

U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 3.71, REVISION 3



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NUCLEAR CRITICALITY SAFETY STANDARDS FOR NUCLEAR MATERIALS OUTSIDE REACTOR CORES

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes methods that the U.S. Nuclear Regulatory Commission (NRC) considers acceptable in criticality safety standards associated with nuclear materials outside reactor cores. The standards describe procedures for preventing nuclear criticality accidents in operations that involve handling, processing, storing, or transporting special nuclear materials (or a combination of these activities).

Applicability

This RG applies to applicants, licensees, and certificate holders authorized under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 70, “Domestic Licensing of Special Nuclear Material” (Ref. 1); 10 CFR Part 71, “Packaging and Transportation of Radioactive Material” (Ref. 2); and 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste” (Ref. 3).

This revision is not intended for use by nuclear reactor licensees. The staff has not evaluated the applicability to reactor facilities licensed under 10 CFR Part 50 because these facilities have cores that are designed to operate at criticality, and the facilities have other regulatory requirements that address criticality safety outside the reactor core.

Applicable Regulations

- 10 CFR 70.20, “General license to own special nuclear material,” requires a specific license to acquire, deliver, receive, possess, use, transfer, import, or export special nuclear material.
- 10 CFR 70.22(a)(8), “Contents of applications,” requires applications for such licenses to include proposed procedures to avoid nuclear criticality accidents.

Written suggestions regarding this guide or development of new guides may be submitted through the NRC’s public Web site in the NRC Library at <https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/>, under Document Collections, in Regulatory Guides, at <https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/contactus.html>.

Electronic copies of this RG, previous versions of RGs, and other recently issued guides are also available through the NRC’s public Web site in the NRC Library at <https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/>, under Document Collections, in Regulatory Guides. This RG is also available through the NRC’s Agencywide Documents Access and Management System (ADAMS) at <http://www.nrc.gov/reading-rm/adams.html>, under ADAMS Accession Number (No.) ML18169A258. The regulatory analysis may be found in ADAMS under Accession No. ML17055B588. The associated draft guide DG-3053 may be found in ADAMS under Accession No. ML17055B591, and the staff responses to the public comments on DG-3053 may be found under ADAMS Accession No. ML18169A253.

- 10 CFR 71.3, “Requirement for license,” requires a specific licensee to deliver licensed material to a carrier for transport or to transport licensed material, which must include compliance with the criticality requirements of 10 CFR Section 71.55, “General requirements for fissile material packages.”
- 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste,” requires a general or specific license for an independent spent fuel storage installation (ISFSI) or a certificate of compliance for a spent fuel storage cask and requires the facility or cask design to meet the regulations in 10 CFR 72.124, “Criteria for nuclear criticality safety.”

Related Guidance

- NUREG-1520, “Standard Review Plan for Fuel Cycle Facilities License Applications” (Ref. 4), provides guidance on the licensing of nuclear fuel cycle facilities under 10 CFR Part 70.
- NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility” (Ref. 5), provides guidance for the licensing of a MOX fuel fabrication facility under 10 CFR Part 70.
- NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems” (Ref. 6), provides guidance for the review of a certificate of compliance for a dry storage system at a general license facility under 10 CFR Part 72.
- NUREG-1567, “Standard Review Plan for Spent Dry Fuel Storage Facilities” (Ref. 7), provides guidance for the licensing of an ISFSI under 10 CFR Part 72.
- NUREG-1617, “Standard Review Plan for Transportation Packages for MOX Spent Nuclear Fuel,” Supplement 1 (Ref. 8), provides guidance for the review of a certificate of compliance for a transportation package under 10 CFR Part 71.
- NUREG-1927, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel” (Ref. 9), provides guidance for renewing the license of an ISFSI and review of a renewed certificate of compliance for a dry storage system under 10 CFR Part 72.
- NUREG/CR-7108, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions” (Ref. 10), provides guidance for the review of the application of burnup credit for a spent fuel transport or storage system under 10 CFR Part 71 and 10 CFR Part 72.
- NUREG/CR-7109, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions” (Ref. 11), provides guidance for the review of the application of burnup credit for a spent fuel transport or storage system under 10 CFR Part 71 and 10 CFR Part 72.

Purpose of Regulatory Guides

The NRC issues RGs to describe to the public methods that the 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 events, and to provide guidance to applicants. RGs are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG provides guidance for implementing the mandatory information collections in 10 CFR Parts 70, 71, and 72 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), under control numbers 3150-0009, 3150-0008, and 3150-0132. Send comments regarding this information collection to the Information Services Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011, 3150-0151), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

B. DISCUSSION

Reason for Revision

This revision of the guide (Revision 3) endorses the most recent American National Standards Institute (ANSI)-approved versions of American Nuclear Society Subcommittee-8 (ANS-8) standards listed in Staff Regulatory Guidance Position C.1 of this guide. In addition, the scope of this guide is expanded beyond 10 CFR Part 70 fuel facilities to include packaging and transportation under 10 CFR Part 71 and storage facilities under 10 CFR Part 72, because many of the standards are based on broad principles that are not limited solely to fuel processing facilities. The NRC staff has also endorsed International Organization for Standardization (ISO) Standard 7753:1987, “Nuclear Energy—Performance and Testing Requirements for Criticality Detection and Alarm Systems” (Ref. 12), in this revision.

Background

The NRC initially issued RG 3.71 in 1998. The RG consolidated and replaced a number of earlier NRC RGs, thereby incorporating all of the relevant guidance at that time into a single document. RG 3.71 was revised in 2005 and again in 2010. These three previous versions of RG 3.71 endorsed specific safety standards developed by ANS to provide guidance, criteria, and best practices for use in preventing and mitigating criticality accidents during operations that involve handling, processing, storing, or transporting special nuclear material at fuel and material facilities (or a combination of these activities).

The ANS Nuclear Criticality Safety Consensus Committee (formerly N16) developed the ANSI/ANS-8 standards that are identified in Staff Regulatory Guidance Position C.1 of this RG. Each ANSI/ANS-8 standard has been developed by a working group of expert practitioners in this area and is reviewed every 5 to 10 years to ensure that the standard can be revised, reaffirmed, or withdrawn, as appropriate, to reflect up-to-date information. New standards are also developed when the need arises.

Consequently, the NRC staff has updated this guide to provide guidance on changes that have occurred since the last revision of RG 3.71 in 2010, and endorses several ANSI/ANS-8 nuclear criticality safety standards that have been developed, reaffirmed, or revised. Because the ANSI/ANS-8 and ISO standards are frequently updated, the NRC may revise this guide to endorse the updated standards, if warranted.

The ANSI/ANS-8 nuclear criticality safety standards and those developed by ISO provide criteria and practices that the NRC staff considers generally acceptable for use in preventing and mitigating nuclear criticality accidents. However, use of the nuclear criticality safety standards is not a substitute for detailed nuclear criticality safety analyses for specific operations.

Harmonization with International Standards

The NRC has a goal of harmonizing its guidance with international standards to the extent practical. The International Atomic Energy Agency (IAEA) and ISO have established a series of safety guides and standards that address good practices for nuclear criticality safety at nuclear fuel cycle facilities and spent fuel transportation and storage. These documents include:

- IAEA Safety Standard No. SF-1, “Fundamental Safety Principles” (Ref. 13); IAEA Safety Requirement NS-R-5, “Safety of Nuclear Fuel Cycle Facilities” (Ref. 14);

- IAEA Specific Safety Guide (SSG)-27, “Criticality Safety in the Handling of Fissile Material” (Ref. 15); and
- IAEA SSG-15, “Storage of Spent Nuclear Fuel” (Ref. 16). This RG incorporates similar concepts and is consistent with the basic safety principles provided in these IAEA guides, as well as the ISO standards listed below.

The ISO Technical Committee 85, Subcommittee 5, Working Group 8 (TC85/SC5/WG8) on nuclear criticality safety produced the following additional valuable criticality standards:

- ISO 1709:1995, “Nuclear Energy—Fissile Materials—Principles of Criticality Safety in Storing, Handling, and Processing” (Ref. 17);
- ISO 7753:1987, “Nuclear Energy—Performance and Testing Requirements for Criticality Detection and Alarm Systems;”
- ISO 11311:2011, “Nuclear Criticality Safety—Critical Values for Homogeneous Plutonium-Uranium Oxide Fuel Mixtures Outside of Reactors” (Ref. 18);
- ISO 11320:2011, “Nuclear Criticality Safety—Emergency Preparedness and Response” (Ref. 19);
- ISO 14943:2004, “Nuclear Fuel Technology—Administrative Criteria Related to Nuclear Criticality Safety” (Ref. 20);
- ISO 16117:2013, “Nuclear Criticality Safety—Estimation of the Number of Fissions of a Postulated Criticality Accident” (Ref. 21);
- ISO 27467:2009, “Nuclear Criticality Safety—Analysis of a Postulated Criticality Accident” (Ref. 22); and
- ISO 27468:2011, “Nuclear Criticality Safety—Evaluation of Systems Containing PWR [pressurized water reactor] UOX [uranium oxide (e.g., uranium dioxide, uranium trioxide, triuranium octoxide)] Fuels—Bounding Burnup Credit Approach” (Ref. 23).

Documents Discussed in Staff Regulatory Guidance

This RG endorses 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 other third party guidance documents (“secondary references”). If a secondary reference has itself been incorporated by reference into an NRC regulation as a requirement, then licensees and applicants must comply with that standard as set forth in the regulation. If the secondary reference has been endorsed in an RG as an acceptable approach for meeting an NRC requirement, then the standard constitutes a method acceptable to the NRC staff for meeting that regulatory requirement as described in the specific RG. If the secondary reference has neither been incorporated by reference into NRC regulations nor endorsed in an RG, then 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 appropriately justified, consistent with current regulatory practice, and consistent with applicable NRC requirements.

C. STAFF REGULATORY GUIDANCE

This section endorses, or endorses with clarifications or exceptions, standards that describe methods, approaches, or data that the staff considers acceptable for meeting the requirements of the regulations cited in the Introduction to this guide. As used within the body of these standards, the term “shall” in a standard denotes a requirement of the standard, the word “should” denotes a recommendation, and the word “may” denotes permission (neither a requirement nor a recommendation). When a licensee or applicant commits to a standard cited in this RG in full, the licensee or applicant commits to perform all operations in accordance with the requirements of that standard, but not necessarily with the standard’s recommendations. Applicants, licensees, or certificate holders may follow the recommendations given in the standards and could use a combination of standards (e.g., instead of ANSI/ANS-8.3 could use ISO standard 7753:1987 as an alternative), unless an exception is stated in this RG, or may use other acceptable methods.

1. Nuclear Criticality Standards Endorsed by the NRC

The NRC endorses the following ANSI/ANS-8 and ISO nuclear criticality safety standards without exception:

- a. ANSI/ANS-8.5-1996 (Reaffirmed 2017), “Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material” (Ref. 24);
- b. ANSI/ANS-8.6-1983 (Reaffirmed 2017), “Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ” (Ref. 25);
- c. ANSI/ANS-8.7-1998 (Reaffirmed 2017), “Nuclear Criticality Safety in the Storage of Fissile Materials” (Ref. 26);
- d. ANSI/ANS-8.12-1987 (Reaffirmed 2016), “Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors” (Ref. 27);
- e. ANSI/ANS-8.14-2004 (Reaffirmed 2016), “Use of Soluble Neutron Absorbers in Nuclear Facilities Outside Reactors” (Ref. 28);
- f. ANSI/ANS-8.15-2014, “Nuclear Criticality Control of Special Actinide Elements,” (Ref. 29);
- g. ANSI/ANS-8.19-2014, “Administrative Practices for Nuclear Criticality Safety,” (Ref. 30);
- h. ANSI/ANS-8.20-1991 (Reaffirmed 2015), “Nuclear Criticality Safety Training” (Ref. 31);
- i. ANSI/ANS-8.21-1995 (Reaffirmed 2011), “Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors” (Ref. 32);
- j. ANSI/ANS-8.22-1997 (Reaffirmed 2016), “Nuclear Criticality Safety Based on Limiting and Controlling Moderators” (Ref. 33); and

- k. ANSI/ANS-8.26-2007 (Reaffirmed 2016), “Criticality Safety Engineer Training and Qualification Program” (Ref. 34).

2. Nuclear Criticality Standards Endorsed by the NRC with Clarifications or Exceptions

The NRC endorses the following ANSI/ANS-8 and ISO nuclear criticality safety standards with the following clarifications or exceptions:

- a. **ANSI/ANS-8.1-2014, “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors” (Ref. 35)**

Table 1 of the standard contains a single-parameter fissile concentration limit for $^{239}\text{Pu}(\text{NO}_3)_4$. Section 5.1 of the standard states that ^{239}Pu subcritical limits apply to mixtures of plutonium isotopes, which are always present, under certain conditions. A clarification is that licensees or applicants wishing to use a fissile concentration limit for $\text{Pu}(\text{NO}_3)_4$ should ensure that the mixture of fissile and non-fissile (e.g., ^{240}Pu) plutonium isotopes is conservative relative to the limit with sufficient subcritical margin, or should calculate it using an approved calculational method. As discussed in Section 5 of the standard, this should include sufficient margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.

- b. **ANSI/ANS-8.3-1997 (Reaffirmed 2017), “Criticality Accident Alarm System” (Ref. 36)**

Section 4.2.1 of the standard requires an evaluation of the need for a criticality alarm system in each area where threshold quantities of special nuclear material are handled, used, or stored. An exception is that 10 CFR Section 70.24, “Criticality Accident Requirements,” takes precedence and requires a criticality alarm system in each area where threshold quantities of special nuclear material are handled, used, or stored.

Section 4.2.2 of the standard states that a criticality alarm system is not required in areas where personnel would not be subject to an excessive radiation dose (defined as greater than 0.12 gray (Gy) (12 rad) in free air). A clarification is that 10 CFR Section 70.24 requires placement of detectors in areas where threshold quantities of special nuclear material are present, but that audible or visual alarms may be located in areas where immediate evacuation is determined to be necessary based on the potential for an excessive dose.

Section 4.4.1 of the standard permits coverage by a single reliable detector for each area monitored. An exception is that 10 CFR Section 70.24 takes precedence and requires that two criticality detectors cover each monitored area.

Section 5.5 of the standard requires that the system be designed to produce a criticality alarm signal within 0.5 seconds of detector recognition of a criticality accident. A clarification is that the specific timing is not important if the delay between detection and alarm is effectively instantaneous relative to the time it takes a person to respond and evacuate.

Section 5.6 of the standard states that the minimum accident of concern may be assumed to deliver an absorbed dose in free air of 0.20 Gy (20 rad) in 1 minute at 2 meters from the reacting material, or otherwise justified. An exception is that 10 CFR Section 70.24

requires that a monitoring system be capable of detecting a nuclear criticality that produces an absorbed dose in soft tissue of 0.20 Gy (20 rad) of combined neutron and gamma radiation in 1 minute at an unshielded distance of 2 meters from the reacting material. The detection threshold in 10 CFR Section 70.24 takes precedence.

c. ANSI/ANS-8.10-2015, “Criteria for Nuclear Criticality Safety Controls in Operations With Shielding and Confinement” (Ref. 37)

Section 4.1 of the standard states that the provisions of the standard may be applied in facilities where operations are conducted remotely by persons located outside the shielded area and where shielding and confinement are adequate to meet the radiation dose limits of the standard. A clarification is that the provisions of the standard may be applied in any areas satisfying those conditions (i.e., may be applied on a per-area rather than per-facility basis).

Section 4.2.1 of the standard states that, to apply the provisions of the standard, shielding and confinement should be such that the total effective dose to any individual outside the shielded and confined area will not exceed 100 millisievert (mSv) (10 rem), and that the total effective dose to an individual outside the restricted area will not exceed 0.5 rem. A clarification is that, for the purpose of applying the provisions of the standard, the dose limits for an intermediate consequence event in 10 CFR Section 70.61, “Performance Requirements,” may be used in lieu of the dose limits specified in Section 4.2.1 of the standard. The dose limits in the standard are lower than those in 10 CFR 70.61(c), which defines an intermediate consequence event as a dose to workers of less than 0.25 sievert (Sv) (25 rem), or a dose to individuals outside the controlled area of less than 50 mSv (5 rem). The dose limits in Section 4.2.1 of the standard and the regulations in 10 CFR 70.61 are used for different purposes. The dose limits in Section 4.2.1 are used to provide flexibilities in various applications. The use of the higher dose limits in 10 CFR 70.61(c) for an intermediate consequence event is consistent with the NRC’s risk overall approach to risk.

d. ANSI/ANS-8.17-2004 (Reaffirmed 2014), “Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors” (Ref. 38)

Section 4.10 of the standard states that credit may be taken for fuel burnup in establishing fuel unit reactivity; assurance may be provided by measurement or by an analysis and verification of the exposure history of the fuel. An exception is that licensees and applicants may take credit for fuel burnup only when the amount of burnup is confirmed by physical measurements that are appropriate for each type of fuel assembly in the environment in which it is to be stored. Alternatively, licensees and applicants may perform a misload analysis, along with additional administrative loading procedures to reduce the likelihood of a misload, in lieu of a quantitative measurement.

e. ANSI/ANS-8.23-2007 (Reaffirmed 2012), “Nuclear Criticality Accident Emergency Planning and Response” (Ref. 39)

Section 4.1(9) of the standard requires provision for nuclear accident dosimeters meeting ANSI N13.3-1969 (Reaffirmed 1981), “Dosimetry for Criticality Accidents.” A clarification is that nuclear accident dosimeters may be used that do not necessarily comply with ANSI N13.3-1969 (R1981). This RG does not endorse that secondary

reference. In addition, ANSI N13.3 was revised in 2013, so this references an obsolete version of the standard.

f. ANSI/ANS-8.24-2017, “Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations” (Ref. 40)

Section 4.2 of the standard requires that verification of the computer code system be completed prior to validation. A clarification is that provision should also be made for periodic (e.g., annual) reverification, and not merely before validation.

Section 6.1.3 of the standard requires that if a positive bias (i.e., overestimation of the effective neutron multiplication factor, k_{eff}) is used in determining calculational margin, its use shall be justified based on the close ability of the benchmarks. The NRC has not historically allowed any credit for positive bias, but may choose to evaluate its use on a case-by-case basis with suitable demonstration that the causes of the bias are known and in accordance with Section 6.1.3.

g. ANSI/ANS-8.27-2015, “Burnup Credit for LWR Fuel” (Ref. 41)

Section 5.2 of the standard allows use of a combined validation approach. An exception is that a combined validation approach should not be used without addressing several key differences between benchmark data for combined validation approaches and spent nuclear fuel storage and transportation systems. Existing benchmark data for combined validation approaches consist of data from power reactors, which have significant physical differences from spent nuclear fuel storage and transportation systems.

One significant difference is fuel temperature, which will be significantly higher for spent fuel in a reactor compared to spent fuel in a storage or transportation system. Another significant difference is that the axial neutron flux distribution in a reactor is near the center of the fuel, whereas most of the neutron flux in a storage or transportation system will be in the top of the fuel, which may affect the neutron energy spectrum. Additional physical differences between power reactors and spent fuel storage and transportation systems include fresh water in the storage or transportation system versus borated water in the reactor (for PWRs), high-worth neutron absorber plates in storage and transportation systems versus none in a reactor, and full-density water in storage and transportation systems versus low-density water in a reactor. All of these parameters significantly affect neutron energy spectrum, and their effects on code bias and bias uncertainty need to be accounted for in a combined validation approach.

Section 7.2 of the standard allows assigned fuel burnup to be obtained by measurement or records of the nuclear facility. An exception is that licensees and applicants may take credit for fuel burnup only when the amount of burnup is confirmed by physical measurements that are appropriate for each type of fuel assembly in the environment in which it is to be stored. Alternatively, licensees and applicants may perform a misload analysis, along with additional administrative loading procedures to reduce the likelihood of a misload, in lieu of a quantitative measurement.

h. **ISO 7753:1987, “Nuclear Energy – Performance and Testing Requirements for Criticality Detection and Alarm Systems First Edition”**

Section 3.2.1 of the standard requires an evaluation of the need for a criticality alarm system in each area with threshold quantities of fissile nuclides, and Section 3.1 states that alarms shall be provided in areas where they will reduce total risk. An exception is that 10 CFR Section 70.24 takes precedence, and requires a criticality alarm system in each area where threshold quantities of special nuclear material are handled, used, or stored.

Section 3.2.1 of the standard also requires that attention be given to moderators or reflectors that are more effective than water. A clarification is that 10 CFR Section 70.24 requires that the threshold quantities be halved where graphite, heavy water, or beryllium are present.

Section 3.2.2 of the standard states that a criticality alarm system is not required in areas where the maximum foreseeable dose in free air is less than 0.12 Gy (12 rad). A clarification is that 10 CFR Section 70.24 requires placement of detectors in areas where threshold quantities of special nuclear material are present, but that audible or visual alarms may be located in areas where immediate evacuation is determined to be necessary based on the potential for an excessive dose.

Section 3.5.1 of the standard permits coverage by a single reliable detector for each area monitored. An exception is that 10 CFR Section 70.24 takes precedence and requires that two criticality detectors cover each monitored area.

Section 4.2 of the standard states that the minimum accident of concern may be assumed to deliver an absorbed dose in free air of 0.20 Gy (20 rad) in 1 minute at 2 meters from the reacting material. An exception is that 10 CFR Section 70.24 requires that a monitoring system be capable of detecting a nuclear criticality that produces an absorbed dose in soft tissue of 0.20 Gy (20 rad) of combined neutron and gamma radiation in 1 minute at an unshielded distance of 2 meters from the reacting material. The detection threshold in 10 CFR Section 70.24 takes precedence.

D. IMPLEMENTATION

The purpose of this section is to provide information on how applicants and licensees under 10 CFR Part 70, 10 CFR Part 71, and 10 CFR Part 72 may use this guide and information regarding the NRC's plans for using this RG. In addition, it describes how the NRC staff complies with the backfitting provisions in 10 CFR Section 70.76, "Backfitting," and 10 CFR 72.62, "Backfitting."

Use by Applicants and Licensees

Applicants and licensees may voluntarily¹ use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this RG may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations. Current licensees may continue to use guidance the NRC found acceptable for complying with the identified regulations as long as their current licensing basis remains unchanged.

Licensees may use the information in this RG for actions that do not require NRC review and approval such as changes to a facility design under 10 CFR 70.72 or 10 CFR 72.48. Licensees may use the information in this RG 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 RG. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this RG, 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 RG 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 RG. Examples of such unplanned NRC regulatory actions include issuance of an order requiring the use of the RG, generic communication, or promulgation of a rule requiring the use of this RG without further backfit consideration.

During facility-specific regulatory discussions, the staff may discuss with licensees various actions consistent with staff positions in this RG, 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 RG are part of the licensing basis of the facility. However, unless this RG 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 RG constitutes a violation.

If an existing licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this RG and (2) the specific subject matter of this RG is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the guidance in this RG 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 Section 70.76 or 10 CFR Section 72.62. Backfit and issue finality considerations do not apply to licensees and applicants under 10 CFR Part 71.

¹ In this section, "voluntary" and "voluntarily" means that the licensee is seeking the action of its own accord, without the force of a legally binding requirement or an NRC representation of further licensing or enforcement action.

Additionally, an existing applicant may be required to comply to new rules, orders, or guidance if 10 CFR Paragraph 70.76(a)(3) or 10 CFR Paragraph 72.62(c) applies.

If a licensee believes that the NRC is either using this RG or requesting or requiring the licensee to implement the methods or processes in this RG in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in NUREG-1409, "Backfitting Guidelines," (Ref. 42) and the NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 43).

REFERENCES²

1. *U.S. Code of Federal Regulations* (CFR), “Domestic Licensing of Special Nuclear Material,” Part 70, Chapter 1, Title 10, “Energy.”
2. CFR, “Packaging and Transportation of Radioactive Material,” Part 71, Chapter 1, Title 10, “Energy.”
3. CFR, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste,” Part 72, Chapter 1, Title 10, “Energy.”
4. U.S. Nuclear Regulatory Commission (NRC), NUREG-1520, “Standard Review Plan for Fuel Cycle Facilities License Applications,” Revision 2, Washington, DC.
5. NRC, NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility,” Washington, DC.
6. NRC, NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems,” Revision 1, Washington, DC.
7. NRC, NUREG-1567, “Standard Review Plan for Spent Dry Fuel Storage Facilities,” Washington, DC.
8. NRC, NUREG-1617, “Standard Review Plan for Transportation Packages for MOX Spent Nuclear Fuel,” Supplement 1, Washington, DC.
9. NRC, NUREG-1927, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel,” Revision 1, Washington, DC.
10. NRC, NUREG/CR-7108, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions,” Washington, DC.
11. NRC, NUREG/CR-7109, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions,” Washington, DC.
12. International Organization for Standardization (ISO) 7753:1987, “Nuclear Energy—Performance and Testing Requirements for Criticality Detection and Alarm Systems First Edition,” Geneva, Switzerland, 1987.³

² Publicly available NRC documents are available electronically through the NRC Library on the NRC’s public Web site at <http://www.nrc.gov/reading-rm/doc-collections/> and through the NRC’s Agencywide Documents Access and Management System (ADAMS) at <http://www.nrc.gov/reading-rm/adams.html>. The documents can also be viewed online for free or printed for a fee in the NRC’s Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

³ Copies of International Organization for Standardization (ISO) standards may be purchased from the ISO Web site at <http://www.iso.org>; by writing to International Organization for Standardization, Chemin de Blandonnet 8, CP 401, 1214 Vernier, Geneva, Switzerland; or by telephone +41 22 749 01 11.

13. International Atomic Energy Agency (IAEA), Safety Standard No. SF-1, “Fundamental Safety Principles,” Vienna, Austria.⁴
14. IAEA, Safety Requirement No. NS-R-5, “Safety of Nuclear Fuel Cycle Facilities,” Vienna, Austria.
15. IAEA SSG-27, “Criticality Safety in the Handling of Fissile Material,” Vienna, Austria.
16. IAEA SSG-15, “Storage of Spent Nuclear Fuel,” Vienna, Austria.
17. ISO 1709:1995, “Nuclear Energy—Fissile Materials—Principles of Criticality Safety in Storing, Handling, and Processing,” Geneva, Switzerland, 1995.
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