U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

6.2.1.1.C PRESSURE-SUPPRESSION TYPE BWR CONTAINMENTS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of Containment Integrity

Secondary - None

I. AREAS OF REVIEW

The specific areas of review are as follows:

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.

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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.

13. Inspection, Test, Analysis, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the applicant's proposed information on the ITAAC associated with the systems, structures, and components (SSCs) related to this SRP section is reviewed in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria - Design Certification." The staff recognizes that the review of ITAAC is performed after review of the rest of this portion of the application against acceptance criteria contained in this SRP section. Furthermore, the ITAAC are reviewed to assure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

14. COL Action Items and Certification Requirements and Restrictions. COL action items may be identified in the NRC staff's final safety evaluation report (FSER) for each certified design to identify information that COL applicants must address in the application. Additionally, DCs contain requirements and restrictions (e.g., interface requirements) that COL applicants must address in the application. For COL applications referencing a DC, the review performed under this SRP section includes information provided in response to COL action items and certification requirements and restrictions pertaining to this SRP section, as identified in the FSER for the referenced certified design.

Review Interfaces

The listed SRP sections interface with this section as follows:

1. Review of the functional capability of the post-accident monitoring instrumentation and recording equipment under SRP Section 7.3.

2. Review of the qualification test program for the plant protection system and the post-accident monitoring instrumentation and recording equipment under SRP Section 3.11.

3. Review of postulated pipe break sizes and locations and guard pipe designs under SRP Section 3.6.2. Review of the design of piping and other components for the appropriate combination of pool dynamic loads and other loads in the wetwell under SRP Sections 3.9.2, 3.9.3, and 3.10. Review of the seismic design and quality group classification under SRP Sections 3.2.1 and 3.2.2.

4. Review of 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 under SRP Section 3.8.3 (NUREG-0763, Reference 57).

5. Review of fission product control features of containment heat removal systems under SRP Section 6.5.2.

6. Review of proposed technical specifications at the operating license or design certification stage of review pertaining to the bypass leakage surveillance under SRP Section 16.0.

7. Review of shutdown risk assessment reviews, including containment analysis issues, as part of its primary review responsibility for SRP Section 19.1

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

II. <u>ACCEPTANCE CRITERIA</u>

<u>Requirements</u>

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. General Design Criterion (GDC) 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. GDC 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. GDC 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.

4. GDC 13, as it relates to instrumentation and control, requires instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions as appropriate to ensure adequate safety.

5. GDC 64, as it relates to monitoring radioactivity releases, requires that means be provided for monitoring the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents.

6. 10 CFR 52.47(a)(1)(vi), as it relates to ITAAC (for design certification) sufficient to assure that the SSCs in this area of review will operate in accordance with the certification.

7. 10 CFR 52.97(b)(1), as it relates to ITAAC (for combined licenses) sufficient to assure that the SSCs in this area of review have been constructed and will be operated in conformity with the license and the Commission's regulations.

8. 10 CFR 50.34(f)(2)(xvii) as it relates to instrumentation capable of operating in the post-accident environment requires instrumentation to measure and record containment pressure, containment water level, containment hydrogen concentration, containment radiation intensity (high level), and noble gas effluents at all potential accident release points.

9. 10 CFR 50.34(f)(3)(v)(A)(1) as it relates to considering containment loads generated by the release of hydrogen from 100% fuel clad metal-water reaction accompanied by hydrogen burning.

<u>SRP Acceptance Criteria</u>

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for review described in Subsection I of this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1. In meeting the requirements of GDC 16 and 50 regarding the design margin for BWR pressure-suppression 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 BWR pressure-suppression containment environmental response to loss-of-coolant accidents are found in NUREG-0588 (Reference 14).

   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 (Reference 39) that was used in the ABWR and Grand Gulf analyses. 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 has 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 ABWR plants, the calculated results for containment short-term and long-term response to postulated line breaks are based on the General Electric Mark III (ABWR) analytical model that was used in the ABWR standard plant analysis evaluated by the NRC in the ABWR FSER (Reference 21).

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 BWR pressure-suppression 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 GDC 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 (Reference 17).

   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 (Reference 55), and Appendix B to this SRP section. 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-0802 (Reference 58).

   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 (Reference 60). 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 (Reference 58).

   The acceptability of pool dynamic loads for plants with ABWR containments is based on the GE analytical model provided in Appendix 3B of the ABWR SSAR (Reference 40) which, in part, conforms with NUREGS 0802, 0808, and 0978. This model was used in the standard plant analysis and evaluated by the NRC in the ABWR FSER (Reference 21).

3. In meeting the requirements of GDC 16 and 50 regarding the containment design margin for Mark III and ABWR 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 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 GDC 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 discussed in Appendix A. Plants with ABWR containments should follow the specific acceptance criteria for Mark II containments.

5. In meeting the requirement of GDC 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. For plants with ABWR containments, the design leakage rate for primary containment should be assumed for the duration of the loss-of-coolant accident consistent with Regulatory Guide 1.3.

6. In meeting the requirement of GDC 16, provisions should be made in one of the following ways to protect the drywell and wetwell (or containment) of Mark I, II, III, and ABWR 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 (Reference 29), 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 GDC 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. In meeting the requirements of GDC4, 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 (Reference 57). 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. NUREG-0783 (Reference 59) specifies that, for BWR pressure-suppression plants, the local suppression pool temperature should not exceed 93 C (200 F) or the acceptance criteria specified in Section 5.1 of NUREG 0783.

   This criterion may be eliminated provided that the SRV discharges are delivered to the suppression pool through a "T" or "X" quencher device previously approved by the staff and described in NUREG-0802 (Reference 58) and NUREG-0978 (Reference 60). NEDO-30832 concluded that unstable condensation oscillation loads due to suppression pool temperatures approaching the saturation temperature are bounded by air clearing hydrodynamic loads when the "T" or "X" quencher is used. The NRC review and approval of this conclusion is documented in a August 29, 1994 safety evaluation (Reference 52).

This NRC safety evaluation also stated that there was no basis for permitting the deletion of local pool temperature requirements when a plant has an Emergency Safety Feature (ESF) pump inlet located at or above the quencher elevation due to concern that steam discharged from the quencher may be ingested at the pump inlet and cause pump cavitation or a water hammer. An analysis based on the plant specific geometry of the quenchers and pump intakes may be used to demonstrate that a steam plume discharged from the quencher will not be ingested by the pump intakes.

10. In meeting the requirements of General Design Criteria 13 and 64, and 10 CFR 50.34(f)(2)(xvii) (for those applicants subject to 10 CFR 50.34(f)), 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. Regulatory guidance is contained in Branch Technical Position 7-10, "Guidance on Application of Regulatory Guide 1.97."

11. In meeting the requirements of 10 CFR 50.34(f)(3)(v)(A)(1), applicants subject to this section should evaluate an accident that releases hydrogen generated from a 100% fuel clad metal-water reaction. The evaluation should demonstrate that the appropriate articles for service level C limits (considering pressure and dead load only), for either concrete or steel containments, from ASME Boiler Pressure Vessel Code, Section III, are met. In addition to the containment pressurization caused directly by this accident, the increase in pressure from hydrogen burning in containment should be analyzed.

Technical Rationale

The technical rationale for application of these requirements to reviewing this SRP section is discussed in the following paragraphs:

1. GDC 4 requires that structures, systems, and components important to safety be designed to withstand the environmental conditions and dynamic effects associated with normal operations, maintenance, testing, and postulated accidents. This SRP Section reviews containment design and related analyses of postulated accident conditions. Containment is the final barrier against the spread of contamination that is released from the reactor or its systems during an accident. It must be designed to function under the harsh environmental conditions and severe dynamic effects associated with accidents such as a LOCA or steam rupture. Meeting GDC 4 provides assurance that containment will prevent the release of radioactivity to the environment under the most challenging conditions it is expected to face.

2. GDC 16 requires containment to be designed as a leak tight barrier that will withstand the most extreme accident conditions for the duration of any postulated accident. Containment must be leak tight and withstand accidents because it is the final barrier against the release of radioactivity to the environment. Meeting GDC 16 provides assurance that radioactivity will not be released to the environment.

3. GDC 50 requires the containment structure and associated heat removal systems to be designed with margin to accommodate any loss-of-coolant accident such that the containment design leak rate is not exceeded. A loss-of-coolant accident potentially causes the greatest pressure surge and release of fission products when compared to

any other accident. Since it is the most severe challenge expected, containment must be designed to definitively withstand this accident. Meeting GDC 50 will ensure that containment integrity is maintained under the most severe accident conditions, thus precluding the release of radioactivity to the environment.

4. GDC 53 requires that containment be designed to permit periodic testing and inspection so that its functionality can be confirmed. Since containment is the final barrier against the release of contamination, it is vital that its ability to carry out its design function be maintained and verified throughout the life of the plant. A design that allows periodic verification of containment operability will help ensure that radioactivity is not released to the environment.

5. GDC 13 requires that instrumentation be provided to monitor all expected parameters of normal operation, anticipated operational occurrences, and accidents to assure adequate reactor safety is maintained. Since containment plays a vital safety role, appropriate instrumentation, such as temperature and pressure, must be provided so that operators can verify containment is properly fulfilling its function. Meeting GDC 13 will help ensure that containment accomplishes its mission of precluding the release of radioactivity to the environment.

6. GDC 64 requires that the containment atmosphere be monitored for the release of radioactivity from normal operations, anticipated operational occurrences, and accidents. In order to ensure that containment functions properly, operators must be aware of any radioactive releases within containment so that they can take appropriate manual action or monitor automatic action. Regulatory Guide 1.97 provides specific criteria for the design of containment instrumentation which have been found acceptable by the NRC as fulfilling the requirements of GDC 64. Meeting GDC 64 and the specific guidance of Regulatory Guide 1.97 will assist operators in ensuring that the containment meets its safety function of preventing the release of radioactivity to the environment.

7. 10 CFR 50.34(f)(3)(v)(A)(1) requires that the containment be designed to withstand either hydrogen burning during an accident that releases hydrogen from a 100% fuel clad metal-water reaction. During the accident at TMI-2, metal-water reactions generated hydrogen in excess of the amounts originally anticipated. As a result of this finding, the Commission issued requirements on hydrogen control in 10 CFR 50.34(f). Other criteria require the containment to be designed to withstand postulated accidents. If such a postulated accident releases or generates hydrogen, an added containment pressurization effect beyond the initial accident may be experienced due to burning of hydrogen. In accordance with this regulation, the containment must be designed to withstand this additional pressure to ensure that its integrity is maintained, thus precluding the release of radioactivity to the environment.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review specified in Subsection I of this SRP section, the review procedure is identified below. These review procedures are based on the identified SRP acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. Upon request from the primary reviewer, other 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.

2. The primary review branch reviews the analyses of the drywell and wetwell temperature and pressure response for BWR pressure-suppression containments. The primary review branch performs confirmatory analyses, when necessary. Input data for the code, including mass and energy release data, are generally provided by the applicant.

   The primary review branch 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. For ABWR plants, the feedwater line break has been determined to be the design basis loss-of-coolant accident.

   The primary review branch evaluates analyses of both the short-term and long-term pressure and temperature responses of Mark III and ABWR 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 primary review branch reviews the containment response analysis presented in the safety analysis report to determine whether the acceptance criteria in Subsection II have been satisfied.

   The NRC has reviewed the General Electric Mark III analytical model and have determined that the code appears to calculate the drywell pressure response for both Mark III and ABWR plants in an acceptable manner. The code has been verified by the General Electric Mark III test program.

   The primary review branch 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 primary review branch will initiate a detailed review of the methods. In this case, 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 are reviewed.

   The primary review branch reviews analyses of the drywell response to a recirculation line rupture, a steam line rupture, or a main feedwater line break as presented in the safety analysis report. The primary review branch 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.

   The primary review branch verifies that the containment is designed to withstand hydrogen burning during an accident that releases hydrogen from a 100% fuel clad metal-water reaction as described in Acceptance Criterion II. of this SRP section.

3. The review of the dynamic loads associated with a LOCA has 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 has been concluded with the issuance of NUREG-0661 for Mark I plants, and NUREG-0802 for Mark II and Mark III plants.

The review of dynamic loads for ABWR plants has been concluded with issuance of Appendix 3B of the ABWR SSAR.

4. For Mark III and ABWR plants, the primary review branch 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 primary review branch reviews analyses of the consequences of postulated ruptures in these lines. The primary review branch 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 branch responsible for SRP Section 15.6.5 Appendix A (for operating reactors) or SRP Section 15.0.3 (for advanced light water reactors) evaluates the effectiveness of the detection instrumentation and isolation devices to mitigate the consequences of a pipe rupture and to meet the electrical design criteria for these systems under SRP Section 7.3.

5. The primary review branch reviews the analyses of the suppression pool temperature for transients involving the actuation of reactor coolant system safety/relief valves in BWR pressure-suppression plants. The primary review branch evaluates the assumptions and conservatisms employed in the analyses to assure that the acceptance criteria set forth in NUREG-0783 are met.

   These criteria may be eliminated provided that the SRV discharges are delivered to the suppression pool through a "T" or "X" quencher device previously approved by the staff and described in NUREG-0802 (Reference 58) and NUREG-0978 (Reference 60). NEDO-30832 concluded that unstable condensation oscillation loads due to suppression pool temperatures approaching the saturation temperature are bounded by air clearing hydrodynamic loads when the "T" or "X" quencher is used. The NRC review and approval of this conclusion is documented in a August 29, 1994 safety evaluation (Reference 52).

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

6. The primary review branch evaluates analyses of bypass leakage capability. The primary review branch 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. The primary review branch will advise the branch responsible for SRP Section 15.6.5 Appendix A (for operating reactors) or SRP Section 15.0.3 (for advanced light water reactors) of the bypass leakage.

7. The primary review branch evaluates the conservatism of potential depressurization transients. In evaluating surveillance and test programs for vacuum relief systems, the primary review branch uses the results of previous reviews and operating experience with similar systems to determine their adequacy. At the operating license or design certification stage, the review branch responsible for SRP Section 16.0 reviews the proposed technical specifications to assure that adequate surveillance and administrative control will be maintained over the vacuum relief devices.

8. Upon request, the review branch responsible for Standard Review Plan Section 3.9.2 will review the design of unique flow-limiting devices which are identified during the primary review branch's review of the containment subcompartments.

9. The primary review branch reviews the accuracy and range of the instrumentation provided to monitor the post-accident environment. The review branch responsible for Standard Review plan Section 7.5, and the review branch responsible for Standard Review Plan 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.

10. For new plant applicants, 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 during shutdown operations. The analyses should encompass shutdown thermodynamic states and physical configurations to which the plant can be subjected during shutdown conditions (such as closure times, temperature, radiological conditions and time to uncover the core during loss of decay heat removal).

11. For reviews of DC and COL applications under 10 CFR Part 52, the reviewer should follow the above procedures to verify that the design set forth in the safety analysis report, and if applicable, site interface requirements meet the acceptance criteria. For DC applications, the reviewer should identify necessary COL action items. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a DC, an ESP or other NRC-approved material, applications, and/or reports.

    After this review, SRP Section 14.3 should be followed for the review of Tier I information for the design, including the postulated site parameters, interface criteria, and ITAAC.

IV. 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.

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

For DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections) the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, and interface requirements and combined license action items relevant to this SRP section.

V. 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 docketed six months or more after the date of issuance of this SRP section, unless superceded by a later revision.

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

1. Revision 3 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. <u>REFERENCES</u>

References for this SRP section as those listed in SRP Section 6.2.1.

Appendix A to SRP Section 6.2.1.1.C
Steam Bypass for Mark I, II, and III Containments

A. Background

This appendix pertains to steam bypass from the drywell to the suppression pool air volume in the Mark I, II, and III containment design. In a pressure suppression-type containment, steam released from the primary system following a postulated LOCA is collected in the containment drywell volume and directed through connecting vents to the suppression pool in the containment wetwell volume and steam is condensed as it enters into the suppression pool. Thus, no steam enters the wetwell air volume. The potential exists for steam to bypass the suppression pool by leakage through the vacuum breakers or directly from leak paths in the drywell-to-suppression chamber vent pipes, the diaphragm-wall seal around diaphragm penetrations, or cracks in the concrete diaphragm.

The capability for steam bypass for small primary system breaks in the Mark I, II, and III containment design are as follows: the Mark I design is of the order of 18.6 cm^2 (0.02 ft^2), the capability of the Mark II containment is approximately 46.5 cm^2 (.05 ft^2), and the Mark III design has a capability of $A/\sqrt{K}$ = 929 cm^2 (1 ft^2).

This steam bypass position was developed to assure that containment integrity will be maintained following the onset of small breaks in the drywell. This can be achieved by upgrading the wetwell spray to an engineered safety feature and requiring automatic actuation of the wetwell spray 10 minutes following a break (Mark II and Mark III).

To provide assurance that the bypass leakage is not substantially increased over the life of a plant, this position includes requirements for leakage tests. The leakage tests include both periodic low-pressure leak tests and a preoperational high-pressure leak test (Mark II and Mark III containments). In addition, Mark I containments have been operating with a positive pressure differential between the drywell and wetwell which provides a mechanism for continuously monitoring the amount of bypass leakage.

B. Position

The system used to quench steam bypassing the suppression pool should be designed such that the steam bypass capability for small breaks satisfies the criteria described below. Any proposed alternative criteria must be suitably justified by the applicant and reviewed by the NRC staff.

1. Bypass Capability (Mark II and Mark III)
The containment should have a steam bypass capability for small breaks of the order of: 46.5 cm^2 (.05 ft^2) ($A/\sqrt{K}$) for Mark II plants and 929 cm^2 (1 ft^2) ($A/\sqrt{K}$) for Mark III plants.

a. Containment Wetwell Sprays

The wetwell spray system, including the electrical instrumentation and controls, should meet the standards appropriate to engineered safety features; i.e., quality, redundancy, testability, and other appropriate criteria. The wetwell spray should be automatically actuated 10 minutes following a LOCA signal and an indication of pressurization of the wetwell. In addition, the instrumentation and control systems provided to actuate the wetwell spray should be actuated by diverse parameters.

If the existing wetwell spray system is to be used to improve the bypass capability, the consequences of actuation of the wetwell spray system on ECCS function and long-term pool cooling considerations should be evaluated to show that minimum ECCS and pool cooling requirements are met.

b. Transient Bypass Capability Analyses

Transient analyses should be provided to establish the capability for a small break. A normal plant shutdown time of 6 hours should be assumed. The results and bases for the analyses should be provided including the following: the pressure history in the drywell and the wetwell; identification and quantification of the static heat sinks and the condensing heat transfer coefficient; spray capacity, efficiency, coverage, start time and temperature history; identification and quantification of heat sources.

2. Leakage Tests and Surveillance Requirements

a. High-Pressure Leak Test

A single preoperational high-pressure leakage test should be performed on each (Mark II and Mark III) unit. The purpose of this test is to detect leakage in the drywell to suppression chamber vent piping, penetrations, downcomers, vacuum breakers, floor seals, vent seals, and the diaphragm. This test should be performed at approximately the peak drywell to wetwell differential pressure following the high-pressure structural test of the diaphragm.

b. Low-Pressure Leak Tests

A post-operational low pressure leakage test should be performed on each Mark I, II, and III unit to detect leakage in the drywell to suppression chamber vent piping, penetration downcomers, vacuum breakers, floor seals, vent seals, and the diaphragm. This test should be performed at each refueling outage at a differential pressure corresponding to approximately the submergence of the vents.

c. Acceptance Criteria for Leakage Tests

The Mark II and Mark III acceptance criteria for both the high and low pressure leakage tests shall be a measured bypass leakage which is less than 10% of the capability of the containment as defined in Position B.1 above. For Mark I containment the acceptance criterion is that the measured leakage is not greater than the leakage that could result from a 2.54 cm (one inch) diameter opening.

d. Surveillance Requirements

A visual inspection should be conducted to detect possible leak paths at each refueling outage. Each vacuum relief valve and associated piping should be checked at this time to determine that it is clear of foreign matter.

3. Vacuum Relief Valve Requirements

a. Position Indicators and Alarms

Redundant position indicators should be placed on all vacuum breakers with redundant indication and an alarm in the control room. The vacuum breaker position indicator system should be designed to provide the plant operators with continuous surveillance of the vacuum breaker position. The indicators should have adequate sensitivity to detect a total valve opening, for all valves, that is less than the bypass capability for a small break (Note for Mark I: this corresponds to the acceptance criteria described in 2.c above). The detectable valve opening should be based on the assumption that the valve opening is evenly divided among all the vacuum breakers.

b. Vacuum Valve Operability Tests

All vacuum breakers should be operability tested at monthly intervals to assure free movement of the valves.

C. Implementation

This position will be applied in the review of all CP, DC, and COL applications with Mark I, Mark II and Mark III containments (see also subsection V of this SRP section). The positions of this revision to Appendix A of this SRP section do not apply to plants with an operating license issued prior to January 1983.

Appendix B to SRP Section 6.2.1.1.C
Summary of Mark II LOCA-Related
Pool Dynamic Loads[115]

The Mark II program to establish LOCA-related pool dynamic loads has been in existence since April 1975. Since that time, a number of different load specifications have been developed. The purpose of this appendix is to identify, in one location, those generic load specifications that the staff finds acceptable.

A summary of generic loads acceptable to the NRC is provided in Table B-1. This table includes the following information: load identification, a summary of the load specification, load specification clarifying criteria and reference to the NRC NUREG section that describes the NRC specific load evaluation.

The staff finds most of the generic LOCA-related pool dynamic load specifications proposed by the Mark II owners acceptable. For the few cases where the staff was unable to conclude that a proposed load was acceptable, the staff developed acceptance criteria. The criteria provide load specifications that are acceptable to the staff.

The staff finds that the detailed loads specifications referenced in Table B-1, along with the criteria that further clarify these loads specifications, constitute a complete set of acceptable LOCA-related pool dynamic loads.

Table B-1 Mark II LOCA-related hydrodynamic loads
Summary table

Load or phenomenon	Load specification	Load specification clarifying footnotes	NRC Evaluation	(Foot-note)
A. Submerged Boundary Loads During Vent Clearing	24 psi overpressure added to local hydrostatic below vent exit (walls and basemat) - linear attenuation to pool surface.	----	II.A.1	(1b)
B. Pool-Swell Loads				
1. Pool-Swell Analytical Model				
a) Air- Bubble Pressure	Calculated by the pool-swell analytical model (PSAM) used in calculation of submerged boundary loads.	----	III.B.3.a.1	(1a)
b) Pool-Swell Evaluation	Use PSAM with polytropic exponent of 1.2 to a maximum swell height which is the greater of 1.5 vent submergence or the evaluation corresponding to the ΔP=2.5 psid.	----	II.A.2	(1b)

Table B-1 Mark II LOCA-related hydrodynamic loads
Summary table

Load or phenomenon	Load specification	Load specification clarifying footnotes	NRC Evaluation	(Foot-note)
c) Pool-Swell Velocity	Velocity history vs. pool elevation predicted by the PSAM used to compute impact loading on small structures and drag on gratings between initial pool surface and maximum pool elevation and steady-state drag between vent exit and maximum pool elevation. Analytical velocity variation is used up to maximum velocity. Maximum velocity applies there-after up to maximum pool swell. PSAM predicted velocities multiplied by a factor of 1.1.	-----	III.B.3.a.3	(1a)
d) Pool-Swell Acceleration	Acceleration predicted by the PSAM. Pool acceleration is utilized in the calculation of acceleration loads on submerged components during pool swell.	----	III.B.3.a.4	(1a)
e) Wetwell Air Compression	Wetwell air compression is calculated by PSAM consistent with maximum pool swell elevation in B.1.b above.	----	II.A.2	(1b)

Table B-1 Mark II LOCA-related hydrodynamic loads
Summary table

Load or phenomenon	**Load specification**	**Load specification clarifying footnotes**	**NRC Evaluation**	**(Foot-note)**
f) Drywell Pressure	Methods of NEDM-10320 and NEDO-20533 Appendix B. Utilized in PSAM to calculate pool swell loads.	----	III.B.3.a.6	(1a)
2. Loads on Submerged Boundaries	Maximum bubble pressure predicted by the PSAM added uniformly to local hydrostatic below vent exit (walls and basemat) linear attenuation to pool surface. Applied to walls up to a maximum pool swell elevation.	----	III.B.3.b	(1a)
3. Impact Loads				
a) Small Structures	1.35 x Pressure-Velocity correlation for pipes and I beams based on PSTF impulse data and flat pool assumption. Variable pulse duration.	Note 3	III.B.3.c.1	(1a)
b) Large Structures	None - Plant unique load where applicable.	----	III.B.3.c.6	(1a)
c) Grating	P drag vs. grating area correlation and pool velocity vs. elevation. Pool velocity from the PSAM. P drag multiplied by dynamic load factor.	Note 2	III.B.3.c.3	(1a)
4. Wetwell Air Compression				

Table B-1 Mark II LOCA-related hydrodynamic loads
Summary table

Load or phenomenon	Load specification	Load specification clarifying footnotes	NRC Evaluation	(Foot-note)
a) Wall Loads	Direct application of the PSAM calculated pressure due to wetwell compression.	----	III.B.3.d.1	(1a)
b) Diaphragm Upward Loads	5.5 psid for diaphragm loadings only.	----	2.1.2.7	(1c)
5. Asymmetric LOCA Pool	Use 20% of maximum bubble pressure statistically applied to one-half of the submerged boundary. This load is to be applied statically together with normal hydrostatic pressure to the submerged portion of the containment.	----	II.A.3	(1b)C. Steam Condensation and Chugging Loads
1. Downcomer Lateral Loads				
a) Single-Vent Loads (24 in.)	Dynamic load to end of vent. Half sine wave with a duration of 3 to 6 ms and corresponding maximum amplitudes of 65 to 10 klbf.	Note 4	2.3.3.2	(1c)
b) Multiple-Vent Loads (24 in.)	Prescribed variation of load per vent vs. number of vents. Determined from single vent dynamic load specification and multivent reduction factor.	----	2.3.3.3	(1c)
c) Single/Multiple Vent Loads (28 in.)	Multiply basic 24" vent loads by factor f=1.34	----	2.3.2.1	(1c)
2. Submerged Boundary Loads				

Table B-1 Mark II LOCA-related hydrodynamic loads
Summary table

Load or phenomenon	**Load specification**	**Load specification clarifying footnotes**	**NRC Evaluation**	**(Foot-note)**
a) High/Medium Steam Flux Condensation Oscillation Load	Bounding CO pressure histories observed in 4TCO tests. Inphase application.	----	2.2.1.3	(1c)
b) Low Steam Flux Chugging Load	Conservative set of 10 sources derived from 4TCO tests. 7 sources are obtained by averaging each individual key chug and its largest adjacent chugs, the other 3 chugs obtained from 4TCO are used without averaging. Alternate load using 7 sources derived from the 4TCO key chugs without averaging. Sources are applied to plants using the IWEGS/MARS acoustic model assuming source desynchronization of 50 ms.	----	2.2.2.3	(1c)
-Symmetric Load	All vents utilize source of equal strength for each of the sources.			

Table B-1 Mark II LOCA-related hydrodynamic loads
Summary table

Load or phenomenon	**Load specification**	**Load specification clarifying footnotes**	**NRC Evaluation**	**(Foot-note)**
-Asymmetric Load Case	Source strengths S± = S (1±α) applied to all vents on + and - side of containment. Sources based on the symmetric sources. Asymmetric parameters α based on rms moment method of interpreting experimental 4TCO single-vent and JAERI multivent data.			
D. Secondary Loads				
1. Sonic Wave Load	Negligible load	----	III.E.1	(1a)
2. Compressive Wave Load	Negligible load	----	III.E.2	(1a)
3. Fallback Load on Submerged Boundary	Negligible load	----	III.E.5	(1a)
4. Thrust Loads	Momentum balance	----	III.E.6	(1a)
5. Friction Drag Loads on Vents	Standard friction drag calculations	----	III.E.7	(1a)
6. Vent Clearing Loads	Negligible load	----	III.E.8	(1a)

Footnotes For Table B-1

NOTE 1 NRC NUREG sections that describe the NRC specific load evaluation. Specific NUREGs are (a) NUREG-0487 (b) NUREG-0487 Supplement 1 and © NUREG-0808.

NOTE 2 Impact Drag Loads on Grating

The static drag load on grating in the pool-swell zone of the wetwell shall be calculated for grating with open area greater than or equal to 60% by forming the product of the pressure differential as given in Figure 4-40 of NEDO-21061, Revision 2, and the total area of the grating. To account for the dynamic nature of the initial loading, the load shall be increased by a multiplier given by:

Table B-1 Mark II LOCA-related hydrodynamic loads
Summary table

$$F_{SE} / D = 1 + \sqrt{1 + (0.0064\ Wf)^2}\ :\ for\quad Wf < 2000\ inch/\mathrm{sec},$$

where:

F_{SE}= static equivalent load
W = width of grating bars, in.
f = natural frequency of lowest mode, Hz
D = static drag load

NOTE 3 Impact Loads on Small Structures

The hydrodynamic loading function that characterizes pool impact on small horizontal structures shall have the versed sine shape.

Small structures are defined as pipes, I-beams, and other similar structures having one dimension less than or equal to 20 inches. The acceptance criteria are not applicable to the determination of ovaling stresses in cylindrical pipes.

$$p(t) = P_{\max}/2(1 - \cos 2\pi \frac{t}{\tau})$$

where:

p = pressure acting on the projected area of the structure, psi
P_{max} = the temporal maximum of pressure acting on the projected area of the structure, psi
t = time, sec
τ = duration of impact, sec

For both cylindrical and flat structures, the maximum pressure P_{max} and pulse duration τ will be determined as follows:

(a) The hydrodynamic mass per unit area for impact loading will be obtained from the appropriate correlation for a cylindrical or flat target in Figure 6-8 of NEDE-13426P.

(b) The impulse will be calculated using the equation

$$I_p = \frac{M_H}{A} V x \frac{1}{(32.2)(144)}$$

where:

I_p = impulse per unit area, psi-sec

$\frac{M_H}{A}$ = hydrodynamic mass per unit area, lbm/ft^2, from (a) above

V = impact velocity, ft/sec, determined according to Section A.2.

(c) The pulse duration will be obtained from the equation

$$\tau = \frac{0.0463D}{V} \quad \text{(cylindrical target)}$$

$$\tau = \frac{0.0011W}{V} \quad V \geq 7 \text{ ft/sec (flat target)}$$

$$\tau = \frac{0.0016W}{V} \quad V < 7 \text{ ft/sec (flat target)}$$ [116]

where:

- τ = pulse duration, sec
- D = diameter of cylindrical pipe, feet
- W = width of the flat structure, feet
- V = impact velocity, ft/sec

(d) The value of P_{max} will be obtained using the following equation:

$$P_{max} = 2I_p/\tau$$

For both cylindrical and flat structures, a margin of 35% will be added to the P_{max} values (as specified above) to obtain conservative design loads.

The load acceptance criteria, as specified above, corresponds to impact on rigid structures. The effect of finite flexibility of real structures will be accounted for in the following manner: When structural dynamic analysis is performed, the "rigid body" impact loads will be applied; however, the masses of the impacted structures will be adjusted by adding on the hydrodynamic masses of impact. The numerical values of hydrodynamic masses will be obtained from the appropriate correlations for cylindrical and flat structures in Figure 6-8 of NEDE-13426P.

NOTE 4 Steam Condensation and Chugging Loads

Single-Vent Lateral Loads

The following dynamic single-vent load specification will be used:

A tip lateral force given by:

$$F(t) = A(\tau)\sin(\pi t/\tau) \quad 0 \geq t \leq \tau$$

where $A(\tau) = (50 - 20\tau/3)$klbf for 3 ms $\leq \tau \leq$ 6 ms shall be applied to each downcomer with τ varied between 3 and 6 ms as indicated.

In addition, a separate assessment shall be made for a load with a tip lateral force of

$$F(t) = 65\sin\left(\frac{\pi t}{\tau}\right) \ klbf \quad 0 \leq t \leq 3 \ ms$$

for each downcomer.

SRP Section 6.2.1.1.C
Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in (Draft) Revision 7, dated April, 1996 of this SRP. See ADAMS accession number ML061710393.

In addition this SRP section was administratively updated in accordance with NRR Office Instruction, LIC-200, Revision 1, "Standard Review Plan (SRP) Process." The revision also adds standard paragraphs to extend application of the updated SRP section to prospective submittals by applicants pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 2, dated 200X.

Review Responsibilities - Reflects changes in review branches resulting from reorganization and branch consolidation. Change is reflected throughout the SRP.

I. AREAS OF REVIEW

Reformatted the section with new numbering system. Incorporated reference to 10 CFR Part 52 from draft revision 7 - April 1996. Incorporated generic paragraphs relating to certified designs, ESPs, and COLs.

II. ACCEPTANCE CRITERIA

Reformatted the section with new numbering system. Incorporated reference to 10 CFR Part 52 from draft revision 7 - April 1996. Incorporated generic paragraphs relating to certified designs, ESPs, and COLs.

III. REVIEW PROCEDURES

Reformatted the section with new numbering system. Incorporated reference to 10 CFR Part 52 from draft revision 7 - April 1996. Incorporated generic paragraphs relating to certified designs, ESPs, and COLs.

IV. EVALUATION FINDINGS

None

V. IMPLEMENTATION

None

VI. REFERENCES

None