ABSTRACT

LEE, DAEHO. A Probabilistic Risk Assessment Model for the Transportation of Fission Batteries with Preliminary Licensing Considerations for Civilian Applications and Deployment Considerations for Military Applications (Under the direction of Dr. Mihai A. Diaconeasa).

Nuclear energy is currently in the spotlight as a future energy source all over the world amid the global warming crisis. Moreover, the United States (U.S.) Military has shown their willingness to use of microreactors through the Pele project. In line with this trend, Republic of Korea (ROK) Military is preparing to switch the energy paradigm to new energy sources.

A fission battery (FB) is inspired by the idea that nuclear energy can be used by ordinary people using the "plug-and-play" concept like chemical batteries. As design requirements, FBs must be economic, standardized, installed, unattended, and reliable.

Meanwhile, the commercialization of reactors is regulated by national bodies, such as the U.S. Nuclear Regulatory Commission (NRC). At an international level, the International Atomic Energy Agency (IAEA) oversees the safe and peaceful use of nuclear power. However, regulations currently face a significant gap in terms of their applicability to advanced non-light water reactors (non-LWRs). Therefore, this study investigates the regulatory gaps for the licensing of FBs concerning safety in terms of siting, autonomous operation, and transportation, and suggests response strategies to supplement it.

To figure out the applicability of the current licensing framework to FBs, we reviewed the U.S. NRC Title 10, Code of Federal Regulations (CFR) and IAEA INSAG-12. To addressing siting issues, we explored the non-power reactor (NPR) approach for site restrictions and the permit-by-rule (PBR) approach for excessive time burdens. In addition, we discussed how the development of an advanced human-system interface (HSI) augmented with artificial intelligence and monitored by personnel for FBs may enable successful exemptions from the current regulatory requirements.

Most importantly, we discovered that no transportation regulatory challenge exists. Accordingly, a study was conducted using the general theory of probabilistic risk assessment (PRA) to numerically evaluate the risk of FB transportation. In this process, the NUREG-2125 "Spent Fuel Transportation Risk Assessment" issued by the U.S. NRC was used as a reference. We used the spent nuclear fuel (SNF) assumptions from NUREG-2125 as input parameters to the NRC-RADTRAN code for realistic calculations of the dose consequences for FBs. Also, fission products inventories of FBs were determined using ORIGEN code assuming 3 months, 6 months, and 1 year of operation with 100% power operation at 10 MWth and 20 MWth. We assumed that the radionuclides and release fractions of FBs during accidents are the same as the current SNF of LWRs. We developed scenarios involving the transportation by rail of one FB right after shutdown through urban areas from Maine Yankee to the Oak Ridge National Laboratory (ORNL) site. For scenarios with the rail accident frequency of one in 787,401 km (1.27E-06 per km), the collective dose risk was calculated to be between 5.28E-08 person-rem and 1.61E-07 person-rem. Also, to evaluate the FB transportation risk independent of the route, we obtained the individual dose risk excluding route information between 1.30E-13 rem and 3.96E-13 rem. To get a better understanding of the consequences associate with the FB transportation compared to SNF of LWRs, we compared their conditional individual dose risk excluding accident rate, and results were between 1.02E-07 rem and 3.12E-07 rem are comparable to the result obtained in NUREG-2125 of 1.29E-07 rem, we demonstrated that FB transportation would be as safe as the current SNF transportation.

Finally, we discussed on the deployment of FBs for the ROK Military. The requirements of the ROK Military and the characteristics of FBs were comprehensively considered. We conclude that FBs are expected to be a potential energy sources with the characteristics, such as site independence, transportable size, and economic competitiveness compared to the current technologies.

A Probabilistic Risk Assessment Model for the Transportation of Fission Batteries with Preliminary Licensing Considerations for Civilian Applications and Deployment Considerations for Military Applications

by
DaeHo Lee

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APPROVED BY:

Dr. Mihai A. Diaconeasa Committee Chair	Dr. Nam T. Dinh

Dr. Maria N. Avramova

DEDICATION

To my family, SeEun Park, SeoHyun, and SeoYoon. I would not be here without you.

BIOGRAPHY

DaeHo Lee was born in Seoul, Republic of Korea. He attended the Korea Military Academy from 2009-2013 where majored in Applied Physics. After graduated in 2013, he was commissioned a Second Lieutenant of Infantry officer and started his military career in the DMZ Reconnaissance Battalion. After being promoted to captain, he served as a Republic of Korea Army staff officer in the U.S. 8th Army and was awarded U.S. Army Commendation Medal in recognition of his contribution to enhancing the ROK-US alliance. Afterwards, while performing the duty of a company commander at the Capital Defense Command, he was selected for a master's degree program supported by the ROK Army and admitted to Probabilistic Risk Assessment Group under Dr. Mihai A. Diaconeasa at North Carolina State University's Nuclear Engineering Department. Following his graduation from NCSU, he would be assigned to a department related to nuclear energy in ROK Army. He is married to SeEun Park and has two daughters, SeoHyun and SeoYoon.

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I would like to express my sincere thanks to Dr. Mihai A. Diaconeasa for his guidance and consideration. He has guided me through two years of successful completion of master's program. It was a great fortune to be his disciple. I am also grateful to my colleagues at PRAG. Thanks to their help, I was able to adjust to NCSU and study well. I have no doubts that they will surely succeed in their field. I would also like to express my gratitude to the ROK Army for giving me the opportunity to study. Finally, thanks to the trust and support of my family, most important for me, I was able to successfully complete the course.

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CHAPTER 1. Introduction

1.1 Background and Motivation

Entering the era of global warming, nuclear energy is one of the eco-friendly and low-carbon energy sources for mankind who are struggling with severe climate change and the resulting natural disasters. Recently, nuclear energy has been in the spotlight as an eco-friendly energy by being included in the Green Taxonomy. Historically, the power of nuclear energy was recognized and started to use it in the 1940s, and through continuous research and development, it has become a major energy source accounting for 10% of global electricity production and 20% of U.S. electricity production[1].

However, the misfortune of the development of nuclear weapons using nuclear energy and the historical tragedy of nuclear accidents such as Chernobyl and Fukushima, have taken away trust in nuclear energy and instilled fear. As a result, some countries such as Germany and Belgium decided the closure of nuclear power plants (NPPs), while the United Kingdom, France, South Korea, and Japan declared a gradual reduction. Even the United States is forcing early retirement of NPPs and weakening its national supply chain[2]. Nevertheless, the International Energy Agency (IEA) predicts that the global electricity production using nuclear energy will be more than triple its current rate from 210TWh (2000~2019) to 650TWh (2019~2040)[3].

In this trend, nuclear experts are conducting research on miniaturization of NPPs to reduce the economic burden and enormous accident damage of the current large-scale NPPs. Accordingly, advanced reactors such as Small Modular Reactors (SMRs) and microreactors are under development, where SMRs are expected to be commercialized in 2029[1]. Going one step further, Idaho National Laboratory (INL) took the idea from battery and established the fission battery (FB) Initiative to make nuclear energy accessible by the public, just like chemical batteries. In this trend, it is noteworthy that the U.S. Military has shown its interest in the use of nuclear reactors through the Pele project, and Republic of Korea (ROK) Military is preparing to switch the energy paradigm to new energy sources.

Meanwhile, the commercialization of reactors is regulated by the U.S. NRC, and safe and peaceful use of nuclear energy in terms of safety, security, and safeguards is supervised by IAEA. However, since advanced reactors like FBs have presented unprecedented innovative ideas, current regulations focusing on current LWRs face a significant regulatory gap of applicability to them and impose licensing challenges.

1.2 Fission Batteries

The battery has been used as a portable and reliable energy source supplying power to various fields since its invention in the 1800s. Nuclear energy industries, one of the main global energy sources, focused on the characteristics of batteries to meet the current need for remote, expansive, and self-contained power sources, away from the existing massive plant system. Consequently, the notation of a “fission battery” was created with an innovative vision of “plug-and-play”[4]. The FB aims to break away from the current concept of large scale NPPs and make it to be used by any user in any location, similar to chemical batteries.

1.2.1 Attributes of FBs

The following five attributes support the vision of FBs and suggest the direction of research and development.

(1) Economic: FBs will have cost competitiveness, compared to energy sources that operate only on a specific platform, through a wide range of use and multiple deployment.

(2) Standardized: FBs will be developed in standardized sizes, power outputs, and manufacturing processes for extensive use, and will be fully assembled in the factory to ensure low-cost and quality assurance.

(3) Installed: FBs will be ready for deployment to implement “plug-and-play”.

(4) Unattended: FBs will be operated without the need for on-site operators based on a resilient and autonomous system.

(5) Reliable: FBs will have high reliability during its lifetime based on a robust, resilient, fault tolerant, and durable system to achieve fail-safe operation.

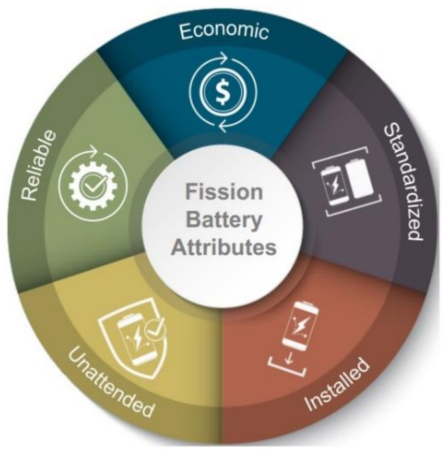

Figure 1. Fission Battery Attributes[4]

1.2.2 Research and Development (R&D) Plan

Figure 2 shows the Research and Development (R&D) plan necessary to achieve the FB attributes mentioned above. FBs are the most advanced forms of making nuclear energy available to anyone, going beyond advanced reactors, including microreactors, which are currently under development. Therefore, technologies including digital instrumentation and control (I&C), autonomous controls, remote monitoring, transportation and siting, manufacturing, safeguards and security, reliability analysis, data science including machine learning (ML)/artificial intelligence (AI), modeling and simulation, data and communication architecture, risk assessment tools, capabilities for proof-of-concept demonstration, and verification & validation (V&V) technologies would be studied and applied to the development of the advanced nuclear reactors.

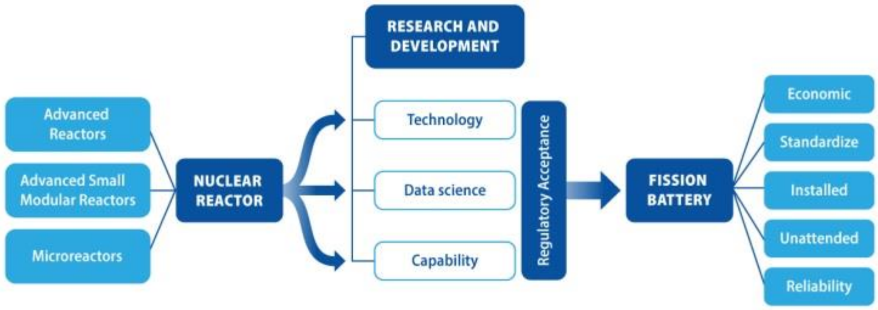

Figure 2. Fission Battery R&D Plan[4]

1.2.3 Reactor Design

The reactor design of FBs is expected to follow the microreactor design features based on its attributes. Mainly, the design will either be gas-cooled reactors with Tri-structural isotropic (TRISO) fuel or heat-pipe reactors with metal, oxide, or TRISO fuels, with target markets such as isolated grid, military bases, and electricity supply to electric-vehicles[5]. Especially, FBs will be designed to be used for less than 1 year with an output of less than 25MWth and cheaper than $0.1B to meet midsize customer energy demands[6]. An example of the design and size of a microreactor that is currently being researched and developed by various institutions such as Westinghouse and X-energy is shown in Figure 3 below.

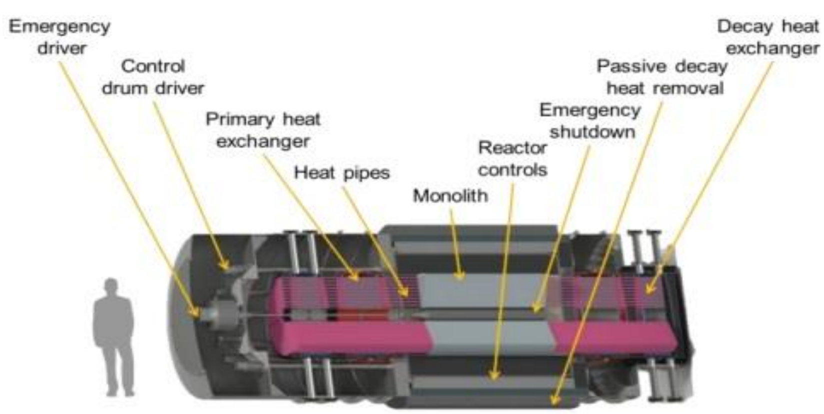

Figure 3. eVinci Microreactor cutaway from Westinghouse[7]

(1) High Temperature Gas-cooled Reactor (HTGR)

HTGR uses TRISO fuel, ceramic core structure, graphite moderator, and helium coolant. HTGRs have safety advantages such as high heat capacity and structural stability even at high temperatures due to use of those materials compared to conventional water-cooled reactors. It is expected for HTGRs to have favorable inherent safety features since it has the design concepts of passive and inherent safety, small size, and modularity.

(2) Heat Pipe reactor (HP reactor)

HP reactor uses various types of fluids such as helium and water that flow through installed heat pipes to remove heat from the core. It is expected to have favorable inherent safety features due to having passive heat removal systems and compact design with fewer components, however, it lacks long term operation history which is key for reliability analysis[8].

1.2.4 Fuel type

Fuel with high-assay low-enriched uranium (HALEU), enriched from 5% to 20% is most typically considered as the fuel type for advanced reactors. TRISO fuel is one of the representative fuels using HALEU with the uranium form of UCO or UO_2 and encapsulated with three layers of carbon and ceramic based materials that prevent the release of radioactive fission products and withstand very high temperatures without melting. So, it is considered as the most robust nuclear fuel and expected to be used for the FBs. Figure 4 shows the cutaway of a TRISO fuel particle; it can be seen that three layered shell surrounds the fuel kernel composed of UCO or UO_2, ensuring safety.

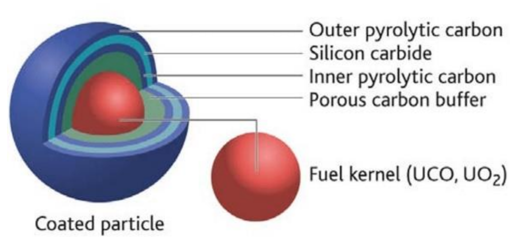

Figure 4. Cutaway of TRISO fuel[9]

1.3 Scope

Chapter 2 describes the process of two methodologies used in this study: a literature review of figuring out the regulatory gap against the current licensing framework for NPPs and suggesting appropriate response strategies, and probabilistic risk assessment (PRA) to numerically evaluate the risk of FB transportation under current regulations.

For civilian application, in Chapter 3, a study on the current regulatory licensing framework stipulated by the U.S. NRC in terms of safety and its applicability to FBs was conducted. Also, appropriate response strategies were suggested to supplement or be exempted from current regulatory requirements.

In addition, in Chapter 4, PRA was utilized to evaluate incident free transportation, loss of shielding accidents, and release of radioactive materials, and the results were respectively compared to radiation dose of chest X-ray, regulatory dose limit in severe accidents, and dose risk of spent fuel transportation.

In the military application, meanwhile, U.S. Military has launched the development of microreactors through Pele project, and ROK Military is in need of new energy sources for future warfare. In Chapter 5, therefore, a discussion was presented on the advantages of deploying FBs to ROK Military preparing for the transformation to new energy sources.

CHAPTER 2. Methodology

Chapter 2 describes the methodology used in this study and provides a logical understanding of the processes during this study. This study used two methodologies: a literature review of the current nuclear regulatory frameworks and a limited-scope PRA. The literature review is used for reviewing and applying current regulatory licensing framework to FBs, as well as suggesting response strategies to discovered challenges. The PRA methodology is used to numerically evaluate the risk of FB transportation.

2.1 Literature Review of Nuclear Regulatory Frameworks

The literature review used in this research proceeds as follows.

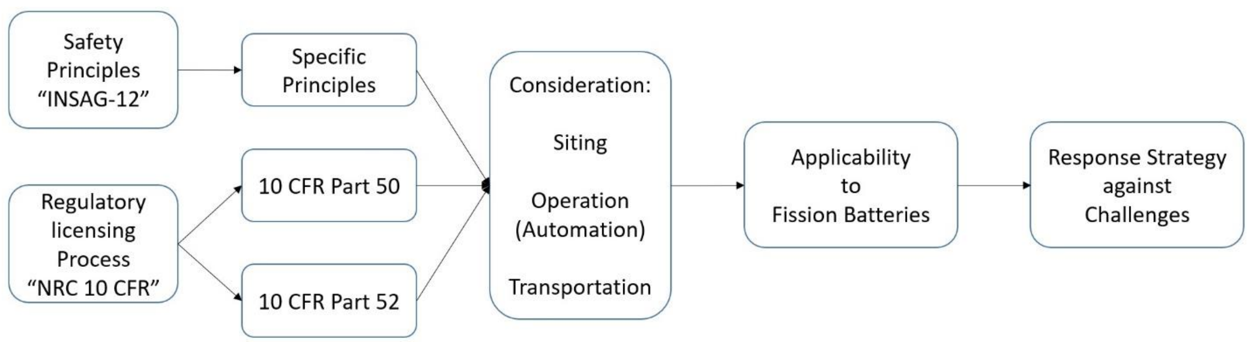

Figure 5. Literature Review process in this study

As shown in Figure 5, INSAG-12 "Basic Safety Principles for Nuclear Power Plants"[10] and U.S. NRC Title 10, Code of Federal Regulations (CFR)[11] were reviewed to introduce the specific safety principles essential for the licensing of NPPs and provide regulatory information about licensing process of 10 CFR Part 50 and 52. The specific U.S. NRC 10 CFR parts related to licensing process described in this study are listed in Table 1.

The next step is applying the current regulatory licensing framework to FBs in terms of siting, operation, and transportation, and figuring out regulatory challenges, considering the characteristics of FBs.

Table 1. List of U.S. NRC 10 CFR Part reviewed in this research

PART	TITLE
PART 50	Domestic Licensing of Production and Utilization Facilities
PART 51	Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions
PART 52	Licenses, Certifications, and Approvals for Nuclear Power Plants
PART 53 (Reserved)	Licensing and Regulations of Advanced Nuclear Reactors

Finally, response strategies were presented to support the licensing of FBs against the challenges derived in the previous step. In this step, Non-Power Reactor (NPR) approach was cited from the "Regulatory Review of Microreactors-Initial Considerations"[12] and the Permit-by-Rule (PBR) approach was referenced from "Key Regulatory Issues in Nuclear Microreactor Transport and Siting"[13]. In addition, an idea on advanced Human-System Interface (HSI) for autonomous operation was derived from "Human-system Interface to Automatic Systems: Review Guidance and Technical Basis"[14] and "A method to Select Human-system Interface for Nuclear Power Plants"[15].

2.2 Probabilistic Risk Assessment

PRA is a comprehensive methodology to evaluate risks and express with numerical values by the product of accident probabilities and consequences. The PRA process used in this study is shown in Figure 6.

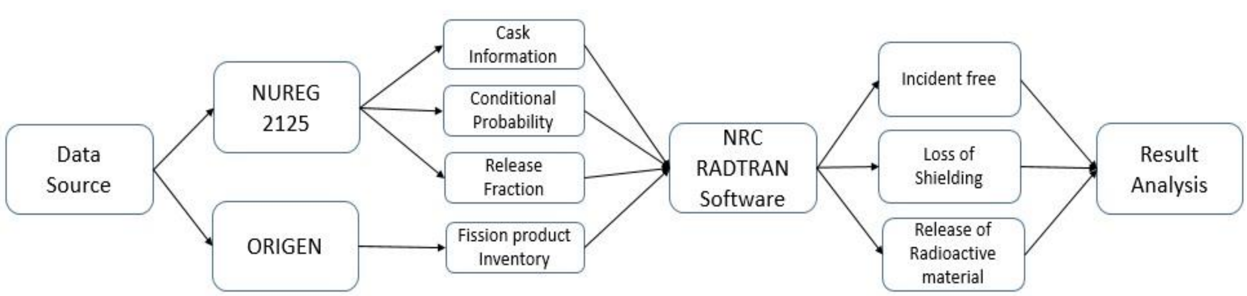

Figure 6. Probabilistic Risk Assessment process in this study

2.2.1 Data Sources

The data required for this study is obtained from NUREG-2125 "Spent Fuel Transportation Risk Assessment"[16] and the Oak Ridge Isotope Generation (ORIGEN)[17] code. NUREG-2125 included extensive research from tests on cask performance used in Type B packaging to analyze the radiation dose risk resulting from severe transportation accidents. Since the objective of this study is to judge the dose risk of FB transportation scenarios through the comparison with dose from chest X-ray, regulatory dose rate, and the dose risk of SNF transportation; scenarios such as cask information, conditional probability on severe accidents, and release fraction of radionuclides analyzed in NUREG-2125 were applied equally. The ORIGEN code was used to determine fission product inventories of FBs obtained through power output, fuel enrichment, and operation time. ORIGEN has a limitation in that it uses LWR fuel information, but it was judged to be a reasonable alternative in current situation with lack of information on FBs.

2.2.2 Program code

The program code used for PRA in this study is NRC-RADTRAN[18] since it is used for risk and consequence analysis of radioactive material transportation. It shows the results by dividing it into incident free, loss of shielding (LOS) accidents, and release of radioactive materials. The models used during this study are shown below and its terms are described in Table 2.

(1) Incident free

(a) Step 1: Calculate dose rate at 1 meter from the cask (Transport Index (TI)).

$$DR_{1m} = DR_{2m} \times \frac{2 + 0.5\, d_e}{1 + 0.5\, d_e} \tag{1}$$

(b) Step 2: Obtain maximum individual dose by calculating total individual dose absorbed at x.

$$D(x) = I(x) \times \frac{2k_0 \times DR_{1m}}{V} \tag{2}$$

$$I(x) = \int_{x}^{\infty} \frac{e^{-\mu r} \times B(r)}{r \times (r^2 - x^2)^{1/2}} dr \tag{3}$$

(2) Loss of Shielding (LOS) accidents

(a) Step 1: Calculate the conditional probability of LOS accidents from rail accident event tree in Figure 7. It should be noted that an assumption 5% of impacts are greater than 145kph and 95% occurs between 113kph and 145kph was made[16].

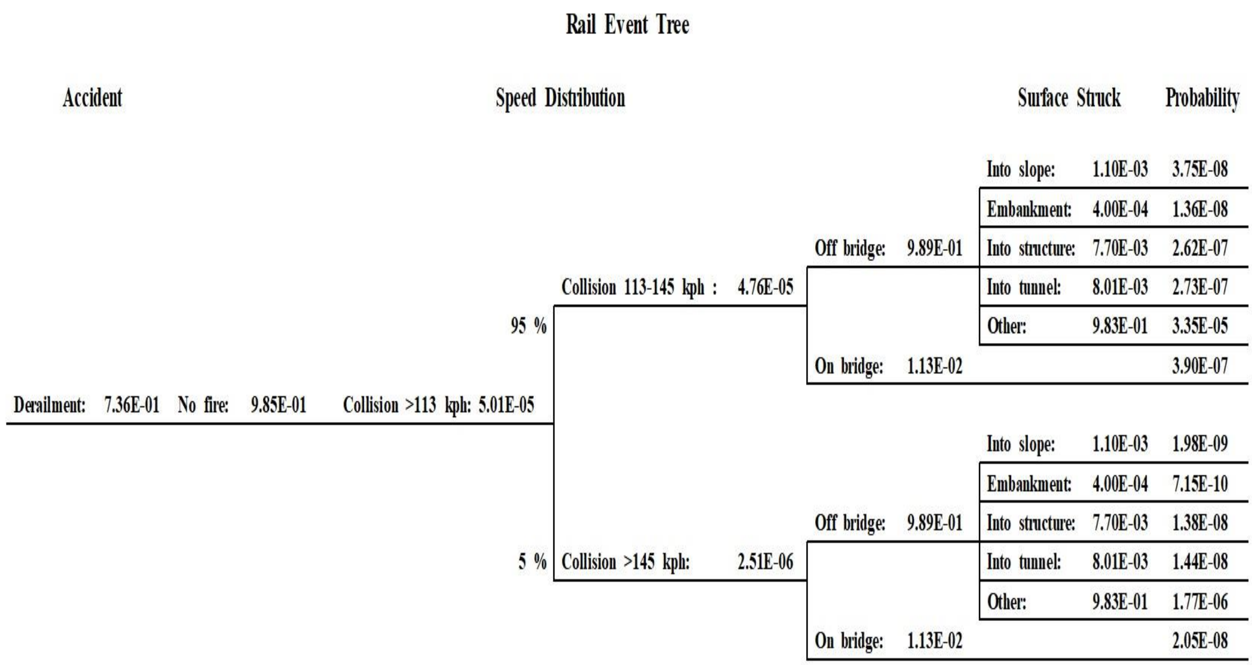

Figure 7. Simplified Rail Accident Event Tree (Reproduced from [16])

(b) Step 2: Calculate individual dose rate per LOS accident at 1 meter and 2 meters to compare with regulatory dose rate.

$$D_{Los}(z) = S_L \times R(E_0) \int_{Z_{min}}^{Z_{max}} \frac{B\left(E_0, \mu_{pb} x_{pb}\right) \times \exp\left[-\mu_{pb}(E_0)\, x_{pb}(z, z') - \mu_{ss}(E_0) x_{ss}(z, z')\right]}{4\pi r^2} dz \tag{4}$$

(c) Step 3: Calculate individual dose risk resulted from LOS accidents.

$$Risk = \sum_{j=1}^{6} \gamma_{j,L} \times D_{LOS} = AR_L \times SV_{j,L} \times DIST_L \times D_{LOS} \tag{5}$$

(3) Release of radioactive materials

(a) Step 1: Determine fission product inventories of FBs right after shutdown. ORIGEN produces fission products in gram, so it should be converted to Curies.

$$Activity(Ci) = \frac{M \times N_A \times \lambda}{A} \times \frac{1(Ci)}{(3.7E+10)(Bq)} \tag{6}$$

(b) Step 2: Calculate population dose from inhalation, cloud-shine, resuspension, and ground-shine.

Ex) Population dose from cloud-shine

$$D_{p-cld-u} = \sum_{i=1}^{62} \sum_{j=1}^{7} Q_7 \times Ci_i \times PPS_{L,m} \times RF_{i,j} \times AER_{i,j} \times CDF_i \times IF \times PD_L \times (UBF \times BDF + USWF \times RPD), \tag{7}$$

$$D_{p-tot-u} = D_{p-inh-u} + D_{p-res-u} + D_{p-cld-u} + D_{p-gnd-u}$$

(c) Step 3: Calculate population total dose risk (Collective dose risk)

$$Risk_{p-tot-u} = \gamma_{j,L} \times D_{p-tot-u} \tag{8}$$

(d) Step 4: Calculate route independent individual dose risk.

$$Individual\ Dose\ Risk = \frac{Risk_{p-tot-u}}{DIST_L \times PD_L} \tag{9}$$

(e) Step 5: Calculate conditional individual dose risk excluding accident rate to compare with that of NUREG-2125.

$$Conditional\ Individual\ Dose\ Risk = \frac{Risk_{p-tot-u}}{AR_L \times DIST_L \times PD_L} \tag{10}$$

2.2.3 Result Analysis

The purpose of this part is to provide a better understanding of the safety of FB transportation by comparing the RADTRAN results with objective guidelines. So, dose consequences and risks computed in incident free, LOS accidents, and release of radioactive materials were compared to chest X-ray dose in ordinary life, regulatory dose rate from 10 CFR 71.47[19] and 71.51[20], and dose risk of SNF transportation from NUREG-2125, respectively to judge the safety of FB transportation.

Table 2. Definition of Terms used in Formulas for RADTRAN model

Terms	Definition
$DR_{1m}(TI)$	Total dose rate at 1 meter from the cask (mrem/h)
DR_{2m}	Total dose rate at 2 meters from the cask (mrem/h)
d_e	Effective dimension of shipment vehicle (m)
$D(x)$	Total integrated dose absorbed by an individual at distance x (rem)
k_0	Point-source package shape factor (m2)
V	Rail speed (km/h)
X	Perpendicular distance of the receptor from shipment (m)
μ	Attenuation coefficient (m-1)
$B(r)$	Buildup factor
R	Distance between shipment and receptor along the route (m)
$D_{LOS}(z)$	Dose rate to a receptor located at z from the source
S_L	Source photon emission rate per unit length of the source
$R(E_0)$	Dose response function
$B(E_0, \mu_{pb}x_{pb})$	Buildup factor for photons in lead
X	Photon path length through each layer
$\gamma_{j,L}$	Probability of an accident j on link L
AR_L	Accident rate on link L: 1.27E-06/km[21]
$SV_{j,L}$	Conditional probability of an accident of severity j on link L
$DIST_L$	Route length L
M	Mass of product
N_A	Avogadro constant
λ	Decay constant
$D_{p\text{-}cld\text{-}u}$	Integrated population dose from cloud-shine in urban area

Table 2 (continued).

Q_7	Unit conversion factor
Ci_i	Activities of radionuclide i
$PPS_{L,m}$	Number of packages
$RF_{i,j}$	Released fraction of radionuclide i in accident severity j
$AER_{i,j}$	Aerosol fraction of radionuclide i in accident severity j
CDF_i	Cloud-shine dose factor
IF	Integral of time-integrated atmospheric dilution factors over all downwind areas computed by the AVINT subroutine of RADTRAN
PD_L	Population density on link L
UBF	Fraction of people indoors
BDF	Building dose factor
$USWF$	Fraction of people out of doors
RPD	Ratio of pedestrian density to residential density
$D_{p\text{-}inh\text{-}u}$	Population dose from inhalation in urban area
$D_{p\text{-}res\text{-}u}$	Population dose from resuspension in urban area
$D_{p\text{-}gnd\text{-}u}$	Population dose from ground-shine in urban area
$D_{p\text{-}tot\text{-}u}$	Total population dose from four pathways in urban area
$Risk_{p\text{-}tot\text{-}u}$	Total population dose risk in urban area

CHAPTER 3. Applicability of Current Regulatory Framework to the Licensing of FBs

As an independent international organization related to the United Nations, the IAEA is sanctioned to observe the safe and peaceful use of nuclear energy internationally and the U.S. NRC regulates the domestic use of nuclear reactors. Therefore, to figure out the current licensing framework in the U.S. in terms of safety, IAEA INSAG-12 “Basic Safety Principles for NPPs” and U.S. NRC 10 CFR were studied. After that, discussion on its applicability to licensing of FBs and possible response strategies if not, were conducted.

3.1. Basic Safety Principles for NPPs (INSAG-12)[10]

The IAEA oversees the safe and peaceful use of nuclear energy. It has published many authoritative documents, ensuring the protection of people and the environment from harmful effects of radiation. The International Nuclear Safety Advisory Group (INSAG) is an advisory group to the Director General of the IAEA that identifies, advises, and shares safety related matters. INSAG-12 “Basic Safety Principles for Nuclear Power Plants” provides the objectives and principles of safe design and operation. It is composed of three safety objectives, three fundamental safety principles, and eight specific principles. The specific principles present eight kinds of safety principles applied during the main stages, from beginning to the end of NPPs lifetime: siting, design, commissioning, manufacturing and construction, operation, accident management, decommissioning, and emergency preparedness.

3.2 U.S. NRC Title 10 of the Code of Federal Regulations (CFR)[11] for Licensing

The U.S. NRC Title 10 CFR is the requirements for all people and organizations to use nuclear materials or operate nuclear facilities. Currently, there are two ways regulated by the U.S. NRC to achieve a license; 10 CFR Part 50 and 10 CFR Part 52[22]. According to Figure 8, initial public hearings, environmental report, and NRC review of the preliminary design for a construction permit (CP) are required within the Part 50 process. After achieving CP, according to Figure 9, final mandatory public hearings are conducted again, and final safety and environmental requirements should be met to obtain operating license (OL). Through this process, it normally takes more than 10 years to obtain a license[23]. In order to reduce the various economic and regulatory risks that may arise during the long period of licensing process with Part 50, the process of Part 52 was introduced[23], combining the CP and OL steps. According to Figure 10, Part 52 process is conducted with an early site permit (ESP) and a design certification (DC) together prior to issuing construction and operating license (COL). Through this streamlined process, it shortens the period within 10 years[23], and reduces financial, regulatory uncertainties[24]. Figure 11 describes two kinds of licenses that could be achieved with the process of Part 50 and Part 52.

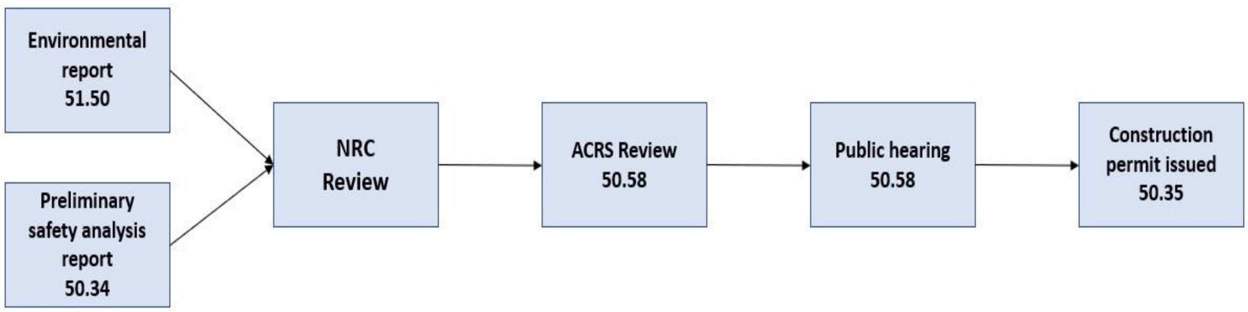

Figure 8. Construction permit process of 10 CFR Part 50 (Reproduced from[24])

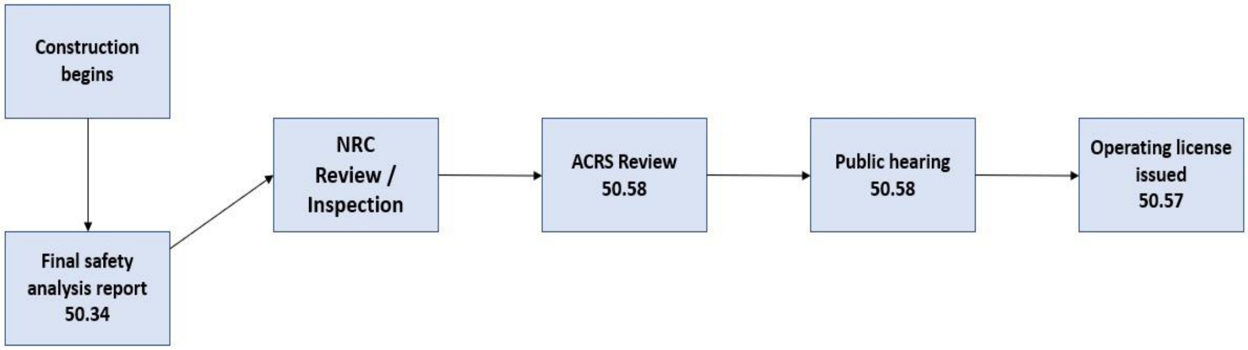

Figure 9. Operating license process of 10 CFR Part 50 (Reproduced from[24])

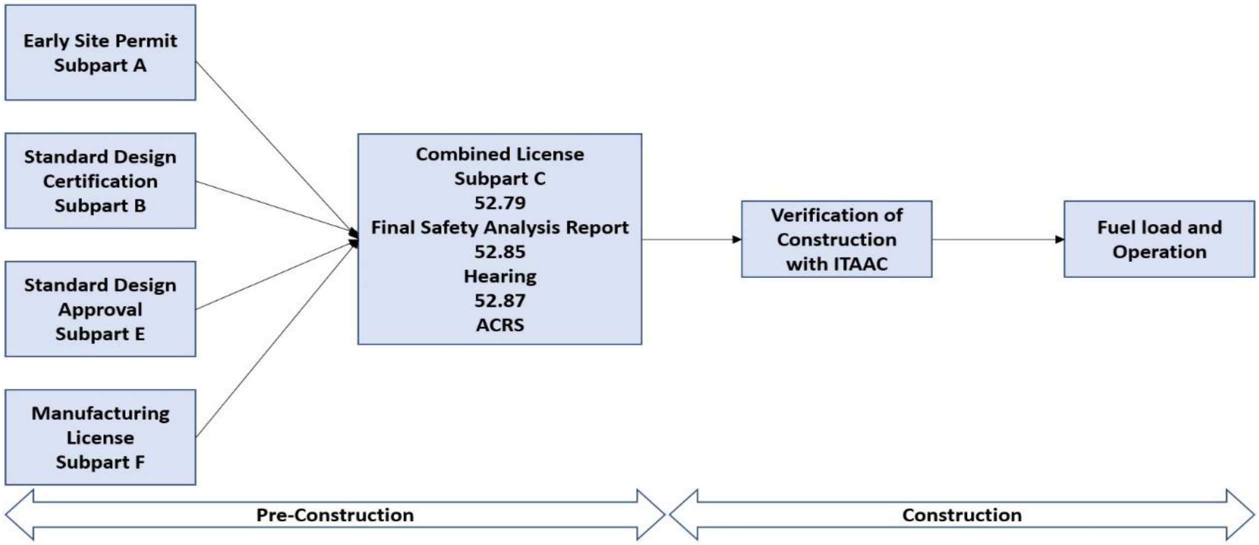

Figure 10. COL process of 10 CFR Part 52 (Reproduced from[24])

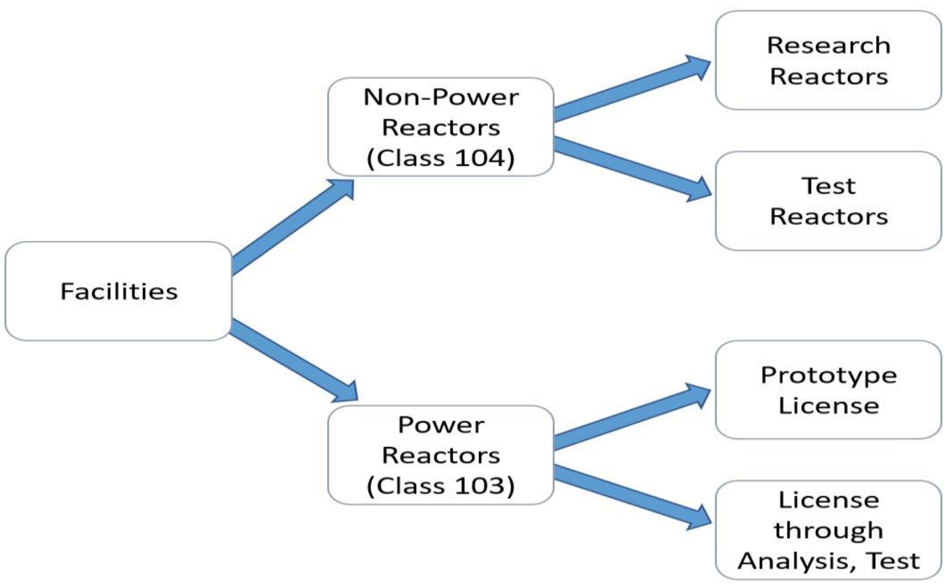

Figure 11. U.S. NRC licensing structure (Reproduced from[25])

3.2.1 10 CFR PART 50

Part 50 provides the requirements for the licensing of production and utilization facilities and notice to all people relating to nuclear activities.

(1) 50.34: Contents of applications; technical information

(a) Preliminary Safety analysis report

Each application for a construction permit should include a preliminary safety analysis report. It should comply that a description and safety assessment of the site, with appropriate attention to features affecting facility design.

(b) Final Safety analysis report

Each application for an operating license should include a final safety analysis report. It should include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components.

(2) 50.58: Hearings and report of the Advisory Committee on Reactor Safeguards

(a) Each application for a facility should be referred to the Advisory Committee on Reactor Safeguards (ACRS) for a review and report.

(b) The commission will hold a hearing after at least 30-days' notice and publication once in the FEDERAL REGISTER on each application for a construction permit. After CP issued, commission may hold a hearing after at least 30-days' notice and publication once in the FEDERAL REGISTER for operating permit.

(3) Appendix N to Part 50: Standardization of NPP designs; Permits to construct and licenses to operate nuclear power reactors of identical design at multiple sites

Appendix N sets the requirements and provisions for licenses of one or more construction and operation of the same nuclear power reactors at different sites.

3.2.2 10 CFR PART 51

Part 51 includes environmental protection regulations applicable to NRC's domestic licensing and related regulatory functions. It requires environmental reports and impact statements which consist of production and utilization facilities, materials licenses, and rulemaking. The environmental review includes analysis of the need for power, socioeconomic impacts, transportation impacts, human-health issues, transmission lines and related human-health impacts, alternatives to the proposed action including the alternative to take no action, accident reviews and severe-accident mitigation alternatives, and review of decommissioning actions[13]. All review and analysis in the document should comply with National Environmental Policy Act (NEPA).

3.2.3 10 CFR PART 52

(1) Subpart A: Early Site Permits

This subpart sets the requirements and procedures applicable to commission issuance of an early site permit for approval of a site for one or more nuclear power facilities.

(2) Subpart B: Standard Design Certifications

This subpart sets the requirements and procedures applicable to commission issuance of rules granting standard design certifications for NPP. Any person could apply a standard design certification for a NPP design which is an evolutionary change from LWR designs.

(3) Subpart C: Combined License

(a) 52.79: Contents of applications; technical information in final safety analysis report

The application must contain a final safety analysis report that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the facility as a whole.

(b) 52.85: Administrative review of applications; hearings

A proceeding on a combined license is subject to all applicable procedural requirements including for docketing and issuance of a notice of hearing.

(c) 52.87: Referral to the Advisory Committee on Reactor Safeguards (ACRS)

The commission should refer a copy of the application to the ACRS.

(4) Subpart E: Standard Design Approvals

This subpart sets the procedures for the filing, NRC staff review, and referral to the Advisory committee on Reactor Safeguards of standard for a nuclear power reactor of the type.

(5) Subpart F: Manufacturing Licenses

This subpart sets the requirements and procedures manufacturing nuclear power reactors to be installed at sites not identified in the manufacturing license application.

3.2.4 10 CFR PART 53 (Reserved)

The current licensing requirements for NPPs with 10 CFR Part 50 and Part 52 do not fully consider the features of advanced nuclear reactors, and so the President Trump signed the Nuclear Energy Innovation and Modernization Act (NEIMA) into law[26] in 2019. The NEIMA prescribes a risk-informed and performance-based regulatory approach that is technology-inclusive which have to fit all potential designs[27]. This act requires the U.S. NRC to make a new licensing pathway for the advanced reactors, and so the U.S. NRC is taking process for 10 CFR Part 53. Currently, preliminary rule language is released to interact with stakeholders and advisory committee. According to preliminary proposed rule language released on Jan 19, 2021[28], part 53 consists of 10 subparts from subpart A "General provisions" to subpart J "Reporting and other administrative requirements". This research covers subpart B, C, D, and F, already released on the document.

(1) Subpart B: Technology-inclusive Safety Requirements

(a) 53.21: Safety Functions

The primary safety function is limiting the release of radioactive materials. Additionally, it is supporting the retention of radioactive materials during routine operation and Licensing Bases Events (LBEs). So, design features and programmatic controls serve to fulfill the safety functions must be maintained over the life of the plant.

(b) 53.22: First Tier Safety Criteria

Design features and programmatic controls must be provided for advanced NPPs, and its effective dose equivalent does not exceed 0.1 rem/yr.

(c) 53.23: Second Tier Safety Criteria

Design features and programmatic controls must be provided to ensure maintain the necessary reliability to address LBEs and provide measures for defense in depth.

(d) 53.24: Licensing Basis Events

Licensing Basis Events (LBEs) must be identified for each advanced nuclear plant and address malfunctions of plant Structures, Systems, and Components (SSCs), human errors, and the effects of external hazards.

(e) 53.25: Defense in depth

Defense in depth (DID) must be provided to compensate for epistemic and aleatory uncertainties, which include the ability of barriers to limit the release of radioactive materials and performance of plant SSCs and personnel, and programmatic controls.

(2) Subpart C: Design and Analysis Requirements

(a) 53.41: First Tier Safety Criteria

Functional design criteria and the first-tier safety criteria required in 53.22 are not exceeded.

(b) 53.42: Second Tier Safety Criteria

Doses from effluents during normal operation are as low as reasonably achievable.

(c) 53.44: Design Requirements

Design features must be demonstrated capable of accomplishing safety functions without adverse effects to other safety features: Analysis, Test programs, Prototype testing, and Operating experience. Also, safety and security must be considered together in design process.

(d) 53.45: Analysis Requirements

PRA must be performed to identify potential failures, degradation mechanisms, susceptibility to hazards, and other risks to safety functions and used to determine LBEs, classify safety significance and human actions, and evaluate defense in depth.

(3) Subpart D: Siting Requirements

(a) 53.50: General Siting

A siting assessment for advanced NPPs must be performed and ensure that external hazards and site characteristics that might contribute to the initiation, progression, or consequences of LBEs.

(b) 53.51: External Hazards

SSCs must be designed to withstand the effects of natural phenomena and man-related hazards. The design bases external hazard level must be identified and address a range of estimated external hazard frequencies. Especially, Safe Shutdown Earthquake Ground Motion must be designed.

(c) 53.52: Site Characteristics

Characteristics of the site and surrounding area should be identified, estimated, and considered in the design and analyses requirement.

(d) 53.53: Population-related considerations

Every site must have an exclusion area, low population zone, and provide population center distance as defined in 10 CFR Part 100.3. Reactor sites should be located away from public.

(4) Subpart F: Requirements for Operation

(a) 53.80: Facility Safety Program

Each licensee must establish and implement a Facility Safety Program (FSP) that routinely and systematically evaluates potential hazards, operating experience relate to plant SSCs, human actions, and programmatic controls affecting the safety functions.

(b) 53.81: Facility Safety Program Performance Criteria

Each licensee for an advanced NPPs must take measures to protect public health and minimize danger to life or property. Risk reduction measures must be assessed related to release of radioactive materials in plant effluents during normal operation, not excessing 0.3 mrem from liquid effluents or 1 mrem from gaseous effluents[28]. Also, potential risk reduction measures must be assessed related to LBEs, identified hazards, or other specific contributors to the overall cumulative risk from unplanned events.

3.3 Applicability of current regulatory framework to FBs

In this section, the discussion on the applicability of current regulatory framework to FBs was conducted, and possible countermeasures were suggested to solve the discovered regulatory challenges. In this study, issues on siting, autonomous operation, and transportation of FBs were described.

3.3.1 Siting

Siting is used to select an appropriate location for the safe operation including the process of analyzing nature or human made hazards, radiological impact on the public, and the local environment, which is an important process since local circumstances can affect safety[10]. Accordingly, during the licensing process for construction permit (CP), NRC requires an environmental report (ER) which takes several years for site investigation requiring detailed site-specific information. The detailed site-specific information includes impacts on area populations and surrounding environmental conditions as identified in RG.4.2 guidance[13]. To minimize the impacts on the environment of site, regulations and guidance for siting is described in detail in 10 CFR Part 50, 10 CFR Part 52, 10 CFR Part100, RG4.7, NUREG series, and the IAEA also provide general nonbinding siting regulations and guidance[29]. Table 3 shows the summary of regulations for site environments specified in the various documents above. It can be seen that a considerable number of site restrictions were imposed in order to achieve the safety objective by protecting people and important infrastructures near NPPs.

Table 3. Regulations for site environment

Source	Contents
10 CFR Part 50.34 & Part 52.79	An individual located on the boundary of the exclusion area for any 2-hour period (or low population zone) would not receive a radiation dose in excess of 25 rem total effective dose equivalent with the onset of the postulated fission product release (or with the radioactive cloud).
10 CFR Part 100.3	Exclusion area means area surrounding the reactor and residence within the exclusion area shall normally be prohibited. Low population zone means the area surrounding the exclusion area where an appropriate protective measure could be taken in the event of a serious accident.
10 CFR Part 53.53	Every site must have an exclusion area, low population zone, and provide population center distance as defined in 10 CFR Part 100.3. Reactor sites should be located away from public.
10 CFR Part 100.21	Located more than 1 mile from any commercial rail line.
10 CFR Part 100 Appendix A	Located more than 5 miles from surface faults and capable tectonic structures.
RG 4.7	Located within 20 square miles, not exceeding 500 people per square mile.
RG 1.7	Located away from population centers of more than 25,000 people.
NUREG-0800	Located more than 10 miles from an airport and 5 miles from a hazardous site.

(1) Applicability to FBs

According to FB attributes discussed in Chapter 1, FBs are expected to be developed for multisite deployments for universal use based on the concept of "plug and play". By developing M&S tools capturing the dynamic behavior of FBs coupled with environmental conditions, ground and soil interactions, and other external factors[4] FBs will embrace all characteristics of the "plug and play" concept. The specific characteristics that the FBs are expected to have for their use are an isolated grid, emergency energy supply to areas hit by natural disaster, harsh locations, military bases, space, and electricity supply to electric vehicles. The target markets mentioned above show the expansive use of FBs in anywhere like chemical batteries currently used widely.

However, the current regulations and guidance presented above, directly contradict the expression "expansive use" by placing numerous restrictions on the site to prevent the damage in emergencies. In response, technology suppliers are designing enhanced safety systems for advanced reactors to prevent and mitigate accidents such that the consequences remain within the regulatory limits under all possible abnormal conditions. With the improved safety systems, FBs are expected to have attributes that reactor accidents will result in a significantly reduced source term and limit any radioactive material fallout to within the site boundary or be limited to within short distance of the Exclusion Area Boundary (EAB)[29]. Therefore, excessive restrictions on the site may not be suitable for the universal use of FBs equipped with enhanced safety system.

Also, the time consumed for site investigation cannot be applied to FBs. As discussed in the licensing process, it takes several years, and it would interfere with the efficient deployment of FBs to multisite and may not meet the needs for FBs demand such as military usage where urgent siting permission may be required.

Therefore, current regulations and licensing process in siting do not apply to the characteristics of FBs in terms of site restrictions and excessive time burdens on site investigation.

(2) Response Strategies

(a) Site Restrictions: Non-Power Reactor (NPR) Approach[12]

Table 4 shows the emergency planning zone (EPZ) for current NPPs where preparations are made to promptly shelter in place to perform environmental monitoring and to implement urgent protective actions, which is suggested by the IAEA safety standards GS-G-2.1 "Arrangements for preparedness for a nuclear or radiological emergency"[30] based on thermal output. Even though the output of FBs have not been decided yet, according to the discussion so far, the output would be less than 25 MWth[6], so an EPZ size would be within 1.5 km.

Table 4. Suggested Emergency Planning Zone size for NPPs (Reproduced from[30])

Authorized Power Level	Acceptable EPZ size
2 MW < Output < 10 MW	500 meters
10 MW ≦ Output < 100 MW	0.5 ~ 5 km
100 MW ≦ Output ≦ 1000 MW	5 ~ 30 km
Output > 1000 MW	5 ~ 30 km

However, since EPZ size was determined to minimize radiological impacts in emergencies considering safety features of current NPPs, it is reasonable to re-analyze the EPZ size of FBs that is expected to have enhanced safety systems and so whose dose could be under the regulatory limit for NPRs (1 rem)[12]. Accordingly, applying the research on Non-Power Reactor (NPR) seems to be appropriate. NUREG-1537 “Guidelines for preparing and reviewing applications for the licensing of NPRs” addresses many aspects of NPRs licensing including construction permit, initial operating license, and license amendment[31]. Table 5 shows the EPZ size of NPRs studied in NUREG-1537, and it can be seen that EPZ size of FBs would be reduced to about 400 meters if it is applied. Although there is still EPZ areas, compared to the size applied to LWRs, it is significantly improved and seems to be closer to the flexibility of siting.

Table 5. Emergency Planning Zone size for NPRs (Reproduced from[31])

Authorized Power Level	Acceptable EPZ size
Output ≦ 2 MW	operations boundary
2 MW < Output ≦ 10 MW	100 meters
10 MW < Output ≦ 20 MW	400 meters
20 MW < Output ≦ 50 MW	800 meters
Output > 50 MW	case-by-case

Meanwhile, according to World Nuclear Association (WNA) report, Small Modular Reactors (SMRs) developers insist that the EPZ size for SMRs with an output of 300 MWth should be within 300 meters, since advanced reactors with enhanced safety features may replace fossil fuel power plants located near big cities[32]. If their argument is accepted, the EPZ size of FBs could be reduced to 25 meters. However, the sudden request for a big regulatory change can be burdensome for the regulatory authorities. Therefore, at this point, starting with the NPR approach it is desirable to request gradual deregulation, as the design of FBs is materialized, and its safety system is verified. Finally, the zero-EPZ concept should be applicable in the future[33] in order for FBs to be competitive against other energy sources. However, to apply NPR approach, information on power conversion should be considered first because FBs are power reactors unlike NPRs.

(b) Excessive Time burdens: Permit-by-Rule (PBR) approach[13]

Permit-by-Rule (PBR) is a preconstruction permit issued by a reviewing authority that may be applied to a number of similar emissions units or sources within a designated category[34]. They are used by many states and governments for an optional permitting provision for safety guaranteed facilities. For example, on-site power generation and site for fuel-burning equipment such as natural gas or oil are applied to PBR in Georgia State[35].

The key to applying PBR is to prove high levels of safety reliability at the design stage. Considering the enhanced safety features of FBs, PBR would be a fast and reliable regulatory approach for achieving multisite deployment and urgent site transfer by reducing the time for siting to few days or weeks instead of several years with current regulation[13].

Meanwhile, PBR approach would be conducted with the analysis of plant parameter envelope (PPE) and site parameter envelope (SPE). Major steps for it may include[13]:

- Defining a safety design PPE of mandatory requirement for construction and operation under PBR.
- Defining a SPE related to plant design safety parameters.
- Safety assessment with PPE and SPE.
- PBR site acceptance criteria would be made based on above steps.

So, defining well developed PPE and hypothetical SPE is essential for PBR approach. PPE would be analyzed in design process and SPE may be conducted by M&S tools and ML/AI techniques for expected areas. Also, safety assessment would help to reduce the uncertainty about the site characterization. The followings shows important issues for considering the PBR approach[13]:

- The detailed definition and methodologies of PPE and SPE for PBR safety assessment.
- The level of confidence required in site field-characterization information.
- Acceptable means for mitigating key envelope such as sites experiencing a natural hazard.

As a result, when the design of FBs is materialized and high-quality enhanced safety system is proven, it is expected that PBR may be a reasonable approach that would sufficiently replace or complement current years-long siting evaluation process for FBs that may require hundreds or thousands to be deployed at the same time.

3.3.2 Autonomous Operation

Operation is a key step for the life cycle of NPPs, so the plant owner ensures that all structures, components, and the number of experienced personnel related to safe operation. Once NPPs begin operation, the safety performance depends on the capabilities of the facility personnel, especially in emergencies. As shown in Table 6, therefore, the composition of the control room and related staffing regulations are specified in Part 50 and Part 55. What stands out is regulations set the minimum required number of licensed operators on site and emergency standby status of staffs in detail. Even in 10 CFR Part 53 for advanced reactors, it still requires the action of a licensee. As such, operation and response to emergencies for NPPs are highly dependent on the human personnel.

Table 6. Operating regulations related to operators

<table>
<tr><th>Source</th><th colspan="3">Contents</th></tr>
<tr><td>10 CFR Part 55.4</td><td colspan="3">An operator is any individual licensed under this part to manipulate a control of a facility. A senior operator is any individual licensed under this part to manipulate the control of a facility and to directly the licensed activities of licensed operators.</td></tr>
<tr><td>10 CFR Part 50.54(k)</td><td colspan="3">An operator or senior operator shall be present at the controls at all times during the operation.</td></tr>
<tr><td>10 CFR Part 50.54(m)(1)</td><td colspan="3">A senior operator shall be present at the facility or readily available on call at all times during its operation and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shut-down or significant reduction in power, and refueling.</td></tr>
<tr><td>10 CFR Part 50.54(m)(2)(ii)</td><td colspan="3">Each licensee shall have at its site a person holding a senior operator license for all fueled units at the site who is assigned responsibility for overall operation at all times there is fuel in any unit.</td></tr>
<tr><td>10 CFR Part 50.54(m)(2)(iii)</td><td colspan="3">When a nuclear power unit is in an operational mode other than cold shutdown or refueling, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times.</td></tr>
<tr><td rowspan="5">10 CFR Part 50.54(m)(2)(i)</td><td>Number of operating units</td><td>Position</td><td>One unit</td></tr>
<tr><td rowspan="2">None</td><td>Senior operator</td><td>1</td></tr>
<tr><td>Operator</td><td>1</td></tr>
<tr><td rowspan="2">One</td><td>Senior operator</td><td>2</td></tr>
<tr><td>Operator</td><td>2</td></tr>
<tr><td>10 CFR Part 53.80</td><td colspan="3">Each licensee must establish and implement a Facility Safety Program (FSP) that routinely and systematically evaluates potential hazards, operating experience relate to plant SSCs, human actions, and programmatic controls affecting the safety functions.</td></tr>
<tr><td>NUREG-0654</td><td colspan="3">Addresses the minimum staffing requirements for emergencies, 10 staff onsite, and 11 additional staff within 30 minutes, and 15 additional staff within 60 minutes.</td></tr>
</table>

(1) Applicability to FBs

According to FB attributes discussed in Chapter 1, FBs are expected to be developed in the way of unattended operation that means automation. Reviewing important insight from studies of the Three Mile Island (TMI), Chernobyl, and other NPP accidents is that errors were resulted from human factors deficiencies[36], therefore the development to automation should be attained, which would be achieved with advanced design of passive safety systems, simplicity of operation, automation, and limited important human action based on innovative ML and AI technology.

As shown above, however, current regulations about control rooms and licensed operators seem to be highly dependent on personnel and do not fully capture automation capabilities. Even the current staffing level for a 1,000 MWe plant is about 1,000 which is essentially one person per 1 MWe including security forces[25]. Also based on the Table 6 above, if one FB is considered as one unit, it can be seen that at least 4 operators are required under current regulations. If security forces are also considered, the number of personnel would increase more.

Meanwhile, considering the characteristics of advanced reactors and improved technology, several studies such as NUREG/IA-0137 "A study of Control Room Staffing Levels for Advanced Reactors"[37] have been conducted, but a minimal control room was still required. Even the NuScale SMRs, which have shown the most progress at this point and would be deployed in the near future, successfully obtained an exemption to reduce the number of staff presented above against the current regulations, but it was also not a complete elimination of the control room staff[25].

Nevertheless, according to Figure 12 from FB initiatives workshop, it is still expected to develop FBs into high automation with remote monitoring and no operator control, at least partial control. It is because autonomous operation with AI is the most fundamental value of FBs that is aiming to be used by ordinary people like chemical batteries. Therefore, current regulations related to operators cannot be applied to autonomous operation of the most advanced reactors, FBs, and an exemption from regulations is required.

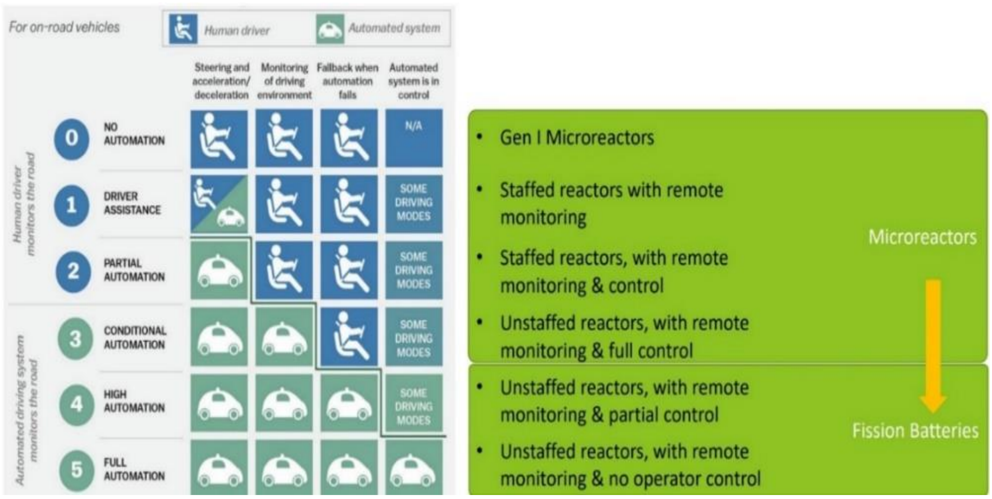

Figure 12. Levels of Autonomy[38]

(2) Response Strategies

Human-system interface (HSI) technology is defined that part of NPPs through which personnel interact to perform their functions and tasks with the systems as shown in Figure 13. Major HSIs include alarms, information displays, controls, and procedures[39]. The primary purpose of the HSI is to provide the operator with a means to monitor and control the NPPs and to restore them to a safe state when adverse conditions occur[15], and so it is widely used at present.

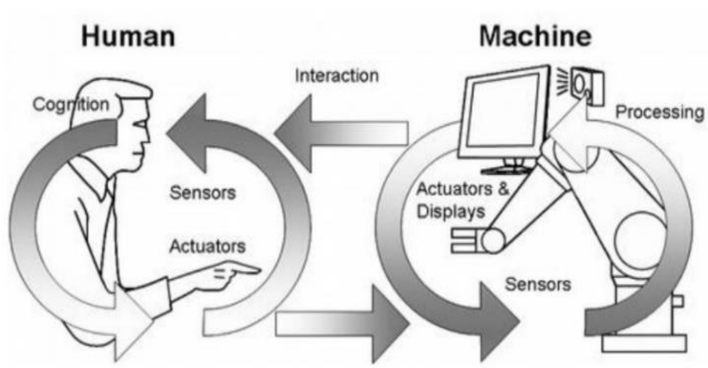

Figure 13. Working of HSI[40]

In advanced HSI systems with improved telecommunication technologies, an offsite space equipped with a set of computer displays and input devices may replace the control rooms and make it feasible to remote monitoring and control. Even, the enhanced safety systems and simplified design may allow one controller to manage multiple reactors.

According to Table 7 “General HSI Requirements for Levels of Operation”, HSI technology still requires human functions and tasks in the current state, however for FBs equipped with AI technology, AI will replace human functions required by HSI. Therefore, advanced HSI managed by AI would be the core technology required for autonomous operation of FBs.

Table 7. General HSI requirements for levels of automation[14]

Level	Automation Functions	Human Functions	HSI Requirements
Shared Operation	Some functions or tasks	Manual performance	See requirements for autonomous operations
Operation by Consent	Under close monitoring and supervision by operators	Monitor, approve actions, and intervene	Provide information on task status, authorized task and Identify valid options
Operation by Exception	Autonomous except for specific situation are encountered	Must approve of important decisions and may intervene	Provide information on reason for exception
Adaptive Automation	Flexible task assignment	Flexible task assignment	Operator control of automation shifts, notification of impending shifts and shift conformation

Meanwhile, the advancements in automation systems and the development of computer performance, such as AI/ML dedicated architectures, have had a tremendous impact on the deployment of automation technologies and systems in many industries including the nuclear industry over the past few years, one of them is a Nearly Autonomous Management and Control (NAMAC) system. The NAMAC system, where the Digital Twin (DT) plays a key role, is processed with knowledge base, DT development layer, and NAMAC operational layer and requires advanced ML algorithms[41]. As such, several systems for the automation of NPPs are being developed, and AI/ML technology development is the key to fully replace personnel. Therefore, expectations on the progress of a complete remote-control system with advanced telecommunication technologies and HSI operated by AI could make it possible for FBs to be exempted from current regulations related to staffing.

If autonomous operation of FBs is achieved and remote monitoring of multiple FBs becomes possible, it may require the change of roles in the qualified licensees, from controlling NPPs directly to monitoring display and confirming AI/ML-driven control software.

3.3.3 Transportation

Transportation in nuclear industry means moving radioactive materials to the desired places, and related regulations are co-managed by 10 CFR 71 of the U.S. NRC and 49 CFR 173 of the U.S. Department of Transportation (DOT).

Table 8 shows the kinds of regulated packaging types for the transportation of radioactive materials, and category of Type B packaging. The information of A_1 and A_2 is described in 10 CFR 71 Appendix A “Determination of A_1 and A_2”[42].

Packaging type is determined by the quantities of radioactive materials and each packaging is required to resist against certain conditions. In order to verify the safety performance of each packaging, the U.S. NRC requires specific tests on the Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC). Especially, tests on the HAC for Type B packaging assuming possible severe transportation accidents are specified in 10 CFR 71.73[43]: 9 m free drop onto a flat surface, 1 m free drop onto a 15 cm diameter vertical cylinder (puncture test), fully engulfing fire of 800 °C (thermal test), and 0.9 m for fissile material and 15 m for all packages water immersion test. The reason for the packaging performance test is to prevent the leakage of radioactive material or to control it below a prescribed regulatory dose rate described in Table 9. Therefore, even in the severe transportation accidents, packaging should be able to maintain its function and prevent release of radioactive materials under regulatory limit.

Table 8. Classification of Packaging Type and Specific Category of Type B

Packaging Type		
Industrial	**Type A**	**Type B**
Materials with little hazard from radiation exposure (e.g., contaminated clothing, laboratory samples, and smoke detectors)	Small quantities of radioactive materials with higher specific-activity levels than industrial packages. Require the resistance against moderate heat, cold, and various impact. (e.g., medical use, limited quantities of radioactive materials)	Large quantities and the highest levels of radioactivity materials. Require the resistance against severe weather and accident conditions. (e.g., spent fuel, large amount of specified radioactive materials)

Category of Type B (RG 7.11)			
Contents Form / Category	**Category Ⅰ**	**Category Ⅱ**	**Category Ⅲ**
Special Form	Greater than 3,000A_1 or greater than 1.11 PBq (30,000 Ci)	Between 3,000A_1 and 30A_1, and not greater than 1.11 PBq (30,000 Ci)	Less than 30A_1 and less than 1.11 PBq (30,000 Ci)
Normal Form	Greater than 3,000A_2 or greater than 1.11 PBq (30,000 Ci)	Between 3,000A_2 and 30A_2, and not greater than 1.11 PBq (30,000 Ci)	Less than 30A_2 and less than 1.11 PBq (30,000 Ci)

Table 9. Regulated radiation dose rate for packages

Source	Contents
10 CFR Part 71.47	2 mSv/h on the external surface package 10 mSv/h allowed in the specific condition such as closed transport vehicle 0.1 mSv/h at any point 2 meters from the outer lateral surfaces of vehicle 0.02 mSv/h at normally occupied space
10 CFR Part 71.51 ***Additional for Type B**	10 mSv/h at 1 meter from the external surface of the package on the HAC

(1) Applicability to FBs

Among the attributes of FB discussed in Chapter 1, "installed" is meaning that FBs would be fully manufactured at the factory and delivered to users in a condition that fuel is also inserted. Considering the characteristics, three transportation scenarios could be made: 1) Transporting fresh fuel to the manufacturing factory, 2) Deploying new FBs to users, 3) Transferring FB's location in use. Among them, the second and third scenarios are new ideas of moving reactors equipped with fuel devised from the characteristics of FBs, and so technical and regulatory challenges are expected.

Against technical challenges, the key for FB transportation is to achieve complete safety reliability by preventing release of radioactive materials, successful radiation shielding, decay heat removal, and maintaining subcriticality with the combination of FB's own DID strategy and performance of packaging. Among them, maintaining subcriticality is the biggest challenge[44], where many parameters work on it: 1) type, mass, and form of the fissile material, 2) moderator to fissile material ratio, 3) amount and distribution of absorber materials, 4) package geometry, 5) reflector effectiveness[45]. However, when FBs are fully developed and commercialized, technical challenges such as maintaining subcriticality issue should have already been resolved. Therefore, technical challenges are not considered in this study under the assumption that they can be completely solved by design.

For regulatory issues, current regulations for transporting radioactive materials described above stipulate the packaging type and the limit of dose rate for packages. Currently, one of the most hazardous radioactive materials in transportation at present is SNF, which requires the use of Type B packaging. SNF transportation is likely to be similar to FBs transportation in that it moves highly radioactive and irradiated fuel after shutdown. However, since SNF is extremely hot and radioactive right after shutdown, it is transported after several years of safety measures unlike FBs such as cooling down in the pool, so it could be evaluated that FB transportation would be more hazardous. So, it seems reasonable to use Type B packaging for FB transportation, which is the safest and robust at current state. Accordingly, if FB packages using Type B packaging keep the regulatory dose limit, licensing issues related to FB transportation would not occur[44], which means FB transportation is possible under current regulations.

CHAPTER 4. Risk Analysis of FB transportation

In Chapter 4, risk analysis was conducted to numerically evaluate the risk of FB transportation against the current regulations. Background knowledge about casks used for Type B packaging, conditional probabilities of accidents, radionuclides, and its release fractions required for this study were referred to NUREG-2125 “Spent Fuel Transportation Risk Assessment”. Calculated dose risk using NRC-RADTRAN program was used for judging the safety of FB transportation.

4.1 Main Results from NUREG-2125

The U.S. NRC has conducted several risk assessments of radioactive material transportation, and NUREG-2125 is the most recent version of SNF transportation following NUREG-0170 and NUREG-6672. NUREG-2125 conducted the research by dividing into three categories: incident free transportation, loss of shielding (LOS) accidents, and release of radioactive materials.

During incident free transportation, the radiation dose risk was compared with the U.S. background radiation dose 0.0036 Sv/year (0.36 rem/year) or 4.1E-07 Sv/h (4.1E-05 rem/h)[16]. As a result, the risk of SNF transportation during the incident free situation was significantly small compared to the U.S. background radiation dose.

Meanwhile, this document extensively studied the performances of three NRC-certified casks of HI-STAR 100 Rail-Steel Cask, NAC-STC Rail-Lead Cask, and GA-4 Truck-DU Cask[16] to provide objective reliability on the test. The analysis on the casks were conducted assuming accident situations in collision and fire that may occur during truck and rail transportation. The casks that passed the tests assuming on the HAC specified in 10 CFR Part 71 maintained most of their functions even in the severe accident. However, it should be noted that when transporting uncanistered SNF using rail-lead cask (NAC-STC), radioactive materials were released following loss of lead shielded cask accident. However, total collective dose risk of an accident resulting in loss of lead shielded cask while transporting uncanistered SNF was between 1.2E-12 person-rem and 9.4E-12 person-rem, smaller than one in billions. So, the results of NUREG-2125 showed that the risk of SNF transportation was negligible.

4.2 FB NRC-RADTRAN Models

RADTRAN is a risk analysis program developed by Sandia National Laboratories (SNL), but now managed by the U.S. NRC, for analyzing the frequencies and consequences of transporting radioactive materials. RADTRAN evaluates the risk during incident free transportation by evaluating the accidents that might occur. Transportation risk analysis is based on the risk triplet: what can happen (scenario), how likely it is to happen (frequency of scenario), and what are the consequences[18]. Therefore, RADTRAN subdivides the transportation scenarios into highway, rail, and waterway, requiring each accident probabilities and estimates the dose risk by multiplying accident frequencies and consequences for each scenario. With the result, users can make a judgment on the dose risk of specific transportation by comparing it with the regulatory limits. Examples of RADTRAN input and output file of this study are attached in Appendix A.

4.3 FB Transportation Scenario

The goal of FBs is “universal use” of nuclear energy by the ordinary people in any location. So specifying a transportation route may not be appropriate for FBs. However, since RADTRAN requires specific route information, this study created a hypothetical scenario that one FB packaged by NAC-STC (rail-lead cask) would be transported using railroad only through urban area on the route from Maine Yankee to ORNL described in NUREG-2125 (Length: 161km, Population density: 2527 people/km^2) right after the shutdown. The reason why the rail transportation using rail-lead cask was the only case considered is because severe accidents that may cause LOS followed by release of radioactive material could only occur during this scenario according to

NUREG-2125. Therefore, the key is to calculate the dose risk while one FB is transported using railroad during incident free, LOS accidents, and release of radioactive materials, and compare them to the dose in everyday life (chest X-ray), regulatory dose rate, and dose risk of SNF respectively.

In this transportation scenario, five assumptions were made:

1) Risks related to security and safeguard during transportation are not considered.
2) Technical issues such as maintaining subcriticality do not occur during transportation.
3) Safety structure of FB packages is the same as spent fuel packages (i.e. Safety features of FBs such as its containment system, are not considered).
4) Radionuclides and its release fractions of FBs are the same as LWR spent fuel.
5) The standardized size of FBs is 5 meters.

4.3.1 Incident free Transportation

Incident free transportation analysis provided in RADTRAN is modeled using Figure 14 as a basic concept.

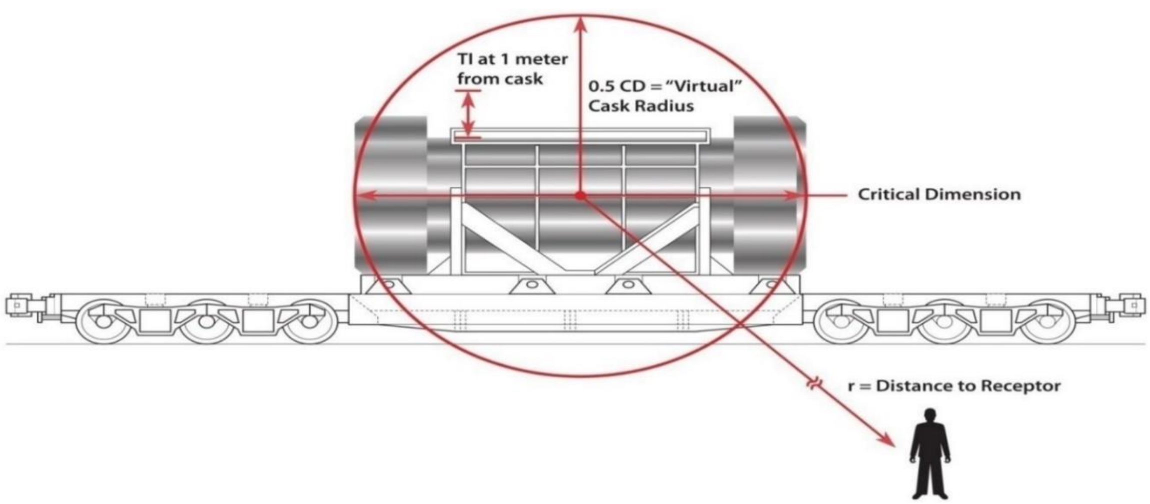

Figure 14. RADTRAN model of the incident-free transportation[18]

RADTRAN models the cask as a sphere which emits radiation uniformly in all directions and recognizes the length of cask as the same value as its diameter, which is named a critical dimension. When receptor is located farther than critical dimension, RADTRAN models the dose to the receptor as proportional to $1/r^2$ and $1/r$ when it is closer[16].

During incident free transportation, maximum individual in transit dose is proportional to TI calculated in equation (1). Since the actual dose rate at 2 meters from the cask of FBs is not known, regulatory limit that is 0.1 mSv/h (10 mrem/h) at 2 meters specified in 10 CFR 71.47[19] was used conservatively. Also, as the standardized size of FB was assumed to be 5 meters, d_e was computed as 4.57 meters on the RADTRAN program. Finally, TI was determined to be 13 mrem/h. Accordingly, maximum individual in transit dose was computed, as seen in Table 10 using equation (2) and (3) with RADTRAN input parameters shown in Table 11.

Table 10. Maximum Individual Dose during incident free transportation (mrem)

Maximum Individual Dose during incident free transportation (mrem)	Radiation dose of chest X-ray (mrem)
5.36E-04	10

As a result, maximum individual dose is very small compared to radiation dose of chest X-ray.

Table 11. RADTRAN Incident-free input parameters

Input Parameters	Size(CD)	Dose rate at 1 meter	Gamma Fraction	Neutron Fraction	Crew Size	Vehicle Speed	Crew Distance	Number of Shipments
Value	5 meters	13 mrem/h	0.89	0.11	3	64 km/h	150m	1
Source	-	-	NUREG-2125		RADTRAN Manual			-

It should be noted that regulatory dose limit was used for TI in this study. Therefore, it is expected that TI of FBs would be reduced with its own advanced safety features, and so consequently maximum individual dose would decrease.

4.3.2 Loss of Shielding (LOS) Accident

LOS accidents occur when some parts of packaging for radiation shielding are degraded during severe accidents. According to NUREG-2125, LOS occurs only in case of using rail-lead cask that is relatively soft and melts at a relatively low temperature.

Meanwhile, dose risk in accidents is calculated as the product of accident probabilities and dose consequences, so calculating accident probabilities that result in LOS should be the first step. According to NUREG-2125, the result of cask performance test shows that from an impact speed of 145 kph, damage to the cask and canister begins with the significant slumping of the lead gamma shielding material. So, one example of conditional probability at the impact speed of 145 kph was calculated referring to the rail accident event tree in Figure 7.

- Rail event tree impact into slope: 3.75E-08 (a)
- Rail event tree impact into embankment: 1.36E-08 (b)
- Impact into slope and embankment for the hard rock: 0.055 (c)
- Impact angle greater than 45 degrees (significant degree for severe lead slump): 0.333 (d)
- Rail event tree on bridge: 3.9E-07 (e)
- Falling off the bridge: 0.02 (f)
- Falling off the bridge for the hard rock: 0.05 (g)
- Impact angle greater than 45 degrees during the accidents on bridge: 0.5 (h)
- Conditional Probability = [(a+b)·c·d]+(e·f·g·h) = 1.13E-09

Table 12 shows the calculated conditional probabilities of LOS accidents from severe impact and their respective slumped fraction analyzed in NUREG-2125, which were directly used as RADTRAN input parameters.

Table 12. Parameters of LOS accidents from severe impact

Collision Point	Impact Speed (kph)	Slumped fraction	Conditional Probability from Event tree	Final Conditional Probability considering Collision Point
End (0.1)	193	0.0725	5.96E-11	5.96E-12
	145	0.0170	1.13E-09	1.13E-10
Corner (0.6)	193	0.0634	5.96E-11	3.57E-11
	145	0.0234	1.13E-09	6.79E-10
Side (0.3)	193	0.00316	5.96E-11	1.79E-11
	145	0.00426	1.13E-09	3.4E-10

Moreover, shielding factor that is the fraction of ionizing radiation to which the receptors are exposed acts an important variable from 0 (100% shielding) to 1 (no shielding). In general, the shielding factor is determined for each area (Urban: 0.018, Suburban: 0.87, Rural: 1)[46], but in this study, considering that no public would be located within at least 10 meters and only emergency responders operate, the shielding factor of 0.1 for them was applied, since radiation protective equipment using lead for emergency responders prevents over 90%[47].

Table 13. Average and Maximum Individual Dose Rate during LOS accidents

Slumped Fraction	Average Individual Dose Rate (mrem/h)		Maximum Individual Dose Rate (mrem/h)
	1 meter	2 meters	
0.0725	425	190	717
0.017	80	35.9	169
0.0634	364	163	627
0.0234	115	51.7	232
0.00316	12.5	5.65	32
0.00426	17.1	7.72	42.9
Regulatory Dose Rate	**1000 mrem/h on the HAC (10 CFR 71.51)**	**10 mrem/h (10CFR 71.47)**	

Finally, the average and maximum individual dose rate at 1 meter and 2 meters calculated using equation (4) during LOS accident scenarios is shown in Table 13. Compared to regulatory dose rate at 2 meters (10 mrem/h) in normal condition, almost all LOS accident scenarios exceeded it. However, since LOS accidents should be considered in the same condition as the HAC, it is appropriate to compare it to additional requirements for Type B packaging on the HAC (1000 mrem/h). From this point of view, it can be seen that no LOS accidents exceeded the regulatory dose rate.

Transport index affects the individual dose rate during LOS accidents. As seen in Figure 15, maximum individual dose rate at 2 meters significantly reduced over TI decreased, and finally some cases are smaller than regulatory dose rate in normal condition. Therefore, it can be inferred that minimizing the TI is an effective way to lower the dose rate for people during LOS accidents.

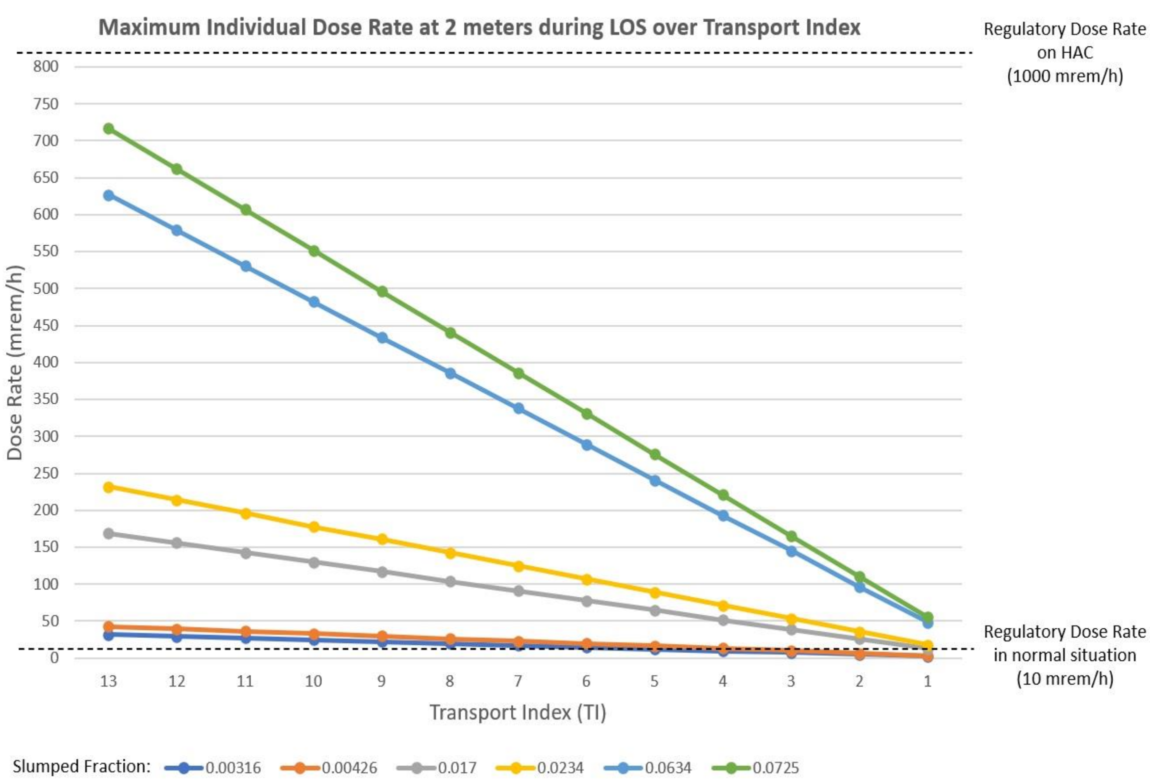

Figure 15. Maximum Individual Dose Rate at 2 meters over TI

Meanwhile, since Table 13 shows only consequences, dose risk was calculated by the product of consequences and accident frequencies using equation (5). Consequently, individual dose risk during LOS accidents when one FB is being transported through the urban area from Maine Yankee to ORNL was following Table 14, maximum 4.45E-11 mrem/h at 2 meters.

Table 14. Individual Dose Risk resulted from LOS accidents (mrem/h)

Accidents	Conditional Probability	Slumped Fraction	Average Individual Dose Risk at 1 meter (mrem/h)	Average Individual Dose Risk at 2 meters (mrem/h)	Maximum Individual Dose Risk at 2 meters (mrem/h)
Accident Rate: 1.27E-06 Length: 161km	5.96E-12	0.0725	5.17E-13	2.31E-13	8.72E-13
	1.13E-10	0.017	1.84E-12	8.28E-13	3.9E-12
	3.57E-11	0.0634	2.65E-12	1.19E-12	4.57E-12
	6.79E-10	0.0234	1.59E-11	7.16E-12	3.21E-11
	1.79E-11	0.00316	4.56E-14	2.06E-14	1.17E-13
	3.4E-10	0.00426	1.19E-12	5.35E-13	2.98E-12
Total Dose Risk			**2.21E-11**	**9.96E-12**	**4.45E-11**

4.3.3 Release of Radioactive Materials

As mentioned above, release of radioactive materials following LOS accidents only occurs when transporting uncanistered fuel using rail-lead cask. Since the degree of release is determined by the radionuclide inventory of FBs, it is necessary to determine it in advance. To figure out the kinds of radionuclides and their respective activities the ORIGEN code was used. ORIGEN code is a software that calculates time-dependent concentrations, activities, and radiation source terms for a large number of isotopes simultaneously generated or depleted by neutron transmutation, fission or radioactive decay[45]. The reason of using ORIGEN code is that this code computes isotopic compositions for LWR containing UO_2 fuel, which is the same form using in TRISO fuel, and requires uranium enrichment, thermal output, and operation time as input parameters which could be inferred to some extent at the present time. However, using ORIGEN code still has a limitation in that it uses information from the current LWR fuel assemblies. Nevertheless, it was a reasonable alternative at the time when no software for non-LWRs has not been developed yet.

Considering that FBs may be used for 1 year with an output of less than 25 MWth mentioned in Chapter 1, the input parameters could be assumed to 3 months, 6 months, and 1 year of operation with 100% power operation at 10 MWth and 20 MWth. Also, uranium enrichment could be assumed to 19.75% according to eVinci-like special purpose reactor reference parameters[48]. Since ORIGEN outputs the amount of fission products in grams, it should be converted to activity in Curie using equation (6), and Table 15 shows the determined fission product inventories of FBs right after shutdown.

Table 15. Fission product inventories of FBs right after shutdown

Radionuclide	Form	Activity (Ci)					
		10MWth			20MWth		
		3 months	6 months	1 year	3 months	6 months	1 year
^{3}H	Gas	1.265E+01	2.511E+01	5.021E+01	2.528E+01	5.022E+01	1.004E+02
^{10}Be	Particle	8.038E-08	1.608E-07	3.267E-07	1.608E-07	3.217E-07	6.521E-07
^{14}C	Gas	3.242E-06	6.483E-06	1.315E-05	6.483E-06	1.297E-05	2.63E-05
^{85}Kr	Gas	3.579E+02	7.112E+02	1.42E+03	7.155E+02	1.422E+03	2.838E+03
^{86}Rb	Volatile	6.151	6.428	6.584	1.239E+01	1.307E+01	1.368E+01
^{87}Rb	Volatile	7.884E-07	1.578E-06	3.2E-06	1.577E-06	3.155E-06	6.399E-06
^{89}Sr	Particle	2.867E+05	3.7E+05	4.011E+05	5.734E+05	7.398E+05	8.011E+05
^{90}Sr	Particle	2.829E+03	5.643E+03	1.137E+04	5.659E+03	1.128E+04	2.275E+04
^{90}Y	Particle	2.848E+03	5.657E+03	1.138E+04	5.695E+03	1.132E+04	2.278E+04
^{91}Y	Particle	3.214E+05	4.335E+05	4.855E+05	6.426E+05	8.665E+05	9.697E+05
^{93}Zr	Particle	6.028E-02	1.209E-01	2.457E-01	1.205E-01	2.419E-01	4.914E-01
^{95}Zr	Particle	3.346E+05	4.609E+05	5.266E+05	6.692E+05	9.214E+05	1.052E+06
^{94}Nb	Particle	6.391E-07	1.279E-06	2.594E+05	1.279E-06	2.558E-06	5.192E-06
^{95}Nb	Particle	1.986E+05	3.866E+05	5.266E+05	3.972E+05	7.733E+05	1.052E+06
^{95m}Nb	Particle	2.355E+03	3.239E+03	3.698E+03	4.711E+03	6.474E+03	7.388E+03
^{99}Mo	Particle	5.059E+05	5.054E+05	5.049E+05	1.011E+06	1.01E+06	1.007E+06

Table 15 (continued).

^{99}Tc	**Particle**	3.867E-01	7.928E-01	1.627	7.734E-01	1.586	3.255
^{103}Ru	**Particle**	2.09E+05	2.516E+05	2.619E+05	4.178E+05	5.03E+05	5.23E+05
^{106}Ru	**Particle**	5.094E+03	9.392E+03	1.624E+04	1.018E+04	1.879E+04	3.249E+04
^{107}Pd	**Particle**	3.587E-04	7.179E-04	1.456E-03	7.174E-04	1.436E-03	2.914E-03
^{111}Ag	**Particle**	1.648E+03	1.648E+03	1.647E+03	3.295E+03	3.293E+03	3.29E+03
^{113m}Cd	**Particle**	2.86E-01	5.693E-01	1.141	5.72E-01	1.139	2.283
^{115m}Cd	**Particle**	5.79E+01	7.216E+01	7.651E+01	1.158E+02	1.443E+02	1.528E+02
^{114m}In	**Volatile**	1.338E-06	4.362E-06	1.988E-05	3.735E-06	1.539E-05	7.753E-05
^{119m}Sn	**Particle**	2.538	4.508	7.271	5.077	9.012	1.454E+01
^{121m}Sn	**Particle**	1.543E-03	3.082E-03	6.299E-03	3.087E-03	6.164E-03	1.246E-02
^{123}Sn	**Particle**	1.406E+02	2.272E+02	3.15E+02	2.811E+02	4.544E+02	6.295E+02
^{125}Sn	**Particle**	1.122E+03	1.123E+03	1.122E+03	2.245E+03	2.246E+03	2.24E+03
^{126}Sn	**Particle**	8.115E-03	1.623E-02	3.293E-02	1.623E-02	3.247E-02	6.585E-02
^{124}Sb	**Particle**	3.039E-01	4.171E-01	4.86E-01	6.137E-01	8.527E-01	1.025
^{125}Sb	**Particle**	1.385E+02	2.785E+02	5.407E+02	2.771E+02	5.57E+02	1.081E+03
^{126}Sb	**Particle**	7.364E+01	7.409E+01	7.402E+01	1.472E+02	1.481E+02	1.478E+02
^{127}Sb	**Particle**	1.093E+04	1.093E+04	1.091E+04	2.185E+04	2.183E+04	2.179E+04
^{123m}Te	**Particle**	1.712E-06	2.742E-06	3.799E-06	3.439E-06	5.568E-06	8.101E-06
^{125m}Te	**Particle**	1.195E+01	3.826E+01	9.815E+01	2.389E+01	7.654E+01	1.964E+02
^{127}Te	**Particle**	1.002E+04	1.039E+04	1.072E+04	2.004E+04	2.077E+04	2.14E+04
^{127m}Te	**Particle**	6.233E+02	1.007E+03	1.354E+03	1.247E+03	2.014E+03	2.706E+03
^{129m}Te	**Particle**	6.65E+03	7.695E+03	7.888E+03	1.330E+04	1.538E+04	1.574E+04
^{132}Te	**Particle**	3.536E+05	3.536E+05	3.530E+05	7.072E+05	7.063E+05	7.048E+05
^{129}I	**Volatile**	5.605E-04	1.156E-03	2.391E-03	1.121E-03	2.312E-03	4.782E-03
^{131}I	**Volatile**	2.366E+05	2.365E+05	2.361E+05	4.729E+05	4.724E+05	4.712E+05
^{133}Xe	**Gas**	5.672E+05	5.668E+05	5.663E+05	1.134E+06	1.132E+06	1.13E+06
^{133m}Xe	**Gas**	1.651E+04	1.65E+04	1.649E+04	3.3E+04	3.297E+04	3.289E+04
^{134}Cs	**Volatile**	1.238	4.569	1.732E+01	4.48	1.738E+01	6.772E+01
^{135}Cs	**Volatile**	3.951E-02	7.943E-02	1.615E-01	7.701E-02	1.548E-01	3.147E-01
^{137}Cs	**Volatile**	2.913E+03	5.81E+03	1.171E+04	5.826E+03	1.162E+04	2.343E+04
^{140}Ba	**Particle**	5.207E+05	5.243E+05	5.237E+05	1.041E+06	1.047E+06	1.045E+06
^{141}Ce	**Particle**	4.193E+05	4.814E+05	4.917E+05	8.384E+05	9.62E+05	9.814E+05
^{144}Ce	**Particle**	8.979E+04	1.619E+05	2.685E+05	1.796E+05	3.237E+05	5.367E+05
^{143}Pr	**Particle**	4.821E+05	4.977E+05	4.97E+05	9.641E+05	9.944E+05	9.917E+05
^{147}Nd	**Particle**	1.905E+05	1.91E+05	1.908E+05	3.807E+05	3.816E+05	3.807E+05
^{147}Pm	**Particle**	9.905E+03	2.113E+04	4.212E+04	1.98E+04	4.224E+04	8.42E+04
^{148m}Pm	**Particle**	3.3	3.165E+01	7.704E+01	3.759E+01	1.248E+02	3.026E+02
^{147}Sm	**Volatile**	6.92E-09	3.48E-08	1.438E-07	1.385E-08	6.962E-08	2.876E-07

Table 15 (continued).

^{151}Sm	**Volatile**	6.461E+01	1.287E+02	2.553E+02	1.276E+02	2.512E+02	4.858E+02
^{152}Eu	**Volatile**	1.665E-03	1.204E-02	9.317E-02	6.589E-03	4.724E-02	3.578E-01
^{154}Eu	**Volatile**	8.559E-02	3.402E-01	1.38	3.37E-01	1.35	5.497
^{155}Eu	**Volatile**	9.323E+01	1.827E+02	3.554E+02	1.858E+02	3.63E+02	7.015E+02
^{156}Eu	**Volatile**	1.118E+03	1.165E+03	1.198E+03	2.264E+03	2.399E+03	2.529E+03
^{153}Gd	**Particle**	6.463E-06	8.532E-05	1.214E-03	5.017E-05	6.71E-04	9.435E-03
^{160}Tb	**Particle**	2.785E-03	6.106E-03	1.352E-02	7.993E-03	1.996E-02	4.89E-02
^{166m}Ho	**Particle**	4.492E-09	9.114E-09	1.901E-08	9.112E-09	1.874E-08	4.02E-08

Meanwhile, when fuel rods are damaged, fuel particles start to flow out due to the pressure difference in the form of gas, volatiles, and particles, which have its own release fractions for each accident scenario. The conditional probabilities of accident and its level of severity resulting in release of radioactive materials and their respective release fractions analyzed in NUREG-2125 are shown in Table 16.

Table 16. Input parameters for release of radioactive materials

Collision Point		**End**	**Corner**	**Side**	**Side**	**Side**	**Side**	**Corner**
Impact Speed(kph)		**193**	**193**	**193**	**193**	**145**	**145**	**145**
Seal		**Metal**	**Metal**	**Elastomer**	**Metal**	**Elastomer**	**Metal**	**Metal**
Release Fraction	**Gas**	0.8	0.8	0.8	0.8	0.8	0.8	0.8
	Particle	0.7	0.7	0.7	0.7	0.7	0.7	0.64
	Volatile	0.5	0.5	0.5	0.5	0.5	0.5	0.45
Conditional Probability		5.96E-12	3.57E-11	1.79E-11	1.79E-11	3.4E-10	3.4E-10	1.13E-10
Severity		1	2	3	4	5	6	7

When radionuclides are released by an accident, the public could be exposed through the air, and four possible exposure pathways of inhalation, cloud-shine, resuspension, and ground-shine are computed by the RADTRAN. An example of calculating population dose from cloud-shine and total population dose from four pathways are in equation (7). Finally, total population dose risk living near the route in this study was calculated using equation (8), and Table 17 shows the total population dose risk (collective dose risk) on the FB transportation scenario from Maine Yankee to ORNL through urban area using railroad resulted from the release of radioactive materials following severe accidents in Table 16.

Table 17. Collective Dose Risk from Release of Radioactive Materials (p-rem)

Risk	10 MWth			20 MWth		
	3 months	6 months	1 year	3 months	6 months	1 year
Risk$_{p\text{-}inh\text{-}u}$	1.69E-08	2.23E-08	2.78E-08	3.38E-08	4.46E-08	5.55E-08
Risk$_{p\text{-}res\text{-}u}$	1.41E-10	1.86E-10	2.32E-10	2.82E-10	3.72E-10	4.63E-10
Risk$_{p\text{-}cld\text{-}u}$	2.82E-09	3.6E-09	4.1E-09	5.64E-09	7.21E-09	8.2E-09
Risk$_{p\text{-}gnd\text{-}u}$	3.29E-08	4.26E-08	4.86E-08	6.57E-08	8.5E-08	9.71E-08
Risk$_{p\text{-}total\text{-}u}$	**5.28E-08**	**6.86E-08**	**8.07E-08**	**1.06E-07**	**1.37E-07**	**1.61E-07**

However, since FB transportation should be route independent, it is appropriate to calculate the route independent individual dose risk rather than collective dose risk in specific route. Therefore, the individual dose risk excluding route parameters is shown in Table 18, calculated using equation (9).

Table 18. Individual Dose Risk resulted from Release of Radioactive Materials (rem)

Dose Risk	10 MWth			20 MWth			SNF (NUREG-2125)
	3 months	6 months	1 year	3 months	6 months	1 year	
Individual Dose Risk	1.30E-13	1.69E-13	1.98E-13	2.61E-13	3.37E-13	3.96E-13	1.70E-14

Table 18 shows that route independent individual dose risk caused by severe rail accidents that result in release of radioactive materials was between 1.30E-13 rem and 3.96E-13 rem.

According to NUREG-2125, conditional maximally exposed individual (MEI) dose risk of SNF transportation resulted from release of radioactive materials is 1.29E-07 rem and individual dose risk is 1.70E-14 rem. Since this study used more recent accident rate data of 1.27E-06/km[21] than that of NUREG-2125, a comparison of the conditional individual dose risk excluding accident rate as in equation (**10**) was appropriate, and Table 19 shows the results. Figure 16 shows the comparison of conditional individual dose risk with that of NUREG-2125 at a glance.

Although only one case of 3 months operation with 10MWth output had a smaller dose risk than that of SNF, the dose risks from other cases were comparable to that of SNF. So, it was demonstrated that FB transportation is as safe as SNF transportation if we followed the five assumptions above.

Table 19. Conditional Individual Dose Risk (rem)

Dose Risk	10 MWth			20 MWth			SNF (NUREG-2125)
	3 months	6 months	1 year	3 months	6 months	1 year	
Conditional Individual Dose Risk	1.02E-07	1.33E-07	1.56E-07	2.05E-07	2.65E-07	3.12E-07	1.29E-07

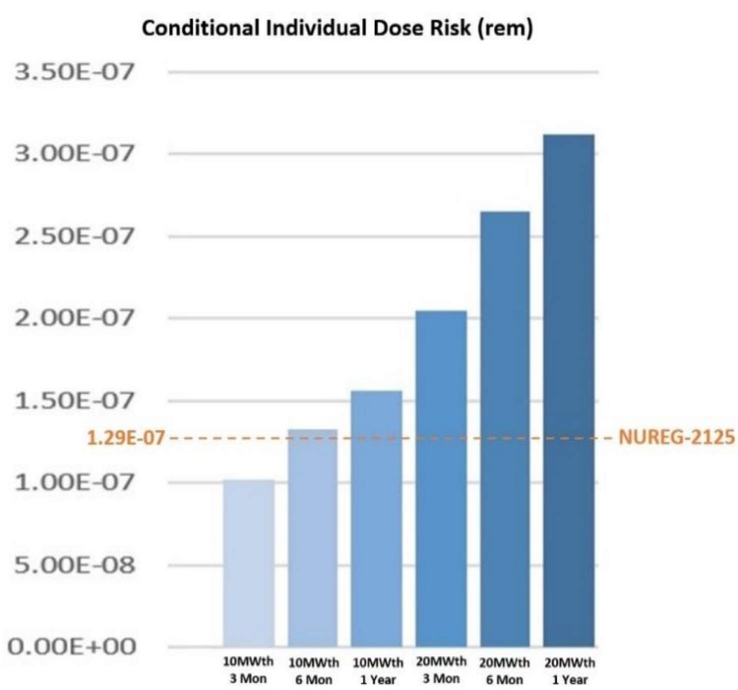

Figure 16. Comparison of Conditional Individual Dose Risk to NUREG-2125

Chapter 5. Discussion on Deploying FBs for ROK Military Applications

The ministry of national defense of the Republic of Korea (ROK) decided to use renewable energy such as solar or geothermal energy for 25% of the total military power usage by 2030 in accordance with net-zero policy of government to respond to energy depletion and climate change[49]. This is a desirable decision because the ROK military, ranked 6th in the world military power index and will now join the globalization of eco-friendly energy policy. Also, it can be seen as a change in energy sources that support national defense. In this context, in this chapter, the importance of the inversion of the energy paradigm to a new energy source in future military operations and logistical support, and why FBs should be at the center would be discussed.

5.1 Project Pele

In 2016, the United States Defense Science Board (DSB) identified critical energy challenges in future military operations. Especially, increment of energy delivery and management following the exponential growth in energy demand of Department of Defense (DOD) energy logistics made the conclusion that DOD would deploy reliable, abundant, and continuous nuclear energy power systems[50]. Consequently, the DOD's Strategic Capabilities Office (SCO) launched Project Pele in 2020. The reactor design that SCO is aiming for is 1~5 MW_e, longer than 3 years lifetime, assembled in 3 days, dissembled and transportable in 7 days, and minimally operated. This reactor design is expected to be used in off-grid military bases and state-of-the-art warfighting system such as lasers and UAVs.

In fact, a critical reason why DOD is pursuing a reactor with the above special characteristics is supported by Figure 17 that emphasizes the criticality of energy logistics in a war zone. The U.S. Military had a bitter experience that 52% of casualties occurred during fuel and water supply mission that account for 70% ~ 90% of land transport missions during Iraq War. For this reason, U.S. DOD is envisioning the outcome of Pele Project as a game changer in the future battlefield by ensuring the energy independence of the deployed forces that depend on energy supply chain, combining it with new weapons systems, and spurring development with the goal of installing a test reactor before 2027[51].

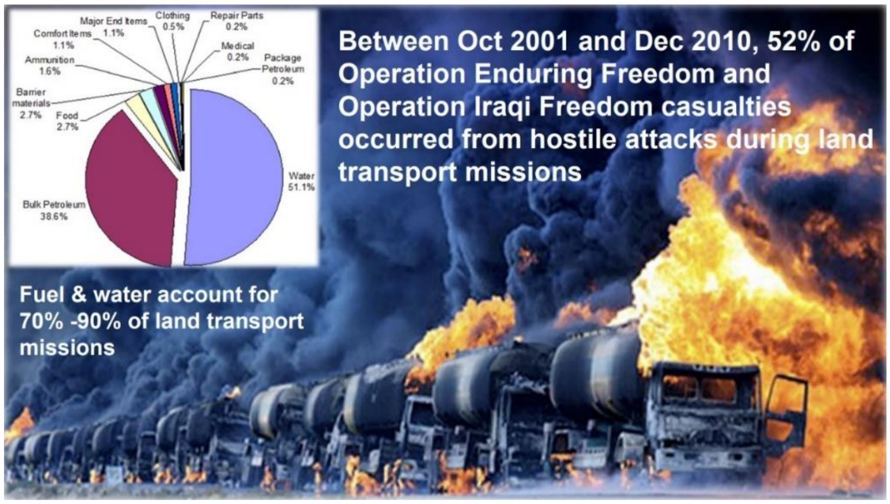

Figure 17. Criticality of Energy Logistics in a War Zone[51]

5.2 The Need of New Energy Source in the ROK Defense Environment

The ROK is the only divided country in the world and has a special security status facing a border with Democratic People’s Republic of Korea (DPRK) in a cease fire. Accordingly, about 70 years after the Korean War, the ROK has developed as a military power by confronting DPRK. It is now the 6th ranked military according to the Global Fire Power (GFP). The increase in their GFP is due to their continual maintenance of international peace by dispatching peacekeeping forces.

Meanwhile, the ROK Military is facing a new paradigm in the defense environment with the development of science and technology in a rapidly changing world. Future warfare would be transformed from a one-dimensional platform war centered on manpower in the past to a multi-dimensional network-centric war that extends beyond outer space to cyber space centered on unmanned and state-of-art weapons system. In the expanded and complicated battlefield, operational areas would be decentralized, and so each unit must have independence of operational continuity to ensure combat performance. Thus, energy supply system needs to support units that could use new defense weapons system requiring high power, carry out a stable operation independently and maintain network system with superior forces.

5.2.1 Individual Combat System (Warrior platform)

The ROK military introduced the warrior platform that can enhance individual combat performance to prepare the future warfare. Figure 18 shows the concept of warrior platform and the required energy specifications.

Figure 18. Concept and required energy of Warrior platform[52]

As the development progresses, the weight of the currently used battery has increased dramatically up to 480 kg, an impossible lifting weight following sustained exertion. For this reason, it can be seen that successful development and implementation of a light and high-output new energy source is the core of Warrior platform.

5.2.2 Independence of Operational Continuance Support

With the development of science and technology, an environment that relies on equipment rather than manpower is created, which may require much more electric power than in the past. In the current state, the wartime electricity is mainly supplied by diesel generators, and 64 drums/day and 24,000 drums/year are consumed to operate a 1 MW diesel generator for all day[33]. As mentioned above, enormous use of fuel poses major challenges in fuel transportation and storage, and wartime transportation missions may become enemy targets inflicting enormous damage. Therefore, it is necessary to secure the independence of operational continuance support that does not need to be supplied with energy sources from superior units.

5.2.3 Unmanned Combat System

In the future warfare, unmanned combat system may be widely used to prevent reckless casualties. Especially Unmanned Air Vehicles (UAVs) and Drone-bot combat system are being actively operated now, and their usefulness will only increase with the development of technology. In the ROK Army, Unmanned Combat System with Drone-bot is the core power in Army-Tiger 4.0. Figure 19 shows the missions of Drone-bot combat systems, and its support to preserve the combat power of soldiers by performing the mission as a replacement for current combatants. In unmanned systems, the key factor for effective mission execution is operational time, which depends completely on battery technology. Recently, it succeeded in flying for about 10 hours using a ‘liquid hydrogen’ fuel cell, but the longer it operates, the more effective it is, so drones using new energy sources are being actively developed[52].

Figure 19. Missions of Drone-bot combat system[52]

5.3 Deployment of FBs for ROK Military applications

As mentioned above, the ROK Military is increasingly dependent on electric power following the change of future warfare and development of future weapons systems, and so continues to study new energy sources to increase tactical utility. In this trend, the characteristics of FBs for military use compared to other energy sources are site independence, transportable size, and economically competitive.

First of all, the biggest disadvantage of renewable energy is strongly site dependent. For example, constructing 1 MW solar power plant requires about 10,000 m^2 of land, which is impossible to build on an off-grid area, in which a constant power system is desperately needed in a limited area such as General Out Post (GOP), Guard Post (GP), and Air defense base. Moreover, during wartime the Command Post (CP) should be located at the tactically best area, but site dependent energy sources may impede it by limiting to a place where energy is normally supplied such as open area for solar power.

FBs, however, do not require vast land, and can be deployed to any off-grid area, requiring sustainable energy sources. Also, it is irrespective of site, which guarantees tremendous tactical flexibility in terms of being able to dispatch and operate in any place. Therefore, CPs like in the above example, can be constructed in small locations, safe from enemy observation and bombardment. As such, the site independent FB is essential because it has high utility during both peacetime and wartime in the military.

Next, transportable size is another strong point compared to other energy sources. FBs can be easily transported as mentioned above. The easy to transport characteristic allows for the CP to tear down and set back up quickly. The capability to quickly move locations is an asset that solar energy and diesel generators do not provide. Transportable FBs also allow for modular command post installations. In addition, FBs, targeting to be used for electric vehicles, could be used for batteries of future weapons system. For example, they can be used as a battery for the unmanned systems, a power supply for warrior platform or high-energy weapon systems.

As such, the size advantage is expected to greatly contribute to operations of mobile CPs in wartime and strengthening of combat power as a power supplier to various future weapon systems. Finally, FBs will have economic competitiveness. For example, a solar power plant requires a huge size of land, and it costs 1.5 million dollars to construct a 1 MW plant[53]. Also, 24,000 drums of fuel, that is, about 4.8 million dollars, assuming the price of fuel 1 liter is $1 and 1 drums is equal to 200 liters is consumed to operate a 1 MW diesel generator for one year. Considering land prices, huge plant security personnel and costs, and damages during fuel transportation, drastic losses beyond economic value are expected.

On the other hand, as described in 1.2.3 Reactor Design, FBs that can produce around 25MWth can be built for less than $0.1B, which could mean that it would cost 4 million dollars to manufacture one FB of 1 MWth. Even though the price of a FB looks expensive now, it could be expected that the price would be lower with mass production in standardized sizes, making it affordable enough. Also, extensive use of FBs and exclusion of other costs or casualties resulted from unnecessary transport missions would guarantee beyond economic benefits. Furthermore, it may be possible to create economic profits with the export of state-of-the-art weapons system combining advanced energy sources. Therefore, it can be seen that FBs are economically competitive compared to other energy sources.

Chapter 6. Conclusions and Future Work

6.1 Conclusions

This research aims to identify regulatory challenges facing the licensing of FBs for civilian applications, consider its military application, and evaluate the risk of FB transportation using a PRA model. As discussed in the literature review, multiple regulatory gaps were identified, and regulatory supplementation was required to support the innovative development and commercialization of FBs. To solve siting challenges, the NPR approach for site restrictions and the PBR approach for excessive time burdens were suggested. Moreover, discussion on the exemption from regulations on operators through expectation on the development of advanced HSI was presented for autonomous operation.

Most importantly, it was discovered that, once complying with the regulatory dose rate, no transportation challenge exists. Accordingly, risk analysis was conducted to numerically evaluate the risk of FB transportation under current regulations using a PRA model. The PRA model was divided into incident free transportation, LOS accidents, and release of radioactive materials. As a result, individual dose during incident free transportation was at a maximum of 5.36E-04 mrem, which was significantly smaller than 10 mrem of radiation dose from a chest X-ray. The maximum and average dose rate at 2 meters and 1 meter to an emergency responder during LOS accidents were between 42.9 mrem/h and 717 mrem/h and between 17.1 mrem/h and 425 mrem/h respectively. The maximum and average dose rates were under the regulatory dose rate of 1000 mrem/h on the HAC. The individual dose risk to the public resulted from release of radioactive materials following LOS accidents were between 1.30E-13 rem and 1.70E-14 rem. For better understanding, the comparison with the risk of SNF transportation was carried out, and it was concluded that FB transportation is as safe as SNF transportation if we followed the assumptions that we set up for this study.

Finally, the discussion on the deployment of FBs to the ROK Military was presented. It was emphasized that the ROK Military is in need of new energy sources and the FB characteristics of site independence, transportable size, and economic competitiveness can meet the need for the ROK Military and enhance the combat power.

6.2 Future work

This research handled a small part of the licensing frameworks, especially, safety aspects. However, licensing issues from security and safeguards should be discussed and studied further, since FBs are pursuing universal use by ordinary people, physical security, cyber security, or safeguards could be more challengeable.

Moreover, during the risk analysis of FB transportation, fission product inventories were determined with the fuel information of LWRs, but it should be developed in accordance with the characteristics of non-LWRs.

In addition, this study only considered railroad transportation, however, further analysis for air and waterway transportation should be conducted, which may be more important transportation modes.

Finally, FBs have its own safety features such as radiation shielding and containment systems. Stipulating the use of Type B packaging for FBs may be a burden on vendors who may feel it is excessive safety measures, and it can be seen as uneconomical. Therefore, it is necessary to carry out a study on the development of dedicated packaging for FBs and figure out which packaging is more reasonable to use in terms of safe and economic use of FBs.

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APPENDICES

Appendix A. RADTRAN Input & Output file (10MWth, 1 year)

A.1 Incident-free transportation

(a) Input file

```
                 VEHICLE CHARACTERISTICS INPUT DATA
                 **********************************

VEHICLE NAME                  VEHICLE_1
MODE TYPE                      RAILWAY
EXCLUSIVE USE                   YES
DOSE RATE (mrem/hr)           1.30E+01
FRACTION OF GAMMA FOR VEH     8.90E-01
FRACTION OF NEUTRON FOR V     1.10E-01
K(0) (SQ. METERS)             1.08E+01
VEHICLE SIZE (M)              5.00E+00
EFFECTIVE SIZE (M)            4.57E+00
NUMBER OF SHIPMENTS           1.00E+00
NUMBER OF CREW                3.00E+00
CREW DISTANCE (M)             1.50E+02
CREW DOSE ADJUSTMENT FACT     5.00E-01
CREW EXPOSER WIDTH (M)        5.00E+00
EFFECTIVE EXPOSER WIDTH       4.57E+00
K(0) (SQ M) CREW EXPOSURE     1.08E+01

  VEHICLE      MATERIAL    NO.PACKAGES
VEHICLE_1
             PACKAGE_1     1.00E+00
```

(b) Output file

```
MAXIMUM INDIVIDUAL IN-TRANSIT DOSE
      VEHICLE_1   5.36E-07 REM
```

A.2 Loss of Shielding Accident

(a) Input file.1

```
LOS_SHIELD
   NPOP=1
     ACIDNT_PRB
      5.96E-12  1.13E-10  3.57E-11  6.79E-10  1.79E-11  3.4E-10
     FRAC_LOST
      0.0725  0.017  0.0634  0.0234  0.00316  0.00426
   NPOP=2
     ACIDNT_PRB
      5.96E-12  1.13E-10  3.57E-11  6.79E-10  1.79E-11  3.4E-10
     FRAC_LOST
      0.0725  0.017  0.0634  0.0234  0.00316  0.00426
   NPOP=3
     ACIDNT_PRB
      5.96E-12  1.13E-10  3.57E-11  6.79E-10  1.79E-11  3.4E-10
     FRAC_LOST
      0.0725  0.017  0.0634  0.0234  0.00316  0.00426
```

(b) Input file.2

```
                         LOSS OF SHIELDING STOP RELATED INPUT DATA
                         *****************************************
LOSS SHIELDING STOP            LOSU
VEHICLE                      VEHICLE_1
ZONE                         RURAL
NUMB PEOLPE EXPOSED          1.00E+00
MINIMUM DISTANCE(M)          1.00E+00
MAXIMUM DISTANCE(M)          8.00E+02
SHIELDING FACTOR             1.00E-01
EXPOSURE TIME (HR)           1.00E+00
```

(c) Output file

LOSS OF SHIELDING STOP = LOSU FOR LINK = LINK_U
AVERAGE RADIOLOGICAL DOSE (P_REM)

SLMP FRAC	1.0 m TO CASK	2.0 m TO CASK	3.0 m TO CASK	4.0 m TO CASK	5.0 m TO CASK	10.0 m TO CASK	20.0 m TO CASK
3.16E-03	1.25E-02	5.65E-03	3.56E-03	2.56E-03	1.99E-03	7.63E-05	1.91E-05
4.26E-03	1.71E-02	7.72E-03	4.85E-03	3.49E-03	2.70E-03	7.76E-05	1.94E-05
1.70E-02	8.00E-02	3.59E-02	2.24E-02	1.61E-02	1.24E-02	1.28E-04	3.08E-05
2.34E-02	1.15E-01	5.17E-02	3.23E-02	2.32E-02	1.79E-02	1.79E-04	4.25E-05
6.34E-02	3.64E-01	1.63E-01	1.02E-01	7.29E-02	5.63E-02	9.45E-04	2.17E-04
7.25E-02	4.25E-01	1.90E-01	1.19E-01	8.51E-02	6.57E-02	1.23E-03	2.83E-04

SLMP FRAC	50.0 m TO CASK	100.0 m TO CASK	800.0 m TO CASK
3.16E-03	3.04E-06	7.60E-07	1.18E-08
4.26E-03	3.08E-06	7.69E-07	1.19E-08
1.70E-02	4.71E-06	1.14E-06	1.64E-08
2.34E-02	6.37E-06	1.52E-06	2.09E-08
6.34E-02	3.12E-05	7.20E-06	8.86E-08
7.25E-02	4.06E-05	9.33E-06	1.14E-07

LOSS OF SHIELDING STOP ON LINK = LINK_U
MAXIMUM RADIOLOGICAL DOSE (P_REM) AT 2 M FROM CASK

SLMP FRAC	SHIELDING STOP LOCATION LOSU
3.16E-03	3.20E-02
4.26E-03	4.29E-02
1.70E-02	1.69E-01
2.34E-02	2.32E-01
6.34E-02	6.27E-01
7.25E-02	7.17E-01

A.3 Release of Radioactive Material

(a) Input file.1

ISOTOPE RELATED INPUT DATA
****

NUCLIDE	CURIES PER PKG	WASTE LIMIT (CI/M^3)	RELEASE GROUP	RESUSPENSION FACTOR
PACKAGE_1				
H3GAS	5.02E+01	4.00E+01	GAS	1.00E+00
C14GAS	1.32E-05	8.00E-01	GAS	1.00E+00
KR85	1.42E+03	7.00E+02	GAS	1.00E+00
XE133	5.66E+05	7.00E+02	GAS	1.00E+00
XE133M	1.65E+04	7.00E+02	GAS	1.00E+00
		⋮		
TB160	1.35E-02	7.00E+02	PARTICLE	1.01E+00
HO166M	1.90E-08	7.00E+02	PARTICLE	1.01E+00

NUCLIDE	HALF LIFE	GAMMA ENERGY	AIR IMMERISON (SHINE) DCF CLOUD	AIR IMMERISON (SHINE) DCF GROUND	INGESTION NUCLIDE	NEUTRON EMISSION
PACKAGE_1	(Days)	(MeV/nt)	(rem-m^3/Ci-s)	(rem-m^2/uCi-d)		neu/sec/CI
H3GAS	4.51E+03	0.00E+00	1.22E-06	0.00E+00	NONE	0.00E+00
C14GAS	2.09E+06	0.00E+00	8.29E-07	5.15E-09	NONE	0.00E+00
KR85	3.91E+03	2.21E-03	4.40E-04	8.44E-07	NONE	0.00E+00
XE133	5.25E+00	4.60E-02	5.77E-03	1.47E-05	NONE	0.00E+00
XE133M	2.19E+00	4.07E-02	5.07E-03	1.30E-05	NONE	0.00E+00
				⋮		
TB160	7.23E+01	1.12E+00	2.05E-01	3.45E-04	Tb-160	0.00E+00
HO166M	4.38E+05	1.74E+00	3.13E-01	5.43E-04	Ho-166m	0.00E+00

ISOTOPE RELATED INPUT DATA

NUCLIDE	INHALATION DOSE CONVERSION FACTORS LUNG	MARROW	THYROID	EFFECTIVE
PACKAGE_1	(REM/CI)	(REM/CI)	(REM/CI)	(REM/CI)
H3GAS	6.66E-03	6.66E-03	0.00E+00	6.66E-03
C14GAS	2.29E+01	2.29E+01	0.00E+00	2.29E+01
KR85	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE133	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE133M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
		⋮		
TB160	1.67E+05	1.04E+04	0.00E+00	2.59E+04
HO166M	2.07E+05	3.55E+04	0.00E+00	4.44E+05

(b) Input file.2

```
                     RELEASE RELATED INPUT DATA
                     **************************


                         RELEASE FRACTIONS
                     FOR EACH ACCIDENT SCENARIO

GROUP       ACCIDNT: 1 ACCIDNT: 2 ACCIDNT: 3 ACCIDNT: 4 ACCIDNT: 5 ACCIDNT: 6 ACCIDNT: 7
GAS          8.00E-01   8.00E-01   8.00E-01   8.00E-01   8.00E-01   8.00E-01   8.00E-01
PARTICLE     7.00E-01   7.00E-01   7.00E-01   7.00E-01   7.00E-01   7.00E-01   6.40E-01
VOLATILE     5.00E-01   5.00E-01   5.00E-01   5.00E-01   5.00E-01   5.00E-01   4.50E-01

                      DEPOSITION VELOCITIES
                        GROUP      M/SEC
                         GAS      0.00E+00
                        PARTICLE  1.00E-02
                        VOLATILE  1.00E-02


              ACCIDENT PROBABILITIES FOR RAILWAY   SCENARIOS

ZONE        ACCIDNT: 1 ACCIDNT: 2 ACCIDNT: 3 ACCIDNT: 4 ACCIDNT: 5 ACCIDNT: 6 ACCIDNT: 7
RURAL        0.00E+00   0.00E+00   0.00E+00   0.00E+00   0.00E+00   0.00E+00   0.00E+00
SUBURBAN     0.00E+00   0.00E+00   0.00E+00   0.00E+00   0.00E+00   0.00E+00   0.00E+00
URBAN        5.96E-12   3.57E-11   1.79E-11   1.79E-11   3.40E-10   3.40E-10   1.13E-10

                AEROSOLIZED FRACTION OF RELEASED MATERIAL


GROUP       ACCIDNT: 1 ACCIDNT: 2 ACCIDNT: 3 ACCIDNT: 4 ACCIDNT: 5 ACCIDNT: 6 ACCIDNT: 7
GAS          1.00E+00   1.00E+00   1.00E+00   1.00E+00   1.00E+00   1.00E+00   1.00E+00
PARTICLE     1.00E+00   1.00E+00   1.00E+00   1.00E+00   1.00E+00   1.00E+00   1.00E+00
VOLATILE     1.00E+00   1.00E+00   1.00E+00   1.00E+00   1.00E+00   1.00E+00   1.00E+00

            RESPIRABLE FRACTION OF AEROSOLS (BELOW 10 MICRONS AED)


GROUP       ACCIDNT: 1 ACCIDNT: 2 ACCIDNT: 3 ACCIDNT: 4 ACCIDNT: 5 ACCIDNT: 6 ACCIDNT: 7
GAS          5.00E-02   5.00E-02   5.00E-02   5.00E-02   5.00E-02   5.00E-02   5.00E-02
PARTICLE     5.00E-02   5.00E-02   5.00E-02   5.00E-02   5.00E-02   5.00E-02   5.00E-02
VOLATILE     5.00E-02   5.00E-02   5.00E-02   5.00E-02   5.00E-02   5.00E-02   5.00E-02
```

(c) Input file.3

```
               OTHER DISPERSAL ACCIDENT INPUT PARAMETERS
               *****************************************

BUILDING DOSE FACTOR                                  (BDF) = 5.000E-02
CONTAMINATION CLEAN UP LEVEL(micro-CI/M^2)(CULVL) = 2.000E-01
BREATHING RATE (M**3/SEC)                           (BRATE) = 3.300E-04
INTERDICTION THRESHOLD (Ci/micro-Ci)   (INTERDICT) = 4.000E+01
EVACUATION TIME (DAYS)                       (EVACUATION) = 1.000E+00
SURVEY INTERVAL (DAYS)                            (SURVEY) = 1.000E+01
CAMPAIGN LENGTH (YEARS)                           (TIMEYR) = 1.000E+00
FRACTION OF URBAN AREAS WITH BUILDINGS           (UBF) = 9.000E-01
FRACTION OF URBAN AREAS WITH SIDEWALKS          (USWF) = 1.000E-01
RATIO OF SIDEWALK PEDESTRIAN DENSITY              (RPD) = 6.000E+00
MAXIMUM IN-TRANSIT DOSE DISTANCE (M)         (DMDIST) = 3.000E+01
MAXIMUM IN-TRANSIT DOSE VELOCITY (KM/H)       (DMVEL) = 2.400E+01
IACC VALUE: 1=NON-DISPERSAL, 2=DISPERSAL                = 2
REGULATORY CHECK, 1=DO CHECKS, 0=NO CHECKS               = 0
BUILDING SHIELDING OPTION                          (IUOPT) = 2
    RURAL SHIELDING FACTOR                               = 1.000E+00
    SUBURBAN SHIELDING FACTOR                            = 8.700E-01
    URBAN SHIELDING FACTOR                               = 1.800E-02
```

(d) Output file

```
LINK: LINK_U             EXPECTED VALUES OF POPULATION RISK IN PERSON-REM
 MATERIAL    ISOTOPE   INHALATN    RESUSP    CLOUDSH     GROUND     TOTAL
 PACKAGE_1     H3GAS   1.84E-19  0.00E+00  2.05E-18  0.00E+00  2.23E-18
 PACKAGE_1    C14GAS   1.66E-22  0.00E+00  3.64E-25  0.00E+00  1.66E-22
 PACKAGE_1      KR85   0.00E+00  0.00E+00  2.09E-14  0.00E+00  2.09E-14
 PACKAGE_1     XE133   0.00E+00  0.00E+00  1.09E-10  0.00E+00  1.09E-10
 PACKAGE_1    XE133M   0.00E+00  0.00E+00  2.79E-12  0.00E+00  2.79E-12
 PACKAGE_1      RB86   4.32E-15  3.60E-17  1.35E-15  1.58E-14  2.15E-14
 PACKAGE_1      RB87   1.13E-21  9.42E-24  2.49E-25  7.36E-24  1.15E-21
                                    ⋮
PACKAGE_1     ND147   3.96E-10  3.31E-12  7.08E-11  9.42E-10  1.41E-09
PACKAGE_1     PM147   2.08E-10  1.74E-12  1.75E-15  5.27E-14  2.10E-10
PACKAGE_1    PM148M   3.89E-13  3.25E-15  4.47E-13  5.50E-12  6.34E-12
PACKAGE_1     GD153   2.52E-18  2.11E-20  2.69E-19  4.72E-18  7.53E-18
PACKAGE_1     TB160   9.36E-17  7.82E-19  4.49E-17  5.33E-16  6.72E-16
PACKAGE_1    HO166M   2.26E-21  1.88E-23  9.64E-23  1.18E-21  3.56E-21
                                                       TOTAL:  8.07E-08
```

Appendix B. ORIGEN Output file (10MWth, 1 year)

```
   *   IRRADIATION OF FE
+                                                                                   FISSION PRODUCTS
    POWER= 1.00000E+01 MW,  BURNUP= 3.65000E+03 MWD,  FLUX= 2.16E+11 N/CM**2-SEC
0                    5 SUMMARY TABLE:  CONCENTRATIONS, GRAMS
                          CALCULATION IS BASED ON 1 FE
          Initial   Irr_1      Dec_3
 H003      0.000E+00 5.201E-03 5.201E-03
 LI006     0.000E+00 4.846E-05 4.846E-05
 LI007     0.000E+00 1.134E-06 1.134E-06
 BE009     0.000E+00 2.188E-06 2.188E-06
 BE010     0.000E+00 1.459E-05 1.459E-05
 C014      0.000E+00 2.949E-06 2.949E-06
 ZN066     0.000E+00 4.965E-11 4.965E-11
 GA071     0.000E+00 4.180E-09 4.180E-09
 NI072     0.000E+00 1.300E-11 1.300E-11
 CU072     0.000E+00 7.126E-11 7.126E-11
 ZN072     0.000E+00 2.214E-06 2.214E-06
 GA072     0.000E+00 6.719E-07 6.719E-07
 GE072     0.000E+00 2.871E-04 2.871E-04
 CU073     0.000E+00 9.793E-11 9.793E-11
 ZN073     0.000E+00 1.302E-09 1.302E-09
 GA073     0.000E+00 1.002E-06 1.002E-06
 GE073     0.000E+00 1.249E-03 1.249E-03
 GE073M    0.000E+00 3.023E-11 3.023E-11
```

⋮

```
TB162     0.000E+00 1.509E-08 1.509E-08
TB162M    0.000E+00 6.945E-09 6.945E-09
DY162     0.000E+00 7.576E-04 7.576E-04
EU163     0.000E+00 5.927E-11 5.927E-11
GD163     0.000E+00 1.063E-09 1.063E-09
TB163     0.000E+00 1.409E-08 1.409E-08
DY163     0.000E+00 2.632E-04 2.632E-04
GD164     0.000E+00 3.731E-09 3.731E-09
TB164     0.000E+00 6.089E-10 6.089E-10
DY164     0.000E+00 7.348E-05 7.348E-05
GD165     0.000E+00 1.066E-10 1.066E-10
TB165     0.000E+00 5.324E-11 5.324E-11
DY165     0.000E+00 1.453E-08 1.453E-08
DY165M    0.000E+00 6.614E-11 6.614E-11
HO165     0.000E+00 3.724E-05 3.724E-05
DY166     0.000E+00 2.183E-07 2.183E-07
HO166     0.000E+00 7.194E-08 7.194E-08
HO166M    0.000E+00 1.059E-08 1.059E-08
```