CHAPTER 7

# Environmental Impacts of Postulated Accidents Involving Radioactive Materials

The purpose of this section is to review and analyze a sufficiently robust spectrum of design basis accidents (DBA) and severe accidents to bracket the postaccident radiological consequences for the spectrum of reactors under consideration and provide results for use in this report.  Analysis of severe accidents and mitigation of those accidents will be deferred until the COL stage.

# 7.1 Design Basis Accidents

The radiological consequences of potential 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 (USNRC, 1983) site dispersion factors at the exclusion and LPZ 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.

## 7.1.1 Selection of Design Basis 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, in order to permit quantitative assessment of the spectrum of potential risks at the EGC ESP Site.

It is not necessary or practical to analyze 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 considered spectrum of accidents 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, 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 are 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. 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 postaccident design assessment. 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 source term methods and NUREG-0800 guidance used for the ABWR certification. To demonstrate EGC ESP Site suitability, a conservative ESBWR LOCA assessment is provided.

  – IRIS: The low core power level and advanced design features (such as the elimination of 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 well finalized for this advanced concept, the vendor anticipates that postaccident radiological consequences will be well 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 USNRC publications are directed toward LWR technology and are not typically applicable to the assessment of the gas-cooled reactors.

  Depressurization events are usually the critical considerations for gas-cooled reactors. The terms coolant, primary coolant, and pressure boundary when used with gas reactor technology differ 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.

  The GTMHR and PBMR use mechanistic accident source terms and postulate relatively small environmental releases compared with the water reactor technologies. The limiting DBA environmental releases specified by the gas reactors vendors are provided in Table 7.1-1. Based on these projections of limiting environmental releases, the postaccident radiological dose consequences would result in less than 0.2 percent of the 10 CFR 50.34 acceptance criteria limits. 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.

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)

### 7.1.2 Evaluation of Radiological Consequences

Doses for the selected DBAs were 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 meltdowns of the core 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 DBAs of greater frequency but with less magnitude. For these accidents, the more restrictive dose limits in Regulatory Guide 1.183 and the NUREG-0800, Standard Review Plan, were used to make certain that the accidents were acceptable from an overall risk perspective (USNRC, 2000 and USNRC, 1987).

The evaluations used short-term accident chi/Qs. The chi/Qs were determined using Regulatory Guide 1.145 methods with on-site meteorology data (USNRC, 1983). The site 50th percentile chi/Qs from Table 2.7-52 of the SSAR were used in these evaluations.

The 0- to 2-hour Chi/Q value is used for the 2-hour release duration with the greatest dose consequences at the EAB.

- EAB
  - 0 to 2 hrs
- LPZ
  - 0 to 8 hrs
  - 8 to 24 hrs
  - 1 to 4 days
  - 4 to 30 days

The accident doses are expressed as total effective dose equivalents (TEDEs) 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, 1988). The DDE is taken the same as the effective dose equivalent from external exposure and the dose conversions in Federal Guidance Report 12 (USEPA, 1993) are applied.

### 7.1.3 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 Technical Information Document (TID)-14844 (USAEC, 1962).

The ESBWR and the AP1000 source term and approach to assessing accidents are based on the AST methods 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.

## 7.1.4 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 7.1-2. This table also compares the environmental doses to the recommended limits based on Regulatory Guide 1.183 and NUREG-0800. Table 7.1-2 shows that the evaluated dose consequences meet the accident-specific acceptance criteria invoked in Section 7.1.2.

The analysis approach for evaluating the AP1000 design basis accidents discussed in the following subsections is based upon the EAB and LPZ doses provided by Westinghouse and given in Chapter 15 of the AP1000 Design Control Document, Tier 2, Revision 2 and the ratio of the ESP Site Chi/Q value to the AP1000 representative site Chi/Q value for each post accident time period. The AP1000 representative site Chi/Q values used in the evaluations are given in Table 7.1-2A. Based upon the revisions made to the Chi/Q values by Westinghouse to support the final AP1000 design certification, the EAB doses presented in Tables 7.1-2, 7.1-5, 7.1-6, 7.1-11, 7.1-13, 7.1-16, 7.1-17, 7.1-19, 7.1-23 and 7.1-31 will increase by approximately 3.6% and the LPZ doses will remain bounding.

### 7.1.4.1 Main Steam Line Break Outside Containment (AP1000)

The bounding AP1000 steam line break for the radiological consequence evaluation occurs outside containment. The facility is designed so that only one steam generator experiences an uncontrolled blowdown even if one of the main steam isolation valves fail 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 the 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 would be no significant accumulations of the noble gases in the steam generators prior to the accident since they are rapidly released during normal service. Noble gases released during the accident would primarily be due to the increase in primary to secondary side leakage assumed during the event. Reactor coolant leakage to the intact steam generator would mix with the existing inventory and increase the secondary side concentrations. This effect would normally be offset by alkali and iodine partitioning in the generator. However, for conservatism, the calculated activity release assumes the primary to secondary side activity in the intact generator that is also leaked directly to the environment. The calculated doses are based on activity releases that assume:

- Duration of accident – 72 hrs

- 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 microcurie per gram (μCi/g) dose equivalent Xe-133
- Accident initiated iodine spike – 500 times the fuel release rate that occurs when the reactor coolant equilibrium activity is 1.0 μCi/g dose equivalent Iodine-131
- Preexisting iodine spike – reactor coolant at 60 μCi/g dose equivalent Iodine-131
- Fuel damage - none

The activities released to the environment for the accident initiated and preexisting iodine spike cases are shown in Tables 7.1-3 and 7.1-4, respectively.

The vendor calculated time-dependent off-site doses for a representative site. The doses were reevaluated using the EGC ESP Site short-term accident dispersion characteristics in Table 2.3-52 of the SSAR.

The TEDE doses for the accident initiated iodine spike are shown in Table 7.1-5. The doses at the EAB and LPZ are a small fraction of the 25-roentgen equivalent man (rem) TEDE identified in 10 CFR 50.34 (USNRC, 2000). A "small fraction" is defined as 10 percent or less in the Standard Review Plan and Regulatory Guide 1.183. The doses for the preexisting iodine spike are shown in Table 7.1-6. These doses also meet the TEDE dose guidelines of 10 CFR 50.34.

### 7.1.4.2 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 would initially be limited by the flow restrictor in the upstream reactor steam nozzle and the remaining flow restrictors in the three unbroken main steam lines feeding the downstream end of the break. Closure of the main steam isolation valves would terminate the mass flows out of the break.

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

- Duration of accident – 2 hrs
- Main steam isolation valve closure – 5 seconds

- Mass releases from break – steam 12,870 kilograms; water 21,950 kilograms
- Reactor coolant maximum equilibrium activity – corresponding to an offgas release rate of 100,000 μCi/s referenced to a 30 minute decay
- Preexisting iodine spike – corresponding to an offgas release rate of 400,000 μCi/s referenced to a 30 minute decay
- Fuel damage – none

The activity released to the environment for the maximum activity and preexisting spike cases is shown in Table 7.1-7.

The calculated doses for the maximum allowed equilibrium activity at full power operation are shown in Table 7.1-8. The calculated doses for the preaccident iodine spike are shown in Table 7.1-9. The EAB and LPZ doses are a small fraction of the 25-rem TEDE dose guidelines of 10 CFR 50.34.

#### 7.1.4.3 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, which results 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 hrs
- Steam released – 6.48E+05 lbm
- Primary/secondary side coolant masses – 3.7E+05 lbm/6.06E+05 lbm
- Primary to secondary leak rate – 350 lbm/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

- Preexisting iodine spike – reactor coolant at 60 μCi/g dose equivalent Iodine-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 preexisting 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 7.1-10.

The vendor calculated the time-dependant off-site doses for a representative site. The doses were reevaluated using the EGC ESP Facility short-term accident dispersion characteristics in Table 2.3-52 of the SSAR. The TEDE doses for the locked rotor accident are shown in Table 7.1-11. The doses at the EAB and LPZ are a small fraction of the TEDE identified in 10 CFR 50.34.

#### 7.1.4.4 Control Rod Ejection (AP1000)

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

Two activity release paths contribute to this event. First, the equilibrium activity in the reactor coolant and the activity from the damaged fuel are blown down through the failed pressure housing to the containment atmosphere. The activity can leak to the environment over a relatively long period due to the containment'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 the 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 unlashed portion mixes with secondary side fluids where partitioning occurs prior to the release as steam.

The dose consequences analyses are performed using guidance in Regulatory Guides 1.77 and 1.183 (USAEC, 1974 and USNRC, 2000). 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
- Preexisting iodine spike – reactor coolant at 60 μCi/g dose equivalent Iodine-131
- Fraction of rods with cladding failures – 0.10
- Fission product gap activity fractions:
  - Iodines – 0.10
  - Noble gases – 0.10
  - Alkali metals – 0.12
- Fraction of fuel melting – 0.0025
- 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 preexisting 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 7.1-12.

The vendor calculated the time-dependent off-site doses for a representative site. The doses were reevaluated using the EGC ESP Site short-term accident dispersion characteristics in Table 2.3-52 of the SSAR. The doses at the EAB and LPZ shown in Table 7.1-13 are well within the 25-rem TEDE identified in 10 CFR 50.34.

#### 7.1.4.5 Rod Drop Accident (ABWR)

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

### 7.1.4.6 Steam Generator Tube Rupture (AP1000)

The AP1000 steam generator tube rupture accident assumes the complete severance of one steam generator tube. The accident causes an increase in the secondary side activity due to reactor coolant flow through the ruptured tube. With the loss of 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 through the tube break flashes to steam while the unflashed portion mixes with the secondary side inventory. Iodines and alkali metals in the flashed fluid are not partitioned during steam releases while activity in the secondary side of the faulted generator is partitioned prior to release as steam. The following assumptions have been used:

- Duration of accident – 24 hrs
- Total flow through ruptured tube – 3.85E+05 lbm
- Steam release from faulted steam generator – 3.32E+0+5 lbm
- Steam released from intact steam generator – 1.42E+06 lbm
- Steam release duration – 13.2 hrs
- Primary/secondary side initial coolant masses – 3.8E+05 lbm/3.7E+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
- Preexisting iodine spike – reactor coolant at 60 μCi/g dose equivalent Iodine-131
- Accident initiated iodine spike – 335 times the fuel release rate that occurs when the reactor coolant equilibrium activity is 1.0 μCi/g dose equivalent Iodine-131
- Partition coefficients in steam generators – 0.01 for iodines and alkali metals
- Off-site power and condenser – lost on reactor trip

- Fuel damage - none

The activities released to the environment for the accident-initiated and preexisting iodine spike cases are shown in Tables 7.1-14 and 7.1-15, respectively.

The vendor calculated the time-dependent off-site doses for a representative site. The doses were reevaluated using the EGC ESP Site short-term accident dispersion characteristics in Table 2.3-52 of the SSAR. The TEDE doses for the steam generator tube rupture accident with the accident-initiated iodine spike are shown in Table 7.1-16. The preexisting iodine spike doses are shown in Table 7.1-17. The doses at the EAB and LPZ are a small fraction of the 25-rem TEDE identified in 10 CFR 50.34.

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

Small lines carrying reactor coolant outside the AP1000 containment include the reactor coolant system sample line and the chemical and volume control system discharge line to the radwaste system. These lines are not continuously used. The 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, 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 to be 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 hrs
- 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 Iodine-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 Iodine-131
- Fuel damage - none

The activities released are shown in Table 7.1-18.

Based on the vendor calculated off-site doses for a representative site, the time-dependent doses were reevaluated using the EGC ESP Site short-term accident meteorology in Table 2.3-52 of the SSAR. The results are shown in Table 7.1-19. The resulting doses at the EAB and LPZ are a small fraction of the 25-rem TEDE in 10 CFR 50.34.

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

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

The iodine in the flashed water is assumed to be transported to the environs by the heating, ventilation and air conditioning (HVAC) system without credit for treatment by the standby gas treatment system. 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 hrs
- Standby gas treatment system – not credited
- Reactor building release rate – 200 percent/hr
- Mass of reactor coolant released – 13,610 kilograms
- Mass of fluid flashed to steam – 2,270 kilograms
- Iodine plateout fraction – 0.5
- Reactor coolant equilibrium activity – maximum permitted by technical specifications corresponding to an offgas release rate of 100,000 μCi/s referenced to a 30-minute decay.
- Iodine spiking – accident initiated spike
- Fuel damage – none

The activity released to the environs is shown in Table 7.1-20. The calculated EAB and LPZ doses are shown in Table 7.1-21. The doses are a small fraction of the 25-rem TEDE limit in 10 CFR 50.34.

#### 7.1.4.9 Large Break Loss of Coolant Accident (AP1000)

The core response analysis for the AP1000 demonstrates that the reactor core maintains its integrity for the large break LOCA. However, significant core degradation and melting is assumed in this DBA. The assumption of major core damage is intended to challenge various accident mitigation features and provide a conservative basis for calculating 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 (USNRC, 1995; USNRC, 2000; and Westinghouse, 2002).

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

The core release starts after the 10-minute blowdown of reactor coolant. The fuel rod gap activity is released over the next half hour followed by an in-vessel core melt that lasts 1.3

hrs. 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 Iodine-131
- Reactor coolant mass – 3.7E+05 lbm
- Containment purge flow rate – 8,800 cfm for 30 seconds
- Containment leak rate – 0.1 percent per day
- Core activity group release fractions – Regulatory Guide 1.183, 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 decontamination factor (DF) of 200 is reached
- Containment atmosphere particulate removal rate – 0.43/hr to 0.7/hr during first 24 hrs

Table 7.1-22 gives the activities released to the environment for the AP1000 large break LOCA.

Based on the vendor calculated off-site doses for a representative site, the time-dependent doses were reevaluated using the EGC ESP Site short-term accident meteorology in Table 2.3-52 of the SSAR. Table 7.1-23 provides 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 7.1-23 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.

### 7.1.4.10 Large Break Loss of Coolant Accident (ABWR)

This ABWR event postulates piping breaks inside containment of varying sizes, types, and locations. The break type includes steam and liquid process lines. The emergency core cooling analyses show that the core temperature and pressure transients caused by the breaks are insufficient to cause fuel cladding perforation. Although no fuel damage occurs, conservative assumptions from Regulatory Guide 1.3 (USAEC, 1974a) are invoked in order to conservatively assess postaccident fission product mitigation systems and the resultant off-site doses.

One hundred percent of the core-inventory noble gases and 50 percent of the iodines are instantaneously released from the reactor to the drywell at the beginning of the accident. Of the iodines, 50 percent are assumed to immediately plateout, which leaves 25 percent of the inventory airborne and available for release. Following the break and depressurization of the reactor, some of the noncondensable 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.

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

- Duration – 30 days
- Core power level – 4,005 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 7.1-24. The doses at the EAB and LPZ are summarized in Table 7.1-25. The doses are within the 25-rem TEDE guidelines of 10 CFR 50.34.

#### 7.1.4.11 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 postaccident fission product mitigation systems and the resultant off-site doses.

One hundred percent of the core-inventory noble gases, 30 percent of the iodines, 25 percent of the core cesium, and minor fractions (less than 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.

Postaccident 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 percent), the external events shield building (45 percent), and a small fraction is released directly to the environment (5 percent). 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 – 4,000 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 7.1-26. The doses at the EAB and LPZ are summarized in Table 7.1-27. The doses are within the 25-rem TEDE guidelines of 10 CFR 50.34.

#### 7.1.4.12 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, which causes 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 severe accident, the heavy water in the moderator acts as the heat sink and the heat is transferred to the service water. The integrity of the pressure tube, calandria tube, and the heat transfer system core cooling geometry are maintained.

The 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 CsI and dissolves as non-volatile iodine in water. The remaining 5 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 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 postaccident period. In addition, a passive component failure (such as an 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 in 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 offers a very high reliability of containment isolation and reduces this potential impairment mechanism.

The containment isolation systems are credited with isolating 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, effectively promoting 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, paragraph 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, paragraph 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 during the large LOCA is shown in Table 7.1-28. The resulting doses at the EGC ESP Site EAB and LPZ are summarized in Table 7.1-29. The EAB and LPZ doses are within the 25-rem TEDE guidelines in 10 CFR 50.34.

#### 7.1.4.13 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 FHAs inside containment, the release to the environment can be mitigated by automatically closing the containment purge lines after detection of radioactivity in the containment atmosphere. For accidents in the spent fuel pool, activity is released through the auxiliary building ventilation system to the environment.

The refueling and fuel transfer systems are designed such that the damaged fuel has a minimum depth of 23 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 reevolve as elemental iodine immediately after damage to the fuel assembly. The dose activity released presumes:

- Core thermal power – 3,468 MWt
- Decay time after shutdown – 100 hrs
- Activity release period – 2 hrs
- 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 (USNRC, 2000)
- 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 7.1-30.

The resulting doses at the EAB and LPZ are summarized in Table 7.1-31. The doses are applicable to fuel handling accidents inside containment and in the spent fuel pool in the auxiliary building (10 CFR 50). 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, Standard Review Plan (USNRC, 2000 and 1987).

### 7.1.4.14 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 minutes. Pool water is credited with removal of elemental iodine released from the failed rods. Guidance from Regulatory Guide 1.25 is used in performance of the analysis. Key aspects include:

- Core thermal power – 4,005 MWt
- Decay time after shutdown – 24 hrs
- Activity release period from pool – 2 hrs
- 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.e
- Pool decontamination for iodine – Regulatory Guide 1.25, regulatory position C.1.f
- 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 – 99 percent efficiency
- Standby gas system exhaust rate – 50 percent/day

The radioactivity released to the environment is provided in Table 7.1-32.

The doses at the site EAB and LPZ are summarized in Table 7.1-33. Activity remaining in the reactor building after two hours is assumed filtered and released without benefit of decay over the next six hours 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 (USAEC, 1972 and USNRC, 2000). The EAB and LPZ doses are shown to be well within the 25-rem TEDE guidelines of 10 CFR 50.34.

# 7.2 Severe Accidents

This section discusses the probabilities and consequences of accidents of greater severity than the design basis accidents. As a class, they are considered less likely to occur, but because their consequences could be more severe, they are considered important both in terms of impact to the environment and off-site costs. These severe accidents, can be distinguished from design basis accidents in two primary respects: (1) they involve substantial physical deterioration of the fuel in the reactor core, including overheating to the point of melting, and (2) they involve deterioration of the capability of the containment system to perform its intended function of limiting the release of radioactive materials to the environment. In NUREG-1437, the *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* [GEIS], the USNRC generically assessed the impacts of severe accidents during license renewal periods, using the results of existing analyses and site-specific information to conservatively predict the environmental impacts of severe accidents for each plant during the renewal period (USNRC, 1996). This methodology is used as a basis for evaluating the severe accident environmental impacts of a new nuclear power plant that may be built on the EGC ESP Site.

## 7.2.1 Applicability of Existing Generic Severe Accident Studies

Section 5.3.3 of NUREG-1437 presents a thorough assessment of impacts of severe accidents during the license renewal period by the USNRC staff. Methodologies therein were developed to evaluate each of the dose pathways by which a severe accident may result in adverse environmental impacts and to estimate off-site costs of severe accidents. This assessment methodology and the resulting conclusions are considered, for reasons discussed below, broadly applicable beyond the license renewal context, including evaluation of severe accident impacts associated with determining site suitability for a nuclear power plant. The three NUREG-1437 pathways for release of radioactive material to the environment from severe accidents, i.e., atmospheric, air to surface water, and groundwater to surface water, are discussed in this section. The economic impacts from severe accidents are also comparatively evaluated in this section.

The GEIS evaluations and conclusions are based on existing assessments of severe accident impacts presented in numerous Final Environmental Statements (FES) published after 1980 and for a representative set of U.S. plants and sites in NUREG-1150. The GEIS results are expressed as a range of values in terms of risk of severe accident impact per reactor-year of operation. The USNRC later confirmed, in 61 FR 28480, that "the analyses performed for the GEIS represent adequate, plant-specific estimates of the impacts from severe accidents…" (USNRC, 1996a).

As described in the GEIS, the purpose of the evaluation of severe accidents was "to use, to the extent possible, the available severe accident results, in conjunction with those factors that are important to risk and that change with time to estimate the consequences of nuclear plant accidents for all plants for a time period that exceeds the time frame of existing analyses." This estimation process was completed by predicting increases or decreases in consequences as the plant lifetime was extended past the normal license period by considering the projected changes in the risk factors. The primary assumption in this analysis was that regulatory controls ensure that the physical plant condition (i.e., the predicted probability of and radioactive releases from an accident) is maintained at a constant level during the renewal period; therefore, the frequency and magnitude of a release remains relatively constant. In other words, significant changes in consequences would result only from changes in the plant's external environment. The logical approach, then, would be to incorporate the most significant environmental factors into calculations of consequences for subsequent correlation with existing analyses (which use the consequence computer codes).

The staff concluded in NUREG-1437 that the primary factors affecting risk are the site population (which reflects the number of people potentially at risk to severe accident exposure) and wind direction (which reflects the likelihood of exposure). Secondary factors, such as terrain, rainfall, and wind stability, also have some effect on risk, but their impact was judged to be much smaller than the effects of population and wind direction. These factors were included in the FES analyses whose results are the bases for the GEIS analyses. Consequently, their effects are indirectly considered in the prediction of future risks and are reflected within the uncertainty bounds generated by the regression of the FES risk values. To ensure that the existing FES analyses covered a range of secondary factors representative of the total population of plants, the more significant secondary factors were also examined in the GEIS. Variations in these factors (precipitation, 50-mi population, 0-mi population in the direction of highest wind frequency, general terrain and emergency planning) were found to be enveloped by the FES analyses and thus reasonably accounted for in the GEIS evaluation of severe accidents.

Detailed severe accident consequence (early and latent fatalities and total dose) evaluations were not available for all plants considered in the GEIS. Therefore, a predictor for these consequences was developed using correlations based upon the calculated results from the existing FES severe accident analyses. This predictor was then used to infer the future consequence level of all individual nuclear plants. Correlations were developed using two environmental parameters that are available for all plants. This correlation process was well described in NUREG-1437.

While the NUREG-1437 discussions dealt with the environmental impacts of accidents during operation after license renewal. The primary assumption for this evaluation was that the frequency (or likelihood of occurrence) of an accident at a given plant would not increase during the plant lifetime (inclusive of the license renewal period) because regulatory controls ensure the plant's licensing basis is maintained and improved, where warranted. The GEIS use of severe accident risk per reactor-year of operation as the principal metric for evaluating severe accident environmental impacts and the assumption that this risk remains constant over the life of the plant are equally applicable and appropriate in both the license renewal and ESP/COL context. Therefore, the thorough generic analysis of severe accident impacts presented in the GEIS also provides an appropriate basis and method for evaluating severe accident impacts for early site permitting.

However, it was recognized that the changing environment around the plant is not subject to regulatory controls and introduces the potential for changing risk. Thus, the site-specific environmental considerations, i.e., population and meteorology, were evaluated in the GEIS and are considered in the following sections.

Specifically, the following evaluation of the significant factors associated with the environment shows these factors for the EGC ESP Site are not substantially different from those factors identified for previously analyzed sites. Thus, it follows that the environmental impacts for the EGC ESP Site will not be substantially different from the acceptable environmental impacts identified for the previously analyzed sites.

## 7.2.2 Evaluation of Potential Severe Accident Releases

EGC has identified the significance of the impacts associated with each issue as either Small, Moderate, or Large, consistent with the criteria that USNRC established in 10 CFR 51, Appendix B, Table B-1, Footnote 3 as follows:

- SMALL - Environmental effects are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource. For the purposes of assessing radiological impacts, the Commission has concluded that those impacts that do not exceed permissible levels in the Commission's regulations are considered small.
- MODERATE - Environmental effects are sufficient to alter noticeably, but not to destabilize, any important attribute of the resource.
- LARGE - Environmental effects are clearly noticeable and are sufficient to destabilize any important attributes of the resource.

In accordance with National Environmental Policy Act practice, EGC considered ongoing and potential additional mitigation in proportion to the significance of the impact to be addressed (i.e., impacts that are small receive less mitigative consideration than impacts that are large).

### 7.2.2.1 Evaluation of Potential Releases via Atmospheric Pathway

The site-specific significant factors of demography and meteorology are considered in the evaluation of the atmospheric exposure pathway for the EGC ESP Site. For this evaluation, NUREG-1437 calculates an exposure index (EI) for use in comparing the relative risk for the current fleet of nuclear power plants.

NUREG-1437 provides the following discussion of EI:

> *"Population, which changes over time, defines the number of people within a given distance from the plant. Wind direction, which is assumed not to change from year to year, helps determine what proportion of the population is at risk in a given direction, because radionuclides are carried by the wind. Therefore, an EI relationship was developed by multiplying the wind direction frequency (fraction of the time per year) for each of 16 (22.5°) compass sectors times the population in that sector for a given distance from the plant and summing all products.…Population varies with population growth and movement, and with the distance from any given plant. As the population changes for that plant, the EI also changes (the larger the EI, the larger the number of people at risk). Thus, EI is proportional to risk and an EI for a site for a future year can be used to predict the risk to the population around that site in that future year."*

Thus, the EI is a function of population surrounding the plant, weighted by the site-specific wind direction frequency, and is, therefore, a site-specific parameter. Because meteorological patterns, including wind direction frequency, tend to remain constant over time, the site meteorology will not be significantly different for the EGC ESP Site than the meteorology considered in NUREG-1437 for the Clinton site and only population can significantly affect the resulting risk in any given year of reactor operation.

However, the 50-mi population projections for the EGC ESP Site (i.e., ~914,000) are not significantly different than for the Clinton site as projected for the year 2050 in Table 5.3 of NUREG-1437, (i.e., ~870,000). Thus, the EGC ESP Site EI will not be significantly different from those established in NUREG-1437 for the Clinton site**.**

Two EIs were evaluated in NUREG-1437. A 10-mi EI was found to best correlate with early fatalities, and a 150-mi EI was found to best correlate with latent fatalities and total dose. Using these indices, it was determined that the risk of early and latent fatalities from individual nuclear power plants is small and represents only a small fraction of the risk to which the public is exposed from other sources.

The 10-mi EI for the Clinton site was 760, as shown in NUREG-1437, Table 5.7, for the year 2050. The 10-mi EI range provided (in Table 5.7 of NUREG-1437) for the current generation of nuclear power plant sites has a low of 96 and a high of 18,959. Thus, the EGC ESP Site is expected to be within the range of risk calculated for the existing fleet of nuclear power plants.

The 150-mi EI for the CPS Site was 1,418,383, as shown in NUREG-1437, Table 5.8, for the year 2050. The 150-mi EI range provided (in Table 5.8 of NUREG-1437) for the current generation of nuclear power plant sites has a low of 132,195 and a high of 2,863,844. Thus, the EGC ESP Site is expected to be within the range of risk calculated for the existing fleet of nuclear power plants.

Thus, the EGC ESP Site risks for the atmospheric exposure pathway will be within the range of those considered as "Small" in NUREG-1437. Section 5.5.2.1 of NUREG-1437 indicated these predicted effects of a severe accident "are not expected to exceed a small fraction of that risk to which the population is already exposed."

#### 7.2.2.2 Evaluation of Potential Releases via Atmospheric Fallout onto Open Bodies Of Water

This section examines such radiation exposure risk for a nuclear power reactor at the EGC ESP Site in the event of a severe reactor accident in which radioactive contaminants are released into the atmosphere and subsequently deposited onto open bodies of water. In the GEIS, the drinking water pathway was treated separately while the aquatic food, swimming, and shoreline pathways were addressed collectively. Population dose estimates for both the drinking water and aquatic food pathways were then compared with estimates from the atmospheric pathway.

As reported in NUREG-1437, analyses for both the drinking water and aquatic food pathways were performed with and without considering interdiction. In the case of the drinking-water pathway, the Great Lakes and the estuarine sites are bound by those of a previous site evaluation (i.e., Fermi); while small river sites with relatively low annual flow rates, long residence times, and large surface-area-to-volume ratios may potentially not be bound by the previous analysis. In all cases, however, interdiction can reduce relative risk to levels at or below that of the previous acceptable analysis and significantly below that for the atmospheric pathway. River sites that may have relatively high concentrations of contaminants but which remove contaminants within short periods of time (hours to several days) are amenable to short-term interdiction. A similar level of reduced risk can be achieved at those sites with longer residence times (months) by more extensive interdictive measures.

For the aquatic food pathway, population dose and population exposure per reactor-year are directly related to aquatic food harvest. For river sites, un-interdicted population exposure is an order of magnitude lower than that for the atmospheric pathway. For Great Lakes sites, the un-interdicted population exposure is a substantial fraction of that predicted for the atmospheric pathway but is reduced significantly by interdiction. For estuarine sites with large annual aquatic food harvests, dose reduction of a factor of 2 to 10 through interdiction provides essentially the same population exposure estimates as the atmospheric pathway.

For these reasons, population dose for the drinking-water pathway was found to be a small fraction of that for the atmospheric pathway. Risk associated with the aquatic food pathway was found to be small relative to the atmospheric pathway for most sites and essentially the same as the atmospheric pathway for the few sites with large annual aquatic food harvests.

Environmental parameters important for input in performing the above analyses, and for use in analyses of additional sites, are (1) the surface area of the receiving body, (2) the volume of water in the body, and (3) the flow rate. In the absence of rigorous site-specific analyses, these data can provide estimates of the extent of contamination in the receiving water body and the residence time of the contaminant in the affected water body. Comparing these estimates and site environmental parameters with those for the previously evaluated site, i.e., Fermi, can provide some indication of the comparative hazard associated with drinking contaminated surface water among sites and the need for site-specific analyses. Accounting for population and meteorological data in the comparison can provide further indication of relative risk among sites.

The above-identified environmental parameters have been identified in the GEIS for the Clinton site. These same parameters are applicable for the EGC ESP Site (since these environmental parameters are generally constant for a given site and no major changes have been identified that would impact these parameters), thus, the drinking-water pathway and the aquatic food, swimming, and shoreline pathways for the EGC ESP Site are comparable to those considered in the GEIS evaluation. Therefore, the risk from the air fallout to a water body exposure pathway generally compares favorably with the risk to the population from atmospheric releases and the EGC ESP Site risks for the water body exposure pathway will also be within the range of those considered as “Small” in NUREG-1437.

#### 7.2.2.3 Evaluation of Potential Releases to Groundwater

This section discusses the potential for radiation exposure from the groundwater pathway as the result of postulated severe accidents at a nuclear reactor on the EGC ESP Site. Severe accidents are the only accidents capable of producing significant groundwater contamination.

As identified in NUREG-1437, groundwater contamination due to severe accidents has been evaluated generically in NUREG-0440, Liquid Pathway Generic Study (LPGS) (USNRC, 1978). The LPGS assumes that core melt with subsequent basemat melt-through occurs, and evaluates the consequences. The LPGS examines six generic sites using typical or comparative assumptions on geology, adsorption factors, etc.

Per NUREG-1437, the LPGS results are believed to provide generally conservative uninterdicted population dose estimates in the six generic plant-site categories. Five of these categories are site groupings in common locations adjacent to small rivers, large rivers, the Great Lakes, oceans, and estuaries. In a severe accident, contaminated groundwater could reach nearby surface water bodies, and the population could be exposed to this source of contamination through drinking of surface water, ingestion of finfish and shellfish, and shoreline contact. Exposure by drinking contaminated groundwater is considered to be minor or nonexistent in these five categories because of a limited number of drinking-water wells. The sixth category is a "dry" site located either at a considerable distance from surface water bodies or where groundwater flow is away from a nearby surface water body. In this case, the only population exposure results from drinking contaminated groundwater.

NUREG-1437 concludes that the risk from the groundwater exposure pathway generally contributes only a small fraction of that risk attributable to the population from the atmospheric pathway but in a few cases may contribute a comparable risk.

In the GEIS analysis, site-specific information on groundwater travel time; retention-adsorption coefficients; distance to surface water; and soil, sediment, and rock characteristics is compared with previous groundwater contamination analyses. Previous analyses are contained in the LPGS and site-specific FESs. These environmental parameters have been identified in the GEIS for the Clinton site. These same parameters are applicable for the EGC ESP Site (since these environmental parameters are generally constant for a given site and no major changes have been identified that would impact these parameters); thus, the groundwater pathway for the EGC ESP Site is comparable to those considered in the GEIS evaluation.Therefore, the risk from the groundwater exposure pathway generally compares favorably with the risk to the population from atmospheric releases and the EGC ESP site risks for the groundwater exposure pathway will also be within the range of those considered as "Small" in NUREG-1437.

### 7.2.3 Evaluation of Economic Impacts of Severe Accidents

This section discusses the potential economic impact as the result of postulated severe accidents at a nuclear reactor on the EGC ESP Site. Similar to Section 7.2.2.1, the EI is used as a predictor of cost because, as identified in the GEIS, the cost should be dependent upon the economic impact in the same way and for the same reason that population dose estimates are dependent on the EI values.

As noted in NUREG-1437, FES analyses used the "Calculation of Reactor Accident Consequences" (CRAC) computer code to calculate off-site severe accident costs for the area contaminated by the accident. The off-site costs that were considered relate to avoidance of adverse health effects and are categorized as follows:

- Evacuation costs;
- Value of crops contaminated and condemned;
- Value of milk contaminated and condemned;
- Costs of decontamination of property where practical; and

- Indirect costs resulting from the loss of use of property and incomes derived therefrom (including interdiction to prevent human injury).

For those FES analyses that addressed severe accidents, the off-site accident costs were estimated to be to be as high as 6 billion dollars to 8 billion dollars (1994 dollars) but with accident probabilities that were extremely low (1E-6 years), as would be expected for this class of events. Because key variables (used in the FES cost analyses) are strongly related to population density, NUREG-1437 further evaluated the FES results using normalization techniques and the 150-mile EI values. This evaluation, which included the Clinton site, demonstrated that the FES cost predictions remained valid, even considering population changes represented by the EI values.

In addition, the generic NUREG-1437 predicted conditional land contamination is small (10 ac/yr at most). This is also consistent with (USNRC 1975) and a 1982 study on siting criteria (USNRC, 1982) which predicts small conditional land contamination values. The GEIS concluded that land contamination values for the evaluated plants can be considered representative of all plants since they cover the major vendor and containment types and include sites at the upper end of annual rainfall. However, even considering that land contamination values can vary at other sites, it is not expected that predicted land contamination from plants at other sites would vary more than 1 or 2 orders of magnitude from the values listed above and would, therefore, still be a small impact.Based on the evaluations of the expected economic costs and land contamination as a result of a severe accident, the GEIS concludes in Section 5.5.2.4 that the conditional impacts in both cases are of small significance for all plants. As for other aspects of the GEIS evaluation of severe accident impacts, this evaluation and conclusion is broadly applicable to beyond the license renewal context. Thus the economic impacts and land contamination resulting from postulated severe accidents at a new nuclear reactor or reactors on the EGC ESP Site should be comparable as well (i.e., within the range of those considered as “Small” in NUREG-1437).

## 7.2.4 Consideration of Commission Severe Accident Policy

In 1985, the USNRC adopted a Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (USNRC, 1985). This policy statement indicated:

> *“The Commission fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs. This expectation is based on:*
>
> *The growing volume of information from industry and government-sponsored research and operating reactor experience has improved our knowledge of specific severe accident vulnerabilities and of low-cost methods for their mitigation. Further learning on safety vulnerabilities and innovative methods is to be expected.*
>
> *The inherent flexibility of this Policy Statement (that permits risk-risk tradeoffs in systems and sub-systems design) encourages thereby innovative ways of achieving an improved overall systems reliability at a reasonable cost.*
>
> *Public acceptance, and hence investor acceptance, of nuclear technology is dependent on demonstrable progress in safety performance, including the reduction in frequency of*

> *accident precursor events as well as a diminished controversy among experts as to the adequacy of nuclear safety technology."*

Thus, implementation of the Commission's Severe Accident Policy can be expected to show that the environmental impact of any new reactor(s) on the EGC ESP Site will be within the range of risk previously determined to be "Small."

A significant factor in the risk associated with the plant design is the frequency of the considered accident sequences. As indicated above, the designs certified in accordance with 10 CFR 52 are expected to exhibit a "higher standard of severe accident safety performance than the prior designs." The ABWR is a currently certified design under 10 CFR 52, Appendix A, and is considered to be representative of advanced light water reactor standard designs. The USNRC Safety Evaluation Report (SER) for the ABWR states "the ABWR design and the submittals made for the ABWR in the SSAR meet the intent of the Commission's Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants" (USNRC, 1994). Similar findings have been made for the other currently certified designs, i.e., the System 80+ and the AP-600. Thus, the Severe Accident Policy Statement expectations have been met for each of the three advanced standard designs considered to-date by the USNRC and are expected to continue to be met for future design certifications and COL approvals.

## 7.2.5 Conclusion

- The GEIS concludes, based on the generic evaluations presented, that the probability-weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to ground water and societal and economic impacts from severe accidents are "Small" for all plants.
- As described above, the methodology and evaluations of the GEIS are applicable to the consideration of new plants in the ESP and/or COL context. Evaluation of site specific factors for purposes of this application have shown that the EGC ESP Site is within the range of sites considered in the GEIS. Thus we conclude that the GEIS conclusion is applicable to the EGC ESP Site.
- Use of pertinent site specific information to confirm the applicability of existing generic analyses is consistent with USNRC staff plans for addressing severe accident environmental impacts at ESP as identified in SECY-91-041 (USNRC, 1991).

In summary, the environmental impacts considered in NUREG-1437 evaluations include potential radiation exposures to individuals and to the population as a whole, the risk of near- and long-term adverse health effects that such exposures could entail, and the potential economic and societal consequences of accidental contamination of the environment. These impacts could be severe, but due to their low likelihood of occurrence, the impacts are judged to be small. This conclusion is based on (1) considerable experience gained with the operation of similar facilities without significant degradation of the environment; (2) the requirement that in order to obtain a license the applicant must comply with the applicable Commission regulations and requirements; and (3) a previously analyzed assessment of the risk of design-basis and severe accidents (USNRC, 1999).

Specifically, based on the USNRC and industry implementation of the 1985 policy statement, the generic NUREG-1437 risk evaluations, and the EGC ESP Site specific demography and meteorology, the probability weighted consequences of atmospheric and (surface and ground) water pathways, and the societal and economic impacts for severe accidents for a future nuclear power plant on the EGC ESP Site will also be "Small."

# 7.3 Severe Accident Mitigation Alternatives

The purpose of severe accident mitigation alternatives (SAMA) is to review and evaluate plant-design alternatives that could significantly reduce the radiological risk from a severe accident by preventing substantial core damage (i.e., preventing a severe accident) or by limiting releases from containment in the event that substantial core damage occurs (i.e., mitigating the impacts of a severe accident) (USNRC, 1999).

No design has been selected and SAMAs cannot be meaningfully discussed in this ESP application. SAMAs are design issues evaluated during standard design certification, and any discussion is more appropriately developed when a certified design is selected and submitted in a COL application. The design of the reactor and analyses of projected severe accidents are major contributing factors in the determination of SAMAs. In order to determine whether mitigation alternatives are cost beneficial, severe accident analyses must be included in these evaluations. A design has not been selected; therefore, these mitigation alternatives cannot be meaningfully evaluated in this Application for the EGC ESP.

# 7.4 Transportation Accidents

The assessment of transportation accidents is provided in Section 3.8.

CHAPTER 7

# References

## Chapter Introduction

None

## Section 7.1

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## Section 7.2

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## Section 7.3

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# Section 7.4

None

CHAPTER 7

# Tables

**TABLE 7.1-1**

PBMR 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 |

**TABLE 7.1-1**

PBMR Design Basis Event Curies Released to Environment by Interval

| Isotope | 0 to 2 hr | 2 to 720 hr |
|---|---|---|
| Sr-90 | 2.00E-05 | 0 |
| Cs-137 | 3.00E-04 | 0 |

Note: Bounding activities released based on PBMR and GT-MHR.

**TABLE 7.1-2**
Comparison of Reactor Types for Limiting Off-Site Dose Consequences

| Design Basis Accident | ESP EAB Dose TEDE (rem) | Vendor EAB Dose TEDE (rem) | ESP/Vendor EAB X/Q Ratio | ESP LPZ Dose TEDE (rem) | Vendor LPZ Dose TEDE (rem) | ESP/Vendor LPZ X/Q Ratio |
|---|---|---|---|---|---|---|
| **AP1000 Reactor** | | | | | | |
| **Main Steam Line Break** | | | | | | |
| *Accident-initiated Iodine Spike* | | | | | | |
| 0 -2 hrs | 4.75E-02 | 8.00E-01 | 5.93E-02 | | | |
| 0 - 8 hrs | | | | 1.61E-02 | 6.4E-01 | 2.52E-02 |
| 8 - 24 hrs | | | | 1.20E-02 | 4.2E-01 | 2.85E-02 |
| 24 -96 hrs | | | | 2.16E-02 | 6.3E-01 | 3.43E-02 |
| **Total** | **4.75E-02** | **8.00E-01** | | **4.97E-02** | **1.69E+00** | |
| *Preexisting Iodine Spike* | | | | | | |
| 0 -2 hrs | 4.15E-02 | 7.00E-01 | 5.93E-02 | | | |
| 0 - 8 hrs | | | | 6.04E-03 | 2.40E-01 | 2.52E-02 |
| 8 - 24 hrs | | | | 2.28E-03 | 8.00E-02 | 2.85E-02 |
| 24 -96 hrs | | | | 4.45E-03 | 1.30E-01 | 3.43E-02 |
| **Total** | **4.15E-02** | **7.00E-01** | | **1.28E-02** | **4.50E-01** | |
| **Reactor Coolant Pump Locked Rotor** | | | | | | |
| 0 -2 hrs | 1.48E-01 | 2.50E+00 | 5.93E-02 | | | |
| 0 - 8 hrs | | | | 1.51E-02 | 6.00E-01 | 2.52E-02 |
| **Total** | **1.48E-01** | **2.50E+00** | | **1.51E-02** | **6.00E-01** | |
| **Control Rod Ejection Accident** | | | | | | |
| 0 -2 hrs | 1.78E-01 | 3.00E+00 | 5.93E-02 | | | |
| 0 - 8 hrs | | | | 3.53E-02 | 1.4E+00 | 2.52E-02 |
| 8 - 24 hrs | | | | 7.41E-03 | 2.6E-01 | 2.85E-02 |
| 24 -96 hrs | | | | 1.58E-03 | 4.6E-02 | 3.43E-02 |
| 96 - 720 hrs | | | | 5.45E-04 | 1.2E-02 | 4.55E-02 |
| **Total** | **1.78E-01** | **3.00E+00** | | **4.48E-02** | **1.72E+00** | |
| **Steam Generator Tube Rupture** | | | | | | |
| *Accident-initiated Iodine Spike* | | | | | | |
| 0 -2 hrs | 8.90E-02 | 1.50E+00 | 5.93E-02 | - | - | - |

**TABLE 7.1-2**

Comparison of Reactor Types for Limiting Off-Site Dose Consequences

| Design Basis Accident | ESP EAB Dose TEDE (rem) | Vendor EAB Dose TEDE (rem) | ESP/Vendor EAB X/Q Ratio | ESP LPZ Dose TEDE (rem) | Vendor LPZ Dose TEDE (rem) | ESP/Vendor LPZ X/Q Ratio |
|---|---|---|---|---|---|---|
| 0 – 8 hrs | | | | 4.53E-03 | 1.80E-01 | 2.52E-02 |
| 8 – 24 hrs | | | | 2.05E-03 | 7.2E-02 | 2.85E-02 |
| **Total** | **8.90E-02** | **1.50E+00** | | **6.60E-03** | **2.52E-01** | |
| *Preexisting Iodine Spike* | | | | | | |
| 0 –2 hrs | 1.78E-01 | 3.00E+00 | 5.93E-02 | - | - | |
| 0 - 8 hrs | | | | 8.06E-03 | 3.20E-01 | 2.52E-02 |
| 8 - 24 hrs | | | | 7.41E-04 | 2.60E-02 | 2.85E-02 |
| **Total** | **1.78E-01** | **3.00E+00** | | **8.80E-03** | **3.46E-01** | |
| **Small Line Break** | | | | | | |
| 0 -2 hrs | 7.71E-02 | 1.30E+00 | 5.93E-02 | | | |
| 0 - 8 hrs | | | | 7.56E-03 | 3.00E-01 | 2.52E-02 |
| **Total** | **7.71E-02** | **1.30E+00** | | **7.56E-03** | **3.00E-01** | |
| **Fuel Handling Accident** | | | | | | |
| 0 -2 hrs | 1.42E-01 | 2.40E+00 | 5.93E-02 | | | |
| 0 - 8 hrs | | | | 1.51E-02 | 6.00E-01 | 2.52E-02 |
| **Total** | **1.42E-01** | **2.40E+00** | | **1.51E-02** | **6.00E-01** | |
| **Loss of Coolant Accident** | | | | | | |
| 1 - 3 hrs | 1.47E+00 | 2.48E+01 | 5.93E-02 | | | |
| 0 - 8 hrs | | | | 2.32E-01 | 9.20E+00 | 2.52E-02 |
| 8 - 24 hrs | | | | 9.41E-03 | 3.30E-01 | 2.85E-02 |
| 24 -96 hrs | | | | 1.06E-02 | 3.10E-01 | 3.43E-02 |
| 96 - 720 hrs | | | | 1.32E-02 | 2.90E-01 | 4.55E-02 |
| **Total** | **1.47E+00** | **2.48E+01** | | **2.65E-01** | **1.01E+01** | |
| **ABWR** | | | | | | |
| **Main Steam Line Break** | | | | | | |
| *Max Equilibrium Iodine Activity* | | | | | | |
| 0 -2 hrs | 3.43E-03 | 1.32E-01 | 2.60E-02 | | | |

**TABLE 7.1-2**
Comparison of Reactor Types for Limiting Off-Site Dose Consequences

| Design Basis Accident | ESP EAB Dose TEDE (rem) | Vendor EAB Dose TEDE (rem) | ESP/Vendor EAB X/Q Ratio | ESP LPZ Dose TEDE (rem) | Vendor LPZ Dose TEDE (rem) | ESP/Vendor LPZ X/Q Ratio |
|---|---|---|---|---|---|---|
| 0 - 8 hrs | | | | 3.28E-04 | 1.50E-02 | 2.18E-02 |
| **Total** | **3.43E-03** | **1.32E-01** | | **3.28E-04** | **1.50E-02** | |
| *Preexisting Iodine Spike* | | | | | | |
| 0 -2 hrs | 6.85E-02 | 2.63E+00 | 2.60E-02 | | | |
| 0 - 8 hrs | | | | 6.54E-03 | 3.00E-01 | 2.18E-02 |
| **Total** | **6.85E-02** | **2.63E+00** | | **6.54E-03** | **3.00E-01** | |
| **Control Rod Drop Accident** | *Not Applicable to the ABWR design* | | | | | |
| **Small Line Break** | | | | | | |
| 0 -2 hrs | 2.97E-03 | 1.14E-01 | 2.60E-02 | | | |
| 0 - 8 hrs | | | | 5.75E-04 | 2.64E-02 | 2.18E-02 |
| **Total** | **2.97E-03** | **1.14E-01** | | **5.75E-04** | **2.64E-02** | |
| **Fuel Handling Accident** | | | | | | |
| 0 -2 hrs | 8.04E-02 | 3.09E+00 | 2.60E-02 | | | |
| 0 - 8 hrs | | | | 9.78E-03 | 4.49E-01 | 2.18E-02 |
| **Total** | **8.04E-02** | **3.09E+00** | | **9.78E-03** | **4.49E-01** | |
| **Loss of Coolant Accident** | | | | | | |
| 0 - 2 hrs | 2.35E-01 | 9.04E+00 | 2.60E-02 | | | |
| 0 - 8 hrs | | | | 3.78E-02 | 1.73E+00 | 2.18E-02 |
| 8 - 24 hrs | | | | 3.20E-02 | 1.08E+00 | 2.97E-02 |
| 24 -96 hrs | | | | 1.65E-01 | 2.99E+00 | 5.51E-02 |
| 96 - 720 hrs | | | | 5.29E-01 | 3.92E+00 | 1.35E-01 |
| **Total** | **2.35E-01** | **9.04E+00** | | **7.63E-01** | **9.73E+00** | |

**TABLE 7.1-2A**

Ratio of EGC ESP 50% Accident Site Chi/Q Values to AP1000 Final Design Approval (FDA) Chi/Q Values

| Post Accident Time Period (hr) | EGC ESP Site Chi/Q Values(sec/m$^3$) | AP1000 Chi/Q Values (sec/m$^3$) | Chi/Q Ratio |
|---|---|---|---|
| | | | EGC Site / AP1000 FDA |
| EAB[1] 0 - 2 | 3.56E-05 | 6.00E-04 | 5.93E-02 |
| LPZ | | | |
| 0 - 8 | 3.40E-06 | 1.35E-04 | 2.52E-02 |
| 8 -24 | 2.85E-06 | 1.00E-04 | 2.85E-02 |
| 24 -96 | 1.85E-06 | 5.40E-05 | 3.43E-02 |
| 96 - 720 | 1.00E-06 | 2.20E-05 | 4.55E-02 |

Note 1: 2 hour period with greatest EAB dose consequences.

**TABLE 7.1-3**
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 7.1-4**

AP1000 Main Steam Line Break Curies Released to Environment by Interval - Preexisting 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 | * | * | * | * |
| 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 |

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

**TABLE 7.1-5**
AP1000 Main Steam Line Break - Accident-Initiated Iodine Spike

| Time | Exclusion Area Boundary Dose TEDE (rem) | Low Population Zone Dose TEDE (rem) |
|---|---|---|
| 0 to 2 hrs | 4.75E-02 | -- |
| 0 to 8 hrs | -- | 1.61E-02 |
| 8 to 24 hrs | -- | 1.20E-02 |
| 24 to 96 hrs | -- | 2.16E-02 |
| 96 to 720 hrs | -- | 0 |
| Total | 4.75E-02 | 4.97E-02 |

**TABLE 7.1-6**
AP1000 Main Steam Line Break - Preexisting Iodine Spike

| Time | Exclusion Area Boundary Dose TEDE (rem) | Low Population Zone Dose TEDE (rem) |
|---|---|---|
| 0 to 2 hrs | 4.15E-02 | -- |
| 0 to 8 hrs | -- | 6.04E-03 |
| 8 to 24 hrs | -- | 2.28E-03 |
| 24 to 96 hrs | -- | 4.45E-03 |
| 96 to 720 hrs | -- | 0 |
| Total | 4.15E-02 | 1.28E-02 |

**TABLE 7.1-7**
ABWR Main Steam Line Break Outside Containment

| Isotope | Maximum Equilibrium Value for Full Power Operation Curies Released 0 to 2 hr | Preexisting Iodine Spike Curies Released 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 7.1-8**
ABWR Main Steam Line Break Outside Containment -Maximum Equilibrium Value for Full Power Operation

| Dose Type | EAB (rem) | LPZ (rem) |
|---|---|---|
| Thyroid | 6.64E-02 | 6.34E-03 |
| Whole Body | 1.46E-03 | 1.39E-04 |
| TEDE | 3.43E-03 | 3.28E-04 |

**TABLE 7.1-9**
ABWR Main Steam Line Break Outside Containment - Preexisting Iodine Spike

| Dose Type | EAB (rem) | LPZ (rem) |
|---|---|---|
| Thyroid | 1.33E+00 | 1.27E-01 |
| Whole Body | 2.89E-02 | 2.76E-03 |
| TEDE | 6.85E-02 | 6.54E-03 |

**TABLE 7.1-10**
AP1000 Locked Rotor Accident Curies Released to Environment

| 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 7.1-11**
AP1000 Locked Rotor Accident, 0 to 1.5 hr Duration - Preexisting Iodine Spike

| Time | Exclusion Area Boundary Dose TEDE (rem) | Low Population Zone Dose TEDE (rem) |
|---|---|---|
| 0 to 2 hr | 1.48E-01 | -- |
| 0 to 8 hr | -- | 1.51E-02 |
| 8 to 24 hr | -- | 0 |
| 24 to 96 hr | -- | 0 |
| 96 to 720 hr | -- | 0 |
| Total | 1.48E-01 | 1.51E-02 |

**TABLE 7.1-12**
AP1000 Control Rod Ejection Accident Curies Released to Environment by Interval - Preexisting 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 7.1-13**
AP1000 Control Rod Ejection Accident - Preexisting Iodine Spike

| Time | Exclusion Area Boundary Dose TEDE (rem) | Low Population Zone Dose TEDE (rem) |
|---|---|---|
| 0 to 2 hr | 1.78E-01 | -- |
| 0 to 8 hr | -- | 3.53E-02 |
| 8 to 24 hr | -- | 7.41E-03 |
| 24 to 96 hr | -- | 1.58E-03 |
| 96 to 720 hr | -- | 5.45E-04 |
| Total | 1.78E-01 | 4.48E-02 |

**TABLE 7.1-14**
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 | * | * | * |
| Cs-134 | 1.65E+00 | 6.35E-02 | 2.27E-01 |
| Cs-136 | 2.45E+00 | 9.30E-02 | 3.30E-01 |
| Cs-137 | 1.19E+00 | 4.58E-02 | 1.64E-01 |
| Cs-138 | 5.71E-01 | 3.07E-06 | 6.00E-07 |

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

**TABLE 7.1-15**
AP1000 Steam Generator Tube Rupture Accident Curies Released to Environment by Interval - Preexisting 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 | * | * | * |
| Cs-134 | 1.65E+00 | 6.35E-02 | 2.27E-01 |
| Cs-136 | 2.45E+00 | 9.30E-02 | 3.30E-01 |
| Cs-137 | 1.19E+00 | 4.58E-02 | 1.64E-01 |
| Cs-138 | 5.71E-01 | 3.07E-06 | 6.00E-07 |

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

**TABLE 7.1-16**
AP1000 Steam Generator Tube Rupture - Accident-Initiated Iodine Spike

| Time | Exclusion Area Boundary Dose TEDE (rem) | Low Population Zone Dose TEDE (rem) |
|---|---|---|
| 0 to 2 hr | 8.90E-02 | -- |
| 0 to 8 hr | -- | 4.53E-03 |
| 8 to 24 hr | -- | 2.05E-03 |
| 24 to 96 hr | -- | 0 |
| 96 to 720 hr | -- | 0 |
| Total | 8.90E-02 | 6.59E-03 |

**TABLE 7.1-17**
AP1000 Steam Generator Tube Rupture - Preexisting Iodine Spike

| Time | Exclusion Area Boundary Dose TEDE (rem) | Low Population Zone Dose TEDE (rem) |
|---|---|---|
| 0 to 2 hr | 1.78E-01 | -- |
| 0 to 8 hr | -- | 8.06E-03 |
| 8 to 24 hr | -- | 7.41E-04 |
| 24 to 96 hr | -- | 0 |
| 96 to 720 hr | -- | 0 |
| Total | 1.78E-01 | 8.80E-03 |

**TABLE 7.1-18**
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 7.1-19**
AP1000 Small Line Break Accident, 0- to 0.5-hr Duration - Accident-Initiated Iodine Spike

| Time | Exclusion Area Boundary Dose TEDE (rem) | Low Population Zone Dose TEDE (rem) |
|---|---|---|
| 0 to 2 hr | 7.71E-02 | -- |
| 0 to 8 hr | -- | 7.56E-03 |
| 8 to 24 hr | -- | 0 |
| 24 to 96 hr | -- | 0 |
| 96 to 720 hr | -- | 0 |
| Total | 7.71E-02 | 7.56E-03 |

**TABLE 7.1-20**
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 7.1-21**
ABWR Small Line Break Outside Primary Containment

| Dose Type | EAB (rem) | LPZ (rem) |
|---|---|---|
| Thyroid | 6.10E-02 | 1.20E-02 |
| Whole Body | 1.14E-03 | 2.16E-04 |
| TEDE | 2.97E-03 | 5.75E-04 |

**TABLE 7.1-22**
AP1000 Design Basis Loss of Coolant Accident Curies Released to Environment by Interval

| Isotope | 0 to 1 hr | 1 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 |
| 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 |

**TABLE 7.1-22**
AP1000 Design Basis Loss of Coolant Accident Curies Released to Environment by Interval

| Isotope | 0 to 1 hr | 1 to 3 hr | 0 to 8 hr | 8 to 24 hr | 24 to 96 hr | 96 to 720 hr |
|---|---|---|---|---|---|---|
| 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 |
| 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 |

**TABLE 7.1-22**
AP1000 Design Basis Loss of Coolant Accident Curies Released to Environment by Interval

| Isotope | 0 to 1 hr | 1 to 3 hr | 0 to 8 hr | 8 to 24 hr | 24 to 96 hr | 96 to 720 hr |
|---|---|---|---|---|---|---|
| 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 7.1-23**
AP1000 Design Basis Loss of Coolant Accident

| Time | Exclusion Area Boundary Dose TEDE (rem) | Low Population Zone Dose TEDE (rem) |
|---|---|---|
| 1 to 3 hr | 1.47E+00 | -- |
| 0 to 8 hr | -- | 2.32E-01 |
| 8 to 24 hr | -- | 9.41E-03 |
| 24 to 96 hr | -- | 1.06E-02 |
| 96 to 720 hr | -- | 1.32E-02 |
| Total | 1.47E+00 | 2.65E-01 |

Notes: 2-hr period with greatest EAB dose shown. LOCA based on Regulatory Guide 1.183 (USNRC, 2000).

**TABLE 7.1-24**
ABWR LOCA Curies Released to Environment by Interval

| Isotope | 0 to 2 hr (Ci) | 0 to 8 hr (Ci) | 8 to 24 hr (Ci) | 24 to 96 hr (Ci) | 96 to 720 hr (Ci) |
|---|---|---|---|---|---|
| I-131 | 2.60E+02 | 3.74E+02 | 9.23E+02 | 8.70E+03 | 6.22E+04 |
| I-132 | 3.52E+02 | 3.85E+02 | 3.24E+01 | 0 | 0 |
| I-133 | 5.41E+02 | 7.43E+02 | 1.18E+03 | 3.32E+03 | 6.76E+02 |
| I-134 | 5.14E+02 | 5.15E+02 | 0 | 0 | 0 |
| I-135 | 5.14E+02 | 6.47E+02 | 3.32E+02 | 1.68E+02 | 0 |
| Kr-83m | 3.26E+02 | 9.00E+02 | 4.32E+01 | 0 | 0 |
| Kr-85m | 8.44E+02 | 3.74E+03 | 4.36E+03 | 7.03E+02 | 0 |
| Kr-85 | 4.09E+01 | 3.49E+02 | 2.19E+03 | 2.18E+04 | 2.86E+05 |
| Kr-87 | 1.20E+03 | 2.17E+03 | 8.92E+01 | 2.70E+00 | 0 |
| Kr-88 | 2.12E+03 | 7.14E+03 | 3.43E+03 | 2.97E+02 | 0 |
| Kr-89 | 1.81E+02 | 1.81E+02 | 0 | 0 | 0 |
| Xe-131m | 2.13E+01 | 1.72E+02 | 1.12E+03 | 9.52E+03 | 6.22E+04 |
| Xe-133m | 3.00E+02 | 2.48E+03 | 1.38E+04 | 7.59E+04 | 7.27E+04 |
| Xe-133 | 7.63E+03 | 6.11E+04 | 3.77E+05 | 2.78E+06 | 8.41E+06 |
| Xe-135m | 4.87E+02 | 4.87E+02 | 0 | 0 | 0 |
| Xe-135 | 9.26E+02 | 5.51E+03 | 1.52E+04 | 1.17E+04 | 0 |
| Xe-137 | 5.14E+02 | 5.14E+02 | 0 | 0 | 0 |
| Xe-138 | 2.00E+03 | 2.00E+03 | 0 | 0 | 0 |

**TABLE 7.1-25**
ABWR Design Basis Loss of Coolant Accident

| Dose Type | EAB (rem) | LPZ (rem) |
|---|---|---|
| Thyroid | 4.96E+00 | 2.15E+01 |
| Whole Body | 1.02E-01 | 1.79E-01 |
| TEDE | 2.35E-01 | 7.63E-01 |

**TABLE 7.1-26**
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 |
| 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 |

**TABLE 7.1-26**
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 |
|---|---|---|---|---|---|---|
| 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 |
| 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 |

**TABLE 7.1-26**
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 |
|---|---|---|---|---|---|---|
| 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 7.1-27**
ESBWR Design Basis Loss of Coolant Accident

| Time | EAB Dose TEDE (rem) | LPZ Dose TEDE (rem) |
|---|---|---|
| 0 to 2 hr | 3.10E-01 | -- |
| 0 to 8 hr | -- | 8.94E-02 |
| 8 to 24 hr | -- | 7.06E-02 |
| 24 to 96 hr | -- | 1.68E-01 |
| 96 to 720 hr | -- | 1.41E-01 |
| Total | 3.10E-01 | 4.69E-01 |

Note: LOCA based on Regulatory Guide 1.183

**TABLE 7.1-28**
ACR-700 Design Basis Large 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 | 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 7.1-29**
ACR-700 Large Loss of Coolant Accident

| Time | EAB Dose TEDE (rem) | LPZ Dose TEDE (rem) |
|---|---|---|
| 0 to 2 hr | 3.77E-01 | - |
| 0 to 8 hr | - | 7.84E-02 |
| 8 to 24 hr | - | 2.56E-02 |
| 24 to 96 hr | - | 2.73E-01 |
| 96 to 720 hr | - | 3.95E-02 |
| Total | 3.77E-01 | 4.16E-01 |

**TABLE 7.1-30**
AP1000 Fuel Handling Accident - Curies Released to Environment

| Isotope | 0 to 2 hrs (Ci) |
|---|---|
| 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 |

Note: Activity is based on a 100-hr shutdown before fuel movement begins. Source term and pool DF are based on Regulatory Guide 1.183 (USNRC, 2000).

**TABLE 7.1-31**
AP1000 Fuel Handling Accident

| Time | Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem) | Low Population Zone Dose Total Effective Dose Equivalent (rem) |
|---|---|---|
| 0 to 2 hrs | 1.42E-01 | -- |
| 0 to 8 hrs | -- | 1.51E-02 |
| 8 to 24 hrs | -- | 0 |
| 24 to 96 hr | -- | 0 |
| 96 to 720 hrs | -- | 0 |
| Total | 1.42E-01 | 1.51E-02 |

**TABLE 7.1-32**
ABWR Fuel Handling Accident - Curies Released to Environment by Interval

| Isotope | 0 to 2 hrs (Ci) | 2 to 8 hrs (Ci) |
|---|---|---|
| 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: Activity is based on a 24-hr shutdown before fuel movement begins. Source term and pool DF are based on Regulatory Guide 1.25 (USAEC, 1972).

**TABLE 7.1-33**
ABWR Fuel Handling Accident

| Dose Type | EAB (rem) | LPZ (rem) |
|---|---|---|
| Thyroid | 1.97E+00 | 1.91E-01 |
| Whole Body | 2.82E-02 | 5.56E-03 |
| TEDE | 8.04E-02 | 9.78E-03 |