NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

6.2.1.3 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED LOSS-OF-COOLANT ACCIDENTS (LOCAs)

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of containment systems

Secondary - None

I. AREAS OF REVIEW

The analyses of the mass and energy release are reviewed to assure that the data used to evaluate the containment and subcompartment functional design are acceptable for that purpose. The review includes the following areas:

1. The energy sources that are available for release to the containment.

2. The mass and energy release rate calculations for the initial blowdown phase of the accident.

3. For pressurized water reactor (PWR) plants, because of the additional steam generator stored energy available for release, the mass and energy release rate calculations for the core reflood and post-reflood phases of the accident.

Rev. 2 - January 2006

4. Inspection, Test, Analysis, and Acceptance Criteria (ITAAC)

For design certification and combined license reviews, the applicant's proposed information on the ITAAC associated with the systems, structures, and components (SSCs) related to this SRP section is reviewed in accordance with SRP Section 14.3. It is recognized that the review of ITAAC is performed after review of the application against acceptance criteria contained in this SRP section. Furthermore, the ITAAC are reviewed to assure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP 14.3.

Review Interfaces

The related SRP section is identified by functional relationship.

Review of the acceptability of piping design criteria, selected break locations and break sizes based on the provisions made to limit pipe motion, for breaks postulated to occur within subcompartments is performed under SRP Section 3.6.2.

The acceptance criteria and review procedures are contained in the referenced SRP section.

II. ACCEPTANCE CRITERIA

The acceptance criteria given below apply to the mass and energy release analysis for postulated LOCAs. If the mass and energy analysis complies with the relevant requirements of General Design Criterion 50 and 10 Code of Federal Regulations (CFR) Part 50, Appendix K, paragraph I.A, then the staff considers the analysis acceptable. The relevant requirements are as follows:

A. General Design Criterion 50, as it relates to the containment and subcompartments being designed with sufficient margin, requires that the containment and its associated systems can accommodate, without exceeding the design leakage rate, and the containment and subcompartment design can withstand the calculated pressure and temperature conditions resulting from any LOCA.

B. 10 CFR Part 50, Appendix K, as it relates to sources of energy during the LOCA, provides requirements to assure that all the energy sources have been considered.

In meeting the requirements of General Design Criterion 50 the following specific criterion or criteria that pertain to the mass and energy analysis are used as included below:

1. Sources of Energy

The sources of stored and generated energy that should be considered in analyses of LOCAs include: reactor power; decay heat; stored energy in the core; stored energy in the reactor coolant system metal, including the reactor vessel and reactor vessel internals; metal-water reaction energy; and stored energy in the secondary system (PWR plants only), including the steam generator tubing and secondary water.

Calculations of the energy available for release from the above sources should be done in general accordance with the requirements of 10 CFR Part 50, Appendix K, paragraph I.A. However, additional conservatism should be included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA. An example of this would be accomplished by maximizing the sensible heat stored in the reactor coolant system (RCS) and steam generator metal and increasing the RCS and steam generator secondary mass to account for uncertainties and thermal expansion.

The requirements of paragraph I.B in Appendix K to 10 CFR Part 50, concerning the prediction of fuel clad swelling and rupture should not be considered. This will maximize the energy available for release from the core.

2. Break Size and Location

   a. The staff's review of the applicant's choice of break locations and types is discussed in SRP Section 3.6.2.

   b. Of several breaks postulated on the basis of a., above, the break selected as the reference case for subcompartment analysis should yield the highest mass and energy release rates, consistent with the criteria for establishing the break location and area.

   c. Containment design basis calculations should be performed for a spectrum of possible pipe break sizes and locations to assure that the worst case has been identified.

3. Calculations

   In general, calculations of the mass and energy release rates for a LOCA should be performed in a manner that conservatively establishes the containment internal design pressure (i.e., maximizes the post-accident containment pressure and the containment subcompartment response). The criteria given below for each phase of the accident indicate the conservatism that should exist.

   a. Subcompartment Analysis

      The analytical approach used to compute the mass and energy release profile will be accepted if both the computer program and volume noding of the piping system are similar to those of an approved emergency core cooling system (ECCS) analysis. The computer programs that are currently acceptable include SATAN-V (Reference 1), CRAFT-2 (Reference 2), CE FLASH-4 (Reference 3), and RELAP4 (Reference 4), when a flow multiplier of 1.0 is used with the applicable choked flow correlation. An alternate approach, which is also acceptable, is to assume a constant blowdown profile using the initial conditions with an acceptable choked flow correlation.

b. Initial Blowdown Phase Containment Design Basis

The initial mass of water in the reactor coolant system should be based on the reactor coolant system volume calculated for the temperature and pressure conditions assuming that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error). An assumed power level lower than the level specified (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.

Mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data.

Calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. For surfaces exposed to steam, heat transfer calculations should be based on forced convection.

Calculations of heat transfer from the secondary coolant to the steam generator tubes for PWRs should be based on natural convection heat transfer for tube surfaces immersed in water and condensing heat transfer for the tube surfaces exposed to steam.

c. PWR Core Reflood Phase (Cold Leg Breaks Only)

Following initial blowdown, which includes the period from the accident initiation (when the reactor is in a steady-state full power operation condition) to the time that the reactor coolant system broken loop pressure equalizes to the containment pressure, the water remaining in the reactor vessel should be assumed to be saturated. Justification should be provided for the refill period, which is the time from the end of the blowdown to the time when the emergency core cooling system (ECCS) refills the vessel lower plenum. An acceptable approach is to assume a water level at the bottom of the active core at the end of blowdown so there is no refill time.

Calculations of the core flooding rate should be based on the ECCS operating condition during the core reflood phase, which begins when the water starts to flood the core and continues until the core is completely quenched, or the post-reflood phase, which is the period after the core has been quenched and energy is released to the RCS primary system by the RCS metal, core decay heat, and the steam generators, that maximizes the containment pressure.

Calculations of liquid entrainment, i.e., the carryout rate fraction, which is the mass ratio of liquid exiting the core to the liquid entering the core, should be based on the PWR full length emergency cooling heat transfer experiments (Reference 5). Liquid entrainment should be assumed to continue until the water level in the core is 61 cm (2 feet) from the top of the core. An acceptable approach is to assume a carryout rate fraction (CRF) of 0.05 to the 46 cm (18-inch) core level, a linearly increasing CRF to 0.80 at the 61 cm (24-inch) level, and a constant CRF of 0.80 until the water level is 61 cm (2 feet) from the top of the core. Above this level, a CRF of 0.05 may be used.

The assumption of steam quenching should be justified by comparison with applicable experimental data. Liquid entrainment calculations should consider the effect on the CRF of the increased core inlet water temperature caused by steam quenching assumed to occur from mixing with the ECCS water.

Steam leaving the steam generators should be assumed to be superheated to the temperature of the secondary coolant.

d. <u>PWR Post-Reflood Phase</u>

All remaining stored energy in the primary and secondary systems should be removed during the post-reflood phase.

Steam quenching should be justified by comparison with applicable experimental data.

The results of post-reflood analytical models should be compared to applicable experimental data.

e. <u>PWR Decay Heat Phase</u>

The dissipation of core decay heat should be considered during this phase of the accident. The fission product decay energy model is acceptable if it is equal to or more conservative than the decay energy model given in Branch Technical Position ASB 9-2 in SRP Section 9.2.5.

Steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with ECCS injection water.

The following methods and computer models are acceptable for calculating the mass and energy releases for containment design basis calculations:

Babcock and Wilcox / Framatome ANP: CRAFT (Reference 6), CRAFT-2 (Reference 2), RELAP5/MOD2-B&W, Revision 1 (Reference 7), and RELAP5/MOD2-B&W, Revision 4 (Reference 8).

Combustion Engineering: CEFLASH-4A (Reference 3), and CESSAR System 80 (Reference 9 and 10).

General Electric: M3CPT (Reference 11), NEDO-20533 (Reference 12), and SHEX (Reference 13, 14, and 15).

Westinghouse: WCAP-8312 (Reference 16), SATAN-V (Reference 1), WCAP-10325 (Reference 17), SATAN-VI (Reference 18), and WREFLOOD (Reference 19).

Theses codes and methods have been referenced in licensee submittals and on a case by case basis have been found to be acceptable for these purposes.

Other methods will be acceptable if they are found to be conservative for these calculations.

C. 10 CFR 52.47(a)(1)(vi) provides the requirement for ITAAC for design certification reviews

D. 10 CFR 52.97(b)(1) provides the requirement for ITAAC for combined license reviews

Technical Rationale

The technical rationale for application of the above acceptance criteria to the mass and energy release analysis for postulated LOCAs is discussed in the following paragraphs.

1. GDC 50 requires the containment structure and associated heat removal system to be designed with margin to accommodate any LOCA such that the containment design leak rate is not exceeded. A LOCA potentially causes the greatest pressure surge and release of fission products when compared to any other accident. Since it is the most severe challenge expected, containment must be designed to definitively withstand this accident. Following GDC 50 will ensure that containment integrity is maintained under the most severe accident conditions thus precluding the release of radioactivity to the environment.

2. Appendix K to 10 CFR 50 provides required and acceptable features of evaluation models used to analyze various circumstances applicable to the ECCS. Section I.A of Appendix K provides a comprehensive list of LOCA heat (energy) sources and the reactor operating history assumptions associated with those heat sources. Since the mass and energy release analysis for postulated LOCAs is used to design containment and containment subcompartments such that they will withstand the worst case LOCA, it is critical that all potential energy sources are taken into account. Following 10 CFR 50 Appendix K will ensure that containment and containment subcompartments are designed to accommodate all energy sources for the worst case LOCA, thus precluding the potential release of radioactivity to the environment following such a LOCA.

III. REVIEW PROCEDURES

The procedures described below are followed for the review of the mass and energy release analysis for LOCAs. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis or by applying the results of previous reviews of similar plants.

The reviewer confirms, with the lead reviewer for SRP Section 3.6.2, the validity of the applicant's analysis of pipe break size, type and locations for subcompartments containing high energy lines by using elevation and plan drawings of the containment showing the routing of lines containing high energy fluids. The reviewer determines that an appropriate reference case for subcompartment analysis has been identified. In the event a pipe break other than a double-ended pipe rupture is postulated by the applicant, the lead reviewer for SRP Section 3.6.2 will evaluate the applicant's justification for assuming a limited displacement pipe break.

The reviewer compares the sources of energy considered in the loss-of-coolant analysis and the methods and assumptions used to calculate the energy available for release from the various sources with the acceptance criteria listed in Section II, above. The reviewer determines the acceptability of the analytical models and the assumptions used to calculate the rates of mass and energy release during the initial blowdown, core reflood, and post-reflood phases of a LOCA. The reviewer also compares energy inventories at various times during a LOCA to ensure that the energy from the various sources has been accounted for and has been transferred to the containment on an appropriate time scale.

The reviewer reviews comparisons made by the applicant to experimental data and makes comparisons to other available experimental data to determine the amount of conservatism in the mass and energy release models.

The reviewer may perform confirmatory analyses of the mass and energy profiles. The purpose of the analysis is to confirm the predictions of the mass and energy release rates appearing in the safety analysis report, and to confirm that an appropriate break location has been considered in these analyses.

For reviews under 10 CFR Part 52, the reviewer should follow the above procedures to verify that the design set forth in the safety analysis report, and as applicable, site interface requirements, and combined license action items, meet the acceptance criteria. Following this review, SRP Section 14.3 should be followed for the review of Tier I information for the design, including the site parameters, interface criteria, and ITAAC.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this SRP section are presented in SRP Section 6.2.1.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plan for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of design certifications and license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52. Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.

VI. REFERENCES

The references for this SRP section are listed in SRP Section 6.2.1.

(As this is the first SRP section of 6.2.1 being updated, references, in sequence, are provided here for completeness)

1. WCAP-7750, "SATAN-V, A Computer Space Time Dependent Analysis of Loss of Coolant," August 1971.

2. BAW-10092, "CRAFT-2 Fortran Program for Digital Simulation of a Multinode Reactor Plant During a Loss-of-Coolant Accident," Babcock and Wilcox Company, December 1974.

3. CENPD-133 Supplement 5-A, "CEFLASH-4A — A Fortran77 Digital Computer Program for Reactor Blowdown Analysis," dated June 1985.

4. W. H. Retting, G. A. Jayne, K. Y. Moore, C. E. Slater, and M. L. Uptmor, "RELAP3 - A Computer Program for Reactor Slowdown Analysts," IN-1321, Idaho Nuclear Corporation, June 1970.

5. F. C. Cadek, et al., "PWR FLECHT (Full Length Emergency Cooling Heat Transfer), Final Report," UCAP-7665, Westinghouse Electric Corporation, April 1971.

6. BAW-10030, "CRAFT - Description of Model for Equilibrium LOCA Analysis Program," Babcock and Wilcox, Lynchburg, VA, October 1971.

7. B&W-10164P, Revision 1, "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," Babcock and Wilcox, October 1988.

8. BAW-10164P-A, Revision 4, "RELAP5-MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," November 2002.

9. Combustion Engineering letter DP-456, F. M. Stern to E. Case, dated August 19, 1974, Chapter 6, Appendix 6B to CESSAR System 80 PSAR.

10. "NRC Safety Evaluation Report - Standard Reference System, CESSAR System 80," Combustion Engineering Inc., December 1975.

11. NEDO-10320, "The General Electric Pressure Suppression Containment Analytical Model," General Electric Company, April 1971; Supplement 1, May 1971; Supplement 2, January 1973.

12. NEDO-20533, "The General Electric Mark III Pressure Suppression Containment System Analytical Model," June 1974.

13. “SHEX Model Description,” PDR ADOCK 050000321, 9807130260, 980706. (Proprietary, not available for public release)

14. NEDE-30911, “SHEX-04 User's Manual,” Class II, General Electric Company, August 1985. And “SHEX-04V User's Manual (Addendum to SHEX-04 User's Manual),” NEDE-30911-1, June 1994.

15. “Use of SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis,” NRC Letter from A. Thadani (NRC) to G. L. Sozzi (GE), July 13, 1993.

16. WCAP-8264-P-A, “Westinghouse Mass and Energy Release Data for Containment Design,” Westinghouse Electric Corporation, March 1974 (Proprietary, not available for public release). WCAP-8312-A (Non-Proprietary).

17. WCAP-10325-P-A “Westinghouse LOCA Mass and Energy Release Model For Containment Design - March 1979 Version,” May 5, 1983 (Proprietary, not available for public release), WCAP-10326-A, May 5, 1983 (Non-Proprietary).

18. WCAP-8302, “SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant,” June 1974 (Proprietary, not available for public release). WCAP-8306 (Non-Proprietary).

19. WCAP-8170, “Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code),” June 1974 (Proprietary, not available for public release). WCAP-8171 (Non-Proprietary).

SRP Draft Section 6.2.1.3
Attachment- Description of Changes

The following summarizes the changes from Revision 1, dated July 1981. Minor editorial changes and formatting changes are not identified by side bars.

1. Review Responsibilities:

   a. Editorial change made to reflect the current primary review function – organization maintained separate from the SRP section.

2. Areas of Review:

   a. Added "Review Interfaces" heading to Areas of Review. Reformatted existing description to describe the relationship to SRP Section 3.6.2.
   b. Added paragraph on ITAAC, SRP Section 14.3

3. Acceptance Criteria:

   a. Section A, changed "subcompartment" was changed to "subcompartments" to reflect the fact that a containment may have more than one subcompartment.

   b. Section B.1, provided an example of additional conservatism.

   c. Section B.3.b, modified to be consistent with measurement recapture under 10 CRF 50 Appendix K . Federal Register: June 1, 2000 (Volume 65, Number 106) [Page 34913-34921].

   d. Section B.3.c, added brief definitions for blowdown, refill, reflood and post-reflood phases of LOCA and clarified that the treatment is to address the core flooding rate for designs which are peak pressure limited during the reflood phase or during the post-reflood phase.

   e. Section B.3, provided an updated list of acceptable models.

   f. Added C and D to refer to ITAAC for design certification and combined license reviews.

   g. Technical rationale associated with GDC 50 and Appendix K to 10 CFR Part 50 was formulated and added to this section in accordance with the specified SPR-UDP format.

4. Review Procedures:

   a. Added paragraph on implementing of 10 CFR 52. A standard paragraph was added to address Review Procedures in design certification reviews.

5. Implementation:

   a. Editorial changes capturing applicability to Part 52 and time-frame in which SRP update goes into effect.

6. References:

   a. Which were provided in Section 6.2.1 are listed in this section for completeness.

   b. Added new computer code references as described above.

NRC FORM 335 (9-2004) NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-0800

2. TITLE AND SUBTITLE

NUREG-0800, Chpt 6, Section 6.2.1.3, Rev. 2, "Mass and Energy Release Analysis for Postulated Loss-of-coolant Accidents," to the Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. LWR Edition.

3. DATE REPORT PUBLISHED

MONTH	YEAR
January	2006

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Containment and Ventilation Branch

6. TYPE OF REPORT

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Risk Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of New Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The revision updates the Revision 1 July 1981 version and includes most of the changes introduced in the draft Revision 2 of April 1996. The changes consist mostly of assigning different responsibilities to the primary and secondary branches because of office reorganizations; editorial and formatting changes as part of the SRP update effort; and updating some references. Although the acceptance criteria in the revised Section 6.2.1.3 did not change and are still based on General Design Criterion (GDC) 50 and 10 CFR Part 50, Appendix K, paragraph I.A, technical rationale were developed and added for acceptance criteria associated with these GDCs, including the proposed revision to address the 65 FR 34921 change to 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," for power measurement uncertainty. The revision also adds standard paragraphs to extend application of the updated SRP section to the design certification reviews as well as to extend implementation of this section to submittals by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

NRC, Standard Review Plan, SRP, NUREG-0800, Section 6.2.1.3, Mass and Energy Release Analysis

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page) unclassified

(This Report) unclassified

15. NUMBER OF PAGES

16. PRICE