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Lower Bound of Dose Constraints in Radiological Protection of the Public Taking into Consideration Dose Criteria for Exemption

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The International Atomic Energy Agency (IAEA) has published a safety guide, RS-G-1.7, that gives specific values of activity concentration for radionuclides of artificial origin that may be used for bulk amounts of material for the purpose of applying exemption. The primary radiological basis for establishing these values is that the effective doses received by individuals should be of the order of 0.01 mSv or less per year. On the other hand, the International Commission on Radiological Protection (ICRP) has published the new concept of a representative person in Publication 101. This representative person is a hypothetical person exposed to a dose that is representative of the most highly exposed persons in the population. On the basis of the new concept of the ICRP, it is reasonable that, theoretically, the 95th percentile of the distribution of the dose received by the population is lower than the dose constraint, which indicates that the main part of the dose distribution is considerably lower than the dose constraint. Taking into consideration the rationale of IAEA/RS-G-1.7 and the new recommendations for probabilistic dose assessment in ICRP/Pub.101, it is possible to propose that the minimum dose constraint for the optimization of radiological protection of the public should be set to 0.1 mSv/y.

KEYWORDS: radiation protection, public, probabilistic distribution, optimization, dose criterion, dose constraint, exemption, clearance

I. Introduction

According to the Basic Safety Standards (BSS)[1] published in 1996, in the case of artificial radionuclides, the basic dose criterion for the exemption and clearance of exposures or materials is of the order of 0.01 mSv/y. The experience gained is increasingly showing that such a low-dose criterion is often not practical. Moreover, it is widely known and recognized that the application of such a low-dose criterion for exposure from natural radioactivity or for naturally occurring radioactive material (NORM) is also often not practicable. For example, recently, some reports have proposed a dose criterion of 0.3 mSv/y (as a measure to exclude or exempt exposure or materials) for industries or activities that generate NORM.[2]

Also, recently, for the purpose of providing guidance to national authorities, including regulatory bodies and operating organizations on the application of the concepts of exclusion, exemption and clearance as established in the BSS, the International Atomic Energy Agency (IAEA) published safety guide, RS-G-1.7[3], which specifies values of activity concentrations for radionuclides of both natural origin and artificial origin that can be founded in bulk materials. For artificial origins, the primary radiological basis for establishing the activity concentrations is that the effective doses to individuals should be of the order of 0.01 mSv or less per year.

On the other hand, the International Commission on Radiological Protection (ICRP) released drafts of new recommendations in June 2004, June 2006 and January 2007 and finally approved a new set of fundamental recommendations on the protection of humans and the environment from ionizing radiation at its meeting in Essen, Germany, 19-21 March 2007. In the new ICRP recommendations, dose constraint is considered an effective tool for the optimization of radiological protection. In the completion of the new ICRP recommendations, the numerical value for the minimum dose constraint for the public remains undetermined.

In this paper, a numerical value for the minimum dose constraint, which is required for the optimization of radiological protection, is proposed, taking into consideration the fundamental concept used in the exclusion, exemption and clearance criteria and the practical influences of the minimum dose constraint on individual effective dose of the public.

II. Dose Criteria of Exclusion, Exemption and Clearance

Two different approaches were developed to establish the activity concentrations given in RS-G-1.7.[4] The first approach applies the concept of exclusion to derive suitable activity concentrations for radionuclides of natural origin. These activity concentrations were selected on the basis of the consideration of the upper bound of the worldwide distribution of soil activity concentrations reported by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR).[5] RS-G-1.7 also states that doses to individuals as a consequence of these activity concentrations would be unlikely to exceed about 1 mSv per

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year, excluding the contribution from the emanation of radon, which is dealt with separately in the BSS. On the other hand, the second approach uses the concept of exemption in order to derive activity concentrations for radionuclides of artificial origin. The primary radiological basis for establishing these activity concentrations is that the effective doses to individuals should be of the order of 0.01 mSv or less per year. To take into account, among other things, of the diversity in the level of control and end use of the diverse situations, this IAEA guideline (RS-G-1.7) includes provisions for activity concentrations that can be up to about ten times higher (or of the order of 0.1 mSv/y: 10 x 0.01 mSv/y) and even up to 1 mSv/y in the case of low-probability events.

As shown above, in seeking greater adaptability to the practical situations encountered, the dose criteria used for exclusion, exemption and clearance have gradually evolved. It follows from this that one key outstanding issue of the radiological protection system is the need for greater coherence and consistency in addressing very low dose situations (i.e., about 1 mSv/y or lower).

III. New Concepts in ICRP

1. Representative Person

The ICRP published a new concept of the representative person in Publication 101 Part I[6]. The representative person is a hypothetical person exposed to a dose that is representative of those of highly exposed persons in a population. In a probabilistic dose assessment, the ICRP recommends that the representative person should be defined such that the probability of exposure occurrence is lower than about 5% that of a person randomly selected from the population receiving a high dose. This concept is used for compliance with the relevant dose constraint. In other words, this recommendation on the representative person can be redefined as the requirement that the 95th percentile of the probability distribution of the assessed individual dose should be lower than the relevant dose constraint.

This concept of the ICRP defines how the dose constraint should be complied with using a probabilistic approach, in addition to the previous approach of compliance with the dose constraint, namely, the deterministic approach, which is based on conservative and simple dose assessments. In other words, the magnitude of the conservativeness of the dose assessment could be mathematically and more accurately accounted for by the concept of the representative person.

When using the concept of the representative person, the following equation theoretically shows that the 95th percentile of the dose distribution for the population is lower than the dose constraint:

$$\text{DC} > \text{95th percentile}, \tag{1}$$

where DC is the dose constraint (mSv/y) and the 95th percentile is that of the dose distribution for the population (mSv/y). In the case of a lognormal distribution, the 95th percentile can be obtained from the geometric mean (GM) and geometric standard deviation (GSD) as

$$\text{95th percentile} = \text{GM x GSD}^{1.645}, \tag{2}$$

Thorough investigations of the actual distributions of external and internal doses due to artificial origin are rather rare or nonexistent. Thus, there is no solid ground for seeking a relevant case study duly representative of the actual public dose distribution on a worldwide basis. On the other hand, experience shows that the dose distributions of radiation workers in nuclear facilities tend to be similar to lognormal distributions[7], which is the same as that in the case of natural background radiation. For this reason, it would be appropriate that the dose distribution of the public to man-made radiation is assumed to be lognormal.

From the above results, the maximum dose distribution of the public can be derived from equations (1) and (2), on the basis of the ICRP concept of the representative person. As an example, the dose distributions based on a dose constraint of 0.6 mSv/y are shown in **Fig. 1**. **Fig. 1** shows that at least 95% of the dose distribution is in the region lower than the dose constraint. It should be noted that the dose to the representative person, to which the dose constraint is compared for compliance, is for more highly exposed people in the population, and that, consequently, the dose actually received by the public in general would be considerably lower.

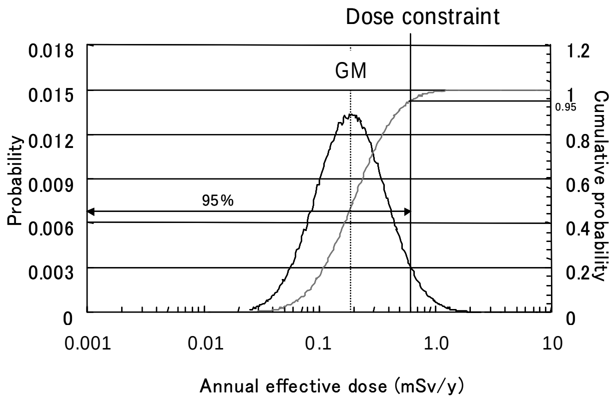


Fig. 1 Relationship between dose constraint and 95th percentile of the dose distribution.

2. Minimum Dose Constraint

Before approving a new set of fundamental recommendations on the protection of humans and the environment against ionizing radiation at the ICRP main commission held in Essen in March 2007, the ICRP released draft reports of its new set of recommendations in June 2004, June 2006 and January 2007 and openly discussed, with many experts, radiation protection through a website consultation system. In the 2004 draft report, the minimum dose constraint was described to be 0.01 mSv/y. In the subsequent 2006 draft reports, this description was removed, but a similar description was noted in the 2007 draft report as an expression of the range of dose constraint with a minimum value of 0.01 mSv/y. Although it has not been officially confirmed yet whether the description has been deleted in the final approved set of recommendations to be published in the upcoming autumn, the numerical value for the minimum dose constraint for the public will be deleted. It is important, for developing a sound radiation protection system and designing the scope of radiological protection, to determine an appropriate minimum dose constraint.

The dose constraint is an effective tool for the optimization of radiological protection. If the minimum dose constraint is determined, no one need to set the dose constraint lower than the minimum dose constraint, which leads to an understanding that the minimum dose constraint is equivalent to the lower bound of optimization in radiological protection.

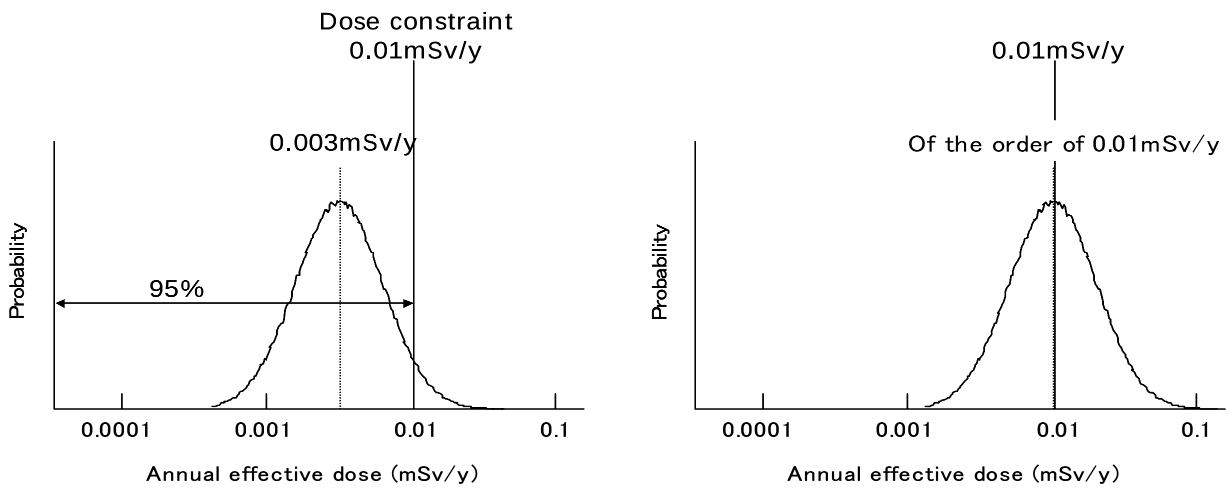
IV. Minimum Dose Constraint

As mentioned above, the ICRP proposed 0.01 mSv/y as a minimum dose constraint in the previous draft recommendations. As an example, dose distribution for the public, setting 0.01 mSv/y as a dose constraint, is compared with the dose criterion of exemption. The result is shown in **Fig. 2**, assuming that the dose distribution is lognormal and the GSD is 2.0. It can be recognized that setting 0.01 mSv/y as a dose constraint leads to a request for further dose reduction to make the public dose lower than the exempted dose region. It is obvious that 0.01 mSv/y is too low and strict as a minimum dose constraint. It should be reminded that the dose to the representative person is for more highly exposed people in the population, and that, consequently, the dose actually received by the public in general would be considerably lower.

In addition to the above result, to discuss how the numerical value for the minimum dose constraint should be set in the radiological protection system, a rationale and the theoretical background of exclusion, exemption and clearance are needed; it is agreed to adopt the IAEA safety guide, RS-G-1.7, and it might be considered to be adopted in the revised BSS. In RS-G-1.7, the exclusion criterion for natural origins has been approximately 1 mSv/y. Exemption criteria for artificial origins have been categorized into two types, the order of 0.01 mSv/y for normal situations and 1 mSv/y in the case of low-probability events. It should also be reminded that RS-G-1.7 permits member states to exceed the relevant activity concentrations for natural and artificial origins by up to ten times (or of the order of 0.1 mSv/y: 10 x 0.01 mSv/y), because regulatory bodies' decisions will depend on the nature of the national regulatory infrastructure.

From the above discussions, it has been clarified that exemption criteria are flexible by up to 1 mSv/y according to the probability of the occurrence of the exposure scenarios. Since the representative person is defined as the 95th percentile of the distribution of the dose received by the population, 1 mSv/y could be the minimum dose constraint. However, 1 mSv/y corresponds to the individual-related dose limit for the public, which serves as the upper bound of the source-related dose constraint. To maintain a balance in the radiological protection system, it is obvious that 1 mSv/y is too high as the minimum dose constraint.

To conclude the above discussions, 0.01 mSv/y is too low and strict and 1 mSv/y is too high as the minimum dose constraint. To make a reasonable radiological protection system, it can be proposed that the minimum dose constraint for the optimization of radiological protection of the public should be set at 0.1 mSv/y.



a) Dose constraint 0.01mSv/y

b) Dose criteria for exemption and clearance

Fig. 2 Comparison between setting 0.01mSv/y as a dose constraint and as an exemption criterion.

V. Conclusion

Dose criteria for exclusion, exemption and clearance adopted in the IAEA safety guide of RS-G-1.7 have been reviewed. A dose distribution of the public has been constructed using new ICRP concepts for a representative person with the dose constraint set at 0.01 mSv/y. As a result of comparison of the dose distribution and dose criteria of exemption and clearance, it has been clarified that 0.01 mSv/y is too low and strict as a minimum dose constraint. Taking into consideration the rationale and theoretical background of dose criteria of exclusion, exemption and clearance presented in RS-G-1.7, it can be proposed that the minimum dose constraint for the optimization of radiological protection of the public should be set to 0.1 mSv/y.

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Activities on Improvement of Radiation Monitoring and Emergency Response Systems in Russian Regions

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Preparedness of the emergency response system to prevent and mitigate consequences of radiation incidents and accidents is one of the most important elements of safe operation of nuclear power facilities. Everyday activities on prevention of emergency situations along with adequate and efficient response are the key factors, which reduce the risks of harmful impact on the population and environment. Both the high engineering level and variety of the nuclear branch facilities impose special requirements to the emergency response system. A powerful scientific and technical support system is required to solve various problems of emergency response.

KEYWORDS: emergency response, radiation monitoring, neutron spectrum, NSS decommissioning

I. Introduction

The emergency response system established in the Russian nuclear branch over the past decade utilizes the scientific potential, expertise and the accumulated experience of participating organizations to provide efficient response.

Decision-making on protection measures for personnel, population and environment in a case of an emergency is based on two main elements. The first one is related to identification of the radiation situation, including environmental measurements, assessment and forecast of accident evolution, environmental contamination and impacts on personnel and population. The second one is planning and evaluation of the efficiency of protective measures and their optimization for the specific conditions with due regard for radiological, economic and social conditions.

Thus, emergency monitoring and assessment of radiation consequences are two important components of the emergency response system, which are differently realized in the nuclear industry and at the regional level of response to radiation accidents.

II. Emergency Response System of Russia

Several factors are required for safe and reliable operation of the nuclear- and radiation-hazardous facilities in accordance with the Russian regulatory documents. These include:

- Assigning Scientific Advisor, Chief Designer, and Chief Planner to each of the facilities constructed or technologies developed;
- Ensuring implementation of regulatory, technical and operational documents and supervision over execution of these requirements;
- Monitoring of technological processes and safety parameters, including environmental safety;
- Qualification of the personnel and safety standards;
- Preparedness of the emergency response system to liquidation of accidents.

Thus the emergency response system is the basic requirement for safe and sustainable development of nuclear energy.

In 1994 a unified Emergency Response System has been established in Russia for management of various emergency situations (See **Fig. 1**).

Emergency response system consists of two components: functional subsystems which include emergency response systems of federal ministries, agencies, services such as Ministry of Emergency Situations, Ministry of Health, Ministry of Agriculture, State Hydrometeorology Service etc. and Territorial subsystems, which include regional, local systems of all 89 Russian regions.

Ministry of Emergency Situations co-ordinates the actions of all subsystems.

The main functional system exists within the Federal Agency for Atomic Energy (Rosatom), which controls all Russian nuclear facilities including 10 nuclear power plants (31 units). This system ensures efficient emergency response at the federal, branch and facility levels in case of a nuclear or radiation accident.

Organization of the professional emergency service was started by Rosatom in 1993 in order to raise the preparedness of the nuclear industry to prevent and mitigate radiological accidents and incidents (See **Fig. 2**). Currently, the service includes 5 emergency technical centers, 7 special emergency teams and 20 special rescue brigades.

Coordination body of the functional subsystem is the Rosatom Commission for prevention and mitigation of emergency situations and provision of fire safety.

Territorial subsystems were established in the Russian regions for management of various emergencies.

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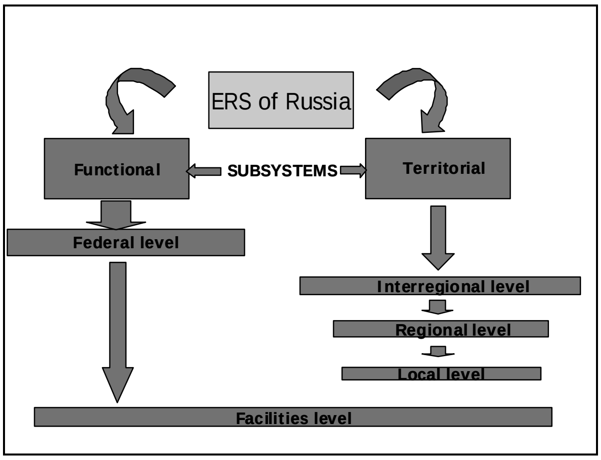

Fig. 1 Emergency Response System of Russia

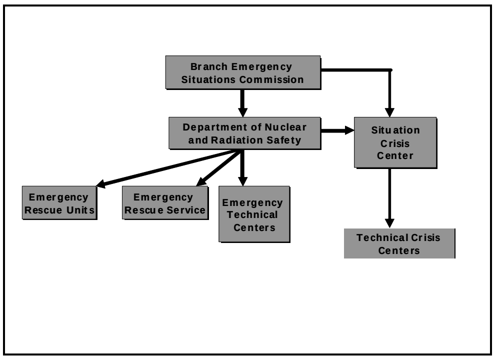

Fig. 2 Emergency Response System of Rosatom

III. Scientific and Technical Support of Emergency Response

There is a comprehensive system of technical and scientific support of emergency response authorities and decision makers.

Assessing and forecasting an accident evolution, impact on personnel, population and environment as well as elaboration of recommendations on counter-emergency and protective measures are important components of the overall emergency response system. Therefore, the development of information-simulation and expert systems for decision-making and scientific and technical support of measures on population and territory protection in case of radiation emergency was started at various nuclear scientific institutes.

Such system was developed at Nuclear Safety Institute of the Russian Academy of Sciences (IBRAE). This work was based on the wide practical experience of the leading IBRAE specialists in the field of nuclear physics, physics of reactors, radiation safety, radiation protection and radioecology gained in the process of assessment and mitigation of the consequences of accidents and incidents at the Chernobyl NPP (1986), Siberian Chemical Combine (1993), Production Association "Mayak" (1957), nuclear weapons testing grounds and nuclear submarine accident in Chazhma Bay of Primorskiy Krai (1985).

This experience was used to establish the Technical Crisis Center (TCC) at IBRAE. The main goals of TCC are as follows:

- Assessment and forecast of the main characteristics of radioactive release sources in accidents and incidents;
- Forecast of environmental contamination taking into account the radiation monitoring data;
- Assessment and forecast of population exposure doses;
- Development of recommendations on population and environmental protection;
- Assessment of the effectiveness of protective measures and their optimization for specific situation, taking into account radiological, economic and social conditions.

To date, the hardware, software and procedures developed by IBRAE includes the following components:

- Databases of radiation-hazardous facilities and regions of their location, including data on population, environmental components and infrastructure;
- Databases of radiological scenarios of potential accidents;
- Computer systems for simulation of radionuclide spreading in the atmosphere;
- Computer models of radionuclide migration in soils and water systems;
- Computer models of radionuclide migration via the soil – plant – animal – human food chain;
- Software for estimation of internal and external exposure doses; and
- Cartographic databank and Geographical Information System (GIS) for sites of radiation-hazardous facilities and regions of their location.

The main task of TCC is a fast response to requests on assessment of possible consequences capable of incidents that cause radioactive contamination of the environment or/and exposure of the population. Every year, TCC experts process about 30 requests from the Crisis Situation Management Center, Rosatom Situation Crisis Center and Rosenergoatom Emergency Crisis Center, e.g. assessment of potential consequences for radiation-hazardous facilities of the earthquake in Turkey; assessment of radionuclide release and population exposure doses caused by the accident at nuclear fuel fabrication plant in Tokaimura (Japan); a forecast of potential radiological consequences of destruction of some NPPs and research reactors during bombing in Yugoslavia, etc.

The level of TCC preparedness is maintained by participation of TCC experts in drills and exercises. Since 1996, TCC took part in 15 large-scale national and international exercises, including exercises at all ten Russian nuclear power plants, 2 national exercises at Saclay Nuclear Research Center (France) in 1996 and in 2000, international exercise at Armenian NPP, etc. During these exercises TCC experts performed real-time assessment of the radiological situation on the basis of input information and proposed recommendations on protection of population and environment.

New approaches and up-to-date technologies are used in development of scenarios and conduct of exercises with IBRAE TCC participation. For example, a new procedure of full-scale simulation of radiological consequences of accidents has been recently developed using geoinfomation technologies.

IV. Development of Regional Emergency Response System

In Russia the regional authorities are responsible for the population protection. Therefore, the regional emergence response systems should include up-to-date tools for radiation monitoring and modern emergency infrastructure. That's why the projects of development of radiation monitoring and emergency response systems were included in the Federal State Program "Nuclear and Radiation Safety of Russia". The pilot project was started in November 2005 in the Murmansk Region of Russia in the framework of Multilateral Nuclear Environmental Program in the Russian Federation.

This Program was launched in 2004. The Agreement is aimed at the practical international cooperation in the field of safe management of spent nuclear fuel and radioactive waste in Northwest Russia. The agreement opens new possibilities in resolving such vital tasks as decommissioning of retired nuclear submarines and nuclear maintenance vessels, establishment of infrastructure of safe storage of spent fuel and wastes, and environmental remediation of the territory of former naval costal maintenance bases.

In December 2004 the Assembly of donor countries from Europe, Russia and Canada made decision to fund top-priority projects, including the Project on enhancement of radiation monitoring and emergency response system of the Murmansk Region of Russia. IBRAE was selected as a main contractor.

Large scale activities aimed at decommissioning of a large number of radiation-hazardous facilities of the Navy are being implemented in the Murmansk Region of Russia throughout the last decade. An important part of these activities is nuclear, radiation, and environmental safety assurance. One of the key elements of the system for safe decommissioning of radiation-hazardous facilities is preparedness to response to possible radiological accidents. Therefore, a modern response system in the Murmansk Region is a necessary factor ensuring protection of population and territories in case of radiological accidents at facilities involved in nuclear submarine decommissioning, spent nuclear fuel and radioactive waste management.

The employer of the works is the Government of the Murmansk Region. The Project duration is expected to be 2 years.

The main objective of the Project is overall enhancement of the system for radiation monitoring and emergency response to potential accidents at radiation-hazardous facilities involved in nuclear submarines decommissioning and management of spent nuclear fuels and radioactive wastes in the Murmansk Region.

The Project is aimed at enhancing the preparedness of the emergency response forces, minimization of the consequences of possible radiological accidents, increasing the effectiveness and efficiency of decision-making and realization of population and environment protection measures.

The Project is unique in Russia in terms of the covered territory, the number of radiation hazardous facilities and facilities involved in international and Russian programs aimed at NS decommissioning and management of SNF and RW.

The implementation of the Project will provide the Murmansk Region with an up-to-date systems of radiation monitoring, informational, analytical and real-time expert support of executive authorities in planning and implementation of adequate protection measures in case of radiation accidents

The main directions of works under the Project are the following:

- Modernization of the existing and setting-up new facility and territorial automatic radiation monitoring systems, including mobile radiation surveillance kits;
- Establishment of the Regional Crisis Centre of the region and the Crisis Centre of nuclear and radiation hazardous facilities (See **Fig. 3**);
- Setting up communication systems for transfer, acquisition, processing, storage and presentation of data for participants of emergency response at the facility, regional and federal levels;
- Development of software and hardware systems for expert support of decision-making on personnel, population and environment protection activities.

The Regional Crisis Centre (RCC) was established to provide informational and technical support for decision-making on protection of population and territories in case of emergency situations at nuclear and radiation-hazardous facilities. Everyday activities of RCC include on-line monitoring of the radiation situation in the territory of the region, planning and verification of actions aimed at prevention of ES in the territory of the Region.

TCC of IBRAE provides expert support to the personnel of RCC and development of recommendations on minimization of the radiological consequences for personnel, population and territories of the Region. It also provides scientific, methodological, and technical support of actions aimed at assuring preparedness of emergency response forces, including participation in exercises and training.

The territorial automatic radiation monitoring system comprises of about 100 stations for dose rate, air and water contamination measurements as well as several weather stations. Territorial and facility systems provide on-line information about the radiation situation in the Murmansk Region, and provide information for regional and federal authorities, as well as for the population.

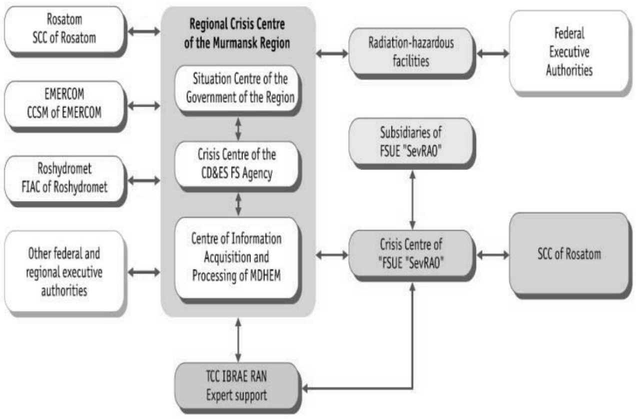

Fig. 3 Emergency Response System of Murmansk Region

V. Conclusions

Preparedness of the emergency response system to prevention of radiation incidents and accidents and mitigation of their consequences is one of the most important elements of ensuring safe operation of nuclear power facilities.

Everyday activities on prevention of emergency situations along with adequate and efficient response are the key factors reducing the risks of harmful impact on the population and environment.

Up-to-date emergency response system of regional and local authorities forms the basis for minimization of consequences of possible radiation accidents.

Acknowledgement

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Warning of and Emergency Response to Nuclear and Radiological Terrorist Incidents

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This paper introduces mainly the research status and developing focuses of warning of and emergency response to nuclear and radiological terrorist incidents and discusses the system and technology of warning and emergency response to nuclear and radiological terrorist incident. The system of warning and response of nuclear and radiation incident is comprised of information acquirement and management, analysis and decision-making, and emergency response. The system is based on threat evaluation. The content and technology of the system is analyzed in the paper, in particular the importance of information flow, such as information collection, estimation, release and so on.

KEYWORDS: nuclear incident, warning system, terrorist incident, emergency response

I. **Introduction**

Prevention and disposition of terrorism have been paid attention great by governments and the public since Sept. 11th terrorist incident in U.S. Prevention and disposition of nuclear and radiological terrorist incidents is important part of preventing and disposing possible terrorist events. U.S.A and international organizations such as IAEA adopt the measures to prevent and respond nuclear and radiological terrorist incidents from beginning to end, according to time order from detecting potential danger to real occurrence of terrorism. The measures include information and surveillance, prevention, defense (or protection), determent (or crisis manage), response and resuming (or consequence management), ascription.

China has begun to consider nuclear and radiological terrorist incidents since Sept. 11th terrorist incident. National Counter-Terrorism Office issued Nuclear and Radiological terrorist Response Plan in 2002. The State Council of China issued National Critical Incident Collectivity Response Plan in January 2006, which established society warning system to cope with natural calamity, accident disaster, public health incident, society safety and so on. With increasing rampancy of international and internal terrorist incidents, nuclear and radiological terrorist incidents would be taken a possible form by terrorist. A credible and effective warning system is important in preventing occurrence of terrorist incidents, mitigating consequence and losses, and safeguarding people's life and wealth.

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II. Threat of Nuclear and Radiological Terrorist Incidents

Nuclear and radiological terrorist incidents are divided into[1)]:

- Illegal obtaining and spiteful dispersion of radioactive materials, or using devices containing radioactive material to disperse radioactive materials with explosive mode (RDD);
- Illegally obtaining nuclear material to make improvised nuclear device (IND), and using or threateningly using improvised nuclear device;
- Attacking important nuclear facilities, such as a nuclear power plant, research reactor, storing facilities of spent fuel or high level radioactive waste liquid, and so on.

After Sept. 11th incident, the possibility of attacking international society by nuclear or radiological terrorists rapidly increases. The analysis[1)] indicates that possible attacking methods would be stealing nuclear weapon, and using improvised nuclear device (IND) and radiological dispersion device (RDD). The acquisition of radiation sources is relatively easy, while the attacking by RDD is most possible.

According to the estimation of IAEA, millions of radioactive sources had been used all over the world, hundred thousands of radioactive sources are used now[1)]. Though most of unused radioactive sources have been held and managed, some radioactive sources actually is uncontrolled or half uncontrolled, which is named Orphan Sources. According to the estimation of American Nuclear Regulatory commission (NRC)[1)], about two hundred radioactive sources uncontrol every year in American. About seventy radioactive sources uncontrol every year in E.U.. Total number of radiation sources are 70,000 ~ 80,000 in mainland of China, unused sources are 25,000 or so. Uncontrolled radioactive sources are more than 2000 in Chinese mainland from 1988 to1998[1)].

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III. Warning and Emergency Respond System of Nuclear and Radiological Terrorist Incidents

1. Focus of Warning and Emergency Response System

Warning and emergency response system of nuclear and radiological terrorist incidents should include two parts:

- One is to strengthen security management of nuclear facility, radiation source and nuclear material to prevent occurrence of nuclear and radiological terrorist incidents from the fountainhead. To strengthen nuclear facility safety, the first point is to determine Design Basis Threat (DBT). The function of DBT is to enhance nuclear safety to prevent inside and outside destruction or collusion. Mostly implementing the measure of safeguard of nuclear materials is controlling and physical protection of nuclear material. For radioactive source, surveillance management should be carried out for whole process[2, 3].
- The other is to increase the capability of detecting radioactive materials to avoid the occurrence of nuclear and radiological terrorist incidents, including border radiation monitoring, internal radiation monitoring and nuclide identification. To strengthen the monitoring capability for radioactive materials, a radiation monitoring system should be established which covers areas radioactive materials appear possibly.

2. Characteristic of Warning and Emergency Response System

The key features of warning and emergency response system for nuclear and radiological terrorist incidents are sensitivity, timeliness, availability, operability, and extensibility.

- Sensitivity is to discover the foreboding of nuclear and radiological terrorist incidents as more as possible;
- Timeliness is to discover occurrence of nuclear and radiological terrorist incidents as early as possible, help win time for start-up measures. Timeliness exists in information collection, transfers, analysis, release, survey and adopting measures.
- Availability means multi-angled and all-dimensional information collection, to make forecast as accurately as possible, to avoid unnecessary measure start-up.
- Operability means that the system accords with real conditions of different areas, which implements easily on the basis of personnel and resources.
- Extensibility means that the system is of adjustable ability, and it has abundant spaces to add and reduce target.

3. Frame of Warning and Emergency Response System

Warning and emergency response system of nuclear and radiological terrorist incidents is a complex system which includes many factors. The flow of the system is from monitoring and information collection, verification and analysis, evaluation and decision to alarm. The system consists of acquisition of information, analysis and decision-making, and emergency response [4].

IV. Acquisition, Management and Analysis of Nuclear and Radiation Warning Information

1. Acquisition of Nuclear and Radiation Warning Information

1) Acquisition of Information.

The information comes from the public and organizations, especially from police agencies.

2) Acquisition of Nationwide Radiation Warning Monitoring Information.

The nationwide radiation warning monitoring information can be acquired from the national radiation warning monitoring information system (which can be compatible with other national radiation monitoring systems), and it is composed of border radiation monitoring and inland radiation monitoring

3) Monitoring Data for Nuclear Facility and Nuclear Activity

Nuclear facility monitoring data come from the routine radiation monitoring system and physical protection monitoring/controlling system for the nuclear facility and nuclear activity

Relevant codes regulate that nuclear facilities should establish corresponding emergency response plan in China. Though the emergency response plan involves the threat analysis from external incidents, it is not enough to determine Design Basis Threat (DBT) right now. The aim of Design Basis Threat (DBT) is designing and building physical protection to prevent nuclear material to be stolen and nuclear facility to be destructed. DBT is not a completely technological problem, it is established by the government, considering policy and society [5,6].

4) Information of national nuclear authorities

Responsibility of national nuclear authorities is to collect the information of nationwide and international nuclear and radiological terrorist incidents, and transfer the information to relevant organizations. The responsibility of nuclear facility competent authority is collecting information from nuclear facility under jurisdiction.

5) International Information

The information for international nuclear and radiological terrorist incidents mostly comes from international cooperation organizations (e.g. IAEA) and relevant countries.

2. Management and Evaluation of Nuclear and Radiological terrorist Incident Information

According to functions of the evaluation system, the database system is divided into 7 subsystems, including system management, geography information, warning notification, rapid forecast evaluation for radiation consequences, decision-making, expert consultation and database management. Of which, the geography information subsystem is a basis flat for other subsystems, and the database management subsystem offers data support for other subsystems.

V. Emergency Response of Nuclear and Radiological terrorist Incidents

Corresponding with the four grades of warning, emergency response actions include pre-incident actions and post-incident activities. The objective of pre-incident activity is to prevent threat becoming reality and to prepare to mitigate the incident consequence. The post-incident action is mainly to deal with the consequence of incident. [7, 8]

The fourth grade response: When some sensitive things containing radioactive material appear, national authorities commence tracking and investigating, and no special action is needed.

The third grade response: Some threat omens have been appeared, but the threat has not been confirmed (potential threat). National relevant authorities commence evaluation and preparedness. National antiterrorism organization will activate corresponding procedures, and demand relevant organizations to be in stand-by.

The second grade response: It is a believable threat. National antiterrorism organization will approve starting emergency response plan. All relevant antiterrorism organizations will prepare for emergency response according to the emergency response plan, and do all endeavor to prevent the threat becoming reality, and mitigate the consequence of threat.

The first grade response: It is the top response grade. National antiterrorism organization commands all the emergency forces to activate emergency response. The in-site emergency response action includes control of contaminated area, rescuing injured person, emergency assessment, taking radiation protection measures, and decontaminating, etc.

Before the incident occurs, the sensitive phenomenon involving nuclear material appear, and meanwhile the Police, information departments and other national relevant departments commence tracking and investigating in normal way. Actually it is the fourth grade threat, no special action is needed, and the emergency plan has not been started-up. The national crisis management agencies merely commence investigating the event according to normal procedures. Sometimes, the expert support is needed in order to identify the characteristics of the incident. As soon as sufficient evidence can prove the incident to be potential threat, the national antiterrorism organizations take actions, start-up emergency response plan according to the obtained information and forecast, and come into emergency response state.

VI. Conclusion

1) Warning System of Nuclear and Radiological Terrorist Incidents emphasizes to forecast and prevent nuclear and radiological terrorist incidents.

2) The system consists of acquisition of information, analysis and decision-making, and emergency response.

3) Emergency response actions may include the four grades during the response to nuclear and radiological terrorist incident.

4) Warning System of Nuclear and Radiological Terrorist Incidents should be in combination with Response System of Nuclear and Radiological Accidents. The national radiation emergency plan should provide for integration with the response to any nuclear or radiation hazards.

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Probabilistic Safety Analysis about the Radiation Risk for the Driver in a Fast-Scan Container/Vehicle Inspection System

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A new Container/Vehicle Inspection System called fast-scan inspection system has been developed and used in some countries, which has a special advantage in scanning efficiency of 200~400 containers per hour. However, for its unique scanning mode, the fast-scan inspection system causes some worries about the radiation risk for the truck drivers, who will drive the container truck to pass through the scanning tunnel and might be exposed by the radiation beam in accidents.

A PSA analysis, which has been widely used to evaluate the safety of nuclear power plant in the past, is presented here to estimate the probability of accidental exposure to the driver and evaluate the health risk. The fault tree and event tree analysis show that the probability of accidental exposure to the driver is pretty low and the main failure contributions are human errors and scanning control devices failures, which provides some recommendations for the further improvement about this product. Furthermore, on the basic of ICRP No.60 and 76 reports, the health risk to the truck driver is only about 4.0×10^{-14}/a. Compared with the exempt level of 5×10^{-7}/a, it can be concluded that the fast-scan system is safe enough for the truck driver.

KEYWORDS: fast-scan system, potential exposure, event tree analysis, fault tree analysis

I. Introduction

In recent years, hundreds of container/vehicle inspection systems have been installed in different countries around the world, to check the contrabands, drugs and explosive materials etc. which may be hidden inside the containers or cargoes. Unfortunately, most of the container/vehicle inspection systems which have been operated in many customs are not efficient enough in the routine safety check, for they just be able to scan 30 to 40 containers per hour. Compared with the huge number of containers which need to be inspected, the throughput of the container/vehicle inspection systems is too low to meet the requirements for container inspection. It is estimated that only several percent of the containers can be inspected by the container/vehicle inspection systems every day in the world. To solve this problem, a new container/vehicle inspection system called fast-scan inspection system has been developed, and successfully attracted the customers by its extremely high efficiency of scanning 200~400 standard containers per hour.

In the fast-scan inspection systems, several measures have been adopted to accelerate the scanning speed of the system. The most important one is to make the system to allow the driver driving the container truck pass through the scanning tunnel instead of stopping the truck and getting out of the scanning tunnel just like the old container/vehicle inspection systems do. Oberviously, this working mode will cause some worries about the drivers inside the container trucks, both in technical view and in psychological view. Then how to estimate the potential risk to the drivers and how to decrease the possibility of accidental exposure to them as much as possible will be the most important problems in radiation protection aspects for the fast-scan inspection system.

II. Methodology

Originally, probabilistic safety analysis is established in the risk evaluation for nuclear power plants and has been widely used in the nuclear industry[1-5]. Compared with the deterministic analysis method, probabilistic safety analysis is more appropriate to analyze the probability of the risk in very complex situations. ICRP publication No.76[6] has recommended and illustrated how to carry out a PSA analysis in large and complicate radiation facilities. So the paper adopts the PSA method to evaluate the radiation risk of the driver, which mainly includes fault tree analysis and event tree analysis. A fault tree analysis is based on the inverse reasoning from the final outcome or failure, called the “top event”. Fault tree assesses the object in a backward way to find out what could cause the top event. Fault tree can be expressed by the Boolean logic relations and get the numerical results which is of significant meaning in the safety evaluation. An event tree analysis shows a sequence of individual events that may cause unexpected result in a graphical way. Event trees begin with an initiating event, then is divided into two branches at each node according the success and failure of the component, finally give all the possible results and the estimation of the possibility[7]. The paper presents a radiation safety analysis about the driver in fast-scan inspection system by the probabilistic safety analysis (PSA) method and provides some recommendations for the improvement of this kind of system.

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III. Fast-scan inspection system features

A fast-scan inspection system mainly consists of several parts, including accelerator subsystem, detector subsystem,

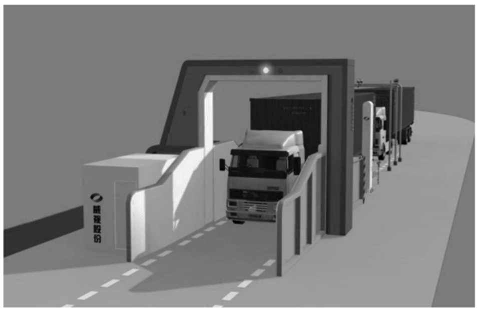


Fig. 1 The layout of fast-scan inspection system

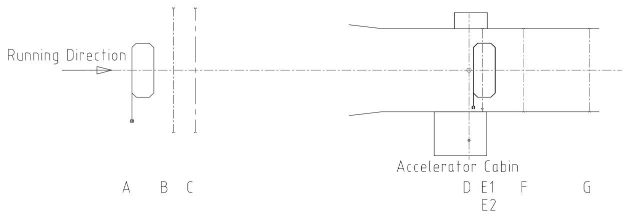


Fig. 2 The location sketch of control sensors

scanning control subsystem, radiation protection subsystem and mechanical structures. Among them, the radiation protection subsystem plays an important role in the radiation safety aspect both to the professionals and to the public, which includes the shielding barriers, collimators, interlock system and emergency devices etc.

The devices which are relative with the radiation safety in working process are shown in **Fig. 2.** Here, A, B, C, D, E, F, G indicate the locations of different kinds of control sensors, e.g. photoelectric sensor, laser curtain, underground coils, respectively.

When a container truck passes through the system, the control sensors will briefly work in this way:

(1) in front of the scanning tunnel, there are some decelerate obstacles to make the truck to keep moving in a speed lower than 15km/h;
(2) the combination of the A and B will give a signal to make the accelerator standby;
(3) the speed sensors B and C are used to measure the speed of the truck, then the system will adjust the pulse rate of the accelerator according to the truck speed;
(4) the combination of the D,E1,E2 and F will give the signal to turn on the accelerator beam;
(5) E2 is also used to turn off the accelerator beam when the
(6) vehicle has thoroughly passed through the accelerator beam;
(7) G works with F to calculate the time to turn the accelerator beam off automatically if the time of beaming is longer than expected;

Besides, in case of any accidentally beam on of the accelerator, an extra radiation detection system is used in the system which will cut off the power of the accelerator immediately if D, E1, E2 and F are not active. This is to reduce the radiation exposure to the driver as low as possible in accidents.

All the devices above are all designed to work in a fail-safe mode that mean any failure of these devices will prohibit the accelerator to send out the radiation beam.

IV. Probabilistic Safety Analysis

There are two main concerns in the probabilistic risk analysis for fast-scan system. One is about the truck driver which will drive the container truck passing through the scanning tunnel; the other is about the foot passenger which may walk through the scanning tunnel by some mistakes. Especially, the truck driver's safety is the pivotal problem in the probabilistic safety analysis.

1. PSA model

The main purpose of PSA for the fast-scan inspection system is to estimate the health risk probability for the truck drivers and foot passengers. The PSA model for the fast-scan inspection system is described in **Fig. 3**. Several parts have been involved, including detail analysis of the system operation sequence, operator regulations, control system functions, then estimating the accident probability and dose level in the accident, finally, making the evaluation about the health risk.

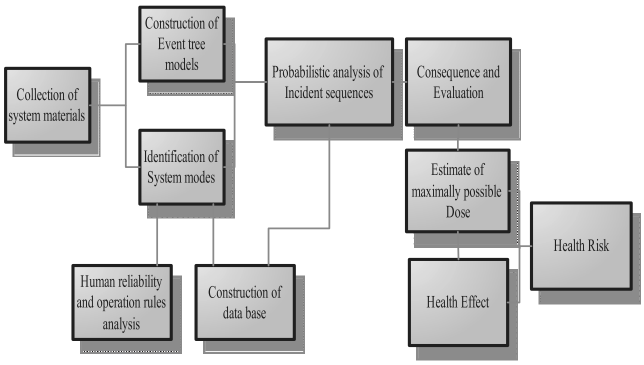
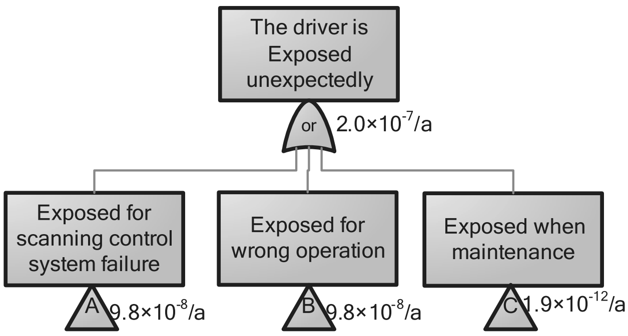


Fig. 3 PSA model for fast-scan inspection system

2. Probabilistic Analysis of the Exposure to Truck Driver



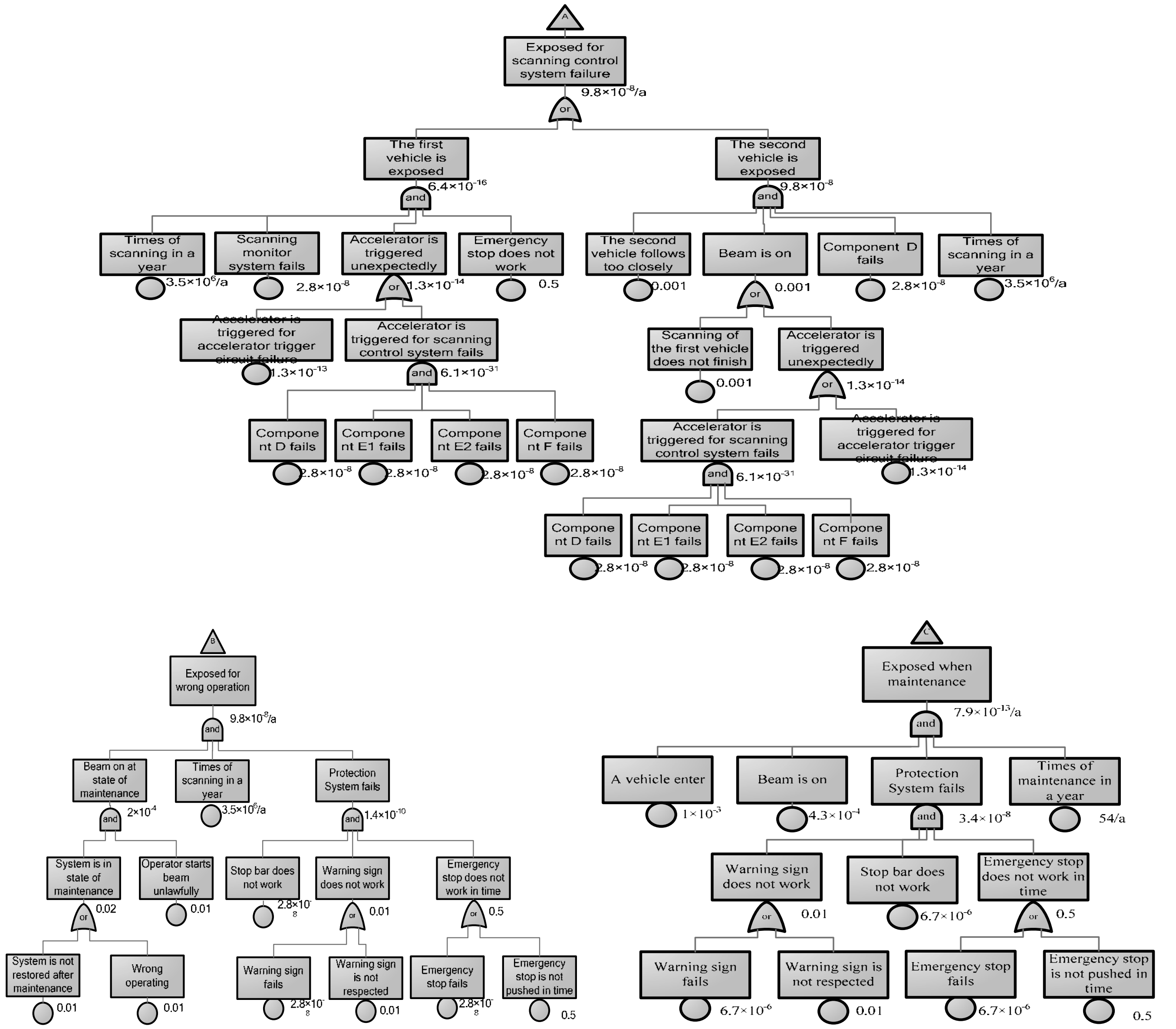


Fig. 4 Fault tree for the accidental exposure to the truck driver

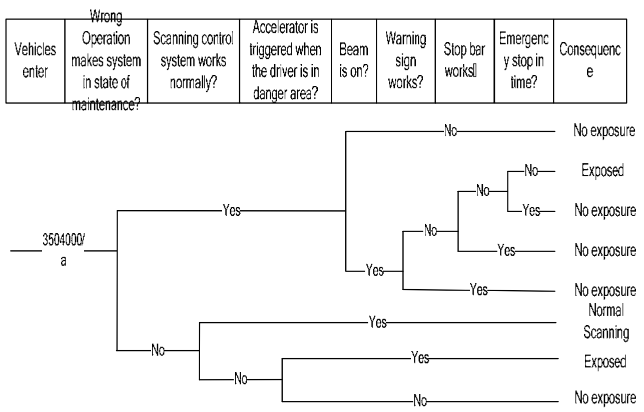


Fig. 5 Event tree for the accidental exposure to the truck driver

Normally, the fast-scan inspection system has two working modes: the operation mode and the maintenance mode. In the operation mode, the accelerator will be beam on only when the control system signals meet a complex logic conditions. While in the maintenance mode, the accelerator can be manually operated, so an extra stop bar is adopted to prevent the trucks getting into the scanning tunnel when the system is in maintenance mode. **Fig. 4** gives the fault tree for the accidental exposure to the truck driver. **Fig. 5** gives the event tree for truck scanning. In normal operation mode, if several control system devices fail at the same time, the truck driver may be exposed; if the first truck scanning has not been finished and the second truck follows too closely, the second truck driver may be exposed also. In maintenance mode, there is some possibility that when the accelerator beam is on, a truck rushed in. Then the driver may be exposed. Certainly, the probability is very low.

Table 1 gives the minimum cut sets for truck driver accidental exposure. It is shown that most of the reasons are related with the human errors.

Table 1 Minimum cut sets for truck driver accidental exposure

No.	Percentage	Expected frequency	Description of the event	***Estimated Dose***
1	50%	9.8×10^{-8}/a	Scanning of the first vehicle does not finish The second vehicle follows too closely Component D fails	<2μSv
2	25%	4.9×10^{-8}/a	Wrong operating Operator starts beam unlawfully Stop bar does not work Warning sign is not respected Emergency stop is not pushed in time	<2μSv
3	25%	4.9×10^{-8}/a	System is not restored after maintenance Operator starts beam unlawfully Stop bar does not work Warning sign is not respected Emergency stop is not pushed in time	<2μSv
4	<0.0001%	7.9×10^{-13}/a	A vehicle enter Beam is on Warning sign is not respected Emergency stop is not pushed in time Stop bar does not work	<2μSv

Actually, the human error is the most complicated one among all the failure reasons. It is quite difficult to give the numerical result for human error concerning the psychology problems, training status, working stress influence etc. For fast-scan inspection system, the operators and maintenance staffs are supposed to work under a mediate working stress to work over time, so the human error probability is taken by 0.01/d. It is a conservative estimation.

Normally, the component failure function is considered to satisfy the exponential distribution. The component failure probability can be calculated as following:

$$P_{\Delta t} = F(t+\Delta t) - F(t) = 1 - e^{-\frac{t+\Delta t}{MTBF}} - (1 - e^{-\frac{t}{MTBF}}) \cong \frac{\Delta t}{MTBF} \tag{1}$$

in the equation, $F(t)$ is the failure probability at t time；$MTBF$ is the mean time interval between two failures. In a container truck scan, the truck speed is about 4m/s, the truck length is 20m, so the Δt for one scan is 5s。

A fast-scan inspection system can scan 400 trucks per hour, one day 24 hours, one year 365 days, then there are 3504000 trucks can be scanned by one system. Obviously, this is a maximum estimation. Under this estimation, the probability of accident exposure to the truck driver is 2.0×10^{-7}/a, even lower than the core damage frequency features for modern nuclear power plant。

From **Table 1**, the biggest minimum cut set contribute almost half of the probability of truck driver exposure, which indicate that the second truck gets into the radiation beam area while the first truck scanning has not completed yet and component D fails at the same time. And the dose for the truck driver is estimated to be less than 2μSv. From here, by the analysis about the minimum cut sets, we can conclude that almost all the accidents are related with the scanning control devices, especially component D is of significance to the driver's safety; and the effective dose to the driver during the accidental exposure is very low.

3. Probabilistic Analysis of Foot Passenger Exposure

The foot passenger is another concern about the accidental exposure. **Fig. 6** gives the fault tree about this situation. It can be seen that because there are several scanning control devices, the probability for foot passenger accidental exposure has been decreased to a very low level.

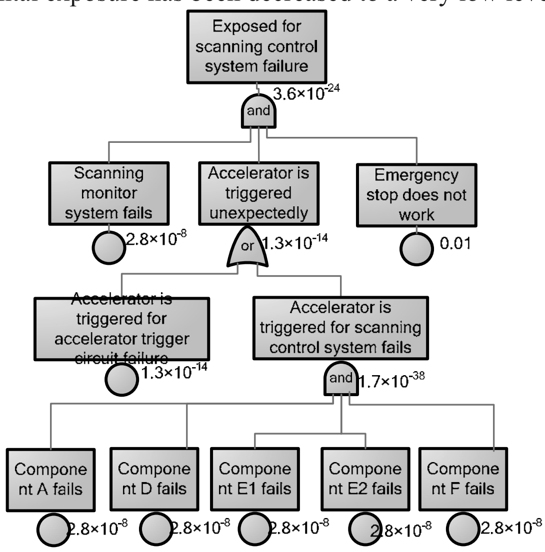


Fig 6. Fault tree for the accidental exposure to the foot passenger

V. Risk Analysis

ICRP No.64 report[8-9] recommended that the constraint of the risk limitation about the potential exposure should be the same in magnitude with that caused by normal exposure. Based on this recommendation, ICRP No.76 report gives the constraint about the potential exposure risk, as following:

$$P \cdot f(E) \leq R \quad (2)$$

in which, P is the probability of event E; $f(E)$ is the probability of death (stochastic effect and deterministic effect) caused by event E; R is the reference risk. If there are many potential exposure possibilities, then it can be written as

$$\sum_i [P_i \cdot f(E_i)] \leq R \quad (3)$$

In ICRP No.76 report, for professions, R is 2×10^{-4}/a; for public, R is 5×10^{-6}/a。

To get the effective dose to the driver, we can assume a very conservative situation in which the driver is exposed to the main radiation beam during scan. The calculation result is 2μSv and the measurement result is 0.02μSv. To estimate the risk, here, the effective dose is adopted as 2μSv.

In ICRP No.60 report[10-11], the nominal probabilistic coefficient is 5×10^{-2}/Sv. Because the dose rate is higher than 0.1Gy/h, so the DDREF（Dose and Dose Rate Effectiveness Factor）is taken as 2. Then the risk to the driver can be calculated:

$$P \cdot f(E) = 2.0\times10^{-7}\times5\times10^{-2}\times2\times10^{-6}\times2 = 4.0\times10^{-14} \quad (4)$$

From here, it is known that the annual risk for the truck driver is about 4.0×10^{-14}/a, far less than the public reference 5×10^{-6}/a and far less than the exempt level of 5×10^{-7}/a. Therefore, it can be concluded that the fast-scan inspection system is safe enough for the truck drivers.

VI. Conclusion

By the PSA analysis about the accidental exposure to the truck driver, the radiation risk for the driver is only about 4.0×10^{-14}/a. Compared with the exempt level of 5×10^{-7}/a, it can be concluded that the fast-scan system is safe enough for the truck driver, although its scanning mode may cause some psychological worries.

Acknowledgement

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Disk Shaped Radiation Sources Fabricated by Compression and Formation of Sinter Powder

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Sinters are deposits found at the bottom of hot springs, some of which contain naturally occurring radioisotopes of the uranium and thorium series. A disk-shaped radiation source was developed by compressing sinter powder. Ten disk-shaped radiation sources were fabricated by this method and their weight, thickness, mass density, and radioactivity were determined. The results indicated that special skills or techniques are not required for production of a radiation source with this method, and the production method is robust. Thus, the method is suitable for the simultaneous fabrication of multiple uniform radiation sources.

To evaluate the ability of this fabrication method to produce sinter radiation sources applicable to courses involving radiation protection or similar investigations, the dependence of the radiation count rate on distance, shielding thickness, and shielding materials was examined using a conventional GM survey meter. The results showed that the sinter radiation source is suitable to comprehend characteristics of radiation and principle of radiation protection related to distance and shielding.

KEYWORDS: sinter, natural radioisotope, uranium, thorium, compressing and forming, radiation source

I. Introduction

Many materials contain naturally occurring radioisotopes, such as monazite, sinter (hot springs deposit), dried seaweed, fertilizer, and lantern mantles[1]. These materials are often used as radiation samples in studies involving natural radioisotopes[2].

Recently, a method for the compression and formation of a radiation source with natural-radioisotope-containing materials has been developed[3]. This method does not affect the amount and concentration of radioactivity in the material, despite reducing the volume of the original material. These materials are referred to as natural radiation sources. In the present study, a natural radiation source was fabricated by compression of sinter powder containing several radioisotopes of the uranium and thorium series.

The performance of the sinter radiation sources fabricated by the method was examined in radiation protection studies, which determined the dependence of radiation count rate on distance, shielding thickness, and shielding material.

II. Fabrication Method of Sinter Radiation Sources

1. Sinter

Sinter consists of hot spring deposits and contains radioisotopes such as ^{212}Pb, ^{212}Bi, ^{214}Pb, and ^{214}Bi[4]. These radioisotopes are daughter nuclides of the thorium or uranium series[5], which occur naturally in the earth's crust[1]. Thus, components in the crust containing the radioisotopes dissolve in hot spring water and then precipitate at the bottom of the hot springs.

The starting nuclide of the thorium series is thorium-232, while that of uranium is uranium-238. The half-lives of thorium-232 and uranium 238 are 1.4×10^{10} years and 4.5×10^{9} years, respectively[5]. These half-lives are very long, since the decrease in radioactivity over the course of the average human lifetime is negligible. Thus, a sinter radiation source can be employed practically forever from the viewpoint of human use.

In the present study, commercially available sinter powder was used to make a sinter radiation source. This sinter powder is produced in the Tamagawa hot spring area in Akita prefecture in Japan for domestic use as a bath agent[6,7]. Sinter powder is safe to handle and is easily soluble in water. Thus, sinter was used to develop a sinter radiation source typical of natural radiation sources.

2. Method of Fabricating Sinter Radiation Source

To fabricate the sinter radiation source, sinter powder (3 - 25 g) was placed in a cylindrical stainless-steel form, with an inner diameter of 35 mm and height of 30 mm. A jack (Osaka Jack Co. Ltd, Model NT20S12.5) and hydraulic hand pump (Osaka Jack Co. Ltd, Model TW-0.7) were used to compress and form the sinter powder into a disc shape in the cylindrical form. Compression force was approximately 160 kN. The fabricated disk-shaped sinter cake had a diameter of 35 mm, equal to the inner diameter of the form. The height of the sinter cake was dependent on the amount of sinter powder.

The method described above was used to fabricate 10 sinter radiation sources using about 20 g of sinter powder. The count rate was measured by 30-minute integration on a source in contact with the center of the head surface of a probe of a GM survey meter (ALOKA, TGS-146) . The setup is shown schematically in **Fig. 1**, in which the distance

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represents the space between the surfaces of the probe and the source. The measurements were performed with the distance set to zero, because the source was in contact with the surface of the probe.

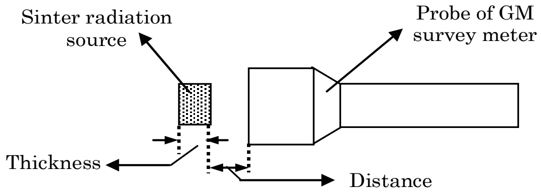

Fig.1 Measuring arrangement of source and detector.

The results are summarized in **Table 1**, which lists weight, thickness, mass density, and radioactivity (count rate) for each of the 10 sources. The average values were 19.92 g, 10.13 mm, 2.01 g/cm^3, and 481.0 cpm, respectively. The relative standard deviations (RSDs) were less than 1% for the first three average values and approximately 3.14 % for the average count rate. The count rate fluctuated more than the other values. However, this level of relative standard deviation is acceptable for sources used in radiation studies related to distance, shielding thickness, and shielding material.

Table 1 Inspection after manufacturing natural radiation sources with compressing and forming method using sinter powder.

Sample No	Weight (g)	Thickness (mm)	Mass Density (g/cm^3)	Radioactivity (Count Rate) (cpm)
1	19.88	10.07	2.019	497.8
2	19.96	10.19	2.003	477.0
3	19.83	10.07	2.014	476.6
4	19.96	10.16	2.007	460.2
5	19.92	10.13	2.013	486.8
6	19.91	10.08	2.022	468.5
7	19.90	10.13	2.011	470.3
8	19.94	10.16	2.008	469.1
9	19.93	10.20	1.998	502.4
10	19.90	10.13	2.009	500.9
Ave.	19.92	10.13	2.010	481.0
RSD	0.192	0.465	0.356	3.139

Ave.: Average,
RSD: Relative standard deviation

These results demonstrate that this compression and formation method is suitable for fabricating numerous uniform sinter radiation sources.

3. Uniform Compressibility

The thickness of the fabricated sinter radiation source was proportional to the amount of sinter powder under a specific pressing pressure, approximately 160 kN, if the powder was compressed uniformly. Uniform compression produced an even distribution of radioactivity across the fabricated source with relative standard deviation of less than a few percent. To confirm this uniformity, six sources were fabricated using approximately 3, 5, 12, 17, 20, and 24 g of sinter powder, and the thickness of the fabricated sources was measured. Figure 2 shows the results, in which the X- and Y-axes represent the weight of sinter powder used and the thickness of fabricated source, respectively. A linear relationship existed between the thickness and the weight of the sinter powder. **Fig. 2** shows that all of the calculated mass densities are about 2.0 g/cm^3. These results show definitively that uniform compressibility is achieved irrespective of the amount of sinter powder, indicating that radioactivity distribution also is uniform throughout the source within a few percent deviation.

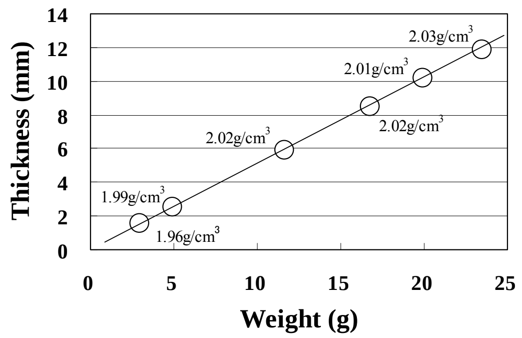

Fig.2 Dependence of source-thickness(density) on amount of material weight.

4. Saturated Thickness of Source

Since compression is uniform, as shown above, the radioactivity of a sinter radiation source should be proportional to the thickness and weight of the sinter powder used. However, the radioactivity evaluated with a GM survey meter may not necessarily correspond to the thickness of a saturated source, due to self-absorption of radiation emitted inside the source by the source material itself. To examine the saturation, the relationship between count rates and source thickness was investigated; weight should be proportional to source thickness under the condition of uniform compression using a specific form.

In this experiment, 15 sources of different thicknesses were fabricated, and their radioactivity measured over 30 minutes using a GM survey meter. During the measurements, each source was in contact with the center of the head surface of the probe of the survey meter.

The relationship obtained between the count rate and thickness is shown in **Fig. 3**, in which the X-axis represents the thickness of the 15 sources and the Y-axis represents count rates, which is the apparent radioactivity measured with the GM survey meter. Figure 3 shows that the count rate gradually increases with thickness initially, then moderates, and finally becomes constant for sources thicker than 23 mm. This saturation thickness of 23 mm corresponds to about 45 g of sinter powder. Thus, sources fabricated using an amount of sinter powder heavier than 45 g does not result in an increase in count rate, but remains constant. In following experiments in the chapter III, the source was fabricated using 20 g of sinter powder. This source had a thickness of about 10 mm, which is less than the saturation thickness, and was used in the following experiments.

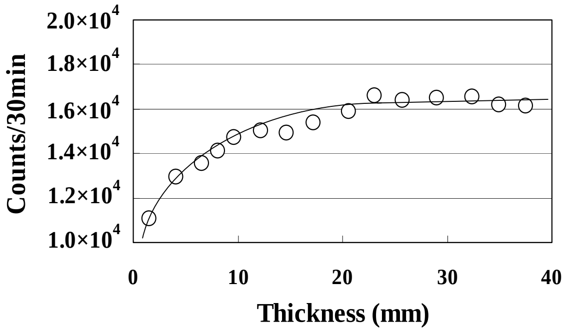

Fig.3 Dependence of radiation counts on source-thickness.

III. Performance Tests of Sinter Radiation Sources

To examine the applicability of the sinter radiation source, dependency tests of count rate on distance and shielding were conducted. In these tests, the count rates were measured over 10 minutes with a GM survey meter.

1. Distance Dependence Test

For the distance dependence test, the distance between the source and probe was set from 0 to 30 cm for 11 sources and the count rates were obtained. A distance of zero means that the source was directly in contact with the surface of the survey meter probe. Results are shown in **Fig. 4**, in which X- and Y-axes are distance and count rate, respectively. As the distance increased, the count rate decreased rapidly at first and then more moderately. This variation in count rate as a function of distance appeared to follow the inverse-square law.

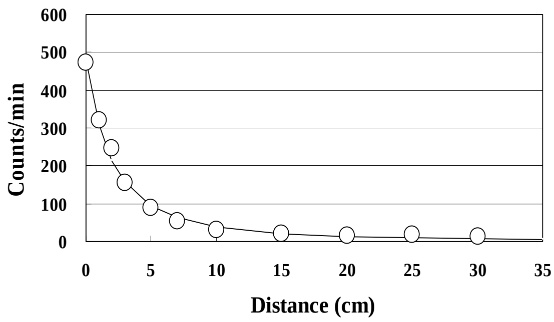

Fig.4 Dependence of radiation counts on distance**.**

To confirm this, a curve represented by the function $Y = A/(a+X)^2$ is shown in **Fig. 4** as a solid line. X and Y are distance and count rate, respectively, and A and a are constants with values of 7.7×10^3 cpm·cm^2 and 4 cm, respectively. The resulting graph shows that measured count rates were similar to calculated values, indicating that a sinter radiation source fabricated from sinter powder is suitable for semi-quantitative studies demonstrating the inverse-square law.

2. Shielding Dependence Test

To examine the dependence of shielding on the radioactive source material and its thickness, four types of thin plates were used as shielding materials. Only materials commonly available were selected as shielding to allow the experiments to be adapted for a basic radiation course appropriate for junior and senior high school students. Kent paper (0.25 mm thick) was the first shielding material used. In the experiment, the paper was cut into plates with 50 mm square and used by stacking an appropriate number of the paper plates to achieve shielding of various thicknesses (0.5, 1, 2, 4, and 8 mm). Similarly, plates of commercially available plastic (vinyl chloride resin, 0.4 mm thick), aluminum (0.5 and 1.0 mm thick), and copper (0.5 and 1.0 mm thick) were selected as the other three materials. For the additional experiments, plates were fabricated by cutting the materials into 50-mm square plates, which were stacked to achieve shielding of various thicknesses.

The distance between the probe and the sinter radiation source was fixed at 15 mm, and various thicknesses of shielding materials were placed between the probe and source. Using a GM survey meter, the count rate was measured with and without shielding materials. By dividing the shielded count rates by unshielded count rates, respective transmission rates were obtained.

Fig. 5 shows the transmission rate results, in which the X- and Y-axes represent the thickness of shielding materials and transmission rate, respectively. Thus, **Fig. 5** shows the transmission rate for the four materials as a function of thickness. The transmission rates for the 0.5- and 1-mm plastic plates were determined by interpolation using the data obtained for plastic plates with thicknesses of 0.4 and 0.8 mm, and 0.8 and 1.2 mm, respectively.

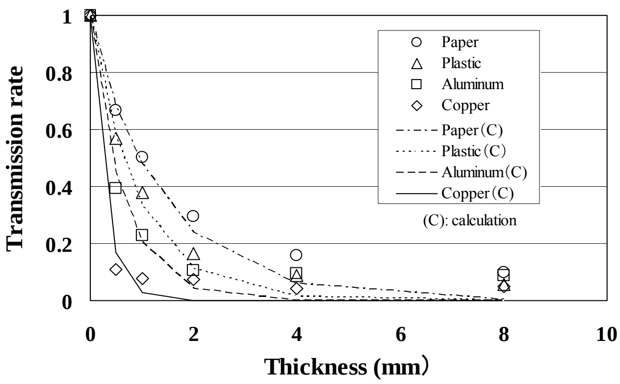

Fig.5 Dependence of radiation counts on shielding thickness of various materials.

All transmission rates decreased exponentially with an increase in thickness in the order from greatest to lowest rate: copper > aluminum > plastic > paper. Since mass densities of copper, aluminum, plastic, and paper are 8.9, 2.7, 1.35, and 0.93 g/cm^3, respectively, the results explain the general principle of radiation shielding that materials of a larger mass density provide greater radiation shielding. This experiment explains, semi-quantitatively, the relationship between mass density and effectiveness of shielding against radiation.

Fig. 5, which shows the curves determined for each shielding material, indicates that the change in transmission rate can be represented by exponential function:

$$Y = e^{-AX} \tag{1}$$

where X and Y are the thickness and transmission rate, respectively, and A is a constant specific for each material. These curves were obtained by estimating values of A. The curves correspond well to the change of transmission rate, except for thicknesses greater than a few millimeters. Results semi-quantitatively demonstrate that the effectiveness of shielding varies exponentially with the thickness of the shielding material. Thus, the value for A estimated for each type of shielding material is a linear absorption coefficient that characterizes effectiveness of the materials. The linear absorption coefficient is originally defined for mono-energetic photons such as gamma-ray, whereas sinter contains various radioisotopes and emits different radiation of various energies. Therefore, A is not strictly a linear absorption coefficient, but a value very similar to it. To differentiate A from a linear absorption coefficient defined for mono-energetic photons, A in Equation (1) is called an "attenuation factor."

The attenuation factor was estimated to be 3.5 mm^{-1} for copper, 1.6 mm^{-1} for aluminum, 1.1 mm^{-1} for plastic, and 0.7 mm^{-1} for paper. From Equation (1), the formula for calculating a half-value layer ($X_{1/2}$) is derived by:

$$X_{1/2} = 0.693/A \tag{2}$$

Using the attenuation factors and Equation (2), half-value layers were determined to be 0.2, 0.43, 0.63, and 0.99 mm for copper, aluminum, plastic, and paper, respectively. Nearly identical values can be derived from **Fig. 5**. The attenuation factor employed here from applying the concept of linear absorption coefficient and half-value layer can be used to demonstrate the characteristics of radiation shielding.

All results in Sections III.1 and III.2 demonstrate that the sinter radiation source is useful for demonstrating the relationships between radiation strength and distance, radiation strength and shield thickness, and radiation strength and mass density of shielding materials using practical experiments.

IV. Summary

Using sinter containing naturally occurring radioisotopes, nuclides of which belong to the thorium or uranium series, a natural radiation source was developed. A compression and formation method was employed to fabricate a natural sinter radiation source.

This fabrication method did not change the original weight and radioactivity of the material, although it did reduce the volume of the material. Therefore, the method does not affect the amount and concentration of radioactivity in the material.

The effect of varying the parameters of the compression and formation method on weight, thickness, mass density, and radioactivity of the natural sinter radiation source was investigated. Results showed that the method was useful for fabricating multiple uniform natural sinter radiation sources. Importantly, no special skills or techniques were required to employ this method successfully.

To verify the effectiveness of the sinter radiation source, the dependence of count rate (measured using a GM survey meter) on distance, shielding thickness, and shielding material was examined.

The test investigating distance dependence demonstrated that the inverse-square law applied to the distance between the source and detector. The shielding thickness dependence test demonstrated that the relationship between shielding effectiveness and thickness of the shielding materials was exponential. In addition, the shielding effectiveness increased with mass density of the material, and an attenuation factor similar to a linear absorption coefficient and a half-value layer could be determined.

The results showed that the sinter radiation source fabricated by compression and formation using sinter powder can be applied to demonstrate characteristics of radiation relating to distance, shielding thickness, and shielding material. The sinter radiation source is a natural substance, not a legal radioisotope, and thus can be safely and easily used in a basic course on radiation protection. This sinter radiation source is a useful teaching aid for courses in understanding radiation and its characteristics.

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Determination and Verification of the Scaling Factor for the Radionuclide Inventory of the Radioactive Waste from Nuclear Power Plants

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The Database of the first Scaling Factor for the radionuclide inventory of the radioactive wastes from the nuclear power plant (NPP) in Korea was obtained in 2005, which consists of the ratios of the radioactivity of Difficult-To-Measure (DTM) radionuclides to key gamma nuclides such as ^{60}Co and ^{137}Cs. The Database of the SF has 255 of radioactivity data on 6 waste streams from 20 NPPs. If the SF can be applicable to any wastes regardless of their generation year, then it can be applied all over the wastes from the past NPP wastes to the future ones. For the purpose, KHNP has planned to obtain the second SF through 337 samples from the same wastes as the first ones from 2007 to 2008. The second SF will be a principal key to verify the validity of the first SF for Korean NPP wastes. The first and the second SF will be unified to one SF in case of completion of verification. The wastes produced before the first SF was derived also have to be verified that it has the same correlation to the unified SF. In this study, 8 solidified evaporated bottom waste drums produced from 1988 to 2003 were cored and the radioactivities were analyzed. The SFs were compared with the first SF and proved that the SF can be applied all over the wastes from the past NPP wastes to the future ones in case of the solidified evaporated bottom waste drums.

KEYWORDS: scaling factor, radioactive waste, radioisotope inventory, nuclear power plant

I. Introduction

In 2005, the regulation on the delivery of low and intermediate level radioactive waste (LILW) to disposal facility was revised. The criteria of examining the radioisotope inventory of the LILW were also decided. The lowest detection limit of radioactivity of the waste was defined in the criteria.

The radioactivities of gamma-ray-emitting radioisotopes can be directly determined by measuring the gamma rays. But the radioactivities of alpha-beta-particle-emitting radioisotopes can not be directly determined by non-destructive way like gamma ray. They can be derived by the Scaling Factor (SF).

The SF is the ratio of the radioactivity of difficult-to-measure (DTM) radionuclide to one of any key gamma nuclide such as ^{60}Co and ^{137}Cs in the radioactive waste. It can be used to estimate nondestructively radioisotope inventory of DTM radionuclide of the radioactive wastes produced from NPPs.

The first SF of NPP in Korea was obtained in 2005 by analyzing the radioactivity ratios of DTM to key gamma nuclides. The first SF was derived through statistical process on 255 samples with 6 waste streams produced from 13 NPP groups.

The SF shall be verified to be independent of the generation year of all over the wastes from the past to the future ones. KHNP planned to obtain the second SF out of 337 samples from the same waste streams and the same NPP groups from 2007 to 2008.

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The second SF will be a principal key to verify the validity of the first SF to all over the wastes in Korea NPP wastes. The first and the second SF will be united one SF in case of completion of verification. The wastes produced before the first SF was derived also have to be verified that it has the same correlation to the unified SF.

In this study, 8 solidified evaporated bottom waste drums produced from 1988 to 2003 were cored and the radioactivities were analyzed. The SFs were compared with the first SF.

II. Measurement

To verify the applicability of the first SF to the past wastes produced before the first SF was derived, 8 solidified evaporated bottom waste drums from 1988 to 2003 were cored. The radioactivities of the cored samples were analyzed and the SFs were obtained from them as follows:

Every drum was cored three times at top, middle and bottom position. The radioactivities of three cored samples of each drum were measured and averaged. Analyzed nuclides were alpha and beta emitting radionuclides such as ^{3}H, ^{14}C, ^{55}Fe, ^{59}Ni, ^{63}Ni, ^{90}Sr, ^{94}Nb, ^{99}Tc, ^{129}I and total alpha including key gamma nuclides such as ^{60}Co and ^{137}Cs.

But radioactivities of alpha emitting nuclides or total alpha were not detected. So, alpha-emitting nuclides were excluded from the SF comparison.

^{90}Sr was measured 5 times in 8 cored samples and compared to the first SF.

The averaged radioactivities of cored samples were corrected considering their radioactive decay.

Verification was done as follows: If the ratio of the sample's SF to the first SF is between 0.1 and 10, then it was concluded that the correlation of the sample's SF and the first SF is good in case that the standard deviation or log mean dispersion of the ratio of the averaged SF to the first SF is less than 10.

If the logs mean dispersion of the ratios of the averaged SF to the first SF is larger than 10, then the ratio comparison can not be a good a yardstick for judgment of correlation. In that case, the correlation between the averaged SF and the first SF will be judged from the E_n value as follows using their relation:

$$E_n=|x-X|/(u_p^2+u_f^2)^{1/2}$$

where, u_p is the standard deviation of the averaged SF and u_f is the standard deviation of the first SF. If E_n value is less than 1, then the correlation of the sample's SF and the first SF can be verified to be good.

III. Results

The ratios of the sample's SF to the first SF for the 8 nuclides except ^{129}I were as follows:

No.	^{3}H	^{14}C	^{55}Fe	^{59}Ni	^{63}Ni	^{90}Sr	^{94}Nb	^{99}Tc
1	0.15	0.03	0.93	0.35	1.28	0.74	1.64	0.00
2	0.14	1.00	0.97	0.41	1.52	1.18	1.23	0.01
3	0.53	0.10	-	-	3.29	1.00	5.68	0.16
4	0.17	0.09	5.13	0.14	0.50	17.4	0.65	0.46
5	0.61	1.08	3.23	1.43	3.38	1.02	2.88	0.07
6	0.04	0.39	5.75	0.47	1.31	-	3.68	0.10
7	0.03	0.62	7.65	0.14	1.72	-	2.05	2.31
8	0.01	0.18	4.18	0.14	0.60	-	1.33	23.13

The averaged ratios of the sample's SF to the first SF were between 0.1 and 2.2 and all of them met the criteria as follows:

nuclide	Ratio of averaged SF to the first SF	Standard deviation of ratio
^{3}H	0.10	0.23
^{14}C	0.25	0.42
^{55}Fe	3.10	2.48
^{59}Ni	0.31	0.46
^{63}Ni	1.40	1.09
^{90}Sr	1.73	7.33
^{94}Nb	1.96	1.64
^{99}Tc	0.16	8.06

From the above results, it was proved that the sample's SF of 8 solidified evaporated bottom wastes had a good correlation to the first SF.

^{129}I does not met the criteria that the log mean dispersion of the ratios of the sample's SF to the first SF is more than 10, namely 32.6. So, the E_n value was calculated for comparison instead of the ratio of the sample's SF to the first SF. The E_n value was:

$$E_n=0.705$$

The E_n value was less than 1. So it was proved that the correlation between the sample's SF and the first SF was good.

From the results, it was proved that the SF is applicable to all wastes regardless of their generation year and it can be applied all over the wastes from the past NPP wastes to the future ones in case of solidified evaporated bottom waste drums.

IV. Conclusion

KHNP has obtained the first Scaling Factor for the radioactive wastes produced from the nuclear power plant in Korea in 2005.

In this study, 8 solidified evaporated bottom waste drums produced from 1988 to 2003 were cored. The radioactivities of the cored samples were analyzed and the SFs were obtained from them. The SFs were compared with the first SFs.

The averaged ratios of the sample's SF to the first SF for the 8 nuclides except ^{129}I were between 0.1 and 2.2 and all of them met the criteria. And for ^{129}I, The E_n value was 0.705 or less than 1 and met the criteria.

From the results, it was proved that SF is applicable to all wastes regardless of their generation year and it can be applied all over the wastes from the past NPP wastes to the future ones in case of solidified evaporated bottom waste drums.

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