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DRAFT REGULATORY GUIDE

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(Proposed Revision 1 of Regulatory Guide 1.65, dated October 1973)

MATERIALS AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS

A. INTRODUCTION

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

General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary," of the same appendix requires, in part, that "[c]omponents that are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical."

General Design Criterion 31, "Fracture prevention of reactor coolant pressure boundary," requires, in part, that "[t]he reactor coolant pressure boundary shall 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 "[m]easures shall 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."

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, 3, metal containment (MC), and concrete containment (CC) nuclear power plant components. A broad spectrum of stakeholders participate in the ASME process. This helps to ensure that the interests of stakeholders such as manufacturers, utilities, and insurers are considered.

Subsection 10 CFR 50.55a(c) provides, in part, that "[c]omponents which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code ["Rules for Construction of Nuclear Power Plant Components (Ref. 2)"] [with exceptions] . . . ."

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

This regulatory guide does not contain any new or amended information collection requirement subject to the Paperwork Reduction Act of 1995 (Ref. 3). 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 Office of Management and Budget control number.

B. DISCUSSION

Sections III (referenced above) and 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 (large-diameter is defined in the ASME BPV Code 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, 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 (AISI) 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 (ASTM) standard A540/A540M-06, "Standard Specification for Alloy-Steel Bolting Materials for Special Applications," addresses regular and special-quality alloy steel bolting materials which may be used for nuclear power plant and other special applications. This standard addresses several grades of steel.

ASME BPV Code, Section III, Article NB, "Class 1 Components," paragraph NB-2333, "Bolting Material," addresses bolting material impact testing. Reactor vessel closure studs and nuts should have a Charpy V (C_v) energy of 61 joules (45 foot-pounds) or greater and a minimum 0.64 mm (25 mils) of lateral expansion, 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 1172 megapascals (MPa) (170 kilopounds per square inch (ksi)) (Ref. 7), 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 measured yield strength level of the material selected will not result in a measured ultimate tensile strength exceeding 1034 MPa (150 ksi). This position was previously established for license renewal and provided in NUREG/CR-1801, "Generic Aging Lessons Learned." NUREG-1801, Chapter IX, "Selected Definitions and Use of Terms for Structures, Components, Materials, Environments, Aging Effects, and Aging Mechanisms," p. 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. 8) 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 482–649 °C (900–1200 °F) range and are somewhat sensitive to variations in the austenitizing treatment. Another shortcoming of this material is temper embrittlement, which may result during cooling through the 399–538 °C (750–1000 °F) temperature range. Certain tempering treatments can also produce alloy compositional gradients within stud bolt structure, which can greatly shorten service life because of a reduction in strength and 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 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 or liquid penetrant method, and 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. (50mm) in Diameter," specifies the examination requirements, method, acceptance standard, and extent and frequency of examination of pressure-retaining bolting greater than 2 inches in diameter. While 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. 9). In addition, bolt thread roots are areas of high stress concentration and are preferential sites for crack initiation. It is therefore necessary for the bolting material 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. Per Examination Category B-G-1, bolting may be examined in place under tension, when the connection is disassembled, or when the bolting is removed. To ensure detection of any cracking, the provisions of Section XI should be supplemented, as discussed below.

The inspection program in Examination Category B-G-1 relies on a combination of volumetric and visual examinations. Visual examinations may fail to reveal defects, especially if the studs are examined in an untensioned condition (Ref. 10). For these types of examinations, reliability may be increased by examining the studs in a stud-tensioning fixture.

Volumetric examination by conventional ultrasonic techniques (i.e., axial scan ultrasonic method) can be difficult to perform, and the results may be inconclusive because of chamfered ends, the lug wrench recess, and threaded surfaces on the stud bolts. A radial scan technique was developed in which a transducer is lowered into the stud bolt center hole and an ultrasonic radial scan is used for the ultrasonic examination. The radial scan technique increases confidence in the ISI for the determination of flaws in the thread region of the stud bolts. This technique formed the basis for Code Case N-307, "Ultrasonic Examination of Class 1 Bolting, Table IWB-2500-1, Examination Category B-G-1, Section XI, Division 1," which was first published in December 1984 (Ref. 11). Revision 3 of the code case was approved on March 28, 2001 (Supplement 1 to the 2007 Edition of the ASME Boiler and Pressure Vessel Code.

For example, 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 pressure vessel flange stud holes to protect against corrosion and contamination following stud removal. Provided that the seal plugs are properly used, and 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 ISI to be performed 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 (Ref. 12) and corrosion. In the past, issues have emerged regarding stud bolt plating 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 can be 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 ASME BPV Code Section III, article NB-2000. Fracture toughness testing is performed in accordance with ASME BPV Code Section III, paragraph NB-2300. In accordance with ASME BPV Code Section III, paragraph NCA-3855, 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 NUREG-0800, "Standard Review Plan," Section 3.13, "Threaded Fasteners ─ ASME Code Classes 1, 2, and 3," the more detailed criteria of ASME BPV Code, Section III, paragraph NB-2200, rather than the material specification 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 ASME BPV Code Section III, article NB-3000. In addition, Electric Power Research Institute (EPRI) document NP-6316, "Guidelines for Threaded-Fastener Application in Nuclear Power Plants," addresses the design of threaded fasteners. The fabrication of reactor stud bolts and nuts must comply with Section III, article NB-4000. Fabrication of threaded fasteners is also addressed by EPRI NP-6316.

NUREG-0800, Section 3.13, "Threaded Fasteners – ASME Code Class 1, 2, and 3," also provides that 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 to be used for any safety-related application. For ferritic steel threaded fasteners, conversion coatings, such as the parkerizing process, are suitable and may be used. If fasteners are plated, low melting point materials, such as zinc, tin, cadmium, etc., should not be used as plating.

C. REGULATORY POSITION

1. Bolting Materials

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. This requirement should be supplemented by the following to ensure that reactor vessel closure stud bolting is designed and tested in an appropriate manner:

- The measured yield strength of the stud bolting material should not exceed 1034 Mpa (150 ksi).

- Charpy V impact testing should be performed according to ASME SA-370, "Methods and Definitions for Mechanical Testing of Steel Products," and to be acceptable, the results must satisfy the requirements of paragraph NB-2333 of the ASME BPV Code, Section III. In case a test fails, one retest may be conducted in accordance with ASME BPV Code, Section III, paragraph NB-2350, "Retests."

- 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, H_2 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 stud bolting are permissible, provided that they are stable at operating temperatures and are compatible with bolting and vessel materials and the surrounding environment.

2. Protection Against Corrosion

- The criteria in ASME BPV Code Section III, article NB-2200, rather than the material specification criteria applicable to mechanical testing should be applied if there is a conflict between the two sets of criteria.

- As provided in NUREG-0800, Section 3.13, 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 to be used for any safety-related application. Low melting point materials, such as zinc, tin, cadmium, etc., should not be used with plated fasteners.

- During 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 regarding the NRC's plans for using this draft regulatory guide. The NRC does not intend or approve any imposition or backfit in connection with its issuance.

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

REGULATORY ANALYSIS

1. Statement of the Problem

This is proposed Revision 1 to Regulatory Guide (RG) 1.65, originally issued in October 1973. The NRC is reviewing and prioritizing updates to its RGs based on anticipated need and to ensure complete, accurate, and current guidance. The NRC is updating this RG to support new reactor licensing activities. An update is necessary because several of the standards referenced in Revision 0 are either out of date or have been withdrawn by the corresponding standards development organization.

2. Objective

The objective of this regulatory action is to update RG 1.65 for use in combined license applications to ensure that the guidance is current and accurate.

3. Alternative Approaches

The NRC staff considered the following alternative approaches:

- Do not revise RG 1.65.
- Update RG 1.65.

3.1 Alternative 1: Do Not Revise Regulatory Guide 1.65

Under this alternative, the NRC would not revise the guidance, and the original version of this regulatory guide would continue to be used. Since several of the standards referenced are either outdated or have been withdrawn, the staff rejected this alternative.

3.2 Alternative 2: Update Regulatory Guide 1.65

RG 1.65 was originally issued in October, 1973. Under this alternative, the NRC would update this guidance so that it is current and accurate. The staff concludes that the proposed action will enhance reactor safety by providing current and accurate guidance.

The costs to the NRC will be the one-time cost (which is expected to be relatively small) of issuing the revised RG. Applicants would incur little or no cost.

4. Conclusion

Based on this regulatory analysis, the staff recommends that the NRC revise RG 1.65. The staff concludes that the proposed action will enhance reactor safety by providing current and accurate guidance. It could also lead to cost savings for applicants, especially with regard to applications for standard plant design certifications and combined licenses.

REFERENCES

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.

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, Vol. 1, "Ferrous Alloys," Syracuse University Press.

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

7. J.H. Gross, "The Effective Utilization of Yield Strength," Transactions of the ASME, Paper No. 71, Pressure Vessel and Piping Conference (PVP)-11.

8. ASM International, *Metals Handbook*, Vol. 2, pp. 245–248, 1964; 9639 Kinsman Road, Materials Park, OH 44073; phone (440) 338-5151.

9. 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.

10. A.G. Pickett and H.C. Burghard, Jr., "An Analysis of the Failure of Nuclear Reactor Pressure Vessel Closure Studs," Southwest Research Institute SWRI-2154-20, December 1971.

11. 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.

12. R.L Scott and P.H. Harley, "Failure of Threaded Fittings and Fasteners at Nuclear Facilities," Nuclear Safety, Vol. 13, No. 1, Jan.–Feb. 1972.