A Comparison of the PARET/ANL and RELAP5/MOD3 Codes for the Analysis of IAEA Benchmark Transients

W. L. Woodruff, N. A. Hanan, R. S. Smith and J. E. Matos

Argonne National Laboratory
Argonne, Illinois 60439-4841 U.S.A.

Presented at the
1996 International Meeting on
Reduced Enrichment for Research and Test Reactors

October 7-10, 1996
Seoul, Republic of Korea

*Work supported by the U.S. Department of Energy,
Office of Nonproliferation and National Security,
under Contract No. W-31-109-ENG-38

A Comparison of the PARET/ANL and RELAP5/MOD3 Codes for the Analysis of IAEA Benchmark Transients

W. L. Woodruff, N. A. Hanan, R. S. Smith and J. E. Matos
Argonne National Laboratory
Argonne, Illinois 60439,USA

ABSTRACT

The PARET/ANL and RELAP5/MOD3 codes are used to analyze the series of benchmark transients specified for the IAEA Research Reactor Core Conversion Guidebook (IAEA-TECDOC-643 Vol. 3). The computed results for these loss-of-flow and reactivity insertion transients with scram are in excellent agreement and agree well with the earlier results reported in the guidebook. Attempts to also compare RELAP5/MOD3 with the SPERT series of experiments are in progress.

INTRODUCTION

The RELAP series of codes have been developed to provide "best estimate" descriptions of the behavior of power reactor systems under off normal conditions. The RELAP5 series was developed by the Idaho National Engineering Laboratory (INEL) for the US Nuclear Regulatory Commission primarily for the analysis and safety assessment of pressurized water reactor systems. The latest in the series, RELAP5/MOD3[1], is a coupled kinetics, one-dimensional heat transfer and two component hydrodynamics code that is capable of modeling all of the components of the system in a very general manner offering the user great flexibility in modeling.

While the RELAP code system has been widely used for light water power reactor systems with extensive testing of the various components against experiments, the application to non-power reactor systems and comparison to experiments has been fairly limited. The earlier work with research reactors was done with a combination of MOD2 and MOD2.5 versions of the code[2-4]. More recent work by Oak Ridge National Laboratory (ORNL) on the Advanced Neutron Source (ANS) reactor design[5] and by Brookhaven National Laboratory (BNL) on the High Flux Beam Reactor (HFBR)[6] did use the current version of the code but with some revisions. The analyses in some of the cases did not include a full kinetics solution, and some of the models reverted back to a homogeneous mode for the thermal-hydraulics. The ORNL and BNL analyses included the introduction of new heat transfer and interfacial drag correlations more appropriate for research reactor applications.

Most research reactors analyses using RELAP5 have been for loss-of-flow (LOF) transients and loss-of-coolant accidents (LOCA) for high performance reactors. To our knowledge, RELAP5 has not been tested against the SPERT reactivity insertion experiments with plate type fuel. Also, the RELAP5/MOD3 code has not been compared with other codes for more conventional research reactor models and transients such as those provided by the 10 MW benchmark reactor in the IAEA Research Reactor Core Conversion Guidebook[7]. These benchmark computations include both LOF and reactivity insertion transients all with reactor scram. The transients were computed by laboratories in four separate countries including Argonne National Laboratory (ANL) in the US. The ANL computations used the PARET/ANL code[8], which is a code that has been used extensively for research reactor analysis and has been compared with the SPERT-I and -II experiments for both light and heavy water systems with plate type fuel[9,10]. The results for benchmark transients with RELAP5/MOD3 and PARET are compared in this report.

THE RELAP5/MOD3 CODE MODELING

The RELAP5/MOD3 version used for these analyses includes the modifications used for the ANSR analysis (including the Petukhov correlation for single phase heat transfer). The benchmark core has no specifications for an external loop, plenums, return flow or bypass flow, so models for the plenums and return flow path are subjects for a sensitivity study. The return flow model is essential only for the LOF transients where it is necessary to provide a return flow loop for natural convection. The active core is modeled with a hot channel with the specified peaking factors applied and with a second channel representing the remainder of the core. The number of axial nodes is limited to 21 to match the maximum allowed by PARET. The LEU feedback coefficients for Doppler, coolant temperature and coolant void/density and other kinetics parameters were chosen to match those determined by ANL and used by the PARET code in the earlier study. The RELAP code provides a full liquid and vapor phase solution with two-phase flow. The RELAP code can also be used to model LOCAs.

THE PARET/ANL CODE MODELING

The PARET/ANL code is essentially the same code used in the earlier studies with some improvements added. The code now provides an ability to follow a LOF transient with down flow initially, through flow reversal and finally through the establishment of natural convection cooling. The Petukhov correlation was also added as an option to the original collection of single phase heat transfer correlations in order to match that used in RELAP. The natural convection correlation in the optional modified heat transfer routine was modified to match that used in RELAP. The active core model is also a two channel model as in the earlier report[7] and uses the same LEU kinetics and feedback parameters. Unless otherwise specified, PARET uses the Petukhov correlation and modified heat transfer routine for these comparisons. The PARET code provides a homogeneous model with two-phase flow and does not require the specification of return flow.

THE BENCHMARK TRANSIENTS

The 10 MW reactor for the benchmark transients is the same reactor model used for the neutronics benchmark computations in IAEA - TECDOC-233, but with the central flux trap of water replaced by a block of aluminum with a 5.0 cm square hole containing water (this was to generate more realistic radial and local power peaking factors for these required computations).

The initial conditions are

- Burnup: Beginning of life
- Hot Channel Factors: Radial x local power peaking factor - 1.4
  Axial power peaking factor - 1.5
  Engineering factor - 1.2
  Overpower factor - 1.2
- Nominal Flow Rate: 1000 m^3/hr
- Coolant Inlet Temperature: 38 Deg. C
- Coolant Inlet Pressure: 1.7 bar absolute
- Thermal Conductivity of the LEU Fuel: 0.5 W/cm K

The scram in each case is a linear reactivity insertion of -$10 in 0.5 seconds.

The benchmark transients for the LEU fuel case only include the following selections:

1. Fast LOF transient, where flow is reduced as $e^{-t/T}$ with T = 1 second, and reactor scram is initiated at 85% of nominal flow with a 200 ms delay before control insertion begins.
2. Slow LOF transient, where the exponential decay now has T = 25 seconds with the same scram conditions as in case 1.
3. Slow reactivity insertion transient, where a $0.09/s ramp is inserted in a critical reactor at an initial power of 1 Watt. The safety system trip point is set at 1.2 P_o (12 MW) with a time delay of 25 ms before control blade insertion is initiated.
4. Fast reactivity transient, where $1.50 is inserted in 0.5 seconds. The scram conditions are the same as for case 3.

RESULTS

While the PARET code was modified to allow conformance with some of the choices of correlations available in RELAP, and the RELAP model was made to conform to the model used in PARET to minimize obvious differences as much as possible, no further changes were made in either code. The LEU benchmark with a $1.50/0.5s reactivity insertion without scram is used here to show the influence of these modifications on the results obtained with the PARET code. The first set of results in Table I are identical to the results reported in the IAEA Conversion Guidebook, the second column shows the influence of the modified heat transfer routine, and the final column shows the additional impact of changing from the Seider-Tate correlation in the original study to the Petukhov correlation. As can be observed, the results of these modifications make very little difference. The largest difference is in the coolant outlet temperature at 0.8s into the transient, where the Petukhov choice gives a slightly higher temperature.

Table I. PARET Code Results for LEU $1.50/0.5s Transient Without Scram

Correlation	Seider-Tate		Petukhov
Heat Transfer Routine	Option 0	Option 2	Option 2
Peak Power, MW	282	282	282
Time of Peak, t_m, s	0.622	0.622	0.622
Energy to t_m, MWs	5.52	5.60	5.58
Clad Temp. at t_m, ^oC	181	184	182
Peak Clad Temp., ^oC	263	257	263
Time to Peak Temp., s	0.643	0.642	0.644
Outlet Temp. at 0.8s, ^oC	101	106	112

Fast LOF Transient

The results for the 1.0s LOF transient with both the RELAP and PARET codes are shown in the top two panels of Fig. 1. The peak temperatures for the fuel and clad are all within a few degrees for the two codes with the RELAP predictions slightly higher and later in time with natural convection cooling. The temperatures for the coolant are predicted to be slightly higher with PARET. For all practical purposes the two codes are predicting the same results.

Slow LOF Transient

The results for the 25.0s LOF transient, shown in the lower two panels of Fig. 1, are very similar to those for the shorter decay time transient with RELAP again predicting a slightly higher fuel and clad temperature. Again the results for the two codes are virtually identical.

While the benchmark transient specifications do not include return flow, the model for RELAP must specify a return flow path and connection of the two channels to the same plenum. The definition for return flow used in the RELAP model has an impact on the results obtained for LOF transients. Figure 2 shows a comparison of the peak temperature for the clad with varying choices of bypass flow. These results show that as the return flow available for natural convection is increased the peak clad temperature decreases and falls slightly below that predicted by PARET. As the return flow is increased still further, little change is noted in the temperature, and the condition where infinite flow is available for cooling by natural convection is approached. This is probably more nearly the conditions modeled in the PARET code (no return loop is modeled in PARET). While the differences in temperature grow with these changes, these differences are still less than 5 ^oC. It is important to note that in a "real" reactor the plenums and return flow path would be well defined in the design.

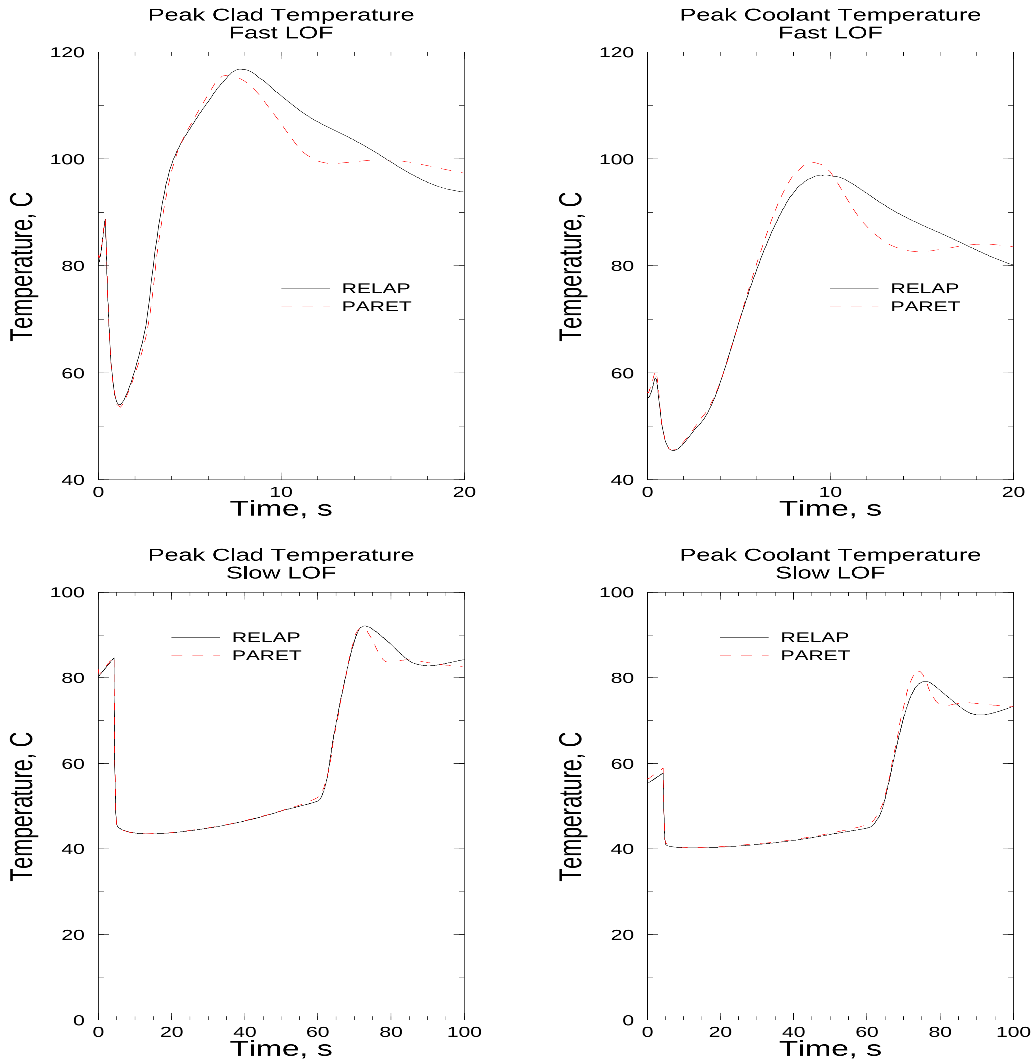

Figure 1. Fast and Slow LOF Transients with Flow Reversal and Natural Convection

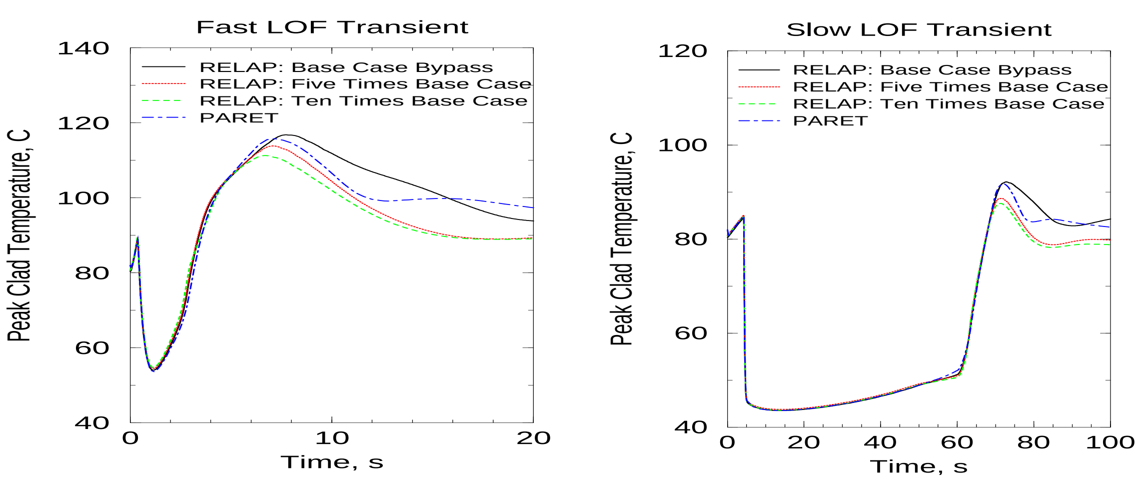

Figure 2. A Comparison of Peak Clad Temperature with LOF and Varying Bypass Flow

Slow Reactivity Insertion Transient

The results for a slow reactivity insertion transient of \$0.09/s are shown in Fig. 3 with peak power, reactivity and temperatures that are all almost identical for the two codes. The largest difference in peak temperature is found in the coolant, and that difference is less than a degree. The peak power predicted by RELAP is 12.3 MW, while PARET predicts a peak power of 12.4 MW. As for the LOF cases, the results from the two codes are virtually identical.

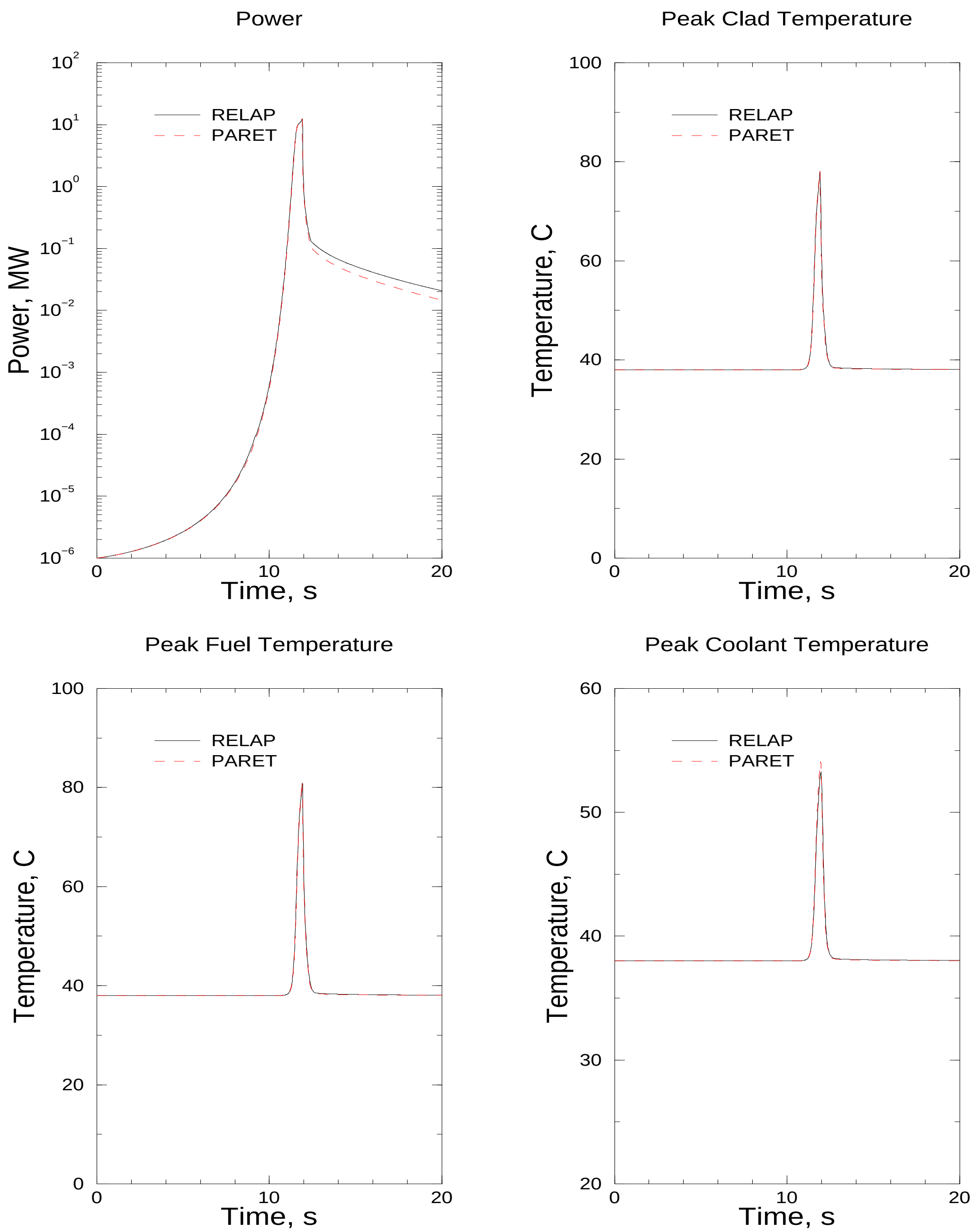

Figure 3. Transient Response of Benchmark Core to Reactivity Insertion of \$0.09/s with an Overpower Scram Trip at 12 MW and a 25 ms Delay.

Fast Reactivity Insertion Transient

The fast reactivity insertion transient results with $1.50/0.5s are given in Fig. 4, and again the two codes agree very well. The largest difference occurs in the prediction of peak clad temperature, where the RELAP code is predicting a temperature that is almost 12 degrees higher than that predicted by the PARET code

.

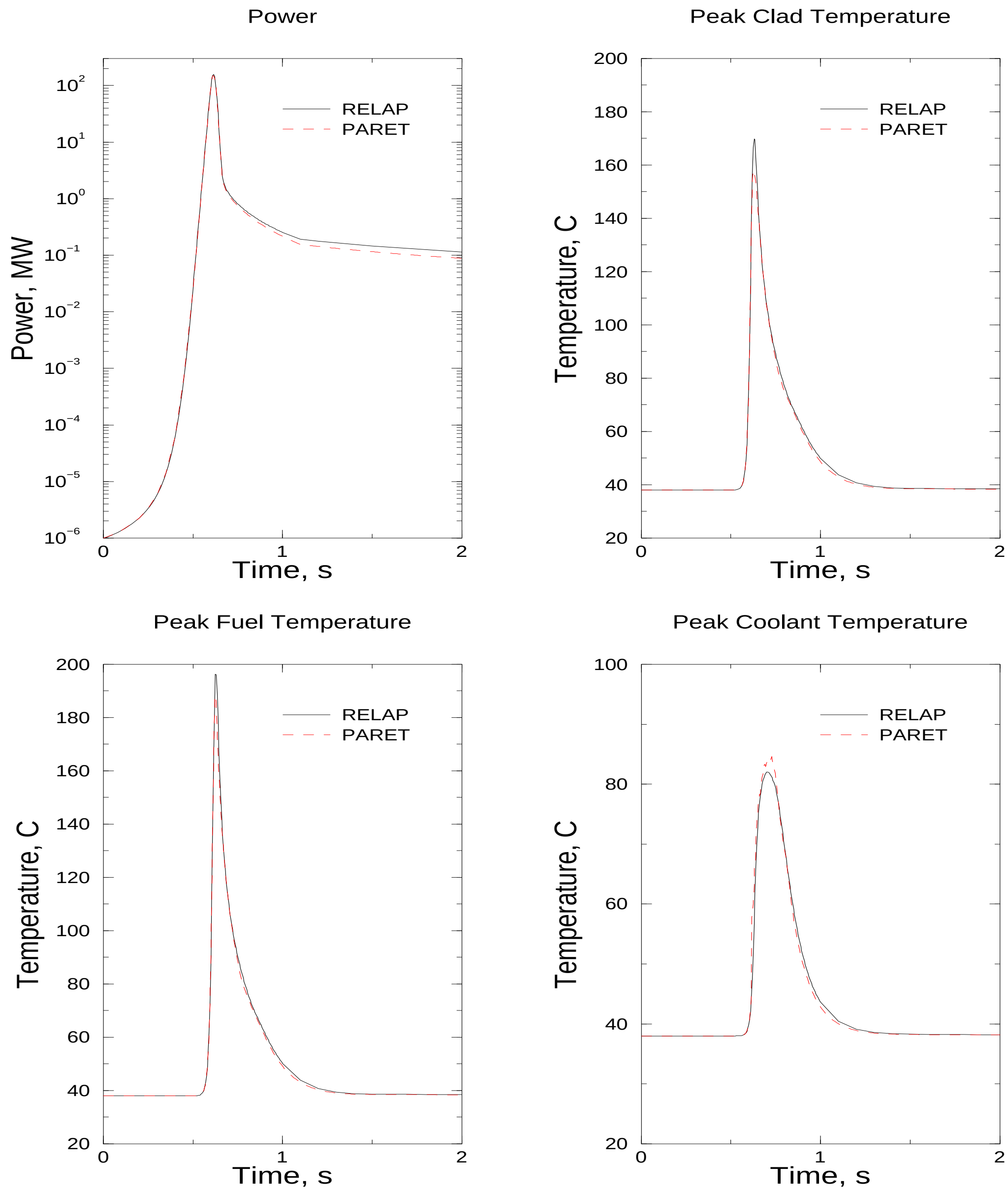

Figure 4. Transient Response of Benchmark Core to Reactivity Insertion of $1.50/0.5s with an Overpower Scram Trip at 12 MW and a 25 ms Delay

The peak temperatures from the Guidebook[7] for the four institutions that provided data and the current data may be compared in the following tables (Tables II and III):

Table II. A Comparison of Peak Temperatures for LOF Transients

LOF Transients	1.0s Pump Decay Peak Temperatures, °C			25.0s Pump Decay Peak Temperatures, °C		
	Fuel	Clad	Coolant	Fuel	Clad	Coolant
ANL, USA	90.3	87.5	60.3	86.8	83.7	58.8
Interatom, FRG	91.9	89.3	56.5	88.2	85.5	55.4
JAERI, Japan	NA	97.1	58.1	NA	96.1	57.5
JEN, Spain	95.4	93.9	59.3	91.9	90.3	58.1
PARET/ANL (Natural Convection)	91.7 (116.0)	88.9 (115.7)	60.3 (99.6)	88.0 (91.8)	84.9 (91.7)	58.8 (81.7)
RELAP5/MOD3 (Natural Convection)	91.7 (117.0)	88.8 (116.8)	59.0 (97.0)	87.8 (92.3)	84.6 (92.2)	57.7 (79.2)

Table III. A Comparison of Peak Temperature for Reactivity Insertion Transients

Reactivity Insertion Transients	$0.09/s Peak Temperatures, °C			$1.50/0.5s Peak Temperatures, °C		
	Fuel	Clad	Coolant	Fuel	Clad	Coolant
ANL, USA	80.6	77.7	53.9	183.4	156.7	82.0
Interatom, FRG	80.8	78.1	51.1	185.3	168.2	63.2
JAERI, Japan	81.2	78.5	52.8	171.0	149.2	62.7
JEN, Spain	73.2	71.9	48.8	166.4	156.6	80.4
PARET/ANL	81.2	78.0	54.1	188.3	157.9	83.3
RELAP5/MOD3	80.8	77.8	53.2	196.2	169.8	82.0

The transient specifications did not distinguish the flow direction, and only JEN attempted to compute flow reversal from down flow and up flow with natural convection. The current predictions for the LOF transients include a second number in parentheses for the peak under natural convection and up flow. Given the assortment of codes that were used in these studies, all of the results are in reasonably good agreement with each other for each of the transients considered. The coolant temperatures from the guidebook data are temperatures at the outlet and may not always depict peak values for comparison.

The ANL contribution to the IAEA guidebook also included an analysis of the benchmark core for the $1.50/0.5s transient without scram, and attempts were made to compare the RELAP and PARET codes for this self-limiting transient. The RELAP code was found to predict a smaller amount of voiding and with less feedback due to voiding, the predicted peak power and temperatures were higher than those predicted by the PARET code (Table I). Some attempts were also made to compare the RELAP5/MOD3 results to the SPERT-I experiments with similar results observed. This behavior was also noted in earlier versions of the code with a comparison of SPERT-IV data.[11] An attempt is being made to resolve these differences.

CONCLUSIONS

The overall agreement between the PARET and RELAP codes for this series of benchmark transients is excellent, and the results agree well with the earlier guidebook results. The benchmark transients each assume only a single mode failure and reactor scram is initiated. The amount of nucleate boiling present (if any) is very limited and the feedback from voiding is not a significant factor in the shutdown of the reactor. When attempts were made to repeat some of the self-limiting cases without scram, such as the case depicted in Table I, the peak power and temperatures predicted by the MOD3 version of the RELAP5 were higher than the PARET results. Preliminary attempts to compare RELAP with SPERT data show a similar trend. We hope to be able to resolve these differences in further work with INEL staff and to be able to successfully compare RELAP5/MOD3 with the SPERT series of experiments.

REFERENCES

1. "RELAP5/MOD3 Code Manual, Volume 1: Code Structure, System Models, and Solution Methods," NUREG/CR-5335, INEL-95/0174, Idaho National Engineering Laboratory (June 1995).

2. Polkinghorne, S. T., "Validation of the ATR-SINDA Fuel Plate Model Using Advanced Test Reactor Startup Data," *Nucl. Technol.*, **108**, 395 (Dec. 1994).

3. Morris, D. G. and M. W. Wendel, "High Flux Isotope Reactor System RELAP5 Input Model," ORNL/TM-11647, Oak Ridge National Laboratory (Jan. 1993).

4. Slovik, G. C., et. al., "RELAP5/MOD2.5 Analysis of the HFBR for a Loss of Power and Coolant Accident," BNL-52243, Brookhaven National Laboratory (May 1990).

5. Fletcher, C. D., et. al., "Conceptual Design Station Blackout and Loss-of-Flow Accident Analyses for the Advanced Neutron Source Reactor,"*Nucl. Technol.,* **106**, 31 (April 1994).

6. Cheng, L. Y., and P. R. Tichler, "Analysis of Flow Reversal Test," Presented at the Fourth International Conference on Nuclear Engineering, ICONE-4, New Orleans, Louisiana, (March 1996).

7. "Research Reactor Core Conversion Guidebook - Volume 3: Analytical Verification," IAEA-TECDOC-643, (April 1992)

8. Woodruff, W. L., :A User Guide for the Current ANL Version of the PARET Code," NESC (1984).
   Obenchain, C. F., "PARET - A Program for the Analysis of Reactor Transients,"IDO-17282, January 1969.

9. Woodruff, W. L., “A Kinetics and Thermal-Hydraulics Capability for the Analysis of Research Reactors,” *Nucl. Technol.*, **64**, 196 (Feb. 1984).

10. Woodruff, W. L., ”Additional Capabilities and Benchmarking with the SPERT Transients for Heavy Water Applications of the PARET Code,” Proc. of 1989 Int. Mtg. on Reduced Enrichment for Research and Test Reactors, Berlin, Germany, pp. 357-365, September 10-14, 1989.

11. Kim, S. S. and J. C. McKibben, “PARET/ANL and RELAP/MOD2 Benchmarking Comparison Against the Spert-IV Test Data,” University of Missouri, Unpublished (1996).