BEPU. Notwithstanding large efforts in development and validation, the current features and the capabilities of those codes are such to envisage unavoidable errors in any prediction. This is specifically true for the application of codes to the transient analysis of accidents in nuclear power plants (NPP).
– Uncertainty. The unavoidable errors expected from the application of computational tools in thermal-hydraulics (also referred above) need to be quantified to determine their acceptability. Then, the word *uncertainty* appears: uncertainty is the prediction of the error expected in NPP calculations results. Specific methods, or procedures or methodologies, are needed and have been developed and qualified to determine the uncertainty; acceptability or the error (*i. e*. of the uncertainty) is directly or indirectly linked with uncertainty evaluation.

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Therefore BEPU implies the use of an *imperfect* BE code where the 'U' (or PU in the acronym) constitutes the substitute for the conservatism. The acceptability of the BEPU approach for the analyses within the licensing process of NPP is the challenge for the industry (the licensee) and, primarily, for the regulators (the licensor).

The features of BEPU are such to make attractive its use to address any design and operational issue in NPP technology. The BEPU is also the tip of the iceberg of competence in nuclear thermal-hydraulics or a cross-way into the thermal-hydraulics universe. The development of BEPU implies interactions between deterministic and probabilistic analyses also recognized as deterministic safety assessment (DSA) and probabilistic safety assessment (PSA).

Progresses in proving the capabilities of the codes continued during the current millennium as well as in further demonstrating the qualification level of uncertainty methods. In the same period institutions in the area of nuclear reactor safety (NRS) issued documents, IAEA, [1] and [2] and USNRC, [3], opening or promoting the application of BEPU. Practical applications of BEPU for the new NPP, even due to the small number of new units entered in operation in the period 2000-2015 are restricted to a few cases; two pioneering efforts were: (a) final safety analysis report (FSAR), Chapter 15 – accident analysis – large break loss of coolant accident (LBLOCA), in Angra-2 NPP in Brazil (1997-2000), [4]; b) FSAR, Chapter 15 – accident analysis – all transients where analytical support is needed, in Atucha-2 NPP in Argentina (2008-2011), [5].

Till the present time, BEPU never became a spreading technology or even an internationally recognized approach for demonstrating the safety of nuclear reactors.

The objective of the present paper is to provide an overall vision for BEPU, *i. e.*, comprehensive as far as possible without entering important details which also characterize the approach. To this aim key elements and framework, are distinguished. Outcomes from applications are summarized together with envisaged developments.

WHAT IS BEPU?

A single or unique definition for BEPU may not be sufficient or may not give the correct picture or may even not address the question *what is BEPU*? Conferences starting from BE-2000, [6], provide broad and dispersed answers to the question; furthermore, a glossary type definition is not even part of more recent publications dealing with BEPU, *e. g.* [7]. Hereafter, the following selected topics are used to provide an overall vision for BEPU:

a) (Outline of a short BEPU) History.
b) BEPU-QA (BEPU – Quality Assurance).
c) ALARA-BEPU-IA (As low As Reasonably Achievable – BEPU – Independent Assessment).
d) The database and the BEPU KM (Knowledge Management), [8].
e) From basics in TH (thermal-hydraulics) and NP (neutron physics) to BEPU.
f) The spectrum of AA (accident analysis): breaking the barrier DSA/PSA.
g) The wild traveler and the ghost city.

The BEPU history

A view for the BEPU history can be drawn from fig. 1; a summary history of nuclear thermal-hydraulics can be found in [9] and in Chapters 1 and 2 of the book in [7]. Reference progress of science and technology and advancement in regulations can be seen on

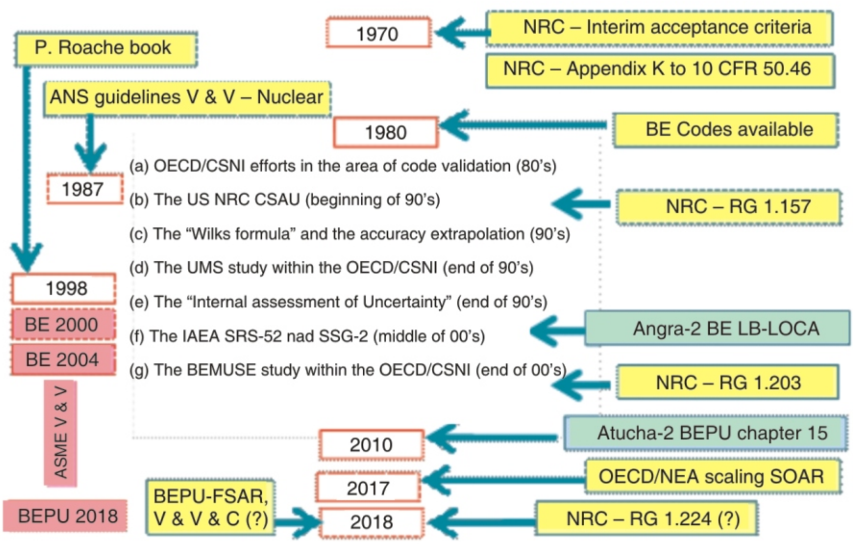

Figure 1. A picture for the BEPU history

the left and right of the central column in fig. 1 where decades starting from 1970 are reported. The pioneering book by P. Roache and the American Nuclear Society (ANS) triggering activities in the '70s are emphasized on the left part; these are followed by Organization for Economic Co-operation and Development / Nuclear Energy Agency / Committee on the Safety of Nuclear Installations (OECD/NEA/CSNI) activities and related reports issued since early 80's till nowadays: the state of art report (SOAR) about thermal-hydraulics of emergency core cooling systems (TECCS) and about scaling can be mentioned together with UMS, BEMUSE, PREMIUM and SAPIUM co-operative projects. The verification and validation (V&V) series of conferences by American Society of Mechanical Engineering (ASME) shall be mentioned together with the BE series of conferences organized by ANS, recently continued with the BEPU conference in Lucca (Italy, May 2018). The progress in licensing (right side of the diagram) is summarized by ECCS rules issued in 1971 by the United States Atomic Energy Commission with subsequent modifications of proposed modalities to perform the (licensing) analysis, *i. e*., Regulatory Guides (RG) 1.157, 1.203 and more recently (still draft) 1.224. The International Atomic Energy Agency (IAEA) vision is summarized by the SSG-2 report. The response of industry is exemplified by the large break loss of coolant accident (LBLOCA) BEPU analysis in Angra-2 (Brazil, around 2000) and the BEPU application to all accidents part of the FSAR in Atucha-2 (Argentina, around 2010): more details related to both cases are given in section *SAMPLE APPLICATIONS*.

The BEPU-QA and ALARA-BEPU-IA

The ALARA principle is at the basis of radioprotection; a straightforward extension to nuclear safety implies an optimization process: the best should be done to minimize the impact of radiations upon the peoples and the environment. BEPU shall be considered a consistent outcome of the ALARA principle in the areas of AA and computation, [10].

The maintenance of the desired level of quality or the way to minimize or prevent mistakes and defects are associated with the acronym QA. The following characteristics apply to QA and BEPU, [10]:
(a) existence of technological (computational) tools to be inserted into an application process,
(b) the key role of qualification (*i. e*., of computation tools),
(c) existence of a regulatory environment (including acceptability requirements), and
(d) relevance and financial value for the outcomes of the technological processes associated with the use of computational tools: noticeably NPP design and safety evaluations.

The QA and QA framework introduce requirements for the BEPU process and at the same time are inherent parts of the BEPU.

The IA constitutes a deep-rooted requisite within NRS established since the industrialization of nuclear technology. The stakes associated with the cost of the NPP, the complexity of the system and the potential impact of radiation upon the humans and the environment justify IA: safety must be demonstrated by experts *independent* from designers and owners of the concerned system who take a benefit from the operation of the same system. The implementation of the IA requirement may reveal unfeasible outside BEPU, [11]; namely this is the only logical approach for the licensor for getting the IA of the licensee submission: the uncertainty in input parameters is derived *independently* and substitutes, and eventually is consistent with, the conservatism of the licensee possibly based on proprietary (best estimate) information.

Thus, BEPU can be imagined as a technological cornerstone supported by QA, IA, and ALARA.

The knowledge and knowledge management

A typically huge database [12] and the knowledge management constitute key term and activity, needed for a BEPU application, respectively. The *house of knowledge* originally produced when issuing the S-SOAR, (scaling state of art report), [8], can easily be extended to the BEPU.

The background for BEPU has been created within nuclear thermal-hydraulics which also constituted a pillar discipline for NRS. Huge investments in research after the (almost simultaneous) events of issuing the ECCS Interim Acceptance criteria by USAEC, section *BEPU-QA and ALARA-BEPU-IA*, and the availability of powerful computers to perform a transient analysis of NPP, had their focus in nuclear thermal-hydraulics, primarily during the years 1970-1995 as previously discussed. The investments, as far as possible, grew as a consequence of the Three Mile accident in 1979. Hundreds of reports and key-documents were issued and constitute the tangible outcomes of those investments; a systematic coverage only for the topics of those documents is well beyond the framework for the present paper; see *e. g*. [6] to [9] for details. Nevertheless, activities and corresponding groups of documents are distinguished in the areas listed below to provide an idea of the database which constitutes the fundaments of BEPU:
– A few thousands *integral* experiments where the reactor coolant system is simulated have been performed: up to around 1000 variables (order of magnitude) are measured in each experiment with typical frequency (per signal) of the order of 1 Hz or higher. Noticeably, 2-D/3-D large scale test fa-

cilities including UPTF have been operated and counterpart tests (CT) have been performed.

– A much higher number (*i. e*. related to *Integral* experiments) of relevant *Separate Effect* or *Basic* experiments have been performed: although the word *relevant* is not defined here, an order of magnitude of (several) 104 tests have been performed which have been used to develop nuclear thermal-hydraulics.
– A dozen system codes, independent among each other, have been developed to calculate the transient performance of NPP following accidents: concerning involved resources, the development and the qualification efforts per each code may be in the order of (a few) 10 man-years and (several) 100 man-years, respectively. Qualification implies the use of the experimental database at the aforementioned items.
– Models and correlations, *e. g*. TPCF, CHF, CCFL, have been developed and qualified and constitute parts of the system codes.
– The qualification (V&V), the scaling and the uncertainty methods and procedures are discussed in section *SELECTED KEY ELEMENTS* and constitute *direct* elements part of the BEPU. The related database exists, and comprehensive reports are published.
– The design and the operation of NPP require the applications of thermal-hydraulics: the reactor coolant system (RCS), specific components (*e. g*. main coolant pimps), core, containment and balance of plant (BoP), imply deployment of thermal-hydraulics knowledge.
– Application of thermal-hydraulics are needed for safety demonstrations of NPP: noticeably, the ESF (ECCS are part of these) and the EoP, are needed to prevent or to limit the radioactivity impact upon the environment should an accident occur.
– Acronyms (easily traceable on the web) for international projects and documents issued by OECD/NEA/CSNI, like UMS, BEMUSE, SOAR, ISP, and IAEA, like SRS (dealing with AA), SSG (dealing with proposed safety evaluation frameworks), TECDOC (dealing with specific aspects of nuclear safety and/or applications of thermal-hydraulics) should be considered.
– Important efforts performed by USNRC are summarized by various reports: key words are *compendium of thermal-hydraulic researches*, NUREG series (several documents), *standard review plan for the review of safety analysis reports for Nuclear Power Plants*, regulatory guides (several documents).
– Selected BEPU applications are discussed in section *SAMPLE APPLICATIONS*

Application of BEPU implies knowledge and consideration of topics listed above (a comprehensive list of documents can be found in [7]). Among other things (these can be seen as important reasons which have limited so far the exploitation of BEPU):

– the overall number of pages corresponding to documents supporting the listed activities can be estimated in the order of several 10-thousands pages),
– about 200 man-years are needed for the full application of BEPU to the analysis of accidents in Chapter 15 of the FSAR.

From the basis in science and technology to BEPU

The BEPU can be seen as an evolution from fundamentals in thermal-hydraulics up to complex phenomena in reactor conditions, [13]; in different terms, assembling and developing the knowledge from basics in TH and NP to derive compound NPP scenarios is a matter for BEPU, [14]. Just as an example in this connection, approximate or reference (minimum or maximum) values for time-scales and length-scales in the area of thermal-hydraulics (much smaller values characterize the lower boundaries of the given ranges in the case of NP) are:

(a) length scale: a BEPU application shall consider phenomena occurring locally as the step-change in heat transfer coefficient typical of quench from advancement in a few mm or the global multidimensional flows in the containment at the scale of 100 m.
(b) time scale: time scales as small as 0.001 s in case of pressure wave propagation or 0.01 s typical of neutron flux excursions shall be dealt with, as well as time-scale values for 1000 s (or even 100000 s) which characterize complex *long-term* transients like SGTR (steam generator tube rupture).

The BEPU analysis of a single complex transient (*e. g*. an LBLOCA) may include the consideration of the overall ranges of time and length scales, *i. e*. not only for thermal-hydraulics.

The spectrum of AA: breaking the barriers among technology sectors

The BEPU is expected to cover the overall spectrum of accidents and accident evolutions in NPP. Several subjects and disciplines, *i. e*., including thermal-hydraulics (TH), radioprotection, structural mechanics (SM), neutron physics and chemistry, part of either DSA or PSA areas, are needed to perform accident analyses (AA). Currently, virtual barriers exist between any couple of subjects or discipline: analyses performed in the context of one subject or discipline have no feedback from another subject or discipline. Within a BEPU framework full connection (or no barrier) is foreseeable among subjects and areas. This is also true if one considers system thermal-hydraulics and CFD codes: coupling may be needed, although the related process may reveal inherently incorrect or physically inconsistent.

The wild traveller and the ghost city

A global image for the role and the importance of BEPU can be derived by a metaphor of the wild traveller who enters a ghost city, [13]. The traveller is an expert (*e. g.* in thermal-hydraulics) who has full knowledge of the fundaments of the concerned subject or discipline. Suddenly he enters a ghost city (no inhabitants) never encountered in his life journey. He knows the elements part of the city, *e. g.* building, cars, trains, *etc*.; however, he has no idea of how to make alive the city. What is missing is the civilization. In the same way, a TH expert may have no idea of what to do with equations, models, phenomena, experiments and numerical methods once an NPP appears: BEPU, which looks like the civilization, is needed to match the expertise and the needs.

SELECTED KEY ELEMENTS

Inherent, characterizing and framework type of elements can be distinguished and all-together constitute the BEPU technology. Any description of the elements which are listed in this section and any related systematic insights are well beyond the purpose and the possible size for the present paper: so, the given references shall be used as guidelines for deriving the needed details.

The *inherent elements* are part of the acronym BEPU:
- The numerical BE code(s).
- The nodalizations or the input decks needed to operate the codes.
- The uncertainty method(s).

Description of suitable codes, of nodalization features and associated issues, can be found in [15] and [16], whereas in the latter document one may find connection with topics of current interest and development. The role of nodalization can be stressed at this point. The nodalization is the interface between the code and the reality to be modeled; this is also the result of a brainstorming process where the code user takes into account of code capabilities and limitations and supplies the code with possible lacking information. One noticeable example of *lacking information* to be supplied by code user is constituted by the pressure drop coefficients at geometric discontinuities: the values of those coefficients may hugely impact the results of predictions. At the end of the process of setting-up of code and a consistent nodalization, BEPU calculation results must compare with reality. In other terms,

$$\text{reality} \quad \text{ideal reality} \quad \text{BEPU} \quad [\text{NO}] \tag{1}$$

$$\text{reality} \quad \text{BEPU} \quad [\text{YES}] \tag{2}$$

Therefore, the results of BEPU calculations are directly comparable with data from experiments or reality. When performing the analysis under the conditions at the second paradigm above one realizes that differences between predictions and reality are unavoidable. This is why uncertainty (third bullet item) methods and evaluations are needed: qualities of codes and connected uncertainty methods shall be so high to cover those differences, ending-up at the same time with values for the differences that are small enough to avoid useless predictions. Description of suitable uncertainty methods can be found in [17].
- The characterizing elements for BEPU are:
- The phenomena.
- The V&V.
- The scaling.
- The code coupling.
- The framework and the requirements for the AA part of the final safety analysis report (FSAR).
- The qualification of the code user or the analyst (or better, of a group of analysts).

The discussion of *characterizing elements* shall start with a key notation which also applies to *framework elements*: current codes and uncertainty methods are not perfect and, perhaps, far from being perfect; they are simply the best tools consistent with the current knowledge. Therefore a continuous check of this condition (*i. e.*, to be the best consistent tools) is needed. This implies a deep understanding of characterizing and framework elements. Furthermore, common understanding and established findings are considered in this section and innovations proposed in the last 1-2 years are discussed in section *INNOVATION AND PERSPECTIVES*.

The thermal-hydraulic phenomena expected in NPP during accident are derived from (scaled) experiments and from the expertise of those who proposed the related lists: the WCNR (Water Cooled Nuclear Reactors), the spectrum of DBA (Design Basis Accident) and the ITF/SETF (Integral Test Facility / Separate Effect Test Facility) experiments constitute reference acronyms for the lists of phenomena [18, 19].

The complexity associated with scenarios part of AA and the insufficient fundamental knowledge imposes V&V for computational tools. A wide variety of documents deals with V&V, *e. g.*, from [20], [21] for pioneering efforts, till [22] (unpublished) for the current summary vision.

Scaling can be associated with numerical codes and with phenomena and/or can be considered part of V&V. The origin of what is called the scaling issue is the lack of experimental data at nuclear reactor conditions, basically full pressure, full geometrical sizes and full power, [23].

Numerical codes developed in different areas relevant to AA, *e. g.*, thermal-hydraulics, mechanics, chemistry radioprotection and neutron physics, need to be coupled. Issues like time step control and feedbacks upon coupled calculation results from different physical mechanisms must be solved, *e. g.*, [15] and [24]. The framework and the requirements for the AA part of the FSAR are discussed in the USNRC Standard Review Plan (SRP), [25]: a huge number of connections with BEPU analyses shall be established.

The availability of rules and conditions for BEPU analyses does not avoid the need for qualified analysts. User-effect and specialized-training constitute established outcomes from the R&D at the basis of BEPU; motivations and key requirements for qualification of analysts can be found in [26].

The framework elements of BEPU are constituted by reports, procedures, findings, *etc*., that support or are complementary to the inherent and the characterizing elements already identified. A non-systematic and non-comprehensive list of key words for framework elements in alphabetic order is provided hereafter (acronyms are given in the *Nomenclature*; references are omitted for simplicity and can be found in [7], in a few cases these are cited in other parts of this paper): a) *Appendix* K to 10CFR50.46; b) ASAMPSA EC project; c) BEMUSE; d) BEPU report for Atucha-2; e) CCVM for ITF; f) CCVM for SETF; g) CFD and NRS; h) Compendium of experimental researches in thermal-hydraulics; i) Conferences selected papers (*e. g.* NURETH, NUTHOS, ICONE, ICAPP); j) CRISSUE-S EC Project; k) CSAU report; l) CT on SBLOCA; m) Databases (NPP, experiments, *etc*.); n) FFTBM for accuracy quantification; o) H2TS; p) IAEA SRS 23; q) IAEA SRS 52; r) IAEA SSG-2; s) Internal Assessment of Uncertainty; t) IRIDM by IAEA; u) ISP in thermal-hydraulics by CSNI; v) I & C modeling; w) J NED Special Issue 1998; x) Journals selected papers (*e. g.* NED, NET, NSE, NT, STNI); y) NC flow map; z) PIE lists; aa) Proceedings of ASME V&V (2006-2017); bb) Proceedings of BE-2000; cc) Proceedings of BE-2004; dd) Proceedings of BEPU-2018; ee) Proceedings of a series of IAEA Workshops (2005-2015) dealing with DSA/PSA integration; ff) Relap5 code manuals; gg) Reports from experimental projects (BETHSY, LSTF, LOBI, LOFT, PKL, PSB, Semiscale, SPES, *etc*.); hh) Roache book on V&V; ii) Safety margins lists; jj) SOAR on TECC; kk) SRP by USNRC; ll) S-SOAR; mm) Textbooks (in thermal-hydraulics / heat transfer, mechanics, neutron physics, radioprotection, PSA, *etc*.); nn) THICKET proceedings; oo) UMS; pp) User effect; qq) USNRC RG 1.157; rr) USNRC RG 1.203; ss) V & V requirements for codes, nodalization, *etc*.; tt) Wilks' formula; uu) 2-D/3-D Program.

Considering of the BEPU complexity, a specific multi-computer, multi-screen, control-room type of system has been designed and constructed, [27]; the driving idea at the basis of the system design is as follows: all identified elements are implemented to identify interfaces and to allow related interactions.

SAMPLE APPLICATIONS

Several BEPU applications have been completed during a nearly 20-year history. Most applications deal with large break loss of coolant accident and a few ones with the part of the remaining scenario of the DBA list. Interactions between licensor and licensee are part of applications and motivations and targets for BEPU application may be different. The comprehensive description of any individual application is well beyond the scope of the paper; however, selected lessons learned may provide an idea of the BEPU strengths and are outlined hereafter.

Angra-2 PWR LBLOCA

Angra-2 is an about 1400 MWe PWR in Brazil. The large break LOCA, double ended guillotine break is concerned in this application, hereafter named A2-LBLOCA. The industry (or the licensee) realized that a conservative analysis would have prevented the compliance with safety requirements under the condition of high linear power for core rods. Therefore, a pioneering application of BEPU was proposed to the regulatory authority (or the licensor) in the country in relation to LBLOCA. The licensor requested the cooperation of international institutions including independent assessment, *e. g.* [4]. This is a BEPU application directly connected with the licensing process.

A large impact upon the calculation results of safety- relevant parameters, *e. g.* peak cladding temperature (PCT) was originated by the models (nodalization) of the upper plenum region of the vessel and the assumption about mixing in the core, namely between the hot channel and the surrounding channels.

A2-LBLOCA BEPU outcomes and messages

High core power density is allowed based on independent BEPU analysis.
The iIndustry makes available recent proprietary findings relevant to safety.

Kozloduy-3 VVER-440 LOCA

Kozoludy-3 is an about 440 MWe VVER in Bulgaria. Two BEPU applications concerned with LOCA are discussed hereafter named KZ3-LOCA and KZ3-LBLOCA, respectively. Both applications are performed under the responsibility of the industry (utility owner of the NPP) to support the licensing process; both applications allowed the continuation of operation of Unit 3 (and of Unit 4) for a few years.

The former application was triggered by the information that Cathare code predicted PCT higher than the values predicted by Relap5 code. A detailed analysis performed with both codes demonstrated, as expected for qualified calculations, that one best estimate code prediction is bounded by the uncertainty bands of other code prediction, [28].

KZ3-LOCA BEPU outcomes and messages

One BE code prediction is bounded by BEPU prediction of another code.

The error associated with nodalization details is part of the uncertainty.

Splitting "time" and "quantity" error is effective (CIAU methodology).

The latter application was triggered by the following question: *Can you demonstrate that a calculation with conservative input values is conservative in terms of prediction of safety parameters*?. The answer shall be based upon a BEPU calculation, *i. e*. demonstrating that qualified and suitable safety margins are predicted by the conservative calculation. The analysis, [29], brought to the distinction between *driven* and *rigorous* conservatism: the first case conservatism in input parameters is driven to prevent calculation results above the licensing limit; in the second case, conservatism in input parameters is fixed on the basis of available information and consequent calculation results overpass the licensing thresholds, *i. e*. the NPP unit is not operable at the concerned power level.

KZ3-LBLOCA BEPU outcomes and messages

Driven and rigorous conservatism shall be distinguished.

Driven conservatism is arbitrary and rigorous conservatism does not allow the licensing of the NPP Unit.

Weak or no connection exists between conservatism in input parameters and errors associated with uncertainty origins

Smolensk-3 RBMK various accident scenarios

Smolensk-3 is an about 920 MWe RBMK, channel type reactor, boiling water-cooled, graphite-moderated in Russia. Various accident scenarios are concerned in this application, hereafter named SM3-DBA; no systematic uncertainty evaluation is performed for each BE code calculation. This is a BEPU application supporting the license renewal process. Following the engagement of about 50 scientists during four years, various research findings have been achieved which have been of use to the industry. The key result is that available safety margins in RBMK are not different from safety margins which characterize other water-cooled reactors, at least concerning AA, [30].

SM3-DBA BEPU outcomes and messages

Safety of RBMK for AA is equivalent to the safety of other wate- cooled reactors.

The rupture of one channel does not propagate to other channels (MPTR issue).

Implementation of Individual Channel Monitoring may further improve safety.

Balakovo-3 VVER-1000 AM (Accident Management)

Balakovo-3 is an about 1000 MWe VVER-1000 in Russia. Various long-lasting accident scenarios are concerned in this application, hereafter named BA3-AM. This is a BEPU application supporting the safety upgrading of the Unit. The accident management for VVER-1000 is concerned part of a project supported by EC-EURATOM. The project included a chain of BEPU activities including design of experiments, demonstration of scaling capabilities for the PSB ITF and a system code, V&V for codes and nodalizations of in relation to VVER-1000, execution of experiments, analysis of experimental data and evaluation of Balakovo-3 NPP performance in case of station blackout with loss of on-site and *off-site* power. Accident management procedures were investigated utilizing all major water reserves available during nominal operation of the unit. The applicability of codes to VVER conditions was demonstrated first. The main achievement is the demonstration of core integrity for more than 11 hours following the full loss of electrical power and based on the water on the side, [31].

BA3-AM BEPU outcomes and messages

Core integrity ensured for 11 hours following the adoption of properly designed AMP.

Scaling capabilities demonstrated by the design of CT in PSB ITF and measured data analysis.

Codes qualified for PWR are qualified for VVER, too.

SG depressurization and injection of FW tank water at the basis of the designed AMP.

Peach Bottom-2 BWR TT (Turbine Trip)

Peach Bottom-2 is an about 1000 MWe BWR in the US. The turbine trip scenario is concerned in this application, hereafter named PB2-TT. This is a BEPU application to establish available safety margin for the Unit. A TT experiment was performed in Peach Bottom-2 Unit starting at about 60 % power. Fission power pulse was measured a consequence of the pressure wave entering the vessel through the steam line. The BEPU application implied the use of a coupled system thermal-hydraulics – three-dimensional neutron physics code. The coupled code demonstrated its capability in predicting the NPP scenario, [32].

PB2-TT BEPU outcomes and messages

Coupled code is capable to predict pressure and fission power rise.

Safety margins are quantified in relation to nuclear fuel integrity.

Impact of condenser bypass upon system response is calculated.

Atucha-2 PHWR DBA envelope

Atucha-2 is an about 800 MWe PHWR in Argentina. All transients parts of Chapter 15 of the FSAR are concerned in this application, hereafter named AT2-DBA. This is a BEPU application part of the licensing process of the Unit. About 50 scientists working during more than six years in cooperation with the owner utility (the licensee) contributed to the issue of Chapter 15 of the FSAR. The BEPU was adopted for the entire spectrum of accidents, [5] and [33]: a BEPU report including the plan for the use of a dozen coupled codes was submitted and approved by the Regulatory Authority (the licensor) before its application. As key results, the existence of suitable safety margins was demonstrated in relation to all accident scenarios and the licensing for full power operation was granted to a reactor with positive void coefficient, [34], see also [35].

AT2-DBA BEPU outcomes and messages

BOT (break opening time) controls FPE (fission power excursion (no FPE if BOT 0.2 s).
BOT connected with BIT (Boron Injection Time).
The need came out to built-up and to operate a scale 1:1 SETF to confirm acceptable BIT values.
Propagation of depressurization wave from break causes steam in core channels, then FPE with timing in the order of 10 ms, different in various parts of the core.
Mechanical loads upon internals were evaluated and did not imply safety concerns.
Pressurized Thermal Shock (PTS) is not a safety issue with BEPU (PTS poses NPP lifetime limitations in case a conservative analysis is adopted).
CCFL (Counter-Current Flow Limitation) due to HL injection does not cause harm to core cooling.
Containment leakages and meteorological conditions considered for radiological impact.

INNOVATION AND PERSPECTIVES

BEPU is an established approach in nuclear technology and led to an impressive improvement in safety understanding as discussed in section *SAMPLE APPLICATIONS*. Still, developments and innovations are part of the BEPU nature. Hereafter, innovation for key BEPU elements and perspectives for the application are distinguished.

Innovation in BEPU elements

Innovations and related interrelations are discussed which are the outcome of recent activities:
(1) List of thermal-hydraulic phenomena.
(2) Standpoints from scaling activity.
(3) Moving V&V towards V&V&C where 'C' stands for Consistency.

Phenomena list and description

A list of 116 phenomena has been derived, [36], with the description provided in ref. [37]. Namely, the new list embeds phenomena from [18] and [19] and takes into account of phenomena discussed in reports by OECD/NEA and IAEA issued till 2017. A picture for the activity which brought to the new list and implication for BEPU can be derived from fig. 2.

Phenomena are derived based on the features of WCNR, the envelope of DBA and the performed experiments (left in the sketch of fig. 2). A dozen reports issued by IAEA and OECD/NEA (all those reports are listed in ref. [36]) are considered in addition to 13 WCNR types and 47 accident sequences (top central in the sketch of fig. 2). Noticeably, phenomena ex-

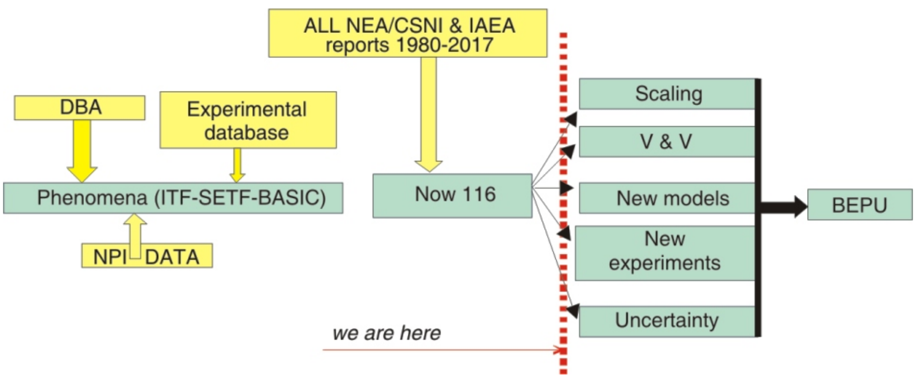

Figure 2. Activity framework for the new list of phenomena in nuclear thermal-hydraulics

pected in reactors equipped with passive systems (including Small Modular Reactors SMR) are included in the list. The gathering process ended with a list of 116 phenomena (central part in the sketch of fig. 2). The new list, possibly to be furtherly endorsed by the international community, might have a huge impact in terms of needed resources for BEPU: scaling and V&V are challenged; the development of new models and the plan of new experiments may become necessary; applicability of uncertainty methods may need a revision (right side in the sketch of fig. 2).

Standpoints from scaling activity

Following a long incubation and activity time (this might be considered as long as 10 years) the results of an attempt to achieve a common understanding in relation to scaling by regulators, industry, research leaders and scientific community was published in 2017, [8] (S-SOAR, already cited above), with two dozen specialists directly involved. The endeavor is formalized in a 400 pages textbook-type of document: any synthesis may reveal inadequate because of the complexity of the topic and the need in each statement part of the document to reflect the views from all the involved sides. Hereafter, a few BEPU-relevant messages are reported from the S-SOAR, as my interpretation:

Scaling activity within licensing does not necessarily coincide with scaling aimed at the design of test facilities or experiments: separate scaling roadmaps have been derived.

No reduced scale (model) experimental data can be meaningfully extrapolated to predict full scale (prototype) performance.

A system thermal-hydraulic code shall be considered as the most effective analytical tool to transpose experimental data and evidence necessarily obtained at a distorted (or small) scale to NPP conditions: the code quality level should be checked at any available scale including the demonstration of acceptability of code performance; the same code should be used with proper confidence (*i. e.* taking into account of errors and uncertainty) at the NPP conditions. Techniques for the code applications to scaling are discussed.

Hierarchy of importance or validity for experiments is established: all experiments are worth; however, priority should be given to large scale, full pressure, full height, full linear power and (as far as possible) resulting unity timescale, experiments.

Moving towards V&V&C

The V&V constitutes a powerful element of BEPU and the nuclear thermal-hydraulics framework provided a challenging environment for the development and the application of V&V techniques. Recently, when reassessing the predictive capability of current codes concerning two-phase critical flow (TPCF), FONESYS network, [16], the idea came to booster current V&V, [38]. The Consistency, *C*, was introduced.

The bases for the *C* proposal are: (a) an expanded role for V&V is foreseen, *i. e.* not only proving model capabilities, but challenging model development and capabilities; (b) bypassing the limitation of a deficient experimental database, both due to unavoidable errors and narrow parameter ranges; (c) achieving (better) the convergence of results for the same phenomenon by different models.

Therefore, rudimentary definitions for *C* are proposed:

Consistency is an activity connected with the development and the qualification of numerical codes which covers topics not considered by current V&V.

Consistency aims at filling gaps in the area of V&V in nuclear thermal-hydraulics, with focus

– to connect the modeling features and technological needs,
– to consider the limitations of the experimental database, and
– to streamline the conditions for developing more powerful models'.

The implementation of *C* needs a complex procedure, outlined in [38], which may need additional resources with a BEPU approach, connecting with the 116 phenomena and with the scaling roadmaps discussed above.

Perspectives for BEPU application

Possible directions for advancing in BEPU applications or perspectives for BEPU approach are:

I Expanding BEPU techniques to cover the entire FSAR.

II Deciding to establish an innovative safety barrier for existing and new NPP and showing the role of BEPU in such a connection.

Introducing BEPU-FSAR

The idea of BEPU-FSAR was spread for a few years, [39], see also [10]. The related proposal can be outlined simply: BEPU-FSAR means adopting BEPU elements and techniques for any Chapter of FSAR where numerical analyses are involved. The implementation is not so simple: if we go back to the wild traveler in section 2.6 we realize that he has to climb high and harsh mountains before encountering a situation where BEPU-FSAR becomes workable. Recent advancements and insights into BEPU-FSAR can be found in [40, 41].

Although resources for implementing BEPU-FSAR are not available, science and technology competencies appear suitable: avoiding the implementation of BEPU-FSAR implies keeping heterogeneous qualities in different NRS areas.

An innovative safety barrier

Rethinking of NRS may reveal an imperative action to restore public trust toward nuclear technology. Accidents like three mile island, Chernobyl and Fukushima are not admissible. Defense in depth (DiD) and safety barriers (SB) constitute recognized concepts and correspond to physical obstacles designed to prevent radiation from impacting the environment and human beings. Unfortunately, one, or more, or all those obstacles failed during the course of the mentioned accidents. Furthermore, in recent years, the weakness of the barrier constituted by nuclear fuel clad became evident.

An activity line started with the publication of [39] and has been furtherly pursued, [42, 43]. Barriers to the release of radioactive fission products can be seen as concentric circles (indicated below as Bi with '*i*' from 0 to 5):

B0 is the intrinsic barrier constituted by the sintered UO2 powder – some technologists never recognized this as an actual barrier (this is why the '0' index is adopted).

B1 is the Zircaloy clad of fuel pins – this is a *design-driven* mechanical barrier: related weaknesses have been recently characterized (see *e. g.* [43]). It may not be considered anymore a barrier to the release of fission products following selected DBA.

B2 is the pressure boundary for the reactor coolant - this is a *design-driven* (resistant-material-thickness based) strong mechanical barrier: the break of B2 is assumed as the origin of some DBA. Namely, the design of ESF (and ECCS) is based upon the rupture of this barrier (*e.g.* LBLOCA and Accumulator design).

B3 is the barrier constituted by the ensemble of ESF (including ECCS) systems and components – this (not usually identified as a barrier and) is a *safety-driven* dynamic barrier.

B4 is the containment this is a (resistant-material-thickness based) *safety-driven* strong barrier overpassed only in the cases of Chernobyl, 1986, and Fukushima, 2011, accidents.

B5 is the added risk-informed barrier, derived from safety analysis relevant elements it is expected that B5 replaces B1 and decreases for 1-2 order of magnitude the probability of core melt.

Five elements constitute B5: ALARA, BEPU, and IA already connected in section 2.2 above, plus the extended safety margins (E-SM) and the emergency rescue team (ERT). Then, the derivation of the set of E-SM signals, expected in the order of ten thousand, shall be supported by the execution of BEPU-FSAR and ERT is an obvious outcome from the lesson learned from the Fukushima event, [44].

The role and the importance of BEPU-FSAR for the implementation of B5 appear from the above summary outline.

CONCLUSIONS

Nuclear thermal-hydraulics and the licensing process for NPP are the roots of the BEPU approach. The multi-face features of the approach (*e. g.* multi-scale, multi-physics/science/discipline, multi-material, *etc.*) make difficult a straightforward definition for BEPU; nevertheless, this has been proposed by introducing three categories of elements: inherent elements (3 selected), characterizing elements (6 selected) and framework elements (a few dozen proposed). In particular, the ALARA can be taken as a principle inspiring BEPU, the independent assessment (IA) requirement is not possible without BEPU and quality assurance (QA) can be taken as a synonymous of BEPU.

The complexity of BEPU corresponds to (large) resources for its implementation: a suitable application to the analysis of transients in Chapter 15 of FSAR for any individual reactors may need an effort as large as 200 man-years.

A sample list of applications of BEPU performed at the University of Pisa has been presented including German, US, and Russian designed reactors; BWR, PWR, PHWR and both VVER-440 and VVER-1000 are distinguished. The outcomes from those applications, a few statements provided in each case, although not substantiated in the present paper, show the relevance of BEPU in nuclear technology: delaying a generalized BEPU use in nuclear technology and reactor safety is not justifiable.

Frontiers and directions for progressing in the area of BEPU in relation to both its bases and the application have been identified and include (supported by provided references):

(a) The newly developed list of 116 thermal-hydraulic phenomena covering the entire spectrum of DBA for existing and under design WCNR: the phenomena are needed to prove the capabilities of computer code.
(b) The roadmaps for performing scaling analyses suitable to demonstrate the applicability of concerned numerical code to the analysis of accidents in WCNR.
(c) The proposal for V&V&C. The Consistency ('C' step) has been added to the original V&V process in order to exploit any source of information and expertise suitable for improving the capabilities of computational tools (*i.e.* not only to demonstrate the validation of available models).
(d) The extension of BEPU procedures and techniques from nuclear thermal-hydraulics and AA to all technological areas part of the formal licensing process for NPP, *i.e.* the topics currently part of the

LIST OF ACRONYMS

AA	Accident Analysis	FSA	Fractional Scaling Analysis	PKL	ITF, PWR simulator
ALARA	As Low As Reasonably Achievable	FSAR	Final Safety Analysis Report	PSA	Probabilistic Safety Assessment
AM	Accident Management	FT	Fault Tree	PSB	ITF, VVER simulator
AMP	Accident Management Procedure	FW	Feed-Water	PTS	Pressurized Thermal Shock
ANS	American Nuclear Society	HL	Hot Leg	PWR	Pressurized Water Reactor
AOO	Anticipated Operational Occurrence	H2TS	Hierarchic 2 Tier Scaling	QA	Quality Assurance
ASAMPSA	EC project	HRA	Human Reliability Analysis	RBMK	Russian BWR, graphite-moderated
ASME	American Society of Mech. Eng.	IA	Independent Assessment	RCS	Reactor Coolant System
ATWS	Anticipated Transient w/o Scram	IAEA	Int. Atomic Energy Agency	RG	Regulatory Guide
BDBA	Beyond DBA	ICAPP	Series of Conferences	R&D	Research & Development
BE	Best Estimate	ICONE	Series of Conferences	SB	Safety Barrier
BEMUSE	OECD/NEA project			SBO	Station Blackout
BEPU	Best Estimate Plus Uncertainty	IET	See ITF	Semiscale	ITF, PWR simulator
BETHSY	ITF, PWR Simulator	IRIDM	Integrated Risk informed Decision Making	SETF	Separate Effect Test Facility
BIT	Boron Injection Time	ISP	International Standard Problem	SGTR	Steam Generator Tube Rupture
BoP	Balance of Plant	ITF	Integral Test Facility	SM	Structural Mechanics
BOT	Break Opening Time	I&C	Instrumentation and Control	SMR	Small Medium Reactor
BWR	Boiling Water Reactor	J NED	Journal	SOAR	State of Art Report
CCFL	CounterCurrent Flow Limitation	KM	Knowledge Management	SPES	ITF, PWR simulator
CCVM	CSNI Code Validation Matrix	LBLOCA	LB LOCA	SRP	Standard Review Plan
CD	Core Damage	LOBI	ITF, PWR Simulator	SRS	Safety Report Series
CDF	CD Frequency	LOCA	Loss of Coolant Accident	SSG	Specific Safety Guide
CFD	Computational Fluid-Dinamics	LOFT	ITF, PWR simulator	S-SOAR	Scaling -SOAR
CFR	Code of Federal Regulations	LR	Large Release	ST	Similar Test
CHF	Critical Heat Flux	LS	Length Scale	STNI	Journal
CIAU	Capability of Internal Assessment of U	MCP	Main Coolant Pump	SYS	System
CRISSUE	EC Project	MPTR	Multiple Pressure Tube Rupture	TECC	TH of Emergency Core Cooling
CSAU	Code Scaling Applicability, U	MSLB	Main Steam Line Break	TH	Thermal-Hydraulics
CSNI	Committee Safety of Nuclear Installations	NC	Natural Circulation	THICKET	OECD/NEA Project, training
CT	Counterpart Test	NEA	Nuclear Energy Agency	TPCF	Two-Phase Critical Flow
DBA	Design Basis Accident	NED	Journal	TS	Time Step
DiD	Defense in Depth	NET	Journal	TT	Turbine Trip
DSA	Deterministic Safety Analysis	NP	Neutron Physics	U	Uncertainty
EC	European Commission	NPP	Nuclear Power Plant	UMAE	U Method by Accuracy Extrapolation
ECCS	Emergency Core Cooling System	NRC	Nuclear Regulatory Commission	UMS	U Method Study
ERT	Emergency Rescue Team	NRS	Nuclear Reactor Safety	USNRC	United States NRC
ESF	Engineered Safety Features	NT	Journal	VVER	Russian PWR
EOP	Emergency Operating Procedures	NURETH	Series of Conferences	V&V	Verification and Validation
ET	Event Tree	NUTHOS	Series of Conferences	V&V&C	V&V and Consistency
E-SM	Expanded Safety Margins	OECD	Organization for Ec. Coop. & Developm.	WCNR	Water Cooled Nuclear Reactors
FFTBM	Fast Fourier Transform Based Method	PCT	Peak Cladding Temperature	2-D/3-D	SETF program
FONESYS	Network of TH code developers	PHWR	Pressurized Heavy Water Reactor		
FPE	Fission Power Excursion	PIE	Postulated initiating Event		

entire FSAR with the main reference to those which require an analytical investigation. It is proposed to built-up what is called BEPU-FSAR.

Tight interconnections exist among the identified directions for development.

Finally, BEPU is the key component for the attempt/proposal to deepen the protection of human beings and the environment against radiations, *i.e.* the outline for an additional safety barrier. The need for the additional safety barrier came from the recent characterization of nuclear fuel weaknesses and from considering the complexity of systems and components, including I&C, which are part of current NPP; the BEPU approach constitutes the logical framework for the development and the design of the new safety barrier.

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We apologize for the great number of acronyms introduced in the text: those acronyms are defined in the table.

AUTHORS' CONTRIBUTIONS

The activity bringing to the paper has been performed during more than forty years engagement of both authors in the field of nuclear thermal-hydraulics design and safety including accident analysis in water-cooled nuclear reactors. Concerning the present manuscript, the first author wrote a draft text that has been amended by the second author.

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Франческо ДАУРИЈА, Ђорђо ГАЛАСИ

МЕТОДА НАЈБОЉЕ ПРОЦЕНЕ СА НЕСИГУРНОШЋУ ЗА ПРИМЕНУ У СИГУРНОСТИ И ЛИЦЕНЦИРАЊУ НУКЛЕАРНИХ РЕАКТОРА
Кратка историја и елементи после лиценцирања

Метода најбоље процене са несигурношћу представља истовремено приступ, процедуру и оквир у нуклеарној термохидраулици, сигурности нуклеарних реактора и лиценцирању нуклеарних реактора. Побуда у основи најбоље процене са несигурношћу лежи у недостатку знања у областима једнофазних и двофазних прелазних стања у термохидраулици. Другим речима, и уз увођење неких поједностављења, недовољно познавање турбуленције намеће нацрт за примену несавршених (термохидрауличких) модела за процену својства дизајна и сигурности комплексних технолошких инсталација или система као што су нуклеарне електране и посебно нуклеарни реактори хлађени водом. Додатно, разматрано је и лиценцирање као правни аспект сигурности нуклеарних реактора, тако да метода најбоље процене са несигурношћу мора узети у обзир и правила која потичу из основних принципа заштите од зрачења о минимизацији утицаја зрачења на људе и животну средину у свим околностима. У овом раду идентификовани су и окарактерисани кључни елементи ове методе. На њих ће се гледати као на подршку доследној примени термохидраулике у дизајну и сигурности нуклеарних реактора хлађених водом.

Кључне речи: најбоља процена, оцена несигурности, најбоља процена са несигурношћу, процес лиценцирања, озакоњење, размера