7.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

7.1 Design Basis Accidents

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

7.1.1 Selection of Design Basis Accidents

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

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

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

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

7.1.2 Evaluation of Radiological Consequences

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

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

Exclusion Area Boundary (EAB)

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

Low Population Zone (LPZ)

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

7.1.3 Source Terms

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

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

(ABWR) source term and releases are based on TID-14844. The AP1000[2] PWR source term and accident analyses approaches are based on the AST methodology in accordance with Regulatory Guide 1.183. The International Reactor Innovative And Secure (IRIS) advanced reactor source term information is preliminary, and based on vendor information the AP600/AP1000 LOCA source terms and releases are expected to bound the worst-case accident release for this advanced reactor concept.

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

7.1.4 Postulated Accidents

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

7.1.4.1 Main Steam Line Break Outside Containment (AP1000)

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

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

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

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

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

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

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

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

7.1.4.2 Main Steam Line Break Outside Containment (ABWR)

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

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

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

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

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

7.1.4.3 Reactor Coolant Pump Locked Rotor (AP1000)

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

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

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

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

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

7.1.4.4 Control Rod Ejection (AP1000)

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

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

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

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

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

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

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

7.1.4.5 Rod Drop Accident (ABWR)

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

7.1.4.6 Steam Generator Tube Rupture (AP1000)

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

7.1.4.9 Large Break Loss of Coolant Accident (AP1000)

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

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

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

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

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

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

7.1.4.10 Large Break Loss of Coolant Accident (ABWR)

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

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

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

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

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

7.1.4.11 Large Loss of Coolant Accident (ACR-700)

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

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

7.1.4.12 Fuel Handling Accidents (AP1000)

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

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

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

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

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

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

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

7.1.4.13 Fuel Handling Accidents (ABWR)

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

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

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

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

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

7.1.5 References

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

7.2 Severe Accidents

7.2.1 Introduction

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, Volume 1, the *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* [GEIS], the NRC 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 or plants that may be built on the GGNS ESP site.

7.2.2 Applicability of Existing Generic Severe Accident Studies

Section 5.3.3 of NUREG-1437, Volume 1, presents a thorough NRC staff assessment of the impacts of severe accidents during the license renewal period. 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. [Reference 5, §5.3.3 /pg 5-12] 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 ground water to surface water, are discussed in this section. The economic impacts from severe accidents are also comparatively evaluated in this section. [Reference 5, §5.3.3 /pg 5-12]

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. [Reference 5, §5.3.3 /pg 5-13] The GEIS results are expressed as a range of values in terms of risk of severe accident impact per reactor-year of operation. [Reference 5, Tables 5.5, 5.6, 5.9, 5.10, 5.11, 5.12, 5.16 & 5.31, and §5.3.3.2.4 /pg 5-44] [Reference 5, §5.3.3.1 /pg 5-12] The NRC 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 GEIS 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). [Reference 5, §5.3.3.2.1 /pg 5-20 & 24] 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. [Reference 5, §5.3.3.2.1 /pg 5-19] 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-mile population, 50-mile 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. [Reference 5, §5.3.3.2.1 /pg 5-25]

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. [Reference 5, §5.3.3.2.1 /pg 5-26]

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) and magnitude 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, or 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 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. [Reference 5, §5.5 /pg 5-114] 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 GGNS ESP site are not substantially different from those factors identified for previously analyzed sites. Thus, it follows that the environmental impacts for the

GGNS ESP Site will not be substantially different from the acceptable environmental impacts identified for the previously analyzed sites.

7.2.3 Evaluation of Potential Severe Accident Releases

The significance of the impacts associated with each severe accident issue have been identified as either Small, Moderate, or Large, consistent with the criteria that NRC 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, ongoing and potential additional mitigation is considered 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).

7.2.3.1 Evaluation Of Potential Releases Via Atmospheric Pathway

The site-specific significant factors of demography and meteorology are considered in this evaluation of the atmospheric exposure pathway for the GGNS ESP site. For this evaluation, NUREG-1437, Volume 1, calculates an exposure index (EI) for use in comparing the relative risk for the current fleet of nuclear power plants.

NUREG-1437, Volume 1, 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." [Reference 5, §5.3.3.2.1 / pg 5-26]

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 GGNS ESP site than the meteorology considered in NUREG-1437, Volume 1, for the GGNS site, and only population can significantly affect the resulting risk in any given year of reactor operation. [Reference 5, §5.3.3.2.1 / pg 5-19]

However, the 50-mi population projections for the GGNS ESP site for the year 2050 (i.e., ~375,988) are less than for the GGNS site as projected for the year 2050 in Table 5.3 of

NUREG-1437, Volume 1 (i.e., ~505,000). Thus, the GGNS ESP site EI will not be significantly different from those established in NUREG-1437 for the GGNS site.

Two EIs were evaluated in NUREG-1437, Volume 1. A 10-mile EI was found to best correlate with early fatalities, and a 150-mile EI was found to best correlate with latent fatalities and total dose. [Reference 5, §5.3.3.2.1 / pg 5-27] 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. [Reference 5, §5.3.3.2.4 / pg 5-44]

The 10-mile EI for the GGNS ESP site was 562, as shown in NUREG-1437, Volume 1, Table 5.7, for the year 2050. The 10-mile 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 GGNS ESP site is expected to be well within the range of risk calculated for the existing fleet of nuclear power plants.

The 150-mile EI for the GGNS ESP site was 388,245, as shown in NUREG-1437, Volume 1, Table 5.8, for the year 2050. The 150-mile 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 GGNS ESP Site is expected to be well within the range of risk calculated for the existing fleet of nuclear power plants.

Thus, the GGNS ESP site risks for the atmospheric exposure pathway will be within the range of those considered as “Small” in NUREG-1437 Volume 1. Section 5.5.2.1 of NUREG-1437 Volume 1, 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 GGNS 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. [Reference 5, §5.3.3.3.1 / pg 5-49 & 50]

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. [Reference 5, §5.3.3.3.3 / pg 5-63 & 5-64]

For the aquatic food pathway, population dose and population exposure per reactor-year are directly related to aquatic food harvest. For river sites, uninterdicted population exposure is orders of magnitude lower than that for the atmospheric pathway. For Great Lakes sites, the uninterdicted 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. [Reference 5, §5.3.3.3.3 / pg 5-64]

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. [Reference 5, §5.3.3.3.3 / pg 5-65]

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. [Reference 5, §5.3.3.3.1 / pg 5-51]

The above identified environmental parameters have been identified in the GEIS for the GGNS site. These same parameters are applicable for the GGNS 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 GGNS 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. [Reference 5, §5.3.3.3.3 / pg 5-64 & 5-65; and §5.5.2.2 & §5.5.2.3 / pg 5-115] The GGNS ESP site risks for the water body exposure pathway will also be within the range of those considered as “Small” in NUREG-1437 Volume 1.

7.2.2.3 Evaluation Of Potential Releases To Ground Water

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

As identified in NUREG-1437, ground water 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. [Reference 5, §5.3.3.4.1 / pg 5-65]

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 ground water 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 ground water 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 ground water flow is away from a nearby surface water body. In this case, the only population exposure results from drinking contaminated ground water. [Reference 5, §5.3.3.4.1 / pg 5-65 & 66]

NUREG-1437 concludes that the risk from the ground water 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. [Reference 5, §5.3.3.4.9 / pg 5-95]

In the GEIS analysis, site-specific information on ground water travel time; retention-adsorption coefficients; distance to surface water; and soil, sediment, and rock characteristics is compared with previous ground water contamination analyses. Previous analyses are contained in the LPGS and site-specific FESs. These environmental parameters have been identified in the GEIS for the GGNS site. These same parameters are applicable for the GGNS 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 ground water pathway for the GGNS ESP Site is comparable to those considered in the GEIS evaluation. Therefore, the risk from the ground water exposure pathway generally compares favorably with the risk to the population from atmospheric releases. [Reference 5, §5.3.3.3.3 / pg 5-64 & 5-65; and §5.5.2.2 & §5.5.2.3 / pg 5-115] The GGNS ESP site risks for the ground water exposure pathway will also be within the range of those considered as “Small” in NUREG-1437.

7.2.4 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 GGNS ESP Site. Similar to Section 7.2.2.1, the EI is used as a predictor of cost because the cost, as identified in the GEIS, 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. [Reference 5, §5.3.3.5 / pg 5-96]

As noted 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: [Reference 5, §5.3.3.5 / pg 5-96]

- 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 quite high ($6 billion to $8 billion in 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 GGNS 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 acres/year at most). This is also consistent with WASH-1400 (NUREG-75/014) and a 1982 study on siting criteria (NUREG/CR-2239) 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. [Reference 5, §5.3.3.5 / pg 5-99]

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 GGNS ESP site should be comparable as well i.e., within the range of those considered as “Small” in NUREG-1437. [Reference 5, §5.5.2.4 / pg 5-115]

7.2.5 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 or reactors on the GGNS 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 Advanced Boiling Water Reactor (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 NRC and are expected to continue to be met for future design certifications and COL approvals.

7.2.6 Conclusion

The following are directly applicable conclusions from the GEIS, and conclusions drawn based on the foregoing analysis:

- 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 context. Evaluation of site specific factors for purposes of this application have shown that the GGNS ESP Site is within the range of sites considered in the GEIS. Thus it is concluded that the GEIS conclusion is applicable to the GGNS ESP Site.
- Use of pertinent site-specific information to confirm the applicability of existing generic analyses is consistent with NRC 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 an operating 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). [Reference 7, §7.2, IV / pg 7.2-6]

Specifically, based on the NRC and industry implementation of the 1985 policy statement, the generic NUREG-1437 risk evaluations, and the GGNS 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 GGNS ESP Site will also be "Small."

7.2.7 References

1. USNRC, 1978, U.S. Nuclear Regulatory Commission, NUREG-0440, Liquid Pathway Generic Study, February 1978.
2. USNRC, 1985, U.S. Nuclear Regulatory Commission, Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, August 8, 1985.

3. USNRC, 1991, SECY-91-0041, Early Site Permit Review Readiness, February 13, 1991.
4. USNRC, 1994, U.S. Nuclear Regulatory Commission, NUREG-1503, Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, July 1, 1994.
5. USNRC, 1996, U.S. Nuclear Regulatory Commission, NUREG-1437, Volume 1, Generic Environmental Impact Statement for License Renewal of Nuclear Plants, May 1996.
6. USNRC, 1996a, U.S. Nuclear Regulatory Commission, 61 FR 28467 - 28497, Environmental Review for Renewal of Nuclear Power Plant Operating Licenses, Final Rule, June 5, 1996.
7. USNRC, 1999, U.S. Nuclear Regulatory Commission, NUREG-1555, Environmental Standard Review Plan, October 1999.

7.3 Severe Accident Mitigation Alternatives

This section is not applicable for an Early Site Permit application.

7.4 Transportation Accidents

Refer to Section 3.8 for a discussion of the fuel cycle transportation accidents and the evaluation of applicability of 10 CFR 51.52 Table S-4 to this report.

TABLE 7.1-1

COMPARISON OF REACTOR TYPES FOR LIMITING OFF-SITE DOSE CONSEQUENCES

PART A, EXCEPT ABWR

Accident	Reactor Type	EAB Dose TEDE (rem)	LPZ Dose TEDE (rem)	Guideline[1] TEDE (rem)
Main Steam Line Break				
Accident-initiated Iodine Spike	AP1000	0.79	1.01	2.5
Pre-existing Iodine Spike	AP1000	0.69	0.28	25
Reactor Coolant Pump Locked Rotor				
	AP1000	2.5	0.4	2.5
Control Rod Ejection Accident				
	AP1000	2.98	1.11	6.3
Steam Generator Tube Rupture				
Accident-initiated Iodine Spike	AP1000	1.49	0.16	2.5
Pre-existing Iodine Spike	AP1000	2.98	0.23	25
Small Line Break				
	AP1000	1.3	0.2	2.5
Loss of Coolant Accident				
	AP1000	24.6	6.50	25
	ACR-700	6.3	5.3	25
Fuel Handling Accident				
	AP1000	2.4	0.4	6.3

NOTES:

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

TABLE 7.1-1 (Continued)

PART B, ABWR

Accident	Affected Organ	EAB Dose (rem)	LPZ Dose (rem)	Guideline[1] (rem)
Main Steam Line Break				
Max Equilibrium Iodine Activity	Thyroid	1.11	1.65E-01	30
	Whole Body	1.7E-02	2.53E-03	2.5
Pre-existing Iodine Spike	Thyroid	22.2	3.29	300
	Whole Body	3.4E-01	5.05E-02	25
Control Rod Drop Accident	Thyroid	Negligible	Negligible	75
	Whole Body	Negligible	Negligible	6
Small Line Break	Thyroid	2.04	0.30	30
	Whole Body	0.027	0.004	2.5
Loss of Coolant Accident	Thyroid	82.5	233	300
	Whole Body	1.78	3.11	25
Fuel Handling Accident	Thyroid	9.78	1.45	75
	Whole Body	0.41	0.06	6

NOTES:

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

TABLE 7.1-2

AP1000 MAIN STEAM LINE BREAK - ACCIDENT-INITIATED IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	0.79	----*
0 to 8 hour	----*	0.42
8 to 24 hour	----*	0.26
24 to 96 hour	----*	0.33
96 to 720 hours	----*	----*
TOTAL	0.79	1.01

NOTES:

* Dose not applicable

TABLE 7.1-3

AP1000 MAIN STEAM LINE BREAK - PRE-EXISTING IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	0.69	----*
0 to 8 hour	----*	0.16
8 to 24 hour	----*	0.05
24 to 96 hour	----*	0.07
96 to 720 hours	----*	----*
TOTAL	0.69	0.28

NOTES:

* Dose not applicable

TABLE 7.1-4

ABWR MAIN STEAM LINE BREAK OUTSIDE CONTAINMENT

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

TABLE 7.1-5

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

Time	Exclusion Area Boundary Dose (rem)		Low Population Zone Dose (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
0 to 2 hour	1.11	1.70E-02	----*	----*
0 to 8 hour	----*	----*	1.65E-01	2.53E-03
8 to 24 hour	----*	----*	----*	----*
24 to 96 hour	----*	----*	----*	----*
96 to 720 hours	----*	----*	----*	----*
TOTAL	1.11	1.70E-02	1.65E-01	2.53E-03

NOTES:

* Dose not applicable

TABLE 7.1-6

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

Time	Exclusion Area Boundary Dose (rem)		Low Population Zone Dose (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
0 to 2 hour	2.22E+01	3.4E-01	----*	----*
0 to 8 hour	----*	----*	3.29E+00	5.05E-02
8 to 24 hour	----*	----*	----*	----*
24 to 96 hour	----*	----*	----*	----*
96 to 720 hours	----*	----*	----*	----*
TOTAL	2.22E+01	3.4E-01	3.29E+00	5.05E-02

NOTES:

* Dose not applicable

TABLE 7.1-7

AP1000 LOCKED ROTOR ACCIDENT – PRE-EXISTING IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	2.5	----*
0 to 8 hour	----*	0.4
8 to 24 hour	----*	----*
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
TOTAL	2.5	0.4

NOTES:

* Dose not applicable

TABLE 7.1-8

AP1000 CONTROL ROD EJECTION ACCIDENT - PRE-EXISTING IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	2.98	----*
0 to 8 hour	----*	0.916
8 to 24 hour	----*	0.160
24 to 96 hour	----*	0.024
96 to 720 hours	----*	0.005
TOTAL	2.98	1.105

NOTES:

* Dose not applicable

TABLE 7.1-9

AP1000 STEAM GENERATOR TUBE RUPTURE - ACCIDENT-INITIATED IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	1.49	----*
0 to 8 hour	----*	0.12
8 to 24 hour	----*	0.04
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
TOTAL	1.49	0.16

NOTES:

* Dose not applicable

TABLE 7.1-10

AP1000 STEAM GENERATOR TUBE RUPTURE - PRE-EXISTING IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	2.98	----*
0 to 8 hour	----*	0.21
8 to 24 hour	----*	0.02
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
TOTAL	2.98	0.23

NOTES:

* Dose not applicable

TABLE 7.1-11

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

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	1.3	----*
0 to 8 hour	----*	0.2
8 to 24 hour	----*	----*
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
TOTAL	1.3	0.2

NOTES:

* Dose not applicable

TABLE 7.1-12

ABWR SMALL LINE BREAK OUTSIDE CONTAINMENT - ACTIVITY RELEASED TO ENVIRONMENT

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

TABLE 7.1-13

ABWR SMALL LINE BREAK OUTSIDE CONTAINMENT

Time	Exclusion Area Boundary Dose (rem)		Low Population Zone Dose (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
0 to 2 hour	2.04	2.68E-02	----*	----*
0 to 8 hour	----*	----*	0.3	0.004
8 to 24 hour	----*	----*	----*	----*
24 to 96 hour	----*	----*	----*	----*
96 to 720 hours	----*	----*	----*	----*
TOTAL	2.04	2.68E-02	0.3	0.004

NOTES:

* Dose not applicable

TABLE 7.1-14

AP1000 DESIGN BASIS LOSS OF COOLANT ACCIDENT

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	24.6	----*
0 to 8 hour	----*	6.02
8 to 24 hour	----*	0.20
24 to 96 hour	----*	0.16
96 to 720 hours	----*	0.12
TOTAL	24.6	6.50

NOTES:

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

TABLE 7.1-15

ABWR LOCA CURIES RELEASED TO ENVIRONMENT BY TIME INTERVAL

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

TABLE 7.1-16

ABWR DESIGN BASIS LOSS OF COOLANT ACCIDENT

	Exclusion Area Boundary Dose		Low Population Zone Dose	
Time	Thyroid (rem)	Whole Body (rem)	Thyroid (rem)	Whole Body (rem)
0 to 2 hour	82.5	1.78	----*	----*
0 to 8 hour	----*	----*	1.75E+01	5.66E-01
8 to 24 hour	----*	----*	1.28E+01	5.13E-01
24 to 96 hour	----*	----*	6.63E+01	9.23E-01
96 to 720 hours	----*	----*	1.36E+02	1.11E+00
TOTAL	82.5	1.78	233	3.11

NOTES:

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

TABLE 7.1-17

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

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

TABLE 7.1-18

ACR-700 LARGE LOSS OF COOLANT ACCIDENT

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	6.3	0.9
2 to 8 hour	----*	1.7
8 to 24 hour	----*	1.6
24 to 96 hour	----*	0.6
96 to 720 hours	----*	0.5
TOTAL	6.3	5.3

NOTES:

* Dose not applicable

TABLE 7.1-19

AP1000 FUEL HANDLING ACCIDENT - CURIES RELEASED TO ENVIRONMENT

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

TABLE 7.1-20

AP1000 FUEL HANDLING ACCIDENT

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	2.4	----*
0 to 8 hour	----*	0.4
8 to 24 hour	----*	----*
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
TOTAL	2.4	0.4

NOTES:

* Dose not applicable

TABLE 7.1-21

ABWR FUEL HANDLING ACCIDENT - CURIES RELEASED TO ENVIRONMENT

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

TABLE 7.1-22

ABWR FUEL HANDLING ACCIDENT

Time	Exclusion Area Boundary Dose (rem)		Low Population Zone Dose (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
0 to 2 hour	9.78	0.41	----*	----*
0 to 8 hour	----*	----*	1.45	0.06
8 to 24 hour	----*	----*	----*	----*
24 to 96 hour	----*	----*	----*	----*
96 to 720 hours	----*	----*	----*	----*
TOTAL	9.78	0.41	1.45	0.06

NOTES:

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

TABLE 7.1-23

AP1000 LOCKED ROTOR ACCIDENT - CURIES RELEASED TO ENVIRONMENT

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

TABLE 7.1-24

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

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

TABLE 7.1-25

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

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

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

TABLE 7.1-26

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

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

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

TABLE 7.1-27

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

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

TABLE 7.1-28

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

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

TABLE 7.1-28 (Continued)

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

TABLE 7.1-28 (Continued)

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