**NUREG-0800**
**(Formerly NUREG-75/087)**

**U.S. NUCLEAR REGULATORY COMMISSION**

# STANDARD REVIEW PLAN

**OFFICE OF NUCLEAR REACTOR REGULATION**

## 6.2.1 CONTAINMENT FUNCTIONAL DESIGN

REVIEW RESPONSIBILITIES

Primary - Containment Systems and Severe Accident Branch (SCSB)[1]

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

INTRODUCTION

The SCSB[2] 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 equations. The basic functional design requirements for containment are given in General Design Criteria 4, 16 and 50 in Appendix A to 10 CFR Part 50 and in 10 CFR 50.46.[3] General Design Criterion 4 provides the basic environmental and dynamic effects design requirements for all structures, systems, and components important to safety.[4] General Design 16 establishes the fundamental requirement to design a containment that is essentially a leak-tight barrier against the release of radioactivity to the environment.[5] 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~~,~~[6] in assessing containment design margins. 10 CFR 50.46 provides methods and criteria for the analysis and design of emergency core cooling systems.[7]
~~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.~~[8]

For new plant applicants and those PWRs subject to the guidance contained in reference 45 (Generic Letter 88-17), the containment analyses should also consider shutdown conditions, when appropriate, to ensure that a basis is provided for procedures, instrumentation, operator response, equipment interactions and equipment response. The analyses should encompass shutdown thermodynamic states and physical configurations to which the plant can be subjected during shutdown conditions (such as time to core uncovery during a loss of shutdown decay heat removal capability) and should provide sufficient depth such that adequate bases can be developed (see Reference 46).[9]

There are a number of different containment types and designs~~,~~[10] 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, and ABWR[11] 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 the[12] "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 4,[13] 16, and 50, ~~52 and 53~~[14]and 10 CFR 50.46.[15] 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~~ have previously been[16] 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.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[17]

## V. IMPLEMENTATION

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

## VI. REFERENCES[18]

~~2~~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."

~~1~~2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and ~~Missile~~Dynamic Effects[19] Design Bases."~~;~~[20]

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."~~; 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~~[21]

9. 10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases."[22]

~~5~~10. RELAP4 MOD5, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems Users Manual, ANCR-NUREG-1335, September 1976.[23]

~~36~~11. U.S. Nuclear Regulatory Commission, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," USNRC Report NUREG-0487, October 1978.

~~37~~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.

~~38~~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.

~~35~~14. NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment."

~~32~~15. NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," January 1981.

~~14~~16. "COMPARE: A Computer Program for the Transient Calculation of a System of Volumes Connected by Flowing Vents," LA-NUREG-6488-MS, September 1976.[24]

~~39~~17. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Mark I Containment Long-Term Program," USNRC Report NUREG-0661, July 1980.

~~30~~18. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," March 1981.

~~27~~19. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

20. U.S. Nuclear Regulatory Commission, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor," USNRC Report NUREG-1503, July 1994.[25]

~~4~~21. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors."

~~34~~22. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following An Accident."[26]

23. Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance."[27]

~~31~~24. NRC Safety Evaluation Report, Babcock and Wilcox Company, Reference Safety Analysis Report, B-SAR-205, May 1978.[28]

~~19~~25. "NRC Safety Evaluation Report - Standard Reference System, CESSAR System 80," Combustion Engineering Inc., December 1975.[29]

~~21~~26. 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.[30]

~~24~~27. Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," attached to SRP Section 6.2.1.5.

~~3~~28. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components," American Society of Mechanical Engineers.[31]

~~5. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."~~ [32]

~~6~~29. 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.[33]

~~7~~30. 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.[34]

~~8~~31. R. J. Wagner and L. L. West, "CONTEMPT-LT Users Manual," Interim Report I-214-74-12.1, Aerojet Nuclear Company, August 1973.

~~9~~32. 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~~33. 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~~34. 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.~~ [35]

~~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.~~ [36]

~~14~~35. 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~~36. "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.[37]

~~18~~37. "Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8312, Westinghouse Electric Corporation, March 1974.[38]

~~20~~38. F. C. Cadek et al., "PWR FLECHT (Full-Length Emergency Cooling Heat Transfer), Final Report," WCAP-7665, Westinghouse Electric Corporation, April l971.[39]

~~22~~39. "Ice Condenser Containment Pressure Transient Analysis Methods," WCAP-8077, Westinghouse Electric Corporation, March 1973.[40]

~~23~~40. "The General Electric Pressure Suppression Containment Analytical Model," NEDO-10320, General Electric Company, April 1971; Supplement 1, May 1971; Supplement 2, January 1973.[41]

~~25~~41. "Long-Term Ice Condenser Containment Code - LOTIC Code," WCAP-8355, Westinghouse Electric Corporation, April 1976. (Non-Proprietary)[42]

~~26~~42. "Long-Term Ice Condenser Containment Code - LOTIC Code," WCAP-8355 Supplement 1, Westinghouse Electric Corporation, April 1976. (Non-Proprietary)[43]

~~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.~~ [44]
~~29. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979.~~ [45]

~~33. NUREG-0694, "TMI-Related Requirements for New Operation Licenses," June 1980.~~ [46]

43. "The General Electric Mark III Pressure Suppression Containment Analytical Model," NEDO-20533, General Electric Company, June 1974; Supplement 1, September 1975.[47]

44. "Advanced Boiling Water Reactor Standard Safety Analysis Report," 23A6100 Rev. 3, General Electric Company, November 1993.[48]

45. NRC Letter to all Holders of Operating Licenses and Construction Permits for Pressurized Water Reactors (PWRs), “Loss of Decay Heat Removal (Generic Letter 88-17),” October 17, 1988.[49]

46. NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States," Final Report, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, September 1993.[50]

**SRP Draft Section 6.2.1**
Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

| Item | Source | Description |
|---|---|---|
| 1. | Current PRB names and abbreviations | Editorial change made to reflect current PRB name and responsibility for this SRP Section. |
| 2. | Current PRB names and abbreviations | Editorial change made to reflect current PRB name and responsibility for this SRP Section. |
| 3. | Editorial | Added citations of General Design Criterion 4 and 10 CFR 50.46. These two references are basic functional design requirements in SRP Sections cited in this section (Sections 6.2.1.2, 6.2.1.3 and 6.2.1.5). (see PIs 24418 and 24522). |
| 4. | Editorial | Added a description of General Design Criterion 4 which is a basic functional design requirement in SRP Section 6.2.1.2 (See PI 24418). |
| 5. | Editorial | Added a description of General Design Criterion 16 which is a basic functional design requirement in SRP Sections 6.2.1.1.A, B, and C. |
| 6. | Editorial | Comma deleted to correct grammar and clarify the sentence. |
| 7. | Editorial | Added a description of 10 CFR 50.46 which is a basic functional design requirement in SRP Sections 6.2.1.3 and 6.2.1.5 (see PI 24522). |
| 8. | Editorial | References to General Design Criteria 52 - 57 were deleted. These requirements are not identified in any of the 6.2.1.X SRP Sections cited in this section that are part of the containment functional design review (see items 1 through 7 in the Introduction). Compliance with these GDCs is reviewed in other SRP Sections as follows: GDCs 52 and 53 - Section 6.2.6, GDCs 54 through 57 - Section 6.2.4. |
| 9. | Integrated Impact 1439 | This paragraph describes the type of containment analyses required during shutdown conditions. Containment interaction and response (including containment closure times for PWRs) will be dependent upon the results of analyses to develop a bases for critical thermodynamic events such as postulated times to core uncovery during a loss of shutdown decay heat removal. |
| 10. | Editorial | Comma deleted to clarify the sentence. |
| 11. | Editorial | Reference to ABWRs was added to SRP Section 6.2.1.1.C under II 926. |
| 12. | Editorial | The article "the" was added to the sentence for clarity and consistency. |

| Item | Source | Description |
|---|---|---|
| 13. | Editorial | Added reference to General Design Criterion 4 which is a basic functional design requirement in Sections and 6.2.1.2 (see PI 24418). |
| 14. | Editorial | References to General Design Criteria 52 and 53 were deleted. These requirements are not identified in any of the 6.2.1.X SRP Sections cited in this section that are part of the containment functional design review (see items 1 through 7 in the Introduction). Compliance with these GDCs is reviewed under SRP Section 6.2.6. |
| 15. | Editorial | Added reference to 10 CFR 50.46 which is a basic functional design requirement in SRP Sections 6.2.1.3 and 6.2.1.5 (see PI 24522). |
| 16. | Editorial | The phrase "that has been previously reviewed" was changed to "that have previously been reviewed" to correct grammar and provide clarity. |
| 17. | 10 CFR 52 applicability related change | Standard design certification (DC) terminology was added to Evaluation Findings section as required by the SRP-UDP Program. |
| 18. | SRP-UDP format item | References were rearranged and renumbered to be consistent with SRP-UDP guidance. |
| 19. | Editorial | The title of GDC 4 was changed to the correct current title of "Environmental and Dynamic Effects Design Bases". |
| 20. | Editorial | Reformatted this and the subsequent GDC references into separate items for clarity and consistency with other SRP Sections. |
| 21. | Editorial | References to GDCs 52 - 57 were deleted. These requirements are not identified in any of the 6.2.1.X SRP Sections cited in this section that are part of the containment functional design review (see items 1 through 7 in the Introduction). Compliance with these GDCs is reviewed in other SRP Sections as follows: GDCs 52 and 53 - Section 6.2.6, GDCs 54 through 57 - Section 6.2.4. |
| 22. | Editorial | The title of GDC 64 was changed to the correct current title "Monitoring Radioactivity Releases". |
| 23. | Unverified reference | This reference cannot be verified to be the most current reference that is still approved by the NRC. |

| Item | Source | Description |
|---|---|---|
| 24. | Unverified reference | This reference cannot be verified to be the most current reference that is still approved by the NRC. In section 6.2.1.7 of the ABWR FSER, the staff discussed use of the computer code COMPARE MOD1A to conduct check calculations (see PI 24399). However, this is a different version of the computer code than is currently cited in SRP 6.2.1.2. |
| 25. | Editorial | This document was added as a reference in Section 6.2.1.1.C as part of the incorporation of II 926. |
| 26. | Editorial | Changed the title of Regulatory Guide 1.97 to the correct current title "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following An Accident". |
| 27. | Editorial | This document was added as a reference in Section 6.2.1.5 as part of the incorporation of II 319. |
| 28. | Unverified reference | This reference cannot be verified to be the most current reference that is still approved by the NRC. |
| 29. | Unverified reference | This reference, which utilizes the computer code CE FLASH-4, cannot be verified to be the most current reference that is still approved by the NRC. The NRC approved CE's use of CEFLASH4A in the System 80+ FSER (see PI 24427). However, CEFLASH4A is not the same version of this computer code that is cited in SRP 6.2.1.3. |
| 30. | Unverified reference | This reference cannot be verified to be the most current reference that is still approved by the NRC. |
| 31. | Editorial | The section of the ASME code was incorrectly listed as section II. This was corrected to section III. |
| 32. | SRP-UDP format item | Format change to make the citation of references consistent with SRP-UDP guidance. This reference is not cited in the 6.2.1.X series of SRP sections. |
| 33. | Unverified reference | This reference cannot be verified to be the most current reference that is still approved by the NRC. The CONTEMPT computer code was cited in Section 6.2.1.1.C as an acceptable code for performing confirmatory containment temperature/pressure response analysis. A version of the CONTEMPT computer code was utilized by the NRC in the ABWR FSER (see PIs 23083 and 24492), but it was not the same version that is cited in SRP Section 6.2.1.1.C. This reference, therefore, cannot be verified to be the most current reference still approved by the NRC. |

| Item | Source | Description |
|---|---|---|
| 34. | Unverified reference | This reference cannot be verified to be the most current reference that is still approved by the NRC. The CONTEMPT computer code was cited in Section 6.2.1.1.C as an acceptable code for performing confirmatory containment temperature/pressure response analysis. A version of the CONTEMPT computer code was utilized by the NRC in the ABWR FSER (see PIs 23083 and 24492), but it was not the same version that is cited in SRP Section 6.2.1.1.C. This reference, therefore, cannot be verified to be the most current reference still approved by the NRC. |
| 35. | SRP-UDP format item | Format change to make the citation of references consistent with SRP-UDP guidance. This reference is not cited in the 6.2.1.X series of SRP sections. |
| 36. | SRP-UDP format item | Format change to make the citation of references consistent with SRP-UDP guidance. This reference is not cited in the 6.2.1.X series of SRP sections. |
| 37. | Unverified reference | This reference cannot be verified to be the most current reference that is still approved by the NRC. |
| 38. | Unverified reference | This reference cannot be verified to be the most current reference that is still approved by the NRC. |
| 39. | Unverified reference | This reference cannot be verified to be the most current reference that is still approved by the NRC. |
| 40. | Unverified reference | This reference cannot be verified to be the most current reference that is still approved by the NRC. |
| 41. | Unverified reference | This reference cannot be verified to be the most current reference that is still approved by the NRC. |
| 42. | Unverified reference | This reference cannot be verified to be the most current reference that is still approved by the NRC. |
| 43. | Unverified reference | This reference cannot be verified to be the most current reference that is still approved by the NRC. |
| 44. | SRP-UDP format item | Format change to make the citation of references consistent with SRP-UDP guidance. This reference is not cited in the 6.2.1.X series of SRP sections. |
| 45. | SRP-UDP format item | Format change to make the citation of references consistent with SRP-UDP guidance. This reference is not cited in the 6.2.1.X series of SRP sections. |
| 46. | SRP-UDP format item | Format change to make the citation of references consistent with SRP-UDP guidance. This reference is not cited in the 6.2.1.X series of SRP sections. |

| Item | Source | Description |
|---|---|---|
| 47. | Editorial | This reference was added to section 6.2.1.1.C to incorporate the updated Mark III analytical model utilized and verified by the staff in the ABWR FSER (see PI 24492). |
| 48. | Editorial | This document was added as a reference in Section 6.2.1.1.C as part of the incorporation of II 926. |
| 49. | Integrated Impact 1439 | Added a reference to Generic Letter 88-17 to support the new introductory paragraph covering containment analyses during shutdown conditions. |
| 50. | Integrated Impact 1439 | Added a reference to NUREG-1449 which documents the NRC staff's evaluation and recommendations for shutdown and low-power operations. |

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**SRP Draft Section 6.2.1**

Attachment B - Cross Reference of Integrated Impacts

| Integrated Impact No. | Issue | SRP Subsections Affected |
|---|---|---|
| 285 | This integrated impact identifies a future work issue which is to consider creating a new SRP Section on severe accident containment performance. This Integrated Impact is not assigned to SRP Section 6.2.1, but has been filed with this section for administrative tracking purposes. | None |
| 1439 | Consideration should be given to adding an introductory paragraph to address containment analyses that may be required to support shutdown operations. | -Subsection I, AREAS OF REVIEW, new 4th paragraph<br>-Subsection VI, REFERENCES, new references 45 and 46 |