**U.S. NUCLEAR REGULATORY COMMISSION**
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Division 1

# DRAFT REGULATORY GUIDE

Contact: S. Shaukat
(301) 251-7646

## DRAFT REGULATORY GUIDE DG-1203

*(New Regulatory Guide)*

## CONTAINMENT PERFORMANCE FOR PRESSURE LOADS

## A. INTRODUCTION

This new regulatory guide is intended to provide licensees and applicants with guidance concerning methods which the staff of the U.S. Nuclear Regulatory Commission (NRC) consider acceptable for demonstrating containment performance in nuclear power plants, in accordance with regulatory requirements and the Commission's performance goals for pressure loadings of containment structures.

The regulatory requirements for the design and performance of nuclear reactor containments are provided in Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the *Code of Federal Regulations* [10 CFR Part 50, Ref. 1]. The "General Design Criteria [GDC] for Nuclear Power Plants" set forth in Appendix A to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), establish minimum requirements for the principal design criteria for water-cooled nuclear power plants. The requirements for combustible gas control within the containment for currently licensed reactors and future water-cooled reactor applicants and licensees, are provided in 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors" (Ref. 1).

The regulatory requirements for early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities are provided in Title 10, Part 52, "Licenses, Certification, and Approvals for Nuclear Power Plants," of the *Code of Federal Regulations* [10 CFR Part 52, Ref. 2]. Design-specific probabilistic risk assessment (PRA) information requirements for standard design certifications can be found in 10 CFR 52.47, "Contents of applications; technical information" (Ref. 2). Plant-specific PRA information requirements for combined

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This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received final staff review or approval and does not represent an official NRC final staff position.

Public comments are being solicited on this draft guide (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rulemaking, Directives, and Editing Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; emailed to nrcrep.resource@nrc.gov; submitted through the NRC's interactive rulemaking Web page at http://www.nrc.gov; faxed to (301) 415-5144; or hand-delivered to Rulemaking, Directives, and Editing Branch, Office of Administration, U.S. NRC, 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. on Federal workdays. Copies of comments received may be examined at the NRC's Public Document Room, 11555 Rockville Pike, Rockville, MD. Comments will be most helpful if received by February 9, 2009.

Electronic copies of this draft regulatory guide are available through the NRC's interactive rulemaking Web page (see above); the NRC's public Web site under Draft Regulatory Guides in the Regulatory Guides document collection of the NRC's Electronic Reading Room at http://www.nrc.gov/reading-rm/doc-collections/; and the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML082050539.

licenses are provided in 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report” (Ref. 2). Requirements for holders of a combined license under Subpart C of 10 CFR Part 52 to develop, maintain, and upgrade PRAs are provided in 10 CFR 50.71(h) (Ref. 1).

Specifically, GDC 1, “Quality Standards and Records,” requires, in part, that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. GDC 16, “Containment Design,” requires that the reactor containment and its associated systems be provided to establish an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment and to ensure that design conditions important to safety are not exceeded for as long as required for postulated accident conditions. GDC 50, “Containment Design Basis,” requires that the reactor containment structure (including access openings, penetrations, and containment heat removal systems) be designed so that the structure and its internal compartments will have the capacity to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculate pressure and temperature conditions caused by a loss-of-coolant accident.

Meeting these requirements provides assurance that containments used for nuclear power plants will be designed to perform their containment function, as long as required, to prevent or mitigate the spread of radioactive material under various containment internal pressure challenges, thus maintaining the plant in a safe condition.

Regulatory Guides 1.7, “Control of Combustible Gas Concentrations in Containment” (Ref. 3), 1.57, “Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components” (Ref. 4), and 1.136, “Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments” (Ref. 5), provide an acceptable approach for meeting the requirements of 10 CFR 50.44 and demonstrating containment structural integrity.

The NRC has also established performance goals for containment structures in nuclear power plants under severe accidents, as specified in SECY-90-016, “Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements” (Ref. 6), and SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs” (Ref. 7), and the corresponding staff requirements memoranda (SRMs).

The NRC issues regulatory guides to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency’s regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required.

This regulatory guide contains information collection requirements by the Office of Management and Budget (OMB) and approved under OMB control number 3150-0011for 10 CFR Part 50 (Ref. 1) and under OMB control number 3150-0151 for 10 CFR Part 52 (Ref. 2). The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays a currently valid OMB control number.

# B. DISCUSSION

## Background

For metal containments, the scope of this regulatory guide includes those components designed and constructed in accordance with the "Rules for Construction of Nuclear Facility Components," in the ASME Boiler & Pressure Vessel Code ("the ASME Code") published by the American Society of Mechanical Engineers (ASME) (Ref. 8). The rules for Class MC components, which include metal containments and appurtenances, as well as metal portions of concrete containments that are not backed by concrete are set forth in Section III, Division 1, Subsection NE of the ASME Code entitled "Class MC Components." The ASME Code classifies components such as piping, pumps, and valves, which are part of the containment system or which penetrate or are attached to the containment vessel, as either Class 1 or Class 2. Such components must meet the criteria of one of the other applicable subsections of the ASME Code.

The ASME and the American Concrete Institute (ACI) have jointly published the "Code for Concrete Containments," which is included in the ASME Code as Section III, Division 2, Subsection CC, and is also known as ACI Standard 359-01, "Code for Concrete Containments" (Ref. 9). For concrete containments, both reinforced and prestressed, the scope of this regulatory guide includes only the components designed and constructed in accordance with the rules of Subsection CC of the ASME Code (i.e., the structural concrete pressure-resisting shells and shell components, steel metallic liners, penetration liners extending the containment liner through the surrounding shell concrete, and tendons and anchorage system, if applicable). Steel parts of concrete containment and their appurtenances, which are not backed by structural concrete (e.g., removable steel head), are subject to the requirements of the rules of Subsection NE of Division I of the ASME Code.

The provisions of this guide may be used for the current light-water reactors, as well as for future advanced reactors, such as the advanced pressurized-water reactors and the economic simplified boiling-water reactors. While this regulatory guide is only directly applicable to metal and concrete containments for light-water reactors, the principles contained herein may be applied to non light-water reactor containments, subject to review by the NRC.

## Ultimate Pressure Capacity of Containment

This draft regulatory guide describes a method and acceptance criteria for the ultimate pressure capacity of concrete and steel containments. NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories - An Overview" (Ref. 10), contains a discussion on the results from a series of tests conducted on containment vessel models made of reinforced concrete, prestressed concrete, and on freestanding steel containments. Conclusions from the containment model tests identified global free-field strains of the order of 2.0 – 3.0 percent for steel containments, 1.5 – 2.0 percent for reinforced concrete containments, and 0.5 – 1.0 percent for prestressed concrete containments, which can be achieved before failure or rupture occur. As noted in the report (Ref. 10), with the added complexity of actual containments, it is reasonable to assume that there is a good probability that local strain risers are present, and may possibly be more severe than in any of the models tested. Therefore, the strain limits discussed in Regulatory Position C.1 of this Draft Regulatory Guide have incorporated some conservatism over the containment model test results and also rely on the results of other studies (Ref. 11 and 12). For noncylindrical containments, and other regions of containment and discontinuities (e.g., containment head or base), this draft regulatory guide also identifies the need for the applicant to develop strain limits for these locations, for staff review on a case-by-case basis.

## Containment Pressure Integrity for Hydrogen-Generated Pressure Loads

As required by 10 CFR 50.44(b)(2)(i), all currently licensed boiling-water reactors with Mark I- or Mark II-type containments must have an inerted atmosphere. Furthermore, 10 CFR 50.44(b)(2)(ii) requires that all currently licensed boiling-water reactors with Mark III-type containments and all pressurized-water reactors with ice condenser containments must have the capability for controlling combustible gas generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region to prevent loss of containment structural integrity. In addition, 10 CFR 50.44(b)(5)(v)(B) requires that, for all currently licensed boiling-water reactors with Mark III-type containments and all pressurized-water reactors with ice condenser containments, an applicant must demonstrate that the systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen, including local detonations, unless such detonations can be shown unlikely to occur.

As required by 10 CFR 50.44(c)(2), containments for all future water-cooled reactors must have an inerted atmosphere or must limit hydrogen concentrations in the containment during and following an accident that releases an amount of hydrogen equivalent to that generated by a 100-percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) while maintaining containment structural integrity and appropriate mitigating features.

For those future water-cooled reactor containments that do not rely upon an inerted atmosphere to control combustible gases, 10 CFR 50.44(c)(3) requires that they have the capability to control combustible gas generated from a metal-water reaction involving 100 percent of the fuel cladding surrounding the active fuel region so that there is no loss of containment structural integrity. These containments for water-cooled reactors must be able to establish and maintain safe shutdown and containment integrity with systems and components capable of performing their intended functions during and after exposure to the environmental conditions created by the burning of hydrogen.

10 CFR 50.44(c)(5) requires that for containments for future water-cooled reactors, an applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is acceptable to the NRC, and include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address accidents that release hydrogen generated from a 100-percent fuel clad-coolant reaction accompanied by the burning of hydrogen. Applicants must also demonstrate that systems necessary to assure containment integrity are able to perform their functions under these conditions.

## Containment Pressure Integrity for More Likely Severe Accident Challenges

NRC's deterministic containment performance goal (CPG) for the passive advanced light-water reactors (ALWRs) states that the containment should maintain its role as a reliable, leak-tight barrier (Ref. 7). For example, containment stresses should not exceed ASME Service Level C limits for metal containments or Factored Load Category for concrete containments, for approximately 24 hours following the onset of core damage under the more likely severe accident challenges. Following this initial period, the containment should continue to provide a barrier against the uncontrolled release of fission products. This deterministic goal should be supplemented with a probabilistic CPG utilizing a conditional containment failure probability (CCFP) of 0.1 (Ref. 6). The applicant's selection of the more likely severe accident challenges are reviewed by the NRC staff.

## Containment Fragility under Pressure Loading

A fundamental objective of the Commission's Severe Accident Policy (Ref. 13) is to take all reasonable steps to reduce the likelihood of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident, should one occur. As stated in 10 CFR 52.47 (a)(27) (Ref. 2) a design-specific Probabilistic Risk Assessment (PRA) must be conducted as part of the application for design certification. The requirements for certifying a new standard plant design includes an evaluation of a completed PRA and consideration of the severe accident vulnerabilities exposed by the PRA, along with the insights that it may add to providing assurance of no undue risk to public health and safety. Pursuant to 10 CFR 52.79, plant-specific information needs to be incorporated into the PRA for a combined license application. The requirements for holders of a combined license under Subpart C of 10 CFR Part 52 to develop, maintain, and upgrade PRAs are stated in 10 CFR 50.71 (h) (Ref. 1).

The Commission issued its policy statement entitled, "Use of Nuclear Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities, Final Policy Statement" (Ref. 14). The PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. The containment fragility analysis provides information on the likelihood of containment failure to provide an effective barrier against the release of fission products to the environment. However, in order to perform a plant-specific PRA, the development of the containment fragility and PRA evaluation should include appropriate sensitivity studies, uncertainty analyses, and importance measures.

# C. REGULATORY POSITION

## 1. Ultimate Pressure Capacity of Containment

Regulatory Guides 1.57 on metal containment (Ref. 4) and 1.136 on concrete containment (Ref. 5) state that a nonlinear finite element analysis should be performed to determine the ultimate pressure capacity of the containment. The purpose of this evaluation is to obtain a measure of the safety margin for the containment structure above the design basis accident pressure. Therefore, the analysis is intended to address the containment structural capacity and not to evaluate the potential effects of the containment's response on other components that it may support.

For concrete and steel containments, the licensee should provide details of the analysis of the internal pressure capacity of the containment, which includes the failure criteria, and the behavior of the containment shell and other containment components under the anticipated pressure loading. General analysis and design criteria should be provided for the shell regions and containment penetrations with special considerations added for steel elliptical and torispherical heads. Generally, a computerized nonlinear finite element model and analysis is required to predict the global response of the containment. Large hatches and penetrations are usually included, while smaller penetrations and other components may be analyzed separately in a localized finite element model or based on test data. The information provided should be sufficient to support a determination of the safety margin above the design-basis accident pressure.

In addition to the items discussed above, the following considerations should be taken into account in the analysis of the containment's ultimate pressure capacity and in the interpretation and evaluation of results:

a. The use of a three-dimensional (3D) finite element model is acceptable. If sufficient symmetry exists, then a two-dimensional (2D) axisymmetric or partial (e.g., half model or wedge) finite element model can be used. In these cases, justification should be provided to allow the use of the 2D or partial finite element model approach.

b. In performing the evaluation of the ultimate pressure capacity of containment, it is expected that the calculation will develop a realistic measure of the safety margin beyond the design-basis accident pressure using a deterministic analysis method. This can be achieved by considering nonlinear material and geometric behavior. The evaluation should consider all failure modes and their locations that form the containment pressure boundary which include localized areas (e.g., penetrations and appurtenances) and regions away from discontinuities. This will generally result in an ultimate pressure capacity prediction greater than that determined by applying the Service C limits of the ASME Code for steel containments (Ref. 8) and Factored Load Category of the ASME Code for concrete containments (Ref. 9). The Service C limits of the ASME Code for steel containments and Factored Load Category of the ASME Code for concrete containments should be used to demonstrate compliance with Regulatory Position C.2 for hydrogen-generated pressure loads and Regulatory Position C.3 for maximum pressure loads caused by the more likely severe accident challenges as stated later in this Draft Regulatory Guide.

c. The nonlinear stress-strain curve for steel materials including reinforcing steel, prestressing tendon, and steel containment shell should be based on the code-specified minimum yield strength for the specific grade of steel and a stress-strain relationship beyond yield that is representative of the specific grade of steel. The stress-strain relationship employed for calculations should correspond to the design-basis accident temperature. For concrete, the tensile strength of concrete should be neglected, and the nonlinear stress-strain curve in compression should be included in the analysis.

d. Since the purpose of this evaluation is to obtain a measure of the safety margin above the design-basis accident pressure, it is acceptable to perform a static pressure analysis; dynamic effects need not be considered. The deterministic nonlinear analysis should begin with the elastic response for dead load and the design pressure load within the containment generated by the design-basis accident. The material properties should correspond to the design-basis accident temperature. The analysis should then increase the internal pressure load until the ultimate pressure capacity is reached, thus allowing an assessment of the safety margin. When performing the ultimate pressure capacity analysis, the pressure(s) corresponding to the yielding of the liner, reinforcing steel, prestressing tendon (if applicable), and steel shell not backed by concrete for concrete containments or yielding of the steel shell for steel containments should also be identified.

e. The analysis for calculating the ultimate pressure capacity of containment and/or its components may be based on closed-form solutions or empirical relationships with adequate technical justification. Test results may also be used; however, sufficient information should be provided demonstrating the validity of the test for the particular containment design and loading condition.

f. Acceptable strain limits for determining the pressure capacity of containments are as follows:

   (1) For cylindrical reinforced concrete containments, the pressure capacity analysis may be based on attaining a maximum global membrane strain away from discontinuities (i.e., the hoop membrane strain in a cylinder) of 1 percent. The specific location of interest is the steel reinforcement in the hoop direction, closest to the inside surface of the concrete. The inside radius of the concrete wall should be used in calculating the strain in the hoop reinforcing steel.

   (2) For cylindrical prestressed concrete containments, the pressure capacity analysis may be based on attaining a maximum global membrane strain away from discontinuities (i.e., the hoop membrane strain in a cylinder) of 0.8 percent. This strain limit is applicable to all materials which contribute to resisting the internal pressure (i.e., tendons, rebars, and liner (if considered)). When calculating the pressure capacity contribution from the tendons, the above-specified strain limit is applicable to the full range of strain (from 0.0 psi at 0.0-percent strain up to the tendon contribution to pressure capacity at 0.8-percent strain). The pressure capacity is to be based on the contribution from each element considered in the analysis at the stress levels corresponding to the nonlinear stress-strain curves.

   (3) For cylindrical steel containments, the pressure capacity analysis may be based on attaining a maximum global membrane strain away from discontinuities (i.e., the hoop membrane strain in a cylinder) of 1.5 percent. This strain limit is considered to be a threshold for avoiding catastrophic failure of a steel containment structure.

g. The methodologies described above apply to the overall containment structure. A complete evaluation of the internal pressure capacity should also address major containment penetrations, such as the removable drywell head and vent lines for BWR designs, equipment hatches, personnel airlocks, and major piping penetrations. The analysis should also address other potential containment leak pathways through mechanical and electrical penetrations.

h. Under internal pressure, a potential failure mode of ellipsoidal and torispherical steel heads is buckling, resulting from a hoop compression zone in the knuckle region. The analysis should evaluate this failure mode to determine if it is the limiting condition for the pressure capacity of the containment. The analysis should consider nonlinear material behavior, and address the effect of initial geometric imperfections either explicitly (direct modeling) or implicitly (through the use of appropriate imperfection sensitivity knockdown factors). With supportable test data, residual post-buckling strength can be considered in determining the pressure capacity.

i. NUREG/CR-6906 (Ref. 10) includes further guidance on computer modeling of concrete containments for internal pressure capacity calculations. Appendix A to NUREG/CR-6906 provides more detailed guidance on developing the finite element models and performing analyses for pressures beyond design-basis accident loads. Based on this appendix, the results at temperatures of up to 400 °F have only a small effect on the ultimate pressure capacity of both reinforced and prestressed concrete containment structures, since the cracked concrete carries no tension regardless of temperature and will have only a minor effect on typical rebar (or steel containment shell) properties. If temperatures above 400 °F exist for a period of time sufficient to heat the embedded rebar or tendons, the effects of the reduction of the elastic modulus and yield strength of the steel elements should be considered in determining the ultimate pressure capacity.

j. NUREG/CR-6706 (Ref. 15) and NUREG/CR-6920 (Ref. 16) provide examples of the implementation of finite element analysis to predict the pressure capacity of steel and concrete containments. Although the objective of these reports was to determine the pressure capacity or fragility of degraded containments, the modeling techniques applied are also applicable to analysis of undegraded containments.

k. The analysis criteria described above address the structural integrity of the containment. The applicant should also perform an analysis to demonstrate that potential leakage from containment components such as penetrations, bolted connections, seals, hatches, or bellows, is sufficiently small for the calculated ultimate pressure and corresponding temperature determined from the finite element analysis. Otherwise, the ultimate pressure capacity should be based on the limiting criteria for leakage from these components. It should be noted that at elevated temperature levels, seals and gaskets at penetrations and connections may not be sufficiently effective in preventing leakage of internal pressure. The acceptance criteria and technical basis for total leakage from the entire containment, typically given in terms of percent of containment volume flow per day at the given pressure or in terms of leakage area, should be provided by the applicant for staff review on a case-by-case basis.

l. The details of the analysis and the results should include, but not be limited to, the following information:

(1) Original design pressure, as defined in the ASME Code;

(2) Calculated static pressure capacity;

(3) Equivalent static pressure response calculated from dynamic pressure;

(4) Associated failure modes;

(5) The stress-strain relationship of all material elements that contribute to the containment capacity such as the liner steel, bolt material, reinforcing steel and prestressing steel (if applicable), and the behavior of the liner under the postulated loading conditions in relation to that of the reinforcing and/or prestressing steel;

(6) Criteria governing the original design and the criteria used to establish failure;

(7) Analysis details and general results which include (1) modeling details, (2) description of computer code(s), (3) material properties and material modeling, (3) loading and loading sequences, (4) failure modes, and (5) interpretation of results. All assumptions made in the analysis and test data (if relied upon) should be stated clearly and be technically justified; and,

(8) Appropriate engineering drawings adequate to allow verification of modeling and evaluation of analyses employed for the containment structure.

m. The licensing application should describe the evaluation which demonstrates that the ultimate pressure capacity of the containment is in accordance with the guidance in Regulatory Positions C.1.a. through C.1.l as stated in this Draft Regulatory Guide. If the evaluation for containment ultimate pressure capacity is combined with one or more of the other containment performance evaluations, then the description of the evaluation should explain how the single evaluation meets each of these separate containment performance requirements and performance goals. The description of the evaluation for the ultimate pressure capacity of the containment should be presented in FSAR or SSAR Section 3.8.1 for steel containment or 3.8.2 for concrete containment, consistent with the requested information identified in Section C.I.3.8.1 or C.I.3.8.2 of Regulatory Guide 1.206 (Ref. 17).

## 2. Containment Pressure Integrity for Hydrogen-Generated Pressure Loads

Regulatory Guide 1.7 (Ref. 3) provides methods acceptable to the staff for meeting the requirements of 10 CFR 50.44 (Ref. 1) on the structural integrity of the containment when hydrogen is generated due to the reaction between fuel cladding and the water coolant at operational temperatures. Regulatory Position C.5 of Regulatory Guide 1.7 provides the acceptance criteria for steel and concrete containments using the requirements in the ASME Code, considering only the pressure and dead load. Steel containments should meet the Service Level C requirements of ASME Code, Section III, Division 1, Subsection NE-3220 (Ref. 8), and concrete containments should meet the Factored Load Category requirements of ASME Code, Section III, Division 2, Subarticle CC-3720 (Ref. 9).

In accordance with Regulatory Guide 1.7 (Ref. 3), the specific ASME Code requirements set forth for each type of containment should be met for a combination of dead load and hydrogen-generated pressure loads, with a minimum of 45 pounds per square inch gauge. For steel containments, an evaluation of instability is not required. Technical justification should be provided for consideration of any deviation from these requirements.

Regulatory Guides 1.57 on metal containment (Ref. 4) and 1.136 on concrete containment (Ref. 5) also provide guidance for meeting the containment structural integrity requirements of

10 CFR 50.44, which are  consistent with Regulatory Guide 1.7 (Ref. 3).  Regulatory Guides 1.57 and 1.136 present the specific loading combinations and design limits associated with the pressure resulting from an accident which releases hydrogen as a result of a 100-percent reaction between fuel cladding and the water coolant at operational temperatures, and the pressure resulting from uncontrolled burning of hydrogen.  In accordance with the guidance presented in these regulatory guides, the evaluation of the containment needs to be demonstrated for the higher pressure arising from the fuel cladding reaction, hydrogen burning, and post accident inerting (if applicable) and 45 psig.

In addition to the technical points discussed above, the analysis should take into account the following aspects, which apply to containments of future water-cooled reactors, in demonstrating containment pressure integrity when subjected to hydrogen-generated pressure loads, and in interpreting and evaluating results:

a. The development of a mathematical finite element model of the containment using the approach described under Regulatory Position C.1, subject to the limitations discussed below, is acceptable.

    (1) ASME Code-specified material properties should be used.  These should correspond to the temperature(s) associated with the hydrogen-generated pressure loads.

    (2) For steel elements, linear elastic material properties may be used.  For concrete, the nonlinear stress-strain relationship or an equivalent linear elastic curve may be used.

    (3) If the pressure transient loading associated with the hydrogen gas generation or the burning of hydrogen is significant over a short duration with respect to the predominant periods (i.e., period = 1/frequency) of the containment-related structures, then the dynamic effects of the pressure transient should be included in calculating the response of the containment.

b. The accident sequence used to determine the pressure load should address the hydrogen mass and energy releases generated from a 100-percent fuel clad-coolant reaction accompanied by the burning of hydrogen.

c. Assumptions used should be clearly justified on pressure peaks resulting by hydrogen generation in the reactor vessel and hydrogen distribution in the containment compartments when released to the containment, including any heat absorptions by water or other heat sinks present within the containment boundary.

d. Regulatory Position 5 of Regulatory Guide 1.7 (Ref. 3) provides the acceptance criteria for the resulting stresses.  As noted earlier in the above discussion and in Regulatory Guide 1.7, instability (buckling) calculation is not required for steel containments.

e. The licensing application should describe the evaluation which demonstrates that the containment pressure integrity for hydrogen-generated pressure loads is maintained in accordance with the guidance in Regulatory Positions C.2.a through C.2.d.  If the evaluation to demonstrate the containment pressure integrity for hydrogen-generated pressure loads is combined with one or more of the other containment performance evaluations, then the description of the evaluation should clearly explain how the single evaluation meets each of these separate containment performance requirements and performance goals. The description of the evaluation for the containment pressure integrity due to hydrogen-generated pressure loads should be presented in

FSAR or SSAR Section 3.8.1 for steel containment or 3.8.2 for concrete containment, consistent with the requested information identified in Section C.I.3.8.1 or C.I.3.8.2 of Regulatory Guide 1.206 (Ref. 17) for steel and concrete containment, respectively.

## 3. Containment Pressure Integrity for More Likely Severe Accident Challenges

Under the more likely severe accident challenges, the containment should maintain its role as a reliable, leaktight barrier for approximately 24 hours following the onset of core damage. Following this initial 24-hour period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

### 3.1 24-Hour Period following the Onset of Core Damage

a. The pressure and temperature transient loads for the more likely severe accident challenges should be analyzed.

b. The development of the global and localized finite element models of the containment using the approach described under Regulatory Position C.1, subject to the limitations discussed below, is acceptable.

   (1) For steel containment elements, linear elastic material properties may be used. For concrete, the nonlinear stress-strain relationship or an equivalent linear elastic curve may be used. All of the material properties should be based on the accident temperatures expected for each severe accident considered.

   (2) For any severe accident loads that occur over a short time duration with respect to the predominant periods (i.e., period = 1/frequency) of the containment structure, the dynamic effects of the pressure transient should be considered in calculating the response of the containment.

c. The use of the ASME Code Service Level C limits for metal containments or Factored Load Category for concrete containments is acceptable to demonstrate the deterministic performance goal for the first 24 hours. This includes the demonstration of the stability/buckling of the containment in accordance with the ASME Code.

### 3.2 Period Following Initial 24 Hours after the Onset of Core Damage

If it can be demonstrated that the pressure and temperature transient for the more likely severe accident events, in the period following the initial 24 hours, are less than the maximum values in the initial 24-hour period, then the initial 24-hour period analysis governs; another set of analyses for the period subsequent to the initial 24 hours need not be performed. However, if the pressure and/or temperature transient in the period following the initial 24 hours are greater than the maximum values in the initial 24-hour period, then an additional evaluation should be conducted that incorporates the following aspects:

a. The development of a finite element model of the containment and the consideration of nonlinear behavior of the containment, using the approach described under Regulatory Position C.1, subject to the limitations discussed below, is acceptable.

(1). The nonlinear stress-strain curve for steel and concrete materials should correspond to the temperature associated with the more likely severe accident events. The effect of elevated temperature on the elastic modulus for all materials should be considered.

(2). For any severe accident loads that occur over a short time duration with respect to the predominant periods (i.e., period = 1/frequency) of the containment structure, the dynamic effects of the pressure transient should be considered in calculating the response of the containment.

b. Acceptance criteria to show that the containment provides a barrier against the uncontrolled release of fission products should be provided for review. The applicant may use the same acceptance criteria as was used in the initial 24-hour period in Regulatory Position C.3.1. Alternatively, the applicant may use the same acceptance criteria presented in Regulatory Position C.1.

c. The analysis criteria described above address the containment structural integrity. The applicant should also perform an analysis to demonstrate that potential leakage from containment components, such as penetrations, bolted connections, seals, hatches, or bellows, are sufficiently small for the calculated pressure and corresponding temperature. Regulatory Position C.1.k provides the guidance for evaluating potential leakage.

### 3.3 <u>Description of Entire Evaluation</u>

The licensing application should describe the evaluation that demonstrates that the containment integrity for the more likely severe accidents is in accordance with the guidance in Regulatory Positions C.3.1 and C.3.2. In addition, if the evaluation to demonstrate the containment integrity for the more likely severe accidents is combined with one or more of the other containment performance evaluations, then the description of the evaluation should clearly explain how the single evaluation meets each of the separate containment performance requirements and performance goals. The description of the evaluation for containment pressure integrity under the more likely severe accident challenges should be presented in Section 19 of the FSAR or SSAR, consistent with the requested information identified in Section C.I.19 of Regulatory Guide 1.206 (Ref. 17).

## 4. Containment Fragility under Pressure Loads

All design certification applications that fall under 10 CFR 52.47 (Ref. 2) must satisfy the requirements of 10 CFR 52.47(a)(27) that a design-specific PRA be conducted as part of the application for design certification. Thus, the estimate of containment fragility as a function of pressure and associated temperature loads needs to be developed for use in the design-specific PRA. Plant-specific information needs to be incorporated into the PRA pursuant to 10 CFR 52.79 for combined license application. In addition, 10 CFR 50.71 (h) provides requirements for holders of a combined license under Subpart C of 10 CFR Part 52 to develop, maintain, and upgrade PRAs.

The applicant should include the following considerations in the analysis of containment pressure fragilities and in the interpretation and evaluation of results:

a. Fragility assessments should be of a quality and depth sufficient to provide adequate insights on the design capability of the containment to withstand postulated severe accident sequences. The assessments should demonstrate that at the highest performance level of pressure and associated temperature load the containment can retain its integrity and there is a reasonable margin in the design.

b. The ultimate pressure capacity of the various containment components (e.g., shell, head, hatches, penetrations, liners, liner anchorages) should be determined.

c. The pressure capacity used in developing the containment fragility may be assumed to have a lognormal distribution.

d. The use of a lognormal distribution requires a determination of the median values of failure pressure for various containment failure modes and the consideration of the variability (aleatory uncertainty) of the associated parameters. To this end, either a simplified fragility method or a sampling method, such as Monte Carlo, may be used to establish the containment fragility.

e. To apply the simplified fragility method, the median failure pressure for various containment failure modes should be calculated first and then the variability (in both aleatory and epistemic terms) about the median failure pressure should be estimated.

f. The sampling method should include the following:

    (1) an identification of all random variables associated with the estimate of the containment failure pressure;

    (2) the selection of the probability distribution for each random variable; and

    (3) sampling analysis to determine the containment pressure fragility.

g. The fragility analyses should be based on detailed 3D finite element modeling, appropriate material constitutive relations, and an assessment of uncertainties within a probabilistic framework. The development of the global and localized finite element models of the containment and the consideration of nonlinear behavior of the containment, using the approach described under Regulatory Position C.1, subject to the limitations discussed in this section, are acceptable. The uncertainties in the analysis results should be associated with the finite element modeling and analysis approach, the material properties of the structure at the time of the accident, failure criteria or limit states used in establishing the pressure capacity, and the loading conditions that lead to pressurization of the containment.

h. The fragility analyses of containment components may use closed-form solutions or empirical relationships with adequate technical justification. Test results may also be used; however, sufficient information should be provided to demonstrate the validity of the test for the particular containment design and loading condition.

i. The analysis criteria described above address the containment structural integrity. The applicant should also perform analyses to demonstrate that potential leakage from containment components such as penetrations, bolted connections, seals, hatches, or bellows, is sufficiently small for the calculated pressure fragility and corresponding temperature. Regulatory Position C.1.k presents the guidance for evaluating potential leakage.

j. The potential failure mode of buckling of containment shell/components (e.g., ellipsoidal and torispherical heads and hatches) resulting from any compression zones should be evaluated to determine whether it is the limiting condition for the pressure capacity of the containment. The analysis should consider nonlinear material and geometric behavior and address the effect of initial geometric imperfections either explicitly (direct modeling) or implicitly (through the use of appropriate imperfection sensitivity knockdown factors). Test data may be used provided the validity of the test for the particular containment design and loading condition can be demonstrated.

k. The analyses for establishing the pressure fragility of the containment may be best-estimate calculations based on median or expected material properties and failure criteria at the applicable temperature(s).

l. For structural analyses, the use of the 95-percent confidence value of the important parameters to assess the effect of uncertainty in the analyses is acceptable. The median and 95-percent confidence values are to be developed for the elastic and plastic material properties and failure criteria, all as a function of temperature.

m. The material property and failure criteria values should cover the entire range of temperatures (e.g., from ambient to the most severe accident temperature condition).

n. The selection of the quantitative failure/rupture values (e.g., rupture strain) for the various materials in the analyses should consider the multiaxis stress state of the material. These values should also correspond to the failure/rupture data based on tests at the expected accident temperature value(s).

o. Failure criteria should be defined to establish limit states on the structural response when the internal pressure is no longer contained by the structure. Since uncertainty exists in defining these failure criteria, median and 95-percent confidence values of the internal pressure should be defined to evaluate the effect of the uncertainty on the analysis results.

p. Accident conditions leading to overpressurization should also include properties and effects at elevated temperatures. Because of temperature-induced stresses and material property degradation at elevated temperatures, the fragility for overpressurization is also a function of temperature. Thus, the fragility analyses should be conducted for three different sets of temperature ranges—steady-state normal operating temperatures (referred to as ambient conditions), steady-state conditions representing long-term accident conditions, and transient thermal conditions, such as a temperature spike representative of direct containment heating conditions.

q. Model uncertainty exists in the analyses for determining the failure pressures for any given set of material properties, geometry, or other dependent parameters. This uncertainty arises from the mesh discretization used in finite element models, the type of element formulations used, the robustness of the constitutive models, the equilibrium iteration algorithms and convergence tolerances, geometric imperfections, allowable fabrication and construction tolerances, rebar placement locations, etc. The fragility calculation should quantify this modeling uncertainty.

r. The licensing application should describe the evaluation that demonstrates that the containment fragility under pressure loads is in accordance with the guidance in Regulatory Positions C.4.a through C.4.q. The evaluation for containment fragility under pressure loads should not be combined with the evaluation for any of the other containment performance evaluations discussed in this draft regulatory guide. The description of the evaluation for containment fragility under pressure loads should be presented in Section 19 of the FSAR or SSAR, consistent with the requested information identified in Section C.I.19 of Regulatory Guide 1.206 (Ref. 17).

# D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC's plans for using this draft regulatory guide. The NRC does not intend or approve any imposition or backfit in connection with its issuance.

The NRC has issued this draft guide to encourage public participation in its development. The NRC will consider all public comments received in development of the final guidance document. In some cases, applicants or licensees may propose an alternative or use a previously established acceptable alternative method for complying with specified portions of the NRC's regulations. Otherwise, the methods described in this guide will be used in evaluating compliance with the applicable regulations for license applications, license amendment applications, and amendment requests.

# REGULATORY ANALYSIS

## 1. Statement of the Problem

Revision 1 to Regulatory Guide 1.57 (Ref. 4) endorsed ASME Code, Section III, Division 1, with some exceptions, for design and loading combinations for metal primary reactor containment system components for nuclear power plants. Regulatory Guide 1.57 gives brief guidance for ultimate capacity of steel containments.

Revision 3 to Regulatory Guide 1.136 (Ref. 5) endorsed ASME Code, Section III, Division 2, with some exceptions for design, loading combinations, materials, construction, and testing of concrete containments for nuclear power plants. Regulatory Guide 1.136 also gives brief guidance for ultimate capacity of concrete containments.

The staff proposes to include the use of deterministic performance goals for pressure loads in the containments. Therefore, the publication of this regulatory guidance is necessary to provide detailed and up-to-date guidance on deterministic performance goals for pressure loads in steel and concrete containments in evolutionary and ALWR designs.

## 2. Objective

The objective of this regulatory action is to provide acceptable methods for demonstrating containment performance in accordance with regulatory requirements and Commission performance goals for pressure loadings of steel and concrete containments, especially for evolutionary and ALWR designs.

## 3. Alternative Approaches

The NRC staff considered the following alternative approaches:

- Take no action.
- Publish a new regulatory guide.

### 3.1 Alternative 1: Take No Action

Under this alternative, the NRC would not publish guidance on meeting the deterministic containment performance goals for pressure loads. This alternative is considered the baseline or "no action" alternative. However, if the NRC does not publish guidance on accepted approaches to meet deterministic containment performance goals for pressure loads, the staff position may not be clear.

### 3.2 Alternative 2: Publish a New Regulatory Guide

Under this alternative, the NRC would publish a new regulatory guide, taking into consideration current regulations, standards, and related staff regulatory positions.

One benefit of this action is that it would enhance reactor safety by incorporating the latest technical information for guidance on the containment performance for pressure loads and would address the potential ambiguity associated with other regulatory guidance documents.

The costs to the NRC would be the one-time cost of issuing a new regulatory guide (which is expected to be relatively small), and applicants would incur little or no cost.

## 4. Conclusion

Based on this regulatory analysis, the staff recommends that the NRC publish this new regulatory guide. The staff concludes that the proposed action will enhance reactor safety by providing updated guidance on the containment performance assessment for pressure loads.

# GLOSSARY

**ultimate pressure capacity:** A deterministically-based estimate of the maximum internal pressure at which the containment structure is still able to maintain its structural and functional integrity.

**pressure fragility:** The containment failure probability as a function of pressure, where failure is generally characterized as either leakage or rupture. The containment pressure fragility depends on the assumptions made regarding the definition of failure used in the analysis, the failure modes and corresponding criteria considered, the methods used to calculate the containment response, and the methods used to incorporate model and aleatory (random) uncertainties.

**structural integrity:** The ability of the containment to withstand applied loading, without rupture, collapse, or uncontrolled deformation. Demonstration of structural integrity alone is not sufficient to ensure the leaktight integrity of the containment.

**functional integrity:** The ability of the containment to maintain its function as a leaktight pressure boundary without significant leakage. This also implies that sufficient containment strength exists which would preclude deformations that would lead to leakage (e.g., tears in the liner, deformation of a removable steel containment head allowing leakage through the bolted connection, deformation in hatches that could cause leakage through seals and gaskets especially at elevated temperatures). Global and local containment response evaluations are required to ensure that leakage of containment does not occur or is shown to be insignificant depending on the intended loading evaluation.

**global containment response:** Overall response based on a global model of the entire containment using finite element models or closed-form solutions, which include consideration of large displacement and strain effects, and nonlinear material properties. For certain containment components (e.g., hatches, doors, bellows), empirical equations appropriately validated against test results may be used to characterize their response. Some global model idealizations may typically ignore the effects of small penetrations, basemat irregularities, and the gravity effect of containment internal structures and other smaller loads on the overall containment response.

- <u>Steel Containments</u>: Global containment response of steel containments involves accurate prediction of the onset of yield, plastic deformation, strain hardening, and buckling, considering the multi-axial nature of the stress state, and the prediction of deformation versus pressure in the final stage of loading.

- <u>Concrete Containments</u>: Global containment response of reinforced and prestressed concrete containments with steel liners involves accurate prediction of cracking of concrete, yielding of concrete and strain softening in compression, yielding of rebars, behavior of tendons, liner tearing and anchor evaluations, buckling of ellipsoidal or torispherical heads, airlock doors, and hatch covers and the prediction of deformation versus pressure in the final stage of loading.

**local containment response:** Localized response based on detailed analysis of stress and strain concentration areas at the geometric and material discontinuities using fine mesh 2D or 3D finite element analysis, empirical formulas validated by test results, and buckling models to calculate critical buckling pressure with test validation to extrapolate theoretical values to actual configurations.

- Steel Containments:  Local containment response of steel containments involves prediction of strain concentrations near geometric and stiffness discontinuities and the local deformation states of containment components such as penetrations and bellows.

- Concrete Containments:  Local containment response of reinforced and prestressed concrete containments with steel liners involves those associated with strain concentrations near geometric and stiffness discontinuities, other influencing factors owing to liner-concrete-rebar-tendon interactions, and difficulties in modeling the shearing response of concrete.

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