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) 50th 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

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