United States Nuclear Regulatory Commission Official Hearing Exhibit
In the Matter of: Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3)
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Table S.3 Continued

Environmental considerations	Total	Maximum effect per annual fuel requirement or reference reactor year of model 1000-MW(e) LWR
I-131	0.83	
Tc-99		Currently under consideration by the Commission.
Fission products and transuranics	0.203	
Liquids		
Uranium and daughters	2.1	Principally from milling—included tailings liquor and returned to ground—no effluents; therefore, no effect on environment.
Ra-226	0.0034	From UF_6 production.
Th-230	0.0015	
Th-234	0.01	From fuel fabrication plants—concentration 10% of 10 CFR 20 for total processing 26 annual fuel requirements for model LWR.
Fission and activation products	5.9×10^{-a}	
Solids (buried on site)		
Other than high level (shallow)	11,300	9100 Ci comes from low-level reactor wastes and 1500 Ci comes from reactor decontamination and decommissioning—buried at land burial facilities. 600 Ci comes from mills—included in tailings returned to ground. Approximately 60 Ci comes from conversion and spent-fuel storage. No significant effluent to the environment.
TRU and HLW (deep)	1.1×10^{6}	Buried at federal repository.
Effluents—thermal, billions of British thermal units	4,063	< 5% of model 1000-MW(e) LWR.

See footnotes at end of table.

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Table S.3 Continued

Environmental considerations	Total	Maximum effect per annual fuel requirement or reference reactor year of model 1000-MW(e) LWR
Transportation, man-rem		
Exposure of workers and general public	2.5	
Occupational exposure	22.6	From reprocessing and waste management.

[a]In some cases where no entry appears, it is clear from the background documents that the matter was addressed and that, in effect, the table should be read as if a specific zero entry had been made. However, other areas are not addressed at all in the table. Table 10.5-3 does not include health effects from the effluents described in the table, estimates of releases of radon-222 from the uranium fuel cycle, or estimates of technetium-99 released from waste management or reprocessing activities. These issues may be the subject of litigation in the individual licensing proceedings.

Data supporting this table are given in WASH-1248; NUREG-0116; NUREG-0216; and in the record of the Docket RM-50-3. The contributions from reprocessing, waste management, and transportation of wastes are maximized for either of the two fuel cycles (uranium only and no recycle). The contribution from transportation excludes transportation of cold fuel to a reactor and of irradiated fuel and radioactive wastes from a reactor, which are considered in Table S-4 of § 5.20(g). The contributions from the other steps of the fuel cycle are given in columns A through E of Table S-3A of WASH-1248.

[b]The contributions to temporarily committed land from reprocessing are not prorated over 30 years, because the complete temporary impact accrues regardless of whether the plant services 1 reactor for 1 year or 57 reactors for 30 years.

[c]Estimated effluents based upon combustion of equivalent coal for power generation.

[d]1.2% from natural gas use and process.

Source: 10 CFR 51.51.

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Table S.4 Environmental impact of transportation of fuel and waste to and from one light-water-cooled nuclear power reactor,[a] normal conditions of transport

		Environmental impact	
Heat (per irradiated fuel cask in transit)		250,000 Btu/hr	
Weight (governed by Federal or State restrictions)		73,000 lb per truck; 100 tons per cask per rail car	
Traffic density			
Truck		Less than 1 per day	
Rail		Less than 3 per month	
Exposed population	Estimated number of persons exposed	Range of doses to exposed individuals[b] (per reactor year)	Cumulative dose to exposed population (per reactor year)[c]
Transportation workers	200	0.01 to 300 mrem	4 man-rem
General public			
Onlookers	1,100	0.003 to 1.3 mrem	3 man-rem
Along route	600,000	0.0001 to 0.06 mrem	

Accidents in transport

	Environmental risk
Radiological effects	Small[d]
Common (nonradiological) causes	1 fatal injury in 100 reactor years, 1 nonfatal injury in 10 reactor years, $475 property damage per reactor year

[a]Data supporting this table are given in the Commission's *Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants*, WASH-1238, December 1972, and Supp. 1 NUREG-75/038, April 1975. Both documents are available for inspection and copying at the Commission's Public Document Room, 2120 L Street N.W., Washington, D.C., and may be obtained from National Technical Information Service, Springfield, VA 22161. WASH-1238 is available from NTIS at a cost of $5.45 (microfiche, $2.25) and NUREG-75/038 is available at a cost of $3.25 (microfiche, $2.25).

[b]The Federal Radiation Council has recommended that the radiation doses from all sources of radiation other than natural background and medical exposures should be limited to 5000 mrem per year for individuals as a result of occupational exposure and should be limited to 500 mrem per year for individuals in the general population. The dose to individuals due to average natural background radiation is about 130 mrem per year.

[c]Man-rem is an expression for the summation of whole body doses to individuals in a group. Thus, if each member of a population group of 1000 people were to receive a dose of 0.0001 rem (1 mrem), or if 2 people were to receive a dose of 0.5 rem (500 mrem) each, the total man-rem dose in each case would be 1 man-rem.

[d]Although the environmental risk of radiological effects stemming from transportation accidents is currently incapable of being numerically quantified, the risk remains small regardless of whether it is being applied to a single reactor or a multireactor site.

Source: 10 CFR 51.52.

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Given current regulatory activities and past regulatory experience, the Commission has no reason to expect that such noncompliance will occur at any significant frequency. To the contrary, the Commission expects that future radiological impacts from the fuel cycle will represent releases and impacts within applicable regulatory limits. Collective doses and associated health effects are calculated and discussed at various places in this chapter. These estimates are provided for perspective only.

Estimates of the magnitude of the human health risks associated with the expected occupational and public dose levels based on the linear effects model should be evaluated relative to the following positions taken by the International Commission on Radiological Protection (ICRP), the National Academy of Sciences/National Research Council (NAS/NRC), and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR): (1) the estimation of health effects at low doses (comparable to external natural background) are based on extrapolation from effects seen at high doses and dose rates, with no threshold (the linear model); however, health effects at these low doses have not been demonstrated by human epidemiological studies; (2) the possibility that there may be no risk from exposures comparable to natural background radiation levels cannot be ruled out by any epidemiological studies; and (3) at low doses and dose rates, it must be acknowledged that the lower limit on the range of uncertainty in the risk estimate extends to zero. Section I.B. of the preamble to final 10 CFR Part 20 (56 FR 23360; May 21, 1991) states: "In the absence of convincing evidence that there is a dose threshold or that low levels of radiation are beneficial, the Commission believes that the assumptions regarding a linear nonthreshold dose-effect model for cancers and genetic effects and the existence of thresholds only for certain nonstochastic effects remain appropriate for formulating radiation protection standards and planning radiation protection programs." Therefore, because the health effects are uncertain at low levels of radiation dose, for regulatory purposes it is prudent to use the linear nonthreshold dose-effect model; accordingly, this model was used to estimate health effects.

Table S-3 states the environmental impacts of the uranium fuel cycle from the mining of uranium ore to the ultimate disposal of spent fuel and other radioactive waste that is generated by the use and management of fuel. Table S-4 states the environmental impacts specific to the transportation of fuel and radioactive waste to and from a reactor.

10 CFR Part 51.51(a) states in part, "Every environmental report prepared for the construction permit stage of a light-water-cooled nuclear power reactor, and submitted on or after September 4, 1979, shall take Table S-3, Table of Uranium Fuel Cycle Environmental Data, as the basis for evaluating the contribution of the environmental effects of uranium mining and milling, the production of uranium hexafluoride, isotopic enrichment, fuel fabrication, reprocessing of irradiated fuel, transportation of radioactive materials and management of low-level wastes and high-level wastes related to uranium fuel-cycle activities to the environmental costs of licensing the nuclear power reactor."

The impacts of the uranium fuel cycle are discussed in Section 6.2. The following sections of the chapter are organized by

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waste type: low level, mixed, and spent fuel. For each waste type, the issues are divided into "baseline," those that are present with or without license renewal, and "effects of license renewal," those that are attributable solely to waste management activities associated with license renewal. In addition, transportation is addressed in a separate section because it applies to all waste types.

6.2 IMPACTS OF THE URANIUM FUEL CYCLE

The following discussion of the environmental impacts of the fuel cycle as related to the operation of an individual nuclear power plant during the license renewal period is based on the values given in Table S-3 and the staff's analysis of the radiological impact from radon and technetium releases. For the sake of consistency, the data presented in Table S-3 have been cast in terms of a model 1000-MW(e) LWR operating at an annual capacity factor of 80%.

Specific categories of natural resource use included in Table S-3 relate to land use, water consumption and thermal effluents, radioactive releases, burial of transuranic and high- and low-level wastes, and radiation doses from transportation and occupational exposures. The contributions in the table for reprocessing, waste management, and transportation of wastes are maximized for either of the two fuel cycles (uranium only and no recycle); that is, the cycle that results in the greater impact is used.

6.2.1 Background of Tables S-3 and S-4

Tables S-3 and S-4 provided a summary of the environmental data, and Table S-4 provided a summary of the environmental impacts related to the LWR fuel-cycle facilities and processing operations. The environmental impact values are expressed in terms normalized to show the potential impacts attributable to processing the fuel required for the operation of a 1000-MW(e) nuclear power plant for one year at an 80 percent availability factor to produce about 800 megawatt-years (0.8 gigawatt-year) of electricity. This is referred to as one reference reactor year (RRY). The RRY fuel replacement requires, as raw material, about 182 metric tons (tonnes) of uranium. Based on U.S. uranium industry averages, which are expected to hold well into the next century, the ore assay is assumed to be 0.1 percent uranium, and the recovery of uranium from the ore to be about 90 percent. Thus the mining of about 202,000 tonnes of ore per RRY would be required. The values in Table S-3 are based on the mining and milling of this quantity of ore and the subsequent processing of related quantities of uranium compounds through all steps of the uranium fuel cycle, including radioactive waste disposal.

6.2.2 Uranium Fuel Cycle Environmental Impact

6.2.2.1 Radioactive Effluents

Radioactive effluents estimated to be released to the environment from reprocessing and waste-management activities and certain other phases of the fuel-cycle process are listed in Table S-3. Using these data, the staff has calculated for 1 year of operation of the model 1000-MW(e) LWR, the 100-year involuntary environmental dose commitment to the U.S. population from the LWR-supporting fuel cycle. The

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100-year environmental dose commitment is the integrated population dose for 100 years (i.e., it represents the sum of the annual population doses for a total of 100 years).

It is estimated from these calculations that the overall involuntary total-body gaseous dose commitment to the U.S. population from the fuel cycle (excluding reactor releases and the dose commitment due to ^{222}Rn) would be about 400 man-rem for each year of operation of the model 1000-MW(e) LWR (RRY). Based on Table S-3 values, the additional involuntary total-body dose commitments to the U.S. population from radioactive liquid effluents resulting from all fuel-cycle operations other than reactor operation would be about 200 man-rem per year of operation. Thus, the estimated involuntary 100-year environmental dose commitment to the U.S. population from radioactive gaseous and liquid releases due to these portions of the fuel cycle is about 600 man-rem (whole body) per RRY. Using risk estimators of 500 cancer deaths per million man-rem for total-body (NUREG/CR-4214, Rev. 1, Part II, Addendum 1, p. 54), the estimated cancer risk would be 0.3 per RRY ($600 \times 500 \times 10^{-6}$).

Currently, the radiological impacts associated with ^{222}Rn and ^{99}Tc releases are not addressed in Table S-3. Principal radon releases occur during mining and milling operations and as emissions from mill tailings, whereas principal ^{99}Tc releases occur from gaseous diffusion enrichment facilities. Estimates of ^{222}Rn release per RRY from these operations are given in Table 6.1. The underlying assumptions are given later. The staff has calculated population-dose commitments for these sources of ^{222}Rn using the RABGAD computer code described in Volume 3 of NUREG-0002, Appendix A, Chapter IV, Section J. The results of these calculations for mining and milling activities prior to tailings stabilization are given in Table 6.2.

For radon releases from stabilized tailings piles, the staff has assumed that the tailings would emit 1 Ci per RRY, with covering fully intact. Based on this radon release rate, the 100-year dose commitments from stabilized tailing piles are estimated to be 2.6 man-rem for total-body, 68 man-rem for bone, and 56 man-rem for lung (bronchial-epithelium). These dose commitments will continue for many years because ^{222}Rn emission source strength will be constant for about 10,000 years and ultimately decline by a factor of 2 every 80,000 years.

The long-term integrity of the coverings must be maintained because the standards in 40 CFR 192 and 10 CFR 40, Appendix A, require certification of stability and the control of average radon flux levels to 20 $pCi/m^2/s$. Under Section 83 of the Atomic Energy Act, a government agency will maintain licensed custody and provide for long-term care of mill tailings sites after closure. Actions the government agency is authorized to conduct include monitoring, maintenance, and emergency measures necessary to protect the public health and safety and other measures necessary to ensure compliance with the standards in 10 Part 40. The NRC has adopted general licenses (10 CFR 40.27 and 40.28) to implement this provision for inactive and active sites, respectively. The general licensee will be the Department of Energy or successor agency, another agency designated by the President, or a state where the disposal site is located. The design and implementation of the radon cover and erosion protection features are the primary reliance for maintaining radon

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Table 6.1 Radon releases from mining and milling operations and mill tailings for each year of operation of the model 1000-MW(e) light-water reactor

Radon source	Quantity released
Mining, Ci	4060
Milling and tailings (during active milling), Ci	780
Inactive tailings (prior to stabilization), Ci	350
Stabilized tailings, Ci/year	1

Table 6.2 Estimated 100-year environmental dose commitment from mining and milling for each year of operation of the model 1000-MW(e) light-water reactor

		Dosage (man-rem)		
Radon source	^{222}Rn release (Ci)	Total body	Bone	Lung (bronchial epithelium)
Mining	4100	110	2800	2300
Milling and tailings (other than stabilized)	1100	29	750	620
Total		140	3600	2900

emissions within the Part 40 limits; significant failure of the covers is considered highly unlikely. However, the indefinite licensed long-term custody and care provide additional assurances. Thus, the NRC staff concludes that any needed repairs will be done and that the most likely future for the closed stabilized tailings piles is conformance with the emission standards in 10 CFR 40. On the other hand, there are inherent uncertainties associated with any reliance on institutional controls. In its recent report (NAS 1995), NAS concluded that there is no technical basis for relying on institutional controls for high-level waste facilities. From a policy/resource perspective, future generations may choose not to fulfill the obligations now specified in law for mill tailings. If such a decision is made, the radon emissions might increase by a factor of two orders of magnitude as centuries. Such a policy decision is not irrevocable and may be reversed so that the covers could be repaired at a later date. In spite of these uncertainties, staff believes that the combination of engineering and institutional controls will most likely result in compliance with the flux emission standards now in place for the foreseeable future.

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These doses and predicted health effects have been compared with those that can be expected from natural emissions of ^{222}Rn. Using data from the National Council on Radiation Protection and Measurements (NCRP 1987), the average indoor ^{222}Rn air concentration in air in the contiguous United States is about 1 pCi/L, and the short-half-lived daughter concentration is 0.004 WL (working level). The NCRP estimates that an annual lung dose from radon of 20 mrem will result in an annual dose to the bronchial epithelium of 2400 mrem as a result of the daughter products. For a stabilized future U.S. population of 300 million, this represents a total lung-dose commitment of 720 million man-rem per year. Using the same risk estimator of 78 lung-cancer fatalities per million lung man-rems used to predict cancer fatalities for the model 1000-MW(e) LWR, estimated lung-cancer fatalities alone from natural ^{222}Rn in the indoor air can be calculated to be up to 56,000 per year.

The staff has assumed that after completion of active mining, underground mines will be sealed, returning releases of ^{222}Rn to background levels. For purposes of providing an upper-bound impact assessment, the staff has assumed that open-pit mines will be unreclaimed and has calculated that if all ore were produced from open-pit mines, releases from them would be 110 Ci per RRY. However, because the distribution of uranium ore reserves available by conventional mining methods is 66 percent underground and 34 percent open-pit [GJO-100(78)], the staff has further assumed that uranium to fuel LWRs will be produced by conventional mining methods in these proportions. This means that long-term releases from unreclaimed open-pit mines will be 37 Ci/year (0.34 × 110) per RRY. In 1994, 100 percent of the domestic uranium came from in situ mining and other sources. None came from underground or open pit (conventional) mining (DOE/EIA-0478, 1995).

Based on these assumptions, the radon released from unreclaimed open-pit mines over 100- and 1000-year periods would be about 3,700 and 37,000 Ci per RRY, respectively. The total dose commitments for a 100- to 1,000-year period would be as shown in Table 6.3. These commitments represent a worst-case situation in that no mitigating circumstances are assumed. However, state and federal laws currently require reclamation of strip and open-pit coal mines, and it is very probable that similar reclamation will be required for open-pit uranium mines. If so, long-term releases from such mines should approach background levels.

For long-term radon releases from stabilized tailings piles, the staff has assumed that the tailings would emit, per RRY, 1 Ci/year for 100 years (covering fully intact), 10 Ci/year for the next 400 years (covering partially failed), and 100 Ci/year for periods beyond 500 years (covering failed). With these assumptions, the cumulative radon-222 release from stabilized tailings piles per RRY would be 100 Ci in 100 years, 4090 Ci in 500 years, and 53,800 Ci in 1000 years (NRC Docket No. 50-488). The total-body, bone, and bronchial-epithelium dose commitments for these periods are as shown in Table 6.4.

It should be noted that there would be global radiological impacts from ^{222}Rn. The number of potential health effects within the U.S. is estimated to be about 90 percent of the total continental health effects. Mexico and Canada would account for the remaining 10 percent of the

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Table 6.3 Population-dose commitments from unreclaimed open-pit mines for each year of operation of the model 1000-MW(e) light-water reactor

Time period (years)	^{222}Rn release (Ci)	Population-dose commitments (man-rem)		
		Total body	Bone	Lung (bronchial epithelium)
100	3,700	96	2,500	2,000
500	19,000	480	13,000	11,000
1,000	37,000	960	25,000	20,000

Table 6.4 Population-dose commitments from stabilized tailings piles for each year of operation of the model 1000-MW(e) light-water reactor

Time period (years)	^{222}Rn release (Ci)	Population-dose commitments (man-rem)		
		Total body	Bone	Lung (bronchial epithelium)
100	100	2.6	68	56
500	4,090	110	2,800	2,300
1,000	53,800	1,400	37,000	30,000

continental health effects. Exposure in Europe and Asia would add about 25 percent more potential health effects to the number of effects predicted for North America (NUREG-0706, p. 6-68).

The staff also considered the potential health effects associated with the release of ^{99}Tc. The release per RRY of ^{99}Tc is 0.007 Ci from chemical reprocessing of recycled UF_6 before it enters the isotope enrichment cascade and 0.005 Ci into the groundwater from a federal repository. The major risks from ^{99}Tc are from exposure of the gastrointestinal tract and kidney, although there is a small risk from total-body exposure. Using organ-specific risk estimators, these individual organ risks can be converted to a total-body 100-year dose commitment of 100 man-rem per RRY. These calculations are based on the gaseous and the hydrological pathway model systems described in Volume 3 of NUREG-0002, *Final Generic Environmental Statement on the Use of Mixed Oxide Fuel in Light Water Cooled Reactors-Health, Safety, and Environment*, Chapter IV, Section J, Appendix A.

The consideration of risks to large populations over long periods of time from exposures to very low concentrations of radionuclides involves many uncertainties. The issue of estimating risks from radon

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and daughters at very low levels continues to be studied. For example, in a June 7, 1995, article in the Journal of the National Cancer Institute (Lubin et al. 1995), the authors reexamined data on miner exposures and described the uncertainties in projecting risks to indoor radon levels. The indoor concentrations are generally about an order of magnitude lower than the miner exposures, but there is some overlap when comparing lifetime exposures to the exposures of the worker cohorts. The authors concluded that much uncertainty still exists in projecting risks at indoor levels from miner data, including the exposures of miners to agents such as arsenic and diesel exhaust, but that reduction of radon levels in homes above the U.S. Environmental Protection Agency's (EPA's) recommended action level of 4 pCi/L "*may* [emphasis added] reduce lung cancer deaths about 2%–4%." Average U.S. indoor levels are about an order of magnitude higher than ambient outdoor levels. Radon releases from tailings for hundreds and thousands of years are undetectable from background levels at a few km, or less than one km in some cases (NRC Docket 50-488 1986). Thus, in the staff's view, projecting risks from levels another order of magnitude or more lower involves even greater uncertainties. However, the linear nonthreshold assumption continues to be used to calculate potential health effects in documents such as this GEIS where the agency is airing the impacts and potential impacts of activities under consideration.

When added to the 500 man-rem total-body dose commitment for the balance of the fuel cycle, the overall estimated total-body involuntary 100-year environmental dose commitment to the U.S. population from the fuel cycle for the model 1000-MW(e) LWR is about 740 man-rem (500 + 140 + 2.6 + 100). Over this period, this dose is equivalent to 0.0008 percent of the natural total-body dose of about 90 million man-rem to the U.S. population. This estimate is based on an annual average natural individual dose commitment of 300 mrem (includes radon) and a stabilized (assumed constant) U.S. population of 300 million.

Using risk estimators of 500, 0.6, and 78 (NUREG/CR-4214, Rev. 1, Part II, Addendum 1, p. 54; Addendum 2, pp. 38 and 49) cancer deaths per million man-rem for total-body, bone, and lung exposures, respectively, the estimated risk of cancer mortality resulting from fuel cycle from emissions of radioactive material is about 0.6 cancer fatality per RRY [$(740 \times 500 + 3668 \times 0.6 + 2956 \times 78) \times 10^{-6}$].

Using the estimates above, the 100-year environmental dose commitment to the U.S. population from the fuel cycle, high-level-waste and spent-fuel disposal excepted, is calculated to be about 14,800 man-rem, or 12 cancer fatalities, for each additional 20-year power reactor operating term. Much of this, especially the contribution of radon releases from mines and tailing piles, consists of tiny doses summed over large populations. This same dose calculation can theoretically be extended to include many tiny doses over additional thousands of years as well as doses outside the United States. The result of such a calculation would be thousands of cancer fatalities from the fuel cycle, but this result assumes that even tiny doses have some statistical adverse health effects that will not ever be mitigated (for example, no cancer cure in the next thousand years), and that these dose projections over thousands of years are meaningful. However, these assumptions are questionable. In particular, science

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cannot rule out the possibility that there will be no cancer fatalities from these tiny doses. For perspective, the doses are very small fractions of regulatory limits, and even smaller fractions of natural background exposure to the same population.

Although collective doses and associated potential health effects are calculated and discussed at various places in this chapter, no conclusion is drawn as to the significance of the collective doses or potential health effects. These collective doses are provided for information purposes.

Uranium fuel cycle facilities must comply with NRC, EPA, other federal and state regulations regarding, among other things, the dose limits to the members of the public. Table 6.5 lists types of facilities, the governing regulatory requirements, and the applicable dose limits for individual members of the public. All licensees must provide reasonable assurance that these dose limits are being met for all unrestricted areas. Since each licensee must ensure that the dose is within the limit and be as low as reasonably achievable (ALARA), the dose to individual members of the public is considered by the staff to be small. More detailed discussions on regulatory limits and compliance are presented below.

In the 1989 National Emission Standards for Hazardous Air Pollutants (NESHAP) rulemaking for radionuclides, discussed elsewhere, EPA examined the uranium fuel-cycle (UFC) licensees (see 54 FR 51668; December 15, 1989) as a separate category. Nonradon emissions from uranium mill tailings, uranium hexafluoride conversion plants, fuel fabrication plants, and power plants (all types of facilities in operation subject to 40 CFR 190 and licensed by NRC at the time) were evaluated and combined. The results of EPA's risk assessment was that:

> the most exposed individual receives a dose associated with an increased risk of fatal cancer of 1.5×10^{-4}. There is a predicted incidence of 0.1 fatal cancer per year in the population, with almost all the population risk received by people with a lifetime risk of less than 1×10^{-6}. Virtually the entire U.S. population lives within 80 km of at least one UFC facility.

EPA found that current emissions were at levels that provided an ample margin of safety but decided to regulate this category to ensure that "the current levels of emissions are not increased." The UFC licensees were included in the licensees subject to Subpart I of 40 CFR 61 and its 10 mrem/year annual dose standard. The UFC licensees other than power reactors have been required to comply with Subpart I since November 1992. Reports to EPA are required if emissions exceed 10% or more of the standard. Based on discussions with EPA staff, NRC understands that no fuel cycle licensees exceeded the standard in 1993 and that the same result is likely for 1994. Note that EPA rescind Subpart I for power reactors on September 5, 1995 (60 FR 46206). EPA evaluated enrichment facilities as part of the Department of Energy category and made similar findings and made them subject to the same 10 mrem/year limit.

The NRC dose limits for individual members of the public are found in 10 CFR 20.1301. The general limit is 100 mrem/year [20.1301(a)(1)]. The risk of fatal cancer to an individual receiving this limit is 5×10^{-5} per year of exposure.

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Table 6.5 Dose limits for most exposed members of the public from uranium fuel cycle facilities

Facility	NRC 10 CFR 20 100 mrem/year and ALARA	EPA (UFC) 40 CFR 190 25/75/25 mrem/year[a]	EPA (CAA) 40 CFR 61 10 mrem/year[a]	Other selected standards
Mines				States, other federal agencies
Surface	No	No	No	
Underground	No	No	Yes	
Milling	Yes	Yes	Yes	10 CFR 40[b]
Mill tailings	Yes[c]	Yes[c]	Yes[c]	10 CFR 40 40 CFR 61, Subpart W[b] 40 CFR 192[b]
UF_6 production	Yes	Yes	Yes	10 CFR 40
Enrichment	Yes	Yes	Yes	10 CFR 76
Fuel fabrication	Yes	Yes	Yes	10 CFR 70
Reactor	Yes	Yes	No	10 CFR 50, Appendix I
Spent-fuel storage	Yes	Yes	Yes	10 CFR 72
Reprocessing	Yes	Yes	Yes	10 CFR 50
High-level waste and spent-fuel disposal	Yes[d]	No	No	10 CFR 60 40 CFR 191
Low-level waste disposal	Yes[d]	No	Yes	10 CFR 61
Transportation	No	No	No	10 CFR 71 DOT-49 CFR

[a]Does not include radon and decay products.
[b]Limiting radon flux to <20 pCi $\cdot$ m^{-2} $\cdot$ s^{-1}, no limit on dose.
[c]Until under general license or closed.
[d]Until closure.

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Other limits are 2 mrem/hour in unrestricted areas [20.1301(a)(2)], case-by-case approvals of up to 500 mrem/year [20.1301(c)], and compliance with EPA's 40 CFR 190, if applicable [20.1301(d)]. Licensees are also required to maintain doses to members of the public ALARA by 10 CFR 20.1101(b). The NRC limits apply to all sources under the control of the licensee and to all pathways combined. The individual dose standards in 40 CFR 190 (25 mrem whole body, 75 mrem thyroid, and 25 mrem to other organs) apply to all pathways of exposure from most fuel-cycle facilities, although doses from radon are excluded. NRC generally implements 40 CFR 190 by means of license conditions and has incorporated it by reference in 10 CFR Part 20. Licensees are required to submit reports every six months on radionuclide emissions under 10 CFR 40.65, 70.59, and 76.35 and explicitly required to report exceeding 40 CFR 190 or violations of implementing license conditions by 10 CFR 20.2203(a)(4).

Procedures are in place for inspecting and ensuring licensee compliance with the regulations and license conditions related to public exposures and environmental protection. For example, Inspection Procedures 83822 ("Radiation Protection"), 88035 ("Radioactive Waste Management"), and 88045 ("Environmental Protection") address inspection and verification of matters such as ALARA for emissions, compliance with procedures and limits in license conditions related to releases and public doses, review of environmental monitoring data, and reporting incidents as required by 10 CFR 20.2203. Fuel cycle facilities are inspected at 6- to 12-month intervals, and compliance with public dose limits and license conditions are normally included in the scope of review. (Note that NRC is still in an observation mode only at the enrichment facilities. NRC certification of compliance with standards is still in process as of September 15, 1995.) NRC inspects uranium recovery facilities, including both mills and in situ operations, annually, and all were inspected in 1995. Inspections include review of effluent and environmental monitoring data and compliance with Part 20 and 40 CFR 190. Inspections have found no evidence that limits are being exceeded. The agreement states of Colorado, Texas, and Washington also license and inspect uranium recovery facilities. They have historically inspected for compliance with 40 CFR 190, and no significant problems have been identified. Most mill tailings are in reclamation. Fuel-cycle licensees are generally found to be in compliance, and many of the fuel-fabrication facilities are operating at small fractions of the limits. Violations found and reported involving the exceeding of public dose limits are serious matters. As a matter of policy, exceeding the public dose limits or license conditions related to the limits is at least a severity level III violation and is subject to escalated enforcement actions. The absence of major enforcement actions related to exceeding public dose limits by fuel-cycle facilities readily discernable from the enforcement data base since 1985 suggests that these facilities are operating within the limits. One mill licensee case involved exceeding 10 CFR 20.106 unrestricted area concentration limits for radon, but no actual overexposure of members of the public occurred. There were no other cases for mill licensees since 1985 readily discernable from the data base that might gave involved overexposure of members of the public.

As noted in the preamble to the final rule revising 10 CFR Part 20 in its entirety (56 FR 23374; May 21, 1991), 40 CFR 190

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limits "apply to the total dose from all sources within the uranium fuel cycle. However, in its practical implementation, the sources would have to be located within a few miles of each other for the combined dose contributions to be significantly different from the dose from either facility alone." Thus, in the unlikely event that facilities should be near each other, each licensee would have to determine that the combined doses do not exceed the limits.

NRC regulatory authority does not include underground or surface/open pit mining. The states and other federal agencies regulate these activities. They are not subject to 40 CFR 190. EPA considered radon emissions from both types of uranium mining in the 1989 Clean Air Act rulemaking. For surface uranium mines, EPA found that the current situation protected the public with an ample margin of safety. Further, EPA noted that

> In addition, this source category is already regulated by a host of state and federal mine reclamation laws. Due to the depressed state of the uranium mining industry, there is no reason to believe that new surface mines will be constructed. The presence of these laws, the very low maximum individual risk and incidence level associated with this category, and the depressed nature of the industry lead EPA to the decision that it is unnecessary for EPA to set a NESHAP for this source category.

For underground mines, EPA found that a NESHAP of 10 mrem/year was necessary and provided an ample margin of safety. As noted elsewhere, no production from either type of uranium mining occurred in 1993 and 1994.

Consideration of EPA's target risk goal for regulatory actions provides perspective and further illuminates the significance of the public doses being estimated and received. The EPA target risk goal is 10^{-4} to 10^{-6} individual lifetime risk. This policy developed over a number of years and is used in many EPA programs, including corrective actions for hazardous waste sites, site cleanups under Superfund, drinking water maximum concentration limits (MCLs) for tap water, and for air emissions under the Clean Air Act. For example, in a 1991 proposed rule to modify the radionuclide MCLs (56 FR 33058; July 18, 1991), EPA stated that "Longstanding and carefully considered EPA policy for regulating carcinogens in drinking water is that the lifetime individual risk target is one in 10,000 (10^{-4}) to one in 1,000,000 (10^{-6}) risk."

The 1989 Clean Air Act (CAA) regulations establishing NESHAPs for radionuclides are based on the target risks. NESHAPs were established for NRC licensees (40 CFR 61, Subpart I) and for radon emissions from operating tailings piles (40 CFR 61, Subpart W) and the disposal of tailings (40 CFR 61, Subpart T). Risks to individuals from high-level waste repository emissions were found to be sufficiently low (less than 1 in 1 million) that no NESHAP was needed. In the final rule (54 FR 51654; December 15; 1989), EPA stated the EPA NESHAPs Policy concerning the risk goals as follows:

> This section provides a description of the EPA's approach for the protection of public health under section 112. In protecting public health with an ample margin of safety under section 112, EPA strives to provide maximum feasible protection against risks to health from hazardous air pollutants by

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(1) protecting the greatest number of persons possible to an individual lifetime risk level no higher than approximately 1 in 1 million and (2) limiting to no higher than approximately 1 in 10 thousand the maximum estimated risk that a person living near a plant would have if he or she were exposed to the emitted pollutant for 70 years. Implementation of these goals is by means of a two-step standard-setting approach, with an analytical first step to determine an "acceptable risk" that considers all health information, including risk estimation uncertainty, and includes a presumptive limit on maximum individual lifetime risk (MIR) of approximately 1 in 10 thousand. A second step follows in which the actual standard is set at a level that provides "an ample margin of safety" in consideration of all health information, including the number of persons at risk levels higher than approximately 1 in 1 million as well as other relevant factors including costs and economic impacts, technological feasibility, and other factors relevant to each particular decision. Applying this approach to the radionuclide source categories in today's notice results in controls that protect over 90 percent of the persons within 80 kilometers (km) of these sources at risk levels no higher than approximately 1 in 1 million.

The 1990 CAA amendments preserved these radionuclide NESHAPS and included a general 10^{-6} lifetime risk threshold for when EPA should consider developing standards or additional requirements on air emissions from sources or removing sources from the list [sections 112(f) and (c)(9), respectively]. The legislative history for the 1990 amendments also included the risk approach as expressed by the quote above as the acceptable basis for EPA not to regulate NRC licensees.

The limits in 40 CFR 190 equate to a maximum individual risk of about 5×10^{-4} per EPA's 1993 rulemaking related to 40 CFR 191 [derived from a lifetime (70-year) individual dose of 15 mrem/year and using risk factor of 500×10^{-6} per rem]. Because ALARA must be applied by licensees, few, if any, individuals would be exposed at this limit. Thus, individual doses expected should fall within EPA's target risk range.

6.2.2.2 Radioactive Wastes

The quantities of buried radioactive waste material (low-level, high-level, and transuranic wastes) associated with the uranium fuel cycle are specified in Table S-3. For low-level waste disposal at land-burial facilities, the Commission notes in Table S-3 that there will be no significant radioactive releases to the environment. The Commission notes that high-level and transuranic wastes are to be buried at a federal repository and that no release to the environment is associated with such disposal, although it has been assumed that all of the gaseous and volatile radionuclides contained in the spent fuel are released to the atmosphere prior to the disposal of the waste. NUREG-0116, which provides background and context for the high-level and transuranic Table S-3 values established by the Commission, indicates that these high-level and transuranic wastes will be buried and will not be released to the biosphere. The generation, storage, and ultimate disposal of low-level waste, mixed waste, and spent fuel from power reactors is addressed in greater detail later in this chapter.

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Waste disposal facilities are not covered by 40 CFR 190, but 10 CFR 60 applies to disposal of high-level waste, and 10 CFR 61 applies or is applied to low-level-waste disposal facilities. The NRC regulations for geologic disposal of high-level radioactive waste in 10 CFR 60 limits the releases of radioactive material to the accessible environment. In addition to satisfying an overall performance objective to be established by EPA, the basic requirements are that containment of high-level waste within the waste packages will be substantially complete for a period between 300 and 1,000 years (to be determined by the Commission) and that the annual releases from the engineered barrier system thereafter should not exceed one part in 100,000 of the total inventory of each radionuclide calculated to be present 1,000 years following permanent closure of the repository. For high-level waste, 10 CFR 60.111 requires compliance with 10 CFR 20 and with EPA general environmental standards in 40 CFR 191.

For the high-level-waste and spent-fuel disposal component of the fuel cycle, there are no current regulatory limits for off-site releases of radionuclides for the candidate repository at Yucca Mountain. If we assume that limits are developed along the lines of the 1995 National Academy of Sciences (NAS) report, *Technical Bases for Yucca Mountain Standards*, and that in accordance with the Commission's Waste Confidence Decision, 10 CFR 51.23, a repository can and likely will be developed at some site that will comply with such limits, peak doses to virtually all individuals will be 100 mrem/year or less. While the NRC has reasonable confidence that these assumptions will prove correct, there is considerable uncertainty because the limits are yet to be developed, no repository application has been completed or reviewed, and uncertainty is inherent in the models used to evaluate possible pathways to the human environment. The National Academy report indicated that 100 mrem/year should be considered as a starting point for limits for individual doses, but notes that some measure of consensus exists among national and international bodies that the limits should be a fraction of the 100 mrem/year. The lifetime individual risk from 100 mrem/year dose limit is about 3×10^{-3}.

Estimating cumulative doses to populations over thousands of years is more problematic. The likelihood and consequences of events that could seriously compromise the integrity of a deep geologic repository were evaluated by the Department of Energy in the *Final Environmental Impact Statement: Management of Commercially Generated Radioactive Waste*, October 1980. The evaluation estimated the 70-year whole-body dose commitment to the maximum individual and to the regional population resulting from several modes of breaching a reference repository in the year of closure, after 1,000 years, after 100,000 years, and after 100,000,000 years. The release scenarios covered a wide range of consequences from the limited consequences of humans accidentally drilling into a waste package in the repository to the catastrophic release of the repository inventory by a direct meteor strike. Subsequently, the NRC and other federal agencies have expended considerable effort to develop models for the design and for the licensing of a high-level-waste repository, especially for the candidate repository at Yucca Mountain. More meaningful estimates of doses to population may be possible in the future as more is understood about the performance of the proposed Yucca Mountain

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repository. Such estimates would involve very great uncertainty, especially with respect to cumulative population doses over thousands of years. The standard proposed by the NAS is a limit on maximum individual dose. The relationship of potential new regulatory requirements, based on the NAS report, and cumulative population impacts has not been determined, although the report articulates the view that protection of individuals will adequately protect the population for a repository at Yucca Mountain. However, EPA's generic repository standards in 40 CFR 191 generally provide an indication of the order of magnitude of cumulative risk to population that could result from the licensing of a Yucca Mountain repository, assuming the ultimate standards will be within the range of standards now under consideration. The standards in 40 CFR 191 protect the population by imposing "containment requirements" that limit the cumulative amount of radioactive material released over 10,000 years. The cumulative release limits are based on EPA's population impact goal of 1,000 premature cancer deaths worldwide for a 100,000-metric tonne (MTHM) repository.

6.2.2.3 Occupational Dose

The annual occupational dose attributable to all phases of the fuel cycle for the model 1000-MW(e) LWR is about 600 man-rem.

6.2.2.4 Transportation

The transportation dose to workers and the public totals about 25 man-rem/RRY.

6.2.2.5 Fuel Cycle

The NRC staff analysis of the uranium fuel cycle did not depend on the selected fuel cycle (no recycle or uranium-only recycle) because the data provided in Table S-3 include maximum recycle-option impact for each element of the fuel cycle, and therefore the environmental impacts of the fuel cycle are not affected by the specific fuel cycle selected.

6.2.2.6 Land Use Impacts

The total annual land requirement for the fuel cycle supporting a model 1000-MW(e) LWR is about 46 ha (113 acres). About 5.3 ha (13 acres) are permanently committed, and 41 ha (100 acres) are temporarily committed. (A "temporary" land commitment is a commitment for the life of the specific fuel-cycle plant; e.g., mill, enrichment plant, or succeeding plants. On abandonment or decommissioning, such land can be used for any purpose. "Permanent" commitments represent land that may not be released for use after permanent storage, plant shutdown, and/or decommissioning.) Of the 41 ha per year of temporarily committed land, 32 ha (79 acres) are undisturbed and 9 ha (22 acres) are disturbed. Considering common classes of land use in the United States, fuel-cycle land-use requirements to support the model 1000-MW(e) LWR do not represent a significant impact. As a comparison, a coal-fired power plant of 1000-MW(e) capacity using strip-mined coal requires the disturbance of about 81 ha (200 acres) per year for fuel alone.

6.2.2.7 Water Use Impacts

The principal water-use requirement for the fuel cycle supporting a model 1000-

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MW(e) LWR is that required to remove waste heat from the power stations supplying electrical energy to the enrichment step of this cycle. Of the total annual requirement of 43×10^6 m^3 (11.4×10^9 gal), about 42×10^6 m^3 (11.1×10^9 gal) are required for this purpose, assuming that these plants use once-through cooling. Other water uses involve the discharge to air (e.g., evaporation losses in process cooling) of about 0.6×10^6 m^3 (160×10^6 gal) per year and water discharged to ground (e.g., mine drainage) of about 0.5×10^6 m^3 (130×10^6 gal) per year.

On a thermal-effluent basis, annual discharges from the nuclear fuel cycle are about 4 percent of those from the model 1000-MW(e) LWR using once-through cooling. The consumptive water use of 0.6×10^6 m^3/year is about 2 percent of that from the model 1000-MW(e) LWR using cooling towers. The maximum consumptive water use (assuming that all plants supplying electrical energy to the nuclear fuel cycle used cooling towers) would be about 6 percent of that of the model 1000-MW(e) LWR using cooling towers. Under this condition, thermal effluents would be negligible. The staff finds that these combinations of thermal loadings and water consumption are acceptable relative to the water use and thermal discharges.

6.2.2.8 Fossil Fuel Impacts

Electrical energy and process heat are required during various phases of the fuel-cycle process. The electrical energy is usually produced by the combustion of fossil fuel at conventional power plants. Electrical energy associated with the fuel cycle represents about 5 percent of the annual electrical power production of the model 1000-MW(e) LWR. Process heat is generated primarily by the combustion of natural gas. This gas consumption, if used to generate electricity, would be less than 0.4 percent of the electrical output from the model plant. The staff finds that the direct and indirect consumptions of electrical energy for fuel-cycle operations are small and acceptable.

6.2.2.9 Chemical Effluents

The quantities of chemical, gaseous, and particulate effluents associated with fuel-cycle processes are given in Table S-3. The principal species are sulfur oxides, nitrogen oxides, and particulates. Judging from data in a Council on Environmental Quality report (seventh annual report), these emissions constitute an extremely small additional atmospheric loading in comparison with these emissions from the stationary fuel-combustion and transportation sectors in the United States (i.e., about 0.02 percent of the annual national releases for each of these species). These emissions can also be compared with those from coal-fired generation of electricity. As an example, one paper reported that in comparison with a coal-fired power plant of the same size with abatement system, a 1300-MW(e) nuclear power plant eliminates annually emission to the air of about 2,000 tons of particulates, 8.5 million tons of CO_2, 12,000 tons of SO_x, and 6,000 tons of NO_x. (Souza and Bennett 1989). The staff believes that such small increases in releases from the nuclear fuel cycle of these pollutants are acceptable.

Impacts from the chemical and physical properties of the materials handled by fuel-cycle licensees can also occur. For example, on January 4, 1986, an overfilled cylinder containing UF_6 ruptured while it was being heated in a steam chest at the Sequoyah

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Fuels Conversion Facility near Gore, Oklahoma (NUREG-1179). One worker died because he inhaled hydrogen fluoride fumes, a reaction product of UF_6 and airborne moisture. Several other workers were injured, but none seriously, and there was on-site and off-site contamination with hydrogen fluoride and uranyl fluoride, a second reaction product.

Liquid chemical effluents produced in fuel-cycle processes are related to fuel enrichment, fabrication, and reprocessing operations and may be released to receiving waters. These effluents are usually present in dilute concentrations such that only small amounts of dilution water are required to reach levels of concentration that are within established standards. The flow of dilution water required for specific constituents is specified in Table S-3. Additionally, all liquid discharges into the navigable waters of the United States from plants associated with the fuel-cycle operations will be subject to requirements and limitations set forth in the NPDES permit. Tailings solutions and solids are generated during the milling process. Based on Table S-3, these solutions and solids are not released in quantities sufficient to have a significant impact on the environment.

6.2.3 Sensitivity to Recent Changes in the Fuel Cycle

The values given in Tables S-3 (10 CFR 51.51) and S-4 (10 CFR 51.52) were calculated from industry averages for the performance of each type of facility or operation within the fuel cycle. Recognizing that this approach meant that there would be a range of reasonable values for each estimate, the staff followed the policy of choosing the assumptions or picking the factors to be applied so that the calculated values would not be underestimated. This approach was intended to ensure that the actual environmental impacts would be less than the quantities shown in Tables S-3 and S-4 for all nuclear power plants within the widest range of operating conditions.

This discussion on the sensitivity of the estimates to changes in assumptions or factors used by the staff in making the environmental impact analyses is provided below in examples to show the degree of conservatism used in developing estimates and thus to give an indication of the uncertainty of the estimates when they are applied to a particular nuclear power plant or to the plant's operations within the applicable regulations. The methodology was deliberately constructed to estimate impacts closer to the upper bound than to the mathematical average or median. Considering this approach, one can judge that the level of precision in the estimates is about 10% at best, probably no more than single-significant-digit accuracy in most cases. For this reason, and to simplify the presentation, many subtle fuel-cycle parameters and interactions were recognized by the staff as being less than the precision of the estimates and were ignored or mentioned briefly to show that they were considered but had no effect on the Table S-3 and S-4 calculations. The following example shows the conservatism of Tables S-3 and S-4 with respect to impacts on the environment.

To determine the quantity of fuel required for a year's operation of a nuclear power plant, the staff defined the model reactor as a 1000-MW(e) light-water-cooled reactor operating at 80 percent capacity with a 12-month fuel reloading cycle and an average fuel burnup of 33,000 MWd/MTU. This is a reactor

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reference year (RRY). The sum of the initial fuel loading plus all of the reloads for the lifetime of the reactor can be divided by the now assumed 60-year (40-year initial license term and 20-year renewal license term) lifetime to obtain an average annual fuel requirement. This was done for both boiling-water reactors (BWRs) and pressurized-water reactors (PWRs), and the higher annual requirement, 35 metric tonnes (MT) of uranium made into fuel for a BWR, was chosen as the basis for the RRY. Since the original estimates in 1979 were made for Table S-3, a number of fuel management improvements have been adopted by nuclear power plants to achieve higher performance and to reduce fuel and separative work (enrichment) requirements. These improvements reduce the annual fuel requirement by 10 to 15 percent. Further, the average plant capacity factor achieved by reactors operating in the United States has been below the assumed 80 percent capacity factor in every reporting period to date, meaning that the consumption of fuel has been below estimated amounts. Some more recent studies have assumed average capacity factors of 70 to 75 percent, indicating a reduction of 6 to 12 percent in annual fuel consumption.

Today's once-through fuel cycle could be expected to require 15 to 20% more uranium from mining and milling to compensate for no recovery and recycle of uranium from spent fuel. However, this increase in requirements is assumed to be offset by the decreases from improved fuel management and the lower average operating capacity factor; and the average fuel requirement for 1 RRY is still estimated to be 182 MT of U_3O_8 (35 MTU), as it was in WASH-1248. However, there has been another change of even greater significance in the elimination of U.S. restrictions on importation of foreign uranium. The economic conditions of the uranium market now and in the foreseeable future favor full utilization of foreign uranium at the expense of the domestic uranium industry. These market conditions have forced the closing of most U.S. uranium mines and mills, substantially reducing the environmental impacts in the U.S. from these activities. However, the Table S-3 estimates have not been reduced accordingly to ensure that these impacts, which have been experienced in the past and may be fully experienced again in the future, are considered. This fact suggests that the environmental impacts of mining and milling could drop to levels far below those given in Table S-3.

In a somewhat similar situation, the Table S-3 estimates for enrichment are based on the gaseous diffusion process, which has been used in the United States since the earliest days of the nuclear power program. In this process, there can be significant changes in uranium feed requirements as a result of changes in the quantity of ^{235}U left in the process tails. The range of tails assay is generally from 0.16 to 0.30 wt-percent ^{235}U, and the value assumed in making Table S-3 estimates is 0.25 percent. If the value of 0.16 percent had been chosen, 16 percent less uranium feed would be required, and environmental impacts (except those associated with enrichment) would be correspondingly lower. At 0.30 percent tails, uranium requirements would increase by 11 percent and environmental impacts would be higher. Far greater potential changes would come from the use of enrichment services from overseas or from the use of centrifuge technology for enrichment in the United States. The largest impacts of

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the gaseous diffusion process are attributable to the large requirement for electric energy to run the plant (especially to the assumption that the electricity will come from coal-fired power plants) and to the large amount of cooling water used in the gaseous diffusion process equipment. The centrifuge process uses 90 percent less electrical energy and therefore would have far lower impacts attributable to coal-fired power plants and the use of cooling water. Clearly, when overseas enrichment services are utilized, domestic impacts from U.S. enrichment plants would drop nearly to zero. These potential reductions are not reflected in Table S-3 estimates. Because there are currently no centrifuge enrichment plants in the United States, this potential reduction is not reflected. However, there is an application pending with the NRC to construct such a plant in the U.S. The assumption of continued use of United States diffusion enrichment services ensures that environmental impacts are not underestimated.

It may be noted that the recycling of uranium in spent fuel would have only minor effects on enrichment because the recycled uranium has about the same ^{235}U assay as fresh natural uranium and would thus require about the same amount of enrichment. There is an increase in the concentration of the ^{236}U isotope in recycled uranium. This acts as a "poison" in the nuclear fuel, requiring more ^{235}U to overcome it. Each kilogram of ^{236}U that is present in the recycled fuel requires an additional 0.3 kg of ^{235}U to compensate for it. In total, the few kilograms of ^{236}U in the fuel cause increases of about 2 to 4 percent in the enrichment impacts.

There is only one U.S. plant currently converting uranium oxide product from the mills to UF_6 feed for the enrichment plant. The UF_6 conversion plant uses a "dry" process using gaseous reagents. Formerly, a "wet" process that starts with dissolving the yellow cake in nitric acid and purifying it by solvent extraction was also used. In the "dry" process, final purification is accomplished by fractional distillation of the UF_6; impurities are eliminated as volatile compounds or as solid wastes. In the "wet" process, many impurities are eliminated in the aqueous phase from solvent extraction. In both cases, environmental releases are so small that changing from 100 percent use of one process to 100 percent use of the other would make no significant difference in the totals given in Table S-3 or S-4. The assumption that half is processed by each method does not contribute significantly to the error band of the totals.

In the fuel fabrication plants, it has been assumed that the UF_6 from enrichment will be converted to UO_2 by the ammonium diuranate "wet" process. An alternative "dry" process for direct conversion of UF_6 to UO_2 powder is being introduced as obsolete facilities are replaced or as new capacity is added. This change reduces environmental impacts, but the impacts from fuel fabrication are so small that the changes are not significant.

Factors related to reactor operation can have a significant effect on the fuel cycle. The original Tables S-3 and S-4 were based on a 12-month fuel reloading cycle. Current practice favors an 18-month cycle although in certain circumstances, the original 12-month cycle or a longer 24-month cycle might be favored. Parametric studies show that producing the higher enrichment fuel needed for higher burnup requires an increase of about 5 percent in the natural uranium feed stream for each 6-month extension of the

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reload interval. Similarly, enrichment impacts increase by about 5 percent with each 6-month extension of the reload cycle. However, the higher burnup of fuel achieved in the longer reload cycles reduces the average annual output of spent fuel by as much as 45 percent.

The values shown in Tables S-3 and S-4 of 10 CFR Part 51 are conservative estimates originally developed on the basis of an average fuel irradiation (burnup) of 33,000 MWd/MTU. Discussions and analyses in NUREG/CR-5009 (PNL-6258), *Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors*, February 1988, show that the burnup level of fuel up to 60,000 MWd/MTU will not result in environmental impacts that are greater than the values currently in Tables S-3 and S-4, and, in many instances, are less (for example, see Table S.1 on p. viii of NUREG/CR-5009). Thus no revision to these tables would be required as a result of extended fuel burnup up to 60,000 MWd/MTU. Experience in handling fuel with burnups over 55,000 MWd/MTU and up to 5.5 percent ^{235}U enrichment has not revealed any unresolved safety concerns (NUREG/CR-5009 p. 1-7).

The reduction in the annual output of spent fuel at high burnup would correspondingly reduce the environmental impacts associated with transportation of spent fuel, with reprocessing, and with waste disposal on or off site. There would also be a decrease in occupational exposure to radiation because of the reduction in processing and handling requirements. Population radiation doses would be lower because of the reduced number of shipments per year.

There are other significant changes that would apply to reprocessing if fuel recycle were to be undertaken in the United States in the future. Estimates for reprocessing impacts were based on the Barnwell and Exxon reprocessing plant designs of the 1970s. The radioisotope release fractions used in the 1976 report (NUREG-0116) are now considered to be conservative by at least two orders of magnitude in comparison to current design values. Also, the original Table S-3 assumption that 100 percent of the volatile radioisotopes and compounds would be released is no longer valid. EPA regulations in 40 CFR Part 190 require that, after 1983, releases of ^{85}Kr and ^{129}I be limited to 50,000 Ci/GW-year and 5 mCi/GW-year, respectively. Because the model reactor that is the basis for Tables S-3 and S-4 values produces 0.8 GW-years of electricity, the EPA limits translate to 40,000 Ci/RRY and 4 mCi/RRY, respectively. Because plants will not be permitted to operate in violation of the EPA requirements, the current Table S-3 values are even more conservative, taking into account compliance with the new EPA requirements. A further EPA requirement is that releases of alpha-emitting transuranic elements with half-lives longer than 1 year must be limited to 0.5 mCi/GW-year, or 0.4 mCi/RRY. This limit for transuranic elements required no change in the Table S-3 estimate, which was already well below the new standard.

Another conservatism in the NUREG-0116 estimates for Table S-3 is an assumption of a cooling time of 160 d between the discharge of spent fuel from the reactor and the reprocessing of the fuel. This 160-d cooling period was based on the optimum for recycling plutonium as well as uranium. With the recycling of uranium only or with the present once-through mode of operation, there is no incentive to

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keep the cooling time short, and, indeed, virtually all spent fuel in storage today has been cooling for years. In comparison to 160-d-old spent fuel, fuel that has been cooled 1 year or more would have its radioactivity reduced by at least 50 percent and its radioactive decay heat emission similarly reduced. The effect of cooling for 5 years or more on site, the age range of most spent fuel today, is to reduce the radioactivity and the decay heat by more than 90 percent; therefore certain radioisotope inventory may be as low as 10 percent of the amount shown in Table S-3, in which case dose commitments and potential health effects calculated for Table S-3 releases would be overestimated.

One effect of going to higher fuel burnup is to increase the formation of transuranic elements, with the result that spontaneous neutron emission from transuranic elements becomes an important shielding consideration along with shielding for the gamma radiation. This fact has potential effects on the transportation of spent fuel. At the time of discharge from the reactor, the radioactivity and decay heat of high-burnup fuel may be up to 25 percent higher at 60,000 MWd/MTU, but this increase diminishes as the cooling time is lengthened. The emission of neutrons also decreases with longer cooling. Gamma radiation is shielded with lead or other dense materials, while neutrons are best shielded by water and neutron-absorbing materials such as boron or cadmium. It has been shown that present spent fuel transportation casks can be made safe for high-burnup fuel by adding boron to the cooling water in the casks. Longer cooling times would increase the margin of safety. With the large inventory of spent fuel that has accumulated, the age of any spent fuel that is reprocessed or transported to a repository is likely to be many years. At the conclusion of the hearings on reprocessing and waste management (Dockets 50-277, 50-278, 50-320, 50-354, and 50-355, Consolidated Hearing on Radon Before the Appeal Board), the Hearing Board concluded that 5 years would be a reasonable value to use in making estimates. The scenario that is visualized today for emplacement of spent fuel and high-level waste in a geologic repository calls for this final disposal to occur after the spent fuel or waste is at least 10 or more years old. Longer cooling times on site reduce the impact on the environment and increase the margin of safety once the fuel is being transported.

The NRC regulations for geologic disposal of high-level radioactive waste (10 CFR Part 60) limit the releases of radioactive material to the accessible environment. In addition to satisfying an overall performance objective to be established by the EPA, the basic requirements are that there should be no leakage from the waste packages in the first 300 to 1,000 years and that the annual releases from the engineered barrier system thereafter should not exceed one part in 100,000 of the total inventory of each radionuclide calculated to be present 1,000 years following permanent closure of the repository. These values are conservative because no credit is taken for chemical or physical retardation and for additional decay of radionuclides during transport through groundwater within the controlled zone prior to release to the accessible environment. In summary, the discussion above shows that the Table S-3 estimates of environmental impacts are higher than the actual impacts would be under any foreseeable combination of reactor and fuel cycle operating conditions, including higher fuel burnup, and any future license renewal activities. One of the greatest changes

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would come from the use of foreign uranium and foreign enrichment services, which could easily reduce U.S. environmental impacts from the front end of the fuel cycle by factors of 10 to 100. Significant uncertainties are also associated with the estimates of environmental releases from high-level waste handling, storage, and disposal. Lacking knowledge of the actual operation of the facilities that will be licensed or relicensed in the future, the staff has estimated that releases will be at the maximum levels permitted by NRC and EPA regulations for releases from the engineered barriers to repository groundwater, whereas some engineering tests indicate that it may be possible to keep releases to much lower levels. The Table S-3 estimates could easily be high under any regulated change in operating, handling or storage conditions of high-level waste; they are not likely to be low.

6.2.4 Conclusions

The radiological and nonradiological environmental impacts of the uranium fuel cycle have been reviewed. The review included a discussion of the values presented in Table S-3, an assessment of the release and impact of ^{222}Rn and of ^{99}Tc, and a review of the regulatory standards and experience of fuel cycle facilities. Although the radiological release values presented in Table S-3 and in the discussion of ^{222}Rn and ^{99}Tc were intended to be within regulatory limits (except for ^{85}Kr and ^{129}I), not attempt was made to use realistic assumptions that would reflect the success of ALARA programs and the history of releases and doses generally being well under regulatory requirements. For assessing the radiological impacts of license renewal, the Commission has indicated that impacts are of small significance if doses and releases do not exceed permissible levels in the Commission's regulations.

The radiological impacts of the uranium fuel cycle on individuals off site have been considered within the framework of Table S-3 and supplemental analyses of ^{222}Rn and ^{99}Tc. Given the available information applicable regulatory requirements, the Commission has concluded that, other than for the disposal of spent fuel and high-level waste, these impacts on individuals from radioactive gaseous and liquid releases will remain at or below the Commission's regulatory limits. Accordingly, the Commission concludes that off-site radiological impacts of the fuel cycle (individual effects from other than the disposal of spent fuel and high-level waste) are small. ALARA efforts will continue to apply to fuel-cycle activities. This is a Category 1 issue.

The radiological impacts of the uranium fuel cycle on human populations over time (collective effects) have been considered within the framework of Table S-3. The 100-year environmental dose commitment to the U.S. population from the fuel cycle, except for high-level waste and spent-fuel disposal, is calculated to be about 14,800 person-rem, or 12 cancer fatalities, for each additional 20-year power reactor operating term. Much of this, especially the contribution of radon releases from mines and tailing piles, consists of tiny doses summed over large populations. This same dose calculation can theoretically be extended to include many tiny doses over additional thousands of years as well as doses outside the U.S. The result of such a calculation would be thousands of cancer fatalities from the fuel cycle, but this result assumes that even tiny doses have some statistical adverse health effect that will not ever be mitigated (for example no cancer

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cure in the next thousand years) and that these dose projection over thousands of years are meaningful. However, these assumptions are questionable. In particular, science cannot rule out the possibility that there will be no cancer fatalities from these tiny doses. For perspective, the doses are very small fractions of regulatory limits, and even smaller fractions of natural background exposure to the same populations. No standards exist that can be used to reach a conclusion as to the significance of the magnitude of the collective radiological effects. Nevertheless, some judgment as to the regulatory NEPA implication of this issue should be made, and it makes no sense to repeat the same judgment in every case. The Commission concludes that these impacts are acceptable in that these impacts would not be sufficiently large to require the NEPA conclusion, for any plant, that the option of extended operation under 10 CFR 54 should be eliminated. Accordingly, while the Commission has not assigned a single level of significance for the collective effects of the fuel cycle, this issue is considered Category 1.

The impacts associated with the high-level-waste and spent-fuel disposal component of the fuel cycle also involve a level of uncertainty. There are no current regulatory limits for off-site releases of radionuclides for the current candidate repository site. However, if we assume that limits are developed along the lines of the 1995 National Academy of Sciences report and that, in accordance with the Commission's Waste Confidence Decision, a repository can and likely will be developed at some site that will comply with such limits, peak doses to virtually all individuals will be 100 mrem/year or less. However, while the Commission has reasonable confidence that these assumptions will prove correct, there is considerable uncertainty because the limits are yet to be developed and no repository application has been completed or reviewed. In addition, uncertainty is inherent in the models used to evaluate possible pathways to the human environment. The National Academy report indicated that 100 mrem/year should be considered as a starting point for limits for individual doses, but notes that some measure of consensus exists among national and international bodies that the limits should be a fraction of the 100 mrem/year. The lifetime individual risk from 100 mrem/year dose limit is about 3×10^{-3}. Doses to populations from disposal cannot now (or possibly ever) be estimated without very great uncertainty.

Estimating cumulative doses for high-level-waste and spent-fuel disposal to populations over thousands of years is more problematic. The likelihood and consequences of events that could seriously compromise the integrity of a deep geologic repository were evaluated by the Department of Energy in the *Final Environmental Impact Statement: Management of Commercially Generated Radioactive Waste*, October 1980. The evaluation estimated the 70-year whole-body dose commitment to the maximum individual and to the regional population resulting from several modes of breaching a reference repository in the year of closure, after 1,000 years, after 100,000 years, and after 100,000,000 years. The release scenarios covered a wide range of consequences from the limited consequence of humans accidentally drilling into a waste package in the repository to the catastrophic release of the repository inventory by a direct meteor strike. Subsequently, the NRC and other federal agencies have expended

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considerable effort to develop models for the design and for the licensing of a high-level-waste repository, especially for the candidate repository at Yucca Mountain. More meaningful estimates of doses to population may be possible in the future as more is understood about the performance of the proposed Yucca Mountain repository. Such estimates would involve very great uncertainty, especially with respect to cumulative population doses over thousand of years. The standard proposed by the NAS is a limit on maximum individual dose. The relationship of potential new regulatory requirements, based on the NAS report, and cumulative population impacts has not been determined, although the report articulates the view that protection of individuals will adequately protect the population for a repository at Yucca Mountain. However, EPA's generic repository standards in 40 CFR 191 generally provide an indication of the order of magnitude of cumulative risk to population that could result from the licensing of a Yucca Mountain repository, assuming the ultimate standards will be within the range of standards now under consideration. The standards in 40 CFR 191 provide for the protection of the population by imposing "containment requirements" that limit the cumulative amount of radioactive material released over 10,000 years. The cumulative release limits are based on EPA's population impact goal of 1,000 premature cancer deaths worldwide for a 100,000-metric tonne (MTHM) repository.

Despite all the uncertainty surrounding the effects of the disposal of spent fuel and high-level waste, some judgment as to the regulatory NEPA implications of these matters should be made, and it makes no sense to repeat the same judgment in every case. Even taking the uncertainties into account, the Commission concludes that these impacts are acceptable in that these impacts would not be sufficiently large to require the NEPA conclusion, for any plant, that the option of extended operation under 10 CFR 54 should be eliminated. Accordingly, while the Commission has not assigned a single level of significance for the impacts of spent-fuel and high-level-waste disposal, this issue is considered Category 1.

Data on the nonradiological impacts of the fuel cycle are provided in Table S-3. These data have not been adjusted for the permanent disposal of high-level waste and spent fuel at the candidate site at Yucca Mountain. However, any changes in the data specific to Yucca Mountain would be minor. The nonradiological environmental impacts from any resource use or effluent datum specified in Table S-3 are assumed to be proportional to the magnitude of the datum. The standard of significance of the environmental impacts associated with Table S-3 is based on one of several relative comparisons.

Land requirements are compared to those for a coal-fired power plant that may be assumed to replace the nuclear capacity if the operating license is not renewed. The LWR fuel cycle requires only 10 percent of the temporarily committed land and 9.5 percent of the permanently committed land that would be required by replacement with coal-fired capacity. A conclusion on the relative land use impact between nuclear and coal should take into consideration differences in the quality and the opportunity cost of the land involved in each fuel cycle. If the quality and opportunity cost of the land were equivalent, then it would be reasonable to say that land requirements for the uranium fuel cycle (at 20 to 30 percent of those for

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the coal fuel cycle) are relatively small. If much of the land involved in the uranium fuel cycle is of lower quality and has a lower opportunity cost than does the land used in the coal fuel cycle, then the land has relatively lower value, and land requirements could be considered small for relative requirements beyond 30 percent. If these postulates are accepted, then the land requirements of about 10 percent given in Table S-3 are clearly small.

Water requirements for the uranium fuel cycle are compared to the annual requirements for an LWR. The amount of water withdrawn from surface and ground water and discharged to air by activities within the fuel cycle represents only 2 percent of the annual discharges to air of an LWR with cooling towers. The fuel cycle discharges are spread among facilities involved in the various stages of the fuel cycle; thus the water discharge to air from any one of these facilities will be less than the 2 percent. The environmental impacts of water withdrawal, use, and discharge from LWRs with cooling towers is reviewed in Chapter 3, and these discharges are found to have only small, or in special but unusual circumstances moderate, environmental impacts. Given that the water discharged to the air from other fuel cycle facilities for a RRY is only a small fraction of the discharge from an LWR, the environmental consequences will be even smaller. The amount of water withdrawn from surface and ground water and discharged to water bodies and to the ground represents only 4 percent of the annual discharges to water bodies and the ground of an LWR with once-through cooling. The fuel cycle discharges are spread among facilities involved in the various stages of the fuel cycle; thus the water discharges from any one of these facilities will be less than the 2 percent. The environmental impacts of water withdrawal and discharge from LWRs with once-through cooling is reviewed in Chapter 3, and these discharges are found to have small environmental impacts. Given that the water discharged to water bodies and to the ground from other fuel cycle facilities for an RRY is only a small fraction of the discharge from an LWR, the environmental consequences will be even smaller.

The fossil fuel (coal and natural gas) consumed to produce electrical energy and process heat during the various phases of the uranium fuel cycle results in a considerable net saving in the use of resources and chemical effluents over the use that would occur if the electrical output from the LWR were supplied by a coal-fired plant. The use of coal and natural gas in the uranium fuel cycle allows the production of electricity with nuclear fuel, which results in a substantial reduction in the requirements for coal and natural gas as fuels to produce electricity. Not only are the fossil fuel requirements small per RRY; there is a net saving in the use of fossil fuel compared to replacing the nuclear-generating capacity with coal-fired capacity.

The gaseous effluents SO_x, NO_x, hydrocarbons, CO, and particulates listed in Table S-3 are the consequence of the coal-fired electrical energy used in the uranium fuel cycle. The volume of effluent is equivalent to that of a quite small [45-MW(e)] coal-fired plant; thus the contribution to the degradation of air quality is small. The generation of electricity with nuclear rather than coal-fired power will result in a net improvement in air quality. For these reasons the impact of these effluents is considered small. Gaseous releases of

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fluorine and hydrogen chloride are at concentrations below state standards and below levels that impact human health. The impact of these effluents is small.

The liquid effluents listed in Table S-3 are present in dilute concentrations and are readily dilutable to meet effluent standards such that environmental impacts are negligible. The impacts from these liquid effluents are considered small.

Tailings solutions and solids generated during the milling process are not released in quantities sufficient to have a significant impact on the environment. Their impact on the environment is considered small.

The aggregate nonradiological impact of the uranium fuel cycle resulting from the renewal of an operating license for any plant is found to be small. License renewal of an individual plant is so indirectly connected to the operation of fuel-cycle facilities that it is meaningless to address the mitigation of the impacts identified above. This is a Category 1 issue.

6.3 TRANSPORTATION

6.3.1 Introduction

This section addresses both the radiological and nonradiological environmental impacts resulting from shipments of low-level radioactive waste (LLW) and mixed waste to off-site disposal facilities and of spent fuel to a monitored retrievable storage (MRS) or permanent repository. The nonradiological impacts are traffic density, weight of the loaded truck or railcar, heat from the fuel cask, and transportation accidents. The radiological impacts include possible exposures of transport workers and the general public along transportation routes. Radiation exposure to these groups also may occur through accidents along transportation corridors. Generic values for the environmental effects of transporting fuel and waste to and from reactors are provided in Table S-4.

Table S-4 "Environmental impact of transportation of fuel and waste to and from one light-water-cooled nuclear power reactor," is a summary impact statement concerning transportation of fuel and radioactive wastes to and from a reactor. The table is divided into two categories of environmental considerations: (1) normal conditions of transport and (2) accidents in transport. The normal conditions of transport consideration are further divided into environmental impact, exposed population, and range of doses to exposed individuals per reactor reference year. The "accidents in transport" consideration is concerned with environmental risk. Under "normal conditions of transport," the environmental impacts of the heat of the fuel cask in transit, weight, and traffic density are described. Also the number and range of radioactive doses of the transportation workers and of the general public are described. Under "accidents in transport," the environmental risk of radiological effect and common nonradiological causes such as fatal and nonfatal injuries and property damage are described. To indicate that Table S-4 adequately describes the environmental effects of the transportation of fuel and waste to and from the reactor, the reactor licensee must state that the reactor and this transportation either meet all of the conditions in paragraph (a) of 10 CFR 51.52 or all of the conditions in paragraph (b) of 10 CFR 51.52. Subparagraphs 10 CFR 51.52(a)(1) through (5) delineate specific conditions the reactor must meet to use Table S-4 as part of its

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environmental report. Subparagraph 10 CFR 51.52(a)(6) states, "The environmental impacts of transportation of fuel and waste to and from the reactor, with respect to normal conditions of transport and possible accidents in transport, are as set forth in Summary Table S-4 in paragraph (c) of this section; and the values in the table represent the contribution of the transportation to the environmental costs of licensing the reactor." Paragraph 10 CFR 51.52(b) states that reactors not meeting the conditions of 10 CFR 51.52(a) shall make a full description and detailed analysis for their reactor equivalent to Table S-4. Because the reactor must continuously meet the conditions of 10 CFR 51.52(a) to use Table S-4 in its environmental statement or make a full description and detailed analysis for their reactor equivalent to Table S-4, the conditions requiring the use of Table S-4 or its equivalent as an environmental transportation statement are continuously reviewed and therefore will be the same in the relicensing period as in the initial licensing period. Rail and truck transport corridors should safely accommodate increased shipments of radioactive waste associated with license renewal. The radiological and nonradiological environmental impacts of transportation of fuel and waste from an LWR are shown to be small.

To address the impacts from transportation of LLW associated with license renewal, the additional volumes of LLW from refurbishments related to license renewal, as well as from an additional 20 years of normal operations, have been compared with baseline information on current volumes of LLW that are shipped from nuclear power plants to licensed disposal facilities. In the case of mixed waste that is currently stored on site, the amount of waste that is likely to require off-site disposal when an off-site disposal facility becomes available is considered. Finally, in the case of spent fuel from nuclear power plants, estimates have been made of the additional amount of spent fuel that will be generated, assuming that all currently operating reactors continue to operate and store-spent fuel on site until a spent-fuel repository (or MRS) is made available, which probably will not occur before 2010 (DOE/RW-0006).

6.3.2 Table S-4—Environmental Impact of Transportation of Fuel and Waste to and from One Light-Water-Cooled Nuclear Power Reactor

6.3.2.1 Low-Level Waste

The volume of LLW disposed of annually (at licensed disposal facilities) from nuclear power plants varies by type of reactor. In 1987, the average PWR disposed of approximately 8,800 ft^3/year (250 m^3/year), while the average BWR disposed of about 19,700 ft^3/year (558 m^3/year). These volumes were disposed of through approximately 35 annual shipments from PWRs and 59 annual shipments from BWRs (EPRI NP-5983).

The majority of routine LLW transported from plants is made up of Class A wastes such as contaminated trash and other compacted material packaged in 55-gal drums. Normally, these drums contain less than 20 Ci of radioactivity. A small percentage contain more than 100 Ci and are shipped as Class B waste. Accident data have been compiled since 1971 involving commercial LLW. During this time, only four transportation accidents for all categories of shippers have involved the release of commercial LLW. None of these

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accidents involved serious injuries or fatalities attributable to the radioactive content of the shipments (Garcia 1992).

As a result of environmental impact studies on waste transportation that were undertaken in the 1970s, measures to minimize occupational and population exposure from all forms of radiological waste have been widely implemented by the nuclear industry, including operational restrictions on transport vehicles, ambient radiation monitoring, special packaging requirements, imposition of licensing standards by NRC (which ensure proper waste certification by testing and analyzing packages), changes in waste form to minimize the release of radioactivity in transit (such as cementing or solidifying liquid wastes), and training of emergency personnel to respond to mishaps (NUREG-0170; Levin 1981; ORNL/TM-9780/V3; DOE 1989). In accordance with licensing requirements and international standards, protection against worker and population exposure is provided principally by the waste packaging (NUREG-0170; O'Sullivan 1988).

In regard to footnote d of Table S-4, although the environmental risk of radiological effects stemming from a transportation accident has not been numerically quantified, the study *Shipping Container Response to Severe Highway and Railway Accidents* (NUREG/CR-4829) confirms that the radiological risk of transportation accidents is small. NUREG/CR-4829 is summarized in NUREG/BR-0111, in which the study results are compared with a 1977 study (NUREG-0170). The 1977 study evaluated the risk for all radioactive material shipments, including spent fuel. The evaluations indicated a radiological risk from transportation accidents of one latent cancer fatality every 59 years for all projected 1985 radioactive material shipments. Most of this risk was associated with shipments of medical radioisotopes. The contribution from spent fuel shipments was 2.5 percent of this estimate. The 1987 study included a more detailed approach to the calculation of radiological hazards and concluded that the hazard from spent fuel shipments is less than one-third of that estimated in the 1977 study.

6.3.2.2 Mixed Waste

Mixed waste from reactors accounts for only a small percentage of annual LLW generation. Only very limited off-site disposal facilities for mixed waste have been available since 1985 (NUREG/CR-5938). Utilities are finding ways to treat some of their mixed waste so that it is no longer hazardous, thus making it possible to dispose of the radioactive component along with other LLW (NUREG/CR-5938). The remainder of mixed waste, however, is currently stored on site.

6.3.2.3 Spent Fuel

The only spent-fuel shipments from nuclear power plants have been from one plant to another or to Department of Energy (DOE) facilities or fuel reprocessing plants. Table S.4 Rule (54 FR 187; 10 CFR 51.52) reflects the state and federal restrictions that trucks used to ship spent fuel are limited to 73,000 lb of capacity, while rail cars are limited to 100 tons/cask/car. Also, under Table S-4, each reactor is expected to make less than one shipment per day by truck and fewer than three per month by rail.

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A concern in spent-fuel shipment is the risk of theft or sabotage leading to a release that could pose a major risk of occupational and population exposure and to the environment. Spent-fuel shipments, however, have proven to be unattractive targets for theft or sabotage. Shipping casks are designed to withstand severe transportation accidents and are resistant to small-arms fire and high-explosive detonation (NUREG-0170; DOE/RW-0065).

States and communities along transportation corridors may impose additional restrictions on the transport of nuclear waste. These restrictions, which are designed to protect health, safety, and the environment, are permitted if they afford an equal or greater level of protection, do not cause undue burdens on commerce, and do not discriminate against particular transporters (Pub. L. 93-633; OTA-SET-304). There has been dramatic growth in local ordinances pertaining to special permitting, prenotification, centralized dispatching and communications, and vehicle manifesting (Anderson 1981; Smith 1982; ORNL/TM-9563).

6.3.3 Effects of License Renewal

During the license renewal period, LLW annual shipments from BWRs and PWRs are expected to show a temporary increase (Section 6.3) in the generally decreasing trend. During the 10-year refurbishment period for the conservative license-renewal scenario, the average annual LLW shipments could increase by 96 percent (for a total of 69 shipments) per PWR and 31 percent (for a total of 77 shipments) per BWR. The increase in waste shipments would be considerably less during the subsequent 20 years of operation. In either case, however, rail and truck transport corridors should easily accommodate the increase in waste shipments.

Within the study sample, Cook, Washington Nuclear Project 2 (WNP-2), San Onofre, Vermont Yankee, and Comanche Peak are under the jurisdiction of compacts and agreement states that do not anticipate additional regulations governing the shipment of LLW from reactors to licensed disposal sites. In the case of WNP-2, there is a relatively short distance between the plant and the Hanford site, the licensed LLW facility for the Northwest Compact. The same applies to San Onofre, which expects to ship LLW to a facility in Ward Valley (25 miles west of Needles), to be operated by U.S. Ecology (*Radioactive Exchange* 1989). Greater distances between plants and disposal facilities will complicate the establishment of a corridor-by-corridor management system because of the number of jurisdictions involved. Staff estimates that transportation distances for some plants to disposal facilities could be as much as 2000 miles. Mixed-waste shipments to a licensed disposal facility (when available) should continue to be a small percentage of total LLW volumes shipped for disposal.

The original analysis on which Table S-4 was based is the report NUREG-75/038, *Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants, Supplement 1*, published in April 1975. This report assumed transportation distances for spent fuel and high-level wastes to be less than 1000 miles for almost all reactors except those located in the far western states. Shipping distances for other packaged wastes were assumed to be even shorter. In accordance with the Nuclear Waste Policy Amendments Act of

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1987 (Pub. L. 100-203, Title V, Subtitle A, 101 stat. 1330-227, December 22, 1987), a possible high-level waste repository is being characterized at Yucca Mountain in Nevada. Therefore, the reactors in the eastern half of the country will have shipping distances near 2000 miles for spent fuel and high level wastes. However, the environmental impacts of transportation are so small that even an increase by a factor of 10 would not significantly change the total environmental impacts of the whole fuel cycle. Therefore, the values in Table S-4 do not need to be updated, because the conservatism (see Section 6.2.3) built into these estimates ensures that the total fuel cycle environmental impacts per reactor are not underestimated. Table S-4 would bound the proposed renewal of licenses currently permitted under 10 CFR 2.109 and 50.51.

For spent fuel, DOE has decided that when a permanent repository or MRS becomes available, acceptance allocation priority will go to the oldest spent fuel, on an industry-wide basis (DOE/RW-0220). For purposes of estimation, DOE assumes that if a spent-fuel repository opened in 2003, total storage requirements from all utilities (based on the above-mentioned priority system) would range from 12,200 metric tons (from 83 reactors) to 20,000 metric tons (from 107 reactors) (DOE/RW-0220). Thus, total spent-fuel volume per reactor would range from 147 to 187 metric tons. Assuming that license renewal would, on average, increase spent-fuel generation per reactor by 50 percent, the incremental amount of spent fuel that would be added to this volume would range between 73 and 93 metric tons. This would further translate into an additional 2 to 3 days of shipments for each reactor's spent fuel under the conditions specified in Table S-4 (10 CFR 51.52). If disposal without license renewal would have required between 330 and 550 days for shipment of all spent fuel by truck to a repository, license renewal could require from 170 to 270 additional days for shipment for all relevant plants (DOE/RW-0220). Transportation distances could range up to 2000 miles.

The public radiation exposure and other potential transportation impacts resulting from radioactive waste transport have been addressed generically in Table S-4 (10 CFR 51). All dose projections in Table S-4 are presented on the basis of a reference reactor year, so estimates are dependent on neither the number of years of operation nor the particulars of operation. The doses are small, as are the potential health effects. If these impacts are increased by a factor of 2 (as would occur if waste shipments were doubled, as is approximately projected for PWR refurbishment waste), the impacts would remain small. If occupational doses for transportation workers were 8 man-rem for 200 workers, the average worker exposure of 0.04 rem/year would be only 1 percent of the occupational dose limit guidelines of 10 CFR Part 20. Individual members of the public are exposed to doses from transportation of waste far lower than the standards discussed previously in this document. The average dose received per reactor-year by members of the exposed population is estimated to be 0.0027 rem (3 rem/1100), which is far lower than the individual annual total dose standard of 0.1 rem in 10 CFR 20.1301. Assuming this exposed population received doses from transportation of waste for 20 years, the cumulative average dose would be 0.054 rem, still less than 0.1 rem. Using the risk factor of 5 cancer deaths per 10,000 rem, the risk of death from cancer by an average member of the exposed

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population would be 2.7×10^{-6}, which is also an acceptable risk under the EPA risk standard. The potential for adverse impacts from low-probability accidents is discussed in the previous section. Design refinements to LLW shipping containers and changes to transport systems will continue to occur during the license renewal period, which should result in even lower probability of accidents and lower releases. Changes to the latter will result from continued interaction with states, communities, and tribal nations in emergency preparedness, vehicle inspection, and manifesting on a corridor-by-corridor basis (Anderson 1981; ORNL/TM-9563; Smith 1982; EPRI NP-5933; Sprecher 1990).

6.3.4 Conclusion

The environmental impacts from the transportation of fuel and waste attributable to license renewal are found to be small when they are within the range of impact parameters identified in Table S-4. The estimated radiological effects are within regulatory standards. The nonradiological impacts are those from periodic shipments of fuel and waste by individual trucks or rail cars and thus would result in infrequent and localized minor contributions to traffic density. Programs designed to reduce risk further, which are already in place, provide for adequate mitigation. Table S-4 should continue to be the basis for case-by-case evaluations of transportation impacts of fuel and waste until such time as a detailed analysis of the environmental impacts of transportation to the proposed repository at Yucca Mountain becomes available. This issue is Category 2.

6.4 GENERATION AND STORAGE OF RADIOACTIVE WASTE DURING THE TERM OF THE RENEWED LICENSE

6.4.1 Introduction

The following discussion considers in greater detail the generation of radioactive waste at nuclear power plants during the term of a renewed operating license and the issues associated with the management of that waste. The discussion addresses LLW, including greater than Class C (GTCC) LLW; mixed waste; and spent fuel.

6.4.2 Low-level Waste

LLW is defined as radioactive waste not classified as high-level waste, transuranic waste, spent nuclear fuel, or byproduct material as defined in Section 11e.(2) of the Atomic Energy Act of 1954 as amended. LLW is classified as A, B, C or GTCC according to the half-lives and concentrations of key radionuclides. Detailed descriptions of these classes are given in 10 CFR 61. In general, requirements for waste form, stability, and disposal methods become more stringent when going from class A to GTCC.

In 1992 approximately 65 percent of the total volume of LLW in the United States was generated by non-utility sources, including academic, medical, industrial and government sectors; 35 percent was generated by nuclear power plants (de Planque 1994). However, percentages vary and have been closer to 50 percent. For example, in 1993, 51 percent was generated by utilities (NUREG-1350, Vol. 7); in 1990, 56 percent (NUREG-1350, Vol. 4); in 1989, 52 percent (NUREG-1350, Vol. 3). LLW from nuclear power plants ranges from trash suspected of being

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slightly contaminated to highly radioactive material such as activated structural components found within or in close proximity to the reactor. LLW includes reactor components, tools, spent demineralizer resins, evaporator concentrates, used filters, and miscellaneous contaminated wastes such as rags, mops, paper, and protective clothing.

Additional LLW will be generated by the license renewal of nuclear power plants requiring interim on-site storage. The following discussion focuses on how projected volumes of LLW generated could affect on-site storage requirements, particularly in light of state-to-state variations in the availability of LLW disposal facilities. In this regard, it is pertinent to note that when storage of LLW extends beyond the 5-year guideline established by the NRC, and perhaps in excess of SAR projected curie or volume limits, licensees may have to secure additional license authority under 10 CFR 30 and/or 10 CFR 50 (EPRI TR-100298). Although the demand for on-site interim storage for refurbishment waste and license renewal operations is an important issue, it is one that relates primarily to the uncertainties associated with additional license authority to store the increased activity of radioactive material. Of less concern is the potential for significant environmental impacts from the storage of radioactive material.

Off-site disposal issues relevant to license renewal are also discussed below; however, it is important to stress that these issues are framed in terms of how they could affect the need for on-site interim storage rather than in terms of potential environmental impacts associated with off-site disposal facilities. Operating licenses for disposal facilities ensure minimal impacts to disposal-site workers, adjacent workers, and the environment. Under 10 CFR 61, licensed LLW disposal facilities must protect the general population from release of radioactivity, protect workers, protect inadvertent intruders after institutional controls cease, and ensure disposal site stability.

6.4.3 LLW Baseline

The following section provides information on past and present volumes of LLW generated at nuclear power plants and current practices and trends in volume reduction (VR). Subsequent sections present information pertaining to on-site storage conditions and the status of off-site disposal facilities. This baseline information is used to frame potential issues associated with LLW generated from operations and refurbishments under current licenses as well as from operations and refurbishments for an additional 20 years.

6.4.3.1 LLW Waste Generation and Volume Reduction

The first step in assessing potential issues associated with LLW generation during both the initial license period and from extended operations is to determine primary trends in the production of LLW. These trends suggest both future on-site storage and off-site disposal requirements. In general, the following analysis shows that the production of LLW is on the decline and indicates that VR activities will continue to reduce the amount of waste requiring storage and disposal during the period of the initial license and beyond.

Specifically, Table 6.6 shows historic and projected routine LLW volumes and radioactivity for BWRs and PWRs through

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Table 6.6 Historic and projected volume and radioactivity of routine compacted low-level radioactive waste shipped for disposal from boiling-water reactors and pressurized-water reactors: 1980–2020

End of calendar year	Volume (10^3 m^3)[a]		Radioactivity[b] (10^3 Ci)	
	Annual	Accumulated[c]	Annual	Accumulated[d]
		Boiling-water reactors		
1980	26.1	141	41	128
1981	23.0	164	42	144
1982	25.5	190	38	155
1983	22.6	212	56	183
1984	24.4	237	29	178
1985	23.1	260	28	177
1986	17.3	277	32	182
1987	14.3	292	28	183
1988	11.7	303	29	185
1989	14.2	317	32	190
1990	10.3	328	34	196
1991	15.6	343	53	221
1992	9.6	353	34	220
1993	9.5	362	34	223
1994	9.5	372	34	225
1995	9.5	381	34	228
1996	9.5	391	34	231
1997	9.5	401	34	233
1998	9.5	410	34	235
1999	9.5	420	34	238
2000	9.5	429	34	240
2001	9.5	439	34	242
2002	9.5	448	34	244
2003	9.5	458	34	246
2004	9.5	467	34	248
2005	9.5	477	34	250
2006	9.5	486	34	252
2007	9.5	496	34	254
2008	9.5	505	34	256
2009	9.4	514	34	257
2010	9.1	523	33	258
2011	8.5	532	30	256
2012	8.4	540	30	255
2013	7.8	548	28	253
2014	6.6	555	24	247
2015	5.8	560	20	239
2016	5.4	566	20	233
2017	5.1	571	18	227
2018	4.9	576	18	222
2019	4.8	581	17	217
2020	4.8	585	17	213

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Table 6.6 (continued)

End of calendar year	Volume (10^3 m^3)[a]		Radioactivity[b] (10^3 Ci)	
	Annual	Accumulated[c]	Annual	Accumulated[d]
		Pressurized-water reactors		
1980	22.4	124	24	81
1981	22.8	147	31	102
1982	20.8	168	34	122
1983	21.4	189	32	138
1984	21.0	210	41	163
1985	18.7	229	29	171
1986	11.6	241	22	175
1987	12.2	253	25	184
1988	10.9	264	36	203
1989	13.4	277	48	231
1990	7.8	285	27	234
1991	8.1	293	24	237
1992	7.8	301	29	247
1993	7.9	309	30	257
1994	8.0	317	30	267
1995	8.1	325	30	277
1996	8.2	333	31	287
1997	8.2	341	31	297
1998	8.2	349	31	307
1999	8.3	358	31	316
2000	8.3	366	31	326
2001	8.3	374	31	334
2002	8.3	383	31	343
2003	8.3	391	31	352
2004	8.3	399	31	360
2005	8.3	407	31	368
2006	8.3	416	31	378
2007	8.3	424	31	384
2008	8.3	432	31	392
2009	8.3	440	31	400
2010	8.2	449	31	407
2011	8.1	457	31	414
2012	8.0	465	30	420
2013	7.6	472	28	425
2014	6.9	479	26	427
2015	6.5	485	24	429
2016	6.3	492	24	430
2017	6.0	498	22	431
2018	5.8	504	22	431
2019	5.7	510	21	432
2020	5.7	515	21	433

[a]One cubic meter = approximately 35.3 ft^3.
[b]Decayed from year of addition using ORIGEN 2 code; 1 Ci (curie) = 37×10^9 becquerels.
[c]Volume accumulation means total waste volume shipped from all plants up to and including the current year.
[d]Radioactivity accumulation is the total curies from low-level waste disposal up to and including the current year.
Source: DOE/RW-0006, Rev. 7 (1980–1989 data) and Rev. 8 (1990–2020 data).

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2020; Table 6.7 shows historic volume and radiological trends for the 10 plants in the study sample from 1985 to 1990. Estimates provided by both tables do not include the additional radiological waste generated for activities associated with license renewal. Table 6.8 depicts total generated solid LLW shipped for off-site disposal after compaction or other predisposal VR treatment. Declining disposal volumes in the plant sample generally reflect those of the industry as a whole, with a few exceptions. The exceptions are accounted for by waste-producing refurbishment activities such as steam generator replacements.

Data in Table 6.6 are provided by the Integrated Radioactive Waste Inventory database prepared for the U.S. Department of Energy (DOE) by Oak Ridge National Laboratory (DOE/RW-0006). Sources of data for this inventory include annual figures from the nuclear industry, trade organizations [e.g., Electric Power Research Institute (EPRI)], and a variety of scientific organizations. LLW volume figures provided by this integrated data, while exhibiting a consistent downward trend, are larger than those provided by the industry through the Institute for Nuclear Power Operations (INPO).

VR and waste minimization efforts have been undertaken by utilities in response to increased disposal costs for LLW. These efforts include segregation, decontamination, minimizing exposure of materials and tools to the contaminated environment, sorting potential contaminated materials, and dewatering and evaporation (Strauss 1987; Coley 1987; EPRI NP-5526 Vols. 1 and 2). Some of the most effective VR strategies are compacting, consolidating, and monitoring waste streams to reduce the volume of LLW requiring storage, as well as reducing the exposure of routine equipment to the reactor environment (Strauss 1987; Taylor 1987; EPRI NP-6163; EPRI NP-5526 Vols. 1 and 2; Shaw 1988). As shown in Tables 6.6 and 6.7, which pertain to wet and dry wastes, significant average VRs have been achieved for LLW shipped for disposal from nuclear power plants. According to EPRI, during the 1980s, BWRs achieved an average disposal VR of 24 percent. Between 1981 and 1985, there was a 50 percent reduction in dry waste, a 42 percent reduction in wet waste, and a 48 percent reduction in total waste volume from PWRs. In 1987, the median total LLW volumes shipped for disposal from BWRs and PWRs was 19,700 ft^3 (558 m^3) and 8800 ft^3 (250 m^3), respectively (EPRI NP-5526, Vols. 1 and 2).

According to 1993 performance indicators published by the INPO, the level of LLW per power plant unit has continued to decrease since the 1980s. During 1993, for example, the median value of the low-level solid radioactive waste per BWR unit was about 5,620 ft^3 (159 m^3) as compared to the industry median of 10,800 ft^3 (306 m^3) in 1988. Similarly, the median values for 1993 are below industry 1995 goals of 8,650 ft^3 (245 m^3) for BWRs and 3,880 ft^3 (110 m^3) for PWRs. Technological advances, as well as major reductions in the extent of contaminated areas within power plants, have contributed to the decrease in waste quantities generated over the past several years. Although volumes have declined, radioactivity levels have remained the same (INPO 1994).

The 10 plants in the study sample illustrate current industry-wide VR practices, including ultra-high-pressure compaction of waste drums, incineration of waste oils and

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Table 6.7 Solid low-level radioactive waste generated by 10 power plants: 1985–1990

Year	1985	1986	1987	1988	1989	1990
Pressurized-water reactor plant volume[a]						
Comanche Peak	NA[b]	NA	NA	NA	NA	0
D. C. Cook 1 and 2	8.28+02	5.28+02	4.63+02	2.46+02	3.88+02	1.95+02
H. B. Robinson	6.42+02	4.53+02	1.01+02	8.42+01	9.69+01	6.99+01
Indian Point 1 and 2	6.89+02	5.30+02	2.30+02	2.41+02	4.78+02	2.60+02
Indian Point 3	2.39+02	8.29+01	3.17+02	1.82+02	5.77+02	6.66+02
San Onofre 1	1.80+02	2.51+02	3.69+02	3.08+01	1.19+02	5.81+01
San Onofre 2 and 3	5.45+02	2.94+02	2.45+02	2.60+02	3.28+02	1.75+02
Surry 1 and 2	2.02+03	6.39+02	5.15+02	7.30+02	5.38+02	1.48+02
Pressurized-water reactor plant activity[c]						
Comanche Peak	NA	NA	NA	NA	NA	NA
D. C. Cook 1 and 2	2.00+03	1.59+03	2.30+03	5.58+02	1.17+03	1.44+02
H. B. Robinson	3.35+03	1.58+02	2.59+02	3.76+02	1.86+02	1.44+01
Indian Point 1 and 2	5.75+02	2.52+02	8.34+02	4.67+02	3.60+02	2.08+03
Indian Point 3	5.49+02	2.58+01	3.33+02	3.57+02	3.50+02	1.50+02
San Onofre 1	6.04+00	3.82+02	4.98+01	4.06+00	1.72+03	1.27+01
San Onofre 2 and 3	1.72+03	1.93+02	2.71+02	2.55+03	2.72+03	3.34+01
Surry 1 and 2	1.21+03	1.16+03	2.94+04	1.94+02	1.31+03	1.13+03
Boiling-water reactor plant volume[a]						
Hatch 1	2.04+03	1.35+03	7.78+02	8.36+02	8.53+02	1.38+03
Hatch 2	*d*	*d*	*d*	*d*	*d*	*d*
Vermont Yankee	5.43+02	3.10+02	2.23+02	1.73+02	4.84+00	0
Limerick	3.06+02	5.76+02	3.81+02	8.95+02	5.76+02	6.86+02
WNP-2[e]	4.02+02	3.02+02	3.75+02	4.70+02	3.64+02	3.34+02
Boiling-water reactor plant activity[c]						
Hatch 1	3.83+04	8.82+02	1.82+03	2.02+03	1.91+03	2.85+04
Hatch 2	*d*	*d*	*d*	*d*	*d*	*d*
Vermont Yankee	1.73+04	3.33+02	1.19+04	4.25+02	2.15+00	0
Limerick	2.06+01	7.53+02	2.15+03	9.70+02	3.40+04	1.24+03
WNP-2	2.96+02	5.07+02	1.09+03	1.01+03	1.10+03	1.29+03

[a]In exponential notation (m^3); 1 m^3 = approximately 35.3 ft^3.
[b]NA = not available; in most cases because the plant was not on line.
[c]In curies; 1 Ci = 37 × 10^9 becquerels.
[d]Included with Hatch 1 totals.
[e]WNP-2 = Washington Nuclear Project 2.
Source: NUREG/CR-2907.

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Table 6.8 Percent of low-level radioactive waste treated by volume reduction methods in current use in the plant sample[a]

Plant	Waste compaction	Sorting before segregation	Decontamination	Shipment	Other
Comanche Peak		100			*b*
D. C. Cook	85	85	85	85	15[c]
Hatch	85–90			80–90	
Indian Point 2	30	55	50		15[d]
Indian Point 3	90		5	1	
Limerick	55		5	35	5[e]
Robinson		60–75	5	70	75[f]
San Onofre					
Surry	60	5	25	5	5[g]
Vermont	95	40	10	40	*h*
Yankee	55				45[j]
WNP-2[i]					
Sample average	50–51	31–36	17	29–30	15
Industry-wide average (n)[k]	46(49)	30(33)	15(41)	35(31)	42(44)

[a]In percent of waste volume treated. Because of multiple volume reduction methods, totals may not equal 100 percent.

[b]Comanche Peak color codes all hazardous and radioactive waste and does limited sorting prior to shipment. An off-site contractor performs incineration, compaction, and decontamination of tools.

[c]D. C. Cook dewaters 15 percent of its wet waste.

[d]Indian Point 2 employs sand blasting, steam cleaning, freon cleaning, and sectioning as well as direct burial of resins, filters, and sludges and is also examining plans for chemical decontamination of reactor coolant.

[e]Limerick incinerates contaminated oils and reagents.

[f]Robinson relies heavily on decontamination of material to the reactor environment and has seen steady improvements in use of this method.

[g]Surry employs an off-site vendor for incineration of waste oil and for supercompaction. The plant is completing an interim storage facility for storage of one year's LLW stream. This facility will employ asphalt solidification, high-pressure compaction, and decontamination of waste.

[h]Vermont Yankee employs a survey process for dry active waste that results in the sorting of high- and low-activity wastes and supercompaction of the former followed by off-site disposal. Irradiated reactor components are stored in the spent-fuel pool. High reliance on compaction has resulted from LLW being stored on site between 1989 and 1991 (Vermont has no available compact).

[i]WNP-2 = Washington Nuclear Project 2.

[j]WNP-2 dewaters spent resins.

[k](n) = number of plants responding to survey. In updating data for the Final GEIS, the sample plants were resurveyed and published updates were used. However, no new industry-wide survey was undertaken because neither published updates nor sample plant data showed significant departures from previous trends.

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resins, mobile thin-film evaporation, waste crystallization, and asphalt solidification of resins and sludges. These are aided by formal establishment of VR goals, assignment of responsibilities to each plant division, special training, and monitoring of procedures (EPRI NP-3763; Riales 1985; Taylor 1987; EPRI NP-5526 Vols. 1 and 2; EPRI NP-5983; Shaw 1988). Some VR activities are performed on site; others are undertaken by off-site contractors. As shown in Table 6.8, the most common VR techniques used for plants in the study sample are compaction, waste segregation through use of radiation control zones and control points, and sorting of wastes into radioactive and nonradioactive batches prior to off-site shipment. The proportions of waste volumes treated by these techniques in the plant sample reflect those of the industry as a whole.

The efforts of three BWRs (WNP-2, Limerick, and Hatch) and two PWRs (Surry and Robinson 2) are representative of emerging VR trends, especially the growing reliance on off-site waste management vendors. VR techniques utilized by these plants typify the state of the art for the industry as a whole. Utilization of these VR methods will support NRC policy encouraging LLW volume reduction to alleviate concern for adequate LLW disposal capacity (46 FR 51100). WNP-2 has hired a radioactive-waste-processing service for segregation, compaction, and waste packaging (Macbeth and Allen 1986). Limerick has hired a vendor to establish a plant database and surveillance program for radionuclide management (Trinoskey et al. 1987).

Following licensed operation, Surry adopted a filtration system in conjunction with demineralization and evaporation processes for liquid waste that are provided by an off-site vendor. Hatch employs a box compactor on site and uses an off-site vendor for supercompaction and incineration (EPRI TR-101160). Robinson relies principally upon segregation, sorting of waste before shipment, and source control to minimize plant areas devoted to LLW storage. In 1985, Robinson averaged 5200 ft^3 (150 m^3) of LLW storage area plant-wide. By 1986, this was reduced to 2300 ft^3 (65 m^3) through sorting, segregating, and retrieving usable nonradioactive wastes. Segregated refuse that emits less than 5 mrem/h is sent to a trash-monitoring facility. Trash that demonstrates an activity level less than 100 counts per minute above background is then sent to the plant landfill. The remainder is currently disposed of at Barnwell (EPRI NP-5934).

6.4.3.2 Interim LLW On-Site Storage

LLW is normally stored on site on an interim basis before being shipped off site for permanent disposal. On-site storage facilities are designed to minimize personnel exposures. High-dose-rate LLW is isolated in a shielded storage area and is easily retrievable. The lower-dose-rate LLW is stacked or stored to maximize packing efficiencies. NRC requirements and guidelines ensure that LLW is stored in facilities that are designed and operated properly and that public health and safety and the environment are adequately protected (EPRI NP-7386). NRC requirements and guidelines include the following:

- The amount of material allowed in a storage facility and the shielding used should be controlled by dose rate criteria for both the site boundary and any adjacent off-site areas. Direct

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radiation and effluent limits are restricted by 10 CFR Part 20 and 40 CFR Part 190. The exposure limits given in 10 CFR 20.1301 apply to unrestricted areas.

- Containers and their waste forms should be compatible to prevent significant corrosion within the container. After a period of storage, the subsequent transportation and disposal should not cause a container breach.
- Gases generated from organic materials in waste packages should be evaluated periodically with respect to container breech. After a period of storage, the subsequent transportation and disposal should not cause a container breach.
- Gases generated from organic materials in waste packages should be evaluated periodically with respect to container breech. High-activity resins should not be stored more than 1 year unless they are in containers with special vents.
- A program of at least quarterly visual inspection should be established.
- A liquid drainage collection and monitoring system should be in place. Routing of the drain should be to a radwaste processing system (EPRI NP-7386).

NRC has historically discouraged the use of on-site storage as a substitute for permanent disposal. NRC Generic Letter 81-38 (NRC 1981) states that no facility should be built to store waste for longer than 5 years under a licensee's 10 CFR 50.59 evaluation. Specific NRC approval should be obtained. This limitation was based in part on safety considerations but was aimed at encouraging the development of permanent LLW disposal facilities. However, recognizing that the 5-year limit has not influenced the development of new waste disposal facilities and that the states continue to make slow progress, NRC has eliminated in its guidance any language that the 5-year term is a limit beyond which storage would not be allowed.

Regarding nuclear power reactors, the 5-year limit is associated with the need to obtain a separate Part 30 license to store LLW. Generic Letter 81-38 states that under certain conditions, Part 50 licensees should obtain a Part 30 materials license to store LLW. These conditions are that (1) there exists an unreviewed safety question with the proposed storage facility, (2) the existing license conditions or technical specifications prohibit increased storage, or (3) the planned storage time exceeds 5 years. Other than for the conditions noted, NRC regulations and procedures do not call for a separate Part 30 license for power reactors for LLW storage, because power reactor licensees are already authorized under Part 30 to possess by-product materials produced by the operation of the facility within the limits of their operating license.

Generic Letter 81-38 states that the application for a Part 30 license is for the administrative convenience of the Commission and is not intended to be substantively different from an application for amendment of the facility operating license (i.e., the Part 50 license). Because Part 50 licensees are already authorized under Part 30 to possess their LLW, NRC staff revised the guidance to state that these licensees should amend their Part 50 licenses when the storage of LLW is not within the limits of their current operating license. On February 1, 1994, the Commission, in responding to SECY-93-323, which recommended withdrawal of the on-site storage rulemaking, directed the staff to eliminate the requirement for power reactor

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licensees to obtain a separate Part 30 license (SECY-94-198). Agreement states are currently reviewing proposed changes to existing guidance.

Several events have increased the trend towards longer on-site storage. These events include the closure of the Beatty, Nevada, site in 1992; the restriction of the Richland, Washington, facility to Northwest Compact and Rocky Mountain Compact states and the restriction of the Barnwell, South Carolina, site to waste generated by Southeast Compact states. As of July 1994, 33 states were without access to licensed full-service disposal facilities. The status of state efforts to form compacts and identify new disposal sites is discussed in Section 6.4.3.3. However, as of July 1, 1995, all states except North Carolina have access to the Barnwell site. The Envirocare site in Utah takes limited types of waste from certain generators.

Most utilities have adequate on-site space for at least the near term, and many facilities are already in place (de Planque 1994). For example, the Cook nuclear plant in Michigan has built an on-site disposal facility (completed in 1992) that will not reach capacity until about 2003. Moreover, the facility is designed so that it can be easily expanded to double its storage capacity after that date (McRae 1994). Vermont Yankee has stored LLW on site since 1989, when it was denied access to off-site disposal space. Vermont is not a member of a compact, nor does it have official "unaffiliated state" status. However, pending Congressional approval, Vermont will become a member of the Texas Compact. Vermont Yankee has one of the smallest plant sites in the study sample [125 acres (50.6 ha)]. Typically, for on-site storage of LLW or spent fuel, a few tenths of an acre are disturbed and occupied by the storage facilities themselves. In addition, a few acres are maintained as exclusion areas surrounding the facilities.

Indian Point is planning to store routine LLW on site if use of an out-of-state disposal facility is denied to New York State after 1994 and if New York fails to establish its own repository. Current (1994) plans are to store LLW for up to 5 years in a qualified interim facility. For two operating units and a 239-acre (96.7-ha) site, such a facility would probably utilize no more than 1 to 2 acres (0.4 to 0.8 ha), or 0.8 percent of the site. Surry is completing an interim facility that will be able to store packaged LLW for up to 1 year. It will employ asphalt solidification, high-pressure compaction, and decontamination to reduce the volume of LLW requiring storage until a new Southeast Compact disposal facility becomes available. Limerick can store LLW for 5 years after closure of the plant site (a contingency undertaken in contemplation of decommissioning). It is investigating the possibility of shipping LLW to two other plants owned by Philadelphia Electric Company. One of these facilities has a 5-year storage capacity, and the other can store 3 months' volume. No plant has had to acquire additional land for interim waste storage.

The environmental impacts of on-site LLW management activities, including interim storage, result principally from exposure to radioactivity. Workers receive external doses from exposure to radiation while handling and packaging the waste materials and from periodic inspections of the packaged materials and any other handling operations required during interim storage. Such doses, however, account for a small fraction of the total radiation dose

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commitment to workers and, as discussed in Section 4.6.3, the total dose commitment is well within regulatory limits. Radiation doses to off-site individuals and biota from interim LLW storage are insignificant. The principal exposure pathway is direct radiation from steam generator assemblies and other large assemblies stored in shielded buildings, but this pathway results in a very small dose commitment (Section 3.8.1.6).

6.4.3.3 LLW Disposal

In 1992 approximately 35 percent of LLW disposed of in the United States was generated by nuclear power plants (DOE/LLW-181) although, as noted earlier, percentages vary and have been closer to 50 percent. Compacted dry waste is the largest single form of LLW disposed of from nuclear power plants, accounting for approximately 46 and 36 percent of total average annual volumes from PWRs and BWRs, respectively (EPRI NP-5526, Vols. 1 and 2; Shaw 1988). Through 1994, the two remaining commercial disposal sites—Barnwell, South Carolina, and Hanford, Washington—acquired a total LLW volume of 39,576 ft^3 (1,121 m^3) (Radioactive Exchange 1994).

The NRC emphasizes an integrated-systems approach to LLW disposal, including consideration of site selection, site design and operation, waste form, and disposal facility closure (10 CFR 61). NRC specifies requirements that must be met by the waste generator, including requirements for waste form and content, waste classification, and waste manifests. The NRC's approach emphasizes passive rather than active systems to minimize and retard releases to the environment over the extremely long periods of time contemplated for the control of radioactive material. The performance objectives given in 10 CFR 61 assume no active controls at the disposal site after 100 years and, further, depending on the waste classification, site stability for up to 300–500 years. The site itself, including subsurface zones, is considered to be part of the containment mechanism, which by design (e.g., clay liners and covers or engineered surface barriers) slows the expected release of acceptably small quantities of radioactivity (NUREG-0945).

Waste disposal facilities sited and operated consistent with 10 CFR 61 and other appropriate regulations would result in minimal environmental impact. Licensing of these facilities requires environmental documentation, including assessment of potential environmental impacts. Procedures are established to ensure that performance objectives are met. Waste generators must meet the waste acceptance criteria established for the facility and adhere to packaging requirements.

With the passage of Pub. L. 96-573 in 1980, the Low-Level Radioactive Waste Policy Act, states were assigned responsibility for the disposal of certain LLW generated within their borders (except for LLW generated by the federal government). This act was amended by the Low-Level Radioactive Waste Policy Amendments Act of 1985 (Pub. L. 99-240), which mandated that from January 1, 1986 through December 1992, states were to divide their volume allocations among waste generators. Volume allocations for the 10 plants in the study sample are depicted in Table 6.9. Volumes due to unusual activities ("unusual volumes") denote a special disposition allocation reserved for refurbishment, decommissioning, and core-related accident waste. As of December 31, 1992, DOE had

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Table 6.9 Profile of allocation of low-level radioactive waste for 10 power plant sites: current and projected allocations[a]

Plant	Compact or unaffiliated state	Unusual volumes[a] (ft^3/year)[b]	Total allocation volume (ft^3/year)	Volume received at disposal[c] used (ft^3/year)	Percentage of allocation used 1986–1992
Comanche Peak	Texas	0	12,330	7,746	63
D. C. Cook 1 and 2	Midwest	46,538	179,474	64,437	36
Hatch 1 and 2	Southeast	0	371,352	237,532	64
Indian Point 1 and 2	New York	0	132,936	71,313	54
Indian Point 3	New York	0	66,468	24,672	37
Limerick 1	Appalachian	0	82,920	137,265	81
Robinson 2	Southeast	0	165,840	32,885	40
Surry 1 and 2	Southeast	0	199,404	89,184	54
San Onofre 1, 2, and	Southeast	0	148,836	86,407	43
3	Vermont	0	148,836	48,546	33
Vermont Yankee	Northwest	0	185,676	94,196	51
WNP-2[d]					

[a]Volumes due to unusual activities (e.g., steam generator replacement).
[b]1 ft^3 = 0.028317 m^3.
[c]To Barnwell, South Carolina; Beatty, Nevada; and Hanford, Washington.
[d]WNP-2 = Washington Nuclear Project 2.
Source: DOE/EM-0143P.

granted six requests for unusual volume allocation, totaling 190,283 ft^3 (5,388 m^3); two of the six requests were returned to the requestor for additional information and no requests were denied. D. C. Cook, Unit 2, which requested an additional 46,538 ft^3 (1,318 m^3), was the largest of the six requests. A balance of 609,717 ft^3 (17,265 m^3) of the original 800,000 ft^3 (22,653 m^3) of unusual volume allocation remains undistributed. No petitions for unusual volume allocations were submitted to the DOE in 1992.

Between 1986 and 1992, all plants in the study sample stayed within allocated ceilings, averaging 51-percent utilization of disposal capacity. The plant that came closest to exceeding its allocated ceiling was Limerick, which used 81 percent of its volume allocation by 1992. Nuclear power plants have generally shipped far less LLW for disposal than had been anticipated under Pub. L. 99-240. Even those few reactors that received unusual volume allocations could have disposed of the waste generated by the unusual activities using only their regular allocations specified under the law. From 1986 through 1992, commercial power reactors used only 49.5 percent of the total regular allocations issued through 1992 (DOE/EM-0143P). While historical patterns of disposal provide no guarantees regarding future disposal, the fact that these ceilings remained intact suggests that compact planners will design new disposal facilities that can accommodate a wide range of disposal scenarios.

Pub. L. 99-240 also provides milestones, incentives, and penalties to promote the

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states' continuous progress toward new LLW disposal facility development. States must ensure their own disposal capacities by forming waste compacts or siting their own disposal facilities. Table 6.10 identifies current and future host states for LLW disposal facilities.

Figure 6.1 shows the geographic arrangement of current compacts and their respective state members. Also shown in Fig. 6.1, incremental progress is being made in forming enduring compacts and siting new LLW disposal facilities. Recent examples of progress include the formation of a new Texas Compact that includes Maine and Vermont, which is pending before the U.S. Congress for consent. Also, the Southeast Compact has selected a new site in North Carolina that is expected to be operational by mid-1998. Moreover, the process of site selection is progressing in the Appalachian, Central, and Midwest compacts. Facilities in the host states of Connecticut, Illinois, Massachusetts, Nebraska, New Jersey, Pennsylvania, and New York are scheduled for operation in the period 1999 to 2002. In addition, site activity in the host state of Ohio can begin when the Ohio General Assembly enacts enabling legislation. Michigan, New Hampshire, Rhode Island, District of Columbia, and Puerto Rico are unaffiliated states and have no plans to develop an LLW-disposal facility. They may be able to fulfill their responsibilities through the contracting and/or compact process. Envirocare, in Utah, takes limited types of LLW from certain generators. Despite evidence of incremental progress in siting new facilities, the lack of access of 33 states to LLW disposal sites until Barnwell became available to all states except North Carolina on July 1, 1995 shows the potential to affect all classes of waste generators, including nuclear power plants. Specifically, about 50 percent of the nation's LLW was expected to require on-site storage after 1994, much of it at the point of generation (DOE/EM-0143P).

6.4.4 Effects of License Renewal on LLW

Additional quantities of LLW will result from refurbishment and extended power plant operations under renewed licenses. These activities could, in turn, require the development of additional on-site storage facilities for LLW, including GTCC waste, especially if the power plant is located in a compact or unaffiliated state that has not developed adequate disposal facilities.

6.4.4.1 Generation

Table 6.11 shows that the annual incremental increase in LLW generation for BWRs would be approximately 6163 ft^3 (175 m^3) during the 10-year refurbishment period (assuming four current-term refurbishment outages and one major refurbishment outage; see Section 2.4 for refurbishment scenario). For PWRs, the average incremental increase in LLW generation would be approximately 8410 ft^3 (238 m^3).

The considerably greater volume of refurbishment-associated LLW in the latter case is largely caused by the potential need for steam-generator replacements in PWRs. For the 20 years of operation after refurbishment, annual average LLW generation rates from the eight refueling outages and three in-service inspections would increase approximately 205 ft^3 (5.8 m^3) for BWRs and 145 ft^3 (4.1 m^3) for PWRs (Table 6.6). These conservative-case estimates include waste from refueling operations and represent about 4 percent of the median waste volume generated by BWRs during 1993 and about 9 percent of

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Table 6.10 Actual and estimated dates for completing steps in facility development (estimated dates obtained from compacts/states), April 1995

Compact/host state	Select site	Submit license application	Operate facility
Appalachian/Pennsylvania	1995	Early 1997	Mid-1999
Central/Nebraska	Dec. 1989	July 1990	Fall 1999
Central Midwest/Illinois	Unscheduled	Nov. 1997	July 2000
Midwest/Ohio	Unscheduled	Unscheduled	Unscheduled
Northeast/Connecticut	Unscheduled	1999	2002
New Jersey	Unscheduled	Jan. 1998	July 2000
Southeast/North Carolina	Dec. 1993	Dec. 1993	Mid-1998
Southeast/California	Mar. 1988	Dec. 1989	Mid-1997
	Unaffiliated States		
Maine[a]			
Massachusetts	Unscheduled	Feb. 1998	2000/2001
Michigan	Unscheduled	Unscheduled	Unscheduled
New York	Unscheduled	June 1999	Nov. 2001
Texas	Aug. 1991	Mar. 1992	Mid-1997
Vermont[a]			
District of Columbia, New Hampshire, Rhode Island, Puerto Rico[b]			

[a]Formation of a compact pending with Texas as the host state.
[b]Currently not planning to develop a low-level radioactive waste disposal facility.

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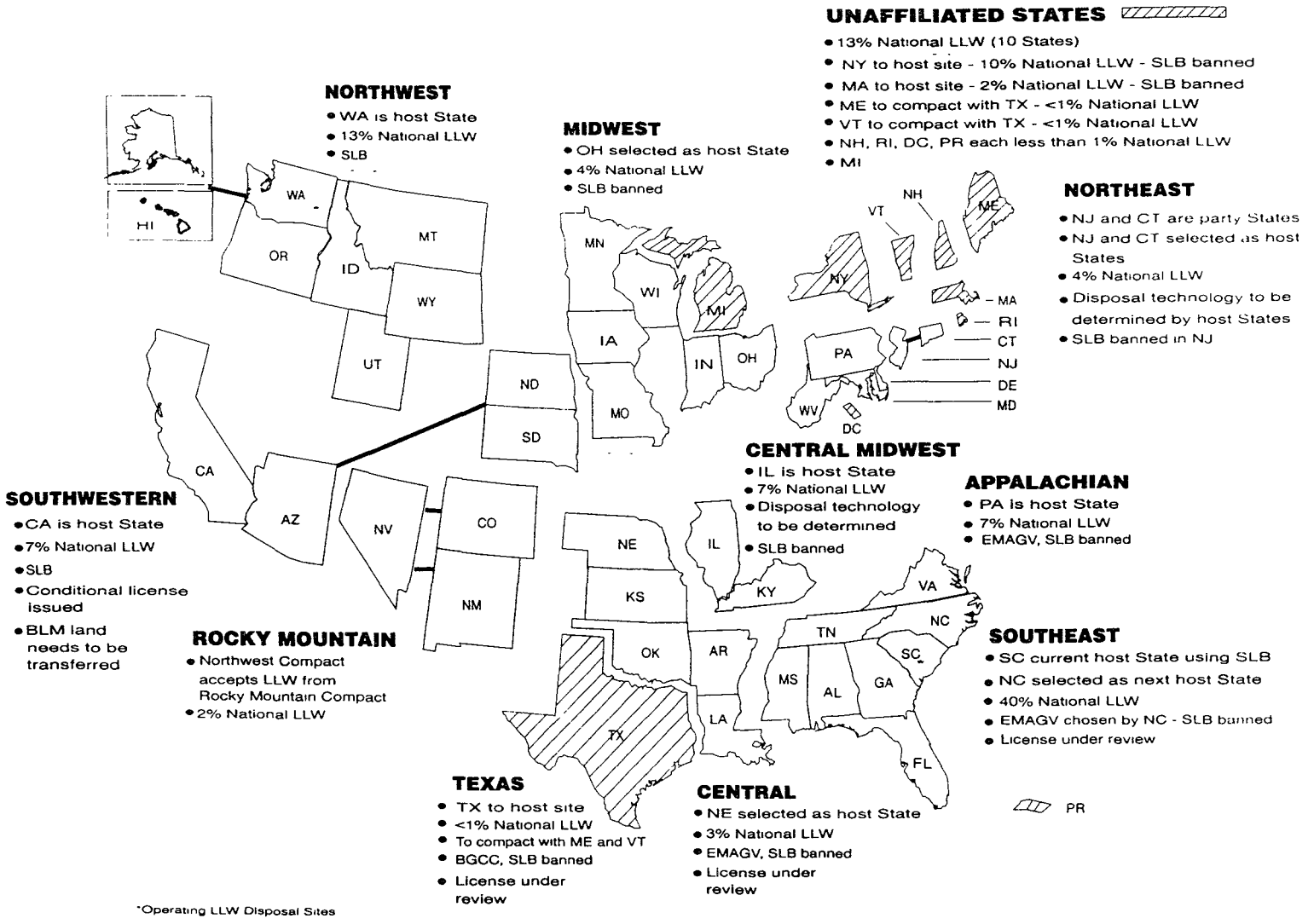
LOW-LEVEL RADIOACTIVE WASTE COMPACT STATUS
APRIL 1995



*Operating LLW Disposal Sites

Note: National LLW volume for 1994 = 0.9 million cube feet disposed
SLB = shallow land burial
EMAGV = Earth mounded above grade vault
BGCC = below ground concrete canisters

Fig. 6.1. Low-level radioactive waste compact status.

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Table 6.11 Total estimated incremental[a] low-level radioactive waste generation during 10-year renewal refurbishment activities and 20-year post-refurbishment operations (ft^3),[b] as shipped

Activity	Boiling-water reactor	Pressurized water reactor
LLW attributable to refurbishments prior to year 41		
Current-term refurbishment outages (4)[c]	8,2000 × 4 = 32,880	9,340 × 4 = 37,360
Major refurbishment outage (1)	28,750	46,740
LLW attributable to plant life extension period (41–60 years)		
Refueling outages (8)	160 × 8 = 1,280	116 × 8 = 928
5-year in-service inspections (2)	731 × 2 = 1,462	466 × 2 = 932
10-year in-service inspection (1)	1,348	1,035
Total LLW	66,000	87,000

[a]Incremental wastes are those in addition to baseline annual for routine operations for the 20 years.
[b]1 ft^3 = 0.028317 m^3.
[c]Number in parentheses is the number of times the activity is performed.
Source: SEA 93-461-10-A:3.

the median value of LLW produced by PWRs during 1993. The total projected increases in LLW generation [66,000 ft^3 (1,870 m^3) for BWRs and 87,000 ft^3 (2,460 m^3) for PWRs] amount to a volume of about 3 percent of the total nuclear power plant waste allocation volume used from 1983 through 1992. Although this scenario assumes distinct LLW streams related to license renewal refurbishments, in fact refurbishments will probably occur as a continuation of normal operations under the original license. Consequently, it will be difficult to distinguish LLW generated by continuing operations (including waste from steam generator replacement) from waste streams related to license renewal. Also, continued progress in VR should slow the accumulation of waste volumes either on site or at a permanent waste disposal facility. Although utilities would rely on the same VR techniques used under original licenses during license renewal (Table 6.12), plants in the study sample anticipate less reliance on waste compaction for VR, because the greatest VR gains from this method have already been achieved. Utilities expect to rely increasingly on incineration, resin drying (dewatering), and off-site contracting for the disposal of large contaminated components through a variety of methods (Efremenkov 1989; DOE/RW-0220).

Certain activities may also produce increased volumes of GTCC waste as a result of removing neutron-activated materials from the reactor vessel or removing materials that are located sufficiently close to the reactor core such that activation results (SEA 93-461-10-A:3). The current inventory of GTCC waste from nuclear utilities consists of a small volume of startup sources, stellate bearings, and other wastes that are being stored at utility sites until a disposal option is available. These wastes bring the total

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Table 6.12 Percent of anticipated low-level waste volumes that will be treated by five types of volume-reduction methods during license renewal for the sample plants[a]

Plant	Waste compaction	Sorted before segregation	Decontamination	Shipment	Other
Comanche Peak		100			
D. C. Cook	85	85	85	85	85
Hatch	85–90			80–90	
Indian Point 2	30	55	50		15[b]
Indian Point 3	90				20[c]
Limerick	20		5	35	40[d]
Robinson		>75	1–5	<70	75[e]
San Onofre					
Surry	35	5	25	5	30[f]
Vermont Yankee	95	40	10	40	
WNP-2[g]	25		5	25	45[h]
Sample average	42–43	33	16–17	31–32	28
Industry-wide average (n)[i]	34(50)	32(35)	19(41)	42(30)	50(43)

[a]In percent of waste volume treated. Because of multiple volume-reduction methods, totals may not add up to 100 percent.

[b]Under license renewal, Indian Point 2 anticipates moving toward greater waste incineration because of increasing volume-reduction costs and uncertainties surrounding availability of a New York state disposal facility. It can keep low-level waste on site in a qualified interim facility.

[c]Indian Point 3 plans to employ a combination of resin drying, incineration of dry active waste, and metals smelting off site during license renewal.

[d]Limerick will move increasingly toward incineration. It will be able to store up to 5 years of low-level waste on site after closure of site.

[e]Robinson will place a greater emphasis on minimizing contamination during license renewal.

[f]Surry will move toward more incineration.

[g]WNP-2 = Washington Nuclear Project 2.

[h]WNP-2 will move toward increasing dewatering of resins. It also anticipates contracting for the disposal of large turbine components and moisture separator reheaters during refurbishment, as well as constructing interim refurbishment-waste warehouse space.

[i]n = number of plants responding to survey. In updating data for the Final GEIS, the sample plants were resurveyed and published updates were used. However, no new industry-wide survey was undertaken because neither published updates nor sample plant data showed significant departures from previous trends.

inventory of GTCC waste from nuclear utilities to 364 ft^3 (10.3 m^3). The activity of the current inventory is estimated to be 3,873,000 Ci (DOE/LLW-114).It is difficult to predict amounts of GTCC waste likely to be produced by refurbishment activities without knowing if reactor core components have to be removed and the extent of irradiation-induced activity. The current conservative estimate is that about 1540 ft^3 (44 m^3) of refurbishment-associated GTCC waste will be generated

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by BWRs and about 500 ft^3 (14 m^3) of GTCC waste will be generated by PWRs (SEA 93-461-10-A:3).

The Low-Level Radioactive Waste Policy Amendments Act of 1985 assigned the responsibility for disposal of GTCC LLW to DOE. Disposal by DOE must be in a facility licensed by NRC; however, states may allow disposal of GTCC waste at LLW sites. NRC's current LLW regulations (10 CFR 61.55) require disposal of GTCC waste by DOE in a geologic repository as defined in 10 CFR Part 60, unless a specific alternative is approved by the Commission. The combined impacts of all waste buried at the repository will be required to meet the applicable standards.

Both the activity and volume of GTCC waste is small when compared to spent fuel. DOE estimates of GTCC waste to be generated through the year 2055 are on the order of 2000 m^3 containing about 90 million curies (DOE/LLW-114). Utility wastes contribute about 66 percent of the volume and 90 percent of the activity of the GTCC waste; the predominant waste form is activated metal components. The volume estimates vary, depending on factors such as assumed packaging, averaging methods, and nonfuel components that may be determined to be covered by HLW contracts; a range of 1000–8700 m^3 is possible for total GTCC waste generation through 2055 (DOE/LLW-114). These volumes and activities can be compared to commercial spent-fuel inventories as of December 31, 1992 of about 10,000 m^3 (just the fuel rods and space between, no packaging) and 26,000 million curies and projections of about 34,000–41,000 m^3 and 25,000–52,000 million curies in the year 2030 (DOE/RW-0006). Spring of 1995 DOE estimates for the Yucca Mountain repository are 147,000 m^3 of waste, a peak activity of 43,000 million curies, and an excavated volume of 4–6 million/m^3. Based on the data in Section B.4.1.2 of Appendix B to this GEIS and assuming 72 PWRs and 37 BWRs, the total volumes of GTCC that would result from renewal activities and plant life extension from all 109 plants would be 2,044 m^3 for the typical case and 2,636 for the conservative case. These volumes appear to be consistent with the other estimates.

The staff also notes that in the 1989 rulemaking to require disposal of GTCC in a deep repository unless disposal elsewhere has been approved by the Commission (54 FR 22578; May 25, 1989), the Commission stated:

> The fact that the expected volume of GTCC waste is very low was an important factor in the Commission's decision to propose the Part 61 amendments. Current evidence shows that the expected volume of GTCC waste is very small relative to volumes of HLW and Class A, B, and C LLW. It is projected that 2000–4800 cubic meters of commercially-generated GTCC waste will need disposal through the year 2030 [U.S. Department of Energy estimates]. This amount of waste is smaller than the anticipated excavated volume of a single emplacement room of a repository, and would not present a significant burden on the capacity of the repository to receive HLW. It would not be a significant factor underlying the need for a second repository.

Based on the Waste Confidence finding reflected in 10 CFR 51.23 concerning the safety and availability of a HLW repository, the Commission has previously determined

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that there is reasonable assurance at least one geologic HLW repository will be available within the first quarter of the twenty-first century, and that sufficient repository capacity will be available within 30 years beyond the licensed life of operation of any reactor. Although the Waste Confidence finding did not expressly encompass GTCC waste, the staff concludes that the Waste Confidence reasoning and information on repository availability, together with relatively small incremental volume and hazard associated with GTCC waste, support a finding that there is reasonable assurance that sufficient GTCC LLW disposal capacity will be made available when needed for facilities to be decommissioned consistent with NRC decommissioning requirements. Off-site disposal capacity for these wastes can reasonably be expected to be available when the HLW repository begins operating.

6.4.4.2 Interim LLW Storage

If compact and unaffiliated states are able to site disposal facilities and accept waste in normal increments (i.e., in accordance with the assigned allocations for each plant in the compact or unaffiliated state), there should be no significant issues or environmental impacts associated with interim storage of LLW generated by nuclear power plants with renewed licenses. Interim storage facilities would be utilized until these wastes could be safely shipped to licensed disposal facilities (EPRI NP-6163). While on-site land will be needed to store the waste, measures taken by the industry appear to be adequate to encompass these additional volumes. For example, Indian Point 2, Limerick, Robinson, and WNP-2 are contemplating construction of additional interim storage facilities associated with license renewal. Indian Point expects to store resin and filter wastes after dewatering in a special Butler-style building. A separate facility may be required for storage of Indian Point 3's old steam generators. Limerick's additional 165,000-ft^2 (15,300-m^2) facility for spent fuel and LLW, to be completed by 1999, will be used for refurbishment-associated wastes. WNP-2 is planning to construct a special warehouse for storage of its low-pressure turbines. The incremental volumes of LLW requiring interim on-site storage would result in increased worker exposure and external radiation dose commitments from waste handling, packaging, and inspection activities. However, such incremental dose commitments would be small and pose a low risk (Section 4.6.3.2).

If off-site disposal facilities are unavailable to accept waste in normal increments, then on-site interim storage may have to take place longer than the 5-year time frame once envisioned by NRC, and additional on-site storage capacity may be needed. As discussed in more detail in Section 6.4.4.6, the radiological implications of storing larger quantities of LLW on site for relatively longer periods of time would be minimal. The external dose commitments to workers would increase slightly because of periodic inspections of the waste and perhaps some handling, but this incremental dose would not be significant. Emergency response capabilities already in place would be adequate for any additional LLW storage capacity that may be required.

6.4.4.3 LLW Disposal

During the 20-year period for which the renewed licenses are granted, most utilities may have uncertain access to currently operating disposal facilities. Beatty closed

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in 1992, and Barnwell was closed to noncompact facilities in June 1994 but made available again in July 1995. The Hanford disposal site is limited to the Pacific Northwest and Rocky Mountain compact states. The additional quantities of LLW resulting from refurbishment and extended power plant operations under renewed licenses will be stored on site or shipped to existing or future disposal facilities.

Three issues associated with off-site disposal may be faced by LLW compacts, compact host-site disposal states, and unaffiliated states during license renewal. First, although routine waste stream volumes continue to decline as VR measures are implemented, major refurbishments that occur during license renewal could produce additional short-term volumes that could tax available off-site disposal space in some compact and unaffiliated states. Although unlikely, refurbishments for some nuclear power plants could be concentrated in a 1-year period before license expiration, thereby exceeding the waste volume acceptance criteria established by the host state.

Most compacts, however, are expected to develop facility design specifications that can encompass multiple, simultaneous refurbishment activities. For example, the Southeast Compact assumes an initial licensing volume of 11 million ft^3 (311,500 m^3) that will be disposed of in its North Carolina site scheduled to be operational in mid-1998. Current Southeast Compact policy limits the annual volume of waste to be received at this facility to 1.6 million ft^3 (45,300 m^3) per year for all categories of waste generators.

Assuming that a major refurbishment (involving the replacement of a steam generator) would generate about 45,000 ft^3 (1,275 m^3) of LLW and that annual regional waste disposal volumes would average 350,000 ft^3 (9,900 m^3) for all categories of generators, it would take about 28 major refurbishments to reach the 1.6 million ft^3 (45,300 m^3) annual ceiling for the Southeast Compact. Because it is unlikely that this many refurbishments will take place in any one year, disposal capacity problems are not likely to surface (data for scenario provided by D. G. Ebenhack Clean-Nuclear Systems, Inc., in letter to J. MacMillan, North Carolina Low-Level Radioactive Waste Management Authority, July 21, 1993; SCCLLRWM 1994). In situations where there is the likelihood that the specified annual compact waste acceptance ceilings will be reached, refurbishment could be staggered over several years. The expected waste from refurbishment is significantly less than the waste generated by decommissioning, which ranges from 650,000 ft^3 (18,400 m^3) for a PWR to 672,000 ft^3 (19,000 m^3) for a BWR (SCCTAC 1994). If 12 nuclear units in the Southeast Compact are decommissioned between 2009 and 2017, approximately 7,800,000 to 8,064,000 ft^3 (220,900 to 228,300 m^3) of LLW would be generated. This amount of waste can still be accommodated within the maximum operating life of a facility with a capacity of 32 million ft^3 (906,000 m^3), depending on the future volumes of LLW produced by other categories of generators.

Second, some host states have advised waste generators that they will be responsible for their own interim storage until disposal facilities can be opened; and those facilities may not open until *some* current licenses[1] have expired. However, for most nuclear power plants, new LLW disposal facilities are scheduled to open

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well before the expiration date for current licenses. An analysis of the expiration dates of nuclear power plant licenses and the expected dates for new LLW waste facilities to be available[2] reveals that existing power plant licenses will expire, on average, about 19 years after planned LLW facilities are currently scheduled to begin operations. Nevertheless, any nuclear power plant can start the license renewal process after 20 years of operation; consequently, the flexibility afforded by the 19-year cushion may be more apparent than real.

Third, an agreement state under the Low-Level Radioactive Waste Policy Amendments Act of 1985 may use averaging criteria that are different from those of the NRC to determine if a particular container of LLW is class C or GTCC wastes. Waste exceeding class C limits may be unacceptable for disposal in a licensed facility (Hutchison and Magleby 1990; Newberry and Coleman 1990). If a particular host state classifies waste as GTCC, it may refuse to accept that waste for disposal. This would increase the volume of GTCC that must be stored on-site by nuclear power plant operators until DOE provides disposal capacity at an HLW repository or other licensed facility.

The environmental impacts associated with LLW disposal during the terms of a renewed license of nuclear power plants should not be different in kind or magnitude from that during the terms of the initial 40-year license. The disposal facilities would be licensed and in compliance with appropriate regulations (10 CFR 61). The waste generators would have to meet the packaging and waste acceptance criteria for the specific disposal facility. Thus, measures would be in place to ensure that the performance objectives for the facility are met and that public exposures will be within regulatory limits.

6.4.4.4 Regulations Applicable

10 CFR Parts 20, 60, 61, and 62. 10 CFR 50.59.

6.4.4.5 Impacts of Extended On-Site Storage of LLW

6.4.4.5.1 Introduction

Preceding sections have discussed LLW from refurbishment and continued operations and presented the more likely events and impacts. The earlier discussion indicated that LLW treatment and disposal capacity are expected to become available before or during the license renewal period, although delayed from the Congressional timetables in the LLW statutes. This additional section separately addresses the more unlikely scenario of on-site storage of both refurbishment and operational low-level radioactive wastes (LLW) for the renewal period of approximately 20 years. Summary data are provided and radiological and nonradiological impacts are addressed. Radiological impacts to members of the public and workers are considered.

First, the nature of the problem is reviewed. As discussed in Chapter 3 and Appendix B, the refurbishment of the plants will produce additional LLW. For the typical case, increased "as-shipped" volumes for a BWR are estimated to be 220 m^3/year, and for a PWR 170 m^3 (Table B.4 of Appendix B). For the conservative case, higher volumes are estimated because of more extensive refurbishment. Major refurbishment activities are replacement of recirculating piping at BWRs and steam generator

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replacement for a PWR. The conservative case estimate for refurbishment waste is 1900 m^3 for a BWR and 2500 m^3 for a PWR. The analysis of refurbishing impacts included preparing wastes for shipment during the outages (e.g., see B.3.2.3 of Appendix B) but did not include inspection and any additional treatment or repackaging activities that might occur with extended on-site storage; however, on-site storage of steam generators was assumed and analyzed. Annual operating LLW that would be generated during the license renewal term is estimated to be about the same as current levels: 560 m^3 for BWRs and 250 m^3 for PWRs for the typical case (see Section 2.6.3.2) and for the conservative case, 11 percent higher volumes for BWRs and 30 percent higher for PWRs (see Section 2.6.4.2). For the typical case, the maximum stored volumes for the license renewal period, assuming no off-site capacity becomes available, would be about 20 times the annual value, or 11,200 m^3 for BWRs and 5,000 m^3 for PWRs. As Table 6.6 shows, based on historical data and DOE projections, the accumulated activities should remain about constant when decay is taken into account.

Extended storage is covered by the existing regulatory framework. Long-term storage of LLW at reactor sites has become necessary because of the slow pace of development of new off-site disposal facilities. In addition, utilities have opted for on-site storage for economic or other reasons. Utilities also store LLW that has been shipped off site for treatment (e.g., compaction, incineration) at commercial treatment facilities and returned for extended storage. Before licensees can build new storage facilities or make changes to the design or operation of the facility as described in the Safety Analysis Report (SAR), they must perform written safety evaluations under 10 CFR 50.59. This requirement applies to activities related to LLW, including long-term storage of LLW and mixed LLW. Under 10 CFR 50.59, licensees are allowed to make changes to their facility without permission of the NRC if the evaluation indicates that a change in the technical specifications is not required or that an unreviewed safety question does not exist. Licensees would have to ensure that the new LLW activities would not represent an unreviewed safety concern for routine operations or because of potential accidents. Both on-site and off-site impacts would have to be considered. If the LLW or mixed-LLW activity fails either of these tests, a license amendment is required. Thus, extended storage would be evaluated by the licensee, subject to inspection by NRC, or approved by NRC. Both licensee and NRC evaluations would include evaluation of anticipated compliance with applicable standards and requirements.

6.4.4.5.2 LLW Off-Site Radiological Impacts to Members of the Public

The storage of LLW is subject to several regulatory requirements related to potential public exposures. An overview of the regulatory requirements is given in Sections 3.8.1.1 and 6.2.2.5 and in Appendix E, but is repeated here. The basic provisions of 10 CFR Part 20 apply to all activities at the site. Part 20 contains both occupational and public dose limits, requirements for radiation safety programs that keep occupational and public doses and releases ALARA, survey and monitoring requirements, and reporting and record-keeping requirements. Part 20 also incorporates 40 CFR 190 [10 CFR 20.1301(d)], the EPA's general environmental standards for the uranium fuel cycle. NRC implements and enforces

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the limits in 40 CFR 190, which cover storage of LLW at the reactor site. The standards in 40 CFR 190 apply to the combined impacts from all uranium fuel cycle facilities and are expressed as annual dose limits for individual members of the public and annual quantity limits on certain radionuclides.[3] Therefore, any LLW activities should not result in doses and releases that would result in the site's failing to meet Part 190. Licensees are further limited by technical specifications to ensure compliance with the design objectives in Appendix I to 10 CFR Part 50. Appendix I design objectives are fractions of the limits in 10 CFR Part 20 and 40 CFR Part 190 and the dose range for doses to the whole body and organs is from 3 to 20 mrem/year. The numerical objectives in Section II of Appendix I state that:

> A. The calculated annual total quantity of all radioactive material above background [Footnote text: Here and elsewhere in this appendix background means radioactive materials in the environment and in the effluents from light-water-cooled power reactors not generated in, or attributable to, the reactors of which specific account is required in determining design objectives]. to be released from each light-water-cooled nuclear power reactor to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 3 millirems to the total body or 10 millirems to any organ.
>
> B.1. The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation.
> 2. Notwithstanding the guidance of paragraph B.1.
> (a) The Commission may specify, as guidance on design objectives, a lower quantity of radioactive material above background to be released to the atmosphere if it appears that the use of the design objectives in paragraph B.1 is likely to result in an estimated annual external dose from gaseous effluents to any individual in an unrestricted area in excess of 5 millirems to the total body; and
> (b) Design objectives based upon a higher quantity of radioactive material above background to be released to the atmosphere than the quantity specified in paragraph B.1. will be deemed to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as in reasonably achievable if the applicant provides reasonable assurance that the proposed higher quantity will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 millirems to the total body or 15 millirems to the skin.
>
> C. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form above background to be released from each light-water-cooled nuclear power reactor in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such

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radioactive iodine and radioactive material in particulate form for any individual in an unrestricted area from all pathways of exposure in excess of 15 millirems to any organ.

D. In addition to the provisions of paragraphs A, B, and C above, the applicant shall include in the radwaste system all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. As an interim measure and until establishment and adoption of better values (or other appropriate criteria), the values $1000 per total body man-rem and $1000 per man-thyroid-rem (or such lesser values as may be demonstrated to be suitable in a particular case) shall be used in this cost-benefit analysis.

The requirements of this paragraph D need not be complied with by persons who have filed applications for construction permits which were docketed on or after January 2, 1971, and prior to June 4, 1976, if the radwaste systems and equipment described in the preliminary or final safety analysis report and amendments thereto satisfy the Objectives on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Docket-RM-50-2 dated February 20, 1974, pp. 25–30, reproduced in the Annex to this Appendix.

Reactor licensees conduct and are required to conduct extensive monitoring and surveillance programs to demonstrate compliance with the limits in the regulations and in technical specifications. Releases and direct radiation from LLW activities, including storage, would be evaluated and included in demonstrating compliance with these standards, which apply to the exposures from all activities at the site combined. The effectiveness of licensee ALARA and compliance efforts in response to these regulatory requirements are demonstrated by the low average doses received by members of the public from reactor operations. Inspection data since 1982 shows that effluents and direct radiation dose rates continue to decline. As doses to members of the public are calculated from this information, it is reasonable to assume that public doses have continued to decline as well.

Appendix E of this document presents historical data on effluents and doses to members of the public. While those who live nearest to NRC-licensed fuel-cycle facilities are in principle allowed to receive up to the 10 CFR Part 20 limit of 100 mrem/year, modified by the 25/75/25 mrem/year dose limits of 40 CFR 190, most receive only a small fraction of the allowable exposure. The ALARA programs in place to supplement the dose limit result in a system of dose control which achieves doses significantly below the limits. As a consequence of this approach, the average dose to most members of the public from NRC-licensed power reactor facilities is well below 1 mrem/yr (e.g., see Table 4.6, which shows that even for maximally exposed individuals, there are few instances of doses above 1 mrem/year, and Table 4.9, which presents 1988 data), and the contributions to this average dose from

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LLW storage activities are significantly less than this average.

The effectiveness of controls at power reactors was also confirmed in EPA's proposed rule to rescind Subpart I of EPA's Clean Air Act regulations in 40 CFR 61 (56 FR 37196; August 5, 1991). The annual dose limits in 40 CFR 61 are 10 mrems total effective dose equivalent, of which no more than 3 mrems effective dose equivalent can be from radioiodines. EPA stated:

> Upon reconsideration of the standard, EPA conducted a review of the nuclear power reactor sector of the uranium fuel cycle and determined that the individual doses associated with nuclear power reactors are even lower than was previously estimated. This latest analysis revealed that the most exposed individual from emissions of nuclear power plants would be expected to receive a dose of less than 1.0 mrem/year EDE (effective dose equivalent) from all radionuclides and a dose of less than 0.01 mrem/year EDE from radioiodine. The estimated doses for these facilities are a factor of 10 less than the standard and are likely to remain low in the future.

Section 3.8 includes occupational and effluent data from actual steam generator replacements. The conclusion in 3.8.1.5 is that the effluents and doses have not been seen to differ significantly from normal operations. It is reasonable to conclude, then, that the incremental effluents and associated potential doses should be negligible from wastes after placement in storage facilities and/or packaging. Section 3.8.1.6 includes an estimate of 0.1 mrem/year as a maximum dose to an off-site person due to direct gamma dose from stored steam generators. This estimate should bound the potential public doses from on-site storage of refurbishment wastes. Section 3.8.1.6 also indicates that past storage facilities for the generators have provided sufficient shielding to limit the dose rate to less than 1 mrem/h outside the building.

Given its experience in inspecting licensees and in making determinations regarding compliance with existing requirements in this area, NRC has found that the actual doses and releases from LLW storage at plants have fallen within the applicable standards discussed above. Based on this past experience and the fact that NRC's regulatory program will continue to require compliance with the applicable regulations, including ALARA, NRC expects that the radiological impacts from LLW storage resulting from license renewal will neither deviate significantly from the kinds of impacts identified in the past nor exceed current regulatory requirements. NRC believes that doses and releases from LLW storage that fall within the range of current regulatory requirements should be considered small. The expected impact from on-site storage facility radiological effluents have been demonstrated to be a small fraction of those impacts allowed by regulation (Appendix I and 40 CFR 190).

6.4.4.5.3 LLW—Occupational

Tables B.4 and B.5 of Appendix B show total incremental occupational doses for refurbishing a BWR and a PWR. For the typical case, the values are 457 man-rem (4.57 man-sieverts) and 261 man-rem (2.61 man-sieverts), respectively. For the conservative case, the estimates are 2666 and 2374 man-rem (26.66 and 23.74 man-sieverts), respectively. These two sets of estimates compare to the actual exposure

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ranges for refurbishing projects mentioned in Section 3.8.2.2 of 2 to 3500 man-rem. Baseline data through 1992 is presented in Table 3.11 for the collective occupational dose per plant and average individual whole body dose. Anticipated average individual doses for refurbishing activities are based on experience in the early 1980s when significant post-TMI refurbishment took place and are estimated to be between 0.4 and 0.8 rem; the 1992 average dose was about 0.3 rem. It is reasonable to assume that the doses due to emplacement of the steam generators or piping or other LLW into on-site storage would be undetectable in view of the nature of the activities and the range of uncertainty in estimating doses. Doses from inspection in storage and further handling after significant decay should be similarly undetectable. Assuming the continued application of ALARA to mitigate and reduce occupational exposures, the staff is unaware of any reason that average occupational exposures from on-site storage of refurbishment LLW would not also be well within regulatory limits.

For routine operations, Section 4.6.3 presents baseline data and projected doses for license renewal. Baseline data (1992) includes a 0.28-rem average occupational dose and the fact that less than 0.5 percent of the workers received doses in excess of 2 rem. As plants age, slight increases in inventories and added maintenance, testing, and inspection would result in slight increases in occupational doses. Doses are projected to increase by 5 percent for the typical case and 8 percent for the conservative case. Considering the range and uncertainties of doses and the projected annual increase, occupational doses during the license renewal term are estimated to remain well within current regulatory limits (Section 4.6.3.3). The staff does not believe that on-site occupational exposures from LLW extended storage activities would be detectable in view of the range of doses and associated uncertainties.

Assuming the continued application of ALARA to mitigate and reduce occupational exposures, staff is unaware of any reason that occupational exposures from extended on-site storage of operational LLW would not continue to be well within current regulatory limits.

6.4.4.5.4 Nonradiological Impacts

Potential nonradiological impacts to be considered for extended on-site storage of LLW are the same as those considered for refurbishment in Sections 3.1 through 3.6 and include land use, fugitive dust, air quality impacts, erosion, sedimentation, and disturbance of ecosystems. Section 3.2 indicates that land use during a recent steam generator replacement was about 1 ha (about 2.5 acres) and that up to 4 ha (10 acres) may be needed. This calculation included training areas and other operational needs in addition to temporary storage of the steam generator. This disturbed area might be used for extended storage, or more remote locations on the site may be used to keep occupational exposures ALARA. Only a fraction of the area should be needed for extended storage of refurbishment or operational LLW. Earlier in this chapter, areas are estimated to be a few tenths of an acre for a storage facility and a few acres for a buffer zone around the facility, based on current experience at several reactors. The facilities might need to double or quadruple the storage volumes for the operational waste during the renewal term, but land use would still be small. Any land

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used would already be under the control of the utility.

Section 3.3 concludes that the small size of the area to be disturbed for refurbishing activities and the likely mitigating management practices should result in minimal fugitive dust. Because the amount of land for extended storage facilities should be even smaller and the mitigating practices should apply, fugitive dust is not a concern for storage. Section 3.3 concludes that vehicle exhaust emissions could be a concern for a large number (e.g., 2300) of additional worker vehicles in nonattainment zones. Extended on-site storage facility construction would involve far fewer workers, and inspection and maintenance even fewer. As noted in Section 3.4.1, only modest amounts of site excavation and grading should be involved in construction of any LLW storage facility, and no unusual practices are involved. Mitigating measures routinely practiced at the plants should mitigate surface water impacts, including erosion and sedimentation. Ecological impacts could be associated with any new construction at a site, including LLW storage facilities. However, with the small size of the facilities and the flexibility of location for extended storage facilities (as opposed to short-term storage near the plant during refurbishing), it is expected that ecological concerns will be routinely addressed by the applicant in the design of the project. Routine monitoring and inspection of LLW extended storage facilities during the license renewal period should not have any significant nonradiological impacts.

6.4.4.6 LLW Conclusions

The comprehensive regulatory controls that are in place and the low public doses being achieved at reactors ensure that the radiological impacts to the environment will remain within regulatory standards and therefore will be small during the term of a renewed license. The maximum additional on-site land that may be required for LLW storage during the term of a renewed license and associated impacts will be small. Nonradiological impacts on air and water will be negligible. The radiological and nonradiological environmental impacts of long-term disposal of LLW from any individual plant at licensed sites are small. The need for the consideration of mitigation alternatives within the context of renewal of a power reactor license has been considered, and the Commission concludes that its regulatory requirements already in place provide adequate mitigation incentives. In addition, the Commission concludes that there is reasonable assurance that sufficient LLW disposal capacity will be made available when needed for facilities to be decommissioned consistent with NRC decommissioning requirements. LLW storage and disposal will have small environmental impacts. This is a Category 1 issue.

Although the impacts of limited and extended on-site storage of LLW that would be generated during the renewal period have been evaluated and found to be small, concern has been expressed that the slow progress that has been made in developing disposal capacity for LLW could result in extended storage of LLW at nuclear power plants. NRC recognizes that no state or compact has completed its work in developing new LLW disposal facilities. However, in NRC's view, there are no unsolvable technical issues that will inevitably preclude successful development of new sites or other off-site disposal capacity by the time they will be needed. NRC's experience in developing the

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requirements and guidance for licensing LLW disposal facilities under 10 CFR Part 61, as well as the successful licensing of the Envirocare disposal facility by the state of Utah, support the conclusion that safe LLW disposal is technically feasible. Opening the Barnwell site to all states but North Carolina in July 1995 also reflects the lack of insolvable technical issues. There are uncertainties in the licensing process and in the length of time needed to resolve technical issues, but we would expect that they are resolvable. For example, in California, the proposed Ward Valley disposal facility was unexpectedly delayed by the need to resolve technical issues raised by several scientists independent of the project after the license was issued. These were recently reviewed and largely resolved by an independent review group. In North Carolina, Texas, and Nebraska, the license application review period has been longer than what is required by the Low-Level Radioactive Waste Policy Amendments Act of 1985, but many issues for each of the proposed facilities have been resolved, and progress on other issues continues to be made. Further, it should not be unexpected today that states and compacts would face difficult obstacles of a political and legal nature as they seek to fulfill their responsibilities for providing disposal capacity. Nonetheless, we believe that the states and compacts possess the determination and the processes to address these obstacles, and we are not prepared to say they will be unsuccessful in doing so; on the contrary, their progress, although slow, supports our conclusion of eventual success. Therefore, the staff conclusion that either on-site or off-site storage of LLW as a Category I issue is appropriate because states are proceeding, albeit slowly, with the development of new disposal facilities and because LLW has been and can be safely stored at reactor sites until new disposal capacity becomes available.

6.4.5 Mixed Waste

Mixed waste contains both hazardous waste and source, special nuclear, or byproduct material as defined in the Atomic Energy Act (AEA) of 1954 (42 U.S.C. 2011 et seq.). Although nuclear power plants, on average, are not significant generators of mixed waste, the management of this waste is problematic because of a lack of sufficient waste treatment and disposal capacity for specific types of mixed wastes. The current situation may be complicated by a lack of economic incentives (i.e., sufficient market demand for commercial mixed waste treatment) necessary to accelerate the development of new treatment or disposal capacity. Currently, there is only one facility providing disposal for certain types of mixed waste, while four other companies provide treatment for a limited number of mixed-waste streams. A lack of treatment capabilities and technologies, in combination with a complex regulatory system, makes the environmentally sound management of mixed-waste a significant challenge for all commercial mixed-waste generators, including nuclear power plant operators.

The management of mixed waste at nuclear power plants is jointly regulated by NRC under the AEA and by EPA or authorized states under the Resource Conservation and Recovery Act of 1976 (RCRA). The NRC or the NRC agreement states and EPA or EPA authorized states regulate off-site disposal. Nuclear power plants managing mixed waste must meet the NRC requirements for general radiation protection and emission control requirements and for LLW specified in

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10 CFR 61 and EPA's requirements for hazardous waste in 40 CFR Parts 261, 264, and 265 (DOE/RW-0006) before final transfer off site in route to burial. Mixed wastes are also subject to land disposal restrictions (LDRs) in 40 CFR 268, except for newly listed hazardous wastes that are mixed with radioactive material and do not yet have EPA standards. The requirement for treating specific hazardous constituents of mixed waste (chlorinated fluorocarbons, lead, etc.) before land disposal is a contingency not faced in the management of LLW.

6.4.5.1 Generation

U.S. commercial low-level mixed waste consists of a variety of waste streams from a diverse set of generators, including government, academic, and industrial sectors, as well as nuclear utilities and medical facilities. Mixed-waste generation in the United States for 1990 was estimated at 139,441 ft^3 (3,949 m^3), of which nuclear power plants produced about 13,626 ft^3 (385.8 m^3) (less than 10 percent).

Mixed waste generated by nuclear power plants covers a broad spectrum of waste types. As shown in Table 6.13, the vast majority of mixed waste in storage at nuclear power plants was chlorinated fluorocarbons (CFCs) and waste oil. These wastes represented approximately 40 percent and 23 percent of the total stored mixed waste, respectively. In contrast to other commercial mixed-waste generators, nuclear power plants produce relatively small volumes of liquid scintillation fluids. Overall, mixed waste from nuclear power plants represented approximately 34 percent of the total mixed-waste volume in storage at the end of 1990. Table 6.13 is based on data from 76 of 78 nuclear facilities surveyed for the NRC and EPA (a facility may contain one or more reactors with common waste handling).

Based on data from the *National Profile on Commercially Generated Low-Level Radioactive Mixed Waste* (NUREG/CR-5938), mixed waste is not distributed uniformly among all nuclear power plants but is concentrated at a relatively few power plants. The average mixed waste generation in 1990 for the category "nuclear utilities" was 175 ft^3 (5 m^3) per facility. Twenty-four facilities reported no mixed-waste generation in 1990, while four facilities reported over 1000 ft^3 (28 m^3) of mixed-waste generation. One facility was responsible for approximately 24 percent of the mixed waste generated, while that facility and three others were responsible for over 60 percent of the mixed waste generated during 1990 (J. Klein, Oak Ridge National Laboratory, letter to L. N. McCold, Oak Ridge National Laboratory, July 3, 1993).

Variability in volume and type of mixed waste produced by nuclear power plants means that those plant operators that produce relatively large mixed-waste volumes annually may find it more difficult to comply with mixed-waste storage regulations. Specifically, the current EPA policy of ascribing a lower enforcement priority for violations of its storage prohibitions for LDR mixed waste expressly excludes generators that produce more than 1000 ft^3/year (28 m^3/year) of hazardous and mixed waste (L-S/488364).

6.4.5.2 Storage

The current lack of mixed-waste treatment and disposal capacity requires nuclear

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Table 6.13 Nuclear power plant mixed waste generation profile for 1990, in cubic feet

Hazardous stream	Amount generated in 1990	Amount treated on site in 1990	Amount treated off site in 1990	Amount generated in 1990 that cannot be currently treated	Amount in storage at the end of 1990
Organics					
Liquid scintillation fluids	11	0	4	0	168
Waste oil	4,709	4,326	562	303	5,061
Chlorinated organics	50	0	0	5	512
Fluorinated organics	0	0	0	0	0
Chlorinated fluorocarbons	3,679	118	12	889	8,600
Other organics	1,154	15	7	79	1,284
Metals					
Lead	1,231	0	8	123	4,451
Mercury	4	0	0	2	416
Chromium	254	138	0	38	757
Cadmium	8	3	0	0	11
Aqueous corrosives	156	24	0	23	361
Other hazardous materials	2,369	168	2,274	8	363
Total	13,625	4,792	2,867	1,470	21,984

Note: Treatment and storage data are not necessarily additive because waste in either category may have been generated before 1990. Mixed waste that currently cannot be treated represents waste that may be difficult or even impossible to dispose of because of a lack of acceptable treatment capability or disposal capacity.
Source: NUREG/CR-5938.

power plant operators to store much of their mixed waste on site. As noted above, only one company in the United States (Envirocare of Utah, Inc.) currently provides disposal capacity for certain types of mixed waste, while four companies have treatment capabilities for certain types of mixed-waste constituents. The joint EPA and NRC survey referenced above estimated a treatment capacity shortfall of at least 12,000 ft^3 (340 m^3) based on treatment demand in 1990. The shortfall particularly affects CFCs, solid lead, and mercury mixed-waste streams. In 1991, EPA officially recognized that a treatment shortfall exists for many commercial low-level mixed-waste streams, including those generated by nuclear power reactors. Subsequent interviews by EPA with waste treatment vendors revealed that there has been little change in the availability of treatment capacity since EPA announced in 1991 that it would not pursue civil fines for those mixed-waste generators where sufficient treatment capacity of LDR-prohibited waste was not available.

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Occupational exposures occur during the testing of mixed wastes (particularly decontamination wastes and ion exchange resins) to determine if constituents are chemically hazardous. A second occupational exposure impact of mixed waste is from on-site storage and handling (Rogers 1990). It has been estimated that the largest single exposures result from samples being collected when lead blankets have not been used to shield pipes and valves (Rogers 1990).

Occupational exposures from on-site storage have been shown to be reduced by the application of waste-minimization technologies and procedures (Rogers 1990). In addition, the potential for exposure can be reduced by remote sampling methods currently under development. Remote evaluation methods include classifying waste streams as mixed through the application of knowledge about processes that generate waste streams and substituting closed-circuit television using high-resolution monitors for weekly inspections by facility personnel (Rogers 1990). The latter method can determine if sufficient deterioration of a container has occurred to warrant proximate visual inspections.

Pursuing environmentally responsible management of mixed wastes is critical to minimizing occupational exposures as well as preventing waste from entering the accessible environment through various air and groundwater pathways. Specifically, records must be maintained identifying each physical location or unit where mixed waste is stored and identifying the method of storage [40 CFR 264.73(b) and 265.73(b)]. An inspection of these storage areas for compliance with applicable RCRA standards for storage methods, including an assessment of compliance with storage facility standards of 40 CFR 264 or 265 (interim status) should be performed regularly (see 40 CFR 264.15 and 265.15).

Facility owners/operators are required by RCRA regulations to maintain sufficient information to identify their mixed wastes. The information required includes RCRA waste codes for the hazardous components, the source of the hazardous constituents and discussion of how the waste was generated, the generation rate and volumes of mixed waste in storage, and any information relied upon to identify mixed wastes or make determinations that the wastes are prohibited by LDRs.

Finally, under RCRA regulations, each facility owner/operator is required to develop a waste-minimization plan that identifies process changes that can be made to reduce or eliminate mixed wastes, methods to minimize the volume of regulated wastes through better segregation of materials, and the substitution of nonhazardous materials. The plan must include a schedule for implementation, projections of volume reductions to be achieved, and assumptions that are critical to the accomplishment of the projected volume reductions (L-S/488364).

6.4.5.3 Disposal

There is currently only one facility that provides disposal capacity for certain types of mixed waste: Envirocare of Utah, Inc. Envirocare has a RCRA Part B permit from the Utah Division of Solid and Hazardous Waste, allowing the receipt, storage, and disposal of certain types of low-activity mixed wastes that are both radioactive and hazardous at its South Clive facility. The combination of stringent LDRs for hazardous waste constituents,

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the associated lack of treatment capacity for particular types of mixed wastes, and the absence of permitted disposal facilities all contribute to the need for many utilities to store their mixed wastes on site.

6.4.5.4 Effects of License Renewal

Mixed waste will continue to be generated by routine maintenance activities, refueling outages, health physics activities, and radiochemical laboratory activities both before and after the completion of license renewal. However, plant refurbishments and extended power plant operations are not expected to increase volumes of mixed waste generated significantly because of continued progress in reducing mixed-waste generation (Rogers 1990). Because refurbishments and nuclear power plant operations are conducted in compliance with applicable NRC and EPA regulations governing the storage and disposal of mixed wastes, exposures will be minimized (10 CFR 20; 10 CFR 61; 40 CFR 264 and 268).

The development and commercialization of noninvasive mixed-waste characterization technologies and treatment capacity would produce several benefits: it would reduce (1) the generation of secondary waste streams, (2) worker exposures to hazardous and radioactive materials, and (3) on-site inventories of untreated mixed waste. While there is reason to be optimistic that lower generation rates and new treatment capabilities will reduce on-site inventories, certain inventories will continue to grow because the relatively small amount of mixed waste generated across all generator categories has not provided sufficient economic incentives (i.e., market demand) required to stimulate a rapid expansion in treatment capabilities.

Despite the current lack of mixed-waste treatment and disposal capacity, new-mixed waste treatment and disposal capacity may still occur prior to license renewal activities. Specifically, DOE's need to develop extensive new mixed-waste treatment capabilities should benefit utilities requiring additional off-site treatment capabilities. DOE generates approximately 2,860,000 ft^3 (81,000 m^3) of mixed waste per year and has approximately 6,320,000 ft^3 (179,000 m^3) of mixed waste in storage, dwarfing the mixed-waste management requirements of other commercial generators (DOE/LLW-180). The mixed-waste inventory conducted by NRC and EPA has revealed that the characteristics of commercial mixed wastes are, for the most part, very similar to those produced by DOE. The development by DOE of new mixed-waste technologies and/or its willingness to accept nuclear utility low-level mixed waste for treatment and disposal could dramatically reduce on-site waste inventories associated with license renewal as well as produce significant economies of scale.

6.4.5.5 Regulations Applicable

NRC (10 CFR Part 20 and LLW requirements in 10 CFR Part 61) and EPA RCRA regulations.

6.4.5.6 Impacts of Extended On-Site Storage of Mixed Low-Level Waste

Preceding sections have discussed mixed waste from refurbishment and continued operations and presented the more likely events and impacts. The earlier discussion indicated that mixed-waste treatment and disposal capacity may become available before or during the license renewal period and that DOE acceptance of commercial

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LLW for treatment and disposal may provide relief. This additional section separately addresses the less likely scenario of on-site storage of both refurbishment and operational mixed LLW for the renewal period of approximately 20 years. Summary data are provided and radiological and nonradiological impacts are addressed. Radiological impacts to members of the public and workers are considered.

6.4.5.6.1 Mixed Waste Off-Site Radiological Impacts to Members of the Public

Mixed LLW generation is highly variable but projected to be about 5 m^3/year per plant, which is less than 3 percent of the LLW volumes (see Section 2.3.7.3 and the discussion in earlier in this chapter). Mixed waste is subject to additional regulatory requirements on containment. For example, RCRA hazardous regulations require maintenance of container integrity, berms, and other catchment means for capturing leaks to prevent or minimize releases of hazardous materials to the environment. Based on the results of the national mixed-waste profile discussed earlier, the predominant waste forms generated by the utilities were slightly contaminated waste oil (35 percent), chlorofluorocarbons (27 percent) and others (38 percent). The Rogers report (Rogers 1990) evaluated potential mixed-waste forms, including three types of resins, sludges, dry active waste, and absorbed liquids. While all mixed waste generated might be stored on site for the license renewal period if adequate treatment and disposal capacities or DOE acceptance of commercial mixed waste are delayed until near the end of the renewal period, the accumulated volumes will be small when compared to LLW volumes. Incremental effluents and doses to members of the public should be minimal and are subject to the same regulatory limits and enforcement as LLW and are included in the overall facility performance findings.

Off-site disposal impacts, as well as the impacts of limited and extended on-site storage of mixed waste that would be generated during the renewal period have been evaluated and found to be small. However, concern has been expressed that the limited progress that has been made in developing disposal capacity for LLW and mixed waste could result in extended storage of mixed waste at nuclear power reactors. Mixed-waste-disposal facility developers face the same types of legal and political challenges as LLW site developers. In addition, the administrative uncertainties of joint regulation and the economics of developing treatment and disposal capacity for the small volumes of mixed waste that are generated at licensed facilities have proven to be disincentives to the development of mixed-waste-disposal facilities.

In NRC's view, however, there are no technical reasons why off-site disposal capacity for all types of mixed waste should not become available when needed. NRC and EPA have developed guidance on the siting of mixed-waste-disposal facilities as well as a conceptual design for a mixed-waste-disposal facility. The agencies are currently cooperating on developing additional guidance on testing and storage of mixed waste. A disposal facility for certain types of mixed waste has been developed by Envirocare in Utah. Depending on the characteristics of the mixed waste, on-site or off-site treatment may allow disposal of certain mixed wastes as purely LLW. As discussed above, DOE is working to establish treatment

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technologies for its mixed wastes, many of which have characteristics similar to commercial mixed waste, and EPA is issuing treatment standards that will permit mixed wastes to be land disposed. In NRC's view, the foregoing activities support the conclusion that safe disposal of mixed waste is technically feasible. Further, states have begun discussions with DOE about accepting commercial mixed for treatment and disposal at DOE facilities. Although these discussions have yet to result in DOE's accepting commercial mixed waste at DOE facilities, it appears that progress is being made towards DOE's eventual acceptance of some portion of commercial mixed waste at its facilities.

Given the technical feasibility of mixed-waste disposal, the states' responsibilities for providing LLW (and thus mixed-waste) disposal capacity and DOE's obligations under the FFCA to develop treatment and disposal capacity for its mixed waste, NRC believes that there will eventually be sufficient economic incentives to overcome nontechnical obstacles and to find cost effective ways to dispose of mixed waste. While the NRC understands that there have been some delays and that uncertainties exist, the staff concludes that there is reasonable assurance that sufficient mixed-LLW-disposal capacity will be made available when needed for facilities to be decommissioned consistent with NRC decommissioning requirements. Thus, in summary, mixed LLW will result in only a small environmental impact, taking into account both storage at a reactor site and disposal at an appropriate disposal site.

6.4.5.6.2 Occupational

Estimates of incremental occupational exposures from short-term and extended storage of mixed LLW have been made (Rogers 1990). The estimates were developed to evaluate ALARA problems for radiation exposures from compliance with RCRA sampling and inspecting requirements. When mixed LLW can be shipped immediately, doses from inspections were estimated to be about 3 man-rem per plant. With five years of accumulated mixed wastes, inspection exposures could rise to 100 man-rem/year per plant. Mitigating measures, including remote inspection, were considered essential to meet ALARA requirements. The doses in these estimates were based on assumed volumes and activities that should bound potential doses, since "these inventories are believed to represent conservatively high estimates of reactor-generated mixed wastes." (Rogers 1990). While sampling and handling were estimated to potentially result in significant doses in the 1990 study, absent ALARA mitigation such as use of lead blankets on contaminated piping with high exposure rates, they are included in current baseline exposures. The staff concludes that ALARA mitigating measures will continue to be developed and implemented by the utilities and RCRA regulatory authorities and that, even with the contribution of incremental occupational doses from extended storage, total individual occupational doses will continue to be will within regulatory limits and thus will be small.

6.4.5.6.3 Nonradiological

Because the volumes of mixed waste represent 3 percent or less of LLW volumes and because no significant emissions or releases of hazardous materials are expected, the staff concludes that the findings for LLW remain valid

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when both LLW and mixed-LLW impacts are considered.

6.4.5.6.4 Conclusion

The storage and disposal of mixed waste will continue to be accomplished well within regulatory limits. The comprehensive regulatory controls and the facilities and procedures that are in place ensure proper handling and storage, as well as negligible doses and exposure to toxic materials for the public and the environment at all plants. License renewal will not increase the small, continuing risk to human health and the environment posed by mixed waste at all plants. The radiological and nonradiological environmental impacts of long-term disposal of mixed waste from any individual plant at licensed sites are small. The need for consideration of mitigation alternatives within the context of renewal of a power reactor license has been considered and the Commission concludes that its regulatory requirements already in place provide adequate mitigation incentives for on-site storage of mixed waste and that, for off-site disposal, mitigation would be a site-specific consideration in the licensing of each facility. In addition, the Commission concludes that there is reasonable assurance that sufficient mixed-waste-disposal capacity will be made available when needed for facilities to be decommissioned consistent with NRC decommissioning requirements. The environmental impacts of mixed-waste storage and disposal will continue to be small during the license renewal period. This is a Category 1 issue.

6.4.6 Spent Fuel

Spent nuclear fuel is fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated. When spent fuel is removed from reactors, it is stored in racks placed in pools to isolate it from the environment and to allow the fuel rods to cool. Licensing plans contemplate disposal of spent fuel in a deep geological repository. Delays in siting an interim monitored retrievable storage (MRS) facility and permanent repository, as required by the Nuclear Waste Policy Act of 1982 as amended (Pub. L. 97-425 and 100-123), coupled with rapidly filling spent-fuel pools at some plants, have led utilities to seek means of continued on-site storage. These include expanded pool storage (through repacking and double tiering), above-ground dry storage, longer fuel burnup to reduce the amount of spent fuel requiring interim storage, and shipment of spent fuel to other plants. The total inventory of spent fuel in storage in the United States as of December 31, 1992, was 91,039 assemblies. Of these, 87,591 assemblies were in storage at 118 reactors that have been or are discharging and/or storing nuclear fuel assemblies, including 903 assemblies in independent spent-fuel storage installations (ISFSIs) at Virginia Power's Surry plant, Carolina Power and Light Company's Robinson 2 plant, and Duke Power Company's Oconee plant. An additional 3,448 assemblies have been shipped to away-from-reactor storage facilities. This compares to the total licensed capacity for storage of spent nuclear fuel in the U.S. of 205,731 assemblies (SR/CNEAF/94-01). This section addresses the availability of interim on-site storage capacity for spent fuel (until an MRS or permanent repository is

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available) and the potential environmental effects of that interim storage.

6.4.6.1 Baseline

DOE is responsible for taking possession of spent fuel from nuclear power plants in 1998 for interim storage in an MRS, followed by permanent disposal in an underground repository (Pub. L. 97-425; Pub. L. 100-123; Gerstberger; DOE March 1987; Parker; Bartlett). However, the original 1998 target date for opening the repository will not be met, and the availability of an MRS on that date is also in serious doubt. DOE now expects to complete site characterization work at Yucca Mountain, the only location being investigated as a permanent repository, by 2002 and expects that a geologic repository will be ready no sooner than 2010 (NWTRB 1993; DOE/RW-0307P-6). Many plants have limited in-pool storage capacities and are turning to fuel pool expansion, above-ground dry storage, and longer fuel (Gilbert et al. 1990). Industry-wide, 24 plants may run out of pool storage space by the year 2000, and 81 will have run out of pool storage space by 2010, if DOE is unable to accept spent fuel in an MRS or for disposal in a permanent repository (SR/CNEAF/94-01). Of the ten sample plants, three will have exhausted their pools by 2000. The projected year of pool storage space exhaustion for the ten sample plants is given in Table 6.14. Deferral of an MRS or permanent repository would necessitate longer at-reactor storage and would exacerbate current storage capacity limitations (SR/CNEAF/94-01).

Table 6.14 Projected year of pool storage space exhaustion for the ten sample plants

Plant	Year of storage space exhaustion
Hatch 1	2003
Hatch 2	2004
Limerick 1	2000
Limerick 2	1996
Vermont Yankee	2004
WNP-2[a]	1999
Comanche Peak 1	2020
Comanche Peak 2	2021
D. C. Cook 1 and 2	2011
Indian Point 2	2003
Indian Point 3	2006
Robinson	2002
San Onofre 2 and 3	2005
Surry 1	2012
Surry 2	2013

[a]WNP-2 = Washington Nuclear Project 2.

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Although plants running out of storage space may enter into agreements with others that have space for sale or lease, this approach is widely viewed as an interim measure practical only for utilities that own more than one nuclear plant (Asselstine 1985; DOE/RW-0187). Interim storage needs vary among plants, with older units likely to lose pool storage capacity sooner than newer ones. Robinson, for example, owned by Carolina Power and Light, has shipped some spent fuel to Shearon Harris, which is owned by the same utility. Transfer of spent fuel from one nuclear plant site to another requires authorization by the receiving plant's operating license (55 FR 29181).

Table 6.15 lists historic and projected trends for spent-fuel discharges and radioactivity levels for LWRs. Projections in Table 6.15 are based on the assumptions that (1) no new units will enter operation, (2) installed capacity will gradually decline, (3) no spent fuel removed from reactors will be reinserted for further irradiation later, and (4) average burnup rate of spent fuel at all LWRs will increase by nearly one-third by 2000 (DOE/RW-0006). In the conservative scenario depicted in Table 6.15, annual spent-fuel discharges are expected to decline for BWRs and PWRs early in the next century. However, total accumulated spent-fuel volumes will more than triple between 1990 and 2020. Thus, continued storage of spent fuel on site may be an issue for some utilities regardless of their license renewal plans. At-reactor pool storage capacity has been increased under original operating licenses through (1) enlarging the capacity of spent-fuel racks, (2) adding racks to existing pool arrays ("dense-racking"), (3) reconfiguring spent fuel with neutron-absorbing racks, and (4) employing double-tiered storage (installing a second tier of racks above those on the pool floor). Each of these methods requires both the repackaging of spent-fuel rods and the handling associated with fuel bundles and racks.

Zircalloy-clad fuel bundles do not appear to degrade as a result of long-term pool storage (Gilbert et al. 1990), and accidental damage to spent-fuel bundles through mishandling or component failure during emplacement or removal from pools has occurred infrequently. A few spent-fuel assemblies have been inadvertently dropped or mishandled. A small fraction of these assemblies has suffered major mechanical damage through such incidents. In most cases, when spent-fuel assemblies were damaged during handling (mostly during refueling operations, with only 10 percent occurring within the spent-fuel pool), only minor degradation of fuel-bundle components occurred. No cases of breaching of fuel cladding or release of radioactive gases or solids to the environment have been reported (EPRI NP-3765; Bailey 1990). Operational incidents involving spent-fuel pools have occurred infrequently. One incident, at Hatch in December 1986, took place during an exceptional handling procedure in a transfer canal between two pools. At Turkey Point, the failure of a circulation pump in August 1988 led to a breach of pool containment and the flow of water into a closed-loop canal, confining the radiation release on site. While the safety significance of both events appears to have been low, subsequent inspection and enforcement actions have been instituted by NRC to reduce the likelihood of such occurrences in the future (55 FR 38472). NRC has also found that, even under the worst probable cause of a loss of spent-fuel pool coolant (a severe seismic-generated accident causing a catastrophic failure of

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Table 6.15 Historic and projected spent-fuel inventories from commercial light-water reactors, 1970–2030 (not including license renewal)

Year	Fuel assemblies		Mass (MTIHM)[a]		Radioactivity (10^6 Ci)[b]	
	Annual	Total	Annual	Total	Annual	Total
			Boiling-water reactors			
Historic						
1970		6	16	1	11	
1971		64	80	190	197	
1972		142	222	431	466	
1973		95	317	349	441	
1974		245	561	908	1,042	
1975		226	787	920	1,218	
1976		297	1,084	1,151	1,581	
1977		383	1,467	1,566	2,129	
1978		383	1,850	1,618	2,412	
1979		400	2,250	1,734	2,728	
1980		620	2,870	2,685	3,888	
1981		459	3,329	2,014	3,664	
1982		357	3,686	1,582	3,362	
1983		491	4,177	2,218	4,015	
1984		498	4,675	2,211	4,283	
1985		515	5,190	2,246	4,519	
1986		458	5,648	1,963	4,404	
1987		699	6,347	2,919	5,411	
1988		536	6,883	2,363	5,177	
1989		715	7,598	3,090	6,038	
1990		633	8,231	2,821	6,101	
1991		588	8,819	2,696	6,186	
1992		729	9,547	3,359	7,037	
Projected						
1995	4,700	64,600	800	11,700	4,000	8,600
2000	3,900	82,400	700	14,800	3,300	9,100
2005	3,100	100,500	500	18,000	2,700	9,600
2010	3,800	120,500	700	21,500	3,200	11,100

See footnotes at end of table.

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Table 6.15 Historic and projected spent-fuel inventories from commercial light-water reactors, 1970–2030 (not including license renewal)

Year	Fuel assemblies		Mass (MTIHM)[a]		Radioactivity (10^6 Ci)[b]	
	Annual	Total	Annual	Total	Annual	Total
2015	2,100	139,600	400	24,800	1,900	10,800
2020	1,700	150,000	300	26,700	1,500	9,600
2025	2,200	162,000	400	28,800	1,900	10,000
2030	0	165,900	0	29,500	0	7,000
			Pressurized-water reactors			
Historic						
1970		39	39	204		204
1971		44	83	247		296
1972		100	183	545		638
1973		67	250	374		571
1974		208	458	1,098		1,320
1975		322	780	1,683		2,098
1976		401	1,181	2,222		2,894
1977		467	1,648	2,660		3,677
197[illegible]		699	2,347	4,030		5,428
1979		721	3,068	4,185		6,254
1980		618	3,686	3,667		6,248
1981		676	4,362	4,025		6,887
1982		640	5,002	3,797		7,037
1983		772	5,775	4,590		8,077
1984		842	6,616	4,978		8,943
1985		861	7,478	5,196		9,641
1986		1,001	8,478	5,969		10,909
1987		1,114	9,592	6,687		12,240
1988		1,125	10,717	6,865		13,132
1989		1,227	11,944	7,422		14,347
1990		1,532	13,476	9,405		17,026
1991		1,298	14,774	8,049		16,881
1992		1,601	16,375	10,032		19,374

See footnotes at end of table.

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Table 6.15 Historic and projected spent-fuel inventories from commercial light-water reactors, 1970–2030 (not including license renewal)

Year	Fuel assemblies		Mass (MTIHM)[a]		Radioactivity (10^6 Ci)[b]	
	Annual	Total	Annual	Total	Annual	Total
Projected						
1995	3,500	48,200	1,500	20,700	9,800	21,400
2000	3,300	63,400	1,400	27,300	9,400	23,700
2005	2,900	78,700	1,300	33,800	8,500	25,500
2010	2,500	93,600	1,100	40,200	7,400	26,900
2015	1,900	106,900	800	46,000	5,600	26,800
2020	1,600	116,000	700	50,000	4,800	24,900
2025	1,200	123,200	500	53,100	3,500	23,000
2030	300	127,000	100	54,800	900	18,000
		Total spent fuel (all light-water reactors)—projections				
1995	8,200	112,800	2,300	32,400	13,800	29,900
2000	7,200	145,800	2,100	42,100	12,700	32,800
2005	6,100	179,200	1,800	51,800	11,200	35,100
2010	6,400	214,100	1,800	61,700	10,600	38,000
2015	4,000	246,400	1,200	70,800	7,500	37,600
2020	3,300	266,000	1,000	76,700	6,300	34,500

[a]MTIHM = metric tons of initial heavy metal; 1 metric ton equals 2204.62 lb.
[b]Curies; 1 curie = 37×10^9 becquerels.
Source: DOE/RW-0006, Rev. 9.

the pool), the likelihood of a fuel-cladding fire is highly remote (55 FR 38474).

Inadvertent criticality and acute occupational exposure are remote risks of dense-racking (DOE/RW-0220). NRC requires licensees to ensure against inadvertent criticality in fuel storage facilities by limiting quantities of stored fuel and by regulating the configuration of fuel bundles (NUREG-0575; 10 CFR 50). The latter includes regulating proper spacing between spent-fuel assemblies and using boron carbide in storage racks (DOE/RW-0220).

Dry storage technologies such as casks, silos, dry wells, and vaults have been developed in conjunction with dry-rod consolidation (EPRI NP-3765; Gilbert et al. 1990; Schneider et al. 1992). Monitoring of occupational exposure in pilot studies of dry-rod consolidation indicates that, because of reliance on remote manipulation techniques, doses received by workers are similar to those from normal fuel movement, in-service inspection, and

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repair activities (Gerstberger 1987; Zacha 1988; Johnson 1989). In addition, dry storage generates no LLW. Ten countries have at least small amounts of spent nuclear fuel in dry storage, with Canada, the United Kingdom, and the United States having industrial-scale facilities (Schneider et al. 1992). Dry storage appears to be a safe, economical method of spent-fuel storage (Roberts 1987; Johnson 1989). Fuel rods in dry storage appear to be environmentally secure for long periods of time (EPRI NP-3765). Dry storage is also simpler and more readily maintained than spent-fuel pools (DOE/RW-0220; 55 FR 38472).

All U.S. commercial nuclear reactors that are storing or planning to store nuclear fuel assemblies in an ISFSI are covered in Table 6.14, which lists data for each of these utilities and affected reactors. Utilities are listed by the date the dry storage license was issued. Environmental assessments for operational ISFSIs at these plants (in a number of different regions) indicate that long-term material and system degradation effects are minimal and that licensees can ensure the use of such systems in full compliance with health, safety, environmental, and safeguards and security criteria (55 FR 29181).

The three utilities that currently use the Nutec Horizontal Modular Storage (NUHOMS) Spent Fuel Storage System are Baltimore Gas and Electric Company, Carolina Power and Light Company, and Duke Power Company. Both GPU Nuclear Corporation and Sacramento Municipal Utility District plan to employ the NUHOMS system. The system consists of three major safety-related components: a dry shielded canister (DSC), which provides a high-integrity containment boundary; a controlled concrete horizontal storage module (HSM), which houses the stored DSC and provides radiation shielding, protection against natural phenomena, and an efficient means for decay heat removal; and a transfer cask, which provides for the safe shielded transfer of the DSC from the plant spent-fuel pool to the storage module. The NUHOMS system is designed and licensed to meet the requirements of 10 CFR 72 and ANS/ANSI 57.9 for ISFSIs.

From the standpoint of emergency preparedness, the impacts of dry cask storage installations should be minor for three reasons. First, because of the reduced radioactive inventory in the fuel stored in dry cask facilities, accidents involving such storage facilities are likely to develop more slowly than those involving the nearby operating reactors. Second, accident impacts should be low, again because of the reduced inventories of radioactive materials in the stored fuel but also because of the correspondingly reduced level of decay heat compared with fuel still in-reactor. Thus, emergency plans formulated for operating reactors should encompass accidents at dry cask storage facilities. Third, it is NRC policy that plants with dry cask storage facilities incorporate the potential sources of hazard from these storage facilities in their emergency plans, as well as the potential hazard from all radiological source terms at the plant site.

Table 6.16 shows present and anticipated spent-fuel management methods in 8 of the 10 plants in the study sample. Practices in these eight plants are illustrative of industry-wide trends. While pool storage remains the most widespread method of spent-fuel management, dry storage and extended burnup are actively under development, mirroring national trends. NRC-licensed, full-scale demonstrations of

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Table 6.16 Spent-fuel management in eight sample plants and industry-wide: present and anticipated

Plant	Reracking[a]	Dry storage	Longer burnup[b]	Other[c]	Additional[d] construction
		Plant sample			
Hatch	Yes	No	Yes	No	No
Robinson	Yes	Yes	Yes	Yes	Yes
Indian Point 2	Yes	Under study	Under study	Yes	No
Surry	Yes	Yes[e]	Yes	No	Yes
Vermont Yankee	Yes	Under study	Yes	No	Yes
Limerick	Yes	Under study[f]	Yes	Yes	Yes
WNP-2[g]	No[h]	In planning	No	No	Yes
Cook	Yes	Under study[i]	Yes	No	Yes
		Industry-wide survey			
Industry-wide response, percent (n = 64)[j]	90.6	4.7	43.8	7.8	37.5
Industry-wide anticipated reliance on techniques until off-site disposal space becomes available, percent (n = 64)[j]	34.4	73.4	40.6	37.4	

[a]Indian Point 2's reracking is good through 2007; Vermont Yankee's, through 1998; and Limerick's, through 2011 and 2012 (Units 2 and 1, respectively). Cook reracked its spent-fuel pool in 1979 and plans to rerack its pool again in the 1993–1994 time frame. This is expected to yield sufficient storage until 2009.

[b]Surry and Vermont Yankee employ an 18-month fuel burnup cycle; Limerick is planning to be the first plant on a 24-month cycle.

[c]Robinson is planning transshipment, Indian Point will employ rod consolidation, and Limerick intends to employ a combination of high-enriched fuel, smaller reload batches, and rod consolidation.

[d]Robinson, Surry, Vermont Yankee, Limerick, Washington Nuclear Project 2, and Cook are planning either to build above-ground dry storage or to expand current storage facilities. Surry is building two additional storage "pads," and Limerick is planning a 165,000-ft^2 (15,300-m^2) facility for pool and dry storage.

[e]Surry's current dry storage facility will be full in 2010.

[f]Limerick will decide on the dry storage option in 2008.

[g]WNP-2 = Washington Nuclear Project 2.

[h]WNP-2 employs high-density racks.

[i]If pool storage proves insufficient for Cook after 2009, dry storage will be pursued.

[j]In updating data for the Final GEIS, the sample plants were resurveyed and published updates were used. However, no new industry-wide survey was undertaken because neither published updates nor sample plant data showed significant departures from previous trends.

Note: Of the 10 plants depicted in Section 6.1.1, Comanche Peak and San Onofre did not respond. Comanche Peak has not discharged spent fuel from its reactor as of February 1991 (SR/CNEAF/94-01). Because of multiple answers, percentages do not add up to 100 percent.

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dry storage techniques at two plants (Surry and Robinson 2) provide insight into measures taken to reduce worker and population exposures under current operations.

To meet the demand for additional storage space, Robinson has built an ISFSI with eight concrete HSMs to provide radiation shielding, protection against natural phenomena, and an efficient means of decay heat removal. The ISFSI is located inside the fence area of the Robinson 2 plant site. Each HSM is a steel-reinforced structure that holds seven intact assemblies in each module. The ISFSI was licensed by the NRC in August 1986.

Virginia Power was the first U.S. utility to use dry storage for spent nuclear fuel. The Virginia Power ISFSI located at the Surry Power Station, Surry, Virginia, houses metal storage casks. It was licensed by the NRC in July 1986. Each cask is 16 ft (4.9 m) high and 8 ft (2.4 m) in diameter, weighs 110 to 120 tons when loaded with fuel, and holds between 21 and 28 fuel assemblies. The casks sit on a reinforced-concrete pad 230 ft (70 m) long, 32 ft (9.7 m) wide, and 3 ft (1 m) thick. The facility and casks have been evaluated for extreme temperatures, extreme wind, snow and ice, loss of electrical power, loss of cask radiation shielding, tornadoes, gas pipeline explosions, and cask seal leakage and drops. By the end of 1990, a total of 252 assemblies had been stored in the ISFSI. By the end of 1991, 53 more assemblies were stored. In 1992, 63 more assemblies were stored, increasing the total number of assemblies in dry storage at Surry to 367 by the end of 1992. By 1995, an additional 250 assemblies will be in storage. The ISFSI has been licensed to hold up to 1764 assemblies.

Before these casks are placed into the ISFSI, they are filled with water and then submerged in the fuel pool to be loaded with spent-fuel assemblies (Godlewski 1987; Wakeman 1989). This procedure limits occupational exposure because the water is a radiation barrier. Individual and collective radiation doses to workers and the public are small (NRC Docket No. 72-2). Also, because the filling operation takes place within the pool containment area, contact with groundwater or surface water and other resources is also prevented. After filling, the casks are fastened with lids, water is pumped out, and the casks are backfilled with helium to prevent corrosion.

For the ISFSI facility itself, a few tenths of an acre are disturbed and occupied; in addition, a few acres are maintained for "intruder" exclusion or controlled access, as well as to limit worker dose. This additional acreage is still relatively small. At Surry, the ISFSI is designed to hold about 63 casks in an area of about 15 acres (6 ha), while Prairie Island will be able to store 48 casks on about 10 acres (4 ha) (Minnesota EQB 1991). Exclusion areas (included in these totals) usually occupy an already disturbed plant site and do not entail additional construction.

Longer fuel burnup reduces the volume of spent fuel removed from the core, deferring the need for additional storage space. An increase in fuel burnup to a maximum of 45 GWd/MTU for PWRs and 38 GWd/MTU for BWRs could halve the amount of spent fuel requiring off-site disposal (Gilbert et al. 1990). Increased burnup can also increase the specific activity of activation products in the radioactive waste system as well as fission products and transuranic-waste concentrations in plant waste streams

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(AIF/NESP-032; EPRI NP-5983). Extended burnup has not resulted in a higher incidence of failed fuel rods or breached cladding (EPRI NP-3765; SR/CNEAF/94-01). Several plants in the study sample are using or contemplating longer burnup (see Table 6.16).

Indian Point 2 is reracking its fuel pool for storage through 2007. Dry storage, rod consolidation, and longer burnup also will be considered. Vermont Yankee and Cook have reracked their pools to provide higher-density packing and are considering additional reracks. Limerick intends to rerack its pool to permit storage until 2011 at Unit 2 and until 2012 at Unit 1. If dry storage is undertaken, current economics favor the use of concrete casks at Limerick. If no repository is available after 2011, Limerick will employ a combination of dry storage and rod consolidation. Because of initial use of high-density fuel racks, WNP-2 plans no reracking. Surry's current ISFSI will be full by 2010, necessitating consideration of other options during the remainder of the plant's current license, including longer fuel burnup (the plant currently operates on an 18-month cycle) and possible construction of two additional storage pads for dry storage of spent fuel.

6.4.6.2 Effects of License Renewal

During the period encompassed by plant life extension, the amount of spent fuel generated annually by nuclear power plants will be a function of each plant's refueling schedule. The amount of spent fuel generated will be roughly proportional to the electrical energy produced by each plant. If all currently operating plants were to request renewed licenses, annual spent fuel generation should be comparable to those amounts generated under original licenses. Thus, total accumulated volumes of spent fuel after an additional 20 years of operation would amount to 50 percent more fuel than at the end of 40 years of operation (DOE/RW-0006). Projections of spent-fuel generation depicted in Table 6.15 are conservative estimates that do not account for nuclear plant life extension.

Under the Waste Confidence Rule, NRC has determined that spent fuel can be stored on-site for at least 30 years beyond the licensed (and license renewal) operating life of nuclear power plants safely and with minimal environmental impact (54 FR 39765; 55 FR 38472). This decision does not address the environmental impacts of storage during the additional 20 years of operation after license renewal. The additional spent fuel generated during this 20-year period poses three potential issues.

First, under the Nuclear Waste Policy Act of 1982 (NWPA) as amended, DOE is authorized to dispose of up to 70,000 metric tonnes of heavy metal (MTHM) in the first repository before granting a construction authorization for a second. Under existing licenses, projected spent-fuel generation could exceed 70,000 MTHM as early as the year 2010. Possible extensions or renewals of operating licenses also need to be considered in assessing the need for and scheduling the second repository. It now appears that unless Congress lifts the capacity limit on the first repository—and unless this repository has the physical capacity to dispose of all spent fuel generated under both the original and extended or renewed licenses—it will be necessary to have at least one additional repository. Assuming that the first repository is available by 2025 and has a capacity on the order of 70,000 MTHM,

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additional disposal capacity would probably not be needed before about the year 2040 to avoid storing spent fuel at a reactor for more than 30 years after expiration of reactor operating licenses.

Second, the NWPA prohibits the opening of an MRS until a permanent repository has been selected and constructed (Pub. L. 97-425). Moreover, the findings of environmental assessments for the MRS and permanent repository must be incorporated in facility design (DOE/RW-0187; GAO/RCED-90-103). Both of these requirements could cause additional delays in the availability of an MRS or permanent repository, necessitating longer on-site storage of the additional spent fuel. Current efforts to identify a host site for an MRS are unlikely to provide for a completed facility by 1998 (GAO/RCED-91-194).

Third, plant refurbishment during license renewal may also adversely affect spent-fuel storage capacity. Utilities may use fuel pools for interim storage of reactor components, as is being done at Vermont Yankee.

During the license renewal period, utilities will focus increasingly on dry storage methods for spent fuel. Either wet or dry storage would meet NRC's Waste Management Confidence Decision Review (49 FR 171; 10 CFR 50 and 51; 54 FR 187), but dry storage is growing in favor because it is more stable. Enlarging spent-fuel racks, adding racks to existing pool arrays, reconfiguring spent fuel with neutron-absorbing racks, and employing double-tiered storage will continue to be pursued; however, above-ground dry storage, utility sharing of spent fuel, and increased fuel burnup to reduce spent-fuel volumes will be the most favored methods until a permanent off-site repository or MRS becomes available, as shown by the study sample and industry-wide survey (Roberts 1987; Mullen et al. 1988; Zacha 1988; Johnson 1989; Fisher 1988).

Industry experience with spent-fuel storage, coupled with supplemental studies of the integrity of pool and dry storage systems, indicates that spent fuel generally can be stored safely on site with minimal environmental impacts (55 FR 38474; NUREG-1092). However, a maintenance concern with spent-fuel pools at permanently closed power plants was identified recently (*Nuclear Waste News* 1994). In January 1994, at the permanently shutdown (since 1978) Dresden Unit 1, a large amount of pool water leaked from a frozen service-water pipe located in the unheated containment building. Because the spent fuel had cooled for 15 years, lowering the pool water depth in this case did not cause significant increases in worker exposure. However, this incident has led to additional safety precautions' being implemented at all permanently shutdown plants.

Extended pool storage provides a benign environment that does not lead to degradation of the integrity of spent-fuel rods. Moreover, continuing advances in dry storage techniques, particularly in standardization of procedures and equipment, indicate that these systems are simple, passive, and easily maintained (53 FR 31651; NUREG-1092; Mullen et al. 1988).

For pool storage, while plant life extension could possibly increase the likelihood of inadvertent criticality through dense-racking or spent-fuel handling accidents, NRC regulations are in place to satisfactorily address this problem. In

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addition, studies of fuel rod or cladding failures indicate that fuel rods should remain secure well beyond the period of plant life extension, if it becomes necessary to continue pool storage on site (EPRI NP-3765; AIF/NESP-032; EPRI NP-5983; Bailey 1990; Gilbert et al. 1990; 55 FR 38474).

As a result of the operational experience demonstrated by Surry, Robinson, Oconee, and Ft. St. Vrain, NRC has determined that ISFSI methods of dry storage are sufficiently well developed, safe, and dependable to permit the generic licensing for any nuclear plant licensee (provided the plant licensee notifies NRC of the intent to use an ISFSI, uses NRC-certified casks, follows all specified conditions for their use, and provides a full description and safety assessment of the proposed site for an ISFSI) (55 FR 29181; 53 FR 31651). Worker and population exposures are minimal, and ISFSIs use only a small fraction of available land. Environmental assessments undertaken for all ISFSIs have resulted in issuance of findings of no significant impact (NRC Dockets 72-2, 72-3, 72-4, and 72-9).

The principal occupational exposures from spent-fuel management will occur during repackaging of spent-fuel rods and during construction and handling activities associated with moving and storing spent-fuel bundles and racks. While these impacts are expected to vary by the amount of fuel requiring storage, occupational doses during the period of license renewal are not expected to result in doses in excess of present levels (Section 4.6.3). Environmental impacts to on-site land availability should be minimal, given the small amount of land required for expanded spent-fuel pools and dry storage facilities.

6.4.6.3 On-Site Storage of Spent Fuel

Current and potential environmental impacts from spent-fuel storage have been studied extensively and are well understood. Storage of spent fuel in spent-fuel pools was considered for each plant in the safety and environmental reviews at the construction permit and operating license stage. The Commission has studied the safety and environmental effects of the temporary storage of spent fuel after cessation of reactor operation and published a generic determination of no significant environmental impact in its regulations at 10 CFR 51.23. The environmental impacts of storing spent fuel on site in a fuel pool for 10 years prior to shipping for off-site disposal were assessed and are included within the environmental data given by Table S-3, found in the Commission's regulations at 10 CFR 51.51. Environmental assessments (EA) for expanding the fuel-pool storage capacity have been conducted for more than 50 plants. A finding of no significant environmental impact was reached for each fuel-pool capacity expansion. Dry cask storage at an ISFSI is the other technology used for spent-fuel storage on site. The Commission has conducted EAs for seven site-specific licensed ISFSIs and has reached a finding of no significant environmental impact for each. The Commission has recently amended its regulations in 10 CFR 72 to allow power reactor licensees to store spent fuel on their sites under a general license. The environmental impacts of implementing this rule were analyzed in an EA that incorporated EAs performed for previous rulemakings related to 10 CFR 72 and for the Commission's Waste Confidence Decision.

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At the construction permit and operating license stage, both the 10 CFR 50 safety review and the 10 CFR 51 environmental review contributed to understanding the potential radiological and nonradiological environmental impacts of fuel-pool construction and operation. The design and operating conditions of spent-fuel pools and their various auxiliary systems were reviewed to ensure that the design criteria of Appendix A to 10 CFR 50 are met. These criteria address (1) control of releases of radioactive materials to the environment, (2) fuel storage and handling and radioactivity control, (3) prevention of criticality in fuel storage and handling, (4) monitoring fuel and waste storage, and (5) monitoring radioactive releases. These criteria ensure that radioactive releases to the environment are controlled and acceptable and that effluent discharge paths and the plant environs are monitored for radioactivity. Appendix I to 10 CFR 50 provides the numerical objectives for the design objectives and limiting conditions for operation required to meet the ALARA criterion for radioactive material in the total effluent from an LWR. The objectives were quoted earlier in this chapter and include an objective that total radioactive material in liquid effluent should not result in an annual dose or dose commitment to the total body or to any organ of an individual in an unrestricted area for all pathways of exposure in excess of 5 mrem. In addition, the calculated annual total quantity of radioactive material, except tritium and dissolved gases, should not exceed 5 Ci for each reactor at a site. Appendix I objectives for annual total gaseous effluent of radioactive material for all reactors at a site is that gamma radiation doses should not exceed 10 mrad and beta radiation doses should not exceed 20 mrad for an individual located at or beyond the site boundary.

Radioactive materials from the spent-fuel pool contribute a small fraction of the total radioactive materials released from a plant. It is the total releases that need to meet Appendix I numerical objectives. In the construction permit and operating license review for each plant, a thorough assessment is made of calculated releases of curies per year of radioactive materials in both liquid effluent and in gaseous effluent, the exposure pathways, and the impacts to man and biota other than man.

The Commission has considered whether radioactive wastes generated in nuclear power reactors can be subsequently disposed of without undue risk to the public health and safety and the environment. As stated in its regulations at 10 CFR 51.23:

> (a) The Commission has made a generic determination that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impact for at least 30 years beyond the licensed life for operation (which may include the term of a revised or renewed license) of that reactor at its spent-fuel storage basin or at either on-site or off-site independent fuel storage installations. Further, the Commission believes that there is reasonable assurance that at least one mined geological repository will be available within the first quarter of the twenty-first century, and sufficient repository capacity will be available within 30 years beyond the licensed life for operation of any reactor to dispose of the commercial high-level waste and spent fuel originating in such reactor and generated up to that time.

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In accordance with this determination the rule also provides that no discussion is required concerning environmental impacts of spent-fuel storage for the period following the term of the reactor operating license, including a renewed license. The waste confidence determination was first published in 1984 at 49 FR 34694, August 31, 1984 and was amended in 1990 at 55 FR 38474, September 18, 1990. Additional information and explanation of the safety and environmental considerations supporting the waste confidence determination are given in the notice of the proposed rule amendment, 54 FR 39767, September 28, 1989.

The environmental impacts of storing spent fuel on site in a fuel pool for 10 years prior to shipping for off-site disposal are incorporated in the data presented in Table S-3. The environmental impacts of storage of spent fuel in a fuel pool are given in Table 2.5 of NUREG-0116, *Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle*. Commitment of land, water consumption, chemical effluent, gaseous, liquid and solid radiological effluent, and thermal effluent are all negligible.

Since 1984, licensees have continued to provide safe and environmentally innocuous additional reactor-pool storage capacity through reracking. Over 50 reviews for the expansion of fuel-pool capacity have been completed by the Commission. Each review has resulted in a finding of no significant environmental impact. The reracking activities take place within existing structures and already disturbed land areas, and the changes in radiological, nonradiological, and thermal effluent are negligible.

Dry storage of spent fuel at ISFSI has been extensively studied by the Commission, and the environmental impacts are well understood. Licensing requirements for the independent storage of spent fuel and HLW are given in 10 CFR 72. In part, these regulations cover siting evaluation factors, general design criteria, general license for storage of spent fuel at power reactor sites, and approval of spent-fuel storage casks.

6.4.6.4 On-Site Dry Cask Storage

On-site dry cask storage of spent fuel can be accomplished either by a specific license issued under 10 CFR 72.40 or by the provisions of a general license issued under 10 CFR 72.210 for an ISFSI at operating power reactors. To date, seven specific licenses have been issued under 10 CFR 72.40 and one general license issued under 10 CFR 72.210 is operational. For each specific license the Commission has prepared an EA and a finding of no significant impact. Each EA addressed the impacts of construction, use, and decommissioning, including fugitive dust, erosion, noise, heat, and radiological impacts. The Commission also prepared an EA for the general license issued on July 18, 1990 (55 FR 29191). The Commission does not prepare an EA for each general licensee but does prepare an EA for each dry storage cask listed under 10 CFR 72.214 which is approved for use by general licensees. Currently seven casks are listed under 10 CFR 27.214 and it is anticipated that more will be added. General licensees can use only casks listed under 10 CFR 72.214.

EAs prepared for site-specific licenses include site description, need for action, alternatives, site and environment, description of the ISFSI, environmental

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impacts of proposed action, safeguards for spent fuel, decommissioning, and finding of no significant impact. Under the environmental impacts of the action, the following are considered: land use and terrestrial resources, water use and aquatic resources, noise and air-quality impacts of construction, socioeconomic impacts of construction, radiological impacts of construction, radiological impacts of routine operations, off-site dose, collective occupational dose, radiological impacts of off-normal events and accidents, land use and terrestrial resources, water use and aquatic resources, other effects of operation, and resources committed.

Using the Calvert Cliffs Nuclear Power Plant Site ISFSI EA as typical, the following impacts are evaluated. Land use is about six acres, which is within the owner-controlled area of 2300 acres. During construction of the pad, water for cleaning, drinking, and fugitive dust control was transported to the site by truck. Storm-water runoff and sediment were controlled according to local codes. Air quality had a temporary increase of suspended particulate material, hydrocarbons, carbon monoxide, and oxides of nitrogen from construction activities. The size of the work force was not expected to exceed 50 people. This expanded work force had little impact in the area with large population growth. During initial construction there were no radiological impacts. As construction proceeded, after filling some storage modules, radiation was controlled with temporary shielding to meet NRC and ALARA exposure requirements. Dry storage of spent fuel in welded canisters has no gaseous or liquid effluents. The exposure of the nearest resident, 4705 ft from the facility, when the facility is filled with design-basis spent fuel in 120 modules, the license limit, is less than one mrem/year. The exposure of that resident from other operations at the site is 13.5 mrem/year. These exposures are well within the requirements of 10 CFR 72.104 and 40 CFR Part 190 limits of 25 mrem/year. By year 2010 there are projected to be about 500 people living between 1 and 2 miles of the Calvert Cliffs Station. The collective dose is estimated to be about 101 man-rem/year. Occupational exposure in constructing additional modules after the initial set has been loaded is expected to total about 4 man-rem. Once all 120 modules are loaded, the radiation exposure from the ISFSI is expected to be less than 5 percent of the total site yearly exposure of 350 man-rem. Worst-case accident dose was calculated to be 23 mrem to the whole body and 111 mrem to the thyroid at the nearest residence. Heat from the modules is not expected to be high enough to affect vegetation growth. Fences will discourage some wildlife species from using the area adjacent to the modules. There is no planned use of water or liquid discharge to local surface or groundwater supplies. Surface runoff from precipitation will enter the Chesapeake Bay under existing drainage routes, but it is not expected to result in negative impact to water quality. Rain may vaporize and form a localized fog over the modules that would not extend beyond the plant exclusion boundary. Noise during construction and movement of fuel would not be distinguishable from other operational noise at the site or to result in adverse impact to local residents. The Commission believes that the impacts discussed above reasonably describe the impacts from existing dry cask storage facilities, as well as the likely impacts from those dry cask storage facilities that are expected to be constructed as a result of license renewal.

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The Commission prepares an EA for each approved cask listed in 10 CFR 72.214. These EAs are tiered off the "Final Waste Confidence Decision," August 31, 1984 (49 FR 34688), the *Environment Assessment for 10 CFR 72 "Requirements for the Independent Storage of Spent Fuel and High-Level Radioactive Waste,"* NUREG-1092 (August 1984), and the "Environmental Assessment for Proposed Rule Entitled 'Storage of Spent Nuclear Fuel in NRC-Approved Storage Casks at Nuclear Power Reactor Sites,'" for the proposed rule published on May 5, 1989 (54 FR 19379). Additional impacts evaluated are those associated with the construction, use, and disposal of the cask. These impacts are very small compared to the total impact of the steel industry, plastics industry, and the concrete industry. The incremental impacts of cask use are considered small. No effluents, either gaseous or liquid, are expected from the sealed casks. Incremental radiation doses off site are also considered to be small compared to those from the other operations on the site. Based on the above summary a finding of no significant impact is appropriate. This finding has been made for each of the seven casks listed in 10 CFR 72.214. Power reactor licensees using one of the listed casks under a general license do not need to prepare an environmental report, nor does the NRC have to prepare an EA.

6.4.6.5 Expanding Fuel-Pool Capacity

The Commission prepares an EA for each request to expand the capacity of a spent-fuel pool. The EA prepared for the increase in the allowed fuel assembly storage for the Pilgrim Nuclear Power Station is a typical example of this type of action. Alternatives looked at include (1) shipment of fuel to a permanent federal fuel-storage/disposal facility, (2) shipment of fuel to a reprocessing facility, (3) shipment of fuel to another utility or site for storage, (4) reduction of spent-fuel generation, (5) construction of a new independent spent-fuel storage installation, and (6) no action. After evaluating the alternatives, the proposed action of increasing the capacity of the spent-fuel pool is the best one at the time; however, in the longer term, an ISFSI is the solution. Radioactive exposures, waste generation, and releases were evaluated and found to be incrementally small. The only nonradiological effluent is additional heat rejected from the plant. This additional heat is small compared to the total rejected by the rest of the plant, and it will have a negligible effect on the environment. The risks due to accidents and their environmental effects are found to be not significant.

6.4.6.6 Regulations Applicable

10 CFR Parts 72, 60, and 61.

6.4.6.7 Conclusion

The Commission's waste confidence finding at 10 CFR 51.23 leaves only the on-site storage of spent fuel during the term of plant operation as a high-level-waste storage and disposal issue at the time of license renewal. The Commission's regulatory requirements and the experience with on-site storage of spent fuel in fuel pools and dry storage has been reviewed. Within the context of a license renewal review and determination, the Commission finds that there is ample basis to conclude that continued storage of existing spent fuel and storage of spent fuel generated during the license renewal period can be accomplished safely and without significant environmental impacts. Radiological

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impacts will be well within regulatory limits; thus radiological impacts of on-site storage meet the standard for a conclusion of small impact. The nonradiological environmental impacts have been shown to be not significant; thus they are classified as small. The overall conclusion for on-site storage of spent fuel during the term of a renewed license is that the environmental impacts will be small for each plant. The need for the consideration of mitigation alternatives within the context of renewal of a power reactor license has been considered, and the Commission concludes that its regulatory requirements already in place provide adequate mitigation incentives for on-site storage of spent fuel. On-site storage of spent fuel during the term of a renewed operating license is a Category 1 issue.

6.5 NONRADIOLOGICAL WASTES

Nonradiological wastes from routine plant operations include those from cooling system blowdown (continual or periodic purging of impurities from cooling systems), water treatment wastes (sludges and high-saline streams whose residues are disposed of as solid waste), boiler metal cleaning, floor and yard drains, storm-water runoff, sewage wastes, and cleaning solvents (NUREG-0020). Descriptions of these waste-generating systems are provided in Section 2.1.6. If nonradiological sanitary wastes cannot be processed by on-site water treatment systems, they are collected by independent contractors and trucked to off-site treatment facilities. If wastes have hazardous constituents, proper handling and disposal are required to minimize potential contamination of surface water and groundwater. In this section, a review of literature on nonradiological waste management throughout the industry was used to depict baseline conditions and to infer the effects of license renewal.

6.5.1 Baseline

Stringent regulations governing the generation of nonradioactive solid waste and the resulting efforts of utilities to establish waste minimization and pollution prevention programs are expected to produce a general decline in the general production of waste by nuclear power plants during the period prior to license renewal. Nonradioactive hazardous solid waste disposal from all nuclear power plants is governed by RCRA (Pub. L. 94-580). RCRA requires EPA and state agencies to establish a permit system for disposal of these wastes in licensed landfills. Utilities have undertaken changes in operation to ensure proper handling and disposal of these wastes in accordance with RCRA, including periodic removal of septic tank sludge by a licensed contractor and disposal on or off site in an approved sanitary system. Construction-related solid wastes are discharged to holding ponds until chemical discharges and runoff are suitable for discharge to surface waters on a batch basis. These latter discharges must comply with allowable standards under RCRA permits.

6.5.2 Effects of License Renewal

Solid nonradiological waste from blowdown, water treatment, boiler metal cleaning, floor and yard drains, storm-water runoff, and sewage wastes will likely remain of limited concern during license renewal for three reasons. First, no changes to the systems that generate these wastes are anticipated as a result of license renewal for all plants. Second, existing regulations, including National Pollutant

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Discharge Elimination System permitting for low-volume wastewater and RCRA permitting for solid wastes such as chemical solvents, are also likely to become increasingly stringent through further amendment (OTA-O-426). Third, statutorily mandated waste-minimization programs, which are expected to incorporate new waste-management technologies, should reduce further the volume of solid nonradioactive waste produced by nuclear power plants.

Some plants may require construction of interim storage facilities for LLW and spent fuel. Construction of these facilities would generate rubble and other debris on a short-term basis. This temporary increase of waste would be typical of that generated by any construction activity in an industrial complex and would be controlled by federal and state regulations. Hence, management of this waste stream would not pose any new or unique issues and would not be expected to result in impacts of concern.

6.5.3 Conclusion

Generation and management of solid nonradioactive waste during the terms of an extended license are not expected to result in significant environmental impacts. No changes to plant systems or mode of operation have been identified that would increase the quantities of waste generated or change the nature and types of waste in a manner that would be of environmental concern. In fact, regulatory and operational trends suggest a gradual decrease in quantities generated annually and the impacts during the terms of renewed licenses. Facilities and procedures are in place to ensure continued proper handling and disposal at all plants. Consequently, the generation and management of solid nonradioactive waste for up to 20 years beyond the terms of the original 40-year license of nuclear power plants is anticipated to result in only small impacts to the environment. Because the facilities and procedures that are in place are expected to ensure continued proper handling and disposal at each plant, additional mitigative measures are not a consideration in the context of a license renewal review. This is a Category 1 issue.

6.6 SUMMARY

The following conclusions have been drawn with regard to the environmental impacts associated with the uranium fuel cycle and with the management of waste generated during nuclear power plant operations beyond the terms of their original 40-year licenses.

- The radiological and nonradiological environmental impacts of the uranium fuel cycle have been reviewed. The review included a discussion of the values presented in Table S-3, an assessment of the release and impact of ^{222}Rn and of ^{99}Tc, and a review of the regulatory standards and experience of fuel-cycle facilities. For the purpose of assessing the radiological impacts of license renewal, the Commission uses the standard that the impacts are of small significance if doses and releases do not exceed permissible levels in the Commission's regulations. Given the available information regarding the compliance of fuel-cycle facilities with applicable regulatory requirements, the Commission has concluded that, other than for the disposal of spent fuel and high-level waste, these impacts on individuals from radioactive gaseous and liquid releases will remain at or

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below the Commission's regulatory limits. Accordingly, the Commission concludes that off-site radiological impacts of the fuel cycle (individual effects from other than the disposal of spent fuel and high-level waste) are small. ALARA efforts will continue to apply to fuel-cycle activities. This is a Category 1 issue.

- The radiological impacts of the uranium fuel cycle on human populations over time (collective effects) have been considered within the framework of Table S-3. The 100-year environmental dose commitment to the U.S. population from the fuel cycle, high-level-waste and spent-fuel disposal excepted, is calculated to be about 14,800 man-rem, or 12 cancer fatalities, for each additional 20-year power reactor operating term. Much of this, especially the contribution of radon releases from mines and tailing piles, consists of tiny doses summed over large populations. This same dose calculation can theoretically be extended to include many tiny doses over additional thousands of years as well as doses outside the U.S. The result of such a calculation would be thousands of cancer fatalities from the fuel cycle, but this result assumes that even tiny doses have some statistical adverse health effect that will not ever be mitigated (for example, no cancer cure in the next thousand years) and that these dose projections over thousands of years are meaningful. However, these assumptions are questionable. In particular, science cannot rule out the possibility that there will be no cancer fatalities from these tiny doses. For perspective, the doses are very small fractions of regulatory limits, and even smaller fractions of natural background exposure to the same populations. No standards exist that can be used to reach a conclusion as to the significance of the magnitude of the collective radiological effects. Nevertheless, some judgment as to the regulatory NEPA implication of this issue should be made, and it makes no sense to repeat the same judgment in every case. The Commission concludes that these impacts are acceptable in that these impacts would not be sufficiently large to require the NEPA conclusion, for any plant, that the option of extended operation under 10 CFR 54 should be eliminated. Accordingly, while the Commission has not assigned a single level of significance for the collective effects of the fuel cycle, this issue is considered Category 1.

- There are no current regulatory limits for off-site releases of radionuclides for the current candidate repository site. However, if we assume that limits are developed along the lines of the 1995 NAS report and that, in accordance with the Commission's Waste Confidence Decision, a repository can and likely will be developed at some site that will comply with such limits, peak doses to virtually all individuals will be 100 mrem/year or less. However, while the Commission has reasonable confidence that these assumptions will prove correct, there is considerable uncertainty because the limits are yet to be developed, no repository application has been completed or reviewed, and uncertainty is inherent in the models used to evaluate possible pathways to the human environment. The National Academy report indicated that 100

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mrem/year should be considered as a starting point for limits for individual doses, but notes that some measure of consensus exists among national and international bodies that the limits should be a fraction of the 100 mrem/year. The lifetime individual risk from 100 mrem/year dose limit is about 3×10^{-3}. Doses to populations from disposal cannot now (or possibly ever) be estimated without very great uncertainty.

Estimating cumulative doses to populations over thousands of years is more problematic. The likelihood and consequences of events that could seriously compromise the integrity of a deep geologic repository were evaluated by the Department of Energy in the *Final Environmental Impact Statement: Management of Commercially Generated Radioactive Waste*, October 1980. The evaluation estimated the 70-year whole-body dose commitment to the maximum individual and to the regional population resulting from several modes of breaching a reference repository in the year of closure, after 1,000 years, after 100,000 years, and after 100,000,000 years. The release scenarios covered a wide range of consequences from the limited consequences of humans accidentally drilling into a waste package in the repository to the catastrophic release of the repository inventory by a direct meteor strike. Subsequently, the NRC and other federal agencies have expended considerable effort to develop models for the design and for the licensing of a high-level-waste repository, especially for the candidate repository at Yucca Mountain. More meaningful estimates of doses to population may be possible in the future as more is understood about the performance of the proposed Yucca Mountain repository. Such estimates would involve very great uncertainty, especially with respect to cumulative population doses over thousands of years. The standard proposed by the NAS is a limit on maximum individual dose. The relationship of potential new regulatory requirements, based on the NAS report, and cumulative population impacts has not been determined, although the report articulates the view that protection of individuals will adequately protect the population for a repository at Yucca Mountain. However, EPA's generic repository standards in 40 CFR 191 generally provide an indication of the order of magnitude of cumulative risk to population that could result from the licensing of a Yucca Mountain repository, assuming the ultimate standards will be within the range of standards now under consideration. The standards in 40 CFR 191 protects the population by imposing "containment requirements" that limit the cumulative amount of radioactive material released over 10,000 years. The cumulative release limits are based on EPA's population impact goal of 1,000 premature cancer deaths worldwide for a 100,000-metric tonne (MTHM) repository.

Nevertheless, despite all the uncertainty surrounding the effects of the disposal of spent fuel and high-level waste, some judgment as to the regulatory NEPA implications of these matters should be made, and it makes no sense to repeat the same judgment in every case. Even taking the uncertainties into account, the

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Commission concludes that these impacts are acceptable in that these impacts would not be sufficiently large to require the NEPA conclusion, for any plant, that the option of extended operation under 10 CFR 54 should be eliminated. Accordingly, while the Commission has not assigned a single level of significance for the impacts of spent-fuel and high-level-waste disposal, this issue is considered Category 1.

- With respect to the nonradiological impact of the uranium fuel cycle, data concerning land requirements, water requirements, the use of fossil fuel, gaseous effluent, liquid effluent, and tailings solutions and solids, all listed in Table S-3, have been reviewed to determine the significance of the environmental impacts of a power reactor operating an additional 20 years. The nonradiological impacts attributable to the relicensing of an individual power reactor are found to be of small significance. License renewal of an individual plant is so indirectly connected to the operation of fuel-cycle facilities that it is meaningless to address the mitigation of impacts identified above. This is a Category 1 issue.

- The radiological and nonradiological environmental impacts from the transportation of fuel and waste attributable to license renewal of a power reactor have been reviewed. Environmental impact data for transportation are provided in Table S-4. The estimated radiological effects are within the Commission's regulatory standards. Radiological impacts of transportation are therefore found to be of small significance. The nonradiological impacts are those from periodic shipments of fuel and waste by individual trucks or rail cars and thus would result in infrequent and localized minor contributions to traffic density. These nonradiological impacts are found to be small. Programs designed to further reduce risk, which are already in place, provide for adequate mitigation. However, because a detailed analysis of the environmental impacts of transportation to the proposed repository at Yucca Mountain is not yet available, transportation of fuel and waste is Category 2.

- The radiological and nonradiological environmental impacts from the storage and disposal of low-level radiological waste attributable to license renewal of a power reactor have been reviewed. The comprehensive regulatory controls that are in place and the low public doses being achieved at reactors ensure that the radiological impacts to the environment will remain small during the term of the renewed license. The maximum additional on-site land that may be required for low-level waste storage during the term of a renewed license and associated impacts will be small. Nonradiological environmental impacts on air and water will be negligible. The radiological and nonradiological environmental impacts of long-term disposal of low-level waste from any individual plants at licensed sites are small. The need for the consideration of mitigation alternatives within the context of renewal of a power reactor license has been considered, and the Commission concludes that its regulatory requirements already in place provide adequate mitigation incentives for on-

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site storage of low-level waste and that for off-site disposal mitigation would be a site-specific consideration in the licensing of each facility. In addition, the Commission concludes that there is reasonable assurance that sufficient low-level waste disposal capacity will be made available when needed for facilities to be decommissioned consistent with NRC decommissioning requirements. Low-level waste is a Category 1 issue.

- The radiological and nonradiological environmental impacts from the storage and disposal of mixed waste attributable to license renewal of a power reactor have been reviewed. The comprehensive regulatory controls and the facilities and procedures that are in place ensure proper handling and storage, as well as negligible doses and exposure to toxic materials for the public and the environment at all plants. License renewal will not increase the small, continuing risk to human health and the environment posed by mixed waste at all plants. The radiological and nonradiological environmental impacts of long-term disposal of mixed waste from any individual plant at licensed sites are small. The maximum additional on-site land that may be required for mixed waste is a small fraction of that needed for low-level waste storage during the term of a renewed license, and associated impacts will be small. Nonradiological environmental impacts on air and water will be negligible. The radiological and nonradiological environmental impacts of long-term disposal of mixed waste from any individual plants at licensed sites are small. The need for the consideration of mitigation alternatives within the context of renewal of a power reactor license has been considered and the Commission concludes that its regulatory requirements already in place provide adequate mitigation incentives for on-site storage of mixed waste and that for off-site disposal mitigation would be a site-specific consideration in the licensing of each facility. In addition, the Commission concludes that there is reasonable assurance that sufficient mixed-waste-disposal capacity will be made available when needed for faculties to be decommissioned consistent with NRC decommissioning requirements. Mixed waste is a Category 1 issue.

- The Commission's waste confidence finding at 10 CFR 51.23 leaves only the on-site storage of spent fuel during the term of plant operation as a high-level waste storage and disposal issue at the time of license renewal. The Commission's regulatory requirements and the experience with on-site storage of spent fuel in fuel pools and dry storage has been reviewed. Within the context of a license renewal review and determination, the Commission finds that there is ample basis to conclude that continued storage of existing spent fuel and storage of spent fuel generated during the license renewal period can be accomplished safely and without significant environmental impacts. Radiological impacts will be well within regulatory limits, thus radiological impacts of on-site storage meet the standard for a conclusion of small impact. The nonradiological environmental impacts have been shown to be not significant; thus they are classified as small. The overall conclusion for on-site storage of spent fuel during the term of a renewed

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license is that the environmental impacts will be small for each plant. The need for the consideration of mitigation alternatives within the context of renewal of a power reactor license has been considered, and the Commission concludes that its regulatory requirements already in place provide adequate mitigation incentives for on-site storage of spent fuel. On-site storage of spent fuel during the term of a renewed operating license is a Category 1 issue.

- The environmental impacts from the storage and disposal of nonradiological waste attributable to the license renewal of a power reactor have been reviewed. Regulatory and operational trends suggest a gradual decrease in quantifies generated annually and the impacts during the terms of renewed licenses. Facilities and procedures are in place to ensure continued proper handling and disposal at all plants. Consequently, the generation and management of solid nonradioactive waste during the term of a renewed license is anticipated to result in only small impacts to the environment. Because the facilities and procedures that are in place are expected to ensure continued proper handling and disposal at each plant, additional mitigative measures are not a consideration in the context of a license renewal review. Nonradiological waste is a Category 1 issue.

6.7 ENDNOTES

1. The expiration dates of the 109 operating reactor licenses are presented in Table 12 of NUREG-1350, Vol 7. Nine expire in the period 2000–2009, 55 in 2010–2019, 43 in 2020–2029, 1 in 2030, and 1 in 2033.
2. The first new LLW sites are forecast in 1997 and 1998 (California, North Carolina, and Texas) and seven in the period 1999–2002.
3. 40 CFR 190.10 Standards for normal operations—"Operations covered by this Subpart shall be conducted in such a manner as to provide reasonable assurance that:
   (a) The annual dose equivalent does not exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials, radon and its daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations.
   (b) The total quantity of radioactive materials entering the general environment from the entire uranium fuel cycle, per gigawatt-year of electrical energy produced by the fuel cycle, contains less than 50,000 curies of krypton-85, 5 millicuries of iodine-129, and 0.5 millicuries combined of plutonium-239 and other alpha-emitting transuranic radionuclides with half-lives greater than one year."

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7. DECOMMISSIONING

7.1 INTRODUCTION

Decommissioning is defined as the safe removal of a nuclear facility from service and the reduction of residual radioactivity to a level that permits release of the property for unrestricted use and termination of the license (10 CFR Part 50.82). Decommissioning must occur because a licensee is not permitted to abandon a facility after ceasing operation. Decommissioning activities do not include the removal of spent fuel, which is considered to be an operational activity; the storage of spent fuel, which is addressed in the Waste Confidence Rule (10 CFR Part 51.23); or the removal and disposal of nonradioactive structures and materials beyond that necessary to terminate the U.S. Nuclear Regulatory Commission (NRC) license. Disposal of the nonradioactive hazardous waste that is not necessary for NRC license termination is not considered part of the decommissioning process for which NRC is responsible.

The purpose of this chapter is to determine whether license renewal of nuclear power plants would change the impacts of decommissioning to such an extent that those impacts would need to be assessed and mitigative measures considered as part of the environmental review for license renewal. Current licenses allow nuclear power plants to operate for as long as 40 years. License renewal would extend the period of operation by as much as 20 years. This chapter addresses incremental impacts of decommissioning after a 20-year license renewal compared with operating for 40 years.

The following potential impacts are addressed: radiation exposures to workers and the public, socioeconomic effects, waste management impacts, air and water quality impacts, and ecological impacts. The principal impacts of decommissioning are expected to result from radiation exposures to workers and from disposal of radioactive materials. Decommissioning is expected to have only minor radiological impacts on the public (primarily as a result of transporting radioactive waste). Socioeconomic impacts of decommissioning would result from the demands on, and contributions to, the community by the workers employed to decommission a power plant. As shown in this chapter, the air quality, water quality, and ecological impacts of decommissioning are all expected to be substantially smaller than those of power plant construction or operation because the level of activity and the releases to the environment are all expected to be smaller during decommissioning than during construction and operation. The effect of license renewal on the costs of decommissioning are also examined because the costs of decommissioning continues to be a public concern; however, no category conclusion is reached because the impact of license renewal on decommissioning cost is not a consideration in the environmental review and the decision to renew a license.

The impacts resulting from decommissioning at 40 years (baseline) are taken from NUREG-0586, the two source documents NUREG/CR-0130 and NUREG/CR-0672, and updates to those source documents such as draft reports NUREG/CR-5884 and NUREG/CR-6174. The same methods used in those

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documents were used to project the impacts of decommissioning after 60 years of operation. Where the source documents did not address a potential impact, other available data and staff members' professional judgments were used to assess the potential for impacts to change as a result of extended operation. The analysis in this chapter is based on large "reference" pressurized-water reactor (PWR) and boiling-water reactor (BWR) nuclear power plants; consequently, the impacts of decommissioning all U.S. nuclear power plants that reach the end of their operating lives without a serious accident should be encompassed by those described here. The changes in impacts resulting from the extended operation and in the environment at the time of decommissioning were considered. [The discussion is built around a "reference" PWR identified by NUREG/CR-0130, the 1175-MW Trojan Nuclear Plant at Rainier, Oregon, and a "reference" BWR, the 1155-MW(e) Washington Public Power Supply System Nuclear Project 2, which was being built near Richland, Washington (NUREG/CR-0672).]

7.2 THE DECOMMISSIONING PROCESS

This section describes the locations of radioactive materials in nuclear power plants, notes the three commonly discussed decommissioning methods, summarizes experience to date with decommissioning nuclear power plants, and provides information on the wastes generated during decommissioning. Except as noted, the information for this section is from NUREG-0586.

7.2.1 Nuclear Power Plants

Nuclear power plants in the United States use two types of nuclear reactors (Chapter 2); the most common type is the PWR. Most of the 118 licensed power reactors in the United States are PWRs. The other type is the BWR. The locations of radioactive components in these two types of power plants are described briefly to aid the reader's understanding of decommissioning.

7.2.1.1 Pressurized-Water Reactors

Buildings or structures associated with a typical large PWR (Figure 7.1) include (1) the heavily reinforced concrete containment building, which houses the pressure vessel, the steam generators, and the pressurizer system; (2) the turbine building, which contains the turbines and the generator; (3) the cooling water system, which may include the cooling tower and other components; (4) the fuel building, which contains fresh and spent fuel, fuel handling facilities, the spent-fuel storage pool and its cooling system, and the solid radioactive waste system; (5) the auxiliary building, which contains the liquid radioactive waste treatment systems, the filter and ion exchanger vaults, the gaseous radioactive waste treatment system, and the ventilation systems for the containment, fuel, and auxiliary buildings; (6) the control building, which houses the reactor control room and personnel facilities; (7) water intake structures; (8) the administration building; and (9) other structures such as warehouses and nonradioactive shops.

The major radioactive components encountered in decommissioning are associated with the reactor itself—the primary coolant loop, the steam generators, the radioactive waste handling systems, and the concrete biological shield that surrounds the pressure vessel. The reactor core, pressure vessel, steam generators, and piping between the reactor and steam generators are highly radioactive. Because some primary-to-secondary leakage is

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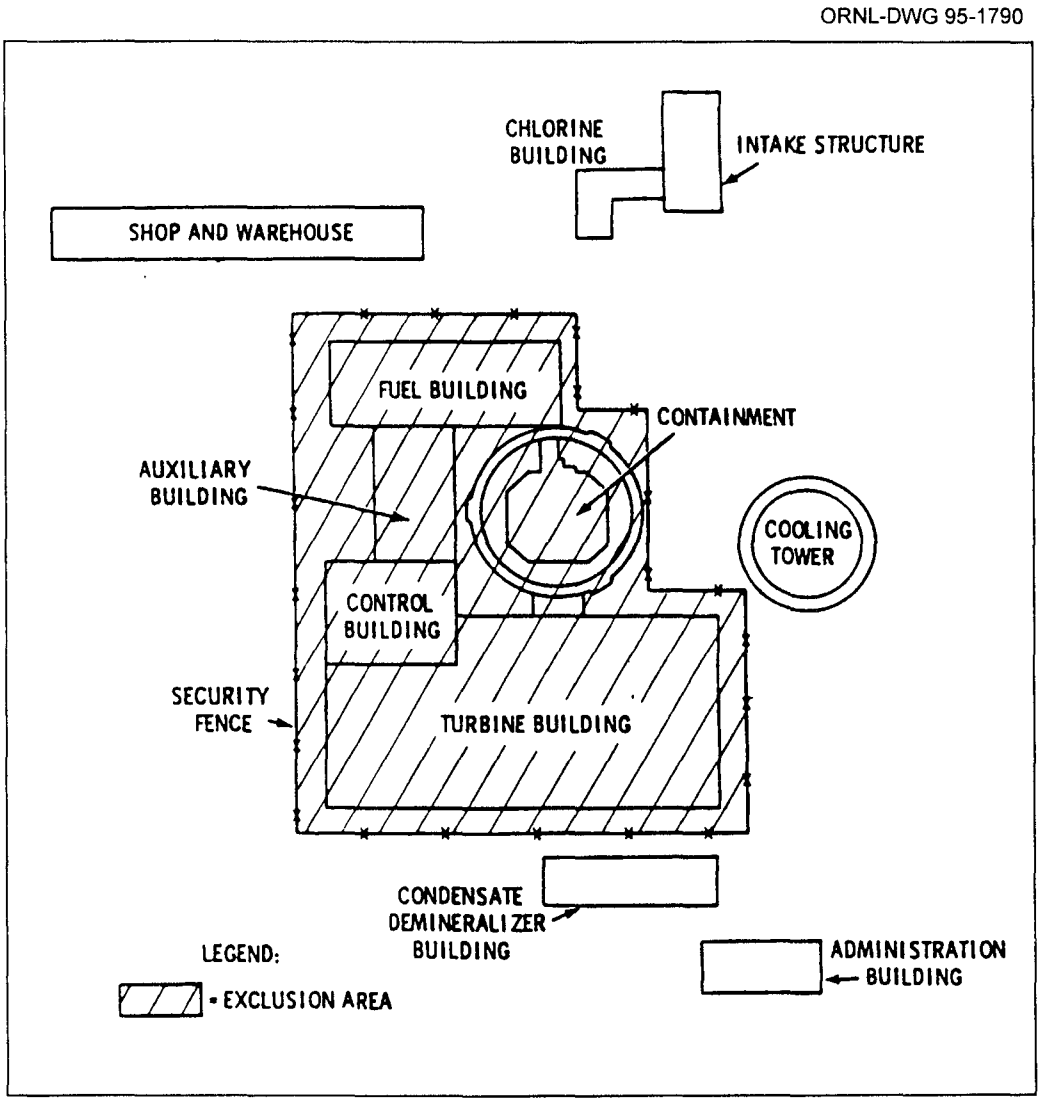

Figure 7.1 **Typical pressurized-water reactor generating station layout.** Adapted from NUREG/CR-0130.

impossible to avoid, the secondary loop, including the turbines, is slightly contaminated. Because of leakage and blowdown, the cooling water is very slightly contaminated. Much equipment in the auxiliary building is contaminated, as is the spent-fuel storage pool and its associated equipment.

7.2.1.2 Boiling-Water Reactors

Buildings and structures associated with a typical large BWR (Figure 7.2) include (1) the reactor building, which houses the reactor pressure vessel, the containment structure, the biological shield, the spent-fuel pool, and fuel handling equipment; (2) the turbine building, which houses the

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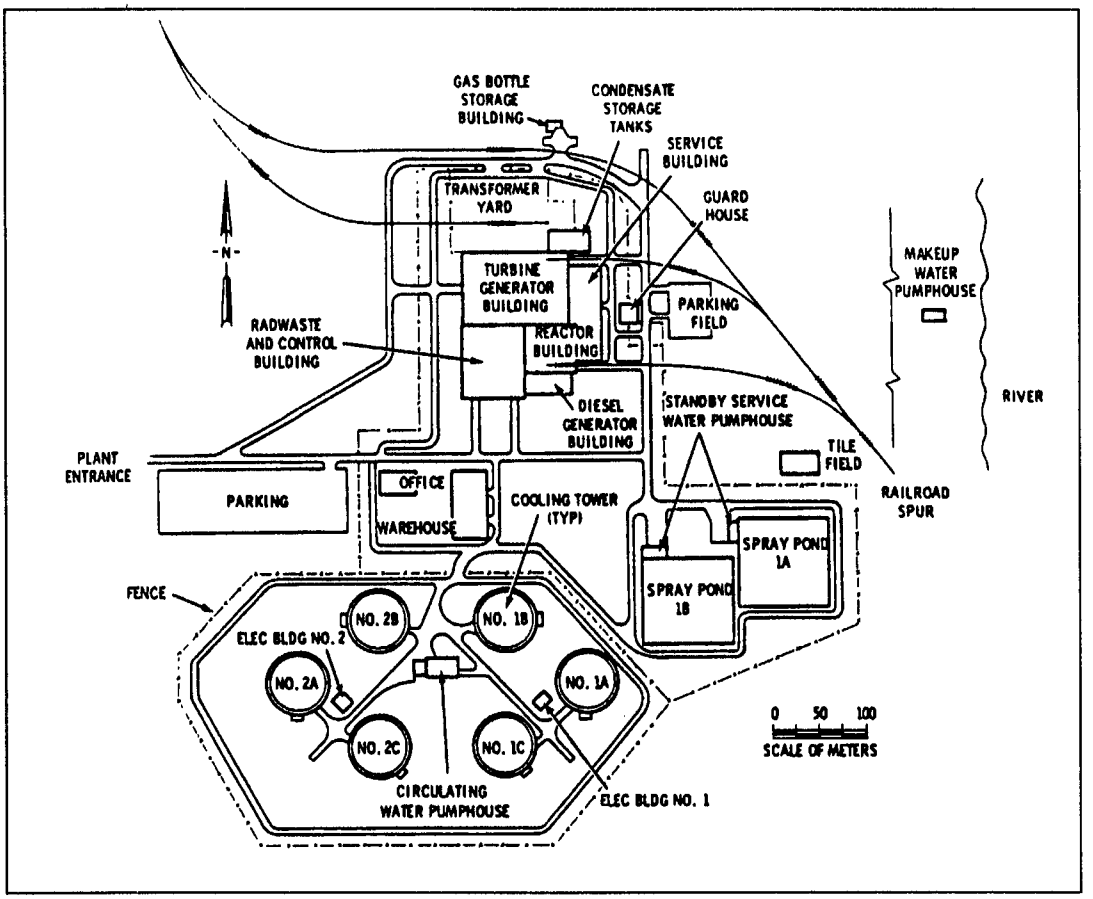

Figure 7.2 Site layout on a typical boiling-water reactor power plant. Adapted from NUREG-0672.

turbine and electric generator; (3) the radioactive waste and control building, which houses the solid, liquid, and gaseous radioactive waste treatment systems and the main control room; (4) the cooling system; (5) water intake structures and pump houses; (6) the service building, which houses the makeup water treatment system, machine shops, and offices; and (7) other minor structures.

The major sources of radiation in decommissioning a BWR are associated with the reactor itself, the containment structure, the concrete biological shield, the primary coolant loop, the turbines, and the radioactive waste handling systems. The reactor building, the turbine generator building, and the radioactive waste building are the only buildings containing radioactive materials. The reactor core and its pressure vessel are highly contaminated, as is the piping to the turbines. The turbines are also contaminated, but the cooling towers and associated piping are not. Much equipment in the radioactive waste building is contaminated, as is the spent-fuel pool in the reactor building.

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7.2.2 Decommissioning Methods

In the NRC's original decommissioning studies (NUREG/CR-0130 for PWRs and NUREG/CR-0672 for BWRs), three alternatives were defined: DECON (decontamination/dismantlement as rapidly after reactor shutdown as possible to achieve termination of the nuclear license); SAFSTOR (a period of safe storage of the stabilized and defueled facility followed by final decontamination/dismantlement and license termination); and ENTOMB (immediate removal of the highly activated reactor vessel internals for disposal and relocation of the remainder of the radioactively contaminated materials to the reactor containment building, which is then sealed. With sufficient time, the radioactivity on the entombed materials will have decayed to levels that permit termination of the nuclear license). However, because current regulations require decommissioning to be complete within 60 years, ENTOMB may not be a viable option.

Changes in the industrial and regulatory situation in the United States since the late 1970s have forced revisions to the scenarios of the NRC's original decommissioning alternatives. The most recently revised decommissioning scenarios are described for PWRs in NUREG/CR-5884 and for BWRs in NUREG/CR-6174. There are two principal changes in the revised scenarios. One is the delay of major decommissioning actions for at least 5 to 7 years following reactor shutdown because of a Department of Energy (DOE) requirement to cool the spent fuel in the reactor pool to avoid cladding failures in dry storage. The other is the assumption that decommissioning will be complete within 60 years, as required by current regulations. This delay results in an increase in decommissioning costs during the short safe storage period while the spent fuel pool continues to operate. Changes in cumulative occupational radiation doses also result from the decommissioning scenario changes.

The basic concept of the three alternatives remains unchanged. However, because of the accumulated inventory of spent fuel in the reactor storage pool and the requirement for at least 5 years of storage for the spent fuel before transfer to DOE for disposal, the timing and steps in the process for each alternative have been adjusted to reflect present conditions and possibilities. For the DECON alternative, it is assumed that the owner has strong incentives to decontaminate and dismantle the retired reactor facility as promptly as possible [i.e., future availability and cost of low-level radioactive waste (LLW) disposal and the need to reuse or dispose of the site, necessitating transfer of the stored spent fuel from the pool to a dry storage facility on the reactor site]. Although continued storage of spent fuel in the pool would be acceptable, the modified Part 50 license could not be terminated until the pool was emptied. It is also assumed that an acceptable dry transfer system would be available to remove the spent fuel from the dry storage facility and place it into licensed transport casks when the time came for DOE to accept the spent fuel for disposal. Similar assumptions are made for the SAFSTOR and ENTOMB alternatives for convenience of analysis, even though extended use of the spent fuel pool might be more cost-effective for SAFSTOR.

7.2.2.1 DECON

DECON is the decommissioning method in which the equipment, structures, and portions of the facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after cessation of operations. It is

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the only decommissioning alternative that leads to termination of the facility license and release of the facility and site for unrestricted use shortly after cessation of facility operations. DECON activities are expected to require about 9 years for large light-water reactors; less time should be required for smaller facilities.

Because DECON operations are expected to be completed within a few years following shutdown, radiation exposures to workers generally are higher than for decommissioning methods that allow for radioactive decay by delaying or extending the work over a longer period. DECON also requires larger commitments of money and commercial waste disposal site space than do other decommissioning methods. The principal advantage of DECON is that the site is available for unrestricted use promptly.

Nonradioactive equipment and structures need not be dismantled or removed for termination of the NRC license and release for unrestricted use. Once the facility's radioactive structures are decontaminated to levels permitting unrestricted use of the facility, nonradioactive facilities may either be put to some other use or demolished at the owner's discretion. [NRC has issued proposed amendments to 10 CFR Part 20 containing radiological criteria for decommissioning of NRC-licensed nuclear facilities (FR 59, 43200, August 22, 1994). Currently, NRC uses, on a case-by-case basis, criteria and practices contained in Regulatory Guide 1.86 and in a letter to Stanford University from J. Miller, Office of Nuclear Reactor Regulation, NRC, dated April 21, 1982.]

DECON, as defined by NUREG/CR-5884 and NUREG/CR-6174, comprises four distinct periods of effort: (1) preshutdown planning/engineering and regulatory reviews, (2) plant deactivation and preparation for storage (no dismantling activities are conducted during this period that would affect the safe operation of the spent fuel pool), (3) plant safe storage with concurrent operations in the spent-fuel pool until the pool inventory is zero, and (4) decontamination and dismantlement of the radioactive portions of the plant, leading to license termination. Because of the delays in development of the federal waste management system, it may be necessary to continue operation of a dry fuel storage facility on the reactor site after the reactor systems have been dismantled and the reactor nuclear license terminated. However, these latter storage costs are considered operations costs under 10 CFR 50.54(b)(b) and are not considered part of decommissioning.

7.2.2.2 SAFSTOR

SAFSTOR is the decommissioning method in which the nuclear facility is placed and maintained in a condition that allows the safe storage of radioactive components of the nuclear plant and subsequent decontamination to levels that permit release for unrestricted use. SAFSTOR was initially conceived of as having three successive stages: (1) a short period of preparation for safe storage (expected to be up to 2 years after final reactor shutdown); (2) a variable safe storage period of continuing care consisting of security, surveillance, and maintenance during which much of the reactor's radioactivity decays; and finally, (3) a relatively short period of decontamination (NUREG-0586). In NUREG/CR-5884 and NUREG/CR-6174, SAFSTOR is described as five distinct periods of effort, with the initial three periods identical to those of DECON. The fourth period is extended safe storage (50 years) with no fuel in the reactor storage pool, and the fifth period is

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decontamination and dismantlement of the radioactive portions of the plant.

The radioactive or contaminated material must be decontaminated or removed, packaged, and disposed of at a regulated disposal facility. After it has been determined that residual radioactivity is at acceptable levels, the license will be terminated and the facility can be released for unrestricted use. After termination of the NRC license, disassembly or demolition of nonradioactive facilities would be performed at the owner's discretion.

SAFSTOR may be used as a means of satisfying requirements for protection of the public while minimizing the initial commitments of time, money, radiation exposure, and waste disposal capacity. SAFSTOR may also have some advantage where there are other operational nuclear facilities at the same site or where a shortage of radioactive waste disposal capacity occurs. The disadvantages of SAFSTOR are that the site is unavailable for other uses for an extended time; maintenance, security, and surveillance are required until the final decontamination is complete; and few, if any, personnel familiar with the facility are available at the time of decontamination (up to 60 years after plant shutdown).

7.2.2.3 ENTOMB

ENTOMB is the alternative in which radioactive contaminants are encased in a long-lasting material, such as concrete. The entombed structure is maintained and surveillance is performed until the radioactivity decays to a level permitting release of the property for unrestricted use. ENTOMB also comprises five distinct periods of effort, with the initial three periods identical to those of DECON (NUREG/CR-5884 and NUREG/CR-6174). The fourth period is preparation for entombment, when all of the radioactive materials are consolidated within the containment building and entombed. The fifth period is entombed storage for an extended time, between 60 and 300 years.

ENTOMB is intended for use where the residual radioactivity will decay to levels permitting unrestricted release of the facility within reasonable time periods (100 years). However, a few radioactive isotopes produced in nuclear reactors have long half-life periods (Section 7.3.1) that prevent the release of the facilities for unrestricted use within the foreseeable lifetime of any man-made structure. ENTOMB would be a viable alternative only for facilities where radioactive isotopes would be expected to decay to safe levels within the expected lifetime of the entombment structure. This condition likely would not pertain to nuclear power reactors. In addition, the use of the ENTOMB alternative contributes to problems associated with increased numbers of sites dedicated to "interim" storage of radioactive materials for long periods of time.

7.2.3 Decommissioning Experience

U.S. commercial nuclear power reactors that have been shut down through 1992 are listed in Table 7.1. An additional 24 reactors have been or are being decommissioned in France, West Germany, Canada, the United Kingdom, Sweden, and Japan (Gaunt et al. 1990).

7.2.4 Inventory and Disposition of Radioactive Materials

Radioactive materials can be classified as activated or radioactively contaminated materials. Materials become activated when they have been exposed to (irradiated by) high levels of neutron radiation (such as in a reactor). When normal (stable) atoms in

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Table 7.1 U.S. commercial nuclear power reactors formerly licensed to operate

Unit/ location	Construction type[a]/ MW(t)	Operating license issued/ shut down	Decommissioning alternative selected/ current status
Bonus[b] Punta Higuera, PR	BWR/50	04/02/64 06/01/68	ENTOMB ENTOMB
Carolina Virginia Tube Reactor[c] Parr, SC	PTHW/65	11/27/62 01/01/67	SAFSTOR SAFSTOR
Dresden 1 Morris, IL	BWR/700	09/28/59 10/31/78	SAFSTOR SAFSTOR
Elk River[b] Elk River, MN	BWR/58	11/06/62 02/01/68	DECON DECON completed
Fermi 1 Lagoona Beach, MI	SCF/200	05/10/63 09/22/72	SAFSTOR SAFSTOR
Fort St. Vrain Platteville, CO	HTG/842	12/21/73 08/18/89	DECON DECON in progress
GE Vallecitos Boiling Water Reactor Pleasanton, CA	BWR/50	08/31/57 12/09/63	SAFSTOR SAFSTOR
Hallam[b] Hallam, NE	SCGM/256	01/02/62 09/01/64	ENTOMB ENTOMB
Humboldt Bay 3 Eureka, CA	BWR/200	08/28/62 07/02/76	SAFSTOR SAFSTOR
Indian Point 1 Buchanan, NY	PWR/615	03/26/62 10/31/74	SAFSTOR NRC review
La Crosse Genoa, WI	BWR/165	07/03/67 04/30/87	SAFSTOR SAFSTOR
Pathfinder Sioux Falls, SD	BWR/190	03/12/64 09/16/67	SAFSTOR DECON in progress
Peach Bottom 1 Peach Bottom, PA	HTG/115	01/24/66 10/31/74	SAFSTOR SAFSTOR
Piqua[b] Piqua, OH	OCM/46	08/23/62 01/01/66	ENTOMB ENTOMB
Rancho Seco Herald, CA	PWR/2772	08/16/74 06/07/89	SAFSTOR NRC review

See notes at end of table.

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Table 7.1 (continued)

Unit/ location	Construction type[a]/ MW(t)	Operating license issued/ shut down	Decommissioning alternative selected/ current status
San Onofre 1 San Clemente, CA	PWR/1347	03/27/67 11/30/92	SAFSTOR[d]
Shippingport[b] Shippingport, PA	PWR/236	N/A 82	DECON DECON completed
Shoreham Wading River, NY	BWR/2436	04/21/89 06/28/89	DECON DECON in progress
Three Mile Island 2 Londonderry Township, PA	PWR/2770	02/08/78 03/28/79	*e*
Trojan Portland, OR	PWR/3411	11/21/75 11/09/92	*f*
Yankee-Rowe Franklin County, MA	PWR/600	12/24/63 10/01/91	*g*

[a]BWR = boiling-water reactor; HTG = high-temperature gas-cooled; OCM = organically cooled and moderated; PTHW = pressure tube, heavy water cooled and moderated; PWR = pressurized-water reactor; SCF = sodium cooled,
fast; SCGM = sodium cooled, graphite moderated.
[b]Atomic Energy Commission/Department of Energy owned; not regulated by the Nuclear Regulatory Commission.
[c]Holds by-product license from state of South Carolina.
[d]San Onofre 1 decommissioning plan was due to the Nuclear Regulatory Commission in November 1994.
[e]Three Mile Island 2 has been placed in a monitored storage mode. The licensee plans to maintain the facility in monitored storage until Three Mile Island 1 permanently ceases operation, at which time both units are to be decommissioned simultaneously.
[f]Trojan received a possession-only license on 05/05/93. The license is evaluating SAFSTOR and DECON decommissioning alternatives. A decommissioning plan was due to the Nuclear Regulatory Commission in January 1995.
[g]Yankee Rowe received a possession-only license on 08/05/92. The licensee submitted a decommissioning plan on 12/20/93. Decommissioning alternative depends on the availability of low-level waste disposal facilities.
Source: DOE/RW-0006, rev. 6.

a material absorb neutrons, they become unstable (radioactive) and subsequently emit energy in the form of radiation. Radioactive contamination is radioactive material in the form of fine particles, liquids, or gases that are deposited on the surface of, or mixed with, materials that otherwise are not radioactive. Contaminated materials can generally be decontaminated to various degrees by several techniques. These techniques range from simply washing with soap and water to sandblasting contaminated surfaces. Decontamination techniques for liquids and gases include filtration and chemical ion exchange. Activated materials cannot be decontaminated; they remain radioactive until the radioactive constituents decay to stable isotopes.

Reactor components are generally both activated and contaminated. The principal

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activated components of a power plant are the reactor internals and the biological shield. Other reactor system components, such as the primary and possibly the secondary coolant loops, the turbines in BWRs, and the radioactive waste handling systems, are not activated but are highly contaminated by the contaminated fluids they contain. The major source of contamination in reactor coolant is the plant corrosion and wear material suspended in the coolant that becomes activated as it passes through the reactor core. Surface contamination can also be found in areas of the plant where leaks from contaminated systems have occurred.

The inventory of radionuclides for PWRs and BWRs is slightly different. A typical large PWR would have a radioactivity level of about 4.8 million Ci (1Ci = 3.7×10^{10} Bq) in the major reactor components, 4800 Ci of radioactive corrosion products in the primary coolant system, and 1200 Ci of radioactivity in the concrete biological shield at the time of shutdown (NUREG/CR-0130). A typical large BWR would have a radioactivity level of about 6.3 million Ci in the major reactor components, 8600 Ci of radioactive corrosion products in the primary coolant system, and 1000 Ci of radioactivity in the concrete biological shield at the time of shutdown (NUREG/CR-0672).

The principal radioactive isotopes from irradiated steel and concrete, with their modes of decay and their half-lives, are listed in Table 7.2. By the end of 40 years of operation, the radionuclides with half-lives of less than about 5 years are at equilibrium, because their rates of decay equal their rates of generation. No matter how much longer a power plant is operated, the concentration of short-half-life radionuclides will not increase. The longer-lived radionuclides are generated much faster than they decay; thus their concentrations increase approximately in proportion to the reactor operating time. Figure 7.3 illustrates the buildup of some important radionuclides as a function of nuclear plant operating life.

Radioactive isotopes that are mainly beta emitters or that have very short half-lives do not contribute significantly to the personnel radiation dose associated with decommissioning. Because beta radiation is weakly penetrating, it can be shielded easily and presents a hazard mainly if ingested or inhaled by operations personnel. Isotopes with very short half-life periods can be allowed to decay to insignificant levels before decommissioning operations begin.

At the time of decommissioning, radioactive materials are found in the reactor building, the auxiliary building, and the fuel building (Section 7.2.1). Immediately after operations are terminated, these parts of the plant are highly radioactive because of short-lived activation products. The highest levels of radioactivity subside very quickly as short-lived radionuclides decay and progressively longer-lived radionuclides dominate the overall radioactivity. After about a year, ^{60}Co dominates the radiation dose to workers. After about 100 years, ^{94}Nb dominates the radiation dose to workers or persons in the vicinity (Figure 7.4). For all practical purposes, the radiation dose to workers will not decrease further because ^{94}Nb has a 20,000-year half-life. Because ^{60}Co and ^{94}Nb dominate the radiation dose during the time of decommissioning, their characteristics affect the decommissioning process.

Both ^{60}Co and ^{94}Nb are activation products—isotopes created when neutrons from nuclear fission convert nonradioactive elements (^{59}Co and ^{93}Nb) in the structural components of the plant into radioactive

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Table 7.2 Principal activated radioactive isotopes found in operating nuclear power plants (excluding fuel)

Element	Isotope	Decay mode[a]	Half-life (years)
Hydrogen	^{3}H	β	1.23×10^{1}
Carbon	^{14}C	β	5.73×10^{3}
Phosphorus	^{33}P	β	6.9×10^{-2}
Silicon	^{35}S	β	2.4×10^{-1}
Chlorine	^{36}Cl	β, γ	3.01×10^{5}
Argon	^{37}Ar	γ	9.5×10^{-2}
Argon	^{39}Ar	β	2.99×10^{2}
Potassium	^{40}K	β, γ	1.28×10^{9}
Calcium	^{41}Ca	γ	8.0×10^{4}
Calcium	^{45}Ca	β	4.5×10^{-1}
Scandium	^{46}Sc	β	2.3×10^{-1}
Chromium	^{46}Cr	γ	7.6×10^{-2}
Manganese	^{54}Mn	γ	$8. \times 10^{-1}$
Iron	^{55}Fe	γ	2.7×10^{0}
Iron	^{59}Fe	β, γ	1.2×10^{-1}
Cobalt	^{58}Co	γ	2.1×10^{-1}
Cobalt	^{60}Co	β, γ	5.27×10^{0}
Nickel	^{59}Ni	γ	8.0×10^{4}
Nickel	^{63}Ni	β	9.2×10^{1}
Zinc	^{65}Zn	γ	6.7×10^{-1}
Niobium	^{93m}Nb	γ	1.36×10^{1}
Niobium	^{94}Nb	β, γ	2.03×10^{4}
Niobium	^{95}Nb	β, γ	9.6×10^{-2}
Molybdenum	^{93}Mo	γ	3.5×10^{3}
Zirconium	^{95}Zr	β, γ	1.8×10^{-1}
Technetium	^{99}Tc	β	2.13×10^{5}
Silver	^{108m}Ag	β, γ	1.27×10^{2}
Silver	^{110m}Ag	β, γ	6.8×10^{-1}
Cadmium	^{109}Cd	γ	1.3×10^{0}
Samarium	^{151}Sm	β, γ	9.0×10^{1}
Europium	^{152}Eu	β, γ	1.33×10^{1}
Europium	^{154}Eu	β, γ	8.8×10^{0}
Holmium	^{166m}Ho	γ	1.2×10^{3}

[a]β = beta, γ = gamma (including x-rays).
Source: R. C. Weast, ed. *Handbook of Chemistry and Physics*, 53rd ed. 1972–73, Chemical Rubber Company, Cleveland, 1972.

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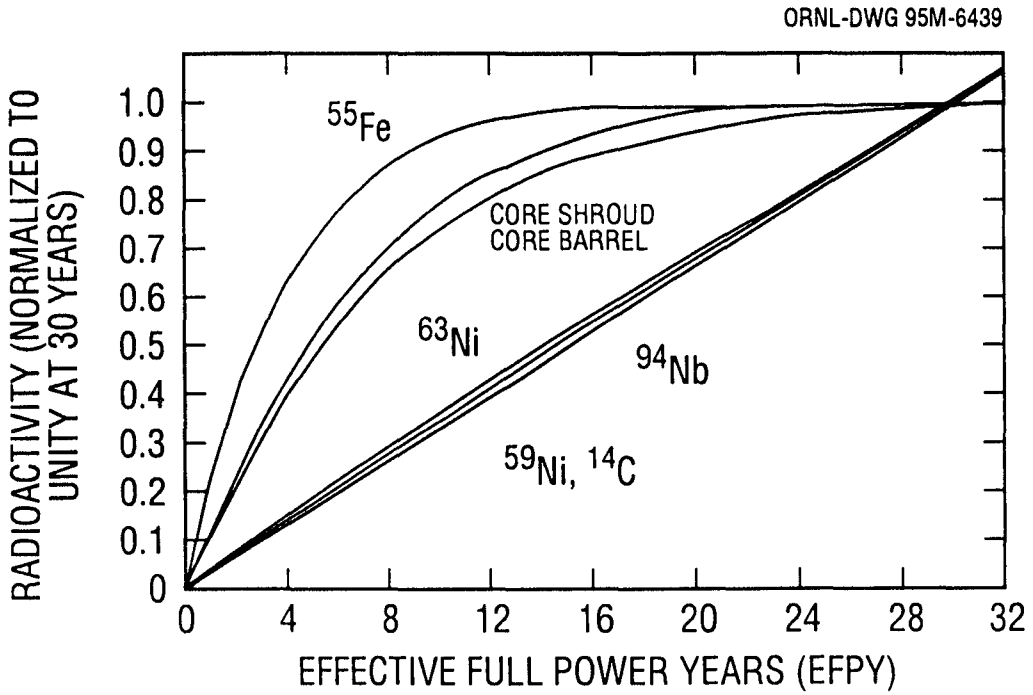

Figure 7.3 **Buildup of activation products in pressurized-water reactor internal components as a function of effective full-power years.** *Source:* NUREG/CR-0130.

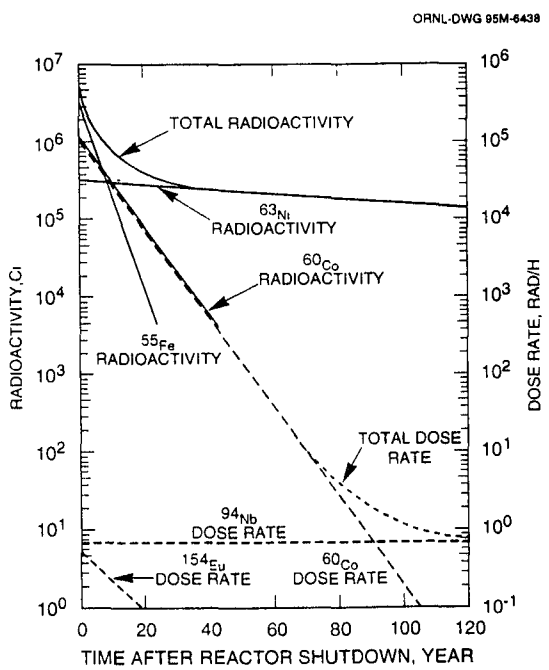

Figure 7.4 **Time dependence of radioactivity and dose rate in a boiling-water reactor core shroud after 40 years of operation.** *Source:* NUREG/CR-0672.

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isotopes. An important difference is that ^{94}Nb in the steel reactor vessel and components, formed by activation of ^{93}Nb, is not subject to corrosion and movement throughout the primary system to the extent that ^{60}Co is. Consequently, equipment in the reactor containment building that is not exposed to high neutron fluxes and parts of the fuel and auxiliary buildings may be highly contaminated with ^{60}Co but only slightly so with ^{94}Nb.

Extending operations to 60 years would not increase the shutdown radioactivity level of either a PWR or BWR to any appreciable extent. This is because most of the radioactivity at shutdown results from short-half-life radionuclides, such as ^{60}Co, that are already in equilibrium by the time 40 years of operations have transpired. The only change in radioactive inventory resulting from the additional 20 years of operations is the further accumulation of long-half-life radionuclides such as ^{63}Ni and ^{94}Nb, but these long-half-life radionuclides produce only a small fraction of the total radioactivity at shutdown. Of the long-half-life radionuclides, ^{63}Ni contributes most at shutdown but composes less than 3 percent of the total radioactivity. Twenty additional years of operation would increase its contribution to about 4 percent of total shutdown radioactivity. Because ^{63}Ni is a beta emitter, it contributes only a very small part of the dose to workers or the public. Gamma-emitting ^{94}Nb is the most important long-half-life radionuclide with regard to producing external radiation exposure. Based on Figure 7.4, it can be determined that at shutdown ^{94}Nb contributes less than 0.001 percent of the total potential dose. Even though 20 additional years of operation would increase the amount of ^{94}Nb by 50 percent, it would not increase its contribution to the dose much above 0.001 percent.

7.2.5 Waste Generated During Decommissioning

This section summarizes the quantities and types of radioactive waste and emissions generated in decommissioning after 40 and 60 years of operation, respectively. Because the demolition and disposal of nonradioactive parts of nuclear facilities are not considered part of decommissioning, almost all waste generated during decommissioning is radioactive. Although the demolition and disposal of the nonradioactive parts may continue during and after decommissioning, these activities are not regulated by NRC. The impacts of radioactive wastes and emissions are described in Section 7.3. This section does not take into account volume reduction or aggressive processing that could allow release for unrestricted use.

7.2.5.1 Atmospheric Emissions

As shown in Table 7.3, the total atmospheric releases for decommissioning are less than 100 mCi, whereas normal operations average about 3000 Ci/year. Atmospheric releases are expected to consist largely of dust, aerosols, and smokelike particulates produced during the dismantling and handling of reactor components. These releases were estimated by assuming that the airborne concentrations of radionuclides will be a fraction of the contamination level on and in the radioactive components (NUREG/CR-0130 and NUREG/CR-0672). Because the radioactive inventory would be nearly unchanged by operations during a 20-year license renewal term, no difference exists between the base case and 20 years of additional operation.

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Table 7.3 Airborne radioactive releases resulting from decommissioning typical pressurized-water reactors (PWRs) and boiling-water reactors (BWRs) with normal operating releases, base case (40 years of operation)[a]

	DECON (mCi)	SAFSTOR (mCi)	ENTOMB (mCi)	Normal operations (Ci/year)
PWR	0.86[b]	0.003[b]	NA[c]	2,600[d]
BWR	87[e]	0.21d[d]	2.25[d]	3,400[d]

[a]Decommissioning releases are for 40 years of operation. Releases for 60 years of operation would be essentially the same.
[b]*Source:* NUREG/CR-0130, Table 11.2-2.
[c]Not available.
[d]*Source:* DOE/EP-0093.
[e]*Source:* NUREG/CR-0672, Tables N.2-12, N.3-4, N.4-4, E.2-11. Decommissioning is assumed to last 5 years.

7.2.5.2 Liquid Effluents

No estimates of liquid waste releases are available for decommissioning nuclear power plants. However, liquids will be produced by decontamination procedures (e.g., some cutting operations and possibly some chemical decontamination procedures) and by disposal of plant fluids (e.g., cooling water and water from fuel storage pools). Filtration and ion exchange methods will be used to decontaminate liquids, as would be done during normal operations. Some liquid effluents may be contaminated with chelating agents and may require further processing. These methods are expected to keep waterborne effluents of most radionuclides within the values of normal operations. Tritium (^3H) is the only radioactive isotope that cannot be removed from waste water by these means.

Tritium is found principally in the primary coolant-loop water. Tritium cannot be removed from water except by extraordinary means and is normally discharged to a surface water body. Normal ^3H discharges from PWRs range from a few hundred to a few thousand curies per year. BWR ^3H discharges are generally only about 10 percent as high as ^3H discharges from PWRs. About 500 Ci of ^3H can be found in PWR primary coolant-loop water. Discharge of the entire volume of primary coolant-loop water over a period of 1 to 5 years after shutdown would be feasible without exceeding normal operating period discharge rates. The amounts or characteristics of liquid effluents discharged during decommissioning would not be changed by operation during a 20-year license renewal term. Discharge of primary coolant water during normal operations limits the accumulation of ^3H in the primary coolant loop; thus ^3H is in equilibrium in the primary coolant water well before 40 years of operation.

7.2.5.3 Solid Waste

Table 7.4 summarizes the quantities of LLW generated by decommissioning of large PWRs and BWRs. The table shows that the largest amount of LLW is

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Table 7.4 Estimated burial volume of low-level waste and rubble for large pressurized-water reactor (PWR) and boiling-water reactor (BWR) decommissioning, base case (40 years of operation)

Decommissioning alternative	PWR (m^3)[a]	BWR (m^3)
DECON	6,992	14,282
SAFSTOR 1	763	1,117
SAFSTOR 2	6,992	14,282
ENTOMB 1	305	490
ENTOMB 2	754	1,139
ENTOMB 3	305	490

[a]1 m^3 = 35.3 ft^3
Source: NUREG/CR-5884, Table ES.1 and NUREG/CR-6174, Table ES.1.

generated by the DECON method and the least is generated by the SAFSTOR method. The quantities listed for the ENTOMB method do not include the volume of the entombing structure or the wastes within.

The decommissioning waste volumes for all three methods of decommissioning also would not be affected by extending the volume of radioactive materials would not increase. (Operational waste quantities would continue, but they do not affect the amount of decommissioning waste.) An additional 20 years of operation would slightly affect the waste characteristics. As discussed in Section 7.2.4, the quantity of long-lived activation products such as ^{94}Nb would continue to increase, essentially in proportion to the additional operational time. As a result, the long-half-life radionuclides in the waste would increase by 50 percent if the plants were operated an additional 20 years. However, as explained earlier, these long-lived radionuclides contribute only a small fraction of the shutdown radioactivity level.

7.3 DECOMMISSIONING IMPACTS AND CHANGES RESULTING FROM LIFE EXTENSION

Estimated decommissioning impacts for 40 years of operation—the base case (taken primarily from NUREG-0586, NUREG/CR-0130, and NUREG/CR-0672)—and the change in impacts caused by continued operations for an additional 20 years under license renewal are reported for each impact area in the following sections. These impacts are estimated for PWRs and BWRs. The per-reactor impacts of decommissioning at multiple-reactor sites are not expected to be significantly different from those at single-reactor sites. [The impacts would be smaller at multiple reactor sites if the reactor decommissionings were staggered and if LLW were stored on the site (NUREG-0586)].

7.3.1 Radiation Dose

The estimated occupational and public radiation doses resulting from the three decommissioning methods after 40 years of operation (base case) are summarized

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in this section. Occupational dose estimates were presented in draft reports NUREG/CR-5884 and NUREG/CR-6174. These reports do not provide estimates of doses to the public. The Atomic Energy Act requires the Nuclear Regulatory Commission to promulgate, inspect, and enforce standards that provide an adequate level of protection of the public health and safety and the environment. These responsibilities, singly and in the aggregate, provide a margin of safety. For the purposes of assessing radiological impacts, the Commission has concluded that impacts are of small significance if doses and releases do not exceed permissible levels in the Commission's regulations.

7.3.1.1 Occupational Dose

For both PWRs and BWRs, there are substantial differences among the occupational radiation doses for the decommissioning methods (Table 7.5). The DECON method has the highest doses, followed by ENTOMB and then SAFSTOR. Although extending operations 20 years would increase the doses from ^{94}Nb and other less-important long-half-life radionuclides, these doses would not have any appreciable effect on the occupational dose because short-lived radionuclides (primarily ^{60}Co) are the principal sources of worker exposure. For each decommissioning method, the bulk of the dose comes during activities in the first few years after termination of plant operations (period four begins less than 5 years after terminating operations for DECON), when the radioactivity level of ^{60}Co is still significant. At the end of 60 years of SAFSTOR, the dose rate would have decayed to about 0.01 percent of the dose rate at the end of operations, at which time ^{94}Nb would contribute only about 2 percent of the total (Figure 7.4). An additional 20 years of operation before 60 years of SAFSTOR would increase the amount of ^{94}Nb by approximately 50 percent. During period 5, occupational exposures from SAFSTOR activities would be no more than 10 person-rem. (Section E.A.3 of Appendix E discusses the International System units used in measuring radioactivity and radiation dose. The contribution from ^{94}Nb would be less than 0.2 person-rem. The increase in dose during decommissioning after 20 additional years of operation would be no more than about 0.1 person-rem.

Although total doses to the decommissioning workforce may increase slightly as a result of an additional 20 years of plant operation, the exposure of individual workers will be maintained well below the existing regulatory limits of 10 CRF Part 20. Accordingly, the Commission concludes that radiological impacts to the decontamination workforce as a result of license renewal is of small significance.

The potential increase in total dose to the decommissioning work force may be mitigated by programs that are responsive to 10 CFR 20.1101(b), which requires that "The licensee shall use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA)." The ongoing ALARA programs within the industry already employ measures that would be considered for mitigating the generation or the accumulation of long-lived activation products during 20 additional years of operation. Two examples of mitigation measures that are already in use are (1) replacing components using cobalt alloys with those using low-cobalt or cobalt-free alloys and (2) full system decontamination (e.g., see

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Table 7.5 Estimated occupational radiation doses for decommissioning a large reactor (person-rem), base case (40 years of operation)[a]

Decommissioning period[b]	DECON[c,d]	SAFSTOR[c,e]	ENTOMB[c,f]
	Pressurized-water reactor[g]		
1	—	—	—
2	207	207	207
3	21	21	21
4	704	88	562–589
5	NA	0-6	0
Totals[h]	931	315–322	790–816
	Boiling-water reactor[i]		
1	—	—	—
2	425	425	425
3	10	10	10
4	528	123	166–230
5	NA	0–10	0
Totals[h]	962	558–568	601–665

[a]Occupational radiation exposures are for decommissioning after 40 years of operations.
[b]Decommissioning periods are defined in NUREG/CR-6174 and NUREG/CR-5884.
[c]DECON, SAFSTOR, and ENTOMB are defined differently by NUREG/CR-5884 and NUREG/CR-6174 than by previous analyses.
[d]Table 3.1.
[e]Table 4.1.
[f]Table 5.2.
[g]*Source:* NUREG/CR-5884.
[h]Totals may not equal sum of entries because of rounding.
[i]*Source:* NUREG/CR-6174.

Moore 1995). No additional mitigation measures warranted. This is a Category 1 issue.

7.3.1.2 Dose to the Public

For both PWRs and BWRs, the radiation dose to the public results primarily from waste shipment (Table 7.6). Furthermore, the dose is almost exclusively caused by shipment of ^{60}Co and shorter-lived radionuclides; for truck shipments, the SAFSTOR 100-years alternative shows negligible dose to the public. Because only the quantities of long-lived radionuclides would increase if plants were operated an additional 20 years, only the dose caused by the long-lived radionuclides would increase. Because the dose to the public from long-lived radionuclides after 40 years

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Table 7.6 Estimated radiation dose to the public for decommissioning a large reactor (person-rem), base case (40 years of operation)[a,b]

	DECON	SAFSTOR 30 years	SAFSTOR 100 years	ENTOMB
Pressurized-water reactor				
SAFSTOR preparation	NA	neg	neg	NA
Continuing care	NA	neg	neg	neg
Decontamination	neg[c]	neg[c]	neg[c]	NA
Entombment	NA	NA	NA	neg
SAFSTOR preparation truck shipments	NA	2	2	NA
Decontamination truck shipments	21[c]	0.4[c]	neg[c]	NA
Entombment truck shipments	NA	NA	NA	4
Totals	21	3	2	4
Boiling-water reactor				
SAFSTOR preparation	NA	neg	neg	NA
Continuing care	NA	neg	neg	neg
Decontamination	neg[c]	neg[c]	neg[c]	NA
Entombment	NA	NA	NA	neg
SAFSTOR preparation truck shipments	NA	2	2	NA
Decontamination truck shipments	10[c]	neg[c]	neg[c]	NA
Entombment truck shipments	NA	NA	NA	5–7[d]
Totals	10	2	2	5–7[d]

[a]Public radiation exposures are for decommissioning after 40 years of operation (NUREG-0586). Decommissioning exposures after 60 years would be identical, except as noted. Draft reports NUREG/CR-5884 and NUREG/CR-6174 do not provide updates for this information.
[b]NA means not applicable and neg means negligible.
[c]Decommissioning after 60 years of operation would increase occupational and public exposure during (1) decontamination and (2) decontamination truck shipments by only negligible amounts.
[d]Ranges are for removing or leaving internal components or leaving them in place. The higher exposures are associated with removing the internals.
Note: To convert person-rem to person-sievert, multiply by 0.01.

of operation is negligible (see the SAFSTOR 100-years alternative in Table 7.6), an increase of 50 percent of this negligible amount would still remain a negligible dose (less than 0.1 person-rem).

The negligible public radiation exposures for SAFSTOR preparation, continuing care, and decontamination (Table 7.6) include exposures from atmospheric and liquid releases during routine decommissioning operations. There are no historical records of significant releases during decommissioning, and no reliable estimates can be made of the probability and consequences of such events.

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However, the probability and consequences of such releases are not expected to be different for decommissioning a base case facility versus decommissioning a facility that has had 20 years of additional operation.

Extending reactor operating life from 40 to 60 years is expected to increase the concentration of long-half-life radionuclides in the facility by up to 50 percent. By the end of the initial 40 years of operation, the radionuclides with half-lives of less than about 5 years are at equilibrium because their rates of decay equal their rates of generation. The release of radioactivity to the atmosphere during decontamination is negligibly small and primarily involves short-lived nuclides. Public exposure even with the increased concentration of long-lived nuclides would remain negligible. The exposure of individual members of the public will be maintained well below existing regulatory limits. Accordingly, the staff concludes that the contribution of license renewal to radiological impacts from decontamination is of small significance. As discussed in Section 7.3.1.1, measures that can reduce possible dose levels to the public are available and are being employed in pursuit of ALARA.

Radiation doses (public and occupational) from decommissioning that are attributable to license renewal are a Category 1 issue.

7.3.2 Waste Management Impacts

An operating 1000-MW(e) reactor generates about 38 m^3 (1300 ft^3) of spent fuel and about 52,000 m^3 (1,800,000 ft^3) of LLW over its 40-year life (NUREG-0586, pp. 2–21). (LLW is defined in Chapter 6.) The reference PWR and BWR are about 15 percent larger, so they would be expected to generate about 15 percent more waste than a 1000-MW(e) plant. As shown by Table 7.4, decommissioning either type of plant after 40 years of operation (base case) would generate less than 15,000 m^3 (530,000 ft^3) of LLW for DECON or short-term SAFSTOR and less than 1,200 m^3 (42,000 ft^3) of LLW for SAFSTOR of 50 years or longer. These waste volumes include spent chelating agent used to decontaminate liquids. The 15,000 m^3 (530,000 ft^3) of decommissioning LLW is about 25 percent, and 1,200 m^3 (42,000 ft^3) is only about 2 percent, of the LLW generated by 40 years of operations. None of these estimates of waste volume includes waste generation during refurbishment.

Extending operations by 20 years would not increase decommissioning waste volumes, so the ratio of decommissioning waste volume to operating waste volume would be even lower. After 60 years of operation, decommissioning LLW would be less than about 20 percent of the operational LLW. If SAFSTOR were used, the decommissioning LLW would be only about 1 percent of the LLW generated by operations.

While the volume of decommissioning waste will not increase with 20 years of additional operating time, the concentration of long-half-life radionuclides will increase. LLW is classified by 10 CFR Part 61 into three waste classes denoted A, B, and C and a category of LLW designated "greater than Class C" (GTCC). Classes A and B are wastes that are contaminated with relatively short-half-life radionuclides and may be safely disposed of near the earth's surface because they will decay to a nonhazardous condition within about 100 years. Class C waste can be disposed of at a moderate depth or near the earth's surface with engineered barriers to prevent inadvertent intrusion into the wastes. GTCC waste cannot safely be disposed of

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near the earth's surface (Section 6.2.2.2; 10 CFR Part 61.7).

Table 7.7 gives the estimated decommissioning LLW breakdown (DECON scenario) for the base case by waste class per 10 CFR Part 61. Items classified as C and GTCC consist of highly activated metal located in the high-flux neutron field. For the PWR, the GTCC items include the lower core barrel, the thermal shields, the core shroud, and the separator assembly, the reactor vessel wall, and portions of the clean-up wastes.

The radionuclides of most importance for determining the classification of these LLWs are those that have relatively long half-life periods, such as ^{59}Ni and ^{94}Nb. These are also the radionuclides that accumulate in proportion with the length of reactor operation. The estimates in Table 7.7 are made for a plant that has operated 40 years. A plant that has operated 60 years would have essentially the same decommissioning waste volumes and classifications. Because the radionuclide concentration differences among waste classes are large (factors of 10 or more) and because the concentrations of radionuclides increase by no more than 50 percent, few components would be advanced to a higher classification by an additional 20 years of operations. Because the decommissioning waste volumes and classifications are essentially unchanged by an additional 20 years of plant operation, the Commission finds that the environmental impacts of decommissioning waste due to license renewal are of small significance. Measures employed within the context of ALARA, as discussed in Section 7.3.1.1, have the potential to reduce slightly the volume of LLW generated by decommissioning. The impact on decommissioning waste

lower grid plate. The class C items are the upper grid plate and the lower support column. The class B wastes consist of spent resins used during decommissioning, part of the combustible contaminated wastes, and part of the cylindrical pressure vessel wall. The only GTCC wastes from a BWR are the core shroud and top fuel guide. BWR class C wastes are from the control rods and in-core instrumentation, jet pump assemblies, and the top fuel guide. The class B wastes are from the steam

Table 7.7 Decommissioning waste volumes for reference pressurized-water reactor (PWR) and boiling-water reactor (BWR) after 40 years of operation[a]

	Class A	Class B/C	GTCC[b]
PWR	6,797 m^3	184 m^3	11 m^3
BWR	13,903 m^3	372 m^3	6.9 m^3

[a]DECON decommissioning method. Other methods would have smaller volumes of Class A and B wastes; Class C and GTCC wastes volumes would not change for other methods. A plant that has operated 60 years would have essentially the same waste volumes and classifications.
[b]GTCC = greater than Class C.
Source: NUREG/CR-5884 and NUREG/CR-6174.
Note: 1 m^3 ≈ 35.3 ft^3.

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management attributable to license renewal is a Category 1 issue.

7.3.3 Air Quality Impacts

Air quality impacts of decommissioning are expected to be negligible. No major land disturbance for construction laydown or temporary waste storage areas is anticipated. The principal air quality impacts would result from motor vehicles operated by workers for transportation on-site and for movement of people and materials to and from the site. Most decommissioning activities would be conducted inside the containment, the auxiliary building, and the fuel-handling buildings. Because there would be a possibility of airborne releases of radioactivity within these buildings during decommissioning, releases to the ambient environment would be controlled. These impacts would be much smaller than those associated with construction or demolition of the facilities on-site and would not change with 20 additional years of operation. License renewal and an additional 20 years of reactor operation will have no impact on air quality during decommissioning; thus the impact of license renewal on decommissioning air quality impacts is of small significance for all plants. Because license renewal does not affect the level of air pollution during decommissioning, there is no need for the consideration of mitigation as part of the license renewal environmental review. The impact of decommissioning on air quality attributable to license renewal is a Category 1 issue.

7.3.4 Water Quality Impacts

The principal water quality impacts expected from decommissioning are those associated with sanitary sewer operations. Because the decommissioning work force is likely to be smaller than those of construction and certain operational activities (see Section 7.3.7), no increase in water quality impacts is expected. Soil erosion and chemical spills associated with increased site activities during decommissioning have the potential to degrade water quality, but such effects are readily controllable. The potential for significant water quality impacts from erosion or spills is no greater if decommissioning occurs after a 20-year license renewal instead of after the original 40 years of operation. Measures to minimize occupational and public radiation exposure will also protect water quality. License renewal and an additional 20 years of reactor operation will have no impact on water quality during decommissioning; thus the impact is of small significance. Because license renewal does not affect water quality impacts during decommissioning, there is no need for the consideration of mitigation as part of the license renewal environmental review. The impact of decommissioning on water quality impacts attributable to license renewal is a Category 1 issue.

7.3.5 Ecological Impacts

Terrestrial biota impacts, if any, would be associated with land disturbance for laydown or temporary waste storage areas, and no such land disturbance is anticipated. No direct impacts to aquatic biota are expected from routine decommissioning activities. Measures employed to protect water quality will also prevent toxic effects to aquatic organisms from liquid effluents. Therefore, the ecological impacts associated with decommissioning are not expected to vary with the length of time the plant is operated. Decommissioning after a 20-year license renewal would have the same ecological impacts, if any, as decommissioning after 40 years of operation; thus the impact is of small significance. Because license renewal does

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not affect ecological impacts during decommissioning, there is no need for the consideration of mitigation as part of the license renewal environmental review. The impact of decommissioning on ecological resources attributable to license renewal is a Category 1 issue.

7.3.6 Economic Impacts

In general, the nature of the activities and the elements of the costs associated with decommissioning are well understood, and the necessary skills and equipment should be readily available when needed. Table 7.8 lists percentage estimates of total costs for decommissioning large PWR and BWR reactors by the DECON method.

A 1991 national survey had estimates that averaged $218 million per 1000 MW for a PWR reactor and $283 million per 1000 MW for a BWR. The standard deviation was $74 million for PWRs and $144 million for BWRs. For both types of reactors, the range for plus and minus one standard deviation was $131 million to $350 million (OTA-E-575). These varying estimates reflect the uncertainty of projecting costs well into the future. Additionally, the unique aspects of a plant's design and operating history can affect decommissioning costs (e.g., Three Mile Island Unit 2 and Fort St. Vrain).

The largest cost category is "undistributed"; the largest component of this cost is utility support staff. The timing of decommissioning could influence disposal costs depending on the price of disposal services. The current trend is steeply increasing cost per units of radioactive waste disposal. If this trend continues over the long run, then one effect of license renewal could be to increase decommissioning costs. However, disposal costs should stabilize by the time that most existing plants would be eligible for license renewal. If this is the case, license renewal would have a minimal effect on the undiscounted costs of decommissioning after a 20-year extended operation period, compared with after 40 years of operation.

For the cost estimates included in Table 7.8, doubling the cost per cubic foot of waste disposal would increase total decommissioning costs by about 13 percent for PWRs and 20 percent for BWRs. The assumed rate charged for disposal would have to increase by a factor of about 6 to double the total cost of decommissioning. If the rate of disposal costs turns out to be significantly more than has been assumed in decommissioning cost estimates, there would tend to be significantly more attention devoted to volume reduction; thus, total cost of disposal would tend to increase less than the proportional increase in the rate charged per cubic foot (NUREG/CR-5884, vol. 1, pp. 3.56, 3.57, and NUREG/CR-6174, vol. 1, p. 3.55).

The timing of decommissioning could also affect costs if progress in robotics technology reduces costs and worker radiation exposure. This progress would affect a relatively small part of the decommissioning process and thus is unlikely to reduce the total cost of decommissioning significantly; however, it could result in substantial dose reductions.

The preceding sections show that there is no reason to expect the physical requirements of decommissioning to be materially different when comparing the base case to a 20-year extended operation period. The undiscounted economic costs, although uncertain, should also be relatively stable and thus unaffected by license renewal. However, because of financial considerations, the timing of

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Table 7.8 Summary and distribution of decommissioning costs for large pressurized-water reactors (PWRs) and boiling-water reactors (BWRs) (thousands of 1993 dollars)

Decommissioning alternative	Duration[a] (years)	Decon[b] (%)	Removal[c] (%)	Packaging[d] (%)	Transport[e] (%)	Disposal[f] (%)	Undistributed[g] (%)	Present value[h] of total cost ($ × 10^3)	Present value[h] of savings[i] for license renewal ($ × 10^3)
				Pressurized-water reactor					
DECON	11	16.7	9.5	1.6	3.3	17.0	51.9	101,600	41,032
SAFESTOR1	59	11.0	0.5	0.3	1.0	3.4	83.8	93,000	37,559
SAFESTOR2	60	9.1	5.2	0.9	1.8	9.1	74.0	101,900	41,153
ENTOMB1	60	NA	NA	NA	NA	NA	NA	104,300	42,123
ENTOMB2	60	NA	NA	NA	NA	NA	NA	106,100	42,850
ENTOMB3	300	NA	NA	NA	NA	NA	NA	109,500	44,223
				Boiling-water reactor					
DECON	9	11.1	9.2	2.6	0.9	27.3	48.9	133,250	53,814
SAFESTOR1	59	7.6	1.0	0.2	0.5	3.1	87.5	121,600	49,109
SAFESTOR2	60	5.8	4.8	1.4	0.5	14.1	73.5	134,200	54,198
ENTOMB1	60	NA	NA	NA	NA	NA	NA	151,900	61,346
ENTOMB2	60	NA	NA	NA	NA	NA	NA	155,200	62,679
ENTOMB3	300	NA	NA	NA	NA	NA	NA	164,500	66,435

[a]Preshutdown period not included in duration total.
[b]Includes direct decommissioning labor and materials for chemical decontamination of systems, cleaning of surfaces, and waste water treatment.
[c]Includes direct labor and materials costs of removal.
[d]Includes direct costs of waste disposal packages.
[e]Includes cask rental costs and transportation costs.
[f]Includes all costs of disposal at the LLW disposal facility.
[g]Includes all costs that are period-dependent—e.g., decommissioning operations contractor (DOC) mobilization/demobilization, utility and DOC overhead staff, nuclear insurance, regulatory costs, plant power usage, taxes, laundry services, environmental monitoring. Most of the undistributed costs are for staffing.
[h]At 3 percent discount rate.
[i]The decommissioning costs have been discounted at a rate of 3 percent real (assumes no inflation). At this rate, delaying decommissioning by the 20-year period of license renewal saves about 45 percent of the decommissioning cost; however, present value total costs have been figured at 2.5 years from final plant shutdown, resulting in savings from license renewal of about 40 percent.
Source: Tables 3.1 and 4.1 and pp. 3.59, 4.13, and 5.13 of NUREG/CR-5884, Vol. 1; Tables 3.1 and 4.1 and pp. 3.58, 4.12, and 5.11 of NUREG/CR-6174, Vol. 1.

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decommissioning costs is important. To compare costs of activities that occur at different times, it is necessary to discount these costs to a common point in time. This is accomplished through present worth calculations, which account for the real opportunity cost or time value of money. Delaying decommissioning will allow any funds accumulated for this purpose to earn a return over the additional 20 years of license renewal and thus to reduce the present value of the decommissioning costs. The reduction in the present value is a function of the delay (license renewal period) and the time value of money, so the present value would be reduced by the same amount even if no fund were established and decommissioning were financed with borrowed money at the end of the plant operations. Regardless of how it is financed, the present value of delaying decommissioning costs will result in significant financial cost savings if a positive real discount rate is assumed.

Because total decommissioning costs are uncertain, the amount of financial savings that results from delaying decommissioning is also uncertain. Higher-than-expected decommissioning costs would result in higher cost savings resulting from delaying these costs, and vice versa. At a 3 percent real (i.e., above general inflation) discount rate, the present value savings associated with license renewal is about 40 percent of decommissioning costs (Table 7.8). Real cost increases, which might occur for waste disposal costs, could reduce the cost advantage of license renewal, but waste disposal costs are expected to stabilize before the current licenses of most plants expire. The impact of license renewal on decommissioning costs is not a consideration in the environmental review and decision whether to renew a license.

7.3.7 Socioeconomic Impacts

Socioeconomic impacts associated with decommissioning will be induced by the net change in the labor force as incoming decommissioning workers replace emigrating operations workers. The nature of these impacts will depend on the vitality of local economic activity at the time of decommissioning.

One of the difficulties of attempting to evaluate the socioeconomic impacts of decommissioning in year 40 of a plant's life compared with decommissioning in year 60 relates to the uncertainties about the size of the work force required. The largest nuclear power plant decommissioned to date has been the 150-MW(e) Shippingport Station (Section 7.2.3), which required an average work force during the peak year of approximately 230 workers (DOE/SSDP-0081); this work force was larger than the estimated work forces for very large power plants examined in studies prepared before the Shippingport experience (NUREG/CR-0130, Table 9.1-1; NUREG/CR-0672, Table 9.1-3). Because more-recent manpower estimates for large nuclear power plants are not available, the actual work force required in the future might be substantially larger than currently expected.

If the decommissioning process requires a smaller work force than the on-site operating staff and if the local economy is stable or declining, the result could be economic hardships, including declining property values and business activity, and problems for local government as it adjusts to lower levels of tax revenues. However, even this reduced work force will tend to mitigate temporarily the full adverse socioeconomic effects of terminating operations.

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If there is a net reduction in the community work force but the economy is growing, the adverse impacts of this ongoing growth (e.g., housing shortages and school overcrowding) could be reduced.

If the decommissioning work force were substantially larger than the operational work force, the result could be increased demand for housing and public services but also increased tax revenues and higher real estate values. If the economy is characterized by decline, decommissioning could temporarily reverse the adverse economic effects.

In a stable economy, a net increase in the community work force could lead to some shortages in housing and public services, as well as to the higher tax revenues and real estate values mentioned previously. In a growing economy, decommissioning could act as an exacerbating factor to the ongoing shortages that already might exist.

Although the staff cannot project with certainty either the size of the required decommissioning work force or the state of the local economy at the time of decommissioning, the staff has assumed that the baseline conditions will be negligibly different in year 40, compared with year 60. Therefore, the staff expects that the socioeconomic impacts of decommissioning would be essentially similar whether that action were taken in year 60 or in year 40. The impact of license renewal on the socioeconomic impacts of decommissioning are of small significance. Because license renewal does not affect the socioeconomic impacts that will occur at the time of decommissioning, there is no need for the consideration of mitigation as part of the license renewal environmental review. The impact of decommissioning on socioeconomic resources attributable to license renewal is a Category 1 issue.

7.4 CONCLUSIONS

The physical requirements and attendant effects of decommissioning nuclear power plants after a 20-year license renewal are not expected to differ from those of decommissioning at the end of 40 years of operation. Decommissioning after a 20-year license renewal would increase the occupational dose no more than 0.1 person-rem (compared with 7,000 to 14,000 person-rem for DECON decommissioning at 40 years) and the public dose by a negligible amount (Section 7.3.1). License renewal would not increase to any appreciable extent the quantity or classification of LLW generated by decommissioning (Section 7.3.2). Air quality, water quality, and ecological impacts of decommissioning would not change as a result of license renewal (Sections 7.3.3, 7.3.4, and 7.3.5). There is considerable uncertainty about the cost of decommissioning; however, while license renewal would not be expected to change the ultimate cost of decommissioning, it would reduce the present value of the cost (Section 7.3.6). The socioeconomic effects of decommissioning will depend on the magnitude of the decommissioning effort, the size of the community, and the other economic activities at the time, but the impacts will not be increased by decommissioning at the end of a 20-year license renewal instead of at the end of 40 years of operation (Section 7.3.7). Incremental radiation doses, waste management, air quality, water quality, ecological, and socioeconomic impacts of decommissioning due to operations during a 20-year license renewal term would be of small significance. No mitigation measures beyond those provided by ALARA are warranted within the context of the license

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renewal process. The impacts of license renewal on radiation doses, waste management, air quality, water quality, ecological resources, and socioeconomics impacts from decommissioning are Category 1 issues.

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8. ALTERNATIVES TO LICENSE RENEWAL

8.1 INTRODUCTION

The Nuclear Regulatory Commission's (NRC's) environmental review regulations implementing the National Environmental Policy Act (NEPA) (10 CFR Part 51) require that the NRC consider all reasonable alternatives to a proposed action before acting on a proposal, including consideration of the no-action alternative. The intent of such a consideration is to enable the agency to consider the relative environmental consequences of an action given the environmental consequences of other activities that also meet the purpose of the action, as well as the environmental consequences of taking no action at all. The information in this chapter does not constitute NRC's final consideration of alternatives to license renewal. Therefore, the rule accompanying this Generic Environmental Impact Statement (GEIS) does not contain any conclusions regarding the environmental impact or acceptability of alternatives to license renewal. Accordingly, the NRC will conduct a full analysis of alternatives at individual license renewal reviews. NRC expects that information contained in this chapter will be used in the analysis of alternatives for the supplemental environmental impact statements prepared for individual license renewals. As defined in Chapter 1, the proposed action is the granting of a renewed license. Additionally, the purpose of such a proposal is to provide an option that allows for power generation capability beyond the term of a current nuclear power plant operating license in order to meet future system generating needs as such needs may be determined by state, utility, and, where authorized, federal (other than NRC) decision makers. This chapter examines the potential environmental impacts associated with denying a renewed license (i.e., the no action alternative); the potential environmental impacts from electric generating sources other than nuclear license renewal; the potential impacts from instituting additional conservation resources to reduce the total demand for power; and the potential impacts from power imports.

The no-action alternative is the denial of a renewed license. In general, if a renewed license were denied, a plant would be decommissioned and other electric generating sources would be pursued if power were still needed. It is important to note that NRC's consideration of the no-action alternative does not involve the determination of whether any power is needed or should be generated. The decision to generate power and the determination of how much power is needed are at the discretion of state and utility officials.

While many methods are available for generating electricity, and a huge number of combinations or mixes can be assimilated to meet a defined generating requirement, such expansive consideration would be too unwieldy to perform given the purposes of this analysis. Therefore, NRC has determined that a reasonable set of alternatives should be limited to analysis of single, discrete electric generation sources and only electric generation sources that are technically feasible and commercially viable.

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To generate this reasonable set of alternatives, NRC included commonly known generation technologies and consulted various state energy plans to identify the alternative generation sources typically being considered by state authorities across the country. From this review, NRC has established a reasonable set of alternatives to be examined in this chapter. These alternatives include wind energy, photovoltaic (PV) cells, solar thermal energy, hydroelectricity, geothermal energy, incineration of wood waste and municipal solid waste (MSW), energy crops, coal, natural gas, oil, advanced light water reactors (LWRs), and delayed retirement of existing non-nuclear plants. NRC has considered these alternatives pursuant to its statutory responsibility under NEPA. NRC's analysis of these issues in no way preempts or displaces state authority to consider and make decisions regarding energy planning issues.

This chapter also includes a discussion of conservation and power import alternatives. Although these alternatives do not represent discrete power generation sources, they represent options that states and utilities may use to reduce their need for power generation capability. In addition, energy conservation and power imports are possible consequences of the no-action alternative. While these two alternatives are not options that fulfill the stated purpose and need of the proposed action *per se* (i.e., options that provide power generation capability), they nevertheless are considered in this chapter because they are important tools available to energy planners in managing need for power and generating capacity.

The potential environmental impacts evaluated include land use, ecology, aesthetics, water quality, air quality, solid waste, human health, socioeconomics, and culture. These impacts are addressed in terms of construction impacts and operational impacts (Tables 8.1 and 8.2, respectively). This chapter occasionally mentions economic costs of particular alternatives for descriptive purposes; they do not provide a basis for an NRC decision on license renewal. In addition such economic costs may change prior to specific license renewal decisions as improvements occur to particular technologies. Additionally, this chapter discusses the relative construction and operating costs of various technologies where available.

8.2 ENVIRONMENTAL IMPACTS OF THE NO-ACTION ALTERNATIVE

As discussed in the introduction, the no-action alternative is denial of a renewed license. Denial of a renewed license as a power generating capability may lead to a variety of potential outcomes. In some cases, denial may lead to the selection of other electric generating sources to meet energy demands as determined by appropriate state and utility officials. In other cases, denial may lead to conservation measures and/or decisions to import power. In addition, denial may result in a combination of these different outcomes. Therefore, the environmental impacts of such resulting alternatives would be included as the environmental impacts of the no-action alternative. Additionally, a denial of a renewed license would lead to facility decommissioning and its associated impacts; these impacts would also represent impacts of the no-action alternative.

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Table 8.1 Environmental impacts of constructing 1000-MW(e)-equivalent electric power plants for non-nuclear alternative generating technologies

	Resource								
Alternative	Land use	Ecology	Aesthetics	Water quality	Air quality	Waste	Human health	Socioeconomic	Cultural
Wind	61,000 ha (150,000 acres) (Pimentel 1994)	Loss of thousands of acres of natural habitat (Pimentel 1994); some stream sedimentation; erosion	Substantial visual impact in any location (Pimentel 1994; SERI/TP-260-3674)	High potential for sedimentation/erosion damage	Considerable vehicle exhaust, dust from earth moving	Considerable amount of vegetation debris from land clearing	Some accident risks for workers (Grubb and Meyer 1993)	No known estimates but believed to be relatively small peak work force—little potential for adverse impacts	High potential for impacts because of large land area
Photovoltaic cells	14,000 ha (35,000 acres) (Pimentel 1994; Pace 1991)	Loss of 14,000 ha (35,000 acres) of natural habitat, some farm land (Pimentel 1994); some stream sedimentation; erosion is a particular threat to arid areas, fragile soil, and plant communities	Substantial visual impact in any location (Hamrin and Rader 1993)	High potential for sedimentation/erosion damage	Considerable vehicle exhaust, dust from earth moving (Pace 1991)	Considerable amount of vegetation debris from land clearing	Some accident risks for workers	No known estimates but believed to be moderate size peak work force—little potential for adverse impacts	High potential for impacts because of large land area

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Table 8.1 (continued)

Alternative	Resource								
	Land use	Ecology	Aesthetics	Water quality	Air quality	Waste	Human health	Socioeconomic	Cultural
Solar thermal	5,700 ha (14,000 acres) (Pimentel 1994; Pace 1991)	Loss of 5,700 ha (14,000 acres) (Pimentel 1994); some stream sedimentation; erosion is a particular threat to arid areas, fragile soil, and plant communities (Pace 1991)	Substantial visual impact to 5,700 ha (14,000 acres) affected (Pimentel 1994; Pace 1991; Hamrin and Rader 1993)	High potential for sedimentation/erosion damage	Considerable vehicle exhaust, dust from earth moving (Pace 1991)	Considerable amount of vegetation debris from land clearing	Some accident risks for workers	No known estimates but believed to be moderate size peak work force—little potential for adverse impacts	High potential for impacts because of large land area
Hydroelectric	400,000 ha (1 million acres) (Pimentel 1994)	Loss of 400,000 ha (1 million acres) of natural habitat, farm land (Pimentel 1994); stream sedimentation, erosion	400,000 ha (1 million acres) visually impacted (Pimentel 1994; Hamrin and Rader 1993)	Considerable sedimentation/ erosion	Considerable vehicle exhaust, dust from earth moving	Considerable amount of vegetation debris from land clearing	Some accident risks for workers; spread of diseases from reservoir filling (Moreira and Poole 1993)	Large work force, moderate potential for adverse community impacts; dislocation of residents (Hamrin and Rader 1993)	Almost unavoidable destruction of cultural sites, artifacts typically located on natural edges of water bodies
Geothermal	2800 ha (7000 acres) (DOE/EP-0093)	Loss of 2800 ha (7000 acres) of natural habitat (DOE-0093); some stream sedimentation, erosion	Visual impacts to 2800 ha (7000 acres) (DOE-0093)	High potential for sedimentation/erosion damage	Considerable vehicle exhaust, dust from earth moving	Considerable amount of vegetation, some construction debris	Some accident risks for workers	Moderate size work force; some potential adverse impacts	Moderate potential unless important site-specific resource affected by plant or transmission lines

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Table 8.1 (continued)

Alternative	Resource								
	Land use	Ecology	Aesthetics	Water quality	Air quality	Waste	Human health	Socioeconomic	Cultural
Wood wastes	High variable and site specific, perhaps 160,000 to 320,000 ha (400,000 to 800,000 acres) for forest residue recovery. For plant, about 30 acres for each 20-MW facility	Considerable potential for loss of natural habitat and biodiversity; increased soil erosion and nutrient loss (ECO Northwest et al.)	Substantial visual impacts from land clearing. Localized visual impacts with plant construction	High potential for sedimentation/erosion damage. Small sedimentation/erosion damage at plant site (ECO Northwest et al.)	Considerable vehicle exhaust and fugitive dust impacts from earth moving	Considerable amount of vegetation debris and some construction debris	Some accident risks for workers	Source of income and employment in rural areas. Moderate size work force at plant site	High potential for impacts because of large land area
Municipal solid waste (MSW)	For plant, about 12 ha (30 acres) for each 20 MW facility	Small impact—few acres affected and in urban area. Potentially positive impacts if landfills displaced (ECO Northwest et al.)	Localized visual impacts with plant construction	Small sedimentation/erosion damage at plant site (ECO Northwest et al.)	Considerable vehicle exhaust and fugitive dust impacts from earth moving	Moderate amount of vegetation and construction debris	Some accident risks for workers	Moderate size work force at plant site	Relatively small unless important site-specific resource affected by plant or transmission lines

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Table 8.1 (continued)

Alternative	Resource								
	Land use	Ecology	Aesthetics	Water quality	Air quality	Waste	Human health	Socioeconomic	Cultural
Energy crops	About 400,000 ha (1 million) acres for crop production. For plant, about 12 ha (30 acres) for each 20 MW facility	Impacts depend on prior land use; if conversion of cropland, then more environmentally benign and would improve biodiversity (OTA; Ranney and Mann)	Minor visual impacts with energy crop establishment. Localized visual impacts with plant construction	Energy crops lower sedimentation, soil erosion, and chemical use relative to agriculture (Ranney and Mann). Small sedimentation and erosion damage at plant site	Moderate vehicle exhaust and fugitive dust impacts from earth moving at plant site	Considerable amount of vegetation debris and some construction debris at plant site	Some accident risks for workers	Source of income and employment in rural areas. Moderate size work force at plant site	Relatively small impacts if cropland and pasture converted to energy crops
Coal	700 ha (1,700 acres) for plant site (DOE/EP-0093)	Loss of 700 ha (1,700 acres) habitat; some erosion, stream sedimentation	Localized visual impacts from land clearing	Potential sedimentation/ erosion damage	Moderate vehicle exhaust, dust from earth moving	Considerable construction debris	Accident risk for workers	1,200–2,500 peak work force (UDI-021-89)	Relatively small unless important site-specific resource affected by plant or transmission lines
Natural gas	45 ha (110 acres) for plant site (DOE/EP-0093)	Loss of 45 ha (110 acres) varied habitat; some erosion, stream sedimentation	Localized visual impacts from land clearing	Potential sedimentation/ erosion damage	Some vehicle exhaust, substantial dust from earth moving	Considerable construction debris	Accident risk for workers	1,200 peak work force (UDI-021-89)	Relatively small unless important site-specific resource affected by plant or transmission lines
Oil	50 ha (120 acres) for plant site (DOE/EP-0093)	Loss of 50 ha (120 acres) varied habitat; some erosion, stream sedimentation	Localized visual impacts from land clearing	Potential sedimentation/ erosion damage	Some vehicle exhaust, substantial dust from earth moving	Considerable construction debris	Accident risk for workers	1,700 peak work force (UDI-021-89)	Relatively small unless important site-specific resource affected by plant or transmission lines

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Table 8.1 (continued)

Alternative	Resource								
	Land use	Ecology	Aesthetics	Water quality	Air quality	Waste	Human health	Socioeconomic	Cultural
Advanced light-water reactor	200–400 ha (500–1,000 acres) for plant site plus exclusion area	Loss of 200–400 ha (500–1,000 acres) of habitat; some erosion, stream sedimentation	Localized visual impacts from land clearing	Potential sedimentation/ erosion damage	Moderate vehicle exhaust, dust from earth moving	Considerable construction debris	Accident risk for workers	2,000–5,500 peak work force (UDI-021-89)	Relatively small unless important site-specific resource affected by plant or transmission lines
Conservation	Unquantified land lost to resource extraction for conservation technologies	Adverse impacts from resource extraction	Minimal for resource recovery and processing	Minimal for resource recovery and processing	Minimal for resource recovery and processing	Minimal for resource recovery, processing	Some risks from resource recovery	Minor employment, tax revenues from conservation industry	Minimal
Imported power	If excess Canadian capacity is insufficient, impacts will be similar to U.S. coal or hydro plants	If excess Canadian capacity is insufficient, impacts will be similar to U.S. coal or hydro plants	If excess Canadian capacity is insufficient, impacts will be similar to U.S. coal or hydro plants	If excess Canadian capacity is insufficient, impacts will be similar to U.S. coal or hydro plants	If excess Canadian capacity is insufficient, impacts will be similar to U.S. coal or hydro plants	If excess Canadian capacity is insufficient, impacts will be similar to U.S. coal or hydro plants	If excess Canadian capacity is insufficient, impacts will be similar to U.S. coal or hydro plants	If excess Canadian capacity is insufficient, impacts will be similar to U.S. coal or hydro plants	Same impacts as U.S. except northern Canada, where social conflict between tribes and government is substantial
Delayed retirement	Very few acres affected (DOE/EIS-0146)	Very few acres affected—no impact (DOE/EIS-0146)	Minimal changes	Incidental use	Small exhaust, fugitive dust (DOE/EIS-0146)	Moderate construction debris	Potential accidents to workers	Estimated one-half of normal construction work force	Minimal impact

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Table 8.2 Environmental impacts of operating 1000-MW(e)-equivalent electric power plants for non-nuclear alternative generating technologies

	Resource								
Alternative	Land use	Ecology	Aesthetics	Water quality	Air quality	Waste	Human health	Socioeconomic	Cultural
Wind	61,000 ha (150,000 acres) of which 3,000 acres occupied by turbines, rest available for agriculture (Pimentel 1994)	Bird collisions, loss of much of thousands of acres of habitat (Pimentel 1994); interference with animal migration routes (Pace 1991)	Substantial visual and some noise impact in any location (Pace 1991; SERI/TP-260-3674; Rader 1989)	Negligible (Pace 1991)	Negligible (Pace 1991)	Very minor amounts from maintenance of equipment, vegetation	Very minor risks from accidents, noise	Relatively low work force, assessed plant value—fewer potential long-term community benefits than large baseload plants	Relatively small unless important site-specific resource affected by plant or transmission lines
Photovoltaic cells	14,000 ha (35,000 acres); no other compatible uses (Pimentel 1994; Pace 1991)	Loss of 14,000 ha (35,000 acres) of natural habitat and some agricultural land (Pimentel 1994)	Substantial visual impact in any location (Hamrin and Rader 1993)	Small runoff from panels could cause sedimentation	Negligible	Very minor amounts from maintenance of equipment, vegetation; some toxics	Some risk to maintenance workers	Relatively small work force, assessed plant value—fewer long-term community benefits than large baseload plants	Relatively small unless important site-specific resource affected by plant or transmission lines
Solar thermal	5,700 ha (14,000 acres); no other uses (Pimentel 1994; Pace 1991)	5,700 ha (14,000 acres) of natural habitat lost and some agricultural land (Pimentel 1994)	Substantial visual impact; reflected sunlight (Pimentel 1994; Pace 1991; Hamrin and Rader 1993)	Minor amounts used except where water is cooling agent (Rader 1989); possible contamination from cleaning agents (Rader 1989); some runoff potential	Minor emissions of pollutants during normal operations, greater risks with accidents (Pimentel 1994)	Very minor amounts from maintenance of equipment, vegetation	Possible eye damage from reflected sunlight; occupational hazards from exposure to heat transfer fluids (Pace 1991); some risk to maintenance workers	Relatively small work force, assessed plant value—fewer long-term community benefits than large baseload plants	Relatively small unless important site-specific resource affected by plant or transmission lines

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Table 8.2 (continued)

Alternative	Resource								
	Land use	Ecology	Aesthetics	Water quality	Air quality	Waste	Human health	Socioeconomic	Cultural
Hydroelectric	400,000 ha (1 million acres); no other uses (Pimentel 1994)	400,000 ha (1 million acres) of natural habitat and agricultural lands lost; disruption of spawning, migration routes (Rader 1989); killing of fish thru eutro-phication, passage through dam, water temperature change (Moreira and Poole 1993); altered flora, fauna populations	1 million acres visually impacted (Pimentel 1994; Hamrin and Rader 1993)	Increased sedimentation (Moreira and Poole 1993); temperature changes, competition for water and arid regions (Rader 1989)	Negligible	Minor amounts from equipment replacement, reservoir clearing	Some risks for recreational boating, swimming deaths; risk of dam failure; some risk to maintenance workers	Small work force, high assessed value—some potential long-term economic/ community impacts, changes in recreation (free-flowing stream to lake)	Relatively small unless important site-specific resource affected by plant or transmission lines

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Table 8.2 (continued)

Alternative	Resource								
	Land use	Ecology	Aesthetics	Water quality	Air quality	Waste	Human health	Socioeconomic	Cultural
Geothermal	2800 ha (7000 acres)—possible subsidence; potential for other uses on unused land (DOE/EP-0093; OECD 1987)	Loss of much of 2800 ha (7000 acres) of natural habitat and some agricultural land (DOE/EP-0093)	Visual impacts to portions of affected areas (Rader)	Potential contamination of surface and groundwater from disposal of geothermal fluid (OECD 1987)	Potential release of various toxic gases to atmosphere, especially H_2S; CO_2 is greatest emission (Pace 1991, Brower 1992)	Minor amounts from equipment replacement, vegetation maintenance, heavy metals sludge (Brower 1992)	Very minor risks from toxic gas released, accidents to workers; noise (Brower 1992)	Relatively small work force, assessed plant value—fewer long-term community benefits than large baseload plants	Relatively small unless important site-specific resource affected by plant or transmission lines
Wood wastes	About 160,000 to 320,000 ha (400,000 to 800,000 acres) for forest residue recovery. About 12 ha (30 acres) per 20 MW of facility operated (OTA 1993)	Considerable potential for loss of natural habitat and biodiversity; increased soil erosion and nutrient loss (OTA 1993)	Some visual impacts from residue recovery. Limited visual impacts from plant structure	Approximately same water requirements as coal	Not significant with residue recovery. Emission of regulated pollutants, can be effectively controlled	Considerable fly ash, can be used as fertilizer and soil conditioner	Occupational risks high, same as for agriculture. Particulates important, but can be controlled	Source of income and employment in rural areas. Moderate size work force at plant site	Relatively small unless important site-specific resource affected by residue recovery area, plant or transmission lines
Municipal solid waste	About 12 ha (30 acres) per 20 MW of facility operated	Potentially positive impacts if landfills are displaced	Limited visual impacts from plant structure. Potential odors	Approximately same water requirements as coal	Emissions of regulated pollutants more significant than other technologies	Considerable fly ash, must meet regulations	Risks from toxics and particulates, safety of municipal solid waste handlers	Moderate size work force at plant sites	Relatively small unless important site-specific resource affected by plant or transmission lines

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Table 8.2 (continued)

Alternative	Resource								
	Land use	Ecology	Aesthetics	Water quality	Air quality	Waste	Human health	Socioeconomic	Cultural
Energy crops	About 400,000 ha (1 million acres) for crop production. About 12 ha (30 acres) per 20 MW of facility operated	Impacts depend on prior land use, may either enhance or reduce biodiversity, habitat (Wright 1994; Ranney and Mann 1994)	Some visual impacts from harvesting. Limited visual impacts from plant structure	Irrigation not used for growing. Approximately same water requirements as coal	Not significant with production of energy crops. Emissions of regulated pollutants, can be effectively controlled (Wright 1994)	Considerable fly ash, can be used as fertilizer and soil conditioner	Occupational risks high, same as for agriculture. Particulate important, but can be controlled (Rader 1989)	Source of income and employment in rural areas. Moderate size work force at plant site	Relatively small unless important site-specific resource affected by cropping area, plant or transmission lines
Coal	700 ha (1,700 acres) for plant site (DOE/EP-0093) and 9070 ha (22,400 acres) for entire fuel cycle (WASH-1224)	Habitat loss (including nationally from acid precipitation; DOE/EIS-0146); impingement, entrainment; waste heat to receiving water body; cooling tower drift, fogging; bird collisions	Limited visual impacts from plant structure, additional from plume	860,000 m^3 (700 acre-ft) per quad (10^{12} Btu) energy produced (based on thermal efficiency relative to nuclear)	Emission of CO_2, regulated pollutants, more than other technologies (Loftness 1984); also radionuclides	Large amounts of fly ash, scrubber sludge, other solid waste—must meet regulations (DOE/EP-0093)	Public risks (cancer, emphysema) from inhalation of toxics and particulates; safety risk to workers	250 workers—moderate long-term economic community benefits (UDI-021-89)	Relatively small unless important site-specific resource affected by plant or transmission lines

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Table 8.2 (continued)

Alternative	Resource								
	Land use	Ecology	Aesthetics	Water quality	Air quality	Waste	Human health	Socioeconomic	Cultural
Natural gas	45 ha (110 acres) for plant site (DOE-EP-0093) and 1500 ha (3,600 acres) for entire fuel cycle (WASH-1224)	Habitat loss, impingement, entrainment; waste heat to receiving water body; cooling tower drift, fogging; bird collisions	Limited visual impacts from plant structure, some from plume	817,000 m^3 (662 acre-ft) water used per quad (10^{12} Btu) energy produced (based on thermal efficiency relative to nuclear)	Emissions of CO_2 and NO_x regulated pollutants, radionuclides less than coal, no SO_2 (Loftness 1984)	Some solid waste produced—must meet regulations (DOE/EP-0093)	Some public risks (cancer, emphysema) from inhalation of toxics and particulates; safety risk to workers	150 workers—moderate long-term economic, community benefits (UDI-021-89)	Relatively small unless important site-specific resource affected by plant or transmission lines
Oil	50 ha (120 acres) for plant site (DOE/EP-0093) and 650 ha (1,600 acres) for entire fuel cycle (WASH-1224)	Habitat loss (including nationally from acid precipitation; DOE/EIS-0146); impingement, entrainment; waste heat to receiving water body; cooling tower drift, fogging; bird collisions	Limited visual impacts from plant structure, some from plume	860,000 m^3 (700 acre-ft) water per quad (10^{12} Btu) energy produced (based on thermal efficiency relative to nuclear)	Emissions of CO_2, SO_2, and NO_x, regulated pollutants, radionuclides less than coal (Loftness)	Moderate (<coal) amounts of scrubber sludge, particulates—must meet regulations (DOE/EP-0093)	Some public risks (cancer, emphysema) from inhalation of toxics and particulates; safety risks to workers	200 workers—moderate long-term economic community benefits (UDI-021-89)	Relatively small unless important site-specific resource affected by plant or transmission lines

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Table 8.2 (continued)

Alternative	Resource								
	Land use	Ecology	Aesthetics	Water quality	Air quality	Waste	Human health	Socioeconomic	Cultural
Advanced light-water reactor	80–200 ha (500–1000 acres) for plant site, plus exclusion acres and 400 ha (1500–2000 acres) for entire fuel cycle	Habitat loss, impingement, entrainment; waste heat to receiving water bodies; cooling tower drift and fogging; bird collisions	Limited visual impacts from plant structure, some from plume	910,000 m^3 (740 acre-ft) water per quad (10^{12} Btu) energy produced (based on thermal efficiency relative to nuclear)	Very little CO_2 or regulated pollutants—from vehicles not facility	Some spent fuel, slightly more mixed waste and low-level waste than license renewal	<1% of natural radiation sources; safety risks to workers	700 workers—substantial long-term economic, community benefits (UDI-021-89)	Relatively small unless important site-specific resource affected by plant or transmission lines
Conservation	Minimal	Minimal	Minimal	Minimal	Minimal	Minimal	Minor impacts regarding radon, perhaps other contaminants (Pace 1991)	Increased jobs in conservation technologies	Minimal
Imported power	Operating impacts of hydro and coal plants similar to those in U.S.	Operating impacts of hydro and coal plants similar to those in U.S.	Operating impacts of hydro and coal plants similar to those in U.S.	Operating impacts of hydro and coal plants similar to those in U.S.	Operating impacts of hydro and coal plants similar to those in U.S.	Operating impacts of hydro and coal plants similar to those in U.S.	Operating impacts of hydro and coal plants similar to those in U.S.	Operating impacts of hydro and coal plants similar to those in U.S.	Cultural impacts to tribes in northern Canada could produce more social conflict than in United States

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Table 8.2 (continued)

Alternative	Resource								
	Land use	Ecology	Aesthetics	Water quality	Air quality	Waste	Human health	Socioeconomic	Cultural
Delayed retirement	Very few acres affected (DOE/EIS-0146)	Very few acres affected—no impact	Minimal changes unless cooling tower installed	Substantial improvement if closed-cycle system replaces once-through (Bretz 1994), otherwise little change. Improvement to distant water bodies adversely affected by acid precipitation (DOE/EIS-0146)	> 90% SO_2 and NO_x emissions of conventional coal plant removed (DOE/EIS-0146, Bretz 1994)	For integrated gasification combined cycle: 40% waste of pulverized coal plant; for atmospheric fluidized bed: possibly double the amount from pulverized coal plant (DOE/EIS-0146)	Substantial public health improvement compared with conventional, pulverized coal plant; safety risks to workers	Moderate employment and tax revenue from first coal plant extended for longer period	No change

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The environmental impacts expected from decommissioning are analyzed in NUREG-0586, *Final Generic Environmental Impact Statement of Decommissioning of Nuclear Facilities* (1988). Consequently, NUREG-0586 represents some of the environmental impacts associated with denial of a renewed license. The analysis in Section 8.3 is equally applicable to the no-action alternative in that the alternatives analyzed in this section are all possible actions resulting from denial of a renewed license. Therefore, Section 8.3 represents additional impacts of the no-action alternative.

8.3 ENVIRONMENTAL IMPACTS OF ALTERNATIVE ENERGY SOURCES

This section describes the technologies and evaluates the environmental impacts of 13 energy supply or demand alternatives identified by NRC as capable of satisfying the purpose and need of the proposed action [i.e., to provide an option that allows for power generation capability beyond the term of a current nuclear power plant operating license to meet future system generating needs as such needs may be determined by state, utility, and, where authorized, federal (other than NRC) decision makers]. The technologies were selected because they correspond with those generally considered in state energy plans as potential generating technologies, or they were proposed as alternatives to nuclear license renewal in comments to the Draft GEIS. Many of these technologies differ dramatically from nuclear, and it is important to evaluate them using a consistent standard. A reference generating capacity of 1000 MW(e) is used in evaluating environmental impacts, because this is the approximate generating capacity of many nuclear plants.

The section evaluates impacts that could occur during construction (Table 8.1) or operation (Table 8.2) of each alternative technology. Environmental resources considered include land use, ecology, aesthetics, water quality, air quality, human health, socioeconomics, and cultural resources. The tables provide more detailed information, and the text highlights the more important impacts. References are omitted in the text when they are included in the impact tables.

License renewal decisions may vary considerably among states and utilities based on numerous factors, of which environmental factors are but one set. These decisions may be reached by utilities and states prior to NRC involvement. NRC staff evaluated the process used by 10 states with nuclear power plants to decide which electricity supply and demand options to implement. (NRC examined state energy plans of California, Florida, Illinois, Massachusetts, Michigan, Minnesota, New York, Texas, Vermont, and Wisconsin.) NRC determined that integrated resource planning in some form is used in almost all of these states. Nuclear technology and license renewal are not emphasized in most of these plans, which are developed by either state energy offices or state public service commissions. It is apparent in the plans that nuclear generating plants submitted for license renewal would be required to demonstrate the overall benefits of license renewal over alternative technologies before states would approve renewal. The options would include large, central generating stations powered by nonrenewable sources of energy, probably coal or natural gas, or advanced technologies powered by those

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same fuels. Some states not enamored of conventional nuclear power may be amenable to considering advanced nuclear technologies. Renewable energy sources have the potential to replace at least some of the generating capacity lost through decommissioning nuclear plants. Solar thermal energy, PV cells, wind energy, hydroelectricity, energy crops, and incineration of MSW and wood waste have some potential in most states surveyed. Geothermal energy has potential in states like California where the resource is prevalent.

Besides sources of power generation, other alternatives are mentioned in state energy plans. Demand-side management (DSM) is viewed in every state as a means to help meet electricity forecasts. Other alternatives include end-use conservation and purchases of power from other utility systems in the United States, Canada, or Mexico. While these two alternatives are not options that fulfill the stated purpose and need of the proposed action *per se* (i.e., options that provide power generation capability), they nevertheless are considered in this section because they are important tools available to energy planners in managing needs for power and generating capacity.

Every technology discussed in this section could generate power in much smaller facilities than 1000 MW(e) in dispersed locations throughout a utility's service area. Typically, conservation or demand-side alternatives and renewable technologies lend themselves best to relatively small facilities, whereas conventional, nonrenewable technologies are suited more for large central generating stations. Numerous exceptions to these generalizations exist or are feasible. Thus, multiple alternatives could be selected to replace a single nuclear plant. For example, a utility and state public utility commission could agree that a combination of 500 MW(e) of conventional or advanced-technology coal, 100 MW(e) of conservation, 100 MW(e) of purchased power, 50 MW(e) of wind power, 50 MW(e) of MSW combustion, and 200 MW(e) of combined-cycle-generation would be the preferred set of alternatives to replace a single nuclear plant. This siting scenario would be expected to diffuse over a wider area the construction and operational impacts otherwise expected from a single 1000-MW(e) facility. It also could be feasible to replace a nuclear plant with an equal amount of capacity from a single technology sited in a dispersed fashion. The types and general magnitude of environmental impacts would be about the same as for a central generating facility using that technology, but impacts would be dispersed in smaller concentrations over a wider area.

The following discussion is intended to suggest generic impacts that could occur from each technology as well as approximations of the magnitude of those impacts. In addition, this discussion is intended to address the reasonably foreseeable impacts of the various alternatives and does not attempt to address impacts that are remote or speculative. In the cases of conservation and renewable technologies, where there are no current equivalents to 1000-MW(e) plants, the impact data are less reliable than for nonrenewable technologies. The GEIS depends on data gathered from many studies, and the data may not always be comparable among technologies.

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8.3.1 Wind

Of the approximately 33,000 quads of wind resources available annually in the coterminous United States, only about 170 quads per year can be accessed with current technology, and only about 1/6 quad per year can currently be used cost-effectively to generate electricity (DOE/EIA-0561). Wind speeds of at least 21 km/h (13 miles/h) are considered necessary for generating electricity. As shown in Figure 8.1, regions with such speeds include the Great Plains, the West, coastal areas, and parts of the Appalachians (DOE/EIA-0561).

The average annual capacity factor (i.e., the proportion of actual generation to potential generation at 100 percent capacity utilization) is estimated at 21 percent in 1995 and 29 percent in 2010. This relatively low capacity, compared with current baseload technologies, results from the high degree of intermittency of wind energy in many locations (DOE/EIA-0561). Current energy storage technologies are too expensive to permit wind power plants to serve as large baseload plants. The inability to increase the capacity factors of wind power makes the technology an inappropriate choice for baseload power (Johansson et al. 1993)

In 1992, wind provided 1676 MW(e) of electric generating capacity, produced mostly in California by nonutility generators (Hamrin and Rader 1993). Windfarms in areas around the Altamont Pass, the Tehachapi Mountains, and the San Gorgino Pass have more than 15,000 wind turbines (Pace 1991). The U.S. Department of Energy's (DOE's) Energy Information Administration (EIA) projects that the contribution of wind power will rise to 3600 MW(e) in 2000 and 6300 MW(e) in 2010, all of which would be generated by nonutilities (DOE/EIA-0561).

A recent survey of utilities conducted by UDI/McGraw-Hill indicated that no utilities have announced plans to construct 25 MW(e) or larger wind power plants in the foreseeable future, although some utilities may have unpublished plans (Bergesen 1994). Wind technology can be advanced with many small improvements, as well as larger ones such as development of lighter, stronger blade materials; improved gearing to capture a greater portion of useful wind velocities; improved understanding of wind patterns and siting configurations for wind turbines at a site; and improved electrical storage capabilities (SERI/TP-260-3674).

Wind energy is expected to require the use of approximately 61,000 ha (150,000 acres) or 610 square km (about 235 square miles) of land to generate 1000 MW(e) of power (see Table 8.1 for construction impacts and references). This large land requirement, even in dispersed sites, would eliminate any possibility of co-locating a wind energy facility with a retired nuclear plant, thereby pointing to the need for greenfield siting (siting on undeveloped land). The relatively low capacity factor of wind power means that it would operate less frequently at full power than nuclear, but the impacts associated with land use would still occur. The earth-moving that might be required to clear such a large amount of land would destroy much of the natural environment in affected areas (e.g., coastal, mountainous, or plains), where wind velocities are highest. Erosion and sedimentation, while controllable, would still occur and would adversely affect land and water resources. The visual impact of such extended land clearing would be quite

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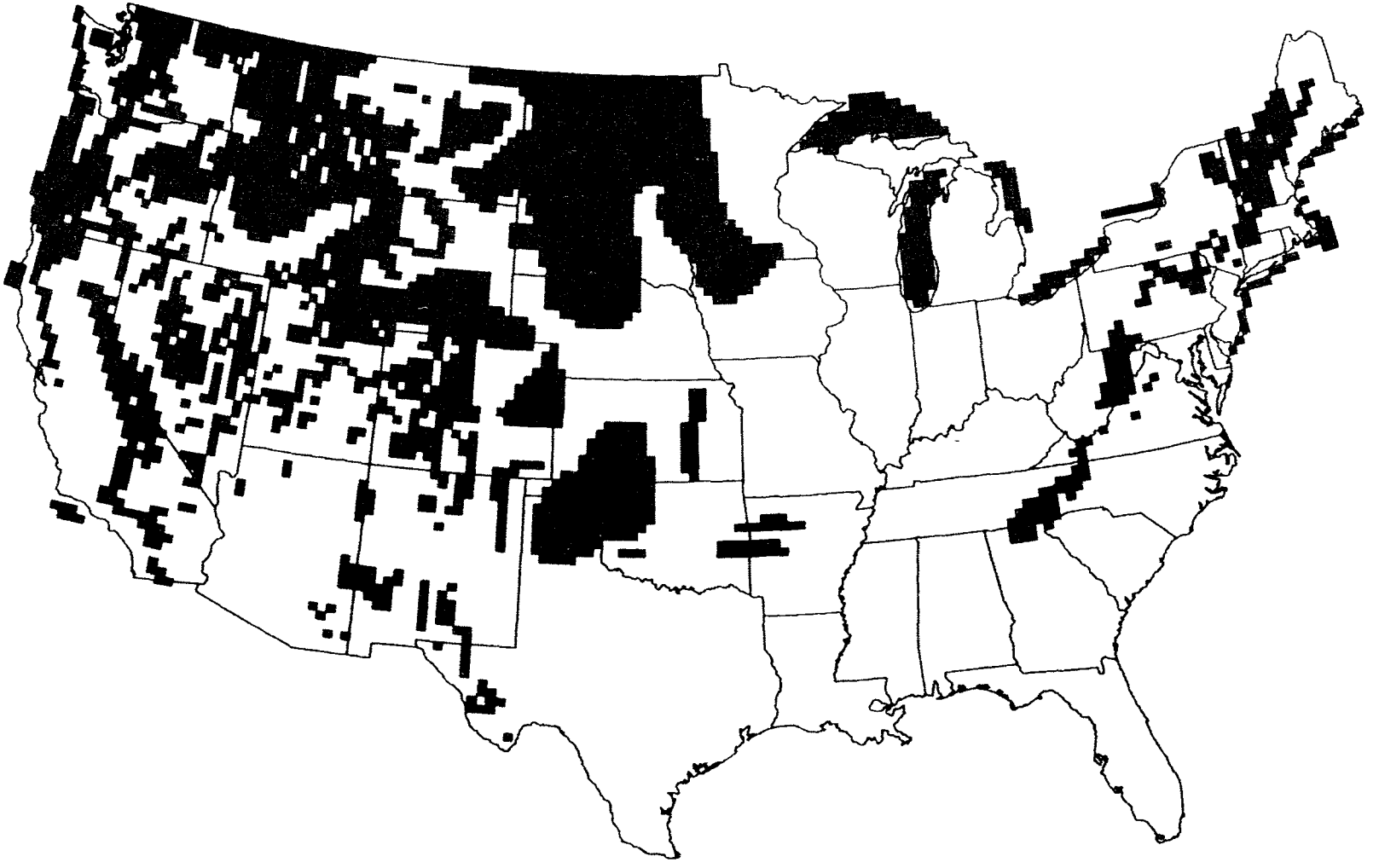

Figure 8.1 U.S. wind energy resources (contiguous states, winds 13 miles per hour or greater).
Source: Adapted from DOE/EIA-0561.

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noticeable and would be a negative aesthetic consequence. Short-term air quality impacts from fugitive dust and equipment exhaust would occur with such extensive activities, and considerable vegetation debris could require disposal. Disturbance of such a large amount of land likely would reveal cultural resources that would require protection. Each of these site impacts would be magnified because of the new transmission lines that are almost always required for greenfield sites. Agricultural land could also be committed to the siting of wind energy facilities in some areas. Adverse impacts could still occur where land is taken out of production, but the acreage lost would likely be less than with natural environments.

The projected impacts of operating wind energy facilities are less than those expected from construction (see Table 8.2 for operational impacts and references). The same amount of land would still be committed to wind generation, but the machines would occupy less than 10 percent of it, freeing up most of the remainder for agricultural or some other compatible use. The aesthetic impact of several thousand wind turbines over a large area likely would strike many observers as obtrusive. The noise from such equipment likely would reinforce these negative opinions. Birds are likely to collide with the turbines, and wind energy developers should consider migration areas and nesting locations when sites for wind energy facilities are selected. In terms of positive environmental impacts, wind power plants would have little effect on water and air quality and would generate very little waste. Human health, except for a potential small number of occupational injuries, would not be affected by operations.

8.3.2 Photovoltaic Cells

PV cells, solid-state devices composed of a thin layer of semi-conductor material (usually single-crystal silicon), convert sunlight directly into electricity. Groups of cells that are mounted on a rigid plate and interconnected to form PV modules have a peak generating capacity of 50 W each (DOE/EH-0077). Usually, groups of modules are permanently attached to a frame and interconnected to form PV arrays or power systems. Power production is proportional to the amount of solar radiation received in a specific geographic area.

The most promising geographic area for the expansion of PV systems is the West; the Midwest and South have some potential (Figure 8.2).

PV power is produced intermittently because solar cells generate electricity only when sunlight is available. The National Association of Regulatory Utility Commissioners indicates an estimated capacity factor of 25 percent (Hamrin and Rader 1993). The largest utility PV system in the United States was built in 1984 on Carrisa Plain in central California by ARCO Solar at a site owned by Pacific Gas and Electric Company (Firor et al. 1993). Until it was dismantled in 1990, it generated 6.5 MW(e) of peak power. Thirty utilities were experimenting with small, grid-connected PV systems as of 1991 (Firor et al. 1993). Use of PV cells for baseload capacity requires very large energy storage devices, such as pumped hydro facilities, batteries, or compressed air chambers. Currently available energy storage devices are too expensive to store sufficient electricity to meet the baseload generating requirements. Thus, while the resource is plentiful, the reserves that

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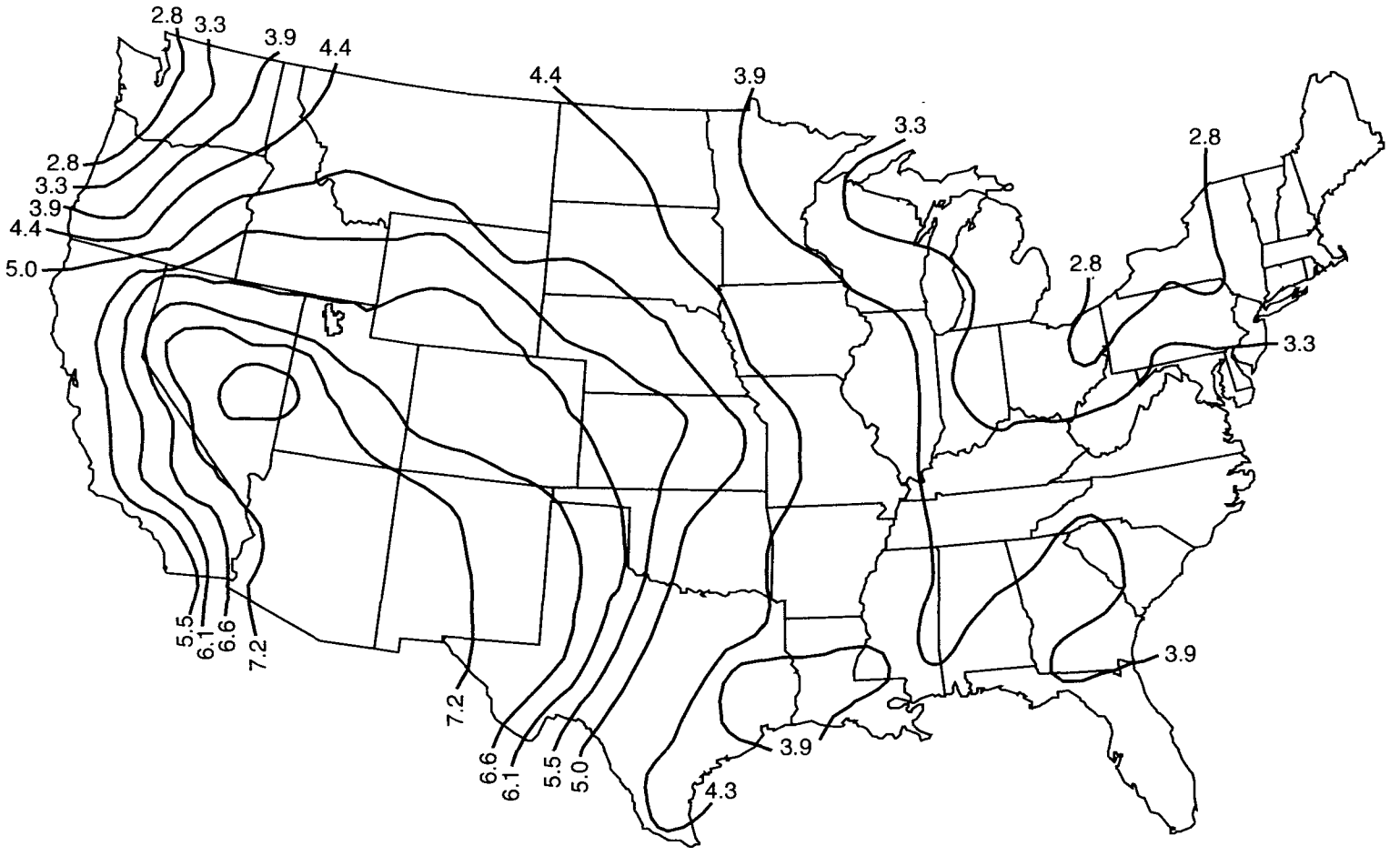

Figure 8.2 Solar resource availability: annual average daily direct normal solar radiation.
Source: Adapted from DOE/EIA-0561.

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currently can be tapped economically for generating electricity in plants of appreciable size are limited.

The high cost of PV systems has been the primary impediment to their more extensive use. These high costs reflect the technical barriers that PV technology must overcome to be competitive. Improvements such as more effective concentrators, use of more easily produced thin-film PV cells rather than silicon cells, and lower module costs could play some part in reducing PV costs. Energy storage technology must become considerably less expensive to enable intermittent technologies like PV to provide reliable electricity. EIA projects that almost no additional PV generating capacity will be added to the electricity grid by 2010, its longest-term forecast (DOE/EIA-0561).

Construction impacts to several resources would be substantial from building a 1000-MW(e) PV facility either at a single site or at numerous smaller dispersed sites. The large land requirement would rule out co-locating a PV facility with an existing nuclear plant, which requires far less land. In addition to these new land requirements, additional land would be required for new transmission lines. No PV plant of this size currently exists, and impacts must be inferred from smaller PV facilities. It is estimated that at least 14,000 ha (35,000 acres) or about 130 km^2 (50 square miles), either at a single site or at multiple sites, would be needed in areas optimal for PV technology to be able to generate as much as 1000 MW(e) of power.Clearing and grading 14,000 ha (35,000 acres) would largely destroy the previous natural or agricultural environment for the life of the facility, with resulting potential impacts to any threatened and endangered species and to aesthetic qualities of the area. Such construction likely would create erosion and resulting stream sedimentation problems. Considerable vegetation debris probably would need to be disposed of as well, which could create short-term air quality problems if it were disposed of through open-air burning. In an area this large, construction impacts to cultural resources would be likely to occur. No work force projections are available for constructing a large PV facility. If prefabricated components and a modular construction approach were used, the work force would probably be smaller than for nonrenewable central generating stations. Such a work force would result in fewer socioeconomic impacts in the form of jobs and local purchases, but the severe impacts of large work forces affecting small communities probably could be avoided.

Adverse operating impacts of PV facilities are associated with the large land requirements. All of the 14,000 ha (35,000 acres) would be lost to other uses for the life of the plant because the land would be covered with PV arrays. Impacts associated with loss of wildlife habitat or agricultural lands would occur, and erosion could develop without proper controls. Water quality could be adversely affected from runoff from PV arrays and drainage unless site engineering included mitigative measures. Substantial visual impacts created from land clearing would be continued in a different form by the extensive PV arrays covering the landscape. The socioeconomic benefits flowing to host communities would be considerably less with PVs than from baseload nonrenewable generating technologies because work forces and plant expenditures would be much less. Tax revenues could be fairly substantial, however, if PV capital costs were

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comparable to nuclear and fossil plant costs and resulted in correspondingly high assessed values. Other impacts, including those to air quality, solid wastes, and human health, either would not occur or would be small.

8.3.3 Solar Thermal Power

Solar thermal conversion systems use reflective materials to concentrate sunlight to heat a fluid that runs a turbine. Both central-receiver and distributed-receiver systems have been used. The parabolic trough, an example of a distributed receiver system, is used in the only large-scale [354-MW(e)] commercial solar thermal power program in the United States, the Luz International facilities located at several sites in the Mojave Desert in California. The Luz facilities, which consist of nine thermal plants [one 13.8-MW(e) unit, six 30-MW(e) units, and two 80-MW(e) units], use natural gas as a backup fuel for generating steam on cloudy days and at night. The company filed for bankruptcy in 1991 because of lower fossil fuel prices and reduced incentives for renewable technologies (DeLaguil et al. 1993). DOE and a consortium of 12 other organizations are retrofitting Solar One, a 10-MW(e) central receiver pilot plant near Barstow, California. It is to come on-line in 1995, renamed Solar Two, and will use a molten-salt heat transfer medium instead of the original oil system to collect and store heat energy. Developers hope that commercial versions of this new Solar Two technology can operate at capacity factors of 60 percent and thus provide dispatchable rather than intermittent power. Based upon solar energy resources (Figure 8.2), the most promising region of the country for this technology is the West.

Solar thermal systems have constraints similar to those of PV systems in that capital costs are higher than for nonrenewable resources, and solar thermal systems lack baseload capability unless combined with natural gas backup. The use of purely solar thermal systems for baseload capacity requires very large amounts of energy storage, such as pumped hydro facilities, compressed air chambers, or batteries. Capacity factors are estimated to be between 25 and 40 percent for future solar thermal plants (Hamrin and Rader 1993). Except for a few older units, most nuclear and baseload coal units generate between 200 and 1000 MW(e) of baseload power and have reached average capacity factors of 65 percent or better in recent years (OTA 1993a).

The construction impacts of building a solar thermal central generating station would stem from the amount of land required to generate 1000 MW(e) of electricity. About 6000 ha (14,000 acres) or 57 km^2 (22 square miles) of land would be cleared either at one site or at multiple locations, with the resulting destruction of whatever wildlife habitat or agricultural values the land provided. A greenfield site or sites, along with new transmission lines, probably would be required because few existing facilities would have sufficient land for such an endeavor. The visual impact of such clearing, even in desert landscapes where solar thermal technology is most competitive, would be regarded by many observers as an obvious negative aesthetic impact. Potential impacts to cultural resources could be considerable because of the large amount of land affected, and care would need to be taken to identify such resources before construction. Some erosion and sedimentation would likely occur during land clearance. Considerable short-term impacts to air quality would

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occur from dust and vehicle exhaust, and vegetation and other debris would require disposal, perhaps through on-site burning. As with PV technology, the size of the construction work force that would be needed is unknown, but it could be reduced through the use of prefabricated components and a modular construction approach. Adverse socioeconomic impacts could be reduced in this fashion.

The operating impacts of a large solar thermal facility also would revolve around land resources dedicated to the plant. No other uses would be compatible since the solar thermal collectors would take up most of the space. Construction-initiated adverse aesthetic impacts and habitat losses and any accompanying risks to threatened and endangered species would continue. There should be few operating impacts to air quality, human health, solid waste, and cultural resources. Water quality should not be affected unless water were used as a cooling agent in an arid environment where it is in short supply or water runoff from the collectors were uncontrolled and sedimentation damaged water bodies. Socioeconomic benefits should be small compared with those going to host communities of large nonrenewable generating stations. Work forces and local purchases would be small. However, the likely high cost—and high assessed value—of solar thermal facilities could lead to substantial property tax revenues.

8.3.4 Hydropower

Currently, the largest electricity contribution from renewable resources is from hydropower. In 1990, conventional hydroelectric plants generated 28 billion kWh of electricity or 83 percent of electricity generated by renewable technologies and about 9.5 percent of electricity generated by all technologies. Hydropower makes up 10 percent of this country's generating capacity. This percentage is expected to decline because new hydroelectric facilities have become difficult to site as a result of public concern over flooding, destruction of natural habitat, and destruction of natural river courses. Hydropower has an average capacity factor of 46 percent, placing it in the middle of the range for renewable technologies (DOE/EIA-0561). Of all renewable and nonrenewable energy resources, hydropower has the fewest resources at 986 quads per year, of which 157 quads are accessible at some cost and 58 quads, or about 6 percent, constitute reserves that are recoverable at current costs (DOE/EIA-0561). Figure 8.3 shows both developed and undeveloped hydropower generating capacity as of January 1992, according to the Federal Energy Regulatory Commission (DOE/EIA-0561).

Impediments to the development of hydropower capacity include environmental concerns and licensing requirements. New dam safety criteria also have affected development. Although it is unlikely that many hydroelectric dams will be constructed in the future, some measures can be taken to increase electrical generation. Older turbines and generators can be upgraded and refurbished. New equipment—such as variable-speed, constant-frequency generators—is being developed which would allow turbines to operate at higher efficiencies (SERI/TP-260-3674).

Although the amount varies, large-scale hydroelectric plants of 1000 MW(e) or greater require an average of almost 400,000 ha (1 million acres). Additional land would be required for transmission, as

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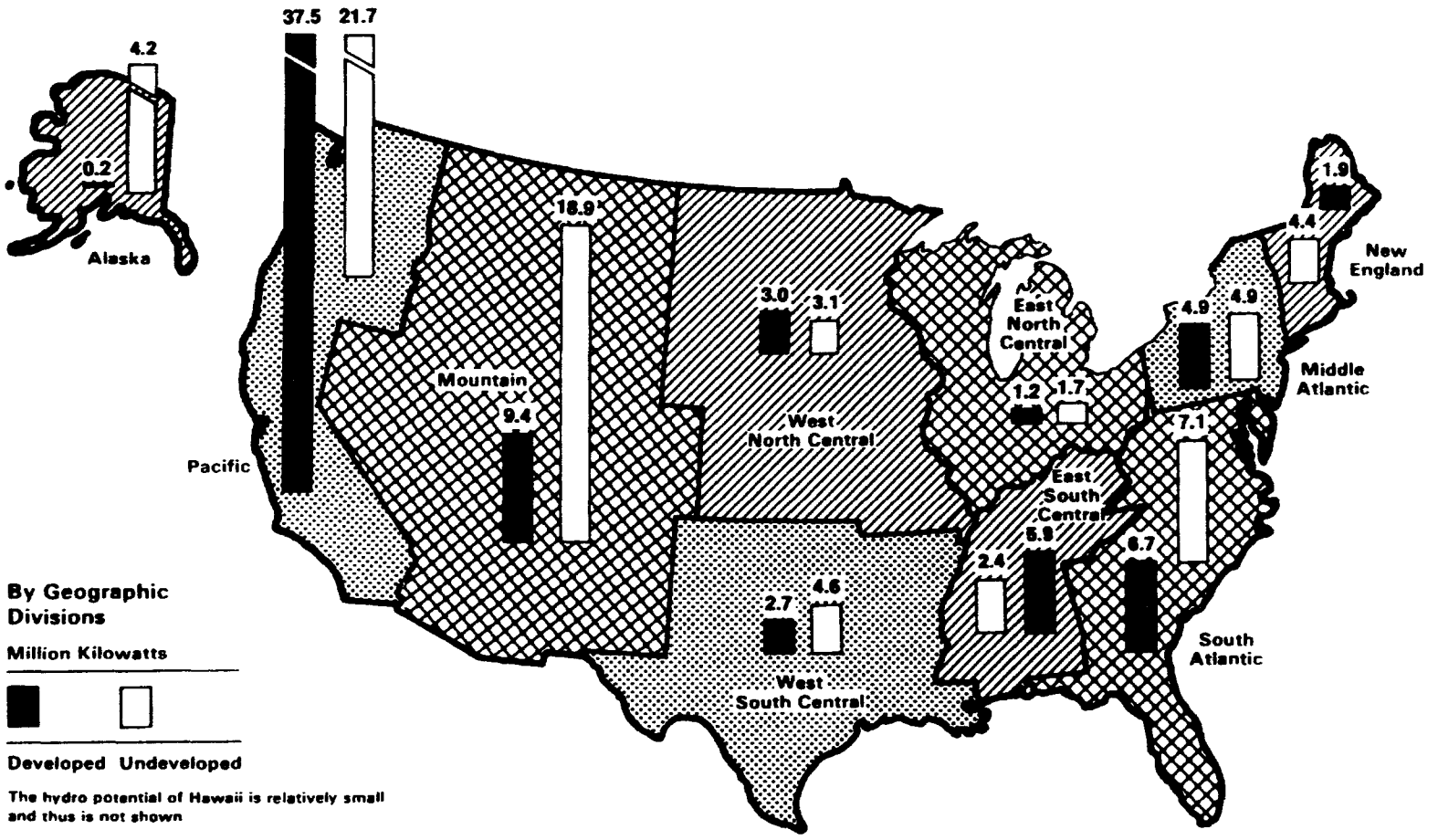

Figure 8.3 U.S. conventional hydroelectric generating capacity, developed and undeveloped (gigawatts). *Source:* Adapted from DOE/EIA-0561.

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