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CHAPTER 15 ACCIDENT ANALYSIS

15.1 ACCIDENT SELECTION

The evaluation of nuclear power plant safety includes analysis of the plant response to a spectrum of postulated disturbances in process variables and postulated equipment failures. However, it is neither practical nor necessary to analyze all historically postulated design basis accidents (DBAs) associated with the small modular reactor (SMR) types under consideration for the Clinch River Nuclear (CRN) Site in the early site permit application (ESPA), as discussed below.

As noted in NEI 10-01, *Industry Guidance for Developing a Plant Parameter Envelope in Support of an Early Site Permit* (Reference 15-1), 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. Different reactor designs have different release pathways, and each pathway has different release rates and different radionuclide removal mechanisms. Given these differences, it is not possible to develop a bounding analysis for use in a plant-parameter-envelope-based ESPA, and accordingly, for the purposes of evaluating offsite post-accident doses, the vendor analysis with the highest resultant post-accident dose was selected for use in the CRN Site-specific dose analysis presented here.

At this time, the site layout and building configuration for each proposed reactor design for the CRN Site has yet to be determined, making it impractical to model near-field atmospheric dispersion around buildings in order to determine doses in the main control room and other areas where habitability is required post-accident. Thus, these types of detailed accident analyses are more appropriately performed at the Combined License Application (COLA) stage, when a technology is selected and the orientation of the plant on the site is known.

Experience with other pressurized water reactor (PWR) designs, as documented in ESPAs to date, has shown that offsite doses due to a postulated loss of coolant accident (LOCA) are expected to more closely approach 10 CFR 52.17 limits than other DBAs that may have a greater probability of occurrence but a lesser magnitude of activity release, as evidenced by the following:

- Clinton ESP Site, ESPA, Site Safety Analysis Report (SSAR), Table 3.3-2 (Reference 15-2)
- Grand Gulf ESP Site, ESPA, SSAR, Table 3.3-1 (Reference 15-3)
- North Anna ESP Site, ESPA, SSAR, Table 15.4-1 (Reference 15-4)
- PSEG Site, ESPA, SSAR, Tables 15.4-2, 15.4-10 and 15.4-19 (Reference 15-5)
- Victoria County Station, ESPA, SSAR, Table 15.1-5 (Reference 15-6)
- Vogtle ESP Site, ESPA, SSAR, Table 15-12 (Reference 15-7)

Each of the four small modular PWR designs under consideration for the CRN Site is expected to include advanced design features that would further minimize accident consequences (see Section 1.11). In particular, based on initial design feedback, 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, analysis of postulated DBAs other than a LOCA is not necessary for the ESPA, 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. The COLA verifies that the accident doses provided in this ESPA are bounded or provides an evaluation of accident radiological consequences.

15.2 SOURCE TERM

The bounding design basis accident (LOCA) source term is provided in Table 2.0-3.

The LOCA source term (radionuclide activity released to the environment) selected for inclusion in the plant parameter envelope (PPE) is based upon vendor input and represents the design with the highest resulting doses at the exclusion area boundary (EAB) and the low population zone (LPZ) boundary from the four SMR designs under consideration. Key input parameters associated with the accident source term in the PPE have been evaluated 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 unit at 800 MW thermal. The methodology and analytical techniques used for development of the source term are similar to those used for large light water reactors, and TVA anticipates that comparable methodologies and techniques will be used in the development of the SMR accident source terms to be presented in the SMR design control documents.

To assess reasonableness, a comparison of the PPE LOCA source term to that of the AP1000 design (as provided in the Vogtle 3 and 4 ESPA, Reference 15-7) was performed, scaling the source term presented in the Vogtle ESPA by a factor of 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 activity release associated with the worst 2-hour time period of the scaled-down AP1000 is approximately 25 percent greater than that for the surrogate plant (as provided in the PPE). This difference is reasonable given that SMR designs contain additional safety features that will result in general improvements over the AP1000 design. The activity release for the 30-day duration of the LOCA is approximately equivalent to that of the surrogate plant and is also considered reasonable.

The source terms developed for the surrogate plant are representative of the potential SMR designs considering core power and average burnup. The surrogate plant assumes a core power that is bounding but representative of the remaining SMR designs being considered. Core burnup was also reviewed. The maximum average burnup assumed for the surrogate plant is 51 GWD/MTU, while the maximum average burnup for the remaining SMR designs is less than 41 GWD/MTU. Although it is recognized that core power and burnup do 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 SMR designs.

15.3 EVALUATION METHODOLOGY AND CONCLUSIONS

Doses for a LOCA are evaluated at the EAB and LPZ boundary.

The evaluation uses the following parameters, as shown in Table 15-1:

- Short-term 95th percentile accident atmospheric dispersion factors (X/Qs) for the CRN Site.
- Bounding vendor-provided LOCA doses.
- X/Q values associated with the bounding vendor-provided LOCA doses.

Doses are calculated based on the amount of activity released to the environment, the dispersion of activity during transport to the receptor (X/Q), the breathing rate at the receptor, and the applicable dose conversion factors. The only parameters that are site-specific are the X/Qs. Hence, it is reasonable to adjust the vendor LOCA doses for site-specific X/Qs values.

For a given time step, the vendor dose is multiplied by the ratio of the site-specific X/Q to the vendor X/Q, as shown in the following equation:

$$Dose_{Site} = Dose_{Vendor}\,[(X/Q)_{Site}/(X/Q)_{Vendor}] \qquad \text{Equation 15-1}$$

The resulting accident doses are expressed as total effective dose equivalent (TEDE), consistent with 10 CFR 52.17. All site LOCA doses meet the 25 rem TEDE limit specified in 10 CFR 52.17 as shown in Table 15-1.

15.4 REFERENCES

15-1. NEI 10-01, *Industry Guideline for Developing a Plant Parameter Envelope in Support of an Early Site Permit*, Nuclear Energy Institute, Rev. 1, May 2012.

15-2. Exelon Nuclear, *Early Site Permit (ESP) Application for the Clinton ESP Site*, Rev. 4, April 14, 2006.

15-3. Entergy, System Energy Resources, Inc., *Grand Gulf Early Site Permit Application*, Rev. 2, October 3, 2005.

15-4. Dominion, Dominion Nuclear North Anna, LLC, *North Anna Early Site Permit Application*, Rev. 9, September 12, 2006.

15-5. PSEG Power, LLC, *Application for Early Site Permit for the PSEG Site*, Rev. 3, March 31, 2014.

15-6. Exelon Generation, Exelon Nuclear Texas Holdings, LLC, *Application for Early Site Permit for Victoria County Station*, Rev. 1, May 1, 2012.

15-7. Southern Company, Southern Nuclear Operating Company, Inc., *Vogtle Early Site Permit Application*, Rev. 5, December 23, 2008.

Table 15-1
Clinch River Nuclear Site Loss of Coolant Accident Doses

Location	Time (hr)	X/Q (s/m^3)		X/Q Ratio (Site/Vendor)	Dose (rem TEDE)	
		Site (95^{th} %)	Vendor		Vendor	Site
EAB	0-2	4.96×10^{-3}	1.0×10^{-3}	4.96	4.35	21.6[a]
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}	0.760	0.05	0.038
	96-720	4.30×10^{-5}	8.0×10^{-5}	0.538	0.06	0.032
	LPZ Total				4.75	2.97[a],[b]

(a) As compared to the 25 rem Total Effective Dose Equivalent (TEDE) limit specified in 10 CFR 52.17.
(b) Column total does not equal sum of individual values due to rounding.

Notes:

LPZ = Low Population Zone