

BEPU AND LICENSING OF NUCLEAR POWER PLANTS

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ABSTRACT

There are different options on accidents calculation area by combining the use of computer codes and data entry for licensing purposes. One is the Best Estimate Plus Uncertainty (BEPU), which considers realistic input data and associated uncertainties. Applications of BEPU approaches in licensing procedures were initiated in the 2000s, first to analysis of Loss of Coolant Accident (LOCA), and then to the accident analysis as a whole, documented in Chapter 15 of the Final Safety Analysis Report (FSAR). The FSAR integrates both the licensing requirements and the analytical techniques. The licensing process of a nuclear power plant is motivated by the need to protect humans and the environment from ionizing radiation and, at the same time, sets out the basis for the design and determining the acceptability of the plant. The analytical techniques can be applied by using a realistic approach, addressing the uncertainties of the results. This work aims to show an overview of the main analytical techniques that can be applied with a Best Estimated Plus Uncertainty methodology, which is 'the best one can do', as well as the ALARA (As Low As Reasonably Achievable) principle. Moreover, the paper intends to demonstrate the background of the licensing process through the main licensing requirements.

1. INTRODUCTION

Demonstration of the safety of Nuclear Power Plants (NPP) is an essential and fundamental requirement for the construction and operation of the plant. When performing the licensing calculations it is expected that availability of the safety and control systems is defined in a conservative way, including the assumption of the single failure and loss of off-site power.

However, uncertainty of the best estimate calculation has to be quantified and considered when comparing the calculated results with the applicable acceptance criteria [1].

The two methods, namely conservative and best estimate can be employed in safety assessment of NPP. These two options are divided in four categories; see Table 1 [2].

N	Option	Computer code	Availability of systems	Initial and boundary conditions
1	Conservative	Conservative	Conservative assumptions	Conservative input data
2	Combined	Best estimate	Conservative assumptions	Conservative input data
3	Best estimate	Best estimate	Conservative assumptions	Realistic plus uncertainty: partly most unfavorable conditions
4	Risk informed	Best estimate	Derived from probabilistic safety analysis	Realistic input data with uncertainties

Table 1 - Possibilities to perform accident analysis.

The first option listed in Table 1 is based upon the use of conservative computational code together with conservative boundary and initial conditions. The second option implies the use of a Best Estimate (BE) code with conservative boundary and initial conditions applied. The third option represents the Best Estimate Plus Uncertainty (BEPU) methodology, which adopts BE code and “realistic” boundary and initial conditions. In this case, the uncertainty quantification of computational simulations is required. The last option is called “Risk informed” and is considered nowadays as a future option for safety assessment of NPP. Similarly to the option 3, it is based upon the use of BE computational tools with “realistic” boundary and initial conditions applied. The main difference is the use of Probabilistic Safety Analysis (PSA) methods to quantify availability of the safety and control systems (so-called “risk-informed approach”) [2].

The application of BEPU methodology to nuclear reactor technology and, in particular to the safety analysis within the licensing process, implies availability of mature and qualified computer codes (e.g. SYS-TH, CFD, etc.) that are able to simulate accurately a wide spectrum of complex single- and two-phase flows and heat transfer phenomena envisaged to occur in Light Water Reactor (LWR) systems under normal, off normal and accidental conditions. Uncertainty quantification of computational simulations is the major processes for assessing and quantifying the confidence of performed analysis, and constitute the basis of the BEPU approach for licensing application, see Figure 1.

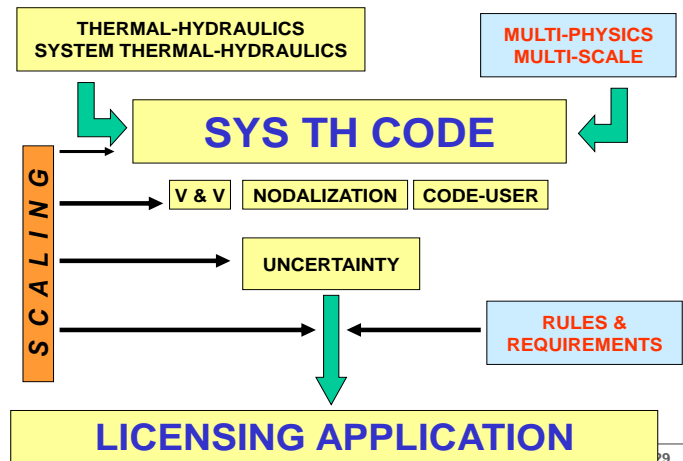


Figure 1- Key elements of BEPU approach.

One of the main objectives of BE approaches with uncertainty quantification is to reduce the level of conservatism in performed safety assessment calculations. The key goal for BEPU in nuclear reactor safety is to provide the analyst with the safety margins according to the latest best estimate methods properly qualified, and regarding to the nuclear reactor design is the consideration of the best available techniques and the identification of the errors associated with their use.

An uncertainty analysis consists of identification and characterization of relevant input parameters (input uncertainty) as well as of the methodology to quantify the global influence of the combination of these uncertainties on calculated results [1]. These two main items are treated in different ways by the various methods, such as, for instance, propagation of input uncertainties or extrapolation of output uncertainties [3].

Uncertainty quantification has been used mainly in two different areas, generally aiming at investigation of the effect of various input uncertainties on the results calculated with the complex thermal-hydraulic codes, and of performing uncertainty analyses for licensing purposes [3].

This work aims to show an overview of the main analytical techniques that can be applied with a Best Estimated Plus Uncertainty methodology, which is ‘the best one can do’, as well as the ALARA (As Low As Reasonably Achievable) principle. Moreover, the paper intends to demonstrate the background of the licensing process through the main licensing requirements.

2. LICENSING

Licensing is motivated by the need to protect humans and the environment from ionizing radiation and, at the same time, sets out the basis for the design and determining the acceptability of nuclear installations. The licensing is the process that guides the life of the NPP from the conceptual design to decommissioning. The licensing objective is to demonstrate the capability of safety systems to maintain fundamental safety functions and it is supported by the International Atomic Energy Agency (IAEA) General Nuclear Safety Objective, which is “to protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defenses against radiological hazards” [4].

A Safety Analysis Report (SAR) should provide the demonstration that the safety objective is met, and it is seen as the compendium of all the information concerning the safety of the plant. The Safety Analysis is part of the licensing process and is documented in the Final Safety Analysis Report (FSAR). The FSAR is composed by 19 Chapters, covering all the information important for the safety of the plant, from the characteristics of the site where the plant will be constructed to the commissioning and the training of the employees [5][6]. The Chapter 15 is where the accident analysis is documented and nowadays is where the designers put the most of effort.

In all countries which use nuclear energy for power production, safety analysis has to be performed and documented in the FSAR (as well as all the important characteristics of the plant), which is reviewed and/or approved by the national regulator. The FSAR should have a predefined structure and content, approved procedures and methodologies, brought out by the regulator by requirements in the form of guides, rules and recommendations.

2.1 Licensing Requirements

For the operating of a commercial nuclear power plant in the United States, and in all the countries with NPP operated by Westinghouse, a license from the United States Nuclear Regulatory Commission (NRC) is necessary. Among other things, the NRC is responsible for licensing and regulating the operation of NPPs [7].

Requirements for obtaining an operating license are observed in the NRC's regulations, which prescribe a two-step process involving issuance of a construction permit and an operating license, according to the 10 Code Federal Regulation Part 50 (10 CFR 50) [8]. An application for a construction permit must contain three types of information: (1) preliminary safety analyses, (2) an environmental review, and (3) financial and antitrust statements. Operating License Final design information and plans for operation are developed during the construction of the nuclear plant. The applicant then submits an application to the NRC for an operating license. The application contains a Final Safety Analysis Report and an updated environmental report. The Safety Analysis Report, as mentioned before, describes the final design of the plant, the safety evaluation, the operational limits, and the anticipated response of the plant to postulated accidents, and the plans for coping with emergencies [7].

In 1989, the NRC established new alternatives for nuclear plant licensing under 10 CFR Part 52, which describes a combined licensing process, an early site permit process, and a standard plant design certification process. An application for a combined license may incorporate by reference a standard design certification, an early site permit, both, or neither [7].

On the one hand the set of Code Federal Regulations are requirements binding on all persons and organizations who receive a license from NRC to use nuclear materials or to operate nuclear facilities and on the other hand there are Regulatory Guides and NUREGs, which play an important role in dealing with recommendations of construction and operation of NPP.

The Regulatory Guides are organized into divisions, which include: Power Reactors (1); Research and Test Reactors (2); Fuels and Materials Facilities (3); Environmental and Siting (4); Materials and Plant Protection (5); Products (6); Transportation (7); Occupational Health (8); Antitrust and Financial Review (9); and General (10). The Regulatory Guide 1.206 - Combined License Applications for Nuclear Power Plants (Light Water Reactor Edition) [5] deals with the content of the FSAR and the information is reflected in the NUREG-0800 [9], which, in turn, is guidance to

NRC staff in performing safety reviews. Both documents contain a description of the content of the 19 chapters of the FSAR.

2.2 BEPU and Licensing

The application of BEPU methodology for licensing purposes is originated from the calculations of Large Break Loss-of-Coolant Accident (LB-LOCA) scenario. Later, this methodology was adopted for analysis of Small Break LOCA (SB-LOCA), as well as for operational transients [10]. Some examples of industrial applications of the BEPU methodology are provided below.

The US Westinghouse developed and licensed a best-estimate LB-LOCA methodology for three- and four-loop designs in 1996 and, later, extended the methodology to two-loop upper plenum injection plants [11].

In France, an accident analysis method was developed based on the use of realistic computer codes called Deterministic Realistic Method (DRM), found on qualification of the calculation uncertainty, which is taken into account deterministically when the results are compared to the acceptance criteria. The DRM was first applied in 1997 to LB-LOCA for a French three-loop pressurized water reactor [12].

In Brazil, the uncertainty analysis of SB-LOCA scenario in Angra-1 NPP was an exercise for the application of an uncertainty methodology. For Angra-2, a LB-LOCA analysis was performed and the treatment of uncertainties was carried out separately in three basic categories: code uncertainty (statistical quantification of the difference between calculated and measured parameters); plant parameters uncertainties (statistical variations); and fuel uncertainty parameters (statistical variations) [13] [14].

For the licensing process of the Atucha-II NPP in Argentina, the BEPU approach was selected and applied to the Chapter 15 of FSAR “Transient and Accident Analysis” in 2008. Thus, the BEPU methodology has been adopted covering the established spectrum of Postulated Initial Events (PIE), wherein procedures have been applied to identify the list of PIE and applicable acceptance criteria, and the application of computational tools produced results related to the Atucha II transient scenarios originated by the PIE [15].

BEPU approach includes the use of the most recent analytical techniques, the existence of validated computational tools, and the characterization of expected errors or the evaluation of uncertainty affecting the results of application.

As defined in Title 10, Section 20.1003, of the Code of Federal Regulations [16] ALARA means making every reasonable effort to maintain exposures to ionizing radiation as far below the dose limits as practical, consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest.

The ALARA principle shall be taken at the origin of BEPU: the as Low as Reasonably Achievable shall be translated into as Accurate as Reasonably Achievable in the case of BEPU [17], and this relation should be the starting point to the connection between BEPU and Licensing.

3. ANALYTICAL TECHNIQUES

Analytical techniques dealing with NPP are the set of methodologies, code computers and approaches to development analysis that ensure the reach of the acceptance criteria and consequently ensure the integrity of barriers to the release of radioactive materials. These analytical techniques are applied to the safety analyses and are documented in the FSAR to demonstrate that the plant is safe.

There is variety of codes that allows predicting the response of the NPPs during accident conditions. In the last decades, several complex system codes have been developed with proven capabilities for simulating the main thermo-hydraulic process that occurs during transient conditions. Originally, system thermal-hydraulic codes were used to support the design of safety systems, but since the publication of the 10 CFR 50.46, in 1978, they start to be applied widely in the licensing process. In parallel, especially after the TMI-2 accident, several realistic or so-called BE codes started being developed in order to switch from the previously-used conservative assumptions to more realistic description of the processes. Since then, BE system codes are used to perform safety analysis of the NPP during accident scenarios, uncertainty quantification, PSA, reactor design, among others. Some examples of BE codes are RELAP5, TRAC, TRACE, CATHARE, and ATHLET [18].

The term Probabilistic Safety Assessment (PSA) has been in use since the issue of the WASH-700 (subsequently WASH-1400) in the early 70's [20]. Three PSA levels are distinguished to estimate the risk. Those levels cover the probability and the consequences (i.e. the radiological impact) of faulting events at any time of the NPP life. Noticeably, the calculation of consequences can only be performed by using Deterministic Safety Assessment (DSA) tools [1].

The term DSA is associated with the availability of qualified BE computational tools or codes, and it has been in use since the 90's. However, conservative DSA constitutes key practice for the design and the safety confirmation of existing reactors. On the other hand, uncertainty is the key-word for the application of BE codes. Both DSA and PSA are needed for the issue of a consistent Safety Analysis Report (i.e. primarily chapters 19 and 15 of the generally accepted FSAR structure). Furthermore, a variety of interactions are envisaged and do exists between the two NST categories [1].

The Risk Informed (regulation) framework or concept was spread into the international nuclear safety community since the '90's: the idea is that the relevance of any action or any component or structure connected with the NPP, including the numerical analyses, shall be evaluated based on its impact upon the safety (or risk). Recently a more robust architecture for the same idea has been formulated [1].

Figure 2 shows the pyramid of licensing competence. As discussed before, the result of a licensing process is the Safety Analysis Report approved and at the bases of the process there are laws, i.e. CFR in the case of US. In-between the bottom and the top there are subjects like Risk Informed Concept, PSA and DSA, Option 3/Option4, and BEPU [1].

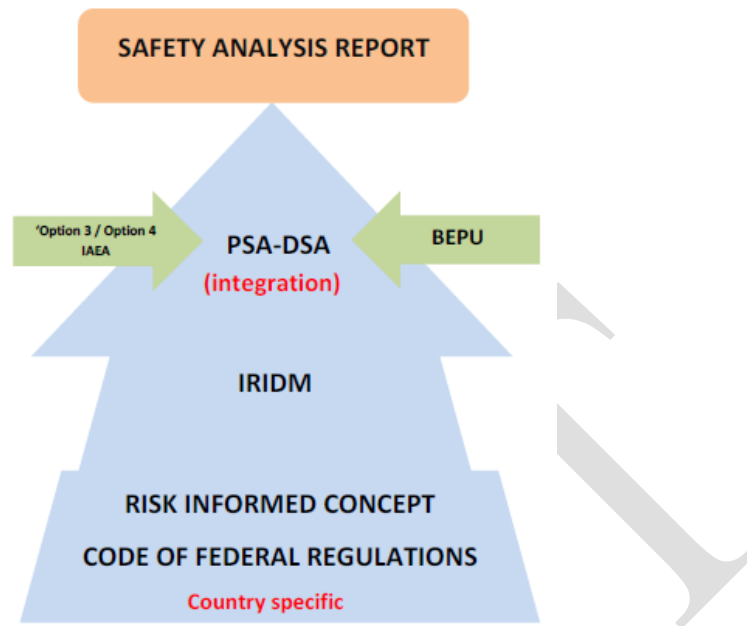


Figure 2 - The pyramid of competence and the licensing process.

During the last decade, attempts were made to integrate DSA and PSA based on the organization of devoted workshops open to specialists in both areas. This is interpreted as the top (or the tip) of the pyramid of competence in the joint area of DSA and PSA. The so-called IAEA ‘Option 3’ or ‘Option 4’ for performing accident analysis may constitute the framework or can provide the bases for the integration between PSA and DSA [1].

4. CONCLUSION

The application of BEPU methods were carried out in several countries; however, the framework to introduce the BE analysis, as well as BEPU methodology, into the licensing process is still an open issue. Notwithstanding over the years more and more applications have proven to be satisfactory, since BE analysis with the evaluation of uncertainties is the only way to quantify existing safety margins, even uncertainty evaluations being considered as a need to improve practicability of methods.

The description of BEPU methodology in nuclear reactor safety and licensing process involves a wide variety of concepts and technological areas. Notwithstanding the considerable growth of BEPU applications over last decades, there is still a margin for further improvements.

Some problems can be associated and addressed within the historical licensing process as high cost, reluctance to innovation and lack of homogeneity. Nowadays, the licensing process is based on a non-homogeneous interpretation of licensing requirements, engaging different groups of experts without coordination, resulting in a lack of homogeneity. Assembling the top level competence in relation to each of the listed topics and disciplines, on the one hand there is an obligation and importance to demonstrate the safety of any nuclear installation and on the other hand there is the difficulty to address the safety in a holistic way.

The idea of a BEPU as a licensing tool is connected with the use of BEPU for qualified computational tools and methods as well as for the analytical techniques that are presented in FSAR. The qualified analytical techniques shall be adopted together with the latest qualified findings from the technology research, thus homogenizing what is in the concern to the safety of nuclear power plants: the analyses including calculation process, not only limited to accident analysis, but all the analysis included on FSAR. For this purpose, it is necessary to establish connections between safety analysis and hardware of the NPP, starting from the connections between the chapters and the disciplines.

The BEPU implies the best exploitation of analytical-numerical techniques consistently with the needs of NPP design and safety. Licensing constitutes an essential component for BEPU once the set of rules and acceptance criteria is established and contributes to ensure the quality in BEPU applications. From the other side, the BEPU approach implies a complexity which is consistent with the complexity of the objective system, the NPP.

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