2017 International Nuclear Atlantic Conference - INAC 2017
Belo Horizonte, MG, Brazil, October 22-27, 2017
ASSOCIAÇÃO BRASILEIRA DE ENERGIA NUCLEAR – ABEN

Agência Brasileira do ISBN
ISBN 978-85-99141-07-6

BEPU-FSAR: ESTABLISHING A BACKGROUND FOR EXTENSION OF NUCLEAR THERMAL HYDRAULIC PRINCIPLES TO NON THERMAL-HYDRAULIC CODES

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ABSTRACT

Nuclear thermal hydraulic and accident analysis are based in three pillar activities, which consists in: Scaling, Coupling and V&V. Each of them are established technology, with key documents to describe and widely used. The final goal of this work is to apply the BEPU methodology in all parts of FSAR where analytical techniques are needed (BEPU-FSAR) and for that the crucial step is the transfer of the BEPU concepts into the other areas. In this sense, the issue is how to adapt to other disciplines the pillar activities presented in the thermal hydraulic area. For that we need to identify which elements can be applied in the other areas, to show that the proposed methodology is feasible. This work aims to discuss the first steps towards a BEPU-FSAR methodology and to show that the Scaling, Coupling and V&V elements, currently done for thermal-hydraulic codes, can be also done for different codes, which are used to perform different analysis included on a FSAR of a generic plant.

1. INTRODUCTION

Demonstration of the safety of Nuclear Power Plants (NPPs) 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 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 [1]. The two methods, namely conservative and best estimate can be employed in safety assessment of NPPs. These two options are divided into four categories, as shown in the Table 1 [2].

Table 1: Possibilities to perform accident analysis

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

The first option listed in Table 1 is based upon the use of conservative computational code 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 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. Scaling, Coupling, Verification and Validation (V&V) and uncertainty quantification of computational simulations are the major processes for assessing and quantifying the confidence of performed analysis, and constitute the basis of the BEPU approach.

The accomplishment of safety requirements in the NPP design is achievable by suitable safety analysis and assessment. 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 exist [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).

This work aims to show the first steps towards a BEPU-FSAR methodology, discussing the key disciplines presented in FSAR and the role of the nuclear thermal hydraulic principles in a system thermal-hydraulic (T-H) codes development, establishing a background for extension of their principles to non T-H codes.

2. SCALING

Scaling is the process of demonstrating the applicability of any parameter value to the reactor conditions. The complexity of a nuclear system brought to the need of current values for system parameters like pressure, geometry and power. Then, the problem was the large difficulty in characterizing the system performance at the conditions of the design: almost unavoidably, again to reduce the cost, experiments aimed at understanding the original system, here called prototype, were performed in small scale systems called models. Models were designed, constructed and operated under downscaled ranges of values for one or more of the listed parameters. This was at the origin of the scaling issue, i.e. the difficulty to demonstrate that a model behaves like the prototype [4].

Scaling is a key step for code development, V&V and application, including uncertainty evaluation. In safety analysis, scaling is an important source of uncertainty. The evaluation model contains numerous experimental correlations that the scaling distortion is embedded. The nodalization could also include scaling effects that affect the results in the reactor simulation. Therefore the applicability and the scalability are two main concerns in the model. Current available approaches to meet safety requirements are focused on these two areas [4].

The thermal-hydraulic code shall be considered as the most powerful tool to perform scaling analyses: this is true if rigorous and traceable procedures are adopted for any computational tool connected with the application of those codes. Scaling methods are essential to achieve independent information and to confirm the application results of thermal-hydraulic codes [4].

2.1 Scaling and Licensing

Within the licensing process of water cooled reactors where best estimate codes are used, a typical request from Regulatory Authority deals with the demonstration of the scaling at different levels. This implies the demonstration of the scaling capabilities of the adopted computational tools including the code, the nodalization and the analyst or code user [4].

3. COUPLING

With the advent of increased computing power has come the capability to couple large codes that have been developed to meet specific needs such as three-dimensional (3-D) neutronics calculations for partial Anticipated Transients Without Scram (ATWS), Computational Fluid Dynamics (CFD) codes to study mixing in 3-D, and others. The ranges of software packages that are desirable to couple with advanced thermal-hydraulics system analysis codes include [5]:

- Multidimensional neutronics;
- Multidimensional CFD;
- Containment;
- Structural mechanics;
- Fuel behavior;

- Radioactivity transport.

There are a number of ways in which two or more codes can be coupled. In essence, the coupling may be either loose or tight. Loose mean that the two or more codes only communicate after a number of time steps and tight is when the codes update one another time step to time step. Whether a loose coupling or a tight coupling is required depends on the phenomena that are being modeled and analyzed [6].

Traditionally the thermal-hydraulic codes and the nuclear kinetics codes were developed to pursue different objectives and with little or no common connections. However, with recent computer developments resulting in the availability of powerful computation capabilities at reasonable costs, the interconnection between the two disciplines has become feasible. It is now possible to perform detailed dynamic T-H reactor system analysis together with coupled detailed dynamic 3-D core kinetics simulation even on a readily available power computational system.

4. VERIFICATION AND VALIDATION

V&V constitutes a critical activity to confirm the quality for any process and any computational tool adopted in nuclear reactor safety. Any calculation method or tool, including computer codes, adopted within the nuclear reactor safety context shall undergo proper V&V.

Verification includes code verification and solution verification. Code verification establishes that the code is free of coding errors and accurately solves the mathematical model incorporated in the code. Solution verification evaluates the numerical accuracy of the algorithms used to solve equations of the physical model [7].

Validation is the process of determining the degree to which a physical model is an accurate representation of the real world from the perspective of intended use of the model. In summary, verification answers the question: “How good are the equations solved” and validation answers the questions: “How good are the equations?” [7].

A typical FSAR covering the accident analysis in Chapter 15, may require the use of different codes when the BE approach is selected. Conservative and simplified safety analysis codes should be changed to thermal-hydraulic codes. Besides these codes, CFD, 3-D neutron kinetics, containment, structural mechanics and sub-channel types of codes could be used for BE approaches.

However, these codes are not the only codes which are used in the licensing process of the plant. There are many other codes that are used in other areas of FSAR, and in order to perform a BEPU-FSAR, the BEPU concepts should be also applied to them.

4.1 V&V and Licensing

In the 1990s, uncertainty methods, i.e. the technological achievement needed to apply BE codes, were proposed [8], and were brought to the attention of the international community.

In the same decade, the United States Nuclear Regulatory Commission (USNRC) issued the Regulatory Guide (RG) 1.157 [9], which opened the way to the use of BE code, although with specific conservative constraints. In this decade, V&V was not yet a matter of licensing processes, although preparatory discussions were held between the Regulatory Authority and the industry and preliminary documents were issued.

In the 2000s, many BEPU methods were applied. The USNRC issued the RG 1.203 "Transient and Accident Analysis Methods" [10], which provide guidance for use in developing and accessing Evaluation Models (EM) for accident and transient analyses. Evaluation models, as describe in the document, provide a more reliable framework for risk-informed regulation and a basis for estimating the uncertainty in understanding transient and accident behavior.

The IAEA issued technical non-binding documents like SRS 23 [11] and 52 [6] and more recently the SSG-2 [2]. The BEMUSE Project was launched by OECD/CSNI, for the study of the LOFT L2-5 transient [12], for the BE calculation and for the uncertainty and sensitivity analyses [13].

At the industrial level, the Large Break Loss of Coolant Accident (LB-LOCA) in the Angra-2 KWU-Siemens NPP in Brazil was licensed based on the BEPU [14]. The V&V for codes came to the attention of regulators as well as the issues of nodalization and user qualification.

The FSAR Chapter 15 dealing with Accident Analysis, according to the USNRC Standard Review Plan [15], basically accepted by the entire nuclear industry worldwide, constitutes the main connection between licensing and V&V. The main purpose of the FSAR Chapter 15 is ensuring the safety level of a NPP. The main tool to ensure the safety is a computer code containing unavoidable errors. To ensure the safety is necessary to cover all possible uncertainties either by conservatism (EM approach) or quantify all possible uncertainties (BEPU). The current status in V&V and Licensing area can be characterized by the requirements for EM and Evaluation Model Development and Assessment Process (EMDAP), proposed in the RG 1.203 [10].

5. BEPU-FSAR

A 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 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 built to the commissioning and the training of the employees [15].

The application of BEPU methodology for licensing purposes is originated from the calculations of LB-LOCA scenario. Later, this methodology was adopted for analysis of Small Break Loss of Coolant Accident (SB-LOCA), as well as for operational transients. Some examples of industrial applications of the BEPU methodology are provided in [16].

Considering all the successful applications of the BEPU methodology for licensing purposes, it is therefore proposed to extend its range of use to each area of FSAR where analytical-computational activities are required, consequently going beyond of the current application which is limited to the accident analysis (Chapter 15).

The first step towards BEPU-FSAR is the identification and characterization of parts of the FSAR where numerical analyses are required. Starting from these parts, so-called BEPU topics, the next step is to create a list of key technological areas, so-called key disciplines and their related key topics. Table 2 shows the list of key disciplines and related key topics which was derived from FSAR content.

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 Format and Content Defense in Depth application Compliance with applicable code
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 Design of reactor Design transients Safety functions
Nuclear Fuel	Nuclear Fuel performance Fuel movement - Loading and unloading machines - Spent fuel cask
Materials	Corrosion Mechanical resistance Radiation damage Creep Analysis Fatigue Analysis Erosion
Neutron Physics	Cross Section Derivation Monte Carlo

Chemical Engineering	Chemistry of nuclear fluids Metal Steam production Zircaloy reactions Boron control Chemical Environmental
Electronic Engineering	Instrumentation and Control (I & C) Nuclear Instrumentation (in-core) Ex-core instrumentation Digital systems Analog systems Safety Systems
Electrical Engineering	Offsite Power System Onsite Power System Station Blackout
Civil Engineering	Containment Foundation
Deterministic Safety Analysis	Accident Analysis Computational tools Thermal-Hydraulic Computational Fluid Dynamics Uncertainty Analysis Severe Accident Consequences
Probabilistic Safety Analysis	Reliability Cost-Benefit Analysis Severe Accident Probability
Human Factors Engineering	Man-Machine interface Simulator Human failure
Occupational Health and Radioprotection	Radiological Protection Doses Impact of Doses 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 Maintenance Power production Financing outcome Administrative Procedures Inspections, Tests, Analyses and Acceptance Criteria
Quality Assurance	Management Procedures Standards
Computational Science	Information Technology Software

The second step toward BEPU-FSAR includes an overview of the current computational activities in each technological area. The term “computational activities” encompasses all the activities conducted within FSAR using different numerical techniques and related methodologies essentially required for the demonstration of the safety of NPP.

The third step to achieve the proposed BEPU-FSAR methodology is the transfer of the BEPU concepts into each technological area of FSAR (e.g. seismic analysis, radioprotection, etc.); this work is the first attempt to discuss the nuclear thermal hydraulic principles in different codes. The next steps include the demonstration of industrial worth and interest of the methodology by means of the comparative analyses with the current approaches (e.g. demonstration of the possibility to reduce abundant conservatism, better understanding of the physical phenomena and code models when performing uncertainty analyses, wider and/or longer operational ranges of NPP components, etc.).

6. CONCLUSIONS

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 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 the technology research, thus homogenizing what is in the concern to the safety of nuclear power plants: the analyses including calculation process, but 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.

In the list of key disciplines and related key topics which was derived from the FSAR content, one can recognize areas for which the specific expertise and knowledge are needed (e.g. Climatology, Instrumentation and Control, etc.), and where the availability of mature and qualified computational tools is required. In order to prepare a BEPU-FSAR, all the technical areas of the FSAR should be logically integrated and all the relations between them properly considered.

An important step to consolidate the BEPU-FSAR is the transfer of BEPU concepts into each technological area of FSAR. The BEPU concepts can be summarized by the pillars of thermal hydraulic field, wherein V&V and uncertainty quantification of computational simulations are the major processes for assessing and quantifying the confidence of performed analysis, constituting the basis of the BEPU approach.

Based on the finalized BEPU applications one can conclude that this methodology is feasible, which encourage to extended its range of use to the other technological areas of FSAR, and therefore to demonstrate the industrial worth and interest.

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.

ACKNOWLEDGMENTS

The authors gratefully acknowledge the contributions of the Gruppo di Ricerca Nucleare San Piero a Grado (GRNSPG) from the Università degli Studi di Pisa, Italy.

REFERENCES

1. F. D'Auria, N. Debrecin, "Perspectives in Licensing and Nuclear Reactor Safety Technology", *Third International Scientific and Technical Conference Innovative Designs and Technologies of Nuclear Power* (ISTC NIKIET-2014), Moscow, Russia, 7-10, October (2014).
2. IAEA, "Deterministic Safety Analysis for Nuclear Power Plants, SSG-2", Vienna (2010).
3. F. D'Auria, H. Glaeser, M. Kim, "A Vision for Nuclear Reactor Safety". Jahrestagung Kerntechnik - *Annual Meeting on Nuclear Technology*, Berlin, Germany, May 5-7 (2015).
4. F. D'Auria, G.M. Galassi, "Scaling in nuclear reactor system thermal-hydraulics", *Nuclear Engineering and Design*, **240**, pp. 3267-3293 (2010).
5. Neutronics/Thermal-hydraulics Coupling in LWR Technology, Vol. 1. CRISSUE-S – WP1: Data Requirements and Databases Needed for Transient Simulations and Qualification 5th EURATOM Framework Programme (1998-2002).
6. IAEA, "Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation. SRS 52", Vienna (2008).
7. J.Y., Sauvage, S. Laroche, "Validation of the Deterministic Realistic Method applied to Cathare on LB LOCA experiments", *ICONE-10*, Arlington, Virginia, ASME (2002).
8. W.L., Oberkampf, T.G. Trucano, "Verification and validation benchmarks", *Nuclear Engineering and Design*, **238**, pp. 716–743 (2008).
9. USNRC, "Best Estimate Calculations of Emergency Core Cooling System Performance, RG 1.157", Washington, DC (1989).
10. USNRC, "Transient and Accident Analysis Methods, RG 1.203", Washington (US) (2005).
11. IAEA, "Accident Analysis for Nuclear Power Plants, SRS 23", Vienna (2002).
12. A. Petruzzi, F. D'Auria (Editors), A. De Crecy, P. Bazin, S. Borisov, T. Skorek, H. Glaeser, J.P. Benoit, E. Chojnacki, K. Fujioka, S. Inoue, B.D. Chung, I. Trosztel, I. Toth, D.Y. Oh, R. Pernica, M. Kyncl, J. Macek, R. Macian, E. Tanker, A.E. Soyer, O. Ozdere, M. Perez, F. Reventos, "BEMUSE Programme. Phase 2 report: Re-Analysis of the ISP-13 Exercise, post test analysis of the LOFT L2-5 experiment". OECD/CSNI Report NEA/CSNI/R (2006).
13. A. de Crécy, P. Bazin (Editors), H. Glaeser, T. Skorek, J. Joucla, P. Probst, B. Chung, D.-Y. Oh, M. Kynck, R. Pernica, J. Macek, R. Meca, R. Macian, F. D'Auria, A. Petruzzi, M. Perez, F. Reventos, K. Fujioka, "BEMUSE Phase III Report, Uncertainty and Sensitivity Analysis of the LOFT L2-5 Test", OECD/CSNI Report NEA/CSNI/R(2007)4, October (2007).

14. M.R.S Galetti, “Regulatory Scenario for the acceptance of uncertainty analysis methodologies for the LB-LOCA and the Brazilian approach”, *Science and Technology of Nuclear Installations*, **2008**, (2008).
15. USNRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG 0800”, Washington, DC (2013).
16. A. Prosek, B. Mavko, “Review of Best Estimate Plus Uncertainty Methods of Thermal-Hydraulic Safety Analysis”. *Proceeding of International Conference Nuclear Energy in Central Europe*, Portorož, Slovenia, 8-11 September (2003).