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# DRAFT REGULATORY GUIDE

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## DRAFT REGULATORY GUIDE DG-1309

*(Proposed Revision 1 of Regulatory Guide 1.207, dated March 2007)*

## GUIDELINES FOR EVALUATING THE EFFECTS OF LIGHT-WATER REACTOR COOLANT ENVIRONMENTS IN FATIGUE ANALYSES OF METAL COMPONENTS

## A. INTRODUCTION

### Purpose

This regulatory guide (RG) describes methods and procedures that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in determining the acceptable fatigue lives of components evaluated by a cumulative usage factor (CUF) calculation in accordance with the fatigue design rules in Section III, "Rules for Construction of Nuclear Power Plant Components," of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereinafter "Code") (Ref. 1), to account for the effects of light-water reactor (LWR) coolant environments.

This guide supports reviews of applications for new nuclear reactor construction permits or operating licenses under *U.S. Code of Federal Regulations*, Title 10, "Energy" (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 2); design certifications under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," (Ref. 3) and combined licenses under 10 CFR Part 52 that do not cite a standard design; and renewed operating licenses under 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (Ref. 4).

### Applicable Rules and Regulations

- General Design Criterion (GDC) 1, "Quality Standards and Records," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 requires, in part, that structures, systems, and components that are important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function performed. In addition, GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires, in part, that components that are part of the reactor-coolant pressure boundary be designed, fabricated, erected, and tested to the highest practical quality standards.

This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received final staff review or approval and does not represent an official NRC final staff position. Public comments are being solicited on this draft guide 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 Docket ID: NRC-2014-0244. Alternatively, comments may be submitted to the Rules, Announcements, and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Comments must be submitted by the date indicated in the *Federal Register* notice.

Electronic copies of this draft regulatory guide, previous versions 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/reg-guides/. The draft 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. ML1417A584. The regulatory analysis may be found in ADAMS under Accession No. ML1417A585.

- Section 55a, “Codes and Standards,” of 10 CFR Part 50 endorses the ASME Code for design of safety-related systems and components. Section 50.55a(c), “Reactor Coolant Pressure Boundary,” requires, in part, that components of the reactor-coolant pressure boundary meet the requirements for Class 1 components in Section III of the ASME Code, except as provided in that section. Specifically, the ASME Class 1 requirements contain provisions, including fatigue design curves, for determining a component’s suitability for cyclic service. These provisions are used for those components that are exposed to reactor coolant and are required by regulation to have a fatigue CUF calculation or have an existing current licensing basis (CLB) fatigue CUF calculation.

**Related Guidance**

- U.S. Nuclear Regulatory Commission (NRC), “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” NUREG-1800 (Ref. 5).

- U.S. NRC, “Generic Aging Lessons Learned (GALL) Report,” NUREG-1801 (Ref. 6).

**Purpose of Regulatory Guides**

The 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 NRC 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. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if such alternatives provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

**Paperwork Reduction Act**

This regulatory guide contains information collections that are covered by the requirements of 10 CFR Parts 50, 52, and 54, which the Office of Management and Budget (OMB) approved under OMB control numbers 3150-0011, 3150-0151, and 3150-0155, respectively. 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

**Reason for Revision**

This revision of the RG (Revision 1) consolidates, updates, and replaces previous NRC staff guidance on the effects of LWR coolant environments on the fatigue lives of nuclear power-plant components. This revision provides an alternative to previous guidance for new reactors provided in Revision 0 of this guide, as well as previous guidance for license renewal of operating reactors in the GALL Report and the Standard Review Plan for License Renewal (SRP-LR). The guidance in Revision 0 of this RG, the GALL Report, and the SRP-LR was developed in response to the closeout of Generic Safety Issue (GSI) 190, “Fatigue Evaluation of Metal Components for 60-Year Plant Life” (Ref. 7), which documented the NRC staff’s selection of the $F_{en}$ method as an acceptable method to properly incorporate LWR environmental effects into fatigue CUF calculations for ASME Code components.

## Background

This RG provides guidance for use in determining the acceptable fatigue lives of components evaluated by a CUF calculation in accordance with the fatigue design rules in Section III of the ASME Code, with consideration of LWR coolant environments. For operating reactors, the NRC staff's closeout of GSI 190 identified that, because of significant conservatism in quantifying other plant-related variables involved in CUF calculations (such as cyclic behavior, including stress and loading rates), the 40-year design of the current fleet of reactors was satisfactory. The NRC staff concluded that there was no basis for a cost/benefit backfit analysis to justify imposition of a new regulatory requirement on operating reactors. However, the calculations that supported the resolution of this issue indicated the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, with the consideration of a risk-informed perspective, the staff concluded that applicants for renewed licenses should address the effects of LWR coolant environments on component fatigue lives as they develop aging-management programs for license renewal, in ways consistent with existing requirements in 10 CFR 54.21, "Contents of Application—Technical Information." These findings were captured in the initial versions of the GALL Report and the SRP-LR as guidance applicable to plants pursuing license renewal. The NRC staff's closeout of GSI 190 and the associated research also led to the development of Revision 0 of RG 1.207 (Ref. 8) to document guidance related to evaluation of environmental effects for new reactors. The guidance provided in Revision 1 of RG 1.207 is a consolidation and update of the staff's previous guidance and is limited to new reactors, operating reactors pursuing license renewal, and plants where evaluation of LWR coolant environments already forms a part of the CLB.

In the late 1960s and early 1970s, ASME developed design fatigue curves for Section III of the ASME Code based on tests conducted in laboratory air environments at room temperature. The original ASME Code developers applied a margin of two on strain (or stress) and a margin of twenty on cyclic life to develop design fatigue curves that accounted for variations in materials, size, surface finish, data scatter, and environmental effects (including temperature differences between specimen test conditions and reactor operating experience). However, the developers lacked sufficient data to explicitly evaluate and account for fatigue life degradation attributable to component exposure to aqueous coolants. Reflecting this circumstance, Paragraph NB-3121, "Corrosion," in Section III of the ASME Code states that the design fatigue curves did not include tests in the presence of corrosive environments that might accelerate fatigue failure. Paragraph NB-3121 further states that provisions for the presence of corrosive environments that might accelerate fatigue failure shall be included in the design or specified life of components. More recent fatigue-test data from the United States (including the results of NRC research activities), Japan, and elsewhere show that LWR coolant environments can have a significant impact on the fatigue lives of components made from carbon, low-alloy, austenitic stainless (both wrought and cast), and nickel-chromium-iron (Ni-Cr-Fe) alloy steels and welds.

In the 1990s, as a part of NRC research activities on fatigue, the staff evaluated two distinct methods for incorporating LWR environmental effects into the fatigue analysis of ASME Code, Class 1 components. The first method involved developing new fatigue curves that were applicable to LWR environments. Given that the fatigue life of ASME Code, Class 1 components in LWR coolant environments is a function of several parameters, this method necessitated the development of several fatigue curves to address potential parameter variations. Alternatively, a single *bounding* fatigue curve could be developed, but this approach might be overly conservative for most applications. The second method involved using an environmental factor ($F_{en}$) to adjust the CUF calculated with the design fatigue curves in Section III of the ASME Code to account for LWR coolant environments. The second method affords the designer greater flexibility to calculate the appropriate impacts for specific environmental parameters. Based on the results of the NRC's efforts, the staff elected to develop guidance that used the $F_{en}$ method.

In developing the underlying $F_{en}$ models, researchers from Argonne National Laboratory (ANL) analyzed existing laboratory data to predict fatigue lives as a function of temperature, strain rate, dissolved oxygen level in water, and sulfur content of the steel. ANL defined an environmental factor, $F_{en}$, as the ratio of the component fatigue life in a room-temperature air environment to its fatigue life in an LWR coolant environment at operating temperature. The resultant $F_{en}$ method also postulated a strain threshold below which environmental effects on fatigue lives did not occur. Calculating CUF using the provisions set forth in Section III of the ASME Code and multiplying the CUF by $F_{en}$ provided a means of incorporating the environmental effects identified in Paragraph NB-3121 of Section III of the ASME Code when warranted.

The NRC staff initially published the $F_{en}$ method in NUREG/CR-6583 for carbon and low-alloy steel materials and NUREG/CR-5704 for austenitic stainless steel materials. In 2001, the NRC staff endorsed the $F_{en}$ methods in NUREG/CR-6583 and NUREG/CR-5704 for use by licensees pursuing license renewal in the initial versions of the GALL Report and the SRP-LR. Additional data evaluation subsequent to the publication of these two documents resulted in a revised $F_{en}$ method for new reactors that the staff documented in Revision 0 of NUREG/CR-6909. The staff published guidance for the revised $F_{en}$ methodology for new reactors in 2007 in Revision 0 of RG 1.207. The staff published Revision 0 of NUREG/CR-6909 as an alternative to NUREG/CR-6583 and NUREG/CR-5704 for use by licensees pursuing license renewal in 2010 as documented in Revision 2 of the GALL Report and the SRP-LR. Because the relationships in NUREG/CR-6583 and NUREG/CR-5704 were generally considered to be conservative compared to the revised expressions in Revision 0 of NUREG/CR-6909, either set of relationships was endorsed for use by license-renewal applicants in Revision 2 of the GALL Report and the SRP-LR.

As part of further NRC-funded research efforts performed since the publication of Revision 0 of RG 1.207, the NRC staff evaluated additional available fatigue data (primarily from Japan), incorporated relevant data in their previously developed database, and updated the fatigue life models. The NRC staff also evaluated and incorporated relevant feedback from interested stakeholders based on the use of the $F_{en}$ methods published in NUREG/CR-6583, NUREG/CR-5704, and Revision 0 of NUREG/CR-6909.

The results of the NRC staff's most recent research efforts on this topic are documented in Revision 1 of NUREG/CR-6909 (Ref. 9). Those results identified the need to revise and consolidate the previously published guidance for incorporating the effects of LWR coolant environments in fatigue life evaluations. Revision 1 of RG 1.207 maintains the previously endorsed methods for establishing fatigue design curves and defines updated $F_{en}$ factors for use in evaluating the fatigue lives of reactor components exposed to LWR coolant environments.

In Revision 0 of RG 1.207, the NRC staff identified a non-conservatism in the ASME Code fatigue design curve with respect to existing fatigue data for austenitic stainless steels and endorsed a separate stainless steel fatigue design curve, as documented in Revision 0 of NUREG/CR-6909. ASME modified the fatigue design curve in Section III for austenitic stainless steels in the 2009b Addenda to adopt the fatigue design curve developed in Revision 0 of NUREG/CR-6909. Section 3.2.11 of Revision 1 of NUREG/CR-6909 provides an updated and comprehensive review of, and technical basis for, continued use of the stainless steel fatigue design curve previously developed in Revision 0 of NUREG/CR-6909. The $F_{en}$ defined for stainless steel in Revision 1 of NUREG/CR-6909 should be used in conjunction with this more recent stainless steel fatigue design curve when evaluating the CUF of ASME components that require a CUF calculation. Use of the austenitic stainless steel design curve also applies to the fatigue analyses for cast austenitic stainless steels, Ni-Cr-Fe alloys (e.g., Alloys 600, 690, 718, and 800) and their associated weld metals, as described in Section 4.3 of Revision 1 of NUREG/CR-6909.

Section 5 of Revision 1 of NUREG/CR-6909 evaluates margins in the ASME Code fatigue design curves. In conducting that evaluation, the ANL researchers reviewed data available in the literature to assess the subfactors (excluding environment) that are necessary to account for the effects of various uncertainties and differences between actual components and laboratory test specimens. The ANL researchers also performed statistical analyses using Monte Carlo simulations to develop fatigue design curves using the "95/95 criterion." In other words, the curves should provide 95% confidence that the fatigue life of 95% of the population of laboratory test specimens will be greater than that predicted by the design curves. The NRC staff deems this criterion acceptable because the NRC staff bases the fatigue design curves on crack initiation, rather than component failure or through-wall crack leakage, and, therefore, additional margin exists between crack initiation and actual component failure or leakage.

The results of the Monte Carlo simulations indicated that for carbon, low-alloy, austenitic stainless, and Ni-Cr-Fe alloy steels, the current ASME Code procedure of adjusting the mean test data by a factor of twenty for cyclic life is conservative compared to the 95/95 criterion. The NRC staff's results indicated that a factor of ten to twelve for cyclic life was sufficient to satisfy the 95/95 criterion for these materials. Figures 36, 37, and 49 of Revision 1 of NUREG/CR-6909 present the resulting new fatigue design curves using margins of twelve for cyclic life and two for strain (whichever is more conservative) for carbon steels, low-alloy steels, and austenitic stainless steels, respectively. This guide uses these new air design curves; thus, an applicant or licensee that chooses to adopt the procedure discussed in this guide to determine the fatigue lives of carbon, low-alloy, austenitic stainless, and Ni-Cr-Fe alloy steels should use the air design curves in Revision 1 of NUREG/CR-6909. The fatigue design curves for carbon and low-alloy steels and austenitic stainless steels in Section III of the 2013 Edition of the ASME Code may also be used with the procedure in this guide to determine the fatigue lives of those materials, because their use will yield the same or more conservative results.

$F_{en}$ calculations for carbon, low-alloy, austenitic stainless, and Ni-Cr-Fe alloy steels need only consider the types of stress cycles or load set pairs that exceed the strain threshold criteria. The evaluation options depend on the complexity of the analyzed transient conditions and the details of the evaluation. For example, in an evaluation in which the results of detailed transient analyses are available to determine the necessary parameters (strain rate, temperature, and others), the "modified rate approach" (presented and cited in Section 4.1.14 of Revision 1 of NUREG/CR-6909) is an acceptable method for determining the $F_{en}$ values. This method involves a strain-based integral to evaluate conditions for which temperature and strain rate change, resulting in the variation of $F_{en}$ over time. This detailed approach calculates the $F_{en}$ values based on the strain history for each load set in the fatigue analysis evaluation, considering the effects of strain rate and temperature variations for each incremental segment in the strain history. Such results may be used to reduce the conservatism in the calculated $F_{en}$ values. For a simplified calculation yielding more conservative results for complex or poorly defined transients, the strain rate is equal to the average strain rate in the transient or segment, and the temperature is equal to the maximum temperature in the transient or segment. For such simplification, care should be taken in the selection of temperature, as discussed in Section 4.1.14 of Revision 1 of NUREG/CR-6909.

Appendix C to Revision 1 of NUREG/CR-6909 provides a sample problem showing one method of calculating and applying $F_{en}$ to a CUF calculation.

**Harmonization with International Standards**

The NRC staff searched for available guidance from the International Atomic Energy Agency (IAEA) and the International Organization for Standardization (ISO), and did not identify any standards that provided additional guidance to NRC staff, applicants, or licensees.

**Documents Discussed in Staff Regulatory Guidance**

This regulatory guide endorses, in part, the use of one or more codes or standards developed by external organizations and other third-party guidance documents. These codes, standards and third-party guidance documents may contain references to other codes, standards or third-party guidance documents ("secondary references"). If a secondary reference has itself been incorporated by reference in NRC regulations as a requirement, 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, 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 in NRC regulations nor endorsed in a regulatory guide, the secondary reference is neither a legally binding requirement nor a "generic" NRC-approved acceptable approach for meeting an NRC requirement. However, licensees and applicants may consider and use the information in the secondary reference if it is appropriately justified, consistent with current regulatory practice, and consistent with applicable NRC requirements.

# C. REGULATORY POSITION

This section describes the methods that the NRC staff considers acceptable for use in performing fatigue evaluations that consider the effects of LWR coolant environments on carbon and low-alloy steels, wrought and cast austenitic stainless steels, Ni-Cr-Fe alloys, and their associated weld metals. Specifically, these methods include calculating the CUF in air using ASME Code, Section III analysis procedures, and then employing the environmental factor ($F_{en}$), as described in NUREG/CR-6909, Revision 1. In particular, Appendix A to that report includes detailed descriptions and additional guidance concerning the overall method and all the required calculations. These methods apply to those components exposed to reactor coolant that are required by regulation to have a fatigue CUF evaluation or have an existing CLB fatigue CUF calculation.

## 1. Carbon and Low-Alloy Steels and Welds

Calculate the environmental CUF for carbon and low-alloy steel components and their associated weld metals exposed to LWR coolant environments using the procedures in Sections 1.1 through 1.3.

### 1.1. <u>CUF in Air</u>

Calculate the CUF in air using ASME Code, Section III analysis procedures (Subarticle NB-3200, "Design by Analysis," or Subarticle NB-3600, "Piping Design") and the fatigue air curves (updated ANL model curves) provided in Figures A.1 and A.2 and Table A.1 in Appendix A to NUREG/CR-6909, Revision 1. Alternatively, the fatigue design curve for carbon and low-alloy steel in Appendix I to Section III of the 2013 Edition of the ASME Code may be used.

### 1.2. <u>Environmental Factor ($F_{en}$)</u>

Calculate the environmental factor $F_{en}$ for carbon and low-alloy steels and associated welds using Equation A.2 in NUREG/CR-6909, Revision 1. In the same document, Equations A.3 through A.6 should be used to calculate the parameters used in Equation A.2; Equation A.7 defines the strain threshold.

### 1.3. Environmental CUF

Calculate the environmental CUF using Equation A.19 in NUREG/CR-6909, Revision 1.

## 2. Wrought and Cast Austenitic Stainless Steels and Welds

Calculate the environmental CUF for wrought and cast austenitic stainless steel components and their associated weld metals exposed to LWR coolant environments using the procedures in Sections 2.1 through 2.3.

### 2.1. CUF in Air

Calculate the CUF in air using ASME Code, Section III analysis procedures (Subarticle NB-3200, "Design by Analysis," or Subarticle NB-3600, "Piping Design") and the stainless steel fatigue air curve (proposed design curve) provided in Figure A.3 and Table A.2 in Appendix A to NUREG/CR-6909, Revision 1. Alternatively, the fatigue design curve for austenitic stainless steel in Section III of the 2013 Edition of the ASME Code may be used.

### 2.2. Environmental Factor ($F_{en}$)

Calculate $F_{en}$ using Equation A.8 in NUREG/CR-6909, Revision 1. In the same document, Equations A.9 through A.11 should be used to calculate the parameters used in Equation A.8; Equation A.12 defines the strain threshold.

### 2.3. Environmental CUF

Calculate the environmental CUF using Equation A.19 in NUREG/CR-6909, Revision 1.

## 3. Ni-Cr-Fe Alloys and Welds

Calculate the environmental CUF for Ni-Cr-Fe alloy components (e.g., Alloys 600, 690, 718, and 800) and their associated weld metals exposed to LWR coolant environments using the procedures in Sections 3.1 through 3.3.

### 3.1. CUF in Air

Calculate the CUF in air using ASME Code, Section III analysis procedures (Subarticle NB-3200, "Design by Analysis," or Subarticle NB-3600, "Piping Design") and the stainless steel fatigue air curve (proposed design curve) provided in Figure A.3 and Table A.2 in Appendix A to NUREG/CR-6909, Revision 1. Alternatively, the fatigue design curve for Ni-Cr-Fe alloys in Section III of the 2013 Edition of the ASME Code may be used.

### 3.2. Environmental Factor ($F_{en}$)

Calculate $F_{en}$ using Equation A.13 in NUREG/CR-6909, Revision 1. In the same document, Equations A.14 through A.16 should be used to calculate the parameters used in Equation A.13; Equation A.17 defines the strain threshold.

### 3.3. Environmental CUF

Calculate the environmental CUF using Equation A.19 in NUREG/CR-6909, Revision 1.

# D. IMPLEMENTATION

The purpose of this section is to provide information on how applicants and licensees[1] may use this guide and information regarding the NRC's plans for using this 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.

**Use by Applicants and Licensees**

Applicants and licensees may voluntarily[2] 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 sufficient basis and information is provided 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 licensing basis remains unchanged.

Licensees may use the information in this regulatory guide for actions that 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 NRC staff does not intend or approve any imposition or 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 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.

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1 In this section, "licensees" refers to licensees of nuclear power plants under 10 CFR Parts 50 and 52 and "applicants" refers to applicants for licenses and permits for (or relating to) nuclear power plants under 10 CFR Parts 50, 52, and 54, as well as applicants for standard design approvals and standard design certifications under 10 CFR Part 52.

2 In this section, "voluntary" and "voluntarily" mean 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.

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, 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 defined in 10 CFR 50.109(a)(1) or a violation of any of the issue-finality provisions in 10 CFR Part 52.

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

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 in a manner inconsistent with the discussion in this Implementation section, the licensee may file a backfit appeal with the NRC in accordance with the guidance in NUREG-1409, "Backfitting Guidelines" (Ref. 10), and NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 11).

# REFERENCES


1. ASME, *Boiler and Pressure Vessel Code*, 2013 edition, Section III, "Rules for Construction of Nuclear Power Plant Components," New York, NY.[3]

2. *U.S. Code of Federal Regulations* (CFR), "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy."[4]

3. CFR, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter I, Title 10, "Energy."

4. CFR, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Part 54, Chapter I, Title 10, "Energy."

5. U.S. Nuclear Regulatory Commission (NRC), "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," NUREG-1800.

6. NRC, "Generic Aging Lessons Learned (GALL) Report," NUREG-1801, Revision 2, December 2010, Agencywide Documents Access and Management System (ADAMS) Accession No. ML103490041.

7. Thadani, Ashok C., Director of the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, memorandum to William D. Travers, Executive Director for Operations, U.S. Nuclear Regulatory Commission, "Closeout of Generic Safety Issue 190, 'Fatigue Evaluation of Metal Components for 60-Year Plant Life'," August 26, 1999, ADAMS Accession No. ML003673136.

8. NRC, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors," Regulatory Guide 1.207, Revision 0, March 2007.

9. NRC , "Effect of LWR Coolant Environments on Fatigue Life of Reactor Materials," NUREG/CR-6909 (ANL-12/60), Revision 1 (DRAFT), March 2014, ADAMS Accession No. ML14087A068.

10. NRC, "Backfitting Guidelines," NUREG-1409, July 1990, ADAMS Accession No. ML032230247.

11. NRC, "Management of Facility-Specific Backfitting and Information Collection," Management Directive 8.4, October 9, 2013, ADAMS Accession No. ML12059A460.


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3 Copies may be purchased from ASME, Three Park Avenue, New York, NY 10016-5990; phone 212-591-8500; fax 212-591-8501; www.asme.org.

4 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at: http://www.nrc.gov/reading-rm/doc-collections/. The documents can also be viewed online 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 800-397-4209; fax 301-415-3548; and e-mail pdr.resource@nrc.gov.