U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

BRANCH TECHNICAL POSITION 11-5

POSTULATED RADIOACTIVE RELEASES DUE TO A WASTE GAS SYSTEM LEAK OR FAILURE

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of effectiveness of radwaste systems and health physics.

Secondary - None

A. BACKGROUND

During normal operation of a nuclear power plant, the reactor generates radioactive fission and activation gases and gases resulting from the radiolytic decomposition of water. These gases are continuously removed from the reactor coolant. After separation, radioactive gases may be treated for volume reduction by the removal of non-radioactive species before being stored for radioactive decay and ultimately released to the environment. The waste gas system accomplishes this separation, reduction, and decay process.

The waste gas system for integral pressurized water reactors (iPWRs) may include volume control tanks, letdown or shim bleed gas separation, gas stripping, cover gas collection, compressors, recombiners, surge and storage tanks, ambient or chilled charcoal adsorbers, moisture separators, condensers, filters, process sampling, process instrumentation and radiation monitoring, and automatic control features to divert or terminate process flows. In all cases, the waste gas system is part of the radioactive gaseous waste management system (GWMS) and information on the system is considered as part of the design information required by Title 10 of the *Code of Federal Regulations* (CFR), Section 50.34a. System operation is required to be in accordance with 10 CFR 50.36a. Design-Specific Review Standard (DSRS) Section 11.3 describes the design acceptance criteria for waste gas systems (as part of the GWMS) and regulatory requirements in controlling and monitoring process flows and associated gaseous effluent releases to the environment.

The basic criterion for reactor accidents, including waste gas system failures, is that offsite doses shall not exceed 25 rem to the whole body (10 CFR 100.11(a)(1) and 10 CFR 50.34(a)(1)(ii)). However, that criterion assumes that the probability of occurrence is very small. The staff assumes that the probability of an accidental release from the waste gas system is relatively high and that lower dose criteria are appropriate.

Generally, the following two kinds of waste gas system failures have been designated as warranting evaluation:

1. Gross system failures, such as rupture of a gas storage decay tank (Regulatory Guide (RG) 1.24) or rupture of a line from a charcoal delay bed (RG 1.98); and

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2. Malfunctions, such as operator errors, valve misalignments, malfunction of attendant equipment, and active component failures.

Both the probabilities and the consequences of a waste gas system leak or failure depend on the kind of accident considered and the characteristics of the system (as defined in Tables 15-1 and 15-4 in Section 15.7.1 of RG 1.70) or RG 1.206, Part I, Section C.I.15, Appendix C.I.15-A and C.I.15-D.

Waste gas system design characteristics differ between plants, but the most important common characteristic among plants is that designs incorporate the guidance of RG 1.143 to withstand the effects of a hydrogen explosion and earthquakes for gaseous wastes produced during normal operation and anticipated operational occurrences (AOOs). As a result, a gross failure of the waste gas system is considered highly unlikely, e.g., such as a failure involving the near total loss of the system's inventory of radioactive materials. However, for present purposes, the most important aspect is that they have been designed in accordance with RG 1.143, and therefore, the U.S. Nuclear Regulatory Commission (NRC) considered a higher dose criterion for evaluating gross failures of such fortified systems. Under Part 50, Appendix A, General Design Criterion (GDC) 61 addresses, in part, the ability of the GWMS design to ensure adequate radiological safety under normal and postulated accident conditions. The goal of this position paper is to minimize potential radiation exposures to the public and to provide reasonable assurance that the radiological consequences of a single failure of an active component in the waste gas system will not result in a dose in excess of a small fraction (i.e., 10 percent) of the 10 CFR Part 100 or 10 CFR Part 20 limit for whole body dose to any offsite individual for a postulated event.

The dichotomy in having dose criteria for systems designed to withstand explosions and earthquakes that differ from those systems that are not designed to withstand such events has led to a dilemma. System malfunctions appear to be the controlling failure mode and resistance to explosions and earthquakes provides no protection against operator error and system malfunction. No specific types of system malfunction failures have been designated as being representative. However, it appears that an event, such as valve misalignment or over-pressure, could result in a release approximating that from the rupture of a tank or pipe. Therefore, it was considered that, for future safety evaluations of waste gas systems, the failures analyzed could be limited to tank or pipe ruptures, but the dose criterion in every case should not exceed 25 mSv (2.5 rem) at the exclusion area boundary, given that such systems are fortified to withstand the effects of a hydrogen explosion and earthquakes. However, for systems not designed to withstand explosions and earthquakes, the criterion is 1 mSv (0.1 rem) at the exclusion area boundary (EAB).

This branch technical position (BTP) provides guidelines on postulated radioactive releases from a radioactive waste gas system leak or failure associated with normal operation and AOOs. The criteria in Section B, below, provide adequate and acceptable design solutions for the concerns outlined above. This BTP sets forth minimum requirements and does not prohibit the implementation of more rigorous design codes, standards, or quality assurance measures than those indicated herein. It also does not require a reevaluation of waste gas systems with limiting conditions for operation based on more conservative analysis and calculational assumptions.

B. BRANCH TECHNICAL POSITION

1. Waste Gas System Leak or Failure Analysis

A. Criteria

The applicant's final safety analysis report (FSAR) (DSRS Section 11.3) should provide an analysis of the radiological consequences of a single failure of an active component in the waste gas system. The analysis should provide reasonable assurance that, in the event of a postulated failure or leak of the waste gas system, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 25 mSv (2.5 rem) for systems designed to withstand internal explosions and earthquakes, or 1 mSv (0.1 rem) for systems not designed to withstand explosions and earthquakes. The bases for the analysis should include the assumption that the waste gas system fails to meet the design intent set forth in 10 CFR 50.34a(c) and GDC 60 of Appendix A to 10 CFR Part 50.

B. Source Term

The analysis on the radiological consequences of a single failure of an active component in the waste gas system should use a system design-basis source term. Source terms are reviewed under DSRS Section 11.1. For iPWR designs, 1 percent of the operating core fission product inventory should be assumed as being released to the primary coolant. The NRC staff method of calculation for this analysis is based on conservative assumptions to maximize the design capacity source term (sustained power operation). For an iPWR, 1 percent of the operating fission product inventory in the core should be assumed as being released to the primary coolant.

The analysis should assume principal parameters and conditions typical of the equipment designed to remove radioactive gases from the coolant and to process and treat these gases during normal operation, including AOOs, by the waste gas system. For this analysis, the NRC staff assumes that no major alteration would occur in the use or performance of gas separation, reduction, and decay equipment before and immediately following this unique unplanned release affected by the waste gas system maximum design capacity source term. The source terms and releases may be developed using methods referred to in RG 1.112 and PWR-GALE Code (NUREG-0017, as modified to reflect the design features of iPWRs) with appropriately justified adjustments made in modeling a specific type of event and designs of the reactor core and fuel.

C. Release

The NRC staff considers that the release to the environment resulting from the postulated event will occur via a pathway not normally used for planned releases, and the release will require a reasonable time to detect and take remedial action to terminate the release. The NRC staff considers that the release of a compressed gas storage tank of a batch-type waste gas system or the inadvertent bypass of the main decay portion of a continuous-type waste gas system will provide a conservative assumption for the release, while the input to the waste gas system is at the system design-basis source term. Only the radioactive noble gases (xenon and krypton) are to be considered since the assumed transit time is long enough to permit major radioactive decay of oxygen

and nitrogen isotopes. Unless determined otherwise by the type of component failure, particulates and radioiodines are assumed to be removed, by pretreatment, gas separation, and intermediate radwaste treatment equipment. The release should be assumed to occur within the building structure housing the waste gas system storage tank or the main decay position of the system. It should further be assumed that the effluent resulting from the postulated event will be released to the environs without continuous effluent radiation monitoring to automatically isolate and/or terminate the effluent release. In addition, ground-level release without credit for a building wake factor should be assumed, using a conservative (5 percent) short-term diffusion estimate (X/Q) for the identified release point and EAB sector, as determined by a method outlined in the acceptance criteria in Standard Review Plan (SRP) Section 2.3.4. Alternatively, the relative concentration at the nearest EAB sector location, given in Figure 1 of RG 1.24, may be used for ground level releases. No deposition is assumed to occur during downwind transport.

2. Staff Method for Analysis

A. Pressurized Gas Storage Decay Tanks: The safety analysis for the radiological consequences of a single failure of an active component in a waste gas system with compressed gas storage (holdup or decay) tanks or cover gas tanks assumes that the tank being filled has a major leak with its inventory of radioactivity being released to the environs. The following general procedural steps should be used for this analysis:

i. The radioactive noble gas inventory in the tank, at 100-percent capacity, should be determined based on the maximum expected radioactive source term and the system design capacity using the parameters and principal components considered for pretreatment and collection of waste gas to the waste gas system tanks during normal operation, including AOOs. The assumptions and parameters used in the analysis should be described and justified to include among others: a description of the event leading to the release, release path from the affected system and building to the environment, type of release, duration of the release, basis for the noble gas and radioiodines (as warranted) source terms, assumed receptor location, atmospheric dispersion parameters, and any modifying factor specific to the event.

ii. The radiological impact should be determined using the noble gas radionuclide inventory determined in Step 1 above, exposure to effective dose conversion factors for submersion of FRG Report No. 11 (Table 2.3) in mrem/hr per uCi/cm^3, any modifying factor specific to the event, and the relative concentration (X/Q, in s/m^3) at the nearest EAB sector location based on a conservative (5 percent) short-term relative concentration for the identified release point, as determined in SRP Section 2.3.4. Alternatively, the diffusion estimate given in Figure 1 of RG 1.24 may be used for ground-level releases.

When radioiodines are assumed to be present in the release, the radiological impact should be determined using the exposure to effective dose conversion factors for inhalation of FRG Report No. 11 (Table 2.1), in mrem per uCi inhaled, RG 1.183 inhalation rate of 3.5 E-04 m^3 per second, any modifying factor specific to the event, and short-term diffusion estimate (X/Q, s/m^3) as noted above.

iii. The dose, summed over all radionuclides, shall not exceed 25 mSv (2.5 rem) for systems designed to withstand explosions and earthquakes, or 1 mSv (0.1 rem) for systems not designed to withstand explosions and earthquakes. Using the same parameters, a corresponding technical specification (TS) can be defined to set a maximum radioactivity inventory limit on a tank, based on the maximum of 25 mSv (2.5 rem) or 1 mSv (0.1 rem) at the nearest exclusion area boundary and same noble gas mixture to assure that the BTP criteria are met at the exclusion area boundary.

iv. If the results of the analysis do not demonstrate compliance with BTP 11-5 acceptance criteria, the applicant is expected to propose a TS limiting the total amount of radioactivity GWMS components. The staff will evaluate the proposed TS limiting the radioactivity content of GWMS components to ensure that the TS is consistent with the safety evaluation. The staff will confirm that DSRS Chapter 16, Section 5.5, "Programs and Manuals," identifies the requirements for this TS and adequately address its implementation in operational programs.

B. Charcoal Delay Units: The safety analysis for the radiological consequences of a single failure of an active component in a waste gas system with charcoal delay or decay tanks assumes that the charcoal unit is bypassed with a 1-hour release to the environs. The staff considers that either a line bypass valve malfunction, control error, or a charcoal decay tank bypass will require a remedial action by isolation and that starting an alternate charcoal unit, if available, or reducing reactor power could take up to 2 hours. The following general procedural steps should be used for this analysis:

i. The radioactive noble gas and radioiodines (as warranted) inventory should be determined based on the maximum expected radioactive source term and the system design capacity using the parameters and principal components considered for pretreatment and collection of waste gas to the waste gas charcoal delay or decay tanks during normal operation, including AOOs. The assumptions and parameters used in the analysis should be described and justified to include among others: a description of the event leading to the release, release pathway from the affected system and building to the environment, type of release, duration of the release, basis for the noble gas and radioidines source terms after 30-minute decay (as appropriate), assumed receptor location, atmospheric dispersion parameters, and any modifying factor specific to the event.

ii. The modeling of noble gases retained in and released from charcoal delay beds may follow the methodology and system parameters given in NUREG-0017 (as modified), Section 2.2.13.1, and assumptions listed in RG 1.98, Regulatory Position C.2, Steps 2.b. to 2.g. In the context of RG 1.98 assumptions, the inclusion of radioiodines in the source term, in addition to noble gases, should be determined by the type of assumed event and whether the failed component could also lead to the release of radioiodines.

iii. The radiological impact should be determined using the noble gas radionuclide inventory determined in Step 1 above, exposure to effective dose conversion factors for submersion of FRG Report No. 11 (Table 2.3)

in mrem/hr per uCi/cm^3, any modifying factor specific to the event, and EAB sector location exclusion area boundary based on a conservative (5 percent) short-term diffusion estimate (X/Q, s/m^3) for the identified release point, as determined by the applicant in SRP Section 2.3.4. Alternatively, the diffusion estimate given in Figure 1 of RG 1.24 may be used for ground-level releases.

When radioiodines are assumed to be present in the release, the radiological impact should be determined using the exposure to effective dose conversion factors for inhalation of FRG Report No. 11 (Table 2.1), in mrem per uCi inhaled, RG 1.183 inhalation rate of 3.5 E-04 m^3 per second, any modifying factor specific to the event, and short-term diffusion estimate (X/Q, s/m^3) as noted above.

iv. The dose, summed over all radionuclides, shall not exceed 1 mSv (0.1 rem). Using the same parameters, a corresponding TS can be defined to set a maximum release rate (Q_i , Bq/s or uCi/s or a maximum tank radioactivity inventory (Bq or Ci) to the waste gas system corresponding to this acceptance criterion, as determine above for an exposure duration of 2 hours at the EAB.

v. If the results of the analysis do not demonstrate compliance with BTP 11-5 acceptance criteria, the applicant is expected to propose a TS limiting the total amount of radioactivity GWMS components. The staff will evaluate the proposed TS limiting the radioactivity content of GWMS components to ensure that the TS is consistent with the safety evaluation. The staff will confirm that DSRS Chapter 16, Section 5.5, “Programs and Manuals,” identifies the requirements for this TS and adequately address its implementation in operational programs.

C. EVALUATION FINDINGS

The staff will document the results of its evaluation and confirm compliance with DSRS Section 11.3 and BTP 11.5 methodology and radiological acceptance criteria in assessing radiological impacts. The staff will describe what was done to evaluate the applicant’s analysis. The staff’s evaluation include verifying the applicant’s results, determining whether the applicant followed applicable regulatory guidance or used an alternative approach, performing independent calculations, and confirming the adequacy of stated assumptions and model parameters used in the analysis.

The staff will summarize the information used in assessing the consequence of a GWMS tank failure or leak, including the assumed failure scenarios, the basis of the radioactive source term, site characteristics and parameters used in modeling the transport of radioactivity to a dose receptor located at the EAB, and resulting doses. The staff will determine the acceptability of special design features, if considered by the applicant, to mitigate the consequences of a GWMS component failure or leak. In instances where the DSRS Section 11.3 and BTP 11.5 acceptance criteria cannot be met, the staff will confirm that the applicant has proposed appropriate TSs limiting the total amount of radioactivity in GWMS components.

In addition to the above, the staff will introduce the appropriate supporting information and conclusions in its evaluation findings, based on the information presented by the applicant and

results of the staff's evaluation. The reviewer may state that certain information provided by the applicant was not considered to be essential to the staff's review and was not reviewed, or that the staff used alternative information or parameters in performing its independent evaluation.

The following provide generic conclusions that will be included, and modified accordingly, in the staff's safety evaluation report:

> With respect to the consequence analysis addressing the radiological impact from a waste gas system leak or failure, the applicant provided the results of a site-specific analysis demonstrating compliance with the acceptance criteria of DSRS Section 11.3 and BTP 11-5. Supporting information on the staff's evaluation of the site's atmospheric dispersion characteristics in dispersing radioactive materials to a dose receptor located at the EAB is presented in SRP Section 2.3.4. The staff concludes that the analysis provided by the applicant is consistent with the guidelines of BTP 11-5 and meets the dose acceptance criteria defined in BTP 11-5 for an individual located at the EAB. The results are also consistent for a case where the waste gas system (is) / (is not) designed to withstand the effects of an internal hydrogen explosion and earthquakes. (*Note: staff to specify the applicable case*)
>
> The staff concludes that the applicant's proposed TSs limiting the total amounts of radioactivity in tanks and components, as described in the application, are adequate based on the results of the staff's review and evaluation. The basis of the staff's acceptance of the TSs is based on the evaluation of the selected GWMS failed component or leak, assumed inventory of radioactive materials in the failed component or leak, assumed failure scenario, methods and assumptions used in modeling the transport of radioactivity to a dose receptor located at the nearest EAB sector, and definition of exposure scenario. Supporting information on the staff's evaluation of the site's atmospheric dispersion characteristics in transporting radioactivity a dose receptor located at the EAB is presented in SRP Section 2.3.4.
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> The evaluation demonstrates compliance with BTP 11-5 acceptance dose criterion of 1 mSv (0.1 rem) for systems not designed to withstand explosions and earthquakes; or a dose of 25 mSv (2.5 rem) for systems designed to withstand explosions and earthquakes. (*Note: staff to specify the applicable case*) The staff confirmed that the proposed TSs limiting the radioactivity content for the stated GWMS tank and components have been incorporated into Chapter 16, Section 5.5, "Programs and Manuals," of the FSAR, and identified as a program element in the Offsite Dose Calculation Manual (ODCM), as addressed in FSAR Sections 11.5 and 13.4 of COL applications.

D. REFERENCES

1. 10 CFR Part 50.34a, "Design Objective for Equipment to Control Releases of Radioactive Materials in Effluents—Nuclear Power Reactors."

2. 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors."

3. 10 CFR Part 50, Appendix A, GDC 60, "Control of Releases of Radioactive Materials to the Environment."

4. 10 CFR Part 50, Appendix A, GDC 61, "Fuel Storage and Handling and Radioactivity Control."

5. 10 CFR Part 100, “Reactor Site Criteria.”

6. RG 1.24 (Safety Guide 24), “Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure.”

7. RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition.”

8. RG 1.98, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor.”

9. 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.”

10. RG 1.112, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluent from Light-Water-Cooled Power Reactors.”

11. RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures and Components Installed in Light-Water-Cooled Nuclear Reactor Power Plants.”

12. RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

13. RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition).”

14. NUREG-0800, Standard Review Plan, Section 2.3.4, “Short-term Dispersion Estimates for Accidental Atmospheric Releases.”

15. NUREG-0017, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWRs) (PWR GALE Code), Revision 1.”

16. Federal Guidance Report No. 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” EPA 520/1-88-020, September 1988.