UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 29, 1978



Ivan W. Smith, Esq., Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Oscar H. Paris
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. Lester Kornblith
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555



In the Matter of
TENNESSEE VALLEY AUTHORITY
(Yellow Creek Nuclear Plant, Units 1 & 2)
Docket Nos. 50-566 & 50-567

Gentlemen:

Recently during the course of certain semiscale experiments, conducted at Idaho Nuclear Engineering Laboratory to model various aspects of ECCS performance, one test run exhibited behavior unanticipated by current ECCS performance models.

Semiscale experiment Mod-3, S-A7-6 was run on September 12, 1978. It was intended to model an integral blowdown-refill-reflood scenario for a double-ended cold-leg break. Some of the detailed results were unanticipated. For example, the heated core simulator was predicted by the RELAP code to quench at 110 seconds. Instead, it dried out again and went through several cycles of dryout and rewet (see enclosed Figure 1). Other portions of the cladding temperature profile also showed discrepancies in that test temperatures in some instances were somewhat above those predicted and in some instances were somewhat below those predicted (see Figure 2 and 3). During the test, the downcomer voided several times in the 100 to 400 seconds period of time. This also was not predicted by RELAP (Figure 4 shows one such void). During the periods of downcomer voiding there was also negative (downward) flow from the heater to the lower plenum.

As indicated above, nearly complete downcomer voiding occurred after downcomer fill. This is not predicted by the ECCS evaluation models (for PWR's) used in connection with 10 CFR §50.46 and Appendix K applications. Also, typical Appendix K calculations do not show successive dryout and rewets over the extended reflood cycle. A quick-look report on this experiment will be published by INEL about October 1, 1978.

The present judgment of INEL is that these unanticipated results are atypical and have been produced by certain characteristics of the experiment which are not typical of reactor systems, particularly the stored heat in the downcomer pipe and in the one-dimensional arrangement of the downcomer.

These matters are under further study by the NRC Staff and INEL. When further information or conclusions concerning this matter become available, we will inform the Boards.

Sincerely,

Edward G. Ketchen

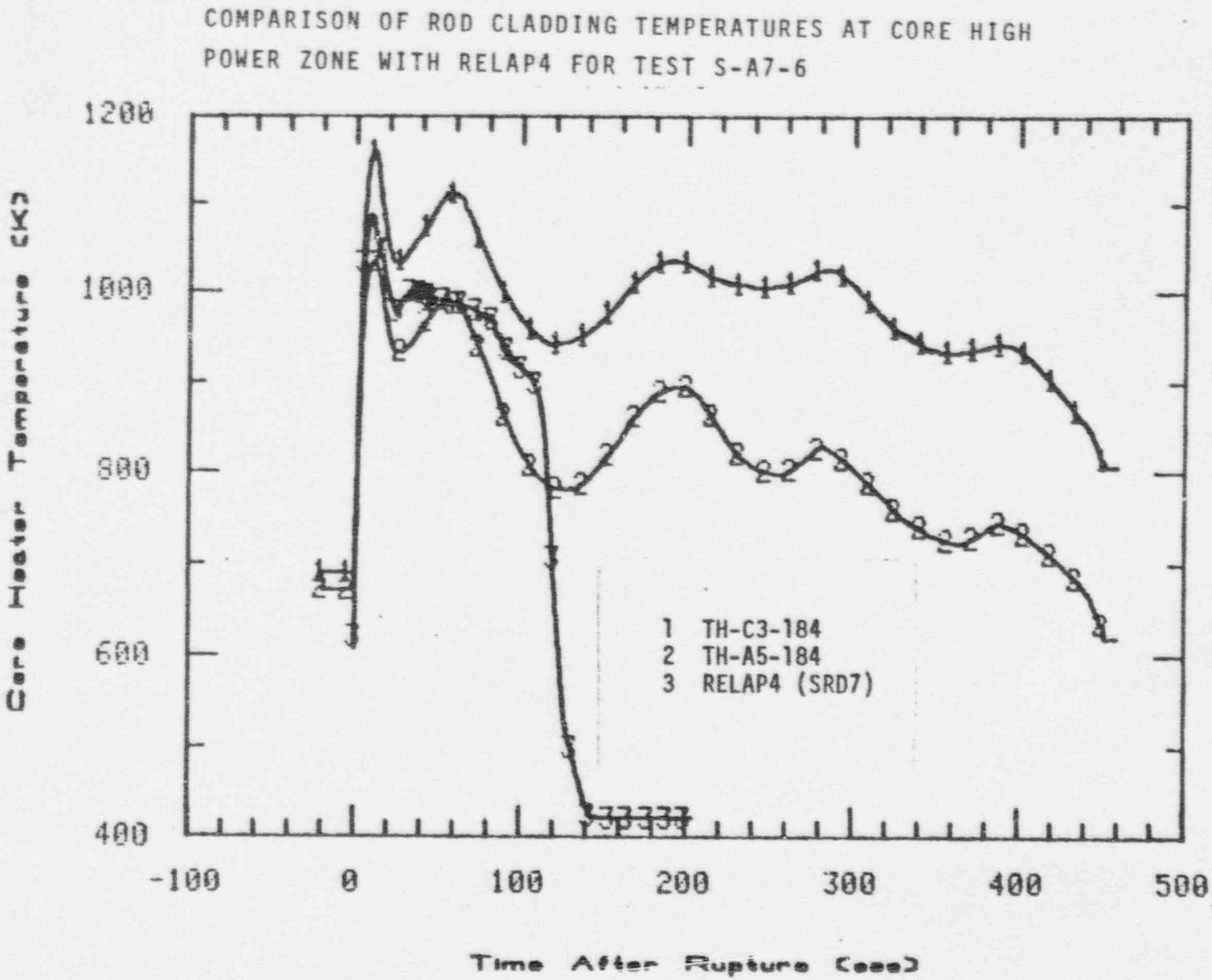
Edward G. Ketchen
Counsel for NRC Staff

Enclosure as Stated

cc (w/encl.):

Ira L. Myers, M.D.
Atomic Safety and Licensing Board Panel
Atomic Safety and Licensing Appeal Panel
Docketing and Service Section
Herbert S. Sanger, Jr., Esq.
Honorable A. F. Summer
Alton B. Cobb, M.D.
William B. Hubbard, Esq.

Figure 1



COMPARISON OF ROD CLADDING TEMPERATURES AT CORE HIGH POWER ZONE WITH RELAP4 FOR TEST S-A7-6
1200
1000
800
600
400
-100
0
100
200
300
400
500
Core Heater Temperature (K)
Time After Rupture (sec)
1 TH-C3-184
2 TH-A5-184
3 RELAP4 (SRD7)

Figure 2

COMPARISON OF MEASURED AND CALCULATED CLADDING TEMPERAUTRE
IN LOWER PART OF CORE FOR TEST S-A7-6

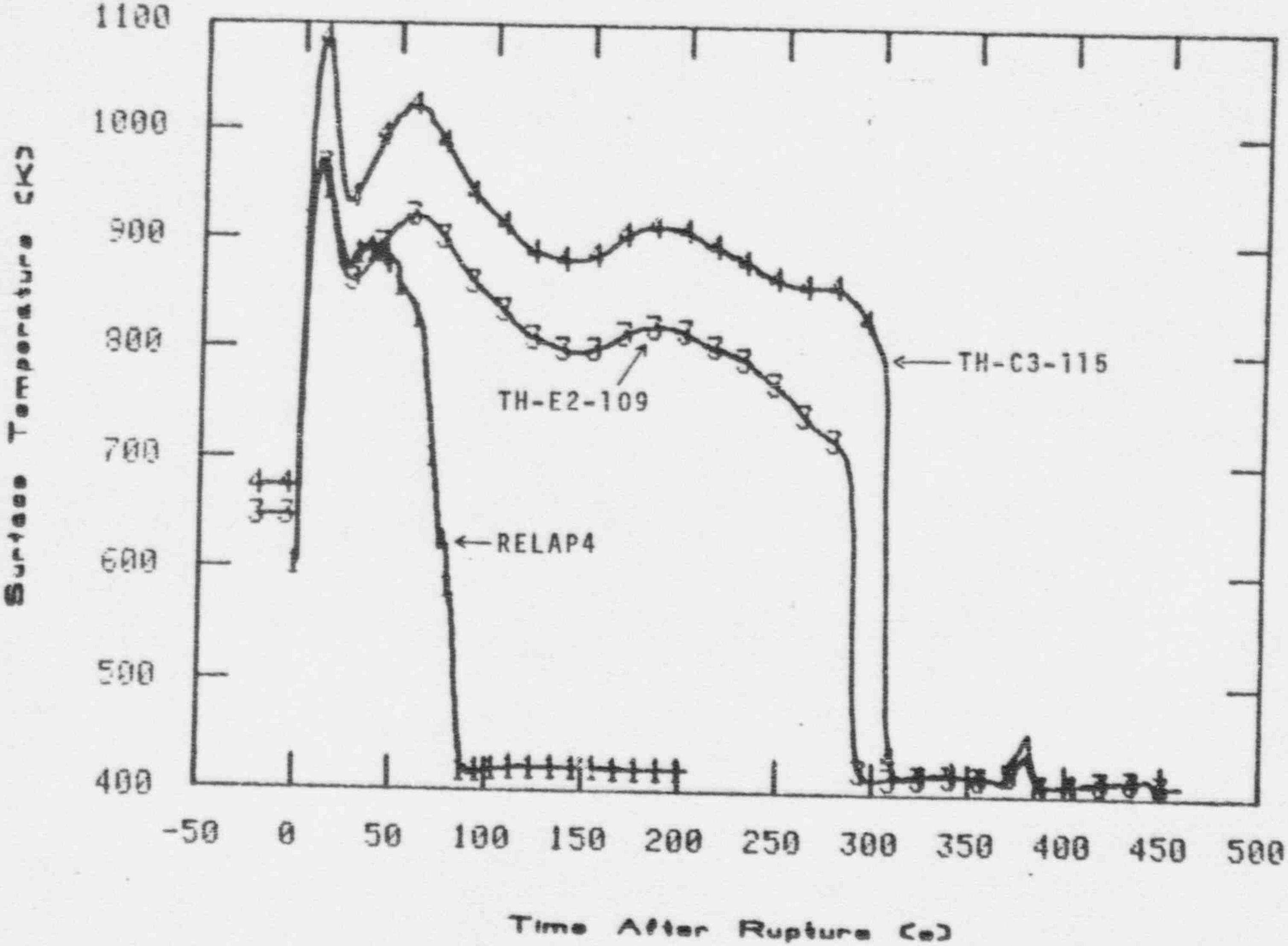
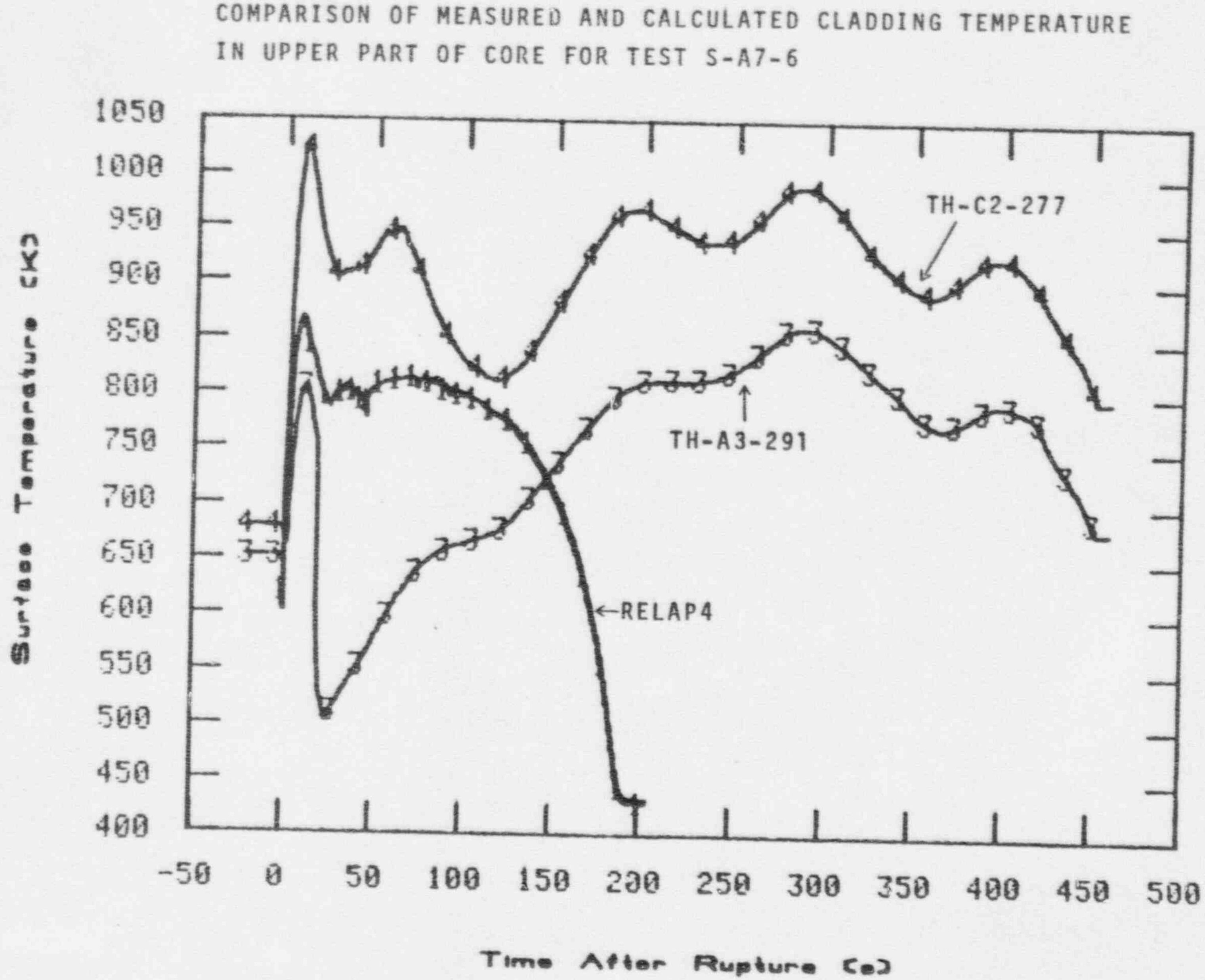


Figure 3



COMPARISON OF MEASURED AND CALCULATED CLADDING TEMPERATURE
IN UPPER PART OF CORE FOR TEST S-A7-6
Surface Temperature (K)
Time After Rupture (s)
TH-C2-277
TH-A3-291
RELAP4
1050
1000
950
900
850
800
750
700
650
600
550
500
450
400
-50
0
50
100
150
200
250
300
350
400
450
500

Figure 4

CALCULATED COLLAPSED DOWNCOMER LIQUID LEVEL FOR TEST S-A7-6