15. ACCIDENT ANALYSES

15.1 Technical Information in the Application

In Chapter 15, "Accident Analyses," of the site safety analysis report (SSAR), the applicant analyzed and provided the radiological consequences of design-basis accidents (DBAs) to demonstrate that new nuclear units could be sited at the proposed early site permit (ESP) site without undue risk to the health and safety of the public, in compliance with the requirements of Title 10, Section 52.17, "Contents of Applications," of the *Code of Federal Regulations* (10 CFR 52.17) and 10 CFR Part 100, "Reactor Site Criteria." The applicant did not identify a particular reactor design to be considered for the proposed ESP site. Instead, the applicant developed a set of reactor DBA source term parameters using surrogate reactor characteristics. The applicant used these parameters in conjunction with site characteristics for accident analysis purposes to assess the suitability of the proposed ESP site. These plant parameters collectively constitute a plant parameter envelope (PPE).

The applicant developed a PPE using seven reactor designs, five water-cooled reactors and two gas-cooled reactors, though it used source terms for only two of these designs as inputs to its DBA analyses. The water-cooled reactors included in the PPE were (1) a version of the Westinghouse Advanced Plant 1000 (AP1000), (2) the certified General Electric Advanced Boiling Water Reactor (ABWR), (3) the Atomic Energy of Canada Advanced CANDU Reactor (ACR-700), (4) the General Electric Economic and Simple Boiling Water Reactor (ESBWR), and (5) the Westinghouse-led International Reactor Innovative and Secure (IRIS) reactor. The ACR-700 is light-water cooled, but heavy-water moderated. The two gas-cooled reactors were (1) the General Atomics Gas Turbine Modular Helium Reactor (GT-MHR) and (2) the Pebble Bed Modular Reactor (PBMR). The applicant stated that the PPE values are not intended to be limited to these reactor designs, but rather to provide a broad overall outline of a design concept and to include other potential reactor designs if they fall within the parameter values provided in the PPE.

In selecting DBAs for dose consequence analyses, the applicant focused on two light-water reactors (LWRs), the certified ABWR and a version of the AP1000,[1] to serve as surrogates. The applicant stated that it selected these two reactor designs because they are (or are based on) previously certified standard designs and have recognized bases for postulated accident analyses. Using source terms developed from these two designs, the applicant performed and provided radiological consequence analyses for the following DBAs:

- pressurized-water reactor (PWR) main steamline break
- PWR feedwater system pipe break
- locked rotor accident
- reactor coolant pump shaft break
- PWR rod ejection accident

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[1] As discussed later in this section, the applicant referenced a version of the AP1000 design available at the time the applicant submitted its ESP application. Westinghouse subsequently revised the AP1000 design prior to the NRC staff's issuance of a final SER for the AP1000 design certification.

- boiling-water reactor (BWR) control rod drop accident
- failure of small lines carrying primary coolant outside containment
- PWR steam generator tube failure
- BWR main steamline break
- PWR and BWR loss-of-coolant accidents
- fuel-handling accident

The applicant calculated site-specific DBA doses by first obtaining DBA dose information from the certified ABWR design control document (DCD) and from the proposed AP1000 DCD. (The reactor designers had obtained such values using assumed atmospheric dispersion factors [χ/Q values].) The applicant then calculated site-specific χ/Q values using onsite meteorological information. (The applicant provided the site-specific χ/Q values used in its radiological consequence analyses in Table 1.9-1, "Site Characteristics and Design Parameters," of the SSAR.) Finally, it multiplied the doses from the two designs by the ratio of the site-specific χ/Q values to the assumed χ/Q values from the DCDs.

The applicant presented the dose consequence assessment results in SSAR Section 15. Table 15.4-1, "Summary of Design Basis Accident Doses," provides a summary of the postulated radiological consequences of the DBAs identified above at the proposed exclusion area boundary (EAB) and the low population zone (LPZ). The table also demonstrates that any potential doses would be within the radiological dose consequence evaluation factors set forth in 10 CFR 50.34(a)(1), "Contents of Applications; Technical Information." The applicant provided the accident-specific source terms (release rates of radioactive materials from the PPE [ESP footprint] to the environment) and resulting site-specific dose consequences for each DBA in Tables 15.4-2 through 15.4-27 of the SSAR.

In request for information (RAI) 15.4-1, the staff noted that Westinghouse has revised its χ/Q values in the AP1000 DCD since the applicant had submitted the North Anna ESP application, and asked whether the applicant planned to use the updated values in revising its application. The applicant responded that it elected not to update the ESP application to incorporate the latest χ/Q values in the AP1000 design certification. It went on to state that site-specific doses would be updated, as necessary, in the combined license (COL) application, after a specific reactor design is selected.

In RAIs 15.4-2 and 15.4-3, the staff noted that SSAR Section 15.4 provides total effective dose equivalent (TEDE) values for the ABWR design, while the ABWR design is certified with the thyroid and whole body doses specified in 10 CFR Part 100. The staff asked the applicant to explain how the doses compare. In its response, the applicant stated that the SSAR would be revised to include the thyroid and whole body doses from the ABWR DCD, in addition to the estimated TEDE values. The applicant provided this information in Revision 3 of its application.

In RAI 15.4-4, the staff asked the applicant to provide references and explain the methodology it used to determine time-dependent activity releases for each DBA. The applicant provided the requested documents. The applicant stated in its response that the methodologies used for calculating time-dependent releases for the ABWR and AP1000 are presented in the respective design certification documents. The staff finds the methodologies used in the respective design certification documents to be acceptable.

In RAI 15.4-5, the staff asked the applicant to provide, for each DBA, the doses it used for the EAB and the LPZ for the AP1000 and the ABWR, as well as the ratios of site-specific χ/Q values to design certification χ/Qs used. In its response, the applicant stated that it would revise the dose tables in SSAR Section 15.4 to show the χ/Q values and doses from the AP1000 and ABWR DCDs, in addition to the ratios of site-specific χ/Q values to design certification χ/Q values.

In RAI 15.4-6, the staff asked the applicant to clarify whether the 0- to 2-hour EAB doses presented in the SSAR are for the 2-hour period with the greatest EAB doses, and, if they are not, to provide the doses for the 2-hour period with the greatest EAB doses. In its response, the applicant stated that the greatest EAB dose occurs during the first 2 hours of the accident for all AP1000 accidents evaluated in SSAR Chapter 15**,** except for a loss-of-coolant accident (LOCA). As indicated in Section 15.6.5.3.8.1 of the AP1000 DCD, the period from 1 to 3 hours yields the greatest EAB dose for a LOCA. The applicant incorporated this information into Revision 3 of its application.

15.2 Regulatory Evaluation

In SSAR Section 1.8, "Conformance to NRC Regulations and Regulatory Guidance," and in SSAR Section 15, the applicant identified the following applicable U.S. Nuclear Regulatory Commission (NRC) regulations and guidance regarding reactor accident radiological consequence analyses:

- 10 CFR 52.17
- 10 CFR Part 100
- 10 CFR 50.34
- Regulatory Guide (RG) 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," issued June 1974
- RG 1.25, "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," issued March 1972
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," issued November 1982
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued July 2000
- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," issued July 1981
- TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," issued March 1962

The staff finds that the applicant correctly identified the applicable regulations and guidance. In its evaluation, the staff used the dose consequence evaluation factors found in 10 CFR 50.34(a)(1) that are a factor in determining the acceptability of the site in accordance with 10 CFR 52.17(a)(1).

The regulations at 10 CFR 52.17(a)(1) require that ESP applications contain an analysis and evaluation of the major structures, systems, and components of the facility that bear significantly on the acceptability of the site under the radiological consequence evaluation factors identified in 10 CFR 50.34(a)(1). In addition, the ESP site characteristics must comply with the requirements of 10 CFR Part 100. The regulations at 10 CFR 50.34(a)(1)(ii)(D) require that for a postulated fission product release based on a major accident:

- An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release would not receive a radiation dose in excess of 25 rem TEDE.

- An individual who is located at any point on the boundary of the LPZ and who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE.

Because the applicant has not selected a reactor design to be constructed on the proposed ESP site, the applicant used a PPE approach to demonstrate that it meets these requirements. A PPE is a set of plant design parameters that are expected to bound the characteristics of a reactor(s) that may be constructed at a site, and it serves as a surrogate for actual reactor design information. As discussed in NRC Review Standard (RS)-002, and in Chapter 1 of this SER, the staff considers the PPE approach to be an acceptable method for assessing site suitability. For the purposes of this analysis, the applicant proposed a fission product release from the PPE (ESP footprint) to the environment, and the staff reviewed the applicant's dose evaluation based on this release.

15.3 Technical Evaluation

The applicant evaluated the suitability of the site under the radiological consequence evaluation factors identified in 10 CFR 50.34(a)(1) using bounding reactor accident source terms and dose consequences as a set of PPE values based on two surrogate designs, as well as site-specific χ/Q values based on the ESP footprint. The following paragraphs describe the staff's review of each aspect of this evaluation.

15.3.1 Selection of DBAs

The applicant selected the DBAs listed in Section 15.1 of this SER, above, based on the proposed AP1000 reactor design and the certified ABWR reactor design, indicating that it chose these two reactor designs because they have (or are based on) previously certified standard designs and have recognized bases for postulated accident analyses. The staff finds that the applicant selected DBAs which are consistent with the DBAs listed and analyzed in NUREG-0800 and RG 1.183. Therefore, the staff finds that the applicant provided an

acceptable DBA selection for evaluating the compliance of the proposed ESP site with the dose consequence evaluation factors specified in 10 CFR 50.34(a)(1). The applicant stated that, because of their greater potential for inherent safety, the DBAs of the other reactors being considered for the proposed ESP site are expected to be bounded by those DBAs analyzed in the proposed AP1000 and certified ABWR DCDs. While the staff has not reviewed these designs in detail, the staff believes that conclusions drawn regarding the site's acceptability based on the AP1000 and ABWR designs are likely to be valid for the other reactor designs the applicant is considering. Whether or not such designs are in fact bounded by the analyses considered here would be subject to review during the staff's consideration of any COL or CP application that might be filed with respect to construction and operation of a reactor design at the North Anna ESP site.

15.3.2 Design-Specific (Assumed) χ/Q Values

To support its accident analyses based on the ABWR as a surrogate design, the applicant used the assumed χ/Q values in the certified ABWR DCD. In evaluating the AP1000, the applicant used those χ/Q values in the proposed AP1000 DCD that were under review by the staff at the time the North Anna ESP application was submitted. Westinghouse subsequently revised the χ/Q values in the AP1000 DCD. Consequently, the assumed χ/Q values and the calculated design-specific doses used in the North Anna ESP application may differ from those associated with a certified AP1000 DCD. However, the staff determined that the PPE values for the assumed χ/Q values associated with the AP1000 design used by the applicant in its accident analyses are reasonable and, therefore, that they are adequate for the purpose of demonstrating that a reactor with design characteristics similar to an AP1000 could be sited at the proposed ESP site. Section 15.4 of the SSAR lists the χ/Q values the applicant used for the version of the AP1000 and the certified ABWR that it considered.

The applicant also listed in Table 1.3-1 of the SSAR a set of design-specific assumed χ/Q values, some of which neither the applicant nor the staff used in their radiological consequence evaluations. The staff finds that the χ/Q values in Table 1.3-1, with the exception of those used in the applicant's dose assessments in Section 15 of the SSAR, are not needed to assess the suitability of the proposed site. Therefore, the staff did not review them.

15.3.3 Site-Specific χ/Qs

The staff reviewed the applicant's site-specific χ/Q values and performed an independent evaluation of atmospheric dispersion in accordance with the guidance provided in Section 2.3.4 of RS-002. The staff finds the χ/Q values to be acceptable, as described in Section 2.3.4 of this SER. The staff intends to include these site-specific χ/Qs in any ESP that the NRC may issue for the North Anna ESP site.

15.3.4 Source Terms and Radiological Consequence Evaluations

To evaluate the suitability of the site using the radiological consequence evaluation factors in 10 CFR 50.34(a)(1), the applicant provided the bounding reactor accident source terms as a set of PPE values based on (1) the surrogate AP1000 and 4300 megawatt thermal (MWt) ABWR designs (as explained below), and (2) the site-specific χ/Qs based on the ESP footprint. The source terms are expressed as the timing and release rate of fission products to the

environment from the proposed ESP site. The dose consequences are then derived from the source terms using established methods.

The AP1000 source terms are based on the guidance provided in RG 1.183. The methodologies and assumptions used by Westinghouse, the AP1000 vendor, in its radiological consequence analyses are consistent with the guidance provided in RG 1.183. The resulting doses calculated for the AP1000 design using assumed site parameters meet the dose consequence evaluation factors specified in 10 CFR 50.34(a)(1), i.e., 25 rem TEDE. The methodologies and assumptions used by General Electric, the ABWR vendor, in its radiological consequence analyses for the ABWR design are consistent with the guidance provided in RGs 1.3 and 1.25. The ABWR source terms are based on the guidance in TID-14844. The resulting doses for the ABWR reactor design using assumed site parameters meet the dose consequence evaluation factors specified in 10 CFR 100.11—300 rem to the thyroid and 25 rem to the whole body. While the requirements of 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," are not applicable to ESPs, the staff notes that the final rule at Appendix A, "Design Certification Rule for the U.S. Advanced Boiling Water Reactor," to 10 CFR Part 52 states the following:

> The Commission has determined that with regard to the revised design basis accident radiation dose acceptance criteria in 10 CFR 50.34, the ABWR design meets the new dose criteria, based on the NRC staff's radiological consequence analyses, provided that the site parameters are not revised.

Therefore, the staff concludes that the certified ABWR design, in conjunction with assumed site parameters, meets the dose consequence evaluation factors specified in 10 CFR 100.11, as well as those specified in 10 CFR 50.34(a)(1).

In its site-specific DBA radiological consequence analyses, the applicant scaled the ABWR source terms and the resulting doses from the 10 CFR Part 52, Appendix A-certified power level of 4005 MWt to 4300 MWt for its version of the ABWR. The applicant used a linear scaling method. Because the fission product release rate is directly proportional to the fission product inventory if mitigating processes remain the same, and because the fission product inventory is directly proportional to reactor power, the staff finds this scaling methodology to be acceptable for the purposes of this evaluation.

The staff has verified the design-specific source terms the applicant provided and finds them to be consistent with those evaluated by the staff as part of the design certification reviews. Further, the staff finds that the references provided by the applicant and the methodology it used to determine timing and release rate of fission product source terms to the environment (and consequent dose consequences) from the proposed ESP site are acceptable. Therefore, the staff finds the source terms from the PPE (ESP footprint) themselves to be reasonable and acceptable. The staff intends to include the source terms in any ESP that the NRC might issue for the North Anna ESP site.

In response to RAI 15.4-6, the applicant stated that the greatest EAB dose occurs during the first 2 hours of the DBAs, except for the AP1000 LOCA. In view of the accident progression sequences for the designs used in the DBA dose assessment, the staff agrees with this conclusion.

Based on its evaluation of the applicant's analysis methodology and inputs to that analysis, the staff finds that the applicant's conclusion that the dose consequences for the chosen surrogate designs comply with the dose consequence evaluation factors of 10 CFR 50.34(a)(1) is correct.

In accordance with 10 CFR 52.79(a)(1), at the COL stage, the staff will evaluate whether the design of the facility falls within the parameters specified in an ESP, should one be issued for the North Anna ESP site. Should the COL applicant reference a certified design as well as the ESP, and should the source term and postulated χ/Qs for the chosen design fall within the PPE source term and the site characteristic χ/Q values specified in the ESP, the staff will conclude that the COL applicant has satisfied this requirement. Should the COL applicant reference the ESP but not a certified design, the staff will evaluate the source term for the chosen design and will use that source term and the site χ/Qs determined at the ESP stage to determine whether the applicable regulations in 10 CFR 50.34 regarding dose consequence evaluation factors have been met. In the event of the filing of a CP referencing the ESP, the staff would evaluate the design's source terms and use the site χ/Qs from the ESP to determine compliance with 10 CFR 50.34.

The staff has identified the following site χ/Q values as appropriate for inclusion in any ESP that the staff might issue for the North Anna ESP site.

Table 15.3-1 Site-Specific χ/Q Values

Location and Time Interval	χ/Q Value
0 to 2 hour EAB	2.26E-4 sec/m^3
0 to 8 hour LPZ	2.05E-5 sec/m^3
8 to 24 hour LPZ	1.36E-5 sec/m^3
1 to 4 day LPZ	5.58E-6 sec/m^3
4 to 30 day LPZ	1.55E-6 sec/m^3

RS-002 calls for the staff to perform a confirmatory radiological consequence calculation. However, the design-related inputs to applicant's dose calculation were directly extracted from design documentation previously submitted to and reviewed by the NRC in connection with design certification applications. Because the applicant simply multiplied these inputs by the ratio of the site χ/Q values to the assumed design χ/Q values, the staff did not consider an independent calculation to be useful or necessary, and therefore did not perform one.

15.4 Conclusions

As set forth above, the applicant submitted its radiological consequence analyses using the site-specific χ/Q values and PPE source term values, and concluded that the proposed site meets the radiological consequence evaluation factors identified in 10 CFR 50.34(a)(1).

Based on the reasons set forth above, the staff finds that the applicant's PPE values for source terms that it included as inputs to the radiological consequence analyses are reasonable.

Further, based on the above, the staff finds that the applicant's site-specific χ/Q values and dose consequence evaluation methodology are acceptable.

Therefore, the staff concludes that the proposed distances to the EAB and the LPZ outer boundary of the proposed ESP site, in conjunction with the fission product release rates to the environment provided by the applicant as PPE values, are adequate to provide reasonable assurance that the radiological consequences of the DBAs will be within the dose consequence evaluation factors set forth at 10 CFR 50.34(a)(1) for the proposed ESP site. This conclusion is subject to confirmation at the COL or CP stage that the design of the facility specified by the COL or CP applicant falls within the ESP PPE values.

The staff further concludes that (1) the applicant has demonstrated that the proposed ESP site is suitable for power reactors with source term characteristics bounded by those of the 4300 MWt ABWR and AP1000 without undue risk to the health and safety of the public and (2) the applicant complies with the requirements of 10 CFR 52.17 and 10 CFR Part 100.