# Mayak Worker Dosimetry Study Project 2.4

# Volume I:

# Overview of Dose Assignment Methodology for Mayak Workers


**Project 2.4:**

**Mayak Production Association**

E. Vasilenko, M. Gorelov, M. Smetanin, V. Knyazev, I. Teplyakov

**Southern Urals Biophysics Institute**

V. Khokhriakov, V. Khokhriakov, Jr., N. Koshurnikova, N. Shilnikova, P. Okatenko, V. Kreslov, M. Bolotnikova, M. Sokolnikov, K. Suslova, S. Romanov

**University of Ekaterinburg**

O. Alexandrova

**University of Utah**

S. Miller, M. Krahenbuhl, D. Choe, L. Anspaugh

**Oak Ridge National Laboratory**

K. Eckerman

**Pacific Northwest National Laboratory**

J. Fix, R. Scherpelz, B. Napier

**Project 2.2:**

**Southern Urals Biophysics Institute**

N. Koshurnikova

**U.S. National Cancer Institute**

E. Gilbert


## March 15, 2007

# Abstract

Project 2.4 Phases I through III, have been conducted to reconstruct external radiation and plutonium intake organ doses for the Mayak worker cohort employed at the Mayak Production Association (Mayak) in Ozyorsk, Russia, at any time between 1948 and 1972 and consisting of 18,831 workers [14,072 males (75%) and 4,759 females (25%)]. Mayak was the site of the first Soviet nuclear weapons production facility and is comparable to the Hanford site in Washington State. The Phase III reconstructed doses are contained in a database called Mayak Worker Doses-2005, herein after referred to as Doses-2005, that will be used by Project 2.2 researchers. The Phase I and II databases are known as Doses-1999 and Doses-2000, respectively.

The Doses-2005 Users Guide is presented in three volumes as follows:

- Volume I (this document) provides an overview of the general methodology; the results of scoping studies of the magnitude of dose to workers from other potential significant exposure pathways identified as neutron radiation, internal intake other than plutonium, airborne effluent, and occupational related medical x-rays; and the structure of the Doses-2005 database analysis files for external radiation and plutonium intake.

- Volume II, "Dose Assignment Methodology Used to Calculate Annual Organ Doses to Mayak Workers from External Radiation.," provides a description of the methodology used to calculate the annual organ dose to MPA workers from external radiation.

- Volume III, "Dose Assignment Methodology Used to Calculate Annual Organ Doses to Mayak Workers from Plutonium Intake," provides a description of the methodology used to calculate the annual organ dose to MPA workers from plutonium intake.

Doses-2005 includes the organ dose from occupational external radiation and from plutonium intake. Two analysis files are provided because of the significant difference in exposure parameters.

- The Doses-2005 External Radiation Analysis File of organ doses contains 250,443 records. Each record contains the historical archive dose entered into the Mayak Production Association (MPA) records at the time of measurement; a reconstructed absorbed dose in air at the point of the dosimeter in consistent units for all years, a reconstructed absorbed dose in air for photon radiation based on radiation field spectral and directional characteristics (i.e., workplace exposure scenario), reconstructed Personal Dose Equivalent for photon, and separately, neutron radiation based on worker orientation (i.e., worker exposure orientation) in the workplace radiation fields and the photon absorbed dose to the 18 organs specified in the dosimetry protocol. All workers have an assigned work group category and a primary exposure scenario for each year of employment.

- The Doses-2005 Plutonium Intake File of organ doses contains the results of analyses of 6,785 workers and an evaluation of autopsy data for 449 workers for a total of 7,234 workers.

The process of dose reconstruction is complex and requires knowledge of the response characteristics of the Mayak dosimeters (based on laboratory energy and angular response measurements); radiation type, spectral and directional characteristics in the workplace (to assess the absorbed dose in air); and the

orientation of the worker in the workplace (to assess the personal dose equivalent and the respective absorbed organ doses). The process requires convolution of point estimates and uncertainties associated with the dosimeter radiation response, workplace radiation field and worker orientation. Primary exposure scenarios were defined and used in the analyses to reconstruct doses and to estimate uncertainties.

In the course of Phase III, scoping studies of other potential sources of significant occupational doses were done. Results of these studies were as follows:

- Neutron radiation dose to Mayak workers is comparatively low on average in relation to the photon radiation dose. However, there are some workers with significant neutron dose. While the corresponding photon dose might in fact be substantially higher, there is a need to further examine the general issue of neutron dose for selected workplaces (plutonium) and work activities (work on reactor while operating).

- Sources of potential occupational internal dose other than plutonium should be considered. The "Original Mayak Worker Cohort" included only selected employees who worked at the primary Mayak reactor, chemical and plutonium chemical-metallurgical facilities from the beginning of operations to the early 1970s based on the quality of their health and dosimetry records, and their work history. Workers exposed to nuclides other than plutonium at the Mayak industrial complex such as tritium, $^{238}$Pu, $^{90}$Sr, $^{241}$Am, and a range of other radionuclides were excluded in the selection of the original Mayak Worker Cohort. For example, production of $^{238}$Pu and $^{90}$Sr for power sources began in the 1960s using production reactor facilities that began production later than the period of the highest occupational exposures in the late 1940s to mid-1950s for workers at the plutonium production reactors, radiochemical and plutonium plants.

- The magnitudes of the environmental doses associated with airborne effluents, although relatively large in comparison with today's environmental standards, are unlikely to result in the need to estimate organ doses for specific workers in the cohort. Assuming continuous lifetime exposures at the rates described in section 7.3, total effective and total selected organ doses environmental pathway doses do not exceed effective doses of about 6 or 7 rem (and for most workers doses should be significantly less than these conservatively estimated screening doses). The only exception to this would be for workers exposed in the late 1940s through late 1950s, for whom thyroid doses approaching 30 rem (0.3 Sv) are possible. As such, epidemiological studies of potential radiation effects on the thyroid should likely consider the environmental exposure to radioiodines.

- The analysis of medical x-ray dose involved 84,982 x-ray examinations for 8,500 workers. The study showed a substantial likelihood of significant doses to Mayak workers from routine medical x-ray examinations. The study demonstrated that the medical x-ray doses to workers are highly variable and for some workers could be greater than the occupational dose. This could be particularly true for workers in the early years with lower occupational exposure and for essentially all workers hired after 1962. A closely related topic concerns determination of the organ doses for individual workers from medical x-ray examinations. X-ray examinations involve partial body irradiation with photon energies that are typically lower than workplace photon irradiation. The analysis of organ doses could have a significant effect on the comparison with the organ doses from occupational radiation in Doses-2005. Approximately 50% of the

available medical x-ray examination data for individual Mayak workers has been computerized. Resolution of this issue is a priority in the analysis of Doses-2005.

Researchers plan in Phase IV to conduct tasks to expand the study cohort to provide Project 2.2 researchers with improved statistical power in analyses, to address the findings noted above, and to improve the accuracy of the dosimetry data, and thus to improve radiation risk estimates. An important finding in the Mayak Worker Study, even with the highest level of protracted occupational radiation doses among available occupational worker studies (i.e., Hanford, Sellafield, IARC 15 Country), concerns the potential importance of radiation exposure from other pathways such as the potential significance of medical x-ray dose based on an analysis of 84,982 examinations for about 8,500 workers.

# Contents

# Tables

# Figures

# Acronyms and Abbreviations

| | |
|---|---|
| α | alpha particle |
| AP | anterior-posterior (or front-to-back) irradiation of the body |
| β | beta particle |
| CMSD | Central Medico-Sanitary Department |
| DCF | dose conversion factor |
| DOE | U.S. Department of Energy |
| dpm | disintegrations per minute |
| DTPA | diethylenetriaminepentaacetic acid |
| ESE | entrance skin exposure |
| EURT | East-Urals Radioactive Trace |
| FIB-1 | First Institute of Biophysics |
| γ | gamma ray |
| GSF | National Research Center for Environment and Health; Munich, Germany |
| GIT | gastrointestinal tract |
| Gy | gray |
| $H_p(d)$ | personnel dose equivalent at depth d in tissue. $H_p(0.07)$ and $H_p(10)$ represent the shallow and deep dose at depths of 0.07 and 10 mm, respectively. |
| ICRP | International Commission on Radiological Protection |
| ID | identification |
| IDC | electronic ionization personnel dosimeter |
| IDM | individual dosimetric monitoring |
| IFK | First Mayak personnel film dosimeter |
| IFK+Pb | Second Mayak personnel film dosimeter |
| IFKU | Mayak multi-element personnel film dosimeter |
| ISO | isotropic |
| JCCRER | Joint Coordinating Committee for Radiation Effects Research |
| keV | kilo (thousand) electron volts, a unit of energy |
| MeV | million electron volts, a unit of energy |
| MPA | Mayak Production Association |
| PA | posterior-anterior |
| PMT | photomultiplier tube |
| R | roentgen, unit of exposure to ionizing photons in air |
| rem | roentgen equivalent man |

| | |
|---|---|
| ROT | rotational |
| SRG | Scientific Review Group |
| SUBI | Southern Urals Biophysical Institute |
| UA | urine analysis |

# 1.0 Mayak Worker Study

## 1.1 Introduction

Since the mid-1990s, the U.S. Department of Energy (DOE) has funded several work tasks to extract data for use in epidemiologic studies of Mayak workers. Numerous studies have already been completed using preliminary dose information (Koshurnikova et al. 1994, 1999, 2000, 2002; Gilbert et al. 2000, 2004). Crucial and often complex tasks have been completed by the Mayak Production Association (MPA), Southern Urals Biophysics Institute (SUBI), and U.S. researchers to improve the respective external radiation, internal radiation, medical, and vital status database information. Unlike the Japan atomic bomb survivors, whose collective experience serves as the primary basis for current radiation protection risk estimates, the Mayak workers were exposed to radiation for protracted periods. In addition, the Mayak workforce was exposed to ionizing radiation at levels significantly higher than those received by the historical DOE workforce and any other known monitored workforce. Accordingly, the Mayak worker cohort offers a unique opportunity to evaluate site-specific cancer mortality risks in a cohort containing both male and female workers under conditions that approximate those of typical nuclear workforces, and therefore might be more likely to help researchers better characterize occupational radiation risk to such workers (BEIR 2006).

MPA was the first industrial complex in the former Soviet Union built for the production of plutonium. The history of the MPA and the associated exposure to workers has been described in a number of publications (Khokhryakov et al. 2000, Romanov et al. 2002). Briefly, construction of the first reactor began in 1945 and construction of the chemical processing plant started the following year. The reactor became operational in 1948. The MPA first produced "finished" plutonium in 1949. The Mayak complex eventually included five nuclear reactors with associated chemical processing and plutonium chemical metallurgical plants for the production of plutonium.

Figs. 1.1 and 1.2 show the location approximately of the MPA in Russia near the city of Ozyorsk. In the early years, the MPA was referred to as Kyshtym (a town actually located to the west of the complex), then as a post office box code of "Chelyabinsk-40", and changed in the early 1990s to "Chelyabinsk-65". In 1994, this was replaced with "Ozyorsk", which was the original name of the town nearest to the MPA complex. Ozyorsk is located about 70 km north of the city of Chelyabinsk.

Several thousand workers in these plants in the early years received comparatively high doses of radiation from external radiation and internal sources, particularly plutonium intakes. There are several reasons for these high exposures, particularly during the early years (1949 through the early 1950s). Reactor, chemical processing plant, and plutonium chemical-metallurgical facility capabilities and technologies were rapidly emerging and the processes were not well understood. In the early years, there was poor understanding of the consequences of relatively high occupational radiation exposures, and there were limitations in MPA resources and capabilities to protect workers.

## 1.2 The Joint Coordinating Committee for Radiation Effects Research

The Joint Coordinating Committee for Radiation Effects Research (JCCRER) was established as a bilateral committee representing Federal agencies from the United States and ministries from the Russian Federation to coordinate the respective project tasks. The original agreement that established this collaboration was signed on January 14, 1994. This agreement identified areas of primary research focus (or direction) as follows:

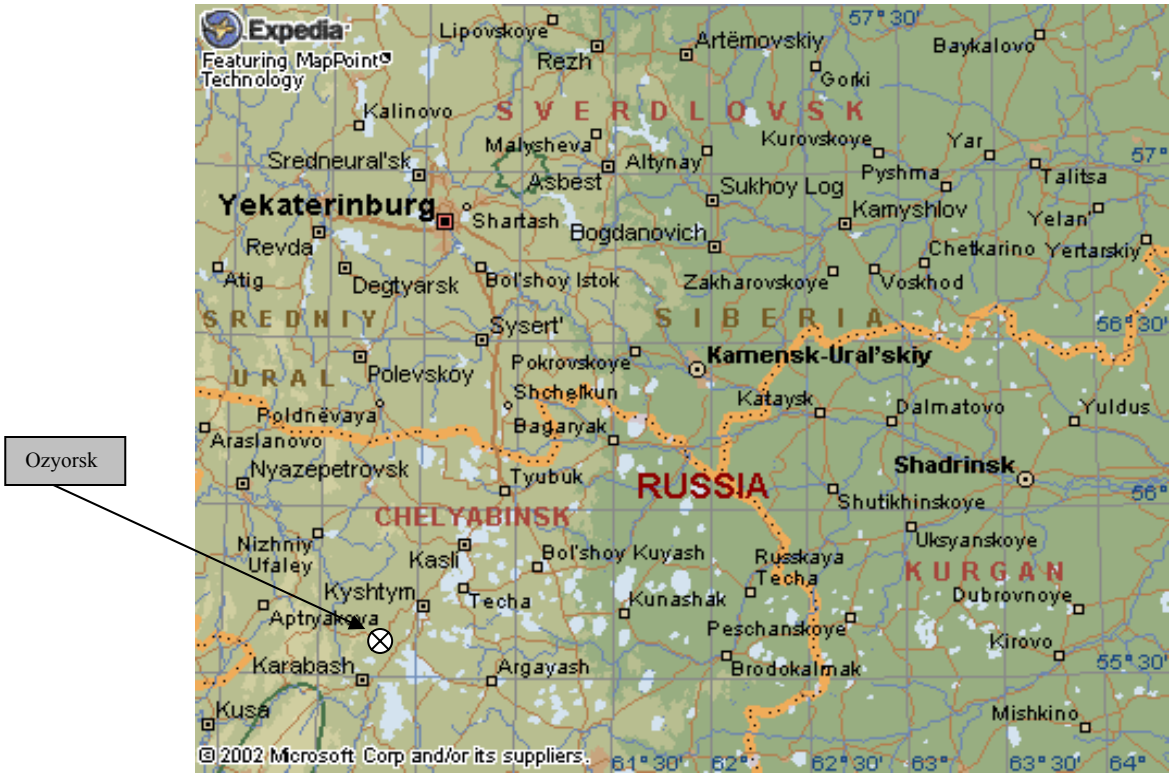



Fig. 1.1. Cheyabinsk region of Russia in southern Urals.

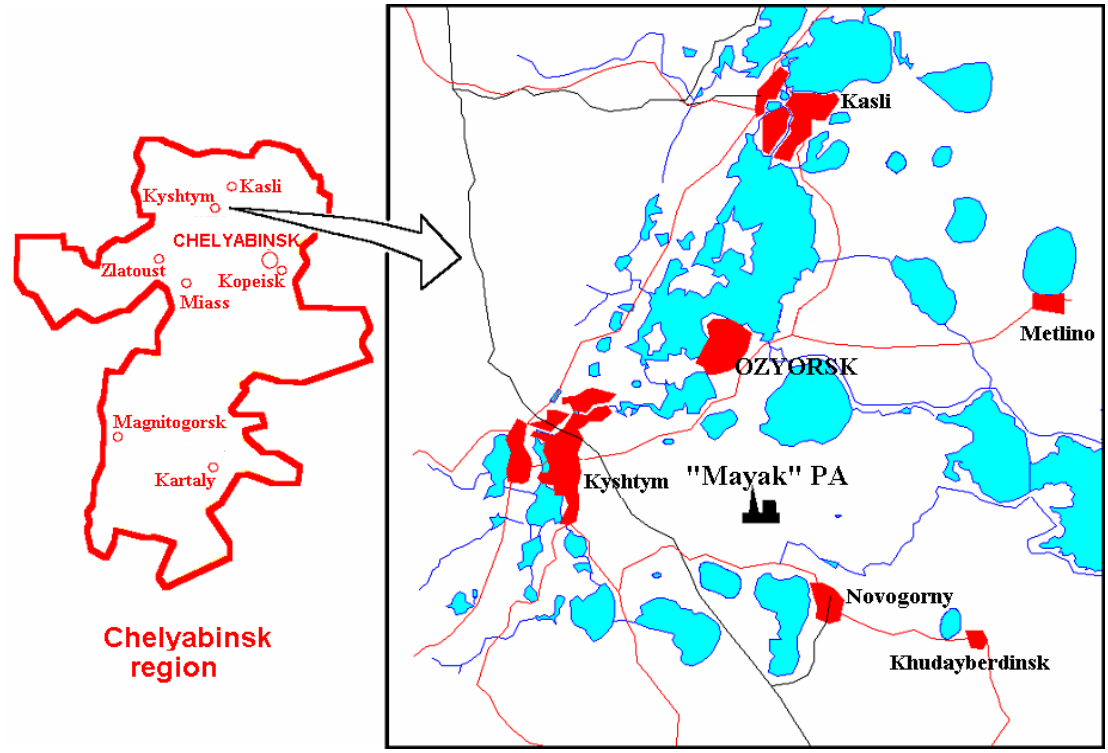



Fig. 1.2. Mayak facility located near Ozyorsk and surrounding region

### 1.2.1 Direction 1

Studies conducted under Direction 1, "The health consequences of radiation exposures downwind and downstream of the MPA (more recently termed the "Techa River studies") have been summarized by Mokrov et al. (2000) and Degteva et al. (2000). Direction 1 Project areas established by the JCCRER to evaluate doses to the population include:

- Project 1.1–Techa River Population Dosimetry.
- Project 1.2a–Physical preservation of data related to the Techa River population at the Urals Research Center for Radiation Medicine.
- Project 1.2b–Techa River Population Epidemiology.
- Project 1.4–Ozyorsk Population Dose Reconstruction from Mayak Atmospheric Releases.

### 1.2.2 Direction 2

Project 2.4, Mayak Worker Dosimetry, is the focus of this report. Descriptions of work done to date in which the radiation dosimetry information is used are presented in Khokhryakov et al. (2000), Romanov et al. (2002), Koshurnikova et al. (1994, 1999, 2000, 2002) and Gilbert et al. (2000, 2004). Direction 2 project areas established by the JCCRER include:

- Project 2.1–Metabolism and Dosimetry of Plutonium Industrial Compounds.
- Project 2.2–Estimation of Risk of Stochastic (Cancer) Effects of Occupational Radiation Exposure, also known as Mayak Worker Epidemiology Study.
- Project 2.3–Deterministic Effects of Occupational Exposure to Radiation.
- Project 2.4–Development of an Improved Dosimetry System for the Workers at the Mayak Production Association, also known as Mayak Worker Dosimetry Study.
- Project 2.5–Microdosimetry for plutonium-induced lung disease.
- Project 2.6–Molecular Markers of Lung Cancer in Mayak Workers.
- Project 2.7–Biomarkers of Radiation Exposure.
- Project 2.8–Mayak Worker Tissue Repository.
- Project 2.9–Database Integration.

### 1.2.3 Direction 3

Studies under Direction 3 involved "Development of information technologies to support decision-making in the event of a radiation accident." This area of research was established by the JCCRER to develop information technologies to support decision-making in the event of a radiation

accident. Work under Directions 1 and 2 was considered to be a priority and thus little work has been performed under Direction 3.

## 1.3 SCIENTIFIC REVIEW GROUPS

Projects conducted under JCCRER auspices are reviewed by independent DOE and Russian Scientific Review Groups (SRGs). The primary purpose of the respective SRGs is to provide scientific peer review, evaluate research progress, and make recommendations to DOE and Designated Russian Federation Ministries on research priorities.

## 1.4 DOSIMETRY PROTOCOL

A Dosimetry Protocol was developed to better ensure collaboration and communication between Project 2.2 epidemiology and 2.4 dosimetry researchers to estimate organ dose. The protocol has been routinely updated to address researcher concerns and expectations. The protocol was collaboratively developed and revised based on input from Project 2.2 and 2.4 researchers. The protocol specifies the study cohort for which doses were provided as available and specifies the organs for which doses are included in Doses-2005. The general methods used to calculate organ doses from the archive external radiation doses are also included.

### Mayak Worker Epidemiology Study Cohort

The focus of MPA dose reconstruction activities in Phase III (Doses 2005) has been to evaluate and reconstruct dose for the original cohort of 18,831 workers of which 15,815 (84%) workers have recorded dose. There were 14,072 (74.7%) and 4,759 (25.3%) male and female workers, respectively. There have been 16 accidents involving 54 workers. These workers are not in the analysis database because of large uncertainties in the dose. The original cohort consisted of workers who were hired before 1972 and worked in the main reactor, chemical processing or plutonium chemical-metallurgical plants. Table 1.1 provides several statistical values for this original cohort of 18,831 workers. For example, the number of facilities among the reactor, radiochemical, and plutonium chemical metallurgical plants were: 76, 75, and 90, respectively. The number of work areas in the three primary plants totaled 188, 242 and 263, respectively. A total of 1,852, 3,005 and 3,277 significant workplaces were identified among the three primary plants. Mayak had a total of 57,027 occupational history and 250,443 annual dose records that were used in the process of reconstructing annual doses for these workers. The highest annual recorded dose, and year in parenthesis, for the primary reactor, chemical and plutonium chemical-metallurgical plants were: 570 R (1949), 844 R (1951) and 408 R (1950), respectively.

Two proposals have been made to expand the Mayak worker cohort since the original cohort selection as follows:

- The addition auxiliary water treatment and maintenance plant workers during the period of 1948 through 1982 to provide relative low-dose within-study controls.

- The addition of workers first employed between 1973 and 1982.

The original cohort plus the two additions would total about 26,000 workers. However, there was not enough time to complete dose reconstruction for these additions in the development of Doses-2005. Doses-2005 contains the measured archive doses, as available for the 15,725 workers with recorded doses, and reconstructed doses for the entire original cohort of 18,831 workers as described in this document.

Table 1.1. Significant values from the Mayak external radiation dosimetry database[a]

| Parameter | Parameter value |
|---|---|
| Production | |
| Number of main plants | 3 |
| Number of facilities | |
| Reactor plant | 76 |
| Radiochemical plant | 75 |
| Plutonium plant | 90 |
| Number of areas | |
| Reactor plant | 188 |
| Radiochemical plant | 242 |
| Plutonium plant | 263 |
| Number of work places | |
| Reactor plant | 1,859 |
| Radiochemical plant | 3,005 |
| Plutonium plant | 3,277 |
| Number of employees | |
| Total | 18,831 |
| Male | 14,072 |
| Female | 4,759 |
| Primarily involved in Reactor plant operation | 6,676[b] |
| Primarily involved in Radiochemical plant operation | 8,561[b] |
| Primarily involved in Plutonium plant operation | 6,540[b] |
| With recorded doses | 15,815 |
| With recorded daily doses | 8,748 |
| With reconstructed doses | 3,016 |
| Number of records | |
| Occupational histories | 65,505 |
| Annual doses | 250,443 |
| Daily doses | 725,350 |
| Maximum recorded total dose (R)/reconstructed air dose (Gy) | 1,129/9.3 |
| Maximum recorded annual dose(R)/reconstructed air dose (Gy)/year of registration | |
| Reactor plant | 570/4.4/1949 |
| Radiochemical plant | 844/3.5/1951 |
| Plutonium plant | 408/1.5/1950 |
| Collective reconstructed air dose (person-Gy) | |
| Reactor plant | 2,904 |
| Radiochemical plant | 7,263 |
| Plutonium plant | 728 |
| Total duration of work /personnel dosimetry monitoring (thousand person-years) | |
| Reactor plant | 90.5 / 77.7 |
| Radiochemical plant | 116.8 / 109.7 |
| Plutonium plant | 103.1 / 60.5 |

(a) Data derived from database query dated 28 April 2006.
(b) The sum of the number of workers at the four plant categories do not equal the total number of workers because some workers were at multiple plants.

## 1.5 PROJECT 2.4, DOSES-2005 DATABASE USERS GUIDE

This Doses-2005 Users Guide is presented in three volumes as follows:

- Volume I - Overview of Phase III Doses-2005
- Volume II - Dose Assignment Methodology used to Calculate Annual Organ Doses to Mayak Workers from External Radiation
- Volume III - Dose Assignment Methodology used to Calculate Annual Organ Doses to Mayak Workers from Plutonium Intake

This document is Volume I and contains overview information as follows:

Chapter 2 - Description of Phases of the Mayak Dosimetry Study

Chapter 3 - Description of MPA Operational History relevant to estimating doses to MPA workers.

Chapter 4 - Description of MPA and SUBI archive record systems relevant to reconstructing dose for MPA workers.

Chapter 5 - Overview of external organ dose methodology from Volume II.

Chapter 6 - Overview of plutonium organ dose methodology from Volume III.

Chapter 7 - Descriptions of bounding analyses conducted to estimate potential MPA worker exposure to neutron radiation, airborne effluent, internal intakes other than plutonium and medical x-rays.

Chapter 8 - Doses-2005 database structure for external radiation organ doses and plutonium intake organ doses, respectively. Also some descriptive information comparing the archive and reconstructed dose.

Chapter 9 - Perceived strengths and weaknesses of Doses-2005 for consideration by researchers using this data in epidemiologic analyses and work tasks underway under Phase IV.

# 2.0 Study Phases

The Mayak Worker Dosimetry Study has progressed through three phases as described in this chapter. A more detailed history of methods used to assess Mayak worker external radiation and plutonium intake doses is presented in Volumes II and III, respectively. Generally, external radiation doses are assessed by the MPA and the internal radiation doses originally by Branch No. 1 of the Moscow Institute of Biophysics (FIB-1) and later by SUBI.

## 2.1 PROGRESSION AND DEVELOPMENT OF EXTERNAL RADIATION DOSES

The general history of three distinct phases of external radiation dose databases used in the evaluation of Mayak worker health effects is presented in the following sections. The databases are identified according to the year of first use in analyses.

### 2.1.1 Phase I - Doses-1999

Initially, external dosimetry data in the Mayak Worker radiation registry at SUBI were extracted for 18,831 workers from paper records, and computerized by SUBI researchers in the early 1990s. At this stage, completed in 1999, the database "Doses-1999" of archive dose data was created and submitted to epidemiologists. In parallel with this effort, work was underway under Mayak Worker Dosimetry Study Project 2.4 by MPA Radiation Safety department personnel to independently computerize its archive data. Subsequently, these two files were checked against each other, and discrepancies were resolved primarily using the MPA Radiation Safety Department dosimetry information. In addition, MPA worker occupational histories and work location data were computerized and also used in the analysis of the worker data. The Occupational History Database contains individual worker work history information.

In the process of creating the Occupational History Database, a list of the MPA plants was made. The three primary plants include reactor, radiochemical, and plutonium/chemical-metallurgical plants. The database contains a list of facilities at each plant and a list of areas at each facility. Furthermore, each area has several workplaces.

At MPA, a "workplace" is defined as the location of the worker's permanent or temporary assignment where he performs his duties. Each workplace has been characterized by the type of operation performed, equipment used; radioactive materials used, and source term. Since it is not possible to perform dose reconstruction for all the workplaces listed in the Occupational History Database, workplaces with similar radiation fields and exposure conditions were grouped into radiation groups, where several radiation groups were identified for each plant. Each radiation group is characterized with its own source term depending on the type of equipment used and radioactive materials used in the process.

In accordance with the administrative and technological structure of the MPA, each worker is associated with a certain area, facility and plant. Over the entire history of the MPA, workplaces were subject to changes. Those changes were connected with modifications in process and administration.

The worker's occupational history represents a chronological list of all the workplaces (plants, facilities, areas) and occupations in which the individual worker was involved. The occupational history takes into account the dates of the person's employment, change in workplace (facility to facility, plant to plant), change of occupation, and termination of employment. As a result, it is possible to reconstruct the history

of worker radiation exposure based on the worker's occupational history and knowledge of radiation sources in the respective plants, facilities and workplaces.

### 2.1.2 Phase II - Doses-2000

Ongoing improvements in the consistency of the Mayak worker dose estimates were incorporated by October 2000 into an improved electronic database entitled "Doses-2000." This database contained reconstructed doses calculated from the original registered doses. These validated doses used adjustments to the original recorded doses to account for:

- Adoption of a single dose quantity since different quantities had been used historically.
- Adjustments to the respective MPA personnel dosimeter anterior-posterior energy response for calculated workplace exposure scenarios.
- Estimates of the neutron dose to workers using workplace exposure scenarios and the use of neutron-to-photon dose ratios.

### 2.1.3 Phase III - Doses-2005

The Phase III "Doses-2005" database, described in this report, contains organ doses from improved reconstructed external doses. For the majority of workers with an external radiation archive dose, the reconstruction is based on analyses of workplace radiation spectral and directional fields, dosimeter response characteristics, etc. For the workers without an external radiation archive dose, estimates of dose were made using the occupational history to determine the availability of relevant coworker dose and/or the use of exposure scenarios based on the methods described in this Users Guide. For all workers, there is an assigned occupational category (i.e., group) and one or more exposure scenarios. Doses-2005 includes enhancements in the reconstructed dose using measured dosimeter energy and angular response characteristics, effects of beta radiation on dosimeter response, and considerations of the workplace radiation field spectral and directional parameters and the geometry of worker exposure.

## 2.2 PROGRESSION AND DEVELOPMENT OF PLUTONIUM INTAKE DOSES

The progression in the assessment of plutonium intake dose can generally be categorized according to:

- Before Project 2.4, doses calculated with the FIB-1 model.
- Doses calculated with the updated FIB-1 model referred to as Doses-1999.
- Doses calculated with the updated Doses-1999 model referred to as Doses-2000.
- Doses, as described herein and in detail in Volume III, calculated with an updated Doses-2000 model referred to as Doses-2005.

# 3.0 MPA Operational History and Processes[1]

## 3.1 INTRODUCTION

The history of MPA relevant to the Mayak Worker Study involved pioneering accomplishments regarding reactor, chemical processing , and plutonium chemical-metallurgical facility design and construction; processes and operations; capabilities to handle large quantities of radioactive materials; and capabilities to detect, measure and control worker radiation exposure. The MPA history also includes relatively high levels of radiation exposure to workers, particularly during the earliest years of operation. Some workers were sufficiently exposed to exhibit clinical symptoms of radiation sickness, particularly during the early years of operations. MPA workers received medical examinations and the examination results were considered in worker job assignments.

## 3.2 MPA FACILITIES

The first MPA reactor and chemical processing and plutonium chemical-metallurgical facilities were constructed in the late 1940s. Constructing these first-of-a-kind facilities, developing the unique technological equipment, and commissioning new processing capabilities were hampered by a general lack of scientific knowledge and process experience. Ongoing upgrades and modifications of the constructed plants and facilities continued during the following approximately 20 yr. The upgrades were performed without stopping facility operations in general and reactor operations in particular, which would have also reduced plutonium production. This situation required workers to perform work in high-radiation fields.

### 3.2.1 Reactor Facilities

The first reactor built to produce plutonium, called Annushka ("A" for short), was a natural-uranium-fueled, single-purpose, water-cooled, graphite-moderated, single-pass reactor. Reactor A became operational on June 19, 1948, and was temporarily shut down during January 1949 for repairs. The reactor was returned to full capacity near the end of March 1949. Operation of four additional plutonium production reactors was commissioned at Mayak during the period from 1950 through 1952 as summarized in Table 3.1. The basic design of these reactors is very similar. Routine refueling operations involved placing new fuel elements in the process channels in the Central Hall area located above the reactor core. The irradiated fuel was normally discharged from the bottom of the core.

[1] More detailed information is presented in P.2.4.2004.14, "Historical Description of Mayak PA Facility Operations and Radiation Protection Practices."

Table 3.1. MPA Reactor Facilities

| Reactor[(a)] | Primary Production[(b)] | Stages | Startup | Shut Down |
|---|---|---|---|---|
| A | $^{239}Pu$ | Operation commissioned: 06/01/48<br>Shutdown for major repairs: 1/49<br>Restart: 3/49<br>Major overhaul began: 10/01/63 | 6/1/48 | 6/16/87 |
| AV-1 | $^{239}Pu$ | Operation Commissioned 4/5/50 | 4/5/50 | 12/8/89 |
| AV-2 | $^{239}Pu$ | Operation commissioned 4/15/51<br>AV-1 and AV-2 reactors were combined into one consolidated plant: 01/01/54 | 4/15/51 | 7/14/90 |
| AI | $^{239}Pu$ | Operation commissioned: 12/22/51<br>Reactor reached its design capacity: 02/14/52<br>Major repair involving partial replacement of the graphite stack: 03/03/56<br>Start of isotope production campaign: 12/24/56 | 12/22/51 | 5/25/87 |
| AV-3 | $^{239}Pu$ | Operation commissioned: 09/15/52<br>Actual startup: 10/04/52<br>Reactor reached its design capacity: 10/30/52<br>Major overhaul: 04/27/54 | 9/15/52 | 11/1/90 |

(a) Reactors A, AI, AV-1, AV-2 and AV-3 are of a water cooled graphite-moderated design.
(b) Irradiation of fuel elements containing uranium in order to generate plutonium.

### 3.2.2 Chemical Processing Facilities

The MPA Plant B, subsequently DB, radiochemical facilities were designed to conduct plutonium extraction from the irradiated fuel elements removed from the reactors. The history of these facilities is summarized in Table 3.2. The extraction process greatly reduced impurities in the plutonium from uranium, fission and/or activation product nuclides. The hold time of irradiated fuel elements before dissolution in Plant B was 45 d during the early years of operation. This was later lengthened to 120 d. The first finished plutonium product was issued from the foot-end process of Plant B in February 1949. Operations at Plant B revealed many shortcomings in facility and process design. Significant contamination was present on Plant B equipment, walls, floors, and air in practically all the rooms. The plant was essentially under continuous modification during 1950 - 1951 to resolve problems with an average annual dose of about 100 rem to workers. The installation of improved corrosion-resistant equipment, the removal of contaminated equipment, improved process, and improved radiation monitoring methods gradually resulted in stable Plant B operation with ongoing repairs.

Plant DB was constructed to provide improved workplace conditions for MPA workers and to install technology adequate to achieve stable radiochemical production. During September 1959, a part of Plant DB, the “northern chain,” was put in operation. On November 2, 1961, a second (equivalent) part of the plant, the “southern chain,” was put into operation. Plant DB incorporated a “Three-Zone” structure involving:

Table 3.2. MPA Radiochemical Facilities

| Plant | Function | Stages of development | Shutdown |
|---|---|---|---|
| B | Extract plutonium from irradiated uranium fuel | Plant startup - 9/1948<br>First load of irradiated blocks received - 12/22/48<br>Production of first product - 2/1949<br>Intensive reconstruction of the plant during 1950-1951 | Reconstructed into plant "DB" |
| DB | | Startup of the northern line of the plant: 09/15/59<br>Startup of the southern line: 11/02/61<br>Acetate-sorption technology application: 1964<br>Extraction-sorption technology application: 06/14/76 | Weapons-grade plutonium production was stopped in 1987 |

- **First Zone** - Process equipment was protected with a thick shield of heavy concrete.
- **Second Zone** - Corridors for piping, valves and chemical sampling were provided for solution transfers and providing chemical reagents.
- **Third Zone** - Personnel occupied rooms containing process control boardrooms, stock rooms, passageways for personnel, etc.

Construction of Plant DB during 1957-1959 was done under conditions of high radioactive contamination because of the 1957 accident (described in Section 7.3). Plant DB was routinely improved to further increase its efficiency, product quality, and safety. An acetate-sorption process was adopted at plant DB in 1964. An extraction-sorption process was adopted on June 14, 1976 and this process was used until 1987 when operations at the plant were stopped.

### 3.2.3 Plutonium Chemical-Metallurgical Facilities

MPA plutonium chemical-metallurgical plants were used to achieve higher purification of plutonium solutions received from the chemical processing plant and to produce highly pure plutonium metal. Construction of the first of these facilities, as summarized in Table 3.3, began in March 1948 with the pilot industrial Facility No. 9 chemical-metallurgical facility. Processing problems were immediately evident, leading to the decision to construct Facility 1. Facility 1 processing equipment was installed in a series of connected sealed gloveboxes. Each glovebox was designed for a certain operation and was made of stainless steel. Foundry equipment also consisted of a series of polished stainless-steel boxes connected with each other on the backside with a duct with an enclosed conveyer to transport objects between the boxes. The holes between the respective boxes and duct were closed with a cover and sealed with a rubber gasket. Lead-glass windows in the gloveboxes provided visual inspection of the interior. Work in gloveboxes was done using rubber gloves through portholes. Operations at Facility 1 to produce plutonium generally continued throughout the 1950s and 1960s without significant stoppage in spite of inadequate facility capabilities, and the ongoing efforts to reconstruct the glovebox assemblies and to replace equipment.

In January 1971, primary chemical-metallurgical production activities were transferred to the new Facility 1B, which also had the improved Three-Zone Design. The Three-Zone Design provided separation between the operator zone and the facility processing and maintenance zones. An "air-lock" design minimized the spread of contamination between zones that handled plutonium and the operator zone that did not handle plutonium. The primary technological equipment and service lines were arranged in the

Table 3.3. MPA Plutonium Chemical-Metallurgical Plant History

| Plant | Function | Stages of Development | Current Status |
|---|---|---|---|
| Facility 9 | Chemical purification of plutonium solutions from high-level impurities and generation of plutonium metal with high purification. | Operation first commissioned: February 1949<br>First batch of liquid plutonium concentrate received: 02/26-27/1949<br>August Transferred production to Facility #1: 1949–1950 | Building was decommissioned (buried) December 12,1982 |
| Facility 1 | | Operation first commissioned: August 1949<br>Compilation of first "Provisional technological instruction": May 1949<br>Equipped with gas purification system: 1962–1969<br>Transferred production to Facility 1B: 1971 | 1971 - Weapons-grade plutonium production stopped |
| Facility 1B | | Operation commissioned: 1971 | Weapons-grade plutonium production |

maintenance zone as well as the engineering support equipment, which was in contact with plutonium. Technological processes were controlled remotely from the operator zone. Radiation exposure of personnel occurred primarily during routine preventive maintenance and repairs, and, occasionally during nonroutine or "incident" situations. For operations in this facility, the Work Access Permission process was required administrative and technical review of work to be performed with the potential for significant dose.

# 4.0 MPA Archive Record Systems

The Mayak Worker Study[2] receives information from Mayak Production Association (MPA) and Southern Urals Biophysical Institute (SUBI) archive record systems as follows:

- MPA Personnel Department Record Archive
- MPA External Dosimetry Record Archive
- SUBI Internal Dosimetry Record Archive
- SUBI Medical Record Archive

These extensive archive record systems provide information for individual workers regarding employment, occupation, occupational history, radiological external dose, radiological internal dose, bioassays and medical examinations as described in the respective sections of this chapter. Records for a worker often extend through many decades.

## 4.1 PERSONNEL DEPARTMENT ARCHIVE

MPA personnel department record archive consists of several sections. The central section is located in the Enterprise Administrative Building. It contains information on staff work assignments at the respective Enterprise facilities throughout their working career beginning in 1948 through the present. The information is maintained in the form of cards and personal files. The information on the cards, illustrated in Fig. 4.1, includes identification (ID) of the worker, the date of the designation to a position and names of the structural subdivisions of the MPA. These cards include information on the transfer of a worker from one plant to another, for example, from the radiochemical plant to the plutonium plant.

The rest of the archive is distributed through the respective MPA plants. The plant cards, as illustrated in Fig. 4.2, record the worker transfer through the workplaces of this plant. In these cards structural subdivisions of the plant are noted including some administrative levels as well as the position to which the worker was assigned.

Information extracted from the cards illustrated in Figs. 4.1 and 4.2 was used to link the worker to the workplace, and based on knowledge of the workplaces to the radiation field characteristics. This type of archival information is used to develop the worker occupational history. Efforts were made to associate the worker occupational history with radiation parameters of the respective workplaces. Current computer records of the primary archive record information are presented in an electronic tablet format that presents a record identical to the informational content of the original paper archive documents. The electronic tablets are stored on the Mayak Worker Study database server.

[2] More detailed information presented in P.2.4.2003.09, "Unified Medical and Dosimetrical Database Containing Information on Personal Doses and Health Conditions of Mayak PA Workers."

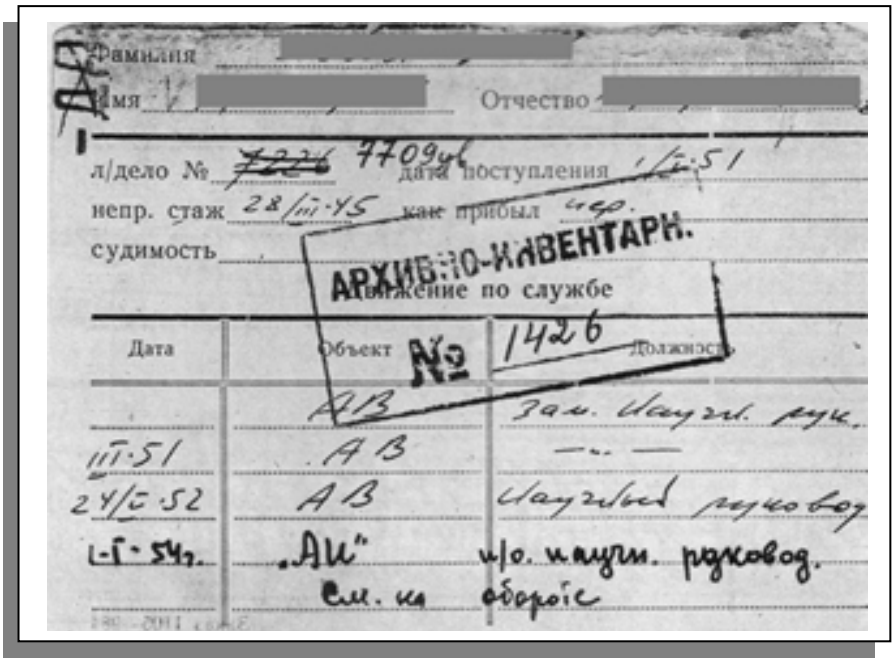



Fig. 4.1. Mayak Personnel Department Central Section Archive card showing site-wide plant assignment of workers

## 4.2 EXTERNAL RADIATION RADIOLOGICAL RECORD ARCHIVE

Radiation and dosimetric monitoring services have been conducted by Mayak Radiation Safety staff from the beginning of operations. Originally, these services were organized at each plant. The service was headed by an engineer or physicist. The service consisted as a rule of a group of workers performing dosimetric device repair, a group performing radiation monitoring and a group conducting individual worker dosimetric monitoring. The labor force of the service depended on the scope of the radiation monitoring to be carried out.

The External Radiation Radiological Record Archive is currently administered in the MPA Radiation Safety Service organization by the Individual Monitoring Group. This record archive contains three types of documents: personal booklets, annual dose logs and individual monitoring cards. The archive contains information for about 100,000 MPA workers, subcontractor institutions and military commands that worked at the MPA site. Two periods can be distinguished in the organization and maintenance history of the archive records as follows: 1948 - 1971, and 1971 till the present.

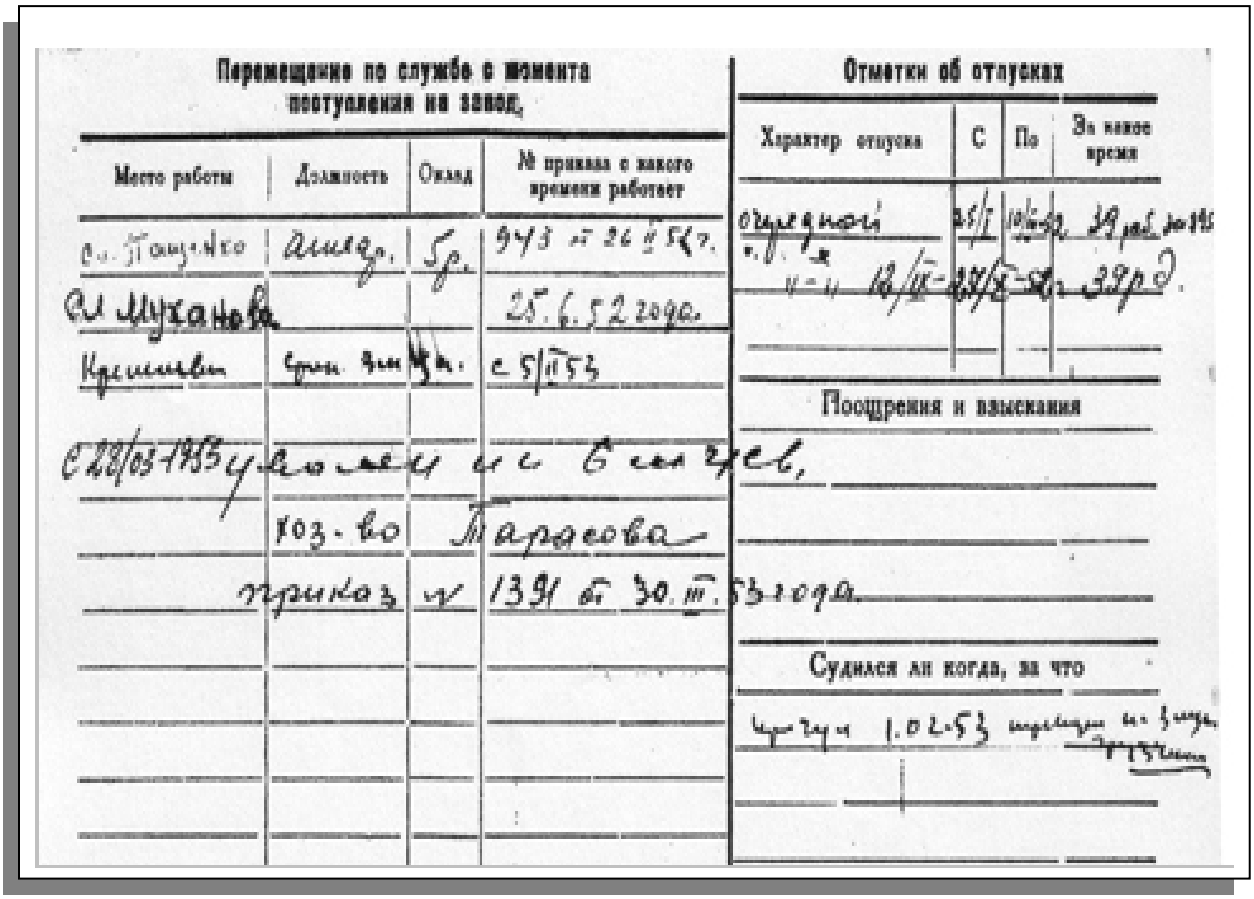



Fig. 4.2. Mayak Personnel Department Plant Archive card showing worker in plant job positions

### 4.2.1 Records 1948 - 1971

During the early period of Mayak operations from 1948 to 1953, workers were exposed to comparatively high doses. Each of the primary reactor, radiochemical, and plutonium plants had its own staff to conduct film dosimeter and ionization condenser dosimeter monitoring. This staff processed dosimeters and recorded doses daily. This arrangement provided capabilities to process dosimeters daily and to use the information to control radiation exposure to workers. Dosimeter processing results were recorded in a personal booklet for each individually monitored worker. These booklets, as shown in Fig. 4.3, contain dates of the individual dosimeter monitoring start and end dates, the number of work shifts during the monitoring period and the calculated dose. As doses to workers decreased with improvements in equipment, operational controls, etc., the film dosimeter monitoring period increased and achieved 1 mo by the end of the 1950s.

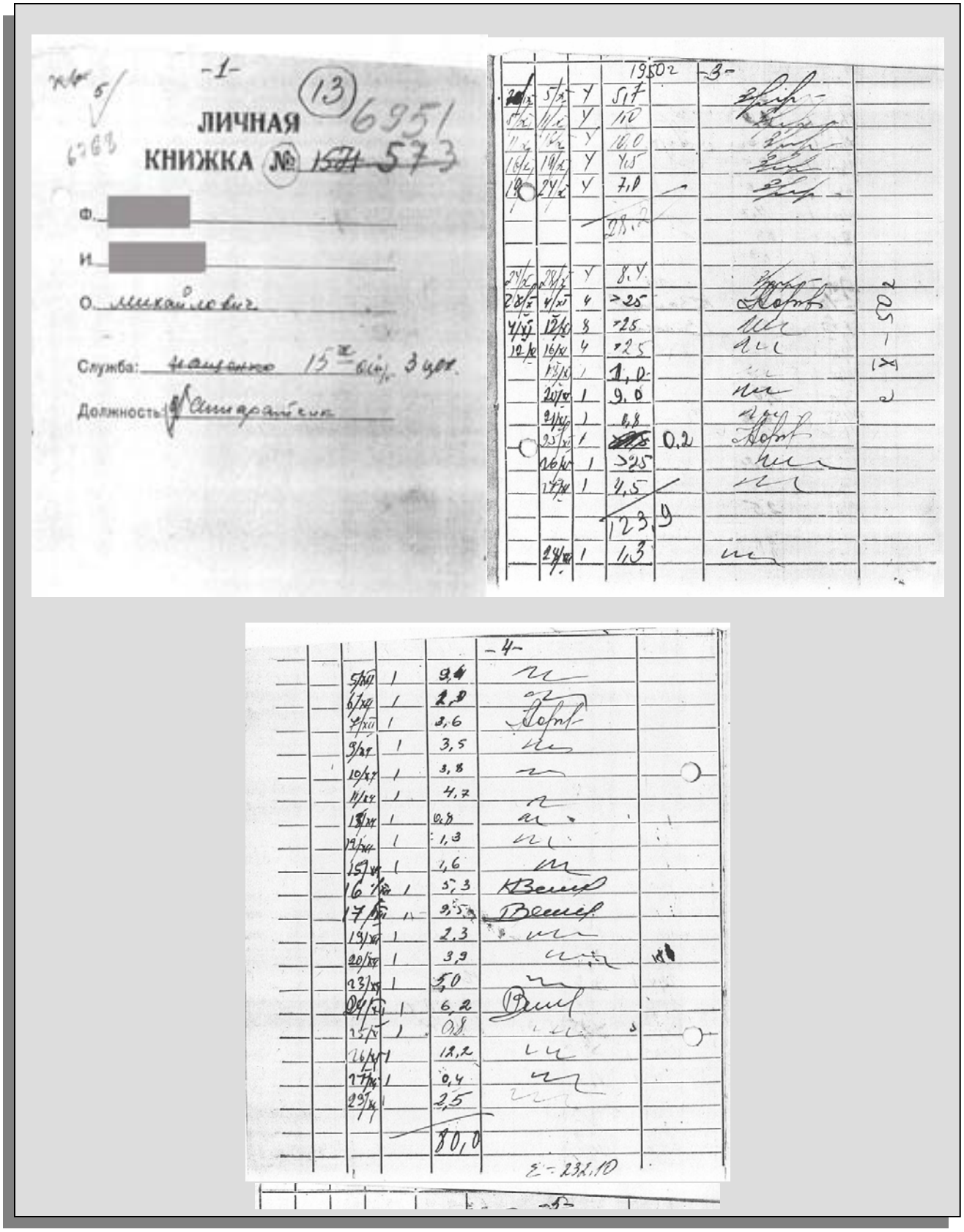



Fig. 4.3. Personal booklet with recorded film dosimetry results

### 4.2.2 Records 1971 - Present

A joint group of the individual dosimetric monitoring was established in MPA in 1971. This group collected all documents available in plants containing data on individual exposure doses to the personnel. At the same time a new form of the document for individual dose recording – the card of the individual monitoring – was introduced as shown in Fig. 4.4. This form provided records on annual exposure doses. A card was provided for each worker subject to individual monitoring in 1971. Data on annual doses exposed to the personnel earlier were entered in these cards from personal booklets and logs.

Nowadays, the archive is available in the group of the individual dosimetric monitoring of the MPA Radiation Safety Service. The archive stores three types of documents – personal booklets, logs, and individual monitoring cards. It contains information on about 100,000 workers of the MPA, contracting institutions, and specialized institutions that carried out activities at the territory of the MPA.

### 4.2.3 Characteristics of Recorded Doses

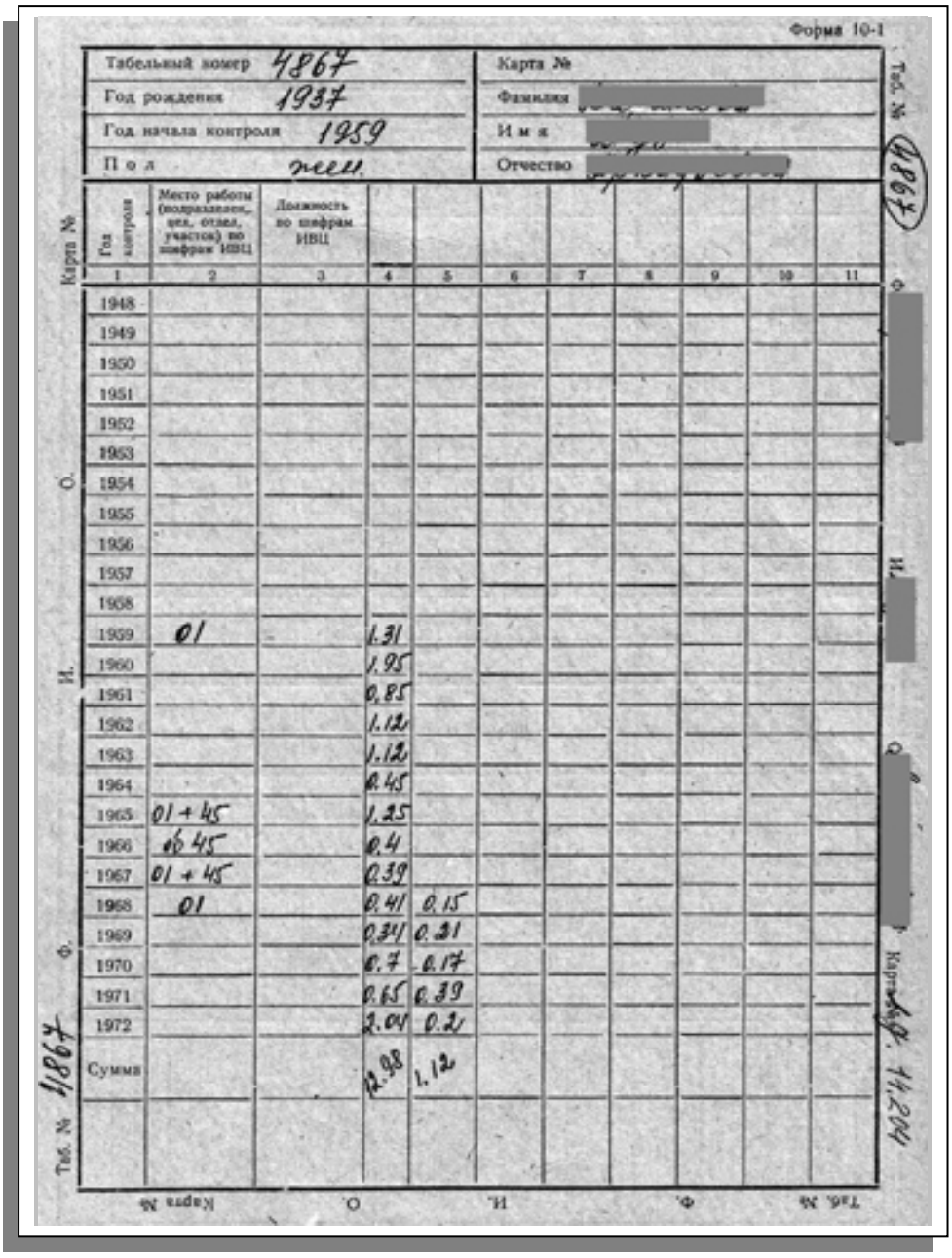
The database consists of tables containing the following information:

- Worker identification information;
- Directories of plants, facilities, locations and positions;
- Occupational histories;
- Tables of correspondence of the workplaces and production areas.;
- Recorded daily, monthly, annual and total doses;
- Correction factors for recorded dose calculation into verified dose with regard to dosimeter type and production area;
- Verified monthly, annual and total exposure doses;
- Uncertainties in verified monthly, annual and total doses;
- Calculated neutron doses (annual and total);
- Uncertainties in calculated doses.

#### 4.2.3.1 Occupational Histories

The information on worker occupational histories is contained in a separate database table. Each record in the table on occupational histories is structured in the following way:

- ID number;
- Date of hire;
- Date of employment termination;
- Plant code;
- Facility code;
- Area code; and
- Position code.



Форма 10-1

Табельный номер 4867
Год рождения 1937
Год начала контроля 1959
Пол жен.
Карта №
Фамилия [redacted]
Имя [redacted]
Отчество [redacted]

| Год контроля | Место работы (подразделение, цех, отдел, участок) по шифрам ИВЦ | Должность по шифрам ИВЦ | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 2 | 3 | | | | | | | | |
| 1948 | | | | | | | | | | |
| 1949 | | | | | | | | | | |
| 1950 | | | | | | | | | | |
| 1951 | | | | | | | | | | |
| 1952 | | | | | | | | | | |
| 1953 | | | | | | | | | | |
| 1954 | | | | | | | | | | |
| 1955 | | | | | | | | | | |
| 1956 | | | | | | | | | | |
| 1957 | | | | | | | | | | |
| 1958 | | | | | | | | | | |
| 1959 | 01 | | 1.31 | | | | | | | |
| 1960 | | | 1.95 | | | | | | | |
| 1961 | | | 0.85 | | | | | | | |
| 1962 | | | 1.12 | | | | | | | |
| 1963 | | | 1.12 | | | | | | | |
| 1964 | | | 0.45 | | | | | | | |
| 1965 | 01 + 45 | | 1.25 | | | | | | | |
| 1966 | 45 | | 0.4 | | | | | | | |
| 1967 | 01 + 45 | | 0.39 | | | | | | | |
| 1968 | 01 | | 0.41 | 0.15 | | | | | | |
| 1969 | | | 0.31 | 0.21 | | | | | | |
| 1970 | | | 0.7 | 0.17 | | | | | | |
| 1971 | | | 0.65 | 0.39 | | | | | | |
| 1972 | | | 2.04 | 0.2 | | | | | | |
| Сумма | | | 12.98 | 1.12 | | | | | | |

Карта № 1867 О. [redacted] И. [redacted] Ф. [redacted] Карта № 1867

**Fig. 4.4**. The form of the individual monitoring card after 1971

The combination of such information as plant, facility, area, positions determine the person's workplace.

#### 4.2.3.2 Mayak Primary Work Areas

The MPA has three main plants (reactor, radiochemical, and plutonium) as well as a radioisotope plant and several support (auxiliary) plants (water treatment, instrumental, mechanical). Each plant is characterized with its own structure and a set of workplaces. The total number of facilities, areas, and workplaces at the MPA is given in Table 4.1.

Table 4.1. Number of primary plant facilities, areas and workplaces

| Plant | Number of facilities | Number of areas | Number of workplaces |
|---|---|---|---|
| Reactor | 40 | 155 | 1,851 |
| Radiochemical | 76 | 243 | 2,543 |
| Plutonium | 63 | 212 | 3,099 |
| Other | 22 | 5 | 1,389 |
| Total | 201 | 615 | 8,882 |

It is clear from Table 4.1 that each plant has several thousand workplaces. With such a large number of workplaces, it is difficult to describe the radiation environment at each. Therefore, several radiation groups were singled out for each plant. A radiation group is a group of workplaces characterized with similar personnel exposure conditions. All workplaces where personnel were exposed were associated with certain radiation groups.

The distribution of the number of records in the "Occupational Histories" section of the database that shows the movement of the personnel in different MPA plants, is shown in Table 4.2 and Fig. 4.5

Table 4.2. Number of records in the database section "Occupational Histories"

| | Male | | Female | | Total | |
|---|---|---|---|---|---|---|
| Plant | No. | Percent | No. | Percent | No. | Percent |
| Reactor | 3,445 | 24.5% | 969 | 20.4% | 4,414 | 23.4% |
| Radiochemical | 5,874 | 41.7% | 2,022 | 42.5% | 7,896 | 41.9% |
| Plutonium | 4,753 | 33.8% | 1,768 | 37.2% | 6,521 | 34.6% |
| Total | 14,072 | | 4,759 | | 18,831 | |

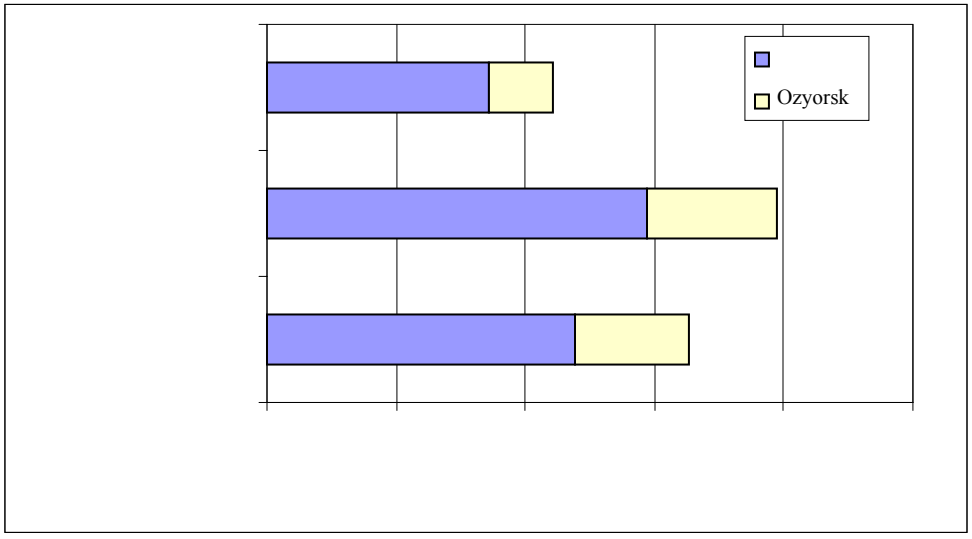



Fig. 4.5. Number of occupational history records

Personnel age distribution based on the age of hire is shown in Table 4.3 and in Fig. 4.6. More detailed data on worker age at the moment of hire is provided in internal project Technical Report P.2.4.2004.23, Tables 13–16 of the attachment. When a worker is moved from one plant into another is considered to be

the date of hire to the second plant. Therefore, the data from Table 4.3 are knowingly less than the total of corresponding data from Technical Report P.2.4.2004.23, Tables 13–16 of the attachment.

Table 4.3. Distribution of worker age and gender

| Age*, years | Male | Percent | Female | Percent | Total | Percent |
|---|---|---|---|---|---|---|
| Up to 20 | 5,117 | 36.36 | 919 | 19.31 | 6,036 | 32.05 |
| From 20 to 29 | 6,791 | 48.26 | 2,684 | 56.40 | 9,475 | 50.32 |
| From 30 to 39 | 1,481 | 10.52 | 805 | 16.92 | 2,286 | 12.14 |
| From 40 to 49 | 592 | 4.21 | 318 | 6.68 | 910 | 4.83 |
| 50 and more | 91 | 0.65 | 33 | 0.69 | 124 | 0.66 |
| Total | 14,072 | | 4,759 | | 18,831 | |

*Age on first hiring at MPA

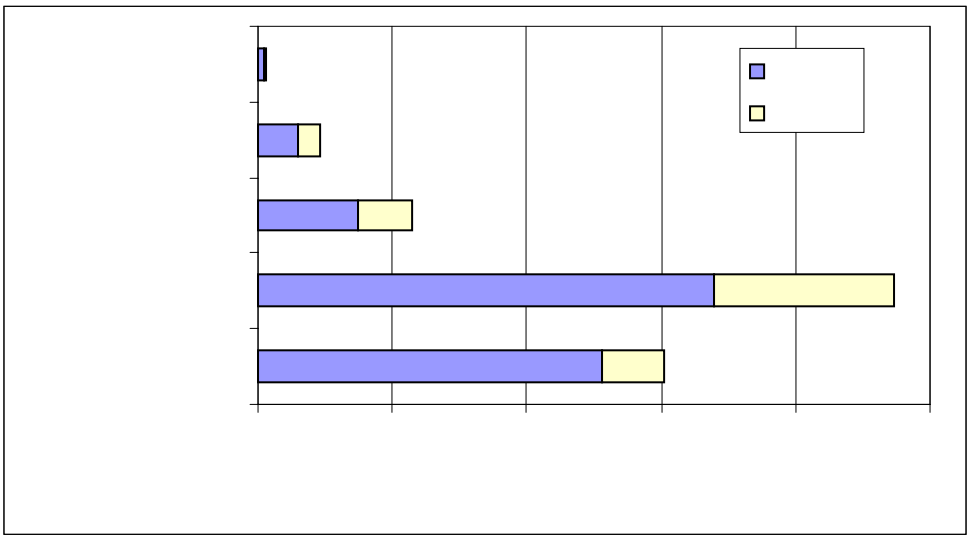


Fig. 4.6. Worker age distribution upon hiring at Mayak

From the data it is clear that 80% of workers at the moment of hire were younger than 30. The data on the total duration of employment of personnel at MPA are listed in Table 4.4 and shown in Fig. 4.7.

Table 4.4. Duration of employment with Mayak PA

| Plant | Total duration of personnel employment, thousand person-years | Percent |
|---|---|---|
| Reactor | 103.1 | 33.22 |
| Radiochemical | 116.8 | 37.63 |
| Plutonium | 90.5 | 29.16 |
| Total | 310.4 | |

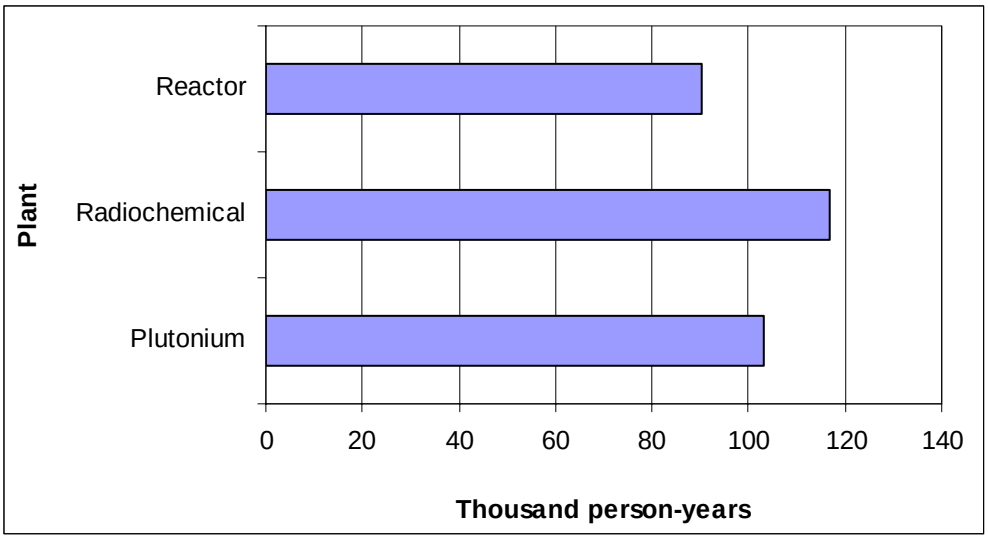


Fig. 4.7. Duration of worker employment at different Mayak plants

**Recorded doses**

Since 1948, the physical values measured by Mayak personal dosimetry have gone through several changes, and the units used for these measurements have changed as well. Table 4.5 lists the physical quantities and units used historically by MPA personnel monitoring.

Table 4.5. Radiation units used historically to record dose

| Years | Physical value used for record | Measurement unit |
|---|---|---|
| 1948 – 1973 | Exposure dose* | roentgen |
| 1974 – 1992 | Absorbed dose in tissue | rad |
| 1992 – 2000 | Equivalent dose in soft biological tissue at 1 g/cm2. Hp(10) | cSv |
| Since 2000 | Dose equivalent in tissue at 1 cm Hp(10) | mSv |

* Terminology used by Mayak dosimetry staff, equivalent to exposure.

The database of radiation safety department "ORB" contains three types of the data on workers exposure doses. These doses were recorded, calculated, and verified.

Recorded doses were determined based on film badge readings. These doses are the foundation of this database. These doses are entered in the database in compliance with doses recorded in the archives.

Calculated doses – worker doses from neutron exposure as well as photon doses for unmonitored individuals. These doses were obtained based on source term conditions at workplaces as a result of application of dose reconstruction methods based on certain conceptual assumptions.

**Daily doses**

The interval of dosimeter exchange was discussed in Section 4.2.1. The recorded dose for a single dosimeter reading is referred to as the "daily dose," even though the exchange interval might have been as long as a month. These daily doses were routinely recorded in individual dosimetry books and are the foundation of Mayak dosimetry archives. Each worker's annual dose was a sum of the recorded daily doses.

Internal project reports described the MPA individual dosimetry system in detail. For individual dosimetry purposes films with high and low sensitivity were used. The type of the film depended on the expected worker dose. Films with low sensitivity could be used for measurement of the dose lower than 25 R.

Selected parameters of the daily doses are listed in Tables 4.6 though 4.8.

Table 4.6. Number of records and workers for selected daily dose parameters

| Parameters | Number of workers | Number of records in database |
|---|---|---|
| Daily doses | 8,764 | 725,350 |
| Daily doses <5 R | 8,741 | 703,408 |
| Daily doses 5-10 R | 3,155 | 14,565 |
| Daily doses 10-25 R | 1,847 | 5,729 |
| Daily dose ≥25 | 790 | 1,648 |

Table 4.7. Distribution of daily doses recorded as ">25" by years

| Year | Number of records | Number of persons |
|---|---|---|
| 1950 | 92 | 132 |
| 1951 | 75 | 80 |

Table 4.8. Daily dose distribution by year.

| Year | Number of records by plants | | |
|---|---|---|---|
| | Reactor | Radiochemical | Plutonium |
| 1949 | 4 | 39 | |
| 1950 | 15,914 | 44,194 | |
| 1951 | 26,469 | 28,107 | |
| 1952 | 35,400 | 25,343 | |
| 1953 | 34,655 | 31,968 | |
| 1954 | 36,488 | 38,293 | |
| 1955 | 23,874 | 30,265 | |
| 1956 | 11,447 | 16,537 | |
| 1957 | 6,405 | 15,922 | |
| 1958 | 6,081 | 16,580 | |
| 1959 | 5,645 | 27,591 | |
| 1960 | 5,317 | 24,896 | |
| 1961 | 2,957 | 24,875 | |
| 1962 | 13 | 22,762 | |
| 1963 | 2 | 19,656 | |
| 1964 | 1 | 8,334 | |
| 1965 | 1 | 7,849 | |
| 1966 | | 7,179 | |
| 1967 | | 11,071 | |
| 1968 | | 10,232 | |
| 1969 | 9 | 9,909 | |

Daily doses recorded as ">25" occurred only in 1950 and 1951. The occurrence of such records in the general volume of daily doses is not high. The number of individuals with such records is limited.

**Annual doses**

Annual doses were obtained from the sum of daily doses. The annual dose is the sum of all recorded daily doses at each workplace, if the person worked at several workplaces during the year.

Selected parameters of annual doses are listed in Table 4.9.

Table 4.9. Data distribution on annual doses at MPA.

| Plant | Number of records | Percent | Number of workers with recorded annual doses | Percent |
|---|---|---|---|---|
| Reactor | 63,941 | 25.53% | 4,415 | 23.45% |
| Radiochemical | 116,829 | 46.66% | 7,895 | 41.93% |
| Plutonium | 69,637 | 27.81% | 6,521 | 34.63% |
| Total* | 250,443 | | 18,831 | |

* Total shows the data that were recorded directly from the database and are not equal to the sum of the given columns.

**Total individual and collective recorded doses**

Recorded doses are available for 15,815 (84%) MPA workers of 18,831 workers in the original cohort. The collective dose in person-sievert is obtained by summation of all worker doses. The recorded dose quantities and units changed during years of operation of MPA, as listed in Table 4.5. Collective dose distribution and distribution of the workers in Mayak plant are listed in Table 4.10. This table also lists the sum of recorded doses.

Table 4.10. Distribution of workers and collective dose in MPA.

| Plant | Collective dose, person-Sv | Percent | Number of workers with recorded doses | Percent |
|---|---|---|---|---|
| Reactor | 2,904 | 26.65 | 4,415 | 23.45 |
| Radiochemical | 7,263 | 66.66 | 7,895 | 41.93 |
| Plutonium | 728 | 6.68 | 6,521 | 34.63 |
| Total* | 10,895 | | 18,831 | |

* Total shows the data that were recorded directly from the database and are not equal to the sum of the given columns.

From Table 4.10, it is clear that the highest collective dose is recorded for the radiochemical plant. The majority of workers worked at this plant. The collective dose received at the radiochemical plant constitutes nearly 70% of the collective dose of MPA workers.

The results of the source term measurements at the production sections were recorded in working logs as shown in Fig. 4.8. These records contain the date, time and place of measurements as well as the measured results. Measurements were typically gamma dose rate, neutron dose rate, beta flux density and alpha flux density. The total concentration of beta and alpha-emitting nuclides in air of the workplaces was typically monitored using gross radioactivity measurements of sampled air filters. Data on the concentration of alpha and beta radiation measured air concentrations were also recorded in the work logs. This information is available in the archives of the radiation safety departments of MPA plants.

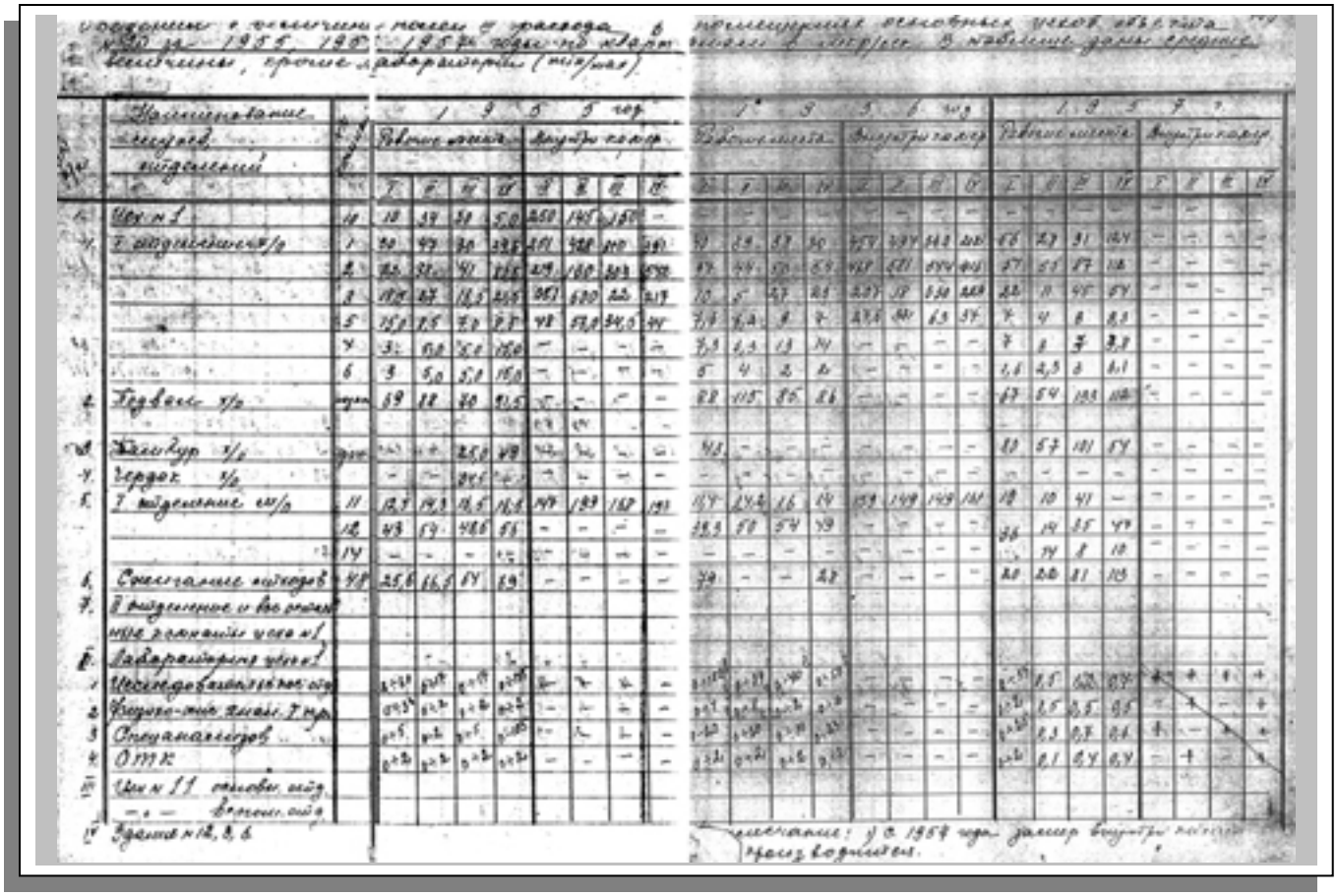


Fig. 4.8. Example of source term measurement log

**Workplace worker groups**

A computer file, "Radiation groups," was established to aid in dose reconstruction based on the list of work sections in MPA production plants with consideration of changes in operations with time. Each plant was subdivided into facilities, which were divided into production areas, which in turn were divided into workplaces. The workplace is a location where the personnel stay constantly or provisionally to carry out their production functions. The criteria for identifying a single workplace included the equipment type or the character of the operations carried out, the radiation source, and places most likely to be occupied by personnel. At the workplaces or near them there are process equipment and service lines used in technical processes. The source term at the workplace is described by radioactive materials in and on the surface of the equipment and service lines. The personnel perform their duties at the workplace subject to radiation exposure. A worker's dose is determined by the source term, nature of the work performed, duration at the workplace and other factors. Workers of different professions could perform their work at different workplaces in the course of production operations.

Several thousand workplaces have been identified at MPA. Specific source terms corresponding to each workplace were identified. The workplaces with similar radiation characteristics were assigned to selected radiation groups and an ID number was assigned. In the reactor, radiochemical, and plutonium plants, two or three groups were distinguished for the most typical conditions of personnel exposure. This analysis resulted in assigning all workplaces to one of eight groups as listed in Table 4.11.

Table 4.11. Mayak worker group assignment

| Group ID | Group description |
|---|---|
| 1 | Radiochemical production: head-end and intermediate nodes |
| 2 | Radiochemical production: foot-end nodes |
| 3 | Plutonium production: radiochemical nodes |
| 4 | Plutonium production: chemical and metallurgical nodes |
| 5 | Reactor production: Central Hall |
| 6 | Reactor production: other, excluding Central Hall |
| 7 | Reactor production: principal exposure at other sections excluding Central Hall and periodical (~25-50%) exposure in Central Hall |
| 8 | Without exposure due to principal production (spectra of the contaminated territories after 1957 accident), increased background |

**Database of "Daily Doses"**

An additional database was completed during September 2004 that contains 725,350 records of daily doses obtained from Worker Personal Logs in the MPA archives for approximately 8,748 workers. The logs contain the dose result for each processed dosimeter, including the date, the number of shifts the dosimeter was worn, and the signature of the dosimetry technician. Analysis of this information has been done to examine the dynamics of the measured dose with the expectation that higher doses for short periods, such as one shift, could be associated with nonroutine exposures.

Personnel exposure scenarios and dose dynamics in nonroutine situations have some unique characteristics. For example, two common nonroutine exposure scenarios involved (1) approximately 50–500 workers in the early years of reactor operation to resolve problems caused by defects of fuel channels and fuel elements, and (2) approximately 30–200 workers to resolve problems associated with coolant spills from equipment seal failures and other radioactive spills. The nonroutine exposure scenarios were analyzed to obtain a more accurate definition of exposure spectra and geometries. The analysis also calculated dose per shift, based on the "daily doses" and the number of shifts that the dosimeter was worn during the exchange period. Table 4.12 lists the number of "daily dose" records in each dose interval.

Table 4.12. Daily dose records

| Dose interval per shift (R) | Number of records |
|---|---|
| <0.1 | 160,441 |
| 0.1-0.3 | 188,174 |
| 0.3-1.0 | 186,840 |
| 1-10 | 182,518 |
| ≥10 | 7,377 |

Dose dynamics for each worker were analyzed throughout the period of employment based on their position and location categories, and categorizing their doses according to the intervals listed in Table 4.12. The analysis through time showed those days with higher exposures. The data were also analyzed according to position/area categories. The results of these analyses provide the number of workers with doses in a particular dose interval, the position/area category, and the date of exposure (i.e., date of dosimeter processing). Of course, the analysis was also done for all dose results for individual workers. The proportion of the annual collective dose from nonroutine exposures was compared to the annual collective dose from nonroutine and routine exposures. This analysis showed that

the percentage of nonroutine exposure significantly declined with time after the earliest years of MPA operation.

**Bioassay record archive**

Direct measurements of plutonium content in the body were derived from autopsy measurements and/or from urine bioassay measurements. Detailed documentation on these programs and methods can be found in the references and literature cited in Volume III of this guide. The original paper records for the bioassay program are kept at the biophysical laboratory at SUBI. These are in the form of log books that contain information on the UAs with data sequentially entered and journals of collated UAs for specific individuals. Individual workers might have had many bioassays during their employment at the MPA. These records include the volume of urine collected, aliquot processed, volume of solution added for precipitation, photomultiplier tube (PMT) identifier, background counts, and the efficiency of counting. From these original data, the UA results were transferred to notebooks that were cataloged by individual worker. These notebooks contain the same information as the log books. An averaged plutonium activity per day was entered into the urinalysis electronic database.

The following example is a transcription and translation from a typical notebook (*Biophysics Journal* 637, page 66, ID: 63766). Privacy information has been deleted and a "unique identifier" assigned.

- Date of collection: 20/11/75
- Amount of urine collected in 24 hr: 1300 mL
- Volume of sample used in assay: 200 mL
- Volume of solution added: 20 mL
- Aliquot used for counting: 10 mL
- Background: 3
- Counting efficiency: 13
- PMT Identifier: 6.26
- Sample counts: 40.7
- Calculated dpm/g: 48
- Final dpm/g: 48

Additional information that is found in the original notebooks and journals includes at the following:

- Name (unique identifier)
- Date of birth
- Beginning of employment with Mayak Production Association
- Sampling date
- Vital status and date of death, if deceased.
- Weight at time of measurement
- If and how diethylenetriaminepentaacetic acid (DTPA) was used
- Plant and subplant codes
- Time between sampling and end of last work period

## 4.3 AUTOPSY RECORDS ARCHIVE

About 1,200 total autopsies have been conducted to date on Mayak workers. Published information on the cases can be found in the references in Volume III of this guide. These records are maintained at the SUBI. The autopsy records contain the gross anatomic observations, histopathological findings, and clinical summary by the pathologist including primary and contributing causes of death. The associated records might also contain the organ concentrations of plutonium (total alpha activity) determined in the post-mortem tissues. Some of the tissues were saved from many of these autopsies for the Tissue Repository and Archive at SUBI. The algorithms used to convert organ plutonium concentrations into dose for various organs are in Volume III.

Total alpha-particle activity was measured in the following organs from some autopsy cases:

| | | | | |
|---|---|---|---|---|
| Lung | Kidneys | Gall bladder | Small intestine | Bone marrow |
| Liver | Heart | Esophagus | Colon | Lymph nodes |
| Skeleton, | Thyroid gland | Stomach | Mammary gland | |
| Skeletal muscle, | Testes | Adrenal glands | Skin | |
| Spleen | Pancreas | Bladder | Blood | |

## 4.4 MEDICAL RECORD ARCHIVE

The medical record archive, which is the property of the Central Medico-Sanitary Department No 71 (CMSD-71), was given to SUBI in the mid-1990s for preservation. It contains about 60,000 outpatient medical records of former MPA workers who left the MPA for various reasons during the period from 1948 to 2004. The CMSD-71 employees continue using the medical records from the archive, mainly for expert judgment in determining cause-effect relationships between health disorders and occupational radiation exposure.

Medical records contain a wide range of information that can be used in worker health research studies. For example, the medical records provide an important source of information to verify the date and place of birth of a Mayak worker, and to abstract information on previous jobs (before employment at Mayak), including information on occupational hazards in these jobs. In addition, medical records contain data on smoking and alcohol consumption, on diseases that the worker had before employment at Mayak and during the entire period of employment, and even diseases for close relatives' diseases.

Quarterly information on workplace and occupation is of particular relevance to the Mayak Worker Study along with the physician's decision about the possibility to continue working in the workplace. If some health disorders were detected, including hematological disorders, a special medical examination was conducted, and on the basis of the results of this special examination, a decision was made on whether the worker could continue working or the worker's removal. There is information on all instrumental medical examinations (such as x-ray examination and endoscopy) and laboratory tests, and on all hospitalizations in the specialized hospital and other medical institutions.

In addition, there are results of biophysical examination of excreta for alpha- and beta-activity. Unfortunately, these data cannot be used for estimation of radionuclide content in the body, because the examinations were conducted during work, and the activity was considerably overestimated due to contamination in transit. However, this information indicates that the worker was in contact with the radionuclide and can be used for development of a surrogate measure of exposure.

Thus, the medical archive contains valuable information, which is useful for verification of occupational histories, occupational and nonoccupational morbidity, and causes of death.

# 5.0 External Dosimetry Organ Dose Methodology

## 5.1 INTRODUCTION

Detailed information concerning the methods used to calculate organ doses from external radiation are provided in Volume II – "Dose Assignment Methodology used to Calculate Annual Organ Doses to Mayak Workers from External Radiation." The early history of MPA nuclear material production is expected to involve classified information and, therefore, a clear description of all events at that time is not publicly available.

## 5.2 DOSIMETRY PROTOCOL

The Mayak Worker Dosimetry Study involves many complex issues in the examination of archive records (see Chapter 4), recorded dose to workers, and the methods to reconstruct dose for external radiation. As noted in Section 1.4, a dosimetry protocol was collaboratively developed by Project 2.4 and Project 2.2 researchers. The protocol defines the original study cohort of 18,831 workers employed before 1972 to be the highest priority for dose reconstruction. These are the workers who are included in the Doses-2005 database. There is interest in extending this cohort in future years. The organs for which the absorbed dose is needed were also specified along with using ICRP (1996) as the source of dose conversion factors (DCFs) for standard geometries. The highest priority stated by Project 2.2 researchers in the protocol concerned examination of the early, high dose records. A database (i.e., referred to as "Daily") was developed that contains 725,350 records of individual dosimeter dose results during the earliest period of Mayak operations. These records were obtained from Worker Personal Logs for approximately 8,748 workers in the Mayak archives. Detailed statistical analyses of these records examined associations of the archive dose with facility, work history, and time. Results of these analyses provide one method to distinguish between worker exposure during routine day-to-day activities and to nonroutine activities typically involving equipment or process failures and higher exposures. The accuracy of records between archive paper records and the Doses-2005 database was examined using a random selection by Project 2.2 researchers of 100 workers whose primary work activities were associated with the reactor, radiochemical, plutonium-chemical-metallurgical and auxiliary areas (i.e., 100 workers for each facility). Results of these evaluations are described in later sections of this chapter.

## 5.3 MPA WORKER MONITORING

Personnel monitoring of Mayak workers was conducted by a dedicated group of radiation protection specialists in the early years according to radiation protection instructions in the Mayak operational guidance "Instructions on individual monitoring of gamma-harm." According to these instructions, workers could access a work zone that had significant radiation levels only if they were assigned an individual dosimetry package. The history of the evolution in limits of worker exposure in these guides is shown in Table 5.1. In 1948, the primary limits of external whole-body exposure were 0.1 R during a 6-hr shift and 30 R during a year. A worker could also receive an "acute" maximum dose of 25 R in a minimum period of 15 min without affecting routine dose limits. This is similar to international guidelines for worker exposure.

Table 5.1. Radiation Protection Guides

| Year | Radiation Protection Guide Description | Mayak Operational Guidance |
|---|---|---|
| 1948 | 30 R/yr, 25R/15 min (accidental), 0.1 R/d | Sanitary Rules and Regulations for Health Protection for Reactor and Radiochemical Plant Workers; No. T-1031; August 24, 1948 |
| 1950 | 30 R/yr, 25R/15 min (accidental), | Temporary General Sanitary Rules and Regulations for Health Protection for People working with radioactive materials; No. 2413; February 10, 1950 |
| 1954 | 15 R/yr (routine), 25R/yr (accidental), 0.05 R/d | Sanitary Rules and Regulations for design of enterprises and laboratories; No. 851; April 11, 1954 |
| 1960 | 5 rem/yr, 100 mR/week, 2,8 mR/hr | Sanitary Regulations No. 333-60; June 25, 1960 |
| 1969 | 5 rem/yr | Radiation safety standard (RSS)-1969; No. 821A-69; August 25, 1969 |
| 1976 | 5 rem/yr | RSS-1976; No. 141-76; June 7, 1976 |
| 1996 | 20 mSv/yr average for 5 sequential years but not to exceed 50 mSv/yr in any 1 year | RSS-1996; No. 2.6.1.054-96; April 19, 1996 |

Major advances in worker safety were achieved over the years and notable events are summarized in Table 5.2.

Table 5.2. Organization of work and protection means

| Period | Description |
|---|---|
| 1949-1953 | Development and realization of Work Access Dosimetric Permission System |
| 1956 | Initial implementation of respiratory protection for individual workers. |
| 1958 | Widespread implementation of respiratory protection. |

The formal “Work Access Permission System” or “Work Access Dosimetric Permission System” began in 1949 and was fully implemented by 1953. This system required examination of unplanned (i.e., incidents) or planned work as reasonable before actually performing the work through a several steps as follows:

- Specification of the main goal of the work to be performed;
- Assessment of the radiation situation;
- Pre-job planning such as identification of the workplace; preparation of a preliminary work scope, designation of the order of planned work activities, identification of individual protective equipment and radiation monitoring support for each and all work stages; and options to improve personnel and production safety; and activities to be done after completion of the work
- Administrative execution of the permission letter to perform the work;
- Admissible dose;

- Admissible work time;
- Final preparations for performing the work including a review of the radiation situation at the work performance site;
- Physical admittance of personnel to perform the work;
- Arrangements for radiation monitoring support during the work and criteria that could cause the work to be suspended;
- Arrangements for dosimeter assignments to personnel who will participate in the work;
- Completion of the planned work and activities to be accomplished to return the facility to normal work activity;
- Analysis to be done to assess the quality of work performance; and
- As necessary, incident analysis, identification of causes and assessment of the condition of the equipment.

The "Work Access Permission" letter for nonroutine work involving an elevated radiation hazard contained the following essential sections:

1. Work performance date and time
2. Unequivocal identification of the workplace
3. Specific description of the content of the work to be done
4. List of protective equipment for individual workers
5. Permitted work duration
6. Permitted dose
7. List of administrative and technical measures to assure safe work performance
8. Name of responsible work performance administrator
9. Names of workers

The "Work Access Permission" letter approval was valid within a limited time period. After this period, the approval became invalid and it was necessary to repeat the review process to obtain further approval to perform the work. The "Permission" system prevented unauthorized modifications in process technology that could expose workers; in addition, personnel exposure was controlled according to Sections 4, 5, and 6 listed above. Judicious selection of individual worker protective equipment reduced the incidence of worker intakes (i.e., inhalation, ingestion, or subliminal transport through the skin) of radioactive substances. Monitoring of workplace conditions after the planned work was completed facilitated assessment of overall performance to limit dose as planned and to the ongoing maintenance of satisfactory radiation exposure and dosimetry in the workplace.

Work Access Permissions were divided in two categories according to the priority of the performed work and to the level of potential radiation exposure of workers. The first category, "Permission No. 1", of increased hazard required obligatory presence of radiation safety service personnel while the work was performed. The second category, "Permission No. 2," required the radiation safety service staff to "open"

and to "close" the permitted work activities (i.e., Permission letter). This required assessing the radiation situation before the work was done and after the work was completed.

Work Access Permissions were approved by plant administration. The level of administration needed for approval depended on the level of the potential radiation dose. Personnel conducting work under a Work Access Permission approval were acquainted with the "approval" conditions before the work began. Operational technology personnel were also acquainted with the "approval" conditions. Coordination of work activities by operational and maintenance personnel reduced the potential for unexpected or unplanned activities that could increase hazards to workers. The Work Access Permission approval conditions were mandatory and activities that could be viewed as not fulfilling these requirements for radiation hazardous work could, some cases, result in administrative expulsion and possibly even criminal responsibility for engineers and administrators.

During 1948–1952, individual monitoring groups performed individual dosimeter processing in shifts (four shifts per day). Later due to the decrease in individual doses, the monitoring period was increased and by 1960 practically all "Mayak" PA sites had a monthly monitoring schedule. During this period, the individual monitoring was subdivided into routine and special operation monitoring. The special operation monitoring was performed with the addition of electronic ionization dosimeters by the radiation protection group, which monitored personnel doses during more hazardous radiation work (i.e., step-by-step support) and prepared monthly dose information. After the film badges were processed, the routine and special operation measured dose results were compared with each other. If there was no discrepancy, the routine monitoring results were recorded in the logs. The IFK (the first Mayak personnel film dosimeter) films were destroyed after use. At some MPA sites there were "Individual monitoring record-books" in addition to the logs that contained weekly and monthly doses of the record-book holder. In 1971 the individual monitoring groups of different sites were united into the centralized individual monitoring group. Since that time, the central group has been providing personnel dosimetry support to MPA personnel as well as those of other agencies or military units working or doing military service on the MPA territory or impact zone.

## 5.4 DOSES-2005 EXTERNAL DOSIMETRY DOSE QUANTITIES

Reconstructed dose quantities provided in the Doses-2005 analysis file for each worker and year of employment are listed in Table 5.3. The dose quantities allow consistent conversion from the archive dose (recorded in different quantities historically) to absorbed dose in air without a phantom based on the radiation field specifications for each exposure scenario, to the personal dose equivalent that considers the worker orientation for each exposure scenario and then to the respective absorbed organ dose. These quantities were selected to provide linkage among the measured archive doses and the quantities used in dose reconstruction and organ dose calculations. The radiation quantities in which the Mayak dose records were recorded changed over time. Consequently, archival dosimeter responses were converted to absorbed dose in air. Conversion factors were used to calculate the absorbed dose in air (without a phantom) for the primary facility work areas. These factors incorporate considerations of the exposure scenario energy and geometry specifications.

Table 5.3. Reconstructed dose quantities

| Dose Quantity | Description |
|---|---|
| $D_{\gamma\ arch}$ | Annual measured dose for a worker from Mayak record archives |
| $D_{\gamma\ dos}$ | Absorbed dose in air in Gy (conversion from historical units) |
| $D_{\gamma\ rec}$ | Absorbed dose in air with consideration of radiation field energy spectra and directional considerations (i.e., based on scenario specifications) , |
| $H_p(10)_\gamma$ | Photon radiation personal dose equivalent at a tissue depth of 10 mm on the trunk of the body with consideration of worker orientation (i.e., photon scenario specifications). |
| $H_p(10)_n$ | Neutron radiation personal dose equivalent at a tissue depth of 10 mm (i.e., based on neutron scenario specifications, photon dose and using neutron dose equivalent-to-photon dose ratio). |
| $D_{\gamma\ org\text{-}i}$ | Absorbed dose for the respective organs |

**Dose reconstruction**

The process of dose reconstruction involves consideration of each of the significant parameters (radiation field energy and directional characteristics, dosimeter response, and worker orientation) involved in the calculation of absorbed organ dose. Eighteen organs were selected by project epidemiological researchers as being of interest in assessing health effects, as listed in Table 5.4. The analytical steps leading from archive dose to absorbed organ dose are summarized in Table 5.5. In Doses-2005, the respective radiation quantities for each worker and year of employment were calculated using tables of conversion factors. For example, the organ DCF is widely used and is used in the Mayak Study to multiplying the absorbed dose in air to determine the absorbed organ dose. Absorbed organ DCFs were calculated for each exposure scenario, using the scenario exposure geometry and photon energy spectrum, and the worker's workplace orientation. To describe the exposure geometry, a standard irradiation orientation such as anterior-posterior (AP), rotational (ROT), or isotropic (ISO), was specified. If a standard orientation will adequately describe the scenario's geometry, then the conversion factors can be derived from standard tabulations of organ dose factors such as the International Commission on Radiation Protection (ICRP) Publication 74 (ICRP 1997) and GSF publication (Zankl et al. 1997).

Table 5.4. List of organs to calculate external radiation absorbed dose

| Organ[a] | Gender | Organ[a] | Gender |
|---|---|---|---|
| Brain[b] | Both | Lung | Both |
| Stomach | Both | Endosteal tissue (bone surface) | Both |
| Lower large intestine[b] | Both | Uterus | Female |
| Thyroid | Both | Bladder (urinary) | Both |
| Red marrow | Both | Liver | Both |
| Ovaries | Female | Colon | Both |
| Breast | Female | Esophagus[c] | Both |
| Kidney[b] | Both | Skin | Both |
| Small intestine[b] | Both | Testes | Male |

a. Gall bladder was identified as an organ of interest by Mayak Study epidemiologists, but no data exist for this organ in either the GSF or the ICRP (1997) compilations.

b. Since the following organs are not included in the ICRP (1997) compilation, the data are extracted from the GSF compilation: brain, kidney, small intestine, lower large intestine

c. Esophagus: dose factor data are derived from ICRP (1997), but the organ does not exist in the phantom model used in MCNP calculations for nonstandard irradiation geometries.

Table 5.5. Dose reconstruction analyses

| Activity | Analysis | Equation |
|---|---|---|
| Determine worker's archive annual dose | Worker archival annual dose, $D_{\gamma\ arch}$, is sum of worker's dosimeter results for year or reconstructed annual dose for worker who was not monitored. $D_{\gamma\ arch}$ is expressed in radiation units used during monitoring period (R, rad, rem, mSv). | $D_{\gamma\ arch}$ |
| Splitting of archive annual dose into routine and nonroutine doses. | Nonroutine daily doses are singled out from daily dose database and all nonroutine daily doses are summed to obtain annual nonroutine archival dose, $D_{\gamma\ arch\text{-}nrout}$. This is subtracted from archival annual dose to obtain annual routine archival dose, $D_{\gamma\ arch\text{-}rout}$. | $D_{\gamma\ arch} = D_{\gamma\ arch\text{-}rout} + D_{\gamma\ arch\text{-}nrout}$ |
| Adjust nonroutine dose for beta response | Using correction factors for effect of high-energy beta radiation on dosimeter readings in nonroutine scenarios, $D_{\gamma\ arch\ nrout}$ is corrected by value of $C_\beta$ appropriate for work location. | $D'_{\gamma\ nrout} = \frac{D\gamma nrout}{C_\beta}$ |
| Conversion of historical radiation quantities to absorbed dose. | $D_{\gamma\ arch}$ (the sum of $D_{\gamma\ arch\ rout}$ and $D'_{\gamma\ arch\ nrout}$) is converted to absorbed dose in air $D_{\gamma\ dos}$ (mGy) using conversion factors. | $D_{\gamma\ dos} = C_{\gamma\ dos} \cdot D_{\gamma\ arch}$ |
| Conversion of archive dose to absorbed dose in air based on exposure scenario radiation field specifications | Conversion factors $C_{\gamma\ rec}$ were determined by MCNP calculations to calculate dose in air, $D_{\gamma\ rec}$ from $D_{\gamma arch}$ for each exposure scenario. Different factors were used for routine and nonroutine exposure scenarios. (units of mGy) | $D_{\gamma\ rec} = D_{\gamma\ dos} / C_{\gamma\ rec}$ |
| Determination of photon radiation personal dose equivalent based on worker orientation in the radiation field for each exposure scenario. | $H_p(10)_\gamma$ is operational quantity defined as dose equivalent delivered at 10-mm depth of tissue. $H_p(10)_\gamma$ is derived from $D_{\gamma\ dos}$ by dividing by conversion factor $C_{\gamma\ Hp}$ that includes effect of radiation attenuation by 10 mm of tissue. Unique values of $C_{\gamma\ Hp}$ were calculated for each exposure scenario. (units of mSv) | $H_p(10)_\gamma = D_{\gamma\ dos} / C_{\gamma\ Hp}$ |
| Determination of neutron radiation personal dose equivalent based on preliminary neutron to photon dose ratios for each neutron exposure scenario. | $H_p(10)_n$ values in the Doses-2005 database were derived from $D_{\gamma\ dos}$ using ratio of neutron dose to gamma dose: $D_n/D_\gamma = K_n$. Value obtained was divided by conversion factor $C_{pn}$ that included effect of radiation attenuation at 10-mm depth of tissue for different exposure geometries and different spectra. (units of mSv) | $H_p(10)_n = K_n \cdot D_{\gamma\ dos} \cdot C_{pn}$ |
| Determination of respective organ doses from absorbed dose in air using spectrum- and geometry-dependent DCFs. | Organ doses received by worker in given year are obtained by multiplying $D_{\gamma\ rec}$ by organ DCFs derived for specific exposure scenarios. DCFs have been derived for each organ in database. Organ DCFs were calculated either by using publicly-available DCFs, or by MCNP calculations specific to Mayak exposure scenarios (units of mGy). | $D_{\gamma\ org\text{-}i} = C_{\gamma\ org\text{-}i} \cdot D_{\gamma\ rec}$ |

For nonstandard irradiation geometries, a radiation transport calculation was performed using the computer code MCNP (Briesmeister 2000). For MCNP calculations, an anthropomorphic phantom was used to simulate radiation transport in the human body and determine absorbed dose in individual organs. Male and female phantoms are used, separately, in the calculations, to evaluate doses for male and female workers. The phantom models are MIRD-type phantoms, with equations taken from Cristy and Eckerman (1987). This approach makes the calculations compatible with the values reported in ICRP (1996). For conversion factors calculated using MCNP and the phantom models, organ doses are calculated in both male and female phantoms. They are reported in a manner compatible with the ICRP (1996) methodology, using an average of the male and female results for organs applicable to both sexes.

An illustration of the calculated factors used to determine absorbed organ dose from Table 5.9, Group No. 5 Table 1, entitled Reactor – Central Hall, used with each of the equations in Table 5.5 is presented in Table 5.6. Analogous factors have been determined for each of the scenarios in Table 5.9 Table 1 and are presented in Volume II of the Doses-2005 documentation.

Table 5.6. Conversion factors for Central Hall scenario

| **Conversion Factor** | **Dosimeter** | **Maximum** | **Mean** | **Minimum** |
|---|---|---|---|---|
| $C_{\gamma\,rec}$, DCFs to find photon dose to air | IFK | 1.779 | 1.13 | 0.802 |
| | IFK +Pb | 0.939 | 0.88 | 0.832 |
| | IFKU | 0.834 | 0.77 | 0.724 |
| $C_{\gamma\,Hp}$, DCFs to find photon personal dose equivalent | IFK | 2.007 | 1.26 | 0.883 |
| | IFK +Pb | 1.047 | 0.97 | 0.913 |
| | IFKU | 0.910 | 0.85 | 0.802 |
| Neutron conversion factors: $K_n$: neutron/photon ratio $C_{pn}$: conversion factor | **Period** | **$K_n = D_n/D_\gamma$** | **$C_{pn}$** | |
| | 1948-1954 | 0.011 | 0.53 | |
| | 1955-1961 | 0.032 | 0.53 | |
| | 1962-1967 | 0.08 | 0.53 | |
| | 1968-1973 | 0.106 | 0.53 | |
| | 1974-1985 | 0.08 | 0.53 | |
| | 1986 – present | 0.096 | 0.53 (0.69) | |
| $C_{\gamma\text{-org}}$ Organ DCFs | **Organ** | **Factor** | **Organ** | **Factor** |
| | Bladder | 0.655 | Esophagus | N/A[(a)] |
| | Bone - marrow | 0.526 | Skin | 0.867 |
| | Bone - surface | 0.853 | Stomach | 0.627 |
| | Breast - female | 0.874 | Thyroid | 0.900 |
| | Colon | 0.576 | Uterus | 0.582 |
| | Ovaries - female | 0.532 | Brain | 0.545 |
| | Testes - male | 0.860 | Lower large intestine | 0.571 |
| | Liver | 0.590 | Kidney | 0.446 |
| | Lung | 0.588 | Small intestine | 0.597 |

a. Not Applicable.

## 5.5 MPA BETA/PHOTON DOSIMETRY TECHNOLOGY

Beginning with operation of Reactor A in 1948, Mayak utilized several dosimetry and instrument methods to monitor individual worker dose to beta and photon (x-ray and gamma) radiation. Three primary methods of measurement were used: (1) portable radiation detection instrumentation, (2) a personal pencil-type electronic ionization (e.g., IDC-1) dosimeter and (3) personnel film dosimeters. Workers were routinely assigned a canvas bag containing an individual dosimetry package (i.e., IFK and IDC-1 dosimeters). Dosimeters were typically clipped to the breast pocket. The Mayak beta/photon film dosimeters of interest to the Mayak Worker Dosimetry Study are listed in Table 5.7. The electronic ionization dosimeter (IDC) was used along with the respective personnel film dosimeters.

Table 5.7. MPA personnel film dosimeter design characteristics

| Dosimeter | Period | Reported Radiation | Filtration |
|---|---|---|---|
| IFK | ~1948–1953 | 1. Penetrating | Plastic[a] |
| IFK + Pb | ~1954–1960 | 1. Penetrating | Plastic + Pb (0.75 mm) |
| IFKU | ~1961–1991 | 1. Nonpenetrating<br>2. Penetrating | Four regions involving:<br>1. plastic[b]<br>2. Al<br>3. Pb+Al<br>4. Pb+Cd+Al |

a. Plastic holder with a density thickness of approximately 600 mg/cm2
b. Plastic holder for nonpenetrating response reduced to a density thickness of approximately 300 mg $cm^{-2}$

### 5.5.1 Studies of MPA Film Dosimeter Radiation Response

The response of the respective MPA personnel dosimeters to photon radiation has been measured using exposures at the German National Research Center for Environment and Health (GSF) calibration facility using in-air and on-phantom measurements to several selected radiation beams, and to beta radiation using GSF irradiations and electron linear accelerator irradiations by the University of Utah.

Mayak dosimeter angular response characteristics were measured using in-air and on-phantom measurements to selected beams during laboratory irradiations at GSF as follows:

June 2002 GSF Exposures (dosimeters in air)

- A10, A15, A20, A30, A40, A60, A80, A100, A120, A150, $^{137}Cs$, $^{60}Co$
- Angles of 30, 45, 60, 80 and 90 degrees

June 2003 GSF Exposures (dosimeters on phantom)

- A60, A100, A150, $^{137}Cs$, $^{60}Co$
- $^{226}Ra$
- $^{90}Sr/Y$ beta radiation
- Horizontal and vertical dosimeter orientations
- Angles of 0, 30, 45, 60, 80, 85, 90, 95, 100, 120, 135, 150 and 180 degrees

### 5.5.2 Photon Energy Response of Dosimeters

Fig. 5.1 shows the photon energy response dependence for the Mayak IFK, IFK + Pb and IFKU film dosimeters.

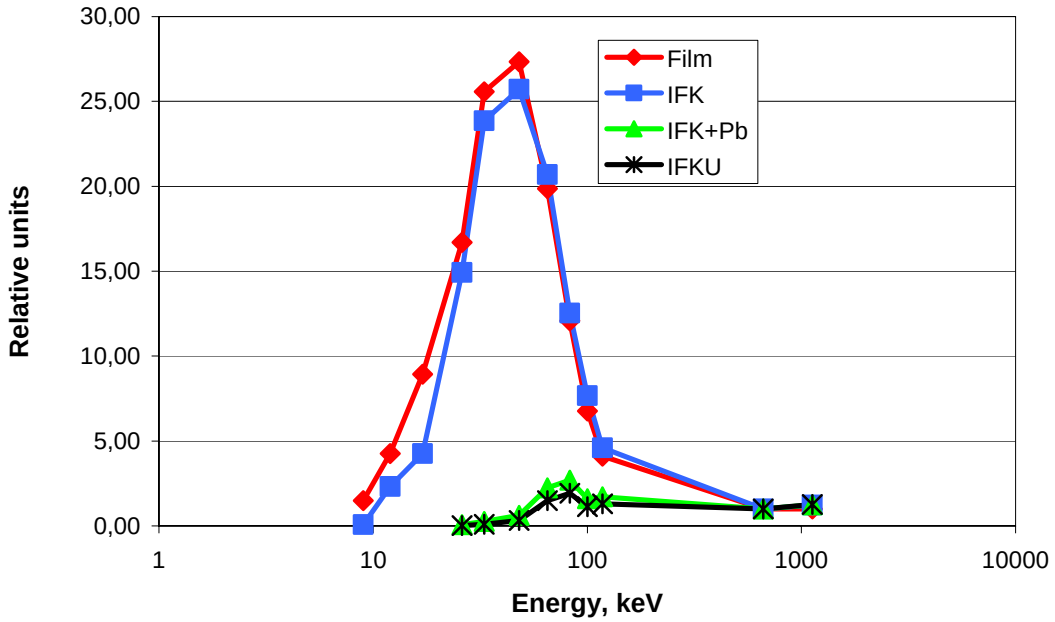



Fig. 5.1. Photon energy response of MPA film dosimeters

The data collected in these radiation studies show that the IFK dosimeter sensitivity to $^{226}$Ra is about 20% higher than to $^{137}$Cs. This is of interest because $^{226}$Ra was used to calibrate Mayak dosimeters during the 1950s and, barring other factors, would be expected to imply a negative bias of 20% in the recorded dose compared to calibrations using $^{137}$Cs.

### 5.5.3 Beta Response of Mayak Film Dosimeters

Volume II describes technical details of evaluations of the contribution of beta radiation to the dose calculated for the Mayak IFK dosimeter for nonroutine scenarios. For radiation fields with a significant beta component, the calculated penetrating dose for the IFK dosimeter will be too high because the readout process assumed that all of the dosimeter response (i.e., beta plus photon) was produced by photon radiation only.

The GSF irradiations with a $^{90}$Sr/Y beta source showed that the IFK dosimeter responds to high-energy beta radiation. Beta radiation from $^{90}$Sr/$^{90}$Y contributes a deep dose response equivalent that is approximately 20% of the magnitude of the deep dose from $^{137}$Cs gamma radiation.

Volume II describes the beta characterization of MPA film dosimeters using electron beams from a linear accelerator and MCNP modeling. Fig. 5.2 shows the response of two dosimeter types to electrons as a function of energy. The curve for the IFK+Pb detector is very similar to the expected behavior of the IFKU dosimeter. For the IFK dosimeter, the curve labeled "IFK Calculation" is the best predictor of IFK dosimeter behavior, as discussed in Volume II.

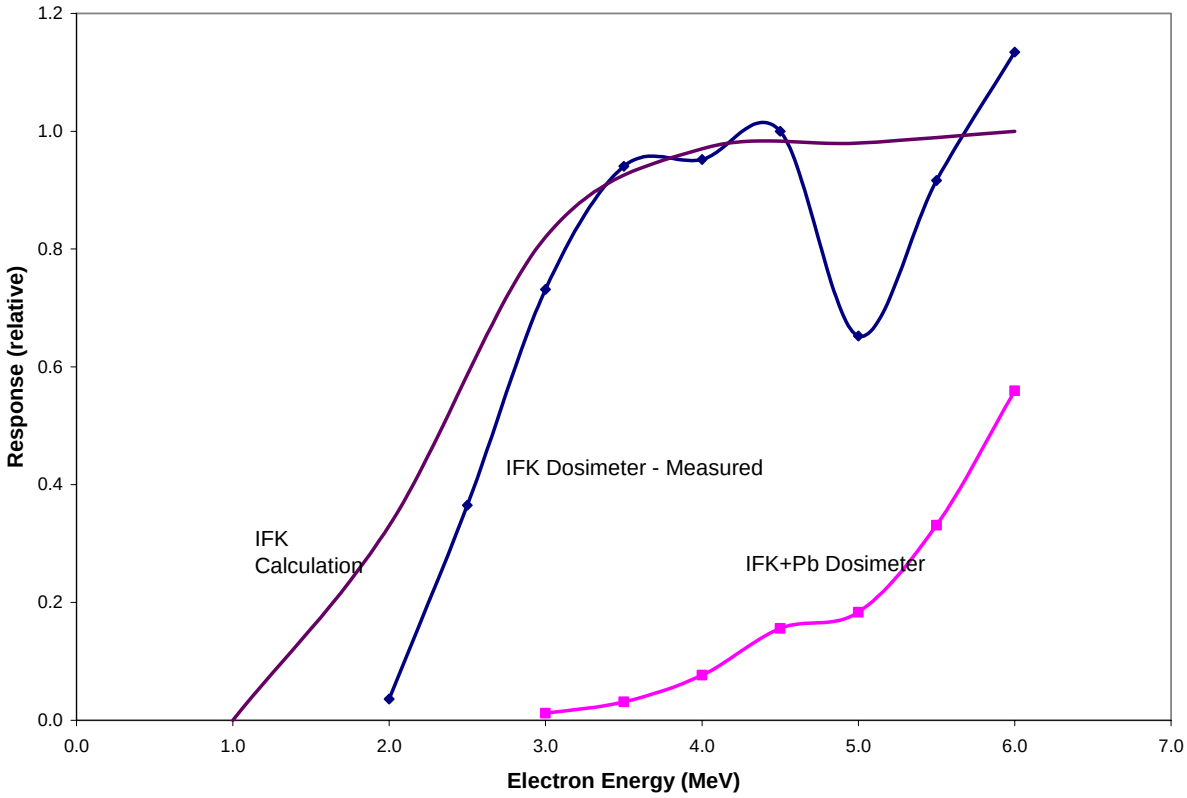



Fig. 5.2. Electron responses of the IFK and IFK+Pb dosimeters, using linac measurements and modeling results

The basic conclusions that can be drawn from the analyses in Volume II are:

- IFK+Pb and IFKU dosimeters are sufficiently insensitive to betas that no correction is required.
- The IFK dosimeter has a minor response to beta radiation with energies below 2 MeV, but no correction to dosimeter reading would be necessary.
- For betas above 2 MeV, response of the IFK dosimeter is sufficient to require a correction.
- For exposure to betas from $^{90}Sr/^{90}Y$, the IFK response to beta is approximately 20% of its response to gammas from $^{137}Cs$

### 5.5.4 Photon Angular Response of Mayak Film Dosimeters

Personal dosimeters have an angular as well as an energy-dependent response. Typical of most organizations, calibration of MPA film dosimeters was performed in an AP geometry using selected reference radiation sources. In the workplace a worker's position can change often in relation to the source of radiation.

To account for the angular effect on the archive dose, a coefficient, $K_a$, was defined to account for the difference in response between specified workplace and calibration conditions. The values of these

coefficients also depend upon spectral and angular characteristics of the respective radiation fields. In a general form, the coefficient is stated as:

$$K_a = D_{\gamma\ arch}/D, \quad (5.1)$$

where: $D_{\gamma\ arch}$ = recorded dose;
D = true absorbed dose in air received in the workplace.

The irradiations at GSF produced data that could be used along with modeling to derive correction factors for standard exposure geometries such as AP beam and isotropic radiation fields. Volume II, Section 2.2.7 contains graphs of these correction factors as a function of energy for four standard geometries and each of the three dosimeter types.

## 5.6 ANALYSES OF RECORDED ARCHIVE DOSE

Analyses of the "daily dose" database of 725,350 records of individual dosimeter dose results for approximately 8,748 workers were done to examine patterns in the recorded dose. These records were obtained from the Worker Personal Logs described in Chapter 4. These records are for the early years beginning in 1948 and ending in 1960. The expectation is that the dynamics of worker dose in nonroutine work scenarios is different from dose received in routine work. For example, differentiation of cases where exposure occurred that was greater than the dose limits would allow more accurate definition of exposure spectra and geometries. An analysis of the range in doses per shift (i.e., total dosimeter dose divided by the number of work shifts) showed the results summarized in Table 4.12.

The data were analyzed further for each worker by examining the distribution of recorded doses per shift in five selected dose intervals as shown in Table 5.8.

Table 5.8. Distribution of doses/shift for a single worker.

| Individual identification | Position index | Location (area) index | Number of "daily dose" measurements | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | | Dose interval, R/d | | | | |
| | | | Career total | 0.1– 0.3 | 0.3 – 1 | 1 – 10 | 10 – 100 | >100 |
| Worker #1 | 0 | 326 | 144 | 19 | 12 | 13 | 0 | 0 |
| Worker #2 | | 223 | 130 | 5 | 33 | 79 | 3 | 0 |

Further analyses were conducted to observe the frequency of doses per shift that occurred for each day for each of the dose intervals during the period from approximately April 1949 through March 1960. These data allow identification of time periods where comparatively high worker doses were recorded. Another type of analysis was done in which workers were assigned to groups according to their position and work area. These analyses were done to identify the number of occurrences of dose values for worker groups that occurred in each of the dose intervals and the time in which work was performed. Fig. 5.3 shows the total number of daily doses that exceeded 0.1 R/day for the identified work groups and day. This type of analysis enables a distinction to be made between routine exposure scenarios that represent typical daily work activities and the nonroutine work where higher recorded doses occurred.

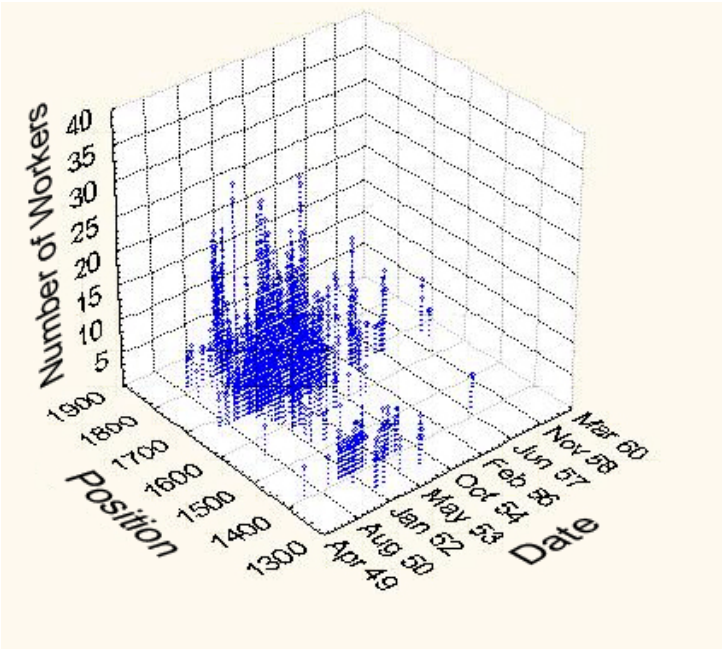



Fig. 5.3. Dependence of the total number of cases of overexposure to doses higher than 0.1 R/d on position/location and a date of work (without cases when $N_{ij} \leq 3$).

A more descriptive illustration of nonroutine exposures is provided in Fig. 5.4 where the daily dose must exceed a dose of between 1 R and 10 R per day. All occurrences were during 1952. Position 1680 approximately showed high doses during March - April and during August.

## 5.7 ANALYSES OF RANDOM WORKER SELECTION

Project 2.2 researchers randomly selected 100 workers for Project 2.4 to examine paper and computer record completeness and accuracy. Workers were selected for the auxiliary, reactor, radiochemical and plutonium chemical-metallurgical plant facilities (i.e., 100 workers for each facility). Comparison of the information on recorded annual MPA worker doses from archival records with data base information. The data are listed in Table 5.9.

Several steps were involved in the verification of these records, which also included analysis of the relocation of workers in workplaces. Information on the recorded doses for these workers is stored in MPA archives in the individual books, logs and cards. Individual books were maintained until the 1970s. These books contain information for certain intervals from one work shift to 1 mo. Dosimetry logs were maintained for annual worker doses. This annual dose for one worker was obtained by summation of doses recorded in worker books for short periods. Sometimes these journals contain the information for the same period. This situation occurs because dose for one individual for the same period of work can be contained in different sources of information. In some cases, the data from different sources do not coincide. In the process of the evaluation some doses values were corrected. For some workers, doses were added for periods when monitoring was not performed. For other workers, some information was deleted due to errors.

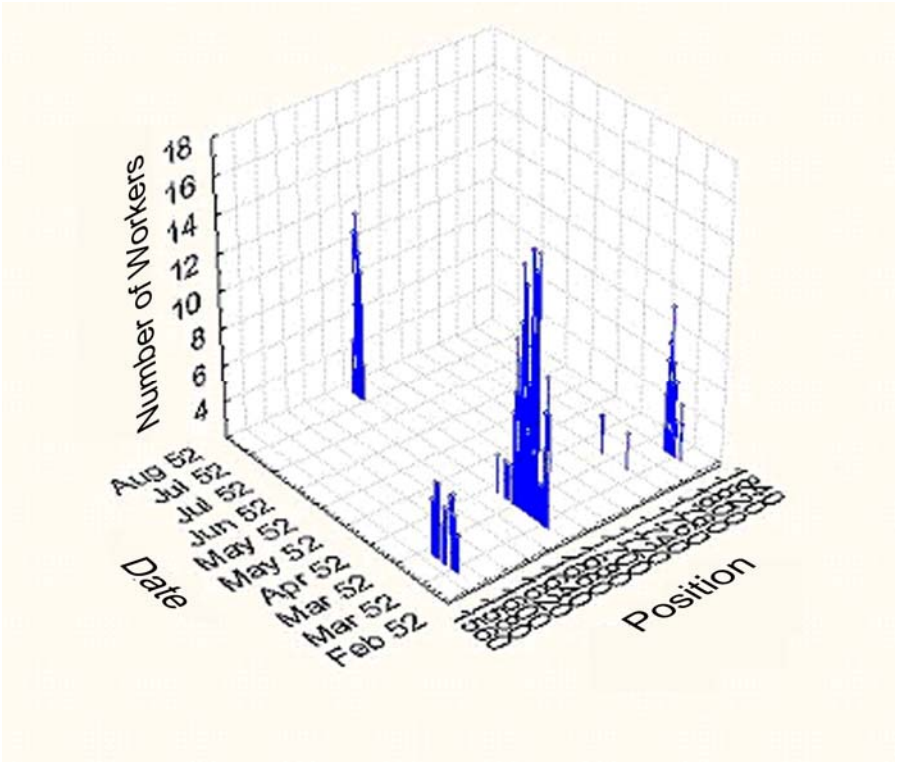



Fig. 5.4. Dependence of the total number of cases of overexposure to doses from 1 to 10 R/day upon position / location and a date of work (for dates with >3 of such cases)

Table 5.9. Records for random worker selections

| Parameter | Reactor plant | Radiochemical plant | Plutonium plant |
|---|---|---|---|
| Number of individuals | 100 | 100 | 100 |
| Number of the annual dose records in the database | 1425 | 1272 | 579 |
| Number of discrepancies in worker doses | 23 | 22 | 7 |
| Rate of discrepancies | 1.6% | 1.7% | 1.2% |

Based on this review, Mayak has taken additional steps to evaluate the accuracy of the Doses-2005 database by:

- Recalculating the monthly worker doses;
- Identifying the preferred dose for different sources of information;
- Verifying further the annual dose based on the data associated with changes of the workplaces by the worker.

In some cases, work with archives did result in a change to the data in documents. These changes are included in Doses-2005. The overall percentage of discrepancies between electronic and archival data is: 1.6; 1.7; and 1.2 % for the reactor, radiochemical, and plutonium plants samples respectively. The number of data discrepancies in archival records and the Doses-2005 database are listed in Table 5.10.

Table 5.10. Identified data discrepancies

| Causes | Reactor plant | Radiochemical plant | Plutonium plant |
|---|---|---|---|
| Total number | 23 | 22 | 7 |
| Entry error | 7 | 3 | 1 |
| Monthly dose summation (for annual dose) | 11 | 16 | 4 |
| Verification from different sources | 5 | 3 | 2 |

**Auxiliary Plant Workers**

A similar analysis was conducted of auxiliary plant workers. Project 2.2 selected 100 workers (i.e., actually 99 workers because records for 2 yrs were identified for one worker) to examine the trend in doses for auxiliary workers because there was consideration of these workers to be in-study controls. A total of 512 annual dose records were identified for 49 of the 99 (i.e., 100 records but 2 records for one worker) workers in the "Mayak" PA archive. There were no data available for the other 50 workers. An additional step in this analysis was to compare the Mayak-determined annual doses with corresponding dose data from Koshurnikova's SUBI laboratory. Discrepancies were found in 30 cases for 11 workers. The analysis of discrepancies between the SUBI data and Mayak archival data is presented in Table 5.11.

Table 5.11. Identified causes of discrepancies between SUBI and Mayak Archival data

| Discrepancy cause | Number of cases |
|---|---|
| Rounding off the doses from 3 to 2 places after the decimal point | 11 |
| Errors of annual dose calculation on the basis of data for shorter periods of time | 5 |
| Consideration of doses not from all places of workers' exposure | 14 |

Annual distribution of average dose for the 49 auxiliary plant workers with registered dose is listed in Table 5.12. Fig. 5.5 is a distribution diagram of the auxiliary worker doses. The distribution of averaged doses by year for reactor, radiochemical and auxiliary plants is presented in Fig. 5.6 for comparison. It is apparent that the same trend in measured dose is observed. As such, auxiliary workers cannot be used as in-study controls without some additional criteria to ensure that these are truly relatively low-dose workers.

The foregoing efforts were done to address the highest stated Project 2.2 need in the Dosimetry Protocol to validate as feasible the early, high dose records.

## 5.8 WORKPLACE EXPOSURE SCENARIOS

Radiation fields in MPA facilities are a complex mix of beta, photon and neutron radiation. Thus, a worker's dosimeter reading might not have been an accurate approximation to the radiation dose actually received in the workplace, particularly in the earliest days of operation when the dosimeters were an early design, and were worn in a radiation field that did not match the calibration conditions. An accurate evaluation of worker dose must include a description of the radiation environment creating the worker exposure. While it is not possible to produce a detailed, individualized description of the radiation environment each worker was in, it is reasonable to develop a limited number of exposure scenarios that describe the radiation environment experienced by large groups of workers. Accordingly, exposure scenarios were applied to the respective worker groups discussed in Chapter 4 (see Table 4.11). For each of these scenarios, correction factors can be developed for application to the worker dosimeter readings. In this way a worker's dosimeter reading can indicate the intensity of the radiation field, compared to

Table 5.12. Annual distribution of average dose for auxiliary plant workers

| Year | Number of persons | Average dose (R) |
|---|---|---|
| 1950 | 2 | 67.76 |
| 1951 | 2 | 64.34 |
| 1952 | 2 | 30.09 |
| 1953 | 1 | 11.21 |
| 1954 | 9 | 3.14 |
| 1955 | 7 | 0.22 |
| 1956 | 3 | 4.28 |
| 1957 | 9 | 20.75 |
| 1958 | 5 | 11.62 |
| 1959 | 6 | 6.45 |
| 1960 | 6 | 3.02 |
| 1961 | 16 | 1.86 |
| 1962 | 16 | 1.45 |
| 1963 | 17 | 1.74 |
| 1964 | 17 | 1.90 |
| 1965 | 21 | 1.68 |
| 1966 | 22 | 1.35 |
| 1967 | 23 | 1.08 |
| 1968 | 22 | 0.83 |
| 1969 | 24 | 0.61 |
| 1970 | 22 | 0.72 |
| 1971 | 19 | 0.93 |
| 1972 | 19 | 0.86 |

other workers in the same exposure scenario category. The exposure scenario correction factor can then be applied to this dosimeter reading to improve the estimate of the dose actually received.

Worker exposures can generally be grouped into routine and nonroutine exposures. Routine exposures were those that were encountered during the regular performance of work in Mayak facilities, and can be described with a knowledge of the facility layout, engineered radiation protection devices such as shields and gloveboxes, worker position and activity, and the nature of the radiation source. This analysis assumes that the routine exposure scenarios stayed constant from day to day for long periods. Nonroutine exposures, on the other hand, occurred in response to a deviation from the normal routine, often caused by equipment failure. Nonroutine exposures usually resulted in higher doses than those received during routine exposures, and were usually of short duration.

### 5.8.1 Reactor Photon Exposure

Much of the routine and nonroutine exposure in the reactor buildings occurred in the Central Hall. Exposure also occurred in auxiliary buildings, and some workers were exposed in both the Central Hall and the auxiliary buildings.

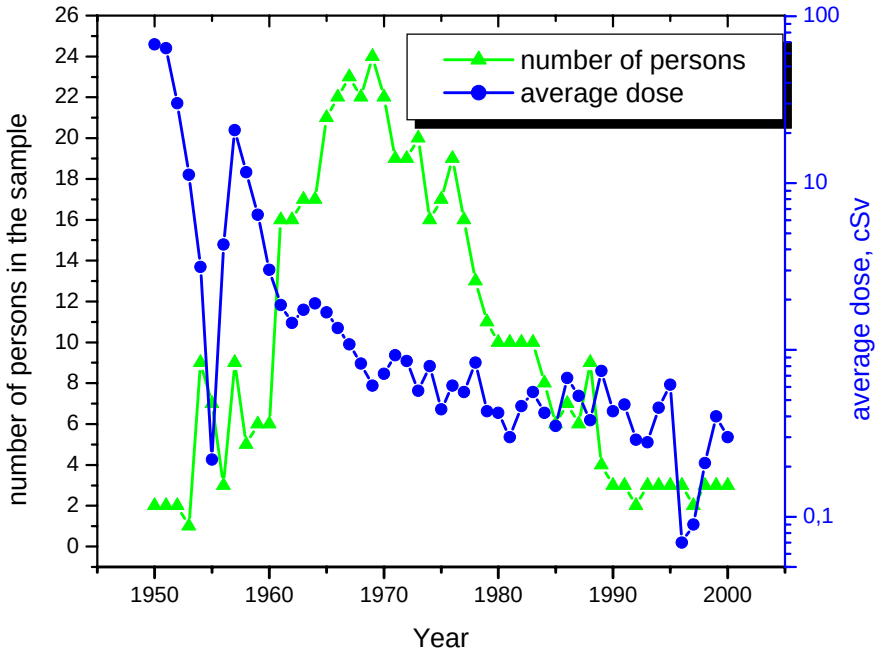



Fig. 5.5. Number of persons and average annual dose for the sample of 49 auxiliary production workers

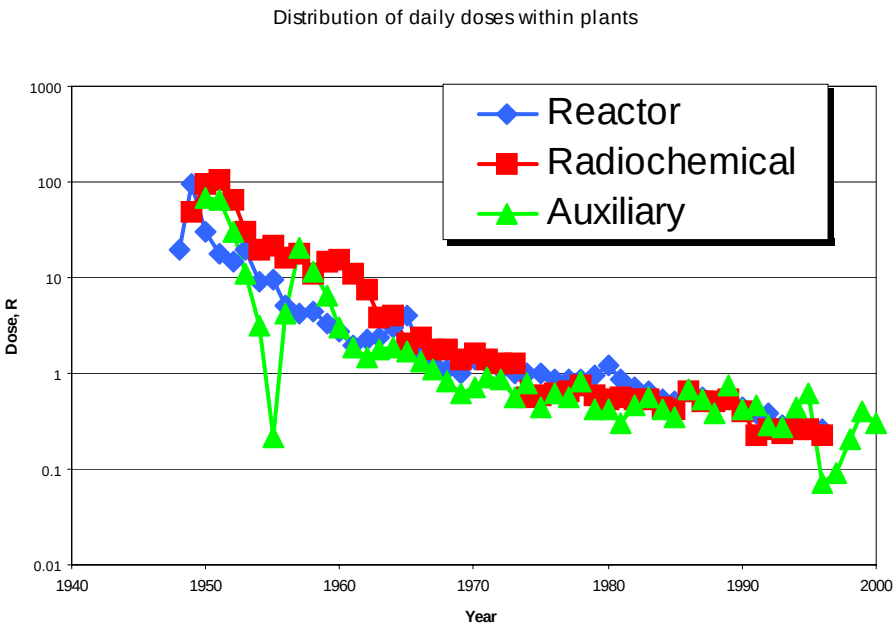



Fig. 5.6. Annual distribution of the average dose for 49 workers of sample of auxiliary production workers compared with average dose of reactor and radiochemical plant personnel

Routine exposure scenarios were defined for workers in the reactor area and in the radiochemical and plutonium production areas, and for "other workers." For nonroutine exposures, two scenarios were defined: one in the reactor plant and one in the radiochemical plant. Early in the study, a set of eight basic scenarios was developed, and these were used for converting archive dose to dose in air. Recently, the set of scenarios was significantly expanded to include more worker orientations and variations in exposure geometry with time. This expanded set of scenarios was used for converting dose in air to organ dose. The expanded set of scenarios is shown in Table 5.13 and described in more detail in Volume II, Section 4.

Table 5.13. Relative importance of worker orientation and photon spectrum for reactor exposure scenario

| Worker orientation | Photon spectrum | Weight |
|---|---|---|
| Worker upright | Reactor spectrum | 0.150 |
| Worker upright | Fission products with 10-hr decay | 0.175 |
| Worker upright | Fission products with 1-yr decay | 0.175 |
| Worker bent at waist | Reactor spectrum | 0.150 |
| Worker bent at waist | Fission products with 10-hr decay | 0.175 |
| Worker bent at waist | Fission products with 1-yr decay | 0.175 |

#### 5.8.1.1 Routine Photon Exposure Scenarios in Reactor Buildings

Three scenarios were developed for routine exposures at the reactors:

1. Exposure in the Central Hall
2. Exposure in the auxiliary rooms of the reactor building
3. Exposure in both the auxiliary rooms and in the Central Hall

#### 5.8.1.2 Exposure in the Central Hall

This scenario is applicable to a worker working full-time in the reactor Central Hall. The scenario assumes that the source of radiation is uniformly distributed on the floor. This source distribution could be caused either by radiation emitted by the reactor core, or by radiation emitted by contamination on the floor. There are three types of sources:

1. A reactor spectrum, with radiation emitted by fission reactions in the reactor core, transported up through the shielding above the core, and emerging as isotropically directed radiation emitted from the floor surface.

2. Fission products with short decay times. In this case, radioactive contamination consisting of a mixture of fission products is assumed to cover the surface of the floor. The mixture of fission products is typical of the radionuclides present only 10 hr after the end of irradiation.

3. Fission products with long decay times. In this case, radioactive contamination consisting of a mixture of fission products, is assumed to cover the surface of the floor. The mixture of fission products is typical of the radionuclides present after 1 yr of radioactive decay after the end of irradiation.

There are two possible orientations for the worker: standing upright and bent at the waist. In the first orientation, the worker's torso and the dosimeter attached to it are vertical with respect to the floor. In the

second orientation, the worker's torso and the dosimeter attached to it are horizontal with respect to the floor. The second orientation is equivalent to a worker who is standing near a contaminated wall.

The Central Hall scenario assumes that a worker spends exactly half of the time in a vertical orientation (standing upright) and the other half in a horizontal orientation (bent at the waist) in this exposure condition. For source term, the scenario assumes that the reactor spectrum accounts for 30% of a worker's exposure, and each of the fission-product spectra account for 35% of exposure. Thus the scenario exposure has six components, with relative importance as shown in Table 5.13.

#### 5.8.1.3 Exposure in the Auxiliary Areas of the Reactor

Workers exposed in auxiliary areas of the reactor building, away from the Central Hall, were exposed to radiation fields generated by a room with radioactive contamination on the walls, floor, and ceiling. The irradiation geometry would be isotropic. The photon spectrum would be fission products with a decay time of 1 yr, identical to the spectrum used for one of the exposure modes in the Central Hall scenario.

#### 5.8.1.4 Exposure in both the Auxiliary Rooms and in the Central Hall

This scenario assumes that a worker spends 70% of his or her time in the auxiliary rooms and 30% in the Central Hall during any period between dosimeter exchanges. A dosimeter correction factor for this exposure scenario can be found by combining 70% of the correction factor from the Auxiliary Rooms with 30% of the correction factor from the Central Hall.

#### 5.8.1.5 Nonroutine Photon Exposure Scenarios in Reactor Buildings

This scenario assumes that the worker is working with freshly-irradiated fuel elements on the floor of the Central Hall. This situation could arise when process tubes containing fuel elements were removed through the top of the reactor rather than discharged through the discharge chute at the bottom of the reactor core. In some cases the tube was damaged and the fuel elements spilled out when it was withdrawn. In other cases, the fuel elements remained in the tubes, creating a radiation field that exposed workers in the Central Hall. One scenario was chosen to represent this class of exposures.

The exposure geometry for this scenario is an AP beam because the worker would be facing the fuel elements.

### 5.8.2 Radiochemical Plant and Plutonium Production Areas Photon Exposure

Workers were exposed in various locations in the radiochemical plant and plutonium operational areas.

#### 5.8.2.1 Routine Photon Exposure Scenarios in Radiochemical Plant and Plutonium Production

Four basic scenarios for routine exposure were identified for the radiochemical plant and the plutonium production areas, and a number of these were subdivided into subscenarios:

- Head-end and intermediate nodes of the radiochemical plant
    - Other occupations
    - Repair workers, laboratory workers and samplers

- Foot-end nodes of radiochemical plant and plutonium production
    - Operators
    - Repair workers
    - Storage workers
    - Others

- Radiochemical areas of the plutonium plant
    - Operators
    - Repair workers
    - Batch workers, material accountants, warehouse workers
    - Others

- Chemical and metallurgical areas of the plutonium plant
    - Operators
    - Batch workers, material accountants, warehouse workers
    - Others

#### 5.8.2.2 Head-End and Intermediate Nodes of the Radiochemical Plant

Exposure scenarios for this class of worker include working in a room with dimensions of approximately 6 m by 8 m by 4 m. Radioactive contamination was typically present on the walls, floor and ceiling.

For the workers called "others," two different photon spectra were assumed to characterize the worker exposure in this room. One assumed that photons were emitted from fission products, decayed 120 d (matching the holdup time of irradiated fuel elements before dissolution in Plant B), and that the radiation was shielded by high-density concrete, 30 cm thick. The second spectrum was assumed to be emitted by fission products decayed for 1 yr, but the radiation was not attenuated by a shield. Thus, the first spectrum would be typical of radiation emitted outside the concrete-walled room, and the second spectrum would be typical of contamination on the inside surfaces of the wall.

The exposure geometry was assumed to be isotropic. The concrete-shielded spectrum would contribute 70% of the worker's dose and the unshielded (surface contamination) spectrum would contribute 30% of the dose.

Repair workers, laboratory workers, and samplers were assumed to be working at gloveboxes and hot cells. The radiation was emitted by fission products with 120-d decay, with the radiation attenuated by 3-mm-thick steel. The geometry was assumed to be a parallel AP beam.

#### 5.8.2.3 Foot-End Nodes of the Radiochemical Plant and Plutonium Production Areas

For operators working in these areas, it is assumed that the worker is facing the radiation source, as might be the situation if the source is in a glovebox. The exposure geometry is an AP beam. The spectra are assumed to vary by year of exposure, so three spectra are used for this scenario. The exposure geometry was assumed to be an AP beam.

Exposure of repair workers in the foot-end nodes assumes that the worker is in a room with contamination on the walls, floor, and ceiling. The worker changes position frequently, rather than consistently facing a single source, so an isotopic exposure geometry is appropriate. For each general period, two possible spectra could produce the exposure, and each had an equal probability of contributing to a worker dose.

In the storage areas of the foot-end nodes, there were two typical exposure geometries: one with a plane source underfoot and one with the operator facing the radiation source, for an AP beam exposure. There were two different spectra in each period, equally probable for contributing to a worker exposure. The AP beam is considered to produce 25% of a typical exposure, and the plane source underfoot is considered to produce 75% of the exposure.

For other workers at the foot-end nodes, the exposure is assumed to be a mixture of the radiation fields for operators and for storage workers. The results of this scenario can be found with a 50% contribution from the operator scenario and a 50% contribution from the storage worker scenario. All photon spectra for this exposure category are presented in Volume II.

#### 5.8.2.4 Radiochemical Areas of the Plutonium Plant

Four subscenarios were considered for exposure in the radiochemical areas of the plutonium plant. The three subscenarios of operators, repair workers, and others are identical to the corresponding subscenario of the foot-end facilities of the radiochemical plant and plutonium production. The third subscenarios for these two facilities are also identical to each other, although for the foot-end facilities of the radiochemical plant and plutonium production the subscenario covers storage workers, while for the radiochemical areas of the plutonium plant the scenario covers batch workers, material accountants, and warehouse workers. The photon spectra for this scenario are presented in Volume II.

#### 5.8.2.5 Chemical-Metallurgical Areas of the Plutonium Plant

Three subscenarios were considered for exposure in the chemical-metallurgical production areas. For operators, exposure was assumed to be an AP beam, as in the other scenarios involving operators. For batch workers, material accountants and warehouse workers, exposure geometry was assumed to be a contaminated floor. For other workers, exposure was assumed to be a 50%/50% mixture of the other two subscenarios. For each scenario, three spectra were considered, depending on the time period of the exposure. The spectra for these scenarios are presented in Volume II.

#### 5.8.2.6 Other Routine Photon Exposure Scenarios

A routine exposure category labeled “Other” includes workers who received doses at facilities other than those in Groups 1.1 through 7. Among these are some workers who responded to the 1957 accident. Exposure to contaminated soil was chosen as a typical radiation exposure condition for this group. The radiation source is assumed to be radioactive contaminants distributed uniformly in the top 2 cm of soil. The source term is assumed to be fission products, decayed by 3 yr. The exposure geometry is assumed to be isotropic.

#### 5.8.2.7 Nonroutine Photon Exposure Scenarios in Radiochemical Plant and Plutonium Production – Process Leaks

Nonroutine radiation exposure occurred in the radiochemical plant as a result of process leaks. Contamination could occur on floors and on walls. Workers would respond to clean up the contamination, resulting in the nonroutine exposures. There were two exposure geometries for this scenario, each one contributing 50% of the worker's exposure. One geometry was for a contaminated floor, with the worker standing above it. The other was for a contaminated wall, so it is a semi-isotropic exposure, facing the worker. The photon spectrum for this scenario is presented in Volume II.

### 5.8.3 Summary of Photon Exposures

Table 5.14 summarizes the scenarios for photon exposure in the MPA facilities. These scenarios were used for calculating organ DCFs. If the photon spectrum is identified by a spectrum number, this refers to the spectra listed in Table 5.15 and described further in Volume II.

### 5.8.4 Neutron Exposure Scenarios

Routine neutron exposure scenarios for the Mayak main reactor, radiochemical processing, and plutonium production plants were used to identify seven groups of workers with potential neutron exposure. In general, structural materials in these plants and the associated equipment provide shielding that has significantly attenuated neutron radiation in most workplaces such that the contribution of the neutron dose component to the total personnel dose was generally less than 1%, with a few exceptions. These exceptions involve specific workplace rooms or operations where there is significant neutron radiation as follows:

- Operators' rooms at the end nodes at the radiochemical plants;
- Rooms in the radiochemical and plutonium chemical-metallurgical plants used for integration and batching of finished plutonium products, where containers are removed from the process line, and batches of product are gathered and sent to the finished-products storage area;
- Rooms in the radiochemical and plutonium chemical-metallurgical plants used for intermediate and long term storage areas for finished products.

The seven groups of workers are summarized in Table 5.16. Evaluation of the neutron source term, spectral and spatial characteristics for these seven groups of workers has been used to evaluate the significance of neutron dose to individual workers.

## 5.9 DOSE RECONSTRUCTION

Reconstruction of Mayak recorded individual worker doses generally involves parameters as follows:

- Assessment of the completeness of the archive dose according to Mayak administrative monitoring practices adopted originally by operating facilities, and since 1971 by Mayak central services, to monitor and record personnel dose based on technical, administrative, and statutory compliance considerations.

Table 5.14. Summary of routine photon exposure scenarios

| Group No. | Facility | Production site | Occupations | Exposure geometry | Photon Spectrum |
|---|---|---|---|---|---|
| 1.1 | Radiochemical Plant | Head-end and Intermediate facilities | Others | Isotropic | Fission products, 120-d decay, shielded by 30-cm concrete (70%) and Fission products, 1-yr decay, no shielding (30%) |
| 1.2 | | | Repair workers, laboratory workers, Samplers | AP beam (facial exposure) | Fission products, 120-d decay, shielded by 3-mm steel |
| 2.1 | Radiochemical Plant and Plutonium Production | Foot-end facilities | Operators | AP beam (facial exposure) | Before 1953: Spectrum 1<br>1953–1961: Spectrum 2<br>After 1961: Spectrum 3 |
| 2.2 | | | Repair workers | Isotropic | Before 1953: Spectra 4 (50%) and 5 (50%)<br>1953–1961: Spectra 6 (50%) and 7 (50%)<br>After 1961: Spectra 8 (50%) and 9 (50%) |
| 2.3 | | | Storage workers | AP beam (25%) and plane source on floor (75%) | Before 1953: Spectra 10 (50%) and 11 (50%)<br>1953–1961: Spectra 12 (50%) and 13 (50%)<br>After 1961: Spectra 14 (50%) and 15 (50%) |
| 2.4 | | | Others | 50% 2.1 and 50% 2.3 | |
| 3.1 | Plutonium Plant | Radio-chemical Areas | Operators | AP beam (facial exposure) | Before 1953: Spectrum 1<br>1953–1961: Spectrum 2<br>After 1961: Spectrum 3 |
| 3.2 | | | Repair workers | Isotropic | Before 1953: Spectra 4 (50%) and 5 (50%)<br>1953–1961: Spectra 6 (50%) and 7 (50%)<br>After 1961: Spectra 8 (50%) and 9 (50%) |
| 3.3 | | | Batch workers, material accountants, warehouse workers | AP beam (25%) and plane source on floor (75%) | Before 1953: Spectra 10 (50%) and 11 (50%)<br>1953–1961: Spectra 12 (50%) and 13 (50%)<br>After 1961: Spectra 14 (50%) and 15 (50%) |
| 3.4 | | | Others | 50% 3.1 and 50% 3.3 | |

Table 5.14. Cont'd

| | | | | | |
|---|---|---|---|---|---|
| 4.1 | Plutonium Plant | Chemical and Metallurgical Areas | Operators | AP beam | Before 1953: Spectrum 16<br>1953–1961: Spectrum 17<br>After 1961: Spectrum 18 |
| 4.2 | | | Batch workers, Material Accountants, Warehouse workers | Plane source on floor | Before 1953: Spectrum 19<br>1953–1961: Spectrum 20<br>After 1961: Spectrum 21 |
| 4.3 | | | Other | 50% 4.1 and 50% 4.2 | |
| 5 | Reactor Plant | Central Hall | All | Plane source on floor (50%), Plane source facing dosimeter (50%) | For each source:<br>Photons from reactor (30%),<br>Fission products, 10-hr decay (35%),<br>Fission products, 1-yr decay (35%) |
| 6 | | Reactor Auxiliary Areas | All | Isotropic | Fission products, 1-yr decay |
| 7 | | Mix of Groups 5 (30%) and 6 (70%) | All | Floor & Isotropic | Mix of spectra from groups 5 (30%) and 6 (70%) |
| 8 | Other | | | Isotropic | Fission products, 3-yr decay, mixed in 2-cm of soil |

- Assessment of the accuracy and limitations of dosimetry technology capabilities to measure dose in the Mayak workplaces. These capabilities have been assessed from response characteristics of the Mayak dosimetry systems to beta and photon radiation of selected energies and angles of irradiation.

- Development of routine exposure scenarios for primary workplace radiation field spectral and directional parameters considered to be most significant to worker dose, including worker exposure orientations.

For Mayak workers without an archive dose, dose reconstruction is done through assigning a work group category and an exposure scenario based on knowledge of the workplace conditions. For all workers, each record of annual dose includes a work group category and an exposure scenario.

The process to reconstruct Mayak worker occupational external radiation dose, which is describe in Volume II, is comprised of several steps as follows:

- Analysis of the potential missed dose for unmonitored workers who are defined to be workers with no recorded dose throughout their employment at Mayak.

- Analysis of the potential missed dose for monitored workers where recorded dose is not available for all work periods in their employment at Mayak.

Table 5.15. Photon spectra used in Scenarios 2.1 through 4.3

| Spectrum number | Period | Source material[1] | Source/shielding configuration | Other |
|---|---|---|---|---|
| 1 | Before 1953 | $PuO_2$ with 100,000 Bq/g fission products | Source inside cask with 2-mm steel walls. Cask inside glovebox with 2-mm steel shielding | Spectrum at location of operator's workplace |
| 2 | 1953–1961 | $PuO_2$ with 1,000 Bq/g fission products | | |
| 3 | After 1961 | $PuO_2$ with no fission products | | |
| 4 | Before 1953 | $PuO_2$ with 100,000 Bq/g fission products | 2-mm steel | Spectrum at location of repair worker's workplace |
| 5 | | | Thin layer of $PuO_2$ on glovebox walls, no shielding | |
| 6 | 1953–1961 | $PuO_2$ with 1,000 Bq/g fission products | 2-mm steel | |
| 7 | | | Thin layer of $PuO_2$ on glovebox walls, no shielding | |
| 8 | After 1961 | $PuO_2$ with no fission products | 2-mm steel | |
| 9 | | | Thin layer of $PuO_2$ on glovebox walls, no shielding | |
| 10 | Before 1953 | $PuO_2$ with 100,000 Bq/g fission products | Cask with 2-mm steel walls | Spectrum in a workplace in storage facility |
| 11 | | | 4-mm steel[2] | |
| 12 | 1953–1961 | $PuO_2$ with 1,000 Bq/g fission products | Cask with 2-mm steel walls | |
| 13 | | | 4-mm steel[2] | |
| 14 | After 1961 | $PuO_2$ with no fission products | Cask with 2-mm steel walls | |
| 15 | | | 4-mm steel[2] | |
| 16 | Before 1953 | Pu metal with 100,000 Bq/g fission products | Pu metal in glovebox, with 2-mm steel shielding | Spectrum at location of operator's workplace |
| 17 | 1953–1961 | Pu metal with 1,000 Bq/g fission products | | |
| 18 | After 1961 | Pu metal with no fission products | | |
| 19 | Before 1953 | Pu metal with 100,000 Bq/g fission products | Pu metal, in cask inside container, 4-mm wall thickness. Array of 86 containers in storage room | Averaged spectrum in workplace in storage facility |
| 20 | 1953–1961 | Pu metal with 1,000 Bq/g fission products | | |
| 21 | After 1961 | Pu metal with no fission products | | |

[1]During the earliest years of plutonium extraction, a large amount of fission products remained in the separated plutonium. As extraction technologies improved, the fission product contamination decreased.
[2]Storage facility containing 86 containers, each holding a cask of $PuO_2$. Total wall thickness of cask and container equals 4 mm steel.

- Assignment of the exposure scenario.
- Assignment of the work group category.
- Calculation of the annual adjusted archive photon dose, $D_{\gamma\,arch}$, for completeness and consistent units used in calibration.
- Calculation of the annual estimated photon dose, $D_{\gamma\,arch}$, for unmonitored workers.

Table 5.16. Groups of Mayak workers subjected to neutron exposure

| Group No. | Facility | Production site | Occupations |
|---|---|---|---|
| Group 1 "N" | Radiochemical Plant | End facilities | Assemblers, end product acceptance |
| Group 2 "N" | | End facilities | Material account and storage |
| Group 3 "N" | Plutonium Plant | Radiochemical facility | Material account and storage |
| Group 4 "N" | | Chemical-metallurgy facility | Assemblers, end product acceptance |
| Group 5 "N" | | Chemical-metallurgy facility | Material account and storage |
| Group 6 "N" | Reactor Plant | Central Hall | All occupations |
| Group 7 "N" | | 30% of time in Central Hall, rest of time in auxiliary areas | All occupations |

- Calculation of the reconstructed absorbed dose in air for photon radiation, $D_{\gamma\,rec}$, using corrections based on the exposure scenario and work group category.

- Calculation of the photon radiation personal dose equivalent, $H_p(10)_\gamma$, based on worker orientation in the workplace field as specified in the exposure scenario.

- Calculation of the neutron radiation personal dose equivalent, $H_p(10)_n$, based on using neutron to gamma ratios.

- Calculation of the absorbed organ dose from photon radiation, $D_{\gamma\,org\text{-}i}$ using the organ DCFs

### 5.9.1 Missed Photon Dose

#### 5.9.1.1 Unmonitored Workers

Recorded external doses are not available in the MPA archives for every worker. For example, in the original 18,831 worker cohort, 3,016 workers have been identified without records, as summarized in Table 5.17.

Table 5.17. MPA Unmonitored Workers

| Facility | Number of Workers |
|---|---|
| Plutonium | 2,457 |
| Reactor | 644 |
| Radiochemical | 285 |
| Mixed[(a)] | 138 |
| Total | 3,016 |

[(a)]Workers who moved among facilities.

Doses are not available for all workers typically for one or more of the following reasons:

- Workers were monitored and doses recorded, but the recorded dose records have been lost;

- Some categories of workers were not monitored because the potential radiation exposure was less than established requirements for assigning dosimeters. The plutonium plant is the most typical example.

- Workers were not assigned dosimeters because of a lack of adequate dosimeters. This occurred primarily during the early years of operation. For example, in the first years of MPA operation, dosimeters were given to workers at the main reactor, chemical, and plutonium facilities because their potential dose was higher than the anticipated doses of workers from auxiliary facilities.

Missed dose to workers during the early period of MPA operations is of particular interest because of the potential for significant missed dose. This period involved work activities to develop capabilities to handle and process irradiated fuel and to recover and refine plutonium. However, at any time, workers with the highest potential for significant radiation dose were preferentially supplied with the available dosimeters. As noted in Table , the largest number of workers without film dosimeter records occurred in plutonium facilities where external doses were less.

The approach used to reconstruct dose in Doses-2005 was developed by MPA dosimetry experts based on the doses measured for workers inside the facility according to worker occupation and work tasks and on the workplace source term, exposure parameters and work activities involved in assigning the individual worker occupation group and scenario category.

#### 5.9.1.2 Monitored Workers

A monitored worker is defined as a worker who was monitored at any time during employment at the MPA. Missed dose for monitored workers can arise from:

- Dose from photon radiation not recorded for all employment periods when work was actually performed. This is likely particularly in the early years because dosimeters were assigned to the workers with the expected highest doses; in later years, dosimeters might not have been assigned unless the expected dose exceeded 30% of allowable levels.

- Unrecorded dose from photon radiation because the issued dosimeter registered a response of less-than the Minimum Detection Level.

- Unrecorded dose from neutron radiation which is expected to be significant for specified workers in specific workplaces as discussed in Section 7.1.

### 5.9.2 Photon Absorbed Dose in Air

The photon absorbed dose in air is a value that is used as an intermediate step between the worker's recorded dosimeter reading and the evaluated personnel dose equivalent or organ doses. It is defined as the dose to air at the location of the dosimeter. It is a quantity that can be derived from dosimeter readings using appropriate corrections, and is also a quantity that can be modeled. For entries in Doses-2005, it is either derived from a worker's dosimeter reading or is found by one of the "missed photon dose" methods. It is used to determine the photon and neutron personal dose equivalent values ($H_p(10)$) and the photon organ doses.

The worker's archival dose, $D_{\gamma\,arch}$, is the sum of a worker's dosimeter responses for a year for a worker who was monitored, or the reconstructed annual dose for a worker who was not monitored. $D_{\gamma\,arch}$ is expressed in the radiation units that were used during the monitoring period. $D_{\gamma\,arch}$ is converted to $D_{\gamma\,dos}$ by making the following corrections:

- Partitioning into routine and nonroutine components,

- Subtracting off a beta component for certain nonroutine scenarios, as discussed in Section 4.2.3 of Volume II.

- Converting the adjusted $D_{\gamma\ arch}$ from the unit of record to mGy using Equation 5-2:

$$D_{\gamma\ dos} = C_{\gamma\ dos} \cdot D_{\gamma\ arch\ adj} \quad (5\text{-}2)$$

$C_{\gamma\ dos}$ is a factor that converts from the unit of record to mGy. The unit of record changed over time, so the value of $C_{\gamma\ dos}$ varies over time as shown in Table 5.18.

Table 5.18. $C_{\gamma\ dos}$, Dose conversion factors for quantities and units

| Year of Monitoring | Quantities and Units of Record for $D_{\gamma arch}$ | Conversion Factor, $C_{\gamma dos}$ |
|---|---|---|
| 1948–1953 | *Exposure*, roentgen | 8.7 |
| 1954–1991 | Absorbed dose in tissue, rad | 9.1 |
| 1992–1999 | $H_p(10)$, rem | 9.1 |
| 2000–present | Effective dose, mSv | 0.91 |

$D_{\gamma\ dos}$ , which has the physical quantity of absorbed dose in air, is further converted to $D_{\gamma\ rec}$ by taking into account the radiation field spectral/directional parameters associated with the exposure scenario assigned to the worker, as shown in Equation 5-3.

$$D_{\gamma\ rec} = \cdot\ D_{\gamma\ dos} / C_{\gamma\ rec} \quad (5\text{-}3)$$

where $C_{\gamma\ rec}$ is a correction factor that accounts for the energy distribution and the exposure geometry of the worker exposure scenario. These correction factors were found using MCNP calculations. $D_{\gamma\ rec}$ also is absorbed dose in air, in mGy. The corrections that are applied to $D_{\gamma\ arch}$ are described below.

#### 5.9.2.1 Adjustments to Recorded Photon Dose for Routine Scenarios

The film dosimeters used at Mayak before 1992 had a dose response that was energy- and angular-dependent. Section 2.2 described how the energy and angular response of the three dosimeter types were characterized during measurement campaigns at GSF in Munich, Germany, and how the dosimeter types were modeled to further characterize the dosimeter responses. Based on these characterizations, conversion factors could be calculated for monoenergetic photon beams in various irradiation geometries. These factors are presented on graphs in Volume II, Section 2.3 (Figs. 2-32 through 2-35). The GSF irradiations were also used to validate the MCNP models used to calculate conversion factors.

The actual conversion factors used to obtain the dose in air, $D_{\gamma\ rec}$ from $D_{\gamma arch}$ were determined by MCNP calculations. For each response scenario, an exposure geometry and photon spectrum was identified. Then for each scenario, calculations were performed to determine the dose to film emulsion and the actual dose in air at worker locations in the facility. The values of the conversion factors in Figs. 2-32 through 2-35 in Volume II were used as weighting factors in the MCNP calculations. The ratio of the dose in air to dose to emulsion is the conversion factor.

The calculated conversion factors are listed in Table 5.19. Table 5.19 includes a maximum and minimum value for each conversion factor. These bounding values were found by performing a set of calculations with the dose position in the room varied to accommodate all the locations that a worker might occupy in the room. This range of calculated values was then used to provide the maximum, mean, and minimum values seen in the table.

Table 5.19. $C_{\gamma\,rec}$, Dosimeter conversion factors for photon energy and angular response

| Group No. | Facility | Production site | Occupations | Time | $C_{\gamma\,rec}$ Minimum | $C_{\gamma\,rec}$ Point Estimate | $C_{\gamma\,rec}$ Maximum |
|---|---|---|---|---|---|---|---|
| 1.1 | Radiochemical Plant | Head-end and intermediate facilities | Others | 1948-53 | 1.107 | 6.739 | 1.354 |
| | | | | 1954-60 | 1.081 | 4.862 | 1.204 |
| | | | | 1961-91 | 0.987 | 4.788 | 1.094 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 1.2 | | | Repair and laboratory workers, samplers | 1948-53 | 1.050 | 1.500 | 3.141 |
| | | | | 1954-60 | 0.670 | 0.760 | 0.929 |
| | | | | 1961-91 | 0.525 | 0.560 | 0.610 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 2.1 | Radiochemical Plant and Plutonium Production | Foot-end facilities | Operators | 1948-53 | 2.125 | 2.870 | 4.339 |
| | | | | 1954-60 | 1.306 | 1.440 | 1.678 |
| | | | | 1961-91 | 1.096 | 1.180 | 1.314 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 2.2 | | | Repair workers | 1948-53 | 1.563 | 3.600 | 7.662 |
| | | | | 1954-60 | 0.844 | 1.140 | 1.724 |
| | | | | 1961-91 | 0.516 | 0.560 | 0.634 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 2.3 | | | Storage workers | 1948-53 | 1.897 | 2.130 | 2.557 |
| | | | | 1954-60 | 0.913 | 0.970 | 1.048 |
| | | | | 1961-91 | 0.654 | 0.690 | 0.729 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 2.4 | | | Others | 1948-53 | 1.800 | 2.400 | 3.582 |
| | | | | 1954-60 | 1.011 | 1.100 | 1.251 |
| | | | | 1961-91 | 0.755 | 0.800 | 0.857 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 3.1 | Plutonium Plant | Radiochemical Areas | Operators | 1948-53 | 2.125 | 2.870 | 4.339 |
| | | | | 1954-60 | 1.306 | 1.440 | 1.678 |
| | | | | 1961-91 | 1.096 | 1.180 | 1.314 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 3.2 | | | Repair workers | 1948-53 | 1.563 | 3.600 | 7.662 |
| | | | | 1954-60 | 0.844 | 1.140 | 1.724 |
| | | | | 1961-91 | 0.516 | 0.560 | 0.634 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 3.3 | | | Batch workers, material accountants, warehouse workers | 1948-53 | 1.897 | 2.130 | 2.557 |
| | | | | 1954-60 | 0.913 | 0.970 | 1.048 |
| | | | | 1961-91 | 0.654 | 0.690 | 0.729 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 3.4 | | | Others | 1948-53 | 1.800 | 2.400 | 3.582 |
| | | | | 1954-60 | 1.011 | 1.100 | 1.251 |
| | | | | 1961-91 | 0.755 | 0.800 | 0.857 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 4.1 | Plutonium Plant | Chemical and Metallurgical Areas | Operators | 1948-53 | 1.178 | 2.000 | 3.635 |
| | | | | 1954-60 | 0.884 | 1.000 | 1.216 |
| | | | | 1961-91 | 0.652 | 0.700 | 0.773 |

| Group No. | Facility | Production site | Occupations | Time | $C_{\gamma\,rec}$ Minimum | $C_{\gamma\,rec}$ Point Estimate | $C_{\gamma\,rec}$ Maximum |
|---|---|---|---|---|---|---|---|
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 4.2 | | | Batch workers, material accountants, warehouse workers | 1948-53 | 0.714 | 1.900 | 4.266 |
| | | | | 1954-60 | 0.811 | 1.000 | 1.367 |
| | | | | 1961-91 | 0.633 | 0.700 | 0.819 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 4.3 | | | Other | 1948-53 | 1.527 | 1.900 | 2.628 |
| | | | | 1954-60 | 0.841 | 1.000 | 1.306 |
| | | | | 1961-91 | 0.641 | 0.700 | 0.800 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 5 | Reactor Plant | Central Hall | All | 1948-53 | 0.802 | 1.130 | 1.779 |
| | | | | 1954-60 | 0.832 | 0.880 | 0.939 |
| | | | | 1961-91 | 0.724 | 0.770 | 0.834 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 6 | | Reactor Auxiliary Areas | All | 1948-53 | 0.743 | 1.110 | 1.837 |
| | | | | 1954-60 | 0.724 | 0.770 | 0.834 |
| | | | | 1961-91 | 0.536 | 0.590 | 0.685 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 7 | | Mix of Groups 5 and 6 | All | 1948-53 | 0.763 | 1.120 | 1.827 |
| | | | | 1954-60 | 0.753 | 0.800 | 0.863 |
| | | | | 1961-91 | 0.588 | 0.640 | 0.727 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 8 | Other | | | 1948-53 | 1.503 | 1.692 | 2.041 |
| | | | | 1954-60 | 0.748 | 0.842 | 1.016 |
| | | | | 1961-91 | 0.495 | 0.557 | 0.672 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |

For each worker, each annual dose in the database is associated with a dosimeter type and an exposure scenario. The conversion factor can be thus be selected from Table 5.19 for the appropriate dosimeter type and scenario. The mean value from the table would be applied to $D_{\gamma\,dos}$ to get $D_{\gamma\,rec}$.

### 5.9.2.2 Adjustments to Recorded Photon Dose for Nonroutine Scenarios

Nonroutine scenarios had radiation fields generated by freshly-irradiated reactor fuel. Determining the dose in air from $D_{\gamma\,arch}$ requires the same corrections for the dosimeter's photon energy and angular response that is used for routine scenarios. In addition to these photon corrections, however, beta radiation must also be considered in nonroutine scenarios.

Irradiated fuel emits high-energy betas from short-lived fission products, so the intensity and spectrum change with time as these fission products decay. Fig. 5.7 shows how the betas decrease with time after removal from the reactor (this time is often called the "decay time").

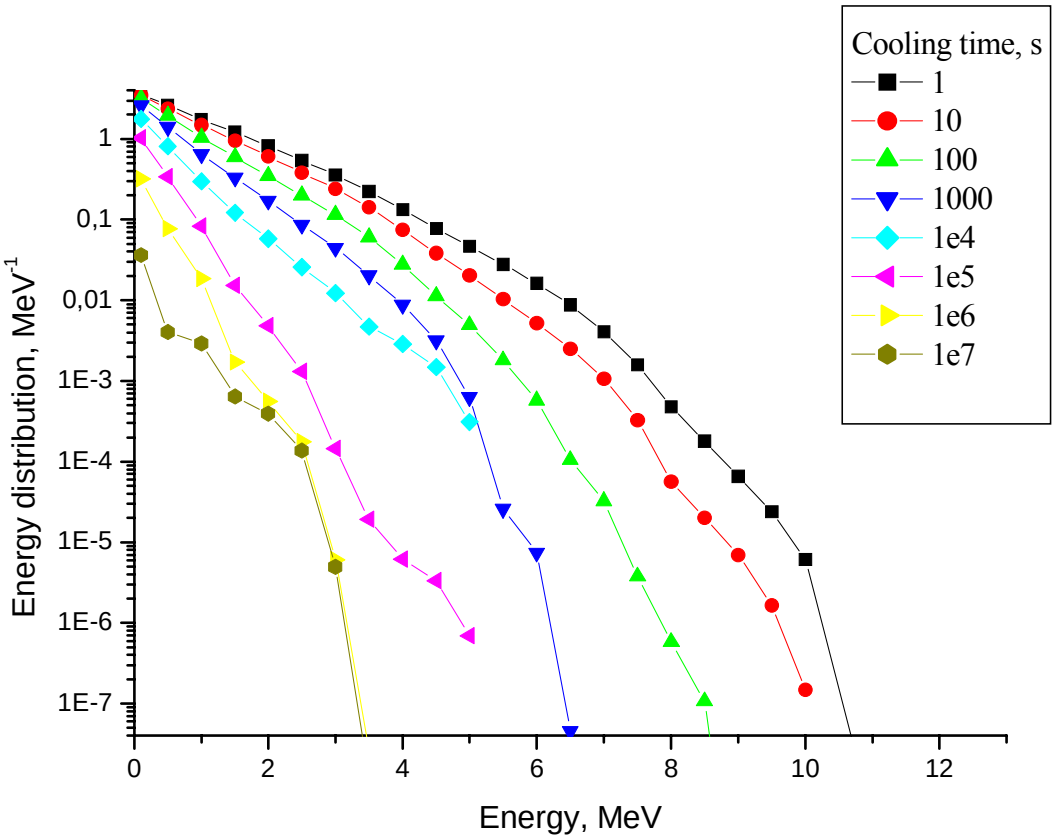



Fig. 5.7. Beta spectra from irradiated reactor fuel

Fig. 5.7 shows that immediately after removal from the reactor there is a substantial amount of high-energy beta radiation that could affect the IFK dosimeter response. With a decay of 1 d (about $10^5$ sec), there are still betas present with energies over 2 MeV, but 11 d (about $10^6$ sec) and 115 d (about $10^7$ sec) there are fewer remaining.

Both nonroutine scenarios have irradiated fuel as the radiation source. In the reactor scenario, however, the fuel elements are intact and clad in aluminum; the self-shielding of the fuel and the aluminum coating prevent most betas from irradiating a worker. In the radiochemical plant nonroutine scenario, however, the fuel has dissolved and betas do not encounter the shielding of the fuel elements. In an analysis of possible correction factors, the emission of the betas was found to depend on the thickness of the film of water covering the fuel particles; a thickness of 1 mm of water was found to be a reasonable attenuator. Using this assumption, a correction of 10% was developed for beta response of IFK dosimeters in the radiochemical plant nonroutine scenario. This beta correction is subtracted from $D_{\gamma arch}$ before the photon corrections are applied.

### 5.9.3 Photon Personal Dose Equivalent, $H_p(10)_\gamma$

The photon personal dose equivalent, $H_p(10)_\gamma$ is an operational quantity defined as the dose equivalent delivered at a 10-mm depth of tissue. This quantity is similar to the older concept of deep dose. The

quantity of dose equivalent differs from the quantity of absorbed dose, because it includes a quality factor. For photon radiation, however, the quality factor is 1.0, so the dose equivalent, in mSv, is numerically equivalent to the dose, in mGy. $H_p(10)_\gamma$ is derived from $D_{\gamma\ dos}$ as shown in Equation 5-4, in the same way that $D_{\gamma\ rec}$ was calculated, but including the effect of radiation attenuation by 10 mm of tissue.

$$H_p(10)_\gamma = D_{\gamma\ dos} / C_{\gamma\ Hp} \qquad (5\text{-}4)$$

where $C_{\gamma\ Hp}$ is a conversion factor that accounts for the energy distribution and the exposure geometry of the worker exposure scenario and also accounts for attenuation by 10 mm of tissue. These conversion factors were derived from calculations that were similar to those used to derive the data in Table 5.16. These conversion factors are listed in Table 5.20.

Table 5.20. $C_{\gamma\ Hp}$, Dosimeter conversion factors to determine $H_p(10)_\gamma$

| Group No. | Facility | Production site | Occupations | Time | $C_{\gamma\ rec}$ Minimum | $C_{\gamma\ rec}$ Point Estimate | $C_{\gamma\ rec}$ Maximum |
|---|---|---|---|---|---|---|---|
| 1.1 | Radiochemical Plant | Head-End and Intermediate facilities | Others | 1948-53 | 1.107 | 6.739 | 1.354 |
| | | | | 1954-60 | 1.081 | 4.862 | 1.204 |
| | | | | 1961-91 | 0.987 | 4.788 | 1.094 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 1.2 | | | Repair & laboratory workers, samplers | 1948-53 | 1.050 | 1.500 | 3.141 |
| | | | | 1954-60 | 0.670 | 0.760 | 0.929 |
| | | | | 1961-91 | 0.525 | 0.560 | 0.610 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 2.1 | Radiochemical Plant and Plutonium Production | Foot-end facilities | Operators | 1948-53 | 2.125 | 2.870 | 4.339 |
| | | | | 1954-60 | 1.306 | 1.440 | 1.678 |
| | | | | 1961-91 | 1.096 | 1.180 | 1.314 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 2.2 | | | Repair workers | 1948-53 | 1.563 | 3.600 | 7.662 |
| | | | | 1954-60 | 0.844 | 1.140 | 1.724 |
| | | | | 1961-91 | 0.516 | 0.560 | 0.634 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 2.3 | | | Storage workers | 1948-53 | 1.897 | 2.130 | 2.557 |
| | | | | 1954-60 | 0.913 | 0.970 | 1.048 |
| | | | | 1961-91 | 0.654 | 0.690 | 0.729 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 2.4 | | | Others | 1948-53 | 1.800 | 2.400 | 3.582 |
| | | | | 1954-60 | 1.011 | 1.100 | 1.251 |
| | | | | 1961-91 | 0.755 | 0.800 | 0.857 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 3.1 | Plutonium Plant | Radiochemical Areas | Operators | 1948-53 | 2.125 | 2.870 | 4.339 |
| | | | | 1954-60 | 1.306 | 1.440 | 1.678 |
| | | | | 1961-91 | 1.096 | 1.180 | 1.314 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 3.2 | | | Repair workers | 1948-53 | 1.563 | 3.600 | 7.662 |
| | | | | 1954-60 | 0.844 | 1.140 | 1.724 |
| | | | | 1961-91 | 0.516 | 0.560 | 0.634 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 3.3 | | | Batch workers, material accountants, warehouse workers | 1948-53 | 1.897 | 2.130 | 2.557 |
| | | | | 1954-60 | 0.913 | 0.970 | 1.048 |
| | | | | 1961-91 | 0.654 | 0.690 | 0.729 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |

| Group No. | Facility | Production site | Occupations | Time | $C_{\gamma\,rec}$ Minimum | $C_{\gamma\,rec}$ Point Estimate | $C_{\gamma\,rec}$ Maximum |
|---|---|---|---|---|---|---|---|
| 3.4 | | | Others | 1948-53 | 1.800 | 2.400 | 3.582 |
| | | | | 1954-60 | 1.011 | 1.100 | 1.251 |
| | | | | 1961-91 | 0.755 | 0.800 | 0.857 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 4.1 | Plutonium Plant | Chemical and Metallurgical Areas | Operators | 1948-53 | 1.178 | 2.000 | 3.635 |
| | | | | 1954-60 | 0.884 | 1.000 | 1.216 |
| | | | | 1961-91 | 0.652 | 0.700 | 0.773 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 4.2 | | | Batch workers, material accountants, warehouse workers | 1948-53 | 0.714 | 1.900 | 4.266 |
| | | | | 1954-60 | 0.811 | 1.000 | 1.367 |
| | | | | 1961-91 | 0.633 | 0.700 | 0.819 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 4.3 | | | Other | 1948-53 | 1.527 | 1.900 | 2.628 |
| | | | | 1954-60 | 0.841 | 1.000 | 1.306 |
| | | | | 1961-91 | 0.641 | 0.700 | 0.800 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 5 | Reactor Plant | Central Hall | All | 1948-53 | 0.802 | 1.130 | 1.779 |
| | | | | 1954-60 | 0.832 | 0.880 | 0.939 |
| | | | | 1961-91 | 0.724 | 0.770 | 0.834 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 6 | | Reactor Auxiliary Areas | All | 1948-53 | 0.743 | 1.110 | 1.837 |
| | | | | 1954-60 | 0.724 | 0.770 | 0.834 |
| | | | | 1961-91 | 0.536 | 0.590 | 0.685 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 7 | | Mix of Groups #5 and #6 | All | 1948-53 | 0.763 | 1.120 | 1.827 |
| | | | | 1954-60 | 0.753 | 0.800 | 0.863 |
| | | | | 1961-91 | 0.588 | 0.640 | 0.727 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |
| 8 | Other | | | 1948-53 | 1.503 | 1.692 | 2.041 |
| | | | | 1954-60 | 0.748 | 0.842 | 1.016 |
| | | | | 1961-91 | 0.495 | 0.557 | 0.672 |
| | | | | 1992-05 | 0.888 | 1.000 | 1.206 |

### 5.9.4 Neutron Personal Dose Equivalent, $H_p(10)_n$

The neutron personal dose equivalent, $H_p(10)_n$ is an operational quantity, very similar to $H_p(10)_\gamma$, also defined as the dose equivalent delivered at a 10-mm depth of tissue. $H_p(10)_n$ values in the Doses-2005 database were derived from $D_{\gamma\,dos}$ as shown in Equation 5-5.

$$H_p(10)_n = K_n \cdot D_{\gamma\,dos} \cdot C_{pn} \qquad (5\text{-}5)$$

where $K_n = D_n/D_\gamma$, the ratio of neutron dose to gamma dose, and $C_{pn}$, conversion factor that includes the effect of radiation attenuation at a 10-mm depth of tissue for different exposure geometries and different spectra (units of mSv). The numerator of the neutron-to-gamma dose ratio $K_n$ is the "equivalent dose," as defined for Russian nuclear facilities. This term is defined as the absorbed dose at the surface of the body resulting from the neutron field, multiplied by a quality factor appropriate for the neutron energy distribution. The quality factors used in this quantity are presented in Volume II, Section 5.5. The denominator of the ratio was gamma dose. $K_n$ values were evaluated using measurements wherever measurement data were available. For cases where measurement data were not available, calculations of the neutron exposure conditions were used. These calculations used an evaluation of the neutron source

term, and energy and spatial characteristics of the neutron radiation field, to determine the neutron equivalent dose to workers exposed by this scenario. Where measurements were available for these ratios, the gamma dose was typically derived from dosimeter readings. When calculations were used, the gamma dose in air was used.

Because the $K_n$ values have both gamma and neutron components, the photon field must be evaluated for each exposure condition along with the neutron field. Thus it was necessary to derive the ratios in three different periods, because the photon intensities changed over time in a way that neutron intensities did not. In the earliest days of plutonium production, the product had a higher level of gamma-emitting impurities than were present after the processes were further developed. Thus the photon fields were higher in the earlier days, but the neutron fields were relatively constant. Table 5.21 presents the $K_n$ and $C_{pn}$ values used in this study.

### 5.9.5 Photon Organ Dose

An important objective of Project 2.4 dosimetry tasks is to provide data for epidemiological analyses, primarily including absorbed dose estimates suitable for assessment of cancer risks in the Mayak worker cohort. The radiation quantity most useful for estimating an exposed worker's cancer risk is the absorbed dose to individual organs. Health physicists are accustomed to working with dose equivalent rather than absorbed dose, so it might seem surprising that the final results of this database would not be dose equivalent. However, while the concept of dose equivalent is most useful for making radiation protection decisions, epidemiologists do not like the additional complication of a quality factor that is embedded in the dose equivalent. Thus, absorbed dose is used rather than dose equivalent for this database. The methodology for calculating organ absorbed doses from external photon radiation is generally based on ICRP Publication 74 (ICRP 1996).

To find the organ doses received by a worker in a given year, the dose to air, $D_{\gamma\,rec}$, is multiplied by conversion factors derived for the specific exposure scenario. Separate conversion factors are derived for each of the organs in the database. The organ DCFs are calculated either by using publicly-available DCFs, or by MCNP calculations specific to the Mayak scenario. Equation 5-6 shows how $D_{\gamma\,org\text{-}i}$ values are calculated.

$$D_{\gamma\,org\text{-}i} = C_{\gamma\,org\text{-}i} \cdot D_{\gamma\,rec} \qquad (5\text{-}6)$$

where $C_{\gamma\,org\text{-}i}$ is a conversion factor that accounts for the energy distribution and the exposure geometry of the worker exposure scenario, and converts absorbed dose in air to absorbed dose to organ *i*. The remainder of this section describes how these conversion factors were determined for this study.

Table 5.21. $K_n$ factor value depending on production area for selected personnel groups

| No. | Exposure facility | Monitoring period | Maximal gamma dose, rad | Exposure geometry | $K_n = Dn/D\gamma$ | $C_{pn}$ |
|---|---|---|---|---|---|---|
| 1 "N" | Radiochemical Plant – Foot-End Facilities | Before 1954 | N/A[(a)] | 50% AP 50% isotropic | 0.076 | 0.76 |
| | | 1954-1961 | N/A[(a)] | | 0.198 | 0.76 |
| | | After 1961 | N/A[(a)] | | 0.205 | 0.76 |
| 2 "N" | Radiochemical Plant – Foot-End Facilities | Before 1954 | N/A[(a)] | 50% from beneath 50% isotropic | 0.205 | 0.69 |
| | | 1954-1961 | N/A[(a)] | | 0.104 | 0.69 |
| | | After 1961 | N/A[(a)] | | 0.269 | 0.69 |
| 3 "N" | Plutonium Plant – Radiochemical Facility | Before 1954 | N/A[(a)] | Isotropic | 0.276 | 0.69 |
| | | 1954-1961 | N/A[(a)] | | 0.276 | 0.69 |
| | | After 1961 | N/A[(a)] | | 0.069 | 0.69 |
| 4 "N" | Plutonium Plant – Chemical-Metallurgical | Before 1954 | N/A[(a)] | 50% AP 50% isotropic | 0.173 | 0.76 |
| | | 1954-1961 | N/A[(a)] | | 0.186 | 0.76 |
| | | After 1961 | N/A[(a)] | | 0.186 | 0.76 |
| 5 "N" | Plutonium Plant – Chemical-Metallurgical | Before 1954 | N/A[(a)] | Isotropic | 0.144 | 0.69 |
| | | 1954-1961 | N/A[(a)] | | 0.319 | 0.69 |
| | | After 1961 | N/A[(a)] | | 0.327 | 0.69 |
| 6 "N" | Reactor – Central Hall | 1948-1954 | 60.0 | From beneath | 0.011 | 0.53 |
| | | 1955-1961 | 20.0 | | 0.032 | 0.53 |
| | | 1962-1967 | 15.0 | | 0.080 | 0.53 |
| | | 1968-1973 | 8.0 | | 0.106 | 0.53 |
| | | 1974-1985 | 7.0 | | 0.080 | 0.53 |
| | | 1986 – present | 4.0 | | 0.096 | 0.53 |
| 7 "N" | Reactor – 30% Central Hall; 70% Auxiliary | 1948-1954 | 60.0 | From beneath during work at CH | 0.003 | 0.53 |
| | | 1955-1961 | 20.0 | | 0.01 | 0.53 |
| | | 1962-1967 | 15.0 | | 0.024 | 0.53 |
| | | 1968-1973 | 8.0 | | 0.032 | 0.53 |
| | | 1974-1985 | 7.0 | | 0.024 | 0.53 |
| | | 1986 – present | 4.0 | | 0.029 | 0.53 |

(a) If a worker's gamma dose exceeds the maximal gamma dose, the maximum gamma dose should be used to find the corresponding neutron dose. An entry of "N/A" indicates that no maximal dose is applied for these scenarios.

The Project 2.4 database contains organ doses only for photon radiation. For beta radiation, there were never sufficient numbers of high-energy betas to reach deeper organs. For neutron radiation, the quantities useful to epidemiologists would be absorbed dose to organs, with separate contributions for high- and low-LET radiation. The ICRP (1996) compilation of organ doses presents neutron absorbed dose to organs, but there was no attempt to separate high- and low-LET components. Other compilations for neutrons present dose equivalent to organs. Calculating separate LET components of neutron absorbed dose to organs would require a large calculation effort, which would not be reasonable given the small contribution of neutrons for the total absorbed dose to organs in these scenarios.

### 5.9.6 Calculation of Organ Dose Conversion Factors

For each exposure scenario, the organ DCFs are calculated using the following information:

- Exposure geometry

- Photon energy spectrum

The organ dose calculation requires a description of the radiation field for each exposure condition. The energy distribution of the gamma radiation is specified as a photon spectrum, which is a set of photon energy bins with associated photon intensities. To describe the orientation, a standard irradiation geometry can be specified if dose factors are to be used for the calculation. If a phantom calculation is to be performed, the irradiation geometry must be specified for the calculation in a way that can be included in an input file for MCNP.

For adjusting the archive dosimeter data, the dosimeter response is corrected for photon radiation energy – and angular – response, and for the beta radiation response in nonroutine scenarios. These dosimeter response adjustments are based on results from the laboratory dosimeter response studies using the exposure scenarios and mathematical modeling of the dosimeter emulsion response relative to the original registered dose. The corrected dosimeter response is then converted to absorbed dose in air at the dosimeter location (i.e., worker location) and then converted to a set of organ doses using the factors calculated in the exposure scenarios.

In each case, a set of organ doses will be calculated by multiplying the dose to air by a set of scenario-specific organ DCFs.

### 5.9.7 Exposure Geometries

The ICRP Publication 74 (ICRP 1996) and the GSF (Zankl et al, 1997) DCF sets were calculated for individual organs exposed to monoenergetic photons in a limited set of exposure conditions, including:

- AP: a parallel beam of radiation incident normally on the anterior surface of the body
- Posterior-anterior (PA): a parallel beam of radiation incident normally on the posterior surface of the body
- Lateral (LAT): a parallel beam of radiation incident normally on a lateral surface of the body, perpendicular to an AP beam. The geometry may be RLAT or LLAT for radiation incident on either the right lateral surface or the left lateral surface, if the dose to the organ is not identical for the two cases
- ROT: radiation directed towards the central vertical axis of the body, equal considering all angles of incidence
- ISO: isotropic radiation

If the scenario has an exposure geometry that matches one in the ICRP Publication 74 list (ICRP 1996), the scenario organ conversion factors can be found by multiplying the photon energy spectrum by the conversion factors in the ICRP and GSF tables. If the exposure geometry is not found in the ICRP Publication 74 list, the conversion factor must be found by a calculation using an MCNP model.

### 5.9.8 MCNP Calculations

For MCNP calculations, an anthropomorphic phantom is used to simulate radiation transport in the human body and determine absorbed dose in individual organs. A male and a female phantom are used separately in the calculations, to evaluate doses for male and female workers. The phantom models are MIRD-type phantoms, with equations taken from (Cristy and Eckerman, 1987). This approach makes the calculations compatible with the values reported in ICRP Publication 74 (ICRP 1996).

For each scenario, two MCNP calculations are performed with phantoms in place (one male, one female). Then another calculation is made a cylinder of air in place of the phantom. The air cylinder has a diameter of 30 cm and is 60 cm high, positioned where the phantom's torso had been. The deposition of energy by photons in this air cylinder is then the dose to air, so the ratio of the calculated organ dose to this dose in air is the organ DCF equivalent to the ICRP Publication 74 value (ICRP 1996).

### 5.9.9 Organs

The organs for which a photon radiation absorbed dose will be included in Doses-2005 are listed in Table 5.22.

Table 5.22. Organs to calculate absorbed dose under Project 2.4

| Organ[(a)] | Gender | Organ[(a)] | Gender |
|---|---|---|---|
| Brain | Both | Lung | Both |
| Stomach | Both | Endosteal tissue (bone surface) | Both |
| Lower large intestine | Both | Uterus | Female |
| Thyroid | Both | Bladder (urinary) | Both |
| Red marrow | Both | Liver | Both |
| Ovaries | Female | Colon | Both |
| Breast | Female | Esophagus | Both |
| Kidney | Both | Skin | Both |
| Small intestine | Both | Testes | Male |

(a) Notes:

- Gall bladder would be an organ of interest to epidemiologists, but no data exist for this organ in either the GSF or ICRP Publication 74 compilations.
- Esophagus: dose factor data are derived from ICRP Publication 74, but the organ does not exist in the phantom model used in MCNP calculations for nonstandard irradiation geometries.
- Because the following organs are not included in the ICRP Publication 74 compilation, the data will be extracted from the GSF compilation: brain, kidney, small intestine, lower large intestine

The GSF compilation lists conversion factors separately for male and female. Some organs obviously apply to only male or female workers, such as the testes, ovaries, and female breast. However, in the GSF compilation, calculations were performed using two different MIRD-type phantoms, one male and one female. The ICRP Publication 74 compilation was also based on calculations in both male and female phantoms; however, it listed only one value for each organ (ICRP 1996). This single value was based on an average of the organ doses calculated in the male and female phantoms. Thus, this study will report only one dose for each organ, which will be applicable to both male and female workers for all organs except ovaries, breast, uterus and testes, which will be used only for male or for female workers as listed in Table 5.22.

For conversion factors calculated using MCNP and the phantom models, organ doses are calculated in both male and female phantoms. They are reported in a manner compatible with the ICRP Publication 74

methodology, using an average of the male and female results for organs applicable to both sexes (ICRP 1996).

Organ DCFs applicable to the routine and nonroutine exposure scenarios are listed in Volume II, Tables 5.5 through 5.12.

### 5.9.10 Uncertainty in Photon and Neutron Dose

A complete estimate of the uncertainty associated with the dose evaluations discussed above, for MPA workers wearing a film dosimeter, would account for the following components:

- Uncertainties in personnel dosimeter readout
    - Errors in control dosimeter calibrations and readout
    - Temperature effects while the dosimeter was worn by the worker and during the processing of the film
    - Variations in the delay period between radiation exposure and readout of the film
    - Variations in the chemicals used to process the film
    - Calibration of the film density reader
    - Variations in duration of film processing steps
    - Variations in other environmental conditions of dosimeter exposure compared to calibration

- Other uncertainties in dosimeter exposure
    - Location of the dosimeter on a worker's body
    - Dosimeter exposed to a radiation field different than radiation exposing worker's body (e.g., partial shielding of dosimeter by a small object)
    - Radiation field exposing worker is different than calibration field (energy spectrum, exposure geometry)

- Development of corrections to dosimeter readings
    - Accuracy of modeling assumptions
    - Experimental errors during characterization irradiations at GSF and a linear accelerator
    - Assumptions of exposure scenarios representing all exposure conditions

- Estimation of dose in air, $H_p(10)_\gamma$, $H_p(10)_n$, and organ doses
    - Realistic assumptions behind the conversion factors compared to the actual worker exposure conditions
    - Differences between the standard phantom used for the conversion factors and the worker's body

Many sources of uncertainty in this list cannot be described quantitatively, but should be acknowledged as possible sources of uncertainty.

#### 5.9.10.1 Dose Estimation Uncertainties from Individual Dosimeter Readings

The uncertainty in estimating the dose in air from individual dosimeter readings, following Equations 5-2 and 5-3, includes an error of dosimeter readout and an uncertainty in the course of dosimeter exposure.

Errors of dose estimation with individual dosimeters are random ones, but the errors in the course of dosimeter exposure appear as systematic errors.

Studying the dosimetry equipment and procedures has determined that the standard deviation of a single measurement for all three dosimeter types is ±30%. This error represents only the random component of the uncertainty.

Investigations during Project 2.4 demonstrated that the random error could be reduced by properly accounting for:

- The number of measurements constituting the annual dose;
- The monitoring period and dosimeter type; and
- Worker occupation.

The last two parameters define the variability of the occupational history and its impact on the dosimeter readings. The impact of these parameters on the evaluated dose was studied using Bayesian statistics, giving the relation between the standard deviations of the annual doses and the number of measurements during a year shown in Fig. 5.8.

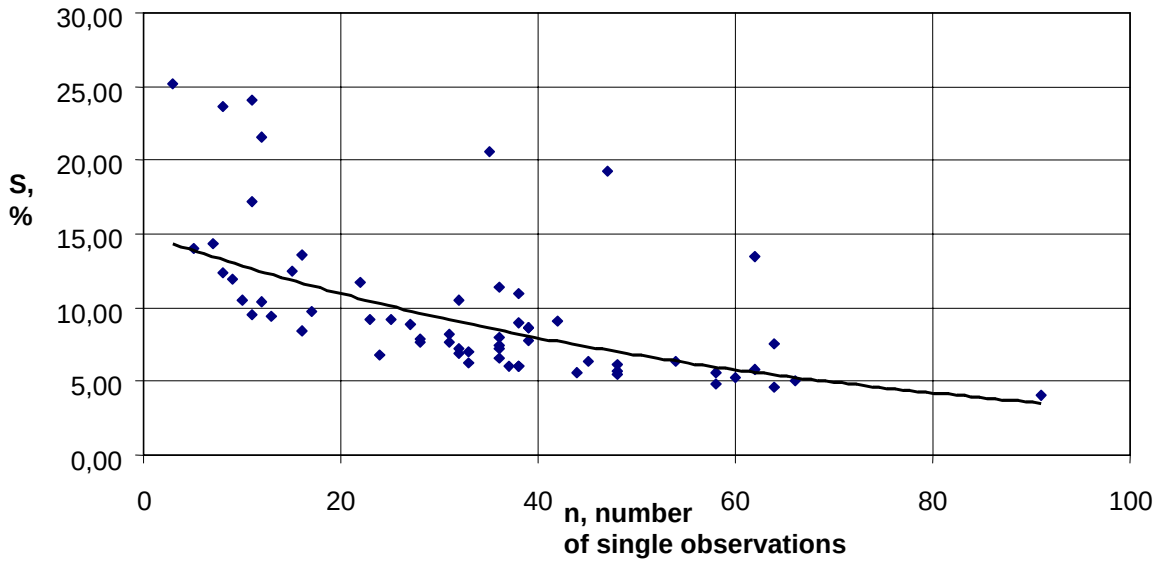



Fig. 5.8. Relation between annual gamma-dose error and number of measurements during a year

The analysis of the error in the annual dose for certain worker groups demonstrated that the random error for workers with uniform samples of individual absorbed doses ranges from 5% to 15% when the standard deviation for a single measurement is 30%. For the case of highly nonuniform samples of single measurements, when the annual dose consists of only one or two individual measurements, the error of the annual dose increases to 25%.

Assuming that the annual dose consists of 10 single measurements with individual dosimeters (monthly monitoring and 2 mo holiday), the random error of the annual dose varied from 4% to 25%

(measurements with IFK dosimeters at the foot-end nodes of radiochemical and plutonium production) during 1948–1974. The systematic error of the annual doses varied from 13% to 27.1%.

The error of the total dose to a worker with 6 yr or more of monitoring is influenced mainly by the systematic component.

Based on these investigations, tables were calculated to assess the error of the annual dose depending on the dosimeter type and the monitoring period as illustrated in Table 5.23, assuming that the annual dose consisted of 10 single measurements with individual dosimeters. The random component of this error could be further reduced by accounting for the number of individual dosimeter measurements used to determine the annual dose from the daily dose database and for the influence of the worker occupation on the random error.

Table 5.23. Interval estimates of the random error of the annual doses with 0.99 confidence. Equations are represented for various time periods, approximating the dependence of the $\psi_{low}$ and $\psi_{upper}$ limits of the confidence interval on the number of single measurements during a year.

| Period | Dosimeter type | Lower limit, $\psi_{low}$ (%) | Upper limit, $\psi_{upper}$ (%) |
|---|---|---|---|
| 1948–1953 | IFK | exp(3.30-0.0087N) | exp(3.53-0.0090N) |
| 1954 | IFK+Pb | exp(3.27-0.0137N) | exp(3.50-0.0168N) |
| 1955–1960 | IFK+Pb | exp(3.27-0.0128N) | exp(3.50-0.0152N) |
| 1961 | IFKU | exp(3.27-0.0230N) | exp(3.61-0.0270N) |
| 1962–1967 | IFKU | exp(3.57-0.0380N) | exp(3.96-0.0550N) |
| 1968–1990 | IFKU | exp(3.58-0.0410N) | exp(3.84-0.0466N) |

Note: N=10

The value of the measured dose $D_{\gamma dos}$ in Doses-2005 corresponds to the dosimeter measurements summed during a year. The error value of the point estimate of $D_{\gamma dos}$ represents a 99% confidence interval. To facilitate use by epidemiologists, the errors are presented not in percents, but in the form of the minimum value (low limit of the 99% confidence interval) and the maximum value (upper limit of the 99% confidence interval).

Errors were calculated with this method for routine dose values measured with individual dosimeters. 99% confidence intervals of estimated annual gamma doses resulting from routine exposure varied from ±60% (for 1948–1953) to ±29% (since 1985).

Nonroutine dose errors were estimated using the assumption that the dose consisted of individual measurements with ±30% error for each. The total error value was determined using Equation 5-7.

$$\delta = \sqrt{\delta_1^2 + \delta_2{}^2 + \ldots + \delta_n^2} \tag{5-7}$$

A point estimate of the neutron dose for nonroutine exposures is based on the sum of individual dosimeter readings measured for a certain worker in the course of all nonroutine exposures during a year. The maximum value corresponds to the dose value $\sum D_{non-routine} + \delta$, and the minimum is $\sum D_{non-routine} - \delta$.

#### 5.9.10.2 Uncertainty in Dose Conversions

The difference between the radiation field used during dosimeter calibration and the actual field affecting the person during work at a production area produces the primary contribution to the uncertainty in dose estimation from dosimeter readings. These uncertainties are systematic and are caused by spectral and angular dosimeter response characteristics, as applied to the photon spectrum and exposure geometry under the real conditions of the work. Photon spectra at workplaces are not constant, but vary by location in a room, area or area group. Thus, the uncertainty of the dose estimate can vary for the same dosimeter because the person moves in a workplace.

To reduce the uncertainties in the dose values recorded in Doses-2005, the adjusted values measured with individual dosimeters, $D_{\gamma dos}$, are corrected. This correction allows a reduction in the systematic dose uncertainty resulting from the dosimeter and radiation field characteristic mentioned above.

As described in the paragraphs above, the values of the conversion factors $C_{\gamma\,rec}$ take into account the dosimeter calibration conditions (calibration exposure geometry and source type) and the workplace exposure conditions (photon spectra at the workplace and variations in the room, area or area group; spectrum and angle dosimeter characteristics, and worker exposure geometry). The conversion factor includes a systematic uncertainty resulting from the difference between the dosimeter calibration conditions and exposure in realistic radiation fields. Applying the conversion factor to $D_{\gamma dos}$ excludes this systematic error. But the conversion factor is not constant; it varies in a room, as mentioned above, depending on the photon spectrum distribution in the room, area or area group, so that $C_{min} < C < C_{max}$.

During the calculation of conversion factors, factors are derived for each dosimeter type for the exposure conditions typical of the workplaces, for each exposure group. These conditions are characterized by different source locations, biological shielding, and distribution of worker locations in a room. In one study of the variability of conversion factors, a large number of MCNP calculations were performed, all applicable to exposure scenario 1.1 for a single dosimeter type, but varying critical modeling parameters: workplace dimensions; shield thickness and material composition; source dimensions, energy characteristics and location; and worker orientation in the workplace. A total of 553 individual MCNP calculations were performed in this study, and the conversion factor $C_{\gamma\,rec}$ was derived from each calculation. The minimum value of $C_{\gamma\,rec}$ was found to be 1.045, and the maximum value was 3.13. The arithmetic mean was found to be 1.57.

A lognormal distribution was found to approximate the frequency distribution of these conversion factors, with a geometric mean of 1.53 and a geometric standard deviation of 1.26. The minimum and maximum values from the set of 553 values were similar to the 99% confidence interval.

Similar conversion factors were calculated for each exposure scenario. The relative uncertainty derived for Scenario 1.1 with the IFK dosimeter was used to find the $C_{min}$ and $C_{max}$ values for the other scenario conversion factors. The same method was applied to conversion factors for $H_p(10)_\gamma$ .

When estimating the individual dose in the database Doses-2005, the uncertainty in the conversion factor is added to the dose measurement error of the individual dosimeter, determined from each dosimeter result, as shown in Equation 5-8:.

$$\delta_{total} = \sqrt{\delta^2 + \delta^2_{syst.non-except.}} \quad (5\text{-}8)$$

So, the value $D_{\gamma rec} = \overline{C}\ D_{\gamma dos}$ was the point estimate of the reconstructed annual dose. The lower limit of the reconstructed annual dose was determined as $D_{rec\ min} = C_{min} \cdot D_{\gamma dos\ min}$, and the upper limit as $D_{\gamma rec\ max} = C_{max} \cdot D_{\gamma dos\ max}$.

#### 5.9.10.3 Uncertainty in Organ Absorbed Dose

The organ absorbed dose to organ *i*, $D_{\gamma org\text{-}i}$, is calculated from the reconstructed dose in air $D_{\gamma rec}$ as shown in Equation 5-6. $C_{\gamma org\text{-}i}$ is calculated from standard compiled factors or MCNP calculations for the given photon spectrum and exposure scenario. Phantom parameters of a reference human are used for calculations.

$D_{\gamma org\text{-}i}$ uncertainties consist of the uncertainty in the estimation of $D_{\gamma rec}$ and the uncertainty in the calculation of $C_{\gamma org\text{-}i}$, which consists generally of the calculation model uncertainty and of differences between factors $C_{\gamma org\text{-}i}$ for a reference human and a person with individual anthropometric data. $C_{\gamma org\text{-}i}$ uncertainties were not available during Phase III of this project. Two possible approaches for uncertainty estimation are:

- Estimate $C_{\gamma org\text{-}i}$ factor uncertainties using a standard phantom, and phantoms with dimensions chosen to produce extreme values for organ doses, then combining the resulting extremes in organ DCFs with the $D_{\gamma rec}$ uncertainties;

- Use the anthropometric characteristics of a number of individual Mayak workers, and calculate organ doses for all the individuals. Then evaluate the statistical distribution for these values. For this approach it would be necessary to enter the information on height and weight and their variations with time of each cohort member studied into the database from medical cards stored in SUBI.

The value of these approaches needs to be studied further.

Organ dose uncertainty in the database Doses-2005 is currently determined only by the estimation of uncertainty in the reconstructed absorbed dose in air $D_{\gamma rec}$. Thus, the value $D_{\gamma org\text{-}i} = C_{\gamma org\text{-}i} \cdot D_{\gamma rec}$ is taken as the point estimate of the absorbed dose value in organ *i*.

The lower organ dose limit is determined as $D_{\gamma org\text{-}i\ min} = C_{\gamma org\text{-}i} \cdot D_{\gamma rec.\ min}$, and the upper limit as $D_{\gamma org\text{-}i\ max} = C_{\gamma org\text{-}i} \cdot D_{\gamma rec.\ max}$.

#### 5.9.10.4 Uncertainty in Neutron Dose Equivalent

The method of calculating neutron personal dose equivalent presented in Section 5.9.4 can also be thought of as a two-step process, where the first step is to calculate the individual neutron equivalent dose, as shown in Equation 5-9:

$$H_{n\,ind} = K_n \cdot D_{\gamma dos} \tag{5-9}$$

where $K_n = \frac{H_{n\,area}}{D_{\gamma area}}$ is the neutron-to-gamma ratio, found by dividing the average neutron equivalent dose in a certain area by the average absorbed gamma dose in the same area. The factor $K_n$ was usually obtained from area monitoring results; for times when neutron monitoring was not performed, it could be determined by calculations.

The individual equivalent dose, $H_{n\,ind}$, is then converted into the neutron personal dose equivalent $H_p(10)_n$ as shown in Equation 5-10:

$$H_p(10)_n = C_{pn} \cdot H_{n\,ind} \tag{5-10}$$

where $C_{pn}$ is the conversion factor considering spectrum and angular neutron characteristics and irradiation geometry for the exposure scenario.

The neutron quantity $H_{n\,ind}$ is similar to the quantity $D_{\gamma dos}$ for photon radiation. As the neutron to gamma ratio, $K_n$, is calculated from the individual neutron dose $H_{n\,ind}$ and the measured gamma dose $D_{\gamma dos}$, the neutron dose uncertainty is calculated as the uncertainty of an indirect measurement, with components of both random and systematic errors:

$$\delta_{\sum} = \sqrt{\delta_{Kn}^2 + \delta_{D\gamma dos}^2} \tag{5-11}$$

where $\delta_{Kn}^2 = \delta_{Hn.area}^2 + \delta_{D\gamma area}^2$. The uncertainty of the factor $C_{pn}$ was not determined during Phase III of this project. The uncertainty in $C_{pn}$ could be calculated like the uncertainty in C, discussed in Section 5.9.10.2. The following value was taken as the point estimate of the neutron dose equivalent:
$Hp(10)_n = \overline{C}_{Hp(10)n} \cdot H_{n\,ind}$.

The lower limit was $H_p(10)_{n\,low} = \overline{C}_{pn\,min} \cdot H_{n\,ind\,min}$.
The upper limit was $H_p(10)_{n\,upper} = \overline{C}_{pn\,max} \cdot H_{n\,ind\,max}$.

# 6.0 Estimation of Plutonium Intake Organ Dose

The direct $^{239}$Pu measurement input data for the Doses-2005 model were obtained from the bioassay (urinalysis) measurements and/or from organ concentrations of plutonium derived from autopsy data. Other input parameters (detailed in Volume III of the Users Guide) include transportability ("solubility") coefficients of the various industrial aerosols, years of employment at the particular plant locations, smoking history, health issues that would affect plutonium distribution among organs (e.g. liver disease) and whether DTPA was used to enhance plutonium excretion for bioassay measurements.

The Doses-2005 model consists of a pulmonary model combined with a systemic model. Both of these components have been recently published and together they comprise the Doses-2005 model for calculations of $^{239}$Pu doses to the various organs. The structure of the entire model with details on the various transfer coefficients are presented in detail in Volume III and in (Khokhryakov et al. 2005 and Leggett et al. 2005).

# 7.0 Scoping Studies of Potential Dose from Other Sources of Occupational Exposure

In the development of the dosimetry protocol it was recognized that tasks were not being done to evaluate other potential significant pathways of exposure. As such, scoping studies were initiated to evaluate identified pathways including neutron radiation, internal intake other than plutonium, airborne effluent, and medical x-rays. These scoping studies were done to assess the potential significance to organ dose in relation to the reconstructed organ dose from occupational external radiation and plutonium intake. The results of these analyses are presented in the following sections.

## 7.1 NEUTRON DOSE

Neutron radiation was undoubtedly present in radiation fields at some MPA facilities and contributed to the worker dose. Monitoring the individual MPA worker exposure to neutron radiation was not done in the early years.

### 7.1.1 MPA Neutron Dosimetry Technology

The first activities to determine neutron doses were begun in reactor facilities in 1970. At that time, such activities were occasional and covered a small group of people. Actual routine monitoring of neutron doses for many MPA workers began in 1973 after development of the individual accidental dosimeter for neutrons (DINA)[3]. The DINA was the first individual dosimeter for neutron radiation manufactured on a wide scale. It took several years to refine manufacturing methods, make design improvements and conduct operational testing of this new dosimeter system. The DINA dosimeter was developed for measurement of accidental level doses. For this purpose its design combined two fundamentally different methods: activation and track methods. The dosimeter design is shown in Fig. 7.1.

The activation method is based on the rhodium foil and is intended for rapid evaluation of the neutron dose. The $^{103m}$Rh isomer activity, generated as a result of the $^{103}$Rh (n, $\gamma$) reaction, is measured. This nuclide decays with a half-life of 57 min and emits characteristic gamma radiation of about 20 keV. A single channel analyzer is used to count the gamma radiation.

The track method is based on using two detectors with a $^{237}$Np target between them. A $^{10}$B filter, 0.1 g/cm$^2$, is used to reduce the dose determination error due to the dependence of the $^{237}$Np fission cross-section on neutron energy. The boron filter is made by pressing a mix of natural boron powder with epoxy resin with subsequent polymerization at 130°C. For fission fragments recording, track detectors of silicate glass or an artificial mica are used.

The size of the detectors is 10 × 10 mm. The fissionable target is made from an alloy of neptunium (14%) and aluminum (86%). Track counting is performed using a microscope after preliminary etching in hydrofluoric acid.

[3] Developed by Biophysics Institute, Ministry of Health, USSR by I.B. Keirim-Markus, T.V. Korolyova, S.N.Kraytor, and L.N.Uspensky.

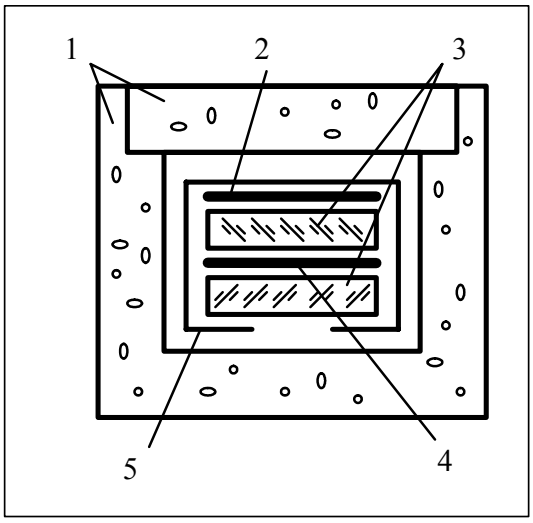



Fig. 7.1. DINA dosimeter design (1 - boron filter, 2 - rhodium foil, 3 - detectors, 4 - neptunium target, 5 - aluminum bracket).

Since 1979, the emergency dosimeter "GNEISS", which includes the DINA dosimeter, was introduced. The lower limit of detection for the DINA detectors was originally 5 rad and, thus, could not be realistically used for routine personnel monitoring. Improvements were made to these dosimeter systems by Mayak and the Moscow Institute of Biophysics, which reduced the lower limit of detection to 10 mrad. In addition, field measurements of the neutron spectrum in selected Mayak workplaces were done with the Bonner sphere spectrometer and spectrometric assembly "DISNEY." A primary observation of the field spectra measurements was that intermediate and fast neutrons were the primary contributors to worker neutron dose. It was also concluded that the neutron dose determination error does not exceed ±10% using the DINA dosimeter at different Mayak reactors with and without shields of water, lead and plastic, iron, and iron with polyethylene, both in air and on the phantom surface. Dose determination error due to angular dependence does not exceed ±3% relative to the angle of 45°. Since 1985, DINA has been used at MPA as the official dosimeter for routine measurement of neutron dose.

In addition to DINA, routine neutron radiation dose rate monitoring has been carried out since 1984 at the radiochemical plant and since 1986 at the plutonium plant using the RUS-8 universal scintillation radiometer. This instrument was designed for measurement of thermal, intermediate and fast neutron flux density, as well as neutron equivalent dose rate.

### 7.1.2 Neutron-to-Photon Dose Ratio

Neutron-to-photon dose ratios can be very useful for describing the significance of neutrons in personnel exposure. These ratios were derived for workers in the seven neutron exposure scenarios, which are discussed in Volume II Section 4.5. The neutron dose quantity used for this ratio was the "equivalent dose," as defined for Russian nuclear facilities. This term is defined as the absorbed dose at the surface of the body resulting from the neutron field, multiplied by a quality factor appropriate for the neutron energy distribution. The denominator of the ratio was gamma dose as measured by the dosimeter.

The neutron-to-photon dose ratios were evaluated using measurements wherever measurement data were available. For cases where measurement data were not available, calculations of the neutron exposure conditions were used. These calculations used an evaluation of the neutron source term and energy and spatial characteristics of the neutron radiation field, to determine the neutron equivalent dose to workers exposed by this scenario. The calculated neutron-to-photon dose ratios are presented in Table 5.16.

### 7.1.3 Significance of Neutron Exposures

The exposure information that is most useful for epidemiologists is the absorbed dose to individual organs. The neutron radiation absorbed dose to organs includes two components of absorbed dose: low- and high-LET; whereas for photon radiation, only low-LET absorbed doses contribute to the organ dose. The neutron organ conversion factors presented in ICRP Publication 74 convert neutron fluence into organ absorbed dose, but the tables do not provide data for neutron exposures that allow a differentiation between low- and high-LET organ doses (ICRP 1996). Determining neutron absorbed dose to organs differentiated into two LET components would have required extensive calculations that were outside the scope of the work leading to Doses-2005. Thus, the only worker exposure information included in the Doses-2005 external dose file is the neutron $H_p(10)$ dose, which is calculated using neutron-to-photon dose ratios.

The neutron component of worker exposure is generally much smaller than the corresponding photon component when described as equivalent dose. The ratios range from 0.02 to 0.9, with only two conditions providing ratios higher than 0.3. For any neutron exposure, the absorbed dose to an organ will be much smaller than the equivalent dose, often by a factor of 10, due to attenuation of the neutrons by body tissue and the fact that absorbed dose does not include a quality factor, which can range from 2 to 10 for neutrons. Thus, most neutron organ doses can be expected to be less than 10% of the corresponding gamma dose.

Even though neutron radiation accounts for a relatively small component of the total organ dose on average for a worker in most Mayak workplaces, the potential for some workers to have significant neutron dose should be further examined.

## 7.2 INTERNAL DOSE (OTHER THAN PLUTONIUM)

Based on the production history of the MPA, the releases from accidents and incidents, and the residual radioactivity found in locations outside the Mayak Plant boundaries, other sources of potential occupational internal dose should be considered. It should be emphasized that the "Original Mayak Worker Cohort" included only a fraction of the total number of employees who worked at the Mayak industrial complex from the beginning of operations to the early 1970s. The selection of the original worker cohort was based on the quality of their health and dosimetry records, and their work history. Workers were specifically selected who worked at the central reactor plant facilities, the radiochemical plants, and the plutonium production plant. Even then, however, only about 40% of the workers in the original cohort had any direct contact with plutonium.

Other isotopes produced at the Mayak industrial complex included tritium, $^{238}Pu$, $^{90}Sr$, $^{241}Am$ and a range of other radionuclides. In the selection of the original Mayak Worker Cohort, workers in some Mayak facilities that produced some of these isotopes were excluded from the cohort.

## 7.3 AIRBORNE EFFLUENT DOSE

### 7.3.1 Introduction

Assessment of the potential increased environmental pathway of exposure to Mayak workers from airborne effluent was done under a collaborative effort between Project 2.4 and 1.4 researchers. Project 1.1 has assessed the dose to members of the public from Mayak releases, a dose also received by workers. Project 1.1 reports should be consulted for greater information concerning environmental releases.

### 7.3.2 Airborne Releases

Several sources of radionuclide releases into the immediate Mayak environment might not have been recorded on Mayak worker radiation records. These are the atmospheric releases from the reactors, radiochemical plant, chemical-metallurgical plant, and isotope plant.

#### 7.3.2.1 Reactors

Doses from the atmospheric releases from the reactors are dominated by $^{41}$Ar. The original Mayak graphite-moderated production reactors had no inert cover gas; air was used. Atmospheric $^{40}$Ar makes up nearly 1% of air; this was activated into $^{41}$Ar in the reactor and released up the ventilation stacks. Reactor stacks are 64 to 100 m tall (it is assumed that the early reactors had 64-m stacks.)

Rovny and Mokrov (2003) reported that an approximate noble gas source term is available, as listed in **Table 7.1**:

Table 7.1. Approximate noble gas source term

| Dates | Releases | Approximate annual release | |
|---|---|---|---|
| 1948–1958 | $1.8 \times 10^8$ Ci | 11,000,000 Ci/year | $4.1 \times 10^{17}$ Bq/year |
| 1958–1999 | $1.8 \times 10^8$ Ci | 4,400,000 Ci/year | $1.6 \times 10^{17}$ Bq/year |

In addition, Rovny and Mokrov (2003) reported, "When the wind blew in the direction of Ozyorsk, the external dose rate from the noble gas cloud was 200-400 μR/hr with a background value of ~10 μR/hr." These exposure rates would have continued until about 1963, when noble gas holdup systems were installed in the reactor effluent systems.

An adequate description of the joint frequency of occurrence of wind speed, direction, and stability for the meteorological tower at Argayash is available (Muzrukov 2005). These meteorological data are sufficient for screening purposes to generate annual average atmospheric dispersion estimates, as it is stated that the Argayash weather station data adequately describe the wind characteristics of the location of the agricultural products suppliers to Ozyorsk (farms of the Argayash Region, Kuluevo, and farms in the 1950s eastward from Ozyorsk in the area of the Techa River and Metlino village).

Using the screening information, the GENII Version 2 code (Napier et al. 2005) was used to estimate annual exposure rates at various distances from an assumed point source. These scoping dose rates, assuming 24 hr/day occupancy, are listed in

Table 7.2. Calculated dose rates downwind of a reactor release

| Avg. Dose rate, mSv/yr | Distance, m | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | **200** | **300** | **500** | **700** | **1,000** | **2,000** | **3,000** | **5,000** | **7,500** | **10,000** |
| 1948–1958 | 3 | 4 | 6 | 6 | 5 | 4 | 3 | 1 | 0.6 | 0.4 |
| 1959–1999 | 1 | 2 | 2 | 2 | 2 | 2 | 0.9 | 0.4 | 0.2 | 0.1 |

These values would be approximately the same for all organs in the body.

The values at the 7,500–10,000 meter range are applicable for the city of Ozyorsk. Measurements made in the late 1950s indicated that when the wind blew toward Ozyorsk, the exposure rates could be as high as 2 - 4 μSv/hr (Rovny and Mokrov 2003). While high, the measurements are not incompatible with this calculation. The results above are about 600 μSv total in Ozyorsk; the wind blows that direction about 2.5% of the time (220 hr/yr) – yielding a peak exposure rate of 2.7 μSv/hr – so 2–4 is not unreasonable for peak values.

To apply these screening results to a specific worker, it would be necessary to determine his or her average distance from the reactor stacks, and assuming about 25% occupancy time at work and the remaining 75% in Ozyorsk, estimate an annual dose. Using the highest dose-rate Mayak location for the early years, this yields annual doses of perhaps 2 mSv/year.

#### 7.3.2.2 Chemical-Metallurgical Plant:

The primary radionuclide of concern from the metallurgical plant is $^{239}$Pu. The metallurgical plant was temporarily housed in unsealed buildings in 1948; the facility was built in 1949 and upgraded later. The stacks heights ranged from 60 to 120 m. It is assumed that the first months of operation produced essentially ground-level releases.

Rovny and Mokrov (2003) report that the following approximate alpha-emitter source term is available:

| Dates: | Releases: | Approximate annual release: | |
|---|---|---|---|
| 1948–1963 | 34 Ci | 2.3 Ci/yr | $8.5 \times 10^{10}$ Bq/year |
| 1964–1999 | 19 Ci | 0.54 Ci/yr | $2.0 \times 10^{10}$ Bq/year |

In addition, Suslova et al. (1995) reported that the mean plutonium body burden of Ozyorsk residents at autopsy, at a distance of about 10 km, ranged from 3.7 to 7.0 Bq.

Using the same meteorological information as for the argon releases, stack heights of 10 m in 1948, 64 m in 1949-1963, and 120 m after 1963, and the reported release rates yields the estimated annual average air concentrations are shown in **Table 7.3**.

Table 7.3. Calculated air concentrations downwind of a release from the Chemical-Metallurgical Plant

| Period | Air concentration, Bq/m3 as a function of distance, m | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | **200** | **300** | **500** | **700** | **1,000** | **2,000** | **3,000** | **5,000** | **7,500** | **10,000** |
| 1948-1949 | $2 \times 10^{-2}$ | $1 \times 10^{-2}$ | $6 \times 10^{-3}$ | $4 \times 10^{-3}$ | $3 \times 10^{-3}$ | $2 \times 10^{-3}$ | $1 \times 10^{-3}$ | $8 \times 10^{-4}$ | $4 \times 10^{-4}$ | $2 \times 10^{-4}$ |
| 1949-1963 | $5 \times 10^{-6}$ | $4 \times 10^{-5}$ | $2 \times 10^{-4}$ | $3 \times 10^{-4}$ | $3 \times 10^{-4}$ | $2 \times 10^{-4}$ | $2 \times 10^{-4}$ | $1 \times 10^{-4}$ | $9 \times 10^{-5}$ | $7 \times 10^{-5}$ |
| 1963-1999 | $2 \times 10^{-10}$ | $2 \times 10^{-7}$ | $2 \times 10^{-5}$ | $2 \times 10^{-5}$ | $3 \times 10^{-5}$ | $6 \times 10^{-5}$ | $4 \times 10^{-5}$ | $3 \times 10^{-5}$ | $2 \times 10^{-5}$ | $2 \times 10^{-5}$ |

If these values are used to estimate the inhalation for an adult individual living in Ozyorsk, the result is 23 Bq over the entire period. Assuming a lung retention of about 30%, this is in reasonable agreement with the measurements of Suslova et al. (1995).

To apply these screening results to a specific worker, it would be necessary to determine his or her average distance from the reactor stacks, and assuming about 25% occupancy time at work and the remaining 75% in Ozyorsk, estimate an annual dose. Using the highest concentration Mayak location for 1948-1949 with an occupational breathing rate of 2 $m^3$/hr and 2000 hr/yr exposure, with the remaining 6,760 hr in Ozyorsk with a breathing rate of 1 $m^3$/hr, this yields annual intakes of perhaps 83 Bq/yr (of which 96% results from exposures at work). Assuming ICRP lung clearance class M, this corresponds to an inhalation effective dose of about 4 mSv/year.

#### 7.3.2.3 Radiochemical Plant:

The primary radionuclide of concern from the dissolution and processing of the irradiated fuel from the reactors in the radiochemical plant is $^{131}$I, as reported by Droshko and Khokhryakov (2003). Stacks on the radiochemical plant are very tall – from 41 to 150 meters.

The source term for the $^{131}$I releases is an objective of Project 1.4; however, a very preliminary monthly source term is available (Mokrov, Anspaugh, and Napier 2004). This can be summarized annually as shown in **Table 7.4**.

Table 7.4. Monthly $^{131}$I releases from the radiochemical plant

| Year | I-131 (Ci) | Year | I-131 (Ci) | Year | I-131 (Ci) |
|---|---|---|---|---|---|
| 1949 | 13,000 | 1962 | 1400 | 1975 | 2.8 |
| 1950 | 22,000 | 1963 | 760 | 1976 | 2.1 |
| 1951 | 39,000 | 1964 | 2500 | 1977 | 1.6 |
| 1952 | 54,000 | 1965 | 1700 | 1978 | 0.7 |
| 1953 | 57,000 | 1966 | 2300 | 1979 | 0.3 |
| 1954 | 61,000 | 1967 | 360 | 1980 | 0.4 |
| 1955 | 57,000 | 1968 | 170 | 1981 | 1 |
| 1956 | 44,000 | 1969 | 100 | 1982 | 0.9 |
| 1957 | 16,000 | 1970 | 97 | 1983 | 0.6 |
| 1958 | 960 | 1971 | 33 | 1984 | 0.3 |
| 1959 | 540 | 1972 | 2 | 1985 | 0.1 |
| 1960 | 4500 | 1973 | 8.1 | 1986 | 0.1 |
| 1961 | 480 | 1974 | 9 | | |

The total release is about 379,000 Ci of $^{131}$I.

It is possible to estimate annual average air concentrations at various distances from the radiochemical plant using the meteorological data of Muzrukov (2005). These can then be used to estimate inhalation doses. Annual averaged air concentrations at various representative distances, in Bq/$m^3$, are listed in Table 7.5.

Table 7.5. Annual average air concentrations at various distances from the Radiochemical Plant

| Year | 300 | 500 | 700 | 1,000 | 2,000 | 3,000 | 5,000 | 7,500 | 10,000 |
|---|---|---|---|---|---|---|---|---|---|
| 1949 | 8.4E-04 | 3.8E-01 | 3.7E-01 | 7.4E-01 | 1.3E+00 | 9.5E-01 | 7.2E-01 | 5.0E-01 | 3.8E-01 |
| 1950 | 1.4E-03 | 6.4E-01 | 6.3E-01 | 1.3E+00 | 2.3E+00 | 1.6E+00 | 1.2E+00 | 8.4E-01 | 6.4E-01 |
| 1951 | 2.5E-03 | 1.1E+00 | 1.1E+00 | 2.2E+00 | 4.0E+00 | 2.9E+00 | 2.2E+00 | 1.5E+00 | 1.1E+00 |
| 1952 | 3.5E-03 | 1.6E+00 | 1.5E+00 | 3.1E+00 | 5.6E+00 | 3.9E+00 | 3.0E+00 | 2.1E+00 | 1.6E+00 |
| 1953 | 3.7E-03 | 1.7E+00 | 1.6E+00 | 3.2E+00 | 5.9E+00 | 4.2E+00 | 3.2E+00 | 2.2E+00 | 1.7E+00 |
| 1954 | 4.0E-03 | 1.8E+00 | 1.7E+00 | 3.5E+00 | 6.3E+00 | 4.5E+00 | 3.4E+00 | 2.3E+00 | 1.8E+00 |
| 1955 | 3.7E-03 | 1.7E+00 | 1.6E+00 | 3.2E+00 | 5.9E+00 | 4.2E+00 | 3.2E+00 | 2.2E+00 | 1.7E+00 |
| 1956 | 2.9E-03 | 1.3E+00 | 1.3E+00 | 2.5E+00 | 4.5E+00 | 3.2E+00 | 2.4E+00 | 1.7E+00 | 1.3E+00 |
| 1957 | 1.0E-03 | 4.7E-01 | 4.6E-01 | 9.1E-01 | 1.6E+00 | 1.2E+00 | 8.9E-01 | 6.1E-01 | 4.6E-01 |
| 1958 | 6.2E-05 | 2.8E-02 | 2.7E-02 | 5.5E-02 | 9.9E-02 | 7.0E-02 | 5.3E-02 | 3.7E-02 | 2.8E-02 |
| 1959 | 3.5E-05 | 1.6E-02 | 1.5E-02 | 3.1E-02 | 5.5E-02 | 3.9E-02 | 3.0E-02 | 2.0E-02 | 1.6E-02 |
| 1960 | 2.9E-04 | 1.3E-01 | 1.3E-01 | 2.6E-01 | 4.6E-01 | 3.3E-01 | 2.5E-01 | 1.7E-01 | 1.3E-01 |
| 1961 | 3.1E-05 | 1.4E-02 | 1.4E-02 | 2.7E-02 | 5.0E-02 | 3.5E-02 | 2.7E-02 | 1.8E-02 | 1.4E-02 |
| 1962 | 9.0E-05 | 4.1E-02 | 4.0E-02 | 7.9E-02 | 1.4E-01 | 1.0E-01 | 7.7E-02 | 5.3E-02 | 4.0E-02 |
| 1963 | 4.9E-05 | 2.2E-02 | 2.2E-02 | 4.3E-02 | 7.9E-02 | 5.6E-02 | 4.2E-02 | 2.9E-02 | 2.2E-02 |
| 1964 | 1.6E-04 | 7.5E-02 | 7.3E-02 | 1.5E-01 | 2.6E-01 | 1.9E-01 | 1.4E-01 | 9.7E-02 | 7.4E-02 |
| 1965 | 1.1E-04 | 4.9E-02 | 4.8E-02 | 9.6E-02 | 1.7E-01 | 1.2E-01 | 9.4E-02 | 6.4E-02 | 4.9E-02 |
| 1966 | 1.5E-04 | 6.6E-02 | 6.4E-02 | 1.3E-01 | 2.3E-01 | 1.6E-01 | 1.3E-01 | 8.6E-02 | 6.5E-02 |
| 1967 | 2.3E-05 | 1.1E-02 | 1.0E-02 | 2.1E-02 | 3.7E-02 | 2.6E-02 | 2.0E-02 | 1.4E-02 | 1.0E-02 |
| 1968 | 1.1E-05 | 4.9E-03 | 4.8E-03 | 9.6E-03 | 1.7E-02 | 1.2E-02 | 9.3E-03 | 6.4E-03 | 4.9E-03 |
| 1969 | 6.7E-06 | 3.0E-03 | 3.0E-03 | 5.9E-03 | 1.1E-02 | 7.6E-03 | 5.8E-03 | 4.0E-03 | 3.0E-03 |
| 1970 | 6.3E-06 | 2.8E-03 | 2.8E-03 | 5.5E-03 | 1.0E-02 | 7.1E-03 | 5.4E-03 | 3.7E-03 | 2.8E-03 |
| 1971 | 2.1E-06 | 9.7E-04 | 9.4E-04 | 1.9E-03 | 3.4E-03 | 2.4E-03 | 1.8E-03 | 1.3E-03 | 9.6E-04 |
| 1972 | 1.3E-07 | 5.9E-05 | 5.7E-05 | 1.1E-04 | 2.1E-04 | 1.5E-04 | 1.1E-04 | 7.6E-05 | 5.8E-05 |
| 1973 | 5.2E-07 | 2.4E-04 | 2.3E-04 | 4.6E-04 | 8.3E-04 | 5.9E-04 | 4.5E-04 | 3.1E-04 | 2.4E-04 |
| 1974 | 5.8E-07 | 2.6E-04 | 2.6E-04 | 5.1E-04 | 9.3E-04 | 6.6E-04 | 5.0E-04 | 3.4E-04 | 2.6E-04 |
| 1975 | 1.8E-07 | 8.2E-05 | 8.0E-05 | 1.6E-04 | 2.9E-04 | 2.0E-04 | 1.6E-04 | 1.1E-04 | 8.1E-05 |
| 1976 | 1.4E-07 | 6.2E-05 | 6.0E-05 | 1.2E-04 | 2.2E-04 | 1.5E-04 | 1.2E-04 | 8.0E-05 | 6.1E-05 |
| 1977 | 1.0E-07 | 4.7E-05 | 4.6E-05 | 9.1E-05 | 1.6E-04 | 1.2E-04 | 8.9E-05 | 6.1E-05 | 4.6E-05 |
| 1978 | 4.5E-08 | 2.1E-05 | 2.0E-05 | 4.0E-05 | 7.2E-05 | 5.1E-05 | 3.9E-05 | 2.7E-05 | 2.0E-05 |
| 1979 | 1.9E-08 | 8.8E-06 | 8.6E-06 | 1.7E-05 | 3.1E-05 | 2.2E-05 | 1.7E-05 | 1.1E-05 | 8.7E-06 |
| 1980 | 2.6E-08 | 1.2E-05 | 1.1E-05 | 2.3E-05 | 4.1E-05 | 2.9E-05 | 2.2E-05 | 1.5E-05 | 1.2E-05 |
| 1981 | 6.5E-08 | 2.9E-05 | 2.9E-05 | 5.7E-05 | 1.0E-04 | 7.3E-05 | 5.5E-05 | 3.8E-05 | 2.9E-05 |
| 1982 | 5.8E-08 | 2.6E-05 | 2.6E-05 | 5.1E-05 | 9.3E-05 | 6.6E-05 | 5.0E-05 | 3.4E-05 | 2.6E-05 |
| 1983 | 3.9E-08 | 1.8E-05 | 1.7E-05 | 3.4E-05 | 6.2E-05 | 4.4E-05 | 3.3E-05 | 2.3E-05 | 1.7E-05 |
| 1984 | 1.9E-08 | 8.8E-06 | 8.6E-06 | 1.7E-05 | 3.1E-05 | 2.2E-05 | 1.7E-05 | 1.1E-05 | 8.7E-06 |
| 1985 | 6.5E-09 | 2.9E-06 | 2.9E-06 | 5.7E-06 | 1.0E-05 | 7.3E-06 | 5.5E-06 | 3.8E-06 | 2.9E-06 |
| 1986 | 6.5E-09 | 2.9E-06 | 2.9E-06 | 5.7E-06 | 1.0E-05 | 7.3E-06 | 5.5E-06 | 3.8E-06 | 2.9E-06 |

To apply these screening results to a specific worker, it would be necessary to determine his or her average distance from the reactor stacks, and assuming about 25% occupancy time at work and the remaining 75% in Ozyorsk, estimate an annual dose. Using the highest concentration Mayak location with an occupational breathing rate of 2 $m^3$/hr and 2,000 hr/yr exposure, with the remaining 6,760 hr in Ozyorsk with a breathing rate of 1 $m^3$/hr, this yields a total inhalation dose over the entire period of under 4 mSv effective dose (about 70 mSv to the thyroid). However, the primary intake of $^{131}I$ by workers is

likely to have been via intake of foods (primarily milk) while living in Ozyorsk away from work. These doses are being estimated by Project 1.4 and are not yet completed. As a rough approximation, it is noted that for adults the inhalation dose is about 10% of the total from all other external and ingestion pathways (Farris et al. 1994). If it is further assumed that the worker consumed all of his or her foods from a local garden 10 km from the Mayak plant site, this approximation can be used to obtain a total effective dose for a worker exposed at the on-site location (inclusive of occupational inhalation exposure) of about 16 mSv (300 mSv to the thyroid). These are likely to be upper-bound estimates and most workers should have received less via the atmospheric iodine pathway.

#### 7.3.2.4 Radioisotope Plant

The main processes at the isotope plant are separation and isotope purification, and fabrication of isotopic heat and radiation sources. The radioisotope plant, as an independent facility of MPA, was brought into operation in 1962. It included two facilities. Facility 1 was used mainly for extraction and purification of radioactive isotopes and for fabrication of sources with high specific activity. Radionuclides were released into the atmosphere via the 150-m stack. Since 1998, the facility was incorporated into the radiochemical plant as Facility 3. The second facility was mainly related to fabrication and packaging of radioactive sources. Radionuclides are released via the 40-m stack. In Facility 2, the radionuclide release was significantly lower than at the reactor and radiochemical plants because of the specifics of fabrication of radioactive sources and sufficient gas-purification systems. It is not likely that worker environmental dose calculations will be needed for the radioisotope plant.

### 7.3.3 1957 Accident

A major environmental release of radioactivity at Mayak occurred on September 29, 1957, because of an accidental explosion in storage tank No. 14 of the C-3 area of complex "C" at the radiochemical plant. The release involved significant spread of contamination inside and outside MPA, and significant doses to personnel and the public. At the time of the explosion, tank No. 14 contained approximately 256 $m^3$ of high-level waste with a total radioactivity of 20 million Ci (740 PBq) (JNREG 1997; Avramenko et al. 1997) accumulated during the period from March 9 until April 10, 1957. After the accident, numerous measurements of the exposure dose rate were made both near the epicenter and in the resulting radioactive trace. These measurements showed that about 90% of the total activity settled in the immediate vicinity of the explosion site (within distances less than 5 km), primarily in the form of coarse particles. The remaining 2 million Ci (74 PBq) was dispersed by the wind and caused the radioactive trace along the path of the plume.

The initial radionuclide composition of the accidental discharge into the atmosphere, derived from different sources of data, is shown in Table 7.6. "The mixture corresponded to fission products formed in a nuclear reactor after a decay time of about 1 year, with a depletion in long-lived $^{137}Cs$, uranium and plutonium. The transuranics had been removed before storage, and the treatment of the radioactive waste involved the extraction of cesium isotopes" (JNREG 1997).

The first radiation survey of MPA land near the accident and at points distant was completed on September 30, 1957. Measurements showed extremely high gamma dose rates. The radioactivity released was later estimated to be about 90% of the total activity contained in the tank; this was generally precipitated on the MPA site. Some radioactivity was scattered through a large territory now called the East-Urals Radioactive Trace (EURT). External exposure significantly prevailed during the first several

Table 7.6. Radionuclide composition of 1957 release, derived from different sources of data (according to Teverovsky et al. 1957; Lyarsky 1962; Ternovsky et al. 1983; Khokhryakov et al. 2002)

| | Contribution to total activity, % | | | | |
|---|---|---|---|---|---|
| **Radionuclide** | **Theoretical evaluation of 1957[(a)]** | **"Close" deposits in 1957[(b)]** | **"Distant" deposits in 1957[(c)]** | **Expert estimates of 1983[(d)]** | **Environmental monitoring of 1998[(e)]** |
| $^{90}Sr+^{90}Y$ | 5.0 | 12 | 7 | 5.4 | 5.4 |
| $^{95}Zr+^{95}Nb$ | 16.5 | 9 | 7 | 24.9 | 24.8 |
| $^{106}Ru+^{106}Rh$ | 5.0 | 3 | 3.5 | 3.7 | 3.7 |
| $^{144}Ce+^{144}Pr$ | 59 | 64 | 70.6 | 66 | 65.8 |
| $^{137}Cs+^{137}Ba$ | [(a)] | 1 | 1.6 | 0.036 | 0.35 |
| $^{89}Sr$ | 2.2 | 2.6 | - | Traces | Traces |
| $^{147}Pm$ | 7.3 | [(b)] | - | Traces | Traces |
| $^{155}Eu$ | 0.1 | [(b)] | - | Traces | Traces |
| $^{239}Pu$ | [(a)] | - | - | Traces | 0.002 |

(a) First theoretical calculation "made by the Mayak CPL specialists in 1957 on the basis of certification data on solutions conveyed to the exploded tank." The mixture also contained "some amount of $^{137}Cs$, about 750 kg uranium and about 0.4 kg plutonium" (Lyarsky 1962).

(b) Radionuclide composition of activity deposited at distances 5-5000 m from epicenter. An average of nine radiochemical analyses of soil samples was collected on 20 October 1957. In addition to those nuclides listed, 4% of ($^{155}Eu+ ^{147}Pm$) and 4.4% of ($^{91}Y$+other rare-earth elements) were determined (Lyarsky 1962; Teverovsky et al. 1957).

(c) Radionuclide composition of activity deposited on the axis of trace at distances 10–15 km from epicenter. Average of three radiochemical analyses of soil samples collected on 20 October 1957 (Lyarsky 1962; Teverovsky et al. 1957).

(d) Expert evaluation made by Mayak CPL specialists in 1983 on the basis of certification data on solutions conveyed to the exploded tank (Ternovsky et al. 1983; JNREG 1997; Avramenko et al. 1998). This estimate became "official" and was given in numerous papers published by the MPA after 1983.

(e) Reconstructed by G. N. Romanov in 1998 on the basis of environmental monitoring data. These findings were published by Khokhryakov et al. (2002).

days after the explosion. The main sources of external exposure were contaminated surfaces of the human environment, human body and clothes. The main pathways of internal exposure were inhalation of radioactive aerosols in air, ingestion of contaminated foodstuffs and water, and intake of radionuclides from contaminated hands.

During the period from October 1957 to April 1958, the main source of human exposure was external exposure from the environment. However, the radiation situation started improving as a result of partial radioactive decay of short-lived radionuclides and the processes of radionuclide sorption and migration in soil. The main contribution to internal exposure was determined by the ingestion of contaminated stored foodstuffs. It is assumed that the radionuclide composition of the ingested foodstuff during this period was the same as that during the first days after the explosion, corrected for radioactive decay.

The period from May 1958 to December 1958 was characterized by qualitative changes in the sources of internal dose formation. The main source of internal exposure gradually became $^{90}Sr$ as a result of radioactive decay of most short-lived radionuclides and $^{90}Sr$ transfer through food chains into agricultural

products during the first vegetation period after the accident. Nevertheless, during this period external exposure remained important due to residual surface contamination of the human environment.

Workers responding to the accident and involved in the cleanup would have been provided with dosimeters. Few are likely to have been downwind and unmonitored at the time of the accident; those who were should be specifically identified and special assessments made if they are retained in the exposure cohort. Otherwise, the main source of exposure would have been foods from nearby regions that were imported into Ozyorsk. For purposes of long-term monitoring, several nonevacuated settlements were selected. The most intensive investigations were performed in Bagaryak, Boulzy, Yushkovo, Tatarskaya Karabolka, and Allaky. In these settlements, examination of $^{90}Sr$ concentrations in food-stuffs was performed in the period from 1958 to 2003. The main source of internal exposure gradually became $^{90}Sr$ as a result of radioactive decay of most short-lived radionuclides and $^{90}Sr$ transfer through food chains into agricultural products during the first vegetation period after the accident. Two to 3 yr after the EURT formation, the radiation situation on the contaminated territory significantly improved as a result of almost complete decay of $^{95}Zr$ and decay of significant amount of $^{144}Ce$ and $^{106}Ru$. The main source of exposure became long-lived $^{90}Sr$ ingested with contaminated foodstuff. $^{90}Sr$ concentrations are listed in Table 7.7

Table 7.7. $^{90}Sr$ Concentrations on Contaminated Land Downwind of the 1957 Release

| Settlement | Distance from site of explosion, km | Distance from sanitary protection zone, km | Initial contamination density, $^{90}Sr$ Ci/km$^2$ (1957)[a] | Initial contamination density, $^{90}Sr$ Bq/m$^2$ (1957)[a] |
|---|---|---|---|---|
| Allaky | 28 | 4 | 0.48* | 18,000 |
| Tat.-Karabolka | 31 | 3 | 0.5* | 18,000 |
| Boulzy | 40 | 7 | 0.4* | 15,000 |
| Yushkovo | 45 | 3 | 0.6 | 22,000 |
| Bagaryak | 65 | 1 | 1.0 | 37,000 |

Internal dose estimates normalized per unit of contamination density with $^{90}Sr$ are used for internal dose calculation for EURT residents. It should be noted that more than 90% of the dose was accumulated during the first 2 or 3 yr after the accident. For gastrointestinal tract (GIT) doses, mainly determined by $^{144}Ce$, the dose values are dependent on what dose coefficient for what GIT department is used. Several variants on this approach are listed below. It is most likely that the dose coefficient for the Upper Large Intestine ( ULI ) was used in the estimates of Ternovsky et al. (1985), whereas the dose coefficient for the Lower Large Intestine (LLI) was used by Romanov et al. (1997) (LLI is the maximal coefficient for GIT). If we assume that residents of Ozyorsk would have received no more than 10% of their food from villages near the EURT, 20,000 Bq m$^{-2}$ $Sr^{90}$ as a typical soil contamination for food-supplying villages, and use the dose conversion values of Romanov et al. (1997), we can make a bounding estimate of doses to Ozyorsk residents/Mayak workers from this route. The estimated doses are shown in Table 7.8.

Table 7.8. Estimated ingestion dose to Ozyorsk residents from the 1957 accident

| Source of information | Period of dose accumulation, yr | Absorbed dose, μGy per Bq $^{90}Sr/m^2$ | | Dosimetric model used |
|---|---|---|---|---|
| | | Red bone marrow | GIT | |
| Burnazyan 1974 | 12 | 0.46 | 0.68 | Not referred |
| Ternovsky et al. 1985 | 25 | 0.65 | 0.51 | ICRP 1983 |
| Romanov et al. 1997 | 70 | 0.78 | 2.6 | ICRP 1990 |

### 7.3.3.1 Lake Karachay

When it became clear that discharging liquid radioactive waste into the Techa River led to increased contamination of the entire river system, the MPA began to dump its liquid waste into the closed water system of Lake Karachay instead. However, the years 1962–1966 were years of relatively low precipitation and the corresponding runoff into the lake was very low. In the spring of 1967, part of Lake Karachay evaporated and 5 ha of land that had formerly lain underwater became exposed. Unusually strong winds swept up radioactive particles from the lake sediments and spread them over an area of 1,800 $km^2$.

According to radiochemical and gamma-spectrometry measurements, the radionuclide compositions of different soil samples were rather similar. The following composition of fallout has been determined: $^{90}Sr$-$^{90}Y$: 32%; $^{137}Cs$: 47%; $^{144}Ce$-$^{144}Pr$: 21%, which differs significantly from the composition of the fallout due to the EURT. Radioactive substances fell out in the form of dust particles. According to investigation of fractional composition of dust, collected from the surface of vegetation, 50% of the fallen mass was represented by fraction 10-50 μm; 11.2% was represented by fraction 1-10 μm; and 1.5% was represented by a silt fraction <1 μm (Peremyslova et al. 2004). Because particles less than about 10 μm are not considered to be respirable, inhalation dose calculations effectively deal with half of the released amount of material.

It should be noted that some references available in the Western literature indicate that the source term from this incident is 600 Ci, not 6,000 Ci. This appears to be a compounding of an earlier typographical error. Review of the original Russian description of the event (Korsakov et al. 1968) and more recent publications (Izrael et al. 2000; Peremyslova et al. 2004) confirm that the value is about 6,000 Ci. This is a rough estimate because of the nature of the event; variable weather conditions, existing contamination of the territories by the 1957 EURT, global fallout, and routine discharges from Mayak all act to make exact determination difficult.

High average daily wind speeds were recorded by the meteorological service in April. Especially strong gusty wind was observed on 18 and 19 April, when the wind speed reached 23 m/s. Table 7.9 shows the data on concentration of β-emitter measurements in air.

Table 7.9. Concentration of β-emitters in air at different distances from Lake Karachay on 18-19 April 1967 (Korsakov et. al. 1968)

| Date | Distance from Lake Karachay, km | Concentration of β-emitters in air, | |
|---|---|---|---|
| | | Ci $l^{-1}$ | Bq $l^{-1}$ |
| April 18, 1967 | 2 | $4 \times 10^{-12}$ | 0.15 |
| April 19, 1967 | 0.5 | $4 \times 10^{-9}$ | 150 |
| April 20, 1967 | 12 | $4 \times 10^{-10}$ | 15 |

An increase in exposure dose rate was observed at the stationary observation points along the perimeter of the MPA area. Wind transfer of radioactive particles was noted in the end of the first and the beginning of the second week of April, both in the territory immediately adjacent to Karachay and in the downwind region (north-northeast sector). In May, additional contamination occurred in the east and east-southeast directions.

If we assume that the resuspension event occurred daily over the course of 1 mo (20 working days), and an average concentration of beta emitters in air of 100 Bq/L, we can conservatively estimate inhalation dose to workers. Using the 50% respirable value with an occupational breathing rate of 2 $m^3$ $hr^{-1}$ and the isotopic mixture described above, the inhalation dose would have been less than 0.2 mSv. Doses were probably considerable less than this, because the wind directions were variable during the period of release and exposure.

### 7.3.4 Comparison of Occupational Environmental Doses to Measured Occupational Doses

The magnitudes of the environmental doses, although relatively large in comparison with today's environmental standards, are unlikely to result in the need to estimate organ doses for specific workers in the cohort. Assuming continuous lifetime exposures at the rates described in previous subsections of Section 7.3, total effective and total selected organ doses are listed in Table 7.10. None of these environmental pathway doses exceed effective doses of about 6 or 7 rem (for most workers, doses should be significantly less than these conservatively estimated screening doses). The only exception to this would be for workers exposed in the late 1940s through the late 1950s, for whom thyroid doses approaching 30 rem (0.3 Sv) are possible. Epidemiological studies indicate that effects on the thyroid should consider the environmental exposure to radioiodines. Efforts are underway in Project 1.4 to develop estimates for these doses, but they are not currently available.

Table 7.10. Screening occupational environmental doses

| Source/primary radionuclide | Effective dose | Specific organ/dose |
|---|---|---|
| Reactors / $^{41}Ar$ | 6 rem (60 mSv) | |
| Chemical-metallurgical plant / $^{239}Pu$ | 0.5 rem (5 mSv) | |
| Radiochemical plant / $^{131}I$ | 1.6 rem (16 mSv) | Thyroid 30 rem (300 mSv) |
| Radioisotope Plant | -- | |
| 1957 Accident / $^{90}Sr$, F.P. | 0.1 rem (1 mSv) | |
| 1967 Lake Karachay / $^{137}Cs$, $^{90}Sr$ | 0.02 rem (0.2 mSv) | |

## 7.4 EXTERNAL AMBIENT DOSE

Mayak dosimetry records indicate that only workers with an anticipated dose greater than 30% of the radiation protection guideline would normally be monitored for radiation exposure. These personnel worked primarily in the Mayak reactor and radiochemical facilities. Employees working in other areas of the site were probably not monitored and probably did receive external dose from the ambient radiation levels on the site.

## 7.5 MEDICAL X-RAY DOSE

X-ray medical doses associated with employment at Mayak can provide a significant source of radiation exposure for MPA personnel particularly during the early years when x-ray exposures were generally higher and examinations were often more frequent. There is the associated issue that medical x-rays could be a relatively significant pathway of exposure for workers with low occupational exposures particularly in later years when occupational doses were much reduced. Lower doses workers are expected to be used as internal controls in the epidemiological analyses.

### 7.5.1 Background

Nina Koshurnikova, SUBI, oversaw collection of information on medical diagnostic procedures for a part of the 18,831-person cohort during 2003–2004. A radiographer of the Medical Department, which treated MPA workers, was involved in the x-ray procedure dose estimation. Detailed information necessary for dose estimation was prepared from the medical records for each examination to each worker as follows: examination type, x-ray device, x-ray tube voltage, current, collimation, beam size, and distance. The radiographer prepared an assessment of the exposure (in Roentgens) at the surface of the patient to include scattered radiation for selected examinations, which includes the backscattered component. This is considered to be equivalent to the entrance skin exposure (ESE) identified in ICRP Publication 34 (1982) and can be compared, for bounding purposes only, to the archival gamma dose $D_g$ from occupational exposure.

The ESE values were examined chronologically for the period beginning in about 1950 reflecting the evolution in x-ray equipment that gradually delivered lower exposures for examinations. The effort to collect medical x-ray dose information was affecting essential work on other scheduled Project 2.2 tasks. Therefore, it was decided to suspend data collection and to analyze the data collected for approximately 8,500 workers to estimate the relative levels of occupational and medical exposures to assess if the medical contribution was significant. The analyzed data represent less-than half of the available medical exposure data. This medical x-ray examination information was presented to Valery Knyazev, Mayak, who oversaw creation of a database containing the medical examination and the occupational dose. A potential future task is to calculate the organ dose to individual workers from ESE and x-ray technique data in this database using the ICRP Publication 34 (1982) organ DCFs.

### 7.5.2 Study of the Type and Frequency of the Diagnostic X-Ray Procedures

MPA workers received x-ray examinations by the SUBI Medical Department. Archive records were maintained and information from these were obtained and used in the analysis of medical x-ray dose. A medical card maintained for each worker contains information for every x-ray examination. Records for fluoroscopy and x-ray procedures were tabulated. Fewer workers received fluoroscopic examinations but there were a greater number of fluoroscopic examinations overall as shown in Table 7.11. For the group

of worker records examined, 21 types of x-ray examinations were identified. ESE values for these examinations are summarized in Table 7.12.

Table 7.11. Cohort description relative to medical examination types

| Examination type | Number of persons | Number of examinations |
|---|---|---|
| Fluorography | 4,453 | 51,283 |
| X-ray procedures | 6,034 | 36,775 |
| Fluorography and x-ray procedures | 6,362 | 84,982 |

Table 7.12. Number of medical examinations and exposed doses

| No. | Procedure | ESE/procedure, R | | | | Number |
|---|---|---|---|---|---|---|
| | | Before 1959 | 1960 – 1969 | 1970 – 1989 | After 1990 | |
| 1 | Fluoroscopy of chest | 8.2 | 7.4 | 3 | 2 | 19,581 |
| 2 | X-ray of chest (plan x-ray film) (biplane) | 1 | 0.78 | 0.7 | 0.68 | 4,588 |
| 3 | Fluorography | 1.8 | 0.98 | 0.8 | 0.71 | 5,128 |
| 4 | Tomography (4 images) | 8 | 6.66 | 5.8 | 5.5 | 49 |
| 5 | Bronchography | 75 | 66.24 | 66.24 | 66.24 | 14 |
| 6 | Fluoroscopy of stomach | 100 | 83.2 | 52 | 41 | 5,344 |
| 7 | Cholecystography (target image) | 3.76 | 3.76 | 3.76 | 3.76 | - |
| 8 | Cholecystography (plan image) | 3.76 | 3.76 | 3.76 | 3.76 | 8 |
| 9 | Cholecystography (total examination) | 17 | 14.85 | 13 | 12 | 166 |
| 10 | Urography (1 image) | 3.1 | 3.1 | 3.1 | 3.1 | 274 |
| 11 | Urography (total examination) | 18 | 16.2 | 15.5 | 15 | 10 |
| 12 | Metrosalpingography | 3 | 3 | 2.8 | 2.8 | - |
| 13 | Abdomen (plan x-ray film) | 3 | 2.39 | 2.2 | 2 | 11 |
| 14 | Passage (colon examination) | 34 | 34 | 34 | 34 | - |
| 15 | Irrigoscopy (colon examination) | 61 | 57 | 31 | 31 | 270 |
| 16 | Teeth (x-ray film) | 5.2 | 4.12 | 3.7 | 3.5 | 34 |
| 17 | Skull (x-ray film) | 3.2 | 2.8 | 2.7 | 2.7 | 1,826 |
| 18 | Femur (x-ray film) | 4.8 | 4.2 | 3.9 | 3.9 | 3 |
| 19 | Pelvis (x-ray film) | 4.8 | 4.1 | 3.9 | 3.9 | 302 |
| 20 | Spinal column (x-ray film) | 4.9 | 4.3 | 4.1 | 4.1 | 1,568 |
| 21 | Distal joints (x-ray film) | 2.5 | 2.2 | 1.9 | 1.9 | 2,709 |

Table 7.13 lists the number of medical examinations per worker. They are subdivided according to fluorography examinations and four selected general categories of radiography examinations as follows:

- X-ray examinations of chest (bronchography, chest etc.);
- X-ray examinations of chest that also included fluoroscopy;
- X-ray examinations of lower body (abdomen, stomach fluoroscopy, pelvis x-ray film, etc.); and
- Skull and teeth x-ray films (to provide dose estimation with electron paramagnetic resonance tooth measurements).

Table 7.13. Number of medical examinations per person

| Examination type | Number of examinations | | |
|---|---|---|---|
| | Minimum | Maximum | Average |
| Fluorography | 1 | 43 | 11.5 |
| X-ray procedures of chest | 1 | 53 | 4.2 |
| X-ray procedures of chest and fluorography | 1 | 75 | 12.0 |
| X-ray procedures of lower part of body | 1 | 26 | 3.3 |
| X-ray film of skull | 1 | 11 | 1.6 |

Exposures resulting from femur and distal joint examinations were not considered because they do not cause exposure to principal body organs. Table 7.13 shows that the fluorography and x-ray examinations of the chest caused the greatest contribution and could be significant, particularly for the lung dose.

### 7.5.3 Comparison of Medical and Occupational Exposure

Tabulated exposures from x-ray and fluoroscopic examinations in Tables 7.13 and 7.14 represent ESE. As a first estimate, the ESE was compared to the Mayak Radiation Dosimetry archive penetrating external dose based on measurements with personnel dosimeters. Table 7.14 lists results of analyses regarding total doses per worker; average, maximum and minimum doses due to fluorography; x-ray examination of chest, lower body, and skull; as well as occupational external gamma dose.

Table 7.14. Total medical and occupational exposure to a worker

| Examination type | Exposure, R | | |
|---|---|---|---|
| | Minimum | Maximum | Average |
| Fluorography | 0.71 | 37.5 | 10.1 |
| X-ray procedures of chest | 0.68 | 274.2 | 25.6 |
| X-ray procedures of chest and fluorography | 0.70 | 283.8 | 31.0 |
| X-ray procedures of abdomen | 2.00 | 1,884.0 | 210.3 |
| X-ray film of skull | 2.70 | 29.9 | 4.6 |
| Medical exposure (total)[(a)] | 6.79 | 2,509.41 | 281.6 |
| Occupational exposure (archive dose) | 0.02 | 1004.2 | 105.2 |

(a) Based on estimated ESE.

Table 7.14 demonstrates that the average exposure to x-ray procedures of the chest contributes about 30% of the average recorded occupational exposure for workers. The average exposure from medical x-ray examinations to the lower body is 2 times higher than the average recorded occupational exposure. Further analysis shows that the average medical exposure to 1,826 of the 5,341 workers who received x-ray examinations of the chest and fluorography is higher than the occupational exposure. Similarly, the medical dose to 1,068 of the 1,853 persons who received lower body x-ray examinations exceeded the recorded occupational exposure. Table 7.15 and Fig. 7.2 provide the distribution of average annual occupational and medical exposures. The average annual ESE from x-ray examinations of the chest exceeds the average annual recorded occupational exposure (dose) since 1962. The average annual ESE from x-ray examinations of the lower body organs exceeds the average annual occupational exposure (dose) by an order of magnitude since 1960 and by 2 orders of magnitude since 1970.

### 7.5.4 Conclusion

The analysis of the significance of medical x-ray exposure showed that, indeed, this can be an important source of confounding in the epidemiologic analyses. Typically, for epidemiologic studies other than Mayak, the history of medical exposures to workers is not available and it is generally assumed that medical exposure is similar to all cohort subjects. The Mayak study demonstrates that medical x-ray doses to workers are highly variable and for some workers could be greater than the occupational dose. This can be particularly true for workers in the early years with lower occupational exposure and for essentially all workers hired after 1962.

An improved study could be done by analyzing the organ dose from medical x-rays and comparing this to the occupational dose estimate. X-ray energies are typically lower than workplace photon irradiation and this may have a significant affect on the comparison. This will require substantially more time than was done for the bounding analysis described above. The steps include:

- Complete data entry for all medical x-ray examinations into the computer database
- Calculate entrance skin exposure for each procedure considering:
    - X-ray equipment type and design
    - Beam filtration
    - Collimation
    - Technique parameters for each examination such as kilovolt peak, milliampere-second, beam size, and distance
- Calculate organ DCFs for each procedure
- Calculate the respective organ dose for each technique
- Consider the chronology of dose based on evolution in equipment and technique parameters

**Table 7.15**. Annual medical examination average exposure compared to recorded occupational exposure

| Year | Occupational | | Chest radiography | | Fluorography | | Chest & fluorography | | Lower body | | Skull | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Dose[a] (R) | No[b] | ESE[c] (R) | No[d] | ESE[e] (R) | No[f] | ESE[g] (R) | No[h] | ESE[i] (R) | No.[j] | ESE[k] (R) | No[l] |
| 1948 | 13.8 | 7 | 8.3 | 212 | 1.8 | 1 | 8.3 | 213 | 100 | 2 | | |
| 1949 | 49.7 | 578 | 8.8 | 101 | 1.8 | 1 | 8.7 | 102 | 100 | 1 | 3.2 | 1 |
| 1950 | 79.1 | 817 | 8.9 | 119 | 1.8 | 1 | 8.8 | 120 | 109.1 | 12 | 3.2 | 3 |
| 1951 | 89.9 | 1089 | 9.3 | 189 | | | 9.3 | 189 | 74.9 | 19 | 3.2 | 4 |
| 1952 | 54.7 | 1447 | 9.3 | 589 | 1.8 | 1 | 9.3 | 590 | 87.2 | 38 | 3.2 | 14 |
| 1953 | 27.1 | 1972 | 9.4 | 953 | 1.8 | 3 | 9.3 | 956 | 106.4 | 73 | 3.6 | 26 |
| 1954 | 17.7 | 2143 | 9.8 | 1090 | 1.8 | 9 | 9.7 | 1099 | 105.7 | 61 | 3.4 | 50 |
| 1955 | 18.7 | 2102 | 9.7 | 959 | 1.8 | 336 | 8 | 1295 | 104.8 | 77 | 3.4 | 36 |
| 1956 | 14.8 | 2044 | 9.7 | 1187 | 1.8 | 15 | 9.6 | 1202 | 101.3 | 82 | 3.4 | 42 |
| 1957 | 16.4 | 2206 | 9.3 | 1291 | 1.8 | 376 | 8 | 1667 | 98 | 99 | 3.3 | 38 |
| 1958 | 9.4 | 2812 | 8.8 | 1596 | 1.8 | 72 | 8.6 | 1668 | 99 | 105 | 3.4 | 43 |
| 1959 | 8.1 | 3097 | 9 | 1386 | 0.7 | 224 | 8 | 1610 | 106.6 | 106 | 3.5 | 65 |
| 1960 | 8.6 | 3229 | 8.2 | 1876 | 1 | 559 | 6.9 | 2435 | 90.3 | 209 | 3.1 | 87 |
| 1961 | 5.8 | 3335 | 7.7 | 1675 | 1 | 303 | 6.8 | 1978 | 82.7 | 201 | 3 | 76 |
| 1962 | 4.3 | 3319 | 7.5 | 1429 | 1 | 1883 | 4.2 | 3312 | 81.6 | 238 | 3.1 | 83 |
| 1963 | 2.7 | 3246 | 7.4 | 1242 | 1 | 2583 | 3.4 | 3825 | 83.1 | 243 | 3 | 87 |
| 1964 | 2.1 | 3080 | 7.2 | 748 | 1 | 2775 | 2.4 | 3523 | 83.2 | 301 | 3 | 109 |
| 1965 | 1.8 | 3040 | 6.6 | 747 | 1 | 2637 | 2.4 | 3384 | 84.8 | 305 | 3 | 114 |
| 1966 | 1.7 | 2919 | 6.6 | 499 | 1 | 2448 | 2 | 2947 | 85.5 | 299 | 3 | 87 |
| 1967 | 1.2 | 2943 | 6 | 335 | 1 | 2311 | 1.6 | 2646 | 84.9 | 245 | 3.1 | 71 |
| 1968 | 1.1 | 2881 | 5.8 | 333 | 1 | 2331 | 1.6 | 2664 | 80.4 | 247 | 2.9 | 71 |
| 1969 | 1 | 2836 | 6.3 | 375 | 1 | 2070 | 1.9 | 2445 | 87.1 | 247 | 3 | 74 |
| 1970 | 1 | 2693 | 2.9 | 346 | 0.8 | 2087 | 1.2 | 2433 | 54.9 | 232 | 2.8 | 55 |
| 1971 | 1 | 2552 | 2.9 | 303 | 0.8 | 1945 | 1.1 | 2248 | 51.5 | 216 | 3.2 | 56 |
| 1972 | 0.7 | 2433 | 2.8 | 297 | 0.8 | 1870 | 1.1 | 2167 | 49.6 | 179 | 2.9 | 34 |
| 1973 | 0.9 | 2271 | 2.7 | 264 | 0.8 | 1740 | 1.1 | 2004 | 50.5 | 173 | 2.8 | 25 |
| 1974 | 0.7 | 2237 | 2.5 | 226 | 0.8 | 1695 | 1.1 | 1921 | 51.9 | 153 | 2.9 | 28 |
| 1975 | 0.6 | 2113 | 2.2 | 248 | 0.8 | 1646 | 1.1 | 1894 | 51.8 | 163 | 2.8 | 28 |
| 1976 | 0.7 | 2076 | 2.3 | 273 | 0.8 | 1560 | 1.1 | 1833 | 51.3 | 167 | 2.7 | 29 |
| 1977 | 0.7 | 1985 | 3.1 | 188 | 0.8 | 1477 | 1.1 | 1665 | 48.7 | 156 | 3.2 | 33 |
| 1978 | 0.8 | 1865 | 2.9 | 183 | 0.8 | 1410 | 1.1 | 1593 | 53.6 | 107 | 2.8 | 42 |
| 1979 | 0.8 | 1781 | 2.2 | 258 | 0.8 | 1342 | 1.1 | 1600 | 50.4 | 117 | 3 | 29 |
| 1980 | 0.6 | 1702 | 2.1 | 271 | 0.8 | 1193 | 1.1 | 1464 | 49 | 124 | 2.9 | 34 |
| 1981 | 0.5 | 1568 | 2 | 239 | 0.8 | 1249 | 1.1 | 1488 | 47.4 | 112 | 3 | 27 |
| 1982 | 0.6 | 1537 | 2.5 | 211 | 0.8 | 1155 | 1.1 | 1366 | 49.5 | 100 | 3 | 22 |

**Table 7.15**. (contd)

| Year | Occupational | | Chest radiography | | Fluorography | | Chest & fluorography | | Lower body | | Skull | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Dose[a] (R) | No[b] | ESE[c] (R) | No[d] | ESE[e] (R) | No[f] | ESE[g] (R) | No[h] | ESE[i] (R) | No.[j] | ESE[k] (R) | No[l] |
| 1983 | 0.6 | 1487 | 2.1 | 211 | 0.8 | 1089 | 1.1 | 1300 | 50.5 | 105 | 2.7 | 14 |
| 1984 | 0.5 | 1380 | 2.9 | 187 | 0.8 | 1040 | 1.2 | 1227 | 52.3 | 76 | 2.7 | 17 |
| 1985 | 0.5 | 1306 | 2.3 | 203 | 0.8 | 988 | 1.1 | 1191 | 54.9 | 97 | 3 | 19 |
| 1986 | 0.8 | 1240 | 2.1 | 253 | 0.8 | 903 | 1.2 | 1156 | 49.1 | 77 | 2.8 | 21 |
| 1987 | 0.5 | 1162 | 2.1 | 257 | 0.8 | 819 | 1.2 | 1076 | 48.2 | 65 | 2.8 | 20 |
| 1988 | 0.5 | 1000 | 2 | 166 | 0.8 | 753 | 1.1 | 919 | 42.6 | 50 | 2.8 | 20 |
| 1989 | 0.6 | 940 | 1.4 | 178 | 0.8 | 645 | 1 | 823 | 41.8 | 46 | 2.8 | 19 |
| 1990 | 0.4 | 871 | 1.6 | 123 | 0.7 | 610 | 0.9 | 733 | 32.6 | 29 | 2.7 | 7 |
| 1991 | 0.3 | 790 | 1.5 | 138 | 0.7 | 512 | 0.9 | 650 | 35.2 | 41 | 2.7 | 11 |
| 1992 | 0.3 | 752 | 3.5 | 76 | 0.7 | 466 | 1.1 | 542 | 38.3 | 24 | 2.7 | 7 |
| 1993 | 0.3 | 706 | 2.1 | 69 | 0.7 | 402 | 1 | 471 | 30.5 | 35 | 2.7 | 13 |
| 1994 | 0.3 | 653 | 1.5 | 74 | 0.7 | 359 | 0.9 | 433 | 26.6 | 23 | 2.7 | 4 |
| 1995 | 0.3 | 602 | 1.6 | 63 | 0.7 | 331 | 0.9 | 394 | 30.7 | 23 | 2.7 | 3 |
| 1996 | 0.2 | 525 | 1.4 | 39 | 0.7 | 236 | 0.8 | 275 | 36.9 | 18 | 2.7 | 4 |
| 1997 | 0.2 | 503 | 2.2 | 41 | 0.7 | 281 | 0.9 | 322 | 33 | 16 | 2.7 | 3 |
| 1998 | 0.2 | 483 | 1.2 | 29 | 0.7 | 224 | 0.8 | 253 | 30.4 | 8 | 2.7 | 6 |

(a) Mean occupational dose (R)
(b) Number of occupational doses
(c) Mean medical ESE of chest examinations (R)
(d) Number of medical chest examinations
(e) Mean fluorography ESE (R)
(f) Number of fluorography examinations
(g) Mean ESE of chest examination and fluorography (R)
(h) Number of chest examinations and fluorography
(i) Mean ESE of lower body medical examinations (R)
(j) Number of lower body medical examinations
(k) Mean ESE of skull examinations (R)
(l) Number of skull examinations

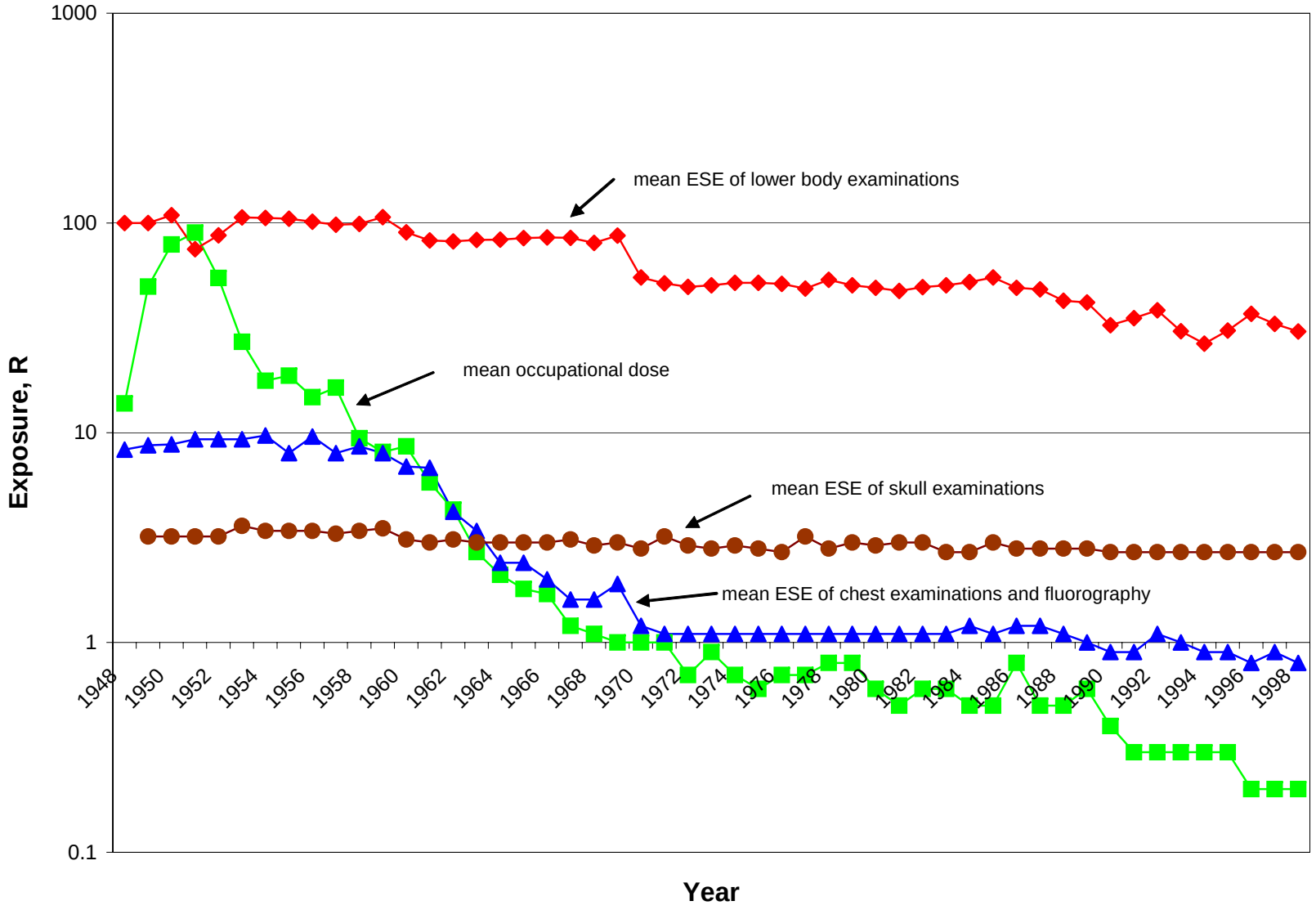



Fig. 7.2. Average annual ESE from medical examinations compared with the recorded occupational exposure (dose)

# 8.0 Doses-2005 Analysis Files

## 8.1 INTRODUCTION

Doses-2005 contains two files of organ doses: one from external radiation and one from plutonium intake. An overview of methods used to calculate the organ doses from external photon radiation is provided in Chapters 5 and 6 of this document. Detailed information is provided in Volume II, *Dose Assignment Methodology used to Calculate Annual Organ Doses to Mayak Workers from External Radiation*, and Volume III, *Dose Assignment Methodology used to Calculate Annual Organ Doses to Mayak Workers from Plutonium Intake*. A common set of organs was used in the respective analyses.

## 8.2 EXTERNAL DOSIMETRY

The External Dosimetry Dose-2005 Analysis File contains the original archive recorded dose for each worker, reconstructed doses consistent with historical methods for workers with no or partial archive doses, reconstructed absorbed dose in air, reconstructed personal dose equivalent, and the calculated absorbed dose in the 18 organs included in the dosimetry protocol. The methods used are described generally in Chapter 5 of this report and in detail in Volume II.

### 8.2.1 Archive Doses

Mayak workers significantly exposed routinely to radiation were monitored and the results recorded as described in Chapter 4. The archive records include the designation of the worker, the monitoring period beginning and ending dates, the number of work shifts in the period, the assigned dose and the signature of the person performing the analysis. Several characteristics of recorded (i.e., archive) doses for the original cohort of 18,831 are presented in Chapter 4. As described in Chapter 5, the archive doses were recorded at the time of measurement using dosimeters with different capabilities that changed with time and using calibration methods and quantities that also changed with time. As such, one step in the development of Doses-2005 was to convert the recorded dosimeter dose for all years to a consistent quantity, the absorbed dose in air in units of mGy.

### 8.2.2 Reconstructed Dose

Annual doses to Mayak workers were reconstructed to achieve year-to-year consistency, improved accuracy, and continuity with current absorbed dose and personal dose equivalent, $H_p(d)$, quantities. Several sources of information were used in the analyses as described generally in Chapter 5 and in detail in Volume II. Although the majority of workers were monitored routinely, there are workers who were not routinely monitored at all times during their employment and there are workers with no recorded dose during their employment. The reconstructed doses in Doses-2005 include consideration of several of the following technical parameters:

- Assigning each worker for each year of employment to a coworker group designation based on tasks performed in the workplace.

- Assigning each worker for each year of employment to a predominant routine exposure scenario based on the workplace and the tasks performed.

- Calculating effects on the recorded dose, and absorbed dose in air, from consideration of the dosimeter response characteristics in the workplace radiation field type, spectral and directional properties.

- Calculating effects on the reconstructed personal dose equivalent from consideration of the workplace radiation field type and orientation of the worker, and the associated absorbed organ dose.

### 8.2.3 Doses-2005 – External Dose Analysis File Structure

The dosimetry protocol identified the general structure of the Doses-2005 External Dose analysis file. The structure of the final file is shown in Fig. 8.1. Each record of this file contains selected dose parameters for each worker for each year of employment. The archive dose is the dose that was recorded historically at the time of measurement. The reconstructed dose has been calculated using methods described generally in Chapter 5 of this report and in greater detail in Volume II. To aid in understanding the process used to create the file shown in Fig. 8.1, sections of the record have been labeled as "blocks." The content of the respective blocks is described below:

**Block 1**

The first data block represents those parameters recorded in individual dosimetric monitoring (IDM) cards or logs at the time of measurement (i.e., archive dose). Unique worker identifications have been developed to allow access to the archive record systems described in Chapter 4. The gender of the worker is included to allow calculation of the absorbed organ dose for male and female workers, using the methods described in Volume II. The recorded annual gamma dose, $D_{\gamma arch}$, is determined either from the measured doses or, in the case of no archive dose, reconstructed from measured archive doses of coworkers using the methods described in Volume II. For most of the original cohort, the values of $D_{\gamma arch}$ are calculated from the original archive records using recorded doses for each processed dosimeter for the daily, weekly, etc., monitoring periods. Different units have been used over the years dating back to 1948. Because the measured dose reflects an archive record in an IDM card or a log, the column "units" represents the unit shown in the original archive records. For those records where the entry is reconstructed, the analysis is based on coworker archive doses. The last field of this block is identified as the "Reliability Index." This field is used to identify if the data in Blocks 2 and 3, derived from Block 1, are based on measured (M), reconstructed (R), or hybrid (H) information since there might be recorded doses in some but not all monitoring periods during the period of employment.

**Block 2**

The second data block represents the adjusted dosimeter interpreted dose, $D_{\gamma dos}$, by conversion of $D_{\gamma\ arch}$ values to the absorbed dose in air in milligrays for all years (see Volume II for conversion factors). This block is used to distinguish between routine and nonroutine exposure components of the annual dose, $D_{\gamma dos}$ based on analyses of the "daily dose" database (Section 4.3.5) for Mayak work areas. The beta exposure correction for nonroutine exposures in specified work areas is also done. The annual $D_{\gamma dos}$ n Fig. 8.1 consists of three components: (1) dose from measured routine photon exposure, (2) dose from reconstructed routine photon exposure, and (3) dose from measured nonroutine exposure. A specific conversion factor is applied to each of these three dose components according to the exposure scenario.

| ID of IDM | ID of SMDR | Monitoring year | Gender | $D_{\gamma\ arch}$ | Units | Reliability Index | $D_{\gamma\ dos}$, mGy | | | $D_{\gamma\ rec}$, mGy | | | $H_p(10)_{\gamma}$, mSv | | | $H_p(10)_n$, mSv | | | $D_{\gamma\ org\text{-}i}$, mGy 1-19 organs | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | min | Point estimate | max | min | Point estimate | max | Min | Point estimate | max | min | Point estimate | max | min | Point estimate | max |
| **Block 1** | | | | | | | **Block 2** | | | **Block 3** | | | **Block 4** | | | **Block 5** | | | **Blocks 6-24** | | |

where:

| | |
|---|---|
| $D_{\gamma\ arch}$ | Recorded annual photon (i.e., gamma) dose from archives or reconstructed annual photon dose for workers with no archive dose from coworker archive doses. |
| Units | Units of original archive records |
| Reliability Index | Measured (M), Reconstructed (R), or a combination (MR) |
| $D_{\gamma\ dos}$ | Adjusted $D_{\gamma\ arch}$ to absorbed dose in air at the location of the dosimeter in consistent units (mGy) |
| $D_{\gamma\ rec}$ | Reconstructed absorbed dose in air for photon radiation based on radiation field spectral/directional parameters (exposure scenario). |
| $H_p(10)_{\gamma}$ | Reconstructed personal dose equivalent for photon radiation with workplace exposure scenario parameters considered. |
| $H_p(10)_n$ | Reconstructed neutron personal dose equivalent based on $D_{\gamma\ dos}$ and neutron-photon ratios |
| $D_{\gamma\ org\text{-}i}$ | Absorbed dose to individual organs |

Fig. 8.1. Structure of external dose analysis file for individual workers

**Block 3**
Data block 3 is the reconstructed absorbed γ dose, $D_{\gamma\ rec,}$ in air at the dosimeter location point in free geometry (without a phantom) calculated from the measured or reconstructed dose in Block 2. The reconstructed absorbed γ dose, $D_{\gamma\ rec}$, is calculated using methods described in Volume II.

In general, the methods are based on considerations of the Mayak personnel dosimeter response characteristics (for measured archive doses) for the workplace radiation field types, spectra and directional characteristics and effects on the recorded dose. These considerations were quantified as conversion factors based on the worker's exposure scenario. The approach for workers with unmeasured dose is the same as for workers with measured dose; the only difference concerns uncertainty. The components of $D_{\gamma\ dos}$ were converted to the corresponding components of $D_{\gamma\ rec}$ using the conversion factors from Table 5.13.

**Block 4**
The gamma personal dose equivalent, $H_p(d)_\gamma$ ( where depth d = 10 mm in tissue), is calculated from $D_{\gamma\ dos}$ using the absorbed dose in air in Block 3 considering the photon radiation field spectral and directional properties and the orientation of the worker in the workplace defined in the predominant exposure scenarios listed in Table 5.10 per the methods described in Volume II. The dosimeter conversion factors $C_{\gamma\ Hp}$ in Table 5.14 are used to calculate the values of $H_p(10)_\gamma$ shown in Fig. 8.1.

**Block 5**
The neutron annual equivalent dose, $D_n$, is calculated from $D_{\gamma dos}$ using the K conversion factors ($K = D_n/D_\gamma$) from Table 5.15 for different monitoring periods and production areas. The personal dose equivalent, $H_p(10)_n$, shown in Fig. 8.1, is calculated from $D_n$ by applying conversion factors described in Volume II to account for spectral and directional characteristics of the neutron field as well as the worker's orientation in the workplace for the predominant exposure scenarios.

**Blocks 6–23**
The data in blocks 6 through 23 are the annual absorbed doses, $D_{\gamma\ org\text{-}I,}$ to each of the 18 organs listed in Table 5.16. These doses are calculated according to the general methods described in Section 5.9 and in more detail in Volume II.

### 8.2.4 Calculations to Determine Data Blocks

Methods used to calculate the respective variables shown in the data blocks in Fig. 8.1 are described in Volume II for each of the identified dose parameters summarized below:

- $D_{\gamma\ arch}$, Recorded annual photon (i.e., gamma) dose from archives or reconstructed annual photon dose for workers with no archive dose from coworker archive doses.

- $D_{\gamma\ dos}$ Absorbed dose in air at the location of the dosimeter, in consistent units (milligrays).

- $D_{\gamma\ rec}$, Reconstructed absorbed dose in air for photon radiation, based on radiation field spectral/directional parameters (exposure scenario).

- $H_p(10)_\gamma$, Reconstructed Personal Dose Equivalent for photon radiation, with workplace exposure scenario parameters considered.

- $H_p(10)_n$, Reconstructed Neutron Personal Dose Equivalent, based on neutron-to-photon ratios applied to $D_{\gamma\ dos}$.

- $D_{\gamma\ org}$, Absorbed dose to individual organs, for each of the 18 organs included in the dosimetry protocol.

For all categories, the exposure scenario is important because the value of a subject's measured archive dose ($D_{\gamma\ arch}$) is dependent on the radiation types and spectral and directional properties; these parameters affect the dosimeter response, and these parameters also affect the calculation of $H_p(10)$ and the organ doses. For subjects with no measured archive dose, these same parameters are essential to the process to define the exposure scenarios and to reconstruct the dose ($D_{\gamma\ rec}$) using MCNP calculations of Mayak dosimeter(s) response in defined workplace radiation (i.e., type, spectral and directional properties) fields. For MCNP calculated dose, the basic process is to define the geometry into cells, then for each cell to calculate the spectrum, normalize fluence to dose conversions, and calculate the arithmetic mean dose, and finally select the maximum and minimum dose values from the cells. The reliability index is based on the judgment of Mayak dosimetry professionals of the significance of these parameters on the recorded dose. Certainly, to the degree feasible, the archive dose records for other subjects with similar work activities in similar workplaces are used. For example, subjects in a category with no archive dose for routine exposures can be reasonably estimated from the records of other subjects with an archive dose in the same category with routine exposures. Typically, the subjects with no archive dose simply received lower exposures and generally were not required to be monitored according to radiation safety regulations. However, there are subjects for whom the archive doses have been lost, at least for some years. Workers with no archive dose and for years with nonroutine workplace exposures are certainly the most difficult for dose reconstruction. The emphasis of the work concerns archive and reconstructed photon dose. Although values are illustrated in Fig. 8.1 for the reconstructed neutron personal dose equivalent [$H_p(10)_n$], these are based on preliminary Doses-2000 $K_n$ factors (neutron-photon ratios). The organ doses are based only on the photon dose.

### 8.2.5 Comparison Between Doses-2005 and Doses-1999

Tables 8.1 and 8.2 list selected parameters of comparison between the Doses-1999 and Doses-2005 databases.

Table 8.1. Parameter comparison between Doses-1999 and Doses-2005

| Parameter | Parameter value | |
|---|---|---|
| | **Doses-1999** | **Doses-2005** |
| Date of database query | August 18, 2004 | April 28, 2006 |
| Production structure | | |
| Number of main productions | 3 | 3 |
| Number of shops | | |
| Reactor production | – | 76 |
| Radiochemical production | – | 75 |
| Plutonium production | – | 90 |
| Number of production areas | | |
| Reactor production | – | 188 |
| Radiochemical production | – | 242 |
| Plutonium production | – | 263 |
| Number of workplaces | | |
| Reactor production | – | 1,859 |
| Radiochemical production | – | 3,005 |
| Plutonium production | – | 3,277 |
| Number of workers | | |
| Total | 18,850 | 18,831 |
| Males | 14,088 | 14,072 |
| Females | 4,762 | 4,759 |
| At reactor plant | 6,619 | 6,676 |
| At radiochemical plant | 8,239 | 8,561 |
| At plutonium plant | 6,539 | 6,540 |
| At other plants | 10,347 | |
| With recorded doses | 15,725 | 15,815 |
| With recorded daily doses | – | 8,748 |
| With reconstructed doses | – | 3,016 |
| Number of records | | |
| Occupational histories | – | 65,505 |
| Yearly doses | 211,757 | 250,443 |
| Daily doses | – | 725,360 |
| Total engagement period/dosimetric monitoring of personnel (thousand man-years) | | |
| Reactor production | 76.9 / 67.6 | 90.5 / 77.7 |
| Radiochemical production | 96.4 / 93.2 | 116.8 / 109.7 |
| Plutonium production | 89.1 / 27.3 | 103.1 / 60.5 |

Table 8.2. Dose parameters for Doses-1999 and Doses-2005

| Parameter | Parameter value | | | |
|---|---|---|---|---|
| | Doses-1999 | Doses-2005, Gamma-dose | | |
| | Archive dose | Measured $D_{\gamma\ dos}$ | Dose in air $D_{\gamma\ rec}$ | Hp(10) |
| Maximum recorded annual dose /recording year | | | | |
| Reactor production | 570.0 R/1949 | 5 Gy/1949 | 4.4 Gy/1949 | 3.7 Sv/ 1949 |
| Radiochemical production | 843.9 R/1951 | 7.3 Gy/1951 | 3.5 Gy/1951 | 3.5 Sv/ 1951 |
| Plutonium production | 407.5 R/1950 | 3.5 Gy/1950 | 1.5 Gy/1950 | 1.4 Sv/ 1950 |
| Collective dose | | (person-years) | | |
| Reactor production | 3,358 | 2,595 | 2,904 | 2,462 |
| Radiochemical production | 9,961 | 7,780 | 7,263 | 6,230 |
| Plutonium production | 1,269 | 1,147 | 728 | 719 |

### 8.2.6 Characteristics of Reconstructed Doses-2005 External Doses

Doses-2005 contains an abundance of data regarding measured and reconstructed dose quantities for each worker, each year of employment, and each of the 18 organs. Selected statistical parameters of the overall cohort are presented in Tables 8.3 through 8.18 as follows:

- Table 8.3 lists the original measured dose [adjusted to consistent units of absorbed dose (milligray)], ($D_{\gamma\ dos}$), as well as the reconstructed absorbed dose (mGy), ($D_{\gamma\ rec}$), and $H_p(10)$ (millisievert). The respective cohort subjects are identified for the primary reactor, radiochemical and plutonium production facilities, respectively, for the period of 1948 through 2004.

- Tables 8.4, 8.5, and 8.6 list characteristics of the average annual $H_p(10)$ and absorbed organ doses for reactor, radiochemical, and plutonium plant workers, respectively, for selected organs.

- Tables 8.7, 8.8, and 8.9 list characteristics of the minimum annual $H_p(10)$ and minimum absorbed organ doses for reactor, radiochemical, and plutonium plant workers, respectively, for selected organs.

- Tables 8.10, 8.11, and 8.12 list characteristics of the maximum annual $H_p(10)$ and maximum absorbed organ doses for reactor, radiochemical, and plutonium plant workers, respectively, for selected organs.

- Tables 8.13, 8.14, and 8.15 list characteristics of the standard deviation of the annual $H_p(10)$ and standard deviation of the absorbed organ doses for reactor, radiochemical, and plutonium plant workers, respectively, for selected organs.

- Tables 8.16, 8.17, and 8.18 list characteristics of the 95% confidence interval of the annual $H_p(10)$ and the 95% confidence interval of the annual absorbed organ doses for reactor, radiochemical, and plutonium plant workers, respectively, for selected organs.

Table 8.3. Characteristics of average doses among production operations

| Year | Reactor plant | | | | Radiochemical plant | | | | Plutonium plant | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Measured dose, mGy | Reconstructed dose, mGy | Hp10, mSv | Neutron Hp10, mSv | Measured dose, mGy | Reconstructed dose, mGy | Hp10, mSv | Neutron Hp10, mSv | Measured dose, mGy | Recon-structed dose, mGy | Hp10, mSv | Neutron Hp10, mSv |
| 1948 | 111.19 | 97.98 | 83.42 | 0.53 | 11.92 | 8.03 | 7.28 | 0.01 | 1.32 | 0.78 | 0.70 | 0.00 |
| 1949 | 525.54 | 462.44 | 391.22 | 1.59 | 402.90 | 263.63 | 246.50 | 0.15 | 10.89 | 5.54 | 5.01 | 0.00 |
| 1950 | 163.66 | 140.20 | 119.43 | 0.59 | 787.08 | 500.22 | 467.00 | 0.50 | 165.29 | 73.00 | 66.96 | 0.01 |
| 1951 | 140.95 | 120.40 | 101.78 | 0.34 | 901.87 | 559.25 | 518.99 | 1.46 | 237.35 | 107.13 | 98.32 | 0.04 |
| 1952 | 217.99 | 192.17 | 161.28 | 0.41 | 550.47 | 350.26 | 323.77 | 0.51 | 144.71 | 64.70 | 60.54 | 0.02 |
| 1953 | 112.07 | 97.77 | 82.39 | 0.38 | 261.35 | 168.59 | 156.01 | 0.15 | 83.39 | 36.63 | 34.92 | 0.01 |
| 1954 | 58.78 | 72.26 | 61.97 | 0.30 | 172.01 | 215.79 | 176.22 | 0.35 | 44.99 | 41.14 | 40.86 | 0.03 |
| 1955 | 46.01 | 56.68 | 48.47 | 0.59 | 189.42 | 237.40 | 193.59 | 0.23 | 27.28 | 24.08 | 24.94 | 0.06 |
| 1956 | 43.26 | 53.46 | 45.68 | 0.50 | 151.70 | 191.88 | 155.84 | 0.20 | 36.84 | 32.50 | 33.91 | 0.03 |
| 1957 | 44.41 | 54.84 | 46.92 | 0.52 | 172.89 | 219.02 | 177.02 | 0.05 | 28.81 | 25.17 | 26.53 | 0.03 |
| 1958 | 34.28 | 42.52 | 36.23 | 0.43 | 96.29 | 124.01 | 99.68 | 0.01 | 27.83 | 24.19 | 25.60 | 0.05 |
| 1959 | 35.28 | 43.02 | 37.31 | 0.41 | 77.74 | 99.49 | 80.19 | 0.00 | 29.43 | 25.38 | 27.04 | 0.05 |
| 1960 | 35.58 | 43.42 | 37.43 | 0.61 | 83.98 | 108.24 | 87.10 | 0.00 | 29.75 | 25.77 | 27.56 | 0.06 |
| 1961 | 23.94 | 36.01 | 30.55 | 0.46 | 55.46 | 96.12 | 76.97 | 0.01 | 16.24 | 19.29 | 19.94 | 0.04 |
| 1962 | 22.20 | 33.64 | 28.38 | 1.05 | 40.31 | 69.94 | 56.01 | 0.01 | 11.99 | 13.75 | 14.64 | 0.04 |
| 1963 | 24.11 | 36.23 | 30.67 | 1.22 | 24.79 | 43.36 | 34.52 | 0.01 | 6.97 | 8.39 | 8.57 | 0.04 |
| 1964 | 28.06 | 42.18 | 35.70 | 1.40 | 19.04 | 33.17 | 26.47 | 0.01 | 6.38 | 8.01 | 7.89 | 0.05 |
| 1965 | 21.09 | 31.75 | 26.87 | 1.05 | 16.52 | 28.83 | 22.97 | 0.01 | 4.33 | 5.44 | 5.36 | 0.05 |
| 1966 | 17.08 | 25.63 | 21.72 | 0.87 | 15.65 | 27.21 | 21.74 | 0.01 | 3.74 | 4.74 | 4.64 | 0.04 |
| 1967 | 16.15 | 24.27 | 20.55 | 0.82 | 10.25 | 17.65 | 14.20 | 0.00 | 2.01 | 2.58 | 2.49 | 0.02 |
| 1968 | 13.16 | 19.74 | 16.72 | 0.90 | 10.60 | 18.25 | 14.68 | 0.00 | 1.04 | 1.35 | 1.30 | 0.01 |
| 1969 | 12.83 | 19.24 | 16.30 | 0.88 | 9.66 | 16.53 | 13.36 | 0.00 | 1.46 | 1.94 | 1.82 | 0.01 |
| 1970 | 12.85 | 19.22 | 16.31 | 0.90 | 9.50 | 16.26 | 13.14 | 0.00 | 1.80 | 2.53 | 2.28 | 0.02 |
| 1971 | 12.82 | 19.12 | 16.25 | 0.92 | 8.75 | 15.00 | 12.11 | 0.00 | 1.23 | 1.67 | 1.55 | 0.02 |
| 1972 | 13.39 | 20.00 | 16.99 | 0.95 | 5.92 | 10.21 | 8.21 | | 1.68 | 2.26 | 2.10 | 0.03 |
| 1973 | 16.14 | 24.17 | 20.51 | 1.12 | 8.40 | 14.40 | 11.61 | 0.00 | 1.31 | 1.79 | 1.64 | 0.02 |
| 1974 | 12.75 | 18.87 | 16.08 | 0.65 | 6.30 | 10.78 | 8.72 | | 2.44 | 3.14 | 3.04 | 0.03 |
| 1975 | 17.22 | 25.49 | 21.75 | 0.73 | 4.93 | 8.39 | 6.81 | 0.00 | 1.75 | 2.42 | 2.20 | 0.03 |
| 1976 | 12.38 | 18.44 | 15.67 | 0.63 | 6.40 | 11.03 | 8.89 | | 2.06 | 2.82 | 2.58 | 0.03 |

| Year | Reactor plant | | | | Radiochemical plant | | | | Plutonium plant | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Measured dose, mGy | Reconstructed dose, mGy | Hp10, mSv | Neutron Hp10, mSv | Measured dose, mGy | Reconstructed dose, mGy | Hp10, mSv | Neutron Hp10, mSv | Measured dose, mGy | Recon-structed dose, mGy | Hp10, mSv | Neutron Hp10, mSv |
| 1977 | 11.14 | 16.48 | 14.05 | 0.61 | 6.40 | 11.05 | 8.89 | 0.00 | 2.05 | 2.82 | 2.58 | 0.04 |
| 1978 | 11.57 | 17.14 | 14.61 | 0.62 | 7.06 | 12.15 | 9.79 | | 2.43 | 3.35 | 3.05 | 0.04 |
| 1979 | 10.67 | 15.80 | 13.46 | 0.58 | 7.58 | 13.09 | 10.53 | | 2.22 | 3.04 | 2.79 | 0.04 |
| 1980 | 11.45 | 17.05 | 14.49 | 0.57 | 5.78 | 9.99 | 8.03 | | 2.20 | 3.04 | 2.76 | 0.04 |
| 1981 | 10.28 | 15.30 | 13.01 | 0.52 | 3.98 | 6.84 | 5.52 | | 1.72 | 2.39 | 2.16 | 0.04 |
| 1982 | 9.78 | 14.48 | 12.34 | 0.52 | 4.85 | 8.37 | 6.73 | | 1.65 | 2.30 | 2.08 | 0.03 |
| 1983 | 8.73 | 12.93 | 11.02 | 0.47 | 5.24 | 9.04 | 7.27 | | 2.35 | 3.29 | 2.96 | 0.04 |
| 1984 | 7.13 | 10.56 | 9.00 | 0.40 | 4.28 | 7.39 | 5.94 | 0.00 | 2.08 | 2.92 | 2.61 | 0.04 |
| 1985 | 6.65 | 9.76 | 8.35 | 0.39 | 4.01 | 6.90 | 5.56 | 0.00 | 1.92 | 2.70 | 2.42 | 0.03 |
| 1986 | 10.00 | 14.78 | 12.61 | 0.65 | 6.54 | 11.30 | 9.09 | 0.00 | 2.81 | 3.90 | 3.53 | 0.04 |
| 1987 | 7.61 | 11.10 | 9.52 | 0.57 | 4.94 | 8.58 | 6.88 | 0.00 | 2.37 | 3.30 | 2.97 | 0.04 |
| 1988 | 6.52 | 9.62 | 8.21 | 0.45 | 4.88 | 8.48 | 6.80 | 0.00 | 1.77 | 2.48 | 2.22 | 0.03 |
| 1989 | 7.67 | 11.49 | 9.75 | 0.48 | 5.23 | 9.09 | 7.28 | 0.00 | 2.53 | 3.47 | 3.17 | 0.03 |
| 1990 | 6.12 | 9.24 | 7.80 | 0.36 | 3.77 | 6.50 | 5.23 | 0.00 | 1.97 | 2.67 | 2.47 | 0.03 |
| 1991 | 3.63 | 5.53 | 4.64 | 0.21 | 2.26 | 3.91 | 3.15 | 0.00 | 1.64 | 2.27 | 2.06 | 0.02 |
| 1992 | 3.13 | 3.14 | 3.14 | 0.13 | 2.51 | 2.51 | 2.51 | 0.00 | 1.10 | 1.11 | 1.10 | 0.01 |
| 1993 | 3.02 | 3.06 | 3.04 | 0.11 | 2.34 | 2.34 | 2.34 | 0.00 | 1.14 | 1.14 | 1.14 | 0.01 |
| 1994 | 2.71 | 2.72 | 2.72 | 0.10 | 2.51 | 2.52 | 2.51 | 0.00 | 1.25 | 1.26 | 1.25 | 0.01 |
| 1995 | 2.48 | 2.48 | 2.48 | 0.09 | 2.22 | 2.22 | 2.22 | 0.00 | 1.70 | 1.70 | 1.70 | 0.01 |
| 1996 | 2.32 | 2.32 | 2.32 | 0.08 | 2.16 | 2.16 | 2.16 | 0.00 | 1.49 | 1.49 | 1.49 | 0.02 |
| 1997 | 2.20 | 2.20 | 2.20 | 0.07 | 1.85 | 1.93 | 1.88 | | 1.51 | 1.51 | 1.51 | 0.01 |
| 1998 | 2.62 | 2.62 | 2.62 | 0.10 | 1.94 | 1.94 | 1.94 | | 2.08 | 2.09 | 2.08 | 0.02 |
| 1999 | 2.69 | 2.69 | 2.69 | 0.12 | 2.20 | 2.20 | 2.20 | | 1.87 | 1.87 | 1.87 | 0.02 |
| 2000 | 2.04 | 2.04 | 2.04 | 0.08 | 2.04 | 2.04 | 2.04 | | 2.16 | 2.16 | 2.16 | 0.02 |
| 2001 | 1.86 | 1.86 | 1.86 | 0.08 | 1.87 | 1.87 | 1.87 | | 2.06 | 2.06 | 2.06 | 0.02 |
| 2002 | 1.92 | 1.92 | 1.92 | 0.08 | 2.38 | 2.38 | 2.38 | | 2.63 | 2.63 | 2.63 | 0.03 |
| 2003 | 1.78 | 1.78 | 1.78 | 0.08 | 2.14 | 2.14 | 2.14 | | 2.37 | 2.37 | 2.37 | 0.02 |
| 2004 | 1.72 | 1.72 | 1.72 | 0.07 | 1.78 | 1.78 | 1.78 | | 1.15 | 1.15 | 1.15 | 0.01 |

Table 8.4. Characteristics of average annual $H_p(10)$ doses and absorbed organ doses for reactor plant workers.

| Year | Hp10, mSv | Absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 99.13 | 64.73 | 57.90 | 95.00 | 58.87 | 62.39 | 65.52 | 90.85 | 64.55 | 84.67 | 63.41 | 57.62 | 54.41 |
| 1949 | 470.76 | 308.53 | 278.67 | 457.87 | 281.54 | 299.43 | 315.59 | 435.37 | 308.96 | 401.62 | 306.61 | 275.25 | 263.98 |
| 1950 | 128.84 | 86.49 | 76.02 | 123.21 | 78.95 | 83.52 | 87.40 | 116.96 | 86.79 | 110.74 | 83.55 | 76.63 | 71.86 |
| 1951 | 107.74 | 73.58 | 64.61 | 103.55 | 67.82 | 71.97 | 75.52 | 96.54 | 74.80 | 91.44 | 71.80 | 65.15 | 62.29 |
| 1952 | 170.45 | 114.91 | 103.43 | 169.06 | 106.41 | 113.96 | 120.57 | 155.02 | 117.78 | 142.63 | 116.06 | 102.06 | 101.19 |
| 1953 | 86.69 | 58.82 | 52.29 | 84.84 | 53.91 | 57.19 | 60.03 | 80.04 | 59.27 | 74.94 | 57.75 | 52.20 | 49.83 |
| 1954 | 64.84 | 41.93 | 38.15 | 62.49 | 38.61 | 41.24 | 43.59 | 58.43 | 42.49 | 53.38 | 42.43 | 37.39 | 36.78 |
| 1955 | 49.81 | 31.87 | 29.26 | 48.04 | 29.38 | 31.44 | 33.32 | 44.91 | 32.31 | 40.67 | 32.58 | 28.49 | 28.28 |
| 1956 | 46.76 | 29.55 | 27.43 | 45.14 | 27.37 | 29.41 | 31.28 | 41.79 | 30.15 | 37.37 | 30.74 | 26.45 | 26.78 |
| 1957 | 47.71 | 29.87 | 27.80 | 45.84 | 27.67 | 29.75 | 31.63 | 42.45 | 30.47 | 37.86 | 31.16 | 26.75 | 27.14 |
| 1958 | 36.71 | 23.02 | 21.44 | 35.34 | 21.31 | 22.89 | 24.34 | 32.81 | 23.45 | 29.27 | 23.99 | 20.63 | 20.88 |
| 1959 | 37.74 | 24.09 | 21.95 | 35.96 | 22.24 | 23.78 | 25.15 | 33.42 | 24.50 | 30.45 | 24.49 | 21.46 | 21.27 |
| 1960 | 37.78 | 24.75 | 22.23 | 36.35 | 22.65 | 24.03 | 25.20 | 34.42 | 24.86 | 31.90 | 24.50 | 21.97 | 21.08 |
| 1961 | 30.92 | 20.26 | 18.32 | 29.99 | 18.59 | 19.79 | 20.82 | 28.25 | 20.42 | 25.97 | 20.29 | 18.01 | 17.51 |
| 1962 | 28.74 | 18.73 | 17.03 | 27.91 | 17.20 | 18.32 | 19.31 | 26.30 | 18.88 | 24.06 | 18.86 | 16.69 | 16.29 |
| 1963 | 31.05 | 20.24 | 18.35 | 30.09 | 18.54 | 19.73 | 20.76 | 28.46 | 20.35 | 26.12 | 20.27 | 18.01 | 17.47 |
| 1964 | 36.14 | 23.63 | 21.46 | 35.21 | 21.65 | 23.05 | 24.27 | 33.29 | 23.77 | 30.51 | 23.71 | 21.05 | 20.44 |
| 1965 | 27.21 | 17.79 | 16.18 | 26.56 | 16.32 | 17.38 | 18.31 | 25.07 | 17.91 | 22.93 | 17.90 | 15.84 | 15.44 |
| 1966 | 21.98 | 14.46 | 13.07 | 21.41 | 13.24 | 14.07 | 14.80 | 20.26 | 14.53 | 18.65 | 14.42 | 12.86 | 12.43 |
| 1967 | 20.81 | 13.65 | 12.38 | 20.31 | 12.51 | 13.31 | 14.01 | 19.20 | 13.73 | 17.61 | 13.69 | 12.15 | 11.79 |
| 1968 | 16.94 | 11.06 | 10.03 | 16.47 | 10.13 | 10.78 | 11.34 | 15.58 | 11.11 | 14.28 | 11.08 | 9.84 | 9.55 |
| 1969 | 16.54 | 10.80 | 9.80 | 16.08 | 9.90 | 10.53 | 11.08 | 15.22 | 10.86 | 13.96 | 10.82 | 9.61 | 9.32 |
| 1970 | 16.54 | 10.84 | 9.82 | 16.12 | 9.92 | 10.55 | 11.10 | 15.27 | 10.89 | 14.02 | 10.84 | 9.65 | 9.33 |
| 1971 | 16.49 | 10.85 | 9.79 | 16.06 | 9.92 | 10.53 | 11.07 | 15.23 | 10.88 | 14.04 | 10.79 | 9.64 | 9.28 |
| 1972 | 17.19 | 11.28 | 10.21 | 16.75 | 10.33 | 10.97 | 11.53 | 15.86 | 11.33 | 14.59 | 11.26 | 10.03 | 9.69 |
| 1973 | 20.70 | 13.58 | 12.32 | 20.21 | 12.44 | 13.23 | 13.92 | 19.12 | 13.65 | 17.53 | 13.61 | 12.08 | 11.72 |
| 1974 | 16.27 | 10.75 | 9.65 | 15.80 | 9.81 | 10.39 | 10.89 | 15.07 | 10.75 | 13.97 | 10.60 | 9.54 | 9.09 |
| 1975 | 21.97 | 14.52 | 13.04 | 21.37 | 13.25 | 14.05 | 14.72 | 20.33 | 14.53 | 18.84 | 14.34 | 12.89 | 12.31 |
| 1976 | 15.82 | 10.42 | 9.40 | 15.41 | 9.52 | 10.11 | 10.61 | 14.64 | 10.44 | 13.50 | 10.35 | 9.26 | 8.90 |
| 1977 | 14.18 | 9.38 | 8.41 | 13.78 | 8.56 | 9.06 | 9.49 | 13.14 | 9.38 | 12.19 | 9.24 | 8.32 | 7.92 |

| Year | Hp10, mSv | Absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1978 | 14.75 | 9.77 | 8.77 | 14.37 | 8.91 | 9.45 | 9.90 | 13.68 | 9.77 | 12.68 | 9.64 | 8.67 | 8.28 |
| 1979 | 13.62 | 9.01 | 8.09 | 13.26 | 8.22 | 8.72 | 9.14 | 12.63 | 9.01 | 11.70 | 8.90 | 8.00 | 7.64 |
| 1980 | 14.65 | 9.64 | 8.70 | 14.27 | 8.82 | 9.36 | 9.83 | 13.55 | 9.67 | 12.49 | 9.59 | 8.57 | 8.24 |
| 1981 | 13.15 | 8.67 | 7.83 | 12.83 | 7.93 | 8.42 | 8.84 | 12.18 | 8.70 | 11.23 | 8.62 | 7.71 | 7.41 |
| 1982 | 12.48 | 8.27 | 7.43 | 12.17 | 7.55 | 8.00 | 8.38 | 11.59 | 8.27 | 10.73 | 8.17 | 7.34 | 7.01 |
| 1983 | 11.14 | 7.37 | 6.62 | 10.85 | 6.73 | 7.13 | 7.47 | 10.33 | 7.37 | 9.57 | 7.28 | 6.54 | 6.25 |
| 1984 | 9.08 | 6.01 | 5.40 | 8.84 | 5.49 | 5.82 | 6.09 | 8.43 | 6.02 | 7.81 | 5.93 | 5.34 | 5.09 |
| 1985 | 8.40 | 5.60 | 4.99 | 8.15 | 5.10 | 5.39 | 5.62 | 7.81 | 5.58 | 7.30 | 5.46 | 4.96 | 4.67 |
| 1986 | 12.71 | 8.43 | 7.56 | 12.38 | 7.69 | 8.14 | 8.53 | 11.80 | 8.43 | 10.96 | 8.30 | 7.48 | 7.12 |
| 1987 | 9.60 | 6.41 | 5.68 | 9.29 | 5.83 | 6.15 | 6.40 | 8.92 | 6.38 | 8.38 | 6.21 | 5.67 | 5.30 |
| 1988 | 8.28 | 5.49 | 4.91 | 8.04 | 5.00 | 5.30 | 5.54 | 7.67 | 5.48 | 7.13 | 5.39 | 4.86 | 4.62 |
| 1989 | 9.82 | 6.45 | 5.85 | 9.60 | 5.91 | 6.28 | 6.61 | 9.08 | 6.48 | 8.33 | 6.46 | 5.74 | 5.57 |
| 1990 | 7.85 | 5.12 | 4.68 | 7.68 | 4.70 | 5.01 | 5.29 | 7.24 | 5.16 | 6.59 | 5.18 | 4.56 | 4.48 |
| 1991 | 4.68 | 3.05 | 2.79 | 4.57 | 2.80 | 2.99 | 3.16 | 4.32 | 3.08 | 3.91 | 3.09 | 2.72 | 2.67 |
| 1992 | 3.15 | 1.74 | 1.58 | 2.59 | 1.59 | 1.70 | 1.78 | 2.46 | 1.75 | 2.25 | 1.75 | 1.55 | 1.50 |
| 1993 | 3.06 | 1.63 | 1.50 | 2.46 | 1.50 | 1.60 | 1.69 | 2.31 | 1.64 | 2.08 | 1.67 | 1.45 | 1.44 |
| 1994 | 2.73 | 1.49 | 1.36 | 2.23 | 1.37 | 1.46 | 1.54 | 2.10 | 1.50 | 1.90 | 1.51 | 1.32 | 1.31 |
| 1995 | 2.49 | 1.35 | 1.24 | 2.04 | 1.25 | 1.33 | 1.41 | 1.93 | 1.37 | 1.74 | 1.38 | 1.21 | 1.19 |
| 1996 | 2.34 | 1.26 | 1.16 | 1.90 | 1.16 | 1.24 | 1.31 | 1.79 | 1.27 | 1.61 | 1.29 | 1.12 | 1.12 |
| 1997 | 2.22 | 1.19 | 1.10 | 1.80 | 1.09 | 1.17 | 1.24 | 1.70 | 1.20 | 1.52 | 1.22 | 1.06 | 1.06 |
| 1998 | 2.63 | 1.43 | 1.31 | 2.15 | 1.31 | 1.40 | 1.48 | 2.03 | 1.44 | 1.83 | 1.45 | 1.27 | 1.25 |
| 1999 | 2.69 | 1.51 | 1.36 | 2.23 | 1.38 | 1.46 | 1.54 | 2.12 | 1.51 | 1.96 | 1.50 | 1.34 | 1.29 |
| 2000 | 2.04 | 1.14 | 1.03 | 1.69 | 1.04 | 1.11 | 1.16 | 1.60 | 1.14 | 1.47 | 1.14 | 1.01 | 0.98 |
| 2001 | 1.86 | 1.05 | 0.94 | 1.55 | 0.96 | 1.02 | 1.07 | 1.47 | 1.05 | 1.35 | 1.04 | 0.93 | 0.89 |
| 2002 | 1.92 | 1.08 | 0.97 | 1.59 | 0.98 | 1.04 | 1.10 | 1.51 | 1.08 | 1.40 | 1.07 | 0.96 | 0.92 |
| 2003 | 1.78 | 1.00 | 0.90 | 1.47 | 0.91 | 0.97 | 1.02 | 1.40 | 1.00 | 1.29 | 0.99 | 0.89 | 0.85 |
| 2004 | 1.85 | 1.02 | 0.93 | 1.53 | 0.94 | 1.00 | 1.05 | 1.44 | 1.03 | 1.32 | 1.03 | 0.91 | 0.89 |

Table 8.5. Characteristics of average annual $H_p(10)$ doses and absorbed organ doses for radiochemical plant workers

| | | Absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 70.39 | 59.39 | 45.29 | 67.48 | 54.10 | 55.77 | 56.48 | 64.34 | 59.73 | 70.80 | 49.17 | 51.19 | 41.73 |
| 1949 | 259.56 | 300.07 | 199.69 | 260.54 | 271.02 | 270.74 | 265.55 | 261.44 | 298.59 | 344.03 | 209.63 | 253.87 | 176.55 |
| 1950 | 486.66 | 556.16 | 370.13 | 485.83 | 501.46 | 500.62 | 490.36 | 487.53 | 552.52 | 641.64 | 387.44 | 470.57 | 325.50 |
| 1951 | 535.78 | 604.68 | 404.64 | 534.18 | 545.58 | 545.01 | 534.51 | 534.94 | 600.72 | 696.19 | 422.70 | 512.30 | 357.19 |
| 1952 | 329.90 | 375.89 | 250.47 | 327.80 | 339.32 | 338.87 | 332.28 | 329.34 | 373.74 | 431.66 | 262.24 | 318.17 | 221.27 |
| 1953 | 157.40 | 178.93 | 118.45 | 153.92 | 161.47 | 161.06 | 157.70 | 155.04 | 177.88 | 205.25 | 123.97 | 151.30 | 104.40 |
| 1954 | 176.57 | 239.57 | 156.92 | 202.87 | 215.91 | 214.91 | 209.87 | 204.53 | 237.93 | 274.60 | 163.79 | 202.18 | 137.63 |
| 1955 | 194.10 | 263.60 | 172.78 | 223.52 | 237.58 | 236.55 | 231.04 | 225.32 | 261.87 | 302.29 | 180.48 | 222.45 | 151.56 |
| 1956 | 156.02 | 214.06 | 140.07 | 180.33 | 192.97 | 192.05 | 187.54 | 182.18 | 212.63 | 245.08 | 146.28 | 180.63 | 122.91 |
| 1957 | 177.63 | 241.09 | 158.96 | 205.44 | 217.50 | 216.85 | 212.13 | 207.35 | 239.65 | 276.42 | 166.46 | 203.65 | 139.89 |
| 1958 | 100.19 | 132.32 | 87.38 | 112.84 | 119.43 | 119.12 | 116.59 | 113.92 | 131.57 | 151.59 | 91.58 | 111.80 | 77.01 |
| 1959 | 80.41 | 103.79 | 69.12 | 89.78 | 93.75 | 93.68 | 91.88 | 90.45 | 103.27 | 119.06 | 72.61 | 87.81 | 61.12 |
| 1960 | 87.23 | 117.45 | 77.32 | 99.48 | 105.98 | 105.63 | 103.32 | 100.60 | 116.75 | 134.45 | 80.98 | 99.20 | 68.07 |
| 1961 | 77.04 | 105.85 | 69.78 | 89.91 | 95.52 | 95.23 | 93.17 | 90.87 | 105.23 | 121.19 | 73.09 | 89.42 | 61.46 |
| 1962 | 56.04 | 78.16 | 51.27 | 65.88 | 70.49 | 70.20 | 68.60 | 66.64 | 77.67 | 89.46 | 53.62 | 65.98 | 45.05 |
| 1963 | 34.56 | 48.44 | 31.78 | 40.83 | 43.69 | 43.51 | 42.52 | 41.32 | 48.14 | 55.44 | 33.25 | 40.89 | 27.94 |
| 1964 | 26.52 | 37.10 | 24.33 | 31.32 | 33.46 | 33.32 | 32.56 | 31.66 | 36.87 | 42.48 | 25.45 | 31.31 | 21.38 |
| 1965 | 23.02 | 31.70 | 20.83 | 26.84 | 28.60 | 28.50 | 27.86 | 27.13 | 31.51 | 36.29 | 21.80 | 26.76 | 18.33 |
| 1966 | 21.79 | 30.06 | 19.79 | 25.54 | 27.12 | 27.03 | 26.44 | 25.79 | 29.88 | 34.43 | 20.72 | 25.39 | 17.42 |
| 1967 | 14.22 | 19.34 | 12.69 | 16.34 | 17.44 | 17.37 | 16.98 | 16.50 | 19.22 | 22.13 | 13.27 | 16.32 | 11.15 |
| 1968 | 14.71 | 20.36 | 13.38 | 17.25 | 18.36 | 18.30 | 17.89 | 17.43 | 20.23 | 23.33 | 14.00 | 17.19 | 11.76 |
| 1969 | 13.37 | 18.27 | 12.04 | 15.55 | 16.48 | 16.43 | 16.08 | 15.69 | 18.16 | 20.92 | 12.62 | 15.43 | 10.61 |
| 1970 | 13.16 | 18.01 | 11.85 | 15.28 | 16.25 | 16.19 | 15.84 | 15.43 | 17.90 | 20.62 | 12.40 | 15.21 | 10.43 |
| 1971 | 12.12 | 16.51 | 10.86 | 14.01 | 14.90 | 14.85 | 14.52 | 14.15 | 16.42 | 18.91 | 11.38 | 13.94 | 9.56 |
| 1972 | 8.23 | 11.35 | 7.47 | 9.61 | 10.24 | 10.21 | 9.98 | 9.71 | 11.29 | 13.00 | 7.82 | 9.59 | 6.57 |
| 1973 | 11.62 | 15.78 | 10.43 | 13.49 | 14.24 | 14.21 | 13.92 | 13.60 | 15.70 | 18.07 | 10.94 | 13.33 | 9.20 |
| 1974 | 8.72 | 12.09 | 7.94 | 10.20 | 10.90 | 10.86 | 10.62 | 10.31 | 12.01 | 13.83 | 8.31 | 10.21 | 6.98 |
| 1975 | 6.82 | 9.41 | 6.18 | 7.95 | 8.49 | 8.46 | 8.27 | 8.04 | 9.35 | 10.77 | 6.47 | 7.95 | 5.44 |
| 1976 | 8.89 | 12.47 | 8.17 | 10.47 | 1125 | 11.20 | 10.94 | 10.60 | 12.39 | 14.27 | 8.54 | 10.53 | 7.18 |
| 1977 | 8.89 | 12.50 | 8.18 | 10.47 | 11.27 | 11.22 | 10.96 | 10.61 | 12.42 | 14.30 | 8.55 | 10.55 | 7.18 |

| | | Absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1978 | 9.80 | 13.71 | 8.98 | 11.52 | 12.37 | 12.31 | 12.03 | 11.66 | 13.62 | 15.69 | 9.40 | 11.57 | 7.89 |
| 1979 | 10.53 | 14.82 | 9.70 | 12.42 | 13.37 | 13.30 | 12.99 | 12.58 | 14.73 | 16.96 | 10.14 | 12.51 | 8.52 |
| 1980 | 8.03 | 11.34 | 7.42 | 9.50 | 10.23 | 10.18 | 9.94 | 9.63 | 11.27 | 12.98 | 7.76 | 9.57 | 6.52 |
| 1981 | 5.52 | 7.74 | 5.07 | 6.50 | 6.98 | 6.95 | 6.79 | 6.58 | 7.69 | 8.86 | 5.31 | 6.54 | 4.46 |
| 1982 | 6.73 | 9.46 | 6.20 | 7.94 | 8.53 | 8.49 | 8.30 | 8.04 | 9.40 | 10.82 | 6.48 | 7.98 | 5.44 |
| 1983 | 7.27 | 10.12 | 6.65 | 8.54 | 9.13 | 9.09 | 8.89 | 8.64 | 10.05 | 11.58 | 6.96 | 8.54 | 5.85 |
| 1984 | 5.94 | 8.31 | 5.45 | 6.99 | 7.50 | 7.47 | 7.30 | 7.08 | 8.26 | 9.51 | 5.70 | 7.02 | 4.79 |
| 1985 | 5.56 | 7.77 | 5.10 | 6.54 | 7.01 | 6.98 | 6.82 | 6.62 | 7.73 | 8.90 | 5.33 | 6.56 | 4.48 |
| 1986 | 9.09 | 12.84 | 8.39 | 10.75 | 11.57 | 11.52 | 11.25 | 10.89 | 12.75 | 14.68 | 8.77 | 10.83 | 7.37 |
| 1987 | 6.88 | 9.84 | 6.42 | 8.20 | 8.87 | 8.82 | 8.61 | 8.31 | 9.78 | 11.26 | 6.70 | 8.30 | 5.63 |
| 1988 | 6.80 | 9.76 | 6.35 | 8.10 | 8.79 | 8.74 | 8.53 | 8.22 | 9.69 | 11.16 | 6.63 | 8.23 | 5.56 |
| 1989 | 7.28 | 10.37 | 6.77 | 8.66 | 9.34 | 9.30 | 9.07 | 8.78 | 10.30 | 11.85 | 7.07 | 8.74 | 5.94 |
| 1990 | 5.24 | 7.46 | 4.86 | 6.22 | 6.72 | 6.69 | 6.52 | 6.30 | 7.41 | 8.53 | 5.08 | 6.29 | 4.26 |
| 1991 | 3.15 | 4.49 | 2.92 | 3.73 | 4.05 | 4.02 | 3.92 | 3.79 | 4.46 | 5.13 | 3.05 | 3.79 | 2.56 |
| 1992 | 2.51 | 2.87 | 1.87 | 2.40 | 2.58 | 2.57 | 2.51 | 2.43 | 2.85 | 3.28 | 1.96 | 2.42 | 1.64 |
| 1993 | 2.34 | 2.64 | 1.73 | 2.23 | 2.38 | 2.37 | 2.32 | 2.25 | 2.63 | 3.03 | 1.81 | 2.23 | 1.52 |
| 1994 | 2.51 | 2.84 | 1.86 | 2.38 | 2.56 | 2.55 | 2.49 | 2.41 | 2.82 | 3.25 | 1.94 | 2.40 | 1.63 |
| 1995 | 2.22 | 2.55 | 1.66 | 2.13 | 2.30 | 2.29 | 2.23 | 2.16 | 2.54 | 2.92 | 1.74 | 2.15 | 1.46 |
| 1996 | 2.16 | 2.48 | 1.62 | 2.07 | 2.24 | 2.22 | 2.17 | 2.09 | 2.47 | 2.84 | 1.69 | 2.09 | 1.42 |
| 1997 | 1.88 | 1.97 | 1.29 | 1.66 | 1.78 | 1.77 | 1.73 | 1.68 | 1.96 | 2.26 | 1.35 | 1.66 | 1.14 |
| 1998 | 1.94 | 2.17 | 1.43 | 1.84 | 1.96 | 1.95 | 1.91 | 1.86 | 2.16 | 2.49 | 1.50 | 1.84 | 1.26 |
| 1999 | 2.20 | 2.49 | 1.63 | 2.10 | 2.25 | 2.24 | 2.18 | 2.12 | 2.47 | 2.85 | 1.71 | 2.10 | 1.43 |
| 2000 | 2.04 | 2.32 | 1.52 | 1.94 | 2.09 | 2.08 | 2.03 | 1.97 | 2.30 | 2.65 | 1.58 | 1.96 | 1.33 |
| 2001 | 1.87 | 2.09 | 1.38 | 1.78 | 1.89 | 1.88 | 1.84 | 1.79 | 2.08 | 2.39 | 1.44 | 1.77 | 1.21 |
| 2002 | 2.38 | 2.71 | 1.77 | 2.28 | 2.44 | 2.43 | 2.37 | 2.30 | 2.69 | 3.10 | 1.85 | 2.28 | 1.56 |
| 2003 | 2.14 | 2.42 | 1.59 | 2.04 | 2.18 | 2.17 | 2.12 | 2.06 | 2.40 | 2.77 | 1.66 | 2.04 | 1.39 |
| 2004 | 1.86 | 2.10 | 1.38 | 1.77 | 1.90 | 1.89 | 1.85 | 1.79 | 2.09 | 2.41 | 1.44 | 1.78 | 1.21 |

Table 8.6. Characteristics of average annual $H_p(10)$ doses and absorbed organ doses for plutonium plant workers

| | | Absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 40.10 | 25.79 | 20.00 | 30.18 | 23.53 | 24.41 | 24.91 | 28.81 | 26.15 | 31.10 | 21.75 | 22.46 | 18.61 |
| 1949 | 16.51 | 13.05 | 9.61 | 14.80 | 11.82 | 12.10 | 12.11 | 13.78 | 13.04 | 15.11 | 10.05 | 11.19 | 8.83 |
| 1950 | 97.53 | 91.54 | 64.22 | 92.08 | 82.98 | 84.35 | 83.81 | 89.02 | 92.44 | 106.94 | 68.22 | 78.25 | 57.89 |
| 1951 | 149.93 | 138.54 | 97.91 | 140.97 | 125.66 | 127.95 | 127.29 | 135.94 | 139.88 | 161.84 | 104.21 | 118.47 | 88.50 |
| 1952 | 96.07 | 88.34 | 62.02 | 89.30 | 80.05 | 81.40 | 80.84 | 85.88 | 89.13 | 103.01 | 65.87 | 75.43 | 55.89 |
| 1953 | 55.45 | 50.75 | 35.39 | 51.00 | 45.93 | 46.66 | 46.23 | 48.95 | 51.16 | 59.13 | 37.52 | 43.25 | 31.76 |
| 1954 | 61.81 | 57.21 | 38.48 | 58.62 | 51.43 | 52.13 | 51.06 | 53.47 | 57.72 | 66.37 | 40.56 | 48.04 | 33.96 |
| 1955 | 40.57 | 37.89 | 24.90 | 38.07 | 33.97 | 34.29 | 33.38 | 34.35 | 38.19 | 43.85 | 26.03 | 31.70 | 21.73 |
| 1956 | 50.63 | 46.61 | 30.72 | 47.17 | 41.80 | 42.26 | 41.18 | 42.33 | 47.04 | 53.95 | 32.20 | 39.00 | 26.85 |
| 1957 | 37.79 | 35.13 | 22.82 | 34.91 | 31.45 | 31.68 | 30.75 | 31.33 | 35.37 | 40.58 | 23.74 | 29.34 | 19.80 |
| 1958 | 34.80 | 32.70 | 21.05 | 32.16 | 29.25 | 29.40 | 28.47 | 28.85 | 32.90 | 37.74 | 21.82 | 27.28 | 18.18 |
| 1959 | 37.30 | 35.27 | 22.59 | 34.57 | 31.53 | 31.65 | 30.63 | 30.94 | 35.48 | 40.68 | 23.37 | 29.40 | 19.47 |
| 1960 | 40.44 | 38.79 | 24.71 | 37.51 | 34.66 | 34.77 | 33.61 | 33.66 | 39.02 | 44.73 | 25.58 | 32.30 | 21.24 |
| 1961 | 35.01 | 33.20 | 21.28 | 33.31 | 29.52 | 29.65 | 28.67 | 29.46 | 33.17 | 38.10 | 21.86 | 27.55 | 18.38 |
| 1962 | 25.86 | 24.65 | 15.58 | 24.23 | 21.88 | 21.95 | 21.17 | 21.39 | 24.66 | 28.28 | 16.02 | 20.39 | 13.36 |
| 1963 | 15.23 | 14.51 | 9.24 | 14.56 | 12.88 | 12.91 | 12.45 | 12.80 | 14.46 | 16.62 | 9.41 | 12.03 | 7.95 |
| 1964 | 13.65 | 13.21 | 8.38 | 13.44 | 11.72 | 11.71 | 11.27 | 11.71 | 13.12 | 15.09 | 8.44 | 10.95 | 7.21 |
| 1965 | 9.08 | 8.64 | 5.56 | 8.87 | 7.68 | 7.69 | 7.43 | 7.79 | 8.59 | 9.89 | 5.63 | 7.18 | 4.81 |
| 1966 | 7.98 | 7.64 | 4.89 | 7.84 | 6.78 | 6.78 | 6.55 | 6.86 | 7.59 | 8.73 | 4.94 | 6.34 | 4.23 |
| 1967 | 4.13 | 4.16 | 2.61 | 4.22 | 3.68 | 3.68 | 3.53 | 3.63 | 4.13 | 4.74 | 2.62 | 3.44 | 2.23 |
| 1968 | 2.30 | 2.31 | 1.45 | 2.34 | 2.05 | 2.05 | 1.97 | 2.02 | 2.30 | 2.64 | 1.47 | 1.91 | 1.24 |
| 1969 | 3.13 | 3.28 | 2.03 | 3.30 | 2.90 | 2.89 | 2.77 | 2.82 | 3.26 | 3.74 | 2.04 | 2.71 | 1.73 |
| 1970 | 4.12 | 4.24 | 2.69 | 4.35 | 3.76 | 3.75 | 3.60 | 3.76 | 4.19 | 4.83 | 2.68 | 3.52 | 2.31 |
| 1971 | 2.49 | 2.38 | 1.55 | 2.48 | 2.12 | 2.12 | 2.06 | 2.18 | 2.37 | 2.73 | 1.57 | 1.99 | 1.35 |
| 1972 | 3.23 | 3.21 | 2.05 | 3.32 | 2.85 | 2.85 | 2.74 | 2.88 | 3.19 | 3.67 | 2.05 | 2.67 | 1.76 |
| 1973 | 2.50 | 2.58 | 1.63 | 2.66 | 2.29 | 2.28 | 2.19 | 2.28 | 2.56 | 2.94 | 1.63 | 2.14 | 1.40 |
| 1974 | 4.63 | 4.53 | 2.90 | 4.70 | 4.02 | 4.03 | 3.90 | 4.08 | 4.52 | 5.20 | 2.94 | 3.76 | 2.50 |
| 1975 | 3.31 | 3.59 | 2.24 | 3.66 | 3.18 | 3.17 | 3.04 | 3.12 | 3.57 | 4.09 | 2.25 | 2.96 | 1.91 |
| 1976 | 3.85 | 4.08 | 2.57 | 4.20 | 3.62 | 3.61 | 3.47 | 3.59 | 4.05 | 4.66 | 2.57 | 3.38 | 2.20 |

| Year | Hp10, mSv | Absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1977 | 3.85 | 4.08 | 2.57 | 4.21 | 3.62 | 3.61 | 3.47 | 3.61 | 4.06 | 4.66 | 2.58 | 3.38 | 2.20 |
| 1978 | 4.52 | 4.82 | 3.01 | 4.95 | 4.27 | 4.25 | 4.07 | 4.23 | 4.78 | 5.49 | 3.00 | 3.99 | 2.57 |
| 1979 | 4.12 | 4.34 | 2.74 | 4.49 | 3.85 | 3.84 | 3.69 | 3.85 | 4.30 | 4.95 | 2.74 | 3.60 | 2.35 |
| 1980 | 4.03 | 4.25 | 2.67 | 4.39 | 3.77 | 3.75 | 3.60 | 3.76 | 4.21 | 4.84 | 2.66 | 3.52 | 2.29 |
| 1981 | 3.12 | 3.31 | 2.11 | 3.47 | 2.93 | 2.92 | 2.81 | 2.98 | 3.27 | 3.76 | 2.08 | 2.75 | 1.82 |
| 1982 | 3.05 | 3.30 | 2.06 | 3.39 | 2.92 | 2.91 | 2.78 | 2.89 | 3.27 | 3.76 | 2.05 | 2.73 | 1.76 |
| 1983 | 4.39 | 4.78 | 2.98 | 4.91 | 4.23 | 4.21 | 4.02 | 4.17 | 4.73 | 5.43 | 2.95 | 3.95 | 2.54 |
| 1984 | 3.89 | 4.27 | 2.65 | 4.37 | 3.77 | 3.76 | 3.59 | 3.70 | 4.23 | 4.86 | 2.63 | 3.53 | 2.25 |
| 1985 | 3.50 | 3.82 | 2.37 | 3.91 | 3.38 | 3.36 | 3.21 | 3.31 | 3.78 | 4.34 | 2.34 | 3.15 | 2.01 |
| 1986 | 5.16 | 5.13 | 3.32 | 5.43 | 4.55 | 4.55 | 4.40 | 4.71 | 5.07 | 5.85 | 3.31 | 4.27 | 2.89 |
| 1987 | 4.31 | 4.44 | 2.80 | 4.64 | 3.93 | 3.90 | 3.73 | 3.95 | 4.37 | 5.03 | 2.73 | 3.68 | 2.40 |
| 1988 | 3.26 | 3.26 | 2.12 | 3.46 | 2.90 | 2.91 | 2.81 | 2.99 | 3.24 | 3.72 | 2.14 | 2.71 | 1.84 |
| 1989 | 4.55 | 4.58 | 2.92 | 4.78 | 4.06 | 4.05 | 3.89 | 4.13 | 4.53 | 5.22 | 2.90 | 3.81 | 2.52 |
| 1990 | 3.52 | 3.64 | 2.31 | 3.74 | 3.23 | 3.22 | 3.10 | 3.24 | 3.61 | 4.16 | 2.31 | 3.02 | 1.98 |
| 1991 | 2.82 | 2.80 | 1.79 | 2.94 | 2.48 | 2.47 | 2.38 | 2.54 | 2.76 | 3.18 | 1.76 | 2.33 | 1.55 |
| 1992 | 1.52 | 1.41 | 0.91 | 1.45 | 1.25 | 1.25 | 1.21 | 1.28 | 1.40 | 1.61 | 0.92 | 1.17 | 0.79 |
| 1993 | 1.59 | 1.45 | 0.94 | 1.51 | 1.29 | 1.29 | 1.25 | 1.32 | 1.44 | 1.66 | 0.95 | 1.21 | 0.82 |
| 1994 | 1.75 | 1.54 | 1.00 | 1.60 | 1.37 | 1.37 | 1.33 | 1.42 | 1.53 | 1.76 | 1.02 | 1.29 | 0.88 |
| 1995 | 2.37 | 2.16 | 1.39 | 2.24 | 1.92 | 1.92 | 1.85 | 1.96 | 2.15 | 2.47 | 1.40 | 1.80 | 1.20 |
| 1996 | 2.08 | 1.90 | 1.23 | 1.97 | 1.69 | 1.69 | 1.63 | 1.74 | 1.88 | 2.17 | 1.23 | 1.58 | 1.07 |
| 1997 | 2.10 | 1.90 | 1.21 | 1.97 | 1.68 | 1.68 | 1.61 | 1.72 | 1.87 | 2.16 | 1.21 | 1.58 | 1.05 |
| 1998 | 2.91 | 2.70 | 1.71 | 2.78 | 2.39 | 2.38 | 2.28 | 2.40 | 2.67 | 3.07 | 1.69 | 2.24 | 1.47 |
| 1999 | 2.69 | 2.52 | 1.61 | 2.60 | 2.24 | 2.24 | 2.15 | 2.26 | 2.50 | 2.88 | 1.61 | 2.10 | 1.39 |
| 2000 | 3.03 | 2.91 | 1.86 | 2.97 | 2.58 | 2.59 | 2.49 | 2.60 | 2.89 | 3.33 | 1.88 | 2.42 | 1.60 |
| 2001 | 2.89 | 2.73 | 1.73 | 2.81 | 2.42 | 2.41 | 2.32 | 2.44 | 2.70 | 3.11 | 1.73 | 2.26 | 1.49 |
| 2002 | 3.74 | 3.49 | 2.22 | 3.61 | 3.09 | 3.08 | 2.97 | 3.13 | 3.45 | 3.97 | 2.21 | 2.90 | 1.91 |
| 2003 | 3.39 | 3.20 | 2.01 | 3.30 | 2.83 | 2.82 | 2.70 | 2.83 | 3.16 | 3.64 | 1.98 | 2.65 | 1.72 |
| 2004 | 2.65 | 2.48 | 1.56 | 2.56 | 2.20 | 2.18 | 2.10 | 2.20 | 2.45 | 2.82 | 1.55 | 2.06 | 1.34 |

Table 8.7. Characterization of minimum $H_p(10)$ and minimum absorbed organ doses for reactor plant workers

| | | Minimum absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 0.13 | 0.08 | 0.08 | 0.13 | 0.08 | 0.08 | 0.09 | 0.12 | 0.08 | 0.11 | 0.09 | 0.07 | 0.08 |
| 1949 | 2.36 | 1.10 | 1.12 | 1.87 | 1.05 | 1.17 | 1.28 | 1.65 | 1.17 | 1.33 | 1.30 | 1.01 | 1.17 |
| 1950 | 1.18 | 0.18 | 0.19 | 0.31 | 0.18 | 0.19 | 0.21 | 0.27 | 0.19 | 0.22 | 0.22 | 0.17 | 0.19 |
| 1951 | 0.68 | 0.21 | 0.21 | 0.36 | 0.20 | 0.22 | 0.24 | 0.31 | 0.22 | 0.25 | 0.25 | 0.19 | 0.22 |
| 1952 | 0.32 | 0.05 | 0.05 | 0.09 | 0.05 | 0.06 | 0.06 | 0.08 | 0.06 | 0.06 | 0.06 | 0.05 | 0.06 |
| 1953 | 0.17 | 0.10 | 0.09 | 0.15 | 0.10 | 0.10 | 0.11 | 0.14 | 0.11 | 0.12 | 0.10 | 0.09 | 0.09 |
| 1954 | 0.18 | 0.02 | 0.02 | 0.03 | 0.02 | 0.02 | 0.02 | 0.03 | 0.02 | 0.02 | 0.02 | 0.02 | 0.02 |
| 1955 | 0.09 | 0.04 | 0.04 | 0.07 | 0.04 | 0.04 | 0.05 | 0.06 | 0.04 | 0.05 | 0.05 | 0.04 | 0.04 |
| 1956 | 0.45 | 0.04 | 0.04 | 0.07 | 0.04 | 0.04 | 0.05 | 0.06 | 0.04 | 0.05 | 0.05 | 0.04 | 0.04 |
| 1957 | 0.09 | 0.03 | 0.02 | 0.04 | 0.02 | 0.02 | 0.02 | 0.04 | 0.03 | 0.04 | 0.02 | 0.02 | 0.02 |
| 1958 | 0.08 | 0.02 | 0.02 | 0.03 | 0.02 | 0.02 | 0.02 | 0.03 | 0.02 | 0.03 | 0.02 | 0.02 | 0.01 |
| 1959 | 0.09 | 0.05 | 0.06 | 0.09 | 0.05 | 0.06 | 0.06 | 0.08 | 0.06 | 0.07 | 0.06 | 0.05 | 0.06 |
| 1960 | 0.37 | 0.22 | 0.20 | 0.34 | 0.20 | 0.22 | 0.23 | 0.31 | 0.22 | 0.28 | 0.23 | 0.20 | 0.20 |
| 1961 | 0.23 | 0.07 | 0.07 | 0.11 | 0.07 | 0.07 | 0.08 | 0.10 | 0.07 | 0.09 | 0.08 | 0.07 | 0.07 |
| 1962 | 0.11 | 0.07 | 0.07 | 0.11 | 0.07 | 0.07 | 0.08 | 0.10 | 0.07 | 0.09 | 0.08 | 0.07 | 0.07 |
| 1963 | 0.11 | 0.07 | 0.05 | 0.09 | 0.06 | 0.06 | 0.06 | 0.09 | 0.07 | 0.09 | 0.06 | 0.06 | 0.05 |
| 1964 | 0.12 | 0.07 | 0.07 | 0.12 | 0.07 | 0.07 | 0.08 | 0.11 | 0.07 | 0.08 | 0.08 | 0.06 | 0.07 |
| 1965 | 0.23 | 0.05 | 0.05 | 0.08 | 0.05 | 0.05 | 0.06 | 0.07 | 0.05 | 0.06 | 0.06 | 0.04 | 0.05 |
| 1966 | 0.11 | 0.07 | 0.07 | 0.11 | 0.07 | 0.07 | 0.08 | 0.10 | 0.07 | 0.09 | 0.07 | 0.06 | 0.07 |
| 1967 | 0.12 | 0.06 | 0.06 | 0.10 | 0.06 | 0.06 | 0.07 | 0.09 | 0.06 | 0.07 | 0.07 | 0.05 | 0.06 |
| 1968 | 0.23 | 0.01 | 0.01 | 0.02 | 0.01 | 0.01 | 0.01 | 0.02 | 0.01 | 0.02 | 0.01 | 0.01 | 0.01 |
| 1969 | 0.12 | 0.07 | 0.06 | 0.10 | 0.06 | 0.07 | 0.07 | 0.10 | 0.07 | 0.08 | 0.07 | 0.06 | 0.06 |
| 1970 | 0.12 | 0.07 | 0.07 | 0.11 | 0.06 | 0.07 | 0.08 | 0.10 | 0.07 | 0.08 | 0.08 | 0.06 | 0.07 |
| 1971 | 0.20 | 0.13 | 0.12 | 0.19 | 0.13 | 0.13 | 0.13 | 0.20 | 0.14 | 0.16 | 0.12 | 0.12 | 0.10 |
| 1972 | 0.22 | 0.02 | 0.01 | 0.02 | 0.02 | 0.02 | 0.02 | 0.02 | 0.02 | 0.02 | 0.01 | 0.01 | 0.01 |
| 1973 | 0.22 | 0.13 | 0.13 | 0.22 | 0.13 | 0.14 | 0.15 | 0.21 | 0.14 | 0.16 | 0.15 | 0.12 | 0.13 |
| 1974 | 0.24 | 0.02 | 0.03 | 0.04 | 0.02 | 0.03 | 0.03 | 0.04 | 0.03 | 0.03 | 0.03 | 0.02 | 0.03 |
| 1975 | 0.24 | 0.10 | 0.09 | 0.16 | 0.09 | 0.10 | 0.11 | 0.15 | 0.10 | 0.12 | 0.11 | 0.09 | 0.09 |
| 1976 | 0.21 | 0.04 | 0.04 | 0.06 | 0.04 | 0.04 | 0.04 | 0.06 | 0.04 | 0.04 | 0.04 | 0.03 | 0.04 |
| 1977 | 0.24 | 0.05 | 0.05 | 0.08 | 0.05 | 0.05 | 0.06 | 0.07 | 0.05 | 0.06 | 0.06 | 0.05 | 0.05 |

| | | Minimum absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1978 | 0.12 | 0.01 | 0.01 | 0.02 | 0.01 | 0.01 | 0.01 | 0.02 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 |
| 1979 | 0.11 | 0.07 | 0.06 | 0.09 | 0.06 | 0.06 | 0.06 | 0.09 | 0.07 | 0.10 | 0.06 | 0.06 | 0.05 |
| 1980 | 0.24 | 0.14 | 0.14 | 0.24 | 0.13 | 0.15 | 0.16 | 0.22 | 0.15 | 0.17 | 0.17 | 0.13 | 0.15 |
| 1981 | 0.12 | 0.02 | 0.02 | 0.03 | 0.02 | 0.02 | 0.02 | 0.03 | 0.02 | 0.02 | 0.02 | 0.02 | 0.02 |
| 1982 | 0.24 | 0.05 | 0.05 | 0.09 | 0.05 | 0.06 | 0.06 | 0.08 | 0.06 | 0.07 | 0.06 | 0.05 | 0.06 |
| 1983 | 0.21 | 0.15 | 0.13 | 0.20 | 0.14 | 0.14 | 0.14 | 0.21 | 0.15 | 0.18 | 0.13 | 0.13 | 0.11 |
| 1984 | 0.24 | 0.04 | 0.04 | 0.06 | 0.03 | 0.04 | 0.04 | 0.05 | 0.04 | 0.04 | 0.04 | 0.03 | 0.04 |
| 1985 | 0.23 | 0.15 | 0.14 | 0.23 | 0.14 | 0.15 | 0.16 | 0.21 | 0.15 | 0.18 | 0.15 | 0.14 | 0.13 |
| 1986 | 0.21 | 0.13 | 0.13 | 0.20 | 0.12 | 0.13 | 0.14 | 0.19 | 0.13 | 0.15 | 0.13 | 0.12 | 0.11 |
| 1987 | 0.43 | 0.21 | 0.17 | 0.28 | 0.19 | 0.19 | 0.19 | 0.28 | 0.20 | 0.29 | 0.18 | 0.19 | 0.15 |
| 1988 | 0.36 | 0.21 | 0.22 | 0.35 | 0.20 | 0.23 | 0.25 | 0.33 | 0.22 | 0.25 | 0.25 | 0.19 | 0.21 |
| 1989 | 0.24 | 0.07 | 0.08 | 0.13 | 0.07 | 0.08 | 0.09 | 0.11 | 0.08 | 0.09 | 0.09 | 0.07 | 0.08 |
| 1990 | 0.24 | 0.08 | 0.06 | 0.10 | 0.07 | 0.07 | 0.07 | 0.10 | 0.07 | 0.11 | 0.07 | 0.07 | 0.05 |
| 1991 | 0.24 | 0.05 | 0.05 | 0.08 | 0.04 | 0.05 | 0.05 | 0.07 | 0.05 | 0.06 | 0.06 | 0.04 | 0.05 |
| 1992 | 0.18 | 0.07 | 0.07 | 0.12 | 0.07 | 0.08 | 0.08 | 0.11 | 0.08 | 0.09 | 0.09 | 0.07 | 0.08 |
| 1993 | 0.18 | 0.09 | 0.09 | 0.15 | 0.08 | 0.09 | 0.10 | 0.13 | 0.09 | 0.11 | 0.10 | 0.08 | 0.08 |
| 1994 | 0.27 | 0.13 | 0.13 | 0.22 | 0.13 | 0.14 | 0.15 | 0.20 | 0.14 | 0.16 | 0.16 | 0.12 | 0.14 |
| 1995 | 0.27 | 0.13 | 0.13 | 0.22 | 0.13 | 0.14 | 0.15 | 0.20 | 0.14 | 0.16 | 0.16 | 0.12 | 0.14 |
| 1996 | 0.18 | 0.08 | 0.08 | 0.13 | 0.07 | 0.08 | 0.09 | 0.12 | 0.08 | 0.09 | 0.09 | 0.07 | 0.08 |
| 1997 | 0.25 | 0.12 | 0.12 | 0.20 | 0.11 | 0.13 | 0.14 | 0.18 | 0.13 | 0.14 | 0.14 | 0.11 | 0.13 |
| 1998 | 0.23 | 0.11 | 0.11 | 0.19 | 0.10 | 0.12 | 0.13 | 0.16 | 0.12 | 0.13 | 0.13 | 0.10 | 0.11 |
| 1999 | 0.61 | 0.29 | 0.30 | 0.50 | 0.28 | 0.31 | 0.34 | 0.44 | 0.31 | 0.35 | 0.35 | 0.27 | 0.31 |
| 2000 | 0.13 | 0.06 | 0.06 | 0.10 | 0.06 | 0.06 | 0.07 | 0.09 | 0.06 | 0.07 | 0.07 | 0.06 | 0.06 |
| 2001 | 0.12 | 0.06 | 0.06 | 0.10 | 0.05 | 0.06 | 0.07 | 0.09 | 0.06 | 0.07 | 0.07 | 0.05 | 0.06 |
| 2002 | 0.14 | 0.07 | 0.07 | 0.11 | 0.06 | 0.07 | 0.08 | 0.10 | 0.07 | 0.08 | 0.08 | 0.06 | 0.07 |
| 2003 | 0.18 | 0.08 | 0.08 | 0.13 | 0.07 | 0.08 | 0.09 | 0.12 | 0.08 | 0.09 | 0.09 | 0.07 | 0.08 |
| 2004 | 0.19 | 0.09 | 0.10 | 0.16 | 0.09 | 0.10 | 0.11 | 0.15 | 0.10 | 0.11 | 0.11 | 0.09 | 0.09 |

Table 8.8. Characterization of minimum $H_p(10)$ and minimum absorbed organ doses for radiochemical plant workers.

| | | Minimum absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Year** | **Hp10, mSv** | **Bladder** | **Red Marrow** | **Bone Surface** | **Colon** | **Liver** | **Lung** | **Skin** | **Stomach** | **Thyroid** | **Brain** | **Lower Lg Intestine** | **Kidney** |
| 1948 | 2.05 | 1.62 | 1.31 | 2.01 | 1.49 | 1.55 | 1.60 | 1.89 | 1.64 | 1.96 | 1.44 | 1.42 | 1.23 |
| 1949 | 0.09 | 0.08 | 0.06 | 0.08 | 0.08 | 0.08 | 0.08 | 0.08 | 0.08 | 0.10 | 0.06 | 0.07 | 0.05 |
| 1950 | 0.46 | 0.12 | 0.09 | 0.12 | 0.11 | 0.11 | 0.11 | 0.12 | 0.12 | 0.14 | 0.09 | 0.10 | 0.08 |
| 1951 | 0.38 | 0.42 | 0.28 | 0.38 | 0.38 | 0.38 | 0.38 | 0.38 | 0.42 | 0.49 | 0.30 | 0.36 | 0.25 |
| 1952 | 0.38 | 0.19 | 0.19 | 0.33 | 0.18 | 0.20 | 0.22 | 0.33 | 0.20 | 0.24 | 0.22 | 0.17 | 0.20 |
| 1953 | 0.21 | 0.06 | 0.06 | 0.09 | 0.06 | 0.06 | 0.07 | 0.09 | 0.06 | 0.07 | 0.07 | 0.05 | 0.06 |
| 1954 | 0.38 | 0.08 | 0.08 | 0.12 | 0.07 | 0.08 | 0.09 | 0.12 | 0.08 | 0.09 | 0.09 | 0.07 | 0.08 |
| 1955 | 0.31 | 0.13 | 0.09 | 0.13 | 0.12 | 0.12 | 0.12 | 0.13 | 0.13 | 0.15 | 0.10 | 0.11 | 0.08 |
| 1956 | 0.46 | 0.06 | 0.04 | 0.05 | 0.05 | 0.05 | 0.05 | 0.05 | 0.06 | 0.07 | 0.04 | 0.05 | 0.03 |
| 1957 | 0.45 | 0.46 | 0.29 | 0.37 | 0.41 | 0.41 | 0.40 | 0.38 | 0.45 | 0.52 | 0.31 | 0.39 | 0.26 |
| 1958 | 0.43 | 0.05 | 0.05 | 0.08 | 0.04 | 0.05 | 0.05 | 0.07 | 0.05 | 0.06 | 0.05 | 0.04 | 0.05 |
| 1959 | 0.38 | 0.18 | 0.14 | 0.17 | 0.18 | 0.19 | 0.18 | 0.17 | 0.19 | 0.22 | 0.14 | 0.17 | 0.12 |
| 1960 | 0.38 | 0.22 | 0.23 | 0.37 | 0.21 | 0.24 | 0.26 | 0.35 | 0.23 | 0.27 | 0.26 | 0.20 | 0.23 |
| 1961 | 0.58 | 0.16 | 0.10 | 0.13 | 0.14 | 0.14 | 0.13 | 0.13 | 0.15 | 0.18 | 0.10 | 0.13 | 0.09 |
| 1962 | 0.25 | 0.11 | 0.09 | 0.12 | 0.11 | 0.12 | 0.12 | 0.12 | 0.12 | 0.14 | 0.10 | 0.10 | 0.08 |
| 1963 | 0.12 | 0.03 | 0.03 | 0.05 | 0.03 | 0.03 | 0.03 | 0.04 | 0.03 | 0.03 | 0.03 | 0.03 | 0.03 |
| 1964 | 0.12 | 0.17 | 0.12 | 0.15 | 0.16 | 0.17 | 0.16 | 0.15 | 0.18 | 0.20 | 0.12 | 0.15 | 0.10 |
| 1965 | 0.23 | 0.06 | 0.06 | 0.10 | 0.06 | 0.07 | 0.07 | 0.10 | 0.06 | 0.07 | 0.07 | 0.06 | 0.06 |
| 1966 | 0.12 | 0.05 | 0.03 | 0.04 | 0.04 | 0.04 | 0.04 | 0.04 | 0.05 | 0.05 | 0.03 | 0.04 | 0.03 |
| 1967 | 0.12 | 0.06 | 0.04 | 0.05 | 0.06 | 0.06 | 0.05 | 0.05 | 0.06 | 0.07 | 0.04 | 0.05 | 0.03 |
| 1968 | 0.12 | 0.01 | 0.01 | 0.02 | 0.01 | 0.01 | 0.01 | 0.02 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 |
| 1969 | 0.12 | 0.07 | 0.07 | 0.09 | 0.06 | 0.07 | 0.08 | 0.09 | 0.07 | 0.08 | 0.08 | 0.06 | 0.06 |
| 1970 | 0.25 | 0.09 | 0.06 | 0.08 | 0.08 | 0.08 | 0.08 | 0.08 | 0.09 | 0.11 | 0.06 | 0.08 | 0.05 |
| 1971 | 0.25 | 0.14 | 0.10 | 0.13 | 0.13 | 0.14 | 0.13 | 0.13 | 0.15 | 0.17 | 0.10 | 0.13 | 0.09 |
| 1972 | 0.20 | 0.02 | 0.01 | 0.02 | 0.01 | 0.01 | 0.01 | 0.02 | 0.02 | 0.02 | 0.01 | 0.01 | 0.01 |
| 1973 | 0.47 | 0.09 | 0.06 | 0.08 | 0.08 | 0.08 | 0.08 | 0.08 | 0.09 | 0.11 | 0.06 | 0.08 | 0.05 |
| 1974 | 0.26 | 0.20 | 0.13 | 0.16 | 0.18 | 0.17 | 0.17 | 0.16 | 0.19 | 0.22 | 0.13 | 0.16 | 0.11 |
| 1975 | 0.24 | 0.03 | 0.02 | 0.03 | 0.03 | 0.03 | 0.03 | 0.03 | 0.03 | 0.04 | 0.02 | 0.03 | 0.02 |
| 1976 | 0.24 | 0.03 | 0.02 | 0.03 | 0.03 | 0.03 | 0.03 | 0.03 | 0.03 | 0.04 | 0.02 | 0.03 | 0.02 |
| 1977 | 0.24 | 0.01 | 0.01 | 0.02 | 0.01 | 0.01 | 0.01 | 0.02 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 |

| | | Minimum absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1978 | 0.12 | 0.08 | 0.07 | 0.10 | 0.08 | 0.09 | 0.09 | 0.10 | 0.09 | 0.10 | 0.08 | 0.07 | 0.07 |
| 1979 | 0.13 | 0.13 | 0.13 | 0.16 | 0.12 | 0.13 | 0.15 | 0.16 | 0.13 | 0.15 | 0.13 | 0.12 | 0.11 |
| 1980 | 0.49 | 0.28 | 0.29 | 0.47 | 0.27 | 0.30 | 0.33 | 0.44 | 0.29 | 0.34 | 0.33 | 0.25 | 0.29 |
| 1981 | 0.26 | 0.39 | 0.25 | 0.32 | 0.35 | 0.35 | 0.34 | 0.32 | 0.39 | 0.45 | 0.26 | 0.33 | 0.22 |
| 1982 | 0.38 | 0.35 | 0.36 | 0.47 | 0.33 | 0.38 | 0.41 | 0.48 | 0.37 | 0.42 | 0.39 | 0.32 | 0.33 |
| 1983 | 0.26 | 0.16 | 0.10 | 0.13 | 0.15 | 0.14 | 0.14 | 0.13 | 0.16 | 0.19 | 0.11 | 0.14 | 0.09 |
| 1984 | 0.24 | 0.09 | 0.10 | 0.16 | 0.09 | 0.10 | 0.11 | 0.15 | 0.10 | 0.11 | 0.11 | 0.08 | 0.10 |
| 1985 | 0.24 | 0.11 | 0.07 | 0.09 | 0.10 | 0.10 | 0.10 | 0.09 | 0.11 | 0.13 | 0.08 | 0.10 | 0.06 |
| 1986 | 0.24 | 0.13 | 0.10 | 0.14 | 0.12 | 0.12 | 0.12 | 0.13 | 0.14 | 0.16 | 0.10 | 0.11 | 0.09 |
| 1987 | 0.36 | 0.21 | 0.22 | 0.32 | 0.20 | 0.23 | 0.25 | 0.32 | 0.22 | 0.25 | 0.25 | 0.19 | 0.22 |
| 1988 | 0.12 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 |
| 1989 | 0.12 | 0.07 | 0.07 | 0.12 | 0.07 | 0.08 | 0.08 | 0.11 | 0.07 | 0.08 | 0.08 | 0.06 | 0.07 |
| 1990 | 0.13 | 0.20 | 0.13 | 0.16 | 0.18 | 0.17 | 0.17 | 0.16 | 0.19 | 0.22 | 0.13 | 0.16 | 0.11 |
| 1991 | 0.22 | 0.08 | 0.09 | 0.14 | 0.08 | 0.09 | 0.10 | 0.13 | 0.09 | 0.10 | 0.10 | 0.08 | 0.09 |
| 1992 | 0.18 | 0.09 | 0.09 | 0.15 | 0.08 | 0.09 | 0.10 | 0.13 | 0.09 | 0.11 | 0.10 | 0.08 | 0.09 |
| 1993 | 0.18 | 0.04 | 0.02 | 0.03 | 0.03 | 0.03 | 0.03 | 0.03 | 0.04 | 0.04 | 0.02 | 0.03 | 0.02 |
| 1994 | 0.18 | 0.22 | 0.14 | 0.18 | 0.20 | 0.19 | 0.19 | 0.18 | 0.22 | 0.25 | 0.15 | 0.18 | 0.12 |
| 1995 | 0.18 | 0.09 | 0.09 | 0.15 | 0.08 | 0.09 | 0.10 | 0.13 | 0.09 | 0.11 | 0.10 | 0.08 | 0.09 |
| 1996 | 0.13 | 0.11 | 0.10 | 0.12 | 0.11 | 0.12 | 0.13 | 0.13 | 0.12 | 0.14 | 0.10 | 0.10 | 0.09 |
| 1997 | 0.23 | 0.12 | 0.12 | 0.20 | 0.12 | 0.13 | 0.14 | 0.18 | 0.13 | 0.15 | 0.14 | 0.11 | 0.12 |
| 1998 | 0.09 | 0.07 | 0.07 | 0.09 | 0.06 | 0.07 | 0.08 | 0.09 | 0.07 | 0.08 | 0.07 | 0.06 | 0.06 |
| 1999 | 0.09 | 0.11 | 0.07 | 0.09 | 0.10 | 0.10 | 0.09 | 0.09 | 0.11 | 0.12 | 0.07 | 0.09 | 0.06 |
| 2000 | 0.09 | 0.07 | 0.07 | 0.09 | 0.07 | 0.08 | 0.08 | 0.09 | 0.07 | 0.09 | 0.07 | 0.06 | 0.06 |
| 2001 | 0.09 | 0.10 | 0.07 | 0.09 | 0.10 | 0.10 | 0.09 | 0.09 | 0.11 | 0.12 | 0.07 | 0.09 | 0.06 |
| 2002 | 0.19 | 0.11 | 0.12 | 0.19 | 0.11 | 0.12 | 0.13 | 0.18 | 0.12 | 0.14 | 0.13 | 0.10 | 0.12 |
| 2003 | 0.29 | 0.29 | 0.22 | 0.28 | 0.28 | 0.31 | 0.30 | 0.29 | 0.30 | 0.35 | 0.23 | 0.27 | 0.20 |
| 2004 | 0.11 | 0.10 | 0.08 | 0.11 | 0.09 | 0.10 | 0.11 | 0.11 | 0.10 | 0.12 | 0.09 | 0.09 | 0.07 |

Table 8.9. Characterization of minimum $H_p(10)$ and minimum absorbed organ doses for plutonium plant workers.

| | | Minimum absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| \Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 32.83 | 17.79 | 16.88 | 27.99 | 16.48 | 17.88 | 19.24 | 25.98 | 18.16 | 22.73 | 18.96 | 16.11 | 16.60 |
| 1949 | 0.09 | 0.05 | 0.05 | 0.08 | 0.05 | 0.06 | 0.06 | 0.08 | 0.05 | 0.06 | 0.06 | 0.05 | 0.05 |
| 1950 | 0.18 | 0.12 | 0.08 | 0.13 | 0.11 | 0.11 | 0.10 | 0.12 | 0.12 | 0.13 | 0.08 | 0.10 | 0.08 |
| 1951 | 0.11 | 0.07 | 0.07 | 0.11 | 0.06 | 0.07 | 0.08 | 0.10 | 0.07 | 0.08 | 0.08 | 0.06 | 0.07 |
| 1952 | 0.10 | 0.09 | 0.06 | 0.08 | 0.08 | 0.08 | 0.08 | 0.08 | 0.09 | 0.10 | 0.06 | 0.07 | 0.05 |
| 1953 | 0.10 | 0.10 | 0.07 | 0.10 | 0.09 | 0.09 | 0.09 | 0.10 | 0.10 | 0.12 | 0.08 | 0.09 | 0.06 |
| 1954 | 0.38 | 0.22 | 0.23 | 0.37 | 0.21 | 0.24 | 0.26 | 0.35 | 0.23 | 0.27 | 0.22 | 0.20 | 0.22 |
| 1955 | 0.16 | 0.11 | 0.08 | 0.13 | 0.10 | 0.10 | 0.10 | 0.12 | 0.11 | 0.13 | 0.07 | 0.10 | 0.07 |
| 1956 | 0.08 | 0.02 | 0.01 | 0.02 | 0.02 | 0.02 | 0.02 | 0.02 | 0.02 | 0.02 | 0.01 | 0.02 | 0.01 |
| 1957 | 0.31 | 0.18 | 0.17 | 0.25 | 0.17 | 0.19 | 0.21 | 0.23 | 0.19 | 0.21 | 0.18 | 0.16 | 0.15 |
| 1958 | 0.08 | 0.06 | 0.04 | 0.06 | 0.05 | 0.05 | 0.05 | 0.06 | 0.05 | 0.06 | 0.04 | 0.05 | 0.04 |
| 1959 | 0.16 | 0.18 | 0.13 | 0.19 | 0.17 | 0.19 | 0.19 | 0.18 | 0.19 | 0.21 | 0.14 | 0.16 | 0.11 |
| 1960 | 0.16 | 0.12 | 0.08 | 0.13 | 0.11 | 0.11 | 0.11 | 0.12 | 0.12 | 0.14 | 0.08 | 0.11 | 0.08 |
| 1961 | 0.11 | 0.17 | 0.10 | 0.16 | 0.15 | 0.15 | 0.14 | 0.13 | 0.17 | 0.20 | 0.10 | 0.14 | 0.08 |
| 1962 | 0.22 | 0.03 | 0.03 | 0.06 | 0.03 | 0.03 | 0.04 | 0.05 | 0.03 | 0.04 | 0.04 | 0.03 | 0.03 |
| 1963 | 0.11 | 0.13 | 0.10 | 0.16 | 0.13 | 0.14 | 0.14 | 0.13 | 0.14 | 0.16 | 0.10 | 0.12 | 0.08 |
| 1964 | 0.22 | 0.14 | 0.13 | 0.21 | 0.13 | 0.14 | 0.15 | 0.20 | 0.14 | 0.17 | 0.15 | 0.12 | 0.12 |
| 1965 | 0.12 | 0.04 | 0.05 | 0.07 | 0.04 | 0.05 | 0.05 | 0.07 | 0.05 | 0.05 | 0.05 | 0.04 | 0.05 |
| 1966 | 0.23 | 0.02 | 0.01 | 0.02 | 0.02 | 0.02 | 0.02 | 0.02 | 0.02 | 0.03 | 0.01 | 0.02 | 0.01 |
| 1967 | 0.11 | 0.07 | 0.06 | 0.10 | 0.06 | 0.07 | 0.07 | 0.08 | 0.07 | 0.08 | 0.05 | 0.06 | 0.05 |
| 1968 | 0.11 | 0.05 | 0.04 | 0.05 | 0.05 | 0.05 | 0.05 | 0.05 | 0.05 | 0.06 | 0.04 | 0.04 | 0.03 |
| 1969 | 0.11 | 0.09 | 0.06 | 0.10 | 0.08 | 0.07 | 0.07 | 0.08 | 0.08 | 0.10 | 0.05 | 0.07 | 0.05 |
| 1970 | 0.12 | 0.07 | 0.07 | 0.12 | 0.07 | 0.08 | 0.08 | 0.11 | 0.07 | 0.09 | 0.07 | 0.06 | 0.06 |
| 1971 | 0.11 | 0.07 | 0.06 | 0.10 | 0.06 | 0.07 | 0.07 | 0.08 | 0.07 | 0.08 | 0.05 | 0.06 | 0.05 |
| 1972 | 0.12 | 0.02 | 0.02 | 0.04 | 0.02 | 0.02 | 0.02 | 0.04 | 0.02 | 0.03 | 0.02 | 0.02 | 0.02 |
| 1973 | 0.17 | 0.10 | 0.10 | 0.16 | 0.10 | 0.10 | 0.11 | 0.14 | 0.11 | 0.12 | 0.11 | 0.09 | 0.09 |
| 1974 | 0.21 | 0.06 | 0.04 | 0.07 | 0.05 | 0.05 | 0.05 | 0.06 | 0.06 | 0.07 | 0.04 | 0.05 | 0.04 |
| 1975 | 0.16 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 |
| 1976 | 0.21 | 0.13 | 0.12 | 0.19 | 0.12 | 0.13 | 0.15 | 0.16 | 0.13 | 0.16 | 0.11 | 0.11 | 0.10 |
| 1977 | 0.21 | 0.05 | 0.04 | 0.06 | 0.05 | 0.05 | 0.05 | 0.05 | 0.05 | 0.06 | 0.03 | 0.05 | 0.03 |

| \Year | Hp10, mSv | Minimum absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1978 | 0.21 | 0.14 | 0.12 | 0.19 | 0.13 | 0.15 | 0.15 | 0.16 | 0.15 | 0.17 | 0.11 | 0.13 | 0.10 |
| 1979 | 0.23 | 0.06 | 0.03 | 0.06 | 0.05 | 0.05 | 0.05 | 0.04 | 0.06 | 0.07 | 0.03 | 0.05 | 0.03 |
| 1980 | 0.21 | 0.10 | 0.07 | 0.12 | 0.09 | 0.09 | 0.09 | 0.10 | 0.10 | 0.12 | 0.07 | 0.09 | 0.06 |
| 1981 | 0.11 | 0.11 | 0.07 | 0.11 | 0.10 | 0.10 | 0.10 | 0.10 | 0.11 | 0.13 | 0.08 | 0.09 | 0.06 |
| 1982 | 0.23 | 0.14 | 0.12 | 0.20 | 0.13 | 0.15 | 0.15 | 0.18 | 0.15 | 0.17 | 0.11 | 0.13 | 0.11 |
| 1983 | 0.21 | 0.02 | 0.02 | 0.04 | 0.02 | 0.02 | 0.02 | 0.04 | 0.02 | 0.03 | 0.02 | 0.02 | 0.02 |
| 1984 | 0.21 | 0.13 | 0.12 | 0.19 | 0.12 | 0.13 | 0.15 | 0.16 | 0.13 | 0.16 | 0.12 | 0.11 | 0.10 |
| 1985 | 0.11 | 0.09 | 0.06 | 0.10 | 0.08 | 0.08 | 0.08 | 0.09 | 0.09 | 0.10 | 0.06 | 0.08 | 0.05 |
| 1986 | 0.11 | 0.05 | 0.03 | 0.05 | 0.05 | 0.05 | 0.04 | 0.04 | 0.05 | 0.06 | 0.03 | 0.04 | 0.02 |
| 1987 | 0.11 | 0.07 | 0.05 | 0.08 | 0.07 | 0.07 | 0.06 | 0.07 | 0.07 | 0.08 | 0.05 | 0.06 | 0.04 |
| 1988 | 0.21 | 0.14 | 0.09 | 0.14 | 0.12 | 0.12 | 0.12 | 0.13 | 0.13 | 0.15 | 0.09 | 0.11 | 0.08 |
| 1989 | 0.24 | 0.02 | 0.01 | 0.02 | 0.02 | 0.02 | 0.02 | 0.02 | 0.02 | 0.02 | 0.01 | 0.02 | 0.01 |
| 1990 | 0.21 | 0.13 | 0.12 | 0.19 | 0.12 | 0.13 | 0.15 | 0.16 | 0.13 | 0.16 | 0.11 | 0.11 | 0.10 |
| 1991 | 0.11 | 0.09 | 0.06 | 0.10 | 0.08 | 0.08 | 0.08 | 0.09 | 0.09 | 0.10 | 0.06 | 0.08 | 0.05 |
| 1992 | 0.11 | 0.06 | 0.05 | 0.09 | 0.05 | 0.06 | 0.06 | 0.08 | 0.06 | 0.07 | 0.06 | 0.05 | 0.05 |
| 1993 | 0.09 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 | 0.01 |
| 1994 | 0.09 | 0.06 | 0.04 | 0.07 | 0.05 | 0.05 | 0.05 | 0.06 | 0.06 | 0.06 | 0.04 | 0.05 | 0.04 |
| 1995 | 0.09 | 0.09 | 0.06 | 0.09 | 0.08 | 0.08 | 0.08 | 0.08 | 0.09 | 0.10 | 0.06 | 0.07 | 0.05 |
| 1996 | 0.13 | 0.05 | 0.06 | 0.09 | 0.05 | 0.06 | 0.06 | 0.09 | 0.06 | 0.07 | 0.06 | 0.05 | 0.06 |
| 1997 | 0.17 | 0.10 | 0.07 | 0.10 | 0.09 | 0.09 | 0.09 | 0.09 | 0.10 | 0.11 | 0.07 | 0.08 | 0.06 |
| 1998 | 0.27 | 0.05 | 0.05 | 0.09 | 0.05 | 0.06 | 0.06 | 0.08 | 0.05 | 0.06 | 0.06 | 0.05 | 0.05 |
| 1999 | 0.27 | 0.11 | 0.11 | 0.20 | 0.10 | 0.11 | 0.13 | 0.19 | 0.11 | 0.13 | 0.13 | 0.10 | 0.11 |
| 2000 | 0.32 | 0.17 | 0.15 | 0.25 | 0.16 | 0.18 | 0.18 | 0.22 | 0.18 | 0.22 | 0.14 | 0.16 | 0.13 |
| 2001 | 0.15 | 0.22 | 0.12 | 0.20 | 0.19 | 0.19 | 0.18 | 0.16 | 0.22 | 0.25 | 0.12 | 0.17 | 0.10 |
| 2002 | 0.17 | 0.12 | 0.08 | 0.13 | 0.11 | 0.10 | 0.10 | 0.12 | 0.11 | 0.13 | 0.07 | 0.10 | 0.07 |
| 2003 | 0.20 | 0.14 | 0.09 | 0.16 | 0.12 | 0.12 | 0.12 | 0.14 | 0.13 | 0.15 | 0.09 | 0.12 | 0.08 |
| 2004 | 0.16 | 0.11 | 0.08 | 0.13 | 0.10 | 0.10 | 0.10 | 0.11 | 0.11 | 0.13 | 0.07 | 0.10 | 0.07 |

Table 8.10. Characteristics of maximum $H_p(10)$ and maximum absorbed organ doses for reactor plant workers

| | | Maximum absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 1765.78 | 1128.05 | 1064.20 | 1766.57 | 1042.91 | 1128.05 | 1213.18 | 1638.86 | 1149.33 | 1447.31 | 1191.90 | 1021.63 | 1042.91 |
| 1949 | 3673.33 | 2346.67 | 2213.84 | 3674.97 | 2169.56 | 2346.67 | 2523.78 | 3409.31 | 2390.95 | 3010.82 | 2479.50 | 2125.29 | 2169.56 |
| 1950 | 2399.79 | 1628.59 | 1365.43 | 2266.62 | 1431.19 | 1455.86 | 1556.59 | 2146.78 | 1554.57 | 2220.81 | 1529.29 | 1406.51 | 1338.13 |
| 1951 | 2627.92 | 2926.38 | 2047.16 | 2767.59 | 2659.80 | 2685.68 | 2670.23 | 2715.84 | 2933.64 | 3370.74 | 2183.62 | 2496.54 | 1859.14 |
| 1952 | 2400.08 | 2146.02 | 1553.76 | 2259.92 | 1908.18 | 1916.02 | 1889.75 | 2310.62 | 2099.96 | 2693.15 | 1613.22 | 1830.30 | 1333.44 |
| 1953 | 2273.40 | 1536.62 | 1233.95 | 1978.98 | 1350.36 | 1373.64 | 1373.64 | 2025.54 | 1466.77 | 2095.39 | 1280.51 | 1327.08 | 1047.69 |
| 1954 | 376.91 | 450.37 | 305.66 | 405.92 | 406.68 | 407.56 | 401.86 | 406.25 | 448.12 | 521.98 | 321.79 | 382.59 | 271.69 |
| 1955 | 460.73 | 557.24 | 364.88 | 467.81 | 502.64 | 500.17 | 488.74 | 472.76 | 553.97 | 637.47 | 381.55 | 470.46 | 320.86 |
| 1956 | 829.18 | 669.81 | 502.55 | 818.13 | 600.96 | 597.45 | 595.95 | 732.43 | 660.96 | 784.28 | 572.86 | 565.70 | 503.40 |
| 1957 | 299.60 | 266.54 | 191.42 | 274.35 | 242.50 | 247.31 | 247.31 | 261.43 | 268.34 | 309.51 | 206.44 | 227.48 | 174.29 |
| 1958 | 237.77 | 172.98 | 138.91 | 222.77 | 152.01 | 154.63 | 154.63 | 228.02 | 165.12 | 235.88 | 144.15 | 149.39 | 117.94 |
| 1959 | 278.42 | 223.05 | 157.63 | 254.41 | 194.92 | 197.25 | 193.72 | 249.11 | 217.27 | 287.85 | 163.90 | 186.24 | 133.28 |
| 1960 | 395.98 | 394.80 | 253.33 | 371.01 | 355.32 | 352.03 | 342.16 | 379.74 | 391.51 | 450.73 | 263.20 | 332.29 | 220.43 |
| 1961 | 216.01 | 194.59 | 131.35 | 200.51 | 175.29 | 174.22 | 170.02 | 188.86 | 193.14 | 223.09 | 146.52 | 164.20 | 126.37 |
| 1962 | 119.54 | 152.49 | 99.41 | 127.18 | 137.36 | 136.53 | 133.23 | 129.05 | 151.36 | 174.82 | 103.67 | 128.67 | 87.03 |
| 1963 | 126.47 | 74.94 | 59.88 | 96.04 | 67.84 | 68.57 | 68.33 | 98.30 | 74.75 | 101.69 | 65.73 | 64.40 | 58.81 |
| 1964 | 187.22 | 119.38 | 112.62 | 186.96 | 110.37 | 119.38 | 128.39 | 173.44 | 121.63 | 153.17 | 126.14 | 108.12 | 110.37 |
| 1965 | 102.86 | 74.94 | 60.18 | 100.48 | 65.86 | 67.00 | 68.62 | 98.79 | 71.54 | 102.20 | 69.85 | 64.72 | 62.49 |
| 1966 | 107.62 | 72.44 | 65.25 | 109.19 | 66.10 | 67.91 | 74.57 | 95.87 | 72.93 | 95.38 | 75.90 | 62.13 | 67.91 |
| 1967 | 137.94 | 79.66 | 81.51 | 133.38 | 75.95 | 85.21 | 92.62 | 124.11 | 83.36 | 96.33 | 94.48 | 72.25 | 83.36 |
| 1968 | 62.44 | 45.49 | 36.53 | 58.58 | 39.97 | 40.66 | 40.66 | 59.96 | 43.42 | 62.03 | 39.67 | 39.29 | 35.50 |
| 1969 | 80.76 | 58.84 | 47.25 | 75.77 | 51.71 | 52.60 | 52.60 | 77.56 | 56.16 | 80.23 | 49.03 | 50.81 | 40.12 |
| 1970 | 59.35 | 37.88 | 35.98 | 60.22 | 33.78 | 37.45 | 41.12 | 52.87 | 37.45 | 51.66 | 41.86 | 32.72 | 37.45 |
| 1971 | 57.68 | 37.14 | 34.97 | 58.52 | 32.83 | 36.40 | 39.97 | 51.39 | 36.40 | 50.64 | 40.68 | 32.07 | 36.40 |
| 1972 | 66.50 | 46.38 | 40.32 | 67.47 | 40.76 | 41.96 | 46.08 | 61.14 | 44.28 | 63.25 | 46.90 | 40.06 | 41.96 |
| 1973 | 71.51 | 42.47 | 43.35 | 72.55 | 40.70 | 45.12 | 49.55 | 63.70 | 45.12 | 53.39 | 50.43 | 38.93 | 45.12 |
| 1974 | 62.95 | 39.70 | 38.17 | 63.87 | 35.83 | 39.72 | 43.62 | 56.08 | 39.72 | 54.14 | 44.40 | 34.29 | 39.72 |
| 1975 | 121.17 | 75.35 | 73.46 | 122.93 | 68.96 | 76.46 | 83.95 | 107.94 | 76.46 | 102.75 | 85.45 | 65.96 | 76.46 |
| 1976 | 54.48 | 38.38 | 33.03 | 55.27 | 33.72 | 34.37 | 37.74 | 50.59 | 36.63 | 52.33 | 38.42 | 33.14 | 34.37 |
| 1977 | 57.59 | 34.79 | 34.92 | 58.43 | 32.78 | 36.34 | 39.90 | 51.31 | 36.34 | 47.44 | 40.62 | 31.35 | 36.34 |

| | | Maximum absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1978 | 49.68 | 36.19 | 29.63 | 49.58 | 31.81 | 32.35 | 33.86 | 47.71 | 34.55 | 49.35 | 34.46 | 31.26 | 30.84 |
| 1979 | 59.37 | 34.28 | 35.08 | 57.40 | 32.69 | 36.67 | 39.86 | 53.42 | 35.88 | 45.10 | 40.66 | 31.09 | 35.88 |
| 1980 | 57.34 | 38.45 | 34.77 | 58.18 | 33.79 | 36.18 | 39.73 | 51.08 | 36.71 | 52.44 | 40.44 | 33.21 | 36.18 |
| 1981 | 60.71 | 38.22 | 36.81 | 61.59 | 34.55 | 38.31 | 42.06 | 54.08 | 38.31 | 52.12 | 42.81 | 33.05 | 38.31 |
| 1982 | 65.00 | 41.45 | 39.10 | 64.91 | 38.32 | 41.45 | 44.58 | 60.22 | 42.23 | 53.39 | 43.79 | 37.54 | 38.32 |
| 1983 | 55.35 | 34.14 | 33.56 | 56.15 | 31.56 | 34.93 | 38.35 | 49.60 | 34.93 | 44.78 | 39.03 | 30.92 | 34.93 |
| 1984 | 43.63 | 28.16 | 26.45 | 44.27 | 24.83 | 27.53 | 30.23 | 38.87 | 27.53 | 38.40 | 30.77 | 24.32 | 27.53 |
| 1985 | 52.98 | 31.46 | 32.12 | 53.75 | 30.15 | 33.43 | 36.71 | 47.20 | 33.43 | 40.21 | 37.36 | 28.84 | 33.43 |
| 1986 | 292.85 | 200.30 | 176.17 | 292.44 | 176.02 | 186.74 | 200.83 | 271.30 | 191.20 | 273.14 | 197.31 | 172.99 | 172.65 |
| 1987 | 88.11 | 64.19 | 51.55 | 82.67 | 56.41 | 57.39 | 57.39 | 84.62 | 61.28 | 87.54 | 53.50 | 55.44 | 43.77 |
| 1988 | 40.25 | 29.33 | 23.55 | 37.77 | 25.77 | 26.22 | 26.22 | 38.66 | 27.99 | 39.99 | 24.44 | 25.33 | 20.00 |
| 1989 | 54.97 | 32.65 | 33.33 | 55.78 | 31.29 | 34.69 | 38.09 | 48.97 | 34.69 | 39.45 | 38.77 | 29.93 | 34.69 |
| 1990 | 56.74 | 41.34 | 33.20 | 53.24 | 36.33 | 36.96 | 36.96 | 54.49 | 39.46 | 56.37 | 34.45 | 35.70 | 28.19 |
| 1991 | 37.26 | 27.14 | 21.80 | 35.42 | 23.85 | 24.27 | 24.27 | 35.78 | 25.91 | 37.01 | 24.39 | 23.44 | 21.58 |
| 1992 | 23.75 | 13.57 | 11.64 | 19.48 | 11.93 | 12.13 | 13.30 | 17.89 | 12.96 | 18.51 | 13.54 | 11.72 | 12.11 |
| 1993 | 36.49 | 24.08 | 19.34 | 31.02 | 21.16 | 21.53 | 21.53 | 31.75 | 22.99 | 32.84 | 20.07 | 20.80 | 16.42 |
| 1994 | 17.29 | 9.37 | 8.47 | 14.18 | 8.50 | 8.82 | 9.68 | 12.45 | 9.37 | 10.78 | 9.86 | 7.97 | 8.82 |
| 1995 | 16.11 | 8.59 | 7.89 | 13.21 | 7.55 | 8.21 | 9.02 | 11.60 | 8.21 | 11.71 | 9.18 | 7.42 | 8.21 |
| 1996 | 22.35 | 14.75 | 11.85 | 19.00 | 12.96 | 13.19 | 13.19 | 19.44 | 14.08 | 20.11 | 12.29 | 12.74 | 10.06 |
| 1997 | 10.91 | 6.30 | 5.35 | 8.95 | 5.54 | 5.63 | 6.11 | 8.30 | 6.01 | 8.59 | 6.22 | 5.44 | 5.56 |
| 1998 | 14.89 | 9.83 | 7.89 | 12.65 | 8.63 | 8.78 | 8.78 | 12.95 | 9.38 | 13.40 | 8.19 | 8.49 | 6.70 |
| 1999 | 13.22 | 8.73 | 7.01 | 11.24 | 7.67 | 7.80 | 7.80 | 11.50 | 8.33 | 11.90 | 7.27 | 7.54 | 6.18 |
| 2000 | 12.51 | 6.01 | 6.13 | 10.26 | 5.76 | 6.38 | 7.01 | 9.01 | 6.38 | 7.26 | 7.13 | 5.51 | 6.38 |
| 2001 | 14.38 | 6.90 | 7.05 | 11.79 | 6.61 | 7.33 | 8.05 | 10.35 | 7.33 | 8.34 | 8.20 | 6.33 | 7.33 |
| 2002 | 9.96 | 6.57 | 5.28 | 8.46 | 5.77 | 5.87 | 5.87 | 8.66 | 6.27 | 8.96 | 5.48 | 5.67 | 4.48 |
| 2003 | 13.25 | 8.74 | 7.02 | 11.26 | 7.68 | 7.82 | 7.82 | 11.53 | 8.35 | 11.92 | 7.29 | 7.55 | 5.96 |
| 2004 | 15.72 | 10.38 | 8.33 | 13.37 | 9.12 | 9.28 | 9.28 | 13.68 | 9.91 | 14.15 | 8.65 | 8.96 | 7.08 |

Table 8.11. Characteristics of maximum $H_p(10)$ and maximum absorbed organ doses for radiochemical plant workers

| | | Maximum absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 257.48 | 242.02 | 173.81 | 249.12 | 220.19 | 224.56 | 224.56 | 237.38 | 243.66 | 281.04 | 187.45 | 206.55 | 158.26 |
| 1949 | 1518.46 | 1595.93 | 1024.05 | 1463.07 | 1436.34 | 1423.04 | 1383.14 | 1394.17 | 1582.63 | 1822.02 | 1100.91 | 1343.24 | 929.44 |
| 1950 | 3141.24 | 3038.90 | 2131.48 | 3213.11 | 2735.01 | 2714.58 | 2662.72 | 3119.67 | 3013.57 | 3787.29 | 2222.70 | 2591.97 | 1817.42 |
| 1951 | 3512.89 | 3789.02 | 2431.29 | 3581.68 | 3410.12 | 3378.55 | 3283.82 | 3434.74 | 3757.45 | 4325.80 | 2526.02 | 3189.10 | 2115.54 |
| 1952 | 2400.08 | 2146.02 | 1553.76 | 2259.92 | 1908.18 | 1916.02 | 1889.75 | 2310.62 | 2087.48 | 2693.15 | 1613.22 | 1830.30 | 1333.44 |
| 1953 | 1147.82 | 1505.94 | 966.31 | 1217.30 | 1355.34 | 1342.79 | 1305.14 | 1242.40 | 1493.39 | 1719.28 | 1003.96 | 1267.50 | 840.81 |
| 1954 | 861.48 | 1305.82 | 837.90 | 1055.54 | 1175.24 | 1164.36 | 1131.71 | 1077.30 | 1294.94 | 1490.82 | 870.55 | 1099.07 | 729.09 |
| 1955 | 1127.10 | 1708.45 | 1096.26 | 1381.00 | 1537.61 | 1523.37 | 1480.66 | 1409.47 | 1694.21 | 1950.48 | 1138.97 | 1437.95 | 953.89 |
| 1956 | 1413.93 | 2143.22 | 1375.23 | 1732.44 | 1928.90 | 1911.04 | 1857.46 | 1768.16 | 2125.36 | 2446.85 | 1428.81 | 1803.88 | 1196.63 |
| 1957 | 1044.37 | 1532.76 | 983.52 | 1285.18 | 1379.48 | 1366.71 | 1328.39 | 1281.16 | 1519.98 | 1749.90 | 1021.84 | 1290.07 | 855.79 |
| 1958 | 599.58 | 908.83 | 583.17 | 734.64 | 817.95 | 810.37 | 787.65 | 749.78 | 901.26 | 1037.58 | 605.89 | 764.93 | 507.43 |
| 1959 | 498.80 | 756.08 | 485.15 | 611.16 | 680.47 | 674.17 | 655.27 | 623.76 | 749.78 | 863.19 | 504.05 | 636.36 | 422.14 |
| 1960 | 593.23 | 899.21 | 577.00 | 726.86 | 809.29 | 801.80 | 779.32 | 741.85 | 891.72 | 1026.60 | 599.48 | 756.84 | 502.06 |
| 1961 | 431.94 | 657.16 | 421.68 | 531.20 | 591.44 | 585.97 | 569.54 | 542.16 | 651.68 | 750.26 | 438.11 | 553.11 | 366.91 |
| 1962 | 262.23 | 398.96 | 256.00 | 322.49 | 359.06 | 355.74 | 345.76 | 329.14 | 395.63 | 455.48 | 265.97 | 335.79 | 222.75 |
| 1963 | 133.20 | 202.65 | 130.03 | 163.81 | 182.38 | 180.69 | 175.63 | 167.18 | 200.96 | 231.36 | 135.10 | 170.56 | 113.15 |
| 1964 | 91.04 | 138.52 | 88.88 | 111.97 | 124.66 | 123.51 | 120.05 | 114.28 | 137.36 | 158.14 | 92.34 | 116.58 | 77.34 |
| 1965 | 78.18 | 118.94 | 76.32 | 96.14 | 107.05 | 106.06 | 103.08 | 98.13 | 117.95 | 135.79 | 79.29 | 100.11 | 66.41 |
| 1966 | 67.76 | 103.10 | 66.15 | 83.34 | 92.79 | 91.93 | 89.35 | 85.05 | 102.24 | 117.70 | 68.73 | 86.77 | 57.56 |
| 1967 | 96.93 | 147.47 | 94.62 | 119.20 | 132.72 | 131.49 | 127.80 | 121.66 | 146.24 | 168.36 | 98.31 | 124.12 | 82.33 |
| 1968 | 61.02 | 92.84 | 59.57 | 75.05 | 83.56 | 82.78 | 80.46 | 76.59 | 92.07 | 105.99 | 61.89 | 78.14 | 51.84 |
| 1969 | 61.15 | 93.03 | 59.69 | 75.20 | 83.73 | 82.95 | 80.62 | 76.75 | 92.25 | 106.21 | 62.02 | 78.30 | 51.94 |
| 1970 | 722.96 | 1099.93 | 705.79 | 889.11 | 989.94 | 980.77 | 953.27 | 907.44 | 1090.76 | 1255.75 | 733.29 | 925.77 | 614.13 |
| 1971 | 71.19 | 108.32 | 69.50 | 87.55 | 97.48 | 96.58 | 93.87 | 89.36 | 107.41 | 123.66 | 72.21 | 91.17 | 60.48 |
| 1972 | 52.94 | 80.54 | 51.68 | 65.10 | 72.48 | 71.81 | 69.80 | 66.44 | 79.87 | 91.95 | 53.69 | 67.79 | 44.97 |
| 1973 | 58.08 | 88.37 | 56.70 | 71.43 | 79.53 | 78.79 | 76.58 | 72.90 | 87.63 | 100.89 | 58.91 | 74.38 | 49.34 |
| 1974 | 53.06 | 80.73 | 51.80 | 65.26 | 72.66 | 71.98 | 69.97 | 66.60 | 80.06 | 92.17 | 53.82 | 67.95 | 45.07 |
| 1975 | 58.96 | 89.70 | 57.56 | 72.51 | 80.73 | 79.98 | 77.74 | 74.00 | 88.95 | 102.41 | 59.80 | 75.50 | 50.08 |
| 1976 | 313.50 | 476.97 | 306.06 | 385.55 | 429.27 | 425.30 | 413.37 | 393.50 | 473.00 | 544.54 | 317.98 | 401.45 | 266.31 |
| 1977 | 55.11 | 83.85 | 53.80 | 67.78 | 75.47 | 74.77 | 72.67 | 69.18 | 83.15 | 95.73 | 55.90 | 70.57 | 46.82 |

| | | Maximum absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1978 | 51.40 | 78.20 | 50.18 | 63.21 | 70.38 | 69.72 | 67.77 | 64.51 | 77.54 | 89.27 | 52.13 | 65.81 | 43.66 |
| 1979 | 49.35 | 75.08 | 48.17 | 60.69 | 67.57 | 66.94 | 65.07 | 61.94 | 74.45 | 85.71 | 50.05 | 63.19 | 41.92 |
| 1980 | 49.47 | 75.27 | 48.30 | 60.84 | 67.74 | 67.12 | 65.23 | 62.10 | 74.64 | 85.93 | 50.18 | 63.35 | 42.03 |
| 1981 | 35.89 | 54.60 | 35.04 | 44.14 | 49.14 | 48.69 | 47.32 | 45.05 | 54.15 | 62.34 | 36.40 | 45.96 | 30.49 |
| 1982 | 37.17 | 56.55 | 36.29 | 45.71 | 50.90 | 50.42 | 49.01 | 46.65 | 56.08 | 64.56 | 37.70 | 47.60 | 31.57 |
| 1983 | 52.16 | 79.37 | 50.93 | 64.15 | 71.43 | 70.77 | 68.78 | 65.48 | 78.70 | 90.61 | 52.91 | 66.80 | 44.31 |
| 1984 | 39.86 | 60.65 | 38.91 | 49.02 | 54.58 | 54.08 | 52.56 | 50.03 | 60.14 | 69.24 | 40.43 | 51.04 | 33.86 |
| 1985 | 43.32 | 65.91 | 42.29 | 53.28 | 59.32 | 58.77 | 57.12 | 54.38 | 65.36 | 75.25 | 43.94 | 55.47 | 36.80 |
| 1986 | 250.31 | 380.84 | 244.37 | 307.84 | 342.75 | 339.58 | 330.06 | 314.19 | 377.66 | 434.79 | 253.89 | 320.54 | 212.63 |
| 1987 | 256.59 | 390.39 | 250.50 | 315.57 | 351.35 | 348.10 | 338.34 | 322.07 | 387.14 | 445.70 | 260.26 | 328.58 | 217.97 |
| 1988 | 45.12 | 68.64 | 44.04 | 55.48 | 61.78 | 61.20 | 59.49 | 56.63 | 68.07 | 78.36 | 45.76 | 57.77 | 38.32 |
| 1989 | 49.47 | 75.27 | 48.30 | 60.84 | 67.74 | 67.12 | 65.23 | 62.10 | 74.64 | 85.93 | 50.18 | 63.35 | 42.03 |
| 1990 | 34.86 | 53.04 | 34.03 | 42.87 | 47.74 | 47.29 | 45.97 | 43.76 | 52.60 | 60.55 | 35.36 | 44.64 | 29.61 |
| 1991 | 51.24 | 69.42 | 44.54 | 56.11 | 62.48 | 61.90 | 60.16 | 57.27 | 68.84 | 79.25 | 46.28 | 58.43 | 38.76 |
| 1992 | 44.32 | 53.18 | 34.12 | 42.99 | 47.86 | 47.42 | 46.09 | 43.87 | 52.74 | 60.71 | 35.45 | 44.76 | 29.69 |
| 1993 | 45.23 | 41.17 | 26.42 | 36.18 | 37.05 | 36.71 | 35.68 | 33.96 | 40.83 | 47.00 | 27.45 | 34.65 | 22.99 |
| 1994 | 35.04 | 42.04 | 26.98 | 33.98 | 37.84 | 37.49 | 36.44 | 34.68 | 41.69 | 48.00 | 28.03 | 35.39 | 23.47 |
| 1995 | 35.49 | 42.59 | 27.33 | 34.43 | 38.33 | 37.97 | 36.91 | 35.14 | 42.23 | 48.62 | 28.39 | 35.84 | 23.78 |
| 1996 | 28.86 | 34.63 | 22.22 | 27.99 | 31.16 | 30.88 | 30.01 | 28.57 | 34.34 | 39.53 | 23.08 | 29.14 | 19.33 |
| 1997 | 24.55 | 24.66 | 15.82 | 19.93 | 22.19 | 21.99 | 21.37 | 20.34 | 24.45 | 28.15 | 16.44 | 20.75 | 13.77 |
| 1998 | 16.46 | 19.75 | 12.68 | 15.97 | 17.78 | 17.61 | 17.12 | 16.30 | 19.59 | 22.55 | 13.17 | 16.63 | 11.03 |
| 1999 | 21.39 | 25.66 | 16.47 | 20.74 | 23.10 | 22.88 | 22.24 | 21.17 | 25.45 | 29.30 | 17.11 | 21.60 | 14.33 |
| 2000 | 18.84 | 22.60 | 14.50 | 18.27 | 20.34 | 20.16 | 19.59 | 18.65 | 22.42 | 25.81 | 15.07 | 19.03 | 12.62 |
| 2001 | 11.81 | 14.17 | 9.10 | 11.46 | 12.76 | 12.64 | 12.28 | 11.69 | 14.06 | 16.18 | 9.45 | 11.93 | 7.91 |
| 2002 | 9.62 | 11.54 | 7.41 | 9.33 | 10.39 | 10.29 | 10.00 | 9.52 | 11.45 | 13.18 | 7.69 | 9.71 | 6.44 |
| 2003 | 13.76 | 16.51 | 10.59 | 13.35 | 14.86 | 14.72 | 14.31 | 13.62 | 16.37 | 18.85 | 11.01 | 13.90 | 9.22 |
| 2004 | 12.87 | 15.44 | 9.91 | 12.48 | 13.90 | 13.77 | 13.38 | 12.74 | 15.31 | 17.63 | 10.29 | 13.00 | 8.62 |

Table 8.12. Characteristics of maximum $H_p(10)$ and maximum absorbed organ doses for plutonium plant workers

| | | Maximum absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 47.38 | 33.78 | 23.11 | 32.36 | 30.58 | 30.94 | 30.58 | 31.65 | 34.14 | 39.47 | 24.54 | 28.80 | 20.63 |
| 1949 | 134.56 | 119.80 | 81.30 | 124.95 | 107.44 | 108.00 | 105.89 | 114.43 | 118.95 | 137.56 | 86.54 | 100.90 | 76.36 |
| 1950 | 1363.59 | 1403.36 | 960.20 | 1344.27 | 1270.41 | 1285.18 | 1270.41 | 1314.73 | 1418.13 | 1639.72 | 1019.28 | 1196.55 | 856.79 |
| 1951 | 1223.26 | 1031.69 | 694.63 | 1000.31 | 928.53 | 940.98 | 922.31 | 954.29 | 1039.40 | 1195.93 | 750.72 | 866.27 | 643.60 |
| 1952 | 982.48 | 612.68 | 610.83 | 1014.07 | 584.85 | 644.60 | 703.44 | 894.56 | 648.71 | 738.59 | 707.53 | 558.84 | 631.35 |
| 1953 | 330.55 | 273.02 | 186.80 | 270.40 | 247.16 | 250.03 | 247.16 | 255.78 | 275.89 | 319.00 | 198.30 | 232.79 | 169.00 |
| 1954 | 360.40 | 441.37 | 288.02 | 367.69 | 397.91 | 395.80 | 386.25 | 373.37 | 438.38 | 504.75 | 300.79 | 372.31 | 252.59 |
| 1955 | 266.72 | 264.49 | 151.70 | 233.38 | 233.38 | 231.43 | 217.82 | 198.37 | 266.44 | 303.39 | 153.64 | 215.87 | 122.52 |
| 1956 | 829.18 | 551.63 | 502.55 | 818.13 | 518.08 | 560.37 | 595.95 | 732.43 | 576.19 | 658.96 | 572.86 | 491.46 | 503.40 |
| 1957 | 464.30 | 420.66 | 291.42 | 422.14 | 380.70 | 386.41 | 382.61 | 395.55 | 423.55 | 487.44 | 311.51 | 356.26 | 260.99 |
| 1958 | 258.76 | 256.61 | 147.17 | 229.06 | 226.42 | 224.53 | 211.32 | 213.15 | 258.49 | 294.34 | 162.25 | 209.44 | 143.16 |
| 1959 | 278.81 | 276.49 | 158.58 | 243.96 | 243.96 | 241.93 | 227.70 | 209.70 | 278.52 | 317.15 | 160.77 | 225.67 | 132.81 |
| 1960 | 361.96 | 548.65 | 352.05 | 443.49 | 493.78 | 489.21 | 475.50 | 452.64 | 544.08 | 626.37 | 365.77 | 461.78 | 306.33 |
| 1961 | 187.16 | 175.62 | 112.69 | 166.84 | 158.05 | 156.59 | 152.20 | 145.45 | 174.15 | 200.49 | 117.08 | 147.81 | 98.05 |
| 1962 | 150.18 | 190.60 | 122.73 | 157.62 | 171.50 | 170.28 | 165.60 | 159.07 | 189.38 | 218.03 | 127.80 | 160.33 | 106.88 |
| 1963 | 86.09 | 112.96 | 71.68 | 92.05 | 101.40 | 100.47 | 97.35 | 92.30 | 112.11 | 128.93 | 74.31 | 94.73 | 61.98 |
| 1964 | 77.09 | 121.80 | 67.86 | 112.23 | 107.01 | 106.14 | 99.18 | 89.61 | 122.67 | 139.20 | 68.73 | 98.31 | 53.94 |
| 1965 | 41.88 | 64.38 | 35.87 | 59.32 | 56.56 | 56.10 | 52.42 | 47.37 | 64.84 | 73.58 | 36.33 | 51.96 | 28.51 |
| 1966 | 63.84 | 62.56 | 41.49 | 61.92 | 56.18 | 56.81 | 55.54 | 57.45 | 63.20 | 72.77 | 44.05 | 52.35 | 36.39 |
| 1967 | 53.97 | 43.33 | 24.27 | 39.92 | 38.06 | 37.76 | 35.28 | 31.88 | 43.64 | 49.52 | 24.58 | 34.97 | 19.60 |
| 1968 | 31.16 | 31.11 | 17.71 | 27.26 | 27.25 | 27.02 | 25.43 | 23.16 | 31.11 | 35.43 | 17.94 | 25.21 | 14.31 |
| 1969 | 31.36 | 43.85 | 24.43 | 40.40 | 38.52 | 38.21 | 35.70 | 32.26 | 44.16 | 50.11 | 24.74 | 35.39 | 19.42 |
| 1970 | 37.77 | 59.68 | 35.08 | 54.99 | 52.43 | 52.01 | 48.60 | 45.51 | 60.11 | 68.21 | 36.65 | 48.17 | 30.79 |
| 1971 | 31.39 | 49.59 | 27.63 | 45.69 | 43.57 | 43.21 | 40.38 | 36.48 | 49.94 | 56.67 | 28.28 | 40.03 | 23.83 |
| 1972 | 42.19 | 55.68 | 31.02 | 51.31 | 48.92 | 48.52 | 45.34 | 40.96 | 56.08 | 63.63 | 31.42 | 44.94 | 24.66 |
| 1973 | 33.92 | 53.59 | 29.86 | 49.38 | 47.08 | 46.70 | 43.64 | 39.43 | 53.97 | 61.25 | 30.24 | 43.26 | 23.73 |
| 1974 | 830.75 | 829.48 | 472.26 | 726.55 | 726.55 | 720.50 | 678.11 | 617.57 | 829.48 | 944.52 | 478.31 | 672.06 | 381.44 |
| 1975 | 55.52 | 87.72 | 48.87 | 80.83 | 77.07 | 76.45 | 71.43 | 64.54 | 88.35 | 100.26 | 49.50 | 70.81 | 38.85 |
| 1976 | 39.63 | 62.61 | 34.88 | 57.69 | 55.01 | 54.56 | 50.98 | 46.06 | 63.06 | 71.55 | 35.33 | 50.53 | 27.73 |
| 1977 | 37.78 | 59.70 | 33.26 | 55.01 | 52.45 | 52.02 | 48.61 | 43.92 | 60.12 | 68.22 | 33.69 | 48.18 | 26.44 |

| | | Maximum absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1978 | 44.46 | 70.25 | 39.14 | 64.73 | 61.72 | 61.22 | 57.21 | 51.69 | 70.75 | 80.29 | 39.64 | 56.70 | 31.11 |
| 1979 | 43.08 | 68.07 | 37.92 | 62.72 | 59.80 | 59.32 | 55.43 | 50.08 | 68.55 | 77.79 | 38.41 | 54.94 | 30.14 |
| 1980 | 42.16 | 66.61 | 37.11 | 61.38 | 58.52 | 58.05 | 54.24 | 49.01 | 67.09 | 76.13 | 37.59 | 53.77 | 29.50 |
| 1981 | 46.88 | 74.07 | 41.27 | 68.25 | 65.08 | 64.55 | 60.32 | 54.50 | 74.60 | 84.66 | 41.80 | 59.79 | 32.80 |
| 1982 | 58.52 | 92.46 | 51.51 | 85.19 | 81.23 | 80.57 | 75.29 | 68.02 | 93.12 | 105.66 | 52.17 | 74.63 | 40.94 |
| 1983 | 58.17 | 91.91 | 51.21 | 84.69 | 80.75 | 80.09 | 74.84 | 67.62 | 92.57 | 105.04 | 51.86 | 74.18 | 40.70 |
| 1984 | 44.23 | 69.89 | 38.94 | 64.40 | 61.40 | 60.90 | 56.91 | 51.42 | 70.39 | 79.87 | 39.44 | 56.41 | 30.95 |
| 1985 | 42.97 | 67.89 | 37.82 | 62.55 | 59.64 | 59.16 | 55.28 | 49.94 | 68.37 | 77.58 | 38.31 | 54.79 | 30.06 |
| 1986 | 283.92 | 278.24 | 184.55 | 275.40 | 249.85 | 252.69 | 247.01 | 255.53 | 281.08 | 323.67 | 195.90 | 232.81 | 161.83 |
| 1987 | 174.27 | 137.42 | 91.61 | 155.34 | 121.49 | 119.50 | 115.51 | 135.43 | 131.45 | 153.35 | 85.64 | 115.51 | 81.66 |
| 1988 | 181.27 | 104.67 | 107.11 | 175.27 | 99.81 | 111.98 | 121.71 | 163.10 | 109.54 | 126.58 | 124.15 | 94.94 | 109.54 |
| 1989 | 34.33 | 54.24 | 30.22 | 49.97 | 47.65 | 47.26 | 44.16 | 39.90 | 54.62 | 61.98 | 30.60 | 43.78 | 24.02 |
| 1990 | 27.07 | 42.77 | 23.83 | 39.41 | 37.58 | 37.27 | 34.83 | 31.47 | 43.08 | 48.88 | 24.13 | 34.52 | 18.94 |
| 1991 | 33.64 | 53.14 | 29.61 | 48.97 | 46.69 | 46.31 | 43.27 | 39.10 | 53.52 | 60.74 | 29.99 | 42.89 | 23.54 |
| 1992 | 14.74 | 20.64 | 11.50 | 19.02 | 18.13 | 17.99 | 16.81 | 15.18 | 20.79 | 23.59 | 11.65 | 16.66 | 9.14 |
| 1993 | 16.20 | 22.68 | 12.63 | 20.90 | 19.92 | 19.76 | 18.47 | 16.68 | 22.84 | 25.92 | 12.80 | 18.30 | 10.04 |
| 1994 | 14.56 | 13.50 | 7.52 | 12.44 | 11.86 | 11.77 | 11.00 | 9.94 | 13.60 | 15.43 | 7.62 | 10.90 | 5.98 |
| 1995 | 19.02 | 14.14 | 8.75 | 14.83 | 12.42 | 12.32 | 11.64 | 12.93 | 14.24 | 16.16 | 9.23 | 11.41 | 7.80 |
| 1996 | 21.22 | 14.64 | 9.76 | 16.55 | 12.94 | 12.73 | 12.31 | 14.43 | 14.29 | 16.34 | 9.13 | 12.31 | 8.70 |
| 1997 | 19.42 | 21.48 | 11.97 | 19.79 | 18.87 | 18.72 | 17.49 | 15.80 | 21.63 | 24.55 | 12.12 | 17.34 | 9.51 |
| 1998 | 34.43 | 48.20 | 26.85 | 44.41 | 42.34 | 42.00 | 39.24 | 35.46 | 48.54 | 55.08 | 27.20 | 38.90 | 21.34 |
| 1999 | 15.18 | 21.25 | 11.84 | 19.58 | 18.67 | 18.52 | 17.30 | 15.63 | 21.40 | 24.29 | 11.99 | 17.15 | 9.41 |
| 2000 | 11.98 | 16.60 | 9.25 | 15.30 | 14.58 | 14.47 | 13.52 | 12.21 | 16.72 | 18.97 | 9.37 | 13.40 | 7.35 |
| 2001 | 16.16 | 22.63 | 12.61 | 20.85 | 19.88 | 19.72 | 18.42 | 16.65 | 22.79 | 25.86 | 12.77 | 18.26 | 10.02 |
| 2002 | 22.07 | 30.89 | 17.21 | 28.47 | 27.14 | 26.92 | 25.16 | 22.73 | 31.12 | 35.31 | 17.43 | 24.94 | 13.68 |
| 2003 | 20.52 | 28.73 | 16.01 | 26.47 | 25.24 | 25.04 | 23.39 | 21.14 | 28.93 | 32.83 | 16.21 | 23.19 | 12.72 |
| 2004 | 17.28 | 24.19 | 13.48 | 22.29 | 21.26 | 21.08 | 19.70 | 17.80 | 24.37 | 27.65 | 13.65 | 19.53 | 10.71 |

Table 8.13. Characteristics of standard deviations of $H_p(10)$ and standard deviations of absorbed organ doses for reactor plant workers

| | | Standard deviation of absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 167.85 | 112.79 | 99.54 | 163.22 | 101.95 | 107.50 | 112.50 | 157.13 | 111.66 | 149.02 | 108.56 | 100.01 | 93.03 |
| 1949 | 494.20 | 331.51 | 295.65 | 485.91 | 300.61 | 318.18 | 334.53 | 464.16 | 329.53 | 436.44 | 324.58 | 294.55 | 279.72 |
| 1950 | 227.38 | 156.56 | 134.20 | 216.58 | 141.92 | 148.74 | 154.47 | 207.60 | 155.76 | 201.57 | 146.59 | 137.77 | 125.73 |
| 1951 | 156.99 | 137.10 | 103.83 | 153.19 | 124.88 | 128.10 | 129.74 | 146.10 | 137.87 | 162.53 | 112.95 | 118.05 | 97.13 |
| 1952 | 251.77 | 183.90 | 153.32 | 249.78 | 168.01 | 177.38 | 184.53 | 227.31 | 187.46 | 225.09 | 171.59 | 159.45 | 149.33 |
| 1953 | 130.91 | 94.74 | 78.06 | 125.02 | 84.74 | 87.54 | 89.40 | 123.02 | 92.69 | 124.40 | 83.63 | 82.47 | 70.62 |
| 1954 | 52.59 | 38.51 | 32.09 | 51.35 | 34.82 | 36.29 | 37.49 | 49.06 | 38.28 | 49.15 | 35.14 | 33.62 | 30.26 |
| 1955 | 53.05 | 37.26 | 32.08 | 51.74 | 33.83 | 35.53 | 37.02 | 49.34 | 37.14 | 47.87 | 35.23 | 32.82 | 30.38 |
| 1956 | 41.20 | 30.35 | 25.48 | 40.46 | 27.74 | 29.08 | 30.19 | 37.79 | 30.59 | 37.59 | 28.30 | 26.52 | 24.60 |
| 1957 | 35.17 | 23.59 | 20.73 | 33.90 | 21.49 | 22.71 | 23.83 | 31.72 | 23.65 | 30.40 | 23.09 | 20.75 | 20.12 |
| 1958 | 25.81 | 17.10 | 15.26 | 25.08 | 15.55 | 16.47 | 17.33 | 23.61 | 17.07 | 22.36 | 16.98 | 15.11 | 14.78 |
| 1959 | 28.86 | 20.33 | 16.99 | 27.28 | 18.46 | 19.30 | 19.98 | 25.59 | 20.38 | 25.51 | 18.82 | 17.68 | 16.32 |
| 1960 | 31.50 | 23.37 | 19.18 | 30.59 | 20.95 | 21.63 | 22.11 | 29.81 | 22.93 | 30.44 | 20.75 | 20.29 | 17.70 |
| 1961 | 22.72 | 16.36 | 13.70 | 21.99 | 14.77 | 15.39 | 15.89 | 20.99 | 16.22 | 21.06 | 15.05 | 14.25 | 13.00 |
| 1962 | 19.44 | 13.66 | 11.74 | 19.07 | 12.31 | 12.87 | 13.37 | 18.29 | 13.48 | 17.97 | 12.89 | 11.97 | 11.12 |
| 1963 | 19.46 | 13.63 | 11.57 | 18.80 | 12.19 | 12.66 | 13.06 | 18.29 | 13.31 | 18.24 | 12.58 | 11.89 | 10.77 |
| 1964 | 23.79 | 16.62 | 14.18 | 23.09 | 14.87 | 15.48 | 16.00 | 22.45 | 16.24 | 22.23 | 15.41 | 14.54 | 13.19 |
| 1965 | 17.23 | 12.00 | 10.25 | 16.72 | 10.73 | 11.16 | 11.53 | 16.22 | 11.71 | 16.10 | 11.17 | 10.48 | 9.59 |
| 1966 | 14.77 | 10.56 | 8.90 | 14.41 | 9.45 | 9.80 | 10.10 | 14.02 | 10.32 | 14.03 | 9.67 | 9.20 | 8.28 |
| 1967 | 15.18 | 10.38 | 9.02 | 14.73 | 9.34 | 9.78 | 10.16 | 14.20 | 10.21 | 13.81 | 9.89 | 9.10 | 8.51 |
| 1968 | 11.52 | 8.05 | 6.82 | 11.10 | 7.18 | 7.45 | 7.66 | 10.86 | 7.83 | 10.84 | 7.39 | 7.02 | 6.30 |
| 1969 | 11.77 | 8.18 | 6.99 | 11.38 | 7.32 | 7.61 | 7.85 | 11.07 | 7.99 | 10.96 | 7.60 | 7.14 | 6.50 |
| 1970 | 11.45 | 7.95 | 6.82 | 11.12 | 7.11 | 7.41 | 7.67 | 10.80 | 7.77 | 10.65 | 7.42 | 6.95 | 6.36 |
| 1971 | 12.54 | 8.71 | 7.44 | 12.12 | 7.79 | 8.11 | 8.37 | 11.80 | 8.51 | 11.66 | 8.08 | 7.61 | 6.90 |
| 1972 | 14.12 | 9.72 | 8.36 | 13.64 | 8.72 | 9.09 | 9.40 | 13.22 | 9.52 | 12.98 | 9.11 | 8.51 | 7.80 |
| 1973 | 12.94 | 9.09 | 7.66 | 12.46 | 8.08 | 8.37 | 8.60 | 12.20 | 8.82 | 12.28 | 8.29 | 7.91 | 7.08 |
| 1974 | 12.25 | 8.71 | 7.24 | 11.73 | 7.72 | 7.95 | 8.10 | 11.63 | 8.41 | 11.79 | 7.74 | 7.56 | 6.53 |
| 1975 | 19.46 | 13.40 | 11.52 | 18.79 | 12.02 | 12.53 | 12.95 | 18.23 | 13.13 | 17.89 | 12.53 | 11.73 | 10.72 |
| 1976 | 11.26 | 7.90 | 6.67 | 10.84 | 7.03 | 7.29 | 7.48 | 10.64 | 7.67 | 10.64 | 7.20 | 6.88 | 6.13 |
| 1977 | 10.51 | 7.43 | 6.21 | 10.08 | 6.60 | 6.81 | 6.96 | 9.97 | 7.19 | 10.05 | 6.66 | 6.46 | 5.63 |

| | | Standard deviation of absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1978 | 9.99 | 7.11 | 5.91 | 9.58 | 6.30 | 6.49 | 6.62 | 9.49 | 6.86 | 9.63 | 6.33 | 6.17 | 5.36 |
| 1979 | 9.40 | 6.66 | 5.54 | 8.99 | 5.90 | 6.09 | 6.21 | 8.92 | 6.43 | 9.04 | 5.94 | 5.79 | 5.02 |
| 1980 | 10.89 | 7.64 | 6.42 | 10.43 | 6.79 | 7.03 | 7.21 | 10.27 | 7.41 | 10.30 | 6.92 | 6.65 | 5.88 |
| 1981 | 11.41 | 7.78 | 6.78 | 11.10 | 7.01 | 7.35 | 7.64 | 10.66 | 7.67 | 10.33 | 7.44 | 6.83 | 6.40 |
| 1982 | 10.56 | 7.30 | 6.26 | 10.21 | 6.54 | 6.81 | 7.04 | 9.93 | 7.14 | 9.76 | 6.80 | 6.39 | 5.81 |
| 1983 | 9.47 | 6.58 | 5.61 | 9.13 | 5.88 | 6.11 | 6.30 | 8.91 | 6.42 | 8.82 | 6.07 | 5.75 | 5.17 |
| 1984 | 7.39 | 5.17 | 4.36 | 7.09 | 4.61 | 4.77 | 4.90 | 6.96 | 5.02 | 6.96 | 4.70 | 4.51 | 3.99 |
| 1985 | 7.39 | 5.28 | 4.35 | 7.03 | 4.67 | 4.79 | 4.87 | 7.03 | 5.08 | 7.16 | 4.62 | 4.58 | 3.87 |
| 1986 | 18.19 | 12.28 | 10.83 | 17.76 | 11.10 | 11.70 | 12.24 | 17.08 | 12.15 | 16.25 | 11.83 | 10.88 | 10.15 |
| 1987 | 9.59 | 6.77 | 5.63 | 9.12 | 6.01 | 6.19 | 6.30 | 9.07 | 6.54 | 9.14 | 6.00 | 5.88 | 5.05 |
| 1988 | 7.04 | 5.02 | 4.15 | 6.70 | 4.44 | 4.57 | 4.64 | 6.70 | 4.84 | 6.81 | 4.41 | 4.36 | 3.70 |
| 1989 | 7.14 | 4.94 | 4.23 | 6.91 | 4.42 | 4.61 | 4.76 | 6.70 | 4.83 | 6.61 | 4.61 | 4.32 | 3.96 |
| 1990 | 4.55 | 3.20 | 2.70 | 4.40 | 2.84 | 2.95 | 3.03 | 4.30 | 3.10 | 4.32 | 2.93 | 2.78 | 2.51 |
| 1991 | 5.07 | 3.45 | 3.03 | 4.94 | 3.12 | 3.29 | 3.42 | 4.73 | 3.42 | 4.53 | 3.33 | 3.04 | 2.86 |
| 1992 | 3.23 | 1.94 | 1.66 | 2.70 | 1.74 | 1.81 | 1.87 | 2.64 | 1.90 | 2.60 | 1.80 | 1.70 | 1.53 |
| 1993 | 3.05 | 1.73 | 1.47 | 2.39 | 1.54 | 1.61 | 1.65 | 2.34 | 1.68 | 2.31 | 1.59 | 1.51 | 1.35 |
| 1994 | 2.23 | 1.31 | 1.13 | 1.84 | 1.19 | 1.24 | 1.29 | 1.77 | 1.30 | 1.72 | 1.24 | 1.15 | 1.07 |
| 1995 | 2.22 | 1.28 | 1.12 | 1.82 | 1.15 | 1.21 | 1.26 | 1.76 | 1.26 | 1.70 | 1.22 | 1.12 | 1.05 |
| 1996 | 2.32 | 1.33 | 1.16 | 1.90 | 1.20 | 1.26 | 1.31 | 1.84 | 1.31 | 1.76 | 1.27 | 1.17 | 1.09 |
| 1997 | 1.82 | 1.03 | 0.90 | 1.48 | 0.93 | 0.98 | 1.02 | 1.42 | 1.02 | 1.36 | 0.99 | 0.90 | 0.85 |
| 1998 | 2.20 | 1.36 | 1.13 | 1.83 | 1.21 | 1.24 | 1.27 | 1.82 | 1.31 | 1.83 | 1.21 | 1.18 | 1.02 |
| 1999 | 2.22 | 1.39 | 1.16 | 1.88 | 1.24 | 1.27 | 1.30 | 1.86 | 1.35 | 1.89 | 1.24 | 1.21 | 1.04 |
| 2000 | 1.93 | 1.11 | 0.98 | 1.61 | 1.01 | 1.06 | 1.11 | 1.54 | 1.10 | 1.47 | 1.08 | 0.98 | 0.93 |
| 2001 | 1.68 | 1.03 | 0.87 | 1.42 | 0.92 | 0.95 | 0.98 | 1.39 | 1.00 | 1.38 | 0.94 | 0.90 | 0.80 |
| 2002 | 1.46 | 0.97 | 0.77 | 1.24 | 0.85 | 0.86 | 0.86 | 1.27 | 0.92 | 1.33 | 0.81 | 0.84 | 0.67 |
| 2003 | 1.51 | 1.00 | 0.80 | 1.28 | 0.87 | 0.89 | 0.89 | 1.31 | 0.95 | 1.36 | 0.83 | 0.86 | 0.69 |
| 2004 | 1.57 | 1.04 | 0.83 | 1.34 | 0.91 | 0.93 | 0.93 | 1.36 | 0.99 | 1.42 | 0.87 | 0.89 | 0.72 |

Table 8.14. Characteristics of standard deviations of $H_p(10)$ and standard deviations of absorbed organ doses for radiochemical plant workers

| | | Standard deviation of absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 74.15 | 71.49 | 50.64 | 72.05 | 65.01 | 66.12 | 65.95 | 68.60 | 71.98 | 82.58 | 54.55 | 60.88 | 46.03 |
| 1949 | 194.13 | 236.00 | 151.54 | 194.03 | 212.24 | 210.05 | 204.16 | 195.99 | 233.88 | 269.22 | 158.07 | 198.44 | 132.69 |
| 1950 | 423.12 | 512.74 | 332.76 | 432.38 | 460.49 | 456.62 | 444.00 | 436.23 | 507.55 | 592.32 | 346.28 | 431.95 | 289.14 |
| 1951 | 462.00 | 569.06 | 366.67 | 475.69 | 510.93 | 505.52 | 490.58 | 479.35 | 562.85 | 653.28 | 378.70 | 479.22 | 318.51 |
| 1952 | 250.71 | 312.02 | 199.74 | 254.96 | 280.41 | 277.33 | 269.19 | 258.56 | 309.03 | 356.68 | 207.74 | 262.28 | 174.06 |
| 1953 | 128.23 | 160.78 | 103.16 | 131.24 | 144.58 | 143.11 | 139.02 | 133.27 | 159.30 | 183.68 | 107.23 | 135.26 | 89.93 |
| 1954 | 134.09 | 205.68 | 130.92 | 165.01 | 184.84 | 182.69 | 177.12 | 168.31 | 203.67 | 234.95 | 135.65 | 172.89 | 113.50 |
| 1955 | 146.85 | 219.68 | 140.21 | 177.49 | 197.39 | 195.19 | 189.33 | 180.82 | 217.54 | 251.46 | 145.41 | 184.71 | 121.56 |
| 1956 | 132.58 | 198.58 | 126.96 | 160.20 | 178.57 | 176.67 | 171.51 | 163.41 | 196.73 | 226.78 | 131.75 | 167.03 | 110.36 |
| 1957 | 141.00 | 210.47 | 134.85 | 171.00 | 189.18 | 187.19 | 181.74 | 174.24 | 208.44 | 240.95 | 140.09 | 177.05 | 117.22 |
| 1958 | 91.43 | 133.10 | 85.24 | 107.79 | 119.71 | 118.49 | 115.09 | 109.81 | 131.92 | 151.92 | 88.63 | 111.94 | 74.26 |
| 1959 | 85.85 | 121.21 | 78.40 | 100.15 | 109.09 | 108.21 | 105.34 | 101.66 | 120.21 | 138.64 | 81.76 | 102.07 | 68.59 |
| 1960 | 89.77 | 134.62 | 86.14 | 108.68 | 121.09 | 119.86 | 116.41 | 110.83 | 133.44 | 153.61 | 89.51 | 113.22 | 74.98 |
| 1961 | 63.54 | 97.90 | 62.11 | 77.92 | 87.98 | 86.90 | 84.22 | 79.63 | 96.96 | 111.64 | 64.38 | 82.24 | 53.86 |
| 1962 | 49.50 | 75.52 | 48.11 | 60.56 | 67.90 | 67.14 | 65.13 | 61.79 | 74.83 | 86.18 | 49.92 | 63.48 | 41.78 |
| 1963 | 21.11 | 33.27 | 20.94 | 26.15 | 29.88 | 29.45 | 28.49 | 26.78 | 32.93 | 37.91 | 21.67 | 27.92 | 18.11 |
| 1964 | 15.55 | 24.40 | 15.34 | 19.22 | 21.90 | 21.58 | 20.87 | 19.65 | 24.14 | 27.82 | 15.87 | 20.47 | 13.26 |
| 1965 | 14.27 | 22.24 | 13.98 | 17.46 | 19.97 | 19.68 | 19.03 | 17.87 | 22.01 | 25.33 | 14.46 | 18.66 | 12.09 |
| 1966 | 12.89 | 20.46 | 12.83 | 16.01 | 18.36 | 18.08 | 17.47 | 16.39 | 20.24 | 23.30 | 13.25 | 17.16 | 11.08 |
| 1967 | 11.19 | 17.23 | 10.88 | 13.67 | 15.47 | 15.28 | 14.79 | 13.94 | 17.06 | 19.64 | 11.26 | 14.46 | 9.41 |
| 1968 | 11.53 | 17.76 | 11.27 | 14.16 | 15.96 | 15.76 | 15.27 | 14.47 | 17.59 | 20.28 | 11.68 | 14.92 | 9.76 |
| 1969 | 8.34 | 13.09 | 8.18 | 10.17 | 11.74 | 11.56 | 11.16 | 10.43 | 12.94 | 14.90 | 8.45 | 10.97 | 7.06 |
| 1970 | 16.56 | 25.36 | 16.19 | 20.33 | 22.81 | 22.57 | 21.91 | 20.77 | 25.14 | 28.94 | 16.80 | 21.33 | 14.06 |
| 1971 | 8.62 | 13.16 | 8.30 | 10.37 | 11.82 | 11.66 | 11.28 | 10.61 | 13.03 | 15.00 | 8.59 | 11.05 | 7.18 |
| 1972 | 6.74 | 10.55 | 6.67 | 8.32 | 9.48 | 9.35 | 9.06 | 8.52 | 10.45 | 12.02 | 6.90 | 8.86 | 5.77 |
| 1973 | 7.10 | 11.49 | 7.15 | 8.84 | 10.31 | 10.13 | 9.77 | 9.08 | 11.36 | 13.08 | 7.36 | 9.63 | 6.14 |
| 1974 | 6.51 | 9.89 | 6.25 | 7.82 | 8.88 | 8.77 | 8.49 | 7.99 | 9.79 | 11.27 | 6.47 | 8.30 | 5.41 |
| 1975 | 6.41 | 9.28 | 5.92 | 7.48 | 8.34 | 8.25 | 8.01 | 7.61 | 9.19 | 10.58 | 6.16 | 7.80 | 5.16 |
| 1976 | 9.32 | 14.27 | 9.09 | 11.40 | 12.83 | 12.69 | 12.31 | 11.66 | 14.14 | 16.27 | 9.43 | 11.99 | 7.89 |
| 1977 | 6.11 | 9.53 | 6.01 | 7.50 | 8.56 | 8.44 | 8.17 | 7.68 | 9.43 | 10.85 | 6.22 | 8.00 | 5.20 |

| | | Standard deviation of absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1978 | 5.80 | 9.03 | 5.66 | 7.04 | 8.11 | 7.98 | 7.71 | 7.22 | 8.93 | 10.28 | 5.85 | 7.57 | 4.89 |
| 1979 | 6.08 | 9.69 | 6.06 | 7.53 | 8.69 | 8.56 | 8.27 | 7.73 | 9.58 | 11.03 | 6.26 | 8.12 | 5.23 |
| 1980 | 4.75 | 7.57 | 4.74 | 5.89 | 6.80 | 6.69 | 6.47 | 6.05 | 7.49 | 8.62 | 4.90 | 6.35 | 4.09 |
| 1981 | 3.91 | 6.07 | 3.83 | 4.78 | 5.45 | 5.38 | 5.20 | 4.90 | 6.01 | 6.91 | 3.96 | 5.09 | 3.31 |
| 1982 | 4.55 | 7.17 | 4.51 | 5.62 | 6.44 | 6.35 | 6.14 | 5.77 | 7.09 | 8.16 | 4.67 | 6.01 | 3.90 |
| 1983 | 5.26 | 8.27 | 5.21 | 6.49 | 7.42 | 7.32 | 7.08 | 6.65 | 8.18 | 9.42 | 5.39 | 6.94 | 4.50 |
| 1984 | 4.60 | 7.18 | 4.54 | 5.68 | 6.45 | 6.37 | 6.17 | 5.82 | 7.11 | 8.18 | 4.71 | 6.03 | 3.94 |
| 1985 | 4.65 | 7.24 | 4.58 | 5.73 | 6.50 | 6.42 | 6.22 | 5.87 | 7.17 | 8.25 | 4.75 | 6.08 | 3.97 |
| 1986 | 12.42 | 18.97 | 12.13 | 15.26 | 17.06 | 16.89 | 16.40 | 15.58 | 18.80 | 21.65 | 12.59 | 15.95 | 10.54 |
| 1987 | 11.91 | 18.06 | 11.57 | 14.56 | 16.25 | 16.09 | 15.63 | 14.86 | 17.90 | 20.61 | 12.01 | 15.19 | 10.06 |
| 1988 | 5.20 | 8.17 | 5.18 | 6.47 | 7.34 | 7.25 | 7.03 | 6.63 | 8.09 | 9.31 | 5.36 | 6.86 | 4.48 |
| 1989 | 5.05 | 8.00 | 5.05 | 6.30 | 7.19 | 7.09 | 6.87 | 6.46 | 7.92 | 9.12 | 5.23 | 6.72 | 4.37 |
| 1990 | 3.79 | 5.89 | 3.73 | 4.66 | 5.29 | 5.23 | 5.06 | 4.77 | 5.83 | 6.71 | 3.86 | 4.95 | 3.23 |
| 1991 | 3.90 | 5.53 | 3.56 | 4.53 | 4.97 | 4.93 | 4.80 | 4.60 | 5.48 | 6.31 | 3.71 | 4.65 | 3.11 |
| 1992 | 3.33 | 3.87 | 2.50 | 3.19 | 3.49 | 3.46 | 3.37 | 3.23 | 3.84 | 4.42 | 2.61 | 3.26 | 2.19 |
| 1993 | 2.87 | 3.09 | 2.04 | 2.68 | 2.79 | 2.78 | 2.72 | 2.67 | 3.07 | 3.54 | 2.15 | 2.61 | 1.81 |
| 1994 | 2.39 | 2.85 | 1.82 | 2.29 | 2.57 | 2.54 | 2.46 | 2.34 | 2.83 | 3.25 | 1.89 | 2.40 | 1.58 |
| 1995 | 2.68 | 3.25 | 2.07 | 2.60 | 2.92 | 2.89 | 2.80 | 2.66 | 3.22 | 3.70 | 2.15 | 2.73 | 1.80 |
| 1996 | 3.07 | 3.71 | 2.37 | 2.98 | 3.33 | 3.30 | 3.21 | 3.04 | 3.67 | 4.23 | 2.46 | 3.12 | 2.06 |
| 1997 | 2.11 | 1.99 | 1.27 | 1.61 | 1.79 | 1.77 | 1.72 | 1.64 | 1.98 | 2.27 | 1.33 | 1.68 | 1.11 |
| 1998 | 1.56 | 1.85 | 1.18 | 1.50 | 1.67 | 1.65 | 1.60 | 1.52 | 1.84 | 2.11 | 1.23 | 1.56 | 1.03 |
| 1999 | 1.82 | 2.20 | 1.40 | 1.76 | 1.98 | 1.96 | 1.90 | 1.80 | 2.18 | 2.51 | 1.45 | 1.85 | 1.22 |
| 2000 | 1.96 | 2.36 | 1.51 | 1.90 | 2.13 | 2.10 | 2.04 | 1.94 | 2.34 | 2.70 | 1.57 | 1.99 | 1.31 |
| 2001 | 1.50 | 1.75 | 1.12 | 1.43 | 1.57 | 1.56 | 1.51 | 1.45 | 1.73 | 1.99 | 1.17 | 1.47 | 0.98 |
| 2002 | 1.60 | 1.93 | 1.22 | 1.54 | 1.73 | 1.71 | 1.66 | 1.57 | 1.91 | 2.20 | 1.27 | 1.62 | 1.06 |
| 2003 | 1.58 | 1.88 | 1.20 | 1.52 | 1.69 | 1.67 | 1.62 | 1.54 | 1.86 | 2.14 | 1.25 | 1.58 | 1.04 |
| 2004 | 1.42 | 1.68 | 1.07 | 1.35 | 1.51 | 1.49 | 1.45 | 1.38 | 1.66 | 1.91 | 1.11 | 1.41 | 0.93 |

Table 8.15. Characteristics of standard deviations of $H_p(10)$ and standard deviations of absorbed organ doses for plutonium plant workers

| | | Standard deviation of absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 10.29 | 11.31 | 4.41 | 3.09 | 9.97 | 9.24 | 8.02 | 4.01 | 11.30 | 11.84 | 3.94 | 8.98 | 2.85 |
| 1949 | 19.29 | 15.21 | 11.79 | 18.44 | 13.85 | 14.35 | 14.57 | 17.15 | 15.31 | 17.77 | 12.87 | 13.06 | 11.19 |
| 1950 | 166.83 | 167.79 | 114.64 | 160.74 | 151.83 | 153.54 | 151.71 | 156.98 | 169.53 | 195.97 | 121.83 | 142.94 | 102.35 |
| 1951 | 170.25 | 166.78 | 114.57 | 161.41 | 150.93 | 152.73 | 151.03 | 157.08 | 168.38 | 194.59 | 122.09 | 141.99 | 102.73 |
| 1952 | 88.15 | 84.02 | 58.28 | 83.54 | 76.00 | 76.92 | 76.24 | 80.00 | 84.76 | 97.75 | 62.50 | 71.45 | 52.96 |
| 1953 | 55.89 | 52.83 | 35.69 | 50.84 | 47.62 | 48.03 | 47.24 | 48.96 | 53.22 | 61.45 | 37.94 | 44.71 | 31.83 |
| 1954 | 62.95 | 61.99 | 39.73 | 59.71 | 55.35 | 55.66 | 53.88 | 54.30 | 62.51 | 71.67 | 41.90 | 51.46 | 34.55 |
| 1955 | 46.29 | 45.54 | 28.26 | 42.94 | 40.50 | 40.59 | 38.97 | 38.31 | 45.96 | 52.58 | 29.59 | 37.57 | 24.15 |
| 1956 | 73.97 | 65.86 | 44.52 | 69.39 | 59.11 | 60.18 | 59.06 | 61.60 | 66.91 | 76.47 | 47.85 | 55.04 | 40.00 |
| 1957 | 52.44 | 49.51 | 31.23 | 47.77 | 44.08 | 44.31 | 42.70 | 42.46 | 50.01 | 57.17 | 32.89 | 40.92 | 27.01 |
| 1958 | 44.00 | 43.05 | 25.82 | 39.47 | 38.15 | 38.08 | 36.25 | 34.51 | 43.45 | 49.59 | 26.71 | 35.33 | 21.56 |
| 1959 | 47.09 | 46.75 | 27.92 | 42.59 | 41.42 | 41.30 | 39.28 | 37.25 | 47.17 | 53.84 | 28.86 | 38.34 | 23.26 |
| 1960 | 60.72 | 62.01 | 37.14 | 55.62 | 55.00 | 54.77 | 52.13 | 49.17 | 62.44 | 71.30 | 38.27 | 50.98 | 30.97 |
| 1961 | 38.17 | 37.90 | 22.74 | 34.70 | 33.43 | 33.30 | 31.70 | 30.59 | 37.95 | 43.36 | 23.35 | 31.01 | 19.11 |
| 1962 | 29.02 | 29.31 | 17.25 | 26.03 | 25.81 | 25.69 | 24.38 | 22.82 | 29.40 | 33.52 | 17.80 | 23.89 | 14.31 |
| 1963 | 14.81 | 15.60 | 9.08 | 13.83 | 13.72 | 13.63 | 12.89 | 12.03 | 15.64 | 17.82 | 9.32 | 12.69 | 7.51 |
| 1964 | 10.52 | 12.24 | 6.97 | 11.13 | 10.74 | 10.66 | 10.02 | 9.31 | 12.29 | 13.97 | 7.11 | 9.90 | 5.70 |
| 1965 | 6.82 | 7.57 | 4.39 | 6.91 | 6.65 | 6.61 | 6.25 | 5.91 | 7.61 | 8.66 | 4.53 | 6.14 | 3.65 |
| 1966 | 5.34 | 6.16 | 3.55 | 5.62 | 5.41 | 5.38 | 5.07 | 4.78 | 6.20 | 7.05 | 3.67 | 4.99 | 2.94 |
| 1967 | 4.42 | 5.24 | 3.02 | 4.92 | 4.61 | 4.58 | 4.31 | 4.06 | 5.27 | 5.99 | 3.07 | 4.25 | 2.48 |
| 1968 | 2.69 | 3.15 | 1.83 | 2.93 | 2.78 | 2.76 | 2.60 | 2.45 | 3.17 | 3.60 | 1.87 | 2.56 | 1.51 |
| 1969 | 4.14 | 5.37 | 3.06 | 5.03 | 4.72 | 4.68 | 4.39 | 4.11 | 5.38 | 6.12 | 3.08 | 4.35 | 2.49 |
| 1970 | 5.07 | 6.35 | 3.78 | 6.08 | 5.60 | 5.56 | 5.26 | 5.15 | 6.33 | 7.23 | 3.78 | 5.20 | 3.17 |
| 1971 | 3.22 | 3.78 | 2.32 | 3.60 | 3.36 | 3.34 | 3.20 | 3.15 | 3.79 | 4.33 | 2.38 | 3.12 | 1.98 |
| 1972 | 3.61 | 4.37 | 2.48 | 4.06 | 3.84 | 3.80 | 3.56 | 3.32 | 4.39 | 4.99 | 2.49 | 3.54 | 2.00 |
| 1973 | 2.63 | 3.80 | 2.14 | 3.54 | 3.33 | 3.31 | 3.09 | 2.85 | 3.82 | 4.33 | 2.16 | 3.07 | 1.72 |
| 1974 | 27.78 | 27.01 | 15.79 | 24.72 | 23.71 | 23.60 | 22.37 | 21.23 | 27.05 | 30.84 | 16.17 | 21.94 | 13.08 |
| 1975 | 5.08 | 7.37 | 4.17 | 6.86 | 6.48 | 6.43 | 6.03 | 5.56 | 7.42 | 8.42 | 4.23 | 5.96 | 3.37 |
| 1976 | 4.67 | 6.77 | 3.85 | 6.38 | 5.96 | 5.91 | 5.54 | 5.16 | 6.81 | 7.74 | 3.90 | 5.48 | 3.12 |
| 1977 | 4.78 | 6.69 | 3.84 | 6.33 | 5.89 | 5.85 | 5.50 | 5.16 | 6.73 | 7.65 | 3.89 | 5.42 | 3.13 |

| | | Standard deviation of absorbed organ dose, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1978 | 4.88 | 7.36 | 4.12 | 6.82 | 6.46 | 6.40 | 5.98 | 5.48 | 7.40 | 8.40 | 4.15 | 5.94 | 3.30 |
| 1979 | 4.53 | 6.40 | 3.69 | 6.11 | 5.63 | 5.59 | 5.25 | 5.02 | 6.43 | 7.31 | 3.70 | 5.20 | 3.03 |
| 1980 | 4.41 | 6.38 | 3.62 | 5.97 | 5.61 | 5.57 | 5.22 | 4.84 | 6.42 | 7.29 | 3.65 | 5.17 | 2.92 |
| 1981 | 4.82 | 6.43 | 3.84 | 6.37 | 5.67 | 5.63 | 5.31 | 5.32 | 6.43 | 7.33 | 3.78 | 5.27 | 3.21 |
| 1982 | 4.53 | 6.82 | 3.83 | 6.34 | 5.99 | 5.94 | 5.56 | 5.11 | 6.86 | 7.79 | 3.86 | 5.51 | 3.08 |
| 1983 | 6.02 | 9.06 | 5.11 | 8.48 | 7.96 | 7.90 | 7.40 | 6.82 | 9.12 | 10.35 | 5.18 | 7.32 | 4.13 |
| 1984 | 5.54 | 7.90 | 4.48 | 7.43 | 6.94 | 6.88 | 6.45 | 6.01 | 7.93 | 9.01 | 4.51 | 6.39 | 3.62 |
| 1985 | 4.76 | 7.03 | 3.95 | 6.56 | 6.18 | 6.12 | 5.73 | 5.28 | 7.07 | 8.02 | 3.98 | 5.68 | 3.18 |
| 1986 | 17.24 | 15.86 | 10.57 | 16.67 | 14.17 | 14.23 | 13.87 | 15.01 | 15.79 | 18.22 | 10.84 | 13.28 | 9.36 |
| 1987 | 11.27 | 10.28 | 6.54 | 11.03 | 9.07 | 8.95 | 8.56 | 9.46 | 10.01 | 11.57 | 6.22 | 8.54 | 5.70 |
| 1988 | 8.67 | 6.51 | 5.40 | 8.85 | 5.97 | 6.35 | 6.54 | 8.01 | 6.67 | 7.65 | 6.06 | 5.60 | 5.28 |
| 1989 | 3.73 | 4.91 | 2.86 | 4.74 | 4.32 | 4.29 | 4.03 | 3.93 | 4.92 | 5.60 | 2.87 | 3.99 | 2.37 |
| 1990 | 3.55 | 4.72 | 2.76 | 4.50 | 4.16 | 4.14 | 3.91 | 3.74 | 4.75 | 5.40 | 2.81 | 3.84 | 2.28 |
| 1991 | 3.51 | 4.11 | 2.48 | 4.11 | 3.63 | 3.60 | 3.41 | 3.46 | 4.10 | 4.68 | 2.45 | 3.37 | 2.10 |
| 1992 | 1.61 | 1.80 | 1.08 | 1.74 | 1.59 | 1.58 | 1.50 | 1.48 | 1.81 | 2.06 | 1.10 | 1.47 | 0.91 |
| 1993 | 1.92 | 2.21 | 1.32 | 2.16 | 1.95 | 1.94 | 1.83 | 1.82 | 2.21 | 2.52 | 1.32 | 1.81 | 1.10 |
| 1994 | 1.50 | 1.48 | 0.91 | 1.45 | 1.31 | 1.30 | 1.24 | 1.26 | 1.48 | 1.69 | 0.91 | 1.22 | 0.77 |
| 1995 | 2.49 | 2.49 | 1.50 | 2.46 | 2.20 | 2.19 | 2.08 | 2.09 | 2.49 | 2.84 | 1.52 | 2.04 | 1.28 |
| 1996 | 2.19 | 2.09 | 1.30 | 2.14 | 1.85 | 1.84 | 1.75 | 1.85 | 2.07 | 2.38 | 1.28 | 1.73 | 1.12 |
| 1997 | 2.43 | 2.53 | 1.48 | 2.48 | 2.22 | 2.20 | 2.07 | 2.04 | 2.51 | 2.87 | 1.46 | 2.06 | 1.23 |
| 1998 | 3.32 | 4.12 | 2.33 | 3.86 | 3.62 | 3.59 | 3.36 | 3.13 | 4.14 | 4.70 | 2.34 | 3.33 | 1.88 |
| 1999 | 2.20 | 2.76 | 1.55 | 2.56 | 2.43 | 2.40 | 2.25 | 2.08 | 2.78 | 3.15 | 1.57 | 2.23 | 1.25 |
| 2000 | 2.42 | 2.96 | 1.72 | 2.74 | 2.61 | 2.60 | 2.45 | 2.30 | 2.98 | 3.40 | 1.76 | 2.41 | 1.41 |
| 2001 | 2.34 | 3.04 | 1.71 | 2.81 | 2.67 | 2.64 | 2.47 | 2.28 | 3.06 | 3.47 | 1.72 | 2.45 | 1.37 |
| 2002 | 3.31 | 4.24 | 2.38 | 3.95 | 3.72 | 3.69 | 3.45 | 3.19 | 4.26 | 4.84 | 2.40 | 3.42 | 1.92 |
| 2003 | 3.52 | 4.40 | 2.49 | 4.14 | 3.86 | 3.83 | 3.58 | 3.34 | 4.41 | 5.01 | 2.48 | 3.56 | 2.01 |
| 2004 | 2.91 | 3.65 | 2.07 | 3.44 | 3.21 | 3.18 | 2.97 | 2.78 | 3.66 | 4.16 | 2.07 | 2.95 | 1.67 |

Table 8.16. Characteristics of 95% confidence intervals for $H_p(10)$ and distribution of 95% confidence intervals for absorbed organ doses for reactor plant workers

| | | 95% confidence interval for absorbed organ doses, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 328.98 | 221.08 | 195.10 | 319.90 | 199.81 | 210.70 | 220.49 | 307.98 | 218.85 | 292.08 | 212.78 | 196.02 | 182.33 |
| 1949 | 968.64 | 649.77 | 579.47 | 952.38 | 589.19 | 623.64 | 655.68 | 909.76 | 645.87 | 855.42 | 636.18 | 577.31 | 548.26 |
| 1950 | 445.67 | 306.85 | 263.04 | 424.51 | 278.16 | 291.54 | 302.76 | 406.89 | 305.30 | 395.08 | 287.31 | 270.03 | 246.43 |
| 1951 | 307.71 | 268.73 | 203.51 | 300.26 | 244.77 | 251.07 | 254.29 | 286.36 | 270.23 | 318.56 | 221.38 | 231.39 | 190.38 |
| 1952 | 493.47 | 360.44 | 300.50 | 489.58 | 329.31 | 347.66 | 361.67 | 445.53 | 367.43 | 441.18 | 336.32 | 312.51 | 292.69 |
| 1953 | 256.59 | 185.69 | 153.00 | 245.05 | 166.08 | 171.58 | 175.22 | 241.12 | 181.68 | 243.83 | 163.92 | 161.65 | 138.42 |
| 1954 | 103.08 | 75.49 | 62.90 | 100.64 | 68.25 | 71.13 | 73.48 | 96.15 | 75.03 | 96.34 | 68.87 | 65.89 | 59.31 |
| 1955 | 103.98 | 73.04 | 62.87 | 101.41 | 66.31 | 69.63 | 72.55 | 96.70 | 72.80 | 93.82 | 69.04 | 64.32 | 59.55 |
| 1956 | 80.76 | 59.48 | 49.93 | 79.30 | 54.37 | 56.99 | 59.17 | 74.06 | 59.95 | 73.68 | 55.46 | 51.99 | 48.22 |
| 1957 | 68.94 | 46.24 | 40.63 | 66.44 | 42.11 | 44.52 | 46.70 | 62.18 | 46.35 | 59.58 | 45.26 | 40.68 | 39.43 |
| 1958 | 50.58 | 33.52 | 29.91 | 49.16 | 30.49 | 32.28 | 33.98 | 46.27 | 33.47 | 43.82 | 33.27 | 29.61 | 28.97 |
| 1959 | 56.56 | 39.85 | 33.30 | 53.46 | 36.19 | 37.83 | 39.16 | 50.15 | 39.94 | 50.01 | 36.88 | 34.66 | 31.99 |
| 1960 | 61.74 | 45.81 | 37.59 | 59.96 | 41.06 | 42.39 | 43.34 | 58.42 | 44.94 | 59.67 | 40.67 | 39.77 | 34.69 |
| 1961 | 44.54 | 32.07 | 26.85 | 43.11 | 28.95 | 30.16 | 31.14 | 41.13 | 31.79 | 41.28 | 29.50 | 27.93 | 25.48 |
| 1962 | 38.10 | 26.78 | 23.01 | 37.37 | 24.13 | 25.23 | 26.20 | 35.85 | 26.42 | 35.21 | 25.26 | 23.46 | 21.80 |
| 1963 | 38.15 | 26.72 | 22.68 | 36.86 | 23.89 | 24.82 | 25.59 | 35.85 | 26.09 | 35.74 | 24.66 | 23.31 | 21.12 |
| 1964 | 46.63 | 32.57 | 27.79 | 45.25 | 29.14 | 30.33 | 31.35 | 44.01 | 31.83 | 43.57 | 30.20 | 28.49 | 25.86 |
| 1965 | 33.78 | 23.52 | 20.09 | 32.76 | 21.03 | 21.88 | 22.61 | 31.80 | 22.96 | 31.57 | 21.89 | 20.54 | 18.79 |
| 1966 | 28.95 | 20.69 | 17.45 | 28.25 | 18.51 | 19.22 | 19.79 | 27.48 | 20.23 | 27.51 | 18.96 | 18.04 | 16.23 |
| 1967 | 29.75 | 20.35 | 17.68 | 28.87 | 18.30 | 19.17 | 19.92 | 27.84 | 20.00 | 27.06 | 19.39 | 17.83 | 16.68 |
| 1968 | 22.59 | 15.78 | 13.37 | 21.76 | 14.07 | 14.59 | 15.01 | 21.28 | 15.35 | 21.25 | 14.48 | 13.75 | 12.36 |
| 1969 | 23.07 | 16.03 | 13.69 | 22.30 | 14.34 | 14.92 | 15.39 | 21.70 | 15.65 | 21.48 | 14.89 | 14.00 | 12.74 |
| 1970 | 22.45 | 15.58 | 13.36 | 21.80 | 13.94 | 14.53 | 15.03 | 21.17 | 15.22 | 20.87 | 14.55 | 13.63 | 12.46 |
| 1971 | 24.58 | 17.07 | 14.58 | 23.75 | 15.27 | 15.89 | 16.40 | 23.13 | 16.67 | 22.85 | 15.83 | 14.92 | 13.52 |
| 1972 | 27.67 | 19.05 | 16.39 | 26.73 | 17.09 | 17.82 | 18.43 | 25.91 | 18.67 | 25.45 | 17.85 | 16.67 | 15.28 |
| 1973 | 25.36 | 17.81 | 15.01 | 24.43 | 15.84 | 16.40 | 16.85 | 23.92 | 17.28 | 24.07 | 16.25 | 15.50 | 13.88 |
| 1974 | 24.00 | 17.06 | 14.18 | 22.99 | 15.13 | 15.58 | 15.89 | 22.80 | 16.48 | 23.11 | 15.16 | 14.82 | 12.79 |
| 1975 | 38.15 | 26.27 | 22.58 | 36.82 | 23.56 | 24.56 | 25.39 | 35.73 | 25.73 | 35.07 | 24.57 | 22.99 | 21.00 |
| 1976 | 22.07 | 15.48 | 13.07 | 21.25 | 13.79 | 14.28 | 14.67 | 20.84 | 15.04 | 20.86 | 14.11 | 13.49 | 12.01 |

| | | 95% confidence interval for absorbed organ doses, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1977 | 20.61 | 14.57 | 12.17 | 19.75 | 12.93 | 13.35 | 13.64 | 19.53 | 14.09 | 19.69 | 13.05 | 12.67 | 11.03 |
| 1978 | 19.58 | 13.93 | 11.58 | 18.78 | 12.34 | 12.72 | 12.98 | 18.60 | 13.45 | 18.88 | 12.40 | 12.10 | 10.50 |
| 1979 | 18.43 | 13.06 | 10.87 | 17.62 | 11.57 | 11.93 | 12.18 | 17.48 | 12.60 | 17.72 | 11.63 | 11.35 | 9.84 |
| 1980 | 21.35 | 14.97 | 12.59 | 20.45 | 13.32 | 13.78 | 14.12 | 20.13 | 14.52 | 20.19 | 13.56 | 13.03 | 11.52 |
| 1981 | 22.36 | 15.25 | 13.30 | 21.75 | 13.74 | 14.40 | 14.98 | 20.89 | 15.03 | 20.24 | 14.58 | 13.39 | 12.54 |
| 1982 | 20.70 | 14.30 | 12.27 | 20.02 | 12.81 | 13.35 | 13.80 | 19.46 | 13.99 | 19.12 | 13.33 | 12.52 | 11.39 |
| 1983 | 18.56 | 12.89 | 10.99 | 17.90 | 11.52 | 11.98 | 12.35 | 17.47 | 12.58 | 17.28 | 11.90 | 11.27 | 10.14 |
| 1984 | 14.48 | 10.13 | 8.55 | 13.90 | 9.03 | 9.35 | 9.59 | 13.65 | 9.85 | 13.64 | 9.22 | 8.83 | 7.82 |
| 1985 | 14.49 | 10.34 | 8.53 | 13.78 | 9.15 | 9.40 | 9.54 | 13.78 | 9.96 | 14.03 | 9.06 | 8.97 | 7.59 |
| 1986 | 35.65 | 24.08 | 21.23 | 34.80 | 21.75 | 22.93 | 23.99 | 33.47 | 23.82 | 31.84 | 23.19 | 21.32 | 19.90 |
| 1987 | 18.79 | 13.27 | 11.04 | 17.87 | 11.78 | 12.13 | 12.36 | 17.77 | 12.83 | 17.92 | 11.77 | 11.53 | 9.89 |
| 1988 | 13.80 | 9.84 | 8.12 | 13.14 | 8.71 | 8.95 | 9.09 | 13.12 | 9.48 | 13.35 | 8.64 | 8.54 | 7.25 |
| 1989 | 13.99 | 9.67 | 8.30 | 13.54 | 8.66 | 9.03 | 9.34 | 13.13 | 9.46 | 12.95 | 9.04 | 8.46 | 7.75 |
| 1990 | 8.93 | 6.26 | 5.29 | 8.62 | 5.58 | 5.78 | 5.95 | 8.42 | 6.08 | 8.47 | 5.75 | 5.45 | 4.93 |
| 1991 | 9.94 | 6.76 | 5.94 | 9.68 | 6.12 | 6.44 | 6.71 | 9.28 | 6.70 | 8.88 | 6.52 | 5.95 | 5.61 |
| 1992 | 6.33 | 3.81 | 3.26 | 5.30 | 3.41 | 3.55 | 3.66 | 5.17 | 3.72 | 5.09 | 3.53 | 3.33 | 3.00 |
| 1993 | 5.97 | 3.38 | 2.88 | 4.69 | 3.03 | 3.15 | 3.24 | 4.59 | 3.30 | 4.53 | 3.12 | 2.96 | 2.65 |
| 1994 | 4.37 | 2.58 | 2.22 | 3.60 | 2.33 | 2.44 | 2.53 | 3.47 | 2.55 | 3.37 | 2.44 | 2.26 | 2.10 |
| 1995 | 4.35 | 2.51 | 2.19 | 3.57 | 2.26 | 2.37 | 2.46 | 3.46 | 2.47 | 3.33 | 2.40 | 2.20 | 2.06 |
| 1996 | 4.54 | 2.61 | 2.28 | 3.72 | 2.36 | 2.47 | 2.57 | 3.60 | 2.57 | 3.46 | 2.50 | 2.29 | 2.14 |
| 1997 | 3.56 | 2.02 | 1.77 | 2.89 | 1.82 | 1.92 | 2.00 | 2.79 | 1.99 | 2.67 | 1.95 | 1.77 | 1.67 |
| 1998 | 4.30 | 2.66 | 2.22 | 3.60 | 2.36 | 2.44 | 2.49 | 3.57 | 2.57 | 3.60 | 2.38 | 2.31 | 2.00 |
| 1999 | 4.35 | 2.73 | 2.27 | 3.68 | 2.42 | 2.50 | 2.54 | 3.65 | 2.64 | 3.70 | 2.43 | 2.37 | 2.04 |
| 2000 | 3.77 | 2.18 | 1.93 | 3.16 | 1.98 | 2.08 | 2.17 | 3.01 | 2.16 | 2.88 | 2.12 | 1.92 | 1.83 |
| 2001 | 3.30 | 2.02 | 1.71 | 2.78 | 1.80 | 1.87 | 1.92 | 2.72 | 1.96 | 2.71 | 1.84 | 1.76 | 1.56 |
| 2002 | 2.87 | 1.90 | 1.52 | 2.44 | 1.66 | 1.69 | 1.69 | 2.49 | 1.81 | 2.61 | 1.58 | 1.64 | 1.31 |
| 2003 | 2.96 | 1.95 | 1.57 | 2.52 | 1.71 | 1.74 | 1.75 | 2.56 | 1.86 | 2.67 | 1.64 | 1.68 | 1.35 |
| 2004 | 3.09 | 2.03 | 1.63 | 2.62 | 1.78 | 1.81 | 1.82 | 2.67 | 1.94 | 2.78 | 1.70 | 1.75 | 1.41 |

Table 8.17. Characteristics of 95% confidence intervals for $H_p(10)$ and distribution of 95% confidence intervals for absorbed organ doses for radiochemical plant workers

| | | 95% confidence interval for absorbed organ doses, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 145.34 | 140.12 | 99.25 | 141.22 | 127.42 | 129.60 | 129.27 | 134.46 | 141.08 | 161.85 | 106.92 | 119.33 | 90.22 |
| 1949 | 380.50 | 462.56 | 297.01 | 380.29 | 415.99 | 411.70 | 400.14 | 384.14 | 458.41 | 527.68 | 309.81 | 388.94 | 260.07 |
| 1950 | 829.32 | 1004.97 | 652.20 | 847.47 | 902.56 | 894.98 | 870.23 | 855.02 | 994.79 | 1160.95 | 678.71 | 846.62 | 566.71 |
| 1951 | 905.51 | 1115.35 | 718.68 | 932.36 | 1001.43 | 990.82 | 961.54 | 939.52 | 1103.19 | 1280.44 | 742.26 | 939.27 | 624.29 |
| 1952 | 491.39 | 611.57 | 391.49 | 499.72 | 549.60 | 543.58 | 527.61 | 506.78 | 605.70 | 699.10 | 407.16 | 514.07 | 341.15 |
| 1953 | 251.33 | 315.12 | 202.19 | 257.23 | 283.38 | 280.49 | 272.47 | 261.21 | 312.23 | 360.02 | 210.18 | 265.11 | 176.26 |
| 1954 | 262.82 | 403.13 | 256.60 | 323.42 | 362.28 | 358.06 | 347.16 | 329.89 | 399.19 | 460.50 | 265.87 | 338.87 | 222.46 |
| 1955 | 287.83 | 430.58 | 274.82 | 347.89 | 386.89 | 382.57 | 371.08 | 354.40 | 426.37 | 492.86 | 285.01 | 362.03 | 238.27 |
| 1956 | 259.86 | 389.22 | 248.84 | 313.98 | 350.00 | 346.27 | 336.16 | 320.29 | 385.59 | 444.49 | 258.22 | 327.38 | 216.30 |
| 1957 | 276.36 | 412.53 | 264.31 | 335.16 | 370.80 | 366.89 | 356.20 | 341.50 | 408.54 | 472.25 | 274.57 | 347.01 | 229.76 |
| 1958 | 179.20 | 260.87 | 167.06 | 211.26 | 234.64 | 232.25 | 225.58 | 215.23 | 258.55 | 297.77 | 173.71 | 219.41 | 145.55 |
| 1959 | 168.26 | 237.57 | 153.67 | 196.29 | 213.82 | 212.10 | 206.46 | 199.25 | 235.60 | 271.73 | 160.25 | 200.05 | 134.43 |
| 1960 | 175.94 | 263.85 | 168.83 | 213.01 | 237.34 | 234.93 | 228.16 | 217.22 | 261.54 | 301.08 | 175.44 | 221.91 | 146.97 |
| 1961 | 124.55 | 191.89 | 121.73 | 152.72 | 172.44 | 170.33 | 165.07 | 156.07 | 190.05 | 218.82 | 126.19 | 161.19 | 105.57 |
| 1962 | 97.02 | 148.03 | 94.29 | 118.70 | 133.08 | 131.59 | 127.66 | 121.11 | 146.67 | 168.90 | 97.84 | 124.42 | 81.88 |
| 1963 | 41.37 | 65.22 | 41.05 | 51.25 | 58.56 | 57.73 | 55.84 | 52.48 | 64.54 | 74.30 | 42.47 | 54.72 | 35.50 |
| 1964 | 30.47 | 47.83 | 30.07 | 37.66 | 42.93 | 42.30 | 40.90 | 38.51 | 47.32 | 54.52 | 31.10 | 40.12 | 25.99 |
| 1965 | 27.97 | 43.59 | 27.40 | 34.22 | 39.13 | 38.56 | 37.29 | 35.03 | 43.13 | 49.64 | 28.34 | 36.57 | 23.69 |
| 1966 | 25.27 | 40.11 | 25.14 | 31.38 | 35.99 | 35.44 | 34.24 | 32.12 | 39.67 | 45.67 | 25.97 | 33.63 | 21.72 |
| 1967 | 21.94 | 33.77 | 21.33 | 26.79 | 30.32 | 29.94 | 28.98 | 27.31 | 33.44 | 38.49 | 22.08 | 28.34 | 18.45 |
| 1968 | 22.60 | 34.82 | 22.09 | 27.75 | 31.28 | 30.89 | 29.93 | 28.36 | 34.47 | 39.74 | 22.88 | 29.25 | 19.14 |
| 1969 | 16.34 | 25.65 | 16.04 | 19.92 | 23.02 | 22.65 | 21.88 | 20.45 | 25.37 | 29.20 | 16.57 | 21.50 | 13.84 |
| 1970 | 32.46 | 49.71 | 31.72 | 39.85 | 44.71 | 44.23 | 42.93 | 40.72 | 49.27 | 56.72 | 32.92 | 41.81 | 27.56 |
| 1971 | 16.90 | 25.80 | 16.26 | 20.32 | 23.17 | 22.85 | 22.11 | 20.80 | 25.53 | 29.39 | 16.83 | 21.65 | 14.08 |
| 1972 | 13.21 | 20.68 | 13.06 | 16.30 | 18.58 | 18.33 | 17.75 | 16.70 | 20.47 | 23.57 | 13.52 | 17.36 | 11.30 |
| 1973 | 13.92 | 22.53 | 14.00 | 17.33 | 20.20 | 19.86 | 19.15 | 17.80 | 22.27 | 25.63 | 14.43 | 18.87 | 12.04 |
| 1974 | 12.77 | 19.39 | 12.25 | 15.32 | 17.41 | 17.18 | 16.63 | 15.67 | 19.19 | 22.09 | 12.68 | 16.27 | 10.61 |
| 1975 | 12.56 | 18.18 | 11.61 | 14.65 | 16.35 | 16.17 | 15.70 | 14.92 | 18.02 | 20.74 | 12.06 | 15.29 | 10.11 |
| 1976 | 18.27 | 27.96 | 17.81 | 22.35 | 25.14 | 24.86 | 24.12 | 22.84 | 27.71 | 31.90 | 18.47 | 23.51 | 15.46 |

| | | 95% confidence interval for absorbed organ doses, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1977 | 11.98 | 18.67 | 11.78 | 14.70 | 16.77 | 16.54 | 16.01 | 15.06 | 18.48 | 21.28 | 12.19 | 15.67 | 10.19 |
| 1978 | 11.37 | 17.70 | 11.10 | 13.81 | 15.89 | 15.65 | 15.12 | 14.16 | 17.51 | 20.15 | 11.47 | 14.84 | 9.58 |
| 1979 | 11.91 | 18.99 | 11.88 | 14.75 | 17.04 | 16.78 | 16.21 | 15.15 | 18.78 | 21.62 | 12.27 | 15.92 | 10.25 |
| 1980 | 9.32 | 14.84 | 9.30 | 11.54 | 13.32 | 13.12 | 12.67 | 11.85 | 14.68 | 16.90 | 9.60 | 12.44 | 8.02 |
| 1981 | 7.67 | 11.89 | 7.50 | 9.37 | 10.68 | 10.54 | 10.20 | 9.60 | 11.77 | 13.55 | 7.77 | 9.98 | 6.49 |
| 1982 | 8.92 | 14.05 | 8.84 | 11.02 | 12.61 | 12.44 | 12.03 | 11.30 | 13.90 | 16.00 | 9.15 | 11.79 | 7.65 |
| 1983 | 10.31 | 16.20 | 10.21 | 12.72 | 14.55 | 14.35 | 13.88 | 13.04 | 16.04 | 18.46 | 10.55 | 13.60 | 8.82 |
| 1984 | 9.01 | 14.07 | 8.91 | 11.14 | 12.65 | 12.49 | 12.09 | 11.40 | 13.94 | 16.04 | 9.23 | 11.82 | 7.72 |
| 1985 | 9.12 | 14.18 | 8.98 | 11.23 | 12.75 | 12.59 | 12.19 | 11.50 | 14.05 | 16.17 | 9.30 | 11.91 | 7.78 |
| 1986 | 24.35 | 37.18 | 23.77 | 29.90 | 33.44 | 33.10 | 32.14 | 30.53 | 36.85 | 42.43 | 24.67 | 31.27 | 20.66 |
| 1987 | 23.34 | 35.39 | 22.67 | 28.54 | 31.84 | 31.53 | 30.64 | 29.13 | 35.09 | 40.39 | 23.54 | 29.78 | 19.72 |
| 1988 | 10.20 | 16.01 | 10.15 | 12.67 | 14.39 | 14.21 | 13.77 | 12.99 | 15.85 | 18.25 | 10.51 | 13.45 | 8.79 |
| 1989 | 9.91 | 15.68 | 9.90 | 12.34 | 14.09 | 13.90 | 13.46 | 12.66 | 15.52 | 17.87 | 10.25 | 13.16 | 8.57 |
| 1990 | 7.43 | 11.54 | 7.31 | 9.13 | 10.37 | 10.24 | 9.92 | 9.35 | 11.43 | 13.16 | 7.57 | 9.70 | 6.33 |
| 1991 | 7.64 | 10.83 | 6.98 | 8.87 | 9.75 | 9.66 | 9.40 | 9.01 | 10.74 | 12.37 | 7.27 | 9.12 | 6.10 |
| 1992 | 6.52 | 7.59 | 4.91 | 6.25 | 6.84 | 6.78 | 6.60 | 6.34 | 7.53 | 8.67 | 5.11 | 6.40 | 4.29 |
| 1993 | 5.62 | 6.06 | 4.01 | 5.24 | 5.47 | 5.44 | 5.33 | 5.23 | 6.02 | 6.93 | 4.22 | 5.12 | 3.56 |
| 1994 | 4.69 | 5.59 | 3.57 | 4.49 | 5.03 | 4.97 | 4.83 | 4.58 | 5.54 | 6.38 | 3.70 | 4.70 | 3.10 |
| 1995 | 5.25 | 6.36 | 4.06 | 5.10 | 5.72 | 5.66 | 5.49 | 5.21 | 6.30 | 7.26 | 4.21 | 5.35 | 3.53 |
| 1996 | 6.02 | 7.26 | 4.65 | 5.84 | 6.54 | 6.47 | 6.28 | 5.97 | 7.20 | 8.29 | 4.82 | 6.11 | 4.04 |
| 1997 | 4.13 | 3.91 | 2.50 | 3.16 | 3.51 | 3.48 | 3.38 | 3.22 | 3.87 | 4.46 | 2.60 | 3.29 | 2.18 |
| 1998 | 3.06 | 3.63 | 2.32 | 2.94 | 3.26 | 3.23 | 3.13 | 2.99 | 3.60 | 4.14 | 2.41 | 3.05 | 2.02 |
| 1999 | 3.56 | 4.32 | 2.75 | 3.45 | 3.88 | 3.84 | 3.72 | 3.52 | 4.28 | 4.93 | 2.85 | 3.63 | 2.38 |
| 2000 | 3.83 | 4.63 | 2.96 | 3.72 | 4.17 | 4.12 | 4.00 | 3.79 | 4.59 | 5.29 | 3.07 | 3.90 | 2.57 |
| 2001 | 2.95 | 3.43 | 2.20 | 2.80 | 3.08 | 3.05 | 2.96 | 2.84 | 3.40 | 3.91 | 2.29 | 2.88 | 1.92 |
| 2002 | 3.14 | 3.77 | 2.40 | 3.02 | 3.39 | 3.35 | 3.25 | 3.08 | 3.74 | 4.30 | 2.49 | 3.17 | 2.09 |
| 2003 | 3.10 | 3.68 | 2.35 | 2.97 | 3.31 | 3.27 | 3.17 | 3.02 | 3.64 | 4.20 | 2.44 | 3.09 | 2.05 |
| 2004 | 2.78 | 3.29 | 2.10 | 2.65 | 2.95 | 2.92 | 2.84 | 2.70 | 3.26 | 3.75 | 2.18 | 2.76 | 1.83 |

Table 8.18. Characteristics of 95% confidence intervals for $H_p(10)$ and distribution of 95% confidence intervals for absorbed organ doses for plutonium plant workers

| | | 95% confidence interval for absorbed organ doses, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1948 | 20.18 | 22.17 | 8.64 | 6.05 | 19.54 | 18.10 | 15.72 | 7.86 | 22.15 | 23.21 | 7.73 | 17.59 | 5.58 |
| 1949 | 37.81 | 29.81 | 23.10 | 36.14 | 27.14 | 28.12 | 28.55 | 33.61 | 30.02 | 34.84 | 25.23 | 25.60 | 21.94 |
| 1950 | 326.99 | 328.88 | 224.69 | 315.05 | 297.60 | 300.95 | 297.35 | 307.69 | 332.28 | 384.11 | 238.78 | 280.16 | 200.61 |
| 1951 | 333.69 | 326.90 | 224.56 | 316.36 | 295.81 | 299.34 | 296.02 | 307.88 | 330.03 | 381.39 | 239.29 | 278.31 | 201.34 |
| 1952 | 172.77 | 164.67 | 114.22 | 163.74 | 148.97 | 150.76 | 149.42 | 156.80 | 166.12 | 191.59 | 122.51 | 140.05 | 103.80 |
| 1953 | 109.55 | 103.55 | 69.94 | 99.65 | 93.33 | 94.13 | 92.58 | 95.96 | 104.30 | 120.44 | 74.37 | 87.62 | 62.39 |
| 1954 | 123.39 | 121.50 | 77.86 | 117.03 | 108.49 | 109.09 | 105.61 | 106.42 | 122.52 | 140.48 | 82.12 | 100.86 | 67.72 |
| 1955 | 90.72 | 89.25 | 55.40 | 84.17 | 79.38 | 79.55 | 76.38 | 75.09 | 90.09 | 103.06 | 57.99 | 73.63 | 47.33 |
| 1956 | 144.98 | 129.08 | 87.26 | 136.01 | 115.86 | 117.94 | 115.75 | 120.74 | 131.14 | 149.89 | 93.78 | 107.89 | 78.40 |
| 1957 | 102.77 | 97.04 | 61.22 | 93.64 | 86.40 | 86.86 | 83.69 | 83.22 | 98.02 | 112.04 | 64.46 | 80.20 | 52.95 |
| 1958 | 86.25 | 84.38 | 50.60 | 77.36 | 74.77 | 74.63 | 71.04 | 67.64 | 85.16 | 97.20 | 52.36 | 69.24 | 42.26 |
| 1959 | 92.29 | 91.63 | 54.72 | 83.47 | 81.17 | 80.96 | 76.99 | 73.01 | 92.46 | 105.53 | 56.56 | 75.15 | 45.59 |
| 1960 | 119.00 | 121.54 | 72.79 | 109.01 | 107.80 | 107.35 | 102.18 | 96.38 | 122.38 | 139.75 | 75.02 | 99.92 | 60.70 |
| 1961 | 74.81 | 74.28 | 44.56 | 68.01 | 65.52 | 65.26 | 62.14 | 59.95 | 74.39 | 84.98 | 45.76 | 60.79 | 37.45 |
| 1962 | 56.87 | 57.45 | 33.80 | 51.02 | 50.58 | 50.35 | 47.78 | 44.72 | 57.62 | 65.71 | 34.90 | 46.82 | 28.05 |
| 1963 | 29.03 | 30.58 | 17.80 | 27.10 | 26.90 | 26.71 | 25.26 | 23.58 | 30.64 | 34.92 | 18.27 | 24.87 | 14.72 |
| 1964 | 20.62 | 23.98 | 13.67 | 21.81 | 21.06 | 20.89 | 19.63 | 18.26 | 24.10 | 27.39 | 13.94 | 19.41 | 11.18 |
| 1965 | 13.37 | 14.83 | 8.61 | 13.53 | 13.04 | 12.96 | 12.25 | 11.59 | 14.92 | 16.97 | 8.88 | 12.03 | 7.16 |
| 1966 | 10.46 | 12.06 | 6.97 | 11.01 | 10.61 | 10.55 | 9.94 | 9.37 | 12.15 | 13.82 | 7.18 | 9.78 | 5.77 |
| 1967 | 8.67 | 10.28 | 5.91 | 9.64 | 9.03 | 8.98 | 8.44 | 7.95 | 10.32 | 11.73 | 6.01 | 8.33 | 4.86 |
| 1968 | 5.28 | 6.18 | 3.59 | 5.73 | 5.44 | 5.41 | 5.10 | 4.81 | 6.21 | 7.06 | 3.66 | 5.02 | 2.95 |
| 1969 | 8.12 | 10.53 | 6.01 | 9.85 | 9.25 | 9.17 | 8.60 | 8.06 | 10.55 | 12.00 | 6.04 | 8.53 | 4.88 |
| 1970 | 9.94 | 12.44 | 7.41 | 11.91 | 10.98 | 10.89 | 10.31 | 10.10 | 12.41 | 14.17 | 7.41 | 10.19 | 6.21 |
| 1971 | 6.30 | 7.42 | 4.55 | 7.05 | 6.58 | 6.55 | 6.26 | 6.18 | 7.42 | 8.48 | 4.66 | 6.11 | 3.88 |
| 1972 | 7.08 | 8.57 | 4.85 | 7.96 | 7.53 | 7.46 | 6.98 | 6.50 | 8.60 | 9.77 | 4.87 | 6.94 | 3.93 |
| 1973 | 5.15 | 7.44 | 4.19 | 6.95 | 6.54 | 6.48 | 6.06 | 5.59 | 7.48 | 8.49 | 4.23 | 6.01 | 3.37 |
| 1974 | 54.44 | 52.94 | 30.94 | 48.44 | 46.47 | 46.25 | 43.84 | 41.60 | 53.02 | 60.45 | 31.69 | 43.00 | 25.63 |
| 1975 | 9.95 | 14.44 | 8.18 | 13.44 | 12.70 | 12.61 | 11.82 | 10.90 | 14.54 | 16.51 | 8.30 | 11.69 | 6.60 |
| 1976 | 9.15 | 13.28 | 7.54 | 12.50 | 11.67 | 11.59 | 10.86 | 10.11 | 13.35 | 15.16 | 7.64 | 10.74 | 6.12 |

| | | 95% confidence interval for absorbed organ doses, mGy | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Year | Hp10, mSv | Bladder | Red Marrow | Bone Surface | Colon | Liver | Lung | Skin | Stomach | Thyroid | Brain | Lower Lg Intestine | Kidney |
| 1977 | 9.37 | 13.12 | 7.52 | 12.41 | 11.54 | 11.47 | 10.78 | 10.12 | 13.20 | 14.99 | 7.62 | 10.63 | 6.13 |
| 1978 | 9.57 | 14.42 | 8.07 | 13.36 | 12.66 | 12.55 | 11.73 | 10.74 | 14.51 | 16.47 | 8.14 | 11.64 | 6.47 |
| 1979 | 8.87 | 12.55 | 7.24 | 11.98 | 11.04 | 10.96 | 10.29 | 9.84 | 12.61 | 14.33 | 7.25 | 10.20 | 5.95 |
| 1980 | 8.64 | 12.51 | 7.09 | 11.70 | 11.00 | 10.91 | 10.22 | 9.49 | 12.59 | 14.29 | 7.16 | 10.12 | 5.73 |
| 1981 | 9.45 | 12.61 | 7.52 | 12.48 | 11.11 | 11.04 | 10.42 | 10.43 | 12.61 | 14.38 | 7.40 | 10.33 | 6.29 |
| 1982 | 8.88 | 13.36 | 7.51 | 12.43 | 11.74 | 11.64 | 10.89 | 10.01 | 13.44 | 15.26 | 7.57 | 10.80 | 6.03 |
| 1983 | 11.81 | 17.76 | 10.02 | 16.62 | 15.61 | 15.49 | 14.50 | 13.36 | 17.88 | 20.29 | 10.14 | 14.36 | 8.09 |
| 1984 | 10.86 | 15.48 | 8.78 | 14.56 | 13.60 | 13.49 | 12.64 | 11.78 | 15.55 | 17.67 | 8.84 | 12.53 | 7.10 |
| 1985 | 9.33 | 13.78 | 7.75 | 12.86 | 12.10 | 12.00 | 11.22 | 10.35 | 13.85 | 15.73 | 7.80 | 11.14 | 6.23 |
| 1986 | 33.79 | 31.09 | 20.72 | 32.67 | 27.77 | 27.90 | 27.18 | 29.43 | 30.94 | 35.71 | 21.24 | 26.03 | 18.35 |
| 1987 | 22.09 | 20.15 | 12.82 | 21.63 | 17.78 | 17.54 | 16.78 | 18.54 | 19.62 | 22.67 | 12.20 | 16.75 | 11.17 |
| 1988 | 16.99 | 12.76 | 10.58 | 17.35 | 11.70 | 12.44 | 12.81 | 15.71 | 13.07 | 15.00 | 11.87 | 10.98 | 10.34 |
| 1989 | 7.32 | 9.61 | 5.60 | 9.29 | 8.46 | 8.40 | 7.91 | 7.70 | 9.65 | 10.98 | 5.62 | 7.82 | 4.65 |
| 1990 | 6.96 | 9.25 | 5.40 | 8.82 | 8.15 | 8.12 | 7.66 | 7.34 | 9.31 | 10.59 | 5.50 | 7.53 | 4.47 |
| 1991 | 6.88 | 8.06 | 4.86 | 8.05 | 7.11 | 7.05 | 6.68 | 6.79 | 8.03 | 9.17 | 4.80 | 6.61 | 4.11 |
| 1992 | 3.16 | 3.53 | 2.12 | 3.40 | 3.12 | 3.10 | 2.95 | 2.90 | 3.54 | 4.04 | 2.15 | 2.89 | 1.78 |
| 1993 | 3.76 | 4.33 | 2.58 | 4.22 | 3.82 | 3.80 | 3.59 | 3.56 | 4.34 | 4.95 | 2.58 | 3.55 | 2.16 |
| 1994 | 2.95 | 2.91 | 1.78 | 2.85 | 2.57 | 2.56 | 2.44 | 2.46 | 2.90 | 3.32 | 1.79 | 2.39 | 1.51 |
| 1995 | 4.88 | 4.88 | 2.95 | 4.82 | 4.31 | 4.29 | 4.08 | 4.10 | 4.88 | 5.57 | 2.98 | 3.99 | 2.50 |
| 1996 | 4.29 | 4.10 | 2.55 | 4.20 | 3.62 | 3.60 | 3.43 | 3.62 | 4.07 | 4.66 | 2.51 | 3.39 | 2.20 |
| 1997 | 4.77 | 4.95 | 2.90 | 4.85 | 4.36 | 4.31 | 4.06 | 4.00 | 4.93 | 5.63 | 2.86 | 4.04 | 2.41 |
| 1998 | 6.51 | 8.07 | 4.57 | 7.57 | 7.10 | 7.03 | 6.59 | 6.13 | 8.11 | 9.21 | 4.60 | 6.54 | 3.69 |
| 1999 | 4.31 | 5.41 | 3.05 | 5.02 | 4.75 | 4.71 | 4.41 | 4.07 | 5.44 | 6.18 | 3.08 | 4.37 | 2.46 |
| 2000 | 4.75 | 5.81 | 3.36 | 5.37 | 5.12 | 5.09 | 4.80 | 4.51 | 5.85 | 6.65 | 3.45 | 4.72 | 2.77 |
| 2001 | 4.59 | 5.96 | 3.34 | 5.51 | 5.23 | 5.18 | 4.84 | 4.46 | 5.99 | 6.80 | 3.37 | 4.81 | 2.69 |
| 2002 | 6.48 | 8.30 | 4.67 | 7.75 | 7.29 | 7.23 | 6.76 | 6.26 | 8.35 | 9.48 | 4.71 | 6.71 | 3.77 |
| 2003 | 6.90 | 8.62 | 4.87 | 8.11 | 7.57 | 7.50 | 7.01 | 6.56 | 8.65 | 9.83 | 4.87 | 6.97 | 3.94 |
| 2004 | 5.70 | 7.15 | 4.05 | 6.74 | 6.28 | 6.23 | 5.83 | 5.46 | 7.18 | 8.16 | 4.05 | 5.79 | 3.28 |

## 8.3 PLUTONIUM ORGAN DOSES

The estimation of plutonium organ and tissue doses is presented in considerable detail in Volume III of this study. The Doses-2005 internal dosimetry model for plutonium is a combined "lung" and "systemic" model. Both of these individual models have been published in the peer-reviewed, public domain literature. The "lung clearance" model describes the transfer of plutonium compounds from the lung to the systemic circulation. This includes consideration of smoking status and the "transportability" of the various industrial aerosols. This is described in detail in Khokhryakov et al. (2005) and summarized in Volume III of this study.

Once in the circulation, plutonium is incorporated into the various organs. The longer-term retention functions of plutonium in these organs are derived from empirical observations and used to derive the appropriate organ doses as a function of time. The transfer coefficients for the movement of plutonium between the various mathematical compartments that represent components of the body are presented in Volume III of this study. This model is also described in detail in Leggett et al. (2005).

The uncertainties of the internal dosimetry depend on a number of factors. Using an approach that was used for the analysis of workers at other nuclear facilities, we have published the various factors that should be considered in the uncertainty calculations. Our approach and the initial presentation of uncertainties for the organ doses derived from the prior dosimetry system (Doses-2000) have been published (see Krahenbuhl et al. 2005).

In addition, the application of ICRP dose coefficients for systemic biokinetics of plutonium is also discussed in detail in another published work from this study (see Leggett 2003).

Uncertainty analyses for the new dosimetry system, Doses-2005, are ongoing. A preliminary correlation of uncertainties for the prior and new dosimetry systems will be presented at a meeting in the summer of 2006.

**Plutonium Organ Dose Analysis File**

The plutonium organ dose analysis file provides output based on the Doses-2005 model described in detail in Volume III and associated peer-reviewed, public domain publications. The Doses-2005 dosimetry system can respond to queries and the output files can be structured as desired by the -user. Typically, for Project 2.2, investigators at SUBI can directly access the Dosimetry system database and transfer output files and records, as needed. Typically, specific file domains are directly exported into programs such as EPICURE or other biostatistical and/or epidemiological software programs.

The organ components for the pulmonary model include three blocks of 13 components representing activity fractions subject to: bound compartments in the lung; rapid dissolution, and; slow dissolution. In addition the model has a gastrointestinal component.

The organ components and compartments for the systemic model include:

- Intermediate, rapid and slow turnover compartments in a variety of soft tissue organs
- Blood compartments

- Liver compartments
- Skeletal compartments, including cortical volume, cortical surface, cortical marrow, trabecular volume, trabecular surface, and trabecular marrow
- Kidney and urine excretion compartments
- GIT, biliary, and fecal excretion pathways
- Gonad; testes and ovaries

Information associated with the specific dosimetry output files includes all input files, including smoking history, late-in-life health effects as they relate to dosimetry, autopsy data (if available), bioassay data (if available), occupational histories and clinical records. Some limited data are also available for some workers on $^{241}$Am content determined by whole-body counting. Similarly, external dosimetry records can also be accessed through the same database, as detailed elsewhere in this report.

# 9.0 Doses-2005 Overview

## 9.1 INTRODUCTION

Doses-2005 contains two analysis files: (1) organ dose from external radiation and (2) organ dose from plutonium intake. Considerations in the analysis of these files are provided in this chapter. Doses-2005 represents a substantial improvement in organ dose estimates compared to Phase I, Doses-1999, and Phase II, Doses-2000. However, limitations in Doses-2005 remain as described in this document. Further improvement in Doses-2005 is readily achievable as noted under proposed Phase IV tasks.

## 9.2 EPIDEMIOLOGICAL ANALYSIS CONSIDERATIONS

This document provides an overview of the methods used to arrive at the respective records in the Doses-2005 Database Analysis Files for each worker and each year of exposure. References are made to several supporting technical Project 2.4 documents and published manuscripts to address specific technical issues. The primary need in epidemiological analyses is the respective total organ radiation dose. The Doses-2005 Database Analysis Files contain organ doses for the 18 organs identified in the dosimetry protocol for each year of worker employment in two analysis files, one for external dose and another for plutonium intake. These are provided as separate files because of the significant differences in the dose parameters. Detailed descriptions of the methods to calculate organ doses from external exposure and plutonium intakes are provided in Volumes II and III, respectively, of the Doses-2005 Users Guide Dose Assignment Methodology documents. As described in Chapter 7, scoping studies were done of other exposure pathways to estimate the potential significance to the occupational dose. Evaluated exposure pathways included: neutron radiation, nuclide intake other than plutonium, airborne effluent and medical x-rays. A summary of the general findings of these studies regarding the potential magnitude of dose is provided in this section. In the following sections, the perceived strengths and limitations of the Mayak Worker Dosimetry Study are described.

## 9.3 DOSES-2005 STRENGTHS

There are notable strengths of the Mayak Dosimetry Study that appear to be unique. Perhaps the most significant are the magnitude of external radiation penetrating photon dose and plutonium intakes. The Mayak worker cohort offers a unique opportunity to compute site-specific cancer mortality risks in a group containing male and female workers exposed to ionizing radiation at levels higher than the DOE workforce. Unlike the Japanese atomic bomb survivors, Mayak workers were exposed like DOE workers for protracted periods and, therefore, might be more likely to help researchers better characterize occupational radiation risk. The US National Academy of Sciences National Research Council Biological Effects of Ionizing Radiation (BEIR) VII Committee, *Health Risks from Low Level Ionizing Radiation*, has identified the Mayak Worker Health Study as important to future assessment of radiation risks to workers and noted this study as *one of the most informative studies of occupational workers* (NAS 2006).

### 9.3.1 Archive Records

As described in Chapter 4, there are extensive archive records for Mayak workers for recorded dose and work history dating to the earliest years of Mayak operations. There are also medical records for many workers that provide a separate means to validate work history records. There are also some records of worker radioassays to assess intakes and levels of radiation in the environs.

### 9.3.2 Dosimeter Data

As described in Chapter 4, in addition to the annual dose records, the early dosimetry data were computerized in a "Daily" database containing 725,350 records obtained from Worker Personal Logs for approximately 8,748 workers in the MPA archives. The logs contain (1) the dose result for each processed dosimeter, (2) the start and end date of the monitoring period, (3) the number of shifts the dosimeter was worn, and (4) the signature of the dosimetry technician. Some of the dose results are for a single monitoring shift. These data provide a means to assess day-to-day patterns in worker exposures.

### 9.3.3 Plutonium Intake

The number and magnitude of plutonium intakes is greater than for any other occupational worker intake from comparable studies. The data are extensive and several further examinations are expected to achieve the full benefits of these data.

## 9.4 DOSES-2005 LIMITATIONS

The primary focus of the original Mayak Worker Dosimetry Study was to assess dose from external radiation and plutonium intakes. Considering organ dose from these two primary pathways of worker exposure seemed incomplete without some analysis of the potential organ dose from other potential sources of significant dose. An overview of the results of scoping analyses of the potential sources of exposure is described in the following sections.

### 9.4.1 Neutron Dose

MPA workers in the reactor and plutonium handling facilities were exposed to neutron radiation in addition to photon radiation. Although workers were assigned ionization and film dosimeters to measure the photon radiation, there was no equivalent monitoring of the neutron dose. Actual routine monitoring of neutron doses for many MPA workers in the reactor facilities began in 1973 using the DINA dosimeter for higher doses, and in 1984 at the radiochemical plant and 1986 in the plutonium facility using a dosimeter capable of measuring intermediate and fast neutron dose at occupational levels. The scoping study conducted showed that the average neutron dose to Mayak workers was relatively low compared to the photon dose. However, for some workers in some work facilities, significant neutron dose is expected. Further analysis is necessary to demonstrate that even in situations with higher neutron doses that the corresponding photon dose is still predominant.

### 9.4.2 Nuclide Intake Dose Other than Plutonium

Sources of potential occupational internal dose other than plutonium should be considered, as briefly examined in Section 7.2. Workers involved in resolving incidents could have received significant dose. Respiratory protection for individual workers was first implemented in 1956 for selected groups of workers and achieved widespread use by 1958. The Original Mayak Worker Cohort included only selected employees who worked at the Mayak industrial complex from the beginning of operations to the early 1970s based on the quality of their health and dosimetry records, and their work history. Workers exposed to nuclides other than plutonium at the Mayak industrial complex such as tritium, $^{238}$Pu, $^{90}$Sr, $^{241}$Am, and a range of other radionuclides were excluded in the selection of the original Mayak Worker Cohort. For example, production of $^{238}$Pu and $^{90}$Sr for power sources began in the 1960s using production reactor facilities that began production later than the period of the highest occupational exposures in the

late 1940s to mid-1950s for workers at the plutonium production reactors, radiochemical and plutonium plants.

### 9.4.3 Airborne Effluent

Assessment of the environmental pathway exposure to Mayak workers involves consideration of their potential increased exposure on the Mayak Site, particularly if they are unmonitored, and through their living in the environs of Mayak. Project 1.1 has assessed the dose to members of the public from Mayak releases. A collaborative effort between Projects 2.4 and 1.4 evaluated the potential significance of exposure to workers on the Mayak site from effluents. This source of worker dose is examined in Section 7.3 of this report.

The magnitudes of the environmental doses from Mayak airborne releases, although large in comparison with today's environmental standards, are unlikely to result in the need to estimate organ doses for specific workers in the cohort. Assuming continuous lifetime exposures at the rates described in Section 7.3, total effective doses range up to 60 mSv. For most workers, doses should be significantly less than these conservatively estimated screening doses. Most organ doses would also be low, except for workers exposed in the late 1940s through late 1950s, for whom thyroid doses approaching 300 mSv are possible. Epidemiological studies of effects on the thyroid should consider the environmental exposure to radioiodines. Efforts are underway in Project 1.4 to develop better estimates for these doses.

### 9.4.4 Medical X-Ray Dose

MPA workers received medical x-ray examinations routinely. Records of these examinations are available from the SUBI medical record archives. A scoping study of the potential dose was conducted as described in Section 7.4. The scoping study identified from the archives for about 8,500 workers, a total of 84,982 diagnostic x-ray procedures. The following results were noted:

- A range of from 1 to 145 procedures for a single worker.
- The average ESE to x-ray procedures of the chest contributes about 30% of the average recorded occupational exposure for workers.
- The average ESE from medical x-ray examinations to the lower body is about 2 times higher than the average recorded occupational exposure.
- The average medical ESE to 1,826 of the 5,341 workers who received x-ray examinations of the chest and fluorography is higher than the occupational exposure.
- The medical ESE to 1,068 of the 1,853 persons who received lower body x-ray examinations exceeded the recorded occupational exposure.
- The average annual ESE from x-ray examinations of the chest exceeds the average annual recorded occupational exposure (dose) since 1962.
- The average annual ESE from x-ray examinations of the lower body organs exceeds the average annual occupational exposure (dose) by an order of magnitude since 1960 and by 2 orders of magnitude since 1970.

Based on the foregoing, consideration of the medical x-ray dose is important to the Mayak Worker Study.

## 9.5 PHASE IV TASKS TO IMPROVE DOSES-2005

The Mayak Worker Dosimetry Study involves analysis of many complex issues, a relatively large number of workers, and extends over a multidecade period beginning in the late 1940s. It is to be expected that resolution has not been achieved for all significant technical issues with the completion of Phase III, Doses-2005. Table 9.1 summarizes the status of Doses-2005 and planned Phase IV tasks leading to a new version (i.e., Doses-2008). Notable remaining tasks identified at this time are summarized in the following sections.

### 9.5.1 Expand Cohort

Project 2.2 researchers have identified a need to include in the study cohort reconstructed dose estimates for several thousand auxiliary plant workers hired during the period of 1948 – 1982 to provide in-study, relatively low-dose controls and workers hired during 1972–1982 and employed at the primary reactor, radiochemical, and plutonium plants. The estimated total final study cohort would be about 26,000 workers.

### 9.5.2 Medical X-ray Organ Dose

The scoping study of medical x-ray doses described in Chapter 7 involving 84,982 x-ray examinations for 8,500 workers showed a substantial likelihood of significant doses to Mayak Workers from routine medical x-ray examinations. The study demonstrated that the medical x-ray doses to workers is highly variable and for some workers might be greater than the occupational dose. This could be particularly true for workers in the early years with lower occupational exposure and for essentially all workers hired after 1962. A closely related need concerns determining organ doses for individual workers from medical x-ray examinations. X-ray examinations involve partial body irradiation with photon energies that are typically lower than workplace photon irradiation. The analysis of organ doses could have a significant effect on the comparison with the organ doses from occupational radiation in Doses-2005. Approximately 50% of the available medical x-ray examination data for individual Mayak workers has been computerized. Resolution of this issue is considered a priority in the analysis of Doses-2005.

**Table 9.1**. Doses-2005 Status and Phase IV Tasks

| Exposure Pathway | Status of available archive data | Project 2.4 Work Accomplished with Doses-2005 | Notable remaining issues | Phase IV Tasks |
|---|---|---|---|---|
| External Beta Radiation | No data with Mayak IFK and IFK + Pb dosimeters used before 1962. There is nonpenetrating dose information with IFKU dosimeter beginning in 1962, but this has not been computerized. | Completed dosimeter response measurements to $^{90}Sr/^{90}Y$ beta radiation at GSF and LINAC beams (University of Utah). Beta radiation response by dosimeter can cause overestimate of deep dose. Doses-2005 reconstructed dose does include corrections to early (pre-1962) archive gamma dose for beta radiation based on work groups and exposure scenarios. | Primary focus of Mayak Study is organ dose from penetrating photon radiation. No attempt has been made to estimate nonpenetrating dose. | Use IFKU dosimeter to determine nonpenetrating dose in addition to current penetrating dose, applicable to skin, breast, testicular, and eye and to extrapolate this to earlier Mayak worker doses. |
| External Photon Radiation | Measured dose for 15,815 subjects | Completed dosimeter photon energy and angular response measurements at GSF. Modeling of Mayak dosimeter photon and angular response characteristics used with exposure scenarios to reconstruct dose in air from archive dose, and to reconstruct $H_p(10)$ based on worker orientation in workplace as specified in exposure scenario | Database of 725,350 records of individual dosimeter dose results for about 8,500 workers from 1948 - 1967 was used to examine trends in routine and nonroutine exposures but not to assess completeness of archive records and characteristics of measured dose with time and worker area/position. | Use these data, based on dosimeters worn by each worker at time of exposure, to examine completeness of exposure record for each worker. |
| | No archive dose for 3,016 subjects. | Reconstructed dosimeter measured dose was done using work history | | Use these data, reconstructed based on work history information, to examine trends in dose accumulation for each worker. |

| Exposure Pathway | Status of available archive data | Project 2.4 Work Accomplished with Doses-2005 | Notable remaining issues | Phase IV Tasks |
|---|---|---|---|---|
| | | information to provide estimate of dose as though dosimeter was worn. Dose reconstruction process for measured and unmeasured dose is identical thereafter. Reliability index is used in Doses-2005 Analysis file to indicate there was no measured dose. | | |
| Transuranic (Pu) nuclide intakes | Biophysics data for 6,785 workers and autopsy data for 449. Total of 7,234 subjects. | There have been significant analyses of available radioassay and autopsy data. These have been presented in several peer reviewed manuscripts. | An improved plutonium dose model was developed using substantial data available. | Further refine plutonium dose model and attempt to identify Mayak workers with unknown significant intakes. |
| External Neutron Radiation | No data before later 1970s. Occupational level monitoring begun in early 1980s. | Scoping study was conducted to assess significance of neutron radiation to organ dose. Doses-2005 Analysis file contains $H_p(10)$ for neutrons using preliminary and highly uncertain neutron to photon dose ratios used in Doses-2000. | Workplace measurements of neutron dose and spectral characteristics at Mayak plutonium storage facility were not done. | Assess neutron dose for those exposure scenarios expected to provide significant neutron dose. |
| Other than plutonium nuclide intake | No nuclide-specific air monitoring data and limited radioassay data available for analysis. | Scoping study conducted to evaluate potential magnitude of dose to organs from airborne concentrations of nuclides in workplace. | Radioassay data of long-lived fission products was used to examine potential dose. There is little information available to examine this pathway. | Pattern in potential nuclide intakes and organ doses needs to be examined. |

| Exposure Pathway | Status of available archive data | Project 2.4 Work Accomplished with Doses-2005 | Notable remaining issues | Phase IV Tasks |
|---|---|---|---|---|
| | | | | |
| Airborne Effluent | Record of airborne effluent by year | Scoping study done in collaboration with Project 1.4 researchers to evaluate potential magnitude of dose using a single release point and annual effluent data. | Restricted analysis using single release point. | Examine significance of dose for lower dosed workers used as in-study controls. |
| Medical x-rays | Extensive archive records | Scoping study conducted using sample of 84,982 diagnostic x-ray procedures for about 8,500 workers. About 50% of data computerized. | For some workers, there appears to be significant exposure to surface of body. Number of examinations in sample ranged from 1 to 145. | Complete computerization of all medical x-ray examination data and evaluate significance to organ doses. |

### 9.5.3 Completeness of Dose Records

There is potential missed dose from at least three types of situations in Doses-2005: (1) workers who have no recorded dose, (2) workers with gaps in annual dose record, and (3) workers who have daily doses but for whom there are gaps in recorded doses. During the early years, workers anticipated to receive less than 30% of the radiation safety limit were not required to be monitored. The "Daily" Doses database contains 725,350 records of individual dosimeter dose results measured during the early period of Mayak operations (from 1948 through 1967) for 8,748 workers in the Mayak archives. While these data were analyzed to distinguish between dose patterns in routine and nonroutine exposures, there was not time to examine dose patterns for individual workers. In addition, it is currently estimated there are 500 and 700 Mayak workers who have doses received at nuclear materials production plants in Tomsk and Krasnoyarsk, respectively, that are not included in Doses-2005. Further analysis of this information and other supporting occupational history records is needed to assess the significance of the missed dose in Doses-2005.

### 9.5.4 Patterns in Dose Accumulation

Patterns in dose accumulation to monitored workers using the "Daily" Doses database described in 9.5.3 should be examined. The daily dose records were obtained from Worker Personal Logs and contain the begin and end wear date for all assigned dosimeters. This effort addresses the identified Project 2.2 researcher need to examine the pattern in dose accumulation and dose rate dependence of health effects.

### 9.5.5 Neutron Radiation

The scoping study of neutron radiation doses described in Section 7.1 was based on workplace scenarios developed in the preparation of Doses-2000 and involved identifying neutron-to-photon dose ratios. Although these ratios might indicate the "average" contribution to Mayak worker neutron doses, the experience at DOE facilities has been historically high contributions of neutron dose for some workers particularly in plutonium facilities and for some specific workplaces at the radiochemical production facilities. The situation at Mayak in which workers performed maintenance on operating reactors (which was not done in the United States) and conducted plutonium separation and refinement similar to DOE workers, requires further evaluation to ensure there are not individual workers with comparatively high neutron doses relative to their photon dose. A related effort concerns the fact that neutron organ DCFs for low and high LET are not currently available from the ICRP. Rather a total (low and high LET) DCF for the organs is available. As such, Doses-2005 organ doses do not contain a neutron dose component. The $H_p(10)$ neutron dose is provided as determined from using the neutron-to-photon dose ratios.

### 9.5.6 Airborne Effluent Dose

The scoping study of airborne effluent radiation doses described in Chapter 7 was based on using the annual release totals and a single Mayak release point. There is a need to incorporate these doses into the individual worker doses of Doses-2005 estimates of organ doses for individual workers received occupationally on the site as well as to members of the public during their residence in Ozyorsk.

### 9.5.7 Beta Radiation Shallow Dose

Early Mayak personnel dosimeters provided a single measurement of photon and higher-energy (~> 1 MeV) beta dose. The IFKU improved personnel dosimeter implemented in 1962 did measure the respective beta and photon dose components. For the early dosimeters the higher energy beta radiation

induced a dosimeter response that resulted in an overestimate of the actual photon radiation dose. This effect was significant for some exposure scenarios and was considered in preparation of Doses-2005. However, worker exposure to the skin from lower-energy (~<1 MeV) beta radiation was not recorded before 1962. Assessment of skin dose from beta radiation to Mayak workers before 1962 was not completed at the conclusion of Phase III (Doses-2005).

### 9.5.8 Workplace Radiation Fields

Reactor, chemical processing, and plutonium chemical-metallurgical operations that caused significant exposure of MPA workers have not existed since the early 1950s. Measurement of the photon and neutron spectra in Mayak facilities that have some relevance to the historical facilities could not be done before the preparation of Doses-2005 because of delays in receiving procured spectroscopy equipment. Efforts were made to obtain information from the Tomsk nuclear reactor facilities, which are similar in design to the Mayak reactors, but this information was not available for consideration before the completion of Doses-2005. Validation of the spectra used in dose reconstruction is an uncompleted activity at the conclusion of Phase III tasks and should be considered under Phase IV tasks.

# References


Avramenko, MI, Averin, AN, Drozhko, EG,; Glagolenko YuV, Loboiko, BG, Mokrov, YuG, Romanov, GN, Kotov, ES, Filin, VP Accident of 1957 and East Urals Radioactive Trace. Radiat. Safety Problems (Mayak Production Association Scientific Journal) 3:18-28; 1997 (in Russian).

Briesmeister, JF, Ed., 2000, MCNP – A General Monte Carlo N-Particle Transport Code – Version 4C, LA-13709-M, Los Alamos National Laboratory, Los Alamos, NM.

Bournazyan, AI Ed. The results of studying and the experience of the liquidation of consequences of accidental contamination of territories by uranium fission products. Moscow: Ministry of Health of USSR; 1974 (in Russian).

Cristy, M, Eckerman KF. 1987. Specific Absorbed Fractions of Energy at Various Ages from Internal Photon Sources. ORNL/TM-8381. Oak Ridge National Laboratory, Oak Ridge, TN USA.

Degteva MO, Kozheurov VP, Tolstykh EI, Vorobiova MI, Anspaugh LR, Napier BA, Kovtun AN. The Techa River dosimetry system: methods for the reconstruction of internal dose. Health Phys. (1): 24-35; 2000.

Drozhko EG, Khokhryakov VV. Reconstruction of Ozyorsk population exposure doses due to iodine-131 atmospheric releases at spent nuclear fuel reprocessing. Ozyorsk: Mayak Production Association; Milestone Report Number 1; 2003.

Eckerman, KF, Leggett, RW, Nelson, CB, Puskin, JS, Richardson, ACB. Cancer Risk coefficients for environmental exposure to radionuclides. Federal Guidance Report No. 13, CD Supplement, U.S. Environmental Protection Agency Report EPA 402-R-99-001; 1999. (National Technical Information Service, Springfield, Virginia)

Farris, WT, Napier, BA, Ikenberry, TA, Simpson, JC, and Shipler, DB. Atmospheric pathway dosimetry report, 1944-1992. Richland: Battelle Pacific Northwest Laboratories, Hanford Environmental Dose Reconstruction Project, PNWD-2228 HEDR; 1994.

Gilbert, ES, Koshurnikova, NA, Sokolnikov, M, Khokhryakov, VF, Miller, S, Preston, DL, Romanov, SA, Shilnikova, NS, Suslova, KG, Vostrotin, VV. Liver cancers in Mayak workers. Radia. Res. 154, 3, 246–252; 2000.

Gilbert, ES, Koshurnikova, NA, Sokolnikov, ME, Shilnikova, NS, Preston, DL, Ron, E, Okatenko, PV, Khokhryakov, VF, Vasilenko, EK, Miller, S, Eckerman, K, Romanov, SA. Lung cancer in Mayak workers. Radia. Res. 162: 505-516; 2004.

International Commission on Radiological Protection. Limits for inhalation of radon daughters by workers. Oxford: Pergamon Press, ICRP Publication 32; 1981.

International Commission on Radiological Protection. Protection of the Patient in Diagnostic Radiology. Oxford: Pergamon Press, ICRP Publication 34; 1982.

International Commission on Radiological Protection. 1996. Conversion coefficients for use in radiological protection against external radiation. Oxford: Pergamon Press, ICRP Publication 74; 1996.

Izrael, YuA, Artemov, EM, Vasilenko, VN, Nazarov, IN, Nakhutin, AI, Uspin, AA, Kyamkin, AM. Radioactive contamination of Urals region by Mayak Production Association. In: Radioactivity under nuclear explosions and accidents. Proceedings of International Conference, Moscow, 24-26 April, 2000. St. Petersburg: Gidrometeoizdat; 2000: 411-424.

Joint Norwegian-Russian Expert Group for Investigation of Radioactive Contamination in the Northern Areas. Sources contributing to radioactive contamination of the Techa River and areas surrounding the "Mayak" Production Association, Urals, Russia. Østerås: Norwegian Radiation Protection Authority; 1997.

Khokhryakov, VV, Drozhko, EG, Romanov, GN, Mokrov, YuG, Kyamkin, AM, Volobuev, PV, Vorobiova, MI, Kostychenko VA, Akleyev, AV. Characteristics of anthropogenic radiation impact on territory and population. In: Consequences of an anthropogenic radiation incident and rehabilitation problems of the Urals Region. Moscow: Comtehprint Publisher; 2002: 8-77 (in Russian).

Khokhryakov, V, Suslova, K, Aladova, E, Vasilenko, E, Miller, SC, Slaughter, DM, Krahenbuhl, MP. Development of an Improved Dosimetry System for the workers at the Mayak Production Association. Health Phys, 79(1):72-76; 2000.

Khokhryakov VF, Suslova KG, Vostrotin VV, Romanov SA, Eckerman KF, Krahenbuhl MP, Miller SC. Adaptation of the ICRP Publication 66 respiratory tract model to data on plutonium biokinetics for Mayak workers. Health Phys. 88:125-132; 2005.

Korsakov, YuD, Fedorov, EI, Romanov, GN, Panteleev, LI, Peremyslova, LM, Dobryakova, GV, Melnikov, YuI, Ivin, IS. Assessments of the radiation situation on the territory contaminated as a result of windblown of radioactive aerosols. Chelyabinsk: Archive of URCRM #532; 1968 (in Russian).

Koshurnikova, NA, Gilbert, ES, Sokolnikov, M, Khokhryakov, VF, Miller, S, Preston, DL, Romanov, SA, Shilnikova, NS, Suslova, KG, Vostrotin, VV. Bone cancers in Mayak workers. Radia. Res., 154, 3, 237–245; 2000.

Koshurnikova, NA, Buldakov, LA, Bysogolov, GD, Bolotnikova, MG, Komleva, NS, Peternikova, VS. Mortality from malignancies of the hematopoietic and lymphatic tissues among personnel of the first nuclear plant in the USSR. The Science of the Total Environment, 142: 19-23; 1994.

Koshurnikova, NA, Gilbert, ES, Shilnikova, NS, Sokolnikov, M, Preston, DL, Kreisheimer, M, Ron, E, Okatenko, P, and Romanov, SA. Studies on the Mayak nuclear workers: Health effects. Radiat Environ Biophys 41(1):29-31; 2002.

Koshurnikova, NA, Shilnikova, NS, Okatenko, PV, Kreslov, VV, Bolotnikova, MG, Sokolnikov, ME, Khokhriakov, VF, Suslova, KG, Vasilenko, EK, Romanov, SA. Characteristics of the cohort of workers at the Mayak Nuclear Complex. Radia. Res. 152, 352-363; 1999.

Krahenbuhl, MP, Bess, JD, Wilde, JL, Vostrotin, VV, Suslova, KG, Khokhryakov, VF, Slaughter, DM, Miller, SC. Uncertainties analysis of doses resulting from chronic inhalation of plutonium at the Mayak Production Association. Health Physics 89:33-45, 2005.

Leggett, RW, Eckerman, KF, Khokhryakov, VF, Suslova, KG, Krahenbuhl, MP, Miller, SC. Mayak worker study: An improved biokinetic model for reconstructing doses from internally deposited plutonium. Radiat. Res. 164:111-122; 2005.

Leggett RW. Reliability of the ICRP's dose coefficients for members of the public. III. Plutonium as a case study of uncertainties in the systemic biokinetics of radionuclides. Radiation Protection Dosimetry 106:103-120, 2003.

Lyarsky, PP. Sanitary consequences of territory contamination by long-lived uranium fission products and organization of sanitary-preventive actions. Moscow: Institute of Biophysics; Doctoral Thesis; 1962 (in Russian).

Mokrov Y, Glagolenko Y, Napier B. Reconstruction of radionuclide contamination of the Techa River caused by liquid waste discharge from radiochemical production at the Mayak Production Association. Health Phys. 79 (1):15-23; 2000.

Mokrov, YG, Anspaugh, LR, Napier, BA. Reconstruction of Dose to the Residents of Ozyorsk from the Operation of the Mayak Production Association: 1948–2002. Final Report on the Feasibility Study for Project 1.4. Mayak Production Association, University of Utah, and Pacific Northwest National Laboratory; 2004.

Muzrukov, VA. Assessment of the possibility to use the data obtained from weather stations located in the nearest vicinity to "Mayak" PA in atmospheric dispersion calculations. Ozyorsk: Mayak Production Association; 2005.

Napier BA, Strenge, DL, Ramsdell, Jr, JV, Eslinger, PW, Fosmire, CJ. GENII Version 2 software design document. Richland: Pacific Northwest National Laboratory, PNNL-14584; 2005.

NAS [National Academy of Sciences, Committee on Biological Effects of Ionizing Radiation (BEIR)]. Health risks from exposure to low levels of ionizing radiation. Washington: BEIR VII; 2006.

Peremyslova, LM, Akleyev, AV, Kostyuchenko, VA, Degteva, MO, Tolstykh, EI. Dietary intakes and internal exposure doses received by residents of the Karachay Trace. In: Proceedings of 11th International Congress of International Radiation Protection Association. Madrid: International Radiation Protection Association. Paper ID 325; 2004.

Romanov SA, Vasilenko EK, Khokhryakov VF, Jacob P. Studies on the Mayak nuclear workers: dosimetry. Radiat Environ Biophys: 41(1):23-8; 2002.

Romanov, GN, Shejn, GP, Aksenov, GM. Exposure doses for the population living on East-Urals radioactive trace: current estimates. Radiat. Safety Problems (Mayak Production Association Scientific Journal) 4:52–67; 1997 (in Russian).

Rovny, SI, Mokrov, YuG. Analysis of the work performed previously and development of proposals for future studies (report on Stage 2). Ozyorsk: Mayak Production Association; 2003.

Suslova, KG, Khokhryakov, VF, Romanov, SA. Map of Plutonium Body Burdens in Chelyabinsk Region Residents based on Radiochemical Analysis of Tissue Samples. In: 1st International Symposium on Chronic Radiation Exposure: Risk of Late Effects. Chelyabinsk; 1995.

Ternovsky, IA, Romanov, GN, Fedorov, EA, Shvedov, VL, Skryabin, AM, Malkin, PM, Melnikov, YuI. Radiation protection of the exposed population. In Study of radioecological, radiation-hygienic and social-economical consequences of massive radioactive contamination of large territories (1958-1984). Technical report MIRAGE, Volume 5. Chelyabinsk: Archive URCRM #117; 1985 (in Russian).

Ternovsky, IA, Romanov, GN, Sokhina, LP, Fedorov, EA, Teverovsky, EN, Korsakov, YuD, Belova, EI, Voronov, AS, Kozhevnikiva, TL, Martyushov, VZ, Mishenkov, NN, Smirnov, EG, Filatova, EV, Volkov, FS, Melnikov, YuI, Antropova, ZG, Baturin, VA, Lebedev, VM, Pitkyanen, GB, Safronova, NG, Ryabcev, IA. Exposure situation and dynamics of radionuclide behaviour in the environment. Ozyorsk: Technical report MIRAGE, Volume 3. Archive of the Mayak Production Association; 1985 (in Russian).

Teverovsky, EN, Ilyin, DI, Volkov, AS, Kiselyov, YaP, Syomova, RV, Rovinsky FYa, et al. Investigation of radioactive contamination of the atmosphere and the region nearby industrial complex 814. Mayak CPL Technical Report 3814. Ozyorsk: Archive of the Mayak Production Association #3814; 1957; (in Russian).

Zankl, M, Petoussi-Henß, N, Drexler, G, Saito, K. The calculation of dose from external photon exposures using reference human phantoms and Monte Carlo methods; Part VII: Organ doses due to parallel and environmental exposure geometries. GSF-Bericht; 1997.

# Glossary

**absorbed dose in air –** $D_{\gamma\ rec}$, the dose in air at the worker location, considering exposure scenario conditions, derived from $D_{\gamma\ dos}$ by multiplying by a scenario-specific conversion factor.

**archive dose –** $D_{\gamma\ arch}$, the dose included in the individual worker files.

**adjusted photon absorbed dose in air –** $D_{\gamma\ dos}$, the dose derived from $D_{\gamma\ arch}$ by partitioning into routine and nonroutine components, adjusting for betas that might have been encountered in nonroutine exposures, and converting to mGy.

**beta (β) dose –** A designation (i.e., beta) on some Pantex external dose records referring to the dose from less-energetic beta, x-ray, or gamma radiation.

**beta radiation –** Radiation consisting of charged particles of very small mass (i.e., the electron) emitted spontaneously from the nuclei of certain radioactive elements. Physically, the beta particle is identical to an electron moving at high velocity.

**curie –** A special unit of activity. One curie (1 Ci) exactly equals $3.7 \times 10^{10}$ nuclear transitions per second.

**deep absorbed dose ($D_d$) –** The absorbed dose at the depth of 1.0 cm in a material of specified geometry and composition.

**deep dose equivalent ($H_d$) –** The dose equivalent at the depth of 1.0 cm in tissue.

**detection limit (lower) –** The minimum quantifiable exposure or neutron flux that can be detected.

**dose equivalent (H) –** The product of the absorbed dose (D), the quality factor (Q), and any other modifying factors. The special unit is the rem. When D is expressed in Gy, H is in sieverts (Sv). (1 Sv = 100 rem).

**dose of record –** The dose included in the individual worker files (see **archive dose**).

**dosimeter –** A device used to measure the quantity of radiation received. A holder with radiation-absorbing elements (filters) and an insert with radiation-sensitive elements packaged to provide a record of absorbed dose or dose equivalent received by an individual. (See **film dosimeter**, **thermoluminescent dosimeter**).

**dosimetry –** The science of assessing absorbed dose, dose equivalent, effective dose equivalent, etc., from external or internal sources of radiation.

**dosimetry system –** A system used to assess dose equivalent from external radiation to the whole body, skin, and extremities. This includes the fabrication, assignment, and processing of dosimeters as well as interpretation and documentation of the results.

**exchange period (frequency) –** Period (weekly, biweekly, monthly, quarterly, etc.) for routine exchange of dosimeters.

***exposure*** **–** As used in the technical sense, a measure expressed in roentgens (R) of the ionization produced by photons (i.e., gamma and x rays) in air.

**extremity –** That portion of the arm extending from and including the elbow through the fingertips, and that portion of the leg extending from and including the knee and patella through the tips of the toes.

**field calibration –** Dosimeter calibration based on radiation types, intensity, and energies present in the work environment.

**film –** Generally means a "film packet" that contains one or more pieces of film in a light-tight wrapping. The film when developed has an image caused by radiation that can be measured using an optical densitometer.

**film density –** See **optical density**.

**film dosimeter –** A small packet of film in a holder that attaches to a wearer.

**gamma rays (γ) –** Electromagnetic radiation (photons) originating in atomic nuclei and accompanying many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Physically, gamma rays are identical to x-rays but with higher energy; the only essential difference is that x-rays do not originate in the nucleus.

**Gray –** SI unit of absorbed dose. Unit symbol, Gy. 1 Gy = 100 rad.

**ionizing radiation –** Electromagnetic or particulate radiation capable of producing charged particles through interactions with matter.

**Minimum Detectable Level (MDL) –** A term used in this document to refer to a statistically determined minimum detection level.

**Minimum Reportable Dose (MRD) –** A general term used to identify the minimum dose recorded and reported, normally based on site-specific policy.

**neutron –** A basic particle that is electrically neutral weighing nearly the same as the hydrogen atom.

**neutron, fast –** Neutrons with energy equal or greater than 10 keV.

**neutron, intermediate –** Neutrons with energy between 0.4 eV and 10 keV.

**neutron, thermal –** Strictly, neutrons in thermal equilibrium with surroundings. Generally, neutrons with energy less than the cadmium cutoff of about 0.4 eV.

**open window –** Designation on film dosimeter reports that implies the use of little shielding. It commonly is used to label the film response corresponding to the open-window area.

**optical density –** The quantitative measurement of photographic blackening with the density defined as $D = \mathrm{Log}_{10} (I_o/I)$.

**organ dose $D_{\gamma\ org\text{-}i}$ –** The absorbed dose to an organ from external gamma radiation.

**personal dose equivalent $H_p(d)$ –** Represents the dose equivalent in soft tissue below a specified point on the body at an appropriate depth d. The depths selected for personnel dosimetry are 0.07 mm and 10 mm, respectively, for the skin and body. These are noted as $H_p(0.07)$ and Hp(10), respectively.

**personal dose equivalent for gammas – $H_p(10)_\gamma$**, the personal dose equivalent from gammas.

**personal dose equivalent for neutrons – $H_p(10)_n$**, the personal dose equivalent from neutrons.

**photon –** A unit or "particle" of electromagnetic radiation consisting of x- or gamma rays.

**photon – x ray –** Electromagnetic radiation of energies between 10 keV and 100 keV whose source can be an x-ray machine or radioisotope.

**quality factor, Q –** A modifying factor used to derive dose equivalent from absorbed dose.

**radiation –** Alpha, beta, neutron, and photon radiation with sufficient energy to ionize atoms. See **ionizing radiation**.

**radioactivity –** The spontaneous emission of radiation, generally alpha or beta particles, gamma rays, and neutrons from unstable nuclei.

**rem –** A special unit of dose equivalent, which is equal to the product of the number of rad absorbed and the quality factor.

**roentgen (R) –** A unit of *exposure* to gamma (or x-ray) radiation. It is defined precisely as the quantity of gamma (or x) rays that will produce a total charge of $2.58 \times 10^{-4}$ coulomb in 1 kg of dry air. An *exposure* of 1 R is approximately equivalent to an absorbed dose of 1 rad in soft tissue for higher (>100 keV) energy photons.

**shallow absorbed dose ($D_s$) –** The absorbed dose at a depth of 0.007 cm in a material of specified geometry and composition.

**shallow dose equivalent ($H_s$) –** Dose equivalent at a depth of 0.007 cm in tissue.

**shielding –** Any material or obstruction that absorbs (or attenuates) radiation and thus tends to protect personnel or materials from radiation.

**skin dose –** Absorbed dose at a tissue depth of 7 mg $cm^{-2}$.

**thermoluminescent –** Property of a material that causes it to emit light as a result of being excited by heat.

**thermoluminescent dosimeter (TLD) –** A holder containing solid chips of material that when heated will release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

**whole-body dose –** Commonly defined as the absorbed dose at a tissue depth of 1.0 cm (1000 mg $cm^{-2}$); however, this term is also used to refer to the recorded dose.

**x ray –** Ionizing electromagnetic radiation that originates external to the nucleus of an atom.