APPENDIX J
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J.1 Introduction

This appendix examines the potential differences in accident impacts if mixed oxide (MOX) fuel were used to partially fuel domestic, commercial nuclear power plants. This appendix provides an assessment of the human health effects related to postulated reactor accidents involving MOX fuel use in the Browns Ferry and Sequoyah Nuclear Plants (Browns Ferry and Sequoyah). The analyses provide a basic comparison of potential safety impacts posed by reactor operations using a partial MOX fuel core (approximately 40 percent MOX fuel and 60 percent low-enriched uranium [LEU] fuel) versus operations using a full LEU fuel core. As part of the licensing process for certifying the use of MOX fuel in any domestic commercial nuclear power facility, the U.S. Nuclear Regulatory Commission (NRC) would also conduct rigorous, independent analyses of the effects of MOX fuel use on reactor safety.

In support of this *SPD Supplemental EIS*, reactor fuel experts at Oak Ridge National Laboratory (ORNL), with input from Tennessee Valley Authority (TVA) nuclear engineers, ran state-of-the-art computer codes to model reactor fuel cores to develop inventories of radioactive materials that would be in the reactor cores at the end of a core's life (ORNL 2012), commensurate with the highest inventory of fission products. These models used actual plant parameters and fuel types for both the Sequoyah and Browns Ferry reactors. Models were run for both typical 100 percent LEU reactor cores and for reactor cores using partial MOX fuel (approximately 40 percent MOX fuel and 60 percent LEU fuel), as is currently anticipated for the potential use of MOX fuel to disposition surplus plutonium.

As these reactors are currently licensed by NRC to operate with LEU fuel, representative accidents were selected from current TVA licensing documents for comparison of the impacts if partial-MOX fuel cores were substituted for the licensed full-LEU fuel cores. It should be noted that, before MOX fuel could be used in these reactors or any commercial reactors in the United States, detailed safety analyses in support of licensing amendment requests would evaluate the probability of the occurrence and consequences of all accident possibilities while using MOX fuel. These analyses would be reviewed by NRC prior to granting licensing amendments to use MOX fuel.

For the purposes of comparison in this *SPD Supplemental EIS*, representative design-basis accidents and beyond-design-basis accidents were selected from TVA safety analyses. Impacts from the potential releases associated with these accidents were examined for both full-LEU and partial-MOX fuel cores for both the Sequoyah and Browns Ferry reactors to see whether the use of MOX fuel made a substantial difference in the projected impacts of design-basis or beyond-design-basis accidents.

J.2 Background

MOX fuel was first used in a thermal reactor in 1963, but did not come into widespread commercial use until the 1980s. From the 1960s to the 1980s, significant amounts of MOX fuel testing were performed at various reactors in the United States. Plutonium was fabricated into MOX fuel, irradiated, and tested in numerous test and commercial reactors in the 1960s and 1970s. In the Saxton Plutonium Program, nuclear fuel was irradiated in a test reactor at the Westinghouse Reactor Evaluation Center (Waltz Mill, Pennsylvania) in 1965 (ORNL 2000). MOX fuel was also tested in Quad Cities Nuclear Power Station (Cordova, Illinois) in the 1970s, and a 1998 report summarized a more recent examination of the Quad Cities Nuclear Power Station irradiations (ORNL 1998). From 1969 to 1976, MOX fuel was used in the Big Rock Point Nuclear Power Plant (Charlevoix, Michigan). Much of the U.S. work ultimately culminated in the *Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors (NUREG-0002)* (NRC 1976).

Beginning in the 1970s, commercial use of reactor-grade MOX fuel occurred in several European countries, as well as in Japan. Introduction of MOX fuel into the fuel cycle has had its own challenges with regards to plutonium feed, production and handling, fabrication and assembly design, and operation and performance (IAEA 2003). Nevertheless, through the years, the use of MOX fuel became a routine part of nuclear reactor operations and much operational experience with this fuel has been gathered. About 40 reactors in Europe (in Belgium, Switzerland, Germany and France) were licensed to use MOX fuel, and over 30 used it regularly. France plans to have all of its 900 megawatt-electric series of reactors running with cores that are at least one-third MOX fuel (WNA 2011). In Japan, about 10 reactors were licensed to use or were planned to use MOX fuel. These reactors generally used cores that were about one-third MOX fuel.

The overall widespread success of the use of MOX fuel in power reactors greatly reduced technical uncertainties regarding the decision to proceed with MOX fuel as a major disposition option for surplus plutonium. Activities involving the use of MOX fuel were restarted in the United States in the mid-1990s, when the feasibility of dispositioning surplus plutonium as MOX fuel was explored by the U.S. Department of Energy (DOE) and the National Academy of Sciences (NAS 1995). A number of evaluations, analyses, and tests involving a wide variety of reactors (pressurized water reactors [PWRs], boiling water reactors [BWRs], and heavy water reactors [HWRs]) were conducted to determine how MOX fuel fabricated with surplus weapons-usable plutonium performs and how it differs from MOX fuel fabricated with reactor-grade plutonium and LEU fuel. Numerous government-, vendor-, and utility-sponsored scoping studies and comprehensive assessments covering the in-core performance of weapons-usable plutonium-based MOX fuel, as well as the reactor operational and accident responses, have been performed in the United States and internationally. Descriptions of U.S. MOX fuel demonstrations and of international experience in the use of MOX fuel have been prepared by NRC and the Electric Power Research Institute (NRC 1999, EPRI 2009).

Recent evaluations included the completion of testing of weapons-usable plutonium-based MOX fuel test rods in the Advanced Test Reactor in Idaho from 1998 to 2004, with subsequent postirradiation examinations of the test rods at ORNL (ORNL 2005; ORNL 2006). That evaluation was performed primarily as a generic test for gallium impurities in the plutonium, which could be a difference between the use of weapons-derived material and the historical and worldwide experience with MOX fuel. After down-selection of a MOX fuel fabricator/utility consortia by DOE (at the time, Duke Power, Inc; Cogema; and Stone & Webster, Inc.), the irradiation of MOX fuel lead test assemblies (LTAs) was approved by NRC and took place in Duke Power's Catawba Nuclear Station Unit 1 PWR (York, South Carolina) beginning in May 2005, as discussed below. Prior to this LTA testing, the last MOX fuel assemblies irradiated in a commercial U.S. nuclear power plant were irradiated in the R.E. Ginna Nuclear Power Plant (Ontario, New York) in 1985.

As part of the DOE SPD MOX Fuel Qualification Program, four PWR LTAs using weapons-usable plutonium MOX fuel were irradiated in the Duke Energy Catawba Nuclear Station Unit 1 between 2005 and 2008. These LTAs were 17×17 fuel assemblies that were similar in design to those used at the TVA Sequoyah reactors. At the end of two cycles, these LTAs had an average assembly and peak fuel rod burnup of 41.8 and 47.3 gigawatt-days per metric ton heavy metal, respectively. Poolside nondestructive examinations were performed on the four LTAs after each cycle of irradiation. After the second cycle, five fuel rods were removed from one of the LTAs and sent to ORNL for hot cell post-irradiation examination. The purpose of this program was to compare post-irradiation examination measurements to computer code predictions and the accumulated experience with reactor-grade MOX fuel and LEU fuel at similar burnup levels (AREVA 2012).

The poolside nondestructive examinations and hot cell fuel rod post-irradiation examination measured the following MOX fuel irradiation thermal, mechanical, and chemical performance behavior and mechanisms: (1) fuel assembly axial growth, bowing, hold-down spring relaxation, and visual appearance; (2) fuel rod external axial growth, oxidation, hydride formation, surface fretting, ridging, crud formation, and integrity; (3) fuel rod internal pressure, void volume, gas analysis, burnup distribution, fuel pellet microstructure, density, and stack height; (4) cladding microstructure; (5) guide tube oxidation; (6) spacer grid width; and (7) migration and impact of fuel pellet gallium on cladding (AREVA 2012).

Measured values were compared to predictions made using the AREVA COPERNIC2 fuel rod design computer code, as well as post-irradiation data from other MOX fuel tests. Most measured parameters were found to be bounded by or similar to the COPERNIC2 calculations and comparable to AREVA's extensive irradiation experience with MOX fuel. The measured maximum fuel assembly axial growth exceeded predicted values by less than 0.05 inches (0.13 centimeters) as compared to the fabricated assembly total axial length of 159.8 inches (406 centimeters), but remained within a range that does not impact safety. This axial growth is due to a change in dimension of the control rod guide tubes and not the MOX fuel rods in the fuel assembly. Similar behavior has been observed in the same design fuel assembly using LEU fuel and is therefore not related to the use of MOX fuel. However, because the axial growth of three of the four LTAs exceeded the criterion for reinsertion for a third cycle of irradiation, the LTAs were discharged to the used fuel pool after the second cycle. In summary, extensive poolside nondestructive examinations and hot cell post-irradiation examination of the four weapons-grade plutonium MOX LTAs showed close agreement with computer code predictions and other MOX fuel experience for most performance behavior. No issues that would affect the safe operation of the core were found, although higher than predicted axial fuel assembly growth in three LTAs prevented a third cycle of irradiation (AREVA 2012).

The principal technical challenges associated with MOX fuel use include reactivity control and maintenance of adequate shutdown margins due to reduced effectiveness of neutron absorber materials (control rods and soluble boron) in the hardened neutron spectrum (i.e., higher-energy neutrons than in an LEU fuel core) resulting from the presence of plutonium (EPRI 2009). In addition, several facility design and operational issues must be addressed for receipt, handling, and storage of fresh MOX fuel and for the management of used MOX fuel due to higher heat loads, increased neutron dose rates, and reduced effectiveness of reactivity control materials in the used fuel pool and in dry storage systems. Experience at reactors in Europe and with the use of LTAs at the Catawba Nuclear Station in the United States have shown how these technical challenges can be met, and how MOX fuel performance and reliability is comparable to those of standard LEU fuel (AREVA 2012; EPRI 2009).

Given the safety margins incorporated into light water reactor designs, most existing U.S. reactor designs could accommodate partial (30 to 40 percent) MOX fuel cores with relatively minor plant modifications and operational changes (EPRI 2009). This mix of MOX and LEU fuel has already been in use in Europe and Japan and has been analyzed by U.S. national laboratories and NRC (INEL 2009; NRC 2005).

U.S. light water reactors using MOX fuel would need to comply with NRC requirements, and would require amendment of the reactor operating license. MOX fuel would be transported in NRC-certified packages using DOE's Secure Transportation Assets or escorted commercial trucks as discussed in Appendix E.

J.2.1 Operation with Mixed Oxide Fuel

There are differences in the design and performance of MOX fuel compared to LEU fuel. The differences in nuclear reactor core physics for plutonium and uranium result in important issues for core reactivity due to (1) overall decreased effectiveness of materials that serve to reduce or suppress reactivity (control/shutdown rods, soluble boron, gadolinium, xenon) and (2) changes in fuel and moderator temperature responses that reduce shutdown margins (EPRI 2009). The reduced effectiveness of reactivity control materials for MOX fuel, notably control/shutdown rods and soluble boron, means that MOX fuel use would likely require one or more of the available options to enhance reactivity control.

For PWRs with partial-MOX fuel cores, reactivity control modifications could include increasing soluble boron concentrations, using enriched boron in coolant systems, replacing partial-length control rods with full-length rods, employing integral burnable absorbers, and/or using higher-worth control rods. It is worth noting that burnable absorber materials and applications were primarily developed for LEU fuel cores. Further development of burnable absorber technology optimized for the MOX fuel core environment could improve core design flexibility, fuel utilization, and overall commercial viability of MOX fuel use in PWRs. Another method to control the reactivity effect of MOX fuel is to locate MOX fuel assemblies away from rod control cluster assembly positions in PWR cores to preserve control rod worth (ORNL 1997). The current analysis assumed the use of 17×17 PWR fuel with 20 gadolina (uranium and gadolinium) burnable poison rods (with 2 weight-percent gadolinium oxide) in each fuel assembly and a fuel-cycle, time-averaged concentration of natural boron of 867 parts per million.

For BWRs, the impact of MOX fuel on control rod worth is less pronounced due to relatively large water gaps between bundles, which allow for recovery of thermal neutron fluxes. Accordingly, BWR cores offer the flexibility of scattering MOX fuel assemblies throughout the core and limiting the number of MOX fuel assemblies assigned to a control blade location to one or two (IAEA 2003). Other reactivity control measures for use in BWRs with partial-MOX fuel cores include incorporation of burnable absorbers or poisons such as gadolinium to provide additional reactivity control early in the irradiation cycle to counteract the effects of fresh fuel, including power peaking. Burnable absorbers can be inserted into the fuel assembly as discrete elements/rods or incorporated into the fuel itself as integral burnable absorbers applied as coatings on fuel pellet surfaces. Integral burnable absorbers in partial-MOX fuel cores provide flexibility for controlling early cycle reactivity. General Electric highlighted its gadolinium-based integral burnable absorber technology as a promising application under active commercial development for use in its international BWR designs (EPRI 2009; ORNL 1997). The current analysis assumed the use of 10 × 10 BWR fuel with individual fuel assemblies comprised of 13 to 28 fuel rods containing 2.2 to 8 weight-percent gadolinium oxide.

The current understanding of MOX and LEU fuel is such that implementing a partial-MOX fuel core is technically reasonable and has been previously accomplished in a number of reactors around the world. It is acknowledged that MOX fuel loadings above certain levels in the reactor core would likely result in modifications to reactivity control systems, worker radiation protection, core fuel management design, technical specifications, and transient behavior. In future NRC licensing applications, licensees would be required to provide the technical bases for remaining within the plant safety envelope, which may involve fuel management, operations, technical specifications, and design modifications. There is ample evidence from the use of MOX fuel in foreign nuclear power reactors that this can be safely accomplished.

The analysis presented in this *SPD Supplemental EIS* is provided to update the analysis presented in the *SPD EIS*. Before MOX fuel can be used in any domestic, commercial nuclear power plant, NRC's approval of its use would be required. The NRC decisionmaking process is based on a set of submittals by the licensee, which would provide detailed safety analyses and include relevant design and operational plant modifications that would allow the licensee to continue to operate its plant safely with partial-MOX fuel cores.

J.2.2 Sequoyah and Browns Ferry Nuclear Plant Low-Enriched Uranium and Partial Mixed Oxide Core Inventory Development

Representative core inventories for both full-LEU and partial-MOX fuel cores were developed for the Sequoyah and Browns Ferry reactors to support the accident analysis presented in this *SPD Supplemental EIS* (ORNL 2012). Models were developed for full-LEU and partial-MOX fuel cores in the Sequoyah and Browns Ferry reactors

Sequoyah. The Sequoyah fuel and reactor parameters were used to develop the new reactor core inventories. The Sequoyah data reflect three different plutonium enrichments in the partial MOX fuel assembly, with an average enrichment of 4.35 weight-percent plutonium. The Sequoyah models for each type of assembly contain 20 gadolina (uranium and gadolinium) rods with 3 weight-percent uranium-235 and 2 weight-percent gadolinium oxide.

To simulate a normal plant refueling cycle at Sequoyah, the MOX fuel portion of the partial-MOX fuel core was assumed to include approximately 50 percent once-burned (i.e., gone through one irradiation cycle), and 50 percent twice-burned (i.e., gone through two irradiation cycles) assemblies with an average enrichment value of approximately 4.35 percent. Approximately 37 percent of the partial-MOX fuel core would include MOX fuel. The LEU portion of the partial-MOX fuel core was assumed to include approximately 40 percent once-burned, 40 percent twice-burned, and 20 percent thrice-burned (i.e., gone through three irradiation cycles) assemblies, with an average enrichment value of 4.39 percent. The full-LEU fuel core was assumed to include approximately 42 percent once-burned, 42 percent twice-burned, and 16 percent thrice-burned assemblies, with an average enrichment value of 4.43 percent. All analyses assumed end-of-cycle inventories to produce the highest consequences. Fuel cycles were based on an 18-month refueling schedule, with about a 40-day downtime period between cycles.

For the Sequoyah MOX fuel core, assembly models for each enrichment in the core were run up to 60 gigawatt-days per metric ton heavy metal to cover expected burnup ranges. The average burnup of the MOX fuel portion of the partial-MOX fuel core was approximately 34 gigawatt-days per metric ton heavy metal. The average burnup of the LEU fuel portion of the partial-MOX fuel core was approximately 39 gigawatt-days per metric ton heavy metal. The average burnup of the full-LEU fuel core was approximately 38 gigawatt-days per metric ton heavy metal (ORNL 2012).

Browns Ferry. For the Browns Ferry case, the analysis used the ATRIUM 10 design. This assembly is heterogeneous and may come in many variations that incorporate rods with gadolina in different percentages and with different numbers of uranium and gadolinium rods in different locations.

To simulate a normal plant refueling cycle at Browns Ferry, the MOX fuel portion of the partial-MOX fuel core was assumed to include approximately 39 percent once-burned, 39 percent twice-burned, and 22 percent thrice-burned assemblies, with an average enrichment value of approximately 4.17 percent. Approximately, 45 percent of the partial-MOX fuel core would include MOX fuel. The LEU portion of the partial-MOX fuel core was assumed to include approximately 48 percent once-burned, 48 percent twice-burned, and 4 percent thrice-burned assemblies, with an average enrichment value of 4.12 percent. The full-LEU fuel core was assumed to include approximately 41 percent once-burned, 41 percent twice-burned, and 18 percent thrice-burned assemblies, with an average enrichment value of 4.11 percent. All analyses assumed end-of-cycle inventories to produce the highest consequences. Fuel cycles were based on a 24-month refueling schedule, with about a 40-day downtime period between cycles.

For the Browns Ferry MOX fuel core, the assemblies have up to 5 axial regions with different average enrichments and lattices to provide a specific average enrichment for the assembly. For each BWR MOX and LEU fuel assembly, two lattices were modeled to represent the assemblies. The average burnup of the MOX fuel portion of the partial-MOX fuel core was approximately 31 gigawatt-days per metric ton heavy metal. The average burnup of the LEU fuel portion of the partial-MOX fuel core was approximately 34 gigawatt-days per metric ton heavy metal. The average burnup of the full-LEU fuel core was approximately 35 gigawatt-days per metric ton heavy metal (ORNL 2012).

Table J–1 presents the results of these core inventory calculations (ORNL 2012). For both Sequoyah and Browns Ferry, the MOX fuel would be fabricated using depleted uranium (approximately 0.25 weight-percent uranium-235). The isotopes used in the accident analysis are also listed in the table.

Table J–1 Partial Mixed Oxide and Full Low-Enriched Uranium Core Inventories for the Sequoyah and Browns Ferry Nuclear Power Plants

	Sequoyah Nuclear Plant		*Browns Ferry Nuclear Plant*	
Isotope [a]	*Partial-MOX Fuel Core (curies)*	*Full-LEU Fuel Core (curies)*	*Partial-MOX Fuel Core (curies)*	*Full-LEU Fuel Core (curies)*
Americium-241	2.79×10^{4}	1.35×10^{4}	3.48×10^{4}	1.95×10^{4}
Americium-242	1.42×10^{7}	8.25×10^{6}	1.48×10^{7}	9.78×10^{6}
Americium-242m	1.39×10^{3}	6.20×10^{2}	1.41×10^{3}	7.75×10^{2}
Americium-243	3.60×10^{3}	2.28×10^{3}	3.53×10^{3}	2.82×10^{3}
Americium-244	1.42×10^{6}	9.20×10^{5}	9.63×10^{5}	8.15×10^{5}
Americium-245	2.30×10^{3}	1.57×10^{3}	1.43×10^{3}	1.34×10^{3}
Barium-137m	1.08×10^{7}	1.05×10^{7}	1.41×10^{7}	1.53×10^{7}
Barium-139	1.61×10^{8}	1.66×10^{8}	1.91×10^{8}	1.97×10^{8}
Barium-140	1.54×10^{8}	1.60×10^{8}	1.84×10^{8}	1.90×10^{8}
Barium-141	1.44×10^{8}	1.50×10^{8}	1.72×10^{8}	1.77×10^{8}
Barium-142	1.33×10^{8}	1.40×10^{8}	1.60×10^{8}	1.67×10^{8}
Bromine-83	9.58×10^{6}	1.09×10^{7}	1.21×10^{7}	1.33×10^{7}
Bromine-84	1.57×10^{7}	1.84×10^{7}	2.03×10^{7}	2.26×10^{7}
Cerium-141	1.45×10^{8}	1.50×10^{8}	1.73×10^{8}	1.79×10^{8}
Cerium-143	1.31×10^{8}	1.40×10^{8}	1.59×10^{8}	1.67×10^{8}
Cerium-144	1.09×10^{8}	1.19×10^{8}	1.35×10^{8}	1.47×10^{8}
Curium-242	8.16×10^{6}	4.53×10^{6}	9.02×10^{6}	5.95×10^{6}
Curium-243	3.48×10^{3}	1.78×10^{3}	3.28×10^{3}	2.18×10^{3}
Curium-244	6.44×10^{5}	3.72×10^{5}	5.07×10^{5}	4.29×10^{5}
Curium-245	1.15×10^{2}	5.45×10^{1}	6.28×10^{1}	4.82×10^{1}
Curium-246	2.45×10^{1}	1.04×10^{1}	1.41×10^{1}	1.22×10^{1}
Cobalt-58	2.56×10^{-11}	1.95×10^{-11}	3.03×10^{-11}	2.56×10^{-11}
Cobalt-60	1.17×10^{-9}	1.19×10^{-9}	1.52×10^{-9}	1.71×10^{-9}
Cesium-134	1.98×10^{7}	1.86×10^{7}	1.94×10^{7}	2.23×10^{7}
Cesium-135	5.95×10^{1}	5.01×10^{1}	8.34×10^{1}	8.27×10^{1}
Cesium-136	5.43×10^{6}	4.60×10^{6}	5.27×10^{6}	5.34×10^{6}
Cesium-137	1.14×10^{7}	1.11×10^{7}	1.48×10^{7}	1.62×10^{7}
Cesium-138	1.69×10^{8}	1.75×10^{8}	2.01×10^{8}	2.06×10^{8}
Europium-154	9.18×10^{5}	7.39×10^{5}	9.32×10^{5}	9.05×10^{5}
Europium-155	4.85×10^{5}	4.08×10^{5}	5.84×10^{5}	5.73×10^{5}
Iodine-129	3.36×10^{0}	2.80×10^{0}	4.26×10^{0}	4.03×10^{0}
Iodine-130	2.00×10^{6}	1.79×10^{6}	2.18×10^{6}	2.24×10^{6}
Iodine-131	9.43×10^{7}	9.26×10^{7}	1.09×10^{8}	1.08×10^{8}
Iodine-132	1.37×10^{8}	1.35×10^{8}	1.58×10^{8}	1.58×10^{8}
Iodine-133	1.87×10^{8}	1.89×10^{8}	2.18×10^{8}	2.22×10^{8}
Iodine-134	2.07×10^{8}	2.11×10^{8}	2.43×10^{8}	2.49×10^{8}
Iodine-135	1.79×10^{8}	1.80×10^{8}	2.09×10^{8}	2.11×10^{8}
Krypton-83m	9.54×10^{6}	1.09×10^{7}	1.20×10^{7}	1.32×10^{7}
Krypton-85	1.09×10^{6}	1.29×10^{6}	1.49×10^{6}	1.89×10^{6}
Krypton-85m	1.97×10^{7}	2.36×10^{7}	2.59×10^{7}	2.93×10^{7}
Krypton-87	3.75×10^{7}	4.53×10^{7}	4.96×10^{7}	5.64×10^{7}
Krypton-88	4.91×10^{7}	5.99×10^{7}	6.54×10^{7}	7.48×10^{7}
Lanthanum-140	1.68×10^{8}	1.74×10^{8}	1.91×10^{8}	1.98×10^{8}
Lanthanum-141	1.45×10^{8}	1.51×10^{8}	1.73×10^{8}	1.78×10^{8}
Lanthanum-142	1.38×10^{8}	1.45×10^{8}	1.66×10^{8}	1.72×10^{8}
Lanthanum-143	1.30×10^{8}	1.39×10^{8}	1.58×10^{8}	1.66×10^{8}
Molybdenum-99	1.69×10^{8}	1.71×10^{8}	1.97×10^{8}	2.01×10^{8}
Niobium-95	1.41×10^{8}	1.53×10^{8}	1.73×10^{8}	1.84×10^{8}

Isotope [a]	Sequoyah Nuclear Plant		Browns Ferry Nuclear Plant	
	Partial-MOX Fuel Core (curies)	Full-LEU Fuel Core (curies)	Partial-MOX Fuel Core (curies)	Full-LEU Fuel Core (curies)
Niobium-97	1.51×10^{8}	1.57×10^{8}	1.80×10^{8}	1.86×10^{8}
Niobium-97m	2.38×10^{5}	2.07×10^{5}	2.52×10^{5}	2.26×10^{5}
Neodymium-147	5.84×10^{7}	5.99×10^{7}	6.91×10^{7}	7.10×10^{7}
Neptunium-237	2.79×10^{1}	3.47×10^{1}	3.15×10^{1}	4.41×10^{1}
Neptunium-238	3.24×10^{7}	3.97×10^{7}	2.63×10^{7}	3.84×10^{7}
Neptunium-239	1.91×10^{9}	1.90×10^{9}	1.88×10^{9}	1.95×10^{9}
Neptunium-240	1.31×10^{6}	1.31×10^{6}	9.13×10^{5}	9.82×10^{5}
Palladium-107	1.65×10^{1}	1.10×10^{1}	2.00×10^{1}	1.52×10^{1}
Promethium-147	1.66×10^{7}	1.74×10^{7}	2.43×10^{7}	2.59×10^{7}
Praseodymium-143	1.27×10^{8}	1.35×10^{8}	1.57×10^{8}	1.65×10^{8}
Praseodymium-144	1.10×10^{8}	1.20×10^{8}	1.36×10^{8}	1.48×10^{8}
Praseodymium-145	9.05×10^{7}	9.56×10^{7}	1.09×10^{8}	1.14×10^{8}
Plutonium-237	7.30×10^{2}	6.49×10^{2}	4.96×10^{2}	5.83×10^{2}
Plutonium-238	3.09×10^{5}	3.14×10^{5}	3.02×10^{5}	3.87×10^{5}
Plutonium-239	5.50×10^{4}	3.46×10^{4}	6.26×10^{4}	4.26×10^{4}
Plutonium-240	8.95×10^{4}	4.67×10^{4}	1.19×10^{5}	6.77×10^{4}
Plutonium-241	2.30×10^{7}	1.35×10^{7}	2.35×10^{7}	1.53×10^{7}
Plutonium-242	2.81×10^{2}	1.84×10^{2}	3.39×10^{2}	2.53×10^{2}
Plutonium-243	5.38×10^{7}	3.75×10^{7}	4.27×10^{7}	3.50×10^{7}
Plutonium-244	1.09×10^{-4}	6.46×10^{-5}	8.02×10^{-5}	6.87×10^{-5}
Plutonium-245	5.24×10^{2}	3.01×10^{2}	2.40×10^{2}	2.17×10^{2}
Rubidium-86	2.34×10^{5}	2.64×10^{5}	2.13×10^{5}	2.80×10^{5}
Rubidium-88	5.03×10^{7}	6.11×10^{7}	6.67×10^{7}	7.61×10^{7}
Rubidium-89	6.58×10^{7}	8.03×10^{7}	8.76×10^{7}	1.00×10^{8}
Rhodium-103m	1.62×10^{8}	1.47×10^{8}	1.78×10^{8}	1.66×10^{8}
Rhodium-105	1.16×10^{8}	9.77×10^{7}	1.20×10^{8}	1.06×10^{8}
Rhodium-106	7.85×10^{7}	5.94×10^{7}	8.41×10^{7}	6.88×10^{7}
Rhodium-107	7.35×10^{7}	5.92×10^{7}	7.42×10^{7}	6.28×10^{7}
Ruthenium-103	1.64×10^{8}	1.49×10^{8}	1.80×10^{8}	1.68×10^{8}
Ruthenium-105	1.24×10^{8}	1.06×10^{8}	1.29×10^{8}	1.15×10^{8}
Ruthenium-106	7.08×10^{7}	5.22×10^{7}	7.71×10^{7}	6.19×10^{7}
Antimony-125	1.13×10^{6}	9.36×10^{5}	1.36×10^{6}	1.24×10^{6}
Antimony-127	1.00×10^{7}	9.00×10^{6}	1.08×10^{7}	1.00×10^{7}
Antimony-129	2.95×10^{7}	2.72×10^{7}	3.23×10^{7}	3.05×10^{7}
Antimony-130	2.70×10^{7}	2.65×10^{7}	3.09×10^{7}	3.06×10^{7}
Samarium-147	1.59×10^{-4}	1.62×10^{-4}	2.75×10^{-4}	3.16×10^{-4}
Samarium-151	4.20×10^{4}	3.39×10^{4}	4.89×10^{4}	4.38×10^{4}
Strontium-89	6.80×10^{7}	8.36×10^{7}	9.04×10^{7}	1.04×10^{8}
Strontium-90	6.69×10^{6}	7.93×10^{6}	9.20×10^{6}	1.19×10^{7}
Strontium-91	8.93×10^{7}	1.06×10^{8}	1.16×10^{8}	1.31×10^{8}
Strontium-92	9.98×10^{7}	1.15×10^{8}	1.27×10^{8}	1.41×10^{8}
Technetium-99	1.42×10^{3}	1.40×10^{3}	1.92×10^{3}	2.11×10^{3}
Technetium-99m	1.49×10^{8}	1.52×10^{8}	1.75×10^{8}	1.78×10^{8}
Technetium-101	1.61×10^{8}	1.60×10^{8}	1.86×10^{8}	1.86×10^{8}
Tellurium-125m	2.48×10^{5}	2.02×10^{5}	3.01×10^{5}	2.74×10^{5}
Tellurium-127	9.77×10^{6}	8.77×10^{6}	1.06×10^{7}	9.82×10^{6}
Tellurium-127m	2.66×10^{5}	2.18×10^{5}	3.00×10^{5}	2.58×10^{5}
Tellurium-129	2.96×10^{7}	2.72×10^{7}	3.23×10^{7}	3.06×10^{7}
Tellurium-129m	1.44×10^{4}	1.22×10^{4}	1.44×10^{4}	1.32×10^{4}
Tellurium-131	8.32×10^{7}	8.28×10^{7}	9.61×10^{7}	9.64×10^{7}

Isotope [a]	Sequoyah Nuclear Plant		Browns Ferry Nuclear Plant	
	Partial-MOX Fuel Core (curies)	Full-LEU Fuel Core (curies)	Partial-MOX Fuel Core (curies)	Full-LEU Fuel Core (curies)
Tellurium-131m	1.49×10^{7}	1.33×10^{7}	1.60×10^{7}	1.48×10^{7}
Tellurium-132	1.33×10^{8}	1.32×10^{8}	1.53×10^{8}	1.54×10^{8}
Tellurium-133	1.05×10^{8}	1.08×10^{8}	1.24×10^{8}	1.28×10^{8}
Tellurium-133m	7.73×10^{7}	7.90×10^{7}	9.11×10^{7}	9.30×10^{7}
Tellurium-134	1.56×10^{8}	1.66×10^{8}	1.89×10^{8}	1.98×10^{8}
Uranium-234	7.38×10^{1}	1.23×10^{2}	1.49×10^{2}	2.08×10^{2}
Uranium-235	1.50×10^{0}	2.59×10^{0}	3.49×10^{0}	4.39×10^{0}
Uranium-236	2.07×10^{1}	3.09×10^{1}	3.15×10^{1}	4.73×10^{1}
Uranium-237	6.48×10^{7}	8.63×10^{7}	5.98×10^{7}	8.32×10^{7}
Uranium-238	2.72×10^{1}	2.74×10^{1}	4.31×10^{1}	4.35×10^{1}
Uranium-239	1.91×10^{9}	1.91×10^{9}	1.88×10^{9}	1.95×10^{9}
Xenon-131m	1.25×10^{6}	1.22×10^{6}	1.42×10^{6}	1.41×10^{6}
Xenon-133	1.80×10^{8}	1.82×10^{8}	2.11×10^{8}	2.14×10^{8}
Xenon-133m	2.48×10^{6}	2.41×10^{6}	2.83×10^{6}	2.81×10^{6}
Xenon-135	6.03×10^{7}	5.30×10^{7}	7.04×10^{7}	6.48×10^{7}
Xenon-135m	3.05×10^{7}	2.92×10^{7}	3.43×10^{7}	3.34×10^{7}
Xenon-138	1.53×10^{8}	1.59×10^{8}	1.83×10^{8}	1.89×10^{8}
Yttrium-90	6.92×10^{6}	8.21×10^{6}	9.49×10^{6}	1.23×10^{7}
Yttrium-91	9.17×10^{7}	1.09×10^{8}	1.19×10^{8}	1.35×10^{8}
Yttrium-91m	5.12×10^{7}	6.06×10^{7}	6.67×10^{7}	7.49×10^{7}
Yttrium-92	1.01×10^{8}	1.16×10^{8}	1.29×10^{8}	1.43×10^{8}
Yttrium-93	1.18×10^{8}	1.32×10^{8}	1.48×10^{8}	1.60×10^{8}
Yttrium-94	1.27×10^{8}	1.40×10^{8}	1.58×10^{8}	1.69×10^{8}
Yttrium-95	1.36×10^{8}	1.47×10^{8}	1.66×10^{8}	1.76×10^{8}
Zirconium-95	1.41×10^{8}	1.52×10^{8}	1.72×10^{8}	1.83×10^{8}
Zirconium-97	1.50×10^{8}	1.56×10^{8}	1.78×10^{8}	1.84×10^{8}

LEU = low-enriched uranium; MOX = mixed oxide.

[a] This is a partial listing of the isotopes that would be in the core at the end of an operational cycle. These are the major isotopes that would contribute to the radiological impacts in the event of an accident.

Source: ORNL 2012.

J.2.3 Meteorological Data

Annual onsite meteorological data for each reactor site from 2005 through 2009 were evaluated. The meteorological data characteristics of the site are described by 1 year of hourly data (8,760 measurements). These data include windspeed, wind direction, atmospheric stability, and rainfall (TVA 2010a). The accident modeling was performed using each year of meteorological data. The years 2006 (Browns Ferry) and 2007 (Sequoyah) were selected for presentation because they result in the highest calculated population doses and therefore provide conservative results.

J.2.4 Population Data

The population distribution around each plant was determined using 2010 and prior decennial census data and projecting to the year 2020. The population was then allocated based on its current location into segments that correspond to a polar coordinate grid. The polar coordinate grid for this analysis consists of 10 radial intervals aligned with the 16 compass directions. For Browns Ferry, the total projected population out to 50 miles (80 kilometers) is about 1.1 million. For Sequoyah, the total projected population out to 50 miles (80 kilometers) is about 1.2 million. Projected population data for the year 2020 corresponding to the grid segments at Browns Ferry and Sequoyah are presented in **Tables J–2** and **J–3**, respectively.

Table J–2 Projected Year 2020 Population near the Browns Ferry Nuclear Plant

	Distance (miles)									
Direction	***0–1***	***1–2***	***2–3***	***3–4***	***4–5***	***5–10***	***10–20***	***20–30***	***30–40***	***40–50***
N	11	46	78	110	142	1,295	2,854	4,000	12,647	8,929
NNE	13	47	78	142	318	2,591	5,952	4,028	9,539	7,084
NE	12	44	66	105	226	6,050	18,358	13,638	8,354	14,249
ENE	8	28	38	53	121	3,333	23,025	41,312	33,507	11,693
E	8	23	38	53	69	1,049	22,441	135,888	104,444	8,163
ESE	8	23	38	53	69	841	2,951	10,851	35,557	13,832
SE	0	0	0	0	0	7,529	33,564	10,186	10,890	26,950
SSE	0	20	35	57	74	7,289	30,659	18,525	29,661	28,354
S	0	10	13	20	44	3,230	7,182	3,406	7,830	11,593
SSW	0	10	13	17	50	1,969	7,880	1,708	3,087	5,601
SW	0	10	13	17	30	810	6,310	2,843	5,047	13,104
WSW	0	10	13	17	22	379	3,411	3,832	18,479	5,861
W	0	10	13	17	22	285	2,547	11,091	30,797	4,239
WNW	0	12	13	17	22	407	3,954	16,886	55,795	7,453
NW	0	0	0	104	87	1,097	5,884	10,127	6,847	4,991
NNW	8	43	78	110	142	1,187	3,394	4,372	16,556	10,247
Total Population										**1,087,041**

Population projected to 2020 using 2010 census data (Census 2011) and prior decennial census data for the area within 50 miles of the Browns Ferry Nuclear Plant.
Note: To convert miles to kilometers, multiply by 1.6093.

Table J–3 Projected Year 2020 Population near the Sequoyah Nuclear Plant

	Distance (miles)									
Direction	***0–1***	***1–2***	***2–3***	***3–4***	***4–5***	***5–10***	***10–20***	***20–30***	***30–40***	***40–50***
N	63	156	72	136	314	2,532	7,186	5,899	5,926	24,236
NNE	0	103	58	81	114	1,276	10,596	9,435	8,709	12,016
NE	0	187	170	216	128	1,226	3,584	7,479	9,536	15,844
ENE	0	227	278	257	293	1,477	6,055	11,689	28,986	28,664
E	0	217	305	347	160	2,103	24,414	8,965	7,901	5,248
ESE	51	163	305	222	135	2,759	53,322	6,896	3,436	17,554
SE	51	161	304	206	251	2,168	12,216	9,113	4,808	14,823
SSE	0	208	273	464	762	4,308	10,817	28,595	62,485	16,564
S	0	207	262	478	771	9,383	40,896	31,620	41,458	22,268
SSW	0	206	282	626	801	8,604	93,860	52,483	20,635	12,969
SW	0	207	564	714	654	10,272	96,974	30,756	14,748	11,089
WSW	50	310	997	1,394	1,387	15,749	41,190	5,527	14,994	8,548
W	51	457	859	1,259	2,019	5,307	4,856	7,445	6,763	7,889
WNW	57	210	350	625	1,092	2,434	4,427	5,214	4,986	5,537
NW	65	210	350	504	1,007	2,419	3,524	4,252	2,182	14,639
NNW	65	210	341	316	358	2,303	1,781	3,504	3,351	6,521
Total Population										**1,211,956**

Population projected to 2020 using 2010 census data (Census 2011) and prior decennial census data for the area within 50 miles of the Sequoyah Nuclear Plant.
Note: To convert miles to kilometers, multiply by 1.6093.

J.3 Reactor Accident Identification and Quantification

As discussed above, the *Supplemental SPD EIS* reactor accident analysis includes an assessment of postulated accidents at TVA reactors at Sequoyah (a PWR) and Browns Ferry (a BWR). The analysis in this *SPD Supplemental EIS* compares the accident results for partial-MOX fuel and full-LEU fuel cores to determine whether the use of MOX fuel in these TVA reactors would make any substantive difference in the potential risks associated with the accidents analyzed.

The postulated accidents include design-basis and beyond-design-basis accidents at each reactor site using both partial-MOX and full-LEU fuel cores. The accidents presented were selected because of their potential to release substantial amounts of radioactive material to the environment. This assessment is patterned after the similar assessment presented in the *SPD EIS* (DOE 1999) and uses similar conventions and assumptions. In the *Final SPD EIS*, design-basis accidents and beyond-design-basis accidents were considered for six PWRs operated by Duke Power and Virginia Power (now Dominion Power). For the current assessment, both a PWR and a BWR were evaluated. Although design features make some of the accident scenarios differ between the PWRs and BWRs, the basic accidents are similar.

Only those accidents with the potential for substantial radiological releases to the environment were evaluated for the purposes of this *SPD Supplemental EIS*. Two design-basis accidents (a loss-of-coolant accident [LOCA] and a used-fuel-handling accident) and four beyond-design-basis accidents (an early containment failure, a late containment failure, a steam generator tube rupture [containment bypass in a PWR], and an interfacing-systems-loss-of-coolant accident [ISLOCA] [containment bypass in a BWR]) meet these criteria. Each of these accidents was analyzed twice: once assuming use of a full-LEU fuel core and once assuming use of a partial-MOX fuel core. As part of its license amendment process, NRC would likely require nuclear reactor plants applying to use MOX fuel to perform additional accident analyses involving other accident scenarios that would likely result in smaller radiological releases to the environment.

These accidents were chosen to highlight differences in the potential impacts to the public due to the use of a partial-MOX fuel core, in the unlikely event that an accident occurred. The LOCA represents a design-basis accident inside the containment that assumes the entire reactor core has failed, thereby releasing a large quantity of fission products to the reactor coolant system and a significant radiological source term to the environment. Similarly, the used-fuel-handling accident represents a design-basis accident outside the containment that releases a significant fraction of fission products within a used nuclear fuel assembly without the ameliorating design features of the containment and its systems. Both the LOCA and used-fuel-handling accident are design-basis accident scenarios that are prescribed by NRC regulations for licensing approval of a commercial nuclear power plant. They do not have any specified annual frequency of occurrence, but are instead used to demonstrate the safety design performance of a specific, sited nuclear power plant and its acceptability with respect to regulatory radiation dose standards. As both the LOCA and used-fuel-handling accident are design-basis accidents that are required by NRC regulations, they are equally applicable to both an LEU and a partial-MOX fuel core. Other NRC prescribed design-basis accidents that could be analyzed would not result in a larger source term than the two selected for this *SPD Supplemental EIS*.

The beyond-design-basis accidents were developed by plant-specific probabilistic risk assessments that postulated a wide spectrum of initiating events followed by different combinations of system and/or component failures, along with operator actions. Some of these events lead to a predicted failure of the reactor core, as in the case of the design-basis LOCA, but with higher release fractions to the environment, different timing of the release, and different plume energies and release heights. Several decades of probabilistic risk assessment experience on the part of the NRC, U.S. national laboratories, licensees, and their contractors have resulted in well-understood, dominant, beyond-design basis accident scenarios.

These are the accident scenarios that were selected for analysis in this *SPD Supplemental EIS*. Each of them results in failure of the entire core, but at different times after reactor shutdown, and different release fractions of groups of fission products, different plume energies, and different release heights. As a group, the selected beyond-design-basis accident scenarios encompass the range of this class of accidents that would be expected to result in the highest radiological consequences to the public. This group of beyond-design-basis accidents also demonstrates the impact and effectiveness of emergency response to ameliorate impacts to the population because differences in the timing of releases allow different emergency response actions such as sheltering and evacuation. As part of its license amendment process, NRC would likely require nuclear reactor plants applying to use MOX fuel to perform additional accident analyses involving other accident scenarios.

The frequencies associated with the accident scenarios evaluated in this *SPD Supplemental EIS* are not expected to be dependent on the fuel type inside the reactor. A recent analysis of severe accidents for reactors using partial-MOX fuel cores determined them to have a similar accident progression (i.e., source term, timing, plume energy) as those for a full-LEU fuel core in a number of scenarios including early and late containment failures (SNL 2010). These frequencies are event-based (e.g., frequency of an initiating event such as loss of offsite power) and depend on systems- and operational-response-related events (mitigation activities with probabilities to accomplish the required actions). For example, an early containment failure at these reactors due to a station blackout (e.g., loss of offsite and onsite [emergency diesel generator] power) as an initiating event and failure to provide emergency and long-term cooling in a timely manner, leading to core melt/containment failure, does not depend on whether the reactor uses a partial-MOX or full-LEU fuel core, but rather on the likelihood of a series of events occurring that are unrelated to the fuel type. The decay heat removal and other control systems that need to be operational in the event of such an accident are the same as those designed for operation with LEU fuel. TVA, as part of its license amendment submittal to NRC, would evaluate and may modify plant operations and core design to allow for the use of MOX fuel and remain within the envelope of accident response for the types of accidents that have been analyzed in the plant's probabilistic risk assessment (PRA), which was the basis for the selection of accidents analyzed in this *SPD Supplemental EIS*.

For this *SPD Supplemental EIS*, postulated design-basis and beyond-design-basis accidents were analyzed using the MACCS2 computer code[1] for each of the proposed reactor sites. Doses (consequences) and risks to the offsite maximally exposed individual (MEI) and the general public within 50 miles (80 kilometers) of each plant, using the population distributions shown in Section J.2.4 for each accident scenario, were calculated for each type of core. Impacts at the time of the accident would be from direct radiation exposure and inhalation of the passing plume. The longer-term effects from radionuclides deposited on the ground and surface waters after the accident were modeled for reactor accidents. Exposure pathways include resuspension and inhalation of plutonium and ingestion of contaminated crops. The MACCS2 code calculates the dose over a number of years, incorporating a number of factors including radioactive decay. In the case of this *SPD Supplemental EIS*, the reactor accident doses were calculated over an 80-year period to represent a typical lifetime. These results were then compared for the partial-MOX and full-LEU fuel cores, by plant, for each postulated accident.

The MEI dose was calculated at the exclusion area boundary of each plant. The exclusion area boundary is that surrounding the reactor within which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided any one of these is not so close to the facility that it interferes with normal operation of the facility, and appropriate and effective arrangements are made to

[1] *MACCS2, version 1.13.1, was used in the analysis. This version of the code is contained in the DOE Office of Health, Safety, and Security safety software "toolbox" of codes. All such codes are compliant with the DOE Safety Software Quality Assurance requirements of DOE O 414.1D and its safety software guidance, DOE G 414.1-4 (http://www.hss.doe.gov/nuclearsafety/qa/sqa/central_registry.htm). MACCS2 is also used by the NRC to calculate impacts from postulated severe accidents in nuclear power plant reactors and support decisionmaking (http://www.nrc.gov/about-nrc/regulatory/research/comp-codes.html).*

control traffic and protect public health and safety on the highway, railroad, or waterway in an emergency. There are generally no residences within an exclusion area. However, if there were residents, they would be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided no significant hazards to the public health and safety would result.

Sources of information. Both design-basis accidents and beyond-design-basis accidents were identified from plant safety analysis documents developed by TVA. Design-basis accidents were selected by reviewing the Updated Final Safety Analysis Report (UFSAR) for each plant (TVA 2007, 2010b). Beyond-design-basis accidents were identified from the submittals in response to NRC's requirements for reactor licensees to perform Individual Plant Examinations (IPEs) for severe accident vulnerabilities, as well as subsequent updates and revisions developed for license renewal (TVA 2002a, 2002b, 2003). Source terms for each accident in terms of the fraction of the reactor core inventory that might be released as a function of time for the full-LEU fuel cores were identified from these documents. These specific, time-dependent release fractions were then applied to the reactor core inventories developed by ORNL for both the full-LEU and partial-MOX fuel cores for Sequoyah and Browns Ferry.

For Sequoyah, a recent Level 3 PRA that developed accident source terms and consequences was not available. However, such an analysis was available for the Watts Bar Nuclear Plant (SAIC 2007), a sister plant to Sequoyah. Sequoyah and the Watts Bar Nuclear Plant are similar plants, both with two Westinghouse PWRs with ice condenser containments. Because of these similarities, the release paths and mitigating mechanisms for the two plants were assumed to be identical.

Key modeling assumptions. It is well known that accident progression and modeling assumptions can make a substantial difference regarding the estimated impacts of an accident. For this reason, the impacts evaluated in this *SPD Supplemental EIS* use the standard accident progression assumptions included in TVA licensing activities and in the standard, NRC-sponsored MACCS2 computer code for evaluation of reactor accident impacts. The accident evaluations presented in this *SPD Supplemental EIS* are used for comparison of the relative impacts of accident scenarios. The real-world impacts associated with any of these accidents, should they occur, would likely be less than those predicted in these accident analyses. This is because conservative values were chosen for a number of the analysis parameters. For example, perpetual rain (resulting in wet deposition) was assumed in the region between 40 and 50 miles (64 and 80 kilometers) from a release point, maximizing the exposure of the population in this region; release plumes were assumed to be neutrally buoyant, resulting in higher concentrations to receptors close to the release point; and all beyond-design-basis accidents were assumed to result in ground-level release. The combined, multiplicative effect of these conservative assumptions is that the impacts shown in this *SPD Supplemental EIS* are likely overestimated.

Assumptions that can substantially influence calculations of close-in doses include how the release occurs and whether there is sufficient energy for plume rise so that close-in locations do not receive high doses. For the analyses in this *SPD Supplemental EIS*, assumptions for beyond-design-basis accidents, such as early containment failure (defined in Section J.3.2), were made that maximize the estimated close-in doses, such as those assessed at the exclusion area boundary, should a member of the public be located there at the time of the accident. More realistically, a plume would likely pass over the exclusion area boundary and, at the point the plume reached the ground, it would be more diluted and doses to individuals in the affected population would be comparatively lower than the dose to an individual at the exclusion area boundary.

For some accidents, such as late containment failure (defined in Section J.3.2), it was assumed that most radioactive material would be released a number of hours after the initial accident. This would allow time for many emergency actions to occur, including evacuation of most of the nearby population. The dose that these members of the population might receive while still located near the reactor, or during evacuation, is highly dependent on the timing of the accident sequence and the timing of evacuation. The analysis for this *SPD Supplemental EIS* used standard, site-specific assumptions used by TVA in NRC licensing activities and local emergency planning preparations. As discussed in Section J.3.2, it was assumed that 95 percent of the affected population within the emergency planning zone would begin to be evacuated shortly after a warning was issued by emergency response officials; however, the timing of accident sequences and evacuation and certain other assumptions differed between the reactor sites. These assumptions differed because of differences in the types of reactors (PWR and BWR) operating at the two sites, as well as the assumptions TVA made to account for differences in local geography, roads, and population distributions.

For reactor accidents, the surrounding population could receive an initial radiation dose from direct radiation exposure and inhalation of the initial plume, and over the longer term, from direct radiation exposure, inhalation of resuspended material, and ingestion of contaminated foods. The MACCS2 computer code estimates not only the acute impacts from the initial cloud passage, but also the longer term, chronic impacts from direct radiation exposure, inhalation, and ingestion of contaminated food. In reality, because contamination in food is easily and relatively inexpensively monitored, most of the contaminated food is unlikely to be consumed. Nevertheless, the doses reported in this *SPD Supplemental EIS* for reactor accidents include those due to the chronic effects associated with the long-term ingestion of contaminated food.

J.3.1 Design-Basis Accidents

Design-basis accidents are identified by NRC, and their impacts are evaluated as a part of the NRC regulatory process to demonstrate that the safety features of the plant provide adequate protection of the public. They are defined by NRC as postulated accidents that a nuclear facility must be designed and built to withstand without loss of the systems, structures, and components necessary to ensure public health and safety. These are the most serious events that reactor plants must be designed against and represent limiting design cases.

The accident analyses presented in the Browns Ferry and Sequoyah UFSARs are conservative design-basis analyses and, therefore, the dose consequences are considered bounding (i.e., a more realistic analysis would result in lower doses and, thus, lower consequences). The results, however, provide a comparison of the potential consequences resulting from design-basis accidents. The consequences also provide insight into which design-basis accidents should be analyzed in this *SPD Supplemental EIS*.

After a review of the UFSAR accident analyses, the LOCA and used-fuel-handling accident were selected as design-basis accidents to be evaluated in this *SPD Supplemental EIS*. When compared to other design-basis accidents, such as a rod ejection or a main steam line break, the LOCA and used-fuel-handling accident constitute scenarios that result in source terms that are larger in magnitude and encompass the broadest spectrum of radionuclides and, therefore, are the best design-basis accidents by which to compare the consequences of a partial-MOX fuel core with a full-LEU fuel core.

The LOCA includes damage to 100 percent of the core and releases involving both the fuel gap and the balance of the fuel while coupling releases to actuation of engineered safety systems and the containment. Another design-basis accident, a rod ejection accident, does not result in failure of 100 percent of the core, but rather a smaller fraction of the fuel that is located around the control rod that was ejected. As both of these design-basis accidents involve fuel failure inside containment and the LOCA results in higher source terms and doses to the public, the LOCA was chosen as the representative design-basis accident inside containment. Furthermore, the licensee can institute plant core and control rod design modifications that ameliorate the fuel damage resulting from a rod ejection accident, whereas the LOCA source term is prescribed by regulation. The main steam line break accident source term is related to the allowable coolant activity levels and not fuel design.

Similarly, other design-basis accidents are postulated outside containment in addition to the used-fuel-handling accident (e.g., waste gas tank failure, dropped used fuel cask), but the used-fuel-handling accident was judged to be the best representative of outside-containment, design-basis accidents for the purpose of comparing the consequences of such an accident involving a partial-MOX fuel core with one involving a full-LEU fuel core because it involves a source term directly related to the fuel design. So the differences between a used-fuel-handling accident involving MOX fuel and one involving LEU fuel can be easily compared, and this accident typically results in larger offsite doses than the other outside-containment, design-basis accidents.

The design-basis accidents evaluated in this *SPD Supplemental EIS* are associated with large source terms. Many design-basis and higher-frequency accident scenarios result in no radiological releases or releases that have no relation to the core fission product inventory and are not expected to result in significant differences in consequences due to the presence of MOX fuel. It is likely that future accident analyses that have yet to be developed would be incorporated into license amendment applications to NRC that are developed by licensees that may desire to use partial-MOX fuel cores in their reactors.

Design-basis LOCA. A design-basis large-break LOCA was chosen for evaluation because it is the limiting reactor design-basis accident at both of the TVA plants evaluated in this *SPD Supplemental EIS*. The large-break LOCA is defined as a break equivalent in size to a double-ended rupture of the largest pipe of the reactor coolant system. Following a postulated double-ended rupture of a reactor coolant pipe, the emergency core cooling system would operate as designed, keeping cladding temperatures well below melting and ensuring that the core remains intact and in a coolable geometry. As a result of the increase in cladding temperature and rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. Thus, a fraction of the fission products accumulated in the pellet-cladding gap may be released to the reactor coolant system and thereby to the containment. Although no core melting would occur for the design-basis LOCA, a gaseous release of fission products is evaluated.

The LOCA source terms, including the specific isotope releases in curies as a function of time after the initiation of the accident, were supplied by TVA from current safety documents for Browns Ferry and Sequoyah (TVA 2010a). These estimated releases were used to calculate the plant- and accident-specific fractions of the core released. These fractions were then applied to the current partial-MOX and full-LEU fuel core inventories developed for Browns Ferry and Sequoyah by ORNL (see Table J–1) to determine the specific releases by isotope and time.

The LOCA radiological consequence analysis for the full-LEU and partial-MOX fuel cores was performed assuming a stack release based on TVA-supplied, plant-specific radioisotope release data. The possible leak paths through containment and bypass were included.

Used-fuel-handling accident. In the postulated used-fuel-handling accident scenario, a used fuel assembly is dropped. The drop would result in a breach of the fuel rod cladding, and a portion of the volatile fission gases from the damaged fuel rods would be released. A used-fuel-handling accident would realistically result in damage to only a fraction of the fuel rods. However, consistent with NRC methodology, all the fuel rods in the dropped fuel assembly were assumed to be damaged.

The accident was assumed to occur at the earliest time that fuel-handling operations may begin after shutdown, as identified in each plant's technical specifications to maximize the potential consequences of such an accident. The accident was assumed to start 72 hours after shutdown at Browns Ferry and 100 hours at Sequoyah, based on previous submittals by TVA to NRC.

Consistent with NRC guidance, the assumption in the TVA safety analyses is that an assembly with extremely high burnup (e.g., for Browns Ferry, 50 percent higher than the average core assembly) is damaged while being removed from the reactor. The values for individual fission product inventories in the damaged assembly were calculated assuming full-power operation at the end of core life immediately preceding shutdown. All of the gap activity in the damaged rods was assumed to be released. Releases would be through the top of the containment building to the environment, but the water in the refueling pool would greatly reduce the iodine available for release to the environment. It was assumed that all of the iodine escaping from the refueling pool is released to the environment over a 2-hour time period through the fuel-handling building ventilation system. The Browns Ferry and Sequoyah UFSARs assumed iodine filter efficiencies of 95 percent for both the inorganic and organic species.

The used-fuel-handling accident source terms, including the specific isotope releases in curies as a function of time after the initiation of the accident, were supplied by TVA from current safety documents for Browns Ferry and Sequoyah (TVA 2010a). These estimated releases were used to calculate plant- and accident-specific fractions of the core released as a function of time. These fractions were then applied to the partial-MOX and full-LEU fuel core inventories developed by ORNL for Browns Ferry and Sequoyah to determine the specific release by isotope and time (see Table J–1).

J.3.1.1 Browns Ferry Design-Basis Accident Analysis

Table J–4 presents the results of this analysis for design-basis accidents at Browns Ferry. Results are presented for a hypothetical individual at the exclusion area boundary for the entire period of release from the accident, as well as for persons in the surrounding population. For both accidents, the doses would be small relative to the NRC requirement that an individual located at any point of the boundary of the exclusion area (referred to as the MEI hereafter) for any 2-hour period following the onset of the postulated accident would not receive a total effective dose equivalent in excess of 25 rem (10 CFR 50.34). For the LOCA at Browns Ferry, the dose to the MEI would be 0.026 rem for a full-LEU fuel core and 0.023 rem for a partial-MOX fuel core. In either case, the dose would be small compared to the NRC limit of 25 rem. For the used-fuel-handling accident at Browns Ferry, the dose to the MEI would be approximately 0.00014 rem for either a partial MOX fuel assembly or an LEU fuel assembly. In either case, the used-fuel-handling accident dose would be negligible compared to the NRC limit of 25 rem.

Table J–4 Browns Ferry Nuclear Plant Design-Basis Accident Impacts

Accident	Full-LEU or Partial- MOX Fuel Core	*Impacts on the MEI at the Exclusion Area Boundary* Dose (rem) [a]	NRC Regulatory Limit (rem) [b]	*Impacts on the Population within 50 Miles* Dose (person-rem) [a]	Average Individual Dose (rem) [c]
Loss-of-coolant accident [d]	LEU	0.026	25	150	1.4×10^{-4}
	MOX	0.023	25	150	1.4×10^{-4}
Used-fuel-handling accident [e]	LEU	0.00014	25	0.086	7.9×10^{-8}
	MOX	0.00014	25	0.086	7.9×10^{-8}

LEU = low-enriched uranium; MEI = maximally exposed individual; MOX = mixed oxide; NRC = U.S. Nuclear Regulatory Commission.

[a] The reactor accident doses were calculated over an 80-year period using the MACCS2 computer code. Eighty years represents a typical person's lifetime.

[b] From 10 CFR 50.34 for design-basis accidents.

[c] Average individual dose to the entire offsite projected population in 2020 (approximately 1,100,000) out to a distance of 50 miles for the indicated accident.

[d] Release would be through a 604-foot stack.

[e] Release was assumed to be through the top of the containment building at 173 feet.

Note: To convert feet to meters, multiply by 0.3048; miles to kilometers by 1.6093.

The results also indicate that the impacts on the surrounding population for a design-basis accident with a partial-MOX fuel core or a full-LEU fuel core would be similar and within the overall analysis uncertainty. The dose to the population from the LOCA at Browns Ferry would be approximately 150 person-rem for either a partial-MOX fuel core or a full-LEU fuel core. The average dose to an individual residing within 50 miles (80 kilometers) of Browns Ferry at the time of the accident, in the unlikely event that it occurred, would be 1.4×10^{-4} rem, regardless of the fuel type in the reactor at the time of the accident. The dose to the population from the used-fuel-handling accident at Browns Ferry would be 0.086 person-rem for either a full-LEU or a partial-MOX fuel core. The average dose to an individual from this accident, in the unlikely event that it occurred, would be 7.9×10^{-8} rem, regardless of the fuel type in the dropped fuel assembly at the time of the accident. Therefore, potential risks presented by the two types of cores are projected to be comparable for the MEI or general population surrounding the plant from these design-basis accidents.

J.3.1.2 Sequoyah Design-Basis Accident Analysis

Table J–5 presents the results of the analysis for design-basis accidents at Sequoyah. Results are presented for a hypothetical individual at the exclusion area boundary for the entire period of release from the accident, as well as for persons in the surrounding population. For a LOCA at Sequoyah, the dose to the MEI would be 0.0023 rem for a full-LEU fuel core and 0.0020 rem for a partial-MOX fuel core. In either case, the dose would be small compared to the NRC limit of 25 rem. For the used-fuel-handling accident at Sequoyah, the dose to the MEI would be approximately 0.000036 rem for either a partial MOX fuel assembly or an LEU fuel assembly. In either case, the used-fuel-handling accident dose would be negligible compared to the NRC limit of 25 rem.

Table J–5 Sequoyah Nuclear Plant Design-Basis Accident Impacts

Accident	Full-LEU or Partial- MOX Fuel Core	Impacts on the MEI at the Exclusion Area Boundary		Impacts on the Population within 50 Miles	
		Dose (rem) [a]	NRC Regulatory Limit (rem) [b]	Dose (person-rem) [a]	Average Individual Dose (rem) [c]
Loss-of-coolant accident [d]	LEU	0.0023	25	0.75	6.2×10^{-7}
	MOX	0.0020	25	0.72	5.9×10^{-7}
Used-fuel-handling accident	LEU	0.000036	25	0.018	1.5×10^{-8}
	MOX	0.000036	25	0.018	1.5×10^{-8}

LEU = low enriched uranium; MEI = maximally exposed individual; MOX = mixed oxide; NRC = U.S. Nuclear Regulatory Commission.

[a] The reactor accident doses were calculated over an 80-year period using the MACCS2 computer code. Eighty years represents a typical person's lifetime.

[b] From 10 CFR 50.34 for design basis accidents.

[c] Average individual dose to the entire offsite projected population in 2020 (approximately 1,200,000) out to a distance of 50 miles for the indicated accident.

[d] Release was assumed to be through the top of the containment building at 171 feet.

Note: To convert feet to meters, multiply by 0.3048; miles to kilometers by 1.6093.

The results also indicate that the impacts on the surrounding population for a design-basis accident with a partial-MOX fuel core or a full-LEU fuel core would be both similar and within the overall analysis uncertainty. The dose to the population from a LOCA at Sequoyah would be approximately 0.75 person-rem for a full-LEU fuel core and 0.72 person-rem for a partial-MOX fuel core. The average dose to an individual residing within 50 miles (80 kilometers) of Sequoyah at the time of the accident, in the unlikely event that it occurred, would be approximately 6.0×10^{-7} rem, regardless of the fuel type in the reactor at the time of the accident. The dose to the population from the used-fuel-handling accident at Sequoyah would be approximately 0.018 person-rem for either a partial MOX fuel assembly or an LEU fuel assembly. The average dose to an individual from this accident, in the unlikely event that it occurred, would be 1.5×10^{-8} rem, regardless of the fuel type in the dropped fuel assembly at the time of the accident. Therefore, potential risks presented by the two types of cores are projected to be comparable for the MEI or the general population surrounding the plant from these design-basis accidents.

J.3.2 Beyond-Design-Basis Accidents

Beyond-design-basis accidents (severe reactor accidents) are less likely to occur than design-basis accidents. In design-basis accidents, mitigating systems are assumed to be available. In beyond-design-basis accidents, even though the initiating event could be a design-basis event (e.g., a large-break LOCA), additional failures of mitigating systems such as the emergency core cooling system cause some degree of physical deterioration of the fuel in the reactor core and a possible breach of the containment structure, leading to the direct release of radioactive materials to the environment.

The beyond-design-basis accident evaluation in the *SPD EIS* included a review of each plant's IPE. In 1988, NRC required all licensees of operating plants to perform IPEs for severe accident vulnerabilities (NRC 1988) and indicated that a PRA would be an acceptable approach to performing the IPE. A PRA evaluates, in full detail (quantitatively), the consequences of all potential events caused by operating disturbances (known as internal initiating events) within the plant. A state-of-the-art PRA uses realistic criteria and assumptions in evaluating the accident progression and the systems required to mitigate each accident.

Beyond-design-basis accidents evaluated in this *SPD Supplemental EIS* include only those scenarios that lead to containment bypass or failure because the public and environmental consequences would be significantly less for accident scenarios that do not lead to containment bypass or failure. Accidents that lead to containment bypass or failure are expected to result in the greatest release of core fission products, which could result in different consequences for the same accident, depending on whether the reactor has a full-LEU or a partial-MOX fuel core. The accidents evaluated consist of an early containment failure, a late containment failure, a steam generator tube rupture (for a PWR), and an ISLOCA (for a BWR).

Early containment failure. This accident is defined as the failure of containment prior to or very soon (within a few hours) after breach of the reactor vessel. A variety of mechanisms, such as direct contact of core debris with the containment, rapid pressure and temperature loads, hydrogen combustion, and fuel-coolant interactions, can cause structural failure of the containment. Early containment failure can be important because it tends to result in shorter warning times for initiating public protective measures, and because radionuclide releases would generally be more severe than if the containment fails later.

Late containment failure. A late containment failure involves structural failure of the containment more than a day after accident initiation and typically a day or more after breach of the reactor vessel. A variety of mechanisms, such as gradual pressure and temperature increase, hydrogen combustion, and basemat melt-through by core debris, can cause late containment failure.

Steam generator tube rupture. A beyond-design-basis steam generator tube rupture induced by high temperatures also represents a containment bypass event. Analyses have indicated a potential for very high gas temperatures in the reactor coolant system during accidents involving core damage when the primary system is at high pressure. The high temperature could cause the steam generator tubes to fail in a PWR (BWRs do not use steam generators). As a result of a severe tube rupture, the secondary side could be exposed to full reactor coolant system pressures (approximately 2,250 pounds per square inch). These pressures would cause relief valves to lift on the secondary side as they are designed to do, resulting in pressure being transferred to the containment structure. If these valves fail to close after venting and the pressure in the containment caused it to be breached, an open pathway from the reactor vessel to the environment could result.

ISLOCA. An ISLOCA refers to a class of accidents in which the reactor coolant system pressure boundary interfacing with a supporting system of lower design pressure is breached. If this occurred, the lower pressure system would be over-pressurized and could rupture outside the containment. This failure would establish a flow path directly to the environment or, sometimes, to another building with a lesser ability to handle increased pressure compared to the containment. An ISLOCA could occur at either a PWR or BWR and has been included in this *SPD Supplemental EIS* as a representative over-pressurization accident for a BWR.

As discussed in Section J.2.2, ORNL developed an end-of-core-life inventory for both full-LEU and partial-MOX fuel cores for Browns Ferry and Sequoyah (see Table J–1). For the source term and offsite consequence analysis of beyond-design-basis accidents, the radioactive species are collected into classes of isotopes that exhibit similar chemical behavior. The following groups represent the isotopes considered to be most important to offsite consequences in the event of a beyond-design-basis accident: noble gases (including krypton and xenon); iodine (including bromine); cesium (including rubidium); tellurium (including selenium and antimony); strontium; ruthenium (including rhodium, palladium, molybdenum, and technetium); lanthanum (including zirconium, neodymium, europium, niobium, promethium, praseodymium, samarium, and yttrium), cerium (including plutonium and neptunium); and barium.

The source term for each accident, taken from data provided for each plant (e.g., PRAs and TVA-provided data), is described by the release height, timing, duration, fraction of each isotope group released, and general emergency declaration (alarm) time (time when preparation for evacuation of persons living closest to the affected reactor is initiated, as discussed below). The PRAs included several release categories for each bypass and failure scenario. These release categories were screened for each accident scenario to determine which release category resulted in the highest risk; those accidents were included in this evaluation. The risk was determined by multiplying the consequences by the frequency for each release category. The highest risk release category source terms for each of the Browns Ferry and Sequoyah beyond-design-basis accidents are presented in **Table J–6.**

Table J–6 Beyond-Design-Basis Accident Source Term Release Fractions

Release Groups [a]	*Release Fractions*								
	Kr, Xe	*I, Br*	*Cs, Rb*	*Te, Sb, Se*	*Sr*	*Ru, Rh, Pd, Mo, Tc*	*La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y*	*Ce, Pu, Np*	*Ba*
Accident: SGTR Release Fractions									
BFN [b]	Not applicable – boiling water reactors do not have steam generators								
SQN [c]	0.91	0.21	0.19	0.0004	0.0023	0.07	0.00028	0.00055	0.0025
Accident: Early Containment Failure Release Fractions									
BFN [b]	1	0.2482	0.2631	0.1711	0.0036	0.01422	0.000324	0.00130	N/A
SQN [c]	0.9	0.042	0.043	0.044	0.0027	0.0065	0.00048	0.004	0.0046
Accident: Late Containment Failure Release Fractions									
BFN [b]	0.95	0.00087	0.0012	0.0053	0.00016	0.000019	0.000015	0.000062	N/A
SQN [c]	0.94	0.0071	0.011	0.0052	0.00036	0.00051	4.2×10^{-6}	4×10^{-6}	0.0013
Accident: Interfacing Systems LOCA Release Fractions									
BFN [b]	0.73	0.0005	0.00059	0.00038	1.3×10^{-6}	4.9×10^{-7}	1.4×10^{-7}	5.8×10^{-7}	N/A
SQN [c]	Not evaluated as a significant risk contributor in the cited reference for SQN								

Ba = barium; BFN = Browns Ferry Nuclear Plant; Br = bromine; Ce = cerium, Cs = cesium, Eu = europium; I = iodine; Kr = krypton; La = lanthanum; LOCA = loss-of-coolant accident; Mo = molybdenum; N/A = not applicable; Nb = niobium; Nd = neodymium; Np = neptunium; Pd = palladium; Pm = promethium; Pr = praseodymium; Pu = plutonium; Rb = rubidium; Rh = rhodium; Ru = ruthenium; Sb = antimony; Se = selenium; SGTR = steam generator tube rupture accident; Sm = samarium; SQN = Sequoyah Nuclear Plant; Sr = strontium; Tc = technetium; Te = terbium; Xe = xenon; Y = yttrium; Zr = zirconium.

[a] Groups of radionuclides with common release fractions.

[b] Browns Ferry release fractions are from TVA 2003, Table II-4, Attachment E-4, page E-410.

[c] Sequoyah release fractions are based on the *Watts Bar Nuclear Plant Severe Accident Analysis* (SAIC 2007).

Evacuation information. Each beyond-design-basis accident scenario has a warning time and a subsequent release time. The warning time is the time at which notification is given to offsite emergency response officials to initiate protective measures for the surrounding population. The release time is the time when the release to the environment begins. The minimum time between the warning time and the release time is one-half hour. The minimum time of one-half hour is enough time to evacuate onsite personnel that are not needed to provide emergency support (i.e., noninvolved workers). This also conservatively assumes that an onsite emergency has not been declared prior to initiating an offsite notification. Intact containment severe accident scenarios, which were not analyzed because of their insignificant offsite consequences, take place over an even longer time frame.

This beyond-design-basis accident analysis assumed that 95 percent of the population within the emergency planning zone (10 miles [16 kilometers] for Sequoyah; 20 miles [32 kilometers] for Browns Ferry) participated in an initial evacuation. It was also assumed that the 5 percent of the population that did not participate in the initial evacuation was relocated within 12 to 24 hours after plume passage, based on the measured concentrations of radioactivity in the surrounding area and the comparison of projected doses with EPA guidelines. Longer-term countermeasures (e.g., crop or land interdiction) were based on EPA Protective Action Guides built into the MACCS2 model (NRC 1998).

J.3.3 Beyond-Design-Basis Accident Analysis

Only beyond-design-basis accident scenarios that lead to containment bypass or failure were evaluated because they are the accidents with the greatest potential consequences. The public health and environmental consequences would be significantly less for accident scenarios that do not lead to containment bypass or failure. Early containment failure, late containment failure, a steam generator tube rupture, and an ISLOCA were chosen as the representative set of beyond-design-basis accidents. As with the design-basis accident, the purpose of the analysis was to compare the potential impacts of using a partial-MOX fuel to those of using a full-LEU fuel core. Differences between the projected impacts for the Browns Ferry and Sequoyah beyond-design-basis accidents are primarily due to accident assumptions and siting, not inherent differences in reactor types.

J.3.3.1 Browns Ferry Beyond-Design-Basis Accident Analysis

Table J–7 shows the potential doses and average individual risks associated with the evaluated beyond-design-basis accidents at Browns Ferry. As shown in this table, of the beyond-design-basis accidents evaluated, the one that presents the highest risk to the MEI and the surrounding population is an early containment failure with an estimated frequency of approximately 1 chance in 9 million of the accident occurring per year of operations. The risks under either accident scenario would be similar regardless of whether the plant was using a full-LEU or partial-MOX fuel core.

For the MEI, the risk of a cancer fatality from an early containment failure would be approximately 1 chance in 10 million per year of operations for either a full-LEU or partial-MOX fuel core. For the average individual residing within 50 miles (80 kilometers) of Browns Ferry, the risk of a cancer fatality from an early containment failure would be approximately 1 chance in 3.3 billion per year of operations for either a full-LEU or partial-MOX fuel core. By comparison, the risk to an individual of developing a fatal cancer from normal background radiation would be approximately 1 chance in 5,200 per year (based on an average annuals natural background radiation dose of 318 millirem [see Chapter 3, Section 3.3.1.2]). The risk of a single latent fatal cancer from exposure to natural background radiation is estimated using the same factor used in the *SPD Supplemental EIS* for the evaluation of human health risk, i.e., 0.0006 latent cancer fatalities per rem.

The results of all of the beyond-design-basis accidents analyzed in this *SPD Supplemental EIS* indicate that, regardless of whether a partial-MOX fuel core or a full-LEU fuel core were used in Browns Ferry, the risk to individuals in the surrounding population would be similar and within the overall analysis uncertainty. Potential risks presented by the two types of cores are projected to be comparable for the MEI or the general population from these beyond-design-basis accidents.

Table J–7 Browns Ferry Nuclear Plant Beyond-Design-Basis Accident Impacts

			Impacts on the MEI at the Exclusion Area Boundary			*Impacts on the Population within 50 Miles*		
Accident	***Frequency (per year)***	***LEU or MOX Core***	***Dose (rem)*** [a]	***Dose Risk (rem/year)*** [b]	***Annual Risk of Fatal Cancer*** [c]	***Dose (person-rem)*** [a]	***Average Individual Dose Risk (rem/year)*** [d]	***Risk of Fatal Cancer to Average individual*** [e]
Early containment failure	1.1×10^{-7}	LEU	11,000	1.2×10^{-3}	1×10^{-7}	5.6×10^{6}	5.7×10^{-7}	3×10^{-10}
		MOX	11,000	1.2×10^{-3}	1×10^{-7}	5.4×10^{6}	5.5×10^{-7}	3×10^{-10}
Late containment failure	3.0×10^{-7}	LEU	190	5.7×10^{-5}	7×10^{-8}	420,000	1.2×10^{-7}	7×10^{-11}
		MOX	200	6.0×10^{-5}	7×10^{-8}	400,000	1.1×10^{-7}	7×10^{-11}
ISLOCA	4.6×10^{-8}	LEU	41	1.9×10^{-6}	2×10^{-9}	220,000	9.3×10^{-9}	6×10^{-12}
		MOX	38	1.7×10^{-6}	2×10^{-9}	210,000	8.9×10^{-9}	5×10^{-12}

ISLOCA = interfacing systems loss-of-coolant accident; LEU = low-enriched uranium; MEI = maximally exposed individual; MOX = mixed oxide.

[a] The reactor accident doses were calculated over an 80-year period using the MACCS2 computer code. Eighty years represents a typical person's lifetime.

[b] Annual dose risk to a hypothetical MEI at the exclusion area boundary (4,806 feet) accounting for the probability of the accident occurring.

[c] Annual risk of a fatality or fatal latent cancer to a hypothetical MEI at the exclusion area boundary (4,806 feet) accounting for the probability of the accident occurring.

[d] Average individual dose risk per year for the entire offsite projected population in 2020 (approximately 1,100,000) out to a distance of 50 miles, given exposure to the indicated dose and accounting for the probability of the accident occurring.

[e] Annual risk of a cancer fatality to the average individual in the entire offsite projected population in 2020 out to a distance of 50 miles accounting for the probability of the accident occurring.

Note: To convert feet to meters, multiply by 0.3048; miles to kilometers by 1.6093.

Source: TVA 2003, Table III-3, Attachment E-4, page E-418 for accident frequencies.

J.3.3.2 Sequoyah Beyond-Design-Basis Accidents

Table J–8 shows the potential doses and risks associated with the evaluated beyond-design-basis accidents at Sequoyah. As shown in this table, of the beyond-design-basis accidents evaluated, the late containment failure accident represents the highest risk to the MEI, with an estimated frequency of approximately 1 chance in 330,000 of the accident occurring per year of operation. The steam generator tube rupture accident represents the highest risk to the population near Sequoyah, with an estimated frequency of approximately 1 chance in 710,000 of the accident occurring per year of operations. The risks under either accident scenario would be similar regardless of whether the reactor was using a full-LEU or a partial-MOX fuel core.

For the MEI, the risk of a cancer fatality from the late containment failure accident would be approximately 1 chance in 330,000 per year of operations for either a full-LEU or partial-MOX fuel core. For the average individual residing within 50 miles (80 kilometers) of Sequoyah, the risk of a latent fatal cancer from a steam generator tube rupture would be approximately 1 chance in 330 million per year of operations for either a full-LEU or partial-MOX fuel core. As discussed in Section J.3.3.1, the risk to the MEI of developing a fatal cancer from normal background radiation would be approximately 1 chance in 5,200 per year.

The results for all of the beyond-design-basis accidents analyzed in this *SPD Supplemental EIS* indicate that, regardless of whether a partial-MOX fuel core or a full-LEU fuel core were used in Sequoyah, the risk to individuals in the surrounding population would be both similar and within the overall analysis uncertainty. Potential risks presented by the two types of cores are projected to be comparable for the MEI or the general population from these beyond-design-basis accidents.

Table J–8 Sequoyah Nuclear Plant Beyond-Design-Basis Accident Impacts

Accident	Frequency (per year)	LEU or MOX Core	Impacts on the MEI at the Exclusion Area Boundary: Dose (rem) [a]	Dose Risk (rem/year) [b]	Annual Risk of Fatal Cancer [c]	Impacts on the Population within 50 miles: Dose (person-rem) [a]	Average Individual Dose Risk (rem/year) [d]	Annual Risk of Fatal Cancer to Average Individual [e]
Early containment failure	3.4×10^{-7}	LEU	27,000	0.0092	3×10^{-7}	2.3×10^{6}	6.5×10^{-7}	4×10^{-10}
		MOX	33,000	0.011	3×10^{-7}	2.4×10^{6}	6.7×10^{-7}	4×10^{-10}
Late containment failure	3.0×10^{-6}	LEU	790	0.0024	3×10^{-6}	1.5×10^{6}	3.7×10^{-6}	2×10^{-9}
		MOX	870	0.0026	3×10^{-6}	1.5×10^{6}	3.7×10^{-6}	2×10^{-9}
Steam generator tube rupture	1.4×10^{-6}	LEU	45,000	0.063	1×10^{-6}	4.0×10^{6}	4.6×10^{-6}	3×10^{-9}
		MOX	56,000	0.078	1×10^{-6}	4.2×10^{6}	4.9×10^{-6}	3×10^{-9}

LEU = low-enriched uranium; MEI = maximally exposed individual; MOX = mixed oxide.

[a] The reactor accident doses were calculated over an 80-year period using the MACCS2 computer code. Eighty years represents a typical person's lifetime.

[b] Annual dose risk to a hypothetical MEI at the exclusion area boundary (1,824 feet) accounting for the probability of the accident occurring.

[c] Annual risk of a fatality or fatal latent cancer to a hypothetical MEI at the exclusion area boundary (1,824 feet) accounting for the probability of the accident occurring.

[d] Average individual dose risk per year for the entire offsite projected population in 2020 (approximately 1,200,000) out to a distance of 50 miles, given exposure to the indicated dose and accounting for the probability of the accident occurring.

[e] Annual risk of a cancer fatality to the average individual to the entire offsite projected population in 2020 out to a distance of 50 miles accounting for the probability of the accident occurring.

Note: To convert feet to meters, multiply by 0.3048; miles to kilometers by 1.6093.

Source: SAIC 2007, for accident frequencies.

J.3.3.3 Consideration of Other Severe Accidents

A wide range of beyond-design-basis accidents is considered by NRC in evaluating the accident risks from the operation of commercial nuclear power reactors, including TVA reactors. Unlikely to very unlikely events are considered in the contingency planning for these plants, including dam failures, hurricanes, flooding, tornadoes, terrorism, and similar events that might cause loss of offsite power (affecting the ability of the plant to provide emergency cooling to the reactors and used fuel pools) and threaten multiple plants. While some of the details of the contingencies to prevent these types of accidents are not made public, NRC requires that the reactor licensees be able to accommodate these kinds of potential disruptions without the plants experiencing severe or beyond-design-basis accidents such as those that occurred in Japan in 2011. TVA anticipates regulatory changes as a result of events that occurred in 2011 in the United States (the East Coast earthquake near Mineral, Virginia) and the earthquake and tsunami in Japan. Regulatory changes are incorporated in the plant design and operations in accordance with implementation requirements included in the regulations (e.g., CFR, NRC order, or 10 CFR 50.54(f) letter).

On March 11, 2011, a magnitude 9.0 earthquake occurred near the northeast coast of Honshu, Japan. This earthquake caused tsunami waves as high as 29.6 meters (97.1 feet) along the coast of Japan. The 14-meter (46-foot) tsunami that occurred at the Fukushima Daiichi nuclear power plant site[2] resulted in extended periods of time when the plant was without emergency system power and emergency cooling water. This, in turn, resulted in significant core damage to three of the six nuclear power plants, including hydrogen explosions that breached the containment. All of the reactors at Fukushima Daiichi are now in a safe shutdown condition with continuing active cooling.

[2] The Fukushima Daiichi nuclear power plant includes six BWRs of the same design as those present in TVA's Browns Ferry Nuclear Plant.

Shortly after this accident began to unfold, NRC formed a Fukushima Near-Term Task Force to conduct a systematic and methodical review of NRC processes and regulations to determine whether the agency should make additional improvements to its regulatory system and to make recommendations to NRC for its policy direction. The Near-Term Task Force issued its report in July 2011 (NRC 2011), which was followed by extensive discussions between NRC, the industry, and the public. Based on the Near-Term Task Force report and subsequent discussions, NRC directed its staff to initiate appropriate regulatory changes through issuance of orders and rulemaking processes.

The Near-Term Task Force has developed three prioritized tiers of recommended actions: Tier 1, which should be started without unnecessary delay; Tier 2, which requires further assessment and depends on Tier 1 issues and resources; and Tier 3, which requires further NRC staff study and is associated with longer-term actions. Tier 1 recommendations include: seismic, flooding, and other external hazard re-evaluations and walk downs; extended station blackout coping capability; reliable hardened vents for some early designs of BWRs; enhanced survival instrumentation for the used fuel pool, nuclear reactor, and containment; strengthening of emergency procedures, as well as severe accident management guidelines, damage mitigation guidelines; and improvements in staffing and communication during an emergency. Tier 2 and 3 recommendations involve additional improvements and enhancements to mitigate the effects of extreme seismic and flooding events in terms of used fuel pool integrity, hydrogen control, long-term station blackout, venting, training, monitoring, decisionmaking, emergency preparedness, and public education.

In February 2012, NRC issued policy guidance to implement the aforementioned actions in the form of proposed orders requiring safety enhancements of operating reactors, construction permit holders, and combined license holders (NRC 2012b). On March 12, 2012, the NRC issued three orders as well as a request for information regarding additional concerns (NRC 2012c). The orders addressed mitigation strategies for beyond-design basis external events (NRC 2012d), reliable hardened containment vents [Mark I and II BWRs] (NRC 2012e), and reliable spent fuel pool instrumentation (NRC 2012f). The request for information directed each reactor licensee to provide specific information following a re-evaluation of seismic and flooding hazards, emergency communications systems and staffing levels. Information from licensees was also requested after the licensees conduct walkdowns of reactor facilities to ensure protections against potential design basis hazards.

The NRC has issued an advance notice of proposed rulemaking for station blackout regulatory actions. It also anticipates issuing an advanced notice of proposed rulemaking on the strengthening and integration of emergency operating procedures, severe accident management guidelines, and extensive damage mitigation guidelines (NRC 2012c).

TVA will institute applicable NRC regulatory updates at Browns Ferry and Sequoyah when they are promulgated in their final approved form. TVA took proactive steps in response to the events at Fukushima, forming a review team to assess early lessons learned and determine their potential applicability to the safety of TVA's reactors, including Browns Ferry and Sequoyah. Based on this assessment, TVA has taken steps to procure additional equipment to further ensure adequate cooling during the extremely unlikely event of an extended loss of offsite power, known as a station blackout, that could affect multiple reactors at TVA sites. In addition, TVA is working with various industry groups such as the Institute for Nuclear Power Operators and the Nuclear Energy Institute to conduct a more comprehensive assessment of the Fukushima events. TVA continues, through its engagement with the Nuclear Energy Institute and the Institute for Nuclear Power Operators, to work with NRC to ensure that the regulations governing the operation of U.S. nuclear plants appropriately protect public health and safety and the environment in light of the Fukushima events.

J.3.4 Overall Modeling Results

Table J–9 shows a comparison of projected radiological impacts from a series of design-basis and beyond-design-basis accidents reactors using partial-MOX fuel cores versus those using full-LEU fuel cores in the unlikely event one of these accidents were to occur. The dose to a member of the general public at the exclusion area boundary (i.e., the MEI) and the general population doses from these accidents, if they were to occur, are expected to be approximately the same for either core as shown in Tables J–4, J–5, J–7, and J–8. The Table J-9 numbers in parentheses are the calculated ratios (impacts for a partial MOX core divided by impacts for an LEU core). A value of less than 1 indicates that the MOX fuel core could result in smaller impacts than the same accident with an LEU fuel core. A value of 1 indicates that the estimated impacts are the same for both fuel core types. A ratio larger than 1 indicates that the MOX fuel core could result in larger impacts than the same accident with an LEU fuel core. Outside the parentheses, the table shows a ratio of 1 for all accident scenarios. This is a rounded value because when modeling and analytical uncertainties are considered, the precision of the results is no more than one significant figure.

Table J–9 Ratio of Accident Impacts for Partial Mixed Oxide Fuel and Full Low-Enriched Uranium Fuel Cores (Partial Mixed Oxide Fuel Doses/Full Low-Enriched Uranium Fuel Doses) [a,b]

Accident	*Browns Ferry Nuclear Plant*		*Sequoyah Nuclear Plant*	
	MEI at the Exclusion Area Boundary	*Population within 50 Miles*	*MEI at the Exclusion Area Boundary*	*Population within 50 Miles*
Design-basis accidents				
LOCA	1 (0.88)	1 (1.00)	1 (0.87)	1 (0.96)
Used-fuel-handling accident	1 (1.00)	1 (1.00)	1 (1.00)	1 (1.00)
Beyond-design-basis accidents				
Early containment failure	1 (1.00)	1 (0.96)	1 (1.22)	1 (1.04)
Late containment failure	1 (1.05)	1 (0.95)	1 (1.10)	1 (1.00)
SGTR [c]	Not applicable	Not applicable	1 (1.24)	1 (1.05)
ISLOCA [d]	1 (0.93)	1 (0.95)	See SGTR	See SGTR

ISLOCA = interfacing systems loss-of-coolant accident; LOCA = loss-of-coolant accident; MEI = maximally exposed individual; SGTR = steam generator tube rupture accident.

[a] Reactor accidents involving the use of partial-MOX fuel cores were assumed to involve reactor cores with approximately 40 percent MOX fuel and 60 percent LEU fuel.

[b] The values in parentheses reflect the ratios calculated by dividing the accident analysis results for a partial MOX fuel core by the results for a full LEU core. When modeling and analytical uncertainties are considered, the precision of the results is no more than one significant figure.

[c] Steam generator tube rupture is not applicable for boiling water reactors because they do not use steam generators.

[d] An ISLOCA was not analyzed in the *Watts Bar Nuclear Plant Severe Reactor Accident Analysis* (SAIC 2007) on which the analysis in this appendix is based because the impacts were bounded by the SGTR accident.

Note: To convert miles to kilometers, multiply by 1.6093.

Regardless of the core type, the estimated doses to the MEI from design-basis accidents would be small compared to the NRC limit, as discussed in Sections J.3.1.1 and J.3.1.2. The estimated doses to the MEI from beyond-design-basis accidents would present similar risks to the MEI, as discussed in Sections J.3.3.1 and J.3.3.2. Based on this evaluation of the potential impacts of accidents with either a full-LEU or a partial-MOX fuel core in either a PWR (Sequoyah) or a BWR (Browns Ferry), the projected radiological impacts of such accidents or the risks associated with the plants' operation are comparable. This conclusion is similar to the conclusion reached in the *SPD EIS* (DOE 1999). The risks to the MEI and the surrounding populations of developing a fatal cancer as a result of one of these accidents, regardless of whether the reactors are using partial-MOX fuel cores or full-LEU fuel cores are small.

J.4 Uncertainties

The purpose of the analysis in this appendix is to compare the potential impacts from accidents related to the use of MOX fuel in domestic, commercial nuclear power plants. The analyses are based on studies, data, and models that introduce levels of uncertainty into the analyses. The following paragraphs address recognized uncertainties in the analyses.

In the application of the MACCS2 v1.13.1 computer code, dose conversion factors from Federal Guidance Report 11 (EPA 1988) were used. A more recent version of dose conversion factors has been developed and is included in Federal Guidance Report 13 (EPA 1999). Using the updated dose conversion factors in Federal Guidance Report 13, the estimated doses from DOE facility accidents and reactor accidents would increase for some key isotopes and decrease for other key isotopes. Overall, these differences are expected to be well within the much larger uncertainties associated with what might actually happen during an accident; for example, the amount of radioactive material that might actually escape a facility, the amount of time the fuel in a reactor may have been irradiated before the accident occurred, or the weather conditions at the time of the accident.

The accident analysis estimated the individual risk of a latent cancer fatality as a result of exposure to radiation by applying a constant factor of 0.0006 LCFs per rem or person-rem to all doses less than 20 rem (the risk factor is doubled for doses equaling or exceeding 20 rem). This linear no-threshold extrapolation is the standard method for estimating health risks. In the unlikely event of an accident, many of the individuals in the affected population could receive such a small dose of radiation that they would not suffer any health effects from the radiation. As discussed in Appendix C (see text box in Section C.3), a number of radiation health scientists and organizations have expressed reservations that the currently used cancer risk conversion factors, which are based on epidemiological studies of high doses (doses exceeding 5 to 10 rem), may not apply at low doses. In addition, because the affected population would receive increased health monitoring in the event of the accidents considered in this *SPD Supplemental EIS*, early detection of cancers may result in a lower number of cancer fatalities in the affected population than in a similar, unmonitored population. Nevertheless, the human health risk analysis in this appendix uses the linear no-threshold dose risk assumption.

A recent beyond-design-basis accident analysis by Sandia National Laboratories (SNL 2011) indicates that release fractions from a 40 percent MOX fuel core are similar to those of a full-LEU fuel core. Differences between the partial-MOX fuel core and full-LEU fuel core release times and source terms for each accident phase and class of radionuclide are within the uncertainty of the calculation methodology. In some cases, full-LEU fuel core release fractions were slightly larger, while in other cases partial-MOX fuel core release fractions were larger. Therefore, the release fractions given in Table J–6 of this appendix are appropriate for accidents involving either a partial-MOX or full-LEU fuel core.

The Sandia National Laboratories beyond-design-basis accident analysis (SNL 2011) was developed as part of an NRC research program to evaluate the impact of using MOX fuel in commercial nuclear power plants. This study was undertaken to evaluate the impact of the usage of a 40 percent MOX fuel core on the consequences of postulated severe or beyond-design-basis accidents. A series of severe accident calculations were performed using MELCOR 1.8.5 for a four-loop Westinghouse reactor with an ice condenser containment (similar to that in Sequoyah). The calculations covered the risk- and consequence-dominant accident sequences in plant-specific PRAs, including early and late containment failures.

The results indicated that the accident progression and source terms for the full-LEU and partial-MOX fuel cores were similar. This was initially unexpected because the experimental data for fission product releases from MOX fuel suggested higher releases than LEU fuel. However, the calculations show that at severe accident fuel temperatures, the volatile fission product releases occur at a very high release rate, regardless of the fuel type. Hence, the differences noted in the experimental results at lower temperature were not prototypical of severe accident conditions in the long term and did not greatly impact the integral source term.

In January 2012, NRC issued draft NUREG-1935, *State-of-the-Art Reactor Consequence Analyses Report (SOARCA Report)* (NRC 2012a), for comment. The *SOARCA Report* presents the results of best-estimate severe (beyond-design-basis) accident analyses for two operating U.S. nuclear power plants, the Surry Power Station (Surry), a PWR in Surry, Virginia, and the Peach Bottom Atomic Power Station (Peach Bottom), a BWR in Delta, Pennsylvania, using current knowledge and computer codes. The *SOARCA Report* work was developed over more than 5 years and has been subject to extensive independent peer review by experts in severe accident phenomena, modeling, and assessment. Using updated and benchmarked plant risk models, the *SOARCA Report* analyzed the following severe accident scenarios: short-term and long-term station blackouts for both Surry and Peach Bottom; a thermally induced steam generator tube rupture for Surry; and an ISLOCA for Surry. Modeling of these severe accident scenarios included input from plant PRAs and senior reactor operators, as well as mitigation measures based on emergency operating procedures and severe accident management guidelines.

The *SOARCA Report* analyzed the timing and magnitude of radioisotope source terms for both mitigated and unmitigated scenarios and compared these results to earlier severe accident studies. The *SOARCA Report* severe accident source terms for risk-dominant radioisotopes of iodine and cesium were calculated to be from 3 to 225 times smaller than those calculated in the 1982 *Technical Guidance for Siting Criteria Development*, NUREG/CR-2239 (NRC 1982). The *SOARCA Report* also confirmed that severe accident and emergency operations procedures and strategies would successfully prevent core damage or large radiological releases to the environment if implemented correctly. The *SOARCA Report* conclusions show that the public risk from severe accidents at current generation nuclear power plants is very small and has benefited by improvements in severe accident management and emergency response procedures.

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