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MASTER

PROGRESS REPORT

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Contract AT(11-1)-2294

"Technical Assessment of Two-Phase Flow Aspects of Nuclear Reactor Safety"

Prepared by the Principal Investigator,
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PROGRESS REPORT

Throughout the period of this contract we have been in continual contact with Dr. Zoltan Rosztoczy, Reactor Systems Branch, Division of Technical Review, Nuclear Regulatory Commission. He has received memoranda describing progress as it occurred. We have also responded to specific requests for review of documents, consultation at NRC and ERDA offices, attendance at working meetings and conferences.

Progress has been made in the following specific areas:

1) Completion of Report COO-2294-4 "Steam/Water Interaction in a Scaled Pressurized Water Reactor Downcomer Annulus", by C.J. Crowley, G.B. Wallis and D.L. Ludwig, September 1974. An outline of this report is given overleaf.

This study investigates the interaction of steam and water in a countercurrent annular flow situation. In particular, a linearly-scaled, "unwrapped" plastic annulus - simulating the downcomer of a Pressurized Water Reactor (PWR) - is used as the model.

Reactor safety analysis programs called for by the AEC require the reactor vendors to postulate breakage of one of the pipes carrying coolant into (or out of) the pressure vessel and to predict the subsequent events. One portion of this Loss of Coolant Accident (LOCA) involves a period of time during which the fluid escaping the pressure vessel is mainly steam, under high pressure, exiting through the break in the piping. When the pressure drops to a certain value (either 600 or 200 psi., depending upon the design), large amounts of water from pressurized storage tanks are rapidly injected through pipes which branch into the cold legs - the inlet pipes. Hence, these unbroken cold legs direct the Emergency Core Coolant (ECC) water into the annulus, where it is intended to fall and fill the lower plenum, in spite of the presence of a strong steam momentum flux upward. It is believed, however, that the steam flow may carry some, perhaps all, of the cooling water out of the break for a time rather than allowing it to fall into the plenum and begin to resubmerge and cool the hot core of the reactor. This is called the "accumulator bypass" phenomenon.

The engineering aspects of accumulator bypass involve calculating the quantity of water which actually is bypassed for a given steam flow. The AEC ruling at the present time regarding the bypass calculations is to assume, in the absence of better information, that all of the ECC water injected into the system prior to the "end-of-bypass" is removed from the system.

This is most likely too conservative since, for instance, a good deal of water may be stored or held up in the annulus, only to reach the lower plenum when the steam flow falls below a critical value. In their calculations, the vendors would like to be able to credit some of the ECC water to the amount of fluid refilling the vessel during the blowdown phase, but the phenomena involving steam/water countercurrent interactions must first be understood before any model can be approved.

None of the previously proposed models takes into account two-dimensional flow patterns available to the steam and the water, or possible asymmetries of pressure vessel design. More importantly, the effects of condensation are also slighted. (The two-dimensionality may actually be advantageous, if the asymmetry allows steam to escape and yet allows the coolant to penetrate.) The applicability of one-dimensional models to the scaled reactor geometry is discussed in this study. Effects of variations in the arrangement of the injection legs, and design modifications such as baffles, are also included.

Briefly, the apparatus consists of a transparent polycarbonate parallel plate "annulus" with a 0.375 inch gap between the plates (1/30 scale). This is mounted on a barrel 22 inches in diameter and 2 feet high (not to scale). Steam enters the barrel, and proceeds upward through the annulus. Water is injected into the annulus via tubes perpendicular to it, as in a reactor. The downward water flow in the gap creates a countercurrent flow condition which is intended to model the accumulator bypass situation.

The types of tests which may be carried out with such an apparatus are as follows: The location or size of the injection pipes may be altered depending upon whether different designs of the pressure vessel are modelled. Baffling by means of straight channels or curved collars may be added, with the intent of helping to direct the water flow downward. The effects of the presence of a simulated thermal shield will be discovered. In the experimental procedure, tests may be conducted in which the water flow is held constant and the steam flow varied, or it may be done vice versa (corresponding more closely to the bypass situation). Finally, the effects of inlet water temperature upon the results may be discovered. The tests were basically steady state and the effects of transients minimal.

The experiment seeks to determine the locus of "flooding" points - that is, under what conditions a bypass of water out the break occurs and the amount of the bypass - in the presence of condensation effects, and demonstrate the effect of the above-mentioned design modifications upon this locus. As the experiments show, when bypass was occurring at the water flow rates tested, the amount of water entering the lower plenum was negligible. It is important, however, to determine whether the modifications will aid or hinder water from reaching the lower plenum.

Mechanisms for the observed behavior are postulated and described. Dimensionless scaling parameters are suggested and used to compare these results with data obtained elsewhere and with the predictions of reactor transient codes.

2) A study of the Hot Wall effect, presented as the B.E. thesis of John E. Allen, "A Study of the Hot Wall Effect in a Loss of Coolant Accident", September 1974. The abstract of that thesis reads as follows:

The hot wall effect is one of the phenomena which could reduce the effectiveness of the emergency core cooling system after a loss of coolant accident in a pressurized water reactor. The magnitude of this effect is studied experimentally and analytically in this paper.

The hot wall effect is the name given to the phenomenon whereby emergency core cooling water is retarded in its entry into the reactor 1) due to levitation by steam generated when the cooling water contacts the hot reactor walls and 2) by the decrease in the amount of cooling water due to evaporation.

Cooling water, subcooling, initial wall temperature, length, and lower plenum venting were found to have significant effects on the delay time caused by hot walls in a very small, unscaled experiment. The response of the experiment was governed by the "flooding" phenomenon for cooling water at saturation temperature and low initial wall temperatures.

A finite difference, two-dimensional, transient heat conduction computer code was developed to model both the experimental apparatus and a full scale pressurized water reactor. The model predicted the delay time of the experimental data for the region where the Wallis flooding correlation governed the response of the experiment. The code showed that the infinite heat transfer coefficient assumption in the Wallis-Block theory is perhaps an excessively conservative assumption.

Finally, based on the assumptions in the analysis, the conclusion of this paper is that the hot walls have a marginal effect in

retarding the flow of the emergency core cooling water into a pressurized water reactor when their effect is considered alone. However, the effect is significant when it is considered with the other effects that would normally be active during a loss of coolant accident.

3) Submission of the draft of report COO-2294-2 "Effect of Hot Walls on the Loss of Coolant Accident". The completion of this report has been delayed in order to allow the incorporation into it of new data and analysis. It should be completed and distributed before the end of this contract period. The abstract of the original draft reads as follows:

This report presents a preliminary analytical model for the effect of heated reactor walls on the processes occurring during the emergency core cooling system (ECCS) injection period of a postulated loss of coolant accident (LOCA) in a pressurized water reactor (PWR). In particular, the investigation reported here represents the first step in our efforts to understand the factors that influence the rate of penetration of the injected water down the annulus and into the lower plenum and core. Because the model developed here contains elements of the more encompassing countercurrent steam/water mixing problem in the downcomer annulus, such as core steam flow, it may represent a useful start on a unifying model of the annulus behavior during blowdown and refill.

In addition to detailing a basic downcomer model with various levels of sophistication, this report presents the results of a series of simple, small scale experiments which tend to confirm the general validity of the model. However, since these experiments only explored several of the numerous important variables and since they were performed in a fairly idealized geometry, considerably more experimental work will be required to bring the model to maturity. The work still required is briefly discussed.

4) Submission of several memoranda and preliminary reports dealing with the "Annulus Bypass" problem and phenomena occurring in the lower plenum of a PWR.

"An Idealized Model of Water Levitation in a PWR Annulus", G.B. Wallis, October 1974.

"Further Models for Liquid Levitation by a Vertical Plane Gas Flow", G.B. Wallis, October 1974.

"Experimental Hydrodynamic Data in a Cylindrical Geometry 1/30 scale PWR During Simulated LOCA Conditions, Robert J. Tobin and Graham B. Wallis, January 1975.

"Lower Plenum Voiding in a Model PWR", G.B. Wallis, February 1975.

"Steam/Water Interaction in a Scaled Pressurized Water Downcomer Annulus", Yoshinobu Hagi, March 1975.

"Effect of ECC Injection Section Oscillations on Penetration of Water to the Lower Plenum of a Model PWR", Gordon Graham, March 1975.

All of this work will be consolidated into a formal report (COO-2294-6) before the end of this contract period.

5) A study of Annulus and Lower Plenum phenomena for the B&W vent valve design has been started.

6) Prof. Wallis has consulted with the Regulatory Staff on the following topics:

Formulation of Two-Phase Flow Momentum Equations

K-factors for B&W vent valves

K-factors for unsteady flow near ECC injection points

Drift flux model for the downcomer

Hot wall delay time

BWR reflood.

In addition, visits have been made to Germantown and Bethesda for attendance at NRC meetings.

Throughout the period of this contract the principal investigator has been involved at the rate of 25% of his time. He will continue this involvement until the end of the contract period.