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

ACRSR-2266

September 26, 2007

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: PROPOSED RECOMMENDATION FOR RESOLVING GENERIC ISSUE 156.6.1, "PIPE BREAK EFFECTS ON SYSTEMS AND COMPONENTS INSIDE CONTAINMENT"

Dear Mr. Reyes:

During the 545th meeting of the Advisory Committee on Reactor Safeguards (ACRS), September 6-8, 2007, we reviewed the analyses performed by the Office of Nuclear Regulatory Research (RES) to resolve Generic Issue (GI) 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment." During our review, we had the benefit of discussions with representatives of RES and of the documents referenced.

CONCLUSION

We concur with the RES recommendation that GI 156.6.1 be closed and that no further actions by the NRC staff or licensees with respect to this issue are necessary.

BACKGROUND AND DISCUSSION

The potential for dynamic effects associated with high energy line breaks such as pipe whip and discharging fluids to disable important safety systems has long been recognized. However, a significant number of plants were licensed before the criteria for postulated pipe break locations, pipe whip restraints, and separation for instrumentation and control systems were established in the Standard Review Plan. These criteria are intended to ensure that a single pipe break does not disable systems and components needed to respond to the event.

The issues associated with pipe break effects in plants licensed before the issuance of the Standard Review Plan were included in the Generic Issues Program in 1991 and assigned a "medium" priority in 1994. In 1999, the Idaho National Engineering and Environmental Laboratory (INEEL) conducted a study to reevaluate its potential significance. This study concluded that the issue could possibly have greater safety significance than estimated by the 1994 prioritization. Subsequent reports by the Boiling Water Reactor Owners Group and an additional NRC sponsored report by Information Systems Laboratories criticized the INEEL report as being overly conservative.

For boiling water reactors (BWRs), the critical problem identified was the potential for penetration of the drywell shell by impact from ruptured piping. The critical problem in pressurized water reactors (PWRs) was a high energy line break inside containment in which pipe whip or jet impingement causes failure of instrumentation and control systems.

To resolve this issue for BWRs, the staff performed a quantitative analysis of the potential for structural failure of the drywell shell. The staff performed elastic–plastic analysis of the pipe breaks most likely to cause structural failure of the drywell shell. The analysis demonstrated that the drywell shell deforms under the impact of the pipe and comes into contact with the concrete shield wall. The shield wall remains intact under the impact and limits the deflections and strains in the drywell shell to levels well below those expected to cause rupture.

For PWRs, the layout of each plant within the scope of the generic issue was examined using drawings and information obtained from licensing project managers, resident inspectors, and licensee personnel. Assessments were made of the separation between electrical penetrations needed for instrumentation and control systems, the locations of high-energy pipes near these penetrations, and the presence of any barriers between the piping and the penetrations. Based on these assessments, the staff concluded that the likelihood of failure of instrumentation and control systems as the result of dynamic effects was acceptably low.

Based on the results of the studies, the staff concluded that further pursuit of Generic Issue 156.6.1 is not justified and recommended that it be closed out. We concur with the staff's recommendation.

Sincerely,

/RA/

William J. Shack
Chairman

<u>References</u>:

1. Memorandum to Frank P. Gillespie, Executive Director, Advisory Committee on Reactor Safeguards from Farouk Eltawila, Director, Division on Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research, "Providing Information on Generic Issue 156.6.1 for ACRS Meeting 545 – September 6-8, 2007," July 18, 2007, ML071780276

2. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (1st Edition) November 1975, (2nd Edition) March 1980, (3rd Edition) July 1981.

3. NUREG/CR-6395, "Enhanced Prioritization of Generic Safety Issue 156.6.1: 'Pipe Break Effects on Systems and Components Inside Containment'," November 1999, ML003732008

4. NEDC-33054, "Conservatism in NRC Prioritization of Pipe Break Effects on Systems and Components," Boiling Water Reactor Owners Group, November 2001, ML072060402

5. “Review of Event Probabilities and Frequencies Used in NUREG/CR-6395, Enhanced Prioritization of Generic Safety Issue 156.6.1, Pipe Break Effects on Systems and Components Inside Containment,” Information Systems Laboratories, December 2002, ML072080390

5. Review of Event Probabilities and Frequencies Used in NUREG/CR-6395, Enhanced Prioritization of Generic Safety Issue 156.6.1, Pipe Break Effects on Systems and Components Inside Containment," Information Systems Laboratories, December 2002, ML072080390

Distribution:

RidsSecyMailCenter	RidsEDOMailCenter	RidsNRROD
RidsNMSSOD	RidsFSMEOD	RidsOCAMailCenter
RidsNSIROD	RidsRESOD	RidsNROOD
RidsOIGMailCenter	RidsOGCMailCenter	RidsASLBPMailCenter
RidsOCAAMailCenter	RidsOPAMailCenter	RidsRGN1MailCenter
RidsRGN2MailCenter	RidsRGN3MailCenter	RidsRGN4MailCenter

BChamp ABates JPerry SDuraiswamy DPelton	CSantos DBessette HNourbakhsh GShukla PTressler	MBanerjee CHammer JFlack CJaegers	SMcKelvin LMike ZAbdullahi DFischer

ADAMS ML072530615

OFFICE	ACRS	ACRS	SUNSI	ACRS	ACRS
NAME	DBessette	CSantos	CSantos	FPGillespie	FPG for WJS
DATE	09/26/07	09/26/07	09/26/07	09/26/07	09/26/07

OFFICIAL RECORD COPY