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CONTAINMENT STRUCTURAL INTEGRITY EVALUATION FOR INTERNAL PRESSURE LOADINGS ABOVE DESIGN-BASIS PRESSURE

A. INTRODUCTION

This guide describes methods that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for (1) predicting the internal pressure capacity for containment structures above the design-basis accident pressure, (2) demonstrating containment structural integrity related to combustible gas control, and (3) demonstrating containment structural integrity through an analysis that specifically addresses the Commission's performance goals related to the prevention and mitigation of severe accidents. It provides guidance on methods for satisfying requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 1). Requirements in 10 CFR 52.47, "Contents of Applications; Technical Information," and in 10 CFR 52.79, "Contents of Applications; Technical Information in Final Safety Analysis Report [FSAR]," relate to the structural integrity of containments under internal pressurization that pertain to the containment structural capacity above design-basis pressures, to combustible gas control, and to the prevention and mitigation of severe accidents. This guidance is intended to be consistent with the Commission's goals and other related guidance, as discussed in the remainder of this section. This regulatory guide does not address requirements and guidance for the structural evaluation of containments for design-basis pressure.

Prediction of Containment Internal Pressure Capacity Above Design-Basis Pressure: 10 CFR 50, Appendix A, General Design Criteria (GDC) 50, "Containment Design Basis," requires that the reactor containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions caused

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The NRC issues regulatory guides to describe and make available to the public methods that the NRC staff considers acceptable for use in implementing specific parts of the agency's regulations, techniques that the staff uses in evaluating specific problems or postulated accidents, and data that the staff needs in reviewing applications for permits and licenses. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

This guide was issued after consideration of comments received from the public.

Regulatory guides are issued in 10 broad divisions—1, Power Reactors; 2, Research and Test Reactors; 3, Fuels and Materials Facilities; 4, Environmental and Siting; 5, Materials and Plant Protection; 6, Products; 7, Transportation; 8, Occupational Health; 9, Antitrust and Financial Review; and 10, General.

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by a loss of coolant accident (LOCA). According to 10 CFR 52.47(a)(9), for design certification (DC) applications, and 10 CFR 52.79(a)(41), for combined license (COL) applications, applications for light water-cooled nuclear power plants shall include an evaluation of the facility against NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the Standard Review Plan (SRP)) (Ref.2). In sections 3.8.1 and 3.8.2 of the SRP, the need for the determination of the internal pressure capacity as a measure of the safety margin above the design-basis accident pressure is described. This regulatory guide addresses methods acceptable to the staff for estimating the margin by predicting the internal pressure capacity for containment structures above the internal pressure for the design-basis LOCA. The internal pressure capacity in this estimation is an internal pressure capacity at which the structural integrity is retained and a failure leading to a significant release of fission products does not occur.

Combustible Gas Control Inside Containment: According to 10 CFR 52.47(a)(12) for DC applications, and 10 CFR 52.79(a)(8) for COL applications, applications for light-water-cooled nuclear power plants shall include an analysis and description of the equipment and systems for combustible gas control, as required by 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors" (Ref. 3), which provides the requirements for combustible gas control within the containment for future water-cooled reactor applications. Specifically, 10 CFR 50.44(c)(5) provides requirements for demonstrating containment structural integrity. This regulatory guide provides an acceptable method for meeting the requirements of 10 CFR 50.44 and demonstrating containment structural integrity.

Commission's Severe Accident Performance Goal: According to 10 CFR 52.47(a)(23) for DC applications, and 10 CFR 52.79(a)(38) for COL applications, applications for light-water reactor (LWR) designs shall include a description and analysis of design features for the prevention and mitigation of severe accidents. Section C.I.19.8 of Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," issued June 2007 (Ref. 5), provides guidance on implementing these requirements. According to Section C.I.19.8, this analysis and description should specifically address the issues and performance goals identified in SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990 (Ref. 6), and SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993 (Ref. 7), which the Commission approved in staff requirement memoranda (SRMs) dated June 26, 1990, and July 21, 1993, respectively. This regulatory guide provides acceptable methods for an analysis that specifically addresses the issues and performance goals identified in SECY-90-016 and SECY-93-087 and related SRMs for containment structures in nuclear power plants under severe accident conditions.

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 covered by 10 CFR Part 50 and 10 CFR Part 52 that the Office of Management and Budget (OMB) approved under OMB control numbers 3150-0011 and 3150-0151. 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

Scope

This regulatory guide addresses deterministic methods for structural integrity evaluations of containment structures and pressure-retaining structural barriers constructed of steel, reinforced concrete, and prestressed concrete.

The scope of this regulatory guide for metal containments includes those components designed and constructed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III, "Rules for Construction of Nuclear Facility Components" (Ref. 8), Division 1, Subsection NE. These components include metal containments and appurtenances, as well as metal portions of concrete containments that are not backed by concrete.

The scope of this regulatory guide for reinforced and prestressed concrete containments includes those components designed and constructed in accordance with Section III, Division 2, Subsection CC, of the ASME Code. These components include 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.

The methods specified in this regulatory guide are applicable to new LWR designs. This regulatory guide is only directly applicable to metal and concrete containments for LWRs. However the principles contained herein may be applied to metal and concrete containments for non-LWRs, subject to review by the NRC.

Prediction of Containment Internal Pressure Capacity above Design Pressure

NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories—An Overview," issued July 2006 (Ref. 9.), reports the results from a series of tests conducted on reinforced and prestressed concrete and free-standing steel containment vessel models. The containment model tests showed that global free-field strains on the order of 2.0–3.0 percent for steel containments, and 1.5–2.0 percent for reinforced concrete containments can be achieved before failure occurs. For prestressed concrete containments with tendon hoop strains of about 0.4 percent before pressurization, analysis of the results presented in NUREG/CR-6810 (Ref. 10) shows that global free-field hoop strains in the containment wall of 0.5 and 1.0 percent can be achieved before the onset of unrestrained wall deformations or rupture, respectively. Analysis of the results in NUREG/CR-6810 also shows that free-field average strains of 0.9 and 1.4 percent (including strains before pressurization and strains from the internal pressurization) can be achieved for the hoop tendons before the onset of unrestrained wall deformations or rupture, respectively. As noted in NUREG/CR-6906, the geometric complexity of actual containments increases the likelihood that local strain risers are present and may be more severe than in any of the models tested. Regulatory Position 1 of this regulatory guide identifies global strain limits for a simplified prediction of the containment's internal pressure capacity. As compared to the strains reached in containment model tests, these limits incorporate reductions that account for strain risers more severe than those on the containment models tested (Ref. 9) and also account for results of other studies (NUREG/CR-6810, "Overpressurization Test of 1:4-Scale Prestressed Concrete Containment Vessel Model," issued March 2003 (Ref. 10); and NUREG/CR-6809, "Posttest Analysis of the NUPEC/NRC 1:4-Scale Prestressed Concrete Containment Vessel Model," issued March 2003 (Ref. 11)). Global strain limits for the containment structures are not applicable to the assessment of large bolted closures (e.g., boiling-water reactor (BWR) steel closure heads, equipment hatches, personnel airlocks). The separate predictions of internal pressure capacity for these containment components are subject to staff review on a case-by-case basis.

Regulatory Position 1 of this regulatory guide identifies methods acceptable to the staff for demonstrating that a concrete or steel containment possesses a significant internal pressure capacity above the containment design pressure.

Combustible Gas Control Inside Containment

As required by 10 CFR 50.44(c)(2), containments for new water-cooled reactors must have an inerted atmosphere, or the 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) must be limited to less than 10 percent (by volume), while maintaining containment structural integrity and appropriate mitigating features.

For new 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 hydrogen generated from a fuel clad-coolant 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 hydrogen burning.

As required by 10 CFR 50.44(c)(5), 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 the analysis must address an accident that releases hydrogen generated from a 100-percent fuel clad-coolant reaction, accompanied by hydrogen burning. Applicants must also demonstrate that systems necessary to ensure containment integrity are able to perform their functions under these conditions. Regulatory Guide 1.7 specifically addresses requirements in 10 CFR 50.44(c)(5) to demonstrate containment structural integrity.

The March 2007 revisions to SRP Sections 3.8.1 and 3.8.2 specifically identify accident load combinations that include pressures related to the generation of hydrogen inside containment. The load combinations specified are consistent with the regulatory position in Regulatory Guide 1.7.

Regulatory Position 2 of this regulatory guide identifies methods acceptable to the staff to demonstrate that the containment structural integrity requirements of 10 CFR 50.44(c)(5) are satisfied.

Commission's Severe Accident Performance Goal

The NRC's deterministic containment performance goal for advanced LWRs states, in SECY-93-087 (Ref. 7) that the containment should maintain its role as a reliable, leak-tight barrier:

> The containment should maintain its role as a reliable, leak-tight barrier (for example, by ensuring that containment stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

Regulatory Position 3 of this regulatory guide identifies methods acceptable to the staff for demonstrating that the Commission's severe accident performance goal is met.

C. REGULATORY POSITION

1. Prediction of Containment Internal Pressure Capacity above Design Pressure

The purpose of this evaluation is to assess the pressure capacity of the containment at which the structural integrity is retained, and a failure leading to a significant release of fission products does not occur. This analysis is intended to address the containment structural capacity and not to evaluate the potential effects of containment structural response on other connected components, such as attached piping and other equipment. An applicant for a DC or COL should submit the details of this containment capacity analysis with its application.

To estimate the pressure capacity of the containment structure, typically, a nonlinear finite element analysis will be needed to examine the overall response. Large penetrations are usually included in the finite element model; smaller penetrations and penetration closure components are typically analyzed using a separate finite element model, test data, or both.

For cylindrical containment structures and axisymmetric components of the containment, it may be feasible to use closed-form solutions or semiempirical methods to estimate the pressure capacity; in such cases, the applicant should provide an adequate technical justification for all simplifications. Test results may also be used; however, the applicant should provide sufficient information to demonstrate the applicability of the test results to the particular containment design and loading condition.

The information submitted should be sufficient for the staff to make a determination of the safety margin above the design-basis accident pressure.

In determining the containment internal pressure capacity, and also in the interpretation and evaluation of results, the applicant should consider the following staff expectations:

a. The use of a three-dimensional finite element model is acceptable. Axisymmetric or partial (e.g., half model or wedge) finite element models can be used, if a sufficient technical basis is provided.

b. For the purpose of estimating the safety margin above the design-basis accident pressure, a static analysis is acceptable. However, if dynamic response effects are important, the static pressure capacity may need to be reduced to account for such effects. One acceptable approach to determine the reduced pressure capacity is a nonlinear dynamic analysis. The NRC will review, on a case-by-case basis, alternative approaches provided by the applicant; for example, the use of an appropriate dynamic amplification factor.

c. The initial condition for the nonlinear analysis of the containment structure should be the linear elastic response caused by dead load and design pressure, at the design temperature. The internal pressure is incrementally increased until a specified failure criterion is reached (e.g., deflection limit; strain limit; solution divergence). When performing this analysis, the applicant should record the pressure(s) corresponding to initial yielding of the liner, reinforcing steel, prestressing tendon (if applicable), and steel components not backed by concrete (e.g., closure head, hatch) for concrete containments; it should record the pressure(s) corresponding to initial yielding of the steel shell and yielding of other steel components (e.g., closure head, hatch) for steel containments.

d. The nonlinear stress-strain curve for steel materials (steel liner, reinforcing steel, prestressing tendons, steel components, steel shell) should be based on the ASME 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 curve used in the analysis should correspond to the design-basis accident temperature.

e. For concrete containments, the tensile strength of concrete should be neglected, and the analysis should include the nonlinear stress-strain curve in compression, corresponding to the design-basis accident temperature. In this regard, Regulatory Position 1i of this regulatory guide contains additional guidance on concrete material properties for the finite element model.

f. The following are acceptable simplified methods for determining the pressure capacity of cylindrical containments:

  (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 may be estimated based on satisfying both of the following strain limits: (1) a total tensile average strain in tendons away from discontinuities (e.g., hoop tendons in a cylinder) of 0.8 percent, which includes the strains in the tendons before pressurization (typically about 0.4 percent) and the additional straining from pressurization; and (2) a global free-field strain for the other materials that contribute to resist the internal pressure (i.e., liner, if considered, and rebars) of 0.4 percent. The pressure capacity is to be based on the contribution from each element considered in the analysis, using the stress-strain curve for each material and the strain level in each material, as determined based on overall strain compatibility between all of the credited structural elements (liner, if considered, tendons and rebars).

  (3) The analysis should consider additional failure modes, such as concrete shear and concrete crushing which may occur near discontinuities, to allow the determination of the controlling containment failure mode. This is necessary to account for failure modes that may occur at pressures lower than those corresponding to membrane yielding.

  (4) 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.

g. The analysis methods 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.

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 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). If supported by test data, residual postbuckling strength can be considered in determining the pressure capacity.

i. Appendix A to NUREG/CR-6906 (Ref. 9) provides more detailed guidance on developing the finite element models and performing analyses for pressures beyond the design-basis accident pressure.

j. The evaluation should also consider the potential for containment leakage at pressure levels below the calculated structural capacity. The applicant should perform analyses to demonstrate that leakage from containment components, such as penetrations, bolted connections, seals, hatches, or bellows, is sufficiently small for the calculated pressure and temperature capacity conditions. Otherwise, the pressure capacity should be based on a defined total leakage limit 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 applicant should provide the acceptance criteria and technical basis for total leakage from the entire containment, typically given in terms of the percentage of containment volume flow per day at the given pressure or in terms of the leakage area. The staff will review the applicant's acceptance criteria and technical basis for total leakage on a case-by-case basis.

k. The applicant should submit details of the analysis and the results in report form, with the following information:

   (1) design internal pressure, as defined in Subarticle NE-3100, "General Design" (Ref. 8), and in Subarticle CC-3200, "Load Criteria" (Ref. 8);

   (2) calculated static pressure capacity;

   (3) dynamic pressure capacity, if applicable (static pressure capacity reduced to account for dynamic amplification effects);

   (4) associated failure modes;

   (5) for concrete containments, the stress-strain relation of the liner steel and reinforcing or prestressing steel and the behavior of the liner under the postulated loading conditions in relation to that of the reinforcing or prestressing steel;

   (6) criteria governing the original design and 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, (4) loading and loading sequences, (5) failure modes, and (6) interpretation of results, with all assumptions made in the analysis and test data (if relied upon) clearly stated and technically justified; and,

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

The description of the evaluation for containment pressure capacity, in accordance with this regulatory position, should be presented in the design certification document (DCD) or FSAR Section 3.8.1, for a concrete containment, or Section 3.8.2, for a steel containment, consistent with the information identified in Sections C.I.3.8.1 or C.I.3.8.2 of Regulatory Guide 1.206 (Ref. 5), respectively. If the evaluation of the containment pressure capacity relies on a single analysis intended to address multiple regulatory requirements and performance goals, then the applicant should clearly explain how the single analysis addresses each specific requirement or performance goal.

2. Combustible Gas Control Inside Containment

This regulatory position provides methods acceptable to the staff for demonstrating the structural integrity of the containment in accordance with the requirements in 10 CFR 50.44 (Ref. 3), related to pressure loadings associated with hydrogen generation caused by the reaction between the fuel cladding and the water coolant.

Regulatory Position 5 of Regulatory Guide 1.7 (Ref. 4) already provides acceptance criteria acceptable to the staff to meet the structural requirements of steel and concrete containments in accordance with 10 CFR 50.44 (Ref. 3). For the required pressure load 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.8).

For metal containments, Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components" (Ref. 12), and, for concrete containments, Regulatory Guide 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments" (Ref. 13), provide guidance for meeting the containment structural integrity requirements of 10 CFR 50.44. This guidance is consistent with the guidance presented in Regulatory Guide 1.7. Regulatory Guides 1.57 and 1.136 also present the specific loading combinations and design limits associated with hydrogen generation caused by the reaction between the fuel cladding and the water coolant.

In accordance with the regulatory guidance described above, the containment should be evaluated for (1) the pressure arising from the fuel cladding-water reaction, hydrogen burning, and post-accident inerting (if applicable), or (2) 45 pounds per square inch gauge, whichever is higher. Regulatory Guides 1.57 and 1.136 define the pressures associated with the fuel cladding-water reaction, hydrogen burning, and post-accident inerting, as well as the load combinations to be considered for the structural integrity assessment.

In addition to the guidance in Regulatory Guides 1.7, 1.57, and 1.136, the analysis to demonstrate the structural integrity of the containment should take into account the following items:

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

    (1) ASME Code-specified material properties should be used. These should correspond to the metal temperature(s) resulting from the hydrogen-generation event.

    (2) For steel containment elements, linear elastic material properties may be used. For concrete containments, the tensile strength of concrete should be neglected while for the concrete in compression the nonlinear stress-strain relationship should be used or justification should be provided for an equivalent linear stress-strain model. The staff will review the justification for an equivalent linear stress-strain model on a case-by-case basis.

(3) The potential structural dynamic amplification effects caused by the pressure transient loading associated with hydrogen gas generation or the burning of hydrogen, if significant, 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, and post-accident inerting (if applicable). For inerted containments, burning does not need to be considered.

c. Regulatory Position 5 of Regulatory Guide 1.7 (Ref. 4) provides the acceptance criteria for the resulting stresses. As noted in Regulatory Guide 1.7, an instability (buckling) calculation is not required for steel containments. For concrete containments, the acceptance criteria are limited to demonstrating that the liner strains satisfy the Factored Load Category requirements presented in ASME Code, Section III, Division 2, Subarticle CC-3720 (Ref. 8).

d. The description of the evaluation for demonstrating the structural integrity of the containment in accordance with this regulatory position should be presented in the DCD or FSAR Section 3.8.1 for a concrete containment or Section 3.8.2 for a steel containment. If the evaluation to demonstrate the containment pressure integrity for the hydrogen-generated pressure loads relies on a single analysis intended to address multiple regulatory requirements and performance goals, then the applicant should clearly explain how the single analysis addresses each specific requirement or performance goal.

3. Commission’s Severe Accident Performance Goal

This regulatory position describes methods acceptable for an analysis that specifically addresses the Commission’s deterministic containment performance goals in accordance with SECY-90-16 (Ref. 6) and SECY-93-087 (Ref. 7), and the corresponding SRMs, dated June 26, 1990, and July 21, 1993, respectively. As specified in SECY-93-087, the containment should maintain its role as a reliable, leak-tight barrier for approximately 24 hours following the onset of core damage, under the more likely severe accident challenges. Following this initial 24-hour period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

3.1 <u>24-Hour Period following the Onset of Core Damage</u>

a. The applicant provides the technical basis for identifying the more likely severe accident challenges. The staff will review the technical basis for identifying the more likely severe accident challenges on a case-by-case basis. An example of an acceptable way to identify the more likely severe accident challenges is to consider the sequences or plant damage states that, when ordered by percentage contribution, represent 90 percent or more of the core damage frequency.

b. From the set of pressure and temperature transient loadings determined from item a above, identify which pair(s) of pressure and corresponding temperature loadings envelop the entire set of pressure and temperature loadings. The reduced pair(s) of pressure and temperature loadings should be used to analyze the containment. For concrete containments, pressure loading generally has a greater influence than temperature loading on the structural integrity challenge. Therefore, it is generally acceptable to analyze concrete containments for the sequence or damage state with the highest pressure load and its coexisting temperature loading.

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

(1) For steel containment elements, linear elastic material properties may be used. For concrete containments, the tensile strength of concrete should be neglected while for the concrete in compression the nonlinear stress-strain relationship should be used or justification should be provided for an equivalent linear stress-strain model. The staff will review the justification for an equivalent linear stress-strain model on a case-by-case basis. All of the material properties should be based on the accident temperatures expected for each severe accident considered. In this regard, see Appendix A to NUREG/CR-6906 for information on material properties for concrete containments.

(2) The potential structural dynamic amplification effects caused by the pressure transients for the severe accident events, if significant, should be included in calculating the response of the containment.

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

3.2 Period following Initial 24 Hours after the Onset of Core Damage

a. Acceptable ways to meet the performance goal that "…the containment should continue to provide a barrier against the uncontrolled release of fission products" (for the more likely severe accident challenges, after the initial 24-hour period), include the following:

(1) The maximum pressure and temperature following the initial 24-hour period are enveloped by the maximum pressure and temperature during the initial 24-hour period; or

(2) The maximum pressure and temperature following the initial 24 hour period meet the applicable Level C or Factored Load acceptance criteria, as described in Regulatory Position 3.1d above; or

(3) The calculated release for the more likely severe accident challenges, following the initial 24-hour period, meets site-specific design criteria for fission product released from the containment, in accordance with the requirements of 10 CFR 100.21, "Non-Seismic Siting Criteria," and 10 CFR 50.34, "Contents of Applications; Technical Information."

An applicant can use an alternative method to meet this performance goal if it provides sufficient justification, acceptable to the staff.

b. The development of a finite element model of the containment using the approach described under Regulatory Position 1 is acceptable, subject to the following limitations:

(1) The 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. To verify acceptance criteria in 3.2.a.2, linear elastic material properties may be used for steel containment elements. To verify acceptance criteria in 3.2.a.2 for concrete

containments, the tensile strength of concrete should be neglected while for the concrete in compression the nonlinear stress-strain relationship should be used or justification should be provided for an equivalent linear stress-strain model. The staff will review the justification for an equivalent linear stress-strain model on a case-by-case basis.

(2) The potential structural dynamic amplification effects caused by the pressure transients for severe accident events, if significant, should be included in calculating the response of the containment.

c. If the approach described in Regulatory Position 3.2a(3) above is used, then the applicant should provide sufficient information to enable the staff to review how the calculated release of fission products was determined and how it compares to the site-specific design criteria for fission product release from containment, in accordance with the requirements of 10 CFR 100.21 (Ref. 14) and 10 CFR 50.34 (Ref. 3). The analysis to determine the fission product release from the containment should consider all possible pathways, including components such as penetrations, bolted connections, seals, hatches, and bellows.

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

The description of the evaluation for containment pressure integrity under the more likely severe accident challenges should be presented in Section 19 of the DCD or FSAR, consistent with the requested information identified in Section C.I.19 of Regulatory Guide 1.206 (Ref. 5). If the evaluation to demonstrate the containment pressure integrity for the more likely severe accident challenges relies on a single analysis intended to address multiple regulatory requirements and performance goals, then the applicant should clearly explain how the single analysis addresses each specific requirement or performance goal.

D. IMPLEMENTATION

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

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.

GLOSSARY

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.

design-basis accident pressure—Pressure used in the design of the containment for the design-basis loss-of-coolant accident, in agreement with General Design Criterion 50, "Containment Design Basis," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

design-basis accident temperature—Temperature used in the design of the containment for the design-basis loss-of-coolant accident, in agreement with General Design Criterion 50 in Appendix A to 10 CFR Part 50.

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 that 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 containment leakage 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. The following describes the global containment response for steel containments and concrete containments:

- steel containments—Global containment response of steel containments involves an accurate prediction of the onset of yield, plastic deformation, strain hardening, and buckling, considering the multiaxial nature of the stress state and the prediction of deformation versus pressure in the final stage of loading.

- concrete containments—Global containment response of reinforced and prestressed concrete containments with steel liners involves an 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 a detailed analysis of stress and strain concentration areas at the geometric and material discontinuities, using fine mesh 2-dimensional or 3-dimensional finite element analysis, empirical formulas validated by test results, and buckling models to calculate the critical buckling pressure, with test validation to extrapolate theoretical values to actual configurations. The following describes the local containment response for steel containments and concrete containments:

- steel containments—Local containment response of steel containments involves the 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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8. ASME Boiler & Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," American Society of Mechanical Engineers, New York, NY.[5]

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[1] All NRC regulations listed herein are available electronically through the Public Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/reading-rm/doc-collections/cfr/part050. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; e-mail pdr.resource@nrc.gov.

[2] All NUREG-series reports listed herein were published by the U.S. Nuclear Regulatory Commission. Copies are available for inspection or copying for a fee from the NRC's Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; e-mail pdr.resource@nrc.gov.

[3] All regulatory guides listed herein were published by the U.S. Nuclear Regulatory Commission. Where an ADAMS accession number is identified, the specified regulatory guide is available electronically through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. All other regulatory guides are available electronically through the Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/.

[4] Commission Papers (SECY) listed herein are available electronically through the Public Electronic Reading Room on the NRC's public Web, site at http://www.nrc.gov/reading-rm/doc-collections/commission/secys/
Copies are also available for inspection or copying for a fee from the NRC's Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; e-mail pdr.resource@nrc.gov.

[5] Copies may be purchased from the American Society of Mechanical Engineers, Three Park Avenue, NewYork, NY 10016-5990; phone (212) 591-8500; fax (212) 591-8501; www.asme.org.

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