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Safety Analysis Reports of First-Of-A-Kind Nuclear Reactors in Brazil: Proposal of Content for Deterministic Safety Analysis

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Abstract: The main documents of Licensing processes for nuclear reactors in Brazil are the Preliminary Safety Analysis Report (PSAR), in the construction phase, and the Final Safety Analysis Report (FSAR), in the operation phase (initial and permanent), as defined by the National Nuclear Energy Commission (CNEN). Thus, this paper presents a proposal for the format and content of the Deterministic Safety Analysis (DSA) chapter of Safety Analysis Reports (SAR) of First-Of-A-Kind (FOAK) nuclear reactors in Brazil, such as Small Modular Reactors (SMR), based on the combination of recommendations of United States Nuclear Regulatory Commission (USNRC) and International Atomic Energy Agency (IAEA) normative and documentary basis. Documentation from the USNRC applies to Nuclear Power Plants (NPPs), specifically to Light Water Reactors (LWR); complementary, IAEA documentation outlines a format and content of a SAR for NPPs but it may, in parts, have a wider applicability to other nuclear facilities. Compared to what is recommended by the USNRC, the IAEA allows for greater scope in terms of the types of events to be considered in a DSA, such as, for example, the Design Extension Conditions (DEC) and, in terms of methodology, it allows for a greater number of options for DSA approaches. The content proposal presented in this work recommends the consideration of DEC without significant fuel degradation (DEC-A), the adoption of the combined approach, analysis of hazards and Pressurized Thermal Shocks (PTSs) analysis, among others.

Keywords: licensing, safety analysis report, deterministic safety analysis.

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Resumo: Os principais documentos do processo de Licenciamento de reatores nucleares no Brasil são o Relatório Preliminar de Análise de Segurança (RPAS), na fase de construção, e o Relatório Final de Análise de Segurança (RFAS), na fase de operação (inicial e permanente), conforme definido pela Comissão Nacional de Energia Nuclear (CNEN). Assim, este artigo apresenta uma proposta para o formato e o conteúdo do capítulo de Análise Determinística de Segurança (ADS) dos Relatórios de Análise de Segurança (RAS) de reatores nucleares "primeiros do tipo" no Brasil, como Pequenos Reatores Modulares (SMR), com base na combinação de recomendações da base normativa e documental da *United States Nuclear Regulatory Commission* (USNRC) e da *International Atomic Energy Agency* (IAEA). A documentação da USNRC aplica-se às usinas nucleares (NPPs), especificamente aos reatores de água leve (LWRs); já a documentação da IAEA especifica o formato e o conteúdo de um RAS para NPPs, mas também pode aplicar-se a outras instalações nucleares. Comparado ao que é recomendado pela USNRC, a IAEA permite um escopo maior em termos dos tipos de eventos a serem considerados em uma ADS, como, por exemplo, as Condições de Extensão de Projeto (DEC) e, em termos de metodologia, permite um maior número de opções para abordagens de ADS. A proposta de conteúdo apresentada neste trabalho recomenda a consideração de DEC sem degradação significativa de combustível (DEC-A), a adoção da abordagem combinada, a análise de perigos e a análise de Choques Térmicos Pressurizados (PTSs), dentre outras recomendações.

Palavras-chave: licenciamento, relatório de análise de segurança, análise determinística.

1. INTRODUCTION

In Brazil, the licensing processes for nuclear reactors involve the preparation of the Preliminary Safety Analysis Report (PSAR) during the construction phase and the Final Safety Analysis Report (FSAR) during the operational phases (initial and permanent), as outlined in the CNEN NE 1.04 [1] standard. These documents are meticulously formulated and prepared following specific formats and content requirements using normative and documentary basis, such as the Standard Formats and Review Plans/Guides recommended by the regulatory authority.

Specifically, for the chapter that deals with Deterministic Safety Analysis (DSA), it is common in Brazil to use as reference the documentation from the United States Nuclear Regulatory Commission (USNRC), such as Regulatory Guide 1.70 (PSAR) [2] and 1.206 (FSAR) [3], and the Standard Review Plan (NUREG-0800) [4], which are applied to the licensing process of Nuclear Power Plants (NPPs), specifically to Light Water Reactors (LWRs).

In 2021, the International Atomic Energy Agency (IAEA) published Specific Safety Guide 61 (SSG 61) [5], which outlines a format and the content of a Safety Analysis Report (SAR) for NPPs; however, it may, in parts, have a wider applicability to other nuclear facilities. Furthermore, compared to what is recommended by the USNRC, SSG-61 [5] allows for greater scope in terms of the types of events to be considered in a DSA, such as, for example, the Design Extension Conditions (DEC) and, in terms of methodology, it allows for a greater number of options for DSA approaches.

In this context, this paper presents a proposal of content for the DSA chapter on safety analysis reports (SAR) of First-Of-A-Kind (FOAK) nuclear reactors in Brazil, such as Small Modular Reactors (SMR), based on the combination of recommendations of IAEA and USNRC.

2. MATERIALS AND METHODS

This proposal of content for the DSA chapter of SAR of FOAK nuclear reactor is based on the guidelines proposed by IAEA [5] and USNRC [2,3,4], maintaining the format/organization/structure recommended by USNRC. Both IAEA and USNRC recommendations consider that DSA is presented in Chapter 15 of the SAR.

This paper outlines the main differences between the IAEA and USNRC recommendations/requirements about content and approaches. Looking deeper into the safety requirements and USNRC practices in safety evaluation, it is clear that the safety issues addressed by the USNRC align with those of the IAEA.

Thus, a comparison is made between the contents of the DSA chapter proposed by the USNRC and the IAEA, discussing the main differences identified and defining the content proposed for Chapter 15 of SAR of a FOAK reactor in Brazil.

3. RESULTS AND DISCUSSIONS

To formulate a proposal of content for the DSA chapter on Safety Analysis Reports (SAR) of FOAK nuclear reactors, it is important to evaluate the main differences identified between IAEA and USNRC recommendations/requirements regarding content and types of approach.

3.1 Identification and evaluation of the main differences between IAEA and USRNC for proposing content for the DSA

The main differences identified between IAEA and USNRC recommendations/requirements regarding content and types of approach for the DSA are described and discussed below:

- IAEA [5] specifies that all safety analyses, that is, DSA and Probabilistic Safety

Analysis (PSA), are considered in one chapter - Chapter 15, while the USNRC [2,3,4] specifies that in Chapter 15 ("Transient and Accident Analysis") only the DSA shall be presented, with PSA being considered in Chapter 19;

- IAEA [5] suggests also to treat the severe accidents independently of PSA; however, PSA shall be considered using deterministic approach and postulating sequences which might result in severe core damage and potential breach of the primary system and challenge to the containment integrity;
- IAEA [5] suggests also to address deterministically for new reactors the problem of potential early and large radioactive releases by analysis of processes which may lead to damage of the containment, i.e. evaluating deterministically the Design Extension Conditions with core melting (DEC-B). The goal is to show that the containment and safety features in the containment are such that off-site measures are limited. The problem, called Practical Elimination, is not considered in USNRC guides;
- USNRC approach does not consider the concept of DEC and Practical Elimination suggested by IAEA. However, some DEC without significant fuel degradation (DEC-A) equivalent conditions are considered, such as Anticipated Transient Without Scram (ATWS). In turn, the IAEA safety requirements specify that all plant states need to be considered in the safety analyses: Normal Operation (NO), Anticipated Operational Occurrences (AOO), Design Basis Accidents (DBA), and DEC-A and additionally a practical elimination of large and early radioactive releases needs to be addressed;
- IAEA also recommends that Pressurized Thermal Shocks (PTS) analysis shall be considered, if applicable, as well as internal and external hazards relevant to the determination of Postulated Initiating Events (PIE); and
- USNRC approach considers using either a fully conservative method (i.e. conservative codes and conservative initial and boundary conditions) or the best

estimate with estimation of uncertainties (best-estimate codes and best estimate initial and boundary conditions). The IAEA presents different options currently available for performing DSA (see Table 1), such as the adoption of combined approach using best estimate code and conservative assumptions for initial and boundary conditions.

Table 1: Options for performing Deterministic Safety Analysis [6].

OPTION	COMPUTER CODE TYPE	ASSUMPTIONS ABOUT SYSTEMS AVAILABILITY	TYPE OF INITIAL AND BOUNDARY CONDITIONS
1. Conservative	Conservative	Conservative	Conservative
2. Combined	Best estimate	Conservative	Conservative
3. Best estimate plus uncertainty	Best estimate	Conservative	Best estimate Partly most unfavorable conditions
4. Realistic*	Best estimate	Best estimate	Best estimate

* For simplicity, the term 'realistic' is used to mean best estimate analysis without quantification of uncertainties.

3.2. Proposed content

Based on the analysis of the main aspects described above, the proposal of content to elaborate the Chapter 15 of a FOAK nuclear reactor in Brazil is presented below:

- Chapter 15 shall encompass only DSA;
- Format/organization/structure shall be as recommended by USNRC [2,3] (the IAEA format is similar to the SAR format required by the USNRC);
- Content shall comply with both USNRC [2,3,4] and IAEA [5,6,7,8] recommendations and requirements for DSA. It is worth noting that IAEA safety standards represent the current understanding and worldwide experience related to safety issues of NPPs and safety demonstration;
- Section 15.0 ("Introduction") shall address:
  - o General considerations: scope of Chapter 15, description and justification of the approach adopted, reference documents related to the methodology used, and

description of the structure of Chapter 15;

- o Identification and Categorization of PIE and accident scenarios. Events shall be divided into categories, according to their frequency of occurrence, and grouped according to type; PTS analysis; analysis of hazards; plant states: NO, AOO, DBA, and DEC-A (DEC-B shall not be considered in Chapter 15);
- o Safety objectives and acceptance criteria. USNRC [5] acceptance criteria are more "prescriptive" than those of the IAEA [6,7,8], apparently more "generic"; it is proposed to use both, if applicable;
- o Human actions shall be considered;
- o Facility characteristics considered in the DSA shall be described, such as: design parameters, power distribution, neutronic data and geometric models of the reactor core, analytical limits and delay times, loss of off-site power (*LOOP*), single failures, and initial conditions;
- o General description of the analysis approach shall be presented, including: a brief description/characterization of the codes; a summary of the verification and validation (V&V) scope of codes; and a description of the models used in the analysis (nodalization schemes/diagrams and their qualification); and
- o Assessment of radiological consequences shall be presented;

- NO analysis shall be presented;
- AOO analysis shall be presented;
- DBA analysis shall be presented;
- DEC-A analysis shall be presented;
- Analysis of PIE and accident scenarios associated with the spent fuel pool and analysis of fuel handling events shall be presented; and
- Interfaces with other SAR chapters shall be considered.

Specifically related to the analysis approach, the proposal presented here suggests to adopt the combined approach using best estimate code and conservative assumptions for systems availability and initial and boundary conditions. This is based on the fact that this approach is conceptually simpler, well understood, and comply with all requirements of the safety analysis.

4. CONCLUSIONS

This paper presents a proposal of content for DSA chapter (Chapter 15) of the SAR aiming to combine USNRC and IAEA recommendations in order to comply with the requirements of the safety analysis for FOAK nuclear reactors in Brazil such as SMR, considering that the licensing process is one of the most important challenges to implement new design of nuclear reactors to face the energetic transition all around the world.

The content proposal presented here is the result of the comparison and evaluation of the main differences identified between the recommendations of USNRC and IAEA, consisting of a combination of both of them, highlighting the consideration of DEC-A, the adoption of the combined approach, analysis of hazards and PTS analysis, among others.

The consideration of IAEA recommendations enables to consider some characteristics of nuclear reactors which are not those of conventional NPPs. It is important to emphasize that the proposal of content must be discussed and agreed with the regulatory authority previously to be applied, and during this process some points can be deeply discussed, taking into account that particular contents of the DSA for FOAK nuclear reactors depend on the specific design of the nuclear facility.

The proposal presented here can help both, applicant and regulatory authority, to drive the DSA for SAR purpose; moreover, this can contribute to a better quality of the safety analysis described in the SAR, and consequently, improve the overall licensing process.

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CONFLICT OF INTEREST

All authors declare that they have no conflicts of interest.

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