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Environmental Effects of Extending Fuel Burnup Above 60 GWd/MTU

Pacific Northwest National Laboratory

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Environmental Effects of Extending Fuel Burnup Above 60 Gwd/MTU

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Abstract

In 1988, the U. S. Nuclear Regulatory Commission published an environmental assessment (EA) of the effects of increasing nuclear reactor peak-rod fuel irradiation (burnup) up to 60 gigawatt days per metric ton of uranium (GWd/MTU). The EA was based, in large part, on the evaluation of environmental impacts of extended fuel burnup in NUREG/CR-5009. This report updates the information in NUREG/CR-5009 using current fuel designs, fuel performance data, and dose computational methods. It contains a best-estimate assessment of the environmental and economic impacts of extending peak-rod burnup above 60 GWd/MTU.

Inventories were calculated for burnup up to 75 GWd/MTU, and gap-release fractions were calculated up to 62 GWd/MTU. Evaluation of gap-release fractions is limited to 62 GWd/MTU by the methods for estimating release fractions. Gap-release fraction estimates for burnup up to 62 GWd/MTU remain below the release fractions assumed in current guidance associated with evaluation of the environmental consequences of potential accidents, although gap-release fractions for current fuel designs are larger than estimated in NUREG/CR-5009. The increase in gap-release fraction may lead to an increase in the potential environmental impacts of normal operation and accidents involving loss of reactor coolant, if the gap activity is released to the environment. However, even though burnup has been increasing, coolant activity has been decreasing as burnup has increased as a result of better control in fuel-rod fabrication.

There will be a reduction in the environmental effects of the front end of the fuel cycle with increased burnup because increases in burnup to 75 GWd/MTU can be achieved without further increases in enrichment.

The environmental consequences of normal operation are expected to remain small as peak-rod burnup increases because the regulatory limits on releases are independent of burnup. The requirements of 10 CFR 50.36a and Appendix I to 10 CFR Part 50 ensure that releases of radioactive materials to unrestricted areas are kept "as low as reasonably achievable." The volume of low-level waste should continue to decrease as the time between refueling outages increases to achieve higher burnups. The reduced fuel throughput associated with increased burnup will also reduce onsite spent fuel storage facility demands.

The potential environmental consequences of postulated accidents are not expected to increase significantly with increased burnup. The changes in potential consequences from postulated loss-of-coolant accidents, pressurized-water reactor (PWR) steam generator tube rupture accidents, boiling-water reactor (BWR) main steam line break accidents, and fuel handling accidents were all evaluated and found to be small. The potential doses from each of these accidents remain well below regulatory limits.

Potential environmental effects of incident-free transportation of spent fuel and the accident risks associated with spent-fuel transportation do not change significantly with increasing burnup up to

75 GWd/MTU, provided that fuel is cooled for at least 5 years before shipment. For all reactors, the estimated environmental impacts of transportation are bounded by the impacts in 10 CFR 51.52, Table S-4.

For those aspects of this assessment not significantly affected by the gap-release fraction, the findings indicate that there are no significant adverse environmental impacts associated with extending peak-rod burnup to 75 GWd/MTU. For those aspects affected by the gap-release fraction, the findings in the report indicate that there are no significant adverse environmental impacts associated with extending peak-rod burnup to 62 GWd/MTU.

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Abbreviations/Acronyms

ALARA	as low as reasonably achievable
ANS	American National Standard
B&W	Babcock and Wilcox
BWR	boiling-water reactor
Bq	Becquerel
CE	Combustion Engineering
CEDE	committed effective dose equivalent
Ci	Curie
DOE/EIA	U.S. Department of Energy, Energy Information Administration
EA	environmental assessment
FGR	fission gas release
g	grams
GE	General Electric
GWd	gigawatt days
IAEA	International Atomic Energy Agency
LOCA	loss-of-coolant accident
mSv	milliSievert
MTU	metric ton (tonne) of uranium
MW	megawatts
NRC	U.S. Nuclear Regulatory Commission
OECD	Nuclear Energy Agency Organization for Economic Co-operation and Development
OMB	U.S Office of Management and Budget
ORNL	Oak Ridge National Laboratory
PNNL	Pacific Northwest National Laboratory
PWR	pressurized-water reactor
RG	Regulatory Guide
Ryr	reactor year
Sv	Sievert
WREBUS	Water Reactor Extended Burn-up Study

1 Introduction

In 1988, the U.S. Nuclear Regulatory Commission (NRC) issued an environmental assessment (EA) (53 FR 30355, August 11, 1988) that concluded "... the environmental impacts of extended irradiation up to 60 GWd/MTU[a] [gigawatt days per metric ton of uranium] and increased enrichment up to 5 weight percent are bounded by the impacts reported in Table S-4 of 10 CFR Part 51." This conclusion was based, in part, on NUREG/CR-5009 (Baker, et al. 1988), which assessed the environmental impacts of the use of extended burnup fuel. There have been changes in fuel design, reactor operation, and methods of assessing radionuclide inventories and release fractions since publication of NUREG/CR-5009. In addition, the approved average burnup for peak rods now ranges from 50 to 62 GWd/MTU (NRC 1999a). Consequently, the NRC staff requested assistance from the Pacific Northwest National Laboratory (PNNL) in assessing the environmental impacts of increasing peak-rod burnup beyond 60 GWd/MTU, by updating the information in NUREG/CR-5009 using current fuel design and test information and dose methodologies.

This report presents results of a best-estimate assessment of the environmental impacts of increasing peak-rod burnup above 60 GWd/MTU. It describes the changes in the radionuclide inventories in fuel and gap-release fractions as burnup increases. Gap-release fraction is the fraction of the radionuclide inventory that is found in the fuel rod between the fuel pellets and the cladding. The report compares the extended burnup fuel inventories and gap-release fractions with the inventories and release fractions considered in NUREG/CR-5009 and assumed in regulatory guidance and other publications, and it evaluates the environmental and economic effects of increasing fuel burnup. Chapters 2 through 4 concentrate on the fuel. Chapter 5 considers the changes in environmental effects of normal operations associated with extended burnup fuel, and Chapter 6 considers the changes in potential environmental effects of postulated reactor accidents. Transportation of spent fuel is considered in Chapter 7, and Chapter 8 considers the economic aspects of increasing fuel burnup. The final chapter, Chapter 9, summarizes the significant findings of Chapters 2 through 8 and concludes that there are no significant adverse environmental or economic impacts of increasing fuel burnup to 62 GWd/MTU. For those aspects of this assessment in which the environmental impacts are not significantly affected by fission gas releases, the findings of Chapters 2 through 8 indicate that there are no significant adverse environmental impacts associated with extending peak-rod burnup to 75 GWd/MTU.

The terms "peak rod" and "core average" are used frequently throughout this report. Peak rod refers to the entire rod. Thus, peak-rod burnup is the burnup of the fuel rod having the highest rod-average burnup in the core. Peak-rod burnup is determined by the power history of the rod during its life in the core. Core-average burnup refers to an average burnup for all rods in the core. Typically, core-average burnup varies from about one half of the peak-rod burnup at the beginning of a fuel cycle to about two thirds of the peak-rod burnup at the end of a fuel cycle.

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[a]GWd (gigawatt day) is a measure of the energy extracted from reactor fuel. 1 GWd = 1000 MWd (megawatt day).

In the evaluation of environmental impacts of increased burnup that follows, numbers are frequently normalized. For example, radionuclide inventories are expressed as becquerels per metric ton of uranium (Bq/MTU),[a] and doses are expressed as sieverts per reactor year (Sv/Ryr).[b] Application of the normalized numbers to a specific reactor requires knowledge of characteristics of the reactor, such as power, mass of uranium in the core, number of fuel assemblies, and number of rods in a fuel assembly. Appendix A contains a compilation of statistics for U.S. reactors. The appendix is divided into two parts, the first part contains statistics for pressurized-water reactors (PWRs), and the second contains statistics for boiling-water reactors (BWRs). The statistics were compiled from Plant Information Books prepared and maintained by the NRC[c] with supplementary information from the 1997 World Nuclear Industry Handbook (NEI 1997).

Where possible, this report provides analysis of the impacts of increasing peak-rod burnup to levels up to 75 GWd/MTU. However, the present analytical methods for assessing fission gas release from fuel have had only limited benchmarking with actual measurement data at burnups greater than 40 GWd/MTU, and have not been benchmarked at burnups greater than 62 GWd/MTU. Therefore, those aspects of the assessment that are significantly affected by fission gas release are evaluated only to a burnup of 62 GWd/MTU. This limitation is discussed in the appropriate sections of this report. As data become available, this limitation will be reevaluated.

This document has been prepared to support environmental assessments of operating plants at which burnup exceeds 60 GWd/MTU. Although conservative assumptions have been made while determining the values in this report, PNNL performed its calculations using assumptions that, in certain cases, were less conservative than those that would have been used while performing a licensing analysis. This is consistent with the procedures for performing environmental assessments for nuclear plants. Users who intend to apply the information presented in this document to other applications should consider the limitations and assumptions used to determine that information and should adjust their analyses accordingly. The limitations and assumptions are discussed in the appropriate sections of this report.

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[a]A Becquerel (Bq) is a measure of radiation; one Becquerel is one disintegration per second. Radiation is also measured in Curies (Ci). 3.7×10^{10} Bq = 1 Ci.
[b]A Sievert is a measure of radiation dose. Doses are also expressed in rem. 1 Sv = 100 rem.
[c]U.S. Nuclear Regulatory Commission at http://www.nrc.gov/AEOD/pib/disclaimer.html (October 25, 2000).

2 Fuel

This chapter describes the calculation of the fuel radionuclide inventories and gap-release fractions used in evaluating the environmental impacts of extending fuel burnup. The first section describes the calculation of core-average and peak-rod inventories. Core-average inventories are used in evaluating the effects of extended burnup on the impacts of normal operation and loss-of-coolant accidents (LOCAs). The peak-rod inventories are used for evaluating the effects of extended burnup on the impacts of fuel-handling accidents, incident-free transportation of spent fuel, and transportation accidents. The next chapter compares fuel inventories and gap-release fractions with previously reported values and values contained in regulatory guides.

2.1 Fuel Inventory and Decay Heat Calculations

Radionuclide inventories were calculated using ORIGEN-ARP (Bowman and Leal 1998). ORIGEN-ARP performs PWR and BWR burnup calculations using libraries (files of radionuclide characteristics such as decay parameters and neutron cross sections) defined for different types of assemblies and different enrichments. ORIGEN-ARP libraries for 15 x 15 PWR and 8 x 8 BWR assemblies provided with the code were used in the radionuclide inventory calculations because the relative increase in inventory with burnup does not change significantly with fuel design. Because the principal purpose of the study is to evaluate the effects of an increase in burnup, the use of existing libraries is reasonable compared with the alternative of creating and verifying special libraries for a 17 x 17 PWR fuel design. Five percent enriched fuel was used in all calculations because enrichment for current high burnup is between 4.5 percent and 5 percent. Cross sections used in the calculations are calculated by interpolation between the cross section in libraries for various fuel exposures.

The initial composition of fuel assemblies (including cladding and fittings) assumed for the ORIGEN-ARP calculations is listed in Table 2.1. The masses of the uranium isotopes are identical to the masses used in Oak Ridge National Laboratory (ORNL) ORIGEN-ARP calculations made for the analysis of the effects of transportation of high burnup spent fuel described in NUREG-1437, Addendum 1 (NRC 1999a). Radionuclide inventories used in evaluating environmental impacts of normal operations, LOCAs, and transportation of spent fuel include activation products and actinides, as well as fission products.

NRC staff have expressed concern that ORIGEN-ARP underestimates the production of actinides, particularly Cm-244, in PWR fuels burned in the presence of burnable poison rod assemblies. The potential environmental impacts of the actinides are small compared to the potential impacts of isotopes or more volatile elements, such as ^{131}I, ^{134}Cs, and ^{137}Cs in postulated loss-of-coolant accidents. They are not expected in the gap and consequently are not on the list of isotopes considered during postulated steam generator tube rupture, main steamline break, and fuel handling accidents. None of the actinides contributes more than one percent of the external dose rate from an iron transportation cask. As a group, the actinides do not contribute significantly to the dose from transportation accidents. In fact, increasing

Table 2.1 Initial Fuel Assembly Composition		
	PWR Assemblies	BWR Assemblies
Element	Mass (g/MTU)	Mass (g/MTU)
U-234	442	442
U-235	50,000	50,000
U-236	230	230
U-238	949,000	949,000
Oxygen	136,000	136,000
Chromium	5,920	3,213
Manganese	330	223
Iron	12,940	9,405
Cobalt	70	28
Nickel	9,870	2,618
Zirconium	221,440	478,737
Niobium	700	1,468
Tin	3,510	7,338
total	1,390,452	1,638,702

the activities of ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am, ^{242}Cm, and ^{244}Cm by more than a factor of 1000 only increased the cumulative dose for a transportation accident during shipment 43 GWd/MTU spent fuel from the northeast to Clark County, NV from 0.0358 to 0.0359 person-mSv/shipment (3.58×10^{-3} to 3.59×10^{-3} person-rem/shipment). Consequently, the potential consequences of underestimation of actinides are negligible and the ORIGEN-ARP inventories are considered acceptable for the present study.

Calculations for PWR and BWR fuel were made with both constant and varying power histories, for burnup ranging from 22 to 75 GWd/MTU, and decay (cooling time) ranging from 0 days to 30 years. Table 2.2 lists the computer runs made for these calculations. The constant-power cases provide core-average fuel radionuclide inventories for use in evaluating potential effects of increased burnup on the environmental impacts from normal operations and LOCAs. The variable power cases provide peak-rod radionuclide inventories for evaluation of the impacts of increased burnup on the potential effects of fuel-handling accidents and transportation of spent fuel. A 72-hour decay time was assumed for the radionuclide inventories used in the fuel-handling accident for PWRs based on the draft of revision 2 to

Table 2.2 ORIGEN-ARP Calculation Cases			
Reactor Type	**Power (MW/MTU)**	**End- of-Cycle Peak-Rod Burnup (GWd/MTU)**	**Decay Times**
PWR	Constant 28.3	22, 24, 25, 42, 43, 46, 48, 50, 60, 62, 65, 70, 75	0 d
PWR	Variable 36.02 - 52.96	22, 24, 25, 42, 43, 46, 48, 50, 60, 62, 65, 70, 75	3 d, 5 yr, 10 yr, 15 yr, 20 yr
BWR	Constant 22.22	22, 24, 25, 42, 43, 46, 48, 50, 60, 62, 65, 70, 75	0 d
BWR	Variable 10.80 - 41.99	22, 24, 25, 42, 43, 46, 48, 50, 60, 62, 65, 70, 75	1 d, 5 yr, 10 yr, 15 yr, 20 yr

standard technical specifications for Combustion Engineering (CE) reactors.[a] Draft revision 2 to standard technical specifications for both Westinghouse and Babcock and Wilcox (B&W) reactors specify 96 hours of decay before moving fuel. A 24-hour decay time was assumed for the BWR fuel-handling accident based on the draft revision 2 to the standard technical specifications for General Electric (GE). Radionuclide inventories for long decay times (5 yr, 10 yr, 15 yr, and 20 yr) are used in evaluating effects and potential mitigation of the effects of increased burnup on impacts from transportation of spent fuel.

Power was varied as a function of time in calculating the inventories for peak rods to provide realistic radionuclide inventories for high burnup fuel. Fuel exposure was assumed to occur in three cycles. Power was highest in the first cycle and decreased in the second and third cycles. The power histories used are given in Appendix B.

ORIGEN-ARP was also used to calculate spent-fuel decay heat as a function of burnup and decay time. For these calculations, PWR fuel was exposed at a constant power of 28.3 MW/MTU until the desired fuel burnup was achieved. Similarly, BWR fuel was exposed at a constant power of 22.22 MW/MTU until the desired fuel burnup was achieved. The results of these calculations, which are summarized in Appendix C, were used to evaluate the effects of increasing burnup on normal operation of the spent-fuel storage pool.

2.2 Gap-Release Fraction Calculations

Peak-rod and core-average radionuclide release fractions (release from fuel pellets to gap) were calculated using the FRAPCON-3 computer code (Lanning et al. 1997a, b; Berna et al. 1997). The

[a]Draft revision 2 to the standard technical specifications for each reactor type is found at http://www.nrc.gov/NRR/sts/sts.htm (October 25, 2000).

FRAPCON-3 code calculates best-estimate release fractions if based on best-estimate values of fuel-rod power histories. Because best estimate release fractions are more appropriate for environmental assessments than bounding values, the release fractions calculated for this study do not include components to account for uncertainties in fuel fabrication and in the prediction of release fractions. Core-average radionuclide release fractions are important for assessing the consequences of steam generator tube rupture accidents and main steam line break accidents that release coolant activity to the environment because coolant activity is determined by many fuel assemblies. Peak-rod release fractions are used for assessing the consequences of fuel-handling accidents because the accident involves only a small number of fuel assemblies.

The power histories used as input for the peak rod for the fuel-handling accident analysis are for typical peak fuel rods from Westinghouse 17 x 17 and General Electric 8 x 8 fuel designs for PWRs and BWRs, respectively. These histories bound the rod powers in the first, second, and third cycle of operation. The 17 x 17 PWR fuel design was selected for these calculations rather than the 15 x 15 design because the 17 x 17 design is currently the most prevalent PWR fuel design. The most prevalent BWR fuel design is the 8 x 8 design. However, 9 x 9 and 10 x 10 fuel designs are replacing the 8 x 8 design in plants that are going to higher burnups.

There can be a significant change in the absolute gap-release fraction depending on fuel design and plant operation. The release fractions given in this study will not be applicable to all designs and operating modes. Two 24-month cycle operation may result in higher release fractions than three 18-month cycles because 24-month cycle operation generally involves higher time-average power levels. However, the relative changes in gap-release fractions with burnup will be similar for the two operating modes. For the same burnup, the gap-release fractions in BWR 9 x 9 and 10 x 10 fuel assemblies will be lower than the gap-release fractions in 8 x 8 assemblies. Again, the relative changes in release fractions will be similar for the three BWR fuel designs.

This analysis is not intended to be a licensing analysis. A licensing analysis would have several conservatisms not included in this analysis; for example, in a traditional licensing analysis uncertainties associated with fuel fabrication, model predictions, and normal (power) operating transients are considered in estimating release fractions. Licensing analyses predict release fractions that are a factor of 1.8 to a factor of 2.5 greater than those predicted in a best-estimate analysis. Fabrication and model predictions generally account for a factor of 1.5 to a factor of 2.0 difference. The remainder of the difference is associated with normal operating transients. Appendix D includes examples of release fractions calculated taking these uncertainties into account. They are more typical of values used in licensing analyses.

The FRAPCON-3 code has two fission gas release (FGR) models: the American National Standard (ANS) 5.4 (ANS 1982) and Massih models (Forsberg and Massih 1985; Lanning 1997a). The ANS 5.4 model, with the GAPCON-THERMAL-2, Revision-2 fuel performance code (Cunningham and Beyer 1984), was used for the analysis of the core-average rod in NUREG/CR-5009 (Baker et al. 1988). The ANS 5.4 model calculates the release fractions for the radioactive isotopes of the noble gases (xenon and krypton), iodine, and cesiums. However, the model has not been verified against a large volume of high burnup data, and it over-predicts noble gas, iodine and cesium releases when used with the FRAPCON-3 code.

The Massih model calculates peak-rod and core-average FGR fractions to the gap for the stable noble gases. It is the primary FGR model used for verifying the FRAPCON-3 code against high burnup data. Therefore, the Massih model in the FRAPCON-3 code has been used to calculate the peak-rod and core-average FGR fractions to the gap for the stable noble gases. Given the release fractions calculated by FRAPCON-3/Massih for the stable noble gases, release fractions for short half-life radioactive isotopes of xenon, krypton, iodine and cesium are estimated using ratios of the short-lived isotope release fractions to the stable noble gas release fractions calculated by the ANS 5.4 model. Gap-release fraction estimates are limited to a burnup of 62 GWd/MTU because the Massih and ANS 5.4 release models have not been verified with data for burnups greater than 62 GWd/MTU. The ANS 5.4 model has not been verified with data for burnups greater than 40 GWd/MTU, and the data available for verification of release models for burnups greater than 40 GWd/MTU are limited.

The ANS 5.4 model over-predicts the release fractions for iodine and cesium isotopes and adds a conservative bias to the release fraction estimates for these isotopes. The diffusion coefficients for iodine and cesium in the ANS 5.4 model are assumed to be factors of 7 and 2 greater than the diffusion coefficient for the noble gases xenon and krypton.. However, more recent release data for iodine and cesium (Hastings et al. 1985; Lewis et al. 1990; Turnbull et al. 2000) demonstrate that the diffusion coefficients for these isotopes are the same as the diffusion coefficients for the noble gases. Therefore, the gap-release fractions calculated using the ANS-5.4 model are conservative by about a factor of 2 for iodine and by about a factor of 1.4 for cesium.

The gap-release fractions calculated are shown in Table 2.3 and 2.4 for those isotopes that are expected to contribute significantly to doses due to releases of gap activity. The noble gases, iodines, and cesiums are the only elements that are sufficiently volatile to have significant gap-release fractions. Isotopes of these elements that are not shown in the tables do not contribute significantly to dose. For example, isotopes of iodine other than ^{131}I are rapidly depleted by decay. The gap-release fraction for ^{131}I provides a conservative estimate for these isotopes.

For PWRs, the core-average gap-release fraction of ^{131}I increases by about 7 percent, from 0.018 to 0.019, as a result of increasing the peak-rod burnup from 60 to 62 GWd/MTU. The gap-release fractions for the peak rod show a slightly greater increase of about 8 percent for the same increase in peak-rod burnup. Given the conservative assumptions in the ANS 5.4 model discussed above, it is likely that the peak-rod gap-release fractions for ^{131}I are about a factor of 2 lower than the values shown in Table 2.3. Similarly, it is likely that the peak-rod gap-release fractions for the isotopes of cesium are about a factor of 1.4 lower than the values in the table.

For BWRs, the increase in the ^{131}I core-average gap-release fraction is about 7.9 percent. In contrast to PWRs, the BWR peak-rod gap-release fractions decrease as burnup increases, although for an increase in burnup from 60 to 62 GWd/MTU the gap-release fractions are nearly constant. Almost all of the high burnup BWR fuel in use is either 9 x 9 or 10 x 10 fuel. The gap-release fractions for these fuel designs are about one half of the gap-release fractions for 8 x 8 fuel. For example, the best estimate peak-rod gap-release fraction for ^{85}Kr would be 0.053 or less for 43 GWd/MTU fuel compared to the 0.10 shown in Table 2.4. The gap-release fractions for ^{85}Kr for both BWR 8 x 8 and BWR 9 x 9 fuel assemblies come from FRAPCON-3 calculations. The reduction in the peak-rod, gap-release fraction for 9x9 fuel assemblies and the over-estimation of the iodine to krypton and cesium to krypton ratios are not related. Therefore, the likely peak-rod gap-release fractions for ^{131}I for extended burnup fuel will be about a factor of 4 lower than shown in Table 2.4, and the likely peak-rod, gap-release fractions for isotopes of cesium will be about a factor of 2.8 lower than shown in the table.

Section 3.2 includes comparisons of the gap-release fractions shown in Tables 2.3 and 2.4 with previously estimated gap-release fractions and gap-release fractions assumed in regulatory guidance.

Table 2.3 Core-Average and Peak-Rod PWR Gap-Release Fractions as a Function of Peak-Rod Burnup in GWd/MTU

	Core-Average Release Fraction		Peak-Rod Release Fraction			
	60	62	43	50	60	62
Kr-85	2.7E-02	2.9E-02	4.3E-02	6.7E-02	7.3E-02	7.9E-02
Kr-87	5.5E-04	5.9E-04	8.8E-04	1.4E-03	3.7E-03	4.0E-03
Kr-88	8.2E-04	8.8E-04	1.3E-03	2.1E-03	5.2E-03	5.6E-03
I-131	1.8E-02	1.9E-02	2.8E-02	4.3E-02	6.3E-02	6.8E-02
Xe-133	5.5E-03	5.9E-03	8.69E-03	1.4E-02	2.6E-02	2.8E-02
Xe-135	1.5E-03	1.6E-03	2.3E-03	3.7E-03	1.0E-02	1.1E-02
Cs-134	3.8E-02	4.1E-02	6.1E-02	9.5E-02	1.0E-01	1.1E-01
Cs-137	3.8E-02	4.1E-02	6.1E-02	9.5E-02	1.0E-01	1.1E-01

Table 2.4 Core-Average and Peak-Rod BWR Gap-Release Fractions as a Function of Peak-Rod Burnup in GWd/MTU						
	Core-Average Release Fraction		**Peak-Rod Release Fraction**			
	60	**62**	**43**	**50**	**60**	**62**
Kr-85	2.7E-02	2.9E-02	1.0E-01	8.9E-02	7.9E-02	7.90E-02
Kr-87	5.4E-04	5.8E-04	5.1E-03	4.4E-03	4.0E-03	4.0E-03
Kr-88	8.1E-04	8.7E-04	7.2E-03	6.3E-03	5.6E-03	5.6E-03
I-131	1.4E-02	1.5E-02	8.7E-02	7.6E-02	6.8E-02	6.8E-02
Xe-133	5.4E-03	5.8E-03	3.6E-02	3.2E-02	2.8E-02	2.8E-02
Xe-135	1.5E-03	1.6E-03	1.4E-02	1.3E-02	1.1E-02	1.1E-02
Cs-134	3.8E-02	4.1E-02	1.4E-01	1.3E-01	1.1E-01	1.1E-01
Cs-137	3.8E-02	4.1E-02	1.4E-01	1.3E-01	1.1E-01	1.1E-01

3 Fuel Comparisons

This chapter describes the radionuclide inventories and the gap-release fractions and places them in the context of previously reported values and values included in regulatory guidance.

3.1 Inventory

The radionuclide inventories calculated by ORIGEN-ARP included activities for more than 300 radionuclides. However, the number of radionuclides making a significant contribution to doses is much smaller than 300. Alpert et al. (1986) list the 60 radionuclides making the largest contribution to doses from reactor accidents. These 60 radionuclides were used in estimating the effect of increasing fuel burnup on doses from postulated LOCAs. Appendix E contains core-average inventories for PWRs and BWRs for peak-rod burnup ranging from 22 to 75 GWd/MTU.

"Constant power" inventories were used in the evaluation of the LOCA. At the time of the postulated accident, the reactor was assumed to be near the end of fuel cycle to maximize inventories of long-lived radionuclides. The core-average inventory was estimated by averaging inventories of fuel at the end of each cycle. For a discharge burnup of 42 GWd/MTU burnup, fuel life was assumed to be 2 cycles, and for the remainder of the burnups, fuel life was assumed to be 3 cycles. Table 3.1 shows the discharge burnup, the core-average burnup corresponding to the discharge burnup, and the burnup assumed in each cycle. The core-average inventories were converted to becquerels per megawatt by multiplying by metric tons of uranium per megawatt. The average mass of uranium in the core for PWRs in the United States is about 0.0283 metric tons per megawatt. For BWRs licensed after 1972, the average mass is about 0.0419 metric tons per megawatt.

Table 3.2 lists those radionuclides considered in NUREG/CR-5009 and gives ratios of the core-average activities from the ORIGEN-ARP calculations to the activities reported in NUREG/CR-5009. With few

Table 3.1 Fuel Burnup Assumed in Determining Core-Average Inventories

Discharge Burnup (GWd/MTU)	Core-Average Burnup (GWd/MTU)	First Cycle Burnup (GWd/MTU)	Second Cycle Burnup (GWd/MTU)	Third Cycle Burnup (GWd/MTU)
42	33	24	42	--
60	41.3	22	43	60
62	42.3	22	43	62
65	45	24	46	65
70	47.3	24	48	70
75	50	25	50	75

Table 3.2 Comparison of Ratios of Core-Average Activities Calculated by ORIGEN-ARP to Activities Reported in NUREG/CR-5009 for Peak-Rod Burnups of 33 and 60 GWd/MTU				
	PWR		**BWR**	
	33GWd	**60GWd**	**33GWd**	**60GWd**
H-3	0.86	0.86	0.84	0.83
Kr-87	0.86	0.77	0.68	0.60
Kr-88	0.86	0.74	0.68	0.59
Sr-89	0.87	0.74	0.68	0.58
Sr-90	1.11	1.01	1.10	0.99
Zr-95	0.80	0.77	0.63	0.60
Nb-95	0.81	0.77	0.63	0.61
Ru-103	0.70	0.76	0.55	0.60
Ru-106	0.69	0.85	0.59	0.70
Te-132	0.76	0.77	0.60	0.61
I-131	0.74	0.76	0.58	0.59
I-132	0.73	0.79	0.57	0.62
I-133	0.78	0.77	0.61	0.60
I-135	0.78	0.77	0.62	0.61
Xe-133	0.78	0.77	0.60	0.59
Xe-135	1.46	1.09	1.31	0.97
Cs-134	0.80	0.81	0.76	0.74
Cs-137	1.07	1.05	1.06	1.03
Ba-140	0.82	0.78	0.64	0.61
La-140	0.79	0.78	0.64	0.62
Ce-141	0.76	0.76	0.59	0.60
Ce-144	0.81	0.74	0.70	0.57
Np-239	0.64	0.80	0.49	0.62
Pu-238	1.20	1.21	1.28	1.26
Pu-241	1.00	1.14	0.96	1.06
Am-241	1.62	1.96	1.93	2.24
Am-242	0.80	1.09	0.77	1.05
Cm-242	0.90	1.21	0.94	1.20
Cm-244	1.03	1.56	1.12	1.62

exceptions, the activities calculated by ORIGEN-ARP for fission products are lower those reported in NUREG/CR-5009. In contrast, the actinide activities calculated by ORIGEN-ARP are generally larger than those reported in NUREG/CR-5009.

3.2 Gap-Release Fractions

The gap-release fractions calculated for this study are presented in Tables 2.3 and 2.4. This section compares those values with gap-release fractions contained in previous studies. Table 3.3 compares the gap-release fractions calculated for 60 GWd/MTU with gap-release fractions contained in NUREG/CR-5009 (Baker et al. 1988), NUREG-1465 (Soffer et al. 1995), Regulatory Guide (RG) 1.25 (NRC 1972), and RG 1.183 (NRC 2000). Note that RG 1.183 is based, in part, on NUREG/CR-5009 and NUREG-1465, and it uses the release fractions from NUREG-1465 for LOCA.

The gap-release fractions calculated in this study for core average conditions are generally about a factor of two larger than the release fractions given in NUREG/CR-5009. This increase is due to the two different codes and fission-gas release models (the newer FRAPCON-3/Massih code versus the older GAPCON-THERMAL-2, Revision2/ANS 5.4 code) and their different predictions of fission gas release for lower power (temperature) rods that are typical of core-average conditions at higher burnups. For example, the newer FRAPCON-3 code with the Massih fission-gas release model predicts higher fission gas releases for these low power rods than the older GAPCON-THERMAL-2, Revision 2 code with the ANS 5.4 release model.

Table 3.3 Comparison of Ratios of Gap-Release Fractions for 60 GWd/MTU Burnup in Tables 2.3 and 2.4 to Gap-Release Fractions Contained in Other Documents					
	60 GWd/MTU Peak-Rod Burnup				
PWR	**Core Avg.**	**Core Avg.**	**Peak Rod**	**Core Avg.**	**Peak Rod**
Isotope	**NUREG/CR-5009 Table 3.1**	**NUREG-1465 Table 3.13[a]**	**RG 1.25 Position C.1.d**	**RG 1.183 Table2**	**RG 1.183 Table 3**
Kr-85		0.54	0.24	0.54	0.73
Kr-87	1.84	0.01	0.04	0.01	0.07
Kr-88	2.05	0.02	0.05	0.02	0.10
I-131	1.94	0.35	0.63	0.35	0.78
Xe-133	1.82	0.11	0.26	0.11	0.52
Xe-135	1.84	0.03	0.10	0.03	0.21
Cs-134	2.24	0.76		0.76	0.86
Cs-137	1.41	0.76		0.76	0.86

BWR	**Core Avg.**	**Core Avg.**	**Peak Rod**	**Core Avg.**	**Peak Rod**
Isotope	**NUREG/CR-5009 Table 3.1**	**NUREG-1465 Table 3.12[a]**	**RG 1.25 Position C.1.d**	**RG 1.183 Table 1**	**RG 1.183 Table 3**
Kr-85		0.54	0.26	0.54	0.79
Kr-87	1.80	0.01	0.02	0.01	0.08
Kr-88	2.03	0.02	0.02	0.02	0.11
I-131	1.56	0.28	0.41	0.28	0.88
Xe-133	1.80	0.11	0.16	0.11	0.56
Xe-135	1.86	0.03	0.04	0.03	0.23
Cs-134	2.24	0.76		0.76	0.93
Cs-137	1.41	0.76		0.76	0.93

[a]Assuming long-term fuel cooling is not maintained.

Table 3.4 presents comparisons of the gap-release fractions calculated for 62 GWd/MTU burnup with fractions given in other documents. Again, the gap-release fractions calculated in this study are lower than the fractions given in other documents.

Table 3.4 Comparison of Ratios of Gap-Release Fractions for 62 GWd/MTU Burnup in Table 2.3 and 2.4 to Gap-Release Fractions Contained in Other Documents				
	62 GWd/MTU Peak-Rod Burnup			
PWR	**Core Avg.**	**Peak Rod**	**Core Avg.**	**Peak Rod**
Isotope	**NUREG-1465 Table 3.13[a]**	**RG 1.25 Position C.1.d**	**RG 1.183 Table 2**	**RG 1.183 Table 3**
Kr-85	0.58	0.26	0.58	0.79
Kr-87	0.01	0.04	0.01	0.08
Kr-88	0.02	0.06	0.02	0.11
I-131	0.38	0.68	0.38	0.85
Xe-133	0.12	[illegible]	0.12	0.56
Xe-135	0.03	0.11	0.03	0.23
Cs-134	0.82		0.82	0.93
Cs-137	0.82		0.82	0.93

BWR	**Core Avg.**	**Peak Rod**	**Core Avg.**	**Peak Rod**
Isotope	**NUREG-1465 Table 3.12[a]**	**RG 1.25 Position C.1.d**	**RG 1.183 Table 1**	**RG 1.183 Table 3**
Kr-85	0.58	0.26	0.58	0.79
Kr-87	0.01	0.02	0.01	0.08
Kr-88	0.02	0.02	0.02	0.11
I-131	0.30	0.41	0.30	0.85
Xe-133	0.12	0.16	0.12	0.56
Xe-135	0.03	0.04	0.03	0.23
Cs-134	0.82		0.82	0.93
Cs-137	0.82		0.82	0.93

[a]Assuming long-term fuel cooling is not maintained.

4 Fuel Cycle Front End

This chapter describes the potential effects of extending fuel burnup from 60 GWd/MTU to 75 GWd/MTU on the environmental impacts of the front end of the fuel cycle. The front end of the fuel cycle includes mining, milling, and enrichment of uranium and the fabrication of fuel assemblies.

The evaluation of environmental impacts of extended burnup fuel in NUREG/CR-5009 was based on 5 percent enriched fuel with a peak-rod burnup of 60 GWd/MTU. Peak-rod burnup can be extended to 75 GWd/MTU without further enrichment. Consequently, increasing fuel burnup will make better use of the uranium resource to the extent that less uranium will be required per gigawatt-day of energy produced. Table 4.1 shows the variation in annual use of 5 percent enriched uranium with increasing fuel burnup for power reactors in the United States. For example, for 60 GWd/MTU burnup, the average use of 5 percent enriched uranium is about 16.4 MTU/yr for US PWRs. However, the use varies from about a minimum of 8.5 MTU/yr to a maximum of 21.6 MTU/yr. About 50 percent of the PWRs use between 14.7 and 19.1 MTU/yr. Figure 4.1 illustrates the decrease in use of 5 percent enriched uranium for an average PWR and an average BWR. Figure 4.2 shows the decrease in total uranium use for U.S. reactors, assuming all reactors continue operation with high burnup fuel.

Table 4.1 Variation in Use (MTU/yr) of 5 Percent Enriched Uranium with Peak-Rod Burnup

		Peak-Rod Burnup (GWd/MTU)						
		43	**50**	**60**	**62**	**65**	**70**	**75**
PWRs	**N = 73**							
	Maximum	29.3	25.5	21.6	20.9	20.0	18.7	17.5
	75%	25.8	22.6	19.1	18.5	17.7	16.6	15.5
	Average	22.3	19.4	16.4	15.9	15.3	14.2	13.3
	Median	23.2	20.2	17.1	16.6	15.9	14.8	13.9
	25%	20.1	17.5	14.7	14.3	13.6	12.7	11.9
	Minimum	11.6	10.1	8.5	8.3	7.9	7.4	6.9
BWRs	**N = 34**							
	Maximum	30.0	26.1	22.0	21.3	20.4	19.0	17.8
	75%	26.5	23.0	19.3	18.7	17.9	16.7	15.6
	Average	23.6	20.5	17.3	16.6	16.0	14.9	14.0
	Median	22.8	19.8	16.7	16.1	15.4	14.4	13.5
	25%	19.2	16.7	14.1	13.6	13.0	12.1	11.4
	Minimum	12.6	10.9	9.2	8.9	8.5	7.9	7.4

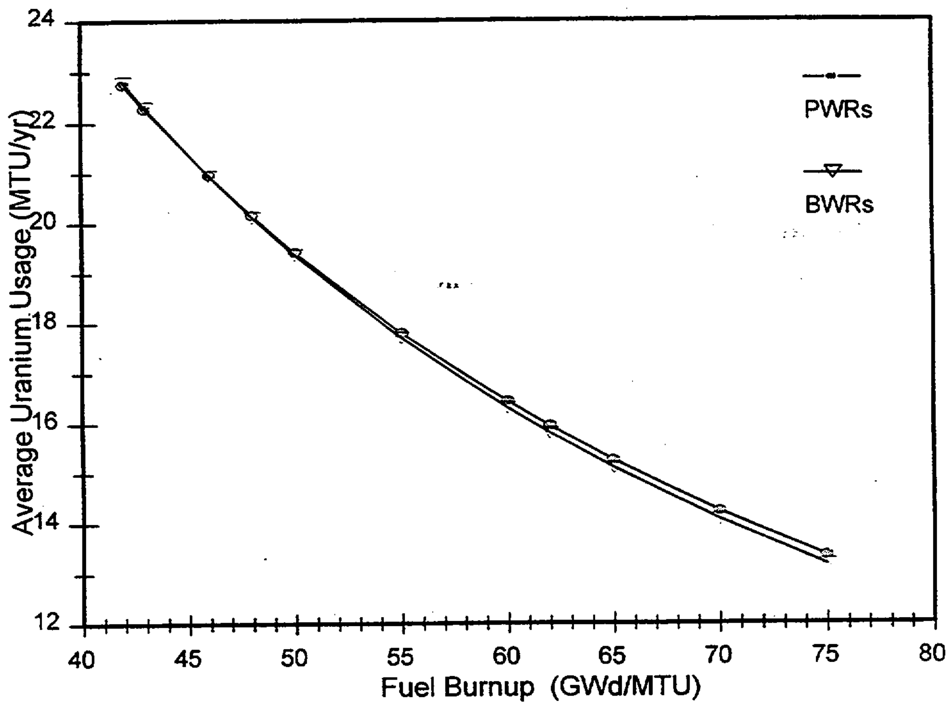

Figure 4.1 Reduction in Uranium Use for Individual Reactors with Increasing Burnup

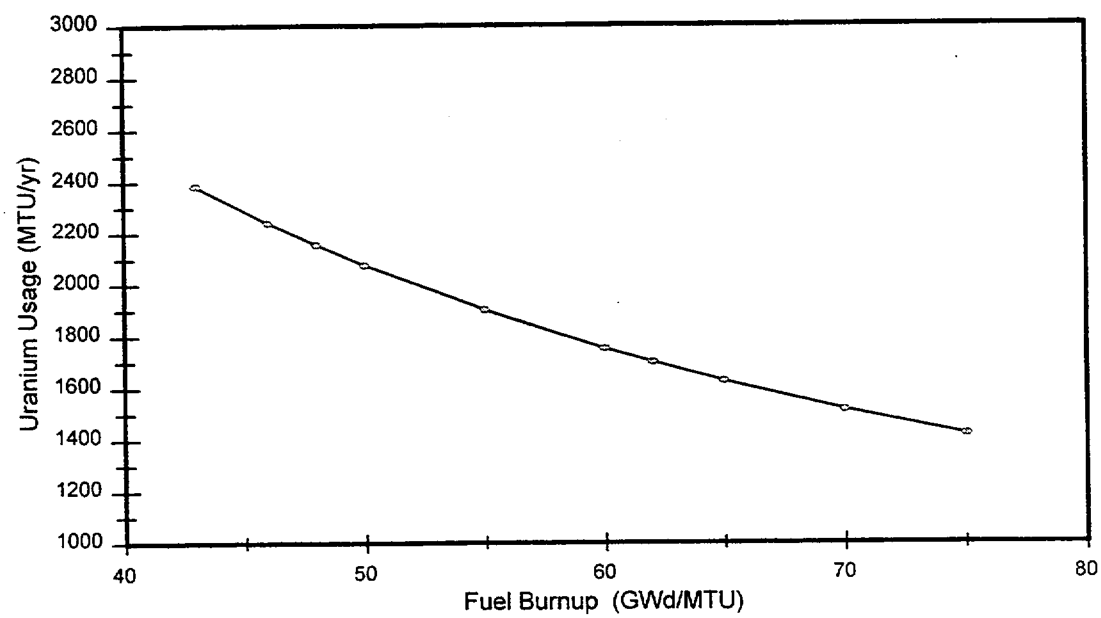

Figure 4.2 Decrease in Total Uranium Use for U.S. Reactors with Increased Burnup

The uranium use statistics in Table 4.1 are based on calculation of potential uranium use for each U.S. reactor assuming continuous operation at 100 percent of maximum power and three 45-day refueling outages during the fuel lifetime. Uranium usage is calculated as

$$U = \frac{365 U_{core}}{B U_{core} / P_{100\%} + 135}$$

where U = uranium use (MTU/yr)
U_{core} = total uranium (MTU) in the reactor
B = burnup (GWd/MTU)
$P_{100\%}$ = maximum reactor power (GW).

This relationship will tend to over estimate uranium use when average reactor power during power operation is less than maximum, which is the usual case. It will tend to underestimate the use if the average burnup at the end of fuel life is less than the burnup used in the calculations or the duration of outages during the fuel life is less than 135 days.

The decrease in uranium use shown in Table 4.1 and Figures 4.1 and 4.2 would be expected to result in a similar decrease in the environmental impacts of all activities in the front end of the fuel cycle including mining, milling, conversion, enrichment, and fuel fabrication. However, Mauro et al. (1985) indicates that the reduction in environmental effects is less than the reduction in uranium use because the grade of ore was assumed to decrease with time. Nevertheless, there should be a reduction in the environmental effects of the front end of the fuel cycle with increased burnup because increases in burnup can be achieved without further increase in enrichment.

5 Normal Operations

This chapter examines the potential impacts of increasing fuel burnup on the environmental effects of normal reactor operation, waste management, and storage of spent fuel in the spent-fuel pool. In considering the effects of normal reactor operation on the environment related to operating license renewal, the NRC staff included the following statement in the *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* (NRC 1996, p. 4-84):

> The Atomic Energy Act requires NRC 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. This definition of "small" applies to occupational doses as well as to doses to individual members of the public. Accidental releases or noncompliance with the standards could conceivably result in releases that would cause moderate or large radiological impacts. Such conditions are beyond the scope of regulations controlling normal operations and providing an adequate level of protection. Given current regulatory activities and past regulatory experience, the Commission has no reason to expect that such noncompliance will occur at a 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.

The NRC staff conducts safety evaluations to ensure that increases in burnup will not result in releases or doses that exceed regulatory limits. Safety evaluations are made using very conservative assumptions that maximize dose estimates. Assumptions in environmental assessments are more realistic and give lower dose estimates. Therefore, the radiological doses from normal operation with extended burnup fuel will remain below regulatory limits. Applying the logic cited above, the radiological impacts of operation with extended burnup fuel are expected to remain small.

5.1 Normal Reactor Operations

The radiological impacts of normal reactor operation are generally related to release of activity in the reactor cooling water. The potential for radionuclide releases to cooling water in normal operation is determined by the fraction of the activity in the gap. Current fuel designs have greater plenum volumes and higher gap-release fractions than the fuel designs considered in NUREG/CR-5009 (Baker et al. 1988). Therefore, the potential for environmental effects of normal operation with current fuel designs is larger than indicated in NUREG/CR-5009.

However, 10 CFR 50.36a imposes conditions on licensees covering effluents from nuclear power reactors. These conditions are intended to keep releases of radioactive materials to unrestricted areas during operations to levels that are "as low as reasonably achievable" (ALARA). Limitations on the maximum cooling water activity are independent of burnup. Plant operators are required to take actions to reduce activity or shut down the reactor if the cooling water activity approaches a limiting value. Consequently, the potential environmental effects of normal operation are independent of burnup.

In reality, plant coolant activities have been decreasing in the last 8 to 10 years because of a decrease in the number of fuel failures (Yang 1997). This decrease is not burnup related; rather, it is due to improved quality control in fuel-rod fabrication. NUREG-1437 (NRC 1996, p.4-85) also notes that:

> Radioactive-waste management systems are incorporated into each plant and are designed to remove most of the fission-product radioactivity that leaks from the fuel, as well as most of the activation- and corrosion-product radioactivity produced by neutrons in the vicinity of the reactor core.... Improved fuel integrity in the 1980s was an important factor in reducing effluents. In addition, the effectiveness of the gaseous and liquid treatment equipment has increased significantly over the past two decades, as is evidenced by the continuously decreasing levels of effluents (NUREG/CR-2907) [Tichler et al. 1993].

The design and operation of effluent control systems are governed, in part, by the requirements of Appendix I to 10 CFR Part 50. The objective of these requirements is to maintain the activity in reactor effluents "As Low As is Reasonably Achievable" (ALARA). Historically, activity releases have been small fractions of the requirements and guidelines of Appendix I. Since these requirements and guidelines are independent of burnup, as are the plant effluent releases, any increase in activity releases would be of small significance.

5.2 Low-Level Waste

The effects of increasing burnup on low-level waste are discussed by Mauro et al. (1985), Baker et al. (1988), and in NUREG-1437 (NRC 1996). Although Mauro et al. conservatively assumed that the impacts of low-level waste were constant, independent of burnup, Baker et al., based on data in Mauro et al., conclude that increasing burnup decreases the impacts of waste management activities, except possibly for activities associated with low-level waste disposal. The general decrease in the quantity of low-level waste with increasing burnup is associated with increased time between refueling outages. Data included in NUREG-1437 indicate that the volume of low-level waste and the activity in the low-level waste has been decreasing with time, not increasing. Similarly, Yang (1997) indicates that the number of fuel failures has decreased in time. There is no reason to expect a reversal of these trends.

5.3 Onsite Storage of Spent Fuel

Two aspects of onsite storage of spent fuel have been considered. These are the effects of increased fuel burnup on available space within the spent-fuel storage pool and the effects of increased fuel burnup on decay heat.

Nuclear power plants have a limited storage capacity for spent fuel. Increasing fuel burnup reduces the number of fuel assemblies that are added to the spent-fuel pool each year. This increases the time during which the plant has the storage capacity to permit the licensee to fully defuel the reactor. It also delays the time at which fuel must be moved from the spent-fuel pool to another facility. This delay has the positive environmental impact of reducing doses associated with incident-free transportation of fuel and the potential consequences of spent-fuel transportation accidents.

Increasing fuel burnup increases the residual activity in fuel elements that are stored in the spent-fuel pool. Table 5.1 shows the variation of decay heat with burnup and decay time (time after discharge). At the time of discharge, the decay heat is nearly independent of burnup because the primary source of the heat is decay of short-lived radionuclides with activity proportional to reactor power rather than burnup. By the end of the first year after discharge, decay heat increases with burnup, as expected because the heat is from decay of radionuclides that have activities proportional to burnup.

Table 5.1 PWR Spent-Fuel Decay Heat (w/MTU) as a Function of Peak-Rod Burnup (GWd/MTU) and Cooling Time (yr)

	Peak-Rod Burnup (GWd/MTU)						
Cooling Time (yr)	**43**	**50**	**60**	**62**	**65**	**70**	**75**
0	1.70E+06	1.69E+06	1.68E+06	1.67E+06	1.67E+06	1.66E+06	1.65E+06
1	1.00E+04	1.12E+04	1.28E+04	1.33E+04	1.38E+04	1.46E+04	1.55E+04
2	5.54E+03	6.35E+03	7.57E+03	7.82E+03	8.20E+03	8.85E+03	9.54E+03
3	3.69E+03	4.32E+03	5.28E+03	5.48E+03	5.79E+03	6.32E+03	6.90E+03
5	2.28E+03	2.74E+03	3.46E+03	3.60E+03	3.84E+03	4.26E+03	4.71E+03
7	1.82E+03	2.20E+03	2.80E+03	2.93E+03	3.13E+03	3.48E+03	3.86E+03
10	1.55E+03	1.87E+03	2.37E+03	2.48E+03	2.65E+03	2.95E+03	3.27E+03
15	1.34E+03	1.61E+03	2.03E+03	2.12E+03	2.26E+03	2.51E+03	2.77E+03
20	1.20E+03	1.44E+03	1.81E+03	1.89E+03	2.01E+03	2.22E+03	2.45E+03
30	9.96E+02	1.19E+03	1.49E+03	1.54E+03	1.64E+03	1.80E+03	1.99E+03

However, when the heat added to the spent-fuel pool is calculated on a per-reactor-year basis, the decrease in fuel usage offsets the increase in decay heat with burnup. Table 5.2 shows the total decay heat released to the spent-fuel pool at the end of 5, 10, 15, and 20 years by fuel required for one reactor year. The decay heat in the first year after discharge is the dominant factor in the total spent-fuel pool heat load. Decay heat in subsequent years is sufficiently small that the increased decay heat of higher burnup fuel is not environmentally significant.

Table 5.2 Total PWR Spent-Fuel Decay Heat (GWd) as a Function of Peak-Rod Burnup (GWd/MTU) and Time Since Discharge for Fuel Discharged in One Reactor Year							
Years After Discharge	**Peak-Rod Burnup (GWd/MTU)**						
	43	**50**	**60**	**62**	**65**	**70**	**75**
5	2.83	2.52	2.20	2.14	2.07	1.96	1.86
10	2.90	2.60	2.29	2.22	2.16	2.05	1.96
15	2.96	2.66	2.35	2.29	2.22	2.12	2.03
20	3.01	2.71	2.41	2.35	2.28	2.18	2.09

6 Postulated Accidents

The potential environmental impacts of extended fuel burnup associated with postulated accidents were evaluated by considering four classes of accidents. The accidents considered were a LOCA, a PWR steam generator tube rupture, a BWR main steam line break, and a fuel-handling accident. The first of these accidents generally represents a bounding case involving fuel damage. The second and third accidents represent accidents in which gap activity is released to the coolant and then the coolant activity is released to the environment, and the fourth accident is an accident in which gap activity is released to the spent-fuel pool and then to the environment. Analysis of each of the first three accidents is based on core-average activity because fuel damage or fuel-rod leaks typically involves many rods and not necessarily the peak rod. The fourth accident assumes peak-rod activity because analysis of fuel-handling accidents has traditionally assumed that the accident involves fuel assemblies containing the peak rod.

The releases modeled in the accident calculations in this report are simple because the emphasis of this review is to determine the change in environmental effects of nuclear power production associated with increasing fuel burnup. The models do not take into account natural processes and engineered safety features that would reduce the releases to the environment, and therefore the results tend to be conservative. In general, the effectiveness of the natural processes and engineered safety features, if included, would be insensitive to small changes in the radionuclide activity present. In particular, their effectiveness would not be sensitive to burnup. Consequently, more attention should be paid to trends in doses than to absolute values.

6.1 Loss-of-Coolant Accident

Potential human health impacts from LOCAs due to increasing fuel burnup were estimated using the MACCS2 computer program (Chanin and Young 1997). This program evaluates individual and population impacts based on user-defined releases, site data, and emergency response scenarios. The source-term characterization for the LOCA is taken from the revised accident source term defined in NUREG-1465 (Soffer 1995) because NUREG-1465 captures "insights available from recent severe accident research on the phenomenology of fission product release and transport behavior." In the design basis accident portion of the revised source term, activity is released from the reactor vessel to the containment during the initial 1.8 hr period for PWRs and the initial 2 hr period for BWRs. The activity in containment then leaks to the environment at the containment design leak rate for the first day, and at one half of the design leak rate for the following 29 days, in accordance with the assumption in Regulatory Guide 1.4 (NRC 1974b). The LOCA analysis used a typical design leak rate of 0.1 percent per day.

An activity release for each individual radionuclide was calculated for each of four time periods—0 to 8 hr, 8 to 24 hr, 24 to 96 hr, and 96 to 720 hr—as

$$Q_{ij} = P \cdot A_i \cdot f_i \cdot L'_j \cdot \Delta t_j$$

where Q_{ij} = activity of isotope i released in time period j (Bq)
P = reactor power, assumed to be 3,000 MW_t
A_i = normalized total activity of i in the reactor core (Bq/MW_t)
f_i = fraction of A_i in core released to containment
L'_j = containment leak rate in time period j (fraction/day)
Δt_j = duration of time period j (days).

Core-average inventories for PWRs and BWRs are presented in Appendix E for peak-rod burnup ranging from 22 MWd/MTU to 75 MWd/MTU. Tables 3.12 and 3.13 of NUREG-1465 (Soffer et al. 1995) provide estimates of the core release fractions for eight radionuclide groups for BWRs and PWRs respectively. The environmental release fractions—products of the core release fractions and leak rates ($f_i L'_j$)—are shown in Table 6.1. MACCS2 accounts for decay in radionuclide inventories in containment but not during transport through the environment.

The atmospheric model in MACCS2 is a segmented plume, Gaussian model that uses hourly meteorological data for transport and dispersion calculations. Ground-level releases were assumed in this analysis, and doses calculations were made beginning at an assumed exclusion area boundary of 0.8 km (0.5 mi). Site data include meteorological data and the population distribution. Hourly meteorological data from Moline, Illinois were used for the analysis because they provide consequence estimates in the middle of the range of consequences estimated using meteorological data sets from around the United States. Table 6.2 shows doses for an individual at the assumed site boundary of 0.8 km (0.5 mi) and the population dose for several meteorological data sets.

Table 6.1 LOCA Environmental Release Fractions for PWRs and BWRs

	PWR				BWR			
	Time Period (hr)				Time Period (hr)			
Radionuclide Group	**0-8**	**8-24**	**24-96**	**96-720**	**0-8**	**8-24**	**24-96**	**96-720**
Noble Gases	2.87E-4	6.67E-4	1.50E-3	1.30E-2	2.83E-4	6.67E-4	1.50E-3	1.30E-2
Halogens	1.16E-4	2.67E-4	6.00E-4	5.20E-4	8.65E-5	2.00E-4	4.50E-4	3.90E-3
Alkali Metals	8.75E-5	2.00E-4	4.50E-4	3.90E-4	7.24E-5	1.64E-4	3.75E-4	3.25E-3
Tellurium group	1.43E-5	3.33E-5	7.50E-5	6.50E-4	1.41E-5	3.33E-5	7.50E-5	6.50E-4
Barium, Strontium	5.71E-6	1.33E-5	3.00E-5	2.60E-4	5.63E-6	1.33E-5	3.00E-5	2.60E-4
Noble Metals	7.14E-7	1.67E-6	3.75E-6	3.25E-5	7.03E-7	1.67E-6	3.75E-6	3.25E-5
Cerium group	1.43E-7	3.33E-7	7.50E-7	6.50E-6	1.41E-7	3.33E-7	7.50E-5	6.50E-6
Lanthanides	5.71E-8	1.33E-7	3.00E-7	2.60E-6	5.63E-8	1.33E-7	3.00E-7	2.60E-6

Table 6.2 Mean Consequences Estimates for Seven Meteorological Data Sets		
Meteorological Data Site	**Individual Dose (Sv[a]) at 0.8 km (0.5 mi)**	**Population Dose (person-Sv)**
Boston, MA	0.095	1.4E+04
El Paso, TX	0.11	1.2E+04
Miami, FL	0.11	1.1E+04
Moline, IL	0.10	1.2E+04
Phoenix, AZ	0.11	1.0E+04
Santa Monica, CA	0.11	1.1E+04
Seattle, WA	0.10	1.2E+04
[a]1 Sv = 100 rem		

The population distribution shown in the Table 6.3 was used in all MACCS2 calculations. For distances within 80 km (50 mi), the population densities are based on the averages of reported densities for operating nuclear power reactors,[a] and beyond 80 km, the mean density is based on the 5 reactor sites included in the severe accident risk study (NRC 1991). The population densities were assumed to be uniform in all directions.

Table 6.3 Population Distribution for MACCS2 Calculations		
Distance Interval, km (1 km = 0.62 mi)	**Population Density, (people/km^2) (1 person/km^2 = 2.59 people/mi^2)**	**Total Population**
0.8 to 3.2	50.8	1,530
3.2 to 8.0	62.5	10,600
8.0 to 16	70.3	42,400
16 to 80	105.0	2,040,000
> 80	19.5	157,000,000

Inhalation dose commitments and external doses were calculated using the MACCS code package. The MACCS2 code package includes dose conversion factors tabulated in Federal Guidance Reports 11 and 12 (Eckerman et al. 1989; Eckerman and Ryman 1993). These dose conversion factors were used in the analyses.

The population doses include exposure from ingestion of contaminated foods. Exposure from contaminated foods was estimated using the MACCS2 option involving unit dose factors generated by the COMIDA2 computer program. This program is based on the food-chain model COMIDA developed

[a]U.S. Nuclear Regulatory Commission at http://www.nrc.gov/AEOD/pib/disclaimer.html (October 25, 2000).

by Abbott and Rood (1993, 1994) with modification by Sandia National Laboratories (Chanin and Young 1997). Food pathway parameters recommended in NUREG/CR-5512 (Kennedy and Strenge 1992) were used.

The MACCS2 dose estimates also include consideration of emergency response measures and long-term interdiction. Modeling of emergency response and long-term interdiction were based on analyses performed for NUREG-1150. In this representation, 95% of the people are assumed to begin moving at the start of the accident with an average travel speed of 1.8 m/s (4 mi/hr). This low speed is believed to represent a reasonable average value for the total population over the first few hours of the accident.

Two sets of analyses were performed using MACCS2 to determine the potential effects of increasing burnup. First, seven analyses were performed for PWRs assuming 62 GWd/MTU burnup to determine the sensitivity of the results to the choice of the meteorological data set. As the values in Table 6.2 indicate, the selection of the meteorological data set has little impact on the estimated consequences. The meteorological data set for Moline, Illinois was selected for use in the remaining analyses of fuel burn-up impacts because it gives results that are approximately in the middle of the range of results.

The second set of analyses involved estimating cumulative consequences (doses) for postulated PWR and BWR LOCAs. Results for these analyses are presented in Table 6.4. The emergency response and long-term interdiction measures in MACCS2 effectively limit the projected doses, both for an individual at the site boundary and to the population.

Table 6.4 Mean LOCA Consequences Based on Core-Average Inventories as a Function of Peak-Rod Burnup For PWRs and BWRs

Reactor Type	Peak-Rod Burnup (GWd/MTU)	Individual Dose at 0.8 km[a] (Sv[b])	Total Population Dose (person-Sv)
PWR	42	1.0E-01	9.4E+03
	50	1.0E-01	1.1E+04
	60	1.0E-01	1.2E+04
	62	1.0E-01	1.2E+04
	65	1.1E-01	1.2E+04
	70	1.1E-01	1.3E+04
	75	1.1E-01	1.3E+04
BWR	60	1.0E-01	1.3E+04
	62	1.0E-01	1.3E+04
	65	1.0E-01	1.3E+04
	70	1.1E-01	1.4E+04
	75	1.1E-01	1.4E+04

[a]0.8 km = 0.5 mi
[b]1 Sv = 100 rem

The consequences of a postulated LOCA increase by about ten percent as peak-rod burnup increases from 60 GWd/MTU to 75 GWd/MTU. The individual doses at 0.8 km remain well below the 0.25 Sv whole body dose limit for an individual at the exclusion area boundary set in 10 CFR 100.11. Therefore, the increases in projected consequences of postulated LOCAs associated with increasing peak-rod burnup to 75 GWd/MTU are not considered to be significant.

6.2 PWR Steam Generator Tube Rupture/BWR Main Steam Line Break

The PWR steam generator tube rupture accident and the BWR main steam line break accident involve direct release of radioactivity from the contaminated reactor coolant to the environment. Factors that increase gap activity in reactor fuel, including extended burnup and fuel design modifications associated with extended burnup, might be thought to increase the potential consequences of these accidents. However, in practice, increased gap activity does not necessarily lead to increased coolant activity. Two factors mitigate the potentially adverse impacts of increasing fuel burnup on the environmental consequences of steam generator tube rupture and main steam line break accidents.

The first of these factors is operational experience; coolant activity has been decreasing as fuel burnup has been increasing (Yang 1997). The decreasing trend in coolant activity is attributed to a decrease in fuel failures resulting from improved quality control in fuel-rod fabrication. The second factor is that maximum coolant activity is regulated through technical specifications that are independent of fuel burnup.

Safety evaluations conducted under NRC Standard Review Plans are based on assumptions that coolant activity is at the technical specification limit at the time of a postulated steam generator rupture or main steam line break and that the accident occurs under adverse meteorological conditions. The consequences associated with a postulated accident are acceptable if the potential consequences of the accident are within regulatory limits. Environmental assessments are based on more typical meteorological conditions for postulated accidents than safety evaluations (NRC 1999b). As a result, the consequences of a postulated accident described in environmental assessments tend to be about an order of magnitude lower than the consequences for the same accident described in the Safety Evaluation Report. Increasing burnup will not lead to environmental consequences that exceed consequences already considered because coolant activity is limited by technical specifications that are independent of burnup.

In licensing analyses, some steam generator tube rupture and main steam line break accident sequences lead to fuel failure. The consequences of the sequences, when evaluated using adverse meteorological conditions, must be below regulatory limits set in 10 CFR 100.11. Evaluation of the consequences of these sequences for environmental reviews using typical meteorological conditions will lead to doses that are well below regulatory limits.

The consequences of increasing burnup are primarily a function of changes in the probability of fuel damage, changes in the extent of fuel failure and increases in the gap-release fraction. As indicated above, increasing fuel burnup has not been accompanied by an increase in fuel failures during normal operation. Although some steam generator tube rupture and main steam line break accident sequences postulate fuel damage, such damage is independent of fuel burnup. While the gap-release fraction may be higher in extended burnup fuel, the fuel damage that might occur during these accidents would be expected to be limited to cladding failure, not fuel melt.

Releases from the fuel in these accidents would be limited to radionuclides in the gap because the accidents do not generally involve a fuel melt. As indicated in Chapters 2 and 3 of this report, the gap-release fraction increases with increasing burnup. Small increases in gap-release fractions will occur as peak-rod burnup increases from 60 GWd/MTU to 62 GWd/MTU. The core-average release fractions at 62 GWd/MTU remain below the gap-release fractions postulated in regulatory guidance.

6.3 Fuel-Handling Accident

The scenario used in evaluation of potential fuel-handling accidents involves a direct release of gap activity to the environment. Regulatory Guide 1.25 (NRC 1972) presents an acceptable methodology for evaluating the consequences of fuel handling accidents. However, Regulatory Guide 1.25 does not include consideration of the isotopes of cesium that NUREG-1465 and the evaluation of release fractions in Chapter 3 of this report indicate should be present in the gap. Regulatory Guide 1.183 (NRC 2000), which has recently been published, includes consideration of cesium and updates assumptions related to fuel-handling accidents.

Inhalation and external doses for the fuel-handling accident were calculated following the general guidance set forth in Regulatory Guide 1.25 and RG 1.183. Inhalation dose commitments (committed effective dose equivalent [CEDE] and thyroid) were calculated using

$$D_{inh} = M_U V_b \left(\chi / Q \right) \sum_i \frac{f_{gi} Q_i DF_i}{F_{pi}}$$

where D_{inh} = inhalation dose (Sv)
M_U = metric tons of uranium (number of assemblies x tons per assembly)
V_b = breathing rate (3.47×10^{-4} m^3/s)
χ/Q = ~50% atmospheric dispersion factor (s/m^3)
f_{gi} = gap-release fraction for isotope
Q_i = isotope activity (Bq/MTU)
DF_i = dose factor for isotope from Federal Guidance Report 11 (Sv/Bq inhaled)
F_{pi} = pool decontamination factor for isotope
i = isotope.

and, external doses were calculated using

$$D_{ext} = M_U \left(\chi / Q \right) \sum_i \frac{f_{gi} Q_i DF_i}{F_{pi}}$$

where DF_i is the dose factor for isotope from Federal Guidance Report 12 (Sv/s)/(Bq/m^3).

For the present analysis, a fuel-handling accident involves two fuel assemblies, one falling upon and damaging another such that the fission products in the gaps of the rods in both assemblies are released to the spent-fuel pool. Thus, the PWR fuel-handling accident involves approximately 1.05 MTU, and the BWR fuel-handling accident involves approximately 0.367 MTU. (The reactor statistics in Appendix A of this report include the mass of uranium in a fuel assembly.)

Technical specifications address the required cooling time between reactor shutdown and the movement of fuel. Based on the generic technical specifications, the fuel-handling accident for PWRs is assumed to occur 72 hours after reactor shutdown, while for BWRs it is assumed to occur 24 hours after shutdown.

The gap activity of concern is based on guidance in RG 1.183 (and NUREG-1465). It consists of the noble gases, iodines, and cesiums. In addition, ^{137m}Ba is assumed to be present in the gaps of both assemblies in equilibrium with ^{137}Cs. Fuel inventories for the accidents are based on peak-rod calculations discussed in Section 2.1 of this report. The inventories used are given in Tables 6.5 and 6.6. Gap-release fractions from Table 2.3 and 2.4 were used in the calculations.

Decontamination factors assumed in the fuel-handling accident are based on guidance in RG 1.183. Pool decontamination factors of 1 and 200 are assumed for noble gases and iodines, respectively. A decontamination factor of 200 for iodines is justified because the temperature of fuel rods entering the spent-fuel pool should be below 100°C, and at that temperature, the pressure should be less than 1200 psi. (Note that FRAPCON-3 calculations performed for higher burnup fuel indicate pressures of about 900 psia compared with pressures in excess of 1200 psia projected by some vendor codes.) RG 1.183 suggests an infinite decontamination factor for cesiums and rubidiums. However, as a conservative measure, a decontamination factor of 1000 has been used for cesiums. A decontamination factor of 1000 has also been used for ^{137m}Ba. The activities of the rubidium isotopes are negligible compared to ^{131}I and ^{134}Cs.

Effluents from the accident were released directly to the environment over a short period of time and doses were calculated at 800 m, the assumed distance to the exclusion area boundary. The ARCON96 code (Ramsdell and Simonen 1997) was used to calculate atmospheric dispersion factors (χ/Qs) for ground-level releases and an 800-m exclusion area boundary using meteorological data from 14 locations around the United States assuming a building area of 2,000 m^2. Fiftieth percentile χ/Qs were determined for each site for the fuel-handling accident dose calculations. They ranged from 1.0×10^{-5} to 2.1×10^{-5} (Bq/m^3)/(Bq/s), with a median value of 1.8×10^{-5} (Bq/m^3)/(Bq/s). This median value was used in the

Table 6.5 Peak-Rod Radionuclide Inventories (Bq/MTU) for PWR Fuel-Handling Accidents				
	Peak-Rod Burnup (GWd/MTU)			
Isotope	42	50	60	62
Kr-85	4.55E+14	5.14E+14	5.74E+14	5.85E+14
Kr-87	2.07E-01	1.89E-01	1.36E-01	1.33E-01
Kr-88	7.22E+08	6.55E+08	4.70E+08	4.59E+08
I-131	3.60E+16	3.52E+16	2.83E+16	2.83E+16
I-132	3.54E+16	3.45E+16	2.76E+16	2.75E+16
I-133	8.73E+15	8.44E+15	6.70E+15	6.66E+15
I-134	8.03E-08	7.62E-08	5.96E-08	5.88E-08
I-135	4.48E+13	4.33E+13	3.46E+13	3.44E+13
Xe-133	7.29E+16	7.22E+16	5.74E+16	5.70E+16
Xe-135	9.66E+14	9.25E+14	7.44E+14	7.40E+14
Cs-134	7.47E+15	9.88E+15	1.28E+16	1.34E+16
Cs-136	2.03E+15	2.46E+15	2.55E+15	2.67E+15
Cs-137	5.11E+15	6.03E+15	7.18E+15	7.40E+15
Ba-137m	4.83E+15	5.71E+15	6.79E+15	7.00E+15

fuel-handling accident dose calculations. A standard breathing rate of 3.47×10^{-4} m^3/s was used in the calculations. Dose factors were taken from Federal Guidance Reports 11 and 12 (Eckerman et al. 1989; Eckerman and Ryman 1993).

The results of the fuel-handling accident calculations are shown in Table 6.7. All doses are well below regulatory limits. Table 6.8 breaks down the dose estimates by isotope for a PWR fuel-handling accident with 62 GWd/MTU fuel. The only isotopes that contribute significant fractions of the CEDE and thyroid doses are ^{131}I and ^{134}Cs. Similarly, the only isotopes that contribute significant fractions of the deep dose are ^{132}I and ^{133}Xe.

Even though the iodine inventory decreases with increasing burnup, the potential doses from fuel-handling accidents increase with fuel burnup for PWRs because of increased gap-release fraction. However, even with an increase in burnup to 62 GWd/MTU, the doses remain well below regulatory limits. The potential doses associated with BWR fuel-handling accidents tend to decrease with increasing burnup.

Table 6.6 Peak-Rod Radionuclide Inventories (Bq/MTU) for BWR Fuel-Handling Accidents				
	Peak-Rod Burnup (GWd/MTU)			
Isotope	42	50	60	62
Kr-85	4.40E+14	4.92E+14	5.40E+14	5.48E+14
Kr-87	3.24E+10	2.60E+10	1.92E+10	1.87E+10
Kr-88	6.07E+13	4.88E+13	3.57E+13	3.46E+13
I-131	2.82E+16	2.47E+16	2.05E+16	2.03E+16
I-132	3.63E+16	3.17E+16	2.60E+16	2.58E+16
I-133	2.89E+16	2.50E+16	2.04E+16	2.02E+16
I-134	1.62E+09	1.38E+09	1.11E+09	1.09E+09
I-135	4.74E+15	4.11E+15	3.37E+15	3.34E+15
Xe-133	5.85E+16	5.11E+16	4.29E+16	4.26E+16
Xe-135	1.60E+16	1.39E+16	1.15E+16	1.14E+16
Cs-134	6.81E+15	8.77E+15	1.10E+16	1.14E+16
Cs-136	1.94E+15	2.12E+15	2.27E+15	2.39E+15
Cs-137	5.07E+15	5.96E+15	7.03E+15	7.22E+15
Ba-137m	4.80E+15	5.64E+15	6.65E+15	6.83E+15

Table 6.7 Exclusion Area Boundary Dose Estimates (mSv) for Fuel-handling Accidents						
Peak- Rod Burnup	PWR			BWR		
GWd/MTU	CEDE (mSv)[a]	Thyroid (mSv)	Deep Dose (mSv)	CEDE (mSv)	Thyroid (mSv)	Deep Dose (mSv)
42	0.33	7.9	0.03	0.34	9.7	0.06
50	0.51	12.	0.04	0.28	7.5	0.04
60	0.69	18.	0.07	0.22	5.5	0.03
62	0.76	19.	0.08	0.22	5.5	0.03

[a]1 mSv = 0.1 rem.

Table 6.8 Exclusion Area Boundary Dose Estimates by Isotope for PWR Fuel-Handling Accidents with 62 GWd/MTU Fuel			
Isotope	**CEDE (mSv)[a]**	**Thyroid Dose (mSv)**	**Deep Dose (mSv)**
Kr-85	0.000	0.000	0.000
Kr-87	0.000	0.000	0.000
Kr-88	0.000	0.000	0.000
I-131	0.558	18.3	0.003
I-132	0.006	0.106	0.020
I-133	0.022	0.718	0.001
I-134	0.000	0.000	0.000
I-135	0.000	0.001	0.000
Xe-133	0.000	0.000	0.047
Xe-135	0.000	0.000	0.002
Cs-134	0.121	0.108	0.002
Cs-136	0.004	0.003	0.001
Cs-137[b]	0.046	0.042	0.000
Total Dose	0.757	19.3	0.076

[a] 1 mSv = 0.1 rem.
[b] ^{137}Cs doses include the contribution from ^{137m}Ba.

7 Transportation

This chapter discusses the potential environmental effects of transportation of reactor fuel related to extended burnup. Section 7.1 briefly discusses the effects of extending fuel burnup related to transportation of fresh fuel, and Section 7.2 discusses the effects of extending fuel burnup related to transportation of spent fuel. Section 7.2 discusses both incident-free transportation and transportation accidents. Environmental effects of the transportation of spent nuclear fuel have been considered in other studies not directly concerned with the consequences of extending fuel burnup. Appendix F presents a comparison of the bases used in this study's assessment of the impacts of transportation spent fuel with burnup above 60 GWd/MTU with the bases used in these other studies. In general, the bases used in this study are consistent with the bases used in the other studies.

7.1 Fresh Fuel

NUREG/CR-5009 (Baker et al. 1988) considered the environmental effects of extending peak-rod burnup from 33 GWd/MTU to 60 GWd/MTU. To accomplish this increase in burnup, it was necessary to increase the ^{235}U enrichment from about 3.5 percent to about 5 percent. The increase in enrichment was determined to have an insignificant effect on radiation exposures associated with transportation of fresh fuel. Therefore, increasing fuel burnup was determined to reduce the environmental effects of transportation of fresh fuel because the number of fuel shipments would decrease in proportion to the increase in burnup.

Fuel burnup can be increased from 60 GWd/MTU to 75 GWd/MTU without increasing enrichment. Increases in burnup in this range will reduce fresh fuel transportation requirements because of the decrease in fuel usage. Consequently, radiation exposure associated with shipment of fresh fuel should decrease in proportion to the increase in burnup.

7.2 Spent Fuel

This section discusses the impact of extending fuel burnup on the environmental effects of transportation of spent fuel. The section is divided into two parts. The first part considers incident-free transportation, and the second part considers transportation accidents.

The analysis is based on shipment of spent fuel by legal-weight trucks in casks with characteristics similar to casks currently available. Each shipment is assumed to consist of a single cask. These assumptions are consistent with assumptions made in the evaluation of the environmental impacts of transportation of spent fuel presented in Addendum 1 to NUREG-1437 (NRC 1999a). As discussed in Addendum 1, these assumptions are conservative because the alternative assumptions involve rail transportation or heavy-haul trucks, which would reduce the number of spent-fuel shipments. A limited evaluation was made of the environmental impacts of shipment of spent fuel by rail to illustrate that these impacts are significantly lower than those for legal-weight trucks. The environmental impacts associated with transportation of spent fuel using heavy-haul trucks should be between those for shipment by legal-weight trucks and rail shipment considering the number of shipments that would be required using heavy-haul trucks.

For PWR fuel, the shipping cask was assumed to be rated for four fuel assemblies (0.452 MTU/ assembly) containing 5 percent enriched fuel irradiated to a burnup of 43 GWd/MTU. The rail shipping cask was assumed to hold 40 fuel assemblies. These ratings are assumed to be based on cask radiation limits. In practice, shipping casks loads may be limited to fewer assemblies for reasons such as crane capacity. Assuming that radiation from the cask is only a function of activity in the cask, the number of assemblies in a cask was reduced as the fuel burnup increased above 43 GWd/MTU, to keep the activity of key radionuclides in the cask at or below the activity of the same radionuclides in a cask full of assemblies at 43 GWd/MTU. The key radionuclides used were ^{60}Co, ^{106}Rh, ^{134}Cs, ^{137m}Ba, ^{144}Pr, and ^{154}Eu. Broadhead et al. (1995) determined that these six key radionuclides contribute more than 70 percent of the total radiation (gamma and neutron) coming from an iron cask. None of the remaining radionuclides contributes more than one percent to the total.

Table 7.1 shows the number of fuel assemblies per cask, relative activity per assembly, and number of shipments per reactor year as a function of burnup. Relative activity for a PWR is the weighted sum of the activities of the six key radionuclides in an assembly to the weighted sum of the activities in an assembly at 43 GWd/MTU burnup. For BWR assemblies, the reference burnup is 35 GWd/MTU. The fractional contribution of the dose rate for the radionuclide to total dose rate given by Broadhead et al., was used as the weight. For example, approximately one third of the dose rate outside an iron cask is associated with ^{60}Co; therefore, ^{60}Co was given a weight of 0.33.

For BWR fuel, the legal-weight truck shipping cask was assumed to be rated for nine assemblies (0.188 MTU/assembly) containing 5 percent enriched fuel irradiated to a burnup of 35 GWd/MTU. As the

Table 7.1 Spent-Fuel Shipment Characteristics

Burnup (GWd/MTU)	PWR					BWR		
	Assemblies per Cask		Relative Activity in Assembly	Shipments per Reactor Year		Assemblies per Cask	Relative Activity in Assembly	Shipments per Reactor Year
	Truck	Rail		Truck	Rail			
35						9	1.00	16.0
43	4	40	1.00	12.3	1.23	7	1.24	17.0
50	3	34	1.16	14.3	1.26	6	1.43	17.2
60	2	28	1.36	18.2	1.30	5	1.66	17.3
62	2	27	1.39	17.6	1.31	5	1.70	16.8
75	2	23	1.48	14.8	1.28	4	2.02	17.5

fuel burnup increased above 35 GWd/MTU, the number of assemblies in a cask was reduced as described above. The number of assemblies and relative activity per cask and number of shipments per reactor year for BWRs is also given in Table 7.1.

The product of the relative activity per assembly, assemblies per cask, and number of shipments per year is nearly constant. It varies less than 5 percent as the burnup is increased from 43 to 75 GWd/MTU for PWR fuel or 35 to 75 GWd/MTU for BWR fuel. This result is in contrast to the result in NUREG/CR-5009 (Baker et al. 1988) and Mauro (1985), where it was assumed that the number of shipments would be inversely proportional to the ratio of burnups because the fuel cooling time would be increased and the casks would improve with burnup to avoid the need to derate the casks to stay within dose guidelines. That is, the number of shipments would be reduced by 43%, assuming the radiation dose rates stay the same.

Environmental impacts of the transportation of spent fuel were calculated using the RADTRAN4 computer code (Neuhauser and Kanipe 1992). Routing and population data for input to RADTRAN for shipment by truck were obtained from the HIGHWAY code (Johnson et al. 1993a). The INTERLINE code (Johnson et al. 1993b) was used to generate rail routing and population information. The population data in the HIGHWAY and INTERLINE codes are based on the 1990 census.

7.2.1 Incident-Free Transportation of Spent Fuel

"Incident -free" transportation refers to transportation activities in which the shipments of radioactive material reach their destination without releasing any radioactive cargo to the environment. The vast majority of radioactive shipments are expected to reach their destination without experiencing an accident or incident or releasing any cargo. The "incident-free" impacts from these normal, routine shipments arise from the low levels of radiation that are emitted externally from the shipping container. Although Federal regulations in 10 CFR Part 71 and 49 CFR Part 173 impose constraints on radioactive material shipments, some radiation penetrates the shipping container and exposes nearby persons to low levels of radiation. The environmental impacts that result from these low-level exposures are quantified in this section as a function of fuel burnup.

Incident-free legal-weight truck transportation of spent fuel has been evaluated by considering shipments from six representative reactor sites to a repository at Yucca Mountain, Nevada, for disposal.[a] This assumption is conservative because it tends to maximize the shipping distance from the east coast and midwest where most of the reactors are located. Therefore, shipment to one or more other sites would reduce the impacts. Rail shipment of spent fuel was evaluated for a single reactor site in the northeast.

Environmental impacts from these shipments will occur to persons residing along the transportation corridors between the reactor sites and the repository, to persons in vehicles passing the spent-fuel

[a] This analysis addressed the impacts of spent nuclear fuel storage to a high level waste repository from a generic perspective. Because Congress has directed the U.S. Department of Energy to study only Yucca Mountain for the proposed repository, the analysis assumed that all spent nuclear fuel would be shipped to that repository.

shipments in the same and opposite directions, to persons at vehicle stops (such as rest areas, refueling stations, inspection stations, etc.), and to transportation crew members. The impacts to these exposed population groups were quantified using the RADTRAN4 computer code (Neuhauser and Kanipe 1992).

For purposes of this analysis, the transportation crew for truck spent-fuel shipments consisted of two drivers. Escorts were considered, but they were not included because their distance from the shipping cask would reduce the dose rates to levels well below the dose rates experienced by the drivers. Stop times were assumed to accrue at the rate of 0.002 hr/km (0.0032 hr/mi). This rate is based on data collected by Hostick et al. (1992). For consistency with the analysis in NUREG-1437 Addendum 1 (NRC 1999a), thirty members of the public were assumed to be within 20 m (66 ft) of the truck for the full duration of each stop. This is considered to be a conservative assumption.

The transportation crew for the rail shipment consisted of 5 members located 152 m (500 ft) from the shipping cask. Stop time accrued at the rate of 0.033 hr/km (0.053 hr/mi). At each stop, 100 members of the public were assumed to be 20 m from the shipment for the full duration of the stop.

The characteristics of specific shipping routes (e.g., population densities, shipping distances) influence the normal radiological exposures. To address the differences that arise from the specific reactor site from which the spent-fuel shipment originates, the United States was divided into five regions. A representative reactor site in each region (two in the southeast region) was chosen to illustrate the impacts of transporting spent fuel from a variety of possible locations. These regions and the representative reactors chosen for each region are:

- Northeast (NE) region—Millstone (BWR)
- Southeast (SE) region—Turkey Point (PWR), Brunswick (BWR)
- Midwest (MW) region—Zion (PWR)
- Southwest (SW) region—San Onofre (PWR)
- Northwest (NW) Region—WNP-2 (BWR)

Input to RADTRAN4 includes the total shipping distance between the origin and destination sites and the population distributions along the routes. This information was obtained by running the HIGHWAY computer code (Johnson et al. 1993a) for the origin-destination combinations of interest for legal weight trucks. It was obtained from the INTERLINE code (Johnson et al. 1993b) for rail shipments. The resulting route characteristics information is shown in Table 7.2. Note that for truck shipments, all the spent fuel is assumed to be shipped to the Clark County, Nevada border. The cumulative impacts of spent-fuel shipments within Clark County for all reactors were examined in NUREG-1437, Addendum 1 (NRC 1999a). Estimates of the impacts of shipping extended burnup fuel within Clark County in this report are based on information presented in NUREG-1437 Addendum 1 with adjustments for differences in assumptions. For rail shipments, the shipment impacts were calculated for shipment to Caliente, NV. Caliente is the first rail stop in Nevada on the route between Millstone and Yucca Mountain.

The radiation emitted from the spent-fuel shipping container (or shipping cask) is limited by Federal regulations. In previous environmental studies of spent-fuel transportation, including NUREG-1437 Addendum 1 (NRC 1999a), the radiation from the cask was assumed to be at or near the maximum allowed by Federal regulations. This is conservative in that the radiation could not be higher but is likely to be lower than Federal regulations allow, particularly if the spent-fuel assemblies are stored at the reactor sites for several years before they are shipped to the repository. Dose rates experienced by truck crews have generally been assumed to be 10 percent of the regulatory limit (AEC 1972; NRC 1977).

Table 7.2 Route Characteristics for Spent-fuel Shipments from Representative Regional Reactors Sites to the Clark County, Nevada Border

Origin Site	One-way Shipping Distance, km	Distance Traveled by Population Zone, km (1 km = 0.62 mi)			Population Density, people/km^2 (1 person/km^2 = 2.59 people/mi^2)		
		Rural	Suburban	Urban	Rural	Suburban	Urban
Southeast - Turkey Point, FL	4558	3606	820	132	7.7	349	2284
Southeast - Brunswick, NC	4001	3336	603	61	8.6	345	2188
Northeast - Millstone, CT Truck	4244	3391	744	109	7.9	348	2309
Northeast - Millstone, CT Rail	4555	3353	994	208	7.5	376	2417
Midwest - Zion, IL	2892	2557	286	49	4.3	366	2092
Southwest - San Onofre, CA	452	303	64	85	3.5	556	2807
Northwest - WNP-2, WA	1582	1373	177	32	4.7	459	2063

The RADTRAN code was run using the above assumptions. Table 7.3 gives the results of the calculations for a single shipment for each region. These doses, which are in person-rem to facilitate comparisons with 10 CFR 51.52 Table S-4, are independent of reactor type and burnup because the dose rates for all shipments are based on regulatory limits.

In NUREG/CR-5009, Baker et al. (1988) assumed that cooling time would be increased so that fully loaded casks could be shipped regardless of burnup. With this assumption, the doses per reactor year resulting from shipment of spent fuel decrease with increasing burnup. The same conclusion is reached if cask design is assumed to improve as burnup increases so that the dose rate remains within regulatory limits. A third, more conservative assumption has been made in this study. Current cask designs and 5-year cooled fuel are assumed, and dose rates are assumed to limit the activity carried per shipment. As the activity in fuel increases as a result of increased burnup, the number of assemblies per cask is reduced,

Table 7.3 Incident-Free Doses for One Shipment of Spent Fuel 5 Years After Reactor Discharge Assuming Dose Rates Based on Regulatory Limits				
	Normal Dose (person-rem/shipment) (1 person-rem = 0.01 person-Sv)			
	Crew	Onlookers	Along Route	Total
Southeast - Turkey Pt	0.027	0.096	0.20	0.32
Southeast - Brunswick	0.011	0.084	0.13	0.22
Southwest	0.0035	0.0095	0.072	0.084
Midwest	0.015	0.061	0.083	0.16
Northeast - Truck	0.025	0.089	0.18	0.29
Northeast - Rail	0.097	0.011	0.028	0.14
Northwest	0.0086	0.033	0.052	0.094

and the number of trips is increased. In addition, the dose rates are adjusted to account for the activity in the cask. Neglecting any potential effect of assembly geometry within the cask, each of these factors affects dose rates and doses linearly. As a result, they can be taken into account when estimating the cumulative doses per reactor year.

The neutron source terms increase with burnup. For a given cask design, higher burnup may mean that a partial shipment is required to ensure the total radiation dose rate (gamma plus neutron dose contributions) is below Federal regulations. However, because neutrons are effectively attenuated by low-density materials such as plastics and water, it is believed that minor modifications can be made to shipping casks to allow them to transport the higher burnup fuel at full load. Therefore, neutron radiation has not been considered further.

Table S-4 of 10 CFR 51.52 summarizes the environmental impacts of the transportation of fuel and waste to and from one light-water-cooled nuclear power reactor, assuming

- thermal power levels do not exceed 3800 MW_t
- reactor fuel is in the form of uranium oxide pellets with ^{235}U enrichment not exceeding 4 percent by weight
- average burnup of irradiated fuel from the reactor does not exceed 33 GWd/MTU
- no irradiated fuel assembly is shipped until at least 90 d after it is discharged from the reactor
- spent-fuel shipment distances do not exceed 1000 mi.

In a series of environmental studies (Mauro et al. 1985; Baker et al. 1988; NRC 1996, 1999a), the description of impacts in Table S-4 has been found to be bounding for five-percent enriched fuel with burnup to 60 GWd/MTU (62 GWd/MTU in Addendum 1 to NUREG-1437), provided that the fuel is shipped at least 5 years after discharge. This discussion addresses whether or not the environmental effects of normal spent-fuel shipments with burnup up to 75 GWd/MTU are within the bounds established in Table S-4 of 10 CFR 51.52. The bounding cumulative doses to the exposed population are:

- Transport workers — 4 person-rem/reactor-year[a]
- General public (onlookers) — 3 person-rem/reactor-year
- General public (along route) — 3 person-rem/reactor-year.

Calculation of the cumulative doses entailed converting the per-shipment risks given in Table 7.3 to estimates of environmental effects per reactor-year of operation. The per-shipment results, which are independent of burnup, were converted to burnup-dependent effects per reactor year by multiplying the values in Table 7.3 by the number of assemblies in the cask, the relative activity in each assembly, and the number of shipments per reactor-year, and dividing the result by the maximum number of assemblies that the cask will hold.

The results of these calculations are shown in Tables 7.4 and 7.5 for spent-fuel shipments from all five regions to the Clark County, Nevada, border. Each shipment was assumed to consist of fuel assemblies with burnup equivalent to the peak-rod burnup. Consequently, the dose estimates in Tables 7.4 and 7.5 are conservative. Note that if spent fuel is cooled for 10 years, derating of the shipping casks is not necessary, and doses will be below those listed in Table 7.4. Based on information in NUREG-1437 Addendum 1, it is estimated that transportation of 62 GWd/MTU spent fuel within Clark County will contribute an additional 0.05 person-rem per reactor-year to crew (transport worker) doses, 0.13 person-rem per reactor-year to onlooker doses, and 0.20 person-rem per reactor-year to along route doses.

The sum of the along route doses to and within Clark County for BWR spent fuel with burnup below 62 GWd/MTU from the northeast region slightly exceeds the value given in Table S-4. All other combined doses are lower than the values in Table S-4. Along route doses are highly sensitive to the transport speed assumed in urban and urban areas. Transport speeds of 24 and 40 km/hr (15 and 25 mph) were assumed for consistency with earlier studies. These speeds are extremely low for the current interstate highway system and wireless communications. A study entitled Reexamination of Spent Fuel Shipment Risks (NUREG/CR-6672) (Sprung et al. 2000) published since the transportation calculations were made for this study assumed a speed of 88 km/hr for rural, suburban, and urban areas. Assuming a speed of 88 km/hr for suburban and urban areas for spent fuel transport reduces the along route doses for BWRs in the northeast region to less than 1 person-rem per reactor-year, and the along route doses within Clark County to less than 0.07 person-rem per reactor-year. In fact, increasing the urban speed to 28 km/hr (17 mph) from 25 km/hr (15 mph) is sufficient to reduce the along route doses for the northeast region to less than 3 person-rem per reactor-year.

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[a] 1 person-rem = 0.01 person-Sv.

Table 7.4 Cumulative Doses (person-rem/reactor year)[a] for Incident-Free Transportation of Spent PWR Fuel								
		Peak-Rod Burnup (GWd/MTU)						
Region	**Population**	**43**	**50**	**60**	**62**	**65**	**70**	**75**
SE	Crew	0.33	0.34	0.34	0.34	0.34	0.34	0.34
Turkey Pt.	Along Route	2.48	2.55	2.56	2.56	2.56	2.56	2.57
	Onlookers	1.18	1.21	1.21	1.21	1.22	1.22	1.22
NE	Crew	0.31	0.31	0.32	0.32	0.32	0.32	0.32
Truck	Along Route	2.17	2.22	2.23	2.23	2.23	2.24	2.24
	Onlookers	1.10	1.13	1.13	1.13	1.13	1.13	1.13
NE	Crew	0.12	0.12	0.13	0.13	0.13	0.13	0.12
Rail	Along Route	0.03	0.04	0.04	0.04	0.04	0.04	0.04
	Onlookers	0.01	0.01	0.01	0.01	0.01	0.01	0.01
MW	Crew	0.19	0.19	0.20	0.20	0.20	0.20	0.20
	Along Route	1.03	1.05	1.06	1.06	1.06	1.06	1.06
	Onlookers	0.75	0.77	0.77	0.77	0.77	0.77	0.77
NW	Crew	0.11	0.11	0.11	0.11	0.11	0.11	0.11
	Along Route	0.64	0.66	0.66	0.66	0.66	0.66	0.66
	Onlookers	0.41	0.42	0.42	0.42	0.42	0.42	0.42
SW	Crew	0.04	0.04	0.04	0.04	0.04	0.04	0.04
	Along Route	0.88	0.90	0.91	0.91	0.91	0.91	0.91
	Onlookers	0.12	0.12	0.12	0.12	0.12	0.12	0.12
[a]1 person-rem = 0.01 person-Sv.								

Given the extremely conservative transport speed assumptions in the suburban and urban areas made in this study and the sensitivity of the along route dose estimates to these assumptions, it is reasonable to conclude that for all regions of the United States, the cumulative doses for shipments of 5-year cooled spent fuel are below the Table S-4 values. In addition, the environmental effects of incident-free transportation of spent fuel are nearly independent of burnup. Thus, increasing fuel burnup up to 75 GWd/MTU fuel will not result in a significant increase in the environmental effects of incident-free transportation of spent fuel.

Table 7.5 Cumulative Doses (person-rem/reactor-year) for Incident-Free Transportation of Spent BWR Fuel									
		Peak-Rod Burnup (GWd/MTU)							
Region	**Population**	**35**	**43**	**50**	**60**	**62**	**65**	**70**	**75**
SE Brunswick	Crew	0.18	0.18	0.18	0.18	0.18	0.18	0.18	0.18
	Along Route	2.03	2.08	2.08	2.03	2.03	2.02	2.00	2.00
	Onlookers	1.34	1.38	1.38	1.35	1.34	1.33	1.32	1.32
NE	Crew	0.40	0.41	0.41	0.40	0.40	0.39	0.39	0.39
	Along Route	2.81	2.87	2.88	2.81	2.80	2.79	2.76	2.76
	Onlookers	1.42	1.45	1.46	1.43	1.42	1.41	1.40	1.40
MW	Crew	0.25	0.25	0.25	0.25	0.25	0.24	0.24	0.24
	Along Route	1.33	1.36	1.36	1.33	1.33	1.32	1.31	1.31
	Onlookers	0.97	0.99	1.00	0.97	0.97	0.96	0.96	0.96
NW	Crew	0.14	0.14	0.14	0.14	0.14	0.14	0.14	0.14
	Along Route	0.83	0.85	0.85	0.83	0.83	0.82	0.82	0.82
	Onlookers	0.53	0.54	0.54	0.53	0.53	0.53	0.52	0.52
SW	Crew	0.06	0.06	0.06	0.06	0.06	0.05	0.05	0.05
	Along Route	1.14	1.17	1.17	1.14	1.14	1.13	1.12	1.12
	Onlookers	0.15	0.15	0.15	0.15	0.15	0.15	0.15	0.15

7.2.2 Transportation Accident Impacts

This section discusses the effect of burnup on spent-fuel transportation accident risks. "Accident risks" are defined here as the product of the likelihood of an accident involving a spent-fuel shipment and the consequences of a release of radioactive material resulting from the accident. Increasing burnup affects both the likelihood of an accident and the potential consequences of a release. The likelihood of an accident is directly proportional to the number of fuel shipments. Table 7.1 gives the number of shipments per reactor year as a function of burnup. As long as the number of assemblies per cask remains constant, increasing the burnup decreases the number of shipments per reactor year. However, increasing the burnup increases the activity in each assembly and, as a result, increases the potential consequences of a release. When the activity capacity of a cask, defined by the dose rate, is reached, further fuel burnup will decrease the activity (number of assemblies) in the cask and increase the number of shipments. The result is a decrease in consequences and an increase in the likelihood of an accident.

Accident risks also include a consequence term. Consequences are represented by the population dose from a release of radioactive material given that an accident occurs that leads to a breach in the shipping cask's containment systems. Consequences are a function of the total amount of radioactive material in the shipment, the fraction that escapes from the shipping cask, the transport of radioactive material to humans, and the characteristics of the exposed population.

To estimate the changes in transportation accident risk that result from increasing burnup, RADTRAN 4 calculations were initially performed for a single shipment from each region using radionuclide inventories for a fully loaded cask. The shipping distances and population distribution information for the regions used the evaluation of the impacts of incident-free transportation (see Table 7.2 of this report) were also used here. Representative shipping casks described above were assumed. Accident rates, release fractions, dispersible fractions, and respirable fractions derived from NUREG/CR-4829 (Fischer et al. 1987) were used.. Key parameter values are shown in Appendix G in the first part of the sample RADTRAN output file where the code input is echoed. The calculations for PWR fuel were then repeated with different burnups for all regions to determine the effect of changes in radionuclide inventory on risks. The differences among regions were insignificant; therefore, the BWR calculations were repeated only for the southeast region. Appendix H gives the isotopic composition assumed for the spent fuel as a function of burnup.

Table 7.6 presents accident risks associated with transportation of spent fuel from a reactor in each region to the Clark County, Nevada boundary. Both per-shipment and per-reactor-year risks are included in the table. The risks for other burnups may be determined by applying correction factors based on the activity in each assembly, the number of assemblies in the cask and, when appropriate, the number of shipments per reactor year. Correction factors for burnup are given in Table 7.7. Note that the number given for Assembly Inventory is based on RADTRAN calculations for single shipments assuming four PWR assemblies per cask; it is not the ratio shown in Table 7.1. The per-shipment total correction factors are always less than or equal to 1.00 because the activity in a cask has been constrained to be less than or equal to the activity for which the cask is rated. The per-reactor-year total correction factors are about 1.0 for PWRs and are all less than 1.0 for BWRs.

Based on risk estimates for transportation of 62 GWd/MTU spent fuel in NUREG-1437 Addendum 1, the population risk related to transportation accidents within Clark County are about 0.001 person-rem per shipment, or 0.02 person-rem per reactor-year. These risks, which are similar in magnitude to the risks associated with transportation to Clark County, are in addition to the risks in Table 7.6.

Considering the uncertainties in the data and computational methods, the overall changes in transportation accident risks due to increasing fuel burnup are not significant. Therefore, no increase in environmental effects of spent-fuel transportation accidents are expected as a result of increasing fuel burnup up to 75 GWd/MTU.

Table 7.6 Transportation Accident Risks for Five Regional Shipments for 5-yr Cooled Fuel				
	Accident Risk (person-rem/shipment)		Accident Risk (person-rem/reactor year)	
Region	PWR (43 GWd/MTU)	BWR (35 GWd/MTU)	PWR (43 GWd/MTU)	BWR (35 GWd/MTU)
Southeast	0.0041	0.0020	0.065	0.032
Northeast (Truck)	0.0036	0.0028	0.057	0.044
Northeast (Rail)	0.028		0.035	
Midwest	0.0015	0.0011	0.024	0.017
Northwest	0.0010	0.00079	0.016	0.013
Southwest	0.0016	0.0012	0.025	0.019

[a]1 person-rem = 0.01 person-Sv.

Table 7.7 Correction Factors to Adjust Legal-Weight Truck Transportation Accident Risks for Increased Burnup						
	Burnup (GWd/MTU)					
Risk Factor	**35**	**43**	**50**	**60**	**62**	**75**
PWR – Truck						
Assembly Inventory		1.00	1.16	1.36	1.39	1.64
Assemblies per Cask		1.00	0.75	0.5	0.5	0.5
Per-Shipment Total		1.00	0.87	0.68	0.70	0.82
Shipments per Year		1.00	1.16	1.48	1.43	1.20
Per-Reactor-Year Total		1.00	1.01	1.01	0.99	0.98
PWR– Rail						
Assembly Inventory		1.00	1.16	1.36	1.39	1.64
Assemblies per Cask		1.00	0.88	0.70	0.68	0.58
Per-Shipment Total		1.00	1.02	0.95	0.95	0.95
Shipments per Year		1.00	1.03	1.05	1.06	1.04
Per-Reactor-Year Total		1.00	1.05	1.00	1.01	0.99
BWR						
Assembly Inventory	1.00	1.20	1.36	1.60	1.63	1.91
Assemblies per Cask	1.00	0.78	0.67	0.56	0.56	0.44
Per-Shipment Total	1.00	0.94	0.91	0.89	0.91	0.85
Shipments per Year	1.00	1.06	1.07	1.09	1.05	1.10
Per-Reactor-Year Total	1.00	0.99	0.98	0.97	0.95	0.93

8 Economics

The primary benefit of using extended burnup fuel is a reduction in the mass of fuel required per unit of electricity generated. The economic impact of using extended burnup fuel (up to a point) is the reduction in overall fuel cycle costs due to the reduction in required fuel. The reduction in required fuel affects both front-end and back-end requirements and costs. However, at some point, the increase in extended burnup is expected to produce diminishing returns or increasing costs due to additional requirements for enrichment services, processing, and back-end services (storage and disposal). Burnup can be extended to 75 GWd/MTU without further enrichment.

A principal finding in NUREG/CR-5009 (Baker et al. 1988), which evaluated the effects of extending peak-rod burnups from 33 GWd/MTU to 60 GWd/MTU, was that there would be an expected net-discounted cost savings on the order of $2 billion (in 1985 dollars). This savings was largely attributable to savings in fuel production costs and other front-end activities of the fuel cycle. The current focus looks at the expected economic benefit of increasing fuel burnup to 75 GWd/MTU.

The economic analysis conducted for NUREG/CR-5009 was based on the review of several key published materials. From those sources, a reasonably clear picture was developed of the likely effects of increasing the fuel burnup from 33 GWd/MTU to 60 GWd/MTU. However, three main issues were encountered while performing the evaluation for NUREG/CR-5009: a lack of detailed information related to fuel production; cost issues stemming from a mixture of discounted and non-discounted sums and cash flows; and changes in the U.S. Department of Energy, Energy Information Administration (DOE/EIA) middle growth forecast for nuclear generating capacity used by different sources. Further, inconsistencies in the assumptions and data from the various sources led to the use of a number of simplifications and adjustments and ultimately to a broad range of possible impacts to particular components of the fuel-cycle costs.

The total discounted fuel-cycle cost savings attributable to the implementation of extended burnup from 33 GWd/MTU to 60 GWd/MTU was determined to range from $1.98 to $2.68 billion (1985 dollars). These savings were dominated by the estimated savings in front-end requirements, which accounted for over 90% of the total savings. The cost impact on the back-end of the fuel cycle was estimated to be a much less significant factor, yet carried the most uncertainty. Savings estimates for back-end services ranged from -$614 to $214 million, with the largest source of uncertainty involving future repository storage costs.

The brief sensitivity analysis in NUREG/CR-5009 indicated that the estimated economic effects of implementing extended burnup are based on a wide set of variable conditions. Front-end savings are directly linked to assumed price levels, particularly the price of uranium and, to a lesser extent, the price of enrichment services. At the back end of the fuel cycle, the most important factor was determined to be the strict requirements for repository design and operation. A significant reduction in back-end costs would occur if these requirements and accompanying costs could be relaxed by aging the spent fuel before disposal or altering repository specifications. Fuel aging remains a viable option, but alteration of repository specifications is not likely in the current political climate. Finally, NUREG/CR-5009 identified the discount rate as another key variable determining the magnitude of the expected savings. Since NUREG/CR-5009 was written in 1988, the real discount rate required by the U.S. Office of Management and Budget (OMB) in the evaluation of time distributed costs and benefits by agencies of the executive branch of the federal government has fallen from 10 percent to less than 3 percent (OMB 1999). This reduction in discount rate increases the economic viability (value) of extended burnup.

The main economic conclusion of NUREG/CR-5009 was that while uncertainties exist that may affect the magnitude of the potential savings, increasing the burnup of nuclear fuel from 33 GWd/MTU to 60 GWd/MTU was expected to provide a substantial cost savings. This study continues the analysis by examining the potential impact of further increasing the allowable burnup level to 75 GWd/MTU.

The scope of this study is limited to the direct costs of electricity production, and no attempt is made to estimate the indirect costs or benefits by imputing dollar values to such factors as changes in radionuclide inventory, accident characteristics, or risk. Where indirect effects are potentially large, it is anticipated that changes is power-generating operations will be instituted to minimize the effects. Therefore, the costs of the indirect effects will be reflected explicitly in direct costs. This study, which picks up where NUREG/CR-5009 left off, consists of the review of materials published since 1988.

Since NUREG/CR-5009 was published over a decade ago, much of the thrust to verify the economic benefits of extending fuel burnups has been satisfied; since then, few reports have been published on the matter. Only a few reports focus on the detailed economic impact of extended burnup fuel. They form the basis for updating the information in NUREG/CR-5009.

As discussed in NUREG/CR-5009, the nationwide aggregate economic effects of using extended burnup nuclear fuel depend on the amount of electricity generated by nuclear power and the proportion of that amount that is generated with extended burnup fuel. The outlook for U.S. nuclear capacity has changed significantly since 1988. The projected (reference case) capacity for the year 2015 has been reduced from 216 gigawatts electric (GWe) to 63 GWe (DOE/EIA 1997). The maximum capacity expected for 2015 is the recent capacity of 101 GWe. However, actual capacity is expected to decline as plants reach the end of their operating licenses and are shut down or are shut down for other reasons. The increase in capacity that was expected has not materialized, and no new plant construction is anticipated in the time frame of the projection. While this will not impact the cost effectiveness of increasing fuel burnup rates, it will have a significant impact on the overall aggregate savings.

In 1990-1991, a detailed economic evaluation of increasing the average burnup of uranium fuel to 55-65 GWd/MTU was undertaken by the International Atomic Energy Agency (IAEA 1992). The study, in which eight nations collaborated, was called the Water Reactor Extended Burn-up Study (WREBUS). It analyzed the expected impact of higher fuel burnup under several different technical boundary conditions and economic scenarios, as well as fuel-cycle cost sensitivities.

The results indicate that for all scenarios but one, fuel-cycle costs continue to decrease as the level of fuel burnup increases over the range of burnups studied. The one exception involved a high discount rate of 10 percent, which neutralized cost savings and then increased costs at the highest burnup levels. However, as Lang explains (IAEA 1992, p. 47), no minimum savings from extending burnup may exist for back-end costs equaling or exceeding front-end costs, due to the nature of the relative timing of front- and back-end charges and the effect of discounting.

The study found that back-end costs are the dominant factor in determining the economic benefit of higher burnup fuels, followed in order of decreasing impact by discount rate, uranium costs, and enrichment and fabrication costs. Increasing burnup to 55 to 65 GWd/MTU was determined to have the potential to reduce uranium consumption by approximately 15 percent (IAEA 1992, p. 45). Longer cycle lengths tend to shift the economic optimum to higher burnup values.

Another study carried out by the Nuclear Energy Agency Organization for Economic Co-operation and Development (OECD) (NEA-OECD 1994) in 1994 analyzed the savings from extended burnup on the same set of technical scenarios used in the WREBUS study, using updated reference prices. The results indicated that there is no optimum burnup level and that fuel-cycle costs continue to decrease as burnup extends through the range evaluated (40-60 GWd/MTU, batch average). Accordingly, from an economic standpoint it can be inferred that it is desirable to proceed to the highest level of burnup technically possible.

The OECD study also examined the sensitivities associated with the various segments of the fuel cycle. Similar to previous studies, it found that the parameters having the greatest influence on the burnup economics are the price of back-end services and the discount rate. The study also concluded that the savings from increasing burnup levels may not be as great in circumstances where storage and disposal costs are tied to the overall fuel quantity to be managed.

The most current study is presented in a 1999 draft report from Duke Power (Duke 1999). The purpose of the study was to determine the batch discharge burnup that would minimize fuel-cycle costs for the Duke reactors and examine the technical difficulties associated with achieving such burnups. Batch-average burnups in the range of 40-80 GWd/MTU were analyzed for two different plant designs and a range of economic parameters and price projections. In each of the eight scenarios run for each plant design, they found that the optimum batch-average burnup fell in the range of 60-70 GWd/MTU (peak-rod burnups of approximately 80 GWd/MTU.

Eighteen-month fuel cycles used by about two thirds of the nation's reactors were assumed by Duke. The remaining reactors are on 24-month cycles. The optimum burnup increases with increasing cycle length. Duke estimated that fuel batch costs at its Oconee plant can be reduced by some $1 million-$4 million (depending on market conditions) by increasing the equilibrium batch-average fuel burnup, relative to the current design and operation. Similarly, savings on the order of $1.5 million-$5 million could be expected per fuel batch at the Catawba and McGuire plants. The largest impact on the potential benefit of extending burnup resulted from varying the costs associated with spent-fuel assembly storage and disposal. These costs consistently hold the most uncertainty and largest impact on the overall savings associated with increasing burnup.

Baker et al. (1988) determined that favorable savings estimates exist for extended burnup from 33-60 GWd/MTU; there is no reason to believe that the principal conclusions would be substantially different for a further moderate extension of burnup. Furthermore, in each of the three more recent studies, increases in peak-rod burnup well beyond 60 GWd/MTU were considered and found to result in fuel-cycle savings. In the OECD study it was found that the economic benefits continue to accrue as batch-average burnup is extended through 60 GWd/MTU (requiring peak-rod burnups of the order of 70+ GWd/MTU). In the Duke Power analysis, minimum costs were found to correlate with batch-average burnups in the range of 60-70 GWd/MTU (corresponding to peak-rod burnups of 80 GWd/MTU or more). It can therefore be concluded that savings will be realized from increasing the burnup to 75 GWd/MTU.

From an economic standpoint, there appear to be no additional barriers to the implementation of extended burnups to 75 GWd/MTU. However, the total aggregate savings expected as a result of increasing burnup are smaller than earlier estimates because the nuclear power industry has not expanded as projected.

9 Summary and Conclusions

The preceding eight chapters have (1) described the effects of increasing fuel burnup on radionuclide inventories in the fuel and the fraction of the inventories released to the gap, (2) compared the radionuclide inventories and gap-release fractions associated with extended burnup fuel with current regulations, guidance, and other publications, (3) considered the effects of increasing fuel burnup on the environmental impacts of normal operations, including transportation of spent fuel, and (4) considered the economics of increasing fuel burnup. Findings from those chapters are summarized below.

Increasing fuel burnup above 60 GWd/MTU

- changes the radionuclide mix in reactor fuel. The activities of short-lived fission products tend to remain constant or decrease slightly, while activities associated with activation products and actinides tend to increase with increasing burnup.

- tends to increase the gap-release fraction. Gap-release fractions for 60 GWd/MTU burnup calculated for this study are generally about a factor of two larger than those given in NUREG/CR-5009. Most of this change is associated with changes in computer codes used to calculate gap-release fractions. Gap-release fractions at 62 GWd/MTU remain below fractions assumed in current guidance.

- reduces the requirement for mining and processing uranium ore and fabrication of fuel assemblies. As a result, increasing fuel burnup should decrease environmental consequences associated with the front end of the fuel cycle.

- is not likely to increase environmental impacts from normal reactor operations because coolant activity has been decreasing as fuel burnup has been increasing. This decrease is attributed to reduction in the number of fuel failures as a result of better quality control in fuel fabrication.

- decreases the annual discharge of fuel to the spent-fuel pool. This decrease preserves space in the spent-fuel pool and postpones the need to remove spent fuel from the pool. It also reduces the total heat load (watts per reactor year) on the pool cooling system.

- will not change limits on coolant activity. Consequently, the potential environmental impacts of postulated LOCAs, PWR steam generator tube rupture accidents, and BWR main steam line break accidents are unchanged.

- increases the potential environmental impacts from a fuel-handling accident. However, the doses calculated for fuel-handling accidents remain well below regulatory limits.

- does not significantly change the potential environmental impacts of incident-free transportation of spent nuclear fuel or the accident risks associated with spent-fuel transportation if the fuel is cooled for five years after discharge from the reactor. Doses associated with incident-free transportation of spent fuel with burnup to 75 GWd/MTU are bounded by the doses given in 10 CFR 51.52, Table S-4 for all regions of the country if dose rates from the shipping casks are maintained within regulatory limits.

- has an appreciable economic benefit.

The findings summarized above indicate that there are no significant adverse environmental impacts associated with extending peak-rod fuel burnup to 62 GWd/MTU. The factor limiting this conclusion to 62 GWd/MTU is uncertainty in changes in the gap-release fraction associated with increasing fuel burnup. Although it was possible to evaluate environmental impacts of other aspects of burnup to a peak-rod burnup of 75 GWd/MTU, the present methods for assessing fission gas releases have not been validated with actual data at peak-rod burnups greater than 62 GWd/MTU. This limitation will be re-evaluated as the methods for assessing fission gas releases are validated with data for higher burnups. For those aspects of this assessment in which the environmental impacts are not significantly affected by fission gas releases, the findings summarized above indicate that there are no significant adverse environmental impacts associated with extending peak-rod fuel burnup to 75 GWd/MTU.

10 References

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10 CFR 50.36a, "Technical specifications on effluents from nuclear power reactors."

10 CFR 51.52, Table S-4, "Environmental Impact of Transportation of Fuel and Waste To and From One Light-Water-Cooled Nuclear Power Reactor."

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Appendix A

U.S. Reactor Characteristics

Appendix A

U.S. Reactor Characteristics

This Appendix lists reactor fuel statistics for U.S. nuclear reactors used in the evaluation of the environmental impacts of increasing fuel burnup in this report. The statistics were compiled from Plant Information Books prepared and maintained by the U.S. Nuclear Regulatory Commission[a] and supplemented by information from the 1997 World Nuclear Industry Handbook (NEI 1997). Table A.1 contains statistics for pressurized-water reactors (PWRs), and Table A.2 contains statistics for boiling-water reactors (BWRs).

Table A.1 PWR Reactor Fuel Statistics

	Unit	OL	Type	Thermal Power (MW)	MTU	MTU/MW	Ass.	Rods	MTU/Rod	MTU/Ass.
Oconee	1	1973	B&W	2568	94.1	0.0366	177	208	2.56E-03	0.532
Oconee	2	1973	B&W	2568	94.1	0.0366	177	208	2.56E-03	0.532
Arkansas	1	1974	B&W	2568	82	0.0319	177	208	2.23E-03	0.463
Three Mile Island	1	1974	B&W	2568	82.1	0.0320	177	208	2.23E-03	0.464
Oconee	3	1974	B&W	2568	94.1	0.0366	177	208	2.56E-03	0.532
Davis Besse	1	1977	B&W	2772	82.9	0.0299	177	208	2.25E-03	0.468
Crystal River	3	1977	B&W	2544	82	0.0322	177	208	2.23E-03	0.463
Palisades	1	1972	CE	2530	81.43	0.0322	204	216	1.85E-03	0.399
Fort Calhoun	1	1973	CE	1500	47.9	0.0319	133	176	2.05E-03	0.360
Maine Yankee	1	1973	CE	2700	81	0.0300	217	176	2.12E-03	0.373
Calvert Cliffs	1	1974	CE	2700	82.5	0.0306	217	176	2.16E-03	0.380
Millstone	2	1975	CE	2700	87.9	0.0326	217	176	2.30E-03	0.405
Calvert Cliffs	2	1976	CE	2700	82.5	0.0306	217	176	2.16E-03	0.380
St Lucie	1	1976	CE	2700	94	0.0348	217	176	2.46E-03	0.433
Arkansas	2	1978	CE	2815	73.6	0.0261	177	236	1.76E-03	0.416
San Onofre	2	1982	CE	3390	89.5	0.0264	217	236	1.75E-03	0.412
St Lucie	2	1983	CE	2700	92.7	0.0343	217	236	1.81E-03	0.427
San Onofre	3	1983	CE	3390	89.5	0.0264	217	236	1.75E-03	0.412
Waterford	3	1985	CE	3390	89.5	0.0264	217	236	1.75E-03	0.412

[a]U.S. Nuclear Regulatory Commission at http://www.nrc.gov/AEOD/pib/disclaimer.html (October 25, 2000).

Table A.1 (continued)										
	Unit	OL	Type	Thermal Power (MW)	MTU	MTU/MW	Ass.	Rods	MTU/Rod	MTU/Ass.
Palo Verde	1	1985	CE	3800	99.03	0.0261	241	236	1.74E-03	0.411
Palo Verde	2	1986	CE	3800	99.03	0.0261	241	236	1.74E-03	0.411
Palo Verde	3	1987	CE	3800	99.03	0.0261	241	236	1.74E-03	0.411
Haddam Neck	1	1967	W	1825	59.8	0.0328	157	204	1.87E-03	0.381
R. E. Ginna	1	1969	W	1520	42.61	0.0280	121	179	1.97E-03	0.352
Point Beach	1	1970	W	1518	50	0.0329	121	179	2.31E-03	0.413
H. B. Robinson	2	1970	W	2300	70	0.0304	157	204	2.19E-03	0.446
Point Beach	2	1972	W	1518	50	0.0329	121	179	2.31E-03	0.413
Surry	1	1972	W	2441	72.38	0.0297	157	204	2.26E-03	0.461
Turkey Point	3	1972	W	2200	79.8	0.0363	157	204	2.49E-03	0.508
Prairie Island	1	1973	W	1650	42.9	0.0260	121	179	1.98E-03	0.355
Turkey Point	4	1973	W	2200	79.8	0.0363	157	204	2.49E-03	0.508
Zion	1	1973	W	3250	87.7	0.0270	193	204	2.23E-03	0.454
Kewaunee	1	1973	W	1650	46.1	0.0279	121	179	2.13E-03	0.381
Surry	2	1973	W	2441	72.41	0.0297	157	204	2.26E-03	0.461
Indian Point	2	1973	W	3071	88.9	0.0289	193	204	2.26E-03	0.461
Zion	2	1973	W	3250	87.7	0.0270	193	204	2.23E-03	0.454
Prairie Island	2	1974	W	1650	42.9	0.0260	121	179	1.98E-03	0.355
D. C. Cook	1	1974	W	3250	88.6	0.0273	193	204	2.25E-03	0.459
Salem	1	1976	W	3411	89.1	0.0261	193	264	1.75E-03	0.462
Beaver Valley	1	1976	W	2652	72.82	0.0275	157	264	1.76E-03	0.464
Indian Point	3	1976	W	3025	89	0.0294	193	204	2.26E-03	0.461
Joseph M Farley	1	1977	W	2652	72.8	0.0275	157	264	1.76E-03	0.464
D. C. Cook	2	1977	W	3250	81	0.0249	193	264	1.59E-03	0.420
North Anna	1	1978	W	2893	72.5	0.0251	157	264	1.75E-03	0.462
Sequoyah	1	1980	W	3411	89.27	0.0262	193	264	1.75E-03	0.463
North Anna	2	1980	W	2893	72.5	0.0251	157	264	1.75E-03	0.462
Sequoyah	2	1981	W	3411	89.27	0.0262	193	264	1.75E-03	0.463
Joseph M Farley	2	1981	W	2652	72.8	0.0275	157	264	1.76E-03	0.464
Salem	2	1981	W	3411	89.1	0.0261	193	264	1.75E-03	0.462
W. B. McGuire	1	1981	W	3411	89	0.0261	193	264	1.75E-03	0.461
V. C. Summer	1	1982	W	2775	65.42	0.0236	157	264	1.58E-03	0.417
W. B. McGuire	2	1983	W	3411	89	0.0261	193	264	1.75E-03	0.461
Callaway	1	1984	W	3565	82	0.0230	193	264	1.61E-03	0.425
Diablo Canyon	1	1984	W	3338	82	0.0246	193	264	1.61E-03	0.425
Catawba	1	1985	W	3411	82	0.0240	193	264	1.61E-03	0.425
Byron	1	1985	W	3411	101	0.0296	193	264	1.98E-03	0.523
Wolf Creek	1	1985	W	3411	89.4	0.0262	193	264	1.75E-03	0.463

Table A.1 (continued)										
	Unit	OL	Type	Thermal Power (MW)	MTU	MTU/MW	Ass.	Rods	MTU/Rod	MTU/Ass.
Diablo Canyon	2	1985	W	3411	82	0.0240	193	264	1.61E-03	0.425
Millstone	3	1986	W	3411	86.57	0.0254	193	264	1.70E-03	0.449
Catawba	2	1986	W	3411	82	0.0240	193	264	1.61E-03	0.425
Beaver Valley	2	1987	W	2652	72.41	0.0273	157	264	1.75E-03	0.461
Byron	2	1987	W	3411	101	0.0296	193	264	1.98E-03	0.523
Shearon Harris	1	1987	W	2775	73.1	0.0263	157	264	1.76E-03	0.466
Braidwood	1	1987	W	3411	101	0.0296	193	264	1.98E-03	0.523
Vogtle	1	1987	W	3565	111.4	0.0312	193	264	2.19E-03	0.577
Braidwood	2	1988	W	3411	101	0.0296	193	264	1.98E-03	0.523
South Texas	1	1988	W	3800	117.82	0.0310	193	264	2.31E-03	0.610
Vogtle	2	1989	W	3565	111.4	0.0312	193	264	2.19E-03	0.577
Comanche Peak	1	1989	W	3341	84.5	0.0253	193	264	1.66E-03	0.438
South Texas	2	1989	W	3800	117.82	0.0310	193	264	2.31E-03	0.610
Seabrook	1	1990	W	3411	101	0.0296	193	264	1.98E-03	0.523
Comanche Peak	2	1993	W	3341	81.85	0.0245	193	264	1.61E-03	0.424
Watts Bar	1	1996	W	3411	88.6	0.0260	193	264	1.74E-03	0.459

All PWRs	**N**		**73**	**73**	**73**			**73**	**73**
	Average		2918	82.8	0.0287			1.98E-03	0.452
	Median		3025	82.9	0.0275			1.98E-03	0.459
	Std Dev		627	16.4	0.0035			2.85E-04	0.056
	Maximum		3800	117.8	0.0366			2.56E-03	0.610
	Minimum		1500	42.6	0.0230			1.58E-03	0.352

Table A.2 BWR Reactor Fuel Statistics										
	Unit	OL	Type	Thermal Power (MTU)	MTU	MTU/MW	Assem.	Rods	MTU/Rod	MTU/Ass.
Big Rock Point	1	1962	GE	240	12.21	0.0509	86	117	1.21E-03	0.142
Nine Mile Point	1	1968	GE	1850	93.6	0.0506	532	62	2.84E-03	0.176
Dresden	2	1969	GE	2527	150	0.0594	724	81	2.56E-03	0.207
Oyster Creek	1	1969	GE	1930	125	0.0648	560	60	3.72E-03	0.223
Millstone	1	1970	GE	2011	103	0.0512	580	62	2.86E-03	0.178
Monticello	1	1970	GE	1670	91.1	0.0546	484	62	3.04E-03	0.188
Dresden	3	1971	GE	2527	150	0.0594	724	81	2.56E-03	0.207
Quad Cities	1	1972	GE	2511	150	0.0597	724	64	3.24E-03	0.207
Pilgrim	1	1972	GE	1998	113	0.0566	580	62	3.14E-03	0.195
Quad Cities	2	1972	GE	2511	150	0.0597	724	64	3.24E-03	0.207
Peach Bottom	2	1973	GE	3458	134.2	0.0388	764	63	2.79E-03	0.176
Vermont Yankee	1	1973	GE	1593	65.4	0.0411	368	63	2.82E-03	0.178
Browns Ferry	1	1973	GE	3293	154	0.0468	764	62	3.25E-03	0.202
Peach Bottom	3	1974	GE	3458	137.8	0.0398	764	63	2.86E-03	0.180
Fitz Patrick	1	1974	GE	2436	100.82	0.0414	560	62	2.90E-03	0.180
Brunswick	2	1974	GE	2436	102.7	0.0422	560	62	2.96E-03	0.183
Browns Ferry	2	1974	GE	3293	149.9	0.0455	764	62	3.16E-03	0.196
Duane Arnold	1	1974	GE	1658	67.4	0.0407	368	60	3.05E-03	0.183
E. I. Hatch	1	1974	GE	2436	114	0.0468	560	62	3.28E-03	0.204
Cooper	1	1974	GE	2381	101.2	0.0425	548	62	2.98E-03	0.185
Brunswick	1	1976	GE	2436	102.7	0.0422	560	62	2.96E-03	0.183
Browns Ferry	3	1976	GE	3293	149	0.0452	764	62	3.15E-03	0.195
E. I. Hatch	2	1978	GE	2436	103.4	0.0424	560	62	2.98E-03	0.185
Susquehanna	1	1982	GE	3293	133.2	0.0404	764	62	2.81E-03	0.174
LaSalle	1	1982	GE	3323	158	0.0475	764	64	3.23E-03	0.207
LaSalle	2	1984	GE	3323	158	0.0475	764	64	3.23E-03	0.207
Grand Gulf	1	1984	GE	3833	144.5	0.0377	800	62	2.91E-03	0.181
WNP	1	1984	GE	3323	128	0.0385	764	72	2.33E-03	0.168
Susquehanna	2	1984	GE	3441	132.3	0.0384	764	62	2.79E-03	0.173
Enrico Fermi	2	1985	GE	3430	140.3	0.0409	764	62	2.96E-03	0.184
Limerick	1	1985	GE	3458	135.52	0.0392	764	62	2.86E-03	0.177
River Bend	1	1985	GE	2894	112.6	0.0389	624	64	2.82E-03	0.180
Perry	1	1986	GE	3679	152.1	0.0413	748	62	3.28E-03	0.203
Hope Creek	1	1986	GE	3293	139.5	0.0424	764	60	3.04E-03	0.183
Clinton	1	1987	GE	2894	115.4	0.0399	624	62	2.98E-03	0.185
Nine Mile Point	2	1987	GE	3323	141	0.0424	764	62	2.98E-03	0.185
Limerick	2	1990	GE	3458	140.1	0.0405	764	62	2.96E-03	0.183

Table A.2 (continued)

	Unit	OL	Type	Thermal Power (MTU)	MTU	MTU/MW	Assem.	Rods	MTU/Rod	MTU/Ass.
	N			37	37	37			37	37
	Average			2739	123	0.0459			2.94E-03	0.188
	Median			2894	133	0.0424			2.96E-03	0.184
	Std Dev			768	31	0.0075			3.81E-04	0.0015
	Maximum			3833	158	0.0648			3.72E-03	0.223
	Minimum			240	12.21	0.0377			1.21E-03	0.142

A.1 Reference

Nuclear Engineering International (NEI). 1997. *1997 World Nuclear Industry Handbook*, Wilmington Business Publishing, Wilmington, United Kingdom.

Appendix B

Peak-Rod Power Histories

Appendix B

Peak-Rod Power Histories

This appendix lists the peak-rod power histories assumed in the ORIGEN-ARP calculation of the fuel radionuclide inventories used in evaluation of the potential environmental impacts of the fuel-handling accident. The PWR and BWR fuel designs modeled for these calculations were 15 x 15 and 8 x 8 fuel design arrays, respectively. These power histories are best-estimate, rod-average power histories for the peak rod in a core and were only for use in evaluating the differences in inventory due to burnup. In the tables, the rod-average power is the average power since the previous time entry.

Table B.1 PWR Peak-Rod Power and Burnup as a Function of Time

Elapsed Time (days)	Rod-Average Power (MW/MTU)	Rod-Average Burnup (MWd/MTU)
415.38	52.96	22,000
453.14	52.96	24,000
472.02	52.96	25,000
528.67	52.96	28,000
610.76	48.73	32,000
694.06	48.02	36,000
778.60	47.31	40,000
821.84	46.25	42,000
843.45	46.25	43,000
908.81	45.90	46,000
930.60	45.90	47,000
952.72	45.20	48,000
996.98	45.20	50,000
1,041.93	44.49	52,000
1,264.06	36.02	60,000
1,319.59	36.02	62,000
1,402.89	36.02	65,000
1,541.72	36.02	70,000
1,680.55	36.02	75,000

Table B.2 BWR Peak-Rod Power and Burnup as a Function of Time		
Elapsed Time (days)	Rod-Average Power (MW/MTU)	Rod-Average Burnup (MWd/MTU)
1.20	10.80	13
5.50	13.90	73
6.90	18.35	98
8.10	28.09	132
10.30	32.29	203
12.40	34.25	275
15.20	38.20	382
50.30	41.99	1,856
90.00	41.99	3,523
130.00	41.99	5,203
170.00	41.99	6,883
210.00	41.99	8,562
250.00	41.99	10,242
290.00	41.34	11,896
332.00	40.69	13,605
368.00	40.04	15,046
404.00	39.38	16,464
443.00	38.73	17,974
486.00	38.08	19,612
526.00	38.08	21,135
549.12	37.43	22,000
573.00	37.43	22,894
603.08	36.77	24,000
622.00	36.77	24,696
630.42	36.12	25,000
673.00	36.12	26,538
722.00	35.47	28,276
829.00	34.82	32,002
873.00	34.17	33,505
880.00	33.51	33,739
921.00	32.86	35,087
941.00	32.86	35,744
973.00	32.21	36,775

Table B.2 (continued)		
Elapsed Time (days)	Rod-Average Power (MW/MTU)	Rod-Average Burnup (MWd/MTU)
999.00	32.21	37,612
1,060.00	31.56	39,537
1,121.00	31.27	41,444
1,139.05	30.78	42,000
1,171.54	30.78	43,000
1,181.00	30.78	43,291
1,243.00	29.84	45,142
1,272.74	28.87	46,000
1,309.00	28.87	47,047
1,343.13	27.93	48,000
1,375.00	27.93	48,890
1,416.19	26.95	50,000
1,437.00	26.95	50,561
1,510.00	26.01	52,460
1,590.00	25.07	54,465
1,670.00	23.16	56,318
1,750.00	22.22	58,096
1,835.71	22.22	60,000
1,925.72	22.22	62,000
2,060.73	22.22	65,000
2,285.76	22.22	70,000
2,510.79	22.22	75,000

Appendix C

Spent-Fuel Decay Heat

Appendix C

Spent-Fuel Decay Heat

This appendix presents the spent-fuel decay heat in watts per metric ton of uranium (w/MTU) as function of decay time and burnup for PWR spent fuel. The decay heats were calculated by ORIGEN-ARP. In addition to listing the total decay heat, the table lists the source—activation products, actinides, and fission products—of the decay heat. Decay heat for BWR spent fuel should be slightly lower than the heat shown in the table.

Table C.1 PWR Spent-Fuel Decay Heat

Decay Time (yr)	Heat Source	Peak-Rod Burnup (GWd/MTU)						
		43	50	60	62	65	70	75
0	Activation	2.02E+03	2.15E+03	2.32E+03	2.38E+03	2.44E+03	2.53E+03	2.62E+03
	Actinides	4.29E+04	4.66E+04	5.12E+04	5.26E+04	5.42E+04	5.65E+04	5.87E+04
	Fission	1.66E+06	1.64E+06	1.63E+06	1.61E+06	1.61E+06	1.60E+06	1.59E+06
	Total	1.70E+06	1.69E+06	1.68E+06	1.67E+06	1.67E+06	1.66E+06	1.65E+06
1	Activation	1.41E+02	1.60E+02	1.87E+02	1.93E+02	2.01E+02	2.14E+02	2.27E+02
	Actinides	7.19E+02	1.03E+03	1.56E+03	1.67E+03	1.85E+03	2.16E+03	2.50E+03
	Fission	9.17E+03	1.00E+04	1.11E+04	1.14E+04	1.17E+04	1.22E+04	1.28E+04
	Total	1.00E+04	1.12E+04	1.28E+04	1.33E+04	1.38E+04	1.46E+04	1.55E+04
2	Activation	1.11E+02	1.27E+02	1.50E+02	1.55E+02	1.61E+02	1.72E+02	1.83E+02
	Actinides	3.88E+02	5.80E+02	9.47E+02	1.03E+03	1.17E+03	1.42E+03	1.71E+03
	Fission	5.04E+03	5.64E+03	6.47E+03	6.63E+03	6.87E+03	7.26E+03	7.65E+03
	Total	5.54E+03	6.35E+03	7.57E+03	7.82E+03	8.20E+03	8.85E+03	9.54E+03
3	Activation	9.60E+01	1.10E+02	1.30E+02	1.34E+02	1.40E+02	1.49E+02	1.59E+02
	Actinides	3.20E+02	4.86E+02	8.10E+02	8.86E+02	1.01E+03	1.24E+03	1.51E+03
	Fission	3.27E+03	3.72E+03	4.34E+03	4.46E+03	4.64E+03	4.93E+03	5.23E+03
	Total	3.69E+03	4.32E+03	5.28E+03	5.48E+03	5.79E+03	6.32E+03	6.90E+03
5	Activation	7.30E+01	8.39E+01	9.89E+01	1.02E+02	1.06E+02	1.14E+02	1.21E+02
	Actinides	3.06E+02	4.59E+02	7.60E+02	8.30E+02	9.48E+02	1.16E+03	1.42E+03
	Fission	1.90E+03	2.20E+03	2.60E+03	2.67E+03	2.79E+03	2.98E+03	3.17E+03
	Total	2.28E+03	2.74E+03	3.46E+03	3.60E+03	3.84E+03	4.26E+03	4.71E+03
7	Activation	5.59E+01	6.42E+01	7.57E+01	7.80E+01	8.15E+01	8.71E+01	9.27E+01
	Actinides	3.09E+02	4.56E+02	7.43E+02	8.08E+02	9.18E+02	1.12E+03	1.36E+03
	Fission	1.46E+03	1.68E+03	1.98E+03	2.04E+03	2.13E+03	2.27E+03	2.41E+03
	Total	1.82E+03	2.20E+03	2.80E+03	2.93E+03	3.13E+03	3.48E+03	3.86E+03

Table C.1 (continued)								
Decay Time (yr)	Heat Source	Peak-Rod Burnup (GWd/MTU)						
		43	50	60	62	65	70	75
10	Activation	3.76E+01	4.31E+01	5.09E+01	5.25E+01	5.48E+01	5.86E+01	6.23E+01
	Actinides	3.13E+02	4.53E+02	7.18E+02	7.79E+02	8.81E+02	1.07E+03	1.28E+03
	Fission	1.20E+03	1.37E+03	1.60E+03	1.65E+03	1.71E+03	1.82E+03	1.92E+03
	Total	1.55E+03	1.87E+03	2.37E+03	2.48E+03	2.65E+03	2.95E+03	3.27E+03
15	Activation	1.94E+01	2.23E+01	2.63E+01	2.71E+01	2.83E+01	3.03E+01	3.23E+01
	Actinides	3.17E+02	4.46E+02	6.81E+02	7.34E+02	8.24E+02	9.86E+02	1.17E+03
	Fission	1.00E+03	1.14E+03	1.32E+03	1.36E+03	1.41E+03	1.49E+03	1.57E+03
	Total	1.34E+03	1.61E+03	2.03E+03	2.12E+03	2.26E+03	2.51E+03	2.77E+03
20	Activation	1.01E+01	1.16E+01	1.37E+01	1.41E+01	1.47E+01	1.57E+01	1.67E+01
	Actinides	3.18E+02	4.37E+02	6.49E+02	6.95E+02	7.73E+02	9.16E+02	1.08E+03
	Fission	8.72E+02	9.90E+02	1.15E+03	1.18E+03	1.22E+03	1.29E+03	1.36E+03
	Total	1.20E+03	1.44E+03	1.81E+03	1.89E+03	2.01E+03	2.22E+03	2.45E+03
30	Activation	2.75E+00	3.16E+00	3.73E+00	3.85E+00	4.02E+00	4.30E+00	4.58E+00
	Actinides	3.15E+02	4.20E+02	5.90E+02	6.27E+02	6.89E+02	8.00E+02	9.23E+02
	Fission	6.78E+02	7.69E+02	8.91E+02	9.14E+02	9.48E+02	1.00E+03	1.06E+03
	Total	9.96E+02	1.19E+03	1.49E+03	1.54E+03	1.64E+03	1.80E+03	1.99E+03

Appendix D

Gap-Release Fractions

Appendix D

Gap-Release Fractions

The gap-release fractions presented in Tables 2.3 and 2.4 are best-estimate values for the peak rod at the end of a fuel cycle. The gap-release fractions for stable gases (long-lived) noble gases (stable Xe) were calculated for Westinghouse 17 x 17 and GE 8 x 8 fuel assemblies using the Massih model (Forsberg and Massih 1985; Lanning 1997a) using typical peak-rod power histories (see Appendix B). The gap-release fractions of other radionuclides, relative to the stable Xe release fraction, were calculated using the ANS 5.4 model (ANS 1982), also using typical power histories. These relative gap-release fractions are conservative because the ANS 5.4 release model was developed to give conservative gap-release fraction estimates. The relative gap-release fractions are functions of fuel design and power history. Variation in the relative gap-release fractions of 20 to 30 percent is possible with changes in fuel design and power history.

The gap-release fractions calculated for this study are lower than gap-release fractions traditionally calculated for licensing applications because the environmental analyses are generally based on more realistic assumptions than licensing calculations. Table D.1 provides an example of the difference in gap-release fraction estimated using realistic assumptions and gap-release fractions estimated using typical licensing assumptions. The table compares the peak-rod gap-release fraction for stable Xe in 62 GWd/MTU fuel calculated for this study with gap-release fractions calculated using licensing assumptions for two different PWR fuel designs. The Massih "best estimate" values were used in this study, and the ANS 5.4 and Massih 95% Upper Bound values are licensing values.

Table D.1 Peak-Rod Gap-Release Fractions for Stable Xe in 62 GWd/MTU Fuel

	Release Models in FRAPCON-3		
	ANS 5.4	Massih	
Vendor Fuel Design[a]	(Conservative)	Best Estimate	95% Upper Bound
Mark B (15 x 15)	0.28	0.11	0.205
Westinghouse (17 x 17)	NC[b]	0.079	0.174

[a]Other fuel design may give higher release fractions.
[b]Not calculated.

The relative gap-release fractions for noble gases and volatile radionuclides, calculated using the ANS 5.4 code, are listed in Table D.2. To estimate the gap-release fraction for a specific radionuclide, multiply the stable Xe gap-release fraction by the relative gap-release fraction for the radionuclide. For example, the best-estimate PWR peak-rod gap-release fraction for ^{131}I in 62 GWd/MTU fuel is 0.079 x 0.86 = 0.0679.

Table D.2 Relative Gap-Release Fractions

Isotope	PWR		BWR	
	Core-Average	Peak-Rod	Core-Average	Peak-Rod
Kr-85	1.00	1.00	1.00	1.00
Kr-87	0.0204	0.050	0.020	0.050
Kr-88	0.0304	0.071	0.030	0.071
I-131	0.647	0.860	0.520	0.860
Xe-133	0.202	0.357	0.200	0.357
Xe-135	0.0545	0.143	0.055	0.143
Cs-134	1.41	1.41	1.41	1.41
Cs-137	1.41	1.41	1.41	1.41

D.1 References

ANSI/ANS 5.4. 1982. "Method for Calculating the Fractional Release of Volatile Fission Products From Oxide Fuel," American Nuclear Society, La Grange Park, Illinois.

Forsberg, K., and A. R. Massih. 1985. "Diffusion Theory of Fission Gas Migration in Irradiated Nuclear Fuel UO_2," *Journal of Nuclear Materials*, Vol. 135, pp. 140-148.

Lanning D. D., C. E. Beyer, and C. L. Painter. 1997a. *FRAPCON-3: Modifications to Fuel Rod Material Properties and Performance Models for High Burnup Applications*. NUREG/CR-6534, Vol 1, U.S. Nuclear Regulatory Commission, Washington, D.C.

Appendix E

Core-Average Radionuclide Inventories

Appendix E

Core-Average Radionuclide Inventories

This appendix lists the core-average radionuclide inventories calculated by ORIGEN-ARP as a function of peak-rod burnup for the 60 radionuclides used in evaluation of environmental impacts of a postulated LOCA. Inventories used in the MACCS code were calculated from these inventories assuming that the accident occurred at the end of a fuel cycle and that the core inventory consisted of equal fractions of fuel at the end of first, second, and third cycles. Table E.1 contains the inventories for PWR fuel, and Table E.2 contains the inventories for BWR fuel.

Table E.1 PWR Core-Average Inventory (Bq/MTU) for Radionuclides Most Important to Dose Calculations as a Function of Burnup

	Peak- Rod Burnup (GWd/MTU)												
	22	24	25	42	43	46	48	50	60	62	65	70	75
Isotope													
H-3	1.16E+13	1.27E+13	1.32E+13	2.16E+13	2.21E+13	2.35E+13	2.45E+13	2.54E+13	3.00E+13	3.09E+13	3.22E+13	3.44E+13	3.66E+13
Co-58	3.55E+14	3.59E+14	3.61E+14	3.92E+14	3.92E+14	3.96E+14	4.00E+14	4.03E+14	4.14E+14	4.18E+14	4.18E+14	4.26E+14	4.29E+14
Co-60	1.80E+14	1.95E+14	2.02E+14	3.26E+14	3.33E+14	3.54E+14	3.68E+14	3.81E+14	4.51E+14	4.66E+14	4.85E+14	5.18E+14	5.51E+14
Kr-85m	8.33E+15	8.18E+15	8.07E+15	6.92E+15	6.88E+15	6.70E+15	6.51E+15	6.40E+15	5.85E+15	5.74E+15	5.55E+15	5.33E+15	5.07E+15
Kr-85	2.62E+14	2.81E+14	2.90E+14	4.29E+14	4.37E+14	4.55E+14	4.70E+14	4.81E+14	5.33E+14	5.44E+14	5.59E+14	5.77E+14	5.96E+14
Kr-87	1.68E+16	1.65E+16	1.63E+16	1.39E+16	1.38E+16	1.34E+16	1.31E+16	1.28E+16	1.17E+16	1.14E+16	1.10E+16	1.05E+16	1.00E+16
Kr-88	2.36E+16	2.31E+16	2.28E+16	1.93E+16	1.91E+16	1.85E+16	1.80E+16	1.76E+16	1.60E+16	1.56E+16	1.51E+16	1.43E+16	1.36E+16
Rb-86	2.66E+13	2.94E+13	3.08E+13	5.74E+13	5.92E+13	6.44E+13	6.77E+13	7.14E+13	8.92E+13	9.25E+13	9.81E+13	1.07E+14	1.17E+14
Sr-89	3.30E+16	3.23E+16	3.20E+16	2.68E+16	2.66E+16	2.57E+16	2.51E+16	2.45E+16	2.19E+16	2.14E+16	2.07E+16	1.95E+16	1.84E+16
Sr-90	2.27E+15	2.44E+15	2.53E+15	3.81E+15	3.89E+15	4.07E+15	4.18E+15	4.33E+15	4.85E+15	4.96E+15	5.11E+15	5.33E+15	5.51E+15
Sr-91	4.03E+16	3.96E+16	3.92E+16	3.37E+16	3.36E+16	3.26E+16	3.18E+16	3.13E+16	2.86E+16	2.80E+16	2.72E+16	2.61E+16	2.49E+16
Sr-92	4.18E+16	4.11E+16	4.07E+16	3.59E+16	3.58E+16	3.49E+16	3.42E+16	3.37E+16	3.14E+16	3.08E+16	3.01E+16	2.91E+16	2.80E+16
Y-90	2.37E+15	2.55E+15	2.64E+15	3.99E+15	4.06E+15	4.25E+15	4.40E+15	4.51E+15	5.11E+15	5.22E+15	5.37E+15	5.60E+15	5.82E+15
Y-91	4.18E+16	4.07E+16	4.03E+16	3.49E+16	3.46E+16	3.36E+16	3.29E+16	3.23E+16	2.95E+16	2.89E+16	2.81E+16	2.69E+16	2.56E+16
Y-92	3.28E+12	3.35E+12	3.37E+12	3.96E+12	4.03E+12	4.14E+12	4.18E+12	4.26E+12	4.59E+12	4.63E+12	4.74E+12	4.92E+12	5.03E+12
Y-93	3.12E+16	3.08E+16	3.05E+16	2.76E+16	2.75E+16	2.69E+16	2.65E+16	2.62E+16	2.48E+16	2.44E+16	2.40E+16	2.34E+16	2.27E+16
Zr-95	5.24E+16	5.21E+16	5.17E+16	4.85E+16	4.86E+16	4.78E+16	4.75E+16	4.71E+16	4.54E+16	4.50E+16	4.43E+16	4.37E+16	4.30E+16
Zr-97	4.86E+16	4.83E+16	4.83E+16	4.67E+16	4.71E+16	4.68E+16	4.65E+16	4.65E+16	4.59E+16	4.56E+16	4.52E+16	4.53E+16	4.46E+16
Nb-95	5.28E+16	5.24E+16	5.21E+16	4.89E+16	4.86E+16	4.83E+16	4.79E+16	4.72E+16	4.55E+16	4.55E+16	4.48E+16	4.41E+16	4.31E+16
Mo-99	5.29E+16	5.29E+16	5.25E+16	5.18E+16	5.22E+16	5.22E+16	5.18E+16	5.18E+16	5.14E+16	5.14E+16	5.14E+16	5.11E+16	5.11E+16
Tc-99m	4.66E+16	4.66E+16	4.66E+16	4.63E+16	4.63E+16	4.63E+16	4.59E+16	4.59E+16	4.59E+16	4.59E+16	4.55E+16	4.55E+16	4.55E+16
Ru-103	3.81E+16	3.89E+16	3.92E+16	4.37E+16	4.40E+16	4.48E+16	4.51E+16	4.59E+16	4.81E+16	4.85E+16	4.88E+16	5.00E+16	5.07E+16
Ru-105	2.25E+16	2.33E+16	2.37E+16	2.96E+16	3.01E+16	3.12E+16	3.17E+16	3.24E+16	3.55E+16	3.59E+16	3.68E+16	3.81E+16	3.92E+16
Ru-106	9.36E+15	1.02E+16	1.07E+16	1.75E+16	1.78E+16	1.89E+16	1.96E+16	2.04E+16	2.37E+16	2.44E+16	2.53E+16	2.68E+16	2.82E+16
Rh-105	2.14E+16	2.23E+16	2.26E+16	2.80E+16	2.85E+16	2.94E+16	2.99E+16	3.06E+16	3.32E+16	3.36E+16	3.43E+16	3.55E+16	3.64E+16
Sb-125	2.05E+13	2.21E+13	2.28E+13	3.50E+13	3.57E+13	3.77E+13	3.89E+13	4.03E+13	4.63E+13	4.77E+13	4.96E+13	5.22E+13	5.51E+13
Sb-127	1.98E+15	2.03E+15	2.05E+15	2.36E+15	2.38E+15	2.43E+15	2.45E+15	2.49E+15	2.64E+15	2.66E+15	2.70E+15	2.76E+15	2.80E+15
Te-127m	3.21E+14	3.32E+14	3.36E+14	3.96E+14	4.00E+14	4.11E+14	4.14E+14	4.22E+14	4.48E+14	4.55E+14	4.63E+14	4.74E+14	4.81E+14
Te-127	1.95E+15	2.00E+15	2.02E+15	2.33E+15	2.36E+15	2.41E+15	2.43E+15	2.47E+15	2.62E+15	2.64E+15	2.68E+15	2.74E+15	2.78E+15
Te-129m	1.57E+15	1.59E+15	1.59E+15	1.70E+15	1.71E+15	1.73E+15	1.74E+15	1.75E+15	1.79E+15	1.80E+15	1.81E+15	1.83E+15	1.84E+15
Te-129	7.81E+15	7.92E+15	7.92E+15	8.40E+15	8.44E+15	8.51E+15	8.55E+15	8.58E+15	8.84E+15	8.84E+15	8.88E+15	8.99E+15	9.03E+15
Te-131m	4.85E+15	4.92E+15	4.96E+15	5.40E+15	5.48E+15	5.55E+15	5.55E+15	5.62E+15	5.85E+15	5.85E+15	5.88E+15	5.99E+15	6.03E+15
Te-132	3.96E+16	3.96E+16	3.96E+16	3.96E+16	4.00E+16	4.00E+16	3.96E+16	4.00E+16	4.00E+16	3.96E+16	3.96E+16	4.00E+16	3.96E+16

Table E.1 (continued)													
	Peak- Rod Burnup (GWd/MTU)												
	22	24	25	42	43	46	48	50	60	62	65	70	75
Isotope													
I-131	2.73E+16	2.73E+16	2.74E+16	2.77E+16	2.78E+16	2.78E+16	2.78E+16	2.79E+16	2.79E+16	2.80E+16	2.80E+16	2.80E+16	2.80E+16
I-132	4.00E+16	4.03E+16	4.03E+16	4.03E+16	4.07E+16	4.07E+16	4.03E+16	4.07E+16	4.07E+16	4.07E+16	4.07E+16	4.07E+16	4.07E+16
I-133	5.81E+16	5.81E+16	5.81E+16	5.70E+16	5.74E+16	5.70E+16	5.70E+16	5.70E+16	5.66E+16	5.62E+16	5.62E+16	5.59E+16	5.55E+16
I-134	6.55E+16	6.51E+16	6.51E+16	6.33E+16	6.33E+16	6.33E+16	6.25E+16	6.25E+16	6.18E+16	6.14E+16	6.14E+16	6.11E+16	6.03E+16
I-135	5.51E+16	5.51E+16	5.48E+16	5.44E+16	5.48E+16	5.44E+16	5.44E+16	5.44E+16	5.40E+16	5.40E+16	5.40E+16	5.37E+16	5.37E+16
Xe-133	5.81E+16	5.81E+16	5.81E+16	5.74E+16	5.74E+16	5.74E+16	5.70E+16	5.70E+16	5.66E+16	5.66E+16	5.62E+16	5.62E+16	5.59E+16
Xe-135	2.39E+16	2.38E+16	2.36E+16	2.06E+16	2.05E+16	2.01E+16	1.97E+16	1.95E+16	1.77E+16	1.74E+16	1.71E+16	1.65E+16	1.62E+16
Cs-134	2.13E+15	2.49E+15	2.68E+15	6.40E+15	6.66E+15	7.40E+15	7.96E+15	8.47E+15	1.11E+16	1.17E+16	1.25E+16	1.39E+16	1.54E+16
Cs-136	9.88E+14	1.08E+15	1.13E+15	2.02E+15	2.08E+15	2.25E+15	2.36E+15	2.48E+15	3.11E+15	3.23E+15	3.43E+15	3.77E+15	4.07E+15
Cs-137	2.68E+15	2.92E+15	3.04E+15	5.00E+15	5.11E+15	5.44E+15	5.70E+15	5.92E+15	6.99E+15	7.22E+15	7.55E+15	8.07E+15	8.58E+15
Ba-139	5.25E+16	5.25E+16	5.22E+16	5.03E+16	5.03E+16	5.03E+16	4.96E+16	4.96E+16	4.88E+16	4.85E+16	4.85E+16	4.81E+16	4.74E+16
Ba-140	5.25E+16	5.22E+16	5.22E+16	5.07E+16	5.07E+16	5.03E+16	5.00E+16	5.00E+16	4.92E+16	4.88E+16	4.85E+16	4.81E+16	4.77E+16
La-140	5.33E+16	5.33E+16	5.29E+16	5.22E+16	5.25E+16	5.22E+16	5.18E+16	5.18E+16	5.18E+16	5.14E+16	5.14E+16	5.14E+16	5.11E+16
La-141	4.81E+16	4.81E+16	4.77E+16	4.59E+16	4.63E+16	4.59E+16	4.55E+16	4.51E+16	4.44E+16	4.40E+16	4.40E+16	4.37E+16	4.29E+16
La-142	4.74E+16	4.74E+16	4.70E+16	4.48E+16	4.51E+16	4.48E+16	4.40E+16	4.40E+16	4.33E+16	4.29E+16	4.26E+16	4.22E+16	4.14E+16
Ce-141	4.85E+16	4.85E+16	4.81E+16	4.66E+16	4.66E+16	4.63E+16	4.59E+16	4.59E+16	4.48E+16	4.48E+16	4.44E+16	4.40E+16	4.37E+16
Ce-143	4.63E+16	4.59E+16	4.59E+16	4.29E+16	4.29E+16	4.26E+16	4.22E+16	4.18E+16	4.07E+16	4.03E+16	4.00E+16	3.92E+16	3.89E+16
Ce-144	3.74E+16	3.81E+16	3.85E+16	4.00E+16	3.96E+16	3.96E+16	3.92E+16	3.92E+16	3.81E+16	3.77E+16	3.74E+16	3.67E+16	3.60E+16
Pr-143	4.55E+16	4.51E+16	4.51E+16	4.22E+16	4.22E+16	4.14E+16	4.14E+16	4.11E+16	3.96E+16	3.96E+16	3.92E+16	3.85E+16	3.81E+16
Nd-147	1.91E+16	1.91E+16	1.91E+16	1.87E+16	1.88E+16	1.87E+16	1.86E+16	1.86E+16	1.85E+16	1.84E+16	1.84E+16	1.83E+16	1.82E+16
Np-239	4.48E+17	4.55E+17	4.59E+17	5.37E+17	5.44E+17	5.59E+17	5.62E+17	5.74E+17	6.18E+17	6.25E+17	6.40E+17	6.59E+17	6.73E+17
Pu-238	2.82E+13	3.52E+13	3.89E+13	1.42E+14	1.50E+14	1.76E+14	1.95E+14	2.14E+14	3.17E+14	3.39E+14	3.70E+14	4.26E+14	4.81E+14
Pu-239	1.17E+13	1.21E+13	1.23E+13	1.41E+13	1.41E+13	1.42E+13	1.42E+13	1.42E+13	1.41E+13	1.41E+13	1.40E+13	1.39E+13	1.39E+13
Pu-240	9.10E+12	1.00E+13	1.05E+13	1.91E+13	1.95E+13	2.06E+13	2.13E+13	2.20E+13	2.55E+13	2.60E+13	2.66E+13	2.75E+13	2.79E+13
Pu-241	2.80E+15	3.22E+15	3.42E+15	5.66E+15	5.81E+15	6.18E+15	6.44E+15	6.66E+15	7.18E+15	7.29E+15	7.47E+15	7.70E+15	7.99E+15
Am-241	2.31E+12	2.87E+12	3.16E+12	8.29E+12	8.55E+12	9.29E+12	9.73E+12	1.02E+13	1.16E+13	1.18E+13	1.20E+13	1.23E+13	1.25E+13
Am-242	7.77E+14	9.81E+14	1.08E+15	3.20E+15	3.34E+15	3.74E+15	3.96E+15	4.22E+15	5.25E+15	5.40E+15	5.66E+15	5.99E+15	6.29E+15
Cm-242	3.05E+14	4.07E+14	4.63E+14	1.93E+15	2.03E+15	2.32E+15	2.52E+15	2.72E+15	3.66E+15	3.81E+15	4.03E+15	4.37E+15	4.66E+15
Cm-244	4.11E+12	6.40E+12	7.84E+12	9.99E+13	1.11E+14	1.51E+14	1.83E+14	2.19E+14	4.66E+14	5.33E+14	6.44E+14	8.55E+14	1.11E+15

Table E.2 BWR Core-Average Inventory (Bq/MTU) for Radionuclides Most Important to Dose Calculations as a Function of Burnup

	Peak-Rod Burnup (GWd/MTU)												
	22	24	25	42	43	46	48	50	60	62	65	70	75
Isotope													
H-3	1.15E+13	1.24E+13	1.30E+13	2.10E+13	2.14E+13	2.28E+13	2.36E+13	2.45E+13	2.88E+13	2.96E+13	3.08E+13	3.28E+13	3.47E+13
Co-58	3.13E+14	3.11E+14	3.12E+14	3.40E+14	3.42E+14	3.39E+14	3.41E+14	3.44E+14	3.59E+14	3.61E+14	3.65E+14	3.69E+14	3.74E+14
Co-60	1.65E+14	1.78E+14	1.85E+14	2.92E+14	2.98E+14	3.16E+14	3.28E+14	3.40E+14	4.00E+14	4.11E+14	4.29E+14	4.59E+14	4.88E+14
Kr-85m	6.51E+15	6.51E+15	6.44E+15	5.48E+15	5.44E+15	5.33E+15	5.22E+15	5.11E+15	4.63E+15	4.51E+15	4.40E+15	4.18E+15	3.96E+15
Kr-85	2.57E+14	2.75E+14	2.85E+14	4.14E+14	4.22E+14	4.40E+14	4.51E+14	4.63E+14	5.11E+14	5.18E+14	5.29E+14	5.44E+14	5.59E+14
Kr-87	1.32E+16	1.32E+16	1.30E+16	1.10E+16	1.09E+16	1.07E+16	1.04E+16	1.02E+16	9.18E+15	8.95E+15	8.70E+15	8.25E+15	7.77E+15
Kr-88	1.85E+16	1.84E+16	1.82E+16	1.52E+16	1.51E+16	1.48E+16	1.44E+16	1.41E+16	1.26E+16	1.22E+16	1.19E+16	1.12E+16	1.05E+16
Rb-86	2.33E+13	2.51E+13	2.63E+13	4.96E+13	5.14E+13	5.40E+13	5.74E+13	6.03E+13	7.73E+13	8.07E+13	8.58E+13	9.40E+13	1.02E+14
Sr-89	2.58E+16	2.54E+16	2.52E+16	2.11E+16	2.08E+16	2.02E+16	1.98E+16	1.94E+16	1.72E+16	1.68E+16	1.62E+16	1.52E+16	1.42E+16
Sr-90	2.25E+15	2.42E+15	2.50E+15	3.77E+15	3.85E+15	4.03E+15	4.14E+15	4.26E+15	4.77E+15	4.88E+15	5.00E+15	5.22E+15	5.40E+15
Sr-91	3.17E+16	3.17E+16	3.13E+16	2.66E+16	2.65E+16	2.60E+16	2.55E+16	2.48E+16	2.26E+16	2.21E+16	2.15E+16	2.05E+16	1.94E+16
Sr-92	3.29E+16	3.28E+16	3.25E+16	2.83E+16	2.82E+16	2.78E+16	2.73E+16	2.67E+16	2.47E+16	2.42E+16	2.37E+16	2.29E+16	2.18E+16
Y-90	2.34E+15	2.50E+15	2.60E+15	3.90E+15	3.98E+15	4.16E+15	4.31E+15	4.42E+15	4.98E+15	5.09E+15	5.24E+15	5.47E+15	5.66E+15
Y-91	3.25E+16	3.21E+16	3.19E+16	2.73E+16	2.71E+16	2.64E+16	2.59E+16	2.55E+16	2.31E+16	2.26E+16	2.20E+16	2.09E+16	1.99E+16
Y-92	3.31E+16	3.31E+16	3.28E+16	2.85E+16	2.84E+16	2.79E+16	2.75E+16	2.69E+16	2.49E+16	2.44E+16	2.39E+16	2.30E+16	2.19E+16
Y-93	2.45E+16	2.45E+16	2.43E+16	2.17E+16	2.16E+16	2.14E+16	2.11E+16	2.07E+16	1.95E+16	1.92E+16	1.89E+16	1.84E+16	1.77E+16
Zr-95	4.12E+16	4.08E+16	4.08E+16	3.80E+16	3.80E+16	3.75E+16	3.73E+16	3.70E+16	3.56E+16	3.53E+16	3.49E+16	3.43E+16	3.35E+16
Zr-97	3.83E+16	3.84E+16	3.83E+16	3.71E+16	3.71E+16	3.71E+16	3.69E+16	3.66E+16	3.62E+16	3.60E+16	3.59E+16	3.57E+16	3.52E+16
Nb-95	4.16E+16	4.12E+16	4.08E+16	3.84E+16	3.80E+16	3.77E+16	3.75E+16	3.72E+16	3.58E+16	3.55E+16	3.51E+16	3.45E+16	3.39E+16
Mo-99	4.14E+16	4.18E+16	4.18E+16	4.11E+16	4.11E+16	4.11E+16	4.11E+16	4.07E+16	4.07E+16	4.03E+16	4.03E+16	4.03E+16	4.00E+16
Tc-99m	3.67E+16	3.70E+16	3.69E+16	3.63E+16	3.64E+16	3.66E+16	3.64E+16	3.62E+16	3.61E+16	3.60E+16	3.60E+16	3.60E+16	3.56E+16
Ru-103	3.02E+16	3.05E+16	3.07E+16	3.43E+16	3.46E+16	3.51E+16	3.56E+16	3.59E+16	3.77E+16	3.81E+16	3.85E+16	3.92E+16	4.00E+16
Ru-105	1.77E+16	1.82E+16	1.84E+16	2.32E+16	2.35E+16	2.44E+16	2.48E+16	2.52E+16	2.78E+16	2.82E+16	2.90E+16	3.00E+16	3.07E+16
Ru-106	8.36E+15	9.03E+15	9.36E+15	1.45E+16	1.48E+16	1.57E+16	1.62E+16	1.68E+16	1.94E+16	1.99E+16	2.06E+16	2.18E+16	2.29E+16
Rh-105	1.71E+16	1.75E+16	1.77E+16	2.22E+16	2.25E+16	2.33E+16	2.36E+16	2.40E+16	2.63E+16	2.66E+16	2.73E+16	2.82E+16	2.89E+16
Sb-125	1.78E+14	1.92E+14	1.99E+14	3.05E+14	3.10E+14	3.26E+14	3.37E+14	3.47E+14	3.95E+14	4.04E+14	4.16E+14	4.37E+14	4.58E+14
Sb-127	1.55E+15	1.58E+15	1.59E+15	1.84E+15	1.86E+15	1.90E+15	1.93E+15	1.94E+15	2.07E+15	2.08E+15	2.12E+15	2.17E+15	2.19E+15
Te-127m	2.57E+14	2.64E+14	2.68E+14	3.13E+14	3.15E+14	3.23E+14	3.28E+14	3.32E+14	3.53E+14	3.57E+14	3.63E+14	3.70E+14	3.81E+14
Te-127	1.53E+15	1.56E+15	1.58E+15	1.82E+15	1.84E+15	1.89E+15	1.91E+15	1.93E+15	2.05E+15	2.07E+15	2.11E+15	2.15E+15	2.18E+15
Te-129m	1.24E+15	1.24E+15	1.25E+15	1.34E+15	1.34E+15	1.35E+15	1.36E+15	1.37E+15	1.41E+15	1.41E+15	1.42E+15	1.44E+15	1.44E+15
Te-129	6.14E+15	6.22E+15	6.25E+15	6.59E+15	6.62E+15	6.70E+15	6.73E+15	6.73E+15	6.92E+15	6.92E+15	6.99E+15	7.07E+15	7.07E+15
Te-131m	3.81E+15	3.85E+15	3.89E+15	4.26E+15	4.26E+15	4.37E+15	4.37E+15	4.40E+15	4.55E+15	4.59E+15	4.63E+15	4.70E+15	4.70E+15

Table E.2 (continued)													
	Peak-Rod Burnup (GWd/MTU)												
	22	24	25	42	43	46	48	50	60	62	65	70	75
Isotope													
Te-132	3.11E+16	3.13E+16	3.13E+16	3.12E+16	3.13E+16	3.14E+16	3.14E+16	3.12E+16	3.13E+16	3.12E+16	3.13E+16	3.13E+16	3.09E+16
I-131	2.15E+16	2.14E+16	2.15E+16	2.18E+16	2.18E+16	2.18E+16	2.18E+16	2.18E+16	2.19E+16	2.19E+16	2.19E+16	2.19E+16	2.19E+16
I-132	3.15E+16	3.17E+16	3.17E+16	3.17E+16	3.19E+16	3.20E+16	3.20E+16	3.18E+16	3.20E+16	3.19E+16	3.19E+16	3.20E+16	3.16E+16
I-133	4.59E+16	4.59E+16	4.59E+16	4.48E+16	4.51E+16	4.51E+16	4.51E+16	4.48E+16	4.44E+16	4.40E+16	4.40E+16	4.40E+16	4.33E+16
I-134	5.14E+16	5.14E+16	5.14E+16	4.96E+16	5.00E+16	5.00E+16	4.96E+16	4.92E+16	4.85E+16	4.81E+16	4.81E+16	4.77E+16	4.70E+16
I-135	4.33E+16	4.37E+16	4.33E+16	4.29E+16	4.29E+16	4.29E+16	4.29E+16	4.26E+16	4.26E+16	4.22E+16	4.26E+16	4.22E+16	4.18E+16
Xe-133	4.59E+16	4.40E+16	4.44E+16	4.51E+16	4.51E+16	4.33E+16	4.37E+16	4.48E+16	4.37E+16	4.44E+16	4.44E+16	4.44E+16	4.37E+16
Xe-135	2.21E+16	2.12E+16	2.11E+16	1.87E+16	1.86E+16	1.78E+16	1.74E+16	1.71E+16	1.58E+16	1.55E+16	1.52E+16	1.46E+16	1.42E+16
Cs-134	2.13E+15	2.44E+15	2.60E+15	5.99E+15	6.22E+15	6.88E+15	7.29E+15	7.77E+15	1.01E+16	1.07E+16	1.14E+16	1.26E+16	1.39E+16
Cs-136	9.58E+14	1.03E+15	1.07E+15	1.92E+15	1.98E+15	2.10E+15	2.22E+15	2.33E+15	2.95E+15	3.06E+15	3.26E+15	3.56E+15	3.89E+15
Cs-137	2.67E+15	2.90E+15	3.02E+15	4.96E+15	5.07E+15	5.40E+15	5.59E+15	5.81E+15	6.88E+15	7.07E+15	7.40E+15	7.88E+15	8.40E+15
Ba-139	4.14E+16	4.14E+16	4.14E+16	3.96E+16	3.96E+16	3.96E+16	3.96E+16	3.92E+16	3.85E+16	3.81E+16	3.81E+16	3.77E+16	3.70E+16
Ba-140	4.11E+16	4.14E+16	4.14E+16	3.96E+16	3.96E+16	3.96E+16	3.96E+16	3.92E+16	3.85E+16	3.85E+16	3.81E+16	3.77E+16	3.74E+16
La-140	4.18E+16	4.40E+16	4.29E+16	4.11E+16	4.11E+16	4.33E+16	4.22E+16	4.07E+16	4.11E+16	4.03E+16	4.07E+16	4.07E+16	4.00E+16
La-141	3.77E+16	3.81E+16	3.77E+16	3.62E+16	3.62E+16	3.62E+16	3.60E+16	3.56E+16	3.50E+16	3.47E+16	3.46E+16	3.42E+16	3.36E+16
La-142	3.74E+16	3.74E+16	3.74E+16	3.54E+16	3.53E+16	3.53E+16	3.51E+16	3.47E+16	3.39E+16	3.36E+16	3.34E+16	3.30E+16	3.23E+16
Ce-141	3.81E+16	3.81E+16	3.81E+16	3.65E+16	3.65E+16	3.63E+16	3.62E+16	3.60E+16	3.52E+16	3.50E+16	3.49E+16	3.45E+16	3.40E+16
Ce-143	3.64E+16	3.65E+16	3.63E+16	3.39E+16	3.39E+16	3.37E+16	3.34E+16	3.30E+16	3.20E+16	3.16E+16	3.14E+16	3.09E+16	3.02E+16
Ce-144	3.14E+16	3.18E+16	3.19E+16	3.16E+16	3.15E+16	3.12E+16	3.10E+16	3.08E+16	2.97E+16	2.95E+16	2.91E+16	2.86E+16	2.80E+16
Pr-143	3.58E+16	3.53E+16	3.52E+16	3.33E+16	3.31E+16	3.26E+16	3.24E+16	3.24E+16	3.11E+16	3.11E+16	3.07E+16	3.02E+16	3.00E+16
Nd-147	1.51E+16	1.51E+16	1.51E+16	1.47E+16	1.48E+16	1.48E+16	1.47E+16	1.47E+16	1.46E+16	1.45E+16	1.45E+16	1.44E+16	1.42E+16
Np-239	3.49E+17	3.50E+17	3.53E+17	4.14E+17	4.18E+17	4.22E+17	4.33E+17	4.40E+17	4.81E+17	4.88E+17	5.00E+17	5.18E+17	5.25E+17
Pu-238	3.10E+13	3.85E+13	4.26E+13	1.51E+14	1.59E+14	1.86E+14	2.05E+14	2.25E+14	3.31E+14	3.53E+14	3.85E+14	4.40E+14	4.92E+14
Pu-239	1.15E+13	1.19E+13	1.21E+13	1.34E+13	1.34E+13	1.34E+13	1.34E+13	1.34E+13	1.31E+13	1.31E+13	1.30E+13	1.28E+13	1.28E+13
Pu-240	9.18E+12	1.04E+13	1.10E+13	1.97E+13	2.01E+13	2.14E+13	2.23E+13	2.32E+13	2.65E+13	2.69E+13	2.76E+13	2.86E+13	2.89E+13
Pu-241	2.81E+15	3.04E+15	3.16E+15	5.44E+15	5.59E+15	5.85E+15	5.92E+15	6.03E+15	6.70E+15	6.81E+15	6.96E+15	7.10E+15	7.36E+15
Am-241	2.94E+12	3.66E+12	4.03E+12	9.62E+12	9.95E+12	1.09E+13	1.14E+13	1.19E+13	1.33E+13	1.34E+13	1.36E+13	1.37E+13	1.38E+13
Am-242	8.07E+14	9.88E+14	1.09E+15	3.06E+15	3.20E+15	3.56E+15	3.81E+15	4.03E+15	5.03E+15	5.18E+15	5.40E+15	5.66E+15	5.85E+15
Cm-242	3.63E+14	4.77E+14	5.40E+14	1.97E+15	2.06E+15	2.36E+15	2.56E+15	2.76E+15	3.65E+15	3.81E+15	4.00E+15	4.26E+15	4.48E+15
Cm-244	4.88E+12	7.55E+12	9.25E+12	1.09E+14	1.21E+14	1.62E+14	1.94E+14	2.30E+14	4.85E+14	5.51E+14	6.62E+14	8.70E+14	1.13E+15

Appendix F

Transportation Analysis Bases

Appendix F

Transportation Analysis Bases

The environmental effects of transportation of spent fuel have been examined in several studies, in addition to those of Mauro et al. (1985) and Baker et al. (1988). The study reported in WASH-1238 (AEC 1972) forms the basis for Table S-4 in 10 CFR Part 51. It evaluated the effects of spent fuel having less than 4 percent enrichment and burnup to 33 GWd/MTU. NUREG-0170 (NRC 1977) reexamined the effects of spent fuel with improved methods and better data and concluded that impacts of transportation of spent fuel in Table S-4 were conservative. Based on the analyses in Mauro et al. (1985) and Baker et al. (1988), the applicability of Table S-4 has been extended to 5 percent enriched fuel with burnup to 60 GWd/MTU provided that the spent fuel is not shipped until at least 5 years elapse after discharge from the reactor. This extension is reasonable as older low-burnup fuel is expected to be shipped to a spent-fuel repository before high-burnup fuel.

Recently, the result of two new studies have been published. Addendum 1 to NUREG-1437 (NRC 1999) describes the cumulative impacts in the vicinity of a single destination of transportation of the current spent fuel inventory to that destination. It considers transportation of 5 percent enriched fuel with 62 GWd/MTU and concludes "... that the values shown in Table S-4 continue to be a reasonable estimate of environmental impacts of transportation of fuel...." The other study by Sprung et al. (2000) uses probabilistic methods to reexamine the fuel shipment risk estimates of NUREG-0170 in the light of the existing spent-fuel inventory and burnup levels. It concludes that the expected population doses for incident-free legal-weight truck shipments are about a factor of 4 lower than the estimates in NUREG-0170. Similarly, the expected population doses for rail shipments are about a factor of 3 lower than the estimates in NUREG-0170.

This study examines the environmental effects of extending fuel burnup above 60 GWd/MTU. It updates and extends the analyses of Mauro et al. and Baker et al., and it has a significantly different focus than the analyses in either Addendum 1 to NUREG-1437 or Sprung et al. However, in general, the bases for the analyses of the environmental effects of the transportation of spent fuel in this study are consistent with the bases in the previous studies. Table F.1 presents a comparison of the bases for the analysis in this study with those in WASH-1238, NUREG-0170, Addendum 1 to NUREG-1437, and the study by Sprung et al.

F.1 References

Baker, D. A., W. J. Bailey, C. E. Beyer, F. C. Bold, and J. J. Tawil. 1988. *Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors*. NUREG/CR-5009, U.S. Nuclear Regulatory Commission, Washington, D.C.

Mauro, J. J., R. Eng, S. Marschke, W. Chang, and T. A. Coleman. 1985. *The Environmental Consequences of Higher Fuel Burn-up*. AIF/NESP-032, Atomic Industrial Forum, Inc., Bethesda, Maryland.

Sprung, J. L., D. J. Ammerman, N. L. Breivik, R. J. Dukhart, F. L. Kanipe, J. A. Koski, G. S. Mills, K. S. Neuhauser, H. D. Radloff, R. F. Weiner, and H. R. Yoshimura. 2000. *Reexamination of Spent Fuel Shipment Risk Estimates*. NUREG/CR-6672 (SAND2000-0234), U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. Atomic Energy Commission (AEC). 1972. *Environmental Survey of Transportation of Radioactive Materials To and From Nuclear Power Plants*. WASH-1238, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1977. *Final Environmental Statement on Transportation of Radioactive Material by Air and Other Modes*. NUREG-0172, Vol. 1, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1999. *Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Main Report, Section 6.3–Transportation, Table 9.1 Summary of findings on NEPA issues for license renewal of nuclear power plants*. NUREG-1437, Vol. 1, Addendum 1, Washington, D.C.

Table F.1 Comparison of Bases for Estimation of Environmental Effects of the Transportation of Spent Fuel

					This Study (2000)				
	WASH-1238 (1972)	NUREG-0170 (1977)	NUREG-1437/A1 (1999)	NUREG/CR-6672 (2000)	Southeast Turkey Pt./ Brunswick	Northeast Truck/Rail	Midwest	Southwest	Northwest
Crew	2	2	4	2	2	2	2	2	2
Onlookers	10 people @ 1 m for 3 min/ shipment	Maximum individual. 3 min @ 25 mr/hr	30 people @ 20 m at each stop	density $=3\times10^4$ people/km^2 between 1 and 10 m	30 people @ 20 m at each stop	30 people @ 20 m at each stop	30 people @ 20 m at each stop	30 people @ 20 m at each stop	30 people @ 20 m at each stop
Trip length (mi)	1000	1000	~150 (Highway)	Variable 0 to 3100 median ~1250	2830/2490 (Highway)	2640/2830 (Highway/ Interline)	1800 (Highway)	281 (Highway)	983 (Highway)
% Rural	90%	90%	<5%	Variable 50% to 100% median ~80%	79%/83%	80%/74%	88%	67%	87%
% Suburban	5%	5%	~20%	Variable 5% to 50% median ~18%	18%/15%	18%/22%	10%	14%	11%
% Urban	5%	5%	~75%	Variable 0% to 10% median ~2%	3%/2%	3%/4%	3%	19%	2%
Rural density (people/mi^2)		15	~5	Variable 0 to 66 mean = 6	20/22	20/19	11	9	12
Suburban density (people/mi^2)	330 east of Mississippi 110 West of Mississippi	1840	~1000	Variable 67 to 1670 mean = 719	903/893	901/973	948	1440	1190
Urban density (people/mi^2)		9880	~5000	Variable > 1671 mean = 3861	5910/5660	5980/6260	5240	7270	5340

Table F.1 (continued)

	WASH-1238 (1972)	NUREG-0170 (1977)	NUREG-1437/A1 (1999)	NUREG/CR-6672 (2000)	This Study (2000): Southeast Turkey Pt./ Brunswick	This Study (2000): Northeast Truck/Rail	This Study (2000): Midwest	This Study (2000): Southwest	This Study (2000): Northwest
Rural stop time	Stop time not used to estimate onlooker doses	2 hr	0.011 hr/km	Variable – 0 to 1 hr median ~ 0.5 hr	0.002 hr/km	0.002 hr/km / 0.033 hr/km	0.002 hr/km	0.002 hr/km	0.002 hr/km
Suburb, stop time		5 hr	0.011 hr/km	None	0.002 hr/km	0.002 hr/km / 0.033 hr/km	0.002 hr/km	0.002 hr/km	0.002 hr/km
Urban stop time		1 hr	0.011 hr/km	None	0.002 hr/km	0.002 hr/km / 0.033 hr/km	0.002 hr/km	0.002 hr/km	0.002 hr/km
Rural speed (mph)	1000 mi/ 20 hr = 50 mph	55	55	55	55	55/40	55	55	55
Sub. speed (mph)		25	25	55	25	25/25	25	25	25
Urban speed (mph)		15	15	55	15	15/16	15	15	15
Fuel assemblies per cask (PWR)	1 to 3	Not specified ~1	4	3	2 to 4 / NA	2 to 4 / 30 to 40	2 to 4	2 to 4	2 to 4
Fuel assemblies per cask (BWR)	2 to 7	Not specified ~3	NA	7	NA / 4 to 9	4 to 9 / NA	4 to 9	4 to 9	4 to 9
Fuel burnup (GWD/MTU)	PWR – 33	Not specified ~33	PWR – 62	PWR – 60 BWR – 50	PWR 43 to 75 / BWR 35 to 75	PWR 43 to 75 / BWR 35 to 75	PWR 43 to 75 / BWR 35 to 75	PWR 43 to 75 / BWR 35 to 75	PWR 43 to 75 / BWR 35 to 75
Cooling time (yr)	0.25	0.42	5	Variable 5 to 25 median ~15	5/5	5/5	5	5	5

Table F.1 (continued)

					This Study (2000)				
	WASH-1238 (1972)	NUREG-0170 (1977)	NUREG-1437/A1 (1999)	NUREG/CR-6672 (2000)	Southeast Turkey Pt./ Brunswick	Northeast Truck/Rail	Midwest	Southwest	Northwest
Crew dose rate (mrem/hr)	0.2	0.2	2	Calculated, not to exceed 2	0.2	0.2	0.2	0.2	0.2
Vehicle surface dose rate (mrem /hr @ 2m)	10	7.6	10	Variable— 2.8 to 13 mrem/hr @ 1 m median ~ 5	10	10	10	10	10

Appendix G

RADTRAN Output Listing Sample

Appendix G

RADTRAN Output Listing Sample

This appendix contains edited listing of RADTRAN output files for representative truck and rail shipments. The truck output file is listed first, followed by the rail output file. The first portion of each listing (Pages 1 and 2) echoes the input data. Among other things, it shows the radionuclide inventory and the transportation route information. The radionuclide inventory was changed, as appropriate, for BWR and PWR fuel, and transportation route information was changed as a function of the region of the country. Accident parameters are also shown. The remainder of the listings contain the results of the incident-free radiation exposure and accident risk calculations performed by RADTRAN. The note under the incident-free population exposure results for the representative truck shipment indicates that the crew doses calculated by RADTRAN were reduced by a factor of 20 to account for a crew of two rather than four and lower dose rates in the truck cab (factor of 10 lower than the regulatory limit).

RADTRAN 4 Output File for Representative Truck Shipment

```
            RUN DATE: [ 18-Aug-00 AT 11:26:00 ]              PAGE    1

 RRRR    AAA   DDDD   TTTTT  RRRR    AAA   N   N
 R   R  A   A  D   D    T    R   R  A   A  NN  N
 R   R  A   A  D   D    T    R   R  A   A  N N N
 RRRR   A   A  D   D    T    RRRR   A   A  N  NN
 R R    AAAAA  D   D    T    R R    AAAAA  N   N
 R  R   A   A  D   D    T    R  R   A   A  N   N
 R   R  A   A  DDDD     T    R   R  A   A  N   N

                            4
                         4  4
                         4  4
                         44444
                            4
                            4
                            4

 RADTRAN 4.0.19.SI    VERSION DATE: MARCH 16, 1999

            MODE DESCRIPTIONS
```

NUMBER	NAME	CHARACTERIZATION
1	TRUCK	LONG HAUL VEHICLE
2	RAIL	COMMERCIAL TRAIN
3	BARGE	INLAND VESSEL
4	SHIP	OPEN SEA VESSEL
5	CARGO AIR	CARGO AIRCRAFT
6	PASS AIR	PASSENGER AIRCRAFT
7	P-VAN	PASSENGER VAN
8	CVAN-T	COMMERCIAL VAN
9	CVAN-R	COMMERCIAL VAN
10	CVAN-CA	COMMERCIAL VAN

ECHO CHECK

```
&& Edited Fri Aug 18 11:25:43 2000
TITLE _Brunswick - 35 GWd - 5-yr_
FORM UNIT
DIMEN 31 6 3 10 18
PARM 1 3 2 1 0
PACKAGE
   LABGRP
      SOLID       GAS     VOLATIL
SHIPMENT
   LABISO
      CR51      MN54      FE55      FE59      CO58      CO60
      KR85      SR89      SR90      Y91       ZR95      NB95
      RU103     RU106     SB125     TE125M    TE127     TE127M
      CS134     CS137     CE141     CE144     PM147     EU154
      PU238     PU239     PU240     PU241     AM241     CM242
      CM244
NORMAL
   NMODE=1
      7.910E-01  1.800E-01  2.900E-02  8.849E+01  4.025E+01  2.416E+01
      2.000E+00  3.100E+00  0.000E+00  2.000E-03  0.000E+00  0.000E+00
      0.000E+00  3.000E+01  2.000E+01  0.000E+00  1.000E+02  1.000E+02
      2.000E+00  8.000E-02  5.000E-02  8.500E-01  4.700E+02  7.800E+02
      2.800E+03
ACCIDENT
   SEVFRC
     NPOP=1
       NMODE=1
       9.94E-01  4.05E-05  3.82E-03  1.80E-03  1.55E-05  9.84E-06
```

```
     NPOP=2
       NMODE=1
       9.94E-01  4.05E-05  3.82E-03  1.80E-03  1.55E-05  9.84E-06
     NPOP=3
       NMODE=1
       9.94E-01  4.05E-05  3.82E-03  1.80E-03  1.55E-05  9.84E-06
RELEASE
   RFRAC
     GROUP=1
       0.00E+00  6.00E-08  2.00E-07  2.00E-06  2.00E-06  2.00E-05
     GROUP=2
       0.00E+00  9.90E-03  3.30E-02  3.90E-01  3.30E-01  6.30E-01
     GROUP=3
       0.00E+00  6.00E-06  2.00E-05  2.00E-04  2.00E-04  2.00E-03
EOF
ISOTOPES  -1    1    1.00   13.000   1.00   0.00  CSNF
         CR51  6.39E-16     SOLID   2
         MN54  4.02E+01     SOLID   2
         FE55  2.35E+03     SOLID   2
         FE59  3.00E-10     SOLID   2
         CO58  3.27E-04     SOLID   2
         CO60  7.90E+03     SOLID   2
         KR85  1.27E+04       GAS  10
         SR89  2.17E-05     SOLID   2
         SR90  1.34E+05     SOLID   2
          Y91  8.41E-04     SOLID   2
         ZR95  7.17E-03     SOLID   2
         NB95  1.63E-02     SOLID   2
        RU103  2.23E-08   VOLATIL   7
        RU106  2.48E+04   VOLATIL   7
        SB125  3.95E+03     SOLID   2
       TE125M  9.65E+02     SOLID   2
        TE127  1.86E-01     SOLID   2
       TE127M  1.91E-01     SOLID   2
        CS134  4.42E+04   VOLATIL   7
        CS137  1.73E+05   VOLATIL   7
        CE141  3.14E-11     SOLID   2
        CE144  2.54E+04     SOLID   2
        PM147  8.94E+04     SOLID   2
        EU154  6.96E+03     SOLID   2
        PU238  4.58E+03     SOLID   2
        PU239  5.91E+02     SOLID   2
        PU240  7.60E+02     SOLID   2
```

```
        PU241   1.60E+05      SOLID    2
        AM241   1.66E+03      SOLID    2
        CM242   3.38E+01      SOLID    2
        CM244   2.49E+03      SOLID    2
LINK   1 3.34E+03 8.80E+01 8.60E+00 4.70E+02 3.15E-07 R 1
LINK   1 6.03E+02 4.00E+01 3.45E+02 7.80E+02 3.15E-07 S 1
LINK   1 6.14E+01 2.40E+01 2.19E+03 2.80E+03 3.66E-07 U 1
PKGSIZ
         CSNF      5.00
EOF
```


Brunswick - 35 GWd - 5-yr

INCIDENT-FREE SUMMARY
******** **** *******

INCIDENT-FREE POPULATION EXPOSURE IN PERSON-REM

	PASSENGR	CREW	HANDLERS	OFF LINK	ON LINK	STOPS	STORAGE	TOTALS
LINK 1	0.00E+00	1.53E-01	0.00E+00	9.52E-04	2.72E-02	7.02E-02	0.00E+00	2.51E-01
LINK 2	0.00E+00	6.08E-02	0.00E+00	1.32E-02	4.15E-02	1.27E-02	0.00E+00	1.28E-01
LINK 3	0.00E+00	1.03E-02	0.00E+00	2.94E-04	4.46E-02	1.29E-03	0.00E+00	5.65E-02
RURAL	0.00E+00	1.53E-01	0.00E+00	9.52E-04	2.72E-02	7.02E-02	0.00E+00	2.51E-01
SUBURB	0.00E+00	6.08E-02	0.00E+00	1.32E-02	4.15E-02	1.27E-02	0.00E+00	1.28E-01
URBAN	0.00E+00	1.03E-02	0.00E+00	2.94E-04	4.46E-02	1.29E-03	0.00E+00	5.65E-02
TOTALS:	0.00E+00	2.24E-01	0.00E+00	1.44E-02	1.13E-01	8.42E-02	0.00E+00	4.36E-01

The crew dose was adjusted to account smaller crew size (2 versus 4 in NUREG-1437 Addendum 1) and for a lower dose rate in the truck cab (0.2 mrem/hr used in NUREG-0170 versus 2 mrem/hr regulatory limit. This adjustment was not made for public doses.

MAXIMUM INDIVIDUAL IN-TRANSIT DOSE

```
LINK  1  5.36E-07 REM
LINK  2  5.36E-07 REM
LINK  3  5.36E-07 REM
```

Brunswick - 35 GWd - 5-yr

EXPECTED VALUES OF POPULATION RISK IN PERSON-REM

	GROUND	INHALED	RESUSPD	CLOUDSH	*INGESTION	TOTAL
LINK 1	1.61E-04	2.60E-06	7.20E-06	7.57E-09	0.00E+00	1.71E-04
LINK 2	1.17E-03	1.88E-05	5.21E-05	5.48E-08	0.00E+00	1.24E-03
LINK 3	5.29E-04	8.54E-06	2.37E-05	2.49E-08	0.00E+00	5.61E-04
RURAL	1.61E-04	2.60E-06	7.20E-06	7.57E-09	0.00E+00	1.71E-04
SUBURB	1.17E-03	1.88E-05	5.21E-05	5.48E-08	0.00E+00	1.24E-03
URBAN	5.29E-04	8.54E-06	2.37E-05	2.49E-08	0.00E+00	5.61E-04
TOTALS:	1.85E-03	2.99E-05	8.30E-05	8.73E-08	0.00E+00	1.97E-03

* NOTE THAT INGESTION RISK IS A SOCIETAL RISK;
THE USER MAY WISH TO TREAT THIS VALUE SEPARATELY.

Brunswick - 35 GWd - 5-yr

EXPECTED RISK VALUES - OTHER

LINK	ECON $$	EARLY FATALITY
1	0.00E+00	0.00E+00
2	0.00E+00	0.00E+00
3	0.00E+00	0.00E+00
TOTAL	0.00E+00	0.00E+00

TOTAL EXPOSED POPULATION: INCIDENT-FREE

LINK 1 4.60E+04 PERSONS
LINK 2 3.33E+05 PERSONS
LINK 3 2.15E+05 PERSONS

TOTAL 5.94E+05 PERSONS

TOTAL EXPOSED POPULATION: ACCIDENT
(PERSONS UNDER PLUME FOOTPRINT FOR A SINGLE ACCIDENT)

```
               LINK  1  1.16E+04 PERSONS
               LINK  2  4.66E+05 PERSONS
               LINK  3  2.96E+06 PERSONS
EOI
END OF RUN
```

RADTRAN 4 Output File for Representative Rail Shipment

```
RUN DATE: [ 18-Aug-00 AT 12:08:26 ]          PAGE     1

          RRRR    AAA   DDDD   TTTTT  RRRR    AAA   N   N
          R   R  A   A  D   D    T    R   R  A   A  NN  N
          R   R  A   A  D   D    T    R   R  A   A  N N N
          RRRR   A   A  D   D    T    RRRR   A   A  N  NN
          R R    AAAAA  D   D    T    R R    AAAAA  N   N
          R  R   A   A  D   D    T    R  R   A   A  N   N
          R   R  A   A  DDDD     T    R   R  A   A  N   N

                             4
                          4  4
                          4  4
                          44444
                             4
                             4
                             4

          RADTRAN 4.0.19.SI   VERSION DATE: MARCH 16, 1999

                     MODE DESCRIPTIONS

           NUMBER  NAME          CHARACTERIZATION
              1    TRUCK         LONG HAUL VEHICLE
              2    RAIL          COMMERCIAL TRAIN
              3    BARGE         INLAND VESSEL
              4    SHIP          OPEN SEA VESSEL
              5    CARGO AIR     CARGO AIRCRAFT
              6    PASS AIR      PASSENGER AIRCRAFT
              7    P-VAN         PASSENGER VAN
```

```
                    8    CVAN-T          COMMERCIAL VAN
                    9    CVAN-R          COMMERCIAL VAN
                    10   CVAN-CA         COMMERCIAL VAN
```


ECHO CHECK

```
&& Edited Fri Aug 18 12:08:15 2000
&& _RAIL4.IN4_
TITLE _Rail - NE Region - 43GWd - 5-yr_
FORM UNIT
DIMEN 31 6 3 10 18
PARM 1 3 2 1 0
PACKAGE
   LABGRP
      SOLID       GAS     VOLATIL
SHIPMENT
   LABISO
       CR51        MN54        FE55        FE59        CO58        CO60
       KR85        SR89        SR90         Y91        ZR95        NB95
      RU103       RU106       SB125      TE125M       TE127      TE127M
      CS134       CS137       CE141       CE144       PM147       EU154
      PU238       PU239       PU240       PU241       AM241       CM242
      CM244
NORMAL
   NMODE=2
       7.910E-01  1.800E-01  2.900E-02  6.437E+01  4.025E+01  2.416E+01
       5.000E+00  1.520E+02  0.000E+00  3.300E-02  0.000E+00  6.000E+01
       2.000E+00  1.000E+02  2.000E+01  0.000E+00  0.000E+00  1.000E+02
       3.000E+00  0.000E+00  0.000E+00  0.000E+00  1.000E+00  5.000E+00
       5.000E+00
ACCIDENT
   SEVFRC
     NPOP=1
       NMODE=2
       9.94E-01  2.02E-03  2.72E-03  5.55E-04  6.14E-04  1.25E-04
     NPOP=2
       NMODE=2
       9.94E-01  2.02E-03  2.72E-03  5.55E-04  6.14E-04  1.25E-04
     NPOP=3
       NMODE=2
```

```
     9.94E-01  2.02E-03  2.72E-03  5.55E-04  6.14E-04  1.25E-04
RELEASE
   RFRAC
     GROUP=1
       0.00E+00  6.00E-08  2.00E-07  2.00E-06  2.00E-06  2.00E-05
     GROUP=2
       0.00E+00  9.90E-03  3.30E-02  3.90E-01  3.30E-01  6.30E-01
     GROUP=3
       0.00E+00  6.00E-06  2.00E-05  2.00E-04  2.00E-04  2.00E-03
EOF
ISOTOPES  2    1     1.00   13.000   1.00   0.00  CSNF
         CR51  8.95E-15     SOLID   2
         MN54  5.51E+02     SOLID   2
         FE55  2.98E+04     SOLID   2
         FE59  4.21E-09     SOLID   2
         CO58  3.92E-03     SOLID   2
         CO60  8.71E+04     SOLID   2
         KR85  1.51E+05       GAS  10
         SR89  2.66E-04     SOLID   2
         SR90  1.59E+06     SOLID   2
          Y91  1.06E-02     SOLID   2
         ZR95  9.44E-02     SOLID   2
         NB95  2.15E-01     SOLID   2
        RU103  3.25E-07   VOLATIL   7
        RU106  3.74E+05   VOLATIL   7
        SB125  5.22E+04     SOLID   2
       TE125M  1.28E+04     SOLID   2
        TE127  2.75E+00     SOLID   2
       TE127M  2.80E+00     SOLID   2
        CS134  6.57E+05   VOLATIL   7
        CS137  2.11E+06   VOLATIL   7
        CE141  4.22E-10     SOLID   2
        CE144  3.37E+05     SOLID   2
        PM147  1.01E+06     SOLID   2
        EU154  9.63E+04     SOLID   2
        PU238  6.62E+04     SOLID   2
        PU239  6.43E+03     SOLID   2
        PU240  8.01E+03     SOLID   2
        PU241  2.35E+06     SOLID   2
        AM241  2.38E+04     SOLID   2
        CM242  4.61E+02     SOLID   2
        CM244  4.27E+04     SOLID   2
 LINK  2 3.35E+03 6.44E+01 7.50E+00 1.00E+00 2.74E-07 R 1
```

```
LINK  2 9.94E+02 4.00E+01 3.76E+02 5.00E+00 2.74E-07 S 1
LINK  2 2.08E+02 2.40E+01 2.42E+03 5.00E+00 2.74E-07 U 1
PKGSIZ
         CSNF      5.00
EOF
```

RUN DATE: [18-Aug-00 AT 12:08:26] PAGE 5

Rail - NE Region - 43GWd - 5-yr

INCIDENT-FREE SUMMARY
******** **** *******

INCIDENT-FREE POPULATION EXPOSURE IN PERSON-REM

	PASSENGR	CREW	HANDLERS	OFF LINK	ON LINK	STOPS	STORAGE	TOTALS
LINK 1	0.00E+00	5.48E-02	0.00E+00	1.14E-03	3.55E-04	0.00E+00	0.00E+00	5.63E-02
LINK 2	0.00E+00	2.59E-02	0.00E+00	2.37E-02	1.37E-03	1.13E-02	0.00E+00	6.22E-02
LINK 3	0.00E+00	1.62E-02	0.00E+00	1.10E-03	7.94E-04	0.00E+00	0.00E+00	1.81E-02
RURAL	0.00E+00	5.48E-02	0.00E+00	1.14E-03	3.55E-04	0.00E+00	0.00E+00	5.63E-02
SUBURB	0.00E+00	2.59E-02	0.00E+00	2.37E-02	1.37E-03	1.13E-02	0.00E+00	6.22E-02
URBAN	0.00E+00	1.62E-02	0.00E+00	1.10E-03	7.94E-04	0.00E+00	0.00E+00	1.81E-02
TOTALS:	0.00E+00	9.69E-02	0.00E+00	2.59E-02	2.51E-03	1.13E-02	0.00E+00	1.37E-01

MAXIMUM INDIVIDUAL IN-TRANSIT DOSE

LINK 1 5.36E-07 REM
LINK 2 5.36E-07 REM
LINK 3 5.36E-07 REM

RUN DATE: [18-Aug-00 AT 12:08:26] PAGE 6

Rail - NE Region - 43GWd - 5-yr

EXPECTED VALUES OF POPULATION RISK IN PERSON-REM

	GROUND	INHALED	RESUSPD	CLOUDSH	*INGESTION	TOTAL
LINK 1	8.58E-04	3.43E-05	9.27E-05	7.21E-08	0.00E+00	9.85E-04
LINK 2	1.28E-02	5.10E-04	1.38E-03	1.07E-06	0.00E+00	1.47E-02
LINK 3	1.04E-02	4.15E-04	1.12E-03	8.73E-07	0.00E+00	1.19E-02

RURAL	8.58E-04	3.43E-05	9.27E-05	7.21E-08	0.00E+00	9.85E-04
SUBURB	1.28E-02	5.10E-04	1.38E-03	1.07E-06	0.00E+00	1.47E-02
URBAN	1.04E-02	4.15E-04	1.12E-03	8.73E-07	0.00E+00	1.19E-02
TOTALS:	2.40E-02	9.59E-04	2.59E-03	2.02E-06	0.00E+00	2.76E-02

* NOTE THAT INGESTION RISK IS A SOCIETAL RISK;
THE USER MAY WISH TO TREAT THIS VALUE SEPARATELY.

Rail - NE Region - 43GWd - 5-yr

EXPECTED RISK VALUES - OTHER

LINK	ECON $$	EARLY FATALITY
1	0.00E+00	0.00E+00
2	0.00E+00	0.00E+00
3	0.00E+00	0.00E+00
TOTAL	0.00E+00	0.00E+00

TOTAL EXPOSED POPULATION: INCIDENT-FREE

LINK 1 4.02E+04 PERSONS
LINK 2 5.98E+05 PERSONS
LINK 3 8.05E+05 PERSONS

TOTAL 1.44E+06 PERSONS

TOTAL EXPOSED POPULATION: ACCIDENT
(PERSONS UNDER PLUME FOOTPRINT FOR A SINGLE ACCIDENT)

LINK 1 1.01E+04 PERSONS
LINK 2 5.08E+05 PERSONS
LINK 3 3.27E+06 PERSONS

EOI
END OF RUN

Appendix H

Spent-Fuel Radionuclide Inventories

Appendix H

Spent-Fuel Radionuclide Inventories

This appendix presents spent-fuel radionuclide inventories calculated by the ORIGEN-ARP computer code assuming the fuel power histories given in Appendix B and a five years cooling time. The radionuclide inventories used in the RADTRAN calculation of the risks associated with transportation accidents were based on these inventories assuming a full shipping cask and the appropriate uranium mass per fuel assembly. Table H.1 contains the inventory for PWR fuel, and Table H.2 contains the inventory for BWR fuel.

Table H.1 PWR Spent-Fuel Radionuclide Inventory (Bq/MTU) After Five Years of Cooling Time					
	Burnup (GWd/MTU)				
Isotope	**43**	**50**	**60**	**62**	**75**
Cr-51	1.98E-05	2.11E-05	1.85E-05	1.88E-05	2.06E-05
Mn-54	1.22E+12	1.30E+12	1.31E+12	1.31E+12	1.37E+12
Fe-55	6.59E+13	7.62E+13	8.88E+13	9.14E+13	1.08E+14
Fe-59	9.29E+00	1.00E+01	9.25E+00	9.47E+00	1.10E+01
Co-58	8.66E+06	8.70E+06	8.18E+06	8.21E+06	8.40E+06
Co-60	1.92E+14	2.23E+14	2.65E+14	2.72E+14	3.24E+14
Kr-85	3.34E+14	3.70E+14	4.14E+14	4.22E+14	4.66E+14
Sr-89	5.88E+05	5.37E+05	3.81E+05	3.69E+05	3.17E+05
Sr-90	3.50E+15	3.92E+15	4.40E+15	4.51E+15	5.03E+15
Y-91	2.33E+07	2.14E+07	1.56E+07	1.52E+07	1.33E+07
Zr-95	2.08E+08	1.98E+08	1.53E+08	1.50E+08	1.41E+08
Nb-95	4.74E+08	4.52E+08	3.49E+08	3.42E+08	3.22E+08
Ru-103	7.18E+02	7.25E+02	6.07E+02	6.11E+02	6.40E+02
Ru-106	8.25E+14	9.47E+14	1.04E+15	1.06E+15	1.18E+15
Sb-125	1.15E+14	1.33E+14	1.52E+14	1.55E+14	1.77E+14
Te-125m	2.82E+13	3.24E+13	3.71E+13	3.79E+13	4.34E+13
Te-127	6.07E+09	6.22E+09	5.51E+09	5.48E+09	5.62E+09
Te-127m	6.18E+09	6.33E+09	5.62E+09	5.59E+09	5.74E+09
Cs-134	1.45E+15	1.84E+15	2.39E+15	2.50E+15	3.24E+15
Cs-137	4.66E+15	5.40E+15	6.40E+15	6.59E+15	7.84E+15
Ce-141	9.32E-01	8.99E-01	6.99E-01	6.96E-01	6.77E-01
Ce-144	7.44E+14	7.40E+14	6.59E+14	6.44E+14	5.77E+14
Pm-147	2.23E+15	2.23E+15	2.11E+15	2.08E+15	1.91E+15
Eu-154	2.13E+14	2.70E+14	3.50E+14	3.66E+14	4.59E+14
Pu-238	1.46E+14	2.08E+14	3.08E+14	3.30E+14	4.66E+14
Pu-239	1.42E+13	1.44E+13	1.43E+13	1.42E+13	1.40E+13
Pu-240	1.77E+13	2.14E+13	2.52E+13	2.57E+13	2.84E+13
Pu-241	5.18E+15	5.18E+15	5.62E+15	5.74E+15	6.22E+15
Am-241	5.25E+13	5.33E+13	5.92E+13	6.07E+13	6.55E+13
Cm-242	1.02E+12	1.39E+12	1.80E+12	1.89E+12	2.35E+12
Cm-244	9.44E+13	1.87E+14	4.07E+14	4.63E+14	9.55E+14

Table H.2 BWR Spent-Fuel Radionuclide Inventory (Bq/MTU) After Five Years of Cooling Time						
	Burnup (GWd/MTU)					
Isotope	35	43	50	60	62	75
Cr-51	1.40E-05	1.41E-05	1.35E-05	1.22E-05	1.24E-05	1.33E-05
Mn-54	8.79E+11	9.44E+11	9.47E+11	8.92E+11	8.81E+11	8.92E+11
Fe-55	5.13E+13	6.14E+13	6.92E+13	7.77E+13	7.92E+13	9.03E+13
Fe-59	6.55E+00	6.81E+00	6.73E+00	6.33E+00	6.48E+00	7.47E+00
Co-58	7.14E+06	7.07E+06	6.81E+06	6.36E+06	6.40E+06	6.51E+06
Co-60	1.42E+14	1.73E+14	1.98E+14	2.28E+14	2.34E+14	2.72E+14
Kr-85	2.77E+14	3.25E+14	3.57E+14	3.92E+14	3.96E+14	4.26E+14
Sr-89	4.73E+05	3.96E+05	3.26E+05	2.31E+05	2.23E+05	1.80E+05
Sr-90	2.94E+15	3.47E+15	3.85E+15	4.33E+15	4.40E+15	4.88E+15
Y-91	1.84E+07	1.56E+07	1.31E+07	9.44E+06	9.14E+06	7.62E+06
Zr-95	1.57E+08	1.39E+08	1.21E+08	9.33E+07	9.15E+07	8.28E+07
Nb-95	3.57E+08	3.18E+08	2.76E+08	2.12E+08	2.08E+08	1.88E+08
Ru-103	4.88E+02	4.77E+02	4.40E+02	3.74E+02	3.77E+02	3.81E+02
Ru-106	5.43E+14	6.51E+14	7.10E+14	7.33E+14	7.36E+14	7.84E+14
Sb-125	8.64E+13	1.04E+14	1.16E+14	1.26E+14	1.28E+14	1.38E+14
Te-125m	2.11E+13	2.54E+13	2.83E+13	3.08E+13	3.12E+13	3.37E+13
Te-127	4.06E+09	4.11E+09	3.92E+09	3.39E+09	3.36E+09	3.43E+09
Te-127m	4.17E+09	4.22E+09	4.03E+09	3.46E+09	3.43E+09	3.50E+09
Cs-134	9.67E+14	1.32E+15	1.64E+15	2.04E+15	2.12E+15	2.67E+15
Cs-137	3.78E+15	4.59E+15	5.29E+15	6.25E+15	6.44E+15	7.62E+15
Ce-141	6.87E-01	6.22E-01	5.44E-01	4.29E-01	4.26E-01	4.00E-01
Ce-144	5.57E+14	5.51E+14	5.07E+14	4.14E+14	4.00E+14	3.41E+14
Pm-147	1.96E+15	2.03E+15	1.97E+15	1.80E+15	1.76E+15	1.57E+15
Eu-154	1.52E+14	2.12E+14	2.67E+14	3.43E+14	3.59E+14	4.59E+14
Pu-238	1.00E+14	1.51E+14	2.13E+14	3.17E+14	3.39E+14	4.85E+14
Pu-239	1.29E+13	1.35E+13	1.34E+13	1.32E+13	1.32E+13	1.29E+13
Pu-240	1.66E+13	2.07E+13	2.36E+13	2.67E+13	2.72E+13	2.95E+13
Pu-241	3.50E+15	4.33E+15	4.77E+15	5.33E+15	5.44E+15	5.77E+15
Am-241	3.62E+13	4.59E+13	5.18E+13	5.96E+13	6.11E+13	6.55E+13
Cm-242	7.39E+11	1.12E+12	1.48E+12	1.95E+12	2.05E+12	2.54E+12
Cm-244	5.44E+13	9.69E+13	1.87E+14	3.96E+14	4.51E+14	9.44E+14

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

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11. ABSTRACT (200 words or less)
In 1988, the U.S. Nuclear Regulatory Commission published an environmental assessment (EA) of the effects of increasing nuclear reactor peak-rod fuel irradiation (burnup) up to 60 gigawatt days per metric ton of uranium (GWd/MTU). The EA was based, in large part, on the evaluation of environmental impacts of extended fuel burnup in NUREG/CR-5009. This report updates the information in NUREG/CR-5009 using current fuel designs, fuel performance data, and dose computational methods. It contains a best-estimate assessment of the environmental and economic impacts of extending peak-rod burnup above 60 GWd/MTU. For those aspects of this assessment not significantly affected by the gap-release fraction, the findings indicate that there are no significant adverse environmental impacts associated with extending peak-rod burnup to 75 GWd/MTU. For those aspects affected by the gap-release fraction, the findings in the report indicate that there are no significant adverse environmental impacts associated with extending peak-rod burnup to 62 GWd/MTU.

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