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

NOV 27 1978

MEMORANDUM FOR: Harold Denton, Director
Office of Nuclear Reactor Regulation

FROM: Saul Levine, Director
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER -#39 - RELAP-4/MOD 6

1.0 INTRODUCTION

Many versions of RELAP (REactor Loss of coolant Accident Program) have been written to aid in the investigation of LOCAs.(1-8) RELAP-4/MOD 6, the latest version to be publicly released, is the subject of this Research Information Letter. MOD 6 retains all of the capability of prior versions and, in addition, contains new Best Estimate (BE) blowdown heat transfer models, BE reflood capability, and other modifications. These are summarized below and are covered in detail in the enclosures.

RELAP-4/MOD 6 was developed in response to a number of requests, including:

NRR: Provide capability for a PWR statistical LOCA study and for sensitivity studies to assess data pertaining to LOCA rule changes.(9)

ACRS: Improve blowdown and reflood understanding and provide quantitative appraisals of ECCS.(10, 11)

with additional consideration to comments of others, such as the American Physical Society comments on quantification of LOCA and ECCS to provide realistic calculations as opposed to setting upper limits.(12)

Formal information of this type was supplemented during code development by continuous contact between the cognizant RES project engineer and NRR technical personnel.

2.0 DISCUSSION

Prior versions of RELAP were intended primarily for EM type calculations with an additional capability for Best-Estimate analysis of blowdown. RELAP-4/MOD 6 was intended to extend the Best Estimate capability to include reflood. The following major additions and improvements, as compared to RELAP-4/MOD 5, were introduced:

- Improved blowdown heat transfer correlations and modeling
- Reflood heat transfer correlations and modeling
- Improved heat transfer logic
- Steam generator natural convection heat transfer correlation for the secondary side
- Best estimate fuel models
- Core superheat model during reflood
- Moving mesh for fuel rod heat conduction calculations
- Implicit and explicit entrainment models, as explained below
- Upper plenum liquid de-entrainment model
- Liquid fallback model for the core to upper plenum junction
- Local mass flux model
- Other improvements (such as addition of the revised Burnell homogeneous equilibrium critical flow model option, improvements in slip calculation, better handling of computer memory, special features such as adjustable coefficients for use in sensitivity and uncertainty studies, trip logic modifications, improved hydrodynamic equation time advancement procedures, and a prescribed best estimate calculation procedure.

The principal additions to and changes in MOD 5 which resulted in MOD 6 are covered in Section 3. Further detail is provided in the MOD 6 manual (17) (enclosed).

As part of this discussion it is important to point out that (a) this code cannot be used for a continuous calculation of a complete (integral) PWR LOCA, mainly because of the difficulty in handling the intermediate (refill) stage; (b) the achieved capability to analyze PWR reflood has not met all our expectations; and (c) the code cannot be used to analyze BWR reflood. More will be said later about these inadequacies.

3.0 RESULTS

3.1 HEAT TRANSFER

New heat transfer logic has been formulated and modularized in MOD 6. An iterative technique is utilized which starts with selection of the wetted surface temperature which, in conjunction with the known fluid properties, allows selection of the correct heat transfer correlation and calculation of heat deposition into the fluid. The surface temperatures are iterated until the calculated fluid convection heat flux is sufficiently close to the conduction heat flux within the solid.

The MOD 6 heat transfer correlations are arranged as illustrated in Figures 1 and 2. Each correlation is utilized over the applicable range of wall and fluid conditions, with suitable interpolations to smooth mis-matches between heat transfer regimes.

These correlations have been selected as a result of several meetings between the staff, consultants, and vendors,* as well as from recommendations of NRC experts.(18, 19)

3.2 REFLOOD MODELING

The major new addition in MOD 6 involves modeling of PWR reflood. The core is divided into several parallel flow channels containing fuel rods, each representing a radial core section (typically hot, average, and cold regions). Local behavior of fluid within channels is determined by further dividing a channel into axial nodes. Fuel rods are nodalized radially and axially, axial nodes corresponding to the adjacent fluid nodes. Finer axial nodalization is employed in the vicinity of the quench front where large axial temperature gradients occur. This treatment is illustrated in Figure 3.

In core channels, a pseudo-volume is assigned corresponding to each of the moving (heat transfer) mesh nodes to perform a quasi-steady-state heat and mass balance involving local quality, local vapor temperature, and local mass flux, with the assumption that liquid is always at saturation conditions. Fractions of the total energy entering

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* The principal meetings were those held in Idaho Falls (7/21-22/75),(20) Germantown (7/19-20/76), Canada (6/21-22/77) and Atlanta (11/27/77).

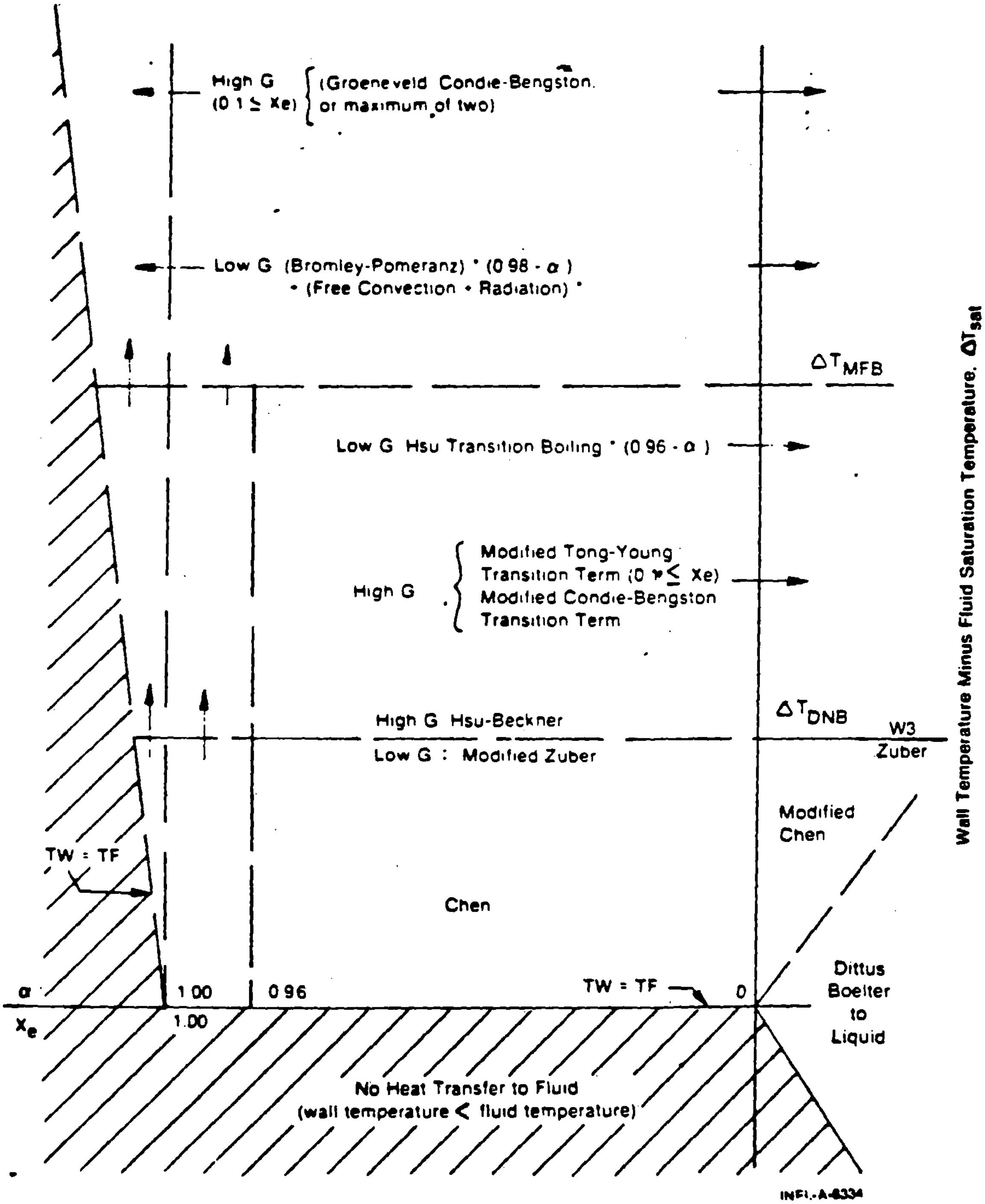

Fig. 1. RELAP4/MOD6 HTS2 blowdown heat transfer correlations and their regions of application.

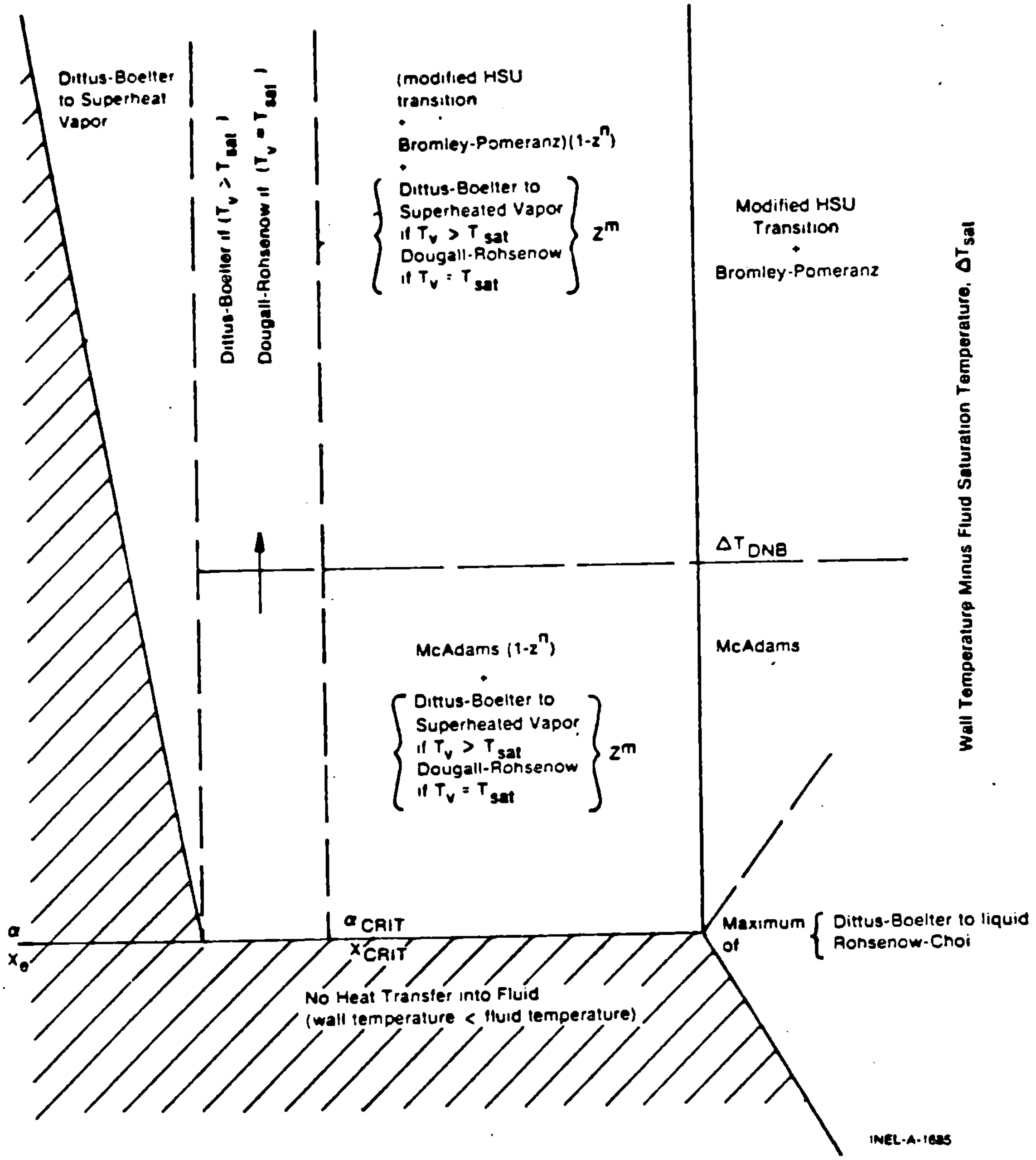

Fig. 2. RELAP4/MOD6 reflood heat transfer correlations and their regions of application.

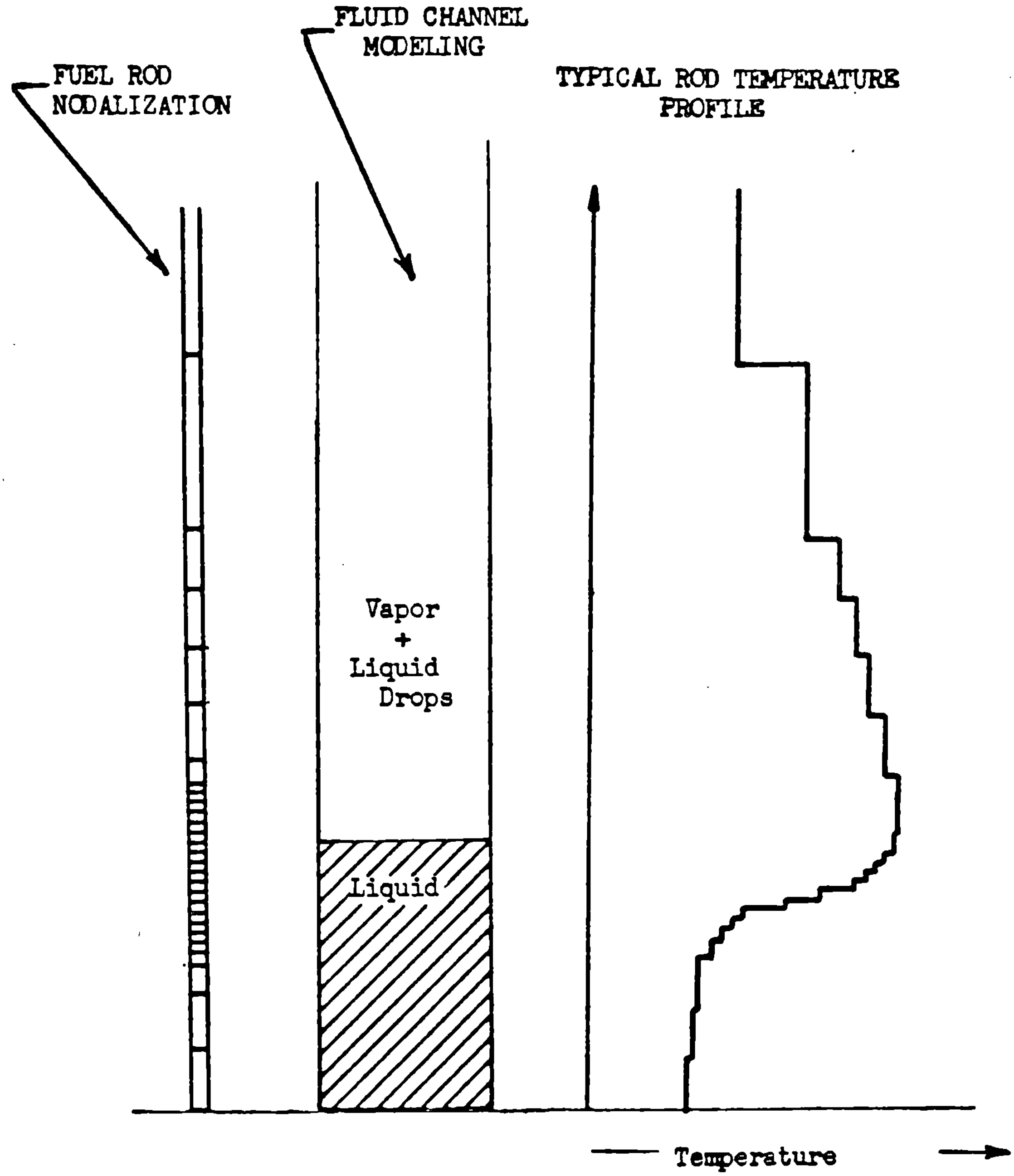

Figure 3. MOD 6 Core Representation

the liquid and vapor phases are either calculated internally or are user specified through input. A practical technique is provided for dynamic partitioning of core heat slabs in the vicinity of the quench front during reflood to more fully represent local conditions. The heat transfer mesh moves as required to follow the propagation of the large axial temperature gradient.

Liquid entrainment (carry-over) during reflood can be modeled either implicitly or explicitly. Implicit modeling is based on a modified bubble rise concept which handles fluid volume conditions including liquid with pure vapor over it (i.e. no entrainment), liquid with a two phase mixture over it, and a homogeneous volume. The last item represents the maximum entrainment situation. The model uses a variety of entrainment correlations to specify the locally entrained liquid fraction at the point of entrainment initiation. The explicit model is equivalent to specification of the carryover rate fraction as in the RELAP-4 FLOOD code.

The upper plenum de-entrainment model considers liquid entrainment from the postulated layer above the core, and de-entrainment due to liquid droplet settling on upper plenum structure. The core exit steam velocity determines, through a CCFL correlation, whether liquid separated with the upper plenum can penetrate the upper core support plate and fall back into the core.

3.3 MOD 6 MODELS NOT PREVIOUSLY DESCRIBED

Other new models introduced in MOD 6 are:

STEAM-GENERATOR NATURAL CONVECTION: Natural convection heat transfer on the secondary side of the steam generator was introduced by employing natural convection correlations for liquid and/or vapor for turbulent or laminar conditions, as determined internally by the code.

BEST-ESTIMATE FUEL MODEL: Considers clad-fuel contact pressure, metal-water reaction, gap pressurization including fuel plenum effects, and fuel expansion. Radial fuel pellet expansion modeling due to aging is that used in FRAP-T3, (21) and the fuel thermal expansion is an adaptation of the modeling employed in MATPRO 07.(22) A conservative metal-water reaction rate model is also available for the EM option.

DYNAMIC STORAGE: Extensive logic and dimensioning changes were made to allow tighter packing of storage and elimination of code subroutines not actually required for the specified calculation sequence.

TRIP LOGIC MODIFICATIONS: Changes were made to allow RELAP to follow logical combinations of several signals to more readily model the trip systems associated with operating reactors.

ADVANCEMENT PROCEDURES FOR HYDRODYNAMIC EQUATIONS: Improvements were made to allow (a) simplified computation of matrix elements, (b) addition of a more accurate advancement algorithm, (c) improvement of the matrix solution technique by using alternating direction for iteration, with direct inversion if the iteration technique does not converge rapidly. Several subroutines were re-written to implement the advancement procedures and to use dynamic rather than fixed storage allocation.

UNCERTAINTY STUDY CAPABILITY: Adjustable (input) coefficients have been added to facilitate uncertainty and sensitivity evaluations.(9)

PHASE SEPARATION: Wilson bubble rise and phase separation models have been added. The Wilson model provides for variable bubble rise velocity based on volume conditions, while the phase separation model yields results synonymous with an infinite bubble rise velocity.

MODIFIED BURNELL/HOMOGENEOUS EQUILIBRIUM CRITICAL FLOW MODEL: This added option utilizes the incompressible form of the Bernoulli equation, with a constraint on back pressure, to calculate subcooled mass flow rates. The homogeneous equilibrium model is used to compute mass flow in the two-phase and superheated flow regimes, with linear interpolation in the transition region between the subcooled and saturated regimes.

VERTICAL SLIP: Provision was introduced to calculate slip in the churn turbulent, transition, dispersed droplet, and dispersed bubble flow regimes.

VARIABLE PUMP MOMENT OF INERTIA: The requirement that the moment of inertia be constant, as coded in MOD 5, was removed.

EXPANDED DEBUG: An option was introduced to assist the user in following flow of the program and locating failure points.

HEAT SLAB NODE TEMPERATURE RESET: This feature allows input setting of all internal heat slab temperatures, as opposed to requiring that the code calculate them by a prescribed procedure.

ROUGH WALL FRICTION: Only the smooth wall assumption exists in MOD 5.

GAMMA HEATING: Allows direct gamma heating of non-fuel heat slabs.

CORE POWER TRIP: A temperature controlled trip was introduced to allow following the power behavior of facilities, such as Semiscale, which have maximum allowable temperature settings that control the power supply to the electrical rods.

ACCUMULATOR POLYTROPIC AIR EXPANSION MODEL: This offers a deviation from the normal RELAP-4 assumption of thermal equilibrium within a fluid volume to allow more accurate calculation of accumulator behavior.

HEAT TRANSFER TIME STEP CONTROL: An option was introduced to uncouple the heat conduction and hydraulic time steps which allows faster calculation.

BUBBLE RISE TRIP OPTION: An option was introduced to avoid excessive calculation time when a volume contains a very small amount of liquid and the bubble rise model has been specified (avoids liquid mass depletion with accompanying excessively small time steps).

3.4 EVALUATION AND APPLICABILITY

Previous RELAP codes were subjected to checkout runs with a series of short, fast running problems, and several LWR calculations were performed prior to release. Only a limited number of comparisons were made with experimental data as part of code development. Two major changes were made in the development approach for MOD 6. First, the code was subjected to a large number of data comparisons prior to release. Second, a program of independent code assessment was initiated when the code was finalized. The first item is reported here; the second will be the subject of a separate report when the independent code assessment task has been completed.

The remainder of this section summarizes experience with MOD 6, assesses the usefulness of the code, and identifies areas where additional work is needed.

Documents which provide a complete description of MOD 6 data comparisons which were performed prior to public release are enclosed.(23-26) These should be referred to for detailed comparisons of calculations with test data. The discussion given below will concentrate on what was learned during the data comparison process.

Bliem's report (23) represents the initial developmental data comparisons performed prior to internal code release in December 1977. It covers comparisons with experimental data from FLECHT, FLECHT-SET, and Semiscale. Good comparisons with data were obtained with run times ranging from good to very good (a few minutes in some cases). The difficulty with Bliem's results is that the entrainment parameters were arbitrarily varied in some cases to provide the best fit to test data.

Davis (24) compared MOD 6 to ORNL THTF test No. 103 data. He obtained relatively accurate comparisons with hydraulic and thermal parameters, but encountered a number of discrepancies with the rewet time. Calculated temperatures were within the experimental error early in the event, but were higher than measured at later times.

Fletcher and Wilson (25) reported on an extensive MOD 6 investigation in which they performed an eight run time-step sensitivity study and a ten run moving mesh study. Following this, code comparisons were made with eight FLECHT tests to determine the effect of the code input specified liquid entrainment and dispersed flow heat transfer parameters. In these cases the code comparisons involved changes in experimental conditions, such as reflood rate, housing temperature, pressure, ECC temperature, power, and initial temperature, to obtain a best estimate specification for code input parameters or options. However, these results are strictly applicable only to the FLECHT forced feed tests, and use of the code for other configurations may give less satisfactory results. The study is also of limited scope in that there is no evaluation of the transition and rewet parameters. In a second series of calculations and comparisons with some of the same data, the authors showed that a good fit to experimental temperatures could be obtained, at selected levels within the core, by variation of the beta term in the Hsu heat transfer correlation. However, when this was accomplished, less satisfactory comparisons were obtained at other elevations. The error was particularly severe for the quench time.

The final data comparisons, applicable to the released version of the code, were reported by Fischer.(26) Thirteen problems were run, including FLECHT-SET, FLECHT-LFT, Semiscale, LOFT, TLTA and PKL. Figure 4 which shows FLECHT-SET quench behavior, is typical of the results. RELAP showed excellent correlation with the experimental upper bound quench times in the lower half of the core, and deviated in the upper core. Also typical are Semiscale test S-03-6 results, illustrated in Figures 5-7, which show quench time and core temperature behavior. Note the mid core behavior in Figures 5 and 6, which show excellent correlation of RELAP with the data, and the discrepancy further up the core as shown in Figure 7. Fischer doesn't provide detail on all of the work, but his report can be supplemented by references (27-30) if more information is desired. Fischer also provides recommendations for the nodalization of a PWR for the 200% cold leg break LOCA analysis.

Calculational error can become significant in some MOD 6 applications, as illustrated in Figure 8 for Flecht Test 0085.(23) This large over prediction of temperature appears to be due to unsatisfactory modeling of heat transfer in the dispersed flow regime.

Conclusions applicable to MOD 6 code assessment are:

1. Good results are generally obtained for blowdown regime. Accuracy in the reflood regime is not as good as desired in all cases.

2. Code run times are variable, ranging from a few minutes for some reflood problems to several (5 to 6) hours for some of the Semiscale blowdown-refill calculations.

3. A preliminary technique has been established for the calculation of reflood in PWRs and in experimental facilities.

4. A number of code weaknesses have been identified. These include:

   - A continuous, integral LOCA calculation cannot be performed with MOD 6, primarily because of poor and unpredictable performance in the refill period. Reflood calculations are performed separately by assuming zero flow in all control volumes at the beginning of reflood.

   - Transition boiling and dispersed flow heat transfer treatment must be improved. This includes a need for more basic data to develop a better understanding of the heat transfer mechanism.

Run number	3105B
Containment pressure	59 psia
Initial clad temperature	1000° F
Peak power	0.84 kW/ft
Average housing temperature	306° F
Coolant temperature	152° F
Injection rate	12.3 lbm/sec for 14 sec Variable to end

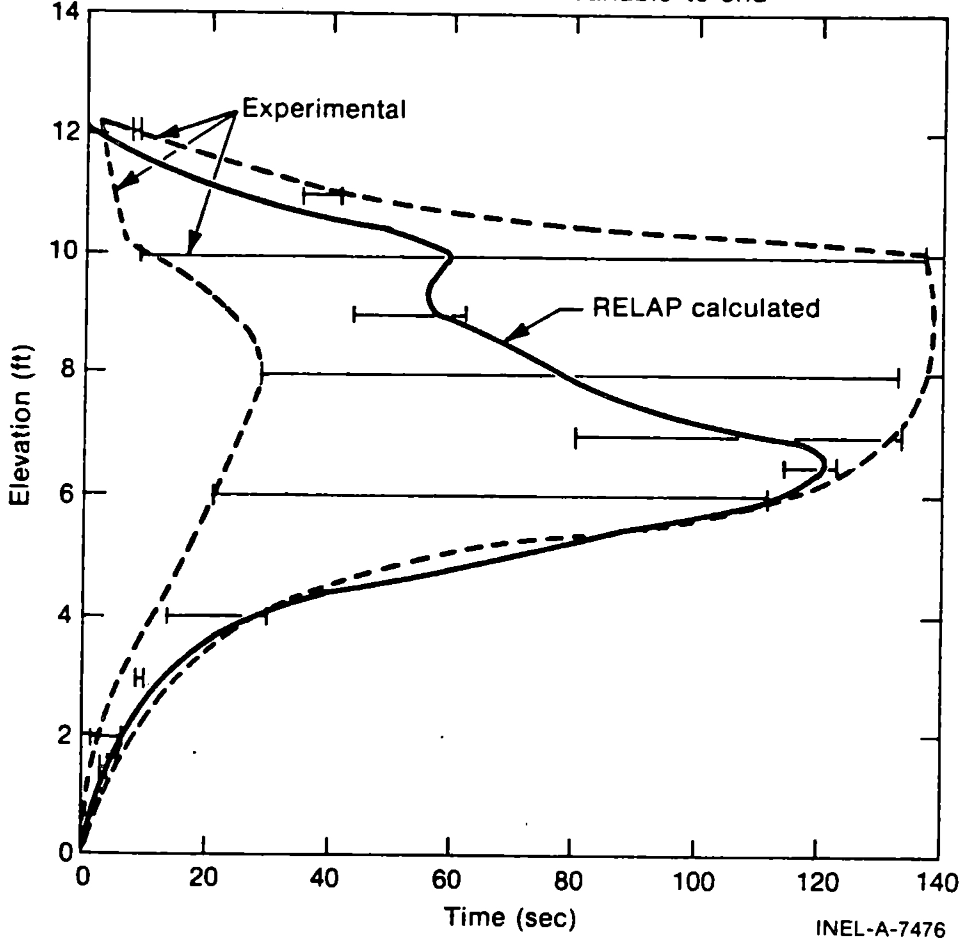

Fig. 4. Quench front for Test 3105B showing maximum and minimum times for each elevation.

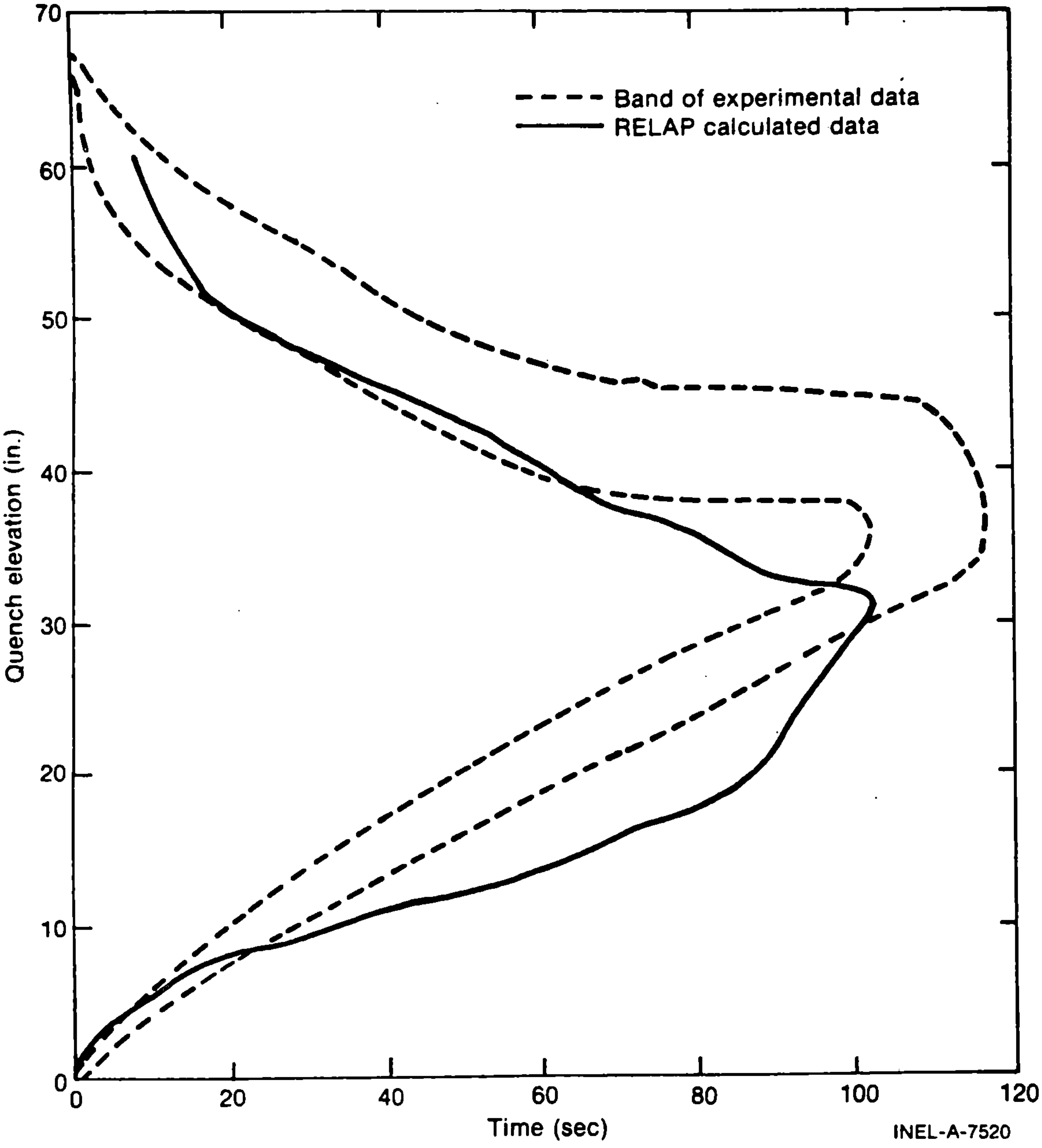

Fig. 5. Quench elevation response.

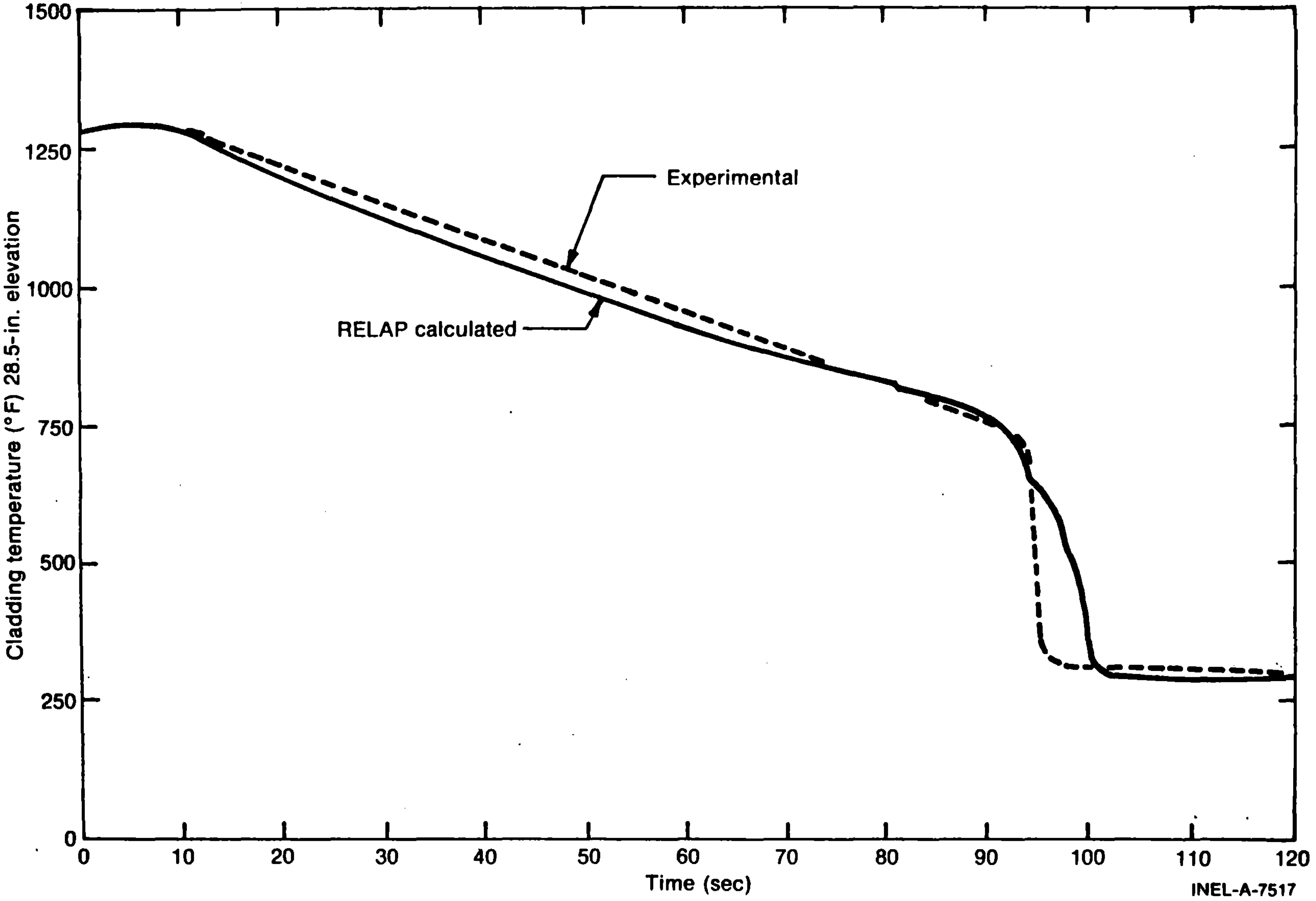

Fig. 6. Clad temperature response at 28.5 in.

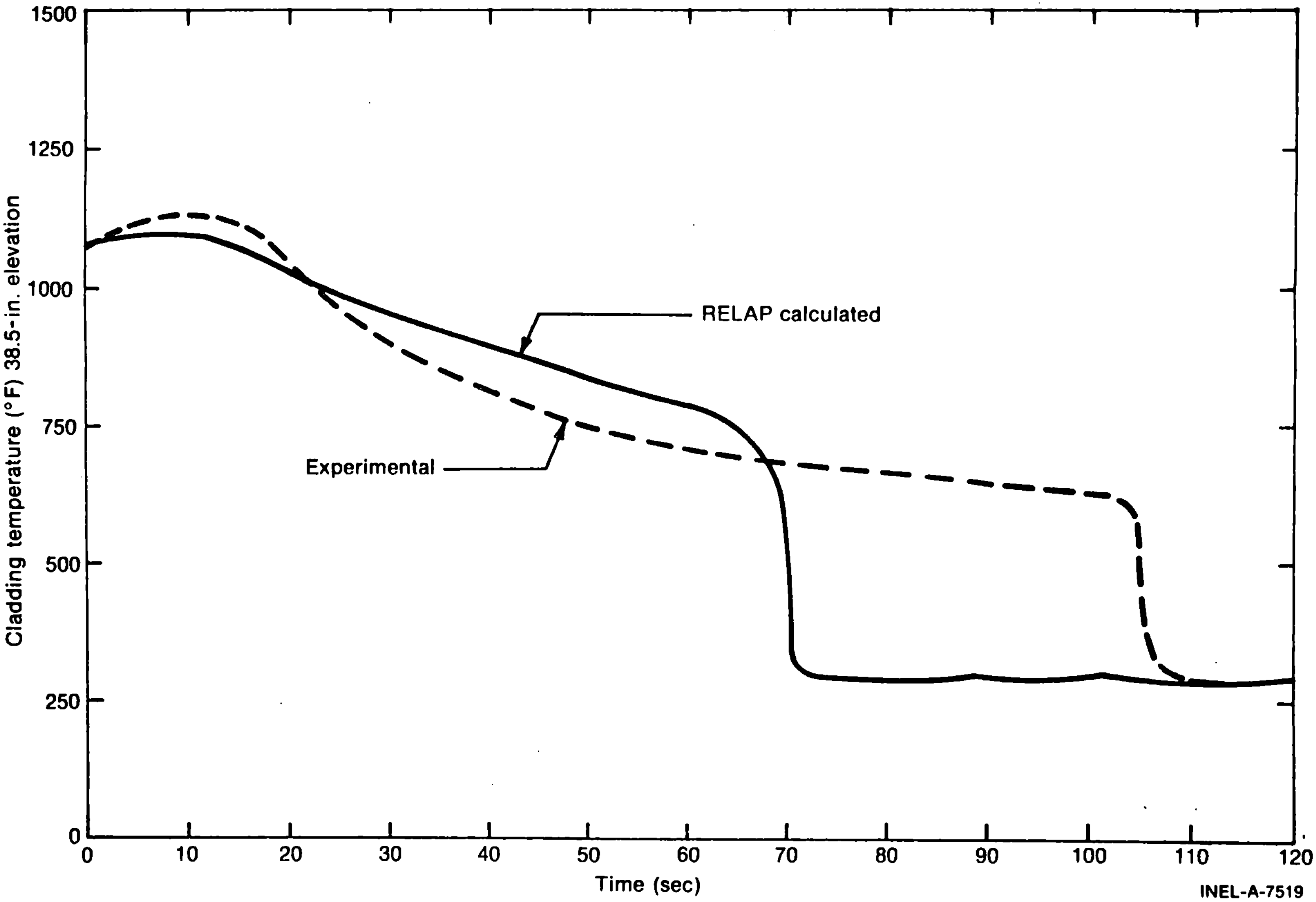

Fig. 7. Clad temperature response at 38.5 in.

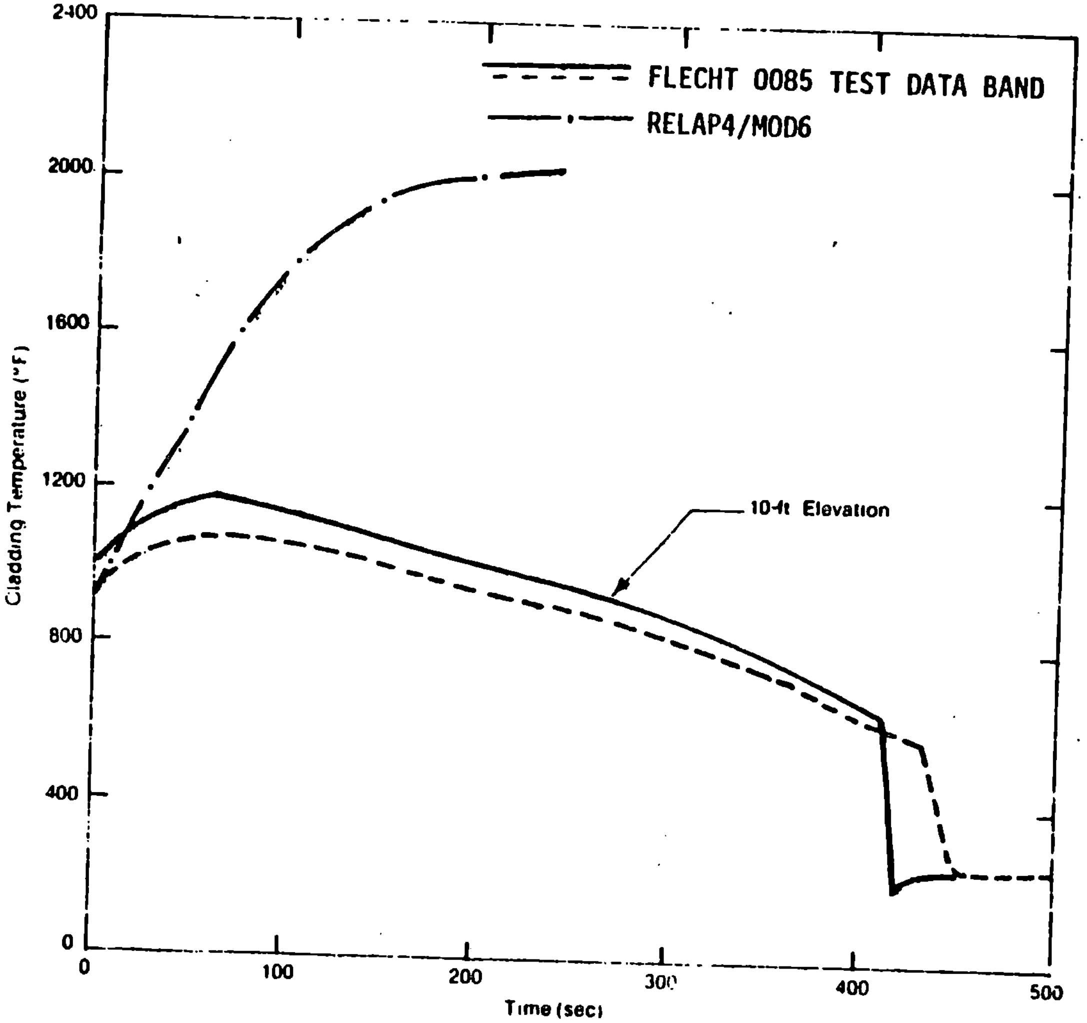

FIGURE 8 ROD CLAD TEMPERATURE HISTORY FOR FLECHT-0085

- The entrainment model must be improved to include local effects and to reduce the number of user selected parameters. User guidelines for the existing model are inadequate.

- De-entrainment and fallback models require better user guidelines. More experimental data are needed to provide a foundation upon which to develop better models.

- The thermal equilibrium assumption introduces particularly severe calculation problems and errors relative to ECC injection and transport

- Modeling of subcooled liquid at the core inlet is necessary

- Reflood initialization is inadequate

- Steam generator heat transfer during refill and reflood requires improvement

- CHF correlations need improvement (CHF is calculated to occur earlier than observed in the upper regions of the core, with a resulting over prediction of core temperature at those locations)

- Downcomer modeling is inadequate.

4.0 RECOMMENDATIONS

MOD 6 is an extension of prior LOCA code capability to allow modeling of LWR and experimental facility reflood phenomena, in addition to calculation of the blowdown phase of the event. RES is applying the code to the uncertainty study requested by NRR,(31) as well as to interpretation of test results from LOCA facilities. The RELAP-4/MOD 6 code is recommended for the following best estimate calculations:

- BLOWDOWN: In general, good temperature calculations will be obtained in the lower and midcore regions, including the hot spot. Agreement between the data and calculation results will not be as satisfactory for locations near the core exit. Most other parameters will be calculated reasonably well.

- REFLOOD: Good results can be obtained if the user properly selects the parameters and options. At present, this selection is an art; an undesirable characteristic which should change as user guidelines improve during independent code assessment.

RELAP-4/MOD 6 is not recommended for refill analyses because of the difficulties associated with the nonequilibrium phenomena which are not modeled. Downcomer modeling is also a problem, and the code does not model nitrogen flow if the accumulator should empty during the ECC bypass and refill phase of the event. As with MOD 5, the code is very slow running during refill, giving calculated results which are not satisfactory. This code is not recommended for steam generator tube break investigations, although user guidelines are being developed to attempt limited investigations with MOD 6.

RES is addressing the MOD 6 limitations, as well as the items identified in the previous NRR request, (9) during the development of RELAP-4/MOD 7.(32) RES is continuing to work closely with NRR technical personnel to provide a RELAP code tailored to NRR needs.

MOD 6 has been sent to the Argonne Code Center, France (NEA for European distribution), Italy, NRC, ORNL and Sandia. It is in use on a number of NRC-funded programs including analysis of Semiscale MOD 3, (13) LOFT,(14) and PKL,(15) and plans are being implemented for its application to Standard Problems.(16) The foreign and domestic recipients of RELAP-4/MOD 6 will be advised of the code assessment.

Saul Levine
Saul Levine, Director
Office of Nuclear Regulatory Research

Enclosures:
1. "RELAP-4/MOD 6, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems," Users Manual, EG&G Idaho, (Idaho National Engineering Laboratory), CDAP-TR-003, January 1978.
2. Bliem, C. J., et al, "RELAP-4/MOD 6, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems," Users Manual, Vol. II, RELAP-4/MOD 6 Developmental Verification, EG&G Idaho (INEL), PG-R-77-06 (Draft), March 1977.
3. Davis, C. B., "RELAP-4/MOD 6 Comparison with PWR-BDHT Test 103 Core Data," EG&G Idaho (INEL), PG-R-77-27, July 1977.
4. Fletcher, C. D., and G. E. Wilson, "Developmental Verification of RELAP-4/MOD 6 UPDATE 1 with FLECHT LFR Cosine Test Data Base," EG&G Idaho (INEL), PG-R-77-24, July 1977.
5. Fischer, et al, "Demonstration Problems for RELAP-4/MOD 6, Update 3," EG&G Idaho (INEL), CDAP-TR-008, April 1978.

See next page for cc's.

cc w/encl: L. Rubenstein
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References

1. Moore, K. V., and W. H. Rettig, "RELAP-2 - A Digital Program for REactor Blowdown and Power Excursion Analysis," IDO-17263, March 1968.

2. Rettig, W. H., et al, "RELAP-3--A Computer Program for Reactor Blowdown Analysis," IN-1321, June 1970.

3. Rettig, W. H., "RELAP-3 - A Computer Program for Reactor Blowdown Analysis," IN-1445, Feb. 1971.

4. Moore, K. V., "ASTEM-A Collection of Fortran Subroutines to Evaluate the 1967 Equation of State for Water/Steam and Derivatives of These Equations," Aerojet Nuclear Company (National Reactor Testing Station), ANCR-1026, Oct. 1971.

5. Moore, K. V., and W. H. Rettig, "RELAP-4 - A Computer Program for Transient Thermal-Hydraulic Analysis," Aerojet Nuclear Company (National Reactor Testing Station), ANCR-1127 (Rev. 1), Mar. 1975 (Originally published Dec. 1973).

6. Young, R. C., and S. D. Matthews, "PLOTR4M, RELAP-4 Version 3 Plot Processing Program," Aerojet Nuclear Company, no report number, May 1974.

7. "WREM: Water Reactor Evaluation Model," Div. of Technical Review, Nuclear Regulatory Commission, NUREG-75/056 (Rev. 1), May 1975.

8. "RELAP-4/MOD 5, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems," Vols 1, 2, and 3, Aerojet Nuclear Company, ANCR-NUREG-1335, Sep. 1976.

9. Case, E. G., "NRR Requirements for Loss-of-Coolant Accident Analysis Computer Programs (RR-NRR-77-5)," NRC Memo to S. Levine, June 23, 1977.

10. Bender, M., "Status of Generic Items Relating to Light-Water Reactors: Report No. 5," Letter from Chairman, ACRS, to M. A. Rowden, Chairman, NRC, Feb. 24, 1977.

11. "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," Report to Congress, NUREG-0410, Jan. 1, 1978.

12. Lewis, H. W., et al, "Report to the American Physical Society by the Study Group on Light-Water Reactor Safety," April 28, 1975 (also published in Review of Modern Physics).

13. "Appendix 7 to the Semiscale Experimental Operating Specification, Test Series 7, Tests S-07-1 Through S-07-7, Semiscale Baseline Test Series," EG&G Idaho (INEL), WR-S-78-002, March 1978.

14. Wells, M. E., "Code Verification Pre Test Prediction of LOFT L1-5," MEW-9-78, Memo to T. R. Charlton, EG&G Idaho (INEL), April 21, 1978.

15. Dearien, J. A., "Test Prediction for KWU PKL Test K5A Using RELAP-4/MOD 6, Update 3," JAD-81-78, EG&G Idaho letter to R. Tiller (DOE), March 1978 (Recd at NRC-Silver Spring March 9, 1978).

16. Phillips, L., (NRR) personal communication to W. Lyon (RES) regarding communication with Jacques Royen (CSNI Secretariat), May 4, 1978.

17. "RELAP-4/MOD 6, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems," Users Manual, EG&G Idaho, (Idaho National Engineering Laboratory), CDAP-TR-003, January 1978.

18. Hsu, Y. Y., "Best-Estimate Recommendations for Blowdown Heat Transfer," (Draft) NRC/RSR, May 19, 1975.

19. Scroggins, R. M., "Best-Estimate Heat Transfer Correlations," NRC Memo to S. Fabic, June 10, 1975.

20. Eckhard, J. D., Y. Y. Hsu, and H. Sullivan, "Trip Report, RELAP-4 Heat Transfer Correlation and Switching Logic Meeting at ANC," NRC Memo to L. S. Tong and V. Stello, July 21-22, 1975.

21. Dearien, J. A., et al, "FRAP-T3, A Computer Code for the Transient Analysis of Oxide Fuel Rods," INEL, TFBP-TR-194, August 1977.

22. MacDonald, P. L. and L. B. Thompson, "MATPRO 07: A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," ANC, ANCR-1263, February 1976.

23. Bliem, C. J., et al, "RELAP-4/MOD 6, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems," Users Manual, Vol. II, RELAP-4/MOD 6 Developmental Verification, EG&G Idaho (INEL), PG-R-77-06 (Draft), March 1977.

24. Davis, C. B., "RELAP-4/MOD 6 Comparison with PWR-BDHT Test 103 Core Data," EG&G Idaho (INEL), PG-R-77-27, July 1977.

25. Fletcher, C. D., and G. E. Wilson, "Developmental Verification of RELAP-4/MOD 6 UPDATE 1 with FLECHT LFR Cosine Test Data Base," EG&G Idaho (INEL), PG-R-77-24, July 1977.

26. Fischer, et al, "Demonstration Problems for RELAP-4/MOD 6, Update 3," EG&G Idaho (INEL), CDAP-TR-008, April 1978.

27. Bliem, C. J., "RELAP-4/MOD 6 Calculation of U.S. Standard Problem 7 and International Standard Problem 5 (LOFT L1-4)," EG&G Idaho (INEL), PG-R-77-33, June 1977.

28. White, J. R., et al, "Experimental Prediction for LOFT Non-Nuclear Experiment L1-4," EG&G Idaho (INEL), TREE-NUREG-1086, April 1977.

29. Cartmill, C. E., "Analysis of Standard Problem 6 (Semiscale Test S-02-6) Data," EG&G Idaho (INEL), TREE-NUREG-1056, August 1977.

30. Transmittal of TLTA Sensitivity Studies and Test 4903 Run 16 Data Comparison," EG&G Idaho (INEL), Letter from R. R. Stiger to P. E. Litteneker (DOE), Stig-68-76, November 1976.

31. "Statistical Analysis," RES Program FIN./189a No. A1205-8, December 1977.

32. "Loss-of-Coolant Accident Analysis," RES 189a No. A6052.