U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of Containment Integrity

Secondary - None

I. AREAS OF REVIEW

The responsible staff 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.

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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 conservation equations. The basic functional design requirements for containment are given in General Design Criteria (GDC) 4, 16 and 50 in Appendix A to 10 CFR Part 50 and in 10 CFR Part 50, Appendix K. GDC 4 provides the basic environmental and dynamic effects design requirements for all structures, systems, and components important to safety including leak-before-break. GDC 16 establishes the fundamental requirement to design a containment that is essentially a leak-tight barrier against the uncontrolled release of radioactivity to the environment. GDC 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. 10 CFR Part 50 Appendix K.I.D.2 requires that the containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a conservatively low value. 10 CFR 50 Appendix K.I.A can be used as guidance in ensuring that all sources of energy are included in determining the mass and energy release from loss-of-coolant accidents and Pressurized Water Reactor (PWR) secondary system pipe ruptures.

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, III, Boiling Water Reactor (BWR), Advanced Boiling Water Rector (ABWR) and Economic Simplified Boiling Water Reactor (ESBWR) 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.

The specific areas of review are described in the "Areas of Review" subsections of the seven SRP sections listed above.

II. ACCEPTANCE CRITERIA

Requirements

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

SRP Acceptance Criteria

Specific SRP acceptance criteria are provided in the referenced SRP Sections.

III. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

1. 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 4, 16, 50 and 10 CFR Part 50 Appendix K. 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 it is 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 have previously been 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.

IV. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission’s regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

V. REFERENCES

1. 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."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects19 Design Bases."

3. 10 CFR Part 50, Appendix A, General Design Criterion 13, "Instrumentation and Control."

4. 10 CFR Part 50, Appendix A, General Design Criterion 16, "Containment Design."

5. 10 CFR Part 50, Appendix A, General Design Criterion 38, "Containment Heat Removal."

6. 10 CFR Part 50, Appendix A, General Design Criterion 39, "Inspection of Containment Heat Removal System."

7. 10 CFR Part 50, Appendix A, General Design Criterion 40, "Testing of Containment Heat Removal System."

8. 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Basis."

9. 10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases."

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

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

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

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

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

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

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

17. NUREG-0661, "Safety Evaluation Report Mark I Containment Long-Term Program," July 1980.

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

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

20. 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.

21. NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor," July 1994.

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

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

24. Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance."

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

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

27. 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.

28. Branch Technical Position 6-2, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation."

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

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

31. 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).

32. 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).

33. "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.

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

35. F. C. Cadek et al., "PWR FLECHT (Full-Length Emergency Cooling Heat Transfer), Final Report," WCAP-7665, Westinghouse Electric Corporation, April I971.

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

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

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

39. "The General Electric Mark III Pressure Suppression Containment Analytical Model," NEDO-20533, General Electric Company, June 1974; Supplement 1, September 1975.

40. "Advanced Boiling Water Reactor Standard Safety Analysis Report," 23A6100 Rev. 3, General Electric Company, November 1993.

41. Code Manual for CONTAIN 2.0: A Computer Code for Nuclear reactor Containment Analysis, K.K. Murata, et al., Sandia National Laboratories, NUREG/CR-6533, December 1997.

42. WCAP-8822 (Proprietary) and WCAP-8860 (Non-proprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-s1-P-A (Proprietary) and WCAP 8860-S1-A (Non-proprietary), "Supplement 1 - Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP 8860–S2-A (Non-proprietary), "Supplement 2 - Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986.

43. GOTHIC: Containment Analysis Package User Manual, Qualification Report and Technical manual, NAI 8907

44. Letter from Anthony C. McMurtray, USNRC, to Thomas Coutu, Site Vice President, Kewaunee Nuclear Power Plant, September 29, 2003 [ADAMS Accession Number ML0326810500]

45. Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design, Volumes 1,2 and 3, NUREG-1793, US NRC, September 2004 (ML043450284)

46. Final Safety Evaluation Report Related to Certification of the AP600 Standard Design, Volumes 1,2 and 3, NUREG-1512, US NRC, September 1998

47. Ice Condenser Containment Pressure Transient Analysis Methods, WCAP-8077, Westinghouse Electric Corporation, March 1973 (Proprietary)

48. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ML003716792)

49. Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design basis Accidents at Light Water Nuclear Power Reactors," May 2003 (ML020160023)

50. Mark I Containment Program Load Definition Report, NEDO-21888 Revision 2, General Electric Company, November 1981

51. Mark III LOCA-Related Hydrodynamic Load Definition, US Nuclear Regulatory Commission, NUREG-0978, August 1984

52. Letter to Robert Pinelli, Chairman, Boiling Water Reactor Owners Group, from Gary M. Holahan, US Nuclear Regulatory Commission, Transmittal of Safety Evaluation of General Electric Co. Topical Reports: NEDO-30832 Entitled "Elimination of Limit on BWR Suppression Pool Temperature for SRV Discharge With Quenchers," and NEDO-31695 Entitled "BWR Suppression Pool Temperature Technical Specification Limits," August 29, 1994

53. J.G.M. Andersen, et al., "TRACG Model Description," GE Energy Nuclear, NEDE-32176P, Revision 3, April 2006

54. SRP Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping."

55. NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria."

56. NUREG-0661, Supplement 1, "Mark I Containment Long Term Program."

57. NUREG-0763, "Guidelines for Confirmatory In-plant Tests of Safety-Relief Discharge for BWR Plants."

58. NUREG-0802, "Safety/Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments."

59. NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments."

60. NUREG-0978, "Mark III LOCA-Related Hydrodynamic Load Definition."

61. NRC Generic Letter 88-17, “Loss of Decay Heat Removal,” US NRC, October 17, 1988.