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U.S. NUCLEAR REGULATORY COMMISSION

NUREG-0800

"STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS"

NOTICE OF ISSUANCE AND AVAILABILITY

REVISED SRP SECTION 6.2.1.1.C

The U. S. Nuclear Regulatory Commission (NRC) has published a revision to Section 6.2.1.1.C, "Pressure-Suppression Type BWR Containments" of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," LWR Edition (SRP).

The revision consists of SRP Section 6.2.1.1.C, Rev. 6 and incorporates the resolution of Generic Issue B-10, "Behavior of Mark III Containments." The acceptance criteria and guidelines incorporated into the SRP section are detailed in Appendix C to NUREG-0978, "Mark III LOCA-Related Hydrodynamic Load Definition." The implementation guidelines for Mark III containment LOCA-related hydrodynamic loads are identified in Section 4 of NUREG-0978. All changes to SRP Section 6.2.1.1.C are identified by a line in the margin of the revised SRP section.

The revised SRP section is effective immediately. A copy is expected to be available in the Public Document Room within 2 weeks. Copies of the revised SRP Section or of the complete Standard Review Plan, NUREG-0800, Accession No. PD-81-920199, are available for purchase from the National Technical Information Service, 5385 Port Royal Road, Springfield, Virginia 22161; telephone (703) 487-4650.

Dated at Bethesda, Maryland this 10th day of September, 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

HR Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants

Section No. 6.2.1.1.C
Revision No. 6

Appendix No. A to 6.2.1.1.C
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PAGES TO BE REMOVED			NEW PAGES TO BE INSERTED		
PAGE NUMBER		DATE	PAGE NUMBER		DATE
6.2.1.1.C-1 thru 6.2.1.1.C-9	Rev. 5	January 1983	6.2.1.1.C-1 thru 6.2.1.1.C-9	Rev. 6	August 1984

The U.S. Nuclear Regulatory Commission's Standard Review Plan, NUREG-0800, prepared by the Office of Nuclear Reactor Regulation, is available for sale by the National Technical Information Service, Springfield, VA 22161.

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U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

6.2.1.1.C PRESSURE-SUPPRESSION TYPE BWR CONTAINMENTS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - None

I. AREAS OF REVIEW

For Mark I, II, and III pressure-suppression type boiling water reactor (BWR) plant containments, the CSB review covers the following areas:

1. The temperature and pressure conditions in the drywell and wetwell due to a spectrum (including break size and location) of postulated loss-of-coolant accidents.

2. The differential pressure across the operating deck for a spectrum of loss-of-coolant accidents including break size and location (Mark II containments only).

3. Suppression pool dynamic effects during a loss-of-coolant accident or following the actuation of one or more reactor coolant system safety/relief valves, including vent clearing, vent interactions, pool swell, pool stratification, and dynamic symmetrical and asymmetrical loads on suppression pool and other containment structures.

4. The consequences of a loss-of-coolant accident occurring within the containment (wetwell); i.e., outside the drywell (Mark III containments only)

5. The capability of the containment to withstand the effects of steam bypassing the suppression pool.

6. The external pressure capability of the drywell and wetwell, and systems that may be provided to limit external pressures.

7. The effectiveness of static and active heat removal mechanisms.

8. The pressure conditions within subcompartments and acting on system components and supports due to high energy line breaks, e.g., the sacrificial shield structure.

9. The range and accuracy of instrumentation that is provided to monitor and record containment conditions during and following an accident.

10. The suppression pool temperature limit during reactor coolant system safety/relief valve operation, including the events considered in analyzing suppression pool temperature response, assumptions used for the analyses, and suppression pool temperature monitoring system.

11. The reactor coolant system safety/relief valve in-plant confirmatory test program.

12. The evaluation of analytical models used for containment analysis.

The CSB will coordinate other branch evaluations that interface with the overall review of the containment as follows:

> Instrumentation and Control Systems Branch (ICSB), as part of its primary responsibility for SRP Section 7.3, will evaluate the functional capability of the post-accident monitoring instrumentation and recording equipment. The Equipment Qualification Branch (EQB), as part of its primary review responsibility for SRP Section 3.11, will review the qualification test program for the plant protection system and the post-accident monitoring instrumentation and recording equipment. The Mechanical Engineering Branch (MEB), as part of its primary review responsibility for SRP Section 3.6.2, will evaluate the postulated pipe break sizes and locations and guard pipe designs. The MEB will review the design of piping and other components for the appropriate combination of pool dynamic loads and other loads in SRP Sections 3.9.2, 3.9.3, and 3.10. The MEB will review the seismic design and quality group classification as part of its primary review responsibility for SRP Sections 3.2.1 and 3.2.2, respectively. The Structural Engineering Branch (SEB), as part of its primary responsibility for SRP Section 3.8.3, will evaluate the structural design of unique flow limiting devices used in subcompartments and certain aspects of guard pipe designs and the structural aspects of the in-plant reactor coolant system safety/relief valve tests (NUREG-0763, Rei 1d). Accident Evaluation Branch (AEB) will review fission product control features of containment heat removal systems as part of its primary review responsibility for SRP Section 6.5.2. The review of proposed technical specifications at the operating license stage of review pertaining to the bypass leakage surveillance is performed by Standardization and Special Projects Branch (SSPB) as part of its primary review responsibility for SRP Section 16.0.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria and their methods of application are contained in the SRP sections identified as the primary review responsibility of those branches.

II. ACCEPTANCE CRITERIA

The acceptance criteria given below applies to the design and functional capability of BWR pressure-suppression type containments. CSB accepts the containment design if the relevant requirements of General Design Criterion 4, 16, 50, and 53 are complied with. The relevant requirements are as follow:

1. General Design Criterion 4, as it relates to the environmental and missile protection design, requires that structures, systems, and components important to safety be designed to accommodate the dynamic effects (e.g., effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures) that may occur during normal plant operation or following a loss-of-coolant accident.

2. General Design Criteria 16 and 50, as they relate to the containment being designed with sufficient margin, require that the containment and its associated systems can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

3. General Design Criterion 53 as it relates to the containment design capabilities provided to assure that the containment design permits periodic inspection, an appropriate surveillance program, and periodic testing at containment design pressure.

Specific criterion or criteria that pertain to design and functional capability of BWR pressure-suppression type containments are indicated below:

1. In meeting the requirements of General Design Criteria 16 and 50 regarding the design margin for Mark I, II and III plants at the operating license stage of review, the peak calculated values of pressure and temperature for the drywell and wetwell should not exceed the respective design values. Also, the peak deck differential pressure for Mark II plants should not exceed the design value. Acceptable methods for the calculation of Mark I, II and III containment environmental response to loss-of-coolant accidents are found in NUREG-0588 (Ref. 35).

   For Mark III plants, the calculated results for drywell pressure and temperature, containment pressure and temperature, and differential pressure between the drywell and containment should be based on the General Electric Mark III analytical model (Ref. 23) that was used in the Grand Gulf analysis and evaluated by CSB. The use of this model at the construction permit stage is acceptable if an appropriate margin (see below) between the calculated and design differential pressures is used. The Mark III analytical model have been verified by the large-scale Mark III test results. If an analytical model other than the General Electric Mark III analytical model identified above is used, the model should be demonstrated to be physically appropriate and conservative to the extent that the General Electric model has been found acceptable. In addition, it will be necessary to demonstrate its performance with suitable test data in a manner similar to that described above.

   For Mark III plants at the construction permit stage, the containment design pressure should provide at least a 15% margin above the peak calculated containment pressure, and the design differential pressure between drywell and containment should provide at least a 30% margin above the peak calculated differential pressure.

For Mark I, II and III plants at the operating license stage, the peak calculated containment pressure and differential pressure should be less than the design values. In general, it is expected that the peak calculated pressures will be about the same as at the construction permit stage. However, it is possible that the margins may be affected by revised or improved analytical models, test results, or minor changes in the as-built design of the plant.

2. In meeting the requirement of General Design Criterion 4, regarding the dynamic effects associated with normal and accident conditions, calculation of dynamic loads should be based on appropriate analytical models and supported by applicable test data. Consideration should be given to loads on suppression pool retaining structures and structures which may be located directly above the pool, as a result of pool motion during a loss-of-coolant accident or following actuation of one or more reactor coolant system safety/relief valves.

   The acceptability of pool dynamic loads for plants with Mark I containments is based on conformance with NRC acceptance criteria found in NUREG-0661 (Ref. 39 and 1c).

   The acceptability of loss-of-coolant accident related pool dynamic loads for plants with Mark II containments is based on conformance with the generic loads previously reviewed and found acceptable by the NRC and NRC acceptance criteria. The loss-of-coolant accident related pool dynamic loads and criteria are as discussed in NUREG-0808 (Ref. 1b), and Appendix B to this SRP section. These loads and criteria supersede those discussed in references 36, 37 and 38. Pool dynamic loads and criteria associated with the actuation of one or more reactor coolant system safety/relief valves are specified in Appendix A of NUREG-0302 (Ref. 1e).

   The acceptability of pool dynamic loads for plants with Mark III containments is based on conformance with the NRC acceptance criteria identified in Appendix C of NUREG-0978. For Mark III plants at the construction permit stage, conformance with the NRC acceptance criteria can be demonstrated if a previously analyzed Mark III plant has sufficient similarity in plant characteristics to make the analyses performed for that plant design applicable to the Mark III plant design under consideration.

   The acceptability of pool dynamic loads associated with the actuation of one or more reactor coolant system safety/relief valves in Mark III containment are specified in Appendix B of NUREG-0802.

3. In meeting the requirements of General Design Criteria 16 and 50 regarding the containment design margin for Mark III plants, high energy lines passing through the containment should be provided with guard pipes or enclosed in other types of protective structures to assure that the suppression pool is not bypassed. If guard pipes are used, they should be designed in accordance with acceptance criteria established by the MEB as set forth in SRP Section 3.6.2. The allowable leakage areas for steam bypass of

the suppression pool should be determined for a spectrum of postulated reactor coolant system pipe breaks. The maximum allowable bypass area of the plant should be based on conservative analyses which consider available energy removal mechanisms and the containment design pressure.

4. In meeting the requirement of General Design Criterion 53 regarding periodic testing at containment design pressure for Mark I, II, and III containments, the maximum allowable leakage area for steam bypass of the suppression pool should be greater than the technical specification limit for leakage measured in periodic drywell-wetwell leakage tests. Specific acceptance criteria for the three types of containments are as discussed in Appendix A.

5. In meeting the requirement of General Design Criterion 50 with respect to the design leakage rate for Mark III containments, justification should be provided for any reduction in the containment leak rate claimed for times less than 30 days after a postulated pipe break accident. This also includes meeting the regulatory position C.1.e of Regulatory Guide 1.3.

6. In meeting the requirement of General Design Criterion 16, provisions should be made in one of the following ways to protect the drywell and wetwell (or containment) of Mark I, II, and III plants, and the operating deck of Mark II plants, against loss of integrity from negative pressure transients or post-accident atmosphere cooldown:

   a. Structures should be designed to withstand the maximum calculated external pressure.

   b. Vacuum relief devices should be provided in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE, to assure that the external design pressures of the structures are not exceeded. The vacuum relief valve guidelines are set forth in Appendix A to this SRP section.

7. In meeting the requirements of General Design Criterion 50, with respect to design margin for item 6 above, the external design pressures of the structures, including the design upward deck differential pressure for Mark II plants, should provide an adequate margin above the maximum calculated external pressures to account for uncertainties in the analyses.

8. The acceptability of the reactor coolant system safety/relief valve in-plant confirmatory test program shall be based on conformance with the guidelines specified in Section 6, 7, and 8 of NUREG-0763 (Ref. 1d). If the applicant/licensee elects not to perform the SRV in-plant tests, the acceptability of this exception shall be determined in conformance with the guidelines specified in Section 4 of NUREG-0763.

9. For Mark I, II, and III plants, the local suppression pool temperature should not exceed 200°F or the acceptance criteria specified in Section 5.1 of NUREG-0783 (Ref. 1f).

10. In meeting the requirements of General Design Criteria 13 and 64, instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature and the suppression pool water level and temperature following

an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked and recorded throughout the course of an accident. Item II.F.1 of NUREG-0737 and NUREG-0718, and Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following An Accident," should be followed.

III. REVIEW PROCEDURES

The procedures described below are followed for the review of BWR pressure-suppression containments. 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 for aspects of functional design common to a class of BWR pressure-suppression type containments or by adopting the results of previous reviews of plants with essentially the same containment functional design.

Upon request from the primary reviewer, the secondary review branches will provide input for the areas of review stated in subsection I of this SRP section. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

1. The CSB reviews the analyses of the drywell and wetwell temperature and pressure response for Mark I, II and III containments. The CSB performs confirmatory analyses, when necessary, using the CONTEMPT-LT computer code. Input data for the code, including mass and energy release data, are generally taken from the safety analysis report.

   The CSB normally analyzes only the design basis loss-of-coolant accident, which has been found from previous reviews to be the recirculation line break for Mark I and II plants. For Mark III plants, the steam line break has been determined to be the design basis loss-of-coolant accident. However, mass and energy releases from the recirculation line break will be evaluated using various flow correlations.

   The CSB evaluates analyses of both the short-term and long-term pressure and temperature responses of Mark III containment plants. For Mark III plants, the peak containment pressure following a loss-of-coolant accident is independent of the postulated pipe break size. The CSB reviews the containment response analysis presented in the safety analysis report to determine that the acceptance criteria in subsection II have been satisfied.

   The CSB and its consultants have reviewed the General Electric Mark III analytical model and have determined that the code appears to calculate the drywell pressure response in an acceptable manner. The code has been verified by the General Electric Mark III test program.

   The CSB verifies from the safety analysis report that the General Electric code has been utilized and that the input assumptions to the code are conservative. If analytical methods other than the General Electric model are used, the CSB, in conjunction with its consultants, will initiate a detailed review of the methods. In this case, the CSB reviews the proposed modeling, analytical methods and assumptions, correlation of results with applicable test data, and comparison with other similar analyses, to determine the acceptability of the proposed model.

The CSB reviews analyses of the drywell response to either a recirculation line rupture or a steam line rupture, as presented in the safety analysis report. The CSB determines from the results of these analyses that the "worst" break has been identified in establishing the drywell-wetwell design differential pressure as well as the design pressure for subcompartments and equipment supports.

Modifications to the CONTEMPT-LT computer code have been made which provide the capability to perform confirmatory analyses of the Mark III drywell pressure response.

2. The review of the dynamic loads associated with a LOCA have been concluded with the issuance of NUREG-0661 for Mark I plants, NUREG-0808 for Mark II plants and NUREG-0978 for Mark III plants.

   The review of the dynamic loads associated with the actuation of one or more primary coolant system safety/relief valves have been concluded with the issuance of NUREG-0661 for Mark I plants, NUREG-0802 for Mark II and Mark III plants.

3. For Mark III plants, the CSB verifies from the safety analysis report that high energy lines which pass through the containment outside the drywell are provided with guard pipes or enclosed in other types of protective structures. If guard pipes are used, the design must meet the acceptance criteria established in SRP Sections 3.6.2 and 3.8.3. For unguarded lines, the CSB reviews analyses of the consequences of postulated ruptures in these lines. The CSB bases its acceptance of the analyses on the conservatism of the methods and assumptions and on the margin provided to assure against exceeding the design pressure of the containment. If leakage detection and isolation equipment are provided, the ICSB evaluates the effectiveness of the detection instrumentation and isolation devices to mitigate the consequences of a pipe rupture and the electrical design criteria for these systems under SRP Section 7.3.

4. The CSB reviews the analyses of the suppression pool temperature for transients involving the actuation of reactor coolant system safety/relief valves in Mark I, II and III plants. The CSB evaluates the assumptions and conservatisms employed in the analyses to assure that the acceptance criteria set forth in NUREG-0783 are met.

   The CSB also reviews the proposed reactor coolant system safety/relief valve in-plant confirmatory test programs or the rationale for not performing such tests.

5. The CSB evaluates analyses of bypass leakage capability. The CSB determines the adequacy of proposed bypass leakage tests and surveillance programs based on the results of previous reviews, operating experience at similar plants, and engineering judgment. CSB will advise the AEB of the bypass leakage.

6. The CSB evaluates the conservatism of potential depressurization transients. In evaluating surveillance and test programs for vacuum relief systems, the CSB uses the results of previous reviews and operating experience with similar systems to determine their adequacy. At the operating license stage, the SSPB reviews the proposed technical specifications to assure

that adequate surveillance and administrative control will be maintained over the vacuum relief devices.

7. Upon request, the SEB will review the design of unique flow-limiting devices which are identified during the CSB review of the containment subcompartments.

8. The CSB reviews the accuracy and range of the instrumentation provided to monitor the post-accident environment. The ICSB, under SRP Section 7.5 and the EQB, under SRP Section 3.11, have review responsibility for the acceptability of, and the qualification test program for the sensing and actuation instrumentation of the plant protection system and the post-accident monitoring instrumentation and recording equipment.

IV. EVALUATION FINDINGS

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

V. IMPLEMENTATION

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

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.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides, NUREGs and the following:
1. Revision 2 to Appendix A of this SRP section does not contain any new criteria or guidelines, therefore implementation remains the same and is as stated in Appendix A. 2. LOCA-related pool dynamic loads criteria are implemented on all plants with Mark I containments in accordance with section 5 of NUREG-0661 and supplement 1 to it; for all Mark II containments in accordance with section 3.1 of NUREG-0808 and/or Appendix B of this SRP section; and for all Mark III containment designs in accordance with Section 4 of NUREG-0978.
3. Reactor coolant system safety/relief valve(s) - related pool dynamic loads criteria are implemented on all plants with Mark I containments in accordance with section 5 of NUREG-0661 and supplement 1 to it, and for all Mark II and III containments in accordance with section 4.1 of NUREG-0802.

VI. REFERENCES

The references for this SRP section are those listed in SRP Section 6.2.1, together with the following:

1a. SRP Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping."

1b. NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria."

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

1d. NUREG-0763, "Guidelines for Confirmatory In-plant Tests of Safety/Relief Discharge for BWR Plants."

1e. NUREG-0802, "Safety/Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments."

1f. NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments."

1g. NUREG-0978, "Mark III LOCA-Related Hydrodynamic Load Definition."

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