REGULATORY SCIENCE AND RADIATION PROTECTION: A STUDY OF DOSE CONSTRAINTS FOR MEMBERS OF THE PUBLIC AND OCCUPATIONALLY-EXPOSED WORKERS AT THE U.S. NUCLEAR POWER PLANTS

A Dissertation

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ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is considering a revision of the existing system of radiation protection regulations with respect to ICRP Publication 103. It is expected that there will be a change in the current NRC regulations to require the implementation of concept of dose constraints for members of the public and for occupationally-exposed workers at the U.S. nuclear power plants (NPPs). Under the paradigm of regulatory science, the use of dose constraints is still highly debatable.

This study addressed two objectives. The first objective was determining whether or not dose constraints are necessary for members of the public and occupationally-exposed workers at the U.S. NPPs. The second objective was determining, if dose constraints were needed, the optimal numerical values of dose constraints at the U.S. NPPs. To achieve these objectives, several areas were investigated and analyzed: 1) the establishment of a regulatory-science framework; 2) a system of radiation protection which would incorporate the concept of dose constraints; 3) methodologies and regulations for public and occupational dose assessment; 4) approaches to the establishment of dose constraints; 5) the actual doses for members of the public living around NPPs; and 6) the range of doses for occupationally-exposed workers in NPPs.

As a result of analysis of exposure data, the annual median and maximum doses to a maximally-exposed individual (MEI) for members of the public were 10^{-4} and 10^{-1} mSv, respectively. The corresponding annual excess risks (*ER)* for the median and maximum doses were calculated to be on the order of 10^{-8} and 10^{-6}, respectively. These excess risks are low and should be considered acceptable. For occupationally-exposed workers, the

average and maximum measurable doses were 1.3 mSv and 24.8 mSv, respectively. The annual excess risks for the average and maximum doses were 10^{-5} and 10^{-3}, respectively. These excess risks are also acceptable from the perspective of occupational risks. This analysis showed that some individuals received relatively higher annual doses than all others. The fraction of the workers in this category was negligible (0.01%) and the economic cost of further dose reduction based on dose constraints will have no net positive benefit. Thus, it is concluded that dose constraints are not necessary for members of the public or occupationally-exposed workers at the U.S. NPPs.

DEDICATION

To my parents, my wife and son

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NOMENCLATURE

ALARA	As low as reasonably achievable
ALI	Annual limits on intake
BSS	Basic Safety Standards
Bq	Becquerel
BWR	Boiling water reactor
CDRH	Center for Devices and Radiological Health
CEDE	Committed effective dose equivalent
CFR	Code of Federal Regulations
Ci	Curie (1 Ci = 3.7×10^{10} Bq)
CINDY	Code for internal dosimetry
cm	Centimeter
d	Day
DAC	Derived air concentration
DCF	Dose conversion factor
DDE	Deep dose equivalent
DDREF	Dose and dose-rate effectiveness factor
DOE	U.S. Department of Energy
EAR	Excess absolute risk
EC	European Commission
EPA	U.S. Environmental Protection Agency
ERR	Excess relative risk

EU	European Union
EURATOM	European Atomic Energy Community
FDA	U.S. Food and Drug Administration
FOIA	Freedom of Information Act
FSAR	Final safety analysis report
ft	Foot
GIT	Gastrointestinal Tract
h	Hour
HEEE	High-end exposure estimate
IAEA	International Atomic Energy Agency
ICR	International Congress of Radiology
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiological Units and Measurements
IXRPC	International X-ray and Radium Protection Committee
kg	kilogram
L	Liter
lb	Pound
LDE	Lens dose equivalent
LNT	Linear-non-threshold
LOCA	Loss of coolant accident
LWR	Light water reactor

MEI	Maximally exposed individual
mL	Milliliter
MPBB	Maximum permissible body burden
MPC	Maximum permissible concentration
mrem	Millirem
mSv	Millisievert
NCRP	National Council on Radiation Protection and Measurements
NESHAP	National emission standards for hazardous air pollutants
NPP	Nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NUREG	NRC technical report designation (NUclear REGulatory)
ODCM	Offsite dose calculation manual
PWR	Pressurized water reactor
RBE	Relative biological effectiveness
rem	rem
REMP	Radiological Environmental Monitoring Program
RETS	Radiological Effluent Technical Specifications
RG	Regulatory guide
RWP	Radiation work permit
SDE	Shallow dose equivalent
SECY	Secretary of the Commission, Office of the NRC
SRM	Staff requirements memorandum

Sv	Sievert (1 Sv = 100 rem)
SWP	Special work permit
TEDE	Total effective dose equivalent
TLD	Thermoluminescent dosimeter
TUBE	Theoretical upper bounding estimate
UN	United Nations
UNSCEAR	UN Scientific Committee on the Effects of Atomic Radiation
w	Week
y	Year
°C	Celsius
°F	Fahrenheit

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INTRODUCTION

The concept of regulatory science

The term “regulatory science” does not originate from radiological or nuclear sciences and its origin is not clear (Jasanoff 2011). However, regulatory science is commonly used by the medical and pharmaceutical industry involving the study of safety and efficacy of drugs and medical devices. According to the U.S. Food and Drug Administration (FDA), regulatory science is defined as the science of developing new tools, standards, and approaches to assess the safety, efficacy, quality, and performance of all FDA-regulated products (USFDA 2010). This definition also includes regulated activities. Based on the FDA definition of regulatory science, efficacy is a key factor. However, regulatory science studies the tradeoff between risk and benefits. If a medical device or pharmaceutical drug were to require the highest level of safety or efficacy without uncertainty, then the R&D costs would be unaffordable. Therefore, in regulatory science it is important to balance these two variables, safety and efficacy, to maximize the benefits of drugs and devices and to minimize the cycle cost.

Sheila Jasanoff discussed in detail the characteristics of regulatory science with regard to differences between research science and regulatory science (Jasanoff 1995). Jasanoff explained that research science is usually determined by theoretical paradigms and clear scientific methodologies. But regulatory science is relatively flexible, contentious and dependent on socio-economic factors and political considerations. Therefore, from the customary point of view of science, the determination of “good” or “bad” research science relies entirely on the stable repetitive result according to an

orthodox rule. On the other hand, regulatory science focuses on the consensus of selecting among different possible evaluations of observations and experiments. Therefore, the determination of goodness or badness of the regulatory science relies on negotiation and compromise, risk analysis and social needs. The differences between regulatory and research science are summarized in Table 1 (Jasanoff 1995).

Table 1. Comparison between regulatory science and research science.

Factors	Regulatory Science	Research Science
Aims	Certainty taking into account policy	Certainty taking into account originality and importance
Institutes	Government agencies, industry	Universities, national laboratories
Outcomes	Data analysis, often unpublished	Published papers
Motivations	Compliance with legal demands	Professional recognition
Time-frame	Legal timetables	Flexible
Choices	Acceptance of data Rejection of data Delay for more data	Acceptance of data Rejection of data
Responsible Institutes	Congress, courts, media	Review peers
Process	Regulatory peer review Inspection and audit Public hearing Legal review and approval	Peer review
Criteria	Absence of deception or distortion Conformity to protocols and agency guidelines Legal tests of sufficiency	Absence of deception or distortion Conformity to methods accepted by peer scientists Statistical significance

Regulatory science and radiological protection under the FDA

Regarding radiation exposure from diagnostic and therapeutic radiation producing devices, the FDA's Center for Devices and Radiological Health (CDRH) has been at the forefront of regulatory science (Leszczynski 2014). Current scientific evidence is very clear regarding acute high doses of radiation. However, little is known regarding the risk and health effects at protracted cumulative low doses. CDRH, under the Clinical Path Initiative and Advancing Regulatory Path Initiative, is currently seeking and implementing new tools and methods to provide scientific evidence on the early and late effects of cumulative low doses of ionizing radiation by means of predictive omics approaches, including from multiple combined hazards, such as radiation and chemical exposures (Chen and McKone 2001). Thus, the use of regulatory science in radiological protection is a well-established area of research in CDRH.

Exposure of patients or subjects to ionizing radiation is regulated by the FDA under 21 CFR Part 361, which is limited in scope as shown in Table 2, and it is based on a clear benefit-analysis of the radiographic study (USFDA 2015). NCRP Report 160 provides a succinct description of the absorbed doses received by the U.S. population (cumulative and per capita) due to medical exposures indicating that doses have increased steadily due to the predominant use of CT and other interventional modalities, such as fluoroscopy (NCRP 2009). The effective dose per individual (E_{US}) was estimated at 3 mSv in 2006, a factor 5.7 higher than that received between 1980 and 1982. Therefore, medical patients implicitly accept the risk from medical exposures due to their immediate health benefits. However, there are no restrictions on the number of times a radiological device

can be used on an individual patient as it is under the prerogative of a physician to assess the need for a diagnostic or therapeutic radiological study or treatment.

Table 2. Limits on radiation absorbed dose established by the FDA as described in 21 CFR Part 361 for diagnostic procedures.

Organ	Dose (mSv)
Whole body, active blood-forming organs, lens of the eye, and gonads	
Single dose	30
Annual and total dose commitment	50
Other organs:	
Single dose	50
Annual and total dose commitment	150

The lack of knowledge in risk assessment for patients and medical occupational workers has led the National Cancer Institute (NCI) to establish the Radiation Epidemiology Branch (REB) under the Division of Cancer Epidemiology and Genetics. The Radiation Epidemiology Branch (REB) was established to identify, understand, and quantify the risk of cancer in populations exposed to medical, occupational, or environmental radiation, and to advance understanding of radiation carcinogenesis, including occupational exposures such as those from NPPs. The methods for risk analysis used by the REB vary considerably depending on multiple factors including environmental exposures, such as chemical and biologics. Thus, the REB has established the Radiation Risk Assessment Tool – Lifetime Cancer Risk from Ionizing Radiation, among many other risk assessment programs, which is based on the BEIR VII Report for

the eleven cancer sites described in the report (Preston et al. 2002; National Cancer Institute 2003; Kocher et al. 2008; Gonzalez et al. 2012). For occupational radiation exposure, the REB is studying four population groups for assessing cancer risk. These are 1) Chernobyl Clean-up Workers, 2) Mayak Nuclear Facility Workers, 3) U.S. Radiologic Technologists, and 4) Interventional Fluoroscopists. These epidemiological studies will serve to address the risk associated with protracted low-radiation exposures compared to those of the general population.

Regulatory science and radiological protection from NPPs

The application of ionizing radiation, including that from NPPs, provides both risks and benefits. Although NPPs provide significant benefits to the society by electricity production, the general public is very sensitive to exposure from radiological sources. Radiation exposure is the unavoidable byproduct of nuclear-electric generation. It is a well-known fact that high radiation exposure causes severe health effects in humans and can lead to development of different types of cancer that may be fatal. Therefore, the operation of NPPs has been considered to need special regulatory requirements to ensure that it is properly used and controlled.

On the other hand, most of activities during the operation of NPPs involve very low radiation doses. Nothing can be said with certainty about the effects of low dose levels. Thus, it is debatable whether or not it is reasonable to apply the precautionary principle to nuclear industry, especially for NPPs. The precautionary principle recommends that action should be taken to prevent serious potential hazard despite the lack of full scientific

evidence (Reis et al. 2007). This principle has been used widely by international organizations, such as ICRP and IAEA, and European nations to prevent environmental problems due to radiation exposure from nuclear facilities. Most of the standards associated with radiation exposure are not based on this precautionary approach but consider the probability of an effect due to radiation exposure. Since there are still no definite data to prove the risk at low doses of radiation, the use of the precautionary principle for decision-making is controversial issue. The question is does the precautionary principle and regulatory science have a place in radiological protection at nuclear power facilities?

For radiological safety from all nuclear industrial activities, the International Commission on Radiological Protection (ICRP) introduced the use of cost-benefit analysis to determine the reasonable level of dose reduction below the recommended limits and introduced the concept of collective dose equivalent to facilitate cost-benefit analysis (OECD NEA 2011). The collective dose equivalent is the sum of all the dose equivalents received by the individual members of a population and is generally used for calculating the stochastic effects of radiation exposure of a large group or a population (Cember and Johnson 2009). According to ICRP Publication 26, the net benefit is calculated using the following equation:

$$B = V - (P + X + Y) \quad (1)$$

where B is the net benefit of a practice involving radiation exposure, V is its gross benefit, P is the basic practice cost, X is the cost of achieving a selected level of protection, and Y

is the cost of the detriment involved in the practice (ICRP 1977). The ICRP also recommended the use of differential cost-benefit analysis to find the optimum net benefit from the practice using the collective dose equivalent as the independent variable. The optimum net benefit can be calculated using the following equation:

$$\frac{dV}{dS} - \left(\frac{dP}{dS} + \frac{dX}{dS} + \frac{dY}{dS} \right) = 0 \qquad (2)$$

where S is the collective dose equivalent from the practice (ICRP 1977). Equation (2) is also reduced to Equation (3) because V and P can be regarded as a constant with S for a given practice:

$$\left(\frac{dX}{dS} \right)_{S^*} = - \left(\frac{dY}{dS} \right)_{S^*} . \qquad (3)$$

Therefore, the increase in the cost of protection per unit dose equivalent balances the decrease of detriment per unit dose equivalent at a certain collective dose equivalent S^* (ICRP 1977). This cost-benefit analysis can be more useful if a monetary value is assigned to the unit of collective dose equivalent, but the ICRP does not provide such a monetary value, since it is very complicated to quantify some of the components of the detriment, taking into account socio-economic factors (OECD NEA 2011).

This is an optimization process, and it is used by regulatory bodies to provide reasonably achievable regulatory guides for the nuclear power industry. In the Publication 60, the ICRP emphasized the importance of optimization again, especially taking into

account economic and social factors (ICRP 1991). Therefore, regulatory science related to radiological protection is not limited to the pure research outcomes, but it involves the social and economic considerations.

Issues of regulatory science for radiation protection in the nuclear power industry

One of the most important issues in the nuclear power industry is the implementation of new ICRP recommendations on radiological protection published in ICRP Publication 103. In 2007, the ICRP revised its previous recommendations stated in ICRP Publication 60 to the new ICRP Publication 103. This revision adopted new biological and physical findings and trends in the setting of radiation safety standards, including the radiological protection of the environment (ICRP 2007). ICRP Publication 103 provided consolidated standards for all controllable exposure situations in accordance with new technical findings published in supplementary reports after ICRP Publication 60. The new recommendations updated radiation and tissue weighting factors and radiation risk coefficients based on the latest available scientific data. In terms of protection methodology, the previous process-based protection approach using practices and interventions was changed into three exposure situations: planned, emergency, and existing (ICRP 2007). Furthermore, the revised recommendations emphasized the use of dose constraints for planned exposure situations and reference levels for emergency and existing exposure situations as the most important features of the principle of optimization (ICRP 2007).

According to ICRP Publication 103, dose constraints are defined as prospective and source-related limits on individual doses from a radiological source under planned exposure situations (ICRP 2007). This application of ICRP Publication 103 will probably be the most difficult for the nuclear power industry to implement in the field (Kong et al. 2014). Dose constraints are expected to have an impact on NPP operation since constraints are involved in radiation dose to both occupational workers and members of the public. For example, it is expected to provide more measures or equipment to reduce the radiation dose to workers. Moreover, a nuclear licensee may have a burden to more strictly control the release of radioactive materials to the environment to meet the dose constraint for members of the public living around a NPP. However, it is important to point out that the ICRP 103 recommendations, including dose constraints, are not yet part of the Code of Federal Regulations.

The IAEA has continuously provided its basic safety standards (BSS) on radiation safety to establish basic requirements for protecting people and the environment against the risks associated with radiation exposure (IAEA 1996). These BSSs have been used by IAEA member nations to make their own national regulatory programs even though the approval of BSSs within their national legal system is not mandatory. Many member nations used the BSS published in 1996 (hereinafter referred to as the 'BSS of 1996') as a benchmark for regulations on radiological protection prior to the issue of new ICRP recommendations in 2007. This new publication let the IAEA Commission on Safety Standards and IAEA sponsoring organizations, such as the European Commission and the OECD Nuclear Energy Agency, request the IAEA Secretariat to revise the BSS of 1996,

ensuring consistency with the new ICRP recommendations (IAEA 2014a). In 2014, the IAEA finally published the new BSS as General Safety Requirements No. GSR Part 3, which supersedes the BSS of 1996, with regards to the new findings of UNSCEAR and the new ICRP recommendations (IAEA 2014a). The IAEA expects that the new BSS will also play a major role as an international benchmark for radiation safety requirements. The acceptance and implementation by IAEA member nations will create better consistency in radiation safety of different member nations. In particular, the new BSS is aimed at the use by regulatory bodies in IAEA member nations; thus, the nuclear power industry, including NPPs, whose activities are under the regulations, will be greatly influenced by the issue of new IAEA BSS.

To establish practical requirements for radiation safety, the new BSS incorporates three different types of exposure situations: planned, emergency, and existing, which were introduced in ICRP Publication 103. In particular, planned exposure situations arise from deliberately introduced and fully controlled activities that result in exposures due to radiation sources (ICRP 2007; IAEA 2014a). Therefore, the routine radiation exposures involving activities in NPPs, including maintenance and the discharge of radioactive effluents, belong to planned exposure situations. In terms of planned exposure situations, the new BSS specifies that dose constraints should be applied to occupational and public exposure as optimization tools for radiation safety (IAEA 2014a). According to the new BSS, dose constraints will play a role as boundary settings in outlining the range of options for the purposes of optimization of radiation safety. Therefore, implementation of dose constraints in radiation protection activities in NPPs is inevitable for nuclear licensees.

However, it is important to understand that implementation of the new BSS, including dose constraints, has not yet been required by the U.S. NRC regulations.

In 2013, the European Commission (EC) provided the Basic Safety Standards (BSS) Directive for radiation protection, as Council Directive 2013/59/Euratom which combines five existing Euratom Directives: 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom (Council of the European Union 2013). The European Atomic Energy Community (Euratom) is a legal organization separate from the European Union (EU), and was established in 1957 by the Euratom Treaty for the peaceful uses of nuclear energy (EC 2015). According to the Euratom Treaty, one of the primary tasks of Euratom is instituting uniform safety standards to protect the health of workers and the general public from ionizing radiation (EU 2012). Therefore, the Euratom is closely connected with regulations for radiation safety in the EU nuclear power industry. The revision of Euratom Directives originates from three motives (HSE 2015). First, the ICRP issued the new recommendations on radiological protection in 2007. Second, the IAEA has revised its own BSS based on the new ICRP recommendations. Finally, the Euratom also participated in the revision of IAEA BSS as a cosponsor, with the purpose of working towards harmonization between the IAEA BSS and the Euratom BSS Directives. In other words, the new BSS Directive integrates the latest ICRP recommendations and harmonizes the EU regulations with the IAEA BSS (HSE 2015).

According to Council Directive 2013/59/Euratom, all EU member nations must establish dose constraints with the objective of optimizing radiological protection, taking into account their social criteria (Council of the European Union 2013). A nuclear licensee

can voluntarily set dose constraints for occupational exposure as a prospective upper bound of individual doses and implement the dose constraint in practice under the supervision of a regulatory body. On the other hand, a regulatory body should set dose constraints for members of the public living around nuclear facilities, assuming the planned operation of a specified radiation source. However, the BSS Directive does not provide a specific dose constraint for each type of exposure (Council of the European Union 2013). Therefore, the establishment of dose constraints with consideration of international guidance and good practices elsewhere will be an imperative assignment for both regulatory bodies and nuclear licensees, including NPP operators.

In 2001, the staffs of the U.S. NRC submitted a Policy Issue Notation Vote Paper, SECY-01-0148, to their Commissioners. They requested the revision of the current Title 10 Code of Federal Regulation (CFR) Part 20 with regard to the adoption of ICRP Recommendation in 1990 (Publication 60) on occupational dose limits and dosimetric models and parameters (USNRC 2001b). The next year, the NRC made a decision to postpone the revision of the current 10 CFR Part 20 until the ICRP completes the issuing of the new recommendations (Publication 103) (USNRC 2002). After the issuance of the ICRP new recommendation in 2007, the NRC staff proposed several ways to revise the basic system of radiation protection regulations, taking into account ICRP Publication 103 (USNRC 2008b). As a result, in the SRM to SECY-08-0197, the NRC finally determined that they would revise the whole system of radiation protection regulation and guidance with respect to ICRP Publication 103. The NRC also decided to immediately begin engagement with stakeholders and interested parties to initiate development of the

technical basis for possible revision of the NRC radiation protection regulations (USNRC 2009b). In particular, this revision would affect the NRC regulatory framework, including 1) 10 CFR Part 20, "Standards for Protection Against Radiation," 2) 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and 3) Appendix I to 10 CFR Part 50, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents" (USNRC 2015e). In 2012, the staff of the NRC proposed policy and technical directions to revise the NRC radiation protection regulations and guidance by SECY-12-0064 (USNRC 2012d). In the SRM to SECY-12-0064, the NRC accepted the NRC staff's development of a draft regulatory basis for a revision to 10 CFR Part 20 to harmonize with the most recent methodology and terminology in ICRP Publication 103 (USNRC 2012g). However, the NRC rejected the NRC staff's recommendations to reduce the annual occupational dose limit to 20 mSv y^{-1} and the elimination of traditional units, such as rem and Ci, from the NRC regulations (USNRC 2012g). Rather, the NRC decided to use both traditional and International System units in the NRC regulations (USNRC 2015e). Furthermore, since the ICRP was currently developing new biokinetic and dosimetric models and dose coefficients for both workers and public exposure to radionuclides based on ICRP Publication 103, and it is predicted that this information will be available after 2015, the NRC would not initiate rulemaking to reflect these changes until the new ICRP methodology for dosimetry was available (USNRC 2014e, 2015f).

In the United States, the term “dose constraint” is already used in the NRC regulations, but it has a somewhat different meaning than the ICRP concept (McGarry 2011). The term “constraint” is defined in 10 CFR Part 20 as a control of the air emission of radioactive material from non-reactor facilities to the environment. Title 10 CFR Part 20 stipulates that this constraint is the value at which the licensee should take appropriate corrective action to ensure the dose limit to members of the public was not exceeded (USNRC 1991c). Contrary to the licensees of non-reactor facilities, the NRC does not require the licensees of reactor facilities to establish dose constraints (USNRC 2010a). In other words, the dose constraint used by the NRC focuses on the control of radioactive materials from non-reactor facilities released to the environment to reduce the dose to members of the public. However, the ICRP recommends that dose constraints be implemented for all nuclear facilities and for both occupational and public exposure. Although the NRC does not currently require NPP licensees to set dose constraints, some licensees in the United States are familiar with the concept of dose constraint since they establish planning values, such as self-imposed administrative limits, in their radiation protection programs or ALARA analysis (USNRC 2010a; USNRC 2014a). In general, some licensees used planning values voluntarily as an operational tool to reduce the dose. However, since the NRC staff recommended revision of the radiation protection regulations and guidance with respect to ICRP Publication 103, it is expected that there will be a change in the current regulations to require that all nuclear licensees establish and apply constraints to their radiation protection program. Therefore, the appropriate method to implement dose constraints into regulatory programs will be actively discussed

among regulatory bodies, stakeholders, and interested parties (USNRC 2014a). One of the most important issues in this discussion will probably be setting numerical values for the dose constraints for both occupational and public exposure.

Objectives of present research

This research was conducted as a preparation of the revision of current NRC regulations according to ICRP Publication 103, which requires the implementation of concept of dose constraints for members of the public and for occupationally-exposed workers at the U.S. NPPs. Under the paradigm of regulatory science, the use of dose constraints is still highly debatable. This research addressed two objectives. The first objective was determining whether or not dose constraints are necessary for members of the public and occupationally-exposed workers at the U.S. NPPs. The second objective was determining, if dose constraints were needed, the optimal numerical values of dose constraints at the U.S. NPPs. To achieve these objectives, several areas were investigated and analyzed: 1) the establishment of a regulatory-science framework; 2) a system of radiation protection which would incorporate the concept of dose constraints; 3) methodologies and regulations for public and occupational dose assessment; 4) approaches to the establishment of dose constraints; 5) the actual doses for members of the public living around NPPs; and 6) the range of doses for occupationally-exposed workers in NPPs.

REGULATIONS

The concept of dose constraints

ICRP Publication 103 replaced the previous process-based approach of practices and interventions to the situation-based approach to cover all possible radiation exposure situations, including planned exposure situations, emergency exposure situations, and existing exposure situations (ICRP 2007; OECD NEA 2011). Planned exposure situations involve the intentional introduction and operation of sources. These situations include typical licensed activities where planning and controls are prepared in advance of radiation exposure (ICRP 2007). Therefore, these situations correspond to the concept of practice, and the routine radiation exposures involving activities in NPPs, including maintenance and the discharge of radioactive effluents. Emergency exposure situations are unpredicted situations which may arise during the operation of a planned exposure situation or from a malevolent action (ICRP 2007). It is necessary to take immediate action to prevent the spread of the radiation exposure. Existing exposure situations indicate that radiation exposure already exists before a measure of control has to be taken (ICRP 2007). These situations normally involve radon exposure in homes and, in aspects of nuclear operation, decommissioning NPP sites, which need remedial action to reduce radiation exposure.

The three principles of radiological protection in ICRP Publication 60 were continuously used in ICRP Publication 103. However, in the revised recommendations, the ICRP categorized the principles to the source-related and individual-related principles to show how these principles apply to sources and to the individual (ICRP 2007). Two principles, including justification and optimization of protection, belong to a source-

related approach and apply to all radiation exposure situations. In particular, the ICRP underlined that dose constraints and reference levels are the main parts in the optimization process and should be used to ensure that all radiation exposures are kept as low as reasonably achievable (ALARA), taking into account societal and economic factors (ICRP 2007). One principle, application of dose limits, belongs to an individual-related approach and applies only in planned exposure situations (ICRP 2007). The differences in concept between the use of individual dose limits in planned exposure situations and the use of dose constraints or reference levels for protection from a source in all radiation exposure situations are displayed in Fig. 1 (ICRP 2007). The source-related approach focuses on the radiation exposure of all the individuals exposed from a single source, while the individual-related approach focuses on the radiation exposure of one individual from several sources. Dose limits are applicable to individual-related approach, and dose constraints or reference levels are applicable to source-related approach.

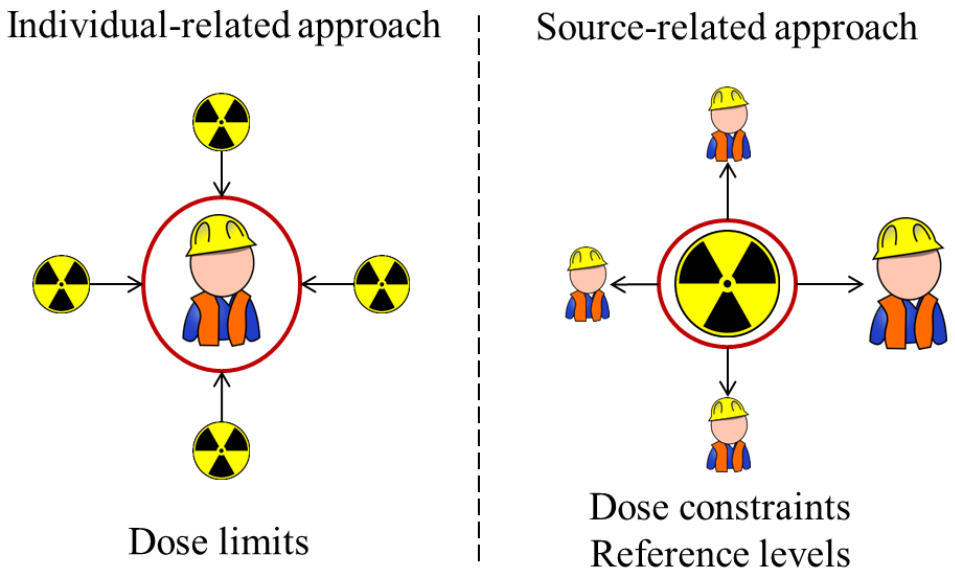

Fig. 1. Differences between individual-related dose limits and source-related dose constraints or reference levels.

In ICRP Publication 103, the ICRP regards the source-related principle of optimization below the dose constraint or reference level as the most effective tool for protection regardless of exposure situations (ICRP 2007). Dose constraints are prospective and source-related restrictions on the individual dose from a source in planned exposure situations except in radiation exposure for medical treatment. In particular, the doses to be compared with the dose constraint are potential doses that may be received in the future and can be affected by decisions and protective measures (OECD NEA 2011). For occupational exposure, a dose constraint can be used to select better optimization options in the field to achieve ALARA. For public exposure, a dose constraint becomes a maximum value of the annual dose limit for the members of the public during the planned operation of any controlled source. Therefore, licensees should take possible measures to

keep the dose for the members of the public below the dose constraint. The ICRP emphasized that dose constraints are not to be used or understood as strict regulatory limits (ICRP 2007). Although, contrary to ICRP Publication 60, ICRP Publication 103 recommends using dose constraints as a requirement for licensees, the ICRP focuses on the intent of dose constraints to reduce the dose, without penalty for exceeding dose constraints. That is, a dose constraint is an operational tool for licensees to help themselves to reduce the dose. Therefore, exceeding a dose constraint is not a violation of regulations. If a dose constraint is exceeded, a licensee is required to take measures to check whether the protection had been optimized, whether the appropriate dose constraint had been set up, and whether additional steps to reduce doses to acceptable levels would be proper.

Similar to dose constraints in planned exposure situations, reference levels play a role as the level of dose or risk that protection measures should be planned and optimized to reduce the dose, in emergency or existing controllable exposure situations (ICRP 2007). The use of different terminology, dose constraints for planned exposure situations and reference levels for emergency or existing exposure situations, is due to the characteristics of exposure situations. That is, in planned exposure situations, the constraint on individual dose can be applied at the planning stage, and the dose can be estimated to confirm that the dose will not exceed the constraint. On the other hand, in emergency or existing exposure situations, radiation exposure can exist with wide ranges of individual doses, and optimization actions may apply to initial levels of individual doses beyond the reference level. The different types of restrictions on individual doses, used in ICRP Publication 103,

depend on exposure situations and categories of exposure as shown in Table 3 (ICRP 2007).

Table 3. Application of dose constraints and reference levels depending on exposure situations and categories of exposure.

Exposure situations	Occupational exposure	Public exposure	Medical exposure
Planned	Dose limit Dose constraint	Dose limit Dose constraint	Diagnostic reference level[a] (Dose constraint[b])
Emergency	Reference level[c]	Reference level	Not applicable
Existing	Not applicable[d]	Reference level	Not applicable

[a] Patients.
[b] Carers, comforters, and volunteers in research only.
[c] Long-term recovery operations belong to planned occupational exposure.
[d] Exposures resulting from long-term remediation operations or from extended work in contaminated areas should be treated as part of planned occupational exposure, although the source of radiation is 'existing.'

The dose constraint for occupational exposure is set voluntarily by a nuclear licensee, taking into account the optimization process necessary to lower individual doses below the dose constraint. For public exposure, the dose constraint should be set by the regulatory body, taking into account an upper bound of the annual doses for members of the public during the planned operation of a specified controlled source. In terms of the assignment of specific values for dose constraints or reference levels, ICRP Publication 103 provides guidance to help licensees or regulatory bodies to select appropriate values for their purposes (ICRP 2007). First, it is necessary to characterize the related exposure situation with regard to its nature, the benefits to individuals and society, and the feasibility

of reducing or avoiding the exposure. These characteristics can be compared with the features in Table 4, which is provided in ICRP Publication 103, to choose the appropriate range for the dose constraint or reference level (ICRP 2007; OECD NEA 2011). Finally, the specific value for the dose constraint or reference level can be determined by an optimization process with regards to national or regional features together with a consideration of international guidance and good practice elsewhere (ICRP 2007). For example, the ICRP suggests that acceptable dose constraint levels for occupational and public exposure in planned exposure situations are less than 20 and 1 mSv y^{-1}, respectively (ICRP 2007). In the case of radioactive waste disposal and prolonged exposure, levels of less than 0.3 mSv y^{-1} and less than 0.3-1 mSv y^{-1} are suggested, respectively, as a dose constraint for public exposure (ICRP 1998; ICRP 2007).

Table 4. ICRP guidance on the selection of appropriate ranges of dose constraints and reference levels.

Ranges of constraints and reference levels[a] (mSv)	Features of the exposure situation	Radiological protection requirements	Examples
Over 20 to 100[b, c]	Not controllable sources. Controlled by action on the exposure pathways.	Information on radiation risk and on actions to reduce doses. Assessment of individual doses.	Reference level for evacuation in a radiological emergency.
Over 1 to 20	Direct individual benefit from the situation but not from the exposure itself. Controlled at source or by action in the exposure pathways.	Available information on dose reduction. Assessment of individual doses and training for planned situations.	Constraints for occupational exposure. Constraints for comforters and caregivers. Reference level for radon in dwellings.
1 or less	No individual benefit but benefits to society in general. Controlled by action on the source.	Available information on the level of exposure. Periodic checks on the exposure pathways as to the level of exposure.	Constraints for public exposure in planned situations.

[a] Acute or annual effective dose.

[b] In exceptional situations, informed volunteer workers may receive doses above this band to save lives, prevent severe radiation-induced health effects, or prevent the development of catastrophic conditions.

[c] Situations in which the dose threshold for deterministic effects in relevant organs or tissue could be exceeded should always require action.

In terms of a result of the optimization process, it is imperative to know the reason that the ICRP introduced the concept of dose constraints and reference levels in the principle of optimization of protection. The ICRP recognized that the benefits and

detriments from radiation exposure are not likely to be distributed equally through individuals and a considerable inequity of radiation exposure between one individual and another does occur (ICRP 1991; ICRP 2007). The ICRP states that this inequity can be limited by integrating source-related restrictions on individual doses into the process of optimization (ICRP 1991; ICRP 2007). Therefore, dose constraints and reference levels will play a key role as a tool to reduce the doses of some individuals who are subject to much more radiation exposure than the average. The use of dose constraints or reference levels is illustrated in Fig. 2, which shows that the distribution of individual doses with time shift to low levels as a result of the optimization process (ICRP 2007). In Fig. 2, Step 1, 2, and 3 indicate the dose distribution before, during, and after implementing dose constraints or reference levels, respectively. The dose levels beyond a dose constraint or a reference level indicate that the corresponding radiation risk can be tolerated, but if you make an additional effort, the risk can be reduced. Fig. 2 shows that the distribution becomes narrower after implementing dose constraints.

ALARA and dose constraints generally have the same goal to reduce the radiation exposure as low as reasonably achievable. From the above it should be clear that ALARA is a goal or objective of radiation protection, and dose constraints are one of the operational tools to achieve ALARA. Specifically, under the principle of ALARA, the minimization of all radiation doses has been pursued to ensure an adequate level of radiation protection. On the other hand, dose constraints are aimed at minimizing radiation doses that are relatively higher than the average using the analysis of dose distribution. It should also be

noted that the application of ALARA and dose constraints involves highly subjective value judgments, which may include economic and other societal factors.

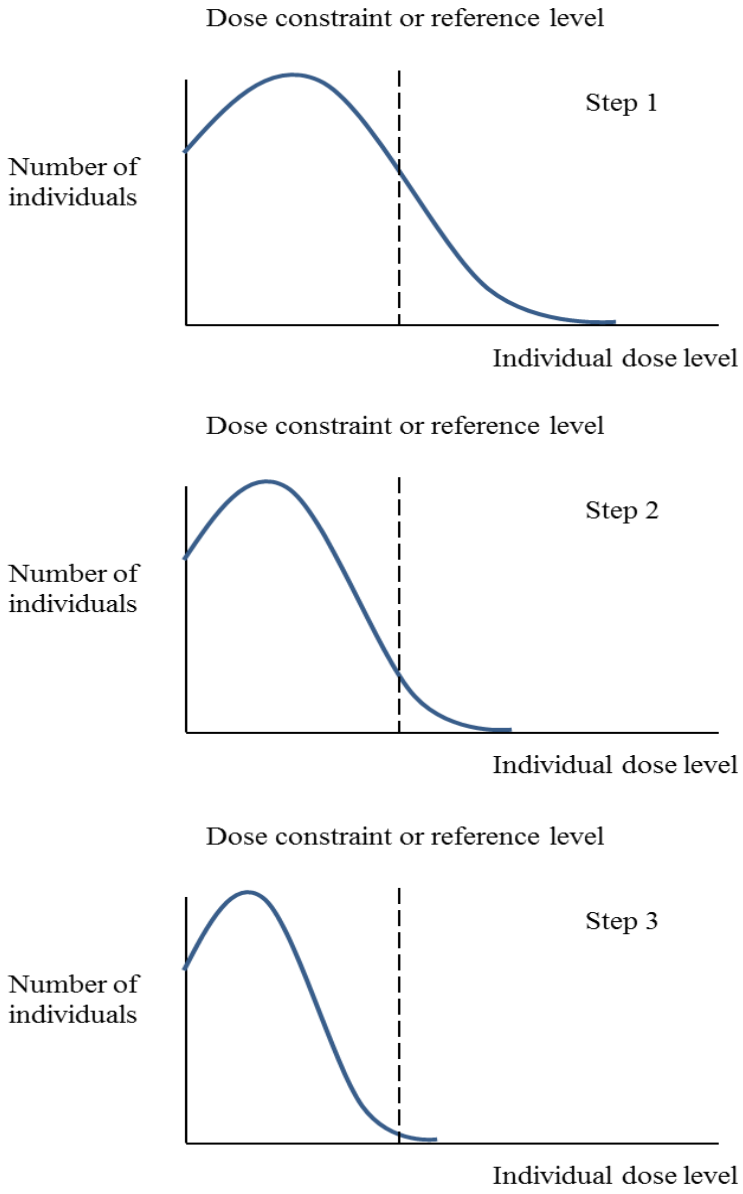

Fig. 2. Evolution of the distribution of individual doses with time as a result of the use of dose constraints or reference levels. Step 1, 2, and 3 indicate the dose distribution before, during, and after implementing dose constraints or reference levels, respectively.

Dose estimation for members of the public due to releases at nuclear power plants

To estimate the dose to members of the public arising from radiation sources, it is imperative to determine the number of factors that affect individual doses, including exposure time, location, transport of radionuclides through the environment, and the characteristics of individuals. The characteristics of individuals include information such as physiological parameters, dietary data, residence data, recreational activities, and many other individual-specific data. In general, a specific set of these characteristics can be referred to as an exposure scenario (ICRP 2006). Therefore, the estimated dose to individual members of the public can be very different depending on the selected exposure scenario, and to have accurate dose estimation, it is important to develop a reasonable exposure scenario and to obtain the reliable data for that scenario. However, in the real world, gathering of actual data is impractical or extremely expensive since individuals have a great variety of characteristics, which are very difficult to quantify. Despite these difficulties, regulatory bodies or related institutes have tried to develop a quantitative exposure scenario from an environmental regulation perspective to facilitate regulatory decision-making for protecting the public from unnecessary radiation exposure. As a part of this effort, the Environmental Protection Agency (EPA) developed the concept of a hypothetical "maximally exposed individual" (MEI) as a means of dealing with dose estimation for members of the public in the United States (USEPA 1992). The Nuclear Regulatory Commission (NRC) also adopted this concept to estimate the dose to members of the public living around nuclear facilities, including NPPs.

Maximally exposed individual (MEI)

The concept of the MEI was first introduced to assess the potential human health risks associated with exposure to environmental contaminants with regard to the national emission standards for hazardous air pollutant (NESHAP) regulations. These standards were established in 1989 by the EPA to protect the public from exposure to airborne contaminants (Hawkins 1991). According to the EPA guidelines for exposure assessment, the MEI is defined as the single individual with the highest exposure in a given population (USEPA 1992). The MEI has often been used as a worst case that represents an extreme set of exposure conditions. Therefore, the calculation of the MEI exposure has involved a variety of conservative assumptions. The most conservative and controversial of the assumptions was that the MEI lived for 70 years at the location considered by the dispersion model to receive the heaviest annual average concentration, that the person stayed there 24 hours per day, and that there is no difference between outdoor and indoor concentrations (NRC 1994). In practice, it is hard to imagine such a person. In terms of the individual exposure distribution, the MEI exposure level falls at the 99.99^{th} percentile of the distribution, which means the exposure is the highest in a given population (Hawkins 1991).

Due to the somewhat unclear definitions of the MEI, it was once replaced with two other estimators of the upper end of the individual exposure distribution, a high-end exposure estimate (HEEE) and the theoretical upper bounding estimate (TUBE) (USEPA 1992). These concepts were introduced to reflect the more reliable distribution of exposure, not implausible estimates of exposure. In the distribution of exposure, the high end extends

beyond the 90th percentile of the population distribution, but not higher than the individual in the population who has the highest exposure (USEPA 1992). This restriction implies that the use of an individual exposure distribution is necessary for estimating the dose to members of the public and the HEEE is a value in the upper tail of that distribution. However, the EPA did not provide the precise percentile for the HEEE, but recommended selecting the percentile, taking into account the population size in the specific application (NRC 1994). Meanwhile, the TUBE was introduced as a bounding estimate for the purpose of easy calculation of exposure. According to the EPA guidelines, the TUBE is calculated by assuming limits for all the variables used to calculate exposure and dose that, when combined, will result in the mathematically highest exposure or dose (USEPA 1992). However, in terms of the use of these concepts, the National Research Council raised the concern that the EPA did not provide sufficient technical background to determine the desired percentile of the individual exposure distribution, especially when the data are sparse (NRC 1994; 2009). Finally, as a response to this concern, the EPA returned its approach to the MEI (NRC 1994).

As mentioned above, the NRC also applies the MEI for estimating the dose to members of the public living around nuclear facilities. Since the MEI represents an individual who may be exposed to the highest concentrations of radioactive materials from radioactive effluents, the parameters and postulations used in this estimation normally include conservative assumptions that are inclined to overestimate the dose. Therefore, the real dose received by a real individual is often much less than that estimated (USNRC 2013d). In terms of regulations, 10 Code of Federal Regulation Part 63.312, “Required

characteristics of the reasonably maximally exposed individual," provides specific conditions for the MEI and defines the MEI as a hypothetical person who meets the following criteria: 1) living in the accessible environment beyond the highest concentration of radionuclides in the plume of contamination, 2) having a diet and life style which represents the residents in the town of Amargosa Valley, Nevada, 3) using well water with average concentrations of radionuclides based on an annual water demand of 3000 acre-feet, 4) drinking 2 liters of water per day from wells drilled into the ground water at the town of Amargosa Valley, Nevada, and 5) being an adult with metabolic and physiological respects consistent with current knowledge of adults (USNRC 2001a). The interesting thing here is that the MEI represents the diet and living style of residents in the town of Amargosa Valley, Nevada. The reason why the MEI focuses on this specific location is that the MEI was originally developed to estimate the potential human exposure resulting from release of radioactive materials from a geologic repository at Yucca Mountain. Therefore, the MEI was selected to represent those persons in the vicinity of Yucca Mountain who are expected to receive the highest exposure to radioactive materials released from a geologic repository.

On the other hand, the ICRP recommends using a somewhat different concept, the "representative person," for the purpose of protection of the public. According to ICRP Publication 101, the representative person is defined as a hypothetical individual who receives the dose that represents the more highly exposed individuals in the population (ICRP 2006). This representative person replaces the previous concept of the critical group, which is a group of people who receive the highest exposure from a particular radiation

source or set of sources. In terms of characteristics of the representative person, the ICRP recommends selecting reasonable individual habits (e.g., consumption of foodstuffs, breathing rate, location, etc.) of highly exposed individuals, but not the extreme habits of a single member of the population, like the MEI. The information and values can be derived from the national or regional population data. ICRP Publication 101 suggests using the 95^{th} percentile of behavior in deterministic calculations. In other words, in a prospective probabilistic assessment of dose to individuals, the ICRP recommends that the representative person should be defined such that the probability is less than around 5% that a person drawn at random from the population will receive a higher dose. Therefore, the difference between the representative person and the MEI is that the exposure level of the representative person falls at the 95^{th} percentile of its distribution, whereas the MEI's exposure is the highest exposure in a given population. Methods used for determining the dose to the representative person are summarized in Table 5 (ICRP 2006).

Table 5. Methods used for determining the dose to representative person.

Factors	Calculation method	
	Deterministic	Probabilistic
Environmental concentration data	Single values for parameters	Distribution of estimated or measured concentration
Habit data	Mean value for the more highly exposed group or 95th percentile of national or regional data	Range or fixed values for habit data
Dose coefficient	Fixed value based on age	Fixed value based on age
Dose to the representative person	Product of above values	Method selected by licensee or regulator; Probability of less than about 5% that a person drawn at random from the population will receive a higher exposure

To calculate potential doses to members of the public from exposure to radioactive materials during the routine operation of NPPs, it is necessary to characterize the transport of radioactive materials from the facility to the individual through the environment. This information includes the nuclides, activities, release rates, types of radiation emitted and migration to humans by the various possible exposure routes. The routes through which individuals may be exposed to radioactive materials are called exposure pathways (NRC 2012). In terms of radioactive effluents from NPPs, the effluent pathways are identified in NRC regulatory guide (RG) 1.109, "Calculation of annual doses to man from routine release of reactor effluents for the purpose of evaluating compliance with 10 CFR Part 50, Appendix I" (USNRC 1977b). In general, surface water and atmosphere are the main

pathways for radioactive effluents from NPPs to the public (USNRC 1977b, 2013d). All liquid and gaseous radioactive effluents from NPPs are monitored before release to the environment to determine whether the limits on release can be met or not.

When individuals are exposed to these radioactive effluents by water or air pathways, their doses are estimated to some extent by the amount of time spent in the vicinity of the radiation source or the amount of time that the radionuclides inhaled or ingested are retained in their bodies. The main exposure pathways for residents living around NPPs are as following: 1) inhalation of radioactive gases, 2) drinking milk or eating meat from animals that graze on open pasture on which radioactive materials may be deposited, 3) eating vegetables grown near the NPP site, and 4) drinking water or eating fish caught near the point of release of liquid radioactive effluents (USNRC 2013b). The dose is estimated based on the MEI, and site-specific data are typically used for dose estimation. If site-specific data are not available, conservative parameters are used to estimate the dose. In addition, if other exposure pathways, which contribute significantly to the individual dose are found by studies and observations of radionuclide movement through the environment or food chains, those exposure pathways should be considered for estimating the dose to members of the public.

Radioactive effluents released from NPPs are generally classified as several groups based on their physical characteristics and their contribution on internal or external radiation doses (USNRC 2013b). First, liquid radioactive effluents include tritium, fission products (e.g., strontium, iodine), and activation and corrosion products (e.g., sodium, iron, cobalt). These radionuclides usually contribute to the internal radiation dose through water

pathways, such as fish consumption, drinking water, and consumption of meat or vegetables grown near a NPP. Second, noble gases (e.g., krypton, xenon, argon) in gaseous radioactive effluents neither deposit on the ground nor are absorbed and accumulated within living organisms. Therefore, these radionuclides act mainly as radiation sources of direct external exposure emanating from the effluent plume. Dose estimation for noble gases is conducted for the site boundary, where the highest external radiation doses to members of the public are expected to occur (USNRC 2013b). Third, radioiodine and tritium in gaseous radioactive effluents can be deposited on the ground or inhaled during respiration. For these effluents, both external and internal exposure may occur. A main pathway of external radiation dose is the direct exposure from ground deposits of radioiodine. For internal exposure, inhalation of contaminated air or consumption of meat, milk, or vegetables are the main pathways. Lastly, particulates in gaseous radioactive effluents, including fission products (e.g., cesium, strontium) and activated corrosion products (e.g., cobalt, chromium) contribute to both direct external and internal radiation doses. Pathways of particulates are similar to those of radioiodine. Calculations for most pathways are limited to a radius of 80 km from the NPP site (USNRC 2013b).

According to the NRC Regulatory Guide (RG) 1.109 methodology, the doses are calculated for the individuals receiving the highest whole body and organ doses (USNRC 1977b, 2013d). Therefore, these doses are frequently referred to as the maximum whole body and the maximum organ doses. In terms of doses to the whole body or total body, the NRC uses somewhat different terminology, “total effective dose equivalent (TEDE),” in 10 CFR Part 20 and 50 compared to ICRP Publication 26. This term was defined by the

NRC to refer to the summation of internal and external exposure (USNRC 2015f). On the other hand, ICRP Publication 26 recommended using the phrase "the sum of the dose-equivalent from external exposure and the committed effective dose equivalent from the intake of radionuclide" (ICRP 1977). In terms of organ doses, nuclear licensees are required to calculate these for 6 separate organs in the human body: bone, liver, thyroid, kidney, lung, and intestines (USNRC 2013d). The organ which is most susceptible to radiation damage is referred to as the critical organ. The concept of critical organ is based on ICRP Publication 2, and the whole body or total body can also be a critical organ, if it receives the largest radiation dose (ICRP 1977). To control the dose to the whole body and any organ of members of the public, the NRC provides the dose criteria for the whole body and corresponding organs depending on the types of radioactive effluents.

In terms of dose estimation, it was recognized that the MEI may not be an adult because infants, children, and teenagers may have higher rates of intake per unit body mass (USNRC 1977a). Therefore, numerous studies were conducted to find dose coefficients for specific age groups in the population. ICRP also began developing age-dependent dose coefficients for members of the public in the mid-1980s (ICRP 2006). A series of publications was issued giving dose coefficients for specific age groups, based on biokinetic and dosimetric models. As a result, the available dose coefficients for different age groups offered the chance to calculate the numerical values for public exposure to radioactive effluents in a more comprehensive manner, as compared to the previous calculations based primarily on an adult member of the public. The NRC currently uses four age groups: infant (< 1 year), child (1-10 years), teenager (11-17 years),

and adult (> 17 years) (NRC 2012). The child is represented by a typical 4-year old, the teenager by a 14-year old and the adult by the definition for Standard Man as described in ICRP Publication 2 (ICRP 1959; USNRC 1977a). On the other hand, ICRP Publication 60 recommended using six age groups: newborn (<1 year), infants (1-2 years), young children (3-7 years), older children (8-12 years), teenagers (13-17 years), and adults (> 17 years) (ICRP 1991, 2006). However, ICRP Publication 101 recommended using more simple age groups for assessing doses in comparison with the relevant dose constraint, especially in prospective assessments (ICRP 2006). These groups are infant (< 5 years), child (6-15 years), and adult (16-70 years). For practical application of these age groups, the use of dose coefficients and metabolism data for a 1-year-old infant, a 10-year-old child, and an adult were recommended to represent the three age categories (ICRP 2006). The differences between the MEI and the representative person are summarized in Table 6 (ICRP 1959; USNRC 1977a; Hawkins 1991; USEPA 1992; NRC 1994; ICRP 2006).

Table 6. Comparison between the maximally exposed individual and the representative person.

Factors	Maximally exposed individual	Representative person
Aim	Assessment of the potential human health risks associated with exposure to environmental contaminants	Dose estimation for members of the public
Institute	U.S. EPA and U.S. NRC	ICRP
Definition	Single individual with the highest exposure in a given population	Hypothetical individual who receives the dose that represents the more highly exposed individuals in the population
Assumption	Living for 70 years at the location considered to receive the heaviest annual average concentration (24 hours per day)	95^{th} percentile of behaviors of residents
Habit data	Person in the vicinity of Yucca Mountain who are expected to receive the highest exposure to radioactive materials released from a geologic repository	Mean value for the more highly exposed group or 95^{th} percentile of national or regional data
Dose quantity	Total effective dose equivalent (TEDE)	Effective dose
Percentile	99.99^{th} percentile of the distribution or the highest	95^{th} percentile of the distribution
Age groups	Four age groups: infant (< 1 year), child (1-10 years), teenager (11-17 years), and adult (> 17 years)	Six age groups: newborn (<1 year), infants (1-2 years), young children (3-7 years), older children (8-12 years), teenagers (13-17 years), and adults (> 17 years)
Physiological model	Standard Man (ICRP Publication 2)	Reference Person (ICRP Publication 89)
Dose coefficient	Fixed value based on age groups (NRC NUREG-0172)	Fixed value based on age groups (ICRP Publications 56, 67, 69, 71, 72)

U.S. Regulation and guidance for public dose estimation

In the United States, dose estimation to members of the public living around NPPs is generally controlled by three basic regulations: 1) 10 CFR Part 20, "Standards for Protection Against Radiation," 2) 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," including Appendix I to 10 CFR Part 50, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," and 3) 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations" (USNRC 2015f). These regulations are based on a common concept that radiation exposure for members of the public is due to the release of radioactive effluents from nuclear facilities. Other dose contributions by natural radiation background, medical exposure, etc. are not considered in the consideration of the public dose. Therefore, it is important to control the release of radioactive effluents from nuclear facilities in order to keep the public doses as low as reasonably achievable (ALARA).

The NRC main radiation protection regulations are in 10 CFR Part 20. The purpose of these regulations is to establish the criteria of protection for both members of the public and occupationally-exposed workers from nuclear activities conducted under licenses issued by the NRC. The current 10 CFR Part 20 is based on ICRP Publication 26, published in 1977, which introduced a risk-based system of radiation protection with three principles: justification, optimization, and dose limitation (USNRC 2014e). The summation of internal and external exposure in 10 CFR Part 20 also was first introduced

in ICRP Publication 26. In addition, the technical basis for the annual limits on intake (ALIs) and derived air concentrations (DACs) for a large number of radionuclides was based on ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers," and ICRP Publication 32, "Limits for Inhalation of Radon Daughters by Workers" (USNRC 2014e).

In terms of public dose estimation, 10 CFR Part 20.1301, "Dose limits for individual members of the public," established the annual dose limit for members of the general public of 1 mSv (USNRC 1991c). In particular, in 10 CFR Part 20.1301(e) the NRC requires that a licensee comply with the EPA environmental standards for uranium fuel-cycle facilities, 40 CFR part 190 (USNRC 1991c). 10 CFR Part 20.1302, "Compliance with dose limits for individual members of the public," also requires that a licensee demonstrate compliance with the public dose limit set forth in 10 CFR Part 20.1301 by showing that the annual average concentrations of radioactive material released in gaseous and liquid effluents at the boundary of the unrestricted area do not exceed the values specified in Appendix B to 10 CFR Part 20. Appendix B to 10 CFR Part 20 provides concentration limits for gaseous and liquid radioactive effluents released to the environment. These values correspond to the radionuclide concentrations which, if inhaled or ingested continuously over the course of a year, would produce a TEDE of 0.5 mSv per year (USNRC 1991c).

As stated earlier, the term "dose constraint" is also used in 10 CFR Part 20, but its purpose is slightly different from that of the ICRP (McGarry 2011). 10 CFR Part 20.1101, "Radiation protection programs," requires that a licensee should set a constraint on

gaseous radioactive effluents released from non-reactor facilities to the environment. This constraint ensures that the MEI of the public will not be expected to receive a TEDE in excess of 0.1 mSv per year from these effluents (USNRC 1991c). The dose constraint used by the NRC is focused on the control of radioactive materials from non-reactor facilities to the environment to reduce dose to members of the public. The ICRP uses the dose constraint in a different way. It is used as an operational tool to restrict both occupational and public exposure and to apply to all nuclear facilities.

In the United States, all applicants or licensees should comply with 10 CFR Part 50 to operate their nuclear facilities with regulatory approvals (USNRC 1956a). In particular, Appendix I to 10 CFR Part 50 provides numerical guides for design objectives and limiting conditions for operation to keep levels of radioactive effluents released from NPPs to the environment as low as reasonably achievable (USNRC 1956b). Appendix I was first published in 1975, and its terminology and methodology for dosimetry are based on ICRP Publication 2, published in 1959 (USNRC 2015f). Since Appendix I is based on ICRP Publication 2, its methodology for dosimetry is somewhat different from that of the current NRC general radiation protection regulations in 10 CFR Part 20. This difference causes some concern with the current 10 CFR Part 50, including its Appendix I in that the regulations, guidance, and computer software do not align with that used in 10 CFR Part 20 (USNRC 2015f).

According to 10 CFR Part 50, the evaluation of off-site radiological consequences is required during the licensing process for the construction and the operation of a new NPP (USNRC 1956a, 2004a). For a construction permit, dose levels at the boundary of

the exclusion area, from postulated accidents, such as loss of coolant accidents (LOCA), should comply with specific dose criteria. For postulated accidents, the evaluation must meet the following two criteria in 10 CFR Part 50.34, "Contents of applications; technical information." First, an individual positioned at any point on the boundary of the exclusion area for any 2-hour period following the beginning of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv TEDE (USNRC 1956a). Second, an individual positioned at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release during the entire period of its passage would not receive a radiation dose in excess of 0.25 Sv TEDE (USNRC 1956a).

On the other hand, the dose standards for postulated normal operation, at the boundary of the exclusion area, are 0.75 mSv y^{-1} to the thyroid and 0.25 mSv y^{-1} to the whole body and any other organ (USEPA 1977). The values of dose standards originate from the EPA environmental protection standards, 40 CFR Part 190 (USEPA 1977; USNRC 1991c). These dose standards serve as practical dose constraints for a new NPP license and for its routine operation (Kong et al. 2014).

In terms of effluent management, 10 CFR Part 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents - nuclear power reactors," requires that a licensee establish design objectives for equipment to control release of radioactive effluents produced during the normal operation of NPPs (USNRC 1956a). These releases should be reported to the NRC regularly in accordance with requirements set forth in 10 CFR Part 50.36a, "Technical specifications on effluents from nuclear power

reactors" (USNRC 1956a). Even though a design objective is different from a dose constraint in 10 CFR Part 20.1101, it has a similar function as a constraint on gaseous radioactive effluents released from non-reactor facilities to the environment. The difference between a design objective in 10 CFR Part 50 and a dose constraint in 10 CFR Part 20 is that the former covers both gaseous and liquid radioactive effluents released from reactor facilities while the latter focuses on gaseous radioactive effluents only released from non-reactor facilities. The design objectives of Appendix I in 10 CFR Part 50 are summarized in Table 7 (USNRC 1956b). These objectives are also called system operating limits (USNRC 2013d). All numerical values in Table 7 apply to the highest off-site dose calculated for a MEI within the model framework (Bevelacqua 2009). These design objectives are more restrictive than either the 1 mSv y^{-1} dose limit for members of the public in 10 CFR Part 20.1301(a) or the effluent concentration limits in Appendix B of 10 CFR Part 20 which corresponds to 0.5 mSv y^{-1} (USNRC 2015f). However, as indicated in 10 CFR Part 50.34a(a), the design objectives of Appendix I of 10 CFR Part 50 are not radiation protection standards, but are design criteria to ensure that equipment designs keep radioactive effluents ALARA (USNRC 1956a). NPPs that meet these objectives are considered to be reducing dose to members of the general public from radioactive effluents to levels which are ALARA (USNRC 2014b).

Table 7. Design objectives of Appendix I of 10 CFR Part 50 for radioactive effluents released from each nuclear power reactor.

Dose	Type of effluents	Limit per unit
Total body from all pathways	Liquid	0.03 mSv y^{-1}
Any organ from all pathways	Liquid	0.1 mSv y^{-1}
Gamma dose in air	Noble gas	0.1 mGy y^{-1}
Beta dose in air	Noble gas	0.2 mGy y^{-1}
External dose to total body of an individual	Noble gas	0.05 mSv y^{-1}
External dose to the skin of an individual	Noble gas	0.15 mSv y^{-1}
Dose to any organ from all pathways	Radioiodine & particulates in gas (including ^{3}H and ^{14}C)	0.15 mSv y^{-1}

In 1979, the EPA established dose standards for uranium fuel-cycle facilities, including NPPs of 0.25 mSv y^{-1} to the whole body, 0.75 mSv y^{-1} to the thyroid, and 0.25 mSv y^{-1} to any other organ of any member of the public (USEPA 1977). These standards apply to the whole site or facility, whether it has a single unit or multiple units (USNRC 2012a). The values of these EPA standards are based on dosimetry concepts and dose calculation methods in ICRP Publication 2 (USNRC 2015f). The NRC integrated these EPA standards into its regulations 10 CFR Part 20.1301(e) in 1981, and all current NPPs should comply with these dose standards (USNRC 1991c; 2014b). The EPA established these standards by comparing the cost-effectiveness of various dose levels in reducing potential health risks from the operation of uranium fuel-cycle facilities. The EPA also presumed that the standards would be able to be met for up to four fuel-cycle facilities, for example four reactors, at a single site (USNRC 2014b). Table 8 shows the dose standards for members of the public as applied in the design of uranium fuel-cycle facilities (USEPA

1977). Although the EPA regulation uses the term dose standards instead of dose constraints, the terms have the same purpose and meaning.

Table 8. Dose standards of 40 CFR Part 190 for Uranium Fuel-cycle Facilities.

Dose	Limit per site
Whole body	0.25 mSv y^{-1}
Thyroid	0.75 mSv y^{-1}
Any other organ	0.25 mSv y^{-1}

To facilitate licensee implementation of the ALARA requirements of Appendix I to 10 CFR Part 50, the NRC provides a series of regulatory guides which describe methods for estimating the activity released in radioactive effluents, dispersion of effluents in the air and water, estimating potential doses to members of the public living around nuclear facilities, etc. Among these guides, the main guidance report is NRC RG 1.109, "Calculation of annual doses to man from routine release of reactor effluents for the purpose of evaluating compliance with 10 CFR Part 50, Appendix I" (USNRC 2015f). NRC RG 1.109 provides mathematical models and postulations for estimating doses to members of the public from liquid and gaseous radioactive effluents (USNRC 1977b). NRC RG 1.109 is also applicable to demonstrate compliance with 10 CFR Part 20 and 40 CFR Part 190. NRC RG 1.109 was first published in 1976, and the modeling approach used was based on ICRP Publication 2. The dose factors of NRC RG 1.109 also originated from ICRP Publication 2; thus, it is expected to be revised sometime later as part of effort to align the NRC regulations with ICRP Publication 103 (USNRC 2015f). To estimate the

dose from each type of radioactive effluent, NRC RG 1.109 has two supporting computer codes: “LADTAP II: Technical Reference and User Guide” for liquid effluents and “GASPAR II: Technical Reference and User Guide” for gaseous effluents (USNRC 1986, 1987). NRC RG 1.109 consists of seven sections, including 1) regulatory positions providing main equations for estimation of all liquid and gaseous exposure pathway doses, 2) Appendix A for liquid effluent pathways, 3) Appendix B for noble gas pathways, 4) Appendix C for pathways of gaseous effluent particulate, iodine, tritium, and carbon-14, 5) Appendix D for population doses which is no longer required, 6) Appendix E for numerical constants, dose conversion factors (DCFs), etc., and 7) Appendix F for the function used in computing gamma doses from noble gas releases (USNRC 1977b).

The general approach to estimating dose in NRC RG 1.109 consists of three simple steps: 1) using effluent releases to calculate nuclide concentrations in environmental media of interest, usually air, water, and food stuffs, 2) multiplying media nuclide concentration by human intake and consumption rates to calculate radionuclide intake, and 3) multiplying nuclide intake by a DCF to determine resulting dose (Sejkora 2013). In NRC RG 1.109, four pathways are considered for liquid radioactive effluents, including potable water, aquatic foods, shoreline deposits, and irrigated foods (USNRC 1977b). All pathways except shoreline deposits contribute to internal radiation dose. For gaseous radioactive effluents, there are four pathways, such as noble gas exposure, ground deposition exposure, inhalation, and crop ingestion (USNRC 1977b). The former two pathways contribute to external radiation dose, while the latter two pathways contribute to internal radiation dose. NRC RG 1.109 also provides internal DCFs for four age groups

(infant, child, teen, and adult) and for six critical organs (bone, liver, thyroid, kidney, lung, and gastrointestinal tract), including total body. All these numerical values are derived from ICRP Publication 2 (Sejkora 2013). NRC RG 1.109 provides DCFs for only 72 radionuclides, but two supporting computer codes, LADTAP and GASPAR, provide DCFs for 196 nuclides (Sejkora 2013).

There is a series of regulatory guides supporting NRC RG 1.109 technically. First, NRC RG 1.110, "Cost-benefit analysis for radwaste systems for light-water-cooled nuclear power reactors," provides methods to carry out cost-benefit analyses in assessing the performance of radioactive waste systems used in light water reactors (USNRC 2013a). NRC RG 1.111, "Methods for estimating atmospheric transport and dispersion of gaseous effluents in routine release from light-water-cooled reactors," describes mathematical and postulations for estimating atmospheric transport, dispersion, and deposition of airborne effluents during normal NPP operation (USNRC 1977d). NRC RG 1.112, "Calculation of releases of radioactive materials in gaseous and liquid effluents from light-water-cooled nuclear power reactors," demonstrates methods for calculating radioactive source terms for evaluating radioactive waste treatment systems (USNRC 2007a). NRC RG 1.113, "Estimating aquatic dispersion of effluents from accidental and routine reactor release for the purpose of implementing Appendix I," describes mathematical models and methods for estimating aquatic dispersion of both routine and accidental releases (USNRC 1977c). NRC RG 1.21, "Measuring, evaluating, and reporting radioactive material in liquid and gaseous effluents and solid waste," provides guidance for sampling and analysis of effluents from NRC-licensed NPPs (USNRC 2009a). In particular, NRC RG 1.21

recommends that licensees monitor all locations in the plant where greater than 1% of activity is discharged as liquid effluents, noble gases into the atmosphere, or anything else into the atmosphere (USNRC 2009a; NRC 2012).

The NRC has also published some technical reports as a NUREG-series to support NRC RG 1.109 and Appendix I to 10 CFR Part 50. First, NRC NUREG-1301, "Offsite dose calculation manual guidance standard radiological effluent controls for pressurized water reactors," and NRC NUREG-1302, "Offsite dose calculation manual guidance standard radiological effluent controls for boiling water reactors," describe how to control radioactive effluent monitoring instrumentation, effluent releases, and radiological environmental monitoring for pressurized water reactors (PWRs) and boiling water reactors (BWRs), respectively (USNRC 1991b, a). NRC NUREG-0543, "Methods for demonstrating LWR compliance with the EPA uranium fuel-cycle standard (40 CFR Part 190)," presents the specifications that implement 40 CFR Part 190 and Appendix I to 10 CFR Part 50 (USNRC 1980). NRC NUREG-0133, "Preparation of radiological effluent technical specifications for nuclear power plants," describes methods for the calculation of certain key values required in the preparation of proposed radiological effluent technical specifications for light-water-cooled NPPs (USNRC 1978). NRC NUREG-1301 and NRC NUREG-1302 are the basic guides for the offsite dose calculation manual (ODCM), which includes locations of monitoring instrumentation with respect to plant effluent systems, inspection requirements for effluent monitoring systems, sampling frequency and analysis for effluent monitoring, types of activity analysis of effluent samples, detection limits, etc. (USNRC 1991a, b). Since these reports are basic guides, site-specific effluent monitoring

programs can differ from the guidance in these NUREGs with suitable justifications and approvals (NRC 2012). Therefore, all NPP licensees have their own site-specific ODCMs.

The NRC requires that licensees of NPPs and fuel-cycle facilities monitor and report releases of radioactive effluents. The monitoring and reporting systems for NPPs are specified in the Radiological Effluent Technical Specifications (RETS) (NRC 2012). These specifications require the operator to monitor the discharge of radioactive effluents at every significant release point at the NPP. Effluent monitoring is composed of continuous measurements of some effluent streams, periodic measurement of radioactive particles trapped on filters, and measurement of samples from effluents released in batches (NRC 2012). Among those measurements, continuous and batch monitoring are the major processes in the RETS program at NPPs. Continuous releases of radioactive effluents are normally monitored by measuring gross activity with a continuously indicating radiation monitoring system (e.g., sodium iodide detector) (NRC 2012). These measurements can be used to sound alarms and halt effluent releases if the levels of activity exceed permissible limits. In addition, continuous measurements are combined with analyses of radioactive particulates trapped on filters or air samples from the effluent stream to acquire estimates of the radionuclide concentrations in the effluent stream (NRC 2012). Such samples are typically taken at specified frequencies, the period of which is determined by the variability of activity in the effluent stream. In contrast, effluent samples of batch releases are taken prior to purging or discharging. Analysis for "hard-to-detect" radionuclides, such as iron-55, strontium-89, and strontium-90, can be conducted after the effluent release (NRC 2012). Detailed information about the RETS program for a specific

NPP is incorporated into the licensee's ODCM, which is part of an operator's application for a NRC license.

In addition to RETS, the NRC requires that all NPPs have the Radiological Environmental Monitoring Program (REMP) to ensure that there are no harmful effects to the environment from NPP operations (NRC 2012). Therefore, all NPP licensees should submit an annual report to the NRC on the results of their monitoring programs. To determine if the releases of radioactive materials are detectable in the offsite environment, the NRC requires the NPP operators to collect air samples at numerous locations around the NPP. In general, measurements are conducted at five locations: three near the NPP boundary in the direction of most likely wind transport, one in the residential area likely to have the highest chance of radiation exposure, and one at the control location 15 to 30 km distant in the upwind direction of prevailing winds (USNRC 1991b; NRC 2012). Measurement for radioiodine is conducted weekly, and gross beta activity of particulates trapped on filters is measured quarterly. Weekly analysis is also conducted to identify gamma-ray-emitting radionuclides. In terms of flexibility of REMP, licensees are allowed to make changes to their programs without prior NRC approval (NRC 2012). However, licensees should notify the NRC of any changes of their REMP.

As mentioned earlier, the evaluation of the off-site radiological consequences is also required during the licensing process for applicants of a new NPP (USNRC 1956a, 1989, 2004a). NPPs were originally licensed under the two-step licensing process: construction permit and operating license. In 1989, the NRC also established a combined license which combines a construction permit and an operating license into a single license

(USNRC 2004a). For the two-step process, the requirements for acquiring an operating license are described in 10 CFR Part 50. In this process, the applicant submits a final safety analysis report (FSAR) to the NRC, and this FSAR includes the NPP's final design, safety evaluation, operational limits, anticipated response of the NPP to postulated accidents, and plans for coping with emergencies (USNRC 2004a). In terms of radioactive waste management in the FSAR, Appendix I to 10 CFR Part 50 provides the numerical guides for design objectives and limiting conditions to meet the criteria of radioactive effluents (USNRC 1956b). On the other hand, a combined license approves construction and conditional operation of a NPP at the same time. The requirements for acquiring a combined license are described in 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants" (USNRC 1989). However, the application for a combined license should basically include the same information required in an application for an operating license issued under 10 CFR Part 50 (USNRC 2004a). Therefore, 10 CFR Part 50 includes the basic requirements for the NPP licensing process regardless of the types of licensing processes. From the perspective of dose criteria, an applicant for NPP license should meet the requirements of 10 CFR Part 50.34 under the conditions of postulated accidents and the requirements of 10 CFR Part 20.1301(e) and design objectives in Appendix I to 10 CFR Part 50 under the condition of a postulated normal operation. In particular, the values of 10 CFR Part 20.1301(e) originate from the EPA dose standards (USEPA 1977; USNRC 1991c). In practice, these dose standards can serve as dose constraints for the license of new NPPs and for the normal operation of existing NPPs (Kong et al. 2014).

Computer codes for public dose estimation

The NRC uses computer codes to model and evaluate the release of radioactive effluents and the dose to members of the public during various operating conditions in NPPs. In terms of the calculation of radioactive effluents discharged from NPPs, the GALE code is normally used to estimate the quantities of activity released by a NPP through liquid and gaseous discharges during routine operation of PWRs and BWRs (USNRC 2015b). The GALE code, which stands for gaseous and liquid effluents, is based on the Fortran programming language. The NRC developed two GALE codes depending on the reactor type: 1) PWR-GALE code given in NRC NUREG-0017, Revision 1, "Calculation of releases of radioactive materials in gaseous and liquid effluents from pressurized water reactors (PWR-GALE Code)" and 2) BWR-GALE code given in NRC NUREG-0016, Revision 1, "Calculation of releases of radioactive materials in gaseous and liquid effluents from boiling water reactors (BWR-GALE Code)" (USNRC 1979, 1985). These codes are mainly used for the NRC to determine the license's conformance with the requirements of Appendix I to 10 CFR Part 50. These codes are also used for NPP design certification and combined license applications submitted under 10 CFR Part 52. In particular, NRC NUREG-0016 and NUREG-0017 identify liquid and gaseous source terms and specific analytical parameters used by the codes, including types of processing methods and flow rates, characteristics of filtration, ion-exchange resins, adsorbent media to treat process and effluent streams, decontamination factors, etc. (USNRC 1979, 1985). These parameters are based on accumulated data generated from operating reactors, field tests, laboratory tests, and plant-specific design considerations to

decrease the amount of radioactive materials which may be released to the environment. The results of GALE code, which are calculated activities of effluents, are used as input data for estimating the dose to the public.

NRCDose 2.3.20, "Code system for evaluating routine radioactive effluents from NPPs with Windows interface" can be used for the evaluation of the dose to members of the public (RSICC 2015). The NRCDose program was developed to conduct dose calculations for up to 169 radionuclides, six organs (bone, liver, thyroid, kidney, lung, and gastrointestinal tract) including the total body, and four age groups (infant, child, teenager, and adult). The DCFs used in the code are based on NRC RG 1.109 which is supplemented with additional DCFs from NRC NUREG-0172 (USNRC 1977b; 1977a; RSICC 2015). NRCDose consists of three following codes: LADTAP II for dose calculations due to liquid effluents, GASPAR II for dose calculations due to gaseous effluents, and XOQDOQ for calculations of the atmospheric diffusion parameters. Originally, LADTAP II, GASPAR II, and XOQDOQ were developed for mainframe computers used for bulk data processing, and written using the Fortran programming language. The methods of LADTAP II and GASPAR II are based on NRC RG 1.109, and XOQDOQ methods are based on NRC RG 1.111 (USNRC 1977b; 1977d; RSICC 2015). To facilitate users of these codes, the NRC provides additional technical reference and user guides, NRC NUREG/CR-4013, NUREG/CR-4653, and NUREG/CR-2919 for LADTAP II, GASPAR II, and XOQDOQ, respectively (USNRC 1982, 1986, 1987). The simplified chart of the ODCM and its process, including computer codes, is summarized in Table 9 and Fig. 3 (USNRC 1982; 1986, 1987; USNRC 1991b, 1991a; CEC1999).

Table 9. Simplified chart of offsite dose calculation manual (ODCM).

Category	Nuclides	Pathway	Receptor	Limits	Frequency
Gaseous effluents	Noble gases	Plume γ[a]	Total body	5 mSv y^{-1} Instantaneous[g]	As required by NPP procedure
		Plume γ[a] and β[b]	Skin	30 mSv y^{-1} Instantaneous[g]	
		Plume γ[a]	Air	0.05 mGy q^{-1}, 0.1 mGy y^{-1h}	Monthly
		Plume β[b]		0.1 mGy q^{-1}, 0.2 mGy y^{-1h}	
	Non-noble gases	Inhalation[b]	Adult (any organ)	15 mSv y^{-1} Instantaneous[g]	As required by NPP procedure
		Ground deposition[c]	Total body	75 μSv q^{-1}, 0.15 mSv y^{-1h}	Monthly and annually
		Inhalation	4 Age groups		
		Leafy vegetables[c]	(all organs)		
		Produce[c], milk[d], meat[d]			
Liquid effluents	All	Water	Total body	10 times 10 CFR Part 20 Appendix B and RETS	As required by NPP procedure
	Non-noble gases	Water[e] and fish[f]	Total body	15 μSv q^{-1}, 0.3 mSv y^{-1h}	Monthly
		Water[e] and fish[f]	4 Age groups (all organs)	50 μSv q^{-1}, 0.1 mSv y^{-1h}	
		Water[e]	Adult (whole body and all organs)	40 μSv y^{-1i}	When required by RETS
Uranium fuel cycle	All	All release plus direct radiation from contained sources	Whole body	0.25 mSv y^{-1j}	Annually
			Thyroid (adult)	0.75 mSv y^{-1j}	
			All other organs (Adult)	0.25 mSv y^{-1j}	
TEDE	All	External (DDE) + internal (CEDE)	Total body + organs (adults)	0.25 mSv y^{-1k}	Annually

[a] Evaluated at the unrestricted area boundary; [b] Evaluated at the location of maximum atmospheric dispersion parameter (X/Q). [c] Evaluated at the location of maximum offsite deposition parameter (D/Q). [d] Evaluated for the nearest producer within 8 km. [e] Evaluated for the nearest downstream community water supply. [f] Evaluated for fish caught in the near-field region downstream. [g] Radiological effluent technical specifications; [h] 10 CFR Part 50; [i] 40 CFR Part 141; [j] 40 CFR Part 190; [k] 10 CFR Part 20.

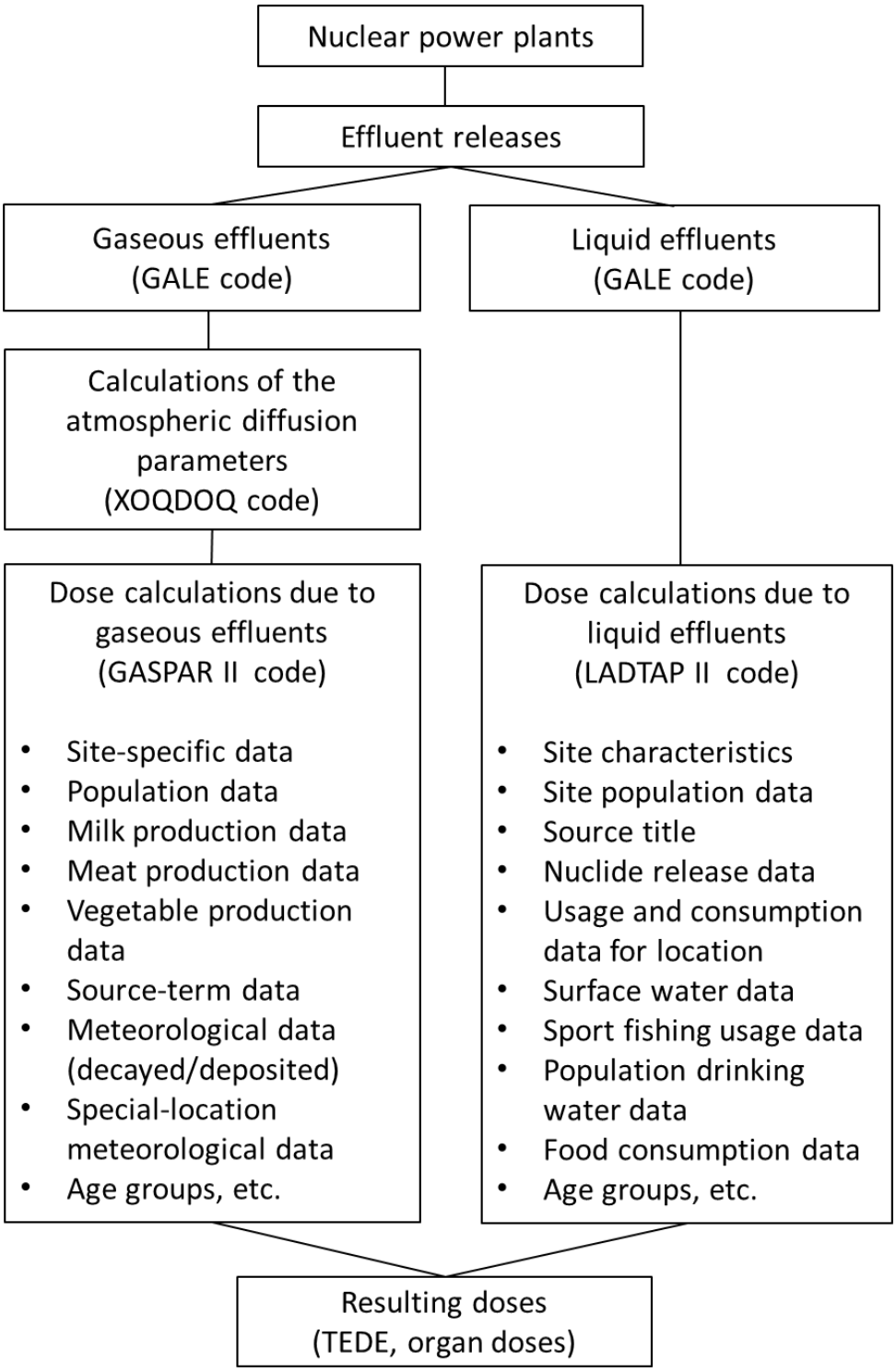

Fig. 3. Process of offsite dose calculation (ODCM) based on multiple codes for liquid and gaseous effluents.

Dose calculation for occupational workers at nuclear power plants

To calculate radiation dose in the human body, it is essential to identify the exposed individual and obtain his or her physiological data, regardless of types of exposure: external or internal radiation. To calculate the external radiation dose, the individual's mass and dimensions, and basic compositions of the organs is generally enough for most circumstances. On the other hand, for internal radiation exposure, much more biological information of the exposed individual is required to calculate the dose due to intake or inhalation of radioactive materials. This information includes respiration rates, excretion rates, removal rates of the radioactive material from various organs or from the body, etc. Due to the wide variety of data for each individual, it is difficult to use accurate physiological data of each individual for dose calculation. However, the levels of dose to individuals during routine NPP operations are sufficiently low so that the differences among individuals can be neglected for dose calculation (ICRP 1975). Furthermore, if a reference individual is defined well enough to represent a typical worker in the nuclear industry, this reference individual can facilitate routine dose calculation for occupational workers. Therefore, to meet the need of identifying a reference individual for the calculation of low-level doses, the ICRP issued ICRP Publication 23, "Report of the Task Group on Reference Man," which provided detailed information on the anatomical, morphological and physiological characteristics of humans related to the biokinetics or dosimetry of internally deposited radionuclides (ICRP 1975).

Reference man

The NRC uses the concept of the Reference Man, which originated from ICRP Publication 23, for dose calculation, especially for the calculation of ALIs and DACs for occupational exposure (ICRP 1975; USNRC 1991c). According to 10 CFR Part 20, the Reference Man is defined as "a hypothetical aggregation of human physical and physiological characteristics arrived at by international consensus" (USNRC 1991c). In addition, the Reference Man is defined in more detail as "a person with the anatomical and physiological characteristics of an average individual that is used in calculations assessing internal dose" (USNRC 2015g). These definitions indicate that the Reference Man is not a real person, and it was made mainly for calculating internal dose.

The Reference Man was originally called the Standard Man. In 1949, the concept of the Standard Man was first introduced at the Chalk River Conference on Permissible Dose to represent a typical or average adult who is exposed occupationally (NRC 2012). Specific data for organ masses and for the chemical composition of the total body and the various organs were attained from several previous studies. The characteristics of intake and excretion and the duration of occupational exposure were determined at the Chalk River Conference such that these data, including water balance, respiration, duration of exposure, and retention of particulates in lungs, should be average for normal activity in the temperate zone (ICRP 1975). These characteristics led to the use of the Standard Man as the basis for ICRP Publication 2 on permissible doses to workers from internally deposited radionuclides (ICRP 1959).

In 1963, a task group of ICRP Committee 2 was established to revise the concept of the Standard Man, and the term was also altered later from the Standard Man to the Reference Man (ICRP 1975). The ICRP stated that the characteristics of radionuclides in the human body must be specified clearly to apply the ICRP body burden and MPC correctly. Similar to the Standard Man, the Reference Man represented a typical radiation worker. Furthermore, to extend the concept of the Reference Man, ICRP Publication 23 provided information for individual variation depending on age, sex, and some other factors (ICRP 1975). Even though the data for the Reference Man were acquired from a wide variety of sources of various geographical areas and races, the ICRP selected data principally to represent a typical Western European or North American individual. According to ICRP Publication 23, the Reference Man is defined as a person who is a Caucasian, between 20-30 years old, 170 cm (5.58 ft.) in height, and 70 kg (154.32 lb.) in weight, and lives in an area of Western Europe or North America with an average temperature between 10-20 °C (50-68 °F) (ICRP 1975).

Although the Reference Man is still used in radiation regulations in the United States, numerous data related on the biokinetics and dosimetry of radionuclides have been accumulated since the issue of ICRP Publication 23 in 1975. In addition, there was a strong need to establish reference characteristics for children or other subgroups of the population. In 1984, the ICRP organized the new task group on the Reference Man and started to update its reference values. As a result, the ICRP issued, in 2002, ICRP Publication 89, "Basic anatomical and physiological data for use in radiological protection: reference values," which provides reference values for both male and female in six age groups:

newborn, 1-year, 5-years, 10-years, 15-years, and adult (ICRP 2002). The reference values given in this publication are based on three general sources of data: 1) anatomical and physiological data not published previously by the ICRP; 2) data in ICRP Publication 66 on the human respiratory, ICRP Publication 70 on the skeleton, and ICRP Publication 88 on the human embryo and fetus; and 3) data in Publication 23, which is still regarded effective for radiation dosimetry purposes (ICRP 1994; ICRP 1995; ICRP 2001; ICRP 2002). The name of the Reference Man was also changed to the Reference Person. According to ICRP Publication 103, the Reference Person is defined as a hypothetical person who uses the reference data of the Reference Male and the Reference Female in ICRP Publication 89 for the calculation of organ or tissue equivalent doses and effective dose (ICRP 2007). Most nations in the world currently use the concept of the Reference Person for dosimetry calculations, and the NRC is also considering adopting the Reference Person for the basis of dose calculation in radiation regulations (USNRC 2014e). The differences between the MEI and the representative person are summarized in Table 10 (ICRP 1975; USNRC 1991c; ICRP 2007).

Table 10. Comparison between the Reference Man and the Reference Person.

Factors	Reference Man	Reference Person
Aim	Occupational dose calculation, especially for internal radiation exposure	Dose calculation for an individual
Institute	ICRP	ICRP
Definition	Hypothetical person with the anatomical and physiological characteristics of an average individual that is used in calculations assessing internal dose	Hypothetical person who uses the reference data of the Reference Male and the Reference Female in ICRP Publication 89 for the calculation of organ or tissue equivalent dose and effective dose
Characteristics	Typical radiation worker	Idealized person for whom the organ or tissue equivalent doses are calculated by averaging the corresponding doses of the Reference Male and Reference Female
Anatomical criteria	ICRP Publication 23	ICRP Publication 89
Dose coefficients	ICRP Publications 30, 32	ICRP Publications 56, 67, 69, 71, 72

U.S. Regulation and guidance for occupational dose calculation

Occupational dose indicates the dose received by an individual during work involving radiation or radioactivity. Occupational dose does not include the dose received from background radiation, from medical treatments, and as a member of the public. In terms of NPP operation, regulations for occupational dose are mainly provided by 10 CFR Part 20 (USNRC 1991c). The dose to the whole body in 10 CFR Part 20 is the weighted dose for tissues or organs (USNRC 2014e). The dose to the whole body is expressed as

TEDE, which represents a single value obtained from summation of external and internal radiation dose (USNRC 1991c). However, the dose equivalents for the lens of the eye, the skin, and the extremities are not applicable to this summation rule, and they have their own separate limits. In terms of external dosimetry, 10 CFR Part 20 stresses that if the external radiation dose is measured by an external personal monitoring device, such as a thermoluminescent dosimeter (TLD), the effective dose equivalent should be the deep-dose equivalent (DDE) (USNRC 1991c). This DDE also indicates the highest dose that the body receives. The occupational dose limits for adults, minors, and embryos or fetuses are described in 10 CFR Part 20.1201 through Part 20.1208, and these are summarized in Table 11 (USNRC 1991c; Cember and Johnson 2009).

Table 11. Occupational dose limits in 10 CFR Part 20.

Exposed subjects	Limits
Whole body	50 mSv y^{-1}
Lens of the eye	150 mSv y^{-1}
Any other organ or tissue	500 mSv y^{-1}
Extremities (Limbs, fingers, etc.)	500 mSv y^{-1}
Skin (averaged over 10 cm^2)	500 mSv y^{-1}
Minors	10 % of adult dose
Embryo or fetus	5 mSv [a]

[a] Over the gestation period.

In terms of internal dosimetry, the regulations in 10 CFR Part 20 are primarily based on several ICRP reports: first, ICRP Publication 30, “Limits for intakes of radionuclides by workers,” including its four parts, four supplements, and index, which

were published during the period of 1979 through 1988 and second, ICRP Publication 32, "Limits for inhalation of radon daughters by workers" (ICRP 1979b; 1979a, 1980, 1981a, c, b, 1982b, a, 1988; USNRC 2014e). These ICRP Publications are used to guide the calculation of the inhalation values for the ALIs and DACs of radionuclides in Appendix B to 10 CFR Part 20. In particular, Table 1, "Occupational Values," in Appendix B to 10 CFR Part 20 provides the values for oral ingestion ALI, inhalation ALI, and DAC for the Reference Man (USNRC 1991c). All of these values assumed that the radioactive material taken into the body is an aerosol with an activity median aerodynamic diameter of 1 μm. In addition, this has three clearance half-times from the respiratory system. These are Class D, Class W, and Class Y, which indicate retention times in the pulmonary region of the lung, approximately days, weeks or years, respectively (USNRC 1991c).

The NRC has issued several regulatory guides which provide methods for calculating and reporting occupational dose to facilitate the licensee's compliance with the requirements of 10 CFR Part 20. Among these guides, NRC RG 8.34, "Monitoring criteria and methods to calculate occupational radiation doses," describes in detail how to calculate external and internal radiation doses for workers and how to report those results to the NRC (USNRC 1992). All methods described in NRC RG 8.34 are based on ICRP Publication 26 and 30, which are also the basis of 10 CFR Part 20. According to 10 CFR Part 20.1502, "Conditions requiring individual monitoring of external and internal occupational dose," external radiation monitoring for a worker is required if the external occupational dose is expected to exceed 10% of the dose limits or an individual enters a high or very high radiation area (USNRC 1991c). Internal radiation monitoring for a

worker is also required if the intake of radioactive material is likely to exceed 0.1 ALI during the year or if committed effective dose equivalent (CEDE) is expected to surpass the 0.5 mSv for the occupationally-exposed minor or declared pregnant woman (USNRC 1991c). As stated earlier, the dose to the whole body in the NRC regulations is expressed as TEDE. This TEDE is obtained through the summation of DDE for external radiation exposure and CEDE for internal radiation exposure.

According to 10 CFR Part 20.1502, individual monitoring devices are required to monitor external radiation exposure to workers, whereas they are not necessary for internal dose monitoring (USNRC 1991c). The requirements of external radiation exposure are applied individually depending on the types of external radiation dose: DDE to the whole body, shallow dose equivalent (SDEs) to the skin or extremities, and lens dose equivalent to the eye (LDE). To determine whether external dose monitoring is required or not, the prospective occupational dose to the worker for the year is used as the basis for evaluating the likelihood of doses exceeding 10% of the dose limit (USNRC 1992). If the conditions of radiation exposure to the worker change during the year, the necessity for external dose monitoring should be reevaluated. Although some doses are measured below 10% of the dose limit, all doses measured during the monitoring must be recorded since the monitoring was conducted to meet 10 CFR Part 20.1502 (USNRC 1992).

In terms of external dose monitoring, 10 CFR Part 20 provides three dose limits: 1) 50 mSv of DDE to the whole body, 2) 500 mSv of SDE to the skin or extremities, and 3) 150 mSv of LDE to the lens of the eye (USNRC 1991c). The definitions of these doses are described in 10 CFR Part 20.1003, “Definitions” (USNRC 1991c). First, the DDE is

defined as the external whole-body dose equivalent at a tissue depth of 1 cm (1000 mg cm^{-2}). The SDE is defined as the dose to the skin or extremities at a tissue depth of 0.007 cm (7 mg cm^{-2}). Finally, the LDE is defined as the dose to the eye at a tissue depth of 0.3 cm (300 mg cm^{-2}). As mentioned earlier, external radiation dose is generally measured by individual monitoring devices, such as TLDs. The monitoring device for the whole body dose is typically placed on the chest of a worker since the chest is expected to receive the highest dose during work when the whole body is exposed homogeneously (USNRC 1992). If the radiation field is not homogeneous, the monitoring device should be placed on the location where a specific part is expected to receive the highest dose. In some cases when the radiation field is highly heterogeneous, two or more dosimeters are used to determine the DDE to the whole body. Several dosimeter algorithms were developed from previous studies to calculate the real DDE, but NRC RG 8.34 recommends using the highest value as the DDE among several measured values of multiple dosimeters (USNRC 1992; Kim and Kong 2010). If the local radiation exposure to extremities, such as fingers, is expected, an extremity dosimeter may be used to monitor the SDE to the extremities. However, in a normal case when exposure is uniform, especially for NPPs, the SDE to extremities can be replaced by the SDE measured by a chest dosimeter, and additional extremity monitoring would not be necessary unless the whole body dose exceeds its limit (USNRC 1992; Kim and Kong 2013).

In internal dose monitoring, the dose is expressed as CEDE. The CEDE is defined as the 50-year effective dose equivalent for intake of radioactive material into the body through inhalation, ingestion, absorption through the skin, accidental injection, or

introduction through a wound (USNRC 1992). To calculate the total CEDE, all occupational intakes through these modes of intake are added over the yearly time period. According to 10 CFR Part 20.1204, "Determination of internal exposure," it is assumed that the inhalation of one ALI, or an exposure of 2,000 DAC-hours, results in a CEDE of 50 mSv for radionuclides that have their ALIs or DACs based on this limit (USNRC 1991c). To calculate the CEDE for the inhalation of radioactive material, NRC RG 8.34 provides the five acceptable methods: 1) use of Federal Guidance Report No.11, 2) use of stochastic inhalation ALIs from 10 CFR Part 20, 3) use of DACs from 10 CFR Part 20, 4) use of ICRP Publication 30, and 5) use of individual or material-specific information (USNRC 1992).

First, Federal Guidance Report No. 11, "Limiting values of radionuclide intake and air concentration and dose conversion factors for inhalation, submersion, and ingestion," provides the CEDE per unit intake by inhalation in its Table 2.1 (USEPA 1988). Second, Table 1 in Appendix B to 10 CFR Part 20 provides ALIs for individual radionuclides (USNRC 1991c). The CEDE can be calculated using these ALI values using Equation (4):

$$CEDE_i = \frac{5 \times I_i}{ALI_i} \tag{4}$$

where $CEDE_i$ is the CEDE for radionuclide i (rem), I_i is the intake of radionuclide i by inhalation during the calendar year (μCi), ALI_i is the value of the stochastic inhalation ALI from Column 2 of Table 1 in Appendix B to 10 CFR Part 20 (μCi), and 5 is the CEDE

from intake of 1 ALI (rem) (USNRC 1992). The stochastic ALIs in Appendix B to 10 CFR Part 20 were derived to result in a risk due to irradiation of organs and tissues, comparable to the risk related with the DDE to the whole body of 50 mSv. The non-stochastic ALIs were derived to prevent deterministic effects, such as immediate damage to tissues (USNRC 1991c). For an intake of more than one radionuclide, the total CEDE is calculated as the sum of the CEDEs for all radionuclides. Third, Table 1 in Appendix B to 10 CFR Part 20 also provides stochastic DACs for individual radionuclides (USNRC 1991c). The CEDE can be calculated using these DAC values using Equations (5) and (6):

$$DAC_i = \frac{ALI_i}{2.4 \times 10^9} \quad (5)$$

$$CEDE_i = \frac{5 \times C_i t}{2000 \times DAC_i} \quad (6)$$

where DAC_i is the stochastic DAC for radionuclide i (μCi cm^{-3}), ALI_i is the stochastic ALI for radionuclide i (μCi), 2.4×10^9 is the volume of air inhaled by a worker in a work year (cm^3), $CEDE_i$ is the CEDE from radionuclide i (rem), C_i is the airborne concentration of radionuclide i to which the worker is exposed (μCi cm^{-3}), t is the duration of the exposure (h), 2000 is the number of hours in a working year (8h×5d×50w), and 5 is the CEDE from annual intake of 1 ALI or 2000 DAC-hours (rem) (USNRC 1992). According to 10 CFR Part 20.1204, when radionuclides exist as a mixture in the air, certain radionuclides in the mixture can be disregarded if the following conditions are met: 1) the concentration of radionuclide ignored is below 10% of its DAC, 2) the sum of these percentages for all of the radionuclides ignored in the mixture does not surpass 30%, and 3) the licensee uses

the total activity of the mixture in demonstrating compliance with the dose limits and monitoring requirements (USNRC 1991c). Fourth, the supplements to ICRP Publication 30 provide the weighted CEDE to target organs or tissues per unit intake by inhalation (ICRP 1979a, 1981b, 1982a, b). According to ICRP Publication 30, the weighted CEDE is defined as the product of the weighting factor and the CEDE for a specified organ or tissue (ICRP 1979b). Finally, it is possible to use individual or material-specific information to calculate the CEDE for the inhalation of radioactive material. 10 CFR Part 20.1204 allows the licensee to use specific data on the physical and biochemical properties of the radionuclides taken into the body or the behavior of the material in an individual without prior NRC approval (USNRC 1991c). However, since this approach requires the licensee to do significantly more work, it is not generally used for small routine inhalation (USNRC 1992).

Ingestion is another main intake pathway for radioactive material into the body. Contrary to inhalation, only one ingestion ALI is provided for each radionuclide, and this value is used for all chemical forms of that radionuclide (USNRC 1992). Similar to the methods used for calculating the CEDE by inhalation, NRC RG 8.34 provides the four acceptable methods: 1) use of Federal Guidance Report No.11, 2) use of stochastic ingestion ALIs from 10 CFR Part 20, 3) use of ICRP Publication 30, and 4) use of individual or material-specific information (USNRC 1992). First, Federal Guidance Report No. 11 provides the CEDE per unit intake by ingestion in its Table 2.2 (USEPA 1988). Second, Table 1 in Appendix B to 10 CFR Part 20 provides stochastic ingestion ALIs for individual radionuclides (USNRC 1991c). The CEDE can also be calculated

using Equation (4) using the ingestion ALI value instead of the inhalation ALI value. Third, the supplements to ICRP Publication 30 list the weighted CEDE to target organs or tissues per unit intake by ingestion (ICRP 1979a, 1981b, 1982a, b). Finally, the use of individual or material-specific information is also used to calculate the CEDE for the ingestion of radioactive material.

To demonstrate compliance with dose limits for an individual organ or tissue in 10 CFR Part 20.1201, the licensee should calculate the organ-specific CEDE (USNRC 1991c). The calculation of organ-specific CEDE is necessary only if the CEDE exceeds 10 mSv or if an over exposure has occurred (USNRC 1992). If the CEDE is below 10 mSv and no overexposure has occurred, the organ-specific CEDE cannot surpass its limit of 500 mSv (USNRC 1992). To calculate the organ-specific CEDE, NRC RG 8.34 provides the five acceptable methods: 1) use of Federal Guidance Report No.11, 2) use of non-stochastic inhalation ALIs from 10 CFR Part 20, 3) use of non-stochastic DACs from 10 CFR Part 20, 4) use of ICRP Publication 30, and 5) use of individual or material-specific information (USNRC 1992).

First, Federal Guidance Report No. 11 provides the organ-specific exposure-to-dose conversion factors in Table 2.1 for inhalation and Table 2.2 for ingestion (USEPA 1988). Then, the organ-specific CEDE can be calculated using these factors and Equation (7):

$$CEDE_T = I_i \times DCF_i \times 3.7 \times 10^6 \qquad (7)$$

where $CEDE_T$ is the CEDE to the tissue or organ from radionuclide *i* (rem), I_i is the intake of radionuclide *i* (μCi), DCF_i is the dose conversion factor for radionuclide i from Table 2.1 or 2.2 in Federal Guidance Report No. 11 (Sv Bq^{-1}), and 3.7×10^6 is the conversion factor to convert from Sv Bq^{-1} to rem μCi^{-1} (USNRC 1992). Second, Table 1 in Appendix B to 10 CFR Part 20 provides non-stochastic inhalation ALIs for individual radionuclides (USNRC 1991c). The CEDE can also be calculated using Equation (4) and both the non-stochastic ALI value instead of the stochastic ALI value and 50 rem instead of 5 rem (USNRC 1992). Third, the corresponding DAC in Appendix B to 10 CFR Part 20 for the non-stochastic ALI of a radionuclide can be used to calculate the organ-specific CEDE (USNRC 1991c). The CEDE can also be calculated using Equation (6) and both these DAC values instead of the stochastic inhalation DAC value and 50 rem instead of 5 rem (USNRC 1992). Fourth, the supplements to ICRP Publication 30 provide the CEDE in target organs or tissues per unit intake to significantly exposed organs (ICRP 1979a, 1981b, 1982a, b). Finally, the use of individual or material-specific information is also allowed in calculating the organ-specific CEDE.

In terms of calculating the dose from intakes through wounds or skin, the intake by skin absorption of airborne radioactive materials is generally not required since, in practice, it is trivial compared to the intake from inhalation (USNRC 1992). However, if the radionuclide is in a solution containing dissolved radioactive material, such as tritium, absorption through the skin is not negligible and the calculation of its dose is necessary. Internal radiation exposure from tritium at pressurized, heavy-water reactors, such as the

CANDU reactor, accounts for 20-40% of the total occupational dose (Kim and Kong 2012b).

10 CFR Part 20.2106, "Records of individual monitoring results," requires that the occupational dose recording be done on NRC Form 5 or equivalent (USNRC 1991c). This recording form is based on annual dose. The monitoring period for occupational dose, especially for NPPs, is generally from January 1 to December 31 (USNRC 1992). Detailed instructions for filling out NRC Form 5 are given in NRC RG 8.7, Revision 2, "Instructions for recording and reporting occupational exposure data" (USNRC 2005a). According to 10 CFR Part 20.1202, "Compliance with requirements for summation of external and internal doses," the TEDE is obtained through the summation of external radiation dose (DDE) and internal radiation dose (CEDE) (USNRC 1991c). However, this summation method is not applicable to the SDE to the skin or extremities or to the eye dose equivalent. The minimum value of recording on NRC Form 5 is 0.1 mSv, since such small values are trivial relative to the dose limits (USNRC 1992).

Computer codes for occupational dose calculation

Contrary to public dose estimation, computer codes are not often used to calculate the occupational dose, especially for external radiation exposure, because most workplace conditions are well known and personal monitoring devices provide relatively precise radiation measurements. In NPPs, the whole body dose is generally determined from personal dosimeter readings which provide the DDE unless the worker has intakes of radioactive materials through inhalation, ingestion, or skin absorption. Regarding

compliance with the requirements of SDE in 10 CFR Part 20, the NRC provides the computer code VARSKIN to be used to calculate skin dose from both beta and gamma radiation sources (USNRC 2015b). NRC NUREG/CR-6918, "VARSKIN 5: A computer code for skin contamination dosimetry," describes how to use VARSKIN for the calculation of tissue dose at various depths as the result of skin contamination (USNRC 2014h). The area of 10 cm^2 is set as a default for skin dose calculations in VARSKIN to meet requirements of 10 CFR Part 20.1201. However, the user can select any dose-averaging area for specific uses. In general, the level of radiation exposure to a worker during normal NPP operation is low enough to neglect the SDE to the skin. Therefore, skin dose calculations are rare and the use of VARSKIN is also not common in the field at NPPs (Kim and Kong 2012a).

In terms of internal radiation exposure, several computer codes are used for bioassay program in NPPs. The code for internal dosimetry (CINDY) is one of the codes used for bioassay calculation, which is based on the respiratory system and GI tract models in ICRP Publication 30 (ICPR 1979b; 1979a, 1980, 1981a, c, b, 1982b, a, 1988; PNNL 2009). This code can be used to calculate intake using curve-fitting of bioassay data; committed organ, tissue, and effective dose equivalents; and bioassay projections. Although CINDY was originally developed by the Pacific Northwest National Laboratory in the late 1980s and early 1990s with the U.S. DOE funding, it is now commercially available through Canberra Nuclear, Incorporation (PNNL 2009). Since the CINDY code is based on ICRP Publication 30, it is not widely used in the world except the United States. One of the other popular codes is the integrated modules for bioassay analysis (IMBA)

developed by the Public Health England in the UK to evaluate bioassay data and calculate internal radiation doses from intakes of radioactive materials (PHE 2015). This code is based on the Human Respiratory Tract Model (HRTM) in ICRP Publication 66, biokinetic models in ICRP Publication 78, and the GI tract model in ICRP Publication 30 (ICRP 1979a, 1980, 1981a, c, b, 1982b, a, 1988, 1994; 1997). The code consists of a series of independent modules that are used for specific tasks and communicate with each other though input and output files. IMBA is widely used in the world for bioassay calculations due to its technical basis of 1990 ICRP Recommendation on Radiological Protection (ICRP Publication 60).

METHODOLOGY

Risk attributable to ionizing radiation

Risk assessment is associated with evaluating probability and impact of individual detriment. Probability is the likelihood of a certain outcome. Impact is the severity or effect of the outcome. Risk assessment is widely used in evaluating the health effects of radiation exposure. Most industrial activities including the operation of NPPs involve low radiation doses. For several decades, an effort has been exerted to discover the dose-response curve at low radiation doses, but nothing can be said with certainty about low dose levels (Romerio 2002). There are still no definite data to prove the risk of low doses, below approximately 100 mSv. However, it is generally acceptable to linearly extrapolate the risk at low doses to estimate the probability of incurring cancer or genetic effects attributable to radiation. The linear-non-threshold (LNT) model implies that a given increment in dose will produce a directly proportionate increment in risk (ICRP 2007). ICRP considers that the LNT model is the best practical approach to managing risk from radiation exposure (ICRP 2007).

In addition to the LNT model, a dose and dose-rate effectiveness factor (DDREF) is used for risk estimation to provide realistic estimates of the effects of radiation exposure at low annual doses. The DDREF is defined as a factor that simplifies the lower biological effectiveness (per unit of dose) of radiation exposures at low doses and low dose rates as compared with exposures at high doses and high dose rates (ICRP 2007). This means that the risk estimates at low doses are not based on a simple extrapolation of effects seen with the range of high doses, such as 1 to 2 Sv, and the cancer risk at these low doses and low

dose rates is judged to be reduced by the value of the factor ascribed to DDREF. A DDREF of 2 is currently used by the ICRP to derive the nominal risk coefficients for stochastic effects (ICRP 1991, 2007).

The risk is expressed as the excess risk per Sv. This excess risk can be calculated using either excess relative risk (*ERR*) or excess absolute risk (*EAR*) models (ICRP 2007). The ERR is the proportional increase in risk over the background absolute risk in the absence of exposure, and it is expressed as

$$ERR = \frac{R_{Radiation}}{R_{Background}} - 1. \quad (8)$$

The EAR is the additional risk above the background absolute risk given by

$$EAR = R_{Radiation} - R_{Background} \quad (9)$$

where $R_{Radiaiton}$ is the risk of radiation exposure and $R_{Background}$ is the risk of background. A weighting of the ERR and EAR was used to estimate radiation risk for each organ: ERR:EAR weights of 0:100% for breast and bone marrow, 100:0% for thyroid and skin, 30:70% for lung, and 50:50% for all others organs (ICRP 2007).

On the basis of the above assumptions, the detriment-adjusted risk is calculated using Equation (10) (ICRP 2007).

$$R_i = R_F + R_{NF}$$
$$R_D = [(R_I \times q + R_I \times (1-q) \times ((1-q_{\min}) \times q + q_{\min})] \times l \quad (10)$$

where R_I is nominal risk coefficient (cases per 10,000 person per Sv) from fatal (R_F) and non-fatal (R_{NF}) cancers, R_D is the detriment-adjusted risk, q is the lethality, $((1-q_{\min}) \times q + q_{\min})$ is the weight given to non-fatal cancers, and l is relative cancer-free life lost. Here, q_{min} is the minimum weight for non-fatal cancers. Nominal risk coefficients R_I result from averaging sex and age-at-exposure lifetime risk estimates in representative populations. Radiation risk estimates are derived for incidence data for specific tumor sites using dose response data from the Japanese Life Span Study (LSS). These risk estimates were reduced by a factor of 2 with respect to DDREF. These nominal risks are calculated for each site of interest and summed to give the population total nominal risk. A summary is provided in Table A.4.1 in ICRP Publication 103 (ICRP 2007).

A detriment-adjusted risk R_D indicates the probability of the occurrence of a stochastic effect, modified to allow for the different components of the detriment to express the severity of the consequence. ICRP also notes that these risk coefficients are based on direct human epidemiological data. ICRP risk coefficients for stochastic effects after radiation exposure at low dose rates are provided in Table A.4.4 in ICRP Publication 103. Table 12 presents a summary of the detriment-adjusted nominal risk for the adult and whole populations, which will be used in future analysis to assess the excess risk for members of the public and occupational workers in NPPs (ICRP 2007). It is important to mention that the ICRP does not provide an uncertainty of the risk coefficients shown in Table 12; however, the uncertainty was estimated by the United Nations Scientific Committee on

the Effects of Atomic Radiation to be a factor of approximately 2 (a ± 2a) (UNSCEAR 2000; ICRP 2005; 2007). The excess risk incurred by a given dose will be given as

$$ER = D \times R_D \tag{11}$$

where ER is the excess risk, D is the radiation dose, R_D is the detriment-adjusted nominal risk coefficient as shown in Table 12.

Table 12. Detriment-adjusted nominal risk coefficients (per mSv) for stochastic effects after radiation exposure at low dose rates as determined in ICRP Publication 103.

Exposed subject	Whole population[a]	Adult population[b]
Cancer	5.5×10^{-5}	4.1×10^{-5}
Heritable effects	0.2×10^{-5}	0.1×10^{-5}
Total	5.7×10^{-5}	4.2×10^{-5}

[a] Whole population, including members of all ages.
[b] Adult age population (18 – 64 years).

Determination of a single source

The definition of dose constraints indicates that a dose constraint originates from a single source. Therefore, the first step is to determine a single source, which will be a target for applying a dose constraint prior to the determination of numerical value for that constraint. A single source can be a physical subject or an activity which results in radiation exposure (ICRP 2007). In general, a single source is determined broadly, such as a NPP or a radiation generating device or routine operation in nuclear facilities such as

non-destructive inspection, to prevent subdividing the source for the sake of avoiding the protection requirements. If a single source for applying a dose constraint is a radiation source with very small activity or a small part of the work, which results in trivial radiation exposure, it is not necessary to implement a dose constraint for that source, because its radiation level is already low enough to be neglected.

In terms of the dose to members of the public living around NPPs, a single source can be a single unit, which has only one reactor, or a site, which includes multiple units. In the United States, NPPs generally consist of two units on a single site, and the number of NPP units in some single sites is expected to increase up to four (WNA 2015b). In some nations, the number of NPP units increases up to six due to the difficulty of finding new NPP sites (Kong et al. 2014; WNA 2015a). In this situation, if a single unit is determined as a single source for establishing a dose constraint for the public, several single sources can exist in a single site. It will be difficult to control each single source independently since, for some NPPs, two units share some radioactive waste processing facilities, and it is difficult to distinguish the origin of the radioactive wastes from two units (USNRC 2013d). This also makes it difficult to control the public dose. The way to prevent these problems is to assume that a site is a single source. Furthermore, the dose standards for NPP operation in NRC regulations are based on a particular site. Specifically, the NRC reports that the EPA dose standards, which are stipulated in 10 CFR Part 20.1301(e), are applicable for up to four reactors at one location or a single site (USNRC 2014b). That is, the EPA dose standards are also based on a particular site.

A single site is also a better choice as a single source for an occupational dose constraint. In NPPs, it is common that a worker conducts maintenance jobs at two units, and he or she occasionally conducts radiation work at other units in the same site. In these circumstances, if a single unit is chosen as a single source for establishing a dose constraint for workers, the control of occupational doses will be more complicated since it is almost impossible to distinguish the origin of the radiation exposure of workers from each unit.

In some ways, the determination of a single site as a single source can cause confusion in that it is difficult to distinguish between the concepts of source-related dose constraints and individual dose limits. However, the targets for the implementation of dose constraints and dose limits are different. For the source-related approach, a dose constraint focuses on the control of the radiation source to prevent exceeding a dose constraint, such as the use of other low-level radiation sources. For the individual-related approach, a dose limit focuses on control of the exposed person to prevent exceeding a dose limit, which might require the replacement of the worker.

In practice, selecting a single site as a single source for the dose constraint is expected to facilitate management of occupational doses in NPPs. It is also possible to select certain jobs, which result in very high radiation exposure to workers, as additional single sources for dose constraints to reduce their dose levels. However, this approach can cause complexity in the management of occupational doses in NPPs. In addition, since NPPs usually issue a radiation work permit (RWP) or special work permit (SWP) prior to conducting radiation work to protect the worker from receiving unnecessary doses, selecting certain jobs as additional single sources with a single site is not necessary.

Dose distribution related to the operation of nuclear power plants

NPP licensees are required, under the regulations in 10 CFR Part 20, to evaluate public and occupational doses and to report their doses to the NRC annually. Therefore, NPPs conduct routine monitoring of public and occupational radiation exposure to demonstrate compliance with dose limits. All records of these results are maintained until the NRC terminates their NPP licensees. Since the data of public and occupational doses are available, the analysis of past and present dose records can be a useful tool to evaluate the trend of radiation exposures related to NPP operation. These data will also be helpful to understand a dose distribution and to determine numerical values for dose constraints. In terms of public exposure, the NRC publishes the annual report of radioactive effluents from NPPs as a series of NUREG/CR-2907, which includes the dose to members of the public due to liquid and gaseous effluents (USNRC 2013d). For occupational exposure, the annual report of occupational radiation exposure at commercial nuclear power reactors and other facilities, which is a series of NUREG-0713, is issued by the NRC (USNRC 2014d). In these reports, TEDE is mainly used to express the dose quantity and to demonstrate compliance with the dose limits since TEDE or effective dose is intended for ALARA and regulatory purposes (ICRP 2007). Therefore, it is also appropriate to use TEDE as a dose quantity for dose constraints.

Dose estimation for members of the public basically originates from the release of radioactive effluents from NPPs. Other dose contributions by natural radiation background, medical exposure, etc., are not considered in the estimation of public dose. Therefore, the activities of radionuclides discharged from NPPs to the environment in liquid and in

gaseous effluents were analyzed to understand their effects on maximum annual doses to the public resulting from these effluent releases. The NRC has started providing annual summary reports, NRC NUREG/CR-2907, "Radioactive effluents from nuclear power plants," since 2007, and the 2007, 2008, and 2009 summary reports are currently available (USNRC 2011b, 2012e, 2013d). All the radioactive effluents data are based on a single NPP site. Common radionuclides in radioactive effluents will be identified through the analysis, and their contributions to the total activities of radioactive effluents will also be evaluated. Since some NPPs have applied a specific method to reduce their radioactive effluents, this approach and its effect on the decrease of total activities of radioactive effluents will be analyzed.

The annual TEDE to the public estimated for each NPP site was used for the analysis of dose distributions for public exposure during the years 2007-2009. According to the NRC annual summary reports for the public dose, NRC NUREG/CR-2907, the total number of NPP sites in the United States is currently 65, and its dose level is approximately 10^{-4} mSv (USNRC 2011b, 2012e, 2013d). Since outliers that have extremely high or low values occur due to the dose estimation using the specific exposure scenario for each NPP site, and the sample size is not large enough to neglect the influence by these outliers, the standard deviation of estimated doses to members of the public is normally larger than its average. In terms of probability distribution, measurements of radiation exposure are often treated as a skewed distribution (ICRP 2006). The skewed distribution is common when mean values are low, variances are large, and values of samples cannot be negative (Limpert et al. 2001). In particular, this skewed distribution

indicates that the tail on the right side is longer than the left side. Such skewed distributions are often well fitted by the lognormal distribution (Limpert et al. 2001). In this study, the lognormal distribution, which takes the natural logarithm and applies the normal distribution, is used for the analysis of dose distribution for public exposure.

For occupational exposure, the annual total body doses to workers in NPPs were used for the analysis of occupational dose distributions during the years 2003-2012. According to the NRC annual reports for occupational exposure, NUREG-0713, approximately 200,000 workers were reported as the total number of monitored individuals in NPPs, and their dose levels were approximately 1-2 mSv (USNRC 2014d). Although the NRC annual reports provide the information on occupational doses in NPPs, they do not provide the raw data of occupational exposure due to requirement of maintaining the privacy of individual workers. Therefore, the exact figures of occupational dose and the standard deviation of the dose distribution cannot be known in the reports. However, this study requested the NRC to provide the raw data of occupational TEDEs during the years 2003-2012 through the Freedom of Information Act (FOIA). Therefore, instead of using the data from the NRC annual reports, the data of occupational exposures at commercial NPPs for years between 2003 and 2012 were used in this study to represent the typical doses to the occupational workers (USNRC 2016). The dose distribution for each year was calculated by counting the number of individuals in each dose range.

The occupational dose distribution with regards to three classifications of individuals in NPPs was the focus of this study. The total number of monitored individuals during each year includes office workers, maintenance workers, contract individuals, and

even visitors to NPPs. However, since most office workers and visitors are not involved in radiation work in the field, their exposure levels are usually extremely low, and their radiation exposure may not be measured. Therefore, the number of individuals whose radiation exposure is measurable is needed to analyze the actual radiation dose due to radiation work in NPPs. According to the NRC annual report for occupational exposure, the percent of individuals with measurable dose to the total number of monitored individuals in NPPs is approximately 40-50% (USNRC 2014d). It means that only half of individuals in NPPs received a dose greater than the dose recording level (0.1 mSv). In reality, the total number of monitored individuals is overcounted because of transient individuals. Transient individuals are defined as individuals who worked at more than one nuclear facility during the year (USNRC 2014d). Each NPP licensee reports the occupational doses received by transient individuals at each facility separately to the NRC. These data look like separate individual doses although they belong to the same individual. To obtain the actual dose information, it is necessary to combine these dose records per individual. Contrary to the occupational exposure data obtained directly from the NRC through the FOIA, the data provided by the NRC annual reports for occupational exposure, taking into account transient individuals, were used to analyze the occupational dose distributions at the U.S. NPPs (USNRC 2004b, 2005b, 2006, 2007b, 2008a, 2010b, 2011a, 2012c, 2013c; 2014d). The reason this study used the data provided by the NRC annual report is that it is necessary to access the individual's identification information, such as a social security number, to combine occupational dose records per individual. However, the privacy act restricts disclosure of personally identifiable records maintained by federal

agencies (USDOJ 1974). Therefore, this study used the data from the NRC annual report for analyzing the occupational dose distributions with regards to transient individuals.

In the analysis of individual dose, a dose distribution can be expressed in two general ways. First, the weighting of a dose distribution can be understood as a probability (ICRP 2006). For example, the probability that the annual dose to the particular individual does not exceed 20 mSv is 0.95 or 95%. This approach is useful to define a specific individual in the unknown exposed population, such as a maximally exposed individual (MEI). The other approach is the use of the fraction of the exposed population receiving a dose within that interval (ICRP 2006). In this case, the percentile can be used, such as 'the 95^{th} percentile of the dose distribution is 20 mSv or 20 mSv is the dose that is not exceeded in 95% of the population.' The second approach is often used to quantify the specific value in the dose distribution. It is also useful to divide parts or pieces of the distribution. The distribution can be divided elaborately by computing percentiles of the distribution. These quantities make it possible to find the specific value that is relatively extreme, either small or large in the distribution (Walpole et al. 2011). For instance, the 95^{th} percentile divides the highest 5% from the bottom 95%. In this study, the percentile method was used to determine dose constraints in the dose distributions of public and occupational exposure at NPPs.

To determine dose constraints using the dose distributions of public and occupational exposure at NPPs, appropriate numerical percentiles are necessary to find the specific value that is relatively inequitable in the distribution. However, there are no documents which provide the quantitative definition of inequitable or equitable doses.

According to the ICRP, inequity or equity is associated with the ethical concept of distributive justice which refers to social fairness (Lochard 2014). In this study, the two concepts of the EPA's MEI and the ICRP's representative person, which are used to define the reference individual for estimating the dose to members of the public, are used for a quantitative approach for determining inequitable dose. This is the first trial to quantify the "inequitable dose."

In particular, the EPA used the 99.99^{th} percentile of the distribution to establish the concept of the MEI for the public dose estimation (Hawkins 1991). The EPA considered the exposure level of the public could even fall to the 99.99^{th} percentile of the distribution, although it is a very conservative assumption. Another interpretation might be that there is only a 0.01% chance that the radiation exposure is unexpected in the distribution. From the perspective of the optimization process, this highest 0.01% can be an inequitable radiation exposure in the distribution. On the other hand, the ICRP used the 95^{th} percentile of the distribution to introduce the concept of the representative person that characterizes a hypothetical individual who is more highly exposed in the population (ICRP 2006). The ICRP observed that the exposure level of the public could even fall to the 95^{th} percentile of the distribution, and the highest 5% can be an inequitable radiation exposure in the distribution. Although two organizations, the EPA and the ICRP, provided numerical percentiles of the exposure levels in the individual exposure distribution for a prospective assessment of dose to individuals, it is necessary to consider what numerical percentile in the individual exposure distribution is appropriate for the percentile of dose constraints at NPPs, taking into account national features.

RESULTS OF ANALYSIS

Dose for members of the public living around nuclear power plants

There are 104 commercial nuclear power plants (NPPs) located 65 sites in the United States, which are displayed in Fig. 4 (NRC 2012; USNRC 2013d). All NPPs monitor their radioactive effluents discharged to the environment regularly, and these monitoring results are reported to the Nuclear Regulatory Commission (NRC) annually. In particular, 10 CFR Part 50.36a requires that NPP licensees submit a report, including the activity of each of the main radionuclides discharged to the environment in liquid and in gaseous effluents during the past 12 months and maximum annual doses to the public or the maximally exposed individual (MEI) resulting from these effluent releases (USNRC 1956a). All these data are used to demonstrate licensee compliance with applicable regulations, such as the NRC design objectives and dose limits and the Environmental Protection Agency (EPA) dose standards.

In 2006, the NRC issued the staff requirement memorandum, SECY-06-0212, which directed the NRC staff to summarize the data of radioactive effluents, including the dose to members of the public, from all NPP sites (USNRC 2012f). In compliance with the SRM to SECY-06-0212, the NRC staff has started providing annual summary reports, NRC NUREG/CR-2907, since 2007, and the 2007, 2008, and 2009 summary reports are currently available (USNRC 2011b, 2012e, 2013d). These effluent data are also available through the NRC database website (http://www.reirs.com/effluent/) (USNRC 2015c).

1 Browns Ferry, Alabama (3)
2 Farley, Alabama (2)
3 Palo Verde, Arizona (3)
4 Arkansas Nuclear, Arkansas (2)
5 Diablo Canyon, California (2)
6 San Onofre, California (2)[a]
7 Millstone, Connecticut (2)
8 Crystal River, Florida (1)
9 St. Lucie, Florida (2)
10 Turkey Point, Florida (2)
11 Edwin I. Hatch, Georgia (2)
12 Vogtle, Georgia (2)
13 Braidwood, Illinois (2)
14 Byron, Illinois (2)
15 Clinton, Illinois (1)
16 Dresden, Illinois (2)
17 LaSalle, Illinois (2)
18 Quad Cities, Illinois (2)
19 Duane Arnold, Iowa (1)
20 Wolf Creek, Kansas (1)
21 River Bend, Louisiana (1)
22 Waterford, Louisiana (1)
23 Calvert Cliffs, Maryland (2)
24 Pilgrim, Massachusetts (1)
25 Cook, Michigan (2)
26 Palisades, Michigan (1)
27 Fermi, Michigan (1)
28 Monticello, Minnesota (1)
29 Prairie Island, Minnesota (2)
30 Grand Gulf, Mississippi (1)
31 Callaway, Missouri (1)
32 Cooper, Nebraska (1)
33 Fort Calhoun, Nebraska (1)
34 Seabrook, New Hampshire (1)
35 Hope Creek, New Jersey (1)
36 Oyster Creek, New Jersey (1)
37 Salem, New Jersey (2)
38 Fitzpatrick, New York (1)
39 Ginna, New York (1)
40 Indian Point, New York (2)
41 Nine Mile Point, New York (2)
42 Brunswick, North Carolina (2)
43 McGuire, North Carolina (2)
44 Harris, North Carolina (1)
45 Davis-Besse, Ohio (1)
46 Perry, Ohio (1)
47 Beaver Valley, Pennsylvania (2)
48 Limerick, Pennsylvania (2)
49 Peach Bottom, Pennsylvania (2)
50 Susquehanna, Pennsylvania (2)
51 Three Mile Island, Pennsylvania (1)
52 Oconee, South Carolina (3)
53 Robinson, South Carolina (1)
54 Summer, South Carolina (1)
55 Catawba, South Carolina (2)
56 Sequoyah, Tennessee (2)
57 Watts Bar, Tennessee (1)
58 Comanche Peak, Texas (2)
59 South Texas Project, Texas (2)
60 Vermont Yankee, Vermont (1)[a]
61 North Anna, Virginia (2)
62 Surry, Virginia (2)
63 Columbia, Washington (1)
64 Kewaunee, Wisconsin (1)[a]
65 Point Beach, Wisconsin (2)

The number inside the parentheses indicates the number of units in a nuclear power plant.
[a]The nuclear power plants are currently closed.

Fig. 4. Operating NPPs in the United States.

Analysis of radioactive effluents released from nuclear power plants

In general, NPP operation produces various radioactive materials. The amount of radioactive materials is expressed as activity whose units are Bq or Ci. Most of these radioactive sources originate from the fission of nuclear fuel. During the normal operation of NPPs, a small fraction of these radioactive sources is generally discharged to the environment through liquid and gaseous effluents. These radioactive effluents typically originate from several sources: 1) fission of tramp uranium which is dissolved from exposed fuel rods and plated out onto the structure of the coolant system, 2) leaks from failed fuel rods, 3) diffusion of radioactive gases through intact fuel rods, 4) activation of materials in the reactor cooling water, and 5) erosion of activated materials from pipes, valves, pumps and ancillary equipment (NRC 2012). According to a series of NRC NUREG/CR-2907, some radionuclides are typically reported in effluent releases from NPPs, and these are shown in Table 13 (USNRC 2013d).

Table 13. Common radionuclides in radioactive effluents released from NPPs[a].

Type	Category	Radionuclide
Liquid	Mixed fission and activation products	Iron (^{55}Fe) Cobalt (^{58}Co, ^{60}Co) Cesium (^{134}Cs, ^{137}Cs) Manganese (^{54}Mn) Zirconium (^{95}Zr) Niobium (^{95}Nb) Iodine (^{131}I, ^{133}I, ^{135}I)
	Tritium	Hydrogen (^{3}H)
	Dissolved and entrained noble gases	Krypton (^{85}Kr, ^{85m}Kr, ^{87}Kr, ^{88}Kr) Xenon (^{131}Xe, ^{133}Xe, ^{133m}Xe, ^{135}Xe, ^{135m}Xe)
	Gross alpha	Total alpha activity
Gaseous	Fission and activation gases	Krypton (^{85}Kr, ^{85m}Kr, ^{87}Kr, ^{88}Kr) Xenon (^{131}Xe, ^{131m}Xe, ^{133}Xe, ^{133m}Xe, ^{135}Xe, ^{135m}Xe) Argon (^{41}Ar)
	Iodine and halogens	Iodine (^{131}I, ^{132}I, ^{133}I, ^{134}I, ^{135}I) Bromine (^{82}Br)
	Particulates	Cobalt (^{58}Co, ^{60}Co) Cesium (^{134}Cs, ^{137}Cs) Chromium (^{51}Cr) Manganese (^{54}Mn) Niobium (^{95}Nb)
	Tritium	Hydrogen (^{3}H)
	Gross alpha	Total alpha activity from all alpha emitters

[a] (USNRC 2013d).

For analysis purposes, the median value among all NNPs for each radionuclide was used to indicate the activity of radioactive effluents in this study. The median is the number separating the higher 50% of the data sample from the lower 50%. That is, the amount of radioactive effluents discharged from half of the NPPs will be greater than the median and half will be lower than the median. Since the average can be distorted by

outliers that have extremely high or low values, the median is used to indicate the typical value of the activity of radioactive effluents. The NRC annual summary reports also use the median to estimate the typical value of NPP effluents (USNRC 2011b, 2012e, 2013d). The median will be zero if the majority of NPPs did not detect radionuclides.

The activity of radioactive effluents released from NPPs, including both boiling water reactor (BWRs) and pressurized water reactors (PWRs) in the United States, was analyzed using the data from the NRC annual summary reports and the database website. The total activities of radioactive effluents discharged from BWRs and PWRs in 2007 were 1.30×10^{12} Bq and 2.16×10^{13} Bq, respectively (USNRC 2011b, 2015c). The activity in effluents released from PWRs was approximately 17 times higher than that from BWRs. This phenomenon results from the higher production of tritium in PWRs. In general, boric acid is added to the PWR reactor coolant system as a chemical shim to control reactivity, and tritium is produced primarily from neutron capture by ^{10}B (Martin 2009). On the other hand, tritium production is significantly lower in BWRs since boric acid is not used to control reactivity in BWRs. The activities in liquid and gaseous effluents discharged from BWRs in 2007 were 7.55×10^{8} Bq and 1.30×10^{12} Bq, respectively (USNRC 2011b, 2015c). The percentages of each activity from liquid and gaseous effluents of the total activity were 0.06% and 99.94%, respectively. Therefore, most of the activity in radioactive effluents released from BWRs in 2007 resulted from gaseous effluents. For PWRs, the activities in liquid and gaseous effluents released in 2007 were 2.04×10^{13} Bq and 1.22×10^{12} Bq, respectively (USNRC 2011b, 2015c). The percentages of each activity from liquid and gaseous effluents of the total activity were 94.35% and 5.65%, respectively.

Therefore, contrary to BWRs, most of the activity in radioactive effluents released from PWRs in 2007 resulted from liquid effluents. The main reason why BWRs and PWRs have different contributions of liquid and gaseous effluents to the total activity is the recycling of liquid waste at BWRs (USNRC 2013d). Many BWRs in the United States reuse either some or all of their liquid waste (USNRC 2013d). To decontaminate liquid radioactive wastes, BWRs use several waste treatment methods such as evaporation, chemical precipitation, and ion exchange. Furthermore, BWRs do not use boron in the reactor coolant unlike PWRs, and this also contributes to a reduction in liquid effluents from BWRs (USNRC 2013d). The main radionuclide, which primarily contributed to the activity in both liquid and gaseous effluents, was tritium. The activities of radionuclides in liquid and gaseous effluents for 2007 are displayed in Table 14 (USNRC 2011b). Table 15 shows the total activities depending on the types of effluents and reactors. The percent contribution of each activity from liquid and gaseous effluents of the total activity in BWRs and PWRs are also shown in Table 15.

Table 14. Activities of nuclides in gaseous and liquid effluents released from NPPs during 2007[a].

Effluent	Nuclide	BWR (Bq)				PWR (Bq)			
		Median	Minimum	Maximum	%[b]	Median	Minimum	Maximum	%[b]
Liquid[b]	^{58}Co	3.47×10^{5}	7.84×10^{2}	5.18×10^{8}	0.05	1.36×10^{8}	1.00×10^{6}	4.18×10^{9}	0.00
	^{60}Co	6.62×10^{6}	3.12×10^{3}	6.14×10^{9}	0.88	8.21×10^{7}	5.14×10^{5}	1.39×10^{9}	0.00
	^{134}Cs				0.00	2.20×10^{5}	1.49×10^{4}	1.97×10^{8}	0.00
	^{137}Cs	2.46×10^{5}	5.92×10^{2}	4.51×10^{8}	0.03	4.63×10^{6}	3.81×10^{3}	2.84×10^{8}	0.00
	^{55}Fe				0.00	1.07×10^{8}	3.28×10^{6}	1.67×10^{9}	0.00
	^{3}H	7.47×10^{8}	8.40×10^{5}	4.85×10^{12}	99.04	2.04×10^{13}	5.51×10^{12}	6.55×10^{13}	100.00
	^{131}I				0.00	2.54×10^{4}	2.54×10^{4}	4.44×10^{7}	0.00
	Total	7.55×10^{8}	8.44×10^{5}	4.85×10^{12}	100.00	2.04×10^{13}	5.51×10^{12}	6.55×10^{13}	100.00
Gaseous[c]	^{58}Co	1.23×10^{5}	4.85×10^{4}	2.62×10^{6}	0.00				0.00
	^{60}Co	5.33×10^{6}	1.14×10^{4}	1.39×10^{8}	0.00				0.00
	^{3}H	9.58×10^{11}	1.21×10^{11}	5.11×10^{12}	73.56	1.17×10^{12}	1.10×10^{11}	2.81×10^{13}	95.53
	^{131}I	3.02×10^{7}	4.29×10^{4}	1.58×10^{9}	0.00	1.61×10^{5}	7.36×10^{2}	1.21×10^{9}	0.00
	^{85}Kr				0.00	9.62×10^{8}	2.36×10^{7}	2.65×10^{12}	0.08
	^{133}Xe	2.21×10^{11}	1.03×10^{10}	1.35×10^{13}	16.93	5.25×10^{10}	9.95×10^{5}	1.22×10^{13}	4.31
	^{135}Xe	1.24×10^{11}	1.71×10^{9}	5.92×10^{12}	9.51	1.02×10^{9}	1.49×10^{3}	4.88×10^{11}	0.08
	Total	1.30×10^{12}	1.33×10^{11}	2.45×10^{13}	100.00	1.22×10^{12}	1.10×10^{11}	4.35×10^{13}	100.00

[a] (USNRC 2011b).
[b] Blank cells are where the majority of NPPs did not detect a radionuclide.
[c] Percentages are based on median values.

Table 15. Total activities of liquid and gaseous effluents released from NPPs during 2007[a].

Effluents	BWR		PWR	
	Bq[b]	%	Bq[b]	%
Liquid	7.55×10^{8}	0.06	2.04×10^{13}	94.35
Gaseous	1.30×10^{12}	99.94	1.22×10^{12}	5.65
Total	1.30×10^{12}	100.00	2.16×10^{13}	100.00

[a] (USNRC 2011b).
[b] Median values are used to indicate the activities of radioactive effluents.

In 2008, the total activities in gaseous and liquid radioactive effluents released from BWRs and PWRs were 1.31×10^{12} Bq and 2.07×10^{13} Bq, respectively (USNRC 2012e, 2015c). The activity in effluents discharged from PWRs was approximately 16 times higher than that from BWRs due to the higher production of tritium in PWRs. The activities in liquid and gaseous effluents released from BWRs in 2008 were 9.07×10^{10} Bq and 1.22×10^{12} Bq, respectively (USNRC 2012e, 2015c). The percentages of each activity from liquid and gaseous effluents of the total activity were 6.92% and 93.08%, respectively. Therefore, most of activity in radioactive effluents released from BWRs in 2008 resulted from gaseous effluents. For PWRs, the activities in liquid and gaseous effluents released in 2008 were 1.95×10^{13} Bq and 1.22×10^{12} Bq, respectively (USNRC 2012e, 2015c). The percentages of each activity from liquid and gaseous effluents of the total activity were 94.11% and 5.89%, respectively. Therefore, contrary to BWRs, most of the activity in radioactive effluents released from PWRs in 2008 resulted from liquid effluents. Tables 16 and 17 show the activities depending on the types of effluents and reactors during 2008 (USNRC 2012e).

Table 16. Activities of nuclides in gaseous and liquid effluents released from NPPs during 2008[a].

Effluent	Nuclide	BWR (Bq)				PWR (Bq)			
		Median	Minimum	Maximum	%[b]	Median	Minimum	Maximum	%[b]
Liquid[b]	^{58}Co				0.00	2.21×10^{8}	2.41×10^{6}	4.85×10^{9}	0.00
	^{60}Co	7.18×10^{6}	1.24×10^{6}	7.88×10^{8}	0.01	1.14×10^{8}	4.11×10^{6}	1.01×10^{9}	0.00
	^{134}Cs				0.00	4.66×10^{5}	2.59×10^{3}	1.67×10^{8}	0.00
	^{137}Cs				0.00	8.07×10^{6}	3.70×10^{4}	5.00×10^{8}	0.00
	^{55}Fe				0.00	1.02×10^{8}	2.57×10^{7}	3.01×10^{9}	0.00
	^{3}H	9.07×10^{10}	4.18×10^{7}	4.70×10^{12}	99.99	1.95×10^{13}	5.88×10^{12}	6.14×10^{13}	100.00
	^{131}I				0.00	7.47×10^{4}	1.72×10^{4}	8.18×10^{7}	0.00
	Total	9.07×10^{10}	4.30×10^{7}	4.70×10^{12}	100.00	1.95×10^{13}	5.88×10^{12}	6.14×10^{13}	100.00
Gaseous[c]	^{58}Co	2.96×10^{4}	3.01×10^{3}	3.81×10^{7}	0.00	3.77×10^{4}	1.62×10^{3}	3.66×10^{7}	0.00
	^{60}Co	2.19×10^{6}	1.39×10^{4}	2.22×10^{6}	0.00				0.00
	^{137}Cs	7.25×10^{4}	1.28×10^{4}	6.99×10^{6}	0.00				0.00
	^{3}H	1.11×10^{12}	1.33×10^{8}	5.48×10^{12}	90.74	1.13×10^{12}	9.77×10^{10}	3.22×10^{13}	92.98
	^{131}I	2.69×10^{7}	1.31×10^{6}	2.19×10^{9}	0.00	6.96×10^{5}	2.16×10^{1}	3.20×10^{8}	0.00
	^{85}Kr				0.00	7.70×10^{9}	2.53×10^{7}	1.94×10^{12}	0.63
	^{133}Xe	2.96×10^{10}	8.44×10^{8}	1.05×10^{13}	2.42	7.70×10^{10}	1.17×10^{8}	2.31×10^{13}	6.32
	^{135}Xe	8.33×10^{10}	1.26×10^{6}	3.27×10^{12}	6.83	8.84×10^{8}	5.99×10^{5}	4.63×10^{11}	0.07
	Total	1.22×10^{12}	9.79×10^{8}	1.92×10^{13}	100.00	1.22×10^{12}	9.78×10^{10}	5.76×10^{13}	100.00

[a] (USNRC 2012e).

[a] Blank cells are where the majority of NPPs did not detect a radionuclide.

[b] Percentages are based on median values.

Table 17. Total activities of liquid and gaseous effluents released from NPPs during 2008[a].

Effluents	BWR		PWR	
	Bq[b]	%	Bq[b]	%
Liquid	9.07×10^{10}	6.92	1.95×10^{13}	94.11
Gaseous	1.22×10^{12}	93.08	1.22×10^{12}	5.89
Total	1.31×10^{12}	100.00	2.07×10^{13}	100.00

[a] (USNRC 2012e).
[b] Median values are used to indicate the activities of radioactive effluents.

In 2009, the activities in radioactive effluents released from BWRs and PWRs were 1.26×10^{12} Bq and 2.21×10^{13} Bq, respectively (USNRC 2013d, 2015c). The activity in effluents discharged from PWRs was approximately 18 times higher than that from BWRs due to the higher production of tritium in PWRs. The activities in liquid and gaseous effluents released from BWRs in 2009 were 3.43×10^{11} Bq and 9.14×10^{11} Bq, respectively (USNRC 2013d, 2015c). The percentages of each activity from liquid and gaseous effluents of the total activity were 27.31% and 72.69%, respectively. Therefore, activity in gaseous effluents was much higher than that in liquid effluents. For PWRs, the activities in liquid and gaseous effluents released in 2009 were 2.06×10^{13} Bq and 1.54×10^{12} Bq, respectively (USNRC 2013d, 2015c). The percentages of each activity from liquid and gaseous effluents of the total activity were 93.03% and 6.97%, respectively. Therefore, contrary to BWRs, most of the activity in radioactive effluents released from PWRs in 2009 resulted from liquid effluents. The activities of radionuclides in liquid and gaseous effluents for 2009 are displayed in Table 18 (USNRC 2013d). Table 19 shows the total activities depending on the types of effluents and reactors.

Table 18. Activities of nuclides in gaseous and liquid effluents released from NPPs during 2009[a].

Effluent	Nuclide	BWR (Bq)				PWR (Bq)			
		Maximum	Median	Minimum	%[b]	Maximum	Median	Minimum	%[b]
Liquid[b]	^{58}Co				0.00	4.07×10^{9}	1.90×10^{8}	3.46×10^{5}	0.00
	^{60}Co	1.72×10^{9}	2.44×10^{7}	3.96×10^{6}	0.01	4.88×10^{8}	7.70×10^{7}	2.76×10^{6}	0.00
	^{134}Cs				0.00	1.41×10^{8}	5.85×10^{4}	7.84×10^{3}	0.00
	^{137}Cs				0.00	7.22×10^{8}	7.77×10^{6}	3.89×10^{4}	0.00
	^{55}Fe				0.00	2.03×10^{9}	5.14×10^{7}	8.33×10^{6}	0.00
	^{3}H	3.89×10^{12}	3.43×10^{11}	4.48×10^{5}	99.99	7.66×10^{13}	2.06×10^{13}	3.56×10^{12}	100.00
	^{131}I				0.00	4.18×10^{7}	4.96×10^{4}	1.76×10^{3}	0.00
	Total	3.89×10^{12}	3.43×10^{11}	4.41×10^{6}	100.00	7.66×10^{13}	2.06×10^{13}	3.56×10^{12}	100.00
Gaseous[c]	^{58}Co	1.47×10^{7}	2.02×10^{5}	1.23×10^{4}	0.00	2.86×10^{8}	9.99×10^{3}	4.03×10^{3}	0.00
	^{60}Co	1.14×10^{8}	3.41×10^{6}	1.83×10^{5}	0.00				0.00
	^{137}Cs	9.32×10^{6}	1.34×10^{4}	4.26×10^{3}	0.00				0.00
	^{3}H	4.40×10^{12}	7.25×10^{11}	6.48×10^{10}	79.35	2.75×10^{13}	1.51×10^{12}	5.14×10^{10}	97.96
	^{131}I	9.77×10^{8}	1.67×10^{7}	1.62×10^{6}	0.00	8.84×10^{8}	1.92×10^{5}	2.10×10^{-3}	0.00
	^{85}Kr				0.00	1.55×10^{12}	8.25×10^{8}	4.37×10^{7}	0.05
	^{133}Xe	2.50×10^{13}	4.85×10^{10}	5.96×10^{8}	5.30	1.26×10^{14}	2.92×10^{10}	7.99×10^{6}	1.89
	^{135}Xe	6.51×10^{12}	1.40×10^{11}	2.66×10^{9}	15.34	3.10×10^{12}	1.40×10^{9}	4.14×10^{5}	0.09
	Total	3.60×10^{13}	9.14×10^{11}	6.80×10^{10}	100.00	1.58×10^{14}	1.54×10^{12}	5.14×10^{10}	100.00

[a] (USNRC 2013d).
[b] Blank cells are where the majority of NPPs did not detect a radionuclide.
[c] Percentages are based on median values.

Table 19. Total activities of liquid and gaseous effluents released from NPPs during 2009.

Effluents	BWR		PWR	
	Bq[b]	%	Bq[b]	%
Liquid	3.43×10^{11}	27.31	2.06×10^{13}	93.03
Gaseous	9.14×10^{11}	72.69	1.54×10^{12}	6.97
Total	1.26×10^{12}	100.00	2.21×10^{13}	100.00

[a] (USNRC 2013d).
[b] Median values are used to indicate the activities of radioactive effluents.

During the years 2007-2009, all the effluents discharged from NPPs met the NRC safety limits for radioactive effluents (USNRC 2011b, 2012e, 2013d). Although all the activities in effluents released from both BWRs and PWRs were lower than the limits, the total activities in radioactive effluents released from PWRs were approximately 16-18 times higher than those from BWRs. The main reason for this phenomenon was the higher production of tritium in PWRs due to the use of boric acid to control reactivity. For BWRs, most of the activities resulted from gaseous effluents, while the contribution of gaseous effluents released from PWRs to the total activities in radioactive effluents were minor compared to liquid effluents. Since BWRs recycle some or all of their liquid radioactive waste, the contribution of liquid effluents to the total activity in radioactive effluents was limited. This is called “zero-release.” For many decades, a “zero-release” approach for liquid effluents has been very popular among BWRs (USNRC 2013d). Due to this approach, BWRs could reduce the activity and the dose from radioactive effluents very effectively. However, as a result of analysis for effluent data, it was found that the total activities in liquid effluents released from BWRs have increased gradually in recent years, which is shown in Fig. 5. Tritium was the primary source for this gradual increase of total

activity in liquid effluents. Based on 10 years of operating experience, these zero-release BWRs have shown a steady increase in the contribution of tritium to occupational exposure at some NPP sites (USNRC 2013d). As liquid radioactive wastes have been reused in BWRs, the tritium concentration in the water has accumulated over time (USNRC 2013d). Therefore, previously zero-release BWRs have begun to discharge liquid effluents again to avoid build-up of their tritium inventory in the facility (Harris 2011). The activities of radionuclides in liquid and gaseous effluents during the years 2007-2009 are displayed in Table 20 (USNRC 2011b, 2012e; 2013d).

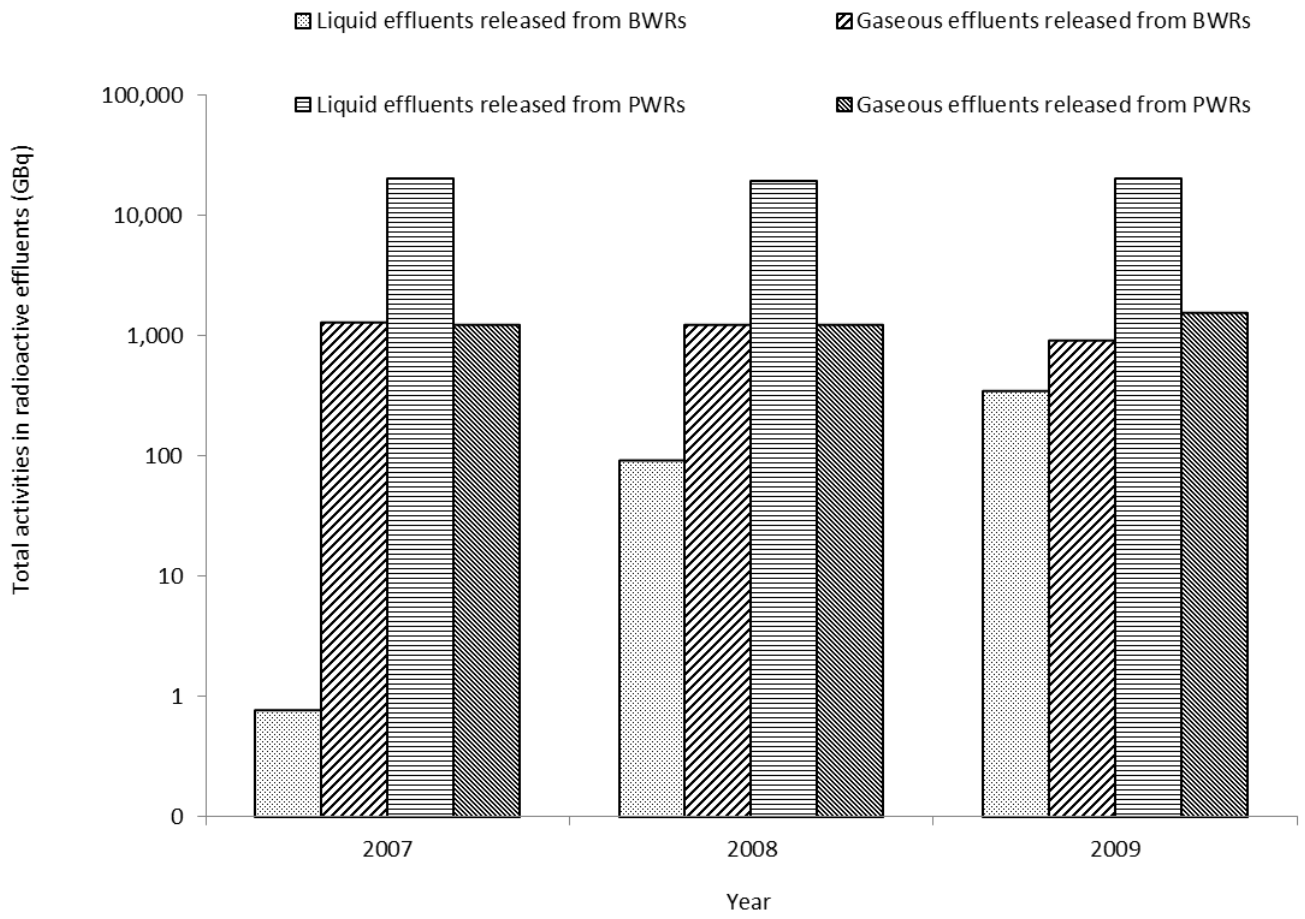

Fig. 5. Activities in liquid and gaseous effluents released from NPPs during the years 2007-2009 (USNRC 2011b, 2012e; 2013d).

Table 20. Total activities of liquid and gaseous effluents released from NPPs during 2007-2009[a].

Year	Effluents	BWR (Bq)				PWR (Bq)			
		Median	Minimum	Maximum	%[a]	Median	Minimum	Maximum	%[a]
2007	Liquid	7.55×10^{8}	8.44×10^{5}	4.85×10^{12}	0.06	2.04×10^{13}	5.51×10^{12}	6.55×10^{13}	94.35
	Gaseous	1.30×10^{12}	1.33×10^{11}	2.45×10^{13}	99.94	1.22×10^{12}	1.10×10^{11}	4.35×10^{13}	5.65
	Total	1.30×10^{12}	1.33×10^{11}	2.94×10^{13}	100.00	2.04×10^{13}	5.62×10^{12}	1.09×10^{14}	100.00
2008	Liquid	9.07×10^{10}	4.30×10^{7}	4.70×10^{12}	6.92	1.95×10^{13}	5.88×10^{12}	6.14×10^{13}	94.11
	Gaseous	1.22×10^{12}	9.79×10^{8}	1.92×10^{13}	93.08	1.22×10^{12}	9.78×10^{10}	5.76×10^{13}	5.89
	Total	1.31×10^{12}	1.02×10^{9}	2.39×10^{13}	100.00	2.07×10^{13}	5.98×10^{12}	1.19×10^{14}	100.00
2009	Liquid	3.43×10^{11}	4.41×10^{6}	3.89×10^{12}	27.31	2.06×10^{13}	3.56×10^{12}	7.66×10^{13}	93.03
	Gaseous	9.14×10^{11}	6.80×10^{10}	3.60×10^{13}	72.69	1.54×10^{12}	5.15×10^{10}	1.58×10^{14}	6.97
	Total	1.26×10^{12}	6.80×10^{10}	3.99×10^{13}	100.00	2.21×10^{13}	3.61×10^{12}	2.35×10^{14}	100.00

[a] (USNRC 2011b, 2012e; 2013d).
[b] Percentages are based on median values.

Analysis of the dose for members of the public living around nuclear power plants

To ensure compliance with the requirements of Appendix I to 10 Code of Federal Regulation (CFR) Part 50, all NPP licensees regularly estimate the public dose from radioactive effluents released from NPPs (USNRC 1956b). This estimation is based on both measurements and theoretical models, including 1) real measurements of radioactive effluents in the environment, 2) models for the dispersion and dilution of radioactive materials in the environment, 3) models for the incorporation of radioactive materials into animals, plants, and soil, and 4) biokinetic models for the human uptake and metabolism of radioactive materials (USNRC 2013d). These models intend to estimate the dose to a MEI who may be exposed to the highest activities from effluents. Therefore, the estimated dose is often much higher than the actual dose to the residents living around NPPs. All NPP licensees have established their procedures for estimating the public dose according to the NRC Regulatory Guide (RG) 1.109, and those procedures are combined into their offsite dose calculation manuals (ODCMs) (USNRC 1977b, 2013d).

Similar to the analysis of effluent data, median values are also used to demonstrate the typical dose to members of the public in this study. The median is the midpoint of the data, and it is used to minimize the influence by outliers that have extremely high or low values. The median is also used in the NRC annual summary reports to estimate the typical value of dose to members of the public (USNRC 2011b, 2012e, 2013d).

The total effective dose equivalent (TEDE) is often called “total body dose,” and it results from radioactive effluents released from NPPs in the United States. The TEDE

was analyzed using the data from the NRC annual summary reports and the database website. The median value of TEDE for members of the public living around NPPs in 2007 was 9.69×10^{-4} mSv for all NPPs, specifically 3.33×10^{-4} mSv for BWRs and 1.07×10^{-3} mSv for PWRs (USNRC 2011b, 2015c). The median TEDE for members of the public was much lower than the EPA dose standard for a site, 0.25 mSv y^{-1}. In comparison with the NRC dose limit, the TEDE for members of the public in 2007 was only 0.1% of the annual TEDE limit for members of the public, 1 mSv y^{-1}. The maximum and minimum TEDEs for the public in 2007 were reported to be 1.72×10^{-1} mSv and 1.98×10^{-7} mSv, respectively. The 95th percentile of the dose distribution for members of the public was 5.59×10^{-2} mSv. The average radiation dose received by an individual from the natural radiation background in the United States is approximately 3.1 mSv y^{-1}. The TEDE to the public due to radioactive effluents from NPPs in 2007 was only 0.03% of what the average person receives each year from natural background radiation (USNRC 2011b). The TEDE for members of the public living around NPPs in 2007 are displayed in Fig. 6 (USNRC 2011b). Fig. 6 also indicates the NRC dose limit and EPA dose standard for the purpose of comparison.

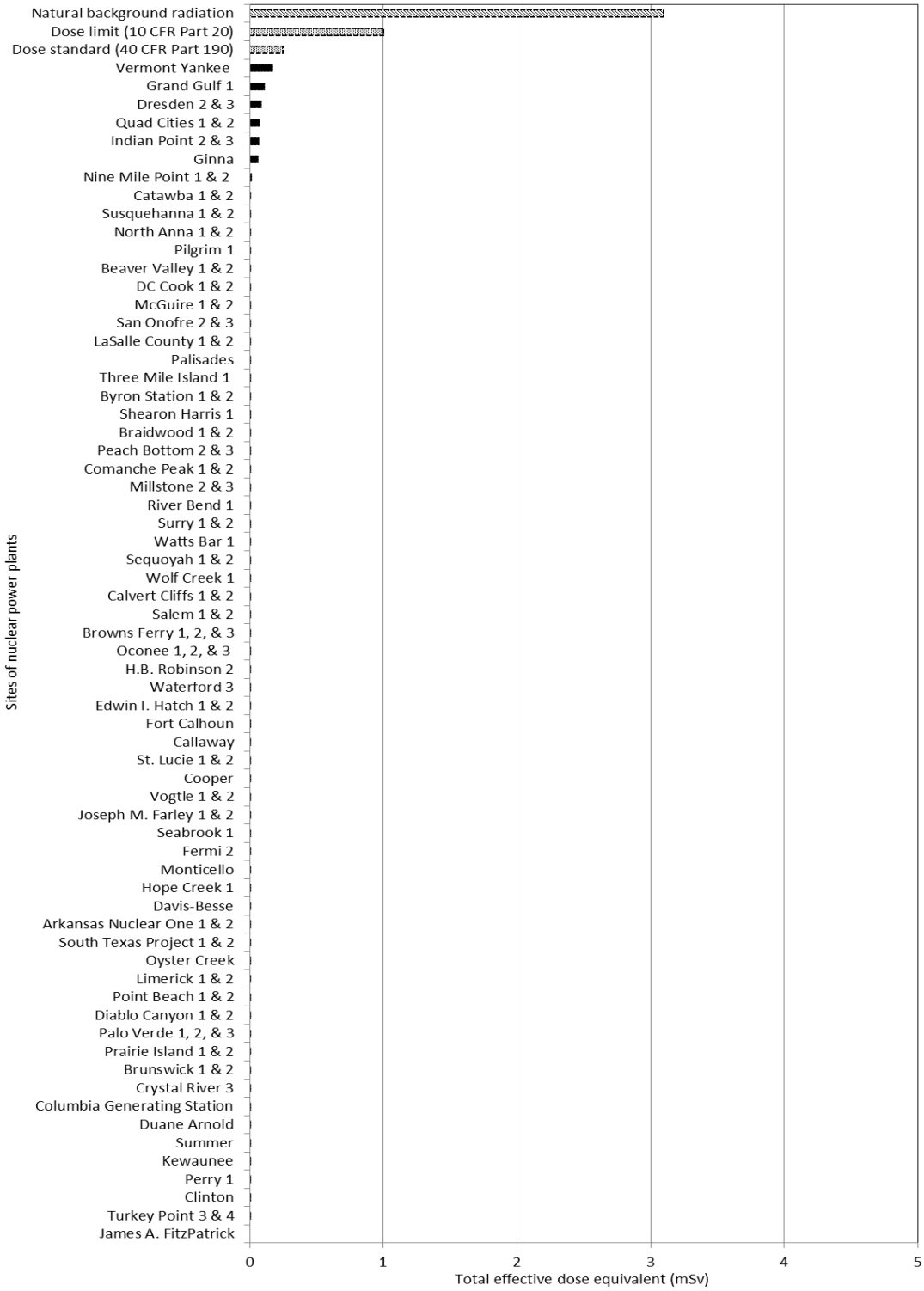

Fig. 6. Total effective dose equivalent for members of the public living around NPPs during 2007 (USNRC 2011b).

In 2008, the median value of TEDE for members of the public living around NPPs was 7.17×10^{-4} mSv for all NPPs, specifically 4.11×10^{-4} mSv for BWRs and 7.56×10^{-4} mSv for PWRs (USNRC 2012e, 2015c). The TEDE for members of the public was much lower than the EPA dose standard for a site, 0.25 mSv y^{-1}, accounting for only 0.29%. In comparison with the NRC dose limit, the TEDE for members of the public in 2008 was only 0.07% of the annual TEDE limit, 1 mSv y^{-1}. The maximum and minimum TEDEs for the public in 2008 were reported to be 1.63×10^{-1} mSv and 1.63×10^{-7} mSv, respectively. The 95th percentile of the dose distribution for members of the public was 4.50×10^{-2} mSv. Compared to the average radiation dose, 3.1 mSv y^{-1}, received by an individual in the United States, the TEDE to the public due to radioactive effluents from NPPs in 2008 was only 0.02% of what the average person receives each year from natural background radiation (USNRC 2012e). The TEDE for members of the public living around NPPs in 2008 are shown in Fig. 7, including the NRC dose limit and EPA dose standard for the purpose of comparison (USNRC 2012e).

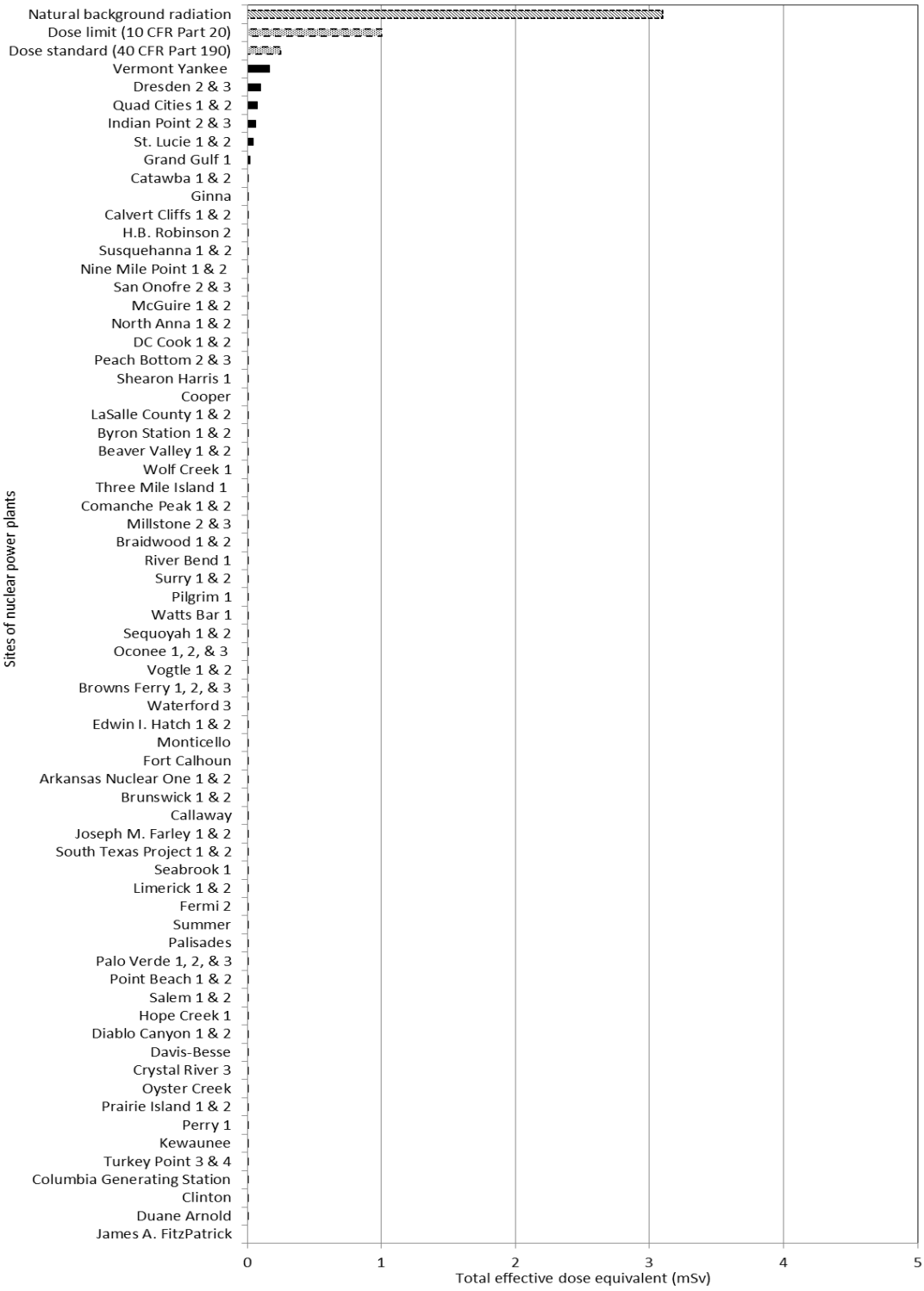

Fig. 7. Total effective dose equivalent for members of the public living around NPPs during 2008 (USNRC 2012e).

In 2009, the median value of the TEDE for members of the public living around NPPs was 5.42×10^{-4} mSv for all NPPs, specifically 4.19×10^{-4} mSv for BWRs and 6.39×10^{-4} mSv for PWRs (USNRC 2013d, 2015c). This TEDE for members of the public was much lower than the EPA dose standard for a site, 0.25 mSv y^{-1}, accounting for only 0.22%. In comparison with the NRC dose limit, the TEDE for members of the public in 2009 was only 0.05% of the annual TEDE limit for members of the public, 1 mSv y^{-1}. The maximum and minimum TEDEs for the public in 2009 were reported to be 1.68×10^{-1} mSv and 1.66×10^{-7} mSv, respectively. The 95^{th} percentile of the dose distribution for members of the public was 5.21×10^{-2} mSv. Compared to the average radiation dose, 3.1 mSv y^{-1}, received by an individual in the United States, the TEDE to the public due to radioactive effluents from NPPs in 2009 was only 0.02% of what the average person receives each year from natural background radiation (USNRC 2013d). The TEDE for members of the public living around NPPs in 2009 are shown in Fig. 8 with the NRC dose limit and EPA dose standard for the purpose of comparison (USNRC 2013d).

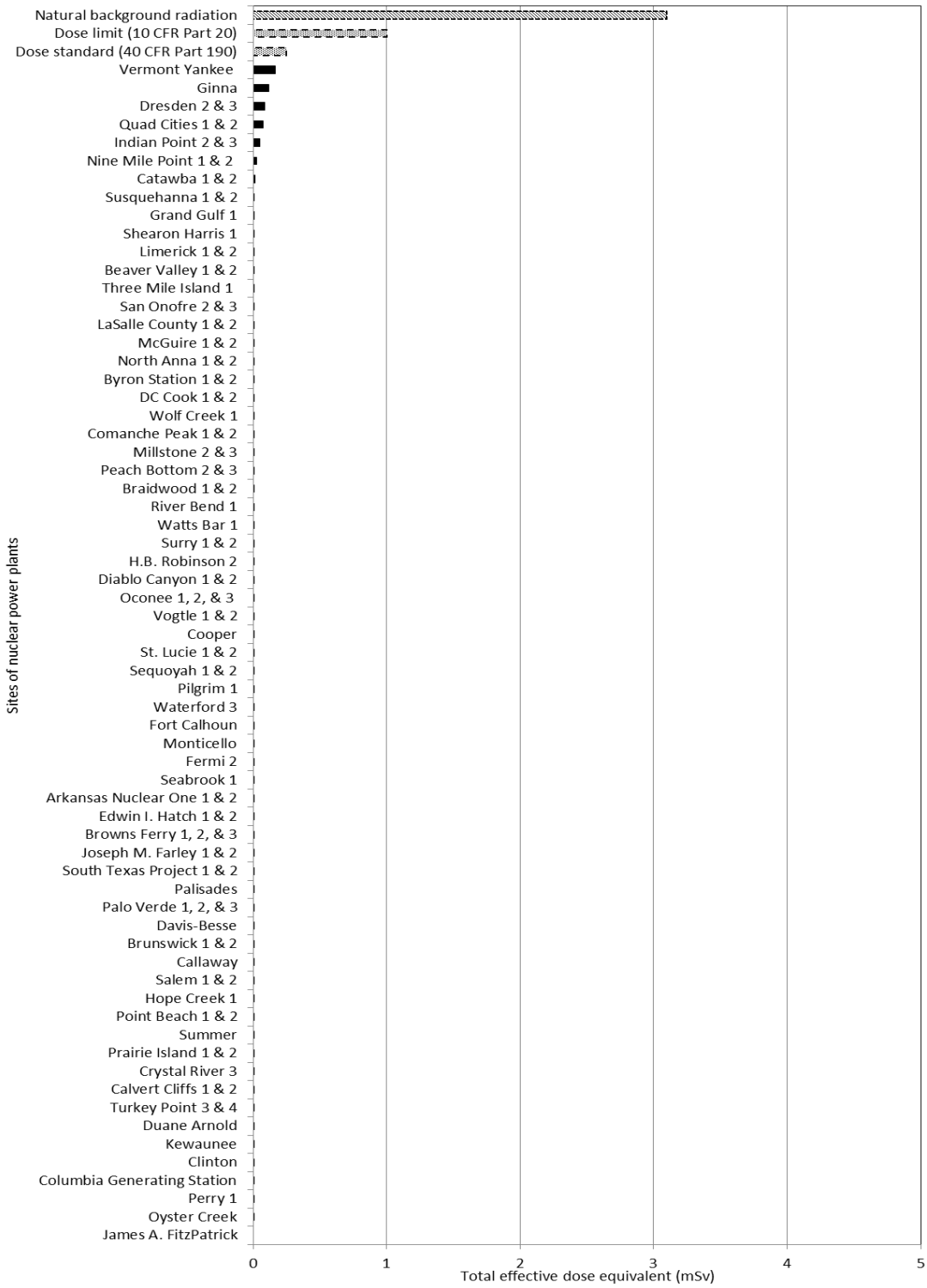

Fig. 8. Total effective dose equivalent for members of the public living around NPPs during 2009 (USNRC 2013d).

During the years 2007-2009, the TEDEs for members of the public due to radioactive effluents released from NPPs met both the EPA dose standard and the NRC dose limit (USNRC 2011b, 2012e, 2013d, 2015c). Even though all TEDEs resulting from both BWRs and PWRs were much lower than the limits, the TEDEs from PWRs were approximately 2 or 3 times higher than those from BWRs. Since BWRs have discharged relatively small amounts of radioactive effluents compared to PWRs due to a zero-release approach, the TEDEs from PWRs were relatively higher than those from BWRs (USNRC 2013d). Compared to the EPA dose standard for a site, 0.25 mSv y^{-1}, all TEDEs for members of the public during the years 2007-2009 accounted for only 0.13-0.43%. In comparison with the NRC dose limit, 1 mSv y^{-1}, the TEDEs for members of the public were only 0.03-0.11%. Therefore, the dose to members of the public due to radioactive effluents released from NPPs has been kept very low due to radioactive effluent control programs in NPPs. The TEDEs for members of the public during the years 2007-2009 depending on the types of reactors were shown in Fig. 9 and Table 21 (USNRC 2011b, 2012e, 2013d, 2015c). The percentages of each TEDE from all NPPs, BWRs, and PWRs to the EPA dose standard and the NRC dose limit are also displayed in Table 21.

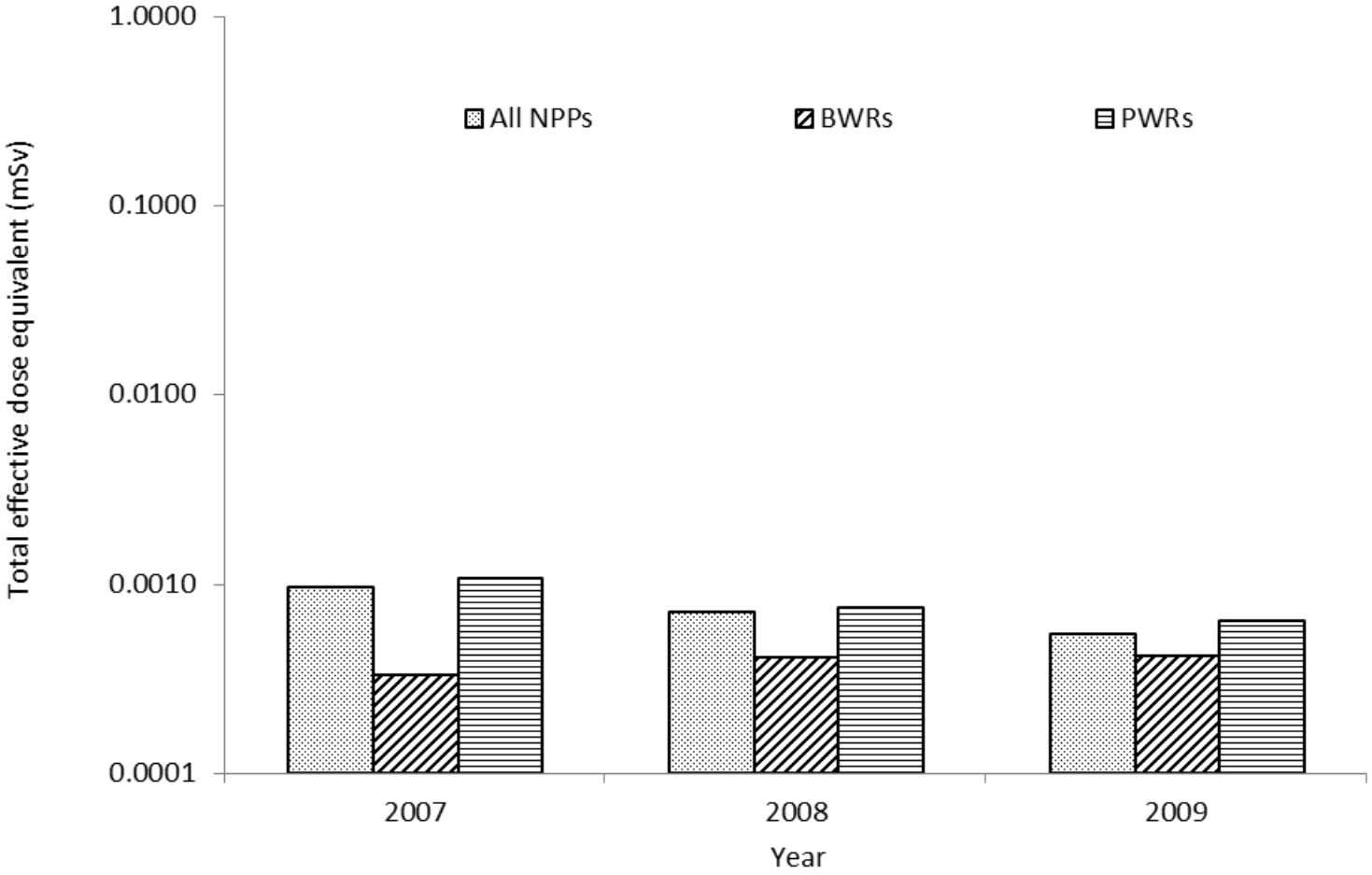

Fig. 9. Total effective dose equivalent for members of the public due to radioactive effluents released from NPPs during the years 2007-2009 (USNRC 2011b, 2012e, 2013d).

Table 21. Comparison of total effective dose equivalent for members of the public during the years 2007-2009 with the EPA dose standard and the NRC dose limits[a].

Year	Type of reactors	Total body dose (mSv)			Percent of the EPA dose standard[a,b] (%)	Percent of the NRC dose limit[a,c] (%)
		Maximum	Median	Minimum		
2007	All NPPs	1.72×10^{-1}	9.69×10^{-4}	1.98×10^{-7}	0.39	0.10
	BWRs	1.72×10^{-1}	3.33×10^{-4}	1.98×10^{-7}	0.13	0.03
	PWRs	7.06×10^{-2}	1.07×10^{-3}	3.81×10^{-6}	0.43	0.11
2008	All NPPs	1.63×10^{-1}	7.17×10^{-4}	1.63×10^{-7}	0.29	0.07
	BWRs	1.63×10^{-1}	4.11×10^{-4}	1.63×10^{-7}	0.16	0.04
	PWRs	6.06×10^{-2}	7.56×10^{-4}	1.18×10^{-5}	0.30	0.08
2009	All NPPs	1.68×10^{-1}	5.42×10^{-4}	1.66×10^{-7}	0.22	0.05
	BWRs	1.68×10^{-1}	4.19×10^{-4}	1.66×10^{-7}	0.17	0.04
	PWRs	1.17×10^{-1}	6.39×10^{-4}	1.55×10^{-5}	0.26	0.06

[a] (USNRC 2011b, 2012e, 2013d).
[b] Percentages are based on median values.
[c] The EPA dose standard to the whole body for a site is 0.25 mSv y^{-1} .
[d] The NRC dose limit for members of the public is 1 mSv y^{-1} .

Analysis of the dose for members of the public living around nuclear power plants for licensing

As stated earlier, all applicants for acquiring a license for new NPP operation are required to evaluate the off-site radiological consequences during the licensing process (USNRC 1956a, 1989, 2004a). The NRC has established a combined license, including both a construction permit and an operating license (USNRC 2004a). This combined license approves construction and conditional operation of a NPP at the same time. The requirements for acquiring a combined license are described in 10 CFR Part 52, but these requirements basically include the same information required in an application for an operating license issued under 10 CFR Part 50 (USNRC 1989; 2004a). From the viewpoint of dose criteria, an applicant for NPP license should comply with the requirements of 10 CFR Part 50.34 under the conditions of postulated accidents and the requirements of 10 CFR Part 20.1301(e) and design objectives in Appendix I to 10 CFR Part 50 under the condition of a postulated normal operation. To estimate the dose for a postulated normal operation, dose is calculated by summing up the dose using the predictable source term for the main reactor and the maximum dose from the history of other reactors. In practice, the values of 10 CFR Part 20.1301(e), which originate from the EPA dose standards, play a key role in evaluating the licensee's implementation of the requirements of as low as reasonably achievable (ALARA) for the safety of the public. Therefore, all applicants for acquiring a license for new NPP operation should demonstrate that their calculated doses, under the condition of a postulated normal operation, comply with the EPA dose standards.

In terms of NPP licensing, several applicants have submitted their applications for new reactors. Table 22 shows the current applications for a combined license which the NRC has received to date (USNRC 2015a). In total 18 applications were submitted to the NRC, and three applicants recently acquired a license to construct and operate NPPs from the NRC. The site locations of projected new NPPs, whose applications are currently under review or suspended, are displayed in Fig. 10 (USNRC 2015d). The early site permit application by PSEG (Public Service Electric and Gas) Power is also included in Fig. 10.

With regards to issuing a combined license, the NRC currently authorized the three licensees to construct and operate NPPs, consistent with established laws and regulations. In 2008, South Carolina Electric & Gas Company submitted its application for combined licenses for two PWRs for the Virgil C. Summer Nuclear Station Units 2 and 3. In the same year, Southern Nuclear Operating Company submitted its application for combined licenses for two PWRs for Vogtle Electric Generating Plant Units 3 and 4. These two applications were passed by the NRC review, and combined licenses were finally issued in 2012 (USNRC 2015a). The latest combined license was issued in 2015 for Fermi Unit 3 (USNRC 2015a). DTE Energy Company (previously Detroit Edison Company) also submitted its application for a combined license for a BWR designated as Fermi Unit 3 in 2008.

Table 22. New NPP applications for combined license to the U.S. NRC[a].

Proposed new NPPs	Design	Applicant	Status
Bell Bend Nuclear Power Plant	U.S. EPR	PPL Bell Bend, LLC	Under Review
Bellefonte Nuclear Station, Units 3 and 4	AP1000	Tennessee Valley Authority (TVA)	Suspended
Callaway Plant, Unit 2	U.S. EPR	AmerenUE	Suspended
Calvert Cliffs, Unit 3	U.S. EPR	Calvert Cliffs 3 Nuclear Project, LLC and UniStar Nuclear Operating Services, LLC	Withdrawn
Comanche Peak, Units 3 and 4	US-APWR	Luminant Generation Company, LLC (Luminant)	Suspended
Fermi, Unit 3	ESBWR	Detroit Edison Company	Issued
Grand Gulf, Unit 3	ESBWR	Entergy Operations, Inc. (EOI)	Suspended
Levy County, Units 1 and 2	AP1000	Duke Energy Florida, Inc. (DEF)	Under Review
Nine Mile Point, Unit 3	U.S. EPR	Nine Mile Point 3 Nuclear Project, LLC and UniStar Nuclear Operating Services, LLC (UniStar)	Withdrawn
North Anna, Unit 3	ESBWR	Dominion Virginia Power (Dominion)	Under Review
River Bend Station, Unit 3	ESBWR	Entergy Operations, Inc. (EOI)	Suspended
Shearon Harris, Units 2 and 3	AP1000	Progress Energy Carolinas, Inc. (PEC)	Suspended
South Texas Project, Units 3 and 4	ABWR	Nuclear Innovation North America, LLC (NINA)	Under Review
Turkey Point, Units 6 and 7	AP1000	Florida Power and Light Company (FPL)	Under Review
Victoria County Station, Units 1 and 2	ESBWR	Exelon Nuclear Texas Holdings, LLC (Exelon)	Withdrawn
Virgil C. Summer, Units 2 and 3	AP1000	South Carolina Electric & Gas (SCE&G)	Issued
Vogtle, Units 3 and 4	AP1000	Southern Nuclear Operating Company (SNC)	Issued
William States Lee III, Units 1 and 2	AP1000	Duke Energy	Under Review

a (USNRC 2015a).

Fig. 10. Locations of projected new NPPs (ABWR: Advanced Boiling Water Reactor, AP1000: Advanced Passive 1000, EPR: Evolutionary Pressurized Reactor, ESBWR: Economic Simplified Boiling Water Reactor, USAPWR: US Advanced Pressurized Water Reactor, ESP: Early Site Permit) (USNRC 2015d).

According to final safety analysis reports (FSARs) from these three applicants, their estimated doses for postulated accidents, including loss of coolant accidents (LOCA), complied with the requirements for a combined license. However, the values are not open to the public for the security reasons (SCE&G 2011; SC 2011; DTE Energy 2014). For postulated normal operations, estimated doses were shown in their FSARs, and all doses were less than those limits in the regulations. In particular, each site dose for the whole body from Virgil C. Summer Units 2 and 3, Vogtle Units 3 and 4, and Fermi Unit 3 accounted for 8.8%, 95.2%, and 22.64% of the EPA dose standard, respectively. For the site dose of Virgil C. Summer Units 2 and 3, the estimated dose included the dose from the currently operating single reactor Unit 1 (SCE&G 2011). The estimated site doses of Vogtle Units 3 and 4 and Fermi Unit 3 also incorporated the doses from existing reactors Units 1 and 2 (SC 2011; DTE Energy 2014). Table 23 shows the result of dose estimation with the EPA dose standards (SCE&G 2011; SC 2011; DTE Energy 2014). The projected doses due to liquid and gaseous effluents released from new NPPs were also compared with corresponding design objectives in Table 24 (SCE&G 2011; SC 2011; DTE Energy 2014).

Table 23. Comparison of site doses for the combined license of new NPP applications with the EPA dose standards[a].

Dose	Dose standards[b] (mSv y^{-1})	Virgil C. Summer Units 2 and 3 (mSv y^{-1})	Vogtle Units 3 and 4 (mSv y^{-1})	Fermi Unit 3 (mSv y^{-1})
Whole body	0.25	2.20×10^{-2}	2.38×10^{-1}	5.66×10^{-2}
Thyroid	0.75	1.40×10^{-1}	1.40×10^{-2}	1.39×10^{-1}
Other organ	0.25	3.50×10^{-2}[c]	2.60×10^{-2}[d]	2.32×10^{-2}[d]

[a] (SCE&G 2011; SC 2011; DTE Energy 2014).
[b] The values are based on 40 CFR Part 190 and 10 CFR Part 20.1301(e). Limits are applicable to a single site.
[c] Dose for bone.
[d] Dose for skin.

Table 24. Comparison of projected doses to the public of new NPP applications for the combined license with design objectives in Appendix I to 10 CFR Part 50[a].

Dose	Design objectives[b] (mSv y^{-1})	Virgil C. Summer Units 2 and 3[c] (mSv y^{-1})	Vogtle Units 3 and 4[c] (mSv y^{-1})	Fermi Unit 3 (mSv y^{-1})
Total body dose from liquid effluents	0.03	1.40×10^{-3}	2.50×10^{-4}	6.48×10^{-5}
Organ dose from liquid effluents	0.1	1.90×10^{-3}[d]	3.70×10^{-4}[c]	8.77×10^{-4}[e]
Total body dose from gaseous effluents	0.05	5.80×10^{-3}	8.10×10^{-3}	9.76×10^{-3}
Organ dose from radioactive iodine and particulates in gaseous effluents	0.15	7.00×10^{-2}[f]	8.10×10^{-3}[f]	1.13×10^{-1}[f]

[a] (SCE&G 2011; SC 2011; DTE Energy 2014).
[b] The values are based on Appendix I to 10 CFR Part 50. Limits are applicable to a single unit.
[c] The values are based on a single unit.
[d] Dose for GI-LLI (Gastrointestinal - Lower Large Intestine).
[e] Dose for bone.
[f] Dose for thyroid.

Dose for occupational workers in nuclear power plants

Approximately 200,000 people are working at 104 commercial NPPs in the United States. These individuals are exposed to direct or indirect radiation (USNRC 2014d). For the safety of workers, all NPPs monitor individual doses regularly using radiation monitoring devices, such as thermoluminescent dosimeters (TLDs) and whole body counters. Similar to reporting radioactive effluents and the dose for members of the public, these occupational doses are reported to the NRC annually. In particular, 10 CFR Part 20.1502 provides the conditions requiring individual monitoring of external and internal doses (USNRC 1991c). The occupational dose recording is based on the calendar year; thus, the monitoring period of occupational dose, especially for NPPs, is normally from January 1 to December 31 (USNRC 1992). According to 10 CFR Part 20.1202, the TEDE is calculated by the summation of external radiation dose (DDE) and internal radiation dose (CEDE) (USNRC 1991c). However, doses to the skin or extremities or to the eyes are not applicable to this summation method. The minimum limit for recording is 0.1 mSv because such a small value is negligible compared to the dose limit (USNRC 1992). All these recordings are used to demonstrate compliance with the NRC regulations, such as dose limits, and are maintained until the NRC terminates the NPP's license. To disseminate radiation exposure information to the public, the NRC has published annually a series of NUREG-0713, "Occupational radiation exposure at commercial nuclear power reactors and other facilities" (USNRC 2014d). In these reports, TEDE is primarily used to express the quantity of occupational dose and to demonstrate compliance with dose limits (USNRC 2014d). For analyzing occupational exposures in the US NPPs, the study

described here used two types of data. First, the data of occupational exposures at commercial NPPs for the years between 2003 and 2012 obtained from the NRC through the Freedom of Information Act (FOIA) were used to analyze the typical occupational doses (USNRC 2016). Second, the data provided by the NRC annual reports for occupational exposure, taking into account transient individuals, were used to analyze the occupational dose distributions in NPPs (USNRC 2004b, 2005b, 2006, 2007b, 2008a, 2010b, 2011a, 2012c, 2013c; 2014d).

Analysis of occupational dose in nuclear power plants

There are 104 NPPs operating in the United States, including 35 BWRs and 69 PWRs. During the years 2003-2012, approximately between 164,000 and 200,000 people were reported each year to the NRC as the number of monitored individuals in NPPs (USNRC 2016). These reports included office workers, maintenance workers, contract individuals, and even visitors in NPPs. Since most office individuals and visitors are not involved in radiation work in the field, their exposure levels are usually extremely low, and their radiation exposure may not be measured due to the detection limits of monitoring devices. Therefore, the number of individuals whose radiation exposure was measurable is provided to analyze the actual radiation dose due to radiation work in NPPs. According to the NRC data for occupational exposure, the fraction of individuals with measurable dose to the total number of monitored individuals in NPPs was approximately 40-50% during the years 2003-2012 (USNRC 2016). This result means that only half of individuals in NPPs received a measurable dose.

Contrary to the use of median value for the analysis of public dose, an average value is used to demonstrate the typical occupational dose in this study. The number of samples, which is the number of monitored individuals, for occupational dose is much greater than those for public dose. Therefore, the average occupational dose is less affected by outliers, which may have extremely high or low values. The average is also used in the NRC annual report to estimate the typical occupational exposure (USNRC 2014d). Furthermore, three types of dose quantities are used to express occupational dose in NPPs. First, the collective dose is used to estimate the total dose to the exposed individuals in NPPs. The collective dose is calculated as the sum of all individual doses during the NPP operation. Second, the average individual dose is used to estimate the average dose to the total number of monitored individuals in NPPs. This dose is calculated by dividing the collective dose by the total number of monitored individuals. Finally, the average measurable dose is used to estimate the actual average dose to individuals with a measurable dose. In this case, it is calculated by dividing the collective dose by the number of individuals with a measurable dose. The average measurable dose is most commonly used for analyzing trends of occupational dose. Since the total number of monitored individuals includes the number of individuals whose dose is not measurable, the average individual dose is generally lower than the average measurable dose.

The TEDE for individuals in NPPs, including both BWRs and PWRs in the United States, was analyzed using the data from the NRC for occupational exposure. The average individual dose during the years 2003-2012 was 0.4-0.8 mSv for all NPPs (USNRC 2016). For the average measurable dose, the dose range was distributed between 1.0-1.6 mSv for

all NPPs. The average measurable dose was 1.1-1.9 mSv for BWRs and 0.9-1.4 mSv for PWRs (USNRC 2016). The occupational doses in BWRs were higher than those in PWRs since the steam produced directly from the BWR reactor is used to drive turbines to produce electricity, which results in activity being present in both the reactor and the turbine systems, while PWR systems are designed to retain the activity within the reactor vessel and primary system and not in the turbine systems. In comparison with the NRC dose limit, the average measurable dose in NPPs accounted for only 2.0-3.2% of the annual occupational dose limit. The maximum and minimum average measurable doses during the years 2003-2012 were reported to be 1.6 and 1 mSv, respectively. The occupational doses in NPPs during the years 2003-2012 are shown in Table 25 and Fig. 11 (USNRC 2016). The occupational doses depending on the types of reactors are also shown in Table 26 and Fig. 12 (USNRC 2016).

Table 25. Occupational doses in NPPs during the years 2003-2012[a].

Year	Number of NPPs	Total number of monitored individuals[b]	Number of individuals with measurable dose[c]	Collective dose (Person-mSv)[c]	Average individual dose (mSv)[d]	Percent of limit (%)[e]	Average measurable dose (mSv)[d]	Percent of limit (%)[e]
2003	104	163,802	78,206	123,291	0.8	1.5	1.6	3.2
2004	104	159,013	72,901	105,928	0.7	1.3	1.5	2.9
2005	104	171,431	81,395	115,845	0.7	1.4	1.4	2.8
2006	104	168,831	80,619	109,876	0.7	1.3	1.4	2.7
2007	104	170,516	80,803	101,771	0.6	1.2	1.3	2.5
2008	104	176,896	80,890	92,067	0.5	1.0	1.1	2.3
2009	104	181,698	82,677	100,274	0.6	1.1	1.2	2.4
2010	104	186,452	76,178	86,436	0.5	0.9	1.1	2.3
2011	104	198,096	83,183	88,135	0.4	0.9	1.1	2.1
2012	104	200,634	81,174	81,350	0.4	0.8	1.0	2.0

[a] (USNRC 2016).
[b] Total number of individuals that NPP licensees reported as being monitored for external and internal radiation exposure during each year.
[c] Number of individuals whose radiation exposure was measurable.
[d] Total effective dose equivalent (TEDE).
[e] NRC annual occupational dose limit: 50mSv.

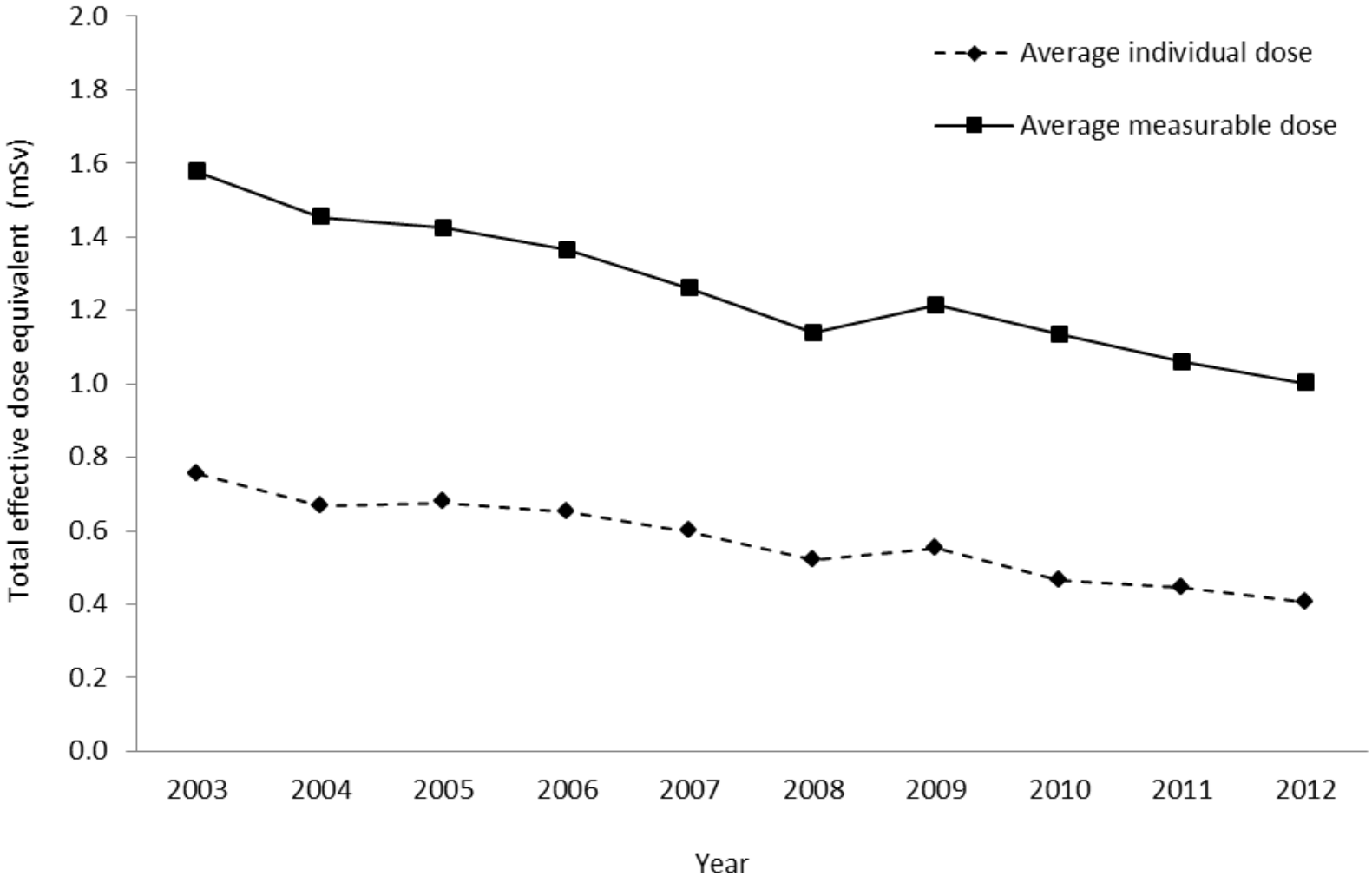

Fig. 11. Average individual and measurable occupational doses in NPPs during the years 2003-2012 (USNRC 2016).

Table 26. Occupational doses in BWRs and PWRs during the years 2003-2012[a].

Year		2003	2004	2005	2006	2007	2008	2009	2010	2011	2012
BWR	Number of reactors	35	35	35	35	35	35	35	35	35	35
	Number of individuals with measurable dose[b]	31,216	33,093	34,635	32,528	36,113	33,967	34,540	35,839	38,479	36,651
	Collective dose (Person-mSv)[c]	58,559	53,604	61,410	48,603	52,377	44,917	51,154	46,574	49,639	40,699
	Average measurable dose (mSv)[c]	1.9	1.6	1.8	1.5	1.5	1.3	1.5	1.3	1.3	1.1
	Percent of limit (%)[d]	3.8	3.2	3.5	3.0	2.9	2.6	3.0	2.6	2.6	2.2
PWR	Number of reactors	69	69	69	69	69	69	69	69	69	69
	Number of individuals with measurable dose[b]	46,990	39,808	46,760	48,091	44,690	46,923	48,137	40,339	44,704	44,523
	Collective dose (Person-mSv)[c]	64,732	52,324	54,435	61,273	49,394	47,150	49,121	39,862	38,496	40,651
	Average measurable dose (mSv)[c]	1.4	1.3	1.2	1.3	1.1	1.0	1.0	1.0	0.9	0.9
	Percent of limit (%)[d]	2.8	2.6	2.3	2.5	2.2	2.0	2.0	2.0	1.7	1.8

[a] (USNRC 2016).
[b] Number of individuals whose radiation exposure was measurable.
[c] Total effective dose equivalent (TEDE).
[d] NRC annual occupational dose limit: 50mSv.

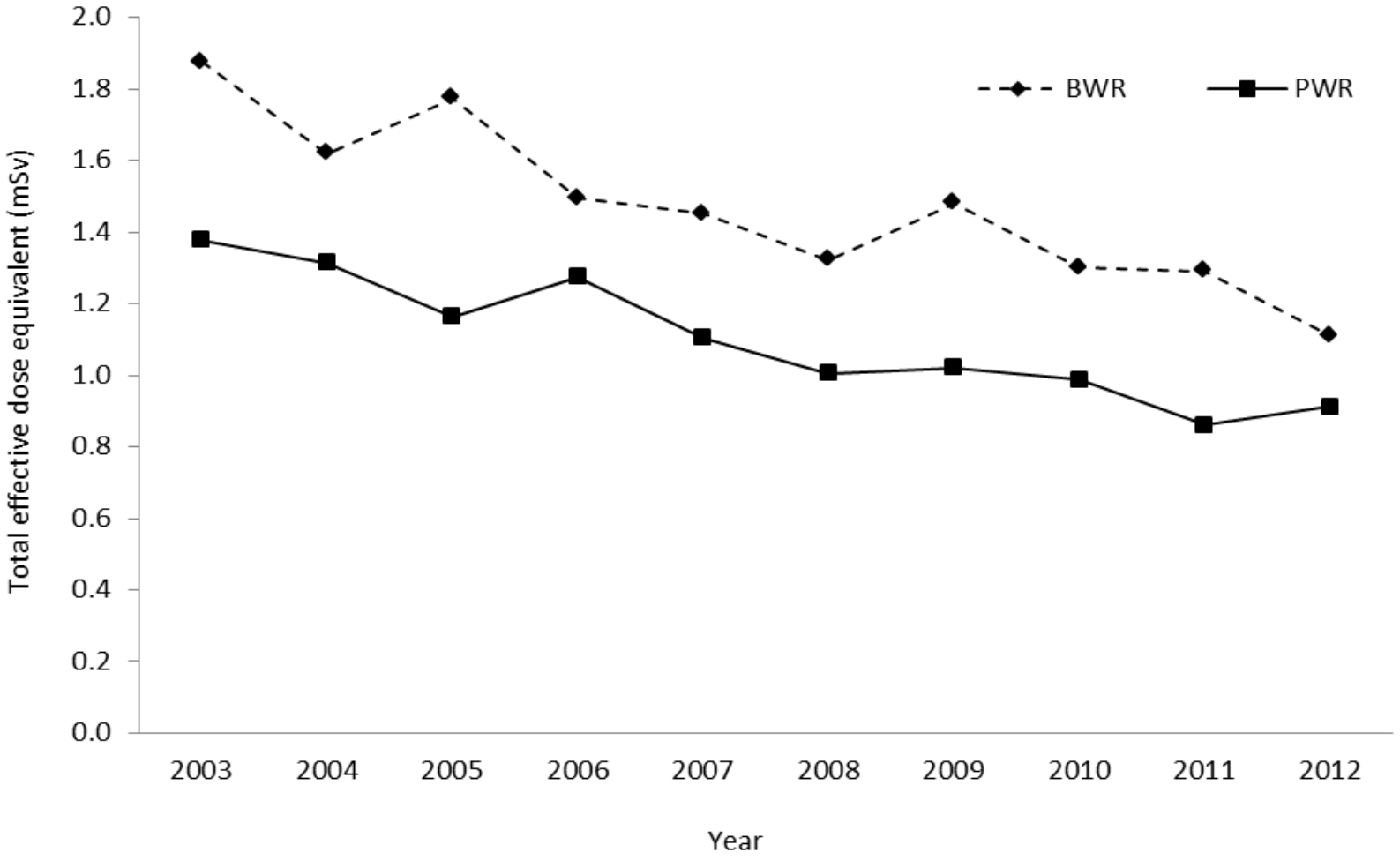

Fig. 12. Average measurable occupational doses in BWRs and PWRs during the years 2003-2012 (USNRC 2016).

The annual dose distributions in NPPs during the years 2003-2012 were also analyzed using the data from the NRC annual reports for occupational exposure to compare the number of individuals in certain dose ranges. These dose distributions were obtained by counting the number of individuals in certain dose ranges. As a result, the fractions of the number of individuals whose doses were less than 1 mSv to the number of individuals with measurable doses were approximately 58-70%. These percentages increased continuously from 58% in 2003 to 70% in 2012 (USNRC 2016). There was no individual who received the dose more than the NRC annual dose limit of 50 mSv. In particular, the fractions of the number of individuals whose doses exceeded 20 mSv to the number of individuals with measurable doses were less than 0.05%. The occupational dose distributions in NPPs during the years 2003-2012 are shown in Table 27 and Fig. 13 (USNRC 2016).

Table 27. Occupational dose distributions in NPPs during the years 2003-2012[a].

Year	Total number of individuals[b]	Number of individuals with non-measurable dose[c]	Number of individuals with measurable dose[d]	Number of individuals in the dose ranges (mSv)[e]								
				< 1	[1-2.5)	[2.5-5)	[5-7.5)	[7.5-10)	[10-20)	[20-30)	[30-40)	[40-50)
2003	163,802	85,596	78,206	45,161	17,770	9,805	3,226	1,278	928	38	0	0
2004	159,013	86,112	72,901	43,889	16,142	8,443	2,735	1,014	663	15	0	0
2005	171,431	90,036	81,395	49,047	18,389	9,301	2,911	1,069	659	19	0	0
2006	168,831	88,212	80,619	48,983	18,321	9,237	2,652	899	525	2	0	0
2007	170,516	89,713	80,803	51,119	17,836	8,297	2,317	817	404	13	0	0
2008	176,896	96,006	80,890	53,187	17,542	7,499	1,817	572	268	5	0	0
2009	181,698	99,021	82,677	53,503	17,580	8,335	2,117	732	410	0	0	0
2010	186,452	110,274	76,178	50,748	16,083	6,623	1,797	598	324	5	0	0
2011	198,096	114,913	83,183	57,198	16,762	6,729	1,653	570	271	0	0	0
2012	200,634	119,460	81,174	57,190	15,707	6,100	1,525	383	248	21	0	0

[a] (USNRC 2016).

[b] Total number of individuals that NPP licensees reported as being monitored for external and internal radiation exposure during each year.

[c] Number of individuals whose radiation exposure was not measurable.

[d] Number of individuals whose radiation exposure was measurable.

[e] Total effective dose equivalent (TEDE).

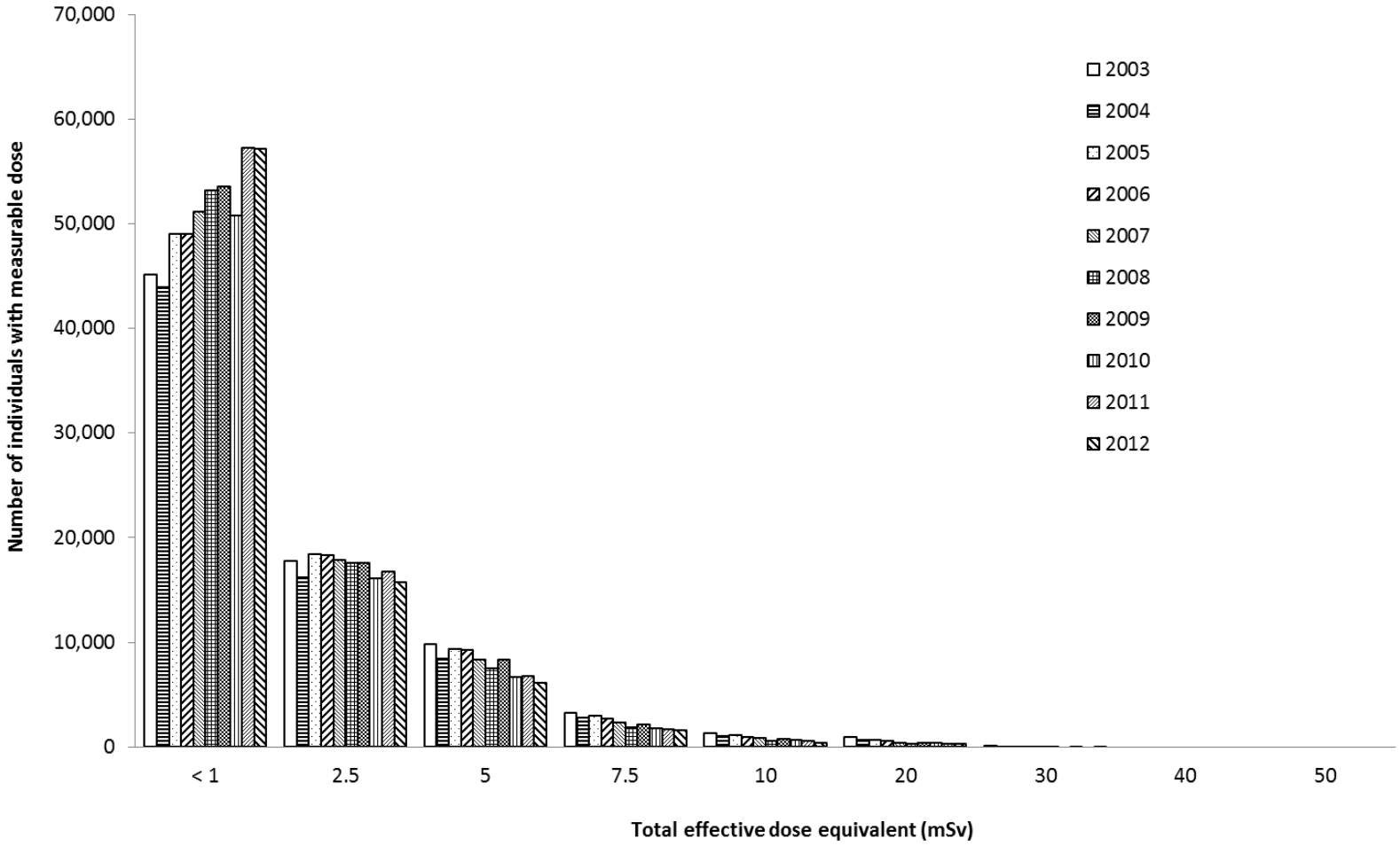

Fig. 13. Measurable occupational dose distributions in NPPs during the years 2003-2012 (USNRC 2016).

Analysis of occupational doses taking into account transient individuals in nuclear power plants

According to 10 CFR Part 20.2206, "Reports of individual monitoring," each NPP licensee reports the occupational dose received by individuals monitored at their NPPs (USNRC 1991c). These data demonstrate the typical individual doses in NPPs; however, in reality, the total number of monitored individuals is overcounted relative to its true level due to transient workers. Transient workers are defined as people who worked at more than one nuclear facility during the monitoring year and their occupational doses, received at each facility, are reported separately to the NRC by each NPP licensee (USNRC 2014d). These data appear to be separate individual doses when dose records are summed for all NPP licensees, although some of the dose records belong to the same individual. To obtain the actual dose information, it is necessary to combine these dose records for each individual.

During the years 2003-2012, approximately 110,000-138,000 workers were reported to the NRC as the total number of monitored individuals except for multiple counting of transient workers in NPPs (USNRC 2004b, 2005b, 2006, 2007b, 2008a, 2010b, 2011a, 2012c, 2013c; 2014d). These data indicated the actual total number of individuals that NPP licensees reported as being monitored for radiation exposure during each year. Therefore, the over-reporting of individuals who worked at more than one nuclear facility during the monitoring year was eliminated. In terms of the number of individuals with measurable doses, the fraction of individuals whose radiation exposure was measurable to the actual total number of monitored individuals in NPPs was approximately 42-51%

during the years 2003-2012 (USNRC 2004b, 2005b, 2006, 2007b, 2008a, 2010b, 2011a, 2012c, 2013c; 2014d). These data indicate that only half of individuals in NPPs received a dose greater than zero.

The TEDE for individuals, taking into account transient workers in NPPs, was analyzed using the data from the NRC annual reports for occupational exposure. The average measurable dose during the years 2003-2012 varied from 1.4 mSv to 2.1 mSv for all NPPs (USNRC 2004b, 2005b, 2006, 2007b, 2008a, 2010b, 2011a, 2012c, 2013c; 2014d). These values were slightly higher than those with no regard for transient individuals, which were 1.0-1.6 mSv. In comparison with the NRC dose limit, the average measurable dose in NPPs was only 2.8-4.2% of the annual occupational dose limit. The maximum and minimum average measurable doses during the years 2003-2012 were reported to be 2.1 mSv and 1.4 mSv, respectively. The occupational doses, taking into account transient individuals in NPPs, during the years 2003-2012 are listed in Table 28 and Fig. 14 (USNRC 2004b, 2005b, 2006, 2007b, 2008a, 2010b, 2011a, 2012c, 2013c; 2014d).

Table 28. Occupational doses taking into account transient individuals in NPPs during the years 2003-2012[a].

Year	Number of NPPs	Actual total number of monitored individuals[b]	Number of individuals with measurable dose[c]	Collective dose (Person-mSv)[d]	Average measurable dose (mSv)[d]	Percent of limit (%)[e]
2003	104	109,990	55,967	119,556	2.1	4.2
2004	104	110,290	52,873	103,679	2.0	4.0
2005	104	114,344	57,566	114,558	2.0	4.0
2006	104	116,354	58,788	110,212	1.9	3.8
2007	104	114,583	57,267	101,200	1.8	3.6
2008	104	118,692	57,356	91,959	1.6	3.2
2009	104	126,767	60,460	100,248	1.7	3.4
2010	104	130,172	55,954	86,314	1.5	3.0
2011	104	137,360	59,268	87,721	1.5	3.0
2012	104	137,762	58,343	80,354	1.4	2.8

[a] (USNRC 2004b, 2005b, 2006, 2007b, 2008a, 2010b, 2011a, 2012c, 2013c; 2014d).

[b] Actual total number of monitored individuals who excluded the multiple counting of individuals who worked at more than one nuclear facility during the monitoring year.

[c] Number of individuals whose radiation exposure is measurable.

[d] Total effective dose equivalent (TEDE).

[e] NRC annual occupational dose limit: 50mSv.

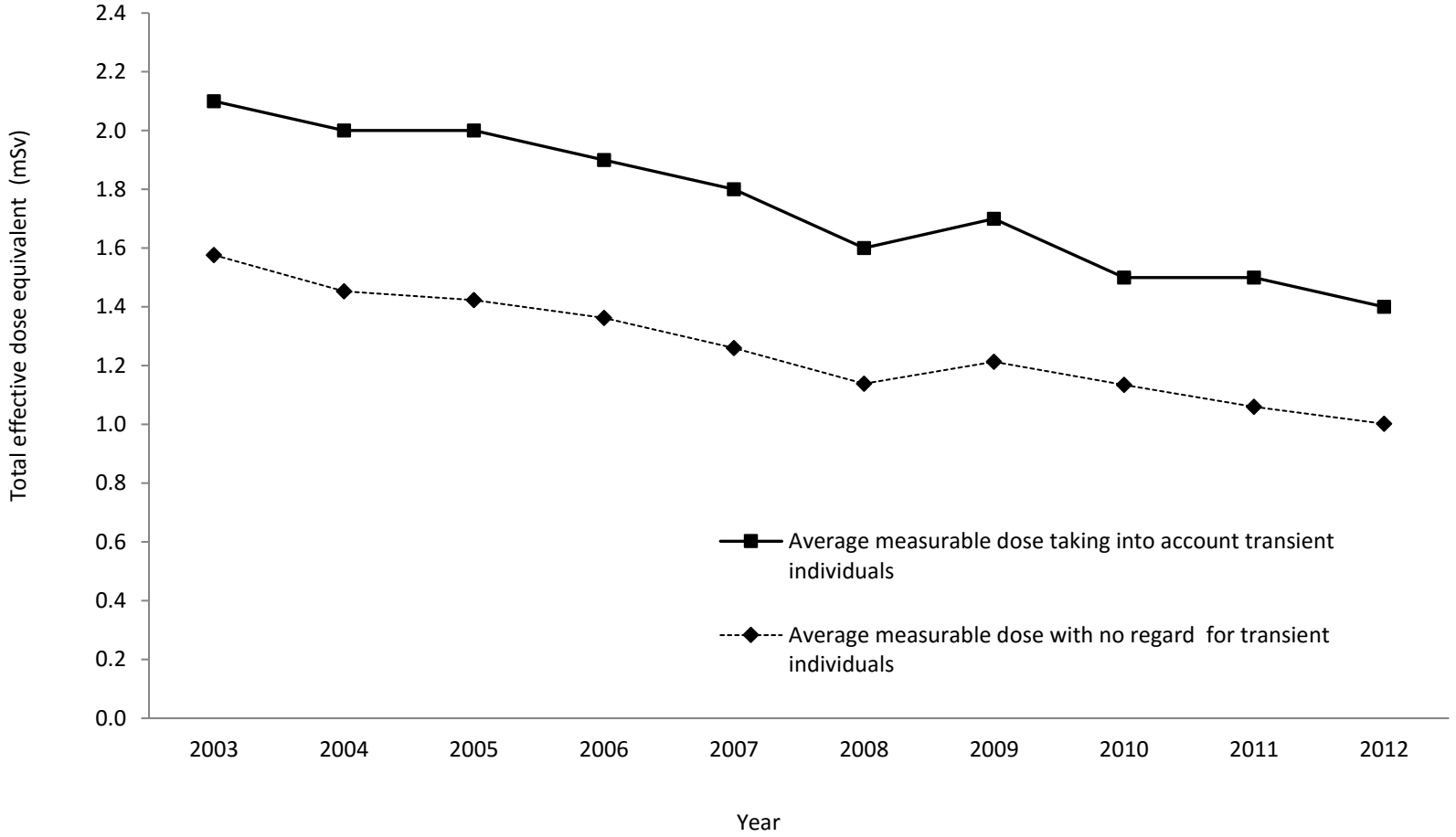

Fig. 14. Comparison of average occupational doses with and without taking into account transient individuals in NPPs during the years 2003-2012 (USNRC 2004b, 2005b, 2006, 2007b, 2008a, 2010b, 2011a, 2012c, 2013c; 2014d).

The occupational dose distributions in NPPs during the years 2003-2012 were also analyzed using the data with regards to transient individuals to compare the number of individuals in certain dose ranges. The dose distribution for each year was also obtained by summing the TEDE from the same individuals, and counting the number of individuals in each dose range. As a result, the fractions of the number of individuals whose dose was less than 1 mSv to the number of individuals with measurable dose were approximately 52-64%. These percentages increased continuously from 52% in 2003 to 64% in 2012 (USNRC 2004b, 2005b, 2006, 2007b, 2008a, 2010b, 2011a, 2012c, 2013c; 2014d). There were no individuals who received a dose above the NRC annual dose limit of 50 mSv. In particular, the percentages of individuals whose doses exceeded 20 mSv to the number of individuals with measurable doses were less than 0.4%. The occupational dose distributions with regards to transient individuals in NPPs during the years 2003-2012 are shown in Table 29 and Fig. 15 (USNRC 2004b, 2005b, 2006, 2007b, 2008a, 2010b, 2011a, 2012c, 2013c; 2014d).

Table 29. Occupational dose distributions taking into account transient individuals in NPPs during the years 2003-2012[a].

Year	Actual total number of individuals[b]	Number of individuals with non-measurable dose[c]	Number of individuals with measurable dose[d]	Number of individuals in the dose ranges (mSv)[e]								
				< 1	[1-2.5)	[2.5-5)	[5-7.5)	[7.5-10)	[10-20)	[20-30)	[30-40)	[40-50)
2003	109,990	54,023	55,967	29,164	11,978	8,199	3,249	1,524	1,651	184	18	0
2004	110,290	57,417	52,873	28,863	11,179	7,334	2,873	1,233	1,190	188	13	0
2005	114,344	56,778	57,566	31,043	12,427	7,815	3,104	1,537	1,490	147	3	0
2006	116,354	57,566	58,788	32,426	12,685	7,796	2,975	1,416	1,406	82	2	0
2007	114,583	57,316	57,267	32,706	11,961	7,396	2,714	1,283	1,101	97	9	0
2008	118,692	61,336	57,356	33,832	12,324	6,786	2,429	1,026	921	38	0	0
2009	126,767	66,307	60,460	35,873	12,319	7,314	2,564	1,174	1,144	68	4	0
2010	130,172	74,218	55,954	33,874	11,670	6,356	2,231	946	832	42	3	0
2011	137,360	78,092	59,268	36,747	12,121	6,308	2,225	1,007	837	23	0	0
2012	137,762	79,419	58,343	37,049	11,946	5,908	1,959	772	672	37	0	0

[a] (USNRC 2004b, 2005b, 2006, 2007b, 2008a, 2010b, 2011a, 2012c, 2013c; 2014d).

[b] Actual total number of monitored individuals who excluded the multiple counting of individuals who worked at more than one nuclear facility during the monitoring year.

[c] Number of individuals whose radiation exposure was not measurable.

[d] Number of individuals whose radiation exposure was measurable.

[e] Total effective dose equivalent (TEDE).

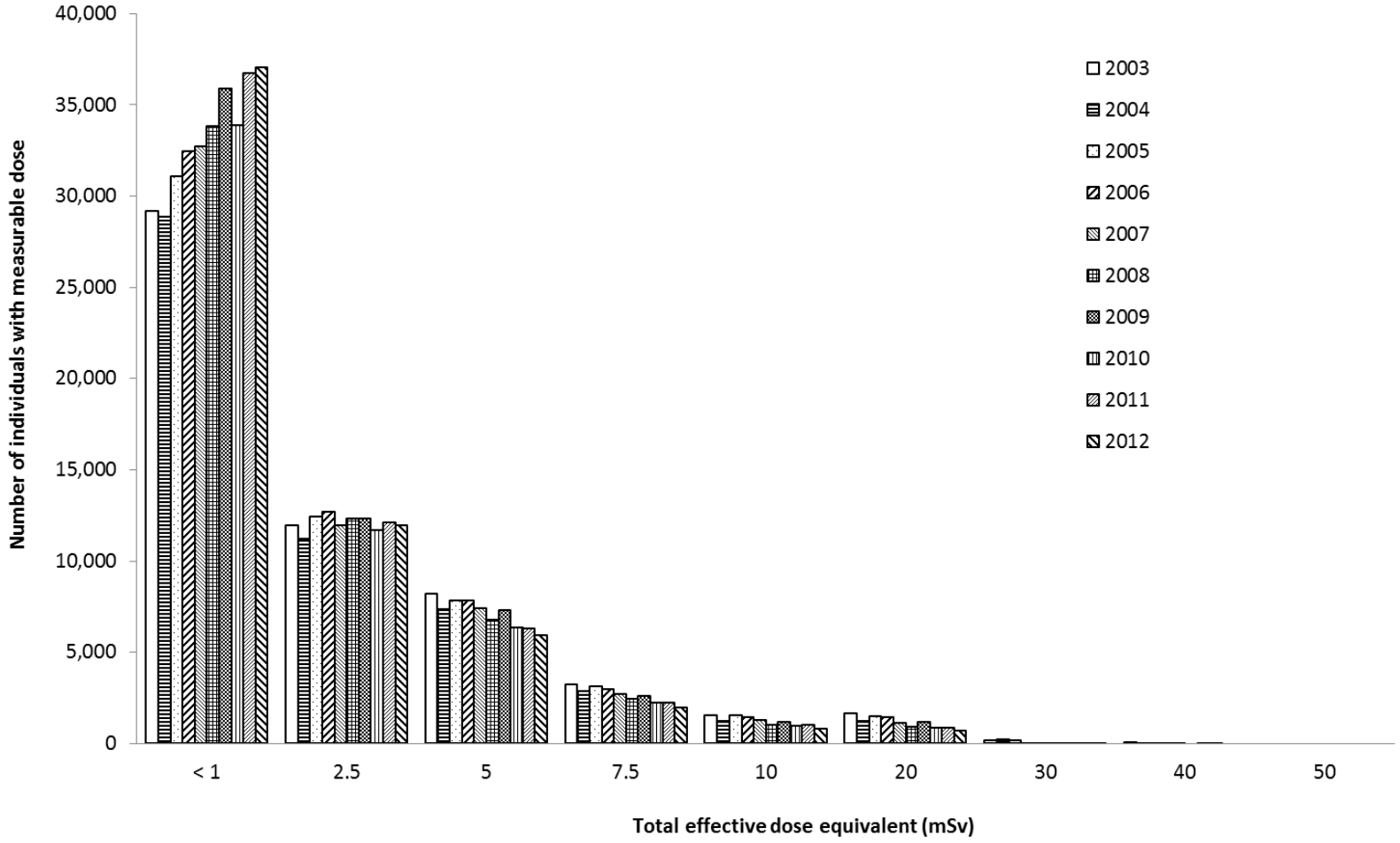

Fig. 15. Occupational dose distributions taking into account transient individuals in NPPs during the years 2003-2012 (USNRC 2004b, 2005b, 2006, 2007b, 2008a, 2010b, 2011a, 2012c, 2013c; 2014d).

DISCUSSION

Determination of the need of dose constraint for members of the public

An analysis of the estimated doses for members of the public (maximum exposed individual, MEI) living around nuclear power plants (NPPs) during the years 2007-2009 was carried out showing that the estimated public doses have been kept very low by radioactive effluent control programs. Even though the estimated total effective dose equivalents (TEDEs) to the public differed slightly depending on the types of reactor, the annual dose level was approximately on the order of 10^{-4} mSv. When compared to the average dose from natural background radiation received by an individual in the United States of 3.1 ± 3.6 mSv y^{-1}, the off-site doses correspond to an increase between 0.02 and 0.03%. Furthermore, when compared to all sources of radiation (medical included) the absorbed dose received by an individual will be 6.2 mSv (NCRP 2009) corresponding to an increase between 0.0087% and 0.0156%. Also, when compared with the annual dose limit of 1 mSv y^{-1}, the TEDEs to the public were between 0.03% and 0.11% of the annual dose limit. Therefore, the risk to the public from radioactive effluents released from NPPs, under normal operating conditions, is negligible compared to the dose limit or the typical background radiation dose.

The data from individual doses in the form of a probability distribution and a cumulative probability distribution were used as the technical basis for finding whether a dose constraint for members of the public is necessary or not. The idea of establishing a threshold of inequitable doses, using numerical percentiles in the dose distribution, is based on the two concepts of the EPA MEI and the ICRP representative person.

Lognormal distributions of individual doses for members of the public during the years 2007-2009 are presented in Fig. 16. The solid and dotted lines indicate the best fit to the probability distribution and the cumulative probability distribution, respectively. Using the cumulative probability distribution, it is possible to find the doses at the 95th percentile (ICRP) and the 99.99th percentile (EPA) of a given dose distribution. As shown in Table 30, the 95th percentiles (ICRP) for the years 2007, 2008, and 2009 were 5.60×10^{-2} mSv, 4.50×10^{-2} mSv, and 5.21×10^{-2} mSv, respectively, and the 99.99th percentiles (EPA) were 1.72×10^{-1} mSv, 1.63×10^{-1} mSv, and 1.68×10^{-1} mSv, respectively. The difference between the values for year from the 95th and the 99.99th percentiles were approximately the order of 10^{-1} mSv, which is negligible when compared when compared to the annual dose limit of 1 mSv.

Table 30. Minimum, median, and maximum doses and individual doses at the 95th and the 99.99th percentiles of public dose distributions during the years 2007-2009.

Year	2007	2008	2009
Minimum dose (mSv)	1.98×10^{-7}	1.63×10^{-7}	1.66×10^{-7}
Median dose (mSv)	9.69×10^{-4}	7.17×10^{-4}	5.42×10^{-4}
Maximum dose (mSv)	1.72×10^{-1}	1.63×10^{-1}	1.68×10^{-1}
Dose at the 95th percentile (mSv)	5.59×10^{-2}	4.50×10^{-2}	5.21×10^{-2}
Dose at the 99.99th percentile (mSv)	1.72×10^{-1}	1.63×10^{-1}	1.68×10^{-1}

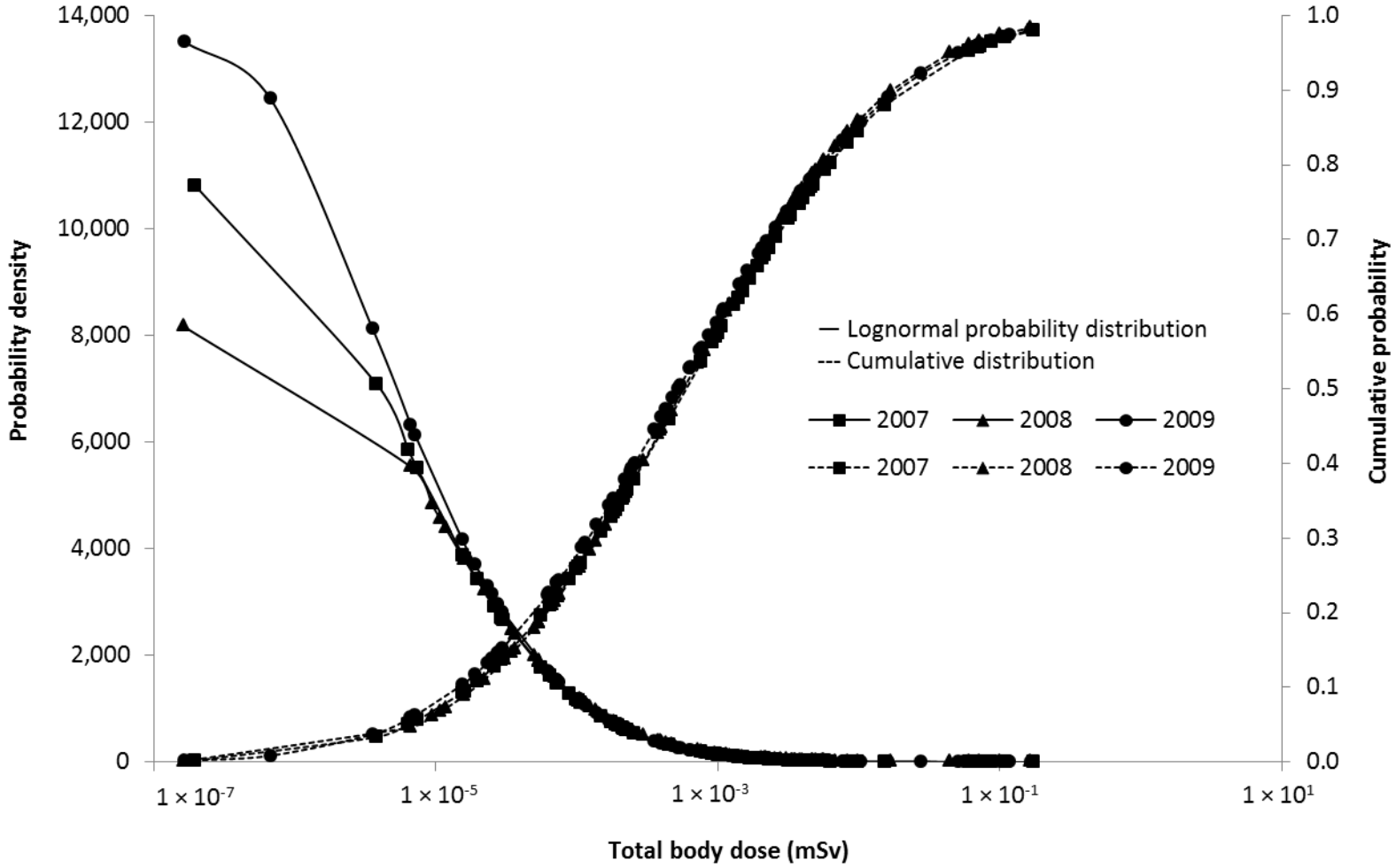

Fig. 16. Lognormal distribution of individual doses for members of the public living around NPPs during the years 2007-2009.

The potential radiation exposure from natural background and NPPs to a MEI is illustrated respectively in Fig. 17 comparing their total doses. Since absorbed dose is proportional to excess risk, $ER = D \times R_D$ (Equation 11), the total dose and corresponding relative increase in excess risk for a *MEI* is given by Equation (12). Under the LNT model, which shows that a given increment in a dose will produce a directly proportional increment in a risk, Equation (12) can be replaced by Equation (13).

$$\frac{D(Background+MEI)}{D(Background)} \sim \frac{R(Background+MEI)}{R(Background)} \quad (12)$$

$$\frac{D(Background)+D(MEI)}{D(Background)} \sim \frac{R(Background)+R(MEI)}{R(Background)} \quad (13)$$

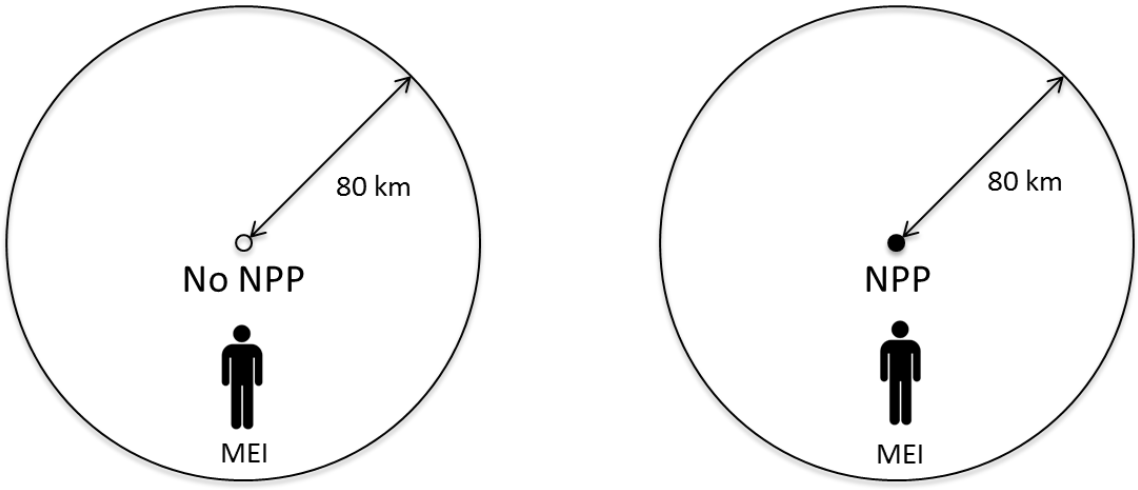

Fig. 17. Potential radiation exposure from natural background (left) and NPPs (right) to MEI.

In ICRP Publication 103 the total detriment-adjusted nominal risk coefficient for individual members of the public was estimated to be 5.7×10^{-5} per mSv (ICRP 2007). If

this risk coefficient is used to estimate the risk of individual doses for members of the public living around NPPs, the corresponding annual excess risk levels at the 95^{th} and the 99.99^{th} percentiles of given individual dose distributions are computed as approximately the order of 10^{-6} using Equation (11). The excess risks for median and maximum doses, individual doses at the 95^{th} and the 99.99^{th} percentiles of a given dose distribution during the years 2007-2009, and the average dose from natural background in the United States are given in Table 31. The corresponding relative increases in excess risk are also calculated using Equation (13), as compared to the risk of the average dose from natural background in the United States (Table 32).

Table 31. Comparison of excess risk, *ER*, for median and maximum doses, individual doses at the 95^{th} and the 99.99^{th} percentiles from public dose distributions during the years 2007-2009, and average dose from natural background in the United States.

Year	2007	2008	2009
Median dose (mSv)	9.69×10^{-4}	7.17×10^{-4}	5.42×10^{-4}
Risk	5.52×10^{-8}	4.09×10^{-8}	3.09×10^{-8}
Maximum dose (mSv)	1.72×10^{-1}	1.63×10^{-1}	1.68×10^{-1}
Risk	9.80×10^{-6}	9.29×10^{-6}	9.58×10^{-6}
Dose at the 95^{th} percentile (mSv)	5.59×10^{-2}	4.50×10^{-2}	5.21×10^{-2}
Risk	3.19×10^{-6}	2.57×10^{-6}	2.97×10^{-6}
Dose at the 99.99th percentile (mSv)	1.72×10^{-1}	1.63×10^{-1}	1.68×10^{-1}
Risk	9.80×10^{-6}	9.29×10^{-6}	9.58×10^{-6}
Average dose from natural background (mSv)	3.10×10^{0}	3.10×10^{0}	3.10×10^{0}
Risk	1.77×10^{-4}	1.77×10^{-4}	1.77×10^{-4}

Table 32. Comparison of relative increase in excess risk for median and maximum doses and individual doses at the 95th and the 99.99th percentiles from public dose distributions during the years 2007-2009, as compared to the risk of average dose from natural background in the United States.

Year	2007	2008	2009
Median dose (mSv)	9.69×10^{-4}	7.17×10^{-4}	5.42×10^{-4}
Relative increase in risk (%)	0.03	0.02	0.02
Maximum dose (mSv)	1.72×10^{-1}	1.63×10^{-1}	1.68×10^{-1}
Relative increase in risk (%)	5.55	5.26	5.42
Dose at the 95th percentile (mSv)	5.59×10^{-2}	4.50×10^{-2}	5.21×10^{-2}
Relative increase in risk (%)	1.80	1.45	1.68
Dose at the 99.99th percentile (mSv)	1.72×10^{-1}	1.63×10^{-1}	1.68×10^{-1}
Relative increase in risk (%)	5.55	5.26	5.42

Annual risk levels of 10^{-5} and below are generally considered to be of low probability or low consequence by most people (Whipple 1988). On the basis of this fact, the ICRP has recommended the annual public dose limit of 1 mSv, whose associated excess risk is 5.7×10^{-5}. The annual excess risk levels for median and maximum doses and individual doses at the 95th and the 99.99th percentiles of the actual individual dose distribution around NPPs were calculated to be on the order of 10^{-8} and 10^{-6}. These results mean there is less than 1 in 1 million chance of an individual fatality per year of exposure (i.e., 10^{-6} y^{-1}). The risk levels of 10^{-6} and below are widely regarded as acceptable risks (Spangler 1987; Martin and Sutton 2015). This level is sometimes referred to as a *de minimis* level, which means that the incremental risk produced by an exposure is sufficiently small so that there is no incentive to reduce the exposure (Whipple 1988). Furthermore, Table 32 shows that the relative increase in excess risk due to absorbed dose

accounted for only 0.03% for median dose and less than 6% for maximum dose. This indicates that the relative increase in risk due to radiation exposure from NPPs is trivial compared to the average dose from natural background in the United States.

A chart of probability of death for an individual per year of exposure to hazards, which was devised by the Royal Commission on Environmental Pollution in the United Kingdom, is shown in Fig. 18. This figure compares risk levels in terms of acceptable or unacceptable risk (Wilson 1984). As is shown, risk levels above 10^{-3} are regarded as unacceptable. In contrast, risk levels of 10^{-6} and below are considered acceptable. The range between risk levels of 10^{-3} and 10^{-6} is designated as the zone where risk-cost-benefit analysis should be conducted by a social decision-making process to determine whether that risk is acceptable or not. Compared to the risk levels in this chart, the results of excess risk for median and maximum doses and individual doses at the 95^{th} and the 99.99^{th} percentiles of the actual individual dose distribution around NPPs would be regarded as acceptable. Although the ICRP has recommended that every reasonable effort should be made to maintain all radiation exposures as low as reasonably achievable, it would be unreasonable to reduce these very low levels of individual risks to levels of 10^{-10} or 10^{-12}. In addition, the public doses realistically have been kept at very low levels by radioactive effluent control programs in the U.S. NPPs. As stated earlier, the annual dose level for members of the public was reported to be the order of 10^{-4} mSv, and such a dose is generally considered negligible. Even the maximum dose level of 10^{-1} mSv is one tenth of the dose limit (1 mSv) and is regarded as trivial compared to the dose limit. Therefore, it

is concluded, in this study, that a dose constraint is not necessary for members of the public living around the U.S. NPPs.

If the NRC were to implement a dose constraint for members of the public, the population fraction that would be affected by such implementation (doses at the 95th or at 99.99th percentiles) will be in fact negligible and the economic cost for such collective dose (S^*) reduction will have no net benefit. This can be expressed as

$$\left(\frac{dX}{dS}\right)_{S^*} > -\left(\frac{dY}{dS}\right)_{S^*} \text{ and} \quad (14)$$

$$\left(\frac{dX}{dS}\right)_{S^*} \approx 0, \quad (15)$$

where X is the cost of protection and Y is the cost of detriment. Therefore, there is no net cost-benefit that can be attained by establishing dose constraints in this particular situation.

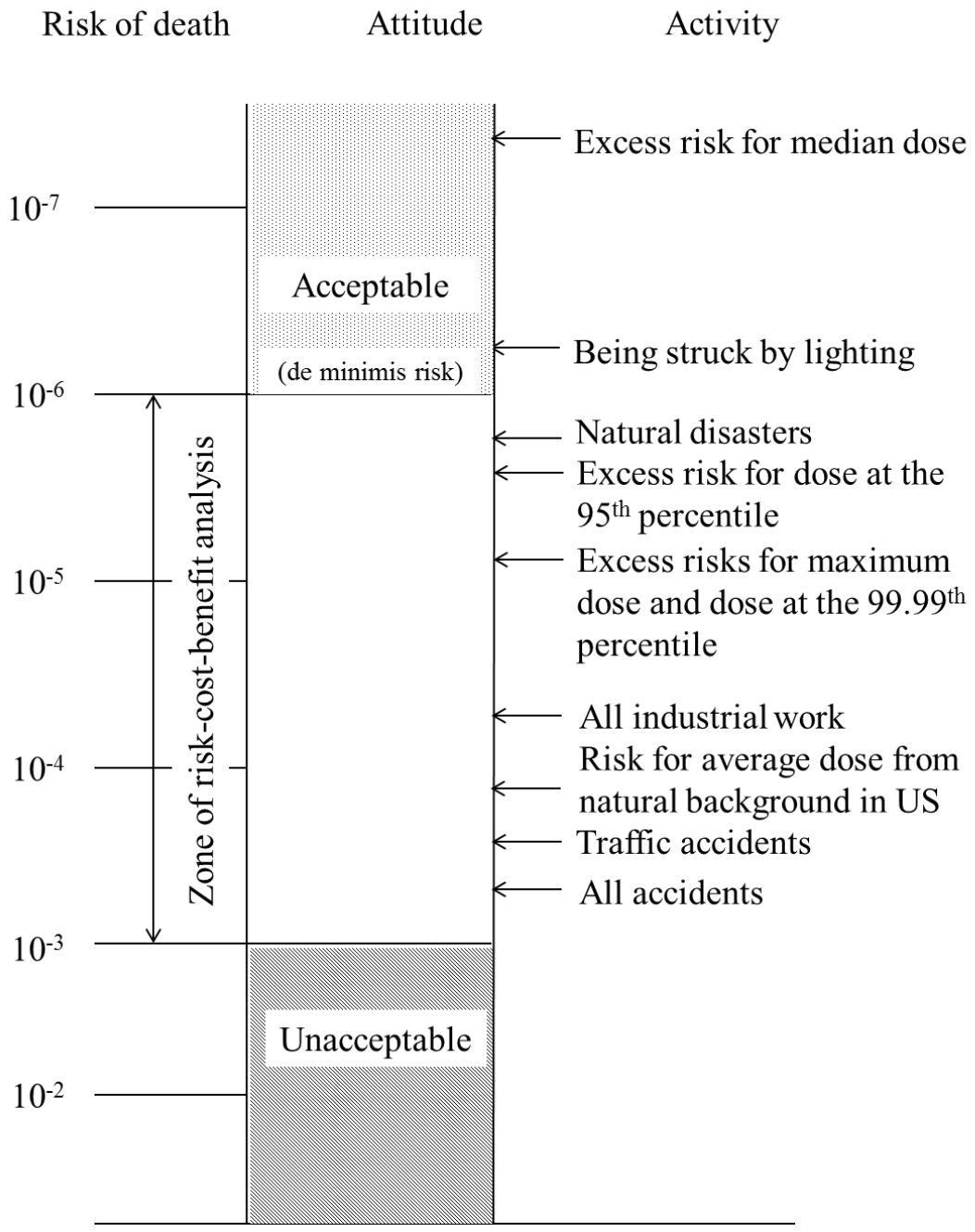

Fig. 18. Risk of death for an individual per year of exposure to hazards in terms of acceptable and unacceptable risk (orders of magnitude) showing the corresponding associated risk for members of the public living around NPPs.

Determination of the need of dose constraint for occupational workers

The occupational exposure data from NPPs for a period of 10 years were investigated to evaluate the annual occupational dose levels resulting from the normal operation of NPPs. The data analysis showed that occupational doses during the period between 2003 and 2012 were kept low by effective radiation safety programs. Although the reported TEDEs for occupational workers differed slightly depending on the types of reactors, the average doses during the years 2003-2012 fell into the range of 0.4 to 0.8 mSv for all NPPs. Since most office individuals and visitors in NPPs are not involved in radiation work in the field, their radiation exposures are generally not measured due to either no radiation exposure or extremely low levels of exposure. Therefore, measurable doses TEDEs greater than zero were used to analyze the actual occupational exposure. The average measurable doses during the years 2003-2012 fell into the range between 1.0-1.6 mSv for all NPPs. These doses were only 16-25% of the average dose of 6.2 mSv y^{-1} received from all radiation sources by an individual in the United States (NCRP 2009). Furthermore, these TEDEs were only 2.0-3.2% of the NRC occupational dose limit. Therefore, annual occupational doses resulting from the routine operation of the U.S. NPPs are considered negligible compared to the typical background radiation dose or the dose limit.

The analysis results showed that the number of individuals with radiation exposure less than 1 mSv y^{-1} accounted for 58-70% of the total number of individuals with measurable doses. Therefore, the majority of individual workers received very little radiation exposure from the normal operation of NPPs, but some of them received

relatively higher radiation doses. This observation means that individual doses for occupational work in NPPs are distributed broadly despite their low average measurable doses.

This study focused on two types of occupational dose distributions; one includes the occupational exposures reported by each NPP licensee, and the other includes the occupational exposures that were combined per individual, taking into account transient individuals who worked at more than one nuclear facility during the monitoring year. Since some of the reported doses belong to the same individual, it is necessary to combine these dose records per individual to obtain the actual occupational dose distribution. However, this study did not use the data, including transient individuals for determining the dose constraint for occupational workers in NPPs, even though the analysis of occupational doses, taking into account transient individuals, was already conducted in the Results of Analysis chapter.

The ICRP introduced two types of assessments for the efficiency of protection: source-related and individual-related assessments (ICRP 1991). The former focuses on the radiation exposure of the individuals exposed from a single source, while the latter focuses on the radiation exposure of an individual from several sources. A dose constraint originates from a single source, and this single source is a single NPP site. That is, the dose records reported by each NPP licensee indicate the source-related radiation exposure resulting from a single source that exists at a single location. On the contrary, the dose records, taking into account transient individuals, are the sum of individual doses resulting from several sources that include multiple facilities at several locations, and these records

indicate the individual-related radiation exposure. Therefore, the occupational exposures that were combined per individual, taking into account transient individuals, are not appropriate to be used as the data for determining a dose constraint.

To apply numerical percentiles to the dose distribution, occupational doses were used in the form of probability distribution and cumulative probability distribution. These dose distributions were obtained by counting the number of individuals in certain dose ranges using the data from the NRC annual reports. The estimated distributions of occupational doses in NPPs during the years 2003-2012 are displayed in Fig. 19. In the figure, the solid and dotted lines indicate the probability distribution and the cumulative probability distribution, respectively. The cumulative probability distribution enables estimating the doses at the 95^{th} and the 99.99^{th} percentiles of a given dose distribution. The 95^{th} percentiles of each dose distribution during the years 2003-2012 fell into the range of 3.8 to 5.9 mSv. On the other hand, relatively higher values, 16.5-24.2 mSv were calculated as the 99.99^{th} percentiles of each dose distribution during the years 2003-2012. Although all values were lower than the NRC annual dose limit, the values from the 99.99^{th} percentiles in each year were approximately 4.5 times higher than those from 95^{th} percentiles of a given distribution. The values of the average measurable and maximum doses and the doses at the 95^{th} and the 99.99^{th} percentiles of occupational dose distributions during the years 2003-2012 are given in Table 33.

Table 33. Comparison of the average measurable and maximum doses and the doses at the 95th and the 99.99th percentiles of occupational dose distributions during the years 2003-2012.

Year	Average measurable dose (mSv)	Maximum dose (mSv)	Dose at 95th percentile (mSv)	Dose at 99.99th percentile (mSv)
2003	1.6	28.9	5.9	24.2
2004	1.5	24.3	5.5	22.4
2005	1.4	29.6	5.3	22.4
2006	1.4	21.9	5.0	17.6
2007	1.3	28.4	4.7	21.8
2008	1.1	26.8	4.2	18.1
2009	1.2	18.5	4.5	17.5
2010	1.1	22.3	4.3	19.3
2011	1.1	18.4	3.9	16.5
2012	1.0	28.7	3.8	21.7
Maximum	1.6	29.6	5.9	24.2
Minimum	1.0	18.4	3.8	16.5
Average	1.3	24.8	4.7	20.1

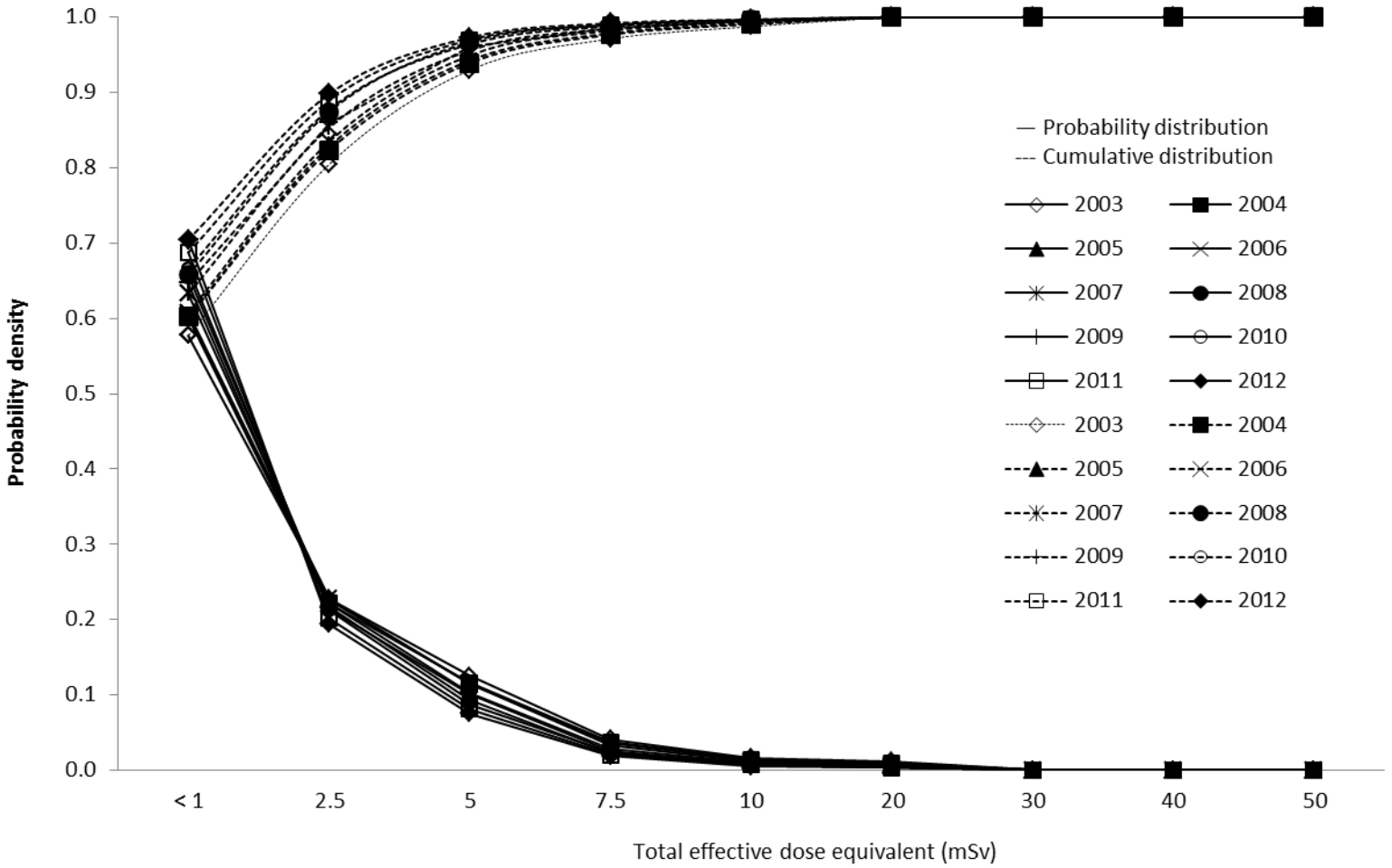

Fig. 19. Probability distribution and cumulative probability distribution of occupational exposure in NPPs during the years 2003-2012.

As shown in Table 33, the average for the doses at 95th percentiles of dose distributions were approximately 5 mSv due to effective radiation safety programs in NPPs. Although Table 33 shows a gradually decreasing trend of the average, it is considered, in this study, inappropriate to use this value as the occupational dose constraint in NPPs. This value is not only one tenth of the current occupational dose limit, 50 mSv, but also even lower than the average dose, 6.2 mSv y^{-1}, from all radiation sources received by a typical individual in the United States (NCRP 2009). Furthermore, even though the fraction of individuals with doses above 5 mSv accounted for only 5% of the total number of individuals with measurable doses in NPPs, the number of individuals whose annual doses were higher than 5 mSv was 2,177 in 2012 due to the very large number of workers in NPPs. Therefore, despite its small percentage in a dose distribution, the number of individuals whose dose levels fall into this highest 5% of the distribution was substantial. That is, if this 5 mSv is used as the occupational dose constraint in NPPs, the costs imposed by this dose constraint, which will be used for lowering individual doses below the dose constraint, can be a financial burden on the routine operation of NPPs.

The doses at 99.99th percentiles of each dose distribution during the years 2003-2012 were calculated to be 16.5-24.2 mSv, and their average was approximately 20 mSv. Since 20 mSv was almost the maximum dose level of occupational workers in NPPs, this value may provide more margin of operating flexibility for NPPs, as compared to the previous average for the doses at 95th percentiles of dose distributions. It is inappropriate to use the dose above 20 mSv y^{-1} as the occupational dose constraint because ICPP Publication 103 recommended 100 mSv in five years as the occupational dose limit.

Although the dose limit for a single year is still 50 mSv, an average of 20 mSv per year over a five-year period will play a role as an upper boundary on the expected dose to comply with the dose limit for occupational workers in NPPs. Therefore, the average dose at the 99.99^{th} percentiles of dose distributions, i.e., 20 mSv, is considered more reasonable for the occupational dose constraint in NPPs than that for the doses at 95^{th} percentiles. However, since the 20 mSv y^{-1} is the same as the average ICRP occupational dose limit over five years, it is concluded, in this study, a dose constraint is not needed for occupational workers in the U.S. NPPs. If the NRC was to implement a dose constraint for occupationally-exposed workers using the dose at 99.99^{th} percentile, the dose constraint would be a redundant regulation.

To analyze the consequences of the given dose level referring to the probability of detriment due to a potential exposure, the corresponding excess risks were estimated using the $ER = D \cdot R_D$ (Equation 11). If the risk coefficient given in ICRP Publication 103, 4.2×10^{-5} per mSv, is used to estimate the excess risk of individual doses for occupational workers in NPPs, the corresponding annual excess risk of 20 mSv y^{-1} is computed as 8.4×10^{-4} (ICRP 2007). The excess risks for the dose limit, the average occupational measurable dose, the average individual doses at the 95^{th} and the 99.99^{th} percentiles of the occupational dose distributions, and the average dose received from all radiation sources by an individual in the United States are given in Table 34.

Table 34. Comparison of excess risks for the occupational dose limit, the average occupational measurable dose, the average individual doses at the 95th and the 99.99th percentiles of occupational dose distributions, and the average dose received from all radiation sources by an individual in the United States.

	Dose (mSv y^{-1})	Risk
Dose limit	50	2.10×10^{-3}[a]
Average occupational measurable dose[b]	1.3	5.46×10^{-5}[a]
Dose at the 95th percentile[c]	5	2.10×10^{-4}[a]
Dose at the 99.99th percentile[d,e]	20	8.40×10^{-4}[a]
Maximum dose	24.8	1.04×10^{-3}[a]
Average dose received from all radiation sources in US	6.2	3.53×10^{-4}[f]

[a] Risk estimates were calculated on the basis of the risk coefficient for adults, 4.2×10^{-5} per mSv, which was given in ICRP Publication 103.
[b] Average dose to occupational individuals with a measurable dose.
[c] The value is the average for the doses at the 95th percentiles of occupational dose distributions during the years 2003-2012.
[d] The value is the average for the doses at the 99.99th percentiles of occupational dose distributions during the years 2003-2012.
[e] Average ICRP dose limit over 5 years.
[f] Risk estimates were calculated on the basis of the risk coefficient for whole population, 5.7×10^{-5} per mSv, which was given in ICRP Publication 103.

Annual risk levels of 10^{-3} are ordinarily accepted by most workers in any industry and are generally regarded as the demarcation between tolerable and unacceptable risk levels for any large part of a work environment (HSE 1992). On this basis, the ICRP has recommended the annual occupational dose limit of 50 mSv, with a limit of 100 mSv in five years (NCRP 1993; ICRP 2007). To compare risk levels associated occupational exposures in NPPs, the previous chart of probability of death for an individual per year of exposure to hazards was used again as shown in Fig. 20 (Wilson 1984). Compared to the risk level of the dose limit in this chart, the excess risks for average measurable and

maximum doses and individual doses at the 95th and the 99.99th percentiles of the actual individual dose distribution around NPPs were regarded as acceptable.

In addition, the fraction of workers that would be affected by the dose at the 99.99th percentiles will be negligible and the economic cost for such collective dose (S^*) reduction will have no net benefit. This can be also expressed as

$$\left(\frac{dX}{dS}\right)_{S^*} > -\left(\frac{dY}{dS}\right)_{S^*} \text{ and} \quad (16)$$

$$\left(\frac{dX}{dS}\right)_{S^*} \approx 0, \quad (17)$$

where X is the cost of protection and Y is the cost of detriment. Therefore, there is no net cost-benefit that can be achieved by implementing a dose constraint for occupational workers in the U.S. NPPs.

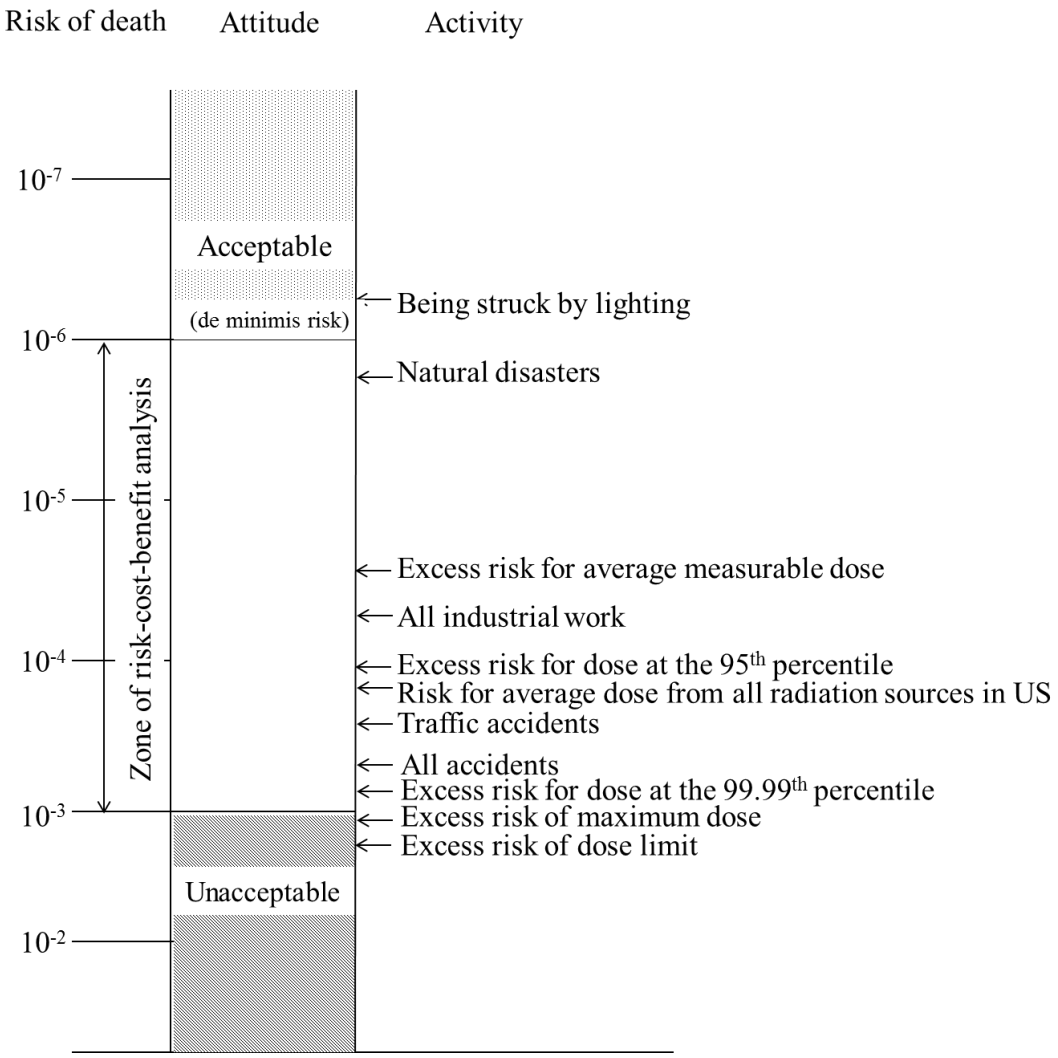

Fig. 20. Risk of death for an individual per year of exposure to hazards in terms of acceptable and unacceptable risk (orders of magnitude) showing the corresponding associated risk for occupationally-exposed workers in NPPs.

CONCLUSION

This research was conducted as a preparation of the revision of current NRC regulations according to ICRP Publication 103, which requires the implementation of concept of dose constraints for members of the public and for occupationally-exposed workers at the U.S. nuclear power plants (NPPs). Under the paradigm of regulatory science, the use of dose constraints is still highly debatable.

This research addressed two objectives. The first objective was determining whether or not dose constraints are necessary for members of the public and occupationally-exposed workers at the U.S. NPPs. The second objective was determining, if dose constraints were needed, the optimal numerical values of dose constraints at the U.S. NPPs. To achieve these objectives, several areas were investigated and analyzed: 1) the establishment of a regulatory-science framework; 2) a system of radiation protection which would incorporate the concept of dose constraints; 3) methodologies and regulations for public and occupational dose assessment; 4) approaches to the establishment of dose constraints; 5) the actual doses and corresponding excess risk for members of the public living around NPPs; and 6) the range of doses and corresponding excess risk for occupationally-exposed workers in NPPs.

An analysis was carried out using exposure data obtained from the NRC from the years 2007 to 2009 for members of the public and from the years 2003 to 2012 for occupational workers. The analysis data finds that the dose distributions for a maximally-exposed individual (MEI) for members of the public and occupationally-exposed workers were lognormal. For members of the public, the annual median and maximum doses to a

MEI were 10^{-4} mSv and 10^{-1} mSv, respectively. The doses at the 95th and the 99.99th percentiles were 10^{-2} mSv and 10^{-1} mSv, respectively. The corresponding annual excess risks (*ER*) for the median and maximum doses were calculated to be on the order of 10^{-8} and 10^{-6}, respectively. These excess risks are very low and should be considered acceptable. The relative increase in excess risk for median dose accounted for only 0.03% and less than 6% for maximum dose. These findings indicate that the relative increase in risk due to radiation exposure from NPPs is trivial compared to the average dose from natural background in the United States of 3.1 mSv. It is concluded that a dose constraint is not necessary for members of the public.

For occupationally-exposed workers, the average and maximum measurable doses were 1.3 mSv and 24.8 mSv. The doses at the 95th and the 99.99th percentiles were calculated as 5 and 20 mSv, respectively. The annual excess risk for the average and maximum doses were on the order of 10^{-5} and 10^{-3}, respectively. These excess risks are also acceptable from the perspective of occupational risks. This analysis showed that some individuals received relatively higher annual doses than others. However, the fraction of the workers in this category was negligible (0.01%) and the economic cost of further dose reduction based on dose constraints will have no net positive benefit. Thus, it is concluded that a dose constraint for occupational workers is not necessary at the U.S. NPPs.

The implementation of dose constraints, as proposed by the ICRP Publication 103, for NPPs was found to be unsupported based on an analysis of the impact of dose constraints in the reduction of annual doses to members of the public and occupationally-exposed workers. The use of dose constraints had no impact on radiation safety at NPPs

in the United States. This conclusion is based on the negligible increase in excess risk. The dose distributions were found to be lognormal and significantly skewed to very low doses. However, it is imperative that dose constraints be put into perspective as the present analysis was made exclusively for the nuclear power industry (NPPs). Dose constraints, however, for other scenarios or industries, such as the medical industry, may require a similar analysis as doses received by occupationally-exposed workers and members of the public (patients) are likely to have different distributions.

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APPENDIX

Comparison of regulatory organizations and their regulations

In terms of nuclear safety, national regulations are influenced by the safety standards of the International Atomic Energy Agency (IAEA). The IAEA is an autonomous organization established through its own international treaty, the IAEA Statute. It provides safety standards to protect the public and environment from harmful effects of nuclear activities (IAEA 2014b). Even though regulating nuclear safety is a national duty, and there is no compulsory obligation for the IAEA member nations to adopt the IAEA safety standards. Nevertheless, regulatory bodies in many member nations use IAEA safety standards in their regulations. The IAEA Safety Standards Series consist of three categories: Safety Fundamentals, Safety Requirements, and Safety Guides (IAEA 2014a). Safety Fundamentals provide the basic objectives and principles of safety. Safety Requirements provide the necessities that must be fulfilled to ensure the protection of people and environment. Last, Safety Guides provide guidance on how to meet the safety requirements. The general process for establishing IAEA safety standards is as follows. First, the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) produces the scientific evaluations of the research data about effects of exposure to ionizing radiation, which is submitted by United Nations (UN) member nations (UNSCEAR 2015). The ICRP uses these scientific evaluations for developing the basic recommendations on radiological protection. These ICRP recommendations are based, not only on scientific results, but also value judgments about balancing of risks and benefits, taking into account societal expectations and experiences (ICRP 2007). The

IAEA applies these recommendations to establish its own safety standards for practical use in the field.

The IAEA safety standards reflect an international agreement on what establishes a high level of safety for protecting people and the environment from detrimental effects of radiation exposure (IAEA 2014a). There are multiple entities involved in the process of developing an IAEA safety standard. These entities are 1) the IAEA Secretariat, 2) the safety standards committees for nuclear safety (NUSSC), radiation safety (RASSC), radioactive waste safety (WASSC) and the safe transport of radioactive material (TRANSSC), and 3) Commission on Safety Standards (CSS). These committees are involved in the review and approval of a standard as shown in Fig. 21 (IAEA 2014a). All IAEA member nations can appoint specialists for the safety standards committees, and they provide remarks on draft standards, taking into account social considerations of their nations. The members of Commission on Safety Standards are also composed of governmental representatives from member nations, and they balance the benefits and risks of new or revised standards for application to their national regulations.

Even though the primary goal of this process is providing impartial criteria for nuclear safety, the judgment is made through a consensus between technical findings and social values. In terms of relationship between the IAEA and the ICRP, especially for radiation protection regulations, the IAEA has established the International Basic Safety Standards (BSS) based on the recommendations of the ICRP. For example, the ICRP has continuously published its recommendations on radiological protection since 1958 with Publication 1. Following updated recommendations were made in 1966 with Publication

9, 1977 with Publication 26, 1990 with Publication 60, and 2007 with Publication 103. The IAEA has also provided the BSS in 1962, 1967, 1982, 1996, and 2014 based on the corresponding editions of the ICRP recommendations (Czarwinski 2011). Many member nations adopt these standards in their regulations. In particular, the application of these standards by EU member nations is required by the European Atomic Energy Community (Euratom) legislation which is regularly updated in collaboration with the development of IAEA BSS (IAEA 2014a).

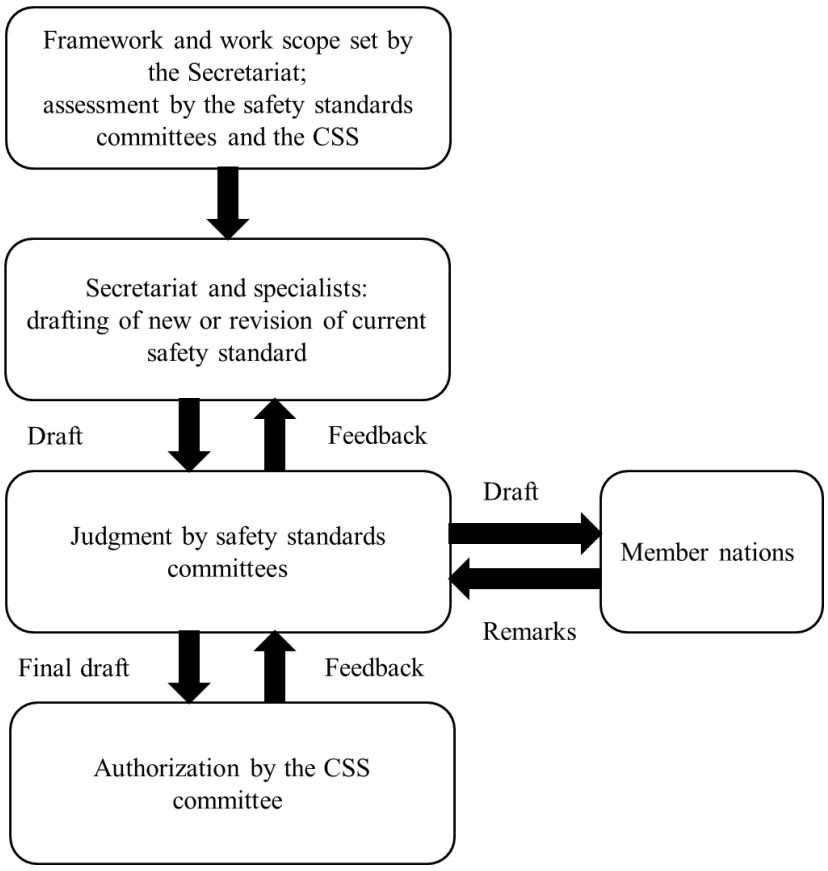

Fig. 21. Development process for IAEA safety standards.

In the United States, the national regulatory system for nuclear safety consists of three bodies: the Department of Energy (DOE), the U.S. Environmental Protection Agency (EPA) and the U.S. Nuclear Regulatory Commission (NRC). The DOE is in charge of the development of nuclear energy policy and the promotion of research programs about reactors, fuel cycles, nonproliferation, etc. (USDOE 2015). The EPA establishes standards and guidance to regulate the offsite releases of radioactive materials from the nuclear facilities (USEPA 2014). Finally, the NRC has a responsibility to regulate commercial application of nuclear materials (USNRC 2012b). Consequently, the NRC regulates most nuclear activities, including NPPs. The NRC provides regulations and guidance to obtain the appropriate protection of workers, the public, and the environment from nuclear related activities. The main components of the development of NRC regulatory requirements are four activities: rulemaking, guidance development, generic communication, and standards development (USNRC 2014c). First, rulemaking is the process of developing and amending regulations with which nuclear licensees must comply to attain or retain a license to use nuclear materials or operate a nuclear facility. Second, guidance development is the process of developing and revising guidance documents, including regulatory guides, standard review plans, and NRC inspection manual to help licensees comply with safety requirements. Third, generic communication is the process of asking applicants and licensees for information about events or regulatory requirements, some of which need response. Last, standards development is the process of developing consensus standards about systems, apparatus, or materials used by the nuclear industry. For developing these standards and guides, the NRC conducts regulatory

research programs for three areas: nuclear reactors, nuclear materials, and nuclear waste (USNRC 2014f). The research conducted by the NRC is normally considered confirmatory research focusing on the regulatory mission, not exploratory research that is research into the unknown. The technical findings and information obtained from the research are published as NUREG-series reports and are used in developing regulatory guides. As mentioned above, a negotiation process between technical findings and social values is necessary for regulatory science to arrive at a consensus on regulations. The NRC also collaborates with standards organizations to have consensus standards (USNRC 2014g). A standards organization uses committees to reach an agreement on the standard. Generally, these committees consist of various stakeholders such as technical specialists from the utilities, vendors, government officials, and, if necessary, the general public. Therefore, it can be said that the regulation is the result of compromising between science and policy.

Evolution of the system of radiation protection

The system of radiation protection has originated in the early attempts to protect people from *X*-rays (Boyd 2012). After the discovery of *X*-rays by Wilhelm Roentgen in 1895, *X*-rays quickly achieved widespread medical and scientific use. Soon after being a popular tool in medicine and scientific research, *X*-rays, especially for large or repeated exposures to human bodies, caused various harmful effects, such as skin burns among patients (Turner 2007). The reports on harmful effects by *X*-ray exposures increased concern about the application of ionizing radiation and many scientists and medical

professionals recognized the need for actions to protect patients and operators from excessive exposure. This is considered the beginning of radiation health protection (Turner 2007). As part of this effort, the foundation of the International Commission on Radiological Units and Measurements (ICRU), which was originally known as the International Committee on *X*-ray and Radium Protection and later as the International Committee for Radiological Units, was proposed by the First International Congress of Radiology (ICR) in 1925 and officially came into being in 1928 (ICRP 2015). The primary goal of the ICRU is to provide recommendations on radiation-related quantities and units, measurement procedures, and reference data for the safe and efficient application of ionizing radiation. In 1928, the ICRP was also established during the second meeting of the ICR, and today, the ICRP plays a crucial role in the development of the system of radiation protection for people and the environment by providing international recommendations and guidance (ICRP 2015a). The most important publications of the ICRP are Publication 2 (the basis of 10 CFR Part 50), Publication 26 (the basis of 10 CFR Part 20), Publication 60 (the basis of current regulations on radiological protection by most IAEA member nations), and Publication 103 (the latest ICRP recommendations, which are supposed to be adopted by most IAEA member nations, including the United States) (USNRC 2015f).

Introduction of the International Commission on Radiological Protection

The ICRP is an international, independent, and non-governmental organization for the radiological protection of people and the environment from ionizing radiations. The

ICRP was formerly the International *X*-ray and Radium Protection Committee (IXRPC), which was established in 1928 to protect people from the harmful effects of *X*-rays (Clarke and Valentin 2009). In 1950, the name of the IXRPC was changed to its present name: ICRP. The ICRP consists of a main commission, a scientific secretariat, five committees, and a series of task groups and working parties (ICRP 2015b). The main commission and the scientific secretariat collaborate to organize the work of the ICRP. In particular, the main commission approves the publication of all ICRP reports, and it is technically advised by five committees: 1) radiation effects, 2) doses from radiation exposure, 3) protection in medicine, 4) application of the ICRP recommendations, and 5) protection of the environment (ICRP 2015b). The committees normally assign the work of task groups that are formed to conduct a specific task such as making a particular ICRP Publication. Sometimes, working parties are established by committee members to solve particular issues. The names of the ICRP committees and their work scope are summarized in Table 35 (Clarke and Valentin 2009; ICRP 2015b).

Table 35. Work scopes of the ICRP Committees.

Number	Name	Work scope
Committee 1	Radiation effects	Risk assessment of the occurrence of cancer and genetic disease (stochastic effects) along with the underlying mechanisms of radiation action; Evaluation of the risks, severity, and mechanisms of occurrence of tissue and organ damage and developmental defects (deterministic effects)
Committee 2	Doses from radiation exposure	Development of dose coefficients for the internal and external radiation dose assessments; Development of reference biokinetic and dosimetric models; Development of reference data for workers and members of the public
Committee 3	Protection in medicine	Development of the protection of persons and unborn children from ionizing radiation during diagnosis, therapy, or biomedical research; Assessment of the medical consequences of accidental exposure
Committee 4	Application of the commission's recommendations	Provision of advice on the application of the recommended system of protection for occupational and public exposure; Communication channel with other international organizations and professional societies focused on protection against ionizing radiation
Committee 5	Protection of the environment	Development of the radiological protection of the environment

The ICRP primary work provides international recommendations and guidance on radiological protection against ionizing radiations. In preparing its recommendations that

cover the overall system of radiological protection, the ICRP focuses on the basic principles and numerical bases on which proper radiological protection measures can be set up, while leaving to national regulatory bodies the duty of developing the specific guidance or regulations that are best suited to the necessities of their individual nations (ICRP 2015b). The ICRP system of radiological protection is a product of regulatory science that involves the social and economic considerations. This system is based on the current knowledge of the science of radiation exposure and value judgements. These value judgements consider socio-economic factors, such as societal expectations, morals, and experience obtained from application of the system. Since the knowledge of the science and the societal expectations has evolved over time, the system of radiological protection has also evolved.

The ICRP provides its recommendations continuously to regulatory bodies and advisory agencies, and these recommendations have been used for providing regulations and guidance for radiation protection in most nations in the world. The ICRP develops recommendations on radiological protection using the technical data of scientific studies provided by the UNSCEAR. The IAEA has used these recommendations as a primary source of information for developing its standards, BSSs. Finally, most of the IAEA member nations adopt the IAEA BSS for making their own regulations for radiation protection. The relationship between different organizations for the use of ICRP recommendations on radiological protection is shown in Fig. 22 (Clarke and Valentin 2009).

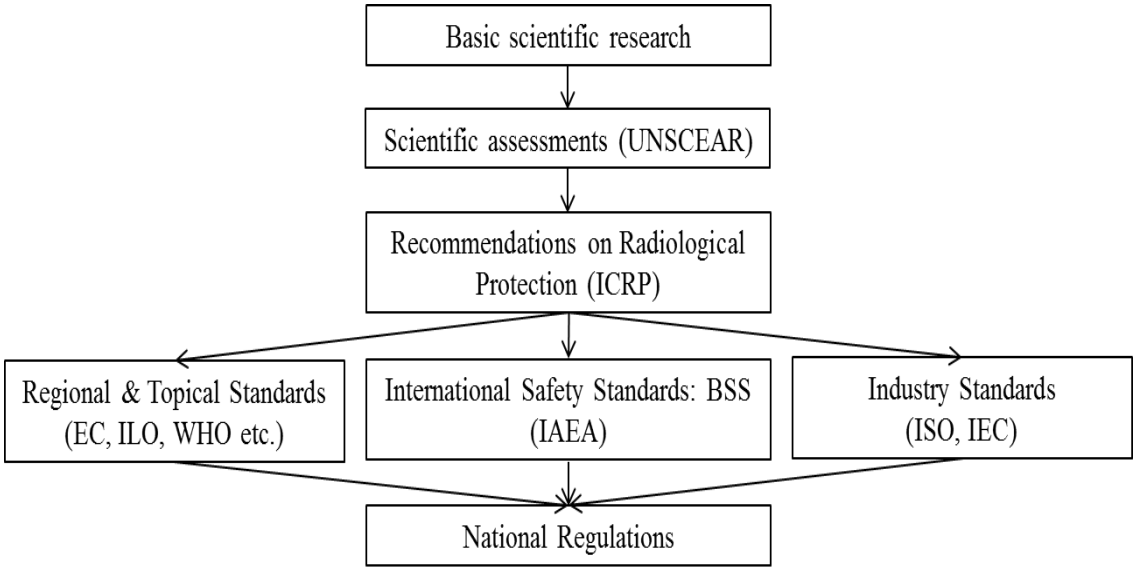

Fig. 22. Development process of ICRP recommendations and their application in various organizations.

Like the ICRP, the National Council on Radiation Protection and Measurements (NCRP) is a non-governmental organization, and it was established by the U.S. Congress in 1964 to distribute information and provide guidance and recommendations on radiation protection and measurements (NCRP 2015). The NCRP also develops its publications to provide guidance for the setting of criteria, standards, and practices of radiation protection in regulatory agencies in the United States (Turner 2007). Therefore, the work of the NCRP work is much similar to that of the ICRP. The main difference between the ICRP and NCRP is that the NCRP aspects of radiation application and exposure environments are unique to the United States (Boice 2012). The NCRP has 100 members and approximately 20 scientific committees that draft recommendations on specific topics

related to radiological protection (NCRP 2015). These recommendations are finally reviewed and approved by the full NCRP council before publication.

Introduction of the major ICRP Publications

The ICRP has issued four major publications of radiological protection: ICRP Publication 2 in 1959, ICRP Publication 26 in 1977, ICRP Publication 60 in 1990, and ICRP Publication 103 in 2007, whose methodology is the basis of current regulations on radiation protection. To estimate the internal radiation exposure, ICRP Publication 2 provided the values of maximum permissible body burden (MPBB) of radionuclide and maximum permissible concentrations (MPC) of these nuclides in air and water (ICRP 1959; PMJ 1960). The MPBB and MPC are defined, respectively, as the maximum amount (inside the body) and the maximum concentration (in air or water) of radioactive material that will produce a dose equivalent to the allowable occupational exposure, such as maximum permissible dose (ICRP 1959; Cember and Johnson 2009). The current requirements of Appendix I to 10 CFR Parts 50 are based on dosimetric methodology in ICRP Publication 2, such as the critical organ dose concept and dosimetric models (USNRC 2015f). The critical organ is the organ that receives the highest dose from the intake of a radionuclide, and this concept was used for the calculation of the MPBB and MPC in ICRP Publication 2. In particular, the MPC values of airborne particulate radioactivity in the workplace were based on the dosimetric model of the respiratory tract, which considered only aerosol radionuclides and two classes of particle solubility, “soluble” and “insoluble,” and modeled the lung as a two-compartment system: the upper

respiratory tract and the deep respiratory tract (Cember and Johnson 2009). All MPC values for occupational exposure were calculated on the condition that the critical organ dose was 3 mSv w^{-1}, corresponding to one-third of the average external occupational exposure 1 mSv w^{-1} or 50 mSv y^{-1} (ICRP 1959). However, in terms of public exposure, the MPC values were not provided in ICRP Publication 2 for members of the public, while it is suggested, in the case of prolonged exposure of a large population, to decrease the permissible level for radionuclides accepted for occupational exposure by a factor of 10 (Clarke and Valentin 2009). The MPC values in ICRP Publication 2 were included in the IAEA's first edition of its BSS in 1962 and in the national regulations for radiation protection, including the United States (Cember and Johnson 2009).

The ICRP issued its 1977 Recommendations as ICRP Publication 26, which first introduced the quantified risks of stochastic effects of radiation and proposed a dose limitation system (ICRP 1977; Clarke and Valentin 2009). In ICRP Publication 26, the scope of radiological protection focused only on activities involving human exposure since the ICRP, at that time, stated that if a man is properly protected then other species are also likely to be adequately protected (ICRP 1977). In terms of protection of human health, the ICRP recognized that most decisions about human activities are based on balancing costs and benefits both in monetary and societal terms (OECD NEA 2011). The ICRP also stated that the formulation of these decision-making procedures could be possible for the radiation protection of all human activities. This led to introduction of the new system of dose limitation with three principles of protection, including 1) no practice shall be adopted unless its introduction produces a positive net benefit, 2) all exposures

shall be kept as low as reasonably achievable, economic and social factors being taken into account, and 3) the doses to individuals shall not exceed the limits recommended for the appropriate circumstances by the Commission (ICRP 1977). Later, these principles became known as justification, optimization, and application of dose limits, respectively. The principle of justification requires that activities involving human exposure should do more good than harm, and that of optimization is used for maximizing the margin of good over harm for the exposed individual or society as a whole. However, the principle of application of dose limits prevents the individual from being exposed to an excessive level of harm even if the benefits exceed the costs. In particular, the application of the principle of optimization was regarded as a more important job by the ICRP, as well as other international and national regulatory bodies, because it required balancing costs and benefits of activities involving human exposure (Clarke and Valentin 2009).

In ICRP Publication 26, the ICRP was also aware that different organs and tissue have different chances of radiogenic cancers occurring. Although the ICRP did not use the term 'effective dose equivalent,' this knowledge of different chances of occurrence led to the introduction of the concept of effective dose equivalent, which reflects the risk of stochastic effects from non-uniform exposure relative to the risk from uniform whole-body exposure (Cember and Johnson 2009). As a result, ICRP Publication 26 recommended the whole body dose-equivalent limits of 50 mSv in a year for occupational exposure and of 5 mSv in a year for public exposure, and these limits include the sum of external and internal radiation dose (ICRP 1977). These values in ICRP Publication 26 were used as the technical basis of the IAEA's second BSS, which was issued in 1982

(Czarwinski 2011). In the United States, the tissue-weighting factors in 10 CFR Part 20 were based on the values in ICRP Publication 26 (Cember and Johnson 2009). In particular, in 10 CFR Part 20, the dose was expressed as the total effective dose equivalent (TEDE) that integrates a risk-based dose, weighted by tissue or organs, as defined in ICRP Publication 26 (USNRC 2015f).

ICRP Publication 26 was superseded by the ICRP 1990 Recommendations as ICRP Publication 60 (ICRP 1991). In the 1980s, there were re-assessments of cancer risk estimates derived from the survivors of atomic bombs at Hiroshima and Nagasaki in Japan. These studies showed that there is a longer latent period for solid tumors compared to leukemia, and the radiation-induced tumors corresponded to a multiplicative model rather than the additive model that had been used previously in ICRP Publication 26 (OECD NEA 2011). This change of model resulted in higher risks of radiation exposure compared to those calculated in ICRP Publication 26. Finally, these new technical findings led to the revision of the previous ICRP recommendations.

In ICRP Publication 60, the ICRP introduced two types of assessments for the efficacy of protection: source-related and individual-related assessments (ICRP 1991). The former focuses on the radiation exposure of all the individuals exposed from a source or group of sources, while the latter focuses on the radiation exposure of an individual from several sources. The ICRP also approved a process-based system of protection, including a "practice," which causes radiation exposure, and "intervention," which decreases exposure (ICRP 1991). For practices, the system of protection recommended by the ICRP was based on the previous general principles given in ICRP Publication 26, but

two new concepts were added to those of ICRP Publication 60: the need to consider risk for potential exposure and the requirement for a constraint in optimization (OECD NEA 2011). In particular, the most important change in the principles was the use of dose constraints in optimization (Clarke and Valentin 2009). According to ICRP Publication 60, dose constraints are defined as the source-related values of individual dose used to restrict the range of choices considered in the process of optimization (ICRP 1991). The ICRP recommended using constraints as individual criteria to ensure that the most exposed individuals are not subjected to unreasonable risk from a single source (Clarke and Valentin 2009). In spite of introducing the concept of a constraint, the ICRP did not provide sufficient guidance on its use and implementation until the issue of ICRP Publication 103 in 2007 (OECD NEA 2011).

In terms of acceptability of risk, ICRP Publication 60 provided three different levels to indicate the degree of tolerability of radiation exposure: "unacceptable," "tolerable," and "acceptable" levels (ICRP 1991; OECD NEA 2011). The first level, unacceptable level, means that the radiation exposure would not be acceptable on any reasonable basis in the routine operation of a practice. The second level, tolerable level, indicates that the radiation exposure was not reviewed as acceptable by the ICRP, but can reasonably be tolerated. The last level, acceptable level, was the condition that the radiation exposure could be accepted without further improvement or with the optimization of protection. In this frame, the dose limit was determined at the boundary between unacceptable level and tolerable level, and dose constraint is determined at the

boundary between tolerable level and acceptable level. The schematic diagram of the acceptability of risk is shown in Fig. 23 (OECD NEA 2011).

In terms of dose limits, ICRP Publication 60 recommended an average of 20 mSv per year over five years (100 mSv in five years) with no more than 50 mSv in a single year for occupational exposure, the values of which corresponds to a risk of death of approximately 1 in 1000 per year (ICRP 1991). For public exposure, the ICRP has recommended a dose limit of 1 mSv in a year, the values of which correspond to a risk of death of approximately 1 in 10,000 per year (ICRP 1991; OECD NEA 2011). The change of numerical values in ICRP Publication 60 led to the revision of IAEA BSS in 1996. The BSS of 1996 had been used by a lot of IAEA member nations to make their own national regulatory programs until the issue of IAEA's fourth BSS in 2014 (IAEA 1996; IAEA 2014a).

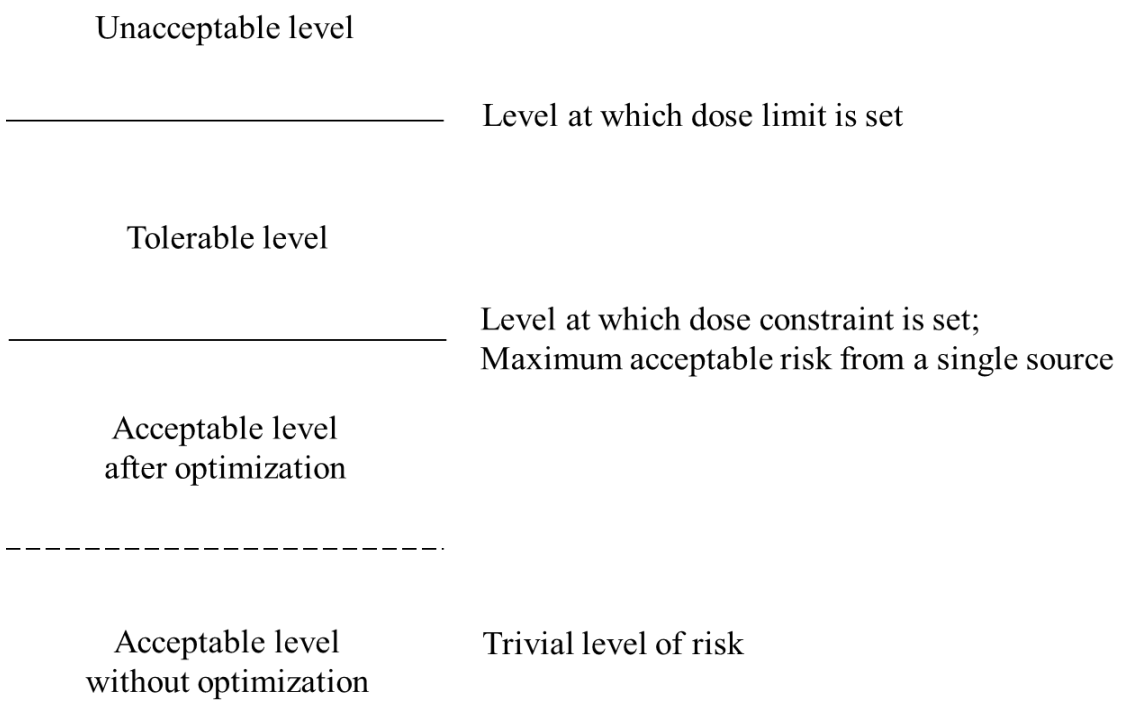

Fig. 23. Schematic diagram of the acceptability of risk.

There were several significant changes in ICRP Publication 60 in comparison with ICRP Publication 26. From the perspective of protection philosophy, the "system of dose limitation" was expanded to the "system of radiological protection" which reinforces the principle of optimization using dose constraints. In terms of technical aspects, the carcinogenic risk factors were increased. New organs and tissues were classified and given their own tissue-weighting factors. The quality factors were also replaced by radiation weighting factors based on relative biological effectiveness (RBE). A summary of the numerical values published by the ICRP Publication 26 to ICRP Publication 60 is provided in Table 36 (OECD NEA 2011).

Table 36. Comparison of protection criteria between ICRP Publication 26 and ICRP Publication 60.

Factors	Exposures	ICRP Publication 26	ICRP Publication 60
Dose limits	Occupational exposure including recovery operation	50 mSv y^{-1} [a]	50 mSv y^{-1} [d] (100 mSv over 5 years)
	Any individual organ, expect:	500 mSv y^{-1} [b]	Dropped
	• lens of the eye	300 mSv y^{-1} [b]	150 mSv y^{-1} [e]
	• skin	20 Sv (in a life time)[b]	500 mSv y^{-1} [e]
	• hands and feet	-	500 mSv y^{-1} [e]
	• pregnant women, remainder of pregnancy	Working condition B[c] (<15 mSv y^{-1} [a])	2 mSv to the surface of abdomen or 1 mSv from intake of radionuclides
	Public exposure	5 mSv y^{-1} [a, f]	1 mSv y^{-1} [d]
	Any individual organ, expect:	50 mSv y^{-1} [b]	Dropped
	• lens of the eye	-	15 mSv y^{-1} [e]
	• skin	-	50 mSv y^{-1} [e]
Dose constraints	Occupational exposure	-	≤20 mSv y^{-1} [d]
	Public exposure	-	-
Accident/ Emergency	Occupational exposure		
	• life saving	-	No dose restrictions
	• other urgent rescue operation	500 mGy whole body 5 Gy individual organ	~500 mSv[d] ~5 Sv[e] (skin)
	• other rescue operation	-	-
	Public exposure		
	• foodstuffs	10 mSv[a, g]	
	• stable iodine	50-500 mSv(thyroid)[b, g]	
	• sheltering	5-50 mSv in 2 days[a, g]	No change
	• evacuation	50-500 mSv in 1 week[b, g]	
	• relocation	100 mSv (1st year)[a, g]	

[a] Effective dose equivalent.
[b] Dose equivalent.
[c] Not exceeding three-tenths of the dose equivalent limits.
[d] Effective dose.
[e] Equivalent dose.
[f] Changed to 1 mSv y^{-1} after 1985 Paris Statement.
[g] Averted dose.

Since ICRP Publication 60, subsequent ICRP reports have continuously been published on radiological protection with new technical findings. As the number of these publications has increased, there is a strong need for the consolidation of these publications to improve the ICRP Recommendations. Finally, the ICRP issued their 2007 recommendations as ICRP Publication 103, which replaced the previous recommendations of ICRP Publication 60 (ICRP 2007). Despite the revision of ICRP recommendations over 17 years, ICRP Publication 103 is similar to the previous ICRP 1990 Recommendations in most respects, and its numerical values for dose limits are not changed from ICRP Publication 60 (USNRC 2008b).

ICRP Publication 60 was superseded by the ICRP 2007 Recommendations as ICRP Publication 103 (ICRP 2007). These recommendations introduced the situation-based approach to cover all potential radiation exposure situations, including planned exposure situations, emergency exposure situations, and existing exposure situations (ICRP 2007; OECD NEA 2011). In particular, ICRP Publication 103 regards the source-related principle of optimization below the dose constraint or reference level as the most effective tool for protection regardless of exposure situations (ICRP 2007).

In terms of biological risks, the fatal cancer and the total detriment nominal risk coefficients are approximately 8-25% lower in ICRP Publication 103 compared to ICRP Publication 60 (ICRP 1991; ICRP 2007). These changes are attributed to two main reasons: the use of incidence data of cancer rather than mortality data and the changes in the estimates of hereditary diseases induced by radiation exposure (OECD NEA 2011). The nominal risk coefficients for all cancers in ICRP Publication 60 and ICRP Publication 103

are given in Table 37 (ICRP 2007). Even though the risk coefficients in ICRP Publication 103 are somewhat lower than those in ICRP Publication 60, the ICRP judged that the existing dose limits continue to offer an appropriate level of protection, and the values of dose limits in ICRP Publication 60 are retained in ICRP Publication 103 (ICRP 2007). The ICRP concluded that these slight differences of risk coefficients are not practically important in the field of nuclear industry.

Table 37. Comparison of detriment-adjusted nominal risk coefficients for stochastic effects between ICRP Publication 60 and ICRP Publication 103.

Exposed population	Cancer		Heritable effects		Total	
	Publ. 60	Publ. 103	Publ. 60	Publ. 103	Publ. 60	Publ. 103
Whole[a, b]	6.0	5.5	1.3	0.2	7.3	5.7
Adult[a, b]	4.8	4.1	0.8	0.1	5.6	4.2

[a] Detriment-adjusted risk indicates the probability of the occurrence of a stochastic effect, modified to allow for the different components of the detriment to express the severity of the consequence. The unadjusted nominal risk coefficients are calculated by averaging estimates of the radiation-associated lifetime risk for cancer incidence for a composite population of equal numbers of males and females.

[b] Unit: percent per Sv (10^{-2} Sv^{-1}).

There are three main differences between ICRP Publication 60 and ICRP Publication 103: risk estimates, the system of protection, and the use of dose constraints and reference levels. The risks from radiation exposure, given in ICRP Publication 103, decreased slightly compared to ICRP Publication 60, but there was no change of dose limits. The system of protection was also changed from a process-based approach to a

situation-based approach with an emphasis on source-related control. Lastly, the application of dose constraints and reference levels was not an obligation in ICRP Publication 60, but it became a requirement in ICRP Publication 103, and the ICRP has strongly recommended using dose constraints and reference levels for decision-making by licensees and regulatory bodies. In terms of values for protection criteria, the comparison of those values between ICRP Publication 60 and ICRP Publication 103 are shown in Table 38 (ICRP 2007; OECD NEA 2011).

Table 38. Comparison of protection criteria between ICRP Publication 60 and ICRP Publication 103.

Factors	Exposures	ICRP Publication 60	ICRP Publication 103
	Planned exposure situations		
Individual dose limits[a]	Occupational exposure including recovery operation	50 mSv y^{-1} (100 mSv over 5 years)	No change No change
	• lens of the eye	150 mSv y^{-1} [b]	50 mSv y^{-1} [b, c] (100 mSv over 5 years)
	• skin	500 mSv y^{-1} [b]	No change
	• hands and feet	500 mSv y^{-1} [b]	No change
	• pregnant women, remainder of pregnancy	2 mSv to the surface of abdomen or 1 mSv from intake of radionuclides	1 mSv to the embryo and fetus
	Public exposure	1 mSv y^{-1}	No change
	• lens of the eye	15 mSv y^{-1} [b]	No change
	• skin	50 mSv y^{-1} [b]	No change
Dose constraints[a]	Occupational exposure	≤20 mSv y^{-1}	No change
	Public exposure		
	• general	-	<1 mSv y^{-1}
	• radwaste disposal	≤0.3 mSv y^{-1}	No change
	• long-lived radwaste disposal	≤0.3 mSv y^{-1}	No change
	• prolonged exposure	≤~1 & ~0.3 mSv y^{-1} [d]	No change
	• prolonged component from long-lived nuclides	≤0.1 mSv y^{-1} [e]	No change
	Medical exposure		
	• volunteers for biomedical research, if benefit to society is;		
	minor	<0.1 mSv	No change
	intermediate	0.1-1 mSv	No change
	moderate	1-10 mSv	No change
	substantial	> 10 mSv	No change
	• comforters and caregivers	5 mSv per episode	No change

Table 38. (continued)

Factors	Exposures	ICRP Publication 60	ICRP Publication 103
	Emergency exposure situations		
Interventional levels[a, f, g]		Interventional levels	Reference levels
	Occupational exposure		
Reference levels[a, g]	• life saving	No dose restrictions	No dose restrictions[i]
	• other urgent rescue operation	~500 mSv; ~5 Sv[h](skin)	1000 or 500 mSv[i]
	• other rescue		≤100 mSv[i]
	Public exposure		
	• foodstuffs	10 mSv	
	• stable iodine	50-500 mSv(thyroid)[h]	
	• sheltering	5-50 mSv in 2 days	
	• evacuation	50-500 mSv in 1 week	
	• relocation	100 mSv (1st year)	
	• overall protection strategy		20-100 mSv y^{-1} (in planning stage)
	Existing exposure situations		
Action levels[a]		Action levels	Reference levels
	Radon		
Reference levels[a, j]	• at home	3-10 mSv y^{-1} (200-600 Bq m^{-3})	<10 mSv y^{-1} (<600 Bq m^{-3})
	• at work	3-10 mSv y^{-1} (500-1500 Bq m^{-3})	<10 mSv y^{-1} (<1500 Bq m^{-3})

[a] Effective dose unless otherwise specified.
[b] Equivalent dose.
[c] Values are based on the ICRP Statement on Tissue Reactions in 2011 (ICRP 2011).
[d] The dose constraint of no more than about 0.3 mSv would be appropriate.
[e] To be considered if dose assessment methodologies to ensure compliance under any conceivable situation of combination of doses are not available.
[f] Averted dose.
[g] Intervention levels refer to averted dose for specific countermeasures.
[h] Equivalent dose.
[i] Effective doses below 1000 mSv should avoid serious deterministic effects; below 500 mSv should avoid other deterministic effects.
[j] Reference levels refer to residual dose and are used to evaluate protection strategies, as opposed to the previously recommended intervention levels which referred to averted doses from individual protective actions.

Tritium effluents discharged from nuclear power plants in the United States

Even though the radionuclides discharged from nuclear power plants (NPPs) differ depending on the types of reactor, the major radionuclide for gaseous and liquid effluents is tritium, and most of the public doses originate from tritium. In a pressurized water reactor (PWR), boric acid is added to the reactor coolant system as a chemical shim to control reactivity, and tritium is produced primarily from neutron capture by ^{10}B in a PWR (Martin 2009). In contrast, tritium production is lower in boiling water reactors (BWRs) than that in PWRs since boric acid is not used to control reactivity in BWRs. For BWRs, tritium is mainly generated by neutron activation of the small amounts of deuterium present in water.

In this study, it is found through the analysis of radioactive effluents released from NPPs during the years 2007-2009 that most of the activities resulted from tritium. As shown in the Results of Analysis chapter, the contribution of tritium activity to the total activity of radioactive effluents was significant in both BWRs and PWRs, accounting for more than 90%. Since tritium is one of the most important radionuclides in NPPs and its contribution to the activity is substantial, the amount of tritium effluents discharged from NPPs to the environment was analyzed in detail to understand its effect on NPP effluent releases. The total activities depending on the types of effluents and reactors are displayed in Table 39 (USNRC 2011b, 2012e, 2013d). As shown in Table 39, the activity in tritium effluents discharged from PWRs was 17-22 times higher than that from BWRs due to the higher production of tritium in PWRs. In terms of type of effluents, for BWRs, the tritium activities in gaseous effluents were higher than those in liquid effluents. Contrary to BWRs,

most of tritium activities in radioactive effluents released from PWRs resulted from liquid effluents. The main reason why BWRs and PWRs have different contributions of liquid and gaseous effluents to the total activity is the recycling of liquid waste at BWRs (USNRC 2013d). Many BWRs in the United States reuse either some or all of their liquid waste. Furthermore, BWRs do not use boron in the reactor coolant unlike PWRs, and this also contributes to reduce liquid effluents from BWRs (USNRC 2013d). Histograms of tritium effluents discharged from NPPs during the years 2007-2009 are presented in Figs. 24-26.

Table 39. Activity in tritium effluents discharged from NPPs during the years 2003-2012.

Year	Effluent	BWR (Bq)				PWR (Bq)			
		Maximum	Median	Minimum	%[a]	Maximum	Median	Minimum	%[a]
2007	Liquid	4.85×10^{12}	7.55×10^{8}	8.44×10^{5}	0.06	6.55×10^{13}	2.04×10^{13}	5.51×10^{12}	94.35
	Gaseous	2.45×10^{13}	1.30×10^{12}	1.33×10^{11}	99.94	4.35×10^{13}	1.22×10^{12}	1.10×10^{11}	5.65
	Total	2.94×10^{13}	1.30×10^{12}	1.33×10^{11}	100.00	1.09×10^{14}	2.16×10^{13}	5.62×10^{12}	100.00
2008	Liquid	4.70×10^{12}	9.07×10^{10}	4.30×10^{7}	6.92	6.14×10^{13}	1.95×10^{13}	5.88×10^{12}	94.11
	Gaseous	1.92×10^{13}	1.22×10^{12}	9.79×10^{8}	93.08	5.76×10^{13}	1.22×10^{12}	9.78×10^{10}	5.89
	Total	2.39×10^{13}	1.31×10^{12}	1.02×10^{9}	100.00	1.19×10^{14}	2.07×10^{13}	5.98×10^{12}	100.00
2009	Liquid	3.89×10^{12}	3.43×10^{11}	4.41×10^{6}	27.31	7.66×10^{13}	2.06×10^{13}	3.56×10^{12}	93.03
	Gaseous	3.60×10^{13}	9.14×10^{11}	6.80×10^{10}	72.69	1.58×10^{14}	1.54×10^{12}	5.15×10^{10}	6.97
	Total	3.99×10^{13}	1.26×10^{12}	6.80×10^{10}	100.00	2.35×10^{14}	2.21×10^{13}	3.61×10^{12}	100.00

[a] Percentages are based on median values.

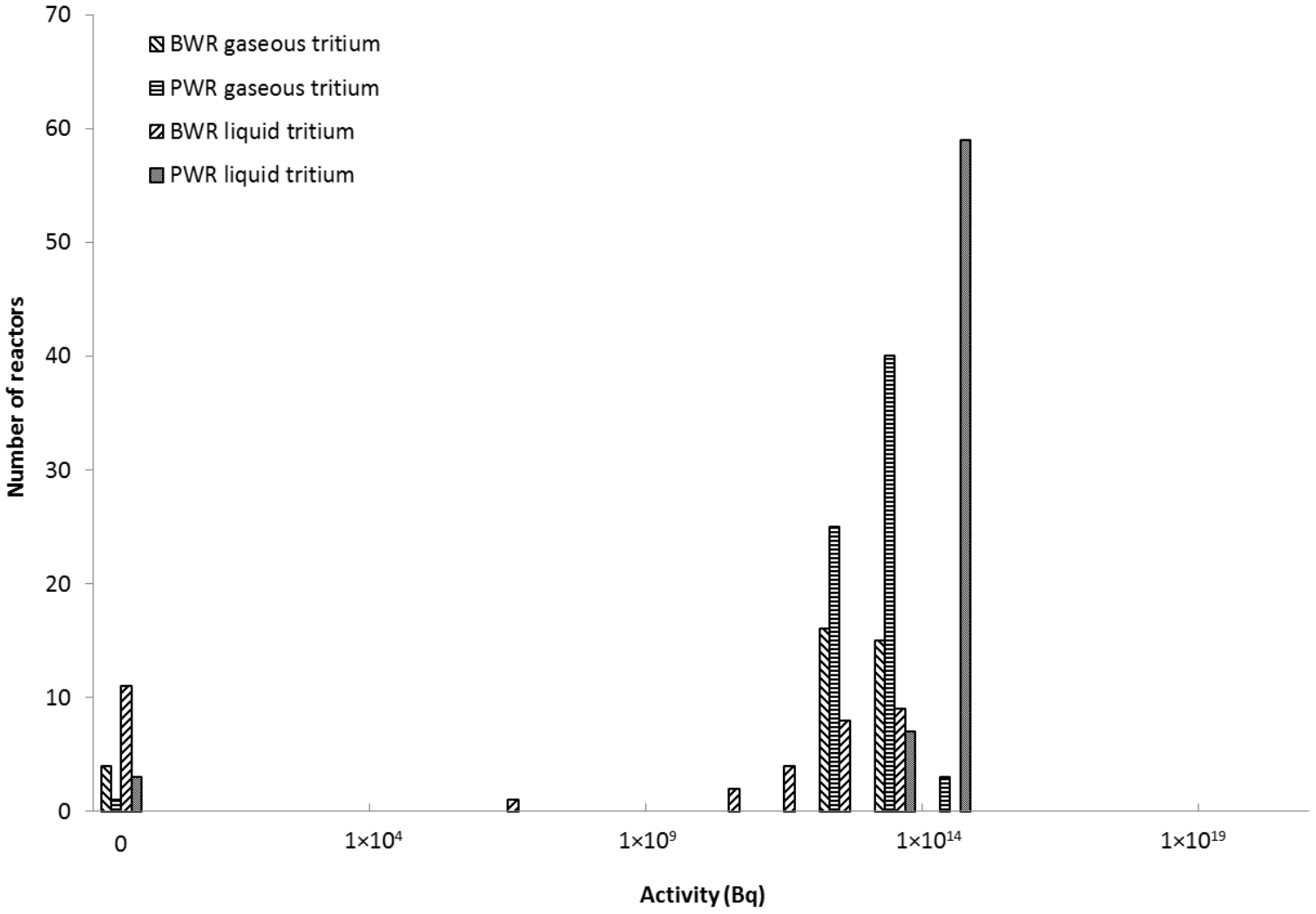

Fig. 24. Tritium effluents discharged from NPPs in 2007.

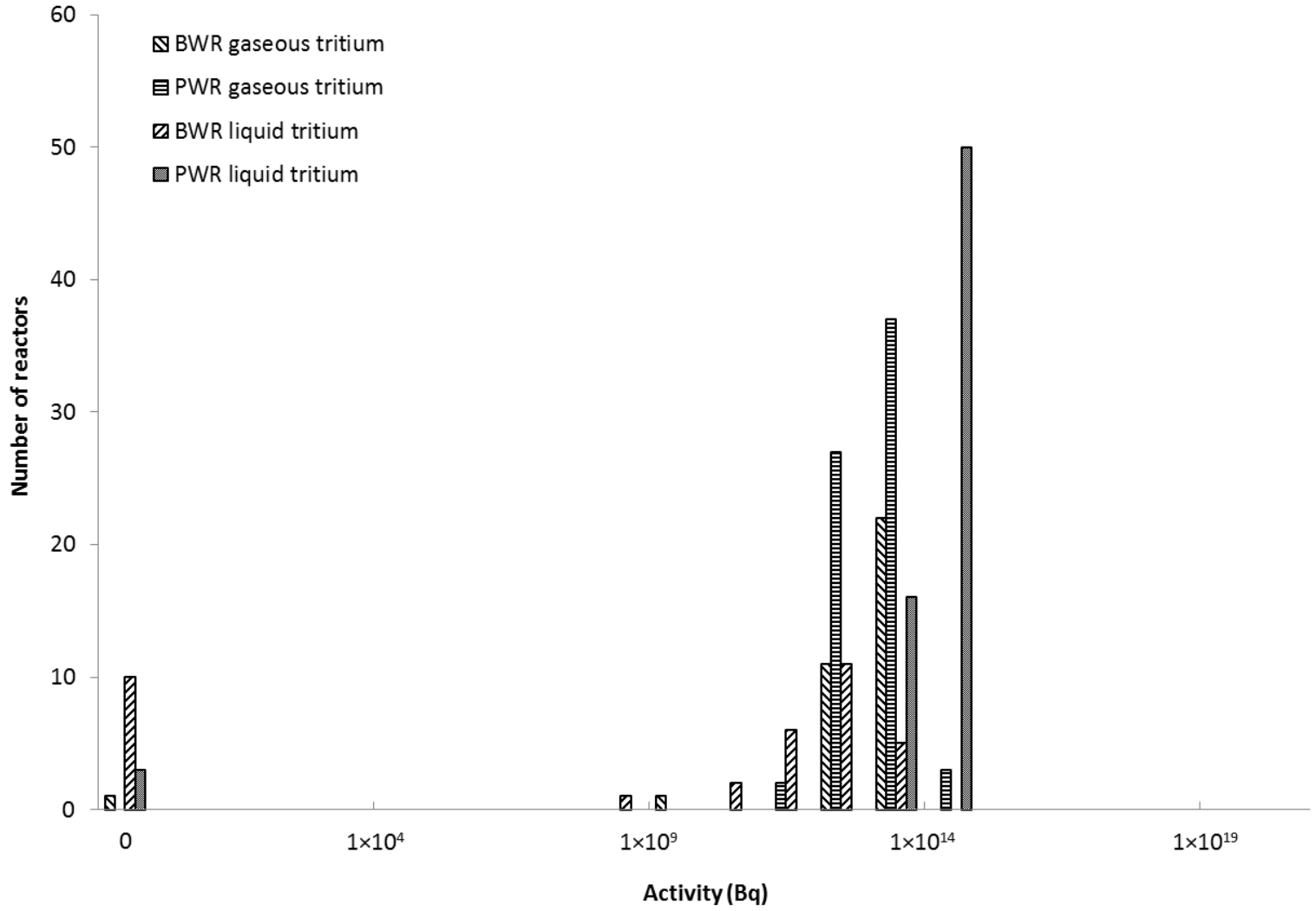

Fig. 25. Tritium effluents discharged from NPPs in 2008.

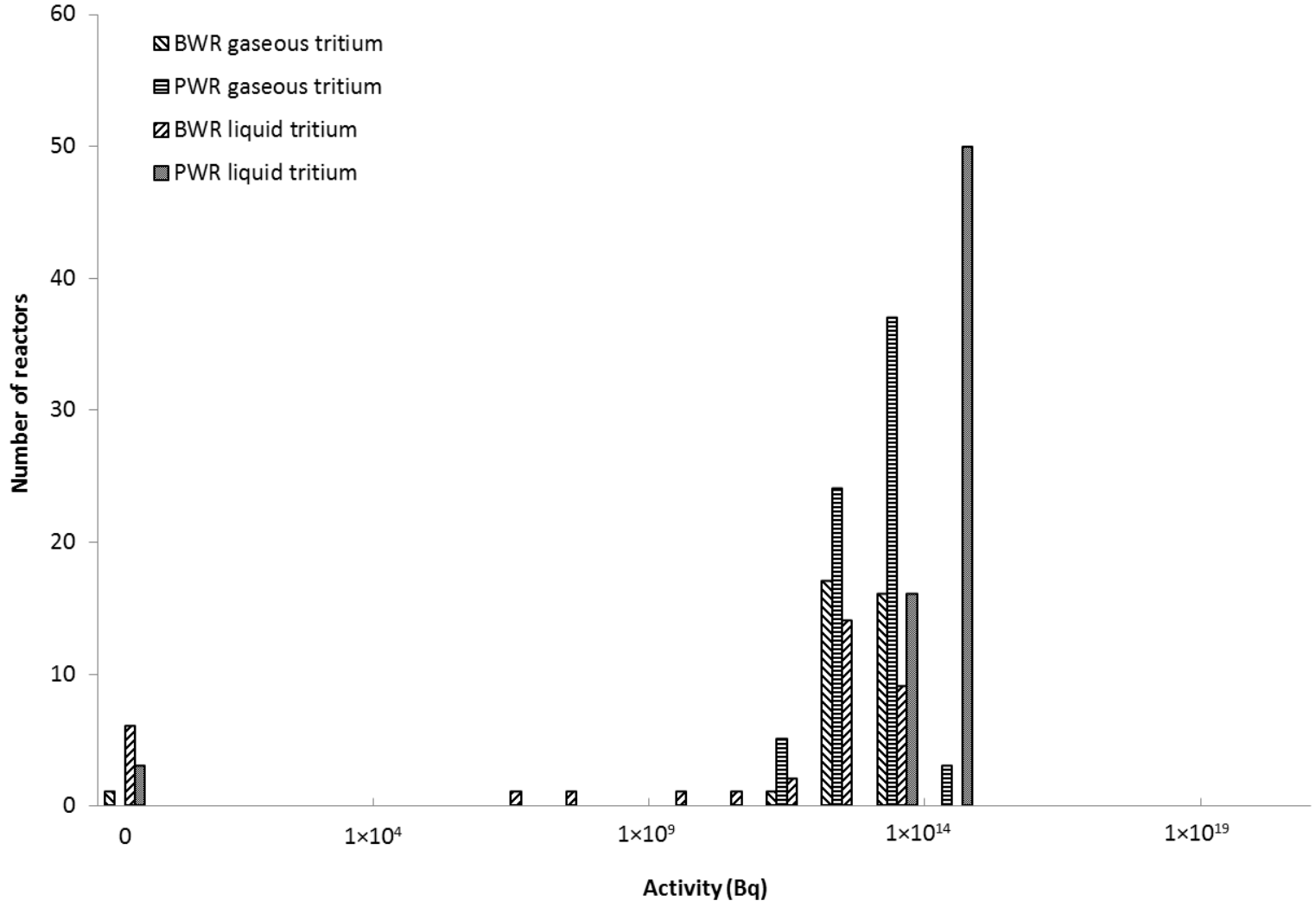

Fig. 26. Tritium effluents discharged from NPPs in 2009.