Appendix G

Radiation Protection Considerations for Nuclear Power Facility Decommissioning

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Radiological issues are associated with the process of decommissioning nuclear reactor facilities, including power reactors, at the end of their operating lives. Both occupational workers and members of the public will be affected by these processes as a result of direct exposures to sources of radiation and as a result of small releases of radioactive materials in gaseous and liquid effluents. This appendix is intended to provide pertinent background information for analyses in this Generic Environmental Impact Statement Supplement.

G.1 Radiation Protection Standards

The primary U.S. Nuclear Regulatory Commission (NRC) standards for protection of workers and members of the public are found in 10 CFR Part 20. These standards are consistent with guidance to Federal agencies prepared by interagency committees and issued by the President. The Federal guidance is based on recommendations published by national and international organizations, such as the National Council on Radiation Protection and Measurements (NCRP), the International Commission on Radiological Protection (ICRP), and the United Nations Scientific Committee on the Effects of Atomic Radiation. Proposed changes to regulations are typically published in the Federal Register for public comment before enactment of the final rule. The most recent major revision to the NRC radiation protection regulations in 10 CFR Part 20 were enacted in 1991, with several amendments issued in the intervening years. Implementation of the regulations became mandatory for NRC licensees in 1994.

G.1.1 Concepts, Terminology, Quantities, and Units Used in Radiation Protection

Title 10 CFR Part 20 was first promulgated in 1957. In 1961, the regulation was amended to add an appendix containing maximum permissible concentrations and a new occupational dose limit structure for whole-body exposure to external radiation (1.25 rem/quarter, or 3 rem/quarter with 5 rem/yr average as a limit on the cumulative dose). The 1991 revision differs considerably from the previous regulations with respect to basic concepts, terminology, radiation dose quantities, and the associated dose units. This section is included to familiarize readers with these concepts.

G.1.1.1 Conventional Quantities and Units

In 10 CFR Part 20, the unit "rad" is usually used for the quantity "radiation absorbed dose" whenever early biological effects are the concern. When latent effects (e.g., cancer and genetic effects) are being considered, the unit "rem" is used for the dose equivalent (DE) quantity. The absorbed dose in rads is multiplied by an overall efficiency factor *Q* to obtain the DE in rem. Each type of radiation has its own value of *Q*, which in a very general way permits adding absorbed doses from different radiations to estimate the probability of stochastic effects. Values of Q in 10 CFR Part 20 are indicated in Table G-1.

These values of *Q* reflect the overall efficiency of a given type of radiation in causing latent effects and are not used for early effects such as acute radiation syndrome. The values were derived in consideration of the ability of the various radiations to ionize atoms in water as well as the relative biological effectiveness factors observed for specific effects.

Table G-1. Quality Factors and Absorbed Equivalents

Radiation	Absorbed Dose, rad	Q	Dose Equivalent, rem
x -, gamma or beta radiation	1	1	1
Alpha particles	1	20	20
Neutron (spectrum unknown)	1	10	10

Note: To convert rem to sievert, multiply by 0.01.

G.1.1.2 International System of Units

The International System (SI) units of particular interest in radiation protection are the gray (Gy), sievert (Sv), and becquerel (Bq), as shown in Table G-2. The SI units are part of the metric system; however, they are not yet widely used in the United States.
Title 10 CFR 20.2101 requires the records to be reported in the units of curie, rad, and rem. The major concern of the NRC staff is that use of both the conventional and SI units would introduce confusion under emergency conditions.

Table G-2. Conventional and SI Units

Quantity	Conventional Unit	SI Unit	SI Unit Conversions
Absorbed dose	rad (100 ergs/gram)	gray (Gy) (10,000 ergs/gram)	100 rad = 1 Gy
Dose equivalent	rem (Q x rad)	sievert (Sv) (Q x gray)	100 rem = 1 Sv
Activity	curie (Ci) (3.7 x 10^{10} disintegrations per second)	becquerel (Bq) (1 disintegration per second)	1 Ci = 3.7 x $10^{(10)}$ Bq

G.1.1.3 Collective Dose

Previous revisions of 10 CFR Part 20 made no use of the collective DE (in person-rem). However, this quantity is used by the NRC in risk analyses and in its decision-making processes. The collective DE may be obtained as the sum of all individual doses or as the product of the average individual dose and the number of people exposed. The linear-nonthreshold hypothesis is accepted by the NRC for purposes of standards setting. Such acceptance means that standards based on the hypothesis, coupled with the “as low as reasonably achievable” (ALARA) concept, are believed to provide an adequate degree of protection.

G.1.1.4 Risks from Radiation Exposure

The current regulations in 10 CFR Part 20 are based on concepts first developed by the ICRP in Publication 26 (ICRP 1977). The ICRP system is based on the recognition of two basic types of radiation-induced health effects: stochastic and nonstochastic. Stochastic effects, such as cancer and hereditary effects, are considered to be probabilistic in nature. For stochastic effects, the probability of the effect, but not the severity, is dose-dependent (i.e., once a malignancy occurs). Its severity is no different if the dose that preceded it were 1 Sv (100 rem), 0.1 Sv (10 rem), or zero. The objective of radiation protection policies is to control the probability of these effects to acceptable levels. In contrast, the severity of nonstochastic effects, but not the probability of occurrence, depends on the radiation dose. Examples of radiation-induced nonstochastic effects include cataracts in the lens of the eye or burns on the skin surface. Nonstochastic effects typically do not occur unless the dose exceeds a threshold, which is specific to each type of effect. Once the threshold dose is exceeded, the effect occurs, and the severity of the effect depends on the dose received by the affected tissue or organ. For example, a radiation-induced cataract caused by a 4-Sv (400-rem) dose to the lens of the

eye would impair vision to a greater extent than one following a dose of 1 Sv (100 rem). Therefore, radiation protection for nonstochastic effects is designed to keep radiological exposures to sensitive tissues below the threshold levels at which the effects would begin to appear.

In January 1990, the National Research Council (NAS 1990) published a report on the health effects of exposure to low levels of ionizing radiation. This report was prepared by the Committee on Biological Effects of Ionizing Radiation (BEIR) known as the BEIR-V Committee, organized by the Council for this purpose. The BEIR-V report concluded that the risk of radiation exposure was greater than estimates published by previous committees (NAS 1972, NAS 1980). In light of this data, the ICRP requested comment from a number of organizations on a draft of its revised recommendations on radiation protection. In 1991, the ICRP issued Publication 60 (ICRP 1991) recommending lower limits for occupational exposures. With regard to this Supplement, the primary importance of these developments lies in the selection of the most appropriate radiation risk coefficients to use for evaluating health effects. For a more complete history of the development of radiological risk estimates, see NRC (1996), Appendix E.

G.1.1.4.1 Stochastic Effects

Stochastic effects refer to health effects, such as cancer and inheritable genetic effects, for which the probability of occurrence is related to radiation dose. Based on the BEIR-V study (1990), the risks were estimated as 4 to 5 excess cancer deaths among 10,000 people receiving 100 person-Sv (10,000 person-rem). The following statement appears in the executive summary of the BEIR-V report (NAS 1990, p. 6):

> On the basis of the available evidence, the population-weighted average lifetime excess risk of death from cancer following an acute dose equivalent to all body organs of 0.1 Sv [0.1 Gy of low-linear energy transfer (LET) radiation] is estimated to be 0.8 percent, although the lifetime risk varies considerably with age at the time of exposure. For low-LET radiation, accumulation of the same dose over weeks or months, however, is expected to reduce the lifetime risk appreciably, possibly by a factor of 2 or more.

The 0.8-percent estimate is equivalent to 800 excess cancer fatalities among 100,000 people, each exposed to 0.1 Sv (10 rem). It is important to note that the risk values tabulated in the report are for a population size of 100,000 and that the 0.8-percent estimate is applicable to instantaneous, uniform irradiation of all organs. With regard to the lower extreme of the dose range over which the estimate is applicable, the Committee observes elsewhere in the BEIR-V report that “in general, the estimates of risk derived in this way for doses of less than 0.1 Gy (10 rem) are too small to be detectable by direct observation in epidemiological studies.” The report does not provide a risk estimate for instantaneous doses of fewer than 0.1 Sv (10 rem). The Committee’s estimate is considered useful for estimating fatalities among large popula-

tions, including all ages, that are irradiated instantaneously and uniformly to individual external radiation doses of 0.1 Sv (10 rem) or more. Risk assessments based on the Japanese experience are subject to substantially greater uncertainty when applied to conditions typically encountered in exposures from normal facility operations, where

- exposures are protracted
- the exposed population is small
- individual doses are much lower than 0.1Sv (10 rem)
- irradiation is caused by internally deposited radionuclides and is not uniform throughout the body
- the exposed population differs significantly from the atomic bomb survivor study group
- some combination of these conditions exists or
- any of an almost infinite list of unknowns applies.

For stochastic effects, the ICRP adopted the risk associated with 0.05 Sv (5 rem) in a year, delivered to every organ, as the basis for its dose-limitation system (ICRP 1977). Therefore, the stochastic annual limit on intake (ALI) for each radionuclide is the quantity that, if inhaled, would cause the same stochastic risk as a uniform, whole-body dose of 0.05 Sv (5 rem) delivered by external sources in 1 year. To establish these ALIs, the ICRP considered the possibility that a given radionuclide taken into the body eventually reaches the bloodstream and is then distributed selectively to the various organs and tissues, where DE is delivered over a time course determined by the retention capabilities of the organ or tissue and the physical characteristics of the radionuclide. Using a radiation risk coefficient specific for each organ or tissue and the 50-year integrated dose equivalent to the tissue, the risk associated with each is estimated. The total risk to the worker per quantity of this radionuclide inhaled is the sum of the individual organ or tissue risks. The intake that will produce the same overall stochastic risk as 0.05 Sv/yr (5 rem/yr) of uniform external radiation can then be readily calculated as the ALI. Of course, a worker may be exposed to several airborne radionuclides and to external radiation as well. In that case, the total risk is still limited to that associated with 0.05 Sv (5 rem) in a year from uniform external radiation. Compliance is achieved if the fraction of the external dose limit that is received, added to the fraction of ALI inhaled for each radionuclide, does not exceed unity.

The risk of hereditary effects is included in a special way that, in the view of the ICRP, renders it additive to the cancer fatality risk. The ICRP considered only detrimental effects that the worker is likely to experience personally, so that effects manifested after the second generation are not included in the genetic risk coefficient used. The coefficient is also limited to very serious genetic effects (i.e., those comparable in severity to premature death).

Although all organs and tissues receive the same DE under uniform exposure conditions, the cancer risks for a given dose in each organ are not the same. Each organ or tissue contributes to the overall risk based on the relative sensitivity of tissue to radiation-induced cancer. This fraction is called the weighting factor, and the sum of the weighting factors for all tissues is

unity. The product of the weighting factor and the DE is the effective dose equivalent (EDE). This quantity is used for both external and internal irradiation and may be used for individual organs and tissues or for the sum of all organs and tissues. The unit used for either quantity is the same as for the DE, namely, the sievert (or rem). In the unique case of uniform irradiation of all organs and tissues, the sum of their EDEs is by definition equal to the whole-body DE. The EDE may be determined irrespective of the degree of uniformity among the organ or tissue doses. The sum of the EDEs is not allowed to exceed 0.05 Sv/yr (5 rem/yr).

The committed dose equivalent (CDE) is a quantity defined as the 50-year integrated DE to a specific organ or tissue following the inhalation of a radionuclide. This quantity is still used, but only in connection with nonstochastic effects. The committed effective dose equivalent (CEDE) is the same quantity as the CDE, with the exception that, in the case of the CEDE, each dose equivalent is multiplied by the tissue or organ weighting factor. The rem (or sievert) is also the unit for both of these quantities.

The mathematical weighting method used by the ICRP is shown in Table G-3. The first column lists the organs, and the second column lists the risk coefficients from ICRP Publication 26 (1977) and their sum, namely, 1.65×10^{-4}. This sum is the total annual risk to the exposed person, assuming exposure to these organs at 0.01 Gy/yr (1 rad/yr).[(a)] The fraction of this risk per rad for each organ can be obtained by dividing its risk coefficient by 1.65×10^{-4}. These fractions represent the relative sensitivity of the organs; they are the weighting factors and are designated by the symbol w_T, where T represents the organ or tissue. The weighting factors appear in column three of the table. If T is the dose equivalent to tissue T, then $w_T H_T$ is the weighted DE. For example, w_T for the lung is 0.12. If a weighted lung dose of H rem is set equal to a highly penetrating, uniform whole-body dose of 5 rem, then

$$0.12\,H = 0.05 \text{ Sv (5 rem) and}$$
$$H = 4.17 \text{ Sv (41.7 rem).}$$

By hypothesis and analogy, an annual DE of 0.417 Sv (41.7 rem) to only the lung would have the same effect as 0.05 Sv (5 rem) to all of the organs combined. For this reason, $w_T H_T$ is called the EDE.

(a) Multiplication by 5 gives the annual risk at 0.05 Gy/yr (5 rad/yr) (i.e., 8.25×10^{-4}/yr). This risk value means that if groups of 10,000 workers were to receive the dose limit every year for their entire careers, data as of the mid-1970s indicate that an average of 8.25 fatal occupational radiation-induced cancers per year would occur within each group. Assuming the approximate worst case of 45 years of exposure, the toll theoretically would be about 370 deaths per group, or almost 4 percent.

Nonstochastic effects have thresholds, and they become more severe as the dose gets larger. The ICRP believes that none of the thresholds will be exceeded if the annual dose to any tissue or organ does not exceed 0.5 Gy (50 rad). This nonstochastic limit is reflected in Table G-3,

Table G-3. ICRP Publication 26 Risk Weighting System

Organs	Risk Coefficients, Effects per Organ-rem	Weighting Factors	Organ DE Causing Same Risk as 5 rem to Whole Body, rem	Annual DE Permitted, Exposure of One Organ, rem/yr
Gonads	4×10^{-5}	0.25	20	20
Breasts	2.5×10^{-5}	0.15	33-1/3	33-1/3
Lung	2×10^{-5}	0.12	41-2/3	41-2/3
Red marrow	2×10^{-5}	0.12	41-2/3	41-2/3
Bone	5×10^{-6}	0.03	166-2/3	50
Thyroid	5×10^{-6}	0.03	166-2/3	50
1st RO[(a)]	1×10^{-5}	0.06	83-1/3	50
2nd RO	1×10^{-5}	0.06	83-1/3	50
3rd RO	1×10^{-5}	0.06	83-1/3	50
4th RO	1×10^{-5}	0.06	83-1/3	50
5th RO	1×10^{-5}	0.06	83-1/3	50
Totals	1.65×10^{-4}	1.0		

(a) The remainder organs (ROs) are the five organs that receive, from a given radionuclide, the highest EDE, integrated over 50 years.
Note: To convert rem to sievert, multiply by 0.01.

where it is evident that nonstochastic effects are controlling for all but four organs that have the largest weighting factors, the most sensitive organs with respect to stochastic effects.

G.1.1.4.2 Nonstochastic Effects

Nonstochastic effects refer to those, such as radiation-induced cataracts, for which the severity of the effect depends on radiation dose. They typically are not observed unless the radiation dose exceeds a minimum threshold, whereas the probability of stochastic effects is assumed to be greater than zero, although very small, even at very low doses. Therefore, radiological protection for nonstochastic effects is based on limiting exposures to levels that prevent the effect, rather than on controlling the probability of occurrence, as discussed previously for stochastic effects. For tissues such as the lens of the eye, the skin, and the extremities,

radiation protection standards are intended primarily to control the dose from external sources. For internal organs, it is necessary to control the dose from internally deposited radionuclides as well. Because radiation can damage or kill cells if the dose is sufficiently high, a nonstochastic dose limit must be established for all tissues, including tissues other than those mentioned above.

ICRP Publication 41 (1983) provides the technical justification supporting the position that, with the exception of the lens of the eye, nonstochastic effects will not be observed among adults if the DE from external and internal radiation combined to every organ and tissue is less than 0.5 Sv/yr (50 rem/yr). The NRC is not aware of later radiobiological information indicating that this dose limit should be changed and notes that the ICRP retained this value in the 1990 revision of its recommendations (ICRP 1991).

G.1.1.4.3 Risk Coefficient Selection for This Supplement

The BEIR-V risk estimate can be arithmetically converted to the more familiar terminology of 8 cancer fatalities among 10,000 people exposed to 10 person-Sv (10,000 person-rem), leading to a convenient risk coefficient of 8×10^{-4} fatalities per person-rem. This coefficient is considered useful for estimating fatalities among large populations irradiated instantaneously and uniformly to individual external radiation doses of 0.1 Sv (10 rem) or more. However, since no dose or dose rate effectiveness factor (DDREF) is included in this risk factor, the fatality estimates become speculative as the individual doses and the size of the exposed population become progressively smaller. A DDREF of 2 has been recommended by the ICRP (1991) for doses below 0.2 Gy (20 rad) and dose rates below 0.1 Gy/h (10 rad/h), which corresponds to a risk coefficient 4.0×10^{-4} fatalities per person-rem.

The risk coefficients used in this Supplement are listed in Table G-4. These coefficients are consistent with the risk factors reported in BEIR-V if a DDREF of 2 is applied. The somewhat higher risk coefficients for the general population as compared to workers reflects the fact that individuals under age 18 at the time of exposure are more susceptible to radiation-induced cancer. A person must be 18 years or older to be employed as a radiological worker. Excess hereditary effects are listed separately because radiation-induced effects of this type have not been observed in any human population, as opposed to excess malignancies that have been identified among people receiving instantaneous and near-uniform exposures of 0.1 Sv (10 rem) or more. As applied to low-level environmental and occupational exposures, risk factors for radiological health effects are subject to substantial uncertainty. The lower limit of the range for these risk coefficients is assumed to be zero because there may be biological mechanisms that can repair damage caused by radiation at low doses and/or dose rates.

G.1.2 Occupational Protection Standards

Occupational radiation protection standards have been in effect since 1947, and have generally been revised downward over the years, from 1.0 roentgen/wk (or about 50 roentgen/yr) in 1947

to the current 0.05 Sv/yr (5 rem/yr) total effective dose equivalent (TEDE). For an historical overview of development of these regulations, see NRC (1996), Appendix E. The current

Table G-4. Nominal Probability Coefficients Used in this Supplement[(a)]

Health Effect	Occupational	Public
Fatal cancer	4	5
Hereditary	0.6	1

(a) Estimated number of excess effects among 10,000 people receiving 100 person-Sv (10,000 person-rem).
Source: ICRP Publication 60 (1991).

regulation implements the concept of TEDE, as developed by ICRP Publication 26 (1977). This methodology accounts for both exposure to radiation from external sources and intakes of radionuclides into the body in assessing compliance with the standards. Standards that were previously in effect applied only to external dose and did not account for dose from intakes of radionuclides by workers, which were assessed separately. In practice, radionuclide intakes account for a small fraction of the total dose received by workers at nuclear power facilities.

Historical dose data for nuclear power plant workers are presented in Section G.2. Table G-5 presents a summary of the occupational standards in the 1991 revision of 10 CFR Part 20. On an annual basis, the whole-body limit has decreased from 12 roentgen (3 roentgen per quarter) in 1957 (external radiation only) to 0.05-Sv (5-rem) TEDE (external plus internal).

Regulatory control over the intake of radioactive materials in the workplace has always been a complex issue. Beginning in 1991, the NRC adopted the method published by the ICRP in Publication 26 (ICRP 1977). Under the ICRP method, the dose to each significantly irradiated organ is weighted according to its radiation sensitivity. The weighted doses are summed to produce an EDE that can be added to the dose from external sources.

The revised 10 CFR Part 20 provides additional flexibility for establishing more accurate dose controls. It allows the use of actual particle-size distribution and physiochemical characteristics of airborne particulates to define site-specific derived air concentration limits. With NRC approval, these modified concentration limits can be used in lieu of generic values provided in 10 CFR Part 20. Such adjustments result in more precise estimates that use actual exposure conditions, as compared to generic assumptions.

The 1991 revision to 10 CFR Part 20 codifies a requirement that licensees implement a program to maintain radiation doses ALARA. Compliance with the commitments is required through the licensing process in 10 CFR Part 50 and the technical specifications. Two Regulatory Guides have been issued to provide guidance on ALARA programs for nuclear power plants: one on ALARA philosophy in NRC Regulatory Guide 8.10, Rev. 1R (NRC 1977), and one on implementation in NRC Regulatory Guide 8.8, Rev. 3 (NRC 1978). Nuclear power plant licensees are required to maintain and implement adequate plant procedures that contain ALARA criteria. During plant licensing, applicants commit to implement ALARA programs consistent with Regulatory Guides 8.8 and 8.10.

Table G-5. Occupational Dose Limits for Adults in 10 CFR Part 20[(a)]

Tissue	External Radiation	Internal Plus External Radiation
Whole Body	0.05 Sv/y (5 rem/yr) total DE,[(b)] not to exceed 0.5 Sv/y (50 rem/yr) total DE to any individual organ or tissue other than the lens of the eye	0.05 Sv/yr (5 rem/year) TEDE,[(c)] not to exceed 0.5 Sv/yr (50 rem/yr) total DE to any individual organ or tissue other than the lens of the eye
Lens	0.15 Sv/yr (15 rem/yr)	
Extremities, Including Skin	0.5 Sv/yr (50 rem/yr)	
All Other Skin	0.5 Sv/yr (50 rem/yr)	

(a) These revised 10 CFR Part 20 standards became effective on January 1, 1994.
(b) The total DE is the sum of the EDE (at 1 cm [0.39 in] depth) and the CDE from nuclides deposited in the body.
(c) The TEDE is the sum of the EDE (at 1 cm depth [0.39 in]) and the CEDE from nuclides deposited in the body.

G.1.3 Public Radiation Protection Standards

For many years, the ICRP and NCRP recommended dose limits for the public that were 10 percent of those for workers. During the 1980s, both organizations adopted a more conservative value of 2 percent. In 1985, the ICRP released a statement that its principal limit for the whole body was 0.001 Sv/yr (0.1 rem/yr) EDE (ICRP 1985). However, a subsidiary limit of 0.005 Sv/yr (0.5 rem/yr) is authorized, provided that the average dose over a lifetime does not exceed 0.001 Sv/yr (0.1 rem/yr). The ICRP limit for the skin and lens of the eye is 0.05 Sv/yr (5 rem/yr). In 1987, the NCRP recommended limits of 0.001 Sv/yr (0.1 rem/yr) EDE for the whole body under conditions of continuous or frequent exposure and 0.005 Sv/yr (0.5/yr) for infrequent exposure (NCRP 1987). The NCRP limit for the lens of the eye, skin, and extremities is 0.05 Sv/yr (5 rem/yr).

The 1991 revision of 10 CFR Part 20 implements guidelines consistent with the recommended limit of 0.001 Sv/yr (0.1 rem/yr) EDE (see Table G-6). Provision is made for temporary increases to 0.005 Sv/yr (0.5 rem/yr) with prior authorization and justification. Hourly and annual dose rate limits for unrestricted areas are also included.

Licensees may also demonstrate compliance with the provisions of 10 CFR Part 20 by showing that annual average concentrations of radioactive material released in gaseous and liquid effluents at the boundary of an unrestricted area do not exceed the values specified in 10 CFR Part 20, Appendix B, Table 2.

The NRC has not established standards for radiological exposures to biota other than humans on the basis that limits established for the maximally exposed members of the public would provide adequate protection for other species. In contrast to the regulatory approach applied to human exposures, the fate of individual nonhuman organisms is of less concern than the

Table G-6. Dose Limits for an Individual Member of the Public under 10 CFR Part 20[a]

Applicability by Pathway	Dose Limits
Annual dose, all pathways[b]	1 mSv/yr (0.1 rem/yr) TEDE[c]
External dose rate, unrestricted areas	0.02 mSv/h (0.002 rem/h) or 0.5 mSv/yr (0.05 rem/yr)
Temporary Annual Dose, all pathways[d]	5 mSv/yr (0.5 rem/yr) TEDE[c]
ALARA dose constraint, air emissions	0.1 mSv/yr (0.01 rem/yr) TEDE[c]

(a) These revised 10 CFR Part 20 standards became effective on January 1, 1994.
(b) Excludes contribution from materials disposed to sanitary sewers.
(c) The TEDE is the sum of the EDE (at 1 cm depth) and the CEDE from nuclides deposited in the body.
(d) Temporary increases in the public dose limit are subject to prior authorization from the NRC and other constraints to ensure the increase is justified and controlled to be ALARA.

maintenance of the endemic population (NCRP 1991). Experience has shown that population stability is crucial to survival of most species. However, in many ecosystems individual members of a species may suffer relatively high mortality rates from natural causes without creating detrimental effects to the population as a whole. The exception might be for threatened or endangered species where protection of the individual may be required in order to avoid detrimental effects on a relatively small population.

Evaluations of radiation exposures to nonhuman biota at nuclear power facilities have not identified exposures that could be considered significant in terms of harm to the species, or which approach the public exposure limits in 10 CFR Part 20. Limiting exposure in humans to 1 mSv/yr (100 mrem/yr) will lead to dose rates to plants in animals in the same area of less than 1 mGy per day (100 mrad per day). The International Atomic Energy Agency (IAEA) concludes that there is no convincing evidence from scientific literature that chronic radiation dose rates below 1 mGy per day (100 mrad per day) will harm plant or animal populations (IAEA 1992). Because of the relatively lower sensitivity of nonhuman species to radiation, and the lack of evidence that nonhuman populations or ecosystems would experience detrimental effects at radiation levels found in the environment around nuclear power stations, effects on these biota are not evaluated in detail for the purposes of this Supplement.

In addition to the basic standards mentioned above, 10 CFR 50.36(a) contains license conditions that are imposed on licensees in the form of technical specifications applicable to effluents from nuclear power reactors. These specifications ensure that releases of radioactive materials to unrestricted areas during normal operations, including expected operational occurrences, remain ALARA. Appendix I to 10 CFR Part 50 provides numerical guidance on dose-design objectives and limiting conditions for operation for light-water reactors (LWRs) to meet the ALARA requirements. As a part of the licensing process, all licensees have provided reasonable assurance that the design objectives will be met for all unrestricted areas even during the decommissioning process. Title 10 CFR Part 20 requires compliance with the U.S. Environmental Protection Agency regulation 40 CFR Part 190, which also contains ALARA limits. The dose constraints are summarized in Tables G-7 and G-8.

Table G-7. 10 CFR Part 50, Appendix I, Design Objectives and Annual Limits on Radiation Doses to the General Public from Nuclear Power Facilities[(a)]

Tissue	Gaseous	Liquid
Total body	0.05 mSv (5 mrem)	0.03 mSv (3 mrem)
Any organ, all pathways	--	0.01 mSv (10 mrem)
Ground-level air dose	0.1 mGy (10 mrad) gamma and	--
	0.3 mGy (30 mrad) beta	--
Any organ,[(b)] all pathways	0.15 mSv (15 mrem)	--
Skin	0.15 mSv (15 mrem)	

(a) Calculated doses.
(b) Particulates, radioiodines.

Table G-8. 40 CFR 190, Subpart B, Annual Limits on Doses to the General Public from Nuclear Power Operations[(a)]

Tissue	Limit	Source
Total body	0.25 mSv (25 mrem)	All effluents and direct radiation from nuclear power operations
Thyroid	0.75 mSv (75 mrem)	"
Any other organ	0.25 mSv (25 mrem)	"

(a) Calculated doses.

Specific radiological criteria for license termination were added to 10 CFR Part 20 in 1997, and the basis for public health and safety considerations is discussed in NUREG-1496 (NRC 1997). These criteria limit the dose to members of the public to 0.25 mSv/yr (25 mrem/yr) from all pathways following unrestricted release of a property. In cases where unrestricted release is not feasible, the licensee must provide for institutional controls that would limit the dose to members of the public to 0.25 mSv/yr (25 mrem/yr) during the control period and to 1 mSv/yr (100 mrem/yr) after the end of institutional controls. These criteria will largely determine the types and extent of activities undertaken during the decommissioning process to reduce the radionuclide inventory remaining onsite.

G.2 Nuclear Power Plant Exposure Data

G.2.1 Occupational Dose Experience

Individual occupational doses are measured by NRC licensees as required by the basic NRC radiation protection standard, 10 CFR Part 20. The exposure pathway of primary interest is from sources that are external to the body. Measurements of the whole-body dose are normally derived from personal dosimeters worn by each worker, and they represent a relatively uniform dose to all organs of the body. Since 1984, many of the nuclear power plants have provided dosimetry programs accredited by the National Bureau of Standards (NBS, now National Institute of Standards and Technology [NIST]). In 1988, NBS/NIST accreditation became an NRC requirement.

Whole-body dose data from NRC-licensed LWRs are shown in Table G-9 for the years 1973 through 1999 (NRC 2000). For each year, the number of reactors, the number of workers

receiving measurable exposures, the average annual dose per worker, the collective dose for all reactors combined, and the number of individuals exceeding 0.05 Sv (5 rem) are listed. Until 1991, the limit for exposure to workers was 0.03 Sv per quarter (3 rem per quarter), or a maximum of 0.12 Sv/yr (12 rem/yr), with an average of 0.05 Sv/yr (5 rem/yr). The collective dose is the sum of doses to workers at all plants. The collective doses to nuclear plant workers decreased from a peak of over 55 person-Sv/yr) (55,000 person-rem/yr) in 1983-1984 to less than 15 person-Sv/yr (15,000 person-rem/yr) in 1998-1999, although there are currently about 25 percent more operating plants than in the mid-1980s. Average annual doses to workers have likewise decreased from just under 0.01 Sv/yr (1 rem/yr) in the early 1970s to less than 0.25 mSv/yr (0.25 rem/yr) after 1997. Whole-body doses exceeding 0.05 Sv/yr (5 rem/yr) have been infrequent since 1985, and no doses at that level have been reported since 1989. Nuclear power plant workers may also be exposed to airborne radioactive material, primarily fission and corrosion products, but such exposures have historically been small in comparison with external doses. A study of intake data indicated that for cobalt-58 and cobalt-60, the most prevalent radionuclides, very few of the workers had organ burdens of more than 1 percent of the maximum permissible (see Baker 1996).

These data indicate that occupational exposures within the nuclear power industry have been significantly reduced since 1973. Individual doses are characteristically far below the regulatory limit, and the annual average is less than 5 percent of the 5 rem per year limit that is now in effect. Effective implementation of the ALARA concept is largely responsible. The range of risks associated with these exposures are discussed in Section G.1.

Occupational doses at reactors that are undergoing decommissioning are a small fraction of those accumulated at operating facilities, as indicated in the Table G-9 data for reactors that are no longer operating. Between 1995 and 1999, the collective dose from shutdown facilities typically amounted to a few hundred person-rem per year, and the annual average dose per worker was comparable to, or lower than, that for operating facilities. A comparison in Table G-10 of the occupational doses at 12 facilities before and after they were shutdown confirms that decommissioning would not be expected to increase occupational doses on average, although some phases of the process may result in temporarily higher collective doses depending on the activities in progress and the number of workers involved.

Tables G-11 and G-12 list available data regarding the distribution of the cumulative collective worker dose among the major types of activities that would occur during a typical decommissioning process. The lack of resolution in much of the data and the small number of facilities involved (10) precludes a detailed analysis. However, it appears that the largest share of occupational doses might be expected for three general classes of activities: (1) large component removal (reactor vessel, steam generators), (2) removal of other plant systems, structures, and components, and (3) the remaining general decontamination activities. Data for removal of the reactor vessel (Table G-12) indicate that the choice of removal method (i.e., intact or segmented) may influence the collective dose associated with the operation. Data for plants electing the SAFSTOR alternative were not substantially different from plants

Table G-9. Occupational Dose at Light Water Reactors (LWRs) - Comparison of Operating Reactors to Reactors No Longer in Operation[a]

Year	Number of Workers with Measurable Exposure[b]	Collective Dose, person-rem[c]	Average Dose per Worker with Measurable Exposure, rem[c]	Total Number with Dose > 5 rem[d]	Number of Reactors	Average Collective Dose per Reactor-Year, person-rem[e]
Operating Reactors						
1973	14,780	13,962	0.945	--	24	582
1974	18,139	13,650	0.753	--	33	414
1975	28,234	20,901	0.740	--	44	475
1976	34,515	26,105	0.756	--	52	502
1977	38,985	32,521	0.834	351	57	571
1978	42,777	31,785	0.743	159	64	497
1979	60,299	39,908	0.662	180	67	596
1980	74,629	53,739	0.720	391	68	790
1981	76,772	54,163	0.706	210	70	774
1982	79,309	52,201	0.658	135	74	705
1983	79,709	56,484	0.709	169	75	753
1984	90,520	55,251	0.610	74	78	708
1985	86,926	43,048	0.495	1	82	525
1986	93,979	42,386	0.451	0	90	471
1987	96,231	40,406	0.420	0	96	421
1988	96,013	40,772	0.425	1	102	400
1989	100,084	35,931	0.359	0	107	336
1990	98,567	36,602	0.371	0	110	333
1991	91,086	28,519	0.313	0	111	257
1992	94,172	29,297	0.311	0	110	266
1993	86,193	26,364	0.306	0	108	244
1994	71,613	21,704	0.303	0	109	199
1995	70,821	21,688	0.306	0	109	199
1996	68,305	18,883	0.276	0	109	173
1997	68,372	17,149	0.251	0	109	157
1998	57,466	13,187	0.229	0	105	126
1999	59,216	13,666	0.231	0	104	131
Average 1973-1999	69,545	32,603	0.514	73		430
Average 1995-1999	64,836	16,915	0.259	0		157
Permanently Shutdown Reactors[f]						
1995	699	262	0.375	0	6	44
1996	974	165	0.169	0	8	21
1997	1144	136	0.119	0	7	19
1998	2178	430	0.197	0	11	39
1999	2856	430	0.151	0	13	33
Average 1995-1999	1,570	285	0.202			31

(a) Data Source: NUREG-0713, Vol. 21 (NRC 2000).
(b) 1973-1976 data are not adjusted for multiple reporting of transient individuals.
(c) To convert rem to sievert, multiply by 0.01.
(d) Number of workers by dose range not available for 1973-1976. The dose limit was 3 rem/quarter (12 rem/yr) before the 1991 revision of 10 CFR Part 20; thereafter, it was reduced to 5 rem/yr.
(e) To convert person-rem to person-sievert, multiply by 0.01.
(f) Includes plants not in operation for a full year as of December 31 of the reporting year.

Table G-10. Occupational Whole-Body Dose at Decommissioning Reactors, Comparison of Dose During Operations to Dose During Decommissioning

						Average Annual Occupational Dose, person-rem/yr			Maximum Annual Occupational Dose, person-rem/yr		
Nuclear Plant	**Reactor Type**	**Capacity, MWe**	**Years in Operation**	**Years Post Shutdown**	**D&D Method**	**Normal Power Operations**	**Post Shutdown**	**Post Shutdown as % of Operations**	**Operations**	**Post Shutdown**	**Post Shutdown as % of Operations**
Ft. St. Vrain	HTGR[a]	330	10	12	DECON	3	106	4076.9	6	210	3500
Big Rock Point	BWR[b]	67	34	2	DECON	166	116	69.7	277	144	52.0
La Crosse	BWR	48	17	13	SAFSTOR	247	19	7.8	313	105	33.5
Humboldt Bay, Unit 3	BWR	63	13	25	SAFSTOR	294	183	62.4	339	1905	561.9
Yankee Rowe	PWR[c]	175	30	8	DECON	159	75	47	246	156	63.4
Haddam Neck	PWR	560	28	3	DECON	355	137	38.5	590	261	44.2
Maine Yankee	PWR	860	25	3	DECON	326	154	47.1	653	173	26.5
Trojan	PWR	1080	17	7	DECON	346	38	11	567	52	9.2
San Onofre, Unit 1	PWR	436	25	8	SAFSTOR	512	16	3.1	880	16	1.8
Rancho Seco	PWR	873	14	10	SAFSTOR	385	9	2.3	787	41	5.2
Zion, Units 1 and 2	PWRs	2080	24	2	DECON	645	8	1.2	1043	12	1.2
Average All LWR						343	75	29	570	287	79.9
Average BWR						235	106	46.6	310	718	215.8
Average PWR						390	62	21.5	681	102	21.6
Average DECON						333	88	35.8	563	133	32.7
Average SAFSTOR						359	57	18.9	580	517	150.6

(a) High-temperature gas-cooled reactor.
(b) Boiling water reactor.
(c) Pressurized water reactor.

Table G-11. Occupational Dose by Activity During Decommissioning

					Percent of Total Cumulative Dose to Completion by Activity					
Nuclear Plant	**Reactor Type**	**Capacity, MWe**	**D&D Method**	**Cumulative Dose Post Shutdown, person-rem[a]**	**Large Component Removal, %**	**Systems, Structures, and Components Removal, %**	**Other Decon Activities, %**	**SNF Management, %**	**Transportation, %**	**SAFSTOR Activities, %**
Fort St. Vrain	HTGR[b]	330	DECON	433	45.1	25.6	13.8		15.5	
Big Rock Point	BWR[c]	67	DECON	700						
Haddam Neck	PWR[d]	560	DECON	996	37	28.7	19.3	8.7	6.1	
Maine Yankee	PWR	860	DECON	946	9.9	12.8	74.2	3		
Trojan	PWR	1080	DECON	556	22.7	50.7	5.4	21.2		
Zion, Units 1 and 2	PWRs	2080	SAFSTOR	637						
Humboldt Bay, Unit 3	BWR	63	SAFSTOR	354			50.8		3.7	45.5
Rancho Seco	PWR	873	SAFSTOR	483	39.1	47.6	5.8			7.5
San Onofre, Unit 1	PWR	436	SAFSTOR	1100						
Average All Plants				689	26.9	28	36.9	8.3	8.4	18.1
Number of Plants				9	6	6	7	4	3	3
Occupational Dose in Decommissioning BWRs										
Average BWR				527			50.8		3.7	45.5
Number of Plants				2			1		1	1
BWR SAFSTOR				354			50.8		3.7	45.5
BWR DECON				700						
Occupational Dose in Decommissioning PWRs										
Average PWR				786	23.2	28.4	38.7	8.3	6.1	4.4
Number of Plants				6	5	5	5	4	1	2
PWR SAFSTOR				792	23.3	25	47.2	0.3		4.4
PWR DECON				784	23.2	30.8	33	11	6.1	

(a) Dose is estimated for activities during decommissioning at plants that have not reached license termination.
(b) High-temperature gas-cooled reactor.
(c) Boiling water reactor.
(d) Pressurized water reactor.

Table G-12. Reactor Vessel Removal Information and Data

Nuclear Plant	Total Bequerels (Curies) Removed	Personnel Exposure person-sievert (person-rem)	Segmented components/ Lineal inches cut	Cutting Methods	Considerations for Planning and Implementation
Haddam Neck (in progress)	27,750 (750,000)	1.77 (177)	• Core baffle • Core former plates • Core barrel in active fuel region • Lower core support plate • Lineal inches cut - 23,251	• Abrasive water • MDM cutting	• Worker exposure • Airborne contamination • Waste form and disposal costs • Cavity cleanup requirements • Schedule
San Onofre, Unit 1 (in progress)	12,210 (330,000)	0.73 (73) 0.14 (14)	• Core region of the core barrel • Core baffles/formers • Lower core support plates • Lineal inches cut - 10,821	• Abrasive water • MDM cutting	
Maine Yankee (in progress)	Not available	(actual to date) 0.24 (24) (projected)	• Upper guide structure • Upper core barrel • Core support barrel • Mid-core region • Thermal shield • Lineal inches cut - 14,000	• Abrasive water jet (AWJ) • Conventional machining	• Avoid thermal processing • Use AWJ and conventional machining vs. plasma arc and MDM/EDM to reduce the occupational dose • Modeled all the cuts in a 3D CAD system before actually performing any of the dismantlement • Segregating, capturing, and confining AWJ cutting waste • Solid waste collection system • Cavity water treatment system • Much Maine Yankee dismantlement done under water and remotely, which cut down the worker dose • Abrasive Feed Assist System (patent pending) • Underwater AWJ Vision Enhancement - remote operability (patent pending) • Minimized amount of secondary waste • For underwater equipment, a maintenance and reliability issue • Sequence of cuts (low to high activity) reduced occupational exposure
Big Rock Point (in progress)	Not available	Not available	N/A	N/A	

Table G-12. (contd)

Nuclear Plant	Total Bequerels (Curies) Removed	Personnel Exposure (person-rem)	Segmented components/ Lineal inches cut	Cutting Methods	Considerations for Planning and Implementation
Trojan (completed)	74,000 (2,000,000)[(a)]	0.72 (72)	N/A	N/A	• Used the fuel transfer crane to lift the reactor vessel and place in the container • Removed reactor vessel with internals intact • The internals were grouted in place with low-density cellular concrete • Placed the reactor vessel on a heavy haul trailer for road transport to the rail • Shipped the reactor vessel with internals to U.S. Ecology, Richland, WA • Eliminated 74,000 Bq (2 million curies) from the Trojan nuclear facility site

(a) The Trojan plant reactor vessel was removed and shipped intact to the disposal facility; reactor vessel internals were not removed as in the other plants listed in this table.

undergoing more immediate DECON. The one exception was at Humboldt Bay, where the plant was maintained in a shutdown condition over an extended period of time. In that case, SAFSTOR activities accounted for a relatively large fraction of the total estimated occupational dose. In all cases, the estimated cumulative doses through the end of decommissioning for these plants were within the estimates presented in the 1988 GEIS (NRC 1988).

G.2.2 Dose to Members of the Public

Doses to members of the public from power reactor effluents were summarized in a series of NRC reports entitled *Dose Commitments Due to Radioactive Releases from Nuclear Power Plant Sites*. The last volume published covers reactor operations during 1992 (NUREG/CR-2850, Baker 1996). Radioactive material is released in gaseous (airborne, and may contain particulates, such as radioiodine) and liquid (aqueous) effluents under stringently controlled conditions in accordance with technical specifications and NRC regulations. The term “dose commitment” indicates that the reported doses come from the inhalation and ingestion of radionuclides, as well as from external radiation from noble gases. The population dose caused by direct radiation from plant facilities is negligible. Table G-13 presents results obtained for the 18-year period ending in 1992. The public doses represent collective person-rem received by those who live within an 80-km (50-mi) radius of a site; data for individual sites also appear in this report. The population dose within 80 km (50 mi) of each plant is calculated for each operating reactor in the United States. The total collective dose is then obtained by combining the doses received by these populations. As with the occupational doses, collective dose to the public from reactor effluents has been decreasing steadily since the mid-1980s. The collective dose to members of the public is smaller by several orders of magnitude than the dose to plant workers.

Data on maximally exposed individuals from gaseous effluents is also reported annually to the NRC by each nuclear utility. Data for the period 1985-1987 were compiled in NUMARC (1989) and summarized in NRC (1996). A summary of the data is presented in Table G-14.

Inspection of this table reveals that the maximum doses to individuals via gaseous effluents are on the order of a few mrem per year, and the dose to an individual is orders of magnitude lower for most plants.

A comparison of more recent effluent release rates from both operating and decommissioning facilities (Table G-15) indicates that the gaseous release rates for many types of effluents are similar. Decommissioning facilities reported no emissions of radioiodine in their gaseous effluents, which would be as expected after the plants are shut down and de-fueled. Most of the iodine isotopes are short-lived and are not present in plants that have been out of operation for any length of time. Releases of longer-lived fission gases and particulate materials in gaseous effluents continue after the end of operation because of the need to maintain plant ventilation systems during activities associated with the decommissioning process. Radionuclide emissions in liquid effluents were typically lower in the shutdown facilities because the reactor core cooling systems were not operating, and the levels of radionuclides in circulating water systems needed to maintain the spent fuel pool are lower than in primary coolant for an operating plant.

Table G-13. Summary of Collective Public and Occupational Doses for All Operating Nuclear Power Facilities Combined[(a)]

		Collective Public Dose, person-rem			
Year	**Number of Operating Reactors[(b)]**	**Liquid Effluents**	**Gaseous Effluents**	**Total**	**Average per reactor-yr, person-rem**
1975	44	76	1300	1300	30
1976	52	82	390	470	9.0
1977	57	160	540	700	12
1978	64	110	530	640	10
1979	67	220	1600	1800	27
1980	68	120	57	180	2.6
1981	70	87	63	150	2.1
1982	74	50	87	140	1.9
1983	75	95	76	170	2.3
1984	78	160	120	280	3.6
1985	82	91	110	200	2.4
1986	90	71	44	110	1.2
1987	96	56	22	78	0.81
1988	102	65	9.6	75	0.74
1989	107	68	16	84	0.79
1990	110	63	15	78	0.71
1991	111	70	17	88	0.79
1992	110	32	15	47	0.43

(a) Collective public dose calculated for those living within an 80-km (50-mi) radius of a nuclear plant site.
(b) Includes plants in operation at least 1 full year at the end of the reporting year.
Source: NUREG/CR-2850 (Baker 1996).
Note: To convert person-rem to person-sievert, multiply by 0.01.

Table G-14. Estimated Doses to the Maximally Exposed Individual from Routine Gaseous Effluents from Operating Facilities, mrem[(a)]

	1985	**1986**	**1987**
Average	2.8E-01	2.6E-01	9.1E-02
Minimum	7.8E-04	4.9E-04	1.0E-06
Maximum	1.8E+00	4.3E+00	8.9E-01
Number of plants reporting	26	33	34

(a) Data compiled from reports submitted to the NRC by each nuclear utility.
Adapted from NUMARC (1989).
Note: To convert millirem to millisievert, multiply by 0.01.

Table G-15. Summary of Effluent Releases Comparison of Operating Facilities and Decommissioning Facilities

Operating Reactors						
Reactor Type	**PWR**			**BWR**		
	Average	**Max**	**Min**	**Average**	**Max**	**Min**
Capacity (MWe)	829	912	760	972	1154	786
Gaseous Effluents - Total (Ci)	5.8E+01	1.5E+02	4.0E-01	9.3E+01	1.7E+02	1.2E+01
Fission and Activation Gases (Ci)	4.4E+01	1.4E+02	7.5E-02	8.3E+01	1.6E+02	1.5E+00
Iodines (Ci)	6.4E-07	1.3E-06	0	2.3E-03	5.1E-03	0
Particulates (Ci)	1.9E-05	3.8E-05	3.3E-07	8.9E-04	1.6E-03	3.0E-04
Gross Alpha (Ci)	--	--	--	--	--	--
Tritium (Ci)	1.4E+01	3.7E+01	3.2E-01	1.0E+01	1.2E+01	6.2E+00
Liquid Effluents - Total (Ci)	5.2E+02	6.7E+02	4.2E+02	1.2E+01	1.9E+01	6.9E+00
Fission and Activation Products (Ci)	1.6E-01	3.7E-01	8.5E-02	6.2E-02	9.4E-02	1.2E-02
Tritium (Ci)	5.2E+02	6.7E+02	4.2E+02	1.2E+01	1.9E+01	6.9E+00
Dissolved and Entrained Gases (Ci)	1.0E-01	3.8E-01	2.2E-04	4.3E-03	6.7E-03	1.8E-03
Gross Alpha (Ci)	1.2E-03	1.9E-03	4.4E-04	2.4E-06	3.8E-06	0
Decommissioning Reactors						
Reactor Type	**PWR**			**BWR**		
	Average	**Max**	**Min**	**Average**	**Max**	**Min**
Capacity, MWe	970	1080	860	65	67	63
Gaseous Effluents - Total (Ci)	2.1E+01	4.0E+01	2.6E+00	1.1E+02	2.1E+02	1.2E+00
Fission and Activation Gases (Ci)	1.6E+01	1.6E+01	1.6E+01	2.1E+02	2.1E+02	2.1E+02
Iodines (Ci)	--	--	--	--	--	--
Particulates (Ci)	0	0	0	1.0E-04	2.0E-04	0
Gross Alpha (Ci)	--	--	--	0	0	0
Tritium (Ci)	1.3E+01	2.4E+01	2.6E+00	1.2E+00	1.2E+00	1.2E+00
Liquid Effluents - Total (Ci)	7.8E-01	1.4E+00	1.2E-01	3.3E-01	1.3E+00	1.0E-03
Fission and Activation Products (Ci)	3.5E-02	6.7E-02	2.6E-03	3.3E-01	1.3E+00	2.0E-04
Tritium (Ci)	7.4E-01	1.4E+00	1.2E-01	9.5E-04	1.1E-03	8.0E-04
Dissolved and Entrained Gases (Ci)	--	--	--	--	--	--
Gross Alpha (Ci)	0	3.0E-05	0	0	0	0

Recent DEs to members of the public from emissions at operating and decommissioning facilities were similar, and the doses from gaseous effluents were within the ranges published in NRC (1996) for operating facilities (see Table G-16). Both individual and collective doses were very low for liquid and gaseous effluents. Although information was available for a relatively small sample of facilities, there does not appear to be any reason to project substantial increases in emissions or public doses from reactors undergoing decommissioning compared to the levels experienced during normal operation of those facilities.

Table G-16. Summary of Public Doses from Operating and Decommissioning Facilities

Operating Reactors	Columbia Generating Station	Turkey Point		ANO		Hatch	
		Unit 3	Unit 4	Unit1	Unit 2	Unit 1	Unit 2
Year	**1999**	**1997**	**1997**	**1999**	**1999**	**1999**	**1999**
Air Pathways							
Collective (person-rem)	1.9E-02	--	--	--	--	--	--
Individual (mrem)	4.3E-03	4.0E-06	3.8E-06	5.7E-03	1.0E-02	1.9E-03	4.4E-03
Water Pathways							
Collective (person-rem)	--	--	--	--	--	--	--
Individual (mrem)	--	9.5E-04	9.5E-04	6.7E-03	1.8E-03	3.9E-02	2.9E-02
Collective Total	1.9E-02	--	--	1.2E-02	1.2E-02	--	--

Decommissioning Reactors	Big Rock Point		Humboldt Bay, Unit 3	
Year	**1998**	**1999**	**1998**	**1999**
Air Pathways				
Collective (person-rem)	1.7E-04	1.5E-04	--	--
Individual (mrem)	--	1.2E-03	4.0E-02	1.0E-02
Water Pathways				
Collective (person-rem)	6.7E-02	1.6E-01	6.4E-04	--
Individual (mrem)	5.7E-02	3.1E-01	4.0E-02	1.0E-02
Collective Total	6.7E-02	1.6E-01	6.4E-04	--

G.3 References

10 CFR 20. Code of Federal Regulations, Title 10, *Energy*, Part 20, “Standards for protection against radiation.”

10 CFR 50. Code of Federal Regulations, Title 10, *Energy*, Part 50, “Domestic licensing of production and utilization facilities.”

40 CFR 190. Code of Federal Regulations, Title 40, *Protection of Environment*, Part 190, “Environmental radiation protection standards for nuclear power operations.”

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G.4 Related Documents

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