6.0 Fuel Cycle, Transportation, and Decommissioning

This chapter addresses the environmental impacts from (1) the uranium fuel cycle and solid waste management, (2) transportation of radioactive material, and (3) decommissioning for the proposed Grand Gulf early site permit (ESP) site. Distinctions between the impacts of advanced light-water reactor (LWR) designs and the gas-cooled reactor designs are discussed. |

In its evaluation of uranium fuel cycle impacts for the Grand Gulf ESP site, System Energy Resources, Inc. (SERI) used the plant parameter envelope (PPE) approach for the advanced LWR designs but not for the two gas-cooled reactors. In its evaluation of the impacts from transportation of radioactive materials, SERI did not use the PPE approach, but rather evaluated each reactor design individually. An applicant for a construction permit (CP) or a combined license (COL) referencing the Grand Gulf ESP would, therefore, have to perform a new evaluation if a different design is proposed at that stage.

6.1 Fuel Cycle Impacts and Solid Waste Management

This section discusses the environmental impacts from the uranium fuel cycle and solid waste management for both the advanced LWR designs and gas-cooled reactor designs. The impacts of the two types of design are presented separately because Title 10 of the Code of Federal Regulations (CFR), Section 51.51 (10 CFR 51.51) provides specific criteria for evaluating the environmental impacts for only LWR designs. Consequently, issues related to fuel cycle impacts and solid waste management are not resolved because of the lack of data to validate impacts from gas-cooled designs.

6.1.1 Light Water Reactors

The regulations in 10 CFR 51.51(a) state that

> Every environmental report prepared for the construction permit stage of a light water cooled nuclear power reactor, and submitted on or after September 4, 1979, shall take Table S–3, Table of Uranium Fuel Cycle Environmental Data, as the basis for evaluating the contribution of the environmental effects of uranium mining and milling, the production of uranium hexafluoride, isotopic enrichment, fuel fabrication, reprocessing of irradiated fuel, transportation of radioactive materials and management of low level wastes and high level wastes related to uranium fuel cycle activities to the environmental costs of licensing the nuclear power reactor. Table S–3 shall be included in the environmental report and may be supplemented by a discussion of the environmental significance of the data set forth in the table as weighed in the analysis for the proposed facility.

The PPE for the new unit or units at the Grand Gulf ESP site uses the bounding input parameters from the following LWR designs:

- Advanced CANDU (CANada Deuterium Uranium Reactor) (ACR-700) – This reactor, developed by Atomic Energy Canada Limited, is an evolutionary extension of CANDU 6 plant using very slightly enriched uranium fuel and light water coolant.

- Advanced Boiling Water Reactor (ABWR) – This reactor, developed by General Electric Company, is a standardized plant that has been certified under the U.S. Nuclear Regulatory Commission (NRC) requirements in 10 CFR Part 52 (Appendix A). The ABWR is fueled with slightly enriched uranium and uses a light water cooling system.

- Advanced Pressurized Water Reactor (AP1000) – This is an earlier version of the AP1000 reactor final design developed by Westinghouse Electric Company and subsequently approved by the NRC, using slightly enriched uranium and a light water cooling system. This design is not the AP1000 that has received final design approval from the NRC; therefore, this design will be referred to as the “surrogate AP1000.”

- Economic Simplified Boiling Water Reactor (ESBWR) – This reactor, developed by General Electric Company, is fueled with slightly enriched uranium and uses a light water cooling system.

- International Reactor Innovative and Secure (IRIS) next-generation pressurized water reactor (PWR) – This reactor, under development by a consortium led by Westinghouse Electric Company, is a modular LWR.

These light water designs all use uranium dioxide fuel; therefore, Table S–3 (10 CFR 51.51(b)) can be used to assess environmental impacts. Table S–3 values are normalized for a reference 1000-MW(e) LWR at an 80 percent capacity factor. The 10 CFR 51.51(b) Table S–3 values are reproduced in Table 6-1. The PPE power rating for the Grand Gulf ESP site is 8600 MW(t) with a net electrical output of up to 3000 MW(e) (SERI 2005a).

Specific categories of natural resource use are included in Table S–3 (see Table 6-1). These categories relate to land use, water consumption and thermal effluents, radioactive releases, burial of transuranic and high-level and low-level wastes, and radiation doses from transportation and occupational exposures. In developing Table S–3, the staff considered two fuel cycle options–no recycle and uranium-only recycle–that differed in the treatment of spent fuel removed from a reactor. “No recycle” treats all spent fuel as waste to be stored at a Federal waste repository; “uranium-only recycle” involves reprocessing spent fuel to recover unused uranium and return it to the system. Neither cycle involves the recovery of plutonium. The contributions in Table S–3 resulting from reprocessing, waste management, and transportation of wastes are maximized for both of the two fuel cycles (uranium-only and no recycle); that is, the identified environmental impacts are based on the cycle that results in the greater impact. The uranium fuel cycle is defined as the total of those operations and processes associated with provision, utilization, and ultimate disposition of fuel for nuclear power reactors.

Table 6-1. Table S–3 from 10 CFR 51.51(b), Table of Uranium Fuel Cycle Environmental Data[1]

Environmental considerations	Total	Maximum effect per annual fuel requirement or reference reactor year of model 1,000 MWe LWR
Natural Resource Use		
Land (acres):		
Temporarily committed[2]	100	
Undisturbed area	79	
Disturbed area	22	Equivalent to a 110 MWe coal-fired power plant.
Permanently committed	13	
Overburden moved (millions of MT)	2.8	Equivalent to 95 MWe coal-fired power plant.
Water (millions of gallons):		
Discharged to air	160	=2 percent of model 1,000 MWe LWR with cooling tower.
Discharged to water bodies	11,090	
Discharged to ground	127	
Total	11,377	<4 percent of model 1,000 MWe LWR with once-through cooling.
Fossil fuel:		
Electrical energy (thousands of MW-hr)	323	<5 percent of model 1,000 MWe LWR output.
Equivalent coal (thousands of MT)	118	Equivalent to the consumption of a 45 MWe coal-fired power plant.
Natural gas (millions of standard cubic feet)	135	<0.4 percent of model 1,000 MWe energy output.
Effluents–Chemical (MT)		
Gases (including entrainment):[3]		
SO_x	4,400	
NO_x[4]	1,190	Equivalent to emissions from 45 MWe coal-fired plant for a year.
Hydrocarbons	14	
CO	29.6	
Particulates	1,154	

Table 6-1. (contd)

Environmental considerations	Total	Maximum effect per annual fuel requirement or reference reactor year of model 1,000 MWe LWR
Other gases:		
F	.67	Principally from UF_6 production, enrichment, and reprocessing. Concentration within range of state standards—below level that has effects on human health.
HCl	.014	
Liquids:		
SO^{-}_4	9.9	From enrichment, fuel fabrication, and reprocessing steps. Components that constitute a potential for adverse environmental effect are present in dilute concentrations and receive additional dilution by receiving bodies of water to levels below permissible standards. The constituents that require dilution and the flow of dilution water are: NH_3–600 cfs., NO_3–20 cfs., Fluoride–70 cfs.
NO^{-}_3	25.8	
Fluoride	12.9	
Ca^{++}	5.4	
Cl^{-}	8.5	
Na^{+}	12.1	
NH_3	10.0	
Fe	.4	
Tailings solutions (thousands of MT)	240	From mills only—no significant effluents to environment.
Solids	91,000	Principally from mills-no significant effluents to environment.
Effluents—Radiological (curies)		
Gases (including entrainment):		
Rn–222	....	Presently under reconsideration by the Commission.
Ra–226	.02	
Th–230	.02	
Uranium	.034	
Tritium (thousands)	18.1	
C–14	24	
Kr–85 (thousands)	400	
Ru–106	.14	Principally from fuel reprocessing plants.
I–129	1.3	
I–131	.83	
Tc–99	....	Presently under consideration by the Commission.
Fission products and transuranics	.203	
Liquids:		
Uranium and daughters	2.1	Principally from milling—included tailings liquor and returned to ground—no effluents; therefore, no effect on environment.
Ra–226	.0034	From UF_6 production.
Th–230	.0015	
Th–234	.01	From fuel fabrication plants—concentration 10 percent of 10 CFR 20 for total processing 26 annual fuel requirements for model LWR.
Fission and activation products	$5.9x10^{-6}$	

Table 6-1. (contd)

Environmental considerations	Total	Maximum effect per annual fuel requirement or reference reactor year of model 1,000 MWe LWR
Solids (buried on site):		
Other than high level (shallow)	11,300	9,100 Ci comes from low level reactor wastes and 1,500 Ci comes from reactor decontamination and decommissioning—buried at land burial facilities. 600 Ci comes from mills—included in tailings returned to ground. Approximately 60 Ci comes from conversion and spent fuel storage. No significant effluent to the environment.
TRU and HLW (deep)	1.1×10^7	Buried at Federal Repository.
Effluents—thermal (billions of British thermal units)	4,063	<5 percent of model 1,000 MWe LWR
Transportation (person-rem):		
Exposure of workers and general public	2.5	
Occupational exposure (person-rem)	22.6	From reprocessing and waste management.

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

Data supporting this table are given in the “Environmental Survey of the Uranium Fuel Cycle,” WASH–1248, April 1974; the “Environmental Survey of the Reprocessing and Waste Management Portion of the LWR Fuel Cycle,” NUREG–0116 (Supp.1 to WASH–1248, NRC 1976); the “Public Comments and Task Force Responses Regarding the Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle,” NUREG–0216 (Supp. 2 to WASH–1248) (NRC 1977b); and in the record of the final rulemaking pertaining to Uranium Fuel Cycle Impacts from Spent Fuel Reprocessing and Radioactive Waste Management, Docket RM–50–3. The contributions from reprocessing, waste management and transportation of wastes are maximized for either of the two fuel cycles (uranium only and no recycle). The contribution from transportation excludes transportation of cold fuel to a reactor and of irradiated fuel and radioactive wastes from a reactor which are considered in Table S–4 of §51.20(g). The contributions from the other steps of the fuel cycle are given in columns A-E of Table S–3A of WASH-1248.

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

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

[4] 1.2 percent from natural gas use and process.

During the Carter administration, the Nuclear Nonproliferation Act of 1978, Pub. L. No. 95-242 (22 USC 3201 *et seq.*), was enacted; it significantly impacted the disposition of spent nuclear fuel by indefinitely deferring the commercial reprocessing and recycling of plutonium produced in the U.S. commercial nuclear power program. While the ban on the reprocessing of spent fuel was lifted during the Reagan administration, economic circumstances changed, reserves of uranium ore increased, and the stagnation of the nuclear power industry provided little incentive for industry to resume reprocessing. During the 109th Congress, the Energy Policy Act of 2005, Pub. L. No. 109-58 (119 Stat. 594 [2005]), was enacted. It authorized the U.S. Department of Energy (DOE) to conduct an advanced fuel recycling technology research and development program to evaluate proliferation-resistant fuel recycling and transmutation technologies that minimize environmental or public health and safety impacts. Consequently, while Federal policy does not prohibit reprocessing, additional DOE efforts would be required before commercial reprocessing and recycling of spent nuclear fuel produced from U.S. commercial nuclear power plants could commence.

The no-recycle option is presented schematically in Figure 6-1. Natural uranium is mined in either open-pit or underground mines or by an in situ leach solution mining process. In situ leach mining, the primary form of mining in the United States today, involves injecting a lixiviant solution into the uranium ore body to dissolve uranium and then pumping the solution to the surface for further processing. The ore or in situ leach solution is transferred to mills where it is processed to produce uranium oxide or “yellowcake.” A conversion facility prepares the uranium oxide by converting it to uranium hexafluoride, which is then processed at an enrichment facility to increase the percentage of the more fissile isotope uranium-235 and decrease the percentage if the non-fissile isotope uranium-238. At a fuel-fabrication facility, the enriched uranium, which is approximately 5 percent uranium-235, is then converted to uranium dioxide (UO_2). The UO_2 is pelletized, sintered, and inserted into tubes to form fuel assemblies. The fuel assemblies are placed in the reactor to produce power. When the content of the uranium-235 reaches a point where the nuclear reactor has become inefficient with respect to neutron economy, the fuel assemblies are withdrawn from the reactor. After onsite storage for sufficient time to allow for short-lived fission product decay and to reduce the heat generation rate, the fuel assemblies would be transferred to a waste repository. Disposal of spent fuel elements in a repository constitutes the final step in the no-recycle option.

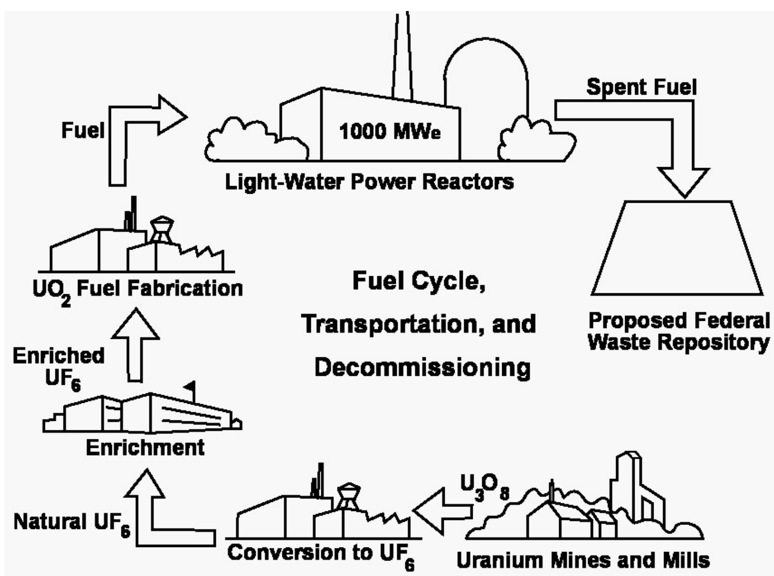

Figure 6-1. The Uranium Fuel Cycle: No-Recycle Option (derived from NRC 1999)

The following assessment of the environmental impacts of the fuel cycle as related to the operation of the proposed project is based on the values given in Table S–3 (see Table 6-1) and the staff's analysis of the radiological impact from radon and technetium. In the *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* (GEIS) (NRC 1996), the staff provides a detailed analysis of the environmental impacts from the uranium fuel cycle. Although the GEIS is specific to the impacts related to license renewal, the information is relevant to this review because the advanced LWR designs considered here use the same type of fuel; the staff's analyses in Section 6.2.3 of the GEIS are summarized and incorporated by reference here.

The fuel cycle impacts in Table S–3 are based on a reference 1000-MW(e) LWR operating at an annual capacity factor of 80 percent for a net electric output of 800 MW(e). In the following review and evaluation of the environmental impacts of the fuel cycle, the staff used the stated capacity factor in the SERI PPE of 96 percent with a total net electric output of 3000 MW(e) for the ESP site (SERI 2005a); this results in approximately four times the impact values in Table S–3 (see Table 6-1). Throughout Chapter 6, this will be referred to as the 1000-MW(e) LWR scaled model, reflecting 3000 MW(e) for the site.

Recent changes in the fuel cycle may have some bearing on environmental impacts; however, as discussed below, the staff is confident that the contemporary fuel cycle impacts below are those identified in Table S–3.

The values in Table S–3 were calculated from industry averages for the performance of each type of facility or operation within the fuel cycle. Recognizing that this approach meant that there would be a range of reasonable values for each estimate, the staff followed the policy of choosing the assumptions or factors to be applied so that the calculated values would not be underestimated. This approach was intended to ensure that the actual environmental impacts would be less than the quantities shown in Table S–3 for all LWR nuclear power plants within the widest range of operating conditions. Many subtle fuel cycle parameters and interactions were recognized by the staff as being less than the precision of the estimates and were not considered or were considered but had no effect on the Table S–3 calculations. For example, to determine the quantity of fuel required for a year's operation of a nuclear power plant in Table S–3, the staff defined the model reactor as a 1000-MW(e) light water reactor operating at 80-percent capacity with a 12-month fuel reloading cycle and an average fuel burnup of 33,000 MWd/MTU. This is a "reactor reference year" or "reference reactor year" depending on the source (either Table S–3 or the GEIS), but it has the same meaning. The sum of the initial fuel loading plus all of the reloads for the lifetime of the reactor can be divided by the now more likely 60-year (40-year initial operating license term and 20-year license renewal term) lifetime to obtain an average annual fuel requirement. This was done for both boiling water reactors (BWRs) and PWRs; the higher annual requirement, 35 metric tonnes (MT) of uranium made into fuel for a BWR, was chosen in the GEIS as the basis for the reference reactor year. A number of fuel management improvements have been adopted by nuclear power plant

| managers to achieve higher performance and to reduce fuel and separative work (enrichment) requirements. Since Table S–3 was promulgated, these improvements have reduced the annual fuel requirement.

Another change is the elimination of the U.S. restrictions on the importation of foreign uranium.
| The economic conditions of the uranium market favor utilization of foreign uranium at the expense of the domestic uranium industry. These market conditions have forced the closing of most U.S. uranium mines and mills, substantially reducing the environmental impacts in the
| United States from these activities. Factoring in changes to the fuel cycle suggests that the
| environmental impacts on mining and tail millings could drop levels below those given in
| Table S–3; however, Table S–3 estimates have not been reduced.

| Section 6.2 of the GEIS discusses the sensitivity to recent changes in the fuel cycle on the environmental impacts in greater detail.

6.1.1.1 Land Use

The total annual land requirement for the fuel cycle supporting the 1000-MW(e) LWR scaled
| model is about 183 ha (452 ac). Approximately 20 ha (52 ac) are permanently committed land,
| and 162 ha (400 ac) are temporarily committed. A "temporary" land commitment is a commitment for the life of the specific fuel cycle plant (e.g., a mill, enrichment plant, or succeeding plants). Following completion of decommissioning, such land can be released for unrestricted use. "Permanent" commitments represent land that may not be released for use after plant shutdown and/or decommissioning because decommissioning activities do not result in removal of sufficient radioactive material to meet the limits in 10 CFR Part 20, Subpart E for
| release of that area for unrestricted use. Of the 162 ha (400 ac) of temporarily committed land,
| 113 ha (280 ac) are undisturbed and 49 ha (120 ac) are disturbed (SERI 2005a). In
| comparison, a coal-fired power plant with the same MW(e) output as the LWR scaled model
| and that uses strip-mined coal requires the disturbance of about 324 ha (800 ac) per year for
| fuel alone. The staff concludes that the impacts on land use to support the 1000-MW(e) LWR scaled model would be small.

6.1.1.2 Water Use

| The principal water use for the fuel cycle supporting a 1000-MW(e) LWR scaled model is that required to remove waste heat from the power stations supplying electrical energy to the enrichment step of this cycle. Scaling from Table S–3, of the total annual water use
| of 1.72×10^8 m^3 (4.55×10^{10} gal), about 1.7×10^8 m^3 (4.44×10^{10} gal) are required for the removal of waste heat, assuming that these plants use once-through cooling. Other water

uses involve the discharge to air (e.g., evaporation losses in process cooling) of about 2.42×10^6 m^3/yr (6.40×10^8 gal/yr) and water discharged to ground (e.g., mine drainage) of about 1.92×10^6 m^3/yr (5.08×10^8 gal/yr).

On a thermal effluent basis, annual discharges from the nuclear fuel cycle are about 4 percent of the 1000-MW(e) LWR scaled model using once-through cooling. The consumptive water use of 2.42×10^6 m^3/yr (6.40×10^8 gal/yr) is about 2 percent of the 1000-MW(e) LWR scaled model using cooling towers. The maximum consumptive water use (assuming that all plants supplying electrical energy to the nuclear fuel cycle use cooling towers) would be about 6 percent of the 1000-MW(e) LWR scaled model using cooling towers. Under this condition, thermal effluents would be negligible. The staff concludes that the impacts on water use for these combinations of thermal loadings and water consumption would be small relative to the water use and thermal discharges.

6.1.1.3 Fossil Fuel Impacts

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

6.1.1.4 Chemical Effluents

The quantities of chemical, gaseous, and particulate effluents from fuel cycle processes are given in Table S–3 (see Table 6-1) for the reference 1000-MW(e) LWR. The quantities of effluents would be approximately four times greater for the reference 1000-MW(e) LWR scaled model. The principal effluents are SO_x, NO_x, and particulates. Based on data in *The Seventh Annual Report of the Council on Environmental Quality*, these emissions constitute a small additional atmospheric loading in comparison with the emissions from the stationary fuel combustion and transportation sectors in the United States. The fuel cycle emissions constitute about 0.08 percent of the annual national releases for each of these effluents (CEQ 1976).

Liquid chemical effluents produced in fuel cycle processes are related to fuel enrichment and fabrication and may be released to receiving waters. These effluents are usually present in dilute concentrations such that only small amounts of dilution water are required to reach levels of concentration that are within established standards. Table S–3 (see Table 6-1) specifies the amount of dilution water required for specific constituents. Additionally, all liquid discharges into the navigable waters of the United States from plants associated with the fuel cycle operations will be subject to requirements and limitations set by an appropriate Federal, State, regional, local, or affected Native American Tribal regulatory agency.

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

The staff determined that the impacts of these chemical effluents would be small.

6.1.1.5 Radioactive Effluents

Radioactive effluents estimated to be released to the environment from waste management activities and certain other phases of the fuel cycle process are set forth in Table S–3 (see Table 6-1). Using these data, the staff calculated for 1 year of operation of the 1000-MW(e) LWR scaled model, the 100-year involuntary environmental dose commitment to the U.S. population from the LWR-supporting fuel cycle. These calculations estimate that the overall whole body gaseous dose commitment to the U.S. population from the fuel cycle (excluding reactor releases and the dose commitments resulting from radon-222 and technetium-99) would be approximately 16 person-Sv (1600 person-rem) per year of operation of the 1000-MW(e) LWR scaled model; this reference reactor year is scaled to reflect the total electric power rating for the site for a year.

The additional whole body dose commitment to the U.S. population from radioactive liquid effluents from all fuel cycle operations other than reactor operation would be approximately 8 person-Sv (800 person-rem) per year of operation. Thus, the estimated 100-year environmental dose commitment to the U.S. population from radioactive gaseous and liquid releases because of these portions of the fuel cycle is approximately 24 person-Sv (2400 person-rem) to the whole body per reference reactor year.

Currently, the radiological impacts associated with radon-222 and technetium-99 release are not addressed in Table S–3. Principal radon releases occur during mining and milling operations and as emissions from mill tailings, whereas principal technetium-99 releases occur from gaseous diffusion enrichment facilities. SERI provided an assessment of radon-222 and technetium-99 in its response to a request for additional information on February 3, 2005 (SERI 2005b). The staff's evaluation in this environmental impact statement (EIS) relied on the information discussed in the GEIS.

In Section 6.2 of the GEIS, the staff estimated the radon-222 releases from mining and milling operation, and from mill tailings for each year of operations of the reference 1000-MW(e) LWR. The estimated releases of radon-222 for the reference reactor year for the 1000-MW(e) LWR scaled model, or for the total electric power rating for the site for a year, is approximately 7.7×10^{14} Bq (20,800 Ci). Of this total, about 78 percent would be from mining, 15 percent from milling operations, 7 percent from inactive tails prior to stabilization. For radon releases from stabilized tailings, the staff assumed that the scaled model would result in an emission of 1.5×10^{11} Bq (4 Ci) per site year; i.e., four times the GEIS estimate for the reference reactor year. The major risks from radon-222 are from exposure to the bone and the lung, although there is a small risk from exposure to the whole body. The organ-specific dose weighting factors from 10 CFR 20.1003 were applied to the bone and lung doses to estimate the 100-year dose commitment from radon-222 to the whole body. The estimated 100-year environmental dose commitment from mining, milling, and tailings prior to stabilization for each site year (assuming the 1000-MW(e) LWR scaled model) would be approximately 37 person-Sv (3700 person-rem) to the whole body. From stabilized tailings piles, the estimated 100-year environmental dose commitment would be approximately 0.71 person-Sv (71 person-rem) to the whole body. Additional insights regarding Federal policy/resource perspectives concerning institutional controls comparisons with routine radon-222 exposure and risk and long-term releases from stabilized tailings piles are discussed in the GEIS. SERI provided an assessment of radon-222 and technetium-99 in its response to a request for additional information on February 3, 2005 (SERI 2005b). The evaluation in this EIS relied on the information discussed in the GEIS.

Also as discussed in the GEIS, the staff considered the potential health effects associated with the releases of technetium-99. The estimated releases of technetium-99 for the reference reactor year for the 1000-MW(e) LWR scaled model are 1.1×10^{9} Bq (0.03 Ci) from chemical processing of recycled uranium hexafluoride before it enters the isotope enrichment cascade and 7.4×10^{8} Bq (0.02 Ci) into the groundwater from a potential repository. The major risks from technetium-99 are from exposure of the gastrointestinal tract and kidney, although there is a small risk from exposure to the whole body. Applying the organ-specific dose weighting factors from 10 CFR 20.1003 to the gastrointestinal tract and kidney doses, the total-body 100-year dose commitment from technetium-99 to the whole body was estimated to be 4 person-Sv (400 person-rem) for the 1000-MW(e) LWR scaled model.

Although radiation may cause cancers at high doses and high dose rates, currently there are no data that unequivocally establish the occurrence of cancer following exposure to low doses below about 100 mSv (10,000 mrem) and at low dose rates. However, radiation protection experts conservatively assume that any amount of radiation may pose some risk of causing cancer or a severe hereditary effect and that the risk is higher for higher radiation exposures. Therefore, a linear, no-threshold dose response model is used to describe the relationship between radiation dose and detriments such as cancer induction. A recent report by the National Research Council (2006), the BEIR VII report, supports the linear, no-threshold dose response model. Simply put, the theory states that any increase in dose, no matter how small, results in an incremental increase in health risk. This theory is accepted by the NRC as a conservative model for estimating health risks from radiation exposure, recognizing that the model probably overestimates those risks.

Based on this model, the staff estimated the risk to the public from radiation exposure using the nominal probability coefficient for total detriment (730 fatal cancers, nonfatal cancers, and severe hereditary effects per 10,000 person-Sv (1,000,000 person-rem)) from International Commission on Radiological Protection Publication 60 (ICRP 1991). This coefficient was multiplied by the sum of the estimated whole body population doses discussed above, approximately 66 person-Sv/yr (6600 person-rem/yr), to calculate that the U.S. population would incur a total of approximately 4.8 fatal cancers, nonfatal cancers, and severe hereditary effects annually. This risk is very small compared to the number of fatal cancers, nonfatal cancers, and severe hereditary effects that would be estimated to the U.S. population annually from exposure to natural sources of radiation using the same risk estimation method.

Radon releases from tailings are indistinguishable from background radiation levels at a few kilometers from the tailings pile (at less than 1 km in some cases) (NRC 1978). The public dose limit in the U.S. Environmental Protection Agency's (EPA's) regulation, 40 CFR 190.10(a), is 0.25 mSv/yr (25 mrem/yr) to the whole body from the entire fuel cycle, but most NRC licensees have airborne effluents resulting in doses of less than 0.01 mSv/yr (1 mrem/yr) (61 FR 65120).

In addition, at the request of the U.S. Congress, the National Cancer Institute conducted a study and published "Cancer in Populations Living Near Nuclear Facilities" in 1990 (NCI 1990). This report included an evaluation of health statistics around all nuclear power plants, as well as several other nuclear fuel cycle facilities, in operation in the United States in 1981 and found "no evidence that an excess occurrence of cancer has resulted from living near nuclear facilities" (NCI 1990). The contribution to the annual average dose received by an individual from the fuel-cycle related radiation and other sources as reported in the National Council on Radiation Protection and Measurements (NCRP) Report 93 (NCRP 1987) is listed in Table 6-2. The nuclear fuel cycle contribution to an individual's annual average radiation is extremely small (less than 0.01 mSv (1 mrem) per year).

Based on the analyses presented above, the staff concludes that the environmental impacts of radioactive effluents from the fuel cycle are small.

6.1.1.6 Radioactive Waste

The quantities of buried radioactive waste material (low-level, high-level, and transuranic wastes) are specified in Table S–3 (see Table 6-1). For low-level waste disposal at land burial facilities, the Commission notes in Table S–3 that there will be no significant radioactive

Table 6-2. Comparison of Annual Average Dose Received by an Individual from All Sources

Source	Dose (mSv/yr)[a]	Percent of Total
Natural		
Radon	2	55
Cosmic	0.27	8
Terrestrial	0.28	8
Internal (body)	0.39	11
Total natural sources	**3**	**82**
Artificial		
Medical x-ray	0.39	11
Nuclear medicine	0.14	4
Consumer products	0.10	3
Total artificial sources	**0.63**	**18**
Other		
Occupational	0.009	<0.30
Nuclear fuel cycle	<0.01	<0.03
Fallout	<0.01	<0.03
Miscellaneous sources	<0.01	<0.03

(a) Multiply millisievert (mSv)/yr by 100 to obtain millirem/yr.
Source: NCRP 1987

releases to the environment. For high-level and transuranic wastes, the Commission notes that these are to be buried at a repository, such as the candidate repository at Yucca Mountain, Nevada, and that no release to the environment is expected to be associated with such disposal, because it has been assumed that all of the gaseous and volatile radionuclides contained in the spent fuel are released to the atmosphere before the disposal of the waste. In NUREG-0116 (NRC 1976), which provides background and context for the high-level and transuranic Table S–3 values established by the Commission, the staff indicates that these high-level and transuranic wastes will be buried and will not be released to the environment.

On February 15, 2002, subsequent to receipt of a recommendation by Secretary Abraham, DOE, the President recommended the Yucca Mountain site for the development of a repository for the geologic disposal of spent nuclear fuel and high-level nuclear waste (White House Press Release 2002).

The EPA developed Yucca Mountain-specific repository standards, which were subsequently adopted by the NRC in 10 CFR Part 63. In an opinion, issued July 9, 2004, the U.S. Court of Appeals for the District of Columbia Circuit (the Court) vacated EPA's radiation protection standards for the candidate repository, which required compliance with certain dose limits over a 10,000-year period (U.S. Court of Appeals 2004). The Court's decision also vacated the compliance period in NRC's licensing criteria for the candidate repository in 10 CFR Part 63. In response to the Court's decision, EPA issued its proposed revised standards on August 22, 2005, that would revise the radiation protection standards for the candidate repository (70 FR 49014). In order to be consistent with EPA's revised standards, NRC proposed revisions to 10 CFR Part 63 on September 8, 2005 (70 FR 53313). The 10 CFR Part 63 rulemaking is titled "Implementation of a Dose Standard after 10,000 years," and the comment period was extended to December 7, 2005. The proposed standards are 0.15 mSv (15 mrem) per year for 10,000 years following disposal and 3.5 mSv (350 mrem) per year for 10,000 years through 1 million years after disposal. RIN 3150 will not be finalized by the time this EIS is issued.

Consequently at this time, for the high-level waste and spent fuel disposal component of the fuel cycle, there is some uncertainty with respect to regulatory limits for offsite releases of radionuclides for the current candidate repository site. However, prior to promulgation of the affected provisions of the Commission's regulations, the staff assumed that limits are developed along the line of the 1995 National Academy of Sciences report, *Technical Bases for Yucca Mountain Standards*, and that in accordance with the Commission's Waste Confidence Decision, 10 CFR 51.23, a repository can and likely will be developed at some site, which would comply with such limits, with peak doses to virtually all individuals of 1 mSv (100 mrem) per year or less (NAS 1995; NRC 1996).

Despite any uncertainty with respect to these rules, some judgement as to National Environmental Policy Act of 1969 (NEPA) implications of offsite radiological impacts of spent fuel and high-level waste disposal should be made. For the reasons stated above, the staff concludes that the environmental impacts of radioactive waste disposal would be small.

6.1.1.7 Occupational Dose

In the review and evaluation of the environmental impacts of the fuel cycle, the staff considered the higher capacity factor in the PPE of 96 percent with a total net electric output of 3000 MW(e) for the ESP site (SERI 2005a). This case is referred to as the 1000-MW(e) LWR scaled model. The annual occupational dose attributable to all phases of the fuel cycle for the 1000-MW(e) LWR scaled model is about 24 person-Sv (2400 person-rem). This is based on a 6 person-Sv (600 person-rem) occupational dose estimate attributable to all phases of the fuel cycle for the model 1000 MW(e) LWR (NRC 1996). The environmental impact from this occupational dose is considered small because the dose to any individual worker is maintained within the dose limits of 10 CFR Part 20, which is 0.05 Sv/yr (5 rem/yr).

6.1.1.8 Transportation

The transportation dose to workers and the public totals about 0.025 person-Sv (2.5 person-rem) annually for the reference 1000-MW(e) LWR per Table S–3 (see Table 6-1).

This corresponds to a dose of 0.1 person-Sv (10 person-rem) for the 1000-MW(e) LWR scaled model. For comparative purposes, the estimated collective dose from natural background radiation to the population within 80 km (50 mi) of the Grand Gulf ESP site is 1020 person-Sv/yr (102,000 person-rem/yr). On this basis of this comparison, the staff concludes that environmental impacts of transportation would be small.

6.1.1.9 Conclusion

The staff evaluated the environmental impacts of the uranium fuel cycle as given in Table S–3 (see Table 6-1), considered the effects of radon-222 and technetium-99, and appropriately scaled the impacts for the 1000-MW(e) LWR scaled model. Based on this evaluation, the staff concludes that the impacts would be SMALL, and mitigation would not be warranted. The staff will verify the continued applicability of all assumptions should an applicant for a CP or COL reference the Grand Gulf ESP.

6.1.2 Gas-Cooled Reactors

As noted earlier, issues related to reactors based on non-LWR designs are not resolved because of the lack of information to validate values and impacts. However, the following analyses were performed using data from SERI for the purposes of estimation only.

The gas-cooled reactors analyzed for the uranium fuel cycle are:

- Gas Turbine Modular Helium Reactor (GT-MHR) – This reactor, developed by General Atomics, is a modular helium-cooled graphite-moderated reactor.

- Pebble Bed Modular Reactor (PBMR) – This reactor, developed by PBMR (Pty) Ltd., is a modular graphite-moderated helium-cooled gas turbine reactor.

Table S–3 from 10 CFR 51.51(a) can be used as a basis for bounding the environmental impacts from the uranium fuel cycle only for LWRs. SERI performed an assessment of the environmental impacts of the fuel cycle for gas-cooled reactor designs by comparing key parameters for these reactor designs to those used to generate the impacts in Table S–3 (SERI 2005a). Key parameters are energy usage, material involved, and number of shipments for each major fuel cycle activity (i.e., mining, milling, conversion, enrichment, fuel fabrication, and radioactive waste disposal). SERI sought to demonstrate in its environmental report that the impacts for the gas-cooled reactor designs were comparable to the environmental impacts identified in the technical basis document, WASH-1248, *Environmental Summary of the Uranium Fuel Cycle*, (AEC 1974) and its Supplement 1 (NUREG-0116) (NRC 1976) for Table S–3.

As discussed in Section 6.1.1, the fuel cycle impacts in Table S–3 (see Table 6-1) were based on a reference 1000-MW(e) LWR operating at an annual capacity factor of 80 percent for a net electric output of 800 MW(e). This is termed the "reference reactor year." For the purposes of evaluating fuel cycle impacts for the Grand Gulf ESP site, it was assumed that the additional LWR site-wide fuel impacts would be based on a total net electric output of 3000 MW(e) at 96 percent annual capacity factor. This was termed the 1000-MW(e) LWR scaled model and resulted in a factor about four times (i.e., 3000/800) the impacts in Table S–3.

One of the other-than-LWRs considered by SERI, the GT-MHR, is a four-module, 2400-MW(t), nominal 1140-MW(e) unit assumed to operate at an annual capacity factor of 88 percent for a net electric output of 1003 MW(e). Therefore, the maximum number of GT-MHR units that could be sited at the Grand Gulf ESP site and remain near or below the 3000 MW(e) total net electric output PPE for the site is three (i.e., 3 x 1003).

The second other-than-LWR considered by SERI, the PBMR, is an eight module, 3200-MW(t), nominal 1320-MW(e) unit assumed to operate at an annual capacity factor of 95 percent for a net electric output of 1253 MW(e). Therefore, the comparable number of PBMR units to remain below the 3000 MW(e) total net electric output PPE for the site is two (i.e., 2 x 1253).

SERI (2005a) compared the impacts in Table S–3 for LWRs with those of the gas-cooled reactor designs. The comparison used an annual fuel loading as a starting point and then proceeded in reverse direction through the fuel cycle (fuel fabrication, enrichment, conversion, milling, mining, radioactive waste). Table 6-3 provides an estimate of the impacts for each phase of the uranium fuel cycle assuming that the Grand Gulf ESP site would host three modular GT-MHR units or two modular PBMR units.

6.1.2.1 Fuel Fabrication

The quantity of UO_2 required for reactor fuel is a key parameter. The more UO_2 required, the greater the environmental impacts (meaning, more energy, greater emissions, and increased water usage). The 1000-MW(e) LWR scaled model described in Section 6.1.1 would require the equivalent of 160 MT of enriched UO_2 annually. This compares to 18 to 19 MT of enriched UO_2 annually for the gas-cooled reactor technologies (see Table 6-3).

GT-MHR fuel consists of microspheres of uranium oxycarbide coated with multiple layers of pryrocarbon and silicon carbide referred to as TRISO coating. Two types of microspheres are used in the GT-MHR fuel, one enriched to 19.8 percent uranium-235 and one with natural uranium. The microspheres and graphite shims are bound together into a rod-shaped compact, which is stacked into graphite blocks referred to as fuel elements. A reactor core consists of 1020 fuel elements.

Table 6-3. Fuel Cycle Environmental Impacts from Gas-Cooled Reactor Designs for the Grand Gulf Early Site Permit Site

Reactor Technology Facility/Activity	GT-MHR 4 Modules (2400 MW(t) total ≈1140 MW(e) total 88 percent capacity: multiplier=3)	PBMR 8 Modules (3200 MW(t) total ≈1320 MW(e) total 95 percent capacity: multiplier=2)
Mining Operations		
Annual ore supply (million MT)	1.01	1.01
Milling Operations		
Annual yellowcake (MT)	909	606
UF_6 Production		
Annual UF_6 (MT)	1137	758
Enrichment Operations		
Enriched UF_6 (MT)	24	25
Annual separative work units (MT)	612	388
Fuel Fabrication Plant Operations		
Enriched UO_2 (MT)	18	19
Annual fuel loading (MT Uranium)	16	17
Solid Radioactive Waste		
Annual low-level waste from reactor operations	3300 Ci[(a)]; 400 m^3	131 Ci; 2400 drums[(a)]
Low-level waste from reactor decontamination and decommissioning (Ci per reference reactor year)	Data not available	Data not available

(a) Multiply curies (Ci) by 3.7 x 10^{10} to obtain becquerel (Bq).

Notes:

- The enrichment separative work units (SWU) calculation was performed using the United States Enrichment Corporation, Inc. (USEC) SWU calculator and assumes a 0.30 percent tails assay.
- The information on the reference reactor (mining, milling, UF_6, enrichment, fuel fabrication values) was taken from NUREG-0116, Table 3.2, (NRC 1976) no recycling.
- The information on the reference reactor (solid radioactive waste) was taken from 10 CFR 51.51, Table S-3.
- The calculated information on the reference reactor uses the same methodology as for the reactor technologies.
- The normalized information is based on 1000 MW(e) and the reactor vendor-supplied unit capacity factor.
- For the new reactor technologies, the annual fuel loading was provided by the reactor vendor.
- The USEC SWU calculator also calculated the kgs of U feed. This number was multiplied by 1.48 to get the necessary amount of UF_6.
- The annual yellowcake number was generated using the relationship 2.61285 lb. of U_3O_8 to 1 kg U of UF_6; 1.185 kgs of U_3O_8 to 1.48 kg.
- The annual ore supply was generated assuming an 0.1 percent ore body and a 90 percent recovery efficiency.
- Co-60 with a 5.26 year half-life and Fe-55 with a 2.73 year half-life are the main nuclides listed for the PBMR decontamination and decommissioning waste.

Source: 10 CFR 51.51(b), Table S-3 Table of Uranium Fuel Cycle Environmental Data

PBMR fuel consists of UO_2 kernels (enriched to 12.9 percent uranium-235) that are TRISO-coated, similar to the GT-MHR fuel. The TRISO-coated particles are imbedded into a graphite matrix to form a fuel sphere that is 60 mm in diameter. Each fuel sphere contains approximately 15,000 TRISO-coated particles. Approximately 260,000 fuel spheres make up a core of a single reactor module.

The fuel described above for gas-cooled reactors is fabricated differently than fuel for LWRs. There are no currently operating large-scale fuel fabrication facilities producing gas-cooled reactor fuels in the United States; thus, a direct comparison of environmental impacts is not possible. Based on some environmental impacts from a small-scale fuel fabrication facility producing gas-cooled reactor fuel, SERI concluded that the environmental impacts from producing gas-cooled reactor fuel would be “not inconsistent” with those of LWRs (SERI 2005a). By comparison with the fuel fabrication impacts for LWR technologies, the staff concludes that the environmental impacts from producing gas-cooled reactor fuel likely would be small. However, these impacts would need to be assessed at the CP or COL stage, when the staff would consider the environmental data that are available on a large-scale, fuel-fabrication facility for gas-cooled reactors should an applicant select one of these designs.

6.1.2.2 Enrichment

SERI (2005a) identified two quantities of interest for enrichment. These were (1) the amount of energy required to enrich the fuel measured in separative work units (SWUs), and (2) the amount of UF_6 needed. A SWU is a measure of energy required to enrich the fuel. The major environmental impacts for the entire uranium fuel cycle are from the emissions of the fossil fuel plants used to supply energy for the gaseous diffusion plants that enrich the uranium. An enrichment technology developed since the impacts in Table S–3 (see Table 6-1) were developed and evaluated includes the gas centrifuge process that uses 90 percent less energy than the gaseous diffusion process (NRC 1996).

To produce 160 MT of enriched UO_2 for the 1000-MW(e) LWR scaled model, the enrichment plant needs to produce about 208 MT of UF_6, which requires approximately 500 MT of SWUs (SERI 2005a). For gas-cooled reactor technologies, the needed enriched UF_6 ranges from 24 to 25 MT of UF_6. The amount of energy to produce these quantities of enriched UF_6 for the gas-cooled reactor designs range from 388 to 612 MT of SWU. The upper range is up to 20 percent higher than the energy required for the reference LWR. SERI (2005a) concluded that the large reduction in energy associated with using an alternate enrichment technology (for example, centrifuge) and its associated environmental impacts would more than offset the increase in SWUs. The staff concludes that, on balance, the environmental impacts of enriching gas-cooled fuels by comparison with the impacts of enriching LWR fuel would likely be small. However, these impacts would need to be assessed at the CP or COL stage, when the staff would consider impacts from the enrichment technology in use at that time, should an applicant select one of these designs.

6.1.2.3 Uranium Hexafluoride Production – Conversion

There are two uranium conversion processes: a wet and a dry process. In the GEIS (NRC 1996), the NRC stated that environmental releases from the conversion facilities are small compared to the overall fuel cycle impacts and that changing from 100 percent use of one process to 100 percent use of the other would make no significant difference in the overall impacts. Similar conversion technologies would be used today to produce UF_6 as were considered when determining the environmental impacts that were part of Table S–3 of 10 CFR 51.51(a) (see Table 6-1).

The conversion facility would need to produce 1440 MT of UF_6 annually for the reference 1000-MW(e) LWR scaled model compared to 758 to 1137 MT of UF_6 for the gas-cooled reactors based on the SWU calculator (SERI 2005a); see Table 6-3, footnote (a) above. The other-than-LWR values are less than the amount of UF_6 required for the 1000-MW(e) LWR scaled model; therefore, the associated environmental impacts are expected to be less. On this basis, the staff concludes that the environmental impacts from producing UF_6 for gas-cooled reactors would be small.

6.1.2.4 Uranium Milling

Annual yellowcake (U_3O_8) production is the metric of interest for uranium milling. Plants requiring less yellowcake production than the reference plant would require less energy, have fewer emissions, and use less water.

The uranium mill for the 1000-MW(e) LWR scaled model would produce about 1200 MT of yellowcake. The uranium mill for the gas-cooled reactor technologies would need to produce 606 to 909 MT of yellowcake, which is less than the amount of yellowcake needed for the 1000 MW(e) LWR scaled model (SERI 2005a). On this basis, the staff concludes that the environmental impacts from uranium milling for the gas-cooled reactors would be small.

6.1.2.5 Uranium Mining

Annual ore supply is the metric of interest for uranium mining. The less ore mined, the smaller the environmental impacts (i.e., less energy used, fewer emissions, less water usage). For the 1000-MW(e) LWR scaled model, 1.09 million MT of raw ore would be required to produce 1200 MT of yellowcake. For the gas-cooled reactor technologies, the scaled ore requirements range from 0.67 to 1.01 million MT of ore, a range that is comparable to the amount of ore required for the reference 1000-MW(e) LWR scaled model. For this reason, the staff concludes that the environmental impacts from uranium mining for the gas-cooled reactors would be small.

6.1.2.6 Solid Low-Level Radioactive Waste – Operations

Table S–3 (see Table 6-1) of 10 CFR 51.51(a) states that there are 3.4 x 10^{14} Bq (9100 Ci) of low-level waste generated annually from operations of the reference LWR; the 1000-MW(e) LWR scaled model would result in 1.35 x 10^{15} Bq (36,400 Ci) of low-level waste annually. Gas-cooled reactor technologies are projected to generate 4.8 x 10^{12} Bq to 1.2 x 10^{14} Bq (131 to 3300 Ci) of low-level waste scaled annually, far below the amounts generated by the reference LWR (SERI 2005a). For this reason, the staff concludes that the environmental impacts from low-level radioactive waste operations for gas-cooled reactors would be small.

6.1.2.7 Solid Low-Level Radioactive Waste – Decontamination and Decommissioning

In Table S–3 (see Table 6-1), the Commission states that 5.6 x 10^{13} Bq (1500 Ci) per reference reactor year "...comes from reactor decontamination and decommissioning - buried at land burial facilities." SERI (2005a) noted that gas-cooled reactor technologies would (1) generate less waste than the reference 1000-MW(e) LWR, and (2) produce less heavy metal radioactive waste because of the higher thermal efficiency and higher fuel burnup. The gas-cooled reactor designs are also more compact than the reference LWR design, which would be expected to result in less decontamination and decommissioning waste (SERI 2005a). SERI concluded that low-level waste impact from decontamination and decommissioning will be comparable to or less than that of the reference LWR (SERI 2005a). On this basis, the staff concludes that the environmental impacts from solid low-level radioactive waste generated during decontamination and decommissioning for gas-cooled reactors would likely be small, but these impacts would need to be assessed again at the CP or COL stage if an applicant selects a gas-cooled design.

6.1.2.8 Conclusions

The staff expects that the environmental impacts from the uranium fuel cycle activities and solid waste management activities for the proposed gas-cooled reactors likely would be small. However, because of the uncertainty in the final design of the gas-cooled reactors and the change in technology that could be applied to uranium fuel cycle activities, this issue is not resolved. Should an applicant reference one of these designs, additional reviews would be needed at the CP or COL stage in the following areas: fuel fabrication, enrichment, and solid low-level waste operation during decontamination and decommissioning.

6.2 Transportation of Radioactive Materials

This section addresses both the radiological and nonradiological environmental impacts from normal operating and accident conditions resulting from (1) shipment of unirradiated fuel to new nuclear units at the Grand Gulf ESP site, (2) shipment of spent fuel to a monitored retrievable storage facility or a permanent repository, and (3) shipment of low-level radioactive waste and mixed waste to offsite disposal facilities. Distinctions between transportation impacts of advanced LWR designs and gas-cooled reactor designs are discussed.

The NRC evaluated the environmental effects of transportation of fuel and waste for light water nuclear power reactors in WASH-1238 (AEC 1972) and NUREG-75/038 (NRC 1975) and found the impact to be SMALL. These documents provided the basis for Table S–4 in 10 CFR 51.52, which summarizes the environmental impacts of transportation of fuel and waste to and from one LWR of 3000 to 5000 megawatts thermal (MW(t)) (1000 to 1500 MW(e)). Impacts are provided for normal conditions of transport and accidents in transport for a reference 1100-MW(e) LWR.

Dose to transportation workers during normal transportation operations was estimated to result in a collective dose of 0.04 person-Sv (4 person-rem) per reference reactor year. The combined dose to the public along the route and dose to onlookers were estimated to result in a collective dose of 0.03 person-Sv (3 person-rem) per reference reactor year. Environmental risks (radiological) during accident conditions were determined to be small. Nonradiological impacts during accident conditions were estimated as one fatal injury in 100 reference reactor years and one nonfatal injury in 10 reference reactor years. Subsequent reviews of transportation impacts in NUREG-0170 (NRC 1977a) and Sprung et al. (2000) concluded that impacts were bounded by Table S–4 of 10 CFR 51.52.

In accordance with 10 CFR 51.52(a), a full description and detailed analysis of transportation impacts is not required when licensing an LWR (i.e., impacts are assumed bounded by Table S–4) if an LWR meets the following criteria:

- The reactor has a core thermal power level not exceeding 3800 MW(t).

- Fuel is in the form of sintered UO_2 pellets having a uranium-235 enrichment not exceeding 4 percent by weight, and pellets are encapsulated in zirconium-clad fuel rods.

- Average level of irradiation of the fuel from the reactor does not exceed 33,000 MWd/MT, and no irradiated fuel assembly is shipped until at least 90 days after it is discharged from the reactor.

- With the exception of irradiated fuel, all radioactive waste shipped from the reactor is packaged and in solid form.

- Unirradiated fuel is shipped to the reactor by truck; irradiated fuel is shipped from the reactor by truck, rail, or barge; and radioactive waste other than irradiated fuel is shipped from the reactor by truck or rail.

The environmental impacts of the transportation of fuel and radioactive wastes to and from nuclear power facilities were resolved generically in 10 CFR 51.52, provided that the specific conditions in the rule (see above) are met; if not, then a full description and detailed analysis is required for initial licensing. Once licensed, the NRC may consider requests to operate at conditions above those in the facility's licensing basis, for example, higher burnups, enrichments, or thermal power levels above 33,000 MWd/MTU, 4 percent, and 3800 MW(t), respectively. Departures from the conditions itemized in 10 CFR 51.52(a) must be supported by a full description and detailed analysis of the environmental effects, as specified by 10 CFR 51.52(b).

SERI has not identified a specific reactor design for the Grand Gulf ESP site but used bounding parameters from seven reactor designs. Five of the designs are LWRs and include the ACR-700 (3964 MW(t)/unit); the ABWR (4300 MW(t)/unit); the surrogate AP1000 (3400 MW(t)/unit); the ESBWR (4000 MW(t)/unit), and the IRIS (3000 MW(t)/unit). For the ACR-700 reactor design, two reactors make up a unit. For the IRIS design, three reactors (modules) make up a unit. For the remaining LWR designs, one reactor makes up a unit.

None of the proposed LWR designs meet all the conditions in 10 CFR 51.52(a); therefore, a full description and detailed analysis are required for each LWR design. This conclusion is based on the following:

- ACR-700, ABWR, and ESBWR designs exceed the 3800-MW(t) core thermal power-level limit.

- ABWR, surrogate AP1000, ESBWR, and IRIS designs require fuel that exceeds the uranium-235 enrichment of 4 percent.

- ABWR, surrogate AP1000, ESBWR, and IRIS designs are expected to exceed the average irradiation level of 33,000 MWd/MTU.

The remaining two designs are gas-cooled reactors: the GT-MHR and the PBMR. Each GT-MHR unit is a four-module, 2400-MW(t), 1140-MW(e) gas-cooled reactor designed to operate at a unit capacity factor of 88 percent. Each PBMR is an eight-module, 3200-MW(t), 1320-MW(e) gas-cooled reactor designed to operate at a unit capacity factor of 96 percent. These compare to the reference reactor in WASH-1238 (AEC 1972), which is a single-unit, 1100-MW(e) LWR with a unit capacity factor of 80 percent. The gas-cooled reactor designs do not meet the conditions in 10 CFR 51.52(a) because these reactors are not LWR designs upon which Table S–4 impacts were based. Therefore, a full description and detailed analysis was required for each gas-cooled reactor design. This was provided by SERI in its response to a request for additional information on September 30, 2004 (SERI 2004f).

SERI used a sensitivity analysis to show that transportation impacts from advanced LWR designs would be bounded by the criteria identified in Table S–4 (SERI 2005a). The GEIS, Addendum 1 (NRC 1999) was referenced as the basis for exceeding 4 percent uranium-235 enrichment and 33,000 MWd/MTU. However, the GEIS, Addendum 1 applies to reactors that are listed in the GEIS, Appendix A, which does not address advanced reactors.

SERI also used a sensitivity analysis to show that transportation impacts from the advanced gas-cooled reactor designs would be bounded by the criteria identified in Table S–4 (SERI 2005a); however, as discussed previously, this type of analysis does not adequately meet the requirements of 10 CFR 51.52. SERI (2005a) identified the major contributors to transportation risk to be the number and type of shipment (shipment risk) and the kind of material being shipped (material risk). Its evaluation of shipment risk showed fewer shipments of unirradiated fuel, spent fuel, and low-level waste would be required for the advanced gas-cooled reactors compared to the reference LWR when averaged over 40 years of operation. Regarding material risk, SERI (2004f) concluded the following:

- The estimated total spent fuel radioactive inventory and fission product inventory was less for the gas-cooled reactors when compared to the reference LWR.

- Actinide inventories would be greater for the gas-cooled reactors (59 to 64 percent greater) because of the increased burnup for these types of reactors; however, because the GT-MHR was assumed to ship about one-third less spent fuel on a MTU basis, SERI (2005a) determined the actinide inventory per shipment would be about one-half of that in the reference LWR shipment. The PBMR is assumed to ship the same amount of spent fuel in a spent fuel shipping cask as the reference LWR so there is about a 60 percent increase in per-shipment actinide inventories from PBMR spent fuel shipments relative to the reference LWR.

- Gas-cooled reactors would generate fewer kilowatts of decay heat per MTU and fewer kilowatts of decay heat per truck cask at the time of shipment.

6.2.1 Transportation of Unirradiated Fuel

The staff performed an independent review of the environmental impacts of transporting unirradiated (fresh) fuel to the Grand Gulf ESP site. Environmental impacts of normal operating conditions and transportation accidents are discussed in this section. Appendix H provides the details of the analysis.

6.2.1.1 Normal Conditions

Normal conditions, sometimes referred to as "incident-free" transportation, are transportation activities in which shipments reach their destination without releasing any radioactive cargo to the environment. Impacts from these shipments would be from the low levels of radiation that penetrate the unirradiated fuel shipping casks.

Truck Shipments

Table 6-4 provides an estimate of the number of truck shipments of unirradiated fuel for each advanced reactor design compared to those of the reference 1100-MW(e) reactor specified in WASH-1238 (AEC 1972). Estimates are normalized for an equivalent 1100-MW(e) electric generating capacity. The basis for the shipment estimates can be found in Appendix H of this EIS. Only the ACR-700, PBMR, and GT-MHR reactor designs would exceed the number of truck shipments of unirradiated fuel estimated for the reference LWR in WASH-1238 (AEC 1972). The largest number of shipments, in excess of 700 shipments over 40 years, is for the GT-MHR. However, the combined annual shipments of unirradiated fuel, spent fuel, and radioactive waste equates to far less than the one truck shipment per day specified in Table S–4 of 10 CFR 51.52 for all reactor types.

Shipping Mode and Weight Limits

In 10 CFR 51.52(a), a condition is identified that states all unirradiated fuel be shipped to the reactor by truck. In information provided by SERI, SERI specifies that unirradiated fuel will be shipped to the reactor site by truck for all reactor designs that it references (INEEL 2003). In addition, 10 CFR 51.52(c) includes a condition that the truck shipments not exceed 33,100 kg (73,000 lbs), as governed by Federal or State gross vehicle weight restrictions. All the advanced reactor designs would meet this weight restriction for unirradiated fuel (INEEL 2003).

Table 6-4. Numbers of Truck Shipments of Unirradiated Fuel for Each Advanced Reactor Type

Reactor Type	Number of Shipments per Reactor Unit: Initial Core[(a)]	Annual Reload	Total[(b)]	Unit Electric Generation, MW(e)[(c)]	Capacity Factor[(c)]	Normalized, Shipments per 1100 MW(e)[(d,e)]
Reference LWR (WASH-1238)	18	6	252	1100	0.8	252
ABWR/ESBWR	30	6.1	267	1500	0.95	165
Surrogate AP1000	14	3.8	161	1150	0.95	130
ACR-700	30	15.4	628	1462[(f)]	0.9	420
IRIS	34	4.3	201	1005[(g)]	0.96	184
GT-MHR	51	20	831	1140[(h)]	0.88	729
PBMR	44	20	824	1320[(i)]	0.95	579

(a) Shipments of the initial core have been rounded up to the next highest whole number.
(b) Total shipments unirradiated fuel over a 40-year plant lifetime (i.e., initial core load plus 39 years of average annual reload quantities).
(c) Unit capacities and capacity factors were taken from INEEL (2003).
(d) Normalized to net electric output for WASH-1238 reference LWR; i.e., 1100-MW(e) plant at 80 percent or net electrical output of 880 MW(e).
(e) Ranges of capacities are given in INEEL (2003) for these unirradiated fuel shipments. The unirradiated fuel shipment data for these reactors were derived using the upper limit of the ranges.
(f) The ACR-700 unit includes two reactors at 731 MW(e) per reactor.
(g) The IRIS unit includes three reactors at 335 MW(e) per reactor.
(h) The GT-MHR unit includes four reactors at 285 MW(e) per reactor.
(i) The PBMR unit includes eight reactors at 165 MW(e) per reactor.
WASH-1238 = ACE 1972
Note: The reference LWR shipment values have all been normalized to 880-MW(e) net electrical generation.

Radiological Doses to Transport Workers and the Public

10 CFR 51.52, Table S–4, includes conditions related to radiological dose to transport workers and members of the public along transport routes. These doses are a function of many variables, including the radiation dose rate emitted from the unirradiated fuel shipments, the number of exposed individuals and their locations relative to the shipment, the time in transit (including travel and stop times), and the number of shipments to which the individuals are exposed. For this EIS, the radiological dose impacts of the transportation of unirradiated fuel were calculated for the worker and the public using the RADTRAN 5 computer code (Neuhauser et al. 2003). Details of the calculations are found in Appendix H.

Table 6-5 presents the radiological impacts to workers, public onlookers (persons at stops and sharing the road), and members of the public along the route (i.e., residents within 800 m (0.5 mi) of the highway) for the advanced reactor designs. The cumulative annual dose estimates in Table 6-5 were normalized to 1100 MW(e). The NRC staff performed an independent review and determined that all dose estimates are bounded by the Table S–4 conditions of 0.04 person-Sv/yr (4 person-rem/yr) to transportation workers, 0.03 person-Sv/yr (3 person-rem/yr) to onlookers, and 0.03 person-Sv/yr (3 person-rem/yr) to members of the public along the route.

Table 6-5. Radiological Impacts of Transporting Unirradiated Fuel to Advanced Reactor Sites

Plant Type	Normalized Average Annual Shipments	Cumulative Annual Dose, person-Sv/yr per 1100 MW(e)[(a)]		
		Workers	Public - Onlookers	Public - Along Route
Reference LWR (WASH-1238)	6.3	1.1×10^{-4}	4.2×10^{-4}	1.0×10^{-5}
ABWR/ESBWR	4.1	7.1×10^{-5}	2.7×10^{-4}	6.6×10^{-6}
Surrogate AP1000	3.3	5.6×10^{-5}	2.2×10^{-4}	5.2×10^{-6}
ACR-700	10.5	1.8×10^{-4}	7.0×10^{-4}	1.7×10^{-5}
IRIS	4.6	7.9×10^{-5}	3.1×10^{-4}	7.4×10^{-6}
GT-MHR	18.2	3.1×10^{-4}	1.2×10^{-3}	2.9×10^{-5}
PBMR	14.5	2.5×10^{-4}	9.6×10^{-4}	2.3×10^{-5}
10 CFR 51.52, Table S–4 Condition	<1 per day	4.0×10^{-2}	3.0×10^{-2}	3.0×10^{-2}

(a) Multiply person-sievert (Sv)/yr by 100 to obtain doses in person-rem/yr.
WASH-1238 = AEC 1972

Although radiation may cause cancers at high doses and high dose rates, currently there are no data that unequivocally establish the occurrence of cancer following exposure to low doses, below about 100 mSv (10,000 mrem). However, radiation protection experts conservatively assume that any amount of radiation may pose some risk of causing cancer or a severe hereditary effect and that the risk is higher for higher radiation exposures. Therefore, a linear, no-threshold dose response model is used to describe the relationship between radiation dose and detriments such as cancer induction. A recent report (National Research Council 2006), the BEIR VII report, supports the linear, no-threshold dose response theory. Simply put, this theory states that any increase in dose, no matter how small, results in an incremental increase in health risk. This theory is accepted by the NRC as a conservative model for estimating health risks from radiation exposure, recognizing that the model probably overestimates those risks.

Based on this model, the staff estimates the risk to the public from radiation exposure using the nominal probability coefficient for total detriment (730 fatal cancers, nonfatal cancers, and severe hereditary effects per 10,000 person-Sv (1,000,000 person-rem)) from International Commission on Radiological Protection Publication 60 (ICRP 1991). All the public doses presented in Table 6-5 are less than or equal to 0.0012 person-Sv/yr (0.12 person-rem/yr); therefore, the total detriment estimates associated with these doses would all be less than 1×10^{-4} fatal cancers, nonfatal cancers, and severe hereditary effects per year. These risks are very small compared to the fatal cancers, nonfatal cancers, and severe hereditary effects that would be expected to occur annually in the same population from exposure to natural sources of radiation.

Maximally Exposed Individuals Under Normal Transport Conditions

A scenario-based analysis was conducted to develop estimates of incident-free radiation doses to maximally exposed individuals (MEI). The analysis is based on information in DOE (2002) and incorporates information about exposure times, dose rates, and the number of times an individual may be exposed to an offsite shipment. Adjustments were made where necessary to reflect the fuel and waste shipments addressed in this EIS. In all cases, it was assumed that the dose rate emitted from the shipping containers is 0.1 mSv/hr (10 mrem/hr) at 2 m (6.6 ft) from the side of the transport vehicle, the maximum dose rate allowed by U.S. Department of Transportation regulations, even though unirradiated fuel and radioactive waste will have much lower dose rates than the regulations allow. An MEI is a person who may receive the highest radiation dose from a shipment to and/or from the advanced reactor site. The analysis is described below.

<u>Truck crew member</u>. Truck crew members would receive the highest radiation doses during incident-free transport because of their proximity to the loaded shipping container for an extended period of time. The analysis assumed that crew member doses are limited to 0.02 Sv (2 rem) per year, which is the DOE administrative control level (DOE 2002). This limit is anticipated to apply to spent nuclear fuel shipments to a disposal facility, as DOE will take title to the spent fuel at the reactor site. Spent nuclear fuel represents the bulk of the fuel and waste shipments to/from advanced reactor sites, and those with the highest radiation dose rates, so crew doses from unirradiated fuel and radioactive waste shipments will be lower than the spent nuclear fuel shipments. The NRC limit for occupational exposures is 0.05 Sv/yr (5 rem/yr).

<u>Inspectors</u>. Radioactive shipments are inspected by Federal or state vehicle inspectors, for example, at state ports of entry. DOE (2002) assumed that inspectors would be exposed for 1 hour at a distance of 1 m (3.3 ft) from the shipping containers. The dose rate at 1 m (3.3 ft) is about 0.14 mSv/hr (14 mrem/hr), so the dose per shipment is about 0.14 Sv (14 mrem). This is independent of the location of the advanced reactor site. Based on this conservative value, the annual doses to vehicle inspectors were calculated to be in the range of 9 to 18 mSv/yr (900 to 1800 mrem/yr), assuming the same person inspects all shipments of fuel and waste to and from

the advanced reactor sites. The high end of the range is the ACR-700 and the low end is the surrogate AP1000. All of the values are less than the 20 mSv/yr (2000 mrem/yr) administrative control level on individual doses.

Resident. The analysis assumed that a resident lives 30 m (100 ft) from the point where a shipment would pass and would be exposed to all shipments along a particular route. Exposures to residents on a per-shipment basis were extracted from RADTRAN 5 output files. These dose estimates are based on an individual located 30 m (100 ft) from the shipments that are traveling 24 km/hr (15 mph). The potential radiation doses to maximally-exposed residents, which are independent of the location of the advanced reactor site, ranged from about 0.00027 mSv/yr (0.027 mrem/yr) for the surrogate AP1000 to 0.00055 mSv/yr (0.055 mrem/yr) for the ACR-700.

Individual stuck in traffic. This scenario addresses potential traffic interruptions that could lead to a person being exposed to a loaded shipment for one hour at a distance of 1.2 m (4 ft). The analysis assumed this exposure scenario would occur only one time to any individual. The dose to the MEI was calculated in DOE (2002) to be 0.016 mSv (1.6 mrem).

Person at a truck service station. This scenario estimates doses to an employee at a service station where all truck shipments to/from the advanced reactors would stop. DOE (2002) assumed this person is exposed for 49 minutes at a distance of 16 m (52 ft) from the loaded shipping container. This results in a dose of about 0.0007 mSv/shipment (0.07 mrem/shipment) and an annual dose in the range from 0.044 mSv (4.4 mrem) for the surrogate AP1000 to 0.09 mSv/yr (9 mrem/yr) for the ACR-700.

6.2.1.2 Accidents

Accident risks are a combination of accident frequency and consequence. Accident frequencies for transportation of fuel to and from future reactors are expected to be lower than those used in the analysis in WASH-1238 (AEC 1972), which forms the basis for Table S–4 of 10 CFR 51.52, because of improvements in highway safety and security and an expected decrease in traffic accident, injury, and fatality rates. There is no significant difference in consequences of accidents severe enough to result in a release of unirradiated fuel particles to the environment between advanced LWRs and current-generation LWRs because the fuel form, cladding, and packaging are similar to those analyzed in WASH-1238. Consequently, the impacts of accidents during transport of unirradiated fuel for advanced LWRs to the Grand Gulf ESP site are expected to be smaller than the impacts listed in Table S–4 for current generation LWRs.

With respect to the advanced gas-cooled reactors, accident rates (accidents per unit distance) and associated accident frequencies (accidents per year) would be expected to follow the same trends as for LWRs (i.e., overall reduction relative to the accident rates used in the WASH-1238 analysis). The consequences of accidents involving gas-cooled reactor unirradiated fuel, however, are more uncertain. The staff assumed that the gas-cooled reactor unirradiated fuel shipments would have the same abilities as LWR unirradiated fuel to maintain functional integrity following a traffic accident. This assumption is considered to be conservative because gas-cooled reactor fuel operates at significantly higher temperatures, and thus maintains integrity under more severe thermal conditions than LWR fuel. Detailed information about the behavior of the gas-cooled reactor fuel under impact conditions was not available. However, packaging systems for unirradiated gas-cooled reactor fuel will need to meet the same performance requirements as unirradiated LWR fuel packages, including fissile material controls to prevent criticality during normal and accident conditions. Consequently, it is expected that packaging systems for unirradiated gas-cooled reactor fuels would provide equivalent protection to those designed for unirradiated LWR fuels. In addition, the fuel forms for the gas-cooled reactors are similar to those for LWRs (i.e., UO_2 for the PBMR and uranium oxycarbide for the GT-MHR versus UO_2 for LWRs) thus, the inherent failure resistance provided by unirradiated gas-cooled reactor fuels should be similar to that provided by LWRs. Based on the assumption that unirradiated gas-cooled and LWR fuels and associated packaging systems would provide similar resistance to various environmental conditions, the staff estimates that the impacts of accidents involving unirradiated gas-cooled reactor fuel likely would not be significantly different from impacts involving unirradiated LWR fuel and would be within the impacts listed in Table S–4 for current-generation LWRs. However, these impacts are not considered to be resolved, and would need to be assessed at the CP or COL stage when specific information is available regarding other-than-LWR fuel performance, if the applicant references such designs.

6.2.2 Transportation of Spent Fuel

The staff performed an independent review of the environmental impacts of transporting spent fuel from the proposed new nuclear unit or units at the Grand Gulf ESP site to a spent fuel disposal repository. The Yucca Mountain, Nevada, site is a possible location for a geologic repository. The staff considers that an estimate of the impacts of the transportation of spent fuel to a possible repository at Yucca Mountain, Nevada to be a reasonable bounding estimate of the transportation impacts to a storage or disposal facility because of the distances involved and the representativeness of the distribution of members of the public in urban, suburban, and rural areas (i.e., population distributions) along the shipping routes. Environmental impacts of normal operating conditions and transportation accidents are discussed in this section.

This analysis is based on shipment of spent fuel by legal-weight trucks in casks with characteristics similar to casks currently available (i.e., massive, heavily shielded, cylindrical metal pressure vessels). Each shipment is assumed to consist of a single shipping cask loaded on a modified trailer. These assumptions are consistent with assumptions made in the evaluation of the environmental impacts of transportation of spent fuel in Addendum 1 to the GEIS (NRC 1999). These assumptions are conservative because the alternative assumptions involve rail transportation or heavy-haul trucks, which would reduce the overall number of spent fuel shipments (NRC 1999).

Environmental impacts of transportation of spent fuel were calculated using the RADTRAN 5 computer code (Neuhauser et al. 2003). Routing and population data used in the RADTRAN 5 for truck shipments were obtained from the TRAGIS routing code (Johnson and Michelhaugh 2000). The population data in the TRAGIS code are based on the 2000 U.S. census.

The staff's evaluation reviewed the impacts of spent fuel shipments originating from the Grand Gulf ESP site and the alternative sites: James A. FitzPatrick Nuclear Power Plant and Pilgrim Nuclear Station. Another alternative site, River Bend Station, was considered by SERI in its environmental report, but was not evaluated by the staff because the route characteristics of distance and population would not be sufficiently different to produce results different from the Grand Gulf ESP site. Appendix H provides the details of the analysis.

6.2.2.1 Normal Conditions

Normal conditions, sometimes referred to as "incident-free" transportation, are transportation activities in which shipments reach their destination without an accident occurring en route. Impacts from these shipments would be from the low levels of radiation that penetrate the heavily shielded spent fuel shipping cask. Radiation exposure would occur to (1) persons residing along the transportation corridors between the Grand Gulf ESP site and the proposed repository; (2) persons in vehicles traveling on the same route as the spent fuel shipment; (3) persons at vehicle stops for vehicle inspections, refueling, and rest; and (4) transportation crew workers.

Shipping casks have not been designed for the advanced reactor designs. Information in INEEL (2003) indicated that advanced LWR fuel designs would not be significantly different from existing LWR designs; therefore, the characteristics of current shipping cask designs were used for the analysis for advanced LWR designs. No information is available on spent fuel shipping cask designs for the gas-cooled reactors. For purposes of this Chapter 6 analysis, their design was assumed to be the same as those used for the existing LWRs. Spent fuel shipping cask designs for gas-cooled reactors have not been defined and, therefore, impacts are not resolved. Impacts would be evaluated at the CP or COL stage if the applicant references such designs.

Radiation doses are a function of many parameters, including vehicle speed, traffic count, dose rate at 1 m from the vehicle, packaging dimensions, number of persons in the truck crew, stop time, and population density at stops. For a listing of the values for these and other parameters, refer to Appendix H. Table 6-6 presents radiation dose estimates to the transport workers and the public for the primary and alternative ESP sites. Doses are presented on a per-shipment basis. The per-shipment dose estimates are independent of reactor technology because they were calculated based on an assumed external radiation dose rate emitted from the cask, which was fixed at the regulatory maximum limit for the advanced reactor designs (i.e., 0.1 mSv/hr (10 mrem/hr) at 2 m).

Table 6-6. Routine (Incident-Free) Radiation Doses to Transport Workers and the Public from Shipping Spent Fuel from Potential Early Site Permit Sites to a Spent Fuel Disposal Facility

	Population Dose, person-Sv/shipment[a]		
Reactor Site	**Crew**	**Onlookers**	**Along Route**
Grand Gulf[b]	8.7×10^{-4}	2.8×10^{-3}	7.0×10^{-5}
FitzPatrick	9.8×10^{-4}	3.5×10^{-3}	9.5×10^{-5}
Pilgrim	1.1×10^{-3}	3.9×10^{-3}	1.2×10^{-4}

(a) Multiply person-sievert (Sv)/yr by 100 to obtain doses in person-rem/yr.
(b) Doses for the River Bend alternative site can be assumed to be bounded by the values for the proposed Grand Gulf ESP site because differences in route characteristics are minimal.

Population dose estimates per reactor year are presented in Table 6-7 for specific advanced reactor designs. Population doses were calculated by multiplying the number of spent fuel shipments per year for each advanced reactor design times the dose per shipment from Table 6-6. Population doses were normalized to the reference LWR design in WASH-1238 (880 net MW(e)) (AEC 1972). This corresponds to an 1100-MW(e) LWR operating at 80 percent capacity. Appendix H provides the basis upon which the number of spent fuel shipments was derived for each advanced reactor design.

The bounding cumulative doses to the exposed population given in Table S–4 (10 CFR 51.51(c)) are

- 0.04 person-Sv (4 person-rem) per reactor-year to transport workers
- 0.03 person-Sv (3 person-rem) per reactor-year to general public (onlookers) and members of the public along the route.

Population doses to the crew and the onlookers for all the reactor types, including the reference reactor designated in Table 6-7, exceed Table S–4 values. Two key reasons for the higher population doses relative to Table S–4 are the higher number of spent fuel shipments estimated for some of the reactor technologies and the longer shipping distances assumed for the analyses (i.e., to a possible repository in Nevada) than were used in WASH-1238. WASH-1238 used a "typical" distance for a spent fuel shipment of 1600 km (1000 mi), whereas the shipping distances used in this assessment ranged from about 3000 km (1800 mi) to 4700 km (2900 mi). The higher numbers of shipments are based on spent fuel shipping casks designed to transport shorter-cooled fuel (150 days out of the reactor). It was assumed in this Chapter 6 analysis that the shipping cask capacities are 0.5 MTU/shipment, roughly equivalent to one PWR or two BWR spent fuel assemblies per shipment.

Newer shipping cask designs are based on longer-cooled spent fuel (5 years out of reactor) and have larger capacities than those used in this assessment. DOE (2002) spent fuel shipping cask capacities were approximately 1.8 MTU/shipment, or up to four PWR or nine BWR fuel assemblies per shipment. Use of the newer shipping cask designs would reduce the number of spent fuel shipments and the associated environmental impacts. On balance, if the population doses are adjusted for the shipping distance and shipping cask capacity, the routine population doses from spent fuel shipments from all reactor types and all sites fall within Table S–4 requirements.

Other conservative assumptions in the staff's calculation include:

- <u>Use of the regulatory maximum dose rate (0.1 mSv/hr or 10 mrem/hr at 2 m) in the RADTRAN 5 calculations</u>. The shipping casks assumed in the EIS prepared in support of the application for a geologic repository at the proposed Yucca Mountain site (DOE 2002) were designed to transport spent fuel that has cooled for five years. In reality, most spent fuel will have cooled for much longer than five years before it is shipped to a possible geologic repository. Sprung et al. (2000) developed a probabilistic distribution of dose rates based on fuel cooling times that indicates that approximately three-fourths of the spent fuel to be transported to a possible geologic repository will have dose rates less than half of the regulatory limit. Consequently, the estimated population doses in Table 6-7 could be divided in half if more realistic dose rate projections are used.

- <u>Use of 30 minutes as the average time at a truck stop in the calculations</u>. Many stops made for actual spent fuel shipments are short-duration stops (e.g., 10 minutes) for brief visual inspections of the cargo (e.g., checking the cask tie-downs). These stops typically occur in minimally populated areas, such as an overpass or freeway ramp in an unpopulated area. Furthermore, empirical data provided in Griego et al. (1996) indicate that a 30-minute stop is toward the high end of the stop-time distribution. Average stop times observed by Griego et al. (1996) are on the order of 18 minutes. Based on these observations, it was concluded that the stop model assumptions used in this study overestimate public doses at stops by at least a factor of two. Consequently, the doses to onlookers given in Table 6-7 could be reduced by a factor of two to reflect more realistic truck shipping conditions.

Table 6-7. Routine (Incident-Free) Population Doses from Spent Fuel Transportation, Normalized to Reference Light Water Reactor

Reactor Type	Reference LWR (WASH-1238)			ABWR/ESBWR			Surrogate AP1000			ACR-700		
No Shipments per Year	60			41			40			90		
	Environmental Effects, person-Sv[(a)] per reference reactor year											
Reactor Site	Crew	Onlookers	Along Route	Crew	Onlookers	Along Route	Crew	Onlookers	Along Route	Crew	Onlookers	Along Route
Grand Gulf	5.2×10^{-2}	1.7×10^{-1}	4.2×10^{-3}	3.5×10^{-2}	1.2×10^{-1}	2.8×10^{-3}	3.4×10^{-2}	1.1×10^{-1}	2.7×10^{-3}	7.8×10^{-2}	2.5×10^{-1}	6.2×10^{-3}
FitzPatrick	5.9×10^{-2}	2.1×10^{-1}	5.7×10^{-3}	4.0×10^{-2}	1.4×10^{-1}	3.9×10^{-3}	3.9×10^{-2}	1.4×10^{-1}	3.8×10^{-3}	8.8×10^{-2}	3.1×10^{-1}	8.5×10^{-3}
Pilgrim	6.5×10^{-2}	2.3×10^{-1}	7.0×10^{-3}	4.4×10^{-2}	1.6×10^{-1}	4.8×10^{-3}	4.3×10^{-2}	1.5×10^{-1}	4.6×10^{-3}	9.8×10^{-2}	3.5×10^{-1}	1.0×10^{-2}

Reactor Type	IRIS			GT-MHR			PBMR		
No Shipments per Year	35			34			12		
	Environmental Effects, person-Sv[(a)] per reference reactor year								
Reactor Site	Crew	Onlookers	Along Route	Crew	Onlookers	Along Route	Crew	Onlookers	Along Route
Grand Gulf[(b)]	3.0×10^{-2}	9.8×10^{-2}	2.4×10^{-3}	2.9×10^{-2}	9.4×10^{-2}	2.3×10^{-3}	9.7×10^{-3}	3.2×10^{-2}	7.8×10^{-4}
FitzPatrick	3.4×10^{-2}	1.2×10^{-1}	3.3×10^{-3}	3.3×10^{-2}	1.2×10^{-1}	3.2×10^{-3}	1.1×10^{-2}	3.9×10^{-2}	1.1×10^{-3}
Pilgrim	3.8×10^{-2}	1.3×10^{-1}	4.0×10^{-3}	3.6×10^{-2}	1.3×10^{-1}	3.9×10^{-3}	1.2×10^{-2}	4.3×10^{-2}	1.3×10^{-3}

(a) Multiply person-sievert (Sv)/yr by 100 to obtain dose in mrem/yr.
(b) Doses for the River Bend alternative site can be assumed to be bounded by the values for the proposed Grand Gulf ESP site because differences in route characteristics are minimal.
WASH-1238 = AEC 1972

SERI performed its own RADTRAN 5 calculations looking at the impact of "incident-free" transport of spent fuel to a spent fuel disposal facility. The assumed transport of spent fuel originated from the Maine Yankee Nuclear Plant (a distance further than the Grand Gulf ESP site) and terminated at a disposal facility assumed to be at Yucca Mountain, Nevada. Dose estimates per shipment were similar to those calculated by the staff.

Although radiation may cause cancers at high doses and high dose rates, currently there are no data that unequivocally establish the occurrence of cancer following exposure to low doses, below about 100 mSv (10,000 mrem), and at low dose rates. However, radiation protection experts conservatively assume that any amount of radiation may pose some risk of causing cancer or a severe hereditary effect and that the risk is higher for higher radiation exposures. Therefore, a linear, no-threshold dose response model is used to describe the relationship between radiation dose and detriments such as cancer induction. A recent report (National Research Council 2006), the BEIR VII Report, supports the linear, no threshold dose response theory. Simply put, this theory states that any increase in dose, no matter how small, results in an incremental increase in health risk. This theory is accepted by the NRC as a conservative model for estimating health risks from radiation exposure, recognizing that the model probably overestimates those risks.

Based on this model, the staff estimated the risk to the public from radiation exposure using the nominal probability coefficient for total detriment (730 fatal cancers, nonfatal cancers, and severe hereditary effects per 10,000 person-Sv (1,000,000 person-rem)) from International Commission on Radiological Protection Publication 60 (ICRP 1991). All the population doses presented in Table 6-7 are less than one person-Sv/yr (100 person-rem/yr); therefore, the total detriment estimates associated with these population doses would all be less than 0.1 fatal cancers, nonfatal cancers, and severe hereditary effects per year. These risks are very small compared to the fatal cancers, nonfatal cancers, and severe hereditary effects that would be expected to occur annually in the same population from exposure to natural sources of radiation.

Dose estimates to the MEI from transport of unirradiated fuel, spent fuel, and wastes under normal conditions are presented in Section 6.2.1.1.

6.2.2.2 Accidents

As discussed previously, the staff used the RADTRAN 5 computer code to estimate impacts of transportation accidents involving spent fuel shipments. RADTRAN 5 considers a spectrum of potential transportation accidents, ranging from those with high frequencies and low consequences (e.g., "fender benders") to those with low frequencies and high consequences (i.e., accidents in which the shipping container is exposed to severe mechanical and thermal conditions). Details of the analysis are discussed in Appendix H.

Radionuclide inventories are important parameters in the calculation of accident risks. The radionuclide inventories used in this Chapter 6 analysis are from *Early Site Permit Environmental Report Sections and Supporting Documentation* (INEEL 2003). This report included hundreds of radionuclides for each advanced reactor type. A screening analysis was conducted to select the dominant contributors to accident risks to simplify the RADTRAN 5 calculations. The screening identified the radionuclides that would contribute more than 99.999 percent of the dose from inhalation of radionuclides released following a transportation accident. The dominant radionuclides are similar regardless of the fuel type (i.e., advanced LWR fuel or gas-cooled reactor fuel). Spent fuel inventories used in the staff analysis are presented in Table 6-8. Note that the list of radionuclides provided in the table includes all of the radionuclides that were included in the analysis conducted by Sprung et al. (2000), which validates the screening process used in this EIS. Also note that the INEEL (2003) analysis relied upon by SERI in its application did not provide radionuclide source terms for radioactive material deposited on the external surfaces of LWR spent fuel rods (commonly called “crud”). In addition, data on activation products was provided for only the advanced BWR. The advanced BWR spent fuel transportation risks were calculated assuming the entire cobalt-60 inventory is in the form of crud. This is very conservative as the source term because it is about two orders of magnitude greater than that given in Sprung et al. (2000). Because crud is deposited from corrosion products generated elsewhere in the reactor cooling system and the complete reactor design and operating parameters are uncertain, the quantities and characteristics of crud deposited on advanced reactor spent fuel are unknown at this time. Consequently, the impacts of crud and activation products on spent fuel transportation accident risks are not resolved and would need to be examined at the CP or COL stage. No radionuclide inventory data were presented in INEEL (2003) for the ACR-700 and IRIS advanced reactors. Because transportation accident risks were not quantified for these reactor types, these accident risks are not resolved and would need to be assessed at the CP or COL stage if the applicant references either of these designs.

Robust shipping casks are used to transport spent fuel because of the radiation shielding and accident resistance required by 10 CFR Part 71. Spent fuel shipping casks must be certified Type B packaging systems, which means the casks must withstand a series of severe postulated accident conditions with essentially no loss of containment or shielding capability. These casks are also designed with fissile material controls to ensure the spent fuel remains subcritical under normal and accident conditions. According to Sprung et al. (2000), the likelihood of encountering accident conditions that would lead to shipping cask failure is less than 0.01 percent (i.e., more than 99.99 percent of all accidents would result in no release of radioactive material from the shipping cask). The staff assumed that shipping casks for advanced reactor spent fuels will provide equivalent mechanical and thermal protection of the spent fuel cargo.

Table 6-8. Radionuclide Inventories Used in Transportation Accident Risk Calculations for Each Advanced Reactor Type, Bq/MTU[(a)]

Radionuclide	ABWR and ESBWR Inventory	Surrogate AP1000 Inventory	GT-MHR Inventory	PBMR Inventory
Am-241	4.96×10^{13}	2.69×10^{13}	8.18×10^{13}	7.55×10^{13}
Am-242m	1.24×10^{12}	4.85×10^{11}	5.03×10^{11}	8.51×10^{11}
Am-243	1.20×10^{12}	1.24×10^{12}	5.14×10^{11}	4.77×10^{12}
Ce-144	4.22×10^{14}	3.28×10^{14}	2.15×10^{15}	1.19×10^{15}
Cm-242	2.04×10^{12}	1.05×10^{12}	1.51×10^{12}	2.78×10^{12}
Cm-243	1.37×10^{12}	1.14×10^{12}	2.02×10^{11}	1.96×10^{12}
Cm-244	1.80×10^{14}	2.87×10^{14}	2.83×10^{13}	5.48×10^{14}
Cm-245	2.43×10^{10}	4.48×10^{10}	1.65×10^{8}	5.29×10^{10}
Co-60	1.01×10^{14}	(b)	(b)	(b)
Cs-134	1.78×10^{15}	1.78×10^{15}	2.21×10^{15}	4.03×10^{15}
Cs-137	4.59×10^{15}	3.44×10^{15}	1.08×10^{16}	1.41×10^{16}
Eu-154	3.81×10^{14}	3.38×10^{14}	3.23×10^{14}	3.74×10^{14}
Eu-155	1.93×10^{14}	1.71×10^{14}	8.77×10^{13}	1.08×10^{14}
Pm-147	1.25×10^{15}	6.51×10^{14}	6.92×10^{15}	5.07×10^{15}
Pu-238	2.27×10^{14}	2.25×10^{14}	1.17×10^{14}	4.55×10^{14}
Pu-239	1.43×10^{13}	9.44×10^{12}	2.25×10^{13}	1.11×10^{13}
Pu-240	2.28×10^{13}	2.01×10^{13}	3.96×10^{13}	3.32×10^{13}
Pu-241	4.51×10^{15}	2.58×10^{15}	8.33×10^{15}	7.18×10^{15}
Pu-242	8.29×10^{10}	6.73×10^{10}	1.56×10^{11}	4.51×10^{11}
Ru-106	6.07×10^{14}	5.74×10^{14}	1.48×10^{15}	1.68×10^{15}
Sb-125	1.99×10^{14}	1.42×10^{14}	2.21×10^{14}	2.51×10^{14}
Sr-90	3.27×10^{15}	2.29×10^{15}	8.95×10^{15}	1.08×10^{16}
Y-90	3.27×10^{15}	2.29×10^{15}	8.95×10^{15}	1.08×10^{16}

(a) Divide bequerel (Bq)/metric tons uranium (Bq/MTU) by 3.7×10^{10} to obtain curies/MTU.
(b) Cobalt-60 is an activation product. Only the ABWR and ESBWR reactor types identified in INEEL (2003) included inventory data for activation products.

The RADTRAN 5 accident risk calculations were performed using unit radionuclide inventories (Bq/MTU) for the spent fuel shipments from the various reactor types. The resulting risk estimates were then multiplied by assumed annual spent fuel shipments (MTU/yr) to derive estimates of the annual accident risks associated with spent fuel shipments from each potential advanced reactor site. As was done for routine exposures, the staff assumed that the numbers of shipments of spent fuel per year are equivalent to the annual discharge quantities.

For this assessment, release fractions for current-generation LWR fuels were used to approximate the impacts from the advanced reactor spent fuel shipments. This assumes that the fuel materials and containment systems (i.e., cladding, fuel coatings) behave similarly to current LWR fuel under applied mechanical and thermal conditions. Because of the lack of experimental data on gas-cooled reactor fuels, it is currently not known if this approach is bounding. However, gas-cooled reactors operate at much higher temperatures than LWRs; therefore, high temperature conditions anticipated in transportation accident fires should have less of an effect on radionuclide releases than they do for LWR fuels. Thus, smaller release fractions are anticipated for advanced gas-cooled reactor fuels than for LWR fuels subjected to thermal transients. However, this issue is not resolved because of the lack of information on these designs.

The NRC staff used RADTRAN 5 to calculate the population dose from the radioactive material released to the environment and assessed for four of five[a] possible exposure pathways. These pathways are:

(1) External dose from exposure to the passing cloud of radioactive material (cloudshine)

(2) External dose from the radionuclides deposited on the ground by the passing plume (groundshine). The staff's analysis included the radiation exposure from this pathway even though the area surrounding a potential accidental release would be evacuated and decontaminated, thus preventing long-term exposures from this pathway.

(3) Internal dose from inhalation of airborne radioactive contaminants (inhalation)

(4) Internal dose from resuspension of radioactive materials that were deposited on the ground (resuspension). The staff's analysis included the radiation exposures from this pathway even though evacuation and decontamination of the area surrounding a potential accidental release would prevent long-term exposures.

Table 6-9 presents the environmental consequences of transportation accidents when shipping spent fuel from the Grand Gulf ESP site and alternative sites to the proposed Yucca Mountain repository. The shipping distances and population distribution information for the routes were the same as those used for the normal "incident-free" conditions (for details, see Appendix H).

(a) Internal dose from ingestion of contaminated food was not considered as the staff assumed evacuation and subsequent interdiction of foodstuffs following a potential transportation accident.

Table 6-9 presents estimates of population dose (person-Sv/ reactor year) for several of the advanced reactor designs. These values are normalized to the WASH-1238 reference reactor (880-MW(e) net electrical generation, 1100-MW(e) reactor operating at 80 percent capacity) (AEC 1972).

Table 6-9. Annual Spent Fuel Transportation Accident Impacts for Advanced Reactors, Normalized to Reference 1000-MW(e) Light Water Reactor Net Electrical Generation

	Advanced Reactor Type			
	ABWR and ESBWR	**Surrogate AP1000**	**GT-MHR**	**PBMR**
MTU/yr	**20.3**	**19.7**	**6.0**	**5.8**
	Population Dose, person-Sv/per reference reactor year[(a)]			
Grand Gulf[(b)]	4.1×10^{-6}	3.7×10^{-7}	1.7×10^{-7}	2.7×10^{-7}
FitzPatrick	3.8×10^{-6}	3.3×10^{-7}	1.5×10^{-7}	2.5×10^{-7}
Pilgrim	8.1×10^{-6}	7.2×10^{-7}	3.3×10^{-7}	5.4×10^{-7}

(a) Multiply person-Sv/yr times 100 to obtain person-rem/yr.
(b) Doses for the River Bend alternative site can be assumed to be bounded by the values for the proposed Grand Gulf ESP site because differences in route characteristics are minimal.

Although radiation may cause cancers at high doses and high dose rates, currently there are no data that unequivocally establish the occurrence of cancer following exposure to low doses below about 100 mSv (10,000 mrem) and at low dose rates. However, radiation protection experts conservatively assume that any amount of radiation may pose some risk of causing cancer or a severe hereditary effect, and that the risk is higher for higher radiation exposures. Therefore, a linear, no-threshold dose response model is used to describe the relationship between radiation dose and detriments such as cancer induction. A recent report (National Research Council 2006), the BEIR VII report, supports the linear, no-threshold dose response theory. Simply put, this theory states that any increase in dose, no matter how small, results in an incremental increase in health risk. This theory is accepted by the NRC as a conservative model for estimating health risks from radiation exposure, recognizing that the model probably overestimates those risks.

Based on this model, the staff estimates the risk to the public from radiation exposure using the nominal probability coefficient for total detriment – 730 fatal cancers, nonfatal cancers, and severe hereditary effects per 10,000 person-Sv (1,000,000 person-rem) – from International Commission on Radiological Protection Publication 60 (ICRP 1991). All the population doses presented in Table 6-9 are less than 1.0×10^{-5} person-Sv (1.0×10^{-3} person-rem) per reference reactor year; therefore, the total detriment estimates associated with these population doses would all be less than 1.0×10^{-6} fatal cancers, nonfatal cancers, and severe hereditary effects per reference reactor year. These risks are quite small compared to the fatal cancers, nonfatal cancers, and severe hereditary effects that would be expected to occur annually in the same population from exposure to natural sources of radiation.

6.2.2.3 Conclusion

The values determined by this analysis represent the contribution of such effects to the environmental costs of licensing the reactor. Because of the conservative approaches and data used to calculate doses, actual environmental effects are not likely to exceed those calculated in the EIS. Thus, the staff concludes that the overall transportation accident risks associated with advanced LWR reactor spent fuel shipments are SMALL and are consistent with the risks associated with transportation of spent fuel from current-generation reactors presented in Table S–4 of 10 CFR 51.52. The fuel performance characteristics, shipping casks, and accident risks for other-than-LWR designs are not resolved and would need to be assessed at the CP or COL stage if the applicant references such designs.

6.2.3 Transportation of Radioactive Waste

This section discusses the environmental effects of transporting waste from advanced reactor sites. The environmental conditions listed in 10 CFR 51.52(a) that apply to shipments of radioactive waste include the following:

- Radioactive waste (except spent fuel) is packaged in solid form.
- Radioactive waste (except spent fuel) is shipped from the reactor by truck or rail.
- The weight limitation is 33,100 kg (73,000 lb) per truck and 90,700 kg (100 tons) per cask per railcar.
- The traffic density limitation is less than one truck shipment per day or three railcars per month.

In INEEL (2003), it is stated that all the radioactive waste will be transported by truck. SERI plans to solidify and package its waste regardless of which advanced reactor technology it chooses. In addition, waste from any of the advanced reactor technologies will be subject to NRC (10 CFR Part 71) and U.S. Department of Transportation (49 CFR Parts 171, 172, 173, and 178) regulations for the shipment of radioactive material. Radioactive waste from any of the advanced reactor technologies are expected to be capable of being shipped in compliance with Federal or State weight restrictions.

Table 6-10 presents estimates of annual waste volumes and annual waste shipment numbers for the advanced reactor types normalized to the reference 1100-MW(e) LWR defined in WASH-1238 (AEC 1972). Annual waste volumes and waste shipments for the advanced reactor technologies were less than the 1100-MW(e) reference reactor that was the basis for Table S–4 for all designs except the PBMR.

Table 6-10. Summary of Radioactive Waste Shipments for Advanced Reactors

Reactor Type	INEEL (2003) Waste Generation Information	Annual Waste Volume, m^3/yr per unit	Electrical Output, MW(e) per unit	Normalized Rate, m^3/1100 MW(e) reactor (880 MW(e) net)[(a)]	Shipments/ 1100 MW(e) (880 MW(e) net) Electrical Output[(b)]
Reference LWR (WASH-1238)	100 m^3/yr per unit	108	1100	108	46
ABWR	100 m^3/yr per unit	100	1500	62	27
ESBWR	100 m^3/yr per unit	100	1500	62	27
Surrogate AP1000	55 m^3/yr per unit	56	1150	45	20
ACR-700	47.5 m^3/yr per unit	95	1462[(c)]	64	28
IRIS	25 m^3/yr per unit	74 (3 units)	1005[(d)]	67	29
GT-MHR	98 m^3/yr (4 unit plant)	98 (4 units)	1140[(e)]	86	37[(g)]
PBMR	100 drums/yr per unit	168 (8 units)	1320[(f)]	118	51[(g)]

(a) Capacity factors used to normalize the waste generation rates to an equivalent electrical generation output are given in Table 6-3 for each reactor type. All are normalized to 880-MW(e) net electrical output (1100-MW(e) plant with an 80-percent capacity factor).
(b) The number of shipments per 1100 MW(e) was calculated assuming the WASH-1238 average waste shipment capacity of 2.34 m^3 per shipment (108 m^3/yr divided by 46 shipments/yr).
(c) The ACR-700 unit includes two reactors at 731 MW(e) per reactor.
(d) The IRIS unit includes three reactors at 335 MW(e) per reactor.
(e) The GT-MHR unit includes four reactors at 285 MW(e) per reactor.
(f) The PBMR unit includes eight reactors at 165 MW(e) per reactor.
(g) SERI states in INEEL (2003) that 90 percent of the waste could be shipped on trucks carrying 28 m^3 (1000 ft^3) of waste and the remaining 10 percent in shipments carrying 5.7 m^3 (200 ft^3) of radioactive waste. This would result in five to six shipments per year after normalization to the reference LWR electrical output.
Conversions: 1 m^3 = 35.31 ft^3, drum volume = 210 liters (0.21 m^3)
WASH-1238 = AEC 1972

As shown in the table, only the PBMR would be expected to generate a larger volume of radioactive waste than the reference LWR in WASH-1238 (AEC 1972). However, the GT-MHR and PBMR information in INEEL (2003) assumed that the applicant would ship wastes using two different packaging systems: one that hauls 28 m^3 per shipment (1000 ft^3 per shipment) and one that hauls 5.7 m^3 per shipment (200 ft^3 per shipment). Under those conditions, the number of shipments of radioactive waste per year, normalized to 1100 MW(e) electric generation capacity, would be about six shipments per year per 1100 MW(e) (880 net MW(e)) for the GT-MHR and seven shipments per year per 1100 MW(e) for the PBMR. These estimates are well below the reference LWR (46 shipments per year per 1100 MW(e)). However, impacts from other than LWR designs are not resolved because of the lack of verifiable information.

The sum of the daily shipments of unirradiated fuel, spent fuel, and radioactive waste is well below the one truck shipment per day condition given in 10 CFR 51.52, Table S–4 for all reactor types. Doubling the shipment estimates to account for empty return shipments of fuel and waste is still well below the one-truck-shipment-per-day condition.

Dose estimates to the maximally-exposed individual from transport of unirradiated fuel, spent fuel, and waste under normal conditions are presented in Section 6.2.1.1.

6.2.4 Conclusions

An analysis was conducted of the impacts under normal operating and accident conditions of transporting unirradiated fuel to advanced reactor sites and spent fuel and wastes from advanced reactor sites to disposal facilities. To make comparisons to Table S–4, the environmental impacts are normalized to a reference reactor year. The reference reactor is an 1100-MW(e) reactor that has an 80-percent capacity factor, for a total electrical output of 880 MW(e) per year. The environmental impacts can be adjusted to calculate impacts per site by multiplying the normalized impacts by the ratio of the total electric output for the advanced reactor sites to the electric output of the reference reactor.

Because of the conservative approaches and data used to calculate doses, actual environmental effects are not likely to exceed those calculated in the EIS. Thus, the staff concludes that the environmental impacts of transportation of fuel and radioactive wastes to and from advanced LWR designs would be SMALL, and would be consistent with the risks associated with transportation of fuel and radioactive wastes from current-generation reactors presented in Table S–4 of 10 CFR 51.52. For gas-cooled designs, the impacts are likely to be small, but this issue is not resolved because of the lack of verifiable information on these designs. At the CP or COL stage, an applicant referencing these designs would need to provide the necessary data and the staff would need to validate the assumptions used in this transportation analysis.

Assumptions that will need validation if a gas-cooled is selected include:

- Verifying that unirradiated and spent fuel from gas-cooled reactors have the same abilities as LWR unirradiated and spent fuel to maintain fuel and cladding integrity following a traffic accident.

- Verifying that shipping cask design assumptions (for example, cask capacities) are equal to or bounded by the assumptions in this analysis.

- Verifying that unirradiated fuel initial core/refueling requirement, spent fuel generation rates, and radioactive waste generation rate assumptions are equal to or bounded by the assumptions in this analysis.

- Verifying that shipping cask capacities and accident source terms, including spent fuel inventories, severity fractions, and release fractions, are equal to or bounded by the assumptions in this analysis.

Should the ACR-700 or IRIS reactors be chosen for the ESP site, a transportation accident analysis will be performed as spent fuel inventories were not available for this analysis.

6.3 Decommissioning Impacts

At the end of the operating life of a power reactor, the NRC regulations require that the facility undergo decommissioning. Decommissioning is the removal of a facility safely from service and the reduction of residual radioactivity to a level that permits termination of the NRC license. The regulations governing decommissioning of power reactors are found in 10 CFR 50.75 and 50.82.

Environmental impacts from the activities associated with the decommissioning of any LWR before or at the end of an initial or renewed license are evaluated in the *Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities*, *Supplement 1, Regarding the Decommissioning of Nuclear Power Reactors*, NUREG-0586, (NRC 2002). If an applicant for a CP or COL referencing the Grand Gulf ESP applies for a license to operate one or more additional units at the Grand Gulf ESP site, there is a requirement to provide a report containing a certification that financial assurance for radiological decommissioning will be provided. At the time an application is submitted, the requirements in 10 CFR 50.33, 50.75, and 52.77 (and any other applicable requirements) would have to be met.

At the ESP stage, applicants are not required to submit information regarding the process of decommissioning, such as the method chosen for decommissioning, the schedule, or any other aspect of planning for decommissioning. SERI did not provide this information in its application.

For the new nuclear unit or units, if LWR designs are chosen or if other-than-LWRs that were considered in NUREG-0586, Supplement 1 are chosen, the impacts from decommissioning are expected to be within the bounds described in NUREG-0586, Supplement 1. In such cases, the staff expects the impact from decommissioning are likely to be small. However, for whatever design that is selected, the impacts from decommissioning are not resolved and would have to be assessed at the CP or COL stage.

6.4 References

10 CFR Part 20. Code of Federal Regulations, Title 10, *Energy*, Part 20, “Standards for Protection Against Radiation.”

10 CFR Part 50. Code of Federal Regulations, Title 10, *Energy*, Part 50, “Domestic Licensing of Production and Utilization Facilities.”

10 CFR Part 51. Code of Federal Regulations, Title 10, *Energy*, Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions.”

10 CFR Part 52. Code of Federal Regulations, Title 10, *Energy*, Part 52, “Early Site Permits, Standard Design Certifications, and Combined Licenses for Nuclear Power Plants.”

10 CFR Part 63. Code of Federal Regulations, Title 10, *Energy*, Part 63, “Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada.”

10 CFR Part 71. Code of Federal Regulations, Title 10, *Energy*, Part 71, “Packaging and Transportation of Radioactive Material.”

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