116. j. m. 116
Conf-700608--8

THE EFFECTS OF CLAD SWELLING ON EMERGENCY CORE COOLING PERFORMANCE

Wayne A. Carbiener
BATTELLE MEMORIAL INSTITUTE
Columbus Laboratories

Presented at

16th Annual Meeting of The American Nuclear Society
Session on Fuel Rod Failure and Its Effects
July 1-2, 1970
Los Angeles, California

MASTER

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

Work Performed Under AEC Contract W-7405-eng-92.

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

[Handwritten signature]

THE EFFECTS OF CLAD SWELLING ON EMERGENCY CORE COOLING PERFORMANCE*

by

Wayne A. Carbiener
BATTELLE MEMORIAL INSTITUTE
Columbus Laboratories

The possibility of degraded Emergency Core Cooling (ECC) performance due to core geometric changes from Zircaloy cladding failures or swelling has been recognized as a potential safety problem in the unlikely event of a loss-of-coolant accident for some time. Extensive investigations of the mechanisms for cladding failure and the resultant modes of failure have been carried out through the Failure Modes of Zircaloy-Clad Fuel Rods Program coordinated by ORNL. These studies have included both single- and multirod tests conducted by both the light-water reactor manufacturers and AEC contractors. The studies to date have served to characterize the geometry of the failures as localized swellings followed by a perforation resulting in partial, local fluid channel blockages. Discussions at periodic meetings held as part of the Fuel Rod Failure Program have stressed the need for both analytical predictions and experimental measurements pertaining to the effect of these blockages on cooling capability.⁽¹⁾

A series of scoping calculations was carried out at Battelle's Columbus Laboratories with the objectives of estimating the blockage effects on ECCS performance and of defining a critical blockage if it exists. The base-line data used in these calculations were the predicted results of a postulated PWR cold-leg break reported previously in BMI-1871⁽²⁾. Two local-radial power factors were studied, 2.31 and 1.61, using the MUCHA-1 computer code for the loop-flow case as defined in BMI-1871. To summarize briefly, the MUCHA-2 code is used to predict the upper and lower plenum pressures and enthalpies as functions of time, based on the leak definition and undamaged loop resistance assumptions. The plenums are coupled by a core

* Work performed under AEC Contract W-7405-eng-92.

consisting of two parallel, heated channels and an unheated bypass, one channel slightly above core average in power generation rate representing approximately one-half the core flow area, and the other slightly below average representing the remainder. In MUCHA-2 the plenum pressures and enthalpies are calculated by the code, considering the feedback from the core heat generation. The resultant plenum histories, corrected for elevation to give core entrance and exit values, are then used as input into MUCHA-1 where only two parallel heated channels coupled at their ends or plenums are considered. In other words, the channel flows are now based on previously defined boundary conditions and channels with higher peaking factors, of which there are fewer in the core, are studied, thereby implicitly assuming that their integral contribution to the mean plenum conditions is small. In this study the plenum boundary conditions obtained through the MUCHA-2 calculations with no geometric distortions were used to study flow blockage effects on channels with local-radial power factors of 2.31, corresponding to the maximum-power rod, and 1.61. It is estimated that approximately 4 percent of the fuel rods in a typical PWR core would have power ratings in excess of that value. Corresponding axial midplane power generation rates are 18.4 and 12.9 kw per ft for these cases. In presenting the results, the "normal" rod curves are included for comparison and are those obtained for identical power ratings and inlet conditions with no flow blockage (BMI-1871).

In the parallel channel formulation in MUCHA, no interchannel coupling is permitted except at the ends, or plenums. Hence it is assumed that pressure drops are equal for each channel, thus no cross flow or mixing at various elevations in the core is analyzed or considered. The anticipated effect of a flow area restriction, then, is an overall reduction or choking of the coolant flow for that particular channel and subsequent accelerated cladding heatup rates, as the heat generation is not reduced. In this regard, a pessimistic appraisal of the blockage effects on the overall channel behavior would be anticipated although interchannel mixing

could affect very localized areas near the blocked region, an aspect that is being investigated experimentally in the PWF-FLECHT Program.

The blockages were characterized in MUCHA-1 as having a calculational effect similar to a spacer grid assembly; that is, a flow resistance between fluid control volumes. in MUCHA.

$$\Delta p = \frac{KG^2}{2g_c \rho}$$

where

- Δp is the local pressure drop
- G is the local mass velocity
- g_c is the gravitational constant
- ρ is the local fluid density.

The K-factor in this relationship is a function of Reynolds number through an equivalent drag coefficient and the ratio of reduced area to normal flow area. It is through the variation in the K-factor at a 4-ft core elevation that the blockage was calculationally represented. A coolant channel is represented in MUCHA-1 by six axial control volumes.

The K-factors were determined using the relationship developed for spacer grids by de Stordeur⁽³⁾. In his work, drag coefficients were measured as a function of Reynolds number and were reasonably constant for values greater than 7×10^4 ; hence, a constant value was used in these calculations. To place the restrictions investigated into perspective, a typical spacer grid represents a flow area reduction of approximately 26 percent, resulting in a K-factor as used in MUCHA of 0.82. The restrictions investigated were area reductions of 62 percent (corresponding to 34 percent diametral expansion and contact of adjacent rods), 90 percent, and 95 percent resulting in K-factors of 7.1, 153, and 645, respectively. The restrictions were introduced beginning at an accident time of 10 sec when the axial midplane cladding temperatures were estimated to be approximately 1800 and

1300 F for the 2.31 and 1.61 power-factor rods, respectively. The restriction was characterized by a linear increase in K-factor over a two second interval. Cladding temperature thresholds for swelling have been analytically predicted to be about 1400 F⁽⁴⁾ and failures in multirod tests have been observed at about 1740 F⁽⁵⁾.

The results of the calculations are presented in terms of mass flow and cladding temperature comparisons in Figure 1 through 4 for the two thermal power factors studied. For both channels, the results indicated little effect on either flow or temperature for the 62 percent case and the calculations were terminated. For the 90 percent case, a decrease in flow by about a factor of two was predicted, resulting in a delay in peak temperature turnover of about 15 sec and an increase of about 300 F in its value for the maximum-power rod. Similar trends, but of a lesser magnitude, were predicted for the lower-power rod; in both cases a temperature turnover at the axial midplane and a lower elevation quench were predicted. For the 95 percent blockage case, a temperature turnover was not predicted, although a lower elevation quench was predicted for the lower-power rod. The predicted hot-spot temperatures were in excess of 2600 F at the termination of the calculations at an accident time of 48 seconds.

Since cladding temperatures well in excess of 2000 F were predicted by these scoping calculations, the amounts of cladding reacted due to exothermic Zircaloy-water reactions were of interest. The NUCHA code includes a metal-water reaction model and calculates the additional heat due to metal-water reaction throughout the transient. Figure 5 shows the predicted volume of cladding reacted for each case. For the lower-power rod, only the 95 percent case is shown as the fraction is less than 1 percent at the termination of the calculations. For the maximum-power rod, the 90 percent case curve in Figure 5 indicates that a maximum in reaction rate has been passed and since the lower elevations have quenched, the volume reacted would be expected to asymptotically approach a limit. For the 95 percent case, the increase would be expected to be limited by the amount of steam available for reaction as the temperatures continue to increase. The fraction reacted at an accident time of 45 sec was 6.5 percent, at which time the

corresponding surface temperature was 2660 F for the maximum-power rod.

The predicted axial temperature profiles at an accident time of 42 sec are compared in Figure 6. No marked effect of the flow restriction being located at 4 ft is apparent, since the MUCHA code does not consider cross flow and would predict only an overall channel flow reduction. The effects of the blockage on local pressure differences are not pronounced under the same boundary conditions or core pressure differences used for the series. In these calculations, the core Δp used as input is less than 1 psi for the early portions of the transient and is between 1 and 2.5 psi for the period beyond 25 seconds when increased steam generation rates are predicted as a result of temperature turnover and lower elevation quenching.

It was concluded from these scoping calculations that flow area reductions up to nearly 90 percent would not present an untenable situation in regard to ECCS performance. The major uncertainty in the calculations is the proper representation of the geometric distortions in terms of flow resistance. The observed failures in both single and multirod⁽⁵⁾ tube burst tests have consisted of bulges about 2 to 3 tube diameters long, hence the characterization of the deformation as axially local in nature appears warranted. The effects of the shape on the flow resistance K-factor are uncertain as are the possible local cross-flow effects; however, the range of area reductions from 62 percent to 90 percent corresponds to K-factors as used in the calculations of 7.1 to 153, a wide band. Initial pressurized deformation tests in a 13-rod cluster showed that in scattered instances individual subchannels could be completely blocked, but the maximum blockage observed at a given elevation averaged over 12 subchannels has been 61 percent⁽¹⁾. Even though these tests were for BWR diameters and spacing, the area reductions are presumed to be representative of square-array assemblies. The average blockage over a few subchannels at a given elevation is believed to be the characteristic most likely to affect overall cooling; since this maximum

value has been in the neighborhood of 60 percent, it provides a considerable margin for uncertainties in the analytical deformation characterization, and further strengthens the conclusion that local deformations of the type observed experimentally should not pose formidable ECCS cooling problems. Flow tests that are planned for distorted multirod arrays as part of the Fuel Rod Failure Program should provide additional information in regard to the geometrical flow resistance characterization. K-factors obtained from these arrays, although obtained from single-phase fluid conditions, can then be compared to those used in the scoping calculations to ascertain if the K-value of about 150 actually represents a 90 percent blockage and hence an effective cooling limit, or additional calculations using the experimental values can be performed to more accurately predict the cooling effects.

FIGURE 1. EFFECTS OF FLOW AREA REDUCTIONS ON CORE FLOW FOR A MAXIMUM-POWER ROD

FIGURE 2. SENSITIVITY OF CLADDING TEMPERATURE HISTORY TO GEOMETRIC CHANGES FOR A MAXIMUM-POWER ROD

FIGURE 3. EFFECTS OF FLOW AREA REDUCTION ON CORE FLOW FOR A 1.61 POWER-FACTOR ROD

FIGURE 4. CLADDING TEMPERATURE SENSITIVITY TO GEOMETRIC DISTORTIONS FOR A 1.61 POWER-FACTOR ROD

FIGURE 5. FRACTION OF CLADDING REACTED AT THE AXIAL MIDPLANE FOR VARIOUS GEOMETRY DISTORTIONS

FIGURE 6. AXIAL CLADDING TEMPERATURE PROFILES AT AN ACCIDENT TIME OF 42 SEC

References

- (1) Cottrell, Wm. B., "ORNL Nuclear Safety Research and Development Program. Bimonthly Report for November-December 1969", ORNL-TM-2829 (March, 1970).
- (2) Carbiener, W. A., and Ritzman, R. L., "An Evaluation of the Applicability of Existing Data to the Analytical Description of a Nuclear-Reactor Accident. Quarterly Progress Report for July through September, 1969", BMI-1871 (October 1969).
- (3) de Stordeur, A. N., "Drag Coefficients for Fuel Element Spacers", NUCLEONICS, Vol. 19, No. 6, p 74 (June 1961).
- (4) Carbiener, W. A., and Ritzman, R. L., "An Evaluation of the Applicability of Existing Data to the Analytical Description of a Nuclear-Reactor Accident. Quarterly Progress Report for October through December, 1969", BMI-1877 (January 1970).
- (5) Cottrell, Wm. B., "ORNL Nuclear Safety Research and Development Program Bimonthly Report for September-October 1969", ORNL-TM-2777 (January 1970).