Entergy

System Energy Resources, Inc.
1340 Echelon Parkway
Jackson, MS 39286-1995

CNRO-2004-00050

August 10, 2004

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Attention: Document Control Desk

DOCKET: 52-009

SUBJECT: Response to Request for Additional Environmental Information Related to Early Site Permit Application (Partial Response No. 4)

REFERENCE:

1. System Energy Resources, Inc. (SERI) letter to USNRC – Early Site Permit Application (CNRO-2003-00054), dated October 16, 2003.
2. USNRC letter to SERI – Request for Additional Information Related to the Staff's Review of the Environmental Report for the Grand Gulf Early Site Permit (ESP) Application (TAC No. MC1379), CNRI-2004-00007, dated May 19, 2004.
3. SERI letter to USNRC – Response to Request for Additional Environmental Information Related to Early Site Permit Application (Partial Response No. 1) (CNRO-2004-00043), dated July 2, 2004
4. SERI letter to USNRC – Response to Request for Additional Environmental Information Related to Early Site Permit Application (Partial Response No. 2) (CNRO-2004-00045), dated July 19, 2004
5. SERI letter to USNRC – Response to Request for Additional Environmental Information Related to Early Site Permit Application (Partial Response No. 3) (CNRO-2004-00047), dated July 22, 2004
6. SERI letter to USNRC - Followup to Early Site Permit Application Environmental Audit - Response 2 (CNRO-2004-00032), dated May 19, 2004

CONTACT:

Name	George A. Zinke
Mailing Address	1340 Echelon Parkway Jackson, MS 39213
E-Mail Address	gzinke@entergy.com
Phone Number	601-368-5381

DOCUMENT COMPONENTS:

One (1) CD-ROM is included in this submission. The CD-ROM contains the following thirty-nine (39) files:

001_GGNSABWR.INP, 49 KB, publicly available
002_GGNSPWR.INP, 48 KB, publicly available
003_GGNSEARLY.INP, 16 KB, publicly available
004_GGNSCHRONC.INP, 12 KB, publicly available
005_METGGNS2001.INP, 163 KB, publicly available
006_METGGNS2002.INP, 163 KB, publicly available
007_METGGNS2003.INP, 163 KB, publicly available
008_GGNSSIT.INP, 13 KB, publicly available
009_Soil Survey_Claiborne County.pdf, 7039 KB, publicly available
010_ESPSE.CXC, 5 KB, publicly available
011_ESPSEL.CXC, 5 KB, publicly available
012_ESPSM.CXC, 5 KB, publicly available
013_ESPSML.CXC, 5 KB, publicly available
014_ESPWE.CXC, 5 KB, publicly available
015_ESPWEL.CXC, 5 KB, publicly available
016_ESPWM.CXC, 5 KB, publicly available
017_ESPWML.CXC, 5 KB, publicly available
018_ESPSE.CX3, 12 KB, publicly available
019_ESPSEL.CX3, 12 KB, publicly available
020_ESPSM.CX3, 12 KB, publicly available
021_ESPSML.CX3, 12 KB, publicly available
022_ESPWE.CX3, 12 KB, publicly available
023_ESPWEL.CX3, 12 KB, publicly available
024_ESPWM.CX3, 12 KB, publicly available
025_ESPWML.CX3, 16 KB, publicly available
026_ESPSE.CXD, 2 KB, publicly available
027_ESPSEL.CXD, 2 KB, publicly available
028_ESPSM.CXD, 2 KB, publicly available
029_ESPSML.CXD, 2 KB, publicly available
030_ESPWE.CXD, 2 KB, publicly available
031_ESPWEL.CXD, 2 KB, publicly available
032_ESPWM.CXD, 2 KB, publicly available
033_ESPWML.CXD, 2 KB, publicly available
034_GGNSABWR2001.OUT, 922 KB, publicly available
035_GGNSABWR2002.OUT, 921 KB, publicly available
036_GGNSABWR2003.OUT, 915 KB, publicly available
037_GGNSPWR2001.OUT, 736 KB, publicly available
038_GGNSPWR2002.OUT, 736 KB, publicly available
039_GGNSPWR2003.OUT, 731 KB, publicly available

In the referenced May 19, 2004, letter (Reference 2) the U.S. Nuclear Regulatory Commission requested additional information to support review of the SERI ESP Application. This letter transmits information as outlined in Attachment 1 to this letter and includes responses to:

E4.1-2 (final), E5.3-1 (Corrected), E7.2-1, E7.2-2, E7.2-3, E7.2-4.

Responses to the following requests for additional information contained in Reference 2 will be submitted at a later date:

S2.1-1, S2.1-2

Should you have any questions, please contact me.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on August 10, 2004.

Sincerely,

George A. Zinke
Project Manager
System Energy Resources Inc.

Enclosure: One CD-ROM

Attachment: Attachment 1

cc: Mr. R. K. Anand, USNRC/NRR/DRIP/RNRP
Mr. C. Brandt, PNL
Ms. D. Curran, Harmon, Curran, Spielberg, & Eisenberg, L.L.P.
Mr. W. A. Eaton (ECH) (w/o enclosure)
Mr. B. S. Mallett, Administrator, USNRC/RIV
Mr. J. H. Wilson, USNRC/NRR/DRIP/RLEP

Resident Inspectors' Office: GGNS

ATTACHMENT 1

SECTION 4.1, LAND-USE IMPACTS

Request:

E4.1-2 **Section 4.1 of ER (Land Use Impacts).** The following is stated: "Review of the Claiborne County Soil Survey issued in 1963 and inquiry with the Claiborne County Natural Resources Conservation Service (NRCS) indicates the presence of soil types, which may be considered "Prime Farmland" at the GGNS site (Reference 4). However, some exclusions apply. If land is frequently flooded during the growing season or is already in or committed to urban development or water storage, it is not considered "prime farmland" (References 4, 5, 6 and 7)." References 4, 6, and 7 do not appear to be publicly available or are not cited completely enough to permit acquiring them. During the site audit the applicant indicated that these references would be made available:

4. United States Department of Agriculture, Soil conservation Service, in Cooperation with the Mississippi Agricultural Experiment Station, "Claiborne County Soil Survey," issued July 1963.
6. Carver, A.D. and J.E. Yahner, Defining Prime Agricultural Land and Methods of Protection Purdue Cooperative Extension Service, AY-283.
7. United States Department of Agriculture, Soil Conservation Service, May 28, 1992, Obtained from the Claiborne County NRCS, Port Gibson, MS, February 21, 2003.

Response:

(Final Response) Attached is Reference 4. (References 6 and 7 above were provided in Cover Letter Reference 5)

See file: 009_Soil Survey_Claiborne County.pdf

SECTION 5.3, COOLING SYSTEM IMPACTS

Request:

E5.3-1 **Section 5.3.2.1 (Thermal Description and Physical Impacts).** Provide input files (electronic) for CORMIX model simulations.

Response:

CORRECTED RESPONSE: The CORMIX input files provided with Cover Letter Reference 6 (and referenced in Cover Letter Reference 4) were not the latest files used for the application. Included with this response are the latest input files. Also included are the output files.

See files:	Input Files:	010_ESPSE.CXC
		011_ESPSEL.CXC
		012_ESPSM.CXC
		013_ESPSML.CXC
		014_ESPWE.CXC
		015_ESPWEL.CXC
		016_ESPWM.CXC
		017_ESPWML.CXC
	Output Files:	018_ESPSE.CX3
		019_ESPSEL.CX3
		020_ESPSM.CX3
		021_ESPSML.CX3
		022_ESPWE.CX3
		023_ESPWEL.CX3
		024_ESPWM.CX3
		025_ESPWML.CX3
		026_ESPSE.CXD
		027_ESPSEL.CXD
		028_ESPSM.CXD
		029_ESPSML.CXD
		030_ESPWE.CXD
		031_ESPWEL.CXD
		032_ESPWM.CXD
		033_ESPWML.CXD

SECTION 7.2, SEVERE ACCIDENTS

Request:

E7.2-1 Section 7.2.2. Please provide an up-to-date, site-specific assessment of the adverse health effects from fallout onto open bodies of water, considering the ESP site parameters (e.g., water flow rates and containment residence times). Justify that the generic conclusion with respect to such matters that was reached in NUREG-1437 is valid for a future reactor at the ESP site.

Response:

In NUREG-1437 (GEIS), Grand Gulf is one of just four sites described as a "large river site" for the purpose of evaluating fallout into open bodies of water. Table 5.16 of NUREG-1437 shows that large river sites are generically the most advantageous in terms of annual edible aquatic food harvest, whole body population dose, and total exposure per reactor-year in person-rem. This is due to the high dilution effect and low residence times associated with a large river. Table 5.15 of NUREG-1437 shows Grand Gulf is bounded by analyses performed at Fermi, and Table 5.14 shows the following results in comparison to that site (the far right column is from Table 5.16 of NUREG-1437):

Plant	Type of site	Residence time (years)	Surface area to volume ratio	Potentially affected population [1]	Percentage of population likely to be affected	Population Exposure per Reactor Year
Fermi	Lake	2.6	5.6×10^{-2}	6,647,763	41	1400
Grand Gulf	Large River	1.2×10^{-3}	1.7×10^{-1}	504,930	18	0.4

These conclusions are applicable to a future reactor at the Grand Gulf site. The water flow rates[2] have not changed significantly. Contaminant resident times are not expected to change relative to that documented in NUREG-1437. However, it is recognized that a large river has the ability to remove contaminants rapidly; that is, in terms of days or weeks rather than years (for a large river site such as Grand Gulf).[3] Given this characteristic, variations over time in parameters important to residence time (for a large river site) can be expected to have little impact on overall results and conclusions.

The population in the area surrounding the GGNS site has not grown significantly[4] and would continue to be much smaller than the Fermi site values. Therefore, the GEIS generic conclusions that:

- doses due to fallout to surface water at Grand Gulf will be bounded by a large margin in comparison to the NUREG-1437 documented Fermi analysis, and
- doses due to fallout to surface water are expected to be a small fraction of the atmospheric dose path at Grand Gulf continues to remain valid for a future reactor at the ESP site.

---

[1] Per NUREG-1437, this is a projected population (2050) for the 50 mile population around the site (NUREG-1437 Tables 5.5, 5.7, and 5.14b).

[2] River flow history was reviewed in support of the ESP ER. As noted in ER Section 2.3.1.1.4, with data updated through 2000, the river's minimum flow value continues to be set by flow measurements taken in the 1930's (i.e., approximately 100,000 cfs).

[3] NUREG-1437, Section 5.3.3.3.2.

[4] Comparing 50 mile populations from the GGNS UFSAR Table 2.1-3 (1970 census; approximately 270,000) and ESP ER Tables 2.5-1, 2.5-6; approximately 332,000), the population growth rate is <10% per decade for this 30 year period. Further, the updated projections for growth rate established in the cited ESP ER tables (from the 2000 census through 2070) confirm that population growth within 50 miles is expected to be <10% per decade through 2070.

Request:

E7.2-2 **Section 7.2.2.** Please provide an up-to-date, site specific assessment of the adverse health effects from potential releases to groundwater, considering the ESP site parameters. Justify that the generic conclusion with respect to such matters that was reached in NUREG-1437 is valid for a future reactor at the ESP site.

Response:

In NUREG-1437 (GEIS), Grand Gulf is one of just four sites described as a "large river site" for the purpose of evaluating potential releases to groundwater. Table 5.17 of NUREG-1437 shows that large river sites are generically the most advantageous in terms of groundwater ingestion total dose, even in comparison to coastal sites, which have higher doses from seafood ingestion. In addition, pathway interdiction can reduce the dose by an order of magnitude. This is particularly possible at Grand Gulf due to the low ground water velocities and the distance to the river. The conclusions in NUREG-1437 estimate the groundwater doses at large river sites to be about 12 person-rem per reactor year (RY). By comparison, the dose at small river sites is estimated to be 1000 person-rem/RY, and at estuarine without interdiction, 17,700 person-rem/RY.

The NUREG-1437 conclusions are based on consideration of site-specific information on groundwater travel time; retention-adsorption coefficients; distance to surface water; and soil, sediment, and rock characteristics. None of these parameters would be expected to change significantly over the life of the future ESP plant. And, as indicated in response to RAI E7.2-1, the population and predicted growth rate are relatively small such that the Grand Gulf site continues to have a relatively low population exposure.

In addition, as noted in the response to RAI E7.2-3, a MACCS2 severe accident consequence analysis has been performed for the GE Advance Boiling Water Reactor design and the Westinghouse AP 1000 design for the Grand Gulf ESP site. Due to the low site population and the low release frequencies of these designs, the total water ingestion dose risk was estimated at less than 0.005 person-rem/RY (as compared to NUREG-1437 estimates of 0.4 person-rem/RY for open bodies of water and 12 person-rem/RY for groundwater). Thus the MACCS2 analysis indicates that the generic NUREG-1437 analysis is conservative and bounding for these advance reactor designs at Grand Gulf.

Request:

E7.2-3 Section 7.2. Provide a site-specific analysis of the environmental consequences of a potential severe accident at a new reactor located on the ESP site using a Level 3 probabilistic risk assessment (PRA) consequence code such as the MACCS2 code. This could involve characterizing the spectrum of credible releases from candidate future plant designs, in terms of representative source terms and their respective frequencies, and using these release characteristics in conjunction with site-specific population and meteorology to determine site-specific risk impacts for the potential design. Release characteristics could be developed through a survey of severe accident analyses for previously certified advanced LWRs and/or operating reactors. The following information should be provided as part of this analysis:

a. a description of the computer code used as the basis for the calculations, including any modifications to the officially released version of the code and important deviations from recommended or default code input values;

b. a description of the site-specific meteorology data used in the calculation, including the treatment of rain/precipitation events and the degree to which the data represents or bounds year-to-year variations in weather at the ESP site;

c. a description of the site-specific population data used in the calculation and justification that this data is representative of the time period through which new unit operations could extend;

d. a description of the major input assumptions for modeling economic impacts, including farm and non-farm values, evacuation costs, value of crops and milk contaminated or condemned, costs of decontamination of property, and costs associated with loss of use of property as a result of the accident (including contamination and condemnation of property);

e. a description of the protective actions considered in the evaluation, including criteria for sheltering and evacuation, criteria for interdiction and condemnation of property and/or crops and the assumed level of medical support to aid the exposed population;

f. a description of the source terms used to represent the reference or surrogate plant design(s), including the radionuclide inventory and the release frequency and characteristics for each release category, including release fractions for the major radionuclide groups, release times and durations, and elevation and energy of release,

g. the results of the calculations in terms of probabilistically-weighted population dose, early and latent fatalities, economic costs, and contaminated and condemned land areas, for the reference or surrogate plant design(s) (Sufficient information should be provided to enable results to be displayed in a manner similar to later final environmental statements [FESs, e.g., Tables 5.10 through 5.13 in NUREG-0921].); and

h. a listing of the input file for the ESP site (including weather data).

Response:

A severe accident consequence analysis was calculated using the Level 3 probabilistic risk assessment (PRA) MACCS2 (Melcor Accident Consequence Code System) code. An attempt was made to be consistent in terms of input and analysis methodology with a recently completed severe accident analysis of a proposed future reactor at the North Anna ESP site (Reference 1 to this response). The same types of reactors were evaluated for the Grand Gulf ESP site, using the same vendor input information.

a. Code: The analysis was performed with the MACCS2 version designated as Oak Ridge National Laboratory RSICC Computer Code Collection MACCS2 V.1.13.1, CCC-652 Code Package. MACCS2, Version 1.13.1, released in January 2004, simulates the impact of severe accidents at nuclear power plants on the surrounding environment. The principal phenomena considered in MACCS2 are atmospheric transport, mitigating actions based on dose projections, dose accumulation by a number of pathways including food and water ingestion, early and latent health effects, and economic costs. The basis model had no important deviations from the default code input values, except for site-specific values and reactor design information. The code values modified for the future designs were primarily the source term data from vendor Level 2 probabilistic safety analyses. The respective reactor vendors provided the Level 2 data for the AP1000 and ABWR designs. This data includes the radionuclide inventory, power level, release fractions and corresponding frequencies, plume release start time, plume release height, delay and duration. Values for the ATMOS input data file (one of the five input files used by MACCS2) was modified, as necessary, to use data appropriate for the ABWR or AP1000 source terms and probability frequencies. (Refer to the response to Part f.) The remaining four MACCS2 input files were reviewed and modified as necessary. All MACCS2 GGNS input files are provided per Part h. below.

b. Meteorology: Three years (2001 -2003) of site-specific hourly meteorological data were used in the analyses. These three recent, consecutive years are considered to be a representative set of data for the site and represents a reasonable bound of year-to-year variations at the ESP site. The three years are each analyzed separately. The results reported below are based on the limiting year for each result. It is noted that the year-to-year variation in meteorology data does not have a significant impact on the MACCS2 output (about 6% variation).

The hourly data (wind direction, wind speed, and precipitation) were collected on-site at the GGNS met tower. These data and their collection are described in the ESP Application SSAR. Stability class was calculated using the GGNS site meteorological data and the methodology of Regulatory Guide 1.23, Table 2 (Reference 5). Missing data were replaced by data from adjacent hours consistent with the recommendations of the EPA in Reference 6; however, when the data gap involved a long sequential period, the entire period was modeled by data from another year. This is not believed to be significant for the following reasons:

(1) The replacement data were reviewed and found to be consistent with that from adjacent periods.

(2) The volume of missing data was small (i.e., 483 hours out of 26,280 hours).

(3) The three separate yearly analyses show relatively consistent results.

Morning and afternoon mixing height values were taken from Table 2.3-125, Mixing Heights at Jackson International Airport, of the ESP SSAR, with the median values selected from Jan-Feb-March for winter season, and so on. The treatment of rain/precipitation events follows the default recommend parameter values given in the ATMOS file supplied with the MACCS2 code.

c. Population: The population distribution and land use information for the region surrounding the ESP site are specified in the SITE input data file. Contained in the SITE input data file are the geometry data used for the site (spatial intervals and wind directions), population distribution, fraction of the area that is land, watershed data for the liquid pathways model, information on agricultural land use and growing seasons, and regional economic information. Some of the detailed data in this input file supercedes certain data in the EARLY input data file.

A 50-mile radius area around the site was divided into sixteen directions that are equivalent to a standard navigational compass rosette. This rosette was further divided into inner radial rings consistent with the ESP ER Figures 2.5-1 and 2.5-2.

It is noted that this population data is associated with the year 2002. In order to extrapolate results to other years, the results can be multiplied by population growth ratios contained within NUREG-1437. The Exposure Index (EI), defined in that NUREG, was verified to be consistent with the above population and meteorology data. The average population out to 10 miles is 453 people per each of the 16 wind segments; however, the estimated EIs (10 and 150 miles) for 2000 are slightly less than the NUREG-1437 values because the prevailing winds are away from population centers. The following estimated EIs are generated for the GGNS site based on NUREG-1437 population ratios and extrapolations.

	1990	2000	2010	2030	2050	2065
Population within 50 miles	350,000	380,000	410,000	450,000	500,000	540,000
Multiplier	0.92	1.00	1.08	1.18	1.32	1.42
10-mile EI	393	427	461	506	562	607
150-mile EI	271772	295066	318361	349421	388245	419305

d. Major site assumptions other than met data and population data:

(1) The land fractions are interpolated off of ER Figures 2.5-1 and 2.5-2 (and can be seen in the input files). However, for watershed definitions in terms of ingestion factors for Sr-89, Sr-90, Cs-134, and Cs-137, it is conservative to ignore the Mississippi River and treat all segments as land.

(2) Regional indices are identified as either Mississippi or Louisiana for region indexing. The two states have similar fractional dairy, total annual farm sales in dollars per hectare, property values in dollars/hectare, and non-farm property values in dollars/person, but the land fraction devoted to farming is different within a 50 mile radius of the plant. Most of the Mississippi side of the river is forested land within this range of the site. The default economic values supplied by the code were increased by the Consumer Price Index ratio of the average value of 109.6 for 1986 (when the NUREG-1150 data above was generated) to 189.1 for May, 2004. Details regarding farm acreage for the counties within a 50-mile radius of the plant were taken from federal statistics in Reference 7.

Region #	State	Fraction farm	Fraction Dairy	Farm Sales ($/hectare)	Property value ($/hectare)	Non-farm property values ($/person)
16	LA	.655	.074	792	5665	105225
22	MISS	.284	.054	695	3595	91425

(3) The crop information required by MACCS2 input are of a slightly different format than similar information provided in the ESP ER. Values were collected from county statistics in Reference 8 for the Louisiana side of the River, and for Districts 2 and 4 on the Mississippi side. These were combined weighted by the total farmland area within the 50-mile radius to produce a single composite measure.

	LA	MS-2	MS-4	Composite
Pasture	0.253	0.291	0.595	0.310

Stored Forage	0.039	0.042	0.337	0.083
Grains	0.093	0.108	0.032	0.087
Green Leafy	0	0	0	0.000
Other	0.200	0.194	0.000	0.170
Legumes/seeds	0.415	0.365	0.036	0.350
Roots/tubers	0	0	0	0.000

(4) The growth season assumed in other GGNS ESP dose calculations was conservatively assumed to be all year long. This assumption was also applied to the MACCS2 analysis.

e. Protective actions: The EARLY module of the MACCS2 code models the time period immediately following a radioactive release. This period is commonly referred to as the emergency phase. It may extend up to one week after the arrival of the first plume at any downwind spatial interval. The subsequent intermediate and long-term periods are treated by CHRONC module of the code. In the EARLY module the user may specify emergency response scenarios that include evacuation, sheltering, and dose-dependent relocation. The EARLY module has the capability for combining results from up to three different emergency response scenarios. This is accomplished by appending change records to the EARLY input data file. The first emergency-response scenario is defined in the main body of the EARLY input data file. Up to two additional emergency-response scenarios can be defined through change record sets positioned at the end of the file.

This analysis used the same assumptions as Reference 1 and the default-supplied data. The emergency evacuation model has been modeled as a single evacuation zone extending out 10 miles from the site. For the purposes of this analysis, an average evacuation speed of 1.8 m/s is used with a 7200 second delay between the alarm and start of evacuation, with no sheltering for the base case. Once evacuees were more than 20 miles from the site, they disappear from the analysis. The evacuation scenario is weighted 95%, compared to no evacuation for the purpose of composite results.

f. Source terms: The ATMOS input data file calculates the dispersion and deposition of material released "source terms" to the atmosphere as a function of downwind distance. Source term release fractions (RELFRC) for the ABWR and AP1000 are shown below, as are plume characterizations, respectively. These data include the source term inventory, power level, release fractions, plume start time, plume release height, delay and duration.

The ABWR shows 10 different source term categories (STCs). See Table 1. The release times and durations, and elevation and energy of release for the ABWR were extracted from the GE ABWR licensing submittal document (Reference 2). Parameters are assigned to each source term according to STC number. Each release plume is assumed to have only one segment. See Table 2. The scaling factor (CORSCA) was used to adjust the ABWR core inventory for a power level of 4300 MWt. The core inventory was based on the discharge exposure burnup of 35,000 MWD/MT.

Vendor data was also used to characterize the AP1000 source term category release fractions and corresponding frequencies for the MACCS2 element groups (References 3 and 4). Four plume segments of release fraction data were originally reported, but were collapsed to two in order to be consistent with the Reference 1 analysis. The process of collapsing the plume data results in the same total releases occurring in the two plumes that the vendor modeled as occurring in four plumes. Table 3 (below) provides the collapsed source term release fractions for 7 different source term categories (STCs). Timing data indicated in the table below was also revised to represent two plume segments. A plume energy level 3.0E+06 W was assigned to the first plume and 2.0E+06 W for the second plume except for the bypass sequence. The ALARM time was selected to be the same as the first plume DELAY time. The balance of the timing data of each plume is taken from the Westinghouse PRA Study document. See Table 4. The scaling factor (CORSCA) used to adjust the AP1000 core inventory for power level was (3415/3412 =) 1.0009. This was determined due to the base 3412 MWI MACCS2 pressurized water reactor default inventory and the actual AP1000 thermal power rating of 3415 MWt. The GGNS input uses slightly more conservative core inventories and slightly different REFTIM data than Reference 1 based on interpretation of Reference 4 material.

g. Results: The results of the dose and dollar risk assessments for the AP1000 and ABWR plant designs are provided in Table 5. These are the results from the year of meteorology that provided the highest risk. Risk is defined in these results as the product of source term category frequency and the dose or cost associated with the STC. The total risk is assumed to be the sum of all scenarios. Since the AP1000 and ABWR plant designs reflect different release/source term categories, use of the total/summed risk provides a common reference point.

The maximum dose risk sensitivity to the meteorological data was shown to differ by approximately 6% from the limiting case for both the AP1000 and ABWR plant designs. A similar sensitivity to the meteorological data was seen for the dollar risk. The highest mean values for affected land areas are shown in Table 5. The mean values for affected land areas are given in hectares and are not totaled for all STCs. Instead, the values reflect the maximum area associated with the worst-case single release scenario. The values for total early and latent fatalities per year were conservatively calculated as the sum of all release scenarios for the limiting meteorological data year. Tables 6, 7, and 8 support the calculated dose/year and dollars/year risks for both advanced reactor designs presented in Table 5. As can be seen from the cited tables and results, consequences from severe accidents from the two advanced reactor designs are products of significantly lower risk factors when compared to existing plant inputs (see response to Request E7.2-4). This is consistent with GEIS findings for existing plants that risk impacts from severe accidents would be small. It is also noted that the relatively low local population reduces the risk even further.

h. Input files: The following are the input file names used in this analysis. All input files have been provided:

001 GGNSABWR.INP	(ATMOS file for the ABWR design)
002 GGNSPWR.INP	(ATMOS file for the AP 1000 design)
003 GGNSEARLY.INP	(EARLY file for the GGNS site)
004 GGNSCHRONC.INP	(CHRONC file for the GGNS site)
005 METGGNS2001.INP	(Year 2001 meteorology data - GGNS site)
006 METGGNS2002.INP	(Year 2002 meteorology data - GGNS site)
007 METGGNS2003.INP	(Year 2003 meteorology data - GGNS site)
008 GGNSSIT.INP	(Data for the GGNS site)

Also provided are the output files.

REFERENCES:

1. Response to 3/12/04 Environmental RAIs for North Anna ESP, E. S. Grecheck, Dominion Nuclear North Anna letter to the NRC DCD, Serial 04-170, Docket No. 52-008, dated May 17, 2004 (ADAMS Accession No. ML041450041)
2. General Electric Advanced Boiling Water Reactor (GE ABWR) Standard Safety Analysis Report 23A6100, Revision 4.
3. AP1000 Design Control Document, Westinghouse Electric Corporation, Revision 8, 2003.
4. AP1000 Probabilistic Risk Assessment Report, Westinghouse Electric Corporation.
5. Regulatory Guide 1.23, "Safety Guide 23, Onsite Meteorological Programs."
6. USEPA document, Dennis Atkinson and Russell F. Lee, "Procedures for Substituting Values for Missing NWS Meteorological Data for Use in Regulatory Air Quality Models," July 7, 1992.
7. US federal and use statistics collected from http://www.fedstats.gov/qf/states/
8. Agricultural Marketing Services branch of the United States Department of Agriculture (USDA) agricultural statistics state summary web pages at http://www.ams.usda.gov/statesummaries/LA/District.htm and http://www.ams.usda.gov/statesummaries/MS/District.htm.

Table 1
ABWR Source Term Release Fractions

STC	Xe/Kr	I-Br	Cs-Rb	Te-Sb	SR	Co-Mo	LA	CE	BA
0	4.40E-2	2.30E-5	2.30E-5	5.30E-6	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
1	1.00E+0	1.50E-7	1.30E-5	3.10E-4	6.30E-6	2.40E-11	7.90E-8	7.90E-8	6.30E-6
2	1.00E+0	5.00E-6	5.00E-6	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
3	1.00E+0	2.80E-4	2.20E-3	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
4	1.00E+0	1.60E-3	1.60E-3	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
5	1.00E+0	6.00E-3	5.30E-4	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
6	1.00E+0	3.10E-2	7.70E-2	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
7	1.00E+0	8.90E-2	9.90E-2	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
8	1.00E+0	1.90E-1	2.50E-1	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0.00E+0
9	1.00E+0	3.70E-1	3.60E-1	1.10E-3	9.30E-3	9.20E-8	2.80E-3	2.80E-3	9.30E-3

NOTE: STC is the source term category and refers to the Base Case 0 and the 9 stacked cases in the ATMOS input for the GE ABWR. Data from Table 19E.3-6, Reference 2, and Reference 1.

Table 2
ABWR Plume Characterization Data

STC	Alarm (s)	Number of Plume Releases	Risk-Dominant Plume	REF TIM	Plume heat (W)	Plume Release Height (m)	Plume Duration (s)	Plume Delay (s)
0	6120.0	1	1	0.0	1.38E+6	37	36000.0	9720.0
1	69120.0	1	1	0.0	1.38E+6	37	3600.0	72000.0
2	65520.0	1	1	0.0	1.38E+6	37	3600.0	68400.0
3	177120.0	1	1	0.0	1.38E+6	37	36000.0	180000.0
4	69120.0	1	1	0.0	1.38E+6	37	3600.0	72000.0
5	69120.0	1	1	0.0	1.38E+6	37	3600.0	68400.0
6	65520.0	1	1	0.0	1.38E+6	37	36000.0	68400.0
7	69120.0	1	1	0.0	1.38E+6	37	36000.0	72000.0
8	4320.0	1	1	0.0	4.19E+6	37	36000.0	7200.0
9	43920.0	1	1	0.0	1.38E+6	37	36000.0	84960.0

NOTE: Alarm time is seconds after accident that emergency conditions are reached as defined in NUREG-0654; since only one plume assumed in each scenario, the risk dominant plume is always 1; a REFTIM of 0 uses the leading edge as a locator for plume contents, the plume delay is the time after SCRAM. Data from Table 19E.3-6 of Reference 2.

Table 3
PWR Source Term Release Fractions

STC	Noble Gases	I	CS	TE	SR	RU	LA	CE	BA
CFI	7.98E-1	3.33E-3	3.32E-3	4.35E-4	2.18E-2	9.28E-3	8.06E-3	4.32E-5	1.65E-2
"	1.22E-1	0.0E+0	0.0E+0	6.04E-6	0.0E+0	0.0E+0	1.12E-2	4.06E-5	0.0E+0
CFE	8.21E-1	5.66E-2	5.49E-2	1.39E-3	3.48E-3	1.42E-2	6.54E-5	1.00E-6	5.28E-3
"	1.42E-1	0.0E+0	0.0E+0	6.04E-7	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
DIRECT	4.43E-3	3.61E-5	3.46E-5	2.42E-6	3.22E-5	3.94E-5	4.06E-6	1.76E-8	3.61E-5
"	3.50E-3	0.0E+0	0.0E+0	5.44E-9	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
IC	1.48E-3	1.20E-5	1.15E-5	8.09E-7	1.07E-5	1.31E-5	1.36E-6	5.88E-9	1.20E-5
"	1.17E-3	0.0E+0	0.0E+0	1.81E-9	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
BP	1.0E+0	2.15E-1	1.96E-1	9.39E-3	3.57E-3	4.48E-2	1.30E-4	3.19E-6	8.93E-3
"	0.0E+0	2.34E-1	7.60E-2	6.89E-3	0.0E+0	0.0E+0	0.0E+0	0.0E+0	1.00E-6
CI	6.86E-1	4.56E-2	2.10E-2	1.65E-3	2.03E-2	4.04E-2	2.39E-4	2.97E-6	3.16E-2
"	8.40E-2	0.0E+0	0.0E+0	9.37E-5	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.00E+0
CFL	1.53E-3	1.21E-5	1.15E-5	1.02E-6	1.67E-5	1.71E-5	1.17E-5	4.79E-8	1.68E-5
"	9.79E-1	2.13E-5	1.19E-5	3.67E-5	2.83E-3	1.42E-3	1.41E-1	5.34E-4	2.60E-3

NOTE: STC is the source term category. The second row for each STC applies to the second plume. Data are developed from Table 49-2 of Reference 4.

Table 4
PWR Plume Characterization Data

STC	Alarm (s)	Number of Plume Releases	Risk-Dominant Plume	REFTIM	Plume heat (W)	Plume Release Height (m)	Plume Duration (s)	Plume Delay (s)
CFI	2924	2	1	0.0	3.0E+6	30	53830	2924
CFI	2924	2	1	0.5	2.0E+6	30	86400	32590
CFE	3004	2	1	0.0	3.0E+6	30	70160	3004.
CFE	3004	2	1	0.0	2.0E+6	30	86400	19810.
DIRECT	4378	2	1	0.5	3.0E+6	30	80432	4378.
DIRECT	4378	2	1	0.0	2.0E+6	30	86400	84810.
IC	4378	2	1	0.5	3.0E+6	30	80432	4378.
IC	4378	2	1	0.0	2.0E+6	30	86400	84810.
BP	31890	2	1	0.5	3.0E+6	30	40050	31890.
BP	31890	2	1	0.0	3.0E+6	30	86400	46440.
CI	100.8	2	1	0.5	3.0E+6	30	86380	100.8
CI	100.8	2	1	0.5	2.0E+6	30	75300	50020.
CFL	2922	2	1	0.5	3.0E+6	30	81640	2922.
CFL	2922	2	1	0.5	2.0E+6	30	86400	26360.

NOTE: Alarm time is seconds after accident that emergency conditions are reached as defined in NUREG-0654; in all cases, the first plume is dominant in terms of risk, the REFTIM value of 0.5 uses the midpoint of the plume as content locator; the plume delay is the time after SCRAM. Data is condensed from Table 49-2 of Reference 4.

Table 5
Results Summary Comparison of Plant Designs

Plant Design	Dose Risk (Person-Rem/yr)	Dollar Risk (per year)	Affected Land (in Hectares)	Early Fatalities (per year)	Latent Fatalities (per year)
ABWR	0.002	2.82	158,000	1.51E-12	1.05E-6
AP1000	0.013	26.7	152,000	$<10^{-12}$	6.94E-6

NOTE: Results are for 0-50 mile radius from the ESP Site.

Table 6
ABWR Mean Value for Total Dose Risk Assessment

STC	STC Freq. (per year)	Case 1A (2001 data)	Case 1B (2002 data)	Case 1C (2003 data)
0	1.34E-07	8.20E-04	1.13E-03	8.60E-04
1	2.08E-08	9.32E-05	1.24E-04	9.82E-05
2	1.00E-10	1.44E-07	1.95E-07	1.61E-07
3	1.00E-10	1.96E-05	2.27E-05	1.32E-05
4	1.00E-10	1.24E-05	1.31E-05	9.16E-06
5	1.00E-10	4.64E-06	4.98E-06	3.82E-06
6	1.00E-10	5.84E-05	5.43E-05	4.71E-05
7	3.91E-10	2.41E-04	2.32E-04	2.13E-04
8	4.05E-10	3.56E-04	3.52E-04	3.39E-04
9	1.70E-10	1.82E-04	1.77E-04	1.68E-04
Total		1.79E-03	2.11E-03	1.75E-03

NOTE:Data is in Person-Rem/year. The three cases refer to the three different years of meteorological data. The worst-case year is used to select the data for Table 5. STC Freq. is from Table 19E.3-6, Reference 2.

Table 7
PWR Mean Value for Total Dose Risk Assessment

STC	STC Freq. (per year)	Case 1A (2001 data)	Case 1B (2002 data)	Case 1C (2003 data)
CFI	1.89E-10	5.82E-05	6.07E-05	6.01E-05
CFE	7.47E-09	2.81E-03	2.72E-03	2.80E-03
IC	2.21E-07	6.98E-04	9.06E-04	7.71E-04
BP	1.05E-08	9.02E-03	8.63E-03	8.90E-03
CI	1.33E-09	4.19E-04	4.12E-04	4.14E-04
CFL	3.45E-13	1.48E-09	1.44E-07	1.43E-09
Total		1.30E-02	1.27E-02	1.30E-02

NOTE:Data is in Person-Rem/year. The three cases refer to the three different years of meteorological data. The worst-case year is used to select the data for Table 5. STC Freq. is from Table 19.59-16 of Reference 3.

Table 8
Dollar Risk Assessment

Design	STC	Case 1A (2001 data)	Case 1B (2002 data)	Case 1C (2003 data)
ABWR	All	2.64	2.82	2.02
PWR	All	26.1	26.1	26.3

NOTE: Data is in Dollars/year.

Response

E7.2-3 **Section 7.2** **(response continued, final page)**

See files:	Input Files:	001_GGNSABWR.INP
		002_GGNSPWR.INP
		003_GGNSEARLY.INP
		004_GGNSCHRONC.INP
		005_METGGNS2001.INP
		006_METGGNS2002.INP
		007_METGGNS2003.INP
		008_GGNSSIT.INP
	Output Files:	034_GGNSABWR2001.OUT
		035_GGNSABWR2002.OUT
		036_GGNSABWR2003.OUT
		037_GGNSPWR2001.OUT
		038_GGNSPWR2002.OUT
		039_GGNSPWR2003.OUT

Request:

E.7.2-4 **Section 7.2.** Provide a comparison of the (probabilistically weighted) environmental risk of severe accidents for a future reactor at the ESP site with:

a. the risks (doses) associated with normal and anticipated operational releases from a future reactor at the ESP site; and

b. the risk of severe accidents for the current generation of operating plants (at their respective sites), as characterized in such studies as NUREG-1150, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, and the plant-specific risk study for Grand Gulf Nuclear Station.

Response:

a. The probabilistically weighted environmental risks of severe accidents are quantified in response to RAI E7.2-3. Due to the relatively low population density, and the extremely low frequency of severe accidents in the ABWR and AP1000 PRA results, the environment risks of severe accidents are extremely low. The weighted total dose risk is less than 0.013 person-rem/year.

   From the ESP ER Section 5.4, the normal and anticipated operational releases from a future reactor at the ESP site are also very low, but still greater than the weighted risk of severe accidents. ER Table 5.4-13 lists the estimated population[5] whole body dose from airborne releases as 3.37 person-rem/yr. The conclusion is that the weighted environmental risks of severe accidents are much less than those associated with normal operation.

   It is emphasized that the environmental risks of normal operation are themselves very low. As stated in the ER Section 5.4.3.2, existing background radiation sources amount to about 130 mrem/yr. The worst case calculated individual dose due to a future reactor is shown in ER Table 5.4-11B to be less than 4 mrem/yr, and that calculated maximum is a bounding value that is not expected to actually occur.

b. The results of severe accidents for current generation reactors as characterized in NUREG-1150, the plant-specific study conducted for Grand Gulf in NUREG/CR-4551, and the ESP submittal for the North Anna site were all reviewed and compared to the severe accident risk calculated in the MACCS2 analysis discussed in RAI E7.2-3. The conclusions are:

   (1) the Grand Gulf ESP site's low population provides low risk (even with current reactor design), and

[5] Annual radiation exposure associated with gaseous releases from the ER Table 5.4-13 is associated with the population with a 50-mile radius of the GGNS site (ER 5.4.3.2).

(2) the low frequency of releases associated with the ABWR and AP1000 designs make the severe accident risk of a future unit at this site extremely low.

Plant	Population Dose (50 miles) (person-rem/yr)
Zion (Reference 1)	5.47E+01
Grand Gulf Existing Unit (Reference 2)	5.2E-01
Surry (Reference 3)	6.E+00
North Anna (Reference 4)	2.51E+01
North Anna AP1000 (Reference 4)	8.28E-02
North Anna ABWR (Reference 4)	5.93E-03
Grand Gulf AP1000	1.3E-02
Grand Gulf ABWR	2.0E-03

REFERENCES:

1. NUREG/CR-4551, "Evaluation of Severe Accident Risks: Zion, Unit 1," U. S. Nuclear Regulatory Commission, Table 5.1-1, Vol. 7, Rev.1, Part 1, March 1993.

2. NUREG/CR-4551, "Evaluation of Severe Accident Risks: Grand Gulf, Unit 1," U. S. Nuclear Regulatory Commission, Table 5.1-1, Vol. 6, Rev.1, Part 1, December 1990.

3. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants Final Summary Report," U. S. Nuclear Regulatory Commission, Table 12, Vol. 1, December 1990.

4. Response to 3/12/04 Environmental RAIs for North Anna ESP, E. S. Grecheck, Dominion Nuclear North Anna letter to the NRC DCD, Serial 04-170, Docket No. 52-008, dated May 17, 2004 (ADAMS Accession No. ML041450041)

TABLE OF CONTENTS (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

7.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

7.1 Design Basis Accidents

The purpose of this section is to review and analyze a robust spectrum of design basis accidents (DBAs) which bracket post-accident radiological consequences for the reactor or reactors considered for the Grand Gulf Nuclear Station (GGNS) site, to demonstrate that a reactor or reactors could be sited at the GGNS ESP Site without undue risk to the health and safety of the public. The safety assessment required by 10 CFR 52.17(a)(1) addresses the acceptability of the site under the radiological consequence evaluation factors identified in §50.34(a)(1). Pursuant to 10 CFR 50.34(a)(1), doses from postulated design basis accidents are calculated for hypothetical individuals, located at the closest point on the exclusion area boundary for a two-hour period (any two-hour period with the greatest EAB doses is used for proposed plants that utilize the Alternate Source Term methodology), and at the outer radius of the low population zone for the course of the accident. Bounding reactor source terms along with site-specific atmospheric dispersion characteristics were used. The selection of accidents evaluated, the conservative source terms used, and use of site-specific meteorology, serve to demonstrate the acceptability of the site with regards to the environmental impact related to off-site dose consequences.

7.1.1 Selection of Design Basis Accidents

A set of postulated accidents was analyzed to demonstrate that a reactor or reactors bounded by parameters defined herein can be operated on the GGNS ESP Site without undue risk to the health and safety of the public. The set of accidents was selected to cover a range of events in Regulatory Guide 1.183 (Reference 6) and NUREG-1555 for various reactor types. Evaluation of this set of accidents provides a basis for establishing site suitability. It is not the intent, nor is it strictly possible, to analyze all possible accidents for each of the reactor types identified in the ESP SSAR Section 1.3. The set of accidents chosen considers those with potential bounding impact, as well as accidents of lesser impact but greater frequency. The bounding accidents selected focus, for the most part, on the LWR designs because they have certified standard designs, and have accepted postulated accident bases.

The representative range of DBAs for the boiling water reactor (BWR), pressurized water reactor (PWR), and other designs include:

- Main Steam Line Breaks (PWR/BWR)
- Reactor Coolant Pump Locked Rotor (PWR)
- Control Rod Ejection (PWR)
- Control Rod Drop (BWR)
- Small Line Break Outside Containment (PWR/BWR)
- Steam Generator Tube Rupture - SGTR (PWR)
- Loss of Coolant Accident – LOCA (PWR/BWR/ACR)
- Fuel Handling Accident – FHA (PWR/BWR)

These accidents include those identified in NUREG-1555, Chapter 7.1 Appendix A as important for assessing the offsite dose consequences.

7.1.2 Evaluation of Radiological Consequences

Doses for selected DBAs were evaluated at the exclusion area boundary (EAB) and low population zone (LPZ) boundary. These doses must meet the site acceptance criteria of 10 CFR 50.34 and 10 CFR 100. Although the emergency safeguard features are expected to prevent core damage and mitigate releases of radioactivity, the surrogate LOCAs analyzed presume substantial meltdown of the core with the release of significant amounts of fission products. For higher frequency accidents, the more restrictive dose limits in Regulatory Guide 1.183 (Reference 6) and NUREG-0800 were used to ensure that the accident doses were acceptable from an overall risk perspective. Where appropriate, the accident doses are expressed as a total effective dose equivalent (TEDE), consistent with 10 CFR 50.34. The TEDE consists of the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The CEDE is determined using dose conversion factors in Federal Guidance Report 11 (US EPA, 1993). The DDE is taken as the same as the effective dose equivalent from external exposure and the dose conversions in Federal Guidance Report 12 (US EPA, 1993a) are applied.

The accident dose evaluations were performed using 0.5 percentile direction dependent atmospheric dispersion (*X*/Q) values for the EAB and LPZ which are based on onsite meteorological data (Section 2.7). The 0.5 percentile direction dependent *X*/Q values were used instead of the less conservative (more realistic) 50^{th} percentile values normally applied in environmental report evaluations for two reasons. Firstly, use of the 0.5 percentile *X*/Q values provides more conservative offsite dose results. Secondly, the use of the 0.5 percentile *X*/Q values allows the dose evaluation results to be used in the safety analysis report which requires the use of more conservative site *X*/Q values. The site specific *X*/Q values are presented in Table 2.7-115 (EAB) and Table 2.7-116 (LPZ). The accident dose estimates were performed using *X*/Q and activity releases for the following intervals:

Exclusion Area Boundary (EAB)

- 0 to 2 hours (any two-hour period with the greatest EAB doses is used for proposed plants that utilize the Alternate Source Term methodology),

Low Population Zone (LPZ)

- 0 to 8 hours
- 8 to 24 hours
- 1 to 4 days
- 4 to 30 days

7.1.3 Source Terms

Time-dependent activities released to the environs were used in the dose estimates, except for the ABWR design basis LOCA. These activities are based on the analyses used to support the reactor vendor's standard safety analysis reports. The released activities account for the reactor core source term and accident mitigation features in the reactor vendor's standard plant designs for certified reactor designs, or as specified by the reactor vendor for non-certified reactor designs. The Advanced BWR[1] (ABWR) source term and releases are based on TID-14844. The

[1] The NRC certified the ABWR design in 1997 (10 CFR Part 52, Appendix A).

AP1000[2] PWR source term and accident analyses approaches are based on the AST methodology in accordance with Regulatory Guide 1.183. The International Reactor Innovative And Secure (IRIS) advanced reactor source term information is preliminary, and based on vendor information the AP600/AP1000 LOCA source terms and releases are expected to bound the worst-case accident release for this advanced reactor concept.

The advanced gas reactor designs (Gas Turbine – Modular Helium Reactor (GT-MHR) and Pebble Bed Modular Reactor (PBMR)) use mechanistic accident source terms and postulate relatively small environmental releases compared to the water-cooled reactor technologies. The light-water-cooled, heavy-water moderated, Advanced CANDU Reactor, ACR-700[3], design uses a non-mechanistic approach based on TID-14844. The source terms and activity releases to the environment are specified by the reactor vendors for these reactor types. Of these advanced reactor designs, the ACR-700 was judged to have the most limiting DBA release.

7.1.4 Postulated Accidents

This section identifies the DBAs, the resultant activity release paths, the important accident parameters and assumptions, and the credited mitigation measures used in the offsite dose estimates. A summary of the accident doses and the associated NRC dose limit guidelines are provided in Table 7.1-1.

7.1.4.1 Main Steam Line Break Outside Containment (AP1000)

The bounding AP1000 main steam line break for offsite radiological dose consequences occurs outside containment. The AP1000 is designed so that only one steam generator experiences an uncontrolled blowdown even if one of the main steam line isolation valves fails to close. Feedwater is isolated after rupture, and the faulted generator dries out. The secondary side inventory of the faulted steam generator is assumed to be released to the environs along with the entire amount of iodine and alkali metals contained in the secondary side coolant.

The reactor is assumed to be cooled by steaming down the intact steam generator. Activity in the secondary side coolant and primary to secondary side leakage contributes to releases to the environment from the intact generator. During the event, primary to secondary side leakage is assumed to increase from the Technical Specification limit of 150 gpd per steam generator to 500 gpd (175 lbm/hour) per steam generator for the intact and faulted steam generators.

The alkali metals and iodines are the only significant nuclides released during a main steam line break. Noble gases are also released; however, there would be no significant accumulations of the noble gases in the steam generators prior to the accident since they are rapidly released during normal service. Noble gases released during the accident would primarily be due to the increase in primary to secondary side leakage assumed during the event. Reactor coolant leakage to the intact steam generator would mix with the existing inventory and increase the secondary side concentrations. This effect would normally be offset by alkali and iodine partitioning in the generator. However, for conservatism, the calculated activity release assumes

[2] The AP1000 design was submitted to the NRC for certification review in March 2002; the NRC review is in progress. The AP1000 standard plant design is based closely on the AP600 design that received NRC certification in December 1999.

[3] AECL have requested the NRC to conduct a pre-application review of the ACR-700 design in June 2002. That review is in progress.

the primary to secondary side activity in the intact generator is also leaked directly to the environment. The calculated doses are based on activity releases that assume:

- Duration of accident - 72 hours
- Steam generator initial mass – 3.03E+5 lbm
- Primary to secondary leak rate – 175 lb/hour in each generator
- Steam generator initial iodine and alkali metal activities – 10 percent of design basis reactor coolant concentrations at maximum equilibrium conditions
- Reactor coolant alkali activity – 0.25 percent design basis fuel defect inventory
- Reactor coolant noble gas activity – limit of 280 microcurie per gram (μCi/g) dose equivalent Xe-133
- Accident initiated iodine spike – 500 times the fuel release rate that occurs when the reactor coolant equilibrium activity is 1.0 μCi/g dose equivalent Iodine-131
- Pre-existing iodine spike – reactor coolant at 60 μCi/g dose equivalent Iodine-131
- Fuel damage – none

The vendor calculated time-dependent offsite dose releases for a representative site (Reference 2). The GGNS ESP-site-specific doses were calculated using the atmospheric dispersion (*X*/Q) values given in Table 2.7-115 (EAB) and Table 2.7-116 (LPZ). The TEDE doses for the accident-initiated iodine spike are shown in Table 7.1-2. The doses at the EAB and LPZ are a small fraction of the 25 rem TEDE of 10 CFR 50.34. A small fraction is defined, in NUREG-0800 Standard Review Plan 15.0.1 and Regulatory Guide 1.183 (Reference 6), as 10 percent or less of the 25 rem TEDE. The doses for the pre-existing iodine spikes are shown in Table 7.1-3. These doses meet the 25 rem TEDE guideline of 10 CFR 50.34.

7.1.4.2 Main Steam Line Break Outside Containment (ABWR)

The ABWR main steam line break outside containment assumes that the largest steam line instantaneously ruptures outside containment downstream of the outermost isolation valve. The plant is designed to automatically detect the break and initiate isolation of the faulted line. Mass flow would initially be limited by the flow restrictor in the upstream reactor steam nozzle and the remaining flow restrictors in the three unbroken main steam lines feeding the downstream end of the break. Closure of the main steam isolation valves would terminate the mass flow out of the break.

No fuel damage would occur during this event. The only sources of activity are the concentrations present in the reactor coolant and steam before the break. The mass releases used to determine the activity available for release presume maximum instrumentation delays and isolation valve closing times. Iodine and noble gas activities in the water and steam masses discharged through the break are assumed to be released directly to the environs without hold-up or filtration. The calculated doses are based on activity releases that assume:

- Duration of accident – 2 hours
- Main steam isolation valve closure – 5 seconds
- Mass release from break – steam 12,870 kilograms; water 21,950 kilograms

- Reactor coolant maximum equilibrium activity – corresponding to an offgas release rate of 100,000 μCi/s referenced to a 30 minute decay
- Pre-existing iodine spike – corresponding to an offgas release rate of 400,000 μCi/s referenced to a 30 minute decay
- Fuel damage – none

The vendor calculated time-dependent radionuclide releases for a main steam line break outside the containment. The GGNS ESP-site-specific doses were calculated using the *X*/Q values given in Table 2.7-115 (EAB) and Table 2.7-116 (LPZ). The activity released to the environment for the maximum activity and pre-existing iodine spike is shown in Table 7.1-4. The calculated doses for the maximum allowed equilibrium activity at full power operation are shown in Table 7.1-5. For this case, the doses at the EAB and LPZ are a small fraction of the 25 rem TEDE guidelines of 10 CFR 50.34 in accordance with NUREG-0800 Standard Review Plan 15.6.4. The calculated doses for the pre-existing iodine spike are shown in Table 7.1-6. The doses at the EAB and LPZ are within the 25 rem TEDE guideline of 10 CFR 50.34.

7.1.4.3 Reactor Coolant Pump Locked Rotor (AP1000)

The AP1000 locked rotor event is the most severe of several possible decreased reactor coolant flow events. This accident is postulated as an instantaneous seizure of the pump rotor in one of four reactor coolant pumps. The rapid reduction in flow in the faulted loop causes a reactor trip. Heat transfer of the stored energy in the fuel rods to the reactor coolant causes the reactor coolant temperature to increase. The reduced flow also degrades heat transfer between the primary and secondary sides of the steam generators. The event can lead to fuel cladding failure resulting in an increase of activity in the coolant. The rapid expansion of the coolant in the core combined with decreased heat transfer in the steam generator causes the reactor coolant pressure to increase dramatically.

Cool down of the plant by steaming off the steam generators provides a pathway for the release of radioactivity to the environment. In addition, primary side activity, carried over due to leakage in the steam generators, mixes in the secondary side and becomes available for release. The primary side coolant activity inventory increases due to postulated failure of some of the fuel cladding with the consequential release of gap fission product inventory to the coolant. The significant releases from this event are the iodines, alkali metals, and noble gases. No fuel melting occurs. The calculated doses are based on activity releases that assume:

- Duration of accident – 1.5 hours
- Steam released – 6.48E+05 lbm
- Primary/secondary side coolant masses – 3.7E+05 lbm/6.06E+05 lbm
- Primary to secondary leak rate – 350 lbm/hour
- Steam generator initial iodine and alkali metal activities – 10 percent of design basis reactor coolant concentrations at maximum equilibrium conditions
- Reactor coolant alkali activity – 0.25 percent design basis fuel defect inventory
- Reactor coolant noble gas activity – limit of 280 μCi/g dose equivalent Xe-133
- Pre-existing iodine spike – reactor coolant at 60 μCi/g dose equivalent Iodine-131

- Fission product gap activity fractions – Regulatory Guide 1.183 (Reference 6), Regulatory Position C.3.2
- Fraction of fuel gap activity released – 0.16
- Partition coefficients in steam generators – 0.01 for iodines and alkali metals
- Fuel damage – none

The pre-existing iodine spike has little impact since the gap activity released to the primary side becomes the dominant mechanism with respect to offsite dose contributions. The vendor calculated time-dependent offsite dose releases for a representative site. The activity released to the environment is shown in Table 7.1-23. The GGNS ESP-site-specific doses were calculated using the *X*/Q values given in Table 2.7-115 (EAB) and Table 2.7-116 (LPZ). The TEDE doses for the locked rotor accident are shown in Table 7.1-7. These doses are a small fraction of the 25 rem TEDE guidelines of 10 CFR 50.34.

7.1.4.4 Control Rod Ejection (AP1000)

This AP1000 accident is postulated as the gross failure of one control rod mechanism pressure housing resulting in ejection of the control rod cluster assembly and drive shaft. The failure leads to a rapid positive reactivity insertion potentially leading to localized fuel rod damage and significant releases of radioactivity to the reactor coolant.

Two activity release paths contribute to this event. First, the equilibrium activity in the reactor coolant and the activity from the damaged fuel are blown down through the failed pressure housing to the containment atmosphere. The activity can leak to the environment over a relatively long period due to the containment design basis leakage. Decay of radioactivity occurs during hold-up inside containment prior to release to the environs.

The second release path is from the release of steam from the steam generators following reactor trip. With coincident loss of offsite power, additional steam must be released in order to cool down the reactor. The steam generator activity consists of the secondary side equilibrium inventory plus the additional contributions from reactor coolant leaks in the steam generators. The reactor coolant activity levels are increased for this accident since the activity released from the damaged fuel mixes into the coolant prior to being leaked to the steam generators. The iodines, alkali metals, and noble gases are the significant activity sources for this event. Noble gases entering the secondary side are quickly released to the atmosphere via the steam releases through the atmospheric relief valves. A small fraction of the iodines and alkali metals in the flashed part of the leak flow are available for immediate release without benefit of partitioning. The unflashed portion mixes with secondary side fluids where partitioning occurs prior to release as steam.

The dose consequence analyses are performed using guidance in Regulatory Guides 1.77 (Reference 10) and 1.183 (Reference 6). The calculated doses are based on activity releases that assume:

- Duration of accident – 30 days
- Steam released – 1.80E+05 lbm
- Secondary side coolant mass – 6.06E+05 lbm
- Primary to secondary leak rate – 350 lbm/hour

- Containment leak rate – 0.1 percent per day
- Steam generator initial iodine and alkali metal activities – 10 percent of the design basis reactor coolant concentrations at maximum equilibrium conditions
- Reactor coolant alkali metal activity – 0.25 percent design basis fuel defect inventory
- Reactor coolant noble gas activity – limit of 280 μCi/g dose equivalent Xe-133
- Pre-existing iodine spike – reactor coolant at 60 μCi/g dose equivalent Iodine–131
- Fraction of rods with cladding failures – 0.10
- Fission product gap activity fractions:
  - Iodines 0.10
  - Noble gases 0.10
  - Alkali metals 0.12
- Fraction of fuel melting – 0.0025
- Activity released from melted fuel:
  - Iodines 0.5
  - Noble gases 1.0
- Iodine chemical form – per Regulatory Guide 1.183 (Reference 6), Regulatory Position C.3.5
- Containment atmosphere activity removal – elemental 1.7/hour; particulate iodine and alkali metals 0.1/hour
- Partition coefficients in steam generators – 0.01 for iodines and 0.001 for alkali metals

The pre-existing iodine spike has little impact since the gap activity released from the failed cladding and melted fuel become the dominant mechanisms contributing to the radioactivity released from the plant. The activity released to the environment is shown in Table 7.1-24. The vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the *X/Q* values given in Table 2.7-115 (EAB) and Table 2.7-116 (LPZ). The TEDE doses for the control rod ejection accident are shown in Table 7.1-8. These doses are well within the 25 rem TEDE guidelines of 10 CFR 50.34. NUREG-0800 Standard Review Plan 15.4.8 defines "well within" as 25 percent or less of the applicable limits.

7.1.4.5 Rod Drop Accident (ABWR)

The design of the ABWR fine motion control rod drive system includes several new unique features compared with current BWR locking piston control rod drives. The new design precludes the occurrence of rod drop accidents in the ABWR. No radiological consequence analysis is required.

7.1.4.6 Steam Generator Tube Rupture (AP1000)

The AP1000 steam generator tube rupture accident assumes the complete severance of one steam generator tube. The accident causes an increase in the secondary side activity due to reactor coolant flow through the ruptured tube. With the loss of offsite power, contaminated steam is released from the secondary system due to turbine trip and dumping of steam via the atmospheric relief valves. Steam dump (and retention of activity) to the condenser is precluded due to assumption of loss of offsite power. The release of radioactivity depends on the primary to secondary leakage rate, the flow to the faulted steam generator from the ruptured tube, the percentage of defective fuel in the core, and the duration/amount of steam released from the steam generators.

The radioiodines, alkali metals, and noble gases are the significant nuclide groups released during a steam generator tube rupture accident. Multiple release paths are analyzed for the tube rupture accident. The noble gases in the reactor coolant enter the ruptured steam generator and are available for immediate release to the environment. In the intact loop, iodines and alkali metals leaked to the secondary side during the accident are partitioned as the intact steam generator is steamed down until switchover to the residual heat removal system occurs. In the ruptured steam generator, some of the reactor coolant flowing through the tube break flashes to steam while the unflashed portion mixes with the secondary side inventory. Iodines and alkali metals in the flashed fluid are not partitioned during steam releases while activity in the secondary side of the faulted generator is partitioned prior to release as steam. The calculated doses are based on activity releases that assume:

- Duration of accident – 24 hours
- Total flow through ruptured tube – 3.85E+05 lbm
- Steam release from faulted steam generator – 3.32E+05 pound mass
- Steam released from the intact generator – 1.42E+06 pound mass
- Steam release duration – 13.2 hours
- Primary/secondary side initial coolant masses – 3.8E+05 lbm/3.7E+05 lbm
- Primary to secondary leak rate – 175 lbm/hour in the intact steam generator
- Reactor coolant noble gas activity – limit of 280 μCi/g dose equivalent Xe-133
- Reactor coolant alkali activity – 0.25 percent design basis fuel defect inventory
- Steam generator initial iodine and alkali metal activities – 10 percent of design basis reactor coolant concentrations at maximum equilibrium conditions
- Pre-existing iodine spike – reactor coolant at 60 μCi/g dose equivalent Iodine-131
- Accident initiated iodine spike – 335 times the fuel release rate that occurs when the reactor coolant equilibrium activity is 1.0 μCi/g dose equivalent Iodine-131
- Partition coefficients in steam generators – 0.01 for iodines and alkali metals
- Offsite power and condenser – lost on reactor trip
- Fuel damage – none

The activity released to the environment for an accident initiated iodine spike and a pre-existing iodine spike are given in Table 7.1-25 and Table 7.1-26, respectively. The vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the *X*/Q values given in Table 2.7-115 (EAB) and Table 2.7-116 (LPZ). The TEDE doses for the steam generator tube rupture accident with the accident-initiated iodine spike are shown in Table 7.1-9. The doses at the EAB and LPZ are a small fraction of the 25 rem TEDE guidelines of 10 CFR 50.34 as per NUREG-0800, Standard Review Plan 15.6.3. The pre-existing iodine spike doses are shown in Table 7.1-10. These doses are within the 25 rem TEDE guidelines of 10 CFR 50.34.

7.1.4.7 Failure of Small Lines Carrying Primary Coolant Outside Containment (AP1000)

Small lines carrying reactor coolant outside the AP1000 containment include the reactor coolant system sample line and the chemical and volume control system discharge line to the radwaste system. These lines are not continuously used.

The discharge line flow (about 100 gpm) leaving containment is cooled below 140 degrees F and has been cleaned by the mixed bed demineralizer. The reduced iodine concentration and low flow and temperature make this break non-limiting with respect to offsite dose consequences.

The reactor coolant system sample line break is the more limiting break. This line is postulated to break between the outboard isolation valve and the reactor coolant sample panel. Offsite doses are based on a break flow limited to 130 gpm by flow restrictors with isolation occurring at 30 minutes.

Radioiodines and noble gases are the only significant activities released. The source term is based on an accident initiated iodine spike that increases the iodine release rate from the fuel by a factor of 500 throughout the event. All activity is assumed released to the environment. The calculated doses are based on activity releases that assume:

- Duration of accident – 0.5 hours
- Break flow rate – 130 gpm
- Reactor coolant noble gas activity – limit of 280 μCi/g dose equivalent Xe-133
- Reactor coolant equivalent iodine activity – 1.0 μCi/g dose equivalent Iodine-131
- Accident initiated iodine spike – 500 times the fuel release rate that occurs when the reactor coolant activity is 1.0 μCi/g dose equivalent Iodine–131
- Fuel damage – none

The activity released to the environment for an AP1000 small line break accident is shown in Table 7.1-27. The vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the *X*/Q values given in Table 2.7-115 (EAB) and Table 2.7-116 (LPZ). The TEDE doses for the failure of small lines carrying primary coolant outside containment are shown in Table 7.1-11. These doses are a small fraction of the 25 rem TEDE guidelines of 10 CFR 50.34 as per NUREG-0800, Standard Review Plan 15.6.2.

7.1.4.8 Failure of Small Lines Carrying Primary Coolant Outside of Containment (ABWR)

This event consists of a small steam or liquid line break inside or outside the ABWR primary containment. The bounding event analyzed is a small instrument line break in the reactor building. The break is assumed to proceed for ten minutes before the operator takes steps to isolate the break, scram the reactor, and reduce reactor pressure.

All iodine in the flashed water is assumed to be transported to the environs by the heating, ventilation and air conditioning (HVAC) system without credit for treatment by the standby gas treatment system. All other activities in the reactor water make only small contributions to the offsite dose and are neglected. The calculated doses are based on activity releases that assume:

- Duration of accident – 8 hours
- Standby gas treatment system – not credited
- Reactor building release rate – 200 percent/hour
- Mass of reactor coolant released – 13,610 kilograms
- Mass of fluid flashed to steam – 2,270 kilograms
- Iodine plateout fraction – 0.5
- Reactor coolant equilibrium activity – maximum permitted by technical specifications corresponding to an offgas release rate of 100,000 μCi/s referenced to a 30-minute delay
- Iodine spiking – accident initiated spike
- Fuel damage – none

The vendor calculated the time-dependent radionuclide releases to the environment as shown in Table 7.1-12. These releases were used along with the *X*/Q values given in Table 2.7-115 (EAB) and Table 2.7-116 (LPZ) to determine the offsite doses. The doses for the failure of small lines carrying primary coolant outside containment are shown in Table 7.1-13. These doses are a "small fraction" of the 10 CFR 100 limit. A "small fraction" is defined to be 10% of the limit (e.g., 30 Rem Thyroid and 2.5 Rem Whole Body) in accordance with NUREG-0800, Standard Review Plan 15.6.2.

7.1.4.9 Large Break Loss of Coolant Accident (AP1000)

The core response analysis for the AP1000 demonstrates that the reactor core maintains its integrity for the large break LOCA. However, significant core damage degradation and melting is assumed in this DBA. The assumption of major core damage is intended to challenge various accident mitigation features and provide a conservative basis for calculating offsite doses. The source term used in the analysis is adopted from NUREG-1465 and Regulatory Guide 1.183 (Reference 6) with nuclide inventory determined for a three-region equilibrium cycle core at the end of life.

The activity released consists of the equilibrium activity in the reactor coolant and the activity released from the damaged core. Because the AP1000 is a leak before break design, coolant is assumed to blowdown to the containment for 10 minutes. One half of the iodine and all of the noble gases in the blowdown steam are released to the containment atmosphere.

The core release starts after the 10-minute blow down of reactor coolant. The fuel rod gap activity is released over the next half-hour followed by an in-vessel core melt lasting 1.3 hours. Iodines, alkali metals and noble gases are released during the gap activity release. During the core melt phase, five additional nuclide groups are released including the tellurium group, the noble metals group, the cerium group, and the barium and strontium group.

Activity is released from the containment via the containment purge line at the beginning of the accident. After isolation of the purge line, activity continues to leak from the containment at its design basis leak rate. There is no emergency core cooling leakage activity because the passive core cooling system does not pass coolant outside of the containment. A coincidental loss of offsite power has no impact on the activity release to the environment because of the passive designs for the core cooling and fission product control systems. The calculated doses are based on activity releases that assume:

- Duration of accident – 30 days
- Reactor coolant noble gas activity – limit of 280 μCi/g dose equivalent Xe-133
- Reactor coolant equilibrium iodine activity – 1.0 μCi/g equivalent Iodine-131
- Reactor coolant mass – 3.7E+05 lbm
- Containment purge flow rate – 8,800 cfm for 30 seconds
- Containment leak rate – 0.1 percent per day
- Core activity group release fractions – Regulatory Guide 1.183 (Reference 6), Regulatory Position C.3.2
- Iodine chemical form – Regulatory Guide 1.183, Regulatory Position C.3.5
- Containment airborne elemental iodine removal – 1.7 per hour until decontamination factor (DF) of 200 is reached
- Containment atmosphere particulate removal – 0.43 per hour to 0.72 per hour during first 24 hours

The activity assumed to be released to the environment for an AP1000 loss of coolant accident is shown in Table 7.1-28. The vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the *X*/Q values given in Table 2.7-115 (EAB) and Table 2.7-116 (LPZ). The TEDE doses for the AP1000 large break LOCA accident are shown in Table 7.1-14. Both EAB and LPZ doses meet the dose guideline of 25 rem TEDE in 10 CFR 50.34. The activity released from the core melt phase of the accident is the greatest contributor to the offsite doses. The EAB dose in Table 7.1-14 is given for the two-hour period during which the dose is greatest at this location. The initial two hours of the accident is not the worst two-hour period because of the delays associated with cladding failure and fuel damage.

7.1.4.10 Large Break Loss of Coolant Accident (ABWR)

This ABWR event postulates piping breaks inside containment of varying sizes, types and locations. The break type includes steam and liquid process lines. The emergency core cooling analyses show that the core temperature and pressure transients caused by the breaks are insufficient to cause fuel cladding perforation. Although no fuel damage occurs, conservative assumptions from Regulatory Guide 1.3 are invoked in order to conservatively assess post-accident fission product mitigation systems and the resultant offsite doses. The source term for this accident is based on TID-14844 (Reference 5).

One hundred percent of the core inventory noble gases and 50 percent of the iodines are instantaneously released from the reactor to the drywell at the beginning of the accident. Of the iodines, 50 percent are assumed to be immediately plateout leaving 25 percent of the inventory airborne and available for release. Following the break and depressurization of the reactor, some of the noncondensable fission product products are purged into the suppression pool. The suppression pool is capable of retaining iodine thereby reducing the overall concentration in the primary containment atmosphere.

Post-accident fission products are released from the primary containment via two principal pathways: leakage to the reactor building and leakage along the main steam lines. The leakage to the reactor building is due to the containment penetrations and emergency core cooling equipment leaks. The iodine activity in the reactor building is filtered through the standby gas treatment system prior to release to the environment. The standby gas treatment system is started and begins removing iodine from the reactor building atmosphere 20 minutes after start of the accident. The main steam line leakage is due to leaks past the main steam line isolation valves that close automatically at the beginning of the accident. The primary leakage path is through the drain lines downstream of the outboard isolation valves to the main condenser. A secondary pathway is through the main steam lines to the turbine. Activity reaching the main condenser and the turbine is held up before leaking from the turbine building to the environment. Iodine plateout occurs in the turbine, main condenser, and the steam lines/drain lines. The calculated doses are based on activity releases that assume:

- Duration of accident – 30 days
- Core power level – 4005 MWt (102 percent of design core power of 3926 MWt)
- Fraction of noble iodine and noble gases released – Regulatory Guide 1.3, Regulatory Positions C.1.a and C.1.b.
- Iodine chemical form – Regulatory Guide 1.3, Regulatory Position C.1.a
- Suppression pool iodine decontamination factor – 2.0 for particulate and elemental iodine (includes allowance for suppression pool bypass)
- Primary containment leakage – 0.5 percent/day
- Main steam isolation valve total leakage – 66.1 liters/minute
- Condenser leakage rate – 11.6 percent/day
- Condenser iodine removal:
  - Elemental and particulate iodine 99.7 percent
  - Organic iodine 0.0 percent
- Delay to achieve design negative pressure in reactor building - 20 minutes
- Reactor building leak rate during draw down – 150 percent/hour
- Standby gas system filtration – 97 percent efficiency
- Standby gas system exhaust rate – 50 percent/day

The vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the *X/Q* values given in Table 2.7-115 (EAB) and Table 2.7-116 (LPZ). The activities released to the environment from the reactor and turbine buildings are listed in Table 7.1-15. The doses for the ABWR large break LOCA accident are shown in Table 7.1-16. Since the vendor evaluation of this postulated accident is based on TID-14844 and Regulatory Guide 1.3 methodology, the offsite dose acceptance criteria of 10 CFR 100 is used. The calculated doses meet the dose guidelines of 300 rem thyroid and 75 rem whole body as specified in 10 CFR 100.

7.1.4.11 Large Loss of Coolant Accident (ACR-700)

The limiting design basis event for the ACR-700 is a large LOCA with coincident loss of emergency cooling. In this accident, the heat transport system coolant is discharged into containment via the break. Without emergency core cooling injection, the fuel bundles start to heat up causing the pressure tube to sag and contact the calandria tube. With contact between the pressure tube and calandria, heat is transferred from the fuel channel to the moderator. In such a severe accident, the heavy water in the moderator acts as the heat sink and the heat is transferred to the service water. The integrity of the pressure tube, calandria tube, and the heat transfer system core cooling geometry are maintained.

The activity released during the large LOCA is shown in Table 7.1-17. The GGNS ESP-site-specific doses were calculated using the *X*/Q values given in Table 2.7-115 (EAB) and Table 2.7-116 (LPZ). The TEDE doses for the ACR-700 LOCA accident are shown in Table 7.1-18. The doses meet the dose guidelines of 25 rem TEDE given in 10 CFR 50.34.

7.1.4.12 Fuel Handling Accidents (AP1000)

The AP1000 fuel handling accident (FHA) can occur inside containment or in the fuel handling area of the auxiliary building. The accident postulates dropping a fuel assembly over the core or in the spent fuel pool. The cladding of the fuel rods is assumed breached and the fission products in the fuel rod gaps are released to the reactor refueling cavity water or spent fuel pool. There are numerous design or safety features to prevent this accident. For example, only one fuel assembly is lifted and transported at a time. Fuel racks are located to prevent missiles from reaching the stored fuel. Fuel handling equipment is designed to prevent it from falling on the fuel, and heavy objects cannot be carried over the spent fuel.

Fuel handling operations are performed under water. Fission gases released from damaged fuel bubble up through the water and escape above the refueling cavity water or spent fuel pool surfaces. For FHAs inside containment, the release to the environment can be mitigated by automatically closing the containment purge lines after detection of radioactivity in the containment atmosphere. For accidents in the spent fuel pool, activity is released through the auxiliary building ventilation system to the environment.

The refueling and fuel transfer systems are designed such that the damaged fuel has a minimum depth of 23 feet of water over the fuel. This depth of water provides for effective scrubbing of elemental iodine released from the fuel. Organic iodine and noble gases are not scrubbed and escape.

The offsite doses are analyzed by only crediting the scrubbing of iodine by the refueling water. Hence, fuel handling accidents inside containment and the auxiliary building are treated in the same manner. Cesium iodide, which accounts for about 95 percent of the gap iodine, is nonvolatile and does not readily become airborne after dissolving. This species is assumed to completely dissociate and re-evolve as elemental iodine immediately after damage to the fuel assembly. The calculated doses are based on activity releases that assume:

- Core thermal power – 3,468 MWt (102 percent of design core power of 3400 MWt)
- Decay time after shutdown – 100 hours
- Activity release period – 2 hours
- One of 157 fuel assemblies in the core is completely discharged
- Maximum rod radial peaking factor – 1.65

- Iodine and noble gas fission product gap fractions – Regulatory Guide 1.183 (Reference 6), Regulatory Position C.3.2
- Iodine chemical form – Regulatory Guide 1.183, Regulatory Position C.3.5
- Pool decontamination for iodine – Regulatory Guide 1.183, Appendix B
- Filtration – none

The radioactivity released to the environment is listed in Table 7.1-19. The GGNS ESP-site-specific doses were calculated using the atmospheric dispersion (*X*/Q) values given in Table 2.7-115 (EAB) and Table 2.7-116 (LPZ). The resulting doses at the EAB and LPZ are summarized in Table 7.1-20. The doses are applicable to fuel handling accidents inside containment and in the spent fuel pool in the auxiliary building. The EAB and LPZ doses are well within the 25 rem TEDE guidelines given in 10 CFR 50.34. "Well within" is taken as being 25 percent of the guideline, consistent with the guidance of Regulatory Guide 1.183 (Reference 6) and NUREG-0800, Standard Review Plan 15.7.4.

7.1.4.13 Fuel Handling Accidents (ABWR)

The ABWR fuel handling accident is postulated as failure of the fuel assembly lifting mechanism resulting in the dropping of a fuel assembly on to the reactor core. Fuel rods in the dropped and struck assemblies are damaged releasing radioactive gases to the pool water.

The activity released in the pool water bubbles to the surface and passes to the reactor building atmosphere. The normal ventilation system is isolated, the standby gas treatment system is started, and effluents are released to the environment through this system. The standby gas treatment system is credited with maintaining the reactor building at a negative pressure after 20 minutes. Pool water is credited with removal of elemental iodine released from the failed rods. Guidance from Regulatory Guide 1.25 was used in performance of the analysis. The calculated doses are based on activity releases that assume:

- Core thermal power – 4,005 MWt (102 percent of design core power of 3,926 MWt)
- Decay time after shutdown – 24 hours
- Activity release period from pool – 2 hours
- Total number of fuel rods damaged – 115 in dropped and struck assemblies
- Radial peaking factor – 15
- Fuel rod fission product gap fractions –Regulatory Guide 1.183 (Reference 6), Regulatory Position C.3.2
- Iodine chemical form – Regulatory Guide 1.183, Regulatory Position C.3.5
- Pool decontamination for iodine – Regulatory Guide 1.183, Appendix B
- Delay to achieve design negative pressure in reactor building – 20 minutes
- Standby gas system filtration – 99 percent efficiency
- Dose conversion factors - Regulatory Guide 1.183, Regulatory Position 4.1

The radionuclide inventory in the damaged fuel is listed in Table 7.1-21. The GGNS ESP-site-specific doses were calculated using the *X*/Q values given in Table 2.7-115 (EAB) and Table 2.7-116 (LPZ). The resulting doses at the EAB and LPZ are summarized in Table 7.1-22. The

LPZ dose is bounded by the EAB dose due to the 2-hour release duration and the lower *X/Q* for the LPZ. All activity released from the fuel is assumed to be released during the first two hours after the accident. The EAB and LPZ doses are well within (less than 25 percent of) the 10CFR100 limits (e.g., 75 rem thyroid and 6.3 rem whole body).

7.1.5 References

1. 23A6100, GE ABWR Standard Safety Analysis Report.
2. Westinghouse AP1000 Design Control Document, Volume 2, Tier 2 Material, Revision 2.
3. U.S. Nuclear Regulatory Commission (NRC), Draft 1996, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Washington, DC.
4. U.S. Nuclear Regulatory Commission (NRC), 1999, Environmental Standard Review Plan, NUREG-1555, Washington, DC.
5. Technical Information Document (TID) 14844, Calculation of Distance Factors for Power And Test Reactor Sites, J.J. DiNunno et al., USAEC TID-14844, U.S. Atomic Energy Commission (now USNRC), March 23, 1962.
6. U.S. Nuclear Regulatory Commission (NRC), July 2000 (draft issued as DG-1081), Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors, Regulatory Guide 1.183, Washington, DC.
7. AECL, Assessment Document, Two-Unit ACR-700, Plant Parameters Envelope for Early Site Permit Application, Advanced Reactor Technology Study, No. 115-01250-050-002, Revision 0
8. U.S. Nuclear Regulatory Commission (NRC), 1974, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss Of Coolant Accident for Boiling Water Reactors, Regulatory Guide 1.3, Revision 2, Washington, DC.
9. U.S. Nuclear Regulatory Commission (NRC), 1972, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, Regulatory Guide 1.25, Washington, DC.
10. U.S. Nuclear Regulatory Commission (NRC), May 1974, Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors, Regulatory Guide 1.77, Washington, DC.

TABLE 7.1-1

COMPARISON OF REACTOR TYPES FOR LIMITING OFF-SITE DOSE CONSEQUENCES

PART A, EXCEPT ABWR

Accident	Reactor Type	EAB Dose TEDE (rem)	LPZ Dose TEDE (rem)	Guideline[1] TEDE (rem)
Main Steam Line Break				
Accident-initiated Iodine Spike	AP1000	0.79	0.79	2.5
Pre-existing Iodine Spike	AP1000	0.69	0.21	25
Reactor Coolant Pump Locked Rotor				
	AP1000	2.5	0.3	2.5
Control Rod Ejection Accident				
	AP1000	2.98	0.84	6.3
Steam Generator Tube Rupture				
Accident-initiated Iodine Spike	AP1000	1.49	0.12	2.5
Pre-existing Iodine Spike	AP1000	2.98	0.17	25
Small Line Break				
	AP1000	1.3	0.1	2.5
Loss of Coolant Accident				
	AP1000	24.5	4.94	25
	ACR-700	6.3	4.1	25
Fuel Handling Accident				
	AP1000	2.4	0.3	6.3

NOTES:

1. 25 rem is the TEDE guideline from Regulatory Guide 1.183. NUREG-0800 Chapter 15 specifies a guideline of "a small fraction" of the limit, defined as 10 percent or less (2.5 rem), and "well within" the guidelines for other events defined as 25 percent or less (6.3 rem).

TABLE 7.1-1 (Continued)

PART B, ABWR

Accident	Affected Organ	EAB Dose (rem)	LPZ Dose (rem)	Guideline[1] (rem)
Main Steam Line Break				
Max Equilibrium Iodine Activity	Thyroid	1.11	1.24E-01	30
	Whole Body	1.7E-02	1.91E-03	2.5
Pre-existing Iodine Spike	Thyroid	22.2	2.48	300
	Whole Body	3.4E-01	3.81E-02	25
Control Rod Drop Accident	Thyroid	Negligible	Negligible	75
	Whole Body	Negligible	Negligible	6
Small Line Break	Thyroid	2.04	0.23	30
	Whole Body	0.027	0.003	2.5
Loss of Coolant Accident	Thyroid	82.5	200	300
	Whole Body	1.78	2.58	25
Fuel Handling Accident	Thyroid	9.78	1.10	75
	Whole Body	0.41	0.05	6

NOTES:

1. ABWR LOCA guideline based on 10CFR100 limits due to use of TID-14844 source term. NUREG-0800 Chapter 15 specifies a guideline of "a small fraction" of the limit, defined as 10 percent or less, and "well within" the guidelines for other events defined as 25 percent or less.

TABLE 7.1-2

AP1000 MAIN STEAM LINE BREAK - ACCIDENT-INITIATED IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	0.79	----*
0 to 8 hour	----*	0.32
8 to 24 hour	----*	0.20
24 to 96 hour	----*	0.27
96 to 720 hours	----*	----*
TOTAL	0.79	0.79

NOTES:

* Dose not applicable

TABLE 7.1-3

AP1000 MAIN STEAM LINE BREAK - PRE-EXISTING IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	0.69	----*
0 to 8 hour	----*	0.12
8 to 24 hour	----*	0.04
24 to 96 hour	----*	0.06
96 to 720 hours	----*	----*
TOTAL	0.69	0.22

NOTES:

* Dose not applicable

TABLE 7.1-4

ABWR MAIN STEAM LINE BREAK OUTSIDE CONTAINMENT

Isotope	Maximum Equilibrium Value for Full Power Operation Megabecquerel Released 0 to 2 hour	Pre-existing Iodine Spike Megabecquerel Released 0 to 2 hour
I-131	7.29E+04	1.46E+06
I-132	7.10E+05	1.42E+07
I-133	5.00E+05	9.99E+06
I-134	1.40E+06	2.79E+07
I-135	7.29E+05	1.46E+07
Total Halogens	3.41E+06	6.81E+07
KR-83M	4.07E+02	2.44E+03
KR-85M	7.18E+02	4.29E+03
KR-85	2.26E+00	1.36E+01
KR-87	2.44E+03	1.47E+04
KR-88	2.46E+03	1.48E+04
KR-89	9.88E+03	5.92E+04
KR-90	2.55E+03	1.55E+04
XE-131M	1.76E+00	1.06E+01
XE-133M	3.39E+01	2.04E+02
XE-133	9.47E+02	5.70E+03
XE-135M	2.89E+03	1.74E+04
XE-135	2.70E+03	1.62E+04
XE-137	1.23E+04	7.40E+04
XE-138	9.44E+03	5.66E+04
XE-139	4.33E+03	2.59E+04
TOTAL NOBLE GASES	5.11E+04	3.07E+05

TABLE 7.1-5

ABWR MAIN STEAM LINE BREAK OUTSIDE CONTAINMENT - MAXIMUM EQUILIBRIUM VALUE FOR FULL POWER OPERATION

Time	Exclusion Area Boundary Dose (rem)		Low Population Zone Dose (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
0 to 2 hour	1.11	1.70E-02	----*	----*
0 to 8 hour	----*	----*	1.24E-01	1.91E-03
8 to 24 hour	----*	----*	----*	----*
24 to 96 hour	----*	----*	----*	----*
96 to 720 hours	----*	----*	----*	----*
TOTAL	1.11	1.70E-02	1.24E-01	1.91E-03

NOTES:

* Dose not applicable

TABLE 7.1-6

ABWR MAIN STEAM LINE BREAK OUTSIDE CONTAINMENT - PRE-EXISTING IODINE SPIKE

Time	Exclusion Area Boundary Dose (rem)		Low Population Zone Dose (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
0 to 2 hour	2.22E+01	3.4E-01	----*	----*
0 to 8 hour	----*	----*	2.48E+00	3.81E-02
8 to 24 hour	----*	----*	----*	----*
24 to 96 hour	----*	----*	----*	----*
96 to 720 hours	----*	----*	----*	----*
TOTAL	2.22E+01	3.4E-01	2.48E+00	3.81E-02

NOTES:

* Dose not applicable

TABLE 7.1-7

AP1000 LOCKED ROTOR ACCIDENT – PRE-EXISTING IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	2.5	----*
0 to 8 hour	----*	0.3
8 to 24 hour	----*	----*
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
TOTAL	2.5	0.3

NOTES:

* Dose not applicable

TABLE 7.1-8

AP1000 CONTROL ROD EJECTION ACCIDENT - PRE-EXISTING IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	2.98	----*
0 to 8 hour	----*	0.69
8 to 24 hour	----*	0.12
24 to 96 hour	----*	0.02
96 to 720 hours	----*	0.01
TOTAL	2.98	0.84

NOTES:

* Dose not applicable

TABLE 7.1-9

AP1000 STEAM GENERATOR TUBE RUPTURE - ACCIDENT-INITIATED IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	1.49	----*
0 to 8 hour	----*	0.09
8 to 24 hour	----*	0.03
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
TOTAL	1.49	0.12

NOTES:

* Dose not applicable

TABLE 7.1-10

AP1000 STEAM GENERATOR TUBE RUPTURE - PRE-EXISTING IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	2.98	----*
0 to 8 hour	----*	0.16
8 to 24 hour	----*	0.01
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
TOTAL	2.98	0.17

NOTES:

* Dose not applicable

TABLE 7.1-11

AP1000 SMALL LINE BREAK ACCIDENT, 0 TO 0.5 HOUR DURATION - ACCIDENT-INITIATED IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	1.3	----*
0 to 8 hour	----*	0.1
8 to 24 hour	----*	----*
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
TOTAL	1.3	0.1

NOTES:

* Dose not applicable

TABLE 7.1-12

ABWR SMALL LINE BREAK OUTSIDE CONTAINMENT - ACTIVITY RELEASED TO ENVIRONMENT

Time	Release from Break (directly to Environment) (MBq)
0 to 2 hour	4.784E+05
0 to 8 hour	4.185E+06
8 to 24 hour	3.288E+06
24 to 96 hour	7.171E+06
96 to 720 hours	4.482E+06
TOTAL	1.960E+07

TABLE 7.1-13

ABWR SMALL LINE BREAK OUTSIDE CONTAINMENT

Time	Exclusion Area Boundary Dose (rem)		Low Population Zone Dose (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
0 to 2 hour	2.04	2.68E-02	----*	----*
0 to 8 hour	----*	----*	2.29E-01	3.00E-03
8 to 24 hour	----*	----*	----*	----*
24 to 96 hour	----*	----*	----*	----*
96 to 720 hours	----*	----*	----*	----*
TOTAL	2.04	2.68E-02	2.29E-01	3.00E-03

NOTES:

* Dose not applicable

TABLE 7.1-14

AP1000 DESIGN BASIS LOSS OF COOLANT ACCIDENT

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	24.6	----*
0 to 8 hour	----*	4.54
8 to 24 hour	----*	0.16
24 to 96 hour	----*	0.13
96 to 720 hours	----*	0.11
TOTAL	24.6	4.94

NOTES:

1. *Dose not applicable
2. Two-hour period with greatest EAB dose shown. LOCA based on Regulatory Guide 1.183.

TABLE 7.1-15

ABWR LOCA CURIES RELEASED TO ENVIRONMENT BY TIME INTERVAL

Isotope	0 to 2 hours	0 to 8 hours	8 to 24 hours	1 to 4 days	4 to 30 days
I-131	2.60E+02	3.74E+02	9.23E+02	8.70E+03	6.22E+04
I-132	3.52E+02	3.85E+02	3.24E+01	0	0
I-133	5.41E+02	7.43E+02	1.18E+03	3.32E+03	6.76E+02
I-134	5.14E+02	5.15E+02	0	0	0
I-135	5.14E+02	6.47E+02	3.32E+02	1.68E+02	0
Kr-83m	3.26E+02	9.00E+02	4.32E+01	0	0
Kr-85m	8.44E+02	3.74E+03	4.36E+03	7.03E+02	0
Kr-85	4.09E+01	3.49E+02	2.19E+03	2.18E+04	2.86E+05
Kr-87	1.20E+03	2.17E+03	8.92E+01	2.70E+00	0
Kr-88	2.12E+03	7.14E+03	3.43E+03	2.97E+02	0
Kr-89	1.81E+02	1.81E+02	0	0	0
Xe-131m	2.13E+01	1.72E+02	1.12E+03	9.52E+03	6.22E+04
Xe-133m	3.00E+02	2.48E+03	1.38E+04	7.59E+04	7.27E+04
Xe-133	7.63E+03	6.11E+04	3.77E+05	2.78E+06	8.41E+06
Xe-135m	4.87E+02	4.87E+02	0	0	0
Xe-135	9.26E+02	5.51E+03	1.52E+04	1.17E+04	0
Xe-137	5.14E+02	5.14E+02	0	0	0
Xe-138	2.00E+03	2.00E+03	0	0	0

TABLE 7.1-16

ABWR DESIGN BASIS LOSS OF COOLANT ACCIDENT

Time	Exclusion Area Boundary Dose		Low Population Zone Dose	
	Thyroid (rem)	Whole Body (rem)	Thyroid (rem)	Whole Body (rem)
0 to 2 hour	8.25E+01	1.78	----*	----*
0 to 8 hour	----*	----*	1.33E+01	4.27E-01
8 to 24 hour	----*	----*	9.93	3.97E-01
24 to 96 hour	----*	----*	5.46E+01	7.60E-01
96 to 720 hours	----*	----*	1.22E+02	9.97E-01
TOTAL	82.5	1.78	2.00E+02	2.58

NOTES:

1. *Dose not applicable
2. LOCA based on Regulatory Guide 1.3 and TID-14844.

TABLE 7.1-17

ACR-700 DESIGN BASIS LARGE LOCA - CURIES RELEASED TO ENVIRONMENT BY INTERVAL

Isotope	0-2 hour	2 to 8 hr	8 to 24 hrs	1 to 4 days	4 to 30 days
I-131	57	170	440	900	3460
I-132	63	120	140	69	69
I-133	117	330	750	830	910
I-134	66	83	83	41	41
I-135	101	250	430	270	270
Kr 83-m	2094	3600	3900	2000	2000
Kr 85-m	5702	13000	19600	10700	10700
Kr 85	45	140	360	820	6900
Kr 87	7977	11600	12000	6000	6000
Kr 88	14474	28900	36700	18700	18700
Kr 89	864	870	860	430	430
Xe 131-m	252	800	2000	4200	19700
Xe133-m	1397	4100	10200	16400	26600
Xe-133	45632	135400	350900	679600	1982700
Xe135-m	1784	1800	1800	900	900
Xe 135	3738	9700	18600	13100	13200
Xe 137	1894	1900	1900	950	950
Xe 138	6774	6800	6800	3400	3400

TABLE 7.1-18

ACR-700 LARGE LOSS OF COOLANT ACCIDENT

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	6.3	0.7
2 to 8 hour	----*	1.3
8 to 24 hour	----*	1.2
24 to 96 hour	----*	0.5
96 to 720 hours	----*	0.4
TOTAL	6.3	4.1

NOTES:

* Dose not applicable

TABLE 7.1-19

AP1000 FUEL HANDLING ACCIDENT - CURIES RELEASED TO ENVIRONMENT

Isotope	Release 0-2 hrs
I-130	3.52E-02
I-131	2.90E+02
I-132	1.54E+02
I-133	1.91E+01
I-134	0
I-135	1.36E-02
Kr-83m	0
Kr-85m	2.68E-03
Kr-85	1.10E+03
Kr-87	0
Kr-88	0
Kr-89	0
Xe-131m	5.36E+02
Xe-133m	1.29E+03
Xe-133	6.94E+04
Xe-135m	4.37E-01
Xe-135	1.32E+02
Xe-137	0
Xe-138	0

TABLE 7.1-20

AP1000 FUEL HANDLING ACCIDENT

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	2.4	----*
0 to 8 hour	----*	0.3
8 to 24 hour	----*	----*
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
TOTAL	2.4	0.3

NOTES:

* Dose not applicable

TABLE 7.1-21

ABWR FUEL HANDLING ACCIDENT - CURIES RELEASED TO ENVIRONMENT

Isotope	Release (Ci)
I131	1.458E+01
I132	1.176E+01
I133	9.430E+00
I134	5.147E-07
I135	1.549E+00
KR 83M	5.563E+00
KR 85	2.568E+02
KR 85M	7.084E+01
KR 87	1.100E-02
KR 88	2.051E+01
XE129M	4.103E-05
XE131M	6.726E+01
XE133	2.272E+04
XE133M	8.907E+02
XE135	5.205E+03
XE135M	2.709E+02

TABLE 7.1-22

ABWR FUEL HANDLING ACCIDENT

Time	Exclusion Area Boundary Dose (rem)		Low Population Zone Dose (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
0 to 2 hour	9.78	0.41	----*	----*
0 to 8 hour	----*	----*	1.10	0.05
8 to 24 hour	----*	----*	----*	----*
24 to 96 hour	----*	----*	----*	----*
96 to 720 hours	----*	----*	----*	----*
TOTAL	9.78	0.41	1.10	0.05

NOTES:

1. Activity is based on a 24-hour shutdown before fuel movement begins.

TABLE 7.1-23

AP1000 LOCKED ROTOR ACCIDENT - CURIES RELEASED TO ENVIRONMENT

Isotope	0 to 1.5 hrs
I-130	4.15E+00
I-131	1.83E+02
I-132	1.33E+02
I-133	2.31E+02
I-134	1.44E+02
I-135	2.04E+02
Kr-85m	4.09E+02
Kr-85	3.77E+01
Kr-87	6.05E+02
Kr-88	1.05E+03
Xe-131m	1.87E+01
Xe-133m	1.02E+02
Xe-133	3.33E+03
Xe-135m	1.63E+02
Xe-135	8.01E+02
Xe-138	6.48E+02
Rb-86	6.69E-02
Cs-134	5.83E+00
Cs-136	1.85E+00
Cs-137	3.42E+00
Cs-138	3.05E+01

TABLE 7.1-24

AP1000 CONTROL ROD EJECTION ACCIDENT - CURIES RELEASED TO ENVIRONMENT BY INTERVAL – PRE-EXISTING IODINE SPIKE

Isotope	0 to 2 hrs	2 to 8 hrs	8 to 24 hrs	24 to 96 hrs	96 to 720 hrs
I-130	5.93E+00	7.28E+00	4.32E+00	4.06E-01	5.88E-04
I-131	1.64E+02	2.45E+02	2.31E+02	6.20E+01	3.33E+01
I-132	1.90E+02	9.94E+01	9.85E+00	1.65E-02	0
I-133	3.29E+02	4.40E+02	3.18E+02	4.56E+01	4.81E-01
I-134	2.18E+02	2.85E+01	1.37E-01	8.96E-08	0
I-135	2.91E+02	2.97E+02	1.19E+02	4.79E+00	1.46E-04
Kr-85m	2.85E+02	6.48E+01	3.87E+01	3.53E+00	5.01E-05
Kr-85	1.24E+01	5.60E+00	1.49E+01	6.70E+01	5.71E+02
Kr-87	4.86E+02	2.60E+01	1.03E+00	1.67E-04	0
Kr-88	7.49E+02	1.18E+02	3.49E+01	7.18E-01	1.68E-08
Xe-131m	1.22E+01	5.46E+00	1.42E+01	5.72E+01	2.31E+02
Xe-133m	6.62E+01	2.81E+01	6.49E+01	1.69E+02	1.06E+02
Xe-133	2.18E+03	9.58E+02	2.40E+03	8.53E+03	1.68E+04
Xe-135m	2.18E+02	5.30E-02	4.33E-09	0	0
Xe-135	5.39E+02	1.72E+02	2.09E+02	8.69E+01	3.58E-01
Xe-138	8.89E+02	1.38E-01	3.19E-09	0	0
Rb-86	3.70E-01	7.27E-01	6.96E-01	1.73E-01	6.79E-02
Cs-134	3.15E+01	6.22E+01	6.03E+01	1.55E+01	1.03E+01
Cs-136	8.98E+00	1.75E+01	1.67E+01	4.10E+00	1.31E+00
Cs-137	1.83E+01	3.62E+01	3.51E+01	9.04E+00	6.05E+00
Cs-138	1.13E+02	7.05E+00	1.68E-03	0	0

TABLE 7.1-25

AP1000 STEAM GENERATOR TUBE RUPTURE ACCIDENT - CURIES RELEASED TO ENVIRONMENT BY INTERVAL - ACCIDENT INITIATED IODINE SPIKE

Isotope	0 to 2 hrs	2 to 8 hrs	8 to 24 hrs
I-130	7.30E-02	1.19E-02	3.13E-02
I-131	4.90E+00	1.15E+00	3.55E+00
I-132	5.79E+00	1.75E-01	2.30E-01
I-133	8.79E+00	1.68E+00	4.73E+00
I-134	1.12E+00	1.18E-03	5.21E-04
I-135	5.15E+00	6.01E-01	1.36E+00
Kr-85m	5.67E+01	1.91E+01	2.50E-02
Kr-85	2.25E+02	1.07E+02	4.44E-01
Kr-87	2.46E+01	3.56E+00	3.02E-04
Kr-88	9.44E+01	2.61E+01	1.80E-02
Xe-131 m	1.02E+02	4.82E+01	1.96E-01
Xe-133m	1.26E+02	5.83E+01	2.19E-01
Xe-133	9.37E+03	4.41E+03	1.75E+01
Xe-135m	3.61E+00	5.78E-03	0
Xe-135	2.51E+02	1.00E+02	2.35E-01
Xe-138	4.78E+00	4.99E-03	0
Rb-86	*	*	*
Cs-134	1.65E+00	6.35E-02	2.27E-01
Cs-136	2.45E+00	9.30E-02	3.30E-01
Cs-137	1.19E+00	4.58E-02	1.64E-01
Cs-138	5.71E-01	3.07E-06	6.00E-07

Note: * = Rb-86 contribution considered negligible for this accident.

TABLE 7.1-26

AP1000 STEAM GENERATOR TUBE RUPTURE ACCIDENT - CURIES RELEASED TO ENVIRONMENT BY INTERVAL – PRE-EXISTING IODINE SPIKE

Isotope	0 to 2 hrs	2 to 8 hrs	8 to 24 hrs
I-130	1.81E+00	6.12E-02	2.90E-01
I-131	1.22E+02	5.97E+00	3.32E+01
I-132	1.43E+02	8.53E-01	2.08E+00
I-133	2.19E+02	8.68E+00	4.41E+01
I-134	2.78E+01	5.16E-03	4.57E-03
I-135	1.28E+02	3.06E+00	1.26E+01
Kr-85m	5.67E+01	1.91E+01	2.50E-02
Kr-85	2.25E+02	1.07E+02	4.44E-01
Kr-87	2.46E+01	3.56E+00	3.02E-04
Kr-88	9.44E+01	2.61E+01	1.80E-02
Xe-131m	1.02E+02	4.82E+01	1.96E-01
Xe-133m	1.26E+02	5.83E+01	2.19E-01
Xe-133	9.37E+03	4.41E+03	1.75E+01
Xe-135m	3.61E+00	5.78E-03	0
Xe-135	2.51E+02	1.00E+02	2.35E-01
Xe-138	4.78E+00	4.99E-03	0
Rb-86	*	*	*
Cs-134	1.65E+00	6.35E-02	2.27E-01
Cs-136	2.45E+00	9.30E-02	3.30E-01
Cs-137	1.19E+00	4.58E-02	1.64E-01
Cs-138	5.71E-01	3.07E-06	6.00E-07

Note: * = Rb-86 contribution considered negligible for this accident.

TABLE 7.1-27

AP1000 SMALL LINE BREAK ACCIDENT - CURIES RELEASED TO ENVIRONMENT - ACCIDENT INITIATED IODINE SPIKE

Isotope	0 to 0.5 hr
I-130	1.90E+00
I-131	9.26E+01
I-132	3.49E+02
I-133	2.01E+02
I-134	1.58E+02
I-135	1.68E+02
Kr-85m	1.24E+01
Kr-85	4.40E+01
Kr-87	7.00E+00
Kr-88	2.21E+01
Xe-131m	1.99E+1
Xe-133m	2.50E+01
Xe-133	1.84E+02
Xe-135m	2.60E+00
Xe-135	5.20E+01
Xe-138	3.60E+00
Cs-134	4.20E+00
Cs-136	6.20E+00
Cs-137	3.00E+00
Cs-138	2.20E+00

TABLE 7.1-28

AP1000 DESIGN BASIS LOSS OF COOLANT ACCIDENT - CURIES RELEASED TO ENVIRONMENT BY INTERVAL

Isotope	0 to 1 hrs	2 to 3 hrs	0 to 8 hrs	8 to 24 hrs	24 to 96 hrs	96 to 720 hrs
Halogen Group						
I-130	5.62E+00	4.92E+01	7.80E+01	2.96E+00	1.11 E+00	1.99E-02
I-131	1.54E+02	1.44E+03	2.36E+03	1.56E+02	3.74E+02	1.12E+03
I-132	1.79E+02	1.18E+03	1.67E+03	7.64E+00	2.29E-02	0
I-133	3.11E+02	2.80E+03	4.51E+03	2.16E+02	1.63E+02	1.62E+01
I-134	1.96E+02	7.51E+02	1.02E+03	1.26E-01	1.07E-07	0
I-135	2.75E+02	2.27E+03	3.50E+03	8.31E+01	9.55E+00	4.95E-03
Noble Gas Group						
Kr-85m	6.74E+01	1.31 E+03	3.77E+03	1.87E+03	1.71E+02	2.43E-03
Kr-85	3.08E+00	7.32E+01	2.96E+02	7.05E+02	3.17E+03	2.70E+04
Kr-87	9.54E+01	1.14E+03	1.94E+03	4.97E+01	8.11E-03	0
Kr-88	1.70E+02	2.95E+03	7.26E+03	1.70E+03	3.49E+01	8.16E-07
Xe-131m	3.07E+00	7.28E+01	2.94E+02	6.79E+02	2.74E+03	1.11E+04
Xe-133m	1.68E+01	3.92E+02	1.54E+03	3.15E+03	8.21E+03	5.15E+03
Xe-133	5.49E+02	1.30E+04	5.19E+04	1.16E+05	4.11E+05	8.10E+05
Xe-135m	1.44E+01	2.14E+01	3.59E+01	2.14E-07	0	0
Xe-135	1.32E+02	2.85E+03	9.64E+03	1.01 E+04	4.21E+03	1.73E+01
Xe-138	5.31E+01	6.69E+01	1.20E+02	1.58E-07	0	0
Alkali Metal Group						
Rb-86	3.32E-01	2.61E+00	4.26E+00	9.37E-02	2.03E-03	1.05E-02
Cs-134	2.81E+01	2.22E+02	3.63E+02	8.06E+00	1.88E-01	1.59E+00
Cs-136	8.01E+00	6.30E+01	1.03E+02	2.25E+00	4.72E-02	2.03E-01
Cs-137	1.64E+01	1.29E+02	2.11E+02	4.70E+00	1.10E-01	9.39E-01
Cs-138	1.06E+02	2.06E+02	3.19E+02	6.92E-04	0	0

TABLE 7.1-28 (Continued)

Isotope	0 to 1 hrs	2 to 3 hrs	0 to 8 hrs	8 to 24 hrs	24 to 96 hrs	96 to 720 hrs
Tellurium Group						
Sr-89	3.23E+00	7.56E+01	1.19E+02	2.87E+00	6.54E-02	4.60E-01
Sr-90	2.78E-01	6.52E+00	1.03E+01	2.48E-01	5.82E-03	4.97E-02
Sr-91	3.77E+00	8.14E+01	1.22E+02	1.74E+00	2.76E-03	1.44E-05
Sr-92	3.45E+00	6.13E+01	8.30E+01	3.26E-01	1.06E-05	0
Sb-127	8.55E-01	1.98E+01	3.11E+01	7.13E-01	1.16E-02	1.60E-02
Sb-129	2.25E+00	4.43E+01	6.28E+01	4.83E-01	1.01E-04	1.00E-09
Te-127m	1.10E-01	2.58E+00	4.06E+00	9.83E-02	2.27E-03	1.77E-02
Te-127	7.99E-01	1.72E+01	2.57E+01	3.65E-01	5.63E-04	2.72E-06
Te-129m	3.76E-01	8.80E+00	1.38E+01	3.33E-01	7.47E-03	4.79E-02
Te-129	1.50E+00	1.89E+01	2.32E+01	8.54E-03	7.27E-10	0
Te-131m	1.15E+00	2.62E+01	4.05E+01	8.29E-01	6.86E-03	1.60E-03
Te-132	1.14E+01	2.65E+02	4.15E+02	9.42E+00	1.44E-01	1.60E-01
Ba-139	3.83E+00	5.30E+01	6.63E+01	4.73E-02	2.03E-08	0
Ba-140	5.71E+00	1.33E+02	2.10E+02	5.00E+00	1.05E-01	4.41E-01
Noble Metals Group						
Mo-99	7.63E-01	1.77E+01	2.76E+01	6.19E-01	8.79E-03	7.72E-03
Tc-99m	6.09E-01	1.26E+01	1.83E+01	1.94E-01	1.08E-04	2.73E-08
Ru-103	6.07E-01	1.42E+01	2.23E+01	5.38E-01	1.21E-02	8.11E-02
Ru-105	3.59E-01	7.08E+00	1.01E+01	7.97E-02	1.82E-05	2.40E-10
Ru-106	2.00E-01	4.67E+00	7.36E+00	1.78E-01	4.16E-03	3.46E-02
Rh-105	3.70E-01	8.48E+00	1.32E+01	2.76E-01	2.64E-03	8.48E-04

TABLE 7.1-28 (Continued)

Isotope	0 to 1 hrs	2 to 3 hrs	0 to 8 hrs	8 to 24 hrs	24 to 96 hrs	96 to 720 hrs
Lanthanide Group						
Y-90	2.90E-03	6.65E-02	1.04E-01	2.32E-03	3.25E-05	2.75E-05
Y-91	4.19E-02	9.71E-01	1.53E+00	3.69E-02	8.43E-04	6.09E-03
Y-92	3.70E-02	6.93E-01	9.64E-01	5.77E-03	5.86E-07	0
Y-93	4.75E-02	1.02E+00	1.53E+00	2.25E-02	4.05E-05	2.91E-07
Nb-95	5.64E-02	1.31E+00	2.06E+00	4.95E-02	1.11E-03	7.23E-03
Zr-95	5.61E-02	1.30E+00	2.05E+00	4.94E-02	1.13E-03	8.29E-03
Zr-97	5.35E-02	1.19E+00	1.81E+00	3.26E-02	1.38E-04	7.58E-06
La-140	6.06E-02	1.38E+00	2.14E+00	4.58E-02	4.84E-04	1.97E-04
La-141	4.69E-02	8.98E-01	1.26E+00	8.69E-03	1.31E-06	0
La-142	3.58E-02	5.15E-01	6.53E-01	6.67E-04	6.96E-10	0
Nd-147	2.19E-02	5.06E-01	7.95E-01	1.89E-02	3.88E-04	1.49E-03
Pr-143	4.93E-02	1.14E+00	1.79E+00	4.27E-02	9.01E-04	3.95E-03
Am-241	4.23E-06	9.81E-05	1.54E-04	3.74E-06	8.75E-08	7.48E-07
Cm-242	9.98E-04	2.31E-02	3.64E-02	8.8 E-04	2.04E-05	1.64E-04
Cm-244	1.22E-04	2.84E-03	4.47E-03	1.08E-04	2.53E-06	2.16E-05
Cerium Group						
Ce-141	1.37E-01	3.19E+00	5.02E+00	1.21E-01	2.71E-03	1.72E-02
Ce-143	1.25E-01	2.85E+00	4.42E+00	9.20E-02	8.29E-04	2.34E-04
Ce-144	1.03E-01	2.41E+00	3.80E+00	9.19E-02	2.14E-03	1.77E-02
Pu-238	3.22E-04	7.51E-03	1.18E-02	2.86E-04	6.71E-06	5.73E-05
Pu-239	2.83E-05	6.60E-04	1.04E-03	2.52E-05	5.90E-07	5.04E-06
Pu-240	4.15E-05	9.69E-04	1.53E-03	3.69E-05	8.65E-07	7.39E-06
Pu-241	9.33E-03	2.17E-01	3.42E-01	8.30E-03	1.94E-04	1.66E-03
Np-239	1.60E+00	3.69E+01	5.76E+01	1.27E+00	1.67E-02	1.17E-02

United States Department of Agriculture

NRCS
Natural Resources Conservation Service
1204 Market St.
Port Gibson, MS 39150

Telephone: (601) 437-8121, ext. 3
Fax: (601) 437-3600

To: *Emily Trice*

From: *Raymond Joyner*

Today's date: *21 FEB 03*

Re: *Prime Farmland /Claiborne County, Miss.*

Notes: *Per your request*

Please telephone or fax if there is a discrepancy in the number of pages, or if the reception is unclear.

TOTAL NUMBER OF PAGES, INCLUDING COVER SHEET: *2*

Retyped from original FAX on 7/22/2004 due to legibility problems

U.S. Department of Agriculture
Soil Conservation Service

PRIME FARMLAND

Survey Area – CLAIBORNE COUNTY, MISSISSIPPI

Map Symbol	Prime Farmland Code	Soil Mapunit Name
Ad	1	ADLER SILT LOAM
CaA	1	CALLOWAY SILT LOAM, 0 TO 2 PERCENT SLOPES
CaB	1	CALLOWAY SILT LOAM, 2 TO 5 PERCENT SLOPES
Cn	1	COLLINS SILT LOAM
Co	2	COMMERCE SILT LOAM
Fa	2	FALAYA SILT LOAM
GrA	1	GRENADA SILT LOAM, 0 TO 2 PERCENT SLOPES
GrB	1	GRENADA SILT LOAM, 2 TO 5 PERCENT SLOPES
GrB2	1	GRENADA SILT LOAM, 2 TO 5 PERCENT SLOPES, ERODED
Hn	2	HENRY SILT LOAM
In	1	INGLEFIELD SILT LOAM (ADLER)
LmA	1	LORING AND MEMPHIS SILT LOAMS, 0 TO 2 PERCENT SLOPES
LmB2	1	LORING AND MEMPHIS SILT LOAMS, 2 TO 5 PERCENT SLOPES, ERODED
MeA	1	MEMPHIS SILT LOAM, 0 TO 2 PERCENT SLOPES
MeB	1	MEMPHIS SILT LOAM, 2 TO 5 PERCENT SLOPES
MeB2	1	MEMPHIS SILT LOAM, 2 TO 5 PERCENT SLOPES, ERODED

Prime Farmland Code	Description
1	All areas are prime farmland.
2	Only drained areas are prime farmland.

Retyped from original FAX on 7/22/2004 due to legibility problems

Defining Prime Agricultural Land and Methods of Protection

Andrew D. Carver and Joseph E. Yahner

This article is a hypertext publication of the Purdue University Department of Agronomy and Cooperative Extension Service.

Citation: Carver, A.D. and J.E. Yahner. 1996. Defining Prime Agricultural Land and Methods of Protection. Purdue Cooperative Extension Service. AY-283.

Statement of the Problem

A significant and critical part of the U.S. agricultural system faces an uncertain future resulting from land use controversy in the urban fringe (rural agricultural land experiencing pressure from suburban development). Urbanization is rapidly moving beyond the suburbs. As a result, competition has developed for incompatible uses of agricultural land. Land allocated to farming provides a flow of both market and nonmarket benefits to society (e.g., crop production and open space). These same lands, on the other hand, are sought by developers for profitable building sites.

Farming is central to the economy of Indiana and its counties. Though the importance of farmland and farmland protection is recognized by federal law, local planning and zoning continues to neglect the issue of prime agricultural land and the conflicts that arise between expanding development and successful farming. The prime farmland issue is of particular importance to extension educators who are often members of local planning commissions and zoning boards.

Estimates of the agricultural land converted annually to non-agricultural uses vary between 800,000 acres to more than 3 million nationwide. More important than the exact rate of conversion is the location of rapidly changing land use. Much of the land being lost is prime or unique farmland, disproportionately located near cities. According to Ralph E. Grossi of the American Farmland Trust, 58 percent of the total U.S. agricultural production comes from counties that the Census Bureau classifies as metropolitan and their adjoining counties. The considerable agricultural land endowments of Indiana have also provided a supply of "vacant" land for development. Ralph Gann of the Indiana Agricultural Statistics Service, estimates 20.2 million acres were under the control of Indiana farmers in 1950. That number has dropped to near 15.6 million acres. Part, but not all, of the decrease is due to development.

Effects of Non-Farm Development

The term "sprawl" can be characterized as the lack of continuity in urban expansion. A sprawling development pattern implies that the urban or suburban area is larger than it otherwise would be because undeveloped tracts remain interspersed among developed parcels and subdivisions. The effects of expanding development in rural and urban-fringe areas can be divided into two primary categories. First, development involves the direct conversion of farmland. Such conversion satisfies the demand for residential, commercial, and industrial land uses. Second, development indirectly reduces the agricultural potential of the remaining farms.

Converting a tract of agricultural land to a non-farm use results in long-term consequences. First, development immediately exhausts the agricultural productivity of the reallocated tract. Unfortunately, development often causes the preferential conversion of highly productive land. Characteristics of quality farmland, (e.g., flat or well drained soils) are often sought for development. Second, loss in terms of the opportunity foregone from the agricultural, open space, and related amenity benefits would be experienced indefinitely. Though a decision to restore the agricultural viability of a residential subdivision may be technically possible, it would not be feasible due to enormous expense.

Development indirectly reduces the productive potential of surrounding agricultural land by limiting its current or future use. In fact, impacts on the converted tract itself may be small in comparison to the current and future consequences impacting adjacent farmland. As an example, restrictions may be imposed on farming activities that affect the health, safety, and welfare of the growing non-farming population. The application of pesticides or manure near residential areas are two such activities for which society may demand new regulation. Much like current laws restricting the location of confined feeding operations, new regulations could require minimum separation distances between these activities and residential areas.

Scattered residential development also increases the potential for nuisance conflicts. Odor, noise, and dust are potential problems associated with agricultural production. These problems can often only be avoided by locating operations (especially confined feeding operations) away from people. Furthermore, even if an area's proportion of agricultural land area remains high, but available only in smaller scattered parcels, farmers may be prevented from employing newer technologies that require more land to achieve full economies of scale. Such restrictions reduce efficiency and increase production costs, perhaps even leading to premature idling of land.

A New Definition of Prime Agricultural Land

The United States Department of Agriculture (USDA) defines prime farmland as the land best suited to food, feed, forage, fiber, and oilseed crops. Prime farmland produces the highest yields with minimal inputs of energy and economic resources, and farming it results in the least damage to the environment. County *Soil Surveys* also follow this productivity-based approach to identifying prime agricultural land. In fact, a county *Soil Survey* not only contains yield data for crops and pasture, but often specifically identifies soils considered prime farmland. Consequently, the county *Soil Survey* provides a preliminary definition of prime agricultural land. However, problems created by direct and indirect effects of development indicate that, within the context of land use planning and zoning, the definition of prime agricultural land must be based on more than the traditional measures of soil productivity and crop yields. Instead, prime or select farmland should be defined by a combination of productivity and location. In the rural and urban fringe areas of today, the distance to residential development is becoming an increasingly important spatial characteristic affecting production.

Location can be incorporated in the definition of prime or select agriculture in the following ways. First, soils of moderate or even low productivity should share the prime agriculture designation if such soils are surrounded by large expanses of undeveloped, highly productive soils. Second, productivity should become secondary to location characteristics if the area in question supports confined feeding operations. Not only does separation by distance reduce the nuisance element associated with this important aspect of agriculture, but separation distance also provides surrounding farmland capable of supporting economical waste assimilation through land application of manure. Third, the designation of prime agriculture should be extended to include unique farmland located within expanding metropolitan areas. The current definition of prime farmland employed by the USDA and the Natural Resources Conservation Service (NRCS) specifically excludes highly productive soils from the "prime" status if they occur in urban or "built-up" areas (see 7 U.S.C.§4201(c)(1)(A)). This exclusion ignores the fact that farmland located within a highly developed area provides market and nonmarket benefits to society. While small "in-town" farming operations often provide higher-valued crops (such as fruits and vegetables) to consumers, they also provide open space, scenic values, and related amenity benefits. Such benefits are important in a planning and zoning context since they are public goods and can contribute to a community's "quality of life."

Retaining Land in Agriculture:

The Zoning Example

Ruled constitutional by the U.S. Supreme Court in 1926 (see Euclid v. Amber Realty Co., 272 U.S. 365), zoning is justified under the police powers of the state to prevent land uses that threaten the safety, health, morals, and general welfare of the public. Zoning ordinances influence urban land use primarily through the physical isolation of uses. While zoning is the primary method used to influence urban land use, relatively little zoning is practiced in rural and urban-fringe areas.

Current planning and zoning practices provide only a weak device for retaining land in agricultural. For example, in some Indiana counties, areas of prime agricultural land are given the AA (Select Agriculture) designation in the zoning ordinance. While such a land use designation may identify areas of agricultural importance, it does little to retain land in agriculture when the ordinances are subject to variances, zoning amendments, and special exceptions. Similarly, minimum lot size is the primary conventional zoning method used to insure low residential density in rural areas. Unfortunately, two, five, or even ten acre residential parcel size restrictions do little more than scatter development and consume or cripple prime farmland. Even if the minimum lot size is forty acres or more, an ordinance does nothing directly to prohibit nonagricultural uses of the tract. Furthermore, minimum lot size restrictions in Indiana primarily address the public health concerns of on-site waste disposal systems, not farmland preservation.

Fortunately, unconventional zoning methods do exist to preserve prime agricultural land. Open space zoning and exclusive agricultural zoning are two of the most promising. The conventional approach to development results in an entire development parcel being covered with houselots and subdivision streets. Open space zoning, on the other hand, relies on the principal of cluster development, whereby new homes are clustered onto part of the development parcel. Clustering allows the remainder to be preserved as productive farmland or unbuilt open space. Since only the density and not the number of houses is changed, open space zoning can permanently protect a substantial portion of every development tract's agricultural productivity without decreasing the development potential for both landowner and developer.

Exclusive agricultural zoning is less frequently used than nonexclusive zoning such as open space zoning, because it prohibits nonagricultural use of the land within the district. The main advantage is that it ensures there will be no conflict between residential and agricultural uses. However, the ordinances are more difficult to adopt because the farmland owners must forego (often reluctantly) the opportunity to sell their land to residential developers.

A more landowner friendly form of exclusive agricultural zoning is the voluntary creation of agricultural districts. The benefits which farmers obtain by voluntarily joining an agricultural district may include differential assessment, protection against nuisance ordinances, and limits on public investments for nonfarm improvements. Basic standards for reviewing district petitions should be outlined in the County Zoning Ordinance, if not at the state level. Like any zoning ordinance, however, its effectiveness can be undermined by a zoning authority's lax supervision of rezoning and variance requests.

The Property Rights Example

In addition to zoning, a county or local government can utilize transferable property rights to provide a more lasting means of preserving prime or select agricultural land. A program for transfer of development rights (TDR) allows landowners to sell their development rights to a developer. In turn, the developer may use them to develop qualified lands at higher densities than allowed under existing zoning laws. A TDR program allows local governments to steer development to desirable areas (such as those with sufficient infrastructure) while assuming little financial burden.

Under a similar program for purchase of development rights (PDR), landowners can sell conservation easements to governmental agencies or nonprofit organizations. PDR involves the purchase of a deed restriction on qualified farmland that restricts the future use of the land to agricultural or open space uses, either permanently or for a specified period of time. While the farmer retains the right to sell or transfer the land, it remains subject to the deed restriction precluding any future development or activities that may negatively impact its agricultural viability. An owner of agricultural land may also donate a conservation easement to a governmental agency or charitable organization and receive a charitable deduction (see 26 U.S.C.. § 170 (h)(4)(A)).

Acquiring the financial resources needed to purchase development rights is the greatest hurdle for implementing a PDR program. Importantly, a planning commission/ordinance committee must carefully establishment criteria from which to determine a farm's eligibility for participation in the program. Criteria should specifically target key parcels that would preserve the county's agricultural potential and open space amenities.

Summary

Prime agricultural land differs from other agricultural land designations in that it generally consists of highly productive soils. However, moderate and low productivity soils should be designated prime if such soils lie within, or are surrounded by contiguous areas identified as prime farmland. The inclusion of these soils may act to discourage development on the less productive or sloping soils of an otherwise prime agricultural area. Should such development occur, remaining prime agricultural land may no longer satisfy the requirements of a prime designation. Productivity is also a secondary factor when considering prime land designations in a rural area with confined feeding operations. Furthermore, "unique" farmland within metropolitan areas can be considered prime if it provides a community with demanded farm produce, open space, or related amenity benefits.

A variety of private and public land protection methods can be employed to protect agricultural operations from the impacts of non-farm development. However, their success ultimately relies on public and political support. Without that support, justification for prime farmland conservation is difficult. Often, environmental, social, and aesthetic effects of prime farmland loss are not readily quantifiable and most protection programs require administrative and financial resources beyond that required for current zoning policies.

Go to the beginning of the article

Back to Home