

How to Minimize Radiological Consequences When Performing Vital Area Identification of Nuclear Facilities

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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).

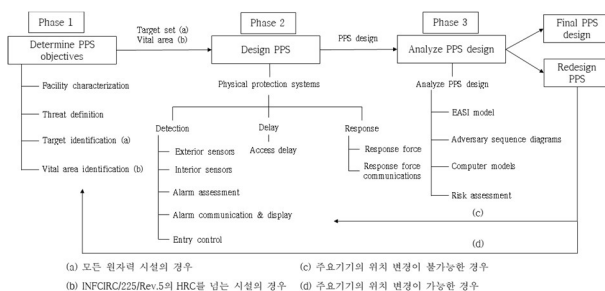


Figure 1. Process for designing and evaluating the PPS[2,3]

Nuclear facilities follow the physical protection system (PPS) design process illustrated in Figure 1. In Phase 1 of the PPS design process, 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 [4].

For nuclear facilities that could result in HRC-level consequences, Vital Area Identification (VAI) is conducted through the following steps: (1) development of a sabotage fault tree, (2) identification of the set of attack combinations, (3) calculation of the set of interdiction combinations, and (4) selection of the

interdiction set that can be protected at the lowest cost, which is then designated as the vital area.

In Korea, core damage has been defined as the threshold for high radiological consequences when identifying vital areas for nuclear power plants. Based on this criterion, a methodology has been developed and applied to commercial nuclear power plants to identify vital areas in practice.

1.2 Quantitative and qualitative approaches

In accordance with the recommendations of IAEA INFCIRC/225/Rev.5[1], some European countries require the implementation of a preliminary PSA at the design stage, as mandated by regulatory authorities. Unlike a completed PSA, a preliminary PSA is an early-stage analysis constructed on the basis of limited design information and is used as a basis for quantitatively assessing the radiological consequences resulting from the compromise of a VA. Such a quantitative approach provides more objective criteria for regulatory authorities in determining the appropriate level of VA protection; however, it is inherently subject to considerable uncertainties due to the limitations of design-stage analysis.

In contrast, the Republic of Korea and the United States refer to PSA in the process of VAI, but conducting a PSA at the design stage is challenging. This is because when the criterion for a VA is established as core damage, a fully developed PSA model is not yet available in the early design phase. Consequently, design-stage VAI is primarily performed through qualitative approaches, such as expert judgment, system function analysis, and adversary pathway analysis, and these results are reflected in regulatory reviews and design adequacy assessments. At the operational stage, PSA is supplemented to incorporate additional quantitative elements into the process.

1.3 Objective

This paper examines the appropriateness of applying core damage, which has been used as the existing criterion for high radiological consequences, in the context of radiological emergency response systems for accidents at domestic nuclear power plants. In addition, from a physical protection perspective, the report proposes improvements to enhance the safety of nuclear

power plant operations by incorporating the concept of defense-in-depth into the methodology for vital area identification.

2. Emergency response system for domestic NPPs

The emergency response system for radiological accidents at large commercial nuclear power plants in Korea is stipulated in the relevant Acts [5], Presidential Decrees [6], Prime Ministerial Decrees [7], and the site-specific radiological emergency plans of each facility. Among these, the criteria for each type of radiological emergency—constituting the most critical factors in determining the scope and specifics of emergency measures—are prescribed in Article 19 of the Enforcement Decree of the Act on Measures for the Protection of Nuclear Facilities, etc., and Prevention of Radiation Disasters (see Table 1).

Table 1. Criteria for each type of radiological emergency

White emergency	An emergency situation in which damage, or the potential for damage, occurs to the containment integrity of radioactive material or to the power supply functions necessary to maintain the safe condition of a nuclear facility. In such cases, the radiological impact from the release of radioactive material is expected to be confined within the buildings of the nuclear facility.
Blue emergency	An emergency situation in which degradation of recovery functions from a white emergency leads to damage, or the potential for damage, to the main safety functions of the nuclear facility. In such cases, the radiological impact from the release of radioactive material is expected to be confined within the site boundary of the nuclear facility.
Red emergency	An emergency situation in which damage, or the potential for damage, occurs to the final barrier of the nuclear facility, such as core damage or melting. In such cases, the radiological impact from the release of radioactive material is expected to extend beyond the site boundary of the nuclear facility.

The determination and declaration of radiological emergencies, which trigger the implementation of emergency measures specified in the Act on Measures for the Protection of Nuclear Facilities, etc., Prevention of Radiation Disasters, its Enforcement Decree, and its Enforcement Rule, are carried out in accordance with the site-specific radiological emergency plans for each nuclear site. These actions are based on accident condition assessments of each operating nuclear power plant within the site. The minimum plant condition requirements for declaring a white, blue, or red emergency at an actual facility can be summarized as described above.

Table 2. Minimum plant condition requirements for declaring a radiological emergency

White emergency	A plant condition involving a significant accident that does not exceed the design basis accident criteria.
Blue emergency	A plant condition involving a design basis accident or a situation in which there is a substantial possibility of core damage if specific systems (safety functions) are not restored.
Red emergency	A plant condition in which core damage is imminent following an accident.

According to the Act on Measures for the Protection of Nuclear Facilities, etc., and Prevention of Radiation Disasters, its Enforcement Decree, its Enforcement Rule, and the site-specific radiological emergency plans, the procedures and scope of actual protective actions for off-site residents in the event of a radiological emergency caused by an accident at a nuclear facility can be summarized as follows:

(1) Occurrence of a red radiological emergency at a nuclear power plant (based on the site-specific radiological emergency plan)

(2) Implementation of protective actions for residents corresponding to a red radiological emergency (based on the site-specific radiological emergency plan)

- The on-site emergency director recommends to the field command center the evacuation of residents within the PAZ in accordance with the radiological emergency plan
- The Site–Radiological Emergency Dose Assessment Program (S-REDAP) is used to evaluate projected public doses and determine the protective action areas

(3) Pursuant to Article 27 of the Act, the on-site emergency director exercises command authority over the head of the municipal/county/district Radiological Emergency Response Headquarters to decide urgent protective actions, including evacuation, relocation, restriction of food consumption, and distribution or administration of thyroid-blocking agents

(4) Under Article 29 (1), Items 3 to 5 of the Act, the head of the municipal/county/district Radiological Emergency Response Headquarters implements the protective actions decided by the on-site emergency director

As described above, the first actual protective action for off-site residents begins after the declaration of a red radiological emergency, through the recommendation to evacuate residents within the PAZ. Subsequently, based on the assessment of off-site impacts as the accident progresses, the content and timing of protective actions for residents within the UPZ are determined.

3. Assessment of the suitability of HRC criteria

3.1 Vital area identification

In various domestic and international methodologies developed for VAI, core damage has been applied as the

reference event or phenomenon that causes HRC. This chapter reviews the validity of using core damage as the criterion for HRC and proposes measures to provide additional defense-in-depth concepts from the perspective of physical protection.

In the context of nuclear power plant physical protection design, the identification of vital areas—an essential element of required design information—is categorized into first-, second-, and third-generation methods, depending on the approach used to develop sabotage fault trees.

- (1) The first-generation VAI method, developed in the United States [8], involves directly constructing a sabotage fault tree that uses compartment failure as its basic events.
- (2) The second-generation method, developed in Korea [9], reuses an integrated PSA model, substituting component failures with compartment failures to develop the sabotage fault tree.
- (3) The third-generation method, also developed in Korea [10], simplifies the VAI process by utilizing PSA event trees.

3.2 Consistency with domestic radiological emergency system

The first practical protective action for offsite residents begins with the recommendation for evacuation within the PAZ following the declaration of a General Emergency. Subsequently, based on the assessment of offsite impacts as the accident progresses, the content and timing of protective actions for residents in the UPZ are determined.

Within Korea's radiological emergency preparedness regime, the first substantive protective action for the offsite population is the evacuation of residents within the PAZ following a Red Emergency declaration based on plant conditions. Among the various Red Emergency declaration criteria specified in each station's Radiological Emergency Plan, the minimum practically applicable threshold is "imminent core damage." Consequently, from the standpoint of maintaining consistency in offsite protective actions, the term "high radiological consequence" closely aligns with the Red Emergency declaration condition—namely, a condition in which core damage is imminent. Therefore, defining "high radiological consequence" as core damage in existing domestic VAI methodologies, and applying that criterion to actual plants, is assessed to be consistent with the emergency preparedness system intended to implement offsite protective actions.

3.3 Strengthening Defense-in-Depth in Physical Protection

In existing vital area identification methodologies, the process involves three main steps: (1) identifying compartment combinations that can cause core damage (i.e., target sets), (2) deriving Boolean complements of

these target sets to obtain denial sets, and (3) selecting the most effective denial set as the vital area. Through such methodologies, physical protection designs can prevent direct core damage resulting from sabotage. However, probabilistic core damage may still occur due to random equipment failures or other physical phenomena. In such cases, maintaining reactor building integrity provides an additional defense-in-depth function by minimizing radioactive release and thus reducing offsite consequences.

In large LWRs, reactor building integrity is maintained by the containment isolation system, which relies on the fail-safe operation of isolation components (e.g., valves) located inside and outside the building. Even under sabotage conditions involving equipment damage and loss of support systems, containment building isolation can still be achieved. Hence, containment building isolation is assured in most plants even when current vital area identification methodologies are applied.

In some plants, however, certain isolation pathways depend on components such as motor-operated valves that require support systems. For these cases, designating the corresponding areas as additional vital areas within the PPS ensures the containment isolation function even if core damage occurs, thereby securing an enhanced defense-in-depth capability to minimize offsite consequences.

4. Conclusion

This study reviewed the radiological emergency preparedness system applied in the event of accidents at domestic nuclear power plants by examining relevant laws, enforcement decrees, regulations, and plant-specific radiological emergency plans. The analysis identified that the first substantial protective action for offsite residents begins with evacuation within the PAZ following the declaration of a General Emergency. Subsequently, the content and timing of protective actions for residents within the UPZ are determined based on the assessment of offsite consequences during accident progression.

In the domestic emergency preparedness system, the first practical protective action for offsite residents is initiated by evacuation within the PAZ following the declaration of a General Emergency, with the minimum applicable criterion being imminent core damage. Therefore, in terms of consistency, HRC is closely aligned with the condition of a General Emergency declaration, namely imminent core damage. Thus, the application of core damage as the reference criterion for HRC in existing VAI methodologies is consistent with the emergency preparedness framework for offsite protective actions.

While PPS designs based on current VAI methodologies can prevent direct core damage caused by sabotage, probabilistic core damage may still occur due to random equipment failures or various physical phenomena. In such cases, maintaining reactor building

integrity provides an additional defense-in-depth function by minimizing radioactive releases and reducing offsite consequences. In large LWRs, reactor building integrity is assured through the isolation system, which relies on the fail-safe function of isolation components (e.g., valves) located inside and outside the building. Even under sabotage conditions involving extensive equipment damage and loss of support systems, reactor building isolation can still succeed. Therefore, the isolation function is generally ensured in most nuclear power plants when current VAI methodologies are applied.

However, in some plants, certain reactor building isolation pathways include components such as motor-operated valves that require support systems. For such plants, if the areas containing non-fail-safe isolation components are additionally designated as vital areas and incorporated into the PPS design, reactor building isolation can be preserved even in the event of core damage. This would secure an enhanced defense-in-depth function to further minimize offsite accident consequences.

In the United States, pursuant to 10 CFR 73.55, the Main Control Room, Spent Fuel Storage Facility, Central Alarm Station, and Secondary Alarm Station are mandated to be designated as vital areas. Therefore, if the reactor isolation system is additionally designated as a vital area, it can achieve the same effect as designating vital areas by considering the HRC.

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