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Radiation Risk Assessment

November 7-8, 2006

Nuclear Science Research Institute,

Tokai Research and Development Center,

Japan Atomic Energy Agency

The Workshop is hosted by
the Nuclear Science and Engineering Directorate, JAEA.

Workshop secretaries

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The United States Environmental Protection Agency

The 4th JAEA-EPA Workshop on Radiation Risk Assessment - Program

Tuesday, November 7, 2006

9:15-	Registration	
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12:00-13:00	Lunch	
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1-1 Research on Radiation Effect and Radiation Protection at JAEA

K. Saito

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This presentation gives an overview concerning researches on biological radiation effects and radiation protection being conducted at JAEA. Biological researches have been carried out in several different groups aiming at different targets. 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 units. Radiation protection research programs were also rearranged after the organizational integration in 2005 inheriting studies carried out in the preceding research laboratories. In the both research fields, our specific potential cultivated in the development of nuclear technology has been utilized sufficiently: one is computational simulation technique and another irradiation technique by different kinds of radiation.

Obviously an important target of the radiation effect research is contribution to clarification of low-dose effect and related risks. In this viewpoint, our researches focus on the examination of radiation effect mechanisms at molecular and cellular levels. The studies cover computational simulation of DNA damage induction and repair processes, modeling of carcinogenesis process, experimental characterization of DNA damages, study on bystander effect utilizing micro beam technique, molecular biology study of radioresistant bacterium *Deinococcus radiodurans*. Obtained knowledge would be expected to contribute not only to the low-dose problem but also to deeper understanding of deterministic effects, to production of novel mutants in terms of plant breeding, and so on.

The main objective of the radiation protection study at JAEA is to develop dosimetry techniques to ensure radiation safety of workers and the public. The expanding human activities in use of nuclear energy and radiation have produced necessity for new studies for exposure conditions which have not been postulated previously. On the basis of the necessity, following studies are carried out at JAEA: external dosimetry for high energy radiation, internal dosimetry for spallation nuclides, dose assessment studies for accidents, development of Japanese phantoms, development of new nuclear decay data, calibration techniques and so on. These studies will be developed to adapt to the new ICRP recommendation.

Interdisciplinary studies between radiation effect and protection have been started, that is, neutron dosimetry for analyzing animal experiments, and dosimetry considering radiosensitive cells in organs. Furthermore, an important application of these studies is medical fields such as radiation therapy and medical diagnosis. Considering these situations the research on radiation effect and protection at JAEA will be extended with wider application scopes.

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

E. Douple and R. Jostes

Nuclear and Radiation Studies Board, The National Academies

The Biological Effects of Ionizing Radiation (BEIR VII) Committee, composed of 18 scientists assembled by the Board on Radiation Effects Research (BRER) in the National Research Council of the National Academies, published an assessment of the potential risks of low-dose, low-LET ionizing radiation in 2006.[1] The BEIR VII Committee reviewed the evidence from numerous biological and epidemiological studies since the 1990 BEIR V report and incorporated its assessment of that evidence into the BEIR VII risk estimates. The committee developed a linear, no-threshold dose-response relationship between exposure to ionizing radiation and the development of cancer in humans for exposures up to 100 mSv, 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. Epidemiological data from studies of persons exposed for medical reasons, and of nuclear workers exposed at low doses and dose rates, were also evaluated. The risk model predicts that in 100 people with an age distribution typical of the U.S. population who receive an acute exposure of 100 mSv, 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. In addition, the committee estimated the risk following radiation exposure for both incidence and mortality for 11 specific cancer sites. It also estimated the total risk of heritable genetic diseases from parents exposed prior to conception to be about 3,000 to 4,700 cases per million progeny per Sv, which is very low (0.4-0.6%) compared to an estimated baseline risk of 738,000 cases per million. The committee concluded that while non-cancer disease outcomes such as cardiovascular disease can result from exposures to high doses of radiation, the data available at this time are not sufficient to develop reliable estimates of risk for these non-cancer outcomes at low doses of radiation. Finally, 12 specific recommendations were presented as needs for future research. This presentation will summarize the BEIR VII report, highlighting similarities and differences between the recent results and the risk assessments of the earlier BEIR V[2] report along with some issues whose resolution might assist future risk assessments.

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1-3 Bystander Effect Studies using Heavy-ion Microbeam

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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. Furthermore, there has recently been an increasing interest in ionizing radiation-induced "bystander effects", that is, radiation effects transmitted from hit cells to neighboring un-hit cells.

Therefore, 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).

Direct investigation of bystander effects using heavy-ion microbeam in Chinese hamster ovary CHO-K1 cells and normal human fibroblast AG01522 cells will be discussed.

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

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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 based on a "draft White Paper" that 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.

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1-5 Molecular Dynamics Simulation of DNA Strand Breaks

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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 the results of computer analysis of a DNA with single strand break, employing both molecular dynamics and quantum simulations.[1-5)]

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 - will be reported. Non-homologous End-Joining mechanism as a direct joining of the broken ends (the primary mechanism of the double strand break repair in mammals) was chosen to be simulated. Ku heterodimer is essential for this mechanism, being the subject of molecular dynamics simulation as a part of enzyme-DNA complex.

As the third area of research, the results from simulations of related problem - clustered DNA damage – will also be presented, focusing on multiple DNA damage containing 8-oxoGuanine and AP (abasic) site situated in various positions in the oligonucleotides.

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1-6 ORNL's DCAL Software Package

K.F. Eckerman

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Oak Ridge National Laboratory has released it's 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 and models representing 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 95/98/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)]. The dose and risk coefficients calculated by this release are consistent with those published in Federal Guidance Reports 12 and 13. The software can be downloaded from the ORNL website[8)]

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

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Neutrons are generally more effective radiation than X and γ rays for induction of neoplasm, and for most other late somatic effects of radiation. In the investigation of the neutron effects for human, it is inevitable to rely on the results obtained from experimental animals (mice and rats etc.), because there is no useful epidemiological data for human. However, there remains the common problem that exists with all data obtained from small animals; the radiation field at the target tissue is quite different in mouse and human, even if the subjects are exposed with an identical radiation field. In order to develop an appropriate method to assess the neutron effects for human from experimental data of small animals, it is very important to understand the behavior of the radiations inside the subjects.

The present study intends to analyze internal radiation field of a typical mouse with volume-pixel (voxel) phantom and radiation transport code. A mouse was imaged by using the dedicated small-animal CT scanner, in which slice pitch was set at 0.1 mm. Each image with the resolution of 0.02 mm was segmented to construct a voxel phantom by using computer tools (JCDS[1)]), which have been applied to process human-head images for 3-dimensional dosimetry in Boron Neutron Capture Therapy at the Japan Atomic Energy Agency. Input files for the Particle and Heavy Ion Transport code System (PHITS[2)]) were prepared from the constructed 3-dimensional voxel-based image data. Absorbed dose distributions and LET spectra in the mouse body were calculated by PHITS on the segmented images.

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

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The International Commission on Radiological Protection (ICRP) may release new primary radiation protection recommendations in 2007. Thus, Committee 2 has underway reviews of the dosimetric and biokinetic models and associated data that it uses to calculation dose coefficients for the intake of radionuclides and exposure 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.

Committee 2 has prepared a chapter and an annex for the forthcoming recommendations on the dosimetric quantities of radiological protection addressing their applications and limitations. 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. A new model of the gastrointestinal or alimentary tract has been adopted and should be published shortly. The DOCAL Task Group has reviewed its computational formulations with the objective of harmonizing the formulation with those of nuclear medicine[2)]. DOCAL has developed new computational phantoms representing the adult male and female reference persons of ICRP Publication 89[3)]. In addition, DOCAL will issue a publication on nuclear decay data to replace ICRP Publication 38[4)]. That effort, a collaborative effort between JAEA and ORNL, has been conducted under the JAEA/EPA agreement (see abstract by Endo and Eckerman). While the current effort have focused on updating the dose coefficients for occupational intake of radionuclide plans are being formulated to address dose coefficients for external radiation fields which will address high energy fields associated with accelerators and space travel.

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

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Under the agreement for cooperation between JAEA (formerly JAERI) and US EPA, JAEA and ORNL have jointly developed a nuclear decay database for 1037 radionuclides, which are significant in medical, environmental and occupational exposures. This collaborative effort has been aimed at providing updated compilations of the Medical Internal Radiation Dose (MIRD) Committee and the International Commission on Radiological Protection (ICRP).[1)]

The decay data were assembled from decay data sets of the Evaluated Nuclear Structure Data File (ENSDF), the latest version as of 2003. Basic nuclear properties in the ENSDF that are particularly important for calculating energies and intensities of radiations were examined and updated by referring to NUBASE2003/AME2003, the database for nuclear and decay properties of nuclides. In addition, modification of incomplete ENSDF was done for their format errors, level schemes, normalization records, and so on. The energies and intensities of emitted radiations by the nuclear transformation and the subsequent atomic process were computed from the ENSDF using the computer code EDISTR04. EDISTR04 is an enhanced version of EDISTR used for assembling the MIRD monograph and ICRP Publication 38 (ICRP38), and incorporates updates of atomic data and computation methods for calculating atomic radiations and spontaneous fission radiations. Quality assurance of the compiled data was made by comparisons with various experimental data and decay databases prepared from different computer codes and data libraries.

A package of the data files, called DECDC2[2)] (Nuclear <u>DEC</u>ay Data for <u>D</u>osimetry <u>C</u>alculation, Version <u>2</u>), was released in 2005 and was then forwarded to the MIRD Committee and the ICRP Task Group DOCAL for review for publication. DECDC2 will succeed the MIRD monograph and ICRP38 and will be extensively used in dose calculation in various applications.

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

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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. Particle transport simulation codes can play an important role in the estimation, since some dosimetric quantities such as fluence to dose conversion coefficients cannot be evaluated experimentally. We are therefore developing a general-purpose Monte Carlo code PHITS,[1)] which can deal with the transports of all kinds of hadrons and heavy ions with energies up to 200 GeV/n.

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 the ionization instead of neutron. This function enables us to evaluate dose equivalent in the unit of Sv directly by employing the Q(L) relationship defined in ICRP60, as well as dose in Gy. Taking this advantage, we have calculated the fluence to effective dose and effective dose equivalent conversion coefficients for protons, neutrons, pions and several kinds of heavy ions with energies up to 200 GeV/n,[2,3)] using the code coupled with an anthropomorphic phantom of the MIRD5 type.[4)]

The detailed features of the PHITS code will be given in the presentation, together with the calculated results of the dose conversion coefficients and their applications.

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

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Organ and tissue doses are fundamental quantity to estimate the risk to radiation exposure. Since the organ doses 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 in conjunction with computational human phantoms.

In recent years, realistic human phantoms have become available on the basis of medical imaging technique, such as computational tomography (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” (volume pixel), 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 many voxel phantoms have been developed for the purposes of radiation protection[1)]. However, most of the voxel phantoms were based on Caucasian anatomical data.

At Japan Atomic Energy Agency (JAEA), a series of Japanese voxel phantoms have been developed to provide a set of basic data for radiation protection for Asian. So far, five adult phantoms for three males (Otoko, JM, and JM2) and two females (Onago and JF) have been completed. The JM and JF phantoms are the male and female phantoms, respectively, whose voxel size is 0.98 mm × 0.98 mm × 1 mm. The small voxel enable us to model realistically the shapes of small or thin organs and tissues. The JM2 phantom was constructed from CT data taken in upright posture of the same subject for employed in the construction with JM. Comparison between JM and JM2 shows the differences in organ doses due to postures.

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

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

A.C. James[1] and B.G. Brooks[2]

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Office of Health, Safety and Security, US DOE

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 109 still-living Registrants, whose average age is now about 76 years (youngest 34.7 y, oldest 95.4 y). This presentation will outline the objectives and progress of 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; (iii) model the effectiveness of chelation therapy, and (iv) examine the adequacy of protection standards utilized for plutonium workers in the early years of the nuclear industry. USTUR's plan to make available via the world-wide web the de-identified health physics, bioassay and tissue analysis case data, together with autopsy findings and summary narratives for all cases (to facilitate wider scientific study) will also be outlined.

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

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On 30th November in 1978, the accident where two workers were short of oxygen occurred at the acid recovery cell in the nuclear spent fuel reprocessing plant of JAEA-NFCEL. They were recovered safely by immediate rescue operation. The subsequent measurements by the whole-body counter found that they were contaminated internally with radioruthenium-106. Excreta analysis and prolonged lung monitoring were carried out for the one of them. An outstanding high value was obtained in the lung monitoring on the day of the accident. The physicochemical characteristics of the materials were not observed. Although pure inhalation was assumed for internal dose assessment at the time of the accident, fractional intake via ingestion is expected from the situation. The bioassay datasets obtained from the subject has been still value of analysis due to a rare case of human accidental intake. In this study, reasonable interpretation of the bioassay datasets of the worker was performed in reference to the guideline demonstrated in the IDEAS project[1,2]. In conclusion, the effective half-life of the materials in the lungs and the f_1 value are evaluated at 120 days and less than 5×10^{-3}, respectively. In addition, simultaneous intake via inhalation and ingestion can be suggested from several pieces of evidence. Although the aerosol size of the materials is not determined in simultaneous intake, the resulting committed effective dose is 1 mSv and the variation of the dose is found to be less than about 5 % with the aerosol size ranging from 1 μm to 20 μm.

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

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Area and personal dosemeters/dose ratemeters used for radiation protection monitoring must be properly calibrated at regular intervals. The FRS, Facility of Radiation Standards, of JAEA has been developing the reference radiation fields of X-, gamma-, beta-rays and neutrons for calibrating the dosemeters 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. In progress are developments for neutron fields of monoenergy and changeable energy spectrum and the field of high-energy gamma-rays (about 6-7 MeV). Until now the established radiation fields have been utilized through a calibration service institute, Institute of Radiation Measurement (IRM) in response to the demands from outside. Otherwise the utilization has been limited only to radiation monitoring and research purposes inside our institute.

At the merger of former JAERI and JNC in October 2005, newly-constituted JAEA positioned FRS as one of the facilities to promote the utilization for the outside demands. In November 2006, we started to open to various institutes, universities and private companies 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 the FRS is the comprehensive secondary standard facility of Japan. However, a laboratory quality assurance is not yet introduced. 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.

In this workshop, we present the outlines of our radiation fields, the planning and provision of services and the approach to the quality assurance.

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3-1 Current Emergency Programs for Nuclear Installations in Japan

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Large effort has been taken for nuclear emergency programs in Japan especially after JCO accident. A special law for nuclear emergency was established just 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, action levels and action procedures are defined based on dose 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 headquarter was designated at each site. The license holders’ responsibilities were also clarified to develop operator’s plan for nuclear emergency preparedness and establish on-site organization for nuclear emergency preparedness, and designate a manager of the organization.

On the basis of the law, emergency preparedness including plans, organizations, systems and materials have been established throughout the country. Local emergency centers (Off-site Centers) were constructed at each area of nuclear facility sites and drills for nuclear emergency are conducted every year by both national and local governments. Periodic performance of emergency drill does highly enhance the capability of emergency response. To support decision making from technical aspect, emergency radiation monitoring and computer-based dose prediction system have been also improved.

This paper describes that structure of emergency program, responsibility of related organization and definition of unusual events for notification and emergency. During nuclear emergency, emergency preparedness, emergency radiation monitoring and computer-based prediction of on- and off-site situation are also addressed.

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

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The U.S. Environmental Protection Agency's (EPA) 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."

A draft revision of the PAG Manual will provide several key updates and additions. It clarifies the use of the existing 1992 protective action guides and protective actions for incidents other than nuclear power plant accidents. The draft revision lowers the projected thyroid dose for administration of stable iodine based on data from the Chernobyl accident. It provides new guidance concerning consumption of drinking water during or after a radiological emergency and limits projected doses from drinking water to 0.5 rem (50 mSv). It also updates the dosimetry basis from ICRP 26[2] to ICRP 60[3] for all derived levels.

Finally, the draft revision includes new guidance for dealing with long-term site restoration following a major radiological release, based on new Department of Homeland Security guidance on implementing PAGs after a radiological dispersal device (RDD) or improvised nuclear device (IND),[4] which was developed by a multi-agency working group that included EPA. The guidance acknowledges that for the broad range of potential impacts from radiation incidents, no single numeric cleanup level can be recommended. Instead, it provides a framework to follow to ensure key stakeholders are involved in a cleanup decision-making process that carefully weighs all relevant factors.

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

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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.

The Radiological Hazards Prevention Law has modified the legal definition of "radioactive materials". The historical definition by "74 Bq/g and 3.7 MBq" was replaced with a set of nuclide-specific radioactivity concentrations and quantities. The exemption levels recommended in the *International Basic Safety Standards*[1)] are adopted as the numerical values for the new definition with some modification.

The Reactors Control Law has been amended to establish clearance levels for releasing radioactive materials from regulatory control. The numerical values of the clearance levels refer to the *IAEA's Safety Guide*[2)].

There are many cases where Japanese industry involves naturally-occurring radioactive materials (NORM) without radiation protection concerns. The Japanese government has a plan to develop guidelines for exclusion and exemption of certain types of NORM. The ICRP's concept of intervention[3)] is considered to be applicable to defining the scope of regulation of NORM in the guidelines.

The amendments and the new guidelines are 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 are 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. The effectiveness of the new regulations should 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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This paper summarizes the 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 this report in terms of collective dose and the distribution of dose among the monitored individuals.

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. 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, which is the same as the value for 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.” 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).

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 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.

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

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Many dosimetric quantities have been defined for radiological protection by the International Commission on Radiological Protection (ICRP) and the International Commission on Radiation Units and Measurements (ICRU). Health effect due to radiation exposure should be appropriately quantified in radiological protection, in addition to imparted energy to human body. As a result, the system of dosimetric quantity has its own complexity, which may not be found in other physical quantities. Thus, the Japan Health Physics Society established the Expert Committee on concepts of Dosimetric Quantities used in radiological protection (ECDQ) to review and clarify issues in dosimetric quantities in April 2005.

ECDQ has mainly made discussions on the following three topics; factors used in weighting absorbed doses (Q(L) and w_R), the protection quantities recommended by ICRP and the operational quantities defined by ICRP. Nowadays, 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. It has been suggested that some of protection quantities have been misapplied to radiological protection. ICRP clearly described the application of effective dose in the draft recommendations, opened for consultation in this June. ECDQ has also made fruitful discussions on meanings, applicability and accompanied uncertainties of the protection quantities and relevant coefficients for various exposure conditions since its establishment.

One of the major goals in ECDQ is to propose a more comprehensive and simpler system of dosimetric quantities. The relation between the protection quantity and the operational quantity has been one of the most disputable points, because this issue directly relate to actual radiation protection practice. A simple system was once proposed to determine directly the protection quantities by adjusting the monitor response to the conversion factor. It is clarified, however, in reviews of ECDQ that the concept of ‘measurable’ operational quantity is quite essential in radiation monitoring. A more comprehensive or simpler system can be introduced in dosimetric quantities, if the proposed system is consistent with actual radiation protection, such as radiation monitoring, control of individual doses and so on.

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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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.[1-3]

The author is challenging to calculate probabilistic effective dose suggested by ICRP[4] 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. It means that probabilistic assessment has an advantage for the dose assessment of a nuclear fuel cycle facility.

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3-7 Updates to EPA's Yucca Mountain Rule: The Post-10,000 Year Standard

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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.[1] The United States Congress divided responsibility for Yucca Mountain among three agencies. The US Department of Energy (DOE) is responsible for constructing and operating the repository. EPA is responsible for establishing public health and environmental protection standards specific to Yucca Mountain, and the US Nuclear Regulatory Commission (NRC) is responsible for licensing and regulating the repository, using EPA's standards as a compliance measure. In setting standards, Congress required EPA to consult with the US National Academy of Sciences (NAS). In the 2001 disposal standards for Yucca Mountain, EPA set a limit of 15 millirems (150 microsieverts) per year committed effective dose equivalent (CEDE) for the reasonably maximally exposed individual (RMEI) living in the accessible environment above the highest level of concentration in the ground water plume. The standard covers all pathways for a period of 10,000 years following closure of the repository.

As a result of a lawsuit, a Federal court upheld a challenge to the 2001 rule, finding that EPA did not follow the advice of the NAS. The court specifically stated that there is no scientific basis for limiting repository performance to 10,000 years, given that the NAS recommended that the performance standard should extend to the period of highest risk (interpreted by EPA to be the time of peak dose). In response, EPA proposed additional standards[2] 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 350 millirems (3.5 millisieverts) per year to the RMEI. Because of the unprecedented challenges required to account for uncertainties in dose projection this far into the future, a dose limit consistent with the range of variation in natural background in the United States was deemed reasonable. EPA is now evaluating comments received on this proposal and expects to issue final standards for Yucca Mountain in late 2006.

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3-8 Status of Decommissioning and Waste Management in the Nuclear Science Research Institute of JAEA

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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. After that we have a few experiences of the decommissioning of research reactors and research laboratories including a reprocessing test facility (JRTF). 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. For example, the dismantlement of JRR-2, which is a heavy water cooled reactor, has been carried out to protect internal exposures due to tritium[1, 2]. The dismantling activities of the JRTF have been carried out to protect internal exposures due to alpha-ray emitting radionuclides such as ^{239}Pu.

As the decommissioning activities of the JPDR, the JRR-2 and the JRTF, about 2,440 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, are 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.

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[3]. 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 clearance system.

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