BEPU-FSAR: A NEW PARADIGM IN NUCLEAR REACTOR SAFETY

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ABSTRACT

To perform an entire FSAR based on BEPU (Best Estimated Plus Uncertainty), a homogenization of the analysis is proposed. The first step towards BEPU-FSAR requires identification and characterization of the FSAR parts where the numerical analyses are needed. The next step consists of creating a list of key technological areas where the relations between so-called key disciplines and the key topics are established. Considering the successful applications of BEPU methodology to the Chapter 15 of FSAR performed in the last two decades (Atucha II NPP, Angra 1 and 2), one can conclude that this methodology is feasible, which encourage to extended its range of use to the other technological areas of FSAR (e.g. seismology, radioprotection, etc.), and therefore to demonstrate the industrial worth and interest. The future step of this work will mainly be focused on the propagation of this expertise into the remaining technical areas of FSAR.

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].

The accomplishment of safety requirements 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 [2]. The national regulator defines the acceptance criteria, and a comprehensive Safety Analysis Report (SAR) for individual NPP provides the demonstration that the safety objective is met and, noticeably, that acceptable safety margins exists [3]. 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].

The FSAR is a compendium for the Nuclear Safety Reactor, and should be made and delivered to the appropriate regulatory body. Accident Analysis consists in a fundamental part of the licensing of the NPP, and should be documented in Chapter 15, on FSAR.

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 with proven capabilities for simulating 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 start 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 [4]:

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.

This work aims at showing the first steps toward an application of a BEPU methodology in all FSAR parts where analytical techniques are needed. The overview of a BEPU methodology is presented below.

2. BEPU Methodology

BEPU approach is characterized by applying the BE code with BE initial and boundary conditions to simulate the intended 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 [2].

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 [5].

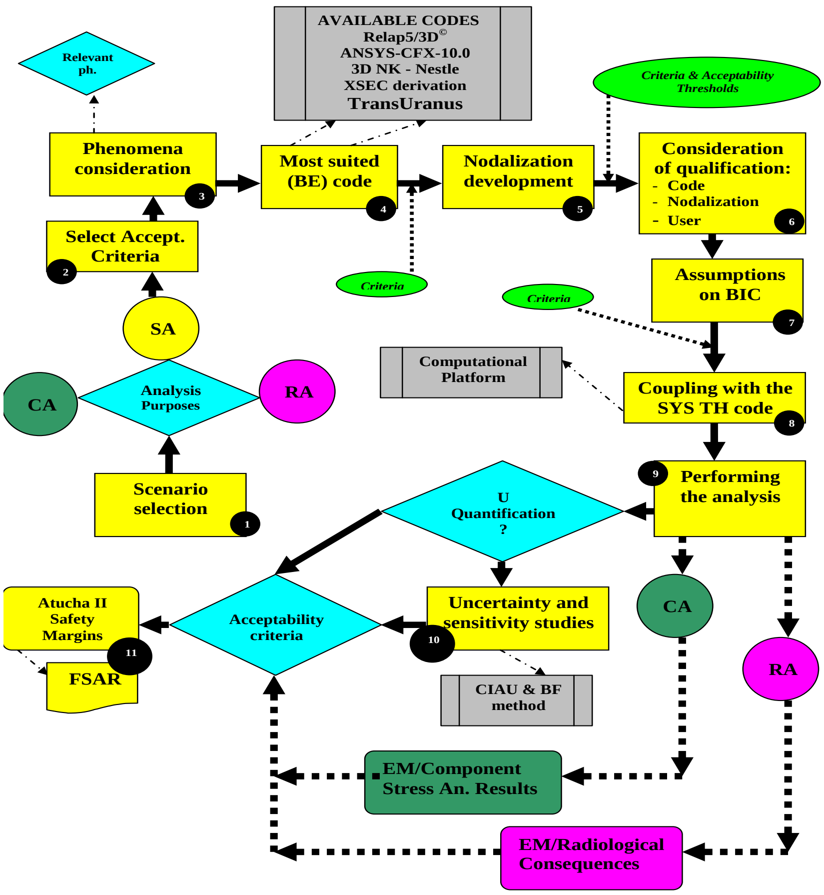

Fig 1. BEPU flow-diagram

The approach takes credit of the concept of evaluation models (EMs), and comprises three separate possible modules depending on the application purposes [5]:

- 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 [2].

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 [6].

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” [7].

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 [8]. 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 [9].

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 [10].

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) [11] [12].

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 [5]. 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 [5].

Considering all successive applications of the BEPU methodology for licensing purposes, it is proposed therefore to extend the implementation area of BEPU covering possibly all the 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.

As defined in Title 10, Section 20.1003, of the Code of Federal Regulations [13] ALARA (As Low As Reasonably Achievable) 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 ALARA shall be translated into as Accurate as Reasonably Achievable in the case of BEPU [3], and this relation should be the starting point to development of a BEPU-FSAR.

To perform a 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.

Due to historical reasons, an accident analysis received considerable attention from the side of NRS actors. However, a sort of accidents can happen in either peripheral areas or following precursory events which may bring the NPP in conditions outside those considered for accident analysis. It may be easily observed by the root-causes of the major nuclear accidents, like Fukushima. Therefore, the homogenization of NRS topics is required: it implies systematic identification of topics and their consideration for the analysis [3].

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 was derived from the FSAR content is provided in Table 1.

Key Disciplines	Key Topics
Legal Licensing Structure	FSAR writing and assessment Knowledge of, IAEA, US NRC, ASME, ANS, IEEE frameworks of requirements Defence 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 (I & 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.

Tab 1: Key disciplines and Key topics in the licensing process of a NPP

4. Conclusions

The description of BEPU methodology in nuclear reactor safety and licensing process involves a wide variety of concepts and areas. In the last decades, the use of BEPU in the licensing process has grown considerably and there is still margin for these grow.

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 BEPU-FSAR concept 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, 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.

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.

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