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USING OF BEPU METHODOLOGY IN A FINAL SAFETY ANALYSIS REPORT

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ABSTRACT

The Nuclear Reactor Safety (NRS) has been established since the discovery of nuclear fission, and the occurrence of accidents in Nuclear Power Plants worldwide has contributed for its improvement. The Final Safety Analysis Report (FSAR) must contain complete information concerning safety of the plant and plant site, and must be seen as a compendium of NRS. The FSAR integrates both the licensing requirements and the analytical techniques. 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 (BEPU) 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

Nuclear Reactor Safety (NRS) 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. NRS has been established for several decades, since the discovery of nuclear fission. 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. Efforts have been completed by the technological community following each of the disasters and ended-up in reinforcements of the Engineered Safety Features (ESF) and of Safety Barriers [2].

The NRS technology 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].

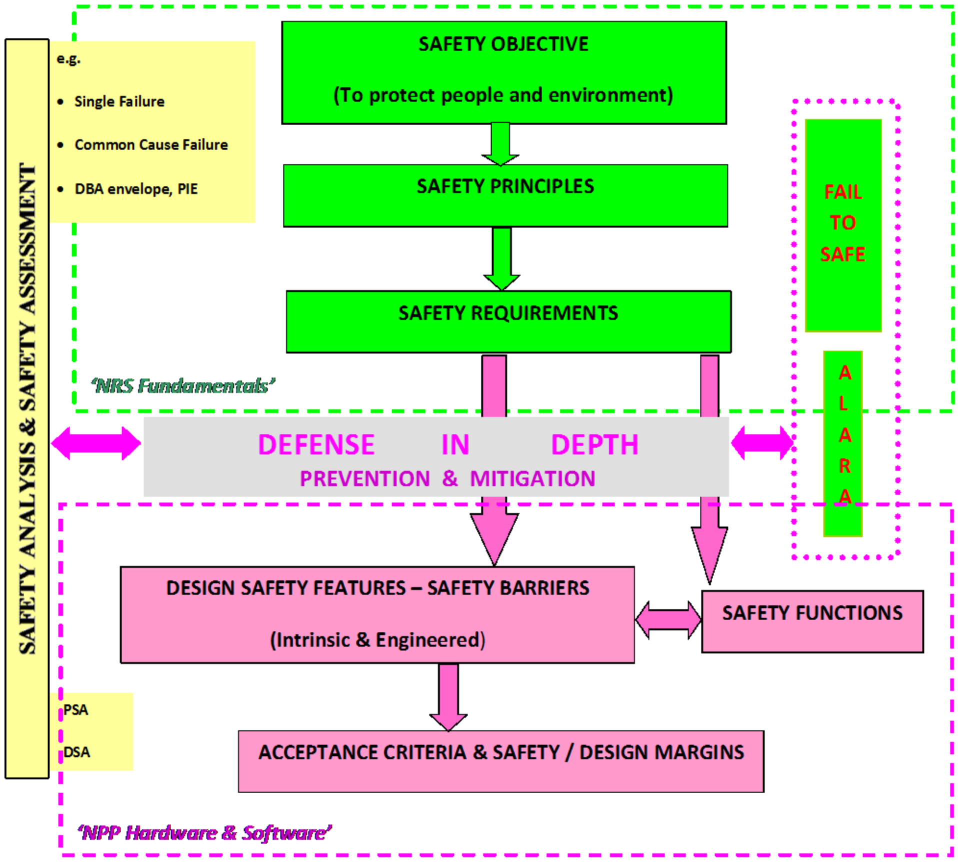

Figure 1: Simplified sketch for Nuclear Reactor Safety Technology.

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 fulfillment 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 acceptance criteria are normally defined by the national regulator, 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 [2].

The SAR shall be seen as the compendium 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) [2].

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 safety analysis tools are broadly used within the framework of the design of new plants and operation of existing plants, including licensing of new NPP projects, periodic safety reviews, development of new emergency operating procedures, analysis of operational events, among others. Significantly, increased capacities of new computation technology made it possible to switch over to the new generation of computer codes, with the use of best estimate codes with treatment of uncertainties, and coupling of computer codes [3].

The survey conducted between 1989-1995 on the evaluation methods of uncertainty led to the development and use of Best Estimated Plus Uncertainty (BEPU) for licensing. Initially BEPU methods were applied mainly to large Loss of Coolant Accident (LOCA). However, later it started to be applied for small LOCA, as well as to operational transients [4].

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. The US Westinghouse developed and licensed a best-estimate Large-Break Loss of Coolant Accident (LB-LOCA) methodology for three and four-loop designs in 1996 and, later, extended the methodology to two-loop upper plenum injection plants [5].

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 [6].

In Brazil, the uncertainty calculation for Small Break Loss of Coolant Accident (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) [7] [8].

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 [9]. 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 [9].

The objective of the present paper is to discuss one entire FSAR based on the BEPU methodology. For this purpose, an overview of the analytical techniques needed in a generic FSAR that can be applied with a BEPU methodology, which is ‘the best one can do’, as well as the ALARA (As Low As Reasonably Achievable) principle will be presented. Furthermore, the paper intends to show a background of the licensing process through the main licensing requirements, as well as the key topics and disciplines in licensing documented on FSAR.

2. LICENSING

The licensing is the process that guides the life of the NPP from the conceptual design to the decommissioning. The licensing objective is to demonstrate the capability of safety systems to maintain fundamental safety functions. The complete licensing process 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” [10].

The licensing process is constituted by NRS technology, imposed by a regulatory authority. The process follows specification and rules that are typically part of the laws of the Country where the NPP is supposed to operate or where it is designed [2].

The legal aspects and the public acceptance of nuclear installations are primarily concerned within the licensing process. The licensing process creates a contest between two main actors: the licensor and the licensee where the licensor is a government institution also identified as regulatory authority, and the licensee is any company owning or managing a nuclear installation. The licensing is a life process for any nuclear installation: licensing must follow innovations and findings from the safety technology [1].

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 U.S. Nuclear Regulatory Commission (NRC) is necessary. Among other things, the NRC is responsible for licensing and regulating the operation of NPPs [11].

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) [12]. 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 [11].

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 [11].

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) [13] deals with the content of the FSAR and the information is reflected in the NUREG-0800 [14], 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. 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.

The safety analyses were initially conservative – the Option 1 in Table 1 – , which is often called Appendix K (of 10 CFR 50), in the case of LOCA. Then after, best-estimated codes have been developed – the Options 2 and 3 emerged – depending on whether only conservative inputs are adopted or a full uncertainty evaluation is being performed (BEPU), respectively. However, it is possible, in principle, to provide a level of flexibility by using probabilistic arguments to take credit for the probability that a system, which was deterministically excluded in Options 1 to 3, is actually available, consisting in Option 4. However, this Option is not yet a part of current licensing practices. This option is connected to future developments in risk informed regulations [2]. Table 1 shows those Options.

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

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 ”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, among others. Some examples of BE codes are RELAP5, TRAC, TRACE, CATHARE, and ATHLET. [15].

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 [16]. 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, already mentioned in the present document, but that will be explained in more detail in the next section [1].

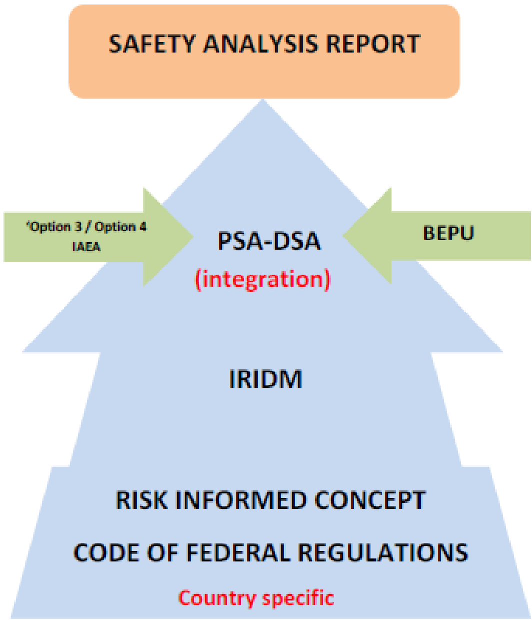

Figure 2: The pyramid of competence and the licensing process.

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].

3. BEPU

BEPU approach (Option 3 shown in Table 1) 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 [3].

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 investigating of the effect of various input uncertainties on the results calculated with complex thermo-hydraulic codes, and of performing uncertainty analyses for licensing purposes [17].

4. 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 [18] 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 [2], 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 [2].

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 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 Analog systems
Electrical Engineering	Transformers Alternators

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

Key Disciplines	Key Topics
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.

5. CONCLUSIONS

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.

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 proposal of a BEPU-FSAR, or a whole FSAR based on the application of BEPU methodology, is to fill this lack by providing the homogenization of analytical techniques and thus to increase the safety of the plant.

Through the key topics and disciplines of licensing showed in the Table 2, we can recognize some areas which need expertise knowledge (e.g. Climatology, Instrumentation and Control). 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.

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