U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - See secondary review responsibilities of the seven SRP sections listed below for the various containment types and aspects.

INTRODUCTION

The CSB reviews information regarding the functional capability of the reactor containment presented in Section 6.2.1 of the applicant's safety analysis report (SAR). The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant, steam line or feedwater line break accidents. The containment structure must also maintain functional integrity in the long term following a postulated accident; i.e., it must remain a low leakage barrier against the release of fission products for as long as postulated accident conditions require.

The design and sizing of containment systems are largely based on the pressure and temperature conditions which result from release of the reactor coolant in the event of a loss-of-coolant accident (LOCA). The containment design basis includes the effects of stored energy in the reactor coolant system, decay energy, and energy from other sources such as the secondary system, and metal-water reactions including the recombination of hydrogen and oxygen. The containment system is not required to be a complete and independent safeguard against a LOCA by itself, but functions to contain any fission products released while the emergency core cooling system cools the reactor core.

The evaluation of a containment functional design includes calculation of the various effects associated with the postulated rupture in the primary or secondary coolant system piping. The subsequent thermodynamic effects in the containment resulting from the release of the coolant mass and energy are determined from a solution of the incremental space and time-dependent energy, mass, and momentum

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Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D C. 20555.

equations. The basic functional design requirements for containment are given in General Design Criteria 16 and 50 in Appendix A to 10 CFR Part 50. General Design Criterion 50, among other things, requires that consideration be given to the potential consequences of degraded engineered safety features, such as the containment heat removal system and the emergency core cooling system, the limitations in defining accident phenomena, and the conservatism of calculational models and input parameters, in assessing containment design margins.

General Design Criteria 52 and 53 provide design requirements to assure that the design can accommodate a periodic integrated leakage rate testing at design pressures, and to assure that the design permits periodic inspections and appropriate surveillance programs. The basic functional design requirements for a leak tight containment barrier for piping systems penetrating the primary reactor containment are given in General Design Criteria 54 thru 57. The General Design Criteria provide design requirements for the installation of containment isolation valves on piping lines that penetrate the containment barrier.

There are a number of different containment types and designs, and several aspects of containment functional design that are within the scope of SAR Section 6.2.1. The various containment types and aspects to be reviewed under this SRP section have been separated and assigned to a set of other SRP sections as follows:

1. Pressurized water reactor (PWR) dry containments, including sub-atmospheric containments (SRP Section 6.2.1.1.A).

2. Ice condenser containments (SRP Section 6.2.1.1.B).

3. Mark I, II, and III boiling water reactor (BWR) pressure-suppression type containments (SRP Section 6.2.1.1.C).

4. Subcompartment analysis (SRP Section 6.2.1.2).

5. Mass and energy release analysis for postulated loss-of-coolant accidents (SRP Section 6.2.1.3).

6. Mass and energy release analysis for postulated secondary system pipe ruptures (SRP Section 6.2.1.4).

7. Minimum containment pressure analysis for emergency core cooling system (ECCS) performance capability studies (SRP Section 6.2.1.5).

A separate SRP section has been prepared for each of these areas.

Areas related to the evaluation of the containment functional capability are treated in other SRP sections; e.g., Containment Heat Removal (SRP Section 6.2.2), Containment Isolation System (SRP Section 6.2.4), Combustible Gas Control (SRP Section 6.2.5), and Containment Leakage Testing (SRP Section 6.2.6). In addition, the evaluation of the secondary containment functional design capability is reviewed in SRP Section 6.2.3.

I. AREAS OF REVIEW

The items reviewed are described in the "Areas of Review" subsections of the seven SRP sections listed above.

II. ACCEPTANCE CRITERIA

The acceptance criteria are given in the "Acceptance Criteria" subsections of the seven SRP sections listed above.

III. REVIEW PROCEDURES

Review procedures are given in "Review Procedures" subsections of the seven SRP sections listed above.

IV. EVALUATION FINDINGS

The results of the reviews under the seven SRP sections listed above are consolidated into a single set of findings. The reviewer verifies that sufficient information has been provided and that the evaluation is adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

Containment Functional Design

The scope of review of the functional design of the containment for the _____ nuclear power plant has included a review of plant arrangement drawings, system drawings, and descriptive information for the containment building, subcompartments, and associated systems, components, and structures that are essential to the functional capability and integrity of the containment. The review has included the applicant's proposed design bases for the containment building and internal structures, and associated structures and systems upon which the containment function depends, and the applicant's analysis of postulated accidents and operational occurrences which support the adequacy of the design bases.

The basis for the staff's acceptance has been conformance of designs and design bases for the containment building, internal structures, and associated systems, components, and structures to the Commission's regulations as set forth in the general design criteria, and to applicable regulatory guides, branch technical positions, and industry codes and standards. (Special problems or exceptions that the staff takes to the design or functional capability of containment structures, systems, and components should be discussed.)

To support the basis for the staff's acceptance of the containment system, the reviewer of the containment system should include in the staff's safety evaluation report, as necessary, the results of the reviews for the seven SRP sections above. The SER writeup should demonstrate conformance with the Commission regulations in the manner indicated. The staff concludes that the containment functional design is acceptable and meets the requirements of General Design Criteria 16, 50, 52 and 53. The conclusion is based on the following: [The reviewer should discuss each item of the regulations or related set of regulations as indicated.]

1. The applicant has met the requirements of (cite regulation) with respect to (state limits of review in relation to regulation) by (for each item that is applicable to the review, state how it was met and why acceptable with respect to regulation being discussed):

a. meeting the regulatory positions in Regulatory Guide _____ or Guides;

b. providing and meeting an alternative method to regulatory positions in Regulatory Guide _____; that the staff has reviewed and found to be acceptable;

c. meeting the regulatory position in BTP;

d. using calculational methods for (state what was evaluated) that has been previously reviewed by the staff and found acceptable; the staff has reviewed the impact parameters in this case and found them to be suitably conservative or performed independent calculations to verify acceptability of their analysis; and/or

e. meeting the provisions of (industry standard number and title) that has been reviewed by the staff and determined to be appropriate for this application.

2. Repeat discussion for each regulation cited above.

3. The temperature/pressure profiles provided in the Final Safety Analysis Report for the spectrum of LOCA and main steam line break accidents are acceptable for use in equipment qualification, i.e., there is reasonable assurance that the actual temperatures and pressures for the postulated accidents will not exceed these profiles anywhere within the specified environmental zones, except in the break zone.

V. IMPLEMENTATION

The implementation schedules are given in the "Implementation" sections of the seven SRP sections listed above.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases"; Criterion 13, "Instrumentation and Control"; Criterion 16, "Containment Design"; Criterion 38, "Containment Heat Removal"; Criterion 39, "Inspection of Containment Heat Removal System"; Criterion 40, "Testing of Containment Heat Removal System"; Criterion 50, "Containment Design Basis"; Criterion 52, "Capability for Containment Leakage Rate Testing"; Criterion 53, "Provisions for Containment Testing and Inspection"; Criterion 54, "Systems Penetrating Containment"; Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment"; Criterion 56, "Primary Containment Isolation"; Criterion 57, "Closed System Isolation Valves"; and Criterion 64, "Monitoring Radioactivity Release."

2. 10 CFR §50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

3. ASME Boiler and Pressure Vessel Code, Section II, Division 1, Subsection NE, "Class MC Components," American Society of Mechanical Engineers.

4. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors."

5. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

6. C. F. Carmichael and S. A. Marks, "CONTEMPT-PS, A Digital Computer Code for Predicting the Pressure-Temperature History Within a Pressure-Suppression Containment Vessel in Response to a Loss-of-Coolant Accident," IDO-17252, Phillips Petroleum Company, April 1969.

7. L. C. Richardson, L. J. Finnegan, R. J. Wagner, and J. M. Waage, "CONTEMPT, A Computer Program for Predicting the Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," IDO-17220, Phillips Petroleum Company, June 1967.

8. R. J. Wagner and L. L. West, "CONTEMPT-LT Users Manual," Interim Report I-214-74-12.1, Aerojet Nuclear Company, August 1973.

9. R. I. Miller, "Evaluation of the Predictive Capabilities of the CONTEMPT-PS Computer Code by Comparison of Calculated Results with the Humboldt Bay and Bodega Bay Pressure Suppression Tests," Interim Report 4.2.1.1, Idaho Nuclear Corporation, September 1970.

10. T. Tagami, "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1)," prepared for the National Reactor Testing Station, February 28, 1966 (unpublished work).

11. H. Uchida, A. Oyama, and Y. Toga, "Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors," Proc. Third International Conference on the Peaceful Uses of Atomic Energy, Volume 13, Session 3.9, United Nations, Geneva (1964).

12. "FLOOD/MOD002 - A Code to Determine the Core Reflood Rate for a PWR Plant with Two Core Vessel Outlet Legs and Four Core Vessel Inlet Legs," Interim Report, Aerojet Nuclear Company, November 2, 1972.

13. "FLOOD/MOD001 - A Code to Determine the Core Reflood Rate for a PWR Plant with Two Core Vessel Outlet Legs and Two Core Vessel Inlet Legs," Interim Report, Aerojet Nuclear Company, October 11, 1972.

14. "COMPARE: A Computer Program for the Transient Calculation of a System of Volumes Connected by Flowing Vents," LA-NUREG-6488-MS, September 1976.

15. RELAP4 MOD5, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems Users Manual, ANCR-NUREG-1335, September 1976.

16. F. J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Jour. of Heat Transfer, Trans. Am. Soc. of Mechanical Engineers, Vol. 87, No. 1, February 1965.

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

18. "Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8312, Westinghouse Electric Corporation, March 1974.

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

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

21. Final Safety Analysis Report for Donald C. Cook Nuclear Plant, Units 1 and 2, Appendices M and N, American Electric Power Company, and the Staff Safety Evaluation Report. AEC Docket Nos. 50-315/316.

22. "Ice Condenser Containment Pressure Transient Analysis Methods," WCAP-8077, Westinghouse Electric Corporation, March 1973.

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

24. Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," attached to SRP Section 6.2.1.5.

25. "Long-Term Ice Condenser Containment Code - LOTIC Code," WCAP-8355, Westinghouse Electric Corporation, April 1976. (Non-Proprietary)

26. "Long-Term Ice Condenser Containment Code - LOTIC Code," WCAP-8355 Supplement 1, Westinghouse Electric Corporation, April 1976. (Non-Proprietary)

27. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

28. NUREG-0660, Vols. 1 and 2, "NRC Action Plan Developed as a Result of the TMI-2 Accident," May 1980, Revision 1, August, 1980.

29. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979.

30. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," March 1981.

31. NRC Safety Evaluation Report, Babcock and Wilcox Company, Reference Safety Analysis Report, B-SAR-205, May 1978.

32. NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," January 1981.

33. NUREG-0694, "TMI-Related Requirements for New Operation Licenses," June 1980.

34. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following An Accident."

35. NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment."

36. U.S. Nuclear Regulatory Commission, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," USNRC Report NUREG-0487, October 1978.

37. U.S. Nuclear Regulatory Commission, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," USNRC Report NUREG-0487, Supplement 1, October 1980.

38. U.S. Nuclear Regulatory Commission, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," USNRC Report NUREG-0487, Supplement 2, February 1981.

39. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Mark I Containment Long-Term Program," USNRC Report NUREG-0661, July 1980.