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DRAFT REGULATORY GUIDE

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DRAFT REGULATORY GUIDE DG-1213

(Proposed Revision 1 of Regulatory Guide 1.141, dated April 1978)

CONTAINMENT ISOLATION PROVISIONS FOR FLUID SYSTEMS

A. INTRODUCTION

This guide describes updated methods that the U.S. Nuclear Regulatory Commission (NRC) staff considers acceptable for use in complying with the Commission's requirements for isolation of fluid systems that penetrate containment.

Title 10, of the *Code of Federal Regulations*, Part 50, "Domestic Licensing of Production and Utilization Facilities" (10 CFR Part 50), Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria (GDC) 54, 55, 56, and 57 establishes that piping systems that penetrate the primary reactor containment be provided with isolation capabilities that reflect the importance to safety of isolating these piping systems. The principal design criteria found in licensees' final safety analysis report (FSAR) should be similar in philosophy and meet the intent of the GDC.

The NRC issues regulatory guides 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 accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required.

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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 (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rulemaking, Directives, and Editing Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; e-mailed to nrcrep.resource@nrc.gov; submitted through the NRC's interactive rulemaking Web page at http://www.nrc.gov; or faxed to (301) 492-3446. Copies of comments received may be examined at the NRC's Public Document Room, 11555 Rockville Pike, Rockville, MD. Comments will be most helpful if received by August 28, 2009.

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B. DISCUSSION

Working Group ANS-56.2 of the American Nuclear Society (ANS) Standards Committee ANS-50, Nuclear Power Plant Systems Engineering, prepared a standard that specifies the minimum design requirements for containment isolation of fluid systems that penetrate the primary containment boundary of light-water-cooled reactors. The American National Standards Institute (ANSI) Committee N18, Design Criteria for Nuclear Power Plants, approved this standard and designated it ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems" (Ref. 2). The provisions of ANSI N271-1976 include minimum design, testing, and maintenance requirements for the isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors. Requirements for the design and testing of power supplies, the qualification of Class 1E equipment, and the design and testing of protection systems are outside the scope of this standard.

ANSI N271-1976 contains requirements (indicated by the word "shall") and recommendations (indicated by the word "should"). The recommendations and the requirements of the standard were evaluated with respect to importance to safety. The NRC considers all recommendations to be of sufficient importance to safety to be endorsed along with the requirements given in the standard.

The previous guide was issued in 1978. The proposed 2009 revision incorporates operating experience from the past three decades. Significantly, the proposed revision includes improved regulatory guidance as a result of the NRC staff's review of the lessons learned from the accident at Three Mile Island Nuclear Generating Station, Unit 2. It also updates the NRC guidance on acceptable design, testing, and maintenance requirements that licensees may use to comply with their plant-specific design criteria, which should meet the intent of GDC 54, 55, 56, and 57 of Appendix A to 10 CFR Part 50 for the isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors.

C. REGULATORY POSITION

The requirements and recommendations for the containment isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors, as specified in ANSI N271-1976, are generally acceptable and provide an adequate basis for use, subject to the following:

1. Section 3.6.4 of ANSI N271-1976 states, "The closed system shall be leak tested in accordance with 5.3 of this standard unless it can be shown by inspection that system integrity is being maintained for those systems operating at a pressure equal to or above the containment design pressure." The system integrity inspections may be applied to closed systems inside the containment in lieu of leak testing.

2. Section 3.6.6 of ANSI N271-1976 states "Relief valves in the backflow direction may be employed as isolation valves provided they satisfy the requirements of this standard." The licensee may use relief valves in the backflow direction or the forward (relief) flow direction as isolation valves provided that the relief set-point is greater than 1.5 times the containment design pressure in a manner consistent with NRC SRP 6.2.4, Subsection SRP Acceptance Criteria Item #7 (Ref. 3).

3. The licensee should provide thermally induced overpressure protection for liquid-filled piping between containment isolation barriers inside containment to prevent damage when the piping is isolated unless the licensee can demonstrate that the pressure between the isolation barriers

cannot exceed the design pressure of the isolation barriers or the design pressure of the piping. Any thermally induced overpressure protection method that the licensee uses should provide such protection inside containment at the maximum back-pressure condition that could exist during a loss-of-coolant accident.

4. Section 4.2.3 of ANSI N271-1976 states, "Sealed closed isolation valves are under administrative controls and do not require position indication in the control room for valve status." Because the containment isolation valves are components of the containment isolation system, which is an engineered safety feature system, all power-operated valves should have position indication in the control room.

5. Section 4.2.4 of ANSI N271-1976 states, "Isolation valve closure shall be completed when an isolation signal is received, and the valve shall not be opened until the signal is removed and deliberate operator action is taken (reset switch)." The reactor operator should not be able to override a containment isolation signal in such a way that would return any isolation valve to its normal (pre-accident) condition by a single action. More specifically, neither the reset/override of the safety injection actuation signal nor the reset/override of a containment isolation actuation signal for a group of valves should cause the reopening of any isolation valve. The licensee should not consider the use of procedural controls to prevent the reopening of a valve upon reset/override as an acceptable design alternative. The design of the reset/override capability should require a deliberate separate operator action, in addition to the reset/override of the signal, for the reopening of each isolation valve. Written, approved procedures should control the reopening of each containment isolation valve. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," (Ref. 4), provides additional guidance on procedures.

6. Section 4.2.5 of ANSI N271-1976 states, "Diversity in means of actuation of automatic isolation valves in series should be considered to preclude common mode failure." The NRC staff's position is that the licensee should provide diversity in the parameters sensed (i.e., types of isolation signals) for the initiation of containment isolation. The licensee may design the containment isolation logic to automatically initiate containment isolation upon the occurrence of an isolation signal derived from the individual coincidence logic of any of the continuously monitored parameters, such as those given in Section A.2 of Appendix A to ANSI N271-1976 for boiling-water reactors or in Section B.2 of Appendix B to ANSI N271-1976 for pressurized-water reactors. As a minimum, the licensee should monitor the following parameters, each with the capability of initiating containment isolation:

   a. high containment pressure;
   b. high radiation level within containment; and
   c. any manual, automatic, or coincident actuation of an engineered safety feature system or subsystem.

7. Section 4.4.2 of ANSI N271-1976 states, "For power-operated isolation valves, which do not receive a containment isolation signal, the primary mode shall be a remote manual initiation signal from the main control room." However, a containment isolation signal should automatically isolate all nonessential systems, as required in 10 CFR 50.34(f)(2)(xiv).

8. Section 4.4.8 of ANSI N271-1976 gives general design requirements for closed systems. In addition, all branch lines and their isolation valves in closed systems both inside and outside the containment should meet the design criteria of Section 3.5 or Section 3.6.7 of ANSI N271-1976 if the branch lines constitute one of the containment isolation barriers.

9. Section 4.6.3 of ANSI N271-1976 cites Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment following a Loss-of-Coolant Accident" (Ref. 5), for guidance in determining radiation exposures for a loss-of-coolant accident. Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants" (Ref. 6), gives more appropriate guidance to determine radiation exposures for a loss-of-coolant accident. For plants that have amended their licensing basis to use an alternate source term, see Appendix I of Regulatory Guide 1.183 (Ref. 7).

10. Section 4.14 of ANSI N271-1976 states, "The piping between isolation barriers or piping, which forms part of isolation barriers, shall meet the requirements of 3.7 and applicable requirements for isolation barriers." Piping between isolation barriers should meet the applicable requirements of Section 3.5 or Section 3.7 of ANSI N271-1976.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC's plans for using this draft regulatory guide. The NRC does not intend or approve any imposition or back-fit in connection with its issuance.

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

REGULATORY ANALYSIS

Statement of the Problem

The NRC staff's review of the lessons learned from the accident at Three Mile Island Nuclear Generating Station, Unit 2, revealed that an isolation signal derived from containment pressure alone was not sufficient to ensure containment isolation in the event of an incident.

The radiation level within containment is an appropriate signal/parameter for taking action to protect the public health and safety; therefore, licensees should monitor these levels. An isolation signal derived from an actuation of an engineered safety feature system or subsystem is a diverse method to ensure containment isolation under those conditions that warrant an engineered safety feature actuation. These three parameters (containment pressure, radiation level, and engineered safety feature actuation) provide diversity for containment isolation to prevent the release of radioactivity beyond the accepted limits under abnormal occurrences or credible accident conditions.

Objective

The objective of this regulatory action is to update the NRC guidance on acceptable design, testing, and maintenance requirements that licensees may use to comply with their plant-specific design criteria, which should meet the intent of GDC 54, 55, 56, and 57 of Appendix A to 10 CFR Part 50 for the isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors.

Alternative Approaches

The NRC staff considered the following alternative approaches:

Do not revise Regulatory Guide 1.141.
Revise Regulatory Guide 1.141.

Alternative 1: Do Not Revise Regulatory Guide 1.141

Under this alternative, the NRC would not revise this guidance, and the current guidance would be retained. If the NRC does not take action, there would be no changes in costs or benefit to the public, licensees, or the NRC. However, the "no-action" alternative would not provide licensees with the current staff positions on acceptable design, testing, and maintenance requirements that licensees may use to comply with their plant-specific design criteria, which should meet the intent of GDC 54, 55, 56, and 57 of Appendix A to 10 CFR Part 50 for the isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors.

Alternative 2: Revise Regulatory Guide 1.141

Under this alternative, the NRC would revise Regulatory Guide 1.141, taking into consideration the improved regulatory guidance resulting from the NRC staff's review of the lessons learned from the accident at Three Mile Island Nuclear Generating Station, Unit 2; and to provide the updated staff positions on ANSI N271-1976 taking into consideration the subsequent years of operating experience since the last revision of the guide.

One benefit of this action is that it would enhance reactor safety by improving regulatory guidance based on the lessons learned from the Three Mile Island accident. In addition, an updated Regulatory Guide 1.141 would provide licensees with the current staff positions on acceptable design, testing, and maintenance requirements that licensees may use to comply with their plant-specific design criteria, which should meet the intent of GDC 54, 55, 56, and 57 of Appendix A to 10 CFR Part 50 for the isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors.

The cost to the NRC would be the one-time cost of issuing the revised regulatory guide (which is expected to be relatively small), and applicants would incur little or no cost.

Conclusion

Based on this regulatory analysis, the staff recommends that the NRC revise Regulatory Guide 1.141. The staff concludes that the proposed action reflects the current staff positions on acceptable design, testing, and maintenance requirements that licensees may use to comply with their plant-specific design criteria, which should meet the intent of GDC 54, 55, 56, and 57 of Appendix A to 10 CFR Part 50 for the isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors.

REFERENCES

1. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission, Washington, DC.

2. ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems," American Nuclear Society, La Grange Park, IL, June 1976.

3. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800.

4. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," U.S. Nuclear Regulatory Commission, Washington, DC.

5. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment following a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission, Washington, DC.

6. Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.

7. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, Washington, DC.