April 3, 2015

MEMORANDUM TO: Brian E. Thomas, Director
Division of Engineering
Office of Nuclear Regulatory Research

FROM: Lawrence E. Kokajko, Director ***/RA/***
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

SUBJECT: RESULTS OF PERIODIC REVIEW OF REGULATORY GUIDE 1.77

This memorandum documents the U.S. Nuclear Regulatory Commission periodic review of Regulatory Guide (RG) 1.77, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (PWRs),” published in May 1974. The RG provides guidance on meeting the requirements in Title 10 of the *Code of Federal Regulations* Part 50, Appendix A, General Design Criteria 28, “Reactivity Limits,” as it pertains to the methods and assumptions that may be used in evaluating the consequence of a control rod ejection accident in uranium oxide-fuel PWRs. As discussed in Management Directive 6.6, “Regulatory Guides,” the staff reviews RGs approximately every five years to ensure that the RGs continue to provide useful guidance. The documentation of the Office of Nuclear Reactor Regulation (NRR) staff review is enclosed.

Based on the results of the periodic review, NRR staff concludes that a revision to RG 1.77 is warranted. NRR staff currently plans to revise RG 1.77 by September 30, 2015.

Enclosure:
Regulatory Guide Periodic Review

CONTACT: Leslie Perkins, NRR/DPR
(301) 415-2375

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DATE	3/19/2015	3/19/2015	3/24/2015	3/25/2015	4/2/2015	4/3/2015

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Regulatory Guide Periodic Review

Regulatory Guide Number: **1.77**

Title: **Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors**

Office/division/branch: **NRR/DSS/SNPB**
Technical Lead: **Paul Clifford**

Recommended Staff Action: **Revise**

1. What are the known technical or regulatory issues with the current version of the Regulatory Guide (RG)?

RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (PWRs)," published in May 1974. The RG provides guidance on meeting the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria 28, "Reactivity Limits," as it pertains to the methods and assumptions that may be used in evaluating the consequence of a control rod ejection accident in uranium oxide-fuel PWRs. The RG identifies three regulatory positions: (1) Reactivity excursion will not result in a radial average fuel enthalpy greater than 280 cal/g at any axial location on any fuel rod; (2) Maximum reactor pressure during any portion of the assumed transient will be less than the value will cause stresses to exceed the Emergency Conditions stress limits as defined in Section III of the *American Society of Mechanical Engineers* (ASME) *Boiler and Pressure Vessel Code*; (3) Offsite dose consequences will remain well within the guidelines of 10 CFR Part 100, "Reactor Site Criteria."

The first regulatory position regarding reactivity excursion is not currently used as an acceptable limit. In more than 30 years since RG 1.77 was issued, there has been extensive experience gained through various studies performed to analyze and model fuel damage. This position is no longer considered acceptable due to peer reviews and experimental results indicating that it is non-conservative. The technical basis for this is stated in a letter dated January 19, 2007, Landry to Martin, "Technical and Regulatory Basis for the Reactivity Accident Interim Acceptance Criteria and Guidance" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070220400).

The second regulatory position regarding the maximum reactor pressure is essentially reiterated in the Standard Review Plan (NUREG-0800) in Section 15.4, which states, "the maximum reactor pressure during any portion of the assumed excursion should be less than the value that result in stresses that exceed the "Service Limit C" as defined in the ASME Bolier and Pressure Vessel Code." This is the current terminology for the same ASME criteria.

The third regulatory position regarding offsite does consequences is not needed because the regulations of Part 100 remain in effect regardless of whether or not it is stated as a position in RG 1.77. Furthermore, acceptance criteria of offsite dose consequences are provided in updated guidance (i.e., RG 1.95, Table 4 and RG 1.183 Table 6).

ENCLOSURE

Additionally, information presented in Appendix A, "Physical and Thermal Hydraulics" of RG 1.77, reflects the state-of-knowledge and technology in 1974 and portions of this appendix are outdated. Approved topical reports define the inputs, assumptions, and analytical methods that each fuel vendor or licensee employs to evaluate the control rod ejection accident. These approved topical reports are incorporated into the plant's licensing basis via reference in the Technical Specifications and/or Updated Final Safety Analysis Report. As the state-of-knowledge and technology evolve, these analytical methods change. For example, WCAP-15806-P-A, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics" (ADAMS Accession No. ML033350177) defines an approved realistic analytical methodology using 3D core neutron kinetics. This improved methodology replaces the conservative 1D axial core neutron kinetics methods which is more representative of the technology described in RG 1.77.

Appendix B, "Radiological Assumptions," in RG 1.77 is also no longer needed because RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design-Basis Accidents at Light-Water Nuclear Power Reactors," and RG 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors," provide updated guidance for the evaluation of the control rod ejection accident. Both of these documents state that the guidance contained within supersede corresponding radiological analysis assumptions provided in RG 1.77.

2. What is the impact on internal and external stakeholders of <u>not</u> updating the RG for the known issues, in terms of anticipated numbers of licensing and inspection activities over the next several years?

Licensees may use unclear guidance, potentially causing delays to reviews.

3. What is an estimate of the level of effort needed to address identified issues in terms of full-time equivalent (FTE) and contractor resources?

The NRC staff requires approximately 1 to 2 FTE to complete documentation of the changes required for RG 1.77

4. Based on the answers to the questions above, what is the NRC staff action for this guide (reviewed with no issues identified, reviewed with issues identified for future consideration, revise, or withdraw)?

Revise.

5. Provide a conceptual plan and timeframe to address the issues identified during the review.

The NRC staff currently plans to revise RG 1.77 by September 30, 2015

NOTE: This review was conducted in January 2015 and reflects the NRC staff's plans as of that date. These plans are tentative and subject to change.