

Development of Quantitative Indicators for the Graded Regulation of Research Reactors with Various Criteria

Yuseung Lee ^a, Jeongkyu Cho ^a, and Ho Jin Park ^{a*}

a Kyung Hee University, 1732 Deokyoungdaero, Giheung-gu, Yongin-Si, Gyeonggi-do, Korea, 17104

**Corresponding author: parkhj@khu.ac.kr*

***Keywords : AGN-201K, Regulation, Research Reactor, Graded Approach, Quantitative Indicator, 10 CFR Part 57**

1. Introduction

It is reasonable that the stringency of the regulatory requirements imposed on a nuclear reactor be proportionate to the risk and complexity of the facility, a principle referred to internationally as the graded approach. In SSG-22 [1], the International Atomic Energy Agency (IAEA) recommends that the application of safety requirements to research reactors be graded according to facility risk while maintaining the same safety objectives. Relevant factors include reactor power, the potential source term, the amount and enrichment of fissile and fissionable material, the type of fuel elements, the types and masses of the moderator, reflector, and coolant, reactivity characteristics, and reactor utilization.

In the Republic of Korea, however, the Nuclear Safety Act [2] classifies reactors into only two categories — power reactors and research reactors — and provides no detailed criteria for power, risk, or hazard level. Consequently, HANARO at 30 MWth and the educational reactor AGN-201K at 10 Wth are managed under the same regulatory category despite a difference in thermal power of approximately six orders of magnitude [3]. The 2024 IAEA Integrated Regulatory Review Service (IRRS) mission to the Republic of Korea likewise issued a Recommendation that a graded regulatory scheme be established for research and educational reactors [4].

A practical difficulty is the lack of quantitative indicators suitable for low-power research reactors. Probabilistic safety assessment (PSA) metrics such as core damage frequency are well established for power reactors, but research reactors generally have sparse component-failure and accident-sequence data; for AGN-201K, only a limited risk-informed study of the reactor protection system has been reported [5]. Deterministic indicators that can be evaluated from design and accident analyses are therefore needed as an alternative screening basis.

In this regard, the proposed rule for 10 CFR Part 57, issued by the U.S. Nuclear Regulatory Commission in May 2026, provides important insights into the development of a graded regulatory framework for research reactors [6]. 10 CFR Part 57 quantifies the eligibility criteria for low-risk reactors as a total effective dose equivalent (TEDE) in the unrestricted area, over the

entire duration of the accident, of no more than 10 mSv based on the maximum hypothetical accident (MHA) or maximum credible accident (MCA) methodology, together with a limit of 10 metric tons of total thorium, uranium and plutonium inventory per reactor, thereby providing a structure that screens low-risk facilities using deterministic indicators alone, without a detailed probabilistic assessment. Accordingly, this study selects thermal power, the maximum accumulated effective dose equivalent to a member of the public for the selected postulated accident, and the amount and enrichment of fissile material and the design of the core cooling system as quantifiable indicators among the SSG-22 graded approach factors and derives a grading scheme from data compiled for fifteen domestic and international research reactors, test reactors, critical assemblies and educational reactors. Finally, we identify what such a framework would additionally require: the incorporation of factors that resist quantification, and expert panel review where the indicators do not converge.

2. Selection of Quantitative Indicators for Graded Regulation

2.1 Graded Approach Factors in IAEA SSG-22

In the IAEA safety standards, SSR-3 establishes safety requirements for research reactors, while SSG-22 provides guidance on the graded application of those requirements [1,7]. The requirements themselves are not waived; grading determines the extent of analysis, documentation and safety measures appropriate to the facility. SSG-22 uses a two-step process: first, the facility is categorized according to potential radiological hazard; second, each requirement and safety related structures, systems, and component (SSC) is graded according to its safety significance. The overall assessment method may be qualitative, quantitative, or a combination of both [1].

The first-step factors include reactor power, potential source term, fissile-material amount and enrichment, fuel type, moderator, reflector, coolant, reactivity-characteristics, confinement, utilization, site characteristics and proximity to population [1,7]. SSG-22 describes high-, medium- and low-hazard facilities qualitatively but does not prescribe universal numerical boundaries. A domestic grading framework therefore

requires numerical factors to be selected and their screening values to be justified separately.

2.2 Selection of Quantifiable Indicators for Graded Approach

In this study, four factors were selected for quantitative screening. *Thermal power* is a stable design parameter that broadly reflects the scale of radioactive inventory. *Radiological consequence* is represented by the maximum accumulated offsite individual TEDE for a selected postulated accident, which combines source term, release pathway, confinement and evaluation location into a consequence metric and is the quantity adopted for the same purpose in the proposed 10 CFR Part 57 [6]. *Fissile material* is represented by ^{235}U fuel loading and enrichment because inventory bounds the material available for release and enrichment is relevant to nuclear-material protection. The *core cooling system* is classified by design attributes — whether forced cooling is required during power operation, whether forced cooling is required following shutdown, and whether a heat sink is required — and these attributes can serve as an indirect indicator of the potential for fuel melting and hence of the potential radiological hazard. The remaining factors were excluded from the quantitative indicators in this study, as they either lack numerically rankable values or vary in significance depending on the facility's design concept.

2.3 Quantitative Screening Criteria of 10 CFR Part 57

If quantifiable indicators are to be used for grading, the boundary values that separate the grades must themselves be traceable to criteria that already carry regulatory meaning. The proposed rule for 10 CFR Part 57 [6], issued as a streamlined licensing framework for microreactors and other reactors of comparably low risk, is instructive in this respect because its eligibility criteria are stated in explicitly numerical terms rather than as descriptive characteristics.

The first criterion is dose based. The latter part of 10 CFR 50.34(a)(1)(i) [8] prescribes a limit of 1 rem TEDE for reactors not subject to Part 100, and Part 57 adopts this value as its screening threshold. An applicant is required to demonstrate, using either the maximum hypothetical accident (MHA) methodology or the maximum credible accident (MCA) methodology that the total effective dose equivalent in the unrestricted area over the entire duration of the accident does not exceed 1 rem (10 mSv). The significance of this construction is that a facility can be identified as low risk on the basis of a bounding deterministic calculation alone, without any probabilistic quantification of accident frequency.

The second criterion concerns fuel inventory. Part 57 limits the total inventory of thorium, uranium and plutonium to 10 metric tons of heavy metal per reactor. This value was obtained by examining the distribution of

special nuclear material inventory against thermal power for existing microreactor designs and confirming that such designs fall below 10 MTHM; the NRC concluded that designs within this inventory limit are likely to retain inherent passive safety functions and to exhibit a favorable safety profile despite variations in core design and thermal power.

This study adopts two key elements from the framework. The first is utilizing the maximum cumulative dose in the unrestricted area derived from global criteria as a directly comparable metric for potential source terms. The second is the rationale of establishing threshold values for fissile material inventory based on case studies of multiple reactors.

3. Application of the New Quantitative Indicators for the Graded Regulation of Research Reactors

The four indicators selected in Section 2.2 were compiled for fifteen research reactors, test reactors, critical assemblies and zero-power reactors, comprising both domestic and international facilities and spanning a thermal power range from 1 W to 50 MW. The data were collected from safety analysis report (SAR), licensing documents and open published literature for each facility.

3.1 Thermal Power

Table I summarizes the thermal power, the maximum accumulated TEDE to an individual offsite for the selected postulated accident. Thermal power spans more than seven orders of magnitude, illustrating the wide range currently contained within the research-reactor category.

Table I: Thermal power, maximum offsite individual TEDE for the selected postulated accident

Reactor	Country	P_{th}	Max. offsite TEDE (mSv)
HANARO	Rep. of Korea	30 MW	52.72
AGN-201K		10 W	† < 0.0032
KJRR		15 MW	44.74
FRM-II	Germany	20 MW	6.5
JRTR	Jordan	5 MW	50
NBSR	United States	20 MW	0.067
MURR (LEU)		12 MW	0.0858
MNRC TRIGA		2 MW	0.0056
OSTR		1.1 MW	0.19
Dow TRIGA		0.3 MW	0.067
GHARR-1		0.03 MW	0.0262
KUCA	Japan	100 W	0.00019
UTR-KINKI		1 W	0.023
JMTR		50 MW	2.78
HTTR		30 MW	1.7

* The accident postulated, the evaluation point and the exposure period differ among facilities; the values are therefore not the product of a harmonized calculation and are used for comparison of order of magnitude only. Some values have been converted from the figures reported in the original sources.

† No offsite evaluation exists for AGN-201K. The value was obtained by applying the attenuation of the shielding configuration to the more

conservative of the two available MHA analyses evaluated at the point adjacent to the reactor.

3.2 Radiological Consequence

The factor in SSG-22 corresponding to radiological consequence is the potential source term, but a source term is expressed as nuclide-wise inventory and release fractions, which makes direct comparison between facilities of differing core configuration difficult, and it is not published for most of them. This study therefore adopts the quantity in which the source term is ultimately manifested: the maximum TEDE received by an individual offsite for the selected postulated accident. This value integrates the source term, the release pathway, the performance of the confinement and the evaluation distance into a single scalar, and since what regulation is ultimately concerned with is not the inventory within the core but its consequence once it reaches the public, it represents most directly whether the objective of safety regulation is achieved. Dose is moreover a quantity for which regulatory thresholds already exist — the postulated accident criterion, the public dose limit and the clearance level — and it is the quantity adopted as the eligibility criterion for low-risk reactors in 10 CFR Part 57.

The values in Table I are not the product of a single harmonized calculation. The postulated accident ranges from a severe accident to a design basis accident, the evaluation point from 5 km to the exclusion area boundary, and the exposure period from two hours to the entire duration of the accident, with some values converted from the original sources. What nonetheless permits their use for grading is the magnitude of the variation. The compiled values span approximately five and a half orders of magnitude, from 1.9×10^{-4} mSv to 52.72 mSv, whereas the deviation introduced by the methodological differences above does not ordinarily exceed one or two orders of magnitude. Given grade boundaries spaced by decades at 10, 1 and 0.01 mSv, the cases in which such variation would alter a grade are limited. Furthermore, the assumption of a severe accident, evaluation at the exclusion area boundary and the MHA methodology are all conservative choices, so the compiled values are of the character of upper bounds, and errors in the direction of assigning too low a grade are unlikely to arise. The value for AGN-201K was derived in the same conservative spirit: no offsite evaluation exists, and the two available MHA analyses at the point adjacent to the reactor, giving 10 mSv (University of New Mexico) and 32 mSv (Idaho State University), were reduced by the attenuation of the shielding configuration, with the more conservative of the two adopted.

However, to establish a consistent legal basis when evaluating values, it is crucial that the calculated spatial positions and time steps are explicitly defined. Therefore, the values in this section have been used only as data for relative ordering at the level of orders of magnitude, not as criteria directly applicable in licensing.

3.3 Amount and Enrichment of Fissile Material

Table II presents the ^{235}U loading, the enrichment of each facility. The loadings range from 690 g for AGN-201K to several tens of kilograms, while the enrichments divide the facilities into a high enriched uranium group (HEU) and a low enriched uranium group (LEU). Recently, GHARR-1 has been converted from HEU to LEU fuel, while conversion of MURR, KUCA and UTR-KINKI is currently under way.

Table II: Amount and enrichment of fissile material of the surveyed reactors

Reactor	U-235	Enrichment (wt%)
HANARO	11.64kg	19.75 (LEU)
AGN-201K	690g	19.5 (LEU)
KJRR	13.84kg	19.75 (LEU)
JMTR	9.63kg	19.75 (LEU)
HTTR	53.1kg	6.0 (LEU)
NBSR	10.5kg	93.0 (HEU)
FRM-II	7.53kg	93.0 (HEU)
MURR (LEU)	12.05kg	19.75 (LEU) *
JRTR	7kg	19.75 (LEU)
MNRC TRIGA	2.8~15.1kg	20.0 (LEU)
OSTR	13.6kg	19.75 (LEU)
Dow TRIGA	2.8kg	20.0 (LEU)
GHARR-1	1.355kg	13.0 (LEU)
KUCA	1~10kg	19.75 (LEU) *
UTR-KINKI	3.07kg	19.75 (LEU) *

* MURR, KUCA and UTR-KINKI are currently being converted to LEU.

Table III: Core Cooling System Design

Reactor	Forced cooling during operation	Forced cooling following shutdown	Heat sink
HANARO	O	O	O
AGN-201K	X	X	X
KJRR	O	X	O
JMTR	O	O	O
NBSR	O	O	O
FRM-II	O	O	O
MURR (LEU)	O	X	O
JRTR	O	O	O
MNRC TRIGA	X	X	O
OSTR	X	X	O
Dow TRIGA	X	X	O
GHARR-1	X	X	O
KUCA	X	X	X
UTR-KINKI	X	X	X

* HTTR could not be classified because it is a helium cooled.

3.4 Core Cooling System Design

The core cooling system is classified by design attributes: whether forced cooling is required during power operation, whether forced cooling is required following shutdown, and whether a heat sink is required. Forced cooling during power operation is judged at the core. Forced cooling following shutdown is judged by whether an active component must continue to operate in

order to remove decay heat immediately after shutdown. The heat sink refers to a pool holding a sufficient inventory of water, and serves as the criterion separating facilities that have a coolant from those that do not. Table III compared the core cooling system designs among various research reactors.

4. Proposed Grading Scheme for the Graded Regulation of Research Reactors

Each indicator was divided into four grades, from Grade 1 for the highest risk to Grade 4 for the lowest. The boundaries were not chosen for convenience of division; each was drawn from a value that already functions as a threshold in an existing domestic or international regulation.

4.1 Grading by Thermal Power

Three boundaries were adopted: 10 MW, 1 MW and 100 W. The 10-MW boundary is consistent with the Japanese research-reactor classification reported in the preceding study [3]. SSG-22 itself does not specify a 10 MW or 1 MW boundary; it describes medium-hazard facilities only as having operating power up to a few MW and leaves numerical categorization to the regulator [1]. The 1 MW boundary is therefore a study-defined screening value used to separate MW-class reactors from sub-MW facilities.

The 100 W boundary is retained because it has been used in the preceding domestic graded-regulation study [3] and appears as the lowest power-based exemption band in NRC emergency-planning guidance for research and test reactors [9]. Its use here is therefore a pragmatic low-power screening boundary, not a threshold prescribed by SSG-22. The resulting grades are given in Table IV.

Table IV: Research reactor categories classified by thermal power

Grade	Thermal power	Reactors
1	$P_{th} \geq 10 \text{ MW}$	HANARO, JMTR, HTTR, NBSR, FRM-II, KJRR, MURR
2	$1 \text{ MW} \leq P_{th} < 10 \text{ MW}$	JRTR, MNRC TRIGA, OSTR
3	$100 \text{ W} < P_{th} < 1 \text{ MW}$	Dow TRIGA, GHARR-1, MNSR
4	$P_{th} \leq 100 \text{ W}$	KUCA, AGN-201K, UTR-KINKI

4.2 Grading by Radiological Consequence

The boundaries were established by applying the method used in Proposed Part 57 to derive its fuel mass limit, reviewed in Section 2.3. The NRC did not prescribe the value of 10 MTHM normatively in advance;

it examined the distribution of special nuclear material inventory against thermal power across existing microreactor designs, confirmed that those designs fall below that value, and then adopted it as the limit. The procedure is thus one of observing the distribution of actual facilities and drawing the line where that distribution supports it. This study follows the same procedure, while first considering existing regulatory thresholds.

Accordingly, 10 mSv, 1 mSv, and 0.01 mSv were selected because they already correspond to established regulatory criteria for non-power reactor accident consequences, public dose limits, and exemption/clearance, respectively [8,10,11]. If these values are consistent with the observed reactor distribution, there is little justification for introducing separate new boundary values.

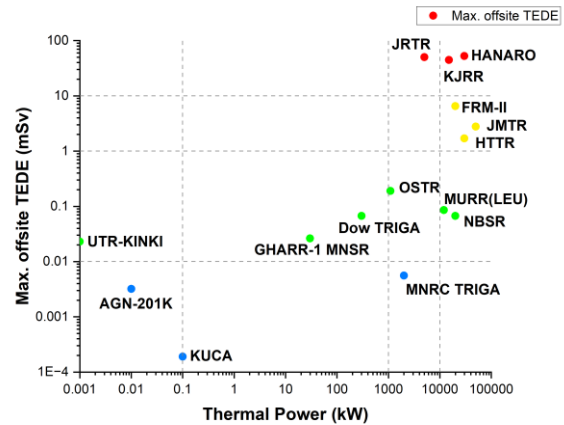


Fig. 1. Maximum offsite individual TEDE versus thermal power for the surveyed reactors. Dashed lines indicate the proposed grade boundaries of Sections 4.1 and 4.2.

Fig. 1 places the fifteen facilities on axes of thermal power and offsite individual TEDE, with the thermal power boundaries of Section 4.1 and the three dose criteria above shown as dashed lines. Both axes are logarithmic. The three criteria are found to pass through the gaps between clusters of facilities rather than sectioning the data arbitrarily. A general tendency for the offsite individual dose to increase with thermal power is also apparent across the scatter. This is to be expected, since the radioactive inventory of the core is broadly proportional to thermal power, and it indicates that the two indicators are not unrelated quantities. The tendency does not hold consistently at the level of individual facilities, however. This is the basis for retaining the two indicators as separate grading axes, and the divergences between grades that arise from it are addressed in Section 4.5. This study adopts the three criteria as they stand rather than deriving new boundary values. The resulting grades are given in Table V.

Table V: Research reactor categories classified by radiological consequence

Grade	Max. offsite TEDE	Reactors
1	≥ 10 mSv	JRTR, HANARO, KJRR
2	> 1 mSv	HTTR, FRM-II, JMTR
3	> 0.01 mSv	OSTR, NBSR, Dow TRIGA, UTR-KINKI, MURR(LEU), GHARR-1
4	≤ 0.01 mSv	AGN-201K, MNRC TRIGA, KUCA

4.3 Grading by Amount and Enrichment of Fissile Material

For this factor SSG-22 gives three categories by way of example: the use of HEU or the storage of a large quantity of fuel for the high category, the limited storage of LEU for the medium category, and the storage of a small quantity of LEU for the low category. This study adopts that division but subdivides the medium band of limited LEU storage into two, giving four grades, because the facilities using LEU under limited storage form the largest group among the fifteen examined and treating the band as a single category would leave the indicator without discriminating power.

The specific assignments were made with reference to the categorization of nuclear material in IAEA INFCIRC/225 [12] and 10 CFR 73 [13]. Facilities using LEU were divided at 10 kg and 5 kg of U-235. Facilities using LEU were divided at 10 kg, the lower bound of Category II in INFCIRC/225 [12] for uranium enriched to 10–20 %, and at 5 kg, a study-defined subdivision introduced to retain discriminating power within the remaining band. The result is given in Table VI.

Table VI: Research reactor categories classified by amount and enrichment of fissile material

Grade	Amount and Enrichment	Reactors
1	HEU	NBSR, FRM-II
2	LEU (≥ 10 kg)	OSTR, MNRC TRIGA, MURR(LEU), HTTR, HANARO, KJRR
3	LEU (5~10kg)	JRTR, KUCA, JMTR
4	LEU (≤ 5 kg)	AGN-201K, Dow TRIGA, GHARR-1, UTR-KINKI

4.4 Grading by Core Cooling System Design

The four grades follow directly from the classification of preceding domestic study [14]. Grade 1 is assigned where forced cooling is required both during power operation and after shutdown, Grade 2 where forced cooling is required during power operation but passive means suffice after shutdown, Grade 3 where the core is cooled by natural convection at power but a heat sink is

required, and Grade 4 where neither cooling nor a heat sink is required. The result is given in Table VII.

Table VII: Research reactor categories classified by core cooling system

Grade	Core Cooling system (forced cooling during operation/ after shutdown/ heatsink)	Reactors
1	O/O/O	JMTR, HANARO, NBSR, FRM-II
2	O/X/O	KJRR, MURR(LEU), JRTR
3	X/X/O	MNRC TRIGA, OSTR, Dow TRIGA, GHARR-1
4	X/X/X	KUCA, UTR-KINKI, AGN-201K

4.5 Integrated Grading Results

Table VIII shows the grades assigned by the four major risk-assessment indicators – thermal power, radiological consequence, amount and enrichment of fissile material, and core cooling system.

Table VIII: Integrated grades for various research reactors based on four key risk-assessment indicators

Reactor	Thermal power	Radiological consequence	Fissile material	Core cooling system
HANARO	1	1	2	1
AGN-201K	4	4	4	4
KJRR	1	1	2	2
FRM-II	1	2	1	1
HTTR	1	2	2	N/A
JMTR	1	2	3	1
NBSR	1	3	1	1
MURR (LEU)	1	3	2	2
JRTR	2	1	3	2
OSTR	2	3	2	3
MNRC TRIGA	2	4	2	3
Dow TRIGA	3	3	4	3
GHARR-1	3	3	4	3
KUCA	4	4	3	4
UTR-KINKI	4	3	4	4

The integrated results show that facilities for which the four indicators diverge outnumber those for which they coincide. NBSR and MURR receive Grade 1 for thermal power but Grade 3 for radiological consequence, which reflects the confinement provided and the accident postulated rather than any reduction in inventory.

Two observations follow. First, no single indicator can stand in for the others. Grading on thermal power alone, the form in which graded regulation is most often proposed domestically, reflects neither accident consequence nor nuclear material protection and therefore falls short of a comprehensive assessment of risk. Second, the number of facilities whose four grades coincide is small, so the assignment of a single overall grade to the remainder cannot be made by the indicators alone. What is required is a determination that considers together the grades assigned by the quantitative indicators and a qualitative expert evaluation of the

factors identified in SSG-22 that do not admit of quantification.

AGN-201K is the only facility placed at Grade 4 by all four indicators. This provides a quantitative basis for the position, advanced in the preceding study on grounds of thermal power alone [3], that zero power educational reactors constitute a distinct regulatory category.

5. Conclusions

This study aimed to develop quantitative indicators for the graded regulation of research reactors. Among the eleven factors enumerated in SSR-3 and detailed in SSG-22, four indicators could be numerically expressed without subjective judgment: thermal power, maximum offsite individual TEDE during selected bounding accidents, fissile material inventory and enrichment, and core cooling system. Adopting the approach of the newly-proposed 10 CFR Part 57, these indicators for 15 domestic and international facilities were categorized into four grades. For the three research reactors in the Republic of Korea, the grades assigned according to the four graded-approach regulation indicators proposed in this study are as follows.

Table IX: Integrated grades for HANARO, KJRR, and AGN-201K reactors in Rep. of Korea

Reactors	Grades	Overall Assessment*
HANARO	1~2	Very high risk research reactor
KJRR	1~2	Very high risk research reactor
AGN-201K	4	Very low risk research reactor

* In this study, research reactors are classified into four grades: very high risk, high risk, low risk, and very low risk research reactors.

Because fewer facilities were assigned consistent grade across all four indicators, it becomes evident that the single-criterion approach based solely on thermal power—the most commonly proposed form of graded regulation in Republic of Korea—fails to achieve a comprehensive risk assessment.

Therefore, the quantitative indicators are best understood as establishing a range for the overall grade, within which “*expert qualitative evaluation*” of the remaining SSG-22 factors would determine the final grade. This maintains the stance of SSG-22 that graded approaches must be grounded in safety analysis and regulatory requirements while being supported by expert judgment, thereby reducing the variability that arises when categorization relies solely on subjective judgment.

Acknowledgement

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT) (No. RS-

2026-25544583).

REFERENCES

- [1] International Atomic Energy Agency, Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors, IAEA Safety Standards Series No. SSG-22 (Rev. 1), IAEA, Vienna, 2023.
- [2] The National Assembly of the Republic of Korea, Nuclear Safety Act and its Enforcement Decree and Enforcement Regulation, 2024.
- [3] J. Hur, M. J. Kim, and H. J. Park, A Study on the Safety Regulatory Framework Analysis and Graded Regulation for Zero Power Research Reactors in South Korea, Transactions of the Korean Nuclear Society Autumn Meeting, Changwon, Korea, October 30-31, 2025.
- [4] International Atomic Energy Agency, Integrated Regulatory Review Service (IRRS) Mission to the Republic of Korea, IAEA, Vienna, 2024.
- [5] I. Ahmed et al., Risk-informed approach to the safety improvement of the reactor protection system of the AGN-201K research reactor, Nuclear Engineering and Technology, Vol.52, No.4, p.764, 2020.
- [6] U.S. Nuclear Regulatory Commission, Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles, Proposed Rule, 91 FR 23628, May 1, 2026.
- [7] International Atomic Energy Agency, Safety of Research Reactors, IAEA Safety Standards Series No. SSR-3, IAEA, Vienna, 2016.
- [8] U.S. Nuclear Regulatory Commission, Domestic Licensing of Production and Utilization Facilities, 10 CFR Part 50.
- [9] U.S. Nuclear Regulatory Commission, NUREG-0849, Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors, 1983.
- [10] U.S. Nuclear Regulatory Commission, Standards for Protection Against Radiation, 10 CFR Part 20.
- [11] Nuclear Safety and Security Commission, Regulations on Self-Disposal of Radioactive Waste, NSSC Notice.
- [12] International Atomic Energy Agency, Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities, INFCIRC/225/Revision 5, IAEA, Vienna, 2011.
- [13] U.S. Nuclear Regulatory Commission, Physical Protection of Plants and Materials, 10 CFR Part 73.
- [14] H. J. Park, J. Hur, and M. J. Kim, “Analysis of the Safety Regulatory for Educational Zero Power Reactors and Proposal of Graded Regulatory Improvements,” MCRPL-TR-005(2025) R0, Kyung Hee University, 2024.