Review of SMR Risk Importance for Risk-informed Safety Classification

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1. Introduction

The emergence of small modular reactors (SMRs) represents a fundamental shift in the nuclear safety paradigm from reliance on active safety systems to passive safety features, and from large, centralized power generation to distributed and modular deployment concepts. Recent SMR designs, including US460 of NuScale Power [1], SMR-160 of Holtec International [2], and innovative small modular reactor (i-SMR) of Korea, are explicitly engineered from the conceptual design stage to prevent or mitigate severe accident scenarios. As a result, these designs are targeting core damage frequencies (CDFs) that are orders of magnitude lower than those of conventional large light-water reactors.

However, such substantial safety improvements may introduce unintended challenges within the existing risk-informed regulation (RIR) of United States, particularly in safety classification based on relative risk importance measures such as risk achievement worth (RAW) under 10 CFR 50.69 [3]. As the baseline plant risk decreases to significantly low levels, RAW values may become large, reflecting a mathematical amplification effect inherent in ratio-based metrics rather than indicating a meaningful increase in actual risk.

This paper demonstrates that the issue is not a numerical artifact, but a structural limitation of applying conventional relative risk metrics to significantly low risk reactor designs. This limitation may undermine both the economic competitiveness of SMRs and the regulatory efficiency intended by risk-informed approaches. To resolve this inconsistency, the paper examines recent regulatory practices and proposes three complementary measures; (1) adoption of absolute risk-based criteria and (2) enhancement of the qualitative authority of the integrated decision-making panel (IDP).

2. Methods and Results

2.1 Limitations of Applying Risk Importance Measures to SMRs

Since the publication of the Reactor Safety Study (WASH-1400) in 1975 [4], probabilistic safety assessment (PSA) has become a standard tool for quantifying nuclear power plant risk. Early PSA applications focused primarily on identifying plant vulnerabilities. Following the U.S. NRC's PRA Policy Statement in 1995 [5], however PSA evolved into a instrument for regulatory decision-making, enabling risk-informed allocation of regulatory resources while alleviating excessive conservatism inherent in deterministic safety analysis (DSA).

For operating large reactors, plant-level CDFs typically fall within the range of 10^{-5} to 10^{-6} /reactor-critical-year(rcyr). Within this range, risk importance measures such as RAW fussell-vesely (FV) provide intuitive and reasonably discriminative indicators of component safety significance.

In contrast, modern SMRs employ extensive passive safety systems, integral primary systems, and reduced source terms, targeting CDF levels as low as 10^{-7} to 10^{-9} /rcyr. While these values represent remarkable safety improvements, often exceeding conventional reactors by two to three orders of magnitude, they also expose inherent limitations of relative importance metrics. The RAW is defined as Equation 1.

$$RAW = \frac{CDF(x_i = 1)}{CDF_{base}} \qquad \text{Equation 1}$$

CDF_{base} represents the total CDF of the plant, that is, sum of all cut-sets corresponding to combinations of component failures that can lead to $CDF(x_i = 1)$ denotes the conditional total CDF assuming the failure of component x_i. In other words, if component x_i is an important component that appears frequently in cut-sets, the variability of the indicator associated with that component can be expected to be large.

From the perspective of applying RAW to SMRs, three issues can be considered as Table I.

Table I: Issues of Applying RAW to SMRs

	Descriptions
1	<u>Explosion of RAW due to very low CDF</u> In SMRs, RAW can become excessively large because the denominator CDF_{base} may be extremely small and approach zero. Even if the conditional risk in the numerator increases only marginally, the resulting ratio can increase explosively by several orders of magnitude.

2	High RAW values for components in passive with low failure rates In SMRs, components in passive systems with very low failure rates can still exhibit very high RAW values. For example, in large reactors, a component with a failure rate on the order of 10^{-2} to 10^{-3} may increase certain cut-sets by roughly a factor of 100. However, for key components in passive systems (e.g., heat exchangers, water storage tanks) with failure probabilities on the order of 10^{-7}, the same effect can increase certain cut-sets by approximately 10^{7} times.
3	Limitations of existing RAW criteria for component classification From a component classification perspective, conventional RAW thresholds have inherent limitations. A RAW value of two for a given component means that, if the component fails, the CDF increases by a factor of two relative to a reference CDF of a typical large reactors, commonly taken as 1.0×10^{-5}/rcyr (i.e., an increase to 2.0×10^{-5} /rcyr). However, for an SMR with a baseline CDF on the order of 1.0×10^{-9}/rcyr, applying the same criterion implies that a component with RAW = 2 increases the CDF only to about 2.0×10^{-9} /rcyr. Despite this very small absolute increase in risk, such a component may still be classified as risk-significant under conventional RAW-based criteria.

Although this conclusion is mathematically correct, from an engineering and regulatory perspective it may convey a distorted message namely that “the safest SSCs are the most dangerous.” This can ultimately lead to unnecessary regulatory burden, increased costs, and inefficient use of resources.

These mathematical characteristics of RAW become particularly problematic when directly embedded into regulatory framework such as 10 CFR 50.69.

2.2 Regulatory Philosophy of 10 CFR 50.69

The 10 CFR 50.69 allows the reclassification of nuclear power plant SSCs based not on traditional safety class, but on their actual risk contribution [3]. The concept of this regulation is that event safety-related SSCs, if their actual risk contribution is low (Low Safety Significant, LSS), may be subject to relaxed quality assurance (QA) and testing requirements. Conversely, non-safety-related SSCs with high risk contribution (High Safety Significant, HSS) are subject to enhanced control and management.

The technical procedures for this process are described in NEI 00-04 [6]

· Sum of F-V for all basic events modeling the SSC of interest, including common cause events > 0.005

· Maximum of component basic event RAW > 2

· Maximum of applicable common cause basic events RAW values > 20

2.3 Comparison of Applicability of Importance Measures between Conventional Large Reactors and SMRs

A example of applying the existing criteria to conventional large reactors and SMRs are shown in Table II.

Table II: Example of RAW calculations

Case	Descriptions
A (Large)	$CDF_{base} = 1.0 \times 10^{-5}$/rcyr $CDF(x=1) = 3.0 \times 10^{-5}$/rcyr $\Delta CDF = 2.0 \times 10^{-5}$/rcyr $RAW = \frac{CDF(x=1)}{CDF_{base}} = 3.0$ ■RAW > 2.0 and HSS $\rightarrow$ *Appropriate*
B (SMR)	$CDF_{base} = 1.0 \times 10^{-9}$/rcyr $CDF(y=1) = 1.0 \times 10^{-6}$/rcyr $\Delta CDF \cong 1.0 \times 10^{-6}$/rcyr $RAW = \frac{CDF(y=1)}{CDF_{base}} = 1{,}000$ ■RAW > 2.0 and HSS $\rightarrow$ *Inappropriate*

In case A, for a conventional large reactor, the target component x may be assumed to be an emergency diesel generator (EDG) with a failure probability on the order of 10^{-2}. For all cut-sets contributing to the base CDF, the ratio of the cut-set assuming EDG failure corresponds to the RAW of the EDG, which exceeds the criterion of 2 and is therefore appropriately classified as high risk.

In case B, even if component fails, to total plant risk remains at the level of 10^{-6}/yr, which is still extremely safe when compared with the typical nuclear safety goal of 10^{-5}/yr. However, the RAW metric leads to an inappropriate classification of HSS. In case B, component y may be assumed to be a heat exchanger, a key component of a passive safety system. The failure probability of such a heat exchanger is on the order of 10^{-7}. In SMRs, where passive safety systems play a dominant role, the proportion of cut-sets including this heat exchanger becomes relatively large. As a result, even when calculated in the same manner as case A, the RAW value becomes excessively large.

That is, as SMR designers introduce passive safety systems and successfully reduce overall plant risk, more SSCs may be classified as HSS. This phenomenon becomes a critical factor affecting the economic viability of SMRs, as it may lead to the concentration of resources on SSCs of limited importance, ultimately reducing overall safety management efficiency.

2.4 Suggestions

2.4.1 Introduction of an Absolute Risk Criterion

To address the issues described above, the most fundamental solution adopted by the U.S. nuclear industry and regulatory authorities is to replace relative measures, such as RAW, with absolute risk increase criteria.

NuScale Power, the first SMR developer to pursue U.S. NRC standard design approval (SDA), encountered this issue directly. The internal events CDF of NuScale was evaluated to be on the order of 10^{-9}/mcyr [7]. Under the conventional RAW criterion (>2.0), this would imply that most pumps and valves in the plant would be classified as HSS, which is with significant regulatory implication.

In response NuScale proposed a new methodology through a technical report entitled Risk Significance Determination [8]. Rather than relying on the magnitude of RAW itself, NuScale adopted the absolute value of the conditional core damage frequency (CCDF) reached when a given component is assumed to fail, as the primary basis for evaluating component importance.

■ If the CCDF under component failure is less than 10^{-6}/yr, the component is not considered risk-significant, regardless of how large its RAW value may be, even if on the order of thousands or tens of thousands.

■ The threshold of 10^{-6}/yr is derived from the NRC's safety goals (CDF and LRF), reflecting the concept that risks below this level are negligible with respect to public health and safety.

The U.S. NRC, after reviewing NuScale's approach, agreed that strictly applying conventional relative risk criteria to designs with significantly low CDF is inappropriate. Furthermore, the NRC concluded that NuScale's approach is consistent with the underlying philosophy of 10 CFR 50.69, which emphasizes focusing regulatory attention and resources on equipment that is truly important to safety, and there fore approved the topical report.

Similar concerns about relative risk importance measures have also been recognized by other SMR developers, including Holtec International who has submitted a related topical report [2]. The key elements of this approach are summarized in Table III.

Table III: Suggestions of Holtec International

SSC Parameter	Descriptions
Component level basic event	$CCDF \geq 3.0 \times 10^{-6}/yr$
System level basic event	$CCDF \geq 1.0 \times 10^{-5}/yr$
Component level basic event	$CLRF \geq 3.0 \times 10^{-7}/yr$
System level basic event	$CLRF \geq 1.0 \times 10^{-6}/yr$
Basic event/contributor	$Total\ FV \geq 0.2$

2.4.2 Strengthening the Qualitative Authority of the Integrated Decision-Making Panel

From a practical perspective, the most important strategy is to strengthen the authority and role of the integrated decision-making panel (IDP), responsible for determining safety significance categories (RISC-1~4). The NEI 00-04 guidance explicitly states that PSA results serve only as the starting point for safety categorization, not as the final determination [6]. The IDP is empowered to review the quantitative PSA results and to adjust the final safety categorization by comprehensively considering deterministic safety analyses, defense-in-depth (DID), safety margins, and other relevant factors.

In SMRs, which rely heavily on passive safety systems, the linkage between the failure of a specific active component and the occurrence of an actual accident is relatively weak. Based on these design characteristics, the IDP may override or discount safety categorizations derived solely from RAW-based criteria.

2.4.3 Example of i-SMR and Insights

To evaluate the impact of basic events or systems included in the i-SMR PSA model, an importance analysis was performed. As importance measures, the FV importance and RAW were applied.

Table IV represents the top 15 basic events ranked by FV importance. The analysis results indicate that all of the top 15 basic events significantly exceed a RAW value of 2.0.

In particular, although the failure probability of the relevant components is as low as 2.44×10^{-7}, the RAW values of component 4 and 5(event number 6 and 7) were evaluated to be on the order of 296,005, representing a typical example of this phenomenon. As shown in the Table IV, the same behavior observed for component 4 and 5 is also evident in other fault trees.

These results confirm the need to reconsider the validity of the RAW-based approach, which derives numerical importance values by deterministically assuming complete failure of a component. This finding demonstrates that conventional RAW-based importance measures may lead to systematic overestimation of risk significance in i-SMR.

The risk importance of i-SMR in Table IV can be different by PSA model updates with design changes.

Table IV: Example Results of Risk Importance for i-SMR

Number	Basic Events		Probability	RAW	FV
1	C	Component 1	1.41×10^{-4}	2,581	0.364
2	C	Component 2	7.64×10^{-3}	45	0.339
3	H	HEP 1	4.61×10^{-3}	39	0.177
4	C	Component 3	3.19×10^{-3}	45	0.141
5	W	CCF 1	2.86×10^{-5}	3,938	0.113
6	C	Component 4	2.44×10^{-7}	296,005	0.072
7	C	Component 5	2.44×10^{-7}	296,005	0.072
8	W	CCF 2	1.05×10^{-5}	5,142	0.054
9	W	CCF 3	1.05×10^{-5}	5,142	0.054
10	W	CCF 4	1.05×10^{-5}	5,142	0.054
11	W	CCF 5	1.05×10^{-5}	5,142	0.054
12	W	CCF 6	1.05×10^{-5}	5,142	0.054
13	W	CCF 7	1.05×10^{-5}	5,142	0.054
14	C	Component 6	1.16×10^{-3}	45	0.051
15	W	CCF 8	9.15×10^{-6}	5,142	0.047
C: Component Failure Probability, H: Human Error Probability, W: CCF probability					

3. Conclusions

The innovative small modular reactor (i-SMR) currently under development in the Republic of Korea features passive safety systems, the elimination of large-break LOCA, diversified systems, and an integrated reactor vessel. As a result, the target CDF is significantly low on the order of 1.0×10^{-9}/mcry. This study has shown that under such condition, the importance evaluation methods developed for conventional nuclear power plants may lead to misleading results. In particular, in PSA models that explicitly represent passive safety systems, SSCs that preserve system functions may appear to have disproportionately high importance, even though their actual contribution to plant risk is minimal.

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