# Safety Analysis of Small Modular Reactors in the context of the Polish regulatory framework

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**Abstract**

Small Modular Reactors (SMR) gained a lot of interest in the Polish industry due to their potential to provide a baseload, carbon-free source of electricity and other commodities like process heat. Poland has a licensing framework and regulatory body, the National Atomic Energy Agency (Państwowa Agencja Atomistyki - PAA), as well as a few nuclear facilities, including a research reactor (MARIA), the decommissioned research reactor Ewa and radioactive waste facilities.

However, there are no commercial nuclear power plants, which creates specific challenges for both the regulatory body and the nuclear industry. One of the main challenges is applying the current licensing framework to advanced reactor technologies, including SMRs. Additionally, there is a need to improve the current framework to make it suitable for new technologies.

Safety analysis of innovative reactors, such as SMRs, is a crucial and emerging topic within the regulatory process. This work focuses on the current Polish regulations related to safety analysis and attempts to place them in the context of advanced reactors. In this paper, selected topics related to safety analysis are discussed, potential obstacles are identified, and conclusions that can be useful for PAA or other involved organizations are drawn.

## INTRODUCTION

The situation in Poland is somewhat unique. The regulatory body was established in 1982. During the 1980s, the construction of the first Nuclear Power Plant in Żarnowiec was underway, and in 1985, PAA issued the first construction permit. Unfortunately, the construction was cancelled in 1990, and a significant amount of human resources and expertise, including in safety analysis, were lost. Despite these setbacks, PAA continued its mission to regulate existing nuclear facilities in Poland, retaining several competencies. The regulatory process was maintained to license the research reactors Maria (since 1974) and Ewa (1959-1995), as well as other nuclear facilities. When plans for new NPPs emerged as early as 2006, it became necessary to renew and rebuild potential. Since then, PAA has been preparing itself for the second attempt to introduce nuclear power in Poland.

Safety analysis is a fundamental tool for ensuring the safety of nuclear power plants (NPP). For the PAA, competence and a thorough approach to safety analysis are crucial. PAA staff will be responsible for reviewing the Safety Analysis Reports (SAR) and related documents. An additional challenge is the introduction of SMRs, which have recently gained substantial interest. This new technology demands a proper approach to safety analysis and necessitates the revision of existing regulations. Considering the dynamic situation in international practices, state-of-the-art regulation, and growing interest in nuclear power, these updates are essential.

## THE Polish regulatory Framework

The National Legal Framework for Nuclear Power in Poland, including all relevant documents, is available on the PAA website [1]. The main Act is the Atomic Law [2]. The hierarchy of Polish atomic-related legal documents is presented in Fig. 1.

The Atomic Law defines principles and basic requirements for activities with radiation sources, including nuclear reactors. The next level in the hierarchy consists of governmental regulations, which provide more detailed guidance on specific topics. Essentially, these documents explain how to apply the principles outlined in the Atomic Law.

Furthermore, lower in the hierarchy are the PAA President Guidelines, which contain detailed instructions or guidelines on how to comply with the regulations.

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*FIG. 1. Hierarchy of the licensing related documents.*

Safety analysis is required by Polish Law (Art. 36d of the Atomic Law [2]). Before applying for a construction license, the investor must prepare safety analyses covering nuclear safety, considering both technology and the environment. These analyses must be verified by organizations that were not involved in the plant design process.

The investor must prepare a Preliminary Safety Analysis Report (PSAR) based on the prepared safety analyses. This report must be submitted to the President of the PAA along with the application for a construction license. It is important to highlight that the Atomic Law [2] explicitly states that the WENRA and IAEA recommendations should be considered when developing national regulations on safety analysis (Art. 36d. 3. [2]).

For the PAA, it is critical to revise the PSAR report, potentially with the support of independent technical organizations. Currently, there is no single Technical Support Organization (TSO). Instead, various external organizations, both national and foreign, can apply for PAA accreditation, allowing them to cooperate with the PAA in the assessment process.

Regulations and requirements for safety analysis, as well as the content of the PSAR, are published in the Regulations of the Council of Ministers related to safety analysis [3] and nuclear facility design [4]. The second document specifies that safety analysis is an inherent part of the design process. Consequently, these two documents form the basis for performing safety analysis for plants to be constructed in Poland. Additionally, there is a regulation related to intervention levels [5], which impacts safety analysis concerning the consequences of accidents.

Regarding PAA guidelines, a single document related to safety analysis was published. This document describes an approach to determine an Emergency Planning Zone, which impacts both the design process and safety analysis (see Reference [6]).

Polish Law does not currently recognize SMRs, with the exception of considerations related to reactor power and the determination of the EPZ. The crucial regulations [3] and [4] were published in 2012, a time when large-scale plants were primarily considered. However, regulations [3] and [4] are currently under review. Consequently, from the perspective of current regulations, there is no substantial difference between SMRs and large-scale NPPs.

## SAFETY ANALYSIS IN THE POLISH FRAMEWORK

### Introduction

Because this paper is limited to current Polish regulations, it must be clear that the same approach applies to both large-scale NPPs and SMRs. This creates a specific challenge in applying the current regulations. Another issue is that contemporary Polish regulations have not previously been applied to license NPPs or issue construction permits. Hence, the PAA, as the regulatory body, and other involved parties face challenges in applying the existing regulations in practice.

Polish regulations related to safety analysis and nuclear facility design were developed over a decade ago and were largely consistent with contemporary international documents, especially those from the IAEA (e.g. SSR-2/1 [9]). However, some practices have changed over the years, and as a result, regulations [3] and [4] are under revision and will likely be updated in the future. Until then, the current regulations remain in force, and the regulatory body must follow them in attempts to license NPPs.

In the next part of this chapter, important contemporary regulations and requirements related to safety analysis will be discussed. Some of these may present substantial challenges in the deployment of SMRs.

In the Polish framework, safety analysis is divided into deterministic and probabilistic parts: Deterministic Safety Analysis (DSA) and Probabilistic Safety Analysis (PSA). Both approaches are required by Polish law. The PSA is limited to Level 1 and Level 2, while Level 3 is not required [3].

### Plant States and Safety Analysis

In Polish regulations, plant states considered within safety analysis are grouped into categories: Normal Operation (NO), Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs), and Design Extension Conditions (DECs). There is also a category for hypothetical severe accidents beyond the design (Ref. [3], App. I).

The Atomic Law [2] considers **accident conditions** as plant states beyond AOOs. **Design (basis) conditions** include NO, AOOs, and DBAs. DBAs in Ref. [2] are defined as accident conditions considered in the design with proper requirements, where fuel damage and radioactive releases are bounded by predefined limits. **Severe accidents** are defined as accidents more serious than design-basis accidents, involving serious core degradation and potential for significant releases. DECs are not explicitly considered in [2]; they are defined in regulations related to safety analysis [3] and design [4]. The term “**considered accidents**” includes DBAs and DECs, while **operational states** are NO and AOOs.

DECs are defined as sequences more serious than DBAs but with acceptable radionuclide releases (Def. 22, in Ref. [4]). DECs are divided into two categories: complex sequences that do not lead to core melt (DEC-A) and core melt sequences without containment failure (DEC-B).

Safety analysis covers both NPP operational states (NO, AOOs) and accident conditions (DBAs, DECs). The main goal and scope of safety analysis is to prove that safety requirements related to radiological doses, consequences, intervention levels, and probabilistic criteria given in the Atomic Law (Art. 36f 2. [2]) and Design Regulation (§9 and §10 in [4]) are fulfilled.

Among other things, safety analysis must assess design solutions, the proper sequence of safety barriers, and internal and external hazards, including extreme environmental hazards, both natural and anthropogenic. Furthermore, it must prove that human factors, aging phenomena, procedures, emergency actions, and the role of SSCs were properly addressed in the design.

Conditions with consequences above DEC-B fall within the hypothetical severe accidents category and must be shown to be improbable. In Polish regulations, the concept of practical elimination is not explicitly present; however, regulations may be interpreted as practical elimination. Essentially, Atomic Law Art. 35. 4. 2 ([2]) demands technical and organizational solutions aimed at avoiding: a) “early releases of radioactive substances requiring intervention measures outside the nuclear facility that could not be performed due to insufficient time” and b) “large releases of radioactive substances requiring intervention measures outside the nuclear facility that could not be contained in time or space”. This largely aligns with the current interpretation of practical elimination (see, e.g., Ref. [8]). These solutions must be considered during the design phase, as well as during the lifetime and decommissioning phases. It can also be interpreted that practical elimination, at least to some extent, is also given in Art. 36c. 2) in Ref. [2] and Ref. [4] §15 and §32.2, and also considered in Appendix 1, 7.4.7.4 of Ref. [3].

### Emergency Planning Zone

One of the tasks to be covered by safety analysis is determining the Emergency Planning Zone (EPZ), which must be identified for a nuclear power plant (NPP). According to Polish regulations, nuclear reactors with more than 100 MWth must have two zones: the precautionary action planning zone (inner zone) and the urgent protective action planning zone (external zone). Reactors with power between 2-100 MWth are required to have only an external zone, and Act [2] does not consider zones for reactors with power lower than 2 MWth ([2]).

Zones must be identified through safety analysis, considering potential consequences of releases characterized by a frequency higher than 1 per 107 years. The inner zone focuses on the prevention of deterministic dose effects, while the external zone is dedicated to reducing the risk of stochastic effects. More details on the EPZ can be found in Art. 86l and Art. 86m [2], and guidance is provided in [6].

### Safety Analysis in the Design Process

Contemporary safety analysis is an integral part of the plant design process, as outlined in Polish regulations related to the design of nuclear facilities [4]. Firstly, the classification of Systems, Structures, and Components (SSC) must be based on deterministic safety analysis, supported by probabilistic analysis if needed (§11.3 in Ref. [4]). Furthermore, the Postulated Initiating Events (PIE) applied to safety analysis are considered in the design process (§16 in Ref. [4]), with PIEs defining a set of design basis accidents (§26, Ref. [4]). The list of PIEs, which includes both internal and external events, serves as an input for safety analysis (§6 and §7, Ref. [3]). Additionally, design limits for the plant must be confirmed by safety analysis ([4], Def. 8). The Defence-in-Depth concept (Ref. [4] §3) and related barrier sequences should incorporate safety systems identified through safety analysis.

For determining design basis conditions, a conservative approach should be applied, using initial and boundary conditions with appropriate safety margins and considering only safety-classified systems. Secondary failures or consequential failures from PIEs are also included. Proven methods are applied with high confidence to exclude serious consequences, core damage, and large doses (§12 [4] and for DSA in §5 [3] and Chapter 3 [3]).

DECs are considered part of the plant design and are included in safety analysis. The design of nuclear facilities should account for DECs, using best estimate methodologies (§13 and §28.2 in Ref. [4], and §5 in Ref. [3]).

The design regulation [4] requires that containment and containment systems withstand severe accident scenarios selected based on engineering judgment and probabilistic safety analysis (§29.2. Ref. [4]). Moreover, the design of an NPP should consider accident sequences with containment bypass and potential releases, even without fuel melt (§32.1 Ref. [4]). According to §32.2 and [4], an NPP must be designed to prevent severe accidents with early primary containment failure. Alternatively, the designer must demonstrate that such accidents are improbable, particularly considering hydrogen explosions, high-pressure melt ejection (HPME), high-energy missile generation, and direct containment heating (DCH) (potentially related to practical elimination – see previous sub-chapter).

The design regulation explicitly mentions DEC complex sequences to be considered during the design process, including ATWS with releases beyond containment, Total Loss of AC Power, Containment Bypass, Total Loss of Feedwater, LB-LOCA with loss of safety injection, Total Loss of Ultimate Heat Sink, Uncontrolled Boric Acid Dilution, SGTR, and others (§30. Ref. [4]).

Regulation §32.4 [4] states that the design must include measures to mitigate severe accident consequences, such as molten core retention and cooling, limiting molten core concrete interactions (MCCI), limiting containment leaks, and allowing extended time for operator actions or actions to control the accident.

Additionally, §33 in Ref. [4] requires that the NPP design considers the impact of a large civilian aircraft. Even with limited operator actions, the reactor core must be cooled, primary containment must remain intact, and cooling of the spent fuel must be maintained or the spent fuel pool must remain intact.

Assessing all the phenomena and design requirements mentioned above demands sophisticated safety analysis using computational tools and possibly experimental work or databases during the design phase.

### Deterministic Safety Analysis

Deterministic safety analysis (DSA) for the design basis conditions (AOOs, DBAs) requires a conservative approach (§5 in [3]), similarly to the determination of design basis conditions discussed in the previous chapter. This aligns with SSR-2/1 (Para. 5.26) [9]. Consequently, BEPU-type methodology is not currently considered for DBCs. For DSA of DECs, best estimate analysis can be applied (§5 in [3]) with proper consideration of uncertainties.

The specific requirements for DSA are detailed in Chapter 3 of Ref. [3]. Notably, §14.1 and 2 in Ref. [3] mandate the consideration of the single failure criterion (SFC) and Loss of Offsite Power (LOOP) after PIE in safety analysis for AOOs, DBAs, and DEC states. This approach is quite conservative and may be reviewed, as current practice tends to avoid LOOP and SFC in DEC states (see Ref. [10]).

Additionally, §20 in [3] requires that safety analysis considers all locations or sources of radioactive materials in a nuclear facility, including the reactor core, reactor coolant system, fuel during handling and transfer, spent fuel storage, and radioactive waste-related facilities.

DSA for DBAs applies two levels of acceptance criteria (§25 in [3]). The first level relates to radiation doses to the public and the absence of intervention actions beyond the limited use area (Ref. [4] §9.1). The second level includes detailed criteria such as ensuring that the PIE does not lead to a more serious state without additional failure, preventing secondary failures due to PIE, maintaining design limits, preserving core geometry, and ensuring accident-rated equipment can withstand existing conditions. Significant acceptance criteria for all states are compared in Appendix 1 of Ref. [3].

For DSA of DECs (§32 in [3]), there are deterministic criteria for a limited radiological impact (Ref. [4] §9.2), expressed through intervention action demands for the EPZ and limited use area. Probabilistic criteria are also provided in Ref. [4] §10, which will be discussed in the next sub-chapter as they relate to PSA.

### Probabilistic Safety Analysis

Probabilistic Safety Analysis specific regulations are detailed in Ref. [3], Chapter 4. There are three probabilistic safety objectives, expressed as frequency limits, as follows (Ref. [4] §10):

1. Core damage frequency must be lower than 1 per 100,000 reactor years.
2. The frequency of releases to the environment that violate any intervention level requiring early or long-term actions, and beyond the EPZ violate intervention levels requiring intermediate-term actions, must be lower than 1 per 1,000,000 reactor years. Intervention levels are defined in Ref. [5].
3. The frequency of accident sequences potentially leading to early containment failure or very large releases to the environment must be significantly lower than 1 per 1,000,000 reactor years.

The last limit pertains to hypothetical severe accidents with containment failure, which are beyond DEC states.

## Conclusions AND DISCUSSION

As mentioned earlier, Polish regulations are currently undergoing substantial revision, particularly in the areas of safety analysis (both DSA and PSA) and design regulations. Consequently, several issues discussed in this paper will be modified and updated. The reader should note that this paper only addresses the current regulations, with a focus on safety analysis rather than design-related topics.

In the current Polish regulations, there is no significant difference between safety analysis for large power reactors and SMRs. However, there is a distinction for reactors with lower thermal power (<100 MWth, potentially ~25 MWe), which may impact the licensing of some SMRs to some extent. The determination of the Emergency Planning Zone (EPZ) is significantly different and may affect safety analysis. However, as of today, this remains an open topic requiring further investigation. As mentioned in the introduction, the current Atomic Law has not yet been applied to new reactors, making potential obstacles difficult to identify.

For water-cooled technologies, SMRs characterized by lower power will have a lower core radionuclide inventory and, consequently, lower source terms, resulting in expected lower consequences for the public and the environment. In this context, limits such as the EPZ or other action areas are reduced. Additionally, for advanced SMRs with passive systems, the scope of safety analysis can differ due to the limited frequency of more serious accidents. However, according to Polish regulations, passive systems are not treated differently from typical active systems. Therefore, their safety analysis was not discussed, as the paper focused on the current regulations.

The situation is different for non-LWR technologies, as the current regulations are primarily oriented towards LWR technology. In this case, each situation is more complex and will likely require updates and changes to the regulations.

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