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Application of Best Estimate Plus Uncertainty (BEPU) Methodology in a Final Safety Analysis Report (FSAR) of a Generic Plant.

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ABSTRACT

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. An important part of the licensing process is the realization of accident analysis related to the design basis, which should be documented in the Final Safety Analysis Report (FSAR). 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 FSAR. This work has as main objective the implementation of BEPU methodology in all analyses contained in FSAR, through the homogenization of the analytical techniques and identification of key disciplines and key topics in the licensing process.

Keywords: *Licensing, Nuclear Power Plants, Safety Analysis, Final Safety Analysis Report, BEPU*

1 INTRODUCTION

Nuclear Reactor Safety Technology (NRST) is the set of materials, components, structures, procedures and numerical tools used to minimize the risk of contamination of humans and environment by radioactive material. NRST has been established for several decades, since the discovery of nuclear fission and since that time, any installation involving the use of radioactive material has been designed according to safety requirements [1].

Nuclear safety has become a technology following extraordinary industrial investments since the 50's. A step impulse to the technology came when powerful computers were available at the beginning of the 80's [1]. Events in the last decades occurring in the Three Mile Island Unit-2, Chernobyl Unit-4 and Fukushima Units1-3 have challenged the sustainability of nuclear technology and undermined the trust of the public, of the decision makers and even of the scientific community toward nuclear safety [2].

The NRST consists of two components – the Fundamentals and the Application – as demonstrated in Figure 1. The first component includes the key safety objective, the related safety principles, and safety requirements developed by the International Atomic Energy Agency (IAEA). The Application refers to the application of those principles and requirements for the design, licensing, construction, operation and decommissioning of any nuclear installation [2].

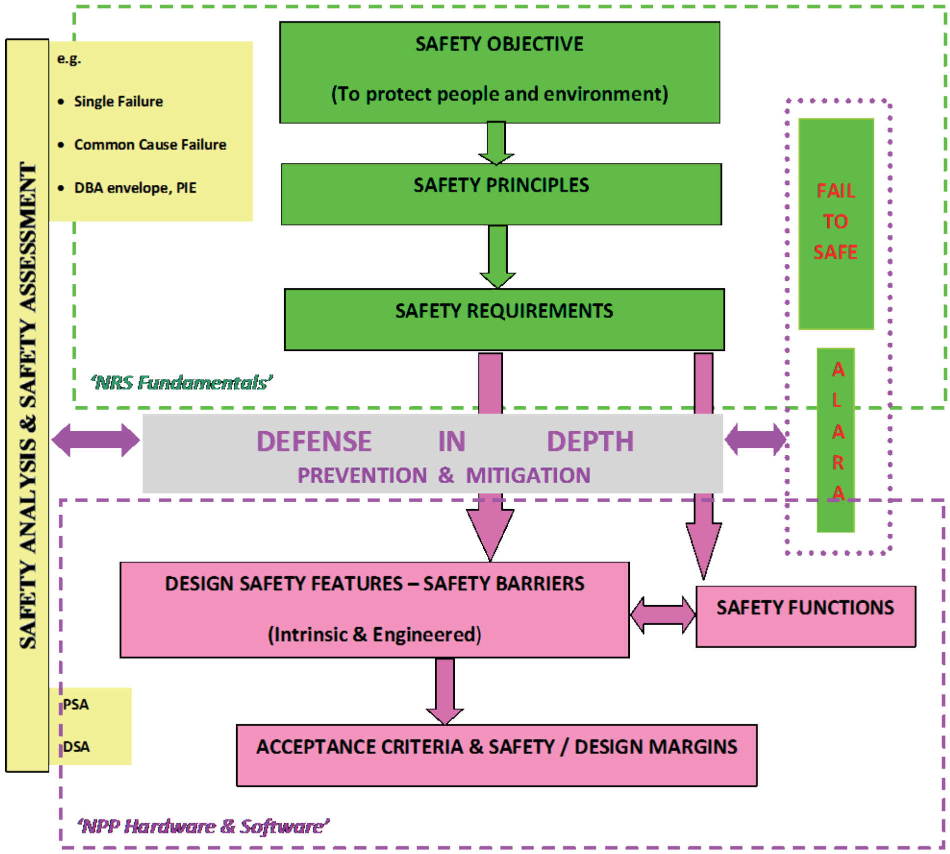

Figure 1: Simplified sketch for Nuclear Reactor Safety Technology.

The accomplishment of safety fundamentals in the Nuclear Power Plant (NPP) design is achievable by suitable safety analysis and assessment. The safety evaluation of the NPP is based on the fulfilment of a set of design acceptance criteria such as maximum peak cladding temperature, maximum pressure in the primary system, among others, to be met under a wide range of plant operating conditions to confirm the preservation of physical barriers [3].

The national regulator normally defines the acceptance criteria, and a comprehensive Safety Analysis Report (SAR) for individual NPP provides the demonstration that the plant is safe and, noticeably, that acceptable safety margins exists [2]. The SAR shall be seen as the survey of information concerning the safety of the specific NPP and includes the demonstration of acceptability of the NPP against the rules and related criteria established for the Country. The Safety Analysis is part of the licensing process and is documented in the Final Safety Analysis Report (FSAR) [3].

In all countries using 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 and approved procedures and methodologies, brought out by the regulator by requirements in the form of guides, rules and recommendations [3].

The accident analysis is an important part of a NRST and should be performed and documented on Chapter 15 – Transient and Accident Analysis, of a FSAR. The Chapter 15 includes the analysis of the following event categories [1]:

1. Increase in heat removal by the secondary side;
2. Decrease in heat removal by the secondary side;
3. Decrease in flow rate in the reactor coolant system;
4. Increase in flow rate in the reactor coolant system;
5. Anomalies in distributions of reactivity and power;
6. Increase in reactor coolant inventory;
7. Decrease in reactor coolant inventory;
8. Radioactive release from a subsystem or component.

Each category of events is typically subdivided into several events that are more specific. Events which are expected to occur during the plant lifetime are called anticipated operational occurrences (anticipated transients). They are also analyzed under the assumption of a complete failure of the fast reactor shutdown system, or Anticipated Transient Without Scram (ATWS).

There is variety of codes that allows predicting the response of the NPP during accident conditions. In the last decades, several complex system codes have been developed to simulate the main thermal-hydraulic phenomena 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 started to be applied widely in the licensing process. In parallel, especially after the TMI-2 accident, several “realistic” or so-called “Best-Estimate” (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, Probabilistic Safety Assessment (PSA), reactor design, etc. Some examples of BE codes are RELAP5, TRAC, TRACE, CATHARE, ATHLET, and others [4].

There are different options on accidents analysis area by combining the use of computer codes and input data for licensing purposes. Four options can be identified [5]:

1. Very conservative approach, shown in Appendix K of 10 Code of Federal Regulations (CFR) 50.46 (USNRC, 1974), for examination in case of Loss Of Coolant Accident (LOCA);

2. Realistic conservative approach, which is similar to the first, except for the fact that best estimate computer codes instead of conservative codes are applied;

3. Initial and boundary conditions taken as realistic considering its uncertainties. In some countries like USA this option would be to Best Estimate Plus Uncertainty (BEPU); and

4. Realistic approach considering the actual installation conditions of the operation and the use of best estimate codes.

These options are represented in the Table 1.

Table 1: Options for combination of a computer code and input data.

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 Plus Uncertainties BEPU)	Best estimate	Conservative assumptions	Realistic plus uncertainty; partly most unfavourable conditions
4. Risk informed (Extended BEPU)	Best estimate	Derived from probabilistic safety analysis	Realistic input data with uncertainties

In the last years, were performed several calculations making use of a BEPU methodology for the LOCA analysis, and most recently, for the others transients present on Chapter 15 of FSAR. However, the FSAR of a generic plant includes more eighteen chapters, totalizing nineteen. Each one relates to the others, addressing different important characteristics of the plant to insure the safety, as the location, training of the employees and meteorological aspects, for example.

Due to historical reasons, the accident analysis part of FSAR received considerable attention in the nuclear reactor safety discipline. However, a set of accidents can happen in peripheral areas or as a consequence of precursory events which can bring the NPP in conditions outside those previously considered for accident analysis. This can be easily observed by the root-causes of the major nuclear accidents, as Fukushima. Therefore, the homogenization of the FSAR topics is required, through the systematic identification of topics and their consideration for the analysis [1].

The objective of the present paper is to discuss one entire FSAR based on the BEPU methodology, through the homogenization of the analytical techniques and identification of key disciplines and key topics in the licensing process.

2 BEPU METHODOLOGY

BEPU approach is characterized by applying the best estimate code with BE initial and boundary conditions to simulate the considered event. When performing the licensing calculations, it is expected that the availability of safety and control components and systems be 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 [3].

The BEPU approach has been adopted as the methodology for accident analyses covering the established spectrum of Postulated Initial Events (PIE). Procedures have been applied to identify the list of PIE and applicable acceptance criteria. Finally, the application of computational tools including nodalizations required suitable boundary and initial conditions and produced results related to the Atucha II transient scenarios originated by the PIE. The proposed BEPU approach follows current practices on deterministic accident analyses, but includes some key features to address particular needs of the application. The BEPU-flow diagram is represented in the Figure 2, where CA means Component Analysis, SA means System Analysis and RA, Radiological Consequences Analysis [6].

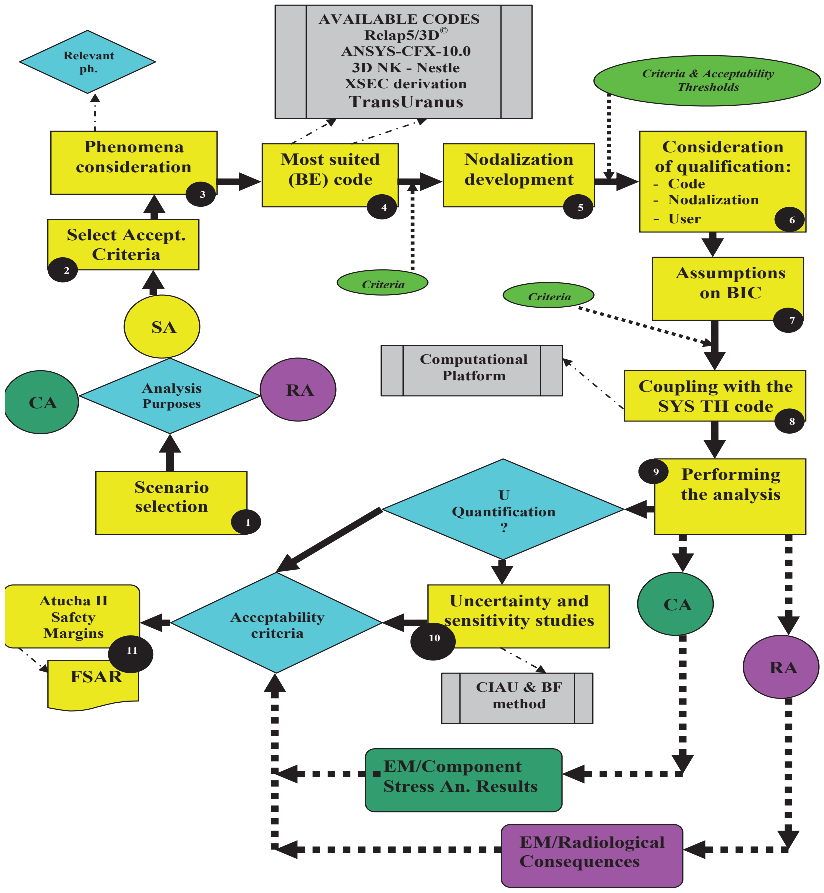

Figure 2: BEPU flow-diagram

The approach takes credit of the concept of evaluation models (EMs), and comprises three separate possible modules depending on the application purposes [6]:

- For the performance of safety system countermeasures (EM/CSA);
- For the evaluation of radiological consequences (EM/RCA);
- For the review of components structural design loadings (EM/CBA), where the acronyms CSA, RCA and CBA stand for 'Core Safety Analysis', 'Radiological Consequence Analysis' and 'Component behaviour Analysis'.

There are several methods for the BEPU application and all of them have the identification and characterization of the relevant uncertainty parameters in common as well as the quantification of the global influence of the combination of these uncertainties on calculated results [3].

BE analysis with evaluation of uncertainties is the only way to quantify the existing safety margins. 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 [7].

2.1 BEPU and 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, guiding 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 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” [8].

Nowadays, in most countries the national regulators allow the use of best-estimate codes to be applied in the licensing process. Some examples of such countries are United States (US), France, Brazil and Argentina. Initially BEPU methods were applied mainly to Large Break Loss–of-Coolant Accident (LB-LOCA). However, later these methods start also to be used for analysis of Small Break LOCA (SB-LOCA), as well as for operational transients [9].

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 [10].

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 [11].

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) [12] [13].

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 [6]. Thus, the BEPU methodology has been adopted covering the established spectrum of 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 [6].

Considering all the successful applications of the BEPU methodology for licensing purposes, it is therefore proposed therefore to extend its range of use to each area of FSAR, principally the chapters and the topics where the analytical techniques are needed.

3 BEPU-FSAR

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.

To perform an entire FSAR bases on BEPU, or so-called "BEPU-FSAR", a homogenization of the analyses is proposed, including calculation processes, that are not limited to accident analysis but cover selected topics that are connected with the design and the operation of the NPP.

Key disciplines and key topics have been defined by areas of knowledge based on the FSAR chapters, the Regulatory Guide divisions, and the IAEA Safety Standard Series. The list of key disciplines and related key topics that were derived from the FSAR content is provided in Table 2.

Table 2: Key disciplines and Key topics in the licensing process of a NPP.

Key Disciplines	Key Topics
Legal Licensing Structure	FSAR writing and assessment Knowledge of, IAEA, US NRC, ASME, ANS, IEEE frameworks of requirements Defense in Depth application
Siting & Environmental	Climatology Seismology Earthquake and Tsunami Geology including stability of slopes Hydrology and Floods Meteorology Catastrophic (including natural and man-originated) events Atmospheric diffusion Loadings Population Distribution
Mechanical Engineering: Design of Structures, Systems and Components	Structural Mechanics Thermodynamic Machinery Control Rod mechanisms
Nuclear Fuel	Nuclear Fuel performance Fuel movement
Materials	Corrosion Mechanical resistance Radiation damage Creep Analysis Fatigue Analysis Erosion
Neutron Physics	Cross Section Derivation Monte Carlo

Chemical Engineering	Chemistry of nuclear fluids Chemistry of water Metal Steam production Zircaloy reactions Boron control
Electronic Engineering	Instrumentation and Control (l & C) Nuclear Instrumentation (in-core) Ex-core instrumentation Digital systems Analogical systems
Electrical Engineering	Transformers Alternators
Civil Engineering	Containment Foundation
Deterministic Safety Analysis	Accident Analysis Computational tools Uncertainty Analysis Severe Accident Consequences
Probabilistic Safety Analysis	Reliability Cost-Benefit Analysis Severe Accident Probability Probability of Meteorite
Human Factors Engineering	Man-Machine interface Simulator Human failure
Occupational Health and Radioprotection	Radiological Protection Accessibility to remote Radioactive Zones Shielding
Physical Security	Fire protection Hazards
Plant Operation and Procedures	Emergency Preparedness Emergency Operating Procedures Plant procedures for normal operation In-service Inspection Administrative Procedures Inspections, Tests, Analyses and Acceptance Criteria
Quality Assurance[1]	Management Procedures Standards
Computational Science[1]	Information Technology Software

[1] Cross Cutting Disciplines, which are presented throughout the FSAR.

4 CONCLUSION

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.

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 the 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.

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 difficulty to address the safety in a holistic way. Therefore, the idea of a BEPU-FSAR proposal is to fill this lack by providing the homogenization of analytical techniques and thus to increase the safety of the plant.

The idea of a BEPU-FSAR 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 technology research, thus homogenizing what is in the concern to the safety of nuclear power plants: the analysis including calculation process, but not only limited to accident analysis, but all the analysis that encompass any FSAR topic. For this purpose, it is necessary to create a connection between safety analysis and the hardware of the NPP, starting from the connections between the chapters and the disciplines.

In the table with the key topics and disciplines that are dedicated to the licensing process, one can recognize areas which need specific expertise knowledge (Climatology and Instrumentation and Control, e.g.). The future steps of this work will concentrate on propagation of this expertise into the remaining areas, thus building a BEPU-FSAR in the most gradual and integrated manner, adding new knowledge and improving plant safety.

One can conclude from the finalized BEPU applications that this methodology is feasible, which encourage to extended the use for other areas and demonstrate the industrial worth and interest. Another point that should be emphasized is the main obstacle in the spread of BEPU, which consists, basically, in the needed of deep expertise, numbers of wide databases and sophistication of computational tools. A lack of expertise in many areas of a FSAR and consequently the nuclear reactor safety technology, results in a simplification of how the safety analysis is conducted nowadays.

The future steps of this work will mainly be focused on the propagation of this expertise into the remaining technical areas of FSAR, adding new knowledge and therefore creating coherent and rigorous background of the BEPU-FSAR methodology.

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