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Integrated Vital Area Identification Method for SMRs and Conventional Nuclear Power Plants

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1. Introduction

1.1 Background

The physical protection design of a nuclear power plant involves identifying vital areas and subsequently developing physical protection systems to safeguard those areas. Prior to the publication of Revision 5 of the International Atomic Energy Agency (IAEA) recommendations (INFCIRC/225/Rev.5)[1], vital areas were designated based on whether a facility could result in Unacceptable Radiological Consequences (URC). However, with the adoption of INFCIRC/225/Rev.5, the criteria have been revised such that vital areas are now identified for facilities that could cause High Radiological Consequences (HRC).

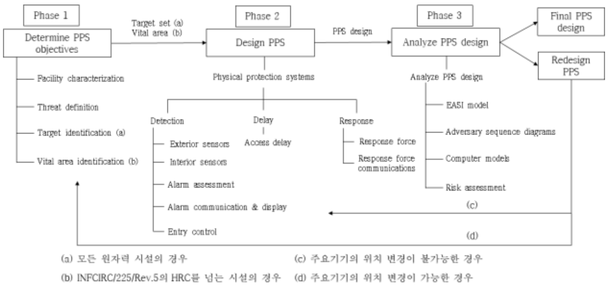

Figure 1. Process for designing and evaluating the PPS[2,3]

Nuclear facilities follow the physical protection system (PPS) design process illustrated in Figure 1[2,3]. In Phase 1, the objectives of physical protection for a nuclear facility are clarified. In Phase 2, (1) for facilities exceeding the HRC threshold, the PPS must be designed to provide focused protection for the identified vital areas, and (2) for facilities not exceeding the HRC threshold, the PPS should be appropriately designed to protect the facility based on the identified target set. In Phase 3, the adequacy of the PPS design is evaluated (see [4]).

Since the 1970s, VAI methodologies have evolved from the first-generation fault-tree approach in the United States to the third-generation event-tree approach in Korea. To meet evolving regulatory requirements, these methods were further enhanced, resulting in the development of a 3.5-generation VAI methodology that ensures both technical completeness and regulatory compliance.

1.2 Objective

Research on the development and application of domestic methodologies, as well as their implementation by nuclear power plant operators, has included evaluations of key elements. In particular, 12 key factors for target set identification were derived, and their application was assessed in large commercial nuclear power plants [5], while potential strategies for small modular reactors (SMRs) were also explored [6].

Nevertheless, no study has yet presented an integrated analysis framework that simultaneously reflects the latest methodologies for vital area identification while also addressing applicability to SMRs. Accordingly, this paper proposes a comprehensive and integrated method for vital area identification, applicable to both SMRs and large commercial nuclear power plants.

2. Existing Advanced Vital Area Identification Method and Additional Considerations

2.1 Existing Advanced Vital Area Identification Method

In nuclear power plant physical protection design, the identification of vital areas constitutes essential information. Methods for identifying vital areas are generally categorized into first-, second-, and third-generation approaches, depending on how the sabotage fault tree is developed. The first-generation method, developed in the United States [7,8,9], directly constructs a sabotage fault tree in which compartment failures are defined as basic events. The second-generation method, developed in Korea [10,11], reuses the integrated PSA model by substituting equipment failures with compartment failures to develop a sabotage fault tree. The third-generation method, also developed in Korea [12,13], simplifies the VAI process by utilizing PSA event trees. Table I summarizes the main features of these methodologies.

Table I: Main features of 1st, 2nd, and 3rd generation VAI methodologies

Generation	Main Features
1st Generation	- 1970s, USA - Developed Direct sabotage fault tree to identify vital areas
2nd Generation	- Early 2000s, Korea - Identified vital areas based on PSA integrated model

	- More efficient than Direct sabotage fault tree method because it reuses already developed PSA models - Limited identification of vital areas due to the large scale of PSA models; only partial identification possible
3rd Generation	- 2017, Korea - Identified vital areas based on PSA event trees - Simplifies PSA event trees by applying basic assumptions and characteristics of sabotage analysis - Expansion of minimal protection sets at the event tree header level into plant-compartment-level protection sets, enabling efficient identification - Does not generate information on compartment-level attack sets

Building on the third-generation PSA event tree–based approach, the existing advanced method, referred to as the 3.5-generation VAI method[14], was developed. Its key feature is the application of basic assumptions and characteristics related to sabotage analysis to PSA models, with the event trees simplified for practical use. By using these simplified event trees, minimal protection sets at the header level are converted into plant-compartment-level protection sets, thereby enabling highly efficient vital area identification.

Furthermore, the 3.5-generation method introduces additional rules for VAI into the third-generation framework. As a result, it provides a more realistic and technically complete methodology, enabling the development of more compact sabotage fault trees. In other words, the latest 3.5-generation method applies not only the basic assumptions and rules but also additional rules for VAI to PSA models, and the overall procedure is implemented through the following steps.

Table II: 3.5-Generation Vital Area Identification Method[5]

Step	Description	Detail
1	Selection of Initial Events for Analysis	Review PSA initial events and exclude those that are unlikely to be caused by sabotage or cannot be modeled as event trees for sabotage scenarios.
2	Simplification of Event Trees	Review PSA event trees and simplify them to reflect the characteristics of physical protection aspects.
3	Calculation of Target Sets at the Event Tree Header Level	Calculate the target sets formed at the event tree header level.
4	Calculation of Protection Sets at the Event Tree Header Level	Calculate the protection sets formed at the event tree header level.
5	Prioritization of Protection Sets at the Event Tree Header Level	Through expert review, select the most cost-effective protection set from among those identified at the event tree header level, considering the physical protection perspective.
6	Transformation into Compartment-Level Protection Sets and Final Selection of Vital Areas	Convert the selected event-tree-level protection sets into compartment-level protection sets, and designate the compartments included in the protection sets as the final vital areas.

As described above, the 3.5-generation advanced VAI method maximizes the use of PSA results in order to minimize the time and effort required for analysis. The basic assumptions, rules, and additional rules applied in the 3.5-generation VAI method are presented in Table 3.

Table III: 3.5-Generation Vital Area Identification Method[12]

Basic Assumption 1	The reactor building and main control room are assumed to be protected against sabotage. This means that all equipment inside the reactor building, including the reactor vessel, is not damaged by sabotage, and that all operator actions performed in the main control room can still be carried out even under sabotage conditions.
Basic Assumption 2	If sabotage results in plant damage, offsite power is assumed to be lost and not recoverable within a short period. In other words, equipment related to offsite power supply is assumed to be damaged by sabotage prior to the damage of other systems within the protected area.
Rule 1	If a component associated with a compartment damaged by sabotage can itself be damaged by sabotage, the corresponding component-failure basic event is replaced with a compartment-failure event and included in the sabotage fault tree.
Rule 2	If a component associated with a compartment damaged by sabotage fails randomly, the corresponding component-failure basic event is excluded from the sabotage fault tree.
Rule 3	If a component associated with a compartment damaged by sabotage operates in a fail-safe manner, the corresponding component-failure basic event is excluded from the sabotage fault tree.
Rule 4	Fire or flooding events that propagate probabilistically are excluded from the sabotage fault tree.
Additional Rule 1	When unavailability of offsite power is already reflected by Basic Assumption 1, initiating events that lead to plant damage conditions identical to those of a loss-of-offsite-power initiating event are excluded.
Additional Rule 2	If an initiating event caused by the failure of a specific safety-related system does not produce a new core damage sequence—i.e., when the system unavailability yields the same effect as in a loss-of-offsite-power accident sequence—the corresponding sequence is excluded from the PSA single-fault tree development.
Additional Rule 3	Event tree headers that are not directly subject to the basic assumptions, rules, or additional rules, but are indirectly affected by their application, should be appropriately reflected in accordance with the influence of these assumptions and rules.
Additional Rule 4	In the case of operator action failures performed outside the main control room, the compartments where the actions are carried out and the paths traversed by operators to perform those actions must be explicitly considered in the development of the sabotage fault tree. In other words, the operator-action basic events must be replaced with the compartments where the operator actions are performed and the compartments included along the access paths.

Details on the key factors identified in VAI methodologies [5] and their application in actual VAI analyses for large domestic commercial nuclear power plants [15] can be found in the corresponding reports [5,15].

2.2 Additional Considerations for Vital Area Selection

Reports presenting methodologies for VAI for SMRs have confirmed that the existing advanced VAI method can be sufficiently applied to the NuScale SMR. Nevertheless, these reports also suggest that VAI for SMRs should consider the unique design and operational characteristics of SMRs, as well as the specific features of SMR PSA models, by taking into account the following possibilities:

・The occurrence of initiating events caused by sabotage beyond those considered in PSA

・The validity of the basic assumptions used in existing VAI methods when applied to SMRs

・The validity of the rules for vital area identification in existing VAI methods when applied to SMRs.

3. Integrated Vital Area Identification Method

3.1 Basic Assumptions and Rules for Vital Area Identification Analysis

In the integrated VAI method applicable to both SMRs and large commercial nuclear power plants, the existing scope of target areas for VAI is applied without modification. The scope of vital areas is presented in Table 4.

Table IV: Scope of vital areas in nuclear power plants[16]

Classification	Vital Area	Remarks
Areas required to be designated as vital areas under 10 CFR 73.55(e)(9)(v)	Main Control Room (MCR)	Main control room for reactor operation
	Spent Fuel Storage Facility	
	Central Alarm Station (CAS)	Central alarm station for physical protection
	Secondary Alarm Station (SAS)	Secondary alarm station for physical protection
Areas required to be designated as vital areas under 10 CFR 73.55(e)(9)(vi)	Protected area for the installation of security alarm systems and backup power supply for non-portable communication equipment	
Vital areas designated through vital area identification analysis	Reactor Building	Selected based on the basic assumptions for each vital area type
	Rooms containing equipment required to prevent core damage from sabotage during power operation	Identified through PSA-based vital area identification methodology
	Rooms containing equipment required to prevent core damage from sabotage during reactor shutdown or refueling	Identified through PSA-based vital area identification methodology

In addition, the basic assumptions, rules, and additional rules applied in the existing VAI methodology are equally applicable to the integrated approach.

3.2 Selection of Sabotage-Initiating Events

According to the results of applying the existing VAI methodology to both large commercial nuclear power plants and the NuScale SMR, the initiating events and event trees modeled in the probabilistic safety assessment (PSA) were found to encompass all initiating events and accident sequences that could potentially be caused by sabotage. However, in the integrated VAI analysis proposed in this study, to ensure methodological completeness, a review is conducted to verify whether all sabotage-initiating events have been modeled in the PSA. If any such events are not represented, a procedure is included to add the missing initiating events to the PSA model. The procedure for selecting sabotage-initiating events, reflecting this addition, is presented below.

Table V: Scope of vital areas in nuclear power plants

Step 1	- Identify initiating events that could be caused by sabotage, considering sabotage-related assumptions and conditions such as the Design Basis Threat (DBT). - Verify whether the initiating events identified as potentially caused by sabotage are modeled in the PSA. - If any identified initiating events are not modeled in the PSA, add them and update the PSA model.
Step 2	Apply the vital area identification (Basic Assumption 1), (Basic Assumption 2), (Additional Rule 1), and (Additional Rule 2) to select the initiating events to be analyzed for use in the vital area identification process.

3.3 Simplification of event trees

For target/prevention set analysis at the PSA event tree heading level, the event trees corresponding to the initiating events selected for VAI analysis are simplified by applying the basic assumptions, rules, and additional rules described below.

Table V: Assumptions and Rules for Event Tree Simplification in Integrated vital area identification method

Item	Related Action
Basic Assumption 1	For event tree headings unrelated to damage of systems other than the reactor building and the main control room, remove the failure branch accident sequences and delete the corresponding heading.
Basic Assumption 2	For event tree headings related to offsite power recovery, remove the success branch accident sequences and delete the corresponding heading.
Rule 2	For event tree headings representing random failures unrelated to sabotage-induced damage, remove the failure branch accident sequences and delete the corresponding heading.
Rule 3	For event tree headings in which equipment located in sabotage-damaged compartments operates in a fail-safe mode, remove the failure branch accident sequences and delete the corresponding heading.
Rule 4	For event tree headings associated with events that occur probabilistically as a result of the effects of sabotage-damaged compartments, remove the failure branch accident sequences and delete the corresponding heading.

Additional Rule 3	Even if the basic assumptions, rules, and additional rules for vital area identification are not directly applicable to a given event tree heading, any headings indirectly affected by their application shall be appropriately modified to reflect their impact.

3.4 Target Set and Protection Set Analysis at the Event Tree Heading Level

For the initiating events selected for VAI analysis, the procedure for deriving target sets and protection sets at the PSA event tree heading level using the simplified PSA event trees is as follows:

・Generation of target sets at the event tree heading level

-Organize the core damage accident sequences from the event trees of the simplified initiating events into the event tree heading level to generate the corresponding target sets.

・Generation of protection sets at the event tree heading level

-Generate the protection sets by taking the Boolean complement of the target sets at the event tree heading level, or by organizing the success-branch headings from the "OK" accident sequences in the event trees for the simplified initiating events.

The target set/protection set identification method at the event tree heading level described above inherently incorporates the application criteria for the following key elements of vital area identification:

・Acceptance criterion for safe plant condition: Safe and stable conditions as defined in the PSA are applied.

・Duration for maintaining the safe plant condition: Mission times for each safety function as defined in the PSA are applied.

・Optimization measures for the VAI procedure: Applied to the generation of target sets and protection sets at the event tree heading level.

Any modification to the current application criteria for these key elements of VAI using PSA requires revision of the PSA model to ensure alignment with the updated criteria.

3.5 Protection Set Analysis at the Compartment Level

The procedure for analyzing protection sets at the compartment level is as follows:

·Connect fault trees to each heading of the event tree–level target sets and develop single-top-event fault trees.

·Replace each basic event of the single-top-event fault trees with a compartment-failure basic event to develop sabotage fault trees.

·Quantify the sabotage fault trees to generate compartment-level target sets, and perform a Boolean complement on these target sets to derive compartment-level protection sets.

The key element in generating compartment-level protection sets is the process of replacing each basic event in the single-top-event fault trees with a compartment-failure basic event. The vital area identification rules that must be applied in this process are as follows:

Application of Rule 1: Replace equipment-failure events with compartment-failure events using external event PSA information. Most equipment-failure basic events are replaced by compartment-failure events.

Application of Rule 2: This rule is applied during the simplification of event trees (e.g., excluding small-break LOCA caused by reactor coolant pump seal failure). For events that can be applied at the fault tree level, additional identification and application are required. For example, common-cause failure events do not occur due to sabotage but occur randomly, and thus they are treated as FALSE.

Application of Rule 3: This rule is applied during the simplification of event trees (e.g., excluding reactor shutdown failure events). If applied at the fault tree level, it must be further identified and applied. For instance, most instruments that generate signals to actuate engineered safety features are designed in a fail-safe manner; therefore, failures of such instruments must be treated as FALSE.

Application of Rule 4: This rule is reflected during the transformation step from equipment-failure events to compartment-failure events.

Application of Additional Rule 3: For operator action failure events carried out in the field, the failures are replaced with the compartments where the operator actions are performed and the compartments included along the operator's movement paths.

Once the above rules and additional rules for vital area identification are applied, the PSA single-fault trees are transformed into sabotage fault trees. From this point onward, the vital area identification process proceeds in the same manner as the conventional second-generation methodology.

3.6 Vital Area Selection and Additional Considerations

Once the compartment-level protection sets are generated, an expert review is conducted to select the most efficient protection set from among them from the perspective of physical protection, and this set is designated as the vital area. For the compartments included in the selected vital area, the required physical protection design is implemented.

Through physical protection design applied to the vital areas identified by the currently developed VAI methodology, direct core damage caused by sabotage can be prevented. However, core damage may still occur due to random failures of equipment located within vital areas or due to various probabilistic physical phenomena. In such cases, if the integrity of the reactor building is maintained, an additional defense-in-depth function can be provided by minimizing the release of radioactive materials and thereby reducing the consequences of offsite accidents. Therefore, it is necessary to incorporate this consideration into the vital area identification process.

In large PWRs, the integrity of the reactor building is ensured through the operation of the containment isolation system, which relies on the fail-safe functions of isolation devices (e.g., valves) located both inside and outside the containment. Thus, even under sabotage conditions where multiple systems are damaged and supporting systems are lost, containment isolation can still be achieved. In other words, containment isolation capability is maintained even when the current VAI methodology is applied to identify and designate vital areas.

However, in some nuclear power plants, containment isolation devices are installed that maintain their as-is status in the event of support system failure. For example, in Westinghouse-type plants, the containment isolation valve on the letdown line remains in its current position when the associated support systems fail. In such cases, additional consideration must be given to the compartments housing containment isolation devices located outside the reactor building, such as the HV-4 valve. These compartments should be designated as vital areas within the VAI methodology to ensure their protection. Moreover, in the event of probabilistic core damage, manual on-site closure of these valves may be required to achieve containment isolation. By incorporating such cases into the vital area identification process, the physical protection design can secure containment isolation capability even under sabotage or equipment failure conditions, thereby reinforcing the defense-in-depth function to minimize offsite accident consequences.

4. Conclusions

This study proposes an integrated VAI methodology applicable to both SMRs and large commercial nuclear power plants, incorporating the latest methodological advancements and reflecting applicability to SMRs. To this end, the existing advanced VAI method and additional considerations were reviewed, and an integrated methodology was developed to ensure that all basic assumptions, rules, and additional rules from previous methods are fully incorporated.

The application of the currently developed VAI methodology enables the prevention of direct core damage from sabotage. Nevertheless, core damage may still occur due to random equipment failures or probabilistic physical phenomena. In such cases, maintaining the integrity of the reactor building can provide an additional defense-in-depth function to minimize radioactive material release and mitigate offsite consequences. Therefore, this study proposes including areas necessary to preserve reactor building performance within the scope of vital areas. This measure is deemed essential for SMRs equipped with small steel containment vessels and containment isolation systems located outside the containment vessel.

This paper provides an integrated set of methodologies and various considerations necessary for conducting VAI for SMRs and large commercial nuclear power plants. Therefore, it is expected to serve as a comprehensive reference for future research on the development of VAI methodologies, as well as for analysts performing actual VAI assessments for SMRs and large commercial nuclear power plants.

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