15 ACCIDENT ANALYSIS

15.1 Accident Analysis

15.1.1 Introduction

This section of the safety evaluation report describes the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the information provided in Chapter 15, "Accident Analysis," of the Site Safety Analysis Report (SSAR), contained in Part 2 of the Clinch River Nuclear (CRN) Site early site permit (ESP) application. The information in Chapter 15 describes the radiological consequences of design basis accidents (DBAs) using information based on four conceptual small modular reactor (SMR) designs under consideration for the site as included in the plant parameter envelope (PPE) to demonstrate that two or more new SMR unit(s) with a maximum rated thermal power for a single unit of 800 thermal megawatts (MWt) could be sited at the proposed ESP Site without posing an undue risk to the health and safety of the public, in compliance with the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR), 52.17, "Contents of Applications," and 10 CFR Part 100, "Reactor Site Criteria."

15.1.2 Summary of Application

In SSAR Section 1.11, "Overview of Reactor Types," the four conceptual SMRs considered in the development of a surrogate plant for the PPE are described. Each of the designs is a pressurized water reactor, with three of the designs being of an integral pressurized water reactor type. All four designs are described as passively safe designs with minimal or no reliance on offsite power, offsite water, or operator action for safety. The four designs and associated vendors are:

- BWXT mPower™ (Generation mPower, LLC);
- NuScale (NuScale Power, LLC);
- SMR-160 (Holtec SMR, LLC); and
- Westinghouse SMR (Westinghouse Electric Company, LLC).

SSAR Chapter 15 describes the applicant's assessment of the offsite radiological consequences of DBAs for a surrogate plant, as bounded by the PPE. The U.S. Nuclear Energy Institute 10-01, "Industry Guidance for Developing a Plant Parameter Envelope in Support of an Early Site Permit," notes that accident analyses model the time-dependent transport of radionuclides out of the reactor core through several pathways, each with different time-dependent removal mechanisms for radionuclides. As the applicant notes in SSAR Section 15.1, "Accident Selection," different reactor designs have different release pathways, and each pathway has different release rates and different radionuclide removal mechanisms. Given these differences, the applicant chose to use the DBA radiological consequence analyses from the design that resulted in the highest post-accident offsite doses in its assessment of the radiological consequences of DBAs at the CRN Site.

The applicant's DBA radiological consequence analysis used, as input, the site characteristic short-term accident atmospheric dispersion factors (χ/Qs) at the Exclusion Area Boundary (EAB) and Low-Population Zone (LPZ) provided in SSAR Section 2.3.4. In SSAR Table 2.0-3, the applicant provided the bounding DBA source term (release rates of radioactive materials to the environment). The applicant also presented the DBA dose assessment results at the proposed EAB and the LPZ in SSAR Table 15-1, which demonstrates that the potential doses

would be within the radiological consequence evaluation factors set forth in 10 CFR 50.34(a)(1) and 10 CFR 52.17(a)(1).

Because the reactor design technology is not selected and the orientations of plant structures on the site are not known, the detailed accident analyses and resulting post-accident doses for control habitability and the Technical Support Center will be performed at the combined license application (COLA) stage, consistent with SRP Guidance, Chapter 2.0, Table 1.

15.1.3 Regulatory Basis

The applicable NRC regulatory requirements for the radiological dose consequences analyses of DBAs include the following:

- 10 CFR 52.17, "Contents of applications; technical information," as it relates to the assessment that must contain 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 paragraphs (a)(1)(ix)(A) and (a)(1)(ix)(B) of this section;

- 10 CFR Part 100, "Reactor Site Criteria," as it relates to considering evaluation factors for stationary power reactor Site Applications on or after January 10, 1997, to demonstrate that the radiological dose consequences of postulated accidents shall meet the criteria set forth in 10 CFR 50.34(a)(1) for type of facility proposed to be located at the CRN Site; and

- 10 CFR 50.34, "Contents of applications; technical information," as it relates to a description and safety assessment of the site and safety assessment of facility.

The acceptance criteria adequate to meet the above requirements are located in the following guidance and reference documents:

- Review Standard (RS)-002, "Guidance for Processing Applications for Early Site Permits," as it relates to providing guidance on the staff's process for reviewing an ESP application and developing the Safety Evaluation Report (SER) with specific technical and format guidance; and

- NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light-Water Reactor (LWR) Edition," as it relates to providing guidance to staff to perform safety reviews of applications to construct or operate nuclear power plants and the review of applications to approve standard designs and sites for nuclear power plants, to assure the quality and uniformity of staff safety review.

As required in 10 CFR 52.17(a)(1), ESP applications must contain an analysis and evaluation of the major systems, structures, and components (SSCs) of the facility that bear significantly on the acceptability of the site under the radiological consequence evaluation factors identified in the requirements of 10 CFR 52.17(a)(1)(ix). In addition, the ESP site characteristics must comply with the requirements of 10 CFR 100.21, "Non-Seismic Siting Criteria," which states that radiological dose consequences of postulated accidents shall meet the criteria set forth in 10 CFR 50.34(a)(1). The radiological dose reference values in 10 CFR 50.34(a)(1) and 10 CFR 52.17(a)(1) for a postulated fission product release based on a major accident are as follows:

- 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 roentgen equivalent man (rem) total effective dose equivalent (TEDE); and

- An individual located at any point on the outer boundary of the LPZ 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.

15.1.4 Technical Evaluation

Using the guidance listed above, the staff reviewed SSAR Chapter 15 for compliance with the applicable regulations. Although the applicant is using the PPE approach, for the DBA radiological consequence analysis with the loss of coolant accident (LOCA) source term is selected based upon vendor information, and presents the design with the highest resulting dose at the EAB and LPZ from four SMR designs under consideration. The applicant evaluated the suitability of the site using this bounding LOCA DBA source terms and radiological consequences, as well as site characteristic atmospheric dispersion factor (χ/Q) values.

15.1.4.1 Selection of Design Basis Accidents and Source Term

The applicant assessed the LOCA DBA in the design control document for an SMR with vendor provided PPE source terms. This DBA is addressed in SSAR Sections 15.1 and 15.2. The staff independently analyzed the information provided by the applicant and finds that the applicant selected DBA is consistent with the DBAs listed in NUREG-0800, Chapter 15 for large light-water reactors.

Each of the four small modular PWR designs under consideration for the CRN Site is expected to include advanced features that would further minimize accident consequences, as addressed in SSAR Section 1.11. As such, TVA anticipates that the consequences of a LOCA will be less than those for large PWR designs and that no events of greater consequence will be identified.

Thus, the analysis of postulated DBAs other than a LOCA is not necessary for this ESP, because the maximum potential offsite doses have been evaluated, demonstrating the ability of the site to comply with the dose limits in 10 CFR 52.17. In accordance with 10 CFR 52.79(b)(1), the COLA will verify that the accident doses provided in the ESPA are bounding or will provide an evaluation of accident radiological consequences. The staff finds this applicant's approach reasonable and acceptable.

The staff's experience with other PWR designs, as documented in other ESPAs to date (examples provided by applicant in SSAR Section 15.1), has shown that offsite doses due to a postulated LOCA are expected to more closely approach 10 CFR 52.17 dose criteria than other DBAs that may have greater probability of occurrence but a lesser magnitude of activity release.

The LOCA source term (radionuclide activity released to the environment) selected for inclusion in the PPE is based upon vendor input and represents the design with the highest resulting dose at the EAB and the LPZ boundary from the four SMR designs under consideration. Key parameters associated with the accident source term in the PPE have been evaluated by the applicant to assess their reasonableness for and representativeness of SMR designs.

The PPE LOCA source term is based on a design that uses standard light-water reactor fuel, which is representative of the SMR designs under consideration, and assumes a core power

level for a single module at 800 MWt. As a comparison, for the other three SMR designs considered in the PPE, the values for the rated core thermal power per single unit are 160, 525 and 530 MWt. TVA anticipates that comparable methodologies and techniques that are used for the development of the source terms for large light water reactors will be used also in the development of the SMR accident source terms to be presented by SMR design documents.

The source terms developed for the surrogate SMR plant (the design parameters represented by the PPE in lieu of a specific reactor design) are representative of the potential SMR designs considering core power and average burnup. The maximum average burnup for the surrogate SMR is 51 gigawatt-day/metric-ton of Uranium (GWD/MTU), while for remaining SMRs is 41 GMD/MTU. Although it is recognized that core power and burnup would not necessarily result in one-to-one ratios to activity releases, it is anticipated the larger core power and burnup would result in larger activity releases than those associated with the remaining designs. The bounding design basis accident LOCA source term is provided in SSAR Table 2.0-3.

To assess reasonableness, the applicant also provided a comparison of the PPE LOCA source term to that of the AP1000 design, scaling the source term by a factor 0.235 (800 MWt/3400 MWt) to account for the smaller core thermal power of the SMR designs being considered for the CRN Site. The worst 2-hour EAB dose is approximately 25 percent greater for the scaled AP1000 design than that for the surrogate plant (as provided in the PPE). The applicant considers that this difference is reasonable given that SMR designs contain additional safety features that are expected to result in enhanced safety features and reduced accident releases as compared to the AP1000 design. The applicant acknowledged approximately 25 percent greater total activity release for the scaled down-AP1000 source term than that for the surrogate plant (as provided in the PPE) for the worst 2-hour period. However, an independent staff evaluation resulted in higher activity release using same 0.235 scale down ratio for all radionuclides for all time periods except for noble gases. This higher release may be attributed to the staff-determined source term that is reduced from the known large LWR release source term by a megawatt ratio that may have not been exactly representative due to unaccounted for fuel and core design differences between the large (AP1000 core) and small (SMR core) fission product inventories. In addition, the bounding SMR source terms used as the basis for the PPE LOCA source term may reflect SMR design enhanced removal mechanisms and advanced engineering features for larger retention times that are not accounted for in the assumption that the accident release source terms from a large LWR can be reduced in direct relationship to the reduction of the core power. Therefore, the staff's assessment potentially overestimates the potential source term for any specific SMR rated at a core power of 800 MWt. However, this evaluation only ensures that the applicant's PPE LOCA source term is representative and not unreasonable. Moreover, the LOCA source term for the selected SMR for the COLA, the applicant is required (10 CFR 52.79 (a)(1)) to demonstrate that the selected SMR LOCA source term is bounded by the PPE LOCA source term. If not the applicant shall opt for variance and demonstrate NRC regulatory compliance with radiological dose criteria for site suitability. As such, the staff finds the applicant's bounding source term based on the ratio is not unreasonable, and the bounding vendor provided SMR source term is also not unreasonable.

Therefore, based on its evaluation of the applicant's information and its own independent analysis to scope the PPE source term, the staff finds the CRN Site SSAR DBA source terms to be not unreasonable as part of the PPE for showing compliance with requirements of 10 CFR 52.17(a)(1)(ix). However, it should be noted that if the bounding DBA source term provided by the applicant does not bound the actual source term for a design selected for the CRN Site, then the radiological consequences of DBAs for the combined license (COL) may exceed the dose results determined in the SSAR. They may also potentially exceed the

regulatory dose reference values in 10 CFR 50.34(a)(1) and 10 CFR 52.79(a)(1). Therefore, the applicant's PPE source term information in SSAR Table 2.0-3 shall be compared with the COL DBA source term for the design that may be selected for the CRN Site at the time of COL review in accordance with 10 CFR 52.79(b)(1). Also, the applicant shall ensure that the radionuclide releases of the PPE bounds the SMR design selected for the CRN Site in meeting the regulatory requirements of 10 CFR 52.79(c)(1) and (d)(1).

15.1.4.2 Site Characteristic Short-Term Atmospheric Dispersion Factors

Site characteristic short-term (accident) atmospheric dispersion factors (χ/Qs) are used in the radiological consequences analyses to characterize the effect of the site-specific meteorological conditions, topography, and distance to either EAB or LPZ on dose at the offsite receptors for purposes of siting. The applicant calculated accident χ/Qs using RG 1.145 methodology and site-specific meteorological data. The staff's evaluation of the site characteristic short term χ/Q values is described in Section 2.3.4 of this report. The site characteristic accident χ/Q values calculated by the applicant are given in SSAR Table 2.0-1.

15.1.4.3 Radiological Consequences

Doses for LOCA are evaluated at the EAB and LPZ boundary using site characteristic short-term (accident) χ/Q values for the CRN Site. Site-specific dose results were calculated by the applicant by adjusting the vendor-provided dose results for each time period by the ratio of the site-characteristic χ/Q values to the vendor-provided χ/Q values, then adding the dose results for each time period together to get a resulting total dose. The CRN Site site-specific doses are presented in SSAR Table 15-1.

For the LOCA radiological consequence analysis referenced by the CRN Site ESP applicant, the vendor's analyses used the design-specific source term assumptions and inputs and reference site parameter values for the accident χ/Qs in lieu of site-specific values. The χ/Q values are the only input to the DBA radiological consequences analysis that are affected by the site characteristics. The estimated DBA dose calculated for a particular site is affected by the site characteristics through the calculated χ/Q input to the analysis; therefore, the resulting dose would be different than that calculated generically by the vendor with assumed reference χ/Qs. Smaller χ/Q values are associated with greater dilution capability, resulting in lower radiological doses. The applicant also stated that all other inputs and assumptions in the radiological consequences analysis remain the same as in the vendor-provided analysis.

To determine the potential doses resulting from DBA at the proposed site, the applicant used the site characteristic χ/Q values in conjunction with the DBA doses calculated using site parameter χ/Q values (based on PPE) that were provided by the vendor for the plant design used in the bounding analysis. The estimated site characteristic χ/Q values for the proposed site are higher for EAB and lower for LPZ than the corresponding site parameter χ/Q values, as summarized in Table 15-1 of this report.

Table 15-1 Site Parameter Short Term χ/Q Values for Vendor Deign Site Parameter and Comparison to Site Characteristic χ/Qs

Location	Release Time (hr)	Site Characteristic χ/Q(sec/m^3)	Vendor Design Site Parameter χ/Q(sec/m^3)	χ/Q Ratio Characteristic/Parameter	Dose(rem TEDE) vendor	Dose(rem TEDE) Site
EAB	0-2	4.96×10^{-3}	1.0×10^{-3}	4.96	4.35	21.6
LPZ	0-8	3.10×10^{-4}	5.0×10^{-4}	0.620	4.44	2.75
	8-24	2.26×10^{-4}	3.0×10^{-4}	0.753	0.20	0.15
	24-96	1.14×10^{-4}	1.5×10^{-4}	0.76	0.05	0.038
	96-720	4.30×10^{-5}	8.0×10^{-5}	0.538	0.06	0.032
LPZ_{total}	-	-	-	-	-	2.97

The radiological consequence results of a LOCA at the CRN Site, using the PPE and site characteristic accident χ/Q values are 21.6 rem TEDE at the EAB and 2.97 rem TEDE total at the LPZ. The calculated radiological consequences at the proposed site are within the regulatory dose criteria of 25 rem TEDE for the maximum 2-hour period at the EAB and 25 rem TEDE at the outer boundary of the LPZ for the duration of the accident release,

Based on its evaluation of the applicant's DBA radiological consequences analysis methodology and the inputs to that analysis, the staff finds that the applicant correctly concluded that the radiological consequences for the considered PPE design technology comply with the radiological dose reference values set forth in 10 CFR 50.34(a)(1) and 10 CFR 52.17(a)(1).

15.1.5 Conclusion

As set forth above, the applicant presented the radiological consequence analysis using PPE values of source terms for the standard design and site characteristic χ/Q values; the applicant concluded that the proposed site meets the radiological dose reference values identified in 10 CFR 50.34(a)(1) and 10 CFR 52.17(a)(1) for the vendor provided PPE source terms and site parameter χ/Q values. Based on the technical evaluation presented in Section 15.1.4 of this report, the staff finds that the applicant's PPE values for source terms are not unreasonable. Furthermore, the staff finds the applicant's dose consequence evaluation methodology acceptable. In accordance with 10 CFR 52.79(b)(1), a COL applicant referencing this ESP must either include or incorporate by reference the ESP SSAR, and the COLA must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the design of the facility falls within the site characteristics and design parameters specified with respect to radionuclide releases and site characteristics provided in the ESP.

The staff further concludes that the applicant's determined site characteristic distances to the EAB and the LPZ outer (i.e., outermost) boundary of the proposed ESP site in SSAR Table 2.0-1, in conjunction with the PPE design parameter source terms, are adequate to

provide reasonable assurance that the radiological consequences of postulated DBA for a SMR design similar to those used as a basis for the PPE will be within the radiological dose reference values set forth in 10 CFR 50.34(a)(1) and 10 CFR 52.17(a)(1).