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MATERIALS AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS

A. INTRODUCTION

General Design Criterion (GDC) 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, of the *Code of Federal Regulations*, Part 50, "Domestic Licensing of Production and Utilization Facilities" (10 CFR Part 50) (Ref. 1), requires, in part, that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Codes and standards that are used shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires, in part, that reactor coolant pressure boundary components be designed, fabricated, erected, and tested to the highest practical quality standards.

GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 requires, in part, that measures be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

10 CFR 50.55a(c) provides, in part, that reactor coolant pressure boundary components meet the requirements for Class 1 components in Section III ["Rules for Construction of Nuclear Power Plant Components" (Ref. 2)] of the ASME BPVCode [with exceptions].

10 CFR 50.55a(g) provides, in part, that Classes 1, 2, 3, MC, and CC components and their supports meet the requirements of Section XI ["Rules for Inservice Inspection of Nuclear Power Plant Components" (Ref. 4)] of the ASME BPV Code [with exceptions].

Provisions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code have been used since 1971 as one part of a framework to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety. Among other things, ASME standards committees develop improved methods for the construction and inservice inspection (ISI) of ASME Class 1, 2, and 3; metal containment; and concrete containment nuclear power plant components. A broad spectrum of stakeholders, such as manufacturers, utilities, and insurers, participates in the ASME process to help ensure that their interests are considered.

This regulatory guide does not contain any new or amended information collection requirement subject to the Paperwork Reduction Act of 1995 (Ref. 3). This regulatory guide does, however, contain existing information collections that are covered by the requirements of 10 CFR Part 50, which 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

Section III (Ref. 2) and Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" (Ref. 4), of the ASME BPV Code specify certain requirements associated with reactor vessel closure stud bolting. Closure stud bolting is defined to include all studs (stud bolts), nuts, and washers used to fasten the pressure vessel head to the pressure vessel. This regulatory guide provides guidance for use in selecting reactor vessel closure stud bolting materials and properties, conducting a preservice inspection, and conducting an ISI.

The reactor vessel closure head flange is fastened to the reactor vessel shell flange by high-strength, large-diameter bolting (the ASME BPV Code defines "large diameter" as over 4 inches (100 millimeters) in diameter). For high-strength, large-diameter bolting, unusual care must be taken to ensure adequate fracture toughness, and it is important that bolting materials possess adequate toughness throughout the reactor operating cycle. Appropriate metallurgical manufacturing practices can increase fracture toughness, which is measured by energy absorption. Control of the stud bolt tempering procedure is very important for this purpose.

High-strength, low-alloy reactor stud bolting is produced by closely controlling quenching and tempering procedures on grades of steel such as American Iron and Steel Institute 4140 and 4340 (Ref. 5). These steels are approved by ASME as bolting materials and are listed under Section II, “Material Specifications” (Ref. 6), of the ASME BPV Code as SA-540 Grade B-23 and B-24 bar, SA-193 Grade B-7 bar, SA-320 Grade L-43 bar, and SA-194 Grade 7 (nuts for bolting). American Society for Testing and Materials Standard A540/A540M-06, “Standard Specification for Alloy-Steel Bolting Materials for Special Applications,” (Ref. 7) addresses regular and special-quality alloy steel bolting materials, which may be used for nuclear power plant and other special applications. The standard addresses several grades of steel.

Article NB, “Class 1 Components,” paragraph NB-2333, “Bolting Material,” of Section III of the ASME BPV Code addresses bolting material impact testing. Reactor vessel closure studs and nuts should have a Charpy V energy of 61 joules (45 foot-pounds) or greater and a minimum lateral expansion of 0.64 millimeters (25 mils) at a temperature no higher than the preload temperature or the lowest service temperature, whichever is less. The above-mentioned stud materials, when tempered to a maximum tensile strength of 1,172 megapascals (MPa) (170 kilopounds per square inch (ksi)) (Ref. 8), are relatively immune to stress-corrosion cracking (SCC). Above this strength level, the alloy becomes increasingly susceptible to SCC. Therefore, design conservatism should be exercised in determining the sizing of the studs so that the strength level of the material selected will not result in a measured yield strength exceeding 1,034 MPa (150 ksi). The NRC previously established this position on page 13 of NUREG-1339, “Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants” (Ref. 9), issued June 1990, and subsequently adopted it for license renewal as provided in NUREG-1801, “Generic Aging Lessons Learned (GALL) Report” (Ref. 10), Chapter IX, “Selected Definitions and Use of Terms For Structures, Components, Materials, Environments, Aging Effects, and Aging Mechanisms,” issued September 2005 (page IX-9).

Use of martensitic stainless steels, such as those of chromium grades from 11–13 percent, should be avoided for reactor stud bolting applications. These steels (Ref. 11) require more closely controlled conditioning parameters than do carbon and low-alloy steels. Small variations in heat treatment can cause large increases in hardness and tensile strength with a corresponding decrease in corrosion resistance and fracture toughness. The tensile properties of this class of material are extremely sensitive to tempering temperatures in the range of 482–649 degrees Celsius (900–1,200 degrees Fahrenheit) and are somewhat sensitive to variations in the austenitizing treatment. Another shortcoming of this material is temper embrittlement, which may occur during cooling through the temperature range of 399–538 degrees Celsius (750–1,000 degrees Fahrenheit). Certain tempering treatments can also produce alloy compositional gradients within the stud bolt structure, which can greatly shorten service life because they can reduce the strength of the material and its resistance to corrosion. Although martensitic stainless steels are more resistant to general corrosion, they are less resistant to SCC than are carbon or low-alloy steels.

Section III of the ASME BPV Code requires that closure stud bolting be examined before service to ensure that unacceptable bolting is not placed into service. Paragraph NB-2580, “Examination of Bolts, Studs, and Nuts,” requires that the bolting be visually examined to detect harmful discontinuities. In addition, bolts of this size must be examined by either the magnetic particle inspection method or the liquid penetrant method, and they must be ultrasonically examined.

Section XI of the ASME BPV Code specifies provisions for the ISI of closure stud bolting. Specifically, Table IWB-2500-1, “Examination Categories, Examination Category B-G-1, Pressure Retaining Bolting, Greater Than 2 in. (50 mm) in Diameter,” specifies the examination requirements, method, acceptable standard, and extent and frequency of the examination of pressure-retaining bolting greater than 2 inches in diameter. Although properly tempered stud materials are relatively immune to SCC, a need still exists for reliable ISI. Corrosion of studs resulting from leakage of reactor coolant has been reported (Ref. 12). In addition, bolt thread roots are areas of high stress concentration and are preferential sites for crack initiation. Therefore, the bolting material needs to possess adequate toughness so that failure will not initiate in the stud thread configuration. If cracks are initiated, however, it is equally necessary to have an ISI program in place that can readily detect cracks before they reach critical size. In accordance with Examination Category B-G-1, bolting may be examined in place under tension when the connection is disassembled or when the bolting is removed.

The inspection program in Examination Category B-G-1 relies on a combination of volumetric and visual examinations. Reactor vessel closure studs and flange threads must be volumetrically examined, and closure head nuts and closure washers/bushings must be visually examined (VT-1). The volumetric examinations must be conducted in accordance with Mandatory Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to Section XI of the ASME BPV Code. Appendix VIII provides provisions for performance demonstration for ultrasonic examination procedures, equipment, and personnel used to detect and size flaws. Supplement 8, "Qualification Requirements for Bolts and Studs," of Appendix VIII addresses the qualification process for reactor vessel closure studs.

ASME approved Code Case N-307-3, "Ultrasonic Examination of Class 1, Bolting, Table IWB-2500-1, Examination Category B-G-1, Section XI, Division 1" (Ref. 13), on March 28, 2001. This revision to Code Case N-307-3 eliminated the surface examination of the center bore hole of reactor vessel studs because operating experience showed that if cracking occurs, it will be initiated on the outside diameter. The NRC approved the use of Code Case N-307-3 in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 15, issued in October 2007 (Ref. 14).

ASME approved Code Case N-652-1, "Alternative Requirements to Categories B-G-1, B-G-2, and C-D Bolting Examination Methods and Selection Criteria, Section XI, Division 1" (Ref. 15), on February 20, 2004. Because cracking would initiate on the outside diameter of closure studs, Code Case N-652-1 provides an alternative to Examination Category B-G-1 that states that either a surface or volumetric examination is acceptable when the closure bolts are removed for examination. The NRC approved the use of Code Case N-652-1 in Regulatory Guide 1.147, Revision 15.

Suppliers of nuclear steam systems for pressurized-water reactors recommend that the vessel closure stud bolts be removed before raising the water level during refueling or other operations involving vessel head removal. The same suppliers also recommend the use of seal plugs for insertion into the stud holes of the pressure vessel flange to protect against corrosion and contamination following stud removal. Provided that the seal plugs are properly used and that the stud bolting is maintained in an area free from corrosion and contamination, the procedure recommended above adequately protects the stud bolting following head removal. This procedure also permits the performance of an ISI on the bolting when it has been removed from the pressure vessel.

Many different plating materials are available to protect reactor stud bolts and nuts from galling and corrosion. In the past, issues about stud bolt plating have emerged at several plants. Metal-plated bolts can be susceptible to plating fracture in the root of the thread after a short period of bolt and nut engagement. Moisture accumulation near coating discontinuities can cause corrosion. Also, the electrolytic plating process can produce hydrogen, which can become entrapped in the parent metal structure and cause embrittlement. A potential combination of hydrogen in the base metal, natural notches in the bolt thread, moisture in the environment, and high stresses in the material creates an ideal condition for cracking. Metallic coatings are prone to seizing between the bolts and nuts, potentially making disassembly difficult.

Material used for reactor stud bolts and nuts must comply with the requirements of Section III, article NB-2000, “Material,” of the ASME BPV Code. Fracture toughness testing is performed in accordance with paragraph NB-2300, “Fracture Toughness Requirements for Material.” In accordance with paragraph NCA-3855, “Control of Purchased Materials, Source Materials, and Services,” a chemical analysis is required for each heat of material, and testing for mechanical properties is required on samples representing each heat of material and, where applicable, each heat treat lot.

As described in Section 3.13, “Threaded Fasteners—ASME Code Class 1, 2, and 3,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (Ref. 16) issued March 2007, the more detailed criteria in Section III, paragraph NB-2200, “Material Test Coupons and Specimens for Ferritic Steel Material,” of the ASME BPV Code, rather than the material specification of criteria applicable to mechanical testing, should be applied if there is a conflict between the two sets of criteria.

The design of reactor stud bolts and nuts must comply with Section III, article NB-3000, “Design,” of the ASME BPV Code. In addition, Electric Power Research Institute (EPRI) NP-6316, “Guidelines for Threaded-Fastener Application in Nuclear Power Plants,” September 1994, addresses the design of threaded fasteners. The fabrication of reactor stud bolts and nuts must comply with Section III, article NB-4000, “Fabrication and Installation,” of the ASME BPV Code. EPRI NP-6316 also addresses the fabrication of threaded fasteners.

As provided in Section 3.13 of NUREG-0800, lubricants with deliberately added halogens, sulfur, or lead should not be used for any reactor coolant pressure boundary components or other components in contact with reactor water. Lubricants containing molybdenum disulfide (disulfide or polysulfide) should not be used for any safety-related application. For ferritic steel threaded fasteners, conversion coatings, such as those applied using the parkerizing process, are suitable and may be used. Fasteners should not be plated with low-melting-point materials, such as zinc, tin, and cadmium.

C. REGULATORY POSITION

1. Bolting Materials

a. In accordance with Section III of the ASME BPV Code, as incorporated by reference into 10 CFR 50.55a, “Codes and Standards,” reactor vessel closure stud bolting must be fabricated from materials that have adequate toughness throughout the life cycle of the reactor. The staff’s position is that applicants can meet the applicable requirements by following this guidance to ensure that reactor vessel closure stud bolting is designed and tested in an appropriate manner:

   i. The measured yield strength of the stud bolting material should not exceed 1,034 MPa (150 ksi).

   ii. Stud bolting should not be metal-plated unless it has been demonstrated that the plating will not degrade the quality of the stud in any significant way (e.g., corrosion and hydrogen embrittlement) or reduce the quality of results attainable by the various required inspection procedures. The stud bolting may have a manganese phosphate (or other acceptable) surface treatment. Lubricants for the stud bolting are permissible, provided that they are stable at operating temperatures and are compatible with the bolting and vessel materials and with the surrounding environment.

2. Protection against Corrosion

a. As provided in Section 3.13 of NUREG-0800, lubricants with deliberately added halogens, sulfur, or lead should not be used for any reactor coolant pressure boundary components or other components in contact with reactor water. Lubricants containing molybdenum sulfide (disulfide or polysulfide) should not be used for any safety-related applications. Fasteners should not be plated with low melting point materials such as zinc, tin, cadmium, etc.

b. During the venting and filling of the pressure vessel and while the head is removed, the stud bolts and stud bolt holes in the vessel flange should be adequately protected from corrosion and contamination.

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 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.

REFERENCES[1]

1. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission, Washington, DC.

2. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, NY.[2]

3. Paperwork Reduction Act of 1995 (Public Law 104-13), *United States Code*, Title 44, "Public Printing and Documents," Chapter 35, "Coordination of Federal Information Policy" (44 U.S.C. 3501 et seq.), 104th Congress of the United States of America, Washington, DC.

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

5. *Aerospace Structural Metals Handbook*, Code 1206, Volume 1, "Ferrous Alloys," Syracuse University Press, Syracuse, NY, 1963.

6. ASME Boiler and Pressure Vessel Code, Section II, "Material Specifications," American Society of Mechanical Engineers, New York, NY.

7. American Society for Testing and Materials Standard A540/A540M-06, "Standard Specification for Alloy-Steel Bolting Materials for Special Applications," ASTM International, West Conshohocken, PA.

8. Gross, J.H. "The Effective Utilization of Yield Strength," *Transactions of the American Society of Mechanical Engineers,* Paper No. 71, Pressure Vessel and Piping Conference (PVP)-11, Vancouver, British Columbia, Canada, July 23-27, 2006.

9. NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC, June 1990.

10. NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," U.S. Nuclear Regulatory Commission, Washington, DC, September 2005.

11. ASM International, *Metals Handbook*, Volume 2, Materials Park, OH, 1964, pp. 245–248.

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[1] Publicly available NRC published documents such as Regulations, Regulatory Guides, NUREGs, and Generic Letters 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/. Copies are also available for inspection or copying for a fee from 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 (800) 397-4209; fax (301) 415-3548; and e-mail PDR.Resource@nrc.gov.

[2] Copies of the non-NRC documents included in these references may be obtained directly from the publishing organization.

12. Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," U.S. Nuclear Regulatory Commission, Washington, DC, March 17, 1988.

13. ASME Boiler and Pressure Vessel Code, Code Case N-307, "Ultrasonic Examination of Class 1, Bolting, Table IWB-2500-1, Examination Category B-G-1, Section XI, Division 1," American Society of Mechanical Engineers, New York, NY, March 2001.

14. Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 15, U.S. Nuclear Regulatory Commission, Washington, DC, October 2007.

15. ASME Boiler and Pressure Vessel Code, Code Case N-652-1, "Alternative Requirements to Categories B-G-1, B-G-2, and C-D Bolting Examination Methods and Selection Criteria, Section XI, Division 1," American Society of Mechanical Engineers, New York, NY, February 2004.

16. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

17. Electric Power Research Institute NP-6316, "Guidelines for Threaded-Fastener Application in Nuclear Power Plants," Palo Alto, CA, September 1994.