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Proceedings of the Fourth JAEA-US EPA Workshop on Radiation Risk Assessment

November 7-8, 2006,
Tokai Research and Development Center, JAEA, Japan

(Eds.) Akira ENDO and Michael BOYD*

Research Group for Radiation Protection
Nuclear Science and Engineering Directorate

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Proceedings of the Fourth JAEA–US EPA Workshop on Radiation Risk Assessment

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(Eds.) Akira ENDO and Michael BOYD*

Division of Environment and Radiation Sciences
Nuclear Science and Engineering Directorate
Japan Atomic Energy Agency
Tokai-mura, Naka-gun, Ibaraki-ken

(Received January 4, 2007)

This report is the proceedings of the fourth workshop jointly organized by the Japan Atomic Energy Agency (JAEA) and the United States Environmental Protection Agency (US EPA) under the terms of agreement for cooperation in the field of radiation protection. The workshop was sponsored by the Nuclear Science and Engineering Directorate and was held at the Nuclear Science Research Institute, the Tokai Research and Development Center, JAEA, on November 7–8, 2006. The objective of the workshop was to exchange and discuss recent information on radiation effects, radiation risk assessment, radiation dosimetry, emergency response, radiation protection standards, and waste management. Twenty-two papers were presented by experts from JAEA, US EPA, the National Academies, Oak Ridge National Laboratory, Washington State University and the US Nuclear Regulatory Commission. Three keynotes addressed research on radiation effects and radiation protection at JAEA, the latest report on health risks from exposure to low levels of ionizing radiation published by the National Research Council (BEIR VII Phase 2), and recent developments in Committee 2 for the forthcoming recommendations of the International Commission on Radiological Protection (ICRP). The workshop provided a good opportunity for identifying future research needed for radiation risk assessment.

Keywords: Radiation Risk Assessment, Radiation Effects, Radiation Dosimetry, Emergency Response, Radiation Protection Standards, Waste Management.

∗ United States Environmental Protection Agency

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第 4 回原子力機構・米国環境保護庁放射線リスク評価に関するワークショップ報文集

2006 年 11 月 7・8 日、日本原子力研究開発機構東海研究開発センター、東海村

日本原子力研究開発機構原子力基礎工学研究部門
環境・放射線工学ユニット
(編) 遠藤　章、Michael BOYD*

(2007 年 1 月 4 日受理)

本報告書は、日本原子力研究開発機構 (原子力機構) 及び米国環境保護庁との放射線防護分野における協力に関する取り決めの下で、合同で開催された第 4 回ワークショップの報文集である。本ワークショップは原子力基礎工学研究部門が主催となり、原子力機構東海研究開発センター原子力科学研究所において、2006 年 11 月 7・8 日に行われた。ワークショップの目的は、放射線影響、放射線リスク評価、線量評価、緊急時対応、放射線防護基準、廃棄物管理に関する最新情報を相互に交換し、議論することであった。原子力機構、米国環境保護庁、米国アカデミー、オークリッジ国立研究所、ワシントン州立大学、米国原子力規制委員会からの専門家により、22 件の講演が行われた。3 件の基調講演では、原子力機構における放射線影響及び放射線防護に関する研究、米国研究評議会が刊行した低レベルの電離放射線の被ばくによる健康リスクに関する最新の報告書 (BEIR VII Phase 2)、国際放射線防護委員会 (ICRP) の新勧告に向けた第 2 専門委員会の活動についての講演が行われた。本ワークショップは、放射線リスク評価に必要な今後の研究を明確にする上で極めて有益な機会となった。

原子力科学研究所 (駐在)：〒 319–1195 茨城県那珂郡東海村白方白根 2–4
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Opening Address

Osamu Oyamada
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Good morning, ladies and gentlemen. As introduced just now, I am Osamu Oyamada, the Director General of the Nuclear Science and Engineering Directorate at the Japan Atomic Energy Agency. It is my pleasure to make an opening address on behalf of JAEA. I am pleased that we have the opportunity to host this workshop, and I would like to express my welcome to all participants from the United States and JAEA.

Since 1986, the Japan Atomic Energy Research Institute, JAERI, and the United States Environmental Protection Agency, EPA, have been exchanging information in the field of radiation protection under the terms of agreement for cooperation between the both organizations. This cooperation was conducted by the former Department of Health Physics at JAERI and provided mutual benefit in research of radiation risk assessment, development of radiation dosimetry techniques, study of residual radioactivity and recycling through the past three workshops.

On October 1, 2005, JAEA was established by the integration of JAERI and the Japan Nuclear Cycle Development Institute, JNC. JAEA is the only institute in Japan dedicated to comprehensive research and development in the field of nuclear energy. At JAEA, the research on radiation effects and radiation protection has been positioned in our directorate and has been promoted as one of important bases for utilization of nuclear energy and radiation. The Nuclear Science and Engineering Directorate takes over the agreement with EPA from JAERI and organizes this fourth workshop, which focuses on the recent developments in radiation effects, radiation risk assessment, radiation dosimetry and radiation protection standards.

I would like to express our gratitude to EPA and the National Research Council of the National Academy of Sciences for the publication of BEIR VII report. The BEIR report has been recognized as a comprehensive report on the health effects of exposure to low radiation doses. I believe that the BEIR VII will be used as scientific bases for radiation effects and radiation protection, and I am pleased to hear presentations about this report and the related topics on radiation effects and radiation risk assessment.

The next important topic is radiation protection standards and radiation dosimetry. The International Commission on Radiological Protection, ICRP, will adopt its new recommendation in 2007. And, to implement this new recommendation, ICRP has been developing a new system for radiation dosimetry. I understand the radiation protection group at JAEA has been contributing to the development of decay data and dosimetry for high-energy radiation. I think it is timely to discuss the recent developments of radiation dosimetry and radiation protection standards.

Emergency response and waste management have been continuous issues for the radiation protection of the public and acceptance of nuclear energy. Both EPA and JAEA have excellent

technical bases, facilities and experiences in these fields. It will be a good opportunity to exchange and share information.

In closing this address, I hope that this workshop will be very informative and fruitful for all participants. I also hope that the participants from the United States will enjoy your stay in Tokai-mura. Thank you for your attention.

Opening Address

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Thank you, Mr. Oyamada, and good morning, ladies and gentlemen. On behalf of the United States Environmental Protection Agency, I would like to thank the Japan Atomic Energy Agency for hosting this fourth joint workshop. Since 1986, these workshops have provided our two agencies with a forum for exchanging current information about our research activities and policies in the field of radiation protection. We use the term radiation protection broadly. Through these workshops we have exchanged information about radioactive waste management and recycling, emergency preparedness and response, internal and external dosimetry, and current epidemiology and its influence on human health risk assessment. Also, as you will hear over the next two days, we have fostered a very successful collaboration between Dr. Endo and Dr. Eckerman that will likely lead to an update of ICRP Publication 38, the primary international reference for radionuclide decay data. This new data is possible only because of the support of JAEA.

I am pleased that the longstanding agreement between EPA and the Japan Atomic Energy Research Institute was transferred to JAEA when it was formed last year. I have personally enjoyed working with my many good friends at JAERI, now JAEA, over the last ten years. I was partly responsible for planning our third workshop, which was held 5 years ago in Las Vegas, Nevada. At that time, I worked with Mr. Shohei Kato, who is here today. That workshop came two months after September 11, 2001, and there was some concern whether we should cancel the workshop. We decided to go ahead with it and I am glad that we did because it turned out to be very successful. I believe that many other people in this room also attended that workshop. Many of the topics that we reported on then, such as the BEIR VII study, have been completed in the last five years, and you will hear about them at this workshop. When we first started planning this workshop, I worked with Dr. Yasuhiro Yamaguchi, whom I have come to know through our work together on the Nuclear Energy Agency's Committee on Radiation Protection and Public Health. I am very pleased that he is also attending this workshop. Over the last two years, I have worked with Dr. Akira Endo. Dr. Endo deserves the credit for making this workshop a reality. I have had a very easy job, because he has taken care of every detail. I am sure that all my American colleagues join me in thanking him and the JAEA for making our travel to Tokai so effortless. I am sure that the next two days will provide us the opportunity to share important information about our current work in radiation protection research and policy development, and that we will form many new friendships. Thank you again, Mr. Oyamada, for demonstrating the commitment of JAEA to this important series of workshops. On behalf of all of us who have traveled here from the United States, thank you to everyone from JAEA for your generous hospitality. I am looking forward to our next three days in Tokai.

Session 1　Radiation Effects and Radiation Risk Assessment

1-1 Research on Radiation Effect and Radiation Protection at JAEA

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Abstract

Researches on radiation effect and radiation protection at JAEA have been carried out in different sections. In recent years, the organizations were rearranged to attain better research circumstances, and new research programs started. At present, radiation effect studies focus on radiation effect mechanisms at atomic, molecular and cellular levels including simulation studies, and protection studies focus on dosimetry for conditions difficult to cover with currently used methods and data as well as the related basic studies. The outlines of the whole studies and also some descriptions on selected subjects will be given in this paper.

Keywords: Radiation biology, Radiation protection, Radiation effect mechanisms, Dosimetry, Computer simulation, Irradiation facilities

1. Introduction

This paper gives an overview concerning researches on biological radiation effects and radiation protection being conducted at JAEA. Life science researches at JAERI have been carried out in several different groups aiming at different targets. However, in some cases their targets are common from wide viewpoints, and it was expected that collaboration would promote the researches much. In 2006, the Research Unit for Quantum Beam Life Science Initiative was organized for effectively performing these studies by collaborating among different groups belonging to different current units[1)]. The Research Unit consists of five newly organized groups shown in Table 1, groups 1 and 2 performing studies on radiation effects as the group names indicate. Radiation protection research programs were also rearranged after the organizational integration in 2005 inheriting studies carried out in the preceding research laboratories. Now radiation protection researches are performed mostly in the Research Group for Radiation Protection, Nuclear Science and Engineering Directorate. A few researchers are performing radiation protection studies in other units, and effective collaboration with these researchers would be desired in future. Nevertheless, as a whole, organizations have been rearranged to attain better circumstances for the researches. In this paper, main studies concerning

Table 1 Research groups newly organized in the Research Unit for Quantum Beam Life Science Initiative

1. Radiobiology mechanism group
2. Repair protein group
3. Drug target protein group
4. Radioisotope drug delivery system group
5. External radiation therapy group

radiation effect and protection will be described briefly.

2. Research potential of JAEA

In implementing these studies, JAEA's research potential should be fully utilized. Two things can be pointed out as the potential useful for the studies: 1) irradiation technology, and 2) computational technology. In the long history of nuclear energy development at JAEA, these technologies have been cultivated in terms of both hardware and software.

Concerning irradiation technology, many different kinds of radiations with different conditions are now available. In those, functional beams like microbeam and coherent X ray are included. JAEA has several R & D centers over Japan at shown in Figure 1, and some of them have specific irradiation facilities. The Tokai Nuclear Research and Development Center has plural neutron irradiation facilities: research reactors (JRR-3, 4), the Facility for Radiation Standard, and large proton accelerators for utilizing neutrons are under construction in J-PARC. At the Takasaki Radiation Chemistry Research Institute, ion beams, electron beams and Co-60 gamma rays are available. The Kansai Research Institute provides laser and synchrotron X ray. These irradiation facilities together with the irradiation techniques are utilized in the studies. While radiation protection is necessary for these facilities, and this has produced new studies concerning radiation protection.

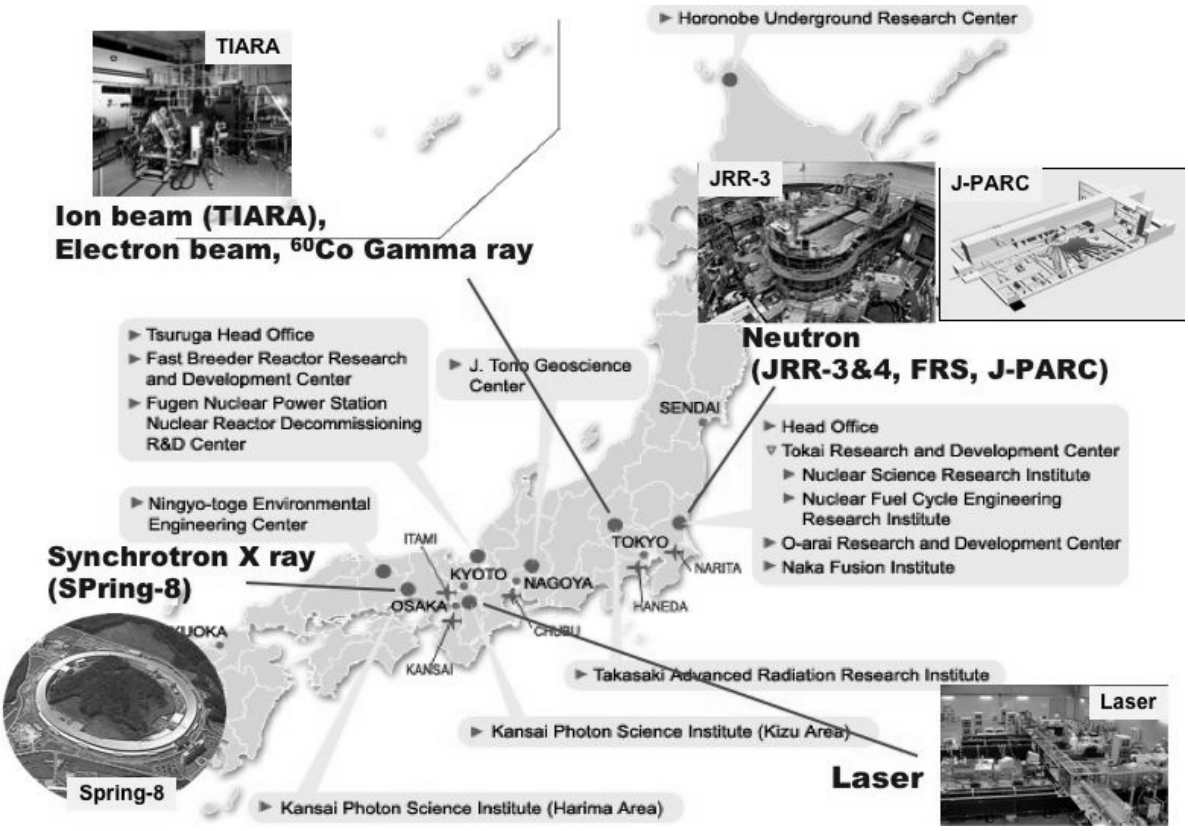


Figure 1 JAEA's R & D centers and irradiation facilities

Computational technology is an important factor in nuclear technology. Many nuclear facilities have been designed with a help of computation simulations; consequently, simulation codes and techniques which can be widely applied to radiation research have been developed like radiation transport calculation codes. According to these backgrounds, JAEA has several high performance computers at different sites, and the related computer techniques have been also developed like grid computing techniques in the ITBL (IT-based laboratory) project. The computational facilities and techniques are effectively used in the studies.

3. Research on radiation effect

Obviously an important target of radiation effect studies is to contribute to elucidation of the low-dose effect and risk. In recent years, some important documents concerning the low-dose effect based on epidemiological studies have been published. One is the BEIR VII report which will be discussed in the next paper. Also, a paper by Cardis et al. made a new analysis summing up different epidemiological data for workers from different countries. Nevertheless, it seems that the low-dose effect below several tens mSv is still not clear, and mechanism studies are considered to be necessary to properly understand the low-dose effect.

A proposed carcinogenesis process due to radiation is shown in Figure 2. Events which happen in a very small region in a very short time, that is ionizations and excitations, will be amplified after a long time, and finally kill the human body. It's a very long process, and a lot of studies are necessary

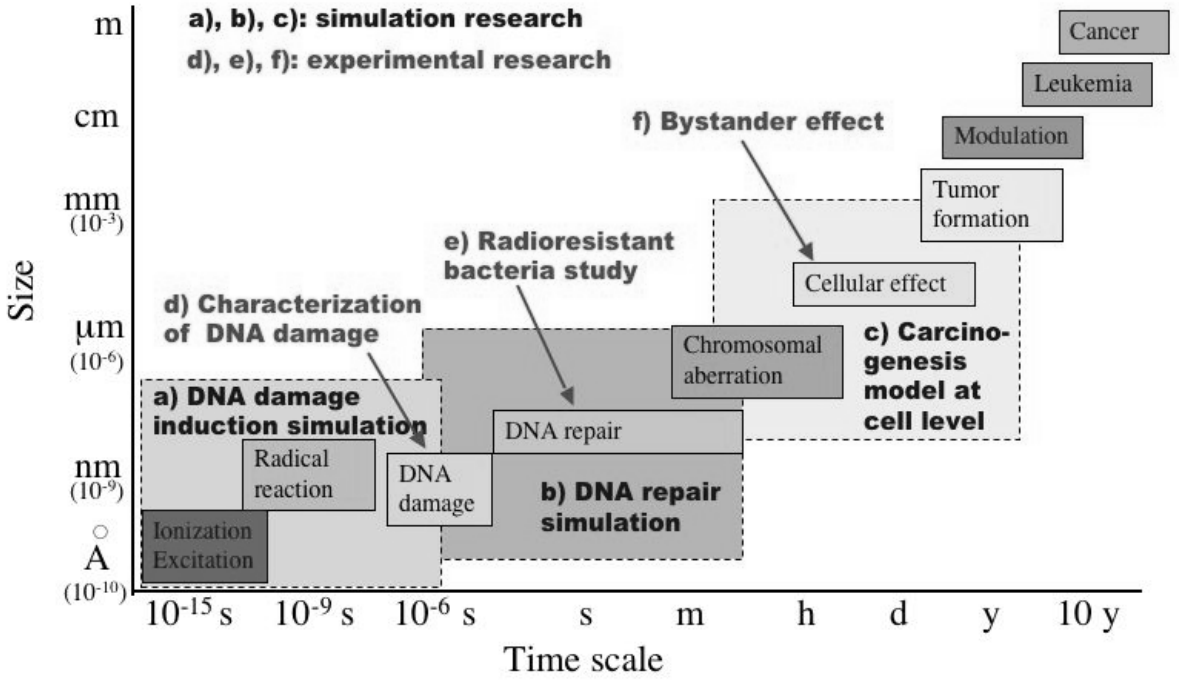


Figure 2 Proposed carcinogenesis process initiated by energy deposition events by radiation and related studies performed at JAEA

to elucidate the process. JAEA concentrates on the early stage of this process at atomic, molecular and cellular levels. Though the early stage of the process is essential to characterize the radiation effect, it is not thoroughly investigated. Both simulation and experimental studies are being carried out. In simulation studies, DNA damage induction simulation, DNA repair simulation, and development a carcinogenesis model at a cellular level have been performed. In experimental studies, characterization of DNA damages, a radioresisant bacteria study, and a bystander effect study have been performed.

3.1 Simulation study

DNA damage induction is simulated using a Monte Carlo method. In this simulation, physical and chemical processes due to radiation are precisely simulated aiming at obtaining systematic data on the relation of radiation quality to DNA damage species and yields, paying special attention for DNA damages difficult to repair. DNA models are assumed to be in water; when a radiation enters the system ionization and excitation events are induced; then, radical species are produced and start diffusing in water. If two radicals get close enough, a chemical reaction happens and a new chemical species is produced. DNA damages are induced by direct energy deposition onto DNA (direct effect) and indirect attacks by radicals produced around DNA (indirect effect). Features of energy deposition events obtained from simulation are shown in Figure 3[2]. It is clear that the energy deposition distribution is different according to radiation type and energy. In case of low LET radiation, especially high-energy gamma rays, the energy deposition happens sparsely. On the other hand, in

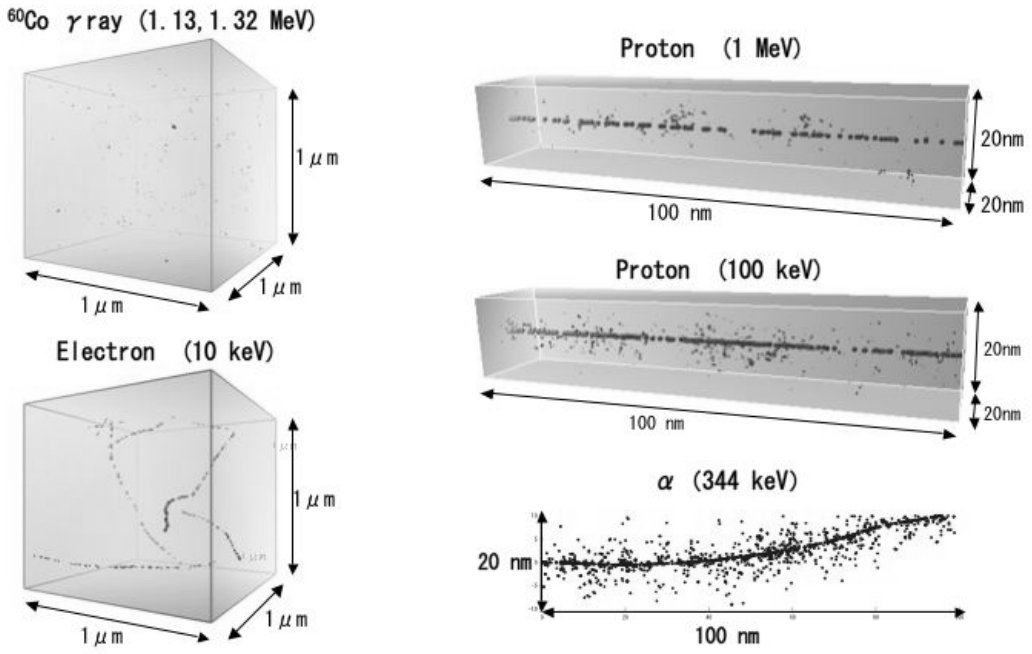


Figure 3 Distributions of ionization and excitation events induced by different radiations obtained from computational simulations

high LET radiation, energy deposition density is very high. Using the simulation, characteristics of DNA damages by radiation have been examined. It was found that low energy electrons and photons below a few keV can produce complex damages even though they are classified into low-LET radiations. It was confirmed that the complexity of DNA damages increases for higher LET up to 100 keV/μm.

DNA repair simulation is performed using a molecular dynamics calculation. Here our target is to understand DNA repair mechanism at a molecular level: especially the onset of repair, that is, damage recognition and binding by a repair enzyme. Further it is intended to find out concrete conditions on what kinds of DNA damages are difficult to repair. Now two kinds of complex damages are being dealt with in the simulation: double strand break (DSB) and clustered damages. DSBs have been believed to be significant from a viewpoint of radiation effect for a long time. While, the clustered damages have been paid attention in recent years. ICRP has taken the clustered damages for a theoretical basis of the Linear Non-Threshold (LNT) hypothesis of radiation effect. But detailed conditions have not been determined yet about DNA damages difficult to repair. From our study it was found that the DNA structure is largely distorted when two base damages exits at next sites on the DNA, suggesting that the difficulty of clustered damages would be attributed to the excess distortion of DNA.

A carcinogenesis model at a cellular level has been developed on the basis of the currently proposed multi-stage model where a normal cell becomes a tumor cell after several stage changes caused by a significant gene mutations[3]. The special feature of our model is to be able to follow the movement of a system consisting of many cells as a function of time. Physical properties of a cell are assumed to change according to the stage in the carcinogenesis process. The whole system alters towards lower energy state with a time step. The growth of tumors can be observed in the simulation using this model. In fact, this model is still primitive; nevertheless, some interesting results have been obtained like the one showing the tumor growth rate would be not proportional to the initial mutation rate. A response of a complex system to input would be often not linear, and this kind of analysis is necessary to elucidate the low-dose effect with consistency.

3.2 Experimental study

Two different studies on characterization of DNA damages are performed. The first study is paying attention on clustered damages. It is believed that radiation induces clustered damages where plural damages are located in a small region on DNA even by a single truck radiation. The clustered damages are considered to lead to significant biological effects. However, the relation of clustered damages to biological responses is not understood well. JAEA is experimentally investigating the biological response to clustered damages using some different approaches. Affinity of repair enzymes to clustered damages is investigated with *in vitro* experiment; the mutation frequencies by clustered damages are being examined with in vivo experiments using *E. coli* cells[4].

Another study is focusing on chemical structure of DNA damages. Chemical conformations of DNA damages by indirect effect are well known; while, those by direct effect are not perfectly known. The damages by direct effect might include unknown significant damages in terms of radiation effect.

From this viewpoint, structures of DNA damages are being analyzed using site-specific enzymes and chromatography[5].

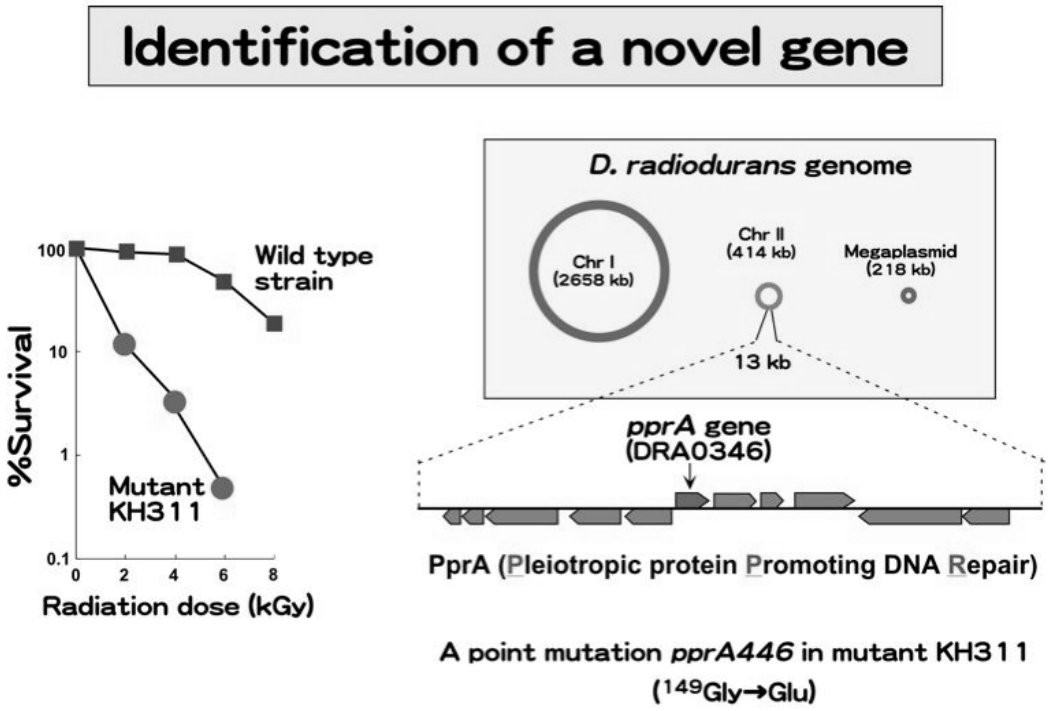


Figure 4 Identification of the novel gene pprA coding the protein working effectively for DNA repair in *Deinococcus radiodurans*

One interesting topic is a study on *Deinococcus radiodurans*. The bacteria are greatly resistant for radiation and other mutagens, since they can repair DNA damages quite effectively. Though the DNA repair mechanisms have been investigated by many researches; they are not thoroughly clarified. A group at Takasaki branch has been studying *Deinocossus radiodurans* using molecular biology. The left figure in Figure 4 shows a survival curve of *Deinococcus radiodurans*. As known from this figure, most of wild type *Deinococcus radiodurans* survive even if they receive 4 kGy dose. A kind of mutant KH311 has less radioresistance. From molecular biology analysis, a novel gene titled pprA which plays an important role in the DNA repair was found. Further, the functions of the protein PprA have been examined, and it was confirmed that it binds to damages DNA at high efficiency and controls DNA repair network[6].

4. Research on radiation protection

The main objective of radiation protection research at JAEA is to develop dosimetry techniques to ensure radiation safety of workers and the public. According to expansion of human activities, various new exposure conditions have come to exist, which cannot be covered by current radiation protection techniques and data. One important problem is high-energy radiations from accelerators and space activities as well as spallation nuclides produced by high-energy radiations. JAEA has developed dosimetry techniques considering these new conditions.

The subjects of protection studies performed at JAEA are listed in Table 2. Several of theses subjects will be presented in other papers. And, here I would like to pick up some subjects which will not be presented by specific papers.

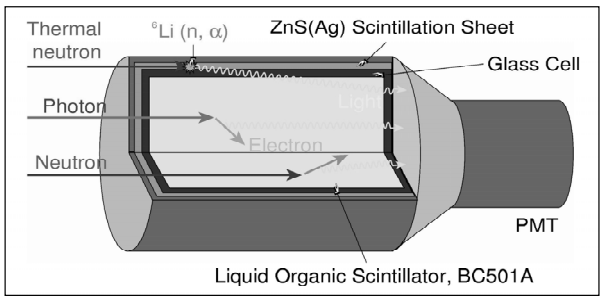
A multi-function radiation monitor is being developed[7]. This monitor utilized a phoswitch detector consisting of two different scintillation detectors, and the Data Storage Oscilloscope (DSO) based data analysis system as shown in Figure 5. The monitor can measure neutrons, photons and muons simultaneously using a pulse-height discrimination method, and give outputs in Sv utilizing the spectrum-dose weighting function method in the energy range up to 1 GeV. The prototype detector has been completed, and is now being upgraded to a commercial product.

In the J-PARC project, large-scale accelerators are under construction. The high-energy proton beams from the accelerators induce spallation radionuclides, and these could lead to internal exposure as shown Figure 6. Particle size distributions and chemical forms of the nuclides necessary for internal dose evaluation have been examined using the iron-beam accelerator TIARA and the research reactors JRR-3 and 4[8]. Experiment at J-PARC is planned in 2007 when the 3 GeV accelerator starts working. Appropriate parameters for internal dose evaluation will be selected based on these experimental results.

Table 2 Subjects of radiation protection studies at JAEA

1. External dosimetry
   a) Computational dosimetry for high-energy radiation
   b) Development of multi-functional radiation monitor
2. Internal dosimetry
   a) Characterization of spallation nuclides
3. Dosimetry for accidents
   a) Dose assessment system for criticality accidents
   b) Dosimetry based on ESR of teeth
   c) Retrospective dosimetry accidental intake of radioactivity
4. Basic studies for dosimetry
   a) Development of Japanese voxel phantoms
   b) Nuclear decay data for dosimetry
   c) Calibration facilities and techniques
5. Interdisciplinary studies
   a) Analysis of radiation fields for animal experiments
   b) Dosimetry considering radiosensitive cells

a) Phoswitch detector



b) DSO-based data analysis system

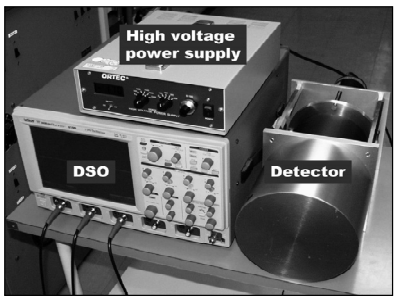


Figure 5 The developed multi-functional radiation motor able to measure neutrons, photons and muons simultaneously

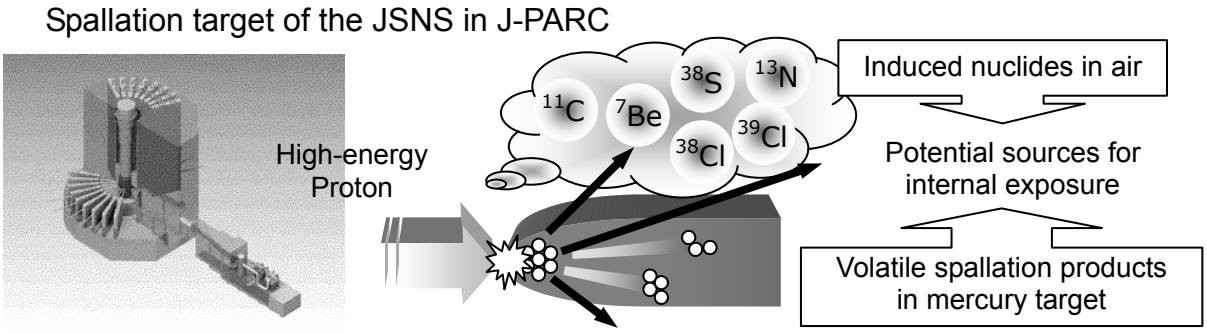


Figure 6 Induction of spallation nuclides by high-energy proton beams in J-PARC

In 1999, the JCO criticality accident happened in Tokai village. This accident initiated development of dose assessment systems for criticality accidents. There are two system having different functions. One is used for rough evaluation of the absorbed dose in the body based on Na-24 activity data[9]. This will help the prompt judgment if or not medical treatments are necessary. The system titled RADAPAS is designed to be user-friendly, and selecting conditions from several typical conditions shown on the PC monitor (Figure 7) would promptly give the evaluated dose. Another system is used for accurate dose evaluation using sophisticated simulation models including phantoms with removable limbs. The accurate dose evaluation data are expected to be utilized in medical treatments. The first system was already completed and available from outside. The second system will be further improved.

ESR dosimetry has been investigated in which a signal from tooth enamel is converted to the absorbed dose. In this study, conversion factors from ESR signal to organ doses were obtained using a mathematical phantom[10]. The simulation was verified by experiment using a physical phantom. Figure 8 shows examples of conversion factors for lung, stomach and bone surface. These data were adopted in IAEA and ICRU reports.

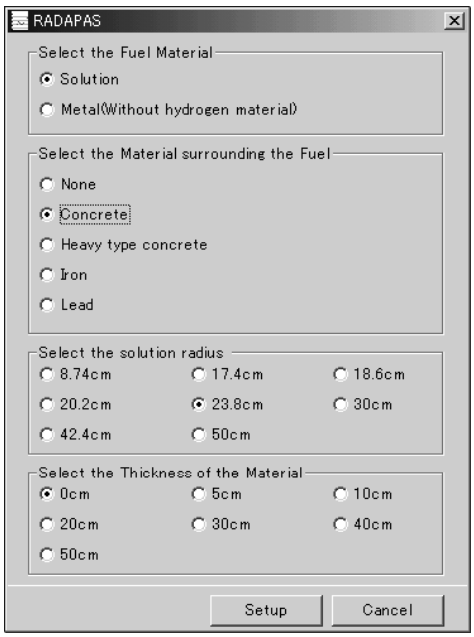


Figure 7 Monitor of RADAPAS

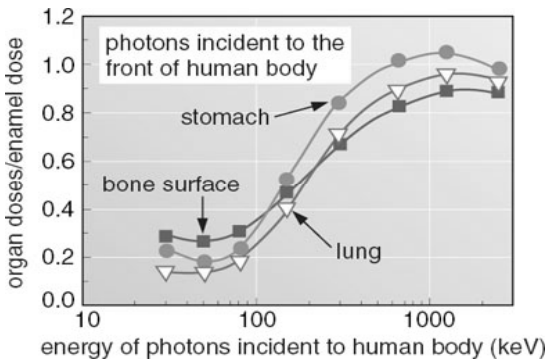


Figure 8 Dose conversion factors for ESR dosimetry

5. Application to medical use

The techniques developed in radiation effect studies and protection studies can be effectively applied to radiation therapy and diagnosis. Concerning radiation therapy, dose calculation systems have been developed for X ray therapy and Boron Neutron Capture Therapy (BNCT). At present in Japan BNCT is performed only at JAEA using the research reactor JRR-4, and the number of treatments is increasing rapidly. The development of accurate dose evaluation method is essential.

A system intended to remotely support radiotherapy using X rays has been developed in collaboration with other institutes being funded by JST (Japan Science and Technology Agency)[11]. In X ray therapy, usually commercial therapy planning systems are used for dose calculation. However, when the conditions are complex they do not give sufficiently accurate doses. Figure 9 gives the schematic representation of the developed system. The dose calculation techniques using voxel phantoms and Monte Carlo simulation have been applied to this system. In this system, the dose calculation is performed exclusively at the dose calculation center. First, the CT pictures of a patient and the treatment plan are transferred from the hospital to the dose calculation center. From the CT pictures, a voxel phantom is constructed immediately; the structure of the irradiation head is fully considered; and the Monte Carlo calculation is carried out utilizing parallel computing. Then, the calculated dose distribution is sent beck to the hospital, and is used for therapy planning

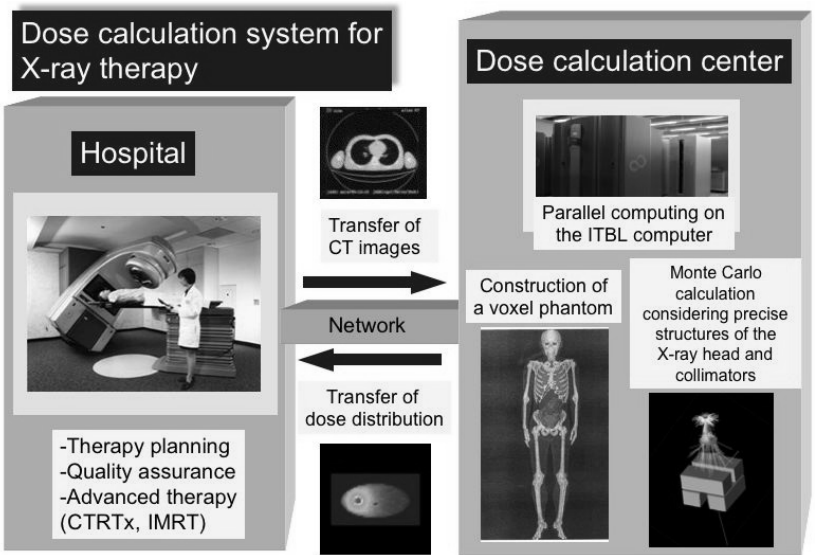


Figure 9 The developed dose calculation system IMAGINE for remotely support X ray therapy

6. Summary

According to the expansion of radiation application in various fields not only in nuclear energy production, studies on radiation effect and radiation protection including interdisciplinary studies are becoming important more and more. Exploiting the specific research potential as described in this paper, JAEA has performed original studies which are difficult to carry out at other institutes. Further, collaboration with other institutes has been promoted though being not mentioned much in this paper. Our researches on radiation effect and protection should be developed steadily considering these situations in future.

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1-2 BEIR VII: What's Old, What's New, and What Challenges Remain?

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Abstract

The Biological Effects of Ionizing Radiation (BEIR VII) Committee reviewed evidence since the 1990 BEIR V report and developed BEIR VII risk estimates, including a linear, no-threshold dose-response relationship between exposure to ionizing radiation and the development of cancer in humans for exposures up to 0.1 Sv, quantifying the lifetime risks for both cancer mortality and incidence as a function of age at exposure and sex, primarily based on the Japanese atomic-bomb survivor data. If 100 people with an age distribution typical of the U.S. population receive an acute exposure of 0.1 Sv, one person would be expected to eventually develop cancer from this exposure, while 42 of the 100 people would be expected to develop cancer from other causes. The committee estimated the risk following radiation exposure for both incidence and mortality for 11 specific cancer sites. The total risk of heritable genetic diseases from parents exposed prior to conception was 3,000 to 4,700 cases per million progeny per Sv, 0.4-0.6% compared to an estimated baseline risk of 738,000 cases per million. Noncancer diseases such as cardiovascular disease can result from exposures to high doses of radiation, but the data available at this time are not sufficient to develop reliable estimates of risk for these noncancer outcomes at low doses of radiation. Twelve specific recommendations were presented as needs for future research.

Keywords: BEIR VII, Radiation Risk Assessment, Radiation Biology, Radiation Carcinogenesis, Radiation Genetics, Noncancer Health Effects, Radiation Epidemiology

1. Introduction

A Phase I committee of the National Academies was asked to review all data since the BEIR V report was published in 1990[1)] and determine whether sufficient new information existed to warrant a Phase II BEIR VII study. In its 1998 report[2)], the Phase I committee recommended that there was sufficient new data for a BEIR VII committee to update risk estimates for exposures to low doses of low-LET ionizing radiation. In particular, that report pointed out that the US Department of Energy (DOE) had initiated a low-dose research program and a new dosimetry was being developed for the Radiation Effects Research Foundation (RERF). The abundant new data included a significant maturing of the RERF data analyses and cancer incidence data was now available from the RERF Life Span Study.

2. Materials and Methods

The 7th committee in a series of the Biological Effects of Ionizing Radiation (BEIR VII) was formed[1] and began a Phase II study in 1999. The committee held 11 meetings and received input from scientists and the public in 6 of those meetings. The BEIR VII report was released June 19, 2005 and the final report was published in January 2006[3)], more than 2 years later than initially anticipated primarily due to a delay in the finalization of the new dosimetry for RERF (DS02).

The primary task for the committee was to develop the best possible risk estimate for human exposure to low-dose, low-LET ionizing radiation. To do that, the committee was charged to conduct a comprehensive review of all relevant biological, physical, and epidemiological data since BEIR V. "Low dose" was defined as 0-0.1 Sv or less than 0.1 mGy/min over months or a lifetime**.** There is data in humans in this dose range since more than 60% of the A-bomb survivors received doses of less than 0.1 Sv. However, proposed regulation levels and levels of interest extend so low that endpoints such as cancer and mutations are not necessarily measurable with statistical significance. One would like to know the shape of the response curve in the low-dose region and several theoretical models have been proposed—including LNT, linear-quadratic, supralinear, hormetic, and threshold. The goal of BEIR VII was not to disprove or prove existing theoretical models; rather, it was to develop a model that best fits the physical, biological, and epidemiological data. The charge to the committee was not to recommend policy decisions related to regulations or radiation protection standards or to conduct cost/benefit analyses.

3. Results and Analysis

The committee found that there was considerable new data since BEIR V in 1990. RERF data had matured where there were now approximately 10,000 cancer deaths compared to 6,000 that were available to BEIR V. RERF cancer-incidence data was now available (13,000 cancers compared to 0 in BEIR V) and there was now evidence for non-cancer health outcomes (such as cardiovascular disease and stroke), albeit at higher exposure levels. Significant progress has been made related to estimating heritable effects of radiation as a result of advances in human molecular biology and it has become possible to project risks for more classes of genetic diseases such as those with more complex patterns of inheritance. Advances in cell and molecular biology have also contributed new information on mechanisms for responses to radiation-induced damage and to the close associations between DNA damage and cancer development.

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[1] BEIR VII committee members included: Richard R. Monson (Chair), James E. Cleaver (Vice-Chair), Herbert L. Abrams, Eula Bingham, Patricia A. Buffler, Elisabeth Cardis, Roger Cox, Scott Davis, William C. Dewey, Ethel S. Gilbert, Albrecht M. Kellerer, Daniel Krewski, Tomas R. Lindahl, Katherine E. Rowan, Krishnaswami Sankaranarayanan, Daniel W. Schafer, Leonard A. Stefanski (through May 2002), and Robert L. Ullrich. Consultants included John D. Boice, Jr., and Kiyohiko Mabuchi. Rick Jostes was study director. Donald A. Pierce served as research advisor.

3.1 Biological and Biochemical Findings

The committee reviewed several major biological advances since 1990. In particular, genetic influences related to the gain of function of certain genes and the loss of function of repressor genes have added understanding to mechanisms of carcinogenesis. Molecular pathways for repair and misrepair of DNA damage such as double-strand breaks (DSBs) have been elucidated and relationships between DNA DSBs, chromosome aberrations, and cancer have been revealed, including implications for genomic instability and telomere involvement. Relationships between locally multiply damaged sites (LMDS) and dose response have been characterized, especially comparing differences between DNA damage resulting from ionizing radiation damage compared to damage resulting from naturally occurring oxidation processes. There is more data available now related to dose and dose rate effectiveness factors (DDREF) and phenomena have been explored such as adaptive responses, bystander effects, and hyper radiation sensitivity (HRS). Those new biological advances are discussed in considerable detail in BEIR VII, including some assessment as to whether or not they should be expected to influence radiation-induced health effects at the low doses of interest in this study.

At low radiation exposures, the number of low-LET radiation traversals of cells should be proportional to the dose and the number of traversals can be very small at the lowest doses. But at the lowest doses, clusters of ionization events may occur. An ionization cluster near a DNA molecule may result in LMDS. So DNA damage resulting from even a single ionizing radiation traversal of a cell is expected to be potentially different from the biochemical damage resulting from normal oxidative processes. A sensitive biomarker of DNA damage and repair (γH2AX) has been identified and used to study changes in chromatin conformation from DNA DSBs, excision repair, and DNA replication. Rothkamm and Löbrich[4] have used this sensitive biomarker to examine the formation and repair of γH2AX foci in normal human cells at very low doses of ionizing radiation, illustrating that DSBs are formed as a linear function of dose down at least to background radiation dose levels and repair of most of the DSBs is complete by 24 hours. Since the BEIR VII committee's assessments, Löbrich and coworkers[5] have shown linearity for DSBs in patients in vivo after CT examinations at doses of 4.8-17.4 mGy.

3.2 Genetic Effects Findings

Radiation-induced heritable diseases have not been demonstrated in humans and studies based on 70,000 children of A-bomb survivors (RERF F1 studies) suggest that radiation doses less than 0.2 Gy are unlikely to double the risk of untoward pregnancies. Studies of nuclear workers' children have also not convincingly linked exposure to heritable diseases, ICRP 1999[6] estimated genetic risk at about 0.2% per Gy (or 1 case in 500 live births per Gy), and UNSCEAR 2001[7] estimated the "doubling dose" at about 1 Sv. Extensive data in mice, however, provide evidence for radiation-induced mutations in mammals. BEIR VII estimates a "doubling dose" using human data on spontaneous mutation rates of disease-causing genes and mouse data on induced mutation rates.

3.3 Epidemiological Findings

For epidemiological evidence, the BEIR VII committee turned to four major groups of

data—the RERF A-bomb survivor studies, the studies of occupational exposures (such as the 3-country pooled study of nuclear workers[8] and the UK National Registry of Radiation Workers study[9]), studies of medically exposed populations, and studies of environmental exposures (such as the populations exposed at Chernobyl, Semipalatinsk, or the Ural Mountains). It should be noted that the International Agency for Research on Cancer (IARC) 15-country pooled study of radiation workers by Cardis and her coworkers[10] was not available in time for thorough assessment by the committee but is discussed in an appendix of the BEIR VII report. In particular, the RERF data was used by BEIR VII because of the strengths of the A-bomb survivor studies; those strengths have been major reasons why the risk assessment work of RERF is often referred to as the "gold standard" for radiation epidemiology. Since BEIR V, RERF has developed an improved dosimetry system (DS02)[11,12], has 15 additional years of mortality follow-up, has cancer incidence data for both Hiroshima and Nagasaki, and has identified an association between non-cancer mortality and radiation exposures. BEIR VII mortality data is based on 10,127 solid cancer deaths (versus 5,588 in BEIR V) and 293 leukemia deaths (versus 202 in BEIR V), and survivors exposed at age 10 or 30 have now entered their most cancer-prone years. One of the strengths of the RERF studies is the range of individual radiation doses reconstructed for the survivors. It should be noted that 62% of the survivors received exposures in the low-dose range of 5-100 mSv. RERF's analyses by Preston and his coworkers[13] have shown that applying the new dosimetry (DS02), in which gamma doses increased slightly and neutron doses decreased in the range of interest, produces a slight decrease (~7%) in the previous RERF cancer risk estimates and has no appreciable impact on dose-response shape, gender risk differences, or age-time patterns. In 80,180 subjects with 2,083,988 person-years follow-up, 13,454 solid cancers have been recorded with an excess of 853 estimated for a radiation-attributable risk of 6.3%. The dose response for the solid cancer incidence does not provide convincing evidence for nonlinearity over weighted colon doses of 0-2 Sv.

3.4 BEIR VII Risk Model

The RERF Life Span Study (LSS) cohort played a principle role in the BEIR VII development of cancer risk estimates. Risk models were developed primarily from cancer incidence data for the period 1958-1998 and were based on DS02. Data from studies involving medical and occupational exposure were also evaluated. Models for estimating risks of breast and thyroid cancer were based on pooled analyses that included data on both the LSS cohort and medically exposed persons. To use models developed primarily from the LSS cohort for the estimation of lifetime risks for the U.S. population, BEIR VII makes assumptions regarding uncertainties such as the DDREF and the transport of risk estimates from the Japanese population. The committee's preferred estimates of the lifetime radiation-attributable risk of incidence and mortality are presented in BEIR VII for all solid cancers and for leukemia, and for males and females, along with 95% subjective confidence limits.

Figure 1 (taken from BEIR VII Figure ES-1) shows estimated excess relative risks (ERRs) of solid cancer versus dose. An example of how the data-based risk models can be used to evaluate the risk of radiation exposure is illustrated in Figure 2 (Taken from BEIR VII Figure PS-4). On average,

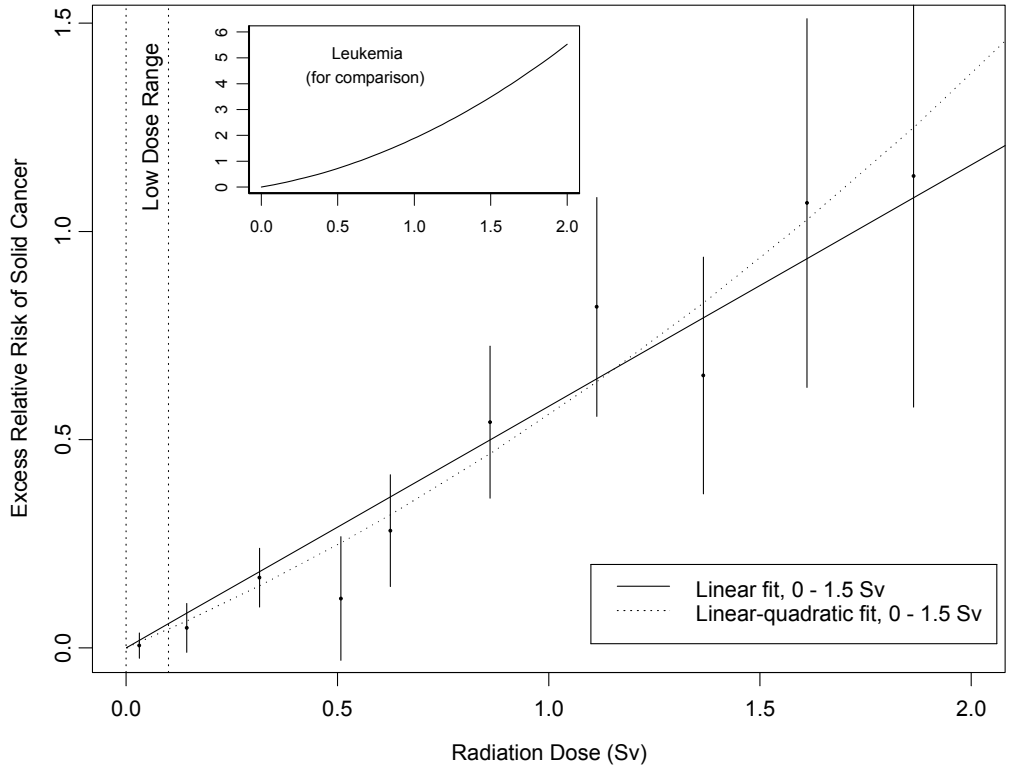

Figure 1 Excess relative risks of solid cancer for the Japanese atomic-bomb survivors. The plotted points are the estimated excess relative risks of solid cancer incidence (averaged over sex, and standardized to represent individuals exposed at age 30 and at attained age 60) for atomic-bomb survivors with doses in each of 10 dose intervals, plotted above the midpoints of the dose intervals. If R(d) represents the age-specific instantaneous risk at some dose d, then the excess relative risk at dose d is [R(d) – R(0)]/R(0) (which is necessarily zero when dose is zero). The vertical lines are approximate 95% confidence intervals. The solid and dotted lines are estimated linear and linear-quadratic models, respectively, for excess relative risk, estimated from all subjects with doses in the range 0 to 1.5 Sv. (These are not estimated from the points; but from the lifetimes and doses of the individual survivors, using statistical methods discussed in BEIR VII's chapter 6). A linear-quadratic model will always fit the data better than a linear model, since the linear model is a restricted special case with quadratic coefficient equal to zero. For solid cancer incidence, however, there is no statistically significant improvement in fit due to the quadratic term. It should also be noted that in the low-dose range of interest the difference between the estimated linear and linear-quadratic models is small relative to the 95% confidence intervals. The limiting slope of the dotted line gives a DDREF of 1.3. The insert shows the fit of a linear-quadratic model for leukemia, to illustrate the greater degree of curvature observed for that cancer.

assuming a sex and age distribution similar to that of the entire U.S. population, the BEIR VII lifetime risk model predicts that approximately 1 person in 100 would be expected to develop cancer (solid cancer or leukemia) from a dose of 100 mSv above background, while approximately 42 of the 100 individuals would be expected to develop solid cancer or leukemia from other causes. The BEIR VII report also presents example estimates for each of several specific cancer sites and for other exposure scenarios.

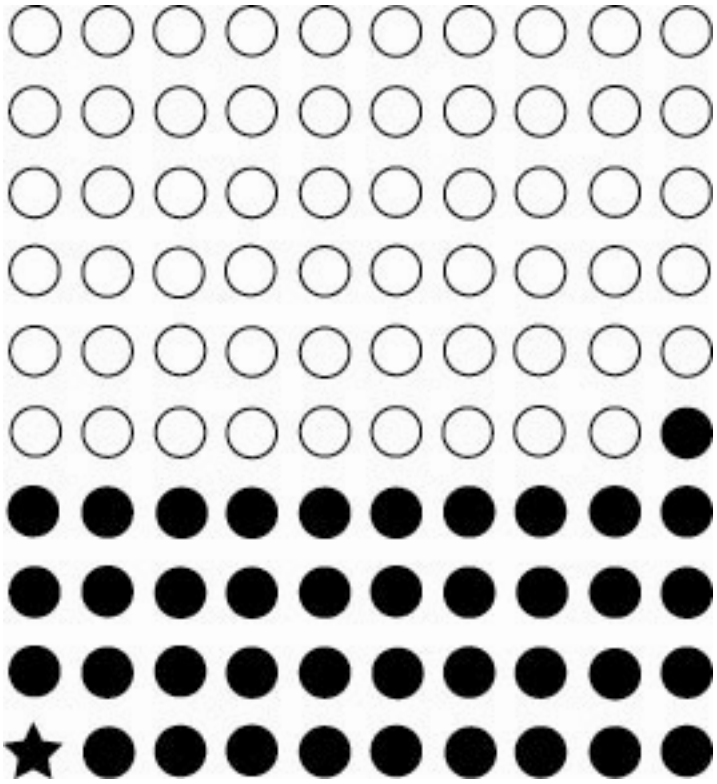

Figure 2 In a lifetime, approximately 42 (solid circles) of 100 people will be diagnosed with cancer from causes unrelated to radiation. BEIR VII calculations suggest that approximately one cancer (star) per 100 people could result from a single exposure to 0.1 Sv of low-LET radiation above background.

4. Discussion of Risk Estimates

4.1 Cancer Risks

Radiation induction of cancer is clearly significant at doses greater than 0.1 Sv for adults exposed to A-bomb radiations in Hiroshima and Nagasaki. Cancer is significant at doses greater than 10 mSv for children exposed in utero[14]. A linear-no-threshold (LNT) model represented a reasonable fit for solid cancers while a linear-quadratic model fit for leukemia. A Bayesian analysis produced estimates for a DDREF from 1.1-2.3; 1.5 was used in the risk analyses. The committee concluded that current scientific evidence is consistent with a LNT dose-response relationship between exposure to

ionizing radiation and the development of solid cancers in humans. ERRs and Excess Absolute Risks (EARs) were estimated for incidence and mortality and with respect to sex, age, and attained age—and for 11 specific cancer sites. In general, the magnitude of estimated risks for total cancer mortality has not changed drastically from past reports. The BEIR VII ERR per Gy is compatible with the estimates from the nuclear worker studies. BEIR VII risk estimates include explicit attention to the transport of risks to the US population and include quantitative evaluation of major sources of uncertainty. The risk estimates are compared to those derived in BEIR V and to other advisory groups.

4.2 Genetic Risks

BEIR VII estimated a "doubling dose" of 1 Sv using human data on spontaneous mutation rates of disease-causing genes and mouse data on induced mutation rates. Those estimates are 3,000-4,700 cases per 10^6 F1 children per Gy or 0.4-0.6% of a baseline of 738,000 cases in 10^6 of which chronic diseases are estimated to be about 650,000 per 10^6. BEIR V had estimated 2,400-5,300 cases per 10^6 F1 per Gy or 5-14% of baseline, but BEIR V did not include chronic diseases in the baseline.

4.3 Other Risks

BEIR VII concluded that radiation appears to increase the risk of diseases other than cancer and genetic risks, particularly cardiovascular disease, following high doses in therapeutic medicine and modest doses in A-bomb survivors. However, there is no direct evidence for increased risk at low doses and data are inadequate to quantify this risk with a model if it exists.

4.4 Research Needs

BEIR VII identified 12 research needs that are recommended for obtaining additional information that would improve understanding of radiation risk assessment. For example, the committee encouraged future medical radiation studies that should rely on exposure information collected prospectively, including cohort and nested case-control epidemiological studies. Those studies should explore effects of modifiers of radiation risk and gene-radiation interactions to provide information on potential sensitive subpopulations. Epidemiological studies were encouraged of persons receiving CT, especially children, infants receiving cardiac catheterization, those receiving recurrent exposures, and premature babies receiving repeated x-rays. It was suggested that there should be consideration of organizing a worldwide consortia for CT, PET, and SPECT data.

5. Conclusions

BEIR VII judged that the balance of evidence from epidemiologic, animal, and mechanistic studies tends to favor a simple, proportionate relationship at low doses between radiation dose and cancer risk. Uncertainties on this judgment are recognized and noted in the BEIR VII report. Current knowledge on adaptive responses, genomic instability, and bystander signaling among cells that may act to alter radiation cancer risk was judged to be insufficient to be incorporated in a meaningful way into the modeling of epidemiologic data at this time. The committee concluded that genetic variation in the population is a potentially important factor in the estimation of radiation cancer risk. But

modeling studies suggest that strongly expressing mutations that predispose humans to cancer are too rare to distort appreciably population-based estimates of risk, although they are a significant issue in some medical radiation settings. The BEIR VII report concludes that the current scientific evidence is consistent with a linear, no-threshold dose-response relationship between exposure to ionizing radiation and the development of cancer in humans, but notes that at low doses that risk will be small. And while adverse health effects have not been observed in the children of exposed parents, extensive data in mice suggest that there is no reason to believe that humans would be immune to this sort of genetic harm, but the risk is also low.

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1-3 Bystander Effect Studies using Heavy-ion Microbeam

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Abstract

We have established a single cell irradiation system, which allows selected cells to be individually hit with defined number of heavy charged particles, using a collimated heavy-ion microbeam apparatus at JAEA-Takasaki. This system has been developed to study radiobiological processes in hit cells and bystander cells exposed to low dose and low dose-rate high-LET radiations, in ways that cannot be achieved using conventional broad-field exposures. Individual cultured cells grown in special dishes were irradiated in the atmosphere with a single or defined numbers of 18.3 MeV/amu ^{12}C, 13.0 or 17.5 MeV/amu ^{20}Ne, and 11.5 MeV/amu ^{40}Ar ions. Targeting and irradiation of the cells were performed automatically according to the positional data of the target cells microscopically obtained before irradiation. The actual number of particle tracks that pass through target cells was detected with prompt etching of the bottom of the cell dish made of ion track detector TNF-1 (modified CR-39).

Keywords: Radiation biology, Heavy-ion microbeam, High-LET, Heavy charged particles, Bystander effect, Low-dose effects, Non-targeted responses, Single-cell/single-particle irradiation

1. Introduction

Heavy charged particles transfer their energy to biological organisms through high-density ionization along the particle trajectories. The population of cells exposed to a very low dose of high-LET heavy particles contains a few cells hit by a particle, while the majority of the cells receive no radiation damage. At somewhat higher doses, some of the cells receive two or more events according to the *Poisson* distribution of ion injections. This fluctuation of particle trajectories through individual cells makes interpretation of radiological effects of heavy ions difficult.

Using microbeams, we will be able to overcome this limitation by delivering a counted number of particles to each cell to study a number of important radiobiological processes in ways that cannot be achieved using conventional “broad-field” irradiation. A microbeam can be used for selective irradiation of individual cells, which can be subsequently observed to ascertain what changes occur to that cell and to neighboring un-irradiated cells. The use of microbeam allows direct investigation of cell-to-cell communications such as “bystander effects”, that is, radiation effects transmitted from hit cells to neighboring un-hit cells. Furthermore, a microbeam with sufficient spatial resolution will be useful for analyzing the interaction of damages produced by separate events in an irradiated cell, the dynamics of cellular repair, and the intra-cellular process such as apoptosis by means of highly localized irradiation of a part of a nucleus or cytoplasm.

The earliest heavy particle microbeam experiments were performed in 1953 using a 2 MV Van de Graaff accelerator and micro collimators to form a proton microbeam to study the process of cell division.[1)] They used two metal plates, with a groove on one, clamped together to make apertures about 2.5 μm in diameter. They also used variable microapertures formed of cross slits, which could be adjusted to any desired width from 0.5 μm to 5 mm. Later, an 11 MeV/amu proton and 22 MeV/amu deuteron microbeam was developed at Brookhaven National Laboratory using a cyclotron. Beams as small as 25 μm were used to investigate the effects of local radiation damage caused by high-energy deuterons to different cells within mouse-brain tissue.[2)] These earlier microbeam systems were very helpful in studying radiation effects in living systems, especially to show that damage to the cytoplasm had a very limited effect on the survival of the cell. However, all of these systems were limited at relatively high doses. To investigate the effects at lower doses, it is necessary to establish single particle irradiation technique.

One microbeam system designed for this purpose was installed on the horizontal beam line of the UNILAC linear accelerator at GSI-Darmstadt where ions of many elements, ranging from carbon to uranium, with energies of 1.4 MeV/amu were available.[3)] Etched tracks of high-energy heavy particles were used to collimate beams. Using this system, the impact parameter dependence of the inactivation of *Baccilus subtilis* spores were measured.[4)] This experiments yield inactivation probabilities of central hits between 40 and 80% depending on the LET and atomic number of the particles showing that the zones of high local ionization seem to be the most important problem causing the low biological efficiency of very high-LET radiation.

The next single-particle microbeam was developed at Pacific Northwest Laboratory.[5)] An electrostatic accelerator was used to produce hydrogen and helium ions. Two sets of four adjustable knife-edges were used to construct two apertures in series to collimate microbeams. A thin plastic scintillator and photomultiplier were used to detect individual particles. Using this system, CHO-K1 cells were exposed to controlled number of 3.2 MeV α-particles and the biological responses of individual cell were quantified.[6)] However, this microbeam apparatus has been removed and reinstalled at Texas A&M University.

Similar single-cell/single-particle irradiation systems have been developed at the Gray Cancer Institute (GCI)[7,8)] in UK, at Columbia University in USA (RARAF: Radiological Research Accelerator Facility) [9)], at the National Institute of Radiological Sciences (SPICE: Single Particle Irradiation

system to CEll)[10] and at the TIARA (Takasaki Ion Accelerators for Advanced Radiation Application) of JAEA-Takasaki, Japan. These are truly operational particle microbeams where single protons and helium ions can be aimed at single cells with a few microns resolution. Using a microbeam at GCI, Prise *et al*. demonstrated that targeting individual α-particles to four cells within a population produced more micronucleated and apoptotic cells than expected on the basis of a direct effect only.[11] It was also reported that when a single cell within a population was targeted by an α-particle, typically an additional 80-100 damaged cells were observed in surrounding population of about 5000 cells.[12] This bystander effect was observed when only a cell was targeted, but not when only the medium was exposed, confirming that a cell-mediated response is involved. At the RARAF microbeam facility, a bystander mutagenic effect has been found in non-traversed cells when a proportion of mammalian cells have suffered a precise number of nuclear traversals by α-particles.[13,14] Furthermore, it was found that targeted cytoplasmic irradiation, of all cells within the population with α-particles, induced mutations in mammalian cells suggesting that cytoplasm is an important target for cellular killing and mutation.[15] Recent studies with the GCI microbeam have also shown that targeting the cytoplasm induces bystander responses to the same extent as that observed from nuclear irradiation.[16]

Protons, helium-3 ions, and α-particles are currently used to study the microbeam irradiation-induced bio-responses. Therefore, we have developed a novel single-cell/single-particle irradiation system using heavy-ion microbeams for targeting cells individually with a specific numbers of particles to elucidate the radiobiological effects of a single high-LET particle traversal. Compared to the single-cell irradiation facilities that use mainly light ions like protons and helium ions, the range of the LET can be extended considerably with heavy ions. Furthermore, higher energies allow larger penetration and better lateral resolution for the microbeam irradiation procedure. Accordingly, there is increased effort to develop heavy charged particle microbeam for single-cell irradiation. Besides our currently using collimated heavy particle microbeam and being developed “second generation” focusing high-energy heavy-ion microbeam[17] at JAEA-Takasaki, at least two facilities have been developed; at the Munich 14 MV tandem accelerator (SNAKE: Superconducting Nanoscope for Applied nuclear (Kern-) physics Elements) [18], and at GSI-Darmstadt.[19]

2. Experimental setup

The cell irradiation system has been incorporated into the collimated heavy-ion microbeam apparatus, which was installed below a vertical beam line of the AVF cyclotron at TIARA in JAEA-Takasaki. The heavy-ion beams delivered from the AVF cyclotron are collimated with a set of apertures. Then the collimated beams are extracted into air through a microaperture on a 100 μm -thick tantalum disk perforated using an electrical discharge machining (spark erosion) method. The smallest microaperture, 5 μm in diameter, was used for cell irradiation with a precise number of 11.5 MeV/amu ^{40}Ar and 13.0 MeV/amu ^{20}Ne ions; and a microaperture of 20 μm in diameter was used for 18.3 MeV/amu ^{12}C ions irradiation.

So far, two inverted optical microscopes are in operation with this system. One of the microscopes is installed below the vertical beam line in the beam room as an “on-line microscope” for cell targeting and for delivery of a certain number of particles. The other microscope, which is called

"off-line microscope", is used in the preparation room for cell finding prior to the irradiation and for cell revisiting and observation during post-irradiation incubation. A local-area-network connects these control systems allowing the object database created at the off-line microscope to be used by the cell-targeting system.

Preparation of the microbeam target-cell dishes and the microbeam irradiation protocol used have been described elsewhere.[20,21] Figure 1 shows the procedure of targeted irradiation of cultured cell with a heavy particle microbeam. Briefly, cells grown in special dish made of ion track detector TNF-1 (modified CR-39) are positioned so that the desired portion of the cell aligns with collimator. The number of ions penetrating the sample is counted with a constant fraction discriminator coupled to a preset counter/timer. A pulse-chopper in the injection line of the cyclotron was used as fast beam switch. The gate output of the counter/timer was fed to the pulse-chopper to turn on the beam until the chosen number of ions had detected. The actual number of particle tracks that passed through cell nuclei was detected with prompt etching of the bottom of the cell dish with alkaline-ethanol solution at 37ºC for 15-30 minutes. After that, the phase-contrast microscopic image of the irradiated cells was overlaid with the image of the etched ion pits obtained at the same field of view. It is possible to revisit each irradiated cell reproducibly during post-irradiation incubation according to the object database.

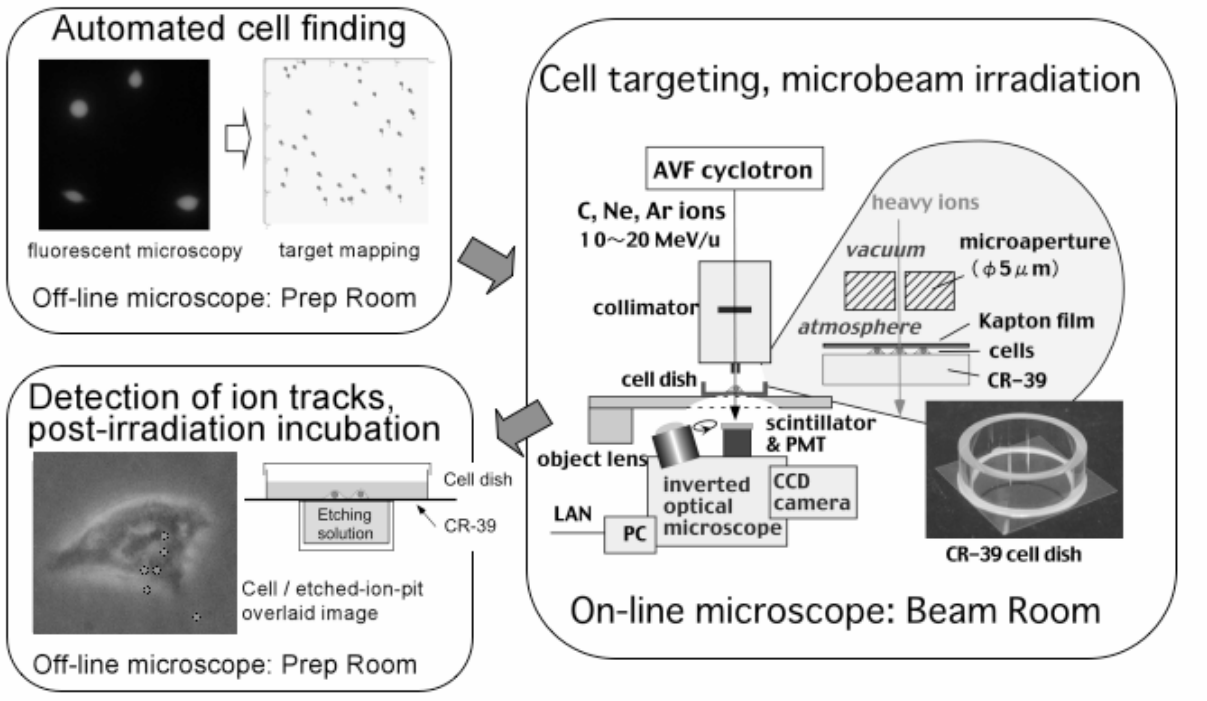

Figure 1 Procedure of targeted irradiation of cultured cells with a heavy-ion microbeam.
Before irradiation, positional data of the individual cells is obtained at the off-line microscope in the preparation room by microscopically searching the cell dish. Using the object database, targeting and irradiation at the on-line microscope are quickly carried out. Immediately after irradiation, the cell dish is refilled with medium, and then the bottom is etched from the opposite side of the cells to detect the accurate position of ion tracks on the cells.

3. Results and Discussion

3.1 Direct nucleus-hit effect and bystander effect on the growth of sparsely inoculated CHO-K1 cells

Chinese hamster ovary (CHO-K1) cells were irradiated individually with counted number of 11.5 MeV/amu ^{40}Ar ions (LET 1,260 keV/μm) using our heavy particle microbeam.[20,21] After irradiation, the effect of direct hit of heavy charged particles was estimated by scoring the number of cells in each colony up to 60 hours. We observed a reduced number of cells per colony in the direct hit cells after 60 hours post irradiation incubation. This reduction of cell number per colony was due to reproductive death of the hit cell, rather than from cell division delay caused by radiation damage. Among the irradiated cells, the cells hit in their nucleus with only one ^{40}Ar ion showed strong growth inhibition, and the percentage of lost cells (detached cells) was increased to more than 40%. The detached cells showed morphological changes within 12-24 hours after irradiation.

In addition, an inhibitory effect on the non-hit cells was also observed. In the non-irradiated cell dish, where no cells in the dish were irradiated but the medium was irradiated, up to 25 cells per colony were observed. Meanwhile, cells in the irradiated dish, which were not irradiated when some of the co-cultured cells in the same dish were irradiated, showed limited cell growth resulting in approximately 12 cells per colony. An increased yield of lost cells was also observed in this cell group. This growth inhibition in the non-hit cells in the cell dish containing co-cultured hit and non-hit cells might be caused by a bystander effect. It had been reported that two different pathways mediate the intercellular signaling of the bystander effects: gap junction[22-24] and medium-mediated molecules.[25-27] In our experiments, the cells were inoculated sparsely in the sample holder, thus medium-mediated molecules may cause this limited growth of the non-hit cells.

3.2 Bystander effect induced by counted high-LET particles in confluent human fibroblasts

Primary human fibroblast (AG01522) cells within a confluent population were individually targeted by a high-LET heavy particle microbeam of 13.0 MeV/amu ^{20}Ne or 11.5 MeV/amu ^{40}Ar with LET values of 380 keV/μm and 1,260 keV/μm, respectively.[23] Even when only a single cell within the confluent culture was hit by one 20Ne or 40Ar particle, a 1.4-fold increase in micronuclei (MN) was detected demonstrating a bystander response. When the number of targeted cells increased, the number of MN biphasically increased; however, the efficiency of MN induction in binucleated (BN) cells per targeted cell markedly decreased. When 49 cells in the culture were individually hit by 1 to 4 particles, the production of MN in the irradiated cultures were ~2-fold higher than control levels but independent of the number and LET of the particles. MN induction in the irradiated culture was partly reduced by treatment with DMSO, a scavenger of reactive oxygen species (ROS), and was almost fully suppressed by a mixture of DMSO and PMA, an inhibitor of gap junction intercellular communication (GJIC). Accordingly, both ROS and GJIC contribute to the above-mentioned bystander response and GJIC may play an essential role by mediating the passage of soluble biochemical factors from targeted cells. To clarify the mechanisms of transduction of bystander signal through the GJIC, dose dependency and LET dependency of bystander effects were studied using this system. Efficiency to produce MN in BN cells were not affected by the number of traversals at the cell

in each position for higher LET particles; i.e., ^{20}Ne or ^{40}Ar ions. When cells were irradiated with 18.3 MeV/amu ^{12}C ion beams having LET of 120 keV/μm, clear particle dependent increment of the MN induction was found up to 6 particles at each position.[28)]

However, GJIC is not always required for the induction of bystander responses. By irradiating non-confluent human fibroblast cells with a precise number of particles, the findings[29)] that targeting of a single cell led to additional 10s of cells being damaged gives direct evidence of non-GJIC involvement and the likelihood of medium-mediated bystander responses. To clarify the mechanisms of transduction of the bystander signal through GJIC, an irradiation method to obtain the distance distribution in the induction of micronuclei in non-irradiated cells from irradiated cells was established by Furusawa *et al*.[30)] Briefly, a 25 mm diameter cell culture cover-slip (NUNC 174985) was attached to the center of a 60 mm plastic dish over a 13 mm hole in bottom of the dish and sealed with white Vaseline. In the middle of the coverslip, a 1-2 mm width of adhesive tape was put on the surface to prevent cells attaching to this “clear zone”, which split the surface of the cover slip into two “confluent zones”. Only a fraction of the cells in the confluent zone in one side of the clear zone were irradiated. Micronuclei (MN) can only be seen in growing cells at the border between the clear zone and the confluent zone. The distributions of the distances over which the bystander effects occurs from the irradiated cells can be obtained by measurement of the distribution of MN cells in the border of the clear zone in the irradiated side. If there are still medium-mediated bystander effects, even if suppressed by inhibitors, we can observe their effect in the opposite side of the irradiated side.

4. Conclusion

In the last ten years, there has been a rapid growth in the number of centers developing microbeams for radiobiological research, and worldwide there are currently about 30 microbeams in operation or under development.[31)] Most of the recent works using microbeams has been to study low-dose effects and “non-targeted” responses, such as bystander effects, genomic instability and adaptive responses. In JAEA, we have developed single-cell/single-particle irradiation methods using collimated energetic heavy-ion microbeams. In addition, a procedure for detection of an ion-hit track within the beam time has been established. With this JAEA-Takasaki heavy-ion microbeam probe, the radiation response in individual cells irradiated with high-LET heavy particles can be analyzed in detail by single-cell assays, and then a cytomolecular biological analysis of irradiated cells can be performed.

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1-4 Modifying EPA Radiation Risk Models Based on BEIR VII

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Abstract

This paper summarizes a "draft White Paper" that provides details on proposed changes in EPA's methodology for estimating radiogenic cancer risks. Many of the changes are based on the contents of a recent National Academy of Sciences (NAS) report (BEIR VII), that addresses cancer and genetic risks from low doses of low-LET radiation. The draft White Paper was prepared for a meeting with the EPA's Science Advisory Board's Radiation Advisory Committee (RAC) in September for seeking advice on the application of BEIR VII and on issues relating to these modifications and expansions. After receiving the Advisory review, we plan to implement the changes by publishing the new methodology in an EPA report, which we expect to submit to the RAC for final review. The revised methodology could then be applied to update the cancer risk coefficients for over 800 radionuclides that are published in EPA's Federal Guidance Report 13.

Keywords: Ionizing radiation, Health effects, Atomic bomb survivors, Risk model, Risk estimate, Cancer, Radiation protection

1. Introduction

In 1994, EPA published a report, referred to as the "Blue Book," which lays out EPA's current methodology for quantitatively estimating radiogenic cancer risks.[1)] A follow-on report made minor adjustments to the previous estimates and presented a partial analysis of the uncertainties in the numerical estimates.[2)] Finally, the Agency published Federal Guidance Report 13 (FGR-13), which used the previously published cancer risk models, in conjunction with ICRP dosimetric models and U.S. usage patterns, to obtain cancer risk estimates for over 800 radionuclides, and for several exposure pathways.[3)]

The National Research Council (NRC) of the National Academy of Sciences (NAS) recently released a report on the health risks from exposure to low levels of ionizing radiation.[4)] Cosponsored by the EPA and several other Federal agencies, *Health Risks from Exposure to Low Levels of Ionizing Radiation BEIR VII Phase 2* (BEIR VII) primarily addresses cancer and genetic risks from low doses of low-LET radiation.

This paper outlines some proposed changes in EPA's methodology for estimating radiogenic cancers, based on the contents of BEIR VII and some ancillary information. The paper is excerpted from a "draft White Paper" (WP),[5] which we prepared for a meeting with the EPA's Science Advisory Board's Radiation Advisory Committee (RAC) in September for seeking advice on the application of BEIR VII and on issues relating to these modifications and expansions. After receiving the Advisory review, we plan to implement changes in our methodology through the publication of a revised Blue Book, which we would expect to submit to the RAC for final review. The revised Blue Book could then serve as a basis for an updated version of FGR-13.

2. Radiation Cancer Risk Models

2.1 Current EPA Risk Models

For most cancer sites, radiation risk models are generally derived from epidemiological data from the life span study (LSS) of the atomic bomb survivors. EPA's models for esophagus, stomach, colon, lung, ovary, bladder, leukemia, and "residual" cancers were adapted from the models published by Land and Sinclair based on a linear, no-threshold fit to the LSS data.[6] For each solid tumor site, gender, and age-at-exposure interval, there is a model providing a coefficient for the excess relative risk (ERR) per Gy for cancer mortality, which is assumed to be constant beginning at the end of a minimum latency period until the end of life. Land and Sinclair present two sets of models – "multiplicative" and "NIH "--- differing in how one "transports" risk from the Japanese LSS population to another population, e.g., to the U.S. population. For the multiplicative model, it is assumed that the ERR/Gy is the same in all populations, whereas, for the NIH model, it is assumed that the excess absolute risk is the same in different populations *for the limited period of epidemiological follow-up*. Given the scarcity of information on how radiogenic cancer risk varies between populations having differing baseline cancer rates, EPA adopted an intermediate model for each site, where the ERR coefficients were taken to be the geometric mean of the corresponding ERR coefficients for the multiplicative and NIH models.[1]

For leukemia, the temporal response in the models was more complex, but the approach for transporting risk to the U.S. population was analogous. Following the approach of Land and Sinclair, EPA also developed a GMC model for kidney from the LSS data. EPA's models for other sites, including breast, liver, thyroid, bone, and skin were based on various authoritative reports.[7–11] Based primarily on ICRP recommendations at that time, for low doses and dose rates, each coefficient was reduced by a factor (DDREF) of 2 from what would be obtained from a linear, no-threshold fit to the LSS data.

2.2 BEIR VII Models

BEIR VII site-specific models derived from the LSS differ from those of Land and Sinclair in several significant ways: (1) they are derived primarily from data on cancer incidence rather than cancer mortality; (2) mathematical fitting is performed to better reflect the functional dependence of solid cancer risk on age at exposure and attained age; (3) a weighted average of risk projection models was used to transport risk from the LSS to the U.S. population; (4) a value for the DDREF of 1.5 was

estimated from the LSS and laboratory data; (5) quantitative uncertainty bounds are provided for the site-specific risk estimates in BEIR VII.

For breast cancer and thyroid cancer, BEIR VII risk models were based on pooled analyses of data from the LSS cohort, together with data on medically irradiated cohorts.[12, 13)]

2.3 Proposed EPA Additions and Modifications to BEIR VII Models

In implementing its revised methodology for estimating radiogenic cancer risks, EPA proposes to adopt many of the recommendations in BEIR VII. One significant extension to be considered is the estimation of risks from exposures to higher LET radiations, especially to alpha particles, but also to lower energy photons and beta particles. Particularly important in this regard is the risk from alpha emitters deposited in the lung and the bone. BEIR VII presents no risk estimates for radiogenic bone cancer. As in the past, we propose to estimate bone cancer risk from data on radium injected patients.

BEIR VII also fails to provide quantitative estimates of risk for skin cancer, both of which might be significant under some exposure conditions. Risks from prenatal exposures are also not fully addressed by the report. BEIR VII presents a model for estimating radiogenic thyroid cancer incidence, but not thyroid cancer mortality. We hope to address these gaps and to consider the findings of an EPA sponsored thyroid report being drafted by the NCRP, when it becomes available.

The WP provides a somewhat detailed discussion of proposed modifications to risk estimates for alpha particle radiation, skin cancer and prenatal exposures. The next section outlines proposed modifications in the applying the BEIR VII risk models. We intend to employ somewhat different population statistics than BEIR VII. For breast cancer, an alternative method is introduced for estimating mortality, which takes into account changes in incidence rates and survival rates over time.

BEIR VII provides quantitative uncertainty bounds for each of its risk coefficients. Nevertheless, in deriving these bounds, it is clear that some sources of uncertainty were not included. Most important, no uncertainty was assigned to the form of the dose-response relationship: it was implicitly assumed that the dose-response relationship is “linear-quadratic”, which allowed the BEIR VII Committee to place uncertainty bounds on the “DDREF”. Mechanisms pertaining to the biological effects of low level ionizing radiation are being investigated, which could eventually mandate a different dose-response model, with resulting large changes in estimates of risk at low doses. Assigning probabilities to alternative models would be highly subjective at this time. We do not propose to quantify the uncertainty pertaining to low-dose extrapolation, beyond what was done in BEIR VII, but we would expect to include a brief discussion of the issue in our revised risk assessment document.

3. Proposed Methods for Projection Radiogenic Risk to the U.S. Population

3.1 Calculating Lifetime Attributable Risk

As in BEIR VII, we propose using lifetime attributable risk (LAR) as our primary risk measure. For a person exposed to dose (x) at age (e), the LAR is given by:

$$LAR(x,e) = \int_{e+L}^{110} M(x,e,a) \cdot S(a)/S(e)da \,. \qquad (1)$$

where

$M(x,e,a)$ is the excess absolute risk at attained age a from an exposure at age e, $S(a)$ is the probability of surviving to age a, and L is the latency period (2 years for leukemia, 5 years for solid cancers). The LAR approximates the probability of a premature cancer death from radiation exposure, and in BEIR VII (approximate) values for the LAR are obtained as weighted sums (over attained ages a up to age 100) of the excess probabilities of radiation-induced cancer incidence or death, $M(x,e,a)$. We intend instead to calculate the integral (Eq. 1) to age 110 (or perhaps 120) using spline approximations – not unlike the approach used to calculate EPA's current risk coefficients.[3)]

The LAR for a population is calculated as a weighted average of the age-at-exposure specific risks discussed above. The weights are proportional to the number of people, $N(e)$, who would be exposed at age e. The population-averaged LAR is given by:

$$LAR(x,pop) = \frac{1}{N^*} \int_{0}^{110-L} N(e) \cdot LAR(x,e) \cdot de \,. \qquad (2)$$

For the BEIR VII approach, $N(e)$ is the number of people from census data in the U.S. population at age e for a reference year – BEIR VII used 1999, and N^* is the total summed over all ages. In contrast, for our primary projection, we propose to use a hypothetical stationary population for which the $N(e)$ are proportional to $S(e)$ based on observed mortality rates for the year 2000. Under the assumption that there would be no appreciable change in future mortality rates, this would approximate the radiogenic risk from a lifetime (chronic) exposure at constant dose rate. A stationary population is being used for our current risk assessment.[3)]

3.2 Solid Cancer Incidence

For most cancer sites, separate evaluations of LAR were made using both an excess absolute risk (EAR) model and an excess relative risk (ERR) model. For most solid cancers (all but thyroid, and breast cancer), the ERR and EAR models were based exclusively on analyses of the atomic bomb survivor incidence data. This differs from the risk models that had been used in previous risk assessments, and had been derived from mortality data.

Except for breast and thyroid cancers, the preferred BEIR VII EAR and ERR models are functions of sex, age at exposure, and attained age, and were of the form:

$$\text{EAR}(x,e,a) \text{ or } \text{ERR}(x,e,a) = \beta_s D \exp(\gamma e^*)(a/60)^{\eta} \,,$$

where $e^* = \frac{\min(e,30) - 30}{10}$.

The values for the parameters β_s, γ, and η depend on the type of model – EAR or ERR. For ERR models for most sites:

β, the ERR per Sv at age-at-exposure 30 and attained age 60, tends to be larger for females than males;

γ = -0.3 implies the radiogenic risk of cancer at age *a* falls by about 25% for every decade increase in age-at-exposure up to age 30; and

η = -1.4 implies the ERR is almost 20% smaller at attained age 70 than at age 60.

Thus, ERR decreases with age-at-exposure (up to age 30) and attained age. In contrast, for EAR models for most sites, γ = -0.41 and η = 2.8. EAR decreases with age-at-exposure but increases with attained age. These patterns are illustrated in BEIR VII (Figure 12–1A, p. 270).

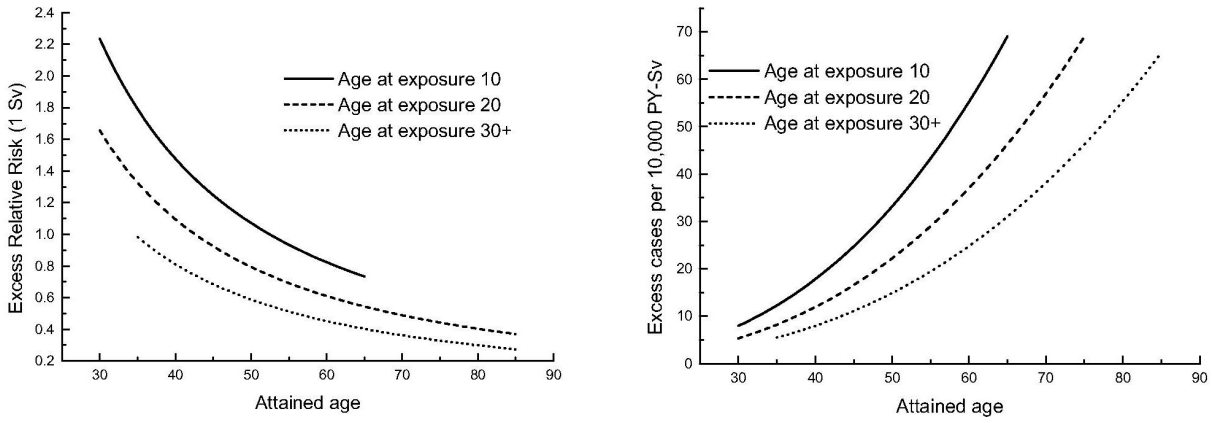

Figure 12–1A Age-time patterns in radiation-associated risks for solid cancer incidence excluding thyroid and nonmelanoma skin cancer. Curves are sex-averaged estimates of the risk at 1 Sv for people exposed at age 10 (solid lines), age 20 (dashed lines), and age 30 or more (dotted lines). Estimates were computed using the parameter estimates shown in Table 12-1 of BEIR VII.

For either type of model, calculating the LAR is relatively straightforward. For the EAR models, note that $M(x,e,a) = \mathrm{EAR}(x,e,a)$. For ERR models,

$$M(x,e,a) = ERR(x,e,a) \cdot \lambda_I(a),$$

where $\lambda_I(a)$ is the baseline cancer incidence rate at age *a*. Values for LAR are then obtained using equations 1 and 2.

Results of LAR calculations for selected cancer sites are given in Table 1. Separate

calculations were made using both census data – weights proportional to the number of people of each age in the year 2000, and a stationary population – based on mortality data for the year 2000. For most sites, the LAR is about 5-10% larger when based on weights from census data.

Table 1 Comparison of the impact of two methods for age-averaging on LAR for solid cancer incidence for selected sites. Projections* are made using the BEIR VII EAR and ERR models. Age-averaging is based on either 2000 census data[14] or a stationary population constructed from 2000 life tables.[15]

		Risk Model Population Weighting			
		EAR		**ERR**	
Site	**Sex**	**Census**	**Stationary**	**Census**	**Stationary**
Stomach	Male	278	259	23	22
	Female	328	308	30	29
Colon	Male	182	169	256	240
	Female	107	100	164	155
Lung	Male	189	179	246	230
	Female	361	344	767	714
Breast	Female	463	423	507	465
Ovary	Female	47	44	75	69
Bladder	Male	121	115	170	160
	Female	101	96	165	155
* Number of cases per 100,000 persons exposed to 0.1 Gy. No adjustment for DDREF.					

3.3 Solid Cancer Mortality

The ERR and modified versions of the EAR models just discussed were used in BEIR VII to calculate LAR for radiation-induced cancer death. For ERR, the same models were used for both incidence and mortality,

$$M(x,e,a) = ERR(x,e,a) \cdot \lambda_M(a).$$

For EAR, BEIR VII used essentially the same approach by assuming

$$M(x,e,a) = \frac{EAR(x,e,a)}{\lambda_I(a)} \lambda_M(a) \quad . \quad (3)$$

Note that the ratio of age-specific EAR to incidence rate is the ERR for incidence – based on the EAR

model. Equations (1) and (2) are then applied to obtain the LAR. This BEIR VII approach, equating the incidence and mortality ERR, ignores the "lag" between incidence and mortality, which could lead to bias in the estimate of mortality risk in at least two different ways.

First, there would be a corresponding lag between the ERR for incidence and mortality, which might result in an underestimate of mortality risk. For purposes of illustration, suppose that a particular cancer is either cured without any potential life-shortening effects or results in death exactly 10 years after diagnosis, and that survival does not depend on whether it was radiation-induced. Then, with subscripts *M* and *I* denoting mortality and incidence:

$$ERR_M(x,e,a) = ERR_I(x,e,a\text{-}10) > ERR_I(x,e,a).$$

The same relationship would hold for EAR, if the baseline cancer rate has the same age-dependence for A-bomb survivors as for the U.S. population.

Second, since current cancer deaths often occur because of cancers that develop years ago, application of the EAR-based ERR for incidence can result in a substantial bias due to birth cohort effects. If age-specific incidence rates increase (decrease) over time, the denominator in Equation 3 would be too large (small). This could result in an underestimate (overestimate) of the LAR.

The BEIR VII approach is reasonable for most cancers, because the time between diagnosis and a resulting cancer death is typically short. An exception is breast cancer, for which we propose an alternative approach with details given in the White Paper. The alternative method for calculating LAR for breast cancer mortality is based on a formula with inputs that include estimates of: (1) age-specific radiogenic breast cancer rates; (2) probabilities of survival from age of exposure to age of cancer incidence; (3) probabilities of survival from cancer diagnosis to age at which death may occur; and (4) breast cancer death rates for breast cancer patients. Preliminary calculations indicate that the projected LAR would be about 30-40% larger using the proposed alternative approach. Much of the discrepancy between the two sets of results seems to be a consequence of observed increases in breast cancer incidence rates.. From 1980 to 2000, age-averaged breast cancer rates increased by about 35% (102.1 to 135.7).[16] The proposed alternative method has limitations. The validity of the projection would depend, for example, on the validity of estimates of (birth-cohort dependent) breast cancer death rates.

3.4 Uncertainty Results and Analysis

The BEIR VII Report includes a quantitative uncertainty analysis with 95% subjective CIs for each site-specific risk estimate. The analysis focused on three sources of uncertainty thought to be most important: (1) sampling variability in the LSS data, (2) the uncertainty in transporting risk from the LSS to the U.S. population, and (3) the uncertainty in the appropriate value of a DDREF for projecting risk at low doses and dose rates from the LSS data.

The BEIR VII analysis neglected other sources of uncertainty, including: (1) errors in dosimetry; (2) errors in disease detection and diagnosis; (3) uncertainty in the age and temporal pattern of risk, especially for individual sites, which was usually taken to be the same as that derived for all

solid tumors; (4) uncertainty in the relative effectiveness of medical x rays in inducing cancer for those sites where data on medically irradiated cohorts were used in deriving the risk models.

It should also be noted that the treatment of uncertainty in projecting risk at low doses and dose rates of low-LET radiation, basically *assumes* the "linear-quadratic" dose-response model in which: (1) the risk from an acute dose, D, is of the form $\alpha D + \beta D^2$ and (2) the risk from low dose rate radiation is then simply αD. The assumption here that one can extrapolate risk linearly from moderate acute doses (≈100 mGy) to very low dose fractions remains contentious.

EPA proposes adopting the BEIR VII quantitative uncertainty bounds for most purposes. It is anticipated, however, that the revised Blue Book would contain an examination of where these uncertainty bounds might fail to adequately capture the overall uncertainty. In addition, we would include a brief discussion of the low dose extrapolation problem, which would acknowledge continuing disagreement on this issue.

Ultimately, the estimates of uncertainties in risk per unit dose can be combined with estimates of uncertainties in tissue doses for internally deposited radionuclides in order to obtain uncertainty estimates for inhaled or ingested radionuclides. For alpha emitting radionuclides this will require additional assessment of risk and uncertainty beyond what is contained in BEIR VII. Estimation of risk from internally deposited alpha emitters is addressed in the WP.

4. Discussion and Conclusions

We anticipate that the revised methodology for calculating radiogenic risk will be applied to update the cancer risk coefficients for over 800 radionuclides that are published in EPA's Federal Guidance Report 13. Most of the revisions would be based on recommendations from BEIR VII. In particular, we intend to use the same ERR and EAR risk models for most cancer sites, and expect to adopt a value of 1.5 for the DDREF. However, we intend to use different population statistics than in BEIR VII, and are considering using an alternative method for estimating breast cancer mortality. In addition, the WP includes proposals for estimating risks not covered in BEIR VII such as: risks associated with alpha particles, lower energy beta/photon emitters, and estimates for skin cancer and prenatal irradiation.

The BEIR VII risk estimates, as compared to current EPA estimates, tend to be somewhat higher for incidence and very similar for mortality. For example, the BEIR VII LAR for cancer incidence (about 10.5% per Gy) is about 25% larger than the current EPA estimate (8.5%). In contrast, for mortality, the BEIR VII LAR (5.7% per Gy) is virtually the same as the current EPA LAR (5.75% per Gy).

As shown in Table 2, the BEIR VII lung cancer LAR estimates for males (1.4% per Gy) and females (3.0% per Gy) are 2 to 3 times as large as current EPA estimates. The very high lung cancer risk estimates in BEIR VII appear to be inconsistent with results from other studies. This is illustrated in the WP, which includes a comparison of the BEIR VII estimates to EPA's estimates of lung cancer risk for indoor radon; the latter were derived from the pooled underground miner studies in BEIR VI. The relative biological effectiveness (RBE) of the radon progeny inferred from the comparison is much lower than what would be expected from animal studies. The WP provides a detailed

discussion of reasons for why the BEIR VII risk estimates are so high. Nevertheless, it appears that the BEIR VII approach for estimating radiogenic lung cancer risks is reasonable, and at this point we anticipate adopting their approach.

Further details on proposed methods for revising EPA radiogenic risk estimates are provided in the WP.

Table 2 BEIR VII and current EPA lung and solid cancer incidence LAR*

Model	**Lung**		**Solid cancers**	
	Male	Female	Male	Female
BEIR VII	140	300	800	1300
(% of solid)	(18)	(23)		
EPA	81	128	650	103
(% of solid)	(12)	(12)		
* Number of cases per 100,000 persons exposed to 0.1 Gy.				

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1-5 Molecular Dynamics Simulations of DNA Strand Breaks

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Abstract

Ionizing radiation such as gamma- and X-rays causes genetic damage and cancer by means of breaking DNA molecule, including both single and double strand breaks. Due to their lethal consequences and relatively high probability of introduction of repair errors and mutations, double strand breaks are among the most important and dangerous DNA lesions. However, the mechanisms of their recognition and repair processes are only poorly known. This presentation reports selected results of computer analysis of a DNA with single strand break, employing both molecular dynamics and quantum simulations. Furthermore, utilizing these results as a template study, the preliminary results of more complex analysis of double strand break and the first stage of its enzymatic repair mechanism – annealing process – are reported.

Keywords: Molecular dynamics, Simulation, Strand break, DNA

1. Introduction

Several metabolic pathways in living organisms along with other oxidative processes in cell generate reactive oxidative radicals which can attack DNA causing both DNA base damage and strand breakage. In many cases the produced specific DNA damage may either block the replication and transcription or generate mutations by miscoding during replication. Oxidants with free-radical character are among the well known instigators of DNA damage.

The most significant consequence of the oxidative stress are thought to be DNA modifications, particularly strand breaks, which can result in mutations during the enzymatic repair and other types of genomic instability. Strand breaks, and double strand breaks (dsb) of DNA in particular, are among the most important lethal effects in cells; moreover, there is a significant probability that the attempted repair of dsb's leads to insertion or deletion errors, and a certain probability of mutation. Strand breaks are caused by the incision of bond at C3' or C5' atoms of DNA pentose and require damage to the sugar-phosphate backbone.

In this work we report results from theoretical modeling of events taking place after a single strand break (ssb) and partially also double strand break. The aim of our work is to provide a molecular view into the structural and electrostatic properties of DNA molecule that may establish the

environment facilitating an enzymatic repair.

2. Materials and Methods

2.1 Molecular Dynamics Protocol

Classical molecular dynamics (MD) was used to study the time evolution of the lesioned DNA, using modules of AMBER7 software package[1]. The NUCGEN module of AMBER7 was used to prepare the native sequence of the 8 base pairs B-DNA duplex – $d(TATGTCTC)_2$ part of oligonucleotides (Fig. 1). The force field used in simulation was parm99.dat[2]. Prior to the MD simulations the systems were neutralized by Na+ counterions, then solvated in a water box, heated to body temperature (310 K), and relaxed with respect to density and pressure. The further details of the used protocol have been reported elsewhere[3].

In the case of ssb lesioned DNA, the phosphodiester bond between hydroxyl group of C3' of cytosine 4 and the OH group of the adjacent phosphate group of adenine at the position 5 was broken. The valences of broken ends were left unoccupied during the first nanosecond (ns) of MD simulation. Throughout the simulation the periodic boundary conditions were employed. Since in a real cellular environment, there is a high probability of hydration of ssb ends, the situation was further modified as follows. After 1 ns of MD simulation the open valences of ssb ends were filled: C3' end by hybridization of O3' atom and phosphate group by attaching an OH group.

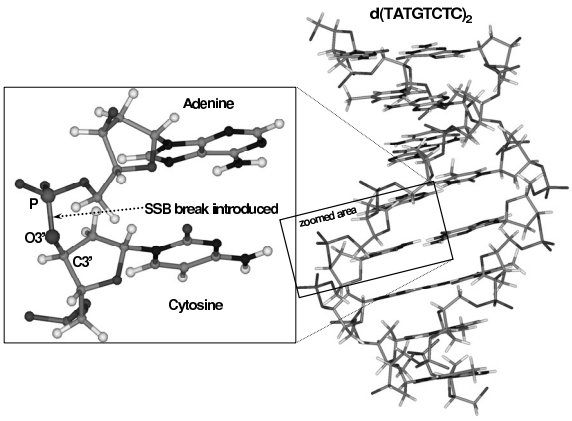

Figure 1 The 8 base pairs B-DNA duplex – $d(TATGTCTC)_2$; zoomed area shows the position where the ssb was introduced.

Double strand break systems were created using experimentally manufactured structures[4] – isolated Ku heterodimer structure (PDB databank code 1JEQ.pdb) and Ku heterodimer in complex with DNA (1JEY.pdb). Instead of the standard molecular dynamics simulation, a special case was applied – the so called simulated annealing. It differs from ordinary MD by using higher temperature environment to fasten the process of annealing, simultaneously also constraining chemical bonds to prevent their splitting. The aim of such simulation is to relatively quickly generate annealing of two independent molecules. Crystal of the isolated Ku heterodimer and isolated DNA sequence identical to the sequence in the 1JEY pdb file were used to create the initial structure (Fig. 2).

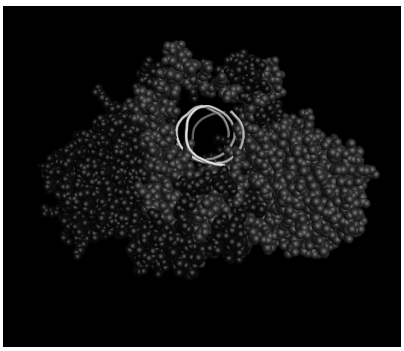

Figure 2 Structure of the Ku heterodimer in complex with DNA molecule. Small spheres denote the atoms of the enzyme, whilst the light curve is the DNA molecule inside of the Ku heterodimer.

In all modified structures atomic point charges were computed employing quantum calculation by Gaussian03 software package[5].

3. Results and Analysis

In order to investigate the structure and dynamics of DNA molecules in the simulated system, the root mean square deviations (r.m.s.d.) of the non-damaged DNA molecule and the DNA molecules with the ssb were calculated. The r.m.s.d. value represents an average deviation calculated for all heavy atoms of the molecule and is usually used as an index of the stability of the simulated system after preparatory steps MD simulation. Weighted fit values of the r.m.s.d. data are plotted in Fig. 3. We

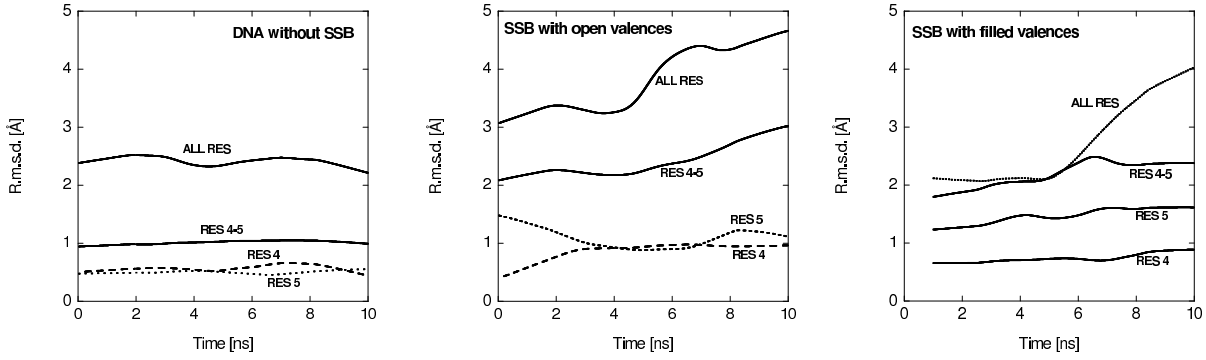

Figure 3 Time dependence of the root mean square values for the non-damaged DNA (left panel), system with single strand break with open valencies (middle panel) as well as filled valencies (right panel). Weighted fit values of the data are plotted. R.m.s.d. was calculated for all residues except the outer terminal base pairs (ALL RES), for the damaged residues (4 and 5) as a doublet (i.e. adenine and cytosine at the ends of single strand break), as well as for each of the residues separately (RES 4 - cytosine, and RES 5 - adenine). All calculations were fitted to the frame of the respective residues, thus calculated values represent realistic evaluation of conformational change of the particular systems.

can see that non-damaged DNA (left panel) is naturally in a stable state. Single strand break with filled valencies shows significant instability, especially after 5 nanoseconds of the simulation. The residues itself remain stable (as can be concluded from the lowest two curves) whilst the distance between the broken residues increases with time. The ssb with filled valencies (right panel) is more stable throughout the simulation; the quickly increasing value for all residues is caused only by a natural distortion of residues at the end of the DNA, and is not connected with the existence of the break.

As a more direct way how to evaluate the structural behavior of the system, Fig. 4 shows time dependence of the geometry for all of the three systems. It supports the results from the r.m.s.d.,

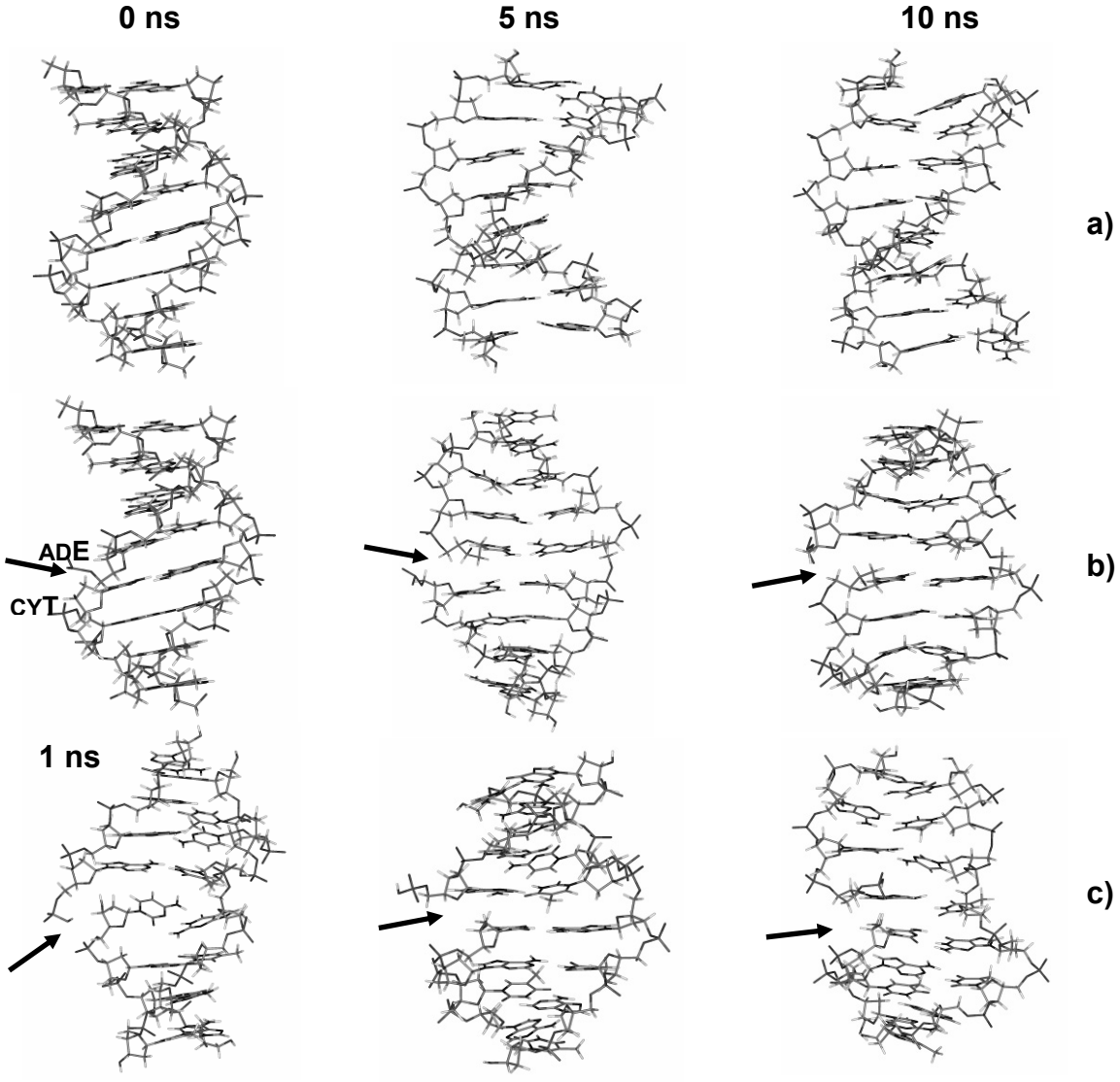

Figure 4 Snapshots of the structural behavior of a) DNA without ssb b) ssb with open valencies and c) ssb with filled valencies. Initial position as well as the structures after 5 and 10 nanoseconds is shown. Black arrows denote the break position.

showing that in the case of open valencies the break is open to the surrounding environment throughout the simulation which makes him relatively easily accessible by a repair enzyme. Filled valences apparently cause less distortion of the structure and although there is similar opening to the outside of the molecule during the first few nanoseconds, after approximately 8 ns the situation differs, and structure partially stabilizes, thus limiting also the docking possibilities of an incoming repair enzyme.

Hydrogen bonds as well as electrostatic interaction were also analyzed (results not shown), supporting the conclusions from the structural analysis. They showed that the relative stability of the ssb with filled valencies is caused by rebounding of the originally broken nucleotides/atoms by a hydrogen bond. Due to relative weakness of such a bond in comparison with the original strong covalent one, this bond is only temporary and more detailed analysis revealed that this bond is broken again in the later stages of the simulation. Thus, although slightly stabilizing the break, the hydrogen bond itself has not the crucial effect on the ssb repair process.

Figure 5 demonstrates structural results from the double strand break simulated annealing. We tried to perform the calculation including explicit water molecules in the system, however, it turned out that AMBER software is not capable to manage such large systems, most probably due to numerical limits of its algorithm. Therefore, we used an alternative approach, the so called general Born approximation, which replaces water molecules by numerically less demanding approximation. Although not as exact as explicit calculation of water, it is the best known and most extensively tested alternative. We obtain reliable results from these simulations, sample of which can be found in Fig. 5. Unfortunately, no annealing has been observed. This is due to the selection of the initial position. There are, of course, large number of possible initial states, and subsequently also many possible outputs. Only some of them are leading to the annealing. Therefore, we obtained realistic results, but those are only a small number of possible states. In order to achieve the annealing, we must more appropriately select and create the initial structure. It is inevitable to perform it using the quantum geometry optimization. However, this is relatively complex and numerically demanding procedure, therefore it has not been realized yet.

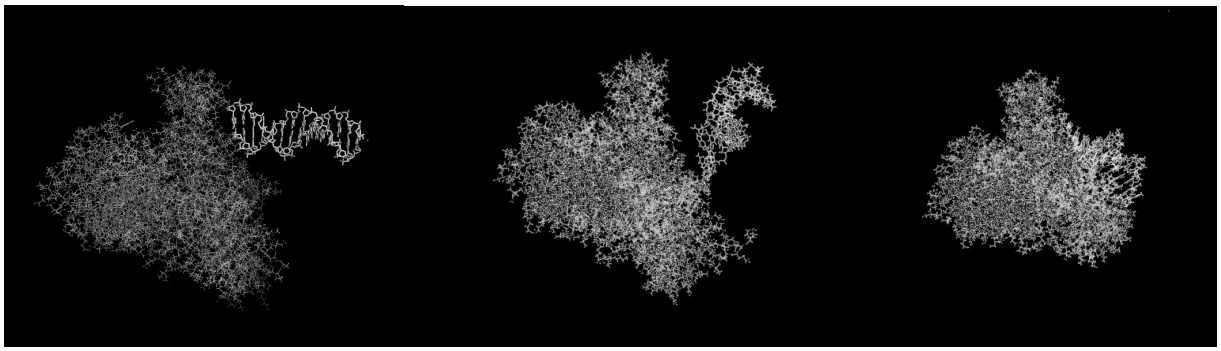

Figure 5 Simulated annealing of the Ku heterodimer and DNA: initial position (left), and two of the results. In the situation showed in the middle panel the DNA did not bind to the heterodimer, and was freely floating near the enzyme. The right panel plots different system, where DNA did bind to the enzyme, however, not in the way which would lead to the annealing and docking of the DNA into the Ku heterodimer.

4. Conclusion

We performed molecular dynamics simulation and subsequent analyses of structural and electrostatic properties of a single strand break and first stages of a double strand break damage using AMBER software. The results indicate that the ssb induces significant distortion of the structure in the vicinity of the break. Ending with open valences is unstable and well exposed to the outside of the molecule, whilst system with filled valencies tends to create temporary hydrogen bond between the broken ends, however, not significantly affecting the whole system..

Double strand simulation with explicit water molecules turned out to be beyond capabilities of AMBER software, no attempt to overcome this limit was successful. Generalized Born approximation (no explicit water) gives realistic results, however, no annealing has been observed, due to inappropriate initial structure - quantum simulations must be employed in order to find the appropriate initial structures.

Acknowledgement

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1-6 ORNL's DCAL Software Package

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Abstract

Oak Ridge National Laboratory has released its Dose and Risk Calculation software, DCAL. DCAL, developed with the support of the U.S. Environmental Protection Agency, consists of a series of computational modules, driven in either an interactive or a batch mode for computation of dose and risk coefficients from intakes of radionuclides or exposure to radionuclides in environmental media. The software package includes extensive libraries of biokinetic and dosimetric data that represent the current state of the art. The software has unique capability for addressing intakes of radionuclides by non-adults. DCAL runs as 32-bit extended DOS and console applications under Windows 98/NT/2000/XP. It is intended for users familiar with the basic elements of computational radiation dosimetry. Components of DCAL have been used to prepare U.S. Environmental Protection Agency's Federal Guidance Reports 12[1)] and 13[2)] and several publications of the International Commission on Radiological Protection[3-7)].

Keywords: Computational dosimetry, Internal dose, External dose, Radiation dose, Radiation risk, Compartment models

1. Introduction

The Dosimetry Research Team at Oak Ridge National Laboratory (ORNL) has developed a comprehensive software system for the calculation of tissue dose and subsequent health risk from intakes of radionuclides or exposure to radionuclides present in environmental media. This system serves U.S. Environmental Protection Agency's current needs in radiation dosimetry and risk analysis. The Dose and Risk Calculation software, called DCAL, has been used in the development of Federal Guidance Reports 12[1)] and 13[2)] and several publications of the International Commission on Radiological Protection (ICRP), specifically in the computation of age-specific dose coefficients for members of the public[3-7)].

DCAL is designed for use on a personal computer (PC) by individuals experienced in scientific computing and computational radiation dosimetry. The system consists of a series of computational modules, 32-bit extended DOS and console applications, driven by a user interface. DCAL may be used either in an interactive mode designed for evaluation of a specified exposure case or in a batch mode that allows non-interactive, multiple-case calculations. The software can be downloaded from the ORNL website[8)].

DCAL performs biokinetic and dosimetric calculations for the acute intake of a radionuclide by

inhalation, ingestion, or injection into blood at a user-specified age at intake. The user may compute either equivalent or absorbed (low and high LET) dose rates as a function of time following intake of the radionuclide. The equivalent dose option enables the generation of a table of age-specific dose coefficients, i.e., committed equivalent doses to organs and committed effective doses per unit intake, such as those published in the ICRP documents on doses to the public from intake of radionuclides. If the endpoint of the calculation is radiogenic risk, the absorbed dose option must be selected because the radiation risk factors used by DCAL are expressed in terms of absorbed dose.

DCAL includes a module for the evaluation of dose rate resulting from external exposure to radionuclides present outside the body in environmental media, i.e., distributed in an airborne cloud, in water, on the ground surface, or to various depths in the soil. That module uses the photon and electron dosimetric data tabulated in Federal Guidance Report 12[1)] for monoenergetic sources to generate radionuclide-specific dose rate coefficients. As is the case for intake of radionuclides, if the endpoint of the calculation is radiogenic risk, DCAL couples the generated absorbed dose rates with radiation risk factors and mortality data to predict organ-specific risk of radiogenic cancer death from chronic exposure to the radionuclide in the environmental medium.

2. Dosimetry of Internal Emitters

For the case of radionuclide intake at a pre-adult age, anatomic and dosimetric parameter values are interpolated in a continuous manner throughout the period of growth. If age dependence is indicated in the biokinetic model, the biokinetic parameter values also are interpolated in a continuous manner throughout life. Age-specific biokinetic models for several radioelements are provided in the ICRP's series on doses to members of the public from intake of radionuclides[3-7)]. The ages at intake considered in that series are 100 d, 1 y, 5 y, 10 y, 15 y, and adult. The beginning age of adulthood is defined in the element-specific systemic biokinetic model. That age is 20 y for most elements but is 25 y for some bone-seeking elements because of substantial changes in the modeled biokinetics of the element between ages 20 and 25 y.

The dosimetric calculations proceed in three main steps:

Step 1: calculation of time-dependent activity of the parent radionuclide and its radioactive progeny present in anatomical regions (source regions) of the body by the ACTACAL module;

Step 2: calculation of *SEE* values for the combinations of source region *S* and target region *T* in the case, where *SEE(T,S)* is the dose rate in *T* per unit activity present in *S* by the SEECAL module;

Step 3: calculation by the EPACAL module of absorbed or equivalent dose rates and committed absorbed or equivalent dose, based on output generated in Steps 1 and 2.

The committed dose coefficients are an integrated organ absorbed dose or dose equivalent, or an effective dose per activity intake. The integration period is 50 y for intake by the adult and from age at intake to age 70 y for intake at a pre-adult age. A utility, HTAB, is provided to tabulate the dose coefficients in the manner seen in the ICRP publications.

Cancer risk coefficients can also be calculated after completion of Step 3 provided the absorbed

dose option was selected. For a given radionuclide and exposure mode, both a mortality risk coefficient and a morbidity risk coefficient are calculated. The mortality risk coefficient is an estimate of the risk, per activity inhaled or ingested for internal exposures or per time-integrated activity concentration in air or soil for external exposures, of dying from a radiogenic cancer. The morbidity risk coefficient is a comparable estimate of the average total risk of experiencing a radiogenic cancer, whether or not the cancer is fatal. Either risk coefficient applies to an average member of the public, in the sense that estimates of risk are averaged over the age and gender distributions of a hypothetical closed "stationary" population whose survival functions and cancer mortality rates are based on recent data for the U.S.

2.1 Data Libraries

DCAL relies on data libraries defining the biokinetic models, nuclear decay data, dosimetric data, anatomic data, radiation risk models, survival data, cancer mortality and morbidity, and various other miscellaneous data. These libraries enable the user to compute dose and risk estimates with minimal input. The software has been designed for easy expansion of its library of systemic biokinetic models and gastrointestinal uptake values (f_1 values) with user-supplied models. The biokinetic, dosimetric, and risk libraries are considered as permanent files, although virtually all portions of these libraries are readily accessible and could be modified by users who are familiar with these data and the organization and structure of the DCAL system.

Two sets of systemic biokinetic models are contained in the DCAL libraries: the models for occupational exposure recommended in ICRP Publication 68[9]; and the age-specific models applied in the production of EPA's Federal Guidance Report 13[2]. The latter set of models is generally the same as that used in the ICRP's series of documents on doses to members of the public, as summarized in ICRP Publication 72[7], although a few of the ICRP's models were modified as described in Federal Guidance Report 13. The biokinetic libraries include: the latest ICRP model of the respiratory tract as described in ICRP Publication 66[10]; the ICRP's gastrointestinal tract model used in calculations for Federal Guidance Report 13; and the urinary bladder voiding model described in ICRP Publication 67[4]. The nuclear decay library contains nuclear decay data currently used by the ICRP[11] and the Medical Internal Radiation Dose (MIRD) Committee of the Society of Nuclear Medicine[12]. The photon specific absorbed fraction library is based on the data of Cristy and Eckerman[13,14] as currently used by the ICRP. Organ masses for adults are taken from ICRP Publication 23[15] and, for children, the values are taken from the phantoms of Cristy and Eckerman[13,14]. The radiation risk models are based on the EPA's current methodology[16], but some parameter values of those models have been modified as described in Federal Guidance Report 13). Gender-specific survival data are from the U. S. Decennial Life Tables for 1989-1991[17].

2.2 ODE Solver of Biokientic Models

DCAL's rapid solution of complex compartment models is accomplished by applying an efficient approximation technique developed by the ORNL dosimetry team[18]. The technique has no restrictions of practical importance on the number of compartments, the network of flows between

compartments, the number of radioactive daughter products, or the paths of movement of chain members. The technique is unconventional in that the model is not viewed as a set of coupled differential equations but rather as a series of isolated compartments. On each time step during the time period of interest, the content of a compartment is approximated by the closed-form solution of the differential equation for an isolated compartment. Any specified level of accuracy can be achieved by selecting time steps sufficiently small. For the biokinetic models included in the DCAL libraries, the set of default time steps used in DCAL typically yields relative errors of at most a few tenths of one percent with regard to instantaneous activities in compartments and virtually exact values for the integrated activities. For a specified level of accuracy, computing time and storage requirements increase roughly in proportion to the number of compartments but with current PCs at most a few seconds is required even for the most complex models.

If one assumes first-order kinetics in an isolated compartment that has a constant activity inflow rate P, a constant removal rate coefficient R, and initial activity Y_0 at time 0, the activity Y at time T is given by the differential equation $\frac{dY}{dT} = -RY + P,$ with solution

$$Y = \left(Y_0 - \frac{P}{R} \right) e^{-RT} + \frac{P}{R} \quad . \tag{1}$$

The integrated activity $\widetilde{Y}$ from time 0 to time T is then

$$\widetilde{Y} = \frac{1 - e^{-RT}}{R} \left(Y_0 - \frac{P}{R} \right) + \frac{PT}{R} \ . \tag{2}$$

Although these equations do not apply directly to compartmental models they may be applied iteratively to approximate the solutions of such models to any desired degree of accuracy.

The calculation proceeds in a series of time steps measured in days, with the k^{th} step defined by a starting time T_k and an ending time $T_{k+1} > T_k$.. The activity Y_i in compartment i at time T_{k+1} is calculated from Eq. 1 by initializing time 0 to T_k, defining the initial activity in the compartment to be the activity calculated at the ending time in the preceding time step, and replacing T with $T_{k+1} - T_k$. The integrated activity $\widetilde{Y}_i$ in the compartment during the same time interval is calculated similarly using Eq. 2. During each time step the flow rate into compartment i, P_i, is taken to be the value that would yield the total activity that flows out of all feeding compartments with an index less than i during the same time step, plus that flowing out from all feeding compartments with higher index during the previous time step. That is, the inflow rate P_i into compartment i during the k^{th} time step is

$$P_i = \frac{1}{(T_{k+1} - T_k)} \sum_{\substack{j=1 \\ j \neq i}}^{NC} R_{j,i} \, \widetilde{Y}_j \tag{3}$$

where $R_{j,i}$ is the transfer coefficient (fractional transfer per day) from compartment j to compartment i at the midpoint of the time step and $\widetilde{Y}_j$ is the last-computed integrated activity in compartment j and NC is the number of compartments. The procedure is repeated until all times of interest have been reached.

There are two main criteria for selection of time steps. First, the steps should be sufficiently short that errors remain within a prescribed limit. Second, times for which estimates are desired should be included in the time grid defining the steps, since computations are made only at the endpoints of time steps. The default time-stepping scheme used in ACTACAL is 1000 steps of length 0.001 d, 900 steps of length 0.01 d, 900 steps of length 0.1 d, 3900 steps of length 1 d, and 2155 steps of length 10 d, giving a total of 8855 steps to reach 70 y (25,550 d) from time of intake.

Numerous checks of the accuracy of the solver have been made through comparison of solutions of this and other solvers for a variety of biokinetic models. Comparisons have been made with published tables of nuclear transformations over 50 y, based on the models of ICRP Publication 30[19]. These tables were produced by the code TIMED[20], which incorporates a widely used adaptive predictor-corrector solver. Checks on the accuracy of the solutions of recycling models have been made by comparisons with two virtually exact solvers with regard to the relatively "non-stiff" models of this series: a computer code called DIFSOL[21], which obtains analytical solutions of linear ordinary differential equations with constant coefficients; and a code developed by Birchall and James[22]. In all instances the solvers were found to be in excellent agreement.

3. External Dosimetry

The DCAL system was developed largely to address the doses and risks resulting from the intake of radionuclides. However, the EXTDOSE module, initially developed during the preparation of Federal Guidance Report 12[1], has been integrated into DCAL.

Photons and electrons are the most important radiations emitted by radionuclides distributed in the environment that can penetrate the body from outside to deposit ionizing energy within its radiosensitive tissues. Some radionuclides produce bremsstrahlung and this contribution to dose was included in Federal Guidance Report 12. The version of EXTDOSE incorporated into DCAL differs from the earlier version in that the bremsstrahlung contribution is computed directly rather than read from a pre-calculated data file. This change can result in minor numerical differences in external coefficients derived using DCAL and those tabulated in Federal Guidance Report 12 for pure beta emitters; e.g., Sr-90. In addition, the radiations associated with spontaneous fission are addressed in the DCAL version and were not considered in Federal Guidance Report 12.

The calculation of organ doses from irradiation of the body by photon emitters distributed in the environment requires the solution of a complex radiation transport problem. It is impractical to solve this problem for the precise spectrum of photons emitted by each radionuclide. Organ dose coefficients, computed for monoenergetic photon sources at twelve energies between 0.01 and 5.0 MeV, were tabulated in Federal Guidance Report 12 and are included as a library in DCAL. EXTDOSE considers exposure to radionuclides distributed:

- in the air surrounding an individual (submersion)
- in the water which the individual is immersed (immersion)
- on the surface of the ground
- in the top 1-, 5-, or 15 cm layer of soil
- uniformly within the soil depth

EXTDOSE uses the organ dose rates for a monoenergetic sources to derive radionuclide-specific dose coefficients.

4. Conclusion

The DCAL software has been used to prepare a number of federal guidance reports and publications of the ICRP. Continued development of DCAL is anticipated to update the dosimetric methods with developments within ICRP, include gender-specific dosimetric and biokinetic models, and update the health risk model. Extension of DCAL to consider *in utero* exposures remains to be addressed. In addition, a module for the interpretation of bioassay measurement may also be included in future editions. Further integration of DCAL into the Windows environment with increased use of graphic user interfaces is planned. It is expected that DCAL will remain the main computational tool for dose and risk estimates of the Oak Ridge Dosimetry Research Team.

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1-7 Simulation Analysis of Radiation Fields inside Phantoms for Neutron Irradiation

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Abstract

Radiation fields inside phantoms have been calculated for neutron irradiation. Particle and heavy-ion transport code system PHITS was employed for the calculation. Energy and size dependences of neutron dose were analyzed using tissue equivalent spheres of different size. A voxel phantom of mouse was developed based on CT images of an 8-week-old male C3H/HeNs mouse. Deposition energy inside the mouse was calculated for 2- and 10-MeV neutron irradiation.

Keywords: Neutron irradiation, Voxel phantom of mouse, Radiation field, Simulation analysis, Relative biological effectiveness

1. Introduction

Neutrons are more effective radiation than X and gamma rays for induction of tumours, and for most other late somatic effects of radiations. The effectiveness of the radiation is indicated with relative biological effectiveness (RBE) that is equal to the ratio of absorbed dose between two types of radiation producing the same specified effect. Recently, the Joint U.S.-Japan Working Group reassessed the absorbed dose to the atomic-bomb survivors of Hiroshima and Nagasaki, and updated the dosimetry system as DS02[1)] from the previous one, DS86[2)]. One of the most important results given by the new dosimetry system is that the dose originated from neutron is considerably lower than that from gamma ray, and the data are insufficient to analyze the dose-response relationship for neutron. This result leads that there is no useful epidemiological data left to derive the RBE values for human carcinogenesis. It is, thereby, necessary to rely on studies in animals with regard to the evaluation of RBE for neutron.

Many radiobiological experiments have been performed for various endpoints in all over the world. The National Institute of Radiological Sciences (NIRS, Japan) has started the biological effect research program on fast neutrons (2 and 10 MeV) to derive the RBE values for carcinogenesis (murine myeloid leukemia, rat mammary tumour, etc.) and for effects on the development of nervous system in rodents. These data are very useful to investigate the effect of neutron, and dedicated to constructing the radiation protection system. However, there still remains the common problem in all experimental data obtained from small animals. The internal radiation field and the energy-deposition process are quite different between small animals and human, even if the subject is exposed with an

identical radiation field. When neutrons are entering into a body, various secondary particles are produced via nuclear reactions, and consequently deposit their energies inside the body. At the final stage of the energy-deposition process in neutron irradiation, the energies are transferred to the receptor with charged heavy particles caused by neutrons (neutron component) or electrons caused by photons (photon component), which are released inside the body by the external neutrons. Therefore, the total absorbed dose of neutron is described by the sum of the doses from the neutron and photon components, and the relative contribution of these components is strongly dependent on the size and sharp of receptor and the position of target region.

In order to assess the effect of neutron for human from RBE data of small animals, it is required that not only radiobiological experiments but also simulation analysis of internal radiation field. To meet this demand, the Japan Atomic Energy Agency (JAEA) is collaborating with NIRS, and developing computational models that represent size, sharp and anatomical structure of mouse and human. In this paper, we report the calculation results of the two components of neutron dose using spherical phantoms of different size in various neutron energies. A voxel phantom of mouse is developed on basis of computer tomographic (CT) images of a real mouse. Deposition energy is also calculated using the voxel phantom for 2- and 10-MeV neutron irradiation.

2. Radiation transport code

Particle and Heavy-Ion Transport code System PHITS[3] was used for the simulation analysis. This code has been developed at JAEA to design the shielding of accelerator facilities. The applicable energy is from thermal region to 200 GeV for neutron. PHITS can treat neutron, photon, electron, proton, meson and heavy-ion transport by use of an optimal combination of theoretical models. Calculation with evaluated nuclear data is also supported by incorporating the components of MCNP[4] (Monte Carlo N-Particle transport code) that is a general purpose transport code developed at Los Alamos National Laboratory. This makes possible to execute more reliable transport compared with that using only theoretical models in the energy region below a few mega-electron volts. In this work, JENDL-3.2[5] (Japan Evaluated Nuclear Data Library, version 3.2) and ENDF/B-VI[6] (Evaluated Nuclear Data File B, version VI) were adopted as the evaluated nuclear data for neutron and photon, respectively.

In the calculation of energy-deposition process, PHITS has a sophisticate model designated as event generator mode. In this mode, actual deposition energies by charged particles can be computed event by event without Kerma approximation. The detail of the event generator mode is described elsewhere[7].

3. Two components of neutron dose

In order to study the energy and size dependences of the two components in neutron dose, absorbed doses have been calculated with PHITS by varying the energy of incident neutron and the size of receptor volume. Spherical phantoms of ICRU tissue[8] were adopted to simplify the calculation. The diameters of the phantom are 30, 5 and 1 cm; they roughly represent human body, mouse and small specimen of tissue. Target regions for calculations of the absorbed dose were set at center of

sphere, and their diameters were 3, 0.5 and 0.1 cm for the 30-, 5- and 1-cm-diameter phantoms, respectively. The phantom data are summarized in Table 1. A neutron source was defined as monoenergetic parallel beam, and the energy was varied from 10^{-8} to 10 MeV.

Figures 1 – 3 shows the calculation results of the absorbed dose in the 30-, 5- and 1-cm-diameter phantoms. In Fig. 1, the absorbed dose from photon component is nearly equal to the total one. At the neutron irradiation for a large receptor such as human body, the neutrons are moderated in energy by multiple scattering and consequently thermalized during the transport. The thermalized neutron is captured by a hydrogen nucleus, and a secondary photon with energy of 2.2 MeV is produced via the H(n, γ)D reaction. On the other hand, the absorbed dose from neutron component is dominant in Figs. 2 and 3. This is because that the moderation of neutron energy is minor in the small receptors, and the relative yield of secondary photons by thermal neutron absorption is suppressed.

The dose from photon component increases in lower energy region below 1 MeV. This reason is that neutrons tend to be moderated and captured by hydrogen nucleus as the incident energy is closing to thermal region. The photon component observed above 1 MeV comes from the photon-emission-decay processes of residual nuclei excited by neutron collision.

4. Voxel phantom of mouse

An 8-week-old male C3H/HeNs mouse has been imaged by using a dedicated small-animal CT scanner (In Vivo 3D Micro X-ray CT System, R_mCT; Rigaku Inc. Japan). The system consists of an X-ray tube and a flat-panel detector. The subject to be imaged is fixed on a central stage, and the tube and the detector are rotated around it. The 3-dimensional images with 552 slices are acquired in 17 sec with 20 μm pixel size.

Voxel data were generated from the CT images by using a computer software JCDS[9] (JAERI Computational Dosimetry System) that has been applied to process human-head images for 3-dimensional dosimetry in the boron neutron capture therapy at JAEA. The CT images are read by JCDS in the format of DICOM. JCDS can define the region of interest (ROI) on the images automatically according to CT values or manually by drawing lines, and assign the materials to the ROI. In the present model, the size of voxel was set at 1 mm × 1 mm × 1 mm by considering mean free path of neutron, and only soft tissue, bone and air were assigned as materials. Organs were not identified, while they will be added in the next revision. The voxel data were provided in the format of general geometry (GG), which is employed in the various transport codes such as MCNP and PHITS. Figures 4 and 5 depict the views of voxel phantom of mouse and its skeleton structure, respectively.

5. Deposition energy inside mouse

Spatial distribution of deposition energy in a body of mouse is shown in Fig. 6. The incident neutron energy was set at 2 MeV; this is the same energy used in the experiment at NIRS, and corresponding to the peak energy of neutrons emitted from the nuclear fission reaction. The upper and lower figures are the cross sections on vertical and horizontal planes passing through the center of the voxel phantom. The grid lines indicate the boundary surface of voxels. Neutrons were incident on the

Table 1 Characteristics of phantoms

Model	Diameter of phantom [cm]	Diameter of target region [cm]
Human body	30	3
Mouse	5	0.5
Small specimen of tissue	1	0.1

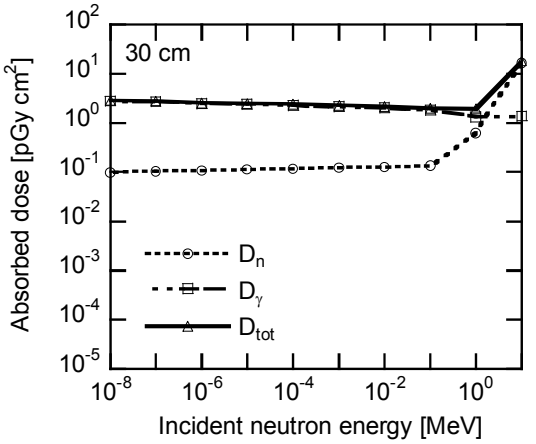

Figure 1 Absorbed dose inside the 30-cm-diameter phantom. D_{tot} denotes total absorbed dose. D_n and D_γ denote doses from neutron and photon components, respectively.

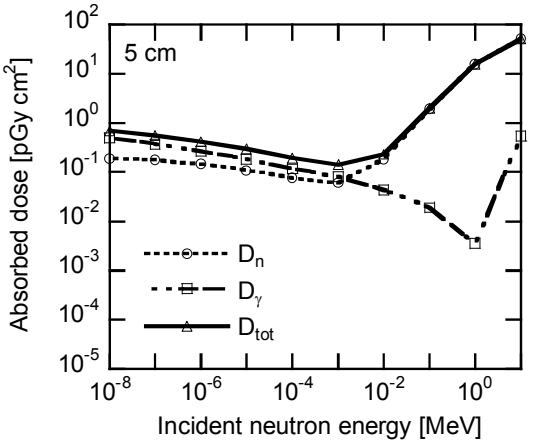

Figure 2 Absorbed dose inside the 5-cm-diameter phantom. D_{tot} denotes total absorbed dose. D_n and D_γ denote doses from neutron and photon components, respectively.

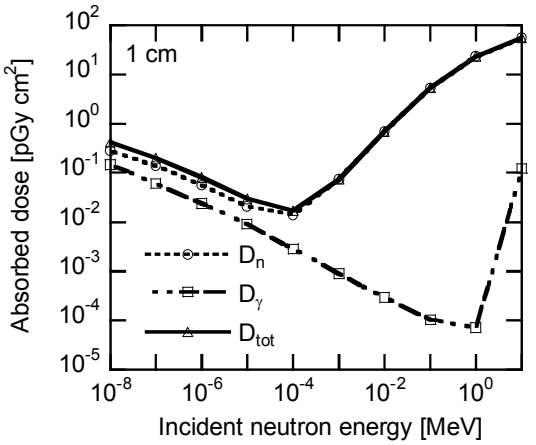

Figure 3 Absorbed dose inside the 1-cm-diameter phantom. D_{tot} denotes total absorbed dose. D_n and D_γ denote doses from neutron and photon components, respectively.

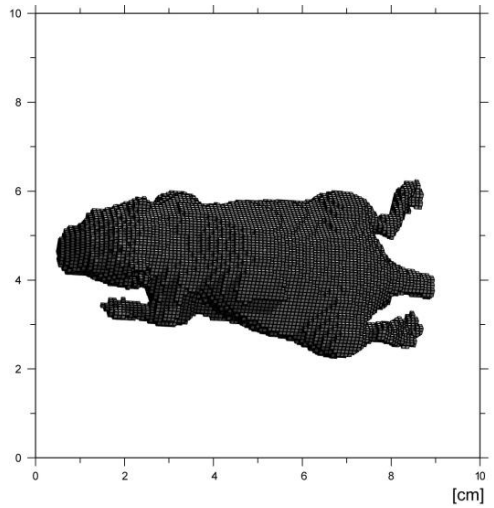

Figure 4 View of voxel phantom of mouse.

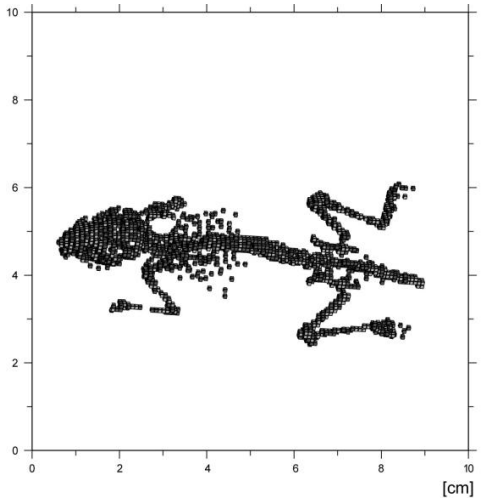

Figure 5 View of skeleton structure inside mouse phantom.

back of mouse, and covered the whole body. From the figure, it is obvious that the deposition energies are distributed uniformly in the body at 2-MeV neutron irradiation.

The contributions of proton, electron and alpha particle to the total deposition energy were also calculated for the same configuration of neutron irradiation described above. Figures 7 and 8 show the results for 2- and 10-MeV neutron irradiation, respectively. For the 2-MeV neutron irradiation, the energies are deposited in the body mainly with protons. The elastic scattering between neutron and hydrogen nucleus is a dominant reaction in this energy. On the other hand, the contribution of electron and alpha particle is observed at 10-MeV neutron irradiation. In this energy, the neutron has a sufficient energy to excite heavier nuclei such as carbon and nitrogen. Photons and alpha particles are produced from the de-excitation process of these nuclei, and subsequently the photons kick out the electrons.

6. Conclusion

Radiation fields inside phantoms were calculated using the radiation transport code PHITS for various external neutron fields. The energy and size dependences of the neutron and photon component in neutron dose were analyzed on tissue equivalent spheres. Neutron moderation and the dose from photon component increase with the size of the receptor volume and with decreasing the neutron energy. A voxel phantom of mouse was developed based on the CT images. The computer software JCDS was used to process the CT image, and the 3-dimensional voxel data were generated in the format readable in PHITS. The voxel size was set at 1 mm × 1 mm × 1 mm. Organs were not identified in the present model. From the simulation of 2-MeV neutron irradiation for a mouse, it was found that the deposition energies are distributed uniformly in the body, and the main carrier of energy is protons. At 10-MeV neutron irradiation, not only protons but also electrons and alpha particles contributed to the energy transfer.

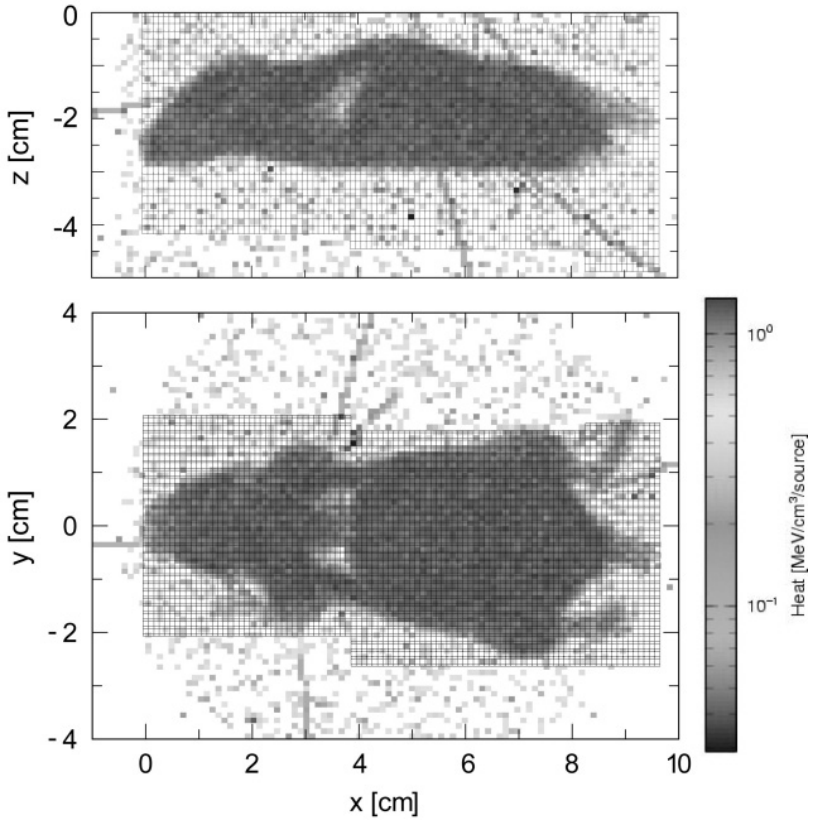

Figure 6 Spatial distribution of deposition energy in voxel phantom of mouse. 2-MeV neutrons were incident on the back of the body.

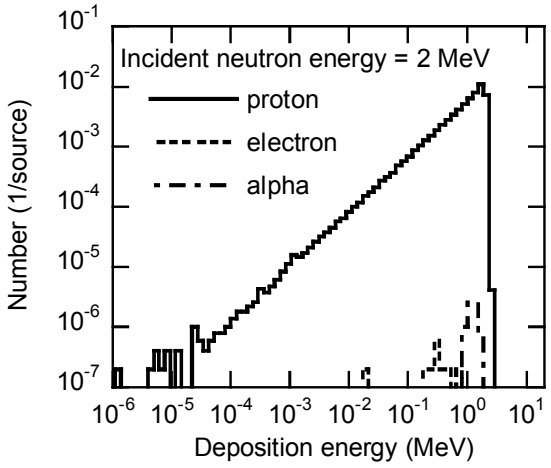

Figure 7 Deposition energies of proton, electron and alpha particle at 2-MeV neutron irradiation for mouse.

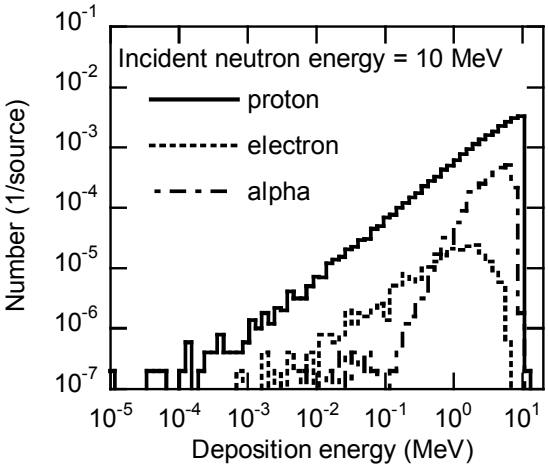

Figure 8 Deposition energies of proton, electron and alpha particle at 10-MeV neutron irradiation for mouse.

In the future work, we will upgrade the voxel phantom of mouse by identifying the organs (heart, liver, kidneys, stomach, intestines, etc.) and calculate the energy spectra of each charged particle at the specified organ. Similar simulation analysis is performed for human using a voxel phantom. Through the detailed analysis of the calculation results between mouse and human, we will construct an appropriate method to assess the effects of neutron for human from experimental data of small animals.

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Session 2　Radiation Dosimetry

2-1 ICRP New Recommendations: Committee 2's Efforts

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Abstract

The International Commission on Radiological Protection (ICRP) may release new primary radiation protection recommendations in 2007. Committee 2 has underway reviews of the dosimetric and biokinetic models and associated data used in calculating dose coefficients for intakes of radionuclides and exposures to external radiation fields. This paper outlines the work plans of Committee 2 during the current term, 2005–2009, in anticipation of the new primary recommendations.

The two task groups of Committee 2 responsible for the computations of dose coefficients, INDOS and DOCAL, are reviewing the models and data used in the computations. INDOS is reviewing the lung model[1)] and the biokinetic models that describe the behavior of the radionuclides in the body. DOCAL is reviewing its computational formulations with the objective of harmonizing the formulation with those of nuclear medicine[2)], and developing new computational phantoms representing the adult male and female reference individuals of ICRP Publication 89[3)]. In addition, DOCAL will issue a publication on nuclear decay data to replace ICRP Publication 38[4)]. While the current efforts are focused on updating the dose coefficients for occupational intakes of radionuclides plans are being formulated to address dose coefficients for external radiation fields which include consideration of high energy fields associated with accelerators and space travel and the updating of dose coefficients for members of the public.

Keywords: ICRP, Radiation protection, Radiation weighting, Tissue weighting, Effective dose

1. Introduction

The International Commission on Radiological Protection (ICRP) may release new primary radiation protection recommendations in 2007. Each of the Commission's committees has been directed to define and address the technical issues associated with implementation of new recommendations. Committee 2 has the role of translating the Commission's primary recommendations into quantities that can be used for planning of work practices. Thus, Committee 2 has underway reviews of the dosimetric and biokinetic models and associated data used in the calculation of dose coefficients for the intake of radionuclides and exposure to external radiation fields. Following a brief discussion of the proposed recommendations, the work plan of Committee 2 in

anticipation of the new primary recommendations during the current term, 2005–2009, is discussed.

2. Proposed Recommendations

The forthcoming recommendations were posted for consultation on the ICRP website and have been characterized by the phrase "Evolution not Revolution". The proposed recommendations provide additional guidance on some topics (e.g., dose constraints) however they largely reflect an updating of the knowledge base that underlies radiation protection; particularly that regarding health risk. The principles of justification, optimization and dose limitation remain as the basic tenets of radiation protection.

Each of the Commission's committees was directed to prepare documents which define and address the technical issues associated with their implementation of new recommendations – these documents are referred to as foundation documents. The four committees of the Commission are:

- Committee 1. Radiation Effects
- Committee 2. Doses from Radiation Exposures
- Committee 3. Protection in Medicine
- Committee 4. Application of Recommendations

The newly formed Committee 5, titled Protection of the Environment, is not expected to prepare a foundation document as it is now just planning its scope of work. Committee 2's role is one of translating the Commission's primary recommendations into quantities that can be used for planning of work practices and thus it is concerned with the dosimetric quantities of radiation protection.

Committee 2 has prepared a foundation document entitled "Dosimetric Quantities" which will be an annex in the forthcoming recommendations. In addition the Committee has prepared a chapter on dosimetric quantities to appear in the new publication. Drafts of the foundation document were posted on the ICRP website for consultation and comments were received on various issues. Some of the issues addressed in the foundation document and subject to comments are noted below.

2.1 Radiation weighting factor - w_R

Committees 1 and 2 have reviewed the information on the radiation weighting factors used in the computation of equivalent dose. Changes have been suggested for the factors applied to protons (extended to also include charged pions) and to neutrons as indicated in Table1. The neutron weighting factor is given as a function of neutron energy as in Eqn 1 where E_n is the neutron energy in MeV (see Fig. 1). As seen in Table 1 for all other radiations no change in the weighting factor is indicated.

$$w_R = \begin{cases} 2.5 + 18.2\, e^{-[\ln(E_n)]^2/6} & , \quad E_n < 1\ \text{MeV} \\ 5.0 + 17.0\, e^{-[\ln(2E_n)]^2/6} & , \quad 1\ \text{MeV} \le E_n \le 50\ \text{MeV} \\ 2.5 + 3.25 e^{-[\ln(0.04E_n)]^2/6} & , \quad E_n > 50\ \text{MeV} \end{cases} \tag{1}$$

As evident from Fig. 1 the above function differs from the step function of ICRP Publication 60[5] at both low and high energy.

Table 1 Recommended Radiation Weighting Factors

Radiation type	**Radiation weighting factor, w_R**
Photons	1
Electrons and muons	1
Protons and charged pions	2
Alphas, fission fragments, heavy ions	20
Neutrons	A continuous function of neutron energy (see Eqn 1)

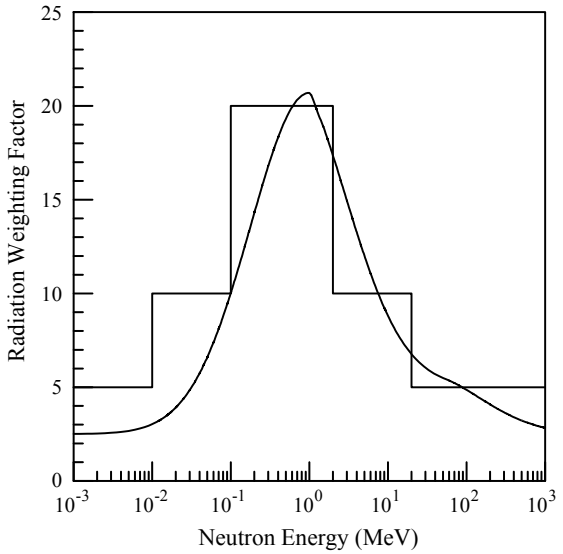

Figure 1 Proposed radiation weighting factor for neutrons (continuous curve) and the current values (step function) of ICRP Publication 60

2.2 Tissue weighting factors - w_T

A revised set of tissue weighting factors has been specified by Committee 1 based on its review of the epidemiological data on radiogenic cancer and hereditary effects. Committee 2 uses these factors in computing effective dose coefficients. The tissue weighting factors are age and gender averaged and the single set of values of Table 2 is to be applied in both occupational and environmental dose assessments. In specifying these values Committee 1 exercised the following judgements:

- The thyroid weighting factor was set to 0.04 reflecting the higher risk of thyroid cancer in childhood.
- Cancer risk due to irradiation of the salivary glands and the brain, whilst not precisely quantified, are judged to be greater than that of the other tissues comprising the remainder fraction, and thus these tissues are assigned a w_T value of 0.01.

The w_T for the remainder tissues (0.12) in Table 2 applies to the weighted mean dose of the 13 organs and tissues listed in the table's footnote for each gender. The so-called splitting rule in the treatment of the remainder[5] is no longer used and hence effective dose is additive.

Table 2 Recommended tissue weighting factors

Tissue	w_T	$\sum w_T$
Bone-marrow, Colon, Lung, Stomach, Breast, Remainder Tissues*	0.12	0.72
Gonads	0.08	0.08
Bladder, Esophagus, Liver, Thyroid	0.04	0.16
Bone surface, Brain, Salivary glands, Skin	0.01	0.04

*Remainder Tissues: Adrenals, Extrathoracic (ET) region, Gall bladder, Heart, Kidneys, Lymphatic nodes, Muscle, Oral mucosa, Pancreas, Prostate (♂), Small intestine, Spleen, Thymus, Uterus/cervix (♀).

2.3 Effective Dose Formulation

The effective dose is computed from the equivalent dose assessed for organ or tissue *T* of the male and female, superscripts M and F, respectively, as:

$$E = \sum_T w_T \left[\frac{H_T^M + H_T^F}{2} \right] \quad . \tag{2}$$

The summation in Eqn 2 extends over the tissues of Table 2, including the remainder. The equivalent dose to the remainder tissue of each gender is computed as the arithmetic mean of the equivalent doses to the tissues listed in the footnotes to Table 2 and is computed as;

$$H_R^M = \frac{1}{13} \sum_T^{13} H_T^M \quad \text{and} \quad H_R^F = \frac{1}{13} \sum_T^{13} H_T^F \quad . \tag{3}$$

This formulation applies to both internal and external components of the irradiation. The effective dose is based on the weighting factors that reflect judgements in selecting the particular values, the averaging of the absorbed energy over the volume of the organ and the averaging of the health detriment over age and gender. The value of the effective dose takes account of the given exposure situation (e.g., chemical form of the radionuclides) however, with the exception of age at intake of a member of the public, no characteristics of the specific individual are considered. Thus the effective dose is not suitable for assessing the risk to a specific individual and it should not be used in epidemiological studies.

3. Committee 2 Activities

In addition to the preparation of its foundation document and a chapter for the forthcoming recommendations, Committee 2 has a number of activities underway for implementation of the new recommendations. These activities are within the INDOS and DOCAL task groups of the committee. Committee 2's charges to these task groups are:

INDOS

Review the human and animal data on the behaviour of radionuclides in the body and develop models that can be used for dosimetry and bioassay interpretation.

DOCAL

Establish the computational methodology and supportive data for the derivation of dose coefficients for radiations incident upon the body (external radiation fields) or emitted within the body (internal emitters).

These two task groups have worked together on various past publications of the ICRP.

3.1 Computational Phantoms

The dose coefficients of ICRP Publications 30[6], 68[7], 72[8], and 74[9] have been based on various versions of the gender-invariant mathematical representation of the body[10-12]; the so-called MIRD phantom. Committee 2 has updated the anatomical and physiological characteristics of reference individuals in ICRP Publication 89[3] and now plans to adopt reference computational phantoms of the adult male and female which are based on medical tomographic images – so called voxel based phantoms[13]. The dimensions of the voxels and their number in these phantoms have been adjusted to approximate the organ masses assigned to the reference adult male and female in ICRP Publication 89. DOCAL is preparing the models for publication and will use these models in future calculations of dose coefficients for external radiation fields and for the intake of radionuclides.

3.2 Nuclear Decay Data

All nuclide-specific dose coefficients of the ICRP issued since 1979 have been based on the nuclear decay data of ICRP Publication 38[4]. An update of the energies and intensities of radiations emitted in nuclear transformations of the radionuclides has been undertaken as described by Endo and Eckerman[14]. DOCAL will prepare a publication of these data to supersede Publication 38. Because of the magnitude of the data the numerical data will be made available in electronic form.

3.3 Review of Biokinetic Data

INDOS is reviewing the biokinetic models to be used in calculating dose coefficients. This effort has focused on the models and parameters describing the absorption from the respiratory and gastrointestinal tract and the models describing the behavior of the absorbed material; the systemic activity. The efforts include recommending element specific parameters for the forthcoming human alimentary tract and a review of the part experience with the respiratory tract model of ICRP Publication 66[1].

3.4 Occupational intake of radionuclides (OIR)

Following the issuance of the new recommendations, INDOS and DOCAL will prepare for Committee 2 a series publications providing dose coefficients for the occupational intake of radionuclides by inhalation and ingestion. Committee 2 plans to implement the wound model described in a forthcoming report by the National Council on Radiation Protection and Measurements[15] and thus some information will be included for wounds. The first publication in the series is expected to address the radioisotopes of 31 elements. Information on the retention of radionuclides within the body (and organs) and excretion rates following an acute intake will be included in the OIR publication. In addition similar information will be included for wounds. Because of the volume of information and users needing access to the numerical values it is anticipated that the information will distributed electronically.

3.5 Guidance Document on Interpretation of Bioassay Data

INDOS, with input from DOCAL, is preparing a guidance document as an aid in the interpretation of bioassay data. A draft of the document was posted on the ICRP website for consultation. The document will provide guidance on the use of the bioassay information (retention and excretion fractions) included in the OIR publication.

3.6 Other activities

A task group on the radiation exposures in space has been established and held its first meeting in April 2006. Initially the task group's consideration were limited to exposures in low-earth orbit but has expanded to include longer flights. The experience in radiation monitoring at the International Space Station has illustrated a number of problems. For example, during solar flares the doses can be very high and thus it is necessary to consider stochastic and non stochastic effects from exposures to protons, pions, neutrons, alpha particles, and heavy ions. Conversion coefficients for mixed high energy fields are needed for calibration of instruments.

Committee 1 has established a task group on alpha epidemiology which involves some members of Committee 2. The task group meet in January 2006. A review of the data on exposures of human populations to alpha emitting radionuclides will be undertaken. These data sets include medical exposures to Thorotrast and radium, occupation exposures to radon, plutonium and uranium.

4. Conclusions

The activities of the ICRP are carried out within its Committees and their Task Groups by voluntary contribution of individual scientists. The occasion of new recommendations by the Commission provides Committee 2 an opportunity to update its dosimetric methodology. Considerable efforts are underway within the DOCAL and INDOS in preparation of the Commission's new primary radiation protection guidance. The efforts by the Task Group members and other specialist outside the ICRP, called upon for their expertise, are critical to the activities of the Committee 2.

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2-2 Development of Nuclear Decay Data for Radiation Dosimetry Calculation

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Abstract

JAEA and ORNL have jointly developed a nuclear decay database for 1037 radionuclides, which are significant in medical, environmental and occupational exposures. The scheme for the development consisted of the following procedures. (1) Consistency of nuclear structure data file, ENSDF, used as input to compile the decay data was examined by referring to the latest nuclear parameters and by comparing computed energies of radiations with total decay energies. (2) The computer code EDISTR04 was developed and adopted for calculating energies and intensities of atomic radiations and spontaneous fission radiations using updated atomic data and computation methods. (3) Quality assurance of the compiled data was undertaken by comparisons with experimental data and other evaluated libraries. A package of the data files, DECDC2, will succeed the MIRD monograph and ICRP Publication 38 and will be extensively used in dose calculation in various applications.

Keywords: DECDC2, Nuclear decay data, Radionuclide, Dose calculation, Radiation exposure, ENSDF, EDISTR04, MIRD monograph, ICRP Publication 38

1. Introduction

Dose calculation for internal or external exposure to radionuclides requires information on mean or unique energies and intensities of emitted radiations. The nuclear decay data of the monograph of the Medical Internal Radiation Dose (MIRD) Committee[1)] and Publication 38 (ICRP38) of the International Commission on Radiological Protection (ICRP),[2)] compiled at the Oak Ridge National Laboratory (ORNL), have been used in dose calculation in medical, environmental and occupational exposures. The two compilations are integrated into a software package NUCDECAY[3)] including full listings of radiations and β particle spectra.

During the last two decades, the needs in dosimetry calculation have been served by the information presented in the two compilations. However, there has been increasing demand to update and enhance the MIRD monograph and ICRP38 to adopt the latest information on decay properties and to evaluate localized dose distribution at cellular dimensions by nuclides used in diagnostic nuclear medicine.[4)]

In order to solve this issue, it was proposed in the 3rd JAERI–EPA Workshop held in 2001 that the Japan Atomic Energy Research Institute (JAERI) and ORNL collaborate to develop a nuclear decay database, which will supersede the existing compilations.[5] The proposed scheme consisted of (1) Evaluation and consistency verification of nuclear structure data used as input, (2) Improvement of computational methods for atomic radiations and spontaneous fission radiations, and (3) Quality assurance of the compiled data.

This paper describes a new decay database resulted from the collaborative effort between the Japan Atomic Energy Agency (JAEA, formerly JAERI) and ORNL aimed at providing updated compilations of the MIRD Committee and ICRP. A package of the data files was named DECDC2 (Nuclear DECay Data for Dosimetry Calculation, Version 2)[6] and was released in 2005 for review in the Committees.

2. Evaluation of ENSDF

Figure 1 shows procedures for the evaluation and compilation of the nuclear decay data. The decay data were assembled using the two fundamental data files, ENSDF[7] and NUBASE2003[8]/AME2003.[9] The ENSDF, Evaluated Nuclear Structure Data File, is a computer-based data file for evaluated data from experiments for half-lives, nuclear ground and excited states, decay γ-rays and other decay characteristics. A set of ENSDF as of January, 2003, and later versions were used for the present compilation. NUBASE2003 and AME2003 are the databases for nuclear and decay properties in ground and isomeric states.

To achieve unified data expression, fundamental decay properties, such as total decay energies (Q values), branching fractions, excitation energies of isomers, half-lives, and spin and parity values of initial and final states, in the ENSDF were referred to those of NUBASE2003/AME2003 and updated. Then, the ENSDF was processed by the computer code EDISTR04 to calculate the energies and intensities of α particles, β particles, γ-rays including annihilation photons, internal conversion electrons, X-rays, and Auger, Coster-Kronig (CK) and super CK electrons. For spontaneously fissioning nuclides, the average energies and intensities of fission fragments, prompt neutrons, prompt γ-rays, delayed γ-rays, and delayed β particles were also computed. EDISTR04 is an upgraded version of the EDISTR code[10] used for the compilation of the MIRD monograph and ICRP38. Several enhancements in EDISTR04 are discussed in Section 3.

The consistency of the computed radiation data were verified by the following indices and criteria.

(1) The total intensity from the parent to the ground and isomeric states of the decay product is 100 ± 10 %.

(2) The total intensity from the parent is 100 ± 5 %.

(3) The deviation D_Q between the calculated Q value and the theoretical Q value is equal to or less than ±5 %.

$$D_Q(\%) = \frac{\text{Calculated } Q \text{ value} - \text{Theoretical } Q \text{ value}}{\text{Theoretical } Q \text{ value}} \times 100 \qquad (1)$$

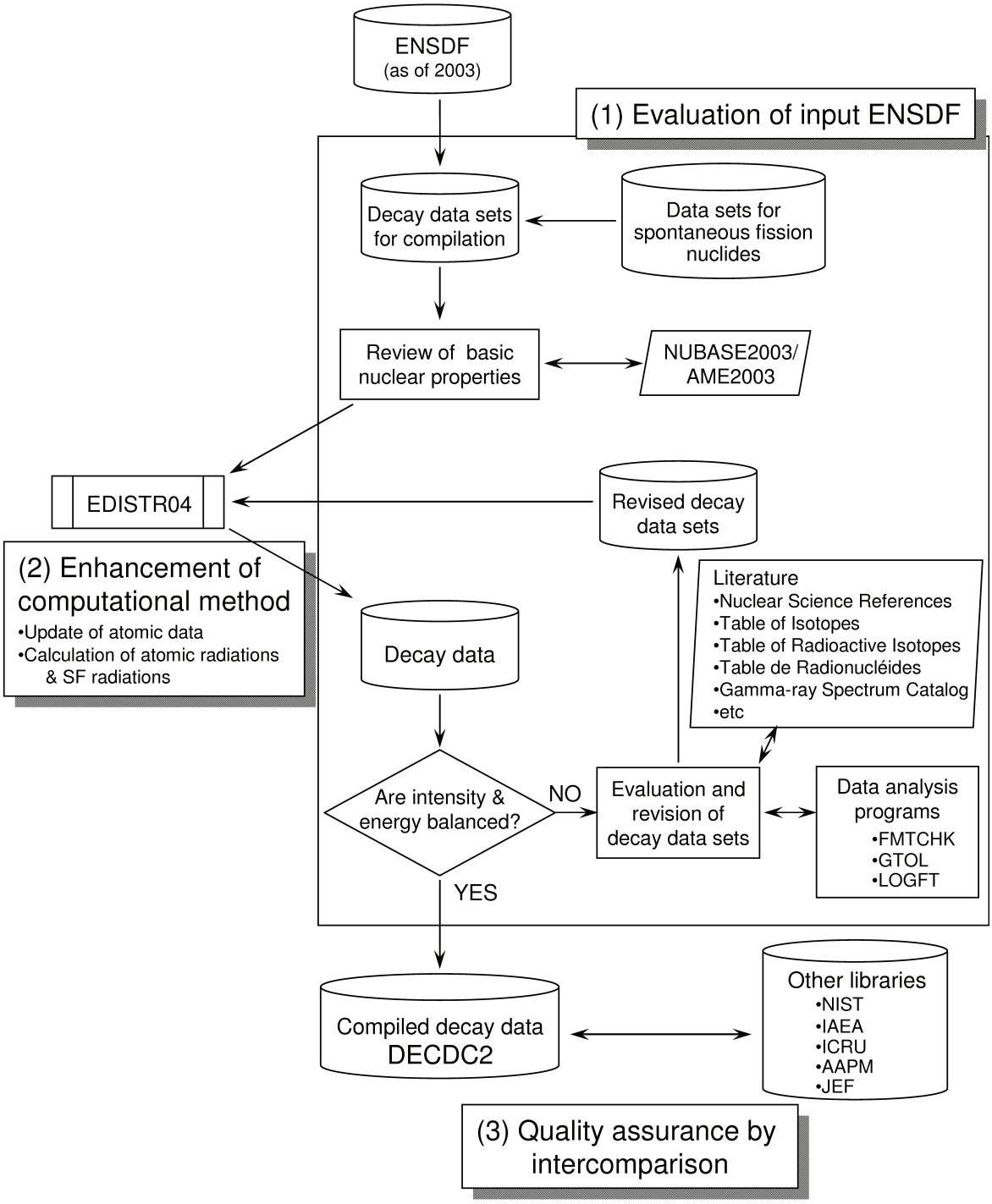

Fig. 1 Flow of the compilation of nuclear decay data

- Calculated Q value $= \sum_{i}^{\text{all }\alpha} E_{\alpha_i} I_{\alpha_i} + \sum_{j}^{\text{all }\beta} E_{\beta_j} I_{\beta_j} + \sum_{k}^{\text{all }\gamma} E_{\gamma_k} I_{\gamma_k} + \cdots$

  where E_{α_i}, E_{β_j}, E_{γ_k}, etc. and I_{α_i}, I_{β_j}, I_{γ_k}, etc. are the energies and intensities of i-th α particle, j-th β particle, k-th γ-ray, etc. from the individual decay process in the decay scheme.

- Theoretical Q value $= \sum_{i=1}^{\text{all }BF} Q_i BF_i$

  where Q_i and BF_i are the Q value and branching fraction of i-th decay mode.

Since the ENSDF data sets are the compilation of experimental data, some data sets are incomplete due to limitations in experimental information and do not satisfy the above criteria. In such cases, the reasons for the imbalance were identified and possible revisions of the ENSDF were made based on the analyses of format and syntax errors, and consistency in net feedings at the ground and excited levels using ENSDF analysis programs.[1] Literature sources relating to the nuclides of interest were searched by Nuclear Science Reference[11)] and were also reflected in the evaluation of the ENSDF. After the analysis and revision, the ENSDF was processed by EDISTR04 again. The process for the production of reliable data has been established by both the consistency check of the ENSDF and the comparison with other libraries discussed in Section 4.

3. Development of Computer Code EDISTR04

The computer code EDISTR was developed at ORNL to compute from the ENSDF the energies and intensities of emitted radiations from nuclear and atomic processes. EDISTR was used for the compilation of both the MIRD monograph and ICRP38. Several new capabilities have been added to EDISTR to compute detailed spectra of X-rays and Auger electrons, which are important in microdosimetry,[12)] for the updated compilation. Methods for calculating the energies and intensities of radiations from spontaneous fission have been updated for dosimetry of transuranium nuclides. The revised computer code is named "EDISTR04" as an upgraded version of EDISTR developed in 2004.

3.1 Enhancement for calculation of energy spectra of X-rays and Auger electrons

Electron capture and internal conversion processes produce inner-shell vacancies in electron orbits in the newly formed atom. This excited atom relaxes to the ground state by migration of the initial vacancy to outer shells via the emission of characteristic X-rays, and Auger, Coster-Kronig (CK) and super CK electrons. EDISTR computes X-rays and Auger electrons originating from electron vacancies in the K- and L-shells, but does not consider a series of radiative and nonradiative transitions as the vacancies move to outer subshells.

In order to calculate the detailed spectra of X-rays and Auger electrons, EDISTR04 uses extensive bound-state electron radial wavefunctions[13)] to assign electron capture subshell ratios and extended internal conversion coefficients.[14,15)] These improvements are essential to determine distributions of vacancies produced in various subshells by the electron capture and internal conversion processes. In

[1]The ENSDF analysis programs, FMTCHK, GTOL, and LOGFT distributed from the National Nuclear Data Center at the Brookhaven National Laboratory.

addition, a method has been introduced into EDISTR04 to calculate X-ray and electron spectra from the atomic process using the relativistic Dirac-Hartree-Slater theory[16] and the evaluated atomic data library EADL.[17] These improvements enable us to calculate high-resolution spectra of X-rays and electrons emitted by Auger and CK transitions, as demonstrated in Fig. 2.

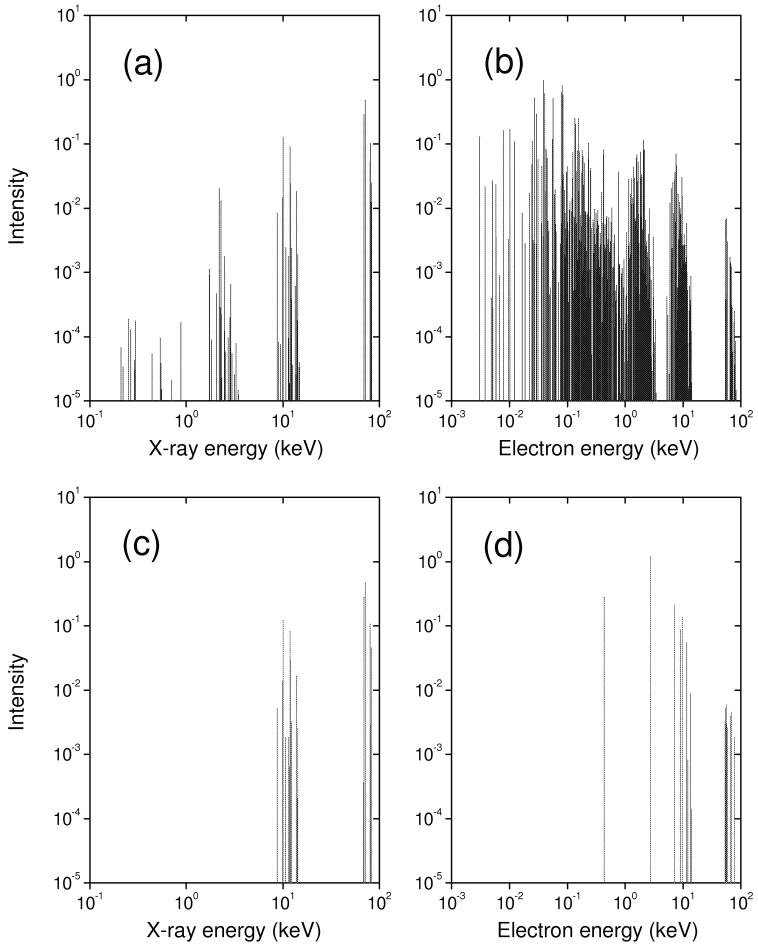

Fig. 2 Comparison of spectra of X-rays and Auger and CK electrons produced by a single electron vacancy in the K shell in Hg. (a) and (b) EDISTR04; (c) and (d) EDISTR.

3.2 Update of calculation methods for spontaneous fission radiations

Spontaneous fission (SF) gives rise to a variety of radiations. From a viewpoint of radiation dosimetry, SF nuclides are of important since even if a nuclide decays via SF with the probability of 1 %, the associated dose will be comparable to or larger than the dose due to all other decay modes.

For fission fragments, the following empirical equation[18] updated by analyzing the latest experimental data is adopted to EDISTR04: $E_{ff} = 0.1189\frac{Z^2}{A^{1/3}} + 7.3$. The equation relates the total kinetic energy of fission fragments, E_{ff}, with the Coulomb parameters, $Z^2/A^{1/3}$, of the fissioning nucleus, where Z and A are the atomic number and pre-fission mass number, respectively, of the fissioning nuclide. The equation predicts the experimental data of E_{ff} within 5 % in the $Z^2/A^{1/3}$ range of 200–2200 covering all SF nuclides.

For prompt neutron, it has been known that the laboratory neutron spectrum shape is well expressed by the Watt spectrum.[19] EDISTR04 incorporates this better approximation and calculates not only the average energy and intensity of prompt neutron but also the energy spectrum.

For prompt γ-ray, a method[20] for predicting the average energy and intensity of prompt γ-ray from fission is employed in EDISTR04. The method enable us to predict the properties of prompt γ-rays depending on *A* and *Z*, while EDISTR adopts the average energy derived from the fission of ^{235}U induced by thermal neutrons to all SF nuclides.

4. Quality Assurance of Compiled Data

Quality assurance of the compiled data in DECDC2 is a significant issue relating to reliability of dose calculation. Equation (1) provides the index for checking internal consistency of the ENSDF data sets. In addition, comparisons with evaluated or reference data are made for the verification of the data in DECDC2.

• Half-life

Half-life values, $T_{1/2}$, of 83 nuclides from DECDC2 have been compared with the two compilations: (1) Measurement in the Radioactivity Group[21] of the US National Institute of Standards and Technology (NIST), and (2) Critical evaluation for detector calibration by a Coordinated Research Project of the International Atomic Energy Agency (IAEA-CRP).[22]

For most radionuclides, the values of $T_{1/2}$ in DECDC2 are in reasonable agreement with those of NIST and IAEA-CRP. Discrepancies beyond the uncertainties are found for several nuclides, such as ^{177}Lu and ^{207}Bi; the maximum difference is 4 % in ^{207}Bi.

• Gamma-ray, X-ray and β particle spectra

The IAEA-CRP has been evaluating a recommended data set for specific radionuclides used for the calibration of equipment used in measurement of γ-ray and X-ray emissions.[22] Good agreements are found in the energies and intensities in most γ-rays between DECDC2 and the IAEA compilation. Disagreement that exceeds the level of 1 % are observed in several emissions with small intensities.

The intensities of X-ray are generally in good agreement between DECDC2 and the IAEA evaluation for elements of low and intermediate atomic numbers, and the disagreement is less than 10 %. However, it should be noted that a discrepancy of about 25 % is found in the L X-ray data of ^{241}Am. The reason is that the theoretical internal conversion coefficients used in EDISTR04 do not reproduce distributions of vacancies in electron orbits that satisfy the intensity of the experimental L X-ray data for ^{241}Am. Although it might not be significant for organ dose calculation, the above discrepancy should be considered when applying the calculated data to quantitative assay of radionuclide content by photon spectroscopy.

Beta particle spectra have been compared for 36 nuclides between DECDC2 and ICRU56,[23] a compendium of data on β particles of the International Commission on Radiation Units and Measurements. It has been found that the β particle spectra in DECDC2 are in reasonable agreement with those of ICRU56.

• SF radiations

The energies released by fission fragments, prompt and delayed radiations from the selected SF nuclides in DECDC2 have been validated by comparison with JEF-2.2[24] (Joint Evaluated File, Version 2.2) and ENDF (Evaluated Nuclear Data File) libraries.[25] The comparison shows that the data are generally consistent between the three compilations.

These comparisons have demonstrated that the compiled data in DECDC2 is reliable and is in consistency with other evaluated and reference data.

5. Impact of New Decay Data

5.1 Comparison with the current ICRP data

To identify the differences in the decay data by the update in the ENSDF and computational methods, the values of $T_{1/2}$ and the energies of radiations have been compared between DECDC2 and NUCDECAY for 832 radionuclides. An index D (%) is used to represent the differences in these values,

$$D(\%) = \frac{V_{\mathrm{NUCDECAY}} - V_{\mathrm{DECDC2}}}{V_{\mathrm{DECDC2}}} \times 100 \qquad (2)$$

where V_{NUCDECAY} and V_{DECDC2} are the values in NUCDECAY and DECDC2, respectively. Table 1 lists the nuclides for which the values of D exceed $\pm$ 25 %. These discrepancies are attributable to the update of the decay energy, $T_{1/2}$ and decay scheme in the ENSDF.

Table 1 Radionuclides with large D values

$T_{1/2}$				Energy of all radiations			
Nuclide	D (%)	Nuclide	D (%)	Nuclide	D (%)	Nuclide	D (%)
^{115}In	1.1E+3	^{123}Te	−98.3	^{190n}Ir[1]	1.8E+3	^{80}Sr	−97.2
^{202}Pb	4.7E+2	^{60}Fe	−93.3	^{123}Te	7.4E+2	^{202}Pb	−75.1
^{32}Si	2.4E+2	^{79}Se	−78.0	^{135}Ce	1.4E+2	^{250}Es	−73.0
^{157}Tb	1.1E+2	^{250}Es	−75.6	^{238}Cm	1.1E+2	^{193}Hg	−72.7
^{41}Ca	37.3	^{108m}Ag	−69.6	^{70}Se	55.2	^{81m}Rb	−72.1
^{218}At	33.3	^{126}Sn	−56.5	^{120m}I	47.6	^{126}Ba	−69.4
^{138}La	32.4	^{194}Hg	−40.9	^{195m}Ir	43.2	^{194}Tl	−46.4
^{121m}Sn	25.3	^{59}Ni	−25.7	^{250}Cm	41.6	^{162}Yb	−42.2
		^{236}Np	−25.3	^{199}Pb	39.4	^{180}Os	−40.8
				^{81}Kr	35.7	^{189}Pt	−35.2
				^{133m}Te	34.3	^{104m}Ag	−33.7
				^{89m}Nb	31.9	^{174}Ta	−31.1
				^{234}Np	29.6	^{173}Lu	−29.7
				^{234}Pa	28.7	^{194}Os	−29.4
				^{170}Hf	25.8	^{116}Te	−27.7
				^{173}Ta	25.5	^{185}Ir	−27.0
						^{153}Tb	−26.8

[1] Isomers are identified in order of increasing excitation energy by appending “m” and “n” to the mass number.

5.2 Influence of update of decay data on dose calculation

To reveal the influence of the decay data revision, effective dose due to inhalation of ^{80}Sr has been calculated using the decay data of DECDC2 and NUCDECAY. Strontium-80 was chosen for this comparison since the significant change in the energy of radiations was found in this nuclide (Table 1). The summary of the decay data for ^{80}Sr is listed in Table 2.

The calculation was carried out using the computer program LUDEP[26] for the conditions of inhalation of a 5 μm AMAD (Activity Median Aerodynamic Diameter) aerosol by a normal nose-breathing adult male worker performing light work. Type S (slow) was selected for the rate of particle dissolution and subsequent uptake in the blood. The integration time following inhalation was 50 y. Strontium-80 produces the radioactive decay product ^{80}Rb. In this calculation, however, the dose from ^{80}Rb was not taken into account in order to determine the discrepancy of the dose due to the update of decay data in ^{80}Sr.

The result is shown in Table 2. The effective dose integrated over 50 y, $E(50)$, per unit activity of ^{80}Sr calculated using DECDC2 is 6.9 times that calculated using NUCDECAY. It is found that the effective dose is obviously increased by the update of the decay data of ^{80}Sr.

Table 2 Comparison of effective dose for ^{80}Sr

	Source of decay data	
	DECDC2	NUCDECAY
$T_{1/2}$	106.3 m	100 m
Energy (MeV) of		
· γ-rays, annihilation photons and X-rays	4.37×10^{-1}	8.00×10^{-3}
· $\beta^{\pm}$ particles, internal conversion electrons and Auger electrons	4.18×10^{-2}	5.40×10^{-3}
$E(50)$ (Sv per Bq)	2.5×10^{-11}	3.6×10^{-12}
Ratio $\frac{E(50)_{\mathrm{DECDC2}}}{E(50)_{\mathrm{NUCDECAY}}}$	6.9	

6. Summary and Outlook

DECDC2 compiles the nuclear decay data of 1037 radionuclides, which consist of 922 radionuclides with $T_{1/2} \geq 10$ min. The proportion that covers the radionuclides with $T_{1/2} \geq 10$ min is extended up to 88 % of the radionuclides in this half-life range, while ICRP38 is limited to 73 %.

The data included are half-lives, decay modes, radiation types and respective energies and intensities, decay chains and branching fractions. Spectrum data are included for β particles, X-rays, Auger and Coster-Kronig electrons, and spontaneous fission neutrons. These data satisfies all requirements for dose calculation from organ to cellular dimensions. The data are prepared in two types of format, ICRP38 and NUCDECAY formats, which are compatible with the previous compilations. It facilitates to update of decay data in computer codes that have been using the ICRP38 and NUCDECAY data.

DECDC2 was released in 2005 and was then forwarded to the MIRD Committee and the ICRP Task Group DOCAL for review for publication. The publication is scheduled to be printed and disseminated from the MIRD Committee in 2006 and from ICRP in 2007.

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2-3 Application of the PHITS Code in High-energy Particle Dosimetry

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Abstract

Estimation of radiation dose from high energy particles is indispensable for the design study of accelerator shielding, radiation therapy, long-term space mission and so on. We therefore developed a general-purpose Monte Carlo code PHITS, which can deal with the transports of all kinds of hadrons and heavy ions with energies up to 200 GeV/n, and calculated fluence to the effective dose and the effective dose equivalent conversion coefficients for high-energy protons, neutrons and several kinds of heavy ions. The effective quality factors were evaluated from the calculated dose conversion coefficients, and compared with the corresponding radiation weighting factors. The comparison result scientifically supported the discussion about the revision of the radiation weighting factors for high-energy particles, which will be included in the next ICRP recommendations.

Keywords: Radiation protection, High-energy particle, Dose conversion coefficient, Computational dosimetry, PHITS

1. Introduction

Workers in high-energy accelerator facilities are supposed to be exposed to high-energy neutrons, which have been barely considered in the radiation protection in nuclear power plants and conventional RI facilities. Furthermore, aircrews and astronauts are subjected to be in an enhanced-level of radiation fields composed of high-energy protons and heavy ions as well as neutrons. Hence, estimations of their radiation doses due to such high-energy particle exposures are indispensable for designing accelerator shielding, keeping the limitation of aircrew-dose and planning long-term manned space missions. Establishment of reliable computational dosimetry is the key issue in the estimation, since measuring dose from high-energy particles has not been easy because of their highly penetrability through radiation monitoring devices. Calculation of fluence to dose conversion coefficients for high-energy particles is of prime importance in the issue.

Several authors have calculated fluence to the effective dose conversion coefficients for high-energy protons[1,2)] and neutrons[2-5)] by performing particle transport simulations in anthropomorphic phantoms. On the other hand, only one set of data has been reported for heavy ions, where the values for α particles were calculated by Yoshizawa et al.[6)] In their simulation, however,

nuclear reactions induced by α particles were simply approximated as the sum of individual neutron and proton reactions. A more precise calculation is therefore required for the α particle as well as for other heavy ions. Estimation of conversion coefficients from fluence to the effective dose equivalent is also requested from space radiation health investigators, since doses for astronauts are generally evaluated in terms of the effective dose equivalent instead of the effective dose.

In order to meet these requests, it is necessary to develop a computer code that can simulate the transport of various particles including heavy ions inside human body. we have therefore developed a general-purpose Monte Carlo code PHITS[7] (Particle and Heavy Ion Transport code system), which can deal with the transports of all kinds of hadrons and heavy ions with energies up to 200 GeV/n. PHITS is based on the high-energy hadron transport code NMTC/JAM,[8] and incorporates the MCNP4 code[9] for low-energy neutron and photon transports, and the JAERI Quantum Molecular Dynamics (JQMD) model[10] for simulating nucleus-nucleus interactions. An advantage of utilizing the code for high-energy particle dosimetry is that it can calculate the deposition energy from neutron without using the Kerma approximation, *i.e.* it can explicitly determine the type and energy of secondary charged particles that cause ionization instead of neutron. This function, referred to "event generator mode" in our papers, enables us to evaluate dose equivalent directly by employing the $Q(L)$ relationship. A detail about this function is given in our forthcoming paper,[11] together with the description of recent development of the code.

Using the code coupled with an anthropomorphic phantom, we have calculated the fluence to effective dose and effective dose equivalent conversion coefficients for protons, neutrons with energies up to 200 GeV,[12] and several kinds of heavy ions up to 3 GeV/n.[13,14] This paper describes the details of the calculation procedure together with the results of the calculated dose conversion coefficients. The discussion about the comparison between the radiation weighting factor and the effective quality factor obtained from the calculated dose conversion coefficients is also presented below.

2. Calculation Procedure

2.1 Fluence to effective dose conversion coefficient

The effective dose is defined in ICRP60,[15] and can be derived from the absorbed doses averaged over a organ (or tissue, this extension of the meaning of "organ" is applicable hereafter) in human body. We therefore performed Monte Carlo simulations to calculate absorbed doses in organs in an anthropomorphic phantom MIRD5[16] using the PHITS code.

In the simulation, the phantom was assumed to be irradiated by protons and neutrons with energies up to 200 GeV, and deuterons, tritons, ^{3}He, α particles, ^{12}C, ^{20}Ne, ^{40}Ar, ^{40}Ca, and ^{56}Fe up to 3 GeV/n with the isotropic (ISO) and anterior-posterior (AP) geometries. The ISO geometry is the most realistic model for simulating radiological situations in space, while the AP geometry is the most conservative model for low energy irradiations. The maximum incident energy for heavy ions, 3 GeV/n, corresponds to the energy up to which PHITS used to be able to treat nuclear reactions of heavy ions. Note that the energy was extended to 200 GeV/n in the current version of the code.

The absorbed dose in each organ was simply obtained from its deposition energy divided by its

mass, except for those in gonads, red bone marrow, bone surface and remainders. The value in the gonads was determined by the arithmetic mean of ovary and testis doses, as defined in ICRP74.[17] The absorbed doses in red bone marrow and bone surface were calculated from those in skeletal tissues by considering the mass ratio of those organs in each skeleton,[18,19] since the skeletal tissue in MIRD5 consists of a homogeneous mixture of those organs. The remainder dose was evaluated from the mass weighted average of the absorbed doses in the 10 organs specified in ICRP60. The absorbed dose in muscle, which predominantly contributed to the remainder dose because of its heavy weight, was determined by calculating the absorbed dose in the remaining parts of the phantom after excluding all specified organs except for the heart.

The fluence to effective dose conversion coefficient E/ϕ can be simply determined by

$$\frac{E}{\phi} = w_{\mathrm{R}} \sum_{\mathrm{T}} w_{\mathrm{T}} \frac{D_{\mathrm{T}}}{\phi}, \tag{1}$$

where w_{R} and w_{T} are the radiation and tissue weighting factors given in ICRP60, respectively, and D_{T}/ϕ denotes the absorbed dose averaged over organ T per unit fluence, which was obtained from the PHITS simulation.

2.2 Fluence to effective dose conversion coefficient

In order to calculate the conversion coefficients from fluence to the effective dose equivalent that is defined in ICRU report 51 (ICRU51),[20] the dose distribution with respect to the LET$_\infty$ of ionizing particle (hereafter, referred to dose-LET distribution) in each organ must be determined. We therefore performed the PHITS simulation for estimating the dose-LET distribution in each organ of the human phantom, employing the event generator mode. The simulation conditions such as incident-particle energies and irradiation geometries are the same as those adopted in the calculation of the effective dose conversion coefficients.

The fluence to effective dose equivalent conversion coefficients H_{E}/ϕ can be obtained by

$$\frac{H_{\mathrm{E}}}{\phi} = \sum_{\mathrm{T}} w_{\mathrm{T}} \int_{L} \frac{D_{\mathrm{L,T}}(L)}{\phi} Q(L)\mathrm{d}L, \tag{2}$$

where L is the LET$_\infty$ of ionizing particle in water, $D_{\mathrm{L,T}}(L)/\phi$ denotes the dose-LET distribution averaged over organ T per unit fluence, and $Q(L)$ indicates the radiation quality factor. Note that $Q(L)$ and w_{T} defined in ICRP60 were also adopted in this calculation.

3. Results and Discussion

3.1 Dose conversion coefficient

The numerical data for the calculated dose conversion coefficients were presented in our previous papers.[12-14] As examples, the calculated E/ϕ and H_{E}/ϕ for neutrons, protons, α particles, and ^{12}C, ^{20}Ne and ^{56}Fe ions for the isotropic geometry are depicted in Figure 1. The statistical uncertainties (fractional standard deviations) are generally small, less than 3% in most cases. The results obtained by using other simulation codes, FLUKA[1,5,21,22] and HETC-3STEP[2,23] are also plotted in the graphs. Note that the FLUKA results of H_{E}/ϕ were estimated from the corresponding E/ϕ and the effective

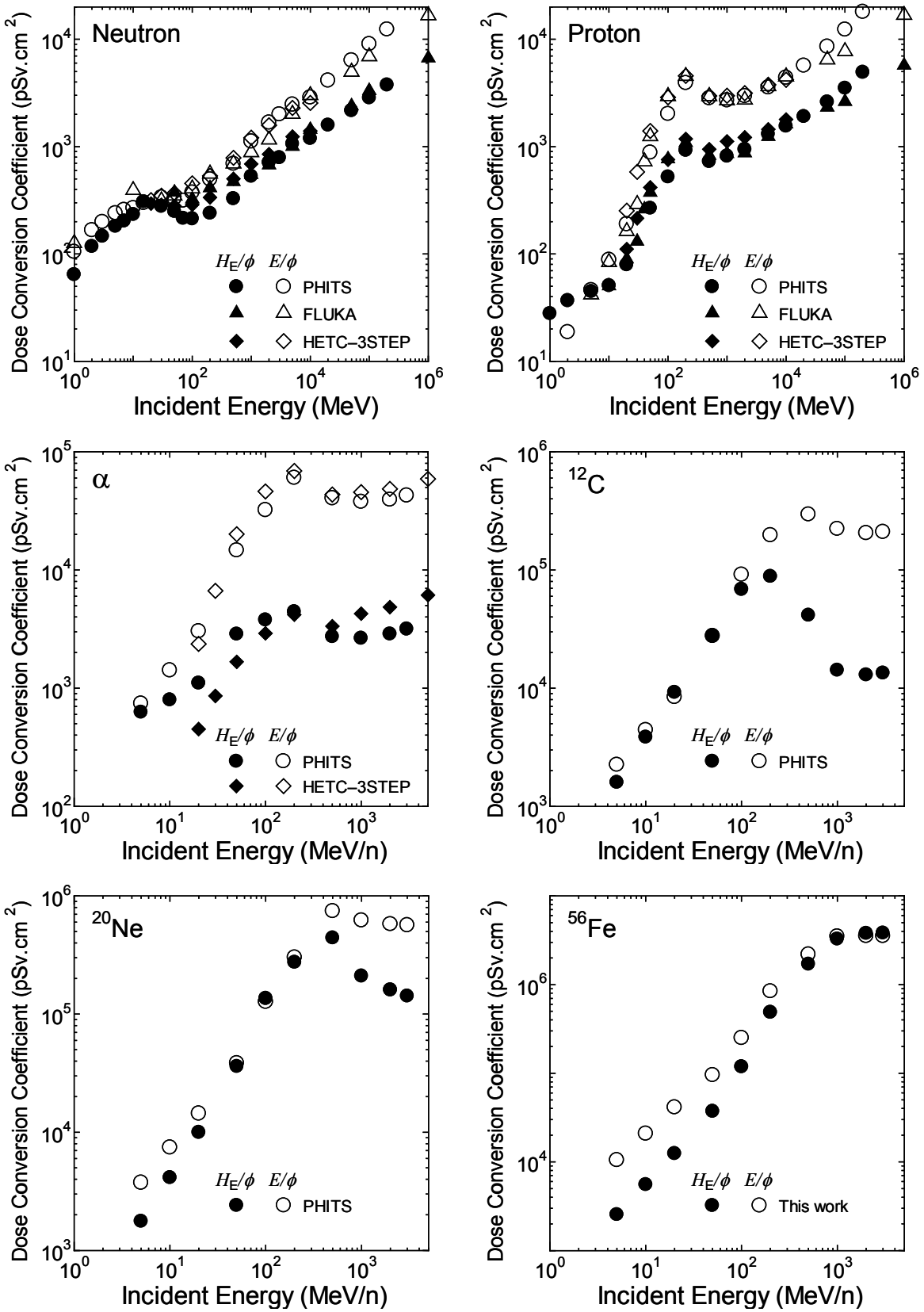

Figure 1 Calculated dose conversion coefficients for neutrons, protons, α particles, and ^{12}C, ^{20}Ne and ^{56}Fe ions for the isotropic geometry

quality factor q_E for the AP irradiation geometry,[24] where the detailed discussion of q_E is given in the next section.

It is found from the figure that our results agree fairly well with those obtained by the other simulation codes, indicating the validity of the calculation methods used not only in the present work but also in the other studies. Our simulation, however, did give greater values than those obtained by FLUKA[1,5] for the dose conversion coefficients for protons and neutrons with energies above 50 GeV. It is possible that this discrepancy is due to the difference of the high-energy nuclear reaction models incorporated in the simulation codes. On the other hand, the values below 200 MeV given by this work are systematically smaller than those obtained from the other studies. This disagreement is mainly attributed to the difference in the phantoms adopted in the simulations, and also that in the calculation methods for the absorbed dose in gonads, since they employed higher values of doses in ovaries and testes. The disagreements observed in the data for α particles obtained by PHITS and HETC-3STEP are predominantly due to the difference in the treatment of α particle-induced nuclear reactions between the codes, since PHITS adopts more sophisticated model, JQMD, for simulating nucleus-nucleus interactions in comparison with HETC-3STEP.

3.2 Effective Quality Factor

After the publication of ICRP60, the concept of w_R together with its numerical values has been extensively discussed among the specialists of radiation protection dosimetry, and the discussion was summarized in the ICRP92.[25] The inconsistency between the numerical values of w_R and the corresponding effective quality factor as defined by Eq. (4.6) in ICRP92 is one of the key issue in the discussion. We therefore evaluated the effective quality factors q_E from the calculated dose conversion coefficients E/ϕ and H_E/ϕ, using the equation

$$q_E = \frac{H_E / \phi}{(E / \phi) / w_R}. \tag{3}$$

Figure 2 shows the calculated q_E for neutrons and protons (panel A), and α particles, and ^{12}C, ^{20}Ne and ^{56}Fe ions (panel B) for the isotropic geometry. The results obtained by using other simulation codes are also plotted in the graphs. Note that the FLUKA data shown in the graph are the values for the AP irradiation geometry, but it is found from our simulation that q_E is almost independent of the irradiation geometry except for lower incident energy cases.

It is evident from the panel A that the effective quality factors for high-energy protons obtained by all the simulation codes are generally less than 2, supporting the proposal of $w_R = 2$ for cosmic-ray protons given in ICRP92. The effective quality factors for neutrons calculated by PHITS are systematically smaller than those by the other codes. This discrepancy is probably attributed to the difference in the calculation methods for the dose-LET distributions due to lower energy neutron transport, where PHITS employs the event generator mode in the calculation.

The effective quality factors for heavy ions are significantly related to the LET of the incident particles, whose graphical presentation is given in Figure 3. In general, q_E becomes the largest – approximately 20 – at the LET of the incident particle around 100 keV/μm at which the $Q(L)$

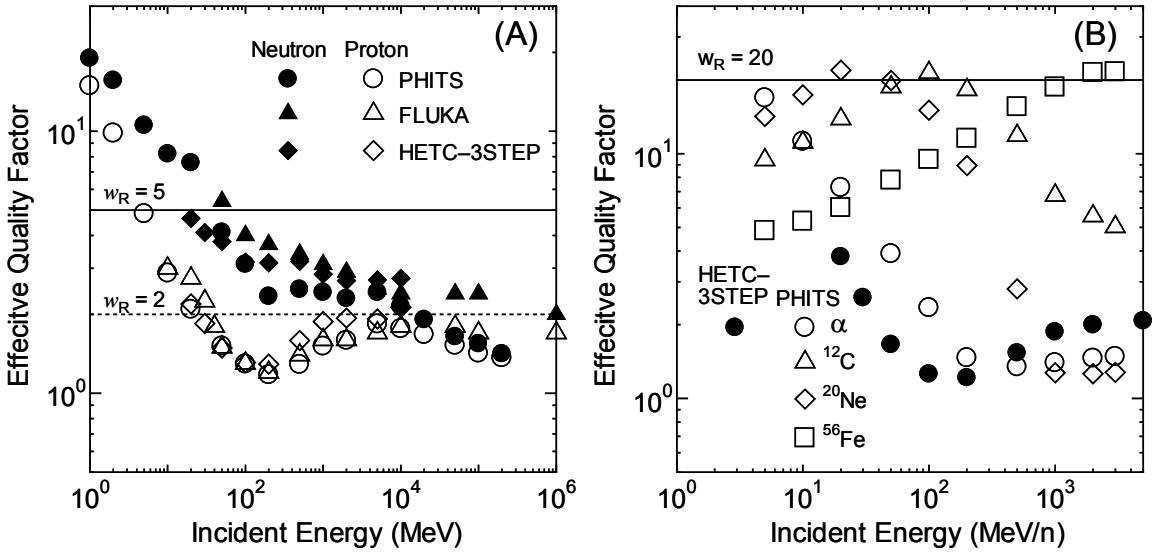

Figure 2 Calculated effective quality factors for neutrons and protons (panel A), and α particles, and ^{12}C, ^{20}Ne and ^{56}Fe ions (panel B) for the isotropic geometry

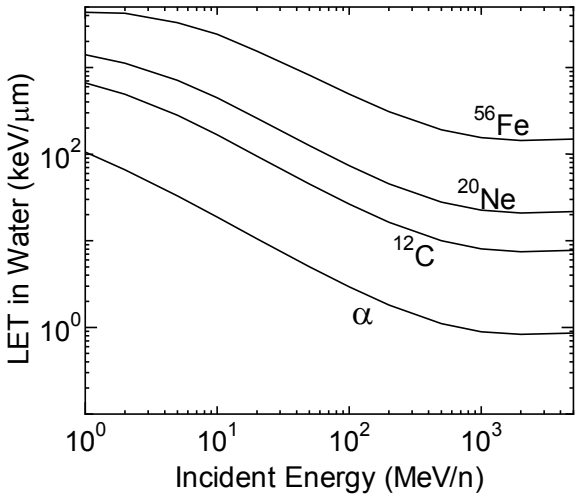

Figure 3 LET$_\infty$ of α particles, and ^{12}C, ^{20}Ne and ^{56}Fe ions in water

relationship has the peak value. Hence, the current definition of w_R = 20 for heavy ions is fairly adequate for the purpose of conservative dose estimation. However, the effective quality factors are generally smaller than 20 especially for lighter ions with higher incident energies, where the values are less than 2 – approximately 1/10 of their w_R value. This inconsistency brings over-conservativeness into the effective dose due to high-energy heavy ion exposure, and this is the primary reason why doses for astronauts are generally evaluated in terms of the effective dose equivalent instead of the effective dose.

4. Conclusions

Fluence to dose conversion coefficients for the effective dose and the effective dose equivalent

were calculated by the PHITS code coupled with the MIRD5 phantom for neutrons, protons and several kinds of heavy ions. The effective quality factors q_E were evaluated from the calculated dose conversion coefficients, and compared with the corresponding radiation weighting factors w_R. The comparison indicates that the current w_R values are fairly adequate for the purpose of conservative dose estimation, but for some cases, they are much larger than the corresponding q_E especially for high-energy protons and lighter ions. These results scientifically supported the discussion about the revision of the w_R values, which will be included in the next ICRP recommendations.

In the future, we plan to calculate the dose conversion coefficients by PHITS, using two voxel phantoms; one is for Caucasian authorized by ICRP and the other for Asian developed in JAEA. The results will be provided to the DOCAL group of ICRP Committee 2, and also released as a compilation database of the dose conversion coefficients for external exposure.

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2-4 Development of Japanese Voxel Models and Their Application to Organ Dose Calculation

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Abstract

Three Japanese voxel (volume pixel) phantoms in supine and upright postures, which are consisted of about 1 mm^3 size voxels, have been developed on the basis of computed tomography (CT) images of healthy Japanese adult male and female volunteers. Their body structures are reproduced more realistically in comparison with most existing voxel phantoms. Organ doses due to internal or external exposures were calculated using the developed phantoms. In estimation of radiation dose from radionuclides incorporated into body, specific absorbed fractions (SAFs) for low energy photon were significantly influenced by the changes in postures. In estimation of organ doses due to external exposures, the doses of some organs of the developed phantom were calculated and were compared with those of a previous Japanese voxel phantom (voxel size: 0.98 × 0.98 × 10 mm^3) and the reference values of ICRP Publication 74.

Keywords: Voxel phantom, Japanese, Organ doses, Specific absorbed fractions, Posture

1. Introduction

Organ doses are fundamental quantity to estimate the radiation risk. Since the organ doses of the living persons cannot be measured directly, dose conversion coefficients that relate a specified dosimetric quantity to organ doses have been used for dose evaluation in radiation protection. The dose conversion coefficients have been calculated using radiation transport codes based on Monte Carlo methods in conjunction with computational human phantoms such as mathematical phantoms[1)].

In recent years, realistic human phantoms have become available on the basis of medical images such as CT and magnetic resonance imaging (MRI) of actual persons. The organs and tissues of such phantoms are defined by aggregate of small rectangular block units called voxel, and the phantom consisting of voxels is called a voxel phantom. The organ shape of a voxel phantom can be modeled with high accuracy using a small voxel size, and so far many voxel phantoms have been developed for the purposes of radiation protection[2-9)]. However, most of the voxel phantoms were based on Caucasian anatomical data[2)]. The International Commission on Radiological Protection (ICRP) will

use the adult male and female voxel phantoms on the basis of updated reference anatomical and physiological data given in ICRP Publication 89[10] to calculate new dose conversion coefficients[11].

The Japan Atomic Energy Agency (JAEA) has been developed a series of CT-based Japanese voxel phantoms to provide a set of basic data of radiation protection for Asian. So far, five adult phantoms for three males (Otoko[3], JM[12] and JM2) and two females (Onago[13] and JF[13]) have been completed. The developed Japanese voxel phantoms were used to calculate organ doses due to diverse exposures[3,4,12,13]. The Otoko and Onago phantoms are the first Asian male and female voxel phantoms, respectively. The voxel size of two voxel phantoms was 0.98 mm × 0.98 mm × 10 mm. In most cases, this voxel is enough size to estimate organ doses. However, in some cases where the deposited energy of small or thin organs is evaluated, more realistic voxel phantom may be needed. The JM, JM2 and JF phantoms are high-resolution phantoms, whose voxel size is 0.98 mm × 0.98 mm × 1 mm. The small voxels make it possible to model realistically the shapes of small or thin organs and to estimate accurately organ doses. The JM2 phantom was constructed from CT data in upright posture of the same subject employed for construction of JM. Thus, it is possible using JM2 and JM to compare directly the differences in organ doses and SAFs due to the change in postures. The five Japanese voxel phantoms, together with radiation transport code system based on the Monte Carlo method, can be applied to calculate the dose conversion coefficients under various exposure conditions.

This paper describes the development of the three Japanese voxel phantoms named JM, JM2 and JF and their applications to organ dose calculations.

2. Materials and Methods

2.1 Volunteers and CT scans

Healthy Japanese adult male and female subjects were recruited to take the CT images in supine or upright postures for the constructions of Japanese voxel phantoms. The male subject for the phantom in supine posture was selected as a volunteer again to obtain the CT data in upright posture. The ages of male and female volunteers were 54-year-old and 54-year-old, respectively.

CT scans were performed after the approval for the plans and objectives by the Ethics Committee of the Fujita Health University Hospital. The whole body CT scans were performed to construct the JM and JF phantoms. A helical CT scanner (Aquilion, Toshiba Medical Systems Co. Ltd., Japan) was used to obtain the CT images (512 × 512 pixels with 1 mm slice) in supine posture. The CT scans in upright postures were carried out using a cone-beam CT scanner (Hitachi Ltd, Japan) to obtain the CT images (512 × 512 pixels with 0.5 mm slice) for the JM2 phantom. The view fields of the CT scanner were limited to a spherical field with a diameter of 25 cm. The four regions with different heights were scanned in order to cover the trunk area of the volunteer. The four scan regions were conjoined on the basis of the position of vertebral column to obtain a data set of the CT images of the trunk. The cone beam CT scanner cannot scan the adipose, bone, muscle and skin located in the periphery of trunk area, because of a spherical scan field. The unscanned trunk areas of JM2 were complemented by using the segmented images of JM. After the completion of trunk area of JM2, the parts of arms, brain and legs of JM were used to construct the whole body of JM2.

CT scans were carried out under the conditions mentioned below. To obtain distinct CT images of gall bladder and urinary bladder, the volunteers had no food intake and urination for several hours before the scanning. They drank 250 ml of warm green tea just before the scans so as to meet the conditions of the stomach content of ICRP reference values[10]. They held their breath and closed the eyes and mouth during the scans

2.2 Constructive methods of voxel phantom

Since image-processing techniques are required for the segmentation, commercial imaging software called Visilog 4 (Noesis Vision Inc, France) and Adobe Photoshop 5.5 for Windows (Adobe Systems Incorporated, USA) are used with SGI O2 workstation (Silicon Graphics Inc, USA) and DELL PowerEdge 600SC (Dell Inc, USA). The segmentations were carried out according to previous method[3,6,8]. The grey values threshold were used to segment organ regions. The grey values are closely related to the electron densities of pixels corresponding to each organ region. If the organs and tissues are unable to segmented by single grey values threshold, the image-processing techniques such as erosion, dilation and filling holes are used to segment the organs and tissues. After segmentations, the region specific identification numbers are assigned to voxels belonging to each organ region in order to identify the segmented regions.

The skin of the JM, JM2 and JF phantoms are assumed to be one voxel layer at the outer surface of the body. The skin thickness in the three phantoms is about 1 mm and is close to the reference value (1.3 mm)[10].

2.3 Calculation of organ doses and SAFs

The EGS4 (Electron Gamma Shower Version 4)[14] - based SAFs calculation system, UCSAF[4], was used to calculate SAFs for photons. The UCSAF was installed on the Kansai ITBL super computer system, PRIMEPOWER (Fujitsu Limited, Japan) of JAEA. Monoenergetic photon sources from 0.01 MeV to 4 MeV were assumed to be distributed uniformly in source organs. The conditions of SAFs calculation were determined to obtain a fractional standard deviation below 5 %.

The EGS4 - based organ doses calculation system, UCPIXEL[3], was use to calculate organ dose due to external exposure of photon. The phantoms placed in a vacuum space were irradiated by monoenergetic parallel photon beams ranged from 10 keV to 10 MeV. Irradiation geometries were anterior-posterior (AP), posterior-anterior (PA), right lateral (RLAT), left lateral (LLAT), isotropic (ISO) and rotational (ROT).

Since the hard bone (cortical bone and trabecular bone) and bone marrow cannot be segmented from CT images, the bone tissues were treated as a composite tissue, “skeleton”, which consists of hard bone and bone marrow with different densities. The organ doses of hard bone and bone marrow were evaluated from deposited energy based on the weight ratios of both tissues. The weight fractions of hard bone and bone marrow in voxels of skeleton region were quantified on the basis of grey value threshold according to the method of the previous papers[3,6,8].

3. Results and discussions

3.1 Physical characteristics

Table 1 shows the physical characteristics of Japanese adult voxel phantoms developed at JAEA and average adult Japanese. The height and weight of the JM and JM2 phantoms are almost the same as the body sizes of the Otoko phantoms and the average Japanese adult male[15]. The height and weight of the JF phantom are smaller than the averages of Japanese adult females[15]. The Onago phantom has large body size as compared with JF and the Japanese averages[15].

Table 1 Physical characteristics of Japanese adult voxel phantoms developed at JAEA and average adult Japanese.

	JM	JF	JM2	Otoko	Onago	Japanese averages[15]	
Gender	Male	Female	Male	Male	Female	Male	Female
Height (cm)	171	152	171	170	162	170	155
Weight (kg)	65	44	65	65	57	64	52
Postures	Supine	Supine	Upright	Supine	Supine	-	-

The body structures of JM, JM2 and JF are reproduced more realistically. It was found that the postures affected the body structures of JM2 and JM. The spine of JM2 was a little backward-bent to keep body balance in contrast with JM. The slight inflation of the lower abdomen of JM2 was observed in comparison with JM. It can be considered that the inflation is caused by the descent of abdomen area organs due to the changes in posture.

Table 2 shows some examples of organ distances between the centers of gravity of organs or organ contents (referred to as organ distances) of the JM2 and JM phantoms. The movement of the esophagus, lower large intestine wall (LLIW), lungs and urinary bladder wall (UBW) are not found in

Table 2 Distances between the centers of gravity of organs or organ contents (organ distance) of the JM2 and JM phantoms.

Organs and organ contents	Organ distances (mm)		Ratios of JM2 to JM
	JM	JM2	
Brain and esophagus	322	318	0.99
Brain and LLIW*	710	713	1.00
Brain and lungs	355	351	0.99
Brain and UBW**	748	748	1.00
Brain and GBW***	510	529	1.04
Brain and kidneys	544	563	1.03
Brain and liver	480	490	1.02
Brain and stomach wall	511	526	1.03
Kidneys and liver	94	98	1.04
Kidneys and LLIW	174	155	0.89
Kidneys and pancreas	64	62	0.97
Stomach content and liver	109	116	1.06
Stomach content and LLIW	206	198	0.96
Stomach content and pancreas	31	47	1.52

*LLIW : Lower large intestine wall
**UBW : Urinary bladder wall
***GBW : Gall bladder wall

JM2 and JM. The locations of the gall bladder wall (GBW), kidneys, liver and stomach wall of JM2 shifted toward the direction of the leg about 10 mm – 20 mm. These results indicate that there are mobile and non-mobile organs at the time of the changes in postures, and mobility of organs would affect the SAFs.

3.2 SAFs for photons

Figure 1 (a) shows the ratios of SAFs of JM2 to those of JM for the kidneys as a source organ and for the liver, LLIW and pancreas as target organs. At the energy ranges from 0.05 MeV to 4 MeV, the SAFs for target organs changes with organ distances. For instances, the SAFs for LLIW of JM2 are higher than those of JM. The differences are caused by the facts that the organ distance from kidneys to LLIW in JM2 is shorter than that in JM (Table 2). At the energy ranges from 0.01 MeV to 0.03 MeV, the differences in SAFs cannot be explained by only the differences in organ distances. In particular, the changes in the SAFs for pancreas cannot be expected from the differences in organ distances. The discrepancy in the relationship between organ distances and the SAFs seems to be caused by the short mean free path of low energy photon and the differences in the 3-D organ arrangement.

Figure 1 (b) shows the ratios of SAFs between JM2 and JM for the stomach content as a source and liver, LLIW and pancreas as target. Similarly to the case that the source organ is the kidneys, the SAFs for high energy photons are primarily dependent on the organ distances. However, the dependence of SAFs on the organ distances is not observed in pancreas with respect to the low energy ranges. The pancreas has the flexibility to change easily its shape and is directly connected to the around organs and tissues such as the LLIW, kidneys, and stomach wall through connective tissue. These anatomical characteristics induce the arrival of the low energy photons to the region of pancreas and contribute to the elevation of SAFs for pancreas of JM2.

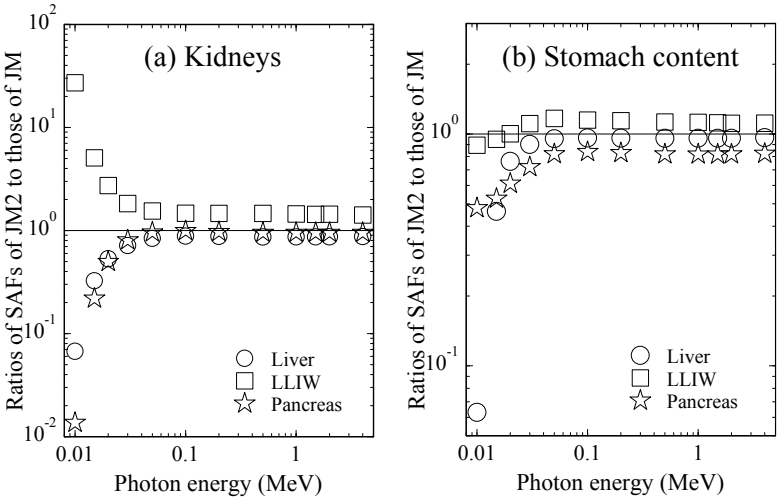

Figure 1 Comparisons of SAFs based on JM2 and JM for sources in (a) kidneys or (b) stomach content and for targets in liver, LLIW and pancreas.

Figure 2 shows the SAFs for (a) esophagus, (b) lungs, (c) LLIW and (d) UBW as target organs of the JM and JM2 phantoms, when 16 organs are assumed as source organs. There are two pronounced

tendencies of the distributions of ratios of the SAFs (JM2 to JM). The tendencies of SAFs are not correlated with photon energy. The SAFs for esophagus and lungs of JM2 tend to be lower than those of JM. The opposite tendency is found in the SAFs for LLIW and UBW. The SAFs for LLIW and UBW of JM2 tend to be higher than those of JM. It is considered that the tendencies are generated from the differences in moved distances of organs.

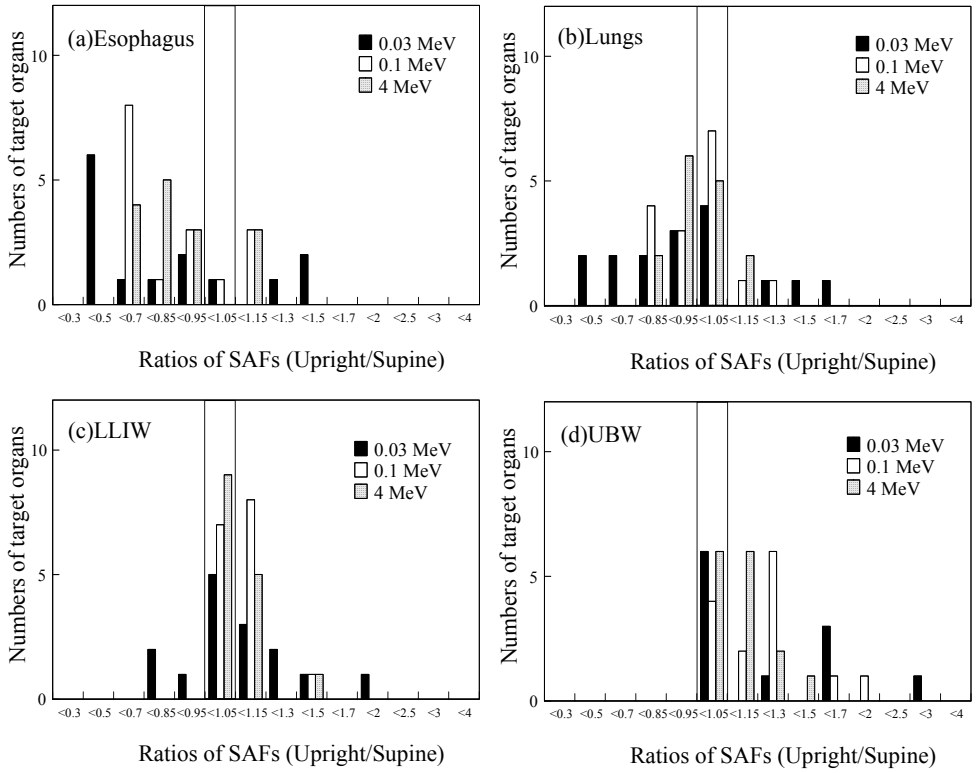

Figure 2 Distribution of the ratios of the SAFs for (a) esophagus, (b) lungs, (c) LLIW and (d) UBW for of JM2 (upright) to those of JM (supine).

3.3 Organ dose due to external exposures

Figure 3 shows the calculated absorbed doses per unit air kerma for (a) kidneys, (b) LLIW, (c) liver and (d) lungs of JM and Otoko[3)] and the reference values of ICRP Publication 74[16)] for AP geometry. The differences in the organ doses between JM and Otoko are found in kidneys and liver at 0.04 MeV and are about 70 % and 50 %, respectively, although the height and weight of JM are very close to those of Otoko (Table 1). The locations of kidneys and LLIW of JM are located deeply in body compared with those of Otoko, and the abdomen of JM is slightly inflated than that of Otoko. The characteristics of their body structures are responsible for the differences in organ doses.

On the other hand, the absorbed doses of liver and lungs of JM for incident photon energy of 0.04 MeV are smaller than those of Otoko about 19 % and 14 %, respectively (Figure 3 (c) and (d)). In energy range from 0.1 MeV to 10 MeV, the doses of JM agree well with those of Otoko and the reference values of ICRP Publication 74 within 10 %. For AP geometry, the organs such as lungs and liver are shielded by only chest. Their chests have no significant differences in thickness. These results

indicate that the organ doses were strongly affected by not only body structures but also irradiation geometry.

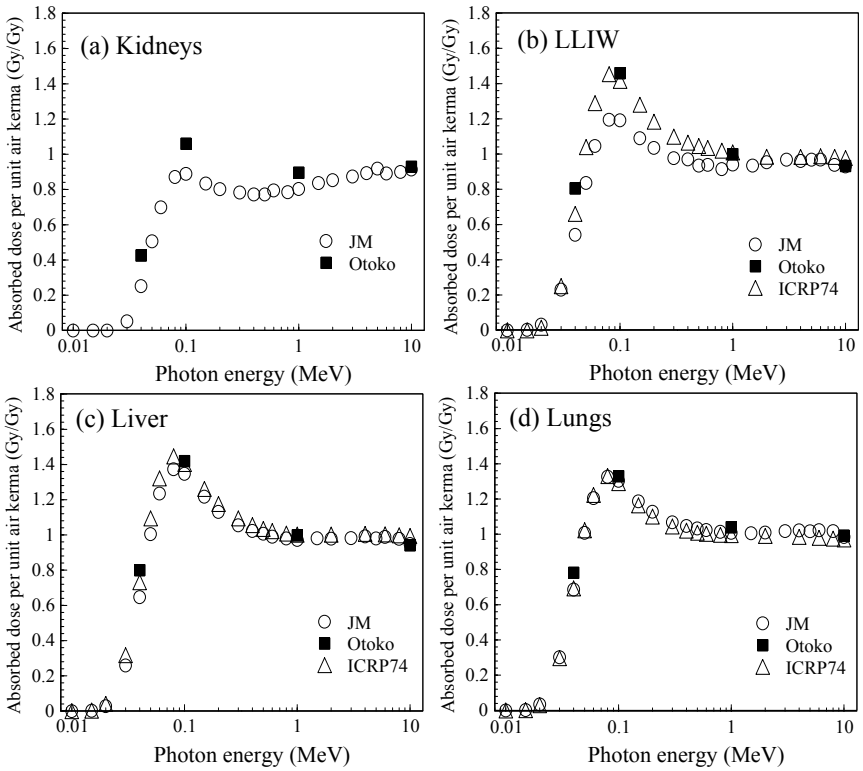

Figure 3 Comparison of absorbed dose per unit air kerma of (a) kidneys, (b) LLIW, (c) liver and (d) lungs between JM, Otoko[3] and the reference values of ICRP Publication 74[16] for AP geometry.

5. Conclusion

In the present study, Japanese adult male and female voxel phantoms named JM (supine), JF (supine) and JM2 (upright) have been constructed on the basis of high-resolution CT-images. It is found that the body structures such as organ locations and body shapes strongly affect the determination of the organ doses due to internal or external exposures. The body structures are significantly dependent on the postures. These results indicate that the posture is important factors for determination of organ doses.

Acknowledgement

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2-5 The United States Transuranium and Uranium Registries (USTUR): Learning from Plutonium and Uranium Workers

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Abstract

Beginning in the 1960's with the mission of acquiring and providing precise information about the effects of plutonium and other transuranic elements in man, the USTUR has followed up to 'old age' almost 500 volunteer Registrants who worked at weapons sites and received measurable internal doses. While failing (despite careful life-time follow-up) to demonstrate deleterious health effects attributable to transuranic elements, USTUR research, based on these real human data from DOE workers, continues its contributions to the development of the biokinetic models used internationally to assess intakes from bioassay data and predict tissue doses.

There is still much to learn from the Registries' 370 deceased tissue donors and the 110 still-living Registrants, whose average age is now about 76 years (youngest < 35 y; oldest > 95 y). This paper illustrates USTUR's current 5-y research program, including the application of registrant case data to (i) quantify the variability in behavior of transuranic materials among individuals; (ii) validate new methodologies used at DOE sites for assessing 'realistic' tissue doses in individual cases; and (iii) model the effectiveness of chelation therapy. These data can also be used to examine the adequacy of protection standards utilized for plutonium workers in the early years of the nuclear industry.

Keywords: USTUR, Transuranium registry, Uranium registry, Pu workers, Pu biokinetic modeling, Pu bioassay, Pu tissue contents, Pu internal dose, Autopsy, Radiation protection

1. Introduction

The United States Transuranium and Uranium Registries (USTUR) began in 1968 with the establishment of the National Plutonium Registry.[1)] In 1970, the name was changed to the United States Transuranium Registry to reflect a broader concern with the entire spectrum of transuranium elements. In 1978, a separate United States Uranium Registry was created to study the uranium decay series. With the goals of understanding the biokinetics, dosimetry, and potential health effects of transuranic elements and uranium based on actual human experience, the two registries were

administratively joined in 1992, when responsibility for USTUR was transferred to Washington State University (WSU).

The USTUR is a human tissue research program studying actinide elements deposited within the body in persons with known, documented exposures to those elements. Voluntary tissue donors allow access to their employment histories, occupational exposure histories, and medical records. That information, together with an autopsy report, and the results of radiochemical analyses of the radionuclide content of major body organs, enables USTUR (the Registries) to compile and maintain a unique and comprehensive collection of scientific data tracing the human experience of accidental exposures to plutonium, americium and uranium over the history of U.S. nuclear materials production. All records of registrants are kept secure to ensure the privacy of USTUR donors. However, the 'de-identified' results of the Registries' and its earlier collaborating laboratories' research are extensively published.[2-10] These publications have contributed critical human data used in the development of the International Commission on Radiological Protection's (ICRP's) current suite of biokinetic and bioassay models for the actinides.[11,12]

The Registries' research program continues to contribute to ICRP's (and the U.S. National Council on Radiation Protection and Measurements') further development and application of these models,[13-17] in particular, to testing their capability to model accurately USTUR's bioassay, health physics and tissue concentration data from individual donor cases.[18-20] This ongoing research also tests with definitive human data the performance of methods for bioassay analysis and actinide internal dose assessment implemented in software[21,22] recently designated by the U.S. Department of Energy for regulatory dose assessments.[23,24]

2. USTUR's Registrants

Currently, USTUR maintains a secure (privacy-protected) file archive containing original paper records (and an electronic database) of administrative, health physics, bioassay, medical records, pathology findings, and results of radiochemical analyses of tissue samples for 370 deceased donors, and employment records for 110 living registrants. Registration is purely voluntarily. As approved by WSU's Institutional Review Board, in order to remain actively registered every potential donor must confirm in writing every 5 years that they still wish to donate their tissues at autopsy for USTUR study. A potential donor can withdraw their permission at any time, as can the donor's family on the donor's death. The majority of USTUR Registrants (actual and potential) are 'routine autopsy' donors, *i.e.*, they had/have permitted USTUR to arrange an autopsy, and to have a licensed medical examiner take samples of their major internal body organs. However, a substantial number had/have permitted USTUR to study their whole body after autopsy.

The first whole body donation was made in 1979: that of a gentleman with a high internal deposition of ^{241}Am. This donation was commemorated by publication of the U.S. Transuranium Registry's report of their study results as a Special Issue of Health Physics.[25] The publication included a detailed description of the Registry's protocol for sampling the complete skeleton.[26] It also included the first systemic physiological (recycling compartment) model for americium, developed from definitive human data.[27]

All donations to the USTUR add significantly to the core scientific 'resource' of human experience from occupational intakes of actinides, and not just the whole body donations. A prime example is the 'partial body' donation by a gentleman who, in 1976, had received a facial wound heavily contaminated with ^{241}Am. This case was also commemorated in a Special Issue of Health Physics,[28] and by several follow-up USTUR studies.[29]

These two cases are also 'special' (and unique) in that the donors wished to be identified. USTUR ensures that all other Registrants (deceased or potential donors) remain anonymous. Of the 370 deceased Registrants, 335 were 'partial body' donors, 30 (8%) were 'whole body' donors, and 5 were 'special study' cases (health physics and bioassay data donations, without tissue samples). Of the 110 living Registrants, 86 are potential partial-body donors, 17 (15%) are potential whole-body donors, and 7 are 'special study' cases. The USTUR's most recent volunteer (< 35-y-old) received a substantial accidental internal deposition of plutonium while working on remediation and clean-up of a contaminated waste burial site. Since USTUR became WSU's responsibility (in 1992), 100 donations have been made, of which 16 are whole-body donations. Over this period, the annual death rate has increased from about 1% per year initially to almost 5%, as the Registrant 'population' has advanced in age.

2.1 Historical profile of USTUR donors' actinide exposures

The earliest plutonium intakes by Registries donors were in 1945. Of the Registries' whole-body donation cases, about two-thirds received their intakes before 1958 (Figure 1).

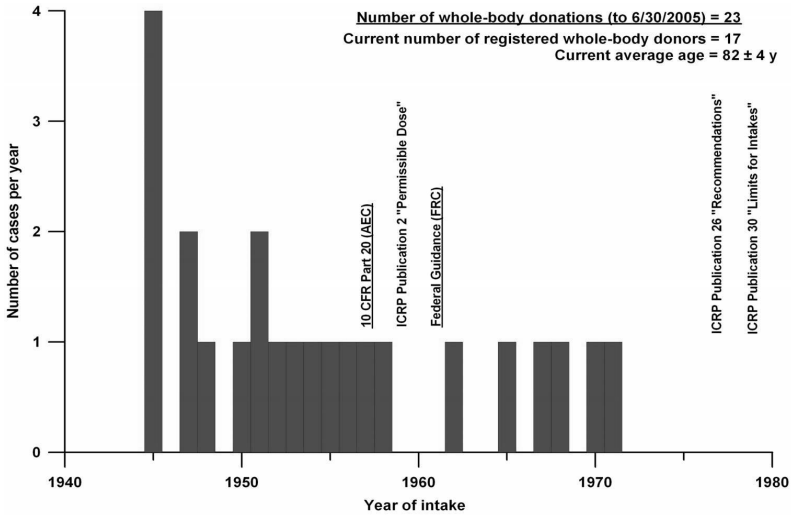

Figure 1 Year of accidental intake for first 23 whole-body donors

Over the three-decades spanning the actinide exposures of USTUR donors, the sensitivity of bioassay monitoring (i.e., limits of detection) improved substantially. Technical details of bioassay monitoring practices of the various DOE work sites are freely available,[30] and this information now enables bioassay data for individual donors to be assessed rigorously in relation to the measured actinide contents of the donor's tissues.[18-21] Table 1 lists the number of USTUR registrant volunteers

by DOE work site. In addition to these 'DOE workers,' the USTUR holds supplementary data from 11 uranium miners, 3 thorotrast cases, 51 Sellafield (UK) plutonium workers, and 9 'miscellaneous' cases.

Table 1 Numbers of USTUR Registrants by DOE work site

Site	Living	Donors	Site	Living	Donors
Mound, OH	6	6	Los Alamos, NM	11	38
Fernald, OH	1	6	Nuclear Test Site, NV	1	1
LLNL, CA	0	1	Chicago, IL	1	1
Hanford, WA	28	112	Oak Ridge, TN	2	7
Rocky Flats, CO	41	119	Savannah River, SC	11	14

3. Results and Analysis

3.1 USTUR donors' tissue burdens

The range of transuranium radionuclide body burdens measured in the USTUR donor population spans almost four orders of magnitude. Figure 2 compares the plutonium in liver concentrations measured in 106 USTUR donors with those from the Russian Federation's Dosimetry Registry of the Mayak Industrial Association (DRMIA).[31] For the respective subsets of cases compared here, the median liver concentration in USTUR donors is approximately 1/200 of that for DRMIA donors, although the ranges of concentration overlap. USTUR and DRMIA are currently updating this inter-comparison to include the lungs, lymph nodes, skeleton and liver from all autopsied cases.

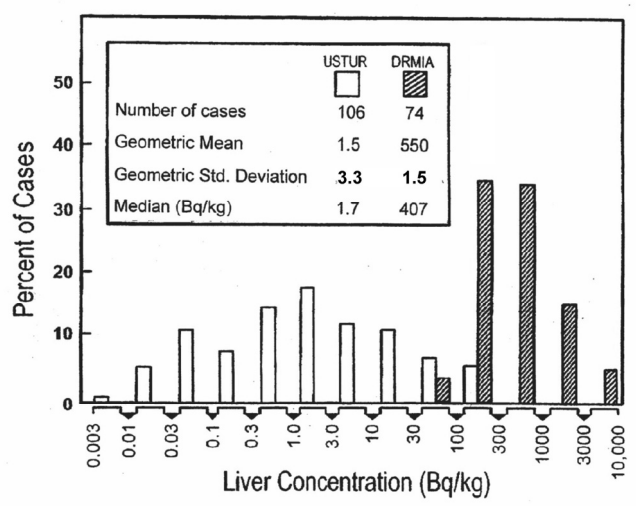

Figure 2 Comparison of Pu concentrations in liver for USTUR and DRMIA donors

3.2 Variability in Pu distribution between body organs

A key objective of USTUR's research program is to quantify the inter-personal variability of biokinetic transfer rates for plutonium and other actinides between organs of the body, and the

resulting variability of tissue doses for a given amount and type of intake. For example, USTUR's results for the ratio of liver:lung Pu concentrations measured at autopsy for 102 Rocky Flats cases demonstrate that this is highly variable, even for a single DOE site (Figure 3).

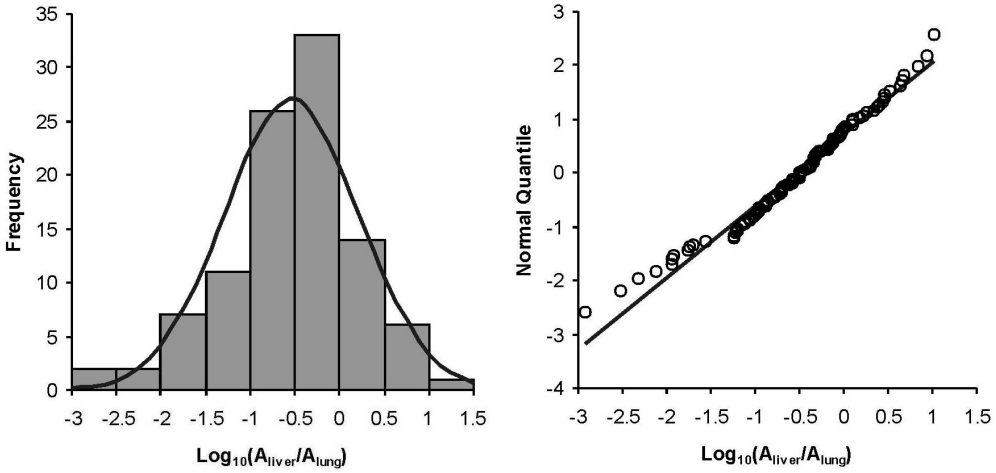

Figure 3 Frequency and normal quartile distributions of $\log_{10}$ liver:lung activity ratio in 102 cases

The observed distribution is approximately log-normal, with a large geometric standard deviation (σ_g) of 5.6. The median value (0.29:1) is significantly lower than expected. For inhalation of ICRP's 'default' Type 'S' material (assumed to represent 'insoluble' forms of plutonium) the ratio would be approximately 1.6:1, while for inhalation of 'soluble' (Type 'M') plutonium, or intake via a skin wound, the liver:lung activity ratio would be orders of magnitude higher. The observed high degree of variability in this tissue activity ratio arises from two discrete components: (i) variability in the physical characteristics/absorption behavior of the Pu material itself, and; (ii) variability in biokinetic behavior of Pu between individual persons. USTUR's objective is to quantify the respective contributions of these components to the observed variability in tissue distribution, by detailed assessment of individual donor cases.

3.3 Quantifying Pu biokinetics – ICRP model framework

ICRP's current biokinetic model for Pu was introduced in Publication 67.[32] This was designed to represent realistically both the internal (systemic) transfer of Pu to and from the blood and organs of retention and its elimination in urine and feces over time.[12] Likewise, ICRP's current lung model (Human Respiratory Tract Model, HRTM, of Publication 66)[33] was designed to represent realistically the competitive nature of uptake to the blood (*via* particle dissolution) and elimination of intact particles to the content of the gastro-intestinal tract and feces. Figure 4 shows how both models are combined to determine organ dose rates, urinary and fecal excretion rates over time, and the resulting committed organ doses: for a given amount of inhaled activity with given aerosol characteristics (activity median aerodynamic diameter, AMAD, and material 'absorption' rates).[34,35] In the case of an intake *via* a wound, an appropriate compartmental representation of the retention of material at the wound site and its translocation to any associated lymph nodes is substituted for the HRTM.

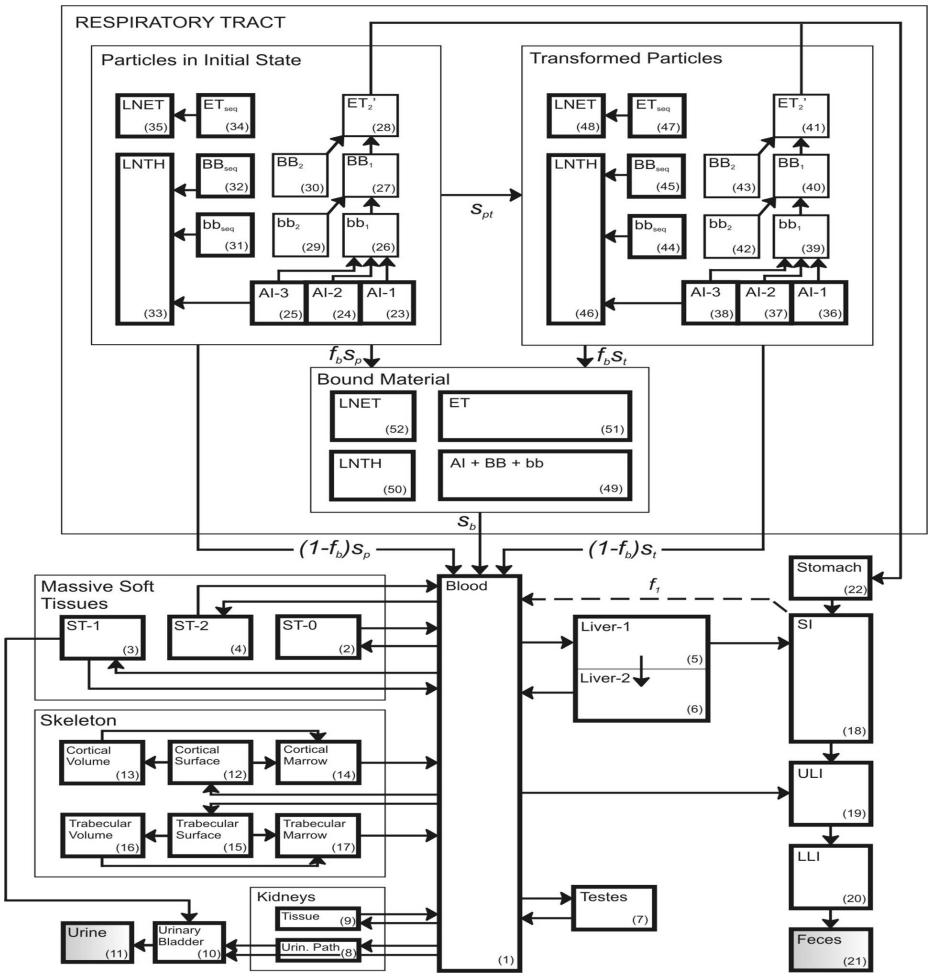

Figure 4 Combination of IC66 respiratory tract model (HRTM) and IC67 systemic Pu model

3.4 Characterizing intakes and Pu systemic biokinetics in individual USTUR cases

The amount and quality of biokinetic information that can be obtained from each USTUR case is determined by the quantity of bioassay data available. Usually, whole-body donations are accompanied by more extensive bioassay data than partial body donations, but not always so. Detailed information on the Pu distribution between body organs (and cortical and trabecular bone) available from whole-body donations, combined with sufficient bioassay data, can enable the values of key systemic transfer rates to be determined for that individual. Table 2 summarizes USTUR's analyses of two whole-body cases.

Case 0259 had an accidental acute inhalation of $^{238}PuO_2$ ceramic particles (in 1971, at Los Alamos), with 17 y of bioassay monitoring (^{238}Pu in urine). He died 17.9 y after the intake, at the age of 54 y, from atherosclerotic cardiovascular disease. Along with six other workers accidentally exposed at the same time, this donor excreted little or no ^{238}Pu in his urine for several months; a previously unknown pattern of Pu excretion. USTUR analyzed this unusual absorption behavior in terms of initial deposition of particles in a highly insoluble form, followed by fragmentation of these particles into moderately soluble (transformed) material, utilizing the HRTM (Figure 4) to determine the 'particle transformation' rate.[18)]

Case 0262 was involved in two suspected $^{239+240}PuO_2$ inhalation intakes (at Hanford in 1956), each indicated by a measurable Pu α-activity in a single urine sample, followed about 1½ y later by a puncture wound to the thumb while working in a Pu glovebox. Urine samples taken after the wound incident had readily measurable Pu α-activity over the next 14 y, before dropping below the minimum detectable excretion rate. The donor died about 33 y after the wound intake, at age 71 y, from hepatocellular carcinoma with extensive metastases. In this case, simultaneous analysis of the Pu-in-urine data and the measured tissue contents at death enabled: (i) the amounts (and absorption characteristics) of both Pu inhalation intakes, and; (ii) the amount of Pu deposited at the wound site, the amount translocated to the associated lymph node, and the subsequent (multi-phased) absorption of Pu into the blood to be determined.[19)]

Table 2 Pu biokinetic behavior quantified for two USTUR cases

Transfer Pathway	Transfer Rate, d^{-1}		
	IC67 Reference	Case-specific Factor	
		Case #0259	Case #0262
Respiratory tract:			
AI_3 to bb_1	0.0001	× 1.00	× 0.918
AI_3 to LNTH	0.00002	× 1.57	× 0.526
Systemic Pu model:			
Blood to Cortical bone surface	0.3235 × 0.4	× 0.515	× 0.444
Cortical bone volume to Marrow	0.0000821	× 0.55	× 0.53
Blood to Trabecular bone surface	0.3235 × 0.6	× 1.1253	× 1.133
Trabecular bone surface to Volume	0.000247	× 1.40	× 1.40
Trabecular bone surface to Marrow	0.000493	× 1.00	× 1.00
Trabecular bone volume to Marrow	0.000493	× 0.64	× 0.35
Trabecular marrow to Blood	0.0076	× 0.605	× 0.605
Blood to Liver 1	0.1941	× 1.61	× 0.928
Liver 2 to Blood	0.000211	× 0.92	× 0.90
Blood to Other kidney tissue	0.00323	× 1.255	× 0.827
Other kidney tissue to Blood	0.00139	× 0.97	× 1.00
Blood to Urinary path	0.00647	× 1.39	× 0.90
Blood to Urinary bladder content	0.0129	× 1.39	× 0.90
Blood to ST-2	0.0129	× 0.87	× 1.84
ST-2 to Blood	0.000019	× 1.00	× 1.00
Blood to Testes	0.00023	× 0.85	× 0.69
Testes to Blood	0.00019	× 1.00	× 1.00

Table 3 compares the measured tissue contents for Case 0262 (multiple inhalations and skin wound) with those 'predicted' from the urine bioassay data using the IMBA Professional Plus software (as recently adopted by DOE for regulatory intake and dose assessments).[22,23)] This software implements the HRTM, together with the ICRP Publication 67 Pu systemic model ('IC67 Reference' parameter values) and a 'generic' multi-exponential-compartment wound absorption model. In this

case, the intake assessment process was constrained to 'fit' the measured lung and whole body contents exactly. As expected, the measured contents of this individual's tissues differ from those predicted for the ICRP 'reference worker.' The measured tissue contents were fitted exactly when the 'case-specific' modifying factors listed in Table 2 were applied to the 'reference' rate constants in the IC67 Pu systemic model.

Table 3 Pu tissue contents predicted using the IC67 systemic model compared with measured values

Tissue	Case 0262 Content of $^{239/240}$Pu, Bq		
	Measured	IC67 Pu Model	Error, %
Wound	68.0	68.0	0
Axillary lymph node	56.0	56.0	0
Lung	2.59	2.59	0
Thoracic lymph nodes	1.05	1.05	0
Skeleton	29.1	33.2	+14
Trabecular bone	17.6	9.2	-48
Cortical bone	11.5	24.0	+109
Red bone marrow	-	0.82	-
Liver	20.7	20.0	-3
Massive soft tissues	8.6	5.3	-38
Testes	0.018	0.025	+39
Kidneys	0.053	0.061	+15

Figure 5 compares the urinary bioassay data for this case with the resulting 'best fits' obtained for ICRP's 'reference' systemic transfer rates and the modified ('case-specific') transfer rates listed in Table 2. It is seen that 'optimization' of the Pu systemic transfer rates (to give exact fits to the measured tissue contents) has a relatively small overall effect on the calculated urinary excretion of Pu. The minimum X^2-sum is reduced from 45.1 (IC67 reference rates) to 42.7 (optimized systemic transfer rates). Thus, the currently recommended ICRP Publication 67 Pu systemic model parameters suffice for the purpose of using the bioassay data to characterize the intakes in this case.

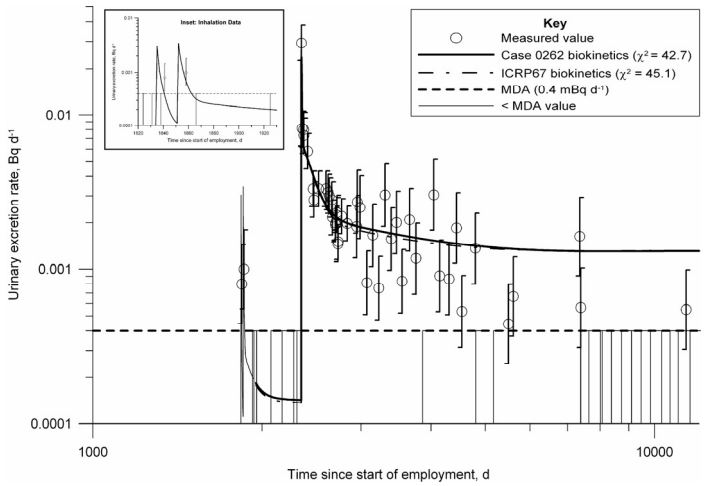

Figure 5 Observed and modeled urinary Pu excretion for Case 0262

3.5 Modeling effectiveness of chelation therapy

Whole body donation Case 0269 involved a single acute inhalation of an acidic $Pu(NO_3)_4$ solution in the form of an aerosol 'mist' (in 1956, at the Hanford site). Chelation treatment with i.v. Ca-EDTA was initiated on the day of intake, and continued intermittently over 6 months. After 2½ years with no further treatment, a course of i.v. Ca-DTPA was administered. A total of 400 $^{239+240}$Pu-in-urine measurements were made; starting on the first day and continuing for 37 years. This sampling included all intervals of chelation. The donor died 38 y after the intake, at age 79 y, with extensive carcinomatosis secondary to adenocarcinoma of the prostate gland. In this case, simultaneous analysis of urine and fecal bioassay data together with the measured tissue contents enabled USTUR to determine the 'chelation enhancement' of transfer rates in the IC67 systemic Pu model achieved by both therapeutic drugs.[19] Figures 6(a) and (b) compare the measured and 'modeled' Pu urinary excretion over the periods influenced by Ca-EDTA and Ca-DTPA therapies, respectively. Figure 6(c) compares the measured and modeled effects of the Ca-DTPA therapy on fecal Pu excretion. The Ca-EDTA therapy had no effect on fecal Pu excretion.

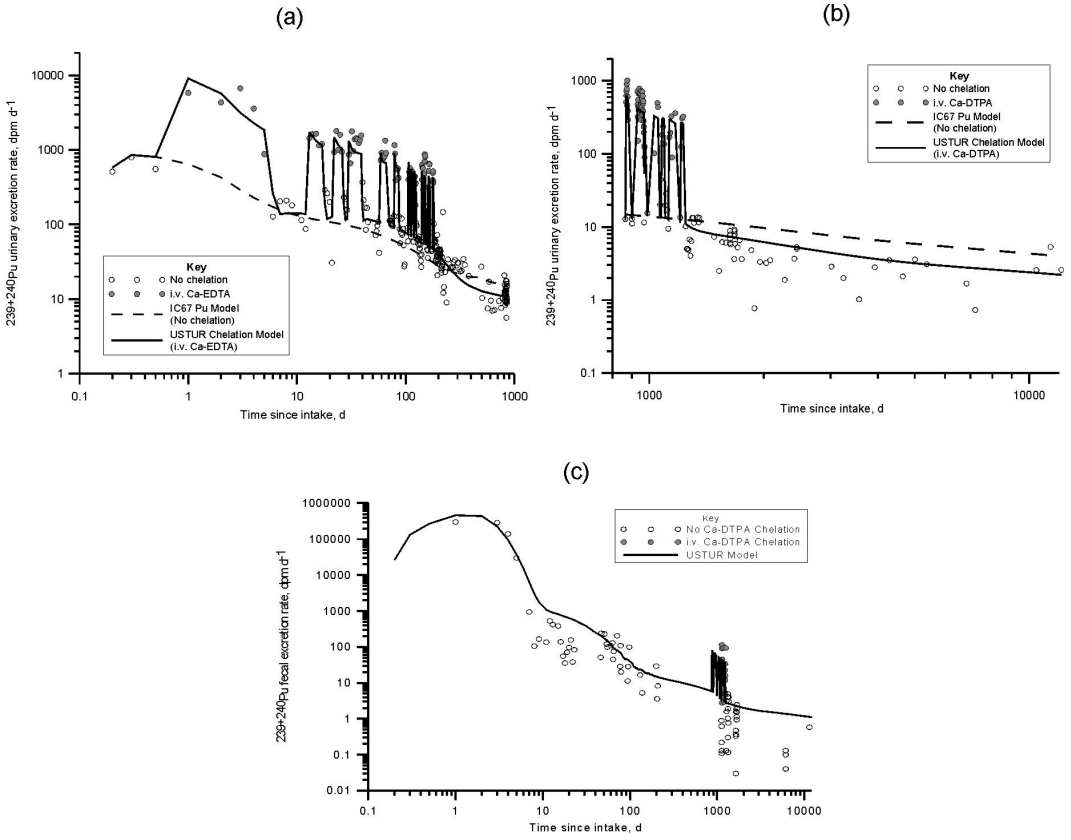

Figure 6 Measured and modeled effects of Ca-EDTA and Ca-DTPA therapies on Pu excretion

Table 4 shows the modeled effects of all therapeutic treatments in reducing tissue burdens in this individual case. As a result of the chelation treatment, the effective dose from the accidental Pu intake was reduced by about 50% (from about 10 Sv to about 5 Sv).

Table 4 Measured and modeled tissue contents resulting from chelation therapy

Tissue	$^{239+240}$Pu Content at Death, kBq			
	Autopsy	USTUR Model		
		Treated	Untreated	Saved
Whole body	2.29	2.29	4.22	46%
Lungs	0.027	0.027	0.027	0%
LNTH	0.00019	0.00021	0.00021	0%
Liver	0.94	0.81	1.62	50%
Skeleton	1.20	1.21	2.18	45%
Muscle, Skin, *etc.*	0.18	0.23	0.38	39%
Testes	0.83	0.83	1.47	44%
Kidneys	0.0017	0.0017	0.0032	47%

These results of USTUR's chelation modeling are preliminary. Further work is in progress to improve prediction of the final liver burden, the late fecal excretion, and the massive soft tissue burden measured in this case. USTUR will finalize model development using the proposed new ICRP Pu biokinetic model[13] as the 'baseline.' This revised model structure includes a more realistic treatment of the early kinetics of Pu in blood and tissue fluid than the IC67 Pu systemic model. In turn, this should improve the modeling of chelation effects.

3.6 Other studies aimed at improving 'field' monitoring and dose assessment

USTUR's first whole body donation case (referred to above, in Section 2) provided the "human half skeleton" incorporated in DOE's human ^{241}Am laboratory inter-calibration phantom (http://www.pnl.gov/phantom/). This phantom is shown in Figure 7. USTUR is now in process of developing a mathematical 'voxel' phantom, based on segmentation of a complete series of high resolution CAT-scan image slices of each part of the physical phantom (http://www.betaustur.org/voxel/index.html). The availability of an actual-human-case ^{241}Am 'virtual' phantom will enable computational simulations of external detection system response to be carried out for an unlimited range of applications, detector types and geometrical configurations.

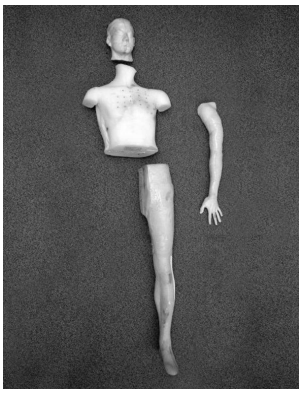

Figure 7 DOE's 'Human ^{241}Am Phantom' incorporating half of Case 0102's skeleton

For suspected substantial intakes of $^{239+240}$Pu, external counting of the ^{241}Am contaminant activity has been used routinely at several DOE sites to estimate the amount of internal $^{239+240}$Pu deposition. Accordingly, USTUR has routinely measured the tissue contents of ^{241}Am in all Pu intake cases, for comparison with the measured $^{239+240}$Pu activity. As part of the current 5-y research program, USTUR plans to extend the Pu biokinetic modeling of whole-body cases to include evaluation of how well ICRP's recommended Am systemic model[32] represents measured tissue contents of 'in-grown' ^{241}Am.

3.7 Web publication of USTUR's case data

As described earlier, for several decades USTUR has published summary results of radiochemical tissue analyses for many Registrant cases in both progress reports and the open literature. However, these tissue analysis results have limited application, unless they can be related to complementary bioassay data and other worksite information. To overcome this limitation, and also to promote timely and wide dissemination of 'de-identified' case data, USTUR is currently developing a comprehensive new website (http://www.betaustur.org/). Early in 2007, this new site will replace USTUR's current http://www.ustur.wsu.edu/ site, which does not publish case data. Figure 8 shows the new site's layout.

Figure 8 New USTUR website to include searchable 'de-identified' case data

In addition to providing background information on the USTUR's research program (as provided by the current site), the new web site will 'index' all (de-identified) case data; by primary radionuclide, type of intake, type of material, work site, length of follow-up, and pathology findings. The tissue analysis data for each case will be linked to the case bioassay data, and any other (summarized and de-identified) supporting health physics data.

4. Conclusion

This paper has outlined the continuous evolution of the USTUR's research program from the primarily 'data collection' and publication activities of the 1968 National Plutonium Registry, through its contributions to the development of ICRP's currently recommended biokinetic models for plutonium, americium and uranium, to its current focus on applying USTUR results to validating practical field methodologies for intake and internal dose assessment; and also contributing to the development of future, more realistic models. A key objective of USTUR research is to quantify the variability of actinide biokinetics between individual workers, and the overall effect of this variability on tissue doses for defined intake conditions. A further key objective is to make USTUR's indexed and coordinated research data readily available for study and application by our international scientific peers.

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2-6 Retrospective Dosimetry of an Accidental Intake Case of Radioruthenium-106 at the Tokai Reprocessing Plant

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Abstract

On November 30, 1978, two workers in the acid recovery cell of the nuclear spent fuel reprocessing plant of JAEA-NFCEL were involved in an accident in which they became unconscious due to lack of oxygen. They were rescued immediately by other workers and were given artificial respiration to restore their normal breathing. Subsequent measurements by the whole-body counter showed that they were contaminated internally with ^{106}Ru. Prolonged lung monitoring was carried out for one of them. A significantly high activity of ^{106}Ru was obtained in the lung monitoring on the day of the accident. The physicochemical characteristics of the incorporated radioactive materials were not observed. In order to perform more reasonable internal dose assessment, the interpretation of the bioassay datasets of the worker was made based on the guideline demonstrated in the EU project IDEAS. The effective half-life of the materials in the lungs was determined to be 140 days which leads to the default Type S absorption type in the HRTM and the f_1 value was estimated to be less than 0.005 which is one-tenth of the default value. Simultaneous intakes via inhalation and ingestion were also suggested from several pieces of evidence although pure inhalation was assumed for internal dose assessment at the time of the accident. The aerosol size of the materials was not determined due to a lack of information if assuming simultaneous intakes; however, the resulting committed effective dose was about 1 mSv and its variation was small against the aerosol size ranging from 1 μm to 20 μm.

Keywords: Reprocessing plant, Radioruthenium-106, Inhalation, Ingestion, Simultaneous intakes, Guideline, IDEAS, Bioassay, Retrospective

1. Introduction

With the aim of establishing a nuclear fuel cycle in Japan, a demonstration spent nuclear fuel reprocessing plant (TRP) was developed at Japan Atomic Energy Agency, Nuclear Fuel Cycle Engineering Laboratories in Tokai-mura (JAEA-NFCEL, former Japan Nuclear Cycle Development Institute, Tokai Works) and its active testing was started in 1977. The TRP uses the PUREX method to reprocess spent nuclear fuels that is used at PWRs, BWRs and an advanced thermal reactor (ATR, namely FUGEN, a heavy water moderated, boiling light water cooled, pressure tube type prototype

reactor) and has the capability of reprocessing 0.7 tons of U per day or annually 210 tons of U[1)]. The accumulated amount of reprocessed fuels reached 1123 tons in October 2006.

The TRP began comprehensive its test operations on August 1, 1978. On August 24, a process monitor gave an alarm due to a leakage of radioactive materials. The cause was pinholes in the heating tube plate inside the acid recovery evaporator. The evaporator is used to recover nitric acid from radioactive nitrate solution exhausted from other processes: the separation process of plutonium/uranium and fission products, the purification process, and the concentration process of highly radioactive liquid waste. On November 30 in the same year, an accident involving two workers becoming lack of oxygen occurred at the cell where the evaporator was installed. A worker (Worker A) entered the cell in order to remove dosimeters on devices for a preparatory survey of ambient dose rate, when he collapsed on the floor and lost consciousness. Another worker (Worker B) who attempted to rescue Worker A also was overcome with lack of oxygen. The concentration of oxygen near the floor was estimated as from 10 % to 15% by volume based on the effect to the workers. A cut-away view of the cell is illustrated in Figure 1.

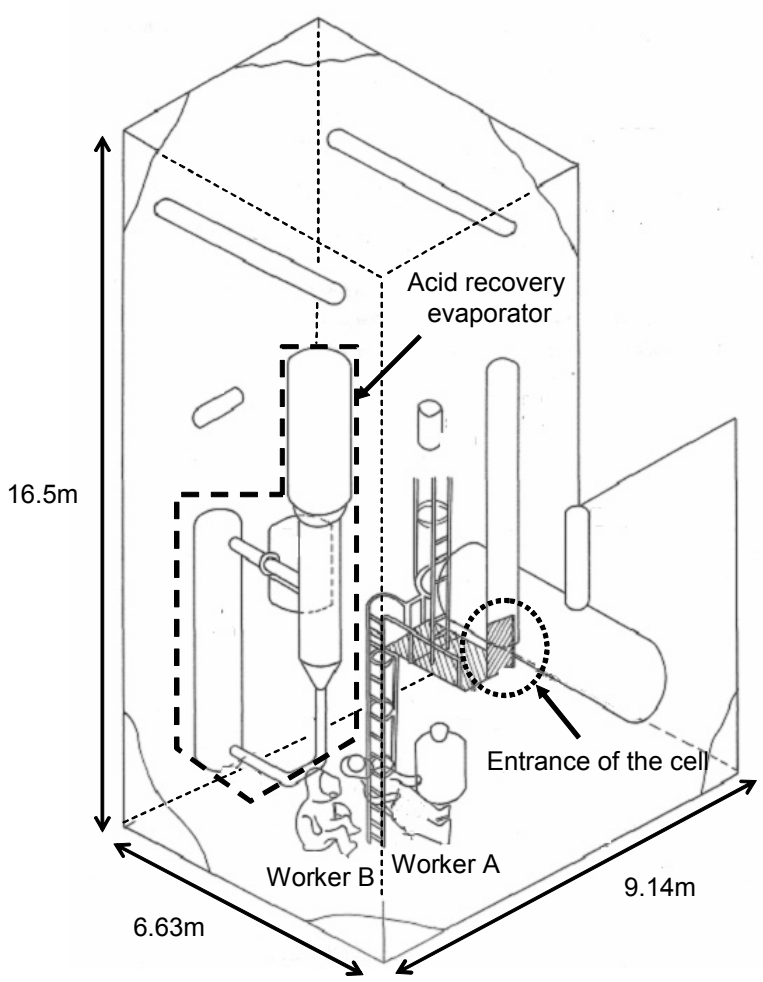

Figure 1 Cut-away view of the cell and positions of the workers involved.

The lack of oxygen was caused by excess frozen carbon dioxide present in the cell. In previous operations, frozen carbon dioxide ("dry ice") had been used to removing blockage in a pipe. However, proper preparation for lack of oxygen had not been done due to overestimation of ventilation rate near the floor of the cell.

Pressurized air was inserted into the cell to prevent other workers from experiencing lack of oxygen and the two overcome workers were rescued by them using a piece of rope. Workers A and B were taken to vinyl houses outside the cell and immediately given artificial respiration which revived them. The two workers were contaminated externally –mainly on their face– during the rescue operation and then were sent to a radiological health service facility outside the TRP for decontamination and medical measures. Their skin contamination was successfully removed up to the background level by a few decontamination procedures including showering.

According to documents concerned with the accident, it was reported that Worker B removed the respiratory protection device (a full face mask) of Worker A for a short time because he first suspected unsuitable fitting of Worker A's mask. Worker B did not wear his mask tightly as he was responding to the situation quickly. It was also described in the documents that artificial respiration was carried out before removal of their external contamination.

Lung monitoring using a whole-body counter was done for Workers A and B and for other workers whose nose swabs were significant. Consequently, internal contamination with ^{106}Ru was found for only Workers A and B. Excreta analysis was also conducted for them. Prolonged lung monitoring was carried out for Worker A. As for Worker B, a significant lung activity of ^{106}Ru was found only on the day of the accident.

Following questions were left about the accident from the viewpoint of internal dosimetry and they have not been discussed so far. (1) What is the actual mode of intake of radioactive materials in unusual situation even though pure inhalation was assumed at the time? (2) Can we perform the reasonable interpretation of the bioassay datasets using the current internal dosimetric models proposed by the International Commission on Radiological Protection (ICRP) although specific physicochemical characteristics of the materials incorporated were not clearly obtained? This study is aimed at obtaining answers to these questions and at summarizing the lessons learned in order to improve internal dosimetry services.

2. Radiation Monitoring Information

2.1 Individual monitoring

The results of individual radiation monitoring of Worker A were described as follows. The beta and gamma activity of a pair of nose swabs and his sputum sample are determined to be 700 Bq and about 3700 Bq, respectively. Skin contamination was confirmed by direct-survey using a GM survey meter, especially on his face with values of: 2000 cpm on the top of the head, 10000 cpm on the nose, 1500 cpm – 2000 cpm on the cheeks. His work clothes were also contaminated extensively. It was likely that Worker A was contaminated externally when he collapsed on the floor in the cell. Lung monitoring was initiated in order to confirm internal contamination and ^{106}Ru (^{106}Rh) was found. Small amounts of radionuclides other than ^{106}Ru were found in his fecal samples, i.e. ^{125}Sb, ^{137}Cs and ^{60}Co.

No significant values were observed in any urinary samples.

The residual radioactivity in the lungs (hereinafter referred to as the lung activity) was evaluated by the bed geometry whole-body counter installed in the shielding chamber whose walls were 20-cm thick iron plates. A NaI(Tl) scintillation detector (12.7 cm diameter and 10.2 cm thickness) was equipped with the whole-body counter and was placed over the bed on which the subject reclined as shown in Figure 2. The detector was able to scan along the body axis and was fixed above the chest of the subject when evaluating lung activity. A collimator and a shield made from lead were attached on the detector as necessary. The photons emitted from ^{106}Rh (622 keV) were measured by the whole-body counter. Excreta analysis was carried out by gamma-spectrometry using a Ge(Li) semiconductor detector.

The bioassay datasets of lung activity and daily fecal excretion for Worker A are shown in Tables 1 and 2, respectively. Day 1 in the tables was the next day after the accident (i.e. Dec 1, 1978). The daily urinary excretion was reported as below the lower limit of detection (<LLD) for all samples taken for a week post intake. The count rate profiles were measured by the whole-body counter in addition to the lung monitoring and these are shown in Figure 3.

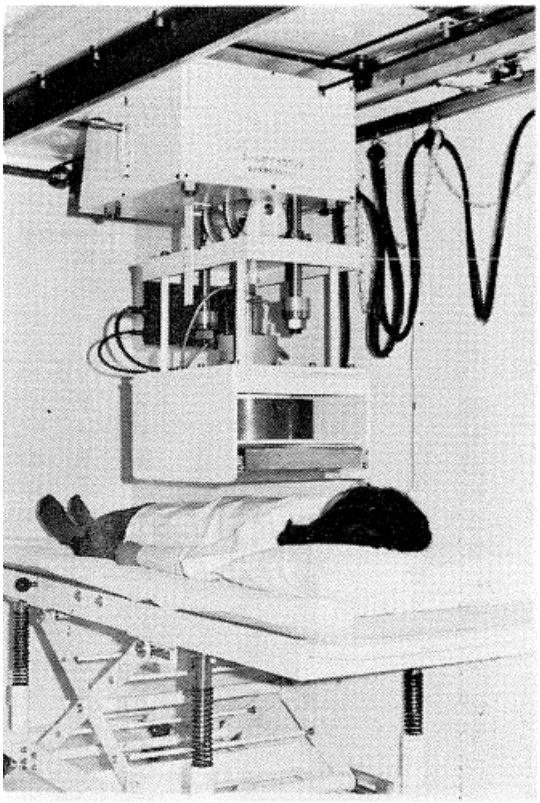

Figure 2 External view of the whole-body counter used.

Table 1 Lung activity of ^{106}Ru for Worker A measured by the whole-body counter.

Day	Activity（Bq）	Day	Activity（Bq）
0.2	6.7E+04*	118	1.0E+03
1	3.7E+03	146	9.6E+02
2	2.4E+03	181	7.4E+02
3	2.0E+03	209	6.3E+02
5	1.7E+03	235	4.8E+02
9	1.6E+03	272	4.4E+02
15	1.7E+03	300	4.4E+02
27	1.5E+03	335	<LLD
62	1.5E+03	364	<LLD
92	1.1E+03	—	—

The lower limit of detection: 3.7E+02 Bq.

* The collimator and shield of the detector equipped with the whole-body counter were removed at the first measurement and were attached at the other measurements.

Table 2 Fecal excretion of ^{106}Ru for Worker A.

Day	Sample weight (g)	Activity（Bq）
1	145	4.4E+04
2	43	1.2E+04
3	93	4.1E+03
4	225	2.7E+02
5	119	3.7E+01
6	51	<LLD

The lower limit of detection: 3.7E+01 Bq. The LLD of urinary analysis is the same as this value.

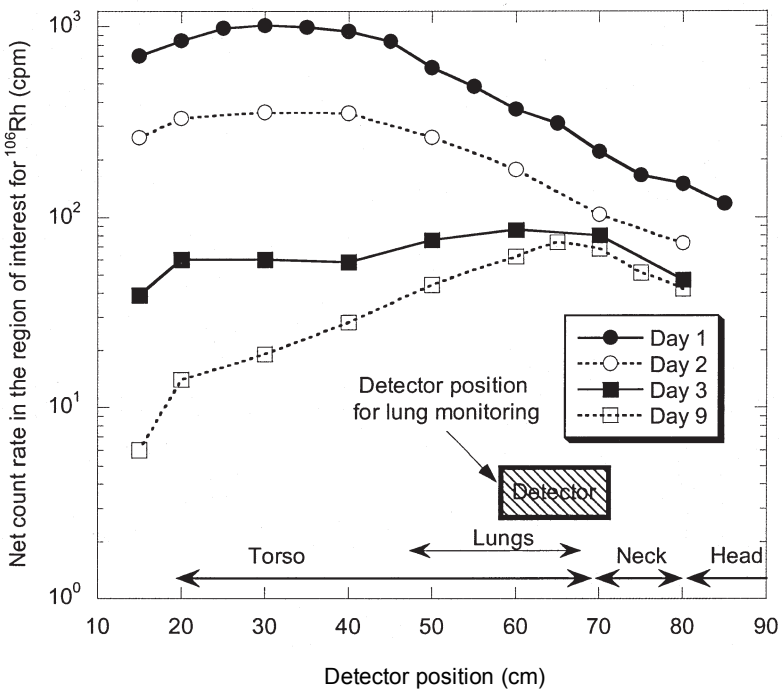

Figure 3 Count rate profile along the body axis of Worker A by the whole-body counter. Day 1 in the figure was the next day after the accident (i.e., Dec.1, 1978).

2.2 Workplace monitoring

The preliminary investigation for the broken evaporator prior to the accident showed that the surface contamination on the floor in the cell was distributed widely and ranged from 3700 Bq/cm^2 to 37000 Bq/cm^2 ($\beta\gamma$). As a decontamination procedure, the floor was filled with water several times and the water was discharged on each time. The decontamination work decreased the surface contamination level on the floor only by a factor of ten and then the floor was covered with sheets of vinyl.

The airborne concentrations of radioactive materials in the cell and the vinyl houses outside the cell were monitored with three dust monitors and a particulate sniffer. They showed a temporary increase at the time of the accident. The radionuclides in the air were found to be ^{106}Ru, ^{137}Cs and ^{125}Sb, predominantly ^{106}Ru.

3. Materials and Methods

The guideline and the related information demonstrated in the EU project IDEAS were used for the interpretation of bioassay data for internal dose assessment[2,3]. A virtual workshop of an intercomparison exercise of internal dose assessment from monitoring data on the Internet was held in collaboration with the IAEA in the project[4]. The usefulness of the guideline was confirmed and further improvements as discussed in the workshop were incorporated. One of the principles of the guideline is “accuracy”, in which the best estimate in intake activity and committed effective dose should be obtained from available data.

The IMBA code was mainly used in the analysis[5]. The IMBA code is capable of calculating the best estimate of intake by fitting between the observed and the predicted data with specific parameter values used in the ICRP dosimetric models if required. It is normally assumed that each measurement is taken from a normal or a lognormal distribution. The maximum likelihood method is recommended for evaluating the best estimate of the intake activity in the guideline. These treatments including simultaneous fits for multiple datasets of different bioassay can be implemented in the IMBA code. As for criteria for reject fits, the null hypothesis in classical statistics using the chi-square test was applied in this study. The test statistic, χ_0^2 is given by the following formula in the case of each monitoring, m_i, being in a lognormal distribution.

$$\chi_0^2 = \sum_{i=1}^{n} \left(\frac{\ln(m_i) - \ln(If(t_i)}{\ln(SF_i)} \right)^2$$

where I is the estimated intake and $f(t_i)$ is the predicted bioassay quantity for unit intake activity. Then the product $I f(t_i)$ is the predicted value. SF_i is a scattering factor, which means the total uncertainty of monitoring in terms of the geometric standard deviation when monitoring data are in a lognormal distribution. The above formula does not apply to data that are reported as below the lower limit of the detection (<LLD). If the predictions are inconsistent with the observed data, then the calculated value of χ_0^2 is inconsistent with the theoretical chi-square (χ^2) distribution with (n-1) degrees of freedom. The probability of observing a larger χ^2-value than χ_0^2 for (n-1) degrees of freedom is denoted by the p-value, which can be obtained from statistical tables. If the p-value is small (e.g., ≤ 5%), then fit to the data is judged to be inadequate.

The in-house code, REIDAC (Retrospective Internal Dose Assessment Code) was supplementarily used for calculation of fractional depositions of “Mouth breather” in the human respiratory tract model (HRTM)[6] and intake retention function in organs not being given by the IMBA code. The REIDAC is capable of internal dose calculation in accordance with the ICRP dosimetric models and has been verified for dose per unit intake (DPUI) of 20 radionuclides including ^{106}Ru by comparison with the ICRP CD-ROM[7] and the retention and excretion function of ^{106}Ru by comparison with the IMBA code. A sophisticated fit function is not available (only the least square fit) but comprehensive parameter values can be modified on the graphical user interface (GUI) in the REIDAC.

4. Results and Discussion

4.1 Route of intake of the radioactive materials in Worker A

Assuming pure inhalation, the effective aerosol size can be determined by the ratio of fecal excretion to lung activity in the case of insoluble or moderately insoluble materials[2,8]. For Worker A, the ratio of total fecal excretion for five successive days post intake to the lung activity on the fifth day was determined to be 36, which led to an effective aerosol size of about 20 μm in aerodynamic mean activity diameter (AMAD) by comparing the ratio calculated with different aerosol sizes for several types of breathing habits as shown in Figure 4. The evaluated aerosol size is relatively large compared to aerosol size of radioactive compounds at typical workplaces in nuclear-related facilities[9] and also

compared to measurements at JAEA workplaces[10]. This evaluated large-sized aerosol suggests a possibility of simultaneous intakes via inhalation and ingestion. That can be simply explained by the fact following. The larger the aerosol size is, the greater the fraction of deposition in the extrathoracic (ET) region is. The ET region is composed of the anterior nose (ET_1) and the posterior nasal passage, larynx and mouth (ET_2) in the HRTM. The residual activity in the latter is cleared rapidly to the stomach due to particle transport. Therefore, the initial distribution of a radionuclide in the body is very similar between pure inhalation with the large-sized aerosol and ingestion. The possibility of simultaneous intakes is also supported by the following two pieces of evidence. The first is the fact that Worker A was contaminated externally on his face and then he received artificial respiration and decontamination. The second is that the ratio of the early fecal excretion to the activity found in nose swabs −86 in this case− was relatively higher than that obtained from reported inhalation cases of insoluble plutonium: the geometric mean of the ratio for insoluble plutonium is about 2 ($\sigma_g = 7$)[11]. In short, the deposition of the aerosols on the nasal cavity was relatively small compared to normal inhalation cases. In the following discussion, simultaneous intakes are also considered as a route of intake in addition to pure inhalation.

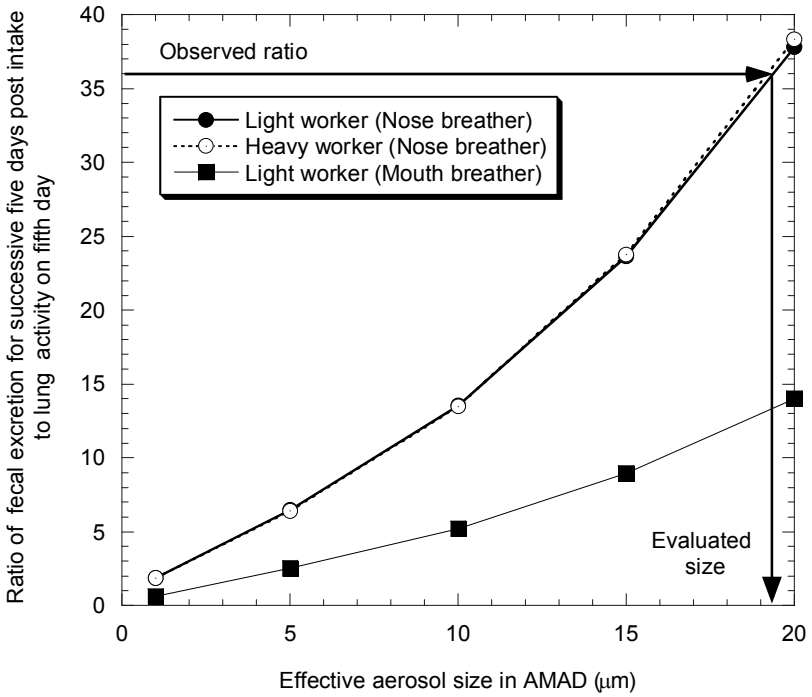

Figure 4 Calculated ratio of total fecal excretion for five days post intake to the lung activity on the fifth day post intake. All calculations in the figure are Type S absorption in the HRTM. As for the light worker, the heavy worker and the mouth breather, see ICRP Publication 66[6].

4.2 Optimization of parameter values in the dosimetric models

A significantly high value for lung activity was found in monitoring on the day of the accident. As proved later, this is caused by overlapping lung activity with residual activity in the gastro-intestine (GI) tract because of the geometry of the whole-body counter in addition to the fact that the detector

was not collimated in the first measurement. As a large part of the residual activity in the GI tract is excreted in feces within a few days post inhalation (early feces), the lung monitoring data within a first few days post intake should be treated as rogue data until the residual activity in the GI tract becomes negligible. The lung activity on the fifth day was treated as the long-term component for the lung as the critical organ in the dose assessment at the time of the accident; the evaluated committed dose for 50 years was 0.7 rem (7 mSv) for the lungs as the critical organ according to the previous ICRP concept.

Both aerosol size and absorption type in the respiratory tract are important factors in internal dose assessment in case of inhalation as a route of intake but neither of them was observed. In terms of aerosol size of the materials, it is difficult to determine that from the monitoring data available in the case of simultaneous intakes. As for absorption type, ICRP assigns oxide and hydroxide forms of radioruthenium with Type S and halides with Type M in the HRTM. Most of the ruthenium in the PUREX waste solution is known to be as nitrosyl compounds such as RuNO-nitrato complex, RuNO-nitro complex, RuNO ion and so on[12]. However, ICRP dose not assign any absorption types to nitrosyl compounds of radioruhenium[13]. It is also known that the behavior of ruthenium in the reprocessing process is very complex and ruthenium tetroxide (RuO_4) vapor is produced under highly oxidizing conditions such as used for dissolution of nuclear spent fuels and nitric acid recovery in the reprocessing process[14]. Webber and Harvey[15] reported in accidental intake case by a human subject that RuO_4 vapor was mainly deposited in the upper region of the respiratory tract before reaching the lungs and stayed there for a certain period. No similarity in behavior of radioruthenium in the body was found in the present case.

A single exponential function was fitted well with the data of lung activity excluding the first four data and the effective half-life was determined to be 140 days (the biological half-life is 220 days). The effective half-life evaluated was shorter than that reported in the actual case of ruthenium oxide (^{106}Ru): an average of 310 days for the subjects involved[16]. Consequently, Type S is the most suitable of default absorption types in the HRTM. As shown below, reasonable fit between the observed and the predicted lung activities with Type S was obtained without using specific absorption parameters in the HRTM.

Almost all of the total fecal excretion was found in the first two days. This trend was earlier than that calculated with the default transit times in the compartments in the GI tract model[17]. A similar pattern of fecal excretion was found in another study on metabolism of ruthenium in oral administration for man[18]. It is appropriate that the accelerated fecal excretion is interpreted as individual metabolic characteristics. The predicted daily fecal excretion in the case of default parameter values of the transit times of upper and lower large intestine (ULI and LLI) being decreased by a factor of 3 agreed well with that observed. This modification affects internal dose assessed very slightly.

The fractional absorption from the GI tract to the transfer compartment, the f_1 value affects urinary excretion. ICRP assigns the same value (0.05) for all chemical forms. Figure 5 shows the predicted daily urinary excretions with different f_1 values for the case of pure inhalation of the compounds with Type S absorption. The f_1 value was estimated to be at least less than one tenth of the

default value as shown in the figure since all urinary samples taken for a week post intake were less than the detection limit of the analysis. In general, f_1 values have been determined from data for ingestion of food. It is suggested that the default f_1 values applied for inhaled material passing through the GI tract are too high in some cases, especially for relatively insoluble materials[19]. The absorption parameter in the HRTM also affects urinary excretion. However, the difference in urinary excretion is very small in early stage post inhalation between compounds with Type S absorption and those with Type M absorption. Therefore, modification of the f_1 value rather than the absorption parameters in the HRTM is reasonable in this case in order to explain no significant detection in the urinary analysis.

The predicted values of lung activity and fecal excretion on the condition of pure inhalation and the modified parameter values above were shown in Figures 6 and 7 with the observed values for comparison. In the calculation, the bioassay data were assumed to be a lognormal distribution and the SFs of each dataset were set at 1.2 for the lung activity and at 3 for the daily fecal excretion, which are provided by the resources in the guideline. As shown in these figures, good agreement between the predicted and the observed values was confirmed. Good agreement was also obtained in runs of simultaneous intakes while varying the aerosol size within the range of realistic values. The best estimates of intake activity and resulting effective dose for each run are shown in Table 3. The aerosol size affected the resulting effective dose slightly even though activities of intakes via inhalation and ingestion changed widely.

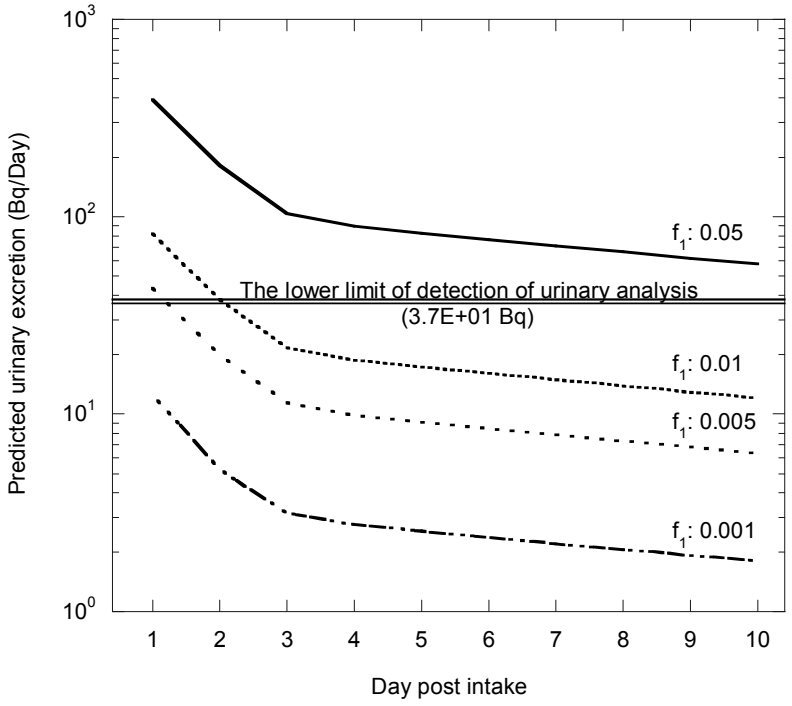

Figure 5 Predicted daily urinary excretion with different f_1 values in the case of pure inhalation of the compounds with Type S absorption.

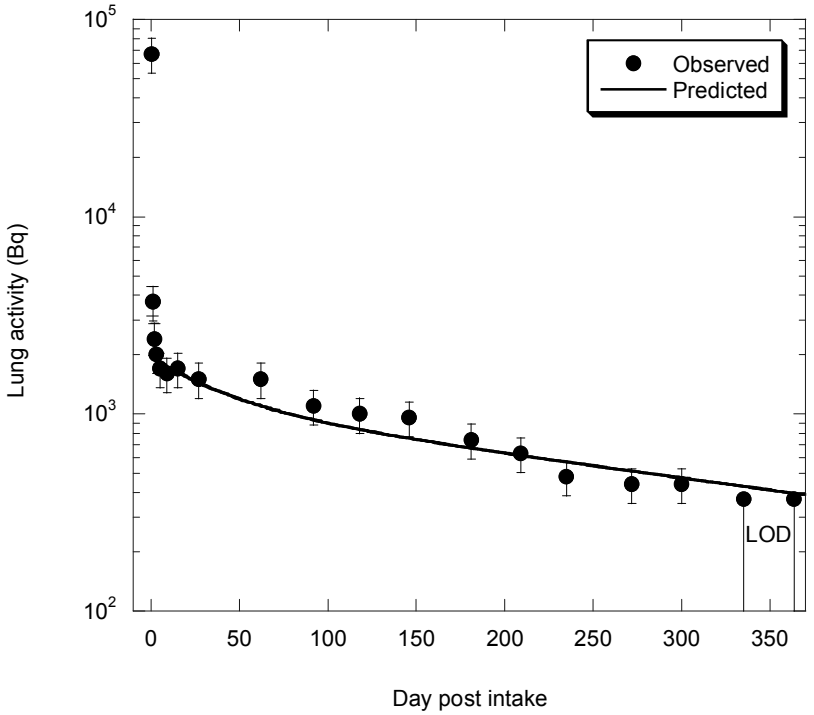

Figure 6 Comparison between the observed and the predicted lung activity in the case of pure inhalation of the compounds with Type S absorption. The first four observed data is not used for fit.

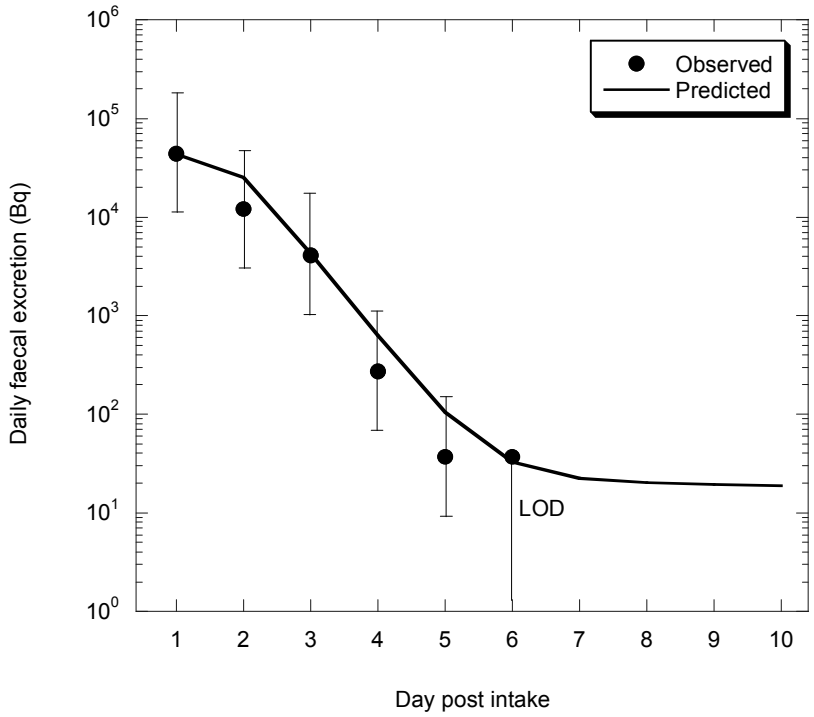

Figure 7 Comparison between the observed and the predicted daily fecal excretion in the case of pure inhalation of the compounds with Type S absorption.

Table 3 Results of the best estimate of intake activity and resulting effective dose with varying aerosol size in pure inhalation and simultaneous intakes.

	RUN 1	RUN 2	RUN 3	RUN 4	RUN 5
AMAD (μm)	20	15	10	5	1
Modified parameters	Transit times of ULI, LLI: decreased by a factor of 3, f_1= 0.005				
Intake activity via inhalation (Bq)	2.18E+05	1.27E+05	6.68E+04	3.00E+04	1.52E+04
Intake activity via ingestion (Bq)	–	–	1.63E+04	3.06E+04	4.00E+04
Effective dose (mSv)	1.19	1.10	1.09	1.08	1.05
χ_0^2-total (P-value)*	11.0 (0.856)	9.82 (0.911)	9.70 (0.916)	9.55 (0.922)	9.40 (0.927)

* Degree of freedom = 17 (13 lung data + 5 fecal data – 1).

4.3 Rogue data of lung monitoring

It seems to be important for internal dosimetrists to examine the significantly high activity in the lung monitoring on the day of the accident. As shown in Figure 3, the count rate profile demonstrated broad activity distribution in the body, and the peak shifted from the lower part of the abdomen on Day 1 to the chest part on Day 9. The first peak was due to the residual activity in the GI tract and the second peak was due to the residual activity in the lungs. The peaks in the count rate profile agreed well with the Gaussian distribution function as shown in Figure 8 and the peak width was almost the same for both peaks. Assuming that the counting efficiencies of the whole-body counter were not largely different in the lungs and the GI tract, the ratio of the integrals of the peaks over its width were regard as that of the residual activity in each organ. The ratio of the integral for the GI tract to that for the lungs was determined to be 15 from the count rate profiles. The assumption of the counting efficiency is roughly reasonable because of attenuation in the soft tissue for the measured photon energy (622 keV of ^{106}Rh): total attenuation coefficient for the photon energy in soft tissue is 0.09 g/cm^3 (half value thickness: about 8 cm).

Table 4 shows the residual activities in the following organs at 0.2 days post intake: the lungs, the GI tract – Total of stomach, small intestine and upper/lower large intestine – and the ET region. The residual activity in the ET region is almost equivalent to that in the ET_1 region because of rapid clearance from the ET_2 region. The aerosol size could be determined if the count rate profile was obtained prior to decontamination of the nasal passage since the initial activity in the ET region is considerably influenced by that. The ratio of the GI tract to the lungs was larger than that obtained from the count rate profiles but the discrepancy would be mitigated if the counting efficiencies of each organ can be evaluated more precisely. The small discrepancy of the lung activity shown in Table 4 also explained small variation of the evaluated doses even using different assumptions about the intake in Table 3 because the equivalent dose of the lungs is the biggest contributor to the evaluated dose (i.e., the committed effective dose).

A comparison of the observed and the predicted lung activity from RUN 2 in Table 3 is shown in Figure 9 up to 10 days post intake. The sum of the predicted values in the lungs and the GI tract was

roughly consistent with the observed lung activity. The rapidly decreasing component observed in the lung monitoring can be explained by overlapping with the residual activity in the GI tract as mentioned before. *In-vivo* measurements with comprehension of biokinetics of the radionuclide concerned are very important for persons in charge of internal dosimetry services.

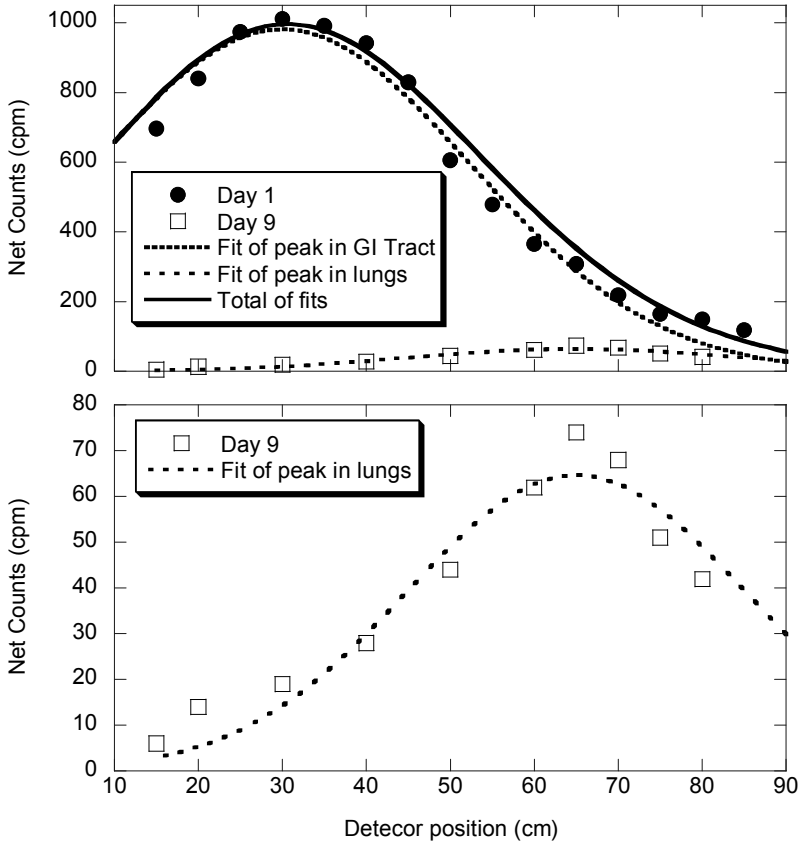

Figure 8 Fit of count rate profile to a Gaussian distribution function. The lower figure gives only the count profile on Day 9.

Table 4 Residual activities in the various organs for pure inhalation and simultaneous intakes.

	RUN 1	RUN 2	RUN 3	RUN 4	RUN 5
Lungs (Bq)	2.41E+03	2.34E+03	2.24E+03	2.11E+03	1.94E+03
GI tract(Bq)*	7.23E+04	4.52E+04	4.15E+04	4.19E+04	4.23E+04
ET region (Bq)	5.67E+04	3.47E+04	1.90E+04	8.31E+03	2.06E+03
GI tract/Lungs	30.0	19.3	18.5	19.9	21.8

* GI tract: Total activity of stomach, small intestine and upper/lower large intestine.

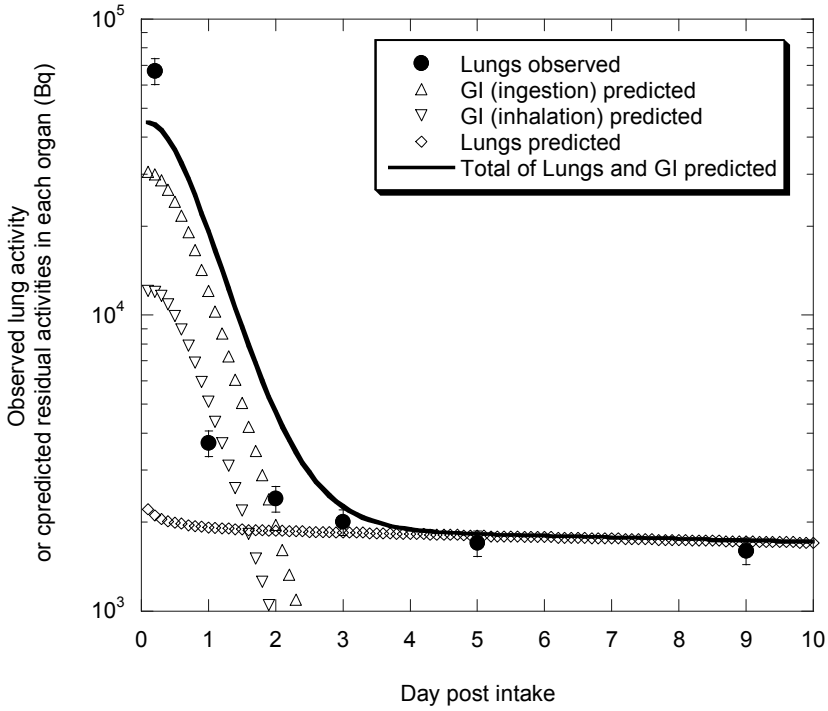

Figure 9 Comparison between the observed lung activity and the predicted residual activities in the lungs and GI tract up to 10 days post intake for the conditions of RUN 4 in Table 3.

5. Conclusions

This paper reviewed an old accidental intake case, in which two workers inhaled radioactive materials when they were experienced lack of oxygen while working in the cell of the reprocessing plant of JAEA-NFCEL. They were contaminated internally with ^{106}Ru as the main radionuclide. The authors analyzed the bioassay data obtained from the worker of them based on the guideline demonstrated in the EU project IDEAS. The findings are summarized as follows.

- The absorption of the materials in the lungs agreed well with the default Type S. Nitrosyl compound was the most likely chemical form according to the process taking place in the cell concerned. The effective half-life in the lungs was determined to be 140 days, which was shorter than that obtained from another case of inhalation of ruthenium oxide. In addition, the f_1 value of the materials was estimated to be at least less than one tenth of the default parameter value presented in the ICRP publication.
- The large-sized effective aerosol size evaluated on the condition of pure inhalation suggested there were simultaneous intakes via inhalation and ingestion. This was also supported by two pieces of evidence: external contamination of the worker and the ratio of the early fecal excretion to the activity found in nose swabs. The effective aerosol size was not able to be determined from the monitoring data available in simultaneous intakes; however, the resulting committed effective dose was slightly dependant on the aerosol size.
- Both the IMBA code and the REIDAC which have a good capability for modification of parameter values in the dosimetric models were found to be very useful in the analysis of the

bioassay datasets and the evaluation of the internal dose. The guideline including the related information was effective and informative. Detailed information on the incident other than individual radiation monitoring such as the process of concern, external contamination and the situation of intake are also important in internal dose assessment.

- It is necessary for internal dosimetrists to provide prompt dose assessment especially in the event of an accident. Scanning measurements by the whole-body counter to provide the activity distribution in the body are essential to avoid overestimation of the lung activity. Application of a simulation technique to evaluating response of *in-vivo* measurement instruments including biokinetics of a radionuclide of concern would be helpful for this task.

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2-7 Strategy on Quality Assurance in Radiation Fields and Calibration Techniques at FRS of JAEA

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Abstract

The FRS, Facility of Radiation Standards, of JAEA has been developing and supplying the fields of X-, gamma-, beta-rays and neutrons for calibrating the radiation measuring devices used for radiation protection monitoring since 1980. Much effort has been devoted in recent years toward the development of the neutron and gamma-ray fields using a 4 MV Van de Graaff accelerator. In November 2006, we just started to open the FRS radiation reference fields to national laboratories, private industry and academia. Almost the fields are traceable to national standards, but a laboratory quality assurance is not yet realized. We plan to obtain the calibration laboratory accreditation of JCSS, Japan Calibration Service System, which is based on ISO-17025. Furthermore, the quality assurance is planed to be extended to the international credibility in the future, and we hope to play a leading role in the calibration facilities in Asia.

Keywords: Calibration standards, Quality assurance, Radiation reference field, Radiation measuring devices, Radiation protection, Calibration techniques, Accelerator-based sources, JCSS, Laboratory accreditation, ISO-17025

1. Introduction

It is quite important to establish the reference radiation fields for calibrating area and personal dosemeters or other radiation measuring devices used for radiation protection and safety. The radiation monitors and personal dosemeters must be properly calibrated at regular intervals against appropriate standards. The Facility of Radiation Standards (FRS) of the Japan Atomic Energy Agency (JAEA) has been developing the reference fields of X-, gamma-, beta-rays and neutrons since 1980.[1)] Much effort has been devoted in recent years toward the developments of the neutron and gamma-ray fields using a 4 MV Van de Graaff accelerator. Until now, the established radiation fields have been utilized through a calibration service institute, the Institute of Radiation Measurement (IRM) in response to the demands from outside. Otherwise, the utilization had been limited only to radiation monitoring and research purposes inside our institute.

At the integration of former JAERI and JNC in October 2005, newly-constituted JAEA positioned FRS as one of the facilities to promote the utilization sharing. In November 2006, we started to open to national laboratories, private industry and academia and the usage is not limited to

calibration purposes, but is extended to research purposes.

Almost all of our radiation fields are traceable to national standards held (in Japan) at the National Institute of Advanced Industrial Science and Technology (AIST) and FRS is the comprehensive secondary standard facility of Japan. However, a laboratory quality assurance is not yet realized. Users of our reference fields would not only obtain information about the supply of radiation dose standards, but also tend to ask for the quality assurance of the given values. To meet their prospective requests, we are planning to obtain the calibration laboratory accreditation of JCSS, Japan Calibration Service System, which is based on ISO-17025. Furthermore, we plan to extend the quality assurance to the international credibility in the future, and we hope to play a leading role in the calibration facilities in Asia.

This paper describes the outlines of our radiation fields, the planning and provision of services and the approach to the quality assurance.

2. Facility of Radiation Standards (FRS)

2.1 General description

FRS was built in 1980 at the Nuclear Science Research Institute, Tokai Research and Development Center of JAEA. Figure 1 shows a cutaway-view of FRS. FRS is two-story building on the ground and one basement floor. There are four gamma-ray irradiation rooms, an X-ray room, a beta-ray room and two neutron rooms in the building. The neutron irradiation rooms are relatively large to reduce scattered neutrons. Neutron sources, except for thermal neutron source, are positioned in the centers and on the gratings at the mid-height of the rooms.

The reference radiation fields of FRS have been used for the JAEA's technology research and development and our routine calibrations of dosemeters, survey meters and area monitors used for radiation protection of workers and the public. On the other hand, the calibration service for external demands has been dealt by the Institute of Radiation Measurements (IRM), which is located next to FRS facility.

A 4 MV Van de Graaff accelerator was installed in 2001. Recent activities are focused on the development of accelerator-based reference radiation fields.

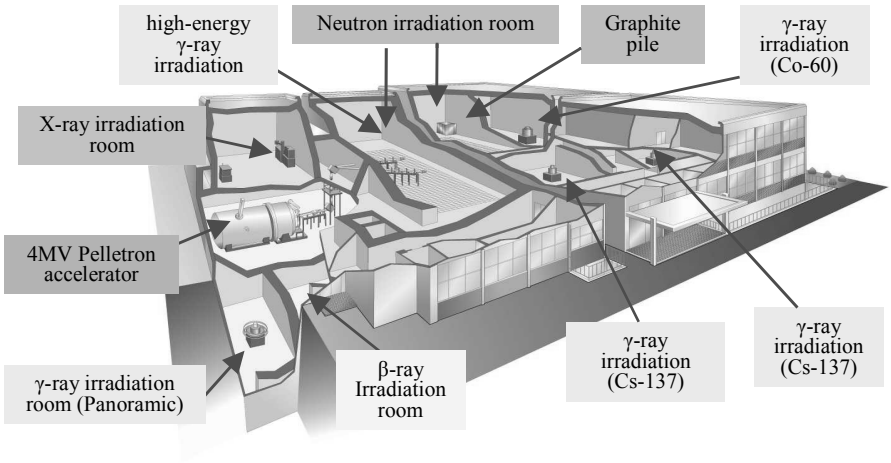

Figure 1 Cutaway view of FRS.

2.2 Established Reference Fields

Detailed information about the established fields at FRS can be found in the literature[1]. Here we pick up the major fields and briefly outline their general specifications.

2.2.1 Gamma-rays

There are four gamma-ray irradiation rooms and three of four rooms are for collimated beam irradiation and one for panoramic irradiation. The apparatuses for collimated beam irradiation contain ^{137}Cs or ^{60}Co point sources of different strength and the gamma-ray beam is collimated to 30 degrees full angle of divergence. Reference air kerma rates at points have been standardized by means of the EXRADIN ion chambers. The ambient dose equivalent rate, $H^*(10)$ and personal dose equivalent rate, $H_p(10)$ are converted from the reference air kerma rate. Our fields cover the dose range from 4 μSv/h to 5 Sv/h. In addition, irradiation of large number of passive dosimeters at a time is available in the panoramic irradiation fields with ^{137}Cs, ^{226}Ra and ^{60}Co. Irradiation by ^{133}Ba, ^{226}Ra and ^{241}Am bare sources are also available.

2.2.2 X-rays

There are four kinds of X-ray fields, “Medium and Hard”, “Fluorescent”, “Soft” and “Pulsed” X-rays, covering the energy from 8 keV to about 200 keV. Among the fields, the “Medium and Hard” X-ray reference field has been mainly used, which is produced by the X-ray generator (tube potential: max. 380 kV, tube current: max. 30 mA). With this generator, three spectrum series of narrow, wide and high air-kerma rate are provided in accordance with ISO 4037. It covers the dose rate range of from 100 μSv/h to 10 mSv/h.

2.2.3 Beta-rays

The beta radiation fields of ^{147}Pm, ^{204}Tl and ^{90}Sr+^{90}Y radionuclide sources specified in the standard ISO 6980 are available for calibration of beta measuring devices. Absorbed dose rates at 7 mg·cm^{-2} of tissue depth are determined by measurement with an extrapolation chamber manufactured by the PTW.

2.2.4 Neutrons from Radionuclide Sources

FRS can provide calibration services with the fast, thermal and heavy water moderated neutron reference fields. For fast neutron field, the radioisotope neutron sources of ^{252}Cf and ^{241}Am-Be specified in ISO 8529 are available. The neutron source emission rates are determined by AIST. The thermal neutron fields are produced with a large mass of graphite moderator. The ^{252}Cf, ^{241}Am-Be and ^{239}Pu-Be sources are placed in the moderator. The thermal neutron fluence rates are determined by the gold foil activation technique at points inside and outside the graphite of 1.50 m×1.64 m×1.50 m(H). A D_2O moderated ^{252}Cf neutron field specified in ISO 8529 is also established for the simulated workplace neutron fields.

To reduce the scattered neutrons with construction materials, the neutron sources are positioned almost in the center of the room. The dimensions of the room are 12.5×12.5×11.7 m^3 and it has a steel grating at the mid-height of the room.

2.3 Currently Developing Reference Fields

With the Van de Graaff accelerator, monoenergetic neutron fields and high energy gamma-ray fields are currently being developed at FRS. The accelerator, Pelletron type, can accelerate proton and deuteron ions up to 4 MeV. The beam current is 50 μA in dc operation. The operation in pulse mode can be done for time-of-flight measurements. Figure 2 shows the irradiation room of the accelerator-based radiation fields.

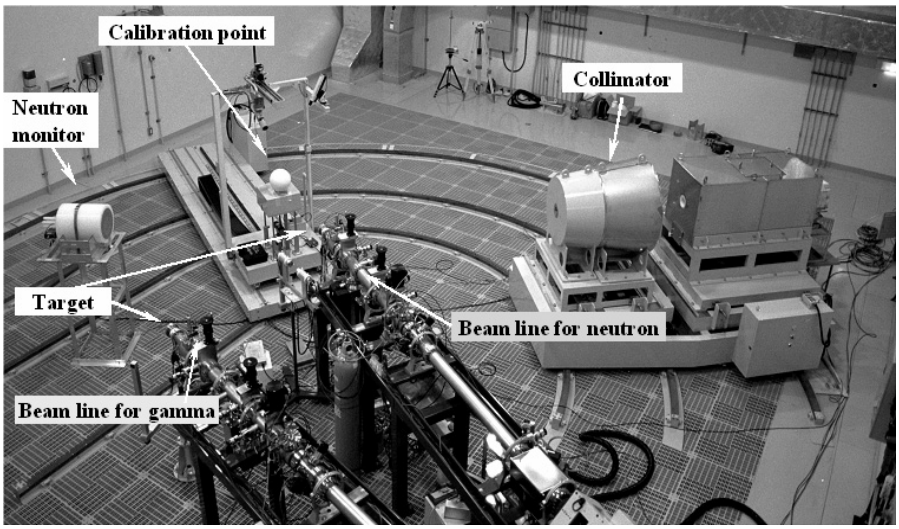

Figure 2 The accelerator-based monoenergetic neutron/high energy gamma-ray fields.

2.3.1 Monoenergetic Neutron Fields

Figure 3 shows the present status of the progress of the monoenergetic neutron fields. The nuclear reactions used for neutron production are also shown in the Figure. As the energy points to be developed, we initially set 10 energy points between 8 keV and 19.0 MeV. We have already established at the points of 144 keV, 250 keV, 565 keV created by ^{7}Li-p reaction, and at the energy points of 5 MeV and 14.8 MeV by D-d and T-d reaction, respectively. The 8 keV and 27 keV monoenergetic neutron fields by the ^{45}Sc-p reaction are presently under development. Almost energy points will be completed next year, after finishing the neutron energy points of the 1.2 MeV and 2.5 MeV by T-p reaction.

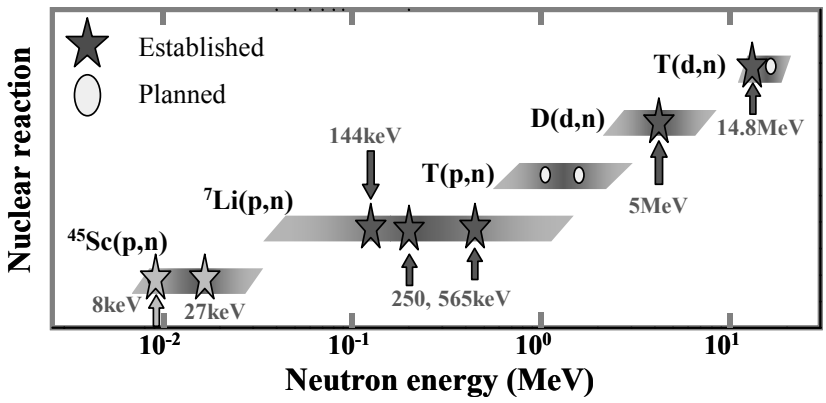

Figure 3 Energy points of monoenergetic neutron fields.

Like the neutron radionuclide source fields, the fields are established in a large room with an aluminum grating floor.

2.3.2 High-energy Gamma-ray Fields

The 6.13 MeV gamma-rays from ^{16}N are produced around the nuclear power plant. The exposure of the workers from these high energy gamma-rays is one of the important problems. However, the energy of general gamma sources is limited below about 3 MeV. Although the reference field of high energy gamma-ray has already established at a reactor JRR-4 of our institute, we are going to construct the field by using the accelerator at FRS for the users' convenience. The de-excitation of ^{16}O created by Fluorine and proton reaction is utilized here.

Created gamma-ray field is dependant on incident proton energy. We are constructing two kinds of gamma-ray fields. One is for purer energy gamma-ray field. The other is for high yield gamma-ray field.

3. Planning and Provision of Services

At the time of the integration of former JAERI and JNC, in October 2005, JAEA was asked to promote a widespread utilization of our facilities for external users. In response to the demand, in November 2006, we started to accept the external usages of the irradiation fields of FRS. At the first stage, we supply monoenergetic and radionuclide neutron fields and gamma fields. In next April, we will supply all of the established fields, adding X-ray and beta-ray fields. However, there are limitations. The request for us to accept will be basically limited to the irradiations for research, because of small number of staff and several regulations. Besides, there is also limitation on available time to offer, because JAEA staff are doing their researches using our reference fields. Regarding the accelerator-based reference fields, we can offer the time of about one week per month in maximum.

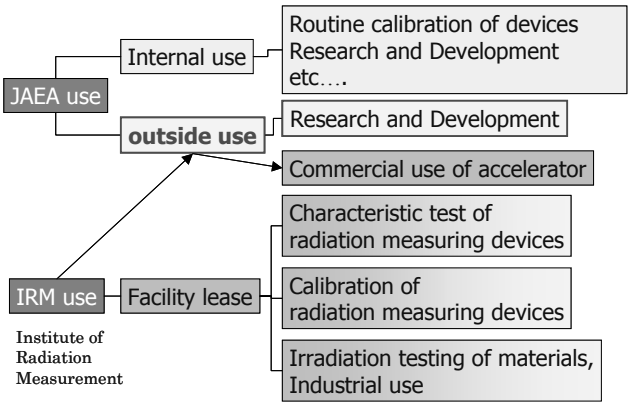

Figure 4 Task-sharing scheme between JAEA and IRM.

On the other hand, IRM has already been dealing with the calibration service for external demands, especially using the gamma and X-rays fields under the lease contract between JAEA and IRM (Figure 4). Except for the use of the accelerator, the utilization of our reference fields is basically as it has been in the past. For the accelerator-base irradiation of the tasks such as the calibration and

test of radiation measuring devices and the irradiation testing of material, IRM will take care of those tasks. It should be noted that the task sharing between JAEA and IRM might change, because we just got off to a start.

Those who want to use our radiation fields for their research purposes should apply for the utilization offerings planned twice a year. There are two ways depending on whether the user discloses the obtained results or not. If the users disclose their results, they can use the fields with the reduced charge. But they have to pass the prior examination by the review committee of outside experts and our staff and have the obligation of result reporting.

4. Quality Assurance (QA)

Most of our fields are traceable to primary national standards, but a laboratory quality assurance is not yet realized. There are mainly two reasons for obtaining the QA accreditation. One reason is that we should assure the traceability to the national measurement standards and a laboratory's technical and operational competence. The other reason is that our fields should be internationally acceptable to play a leading role in Asia on radiation reference field development in the future. As a result, we decide to obtain the accreditation of JCSS, Japan Calibration Service System. This JCSS is internationally acceptable, due to the international comparisons and recognitions among national metrology institutes. The accreditation body in Japan, IAJapan is a signatory to mutual recognition arrangements (MRA) of Asia Pacific Laboratory Accreditation Cooperation (APLAC) as well as International Laboratory Accreditation Cooperation (ILAC) (Figure 5).

Figure 5 Accreditation symbols of ILAC-MRA and JCSS.

Our schedule of obtaining the accreditation depends on a time table for providing the JCSS supply by AIST, national metrology institute in Japan. The radiation standards for X-rays and gamma-rays have already been supplied, and IRM owns the JCSS for the standards. For the neutron standards, AIST is going to start to supply. By the AIST plan, they plan to supply the monoenergetic neutrons standards at 4 energy points – 144 keV, 565 keV, 5.0 MeV and 14.8 MeV – in 2007, the thermal neutron standards in 2008, and the radionuclide neutron source standards in 2010. Concerning the beta-rays, there is no plan at this moment. We at FRS are going to obtain the JCSS, according to the time table.

Acknowledgment

The development and maintenance of the reference radiation fields at FRS involve the work of the past and present FRS colleagues in JAEA. The author wishes to thank all of them.

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Session 3　Emergency Response, Radiation Protection Standards and Waste Management

3-1 Current Emergency Programs for Nuclear Installations in Japan

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Abstract

Large effort has been taken for nuclear emergency programs in Japan especially after the JCO accident. A special law for nuclear emergency was established after the accident. The law extended the scope of emergency preparedness to fuel cycle facilities, research reactors, etc. and clarified the roles and responsibilities of the national government, local governments and license holders. For initial responses, the action levels and action procedures are defined based on environmental doses and specific initial events of NPPs. A senior specialist was dispatched to each site for nuclear emergency and a facility "Off-site center" to be used as the local emergency headquator was designated at each site. This paper describes the structure of emergency program, responsibility of related organizations and the definition of unusual events for notification and emergency. Emergency preparedness, emergency radiation monitoring and computer-based prediction of on- and off-site situation are also addressed.

Keywords: Nuclear emergency, Emergency preparedness, National government, Local government, Off-site center, Monitoring

1. Introduction

In Japan, about 1/3 of electricity is generated by nuclear power and the nuclear safety is essential for electricity supply. People are very sensitive to nuclear safety due to their experience of nuclear bombing and of nuclear emergency in the JCO accident.

Activities for nuclear emergency were highly enhanced after the JCO accident. Special law for nuclear emergency was established after the accident and the role of national government was strengthened. Local emergency centers (Off-site centers) were constructed at the vicinity of nuclear facility sites and exercises for nuclear emergency response are conducted every year by both national and local governments.

Emergency preparedness strengthened after the JCO accident is as follows:

(1) Extend the scope to include fuel cycle facilities, research reactors, etc.
(2) Clarify the roles and responsibilities of the national government, local governments and license holders.
(3) Improve initial responses;
   - define action levels by dose and specific initial events of nuclear power plants (NPPs).

– define the action procedures of national government.

(4) Designate a facility “Off-site center” to be used as a local emergency response headquarters at each site.

The last emergency response strategy is described in detail in the guide issued from Nuclear Safety Commission[1]. This paper outlines this guide.

2. The Extension of Scope to Fuel Cycle and Research Reactors

2.1 Emergency planning zone

The emergency planning zones revised after the JCO accident are shown in Table 1. Before the accident, the emergency planning zone was determined for only commercial nuclear power plants and it was 8 – 10 km. However, due to the extension of the scope of emergency preparedness to fuel cycle facilities and research reactors, the new emergency planning zones for them are determined.

Table 1 Emergency planning zones

Type of installation		Radius of EPZ
Power reactors and research reactors > 50MW(th)		8 – 10 km
Research reactors ≤ 50MW(th)	Power ≤ 1kW	50 m
	1kW < Power ≤ 100kW	100 m
	100kW < Power ≤ 10MW	500 m
	10MW < Power ≤ 50MW	1500 m
	Special design features	Define specifically
Spent fuel reprocessing plants		5 km
Fuel fabrication plants	Liquid, powder or gaseous fuel Enrichment > 5%, Pu fuel	500 m
	Others	50 m
Radioactive waste storage		50 m

2.2 Action level for sheltering and evacuation

The action levels for evacuation and sheltering are also revised after the JCO accident as shown in Table 2. Before the accident, the action levels are based on external dose and thyroid dose due to inhalation of iodine. However, because fuel cycle plants are included in the scope, the internal doses to inhalation of U and Pu are considered. Furthermore, the sheltering and evacuation for direct neutron or gamma rays from the site are recommended. Concerning the limitation of ingestion of foods, the action levels are determined based on concentrations of U, Pu and TRU as well as iodine and cesium.

Table 2 Action levels for sheltering and evacuation

Project doses (mSv)		Countermeasures
External dose	Internal doses ・ Thyroid dose to iodine ・ Lung or bone-surface doses to U or Pu	
10 ~ 50	100 ~ 500	Sheltering （In case of neutron exposure, sheltering in concrete buildings, or evacuation）
50 <	500 <	Sheltering in concrete buildings, or evacuation

3. The Roles and Responsibilities of National Government, Local Government and License Holders

National government (Ministry of Economy, Trade and Industry (METI) or Ministry of Education, Culture, Sports, Science and Technology (MEXT)) should designate a facility as Off-Site center in the vicinity of a nuclear installation, and prepare necessary equipments for the communication among the Prime Minister's Official Residence, the Cabinet, the Emergency Response Center of NISA, the Emergency and Disaster Countermeasure Center of MEXT, and related local government, and should dispatch specialists for nuclear emergency at Off-site center. Nuclear Safety Commission (NSC) is responsible to support the Prime Minister giving technical advices on a) the designation or alteration of the area of emergency measures to be taken, b) implementation of emergency measures, and c) dissolution of nuclear emergency.

For on-site emergency preparedness, a license holder should prepare its emergency action plan for prevention, mitigation and restoration of the emergency, including on-site/off-site cooperation and quick notification, after consulting with related local governments.

Concerning the off-site emergency preparedness, each local government should develop its own regional emergency plan, including environmental radiation monitoring, implementation of evacuation, sheltering and other protective measures of the residents, on receiving direction from the Prime Minister.

4. Improvement of Initial Responses

4.1 Action levels

Initial responses are conducted in two levels. Level 1 is a notification and Level 2 triggering of emergency. The references to determine the level are environmental doses and specific initial events of

NPPs. The reference dose is air dose rate at the boundary of site. When the measured dose exceeds the level of 5 μSv/h for more than 10 min. or two monitoring points observe the doses larger than 5 μSv/h, it becomes Level 1. For Level 2, the reference is 500 μSv/h. The dose for Level 2 was determined so that the countermeasures could be conducted for nuclear accident similar to the JCO accident.

Unusual events for notification and emergency power reactors for Level 1 are:

1) Loss of electric power supply over 5minutes during operation,
2) Failure of reactor shut-down by control rods when needed,
3) Loss of core cooling function (LOCA, Loss of feed water, etc.), and
4) Reduction of spent fuel storage pool water level down to the top of stored fuel assembly.

Unusual events for Level 2;

1) Total loss of electric power supply and core cooling capability,
2) Total loss of reactor shut-down functions when needed,
3) Total loss of ECCS during LOCA, loss of feed water, etc.,
4) Total loss of final heat sink of the rector system,
5) Detection of core melt,
6) Over pressure of containment vessel beyond max. design level, and
7) Reduction of spent fuel storage pool water level below the top of stored fuel assembly.

For research reactors, unusual events for Levels 1 and 2 are also determined.

4.2 Action procedures

Procedure of emergency response is as follows:

- If the event exceeds the Level 1 condition, a license holder immediately notifies METI (NPP) or MEXT (research reactor) and local governments.
- METI or MEXT triggers activity and sends staff to local governments.
- If the event exceeds the Level 2 condition, METI or MEXT reports to the Prime Minister.
- The Prime Minister declares nuclear emergency, establishes Emergency Response Headquarters (ERHQ) in Tokyo and at Off-site center, advises or directs local governments necessary measures to be taken.
- NSC establishes the technical advisory group to support the Prime Minister and local governments.
- Local governments establish their ERHQ, conduct emergency activities such as radiation monitoring, convey information, directions to local residents etc.
- A joint council for nuclear emergency response is established to coordinate the emergency measures at Off-site center.

Figure 1 shows the total structure of emergency response system. National government, NSC, licensee, and local governments (prefecture, city and town) participate in the planning of emergency response. National and local governments establish ERHQ and local ERHQ in Off-site center. NSC calls for

members of advisory group and some members are dispatched to the Off-site center for technical support. Licensee also establishes an emergency response team and some member participate in joint council in the Off-site center.

The plans made in Joint council are realized by police and fire protection agencies, self-defense forces, etc.

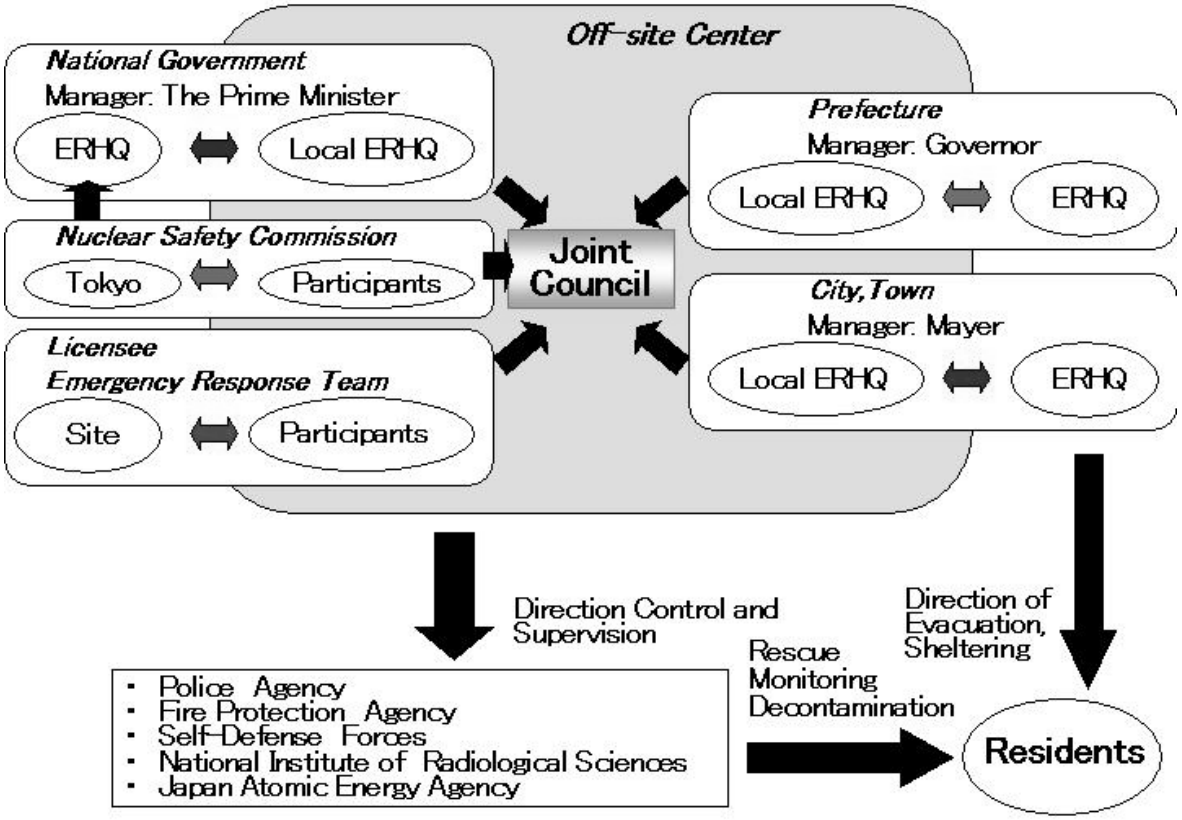

Figure 1 Total structure of emergency response system.

5. Facilities "Off-site Center" to be Used as the Local ERHQ

5.1 Functions in Off-site center

The actions in Off-site center are functioned as shown in Fig.2. The four groups, e.g., On-site, Radiation, Resident and Medical care groups, are established in Off-site center. These groups consist of experts from national government, local governments, NSC and Licensee.

The role of On-site group is the planning of countermeasures against on-site emergency, Radiation group the estimation and forecast of environmental contamination, Resident group the planning of countermeasures to protect residents and medical care group the planning of medical care for exposed personnel. Coordinate meetings are carried out time to time. And some meetings are conducted to exchange information and discuss the Prime minister with the planning.

For supporting these groups, some technical tools are prepared.

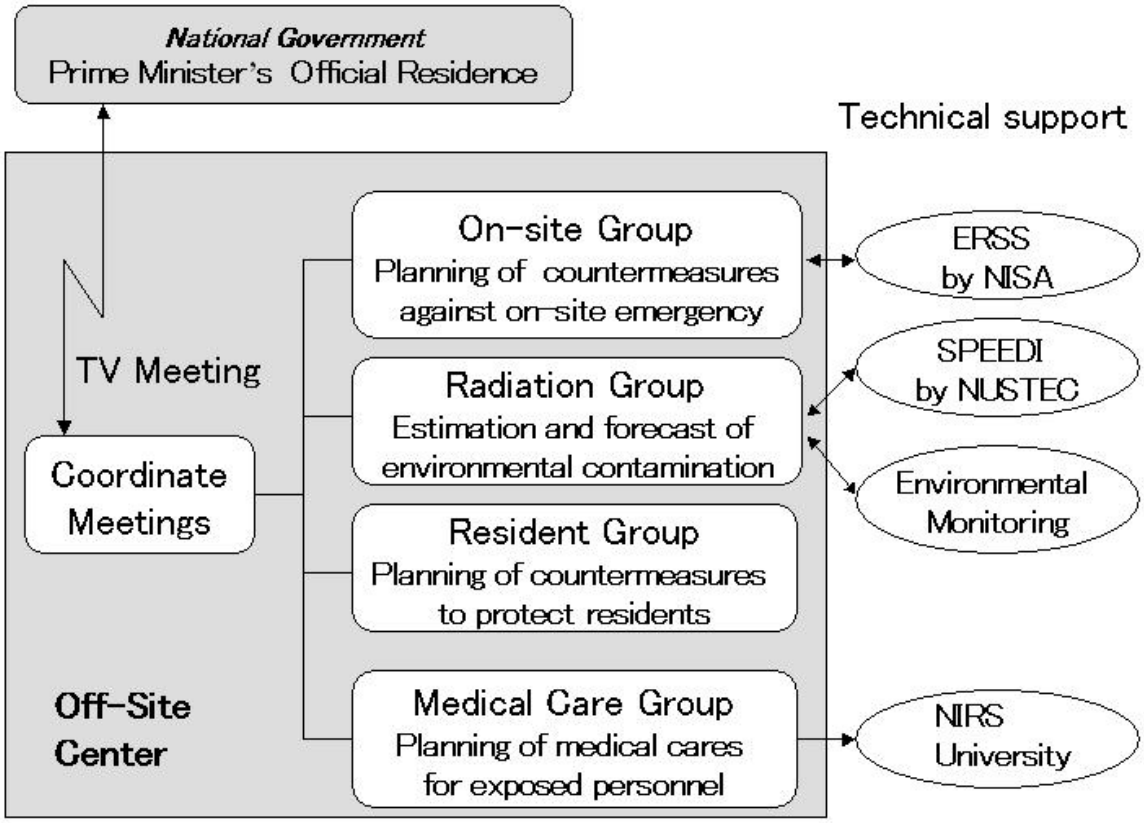

Figure 2 Function of Off-site center.

5.2 Technical support

(1) Emergency radiation monitoring

The radiation monitoring is conducted by monitoring centers of local governments, emergency monitoring team and supporting team dispatched from JAEA, NIRS and nuclear industry.

The monitoring is divided into two stages. The first stage radiation monitoring starts promptly on receiving the report of emergency and makes a monitoring plan depending on meteorological conditions. It measures radiation levels and concentrations of radioactive materials in the air and environmental samples in the vicinity of nuclear facility and, then, assesses the doses of residents for the decision of emergency action

The second stage radiation monitoring is detailed monitoring by expanding measuring points and kind of radionuclides to estimate the actual doses of the local residents and assess the general environmental hazard for the decision of long term preventive action.

(2) Computer-based technical support system

For the On-site group, Emergency Response Support System, (ERSS) operated by the Japanese Nuclear Energy Safety Organization (JNES), provides monitoring data of NPP plant parameters, indicates the state of unusual event and predicts the accident progress by analytical tools.

For the radiation group, System for Prediction of Environmental Emergency Dose Information (SPEEDI) operated by the Nuclear Safety Technology Center (NSTC) provides current and predicted meteorological conditions, geological and social information near the NPP site and performs real-time prediction of environmental and radiological consequences due to accidental discharge.

(3) Radiation emergency medicine

Emergency medical care is carried out for the urgent treatment of workers and local residents exposed in accidents. Basic procedure consists of three parts. The early stage care is conducted at nuclear facility, shelter and local hospitals near the site. Treatment to exposed patients is decontamination and first care. For local residents, surveillance, screening, dose estimation and iodine medication are conducted. The secondary stage care is at central hospitals near the site. Treatment of contaminated patients and dose estimation of high-dose patients are done. The third stage care is at specified governmental and university hospitals, where special treatment of high-dose patients. NIRS and some university hospitals are specified for this care.

5.3 Exercise

The purpose of emergency exercise is to enhance understanding of the nuclear emergency preparedness by responsible personnel of the national government, local governments, the license holder and residents and verify whether emergency measures function in predetermined way and information sharing and cooperation among related organizations are adequate.

Comprehensive nuclear emergency exercise in collaboration with the national government, local governments, license holders and supporting research organizations has been carried out once a year since 2001. Local government also conducts emergency exercise once a year for each government.

6. Conclusions

Large efforts are being taken for nuclear emergency programs especially after the JCO accident. On the basis of the special law for nuclear emergency, emergency preparedness including plans, organizations, systems and materials has been established throughout the country. Periodic emergency exercises highly enhance the capability of emergency response, e.g., sharing the exact information and understanding the responsibility for action among the related personnel and organizations.

Acknowledgement

The author would like to thank Dr. T. Fujishiro of Research Organization for Information Science and Technology for his assistance to providing the author with some materials on emergency response in Japan.

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3-2 Revision of the Protective Action Guides Manual for Nuclear Incidents

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Abstract

EPA's 1992 Manual of Protective Action Guides and Protective Actions for Nuclear Incidents,[1)] referred to as the PAG Manual, is a radiological emergency planning and response tool for emergency management officials at the Federal, state, tribal, and local levels. A Protective Action Guide is defined as, "the projected dose to reference man, or other defined individual, from a release of radioactive material at which a specific protective action to reduce or avoid that dose is recommended." The updated version of the PAG Manual accomplishes these key objectives: applying the existing 1992 protective action guides and protective actions to new radiological and nuclear scenarios of concern; updating the dosimetry basis; lowering the recommended dose for administration of stable iodine; providing new guidance concerning consumption of drinking water during or after a radiological emergency; updating the dosimetry basis for all derived levels, and, adding guidance for dealing with long-term site restoration following a major radiological release.

Keywords: Radiation emergency, Emergency response, Evacuation, Shelter-in-place, Potassium iodide, Nuclear emergency, Emergency planning zone, Protective action, Radiological incident, Dirty bomb

1. Introduction

The Manual of Protective Action Guides and Protective Actions for Nuclear Incidents published by the Environmental Protection Agency (EPA) is being revised. The proposed revision includes guides and recommendations that were developed cooperatively with the Federal Radiological Preparedness Coordination Committee, with representation from the Federal Emergency Management Agency (FEMA), the Environmental Protection Agency (EPA), the Department of Energy (DOE), the Department of Defense (DOD), the Department of Homeland Security (DHS), the Nuclear Regulatory Commission (NRC), the Department of Health and Human Services (HHS) including the Centers for Disease Control and Prevention (CDC) and the Food and Drug Administration (FDA), the Department of Agriculture (USDA), and the Occupational Safety and Health Administration (OSHA) within the Department of Labor.

Executive Order 10831, The Atomic Energy Act, as amended, and Reorganization Plan No.3 of 1970, charge the Administrator of the Environmental Protection Agency to "…advise the President

with respect to radiation matters, directly or indirectly affecting health, including guidance for all Federal agencies in the formulation of radiation standards and in the establishment and execution of programs of cooperation with states." This guidance has historically taken the form of qualitative or quantitative "Federal Radiation Protection Guidance (Federal Guidance), or Presidential Guidance to Federal departments and agencies. FEMA regulations (47 FR 10758, March 11, 1982) also instruct EPA to develop protective action guides (PAGs) and protective actions for nuclear emergencies, and Executive Order 12656 further instructs the Administrator of the Environmental Protection Agency to, "Develop, for national security emergencies, guidance on acceptable emergency levels of nuclear radiation."

EPA's 1992 Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, referred to as the PAG Manual, intended for use by emergency management officials at the Federal, state, tribal, and local levels, forms the basis by which emergency management officials may plan for and respond to radiological emergencies. The proposed revision of the PAG Manual accomplishes these additional key objectives: applying the existing 1992 protective action guides and protective actions to new radiological and nuclear scenarios of concern; updating the dosimetry basis from ICRP 26[2] to ICRP 60[3]; lowering the recommended dose for administration of stable iodine; providing new guidance concerning consumption of water during or after a radiological emergency; and, adding guidance for dealing with long-term site restoration following a major radiological release.

Development of the PAGs was based on four important principles, which also apply to the selection of any protective action before or during an incident:

⇒ Acute effects on health (those that would be observable within a short period of time and which have a dose threshold below which such effects are not likely to occur) should be avoided.
⇒ The risk of delayed effects on health (primarily cancer and genetic effects for which linear non-threshold relationships to dose are assumed) should not exceed upper bounds that are judged to be adequately protective of public health under emergency conditions, and are reasonably achievable.
⇒ PAGs should not be higher than justified on the basis of optimization of cost and the collective risk of effects on health. That is, any reduction of risk to public health achievable at acceptable cost should be carried out.
⇒ Regardless of the above principles, the risk to health from a protective action should not itself exceed the risk to health from the dose that would be avoided.

2. Application

Protective actions may be recommended for a wide range of incidents, but generally would be utilized for incidents involving relatively significant releases of radionuclides. Among the more potentially serious radiological incidents are: a transportation accident involving spent nuclear fuel, a fire in a major facility such as a fuel manufacturing plant, an accident at a Federal weapons complex facility, an accident at a commercial nuclear power plant, or an act of radiological or nuclear terrorism. Each type of incident would pose a unique threat to public health, and must be planned for and

managed accordingly. Thus emergency response planning for a given facility, or potential scenario, must consider the radionuclides that may be expected, the nature of release dynamics, the timing of notification, response and protective action implementation, and the feasibility of executing a particular protective action.

Officials responsible for emergency planning and policies should assess radiological and nuclear facilities, and potential scenarios that could lead to significant releases of radioactive materials, and use these principles and the accompanying PAGs and protective actions to perform emergency planning and exercises in advance. A PAG is defined as "the projected dose to reference man, or other defined individual, from a release of radioactive material at which a specific protective action to reduce or avoid that dose is recommended." A protective action is a recommended action associated with a PAG. Protective actions are those actions that have the effect of reducing or avoiding radiation dose. Examples include evacuating an area, sheltering-in-place within a protective structure, or acquiring an alternate source of drinking water.

The 1992 PAG Manual was written to accommodate the worst release scenario deemed likely at the time - a major accident at a commercial nuclear power plant (NPP) resulting in significant offsite release of radioactive material. Certain characteristics typify NPPs, including: fixed locations at which an accident might occur, a known suite of radionuclides on site that are dominated by short-lived radioisotopes, tight regulatory controls and requirements, skilled operational personnel that plan for and exercise emergency response, state and local involvement in emergency planning, well-developed and zoned emergency evacuation plans and routes, and substantial advance notice (generally hours to days) prior to accidental release of radioactive material into the environment. Therefore, the 1992 guidance provided radiation dose-based PAG values for decision makers for various exposure pathways (such as whole body, skin dose, and food ingestion), and protective actions which were adapted to some extent toward the mix of radionuclides released in NPP accidents, and to the operational environment of a commercial NPP.

Since then, new radiological and nuclear scenarios involving terrorist use of a radiological dispersal device (RDD) or an improvised nuclear device (IND) have gained priority status in radiological emergency response planning. An RDD is a device that combines conventional explosives or other diffusion device with radioactive material to scatter dangerous amounts of radioactive material over a general area. An IND is a crude, yield-producing nuclear weapon fabricated from diverted fissile material. These types of incidents may occur anywhere with likely no warning.

Evaluation of the threat posed by these and other potential incidents, including transportation and nuclear fuel processing accidents, has concluded that the PAGs and protective actions are applicable to all radiological incidents. This guidance therefore applies to all releases of radioactive material to the environment with the potential to impact public health. For purposes of this manual, a radiological incident is defined as "an event or a series of events, whether deliberate or accidental, leading to the release, or potential release, into the environment of radioactive materials in sufficient quantity to warrant consideration of protective actions." The definition includes acts of nuclear or radiological terrorism, but not nuclear war.

PAGs must accommodate all facilities and circumstances potentially confronting emergency

managers, including those that might occur at unpredictable locations, and those that occur with little or no warning. For example, an explosion or fire grants little or no warning and communities are likely to be caught by surprise, as would likely be the case in the event of an RDD or IND attack. Unpredictable locations make advance planning difficult. Sudden release of radioactivity into the environment leaves little time for officials to analyze options, and some protective actions, such as evacuation, may lead to greater net harm. Advance planning on the part of government officials and emergency responders for such cases is critical.

3. Protective Action Guides and Protective Actions for Radiological Incidents

Radiological emergencies, as defined in the PAG Manual, are divided into three incident phases for purposes of planning, preparation and response. The incident phases are defined as follows:

Early Phase – "The period beginning at the projected (or actual) initiation of a release and extending to a few days later, when deposition of airborne materials has ceased and enough information has become available to permit reliable decisions about the need for longer term protection." This phase may last hours to days.

Intermediate Phase – "The period beginning after the source and releases have been brought under control and environmental measurements are available for use as a basis for decisions on protective actions and extending until these protective actions are terminated." This phase may overlap the early phase and late phase and may last from weeks to months.

Late Phase – "The period beginning when recovery action designed to reduce radiation levels in the environment to acceptable levels for unrestricted use are commenced, and ending when all recovery actions have been completed." This phase may extend from months to years.

PAGs and their associated protective actions are applicable for both the early and intermediate phases.

Table 1 provides an overview of exposure routes and various protective actions and other activities based on the phase of the incident. The table shows which exposure pathways are of concern in the earliest time frames, and how the exposure pathways change over time after the incident. Sheltering and evacuation are the principal protective actions in the early phase. These actions are meant to avoid inhalation of gases or particulates in an atmospheric plume, but during this phase, consumption of contaminated food is generally not a priority issue. Administration of prophylactic drugs may be employed depending on the specific radionuclides released; in particular, potassium iodide (KI) may be administered in incidents involving the release of radioactive iodine, such as during NPP accidents.

Some protective actions may begin prior to release of radioactive material in cases in which advance notice is possible. Others, such as reentry and reoccupation in the affected area, would occur well after, perhaps only after lengthy decontamination and cleanup has occurred.

Table 1 Relationship between Exposure Routes, Activities and Time Frames

	EARLY	INTERMEDIATE	LATE
EXPOSURE ROUTE			
DIRECT PLUME	✸ —		
INHALATION PLUME MODEL	✸ —		
CONTAMINATION OF SKIN AND CLOTHES	✸ —	—	—
GROUND SHINE (DEPOSITED MATERIAL)	✸ —	—	—
INHALATION OF RESUSPENDED MATERIAL	✸ —	—	—
INGESTION OF CONTAMINATED WATER	✸	—	—
INGESTION OF CONTAMINATED FOOD	✸	—	—
ACTIVITY			
EVACUATION	✸ —		
SHELTERING	✸ —		
CONTROL OF ACCESS TO THE PUBLIC	✸ —	—	—
ADMINISTRATION OF PROPHYLACTIC DRUGS	✸ —	—	
DECONTAMINATION OF PERSONS	✸ —	—	—
DECONTAMINATION OF LAND AND PROPERTY	✸ —	—	—
RELOCATION	✸	—	—
FOOD CONTROLS	✸ —	—	—
WATER CONTROLS	✸	—	—
LIVESTOCK AND ANIMAL PROTECTION	✸ —	—	—
WASTE CONTROL	✸ —	—	—
REFINEMENT OF ACCESS CONTROL	✸	—	—
RELEASE OF PERSONAL PROPERTY	✸ —	—	—
RELEASE OF REAL PROPERTY	✸	—	—
REENTRY OF NON-EMERGENCY WORKFORCE	✸	—	—
REENTRY TO HOMES	✸	—	—

✸ RADIOLOGICAL RELEASE OCCURS — EXPOSURE OR ACTION OCCURS

PAGs are found in Table 2 with the principal associated protective actions. The PAGs are not meant to be applied as strict numeric criteria, but rather as guidelines to be considered alongside incident-specific factors. PAGs are for use only under emergency circumstances, and imply relatively short time periods during which exposures would occur. They do not imply an acceptable level of risk for normal, non-emergency conditions.

The PAGs and corresponding protective actions for the early and intermediate phases as found in the 1992 PAG Manual remain unchanged in the proposed revision, except that the PAG for administration of stable iodine has been lowered from 25 rem (250 mSv) adult thyroid dose to 5 rem (50 mSv) child thyroid dose. Additionally, this update provides a new PAG for drinking water and new guidance for cleanup during the late phase. PAGs are provided for the early and intermediate phases, where a PAG is "…a projected dose … at which a specific protective action to reduce or avoid that dose is recommended." The late phase, however, constitutes remediation and environmental restoration of the affected area, and thus is not appropriate for a PAG. No set values for late phase clean up can be derived in advance given the enormous breadth of potential consequences and site-specific factors. Rather, risk or dose-based clean up values must be established on an incident-specific basis (or the site would be cleaned up under existing regulatory authority of a responsible government agency).

Table 2 Protective Action Guides

Phase	**Protective Action Recommendation**	**Protective Action Guide**
Early	Sheltering in-place of the public	1 to 5 rem projected dose[a]
	Evacuation of the public	1 to 5 rem projected dose[b]
	Administration of prophylactic drugs - potassium iodide[c,d]	5 rem projected child thyroid dose
	Limit emergency worker exposure	5 rem (or greater under exceptional circumstances)[e]
Intermediate	Relocation of the public	2 rem projected dose first year. Subsequent years, 500 mrem/yr projected dose
	Food Interdiction	0.5 rem/yr projected dose, or 5 rem/yr to any individual organ or tissue, whichever is limiting[f]
	Drinking water interdiction	0.5 rem/yr projected dose
	Limit Worker Exposure	5 rem/yr
Late	Final site clean up and restoration	Site-specific optimization

[a] Should normally begin at 1 rem; however, sheltering may begin at lower levels if advantageous.
[b] Should normally begin at 1 rem.
[c] Provides protection from radioactive iodine only.
[d] For other information on radiological prophylactics and treatment please refer to www.fda.gov/cder/drugprepare/default.htm, or www.bt.cdc.gov/radiation/index/asp, or www.orau.gov/reacts.
[e] In cases when radiation control options are not available or, due to the magnitude of the incident, are not sufficient, doses to emergency workers above 5 rem (50 mSv) may be unavoidable. For further discussion see Chapter 2, Section 2.4.4.
[f] Committed effective dose equivalent

4. Early Phase PAGs and Protective Actions

The early phase is characterized by little or no data on actual releases to the environment and may necessitate crude estimates of airborne releases. Victims are triaged in the early phase. Decision time frames are short and preparation is critical to make prudent decision when data is lacking. Given the short time frames, panic is a major concern in the early phase potentially complicating evacuation orders, and leading to added risk to health due to injuries. Prompt, effective communication with the public, such as an order to shelter-in-place, is another major challenge. Officials should plan for rapid broadcast and dissemination of protective action orders to the public.

The principal PAG for the early phase is a projected dose of 1 to 5 rem (10 to 50 mSv) total effective dose equivalent, where protective actions would normally be initiated at 1 rem (10 mSv). The principal associated protective actions are evacuation and shelter-in-place. In cases where radioiodine may have been released, administration of the radioprotectant potassium iodide (KI, also called stable iodine) may be considered if the committed dose equivalent exceeds 5 rem committed effective dose equivalent to the child thyroid. The lower dose, proposed by FDA, is for protection of children based on new studies of Chernobyl exposure data. Decontamination is another protective action that may be utilized in the early phase and may include washing of contaminated individuals, changing out of contaminated clothing, and surficial decontamination of critical areas and objects. Individuals should also be instructed to cover breathing ways (nose and mouth) with available filtering material when airborne radionuclides may be present.

The decision to evacuate must weigh the anticipated radiation dose to individuals in the affected population against cost, feasibility of evacuation within a determined time frame, and evacuation-related injuries. For example, evacuating a population of 50,000 to avoid or reduce radiation dose to that population carries with it substantial monetary cost, and a statistical risk of injury or death associated with the evacuation. Evacuation also takes a given amount of time. In the case of a NPP accident the necessary time may be available to allow for orderly and relatively safe evacuation. However, when an incident occurs suddenly, as in the case of a fire or an RDD or IND in a dense urban area, evacuating a large group of people may increase the radiation dose to those people if they are caught within the plume or cannot escape a high dose rate outside area. Increased injuries and/or fatalities may ensue due to panic in a poorly executed evacuation. Sheltering in place may be warranted for situations where evacuation poses undue risks.

Limits of exposure for emergency workers are also recommended. The recommendations in the proposed revision remain unchanged from the 1992 PAG Manual. Responsible officials must use judgment when doses exceeding the OSHA annual limit of 5 rem (50 mSv) will be exceeded, and advise workers of the risks involved when doses approach 25 rem (250 mSv). There is no dose limit recommended for emergency workers performing life-saving activities. These emergency worker doses are presumed to be once in a lifetime events.

5. Intermediate Phase PAGs and Protective Actions

The intermediate phase begins when the source is under control and field data become available. Site stabilization and radiological characterization occur, as well as prompt removal and/or

decontamination of highly radioactive “hot-spots.” Intermediate phase activities are intended to reduce or avoid dose to the public, control worker exposures, control the spread of radioactive contamination, and prepare for long term cleanup operations.

Intermediate phase PAGs cover doses received in the first year, and those projected out to 50 years. Decisions must be made concerning the acceptability of occupation of homes and businesses by the public. If radiation doses in an area are deemed too high, temporary relocation should be implemented. The PAG for relocation of the public is 2 rem (20 mSv) projected dose in the first year and 0.5 rem (5 mSv) in any subsequent year (the intermediate phase dose does not include ingestion of food and water, which have separate provisions).

Keeping below the 0.5 rem (5 mSv) PAG for out years, the second year and beyond, may be achieved through allowing for the decay of shorter half-life radioisotopes (as in the case of a NPP accident), through decontamination efforts, or through other means of controlling public exposures (such as limiting access to certain areas). In the case of an RDD, in which a longer half-life radioisotope would likely be utilized, reductions in dose may prove difficult to achieve without full-scale site restoration. If out-year projected doses are estimated to remain above 0.5 rem (5 mSv) relocation should be considered. If, over a period of 50 years, the total dose to the individual is estimated to exceed 5 rem (50 mSv), the public should be relocated.

EPA is proposing a PAG for the consumption of drinking water of 0.5 rem (5 mSv) projected dose in the first year of exposure. If this PAG level is projected to be exceeded, the protective action would be to obtain an alternate source of drinking water. In some cases, water treatment or other actions may help reduce radiation doses received via drinking water. While this PAG applies to all potential sources of drinking water, radiological and nuclear emergencies will generally affect surface water bodies, such as lakes, rivers and reservoirs. Some may cause deeper ground water contamination, but this is less likely. The Water PAG is not intended to set an acceptable level of contamination in water, nor is it intended to serve as a remediation level in water. This PAG dose is in addition to the primary intermediate phase PAG.

The PAGs for the consumption of food and animal feed comes as a recommendation from the Food and Drug Administration (FDA). The PAG for the consumption of food, as well as the PAG for animal feed is 0.5 rem (5 mSv) projected dose in the first year of exposure, or 5 rem (50 mSv) projected dose to any particular organ or tissue (committed effective dose equivalent), whichever is more limiting. When food or animal feed becomes, or may become, accidentally contaminated to a level that can result in exposure to the public exceeding the PAG, protective actions should be considered. Simple protective actions include covering exposed products, moving animals to shelter, and providing protected feed and water to animals. Temporary embargoes on food and agricultural products may be necessary to prevent public consumption of potentially contaminated food.

Finally, during the intermediate phase, government officials begin convening to discuss long-term clean up and site restoration strategies. All actions taken during the early and intermediate phases should consider the impact they may have on long term remediation to avoid actions that will exacerbate or lengthen cleanup operations.

6. Late Phase Cleanup and Site Restoration

A multi-agency federal working group led by Department of Homeland Security studied the application of PAGs to RDDs and INDs and concluded that the existing EPA PAGs for the early and intermediate phase apply. However, late phase guidance did not yet exist. The working group developed guidance during that process, "Application of Protective Action Guides for Radiological Dispersal Devices (RDD) and Improvised Nuclear Devices (IND) Incidents"[4] that is being adopted in EPA's PAG Manual.

The late phase involves the final cleanup of areas and property contaminated with radioactive material. Unlike the early and intermediate phases of an incident, decision-makers will have more time and information during the late phase to allow for better data collection, stakeholder involvement, and options analysis. In this respect, the late phase is no longer a response to an "emergency," and is better viewed in terms of the objectives of site restoration and cleanup.

Because of the extremely broad range of potential impacts that may occur (i.e., ranging from light contamination of a small area, to widespread destruction of a major metropolitan area, in the case of an IND), a pre-established numeric guideline is not recommended as best serving the needs of decision-makers in the late phase. Rather, a process should be used to determine the societal objectives for expected land uses and the options and approaches available, in order to select the most acceptable criteria. For example, if the incident is of limited size, such that the impacted area is small, then it might reasonably be expected that a complete return to normal conditions can be achieved within a short period of time. However, if the impacted area is large, then achieving low cleanup levels for remediation of the entire area and/or maintaining existing land uses may not be practicable. Such a process of determining societal objectives may be called optimization.

Optimization (broadly defined) is a concept that is common to many state, Federal, and international risk management programs that address radionuclides and chemicals, although it is not always identified as such. Optimization is a flexible approach in which a variety of dose and/or risk benchmarks may be identified from state, Federal, or other sources (i.e., national and international advisory organizations). These benchmarks may be useful for analysis of remediation options and levels may move up or down depending on the site-specific circumstances and balancing of other relevant factors.

Optimization activities are quantitative and qualitative assessments applied at each stage of site restoration decision-making, from evaluation of remedial options to implementation of the chosen alternative. The evaluation of options for the late phase of recovery after an incident should balance all of the relevant factors. These factors may include: Areas impacted (i.e., size, location relative to population); Types of contamination (chemical, biological, and radiological); Other hazards present; Human health; Public welfare; Ecological risks; Projected land use; Preservation or destruction of places of historical, national, or regional significance; Technical feasibility; Wastes generated and disposal options and costs; Costs and available resources to implement and maintain remedial options; Potential adverse impacts (i.e., to human health, the environment, and the economy) of remedial options; Long-term effectiveness; Public acceptability, including local cultural sensitivities; and, Economic effects (i.e., tourism, business, and industry). The optimization process provides the best

opportunity for decision-makers to gain public confidence through the involvement of stakeholders. This process may begin during, and proceed independently of, intermediate phase protective actions.

7. Conclusion

The PAG Manual has been used for radiological emergency preparedness and planning for over thirty years, primarily in communities surrounding commercial nuclear power plants. The proposed revision provides updates and additions to aid communities planning and preparing for a wide range of radiological scenarios. The revision will be published for public comment in 2007.

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3-3 Some Aspects in the Compliance with the Japanese Radiation Protection Regulations

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Abstract

This paper gives an overview of the major subjects in the recent amendments to the Japanese radiation protection regulations. These are related to the scope of the application of regulations: exclusion, exemption and clearance. The Radiological Hazards Prevention Law has modified the legal definition of "radioactive materials". The Reactors Control Law has been amended to establish clearance levels for releasing radioactive materials from regulatory control. The Japanese government has a plan to develop guidelines for exclusion and exemption of certain types of naturally-occurring radioactive materials (NORM). Some aspects in the compliance with the regulations are addressed.

Keywords: Radiation protection, Law, Legislation, Regulation, Exclusion, Exemption, Clearance, NORM, ICRP, IAEA

1. Introduction

The national legislation and regulations are established in Japan to protect radiation workers and the general public against the risks associated with exposure to ionizing radiation and to ensure the safety of all types of radiation sources. The Atomic Energy Basic Act of 1955 provides the basic policy for promotion and regulation of the application of atomic energy in Japan. Article 2 of the Act stipulates that "the research, development and utilization of atomic energy shall be limited to peaceful purposes, aimed at ensuring safety and performed independently under democratic management, the results therefrom shall be made public to contribute international cooperation."

Under the Basic Act several specialized Laws are enacted. The Law for the regulations of nuclear source material, nuclear fuel material and reactors (Reactors Control Law) regulates practices in the entire nuclear fuel cycle, research reactors, application and transport of nuclear materials, etc. The Law concerning prevention from radiation hazards due to radioisotopes (Radiological Hazards Prevention Law) regulates radioisotopes used, sold, loaned and disposed of, etc. and radiation generators (particle accelerators). The competent Cabinet ministers issue their Ordinances in order to implement these Laws.

The legal person (operator) responsible to any such practice, radioisotope or radiation generator shall, unless these are beyond the scope of the regulation, apply the competent Cabinet minister for an authorization which shall take the form either a notification or a license. Licensees are usually

required to comply with multiple regulations.

These radiation protection regulations have introduced the recommendations of the International Commission on Radiological Protection (ICRP) and the radiation safety standards of the International Atomic Energy Agency (IAEA). In 2000 extensive amendments were made of the regulations, introducing the ICRP's 1990 recommendations[1]: the new dosimetric quantities were adopted to be applied to individual and workplace monitoring and the numbers of the dose limits for workers and the public were revised. However, several ideas in the 1990 recommendations of ICRP were not incorporated into the amended regulations and were submitted to future discussion. Among them was the concept of "exclusion and exemption from regulatory control".

Some important amendments were made recently to the Japanese radiation protection regulations. One of the major subjects in the amendments is related to the scope of the application of regulations: exclusion, exemption and clearance.

2. Exemption of Radiation Sources

The Radiation Council, one of the advisory bodies for the Government, investigated the applicability of the ICRP's concept of exclusion and exemption from regulatory control, and made a recommendations related to the concept in 2002. The Radiation Council concluded that the scope of regulatory instruments, beyond which regulatory provisions are unnecessary, should be based upon the newest scientific knowledge and international consensus. In radiation protection provisions, an instrument to establish the scope of regulation is the definition of "which is a radioactive source and which is not in a legal sense". The Radiation Council recommended that the ICRP's concept of exemption and the exemption principles provided in the International Basic Safety Standards (BSS)[2] could be applied to develop the definition.

The Radiation Council appreciated the scientific knowledge underlying these principles and then reviewed the exemption levels provided in Schedule I of the BSS. The Radiation Council made a peer review of the methodology of derivation of the exemption levels, re-calculating numerical values of the levels with the reviewed parameters and pathway models. The calculation was executed also for some additional exposure pathways, which were deemed to be specific to Japanese situations of the use of radiation sources. The Radiation Council concluded that these internationally accepted exemption principles and levels of the BSS were applicable to Japanese radiation protection regulations and recommended to adopt the entire set of the exemption levels for legally defining radioactive sources in Japanese regulations.

Based upon the recommendation of the Radiation Council, the Radiological Hazards Prevention Law and the related regulations were amended in 2004 and became effective in June 2005. The previous definition of radiation sources (see Table 1), which had been applied for over 40 years, was replaced by the nuclide-specific exemption levels. In addition to the exemption levels of about 300 nuclides provided in the Schedule I of the BSS, the levels of additional about 400 nuclides were introduced into the Japanese regulations from a report of the National Radiation Protection Board (NRPB), United Kingdom[3]. The NRPB derived these levels by the same method as that of the BSS.

Table 1 Previous definition of radiation source*

Source	Concentration	Quantity
Unsealed source	74Bq/g	Group 1** 3.7kBq Group 2: 37kBq Group 3: 370kBq Group 4: 3.7MBq
Sealed source	74Bq/g	3.7MBq per source

*: Substances for which both the concentration and the quantity exceed the numerical values given in this table are radiation sources.

**: Grouping by radiotoxicity: Nuclides in Group 1 have the highest toxicity.

Table 2 Previous and amended definition of sealed radiation source*

Source	Previous		Amended	
	Quantity (MBq)	Concentration (Bq/g)	Quantity (MBq)	Concentration (Bq/g)
Ni-63	3.7	74	100	1×10^5
Cs-137	3.7	74	0.01	10
Am-241	3.7	74	0.01	1

*: Substances for which both the concentration and the quantity exceed the numerical values given in this table are radiation sources.

Table 2 shows a simple comparison of the previous and amended (current) definition of sealed radiation sources. For a simplified example, any user who applies a sealed single-nuclide source of which the quantity and the concentration exceed the numerical values for the nuclide given in Table 2 shall have a license. A sealed Ni-63 source previously exempted from licensing is still exempted because the current exemption levels of Ni-63 are higher than previous ones. On the contrary a sealed Cs-137 source previously exempted from licensing now requires a license because the exemption levels have become lower.

The latter case can be found in the use of some consumer products. A smoke detector containing Am-241, which was exempted from the previous regulation, could be under radiological regulation because the typical content of Am-241 per detector, about 90kBq, exceeds 10kBq, the current exemption quantity for Am-241. However it would be impossible to put millions of smoke detectors currently in extensive use under radiation control. To mitigate such conflict in the compliance with the new exemption rule, the amended Radiological Hazards Prevention Law and associated regulations have a provision of “Design approval”. In case where the regulatory body authorizes a consumer product be inherently safe and approves its design for radiological protection, the product can be exempted from requirements of licensing and notification even if its contents

exceed the relevant exemption level. The radiological criteria for such authorization and design approval include: external radiation exposure not exceeding 1μSv per hour at 10cm from the accessible surface of the product and no possibility of internal exposure due to its contents.

In addition the full implementation of the amended exemption rules are suspended until March 2007: any product made before April 2007 will be exempted if its contents exceed the relevant exemption level. On the other hand, there can be licensed radioactive sources that shall be exempted from the regulatory requirements because the relevant exemption levels are higher than the previous numerical definition of radiation sources. In this case the licensee shall submit a revised license application to the regulatory authorities, returning the previous license for the sources now to be exempted, in order to comply with the amended regulation.

3. Clearance of Radioactive Materials

"Clearance" is a regulatory instrument to release materials from regulatory control. For example, radioactive waste under the regulatory control can be released to unconditional recycling without radiation protection when the radioactivity concentration of its contents is below prescribed clearance level. The Reactor Control Law and the related Ordinances were amended in 2005 to establish the clearance rule for free release of nuclear waste under the control of the Law. The numerical values of the clearance levels prescribed in these regulations are identical to the values provided in the IAEA's Safety Series document, RS-G-1.7[4]. The first application of the clearance rule is in progress in the decommissioning of the Japan's first commercial nuclear power plant.

On the contrary a possibility of establishing a clearance rule under the Radiological Hazards Prevention Law is still being examined by the Government. In its interim report issued in June 2006 an expert advisory committee in the Government identified two candidates to which a clearance rule could be applied: radioactive waste generated in dismantlement of particle accelerators and radioactive waste contaminated only with short-lived radionuclides.

Dismantlement of large accelerators in some research institutes are expected in near future and the management of associated large amount of radioactive waste to be generated is an issue in Japan. In addition smaller accelerators for medical purposes are also disposed of eventually. The interim report of the expert advisory committee indicated that the application of clearance rules to these wastes could solve the issues. However the interim report identified a difficulty in applying to accelerator wastes the already established methodology to demonstrate the compliance with the clearance rules specific to nuclear wastes. According to the interim report, this is because much more complex formation process of radioactivity in the accelerator wastes (activation process) makes it difficult to estimate activity concentrations in the wastes, which should be compare with the clearance levels.

The interim report also indicated that radioactive wastes contaminated only with short-lived radionuclides could be released from regulatory control after the storage for waiting for the decay-out of the activity. This is a similar concept to "decay-in-storage" provided in 10CFR Part 35[5]. A discussion was made in the interim report about the results from a feasibility study on the application of the concept of decay-in-storage to radioactive wastes generated in the industrial, research and

medical use of radioisotopes in Japan. The interim report underlined that it should be important for the licensees responsible to short-lived radioactive wastes to establish their quality assurance programs to eliminate unforeseen contamination with long-lived radionuclides. The Government plans further technical, legal and administrative investigation on the clearance rule under the Radiological Hazards Prevention Law.

4. Exclusion and Exemption of NORM

Naturally-occurring radioactive materials (NORM) raised another issues related to the scope of regulatory control. The ICRP analyzed this issue in Publication 60: sources that are essentially uncontrollable, such as cosmic radiation at ground level and potassium-40 in the body, can best be dealt with by the process of exclusion from the scope of the regulatory instruments, rather than by an exemption provision forming part of the regulatory instruments.

The Radiation Council published its report concerning regulation regarding NORM in 2003. The report reviewed Japanese industrial application of NORM, and revealed that in most cases the concentrations of NORM in the raw materials were low enough not to warrant radiation control. However the report identified some cases where concentrations of NORM in raw materials were considerably high. For example, Bastnaesite, which Japanese industry import from the United States of America 2,000-3,000 metric tons annually for manufacturing abrasive, contains Th-232 concentration of about 6Bq/g (higher than the BSS exemption level for Th-232, 1Bq/g). The report also identified some consumer products contained relatively high concentration of NORM. For example, a pigment contains about 10Bq/g of U-238 and about 80Bq/g of Th-232.

Most of these cases are not under radiological regulatory control, without recognizing the existence of radiation exposures. The report discussed radiation protection principle regarding NORM as follows. The raw materials the NORM concentration of which is not modified can be excluded from any radiation protection regulation because they are unamenable to control. In the case where an industry extracts target materials from raw materials containing NORM, the extraction process may cause radiation exposure of workers and the residues from the extraction may have condensed NORM contents and may become radiation sources, the exposure to which may be warrant to radiation protection. NORM can be condensed unwillingly in some application of certain raw materials: fry ash from coal-burning in the iron industry or in thermal power generation can be among examples. In other manufacturers consumer products are made from NORM or NORM is deliberately incorporate with non-radioactive raw materials of consumer products. In these cases the exposure pathways are added. In this regard these cases can be deemed to be the ICRP's definition of "Practices", where human activities increase the overall exposure to radiation. On the other hand these industries have already existed: some industries had started their operation even before the radiation protection legislatives were established. In this regard these cases can be regarded as ICRP's definition of "Intervention situation", where de facto exposure situations that are not a matter of choice but are already present.

Thus the report concluded that the NORM industry could have an ambiguous attribute, and could be protected by the radiation protection based upon both a "Practice" and an "Intervention".

As far as the NORM industry is a practice, radiation exposure of workers and the public could be controlled by the individual annual dose limit for the public exposure, 1mSv per year recommended in ICRP Publication 60. As far as the NORM industry is in an "intervention situation", any intervention to the industry might not be warranted when individual annual dose is below the intervention exemption criterion, 1mSv recommended in ICRP Publication 82[6]. In such a way the individual annual dose of 1mSv could serve as a numerical criterion for decision-making for further action upon NORM industry.

Considering this conclusion of the report of the Radiation Council, the Government is developing radiation protection guidelines for the NORM industry. The guidelines will advice the NORM industry to survey radiological situation in their plants for screening to determine whether the radiation levels in their sites exceed 1mSv per year, the practice/intervention criterion recommended by the Radiation Council. When exceeding annual individual dose criterion of 1mSv, the operator will be recommended to remedy the situation by, for example, altering the raw materials, modifying the production process or reducing the scale of production. Each operator in the NORM industry would be recommended to demonstrate the compliance with the guidelines. The government has a plan to develop an official manual for the NORM industry to demonstrate the compliance because most of the operators in this industry must be unfamiliar with radiation survey and radiological analyses.

5. Conclusion

The recent amendments to the Japanese radiation protection regulations deal with the scope of implementing regulations. The exemption levels for radiation sources and the clearance rule for waste management in decommissioning of nuclear facility have been established. The discussion about the exclusion/exemption guidelines for NORM is also related to the scope of a regulatory instrument.

The amendments and the developing guidelines will be appreciated because these are based upon the most recent radiological knowledge and the internationally-agreed radiation protection principles. On the other hand, for example, some radioisotope users may be required to comply with the new requirements of the amended regulations in their license applications, or some industries involving regulated NORM may have to make radiation protection programs in compliance with the guidelines. Operators will be required to be more accountable to the public for their endeavor. The effectiveness of the new regulations will be evaluated in all its aspects.

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3-4 The Latest Occupational Radiation Exposure Data from U.S. Nuclear Regulatory Commission Licensees

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Abstract

This presentation summarizes the latest – 2005 occupational exposure data that are maintained in the U.S. Nuclear Regulatory Commission (NRC) Radiation Exposure Information and Reporting System. The bulk of the information contained in the paper was compiled from the 2005 annual reports submitted by NRC licensees subject to the reporting requirements of U.S. regulations (10 CFR 20.2206). Those licensees subject to reporting include commercial nuclear power plants, industrial radiographers, fuel processors, independent spent fuel storage installations, manufacturers and distributors of by-product material, facilities for low-level waste disposal, and geologic repositories for high-level waste. The annual reports submitted by these licensees consist of radiation exposure records for each monitored individual. These records are analyzed for trends and presented in terms of collective dose and the distribution of doses by licensee category.

Keywords: Radiation exposure information and reporting system, Occupational exposure, Dose limits, NRC licensees

1. Introduction

I will discuss the latest occupational radiation exposure data from U.S. Nuclear Regulatory Commission (NRC) licensees. The information and graphs in this presentation are excerpted from the NRC's Radiation Exposure Information and Reporting System (REIRS) report, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities."[1] I encourage you to visit the REIRS website at www.reirs.com for additional information on the REIRS program and annual report.[2]

2. 2005 Data and Trends

The bulk of the data contained in this talk was compiled from the 2005 annual reports submitted by NRC licensees subject to the reporting requirements of U.S. regulations.[3] Those licensees subject to reporting include commercial nuclear power plants, industrial radiographers, fuel processors, independent spent fuel storage installations, manufacturers and distributors of by-product material, facilities for low-level waste disposal, and geologic repositories for high-level waste. The NRC does not currently license a low-level waste site in our jurisdiction; however, there are three low-level waste

sites in the U.S. that are located in, and regulated by, the States of Utah, South Carolina, and Washington. The annual reports submitted by these licensees consist of radiation exposure records for each monitored individual. These records are analyzed for trends and presented in terms of collective dose, average measurable dose (Figures 1 and 2), and the distribution of dose among the monitored individuals.

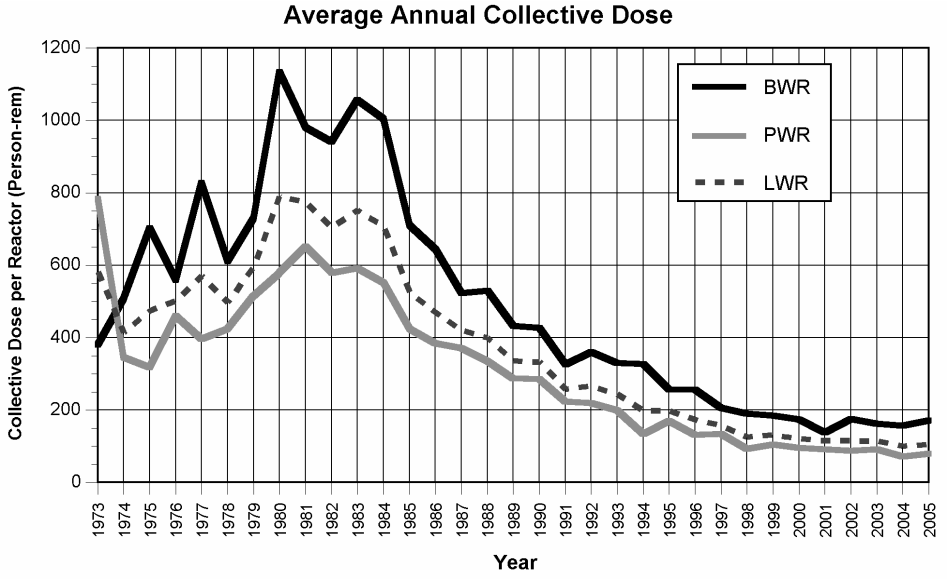

Figure 1 Average Annual Collective Dose (Person-rem): 1973 – 2005

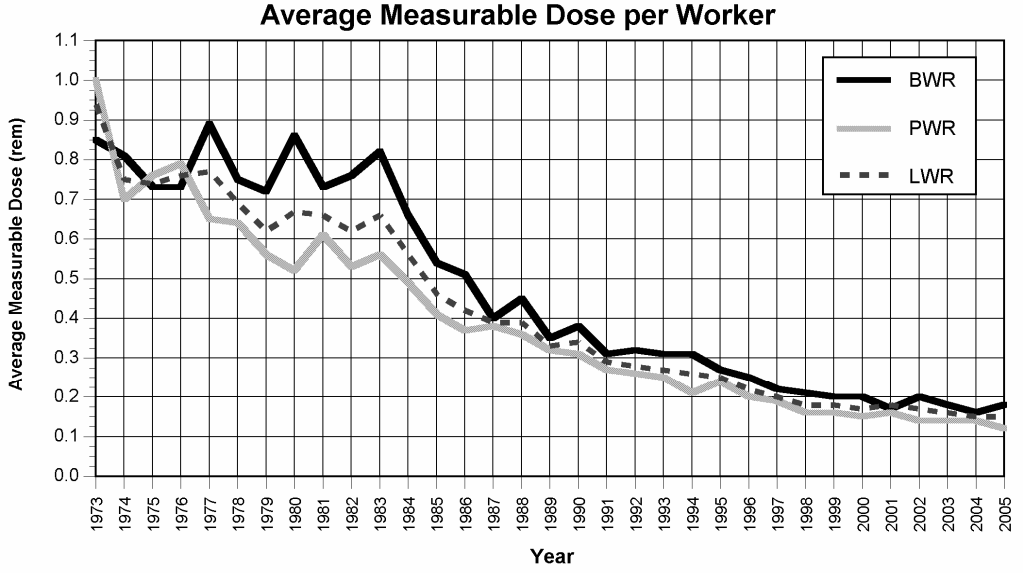

Figure 2 Average Measurable Dose per Worker (rem)

Annual reports for 2005 were received from a total of 218 NRC licensees, of which 104 were operators of nuclear power reactors in commercial operation (Figure 3). Compilations of the reports submitted by the 218 licensees indicated that 126,062 individuals were monitored, 64,246 of whom received a measurable dose. The collective dose incurred by these individuals was 137.33 person-sievert (13,733 person-rem), which represents an 8% increase from the 2004 value. The number of workers receiving a measurable dose also increased, resulting in an average measurable dose of 2.1 millisievert (mSv; 0.21 rem) for 2005; however, the average measurable dose did not increase from 2004. The average measurable dose is defined as the total effective dose equivalent divided by the number of workers receiving a measurable dose. The number of workers with measurable dose includes any individual with a dose greater than zero and does not include doses reported as "not detectable."

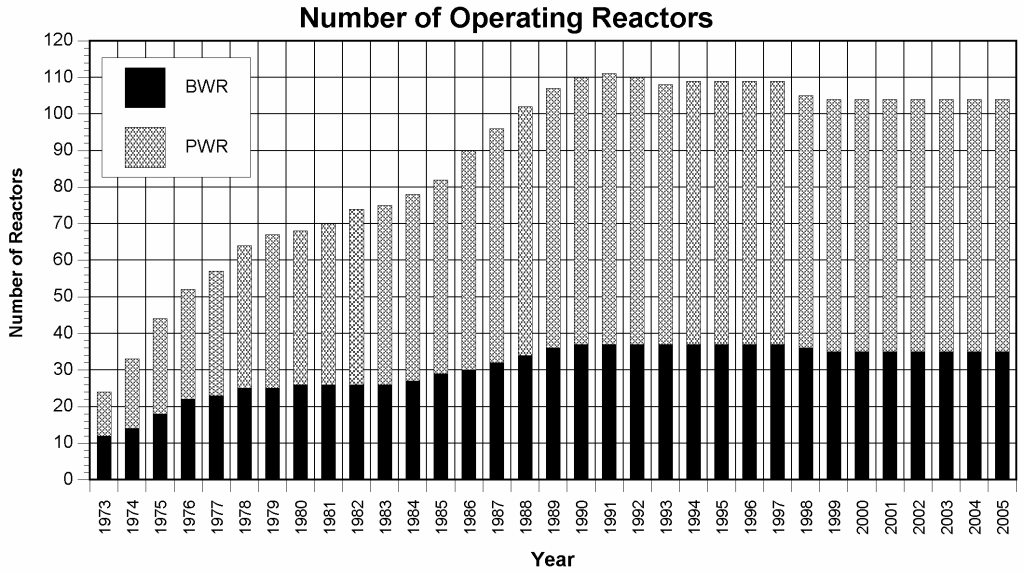

Figure 3 Number of U.S. operating reactors from 1973 – 2005

In calendar year 2005, the annual collective dose per reactor for light water reactor licensees was 1.1 person-sievert (110 person-rem). This represents a 10% increase from the value reported for 2004 of 1.00 person-sievert (100 person-rem). The annual collective dose per reactor for boiling water reactors and pressurized water reactors was 1.71 person-sievert (171 person-rem) and 0.79 person-sievert (79 person-rem), respectively. These figures for commercial reactors have been adjusted to account for transient reactor workers.

Analyses of transient worker data indicate that 26,936 individuals completed work assignments at two or more licensees during the monitoring year. The dose distributions are adjusted each year to account for the duplicate reporting of transient workers by multiple licensees. In 2005, the average measurable dose per worker for all licensees calculated from reported data was 1.6 mSv (0.16 rem).

The adjusted dose distribution for transient workers resulted in an average measurable dose per worker for all licensees of 2.2 mSv (0.22 rem).

The industrial radiographers had the highest average annual dose per licensee class of 5.4 mSv (540 mrem) compared to light water reactors of 2.0 mSv (200 mrem). Table 1 provides the average annual dose per all licensee categories except for low-level waste, because as discussed before, the NRC does not currently license this type of facility.

An interesting trending metric is the ratio of the average collective dose per Megawatt-year (MW-yr) of electricity generated at nuclear power plants. The ratio was calculated by dividing the total collective dose in person-rem by the electric energy generated in MW-yrs and is a measure of the dose incurred by workers at power plants in relation to the electric energy produced. From 1983 to 2005, the collective dose per MW-yr has decreased from 28 to 2 person-mSv (2.8 to 0.2 person-rem), respectively (Figure 4).

Table 1 Average dose by licensee category

Licensee Category	# of Individuals	Average Measurable Dose (mSv)
Industrial Radiography	2,476	5.4
Manufacturing and Distribution	804	3.9
Independent Spent Fuel Storage	30	0.3
Fuel Cycle Licensees	3,370	1.5
Light Water Reactors	57,566	2.0

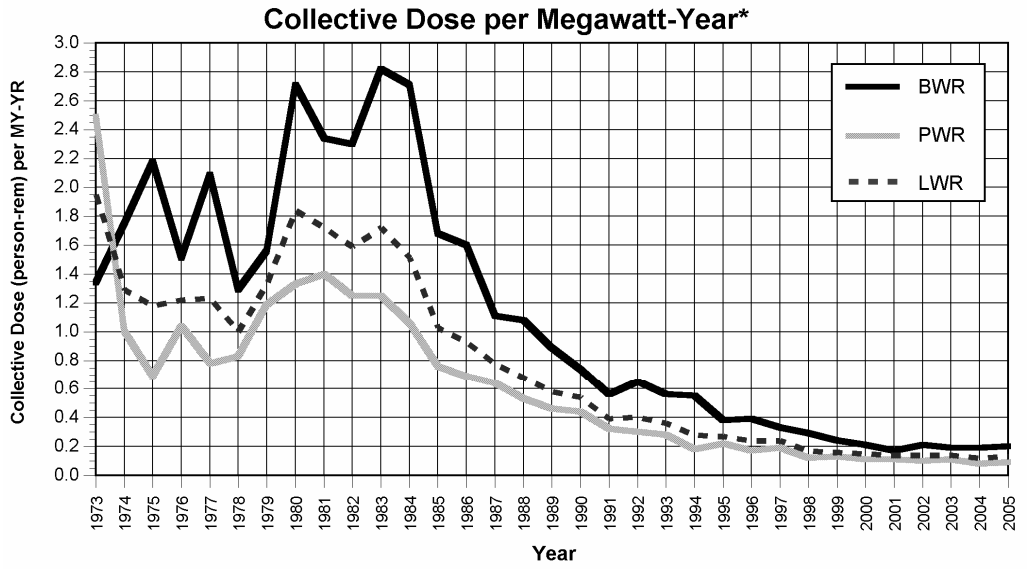

Figure 4 Collective Dose per megawatt-Year 1973 – 2005

3. Comparison to NRC's Annual Occupational Dose Limit

A useful application of the REIRS data is to evaluate exposures in relation to NRC's annual occupational dose limit of 50 mSv (5 rem). No individual in 2004 and 2005 monitored at an NRC licensed facility was reported to receive a dose above the limit. Approximately 95% of the annual exposures between 1968 and 1984 consistently remained < 20 mSv (2 rem). For the past 15 years, the percentage of workers with doses < 20 mSv (2 rem) has been greater than 99%. The number of workers receiving an annual exposure in excess of 5 rem has been <0.01% since 1985.

4. Data Sharing

NRC provides select annual dose data to the Organization of Economic Co-operation of Economically Developed Countries – Nuclear Energy Agency's International System on Occupational Exposure (ISOE). ISOE is co-sponsored by the International Atomic Energy Agency and NRC provides funding to the ISOE North American Technical Center to support additional trending of occupational exposure data at nuclear power plants. ISOE's objectives are to provide broad and regularly updated information on occupational exposure in nuclear power plants and provide methods to improve the protection of workers. Additionally, ISOE provides a mechanism for dissemination of information on these issues, including evaluation and analysis of the data assembled, as a contribution to the optimization of radiation protection.[4] World-wide there are 430 reactor units from 27 countries that participate in the program.

5. Conclusion

I would like to close the talk with examples of how the NRC uses the occupational exposure data discussed today in our regulatory programs:

- The data permit evaluation of trends, both favorable and unfavorable, from the viewpoint of the effectiveness of overall NRC/licensee radiation protection and as low as reasonably achievable (ALARA) efforts by licensees;
- The external dose data assist in the evaluation of the radiological risk associated with certain categories of NRC licensed activities and are used for comparative analyses of radiation protection performance: U.S./foreign, Boiling Water Reactors/Pressurized Water Reactors, civilian/military, facility/facility, and nuclear industry/other industries;
- The data are used as one of the metrics of the NRC's Reactor Oversight Program to evaluate the effectiveness of the licensee's ALARA program and also for inspection planning purposes;
- The data provide for the monitoring of transient workers who may affect dose distribution statistics through multiple counting;
- The data help provide facts for evaluating the adequacy of the current risk limitation system (e.g., Are individual lifetime dose limits, worker population collective dose limits, and requirements for optimization needed?);
- The data permit comparisons of occupational radiation risks with potential public risks when action for additional protection of the public involves worker exposures;

- The data are used in the establishment of priorities for the utilization of NRC health physics resources: research, standards development, and regulatory program development;
- The data provide facts for answering Congressional and Administration inquiries and for responding to questions raised by the public;
- The data are used to provide radiation exposure histories to individuals who were exposed to radiation at NRC licensed facilities;
- The data provide information that may be used in the planning of epidemiological studies; and,
- The data is shared internationally with the Nuclear Energy Agency's Information System on Occupational Exposure (ISOE).

Acknowledgement

Thanks to Ms. Sheryl Burrows, NRC, REIRS Project Manger and Mr. Derek Hagemeyer, Oak Ridge Associate Universities, REIRS Contractor Project Manager for their assistance in preparing this presentation.

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3-5 Discussion on Concepts for Radiological Dosimetric Quantities in the Japan Health Physics Society

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Abstract

Many dosimetric quantities have been used for radiation protection purpose. The International Commission on Radiological Protection (ICRP) has recommended protection quantities and the International Commission on Radiation Units and Measurements (ICRU) has introduced operational quantities to provide a reasonable estimate of the protection quantities. Enthusiastic discussions are continuously made on the issues of the dosimetric quantities, such as basic biological data for the definition of these quantities and applicability of the quantities to actual radiation protection practice. At the moment, some changes are being proposed concerning dosimetric quantities in the draft recommendations of ICRP, opened for consultation in recent years. Thus, the Japan Health Physics Society (JHPS) established the Expert Committee on concepts of Dosimetric Quantities used in radiological protection (ECDQ) in April 2005 to reviewed and discuss issues in the dosimetric quantities.

Keywords: Dosimetric quantity, Radiation protection, ICRP, ICRU, Protection quantity, Operational quantity, Internal exposure, External exposure

1. Introduction

The International Commission on Radiological Protection (ICRP) has published many recommendations for radiological protection. Dose limits have been recommended for public and workers to prevent detriments in health due to radiation exposure in some piblications, such as publications 26[1)] and 60[2)]. The dose limits have been given with dosimetric quantities, which have been introduced to quantify human health effect by exposures to radiations in each publication. The current system recommended in ICRP Publication 60 presents equivalent dose and effective dose as 'protection quantities' [2)]. On the other hand, the International Commission on Radiological Units and Measurements (ICRU) have discussed about quantities and their units for radiation measurements. Basic physical quantities, such as fluence and absorbed dose, have been defined in ICRU reports[3)]. In addition, many quantities named as 'operational quantities' have been defined to enable to estimate reasonably and conservatively protection quantities by radiation monitorings[4–6)]. Many countries,

including Japan, adopt the dosimetric quantities defined by ICRP and ICRU in regulation for the radiological protection.

ICRP and ICRU have amended the quantities, their units and relevant coefficients, if state-of-art scientific knowledge is available. The process of amendment, however, had not been clearly recognized in some past publications and reports. The radiological protection requires appropriate quantification of human health effect by radiation exposure from imparted energy per mass of human body (absorbed dose). Thus, the system of dosimetric quantity has its own complexity, which may not be found in other quantities. It is also suggested that the current system may not be appropriate for radiological protection in high energy accelerator and in space. The system of dosimetric quantity has been sometimes criticized due to the above reasons in Japan. ICRP has taken an open process to publish radiological protection recommendations in these several years and then has put several drafts on the web page. At the moment some changes are being proposed concerning dosimetric quantities in the recently opened drafts.

From these backgrounds, the Japan Health Physics Society established the Expert Committee on concepts of Dosimetric Quantities used in radiological protection (ECDQ) to review and clarify the issues in the current dosimetric quantities. This paper describes principal reviews, discussions and suggestions in ECDQ up to now.

2. Disputable Issues in Dosimetric Quantities

Table 1 lists principal issues on dosimetric quantities, which have been reviewed and discussed in ECDQ. Some of them had been suggested before ECDQ was established. The comments are mainly summarized in two topics; one is concerned with radiological protection and the other is concerned with radiation monitoring. Some of the present issues are described in detail in the next chapter.

Table 1 Principal disputable issue for dosimetric quantities in ECDQ

Quantities or relevant coefficients	Disputable issues
1) Quality factor Q(L) and radiation weigting factor, w_R	/ What biological data are bases for these factors? / Is the value of w_R consistent with the Q-L relationship?
2) Equivalent Dose, H_T	/ Is the $D_{T,R}$ appropriate as the basic quantity? / Is the H_T appeared only in the mid-step of effective dose?
3) Effective dose, E	/ Is this quantity applied to each individual? / Validity of the nominal risk coefficient in low dose range / Consideration of the uncertainty for this quantity
4) Operational quantity, $H^*(10)$, $H'(0.07)$, $H_p(d)$, (Activity of radionuclide within a human body)	/ Are the operational quantities always applicable to radiological protection? / Can we actually measure the operational quantities? / Different systems between external and internal exposures

3. Reviews and Discussions on Dosimetric Quantities in ECDQ

3.1 Protection quantities and relevant factors

ICRP introduces the protection quantities as below in the current recommendations.

$$H_T = \sum_R w_R \cdot D_{T,R} \quad (1)$$

$$E = \sum_T w_T \cdot H_T \quad (2)$$

$D_{T,R}$: Absorbed dose averaged over a tissue or organ T due to radiation R (Organ dose) (unit:Gy)

w_R: Radiation weighting factor of radiation R

H_T: Equivalent dose in a tissue or organ T (unit:Sv)

w_T: Tissue weighting factor of a tissue or organ T

E: Effective dose (unit:Sv)

Since the probability of stochastic effect is taken into account in the definition of the values for two weighting factors, radiation weighting factor (w_R) and tissue weighting factor (w_T), these quantities are considered to quantify human health effect from low dose exposure. Radiation doses inside human body can be calculated more and more precisely with progressed radiation transport codes, while large uncertainty is remained about consideration of biological effectiveness. One of the motivations to establish ECDQ was to clarify the basic biological data for these quantities and relevant coefficients. It was reviewed, however, that most publications have not shown the details of experimental data and a procedure to define the values, except the Q(y) in ICRU Report 40[7].

The protection quantities can be commonly defined for exposures from radionuclide within a human body (internal exposure) and radiations from outside of a human body (external exposure). Thus, ICRP approves to add the protection quantities for the different exposure pathways in radiological protection. Now, averaging absorbed dose over an organ or a tissue (‘organ dose’), $D_{T,R}$, is the base (or starting quantity) to derive the equivalent dose and effective dose. It is pointed out that the protection quantities based upon $D_{T,R}$ can bring about large uncertainty in dose assessments for internal exposure[8]. Dose distributions in organs can be analyzed for exposure from radioactive materials within a human body in detail now with a precisely defined model. On the other hand, it was suggested in ECDQ that introduction of new quantities for internal exposures can bring about an inconsistent element with the quantities in external exposures and then the protection quantities cannot be added for different exposure pathways.

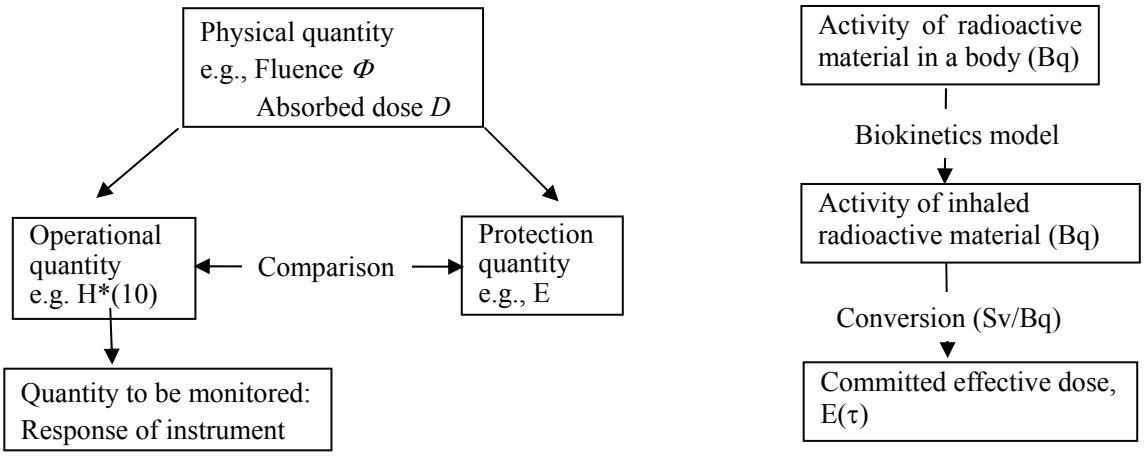
The protection quantities can be derived from the organ dose and the two weighting factors, which are described in above. The equivalent dose is one of the most disputable issues in ECDQ. Dose limits for some tissues are given by the equivalent dose to prevent the detriment effect (recently, ‘tissue reaction), although this quantity has the unit of Sv. The SI unit of ‘J/kg’ is applied to the equivalent dose, in addition to Sv. This can make confusions in radiological protection, because $D_{T,R}$ and H_T for an organ T can have different values with same unit for one exposure. Many members in ECDQ also have states that the equivalent dose is usually appeared in the mid-step of the effective dose calculation.

Since the effective dose can present health risk over a whole body from different types of radiations, this quantity is considered as the most significant quantity for the current radiological system. For example, the dose limits are regulated with the effective dose in Japan. The effective dose is derived with the standard model and parameters, which are presented in ICRP publications. ECDQ also have made reviews and discussions on the application and the meanings of these quantities for various exposure situations.

In addition, some members of ECDQ pointed out the validity of nominal risk coefficient for low dose exposure. ECDQ received a lecture on health effect from low dose exposures by an expert in the field of radiation biology. Through the exchange of comments among experts, it was recognized that difficulty exists in estimation of radiation risk in low dose region by extrapolation from experimental or epidemiologic studies for high dose exposures.

3.2 Quantities used in radiation monitoring

Since the relation between the protection quantity and the operational quantity is one of the most important issues in radiological protection practices, ECDQ has made discussions on this issue. ICRU has defined some operational quantities as the measurable quantities for monitoring of external radiations. Now, the ambient dose equivalent, $H^*(10)$, the directional dose equivalent, $H'(0.07)$, and the personal dose equivalent, $H_p(d)$ are being adopted in environment and individual monitoring against external exposures. These quantities are related with the protection quantities, as described in Fig. 1(a)[9]. On the contrary, dose assessment can be based upon the activity of radioactive material inhaled to a human body for internal exposures. From the view point of measurement, the activity of radionuclide within a human body may play a role as an 'operational quantity' to estimate the protection quantity for internal exposure, as shown in Fig. 1(b). In this figure, the committed effective dose is the protection quantity.



(a) External exposure[9] (b) Internal exposure

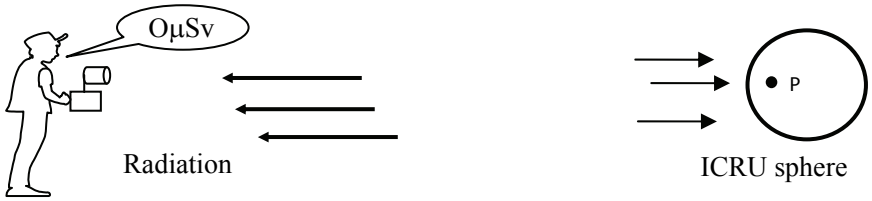
Fig. 1 Relation between the operational quantity (or measurable quantity) and the protection quantity in (a) External exposure and (b) Internal Exposure.

One of the major purposes of the operational quantity is to assess reasonably the protection quantities on the safer side by measurements. This requirement can meet radiological protection in most facilities, which have been principally considered up to now. However, it has been presented also in ECDQ that the effective dose can excess the currently used operational quantities for external exposures in high energy radiation facilities and in space. In addition, the operational quantities have never been actually measured with the ICRU phantom, which is realized based upon its definition.

4. Thinking of a More Comprehensive System for Dosimetric Quantities

It has been suggested in ECDQ that some dosimetric quantities have been misapplied or misunderstood to the radiological protection. If misapplications and misunderstanding arise from the current 'complicated' system, it may be preferable to introduce a 'simpler' system for dosimetric quantities. For example, one idea was once considered to give response of instruments with the conversion coefficient of the protection quantity for external exposures in Fig. 1(a). Since the 'response of instrument' can be moved to below protection quantity here, the proposed dosimetric quantity system may be similar to the current system for internal exposure in Fig. 1(b).

Significant difference, however, exists between the new proposed system for external exposure and the current system for internal exposures from the view point of radiation measurements. The activity of inhaled radioactive material is to be derived from the activity of radionuclide within a human body or the activity of radioactive material in airborne in actual radiological practices for internal exposures. The committed effective dose is therefore based upon the measured activity of radionuclide in Fig. 1(b). The new proposed idea for external exposures intends to measure protection quantities directly with any instruments. All the protection quantities are based upon the organ doses of T from radiation R, $D_{T,R}$, which cannot be principally measured. For radiations from external of human body, the dose can be determined by monitoring at one point, which corresponds to the definition of the operational quantity, as depicted in Fig. 2. The actual radiological protection can be confirmed only by any radiation measurements. Thus, the concept of any measurable dosimetric quantities should be maintained for radiological protection.



(a) Radiation monitoring (b) Definition of the operational quantity

Fig. 2 Image views of (a) radiation monitoring and (b) definition of the operational quantity.

ECDQ has a plan to present suggestions to exclude misapplication and misunderstanding of dosimetric quantities in the radiological protection by reflecting our reviews and discussions as a final report. The things below are to be taken into account here.

/ Further researches should be made on radiological protection with progressed technology and

knowledge. New concepts can be introduced for dosimetric quantities in the process of the researches. Actual practices for radiological protection should be taken into account to introduce a new quantity.

/ It may be difficult to completely replace the current system in Fig. 1 with a new one. On the other hand, the radiological protection relates to various fields of natural science. Thus, the dosimetric quantities should be more comprehensive among the experts concerning radiological protection.

/ Introductions of new term are not recommended except the case, where the current quantity is obviously wrong. This kind of amendment may confuse actual radiological protection practices.

5. Conclusion

ECDQ has made enthusiastic discussions on dosimetric quantities and some relevant factors defined by ICRP and ICRU. A simpler system may be preferable to avoid misapplication and misunderstandings in dosimetric quantities for radiological protections. The dosimetric quantity should be consistent with actual radiation protection, such as radiation monitoring, control of individual doses and so on. Thus, it can be considered that the currently used dosimetric quantity system cannot be replaced with a completely new system. On the other hand, the reviews and discussions in ECDQ can significantly contribute to exclude misapplication and misunderstanding of dosimetric quantities in the radiological protection. ECDQ will present statements about dosimetric quantities used in radiological protection, according to its reviews and discussions.

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3-6 Study on the Estimation of Probabilistic Effective Dose: Committed Effective Dose from Intake of Marine Products Using Oceanic General Circulation Model

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Abstract

The worldwide environmental protection is required by the public. A long-term environmental assessment from nuclear fuel cycle facilities to the aquatic environment also becomes more important to utilize nuclear energy more efficiently. Evaluation of long-term risk including not only in Japan but also in neighboring countries is considered to be necessary in order to develop nuclear power industry.

The author successfully simulated the distribution of radionuclides in seawater and seabed sediment produced by atmospheric nuclear tests using LAMER (Long-term Assessment ModEl for Radioactivity in the oceans). A part of the LAMER calculated the advection- diffusion-scavenging processes for radionuclides in the oceans and the Japan Sea in cooperate with Oceanic General Circulation Model (OGCM) and was validated.

The author is challenging to calculate probabilistic effective dose suggested by ICRP from intake of marine products due to atmospheric nuclear tests using the Monte Carlo method in the other part of LAMER. Depending on the deviation of each parameter, the 95th percentile of the probabilistic effective dose was calculated about half of the 95th percentile of the deterministic effective dose in proforma calculation. The probabilistic assessment gives realistic value for the dose assessment of a nuclear fuel cycle facility.

Keywords: Aquatic environment, Nuclear fuel cycle, Radionuclides, Probabilistic assessment, Deterministic assessment, Effective dose, Atmospheric nuclear tests

1. Introduction

The worldwide environmental protection is required by the public. A long-term environmental assessment from nuclear fuel cycle facilities to the aquatic environment also becomes more important to utilize nuclear energy more efficiently. Evaluation of long-term risk including not only Japan but also neighboring countries is considered to be necessary to develop nuclear power industry.

Predictive computer models for studying the dispersion of radionuclides from authorized discharge, e.g. from the Sellafield and La Hague reprocessing plants, from unpredictable releases, e.g. from radioactive waste dumping sites, from the Mururoa nuclear weapons test sites and from sunken nuclear ships and submarines have been developed and discussed. But, no long-term assessment of radioactive discharge from a site along the ocean, e.g. Tokai-mura and Rokkasyo-mura, was

performed.

On the other hand, the author successfully reproduced the distribution of radionuclides in seawater and seabed sediment produced by atmospheric nuclear tests using LAMER (Long-term Assessment ModEl for Radioactivity in the oceans). The LAMER calculated the advection-diffusion-scavenging processes for radionuclides in the oceans and the Japan Sea in cooperate with Oceanic General Circulation Model (OGCM) and was validated with cesium-137 (^{137}Cs) and plutonium-239,240 (^{239,240}Pu).[1–3] It also became possible to calculate the radionuclides from nuclear fuel cycle facilities by LAMER.

Recently, International Committee of Radiation Protection (ICRP) suggested new recommendation draft.[4] It implies probabilistic assessment as well as traditional deterministic assessment for dose evaluation. It might be economic to assess the probabilistic risk from nuclear fuel cycle facilities. In this study, the deterministic and probabilistic effective dose from the past atmospheric nuclear weapon testing was carried out by LAMER and discussed as preparatory investigations into the assessment from nuclear fuel cycle facilities.

2. Models and parameters

2.1 Outline of LAMER

The conceptual figure of LAMER, which contributes risk assessment of nuclear fuel cycle facilities, is shown in Figure 1. It consists of Part A, which calculates the long- and short- term behavior of radionuclides in marine environment, and Part B, which performs deterministic or probabilistic risk assessment considering concentration factor, intake of marine products, dose coefficient and so on.

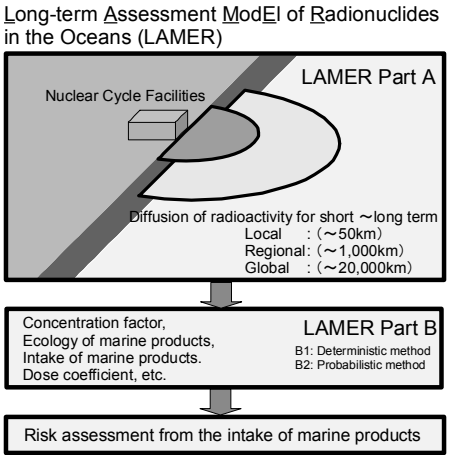

Figure 1 Conception of a Long-term Assessment Model of Radionuclides in the Oceans (LAMER).

2.2 Calculation model of the long- and short- term behavior of radionuclides in marine environment (Part A)

(1) Oceanic general circulation model

Several models of OGCM were published. In this study, diagnostic OGCM developed by Fujio and Imasato[5] was adopted. Forecast models, which analytically solve water temperature and salinity, cannot precisely determine the evaporation from sea surface, rainfall, inflow from river and so on. Thus the calculated value of water temperature and salinity would be different from the real value. And the calculated flow would be inconsistent with real flow. The diagnostic model calculates the density field with the observation value of water temperature and salinity to avoid above difficulties. The diagnostic model numerically solves the equations of motion with the density field. The three-dimensional flow field of the world ocean can be calculated by the diagnostic OGCM. The detail of the OGCM method was shown elsewhere.[1]

(2) Advection-diffusion model

The calculus of finite differences or the random walk method can be applied for modeling of advection-diffusion process. The calculus of finite differences has an advantage for calculation time, but it has some difficulties, namely it cannot be imaged the process of diffusion, cannot be tracked history, induces dummy diffusion up to grid size. On the other hand, the random walk method solves these problems, though longer calculation time is needed. The detail of the random walk method was shown elsewhere.[1)]

(3) Scavenging model

The one-dimensional scavenging model by Perianez[6)] was adopted with some improvement and the behavior of Pu was calculated in cooperate with random walk method. The equations of the reversible exchange model which considered the rapid sedimentation particle are shown in Eqs. (1)–(3) and Figure 2. The detail of the scavenging model was shown elsewhere.[2)]

$$\frac{\partial C_d}{\partial t} = -k_1(z)C_d + k_2 C_s \tag{1}$$

$$\frac{\partial C_s}{\partial t} = (1-f)k_1(z)C_d - k_2 C_s - \frac{\partial}{\partial z}\left(w_z C_s\right) \tag{2}$$

$$\frac{\partial A_s}{\partial t} = \int_{surface}^{bottom} f k_1(z) C_d dz + w_z C_s^* \tag{3}$$

where,

C_d,C_s : radionuclide concentration in the soluble and particulate phase (Bq/m^3)

A_s : radionuclide concentration in seabed sediment (Bq/m^2)

* : water layer which contacts with seabed

k_1, k_2 : adsorption and desorption velocity (/s)

f : rapid sedimentation fraction (-)

w_z : vertical sedimentation velocity (m/s)

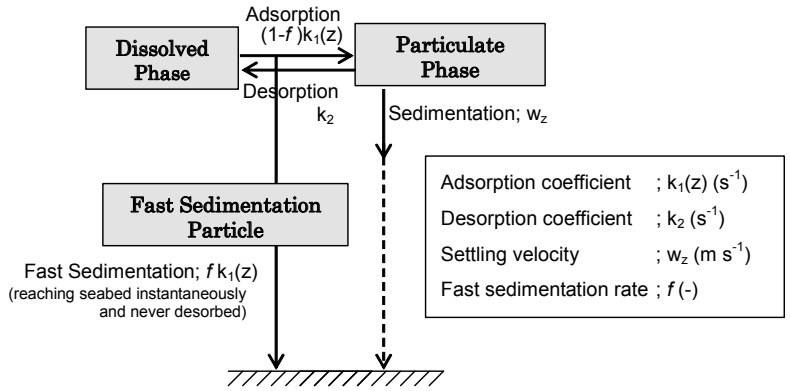

Figure 2 A simplified scavenging model used in this study.

2.3 Deterministic risk assessment of intake of marine products (Part B1)

(1) Preparation for input data

(a) Radionuclide concentration in surface seawater

The radionuclide concentration in seawater from the past atmospheric nuclear weapon tests was calculated in 2 degree grids horizontally by the method described in Part A. The objective nuclides were selected as tritium (^{3}H), carbon-14 (^{14}C), strontium-90 (^{90}Sr), ^{137}Cs and ^{239,240}Pu, which were considered to be important for dose evaluation, i.e. long half-life (approximately over 10 years) and large discharge amount. The radionuclides concentration in surface seawater averaged in the geographic division of fishery defined by United Nations Food and Agriculture Organization (FAO)[7] is shown in Figure 3. The radionuclides concentrations of ^{137}Cs, ^{239,240}Pu in the averaged surface seawater from 1950 to 2003 are shown in Figure 4.

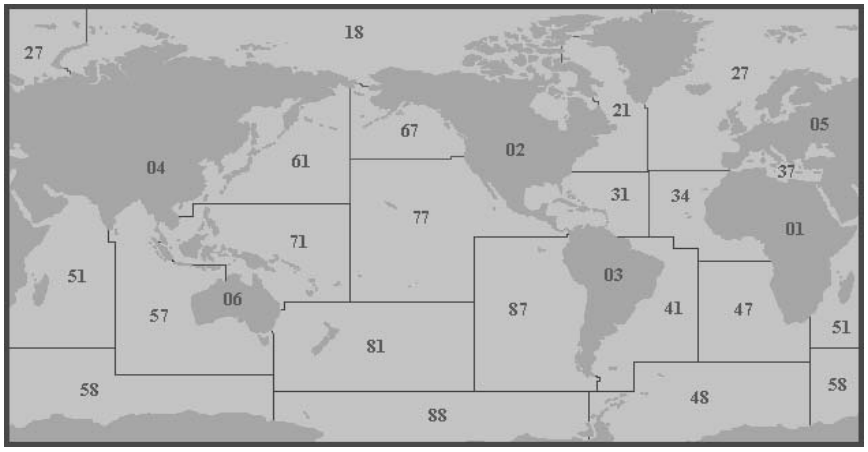

Figure 3 Partitioned world map for fishery statistics by FAO.[7]

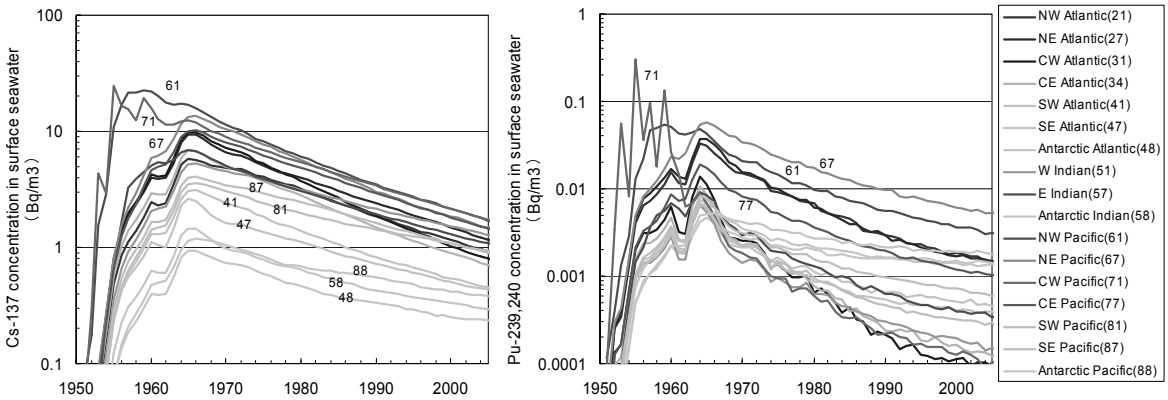

Figure 4 Chronological concentration of ^{137}Cs (left) and ^{239,240}Pu (right) in surface seawater calculated by LAMER.

(b) Intake amount of marine products

The world and Japanese total marine production from 1950 to 2003 were compiled from FAO database[7]. The total production includes non-edible parts such as shell, bone, and lever. And it

includes animal feeding stuff. Thus the intake amount was not the same with the production amount.

As shown in Table 1, the Japanese intake ratio, which was defined by (intake) / (production), was calculated with Japanese total production by FAO and intake in Japan by the national nutrition survey of Ministry of Health, Labor and Welfare (MHLW)[8]. The world intake ratio was assumed to be same with the Japanese intake ratio. The Japanese and world intakes of marine product can be calculated by the Japanese and world production multiplied with the Japanese intake ratio.

Table 1 Estimated intake ratio for each marine product.

species	production in Japan*1 (kt/a)	intake in Japan *2		estimated intake ratio (%)
		per capita (g/d)	total (kt/a)	
fish	4,787	77.8	3,582	75
crustacea, cephalopod	902	15.4	709	79
shellfishery	855	5.0	230	27
seaweed	683	5.2	239	35

*1 Total marine production in 1997 calculated by FAO[7]

*2 Intake of marine products in 1997 by calculated MHLW (Ministry of Health, Labor and Welfare)[8]

(c) Concentration factor

The concentration factors of marine products shown in Table 2 were quoted from IAEA Technical report No.247.[9]

(d) Dose coefficient

The dose coefficients of radionuclides shown in Table 2 were quoted from ICRP publication 72.[10] The most conservative chemical form was applied.

Table 2 Concentration factor and dose coefficient for each nuclide.

Nuclide	concentration factor[9]					dose coefficient[10] (Sv/Bq)
	fish	crustacea	shellfishery	cephalopod	seaweed	
^{3}H	1	1	1	-	1	4.2×10^{-11}
^{14}C	20,000	20,000	20,000	-	10,000	5.8×10^{-10}
^{90}Sr	2	2	1	2	5	2.8×10^{-8}
^{137}Cs	100	30	30	10	50	1.3×10^{-8}
^{239}Pu	40	300	3,000	50	2,000	2.5×10^{-7}

(e) Population

The world population was quoted from “Total Midyear Population for the World: 1950-2050” by U.S. Census Bureau.[11] The Japanese population was quoted from the survey of Bureau of Statistics, Ministry of Public Management, Home Affairs, Posts and Telecommunications.[12]

(2) Calculation of averaged effective dose

The collective doses of each continental group in case of the intake of marine products were calculated by multiplying the value of (a) to (d) of (1) as Eq. (4).

$$S = \sum_i \sum_k \sum_j (DC)_i (Intake)_k (CF)_{i,k} (Cw)_{i,j} \quad (4)$$

where,

S : annual collective dose of objective group (person Sv/a)

$(DC)_i$: dose coefficient of nuclide i (Sv/Bq)

$(Intake)_k$: annual intake of marine product k (kg/a) (= annual production × intake ratio)

$(CF)_{i,k}$: concentration factor of radionuclide i from seawater to marine product k ; (Bq/kg)/(Bq/L)

$(Cw)_{i,j}$: averaged concentration of nuclide i in surface seawater in the grid j (Bq/L)

Then, the averaged effective dose of the group was calculated by Eq. (5), which meant that the collective dose of the group was divided by the population of the group.

$$E = S / pop \quad (5)$$

where,

E : averaged effective dose of the group (Sv/a)

pop : the population of the group (person)

2.4 Probabilistic risk assessment of intake of marine products (Part B2)

(1) Uncertainty of parameters

In the deterministic assessment described in section 2.3., the risk was evaluated by simple equations and parameters that were fixed in conservative. On the other hand, ICRP's draft of new recommendation [4] suggests using composite distribution. It implies the dose distribution based on the emergence probability of the following parameter;

(i) the uncertainty and natural variability in the estimated environmental media concentration (i.e., radionuclide concentration in air, water, soil, and food) and (ii) uncertainty in the habit data (i.e., breathing rate, food and water ingestion rates, time spent at various activities).

ICRP's draft also considers the age dependency and time- and space- scale of model, but only uncertainty of parameters shown in Table 3 was considered in this study. Everybody was assumed to be adult.

(2) Preparation for input data

(a) Radionuclide concentration in surface seawater

The distribution of radionuclide concentration in surface seawater was calculated by the method described in Part A. But the concentration was not averaged. The distribution was used for the probabilistic risk assessment.

(b) Intake amount of marine products

As for intake amount of marine products, very few references described the distribution form of intake amount. It cost lots of money and energy to research the distribution form. According to Byrom et al.[13], established database suggested that 95th percentile of consumption rates for many staple foods

tended to exceed the mean value of the distribution by approximately a factor of 3. Based on this description, the logarithm distribution, that 3 times of average was equal to 95th percentile, was assumed and applied for the probabilistic risk assessment.

(c) Concentration factor

The concentration factors in IAEA Technical report No.247[9] (Table 2) was conservatively designed. It is possible to use the concentration factors with uncertainty. But very few references described the distribution form of concentration factors. Thus the same concentration factor which was used in deterministic assessment was applied in this study.

(d) Dose coefficient

ICRP's draft of new recommendation defined that dose coefficient had no uncertainty. The same dose coefficient which was used in deterministic assessment (Table 2) was applied in this study.

(3) Calculation of effective dose by the Monte-Carlo method

Probabilistic effective dose from intake of marine products was calculated by multiplying (a)–(d) collected in (2). The distribution of the effective dose was calculated by the Monte-Carlo method that parameters were repeatedly extracted in a random order. As an example of result, the probabilistic effective dose in case that Japanese ate several kinds of marine products caught in the northwestern of Pacific Ocean (Area 61 in Figure 3) in 1997 will be presented in this paper.

3. Results and Discussion

3.1 Averaged dose obtained by the deterministic risk assessment

World averaged dose from the intake of marine products due to the atmospheric nuclear weapon testing is shown in Figure 5. Carbon-14 was the most contributing nuclide, and occupied about 3 quarter of all nuclide, and ^{137}Cs was the second one. The averaged dose had a maximum (0.34 μSv/a) in 1963, then gradually decreased. It was 0.12 μSv/a in 2003.

Japanese averaged dose is shown in Figure 6. It was 3.2 μSv/a in 1963, and 0.41 μSv/a in 2003. Japanese averaged dose was larger than world averaged dose throughout the calculation period.

The reason is considered that the radioactive concentration in surface seawater in the northwestern of Pacific Ocean has relatively high as shown in Figure 4., and that Japanese generally has a lot of marine products by dietary habit.

But the diffusion of radionuclides, and downslide of intake of marine products by westernization of dietary habit made the dose ratio between Japan and world to be about 3 times in 2003, though it was about 10 times in 1963. The cumulative dose for 54 years from 1950 to 2003 was 11 μSv for world average and 98 μSv for Japanese average. The effective dose from intake of marine products was confirmed to be sufficiently less than the dose limit defined by ICRP.

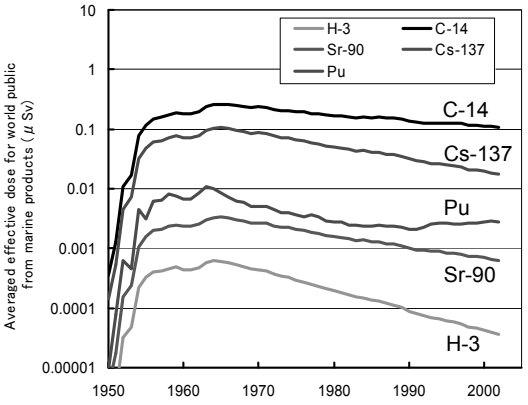

Figure 5 Average effecitve dose for world public from marine products.

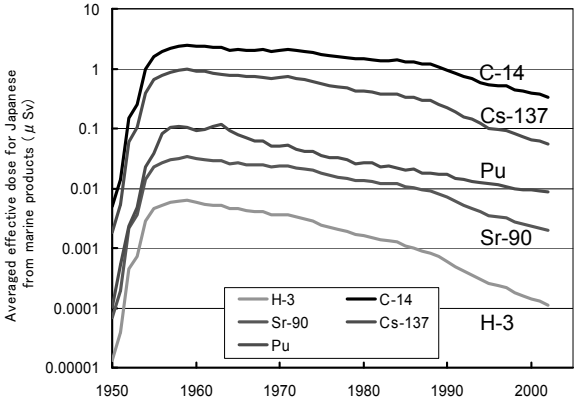

Figure 6 Average effecitve dose for Japanese from marine products.

3.2 Dose distribution obtained by the probabilistic risk assessment

The distribution of Japanese effective dose for ^{137}Cs and for ^{239,240}Pu by the intake of several kinds of marine products caught in the northwestern Pacific Ocean (area 61 in Figure 3) in 1997 is shown in Figure 7.

As for ^{137}Cs, the distribution looked like log-normal, and the total dose of each marine product was 0.022 μSv/a, 0.080 μSv/a and 0.25 μSv/a for 5th, 50 th, 95 th percentile, respectively.

As for ^{239,240}Pu, the distribution did not look like log-normal. Plutonium, which is insoluble element to seawater, is considered to be easily transported downward by scavenging process. The variation of ^{239,240}Pu concentration in the same area (area 61) was larger than that of ^{137}Cs. The total dose of each marine product for ^{239,240}Pu was 0.00021 μSv/a, 0.0089 μSv/a and 0.046 μSv/a for 5th, 50th, 95th percentile, respectively.

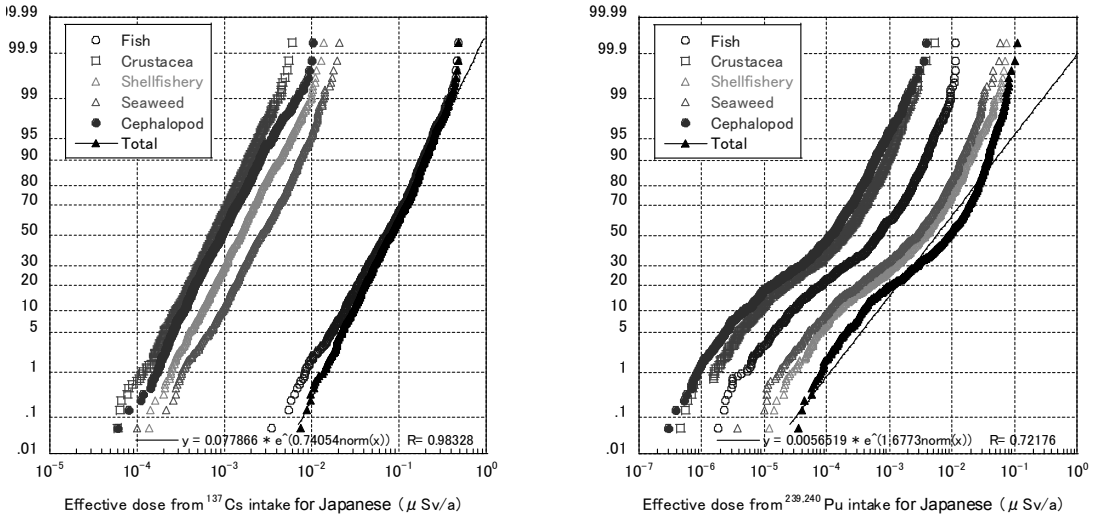

Figure 7 Distribution of the effective dose for Japanese from ^{137}Cs (left) ^{239,240}Pu (right) intake of marine products produced in the northwest Pacific (area 61) during 1997.

3.3 Comparison of deterministic and probabilistic assessment

The result of the deterministic assessment described in section 3.1 showed the averaged figure using the averaged parameters. To compare the result of probabilistic assessment, the deterministic assessment for northwest Pacific (area 61 in Figure 3) in 1997 was performed using the following 95th percentile of the parameters.

^{137}Cs concentration in surface seawater	: 3.6 Bq/m^3 (95th percentile)
^{239,240}Pu concentration in surface seawater	: 12 mBq/m^3 (95th percentile)
Intake of marine products (daily)	: fish 211 g, crustacea 7.1 g, shellfishery: 15.0 g, seaweed: 15.6 g, cephalopod: 27.8 g (95th percentile)
The other parameters	: same with the other methods

As a result, the annual dose from ^{137}Cs and ^{239,240}Pu was 0.39 μSv/a and 0.097 μSv/a, respectively. These values were confirmed to be about double of the 95th percentile of the probabilistic assessment described in section 3.2.

Table 3 comprehensively contains the method and result concerned in this study. The order of the assessed value will be 'averaged value of deterministic < 95th of probabilistic < 95th of deterministic'.

Table 3 Assessment methods and their results for the calculation of the effective dose for Japanese from intake of marine products produced in the northwest Pacific (area 61) during 1997.

	deterministic method (average)	probabilistic method (95th percentile)	deterministic method (95th percentile)
(a) Concentration in seawater	arithmetic average	distribution	95th percentile of distribution
(b) Concentration factor	Constant (Conservative)		
(c) Intake of marine product	geometrical average	lognormal distribution	95th percentile of distribution
(d) Dose coefficient	Constant		
Effective dose for Japanese from intake of marine products produced in the northwest Pacific (area 61) during 1997			
^{137}Cs	0.087 μSv	0.25 μSv	0.39 μSv
^{239,240}Pu	0.011 μSv	0.046 μSv	0.097 μSv

4. Conclusion

In this study, the effective dose from the intake of marine products due to the past atmospheric nuclear tests was calculated using LAMER and discussed.

1) As a result of the proforma calculation, about 3 quarters of the effective dose from the intake of marine products due to the past atmospheric nuclear tests was originated from ^{14}C. The averaged dose for Japanese was very small but about 10 times higher than the world average.
2) The probabilistic assessment suggested by ICRP's draft of new recommendation was applied for Japanese dose from a part of the Pacific Ocean. The order of the assessed value was '(averaged value of deterministic) < (95th of probabilistic) < (95th of deterministic)'.

This method can show the deterministic and probabilistic dose concerned world wide effect due

to the past atmospheric nuclear tests. Also, the practical procedure and control factor of the probabilistic assessment were confirmed in this study. In near future, the deterministic and probabilistic assessment from nuclear fuel cycle facilities will be carried out using the technique developed in this study.

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3-7 Proposed Amendments to the Environmental Radiation Protection Standards for Yucca Mountain, Nevada

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Abstract

In 2001, the United States Environmental Protection Agency (EPA) issued public health and safety standards for the proposed high level waste repository now under construction at Yucca Mountain, Nevada, a site 90 miles west of Las Vegas. In these standards, EPA set a limit of 150 microsieverts (μSv) per year committed effective dose equivalent (CEDE) for the reasonably maximally exposed individual (RMEI) living in the accessible environment near the repository. In July 2004, a Federal court remanded part of the standards to EPA for reconsideration. In response, EPA proposed additional standards in 2005 to protect human health over the anticipated period of geological stability for the repository, i.e. 1 million years. For the period of 10,000 to 1 million years, EPA is proposing a dose limit of 3.5 millisieverts (mSv) per year to the RMEI. In addition, the dose calculation methodology would be updated to an ICRP 60 and 72 basis instead of ICRP 26 and 30.

Keywords: High level waste repository, Spent fuel, Yucca Mountain, Waste management

1. Introduction

The U.S. Environmental Protection Agency (EPA) first issued radiation protection standards for the potential spent nuclear fuel and high-level radioactive waste disposal system in Yucca Mountain, Nevada on 13 June 2001 (the 2001 standards [1]) under the authority of the Energy Policy Act of 1992 (EnPA [2]). (The term "repository" is used in this paper to refer to the mined facility, while the term "disposal system" is used to refer to the entirety of the mined facility, the engineered barriers, and the geologic barrier.) The EnPA also directed EPA to set the standards "based upon and consistent with" the results of a study by the National Academy of Sciences (NAS) "to provide [to EPA]...findings and recommendations on reasonable standards for protection of the public health and safety…." (the NAS Report [3]). The standards are in Part 197 of Title 40 of the Code of Federal Regulations (40 CFR Part 197).

After the standards were issued, petitions for review were filed in Federal courts by the State of Nevada, several environmental and public interest groups led by the Natural Resources Defense Council, and the Nuclear Energy Institute. The standards survived every challenge except one regarding the compliance period. The Court ruled that the 10,000-year compliance period was not

based upon and consistent with a recommendation in the NAS Report [3]. The NAS recommendation was:

> "…there is no scientific reason for limiting the time period of an individual-risk standard in this way [10,000 years]. We believe that compliance assessment is feasible for most physical and geologic aspects of repository performance on the time scale of the long-term stability of the fundamental geologic regimes – a time scale that is on the order of 10^6 years at Yucca Mountain – and that at least some potentially important exposures might not occur until after several hundred thousand years. For these reasons, we recommend that compliance assessment be conducted for the time when the greatest risk occurs, within the limits imposed by long-term stability of the geologic environment." [3]

Notably, NAS also said: "Nevertheless, we note that although the selection of a time period of applicability has scientific elements, it also has policy aspects that we have not addressed. For example, EPA might choose to establish consistent policies for managing risks from disposal of both long-lived hazardous nonradioactive materials and radioactive materials." The Agency's longest-term disposal standards and regulations for both nonradioactive and radioactive hazardous wastes extended only to 10,000 years. Despite EPA's explanations of those factors, the Court ruled that EPA's compliance period for Yucca Mountain was not based upon and consistent with the NAS recommendation and that EPA had not sufficiently justified its decision to set the 10,000-year compliance period on policy grounds.

On 22 August 2005, the Agency proposed amendments to address the Court ruling [4]. The parts of the standards not affecting the extension of the compliance period are not being proposed for change, with the exception of updating the dose methodology. Thus, changes were not proposed to the storage standards, the characteristics of the reasonably maximally exposed individual, and the ground-water protection standards, for example. The comment period ended 21 November 2005. Hearings were held in early October 2005 in Amargosa Valley and Las Vegas, Nevada, and Washington D.C.

In previous papers, EPA has reported the findings and recommendations in the NAS Report, public comments received from the review of the NAS Report, the considerations made while establishing the 2001 standards, and the contents of those standards. This paper discusses the proposed amendments to the 2001 standards.

2. Overview of the 2001 Disposal Standards

Subpart B of 40 CFR Part 197 contains the disposal standards for: (a) protection of individuals; (b) human intrusion; and (c) ground-water protection. The disposal phase is considered to start when the repository is closed. Disposal was the subject of the findings and recommendations of the NAS Report [3].

Individual-protection Standard. The individual-protection standard is 150 μSv (15 millirems; abbreviated as mrem) committed effective dose equivalent (CEDE) per year for 10,000 years after closure. The Agency uses the dose incurred by a reasonably maximally exposed individual (RMEI) to compare with the dose limits. The concept is similar to the critical group approach in that its

purpose is to project doses that are among the highest but still in a reasonably expected range rather than the highest theoretical dose. The location of the RMEI must be assumed to be in the accessible environment above the point of highest concentration of radionuclides in the aquifer. The accessible environment can be no farther down gradient than the southern edge of the Nevada Test Site (NTS), or about 18 kilometers south of the repository.

Ground-water Protection Standards. These standards provide separate protection of ground water. The overall goal is to prevent adverse effects upon human health and the environment by preventing contamination rather than relying upon later mitigation. The limits are the same as the maximum contaminant levels for radionuclides under the Safe Drinking Water Act. The compliance period for these standards is 10,000 years based upon undisturbed performance, i.e., the assumption that the repository is not affected by human intrusion or unlikely features, events, or processes (FEPs).

Human-intrusion Standard. The human-intrusion standard is 150 μSv (15 mrem) CEDE per year for 10,000 years after closure. The required human-intrusion scenario is a single intrusion as a result of exploratory drilling for ground water. The EPA specifies certain borehole parameters that DOE must use to assess the dose received by the RMEI as a result of releases that travel through the borehole, without including the effects of unlikely FEPs. The timing of the intrusion is to be established by NRC based upon the earliest time that current technology and practices could lead to waste package penetration without the drillers noticing it. However, it must not occur sooner than the cessation of active institutional controls. Finally, the standard requires that the human-intrusion analysis be done using the same assumptions and RMEI characteristics as those required for the individual-protection standard.

3. Proposed Amendments to the 2001 Standards

3.1 Scope of the Rulemaking

The rulemaking is limited to those portions of the 2001 standards that were affected by the court ruling, i.e., the compliance period for the individual-protection and the human-intrusion standards and certain supporting items. Even though the ground-water protection standards also have a 10,000-year compliance period of 10,000 years, the Court did not vacate these standards since NAS made no recommendation regarding ground-water standards. Therefore, EPA did not propose changes to the ground-water standards.

The Agency also proposed to update the dose methodology and to revise certain definitions to achieve consistency with the extended compliance period.

3.2 Individual-protection Standard

The Court's decision centered upon the NAS recommendation regarding the compliance period for the individual-protection standard. To address the Court decision, EPA proposed a compliance limit of 3.5 mSv (350 mrem) CEDE/yr to apply for projected performance between 10,000 and 1 millions years. In addition, EPA is retaining the 150 μSv (15 mrem) CEDE/yr standard applicable for the first 10,000 years as established in the 2001 standards.

The Agency believes that the most problematic aspect of extending the compliance period to

peak dose is the uncertainty involved in making projections over such long time frames. Regardless of the level of rigor that can be applied to the technical calculation, it is not possible to place the same level of confidence in performance projections over 10,000 years versus 1 million years.

In addressing how to incorporate extremely long-term projections into a regulatory process and have them be sufficiently reliable to serve as a basis for regulatory decisions, EPA considered guidance and precedents from international and domestic sources. The NAS discussed some technical aspects of uncertainty. For example, NAS stated: "uncertainties in waste canister lifetimes might have a more significant effect on assessing performance in the initial 10,000 years than in performance in the range of 100,000 years." [3] On the other hand, NAS recognized that: "the timing of seismic events is unpredictable." [3] Unfortunately, NAS provided no recommendations on how to deal with such uncertainties, but noted: "No analysis of compliance will ever constitute an absolute proof; the objective instead is a reasonable level of confidence in analyses that indicates whether limits established by the standard will be exceeded." [3] For regulatory compliance within 10,000 years, EPA identified several U.S. regulatory programs as possible precedents, including those for the Waste Isolation Pilot Plant and EPA's underground injection control program, but for a compliance period extending to 1 million years, there are no precedents in U.S. regulation. In response to the Court decision, therefore, important sources for guidance and models for contemplating regulations at such long times were international programs grappling with the same issues. In general, international guidance reinforces two points. The first is that uncertainties generally increase with time. For example, the International Atomic Energy Agency [5], the Nuclear Energy Agency [6], and the Swiss National Cooperative for the Disposal of Radioactive Waste [7] have all concluded that the further into the future projections are made, the greater the uncertainty. The second point is that projections at those longer times cannot be viewed with the same level of confidence as shorter-term projections. As exemplified in statements by IAEA [5], NEA [6], and SSI [8] experts indicate that the uncertainties in quantitative performance projections become so large that the results need to be viewed more as qualitative, rather than quantitative, projections.

A number of international scientific and regulatory bodies and programs suggest natural sources of radioactivity serve as a point of comparison when uncertainties become significant. For example, IAEA has stated that, for time frames extending from about 10,000 to 1 million years, "it may be appropriate to use quantitative and qualitative assessments based on comparisons with natural radioactivity and naturally occurring toxic substances." [9] The IAEA also suggests that "In very long time frames…uncertainties could become much larger and calculated doses may exceed the dose constraint. Comparison of the doses with doses from naturally occurring radionuclides may provide a useful indication of the significance of such cases." [5] Similarly, NEA stated that a key performance indicator could be "comparison with background radiation levels" for times up to just 100,000 years [6].

The proposed rule describes a dose limit – to apply for the period from 10,000 to 1 million years – that will not cause people living near Yucca Mountain to receive a total dose that is more than the natural background radiation which people receive routinely in other parts of the U.S. In order to assess total exposures and derive a dose limit, it is necessary to establish levels of natural background

radiation already experienced in the vicinity of Yucca Mountain. The Agency selected Amargosa Valley as the point of comparison for this analysis since that is where the RMEI will likely live. Combined with the cosmic and terrestrial exposures estimated by DOE, EPA estimated the total annual natural background radiation in Amargosa Valley to be approximately 3.5 mSv (350 mrem) CEDE/yr.

To make the comparison with total exposures, it is also necessary to consider what total exposures provide a reasonable reference point for limiting releases from Yucca Mountain. As noted above, the goal is to ensure that releases from Yucca Mountain will not cause total exposures of the RMEI to exceed natural background levels with which other populations live routinely. The Agency considered several factors in this selection. First, some incremental exposure will be allowed since the standards cannot be expected to reduce natural background exposures. Thus, the reference point would have to have a higher level of background than does the area near Yucca Mountain. Because of the complications in estimating localized background radiation (due primarily to the radon component), statewide averages, which are less uncertain, were examined. Of the States with sufficient data, 32 have average background radiation levels higher than Nevada. The States' characteristics, such as geographic location and population, were then considered. Colorado was selected as a State in the western part of the country that best fit the search criteria – fairly well populated and with characteristics reasonably comparable to Nevada (such as radon potential, surface water/coastal features, or size of major cities). According to population data, Colorado ranks 22nd among all states in total population (Nevada is 35th) [10]. Colorado's average annual background radiation is estimated to be about 7 mSv (700 mrem)/yr [11]. Other States have comparable or higher radon potential and higher background levels with which people live routinely (e.g., background levels in North Dakota, South Dakota, and Iowa, for example, are about 8 mSv (789 mrem)/yr, 10 mSv (963 mrem)/yr, and 8 mSv (784 mrem)/yr, respectively), and might also be used for comparison, but their population and geographic characteristics are much different than Nevada's.

Finally, comparing Colorado's estimated average annual background radiation of 7 mSv (700 mrem) CEDE/yr to the estimate for Amargosa Valley, EPA derived an incremental exposure level of 3.5 mSv (350 mrem) CEDE/yr, which was proposed as the dose limit.

The Agency also considered other possible dose limits to apply out to 1 million years. The first option was 1 mSv (100 mrem) CEDE/yr. This level is based upon international guidance to limit all sources of exposure except natural, accidental, and medical. However, in view of the uncertainties in estimating performance in the very far future, EPA concluded that comparisons with natural background radiation provide a reasonable indication of safety out to 1 million years. As McCombie and Chapman have stated in their authoritative reference on radioactive waste disposal: "There is no logical or ethical reason for trying to provide more protection than the population already has from Earth's natural radiation environment, in which it lives and evolves…it must be recognized that man cannot be expected over infinite times to do much better than nature." [12] The other limit considered was 2 mSv (200 mrem) CEDE/yr. It was derived using an approach that incorporated statewide background levels in all the contiguous States in the U.S. However, EPA concluded that it was most appropriate to use site-specific information related to Amargosa Valley (and the RMEI) rather than generic points of reference. For these reasons, the 3.5 mSv (350 mrem) CEDE/yr dose

limit, including consideration of natural background radiation in Amargosa Valley, was preferred over the other options considered, and was proposed as the regulatory limit.

We recognized that a standard based on variations in natural background radiation would be higher than previous, non-occupational standards in the U.S. In the 2001 rulemaking, the 150 μSv (15 mrem) CEDE/yr dose limit and the 10,000-year compliance period were justified in part because they were consistent with other EPA policies. However, the circumstances in the proposed Yucca Mountain standards – and, in particular, the nature and degree of uncertainty in projecting performance out to 1 million years – are significantly different from the situations addressed under Superfund or any other existing U.S. regulatory program. The approach and the dose limit that EPA proposed for the Yucca Mountain standards are consistent with international guidance on the issue of radioactive waste disposal over extremely long times.

3.3 Human-intrusion Standard

While the Court did not specifically address the human-intrusion standard, the Agency proposed revisions to it to parallel the changes proposed for the individual-protection standard. To do so is consistent with the NAS recommendation that "EPA require that the estimated risk calculated from the assumed intrusion scenario be no greater than the risk limit adopted for the undisturbed-repository case" [3].

The Agency proposed to extend the compliance period from 10,000 to 1 million years and to establish a dose limit of 3.5 mSv (350 mrem) CEDE/yr, which corresponds to the proposed individual-protection dose limit. Other aspects of the human-intrusion standard are unchanged from 2001. The intrusion scenario described in 2001 would still apply because the longer compliance period does not in any way affect the reasoning underlying the selection of this scenario. It remains fully consistent with the NAS conclusion that at Yucca Mountain "there is no scientific basis for estimating the probability of intrusion at far-future times" [3]. Instead, NAS recommended that "the result of the analysis should not be integrated into an assessment of repository performance based on risk, but rather should be considered separately. The purpose of this consequence analysis is to evaluate the resilience of the repository to intrusion" [3].

The intrusion scenario requires consideration of package degradation, premised on the assumption that drillers encountering an intact package would cease drilling and releases would be avoided. We believe that this assumption is equally valid both within and beyond a 10,000-year time frame. In the 2001 standards, DOE was not required to demonstrate compliance with a dose limit if packages did not degrade sufficiently within 10,000 years to permit intrusion (or, in any event, if the consequences of the intrusion were not calculated to occur within 10,000 years). However, the current proposal would require DOE to show compliance with a dose limit regardless of when the consequences of the intrusion occur (within 1 million years). Overall, this scenario continues to represent a reasonable test that "can provide useful insight into the degree to which the ability of a repository to protect public health would be degraded by intrusion" [3].

3.4 Dose Methodology

In 1977 and 1979, ICRP published Reports 26 [13] and 30 [14], respectively. These two reports reflected advances in the state of knowledge of radionuclide dosimetry and biological transport of radionuclides in humans that occurred over the 20 years since ICRP's 1957 dose methodology recommendation (ICRP 2) [15]. The new methodology was called the effective dose equivalent (EDE).

The 2001 standards required DOE to calculate annual doses (as CEDE) to demonstrate compliance with the storage, individual-protection, and human-intrusion standards. The Agency proposed to modify that requirement to incorporate updated scientific factors necessary for the calculation, but would not change the underlying methodology. Specifically, EPA proposed to require DOE to calculate the annual CEDE using the radiation- and organ-weighting factors in ICRP Publications 60 [16] and 72 [17], rather than those in ICRP Publications 26 [13] and 30 [14].

These ICRP factors represent the most recent science and dose calculation approaches in the area of radiation protection. The EPA believes that it is reasonable and desirable to conform the standards to the most recent method approved by the U.S. and international radiation-protection community. The Agency also proposed an updating mechanism since repository closure and license termination may be decades or even more than one hundred years into the future. Therefore, EPA would allow DOE to use, with NRC approval, further updated dose calculation factors in the future, but only if those factors have been appropriately reviewed and accepted by the scientific community and issued by independent scientific bodies (such as ICRP and its successor bodies) and incorporated by EPA into its Federal Guidance.

3.5 Judging Compliance

Under 40 CFR Part 197, EPA requires DOE to complete a probabilistic performance assessment to demonstrate compliance with the individual-protection standard. The results will be a distribution of projected doses since the analysis contains parameters with a range of values, incorporates uncertainties in the models, and uses various expert-judgment assumptions. In 2001, EPA specified the mean of the distribution as the metric to be used for comparison with the standard. In 2005, EPA proposed to retain the mean as the compliance measure for the first 10,000 years. In the unlikely event that the peak dose is found to occur within the first 10,000 years, the mean would be consistent with the statistical measure used in other applications for geologic disposal, i.e., 40 CFR parts 191 and 194 for the 10,000-year compliance period. However, for the period from 10,000 to 1 million years, the Agency believes that the compliance measure should be examined separately to determine if there is a more appropriate measure.

There are significant uncertainties in predicting when discrete events, such as seismic activity, will occur and the effects of these events. Some scenarios incorporating these uncertainties would represent unlikely behavior in that they could show extremely poor or extremely good performance. Such low-probability situations should not be ignored in compliance decisions, but they should not be given undue influence in judging compliance. The NAS stated: "The challenge is to define a standard that specifies a high level of protection but that does not rule out an adequately sited and

well-designed repository because of highly improbable events." [3] The Agency concluded that for the longer compliance period, there should be a measure that represents the "central tendency" in the distribution. Therefore, the compliance measure should represent a central measure that is not strongly affected by extreme input and results.

A difficulty with the mean is that when the bases of the calculations are excessively conservative (or non-conservative), the results suggest that the "most likely" dose is higher (or lower) than if a more reasonable and realistic approach were taken. Therefore, we believe that a regulatory performance measure should not give undue emphasis to high-end or low-end projections which the mean could do.

On the other hand, the median is less affected by the extremes of the distribution and the attendant uncertainty about how close the mean is to the center of the distribution is removed. In this respect, the median is an attractive alternative to the mean as a measure of central tendency since it is not as strongly influenced by high or low-end outliers. Therefore, EPA proposed to use the median for the post-10,000-year compliance period.

3.6 Features, Events, and Processes

The overall purpose of the performance assessment is to provide a reasonable test for compliance with the standards. A major part of providing that reasonable test is determining which features, events, and processes (FEPs) are to be included in the performance assessment. Key to this consideration is EPA's goal of setting standards that provide for a reasonable test of the disposal system under a range of conditions that represent the expected case, as well as relatively less likely (but not wholly speculative) scenarios with potentially significant consequences. As a result, it is neither constructive nor necessary for EPA to require DOE to predict or model every conceivable scenario that could occur at Yucca Mountain.

This implies that some FEPs (or series of FEPs) need not be included in the performance assessment because their probability of occurrence is extremely low. As a means of restricting scenarios, in the 2001 standards, the Agency outlined how to screen FEPs. Without such measures, the list of FEPs would be limitless, bounded only by the imagination. The Agency determined that FEPs that could occur with a probability equal to or greater than 1 in 10,000 over a period of 10,000 years, an annual probability of occurrence of 10^{-8}, would be sufficiently likely to occur that they should be included among the FEPs available for selection in any particular scenario. Any FEPs with lower probabilities could be excluded from the performance assessment.

For the 10,000-year to 1-million-year compliance period, we considered how to address this probability cutoff. If, for example, we required consideration of events with a probability of occurrence of 10^{-4} over 1 million years, an approach that has been suggested by some stakeholders, it would equate to an annual probability of 10^{-10}, which encompasses events nearly as remote as the "Big Bang" that created the Universe. No disposal system, and perhaps not even the Earth, would survive the effects of such an event, and, therefore, EPA did not find such FEPs to be useful indicators to distinguish between safe or unsafe performance of the disposal system. In the end, the Agency proposed to retain the screening criterion without change – except as described below. However,

certain scenarios merit special considerations at extremely long times (beyond 10,000 years)

The Agency also considered what scenarios should be included in the performance assessment. In formulating our approach to the extended compliance period, we began by reviewing the NAS Report. The NAS concluded that volcanism, seismic activity, and climate change have the potential to significantly modify the properties of the repository and the processes by which radionuclides are transported. The NAS also concluded that the probabilities and consequences of modifications generated by volcanism, seismic activity, and climate change are sufficiently boundable that they should be included (along with an undisturbed scenario) in performance assessments that extend over 1 million years. Thus, EPA proposed to include igneous, seismic, and climate change scenarios and have DOE assess the most likely and significant impacts, with appropriate variability incorporated, on dose projections.

Having identified particular natural FEPs, the Agency considered whether there are FEPs that could significantly affect the engineered barrier system that had not been identified for the 10,000-year compliance period. After reviewing DOE's published assessments and other relevant information, the Agency concluded that general corrosion of the waste packages could be a significant failure mechanism at times in the hundreds of thousands of years [18]. Unlike certain other corrosion processes which may be likelier or faster-acting at earlier times, general corrosion may not be a significant factor within 10,000 years and could potentially be removed from consideration at those times because of its limited consequence. This is a situation that EPA found inappropriate and proposed that DOE must project the effects of general corrosion throughout the compliance period.

4. Status and Future Steps

The EPA published the proposed amendments to 40 CFR Part 197 in the 22 August 2005 *Federal Register* [19]. A public comment period was open from then until 21 November 2005. Public hearings were held in Amargosa Valley, Nevada; Las Vegas, Nevada; and Washington, DC. Approximately 2500 comment messages were received.

The Agency does not have a schedule to publish its final amendments. The Agency is considering the comments and will publish its response-to-comments document and the final versions of its technical support documents when the final amendments are published.

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3-8 Status of Decommissioning and Waste Management in the Nuclear Science Research Institute of JAEA

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Abstract

The Nuclear Science Research Institute (NSRI) of JAEA has some experiences of the decommissioning of research reactors and research laboratories including a reprocessing test facility. In order to dismantle those facilities safely, we paid much attention for the radiological protection of radiation workers taking into consideration of characteristics of each facility, especially to protect internal exposures. As the results of decommissioning activities, several thousands tons of solid radioactive wastes were generated. In the near future, we will start the treatment of these stored wastes by a super compactor, metal melting furnace and non-metal waste melting furnace to gain high volume reduction and to prepare stable waste forms for final disposal. In Japan, the clearance system was established in 2005 by amending the Nuclear Regulatory Law. The NSRI plans to release very slightly contaminated concrete debris for recycling, which was generated from the replacement of reactor core of research reactor (JRR-3), according to the clearance system.

Keywords: Decommissioning, Waste management, JAEA, Treatment, Disposal, Clearance, JPDR, JRR-2, JRR-3, JRTF

1. Introduction

In the Nuclear Science Research Institute (NSRI) of JAEA, the decommissioning of the JPDR (BWR, 12.5 MWe), which was the first power demonstration reactor in Japan, was carried out successfully from 1986 to 1996. It showed that nuclear power plants can be dismantled safely. After that we have some experiences of the decommissioning of research reactors and research laboratories including a reprocessing test facility (JRTF). The decommissioning programs for those facilities are planned to reduce the radiation effects to workers and the general public according to the ALARA concept. And also the decommissioning procedures and techniques are selected to minimize the generation of radioactive waste.

The generated radioactive wastes are safely treated and stored at waste treatment facilities in the NSRI according to their physical characteristics, radiation levels, activity concentration levels, and the kinds of radionulides. The NSRI also starts the operation of high volume reduction facility for solid radioactive wastes except for combustible waste in the near future. The release of very slightly contaminated concrete debris for recycling is planned according to the clearance system which was

established in 2005.

In this paper, the main outcomes of the decommissioning activities for a research reactor (JRR-2) and the JRTF, and the present status and future plan of waste management are presented.

2. Decommissioning Activities after Decommissioning of the JPDR

2.1 Decommissioning of the JRR-2[1, 2)]

Japan Research Reactor No.2 (JRR-2), which is a heavy water moderated and cooled tank type research reactor with maximum thermal power of 10 MW, was operated for over 36 years. It was retired from its operation in Dec., 1996. In 1997, the decommissioning plan was submitted to the Science and Technology Agency (present Ministry of Education, Culture, Sports, Science and Technology, MEXT) and then the dismantling activities started. The decommissioning plan is consisted of four phases. Figure 1 shows the outline of the decommissioning program. The 1st, 2nd and 3rd phases have already finished successfully. At present time, the JRR-2 is under the safe storage. The safe storage period continues until the start of operation of a repository which is under the site selection stage, and the decommissioning activities will resume after that.

Phase	F.Y. / Activities	'97	'98	'99	'00	'01	'02	'03	'04～	
Phase 1	Removal of CRDM									
	Drain and storage of D_2O				Transportstion of D_2O to Canada					
Phase 2	Inventory measurement of reactor									
	Isolation of cooling sysvem									
	Removal of Experiment faicility									
	Removal of secondary cooling system									
	Sealing of reactor									
Phase 3	Decontamination test of ^3H									
	Removal of spent fuel storage facility									
	Removal of stack gas monitor									
	Removal of reactor cooling sysvtem									
	Removal of fresh fuel storage facility									
	Removal of reactor control system									
Phase 4	Dismantlement of ractor *								Safe storage	

Figure 1 Decommissioning schedule of the JRR-2

The decommissioning activities have been carried out taking into consideration of internal exposures of workers due to ^3H inhalation. So far, internal exposures of workers were not detected. Table 1 shows the external exposure doses of workers at each phase. The actual individual exposure doses are smaller than planned doses for each phase and of course smaller than the annual dose limit for workers (100 mSv/5 years).

Table 1 External exposure dose of workers who worked for the dismantlement of the JRR-2

Phase	Number of workers (person)	Collective dose (man-mSv)	Average individual dose (mSv)	Maximum individual dose (mSv)
Phase 1	47	9.2	0.20	1.2
Phase 2	117	12.2	0.10	2.2
Phase 3	222	4.6	0.02	0.7
Total	386	26.0	Av. 0.07	-

2.2 Decommissioning of the JRTF*

The Japan Atomic Energy Agency Reprocessing Test Facility (JRTF) was the first reprocessing facility constructed in Japan, and it was operated during 1968 to 1969 to develop basic technologies for reprocessing using spent fuels from the Japan Research Reactor 3 (JRR-3). After closing the facility, the JRTF decommissioning program was started in 1990 to develop decommissioning technologies and to obtain experiences and data for to dismantle nuclear fuel facilities. This project has been carried out under the contract with the MEXT.

Table 2 shows the major specifications of the JRTF. This project consists of three phases; Phase 1 was the study for dismantling of the JRTF and the treatment of liquid waste stored in the facility, Phase 2 was the research and development of decommissioning technologies for dismantling of the JRTF, and Phase 3 is the actual dismantling of the JRTF by using the decommissioning technologies developed through Phase 2. In Phase 3 the data (e.g. number of workers, exposure dose, quantities of radioactive waste) are collected systematically to characterize the dismantling activities of nuclear fuel facilities.

Table 2 Major specifications of the JRTF

Name of building	Total floor area (m^2)	Major components
Main building	ca. 3,000	hot-cave, the plutonium purification cell (Pu cell), the solvent recovery cell, and 11 lead cells for analysis
Annex building A	ca. 160	12 tanks that store low-level liquid radioactive wastes generated from the reprocessing tests
Annex building B	ca. 400	7 tanks that store liquid wastes such as aluminum decladding liquid waste, alpha contaminated liquid waste and high-level liquid waste generated from reprocessing tests

* This study was carried out from 1990 to 2005 by the JAEA as a funded research of "Dismantling Technologies Development for Reprocessing Plant" under a contract with the Ministry of Education, Culture, Sports, Science and Technology (MEXT).

Phases 1 and 2 have already finished successfully and Phase 3 started in 1996. So far, glove boxes, analytical cells, large size vessels (LV-3~LV-6) and the components in cells such as hot cave, solvent recovery cell, Pu cell, pump cell etc. were dismantled. These works have been carried out taking into consideration of the radiological protection, especially for the internal exposures of workers due to alpha-ray emitting radionuclides such as Pu. Figure 2 shows a worker wearing air-line suit who worked to dismantle the contaminated objects with alpha-ray emitting radionuclides.

Figure 2 Dismantling works by a worker wearing air-line suit

As the end of March, 2006, total labor days are 50,243 man-day for 10 years, and collective doses of workers are 73.7 man-mSv. Large parts of external exposures for workers were resulted from the activities to dismantle liquid waste treatment equipment (collective doses: 34.7 man-mSv and maximum individual dose: 4.4 mSv) and a hot cave (collective doses: 15.9 man-mSv and maximum individual dose: 2.0 mSv). No internal exposure of workers was detected during all completed works.

The large size vessel LV-2 in the Annex Building B will be dismantled during 2006 and 2007, and the another vessel (LV-1) will be dismantled in next two years. The decommissioning of the JRTF will be finished in 2014.

2.3 Future decommissioning activities

In our plan, by the end of March, 2010, one critical assembly (VHTRC: Very High Temperature Reactor Critical Assembly) and five research laboratories, where the R&D activities of nuclear fuel materials were carried out, will be dismantled. And after that the decommissioning activities will continue to dismantle old research laboratories and waste treatment facilities in the NSRI.

3. Waste Management in the NSRI

As the results of the decommissioning activities of the JPDR, the JRR-2 and the JRTF, about 3,770 tons, 421 tons and 347 tons of solid radioactive wastes were generated, respectively. At present time, all solid radioactive wastes, except for the very low-level radioactive waste which was generated from the dismantling of the JPDR and disposed of at the near surface disposal facility to demonstrate the safety of trench disposal method[3], are treated and stored on the site. In the near future, we will start the treatment of these stored wastes by a super compactor, metal melting furnace and non-metal waste melting furnace to gain high volume reduction and to prepare stable waste forms for final disposal[4].

Compressible metal wastes from research reactors are treated by the super compactor. This compaction system consists of the diameter reduction unit to compact the 200-liter drums in the diameter direction with 520 ton force and the high-pressure compaction unit to compact the 200-liter drums in the direction of height with 2,000 ton force. Compacted pellets are filled into new 200-liter drums as close as possible to the limit of the height. Metal wastes except for compressible wastes are treated by the metal melting unit of an induction furnace. After melting, the metal wastes are cast to receptacles, which are used at the non-metal melting unit, by centrifugal casting apparatus, or cast to ingots. Noncombustible wastes such as vinyl chlorides are incinerated by the incinerator first, so as to reduce a burden on the off gas cleaning system of the unit, and generated ash and incombustible wastes such as concrete and glass are melted by the plasma melting furnace. The major specifications of these waste treatment equipments are shown in Table 3.

Table 3 Major specifications of advanced high volume reduction facilities

Equipment	Specifications	Volume reduction ratio	Capacity
Super compactor	Compaction force: 2,000 ton	ca. one-third	50 drums (200 liter)/day
Metal melting furnace	Electrical power: Induction furnace 1,200 kW	ca. one-sixth	4 ton/day, 4 ton/batch
Non-metal waste melting furnace	Electrical power: 2 plasma torches, 1,000 kW each	ca. one-third	4 ton/day, 2 ton/batch

4. Plan of Clearance

In Japan, the clearance system was established in 2005 by amending the Nuclear Regulatory Law. The clearance levels are based on the recommended values by the IAEA Safety Guide RS-G1.7[5]. Table 4 show the clearance levels for major radionuclides in Japan.

The basic flow of Japanese clearance system is shown in Figure 3. The operators who wish to release contaminated materials have the main responsibility to demonstrate the compliance with the clearance levels. On the first stage, the operators evaluate the activity levels and amounts of candidate materials and collect the information to perform the monitoring adequately. And they must submit a report how to perform the monitoring of the clearance to the regulatory authority (MEXT). The

regulatory authority checks the adequacy of the monitoring program. If they review the adequacy of the operator's monitoring program, they should approve the monitoring program. On the second stage, the operators must submit the documents of monitoring results to the regulatory authority. The authority verifies the monitoring results based on the submitted report and also they measure the activity levels of samples if necessary.

Table 4 Clearance levels for major radionuclides in Japan

Kinds of radionuclide	Clearance levels (Bq/g)
^{129}I	0.01
^{46}Sc, ^{54}Mn, ^{60}Co, ^{65}Zn, ^{94}Nb, ^{106}Ru, ^{108m}Ag, ^{110m}Ag, ^{134}Cs, ^{137}Cs, ^{133}Ba, ^{152}Eu, ^{154}Eu, ^{239}Pu, ^{241}Am	0.1
^{14}C, ^{36}Cl, ^{59}Fe, ^{58}Co, ^{90}Sr, ^{95}Nb, ^{99}Tc	1
^{241}Pu	10
^{3}H, ^{41}Ca, ^{59}Ni, ^{63}Ni	100
^{55}Fe	1,000

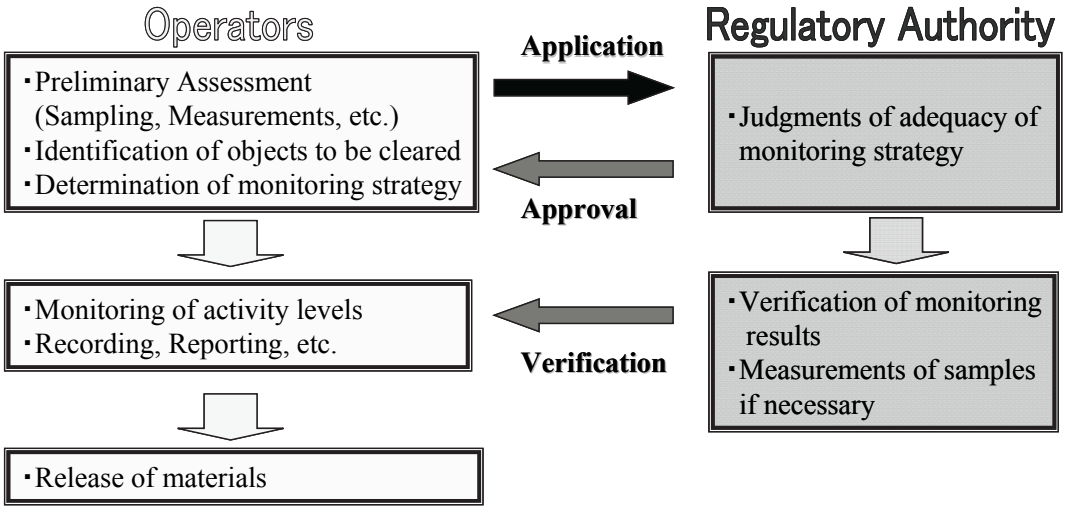

Figure 3 Overview of clearance system in Japan

The NSRI plans to release very slightly contaminated concrete debris (ca. 4,000 tons) for recycling, which was generated from the replacement of reactor core of research reactor (JRR-3), according to the above mentioned clearance system.

5. Conclusion

The NSRI has some experiences of decommissioning of reactors and research laboratories including the test reprocessing plant, and experiences of managing waste arising from the decommissioning of nuclear facilities. These experiences are very useful for the future decommissioning of commercial nuclear power plants and nuclear fuel cycle facilities.

Acknowledgement

We highly appreciate for the sponsorship by the MEXT for R&D activities at the JRTF.

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Appendix 1 The 4th JAEA-US EPA Workshop on Radiation Risk Assessment: Program

Tuesday, November 7, 2006

9:15- Registration

9:45-10:00 Opening addresses
Osamu Oyamada, Japan Atomic Energy Agency
Michael Boyd, United States Environmental Protection Agency

Session 1: Radiation Effects and Radiation Risk Assessment
Session chairs: Miroslav Pinak (JAEA) and David Pawel (US EPA)

10:00-10:45 1-1 Research on Radiation Effect and Radiation Protection at JAEA (Keynote)
K. Saito

10:45-11:30 1-2 BEIR VII: What's Old, What's New, and What Challenges Remain? (Keynote)
E. Douple and R. Jostes

11:30-12:00 1-3 Bystander Effect Studies using Heavy-ion Microbeam
Y. Kobayashi, T. Funayama, T. Sakashita, Y. Furusawa, S. Wada, Y. Yokota, T. Kakizaki, N. Hamada, T. Hara, K. Fukamoto, M. Suzuki and M. Ni

12:00-13:00 Lunch

13:00-13:30 1-4 Modifying EPA Radiation Risk Models Based on BEIR VII
D. Pawel and J. Puskin

13:30-14:00 1-5 Molecular Dynamics Simulation of DNA Strand Breaks
J. Kotulic Bunta, M. Pinak, T. Nemoto, M. Higuchi and K. Saito

14:00-14:30 1-6 ORNL's DCAL Software Package
K.F. Eckerman

14:30-15:00 1-7 Simulation Analysis of Radiation Fields inside Phantoms for Neutron Irradiation
D. Satoh, F. Takahashi, A. Endo, Y. Ohmachi and N. Miyahara

15:00-15:15 Break

Session 2 : Radiation Dosimetry
Session chairs: Keith Eckerman (ORNL) and Akira Endo (JAEA)

15:15-16:00 2-1 ICRP New Recommendations: Committee 2's Efforts (Keynote)
K.F. Eckerman

16:00-16:30 2-2 Development of Nuclear Decay Data for Radiation Dosimetry Calculation
A. Endo and K.F. Eckerman

16:30-17:00 2-3 Application of the PHITS Code in High-Energy Particle Dosimetry
T. Sato, A. Endo and K. Niita

17:00-17:30 2-4 Development of Japanese Voxel Models and Their Application to Organ Dose Calculation
K. Sato, A. Endo and K. Saito

18:30-20:00 Reception

Wednesday, November 8, 2006

Session 2: Radiation Dosimetry – continued

9:15-9:45 2-5 The United States Transuranium and Uranium Registries (USTUR): Learning from Plutonium and Uranium Workers
A.C. James and B.G. Brooks

9:45-10:15 2-6 Retrospective Dosimetry of an Accidental Intake Case of Radioruthenium-106 at the Tokai Reprocessing Plant
O. Kurihara, K. Kanai, C. Takada, K. Ito, T. Momose and K. Miyabe

10:15-10:45 2-7 Strategy on Quality Assurance in Radiation Fields and Calibration Techniques at FRS of JAEA
M. Tsutsumi

10:45-11:00 Break

Session 3 : Emergency Response, Radiation Protection Standards and Waste Management
Session chairs: Yasuhiro Yamaguchi (JAEA) and Michael Boyd (US EPA)

11:00-11:30 3-1 Current Emergency Programs for Nuclear Installations in Japan
M. Chino

11:30-12:00 3-2 Revision of the Protective Action Guides Manual for Nuclear Incidents
S. DeCair

12:00-13:00 Lunch

13:00-13:30 3-3 Some Aspects in the Compliance with the Japanese Radiation Protection Regulations
H. Yamamoto and S. Mizushita

13:30-14:00 3-4 The Latest Occupational Radiation Exposure Data from U.S. Nuclear Regulatory Commission Licensees
T. Brock

14:00-14:30 3-5 Discussion on Concepts for Radiological Dosimetric Quantities in the Japan Health Physics Society
F. Takahashi and K. Oda

14:30-15:00 3-6 Study on the Estimation of Probabilistic Effective Dose: Committed Effective Dose from Intake of Marine Products using Oceanic General Circulation Model
M. Nakano

15:00-15:20 Break

15:20-15:50 3-7 Updates to EPA's Yucca Mountain Rule: The Post-10,000 Year Standard
M. Boyd

15:50-16:20 3-8 Status of Decommissioning and Waste Management in the Nuclear Science Research Institute of JAEA
M. Okoshi and T. Yamashita

16:20-16:30 Closing address
Masamichi Chino, Japan Atomic Energy Agency

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国際単位系（SI）

表１．SI 基本単位

基本量	SI 基本単位	
	名称	記号
長さ	メートル	m
質量	キログラム	kg
時間	秒	s
電流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光度	カンデラ	cd

表２．基本単位を用いて表されるSI組立単位の例

組立量	SI 基本単位	
	名称	記号
面積	平方メートル	m^2
体積	立法メートル	m^3
速さ，速度	メートル毎秒	m/s
加速度	メートル毎秒毎秒	m/s^2
波数	毎メートル	m^{-1}
密度（質量密度）	キログラム毎立法メートル	kg/m^3
質量体積（比体積）	立法メートル毎キログラム	m^3/kg
電流密度	アンペア毎平方メートル	A/m^2
磁界の強さ	アンペア毎メートル	A/m
（物質量の）濃度	モル毎立方メートル	mol/m^3
輝度	カンデラ毎平方メートル	cd/m^2
屈折率	（数の） 1	1

表５．SI 接頭語

乗数	接頭語	記号	乗数	接頭語	記号
10^{24}	ヨタ	Y	10^{-1}	デシ	d
10^{21}	ゼタ	Z	10^{-2}	センチ	c
10^{18}	エクサ	E	10^{-3}	ミリ	m
10^{15}	ペタ	P	10^{-6}	マイクロ	μ
10^{12}	テラ	T	10^{-9}	ナノ	n
10^{9}	ギガ	G	10^{-12}	ピコ	p
10^{6}	メガ	M	10^{-15}	フェムト	f
10^{3}	キロ	k	10^{-18}	アト	a
10^{2}	ヘクト	h	10^{-21}	ゼプト	z
10^{1}	デカ	da	10^{-24}	ヨクト	y

表３．固有の名称とその独自の記号で表されるSI組立単位

組立量	SI 組立単位			
	名称	記号	他のSI単位による表し方	SI基本単位による表し方
平面角	ラジアン(a)	rad		$m \cdot m^{-1}=1$(b)
立体角	ステラジアン(a)	sr(c)		$m^2 \cdot m^{-2}=1$(b)
周波数	ヘルツ	Hz		s^{-1}
力	ニュートン	N		$m \cdot kg \cdot s^{-2}$
圧力，応力	パスカル	Pa	N/m^2	$m^{-1} \cdot kg \cdot s^{-2}$
エネルギー，仕事，熱量	ジュール	J	N・m	$m^2 \cdot kg \cdot s^{-2}$
工率，放射束	ワット	W	J/s	$m^2 \cdot kg \cdot s^{-3}$
電荷，電気量	クーロン	C		s・A
電位差（電圧），起電力	ボルト	V	W/A	$m^2 \cdot kg \cdot s^{-3} \cdot A^{-1}$
静電容量	ファラド	F	C/V	$m^{-2} \cdot kg^{-1} \cdot s^4 \cdot A^2$
電気抵抗	オーム	Ω	V/A	$m^2 \cdot kg \cdot s^{-3} \cdot A^{-2}$
コンダクタンス	ジーメンス	S	A/V	$m^{-2} \cdot kg^{-1} \cdot s^3 \cdot A^2$
磁束	ウエーバ	Wb	V・s	$m^2 \cdot kg \cdot s^{-2} \cdot A^{-1}$
磁束密度	テスラ	T	Wb/m^2	$kg \cdot s^{-2} \cdot A^{-1}$
インダクタンス	ヘンリー	H	Wb/A	$m^2 \cdot kg \cdot s^{-2} \cdot A^{-2}$
セルシウス温度	セルシウス度(d)	℃		K
光束	ルーメン	lm	cd・sr(c)	$m^2 \cdot m^{-2} \cdot cd=cd$
照度	ルクス	lx	lm/m^2	$m^2 \cdot m^{-4} \cdot cd=m^{-2} \cdot cd$
（放射性核種の）放射能	ベクレル	Bq		s^{-1}
吸収線量，質量エネルギー分与，カーマ	グレイ	Gy	J/kg	$m^2 \cdot s^{-2}$
線量当量，周辺線量当量，方向性線量当量，個人線量当量，組織線量当	シーベルト	Sv	J/kg	$m^2 \cdot s^{-2}$

(a)ラジアン及びステラジアンの使用は、同じ次元であっても異なった性質をもった量を区別するときの組立単位の表し方として利点がある。組立単位を形作るときのいくつかの用例は表４に示されている。
(b)実際には、使用する時には記号rad及びsrが用いられるが、習慣として組立単位としての記号“１”は明示されない。
(c)測光学では、ステラジアンの名称と記号srを単位の表し方の中にそのまま維持している。
(d)この単位は、例としてミリセルシウス度m℃のようにSI接頭語を伴って用いても良い。

表４．単位の中に固有の名称とその独自の記号を含むSI組立単位の例

組立量	SI 組立単位		
	名称	記号	SI 基本単位による表し方
粘度	パスカル秒	Pa・s	$m^{-1} \cdot kg \cdot s^{-1}$
力のモーメント	ニュートンメートル	N・m	$m^2 \cdot kg \cdot s^{-2}$
表面張力	ニュートン毎メートル	N/m	$kg \cdot s^{-2}$
角速度	ラジアン毎秒	rad/s	$m \cdot m^{-1} \cdot s^{-1}=s^{-1}$
角加速度	ラジアン毎平方秒	rad/s^2	$m \cdot m^{-1} \cdot s^{-2}=s^{-2}$
熱流密度，放射照度	ワット毎平方メートル	W/m^2	$kg \cdot s^{-3}$
熱容量，エントロピー	ジュール毎ケルビン	J/K	$m^2 \cdot kg \cdot s^{-2} \cdot K^{-1}$
質量熱容量（比熱容量），質量エントロピー	ジュール毎キログラム毎ケルビン	J/(kg・K)	$m^2 \cdot s^{-2} \cdot K^{-1}$
質量エネルギー（比エネルギー）	ジュール毎キログラム	J/kg	$m^2 \cdot s^{-2} \cdot K^{-1}$
熱伝導率	ワット毎メートル毎ケルビン	W/(m・K)	$m \cdot kg \cdot s^{-3} \cdot K^{-1}$
体積エネルギー	ジュール毎立方メートル	J/m^3	$m^{-1} \cdot kg \cdot s^{-2}$
電界の強さ	ボルト毎メートル	V/m	$m \cdot kg \cdot s^{-3} \cdot A^{-1}$
体積電荷	クーロン毎立方メートル	C/m^3	$m^{-3} \cdot s \cdot A$
電気変位	クーロン毎平方メートル	C/m^2	$m^{-2} \cdot s \cdot A$
誘電率	ファラド毎メートル	F/m	$m^{-3} \cdot kg^{-1} \cdot s^4 \cdot A^2$
透磁率	ヘンリー毎メートル	H/m	$m \cdot kg \cdot s^{-2} \cdot A^{-2}$
モルエネルギー	ジュール毎モル	J/mol	$m^2 \cdot kg \cdot s^{-2} \cdot mol^{-1}$
モルエントロピー，モル熱容量	ジュール毎モル毎ケルビン	J/(mol・K)	$m^2 \cdot kg \cdot s^{-2} \cdot K^{-1} \cdot mol^{-1}$
照射線量（X線及びγ線）	クーロン毎キログラム	C/kg	$kg^{-1} \cdot s \cdot A$
吸収線量率	グレイ毎秒	Gy/s	$m^2 \cdot s^{-3}$
放射強度	ワット毎ステラジアン	W/sr	$m^4 \cdot m^{-2} \cdot kg \cdot s^{-3}=m^2 \cdot kg \cdot s^{-3}$
放射輝度	ワット毎平方メートル毎ステラジアン	$W/(m^2 \cdot sr)$	$m^2 \cdot m^{-2} \cdot kg \cdot s^{-3}=kg \cdot s^{-3}$

表６．国際単位系と併用されるが国際単位系に属さない単位

名称	記号	SI 単位による値
分	min	1 min=60s
時	h	1h =60 min=3600 s
日	d	1 d=24 h=86400 s
度	°	1°=(π/180) rad
分	’	1’=(1/60)°=(π/10800) rad
秒	”	1”=(1/60)’=(π/648000) rad
リットル	l、L	1l=1 $dm^3=10^{-3}m^3$
トン	t	$1t=10^3$ kg
ネーパ	Np	1Np=1
ベル	B	1B=(1/2)ln10(Np)

表７．国際単位系と併用されこれに属さない単位でSI単位で表される数値が実験的に得られるもの

名称	記号	SI 単位であらわされる数値
電子ボルト	eV	$1eV=1.60217733(49) \times 10^{-19}J$
統一原子質量単位	u	$1u=1.6605402(10) \times 10^{-27}kg$
天文単位	ua	$1ua=1.49597870691(30) \times 10^{11}m$

表８．国際単位系に属さないが国際単位系と併用されるその他の単位

名称	記号	SI 単位であらわされる数値
海里		1海里=1852m
ノット		1ノット=1海里毎時=(1852/3600)m/s
アール	a	1 a=1 $dam^2=10^2m^2$
ヘクタール	ha	1 ha=1 $hm^2=10^4m^2$
バール	bar	1 bar=0.1MPa=100kPa=1000hPa=10^5Pa
オングストローム	Å	1 Å=0.1nm=10^{-10}m
バーン	b	1 b=100$fm^2=10^{-28}m^2$

表９．固有の名称を含むCGS組立単位

名称	記号	SI 単位であらわされる数値
エルグ	erg	1 erg=10^{-7} J
ダイン	dyn	1 dyn=10^{-5}N
ポアズ	P	1 P=1 dyn・s/cm^2=0.1Pa・s
ストークス	St	1 St =$1cm^2/s=10^{-4}m^2/s$
ガウス	G	1 G ^10^{-4}T
エルステッド	Oe	1 Oe ^(1000/4π)A/m
マクスウェル	Mx	1 Mx ^10^{-8}Wb
スチルブ	sb	1 sb =$1cd/cm^2=10^4cd/m^2$
ホト	ph	1 ph=10^4lx
ガル	Gal	1 Gal =$1cm/s^2=10^{-2}m/s^2$

表10．国際単位に属さないその他の単位の例

名称	記号	SI 単位であらわされる数値
キュリー	Ci	1 Ci=3.7×10^{10}Bq
レントゲン	R	1 R = 2.58×10^{-4}C/kg
ラド	rad	1 rad=1cGy=10^{-2}Gy
レム	rem	1 rem=1 cSv=10^{-2}Sv
X線単位		1X unit=1.002×10^{-4}nm
ガンマ	γ	1γ=1 nT=10^{-9}T
ジャンスキー	Jy	1 Jy=$10^{-26}W \cdot m^{-2} \cdot Hz^{-1}$
フェルミ		1 fermi=1 fm=10^{-15}m
メートル系カラット		1 metric carat = 200 mg = 2×10^{-4}kg
トル	Torr	1 Torr = (101 325/760) Pa
標準大気圧	atm	1 atm = 101 325 Pa
カロリー	cal	
ミクロン	μ	1 μ =1μm=10^{-6}m

（第７版，1998年改訂）