AOO Evaluations

RG 1.206, C.I.11.1 Source Terms

This section addresses the sources of radioactivity that are generated within the core and have the potential of leaking to the RCS during normal plant operation, including AOO, by way of defects in the fuel cladding.

The applicant should provide two source terms. The first source term is a conservative or design-basis source term which assumes a design-basis fuel defect level. The applicant should provide the design-basis reactor primary and secondary coolant fission, activation, and corrosion product activities. The reactor core fission product inventories are determined based on time-dependent fission product core inventories that are calculated by the ORIGEN code. The first source term serves as a basis for (1) radwaste system design capability to process radioactive wastes at design-basis fuel defect level and fission product leakage level, (2) confirmation of compliance with radioactive gaseous and liquid effluent release standards and effluent monitoring requirements under routine operations and AOO, and (3) shielding requirements and compliance with occupational radiation exposure limits.

The second source term is a realistic model which represents the expected average concentrations of radionuclides in the primary and secondary coolant. The application should provide realistic reactor primary and secondary coolant fission, activation, and corrosion product activities. The supporting information should describe expected liquid and gaseous source terms by plant systems, transport or leakage mechanisms, system flow rates, applicable radionuclide partitioning and decontamination factors, and release pathways. For PWRs, the applicant should provide these activities in the steam generator secondary side for the liquid and steam phases. The applicant should determine these values using the model in ANSI/ANS 18.1-1999, “Radioactive Source Term for Normal Operation of Light-Water Reactors,” NUREG-0016, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWRs),” and NUREG-0017, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWRs).”

SRP 11.1, Source Terms

In general, the source terms used as the design basis for **expected releases** have been found acceptable if these values are determined using models and parameters that are consistent with NUREG-0016 (BWR-GALE code) or NUREG-0017 (PWR-GALE code) and the guidance provided in ANSI/ANS 18.1-1999.

Decontamination factors for in-plant control measures used to reduce gaseous effluent releases to the environment, such as iodine removal systems and high-efficiency particulate air (HEPA) filters for building ventilation exhaust systems and containment internal cleanup systems should be consistent with those given in Regulatory Guide 1.140. The building mixing efficiency for containment internal cleanup should be consistent with NUREG-0017.

For normal operations, effluent concentration limits at the boundary of the unrestricted area do not exceed the values specified in Table 2 of Appendix B to 10 CFR Part 20. The source terms result in meeting the design objectives for doses in unrestricted areas as set forth in Appendix I to 10 CFR Part 50.

AOO Evaluations

NUREG-0016, Calculation of Releases of Radioactive Materials in Gaseous Liquid Effluents from Boiling Water Reactors (BWR-GALE CODE)

The calculations performed by the BWR-GALE Code are based on (1) standardized coolant activities derived from American Nuclear Society (ANS) 18.1 Working Group recommendations, (2) release and transport mechanisms that result in the appearance of radioactive material in liquid and gaseous waste streams, (3) plant-specific design features used to reduce the quantities of radioactive materials ultimately released to the environs, and (4) information received on the operation of nuclear power plants.

Source Term: The calculated annual **average** quantity of radioactive material released to the environment from a nuclear power reactor during normal operation **including anticipated operational occurrences**. The source term is the isotopic distribution of radioactive materials used in evaluating the impact of radioactive releases on the environment. Normal operation includes routine outages for maintenance and scheduled refuelings.

RG 1.112 - Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-cooled Nuclear Power Reactors

This RG refers to NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling-Water Reactors (BWR-GALE Code)," Revision 1, dated January 1979, and NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized-Water Reactors (PWR-GALE Code)," Revision 1, dated April 1985.1 These two reports provide acceptable methods for calculating annual average expected releases of radioactive material in gaseous and liquid effluents from light-water-cooled nuclear power reactors.

This RG also refers to the methodology that the American National Standards Institute (ANSI) described in ANSI/ANS-18.1-1999, "Radioactive Source Term for Normal Operation for Light Water Reactors," may be used in calculating radioactive source terms in BWR reactor coolant and reactor steam, as well as PWR primary coolant and secondary water and steam.

SRP BTP 7-19

For each anticipated operational occurrence in the design basis occurring in conjunction with each single postulated common-cause failure, the plant response calculated **using best-estimate (realistic assumptions) analyses** should not result in radiation release exceeding 10 percent of the 10 CFR 100 guideline value or violation of the integrity of the primary coolant pressure boundary.

From RG 1.206, C.I.11.3.3 Radioactive Effluent Releases

The applicant should provide the criteria to be used for releasing gaseous wastes and acceptable release rates. Also describe the parameters, assumptions, and bases used to calculate releases of radioactive material in gaseous effluents, using RG 1.112 (Appendix A applies to BWRs and Appendix B applies to PWRs). If this guidance is not followed, describe the specific alternative methods used. Provide the expected releases of radioactive materials (by radionuclide) in gaseous effluents resulting from normal operation, including AOO, in MBq/yr (Ci/yr) per reactor.

Tabulate the releases by radionuclide for the total system and each subsystem, and indicate effluent concentrations. Demonstrate compliance with regulations by comparing the calculated effluents with the concentration limits in Table 2, Column 1, of Appendix B to 10 CFR Part 20. Calculate doses to members of the public in unrestricted areas, using the guidance in RGs 1.109 and RG 1.111. If this guidance is not followed, describe the specific alternative methods used. Compare the doses due to the effluents with the numerical design objectives of Appendix I to 10 CFR Part 50, compliance requirements of 10 CFR 20.1302, and the EPA environmental standards in 40 CFR Part 190 as they apply in SRP Section 11.5 in determining total dose. Indicate the atmospheric dispersion and deposition factors considered in the evaluation. (The atmospheric dispersion and deposition factors provided to assess the presence of airborne radioactivity at downwind locations depend on site-specific features.)

RG 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I

Equations use terms like:

- Tp which is the **average** transit time required for nuclides to reach the point of exposure.
- Un which is the **mean** wind speed of wind speed class n
- X/Q which is the annual **average** gaseous dispersion factor

SRP 2.3.5, Long-Term Atmospheric Dispersion Estimates for Routine Releases

The χ/Q and D/Q values reviewed in this SRP section are provided as input to the review of the calculated concentrations and dose consequences of routine airborne radioactive releases that is performed in SRP Section 11.3.

The χ/Q and D/Q values to be used for assessment of the consequences of routine airborne radiological releases as described in Section 2.3.5.2 of Regulatory Guide 1.70 (Ref. 9) and Section 2.3.5.2 of RG 1.206 (Ref. 10): Maximum annual average χ/Q values and D/Q values at or beyond the site boundary and at specific locations of potential receptors of interest utilizing appropriate meteorological data for each routine venting location.

Regulatory Guide 1.111 presents criteria for characterizing atmospheric dispersion and deposition conditions for evaluating the consequences of routine releases.

RG 1.111, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors.

The number of elements and the plume spread parameters should be selected such that the resulting concentration estimate is representative of the concentration from a continuous point source release.

Equations use terms like:

- U which is the **mean** windspeed at the height of the effective release point

SRP 15.2.1–15.2.5, Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); And Steam Pressure Regulator Failure (Closed) [these are AOOs in LWRs]

The applicant should analyze these events using an acceptable analytical model. Any other analytical method proposed by the applicant is evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by the appropriate organization for reactor systems.

The values of the parameters in the analytical model should be **suitably conservative**. The following values are acceptable:

A. The reactor is initially at 102 percent of the rated (licensed) core thermal power (to account for a 2 percent power measurement uncertainty unless a lower number can be justified through measurement uncertainty methodology and evaluation or unless the uncertainty otherwise is accounted for (see SAR (or DCD) Section 4.4)), and primary loop flow is at the nominal design flow less the flow measurement uncertainty.

B. Conservative scram characteristics are assumed (i.e., for a PWR maximum time delay with the most reactive rod held out of the core, for a BWR a 0.8 design conservatism multiplier on the predicted reactivity insertion rate) unless (i) a different conservatism factor can be justified through the uncertainty methodology and evaluation or (ii) the uncertainty is otherwise accounted for (see SAR (or DCD) Section 4.4).

C. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with Regulatory Guide 1.105.

From SRP 11.3, Gaseous Waste Management System

Releases of radioactive materials in gaseous effluents to unrestricted areas during normal plant operation and anticipated operational occurrences will not result in offsite radiation doses exceeding the dose objectives specified in Appendix I to 10 CFR Part 50 and concentrations of radioactive materials in gaseous effluents in any unrestricted area exceeding the limits specified in Table 2, Column 1, of Appendix B to 10 CFR Part 20.

Evaluation Findings - Regarding Sections II.B and II.C of **Appendix I**, the staff has considered releases of radioactive material (noble gases, radioiodine, tritium, carbon-14, and particulates) in gaseous effluents for normal operation, including anticipated operational occurrences, based on **expected amounts and concentrations of gaseous wastes** over the life of the plant for each reactor on the site. The staff has determined that the proposed GWMS is capable of maintaining releases of radioactive materials in gaseous effluents such that the calculated individual doses in an unrestricted area from all pathways of exposure are less than 0.05 mSv (5 mrem) to the total body or 0.15 mSv (15 mrem) to the skin and less than 0.15 mSv (15 mrem) to any organ from releases of radioiodines, tritium, carbon-14, and radioactive materials in particulate form.

Appendix I to Part 50

B.1. The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation.

ESBWR DCD, Chapter 15

The 10 CFR 20.1301(a)(1) 1 mSv (0.1 rem) annual dose limit combined with (i.e., subtracting) the 10 CFR 20.1302(b)(2)(ii) 0.5 mSv (0.05 rem) annual limit (for normal airborne releases) is the appropriate radiological acceptance limit for an AOO in combination with an additional SACF or single operator error (i.e., an AOO with an additional single failure). This position is conservatively based on an assumption that an individual at the exclusion boundary annually receives 100% of the 10 CFR 20.1302(b)(2)(ii) 0.5 mSv (0.05 rem) annual limit from normal operations (which is conservative, when compared to the 10 CFR 50, Appendix I 0.1 mGy (10 millirad) as low as reasonably achievable [ALARA] annual airborne gamma dose guideline), and applying the 10 CFR 20.1301(a)(1) 1 mSv (0.1 rem) annual dose limit. Therefore, the radiological acceptance criterion for an AOO with a single failure should generically be (1 mSv – 0.5 mSv) = 0.5 mSv (0.05 rem) total effective dose equivalent (TEDE).