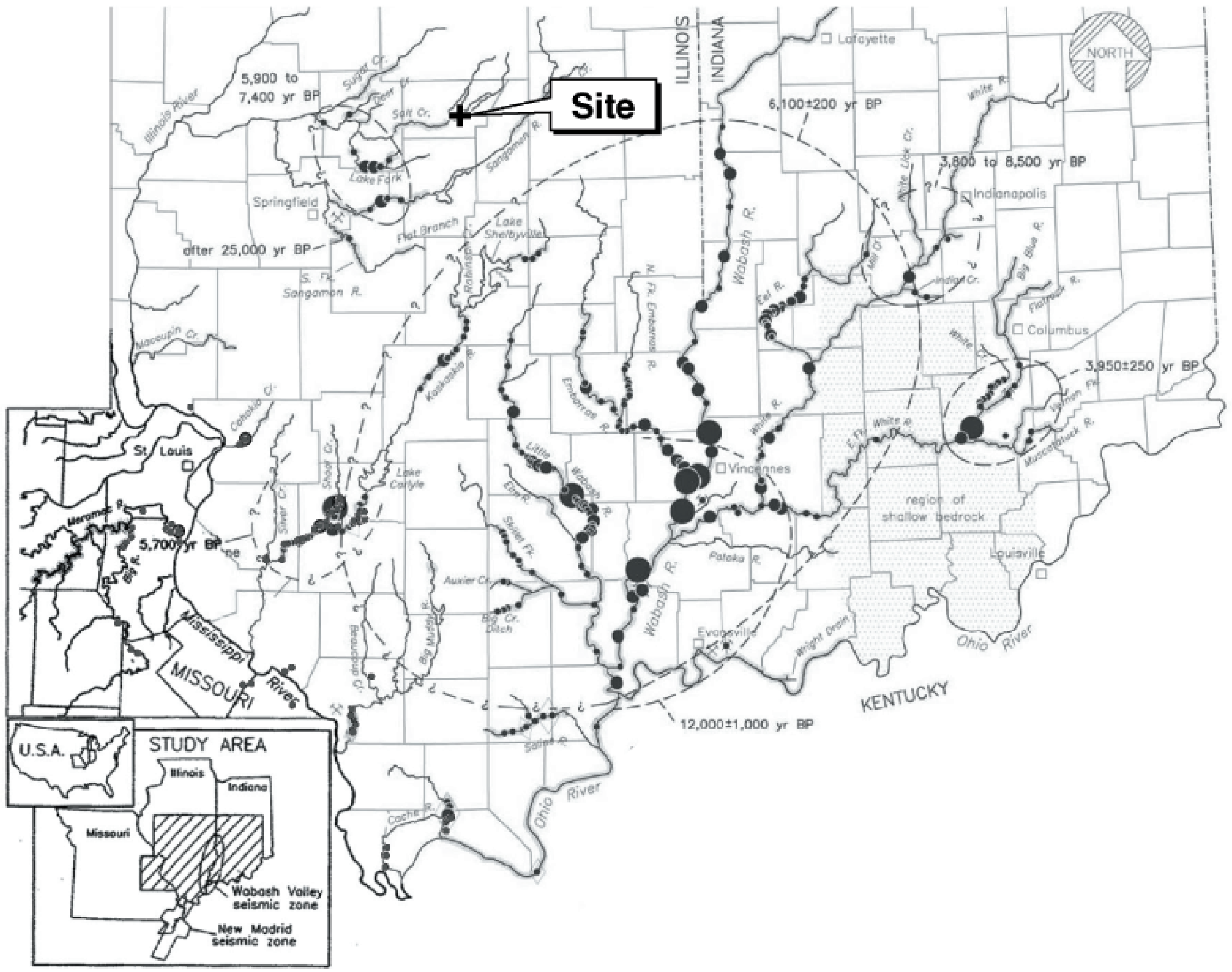
Site Safety Analysis Report for the EGC Early Site Permit

Figure 2.5-6

Location of Paleoliquefaction Sites in Southern Illinois and Indiana



Legend

- Surveyed stream banks
- Stream
- City
- Surveyed pit
- Maximum dike width >0.5m
- Maximum dike width >0.15 - 0.5m
- Maximum dike width <0.15m
- Approximate limit of liquefaction for a paleo-earthquake
- Site with one to tens of dikes, prehistoric Holocene and latest Pleistocene in age.
- Young (1811-127) and ancient dikes along Saline River; unknown ages of dikes along Cache and Big Muddy Rivers; young dikes along Kaskaskia River; young dikes along Ohio River.

Note:
See figure 2.1-15 (Appendix B) for sources of data.

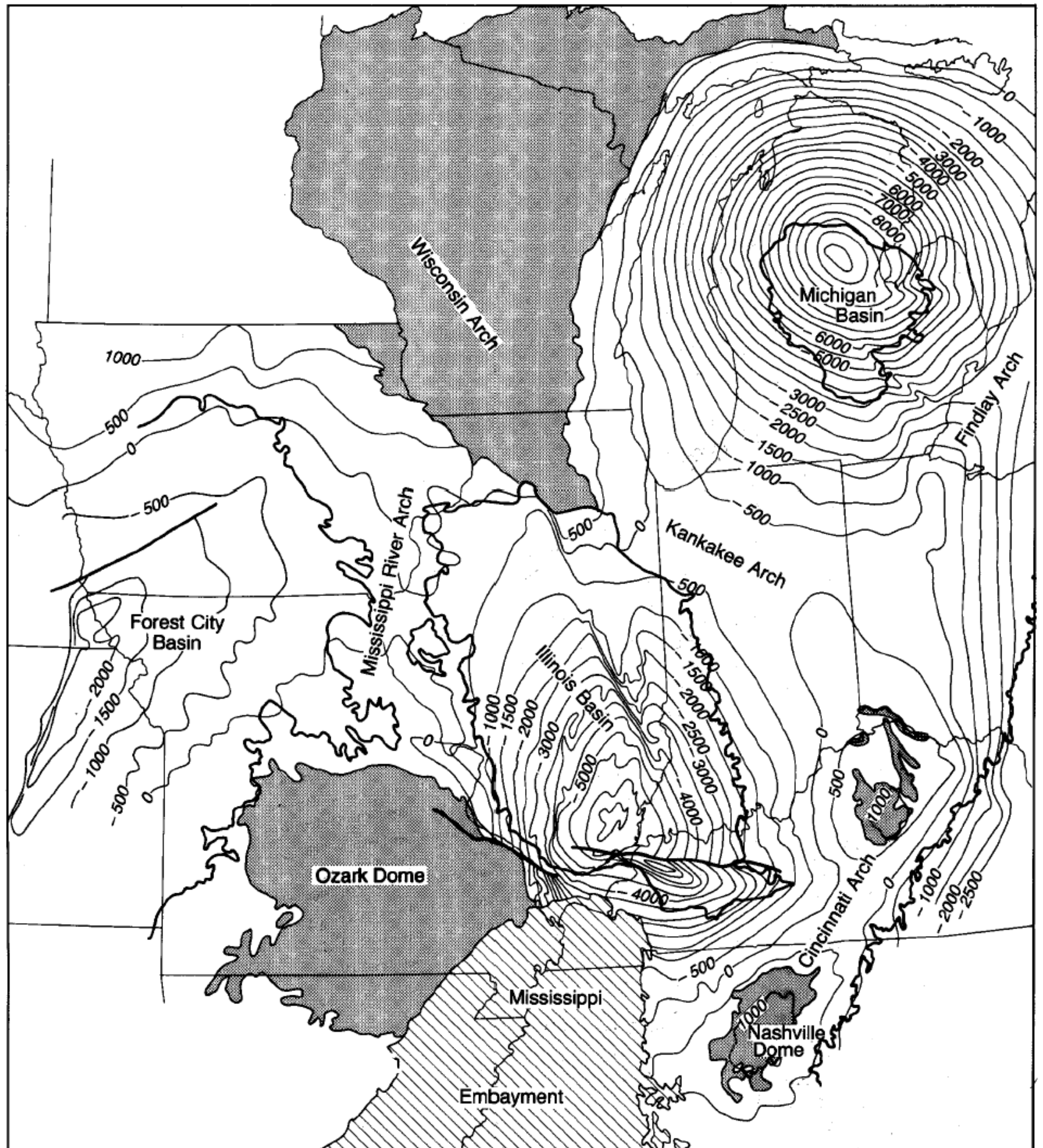
N

0 25 50 75 100 KM

0 30 60 miles

Site Safety Analysis Report for
the EGC Early Site Permit

Figure 2.5-7
Regional Structural Setting in Illinois



Legend

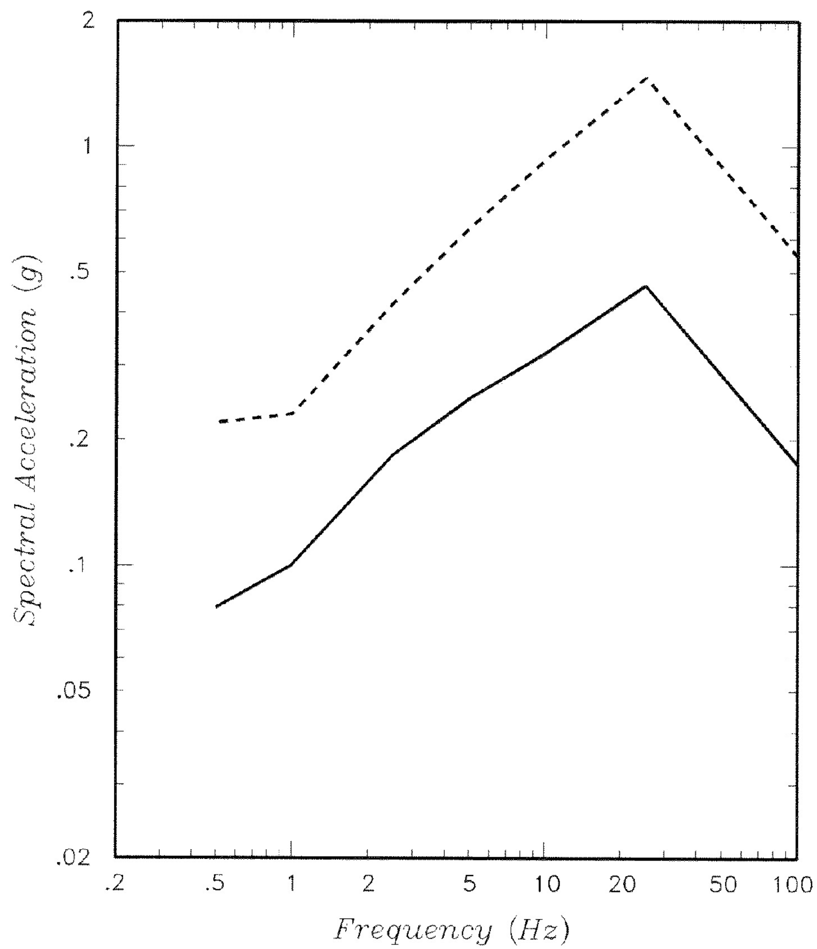
- 500 — Elevation (ft) of the top of Trenton Ls or equivalents
- Outcrop of strata below top of Trenton
- Paleozoic rock overlapped by Mesozoic and younger strata in Mississippi Embayment
- Limit of Pennsylvanian

Data Source
Nelson (1995)

N

0 100 mi
0 200 km

Figure 2.5-8
Uniform Hazard Spectra for Rock



Legend

mean e-4

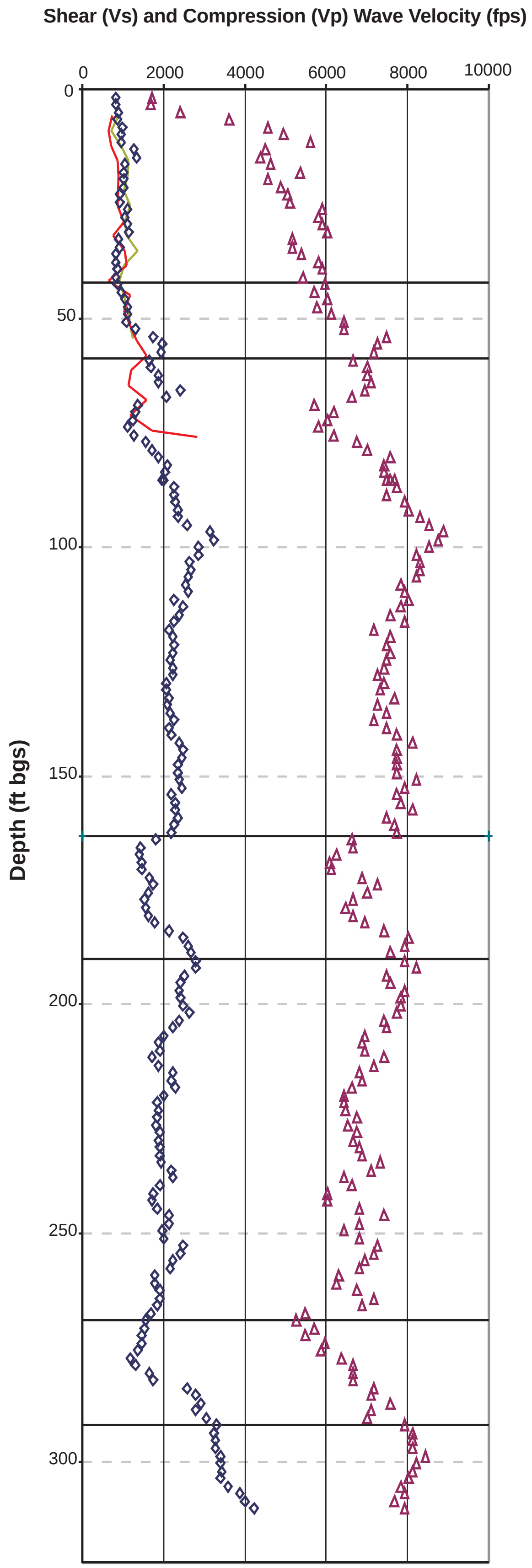
mean e-5

Notes:

1 Spectra established at hard rock level before modification for site effects.

2 See Section 4.1 of Appendix B for additional discussion.

Not to Scale



Unit	Depth (ft. bgs)	Soil Properties - EGC ESP Site *(and CPS Site)*[1] Moist Unit Wt. (pcf)	Moist. Cont. (%)	LL	PL	PI	Comments
Loess & Wisconsinan Till	0 - 42	131 *(131)*	16 *(16)*	35 *(25)*	14 *(14)*	14 *(11)*	Perched water table at ~5 ft bgs
Interglacial	42 - 59	116 *(132)*	39 *(17)*	40 *(26)*	26 *(13)*	14 *(13)*	
Illinoian Till	59 - 163	148 *(147)*	8 *(9)*	18 *(18)*	9 *(11)*	9 *(7)*	
Lacustrine	163-190	133 *(140)*	13 *(11)*	28 *(19)*	11 *(12)*	17 *(7)*	
Pre-Illinoian Till	190 - 269	138 *(137)*	14 *(14)*	29 *(27)*	14 *(14)*	15 *(13)*	
Pre-Illinoian Alluvial/ Lacustrine	269 - 292	N.A. *(N.A.)*	23 *(N.A.)*	48 *(N.A.)*	17 *(N.A.)*	29 *(N.A.)*	
Bedrock	292-322	N.A. *(N.A.)*	N.A. *(N.A.)*	N.A. *(N.A.)*	N.A. *(N.A.)*	N.A. *(N.A.)*	Weathered rock contact at 292 ft bgs

Site Safety Analysis Report for the EGC Early Site Permit

Figure 2.5-9

Shear and Compression Wave Velocities and other Test Results

Legend

◇ Vs at B-2
△ Vp at B-2
— Vs at CPT-02
— Vs at CPT-04

Notes:

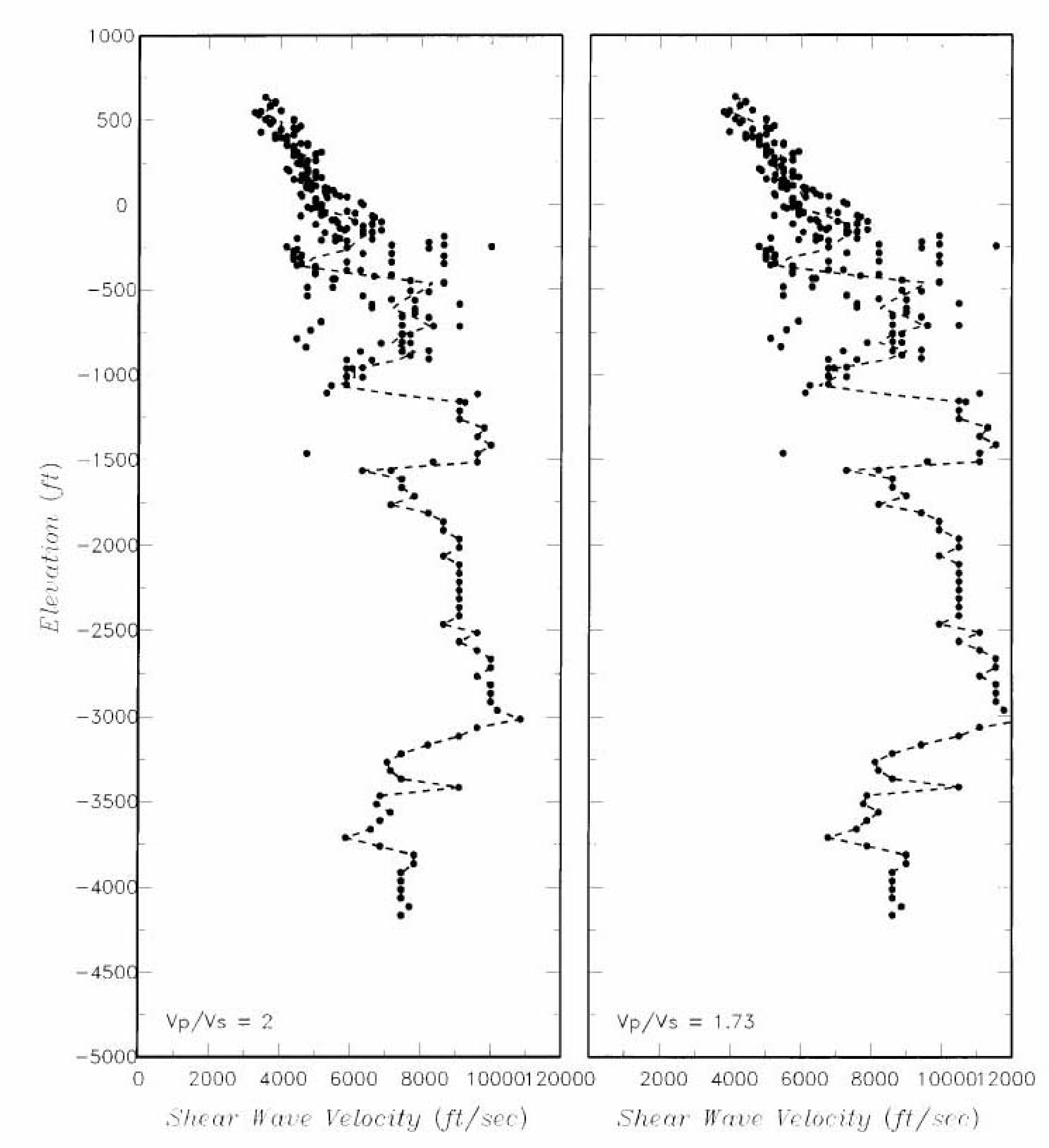
1 Soil properties shown are the arithmetic mean values for all available soil sample results for each stratigraphic unit. Top number is the mean value of applicable EGC ESP Site Investigation data.

(Italic) = Mean value of applicable data from CPS Site P-Series Soil Samples, as reported in Section 2.5 of CPS, 2002

N.A. = Results not available

Site Safety Analysis Report for
the EGC Early Site Permit

Figure 2.5-10
Velocity Data from Regional Deep Boreholes



Legend

Notes:

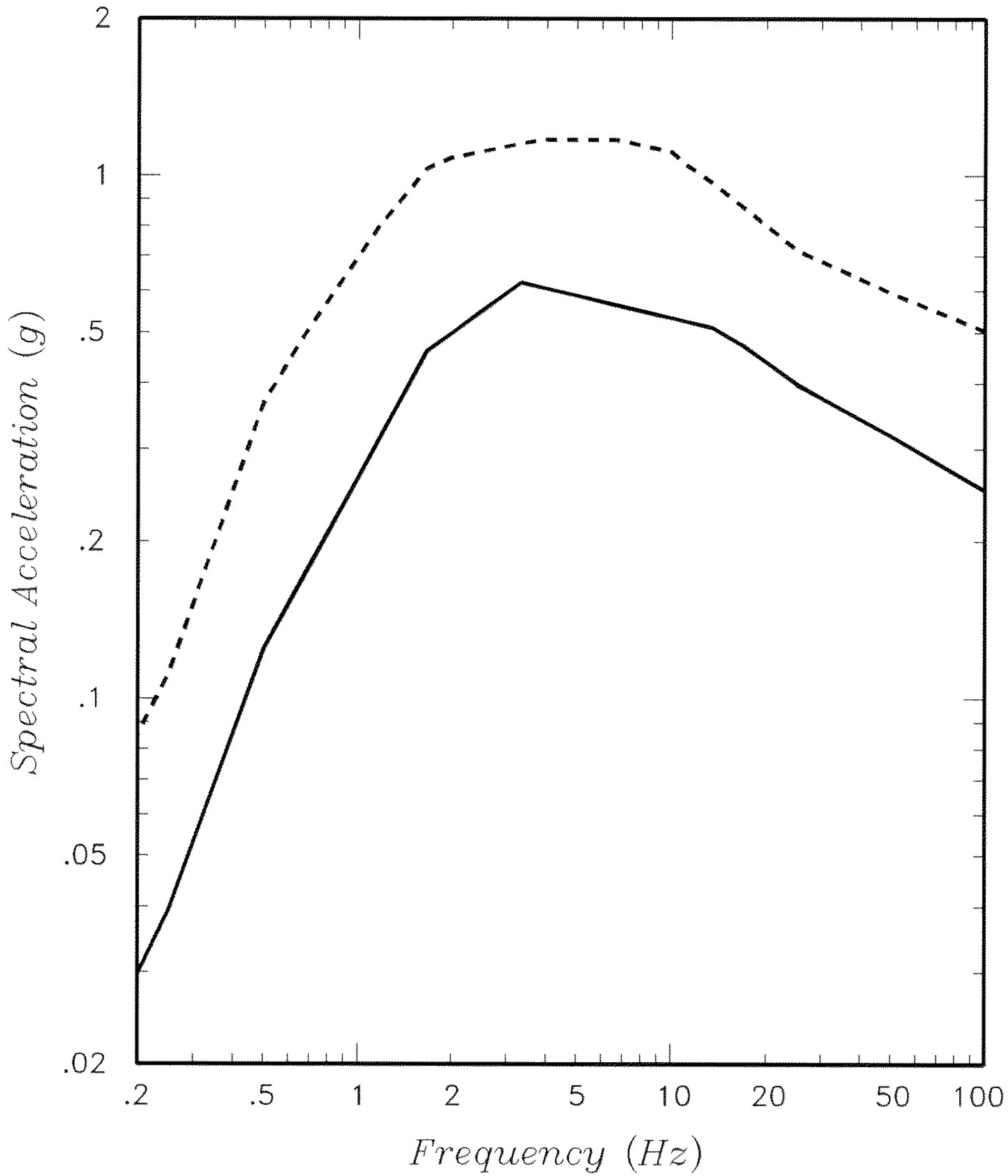
1 Shear wave velocity values estimated from borehole compression wave velocity surveys in area.

2 See Section 4.2 of Appendix B for additional discussion.

Not to Scale

Site Safety Analysis Report for
the EGC Early Site Permit

Figure 2.5-11
Uniform Hazard Spectra at Ground Surface-Horizontal



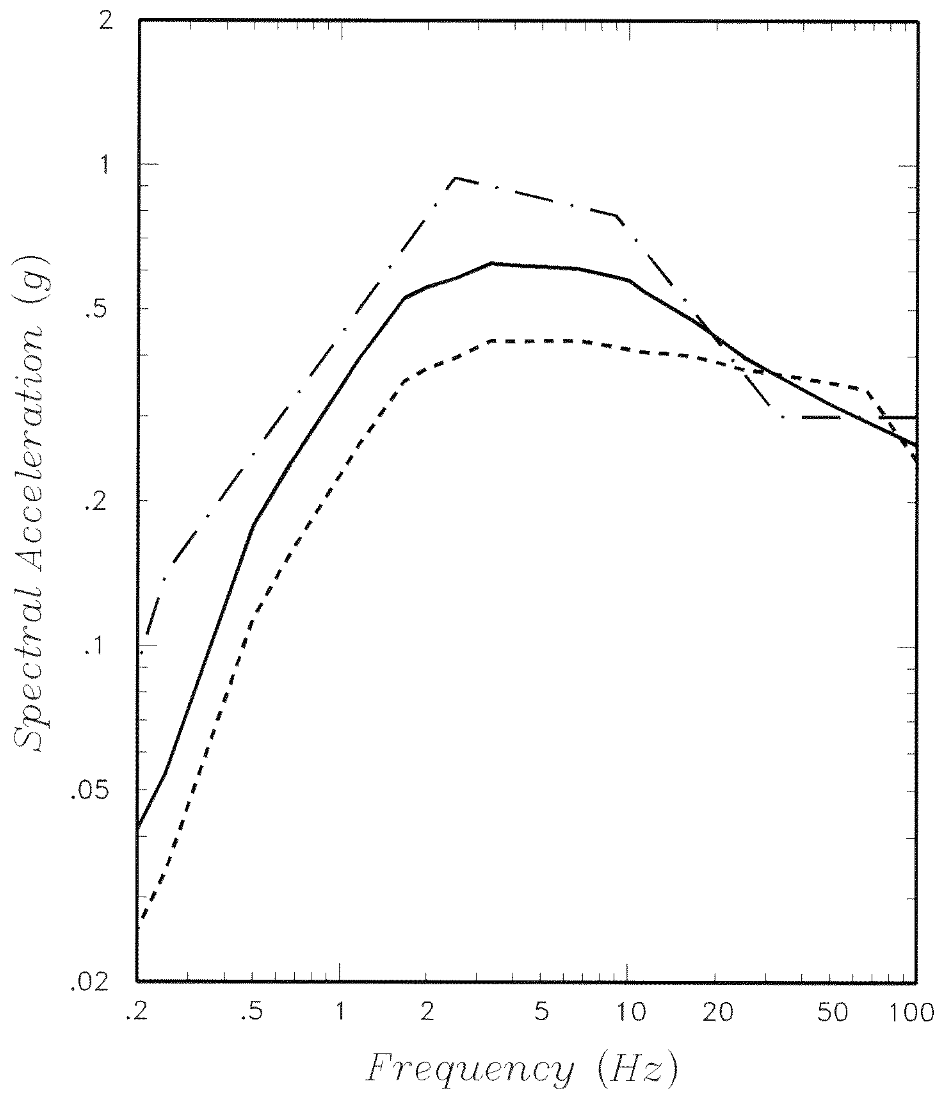
Legend

- Mean e-4 Envelope
- Mean e-5 Envelope

Site Safety Analysis Report for
the EGC Early Site Permit

Figure 2.5-12

Risk Consistent Design Response Spectra



Legend

- Horizontal DRS
- Vertical DRS
- RS 1.60 scaled to 0.3g PGA

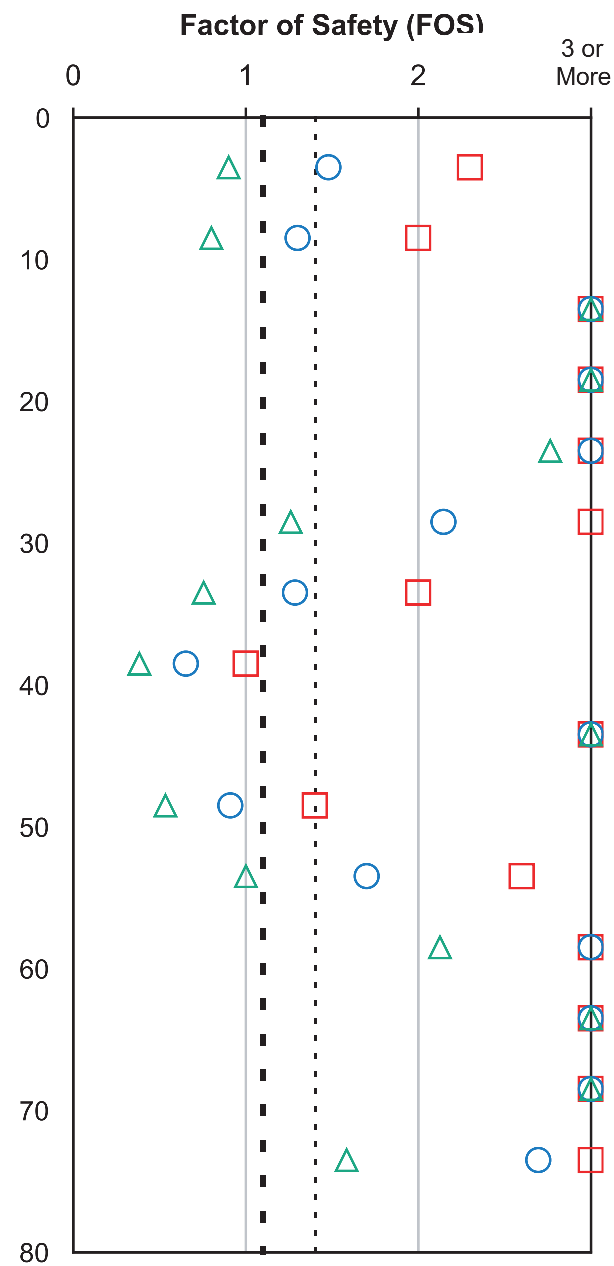
Not to Scale

Site Safety Analysis Report for
the EGC Early Site Permit

Figure 2.5-13

Factor of Safety Against Liquefaction with Depth Borehole B-1

Depth (ft)	USCS		Liquefaction Concern?
3.5	CL		No - Clay
8.5	CL		No - Clay
13.5	CL		No
18.5	CL		No
23.5	SP		No
28.5	SP		FOS < 1.4
33.5	CL		No - Clay
38.5	SW		**FOS < 1.1**
43.5	ML		No
48.5	CL		No - Clay
53.5	ML		**FOS < 1.1**
58.5	ML		No
63.5	ML		No
68.5	ML		No
73.5	ML		No



Legend

- = FOS < 1.1: Liquefaction possible
- = FOS < 1.4: Seismic pore pressures possible
- M=6.5, PGA=0.35
- M=7.75, PGA=0.25
- FOS = 1.1
- FOS = 1.4

Not to Scale

CHAPTER 3

Site Safety Assessment

This chapter provides information on the potential radiological effluents, thermal discharges, major accident doses, and a summary assessment of conformance with 10 CFR 100 - Reactor Site Criteria Subpart B requirements for the EGC ESP Site and Facility. The presentation and discussion of bounding plant radiological effluents, thermal discharges, and accidental radioactive releases also serve as input for the development of the environmental impact analyses that are presented in the Environmental Report.

3.1 Radiological Effluents

Radioactive effluents consist of gaseous, liquid, and solid waste materials that are generated as a normal by-product of nuclear power reactor operations. These radioactive materials are collected, processed, placed in interim storage, discharged in a controlled manner to the local site environment, or transported off-site for long-term storage or disposal. Radioactive waste management systems are provided as part of the EGC ESP Facility design to handle these materials in a manner that minimizes releases to the environment and maintains exposure to the general population and plant personnel during normal plant operation and maintenance as low as reasonably achievable (ALARA).

3.1.1 Gaseous Effluents

The gaseous waste management system of the EGC ESP Facility will control, collect, process, store, and dispose of potentially radioactive gases during plant operation including startup, normal operation, shutdown, refueling, and anticipated operational occurrences. The normal gaseous effluents are released from the plant to the environment via waste gas processing systems designed to minimize the releases to and the impact on the environment. Potentially radioactive gases are also present in the plant buildings as a result of process leakage. These gases are released to the environment via the building ventilation systems. The release of radioactive gaseous effluents from the plant is controlled and monitored to within the regulatory limits specified in 10 CFR 20 and 10 CFR 50 Appendix I.

The bounding quantity of radioactive gases released from the gaseous waste management and the building ventilation systems used in the evaluation of the EGC ESP Site is shown on Table 1.4-3. The gaseous radioactive effluent concentrations are determined based on a composite of the highest activity content of the individual isotopes anticipated to be released from the alternative reactor designs as presented in Section 1.4, Plant Parameters Envelope and the site characteristic average annual atmospheric dilution factor given in Section 2.3.5.

Compliance with the isotopic limits of 10 CFR 20 is based on demonstrating that the bounding average annual concentrations of radioactive material released in the gaseous effluents at the boundary of the restricted area do not exceed the values specified in Table 2 of Appendix B to 10 CFR 20.

The comparison of the Table 1.4-3 releases with the 10 CFR 20 effluent concentration limits is provided in Table 3.1-1.

3.1.1.1 Safety Function

There is no safety function associated with the normal radioactive gaseous effluents.

3.1.1.2 Estimated Doses

The NRCDose computer program (Bland, 2000) is used to model the isotopic activity release and the dilution and uptake of radioactivity via the potential pathways of exposure. The NRCDose code package is a PC-based, software interface for the LADTAP II, GASPAR II, and XOQDOQ program that operates on Microsoft Windows platforms. The methodology contained in the GASPAR II program (NUREG/CR-4653) is used to determine the gaseous

pathway doses (Strange, et. al., 1987). This program implements the radiological exposure models described in Regulatory Guide 1.111, Revision 1 (USNRC, 1977b) for radioactivity releases in gaseous effluent. 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. Doses are calculated for both the maximum individual and for the population and are summarized for each pathway by age group and organ.

Dose rate estimates were made 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.

The parameters used in determining the gaseous pathways doses are provided in Table 3.1-2.

Table 3.1-3 provides the estimated total-body and critical organ doses for the identified gaseous effluent pathways.

Compliance with the dose limits of 10 CFR 20 and 10 CFR 50 Appendix I is demonstrated in Table 3.1-4.

3.1.2 Liquid Effluents

The liquid waste management system of the EGC ESP Facility will control, collect, process, store, and dispose of, as required, potentially radioactive liquids during plant operation including startup, normal operation, shutdown, refueling, and anticipated operational occurrences. This system will typically be operated in a manner that minimizes the release of radioactivity the environment. Normal discharges will be via the existing discharge plume of the CPS.

The CPS Facility currently does not routinely discharge radioactive liquid wastes into the Clinton Lake. It is likely that the EGC ESP Facility will also not routinely discharge radioactive liquid wastes to the environment. However, to provide for operating flexibility a bounding assessment is performed to demonstrate the capability of complying with the 10 CFR 20 and 10 CFR 50, Appendix I regulatory requirements at the EGC ESP Site.

Compliance with the 10 CFR 20 criteria is based on demonstrating that average annual concentrations of radioactive material released in the liquid effluents at the boundary of the restricted area do not exceed the values specified in Table 2 of Appendix B to 10 CFR 20.

The bounding average annual quantity of radioactivity projected to be released is shown on Table 1.4-4. The liquid waste effluent concentrations are determined based on a composite of the highest activity content of the individual isotopes from the alternative reactor designs as presented in Section 1.4, Plant Parameters Envelope. These releases will bound those for

any selected reactor design. The discharge flow is taken as the minimum dilution flow of 2,400 gpm from Table 1.4-1 (Section 10.2.1).

The comparison of the Table 1.4-4 releases with the 10 CFR 20 effluent concentration limits is provided in Table 3.1-5.

3.1.2.1 Safety Function

There is no safety function associated with the normal radioactive liquid effluents.

3.1.2.2 Estimated Doses

The NRCDose computer program (Bland, 2000) is used to model the isotopic activity release, the dilution and uptake of radioactivity via the potential pathways of exposure. The NRCDose code package is a PC-based, software interface for the LADTAP II, GASPAR II, and XOQDOQ programs, which operate on Microsoft Windows platforms. The liquid pathway parameters used for the maximum dose uses the LADTAP II computer program (NUREG/CR-4013). This program implements the radiological exposure models described in Regulatory Guides 1.109, Revision 1 (USNRC, 1977a) and 1.113, Revision 1 (USNRC, 1978) for radioactivity releases in liquid effluent. Doses are calculated for both the maximum individual and for the population and are summarized for each pathway by age group and organ (Strange et. al., 1986).

The pertinent parameters used in determining the maximum dose are provided in Table 3.1-6.

Population doses are not determined due to the radioactive liquid effluents. There are no municipal or industrial water intakes within 50 mi downstream of the plant. Commercial fishing is not allowed on Salt Creek but is allowed on the Sangamon River. Salt Creek joins the Sangamon River 56 mi west of the plant. Therefore, the only possible aquatic pathway to man is due to sport fishing. This is not considered to be a significant contribution to the annual population dose within 50 mi.

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

- Eating fish or invertebrates caught near the point of discharge;
- Using the shoreline for activities such as sunbathing or fishing; and
- Swimming and boating on the lake near the point of discharge.

Table 3.1-7 provides a listing of the calculated doses and demonstrates that the EGC ESP Site satisfies the dose requirements of 10 CFR 20 and 10 CFR 50 Appendix I.

3.1.3 Solid Waste

The solid waste management system of the EGC ESP Facility will control, collect, handle, process, package, and temporarily store prior to off-site shipping the wet and dry solid radioactive waste materials generated during normal plant operations. The solid waste materials may consist of wet waste sludges, dewatered resins, and contaminated solids such as HEPA and cartridge filters, rags, paper, clothing, tools, and equipment. Shipments of

solid radwaste material will be made periodically between the EGC ESP Site and the permanent waste disposal facility.

The average yearly quantity to be shipped per Table 1.4-1, Section 11.2.3 is estimated to be 15,087 ft^3/yr. The maximum curie content of the shipped waste per Table 1.4-5 is estimated at 5,100 curies. Table 1.4-5 provides a compilation of the principal radionuclides that may be present in the solid waste. The waste will be packaged and shipped in accordance with the applicable regulations provided in 10 CFR 71 and 49 CFR 173.

3.1.3.1 Safety Function

There is no safety function associated with the solid waste management system.

3.2 Thermal Discharges

3.2.1 Normal Plant Heat Sink

The normal plant heat sink provides cooling water for condensing turbine exhaust steam and cooling the turbine auxiliaries in a light water reactor plant, provides helium cooling in a gas-cooled reactor plant, and provides the cooling water for other non-safety plant components.

3.2.1.1 Description

The normal plant heat sink provides the cooling water required for the non-safety related station components during normal operation. The cooling water source for the normal plant heat sink is from cooling tower(s). Circulating water and service water pumps take suction from the cooling tower basin(s) and supply water to the components for cooling. The heated water from the components is returned to the cooling tower(s) for rejection of the heat to the atmosphere. The cooling systems that use water from the normal plant heat sink are described in the reactor manufacturer's standard design documentation and this SSAR's content is limited to a description of the supply and discharge of the cooling water external to the standard plant package.

Chemical treatment of the cooling water with biocides, dispersants, molluskicides, and scale inhibitors will be required on a periodic basis. The chemicals used will be subject to review and approval for use by the Illinois Environmental Protection Agency (IEPA). The total residual chemical concentrations in the discharges to Clinton Lake will be subject to discharge permit limits established by the IEPA.

Blowdown, from the discharge of the circulating water and service water system pumps, is used to control the concentration of impurities in the water due to evaporation in the cooling tower(s).

3.2.1.2 Discharge Flows, Heat Loads, and Locations

The maximum discharge flow from the normal cooling system to the cooling tower(s) is 1,200,000 gpm during normal operation.

The maximum heat load on the normal heat sink cooling system is 15.08 E+09 Btu/hr during normal operation.

The discharge from cooling tower blowdown is normally 12,000 gpm with a maximum flow of 49,000 gpm. The temperature of the blowdown discharge to the existing CPS Facility discharge flume is 100°F maximum. The 100°F discharge temperature is based on a maximum wet bulb temperature of 85°F and a maximum cooling tower design approach of 15°F. The maximum wet bulb temperature that is exceeded less than 1 percent of the time is 77.2°F and the maximum wet bulb temperature based on weather data will be 84.7°F with corresponding blowdown temperatures of 92.2°F and 99.7°F with the maximum cooling tower approach of 15°F. The blowdown constituents and concentrations expected are listed below:

Constituent	Concentration (ppm)
Chlorine Demand	10.1
Free Available Chlorine	0.5
Copper	6
Iron	3.5
Zinc	0.6
Phosphate	7.2
Sulphate	3500
Total Dissolved Solids	17,000
Total Suspended Solids	150

3.2.1.3 Water Supply

The makeup water supply for the normal heat sink cooling tower(s) will be taken from Clinton Lake. Pumps for makeup water will be located in a new intake structure located next to the existing CPS Facility intake structure. The intake water will pass through bar racks to remove large debris, and traveling screens to remove smaller debris, before entering the pump suction chamber. The approach velocity to the intake will be limited to a maximum velocity of 0.5 ft/sec at the normal lake level elevation of 690 ft above msl. Trash collection baskets will be provided to collect trash from the screen wash water, for approved disposal, before the wash water is discharged to the lake. Strainers will be provided on the makeup pump discharges and the strainer backwash water is returned to the lake. A combination wet/dry surface cooling tower may be used to reduce makeup water consumption, if required, to match water demand with the available water supply.

The Normal Heat Sink intake structure will be a common structure providing the Ultimate Heat Sink (UHS) intake described in SSAR section 3.2.2.3.

3.2.1.4 Safety Functions

The Normal Plant Heat Sink has no safety function and is not required for shutdown or accident mitigation.

3.2.1.5 Instrumentation

Temperature elements are provided in the cooling tower blowdown discharge pipe to monitor the discharge temperature.

3.2.2 Ultimate Heat Sink

The UHS provides safety-related cooling water, if required, to the various reactor plant cooling water systems and components that are used for accident mitigation, safe shutdown, and to maintain the unit in a safe shutdown condition. Some of the reactor plants being considered for the EGC ESP Facility utilize passive cooling systems which may not require a water-based UHS system. The UHS function for the EGC ESP Facility may be

provided by safety-related cooling towers that will provide the heat rejection from the safety-related cooling water systems. The safety-related cooling water system, if utilized, will be referred to as the essential service water (ESW) system.

3.2.2.1 Description

The ESW pumps water from the ESW cooling tower basins through the components cooled by the system and returns the water to the cooling towers for heat rejection to the atmosphere. Normal makeup water for the ESW cooling tower basins is supplied from Clinton Lake. Pumps for normal ESW makeup water will be located in a new intake structure located next to the existing CPS Facility intake structure. Blowdown, from the discharge of the ESW system pumps, is used to control the concentration of impurities in the water due to evaporation in the cooling tower.

The cooling systems that use water from the ESW are described in the reactor manufacturer's standard design documentation and this SSAR's content is limited to a description of the supply and thermal discharge of the cooling water external to the standard plant package.

The ESW system will consist of a minimum of two redundant cooling divisions (trains) such that adequate cooling is provided with a single failure in accordance with Regulatory Guide 1.27. The quantity of pumps in each division (train) and the number of divisions of safety related cooling water pumps, heat exchangers, and piping will be provided to satisfy the requirements of the reactor manufacturer's standard plant design.

The ESW system design basis will include safe shutdown, earthquake, tornado, flooding, missiles from equipment failure, and the effects of pipe rupture, in accordance with Regulatory Guide 1.27 (USNRC, 1976).

Chemical treatment of the safety related cooling water system with biocides, dispersants, molluskicides, and scale inhibitors will be required on a periodic basis. The chemicals used will be subject to review and approval for use by the IEPA. The total residual chemical concentrations in the discharges to Clinton Lake will be subject to limits that will be established by the IEPA.

3.2.2.2 Discharge Flows, Heat Loads, and Locations

The maximum discharge flow from the ESW cooling system to the cooling tower(s) is 26,125 gpm during normal operation and 52,250 gpm during shutdown.

The maximum heat load on the ESW cooling system is 225 E+06 Btu/hr during normal operation and 411.4 E+06 Btu/hr during shutdown.

The discharge from cooling tower blowdown is 144 gpm normal with a maximum blowdown of 700 gpm. The temperature of the blowdown discharge to the existing CPS Facility discharge flume is 95°F maximum.

3.2.2.3 Water Supply

A safety class supply of makeup water for the ESW cooling towers is provided from Clinton Lake using redundant makeup pumps. Pumps for makeup water will be located in a new intake structure located next to the existing CPS Facility intake structure. The intake water

will pass through bar racks to remove large debris, and traveling screens to remove smaller debris before entering the pump suction chamber. The approach velocity to the intake will be limited to a maximum velocity of 0.5 ft/sec at the normal lake level elevation of 690 ft above msl. Trash collection baskets will be provided to collect trash from the screen wash water, for approved disposal, before the wash water is discharged to the lake. Strainers will be provided on the makeup pump discharges and the strainer backwash water is returned to the lake. The Ultimate Heat Sink intake is part of a common structure also providing the Normal Heat Sink intake function described in SSAR section 3.2.1.3

A backup supply of makeup water for the ESW cooling towers is supplied from the submerged pond located at the bottom of Clinton Lake that was constructed for the existing CPS Facility to provide the UHS function in the event of a failure of the dam on Clinton Lake.

3.2.2.4 Safety Functions

The ESW cooling water system provides cooling water to the closed cycle cooling system heat exchangers that serve the reactor plant systems requiring safety related cooling.

The existing CPS Facility submerged UHS pond contains sufficient water inventory to provide 30 days of shutdown cooling makeup water for the EGC ESP Facility and provide shutdown cooling for the existing CPS Facility under accident conditions. The additional water volume required to provide cooling tower make-up for cooldown of the EGC ESP Facility may reduce the allowable amount of accumulated sediment in the UHS pond.

The UHS pond is monitored for sediment accumulation periodically and after a major flood passes through the cooling lake (CPS, 2002). After the EGC ESP Facility is constructed, the allowable sedimentation accumulation in the UHS pond may be decreased. For example, an allowable sedimentation accumulation of approximately 118 acre-feet would support the largest anticipated additional capacity requirements.

3.2.2.5 Instrumentation

Temperature monitoring instrumentation is provided in the blowdown discharge pipe to monitor the discharge temperature.

3.3 Radiological Consequences of Accidents

The radiological consequences of potential design basis accidents (DBAs) are assessed to demonstrate that the alternative advanced reactors can be sited at the EGC ESP Site without undue risk to the health and safety of the public. The selection and evaluation of accidents is based upon USNRC regulatory guidance to the extent practical. Short -term (accident) site dispersion factors at the exclusion and low population zone boundaries that are based on measured site data are used to perform the assessments. The radioactivity released to the environs for DBAs is provided by the reactor supplier based upon their standard safety analysis reports or as specified in their PPE listing as being representative of the bounding DBA environmental release. The activities released to the environs are considered to be indicative of the performance of major structures, systems, and components intended to mitigate the consequences of accidents.

3.3.1 Selection of Postulated Accidents

Accidents have been selected to cover a spectrum of design basis events and reactor types. Consistent with regulatory objectives for determining site suitability, the selection includes low probability accidents postulated to result in significant releases of radioactivity to the environs. As such, the evaluations include light water reactor (LWR) Loss of Coolant Accidents (LOCAs) that presume substantial fuel damage in the core followed by the release of significant amounts of fission products into a containment building. In addition, accidents of higher frequency but with lower potential for significant releases are considered to permit quantitative assessment of the spectrum of potential risks at the EGC ESP Site.

It is not necessary nor practical to analyze all of the DBAs associated with the alternative reactor types that could be deployed at the EGC ESP Site, but rather to include a bounding and representative set (in terms of frequency and consequences) that can be used to demonstrate site suitability.

The spectrum of accidents considered focused on the LWR designs because of their recognized postulated accident bases and the availability of data. Accidents of lesser severity (and higher frequency) for some of the newer reactor types being considered are not as well defined and the application of accepted analytical conservatisms applied to LWRs through regulatory guides and standard review plans is not applicable based upon their unique design characteristics.

Selected accidents identified in Regulatory Guide 1.183, (USNRC, 2000) vendor design certification packages, vendor technical summary documents, and USNRC standard review plans for safety analyses were reviewed to establish the spectrum of accidents considered.

The following conditions and results were used in selecting DBAs for demonstrating site suitability:

Advanced Reactors for which Design Certification DBA data is available:

- AP1000: The AP1000 Design Control Document (Westinghouse, 2002) provides descriptions of the accidents and the technical data used to determine the radiological consequences for DBAs at a generic site. The AP1000 evaluations consider the major

DBAs identified in Regulatory Guide 1.183 and NUREG-1555 (USNRC, 2001). This information is part of the design certification licensing submittal for the AP1000, and is similar to the required analyses previously submitted for the certified AP600 reactor. The DBA assessments are evaluated to demonstrate EGC ESP Site suitability.

- ABWR: The ABWR Design Control Document (GE, 1997) provides descriptions of the accidents and the technical data used to determine the radiological consequences for DBAs at a generic site. This information was used by GE to obtain the design certification of the ABWR. The technical information and results are extended to the EGC ESP site assessment.

Non-Certified Advanced Reactor Designs:

Non-certified advanced reactor designs are screened and selected for assessment using the DBAs identified by the reactor vendors as having the potential to result in the limiting off-site radiological consequences.

- ESBWR: The DBAs postulated for the ABWR are expected to bound the ESBWR post accident design assessment. However under current regulations, the ESBWR limiting DBAs will be assessed using the alternate source term (AST) methods and guidance contained in Regulatory Guide 1.183 as opposed to the TID-14844 (USAEC, 1962) source term methods and NUREG-0800 (USNRC, 1987) guidance used for the ABWR certification. To demonstrate EGC ESP Site suitability under the current guidance a conservative ESBWR LOCA assessment is provided.

- IRIS: The low core power level and advanced design features (such as the elimination of all large loop piping) of the IRIS will limit the environmental releases of radioactivity after DBAs relative to other LWRs being considered. Although the DBAs are not finalized for this advanced concept, the vendor anticipates that post accident radiological consequences will be bounded by the AP600 and AP1000 evaluations. Therefore, no IRIS-specific dose assessments are performed.

- ACR-700: The LOCA with loss of emergency core cooling is considered the most limiting DBA for the ACR-700. The source term bases and approaches utilized to license this reactor type outside the U.S. have a number of similarities to USNRC regulatory guidance. There are, however, some differences in interpretation and implementation of this guidance. Therefore, the ACR-700 LOCA is analyzed to demonstrate that this reactor plant can be sited at the EGC ESP Site and also to provide a quantitative dose perspective for this design relative to the other alternatives.

Gas Cooled Advanced Reactor Designs:

The regulatory guidance and review standards described in current NRC publications are directed toward LWR technology and are not typically applicable to the assessment of the gas-cooled reactors.

Depressurization events are typically the critical considerations for gas-cooled reactors. The terms coolant, primary coolant, and pressure boundary when used with gas reactor technology differ significantly from the equivalent LWR usage. Coolant in the LWR context implies keeping the core cool in order to avoid fuel damage; maintaining the primary

coolant pressure boundary is a critical safety function. The pressure boundary function in the gas reactors is to contain the helium that removes heat from the core and transfers the energy to the power conversion unit. Core geometry, however, is physically maintained under normal and postulated accident conditions. Thus, loss of helium coolant does not result in significant fuel damage. This fact, and the much lower core power levels and associated fission product inventory for the gas reactors result in bounding post accident environmental releases that are substantially less than the LWRs.

- GT-MHR and PBMR: The gas-cooled reactors use mechanistic accident source terms and predict relatively small environmental releases compared with the water reactor technologies. The limiting DBA environmental releases specified by the gas reactor vendors are provided in Table 3.3-1.

  Based upon these projections of limiting environmental releases the post accident radiological dose consequences would result in less than 0.2 percent of the 10 CFR 50.34 acceptance criteria limits of 25 rem TEDE. Consequently, the DBAs that would be associated with the gas reactor technologies are not considered to be a major factor in assessing EGC ESP Site suitability relative to LWRs.

The above rationale provides the basis for the spectrum of limiting DBAs selected for evaluation in assessing the EGC ESP Site suitability. The selection predominately includes the LWR accidents identified in Regulatory Guide 1.183 and its appendices as important considerations for assessing the safety of nuclear plants at the EGC ESP Site.

- Main steam line breaks (AP1000 and ABWR)
- Reactor coolant pump locked rotor (AP1000)
- Control rod ejection (AP1000)
- Control rod drop (ABWR)
- Small line break outside containment (AP1000 and ABWR)
- Steam generator tube rupture (AP1000)
- LOCA (AP1000, ABWR, ESBWR, and ACR-700)
- Fuel handling accident (AP1000 and ABWR)

3.3.2 Evaluation of Radiological Consequences

Doses for the selected accidents are evaluated at the EAB and LPZ. These doses must meet the site acceptance criteria in 10 CFR 50.34 and 10 CFR 100. Although the emergency safety features are expected to prevent core damage and mitigate releases of radioactivity, the surrogate LOCAs analyzed presume substantial core damage with the release of significant amounts of fission products. The postulated LOCAs are expected to more closely approach 10 CFR 50.34 limits than the other postulated accidents of greater frequency but lesser magnitude. For these accidents the more restrictive dose limits in Regulatory Guide 1.183 and the NUREG-0800 Standard Review Plan are invoked to determine that the accidents are acceptable from an overall risk perspective.

The evaluations use short-term accident Chi/Qs. The Chi/Qs are determined using Regulatory Guide 1.145 (USNRC, 1983) methods with on-site meteorology data. The site 5th percentile worst sector Chi/Qs from Table 2.3-51 are used in these evaluations.

The accident dose evaluations are performed using Chi/Qs and activity releases for the following intervals:

EAB	LPZ
0 to 2 hr	0 to 8 hr
	8 to 24 hr
	1 to 4 days
	4 to 30 days

The accident doses are expressed as 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 (USEPA, 1993a). The DDE is taken as the same as the effective dose equivalent from external exposure and the dose conversions in Federal Guidance Report 12 (USEPA, 1993b) are applied.

3.3.3 Source Terms

Time-dependent activities released to the environs are used in the dose evaluations. These activities are based on the analyses used to support the reactor vendors' standard safety analysis reports. The different reactor technologies use different source terms and approaches in defining the activity releases.

The ABWR source term is based on TID-14844.

The ESBWR and the AP1000 source term and approach to assessing accidents are based on the AST methods as described in NUREG-1465 (USNRC, 1995) and guidance outlined in Regulatory Guide 1.183.

The ACR–700 source term definition is similar to the TID-14844 approach.

As noted, the GT-MHR and PBMR use a mechanistic approach to arrive at their accident source terms.

3.3.4 Postulated Accidents

This section identifies the postulated accidents, the resultant activity release paths, the important accident parameters and assumptions, and the credited mitigation features used in the EGC ESP Site dose consequence assessments. An overall summary of the results of the evaluated accident doses appears in Table 3.3-2. This table also compares the site safety analysis doses to the recommended limits based on Regulatory Guide 1.183 and NUREG-0800. Table 3.3-2 shows that the evaluated dose consequences meet the accident-specific acceptance criteria invoked in Section 3.3.2.

3.3.4.1 Main Steam Line Break Outside Containment (AP1000)

The bounding AP1000 steam line break for the radiological consequence evaluation occurs outside containment. The plant is designed so that only one steam generator experiences an uncontrolled blowdown even if one of the main steam isolation valves fails to close. Feedwater is isolated after the rupture and the faulted steam generator dries out. The secondary side inventory of the faulted steam generator is 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 contribute 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/hr) 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 are no significant accumulations of the nobles in the steam generators prior to the accident since they are rapidly released during normal service. Noble gases released during the accident are primarily 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 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 hr
- Steam generator initial mass – 3.03E+05 lbm
- Primary to secondary leak rate – 175 lb/hr in each steam 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 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 I-131
- Pre-existing iodine spike – reactor coolant at 60 µCi/g dose equivalent I-131
- Fuel damage - none

The activities released to the environment for the accident initiated and pre-existing iodine spike cases are shown in Tables 3.3-3 and 3.3-4, respectively.

The vendor calculated time-dependant off-site doses for a representative site. The doses were re-evaluated using the EGC ESP Site short-term accident dispersion characteristics in Table 2.3-51.

The total effective dose equivalent (TEDE) doses for the accident initiated iodine spike are shown in Table 3.3-5. The doses at the EAB and LPZ are a small fraction of the 25 rem TEDE limit identified in 10 CFR 50.34. A "small fraction" is defined as 10 percent or less in NUREG-0800 and Regulatory Guide 1.183. The doses for the pre-existing iodine spike are shown in Table 3.3-6. These doses also meet the TEDE dose guidelines of 10 CFR 50.34.

3.3.4.2 Main Steam Line Break Outside Containment (ABWR)

This ABWR event 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 line. Mass flow is initially 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 terminates the mass flows out of the break.

No fuel damage occurs 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. The iodine and noble gas activity in the water and steam masses discharged through the break is assumed released directly to the environs without hold-up or filtration. Salient features of the analyzed accident include:

- Duration of accident – 2 hr
- Main steam isolation valve closure – 5 sec
- Mass releases from break – steam 12,870 kg; water 21,950 kg
- 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/sec referenced to a 30 minute decay
- Fuel damage - none

The activities released to the environment for the maximum activity and pre-existing spike cases are shown in Table 3.3-7.

The calculated doses for the maximum allowed equilibrium activity at full power operation are shown in Table 3.3-8. The calculated doses for the pre-accident iodine spike are shown in Table 3.3-9. The EAB and LPZ doses are a small fraction of the 25 rem TEDE dose guidelines of 10 CFR 50.34.

3.3.4.3 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 coolant in the core combined with decreased heat transfer in the steam generator causes the reactor coolant system 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 the postulated failure of some of the fuel cladding with the consequential release of the 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. Analysis of the dose consequences presumes:

- Duration of accident – 1.5 hr
- Steam released - 6.48E+05 lbm
- Primary/secondary side coolant masses – 3.7E+05 lbm/6.06+05 lbm
- Primary to secondary leak rate – 350 lbm/hr
- 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 I-131
- Fission product gap activity fractions – Regulatory Guide 1.183, 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 off-site dose contributions. The activities released to the environment are shown in Table 3.3-10.

The vendor calculated the time-dependant off-site doses for a representative site. The doses were re-evaluated using the EGC ESP Site short-term accident dispersion characteristics in Table 2.3-51. The TEDE doses for the locked rotor accident are shown in Table 3.3-11. The doses at the EAB and LPZ are a small fraction of the 25 rem TEDE identified in 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's 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 a coincident loss of off-site 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 consequences analyses are performed using guidance in Regulatory Guides 1.77 (USAEC, 1974b) and 1.183. Salient features of the analysis of activity releases include:

- Duration of accident – 30 days
- Steam released - 1.08E+05 lbm
- Secondary side coolant mass – 6.06E+05 lbm
- Primary to secondary leak rate – 350 lbm/hr
- Containment leak rate – 0.1 percent per day
- Steam generator initial iodine and alkali metal activities – 10 percent of 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 I-131
- Fraction of rods with cladding failures – 0.10
- Fission product gap activity fractions
    - Iodines - 0.10
    - Noble gases - 0.10
    - Alkali metals - 0.12
- Fraction of fuel melting – 0.0025

- Fraction of activity released from melted fuel
  - Iodines - 0.5
  - Noble gases - 1.0
- Iodine chemical form – per Regulatory Guide 1.183 position C.3.5
- Containment atmosphere activity removal rates – 1.7/hr for elemental iodines, and 0.1/hr for particulate iodines and alkali metals.
- 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 activities released to the environment for the 30-day accident duration are shown in Table 3.3-12.

The vendor calculated the time-dependant off-site doses for a representative site. The doses were re-evaluated using the EGC ESP Site short-term accident dispersion characteristics in Table 2.3-51. The doses at the EAB and LPZ shown in Table 3.3-13 are well within the 25 rem TEDE identified in 10 CFR 50.34.

3.3.4.5 Rod Drop Accident (ABWR)

The design of the ABWR fine motion control rod drive system has 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 off-site power, contaminated steam is released from the secondary system due to the turbine trip and dumping of steam via the atmospheric relief valves. Steam dump (and retention of activity) to the condenser is precluded due to the assumption of loss of off-site 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 pathways 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 out 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 following assumptions have been used:

- Duration of accident – 24 hr
- Total flow through ruptured tube – 3.85E+05 lbm
- Steam release from faulted steam generator – 3.32E+05 lbm
- Steam released from intact steam generator – 1.42E+06 lbm
- Steam release duration – 13.2 hr
- Primary/secondary side initial coolant masses – 3.8E+05 lbm/3.7+05 lbm
- Primary to secondary leak rate – 175 lbm/hr 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 I-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 I-131
- Partition coefficients in steam generators - 0.01 for iodines and alkali metals
- Off-site power and condenser – lost on reactor trip
- Fuel damage - none

The activities released to the environment for the accident-initiated and pre-existing iodine spike cases are shown in Tables 3.3-14 and 3.3-15, respectively.

Based upon the vendor calculated the time-dependant off-site doses for a representative site, the doses were re-evaluated using the EGC ESP Site short-term accident dispersion characteristics in Table 2.3-51. The TEDE doses for the steam generator tube rupture accident with the accident-initiated iodine spike are shown in Table 3.3-16. The pre-existing iodine spike doses are shown in Table 3.3-17. The doses at the EAB and LPZ are a small fraction of the 25 rem TEDE identified in 10 CFR 50.34.

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

Small lines carrying reactor coolant outside 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 failure of the discharge line is neither significant nor analyzed. The flow (about 100 gpm) leaving containment is cooled below 140°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 off-site 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. Off-site 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. The activity is assumed released to the environment without decay or hold-up in the auxiliary building. Conditions used to determine activity releases include:

- Duration of accident – 0.5 hr
- Break flow rate – 130 gpm
- Reactor coolant noble gas activity – limit of 280 µCi/g dose equivalent Xe-133
- Reactor coolant equilibrium iodine activity - 1.0 µCi/g dose equivalent I-131
- 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 I-131
- Fuel damage - none

The activities released are shown in Table 3.3-18. Based upon the vendor calculated off-site doses for a representative site, the time-dependent doses were re-evaluated using the EGC ESP Site short-term accident meteorology in Table 2.3-51. The results are shown in Table 3.3-19. The resulting doses at the EAB and LPZ are a small fraction of the 25 rem TEDE in 10 CFR 50.34.

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, trip the reactor, and reduce reactor pressure.

The iodine in the flashed water is assumed to be transported to the environs by the HVAC system without credit for treatment by the standby gas treatment system. The other activities in the reactor water make only small contributions to the off-site dose and are neglected. The activity release assumes:

- Duration of the accident – 8 hr
- Standby gas treatment system – not credited
- Reactor building release rate – 200 percent/hr
- Mass of reactor coolant released – 13,610 kg
- Mass of fluid flashed to steam – 2,270 kg
- 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 min decay.

- Iodine spiking - accident initiated spike
- Fuel damage - none

The activity released to the environs is shown in Table 3.3-20. The calculated EAB and LPZ doses are shown in Table 3.3-21. The doses are a small fraction of the 25 rem TEDE limit in 10 CFR 50.34.

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 degradation and melting is assumed in this design basis accident. The assumption of major core damage is intended to challenge various accident mitigation features and provide a conservative basis for calculating site radiological consequences. The source term used in the analysis is adopted from NUREG-1465 and Regulatory Guide 1.183 with the nuclide inventory determined for a three-region equilibrium cycle core at 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 min. One half of the iodine and all of the noble gases in the blowdown stream are released to the containment atmosphere.

The core release starts after the 10-min 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 hr. 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 off-site 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. Important bases for determining activity releases and off-site doses include:

- Duration of accident – 30 days
- Reactor coolant noble gas activity – limit of 280 µCi/g dose equivalent Xe-133
- Reactor coolant equilibrium iodine activity - 1.0 µCi/g dose equivalent I-131
- Reactor coolant mass – 3.7E+05 lbm
- Containment purge flow rate – 8,800 cfm for 30 sec
- Containment leak rate – 0.1 percent per day
- Core activity group release fractions – Regulatory Guide 1.183, regulatory position C.3.2

- Iodine chemical form – Regulatory Guide 1.183, regulatory position C.3.5
- Containment airborne elemental iodine removal rate – 1.7/hr until DF of 200 is reached
- Containment atmosphere particulate removal rate – 0.43/hr to 0.72/hr during first 24 hr

Table 3.3-22 gives the activities released to the environment for the AP1000 large break LOCA. Based upon the vendor calculated off-site doses for a representative site, the time-dependent doses were re-evaluated using the EGC ESP Site short-term accident meteorology in Table 2.3-51. Table 3.3-23 gives the EAB and LPZ doses. Both 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 off-site doses. The EAB dose in Table 3.3-23 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 (USAEC, 1974a) are invoked in order to conservatively assess post-accident fission product mitigation systems and the resultant off-site doses.

100 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 immediately plateout leaving 25 percent of the inventory airborne and available for release. Following the break and depressurization of the reactor, some of the non-condensable fission 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 of contaminated steam past the main steam isolation valves. 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 gas treatment system is started and begins removing iodine from the reactor building atmosphere 20 min after start of the accident. The main steam line leakage is due to leaks past the main steam line isolation valves which 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/drain lines. Key features of the analysis of activity released include:

- Duration – 30 days

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

The activities released from the reactor and turbine buildings are given in Table 3.3-24. The doses at the EAB and LPZ are summarized in Table 3.3-25. The doses are within the 25 rem TEDE guidelines of 10 CFR 50.34.

3.3.4.11 Large Break Loss of Coolant Accident (ESBWR)

This ESBWR 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 1.183 are invoked in order to conservatively assess post-accident fission product mitigation systems and the resultant off-site doses.

100 percent of the core-inventory noble gases, 30 percent of the iodines, 25 percent of the core cesium, and minor fractions (< 1 percent) of the remaining core inventory are released from the reactor to the drywell over a 2-hour period at the beginning of the accident. The natural deposition of iodine within the drywell is credited in the analysis for the first day of the event. Following the break and depressurization of the reactor, some of the non-condensable fission products are removed by condensation within the Passive Containment Cooling System (PCCS). The PCCS 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: primary containment leakage and leakage of contaminated steam past the main steam isolation valves. The leakage to the reactor building is due to the containment penetrations. This leakage is distributed between the reactor building (50%), the external events shield building (45%), and a small fraction is released directly to the environment (5%). No credit is taken for any charcoal filtration systems for these paths. The main steam line leakage is due to leaks past the main steam line isolation valves, which 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. Key features of the analysis of activity released include:

- Duration – 30 days
- Core power level – 4000 MWt
- Fraction of iodine, noble gases, and other core isotopes released – Regulatory Guide 1.183, regulatory position 3.2.
- Iodine chemical form – Regulatory Guide 1.183, Appendix A, regulatory position 2.
- Passive Containment Cooling System Decontamination Factor – 1.5 for particulate and elemental iodine.
- Primary containment leakage – 0.5 percent/day.
- Main steam isolation valve total leakage – 150 cfh.
- Condenser leakage rate – 12.0 percent/day

The activities released to the environment are given in Table 3.3-26. The doses at the EAB and LPZ are summarized in Table 3.3-27. The doses are within the 25 rem TEDE guidelines of 10 CFR 50.34.

3.3.4.12 Large Break Loss of Coolant Accident (ACR-700)

The limiting design basis event for the ACR-700 is a large LOCA with coincident loss of emergency core 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 this 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 ACR-700 source term consists of 100 percent of the core-inventory noble gases and 50 percent of the iodines. These quantities are released from the fuel at the beginning of the accident. Ninety-five percent of the iodine enters containment as cesium iodide (CsI) and dissolves as non-volatile iodine in water. The remaining five percent of the iodine is released inside containment as volatile elemental and organic iodines. Under the oxidizing and high radiation environment following an accident, some non-volatile iodide in water would react and become volatile and partition into the gas phase. Elemental iodine,

however, is rapidly removed by adsorption on surfaces inside containment. A net reduction factor of 14 is applied to the elemental iodine based on analysis of the re-evolution and removal mechanisms during the accident.

The emergency core cooling (ECC) pumps and valves, which operate during the accident, are located in the long term cooling rooms outside the reactor containment building. The rooms have a sump to collect ECC leakage and a pump to return the radioactive fluids to the reactor building. Although the rooms' ventilation systems are isolated following a LOCA signal, it is possible that iodine flashed from the ECC leakage can leak past the ventilation dampers to the environment.

The contribution from ECC leakage outside the containment is analyzed assuming 50 percent of the core iodine inventory (as elemental iodine) is uniformly distributed in the containment sump water during recirculation. ECC leakage at greater than design conditions is assumed to occur for the duration of the post-accident period. In addition, a passive component failure (such as a ECC pump seal or valve packing) is assumed to occur 24 hours after start of the LOCA.

The dose contribution from containment bypass following a LOCA is small and may be neglected. Activity can be released from the steam generator main steam relief valves during a crash cool down of the plant during a LOCA. Even under conditions of chronic steam generator tube leakage during the LOCA, the contribution is several orders of magnitude less than the LOCA leakage contribution, and hence is neglected. Containment bypass due to operation of the containment ventilation system is not considered credible. Two independent means of rapidly isolating containment ventilation lines are provided for in the ACR generic design. This dual failure consideration provides a very high reliability of containment isolation and eliminates this potential impairment mechanism.

The containment isolation systems are credited with isolating the fluid systems that are not required to operate during the accident. The design basis includes a double barrier at the containment penetration with automatic closure of redundant valves. The normally sub-atmospheric containment isolates on a high-pressure signal (approximately ½ psig) during the accident thereby effectively assuring isolation prior to fission product release.

Features of the analysis of radioactivity released to the environment include:

- Duration – 30 days
- Core power level – 2059 MWt
- Core noble gas and iodine release fractions to containment – similar to TID-14844
- Iodine chemical form – similar to Regulatory Guide 1.183, regulatory position C.3.5
- Containment leak rate – 0.5 percent per day for 24 hours; 0.25 percent thereafter.
- Containment isolation –within 5 seconds after large LOCA
- Onset of fission product release from core – after containment isolation
- Iodine removal – factor of 14 removal for elemental iodines

- Containment dousing spray – not credited
- Containment ventilation filtration – not credited
- Sump water volume during recirculation – greater than 1000 m^3
- ECC leakage – 1 gal/hour based on Regulatory Guide 1.183, Appendix A, para. 5.2
- ECC passive failure – 50 gpm for 30 minutes at 24 hours
- Flashing fraction – 0.1 based on Regulatory Guide 1.183, Appendix A, para. 5.5
- ECC iodine chemical form – consistent with Regulatory Guide 1.183, Appendix A, paragraph 5.6
- ECC pump room isolation and hold-up – not credited.

The activity released to the environment during the large LOCA is shown in Table 3.3-28. The resulting doses at the EGC ESP Site EAB and LPZ are summarized in Table 3.3-29. The EAB and LPZ doses are within the 25 rem TEDE guidelines in 10 CFR 50.34

3.3.4.13 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 the dropping of 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 to the fuel, and heavy objects cannot be carried over the spent fuel.

Spent 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 the spent fuel pool surfaces. For fuel handling accidents 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 ft 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 off-site 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 dose activity released presumes:

- Core thermal power – 3468 MWt
- Decay time after shutdown – 100 hr
- Activity release period – 2 hr
- One of 157 fuel assemblies in the core is completely damaged
- Maximum rod radial peaking factor – 1.65
- Iodine and noble gas fission product gap fractions - Regulatory Guide 1.183, 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 given in Table 3.3-30.

The resulting doses at the EAB and LPZ are summarized in Table 3.3-31. 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 in 10 CFR 50.34. "Well within" is taken as being within 25 percent of the guideline limit consistent with the guidance in Regulatory Guide 1.183 and NUREG-0800.

3.3.4.14 Fuel Handling Accidents (ABWR)

The ABWR fuel handling accident is postulated as the 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 started, and effluents are released to the environment through this system. The gas treatment system is credited with maintaining the reactor building at a negative pressure after 20 min. Pool water is credited with removal of elemental iodine released from the failed rods. Guidance from Regulatory Guide 1.25 (USAEC, 1972) is used in performance of the analysis. Key aspects include:

- Core thermal power - 4005 MWt
- Decay time after shutdown - 24 hr
- Activity release period from pool - 2 hr
- Total number of fuel rods damaged - 115 in dropped and struck assemblies
- Radial peaking factor - 1.5
- Iodine and noble gas fission product gap fractions - Regulatory Guide 1.25, regulatory position C.1.d

- Iodine chemical form - Regulatory Guide 1.25, regulatory position C.1.f
- Pool decontamination for iodine - Regulatory Guide 1.25, regulatory position C.1.g
- Delay to achieve design negative pressure in reactor building - 20 min
- Reactor building leak rate during draw down - 150 percent/hr
- Standby gas system filtration - 99 percent efficiency
- Standby gas system exhaust rate - 50 percent/day

The radioactivity released to the environment is given in Table 3.3-32.

The doses at the site EAB and LPZ are summarized in Table 3.3-33. Activity remaining in the reactor building after two hours is assumed filtered and released without benefit of decay over the next six hr to determine the LPZ dose. Although assumptions in Regulatory Guide 1.25 are used, the off-site dose conversions are made using the guidance in Regulatory Guide 1.183. The EAB and LPZ doses are shown to be well within the 25 rem TEDE guidelines of 10 CFR 50.34.

3.4 Conformance With 10 CFR 100 – Reactor Site Criteria

3.4.1 10 CFR 100.21 – Non-Seismic Site Criteria

3.4.1.1 Exclusion Area and Low Population Zone

The EGC ESP Site EAB and control thereof is described in Section 2.1.2. The EGC ESP Site EAB includes an area encompassed by a circle of 1025 m radius. The boundary line for the EAB is shown in Figure 1.2-3. The EGC ESP Site exclusion area is in accordance with the definition for an exclusion area provided in 10 CFR 100.3.

The EGC ESP Site LPZ is described in Section 2.1.3.4. The EGC ESP Site LPZ includes an area encompassed by a circle of 2.5 mi radius (4018 m). The boundary line for the LPZ is shown in Figure 1.2-3. The EGC ESP Site LPZ is in accordance with the definition for a LPZ provided in 10 CFR 100.3.

3.4.1.2 Population Center Distance

The EGC ESP Site population center distance is described in Section 2.1.3.5. The closest population center for the Exelon ESP Site is Decatur, Illinois located approximately 22 mi SSW of the site. The EGC 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 reactor to the outer boundary of the low population zone or, in this case, approximately 3.3 mi.

3.4.1.3 Site Atmospheric Dispersion Characteristics and Dispersion Parameters

The site atmospheric dispersion characteristics and dispersion parameters for the EGC 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 EGC ESP Site are provided in Section 3.3. As demonstrated therein, the dose limits at the EAB and LPZ are in accordance with 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 radiological effluent release limits are provided in Section 3.1.1.1, where it is shown that the applicable regulatory limits provided in 10 CFR 20 and 10 CFR 50, Appendix I are satisfied for the EGC ESP Site.

3.4.1.4 Site Characteristics – Meteorology, Geology, Seismology, and Hydrology

3.4.1.4.1 Meteorology

The meteorological characteristics of the EGC ESP Site are described in detail in Sections 2.3.1 and 2.3.2. The regional and local data were used to establish average and extreme meteorological parameters that should be accounted for in the design of the EGC ESP

Facility.

Section 2.3.1 describes the regional meteorological characteristics of the general site based on long-term historical observations from two National Weather Service Observation Stations located in Peoria and Springfield, Illinois, both of which are within 55 mi of the EGC ESP Site. Regional historical information for the site area includes data for temperature, relative humidity, wind, precipitation, and snowfall. Severe weather information for the area is also summarized in this section for thunderstorms (expected frequency of occurrence), hail (expected frequency and size distribution), and lightning (predicted frequency of flashes), all of which have been characterized and bounded for inclusion in the design of site structures and equipment. Tornadoes (predicted frequency and intensity) and severe winds (maximum speed) were characterized to provide the site parameters associated with these events (including maximum linear and rotational wind speeds, pressure drop, and rate of pressure drop). Heavy snow (frequency and intensity), and severe icing (frequency and intensity) were characterized to provide worst-case accumulations of snow and ice to be accounted for in the design of buildings, towers, stacks and other site structures. The frequency of occurrence of fog was determined to facilitate a relative comparison with the occurrence of visible moisture plumes from the facility's cooling towers and the ultimate heat sink.

Section 2.3.2 describes the site-specific meteorological characteristics of the EGC ESP Site as obtained from an on-site meteorological monitoring system operated continuously by CPS 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 were 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 data described above as well as to facilitate the development of site atmospheric dispersion characteristics and dispersion parameters for routine and accidental releases from the EGC 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 and site-specific parameters established to provide representative average and extreme meteorological information characteristic of the site. These data were summarized for use in the design of the EGC ESP Facility to determine that no site parameters would pose an undue risk to the operation of the facility.

3.4.1.4.2 Geology

The regional and site geology for the EGC ESP Site is summarized in Sections 2.5.1 and 2.5.2 of this SSAR and described in more detail in Appendices A and B. These discussions review past and recent published information about the regional and site geology for the EGC ESP and CPS Sites. The literature review included a search of relevant geological information at the Illinois State Geological Survey (ISGS).

The evaluation of geology included a review of results of geotechnical explorations and laboratory testing programs carried out at the CPS and EGC ESP Sites. Geotechnical explorations and laboratory testing programs were performed during the mid-to-late 1970s at the CPS Site to define the soil stratigraphy and to quantify engineering properties of the

soils at the site. Additional explorations and laboratory testing programs were completed at the EGC ESP Site to evaluate the consistency of soil and rock conditions at the EGC ESP Site relative to the CPS Site, and to update dynamic soil property information within the footprint of the EGC ESP Site. The updated dynamic soil property information at the EGC ESP Site included a borehole P-S Suspension Logging test to define the shear and compression wave velocities of soils in situ, as well as laboratory resonant column/cyclic torsional shear tests on soil samples obtained from the EGC ESP Site to quantify the change in shear modulus and material damping with shearing strain amplitude.

It was concluded from the explorations, laboratory testing programs, and literature reviews conducted for the EGC ESP Site that the geology and geotechnical conditions, and particularly site soil profile for the EGC ESP Site are consistent with information presented in the CPS USAR for the CPS Site. The available information indicates that the geologic and geotechnical conditions are relatively uniform within the footprint of the existing CPS and EGC ESP Sites. These conditions consist of 250 to 280 ft of alluvium made up primarily of hard silts and clays and occasional layers of sand above bedrock. The bedrock consists of shale and limestone. No evidence of solution cavities or similar features was detected in the limestone. Groundwater is located within 30 ft of the ground surface.

Results of updated geology and geotechnical evaluations determined that the geotechnical and geologic characteristics pose no undue risk to the EGC ESP Facilities. No geologic hazards from non-seismic faults, slope instability and landslides, or ground subsidence from sinkhole or mine collapse were identified either during the original CPS Site evaluation or during this EGC ESP Site evaluation. Soils were found to have high bearing characteristics and relatively small settlement potential under loads that would normally be associated with facility design. These favorable soil conditions result from the significant thickness of ice that overrode the site during past glaciations.

There have also been no reports of unusual or unacceptable behavior of the existing CPS Facility, relative to geology or geotechnical conditions, during its nearly 20 years of operation.

3.4.1.4.3 Seismology

The seismotectonic environment for the EGC ESP Site has been reviewed in detail, and an updated probabilistic seismic hazards analysis (PSHA) has been performed for the EGC ESP Site. The results of the review and update are summarized in Sections 2.5.2 of this SSAR and in Appendix B. The seismic hazard analysis for the EGC ESP Site included updating components of the existing PSHA conducted by the Electric Power Research Institute (EPRI) in the late 1980s (EPRI, 1989-1991). Results of updated seismic hazard analyses were used to establish risk-consistent design response spectra (DSR) for the EGC ESP Site.

The geotechnical and seismology work for the EGC ESP Site included:

- shear wave velocity measurements in the alluvium and top of rock at the site using in situ velocity measurement methods,
- cyclic laboratory tests to estimate shear modulus and material damping of representative, undisturbed soil samples collected from the site,

- reviews and updates of seismic source zones, maximum magnitude of the source zones, earthquake recurrence rates, and ground motion attenuation models for the region,
- determination of uniform hazard response spectra on rock using the updated EPRI PSHA seismotectonic inputs and ground motion attenuation model,
- evaluation of site response using a one-dimensional wave propagation model to determine hazard consistent response spectra at the ground surface, and
- use of the hazard-consistent response spectra at the ground surface for establishing the risk-consistent DRS.

Results of the field program found that the shear wave velocities in soil and rock at the EGC ESP Site are consistent with information obtained previously for the CPS Site. Shear modulus and material damping results from laboratory resonant column/cyclic torsional shear tests on samples from the EGC ESP Site are consistent with currently accepted modulus and damping properties for soils, including the EPRI soil model (EPRI, 1993a).

Results of the reviews of seismic source zones, source earthquake potential, and ground motions models determined that an update to the EPRI seismic hazard model was required for the EGC ESP Site. The update to the EPRI seismic source and ground motion model was made, and the hazard recalculated at the top of hard rock for a mean 10^{-4} and mean 10^{-5} annual occurrence. A smoothed response spectrum for hard rock was then obtained. Earthquake records consistent with the magnitude and distance combinations from deaggregation of the seismic hazard at 1 to 2 Hz and at 5 to 10 Hz were used for site response evaluations. These earthquake records (that is, time histories) were used in a one-dimensional, equivalent linear computer code, called SHAKE, to obtain hazard consistent response spectra in the free-field at the ground surface. Uncertainties in the soil properties, layering, and appropriate earthquake records were accounted for by conducting multiple realizations of the soil and rock model.

The resulting response spectra from the realizations were enveloped to define the SSE for the mean 10^{-4} and mean 10^{-5} annual occurrence at the EGC ESP Site. The risk-consistent DRS for horizontal and vertical motions were computed following guidance given in the ASCE nuclear facility design guide titled *Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities and Commentary* (ASCE, 2003). This method for obtaining DRS is consistent with the approach described in the U.S. Department of Energy Standard titled *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities* (USDOE, 1996) and in NUREG/CR-6728 titled *Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines* (McGuire et al., 2001) and its companion document NUREG/CR-6769 titled *Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Development of Hazard- and Risk-Consistent Seismic Spectra for Two Sites* (McGuire et al., 2002). A summary of the approach used to determine the risk-consistent DRS is given in Sections 2.5.2.6 and 2.5.4.9 of the SSAR, and details are provided in Appendix B.

The resulting DRS for horizontal and vertical response are lower than the Regulatory Guide 1.60 (USAEC, 1973) spectrum anchored to a peak free-field ground motion of 0.3g except at frequencies between 16 Hz and 50 Hz. These exceedances are considered acceptable based on high-frequency evaluations discussed in EPRI (1993b). EPRI (1993b) presents an

assessment of the significance of high frequency ground motions to the seismic safety performance of nuclear power plants. That study indicates that there are two factors that lead to reduced effectiveness of high frequency motions to adversely affect performance: (1) the increased incoherence of ground motions at frequencies greater than 10 Hz compared to those at lower frequencies and (2) the capacity of structures and equipment in nuclear power plants to in-elastically absorb the small displacements associated with high frequency ground motions without significant effect. The incoherence reductions are consistent with those recommended in ASCE 4 titled *Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures* (ASCE , 1998).

EPRI (1993b) recommends procedures for reducing the high-frequency portion of DRS to account for these effects. The recommended reduction factors for ground motion incoherencey are 10 percent at a frequency of 10 Hz increasing to 20 percent for frequencies of 25 Hz and larger. These factors are appropriate for a building width of approximately 150 ft. For a 75 ft dimension, such as might be associated with a diesel generator building or pump house, these reductions are approximately 50 percent of those for a 150-ft dimension. The reduction factors due to in-elastic absorption of small displacements are of comparable magnitude.

When modified by the high frequency adjustment factors recommended by EPRI (1993b), the vertical and horizontal DRS in Figure 2.5-12 either will be enveloped by the Regulatory Guide 1.60 response spectrum, in the case of a large structure (that is, dimensions equal to approximately 150 ft), or result in only minor exceedances at frequencies in excess of 25 Hz for structures where dimensions are on the order of 75 ft. Based on these results, it is concluded that the high-frequency exceedances of the Regulatory Guide 1.60 response spectrum anchored to 0.3g peak acceleration by the EGC ESP DRS are not significant – indicating that the EGC ESP Site is suitable for any design based on a Regulatory Guide 1.60 response spectrum.

Other risks associated with the ground motions include the potential for ground motion-induced liquefaction, ground motion-induced settlement, and ground motion-induced slope instability. The characteristics of the soil at the EGC ESP Site are such that the potential for each of these occurrences is low, resulting in no undo risk to development at the site.

3.4.1.4.4 Hydrology

The hydrologic conditions of the EGC 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 EGC ESP Facility. These hydrologic engineering characteristics include floods, ice effects, cooling water, low water considerations, accidental releases in surface water, and groundwater.

Section 2.4.2 presents information on the flooding history, flood design considerations, and the effects of local intense precipitation. Section 2.4.2.2 describes the hydrologic analyses and hydraulic design for the dam and Clinton Lake. Safety-related structures at the EGC ESP Facility will need to be outside of the flood elevation or designed to withstand the effect 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 Clinton Lake and

Section 2.4.10 discusses the flooding protection requirements. As described in Section 2.4.3, the attenuation effect of the lake will reduce the expected magnitude of the flood flows downstream. Floods in the lake will not affect the EGC ESP Site at grade elevation of 735 ft above msl.

Section 2.4.7 describes the effects of ice formation and the probable maximum winter flood on the lake water levels. The availability of cooling water and the performance of the ultimate heat sink will not be affected by ice formation.

Section 2.4.8 describes Clinton Lake, the dam, the ultimate heat sink and the station discharge flume. The description of the ultimate heat sink that will provide shutdown-cooling water for the existing CPS and makeup water to the EGC ESP Facility safety-related cooling towers is discussed in Section 2.4.8.1.5.

Section 2.4.11 describes low water considerations including the evaluation of drought effects on the cooling lake, plant requirements and heat sink dependability requirements. A lake drawdown analysis performed at the COL stage will determine the type of cooling tower design and/or load reduction program required to maintain the minimum lake elevation.

Section 2.4.12 describes the possibility of effluents to reach a surface water body. As discussed in Section 2.4.13.3, there would be no hydraulic gradient for effluents accidentally released within the buildings to leak to the outside. Instead, groundwater would be forced into the building to relieve hydrostatic pressure. For tanks located in structures above grade, any released fluid would ultimately reach the lower levels of the building and would also be contained therein. For tanks located outside structures, positive means to collect and prevent releases such as dikes and collection basins will be provided eliminating them as a source of groundwater contamination. Therefore the potential for effluents to reach a surface water body and surface water users is minimal.

Section 2.4.13 provides the regional and site-specific descriptions of groundwater conditions. Section 2.4.13.2.3 describes the site hydrogeologic systems including the aquifers present and their characteristics (depth, permeability, potentiometric levels and velocity). As indicated above, Section 2.4.13.3 describes the potential effects on groundwater from accidental releases. The design basis for subsurface hydrostatic loading is presented in Subsection 2.4.13.5.

The information contained in Section 2.4 on surface water and groundwater conditions was evaluated and site-specific parameters were established to represent the site. These data were summarized for use in the design of the EGC ESP Facility to verify that no site parameters would pose an undue risk to the operation of the facility.

3.4.1.5 Potential Off-Site Hazards

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

Section 2.2.2.5.3 addresses aircraft hazards as they may affect the EGC ESP Facility wherein the area in which the ESP safety related structures would need to be located to meet the 1.0E-07 impact probability criterion is established.

As noted in Section 2.2.3, highway accidents are not a concern for the EGC ESP Site.

Accidents involving railway traffic, while not expected to be a design issue, will require analysis at the COL stage to verify control room habitability is acceptable for the EGC ESP Facility.

Explosions are addressed in Section 2.2.3.1.1. This section discusses pipelines and nearby industrial facilities. Evaluation of the 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 EGC ESP Site. The only industrial facility potentially representing an explosive source is propane storage by Cornbelt FS in DeWitt. Propane storage at this location was also determined to not constitute a design consideration for the EGC ESP Site.

Toxic chemicals are discussed in Section 2.2.3.1.3. Based on the information in this section it is concluded that an analysis specific to the location of the EGC ESP Facility control room evaluating the impact of an ammonia release from the Van Horn–DeWitt facility will be required at the COL stage to determine if ammonia detection and isolation capability will be required in the EGC ESP Facility design. It was also determined that the COL phase for the EGC ESP Site will require a new analysis of the hazards associated with the Gilman Line that considers the control room ventilation design and the specific location of the EGC ESP Facility. Periodic review of hazardous material types and quantities being transported will also be required, at the same or similar frequency to those currently being performed in support of the CPS operation.

No other off-site hazards associated with fires, collisions with the intake structure or liquid spills are an issue for the EGC ESP Site.

3.4.1.6 Site Characteristics - Security Plans

The EGC ESP Facility is located approximately 700 ft south of the existing CPS Facility. The site plot plan is provided in Figure 1.2-3. The detailed security plans developed for the EGC ESP Facility will be established at the COL stage once a reactor design is selected and detailed plant layout information is available.

The current footprint for the EGC ESP Facility is established to fit the various equipment and structures associated with the alternative vendor designs and does not include any specific distance for fencing, etc. Depending on the final selection of the reactor vendor, a distance of 110 meters as recommended in Regulatory Guide 4.7 (USNRC, 1998) from some vital equipment and structures may extend outside of the current EGC ESP Site footprint. The distance of 110 meters is recommended in Regulatory Guide 4.7 as the distance around vital structures and equipment that provides sufficient space to apply satisfactory security measures such as protected area barriers, detection equipment, isolation zones and vehicle barriers. However, sufficient distance is available to implement the criteria of 10 CFR 73.55, including the revised design basis threat. Implementation is anticipated to be equivalent to the implementation for the existing CPS facility.

Since CPS is an existing plant with an existing security plan located on the site property, there are no identified impediments to the eventual development of an adequate security plan for the EGC ESP Facility. For example, an approach from Clinton Lake will be controlled in the same manner as it is controlled for the existing CPS facility.

Potential hazards from the nearby CPS facility have been evaluated as discussed in Section 2.2 of this SSAR.

The location of relevant law enforcement agencies, their geographical jurisdictions, and their ability to respond in force to a security event are equivalent to the descriptions provided in the security response information for the CPS facility.

3.4.1.7 Site Characteristics - Emergency Plans

Information is provided in the EGC ESP Application, Emergency Plan. The CPS evacuation time estimate (ETE) performed in 1993 is valid for current conditions. The estimates of evacuation times for the most limiting conditions of summer weekdays are acceptable. The estimate of evacuation time for one special case of the Apple and Pork Festival is also acceptable. Therefore, there are no geographic or political impediments to the development of an Emergency Plan.

3.4.1.8 Population Density

As described in Section 2.1.3.6, the EGC ESP Site is located in a mostly rural area. The population density in this area is well below the Regulatory Guide 1.70 criteria of 500 people per mi^2. The area between 25 and 37 mi from the site is the most densely populated with an average population density of 110 people per square mi. Based on population projections for this region, this density is not projected to significantly change through 2060.

3.4.2 10 CFR 100.23 - Geologic and Seismic Siting Criteria

3.4.2.1 Geological, Seismological, and Engineering Characteristics

Geological, seismological, and engineering characteristics of the EGC ESP Site and its surroundings have been investigated to allow evaluation of the site, to provide information to support evaluations performed to derive the appropriate risk-consistent DRS, and to permit adequate engineering solutions to actual or potential geologic and seismic effects. Results of these investigations have been discussed previously in more detail in Sections 2.5, 3.4.1.4.2 and 3.4.1.4.3, and in Appendices A and B.

The scope of the geological, seismological, and engineering studies performed for the EGC ESP Site relied heavily on the existing database of information available for the operating CPS facilities. These facilities are located within 700 ft of the EGC ESP Site. Regional geologic conditions affecting the EGC ESP Site are consistent with those previously investigated for the CPS Site; therefore, the scope of the geologic investigations was focused on determining what new information has become available since studies were carried out for the CPS Site. Much of this updated information addressed the tectonic provinces and structures relevant to the EGC ESP Site, the potential magnitudes of seismic events associated with these provinces and structures, and the attenuation of ground motions from these sources. A geotechnical exploration program consisting of drilling and sampling, cone penetrometer testing, and geophysical surveys was also completed at the EGC ESP Site. Results of these efforts confirm that geologic and geotechnical conditions at the EGC ESP Site are consistent with those previously determined for the CPS Site. This similarity in geologic and geotechnical conditions allowed the evaluations of geologic, seismologic, and engineering characteristics to consist of updates to the existing information rather than new

characterizations.

As part of the assessment of vibratory ground motion, the stratigraphy and structural geology for the CPS Site were reviewed for consistency with more recently published information. Tectonic structures and provinces within 200 mi of the site were considered during this review. This review began with information in the CPS USAR and then was supplemented by more recently published information. The listing of historically reported earthquakes used in the EPRI seismic hazard model was also updated to include earthquakes occurring since the mid-1980s. Current evaluations of the recurrence rates and maximum magnitudes of earthquakes in these sources were also made by reviewing literature and through discussions with scientists specializing in the seismic characterization within the site area. A paleoliquefaction reconnaissance was conducted in proximity to the site by inspecting river and stream banks for potential features from pre-historic liquefaction. Results of this work confirmed recently published information about potential magnitudes of earthquakes near the site. Additional discussions of the paleoliquefaction work are summarized in Section 2.5.2 and discussed in detail in Appendix B.

The seismological review concluded that an update of the EPRI seismic hazard analysis was needed. This update was implemented by an independent Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 evaluation of ground motion models for CEUS (EPRI, 2003). The SSHAC evaluation included participation of a team of Technical Integrators and a panel of nationally recognized experts (Panel Experts) in the area of ground motion modeling. This group identified a set of models and weighting factors and developed an updated ground motion model for ground motion hazard studies in CEUS. Seismic sources were revised following the guidance in Regulatory Guide 1.165 (USNRC, 1997), Appendix E.3. The EPRI (1989-1991) hazard results were then updated using the revised seismic source interpretations and new ground motion model.

Results of the new geological and seismological data and revised input models were used in a PSHA calculation to determine the rock level uniform hazard response spectra at the EGC ESP Site. Site-specific determination of hazard-consistent ground motion at the ground surface at the site considered the difference in shear wave velocity at hard rock sites (for example, velocities in excess of 9,000 fps) and shallow rock and soil conditions existing at the EGC ESP Site. Published compression wave velocity information obtained during deep well logging throughout the area was used to estimate the velocity of rock at the EGC ESP Site below the depth of measurement. The effects of the local site soil conditions, which consisted of 280 ft of alluvium over rock, were then accounted for in the determination of the hazard-consistent ground motion spectra in the free-field by modeling the soil response.

Time histories consistent with controlling earthquake magnitudes and distances determined by deaggregating the rock level uniform hazard spectrum at low frequencies (1 to 2.5 Hz) and high frequencies (5 to 10 Hz) were used in these site-response analyses. Results of in situ shear wave velocity measurements and laboratory resonant column/cyclic torsional testing on soil samples obtained from the EGC ESP Site were used to model the dynamic properties of the site materials for the purpose of site response analyses. The site response evaluation considered the uncertainty in soil and rock property characterization when performing these analyses.

Smoothed DRS for horizontal and vertical motions were determined using a risk-consistent

approach described in various documents, including ASCE Standard XXX involving the design criteria for structures, systems, and components of nuclear facilities (ASCE, 2003), the U.S. DOE 1020 Design Standard (USDOE, 1996), and NUREG/CR-6728 and NUREG/CR-6769 (McGuire et al., 2001 and 2002). These spectra are shown in Figure 2.5-12 of Section 2.5 and discussed in more detail in Appendix B.

As discussed in Section 3.4.1.4.3, the resulting risk-consistent DRS exceed the spectra given in Regulatory Guide 1.60 anchored to 0.3g at frequencies from 16 Hz to 50 Hz. High frequency adjustments using procedures described in EPRI (1993b) and in ASCE 4 (1998) result in spectral accelerations at high frequencies which are approximately equal to or lower than those in Regulatory Guide 1.60 for facilities with plan dimensions of 75 ft to 150 ft, respectively. This leads to the conclusion that a nuclear power plant can be constructed and operated at the EGC ESP Site without undue risk to the health and safety of the public and therefore, the EGC ESP Site would be suitable for any design based on a Regulatory Guide 1.60 spectrum.

An operating system has not been selected at this time, and therefore, system-specific soil-structure interaction studies have not been performed. Once an operating system is selected, appropriate design analyses will be conducted, as might be required by the vendor or in light of the site-specific DRS relative to the Regulatory Guide 1.60 anchored to 0.3g.

3.4.2.2 Geologic and Seismic Siting Factors

Geologic and seismic siting factors for other design conditions at the EGC ESP Site were also addressed. These siting factors included soil and rock stability, liquefaction potential, natural and artificial slope stability, cooling water supply, and remote safety-related structure siting. Evaluations were also conducted to address site foundation material and seismically induced floods and water waves.

Results of the field exploration and laboratory testing programs indicate that site foundation materials are essentially the same in terms of consistency and layering as those existing at the CPS Site. These soils are hard silts and clays with very high allowable bearing pressures and very low settlement potential to depths of 280 ft where rock occurs. Soils in the upper 55 ft of soil profile at the CPS Site had higher settlement potential and therefore were removed and replaced. Similar soil conditions occur at the EGC ESP Site, and therefore, it is likely that the upper 55 ft would be removed and replaced with compacted soil that is not settlement prone. By removing soil in the upper 60 ft, the potential for liquefaction of soil during a design earthquake is minimized.

The EGC ESP Site is essentially flat with the closest slope located nearly 800 ft to the north. This location poses little risk from slope instability of any type – even associated with lateral spreading during a design earthquake. Slopes required for construction can be as steep as 1H:1V (horizontal to vertical) and therefore should not result in any undue restrictions. Cooling water supply is currently provided to the CPS Facility from Clinton Lake. The EGC ESP Facility will use cooling towers for cooling with Clinton Lake being used to provide make-up water to the cooling towers. The description of the ultimate heat sink (UHS) that will provide shutdown-cooling water for the existing CPS Facility and makeup water to the EGC ESP Facility safety-related cooling towers is provided in Section 2.4.8.1.5. Other remote safety-related structures are such that they should not require or result in foundation

requirements that are unique or result in unique risk. In light of these conditions there is a reasonable assurance that a nuclear power plant can be constructed and operated at the EGC ESP Site without undue risk to the health and safety of the public.

Other design conditions such as areas of potential collapse, flooding, water waves, and tectonic deformation were also evaluated to determine the potential risk that these phenomena pose to construction and operation. No evidence of soluble deposits, karst terrain, or mining activities was found in proximity to the site. There are no upstream dams or large water reservoirs, and therefore the potential for flooding from failure of a dam or reservoir located upstream of the site is nonexistent. The location of the EGC ESP Site is nearly 800 ft from Clinton Lake, and therefore the potential for water waves associated with seiches is minimal. The shape of Clinton Lake is also such that the likelihood of developing anything more than a small seiche is minimal. No surface tectonic, nontectonic features, or potentially active faults were identified as being located within a 25 mi radius of the EGC ESP Site, resulting in minimal risk from these occurrences.

CHAPTER 3

References

Chapter Introduction

10 CFR 100. Code of Federal Regulations. "Reactor Site Criteria."

Section 3.1

10 CFR 20. Code of Federal Regulations. "Standards for Protection Against Radiation."

10 CFR 50. Appendix I. Code of Federal Regulations. "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As is Reasonably Achievable' for Radioactive Materials in Light Water Cooled Nuclear Power Reactor Effluents."

10 CFR 71. Code of Federal Regulations. "Packaging and Transportation of Radioactive Material."

49 CFR 173. Code of Federal Regulations. "Shippers – General Requirements for Shipments and Packagings."

Bland, J. S. "NRCDose for Windows. User's Guide." November 2000.

Strange, D. L., R. A. Peloquin, and G. Wheelan, "LADTAP II – Technical Reference and User Guide." NUREG/CR-4013 PNL-5270. April 1986.

Strange, D. L., T. J. Bander, and J. K. Soldat, "GASPAR II, - Technical Reference and User Guide." NUREG/CR-4653 PNL-5907. March 1987.

U.S. Nuclear Regulatory Commission (USNRC). *Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purposed of Evaluating Compliance with 10 CFR 50, Appendix I*. Regulatory Guide 1.109, Revision 1. Office of Standards Development. October 1977a.

U.S. Nuclear Regulatory Commission (USNRC). *Estimating Aquatic Dispersion of Effluent from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I*. Regulatory Guide 1.113, Revision 1. Office of Standards Development. July 1978.

U.S. Nuclear Regulatory Commission (USNRC). M*ethods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light Water-Cooled Reactors*. Regulatory Guide 1.111, Revision 1. Office of Standards Development. July 1977b.

Section 3.2

Clinton Power Station (CPS). *Clinton Power Station Updated Safety Analysis Report*. Revision 10. 2002.

U.S. Nuclear Regulatory Commission (USNRC). *Ultimate Heat Sink for Nuclear Power Plants.* Regulatory Guide 1.27, Revision 2. Office of Standards Development. January 1976.

Section 3.3

10 CFR 50. Code of Federal Regulations. "Domestic Licensing of Production and Utilization Facilities."

10 CFR100. Code of Federal Regulations. "Reactor Siting Criteria."

General Electric (GE), ABWR Standard Safety Analysis Report, through Amend. 35. May 1997.

U.S. Atomic Energy Commission (USAEC). *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. Directorate of Regulatory Standards. March 1972.

U.S. Atomic Energy Commission (USAEC). *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors.* Regulatory Guide 1.3. Rev. 2. Directorate of Regulatory Standards. June 1974a.

U.S. Atomic Energy Commission (USAEC). *Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors.* Regulatory Guide 1.77, Revision 2. Directorate of Regulatory Standards. June 1974b.

U.S. Atomic Energy Commission (USAEC). *Calculation of Distance Factors for Power and Test Reactor Sites.* TID-14844. Division of Licensing and Regulation. March 1962.

U.S. Environmental Protection Agency (USEPA). Federal Guidance Report 11. *Limiting Values of Radionuclide Intake and Air concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion.* EPA-520/1-88-020. 1993a.

U.S. Environmental Protection Agency (USEPA). Federal Guidance Report 12. *External Exposure to Radionuclides in Air, Water, and Soil.* EPA-402-R-93-081. 1993b.

U.S. Nuclear Regulatory Commission (USNRC). *Accident Source Terms for Light –Water Nuclear Power Plants.* NUREG-1465. Office of Nuclear Regulatory Research. February 1995.

U.S. Nuclear Regulatory Commission (USNRC). *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*. Regulatory Guide 1.183. Office of Nuclear Regulatory Research. July 2000.

U.S. Nuclear Regulatory Commission (USNRC). *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.* Regulatory Guide 1.145, Revision 1. Office of Nuclear Regulatory Research. February 1983.

U.S. Nuclear Regulatory Commission (USNRC). *Standard Review Plan for the Review of Safety Analysis Report for Nuclear Power Plants.* NUREG-0800. Office of Nuclear Regulatory Research. 1987.

Section 3.4

10 CFR 20. Code of Federal Regulations. "Standards for Protection Against Radiation."

10 CFR 50. Code of Federal Regulations. "Domestic Licensing of Production and Utilization Facilities."

10 CFR 100. Code of Federal Regulations. "Reactor Site Criteria."

American Society of Civil Engineers (ASCE). "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities and Commentary." ASCE Standard XXX, Approval Draft. ASCE, Reston, VA. July 25, 2003.

American Society of Civil Engineers (ASCE). "Seismic Analysis of Safety Related Nuclear Structures and Commentary." ASCE 4. ASCE, Reston, VA. 1998.

Electric Power Research Institute (EPRI). "Probabilistic Seismic Hazard Evaluations at Nuclear Power Plant sites in the Central and Eastern United States," EPRI Report NP-4726, All Volumes, 1989-1991.

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.

Electric Power Research Institute (EPRI). "Analysis of High-Frequency Seismic Effects." EPRI Report TR-102470, 1993b.

Electric Power Research Institute (EPRI). CEUS Ground Motion Project – Model Development and Results, TR-1008910, 2003

McGuire, R. K., W. J. Silva, and C. J. Costantino. "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines." NUREG/CR-6728. U. S. Nuclear Regulatory Commission, Washington, D.C., 2001.

McGuire, R. K., W. J. Silva, and C. J. Costantino. "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Development of Hazard-and Risk-Consistent Seismic Spectra for Two Sites." NUREG/CR-6769. U. S. Nuclear Regulatory Commission, Washington, D.C., 2002.

U.S. Atomic Energy Commission (USAEC). *Design Response Spectra for Seismic Design of Nuclear Power Plants.* Regulatory Guide 1.60, Revision 1. Directorate of Regulatory Standards. December, 1973.

U.S. Department of Energy (USDOE). "DOE Standard: Natural Phenomena Hazards, Design and Evaluation Criteria for Department of Energy Facilities." DOE-STD-1020-94 including Change Notice #1. January 1996.

U.S. Nuclear Regulatory Commission (USNRC). *Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquakes Motion.* Regulatory Guide 1.165. Office of Nuclear Research. March 1997.

U.S. Nuclear Regulatory Commission (USNRC). *General Site Suitability Criteria for Nuclear Power Stations*. Regulatory Guide 4.7. Office of Nuclear Regulatory Research. April 1998.

CHAPTER 3

Tables

TABLE 3.1-1
Comparison of Average Annual Gaseous Releases to 10 CFR 20 Effluent Concentration Limits (ECL)

Isotope	Release[a] Ci/yr	Boundary Conc.[b] μCi/cc	10 CFR 20 ECL μCi/cc	Fraction of ECL
Kr-83m	8.38E-04	6.8E-17	5.0E-05	1.4E-12
Kr-85m	7.20E+01	5.8E-12	1.0E-07	5.8E-05
Kr-85	8.20E+03	6.6E-10	7.0E-07	9.5E-04
Kr-87	3.00E+01	2.4E-12	2.0E-08	1.2E-04
Kr-88	9.20E+01	7.4E-12	9.0E-09	8.3E-04
Kr-89	2.41E+02	1.9E-11	1.0E-09	1.9E-02
Kr-90	3.24E-04	2.6E-17	1.0E-09	2.6E-08
Xe-131m	3.60E+03	2.9E-10	2.0E-06	1.5E-04
Xe-133m	1.74E+02	1.4E-11	6.0E-07	2.3E-05
Xe-133	9.20E+03	7.4E-10	5.0E-07	1.5E-03
Xe-135m	4.05E+02	3.3E-11	4.0E-08	8.2E-04
Xe-135	6.60E+02	5.3E-11	7.0E-08	7.6E-04
Xe-137	5.14E+02	4.2E-11	1.0E-09	4.2E-02
Xe-138	4.32E+02	3.5E-11	2.0E-08	1.7E-03
Xe-139	4.05E-04	3.3E-17	1.0E-09	3.3E-08
I-131	2.59E-01	2.1E-14	2.0E-10	1.0E-04
I-132	2.19E+00	1.8E-13	2.0E-08	8.9E-06
I-133	1.70E+00	1.4E-13	1.0E-09	1.4E-04
I-134	3.78E+00	3.1E-13	6.0E-08	5.1E-06
I-135	2.41E+00	1.9E-13	6.0E-09	3.2E-05
C-14	1.46E+01	1.2E-12	3.0E-09	3.9E-04
Na-24	4.05E-03	3.3E-16	7.0E-09	4.7E-08
P-32	9.19E-04	7.4E-17	1.0E-09	7.4E-08
Ar-41	4.00E+02	3.2E-11	1.0E-08	3.2E-03
Cr-51	3.51E-02	2.8E-15	3.0E-08	9.5E-08
Mn-54	5.41E-03	4.4E-16	1.0E-09	4.4E-07

TABLE 3.1-1
Comparison of Average Annual Gaseous Releases to 10 CFR 20 Effluent Concentration Limits (ECL)

Isotope	Release[a] Ci/yr	Boundary Conc.[b] µCi/cc	10 CFR 20 ECL µCi/cc	Fraction of ECL
Mn-56	3.51E-03	2.8E-16	2.0E-08	1.4E-08
Fe-55	6.49E-03	5.2E-16	3.0E-09	1.7E-07
Co-57	1.64E-05	1.3E-18	9.0E-10	1.5E-09
Co-58	4.60E-02	3.7E-15	1.0E-09	3.7E-06
Co-60	1.74E-02	1.4E-15	5.0E-11	2.8E-05
Fe-59	8.11E-04	6.6E-17	5.0E-10	1.3E-07
Ni-63	6.49E-06	5.2E-19	1.0E-09	5.2E-10
Cu-64	1.00E-02	8.1E-16	3.0E-08	2.7E-08
Zn-65	1.11E-02	9.0E-16	4.0E-10	2.2E-06
Rb-89	4.32E-05	3.5E-18	2.0E-07	1.7E-11
Sr-89	6.00E-03	4.9E-16	2.0E-10	2.4E-06
Sr-90	2.40E-03	1.9E-16	6.0E-12	3.2E-05
Y-90	4.59E-05	3.7E-18	9.0E-10	4.1E-09
Sr-91	1.00E-03	8.1E-17	5.0E-09	1.6E-08
Sr-92	7.84E-04	6.3E-17	9.0E-09	7.0E-09
Y-91	2.41E-04	1.9E-17	2.0E-10	9.7E-08
Y-92	6.22E-04	5.0E-17	1.0E-08	5.0E-09
Y-93	1.11E-03	9.0E-17	3.0E-09	3.0E-08
Zr-95	2.00E-03	1.6E-16	4.0E-10	4.0E-07
Nb-95	8.38E-03	6.8E-16	2.0E-09	3.4E-07
Mo-99	5.95E-02	4.8E-15	4.0E-09	1.2E-06
Tc-99m	2.97E-04	2.4E-17	2.0E-07	1.2E-10
Ru-103	3.51E-03	2.8E-16	9.0E-10	3.2E-07
Rh-103m	1.11E-04	9.0E-18	2.0E-06	4.5E-12
Ru-106	1.56E-04	1.3E-17	2.0E-11	6.3E-07
Rh-106	1.89E-05	1.5E-18	1.0E-09	1.5E-09
Ag-110m	2.00E-06	1.6E-19	1.0E-10	1.6E-09
Sb-124	1.81E-04	1.5E-17	3.0E-10	4.9E-08
Sb-125	1.22E-04	9.9E-18	7.0E-10	1.4E-08
Te-129m	2.19E-04	1.8E-17	3.0E-10	5.9E-08
Te-131m	7.57E-05	6.1E-18	2.0E-09	3.1E-09

TABLE 3.1-1
Comparison of Average Annual Gaseous Releases to 10 CFR 20 Effluent Concentration Limits (ECL)

Isotope	Release[a] Ci/yr	Boundary Conc.[b] µCi/cc	10 CFR 20 ECL µCi/cc	Fraction of ECL
Te-132	1.89E-05	1.5E-18	1.0E-09	1.5E-09
Cs-134	6.22E-03	5.0E-16	2.0E-10	2.5E-06
Cs-136	5.95E-04	4.8E-17	9.0E-10	5.3E-08
Cs-137	9.46E-03	7.6E-16	2.0E-10	3.8E-06
Cs-138	1.70E-04	1.4E-17	8.0E-08	1.7E-10
Ba-140	2.70E-02	2.2E-15	2.0E-09	1.1E-06
La-140	1.81E-03	1.5E-16	2.0E-09	7.3E-08
Ce-141	9.19E-03	7.4E-16	8.0E-10	9.3E-07
Ce-144	1.89E-05	1.5E-18	2.0E-11	7.6E-08
Pr-144	1.89E-05	1.5E-18	2.0E-07	7.6E-12
W-187	1.89E-04	1.5E-17	1.0E-08	1.5E-09
Np-239	1.19E-02	9.6E-16	3.0E-09	3.2E-07
Subtotal (w/o H-3)	2.40E+04	-	-	7.2E-02
Tritium (H-3)	3.53E+03	2.9E-10	1.0E-07	2.9E-03
Total	2.76E+04	-	-	7.5E-02

[a] Total release based on composite of the highest activity content of the individual isotopes from the AP1000 (two units), ABWR/ESBWR(one unit), ACR-700 (two units), IRIS (three units), GT-MHR (four modules) and the PBMR (eight modules).

[b] Boundary concentration determined using the highest annual average sector Chi/Q of 2.04E-06 sec/m^3 at the exclusion area boundary.

TABLE 3.1-2
Parameters Used in Gaseous Pathways Dose Analysis

Parameter	Value/Reference	Notes
Source Term	Table 1.4-3 Table 3.1-1	Average annual composite release
Population Data	Table 2.1-2 and 2.1-4	Data for year 2010 by sector
Meteorological Data:		
Annual Chi/Q Average	Table 2.3-53	By sector
Annual D/Q Average	Table 2.3-54	By sector
Annual Decayed Chi/Q Average	Table 2.3-55	By sector
Annual Decayed D/Q Average	Table 2.3-56	By sector

Table 3.1-3
Gaseous Pathways - Expected Individual Doses from Gaseous Effluents

		Dose Rate per Unit (mrem/year)		
LOCATION[b]	**PATHWAY**	**TOTAL BODY**	**SKIN**	**THYROID[a]**
Nearest Residence	Plume	3.9E-01	1.4E+00	-
(0.73 mile SW)				
	Inhalation			
	Adult	1.2E-01	-	4.8E-01
	Teen	1.2E-01	-	6.0E-01
	Child	1.1E-01	-	7.0E-01
	Infant	6.3E-02	-	6.0E-01
Nearest Garden	Vegetables		-	
(0.93 mile N)	Adult	2.7E-01	-	2.6E+00
	Teen	3.6E-01	-	3.6E+00
	Child	6.8E-01	-	7.0E+00
Nearest Meat Animal	Meat			
	Adult	6.1E-02	-	-
(0.93 mile N)	Teen	4.5E-02	-	-
	Child	7.3E-02	-	-
Nearest Milk Cow[c]	Milk			
(5.0 miles N)	Adult	9.7E-03	-	1.5E-01
	Teen	1.4E-02	-	2.4E-01
	Child	2.7E-02	-	4.7E-01
	Infant	5.0E-02	-	1.1E+00
Nearest Milk Goat	Milk			
(4.4 miles SE)	Adult	1.5E-02	-	1.7E-01
	Teen	2.0E-02	-	2.7E-01
	Child	3.4E-02	-	5.4E-01
	Infant	5.9E-02	-	1.3E+00

[a] Thyroid is the maximum organ for individual dose due to pathway and location shown.
[b] Locations are based on Tables 2.3-53 to 2.3-56.
[c] The nearest milking cow used for human consumption is located beyond five miles.

TABLE 3.1-4
Conformance to Regulatory Dose Limits - Gaseous Releases

Type of Dose	Objective	Point of Evaluation	Calculated Dose	Point of Evaluation
		10 CFR 20.1301 Criteria		
TEDE	0.1 rem	Location of highest dose off-site	0.0023 rem	Nearest residence
		10 CFR 50, Appendix I Criteria		
Gamma air dose	10 mrad	Location of highest dose off-site	1.35 mrad	Location of highest Chi/Q at site boundary
Beta air dose	20 mrad	Location of highest dose off-site	2.89 mrad	Location of highest Chi/Q at site boundary
Total body Dose	5 mrem	Location of highest dose off-site	0.88 mrem	Nearest residence
Skin dose	15 mrem	Location of highest dose off-site	2.94 mrem	Nearest residence
Radioiodines and Particulates				
Dose to any organ from all pathways	15 mrem	Location of highest dose off-site	9.44 mrem	Nearest residence

Table 3.1-5
Comparison of Average Annual Liquid Releases to 10 CFR 20 Effluent Concentration Limits (ECL)

Isotope [b]	Release [a] Ci/yr	Boundary Conc. µCi/cc	ECL µCi/cc	Fraction of ECL
C-14	4.40E-04	1.15E-10	3.0E-05	3.8E-06
Na-24	3.26E-03	8.53E-10	5.0E-05	1.7E-05
P-32	1.80E-04	4.71E-11	9.0E-06	5.2E-06
Cr-51	7.70E-03	2.02E-09	5.0E-04	4.0E-06
Mn-54	2.60E-03	6.81E-10	3.0E-05	2.3E-05
Mn-56	3.81E-03	9.98E-10	7.0E-05	1.4E-05
Fe-55	5.81E-03	1.52E-09	1.0E-04	1.5E-05
Fe-59	4.00E-04	1.02E-10	1.0E-05	1.0E-05
Ni-63	1.40E-04	3.66E-11	1.0E-04	3.7E-07
Cu-64	7.51E-03	1.97E-09	2.0E-04	9.8E-06
Co-56	5.19E-03	1.36E-09	6.0E-06	2.3E-04
Co-57	7.19E-05	1.88E-11	6.0E-05	3.1E-07
Co-58	6.72E-03	1.76E-09	2.0E-05	8.8E-05
Co-60	9.11E-03	2.38E-09	3.0E-06	7.9E-04
Zn-65	8.20E-04	2.15E-10	5.0E-06	4.3E-05
W-187	2.60E-04	6.81E-11	3.0E-05	2.3E-06
Np-239	3.11E-03	8.14E-10	2.0E-05	4.1E-05
Br-84	4.00E-05	1.05E-11	4.0E-04	2.6E-08
Rb-88	5.40E-04	1.41E-10	4.0E-04	3.5E-07
Rb-89	4.41E-05	1.15E-11	9.0E-04	1.3E-08
Sr-89	2.00E-04	5.24E-11	8.0E-06	6.5E-06
Sr-90	3.51E-05	9.20E-12	5.0E-07	1.8E-05
Sr-91	9.00E-04	2.36E-10	2.0E-05	1.2E-05
Y-90	3.11E-06	8.14E-13	7.0E-06	1.2E-07
Y-91	1.10E-04	2.88E-11	8.0E-06	3.6E-06
Sr-92	8.00E-04	2.09E-10	4.0E-05	5.2E-06
Y-91m	2.00E-05	5.24E-12	2.0E-03	2.6E-09
Y-92	6.00E-04	1.57E-10	4.0E-05	3.9E-06
Y-93	9.00E-04	2.36E-10	2.0E-05	1.2E-05
Zr-95	1.04E-03	2.72E-10	2.0E-05	1.4E-05

Table 3.1-5
Comparison of Average Annual Liquid Releases to 10 CFR 20 Effluent Concentration Limits (ECL)

Isotope [b]	Release [a]	Boundary Conc.	ECL	Fraction of ECL
	Ci/yr	µCi/cc	µCi/cc	
Nb-95	1.91E-03	5.00E-10	3.0E-05	1.7E-05
Mo-99	1.14E-03	2.98E-10	2.0E-05	1.5E-05
Tc-99m	1.10E-03	2.88E-10	1.0E-03	2.9E-07
Ru-103	9.86E-03	2.58E-09	3.0E-05	8.6E-05
Rh-103m	9.86E-03	2.58E-09	6.0E-03	4.3E-07
Ru-106	1.47E-01	3.85E-08	3.0E-06	1.3E-02
Ag-110m	2.10E-03	5.50E-10	6.0E-06	9.2E-05
Sb-124	6.79E-04	1.78E-11	7.0E-06	2.5E-05
Te-129m	2.40E-04	6.28E-11	7.0E-06	9.0E-06
Te-129	3.00E-04	7.85E-11	4.0E-04	2.0E-07
Te-131m	1.80E-04	4.71E-11	8.0E-06	5.9E-06
Te-131	6.00E-05	1.57E-11	8.0E-05	2.0E-07
I-131	2.83E-02	7.40E-09	1.0E-06	7.4E-03
Te-132	4.80E-04	1.26E-10	9.0E-06	1.4E-05
I-132	3.28E-03	8.59E-10	1.0E-04	8.6E-06
I-133	1.34E-02	3.51E-09	7.0E-06	5.0E-04
I-134	1.70E-03	4.45E-10	4.0E-04	1.1E-06
Cs-134	1.99E-02	5.20E-09	9.0E-07	5.8E-03
I-135	9.94E-03	2.60E-09	3.0E-05	8.7E-05
Cs-136	1.26E-03	3.30E-10	6.0E-06	5.5E-05
Cs-137	2.66E-02	6.97E-09	1.0E-06	7.0E-03
Cs-138	1.90E-04	4.97E-11	4.0E-04	1.2E-07
Ba-140	1.10E-02	2.89E-09	8.0E-06	3.6E-04
La-140	1.49E-02	3.89E-09	9.0E-06	4.3E-04
Ce-141	1.80E-04	4.71E-11	3.0E-05	1.6E-06
Ce-143	3.80E-04	9.95E-11	2.0E-05	5.0E-06
Pr-143	2.60E-04	6.81E-11	2.5E-05	2.7E-06
Ce-144	6.32E-03	1.65E-09	3.0E-06	5.5E-04
Pr-144	6.32E-03	1.65E-09	6.0E-04	2.8E-06

Table 3.1-5
Comparison of Average Annual Liquid Releases to 10 CFR 20 Effluent Concentration Limits (ECL)

Isotope [b]	**Release** [a] Ci/yr	**Boundary Conc.** µCi/cc	**ECL** µCi/cc	**Fraction of ECL**
Subtotal (w/o H 3)	3.81E-01	-	-	3.7E-02
Tritium (H 3)	3.10E+03	8.12E-04	1.0E-03	8.1E-01
Total	3.10E+03	-	-	0.85

[a] Total release based on composite of the highest activity content of the individual isotopes from the AP1000 (two units), ABWR/ESBWR (one unit), ACR-700 (two units), IRIS (three units), GT-MHR (four modules) and the PBMR (eight modules).

[b] Certain nuclides such as Rh-106, Ag-110 and Ba-137m shown in Table 1.4-4 are not included in the above table due to considerations of holdup and decay (short half-life) prior to discharge.

TABLE 3.1-6
Parameters Used in Liquid Pathways Dose Analysis

Parameter	**Value/Reference**	**Notes**
Source Term	Table 1.4-4 Table 3.1-5	Average annual composite release
Population Data	None used	Clinton Lake water not used as a source of public drinking water
Average minimum dilution flow	2400 gpm	Table 1.4-1 Section 10.2.1
Mean annual discharge	198 cfs	The annual discharge from Clinton Lake dam is 212 cfs. Only the flow from the Salt Creek into the lake is credited for effluent dilution. The diluting flow is 198 cfs based on consideration of the Salt Creek and North Fork drainage areas (Tables 2.4-1 and 2.4-3 Clinton ER). The analysis is conservative since more recent data shows greater flow.
Volume of Clinton Lake	74200 ac-ft	SSAR Section 2.4.8.1
Fish Consumption		
Adult	21 kg/yr	Standard LADTAP II values
Teen	16 kg/yr	Standard LADTAP II values
Child	6.9 kg/yr	Standard LADTAP II values
Infant	0.0 kg/yr	Standard LADTAP II values
Shoreline / Swimming / Boating Exposure		
Adult	12 /12/ 100 hr/yr	LADTAP / assumption / assumption
Teen	67 /67 / 67 hr/yr	LADTAP / assumption / assumption
Child	14 / 14 / 14 hr/yr	LADTAP / assumption / assumption
Infant	0 / 0 / 0 hr/yr	LADTAP / assumption / assumption

TABLE 3.1-7
Conformance to Regulatory Dose Limits - Liquid Releases

Type of Dose	Objective	Point of Evaluation	Calculated Dose	Point of Evaluation
		10 CFR 20.1301 Criteria		
TEDE	0.1 rem	Location of the highest dose off-site.	0.001 rem	Clinton Lake
		10 CFR 50 Appendix I Criteria		
Liquid Effluents				
Dose to total body from all pathways	3 mrem/yr	Location of the highest dose off-site.	0.95 mrem/yr Adult	Clinton Lake
Dose to any organ from all pathways	10 mrem/yr	Location of the highest dose off-site.	1.33 mrem/yr Teen Liver	Clinton Lake

TABLE 3.3-1

Limiting Gas Cooled Reactor Design Basis Event Curies Released to Environment by Interval

Isotope	0 to 2 hr	2 to 720 hr
C-14	3.87E+02	0
Br-83	2.00E-02	0
Br-84	8.00E-02	0
Br-85	4.70E-01	0
I-131	0	2.43E+01
I-132	1.10E-01	5.00E-02
I-133	3.00E-02	8.11E+00
I-134	3.80E-01	0
I-135	7.00E-02	7.90E-01
I-136	1.00E-02	0
Kr-83m	2.42E+00	2.00E-02
Kr-85m	7.14E+00	6.40E-01
Kr-85	2.60E+00	1.96E+00
Kr-87	9.84E+00	2.00E-02
Kr-88	1.69E+01	5.60E-01
Kr-89	5.85E+00	0
Kr-90	2.92E+00	0
Kr-91	1.39E+00	2.88E+00
Xe-131m	4.90E-01	8.19E+00
Xe-133m	1.38E+00	4.72E+02
Xe-133	6.01E+01	0
Xe-135m	2.36E+00	1.90E+00
Xe-135	9.28E+00	0
Xe-137	6.17E+00	0
Xe-138	1.13E+01	0
Xe-139	1.78E+00	0
Xe-140	7.90E-01	0
Sr-90	2.00E-05	0
Cs-137	3.00E-04	0

Bounding activities released based on PBMR and GT-MHR.

TABLE 3.3-2
Design Basis Accident Off-Site Dose Consequences

Accident	Reactor Type	EAB Dose TEDE Rem	LPZ Dose TEDE Rem	Guideline TEDE Rem
Main Steam Line Break				
Accident-initiated Iodine Spike	AP1000	2.5E-01	2.7E-01	2.5
Pre-existing Iodine Spike		2.2E-01	7.5E-02	25
Max Equilibrium Iodine Activity	ABWR	1.8E-02	2.4E-03	2.5
Pre-existing Iodine Spike		3.6E-01	4.8E-02	25
Reactor Coolant Pump Locked Rotor	AP1000	7.7E-01	1.1E-01	2.5
Control Rod Ejection Accident	AP1000	9.3E-01	3.1E-01	6.3
Control Rod Drop Accident	ABWR	N/A	N/A	6.3
Steam Generator Tube Rupture				
Accident-initiated Iodine Spike	AP1000	4.6E-01	4.5E-02	2.5
Pre-existing Iodine Spike		9.3E-01	6.3E-02	25
Small Line Break	AP1000	4.0E-01	5.5E-02	2.5
	ABWR	1.5E-02	4.2E-03	2.5
Loss of Coolant Accident	AP1000	7.6E+00	1.8E+00	25
	ABWR	1.2E+00	2.2E+00	25
	ESBWR	1.6E+00	2.0E+00	25
	ACR-700	2.0E+00	1.9E+00	25
Fuel Handling Accident	AP1000	7.4E-01	1.1E-01	6.3
	ABWR	4.2E-01	7.2E-02	6.3

Note:
1. TEDE guidelines from Regulatory Guide 1.183. Small line break guideline based on NUREG-0800, Chapter 15.6.2.
2. N/A - Not applicable due to design of ABWR, see Section 3.3.4.5.

TABLE 3.3-3
AP1000 Main Steam Line Break Curies Released to Environment by Interval – Accident-Initiated Iodine Spike

Isotope	0 to 2 hr	2 to 8 hr	8 to 24 hr	24 to 96 hr
I-130	6.84E-01	3.33E+00	5.27E+00	3.30E+00
I-131	3.92E+01	1.92E+02	5.18E+02	1.35E+03
I-132	9.12E+01	3.26E+02	7.46E+01	6.00E-01
I-133	7.75E+01	3.81E+02	7.54E+02	8.34E+02
I-134	3.03E+01	6.23E+01	8.85E-01	2.78E-06
I-135	5.57E+01	2.59E+02	2.61E+02	5.82E+01
Kr-85m	2.30E-01	3.82E-01	2.26E-01	2.03E-02
Kr-85	9.47E-01	2.83E+00	7.47E+00	2.17E+01
Kr-87	9.24E-02	4.49E-02	1.76E-03	2.84E-07
Kr-88	3.77E-01	4.59E-01	1.34E-01	2.72E-03
Xe-131m	4.28E-01	1.27E+00	3.26E+00	8.78E+00
Xe-133m	5.31E-01	1.51E+00	3.45E+00	6.69E+00
Xe-133	3.95E+01	1.15E+02	2.87E+02	7.03E+02
Xe-135m	1.02E-02	4.44E-05	0	0
Xe-135	1.04E+00	2.31E+00	2.78E+00	1.11E+00
Xe-138	1.34E-02	3.81E-05	0	0
Cs-134	1.91E+01	6.52E-01	1.72E+00	5.00E+00
Cs-136	2.84E+01	9.57E-01	2.47E+00	6.69E+00
Cs-137	1.38E+01	4.70E-01	1.24E+00	3.61E+00
Cs-138	1.02E+01	3.41E-03	1.48E-06	0

TABLE 3.3-4
AP1000 Main Steam Line Break Curies Released to Environment by Interval
Pre-existing Iodine Spike

Isotope	0 to 2 hr	2 to 8 hr	8 to 24 hr	24 to 96 hr
I-130	4.98E-01	4.74E-01	6.95E-01	4.36E-01
I-131	3.37E+01	4.05E+01	1.03E+02	2.67E+02
I-132	4.02E+01	1.39E+01	2.68E+00	2.16E-02
I-133	6.03E+01	6.35E+01	1.17E+02	1.30E+02
I-134	8.24E+00	5.47E-01	4.77E-03	1.50E-08
I-135	3.56E+01	2.73E+01	2.51E+01	5.60E+00
Kr-85m	2.30E-01	3.82E-01	2.26E-01	2.03E-02
Kr-85	9.47E-01	2.83E+00	7.47E+00	2.17E+01
Kr-87	9.24E-02	4.49E-02	1.76E-03	2.84E-07
Kr-88	3.77E-01	4.59E-01	1.34E-01	2.72E-03
Xe-131m	4.28E-01	1.27E+00	3.26E+00	8.78E+00
Xe-133m	5.31E-01	1.51E+00	3.45E+00	6.69E+00
Xe-133	3.95E+01	1.15E+02	2.87E+02	7.03E+02
Xe-135m	1.02E-02	4.44E-05	0	0
Xe-135	1.04E+00	2.31E+00	2.78E+00	1.11E+00
Xe-138	1.34E-02	3.81E-05	0	0
Rb-86	NA	NA	NA	NA
Cs-134	1.91E+01	6.52E-01	1.72E+00	5.00E+00
Cs-136	2.84E+01	9.57E-01	2.47E+00	6.69E+00
Cs-137	1.38E+01	4.70E-01	1.24E+00	3.61E+00
Cs-138	1.02E+01	3.41E-03	1.48E-06	0

NA = Rb-86 contribution considered negligible for this accident.

TABLE 3.3-5
AP1000 Main Steam Line Break
Accident-Initiated Iodine Spike Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	2.47E-01	-
0 to 8 hr	-	1.18E-01
8 to 24 hr	-	7.06E-02
24 to 96 hr	-	8.38E-02
96 to 720 hr	-	0
Total	2.47E-01	2.72E-01

TABLE 3.3-6
AP1000 Main Steam Line Break
Pre-Existing Iodine Spike Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	2.16E-01	-
0 to 8 hr	-	4.43E-02
8 to 24 hr	-	1.34E-02
24 to 96 hr	-	1.73E-02
96 to 720 hr	-	0
Total	2.16E-01	7.50E-02

TABLE 3.3-7
ABWR Main Steam Line Break Outside Containment Curies Released to Environment

Isotope	Maximum Equilibrium Value for Full Power Operation 0 to 2 hr	Pre-existing Iodine Spike 0 to 2 hr
I-131	1.97E+00	3.95E+01
I-132	1.92E+01	3.84E+02
I-133	1.35E+01	2.70E+02
I-134	3.78E+01	7.54E+02
I-135	1.97E+01	3.95E+02
Kr-83m	1.10E-02	6.59E-02
Kr-85m	1.94E-02	1.16E-01
Kr-85	6.11E-05	3.68E-04
Kr-87	6.59E-02	3.97E-01
Kr-88	6.65E-02	4.00E-01
Kr-89	2.67E-01	1.60E+00
Kr-90	6.89E-02	4.19E-01
Xe-131m	4.76E-05	2.86E-04
Xe-133m	9.16E-04	5.51E-03
Xe-133	2.56E-02	1.54E-01
Xe-135m	7.81E-02	4.59E-01
Xe-135	7.30E-02	4.38E-01
Xe-137	3.32E-01	2.00E+00
Xe-138	2.55E-01	1.53E+00
Xe-139	1.17E-01	7.00E-01

TABLE 3.3-8
ABWR Main Steam Line Break Outside Containment
Maximum Equilibrium Value for Full Power Operation Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	1.78E-02	-
0 to 8 hr	-	2.40E-03
8 to 24 hr	-	0
24 to 96 hr	-	0
96 to 720 hr	-	0
Total	1.78E-02	2.40E-03

TABLE 3.3-9
ABWR Main Steam Line Break Outside Containment
Pre-existing Iodine Spike Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	3.56E-01	-
0 to 8 hr	-	4.79E-02
8 to 24 hr	-	0
24 to 96 hr	-	0
96 to 720 hr	-	0
Total	3.56E-01	4.79E-02

TABLE 3.3-10
AP1000 Locked Rotor Accident Curies Released to Environment
Pre-existing Iodine Spike

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

TABLE 3.3-11
AP1000 Locked Rotor Accident, 0 to 1.5 Hour Duration
Pre-existing Iodine Spike Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	7.71E-01	-
0 to 8 hr	-	1.11E-01
8 to 24 hr	-	0
24 to 96 hr	-	0
96 to 720 hr	-	0
Total	7.71E-01	1.11E-01

TABLE 3.3-12
AP1000 Control Rod Ejection Accident Curies Released to Environment by Interval
Pre-existing Iodine Spike

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

TABLE 3.3-13
AP1000 Control Rod Ejection Accident
Pre-existing Iodine Spike Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	9.25E-01	-
0 to 8 hr	-	2.58E-01
8 to 24 hr	-	4.37E-02
24 to 96 hr	-	6.12E-03
96 to 720 hr	-	1.15E-03
Total	9.25E-01	3.09E-01

TABLE 3.3-14
AP1000 Steam Generator Tube Rupture Accident Curies Released to Environment by Interval
Accident Initiated Iodine Spike

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

NA = Rb-86 contribution considered negligible for this accident.

TABLE 3.3-15
AP1000 Steam Generator Tube Rupture Accident Curies Released to Environment by Interval
Pre-existing Iodine Spike

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

NA = Rb-86 contribution considered negligible for this accident.

TABLE 3.3-16
AP1000 Steam Generator Tube Rupture
Accident-Initiated Iodine Spike Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	4.63E-01	-
0 to 8 hr	-	3.32E-02
8 to 24 hr	-	1.21E-02
24 to 96 hr	-	0
96 to 720 hr	-	0
Total	4.63E-01	4.53E-02

TABLE 3.3-17
AP1000 Steam Generator Tube Rupture
Pre-existing Iodine Spike Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	9.25E-01	-
0 to 8 hr	-	5.90E-02
8 to 24 hr	-	4.37E-03
24 to 96 hr	-	0
96 to 720 hr	-	0
Total	9.25E-01	6.34E-02

Table 3.3-18

AP1000 Small Line Break Accident Curies Released to Environment Accident-Initiated Iodine Spike

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

TABLE 3.3-19
AP1000 Small Line Break Accident, 0 to 0.5 Hour Duration
Accident-Initiated Iodine Spike Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	4.01E-01	-
0 to 8 hr	-	5.53E-02
8 to 24 hr	-	0
24 to 96 hr	-	0
96 to 720 hr	-	0
Total	4.01E-01	5.53E-02

TABLE 3.3-20
ABWR Small Line Break Outside Containment
Activity Released to Environment

Isotope	Curies Released 0 to 2 hr	Curies Released 0 to 8 hr
I-131	1.84E+00	3.81E+00
I-132	1.61E+01	3.22E+01
I-133	1.24E+01	2.55E+01
I-134	2.68E+01	5.14E+01
I-135	1.78E+01	3.62E+01
Total	7.50E+01	1.49E+02

TABLE 3.3-21
ABWR Small Line Break Outside Containment Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	1.54E-02	-
0 to 8 hr	-	4.21E-03
8 to 24 hr	-	0
24 to 96 hr	-	0
96 to 720 hr	-	0
Total	1.54E-02	4.21E-03

TABLE 3.3-22
AP1000 Design Basis Loss of Coolant Accident Curies Released to Environment by Interval

Isotope	0 to 1 hr	2 to 3 hr	0 to 8 hr	8 to 24 hr	24 to 96 hr	96 to 720 hr
Halogen Group						
I-130	5.62E+00	4.92E+01	7.80E+01	2.96E+00	1.11E+00	1.99E-02
I-131	1.54E+02	1.44E+03	2.36E+03	1.56E+02	3.74E+02	1.12E+03
I-132	1.79E+02	1.18E+03	1.67E+03	7.64E+00	2.29E-02	0
I-133	3.11E+02	2.80E+03	4.51E+03	2.16E+02	1.63E+02	1.62E+01
I-134	1.96E+02	7.51E+02	1.02E+03	1.26E-01	1.07E-07	0
I-135	2.75E+02	2.27E+03	3.50E+03	8.31E+01	9.55E+00	4.95E-03
Noble Gas Group						
Kr-85m	6.74E+01	1.31E+03	3.77E+03	1.87E+03	1.71E+02	2.43E-03
Kr-85	3.08E+00	7.32E+01	2.96E+02	7.05E+02	3.17E+03	2.70E+04
Kr-87	9.54E+01	1.14E+03	1.94E+03	4.97E+01	8.11E-03	0
Kr-88	1.70E+02	2.95E+03	7.26E+03	1.70E+03	3.49E+01	8.16E-07
Xe-131m	3.07E+00	7.28E+01	2.94E+02	6.79E+02	2.74E+03	1.11E+04
Xe-133m	1.68E+01	3.92E+02	1.54E+03	3.15E+03	8.21E+03	5.15E+03
Xe-133	5.49E+02	1.30E+04	5.19E+04	1.16E+05	4.11E+05	8.10E+05
Xe-135m	1.44E+01	2.14E+01	3.59E+01	2.14E-07	0	0
Xe-135	1.32E+02	2.85E+03	9.64E+03	1.01E+04	4.21E+03	1.73E+01
Xe-138	5.31E+01	6.69E+01	1.20E+02	1.58E-07	0	0
Alkali Metal Group						
Rb-86	3.32E-01	2.61E+00	4.26E+00	9.37E-02	2.03E-03	1.05E-02
Cs-134	2.81E+01	2.22E+02	3.63E+02	8.06E+00	1.88E-01	1.59E+00
Cs-136	8.01E+00	6.30E+01	1.03E+02	2.25E+00	4.72E-02	2.03E-01
Cs-137	1.64E+01	1.29E+02	2.11E+02	4.70E+00	1.10E-01	9.39E-01
Cs-138	1.06E+02	2.06E+02	3.19E+02	6.92E-04	0	0
Tellurium Group						
Sr-89	3.23E+00	7.56E+01	1.19E+02	2.87E+00	6.54E-02	4.60E-01
Sr-90	2.78E-01	6.52E+00	1.03E+01	2.48E-01	5.82E-03	4.97E-02
Sr-91	3.77E+00	8.14E+01	1.22E+02	1.74E+00	2.76E-03	1.44E-05
Sr-92	3.45E+00	6.13E+01	8.30E+01	3.26E-01	1.06E-05	0

TABLE 3.3-22 (CONTINUED)
AP1000 Design Basis Loss of Coolant Accident Curies Released to Environment by Interval

Isotope	0 to 1 hr	2 to 3 hr	0 to 8 hr	8 to 24 hr	24 to 96 hr	96 to 720 hr
Tellurium Group (continued)						
Sb-127	8.55E-01	1.98E+01	3.11E+01	7.13E-01	1.16E-02	1.60E-02
Sb-129	2.25E+00	4.43E+01	6.28E+01	4.83E-01	1.01E-04	1.00E-09
Te-127m	1.10E-01	2.58E+00	4.06E+00	9.83E-02	2.27E-03	1.77E-02
Te-127	7.99E-01	1.72E+01	2.57E+01	3.65E-01	5.63E-04	2.72E-06
Te-129m	3.76E-01	8.80E+00	1.38E+01	3.33E-01	7.47E-03	4.79E-02
Te-129	1.50E+00	1.89E+01	2.32E+01	8.54E-03	7.27E-10	0
Te-131m	1.15E+00	2.62E+01	4.05E+01	8.29E-01	6.86E-03	1.60E-03
Te-132	1.14E+01	2.65E+02	4.15E+02	9.42E+00	1.44E-01	1.60E-01
Ba-139	3.83E+00	5.30E+01	6.63E+01	4.73E-02	2.03E-08	0
Ba-140	5.71E+00	1.33E+02	2.10E+02	5.00E+00	1.05E-01	4.41E-01
Noble Metals Group						
Mo-99	7.63E-01	1.77E+01	2.76E+01	6.19E-01	8.79E-03	7.72E-03
Tc-99m	6.09E-01	1.26E+01	1.83E+01	1.94E-01	1.08E-04	2.73E-08
Ru-103	6.07E-01	1.42E+01	2.23E+01	5.38E-01	1.21E-02	8.11E-02
Ru-105	3.59E-01	7.08E+00	1.01E+01	7.97E-02	1.82E-05	2.40E-10
Ru-106	2.00E-01	4.67E+00	7.36E+00	1.78E-01	4.16E-03	3.46E-02
Rh-105	3.70E-01	8.48E+00	1.32E+01	2.76E-01	2.64E-03	8.48E-04
Lanthanide Group						
Y-90	2.90E-03	6.65E-02	1.04E-01	2.32E-03	3.25E-05	2.75E-05
Y-91	4.19E-02	9.71E-01	1.53E+00	3.69E-02	8.43E-04	6.09E-03
Y-92	3.70E-02	6.93E-01	9.64E-01	5.77E-03	5.86E-07	0
Y-93	4.75E-02	1.02E+00	1.53E+00	2.25E-02	4.05E-05	2.91E-07
Nb-95	5.64E-02	1.31E+00	2.06E+00	4.95E-02	1.11E-03	7.23E-03
Zr-95	5.61E-02	1.30E+00	2.05E+00	4.94E-02	1.13E-03	8.29E-03
Zr-97	5.35E-02	1.19E+00	1.81E+00	3.26E-02	1.38E-04	7.58E-06

TABLE 3.3-22 (CONTINUED)
AP1000 Design Basis Loss of Coolant Accident Curies Released to Environment by Interval

Isotope	0 to 1 hr	2 to 3 hr	0 to 8 hr	8 to 24 hr	24 to 96 hr	96 to 720 hr
Lanthanide Group (continued)						
La-140	6.06E-02	1.38E+00	2.14E+00	4.58E-02	4.84E-04	1.97E-04
La-141	4.69E-02	8.98E-01	1.26E+00	8.69E-03	1.31E-06	0
La-142	3.58E-02	5.15E-01	6.53E-01	6.67E-04	6.96E-10	0
Nd-147	2.19E-02	5.06E-01	7.95E-01	1.89E-02	3.88E-04	1.49E-03
Pr-143	4.93E-02	1.14E+00	1.79E+00	4.27E-02	9.01E-04	3.95E-03
Am-241	4.23E-06	9.81E-05	1.54E-04	3.74E-06	8.75E-08	7.48E-07
Cm-242	9.98E-04	2.31E-02	3.64E-02	8.81E-04	2.04E-05	1.64E-04
Cm-244	1.22E-04	2.84E-03	4.47E-03	1.08E-04	2.53E-06	2.16E-05
Cerium Group						
Ce-141	1.37E-01	3.19E+00	5.02E+00	1.21E-01	2.71E-03	1.72E-02
Ce-143	1.25E-01	2.85E+00	4.42E+00	9.20E-02	8.29E-04	2.34E-04
Ce-144	1.03E-01	2.41E+00	3.80E+00	9.19E-02	2.14E-03	1.77E-02
Pu-238	3.22E-04	7.51E-03	1.18E-02	2.86E-04	6.71E-06	5.73E-05
Pu-239	2.83E-05	6.60E-04	1.04E-03	2.52E-05	5.90E-07	5.04E-06
Pu-240	4.15E-05	9.69E-04	1.53E-03	3.69E-05	8.65E-07	7.39E-06
Pu-241	9.33E-03	2.17E-01	3.42E-01	8.30E-03	1.94E-04	1.66E-03
Np-239	1.60E+00	3.69E+01	5.76E+01	1.27E+00	1.67E-02	1.17E-02

TABLE 3.3-23
AP1000 Design Basis Loss of Coolant Accident Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
1 to 3 hr	7.65E+00	-
0 to 8 hr	-	1.70E+00
8 to 24 hr	-	5.54E-02
24 to 96 hr	-	4.12E-02
96 to 720 hr	-	2.78E-02
Total	7.65E+00	1.82E+00

Notes:
1. The EAB dose is greatest during the two-hr period between 1 and 3 hours after start of this accident.
2. LOCA based on Regulatory Guide 1.183.

TABLE 3.3-24
ABWR LOCA Curies Released to Environment by Interval

Isotope	0 to 2 hr	0 to 8 hr	8 to 24 hr	24 to 96 hr	96 to 720 hr
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 3.3-25
ABWR Design Basis Loss of Coolant Accident Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	1.22E+00	-
0 to 8 hr	-	2.76E-01
8 to 24 hr	-	1.89E-01
24 to 96 hr	-	6.39E-01
96 to 720 hr	-	1.12E+00
Total	1.22E+00	2.22E+00

Note: LOCA based on Regulatory Guide 1.3 and TID-14844.

TABLE 3.3-26
ESBWR Design Basis Loss of Coolant Accident Curies Released to Environment by Interval

Isotope	0 to 1.4 hr	1.4 to 3.4 hr	0 to 8 hr	8 to 24 hr	24 to 96 hr	96 to 720 hr
Halogen Group						
I-131	9.28E+01	2.85E+02	8.72E+02	1.60E+03	5.09E+03	6.64E+03
I-132	1.21E+02	3.11E+02	7.18E+02	4.42E+02	1.02E+03	4.80E+02
I-133	1.89E+02	5.56E+02	1.62E+03	2.09E+03	2.36E+03	1.50E+02
I-134	1.01E+02	1.09E+02	2.31E+02	0	0	0
I-135	1.66E+02	4.42E+02	1.16E+03	6.90E+02	1.40E+02	0
Noble Gas Group						
Kr-85m	1.09E+02	7.25E+02	2.90E+03	3.83E+03	6.40E+02	0
Kr-85	3.56E+00	2.96E+01	1.75E+02	1.24E+03	1.23E+04	1.99E+05
Kr-87	1.30E+02	5.02E+02	1.09E+03	7.00E+01	0	0
Kr-88	2.43E+02	1.42E+03	4.72E+03	2.82E+03	1.10E+02	0
Xe-133	7.68E+02	6.36E+03	3.70E+04	2.46E+05	1.89E+06	6.68E+06
Xe-135	2.02E+02	1.66E+03	8.14E+03	2.44E+04	1.90E+04	1.00E+02
Alkali Metal Group						
Rb-86	4.50E-02	1.30E-01	4.03E-01	7.37E-01	2.40E+00	2.91E+00
Cs-134	1.36E+01	3.95E+01	1.22E+02	2.28E+02	7.90E+02	1.26E+03
Cs-136	3.64E+00	1.06E+01	3.25E+01	5.90E+01	1.87E+02	2.04E+02
Cs-137	8.14E+00	2.37E+01	7.32E+01	1.37E+02	4.72E+02	7.58E+02
Tellurium Group						
Sr-89	4.70E+00	2.15E+01	6.27E+01	1.19E+02	4.03E+02	5.85E+02
Sr-90	3.33E-01	1.53E+00	4.45E+00	8.55E+00	2.94E+01	4.75E+01
Sr-91	5.62E+00	2.36E+01	6.07E+01	5.03E+01	2.00E+01	0
Sr-92	4.78E+00	1.60E+01	3.30E+01	4.90E+00	1.00E-01	0
Sb-127	9.76E-01	4.43E+00	1.28E+01	2.23E+01	5.73E+01	3.06E+01
Sb-129	2.85E+00	1.08E+01	2.44E+01	8.60E+00	6.00E-01	0
Te-127	9.51E-01	4.36E+00	1.26E+01	2.33E+01	6.51E+01	4.80E+01
Te-127m	1.28E-01	5.89E-01	1.72E+00	3.29E+00	1.14E+01	1.78E+01
Te-129	3.11E+00	1.30E+01	3.19E+01	2.69E+01	6.22E+01	8.50E+01

TABLE 3.3-26 (CONTINUED)
ESBWR Design Basis Loss of Coolant Accident Curies Released to Environment by Interval

Isotope	0 to 1 hr	2 to 3 hr	0 to 8 hr	8 to 24 hr	24 to 96 hr	96 to 720 hr
Tellurium Group (continued)						
Te-129m	8.43E-01	3.87E+00	1.13E+01	2.13E+01	7.14E+01	9.80E+01
Te-131m	1.58E+00	7.02E+00	1.97E+01	2.86E+01	4.23E+01	5.30E+00
Te-132	1.57E+01	7.10E+01	2.04E+02	3.51E+02	8.55E+02	4.00E+02
Ba-139	4.82E+00	1.21E+01	2.15E+01	5.00E-01	0	0
Ba-140	8.33E+00	3.81E+01	1.11E+02	2.06E+02	6.49E+02	7.04E+02
Noble Metals Group						
Co-58	3.24E-03	1.49E-02	4.33E-02	8.27E-02	2.80E-01	4.18E-01
Co-60	3.88E-03	1.78E-02	5.19E-02	9.91E-02	3.43E-01	5.56E-01
Mo-99	1.02E+00	4.61E+00	1.32E+01	2.22E+01	5.11E+01	1.95E+01
Tc-99m	8.91E-01	4.09E+00	1.19E+01	2.14E+01	5.21E+01	2.06E+01
Ru-103	7.81E-01	3.58E+00	1.04E+01	1.98E+01	6.64E+01	9.34E+01
Ru-105	4.37E-01	1.65E+00	3.78E+00	1.37E+00	1.10E-01	0
Ru-106	2.12E-01	9.78E-01	2.84E+00	5.42E+00	1.87E+01	2.97E+01
Rh-105	3.91E-01	1.79E+00	5.17E+00	8.43E+00	1.44E+01	2.40E+00
Lanthanide Group						
Y-90	4.85E-03	3.54E-02	1.90E-01	1.35E+00	1.33E+01	4.16E+01
Y-91	5.78E-02	2.69E-01	8.07E-01	1.72E+00	6.26E+00	9.31E+00
Y-92	4.03E-01	3.88E+00	1.58E+01	1.50E+01	1.10E+00	0
Y-93	6.74E-02	2.84E-01	7.36E-01	6.44E-01	2.80E-01	0
Zr-95	7.55E-02	3.47E-01	1.01E+00	1.92E+00	6.51E+00	9.66E+00
Zr-97	7.42E-02	3.24E-01	8.77E-01	1.04E+00	9.00E-01	2.00E-02
Nb-95	7.14E-02	3.28E-01	9.56E-01	1.83E+00	6.33E+00	1.02E+01
La-140	1.37E-01	1.14E+00	6.70E+00	4.90E+01	4.12E+02	7.42E+02
La-141	6.45E-02	2.38E-01	5.32E-01	1.59E-01	9.00E-03	0
La-142	4.57E-02	1.21E-01	2.21E-01	7.00E-03	0	0
Pr-143	7.23E-02	3.33E-01	9.75E-01	1.92E+00	6.67E+00	7.94E+00
Nd-147	3.22E-02	1.47E-01	4.27E-01	7.93E-01	2.46E+00	2.52E+00
Am-241	3.72E-06	1.71E-05	4.98E-05	9.62E-05	3.37E-04	5.87E-04

TABLE 3.3-26 (CONTINUED)
ESBWR Design Basis Loss of Coolant Accident Curies Released to Environment by Interval

Isotope	0 to 1 hr	2 to 3 hr	0 to 8 hr	8 to 24 hr	24 to 96 hr	96 to 720 hr
Lanthanide Group (continued)						
Cm-242	9.81E-04	4.50E-03	1.31E-02	2.51E-02	8.58E-02	1.34E-01
Cm-244	5.29E-05	2.43E-04	7.08E-04	1.35E-03	4.69E-03	7.55E-03
Cerium Group						
Ce-141	1.89E-01	8.71E-01	2.53E+00	4.79E+00	1.60E+01	2.18E+01
Ce-143	1.80E-01	8.05E-01	2.26E+00	3.37E+00	5.37E+00	8.00E-01
Ce-144	1.23E-01	5.64E-01	1.64E+00	3.14E+00	1.08E+01	1.71E+01
Pu-238	1.67E-04	7.68E-04	2.24E-03	4.28E-03	1.48E-02	2.39E-02
Pu-239	4.24E-05	1.95E-04	5.68E-04	1.09E-03	3.78E-03	6.16E-03
Pu-240	5.31E-05	2.44E-04	7.10E-04	1.36E-03	4.70E-03	7.53E-03
Pu-241	9.14E-03	4.20E-02	1.22E-01	2.34E-01	8.14E-01	1.30E+00
Np-239	2.37E+00	1.07E+01	3.06E+01	5.05E+01	1.09E+02	3.50E+01

TABLE 3.3-27
ESBWR Design Basis Loss of Coolant Accident Off-Site Dose Consequences

Time	**EAB Dose TEDE Rem**	**LPZ Dose TEDE Rem**
0 to 2 hr	1.61E+00	-
0 to 8 hr	-	6.54E-01
8 to 24 hr	-	4.16E-01
24 to 96 hr	-	6.53E-01
96 to 720 hr	-	2.97E-01
Total	1.61E+00	2.02E+00

Note: LOCA based on Regulatory Guide 1.183

TABLE 3.3-28
ACR-700 Design Basis Large Loss of Coolant Accident
Curies Released to Environment by Interval

Isotope	0 to 2 hr	0 to 8 hr	8 to 24 hr	24 to 96 hr	96 to 720 hr
I-131	7.76E+01	3.06E+02	5.84E+02	1.56E+04	4.24E+03
I-132	8.55E+01	1.71E+02	1.61E+01	1.42E+01	0
I-133	1.59E+02	5.78E+02	7.75E+02	1.52E+04	6.20E+01
I-134	8.91E+01	1.12E+02	5.10E-02	0	0
I-135	1.37E+02	4.12E+02	2.49E+02	2.36E+03	0
Kr-83m	2.09E+03	3.76E+03	1.91E+02	0	0
Kr-85m	5.70E+03	1.52E+04	5.67E+03	2.60E+02	0
Kr-85	4.50E+01	1.81E+02	3.63E+02	8.13E+02	6.78E+03
Kr-87	7.98E+03	1.18E+04	1.50E+02	0	0
Kr-88	1.45E+04	3.21E+04	5.20E+03	5.30E+01	0
Kr-89	8.64E+02	8.64E+02	0	0	0
Xe-131m	2.52E+02	1.00E+03	1.94E+03	3.91E+03	1.55E+04
Xe-133m	1.40E+03	5.37E+03	9.16E+03	1.19E+04	7.45E+03
Xe-133	4.56E+04	1.79E+05	3.35E+05	5.94E+05	1.16E+06
Xe-135m	1.78E+03	1.79E+03	0	0	0
Xe-135	3.74E+03	1.21E+04	1.01E+04	2.10E+03	9.00E+00
Xe-137	1.89E+03	1.89E+03	0	0	0
Xe-138	6.78E+03	6.79E+03	0	0	0

TABLE 3.3-29
ACR-700 Large Loss of Coolant Accident Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	1.96E+00	-
0 to 8 hr	-	5.74E-01
8 to 24 hr	-	1.51E-01
24 to 96 hr	-	1.06E+00
96 to 720 hr	-	8.32E-02
Total	1.96E+00	1.87E+00

TABLE 3.3-30
AP1000 Fuel Handling Accident
Curies Released to Environment

Isotope	0 to 2 hr
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

Notes:
1. Activity is based on a 100 hr shutdown period before fuel movement begins.
2. Source term and pool DF based on Regulatory Guide 1.183.

TABLE 3.3-31
AP1000 Fuel Handling Accident Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	7.40E-01	-
0 to 8 hr	-	1.11E-01
8 to 24 hr	-	0
24 to 96 hr	-	0
96 to 720 hr	-	0
Total	7.40E-01	1.11E-01

TABLE 3.3-32
ABWR Fuel Handling Accident
Curies Released to Environment by Interval

Isotope	0 to 2 hr	2 to 8 hr
I-131	1.23E+02	1.82E+00
I-132	1.52E+02	1.29E+00
I-133	1.27E+02	1.77E+00
I-134	6.16E-06	2.13E-08
I-135	2.06E+01	2.52E-01
Kr-83m	6.43E+00	4.57E+00
Kr-85m	8.54E+01	9.14E+01
Kr-85	4.78E+02	6.76E+02
Kr-87	1.23E-02	6.51E-03
Kr-88	2.43E+01	2.21E+01
Kr-89	8.14E-11	1.00E-20
Xe-131m	0	0
Xe-133m	8.35E+01	1.18E+02
Xe-133	1.10E+03	1.52E+03
Xe-135m	2.81E+04	3.95E+04
Xe-135	2.21E+02	2.34E+00
Xe-137	6.38E+03	7.84E+03
Xe-138	2.07E-10	2.81E-19
Xe-138	0	0

Notes:
1. Activity is based on a 24 hr shutdown before fuel movement begins.
2. Source term and pool DF are based on Regulatory Guide 1.25.

TABLE 3.3-33
ABWR Fuel Handling Accident Off-Site Dose Consequences

Time	EAB Dose TEDE Rem	LPZ Dose TEDE Rem
0 to 2 hr	4.18E-01	-
0 to 8 hr	-	7.16E-02
8 to 24 hr	-	-
24 to 96 hr	-	-
96 to 720 hr	-	-
Total	4.18E-01	7.16E-02

Note: LPZ dose includes contribution from activity remaining in reactor building. See Section 3.3.4.13.