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# U.S. NUCLEAR REGULATORY COMMISSION

## DRAFT REGULATORY GUIDE DG-1397

***Proposed Revision 3 to Regulatory Guide 1.57***

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# ~~REGULATORY GUIDE~~

~~**OFFICE OF NUCLEAR REGULATORY RESEARCH**~~

## ~~REGULATORY GUIDE 1.57~~

# DESIGN LIMITS AND LOADING COMBINATIONS FOR METAL PRIMARY REACTOR CONTAINMENT SYSTEM COMPONENTS

## A. INTRODUCTION

**Purpose**

This regulatory guide (RG) describes an approach that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in designing metal primary reactor containment system components and it provides methods for demonstrating containment structural integrity.



This RG is being issued in draft form to involve the public in the development of regulatory guidance in this area. It has not received final staff review or approval and does not represent an NRC final staff position. Public comments are being solicited on this DG and its associated regulatory analysis. Comments should be accompanied by appropriate supporting data. Comments may be submitted through the Federal rulemaking Web site, http://www.regulations.gov, by searching for draft regulatory guide DG-1397. Alternatively, comments may be submitted to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Rulemaking and Adjudications Staff. Comments must be submitted by the date indicated in the *Federal Register* notice.

Electronic copies of this DG, previous versions of DGs, and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC Library at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/. The DG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML21208A048. The regulatory analysis is associated with a rulemaking and may be found in ADAMS under Accession No. ML21159A069.

**~~Applicable Rules~~ Applicability**

This RG applies to non-power and ~~**Regulations**~~power reactor licensees and applicants subject to ~~Appendix A, to~~ Title 10~~, Part 50,~~ of the *Code of Federal Regulations* (10 CFR) Part 50~~)~~, "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 2).

**Applicable Regulations**

- Appendix A, ~~1)~~"General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, provides the general design criteria (GDC) for nuclear power plants. The following GDCs are of importance to the design of metal primary reactor containment system components.

  - GDC 1, "Quality ~~standards~~Standards and ~~records~~Records," requires, in part, that structures, systems, and components (SSCs) important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

  - GDC 2, "Design ~~bases~~Bases for ~~protection~~Protection against ~~natural phenomena~~Natural Phenomena," requires, in part, that ~~structures~~SSCs important to safety shall be designed to withstand the effects of ~~expected~~ natural phenomena ~~when combined~~such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches, with the appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena without loss of capability to perform their safety function. In addition, to ensure that the containment of a nuclear power plant is designed to withstand natural phenomena, it is necessary to specify the most severe natural phenomena event that may occur as a function of the frequency of occurrence.

  - GDC 4, "Environmental and ~~dynamic effects, design bases~~Dynamic Effects Design Bases," requires, in part, that ~~nuclear power plant~~ SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs).

  - GDC 16, "Containment ~~design~~Design," requires, in part, that ~~the~~a reactor containment and its associated systems be provided to establish an essentially leak tight barrier against the 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~~.~~ require.

  - GDC 50, "Containment Design Basis," requires, in part, that the reactor containment structure, (including access openings, penetrations, and the containment heat removal ~~systems)~~system), be designed so that the structure and its internal compartments will have the capability to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions caused by ~~a~~any LOCA.

- 10 CFR ~~Part~~ 50 ~~provides for the domestic licensing of production and utilization facilities.~~

  - ~~10 CFR 50.34(f), 10 CFR 50.34(f)(3)(v)(A) and (B) require specific steel containments to meet specific provisions of the Boiler and Pressure Vessel (B&PV) Code promulgated by the American Society of Mechanical Engineers (ASME), when subjected to loads resulting from fuel damage, metal-water reactions, hydrogen burning, and inerting system actuations.~~

- , Appendix J 10 CFR 50.44 provides the requirements for combustible gas control for currently~~-~~licensed reactors and for future water-cooled reactor applicants and licensees.

- 10 CFR 50.55a incorporates by reference (with conditions) codes and standards applicable to metal primary reactor containment system components.

- 10 CFR 50, Appendix J contains requirements for primary reactor containment leakage testing for water-cooled power Reactors. Leak tightness of the containment structure must be tested at regular intervals during the life of the plant.

- 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," contains the requirements for the operating-basis earthquake (OBE) and safe-shutdown earthquake (SSE). The OBE serves the function as an inspection-level earthquake below which the effect on the health and safety of the public would be insignificant and above which the licensee would be required to shut down the plant and inspect for damage.

- ~~10 CFR Part 52 (Ref. 2) governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities licensed under Section 103 of the Atomic Energy Act of 1954, as amended (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242).~~

  - ~~10 CFR 52.47 provides requirements on the content of technical information for standard design certifications submitted under Part 52.~~

- 10 CFR 52.47 provides requirements on the content of technical information for standard design certifications submitted under Part 52.

- 10 CFR 52.77 and 52.79 provide requirements on the technical content of combined operating license applications~~.~~

Meeting these regulatory requirements provides assurance that steel containments used for nuclear power plants will be designed to be capable of performing their containment function as long as required to prevent or mitigate the spread of radioactive material, and that they can withstand the effects of natural phenomena and other external events and maintain the plant in a safe condition.

**Related Guidance**

- NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 3.8.2, “Steel Containment,” (Ref. 3) provides information on how the staff will review the portions of a license application or a license amendment relating to steel containment.

- NUREG/CR-6906, “Containment Integrity Research at Sandia National Laboratories – An Overview,” (Ref. 4) provides guidance for containment model tests.

- Regulatory Guide 1.29, “Seismic Design Classification,” (Ref. 5) describes a method that the NRC staff considers acceptable for use in identifying and classifying those features of light-water-reactor (LWR) nuclear power plants that must be designed to withstand the effects of the safe shutdown earthquake (SSE).

- Regulatory Guide 1.7, “Control of Combustible Gas Concentrations in Containment” (Ref. 6), describes methods acceptable to the NRC staff for implementing 10 CFR 50.44.

- Regulatory Guide 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” (Ref. 7), identifies the Code Cases that have been determined by the NRC to be acceptable alternatives to parts of ASME Section III.

- Regulatory Guide 1.193, “ASME Code Cases Not Approved for Use,” (Ref. 8), provides tables listing unapproved Code Cases for ASME Sections III and XI.

- American Society of Mechanical Engineers, *Boiler & Pressure Vessel Code*, Section III, Division 1, Subsection NE, Class MC Components, “Rules for Construction of Nuclear Facility Components,” (Ref. 9) provides rules for Class MC components in the construction of nuclear facilities.

- American Society of Mechanical Engineers, *Boiler & Pressure Vessel Code*, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” (Ref. 10) provides rules for inspection of nuclear power plant components during the period they are in use.

~~**Information Collection Requirements**~~

**Purpose of Regulatory Guides**

The NRC issues RGs to describe methods that are acceptable to the staff for implementing specific parts of the agency’s regulations, to explain techniques that the staff uses in evaluating specific issues or postulated events, and to describe information that the staff needs in its review of applications for permits and licenses. Regulatory guides are not NRC regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs are acceptable if supported by a basis for the issuance or continuance of a permit or license by the Commission.

**Paperwork Reduction Act**

This ~~regulatory guide contains~~RG provides voluntary guidance for implementing the mandatory information ~~collection requirements covered by~~collections in 10~~-~~ CFR~~ Part~~ Parts 50 and ~~10 CFR Part~~ 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB~~) approved under~~), approval numbers 3150-0011 and 3150-0151. Send comments regarding this information collection to the Information Services Branch (T6-A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the OMB ~~control number 3150-0011 and 3150-0151, respectively.~~ reviewer at: OMB Office of Information and Regulatory Affairs (3150-0011 and 3150-0151), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW Washington, DC 20503; e-mail: oira_submission@omb.eop.gov.

**Public Protection Notification**

The NRC may ~~neither~~not conduct ~~nor~~or sponsor, and a person is not required to respond to, ~~an information~~ a collection ~~request or requirement~~of information unless the document requesting ~~document~~ or requiring the collection displays a currently valid OMB control number.

# B. -DISCUSSION

**Reason for Revision**

~~RG 1.57 was revised to correct an editorial error in a title found on page 10 of RG 1.57, Revision. 1 which referred to the "Ultimate Capacity of Concrete Containment" when it should have been "Ultimate Capacity of Steel Containments." In addition, the text in this section referred to SRP Section 3.8.2 "Steel Containment" for guidance without further elaboration. That was corrected by importing the guidance found in Section 3.8.2 into Revision 2. This did not change the staff regulatory guidance. In addition, editorial changes were made to improve clarity, and ADAMS Accession Numbers were added in the reference section to facilitate public access the documents.~~

This revision of the guide (Revision 5) makes administrative changes as a result of the final rule for the "Alignment of Licensing Processes and Lessons Learned from New Reactor Licensing" (RIN 3150–AI66). This revised guide deletes reference to 10 CFR 50.34(f) with respect to combustible gas control requirements which are cited in 10 CFR 50.44. It should be noted that Part 52 applicants are exempt from the combustible gas control requirements cited in 10 CFR 50.34(f).

## Background

The American Society of Mechanical Engineers (ASME) publishes the "Rules for Construction of Nuclear Facility Components," as Section III, "Nuclear Components," of the ASME Boiler &Pressure Vessel (B&PV) Code. Sections III and XI of the B&PV Code are incorporated by reference, with conditions, into 10 CFR 50.55a. Section III, Division 1, Subsection NE, sets forth 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. ASME B&PV Code Section III, Division 1, is hereinafter referred to as "the Code." The existing Section III of the Code is based on the current class of light-water reactors and, as such, may not adequately address design and construction features of the next generation of advanced reactors. However, the provisions of this guide may be used for the current light-water reactors, as well as future advanced reactors, such as the Advanced Pressurized-Water Reactor (AP1000) and the Economic Simplified Boiling-Water Reactor (ESBWR).

The NRC is committed to the use of consensus codes and standards for the design, construction, and licensing of commercial nuclear power reactors facilities. Thus, the recent significant advancement in the technology (both in the nuclear industry and the Code) has prompted a need to revise the regulatory guidance for metal containments. Toward that end, this regulatory guide provides guidance on the use of codes and standards for the design of advanced reactors to ensure that SSCs will perform their intended safety functions. While this regulatory guide is only directly applicable to light-water reactor metal containments, the principles may be applied to non-light water reactor containments, subject to review by the NRC.

10 CFR 50.44(b)(2)(i) requires that all currently licensed boiling-water reactors with Mark I or Mark II type containments must have an inert atmosphere. 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 so that there is no loss of containment structural integrity. 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, demonstrate that 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 to be unlikely to occur.

10 CFR 50.44(c)(3) requires that future water-cooled reactors containments that do not rely upon an inert atmosphere to control combustible gases must have the capability for controlling 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. 10 CFR 50.44(c)(5) requires that for future water-cooled reactors containments, an applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by 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 an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning.

To address the requirements of 10 CFR 50.~~34(f) and 10 CFR 50.~~44(b) and (c), Staff ~~-~~Regulatory Guidance Position I. B. 3. (c) "Level C Service Limits,"~~ on page 10~~ provides load combinations for pressure loads that result from a fuel-clad metal-water reaction, an uncontrolled hydrogen burn, and from a post-accident condition inerted by carbon dioxide.

The design conditions and functional requirements of components that provide a pressure boundary for the primary reactor containment function should be reflected in the application of appropriate design limits (e.g., stress or strain limits) for the most adverse combination of loadings to which these components might be subjected. For components constructed in accordance with Subsection NE of the Code (Code Class MC), the NRC requires provision of a design specification, which stipulates the design requirements (e.g., the mechanical and operational loadings) for the components.

In Appendix B to the Code, entitled "Owner's Design Specifications," Paragraph B-2125, "Load Combinations," states, "In order to provide a complete definition of service loads, the combination of specific events must be considered. Since these combinations are a function of specific systems which make up a part of a specific type of nuclear facility, this section does not directly address service loads other than to provide different stress limits for various loadings."

To further provide a consistent basis for the design of metal containment system components, this guide delineates acceptable design limits for appropriate combinations of loadings. The intent is to address only the most adverse combinations of loadings resulting from those events or conditions identified herein (e.g., those combinations of loadings that result in the limiting or controlling design condition). These loadings are associated either with conditions for which the containment function is required in combination with specified seismic events producing possible mechanisms for failure that could affect the function and/or integrity of structures, systems, and components important to safety. Included in the latter are the loadings associated with the vibratory motion of the safe-shutdown earthquake (SSE), design external pressure (if applicable), and other loadings that induce compressive stresses. The effects of natural phenomena other than earthquakes, such as tornadoes, hurricanes, and floods, are not considered in this guide, because a Category I concrete shield building typically protects the steel containment from the effects of tornadoes, hurricanes, and floods occurring outside the shield building. In addition to the loading combinations addressed in this guide, primary reactor containment components enclosed within Seismic Category I structures should be designed to withstand the effects of pertinent natural phenomena not otherwise protected against.

The approach set forth in this guide is directly related to Section III of the Code. Design limits as specified in Section III are adopted to provide assurance of maintaining the pressure-retaining integrity

of the primary reactor containment. The primary reactor containment system of metal construction includes all components that perform a containment function, such as (1) the containment vessel(s), (2) penetration assemblies and access openings, and (3) piping systems attached to the containment vessel nozzles or penetration assemblies out to and including all pumps and valves required to isolate the containment.

The only components that are classified as ASME Code Class MC (i.e., components constructed in accordance with the rules of Subsection NE of the Code) are metal containment vessels, including parts and appurtenances thereof.1 Such parts and appurtenances may include mechanical, electrical, and piping penetration assemblies,2 bellows-type expansion joints, and access openings. Piping, pumps, and valves that are defined as components of primary metal containment systems are constructed in accordance with the rules for either Code Class 1 or Code Class 2 components, as required by Article NE-1120. Any piping penetration assemblies or appurtenances that are not a part of the containment vessel should be constructed in accordance with the rules for Code Class 1 or Code Class 2 components, as required by the intended service function.

In addition to the above discussion, 10 CFR 50.55a also imposes the examination requirements established in Section XI, Subsection IWE, of the ASME B&PV Code ~~(Ref. 10),~~, as they relate to metal containments and liners of concrete containments.

~~**Secondary Reference**~~ ”
**Consideration of International Standards**

The NRC has a goal of harmonizing its regulatory guidance with documents issued by the International Atomic Energy Agency (IAEA) to the extent practical. Although the NRC does not endorse the following IAEA safety standard(s) and/or guide(s), in general this RG incorporates similar guidelines and is generally consistent with the basic safety principles provided in them.

- IAEA Safety Standards Series No. SSG-53, “Design of Reactor Containment and Associated Systems of Nuclear Power Plants,” issued 2019 (Ref. 11), provides guidance relevant to reactor containment and associated systems at a high level. It addresses the containment structure and the systems with the functions of isolation, control and management of mass and energy releases, control and limitation of radioactive releases, and control and management of combustible gases.

**Documents ~~Used~~Discussed in Staff Regulatory Guidance**

This regulatory guide endorses, in part, the ASME Boiler & Pressure Vessel Code (Ref. 9 & 10). The Code contains references to other codes, standards and third party guidance documents (“secondary references”). If a secondary reference has itself been incorporated by reference into NRC regulations as a requirement, then licensees and applicants must comply with that standard as set forth in the regulation. If the secondary reference has been endorsed in a regulatory guide as an acceptable approach for meeting an NRC requirement, then the standard constitutes a method acceptable to the NRC staff for meeting that regulatory requirement as described in the specific regulatory guide. If the secondary reference has neither been incorporated by reference into NRC regulations nor endorsed in a regulatory guide, then the secondary

1 Refer to NCA-9200 of the Code for definitions of “parts” and “appurtenances.”

2 Penetration assemblies are parts or appurtenances that are required to permit piping, mechanical devices, and electrical connections to pass through the containment vessel shell or head and maintain leak tight integrity, while compensating for such things as temperature and pressure fluctuations and earthquake movements.

reference is neither a legally~~-~~ binding requirement nor a “generic” NRC approval as an acceptable approach for meeting an NRC requirement.  However, licensees and applicants may consider and use the information in the secondary reference, if appropriately justified and consistent with current regulatory practice, consistent with applicable NRC requirements ~~such as 10 CFR 50.59~~.

# C. -STAFF REGULATORY GUIDANCE

**I. Code Class MC vessels, electrical and mechanical penetration assemblies, and other penetration assemblies (excluding bellows-type expansion joints) that are parts or appurtenances of the vessel.**

Code Class MC components of primary metal containment systems that are completely enclosed within Seismic Category I structures[3] should be designed to withstand the following loads and loading combinations within the specified design limits.

**A. Loads**

D Dead loads.

L Live loads, including all loads resulting from platform flexibility and deformation and from crane loading, if applicable.

$P_t$ Test pressure.

$T_t$ Test temperature.

$T_o$ Thermal effects and loads during startup, normal operating, or shutdown conditions, based on the most critical transient or steady-state condition.

$R_o$ Pipe reactions during startup, normal operating, or shutdown conditions based on the most critical transient or steady-state condition.

$P_o$ External pressure loads resulting from pressure variation either inside or outside containment.

E Loads generated by the operating-basis earthquake including sloshing effects, if applicable.

E' Loads generated by the SSE, including sloshing effects.

$P_a$ Pressure load generated by the postulated pipe break accident (including pressure generated by postulated small-break or intermediate-break pipe ruptures), pool swell, and subsequent hydrodynamic loads.[4]

$T_a$ Thermal loads under thermal conditions generated by the postulated pipe break accident, pool swell, and subsequent hydrodynamic reaction loads.[5]

$R_a$ Pipe reactions under thermal conditions generated by the postulated pipe break accident, pool swell, and subsequent hydrodynamic reaction loads.[5]

$P_s$ All pressure loads that are caused by the actuation of safety relief valve (SRV) discharge, including pool swell and subsequent hydrodynamic loads.

$T_s$ All thermal loads that are generated by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic thermal loads.

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3 Components of primary reactor containment systems are Seismic Category I for seismic design purposes in accordance with Regulatory Guide 1.29. Seismic Category I SSCs are designed to remain functional if the SSE occurs.

4 For load combinations 1.2.3.1(4), 1.2.3.3(3), and 1.2.3.4(2), a small or intermediate pipe break is postulated. For all other load combinations involving a loss of coolant accident (LOCA), the design-basis LOCA is postulated.

$R_s$ All pipe reaction loads that are generated by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic reaction loads.

$Y_r$ Equivalent static load on the structure generated by the reaction on the broken pipe during the design-basis accident.

$Y_j$ Jet impingement equivalent static load on the structure generated by the broken pipe during the design-basis accident.

$Y_m$ Missile impact equivalent static load on the structure generated by or during the design-basis accident, such as pipe whipping.

$F_L$ Load generated by the post-LOCA flooding of the containment, if any.

$P_{g1}$ Pressure resulting from an accident that releases hydrogen generated from 100% fuel clad metal-water reaction.

$P_{g2}$ Pressure resulting from uncontrolled hydrogen burning.

$P_{g3}$ Pressure resulting from post-accident inerting, assuming carbon dioxide is the inerting agent. See Regulatory Guide 1.7 "Control of Combustible Gas Concentrations in Containment," for additional guidance about the pressure load $P_{g3}$ due to combustible gas concentration.

## B. Loading Combinations and Design Limits

The specified loads and load combinations are acceptable if found to be in accordance with the following guidance. The following load combinations include all loading combinations for which the containment might be designed for or subjected to during the expected life of the plant:

### 1. *Testing Condition*

This includes the testing condition of the containment to verify its leak integrity. In this case, the loading combination includes:

$D + L + T_t + P_t$

### 2. *Design Conditions*

These include all design loadings for which the containment vessel or portions thereof might be designed for during the expected life of the plant. Such loads include design pressure, design temperature, and the design mechanical loads generated by the design-basis accident. In this case, the loading combination includes:

$D + L + P_a + T_a + R_a$

### 3. *Service Conditions*

The load combinations in these cases correspond to and include Level A service limits, Level B service limits, Level C service limits, Level D service limits and the post-flooding condition. The loads may be combined by their actual time history of occurrence, taking into consideration their dynamic effect upon the structure.

#### (a) Level A Service Limits

These service limits are applicable to the service loadings to which the containment is subjected, including the plant or system design-basis accident conditions for which the containment function is required, excepting only those categorized as Level B, C, or D, or Testing Loadings. The loading combinations corresponding to these limits include the following:

(i) Normal operating plant condition

$D + L + T_o + R_o + P_o$

(ii) Operating plant condition in conjunction with multiple SRV actuations

$D + L + T_s + R_s + P_s$

(~~ii~~iii) Loss-of-coolant accident

$D + L + T_a + R_a + P_a$

(iv) Multiple SRV actuations in combination with a small- or intermediate-break accident

$D + L + T_a + R_a + P_a + T_s + R_s + P_s$

(v) Normal operating plant conditions in combination with inadvertent full actuation of a post-accident inerting hydrogen control system [reference 10 CFR 50.~~34(f)(3)(v)(B)(1)]~~44]

$D + L + T_o + R_o + P_o + P_{g3}$

(vi) Pressure test load to ensure that the containment will safely withstand the pressure calculated to result from carbon-dioxide inerting [reference 10 CFR 50.~~34(f)(3)(v)(B)(2)]~~44]

$D + 1.10 \times P_{g3}$

#### (b) Level B Service Limits

These service limits include the loads subject to Level A service limits, plus the additional loads resulting from natural phenomena during which

the plant must remain operational. The loading combinations corresponding to these limits include the following:

(i) Design-basis LOCA in combination with the operating-basis earthquake (if E ≤ one-third E', only its contribution to cyclic loading needs to be considered)

$$D + L + T_a + R_a + P_a + E$$

(ii) Operating plant condition in combination with the operating-basis earthquake (if E ≤ one-third E', only its contribution to cyclic loading needs to be considered)

$$D + L + T_o + R_o + P_o + E$$

(iii) Operating plant condition in combination with the operating-basis earthquake and multiple SRV actuations (if E ≤ one-third E', only its contribution to cyclic loading needs to be considered)

$$D + L + T_s + R_s + P_s + E$$

(iv) Loss-of-coolant accident in combination with a single active component failure causing one SRV discharge

$$D + L + T_a + P_a + R_a + T_s + R_s + P_s$$

**(c) Level C Service Limits**

These service limits include the loads subject to Level A service limits, plus the additional loads resulting from natural phenomena for which safe shutdown of the plant is required. The loading combinations corresponding to these limits include the following:

(i) Loss-of-coolant accident in combination with the SSE

$$D + L + T_a + R_a + P_a + E'$$

(ii) Operating plant condition in combination with the SSE

$$D + L + T_o + R_o + P_o + E'$$

(iii) Multiple SRV actuations in combination with a small- or intermediate-break accident and SSE

$$D + L + T_a + R_a + P_a + T_s + R_s + P_s + E'$$

(iv) Dead load plus pressure resulting from an accident that releases hydrogen generated from 100% fuel clad metal-water reaction

accompanied by hydrogen burning [reference 10 CFR 50.~~34(f)(3)(v)(A)(1); 10 CFR 50.44]~~44]

$D + P_{g1} + P_{g2}$

[NOTE: In this load combination, $P_{g1} + P_{g2}$ should not be less than 310 kPa (45 psig) and evaluation of instability is not required.]

(v) Dead load plus pressure resulting from an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by the added pressure from post-accident inerting, assuming carbon dioxide as the inerting agent [reference 10 CFR 50.~~34(f)(3)(v)(A)(1); 10 CFR 50.44]~~44]

$D + P_{g1} + P_{g3}$

[NOTE: In this load combination, $P_{g1} + P_{g3}$ should not be less than 310 kPa (45 psig) and evaluation of instability is not required.]

**(d) Level D Service Limits**

These service limits include other applicable service limits and loadings of dynamic nature for which the containment function is required. The load combinations corresponding to these limits include the following:

(i) Loss-of-coolant accident in combination with the SSE and local dynamic loadings

$D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + E'$

(ii) Multiple SRV actuations in combination with a small- or intermediate-break accident, SSE, and local dynamic loadings

$D + L + T_a + R_a + P_a + Y_r + Y_j + Y_j + P_s + T_s + R_s + E'$

(iii) Post-LOCA flooding of the containment in combination with the operating-basis earthquake

$D + L + F_L + E$

## C. Design Limits

Total stresses for the combination of loads delineated in Regulatory Position 1.2 (above) are acceptable if found to be within the limits defined by Articles NE-3221.1, NE-3221.2, NE-3221.3 and NE-3221.4 of the Code.

## D. Treatment of Buckling Effects

Earthquake, thermal, and pressure loads require consideration of buckling of the shell. Buckling of shells with more complex geometries and loading conditions than those covered by Article NE-3133 of the Code should be considered in accordance with the criteria

described in ASME Code Case N-284-2[5]. An acceptable approach to this problem is to perform a nonlinear analysis.

## II. Bellows-Type Expansion Joints that are Parts or Appurtenances of ASME Code Class MC Vessels

Bellows-type expansion joints that are parts or appurtenances of Code Class MC components that are completely enclosed within Seismic Category I structures should be designed to withstand the loads and loading combinations within the design limits specified in Regulatory Position 1 (above), as applicable, supplemented by the design limits specified in Article NE-3366.2(b) of the Code.

## III. Ultimate Capacity of Steel Containments

### A. Determination of the Ultimate Capacity of the Containment for New Reactors

A nonlinear finite element analysis should be performed to determine the ultimate capacity of the containment. For new reactors, a determination of the internal pressure capacity for containment structures, as a measure of the safety margin above the design-basis accident pressure is needed.

#### 1. Cylindrical Steel Containments

An acceptable method for cylindrical steel containments is to estimate the capacity based on attaining a maximum global membrane strain away from discontinuities (i.e., the hoop membrane strain in a cylinder) of 1.5 percent. To conduct the necessary analysis, both nonlinear material behavior and nonlinear geometric behavior must be considered. The stress-strain curve for the steel containment material should be based on the code-specified minimum yield strength and a stress-strain relationship above yield that is representative of that specific grade of steel. The stress-strain curve must be developed for the design basis accident temperature.

#### 2. Non-cylindrical Containments

Analyses of non-cylindrical containments and analyses of cylindrical containments that use alternate failure criteria will be subject to detailed staff review, on a case-by-case basis.

### B. Application of the Analysis to Existing Containment Structures

In applying the analysis to existing containment structures, it is permissible to use as-built material properties for the steel containment material. Sufficient material certification data must be available to establish with reasonable confidence a lower bound, a median, and an upper bound value for the important material parameters. These values must be adjusted for

5 Code Case N-284-2, “Metal Containment Shell Buckling Design Methods, Class MC Section III, Division 1,” has been endorsed by RG 1.84. Revision 2 of N-284 corrected errata, misprints, recommendations, and errors identified by the NRC staff in N-284-1 as the use of N-284-1 is unacceptable to the NRC, as discussed in Regulatory Guide 1.193.

the design-basis accident temperature. For deterministic assessments, the lower bound values should be used. For probabilistic risk assessment, calculations of failure probability vs. pressure should consider the statistical distribution of the material properties.

## C. Containment Penetrations

The methods described above apply to the containment structure. A complete evaluation of the internal pressure capacity must 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. Other potential containment leak paths through mechanical and electrical penetrations should also be addressed.

## D. Special Considerations for Steel Ellipsoidal and Torispherical Heads

Under internal pressure, a potential failure mode of steel ellipsoidal and torispherical heads is buckling, resulting from a hoop compression zone in the knuckle region. This potential mode of failure needs to be evaluated, 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 appropriately demonstrated, residual post buckling strength can be considered in determining the pressure capacity.

The details of the analysis and the results should be submitted in a report form with the following identifiable information:

1. Original design pressure, P, as defined in ASME Code, Section III, Division 1, Subsection NE, Sub article NE-3112.1;

2. Calculated static pressure capacity;

3. Equivalent static pressure response calculated from dynamic pressure;

4. Associated failure mode;

5. Criteria governing the original design and the criteria used to establish failure;

6. Analysis details and general results; and

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

## D. IMPLEMENTATION

The ~~purpose of this section is to provide information on how applicants and licensees~~[6] NRC staff may use this ~~guide and information regarding the NRC's plans for using this~~ RG as a reference in its regulatory ~~guide. In addition, it describes how the NRC staff complies with 10 CFR 50.109, "Backfitting" and any applicable finality provisions in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."~~

~~**Use by Applicants and Licensees**~~

~~Applicants and licensees may voluntarily~~ [7]~~use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this regulatory guide may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations. Current licensees may continue to use guidance the NRC found acceptable for complying with the identified regulations as long as their current~~ processes, such as licensing ~~basis remains unchanged.~~

~~Licensees may use the information in this regulatory guide for actions which do not require NRC review and approval such as changes to a facility design under 10 CFR 50.59, "Changes, Tests, and Experiments." Licensees may use the information in this regulatory guide or applicable parts to resolve regulatory or~~, inspection ~~issues.~~

~~**Use by NRC Staff**~~

~~The~~ , or enforcement. However, the NRC staff does not intend to use the guidance in this RG to support NRC staff ~~does not intend or approve any imposition or~~actions in a manner that would constitute backfitting ~~of the guidance in this regulatory guide. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this regulatory guide, unless the licensee makes a change to its licensing basis. The NRC staff does not expect or plan to request licensees to voluntarily adopt this regulatory guide to resolve a generic regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action which would require the use of this regulatory guide. Examples of such unplanned NRC regulatory actions include issuance of an order requiring the use of the regulatory guide, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this regulatory~~

~~[6] In this section, "licensees" refers to licensees of nuclear power plants under 10 CFR Parts 50 and 52; and the term "applicants," refers to applicants for licenses and permits for (or relating to) nuclear power plants under 10 CFR Parts 50 and 52, and applicants for standard design approvals and standard design certifications under 10 CFR Part 52.~~

~~[7] In this section, "voluntary" and "voluntarily" means that the licensee is seeking the action of its own accord, without the force of a legally binding requirement or an NRC representation of further licensing or enforcement action.~~

~~guide, generic communication, or promulgation of a rule requiring the use of this regulatory guide without further backfit consideration.~~

~~During regulatory discussions on plant specific operational issues, the staff may discuss with licensees various actions consistent with staff positions in this regulatory guide, as one acceptable means of meeting the underlying NRC regulatory requirement. Such discussions would not ordinarily be considered backfitting even if prior versions of this regulatory guide are part of the licensing basis of the facility. However, unless this regulatory guide is part of the licensing basis for a facility, the staff may not represent to the licensee that the licensee's failure to comply with the positions in this regulatory guide constitutes a violation.~~

~~If an existing licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this new or revised regulatory guide and (2) the specific subject matter of this regulatory guide is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the guidance in this regulatory guide or provide an equivalent alternative process that demonstrates compliance with the underlying NRC regulatory requirements. This is not considered backfitting as~~ as that term is defined in 10 CFR 50.109~~(a)(1) or a violation~~, "Backfitting," and as described in NRC Management Directive 8.4, "Management of ~~any of~~ Backfitting, Forward Fitting, Issue Finality, and Information Requests" (Ref. 12), nor does the NRC staff intend to use the guidance to affect the issue finality ~~provisions in~~ of an approval under 10 CFR Part 52.

~~Additionally, an existing applicant may be required to comply to new rules, orders, or guidance if 10 CFR 50.109(a)(3) applies.~~

The staff also does not intend to use the guidance to support NRC staff actions in a manner that constitutes "forward fitting" as that term is defined and described in Management Directive 8.4. If a licensee believes that the NRC is ~~either~~ using this ~~regulatory guide or requesting or requiring the licensee to implement the methods or processes in this regulatory guide~~RG in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a ~~backfit~~backfitting or forward fitting appeal with the NRC in accordance with the ~~guidance~~process in ~~NUREG-1409, "Backfitting Guidelines," (Ref. 11) and the NRC~~ Management Directive 8.4~~, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 12)~~.

# REFERENCES[8]


1. *U.S. Code of Federal Regulations*~~,~~ (CFR), "Domestic Licensing of Production and Utilization Facilities," ~~Appendix A, "General Design Criteria for Nuclear Power Plants," Part 50~~Part 50, Chapter 1, Title 10, "Energy."

2. ~~*U.S. Code of Federal Regulations,*~~ CFR, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter 1, Title~~-~~ 10, "Energy."

3. U.S. Nuclear Regulatory Commission~~, "~~ (NRC), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Washington,~~-~~ DC.

4. U.S. Nuclear Regulatory Commission, "Containment Integrity Research at Sandia National Laboratories – An Overview," NUREG/CR-6906, Washington, DC, ADAMS Accession Number ML062440075.

5. ~~U.S. Nuclear Regulatory Commission,~~ NRC, RG 1.29, "Seismic Design Classification," Washington, DC.

~~5.~~ NRC, Regulatory Guide ~~1.29, Washington, DC.~~

6. ~~U.S. Nuclear Regulatory Commission~~RG 1.7, "Control of Combustible Gas Concentrations in Containment," ~~Regulatory Guide 1.7,~~ Washington, DC.

7. ~~U.S. Nuclear Regulatory Commission~~NRC, RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section~~-~~ III," ~~Regulatory Guide 1.84~~Revision 38, Washington, DC~~.~~, October 2019 (ADAMS Accession No. ML19128A276).


[8] Publicly available documents from the U.S. Nuclear Regulatory Commission (NRC) are available electronically through the NRC Library on the NRC's public Web site at ~~http://www.nrc.gov/reading-rm/doc-collections/.~~ http//www.nrc.gov/reading-rm/doc-collections. The documents can also be viewed on-line for free or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD~~;~~, the mailing address is USNRC PDR, Washington, DC 20555~~;~~, telephone (301) 415-4737 ~~or~~0r (800) 397-4209~~;~~, fax (301) 415~~-~~-3548; ~~and~~ e-mail ~~pdr.~~pdf resource@nrc.gov.

8. ~~U.S. Nuclear Regulatory Commission,~~ NRC, RG 1.193, "ASME Code Cases Not Approved for Use," ~~Regulatory Guide 1.193,~~ Washington, DC.

9. American Society of~~-~~ Mechanical Engineers~~,~~ (ASME) Boiler ~~&~~and Pressure Vessel Code,~~-~~ Section III, Division 1, Subsection NE, Class MC Components, "Rules for Construction of Nuclear Facility Components," New York, ~~New York.~~NY.[9]

10. ~~American Society of Mechanical Engineers,~~ASME Boiler ~~&~~and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," New York, ~~New York~~NY.

~~11. U.S.~~

11. International Atomic Energy Agency (IAEA) Specific Safety Guide No. SSG-53, "Design of Reactor Containment and Associated Systems for Nuclear ~~Regulatory Commission, "~~Power Plants," Vienna, Austria, 2019.[10]

12. NRC, Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information ~~Collection, NUREG-1409, July 1990,~~Requests," Washington, DC~~, ADAMS Accession No. ML032230247~~.

~~12. U.S. Nuclear Regulatory Commission, "Management of Facility-specific Backfitting and Information Collection," NRC Management Directive 8.4, October 2004, Washington, DC, ADAMS Accession No. ML050110156.~~



[9] Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Three Park Avenue, New York, ~~New York~~NY 10016-5990; ~~Telephone~~telephone (800) 843-2763.~~-~~ Purchase information is available through the ASME Web site store at ~~http://www.asme.org/Codes/Publications/~~http://www.asme.org/Codes/Publications/.

[10] Copies of IAEA documents may be obtained through its Web site WWW.IAEA.Org/ or by writing the International Atomic Energy Agency, P.O. Box 100 Wagramer Strasse 5, A-1400, Vienna, Austria.

~~Written suggestions regarding this guide or development of new guides may be submitted through the NRC's public Web site under the Regulatory Guides document collection of the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/contactus.html.~~

~~Electronic copies of this regulatory guide and previous version of this guide and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/. The regulatory guide is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML12325A043.~~