U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

BRANCH TECHNICAL POSITION 11-5

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

REVIEW RESPONSIBILITIES

Primary - Organizations responsible for the reviews of waste gas processing systems, and health physics and radiation protection in meeting effluent concentration and dose limits.

Secondary - None

A. BACKGROUND

During operation, a nuclear power plant 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 nonradioactive species before being stored for radioactive decay and ultimately released to the environment. The waste gas system accomplishes this separation, reduction, and decay process.

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The waste gas system in boiling-water reactors (BWRs) may include steam air ejectors, vacuum pumps, decay pipes, moisture separators and driers, condensers, cryogenic distillation, tanks, ambient or chilled charcoal absorber beds, filters, process sampling, instrumentation and radiation monitoring, and automatic control features to divert or terminate process flows.

The waste gas system in pressurized-water reactors (PWRs) 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 absorbers, moisture separators, condensers, filters, process sampling, process instrumentation and radiation monitoring, and associated control features.

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* (10 CFR) 50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents—Nuclear Power Reactors." System operation is required to be in accordance with 10 CFR 50.36a. Standard Review Plan (SRP) Section 11.3, "Gaseous Waste Management System," describes the design acceptance criteria for waste gas systems (as part of the GWMS) and regulatory requirements for 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 higher 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, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized-Water Reactor Radioactive Gas Storage Tank Failure") or the rupture of a line from a charcoal delay bed (RG 1.98, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling-Water Reactor")

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 exemplified 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). These characteristics are illustrative of specific considerations that may be used in describing events and developing accidents.

Waste gas system design characteristics differ between PWR and BWR plants, and among plant designs, but the most important common characteristic among plants is that designs incorporate the guidance in RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled- Nuclear Power Plants," 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, such as a failure involving the near total loss of the system's inventory of radioactive materials, is considered highly unlikely. However, for present purposes, the most important aspect is that such systems have been designed in accordance with RG 1.143; therefore, the U.S. Nuclear Regulatory Commission (NRC) considered a higher dose criterion for evaluating gross failures of such fortified systems. Under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, General Design Criterion (GDC) 61, "Fuel Storage and Handling and Radioactivity Control," addresses, in part, the ability of the GWMS design to ensure adequate radiological safety under normal and postulated accident conditions. The goal of this branch technical position (BTP) 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 percentage (i.e., 10 percent) of the 10 CFR Part 100, "Reactor Site Criteria," or 10 CFR Part 20, "Standards for Protection against Radiation," 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 of the types of expected events and releases. However, it appears that an event, such as a valve misalignment or system over-pressure and leakage, could result in a release approximating that from the rupture of a system tank or piping. Therefore, it was considered that, for safety evaluations of waste gas systems, the failures analyzed could be limited to tank or pipe ruptures, but that the dose criterion in every case should not exceed 25 mSv (2.5 rem) at the exclusion area boundary (EAB), given that such systems are fortified to withstand the effects of a hydrogen explosion and earthquakes. However, for WGS not designed to withstand explosions and earthquakes, the criterion is 1 mSv (0.1 rem) at the EAB.

This BTP provides guidance 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 acceptance criteria and does not prohibit the implementation of more rigorous design codes, standards, or quality assurance measures than those indicated herein. It also does not impose a reevaluation of waste gas systems with limiting conditions for operation based on more conservative analysis and calculational assumptions.

B. BRANCH TECHNICAL POSITION

1. <u>Waste Gas System Leak or Failure Analysis</u>

   A. Criteria

      The applicant's Safety Analysis Report (SAR) based on the guidance of SRP 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 EAB will not exceed 25 mSv (2.5 rem) for systems that are designed to withstand internal explosions and earthquakes, or 1 mSv (0.1 rem) for systems that are not designed to withstand internal 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, “Control of Releases of Radioactive Materials to the Environment,” of Appendix A to 10 CFR Part 50.

B. Source Term

The analysis of 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 using the guidance of SRP Section 11.1, “Coolant Source Terms.” The NRC staff’s method of calculation for this analysis is based on the conservative assumptions to maximize the design capacity source term (sustained power operation). These assumptions are:

i. For a PWR, 1 percent of the operating fission product inventory in the core should be assumed as being released to the primary coolant.

ii. For a BWR, a fission product release rate should be assumed consistent with the noble gas release to the reactor coolant of 100 μCi/s per MWt (after 30-minute decay).

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 waste gas equipment, such as separation, reduction, and decay in storage tanks or charcoal delay beds, before and immediately following an unplanned release, as characterized by the waste gas system maximum design capacity source term. The radioactive source terms and releases may be developed using methods referred to in RG 1.112, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors,” using the BWR gaseous and liquid effluent (GALE) Code and PWR-GALE Code, as modified to reflect specific design features. In all cases, the analysis should justify any adjustments made in modeling a specific type of event, and should describe the design features of the reactor core, type of fuel, and primary coolant inventories that form the basis of the source term. See 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),” for details on input parameters used by BWR-GALE and PWR-GALE codes, respectively.

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 that the release will require a reasonable time to detect and subsequently 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 by a continuous-type waste gas system (e.g., charcoal delay beds or tanks) 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) should be considered since the assumed transit time is long enough to permit major radioactive decay of oxygen and nitrogen isotopes. Unless determined otherwise based on 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 SRP Section 2.3.4, "Short-Term Dispersion Estimates for Accident Releases." 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 radioactive ground deposition is assumed to occur during downwind transport of the release to the location of the assumed dose receptor.

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 (i.e., held up for radioactive decay) tanks or cover gas tanks assumes that the tank being filled has a major leak resulting in 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 accumulation of waste gas into 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 radioiodine (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 2.A.i above, use of effective dose conversion factors for submersion given in Federal Guidance Report (FGR) No. 12, "External Exposure to Radionuclides in Air, Water, and Soil" (U.S. Environment Protection Agency (EPA) 402-R-93-081), issued September 1993 (Table III.1), in mrem/hour 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 using the guidance of SRP Section 2.3.4 in deriving the relative concentration. Alternatively, the relative concentration value given in Figure 1 of RG 1.24 may be used for ground-level releases.

When the radioiodines are assumed to be present in the release, the radiological impact should be determined using the effective dose conversion factors for inhalation from FRG 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), issued September 1988 (Table 2.1), in mrem per uCi inhaled, inhalation rate of 3.5 E-04 m^3 per second as derived from RG 1.183, any modifying factor specific to the event, and short-term diffusion estimate (X/Q, s/m^3) as noted in step 1.C above.

iii. The dose, summed over all radionuclides, shall not exceed 25 mSv (2.5 rem) for systems that are designed to withstand explosions and earthquakes, or 1 mSv (0.1 rem) for systems that are 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 EAB and the same noble gas mixture to assure that the BTP criteria are met at the exclusion area boundary.

iv. If the results of the analysis are not consistent with BTP 11-5 acceptance criteria, the applicant should propose a TS limiting the total amount of radioactivity in 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 results and conclusions of the safety evaluation. The staff will confirm that SRP Chapter 16, "Technical Specifications," TS Section 5.5, "Programs and Manuals," identifies the requirements for this TS and adequately addresses its implementation in operational programs. The associated guidance is presented in standard technical specifications (STS), including NUREG-1430, "Standard Technical Specifications—Babcock and Wilcox Plants," issued April 2012; NUREG-1431, "Standard Technical Specifications—Westinghouse Plants," issued April 2012; NUREG-1432, "Standard Technical Specifications—Combustion Engineering Plants," issued April 2012; NUREG-1433, "Standard Technical Specifications—General Electric Plants (BWR/4)," issued April 2012; and NUREG-1434, "Standard Technical Specifications—General Electric Plants (BWR/6)," issued April 2012.

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 beds 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 radioiodine (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 beds 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 path from the affected system and building to the environment, type of release, duration of the release, basis for the noble gas and radioiodine source terms after 30-minute decay (as appropriate), assumed location of dose receptor, 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 Section 2.2.13.1 of NUREG-0016 or NUREG-0017 (with code input parameters modified to reflect specific system design features), and assumptions listed in RG 1.98, Regulatory Position C.2, steps 2.b. through 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 2.A.i above, use of effective dose conversion factors for submersion given in FGR No. 12 (Table III.1) in mrem/hour per uCi/cm^3, any modifying factor specific to the event, and the EAB sector location exclusion area boundary based on a conservative (5 percent) short-term relative concentration estimate (X/Q, s/m^3) for the identified release point using the guidance of 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 effective dose conversion factors for inhalation given in FGR No. 11 (Table 2.1), in mrem per uCi inhaled, inhalation rate of 3.5 E-04 m^3 per second as derived from RG 1.183, 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 25 mSv (2.5 rem) for systems that are designed to withstand explosions and earthquakes, or 1 mSv (0.1 rem) for systems that are not designed to withstand explosions and earthquakes. Using the same parameters, a corresponding TS may be derived to set a maximum release (Q_i or Ci), release rate (Bq/s or uCi/s, or a maximum tank radioactivity inventory (Bq or Ci)) to the waste gas system corresponding to the 2–hour EAB dose derived in steps B.i through B.iii above.

v. If the results of the analysis are not consistent with BTP 11-5 acceptance criteria, the applicant should 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 results and conclusions of the safety evaluation. The staff will confirm that SRP Chapter 16, TS Section 5.5, identifies the requirements for this TS and adequately addresses its implementation in operational programs. The associated guidance is presented in STS, including NUREG-1430, NUREG-1431, NUREG-1432, NUREG-1433, and NUREG-1434.

C. EVALUATION FINDINGS

The staff will document the results of its evaluation and will confirm that they are consistent with SRP Section 11.3 in describing system components and operation and BTP 11-5 methodology and radiological acceptance criteria in assessing radiological impacts. The staff will describe what it did to evaluate the applicant's analysis. The staff's evaluation includes 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 SRP Section 11.3 and BTP 11-5 acceptance criteria cannot be met, the staff will confirm that the applicant has proposed appropriate TS limiting the total amount of radioactivity in GWMS components.

In addition to the above, the staff will introduce the appropriate supporting information from its independent analysis in evaluation findings, based on the information presented by the applicant. If appropriate, 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 conclusion language that will be included, and modified accordingly, in the staff's Safety Evaluation Report (SER):

A. 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 that are consistent with the acceptance criteria of SRP Section 11.3 and BTP 11-5. Supporting information on the staff's evaluation of the site's atmospheric dispersion characteristics in transporting 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 guidance of BTP 11-5 and with the dose acceptance criteria 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 to staff: Specify the applicable case in the SER.)

B. The staff concludes that the applicant's proposed TS 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 staff' bases its acceptance of the TS 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. SRP Section 2.3.4 presents supporting information on the staff's evaluation of the site's atmospheric dispersion characteristics in transporting radioactive materials to a dose receptor located at the EAB.

C. The evaluation demonstrates that the results are consistent 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. The staff confirmed that the proposed TS 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 Final Safety Analysis Report (FSAR), and identified as a program element in the Offsite Dose Calculation Manual, as addressed in FSAR Sections 11.5 and 13.4 of design certifications, operating licenses, and combined license applications.

D. REFERENCES

1. 10 CFR 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 20, "Standards for Protection against Radiation,"

6. 10 CFR Part 100, "Reactor Site Criteria."

7. FGR 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.

8. FGR No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," (EPA 402-R-93-081), September 1993.

9. NUREG-0016, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling- Water Reactors (BWRs),” (BWR-GALE Code, GALE86), Revision 1.

10. NUREG-0017, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized- Water Reactors (PWRs),” (PWR-GALE Code, GALE86), Revision. 1.

11. NUREG-0800, SRP Section 2.3.4, “Short-term Dispersion Estimates for Accidental Atmospheric Releases.”

12. NUREG-1430, “Standard Technical Specifications - Babcock and Wilcox Plants.”

13. NUREG-1431, “Standard Technical Specifications - Westinghouse Plants.”

14. NUREG-1432, “Standard Technical Specifications - Combustion Engineering Plants.”

15. NUREG-1433, “Standard Technical Specifications - General Electric Plants (BWR/4).”

16. NUREG-1434, “Standard Technical Specifications - General Electric Plants (BWR/6).”

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

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

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

20. 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.”

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

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

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

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

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 52 and were approved by the Office of Management and Budget, approval numbers 3150-0014, 3150-0011, 3150-0151, and 3150-0135.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

SRP Section 11.3, BTP 11-5
Description of Changes

BTP 11-5 "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure"

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in BTP 11-5, as referenced in SRP Section 11.3, Revision 3, dated March 2007. See (ADAMS Accession No. ML070730056).

Editorial changes added new abbreviations in several places throughout this section and corrected grammatical errors. Other changes reflect the removal of redundant information.

Technical changes incorporated in this revision include:

I. AREAS OF REVIEW

The areas of review, as described in the background and technical position sections, were revised by expanding technical guidance in developing radioactive source terms assumed to be present in a failed component.

The guidance also addresses the possible presence of radioiodines and, if radioiodines are present, the applicant or licensee should consider in their dose analysis the associated radiological dose in addition to that contributed by noble gases.

If the results of the dose analyses are not consistent with the BTP acceptance criteria, the applicant should propose technical specifications limiting the total amount of radioactivity in waste gas components. The maximum inventory of radioactive materials, in the event of an uncontrolled release of radioactivity, is based on that quantity of radioactivity that would not exceed the SRP dose acceptance criteria due to inhalation and immersion exposure pathways at the EAB. Chapter 16, Section 5.5, "Programs and Manuals," of the FSAR addresses this commitment in COL applications. The milestones for the development and implementation of such plant- and site--specific requirements are addressed in FSAR Sections 11.5 and 13.4 of COL applications.

II. ACCEPTANCE CRITERIA

The acceptance criteria, as doses determined at the EAB, remain unchanged from the prior version of BTP 11-5.

Compliance with 10 CFR Part 50, Appendix A, GDC 61 was noted in the guidance as it relates to the consequence analyses conducted under SRP Section 11.3 using BTP 11-5.

III. REVIEW PROCEDURES

The review procedures section was updated in recognition of the revisions identified in the areas of review and acceptance criteria sections, as noted in explanations above.

IV. EVALUATION FINDINGS

The evaluation findings section was revised by adding a new subsection. The new subsection discusses the results of the staff's evaluation and conclusion of acceptability with the guidance of the BTP and its acceptance criteria. The revisions address the following factors:

1. With respect to the consequence analysis addressing the radiological impact from a waste gas system leak or failure, the revision provides more specific guidance on how the staff would confirm that the applicant has provided the results of a site-specific analysis that are consistent with the acceptance criteria of SRP 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.

2. If the results of site-specific analyses are not consistent with the BTP acceptance criteria, the staff would confirm that the applicant's proposed TS limiting the total amounts of radioactivity in waste gas components are adequate. The staff would conclude that the applicant's proposed TS limiting the total amounts of radioactivity in waste gas components, as described in the application, are adequate based on the results of the staff's review and evaluation. The staff would also confirm that the proposed TS limiting the radioactivity content for the stated waste gas components have been incorporated into Chapter 16, TS Section 5.5, "Programs and Manuals," of the FSAR, and identified as a program element, as addressed in FSAR Sections 11.5 and 13.4 of COL applications.

V. IMPLEMENTATION

The implementation of BTP 11-5 is addressed in the corresponding part of SRP Section 11.3 on implementation. No new provisions were added on the implementation of BTP 11-5.

VI. REFERENCES

The following references were added in support of the expanded discussions presented in areas of review, acceptance criteria, and review procedures. The added references are:

1. 10 CFR Part 50, Appendix A, GDC 61, "Fuel Storage and Handling and Radioactivity Control."

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

3. RG 1.183, "Alternative Radiological Source Terms for Evaluating Design- Basis Accidents at Nuclear Power Reactors."

4. RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

5. NUREG-0800, SRP Section 2.3.4, “Short- term Dispersion Estimates for Accidental Atmospheric Releases.”

6. NUREG-1430, “Standard Technical Specifications - Babcock and Wilcox Plants.”

7. NUREG-1431, “Standard Technical Specifications - Westinghouse Plants.”

8. NUREG-1432, “Standard Technical Specifications - Combustion Engineering Plants.”

9. NUREG-1433, “Standard Technical Specifications - General Electric Plants (BWR/4).”

10. NUREG-1434, “Standard Technical Specifications - General Electric Plants (BWR/6).”

11. FGR 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.

12. FGR No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," (EPA 402-R-93-081), September 1993.

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