Entergy

System Energy Resources, Inc.
1340 Echelon Parkway
Jackson, MS 39286-1995

CNRO-2004-00053

August 16, 2004

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Attention: Document Control Desk

DOCKET: 52-009

SUBJECT: Early Site Permit - Response to Request for Additional Information Letter No. 2 (Partial Response No. 1)

REFERENCE:

1. System Energy Resources, Inc. (SERI) letter to USNRC – Early Site Permit Application (CNRO-2003-00054), dated October 16, 2003.

2. USNRC letter to SERI – Request for Additional Information Letter No. 2 – System Energy Resources, Inc., Early Site Permit Application for the Grand Gulf ESP Site (TAC No. MC 1378) (CNRI-2004-00012), dated July 15, 2004.

3. USNRC letter to SERI – System Energy Resources, Inc., - NRC Inspection of Applicant and Contractor Quality Assurance Activities Involved with Preparation of the Application for an Early Site Permit, NRC Inspection Report 052000009/2004001, dated March 19, 2004.

4. System Energy Resources, Inc. (SERI) letter to USNRC – Early Site Permit Application Request for Information – Emergency Preparedness (CNRO-2003-00036), dated June 3, 2004.

CONTACT:

Name	George A. Zinke
Mailing Address	1340 Echelon Parkway Jackson, MS 39213
E-Mail Address	gzinke@entergy.com
Phone Number	601-368-5381

DOCUMENT COMPONENTS:

One (1) CD-ROM is included in this submission. The CD-ROM contains the following seven (7) files:

001_Draft-Rev1_Tables 1.3-1_thru_1.3-3_08-16-04.pdf
002_Fig 2.1-5 Draft Rev1.PDF
003_Figure2.1-5.tif
004_Figure2.1-5.DWG
005_ESP-SSAR_DraftRev-1_08-16-04.pdf
006_Draft-Rev1_Tables 3.3-1_thru_3.3-28_8-16-04.pdf
007_PPD ENTO-002_Rev5.PDF

In the referenced July 15, 2004, letter (Reference 2) the U.S. Nuclear Regulatory Commission requested additional information to support review of the SERI ESP Application. This letter transmits information as outlined in Attachment 1 to this letter.

Should you have any questions, please contact me.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on August 16, 2004.

Sincerely,

George A. Zinke
Project Manager
System Energy Resources Inc.

Enclosure: One CD-ROM

Attachment: Attachment 1

cc: Mr. R. K. Anand, USNRC/NRR/DRIP/RNRP
Ms. D. Curran, Harmon, Curran, Spielberg, & Eisenberg, L.L.P.
Mr. W. A. Eaton (ECH) (w/o enclosure)
Mr. B. S. Mallett, Administrator, USNRC/RIV
Mr. J. H. Wilson, USNRC/NRR/DRIP/RLEP

Resident Inspectors' Office: GGNS

ATTACHMENT 1

SSAR Section 1.3 Plant Parameters Envelope (PPE)

Request:

RAI 1.3-1

Please clarify the source term in PPE Sections 9.5.1 and 9.5.2 to reflect effluent release characteristics ("gaseous" vs. "airborne effluents"). Please clarify what is meant by "tritium" in PPE Section 9.5.3 (normal or post-accident). Please clarify how the composite value of "32,699 Ci/yr" in PPE Section 9.5.1 relates to "isotopic values" in Table 1.3-2.

Response:

Table 1.3-1, Section 9.5.1, "Gaseous (Normal)" will be changed to "Airborne Effluents (Normal)" and Section 9.5.2, "Gaseous (Post-Accident)" will be changed to "Airborne Effluents (Post-Accident)" to improve clarity. Section 9.5.3 gives the normal tritium annual release. This section title will be changed to "Tritium Airborne Effluent (Normal)". The composite value (32,699 Ci/yr) is the total airborne effluent release including tritium. This is the same value as is given in Table 1.3-2. Definitions for these terms are also provided in Table 1.3-3.

See file: 001_Draft-Rev1_Tables 1.3-1_thru_1.3-3_08-16-04.pdf

SSAR Section 1.4 Conformance with Regulatory Requirements and Guidance

Request:

RAI 1.4-1

Please provide a comprehensive listing of NRC regulations and regulatory guidance applicable to the Grand Gulf ESP SSAR and the affected SSAR sections. For example, please state whether 10 CFR 100.21(f) and Regulatory Guide 4.7 apply to SSAR Section 3.1.6, "Security Plan."

Response:

In general, 10 CFR 52.1, 52.11, and 52.17 define the scope, applicable regulations, and required content of safety assessments associated with the application for an early site permit. As stated in the ESP application, Part 2, Section 1.1, the application was prepared to meet the requirements of Part 52.17. The application's safety assessment discussion identifies, as appropriate in each SSAR section, those specific regulations and regulatory guidance that are applicable to an early site permit. The ESP application contains no requests for exemption from any applicable regulation.

For example, SSAR Section 3.0 discusses conformance with the applicable requirements of 10 CFR 100. Specific to the RAI's request, SSAR 3.1.6.1 evaluates key site characteristics that should be considered in assessing the ability to develop adequate security plans for the proposed new facility at the GGNS site. Section 3.1.6.1 also specifically indicates the ability of the site to comply with the security related guidance of Regulatory Guide 4.7 (i.e., the minimum distance to vital structures and equipment), sufficient to satisfy the requirements of Part 73.55. 10CFR 100.21(f) does apply to SSAR section 3.1.6.

SSAR Section 1.4 and Table 1.4-1, patterned after the guidance of Regulatory Guide 1.70, Section 1.8, lists those (Division 1) regulatory guides that were considered relevant and applicable to an ESP application, as discussed in the ESP SSAR.

Entergy believes the ESP application as a whole comprehensively addresses compliance with the applicable regulations and identifies regulatory guides applicable to the various technical topics in the ESP Application. If the NRC Staff has other specific regulation or regulatory guide questions, please advise.

<u>**SSAR Section 2.1.1 Site Location and Description**</u>

<u>**Request:**</u>

<u>**RAI 2.1-1**</u>

Please provide the altitude and longitude of the proposed new reactor site complete with Universal Transverse Mercator zone numbers.

<u>**Response:**</u>

From the ESP Application SSAR:
"1.2 General Site Description
.... The Universal Transverse Mercator (UTM) Grid Coordinates for the approximate center of the location of the power block area of a new facility are N 3542873 meters, and E 684021 meters."

These UTM coordinates for UTM Zone 15, correspond to a latitude and longitude of:

Using the International Ellipsoid:
32 deg, 00 min, 23.565415 sec N
91 deg, 03 min, 06.764252 sec W

Using the Average Terrestrial Ellipsoid:
32 deg, 00 min, 25.326755 sec N
91 deg, 03 min, 06.420908 sec W

Using the NAD83 (WGS84) Ellipsoid:
32 deg, 00 min, 25.290236 sec N
91 deg, 03 min, 06.423875 sec W

Request:

RAI 2.1-2

Please provide the potential radioactive gaseous and liquid effluent release points and their locations from the proposed ESP site.

Response:

Liquid Effluent Release Point

The proposed approximate location for the liquid effluent discharge point from the new facility is depicted on ER Figures 2.1-1 and 2.1-2.

The SSAR Sections 2.4.1.1 states: "Effluent from a new facility would be combined with that from the existing GGNS Unit 1 facility, and the combined effluent would be discharged into the river downstream of the intake such that recirculation to the embayment area and intake pipes would be precluded."

The SSAR Sections 2.4.11.4 states: "Effluent from a new facility would be combined with that from the existing GGNS Unit 1 facility, and would be discharged into the river downstream of the new facility intake such that recirculation to the embayment area and intake screens of the new facility would be precluded."

Gaseous Effluent Release Point

The potential gaseous effluent release point is assumed to be within the proposed construction area designated for the new facility power block. This is indicated on Figure 2.1-1 in the SSAR.

From Section 2.1.1.1 of the SSAR: "The exclusion area boundary (EAB) for a new facility consists of a circle of approximately 0.52 miles (841 meters) radial distance from the circumference of a 630 ft. circle encompassing the proposed power block location for a new facility. Thus the minimum distance to the exclusion area boundary from any individual new reactor site within the 630 ft. circle would be 0.52 miles (841 meters). .... Distances from the proposed location of a new facility's reactor site to the nearest plant site property boundary in each of the sixteen sectors are given in Table 2.3-143, which also gives annual average atmospheric dispersion factors."

As stated in SSAR Section 2.3.5.2: "The [diffusion] analysis assumed a combined vent located at the center of the proposed facility location. ... For this analysis, routine releases from a new facility were conservatively modeled as ground level releases."

Request:

RAI 2.1-3

Please state if there are any physical characteristics unique to the proposed ESP site that could pose a significant impediment to the development of site emergency plans.

Response:

As discussed in the ESP Application Part 4, Section 2.2.4, an evaluation (i.e. "...preliminary analysis ... "[1]) of the previous GGNS Unit 1 Evacuation Time Estimate (ETE) was made to identify any physical characteristics that might constitute a significant impediment to the development of site emergency plans. As noted in Section 2.2.4.4, that evaluation concluded that there were no such physical characteristics. This conclusion was based on the ETE evaluation as well as discussion with local Emergency Management Agency and local Department of Transportation officials. In addition, numerous government officials (in Mississippi and Louisiana) involved in supporting emergency preparedness for the current operating GGNS Unit 1 indicated (as documented in correspondence enclosed in Appendix A to Part 4) that they were not aware of any significant impediments to the development of emergency plans should a future nuclear facility be constructed at the existing GGNS site, thus, further confirming the conclusion reached in Part 4, Section 2.2.4.4.

[1] NUREG-0654 Rev. 1, Supplement 2, Section II.A. states "An ESP applicant may identify such unique physical characteristics by performing a preliminary analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations, noting major impediments to the evacuation or the taking of other protective actions."

SSAR Section 2.1.2 Exclusion Area Authority and Control

Request:

RAI 2.2-1

You stated that an arrangement would be made to authorize Entergy Operations to maintain control of ingress to and egress from the new proposed ESP exclusion area and provide for evacuation of individuals from the new proposed ESP exclusion area in the event of an emergency. Please state if you would make such arrangement prior to issuance of the Grand Gulf ESP.

Response:

This arrangement would not be made prior to issuance of the Grand Gulf ESP. Such arrangements would be made associated with a Combined License application.

Request:

RAI 2.2-2

The new proposed Grand Gulf ESP exclusion area extends into bodies of Lake Hamilton water. Please state whether appropriate arrangements will be made with the local, state, Federal, or other public agency having authority over the body of water for the exclusion and ready removal of personnel and property from the area in an emergency, by either the applicant or the public agency in authority, of any persons on those portions of the body of water which lie within the designated exclusion area.

Response:

As shown on Figure 2.1-1 in Part 2, Chapter 2 of the ESP Application (Cover Letter Reference 1), and described in Part 4, Section 2.1.1 of the application, Hamilton Lake is on SERI-owned property. Entergy Operations (i.e. the licensed operator for GGNS Unit 1) currently has the authority and responsibility for persons in that area in accordance with the Grand Gulf Nuclear Station Emergency Plan.

SSAR Section 2.1.3 Population Distribution

Request:

RAI 2.1.3-1

SSAR Figure 2.1-5 illustrates the approximate LPZ, among others. Please provide an expended and legible figure to clearly show the population zone and the distance.

Response:

See files: 002_Fig 2.1-5 Draft Rev1.PDF
003_Figure2.1-5.tif
004_Figure2.1-5.DWG

Request:

RAI 2.1.3-2

Please describe appropriate protective measures that could be taken on behalf of the populace in the low population zone which includes transient population in the Grand Gulf Military Park in the event of a reactor accident.

Response:

Offsite protective measures are the responsibility of the applicable state and local governments and are described in the respective emergency plans. The plans were provided to the NRC on June 3, 2004 by CNRO-2004-00036 (Cover Letter Reference 4).

Request:

RAI 2.1.3-3

Please clarify whether your current and projected population data shown in SSAR Tables 2.1-1 and 2.1-2 includes weighted transient population.

Response:

Population data shown in SSAR Tables 2.1-1 and 2.1-2 does not include weighted transient population.

SSAR Section 3.3 Postulated Accidents and Accident Dose Consequences

Request:

RAI 3.3-1

Section 3.3 of the SSAR stated that

> "....doses from postulated design basis accidents are calculated for hypothetical individuals, located at the closest point on the exclusion boundary for a two-hour period,....."

Please discuss why doses were not calculated for any two-hour period with the greatest exclusion area boundary (EAB) doses Please clarify whether the "0 to 2 hour" time period shown in SSAR Section 3.3.2 indicates that this time period is also for any two-hour period with the greatest EAB doses.

Response:

The doses presented in this section are based on the guidance provided in Regulatory Guide 1.183 for the plant types that use the Alternate Source term methodology. For these plants, the Exclusion Area Boundary dose is determined using a sliding 2-hour dose window. The correct statement for these plants is "... for any two-hour period with the greatest EAB doses." For the ABWR, doses were based on TID-14844, and the EAB doses are calculated for the first two hours post-accident. Section 3.3 and 3.3.2 will be revised to clarify this.

See file: 005_ESP-SSAR_DraftRev-1_08-16-04.pdf

Request:

RAI 3.3-2

Section 3.3.3 of the SSAR stated that time-dependent activities released to the environs were used in dose estimates and they are provided in tables in Section 3.3 for certain design basis accidents (DBAs). Please provide time-dependent activities released to the environs in curies for all DBAs. Please provide the references and the methodology used to determine the time-dependent activity release values in these tables. Also, please ensure that the values in these tables appropriately reflect the certified AP-1000 design χ/Qs as discussed in RAI 3.3-4.

Response:

The time dependent releases were provided by the reactor vendors. The methodology used to determine these releases is provided in the vendor's certified design documents. Time dependent release activities in curies will be added for the remaining design basis accidents evaluated in the ER. These new tables to be added are:

TABLE 3.3-23: AP1000 LOCKED ROTOR ACCIDENT

TABLE 3.3-24: AP1000 CONTROL ROD EJECTION ACCIDENT - CURIES RELEASED TO ENVIRONMENT BY INTERVAL – PRE-EXISTING IODINE SPIKE

TABLE 3.3-25: AP1000 STEAM GENERATOR TUBE RUPTURE ACCIDENT - CURIES RELEASED TO ENVIRONMENT BY INTERVAL - ACCIDENT INITIATED IODINE SPIKE

TABLE 3.3-26: AP1000 STEAM GENERATOR TUBE RUPTURE ACCIDENT - CURIES RELEASED TO ENVIRONMENT BY INTERVAL – PRE-EXISTING IODINE SPIKE

TABLE 3.3-27: AP1000 SMALL LINE BREAK ACCIDENT - CURIES RELEASED TO ENVIRONMENT - ACCIDENT INITIATED IODINE SPIKE

TABLE 3.3-28: AP1000 DESIGN BASIS LOSS OF COOLANT ACCIDENT - CURIES RELEASED TO ENVIRONMENT BY INTERVAL

For the AP1000, the dose estimates were performed by using a ratio of the GGNS ESP X/Q values to the AP1000 X/Q values. This same approach was used for the ABWR LOCA dose estimates. For the remaining ABWR events, the vendor did not, in general, provide the low population zone doses. For these accidents, the isotopic releases provided by the vendor were used to determine the offsite (EAB and LPZ) doses. For these ABWR accidents, the methodology used was based on TID-14844. For the ACR-700, the TEDE doses were calculated as the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure.

Refer to the responses to Requests 3.3-4 and 3.3-7 for discussion of the AP1000 certified X/Q values used in the calculations.

See files: 005_ESP-SSAR_DraftRev-1_08-16-04.pdf
006_Draft-Rev1_Tables 3.3-1_thru_3.3-28_8-16-04.pdf

Request:

RAI 3.3-3

Section 3.3.3 of the SSAR stated that the advanced boiling water reactor (ABWR) accident evaluation (other than a loss-of-coolant accident (LOCA)) used the alternative source term methodology in accordance with Regulatory Guide (RG) 1.183. Because the ABWR design is certified with TID-14844 source term and with the radiological consequence dose criteria in thyroid and whole body doses, justify the use of the alternative source term methodology in accordance with RG 1.183 for evaluating the ABWR radiological consequence analyses. The RG stated in its introduction section that a holder of an operating license issued prior to January 10, 1997, or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued prior to January 10, 1997, is allowed by 10 CFR Part 50.67, "Accident Source Term," to voluntarily revise the source term used in design basis radiological consequence analyses.

Response:

This section will be revised to clarify that the ABWR accident evaluations were based on TID-14844. The results given Table 3.3-1 will be revised to give the offsite doses in terms of thyroid and whole body doses. In addition, these results will be revised to incorporate the updated GGNS ESP X/Q values generated using 2002 - 2003 meteorological data.

See files: 005_ESP-SSAR_DraftRev-1_08-16-04.pdf
006_Draft-Rev1_Tables 3.3-1_thru_3.3-28_8-16-04.pdf

Request:

RAI 3.3-4

Section 3.3.3 of the SSAR stated that the AP-1000 accident evaluation used alternative source term methodology in accordance with RG 1.183. Westinghouse has revised its χ/Qs in the AP-1000 design certification control document since submittal of the Grand Gulf ESP application. Please use the most current χ/Qs in the Westinghouse AP-1000 Design Control Document and revise the site-specific doses and fission product releases for all DBAs in SSAR Section 3.3 accordingly, or please note that the AP1000 values used in the SSAR have been revised but the applicant has elected not to use the updated values in the accident analyses.

Response:

The change in the Westinghouse AP1000 X/Qs is a result of ongoing revisions of the AP1000 LOCA analysis. Since the AP1000 LOCA analysis is currently undergoing review and may undergo additional changes in the future, SERI has elected not to use the updated X/Q values in the accident analysis. However, the results for the AP1000 design basis events will be revised based on the updated GGNS ESP X/Q values generated using 2002-2003 meteorological data.

See files: 005_ESP-SSAR_DraftRev-1_08-16-04.pdf
006_Draft-Rev1_Tables 3.3-1_thru_3.3-28_8-16-04.pdf

Request:

RAI 3.3-5

Please state the χ/Q values used for evaluating the radiological consequences for the ACR-700 LOCA in Table 3.1-1.

Response:

The evaluation of the design basis accident for the ACR-700 was based on the releases provided by the vendor and the GGNS ESP site specific x/Q values. The GGNS site specific X/Q values will be updated using 2002-2003 meteorological data. The GGNS site X/Q values are indicated in SSAR Section 2.3.4.2.

See file: 005_ESP-SSAR_DraftRev-1_08-16-04.pdf

Request:

RAI 3.3-6

Section 3.3.3 of the SSAR stated that the reactor accident source term for the ACR-700 design uses a non-mechanistic approach based on TID-14844 and they are provided by the reactor vendor. Please provide the reactor accident source term used for the ACR-700 design.

Response:

The design basis accident releases for the ACR-700 are given in Table 3.3-17 of the SSAR.

Request:

RAI 3.3.7

SSAR Table 3.1-1 summarizes the resulting doses at the ESP site for postulated design basis accidents using the AP-1000, the ABWR, and the ACR-700 as surrogate reactor designs. Please update the table for each design basis accident to include (1) AP-1000, ABWR, and ACR-700 χ/Q values and doses used for the EAB and LPZ, and (2) the ratios of site-specific χ/Qs to design certification χ/Qs used.

Response:

The AP-1000 dose analysis used a ratio of the X/Qs to determine the GGNS offsite doses. For the AP-1000 design, the X/Q's used are as listed below:

AP-1000 Atmospheric Dispersion Factors

Atmospheric Dispersion (CHI/Q) (Accident)	EAB=0.5 mi LPZ=2 mi
0-2 hr @ EAB	6.0 E-4 sec/m^3
0-8 hr @ LPZ	1.35 E-4 sec/m^3
8-24 hr @ LPZ	1.0 E-4 sec/m^3
1-4 day @ LPZ	5.4 E-5 sec/m^3
4-30 day @ LPZ	2.2 E-5 sec/m^3

The ABWR LOCA analysis also used a ratio of X/Q values to determine the equivalent GGNS offsite dose. The X/Q values used for the ABWR evaluation are as given below:

ABWR X/Q VALUES (sec/m^3)
[From ABWR SSAR Table 15.6-13]

	0 – 2 Hrs	0 – 8 Hrs	8 – 24 Hrs	24 – 96 Hrs	96 – 720 Hrs
EAB	1.37E-03				
LPZ		1.56E-04	9.61E-05	3.36E-05	7.42E-06

The GGNS site specific X/Q values used for the ESP analyses are given in the SSAR in Section 2.3.4.2.

See file: 005_ESP-SSAR_DraftRev-1_08-16-04.pdf

Request:

RAI 3.3.-8

Several tables in Section 3.3 present doses for ABWR design basis accidents in total effective dose equivalent (TEDE) units. Please provide the doses in thyroid and whole body doses in addition to the doses in TEDE units, because the General Electric ABWR design is certified with the thyroid and whole body doses.

Response:

The Chapter 3 tables will be revised to present the offsite doses in terms of thyroid and whole body doses based on the updated GGNS ESP X/Q values using 2002-2003 meteorological data.

See file: 006_Draft-Rev1_Tables 3.3-1_thru_3.3-28_8-16-04.pdf

Programs and Plans - Quality Assurance (QA)

Request:

RAI 17.1-1

Please describe the QA measures used to authenticate and verify data retrieved from internet websites that support information in the SSAR that would affect the design, construction, or operation of structures, systems, and components important to safety.

Response:

The response to Request 17.1-2, part b, provides a copy of Enercon's Quality Assurance Project Planning Document for Entergy Nuclear Potomac Early Site Permitting Project Grand Gulf Nuclear Station Site, Revision 5. Attachment 3 to the QA PPD is Project Instruction ENTO002-PI-02, which describes the controls applied to the collection of data in support of the development of the SSAR for the ESP Application. The PI was prepared for use for collection and review of data which supported those aspects of the SSAR dealing with the safety assessment of the ESP site; specifically hydrological and meteorological data. The PI also indicates the option for its use for other data collection and review, such as demographic data. The PI is applicable to published data, and raw data (e.g., data collected from internet web site databases, etc.). These data sources were documented on Attachment 1 of the PI, as required.

Request:

RAI 17.1-2

Please provide copies of the following documents:

a. Entergy document CNRO-2003-00013, "Quality Assurance Program Manual," Revision 8

b. Enercon Services, Incorporated document PPD ENTO-002, "Quality Assurance Project Planning Document for Entergy Nuclear Potomac Early Site Permitting Project Grand Gulf Nuclear Station Site," Revision 5

c. Enercon Services, Incorporated document, "Quality Assurance Program," Revision 8

Response:

a. The ADAMS Accession No. for the requested document is ML031210008

b. See file: 007_PPD ENTO-002_Rev5.PDF

c. This document will be provided under separate transmittal letter.

Request:

RAI 17.1-3

The QA project planning document identified QA requirements only applicable to the ESP project. Specifically, of the 18 elements in the Enercon QA Program manual, the following elements were not applicable to the ESP project: Section 8.0, "Identification and Control of Material, Parts and Components"; Section 9.0, "Control of Special Processes"; Section 10.0, "Inspections"; Section 11.0, Test Control"; Section 14.0, "Inspection, Test and Operating Status"; Section 15.0, "Nonconforming Materials, Parts or Components." Please describe why these QA measures were not applicable to the development of the ESP application. Alternatively, if these QA measures were applicable to the ESP application, please describe the QA measures used by Entergy and/or the primary contractor (Enercon) for these activities.

Response:

A summary of the reasoning applied to each criteria listed in the RAI is provided below.

8.0 Identification and Control of Materials, Parts and Components

The ESP Project did not require procurement, fabrication, receipt or erection of safety related materials, parts, components or partially fabricated assemblies for installation into a nuclear power plant. The applicable quality controls and the governing codes and national consensus standards for control of all equipment used in laboratory testing of samples are specified in Project Instruction ENTO002-PI-05. This criterion is therefore not applicable to the ESP project.

9.0 Control of Special Processes

The ESP project did not involve any special processes including welding, heat treating, or non-destructive examination. There are no requirements for use of personnel qualified in accordance with specific codes and standards governing nondestructive examination activities. Therefore, this criterion is not applicable to the ESP project.

10.0 Inspection

For the ESP project, no materials or products are being processed that would require inspection. Neither inspection activities nor process monitoring activities were required during the ESP project. In addition, no hold or witness points were required to be established for any of the activities performed under the ESP project. Quality surveillances and audits were performed for activities under the QAPPD as discussed under Criterion XVIII, Audits. Therefore, this criterion is not applicable to the ESP project.

11.0 Test Control

The ESP project did not require a test program to demonstrate that structures, systems and components would perform satisfactorily in service. No testing was performed under the ESP Project that relates to proof testing, preoperational testing or operational testing.

The testing addressed by this criterion is different than laboratory testing performed on samples taken for seismological data collection activities and field geophysical testing. The appropriate quality controls have been specified and implemented for laboratory and geophysical testing and analysis in accordance with the requirements of Criterion III, Design Control. The applicable quality controls and the governing codes and national consensus standards for testing of these samples are specified in Project Instruction ENTO002-PI-05.

Therefore, this criterion is not applicable to the ESP project.

14.0 Inspection, Test and Operating Status

The ESP Project did not entail the design, fabrication, or installation of any safety related components, systems or structures. Accordingly, there were no inspections or tests of safety related components, systems or structures required as part of the ESP project. No measures were required to indicate whether components have passed inspections or tests. Likewise, there are no systems, structures or components that are presumed to operate where tagging or operational controls would be required to indicate status. The applicable quality controls and the governing codes and national consensus standards for control of all equipment used in laboratory testing of samples are specified in Project Instruction ENTO002-PI-05. Therefore, this criterion in 10 CFR Part 50, Appendix B is not applicable to the ESP Project.

15.0 Nonconforming Materials, Parts, or Components

The ESP project did not involve design, procurement, fabrication, delivery or receipt of any safety related components, parts or materials. Therefore no measures were required to prevent the inadvertent use or installation of nonconforming materials, parts or components. Provisions are included in the quality processes in accordance with Criterion XVI, Corrective Actions, to identify any conditions adverse to quality and initiate corrective actions. The applicable quality controls and the governing codes and national consensus standards for control of all equipment used in laboratory testing of samples are specified in Project Instruction ENTO002-PI-05. Therefore, this criterion is not applicable to the ESP project.

Request:

RAI 17.1-4

A special team inspection was conducted from February 9-13, 2004, to review aspects of applicant and contractor quality control activities involved with the preparation of the application for the Grand Gulf ESP. The team identified an open item regarding an issue which was not addressed during the inspection. The open item involves the applicability of 10CFR Part 21, "Report of Defects and Noncompliance," to the Exelon ESP project. Please describe the actions taken to ensure the Entergy ESP project complies with Part 21.

Response:

In response to this request, we have assumed that the reference to "Exelon" is an error, and that the open item referred to is Open Item 052000009/2004001-02, "Applicability of Part 21 to Early Site Permit Application Process," identified in cover letter Reference 3.

All project activities, with the exception of static soil testing conducted by Eustis Laboratories, were conducted under the Entergy, Enercon or EPRI Quality Assurance Programs. The Entergy, Enercon and EPRI QA programs apply Part 21 requirements and therefore comply with Part 21.

The Enercon QA Program qualified, via source audit, the Eustis QA Program for performing static soil testing. Eustis does not maintain a Part 21 reporting process. Assurance of the adequacy of the work provided by Eustis is well documented in project records. This assurance can be summarized as follows:

- All testing was done in accordance with the applicable ASTM standards as prescribed in Enercon QA Procedures.

- In addition to a program review described in Enercon's audit of Eustis, it was noted that just prior to performing these tests for Enercon/Lettis, Eustis had been certified by the US Army Corps of Engineers. This USACE certification was for performing the same ASTM test procedures required by Enercon/Lettis for the ESP project. The verification/certification done by the USACE not only included review of QA controls, but also in some cases involved independent verification of obtained results using Corps of Engineer test equipment which was totally independent of that used by Eustis.

- Results obtained from Eustis tests were compared to geological data obtained for the original Grand Gulf site selection process with generally good agreement. This is documented in Section 10.5 of Engineering Report ENTO002-ER-02, Geologic, Geotechnical, and Geophysical Field Exploration and Laboratory Testing, Grand Gulf Nuclear Station, Early Site Permit.

- During the ESP phase, static soil data provided by Eustis is used in a qualitative, rather than quantitative manner for soil classification, general evaluation of geologic hazards and comparative evaluation with UFSAR data. During the second phase (COL phase) the data may be used to supplement additional information required to be collected during the COL phase (additional bore holes and testing will be required at COL). This COL data would be used to evaluate liquefaction potential, foundation bearing capacities, foundation settlement settlements and excavation design. The work performed for the COL phase, which will serve to further validate the original Eustis work, will require Part 21 requirements to be applied.

XEROX

Administrator

Document Name:	Untitled
Printing Time:	10/19/04 13:41:42
Copies Requested:	1
Virtual Printer:	dt6180/6180hold
Printed For:	Administrator

Administrator

TABLE 1.3-1

PLANT PARAMETERS ENVELOPE (PPE)

PPE Section	Composite Value[1]	Comments	Value[2]
2. Normal Plant Heat Sink			
2.3 Condenser			
2.3.2 Condenser / Heat Exchanger Duty	10.7 E9 Btu/hr		US
2.4 NHS Cooling Towers – Mechanical Draft (Natural Draft) (See Note 3)			
2.4.6 (2.5.6) Cycles of Concentration	4		US
2.4.8 (2.5.8) Height	60 ft (475 ft)		US
2.4.9 (2.5.9) Makeup Flow Rate	47,900 gpm expected (78,000 gpm max)		TP
2.4.12 (2.5.12) Cooling Water Flow Rate	865,000 gpm		US
5. Potable Water/Sanitary Waste System			
5.2 Raw Water Requirements			
5.2.1 Maximum Use	240 gpm		TP
5.2.2 Monthly Average Use	180 gpm		TP
6. Demineralized Water System			
6.2 Raw Water Requirements			
6.2.1 Maximum Use	1440 gpm		TP
6.2.2 Monthly Average Use	1100 gpm		TP
7. Fire Protection System			
7.1 Raw Water Requirements			
7.1.1 Maximum Use	1890 gpm		TP
7.1.2 Monthly Average Use	(30 gpm)		TP
9. Unit Vent/Airborne Effluent Release Point			
9.4 Release Point			
9.4.2 Elevation (Normal)	Ground level		US
9.4.3 Elevation (Post Accident)	Ground level		US
9.4.4 Minimum Distance to Site Boundary	0.52 mi (841 m) exclusion area		US

TABLE 1.3-1 (Continued)

PPE Section		Composite Value[1]	Comments	Value[2]
9.5	Source Term			
9.5.1	Airborne Effluents (Normal)	32,699 Ci/yr	See TABLE 1.3-2	TP
9.5.2	Airborne Effluents (Post-Accident)	Based on limiting DBAs	See Note 4	US
9.5.3	Tritium Airborne Effluent (Normal)	7060 Ci/yr		TP
17.	Plant Characteristics			
17.3	Megawatts Thermal	4300 MWt	Includes allowance for ~10% uprate from 3926 MWt.	US
18.	Construction			
18.4	Plant Population			
18.4.1	Construction	3150 people max		US

NOTES:

1. The "Composite Value" provides an envelope (bounding values) for design parameters for the various plant designs considered for the site. See Site Safety Analysis Report Section 1.3 for a discussion of the basis for parameter values.
2. "Value" pertains to the "Composite Value" for each parameter listed. In this table, a value designated "US" represents a "unit specific" value, meaning that it is applied per unit, or group of units or modules. A designation of "TP" is given to a value that represents total facility requirements. See Site Safety Analysis Report Section 1.3 for a discussion of the basis for parameter values.
3. Several main condenser cooling system alternatives were considered (i.e., mechanical and natural draft cooling towers, cooling ponds, and once-through cooling).
   - The once through cooling option was eliminated due to significant environmental impact.
   - The cooling pond option was eliminated due to insufficient GGNS site acreage to accommodate pond.
4. In general, source terms for any given accident are those used by the vendors in their safety analyses. The methodologies used by the Vendors for establishing source terms include those established in TID-14844 and Regulatory Guide 1.183. See SSAR Sections 3.3.2 and 3.3.3 for additional detail on accident selection and source term methods.

TABLE 1.3-2

NORMAL OPERATIONS GASEOUS RELEASE SOURCE TERM[1]

Radionuclide	Composite Normal Release[2] (Ci/yr)	Radionuclide	Composite Normal Release[2] (Ci/yr)
Kr-83m	1.68E-03	Rb-89	8.65E-05
Kr-85m	7.20E+01	Sr-89	1.14E-02
Kr-85	8.20E+03	Sr-90	3.60E-03
Kr-87	5.03E+01	Y-90	9.19E-05
Kr-88	9.20E+01	Sr-91	2.00E-03
Kr-89	4.81E+02	Sr-92	1.57E-03
Kr-90	6.49E-04	Y-91	4.81E-04
Xe-131m	3.60E+03	Y-92	1.24E-03
Xe-133m	1.74E+02	Y-93	2.22E-03
Xe-133	9.20E+03	Zr-95	3.19E-03
Xe-135m	8.11E+02	Nb-95	1.68E-02
Xe-135	9.19E+02	Mo-99	1.19E-01
Xe-137	1.03E+03	Tc-99m	5.95E-04
Xe-138	8.65E+02	Ru-103	7.03E-03
Xe-139	8.11E-04	Rh-103m	2.22E-04
I-131	5.19E-01	Ru-106	2.34E-04
I-132	4.38E+00	Rh-106	3.78E-05
I-133	3.41E+00	Ag-110m	4.00E-06
I-134	7.57E+00	Sb-124	3.62E-04
I-135	4.81E+00	Sb-125	1.83E-04
C-14	2.19E+01	Te-129m	4.38E-04
Na-24	8.11E-03	Te-131m	1.51E-04
P-32	1.84E-03	Te-132	3.78E-05
Ar-41	1.02E+02	Cs-134	1.24E-02
Cr-51	7.03E-02	Cs-136	1.19E-03
Mn-54	1.08E-02	Cs-137	1.89E-02
Mn-56	7.03E-03	Cs-138	3.41E-04
Fe-55	1.30E-02	Ba-140	5.41E-02
Co-57	2.46E-05	La-140	3.62E-03
Co-58	6.90E-02	Ce-141	1.84E-02
Fe-59	1.62E-03	Ce-144	3.78E-05
Co-60	2.61E-02	Pr-144	3.78E-05
Ni-63	1.30E-05	W-187	3.78E-04
Cu-64	2.00E-02	Np-239	2.38E-02
Zn-65	2.22E-02		
		Total without Tritium	25,639
		Tritium (H-3)	7.06E+03
		Total with Tritium	32,699

NOTES:

1. See PPE Table 1.3-1, Section 9.5.
2. Composite source term based on highest radionuclide release for all plant types considered.

TABLE 1.3-3

PLANT PARAMETERS DEFINITIONS

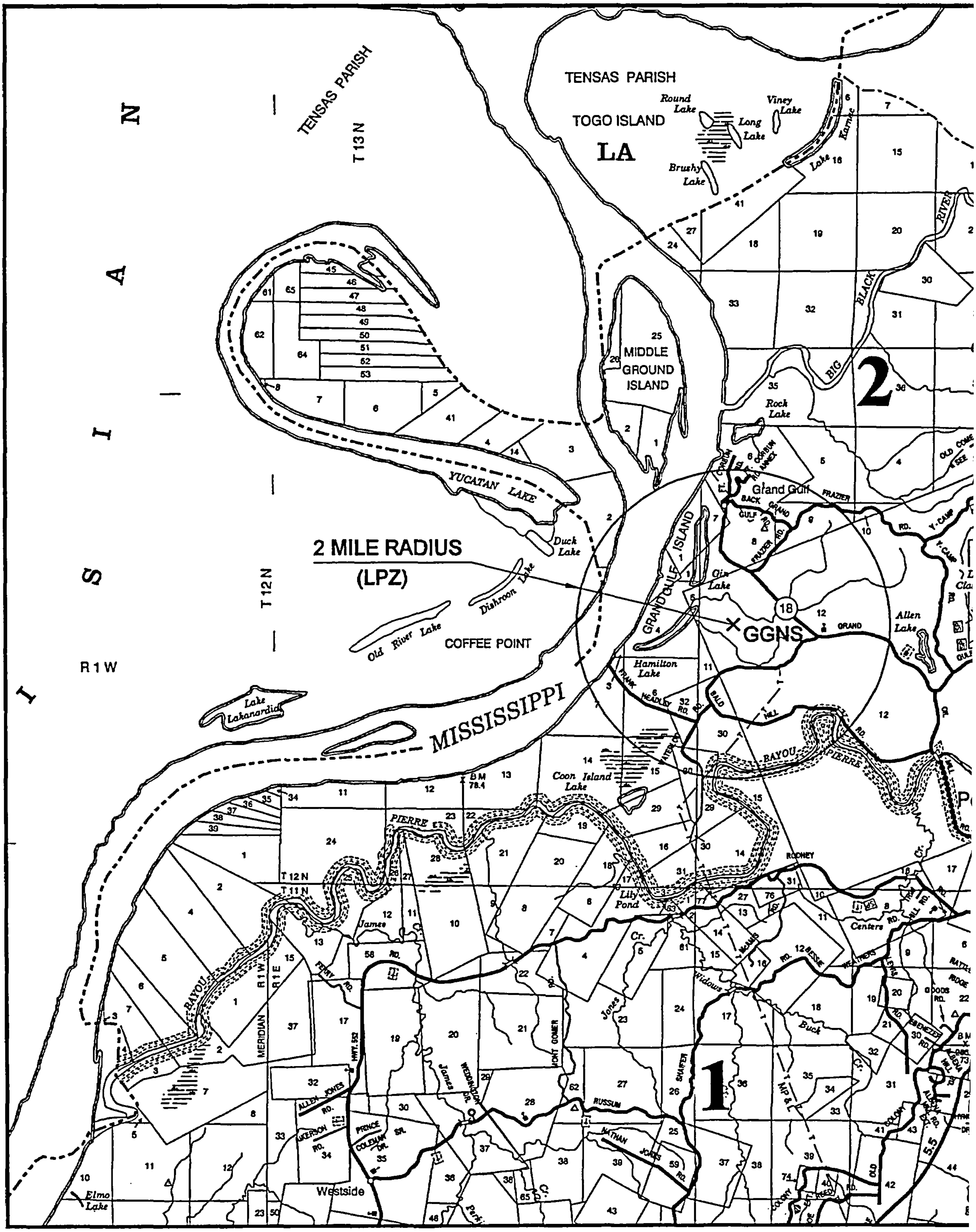
Parameter	Units	Definition	Bounding Value (Notes)
2. Normal Plant Heat Sink			
2.3 Condenser			
2.3.2 Condenser / Heat Exchanger Duty	BTU per hour	Design value for the waste heat rejected to the circulating water system across the normal heat sink condensers	2
2.4 (2.5) NHS Cooling Towers (Mechanical Draft or Natural Draft)			
2.4.6 (2.5.6) Cycles of Concentration	Number of cycles	The ratio of total dissolved solids in the cooling water blowdown streams to the total dissolved solids in the makeup water streams.	1
2.4.8 (2.5.8) Height	Feet	The vertical height above finished grade of either natural draft or mechanical draft cooling towers associated with the cooling water systems.	1
2.4.9 (2.5.9) Makeup Flow Rate	Gallons per minute	The expected (and maximum) rate of removal of water from a natural source to replace water losses from closed cooling water systems.	2
2.4.12 (2.5.12) Cooling Water Flow Rate	Gallons per minute	The total cooling water flow rate through the normal heat sink condensers/heat exchangers.	1
5. Potable Water/Sanitary Waste System			
5.2 Raw Water Requirements			
5.2.1 Maximum Use	Gallons per minute	The maximum short-term rate of withdrawal from the water source for the potable and sanitary waste water systems.	2
5.2.2 Monthly Average Use	Gallons per minute	The average rate of withdrawal from the water source for the potable and sanitary waste water systems.	2
6. Demineralized Water System			
6.2 Raw Water Requirements			
6.2.1 Maximum Use	Gallons per minute	The maximum short-term rate of withdrawal from the water source for the demineralized water system.	2
6.2.2 Monthly Average Use	Gallons per minute	The average rate of withdrawal from the water source for the demineralized water system.	2
7. Fire Protection System			
7.1 Raw Water Requirements			
7.1.1 Maximum Use	Gallons per minute	The maximum short-term rate of withdrawal from the water source for the fire protection water system.	2
7.1.2 Monthly Average Use	Gallons per minute	The average rate of withdrawal from the water source for the fire protection water system.	2
9. Unit Vent/Airborne Effluent Release Point			
9.4 Release Point			
9.4.2 Elevation (Normal Operation)	Feet	The elevation above finished grade of the release point for routine operational releases.	3

TABLE 1.3-3 (Continued)

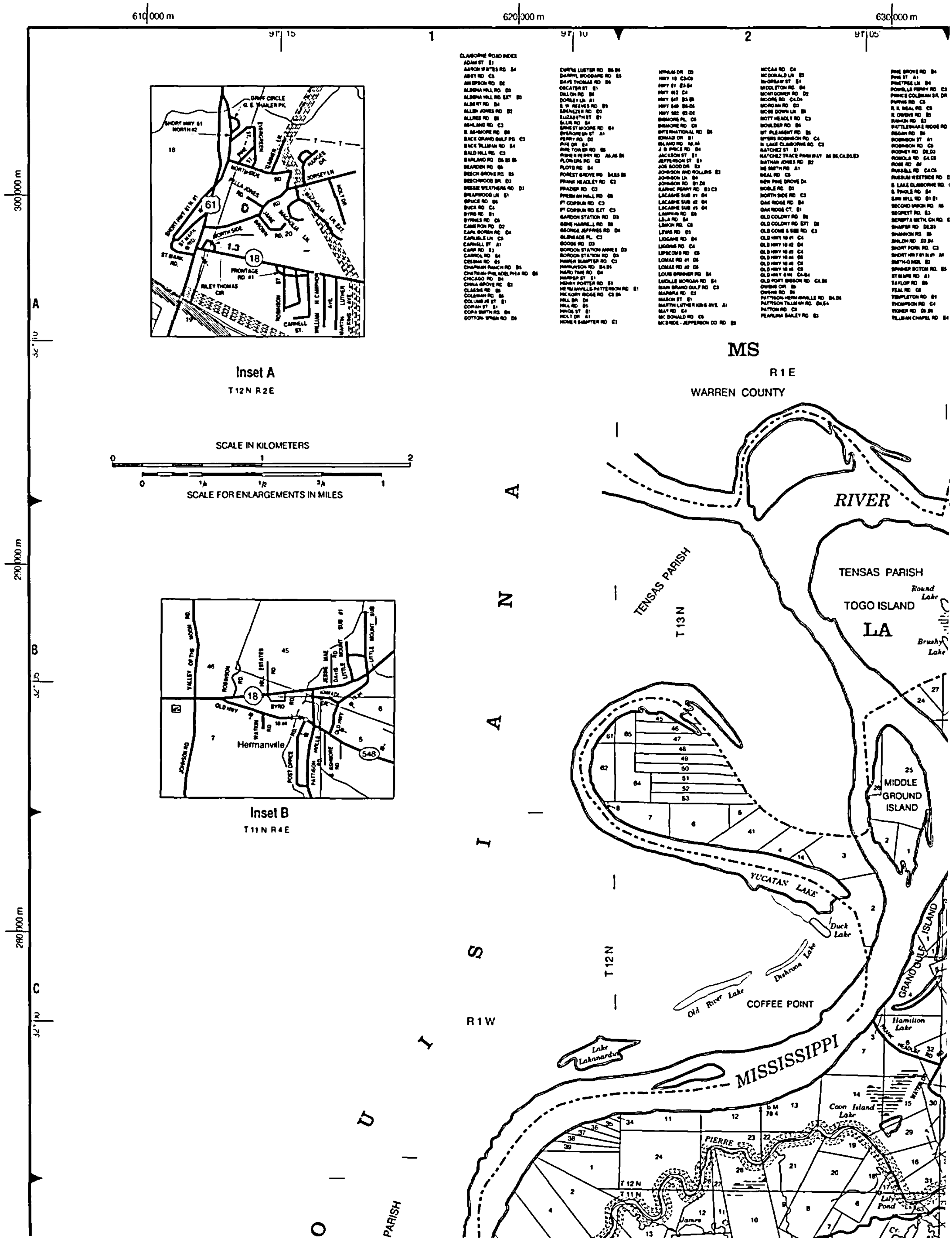
Parameter	Units	Definition	Bounding Value (Notes)
9.4.3 Elevation (Post Accident)	Feet	The elevation above finished grade of the release point for accident sequence releases.	3
9.4.4 Minimum Distance to Site Boundary	Feet	The minimum lateral distance from the release point to the site boundary.	3
9.5 Source Term			
9.5.1 Airborne Effluents (Normal)	Curies per year	The annual activity, by isotope, contained in routine (normal) plant airborne effluent streams.	2
9.5.2 Airborne Effluents (Post-Accident)	Curies	The activity, by isotope, activity contained in post-accident airborne effluents.	1
9.5.3 Tritium Airborne Effluent (Normal)	Curies per year	The annual activity of tritium contained in routine (normal) plant airborne effluent streams.	2
17. Plant Characteristics			
17.3 Megawatts Thermal	Mega-watts	The maximum thermal power generated by a single unit or group of units/modules of a specific reactor plant type.	2
18. Construction			
18.4 Plant Population			
18.4.1 Construction	Persons	The number of people required to construct the plant.	2

NOTES:

1. The Bounding Value is the maximum value for any of the plant designs being considered for the site.
2. The Bounding Value is the maximum value for any of the plant design/number of unit combinations being considered for the site.
3. The Bounding Value is the minimum value for any of the plant designs being considered for the site.



TENSAS PARISH
TOGO ISLAND
LA
Round Lake
Viney Lake
Long Lake
Brushy Lake
Lake Karnac
T 13 N
T 12 N
T 11 N
R 1 W
R 1 E
MERIDIAN
MIDDLE GROUND ISLAND
Rock Lake
BIG BLACK RIVER
YUCATAN LAKE
Duck Lake
2 MILE RADIUS
(LPZ)
Dishroon Lake
Old River Lake
COFFEE POINT
Lake Lakanardia
MISSISSIPPI
Grand Gulf
GRAND GULF ISLAND
Gin Lake
GGNS
Hamilton Lake
Allen Lake
Coon Island Lake
Lily Pond
PIERRE
BAYOU
RODNEY
Centers Rd.
Widows
Buck Cr.
James
Jones
HWY. 552
ALLEN JONES RD.
PRINCE COLEMAN DR.
RUSSUM
NATHAN JONES RD.
SHAFER
Westside
Elmo Lake
2
1



610 000 m
620 000 m
630 000 m
300 000 m
290 000 m
280 000 m
1
2
A
B
C
CLAIBORNE ROAD INDEX
Inset A
T 12 N R 2 E
SCALE IN KILOMETERS
SCALE FOR ENLARGEMENTS IN MILES
Inset B
T 11 N R 4 E
Hermanville
MS
R 1 E
WARREN COUNTY
RIVER
TENSAS PARISH
TOGO ISLAND
LA
Round Lake
Brushy Lake
T 13 N
MIDDLE GROUND ISLAND
YUCATAN LAKE
Duck Lake
Dahroon Lake
Old River Lake
COFFEE POINT
T 12 N
R 1 W
Lake Lakanardu
MISSISSIPPI
Hamilton Lake
GRAND GULF ISLAND
Coon Island Lake
PIERRE
Lily Pond
T 12 N
T 11 N
L O U I S I A N A
PARISH

LIST OF TABLES (Continued)

LIST OF FIGURES

3.3 Postulated Accidents And Accident Dose Consequences

10 CFR 52.17(a)(1) requires a site safety assessment that demonstrates the acceptability of the site under the radiological consequence evaluation factors identified in §50.34(a)(1) and that site characteristics comply with 10 CFR 100. Specifically, 10 CFR 100.21(c)(2) requires that radiological dose consequences of postulated accidents meet the criteria set forth in 10 CFR 50.34(a)(1). This section will review and analyze a robust spectrum of design basis accidents (DBAs) in order to bracket post-accident radiological consequences for the reactor or reactors proposed for the Grand Gulf Nuclear Station (GGNS) site, to demonstrate that a reactor or reactors could be sited at the GGNS ESP Site without undue risk to the health and safety of the public. Pursuant to 10 CFR 50.34(a)(1), doses from postulated design basis accidents are calculated for hypothetical individuals, located at the closest point on the exclusion area boundary for a two-hour period (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 suitability of the site.

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

3.3.1 Selection of Design Basis Accidents

A set of postulated accidents was analyzed to demonstrate that a reactor or reactors bounded by parameters defined herein can be operated on the ESP Site without undue risk to the health and safety of the public. The set of accidents was selected to cover a range of events in Regulatory Guide 1.183 (Reference 6), NUREG-0800 and NUREG-1555 for various reactor types. Evaluation of this set of accidents provides a basis for establishing site suitability. It is not the intent, nor is it strictly possible, to analyze all possible accidents for each of the reactor types identified in Section 1.3. The set of accidents chosen considers those with potential bounding impact, as well as accidents of lesser impact but greater frequency. The bounding accidents selected focus, for the most part, on the LWR designs because various LWR plants have certified standard designs, and they have accepted postulated accident bases.

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

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

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

3.3.2 Evaluation of Radiological Consequences

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

The accident dose evaluations were performed using 0.5 percentile direction dependent atmospheric dispersion (χ/Q) values for the EAB and LPZ which are based on onsite meteorological data (Section 2.3). The site specific χ/Q values are presented in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The accident dose estimates were performed using χ/Q and activity releases for the following intervals:

- Exclusion Area Boundary (EAB)
  - 0 to 2 hours (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

3.3.3 Source Terms

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

[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.

3.3.4 Postulated Accident Analyses

This section identifies the DBAs, the resultant activity release paths, the important accident parameters and assumptions, and the credited mitigation measures used in the offsite dose estimates. A summary of the accident doses and the associated NRC dose limit guidelines are provided in Table 3.3-1.

3.3.4.1 Main Steam Line Break Outside Containment (AP1000)

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

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

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

---

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

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

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

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

The vendor calculated time-dependent offsite dose releases for a representative site (Reference 2). The GGNS ESP-site-specific doses were calculated using the atmospheric dispersion (χ/Q) values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The TEDE doses for the accident-initiated iodine spike are shown in Table 3.3-2. The doses at the EAB and LPZ are a small fraction of the 25 rem TEDE of 10 CFR 50.34. A small fraction is defined, in NUREG-0800 Standard Review Plan 15.0.1 and Regulatory Guide 1.183 (Reference 6), as 10 percent or less of the 25 rem TEDE. The doses for the pre-existing iodine spikes are shown in Table 3.3-3. These doses meet the 25 rem TEDE guideline of 10 CFR 50.34.

3.3.4.2 Main Steam Line Break Outside Containment (ABWR)

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

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

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

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

The vendor calculated time-dependent radionuclide releases for a main steam line break outside the containment. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The activity released to the environment for the maximum activity and pre-existing iodine spike is shown in Table 3.3-4. The calculated doses for the maximum allowed equilibrium activity at full power operation are shown in Table 3.3-5. For this case, the doses at the EAB and LPZ are a small fraction of the 25 rem TEDE guidelines of 10 CFR 50.34 in accordance with NUREG-0800 Standard Review Plan 15.6.4. The calculated doses for the pre-existing iodine spike are shown in Table 3.3-6. The doses at the EAB and LPZ are within the 25 rem TEDE guideline of 10 CFR 50.34.

3.3.4.3 Reactor Coolant Pump Locked Rotor (AP1000)

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

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

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

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

The pre-existing iodine spike has little impact since the gap activity released to the primary side becomes the dominant mechanism with respect to offsite dose contributions. The vendor calculated time-dependent offsite dose releases for a representative site. The activity released to the environment is shown in Table 3.3-23. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The TEDE doses for the locked rotor accident are shown in Table 3.3-7. These doses are a small fraction of the 25 rem TEDE guidelines of 10 CFR 50.34.

3.3.4.4 Control Rod Ejection (AP1000)

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

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

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

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

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

- Containment leak rate – 0.1 percent per day
- Steam generator initial iodine and alkali metal activities – 10 percent of the design basis reactor coolant concentrations at maximum equilibrium conditions
- Reactor coolant alkali metal activity – 0.25 percent design basis fuel defect inventory
- Reactor coolant noble gas activity – limit of 280 μCi/g dose equivalent Xe-133
- Pre-existing iodine spike – reactor coolant at 60 μCi/g dose equivalent Iodine–131
- Fraction of rods with cladding failures – 0.10
- Fission product gap activity fractions:
  - Iodines 0.10
  - Noble gases0.10
  - Alkali metals0.12
- Fraction of fuel melting – 0.0025
- Activity released from melted fuel:
  - Iodines0.5
  - Noble gases1.0
- Iodine chemical form – per Regulatory Guide 1.183 (Reference 6), Regulatory Position C.3.5
- Containment atmosphere activity removal – elemental 1.7/hour; particulate iodine and alkali metals 0.1/hour
- Partition coefficients in steam generators – 0.01 for iodines and 0.001 for alkali metals

The pre-existing iodine spike has little impact since the gap activity released from the failed cladding and melted fuel become the dominant mechanisms contributing to the radioactivity released from the plant. The activity released to the environment is shown in Table 3.3-24. The vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The TEDE doses for the control rod ejection accident are shown in Table 3.3-8. These doses are well within the 25 rem TEDE guidelines of 10 CFR 50.34. NUREG-0800 Standard Review Plan 15.4.8 defines "well within" as 25 percent or less of the applicable limits.

3.3.4.5 Rod Drop Accident (ABWR)

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

3.3.4.6 Steam Generator Tube Rupture (AP1000)

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

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

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

The activity released to the environment for an accident initiated iodine spike and a pre-existing iodine spike are given in Table 3.3-25 and Table 3.3-26, respectively. The vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The TEDE doses for the steam generator tube rupture accident with the accident-initiated iodine spike are shown in Table 3.3-9. The doses at the EAB and LPZ are a small fraction of the 25 rem TEDE guidelines of 10 CFR 50.34 as per NUREG-0800, Standard Review Plan 15.6.3. The pre-existing iodine spike doses are shown in Table 3.3-10. These doses are within the 25 rem TEDE guidelines of 10 CFR 50.34.

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

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

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

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

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

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

The activity released to the environment for an AP1000 small line break accident is shown in Table 3.3-27. The vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The TEDE doses for the failure of small lines carrying primary coolant outside containment are shown in Table 3.3-11. These doses are a small fraction of the 25 rem TEDE guidelines of 10 CFR 50.34 as per NUREG-0800, Standard Review Plan 15.6.2.

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

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

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

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

The vendor calculated the time-dependent radionuclide releases to the environment as shown in Table 3.3-12. These releases were used along with the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ) to determine the offsite doses. The doses for the failure of small lines carrying primary coolant outside containment are shown in Table 3.3-13. These doses are a "small fraction" of the 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.

3.3.4.9 Large Break Loss of Coolant Accident (AP1000)

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

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

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

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

- Duration of accident – 30 days
- Core thermal power of 3468 MWt (102 percent of design core power of 3400 MWt)
- Reactor coolant noble gas activity – limit of 280 μCi/g dose equivalent Xe-133
- Reactor coolant equilibrium iodine activity – 1.0 μCi/g equivalent Iodine-131
- Reactor coolant mass – 3.7E+05 lbm
- Containment purge flow rate – 8,800 cfm for 30 seconds
- Containment leak rate – 0.1 percent per day
- Core activity group release fractions – Regulatory Guide 1.183 (Reference 6), Regulatory Position C.3.2
- Iodine chemical form – Regulatory Guide 1.183, Regulatory Position C.3.5
- Containment airborne elemental iodine removal – 1.7 per hour until decontamination factor (DF) of 200 is reached
- Containment atmosphere particulate removal – 0.43 per hour to 0.72 per hour during first 24 hours

The activity assumed to be released to the environment for an AP1000 loss of coolant accident is shown in Table 3.3-28. The vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The TEDE doses for the AP1000 large break LOCA accident are shown in Table 3.3-14. Both EAB and LPZ doses meet the dose guideline of 25 rem TEDE in 10 CFR 50.34. The activity released from the core melt phase of the accident is the greatest contributor to the offsite doses. The EAB dose in Table 3.3-14 is given for the two-hour period during which the dose is greatest at this location. The initial two hours of the accident is not the worst two-hour period because of the delays associated with cladding failure and fuel damage.

3.3.4.10 Large Break Loss of Coolant Accident (ABWR)

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

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

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

- Duration of accident – 30 days
- Core power level – 4005 MWt (102 percent of design core power of 3926 MWt)
- Fraction of noble iodine and noble gases released – Regulatory Guide 1.3, Regulatory Positions C.1.a and C.1.b.
- Iodine chemical form – Regulatory Guide 1.3, Regulatory Position C.1.a
- Suppression pool iodine decontamination factor – 2.0 for particulate and elemental iodine (includes allowance for suppression pool bypass)
- Primary containment leakage – 0.5 percent/day
- Main steam isolation valve total leakage – 66.1 liters/minute
- Condenser leakage rate – 11.6 percent/day
- Condenser iodine removal:
- Elemental and particulate iodine99.7 percent
- Organic iodine0.0 percent
- Delay to achieve design negative pressure in reactor building - 20 minutes
- Reactor building leak rate during draw down – 150 percent/hour
- Standby gas system filtration – 97 percent efficiency
- Standby gas system exhaust rate – 50 percent/day

The vendor calculated the time-dependent offsite doses for a representative site. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The activities released to the environment from the reactor and turbine buildings are listed in Table 3.3-15. The doses for the ABWR large break LOCA accident are shown in Table 3.3-16. Since the vendor evaluation of this postulated accident is based on TID-14844 and Regulatory Guide 1.3 methodology, the offsite dose acceptance criteria of 10 CFR 100 is used. The calculated doses meet the dose guidelines of 300 rem thyroid and 75 rem whole body as specified in 10 CFR 100.

3.3.4.11 Large Loss of Coolant Accident (ACR-700)

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

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

3.3.4.12 Fuel Handling Accidents (AP1000)

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

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

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

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

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

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

The radioactivity released to the environment is listed in Table 3.3-19. The GGNS ESP-site-specific doses were calculated using the atmospheric dispersion (χ/Q) values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The resulting doses at the EAB and LPZ are summarized in Table 3.3-20. The doses are applicable to fuel handling accidents inside containment and in the spent fuel pool in the auxiliary building. The EAB and LPZ doses are well within the 25 rem TEDE guidelines given in 10 CFR 50.34. "Well within" is taken as being 25 percent of the guideline, consistent with the guidance of Regulatory Guide 1.183 (Reference 6) and NUREG-0800, Standard Review Plan 15.7.4.

3.3.4.13 Fuel Handling Accidents (ABWR)

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

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

- Core thermal power – 4,005 MWt (102 percent of design core power of 3926 MWt)
- Decay time after shutdown – 24 hours
- Activity release period from pool – 2 hours
- Total number of fuel rods damaged – 115 in dropped and struck assemblies
- Radial peaking factor – 15
- Fuel rod fission product gap fractions –Regulatory Guide 1.183 (Reference 6), Regulatory Position C.3.2
- Iodine chemical form – Regulatory Guide 1.183, Regulatory Position C.3.5
- Pool decontamination for iodine – Regulatory Guide 1.183, Appendix B
- Delay to achieve design negative pressure in reactor building – 20 minutes
- Standby gas system filtration – 99 percent efficiency
- Dose conversion factors - Regulatory Guide 1.183, Regulatory Position 4.1

The radionuclide inventory in the damaged fuel is listed in Table 3.3-21. The GGNS ESP-site-specific doses were calculated using the χ/Q values given in Table 2.3-139 (EAB) and Table 2.3-140 (LPZ). The resulting doses at the EAB and LPZ are summarized in Table 3.3-22. The

LPZ dose is bounded by the EAB dose due to the 2-hour release duration and the lower χ/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 10 CFR 100 limits (e.g., 75 rem thyroid and 6.3 rem whole body).

3.3.5 References

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

TABLE 3.3-1

COMPARISON OF REACTOR TYPES FOR LIMITING OFF-SITE DOSE CONSEQUENCES

PART A, ALL PLANTS 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	0.79	2.5
Pre-existing Iodine Spike	AP1000	0.69	0.21	25
Reactor Coolant Pump Locked Rotor				
	AP1000	2.5	0.3	2.5
Control Rod Ejection Accident				
	AP1000	2.98	0.84	6.3
Steam Generator Tube Rupture				
Accident-initiated Iodine Spike	AP1000	1.49	0.12	2.5
Pre-existing Iodine Spike	AP1000	2.98	0.17	25
Small Line Break				
	AP1000	1.3	0.1	2.5
Loss of Coolant Accident				
	AP1000	24.5	4.94	25
	ACR-700	6.3	4.1	25
Fuel Handling Accident				
	AP1000	2.4	0.3	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).

PART B, ABWR PLANT

Accident	Affected Organ	EAB Dose (rem)	LPZ Dose (rem)	Guideline[1] (rem)
Main Steam Line Break				
Max Equilibrium Iodine Activity	Thyroid	1.11	1.24E-01	30
	Whole Body	1.7E-02	1.91E-03	2.5
Pre-existing Iodine Spike	Thyroid	22.2	2.48	300
	Whole Body	3.4E-01	3.81E-02	25
Control Rod Drop Accident	Thyroid	Negligible	Negligible	75
	Whole Body	Negligible	Negligible	6
Small Line Break	Thyroid	2.04	0.23	30
	Whole Body	0.027	0.003	2.5
Loss of Coolant Accident	Thyroid	82.5	200	300
	Whole Body	1.78	2.58	25
Fuel Handling Accident	Thyroid	9.78	1.10	75
	Whole Body	0.41	0.05	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 3.3-2

AP1000 MAIN STEAM LINE BREAK - ACCIDENT-INITIATED IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	0.79	----*
0 to 8 hour	----*	0.32
8 to 24 hour	----*	0.20
24 to 96 hour	----*	0.27
96 to 720 hours	----*	----
Total	0.79	0.79

NOTES:

*Dose not applicable

TABLE 3.3-3

AP1000 MAIN STEAM LINE BREAK - PRE-EXISTING IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	0.69	----*
0 to 8 hour	----*	0.12
8 to 24 hour	----*	0.04
24 to 96 hour	----*	0.06
96 to 720 hours	----*	----*
Total	0.69	0.22

NOTES:

*Dose not applicable

TABLE 3.3-4

ABWR MAIN STEAM LINE BREAK OUTSIDE CONTAINMENT

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

TABLE 3.3-5

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

Time	Exclusion Area Boundary Dose (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	----*	----*	1.24E-01	1.91E-03
8 to 24 hour	----*	----*	----*	----*
24 to 96 hour	----*	----*	----*	----*
96 to 720 hours	----*	----*	----*	----*
TOTAL	1.11	1.70E-02	1.24E-01	1.91E-03

NOTES:

*Dose not applicable

TABLE 3.3-6

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	----*	----*	2.48E+00	3.81E-02
8 to 24 hour	----*	----*	----*	----*
24 to 96 hour	----*	----*	----*	----*
96 to 720 hours	----*	----*	----*	----*
TOTAL	2.22E+01	3.4E-01	2.48E+00	3.81E-02

NOTES:

*Dose not applicable

TABLE 3.3-7

AP1000 LOCKED ROTOR ACCIDENT – PRE-EXISTING IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	2.5	----*
0 to 8 hour	----*	0.3
8 to 24 hour	----*	----*
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
Total	2.5	0.3

NOTES:

*Dose not applicable

TABLE 3.3-8

AP1000 CONTROL ROD EJECTION ACCIDENT - PRE-EXISTING IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	2.98	----*
0 to 8 hour	----*	0.69
8 to 24 hour	----*	0.12
24 to 96 hour	----*	0.02
96 to 720 hours	----*	0.01
Total	2.98	0.84

NOTES:

*Dose not applicable

TABLE 3.3-9

AP1000 STEAM GENERATOR TUBE RUPTURE - ACCIDENT-INITIATED IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	1.49	----*
0 to 8 hour	----*	0.09
8 to 24 hour	----*	0.03
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
Total	1.49	0.12

NOTES:

*Dose not applicable

TABLE 3.3-10

AP1000 STEAM GENERATOR TUBE RUPTURE - PRE-EXISTING IODINE SPIKE

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	2.98	----*
0 to 8 hour	----*	0.16
8 to 24 hour	----*	0.01
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
Total	2.98	0.17

NOTES:

*Dose not applicable

TABLE 3.3-11

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

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	1.3	----*
0 to 8 hour	----*	0.1
8 to 24 hour	----*	----*
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
Total	1.3	0.1

NOTES:

*Dose not applicable

TABLE 3.3-12

ABWR SMALL LINE BREAK OUTSIDE CONTAINMENT - ACTIVITY RELEASED TO ENVIRONMENT

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

TABLE 3.3-13

ABWR SMALL LINE BREAK OUTSIDE CONTAINMENT

Time	Exclusion Area Boundary Dose (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	----*	----*	2.29E-01	3.00E-03
8 to 24 hour	----*	----*	----*	----*
24 to 96 hour	----*	----*	----*	----*
96 to 720 hours	----*	----*	----*	----*
TOTAL	2.04	2.68E-02	2.29E-01	3.00E-03

NOTES:

*Dose not applicable

TABLE 3.3-14

AP1000 DESIGN BASIS LOSS OF COOLANT ACCIDENT

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour[1]	24.6	----*
0 to 8 hour	----*	4.54
8 to 24 hour	----*	0.16
24 to 96 hour	----*	0.13
96 to 720 hours	----*	0.11
Total	24.6	4.94

NOTES:

*Dose not applicable

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

TABLE 3.3-15

ABWR LOCA CURIES RELEASED TO ENVIRONMENT BY TIME INTERVAL

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

TABLE 3.3-16

ABWR DESIGN BASIS LOSS OF COOLANT ACCIDENT[1]

Time	Exclusion Area Boundary Dose		Low Population Zone Dose	
	Thyroid (rem)	Whole Body (rem)	Thyroid (rem)	Whole Body (rem)
0 to 2 hour	8.25E+01	1.78	----*	----*
0 to 8 hour	----*		1.33E+01	4.27E-01
8 to 24 hour	----*		9.93	3.97E-01
24 to 96 hour	----*		5.46E+01	7.60E-01
96 to 720 hours	----*		1.22E+02	9.97E-01
Total	82.5	1.78	2.00E+02	2.58

NOTES:

*Dose not applicable

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

TABLE 3.3-17

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

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

TABLE 3.3-18

ACR-700 LARGE LOSS OF COOLANT ACCIDENT

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	6.3	0.7
2 to 8 hour	----*	1.3
8 to 24 hour	----*	1.2
24 to 96 hour	----*	0.5
96 to 720 hours	----*	0.4
Total	6.3	4.1

NOTES:

*Dose not applicable

TABLE 3.3-19

AP1000 FUEL HANDLING ACCIDENT - CURIES RELEASED TO ENVIRONMENT

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

TABLE 3.3-20

AP1000 FUEL HANDLING ACCIDENT

Time	Exclusion Area Boundary Dose Total Effective Dose Equivalent (rem)	Low Population Zone Dose Total Effective Dose Equivalent (rem)
0 to 2 hour	2.4	----*
0 to 8 hour	----*	0.3
8 to 24 hour	----*	----*
24 to 96 hour	----*	----*
96 to 720 hours	----*	----*
Total	2.4	0.3

NOTES:

*Dose not applicable

TABLE 3.3-21

ABWR FUEL HANDLING ACCIDENT - CURIES RELEASED TO ENVIRONMENT

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

TABLE 3.3-22

ABWR FUEL HANDLING ACCIDENT

Time	Exclusion Area Boundary Dose (rem)		Low Population Zone Dose (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
0 to 2 hour	9.78	0.41	----*	----*
0 to 8 hour	----*	----*	1.10	0.05
8 to 24 hour	----*	----*	----*	----*
24 to 96 hour	----*	----*	----*	----*
96 to 720 hours	----*	----*	----*	----*
TOTAL	9.78	0.41	1.10	0.05

NOTES:

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

TABLE 3.3-23

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 3.3-24

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 3.3-25

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 3.3-26

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 3.3-27

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 3.3-28

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 3.3-28 (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 3.3-28 (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