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HEAT TRANSFER CONSIDERATIONS FOR THE FIRST NUCLEAR BLOWDOWNS

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ABSTRACT

The first nuclear blowdowns were carried out in the Power Burst Facility at the Idaho National Engineering Laboratory as the LOC-11 series of experiments[1]. This test series was designed to simulate a blowdown transient in a pressurized water reactor (PWR) so that nuclear fuel performance could be investigated under conditions representative of the PWR 15 x 15 fuel element design.

Post-test calculations using the RELAP4 computer program[2] were performed for the LOC-11B and LOC-11C tests. Comparisons between calculations and experimental data revealed that the ability to accurately model (1) critical heat flux (CHF) during low core flow conditions, (2) initial stored energy in the fuel rods, and (3) radiative heat transfer between fuel rods and shrouds, was required to adequately represent the fuel rod thermal behavior. Pre-test calculations performed using RELAP4 with licensing-type heat transfer and fuel rod models resulted in peak cladding temperatures several hundred K higher than measured, thus providing further evidence of the need to accurately model heat transfer and fuel rod behavior.

INTRODUCTION

This paper presents results of the first nuclear blowdown tests (LOC-11A, LOC-11B, LOC-11C) ever conducted. The Loss-of-Coolant Accident (LOCA) Test Series is being conducted in the Power Burst Facility (PBF) reactor at the Idaho National Engineering Laboratory, near Idaho Falls, Idaho for the Nuclear Regulatory Commission. The objective of the LOC-11 tests was to obtain data on the behavior of pressurized and unpressurized fuel rods when exposed to a blowdown similar to that expected in a pressurized water reactor (PWR) during a hypothesized double-ended cold-leg break. The data are being used for the development and assessment of analytical models that are used to predict the thermal and hydraulic response of a PWR during a blowdown transient.

TEST DESCRIPTION AND CONDUCT

The tests were conducted with four, separately shrouded, PWR-type fresh fuel rods. The fuel rods were of 15 x 15 design, except for the active

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length, which was 0.91 m. The fuel rod plenum volume was scaled proportionally to the active fuel length. Two rods were initially pressurized to 0.1 MPa (Rods 611-1, 611-4) and one each to 2.11 MPa (Rod 611-3) and 4.82 MPa (Rod 611-2). (However, the 4.82 MPa rod contained a small leak and its posttest pressure was 1.0 MPa). A fluted flow shroud was selected to minimize the chance of complete flow blockage if uniform ballooning occurred. The coolant flow area was about twice the value associated with a single PWR rod. Four screws, located at two axial elevations, centered each fuel rod. Figure 1 illustrates the blowdown system and a test fuel rod within a fluted shroud.

Valves were used to isolate the experimental hardware from the PBF loop coolant system and thereby provide a controllable flow path during blowdown. Test conduct began with PBF loop isolation from the in-pile tube and a simultaneous reactor scram. Blowdown then commenced and was controlled by quick ($\sim$ 100 ms) opening blowdown valves. Valve operation was controlled by a time sequential programmer. The break planes were formed by converging-diverging nozzles with a cylindrical throat section having equal length and diameter measurements. The design was patterned after that used in the Semiscale program at the Idaho National Engineering Laboratory to optimize predictive capability. The throats were sized to control the flow and depressurization rates. The coolant ejected from the system and the fission products carried from the fuel were collected in a blowdown tank. A quench system provided coolant for terminating the cladding temperature excursion and ending the test.

The fuel rod instrumentation consisted of cladding surface thermocouples, fuel centerline thermocouples, linear variable differential transformers, plenum pressure transducers, and plenum temperature thermocouples. The test train instrumentation consisted of flow turbines located at each end of the fuel rod flow shrouds, coolant temperature thermocouples, coolant pressure transducers, and thermocouples on the outer surface of the fuel rod flow shroud.

Piping measurement spools were installed for determination of the initial inlet and blowdown coolant conditions. Each spool contained temperature, pressure, and flow rate measuring devices. The spools in the blowdown piping also contained a shielded and chopped three-beam gamma densitometer to determine coolant density, and inlet screens to straighten and disperse the flow.

The LOC-11 tests consisted of three separate blowdowns from nuclear power operation. The first test (Test LOC-11A) was conducted after a power calibration, two cycles of full power operation for fuel rod preconditioning, and an additional six hours at full power. Initial test conditions were a coolant inlet pressure of 14.9 MPa, temperature of 591 K, flow rate of 0.91 ℓ/s, and a peak power of 39.1 kW/m. Spurious system blowdown and isolation valve cycling occurred because of an inductive feedback from a liquid level indicator in the blowdown tank interrupting the electrical signals required to activate proper valve sequencing. As a result, additional coolant entered the blowdown system from the PBF loop, thus delaying the onset of critical heat flux (CHF) for six to eight seconds after blowdown initiation. Peak measured cladding temperatures did not exceed 830 K. Test LOC-11A served as a facility checkout test and is not considered further.

Tests LOC-11B and LOC-11C were conducted with axial peak powers of 45.5 and 69.9 kW/m, inlet coolant pressures of 15.2 and 15.3 MPa, inlet coolant temperatures of 593 and 596 K, and flow rates per rod of 0.99 and 0.98 ℓ/s, respectively. During Test LOC-11B, blowdown system siolation and reactor scram occurred at time zero, with one blowdown valve opening in the hot- and cold-leg piping at about 0.9 second, as planned. The delay in valve opening allowed for a 0.9-second stagnation period prior to blowdown. Blowdown was programmed to begin about 0.1 second after isolation and reactor scram during Test LOC-11C, rather than the 0.9-second delay used for Test LOC-11B. CHF occurred 3.2 seconds after isolation during Test LOC-11B, and peak measured cladding temperatures reached 880 K. During Test LOC-11C, CHF occurred 1.6 seconds after isolation, and the peak measured cladding temperatures reached 1030 K.

EXPERIMENTAL DATA COMPARISONS

Pretest calculations were performed for the LOC-11 Test at the originally-planned peak power of 55 kW/m, using the RELAP4/MOD5 computer program. In these calculations, a number of the assumptions required for licensing of commercial nuclear reactors were employed. (The principal effects of these assumptions were to force early CHF, to calculate a high value for initial stored energy, and to neglect radiative heat transfer from fuel rod to shroud).

Post-test calculations using a special version of the RELAP4/MOD6 computer program[a] were made at the actual test conditions, as far as possible, and employing assumptions intended to accurately model the heat transfer and fluid flow processes occurring within the reactor system. The following paragraphs discuss comparisons between experimental data and RELAP4 calculations.

Critical Heat Flux

Figure 2 compares measured values for cladding surface temperature, shroud inlet and outlet volumetric flow rate, and cladding elongation for Test LOC-11C. It is seen that the CHF condition, as indicated by a sudden rise in cladding temperature and a sudden elongation of the cladding, occurs at the second occurrence of zero flow within the shroud.

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[a] RELAP4/MOD6, Update 4, EG&G Idaho, Inc. Configuration Control Number H003321B.

In the RELAP4/MOD5 pretest calculations, the calculated occurrence of CHF was at the initial occurrence of zero flow. This was because the CHF correlations used were based on experimental data at high flow, and extrapolating these correlations to low flow gave a too-small CHF value. The licensing assumptions made in this calculation inhibited any tendency there may have been to return to nucleate boiling. In the RELAP4/MOD6 program, the CHF logic was improved over that of RELAP4/MOD5 by using the modified Zuber correlation[3] for low-flow conditions. With RELAP4/MOD6, the CHF condition was calculated to occur at 1.2 s, and the second occurrence of zero flow was calculated to occur at 1.4 s. In RELAP4/MOD6, for mass fluxes greater than or equal to 1356 $kg/m^2 \cdot s$, the high-flow correlations (the W-3 correlation[4] for subcooled fluid, the Hsu-Beckner modification[5] of the W-3 correlation for saturated and two-phase fluid) are used. For mass fluxes less than 271 $kg/m^2 \cdot s$, the low-flow correlation (modified Zuber) is used. For mass fluxes between 271 and 1356 $kg/m^2 \cdot s$, interpolation on mass flux between the appropriate high-flow correlation, evaluated at 1356 $kg/m^2 \cdot s$, and the low-flow correlation is used. Reduction of the value of mass flux below which the low-flow correlation is used to 67.8 $kg/m^2 \cdot s$ did not appreciably improve the time of CHF in the RELAP4 calculations.

The modified Zuber correlation has no mass flux dependence and has only a moderate pressure dependence. The principal parameter in the correlation is the void fraction. Since the development of void fraction depends on the rate of phase separation, the ability of the RELAP4 slip model to calculate phase separation effects is of significance. The early occurrence of CHF in the RELAP4 calculations indicates calculation of a more rapid rate of void formation than actually occurred. Sensitivity studies, in which adjustments are made to the RELAP4 slip model parameters, will help to evaluate whether improved calculation of CHF time can be achieved by simple adjustment of the RELAP slip model parameters.

Additional understanding concerning the physical processes occurring during CHF in the LOC-11 tests is evidently needed. More experimental data and theoretical investigation will lead to such understanding.

As an illustration of the importance of calculating the timing of CHF, a calculation was performed in which the occurrence of CHF in RELAP4 was forced to coincide with that obtained experimentally. Figure 3 shows a comparison of three RELAP4 calculations, the pretest MOD5 calculation, the post-test MOD6 calculation, and the post-test MOD6 calculation with correct CHF time. The improvement in agreement with improved calculation of CHF time is evident. Thus, it is seen that significant progress in calculating CHF time, relative to RELAP4/MOD5, was obtained in the development of RELAP4/MOD6.

Initial Stored Energy

It is well known that initial stored energy within the fuel rod has a significant effect on cladding temperature during a blowdown transient. However, the standard RELAP4 program cannot model the radial power generation profile that exists in the LOC-11 fuel rods. Therefore, the capability to model a radial power generation profile was added to RELAP4. Also of significance is the centerline thermocouple hole within the LOC-11 fuel rods. An early post-test RELAP4/MOD6 calculation was run neglecting the centerline thermocouple hole, and a later calculation was run considering the hole. Table I compares the early post-test with the final post-test calculated values for centerline temperature with the measured value. As can be noted, the final post-test calculation gave the best value for centerline temperature, indicating the necessity of modeling the thermocouple hole.

Radiative Heat Transfer

In Figure 4, calculated cladding temperatures are compared with experimental data for two calculations: one neglecting radiative heat transfer from fuel rod to shroud, and one in which this effect is considered. It is evident that radiative heat transfer significantly affects cladding temperature and that considering radiative heat transfer significantly improves agreement between calculations and experimental data.

CONCLUSIONS

The results of this study support the conclusion that accurate calculation of core thermal response in nuclear fuel during blowdown requires accurate modeling of critical heat flux under low flow conditions, initial fuel rod stored energy, and, when significant, radiative heat transfer. It is further concluded that additional work concerning modeling of critical heat flux is needed to improve agreement between RELAP4 calculations and experimental data.

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TABLE I

CALCULATED AND EXPERIMENTAL FUEL ROD CENTERLINE TEMPERATURES FOR LOC-11C

Case	Centerline Temperature, K
Post-test MOD6 (no centerline hole)	2675
Post-test MOD6 (centerline hole)	2450
Experimental Data	2500

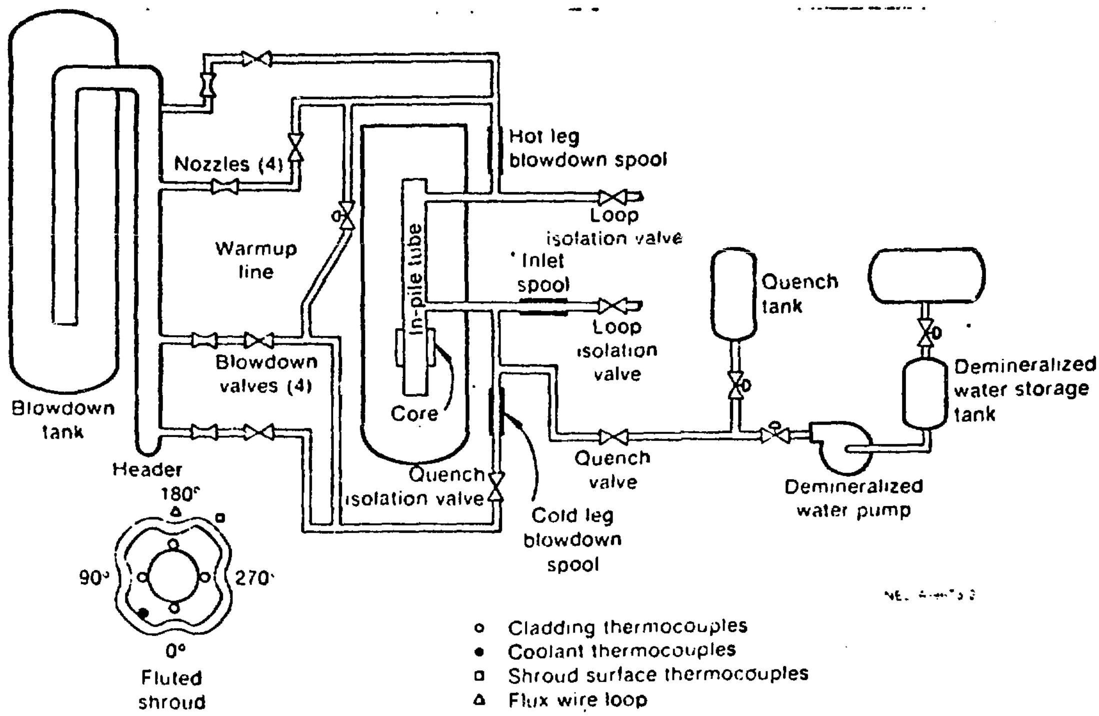

Fig. 1 PBF Blowdown System and Test LOC-11 Fuel Rod Orientation

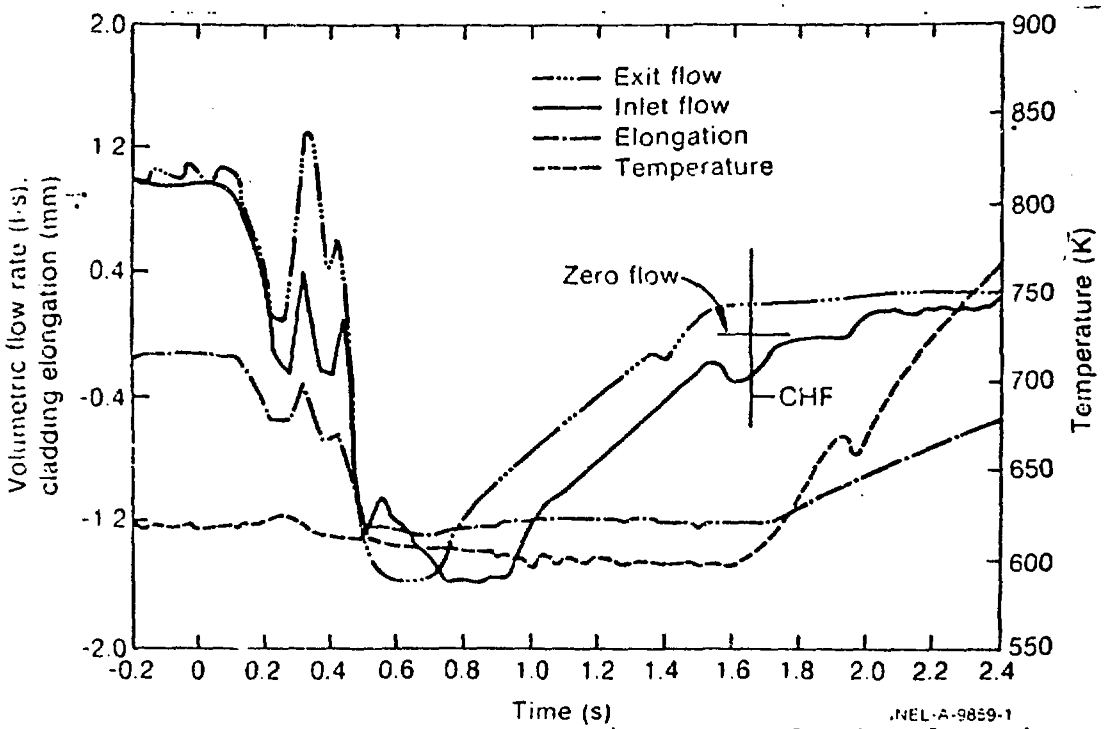

Fig. 2 Comparison of Shroud Volumetric Flow Rate, Cladding Elongation, and Cladding Surface Temperature During LOC-11C

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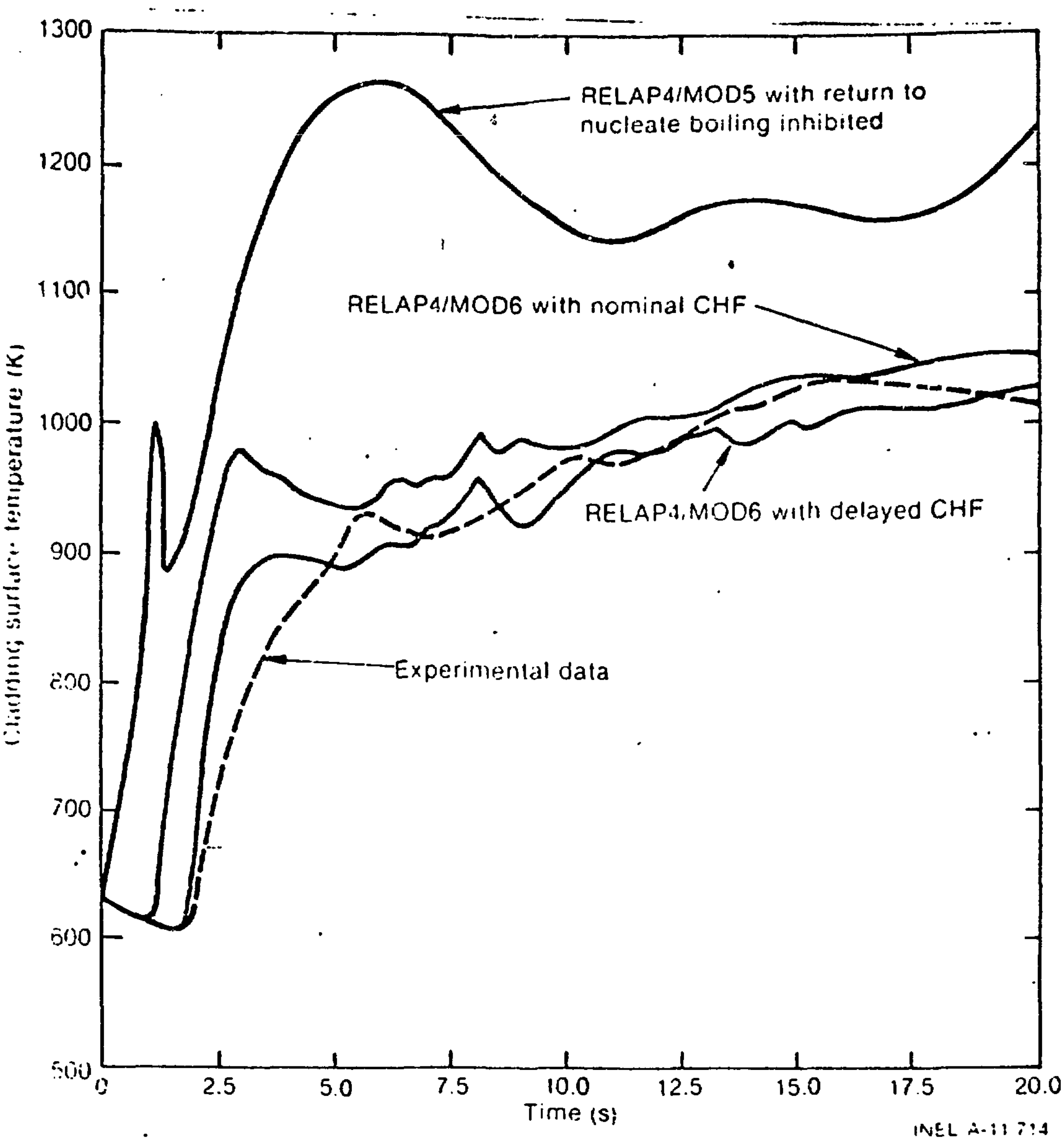

Fig. 3 Effect of CHF Time on Calculated Cladding Temperature

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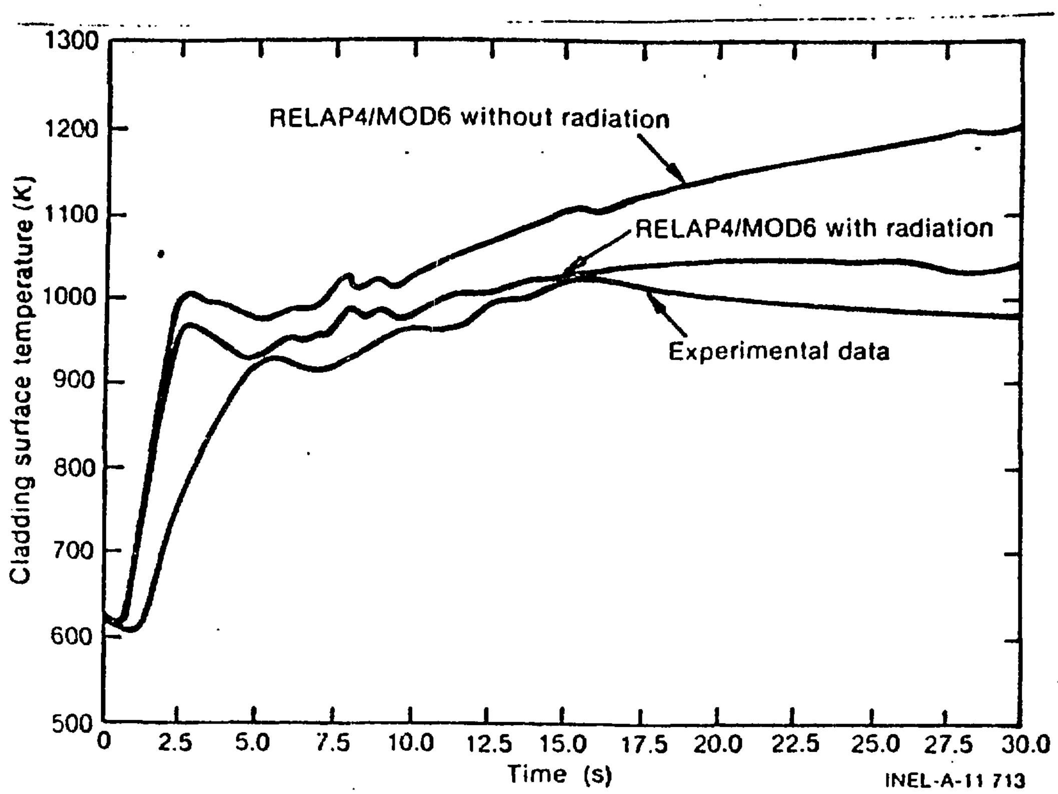

Fig. 4 Effect of Radiative Heat Transfer on Calculated Cladding Temperature