U.S. NUCLEAR REGULATORY COMMISSION

March 2007
Revision 1

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.57

(*Draft was issued as DG-1158, dated October 2006*)

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

A. INTRODUCTION

This regulatory guide describes an approach that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in satisfying the requirements of General Design Criteria (GDC) 1, 2, 4, and 16, as specified in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50), "Domestic Licensing of Production and Utilization Facilities" (Ref. 1). Specifically, GDC 1, "Quality Standards and Records," requires, in part, that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

To augment those requirements, GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that structures important to safety be designed to withstand the effects of expected natural phenomena when combined with the effects of normal accident conditions 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. Similarly, GDC 4, "Environmental and Dynamic Effects, Design Bases," requires that nuclear power plant SSCs important to safety be designed to accommodate the effects of and be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs).

In addition, GDC 16, “Containment Design,” requires that the reactor containment and its associated systems be provided to establish an essentially leaktight barrier against uncontrolled release of radioactivity to the environment and to ensure that design conditions important to safety are not exceeded for as long as required for postulated accident conditions. Finally, GDC 50, “Containment Design Basis,” requires that the reactor containment structure (including access openings, penetrations, and containment heat removal systems) be designed so that the structure and its internal compartments will have the capability to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions caused by a LOCA.

10 CFR 50.44 provides the requirements for combustible gas control for currently-licensed reactors and for future water-cooled reactor applicants and licensees. This regulatory guide describes an approach acceptable to the NRC staff to consider the structural loads involved and determine the containment response to demonstrate the containment structural integrity.

Moreover, leaktightness of the containment structure must be tested at regular intervals during the life of the plant, in accordance with the provisions of 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.” In addition, for certain reactors specified in 10 CFR 50.34(f), 10 CFR 50.34(f)(3)(v)(A) and (B) require 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.

Meeting these criteria 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.

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

B. DISCUSSION

Background

The American Society of Mechanical Engineers (ASME) publishes the "Rules for Construction of Nuclear Facility Components," as Section III of the ASME B&PV Code (Ref. 2).[1] In that section, 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"). However, the existing industry codes and standards are 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. 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 industry 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 inerted 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 inerted 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.

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[1] ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Components," Division 1, including that part of the Summer 2003 Addenda.

To address the requirements of 10 CFR 50.34(f) and 10 CFR 50.44(b) and (c), Regulatory Position C.1.2.3.3 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.[2] Such parts and appurtenances may include mechanical, electrical, and piping penetration assemblies,[3] 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. 3), as they relate to metal containments and liners of concrete containments.

Ultimate Capacity of Steel Containment

New guidelines for the ultimate capacity of steel containments are added in this regulatory guide in order to be consistent with the current staff position. These guidelines are being considered for inclusion in the update to Section 3.8.2, “Steel Containment,” of NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants” (Ref. 4).

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

3 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 leaktight integrity, while compensating for such things as temperature and pressure fluctuations and earthquake movements.

C. REGULATORY POSITION

1. 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.

For earthquake engineering criteria, 10 CFR Part 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” would be applicable for the operating-basis earthquake (OBE) and safe-shutdown earthquake (SSE). In this manner, 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.

Code Class MC components of primary metal containment systems that are completely enclosed within Seismic Category I structures[4] should be designed to withstand the following loads and loading combinations within the specified design limits.

1.1 <u>Loads</u>

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.[5]

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]

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4 Components of primary reactor containment systems are Seismic Category I for seismic design purposes in accordance with Regulatory Guide 1.29, “Seismic Design Classification” (Ref. 5). Seismic Category I SSCs are designed to remain functional if the SSE occurs.

5 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.

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.

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 uncontrlooed hydrogen burning.

P_{g3} ----- Pressure resulting from post-accident inerting, assuming carbon dioxide is the inerting agent.

See Regulatory Guide 1.7 (Ref. 6) for additional guidance about the pressure load P_{g3} due to combustible gas concentration.

1.2 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:

1.2.1 *Testing Condition*

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

$D + L + T_t + P_t$

1.2.2 *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:

$D + L + P_a + T_a + R_a$

1.2.3 *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.

1.2.3.1 <u>Level A Service Limits</u>

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:

(1) Normal operating plant condition

$D + L + T_o + R_o + P_o$

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

$D + L + T_s + R_s + P_s$

(3) Loss-of-coolant accident

$D + L + T_a + R_a + P_a$

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

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

(5) 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)]

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

(6) 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)]

$D + 1.10 \times P_{g3}$

1.2.3.2 <u>Level B Service Limits</u>

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:

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

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

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

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

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

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

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

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

1.2.3.3 <u>Level C Service Limits</u>

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:

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

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

(2) Operating plant condition in combination with the SSE

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

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

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

(4) 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]

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

[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.]

(5) 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]

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

[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.]

1.2.3.4 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:

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

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

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

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

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

$D + L + F_L + E$

1.3 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.

1.4 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, pending endorsement in Regulatory Guide 1.84 (Ref. 7).[6] An acceptable approach to this problem is to perform a nonlinear analysis.

2. 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.

Ultimate Capacity of Concrete Containment

A nonlinear finite element analysis should be performed to determine the ultimate capacity of the containment. Additional information guidance is provided in the SRP 3.8.2.

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6 Code Case N-284, "Metal Containment Shell Buckling Design Methods, Class MC Section III, Division 1," is currently being revised. Revision 1 of N-284 is unacceptable to the NRC, as discussed in Regulatory Guide 1.193 (Ref. 8). Revision 2 of N-284 is correcting errata, misprints, recommendations, and errors identified by the NRC staff, and is expected to be approved when it is published.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. No backfitting is intended or approved in connection with its issuance.

Except in those cases in which an applicant or licensee proposes or has previously established an acceptable alternative method for complying with specified portions of the NRC's regulations, the methods to be described in the active guide will reflect public comments and will be used in evaluating (1) submittals in connection with applications for construction permits, standard plant design certifications, operating licenses, early site permits, and combined licenses; and (2) submittals from operating reactor licensees who voluntarily propose to initiate system modifications if there is a clear nexus between the proposed modifications and the subject for which guidance is provided herein.

REGULATORY ANALYSIS / BACKFIT ANALYSIS

The regulatory analysis and backfit analysis for this regulatory guide are available in Draft Regulatory Guide DG-1158, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components" (Ref. 9). The NRC issued DG-1158 in October 2006 to solicit public comment on the draft of this Revision 1 of Regulatory Guide 1.57.

REFERENCES

1. *U.S. Code of Federal Regulations,* Title 10, *Energy*, Part 50, "Domestic Licensing of Production and Utilization Facilities."[7]

2. ASME Boiler & Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," Division 1, Subsection NE, "Class MC Components," 2001 Edition with 2003 Addenda, American Society of Mechanical Engineers, New York, New York.[8]

3. ASME Boiler & Pressure Vessel Code**,** Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition with 2003 Addenda, American Society of Mechanical Engineers, New York, New York.

4. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[5]

5. Regulatory Guide 1.29, "Seismic Design Classification," U.S. Nuclear Regulatory Commission, Washington, DC.[9]

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

7. Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," U.S. Nuclear Regulatory Commission, Washington, DC.

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[7] All NRC regulations listed herein are available electronically through the Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/reading-rm/doc-collections/cfr/. 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; email PDR@nrc.gov.

[8] Copies of the Code and addenda thereto may be obtained from the American Society of Mechanical Engineers, Three Park Avenue, New York, New York 10016-5990.

[5] 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; email PDR@nrc.gov. Copies are also available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328, telephone (202) 512-1800; or from the National Technical Information Service (NTIS) at 5285 Port Royal Road, Springfield, Virginia 22161, online at http://www.ntis.gov, by telephone at (800) 553-NTIS (6847) or (703) 605-6000, or by fax to (703) 605-6900. NUREG-0800 is also available electronically through the Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/ reading-rm/doc-collections/nuregs/.

[9] 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/. Single copies of regulatory guides may also be obtained free of charge by writing the Reproduction and Distribution Services Section, ADM, USNRC, Washington, DC 20555-0001, or by fax to (301)415-2289, or by email to DISTRIBUTION@nrc.gov. Active guides may also be purchased from the National Technical Information Service (NTIS) on a standing order basis. Details on this service may be obtained by contacting NTIS **Error! Main Document Only.**at 5285 Port Royal Road, Springfield, Virginia 22161, online at http://www.ntis.gov, by telephone at (800) 553-NTIS (6847) or (703) 605-6000, or by fax to (703) 605-6900. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4209, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

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

9. Draft Regulatory Guide DG-1158, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," U.S. Nuclear Regulatory Commission, Washington, DC.[10]

[10] Draft Regulatory Guide DG-1158 is available electronically under Accession #ML063000278 in the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4209, by fax at (301) 415-3548, and by email to PDR@nrc.gov.