Appendix K
Facility Accidents

K.1 COMMERCIAL REACTOR ACCIDENT ANALYSIS

K.1.1 Introduction

Postulated design basis and beyond-design-basis accidents were analyzed using the Melcor Accident Consequence Code System (MACCS2) computer code (NRC 1990, SNL 1997) for each of the three proposed reactor sites, Catawba Nuclear Station, McGuire Nuclear Station, and North Anna Power Station. Only those accidents with the potential for substantial radiological releases to the environment were evaluated. Two design basis accidents, a loss-of-coolant accident (LOCA) and a fuel-handling accident; and four beyond-design-basis accidents, a steam generator tube rupture, an early containment failure, a late containment failure, and an interfacing systems loss-of-coolant accident (ISLOCA) meet this criteria. Each of these accidents was analyzed twice, once using the current low-enriched uranium (LEU) core, and again, assuming a partial (40 percent) mixed oxide (MOX) core. Doses (consequences) and risks to a noninvolved worker, the offsite maximally exposed individual (MEI), and the general public within 80 km (50 mi) of each plant from each accident scenario were calculated. These results were then compared, by plant, for each postulated accident.

The MEI dose is calculated at the exclusion area boundary of each plant. The exclusion area boundary is that area surrounding the reactor in 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 operations of the facility and appropriate and effective arrangements are made to 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 that no significant hazards to the public health and safety would result.

K.1.2 Reactor Accident Identification and Quantification

Catawba and McGuire are similar plants, both with two 3,411 MWt Westinghouse pressurized water reactors (PWRs) with ice condenser containments. Because of these similarities, the release paths and mitigating mechanisms for the two plants are almost identical. The conservative assumptions of the U.S. Nuclear Regulatory Commission (NRC) regulatory guidance produce identical radiological releases to the environment (source terms) for the two plants. However, site-specific population and meteorological inputs result in different consequences from the two plants. The North Anna site has two 2,893 MWt Westinghouse PWRs with subatmospheric containments.

Both the design basis and beyond-design-basis accidents were identified from plant documents. Design basis accidents were selected by reviewing the Updated Final Safety Analysis Report (UFSAR) for each plant (DPC 1996, 1997; VPC 1998). Beyond-design-basis accidents were identified from the submittals (DPC 1991, 1992; VPC 1992) in response to the NRC's Generic Letter 88-20 (NRC 1988), which required reactor licensees to perform Individual Plant Examinations (IPEs) for severe accident vulnerabilities. Source terms for each accident for LEU-only cores were identified from these documents, source terms for partial MOX cores were developed based on these LEU source terms, and analyses were performed assuming both the current LEU-only cores and partial MOX cores containing 40 percent MOX fuel and 60 percent LEU fuel. After the source term is developed, the consequences (in terms of latent cancer fatalities [LCFs] and prompt fatalities) can be determined. To determine the risk, however, the frequency (probability) of occurrence of the accident must be determined. Then the consequences are multiplied by the frequency to determine the risk.

For this analysis, the frequencies of occurrence for the accidents with a 40 percent MOX core are assumed to be the same as those with an LEU core. The National Academy of Sciences reported (NAS 1995) that "any approach to the use of MOX fuel in U.S. power reactors must and will receive a thorough, formal safety review before it is licensed. While we are not in a position to predict what if any modifications to existing reactor types will be required as a result of such licensing reviews, we expect that the final outcome will be certification that whatever LWR type is chosen will be able, with modifications if appropriate, to operate within prevailing reactivity and thermal margins using sufficient plutonium loadings to accomplish the disposition mission in a small number of reactors. We believe, further, that under these circumstances no important overall adverse impact of MOX use ion the accident probabilities of the LWRs involved will occur; if there are adequate reactivity and thermal margins in the fuel, as licensing review should ensure, the main remaining determinants of accident probabilities will involve factors not related to fuel composition and hence unaffected by the use of MOX rather than LEU fuel." Considering the National Academy of Sciences statements, the lack of empirical data, and the degree of uncertainty associated with accident frequencies, this analysis assumes that the accident frequencies are the same for a 40 percent MOX core as those for a 100 percent LEU core.

K.1.2.1 MOX Source Term Development

MOX source terms were developed by applying the calculated ratio for individual radioisotopes present in both the MOX and LEU cores to the source term for each of the LEU accidents. MOX source term development required several steps. The analysis assumes that the initial isotopic composition of the plutonium is that delivered to the MOX facility for fabrication into MOX fuel. The MOX facility includes a polishing step that removes impurities, including americium 241, a major contributor to the dose from plutonium 235. This analysis conservatively assumes that the polishing step reduces the americium 241 to 1 part per million (ppm), then ages the plutonium for 1 year after polishing prior to being loaded into a reactor. Table K–1 provides the assumed isotopic composition for the plutonium source material.

Table K–1. Isotopic Breakdown of Plutonium

Isotope	Prior to Polishing (wt %)	After Polishing and Aging (wt %)
Plutonium 236	<1 ppb	1 ppb
Plutonium 238	0.03	0.03
Plutonium 239	92.2	93.28
Plutonium 240	6.46	6.54
Plutonium 241	0.05	0.05
Plutonium 242	0.1	0.1
Americium 241	0.9	25 ppm

Key: ppb, parts per billion; ppm, parts per million; wt %, weight percent.

The *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) assumes that MOX fuel would be fabricated using depleted uranium (0.25 weight percent uranium 235) (White 1997). The MOX assemblies are assumed to be 4.37 percent plutonium/americium and the LEU assemblies are assumed to be 4.37 percent uranium 235. To simulate a normal plant refueling cycle, the MOX portion was assumed to be 50 percent once-burned and 50 percent twice-burned assemblies. The LEU portion of the MOX was assumed to be 33.3 percent once-burned, 33.3 percent twice-burned, and 33.3 percent thrice-burned assemblies. The LEU-only cores were assumed to be equally divided between once-, twice-, and thrice-burned assemblies. All analyses assumed end-of-cycle inventories to produce the highest consequences. Fuel cycles were based on an 18-month refueling schedule with a 40-day downtime between cycles. The source terms for the LEU-only accident analyses were those identified in plant documents. Source terms for the partial MOX cores were developed using the isotopic ratios in Table K–2 provided by Oak Ridge National Laboratory (ORNL 1999). The MOX core inventory for

Table K–2. MOX/LEU Core Inventory Isotopic Ratios

Isotope	Ratio	Isotope	Ratio	Isotope	Ratio
Americium 241	2.06	Krypton 85m	0.86	Strontium 91	0.86
Antimony 127	1.15	Krypton 87	0.85	Strontium 92	0.89
Antimony 129	1.07	Krypton 88	0.84	Technetium 99m	0.99
Barium 139	0.97	Lanthanum 140	0.97	Tellurium 127	1.16
Barium 140	0.98	Lanthanum 141	0.97	Tellurium 127m	1.20
Cerium 141	0.98	Lanthanum 142	0.97	Tellurium 129	1.08
Cerium 143	0.95	Molybdenum 99	0.99	Tellurium 129m	1.09
Cerium 144	0.91	Neodymium 147	0.98	Tellurium 131m	1.11
Cesium 134	0.85	Neptunium 239	0.99	Tellurium 132	1.01
Cesium 136	1.09	Niobium 95	0.94	Xenon 131m	1.02
Cesium 137	0.91	Plutonium 238	0.76	Xenon 133	1.00
Cobalt 58	0.86	Plutonium 239	2.06	Xenon 133m	1.01
Cobalt 60	0.72	Plutonium 240	2.20	Xenon 135	1.28
Curium 242	1.43	Plutonium 241	1.79	Xenon 135m	1.04
Curium 244	0.94	Praseodymium 143	0.95	Xenon 138	0.96
Iodine 131	1.03	Rhodium 105	1.19	Yttrium 90	0.76
Iodine 132	1.02	Rubidium 86	0.77	Yttrium 91	0.85
Iodine 133	1.00	Ruthenium 103	1.11	Yttrium 92	0.89
Iodine 134	0.98	Ruthenium 105	1.18	Yttrium 93	0.91
Iodine 135	1.00	Ruthenium 106	1.28	Zirconium 95	0.94
Krypton 83m	0.89	Strontium 89	0.83	Zirconium 97	0.98
Krypton 85	0.78	Strontium 90	0.75		

each isotope was divided by the LEU core inventory for that isotope to provide a MOX/LEU ratio for each isotope. These ratios were then applied to LEU releases for each accident to estimate the MOX releases.

The NRC licensing process will thoroughly review precise enrichments and fuel management schemes. The enrichments and fuel management schemes analyzed in the SPD EIS were chosen as realistic upper bounds. The accidents also assumed a maximum 40 percent MOX core. Taken together, these assumptions are sufficiently conservative to account for uncertainties associated with the MOX/LEU ratios.

K.1.2.2 Meteorological Data

Meteorological data for each specific reactor site were used. The meteorological data characteristic of the site region are described by 1 year of hourly data (8,760 measurements). This data includes wind speed, wind direction, atmospheric stability, and rainfall (DOE 1999).

K.1.2.3 Population Data

The population distribution around each plant was determined using 1990 Census data extrapolated to the year 2015. The population was then split into segments which correspond to the chosen polar coordinate grid. The polar coordinate grid for this analysis consists of 12 radial intervals aligned with the 16 compass directions. For Catawba and McGuire, the distances (in kilometers) of the 12 radial intervals are: 0.64, 0.762, 1.61, 3.22, 4.83, 6.44, 8.05, 16.09, 32.18, 48.27, 64.36, 80.45. For North Anna, these distances (in kilometers) are: 0.64, 1.350, 1.61, 3.22, 4.83, 6.44, 8.05, 16.09, 32.18, 48.27, 64.36, 80.45. The first of the 12 segments represents the location of the noninvolved worker and the second is the location of the site boundary. Projected population data for the year 2015 corresponding to the grid segments at Catawba, McGuire, and North Anna are presented in Tables K–3, K–4, and K–5, respectively.

Table K–3. Projected Catawba Population for Year 2015

	Distance in Kilometers From Release Point											
Direction	**0.64**	**0.762**	**1.61**	**3.22**	**4.83**	**6.44**	**8.05**	**16.09**	**32.18**	**48.27**	**64.36**	**80.45**
N	0	0	6	14	73	469	800	2,642	51,540	31,112	49,551	33,306
NNE	0	0	6	112	250	334	362	9,394	173,036	135,229	102,558	66,298
NE	0	0	7	119	239	394	595	6,442	212,814	143,650	22,571	20,108
ENE	0	0	11	81	504	1,409	1,042	5,842	72,488	52,784	32,588	10,919
E	0	0	21	5	863	1,059	570	7,959	12,144	27,800	22,844	10,995
ESE	0	0	23	47	295	388	679	7,449	8,607	18,196	12,293	9,290
SE	0	0	20	25	284	893	1,060	37,300	14,279	14,657	12,776	3,692
SSE	0	0	6	80	278	706	891	16,458	10,249	4,190	1,599	11,376
S	0	0	24	165	275	606	819	4,529	4,457	15,062	1,579	1,874
SSW	0	0	17	137	245	238	346	2,268	3,563	2,093	12,970	4,245
SW	0	0	20	114	162	208	267	5,538	9,559	2,040	11,272	12,302
WSW	0	0	21	84	159	205	257	2,493	4,756	8,947	31,712	80,518
W	0	0	23	113	202	272	345	4,979	6,978	17,182	26,070	35,091
WNW	0	0	23	103	199	283	363	3,011	17,814	32,751	29,031	8,706
NW	0	0	23	96	165	274	363	3,099	65,856	28,474	33,819	45,793
NNW	0	0	21	85	125	1,153	1,296	3,404	48,431	24,219	32,537	52,530

Table K–4. Projected McGuire Population for Year 2015

	Distance in Kilometers From Release Point											
Direction	**0.64**	**0.762**	**1.61**	**3.22**	**4.83**	**6.44**	**8.05**	**16.09**	**32.18**	**48.27**	**64.36**	**80.45**
N	0	0	44	0	269	110	203	3,153	14,870	28,254	12,987	15,726
NNE	0	0	28	0	124	569	1,728	9,493	21,903	12,317	24,826	43,937
NE	0	0	30	0	5	832	1,016	6,944	30,939	44,064	55,186	44,691
ENE	0	0	184	144	405	684	591	4,289	51,928	37,373	13,039	28,160
E	0	0	217	180	448	381	493	7,575	26,495	21,992	16,957	14,635
ESE	0	0	65	69	271	381	507	7,423	119,345	79,039	36,221	26,552
SE	0	0	15	59	130	244	273	8,387	219,183	204,614	46,100	24,527
SSE	0	0	15	59	99	138	100	9,530	90,900	95,688	79,859	15,954
S	0	0	14	83	165	182	165	6,429	35,178	21,241	41,638	9,071
SSW	0	0	18	101	169	240	221	3,261	61,514	29,814	10,774	9,327
SW	0	0	26	101	169	236	305	5,338	20,195	31,064	47,641	43,067
WSW	0	0	19	101	169	236	296	2,741	20,873	17,334	15,815	15,077
W	6	0	14	112	184	252	312	2,048	24,932	11,715	12,705	43,357
WNW	0	0	3	101	444	811	338	2,187	14,985	57,262	74,708	60,953
NW	0	0	0	224	200	1,005	793	4,260	8,528	22,380	26,093	12,511
NNW	0	0	0	0	4	0	36	1,989	8,570	40,993	13,101	10,686

Table K–5. Projected North Anna Population for Year 2015

	Distance in Kilometers From Release Point											
Direction	**0.64**	**1.35**	**1.61**	**3.22**	**4.83**	**6.44**	**8.05**	**16.09**	**32.18**	**48.27**	**64.36**	**80.45**
N	0	0	0	39	98	122	153	576	7,816	5,149	17,803	42,233
NNE	0	0	2	37	58	160	206	1,236	7,634	10,765	25,976	172,658
NE	0	0	2	30	43	94	100	1,122	38,833	90,820	34,429	77,097
ENE	0	0	0	15	103	40	64	1,373	5,822	6,693	11,426	17,324
E	0	0	0	17	112	42	34	1,183	6,128	5,175	1,839	4,296
ESE	0	0	2	7	17	97	135	950	5,595	5,454	5,161	7,909
SE	0	0	1	18	77	9	12	575	2,989	19,343	59,057	76,396
SSE	0	0	3	50	29	27	40	919	5,051	15,259	443,326	392,420
S	0	0	0	42	20	30	40	669	4,413	11,763	20,254	34,375
SSW	0	0	0	10	12	54	65	554	3,098	5,803	5,616	6,222
SW	0	0	0	4	14	54	86	1,186	2,678	2,845	5,482	4,576
WSW	0	0	0	19	42	31	63	1,381	4,402	6,729	8,905	8,094
W	0	0	0	31	24	24	29	466	2,883	4,529	109,205	21,748
WNW	0	0	0	30	79	52	29	606	2,725	8,371	17,931	9,934
NW	0	0	1	35	52	92	81	662	3,327	11,604	11,816	3,090
NNW	0	0	0	28	64	13	25	771	4,725	9,040	25,534	10,041

K.1.2.4 Design Basis Events

Design basis events are defined by the American Nuclear Society as Condition IV occurrences or limiting faults. Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of substantial radioactive material. These are the most serious events which must be designed against and represent limiting design cases.

The accident analyses presented in the UFSARs are conservative design basis analyses and therefore the dose consequences are bounding (i.e., a realistically based analysis would result in lower doses). 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 an environmental impact statement, such as the SPD EIS. After reviewing the UFSAR accident analyses, the design basis accidents chosen for evaluation in the SPD EIS are a large-break LOCA and a fuel-handling accident.

LOCA. A design basis large-break LOCA was chosen for evaluation because it is the limiting reactor design basis accident at each of the three plants. The analysis was performed in accordance with the methodology and assumptions in Regulatory Guide 1.4 (NRC 1974). 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 keeps cladding temperatures well below melting, 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 gross release of fission products is evaluated. The only postulated mechanism for such a release would require a number of simultaneous and extended failures to occur in the engineered safety feature systems, producing severe physical degradation of core geometry and partial melting of the fuel.

Development of the LOCA source term is based on the conservative assumptions specified in Regulatory Guide 1.4. Consistent with this Regulatory Guide, 100 percent of the noble gas inventory and 25 percent of the iodine inventory in the core are assumed to be immediately available for leakage from the primary containment. However, all of this radioactivity is not released directly to the environment because there are a number of mitigating mechanisms which can delay or retain radioisotopes. The principal mechanism, the primary containment, substantially restricts the release rate of the radioisotopes. Following a postulated LOCA, another potential source of fission product release to the environment is the leakage of radioactive water from engineered safety feature equipment located outside containment. The fission products could then be released from the water into the atmosphere, resulting in offsite radiological consequences that contribute to the total dose from the LOCA.

The LOCA radiological consequence analysis for the LEU cores was performed assuming a ground-level release based on offeror-supplied plant-specific radioisotope release data. All possible leak paths (containment, bypass, and the emergency core cooling system) were included. Were a LOCA to occur, a substantial percentage of the releases would be expected to be elevated, which would be expected to reduce the consequences from those calculated in this analysis. To analyze the accident for a partial MOX core, the LEU isotopic activity was multiplied by the MOX/LEU ratios (from Table K–2) to provide a MOX core activity for each isotope. The LEU and MOX LOCA releases for Catawba and McGuire are provided in Table K–6 and for North Anna in Table K–7.

Table K–6. Catawba and McGuire LOCA Source Term

Isotope	LEU LOCA Release (Ci)	MOX/LEU Ratio	40% MOX Core Release (Ci)
Iodine 131	2.42×10^4	1.03	2.49×10^4
Iodine 132	7.76×10^2	1.02	7.92×10^2
Iodine 133	3.22×10^3	1.00	3.22×10^3
Iodine 134	6.55×10^2	0.98	6.42×10^2
Iodine 135	2.51×10^3	1.00	2.51×10^3
Krypton 83m	3.62×10^3	0.89	3.22×10^3
Krypton 85	1.96×10^4	0.78	1.53×10^4
Krypton 85m	1.96×10^4	0.86	1.68×10^4
Krypton 87	1.04×10^4	0.85	8.82×10^3
Krypton 88	3.23×10^4	0.84	2.72×10^4
Xenon 131m	2.79×10^4	1.02	2.84×10^4
Xenon 133	2.33×10^6	1.00	2.33×10^6
Xenon 133m	3.45×10^4	1.01	3.49×10^4
Xenon 135	2.90×10^5	1.28	3.71×10^5
Xenon 135m	1.40×10^3	1.04	1.46×10^3
Xenon 138	7.21×10^3	0.96	6.92×10^3

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

Fuel-Handling Accident. The fuel-handling accident analysis was performed in a conservative manner, in accordance with Regulatory Guide 1.25 methodology (NRC 1972). In the fuel-handling accident scenario, a spent fuel assembly is dropped. The drop results in a breach of the fuel rod cladding, and a portion of the volatile fission gases from the damaged fuel rods is released. A fuel-handling accident would realistically result in only a fraction of the fuel rods being damaged. However, consistent with NRC methodology, all the fuel rods in the assembly are assumed to be damaged.

Table K–7. North Anna LOCA Source Term

Isotope	LEU LOCA Release (Ci)	MOX/LEU Ratio	40% MOX Core Release (Ci)
Iodine 131	3.68×10^2	1.03	3.79×10^2
Iodine 132	3.45×10^2	1.02	3.52×10^2
Iodine 133	5.87×10^2	1.00	5.87×10^2
Iodine 134	5.10×10^2	0.98	5.00×10^2
Iodine 135	5.01×10^2	1.00	5.01×10^2
Krypton 83m	4.26×10^2	0.89	3.79×10^2
Krypton 85	5.06×10^1	0.78	3.95×10^1
Krypton 85m	1.48×10^3	0.86	1.27×10^3
Krypton 87	2.22×10^3	0.85	1.89×10^3
Krypton 88	3.50×10^3	0.84	2.94×10^3
Xenon 131m	3.20×10^1	1.02	3.26×10^1
Xenon 133	6.91×10^3	1.00	6.91×10^3
Xenon 133m	1.70×10^2	1.01	1.72×10^2
Xenon 135	6.37×10^3	1.28	8.15×10^3
Xenon 135m	6.72×10^2	1.04	6.99×10^2
Xenon 138	1.90×10^3	0.96	1.82×10^3

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

The accident is assumed to occur at the earliest time fuel-handling operations may begin after shutdown as identified in each plant's Technical Specifications.[1] The assumed accident time is 72 hr after shutdown at Catawba and McGuire. North Anna Technical Specifications require a minimum of 150 hr between shutdown and the initiation of fuel movement, but assumed an accident time of 100 hr.

As assumed in Regulatory Guide 1.25, the damaged assembly is the highest powered assembly being removed from the reactor. The values for individual fission product inventories in the damaged assembly are calculated assuming full power operation at the end of core life immediately preceding shutdown. All of the gap activity in the damaged rods is assumed to be released to the spent fuel pool. Noble gases released to the spent fuel pool are immediately released at ground level to the environment, but the water in the spent fuel pool greatly reduces the iodine available for release to the environment. It is assumed that all of the iodine escaping from the spent fuel pool is released to the environment at ground level over a 2-hr time period through the fuel-handling building ventilation system. The Catawba and McGuire UFSARs assume iodine filter efficiencies of 95 percent for both the inorganic and organic species. The North Anna UFSAR assumes a filter efficiency of 90 percent for the inorganic iodine and 70 percent for the organic iodine. The LEU and MOX source terms for Catawba and McGuire are provided in Table K–8 and the source terms for North Anna are provided in Table K–9.

The frequencies for the design basis LOCAs, obtained from the IPEs, are Catawba, 7.50×10^{-6}; McGuire, 1.50×10^{-5}; and North Anna, 2.10×10^{-5}. The frequencies of the fuel-handling accidents were estimated in lieu of plant-specific data. For conservatism, a frequency of 1×10^{-4} was chosen for the analysis.

[1] Technical Specifications are plant-specific operating conditions that control safety-related parameters of plant operation. Technical Specifications are part of the operating license and require an operating license amendment to change.

Table K–8. Catawba and McGuire Fuel-Handling Accident Source Term

	LEU	MOX/LEU	40% MOX Core
Nuclide	**Release (Ci)**	**Ratio**	**Release**
Iodine 131	3.83×10^1	1.03	3.94×10^1
Iodine 132	5.55×10^1	1.02	5.66×10^1
Iodine 133	8.00×10^1	1.00	8.00×10^1
Iodine 134	8.80×10^1	0.98	8.62×10^1
Iodine 135	7.55×10^1	1.00	7.55×10^1
Krypton 83m	9.47×10^3	0.89	8.43×10^3
Krypton 85	1.11×10^3	0.78	8.66×10^2
Krypton 85m	2.16×10^4	0.86	1.86×10^4
Krypton 87	4.04×10^4	0.85	3.43×10^4
Krypton 88	5.58×10^4	0.84	4.69×10^4
Xenon 133	1.60×10^5	1.00	1.60×10^5
Xenon 133m	4.81×10^3	1.01	4.86×10^3
Xenon 135	1.65×10^5	1.28	2.11×10^5
Xenon 135m	2.96×10^4	1.04	3.08×10^4
Xenon 138	1.34×10^5	0.96	1.29×10^5

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

Table K–9. North Anna Fuel-Handling Accident Source Term

	LEU	MOX/LEU	40% MOX Core
Nuclide	**Release (Ci)**	**Ratio**	**Release**
Iodine 131	9.05×10^1	1.03	9.32×10^1
Iodine 132	1.37×10^2	1.02	1.40×10^2
Iodine 133	2.01×10^2	1.00	2.01×10^2
Iodine 134	2.36×10^2	0.98	2.31×10^2
Iodine 135	1.82×10^2	1.00	1.82×10^2
Krypton 85	2.60×10^3	0.78	2.03×10^3
Krypton 85m	2.65×10^4	0.86	2.28×10^4
Krypton 87	5.10×10^4	0.85	4.34×10^4
Krypton 88	7.25×10^4	0.84	6.09×10^4
Xenon 131m	4.56×10^2	1.02	4.65×10^2
Xenon 133	1.36×10^5	1.00	1.36×10^5
Xenon 133m	3.46×10^3	1.01	3.49×10^3
Xenon 135	3.70×10^4	1.28	4.74×10^4
Xenon 135m	3.74×10^4	1.04	3.89×10^4
Xenon 138	1.22×10^5	0.96	1.17×10^5

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

K.1.2.5 Beyond-Design-Basis Events

Beyond-design-basis accidents (severe reactor accidents) are less likely to occur than reactor design basis accidents. In the reactor design basis accidents, the mitigating systems are assumed to be available. In the severe reactor accidents, even though the initiating event could be a design basis event (e.g., large-break LOCA), additional failures of mitigating systems would 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, the NRC required all licensees of operating plants to perform IPEs for severe accident vulnerabilities (Generic Letter 88-20) (NRC 1988), and indicated that a Probabalistic Risk Assessment (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 the operating disturbances (known as internal initiating events) within each plant. The state-of-the-art PRA uses realistic criteria and assumptions in evaluating the accident progression and the systems required to mitigate each accident.

A plant-specific PRA for severe accident vulnerabilities starts with identification of initiating events (i.e., challenges to normal plant operation or accidents) that require successful mitigation to prevent core damage. These events are grouped into initiating event classes that have similar characteristics and require the same overall plant response.

Event trees are developed for each initiating event class. These event trees depict the possible sequence of events that could occur during the plant's response to each initiating event class. The trees delineate the possible combinations (sequences) of functional and/or system successes and failures that lead to either successful mitigation of the initiating event or core damage. Functional and/or system success criteria are developed based on the plant response to the class of accident sequences. Failure modes of systems that are functionally important to preventing core damage are modeled. This modeling process is usually done with fault trees that define the combinations of equipment failures, equipment outages, and human errors that could cause the failure of systems to perform the desired functions.

Quantification of the event trees leads to hundreds, or even thousands, of different end states representing various accident sequences that are either mitigated or lead to core damage. Each accident sequence and its associated end state has a unique "signature" because of the particular combination of system successes and failures. These end states are grouped together into plant damage states, each of which collects sequences for which the progression of core damage, the release of fission products from the fuel, the status of containment and its systems, and the potential for mitigating source terms are similar. The sum of all core damage accident sequences will then represent an estimate of plant core damage frequency. The analysis of core damage frequency calculations is called a Level 1 PRA, or front-end analysis.

Next, an analysis of accident progression, containment loading[2] resulting from the accident, and the structural response to the accident loading is performed. The primary objective of this analysis, which is called a Level 2 PRA, is to characterize the potential for, and magnitude of, a release of radioactive material from the reactor fuel to the environment, given the occurrence of an accident that damages the core. The analysis includes an assessment of containment performance in response to a series of severe accidents. Analysis of the progression of an accident (an accident sequence within a plant damage state) generates a time history of loads imposed on the containment pressure boundary. These loads would then be compared against the containment's structural performance limits. If the loads exceed the performance limits, the containment would be expected to fail; conversely, if the containment performance limits exceed the calculated loads, the containment would be expected to survive. Four modes of containment failure are defined: containment isolation failure, containment bypass, early containment failure, and late containment failure.

The magnitude of the radioactive release to the atmosphere in an accident is dependent on the timing of the reactor vessel failure and the containment failure. To determine the magnitude of the release, a containment

[2] Challenges to containment integrity such as elevated temperature or pressure are referred to as containment loading.

event tree representing the time sequence of major phenomenological events that could occur during the formation and relocation of core debris (after core melt), availability of the containment heat removal system, and the expected mode of containment failures (i.e., bypass, early, and late), is developed. A reduced set of plant damage states is defined by culling the lower frequency plant damage states into higher frequency ones that have relatively similar severity and consequence potential. This condensed set is known as the key plant damage states. These key plant damage states would then become the initiating events for the containment event tree. The outcome of each sequence in this event tree represents a specific release category. Release categories that can be represented by similar source terms are grouped. Source terms associated with various release categories describe the fractional releases for representative radionuclide groups, as well as the timing, duration, and energy of release.

Beyond-design-basis accidents evaluated in the SPD EIS included 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. The accidents evaluated consisted of a steam generator tube rupture, an early containment failure, a late containment failure, and an ISLOCA.

Steam Generator Tube Rupture. A beyond-design-basis steam generator tube rupture induced by high temperatures 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 fail the steam generator tubes. As a result of the tube rupture, the secondary side may be exposed to full Reactor Coolant System pressures. These pressures are likely to cause relief valves to lift on the secondary side as they are designed to do. If these valves fail to close after venting, an open pathway from the reactor vessel to the environment can result.

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 late.

Late Containment Failure. A late containment failure involves structural failure of the containment several hours 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.

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 occurs, the lower pressure system will be overpressurized and could rupture outside the containment. This failure would establish a flow path directly to the environment or, sometimes, to another building of small-pressure capacity.

For each of the proposed reactors, an assessment was made of the pre-accident inventories of each radioactive species in the reactor fuel, using information on the thermal power and refueling cycles. For the source term and offsite consequence analysis, the radioactive species were collected into groups that exhibit similar chemical behavior. The following groups represent the radionuclides considered to be most important to offsite consequences: noble gases, iodine, cesium, tellurium, strontium, ruthenium, lanthanum, cerium, and barium.

The LEU end-of-cycle isotopic activities (inventories) were multiplied by the MOX/LEU ratio to provide a MOX end-of-cycle activity for each isotope. The LEU and MOX core activities for Catawba and McGuire are provided in Table K–10. The activities for North Anna are provided in Table K–11.

Table K–10. Catawba and McGuire End-of-Cycle Core Activities

Isotope	LEU Core Activity (Ci)	MOX/ LEU Ratio	40% MOX Core Activity (Ci)	Isotope	LEU Core Activity (Ci)	MOX/ LEU Ratio	40% MOX Core Activity (Ci)
Americium 241	3.13×10^3	2.06	6.45×10^3	Niobium 95	1.41×10^8	0.94	1.33×10^8
Antimony 127	7.53×10^6	1.15	8.66×10^6	Plutonium 238	9.90×10^4	0.76	7.53×10^4
Antimony 129	2.67×10^7	1.07	2.85×10^7	Plutonium 239	2.23×10^4	2.06	4.60×10^4
Barium 139	1.70×10^8	0.97	1.65×10^8	Plutonium 240	2.82×10^4	2.20	6.20×10^4
Barium 140	1.68×10^8	0.98	1.65×10^8	Plutonium 241	4.74×10^6	1.79	8.49×10^6
Cerium 141	1.53×10^8	0.98	1.50×10^8	Praseodymium 143	1.46×10^8	0.95	1.39×10^8
Cerium 143	1.48×10^8	0.95	1.41×10^8	Rhodium 105	5.53×10^7	1.19	6.58×10^7
Cerium 144	9.20×10^7	0.91	8.37×10^7	Rubidium 86	5.10×10^4	0.77	3.93×10^4
Cesium 134	1.17×10^7	0.85	9.93×10^6	Ruthenium 103	1.23×10^8	1.11	1.36×10^8
Cesium 136	3.56×10^6	1.09	3.88×10^6	Ruthenium 105	7.98×10^7	1.18	9.42×10^7
Cesium 137	6.53×10^6	0.91	5.94×10^6	Ruthenium 106	2.79×10^7	1.28	3.57×10^7
Cobolt 58	8.71×10^5	0.86	7.49×10^5	Strontium 89	9.70×10^7	0.83	8.05×10^7
Cobolt 60	6.66×10^5	0.72	4.80×10^5	Strontium 90	5.24×10^6	0.75	3.93×10^6
Curium 242	1.20×10^6	1.43	1.71×10^6	Strontium 91	1.25×10^8	0.86	1.07×10^8
Curium 244	7.02×10^4	0.94	6.60×10^4	Strontium 92	1.30×10^8	0.89	1.16×10^8
Iodine 131	8.66×10^7	1.03	8.92×10^7	Technetium 99m	1.42×10^8	0.99	1.41×10^8
Iodine 132	1.28×10^8	1.02	1.30×10^8	Tellurium 127	7.28×10^6	1.16	8.44×10^6
Iodine 133	1.83×10^8	1.00	1.83×10^8	Tellurium 127m	9.63×10^5	1.20	1.16×10^6
Iodine 134	2.01×10^8	0.98	1.97×10^8	Tellurium 129	2.50×10^7	1.08	2.70×10^7
Iodine 135	1.73×10^8	1.00	1.73×10^8	Tellurium 129m	6.60×10^6	1.09	7.20×10^6
Krypton 85	6.69×10^5	0.78	5.22×10^5	Tellurium 131m	1.26×10^7	1.11	1.40×10^7
Krypton 85m	3.13×10^7	0.86	2.69×10^7	Tellurium 132	1.26×10^8	1.01	1.27×10^8
Krypton 87	5.72×10^7	0.85	4.87×10^7	Xenon 133	1.83×10^8	1.00	1.83×10^8
Krypton 88	7.74×10^7	0.84	6.50×10^7	Xenon 135	3.44×10^7	1.28	4.40×10^7
Lanthanum 140	1.72×10^8	0.97	1.67×10^8	Yttrium 90	5.62×10^6	0.76	4.27×10^6
Lanthanum 141	1.57×10^8	0.97	1.53×10^8	Yttrium 91	1.18×10^8	0.85	1.00×10^8
Lanthanum 142	1.52×10^8	0.97	1.47×10^8	Yttrium 92	1.30×10^8	0.89	1.16×10^8
Molybdenum 99	1.65×10^8	0.99	1.63×10^8	Yttrium 93	1.47×10^8	0.91	1.34×10^8
Neodymium 147	6.52×10^7	0.98	6.39×10^7	Zirconium 95	1.49×10^8	0.94	1.40×10^8
Neptunium 239	1.75×10^9	0.99	1.73×10^9	Zirconium 97	1.56×10^8	0.98	1.53×10^8

Key: LEU, low-enriched uranium.

Table K–11. North Anna End-of-Cycle Core Activities

Isotope	LEU Core Activity (Ci)	MOX/ LEU Ratio	40% MOX Core Activity (Ci)	Isotope	LEU Core Activity (Ci)	MOX/ LEU Ratio	40% MOX Core Activity (Ci)
Americium 241	1.03×10^4	2.06	2.13×10^4	Plutonium 238	1.99×10^5	0.76	1.51×10^5
Antimony 127	6.36×10^6	1.15	7.31×10^6	Plutonium 239	2.70×10^4	2.06	5.57×10^4
Antimony 129	2.41×10^7	1.07	2.58×10^7	Plutonium 240	3.43×10^4	2.20	7.54×10^4
Barium 139	1.39×10^8	0.97	1.35×10^8	Plutonium 241	9.82×10^6	1.79	1.76×10^7
Barium 140	1.37×10^8	0.98	1.34×10^8	Praseodymium 143	1.17×10^8	0.95	1.11×10^8
Cerium 141	1.25×10^8	0.98	1.22×10^8	Rhodium 105	7.22×10^7	1.19	8.59×10^7
Cerium 143	1.18×10^8	0.95	1.12×10^8	Rubidium 86	1.45×10^4	0.77	1.12×10^4
Cerium 144	9.70×10^7	0.91	8.82×10^7	Rubidium 103	1.16×10^8	1.11	1.28×10^8
Cesium 134	1.28×10^7	0.85	1.09×10^7	Rubidium 105	7.84×10^7	1.18	9.25×10^7
Cesium 136	3.42×10^6	1.09	3.72×10^6	Rubidium 106	3.83×10^7	1.28	4.90×10^7
Cesium 137	8.41×10^6	0.91	7.66×10^6	Strontium 89	7.48×10^7	0.83	6.21×10^7
Curium 242	2.72×10^6	1.43	3.88×10^6	Strontium 90	6.22×10^6	0.75	4.66×10^6
Curium 244	2.75×10^5	0.94	2.58×10^5	Strontium 91	9.36×10^7	0.86	8.05×10^7
Iodine 131	7.33×10^7	1.03	7.55×10^7	Strontium 92	1.04×10^8	0.89	9.23×10^7
Iodine 132	1.07×10^8	1.02	1.09×10^8	Technetium 99m	1.26×10^8	0.99	1.25×10^8
Iodine 133	1.52×10^8	1.00	1.52×10^8	Tellurium 127	6.21×10^6	1.16	7.21×10^6
Iodine 134	1.75×10^8	0.98	1.71×10^8	Tellurium 127m	9.87×10^5	1.20	1.18×10^6
Iodine 135	1.49×10^8	1.00	1.49×10^8	Tellurium 129	2.29×10^7	1.08	2.47×10^7
Krypton 85	3.51×10^6	0.78	2.74×10^6	Tellurium 129m	4.20×10^6	1.09	4.58×10^6
Krypton 85m	8.69×10^5	0.86	7.48×10^5	Tellurium 132	1.07×10^8	1.01	1.08×10^8
Krypton 87	3.86×10^7	0.85	3.28×10^7	Xenon 133	1.59×10^8	1.00	1.59×10^8
Krypton 88	5.46×10^7	0.84	4.59×10^7	Xenon 133m	4.69×10^6	1.01	4.73×10^6
Lanthanum 140	1.42×10^8	0.97	1.37×10^8	Xenon 135	4.47×10^7	1.28	5.72×10^7
Lanthanum 141	1.28×10^8	0.97	1.24×10^8	Yttrium 90	6.21×10^6	0.76	4.72×10^6
Lanthanum 142	1.24×10^8	0.97	1.21×10^8	Yttrium 91	9.93×10^7	0.85	8.44×10^7
Molybdenum 99	1.43×10^8	0.99	1.42×10^8	Yttrium 92	1.01×10^8	0.89	8.97×10^7
Neodymium 147	5.12×10^7	0.98	5.02×10^7	Yttrium 93	1.16×10^8	0.91	1.05×10^8
Neptunium 239	1.51×10^9	0.99	1.50x109	Zirconium 95	1.27×10^8	0.94	1.20×10^8
Niobium 95	1.31×10^8	0.94	1.23×10^8	Zirconium 97	1.28×10^8	0.98	1.26×10^8

Key: LEU, low-enriched uranium.

The source term for each accident, taken from each plant's PRA, is described by the release height, timing, duration, and heat content of the plume, the fraction of each isotope group released, and the warning time (time when offsite officials are warned that an emergency response should be initiated). 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. The risk was determined by multiplying the consequences by the frequency for each release category. The release category with the highest risk for each scenario was used in the SPD EIS analysis. The highest risk release category source terms for Catawba, McGuire, and North Anna are presented in Table K–12. Also included in each release category characterization is the frequency of occurrence.

The overall risk from beyond-design-basis accidents can be described by the sum of risks from all beyond-design-basis accidents. The group of accidents derived from the screening process results in the highest risks from the containment bypass and failure scenarios. The screened-out accidents in these categories not only result in lower consequences, but also have much lower probabilities, often resulting in risks several orders of magnitude lower. The other type of severe accident scenario for these reactors results in an intact containment. The risks from these events are several orders of magnitude lower than the risks from the bypass and failure scenarios. Therefore, a summation of the severe accident risks presented in the SPD EIS is a good indicator of overall risk.

Table K–12. Beyond-Design-Basis Accident Source Terms

Accident	Parameters	Release Category	Frequency	Release Fractions: Xe/Kr	I	Cs/Rb	Te/Sb	Sr	Ru/Mo	La	Ce	Ba
CATAWBA												
SG tube rupture[a]	Time: 20 hr Duration: 1.0 hr Energy: 1.0×10^{4} cal/sec (4.2×10^{4} W) Elevation: 10.0 m Warning time: 7.5 hr	1.04	6.31×10^{-10}	1.0	7.7×10^{-1}	7.9×10^{-1}	7.3×10^{-1}	5.0×10^{-3}	9.4×10^{-2}	1.3×10^{-4}	NA	4.0×10^{-2}
Early containment failure	Time: 6.0 hr Duration: 0.5 hr Energy: 2.0×10^{7} cal/sec (8.37×10^{7} W) Elevation: 10.0 m Warning time: 5.5 hr	5.01	3.42×10^{-8}	1.0	5.5×10^{-2}	4.8×10^{-2}	3.0×10^{-2}	2.5×10^{-4}	2.2×10^{-3}	1.2×10^{-4}	NA	1.7×10^{-3}
Late containment failure	Time: 18.5 hr Duration: 0.5 hr Energy: 1.0×10^{7} cal/sec (4.2×10^{7} W) Elevation: 10.0 m Warning time: 18.0 hr	6.01	1.21×10^{-5}	1.0	3.6×10^{-3}	3.9×10^{-3}	1.8×10^{-3}	5.2×10^{-5}	3.8×10^{-4}	2.6×10^{-5}	NA	1.6×10^{-4}
Interfacing systems LOCA	Time: 6.0 hr Duration: 1.0 hr Energy: 1.0×10^{4} cal/sec (4.2×10^{4} W) Elevation: 10.0 m Warning time: 5.5 hr	2.04	6.9×10^{-8}	1.0	8.2×10^{-1}	8.2×10^{-1}	7.9×10^{-1}	5.8×10^{-2}	2.1×10^{-1}	3.1×10^{-2}	NA	1.4×10^{-1}

Table K–12. Beyond-Design-Basis Accident Source Terms (Continued)

Accident	Parameters	Release Category	Frequency	Release Fractions								
				Xe/Kr	I	Cs/Rb	Te/Sb	Sr	Ru/Mo	La	Ce	Ba
McGUIRE												
SG tube rupture	Time: 20.0 hr Duration: 1.0 hr Energy: 1.0×10^{4} cal/sec (4.2×10^{4} W) Elevation: 10.0 m Warning time: 7.5 hr	1.04	5.81×10^{-9}	1.0	7.7×10^{-1}	7.9×10^{-1}	7.3×10^{-1}	5.0×10^{-3}	9.4×10^{-2}	1.3×10^{-4}	NA	4.0×10^{-2}
Early containment failure	Time: 6.0 hr Duration: 0.5 hr Energy: 2.0×10^{7} cal/sec (8.37×10^{7} W) Elevation: 10.0 m Warning time: 5.5 hr	5.01	9.89×10^{-8}	1.0	4.4×10^{-2}	3.5×10^{-2}	2.1×10^{-2}	1.4×10^{-4}	4.3×10^{-3}	2.0×10^{-5}	NA	1.4×10^{-3}
Late containment failure	Time: 32.0 hr Duration: 0.5 hr Energy: 1.0×10^{7} cal/sec (4.2×10^{7} W) Elevation: 10.0 m Warning time: 31.5 hr	6.01	7.21×10^{-6}	1.0	3.2×10^{-3}	2.4×10^{-3}	3.3×10^{-3}	1.0×10^{-8}	5.8×10^{-8}	1.0×10^{-9}	NA	1.8×10^{-7}
Interfacing systems LOCA	Time: 3.0 hr Duration: 1.0 hr Energy: 1.0×10^{4} cal/sec (4.2×10^{4} W) Elevation: 10.0 m Warning time: 2.0 hr	2.04	6.35×10^{-7}	1.0	7.5×10^{-1}	7.5×10^{-1}	6.6×10^{-1}	4.2×10^{-2}	1.5×10^{-1}	2.0×10^{-2}	NA	9.8×10^{-2}

Table K–12. Beyond-Design-Basis Accident Source Terms (Continued)

Accident	Parameters	Release Category	Frequency	Release Fractions								
				Xe/Kr	I	Cs/Rb	Te/Sb	Sr	Ru/Mo	La	Ce	Ba
				NORTH ANNA								
SG tube rupture	Time: 20.3 hr Duration: 1.0 hr Energy: 8.48×10^{3} cal/sec (3.55×10^{4} W) Elevation: 10.0 m Warning time: 7.8 hr	24	7.38×10^{-6}	9.96×10^{-1}	5.2×10^{-1}	5.4×10^{-1}	2.6×10^{-3}/ 6.8×10^{-1}	3.4×10^{-2}	1.4×10^{-1}	5.5×10^{-5}	5.2×10^{-3}	2.1×10^{-2}
Early containment failure	Time: 3.056 hr Duration: 0.5 hr Energy: 1.696×10^{7} cal/sec (7.1×10^{7} W) Elevation: 10.0 m Warning time: 2.556 hr	7	1.60×10^{-7}	9.0×10^{-1}	7.4×10^{-2}	9.7×10^{-2}	1.4×10^{-2}/ 1.3×10^{-1}	1.5×10^{-2}	2.5×10^{-2}	8.1×10^{-6}	9.7×10^{-5}	8.7×10^{-3}
Late containment failure	Time: 8.33 hr Duration: 0.5 hr Energy: 8.48×10^{6} cal/sec (3.55×10^{7} W) Elevation: 10.0 m Warning time: 7.83 hr	9	2.46×10^{-6}	8.2×10^{-1}	2.3×10^{-6}	1.4×10^{-5}	1.6×10^{-5}/ 1.2×10^{-4}	3.2×10^{-4}	3.9×10^{-4}	1.8×10^{-11}	1.4×10^{-11}	1.3×10^{-5}
Interfacing systems LOCA[b]	Time: 5.56 hr Duration: 1.0 hr Energy: 8.48×10^{3} cal/sec (3.55×10^{4} W) Elevation: 10.0 m Warning time: 4.56 hr	23	2.40×10^{-7}	9.4×10^{-1}	2.9×10^{-1}	3.1×10^{-1}	1.6×10^{-5}/ 5.0×10^{-1}	2.3×10^{-1}	2.8×10^{-1}	3.6×10^{-4}	3.7×10^{-2}	1.5×10^{-1}

[a] McGuire data was used for the Catawba steam generator tube rupture event to compare similar scenarios.
[b] McGuire release duration, elevation, and warning time span were used for North Anna in lieu of plant-specific information.

Key: LOCA, loss-of-coolant accident; NA, not applicable; SG, steam generator.

K.1.2.5.1 Evacuation Information

This analysis conservatively assumes that 95 percent of the population within the 16-km (10-mi) emergency planning zone participated in an evacuation. It was also assumed that the five percent of the population that did not participate in the initial evacuation was relocated within 12 to 24 hr after plume passage, based on the measured concentrations of radioactivity in the surrounding area and the comparison of projected doses with Environmental Protection Agency (EPA) guidelines. Longer term countermeasures (e.g., crop or land interdiction) were based on EPA Protective Action Guides.

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 (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 on an even longer time frame.

K.1.2.6 Accident Impacts

Accident impacts are presented in terms of increased risk. Increased risk is defined as the additional risk resulting from using a partial MOX core rather than an LEU core. For example, if the risk of an LCF from an accident with an LEU core is 1.0×10^{-6} and the risk of an LCF from the same accident with a MOX core is 1.1×10^{-6}, then the increased risk of an LCF is 1.0×10^{-7} ($1.1\times10^{-6} - 1.0\times10^{-6} = 1.0\times10^{-7}$).

Tables K–13 through K–18 present the consequences and risks of the postulated set of accidents at Catawba, McGuire, and North Anna, respectively. The receptors include a noninvolved worker located 640 m (0.4 mi) from the release point, the MEI, and the population within an 80-km (50-mi) radius of the reactor site. The consequences and risks are presented for both the current LEU-only and the proposed 40 percent MOX core configurations.

Table K–19 shows the ratios of accident impacts with the proposed 40 percent MOX core to the impacts with the current LEU core. This table shows that the increased risk from accidents to the surrounding population from a MOX core is, on average, less than 5 percent. For the fuel-handling accident at all three plants, the risk is reduced when using MOX fuel.

Severe accident scenarios that postulate large abrupt releases could result in prompt fatalities if the radiation dose is sufficiently high. Of the accidents analyzed in the SPD EIS, the ISLOCA and steam generator tube rupture at Catawba and McGuire, and the ISLOCA at North Anna were the only accidents that resulted in doses high enough to cause prompt fatalities. However, the number of prompt fatalities is expected to increase only for the ISLOCA scenarios. Table K–20 shows the estimated number of prompt fatalities estimated to result from these accidents.

Table K–13. Design Basis Accident Impacts for Catawba With LEU and MOX Fuels

			Impacts on Noninvolved Worker			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	**Frequency (per year)**	**LEU or MOX Core**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (person-rem)**	**Latent Cancer Fatalities[c]**	**Risk of Latent Cancer Fatalities (over campaign)[d]**
Loss-of-coolant accident	7.50×10^{-6}	LEU	3.78	1.51×10^{-3}	1.81×10^{-7}	1.44	7.20×10^{-4}	8.64×10^{-8}	3.64×10^{3}	1.82	2.19×10^{-4}
		MOX	3.85	1.54×10^{-3}	1.86×10^{-7}	1.48	7.40×10^{-4}	8.88×10^{-8}	3.75×10^{3}	1.88	2.26×10^{-4}
Spent-fuel-handling accident[e]	1.00×10^{-4}	LEU	0.275	1.10×10^{-4}	1.78×10^{-7}	0.138	6.90×10^{-5}	1.10×10^{-7}	1.12×10^{2}	5.61×10^{-2}	8.98×10^{-5}
		MOX	0.262	1.05×10^{-4}	1.68×10^{-7}	0.131	6.55×10^{-5}	1.05×10^{-7}	1.10×10^{2}	5.48×10^{-2}	8.77×10^{-5}

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary—given exposure (762 m [2,500 ft]) to the indicated dose.

[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).

[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.

[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).

[e] Postulated design basis accidents at commercial reactors are considered extremely unlikely events. They are estimated to have a frequency of between 1.0×10^{-4} and 1.0×10^{-6} per year. Because a spent-fuel-handling accident does not have a calculated frequency associated with it, it has been estimated to have the highest frequency for the purposes of this analysis.

Key: LEU, low-enriched uranium.

Table K–14. Beyond-Design-Basis Accident Impacts for Catawba With LEU and MOX Fuels

			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	**Frequency (per year)**	**LEU or MOX Core**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (person-rem)**	**Latent Cancer Fatalities[c]**	**Risk of Latent Cancer Fatalities (over campaign)[d]**
Steam generator tube rupture[e]	6.31×10^{-10}	LEU	3.46×10^{2}	0.346	3.49×10^{-9}	5.71×10^{6}	2.86×10^{3}	2.88×10^{-5}
		MOX	3.67×10^{2}	0.367	3.71×10^{-9}	5.93×10^{6}	2.96×10^{3}	2.99×10^{-5}
Early containment failure	3.42×10^{-8}	LEU	5.97	2.99×10^{-3}	1.63×10^{-9}	7.70×10^{5}	3.85×10^{2}	2.11×10^{-4}
		MOX	6.01	3.01×10^{-3}	1.65×10^{-9}	8.07×10^{5}	4.04×10^{2}	2.21×10^{-4}
Late containment failure	1.21×10^{-5}	LEU	3.25	1.63×10^{-3}	3.15×10^{-7}	3.93×10^{5}	1.96×10^{2}	3.79×10^{-2}
		MOX	3.48	1.74×10^{-3}	3.38×10^{-7}	3.78×10^{5}	1.89×10^{2}	3.66×10^{-2}
Interfacing systems LOCA	6.90×10^{-8}	LEU	1.40×10^{4}	1	1.10×10^{-6}	2.64×10^{7}	1.32×10^{4}	1.46×10^{-2}
		MOX	1.60×10^{4}	1	1.10×10^{-6}	2.96×10^{7}	1.48×10^{4}	1.63×10^{-2}

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft])—given exposure to the indicated dose.
[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).
[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.
[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).
[e] McGuire timing and release fractions were used to compare like scenarios.

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

Table K–15. Design Basis Accident Impacts for McGuire With LEU and MOX Fuels

			Impacts on Noninvolved Worker			Impacts at Site Boundaries			Impacts on Population Within 80 km		
Accident	Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Risk of Latent Cancer Fatalities (over campaign)[d]
Loss-of-coolant accident	1.50×10^{-5}	LEU	5.31	2.12×10^{-3}	5.10×10^{-7}	2.28	1.14×10^{-3}	2.74×10^{-7}	3.37×10^{3}	1.68	4.03×10^{-4}
		MOX	5.46	2.18×10^{-3}	5.25×10^{-7}	2.34	1.17×10^{-3}	2.82×10^{-7}	3.47×10^{3}	1.73	4.16×10^{-4}
Spent-fuel-handling accident[e]	1.00×10^{-4}	LEU	0.392	1.57×10^{-4}	2.51×10^{-7}	0.212	1.06×10^{-4}	1.70×10^{-7}	99.1	4.96×10^{-2}	7.94×10^{-5}
		MOX	0.373	1.49×10^{-4}	2.38×10^{-7}	0.201	1.01×10^{-4}	1.62×10^{-7}	97.3	4.87×10^{-2}	7.79×10^{-5}

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual–a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (762 m [2,500 ft])—given exposure to the indicated dose.

[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).

[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.

[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).

[e] Postulated design basis accidents at commercial reactors are considered extremely unlikely events. They are estimated to have a frequency of between 1.0×10^{-4} and 1.0×10^{-6} per year. Because a spent-fuel-handling accident does not have a calculated frequency associated with it, it has been estimated to have the highest frequency for the purposes of this analysis.

Key: LEU, low-enriched uranium.

Table K–16. Beyond-Design-Basis Accident Impacts for McGuire With LEU and MOX Fuels

			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	**Frequency (per year)**	**LEU or MOX Core**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (person-rem)**	**Latent Cancer Fatalities[c]**	**Risk of Latent Cancer Fatalities (over campaign)[d]**
Steam generator tube rupture	5.81×10^{-9}	LEU	6.10×10^{2}	0.610	5.66×10^{-8}	5.08×10^{6}	2.54×10^{3}	2.37×10^{-4}
		MOX	6.47×10^{2}	0.647	6.02×10^{-8}	5.28×10^{6}	2.64×10^{3}	2.45×10^{-4}
Early containment failure	9.89×10^{-8}	LEU	12.2	6.10×10^{-3}	9.65×10^{-9}	7.90×10^{5}	3.95×10^{2}	6.26×10^{-4}
		MOX	12.6	6.30×10^{-3}	9.97×10^{-9}	8.04×10^{5}	4.02×10^{2}	6.37×10^{-4}
Late containment failure	7.21×10^{-6}	LEU	2.18	1.09×10^{-3}	1.26×10^{-7}	3.04×10^{5}	1.52×10^{2}	1.76×10^{-2}
		MOX	2.21	1.11×10^{-3}	1.28×10^{-7}	2.96×10^{5}	1.48×10^{2}	1.71×10^{-2}
Interfacing systems LOCA	6.35×10^{-7}	LEU	1.95×10^{4}	1	1.02×10^{-5}	1.79×10^{7}	8.93×10^{3}	0.091
		MOX	2.19×10^{4}	1	1.02×10^{-5}	1.97×10^{7}	9.85×10^{3}	0.10

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft])—given exposure to the indicated dose.
[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).
[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.
[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

Table K–17. Design Basis Accident Impacts for North Anna With LEU and MOX Fuels

			Impacts on Noninvolved Worker			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Risk of Latent Cancer Fatalities (over campaign)[d]
Loss-of-coolant accident	2.10×10^{-5}	LEU	0.114	4.56×10^{-5}	1.53×10^{-8}	3.18×10^{-2}	1.59×10^{-5}	5.34×10^{-9}	39.4	1.97×10^{-2}	6.62×10^{-6}
		MOX	0.115	4.60×10^{-5}	1.55×10^{-8}	3.20×10^{-2}	1.60×10^{-5}	5.38×10^{-9}	40.3	2.02×10^{-2}	6.78×10^{-6}
Spent-fuel-handling accident[e]	1.00×10^{-4}	LEU	0.261	1.04×10^{-4}	1.66×10^{-7}	9.54×10^{-2}	4.77×10^{-5}	7.63×10^{-8}	29.4	1.47×10^{-2}	2.35×10^{-5}
		MOX	0.239	9.56x10-5	1.53×10^{-7}	8.61×10^{-2}	4.31×10^{-5}	6.90×10^{-8}	27.5	1.38×10^{-2}	2.21×10^{-5}

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft])—given exposure to the indicated dose.
[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft]).
[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.
[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).
[e] Postulated design basis accidents at commercial reactors are considered extremely unlikely events. They are estimated to have a frequency of between 1.0×10^{-4} and 1.0×10^{-6} per year. Because a spent-fuel-handling accident does not have a calculated frequency associated with it, it has been estimated to have the highest frequency for the purposes of this analysis.

Key: LEU, low-enriched uranium.

Table K–18. Beyond-Design-Basis Accident Impacts for North Anna With LEU and MOX Fuels

			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	**Frequency (per year)**	**LEU or MOX Core**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (person-rem)**	**Latent Cancer Fatalities [c]**	**Risk of Latent Cancer Fatalities (over campaign)[d]**
Steam generator tube rupture[e]	7.38×10^{-6}	LEU	2.09×10^{2}	0.209	2.46×10^{-5}	1.73×10^{6}	8.63×10^{2}	0.102
		MOX	2.43×10^{2}	0.243	2.86×10^{-5}	1.84×10^{6}	9.20×10^{2}	0.109
Early containment failure[e]	1.60×10^{-7}	LEU	19.6	1.96×10^{-2}	5.02×10^{-8}	8.33×10^{5}	4.17×10^{2}	1.07×10^{-3}
		MOX	21.6	2.16×10^{-2}	5.54×10^{-8}	8.42×10^{5}	4.21×10^{2}	1.08×10^{-3}
Late containment failure[e]	2.46×10^{-6}	LEU	1.12	5.60×10^{-4}	2.21×10^{-8}	4.04×10^{4}	20.2	7.95×10^{-4}
		MOX	1.15	5.75×10^{-4}	2.26×10^{-8}	4.43×10^{4}	22.1	8.70×10^{-4}
Interfacing systems LOCA[e]	2.40×10^{-7}	LEU	1.00×10^{4}	1	3.84×10^{-6}	4.68×10^{6}	2.34×10^{3}	8.99×10^{-3}
		MOX	1.22×10^{4}	1	3.84×10^{-6}	5.41×10^{6}	2.70×10^{3}	1.04×10^{-2}

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual—the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft])—given exposure to the indicated dose.
[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft]).
[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.
[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).
[e] McGuire release durations and warning time spans were used in lieu of site specific data.

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

Table K–19. Ratio of Accident Impacts for MOX-Fueled and LEU-Fueled Reactors (MOX Impacts/Uranium Impacts)

	Catawba			McGuire			North Anna		
Accident	**Worker**	**MEI**	**Population**	**Worker**	**MEI**	**Population**	**Worker**	**MEI**	**Population**
LOCA	1.019	1.028	1.033	1.028	1.026	1.030	1.009	1.006	1.025
FHA	0.953	0.949	0.977	0.952	0.948	0.982	0.916	0.903	0.939
SGTR	NA	1.061	1.035	NA	1.061	1.039	NA	1.163	1.066
EARLY	NA	1.007	1.049	NA	1.033	1.018	NA	1.102	1.010
LATE	NA	1.071	0.964	NA	1.014	0.974	NA	1.027	1.094
ISLOCA	NA	1.143	1.121	NA	1.123	1.103	NA	1.220	1.154

Key: Early, early containment; FHA, fuel-handling accident; ISLOCA, interfacing systems loss-of-coolant accident; Late, late containment; LEU, low-enriched uranium; LOCA, loss-of-coolant accident; MEI, maximally exposed individual; NA, not applicable; SGTR, steam generator tube rupture.

Table K–20. Prompt Fatalities for MOX-Fueled and LEU-Fueled Reactors

Accident Scenario	LEU	MOX
Steam generator tube rupture		
Catawba	1	1
McGuire	1	1
North Anna	0	0
Interfacing systems loss-of-coolant accident		
Catawba	815	843
McGuire	398	421
North Anna	54	60

Key: LEU, low-enriched uranium.

K.1.2.6.1 Catawba

Design Basis Accidents. Table K–13 shows the risks and consequences associated with a LOCA and spent-fuel-handling accident at Catawba. The greatest risk increase to the surrounding population for a design basis accident with a MOX core configuration is approximately 3.3 percent from the LOCA. If this accident were to occur, the consequences in terms of LCFs in the surrounding population within 80 km (50 mi) would be 1.82 LCF for an LEU core and 1.88 LCF for a partial MOX core. The increased risk to the noninvolved worker is one fatality every 210 million years (4.8×10^{-9} per 16-year campaign); the MEI, one fatality every 420 million years (2.4×10^{-9} per 16-year campaign); and the population, one fatality every 160,000 years (6.4×10^{-6} per 16-year campaign). (The numbers in parenthesis indicate the corresponding risk per year [i.e., one fatality every million years is equivalent to 1.0×10^{-6} fatalities per year].)

Beyond-Design-Basis Accidents. Table K–14 shows the risks and consequences associated with four beyond-design-basis accidents at Catawba. Table K–20 shows prompt fatalities. The greatest risk increase to the surrounding population from a beyond-design-basis accident with a MOX core configuration is approximately 12 percent from the ISLOCA. If this accident were to occur, the consequences in terms of LCFs and prompt fatalities in the surrounding population within 80 km (50 mi) would be approximately 14,000 fatalities for an LEU core and 15,600 fatalities for a partial MOX core. The increased risk to the population is one fatality every 570 years (1.7×10^{-3} per 16-year campaign). The increased risk of a prompt fatality is one every 32,000 years (3.0×10^{-5} per 16-year campaign).

K.1.2.6.2 McGuire

Design Basis Accidents. Table K–15 shows the risks and consequences associated with a LOCA and spent-fuel-handling accident at McGuire. The greatest risk increase to the surrounding population for a design basis accident with a MOX core configuration is 2.9 percent from the LOCA. If this accident were to occur, the consequences in terms of LCFs in the surrounding population within 80 km (50 mi) would be 1.68 LCF for an LEU core and 1.73 LCF for a partial MOX core. The increased risk to the noninvolved worker is one fatality every 69 million years (1.5×10^{-8} per 16-year campaign); the MEI, one fatality every 120 million years (8.0×10^{-9} per 16-year campaign); and the population, one fatality every 78,000 years (1.3×10^{-5} per 16-year campaign).

Beyond-Design-Basis Accidents. Table K–16 shows the risks and consequences associated with four beyond-design-basis accidents at McGuire. Table K–20 shows prompt fatalities. The greatest risk increase to the surrounding population for a beyond-design-basis accident with a MOX core configuration is approximately 10 percent from the ISLOCA. If this accident were to occur, the consequences in terms of LCFs and prompt fatalities in the surrounding population within 80 km (50 mi) would be approximately 9,300 fatalities with an LEU core and 10,300 with a partial MOX core. The increased risk to the population is one fatality every 110 years (9.3×10^{-3} per 16-year campaign). The increased risk of a prompt fatality is one every 4,300 years (2.3×10^{-4} per 16-year campaign).

K.1.2.6.3 North Anna

Design Basis Accidents. Table K–17 shows the risks and consequences associated with a LOCA and spent-fuel-handling accident at North Anna. The greatest risk increase to the surrounding population for a design-basis-accident with a MOX core configuration is approximately 2.5 percent from the LOCA. If this accident were to occur, the consequences in terms of LCFs in the surrounding population within 80 km (50 mi) would be 1.97×10^{-2} LCF for an LEU core and 2.02×10^{-2} LCF for a partial MOX core. The increased risk to the noninvolved worker is one fatality every 7.8 billion years (1.3×10^{-10} per 16-year campaign); the MEI, one fatality every 31 billion years (3.2×10^{-10} per 16-year campaign); and the population, one fatality every 6.2 million years (1.6×10^{-7} per 16-year campaign).

Beyond-Design-Basis Accidents. Table K–18 shows the risks and consequences associated with four beyond-design-basis accidents at North Anna. Table K–20 shows prompt fatalities. The greatest risk increase to the surrounding population from a beyond-design-basis accident with a MOX core configuration is approximately 15 percent from the ISLOCA. If this accident were to occur, the consequences in terms of LCFs and prompt fatalities in the surrounding populations within 80 km (50 mi) would be approximately 2,400 fatalities for an LEU core and 2,800 fatalities for a partial MOX core. The increased risk to the population is one fatality every 730 years (1.4×10^{-3} per 16-year campaign). The increased risk of a prompt fatality is one every 43,000 years (2.3×10^{-5} per 16-year campaign).

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