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

October 18, 2001

NRC REGULATORY ISSUE SUMMARY 2001-19: DEFICIENCIES IN THE DOCUMENTATION OF DESIGN BASIS RADIOLOGICAL ANALYSES SUBMITTED IN CONJUNCTION WITH LICENSE AMENDMENT REQUESTS

ADDRESSEES

All holders of operating licenses for power reactors.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to inform addressees of inadequacies in licensees' documentation of design basis accident (DBA) radiological analyses in license amendment submittals. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate. However, suggestions contained in this RIS are not NRC requirements; therefore, no specific action or written response is required.

BACKGROUND INFORMATION

Under Section 50.59 of Title 10 of the Code of Federal Regulations (10 CFR 50.59), a licensee may make changes to a nuclear facility without prior NRC approval. Changes made under 10 CFR 50.59 must meet certain criteria and must not involve a revision to a technical specification. Revisions to technical specifications and proposed changes that do not meet the criteria of 10 CFR 50.59 are submitted under 10 CFR 50.90 for NRC approval. Under 10 CFR 50.90, a licensee is required to fully describe the changes desired and to follow, as far as applicable, the format prescribed for original applications. The evaluation of postulated radiological consequences often constitutes a significant portion of the safety analyses performed in support of the proposed license amendment. In reviewing these submittals, the NRC staff considers the licensee's description of the analyses performed, the assumptions and inputs, the methodology, and the results obtained. The NRC staff often finds that licensees submit insufficient information for an adequate review. Also, in some cases, the NRC staff has identified deficiencies in analysis assumptions, inputs, and methods that had to be resolved before the amendment was approved. The purpose of this RIS is to discuss the more frequent and more significant deficiencies observed by the NRC staff.

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SUMMARY OF ISSUE

The DBAs were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Many physical processes and phenomena are represented by conservative, bounding assumptions rather than by being directly modeled. The staff has selected assumptions and models that, when used in combination, form a basis for evaluating the facility design to ensure an appropriate and prudent safety margin against unpredicted events in the course of an accident and to compensate for uncertainties in plant parameters, accident progression, human performance, radioactive material transport, and atmospheric dispersion.

1. Facility Design Basis

Radiological consequence analyses performed in support of license amendment requests should use analysis assumptions, inputs, and methods that are consistent with the current facility design basis and with current facility normal and emergency operating procedures. Licensees may take analysis credit for plant features that were included in design basis radiological consequence calculations previously approved by the NRC staff. Such credit should be taken only if assumptions related to equipment operability and performance are consistent with the facility's current design basis and current normal and emergency operating procedures. The NRC staff generally does not accept analyses that credit plant features that (a) are not safety-related, (b) are not covered by technical specifications, (c) do not meet single-failure criteria, or (d) rely on the availability of offsite power unless the assumptions were previously accepted by the NRC in a site-specific licensing action and are therefore part of the facility design basis. Design basis delays in actuation of these features should be considered, especially for those features that rely on manual operator intervention.

Generally, the NRC staff will consider an assumption made in a licensee analysis, supporting a docketed amendment request, to be part of the current design basis if the staff relied upon that assumption when evaluating whether NRC requirements were met in granting the license amendment.

2. Level of Detail in Submittals

The NRC staff reviews licensee amendment requests to ensure that the proposed change will maintain an adequate level of protection of public health and safety. The NRC staff accomplishes these reviews by evaluating the information submitted in the amendment request against the current plant design basis as documented in the Final Safety Analysis Report (FSAR), previously issued staff safety evaluation reports, regulatory guidance, other licensee commitments, and staff experience gained in considering similar requests for other plants. The NRC staff bases its finding on the acceptability of an amendment on its assessment of the licensee's analysis, since it is the licensee's analysis that becomes part of the facility's design basis. Licensees should ensure that adequate information, including analysis assumptions, inputs, and methods is presented in the submittal to support a staff assessment. The NRC staff's assessment may include performance of independent analyses to confirm the licensee's conclusions. Licensees should expect an NRC staff effort aimed at resolving critical differences between analysis assumptions, inputs, and methods used by the licensee and those deemed acceptable to the NRC staff.

Regulatory Guide (RG) 1.70 (Ref. 1) offers guidance on information to be included in accident analysis descriptions in FSARs, and may be useful in determining the minimum information that should be submitted in support of a license amendment. Additional information may be needed, depending on the particular analysis. Licensees may want to consider submitting the affected FSAR pages annotated to reflect the revised analyses and or the actual calculation documentation, in addition to the analysis summary. Licensees who submit electronic FSARS, i.e. CDs may wish to consider submitting any updates electronically and also provide a list of affected FSAR pages.

3. <u>Analysis Inputs</u>

Analysis inputs should be the most restrictive values of plant parameters selected from the range of design values possible during the specific event so that the postulated consequences of the event are maximized. It is generally inappropriate to use values characterized as “best estimates.” Licensee commitments to particular regulatory guides and standard review plan sections may establish the value of certain parameters and should continue to be used where applicable. Other considerations follow:

a. The range of values applicable during an accident may vary from accident to accident, and will likely differ from the range that applies during normal operations. For example, a loss-of-offsite-power assumption may affect ventilation system flow rates.

b. It may be necessary to use different parameter values in different portions of the analyses or to perform a sensitivity analysis to determine the limiting value. In some situations the minimum and maximum value of the range may be applicable in a single analysis. For example, the minimum containment spray flow rate is used in determining the spray removal coefficients, but the maximum flow rate may be appropriate in determining the minimum sump pH.

c. If a plant parameter is associated with a technical specification limiting condition for operation (LCO), the value specified in the technical specification should be used. If the LCO specifies a range, or a value with a tolerance band, the most restrictive value should be used. The technical specifications may also specify numeric values in surveillance requirements or action statements; for example, acceptable emergency core cooling system leakage or transient reactor coolant system (RCS) iodine concentration. These should be used where appropriate.

d. Some parameters may change value during the accident; for example, RCS temperature and pressure decrease during plant cooldown. In these cases, the calculation should either assume the most restrictive value for the entire duration or the calculation should be performed in time steps, with the appropriate parameter values used for each time step. Containment leakage should be modeled as described in RG 1.3 and 1.4.

e. For parameters based on the results of less frequent surveillance testing, for example, nondestructive testing (NDT) of steam generator tubes or efficiency testing of charcoal filters, the degradation that may occur between periodic tests should be considered in establishing the analysis value.

f. Some analysis parameters can be affected by density changes that occur in the process stream. The NRC staff has noted errors made in converting volumes and volumetric flow rates (for example, gpm) to mass units, or vice versa, particularly in analyses involving primary-to-secondary leakage (Ref. 2). Licensees may wish to avoid using volumetric units to the extent possible in these calculations. With regard to the volumetric flow rates specified as LCOs, the density used should be consistent with the density that is assumed in the surveillance procedure that demonstrates compliance with the LCO. These procedures are typically based on cooled water, not on water at RCS operating temperature and pressure. Similarly, for those pressurized-water reactors (PWRs) using alternate repair criteria (ARC), the tube burst flow rate correlations are typically based on measurements of cooled water.

4. Use of Incompatible Assumptions

Licensees should ensure that their analyses do not use assumptions that are incompatible with the accident conditions or with other assumptions. For example:

a. RG 1.3 (Ref. 3) and RG 1.4 (Ref. 4) state that 50 percent of the iodine activity released from the core during a loss-of-coolant accident (LOCA) can be assumed to instantaneously plate out on containment surfaces, leaving 25 percent of the core inventory in the containment atmosphere available for release. Later revisions of the Standard Review Plan (SRP) (Ref. 5) Section 6.5.2 identify a mechanistic treatment of plateout that can be included in the determination of the containment spray coefficients. It would not be appropriate to assume 50 percent instantaneous plateout and to incorporate mechanistic treatment plateout in the same calculation, because this would constitute double credit of iodine plateout.

b. RG 1.25 (Ref. 6) contains a footnote that the assumptions in the guide are acceptable for use if certain fuel parameters, including the amount of burnup, are not exceeded. However, some extended burnup fuel designs may exceed these parameters. NUREG/CR-5009 (Ref. 7) considers the impact of extended burnup fuel and suggests revised isotopic gap fractions for use in fuel handling accidents. Licensees should justify the use of RG 1.25 or propose alternatives if the fuel parameters specified in RG 1.25 are exceeded.

5. Analysis Source Terms

The source terms used in accident analyses should be consistent with the guidance in applicable RGs and SRPs. Several source terms are tabulated in typical FSARs, each intended for specific purposes. Licensees should ensure the proper source terms are used. For analyses performed in support of license amendment requests, the assumed core inventory data should be appropriate for the currently licensed reactor power, fuel enrichment, and fuel burnup. Reactor coolant activity should be based on the technical specification specific activity LCO, including the specified transient specific activity.

6. Atmospheric Dispersion Values

The NRC guidance on short-term atmospheric dispersion values (χ/Q) has changed over time. Many of the early plants were licensed on the basis of analyses that incorporated the conservative and simplistic dispersion methods described in RG 1.3 and RG 1.4. Most control room χ/Qs were based on the guidance of Murphy and Campe (Ref. 8), but other methods have been used. Later plants may have used the guidance in RG 1.145 (Ref. 9) for determining offsite χ/Qs. The NRC staff is currently evaluating whether the ARCON96 (Ref. 10) methodology may be used to determine control room χ/Q. Licensees should use χ/Q values previously approved by the NRC staff and documented in the FSAR. If the licensee chooses to revise the χ/Q values using a methodology different from that accepted by the NRC staff and documented in the FSAR, the amendment submittal should identify this change in methodology and present sufficient information for the staff to make a determination regarding the acceptability of the revised values. Meteorological data used in the offsite and control room assessments should meet the guidance of Regulatory Position C.1.1 of RG 1.145.

7. Control Room Habitability

Many amendments submitted for NRC staff review address changes in the offsite dose consequences, but fail to address the impact of the increased releases on control room habitability. In approving the amendment, the NRC staff is required, under 10 CFR 50.92, to make a finding that the radiological consequences of the proposed amendment, if implemented, would comply with 10 CFR Part 100 and with 10 CFR Part 50 (Appendix A, General Design Criterion 19 (GDC 19)). Some believe that the LOCA dose consequences will be limiting for the control room because of the magnitude of the source term relative to the source term for other accidents. The NRC staff has identified several cases in which the LOCA was not the limiting accident for control room habitability. The following considerations should be evaluated in performing control room habitability analyses:

a. The control room design is often optimized for the DBA LOCA, and the protection afforded for other accident sequences may not be as advantageous. For example, in most designs, control room isolation is actuated by engineered safety feature (ESF) signals such as containment high pressure or safety injection (SI), or radiation monitors, or both. For accidents that rely on radiation monitor actuation, there may be a time delay in isolation that would not occur for the immediate SI signal that would result from a LOCA. In such cases, contaminated air would enter the control room for a longer period preceding isolation than it would for a LOCA.

b. The configuration of radiation monitors has an impact on their sensitivity. Ideally, the radiation monitors would be located outside in air ventilation intake ductwork. However, there are system designs that place the radiation monitor in recirculation ductwork or downstream of filters. There are also designs that use area radiation monitors. In these latter designs, the contaminated air continues to build up in the control room volume until the concentration is large enough to actuate the radiation monitor.

c. In some cases, control room radiation monitor setpoints may have been based on external exposure concerns, for example, 2.5 mrem/hour, rather than thyroid dose from inhalation. The airborne concentration of radioiodines will likely cause elevated thyroid doses before reaching the concentration of all radionuclides necessary to alarm the monitor. This condition is typically seen with accidents that involve a high iodine-to-noble-gas ratio, such as main steam line breaks in PWRs.

d. The distance between the control room and the release point, and the associated wind sectors, may be different for each postulated accident. These differences are usually not significant with regard to offsite doses, but may be significant for control room assessments because of the shorter distances typically involved. The χ/Q for the DBA LOCA may not be applicable to other DBAs. A ground-level release associated with a non-LOCA event may be more limiting than the elevated release associated with LOCAs at plants with secondary containments or enclosure buildings.

e. Licensees should ensure that assumptions regarding control room isolation and infiltration can be supported by appropriate test results or engineering evaluations. Twenty percent of the licensed power reactors have performed tracer gas tests of control room integrity. All of the tests performed identified as-found infiltration rates greater than those assumed in the design basis calculations.

f. The use of personal respirators or the use of potassium iodide (KI) as a thyroid prophylaxis should not be credited as a substitute for process controls or other engineering controls as discussed in 10 CFR 20.1702.

8. Dose Conversion Factors

The dose conversion factors (DCFs) used to convert release rate to doses should be appropriate for use in acute, short-term exposure situations. Whole-body doses have been traditionally based on semi-infinite cloud models, and thyroid doses have been based on DCFs presented in Technical Information Document (TID)-14844 (Ref. 11) (which are based on ICRP-2 (Ref. 12)). The NRC staff considers thyroid dose conversion factors based on ICRP-30 (Ref. 13), such as those tabulated in Federal Guidance Report 11 (Ref. 14), to be an acceptable change in methodology that does not warrant prior review. Licensees using ICRP-30 DCFs in accident calculations should consider revising the technical specification definition for dose equivalent I-131 to reflect the DCFs used. However, total effective dose equivalent (TEDE) is not an acceptable alternative in showing compliance with GDC-19 and Part 100 whole-body and thyroid dose criteria.[1]

1 Although TEDE subsumes both the whole body dose and the thyroid dose, the rule language in GDC-19 and 10 CFR 100.11 specifically identifies *whole-body* and *thyroid* doses. The staff is considering changes to GDC-19 to replace the current dose criterion with one based on TEDE. There are no current plans to revise the §100.11 guidelines due to the synergy that exists between the TID-14844 source terms and the accident dose guidelines. For further information, see the discussion at 64 *Federal Register* 12119.

For control room whole-body dose estimates, it is common to adjust the semi-infinite cloud DCF to account for the finite size of the control room. This correction is not applied to beta skin dose estimates since the range of beta particles in air is less than the typical control room dimensions. It is important to note that the skin dose DCFs presented in the recent literature (e.g., Federal Guidance Report 12 (Ref. 15)) are based on both photon and beta emissions. Without the geometry correction, the photon dose component will be over-estimated. If the geometry correction is included, the beta dose component will be under-estimated. DOE/EH-0070 (Ref. 16) tabulates the beta and photon skin dose DCFs separately. Licensees should ensure that the DCF's used are appropriate for the intended use.

BACKFIT DISCUSSION

This RIS does not require any modification to plant structures, systems, components, or design of facilities, or action or written response; therefore, the staff did not perform a backfit analysis or require OMB clearance.

FEDERAL REGISTER NOTICE

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

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not request any information collection.

If you have any questions about this matter, please contact one of the persons listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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Attachments:
1. List of References
2. List of Recently Issued Regulatory Issue Summaries

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LIST OF REFERENCES

1. USNRC, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” RG 1.70, 1978.

2. USNRC, “Steam Generator Tube Rupture Analysis Deficiency,” IN 88-13, May 25,1988.

3. USNRC, “Assumptions Used for Evaluation the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors,” RG 1.3, 1974.

4. USNRC, “Assumptions Used for Evaluating the Potential Radiological Consequence of a Loss of Coolant Accident for Pressurized Water Reactors,” RG 1.4, 1974.

5. USNRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800, 1987.

6. USNRC, “Assumptions Used for evaluation the Potential Radiological Consequences of a fuel Handling Accident in the Fuel Handling and Storage facility for Boiling and Pressurized Water Reactors,” RG 1.25, 1972.

7. USNRC, “Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors,” NUREG/CR-5009, 1988.

8. K.G. Murphy and K.W. Campe, “Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19,” published in proceedings of the 13th AEC Air Cleaning Conference.

9. USNRC, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” RG 1.145, 1982.

10. J.V. Ramsdell and C.A. Simonen, “Atmospheric Relative Concentrations in Building Wakes,” NUREG/CR-631, Revision 1, 1997.

11. J.J. DiNunno, et al., “Calculation of distance factors for Power and Test Reactors Sites,” USAEC TID-14844, 1962.

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

13. ICRP, “Limits for Intakes of Radionuclides by workers,” ICRP Publication 30, 1978.

14. Eckerman, K.F., et al., “Limiting values of Radionuclide Intake and Air Concentration and dose Conversion factors for Inhalation, submersion, and Ingestion; Federal Guidance Report 11,” EPA-520/1-88-020, 1988.

15. K.F. Eckerman and J.C. Ryman, “External Exposure to Radionuclides in Air, Water, and Soil, Federal Guidance Report 12,” EPA-402-R-93-081.

16. USDOE, “External Dose-Rate Conversion Factors for Calculation of Dose to the Public,” DOE/EH-0070.

LIST OF RECENTLY ISSUED
NRC REGULATORY ISSUE SUMMARIES

Regulatory Issue Summary No.	Subject	Date of Issuance	Issued to
2001-18	Requirements for Oath or Affirmation	08/22/2001	All holders of construction permits or operating licenses for nuclear power reactors and non-power reactors under Part 50 of Title 10 of the Cod of Federal Regulations (10 CFR Part 50), including those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel, holders of licenses issued under 10 CFR Part 72, and holders of certificates issued under 10 CFR Part 76
2001-17	Preparation and Scheduling of Operator Licensing Examinations	08/22/2001	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
2001-16	Update of Evacuation Time Estimates	08/01/2001	All holders of operating licenses for nuclear power plants
2001-15	Performance of DC-Powered Motor-Operated Valve Actuators	08/01/2001	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
2001-14	Position on Reportability Requirements for Reactor Core Isolation Cooling System Failure	07/19/2001	All holders of boiling-water reactor (BWR) operating licenses

OL = Operating License
CP = Construction Permit