Date

[illegible] AND TRANSMITTAL SLIP

January 28, 1980

TO: (Name, office symbol, room number, building, Agency/Post)	Initials	Date
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REMARKS

With regard to our Telecon earlier today, attached is a revised copy of R. G. 1.141. Please review Regulatory Position 4 (item c) so that we can establish a staff consensus regarding the requirement for isolation of containment following an engineered safety feature actuation.

DO NOT use this form as a RECORD of approvals, concurrences, disposals, clearances, and similar actions

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5041-102

☆ U.S. GPO: 1978-0-261-847/3324

OPTIONAL FORM 41 (Rev. 7-76)
Prescribed by GSA
FPMR (41 CFR) 101-11.206

U.S. NUCLEAR REGULATORY COMMISSION

[April 1978]

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

Revision 1
Draft 2
JANUARY 24, 1980

REGULATORY GUIDE 1.141

CONTAINMENT ISOLATION PROVISIONS FOR FLUID SYSTEMS

A. INTRODUCTION

General Design Criteria 54, 55, 56, and 57 of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," require that piping systems penetrating primary reactor containment be provided with isolation capabilities that reflect the importance to safety of isolating these piping systems. This guide describes a method acceptable to the NRC staff for complying with the Commission's requirements with respect to containment isolation of fluid systems.

B. DISCUSSION

Working Group ANS-56.2 of the American Nuclear Society Standards Committee ANS [illegible] Nuclear Power Plant Systems Engineering, h[illegible]d a standard which specifies the minim[illegible]sign requirements for containment isolation of fluid systems that penetrate the primary containment boundary of light-water-cooled reactors. This standard was approved by the American National Standards Institute (ANSI) Committee N18, Design Criteria for Nuclear Power Plants, and designated ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems."*

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, qualifying of Class 1E equipment, and the design and testing of protection systems are outside the scope of this standard. These areas are not completely covered by the references given in ANSI N271-1976.

* Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525.

Domestic

This revised guide includes improved regulatory guidance as a result of NRC staff review of the lessons learned from the Three Mile Island-Unit 2 accident. In particular, the review revealed that an isolation signal derived from containment pressure was not sufficient to ensure containment isolation when necessary. Radiation level within containment is the primary concern in protection of the public health and safety and should be monitored. In addition, this may be the only parameter capable of initiating containment isolation during certain situations (e.g., refuelling operations). An isolation signal derived from actuation of an engineered safety feature system or subsystem ~~provided for the mitigation of a loss-of-coolant accident or its consequences~~ is a reliable backup to ensure containment isolation under those conditions which warrant an engineered safety feature actuation. These three parameters (containment pressure, radiation level, and engineered safety feature actuation) provide diversity for containment isolation such as to prevent the release of radioactivity beyond the accepted limits under abnormal occurrences or credible accident conditions.

The manner in which the NRC staff will implement this regulatory guide is discussed in Section D, Implementation. In an effort to provide concise implementation guidance, Section D has been written in two parts. The first part addresses the implementation of Regulatory Positions 3, 4, and 5 which relate to the recommendations presented in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."[1] The implementation schedule for Regulatory Positions 3, 4, and 5, as described in Section D, corresponds to the schedule presented in NUREG-0578. The second part of Section D addresses the implementation of the remaining Regulatory Positions of the guide which are not directly related to NUREG-0578.

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U.S. NUCLEAR REGULATORY COMMISSION [April 1978]

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 1.141

CONTAINMENT ISOLATION PROVISIONS FOR FLUID SYSTEMS

This standard contains requirements indicated by the verb "shall" and recommendations indicated by the verb "should." The recommendations as well as the requirements of the standard were evaluated with respect to importance to safety. All recommendations are considered to be of sufficient importance to safety to be endorsed along with the requirements given in the standard.

This is acceptable for closed systems both inside and outside the containment.

C. REGULATORY POSITION

The requirements and recommendations for containment isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors as specified in ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems," are generally acceptable and provide an adequate basis for complying with the pertinent containment isolation requirements of Appendix A to 10 CFR Part 50, 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." [~~This exception to system leak testing is also applicable to closed systems inside the containment.~~]

2. 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." Since 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.

3. 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 capability should not exist for the reactor operator to override a containment isolation signal such that any isolation valve will be returned 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 for a group of valves should cause the reopening of any isolation valve.
The use of procedural controls to prevent re-opening of a valve upon reset/override should not be considered an acceptable design alternative.
The design of the reset/override capability should require a deliberate separate operator action, in addition to reset/override of the signal, for the reopening of each isolation valve.

This reopening of each containment isolation valve should be controlled by written procedures which meet the requirements of 10 CFR 50.36(c)(5), "Administrative Controls".
Additional guidance on procedures is given in Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch.

The guides are issued in the following ten broad divisions:

1. Power Reactors
2. Research and Test Reactors
3. Fuels and Materials Facilities
4. Environmental and Siting
5. Materials and Plant Protection
6. Products
7. Transportation
8. Occupational Health
9. Antitrust Review
10. General

Requests for single copies of issued guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Document Control.

Section 4.4.6 of ANSI N271-1976 states: "Diversity in the actuation parameters sensed should be considered."

The containment isolation logic should be designed to automatically initiate containment isolation upon the occurrence of an isolation signal derived from the individual coincidence logic of any continuously monitored parameter, such as given in ANSI N271-1976, Appendix A, Table A.2 (for BWRs) or Appendix B, Table B.2 (for PWRs). As a minimum, the following parameters should be monitored with each capable 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, ~~provided for the mitigation of a loss of coolant accident or its consequences. With regard only to containment isolation provisions, engineered safety feature systems provided for the mitigation of a loss-of-coolant accident include, as a minimum, the emergency core cooling systems, the primary containment isolation system, the auxiliary feedwater system (for PWR plants), and the reactor core isolation cooling system (for BWR plants).~~

4. [7] 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 there should be diversity in the parameters sensed (i.e., types of isolation signals) for the initiation of containment isolation.

6. [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 if the branch lines constitute one of the containment isolation barriers.

7. [9] In Section 4.6.3 of ANSI N271-1976, reference is made to Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," for guidance in determining radiation exposures for a loss-of-coolant accident. [More appropriate] guidance is given in Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants."

8. [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 re-

Additional

5. 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, all non-essential systems 2/ should be automatically isolated by a containment isolation signal.

1NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," was published by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission in July 1979. NUREG-0578 is available from the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 or the National Technical Information Service, Springfield, Virginia 22161.

2/ Appendix A to NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," contains a short term recommendation entitled, "Containment Isolation Provisions for PWRs and BWRs." The position of this section states: "All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essentail system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to the NRC."

quirements 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.

D. IMPLEMENTATION

[The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide.

This guide reflects current NRC staff practice. Therefore, except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein is being and will continue to be used in the evaluation of submittals for construction permit applications until this guidance is revised as a result of suggestions from the public or additional staff review.

For those plants for which the second round of questions (Q2) on the construction permit application has been received by the date of issuance of this guide, the recommendations of this guide will be considered by the staff on a case-by-case basis pursuant to §50.109 of 10 CFR Part 50.]

D. Implementation

1. For Regulatory Positions 3, 4, and 5 of this guide, except for those cases in which an applicant or licensee provides an acceptable alternative method for complying with specified portions of the Commission's regulations, the NRC staff will implement this guide as indicated below:

   a. Post-CP and CP Applicants

      For all plants in CP review and all plants under construction for which an OL has not yet been tendered, the method described herein will be used on a case-by-case basis in the evaluation of the application.

   b. OL Applicants

      The method described herein will be used on a case-by-case basis in the evaluation of the OL application prior to its issuance.

   c. Operating Reactors

      The method described herein will be used on a case-by-case basis in the evaluation of operating reactors as of January 1, 1980. However, this date will be extended to January 1, 1981 with regard to the containment isolation parameter (radiation level within containment) addressed in Regulatory Position C.4.b.

   The above implementation schedule will be used by the NRC staff in review of the following standard design applications to be submitted for review or approved:

a. Preliminary Design Approval (PDA) applications and Preliminary Duplicate Design Approval (PDDA) applications.

b. Final Design Approval, Type 1 (FDA-1) and Type 2 (FDA-2) applications and Final Duplicate Design Approval, Type 1 (FDDA-1) and Type 2 (FDDA-2) applications.

c. Manufacturing License (ML) applications.

d. A base plant design for replication.

2. For Regulatory Positions 1, 2, 6, 7, and 8 of this guide, except for those cases in which an applicant or licensee proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the NRC staff will implement this guide as indicated below:

a. Future Applications and Applications in the Early Stages of the Safety Review

The method described herein will be used in the evaluation of the following applications docketed after [date] or currently under review for which round Q-2 questions have not been issued as of [date]:

(1) Preliminary Design Approval (PDA) applications and Preliminary Duplicate Design Approval (PDDA) applications.

(2) Final Design Approval, Type 2 (FDA-2) applications and Final Duplicate Design Approval, Type 2 (FDDA-2) applications.

(3) Manufacturing License (ML) applications.

(4) Construction Permit (CP) applications, except for those portions of CP applications that reference standard designs (i.e., PDA, FDA-1, PDA-2, PDDA, FDDA-1, FDDA-2, or ML) or that reference qualified base plant designs under the replication option.

In addition, the methods described herein will be used in the evaluation of the following applications on a case-by-case basis:

(1) Final Design Approval, Type 1 (FDA-1) applications, and Final Duplicate Design Approval, Type 1 (FDDA-1) applications.

(2) Operating License (OL) applications.

b. Applications in the Late Stages of the Safety Review

The method described herein will be used on a case-by-case basis in the evaluation of the following applications currently under review for which round Q-2 questions have been issued as of [date].

(1) PDA applications and PDDA applications.

(2) PDA-1 applications and FDDA-1 applications.

(3) FDA-2 applications and FDDA-2 applications.

(4) ML applications.

(5) CP applications, except for those portions of CP applications that reference standard designs (i.e., PDA, FDA-1, FDA-2, PDDA, FDDA-1,

FDDA-2, or ML) or that reference qualified base plant designs under the replication option.

(6) OL applications.

c. Operating Reactors

The method described herein will be used on a case-by-case basis in the evaluation of operating reactors.