NUREG-0800
(Formerly NUREG-75/087)

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 and Severe Accident Branch (SCSB)[1]

Secondary None

I. AREAS OF REVIEW

For ~~Mark I, II, and III~~ [2]pressure-suppression type boiling water reactor (BWR) plant containments, the SCSB[3] 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.

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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.

Review Interfaces:[4]

The SCSB[5] will coordinate other branches'[6] evaluations that interface with the overall review of the containment as follows:

1. The[7] Instrumentation &~~and~~ Control Systems Branch ~~(ICSB)~~(HICB)[8], 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.

2. The ~~Equipment Qualification~~Plant Systems Branch ~~(EQB)~~(SPLB)[9], 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.

3. The Mechanical Engineering Branch ~~(MEB)~~(EMEB), 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~~EMEB 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~~EMEB[10] 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.

4. The ~~Structural~~Civil Engineering and Geosciences Branch ~~(SEB)~~(ECGB)[11], as part of its primary review[12] 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, Reference~~.~~ 1d)[13].

5. ~~Accident Evaluation~~The Materials and Chemical Engineering Branch ~~(AEB)~~ (EMCB)[14] will review fission product control features of containment heat removal systems as part of its primary review responsibility for SRP Section 6.5.2.

6. The review of proposed technical specifications at the operating license or design certification[15] stage of review pertaining to the bypass leakage surveillance is performed by the[16] ~~Standardization and Special Projects~~Technical Specifications Branch ~~(SSPB)~~(TSB)[17] as part of its primary review responsibility for SRP Section 16.0.

7. For new plant applicants, the Probabilistic Safety Assessment Branch (SPSB) coordinates and performs shutdown risk assessment reviews, including containment analysis issues, as part of its primary review responsibility for SRP Section 19.1 (Proposed).[18]

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 apply~~applies~~[19] to the design and functional capability of BWR pressure-suppression type containments. SCSB[20] accepts the containment design if the relevant requirements of General Design Criteria~~on~~[21] 4, 16, 50, ~~and~~ 53, 13, and 64[22], and of 10 CFR Part 50, §50.34(f)(3)(v)(A)(1) and §50.34(f)(3)(v)(B)(1)[23] are complied with. The relevant requirements are as follows:[24]

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.

4. General Design Criterion 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 assure adequate safety.

5. General Design Criterion 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.[25]

6. For those applicants subject to 10 CFR 50.34(f):

   a. 10 CFR 50.34(f)(3)(v)(A)(1) as it relates to containment integrity being maintained during an accident that releases hydrogen generated from a 100-percent fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post accident inerting.[26]

   b. 10 CFR 50.34(f)(3)(v)(B)(1) as it relates to containment integrity being maintained during inadvertent full actuation of the post-accident inerting system, if installed.[27]

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

a~~1~~.[28] In meeting the requirements of General Design Criteria 16 and 50 regarding the design margin for BWR pressure-suppression~~Mark I, II and III~~[29] 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~~Mark I, II and III~~[30] containment environmental response to loss-of-coolant accidents are found in NUREG-0588 (Reference~~.~~ ~~35~~7)[31].

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~~.~~ ~~23~~36)[32] that was used in the ABWR~~Grand Gulf~~ analysis[33] and evaluated by SCSB[34]. 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~~have~~[35] 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 and evaluated by SCSB in the ABWR FSER (Reference 13).[36]

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~~Mark I, II and III~~[37] 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.

b~~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 (References~~.~~ ~~39~~10 and 1c)[38].

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~~.~~ 1b)[39], and Appendix B[40] to this SRP section. These loads and criteria supersede those discussed in references ~~36, 37 and 38~~4, 5 and 6.[41] 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~~.~~ 1e)[42].

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 1g)[43]. 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.

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 37) which, in part, conforms with NUREGS 0802, 0808, and 0978. This model was used in the standard plant analysis and evaluated by SCSB in the ABWR FSER.[44]

c~~3~~. In meeting the requirements of General Design Criteria 16 and 50 regarding the containment design margin for Mark III and ABWR[45] 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 EMEB~~MEB~~[46] 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.

d~~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. Plants with ABWR containments should follow the specific acceptance criteria for Mark II containments.[47]

e~~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. 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.[48]

f~~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, and ABWR[49] plants, and the operating deck of Mark II plants, against loss of integrity from negative pressure transients or postaccident 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 21)[50], 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.

g~~7~~. In meeting the requirements of General Design Criterion 50, with respect to design margin for item ~~6~~f[51] 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.

h~~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 (Reference~~.~~ 1d)[52]. 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.

i~~9~~. For BWR pressure-suppression~~Mark I, II and III~~[53] plants, the local suppression pool temperature should not exceed 93°C (200°F)[54] or the acceptance criteria specified in Section 5.1 of NUREG-0783 (Reference~~.~~ 1f)[55].

j~~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)),[56] 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 (References 11 and 12)[57], 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.

k. In meeting the requirements of 10 CFR 50.34(f)(3)(v)(A)(1), applicants subject to this article 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 either hydrogen burning in containment or initiation of the post-accident inerting system, if installed, should be analyzed. Unless specifically known, the post-accident inerting gas should be assumed to be carbon dioxide.[58]

l. In meeting the requirements of 10 CFR 50.34(f)(3)(v)(B)(1), applicants subject to this article should evaluate the containment design's capability to withstand inadvertent full actuation of the post-accident inerting system, if installed. The peak pressure caused by inadvertent actuation of the post-accident inerting system should be less than the containment design pressure.[59]

<u>Technical Rationale:</u>[60]

The technical rationale for application of the above acceptance criteria to pressure-suppression type BWR containments 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. Following 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 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 or initiation of the post-accident inerting system, if installed, 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 or initiation of the post-accident inerting system, if installed. 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.

8. 10 CFR 50.34(f)(3)(v)(B)(1) requires that the containment be designed to withstand inadvertent actuation of the post-accident inerting system, if installed. 10 CFR 50.34(f) promulgates hydrogen control requirements which include the option of a post-accident inerting system. A post-accident inerting system floods containment with an inert gas, such as carbon dioxide, during a hydrogen releasing accident. If inadvertently actuated during normal operation, containment could potentially be pressurized by the inerting system. In accordance with this regulation, the containment must be designed to withstand this potential inadvertent pressurization to ensure that its integrity is maintained, thus precluding the release of radioactivity to the environment.

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, other~~the secondary~~[61] 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 SCSB reviews the analyses of the drywell and wetwell temperature and pressure response for BWR pressure-suppression~~Mark I, II and III~~[62] containments. The SCSB[63] performs confirmatory analyses, when necessary, using the CONTEMPT-LT computer code (References 22 and 23)[64]. Input data for the code, including mass and energy release data,[65] are generally taken from the safety analysis report.

   The SCSB[66] 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.[67]

The SCSB evaluates analyses of both the short-term and long-term pressure and temperature responses of Mark III and ABWR[68] 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 SCSB[69] reviews the containment response analysis presented in the safety analysis report to determine that the acceptance criteria in subsection II have been satisfied.

Design certification applicants should meet the margins for containment design pressure and containment/drywell design differential pressure specified in specific criterion "a" For Mark III plants. 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 design certification 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.[70]

The SCSB[71] 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 for both Mark III and ABWR plants[72] in an acceptable manner. The code has been verified by the General Electric Mark III test program.

The SCSB 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 SCSB, in conjunction with its consultants, will initiate a detailed review of the methods. In this case, the SCSB[73] 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 SCSB reviews analyses of the drywell response to ~~either~~ a recirculation line rupture, ~~or a~~ steam line rupture, or main feedwater line break[74] as presented in the safety analysis report. The SCSB[75] 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 SCSB verifies that the containment is designed to withstand either hydrogen burning or initiation of the post-accident inerting system, if installed, during an accident that releases hydrogen from a 100% fuel clad metal-water reaction as described in specific criterion II.k of this SRP section.[76]

If a post-accident inerting system is utilized, the SCSB verifies the containment is designed to withstand inadvertent actuation of this system.[77]

Modifications to the CONTEMPT-LT computer code have been made which provide the capability to perform confirmatory analyses of the Mark III and ABWR[78] drywell pressure response.

2. The review of the dynamic loads associated with a LOCA has~~have~~[79] 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~~have~~[80] been concluded with the issuance of NUREG-0661 for Mark I plants, 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.[81]

3. For Mark III and ABWR[82] plants, the SCSB 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 SCSB reviews analyses of the consequences of postulated ruptures in these lines. The SCSB[83] 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 HICB~~ICSB~~[84] evaluates the effectiveness of the detection instrumentation and isolation devices to mitigate the consequences of a pipe rupture and to meet[85] the electrical design criteria for these systems under SRP Section 7.3.

4. The SCSB reviews the analyses of the suppression pool temperature for transients involving the actuation of reactor coolant system safety/relief valves in BWR pressure-suppression~~Mark I, II and III~~[86] plants. The SCSB[87] evaluates the assumptions and conservatisms employed in the analyses to assure that the acceptance criteria set forth in NUREG-0783 are met.

   The SCSB[88] 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 SCSB evaluates analyses of bypass leakage capability. The SCSB 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. SCSB[89] will advise the ~~AEB~~PERB[90] of the bypass leakage.

6. The SCSB evaluates the conservatism of potential depressurization transients. In evaluating surveillance and test programs for vacuum relief systems, the SCSB[91] uses the

results of previous reviews and operating experience with similar systems to determine their adequacy. At the operating license or design certification[92] stage, the TSB~~SSPB~~[93] 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~~ECGB[94] will review the design of unique flow-limiting devices which are identified during the SCSB[95] review of the containment subcompartments.

8. The SCSB reviews the accuracy and range of the instrumentation provided to monitor the post-accident environment. The ~~ICSB~~HICB[96], under SRP Section 7.5, and the ~~EQB~~SPLB[97], under SRP Section 3.11, have review responsibility for the acceptability of, and the qualification test program for,[98] the sensing and actuation instrumentation of the plant protection system and the postaccident monitoring instrumentation and recording equipment.

9. 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).[99]

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[100]

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.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10CFR 52.[101] 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.

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 except as noted below.[102]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides, regulations,[103] NUREGs and the following:[104]

1. Revision ~~2~~3[105] 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~~/~~-[106]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."

Appendix A to SRP Section 6.2.1.1.C
(Formerly~~Formally~~[107] Appendix I)
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),~~;~~[108] the capability of the Mark II containment is approximately 46.5 cm^2 (.05~~0.5~~ ft^2,[109], and the Mark III design has a capability of $A/\sqrt{K}$ = 929 cm^2 (1 ft^2).[110]

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,[111] ($A/\sqrt{K}$) 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)[112] 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[113] and OL applications with Mark I, Mark II and Mark III containments (see also subsection V of this SRP section). The positions of ~~Revision 2~~this revision to Appendix A of this SRP section do~~es~~ not apply to plants with an operating license issued prior to January 1983~~operating reactors~~. [114]

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)
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)

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

Load or phenomenon	**Load specification**	**Load specification clarifying footnotes**	**NRC Evaluation**	**(Foot-note)**
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)
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				
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)

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

Load or phenomenon	**Load specification**	**Load specification clarifying footnotes**	**NRC Evaluation**	**(Foot-note)**
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				
a) High/Medium Steam Flux Condensation Oscillation Load	Bounding CO pressure histories observed in 4TCO tests. Inphase application.	----	2.2.1.3	(1c)

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

Load or phenomenon	Load specification	Load specification clarifying footnotes	NRC Evaluation	(Foot-note)
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.			
-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)

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

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 (c) 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:

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

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.

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

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

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

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.

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

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:

$$F(t) = A(\tau)\sin\left(\frac{\pi t}{\tau}\right) \quad 0 \le t \le \tau$$

where $A(\tau) = (50 - 20\tau/3)$klbf for 3 ms $\le \tau \le$ 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

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

for each downcomer.

SRP Draft Section 6.2.1.1.C
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 to reflect current PRB name and responsibility for this SRP Section.
2.	Editorial	Several Integrated Impacts have been incorporated to address ABWRs. The opening sentence has been revised to generically refer to all BWR containments.
3.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section.
4.	SRP-UDP format item	Added "Review Interfaces" heading to Areas of Review. Reformatted existing description of review interfaces in numbered format to describe how SCSB reviews aspects of Pressure Suppression Type BWR Containments under other SRP Sections and how other branches support the review.
5.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section.
6.	SRP-UDP format item	Changed "branch" to "branches'" to make Review Interface lead in statement consistent with SRP-UDP guidance.
7.	Editorial	"The" was added to clarify the sentence.
8.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for SRP Section 7.3.
9.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for SRP Section 3.11.
10.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for SRP Sections 3.2.1, 3.2.2, 3.6.2, 3.9.2, 3.9.3, and 3.10 (3 identical changes in this paragraph).
11.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for SRP Section 3.8.3.
12.	Editorial	Added the word "review" for consistency with other review interface items.
13.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format requirements.
14.	Current PRB names and abbreviations/Editorial	Editorial change to reflect current PRB name and responsibility for SRP Section 6.5.2. Also, "The" was added to clarify the sentence.
15.	10 CFR 52 applicability related change	Added "or design certification" to indicate that Technical Specifications may be reviewed at the design certification stage.

Item	Source	Description
16.	Editorial	"The" was added to clarify the sentence.
17.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for SRP Section 16.0.
18.	**Integrated Impact 1502**	This review interface identifies reviews conducted to satisfy SECY 93-087 and NUREG-1449 guidance on Shutdown and Low Power Operations. The staff requested that design certification applicants complete an assessment of shutdown and low-power risk. The shutdown and low-power risk assessment must identify design-specific vulnerabilities and weaknesses and document consideration and incorporation of design features that minimize such vulnerabilities. Containment analysis issues related to containment integrity during shutdown conditions are included in the shutdown risk assessments. Consideration of this issue in the shutdown and low-power risk assessment is the responsibility of the SPSB and will be included in the proposed SRP Section 19.1 on risk assessments.
19.	Editorial	"Applies" was changed to "apply". The plural form of the verb is correct in this sentence.
20.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section.
21.	Editorial	Changed "criterion" to "criteria" to reflect that more than one criterion is being discussed.
22.	Editorial	GDCs 13 and 64 were added to the Acceptance Criteria. These two GDCs are invoked in specific criterion j. Also, these two criterion are called out in the other two containment design SRP sections (6.2.1.1.A and 6.2.1.1.B). Therefore, for consistency between the general and specific acceptance criterion within SRP 6.2.1.1.C, and for consistency with other similar sections, GDC 13 and 64 are being added.
23.	**Integrated Impacts 853 and 884**	Added reference to 10 CFR 50, §50.34(f)(3)(v)(A)(1) and §50.34(f)(3)(v)(B)(1).
24.	Editorial	"Follow" was changed to "follows" to correct the word to plural usage.
25.	Editorial	GDCs 13 and 64 were added to the Acceptance Criteria. These two GDCs are invoked in specific criterion j. Also, these two criterion are called out in the other two containment design SRP sections (6.2.1.1.A and 6.2.1.1.B). Therefore, for consistency between the general and specific acceptance criterion within SRP 6.2.1.1.C, and for consistency with other similar sections, GDC 13 and 64 are being added.

Item	Source	Description
26.	**Integrated Impact 884**	Added a general criterion for 10CFR50.34(f)(3)(v)(A)(1) regarding designing containment to meet hydrogen burning or post-accident inerting system actuation during an accident.
27.	**Integrated Impact 853**	Added a general criterion for 10CFR50.34(f)(3)(v)(B)(1) regarding designing containment to withstand inadvertent actuation of the post-accident inerting system, if installed.
28.	Editorial	Numbered specific acceptance criteria were changed to letters for clarity and consistency with other SRP sections. Numbers were already used in the general acceptance criteria above.
29.	**Integrated Impact 926**	The sentence was revised to show its applicability to all types of BWR containments, including ABWRs.
30.	**Integrated Impact 926**	The sentence was revised to show its applicability to all types of BWR containments, including ABWRs.
31.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format requirements. The reference number was revised to reflect changes made to SRP 6.2.1 Reference section.
32.	SRP-UDP format item/reference verification	Format change to make the citation of references consistent with the SRP-UDP format requirements. The reference was revised to incorporate the updated Mark III analytical model utilized and verified by the staff in the ABWR FSER (see PI 24492).
33.	Editorial	The "Grand Gulf analysis" was replaced with the "ABWR analysis" to reflect the updated Mark III analytical model utilized and verified by the staff in the ABWR FSER (see PI 24492).
34.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section.
35.	Editorial	"Have" was changed to "has". The singular form of the verb is correct in this sentence.
36.	**Integrated Impact 926**	A new specific criterion for ABWR short and long term temperature/pressure response was added.
37.	**Integrated Impact 926**	The sentence was revised to show its applicability to all types of BWR containments, including ABWRs.
38.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format requirements. The reference number was revised to reflect changes made to SRP 6.2.1 Reference section.
39.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format requirements.

SRP Draft Section 6.2.1.1.C

Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
40.	Unverified reference	This reference cannot be verified to be the most current reference that is still approved by the NRC.
41.	Editorial	The reference numbers were revised to reflect changes made to SRP 6.2.1 Reference section.
42.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format requirements.
43.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format requirements.
44.	**Integrated Impact 926**	Added a new paragraph for ABWR dynamic load analyses.
45.	**Integrated Impact 926**	Added reference to ABWRs in the specific criterion for guard pipes.
46.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for SRP Section 3.6.2.
47.	**Integrated Impact 926**	Revised the steam bypass specific criterion to address ABWRs.
48.	**Integrated Impact 926**	Added a sentence for ABWRs regarding the assumed duration of the containment leak rate during a LOCA.
49.	**Integrated Impact 926**	Reference to ABWRs was added to the negative pressure transient specific criterion.
50.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format requirements.
51.	Editorial	Changed the reference from "item 6" to "item f" to reflect changes to the numbering/lettering of the specific criteria.
52.	SRP-UDP format item.	Format change to make the citation of referencesconsistent with the SRP-UDP format requirements.
53.	**Integrated Impact 926**	The sentence was revised to show its applicability to all types of BWR containments, including ABWRs.
54.	SRP-UDP format item, Metrication policy implementation	The existing criterion of 200° F for the suppression pool temperature limit was converted to 93°C using the guidance of Federal Standard 376B. See enclosed conversion documentation.
55.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format requirements.
56.	**Integrated Impact 996**	Added citation of 10 CFR 50.34(f)(2)(xvii) related to the existing citation of II.F.1 of NUREG 0737/NUREG 0718.

Item	Source	Description
57.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format requirements.
58.	**Integrated Impact 884**	Added a specific criterion for 10CFR50.34(f)(3)(v)(A)(1) regarding evaluating containment design to meet hydrogen burning or post-accident inerting system actuation during an accident.
59.	**Integrated Impact 853**	Added a specific criterion for 10CFR50.34(f)(3)(v)(B)(1) regarding evaluating containment design to meet inadvertent actuation of the post-accident inerting system if installed.
60.	SRP-UDP format item, Develop Technical Rationale	Added Technical Rationale for GDCs 4, 16, 50, 53, 13, and 64, and 10 CFR 50.34(f)(3)(v), articles (A)(1) and (B)(1). Technical Rationale is a new SRP-UDP format item.
61.	Editorial	The word "other" was substituted for "the secondary". The term "secondary review branch" is inappropriate since there is no designated secondary review branch for this SRP Section.
62.	**Integrated Impact 926**	The sentence was revised to show its applicability to all types of BWR containments, including ABWRs.
63.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section (2 identical changes in this paragraph).
64.	SRP-UDP format item/Unverified reference.	Format change to make the citation of references consistent with the SRP-UDP format requirements. 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 this SRP section. This reference, therefore, cannot be verified to be the most current reference still approved by the NRC.
65.	Editorial	A comma was added to clarify the meaning of the sentence.
66.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section.
67.	**Integrated Impact 926**	Added the ABWR design basis loss-of-coolant accident (feedwater line break) to the review procedure on LOCA analysis.
68.	**Integrated Impact 926**	Added ABWRs to the review procedure on short and long term temperature/pressure response.
69.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section (2 identical changes in this paragraph).

Item	Source	Description
70.	**Integrated Impact 288**	Added a sentence to Review Procedures that DC applicants are reviewed for incorporation of the CP containment design pressure and containment/drywell design differential pressure margins.
71.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section.
72.	**Integrated Impact 926**	Added "for both Mark III and ABWR plants" to show that the Mark III analytical model is also applicable to ABWRs.
73.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section (3 identical changes in this paragraph).
74.	**Integrated Impact 926**	Added the feedwater line break to this review procedure for applicability to ABWRs.
75.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section (2 identical changes in this paragraph).
76.	**Integrated Impact 884**	Added a Review Procedure for 10CFR50.34(f)(3)(v)(A)(1) regarding evaluating containment design to meet hydrogen burning or post-accident inerting system actuation during an accident.
77.	**Integrated Impact 853**	Added a Review Procedure for 10CFR50.34(f)(3)(v)(B)(1) regarding evaluation of the containment design pressure against inadvertent actuation of the post-accident inerting system if such a system is installed.
78.	**Integrated Impact 926**	Added reference to the ABWR. The CONTEMPT code was utilized to review the ABWR containment analysis (see PI 24492).
79.	Editorial	Changed "have" to "has" to correct and clarify the sentence.
80.	Editorial	Changed "have" to "has" to correct and clarify the sentence.
81.	**Integrated Impact 926**	Added a review procedure referring to the reference document for ABWR dynamic load analysis.
82.	**Integrated Impact 926**	Added ABWRs to the review procedure regarding guard pipes.
83.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section (3 identical changes in this paragraph).
84.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for SRP Section 7.3.

SRP Draft Section 6.2.1.1.C

Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
85.	Editorial	The words "to meet" were added to change the phrase to "to meet the electrical design criteria..." Without this change, the verb from the linking prepositional phrase is applied to the subject phrase implying the applicant should "mitigate the electrical design criteria". Adding "to meet" clarifies and gives proper meaning to the sentence.
86.	**Integrated Impact 926**	The sentence was revised to show its applicability to all BWR containments, including ABWRs.
87.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section (2 identical changes in this paragraph).
88.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section.
89.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section (3 identical changes in this paragraph).
90.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for SRP Section 15.6.5.A.
91.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section (2 identical changes in this paragraph).
92.	10 CFR 52 applicability related change	Added "or design certification" to indicate that technical specifications may be reviewed at the OL or DC stage.
93.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for SRP Section 16.0.
94.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for SRP Section 3.8.3.
95.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for this SRP Section
96.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for SRP Section 7.5.
97.	Current PRB names and abbreviations	Editorial change to reflect current PRB name and responsibility for SRP Section 3.11.
98.	Editorial	Two commas were added to the sentence to clarify its meaning.

SRP Draft Section 6.2.1.1.C

Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
99.	**Integrated Impact 1502**	This paragraph describes the type of containment analyses required during shutdown conditions. Containment interaction and response (including containment closure times, temperatures and radiological conditions) will be dependent upon the results of analyses to develop a bases for critical thermodynamic events such as containment temperatures and postulated times to core uncovery during a loss of shutdown decay heat removal.
100.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
101.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
102.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
103.	Editorial	Added "regulations" to indicate that a 10 CFR regulation is now part of the acceptance criteria.
104.	Editorial	Revised format of this implementation section into separate indented items for each numbered schedule reference for clarity and consistency with other SRP sections.
105.	Editorial	Changed the revision number for Appendix A from 2 to 3 to reflect the current revision.
106.	Editorial	Deleted a slash (/) and added a dash (-) to correct the title of the reference.
107.	Editorial	"Formally" was replaced with "Formerly" which is the correct word for this phrase.
108.	Editorial/SRP-UDP format item, Metrication policy implementation	The existing criteria of 0.02 ft^2 for the allowed steam bypass area for Mark I containments was converted to 18.6 cm^2 using the guidance of Federal Standard 376B. See enclosed conversion documentation. Also, a semicolon was changed to a comma for clarity and correctness.
109.	Editorial/SRP-UDP format item, Metrication policy implementation	The existing value of 0.5 ft^2 for the allowed steam bypass area for Mark II containments, which appears in the 2nd paragraph of section A in Appendix A, is assumed to be a typographical error. In section B.1 of Appendix A this same criterion is given as .05ft^2. Also, section 6.2.1.8 of the ABWR FSER gives this same criterion as 46.5cm^2 (.05ft^2) (see PI 24496). The existing criterion of 0.5 ft^2, therefore, was changed to .05 ft^2. This new value was then converted to 46.5 cm^2 using the guidance of Federal Standard 376B. See enclosed conversion documentation.

Item	Source	Description
110.	SRP-UDP format item, Metrication policy implementation	The existing criteria of 1 ft^2 for the allowed steam bypass area for Mark III containments was converted to 929 cm^2 using the guidance of Federal Standard 376B. See enclosed conversion documentation.
111.	SRP-UDP format item, Metrication policy implementation	The existing criteria of .05 ft^2 and 1 ft^2 for the required steam bypass capability for small breaks for Mark II and III containments were converted to 46.5 cm^2 and 929 cm^2 using the guidance of Federal Standard 376B. See enclosed conversion documentation.
112.	SRP-UDP format item, Metrication policy implementation	The existing criterion of one inch diameter for the maximum allowed equivalent containment leak for Mark I containments was converted to 2.54 cm diameter using the guidance of Federal Standard 376B. See enclosed conversion documentation.
113.	10 CFR 52 applicability related change.	Standard design certification (DC) terminology was added to the implementation section of Appendix A as required by the SRP-UDP program.
114.	Applicability statement for operating plants	Changed applicability of Appendix A for operating plants from "revision 2" to "plants with an operating license issued prior to January 1983" in order to show that operating plants need not comply with the provisions of this revision.
115.	Editorial	Table B-1 in Appendix B has been changed from a landscape to a vertical orientation and placed in a boxed table. These changes were made for clarity, readability, and to facilitate use of standard computer word processing capabilities. Additionally, Appendix B cannot be verified to be the most current reference that is still being used by the NRC. Since Appendix B is unverified and applies to Mark II containments (which were designed in English units), and since metrication of Table B-1 would be time consuming and complex, the criteria and formulae are left in English units.
116.	Editorial	By unit analysis and comparison to the other equations for τ, it is evident that this equation must have a "V" in the denominator.

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SRP Draft Section 6.2.1.1.C

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
288	Consider revising SRP 6.2.1.1.C to incorporate Construction Permit stage design margin acceptance criteria into the Design Certification process. The staff enforced the Construction Permit stage design margins in the ABWR and CE System 80+ FSERs.	Section III, Review Procedures, item 1, new paragraph.
289	This II recommended adding the review of hardened wetwell vent capability in Mark I containments. Since the Generic Letter that originated this II handled the hardened wetwell vent issue on a plant-by-plant basis, no generic changes are recommended for this SRP Section.	None
853	Consider revising SRP 6.2.1.1.C to add review of the containment capability to withstand inadvertent actuation of the post-accident inerting system if such a system is installed.	Section II, Acceptance Criteria, general criterion 6.b, specific criterion l. Section III, Review Procedures, item 1.
884	Consider revising SRP 6.2.1.1.C to add review of the containment capability to withstand either hydrogen burning or actuation of the post-accident inerting system (if installed) during an accident that releases hydrogen from a 100% fuel clad metal-water reaction.	Section II, Acceptance Criteria, general criterion 6.a, specific criterion k. Section III, Review Procedures, item 1.
926	Consider revising SRP 6.2.1.1.C to add appropriate references regarding the review of ABWR containments as shown in the ABWR FSER.	Subsection II, Acceptance Criteria, specific criteria a, b, c, d, e, f, and i. Subsection III, Review Procedures 1, 2, 3, and 4.
996	Consider citing 10 CFR 50.34(f)(2)(xvii) related to TMI action plan item II.F.1 in the SRP Section.	Subsection II, Acceptance Criteria, specific criterion j.
1502	Consider revising SRP 6.2.1.1.C to incorporate staff guidance on containment analyses that should be performed to develop sufficient bases for shutdown operations.	Subsection I, Areas of Review, new Review Interface 7. Subsection III, Review Procedures, new item 9.