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MEMORANDUM FOR: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

FROM: Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER NO. 156 "AN INVESTIGATION OF CORE LIQUID LEVEL DEPRESSION IN SMALL BREAK LOSS-OF-COOLANT ACCIDENTS"

References:

1. R. Wayne Houston memorandum for O. E. Bassett, "Request for Assistance in Resolving TMI Action Item II.K.3.30 (SBLOCAs)", July 19, 1983.

2. W. W. Tingle, Test Data Report on Westinghouse Reactor Vessel Level Indicating System Performance During Semiscale Test S-UT-8, EGG-SEMI-5827, March, 1982.

3. G. G. Loomis and J. E. Streit, Quick Look Report For Semiscale Mod-2C Experiments S-LH-1 and S-LH-2, EGG-SEMI-6884, May, 1985.

4. R. Fujita, TRAC-PF1/MOD1 Post Test Analysis of Semiscale Small Break Test S-UT-8, LA-UR-85-961, December, 1985.

5. P. D. Wheatley, et al., RELAP5/MOD2 Code Assessment at the Idaho National Engineering Laboratory, NUREG/CR-4454, March, 1986.

6. G. G. Loomis and J. E. Streit, Results of Semiscale MOD-2C Small Break (5%) Loss-of-Coolant Accident Experiments S-LH-1 and S-LH-2, NUREG/CR-4438, November, 1985.

7. C. D. Fletcher and C. M. Kullberg, Break Spectrum Analyses for Small Break Loss-of-Coolant Accidents in a RESAR-3S Plant, NUREG/CR-4384, March, 1986.

156

This Research Information Letter (RIL) transmits the attached results of a study conducted by the Idaho National Engineering Laboratory (INEL) and the Los Alamos National Laboratory (LANL) to investigate "Core Liquid Level Depression in Small Break Loss-of-Coolant Accidents" (SBLOCAs) in Westinghouse (W)-type four-loop pressurized water reactors (PWR). This culminates the previous transmittal of a series of analysis and test results in

References 1-7. The W-type plants studied were similar to the RESAR-3S plant and will be termed the reference PWR. For the study outlined in the attached results, test data of Semiscale (MOD2C) S-LH-1 and five tests from the ROSA-IV Large Scale Test Facility (LSTF) tests were chosen for TRAC-PF1/MOD1 (version 12.7) benchmark calculations.

Semiscale (MOD2C) was a 1:1705 power and volume scaled and 1:1 elevation scaled facility. ROSA-IV LSTF is the largest non-nuclear facility in the world for the study of SBLOCAs and transients in PWRs. The facility is scaled 1:48 in terms of volume and 1:1 in terms of elevation compared to a full-size plant.

1. Regulatory Issue

During a cold leg SBLOCA, hydraulic "seals" are formed in the pump suction loop U-bend piping as a result of slow gravity dominated depletion of primary coolant, characteristic of small cold break transients. The liquid seals impede the flow of vapor (generated in the core) through the coolant loop piping thus preventing steam venting from the break and this in turn can induce a differential pressure between the reactor core and downcomer, especially when liquid holdup is present in the steam generator tubes. The coolant level in the core is consequently depressed, and core heatup occurs. The phenomena were first calculated by RELAP5, and later observed in the Semiscale S-UT-8 test (i.e., a 5% cold leg SBLOCA) conducted in late 1981. The phenomena were subsequently confirmed in the Semiscale S-LH-1 test conducted in 1985 with improved instrumentation and better defined boundary conditions.

There is considerable uncertainty about the safety significance of the observed core heatup phenomena because: (1) it is not certain that the observed phenomena will occur in a full-scale PWR, and (2) the effects of the complete spectrum of possible design and accident parameters had not been investigated. Because the core heatup phenomena could have direct safety and licensing implications, assistance from NRC/RES was requested by the staff of the Division of Systems Integration, Office of Nuclear Reactor Regulation in mid-1983.[1]

2. Research Results and Conclusions

Since the NRC/NRR request, considerable experimental and analysis work have been accomplished. Following the Semiscale S-UT-8 test[2] conducted in December 1981, the Semiscale S-LH series tests[3] were conducted in 1985 and 1986 in order to better understand the core level depression and core heatup phenomena. Some significant analysis work completed in 1985 and 1986 includes: (1) a TRAC-PF1/MOD1 post-test analysis of the Semiscale S-UT-8 experiment,[4] (2) A RELAP5/MOD2 code assessment analysis of the Semiscale S-UT-8 experiment,[5] (3) a RELAP5/MOD2 code assessment analysis of the Semiscale S-LH-1 experiment[6] and (4) a break spectrum analysis of the W RESAR-3S plant using both TRAC-PF1/MOD1 and RELAP5/MOD2.[7] These previous results have been previously transmitted to your staff.

Following the above contributions, NRC/RES had the unique opportunity to take advantage of the existing USNRC/JAERI ROSA-IV bilateral agreement to investigate the core level depression phenomna. At the request of the NRC, and based on the results in References 1-7, the Japanese agreed to perform a series of cold leg SBLOCA tests in the ROSA-IV Large Scale Test Facility (LSTF) at the Japan Atomic Energy Research Institute (JAERI). The test data of these LSTF tests and that from the previously conducted Semiscale S-LH-1 tests were then used for TRAC benchmark calculations.

The benchmark calculations demonstrated that TRAC-PF1/MOD1 can predict the liquid mass distribution and force balance in the vessel and the loops which lead to core liquid level depression and liquid holdup in both 1:1705 and 1:48 volume scaled facilities reasonably well. In particular, for the LSTF experiment conducted as a counterpart to the Semiscale S-LH-1 test with best-estimate boundary conditions (especially core power decay), reasonable agreement was shown between the data and the benchmark calculation for not only the above phenomena but the core heatup as well. For the high decay power tests, however, the magnitude of the core uncovery and the rod temperature heatup were usually underpredicted by TRAC. In the worst case studied, the magnitude of rod heatup was under predicted by 67°F. This can be considered a worst case underpredictive bias for the TRAC code.

Following the extensive code benchmark calculations, a baseline reference PWR TRAC-PF1/MOD1 calculation was conducted to provide a basis of evaluation to analyze core liquid level depression behavior in W-type plants. The consistent modeling approach used for all benchmark calculations was also used in this reference PWR study. For a 5% cold leg SBLOCA in the reference PWR the calculated core liquid level depression by TRAC showed a 4.3ft minimum core collapsed liquid level (i.e., 1.6ft below the core midplane). No core heatup was calculated.

Several important factors such as core decay power, core bypass geometry, break size, and ECCS flow influencing core liquid level depression have been investigated experimentally and analytically. A number of conclusions and observations regarding these factors are summarized in the attached results.

3. Regulatory Implications

Based on the code underpredictive bias and TRAC calculational results for a 5% cold leg SBLOCA in the reference PWR, either no core heatup or core heatup less than 67°F is expected at the most. Therefore, it is concluded that:

(1) Core heatup in a reference PWR during the core liquid level depression phase of a 5% SBLOCA does not pose a safety concern.

(2) Based on observed core heatup data obtained in the ROSA-IV LSTF experiments conducted at representative or nearly representative core power levels, little or no core heatup is expected in a reference PWR during cold leg SBLOCAs for break sizes ranging from 2.5 to 10%.

4. Further Work

An additional factor which might influence core level depression and core level recovery for a 5% cold leg SBLOCA is the SG secondary-side conditions. An asymmetric loop seal clearing and delayed core level recovery were observed in the Semiscale S-LH-1 test. Asymmetric secondary conditions might occur due to a simultaneous failure of a secondary component or an operator action at the time of the SBLOCA. These considerations are beyond the scope of the study but might warrant future investigations.

Eric S. Beckjord

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research