UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

March 7, 2006

# NRC REGULATORY ISSUE SUMMARY 2006-04 EXPERIENCE WITH IMPLEMENTATION OF ALTERNATIVE SOURCE TERMS

## ADDRESSEES

All holders of operating licenses for nuclear power reactors except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

## INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to update addressees on experience with implementation of alternative source terms (ASTs) in design basis accident (DBA) radiological analyses of currently licensed light water reactors. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate. However, any suggestions this RIS may contain are not NRC requirements; therefore, no specific action or written response is required on the part of an addressee.

In particular, the information in this RIS should be useful to licensing and engineering staffs at currently operating reactors and to contractors supporting implementation of an AST through a license amendment request (LAR). This RIS should aid in the reduction of requests for additional information (RAIs) and help improve the planning for and implementation of an AST.

## BACKGROUND INFORMATION

Many of the LARs for implementation of an AST have lacked the necessary information for the NRC staff to make its safety determination. The purpose of this RIS is to discuss the more frequent and significant issues encountered by the NRC staff during its review of AST submittals and to provide information for licensees to consider when developing submittals for implementation of an AST. Frequently, licensees submit AST analyses using methods that are not simple variations or technically justifiable extensions of those addressed in the NRC guidance documents. These methods require additional NRC staff research and review and frequently result in RAIs. Licensees should consider early interaction with NRC staff during development of submittals that incorporate assumptions, methods, or analyses not specifically discussed in the regulatory documents or guidance, since they may require additional NRC staff review.

**ML053460347**

**SUMMARY OF ISSUE**

**1. <u>Level of Detail Contained in LARs</u>**

An AST amendment request should describe the licensee's analyses of the radiological and non-radiological impacts and provide a justification for the proposed modification in sufficient detail to support review by the NRC staff. For example, the AST amendment request should (1) provide justification for each individual proposed change to the technical specifications (TS), (2) identify and justify each change to the licensing basis accident analyses, and (3) contain enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations. The provision of sufficient detail is necessary for the NRC staff to be able to conclude, with reasonable assurance, whether the licensee's analyses and changes are acceptable. For a previous NRC staff discussion on the level of detail necessary for review, see RIS 2001-19, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests" (Ref 1).

In response to RAIs, some licensees have made changes to originally proposed LARs and their supporting analyses. In some cases, these changes were extensive or involved multiple re-analyses and supplements. Because of the depth and scope of many AST submittals, multiple changes to the original submittal (particularly those with multiple supplements that revise portions of previous supplements) can increase the chance of NRC staff using information that has been superceded during the review. For these cases, NRC staff recommends that licensees identify the most current analyses, assumptions, and TS changes in their submittal and supplements to the submittal.

**2. <u>Main Steam Isolation Valve (MSIV) Leakage and Fission Product Deposition in Piping</u>**

For calculation of aerosol settling velocity in the main steamline (MSL) piping of boiling water reactors, some LARs reference Accident Evaluation Report (AEB) 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term" (Ref. 2). This is acceptable. However, it is important to note that the report was written based on the parameters of a particular plant and, therefore, the removal rate constant is specific to that plant. Any licensee who chooses to reference these AEB 98-03 assumptions should provide appropriate justification that the assumptions are applicable to their particular design.

Both NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Ref. 3) and Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants" (Ref. 4) define AST as a fission product release from the reactor core into the containment. Neither provides sufficient information regarding the amount and composition of fission products in the reactor vessel or the attached piping. As indicated in Appendix A to RG 1.183, Regulatory Position 6.0, the NRC staff accepts the practice of treating fission product concentration in containment (more specifically in the drywell) as representative of that in the vicinity of the MSIV. Some AST amendment requests have reduced the drywell activity levels by assuming mixing with the free air volume of the wetwell. If appropriate justification is provided, the suppression pool free air volume may be included, provided there is a mechanism to ensure mixing between the drywell and wetwell. For example, the NRC staff would expect to see thermal and hydraulic analyses in support of crediting mass exchange between the wetwell and drywell airspace for time periods associated with fission product releases.

The size distribution of airborne particles in the vicinity of the MSIV is, in general, different from that in the containment. Since the piping is attached to the source of fission product releases, the agglomeration process of highly concentrated but small aerosols may substantially differ from that in containment. Modeling of MSL piping may include volumes between the reactor pressure vessel and the inboard MSIV (inboard volume), between the inboard and outboard valves (in-between volume), and outside of the outboard valve (outboard volume). Since a majority of large (i.e., heavier) particles deposit in the inboard volume, the distribution of the aerosol that leaks to the subsequent volume is smaller (i.e., lighter) particles. This particle behavior leads to the conclusion that the choice of an effective settling velocity in any volume should account for the distribution of particle sizes in that volume. The steam flow rate during the accident also affects the removal of particles and should be accounted for in the analysis. For aerosol settling, only horizontal sections of piping should be credited. The effective settling area should be calculated as the length of horizontal piping multiplied by the pipe diameter.

Deposition of gaseous iodine (elemental and organic) in the piping is a frequent point of contention between licensees and NRC staff. Some licensees claim that because of chemical adsorption, a large portion of iodine is deposited on the piping surface. However, this deposition is strongly dependent upon the thermal and hydraulic conditions in the piping. Given the large uncertainty associated with iodine behavior in piping, deposition of gaseous iodine in piping should be omitted unless appropriate justification is provided (including providing estimates of the thermal and hydraulic conditions in the piping).

## 3. <u>Control Room Habitability</u>

When implementing an AST, some licensees have proposed that certain engineered safety features (ESF) ventilation systems not be credited as a mitigation feature in response to an accident. In some cases, the licensee’s revised design basis analysis introduced the assumption that normal (non-ESF) ventilation systems are operating during all or part of an accident scenario. Such an assumption is inappropriate unless the non-ESF system meets certain qualities, attributes, and performance criteria as described in RG 1.183, Regulatory Positions 4.2.4 and 5.1.2. For example, credit for the operation of non-ESF ventilation systems should not be assumed unless they have a source of emergency power. In addition, the operation of ventilation systems establishes certain building or area pressures based upon their flowrates. These pressures affect leakage and infiltration rates which ultimately affect operator dose. Therefore, to credit the use of these systems, licensees should incorporate the systems into the ventilation filter testing program in Section 5 of the TS. In summary, use of non-ESF ventilation systems during a DBA should not be assumed unless the systems have emergency power and are part of the ventilation filter testing program in Section 5 of the TS.

Generic Letter (GL) 2003-01, “Control Room Habitability” (Ref. 5) requested licensees to confirm the ability of their facility’s control room to meet applicable habitability regulatory requirements. In addition, licensees were requested to confirm that control room habitability systems were designed, constructed, configured, operated and maintained in accordance with the facility’s design and licensing bases. The GL placed emphasis on licensees confirming that the most limiting unfiltered inleakage into the control room envelope (CRE) was not greater than the value assumed in the DBA analyses. The tests, measurements, and analyses which were performed for this confirmation were to be described in the response to the GL. Some AST amendment requests proposed operating schemes for the control room and other ventilation systems which affect areas adjacent to the CRE and are different from the manner of operation and performance described in the response to the GL without providing sufficient justification for the proposed changes in the operating scheme. In some cases, licensees proposed new modes of operation that lacked confirmation of the CRE inleakage characteristics.

Measurements[1] of these characteristics are important to confirm inleakage assumptions used in the analyses for an AST amendment, even for those situations in which the air in the control room would appear to be stagnant.

## 4. Atmospheric Dispersion

Licensees may continue to use atmospheric relative concentration (χ/Q) values and methodologies from their existing licensing-basis analyses when appropriate. Licensees also have the option to adopt the generally less conservative (more realistic) updated NRC staff guidance on determining χ/Q values in support of design basis control room radiological habitability assessments provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" (Ref. 6). Regulatory positions on χ/Q values for offsite (i.e., exclusion area boundary and low population zone) accident radiological consequence assessments are provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (Ref. 7).

Based on submittal reviews, the NRC staff identified the following areas of improvement for licensee submittals that propose revision of the design basis atmospheric dispersion analyses for implementing AST. They should include the following information:

- A site plan showing true North and indicating locations of all potential accident release pathways and control room intake and unfiltered inleakage pathways (whether assumed or identified during inleakage testing).

- Justification for using control room intake χ/Q values for modeling the unfiltered inleakage, if applicable.

- A copy of the meteorological data inputs and program outputs along with a discussion of assumptions and potential deviations from staff guidelines. Meteorological data input files should be checked to ensure quality (e.g., compared against historical or other data and against the raw data to ensure that the electronic file has been properly formatted, any unit conversions are correct, and invalid data are properly identified).

When running the control room atmospheric dispersion model ARCON96, two or more files of meteorological data representative of each potential release height should be used if χ/Q values are being calculated for both ground-level and elevated releases (see RG 1.23, "Onsite Meteorological Programs," Regulatory Position 2 (Ref. 8) and Table A-2 in Appendix A to RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants"). In addition, licensees should be aware that (1) two levels of wind speed and direction data should always be provided as input to each data file, (2) fields of "nines" (e.g., 9999) should be used to indicate invalid or missing data, and (3) valid wind direction data should range from 1° to 360°. Licensees should also provide detailed engineering information when applying the default plume rise adjustment cited in RG 1.194 to control room χ/Q values to account for buoyancy or mechanical jets of high energy releases.

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[1] Use of parametric studies in which inleakage rates are varied is not a preferable alternative to CRE inleakage measurements. (see GL 2003-01)

This information should demonstrate that the minimum effluent velocity during any time of the release over which the adjustment is being applied is greater than the 95th percentile wind speed at the height of release.

When running the offsite atmospheric dispersion model PAVAN, two or more files of meteorological data representative of each potential release height should be used if $\chi/Q$ values are being calculated for pathways with significantly different release heights (e.g., ground level versus elevated stack). The joint frequency distributions of wind speed, wind direction, and atmospheric stability data used as input to PAVAN should have a large number of wind speed categories at the lower wind speeds in order to produce the best results (e.g., Section 4.6 of NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations" (Ref. 9), suggests wind speed categories of calm, 0.5, 0.75, 1.0, 1.25, 1.5, 2.0, 3.0, 4.0 5.0, 6.0, 8.0 and 10.0 meters per second).

## 5. Modeling of ESF Leakage

ESF systems that recirculate sump water outside the primary containment may leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (e.g., refueling water storage tank). Appendix A to RG 1.183, Regulatory Position 5, states that "the radiological consequences from the postulated [ESF] leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the [loss-of-coolant accident] LOCA."

The allowable ESF leakage is typically contained in the plant's TS or procedures. The ESF leakage at accident conditions may differ from the ESF leakage at normal operating conditions. Licensees should account for ESF leakage at accident conditions in their dose analyses so as not to underestimate the release rate.

In Appendix A to RG 1.183, Regulatory Position 5.5, the NRC staff provided a conservative value of 10 percent as the assumed amount of iodine that may become airborne from ESF leakage that is less than 212 °F. The NRC staff structured this regulatory position to be deterministic and conservative. The 10 percent value also compensates for the lack of research concerning iodine speciation beyond the containment and the uncertainties of applying laboratory data to the post-accident environment of the plant. Regulatory Position 5.5 states that a smaller flash fraction could be justified. Some licensees have referenced NUREG/CR-5950, "Iodine Evolution and pH Control" (Ref. 10) to justify a smaller flash fraction. However, NUREG/CR-5950 was developed for very specific laboratory conditions and the results have a degree of uncertainty. The mechanism for release of the fluid is also uncertain. Leaked fluid may spray onto surfaces and evaporate, or be sprayed in fine droplets into the air. A value of less than 10 percent can be justified by including considerations for plant-specific variables, including the post-accident environment (e.g., impurities in the water or the presence of organic substances) and the uncertainties in the application of research situations to plant environments.

Figure 3.1 in NUREG/CR-5950 can be used to quantify the amount of elemental iodine as a function of the sump water pH and the concentration of iodine in the solution. In some cases, however, licensees have misapplied this figure. Rather than using the total concentration of

iodine (i.e., stable and radioactive), licensees based their assessment on only the radioactive iodine in the sump water. By using only the radioactive iodine, licensees have underestimated how much iodine evolves during post-accident conditions.

## 6. Release Pathways

Changes to the plant configuration associated with an LAR (e.g., an "open" containment during refueling) may require a re-analysis of the design basis dose calculations. A request for TS modifications allowing containment penetrations (i.e., personnel air lock, equipment hatch) to be open during refueling cannot rely on the current dose analysis if this analysis has not already considered these release pathways. RG 1.194, Regulatory Position 3.2.4.2 supports review of penetration pathways, by stating that "leakage is more likely to occur at a penetration, [and that the] analysts must consider the potential impact of leakage from building penetrations exposed to the environment." Therefore, releases from personnel air locks and equipment hatches exposed to the environment and containment purge releases prior to containment isolation need to be addressed.

Some licensees have identified unique release pathways that had not been previously considered. For example, a recent submittal noted that containment hatches and containment plugs may be removed during refueling. The removal of these barriers creates new release pathways. Licensees are responsible for identifying all release pathways and for considering these pathways in their AST analyses, consistent with any proposed modification.

## 7. Primary to Secondary Leakage

Some analysis parameters can be affected by density changes that occur in the process steam. The NRC staff continues to find errors in LAR submittals concerning the modeling of primary to secondary leakage during a postulated accident. This issue is discussed in Information Notice (IN) 88-31, "Steam Generator Tube Rupture Analysis Deficiency," (Ref. 11) and Item 3.f in RIS 2001-19. An acceptable methodology for modeling this leakage is provided in Appendix F to RG 1.183, Regulatory Position 5.2.

## 8. Elemental Iodine Decontamination Factor (DF)

Appendix B to RG 1.183, provides assumptions for evaluating the radiological consequences of a fuel handling accident. If the water depth above the damaged fuel is 23 feet or greater, Regulatory Position 2 states that "the decontamination factors for the elemental and organic [iodine] species are 500 and 1, respectively, giving an overall effective decontamination factor of 200." However, an overall DF of 200 is achieved when the DF for elemental iodine is 285, not 500.

## 9. Isotopes Used in Dose Assessments

For some accidents (e.g., main steamline break and rod drop), licensees have excluded noble gas and cesium isotopes from the dose assessment. The inclusion of these isotopes should be addressed in the dose assessments for AST implementation.

## 10. Definition of Dose Equivalent $^{131}$I

In the conversion to an AST, licensees have proposed a modification to the TS definition of dose equivalent $^{131}$I. Some have modified the definition to base it upon the thyroid dose conversion factors of International Commission on Radiation Protection (ICRP) Publication 2,

"Report of Committee II on Permissible Dose for Internal Radiation" (Ref. 12) or ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 13). Others have proposed a definition which is a combination of different iodine dose conversion factors, (e.g., RG 1.109, Revision 1, "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" (Ref. 14), ICRP Publication 2, Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 15). Although different references are available for dose conversion factors, the TS definition should be based on the same dose conversion factors that are used in the determination of the reactor coolant dose equivalent iodine curie content for the main steamline break and steam generator tube rupture accident analyses.

## 11. Acceptance Criteria for Off-Gas or Waste Gas System Release

As part of full AST implementation,[2] some licensees have included an accident involving a release from their off-gas or waste gas system. For this accident, they have proposed acceptance criteria of 500 millirem (mrem) total effective dose equivalent (TEDE).

The acceptance criteria for this event is that associated with the dose to an individual member of the public as described in 10 CFR Part 20, "Standards for Protection Against Radiation."[3] When the NRC revised 10 CFR Part 20 to incorporate a TEDE dose, the offsite dose to an individual member of the public was changed from 500 mrem whole body to 100 mrem TEDE. Therefore, any licensee who chooses to implement AST for an off-gas or waste gas system release should base its acceptance criteria on 100 mrem TEDE. Licensees may also choose not to implement AST for this accident and continue with their existing analysis and acceptance criteria of 500 mrem whole body.

## 12. Containment Spray Mixing

Some plants with mechanical means for mixing containment air have assumed that the containment fans intake air solely from a sprayed area and discharge it solely to an unsprayed region or vice versa. Without additional analysis, test measurements or further justification, it should be assumed that the intake of air by containment ventilation systems is supplied proportionally to the sprayed and unsprayed volumes in containment.

## BACKFIT DISCUSSION

This RIS provides guidance for licensees who voluntarily request to amend their licenses to implement an AST. Accordingly, this RIS is not a backfit under 10 CFR 50.109, and the NRC staff did not prepare a backfit analysis.

## *FEDERAL REGISTER* NOTIFICATION

A notice of opportunity for public comment was not published in the *Federal Register* because this RIS is informational and pertains to NRC staff positions that do not represent departures from current regulatory requirements and practice.

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[2] An off-gas or waste gas system release does not need to be addressed for a full AST implementation unless a design change is being proposed for the waste gas tank or systems at the same time.
[3] Branch Technical Position ETSB 11-5, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (Ref. 16).

**SMALL BUSINESS REGULATORY ENFORCEMENT FAIRNESS ACT OF 1996**

In accordance with the Small Business Regulatory Enforcement Fairness act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of the Office of Management and Budget (OMB).

**PAPERWORK REDUCTION ACT STATEMENT**

This Generic Letter (Bulletin, etc.) contains information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the OMB, approval number 3150-0011.

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Please direct any questions about this matter to the technical contacts listed below, or to the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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**REFERENCES**

1. U.S. Nuclear Regulatory Commission, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests," RIS 2001-19, October 18, 2001.

2. J. Schaperow et al., “Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term,” U.S. Nuclear Regulatory Commission, AEB 98-03, December 9, 1998

3. L. Soffer et al., “Accident Source Terms for Light-Water Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, NUREG-1465, February 1995.

4. U.S. Nuclear Regulatory Commission, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants,” Regulatory Guide 1.183, July 2000.

5. U.S. Nuclear Regulatory Commission, “Control Room Habitability,” NRC Generic Letter 2003-01, June 12, 2003.

6. U.S. Nuclear Regulatory Commission, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” Regulatory Guide 1.194, June 2003.

7. U.S. Nuclear Regulatory Commission, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” Regulatory Guide 1.145, Revision 1, November 1982.

8. U.S. Nuclear Regulatory Commission, “Onsite Meteorological Programs,” Regulatory Guide 1.23, February 1972.

9. T.J. Bander, “PAVAN:  An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations,” U.S. Nuclear Regulatory Commission, NUREG/CR-2858, November 1982.

10. E.C. Beahm, et al., “Iodine Evolution and pH Control,” U.S. Nuclear Regulatory Commission, NUREG/CR-5950, December 1992.

11. U. S. Nuclear Regulatory Commission, "Steam Generator Tube Rupture Analysis Deficiency,” IN 1988-31, May 25, 1988.

12. International Commission on Radiation Protection, “Report of Committee II on Permissible Dose for Internal Radiation,” ICRP Publication 2, 1959.

13. International Commission on Radiation Protection, “Limits for Intakes of Radionuclides by Workers,” ICRP Publication 30, 1978.

14. U.S. Nuclear Regulatory Commission, “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,” Regulatory Guide 1.190, Revision 1, October 1977.

15. K.F. Eckerman et al., “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” Federal Guidance Report 11, Environmental Protection Agency, EPA-520/1-88-020,1988.

16. U.S. Nuclear Regulatory Commission, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800, Branch Technical Position ETSB 11-5, July 1981.

## RECENTLY ISSUED REGULATION ISSUE SUMMARIES

1. USNRC, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests," RIS 01-019, October 18, 2001