EDO Principal Correspondence Control

FROM: DUE: 07/26/10

EDO CONTROL: G20100408
DOC DT: 06/24/10
FINAL REPLY:

Said Abdel-Khalik, ACRS

TO:

Borchardt, EDO

FOR SIGNATURE OF : ** GRN ** CRC NO:

Borchardt, EDO

DESC:

Draft Final Regulatory Guide 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure" (OEDO-2010-0527)

ROUTING:

Borchardt
Weber
Virgilio
Ash
Mamish
OGC/GC
Leeds, NRR
Frazier, OEDO

DATE: 06/25/10

ASSIGNED TO: RES

CONTACT: Sheron

SPECIAL INSTRUCTIONS OR REMARKS:

Please prepare resonse to ACRS for the signature of the EDO. Add the Commission and SECY as cc's. USE SUBJECT LINE IN RESPONSE.

Template: EDO-001

E-RIDS: EDO-01

EDATS
Electronic Document and Action Tracking System

EDATS Number: OEDO-2010-0527 — Source: OEDO

General Information

Assigned To: RES — **OEDO Due Date:** 7/26/2010 11:00 PM

Other Assignees: — **SECY Due Date:** NONE

Subject: Draft Final Regulatory Guide 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure"

Description:

CC Routing: NRR

ADAMS Accession Numbers - Incoming: NONE — **Response/Package:** NONE

Other Information

Cross Reference Number: G20100408, R. G. 1.216 — **Staff Initiated:** NO

Related Task: — **Recurring Item:** NO

File Routing: EDATS — **Agency Lesson Learned:** NO

OEDO Monthly Report Item: NO

Process Information

Action Type: Letter — **Priority:** Medium

Sensitivity: None

Signature Level: EDO — **Urgency:** NO

Approval Level: No Approval Required

OEDO Concurrence: NO

OCM Concurrence: NO

OCA Concurrence: NO

Special Instructions: Please prepare response to ACRS for the signature of the EDO. Add the Commission and SECY as cc's. USE SUBJECT LINE IN RESPONSE.

Document Information

Originator Name: Said Abdel-Khalik — **Date of Incoming:** 6/24/2010

Originating Organization: ACRS — **Document Received by OEDO Date:** 6/25/2010

Addressee: R. W. Borchardt, EDO — **Date Response Requested by Originator:** NONE

Incoming Task Received: Letter

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

June 24, 2010

Mr. R.W. Borchardt
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL REGULATORY GUIDE 1.216, "CONTAINMENT STRUCTURAL INTEGRITY EVALUATION FOR INTERNAL PRESSURE LOADINGS ABOVE DESIGN-BASIS PRESSURE"

Dear Mr. Borchardt:

During the 573rd meeting of the Advisory Committee on Reactor Safeguards, June 9-11, 2010, we reviewed Draft Final Regulatory Guide (RG) 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure." Our Regulatory Policies and Practices Subcommittee also reviewed this matter during a meeting on May 19, 2010. During these meetings we had the benefit of discussions with representatives of the NRC staff and their consultants. We also had the benefit of the documents referenced.

RECOMMENDATION

Regulatory Guide 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure," should be issued.

BACKGROUND

RG 1.216 applies to new light water reactor designs with free-standing steel or concrete containments. It describes methods that the staff considers acceptable for: (1) predicting the internal pressure capacity for containment structures above the design-basis accident pressure; (2) demonstrating containment structural integrity related to combustible gas control; and (3) demonstrating that containment structural integrity meets the Commission's performance goals related to the prevention and mitigation of severe accidents. RG 1.216 complements other guidance such as RGs 1.136, 1.57, 1.7, and 1.206 that provide acceptable approaches to meeting regulatory requirements for combustible gas control, containment design limits, and containment load combinations.

General Design Criterion 50 states that the containment structure should accommodate "without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident." Containment structures designed in accordance with American Society of Mechanical Engineers (ASME) Code requirements have margin above the design pressure. Revision 1 of the Standard Review Plan (NUREG-0800) Sections 3.8.1 and 3.8.2 specifies that for a license application to be accepted for review, it must include an explicit demonstration that the containment internal pressure

EDO --G20100408

capacity significantly exceeds the design-basis accident pressure. In the past, licensees used a variety of approaches to do this, which required staff review on a case-by-case basis. RG 1.216 provides acceptable methods for performing these demonstrations.

Section (c)(5) of 10 CFR 50.44, "Combustible gas control for nuclear power reactors," requires that an applicant perform an analysis demonstrating containment structural integrity for an accident that releases hydrogen generated from a 100-percent fuel clad-coolant reaction, accompanied by hydrogen burning. RG 1.7 provides acceptance criteria to meet the structural requirements for steel and concrete containments. RG 1.216 provides additional guidance on the modeling approaches and assumptions for this analysis.

The Commission has established performance goals for containments under severe accident conditions. The Staff Requirements Memorandum (SRM) to SECY-90-16 approved the use of a conditional containment failure probability of 0.1 as a basis for regulatory guidance on containment performance. Subsequently, the SRM to SECY-93-087 stated:

> The containment should maintain its role as a reliable, leak-tight barrier (*for example, by ensuring that containment stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments*) for approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

RG 1.216 provides guidance for identifying the "more likely severe accident challenges" and determining the associated pressure and temperature responses to be used in containment analyses. It also provides guidance on modeling approaches and assumptions for these analyses.

DISCUSSION

Regulatory Position C.1 of RG 1.216 describes staff expectations for analyses of containment internal pressure capacity above the design pressure. The recommended analytical techniques include both the use of nonlinear finite-element analyses and simplified approaches based on experimental studies of steel and concrete containments under internal pressure loads. These experimental studies and corresponding analyses are described in NUREG/CR-6906.

In the simplified approaches, the structural capacities are defined in terms of maximum global membrane strain in regions away from discontinuities. The maximum allowable values for these strains are 1.0 percent for cylindrical reinforced concrete containments, 0.8 percent for cylindrical prestressed concrete containments, and 1.5 percent for cylindrical steel containments. These maximum strain limits are about one half the global strain values observed at containment failure in the tests described in NUREG/CR-6906, but take the structure into the fully plastic range. Failures are unlikely to initiate in the membrane stress regions away from discontinuities, but instead occur near strain risers associated with local features that are difficult to model accurately, even with finite-element methods. The geometric complexity of actual containments increases the likelihood that local strain risers are present and may be more severe than in any of the models tested.

The evaluation of the internal pressure capacity also addresses major containment penetrations, such as the removable drywell head and vent lines for BWR designs, equipment hatches, personnel airlocks, and major piping penetrations. These detailed analyses help designers preclude strain risers that could completely erode the margin provided by use of conservative membrane strain limits. RG 1.216 also references Appendix A to NUREG/CR-6906 for more detailed guidance for developing finite-element models and performing analyses for pressures above the design-basis accident pressure.

RG 1.7 provides the acceptance criteria for complying with the structural integrity requirements in 10 CFR 50.44. For the required pressure and dead loads, steel containments should meet the Service Level C requirements of ASME Code, Section III, Division 1, and concrete containments should meet the Factored Load Category requirements of ASME Code, Section III, Division 2. These criteria are also prescribed in SECY-93-087 and the associated SRM for severe accident performance of the containment. These limits constrain the structure to essentially elastic behavior and are considerably more conservative than the plastic strain limits used to estimate the ultimate pressure capacity. RG 1.216 also includes guidance on the finite-element modeling and the choice of material properties for steel and concrete containments.

RG 1.216 provides an interpretation of "more likely severe accident challenges." The methodology described in the Guide selects sequences or plant damage states that represent 90 percent or more of the core damage frequency. From the set of pressure and temperature transient loadings associated with these sequences, pressure and corresponding temperature loadings that envelope the entire set of pressure and temperature loadings are identified. These pressure and temperature loadings can then be used to analyze the containment to determine whether the acceptance criteria have been met. This set of sequences is unlikely to represent 90 percent of the risk for the plant, but it is relatively easy to develop and, together with the conservative acceptance criteria, is adequate to meet the intent of the SRMs to SECY-90-16 and SECY-93-087.

The Subcommittee discussed with the staff a number of clarifications to the language of RG 1.216. The staff was responsive and presented suggested revisions that adequately addressed the needed clarifications. With the inclusion of these revisions, Regulatory Guide 1.216 should be issued.

Sincerely,

/RA/

Said Abdel-Khalik
Chairman

References:

1. Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design Basis Pressure (Draft Regulatory Guide 1.216) (ML093200703)

2. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (NUREG-0800) (ML052340514)

3. Control of Combustible Gas Concentrations in Containment (Regulatory Guide 1.7) (ML070290080)

4. Combined License Applications for Nuclear Power Plants (LWR Edition) (Regulatory Guide 1.206) (ML062460002)

5. SRM MC900626, "SECY-90-16 - Evolutionary Light-Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements," 06/26/1990 (ML0518100700)

6. SRM MC930721, "SECY-93-087- Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," 07/21/1993 (ML0517501700)

7. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," 04/02/1993, (ML0037080210)

8. SRM MC900615, "SECY-89-102 - Implementation of the Safety Goals," 06/15/1990, (ML0516607120)

9. SECY- 90-0831, "May 22, 1990, Staff Requirements Memorandum on Briefing of Evolutionary Light Water Reactor Certification Issues and Related Regulatory Requirements," 08/31/1990 (ML0517407440)

10. Containment Integrity Research at Sandia National Laboratories (NUREG/CR-6906)

11. Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Regulatory Guide 1.57) (ML070310029)

12. Design Limits and Loading Combinations, Materials, Construction, and Testing of Concrete Containments (Regulatory Guide 1.136) (ML070310045)

13. Overpressurization Test of a 1:4-Scale Prestressed Concrete Containment Vessel Model (NUREG/CR-6810) (ML032040014)

14. Posttest Analysis of the NUPEC/NRC 1:4-Scale Prestressed Concrete Containment Vessel Model (NUREG/CR-6809) (ML031830555)