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General Comment

See attached file(s)

Attachments

10-20-17_NRC_NEI_Industry Comments on NRC Draft Regulatory Guide DG-3053 Nuclear Criticality Safety Standards for Nuclear Materials Outside Reactor Cores + Attachment

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October 20, 2017

Ms. Cindy Bladey
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Submitted via Regulations.gov

Subject: Industry Comments on NRC Draft Regulatory Guide DG-3053, "Nuclear Criticality Safety Standards for Nuclear Materials Outside Reactor Cores" (Docket ID: NRC–2017–0183)

Project Number: 689

Dear Ms. Bladey:

The Nuclear Energy Institute (NEI)[1], on behalf of our fuel cycle facility members, appreciates the opportunity to provide industry comments on DG-3053, "Nuclear Criticality Safety Standards for Nuclear Materials Outside Reactor Cores." We also appreciated staff's update on this draft regulatory guide (DG) during the public meeting on September 26, 2017. We are pleased to provide our written comments in this letter and the attachment.

We understand that the purpose of this DG (which will be issued as Revision 3 to RG 3.71) is to endorse several American National Standards Institute/American Nuclear Society (ANSI/ANS)-8 nuclear criticality safety standards that have been added, reaffirmed, or revised (some containing certain exceptions and clarifications), and make other changes that have occurred since the last revision of RG 3.71 in 2010[2]. The revision would also endorse International Organization for Standardization (ISO) Standard 7753:1987, "Nuclear Energy – Performance and Testing Requirements for Criticality Detection and Alarm Systems."

Two of the guides (ANSI/ANS-8.10-1983 and ANSI/ANS-8.23-2007) that were previously endorsed in RG 3.71, Rev. 2 are now endorsed with added exceptions. These exceptions appear to be an imposition of either new or different regulatory staff positions. The problem is compounded with the changes made to NUREG-1520, Rev. 2 "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." We encourage NRC staff to review Section 5.4.3.1.1 (Use of Industry Standards) of NUREG-1520,

[1] The Nuclear Energy Institute (NEI) is the organization responsible for establishing unified industry policy on matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include all entities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel cycle facilities, nuclear materials licensees, and other organizations and entities involved in the nuclear energy industry.

[2] NEI submitted a comment letter to the NRC on DG-3030 (Revision 2) dated September 23, 2010.

in light of the changes made to DG-3053. Furthermore, we would also like to refer the NRC to NEI's comment letter[3] and attachment on Draft NUREG-1520, Rev. 2, which addresses several relevant items in Chapter 5, Nuclear Criticality Safety.

Furthermore, under Section D (Implementation), the last paragraph on Page 11 states: "If an existing licensee voluntarily seeks a license amendment or change...then 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..." It would be helpful to gain additional context from NRC staff as to what level of justification would be required, in outlining "an equivalent alternative process that demonstrates compliance." Additionally, it is industry's understanding that, pursuant to the NRC's current position regarding so-called "forward-fitting," the application of Revision 3 would fall outside the definition of backfitting only in situations where it: (1) relates directly to a licensee's voluntary request, and (2) is an essential consideration in the NRC staff's determination of the acceptability of the licensee's voluntary request.[4] In other words, it is our understanding that unrelated licensing actions or amendments should not garner new commitments regarding the new referenced standards in Revision 3, nor should it garner subsequent RAIs related to criticality safety, as discussed during the September 26, 2017 public meeting.

The last sentence on Page 11, as referenced above states: "This is not considered backfitting as defined in 10 CFR 70.76 or 10 CFR 72.62." This paragraph notes that license amendments or changes would not constitute a backfit, but the DG is silent on license renewals. This omission leaves considerable uncertainty as to NRC's expectations for a license renewal. For example, an attempt to endorse a particular standard's current version may be problematic during a Part 70 license renewal, in which several decades are likely to pass between renewals. This places an unnecessary burden on the licensee to perform significant gap analyses, demonstrate or justify exceptions to revised standards, or potentially make significant licensed program upgrades.

The applicability of DG-3053 is also proposed to be expanded to include fuel cycle transportation (10 CFR Part 71 - Packaging and Transportation of Radioactive Material). However, this proposed scope expansion could impart unintended burdens on fissile package certificate holders when a renewal is pursued, without a clear regulatory basis, articulated benefit, or safety concern. This too appears to be an imposition of a regulatory staff position that is either new or different from a previous NRC staff position. In addition, the fissile package requirements in 10 CFR 71.55 and RG 7.9, "Standard Format and Content of Part 71 Packages for Radioactive Material" have not changed and there is no equivalent "Use of Industry Standard" statement (similar to Section 5.4.3.1.1 in NUREG-1520, Rev. 2) for Part 71 fissile package reviews, as there is for the reviews of applications to construct, modify or operate Part 70 nuclear fuel cycle facilities. As a result, it is unclear how NRC plans to expand the applicability of DG-3053 to Part 71 reviews, unless through inappropriate use of the RAI process.

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[3] NEI letter to Ms. Cindy Bladey, Docket NRC-2012-0220, dated November 3, 2014.

[4] Letter from S.G. Burns (NRC) to E.C. Ginsberg (NEI), July 14, 2010, at FN 2.

Thank you again for the opportunity to comment on DG-3053 and the related public meeting on September 26, 2017. We look forward to seeing how these comments are addressed in the final guidance. If you have any questions, please contact me or Hilary Lane (202-739-8148; hml@nei.org).

Sincerely,

Nima Ashkeboussi

Nima Ashkeboussi

Attachment

c: Craig Erlanger, NMSS, NRC
Margie Kotzalas, NMSS, NRC
Christopher Tripp, NMSS, NRC

ATTACHMENT

Industry Comments on Draft Regulatory Guide DG-3053, "Nuclear Criticality Safety Standards for Nuclear Materials Outside Reactor Cores"

1) **General Comment:**
   a. This DG endorses and aims to harmonize both ANSI/ANS-8 standards and, for the first time, standards developed by ISO, per routine reviews by the ANSI and ISO standard working group committees. Industry believes that harmonization of these standards through guidance documents is misplaced as an NRC goal, and consideration should be given to the removal of this ISO standard. However, should ISO standards remain, DG-3053 should address the acceptability of a licensee using a combination of both standards. For instance, clarify if it is acceptable for a licensee to endorse all ANSI/ANS-8 standards *except* for ANSI/ANS-8.3-1997, "Criticality Accident Alarm System," and alternatively endorse ISO Standard 7753:1987, "Nuclear Energy—Performance and Testing Requirements for Criticality Detection and Alarm Systems." Otherwise, without further clarification, this DG endorses two different standards for criticality accident alarm systems (2.b ANSI/ANS-8.3 and 2.h ISO 7753:1987), which is not necessary and will certainly cause confusion as to which takes precedent.

   b. As stated in the Federal Register Notice (FRN), "the proposed revision would provide methods that are acceptable to the NRC staff..." and the RG provides one method by which a licensee may demonstrate compliance, and alternative methods (to demonstrate compliance) with appropriate justification may be deemed acceptable by the NRC. DG-3053 incorporates several "shall" statements, understood to denote a requirement of the standard, if and only if the licensee commits to that particular standard. Alternatively, if the standard is not accepted by the licensee, and the licensee choses an alternate method, "shall" statements would not apply.

2) **Specific Comments:**
   a. As noted during the September 26, 2017 public meeting regarding DG-3053, the ANSI/ANS standards are continually being revised and re-affirmed by several working groups approximately every 5 years. However, this is problematic in that the effective dates of issuance are frequently changing and endorsing the current versions of the 18 standards in DG-3053 is not practical. For example, several on the list provided in DG-3053 (issued August 2017), Section C.1 (Page 6) are already outdated and have the following errors:

      i. The referenced citations for ANSI/ANS-8.12 are incorrect – it is now R2016.
      ii. The referenced citations for ANSI/ANS-8.14 are incorrect – it is now R2016.
      iii. The referenced citations for ANSI/ANS-8.17 are incorrect – it is now R2014.
      iv. The referenced citations for ANSI/ANS-8.20 are incorrect – it is now R2015.
      v. The referenced citations for ANSI/ANS-8.22 are incorrect – it is now R2016.
      vi. The referenced citations for ANSI/ANS-8.26 are incorrect – it is now R2016.

Furthermore, NRC stated that RG 3.71 is ultimately a "living document," that will be revised on a "regular basis." While we applaud the NRC for periodically updating its guidance documents to ensure that endorsements of standards remain current, we recommend revisions occur at a frequency of a minimum of every 5 years, in recognizing the Cumulative Effects of Regulation on the fuel cycle facility community. Furthermore, a minimum of 5 years between revisions will ensure a more comprehensive review, versus piecemeal changes. We do note that the last revision was 7 years ago, which we feel is an appropriate timeframe between revisions.

b. Section C.2.b, paragraph 2 (Page 7) states: "Section 4.2.2 of the standard states that a criticality alarm system is not required in areas where personnel would be subject to an excessive radiation dose." We believe there is an error in this sentence and the correct sentence would read "...where personnel would **not** be subject to an excessive radiation dose."

c. The ANSI/ANS-8.3 exceptions endorsement in DG-3053 added the sentence: "A clarification is that 10 CFR 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." Excessive radiation dose is defined in ANSI/ANS-8.3-1997 R2012 as "any dose to personnel corresponding to an absorbed dose from neutrons and gamma rays equal to or greater than 0.12 Gy (12 rad) in free air." However, 10 CFR 70.24(a)(1) requires a monitoring system capable of detecting a criticality that produces an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of 2 meters from the reacting material within one minute. The added sentence in the exceptions endorsement of ANSI/ANS-8.3 is not clear and appears to change the regulatory requirement from detecting 20 rads within one minute to a system capable of detecting 12 rad. Industry recommends using the current Reg Guide 3.71, Revision 2 monitoring system exception endorsements for ANSI/ANS-8.3.

d. The exceptions endorsement of Section C.2.c. (Page 8) is not clear and appear to be misaligned with the regulatory definitions in 10 CFR 70.61(c). The ANSI/ANS standard states shielding and confinement should be applied such that the total effective dose to any individual outside the shielded and confined area will not exceed 10 rem and that the total effective dose to an individual outside the restricted area will not exceed 0.5 rem. The DG then states that the dose limits in ANSI/ANS-8.10 are more conservative than the performance requirements in 10 CFR 70.61 and are applicable (i.e., 10 rem versus 25 rem for workers outside the shielded and confined area or 0.5 rem versus 5 rem to an individual outside the controlled area). As currently written, this DG endorsement appears to lower the regulatory definition of an intermediate consequence event described in 10 CFR 70.61(c) and NUREG-1520.

e. The exceptions endorsement of Section C.2.d (Page 8) states: "...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." The following new sentence states: "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."

Section 4.10 of ANSI/ANS-8.17 allows both physical measurements or appropriate analysis and verification to determine the appropriate fuel burnup credit as follows:

"In performing the criticality safety evaluation, the fuel characteristics ~e.g., material compositions, geometry, temperature that affect reactivity shall be chosen from the range of credible values such that the maximum neutron multiplication factor of the system is obtained. Credit may be taken for fuel burnup by establishing a maximum fuel unit reactivity and assuring that each fuel unit has a reactivity no greater than the maximum established reactivity. Assurance that the reactivity limit is not exceeded may be provided by (1) a measurement that can be related to the reactivity or (2) an analysis and verification of the exposure history of each fuel unit."

NRC currently accepts analytical methods to determine burnup credit in criticality safety analyses in the following documents:

- NUREG/CR 6801 (dated March 2003), "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis"
- NUREG/CR 7109 (dated April 2012), "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses- Criticality Predictions"
- NEI 12-16, Rev 1 (dated April 2014), "Guidance for Performing Criticality Analyses of Fuel Storage at Light Water Reactor Power Plants"

The DG-3053 endorsement of this particular standard does not provide the flexibility to use currently acceptable practices to use a burnup analysis method and appears to be narrowly focused on loading spent fuel casks.

f. Section C.2.f (Page 8) states: "Section 4.1 of the standard requires that verification of the computer code system be completed prior to validation. A clarification is that provisions should be made for routine (e.g., annual) reverification, and not merely before validation." It is not clear why annual reverification is required if there are no changes to the computer code system (i.e., method, hardware, software, or operating system). Industry suggests either removing this new statement, or that reverification should be performed only when there is a change to the configuration of the computer code system (for example, when a patch is installed to fix a glitch in the code, reverification should be performed). Performing reverification annually provides no added value when the software hasn't changed, and appears to be an imposition of a regulatory staff position that is either new or different from a previous NRC staff position. Currently, the computer code system is verified prior to use on the production workstation used to perform said transport calculations.