3.0 SITE SAFETY ASSESSMENT

As required by 10 CFR 52.17(a)(1), an application for an early site permit (ESP) must contain a description and safety assessment of the site on which a new facility would be located. The assessment must contain an analysis and evaluation of the major structures, systems, and components of the facility that bear significantly on the acceptability of the site under the radiological consequence evaluation factors identified in 10 CFR 50.34(a)(1). That site characteristics comply with 10 CFR 100 must also be demonstrated.

Preceding sections provide detailed descriptions, assessments, and analyses of the proposed ESP Site (i.e., the Grand Gulf Nuclear Station (GGNS) site), and the ESP Facility as defined in Chapters 1.0 and 2.0 of this report.

This section provides an assessment of conformance with 10 CFR 100 requirements, including applicable parts of 100.10, 100.11, 100.20, 100.21 and 100.23, with respect to evaluation of the ESP Site for an Early Site Permit under Part 52. Specifically, this section demonstrates that radiological doses from normal operation and postulated accidents will be acceptably low, that natural phenomena and potential man-made hazards important to the design of the plant have been identified, that adequate security measures to protect the plant can be developed, and that there are no physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans for the ESP Facility.

3.1 Non-Seismic Siting Criteria

3.1.1 Exclusion Area and Low Population Zone

The ESP Site exclusion area authority and control thereof is described in Section 2.1.2. The ESP Site exclusion area boundary (EAB) includes an area encompassed by a circle of about 841 meters radius. The boundary line for the proposed EAB is shown in Figure 2.2-1. The ESP Site exclusion area meets the definition for an exclusion area provided in 10 CFR 100.3.

The ESP Site low population zone (LPZ) is described in Section 2.1.3.4. The ESP Site LPZ includes an area encompassed by a circle of approximately 2-mile radius (3219-m). The approximate LPZ is shown in Figure 2.1-3. The ESP Site LPZ meets the definition for an LPZ provided in 10 CFR 100.3.

3.1.2 Population Center Distance

The ESP Site population center distance is described in Section 2.1.3.5. The closest population center for the ESP Site is Vicksburg, Mississippi, located approximately 25 miles north-northeast of the site, with a 2000 population of 26,407. The ESP Site nearest population center is in accordance with the definition of a population center (more than a population of about 25,000 residents) provided in 10 CFR 100.3. In addition, it satisfies the criteria provided in 10 CFR 100.21(b) as being at least one-and-one-third times the distance from the proposed reactor location to the outer boundary of the low population zone or, in this case, approximately 2.7 mi.

3.1.3 Site Atmospheric Dispersion Characteristics and Dispersion Parameters

The site atmospheric dispersion characteristics and dispersion parameters for the ESP Site are described in Section 2.3.4 for the short term diffusion estimates used in assessing the site suitability (radiological consequences) associated with postulated accidents and Section 2.3.5 for the long term diffusion estimates used in evaluating the normal radiological effluent release limits.

The potential consequences and acceptance criteria for the postulated accidents used in the evaluation of the ESP Site are provided in Section 3.3. As demonstrated therein, the dose limits at the EAB and LPZ meet the requirements of 10 CFR 50.34(a)(1)(ii)(D)(1) and 10 CFR 50.34(a)(1)(ii)(D)(2), respectively.

The potential consequences and acceptance criteria for the normal operations gaseous radiological effluent release limits are provided in Section 3.2, where it is shown that the applicable regulatory limits, provided in 10 CFR 20 and 10 CFR 50, Appendix I, are satisfied for the ESP Site.

3.1.4 Physical Site Characteristics – Meteorology, Geology, Seismology, and Hydrology

3.1.4.1 Meteorology

The meteorological characteristics of the ESP Site are described in detail in Sections 2.3.1 and 2.3.2. Regional, local and site data were used to establish average and extreme meteorological parameters for the site.

Section 2.3.1 describes the regional meteorological characteristics of the general site based on long-term historical observations from National Weather Service Stations located in Jackson, Mississippi, and in Vicksburg, Mississippi, both of which are within 55 mile of the ESP Site. Recent data from these weather stations and from the National Oceanographic and Atmosphere Administration (NOAA) National Climatic Data Center (NCDC) data systems are provided as appropriate. Regional historical information for the site area includes data for temperature, relative humidity, wind, and precipitation (rain and snowfall). Severe weather information for the area is also summarized in this section for hurricanes (frequency of occurrence and wind speeds), thunderstorms (frequency of occurrence), hail (frequency and distribution in the region), and lightning (predicted stroke density), all of which have been characterized for consideration in the design of site structures, systems and components as required. Tornadoes (predicted frequency and intensity) and severe winds (maximum speed) were characterized to provide the site parameters to be considered in association with these events (including maximum linear and rotational wind speeds, pressure drop, and rate of pressure drop). Heavy snow (frequency and intensity), and freezing rain / ice (frequency and intensity) were characterized to provide worst-case accumulations of snow and ice to be accounted for in the design of site structures.

Section 2.3.2 describes the local and site-specific meteorological characteristics of the ESP Site as obtained from the Vicksburg weather station, and from an on-site meteorological monitoring system operated continuously by Entergy since 1972. A detailed description of the on-site monitoring system is provided in Section 2.3.3. Data from the on-site monitoring system was used to establish normal and extreme values of wind speed and direction, temperature, atmospheric moisture (wet bulb temperature, relative humidity, and dew point temperature), precipitation, and atmospheric stability. Site-specific meteorological data were also used to supplement the regional and local data, as well as to facilitate the development of site-specific atmospheric dispersion characteristics and dispersion parameters for routine and accidental gaseous releases from the ESP Facility, as described in Sections 2.3.4 and 2.3.5.

The information contained in Sections 2.3.1 and 2.3.2, on regional and local meteorology were evaluated to provide representative average and extreme meteorological information characteristic of the ESP Site.

3.1.4.2 Geology

The geological, seismological and geophysical characteristics of the proposed location of the new facility at the existing GGNS are described in Section 2.5. The geology of the Site Region (200-mile-radius) and Site Vicinity (25-mile-radius) is described in Section 2.5.1.1. The geology of the Site Area (5-mile radius) and Site Location (0.6-mile radius) is described in Sections 2.5.1.2. Descriptions of the geological characteristics of the ESP Site are based on a compilation, review and analysis of existing data, as well as the results of a geological and geotechnical site investigation and laboratory testing program. The evaluation of the site geology included a review of results of geotechnical explorations and laboratory analyses completed as part of the original site evaluations documented in the PSAR for GGNS Units 1 and 2. The previous subsurface exploration program included 275 borings drilled to a maximum depth of 447 feet. The field investigation completed during this ESP investigation is described in Section 2.5.4 and included:

- Drilling and sampling of four borings to depths between 141.5 and 238.0 feet;
- Four Cone Penetrometer (CPT) soundings to depths of between 75 and 98 feet.

The ESP Site is underlain by a sequence of Quaternary eolian and alluvial deposits overlying Miocene Catahoula Formation bedrock. Four units were differentiated at the site, including: (1) an upper layer of late Pleistocene silt and clayey silt (“loess”) ranging from 55- to 70-feet thick; (2) an intermediate layer of stiff to very stiff Pleistocene alluvium ranging from 50- to 100-feet thick; (3) a deeper layer of very stiff to hard older alluvium ranging from 40- to 90-feet thick; and (4) Catahoula formation bedrock.

The results of the data review and site investigations indicate that the geological and geotechnical conditions of the ESP Site are consistent with the information presented in the GGNS UFSAR. The ESP Site soil profile is relatively consistent across the footprint of the existing GGNS Unit 1 facility and the location of the power block for the proposed new facility.

Section 2.5.3 discusses the potential for surface fault rupture in the Site Area. The ESP Site is located within the tectonically quiescent Gulf Coastal Plain province and is underlain by unfaulted deposits of at least Oligocene age. No faults are mapped within the 5-mile radius of the ESP Site. The closest mapped faults in the Study Region occur in southeastern Arkansas, located approximately 90 miles north-northwest from the ESP Site. Deformation associated with salt migration has occurred in the Site Region. However, no salt domes occur within either the 5-mile radius or 0.6-mile radius of the ESP Site.

Results of the geological and geotechnical investigations conclude that the physical characteristics of the site pose no undue risk to the siting of a new facility at the proposed location. No geological hazards from surface fault rupture (Section 2.5.3), slope instability, or ground subsidence from sinkholes or mine collapse were identified either during the original PSAR site evaluations for GGNS Units 1 and 2 or during this ESP Site investigation (Section 2.5.5). Due to the position of the site on topographically high ground, and lack of surface water impoundments, there is no risk to the site from flooding or inundation (Section 2.5.6). There have been no reports of unusual or unacceptable behavior of the existing GGNS facility relative to geologic or geotechnical conditions during its nearly 20 years of operation. Subsurface materials exist beneath the ESP Site that are suitable bearing layers for the foundation of a new facility at the proposed location.

3.1.4.3 Seismology

Section 2.5.1 describes the seismotectonic environment of the Site Region and Section 2.5.2 describes the data and methodology used to develop the Safe Shutdown Earthquake (SSE) ground motion for the proposed location of a new facility at the existing GGNS site.

The Site Region is characterized by extremely low rates of earthquake activity. Only 39 earthquakes of magnitude greater than M_b 3.3 have been recorded within the entire Site Region. No earthquakes of magnitude greater than M_b 3.3 have been recorded within approximately 90-miles of the ESP Site.

Because the ESP Site is underlain by soils, investigations were completed to establish the soil profile, e.g., seismic wave transmission effects, for the site-specific site-response analysis and development of the SSE. In addition to the four borings and four CPT probes, the site investigation included:

- Borehole P-S seismic velocity surveys in three of the exploratory borings;
- Laboratory engineering index testing of sixty ESP borehole samples; and,
- Dynamic resonant column testing of six boring samples.

The average shear wave velocity for the ESP site ground motion site-response analysis was developed by normalizing the three borehole surveys to a common elevation, and then averaging the receiver-to-receiver shear wave velocities. The resulting averaged velocity plot (Section 2.5.4) was visually examined to identify discrete interval velocities that correspond, in part, to the geologic unit layers, and that have relatively distinct average velocity increases or breaks. Four interval velocities were differentiated from the P-S velocity survey profile:

- Loess – 770 fps;
- Upland Complex Alluvium and Loess-Alluvium Interface – 1,004 fps;
- Upland Complex Old Alluvium – 1,378 fps; and,
- Catahoula Formation – 2,118 fps.

The average velocities are within typical ranges for similar materials reported in published literature (e.g., Hunt, 1984).

The P-S datalogger used for the ESP study represents a marked improvement and advancement of technique over the cross-hole seismic velocity techniques and equipment that were used for the initial site evaluation for GGNS in the 1970s. Therefore, a direct comparison cannot be made between the two data sets. However, the velocities for the various geologic layers generally fall within similar ranges, if the GGNS data for the upper Catahoula Formation are compared against the velocity data for the Upland Complex Old Alluvium. A comparison between the shear wave velocities is shown below.

Material	ESP Vs (fps)	UFSAR Vs (fps)
Loess	590 to 1,450	670
Upland Complex Alluvium	740 to 1,750	1,100 to 1,600
Upland Complex Old Alluvium	530 to 3,360	1,640 to 1,720
Catahoula Formation	1,500 to 2,830	1,640 to 1,720

3.1.4.4 Hydrology

The hydrologic conditions of the ESP Site and vicinity are described in detail in Section 2.4. The descriptions include hydrologic features and characteristics that should be accounted for in the design of the ESP Facility. These hydrologic engineering characteristics include floods, ice effects, cooling water supply, low-water considerations, accidental releases in surface water, and ground water.

Section 2.4.2 presents information on the flooding history, flood design considerations, and the effects of local intense precipitation. The probable maximum precipitation event was determined to control facility flood design. Therefore, safety-related structures of the ESP Facility will need to be above the flood elevation or be designed to withstand the effects of flooding. The effects of and development of the probable maximum precipitation are presented in Section 2.4.2.3 and 2.4.3.1.

Section 2.4.3 describes the probable maximum flood characteristics for local streams and for the Mississippi River, and Section 2.4.10 discusses the flooding protection requirements. As described in Section 2.4.3, the maximum flood elevation of the river is about 103 ft msl, based on the height of the flood control levees on the west side of the river. Floods in the river would not affect the ESP Facility, the location of which is proposed at a similar grade elevation as that of the existing GGNS Unit 1 facility, on the bluffs east of the river.

Section 2.4.7 describes the effects of ice formation in the river at the location of the ESP Site, and the probable maximum winter flood on the river level. In Section 2.4.8 of the NRC Safety Evaluation Report (NUREG-0831) for GGNS Unit 1, the NRC concluded that the occurrence of a major ice jam on the Mississippi River is very unlikely, and concurred that ice flooding was not a design basis consideration for the GGNS site. Therefore, ice flooding is similarly not a design basis consideration for the ESP site.

Section 2.4.11 describes low river water considerations for the site, including the evaluation of plant requirements and ultimate heat sink (UHS) dependability requirements. The ultimate heat sink for the ESP Facility would be provided from closed-loop cooling systems utilizing basin type reservoirs, and would not rely on the river intake for cooling capability. Therefore, the UHS would be unaffected by a low river stage.

Section 2.4.13 describes the potential effects on ground water from accidental radiological releases. The evaluation for GGNS Unit 1 in their UFSAR indicated that strontium and cesium isotopic concentrations for a design basis accidental spill would be below the maximum permissible concentration at a distance of 57 feet from the location of the spill. An estimated ground water travel time to the Mississippi River was determined as about 12.5 years. Since the proposed location of the ESP Facility, like the GGNS Unit 1 facility, is approximately 3,200 feet from the Mississippi River, the isotopic concentrations from a similar spill into the ground water should be well below the maximum permissible concentration before they reach the Mississippi River. Therefore, the potential for effluents to reach a surface water body and surface water users is minimal.

Section 2.4.12 describes the regional and local aquifers, their formation, sources and sinks. Section 2.4.12.1 describes plant requirements from the ground water system and describes ground water quality. Section 2.4.12.2 describes the site hydrogeologic systems including the aquifers present and their characteristics (depth, permeability, potentiometric levels and velocity), and present and projected future ground water users. The design basis for subsurface hydrostatic loading is presented in Section 2.4.12.4.

The information contained in Section 2.4 on surface water and ground water conditions was evaluated and was determined to be adequate in support of the ESP Facility. These data would be used as appropriate in the design of the ESP Facility to ensure that no hydrology related site parameters would pose an undue risk to the operation of the ESP Facility.

3.1.5 Potential Off-site Hazards

The potential offsite hazards for the ESP Facility are described in Section 2.2. The description includes nearby industrial, transportation and military facilities.

Sections 2.2.1 and 2.2.2.5 addresses area airports and associated air transportation routes, as they may affect the ESP Facility. No commercial airport facilities are located within 10 miles of the GGNS site. The nearest commercial airport is located in Jackson, MS, approximately 65 miles northeast of the site. There are 5 general/public aviation airports located within the vicinity of the site. These general/public aviation airports are used only for small planes.

As noted in Section 2.2.3, highway accidents are not a concern for the ESP Site. The ESP Site area is accessible by U. S. Highway 61 and State Highway 18 which connect Port Gibson (5 miles southeast of the site) with Natchez, Jackson, and Vicksburg. U. S. Highway 61 passes approximately 4.5 miles east-southeast of the GGNS site at its closest point. The distance beyond which an exploding truck will not have an adverse effect on plant operations, nor prevent safe shutdown, is calculated to be 1,658 feet (0.31 miles). Since the closest point of U. S. Highway 61 to the ESP Site is about 4.5 miles, there is no hazard to the plant due to an accident on U.S. Highway 61.

There are currently no active rail lines in the vicinity of the ESP Site. Therefore, potential accidents involving railway traffic are not evaluated.

The nearest bank of the river is approximately 1.1 miles from the proposed location for the ESP Facility on the GGNS ESP Site. In addition, a new facility would be located on the bluffs to the east of the river, which are approximately 65 feet above the normal river level. As noted above for the GGNS Unit 1 plant, this bluff would provide an earthen shield against possible explosions originating from river barge traffic. Based on the combination of distance from the river bank and the intervening bluff, this would preclude any damage to the structures of the ESP Facility at the proposed location, resulting from an explosion originating from a ship or barge on the river.

Section 2.2.3.1 discusses explosions due to pipelines and nearby industrial facilities. Evaluation of the existing pipelines, their proximity to the site and the materials passing through them resulted in the determination that they do not represent a design concern for facilities at the ESP Site. There are no existing industrial facilities potentially representing an explosive source which would constitute a design consideration for the ESP Site.

Section 2.2.3.1 discusses explosions due to onsite hydrogen storage, and due to liquid-hydrogen delivery truck accidents/explosions. Liquefied hydrogen is delivered to the GGNS site by United States Department of Transportation (USDOT) approved truck, with a maximum capacity of 17,000 gallons. There are no regulations specifying a minimum distance between a liquid-hydrogen delivery truck and a safety-related structure. The current truck route on the GGNS (ESP) Site results in about 400 ft separation from the outer boundary of the proposed location for the power block of the ESP Facility, which is less than the minimum separation distance of 1285 ft calculated per Regulatory Guide 1.91 (Reference 1). However, the probability of an accident resulting in a hydrogen explosion calculated per the Regulatory Guide 1.91 methodology is 4.1×10^{-7} per year. Therefore, according to the guidelines presented in Regulatory Guide 1.91 (criteria is less than 10^{-6} per year), a liquid-hydrogen truck explosion event need not be considered a design basis accident for the ESP Facility on the site.

The presence of the 20,000 gallon liquid-hydrogen storage tank located in the north end of the abandoned GGNS Unit 2 cooling tower basin (Figure 2.2-4) presents a potential hazard of an explosion. An analysis was performed to determine the safe separation distance between the liquid-hydrogen storage tank and any GGNS Unit 1 safety-related structure. These calculations are valid for the ESP Facility at the GGNS ESP Site, so long as the minimum separation distances stated in the report are maintained, or structures are appropriately designed for the expected blast pressure. The proposed area for construction of the ESP Facility is beyond the minimum separation distance requirements given in the calculation for both blast considerations and gaseous cloud considerations.

Toxic chemicals are discussed in Section 2.2.3.1.2. The closest point of U.S. Highway 61 to the GGNS site is 4.5 miles. Therefore, an accidental release of toxic chemicals transported on U. S. Highway 61 would not endanger the safe operation of the ESP Facility at its proposed location on the ESP Site. In the year 2000, the majority of the hazardous materials transported near the GGNS site were fuel products moving on the Mississippi River. The 6-year onsite wind frequency distribution data (1996-2001) reported in Section 2.3 shows that the winds that originated from compass sectors W-SW, W, W-NW and NW, that would carry the hot plume from a fire caused by explosion to the proposed location for a new facility, had speeds generally under 20 mph. An analyses presented in the GGNS Unit 1 UFSAR concluded that a wind speed greater than 70 mph would be required to direct a plume toward GGNS Unit 1. The proposed location for the ESP Facility is on the bluffs above the river and about 1.1 miles inland. Since the proposed location for the ESP Facility is very near to that of the existing GGNS Unit 1, no toxic hazard to the ESP Facility would be expected.

There are no military installations, chemical or munitions plants, stone quarries, or major gasoline-storage areas located within 5 miles of the ESP site. Therefore, they do not need to be considered as a hazard for the ESP Facility on the ESP Site.

Section 2.2.3.1.3 discusses the possible offsite fire hazards to an ESP Facility on the GGNS ESP Site. It was concluded that offsite fires do not pose a design basis threat to a new facility on the site.

A collision (by river traffic) with the proposed cooling system intake is not considered likely and not a design basis event for the ESP Facility as discussed in Section 2.2.3.1.4.

Liquid spills on the Mississippi River do not pose a threat to safe shutdown of the ESP Facility, as the river intake is utilized only for non-safety related water supply. Any potential intrusion of hazardous chemicals or liquids into the proposed embayment and makeup water system could be mitigated by orderly shutdown of the facility, if required.

3.1.6 Site Characteristics - Security Plans

The ESP Facility power block proposed location (approximate center of the power block area) is approximately 1200 ft west and 1000 ft north of the existing GGNS Unit 1 Facility. A site plot plan is provided in Figure 2.1-1.

3.1.6.1 Land Sufficient To Implement The Criteria Of 10 CFR 73.55

Based upon the general location at the GGNS site on which the nuclear unit or units would be located; e.g., in the general vicinity of the GGNS Unit 1, there is sufficient land and distance to the site boundary and appropriate topography to implement the criteria of 10 CFR 73.55 relating to the development of a security plan. This conclusion is based in part on the fact that GGNS Unit 1 has implemented a security plan meeting the requirements of 10 CFR 73.55 and the interim compensatory measures required by the NRC's Order of February 25, 2002. While GGNS Unit 1 is still in the process of implementing the requirements of the revised design basis threat (DBT) Order of April 25, 2003, preliminary evaluations would indicate that neither the amount of land, the particular location of the GGNS site in relation to the topography and site boundaries or the distances to the site boundary or other natural features, would preclude compliance with the revised DBT.

It should be noted that existing commercial nuclear power plants, such as GGNS Unit 1, were designed to meet evolving 10 CFR 73.55 requirements, including effective changes in the DBT and revised DBT, on an "add-on" basis after completion of the initial physical design. Even given these circumstances, plants such as GGNS Unit 1 are capable of meeting the evolving NRC security requirements. For a plant which would be built in the future, security considerations (e.g., barriers, access, fences) would be incorporated as initial design requirements and inputs and integrated into the overall design as an important element, making it reasonable to conclude that such a facility will be able to meet NRC security requirements.

Given the opportunity to design security into a new facility, the distance specified in Regulatory Guide 4.7 would be sufficient to satisfy the criteria of 10 CFR 73.55 although a larger distance could be used at the GGNS ESP site, and even a smaller distance could be accommodated.

3.1.6.2 Site Characteristics That May Require Mitigation

No site characteristics that require significant mitigation in order to control close approaches to the proposed location of a new facility have been identified. As indicated Figure 2.1-1, the nearest public road is about 3000 feet from the general area of the proposed power block building site. The Mississippi river is approximately 1 mile from the proposed power block building site. Safety-related structures necessary for the ultimate heat sink would not be located on an accessible, navigable waterway.

3.1.6.3 Identification of Potential Hazards in the Site Vicinity

Initially, given the successful implementation of a security plan by Entergy Operations for GGNS Unit 1, there are no potential hazards in the site vicinity which would preclude the development of a security plan for the new unit or units. The new reactor or reactors will be sited at some distance from the existing GGNS Unit 1, and provisions will be made such that construction activities at a new facility will not adversely affect the ability of GGNS Unit 1 or any new operating unit to meet NRC security requirements. Similarly, the design of the security plan and defensive strategy will be such that during operation or other activities on site, the security plans of the units on site positively reinforce each other, or will be independent with regard to their individual ability to meet NRC security requirements and the design basis threat, as revised.

3.1.6.4 Law Enforcement Agencies

Given the location of a new facility in relationship to GGNS Unit 1 which has, as part of its security plan, made provisions with relevant local law enforcement agencies, there is high assurance that similar provisions can be made with regard to any new facility, in that the jurisdictions and local law enforcement agencies are the same as for GGNS Unit 1.

In summary, given the proposed location of a new facility near GGNS Unit 1, and the ability to assure compliance with NRC provisions through design, there is a high assurance that NRC security requirements can be met for a new facility.

3.1.7 Site Characteristics - Emergency Plans

Information regarding emergency planning capability is provided in the ESP Application, Emergency Planning Information, Part 4. The GGNS Unit 1 evacuation time estimate (ETE) performed in 1986 was re-evaluated in support of this application. This re-evaluation included an assessment of updated population levels and distributions and transportation networks. As part of the effort, each major roadway was driven and traffic count data was obtained, as appropriate. Improvement in several key roadways was noted, and updated roadway capacities were estimated to support this evaluation. Local Mississippi and Louisiana emergency management agency officials, as well as state department of transportation representatives, were consulted and provided their concurrence regarding the findings. Based on this re-evacuation of the ETE, it was determined that there are no physical characteristics unique to the GGNS site that could pose a significant impediment to the development of the required emergency plans for the ESP Facility.

3.1.8 Population Density

As described in Section 2.1.1 and Section 2.1.3.6, the ESP Site is located in a mostly rural, low population density, area. The most densely populated area within 30 miles of the site is to the north-northeast with an average projected population density of about 238 people per square mile in the year 2030. This population density is projected to increase to only about 268 persons per square mile in the year 2070. The current and projected population density in this area is well below the NUREG-0800 guidance of 500 people per mi^2.

3.1.9 References

1. U.S. Nuclear Regulatory Commission (NRC), February 1978, Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants, Regulatory Guide 1.91, Revision 1, Washington, DC.

3.2 Gaseous Effluent Release Dose Consequences from Normal Operations

The site atmospheric dispersion characteristics and dispersion parameters for the ESP Site are described in Section 2.3.5 for the long term diffusion estimates used in evaluating the normal gaseous radiological effluent release limits.

3.2.1 Exposure Pathway and Source Terms

Operation of the new facility would contribute slightly to the radiation exposure over that of natural background received by individuals living in the vicinity of the site. Radiological exposure due to operation of the new facility is highly dependent on the exposure pathway by which a receptor may become exposed to radiological releases from the facility. The major pathways of concern are those that could result in the highest calculated offsite radiological dose. These pathways are determined from the type and amount of radioactivity released, the environmental transport mechanism, and how the environs surrounding the site are used (e.g., residence, gardens, etc.). Per 10 CFR 100.21(c) and (c)(1), this assessment focuses on gaseous effluents associated with normal operations of the ESP Facility.

For gaseous effluents, the environmental transport mechanism is dependent on the meteorological characteristics of the area. However, the most important factor in evaluating the exposure pathway is the use of the environment by the residents in the area around the GGNS ESP site. Factors, such as location of homes in the area, use of cattle for milk, and gardens used for vegetable consumption, are considerations when evaluating exposure pathways. Radioactive gaseous effluent exposure pathways include direct radiation from plume immersion, deposition on plants and soil, and inhalation by animals and humans.

The description of the exposure pathways herein and the calculational methods utilized to estimate doses to the maximally exposed individual and to the population surrounding the GGNS ESP site are based on USNRC Regulatory Guides 1.109 (Reference 5) and 1.111 (Reference 6). The source terms used in estimating exposure pathway doses are based on the values provided in Table 1.3-2.

3.2.2 Gaseous Pathway Dose Calculation Methodology

The methodology contained in the GASPAR II program (described in NUREG/CR-4653) was used to determine the gaseous pathway doses. This program implements the radiological exposure models described in Regulatory Guide 1.109 (Reference 5) for radioactivity releases in gaseous effluents. The code calculates the radiation exposure to man from:

- External exposure to airborne radioactivity;
- External exposure to deposited activity on the ground;
- Inhalation of airborne activity; and,
- Ingestion of contaminated agricultural products.

Table 3.2-1 and Table 3.2-2 present the gaseous pathway parameters used by the GASPAR II code to calculate doses for both the maximally exposed individual and for the general population.

3.2.3 Radiation Dose to Members of the Public

Dose rate estimates were calculated for hypothetical individuals of various ages exposed to gaseous radioactive effluents through the following pathways:

- Direct radiation from immersion in the gaseous effluent cloud and from particulates deposited on the ground;
- Inhalation of gases and particulates;
- Ingestion of milk contaminated through the grass-cow-milk pathway; and,
- Ingestion of foods contaminated by gases and particulates.

Annual radiation exposures to the maximum exposed individual and the population within a 50-mile radius of the Grand Gulf site via the pathways of submersion, ground contamination, inhalation and ingestion are given in Tables 3.2-3 and 3.2-4, respectively. These doses have been evaluated using the release data given in Table 1.3-2 and atmospheric dilution and deposition factors (χ/Q and D/Q) given in Section 2.3.5. For models and values of required parameters, Regulatory Guide 1.109 (Reference 5) was used. Annual production rates of milk, meat, and vegetables are given in Tables 3.2-6, 3.2-7 and 3.2-8, respectively. The estimated population distribution in the year 2070 within a 50-mile radius of the Grand Gulf site, given in Section 2.1, were used to evaluate the population exposures. As can be seen from Table 3.2-3, the estimated whole-body and critical organ annual doses to the maximum exposed individual due to release of radioactive materials in gaseous effluents from a new facility meet the guidelines of Appendix I to 10 CFR Part 50. Since the guidelines of Appendix I to 10 CFR Part 50 for maximum individual exposures via atmospheric pathways are much more restrictive (by a factor of $\approx$100) than the standards of 10 CFR Part 20, it can be inferred that radioactive releases via gaseous effluents from the new facility meets the standards for concentrations of released radioactive materials in air (at the locations of maximum annual dose to an individual and hence, at all locations accessible to the general public), as specified in Column 1 of Table 2 of 10 CFR Part 20.

As stated in Section 5.2.1 of the GGNS FER (Reference 1), the whole body dose to individuals living in the site region from existing radiation sources is expected to average about 130 mrem/yr. Comparison of the calculated doses listed in Table 3.2-3 shows that there is no significant additional dose to members of the public due to operation of a new facility at the GGNS ESP site.

3.2.4 References

1. Mississippi Power and Light Company, Grand Gulf Nuclear Station Units 1 and 2 Final Environmental Report (FER), as amended through Amendment No. 8
2. NUREG/CR-4013, LADTAP II - Technical Reference and User Guide, PNL-5270, April 1986.
3. NUREG/CR-4653, GASPAR II - Technical Reference and User Guide, PNL-5907, March 1987.
4. Grand Gulf Nuclear Station Updated Final Safety Analysis Report (UFSAR).
5. USNRC Regulatory Guide 1.109, Rev. 1, 1977, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I.

6. USNRC Regulatory Guide 1.111, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors, Revision 1, July 1977.
7. 10 CFR 50, Appendix I, Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents.
8. 10 CFR 20, Standards for Protection from Radiation.
9. 10 CFR 20.1301, Dose Limits for Individual Members of the Public.

3.3 Postulated Accidents And Accident Dose Consequences

10 CFR 52.17(a)(1) requires a site safety assessment that demonstrates the acceptability of the site under the radiological consequence evaluation factors identified in §50.34(a)(1) and that site characteristics comply with 10 CFR 100. Specifically, 10 CFR 100.21(c)(2) requires that radiological dose consequences of postulated accidents meet the criteria set forth in 10 CFR 50.34(a)(1). This section will review and analyze a robust spectrum of design basis accidents (DBAs) in order to bracket post-accident radiological consequences for the reactor or reactors proposed 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. 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, 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 suitability of the site.

The site atmospheric dispersion characteristics and dispersion parameters for the ESP Site are described in Section 2.3.4 for the short term diffusion estimates used in assessing the site suitability (radiological consequences) associated with postulated accidents.

3.3.1 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 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), NUREG-0800 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 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 various LWR plants have certified standard designs, and they have accepted postulated accident bases.

The representative DBAs for the boiling water reactor (BWR), pressurized water reactor (PWR), and other reactor designs evaluated includes:

- 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 Regulatory Guide 1.183 (Reference 6) as important for assessing the offsite dose consequences, and thus site suitability for construction and operation of a reactor or reactors as defined by the PPE.

3.3.2 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 (χ/Q) values for the EAB and LPZ which are based on onsite meteorological data (Section 2.3). The site specific χ/Q values are presented in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The accident dose estimates were performed using χ/Q and activity releases for the following intervals:

- Exclusion Area Boundary (EAB)
  - 0 to 2 hours
- Low Population Zone (LPZ)
  - 0 to 8 hours
  - 8 to 24 hours
  - 1 to 4 days
  - 4 to 30 days

3.3.3 Source Terms

Time-dependent activities released to the environs were used in the dose estimates. 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 for the large break LOCA; otherwise, the ABWR accident evaluations use the Alternate Source Term (AST) methodology

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

in accordance with Regulatory Guide 1.183 (Reference 6). The 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.

3.3.4 Postulated Accident Analyses

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 3.3-1.

3.3.4.1 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

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[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 (χ/Q) values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The TEDE doses for the accident-initiated iodine spike are shown in Table 3.3-2. 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 3.3-3. These doses meet the 25 rem TEDE guideline of 10 CFR 50.34.

3.3.4.2 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. All 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 χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The activity released to the environment for the maximum activity and pre-existing iodine spike is shown in Table 3.3-4. The calculated doses for the maximum allowed equilibrium activity at full power operation are shown in Table 3.3-5. 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 3.3-6. The doses at the EAB and LPZ are within the 25 rem TEDE guideline of 10 CFR 50.34.

3.3.4.3 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 GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The TEDE doses for the locked rotor accident are shown in Table 3.3-7. These doses are a small fraction of the 25 rem TEDE guidelines of 10 CFR 50.34.

3.3.4.4 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 gases0.10
  - Alkali metals0.12
- Fraction of fuel melting – 0.0025
- Activity released from melted fuel:
  - Iodines0.5
  - Noble gases1.0
- 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 vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The TEDE doses for the control rod ejection accident are shown in Table 3.3-8. 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.

3.3.4.5 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.

3.3.4.6 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 vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The TEDE doses for the steam generator tube rupture accident with the accident-initiated iodine spike are shown in Table 3.3-9. 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 3.3-10. These doses are within the 25 rem TEDE guidelines of 10 CFR 50.34.

3.3.4.7 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 vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The TEDE doses for the failure of small lines carrying primary coolant outside containment are shown in Table 3.3-11. 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.

3.3.4.8 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 3.3-12. These releases were used along with the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ) to determine the offsite doses. The TEDE doses for the failure of small lines carrying primary coolant outside containment are shown in Table 3.3-13. 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.

3.3.4.9 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
- Core thermal power of 3468 MWt (102 percent of design core power of 3400 MWt)

- 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 vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The TEDE doses for the AP1000 large break LOCA accident are shown in Table 3.3-14. 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 3.3-14 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.

3.3.4.10 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 iodine99.7 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 χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The activities released to the environment from the reactor and turbine buildings are listed in Table 3.3-15. The doses for the ABWR large break LOCA accident are shown in Table 3.3-16. 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.

3.3.4.11 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 3.3-17. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The TEDE doses for the ACR-700 LOCA accident are shown in Table 3.3-18. The doses meet the dose guidelines of 25 rem TEDE given in 10 CFR 50.34.

3.3.4.12 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.

All 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 3.3-19. The GGNS ESP-site-specific doses were calculated using the atmospheric dispersion (χ/Q) values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The resulting doses at the EAB and LPZ are summarized in Table 3.3-20. 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.

3.3.4.13 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 3926 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 3.3-21. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The resulting doses at the EAB are summarized in Table 3.3-22. This evaluation does not assess the LPZ dose since it is bounded by the EAB dose due to the 2-hour release duration and the lower χ/Q for the LPZ. Since the EAB and LPZ have the same acceptance criteria, compliance with the EAB criteria will ensure compliance with the LPZ criteria. 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 25 rem TEDE guidelines in 10 CFR 50.34.

3.3.5 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.

3.4 Geologic and Seismic Siting Factors

3.4.1 Geologic and Seismic Engineering Characteristics

The geological, seismological, and geotechnical characteristics of the EPS Site and its surroundings have been investigated to evaluate the suitability of the site with respect to geological hazards, to assess whether general foundation conditions are appropriate for placement of a new facility, and to provide the necessary information for developing the SSE ground motions. As discussed in Section 2.5 and Sections 3.1.4.2 and 3.1.4.3, there are no geological hazards that would adversely affect the ESP Site, and suitable foundations materials are present to support a new facility at the proposed location. The geological and geotechnical conditions of the ESP Site are suitable for the development of a new facility. As discussed below and in Section 2.5.2, the SSE ground motions for the ESP Site are lower than the Regulatory Guide 1.60 spectrum anchored to a peak free-field ground motion of 0.3g. Therefore, the ESP Site is also suitable with respect to earthquake ground motions.

Regulatory Guide 1.165 "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion" recommends that the SSE ground motion be developed using either the Electric Power Research Institute (EPRI) Seismicity Owners Group (SOG) project or Lawrence Livermore National Laboratory (LLNL) Probabilistic Seismic Hazard Analyses (PSHA) methodologies (EPRI, 1986; LLNL, 1993), updated through a comprehensive review of the geology, seismology and geophysics of the Site Region. If review of existing data shows a significant change to either the seismic source model or ground motion model (i.e., attenuation relationships), then Regulatory Guide 1.165 recommends that an updated PSHA be performed to develop the SSE ground motion.

For the GGNS ESP Site evaluation, the EPRI SOG methodology was adopted to develop the SSE ground motion. Following review of the data and information developed since publication of the EPRI SOG results in 1986, significant new information regarding seismic sources and earthquake ground motion attenuation in the Site Region was identified. To address new information and approaches for ground motion attenuation modeling, EPRI (2003) developed a new ground motion attenuation model for the central and eastern United States, including the Gulf Coast region. These new relationships were used in the PSHA and are described in Section 2.5.2.3. The seismic source model used to develop the SSE ground motions for the ESP Site was developed following review of data related to active tectonic features in the Site Region (Section 2.5.1).

With two exceptions, the review and analysis shows that all tectonic features in the GGNS Site Region, and the Reelfoot Rift Complex extending north of the Site Region, are adequately characterized by the EPRI SOG seismic source model. The two exceptions are (1) identification of the Saline River source zone as a new source zone, within the Site Region, and (2) revisions in source parameters to the New Madrid Seismic Zone (NMSZ), which lies within the Reelfoot Rift Complex outside of the Site Region. Revisions to the NMSZ include changes in source geometry, maximum magnitude and earthquake recurrence since publication of the 1986 EPRI SOG source model.

Based on the new information on seismic sources and new approaches for ground motion attenuation modeling that have been published since the 1986 EPRI SOG study, the EPRI PSHA methodology has been updated for use in this ESP Application. The EPRI PSHA was updated by revising the seismic source model, adding the ground motion attenuation model developed by EPRI (2003a), and updating the PSHA computational code that was published by EPRI in 1986 (EPRI, 2003b).

Regulatory Guide 1.165 recommends that a PSHA be performed to define the median rock ground motion at the site that has an annual probability of exceedance of not greater than 10^{-5}, and for soil sites, that a site-response analysis be performed to develop the SSE ground motion. The PSHA used to develop the 10^{-5} median rock ground motions is described in Section 2.5.2.2. Because the ESP Site is underlain by soils rather than rock, a site-specific site-response analysis was conducted following the guidelines described in NUREG/CR-6728 (McGuire et al., 2001). The site-specific site-response analysis is described in Section 2.5.2.3 and the data used to develop the soil profile for the site response analysis are presented in Section 2.5.4. The seismic source characterization, site investigation and laboratory analyses, and site-response analysis also are described in Engineering Reports RP-01, RP-02, and Calculation Package CP-01, respectively.

The results of the updated EPRI PSHA were used to obtain the bedrock ground motions for the ESP Site. The results of the PSHA were deaggregated to identify the controlling earthquakes and used to develop a response spectrum for bedrock conditions, scaled at 1 hertz and 10 hertz, that is compatible with the controlling earthquakes. The resulting response spectrum for rock conditions was used in the site response analysis to obtain the SSE response spectrum for free-field conditions at the ground surface. The SSE ground motions for the ESP Site are lower than and are compatible with the Regulatory Guide 1.60 spectrum at all spectral frequencies.

The ESP Site is considered a suitable location for a new facility. The site has negligible risk from surface fault rupture hazards, slope instability, liquefaction-related ground failure, collapse or inundation. The geological and geotechnical conditions are similar to those of the existing GGNS site (of which the ESP Site is a part), which has performed well over the past 20 years. The SSE ground motions for the ESP Site were developed in accordance with the U.S. NRC Regulatory Guide 1.165 methodology, taking into account the most up-to-date information on the locations and characteristics of potential earthquake sources and site-specific seismic wave transmission effects. The SSE ground motions for the Grand Gulf ESP Site are consistent with the U.S. NRC's recommended design spectrum for new nuclear power plants.

3.4.2 References

1. U.S. Nuclear Regulatory Commission (NRC), Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion, Regulatory Guide 1.165, March 1977, Washington, DC.
2. U.S. Atomic Energy Commissioin (USAEC), Design Response Spectra for Seismic Design of Nuclear Power Plants, Regulatory Guide 1.160, Revision 1, December 1973, Washington, DC.
3. Electric Power Research Institute (EPRI), Guidelines for Determining Design Basis Ground Motions – Volume 1: Method and Guidelines for Estimating Earthquake Ground Motion in Eastern North America, EPRI Report TR-102293, 1993a.
4. Electric Power Research Institute (EPRI), Analysis of High-Frequency Seismic Effects, EPRI Report TR-102470, 1993b.
5. U. S. Nuclear Regulatory Commission, McGuire, R. K., W. J. Silva, and C. J. Costantino, NUREG/CR-6728, 2001, Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines, Washington, DC.

TABLE 3.2-1

GASEOUS PATHWAY PARAMETERS

Input Description	Location of Data
Source Term	Table 1.3-7
Population Data	Section 2.1
Meteorological Data	Section 2.1
Consumption Factors	Table 3.2-2
Milk Production	Table 3.2-6
Meat Production	Table 3.2-7
Vegetable Production	Table 3.2-8

TABLE 3.2-2

GASEOUS PATHWAY CONSUMPTION FACTORS

	Maximum Individual Consumption Factors [1]			
Maximum Individual	Vegetables (kg/yr)	Leafy Vegetables (kg/yr)	Milk (L/yr)	Meat (kg/yr)
Adult	520	64	310	110
Teen	630	42	400	65
Child	520	26	330	41
Infant	0	0	330	0

NOTES:

1 Consumption Factors from USNRC Regulatory Guide 1.109, Table E-5.

TABLE 3.2-3A

ANNUAL DOSE TO A MAXIMALLY EXPOSED INDIVIDUAL
FROM GASEOUS EFFLUENTS

Location	Pathway	Dose Rate (mrem/yr) Total Body	Skin	Thyroid
Nearest Residence [1] (N-NE, 0.64 mile)	Plume Exposure	3.26E-01	2.31	3.26E-01
	Inhalation			
	Adult	5.84E-01	5.79E-01	1.92E+00
	Teen	5.90E-01	5.84E-01	2.32E+00
	Child	5.21E-01	5.16E-01	2.63E+00
	Infant	3.00E-01	2.97E-01	2.23E+00
Nearest Garden [1] (E-NE, 0.63 miles)	Vegetable Consumption			
	Adult	1.63E+00	1.54E+00	7.99E+00
	Teen	2.08E+00	1.99E+00	1.00E+01
	Child	3.85E+00	3.73E+00	1.88E+01
Nearest Site Boundary [2] (NE, 0.52 miles)	Inhalation			
	Adult	8.44E-01	8.36E-01	2.85E+00
	Teen	8.53E-01	8.44E-01	3.45E+00
	Child	7.54E-01	7.45E-01	3.93E+00
Nearest Milk Cow [1] (S-SW, 10.0 miles)	Cow Milk			
	Adult	1.88E-02	1.79E-02	1.23E-01
	Teen	2.77E-02	2.66E-02	1.94E-01
	Child	5.35E-02	5.20E-02	3.84E-01
	Infant	9.69E-02	9.47E-02	9.03E-01
Nearest Meat Cow [1] (S, 4.0 miles)	Meat Consumption			
	Adult	2.81E-02	2.75E-02	4.55E-02
	Teen	2.04E-02	2.00E-02	3.30E-02
	Child	3.29E-02	3.24E-02	5.21E-02

NOTES:

1 “Nearest” refers to the location at which the highest radiation dose to an individual from the applicable pathways has been estimated.

2 “Nearest” refers to that site boundary location at which the highest radiation doses due to gaseous effluents have been estimated to occur.

TABLE 3.2-3B

COMPARISON OF MAXIMUM INDIVIDUAL DOSE TO 10 CFR 50, APPENDIX I CRITERIA – GASEOUS PATHWAY

Type of Dose	Design Objective[1]	Point of Evaluation	Calculated Dose
Gaseous Effluents (Noble Gases Only)			
Gamma air dose	10 mrad	Exclusion Area Boundary	8.32E-1 mrad
Beta air dose	20 mrad	Exclusion Area Boundary	5.06 mrad
Total body dose	5 mrem	Exclusion Area Boundary	4.87E-01 mrem
Skin dose	15 mrem	Exclusion Area Boundary	3.37 mrem
Radioiodines and Particulates			
Inhalation Dose	15 mrem	Exclusion Area Boundary	3.93 mrem (thyroid)

NOTES:

1 10 CFR 50, Appendix I

TABLE 3.2-4

ANNUAL POPULATION DOSES - GASEOUS PATHWAY

Pathway	Estimated Doses (Person-rem)	
	Whole Body	Skin (Worst Case Organ)
Plume	5.26E-01	5.03E+00
Ground	1.68E-01	1.97E-01
Inhalation	1.34E+00	1.33E+00
Vegetable Ingestion	4.23E-01	4.15E-01
Cow Milk Ingestion	4.18E-01	4.07E-01
Meat Ingestion	5.26E-01	5.21E-01
Total	3.40E+00	7.90E+00

TABLE 3.2-5

COMPARISON OF MAXIMUM INDIVIDUAL DOSE TO
40 CFR 190 CRITERIA - GASEOUS PATHWAY

Type of Dose (Annual)	Design Objective [1]	Calculated Dose [2]
Whole body dose equivalent	25 mrem	1.34 mrem
Dose to thyroid	75 mrem	4.42 mrem
Dose to skin	25 mrem	4.21 mrem

NOTES:

1 Source 40 CFR 190.
2 Plume + inhalation dose at EAB

TABLE 3.2-6

MILK PRODUCTION BETWEEN 10 AND 50 MILES BY SECTOR

	Milk Production (Liters/yr)			
Sector	10-20	20-30	30-40	40-50
N-NE	0	0	0	0
NE	0	336,480	336,480	336,480
E-NE	0	336,480	672,959	672,959
E	0	854,659	854,659	854,659
E-SE	0	1,722,775	1,722,775	1,722,775
SE	0	343,209	1,009,439	11,776,790
S-SE	0	0	672,959	672,959
S	336,480	336,480	1,009,439	0
S-SW	1,682,399	336,480	336,480	336,480
SW	0	0	0	0
W-SW	0	0	403,776	975,792
W	0	1,514,159	1,514,159	1,514,159
W-NW	0	0	1,514,159	2,691,838
NW	0	0	0	2,691,838
N-NW	0	0	0	672,959
N	0	0	0	0

TABLE 3.2-7

TOTAL MEAT PRODUCTION BETWEEN 0 AND 50 MILES BY SECTOR

	Meat Production (Kg/yr)									
Sector	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
N-NE	0	0	0	0	0	2,814	16,738	39,418	71,871	88,427
NE	0	0	0	0	1,925	5,464	193,395	719,571	1,216,263	1,453,516
E-NE	0	0	0	0	9,603	8,278	331,128	1,052,352	2,045,736	2,045,736
E	0	0	0	0	8,278	11,093	408,240	813,829	1,310,521	1,310,521
E-SE	0	0	0	0	16,556	11,093	408,240	949,909	839,478	977,393
SE	0	0	0	0	0	0	331,128	44,090	60,962	1,118,557
S-SE	0	0	0	0	12,633	77,112	450,336	339,905	220,697	697,843
S	0	0	0	0	8,278	0	450,336	339,905	559,762	27,649
S-SW	0	0	0	0	0	0	450,336	339,905	691,740	597,865
SW	0	0	0	0	0	0	130,135	36,426	220,697	171,461
W-SW	0	0	0	0	0	0	211,425	111,066	326,299	475,942
W	0	0	0	0	2,152	0	43,544	172,232	308,491	314,756
W-NW	0	0	0	0	11,093	0	106,458	243,308	281,007	597,412
NW	0	0	0	0	0	11,093	75,000	75,978	87,794	468,322
N-NW	0	0	0	0	0	0	137,915	120,183	175,316	661,555
N	0	0	0	0	2,814	0	41,391	64,885	110,431	116,101
TOTAL	0	0	0	0	73,332	126,947	3,785,745	5,462,962	8,527,065	11,123,056

TABLE 3.2-8

TOTAL VEGETABLE PRODUCTION BETWEEN 0 AND 50 MILES BY SECTOR

	Total Vegetable Production (kg/yr)									
Sector	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
N-NE	0	0	0	0	0	10,623	27,549	34,568	75,896	84,986
NE	0	0	0	0	0	0	218,838	76,487	648,016	1,202,548
E-NE	0	0	0	0	0	0	276,204	297,450	1,062,322	1,062,322
E	0	0	42,493	84,986	0	127,479	448,572	67,223	84,986	1,064,989
E-SE	0	0	0	42,493	0	0	342,248	84,986	389,732	2,318,642
SE	0	0	0	0	0	6,350	193,523	0	241,112	858,459
S-SE	0	0	0	0	0	15,501	0	106,232	107,259	104,160
S	0	0	0	0	0	1,270	21,247	59,490	42,493	194,374
S-SW	0	0	0	0	0	3,171	21,247	63,739	127,479	63,739
SW	0	0	0	0	0	0	42,493	63,739	0	89,235
W-SW	0	0	0	0	0	0	25,496	212,465	127,479	169,971
W	0	0	0	0	0	25,496	0	169,971	254,957	254,957
W-NW	0	0	0	0	0	0	0	169,971	637,390	212,465
NW	0	0	0	0	0	0	0	0	169,971	37,165
N-NW	0	0	0	0	0	0	84,986	0	0	844,167
N	10,623	0	0	0	0	0	424,929	424,929	254,954	0
TOTAL	10,623	0	42,493	127,479	0	189,889	2,127,330	1,831,251	4,224,047	8,562,180

TABLE 3.2-9

DOSE TO BIOTA FROM GASEOUS EFFLUENTS

Organism	Internal Dose (mrem/yr)	External Dose (mrem/yr)
Fish	N/A	N/A
Invertebrate	N/A	N/A
Algae	N/A	N/A
Muskrat	8.26E-02	2.45
Raccoon	8.26E-02	1.96
Heron	8.26E-02	1.67
Duck	8.26E-02	2.69

TABLE 3.3-1

COMPARISON OF REACTOR TYPES FOR LIMITING OFF-SITE DOSE CONSEQUENCES

Accident	Reactor Type	Affected Organ	EAB Dose TEDE (rem)	LPZ Dose TEDE (rem)	Guideline[1] TEDE (rem)
Main Steam Line Break					
Accident-initiated Iodine Spike	AP1000		0.68	0.87	2.5
Pre-existing Iodine Spike	AP1000		0.60	0.24	25
Max Equilibrium Iodine Activity	ABWR		0.05	<0.05	2.5
Pre-existing Iodine Spike	ABWR		1.0	<1.0	25
Reactor Coolant Pump Locked Rotor	AP1000		2.14	0.34	2.5
Control Rod Ejection Accident	AP1000		2.57	0.96	6.3
Control Rod Drop Accident	ABWR		Negligible	Negligible	6.3
Steam Generator Tube Rupture					
Accident-initiated Iodine Spike	AP1000		1.28	0.14	2.5
Pre-existing Iodine Spike	AP1000		2.57	0.20	25
Small Line Break					
	AP1000		1.11	0.17	2.5
	ABWR		0.31	<0.31	2.5
Loss of Coolant Accident					
	AP1000		21.20	5.64	25
		Thyroid	0.71	2.03	300 Rem
	ABWR [2]	Whole Body	1.54E-02	2.72E-02	25 rem
	ACR-700		5.5	4.6	25
Fuel Handling Accident					
	AP1000		2.05	0.34	6.3
	ABWR		0.30	<0.30	6.3

NOTES:

1. 25 rem is 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).
2. ABWR LOCA guideline based on 10CFR100 limits due to use of TID-14844 source term.

TABLE 3.3-2

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	0.68	----*
0 to 8 hour	----*	0.36
8 to 24 hour	----*	0.22
24 to 96 hour	----*	0.29
96 to 720 hours	----*	0
Total	0.68	0.87

NOTES:

*Dose not applicable

TABLE 3.3-3

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	0.60	----*
0 to 8 hour	----*	0.14
8 to 24 hour	----*	0.04
24 to 96 hour	----*	0.06
96 to 720 hours	----*	0
Total	0.60	0.24

NOTES:

*Dose not applicable

TABLE 3.3-4

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 3.3-5

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

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	0.05	----*
0 to 8 hour	----*	<0.5
8 to 24 hour	----*	
24 to 96 hour	----*	
96 to 720 hours	----*	
Total	0.05	<0.5

NOTES:

*Dose not applicable

TABLE 3.3-6

ABWR MAIN STEAM LINE BREAK OUTSIDE CONTAINMENT - 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	1.0	----*
0 to 8 hour	----*	<1.0
8 to 24 hour	----*	
24 to 96 hour	----*	
96 to 720 hours	----*	
Total	1.0	<1.0

NOTES:

*Dose not applicable

TABLE 3.3-7

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	2.14	----*
0 to 8 hour	----*	0.34
8 to 24 hour	----*	
24 to 96 hour	----*	
96 to 720 hours	----*	
Total	2.14	0.34

NOTES:

*Dose not applicable

TABLE 3.3-8

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	2.57	----*
0 to 8 hour	----*	0.79
8 to 24 hour	----*	0.14
24 to 96 hour	----*	0.02
96 to 720 hours	----*	0.00
Total	2.57	0.96

NOTES:

*Dose not applicable

TABLE 3.3-9

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	1.28	----*
0 to 8 hour	----*	0.10
8 to 24 hour	----*	0.04
24 to 96 hour	----*	0
96 to 720 hours	----*	0
Total	1.28	0.14

NOTES:

*Dose not applicable

TABLE 3.3-10

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	2.57	----*
0 to 8 hour	----*	0.18
8 to 24 hour	----*	0.01
24 to 96 hour	----*	0
96 to 720 hours	----*	0
Total	2.57	0.20

NOTES:

*Dose not applicable

TABLE 3.3-11

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	1.11	----*
0 to 8 hour	----*	0.17
8 to 24 hour	----*	0
24 to 96 hour	----*	0
96 to 720 hours	----*	0
Total	1.11	0.17

NOTES:

*Dose not applicable

TABLE 3.3-12

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 3.3-13

ABWR SMALL LINE BREAK OUTSIDE CONTAINMENT

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	0.31	----*
0 to 8 hour	----*	<0.31
8 to 24 hour	----*	0
24 to 96 hour	----*	0
96 to 720 hours	----*	0
Total	0.31	<0.31

NOTES:

*Dose not applicable

TABLE 3.3-14

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[1]	21.20	----*
0 to 8 hour	----*	5.21
8 to 24 hour	----*	0.18
24 to 96 hour	----*	0.14
96 to 720 hours	----*	0.11
Total	21.20	5.64

NOTES:

*Dose not applicable

1. Two-hour period with greatest EAB dose shown. LOCA based on Regulatory Guide 1.183.

TABLE 3.3-15

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 3.3-16

ABWR DESIGN BASIS LOSS OF COOLANT ACCIDENT[1]

	Exclusion Area Boundary Dose		Low Population Zone Dose	
Time	Thyroid (rem)	Whole Body (rem)	Thyroid (rem)	Whole Body (rem)
0 to 2 hour	7.11E-01	1.54E-02	----*	----*
0 to 8 hour	----*		1.52E-01	4.90E-03
8 to 24 hour	----*		1.11E-01	4.45E-03
24 to 96 hour	----*		5.78E-01	8.05E-03
96 to 720 hours	----*		1.19E+00	9.75E-03
Total	7.11E-01	1.54E-02	2.03E+00	2.72E-02

NOTES:

*Dose not applicable

1. LOCA based on Regulatory Guide 1.3 and TID-14844.

TABLE 3.3-17

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 3.3-18

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	5.5	0.8
2 to 8 hour	----*	1.5
8 to 24 hour	----*	1.4
24 to 96 hour	----*	0.5
96 to 720 hours	----*	0.4
Total	5.5	4.6

NOTES:

*Dose not applicable

TABLE 3.3-19

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 3.3-20

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	2.05	----*
0 to 8 hour	----*	0.34
8 to 24 hour	----*	----*
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
Total	2.05	0.34

NOTES:

*Dose not applicable

TABLE 3.3-21

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 3.3-22

ABWR 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	0.304	----*
0 to 8 hour	----*	<0.304
8 to 24 hour	----*	----*
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
Total	0.304	<0.304

NOTES:

1. Activity is based on a 24-hour shutdown before fuel movement begins.
2. RG 1.183 assumptions are used for values of gap fractions, pool decontamination factors, and gap iodine species.