



Original Article

An integrated risk-informed safety classification for unique research reactors



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ARTICLE INFO

Article history:

Received 24 August 2022

Received in revised form

14 January 2023

Accepted 15 January 2023

Available online 23 January 2023

Keywords:

IRISC

Integrated Risk-Informed Safety Classification

SSC classification

MARIA Research Reactor

ABSTRACT

Safety classification of systems, structures, and components (SSC) is an essential activity for nuclear reactor design and operation. The current regulatory trend is to require risk-informed safety classification that considers first, the severity, but also the frequency of SSC failures. While safety classification for nuclear power plants is covered in many regulatory and scientific publications, research reactors received less attention. Research reactors are typically of lower power but, at the same time, are less standardized i.e., have more variability in the design, operational modes, and operating conditions. This makes them more challenging when considering safety classification. This work presents the Integrated Risk-Informed Safety Classification (IRISC) procedure which is a novel extension of the IAEA recommended process with dedicated probabilistic treatment of research reactor designs. The article provides the details of probabilistic analysis performed within safety classification process to a degree that is often missing in most literature on the topic. The article presents insight from the implementation of the procedure in the safety classification for the MARIA Research Reactor operated by the National Center for Nuclear Research in Poland.

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1. Introduction

Safety classification in the design of a nuclear reactor is used to create the appropriate technical requirements for its systems, structures and components (SSC). Based on the assigned safety class, in order to reach the expected level of safety performance, items important to safety have more strict requirements for their technical characteristics, design, testing and manufacturing process, and maintenance procedures.

Nuclear power plant (NPP) reactors are well understood in terms of design safety and operational risks. The safety classification for NPPs, therefore, is a straightforward process. Research reactors typically reach much lower operational powers with smaller radioactive inventory than NPPs. However, due to unique design they are considered more challenging in terms of analytical effort in the design stage. In addition, research reactors may have one of a kind operational modes and operating conditions. Their design may change over time to satisfy research goals. Also, the operational culture is different (R&D environment, academic culture, research

oriented).

Those factors make the safety classification process for a research reactor a challenging task, especially the part of it that involves probabilistic aspects. The topic that is missing detailed coverage in regulatory documents and scientific publications. While general direction can be recognized in available documentation, many questions arise when dealing with actual implementation of the process.

The article is based on the research material gathered in safety classification project performed for MARIA Research Reactor operated by National Centre for Nuclear Research in Poland. The authors present in depth description of probabilistic methods used for SSC classification, together with a model of analytical process that can be utilized in similar projects and in research reactors of various type.

1.1. Polish conditions

Safety classification requirement for new nuclear facilities was introduced to Polish Atomic Law in the year 2000. It describes factors to be considered in safety classification based on the IAEA safety guide [1] for nuclear power plants.

Currently, in Poland there is only one nuclear reactor in

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operation. It is the 30 MW(th) MARIA Research Reactor located in National Center for Nuclear Research (NCBJ). MARIA is in operation since 1974, with major upgrades done in 1985, 2002 and 2014. There is no relevant experience from NPP operated in Polish regulatory conditions.

Polish regulatory body, which is the National Atomic Energy Agency (PAA) agreed with NCBJ to run the classification process for MARIA despite the fact it was not a new facility. One of the major reasons was to gather experience in the methodology, which was important for both organizations.

The safety classification team was formed from selected members of the reactor personnel and analysts experienced in environmental and probabilistic safety studies. During the initial communication, the representatives of Polish regulatory body asked for precise description of a procedure to implement SSC demand probability (the probability that SSC is required to operate) into classification process. That requirement was critical – MARIA researchers at that time have done multiple safety analyses of deterministic type, but only few of probabilistic nature [2]. As a result, the work on probabilistic models was initiated with a purpose to calculate the SSC demand probability for a set of Initiating Events (IE).

From the beginning it was clear that the SSC safety class should be based on its failure consequences (according to IAEA standards). But it was not clear how SSC demand probability can be used to amend its safety class. The task of the authors was to develop the appropriate procedure that would be accepted by the regulatory body. The additional difficulty was that SSC classification process for research reactors is not as well described as for nuclear power plants. In fact, research reactors, due to their unique design and functions are more challenging in terms of safety analysis.

1.2. Aims and objectives

The objective of MARIA safety classification was to:

1. Establish the classification procedure that is accepted by the regulatory body.
2. Build probabilistic models required to calculate SSC demand probability.
3. Follow the procedure to obtain SSC classes.

The aim of this article is to:

1. Review existing publications on SSC classification process for research reactors and NPPs.
2. Identify the missing know-how of SSC classification process.
3. Explain the process of including demand frequency into SSC classification:
 - a. Explain the frequency thresholds used in the risk matrix,
 - b. Explain the various cases of demand frequency calculation for research reactors,
 - c. Explain how SSC demand frequency may influence the SSC safety class.
4. Present the IRISC procedure and its application.

1.3. Frequency and probability

Frequency and probability are related terms. They can be used interchangeably in certain cases. If a given event has a constant, known frequency, one can use exponential distribution to calculate its probability. And in opposite case, knowing probability of an event, one can obtain its frequency. In the context of safety classification process, the frequency is preferred because it can

differentiate between events that can happen very often e.g., their annual frequency is higher than one. In this article the term “probability” or “probabilistic” is used occasionally when more appropriate but should not be understood as a different approach for the described topics.

1.4. State of the art

There are two main sources of information on the SSC classification process:

- national and international nuclear regulations,
- research papers and reports covering part or the whole process of SSC classification in existing reactors.

Those sources, for the purpose of this work, are grouped by the approach to probabilistic safety analysis:

- a. Sources where probability is utilized in classification,
- b. Sources with probability omitted from classification process (based solely on deterministic safety analysis i.e., on quantitative analysis of failure consequences).

Main international regulatory document describing the SSC classification in research reactors is the IAEA SSR-3 [3]. However, it contains only pure requirements for the process, without a deeper explanation. In much greater detail it is described in the IAEA SSG-30 [1] dedicated to NPPs. Final document from IAEA on classification is the IAEA TECDOC No. 1787 [4], which is a comprehensive guide to classification process (but lacking in terms of probabilistic methods).

US NRC General Design Criteria for NPPs in Criterion 1 [5] is explaining that the whole lifecycle of the safety related SSCs should be covered and tracked by the Quality Assurance Program (QAP) of the newly designed NPP. SSCs are then classified according to the QAP criteria. The same approach is maintained for research reactors according to the Regulatory Guide 2.5 [6]. This guide, however, is pointing to ANSI/ANS 15.8–1995 (R2013) [7] as a source of requirements.

The Quality Assurance Program (QAP) defines different requirements for safety related and non-safety related SSCs. For safety related SSCs, another requirement described by US NRC [8] formulates Risk Informed Categorization (RISC) of SSCs for Nuclear Power Reactors. The requirements given there, state that Probabilistic Risk Assessment (PRA) results should be considered in the process. The RISC categories, from I to IV are defined around severity of failures, described as safety significant (RISC I and II for safety or non-safety related SSCs) or low safety significant (RISC III and IV). The provided explanation of “safety significant” includes “risk” which hints that probabilistic methods are required.

Risk-informed methods for safety assessment (including risk-informed safety classification of SSCs) is a well-established research topic [9–13], that relies on importance measures for basic events (SSC failures), developed from PRA models. Those methods in general require mature PRA models and dedicated software to calculate importance measures e.g., RAW (Risk Achievement Worth), Birnbaum, Fussler-Vesely, that are used next to arrive at safety categorization for SSCs.

One has to mention a methodology that assigns the safety class to equipment based on the function it performs. Ref. [14] describes safety class 1 for equipment used in primary core cooling system or working at elevated temperature, class 2 for equipment in emergency core cooling systems and class 3 for systems needed for plant operation. In a similar way, the SSC classification is defined in Finnish regulations for NPPs [15].

Research papers describing the details of the SSC classification process that does not consider demand frequency include, for example, Japanese HTGR [16]. Its SSCs are classified according to failure consequence assessment i.e. purely deterministic approach. What is unique to this method is that it has three classes for prevention systems and three classes for mitigation systems (similar to IAEA's safety functions and design provisions). Ref. [17] is a comprehensive review of existing SSC classification systems applicable to pool-type research reactors. This work provides the summary of SSC classification performed for the Jordan Research and Training Reactor, which is based on functions provided by SSCs. Another paper on research reactors is about Chinese HTR-10 [18]. Its classification is binary, which means that an SSC is classified as important to safety or not important to safety. The criteria for classification are based on safety functions provided by SSC (or lack of those functions).

In general, two methods of safety classification may be distinguished, that do not consider demand frequency:

- a. Functional approach, where safety classes are assigned considering the safety functions delivered by SSCs,
- b. Deterministic approach, where safety classes are assigned based on SSC failure consequences.

In recent years, several studies were published about SSC classification that is risk-based, or in other way considers SSC demand or failure probability. For new nuclear designs, Ref. [19] proposes a risk-informed approach that is based on the US NRC Top Level Regulatory Criteria (TLRC), which provide a risk threshold for accident radioactive releases. It describes the allowed probability of an event with releases in specified range or above a certain value. If a failure of a given SSC leads to consequences above the threshold value, then that SSC is classified as safety related. An important distinction from the IAEA SSG-30 is that it considers not the SSC demand frequency but both the probability of SSC failure and success.

South African method presented in Refs. [20,21] considers failure frequency of an SCC. The frequency thresholds used in the risk matrix are the same as IAEA frequency range for Anticipated Operational Occurrences (AOOs) and for Design/Beyond Design Basis Accidents (DBA/BDDBA) i.e., $1E-06$.

It's important to note that safety classification process, in most cases, is based on Probabilistic Safety Assessment (PSA) data, and therefore is part of "PSA based knowledge system" as described in Refs. [22,23]. That system connects safety analysis to other major areas of plant management e.g. maintenance and operations. It is shown in Ref. [24] how safety classification of a research reactor in a risk-based approach can be derived from PSA results. Therefore, it makes sense to upgrade safety analyses of existing research reactors to at least PSA level I and Polish case is not an exception [25].

1.5. General procedure

From IAEA documents, a distilled general procedure of safety classification is established:

1. Fundamental safety functions are broken down into dedicated safety functions (consequence mitigation) and design provisions (accident prevention).
2. SSCs used as design provisions are assigned the safety class based on consequences of their failure.
3. Each safety function is assigned first a safety category that depends on:
 - a. The consequences of its failure,

- b. Demand frequency to fulfill the safety function – derived from the Initiating Event frequency,
- c. The role of the function in achieving controlled state (fast/automated response) or safe state (longer reaction time/activated by the operator).
4. All SSCs in the plant are assigned safety classes according to the safety category of the safety function they provide, considering the following factors:
 - a. Provided safety function – is SSC necessary for a given function?
 - b. Consequences of SSC failure – does the failure impact the function?
 - c. The frequency of the safety function demand (already considered in function category),
 - d. The time following the Initiating Event and the duration to perform SSC function.

At this point, one has to explain that Polish Atomic Law was inspired by the IAEA documents, referring to the SSCs directly, rather than to safety functions. In result, the Polish regulator placed a strong emphasis on calculation of demand probability for each SSC. In addition, Polish regulations do not consider factor "3c" in the classification process. However, it includes factors not directly pointed in SSG-30:

- possibility of Initiating Event upon the SSC failure,
- direct impact of SSC failure on safety function during normal operation and accident conditions.

In general sense, those factors are included in point "4b" (consequences of SSC failure) of the above procedure.

1.6. Missing insights

The IAEA documents on the SSC safety classification do not contain clear information on how the demand frequency (probability) should be included in the classification process. The missing information can be expressed in two questions:

1. What are the threshold values for high, medium, and low frequency events in the nuclear facility? Those thresholds are used to create a risk matrix considering consequences in terms of radiation dose thresholds, that are typically varying in different countries.
2. How to use a given demand frequency (probability) to amend a safety class of a given SSC?

During the work on the project, it was also established that in MARIA reactor there are several SSCs performing two or more safety functions. In the authors' assessment, this may be a more common case in research reactors than in NPPs.

In addition, a given safety function is typically used in response to more than one initiating event. According to IAEA SSG-30, each of the events identified for a specific safety function should be analyzed separately and the function should be assigned the highest category resulting from such analyses.

However, it is possible that demand frequency for a given SSC will be very low for each investigated safety function and each investigated IE, but when summarized it may reach high values, according to the risk matrix. The question is if it is necessary to consider the summarized demand frequency for each safety function when establishing safety categories.

As can be observed in the general procedure, the demand frequency is referred twice: first, it is considered for function categorization, and next for the SSC classification. However, according

to Ref. [4], since the probability of occurrence of the initiating event is considered in the categorization of the functions, this factor is already implicitly considered.

2. Integrated risk-informed safety classification (IRISC) procedure

2.1. Risk matrix development

From the SSG-30 it can be deduced what should be the approach to frequency thresholds used in SSC classification. Table 1 of [1] can be seen as a prototype of a risk matrix. The horizontal axis refers to the severity of consequences if the function is not performed, and on the vertical axis, various types of safety functions are broken down by plant states, from most often used (in AOOs) to least used (in design extension conditions). The threshold probability values for y-axis are given in Ref. [4], in Table 2. It is not fully defined and requires interpretation, except the fact that events with frequency above $1\text{E-}02/\text{year}$ are considered anticipated (AOOs). Therefore, that is the first threshold, and the second may be set as $1\text{E-}04/\text{year}$ (defined as design extension conditions without core melt) or $1\text{E-}05/\text{year}$ for the events with core damage. Flexibility in selecting the second threshold is desired as the probability values must make sense when considering allowed radiation doses (severity thresholds) in a given country.

Typically, the consequences of a losing a safety function are described in terms of radiation doses to workers and/or general population. Allowed and dangerous levels of radiation are specified in national regulations and those are used in safety classification process to specify low, medium, or high level of consequences. For an accident sequence with a loss of safety function, a deterministic calculation should be provided that estimates the doses.

An alternative way to describe the consequences is the amount of radioactive substance released to a closed space or environment in a specific scenario. IAEA provides guidelines for the radioactive consequence estimation, both in terms of radioactive dose and amount of radionuclides released [26,27].

Demand frequency, which describes how often a safety function will be called upon, must be calculated from PRA models. Typically, this means fault tree models that describe a logic of events (and their frequency) that lead to a certain IE and the use of analyzed safety function.

The risk matrix proposed in this work for the safety functions categorization is shown in Fig. 1.

The matrix proposed herein is the combination of Table 2 of [4] and Table 1 of [1]. For the AOOs the safety categories are the same as in Table 1 of [1]. For everything that is not considered a normal operation (demand frequency less than $1\text{E-}02/\text{year}$) the matrix is

		Function loss severity		
		High	Medium	Low
Demand frequency [annual]	High $1\text{E-}02$	Safety Category 1	Safety Category 2	Safety Category 3
	Medium $1\text{E-}05$	Safety Category 2	Safety Category 3	Safety Category 3
	Low	Safety Category 2 or 3	Not Categorized	Not Categorized

Fig. 1. Risk matrix for safety function categorization.

using categories as if all functions were used to reach a safe state (slower activation when compared to controlled state functions).

Such an approach makes the matrix compatible with other factors considered in the classification process. For example, functions used to mitigate high and medium consequences of events with medium frequency (mostly DBA) – if they are of faster activation type (to reach the controlled state), their safety category should increase by one.

2.2. Demand frequency summation

Up to this point, the described procedure follows closely the IAEA recommendations. The IAEA recommends [1] that if safety a function is used in case of more than one IE with varying consequences of failure, it receives the highest safety category identified among the events. In this work, a novel extension to this approach is proposed for research reactors.

If a safety function is used in multiple IEs, then the valid question is: are there functions used in multiple IEs that have low individual demand frequency, but when combined, it is in the AOO region (high frequency)?

In other words, the analysis results may suggest that the function is used rarely, but when the demand frequency from multiple IEs is combined it may be revealed otherwise. That is a relevant question because in research reactors it is more common to have safety systems that are multipurpose, as opposed to NPPs, where applicable IEs have dedicated safety devices. And the related question: if that is the case, should the combined demand frequency be used instead in the process of function categorization?

To address that issue, an extension to the procedure is proposed based on the IAEA requirements, where the demand frequency of a given safety function is combined with all initiating events, for which the function availability influences the accident sequences. The procedure is as follows:

$$\lambda_{i_demand} = \sum_{j=1}^m \lambda_{ij},$$

where: λ_{i_demand} is combined demand frequency for the i -th safety function, m refers to the number of IE sequences that use the i -th safety function, and λ_{ij} is the demand frequency for the i -th safety function used in the j -th IE sequence.

In cases where demand probability is used, the formula is:

$$P_{i_demand} = 1 - \prod_{j=1}^m (1 - P_{ij}),$$

where: P_{i_demand} is combined demand probability for the i -th safety function, m refers to the number of IE sequences that use the i -th safety function, and P_{ij} is the demand probability for the i -th safety function used in the j -th IE sequence.

2.3. Integrated procedure for safety classification

The combined demand frequency (probability) can be integrated into the IAEA recommended process by adding additional step to the function categorization procedure. The whole process is presented in Fig. 2. After the function category is assigned from the risk combination that results in highest category (according to matrix in Fig. 1), factors d and e are considered and their impact on functions category. Next, the demand frequency from the worst case IE is compared to the combined frequency value. If the combined frequency belongs to the higher frequency threshold, the function category should be increased. The rationale is that the SSCs

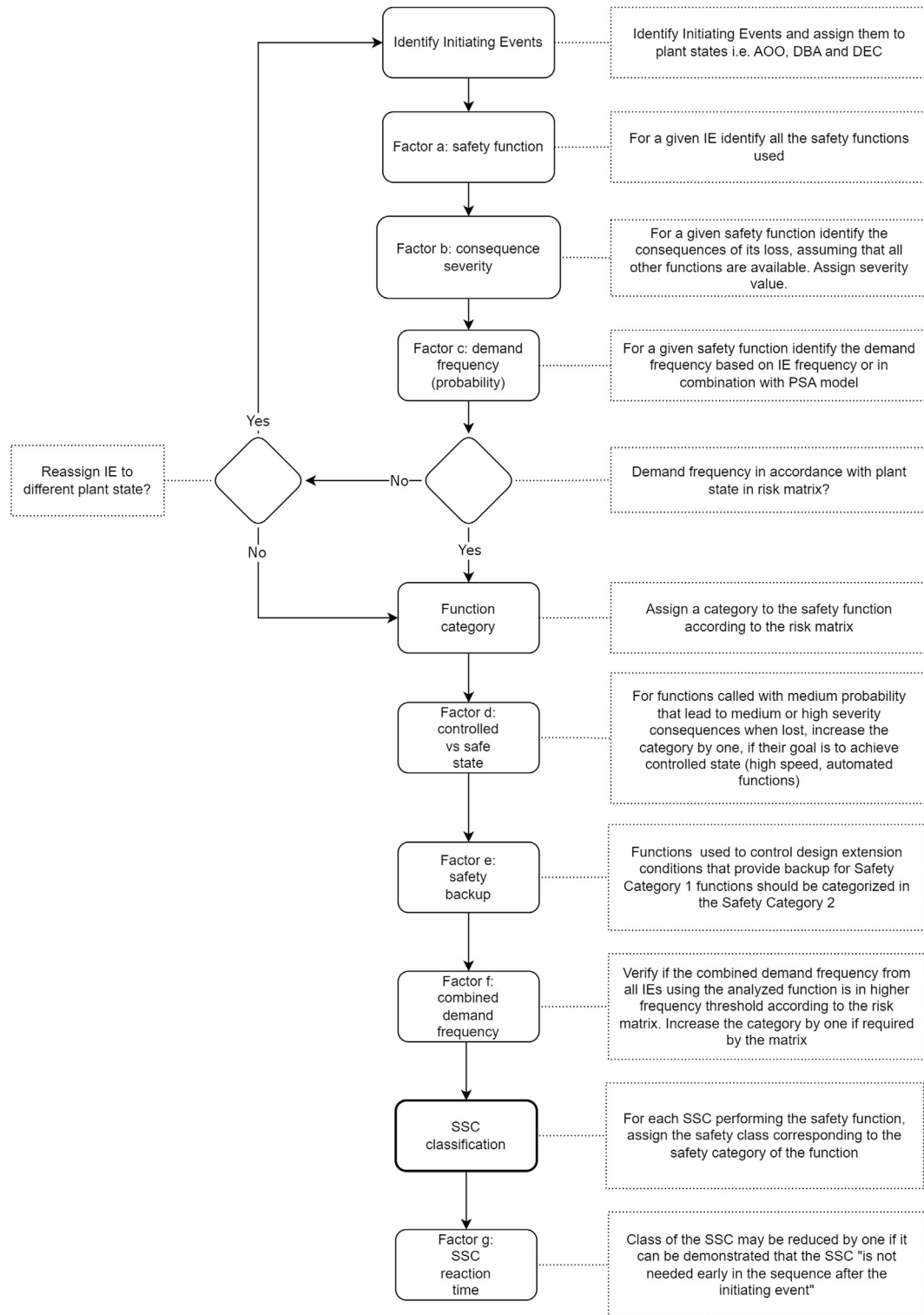


Fig. 2. Integrated procedure for SSCs safety classification.

belonging to the safety function will be activated more often than in worst case IE sequence. Therefore, a more conservative quality and reliability requirements from a higher safety class (following functions safety category) should be applied.

In most cases, safety category should be increased only once if any of the factors (*d*, *e*, *f*) is recognized applicable. For example, if a

safety function category is assigned from risk combination of medium frequency and medium severity (resulting in category 3), but the same function is used in multiple IEs, and combined demand frequency belongs to the high frequency threshold, then its category ought to be increased by one. If that function is used to reach controlled state as described by factor *d*, the safety category will be

increased by one and there is no need to increase it once more.

The second step from the end of the procedure, called “SSC classification” is the breaking point, where safety function category is inherited by all SSCs belonging to that function and becomes the safety class. The IAEA recommends [1] safety class to follow the same pattern as categories i.e., three classes with class one as the most important.

The IAEA suggests [1,4] that some SSCs, not belonging to any safety function, may still affect the safety performance upon failure. In such cases, they are classified according to the consequences of failure. This is a special case that was not included in the procedure in Fig. 2. The authors are aware that there may be more cases of safety solutions that are complex and difficult to classify. The presented flowchart, however, can be used to effectively classify the majority of them.

3. IRISC implementation in MARIA reactor

To demonstrate the applicability of the new procedure, the data from SSC classification for MARIA reactor was used:

1. Safety functions used in multiple initiating events,
2. Severity of consequences of safety function loss,
3. Demand frequency of safety functions (from PRA).

Table 1 provides the list of selected safety functions of MARIA reactor that are used in more than one IE sequence. For a given safety function it provides the demand frequency and loss severity in the worst case IE sequence. It also provides the combined demand frequency from all applicable IEs, as calculated according to the formula described in Section 2. Next, the table provides the safety category for a function, obtained from the risk matrix (Fig. 1), both for individual (the worst case IE) and combined demand frequency.

As can be observed, there are two safety functions that have the combined demand frequency in the higher frequency threshold than in the worst case IE.

Emergency reactor shutdown function is used in multiple (18) IE sequences and has in general high severity of loss. In the worst case scenario, its demand frequency is at medium level, but combined frequency is high, therefore, its safety category should be increased from two to one, in accordance with the risk matrix (Fig. 1). As this function is used to reach controlled state (fast response) its category would be increased before, according to the factor d (Fig. 2).

Reactivity increases rate detection function is used in two IE sequences, both with low severity of loss. Its demand frequency from the worst case is at medium level, but the combined frequency value is high. However, according to the matrix in Fig. 2, the category for functions with low loss severity should remain at three, both for medium and high demand frequency. The function is used

to reach controlled state after IE, but the factor d is applied only in case of medium to high loss severity. Therefore, the assigned category is three both for individual and combined demand frequency case.

The classification process for MARIA Research Reactor is not yet finalized, therefore the authors decided not to reveal the actual values of demand frequency or assigned categories for all safety functions. The results are currently under review from the regulatory body and will be discussed separately.

However, the results that are presented herein suggest that the new approach, as developed under this work, is sensitive to the case of high demand frequency in the multiple-use safety functions, the availability of which influences the accident sequences originated by several initiating events. It may be concluded that the proposed extension provides better risk coverage in the safety classification process.

In this context, the term *better risk coverage* means that the new approach in a more consistent way identifies the cases of the multiple-use functions, for which the demand frequency may be underestimated when the applicable initiating events are analyzed separately. Consequently, the application of the proposed solution, where several initiating events provide their contributions to the demand frequency of the safety functions, may result in some of them being moved to higher thresholds and thus higher safety categories. Finally, the SSCs of a nuclear facility receive the safety class that better reflects the risk associated with the safety functions they provide.

Although, in general, the proposed method for inclusion of the contributions from several initiating events into the functions' demand frequency tends to increase its resultant value, the new solution does not provide overly conservative results. Specifically, not every change of frequency demand originating from the proposed extension will produce higher safety class. As explained above, in some cases, the increased safety class may arise due to the factors other than the demand frequency, according to the original IAEA framework.

These conclusions have been confirmed by the collective results of the SSCs safety classification process performed for the MARIA Research Reactor. In this practical application of the new risk-integrated approach to classification, only in very few cases the categories of the functions have been increased due to the cumulative impact of several initiating events, while the majority of them remained unchanged when compared to the standard IAEA procedure. In this context, the proposed method can be considered useful for the identification and improvement of those cases, that should be treated with higher precision. This is especially important in the case of new and unique research reactors, where the variety of specific design solutions, operating modes, and operation conditions favor the implementation of multiple-use safety functions.

Table 1
Categorization of selected safety functions of MARIA Research Reactor.

Safety function	Loss Severity	Freq. Individual	Freq. Combined (# IE used)	Category (individual)	Category (combined)
Reactor overpower detection	Low	High	High (3)	3	3
Reactivity increases rate detection	Low	Medium	High (2)	3	3
Emergency reactor shutdown	High	Medium	High (18)	2	1
Post shutdown decay heat removal	High	High	High (18)	1	1
Fuel cooling parameters monitoring	High	High	High (11)	1	1
Fire alarm system	Low	High	High (5)	3	3
Manual reactor shutdown	High	High	High (7)	1	1
Fuel channel leakage detection	High	High	High (4)	1	1
Biological protection for reactor and spent fuel pool	Low	Medium	Medium (2)	3	3
Offsite releases prevention	Medium	High	High (5)	2	2

4. Conclusions

The presented work captures details of the probabilistic study, performed in the framework of the safety classification process for the MARIA Research Reactor. The authors present a novel extension of the general procedure for the safety classification process that can be applied to research reactors. The article describes two probabilistic procedures that were missing detailed explanations in the literature on the safety classification of research reactors:

- How demand frequency of a safety function can be considered in the process of assigning categories to safety functions.
- How the frequency thresholds in the safety categorization matrix are derived from the IAEA documents on the safety classification process.

The authors propose a probabilistic extension to the IAEA procedure that addresses a problematic case of safety functions used in multiple initiating events that have high combined demand frequency – the case that is particularly important for the safety of research reactors. This is specifically an important contribution to new research reactor projects that can reuse the presented approach.

The data from safety classification of MARIA Research Reactor show that the proposed extension does not produce dramatically different results when compared to the procedure described by IAEA and can be concluded compatible with the IAEA recommendations.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Acknowledgements

The authors gratefully acknowledge the support provided by Dr. Sławomir Potemski, Dr. Aleksej Kaszko, Bartłomiej Piwowski, Aleksandra Niepokólczycka-Feník, and Paweł Nowakowski from the National Centre for Nuclear Research.

References

- [1] International Atomic Energy Agency (IAEA), IAEA Safety Standards Series No. SSG-30: Safety Classification of Structures, Systems and Components in Nuclear Power Plants, IAEA, Vienna, Austria, 2014.
- [2] M. Borysiewicz, K. Kowal, S. Potemski, An application of the value tree analysis methodology within the integrated risk informed decision making for the nuclear facilities, *Reliab. Eng. Syst. Saf.* 139 (2015) 113–119, <https://doi.org/10.1016/j.res.2015.02.013>.
- [3] International Atomic Energy Agency (IAEA), IAEA Safety Standards Series No. SSR-3: Safety of Research Reactors, IAEA, Vienna, Austria, 2016.
- [4] International Atomic Energy Agency (IAEA), IAEA-TECDOC-1787: Application of the Safety Classification of Structures, Systems and Components in Nuclear Power Plants, IAEA, Vienna, Austria, 2016.
- [5] U.S., Nuclear Regulatory Commission (US NRC), 10FCR50 Appendix A: General Design Criteria for Nuclear Power Plants, US NRC, Washington DC, USA, 2004.
- [6] U.S., Nuclear Regulatory Commission (US NRC), Regulatory Guide 2.5: Quality Assurance Program Requirements for Research and Test Reactors, US NRC, Washington DC, USA, 1989.
- [7] American National Standards Institute (ANSI), ANSI/ANS-15.8-1995 (R2013): Quality Assurance Program Requirements for Research Reactors, ANSI, Washington DC, USA, 2013.
- [8] U.S., Nuclear Regulatory Commission (US NRC), 10CFR50.69: Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors, US NRC, Washington DC, USA, 2004.
- [9] I.S. Kim, S.K. Ahn, K.M. Oh, Deterministic and risk-informed approaches for safety analysis of advanced reactors: Part II, Risk-informed approaches, *Reliab. Eng. Syst. Saf.* 95 (2010) 459–468, <https://doi.org/10.1016/j.res.2009.12.004>.
- [10] M.C. Cheok, G.W. Parry, R.R. Sherry, Use of importance measures in risk-informed regulatory applications, *Reliab. Eng. Syst. Saf.* 60 (1998) 213–226, [https://doi.org/10.1016/S0951-8320\(97\)00144-0](https://doi.org/10.1016/S0951-8320(97)00144-0).
- [11] J.S. Ha, P.H. Seong, A method for risk-informed safety significance categorization using the analytic hierarchy process and bayesian belief networks, *Reliab. Eng. Syst. Saf.* 83 (2004) 1–15, <https://doi.org/10.1016/j.res.2003.08.002>.
- [12] J. Wang, F. Wang, J. Wang, S. Chen, L. Hu, Y. Li, Ch Li, A new importance assessment method for risk-informed SSC categorization, *Int. J. Energy Res.* 42 (2018) 1779–1786, <https://doi.org/10.1002/er.3985>.
- [13] M. Sun, J. Wang, F. Wang, J. Yu, Y. Yin, A new integrated SSCs safety classification method and a case study on CN HCCB TES, *Ann. Nucl. Energy* 145 (2020), 107594, <https://doi.org/10.1016/j.anucene.2020.107594>.
- [14] The American Society of Mechanical Engineers (ASME), ASME Boiler and Pressure Vessel Code Section III, ASME, New York, USA, 2017.
- [15] J.-E. Holmberg, I. Männistö, Risk-informed classification of systems, structures and components, *Rakenteiden Mekaniikka* 41 (2008) 90–98, http://rmseura.tkk.fi/rmlehti/2008/nro2/RakMek_41_2_2008_3.pdf.
- [16] K. Kunitomi, S. Shiozawa, Safety design, *Nucl. Eng. Des.* 233 (2004) 45–58, <https://doi.org/10.1016/j.nucengdes.2004.07.010>.
- [17] T.-R. Kim, Safety classification of systems, structures, and components for pool-type research reactors, *Nucl. Eng. Technol.* 48 (2016) 1015–1021, <https://doi.org/10.1016/j.net.2016.02.009>.
- [18] Z. Wu, S. Xi, Safety functions and component classification for the HTR-10, *Nucl. Eng. Des.* 218 (2002) 103–110, [https://doi.org/10.1016/S0029-5493\(02\)00202-9](https://doi.org/10.1016/S0029-5493(02)00202-9).
- [19] Idaho National Laboratory (INL), INL/EXT-10-19509: Next Generation Nuclear Plant Structures, Systems, and Components Safety Classification White Paper, INL, Idaho Falls, USA, 2010.
- [20] K.J. Kang, S.I. Wu, J. Yoon, Introduction to South Africa's safety classification, in: *Transactions of the Korean Nuclear Society Autumn Meeting*, 2012, Gyeongju, Korea, October 25–26.
- [21] K. Moodley, D.A.H. Arndt, S.M. Malaka, A case study SAFARI-1: implementation of the safety classification in the existing facilities using a graded approach, in: *European Research Reactors Conference Proceedings*, March, Berlin, Germany, 2016, pp. 13–17.
- [22] P.V. Varde, S. Sankar, A.K. Verma, An operator support system for research reactor operations and fault diagnosis through a connectionist framework and PSA based knowledge based systems, *Reliab. Eng. Syst. Saf.* 60 (1998) 53–69, [https://doi.org/10.1016/S0951-8320\(97\)00154-3](https://doi.org/10.1016/S0951-8320(97)00154-3).
- [23] P.V. Varde, M.G. Pecht, Role of prognostics in support of integrated risk-based engineering in nuclear power plant safety, *Int. J. Prognostics Health Manag.* 3 (2012) 1–23, <https://doi.org/10.36001/ijphm.2012.v3i1.1362>.
- [24] D. Grabaskas, J. Andrus, D. Henneke, J. Li, M. Bucknor, M. Warner, Development of the versatile test reactor probabilistic risk assessment, *Nucl. Sci. Eng.* 196 (2022) 278–288, <https://doi.org/10.1080/00295639.2021.2014741>.
- [25] M. Maskin, P.P. Tom, T.A. Lanyau, F.C.M. Brayon, F. Mohamed, M.F. Saad, A.R. Ismail, M.P.H. Abu, Development and methodology of level 1 probability safety assessment at PUSPATI TRIGA reactor, *AIP Conf. Proc.* 1584 (2014) 240–244, <https://doi.org/10.1063/1.4866138>.
- [26] International Atomic Energy Agency (IAEA), IAEA Safety Standards Series No. GSR Part 3: Radiation Protection and Safety of Radiation Sources, International Basic Safety Standards, IAEA, Vienna, Austria, 2014.
- [27] International Atomic Energy Agency (IAEA), EPR-D-VALUES 2006: Dangerous Quantities of Radioactive Material, (D-values), IAEA, Vienna, Austria, 2006.