CONF-950914--9

Paper to be presented at the Safety of Operating Reactors ANS International Topical Meeting, September 17-20, 1995, Hyatt Regency Bellevue, Seattle (Bellevue), Washington.

VALIDATION OF DETAILED THERMAL HYDRAULIC MODELS USED FOR LMR SAFETY AND FOR IMPROVEMENT OF TECHNICAL SPECIFICATIONS

F. E. Dunn

Reactor Analysis Division
Argonne National Laboratory
9700 S. Cass Avenue
(708)252-4692

*Work supported by the U.S. Department of Energy, Nuclear Energy Program under Contract W-31-109-ENG-38.

VALIDATION OF DETAILED THERMAL HYDRAULIC MODELS USED FOR LMR SAFETY AND FOR IMPROVEMENT OF TECHNICAL SPECIFICATIONS

Floyd E. Dunn
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439
(708) 252-4692, E mail fedunn@anl.gov

ABSTRACT

Detailed steady-state and transient coolant temperatures and flow rates from an operating reactor have been used to validate the multiple pin model in the SASSYS-1 liquid metal reactor systems analysis code. This multiple pin capability can be used for explicit calculations of axial and lateral temperature distributions within individual subassemblies. Thermocouples at a number of axial locations and in a number of different coolant sub-channels in the XX09 instrumented subassembly in the EBR-II reactor provided temperature data from the Shutdown Heat Removal Test (SHRT) series. Flow meter data for XX09 and for the overall system are also available from these tests. Results of consistent SASSYS-1 multiple pin analyses for both the SHRT-45 loss-of-flow-without-scram test and the SHRT-17 protected loss-of-flow test agree well with the experimental data, providing validation of the SASSYS-1 code over a wide range of conditions.

I. INTRODUCTION

A detailed multiple pin model[1] has been added to the SASSYS-1 liquid metal reactor (LMR) systems analysis code.[2] This model provides a much more detailed treatment of temperatures and coolant flow rates in the core than the single pin per subassembly model previously used in the code. This more detailed treatment can provide more realistic safety assessments. If properly validated with detailed experimental data, the detailed model can provide the basis for improvement of technical specifications for operating reactors. Detailed experimental data for validation of this model is available from the Shutdown Heat Removal Test (SHRT) series of tests[3-5] in the Experimental Breeder Reactor II (EBR-II). This series of thermal hydraulic tests was performed to demonstrate that a properly designed LMR with metal fuel can survive a number of different anticipated transients without scram. For these tests, detailed temperature data is available from thermocouples in the coolant in the XX09 instrumented subassembly.[6] In addition, coolant flow data is available for these tests from flow meters in XX09 and in the primary and secondary coolant systems. Previously, results have been reported[1,7] for SASSYS-1 multiple pin analyses of the SHRT-45 loss-of-flow-without-scram test. Reference 7 also describes a very detailed SASSYS-1 analysis of EBR-II blankets. This detailed blanket analysis illustrates the excessive conservatism that went into the initial simpler analysis that is the basis for the current technical specification limits for the irradiation of EBR-II blanket subassemblies. Now, consistent multiple-pin analyses for both the SHRT-45 test and the SHRT-17 protected loss-of-flow test have been completed, providing validation of the SASSYS-1 code over a greater range of conditions. The transient temperatures reached in SHRT-45 were higher than those reached in SHRT-17, but significantly lower transient coolant flow rates were reached in SHRT-17. Also, the transient coolant temperature behavior in SHRT-17 was more complex than that in SHRT-45.

SASSYS-1 with the multiple pin model is not the first computer code used to calculate detailed temperature distributions within LMR subassemblies, but it is the first LMR systems code to provide such a detailed steady-state and transient core treatment while also analyzing the behavior of the whole primary and secondary heat transport systems. Also, SASSYS-1 is probably the first LMR code with which it is feasible to do very detailed transient core temperature calculations on a routine basis. In the 1970s various versions of the COBRA code[8,9] were written to provide a detailed sub-channel analysis of the coolant temperatures within a subassembly. Every subchannel in a subassembly could be modeled. Also, the COBRA-WC code[10] was written to provide somewhat less detailed coolant temperatures for the whole core. The COBRA codes strained the capabilities of the computers that were available when the codes were written; if an attempt were made to put much detail into a COBRA model, one would quickly run out of memory. Also, a COBRA case big enough to fill memory would often run a very long time. In order to address the computer limitation problems of COBRA-type

codes, the ENERGY, SUPERENERGY, and SUPERENERGY-2 codes[11] were written. By using some modeling approximations and using an efficient solution method, SUPERENERGY-2 can compute detailed steady-state coolant temperatures for every subassembly in a core in a reasonable amount of computing time; but the SUPERENERGY codes do not do transient calculations. The combination of the use of efficient solution methods in SASSYS-1 and the current availability of fast, relatively cheap engineering work stations with large amounts of memory has now made it feasible to do very detailed calculations. The SHRT-17 and SHRT-45 cases described below each took less than 1000 seconds of computing time on a Sun Sparc-20 workstation to do 600 seconds of transient calculations; even though the whole core, the rest of the primary loop, and the intermediate heat transfer loop were modeled; and even though considerable detail was used in the modeling of the XX09 subassembly and its neighbors.

II. SASSYS-1 CODE

The SASSYS-1 code is an integrated systems analysis code. It contains a point kinetics treatment for core neutronics, with spatially-dependent reactivity feedback. It also contains a multi-channel thermal hydraulics treatment of the core coupled to a thermal hydraulics treatment of the rest of the primary coolant system, thermal hydraulic treatments for the intermediate coolant loops, the steam generators, and the balance-of-plant, plus a control system model. In the multi-channel thermal hydraulics treatment of the core, each subassembly or group of similar subassemblies is represented by one or more channels. A channel models one pin, its associated coolant, and a structure which represents the wrapper wire and/or the subassembly duct wall. The whole length of a subassembly is modeled, including the pin section and reflector regions above and below the pins. If a single channel is used for a subassembly, then it usually models the average pin.

In the multiple pin option, the regions above and below the pin are still represented by a single channel; but a number of channels can be used to represent the pin section, with each channel representing one or more pins and their associated coolant and structure. Coolant-to-coolant heat transfer between adjacent channels is accounted for, including the effects of both conduction and turbulent mixing. The code also accounts for subassembly-to-subassembly heat transfer from the duct wall of a subassembly, through the interstitial sodium, to the duct wall of a neighboring subassembly. No coolant cross flow between channels is calculated, but the effects of turbulent mixing between subchannels are included in channel-to-channel coolant heat transfer coefficients. Each channel representing the pin section of a subassembly has its own time-dependent flow rate. All coolant channels in a subassembly are driven by common inlet and outlet pressures, so transient flow redistribution is accounted for.

III. EXPERIMENTAL VALIDATION OF THE MULTIPLE-PIN MODEL IN SASSYS-1

The SHRT series of tests included a number of different types of transients starting from various initial powers and coolant flow rates. As mentioned above, the SHRT-17 and SHRT-45 tests were used for experimental validation of the multiple pin model in SASSYS-1. SHRT-17 was the most severe protected loss-of-flow test, and SHRT-45 was the most severe loss-of-flow-without-scram test in the series. Both of these tests started from full nominal power and full nominal coolant flow. The normalized powers and flows for these tests are shown in Figure 1. The powers in this figure are based on the sum of measured fission power plus computed decay heat. The decay heat was calculated from the ANSI light water reactor standard[12] using the irradiation history for the core loading. The flows are computed total flows from the two main primary pumps to the high pressure inlet plenum that feeds the core region. EBR-II also has a low pressure inlet plenum that feeds radial shields and blankets. Both SHRT-17 and SHRT-45 started with a trip of the main primary pumps and the intermediate loop pump at time zero. In SHRT-17 the control rods were inserted almost immediately, and the power rapidly dropped to decay heat levels. In contrast, there was no scram in SHRT-45, so the decrease in power was only due to negative reactivity feedback. Even after 600 seconds, the reactor was not completely shut down in SHRT-45: the fission power was still comparable to the decay heat at 600 seconds. The transient flows were lower in SHRT-17 than in SHRT-45 for a number of reasons. The pump coastdown was rapid in SHRT-17, but a slower coastdown was used in SHRT-45. Also, EBR-II has an electromagnetic auxiliary pump in the pipe between the coolant outlet plenum and the intermediate heat exchanger (IHX). In SHRT-17 the auxiliary pump was shut off; but in SHRT-45 it was left running on battery power. Therefore, after the main primary pumps stopped in SHRT-17 there was nothing but natural circulation head driving the flow, whereas in SHRT-45 there was a small pump head from the auxiliary pump. In addition, the transient powers and temperatures were higher in SHRT-45, giving higher natural circulation heads. Note that these transients span a wide range of conditions, from high power, high flow to low power, low flow, and from forced convection with turbulent flow to natural circulation with laminar flow.

In order to provide a clean validation of the thermal hydraulics model in SASSYS-1, the power used in the calculations was specified as a function of time, using the curves shown in Figure 1, rather than being calculated by the code on the basis of the reactivity feedback. This separated out the issue of the accuracy of the reactivity feedback coefficients in SHRT-45. Also, consistent models were used in the SASSYS-1 calculations for both tests. The model for the XX09 instrumented subassembly was the same, and powers and flows were calculated in a consistent manner.

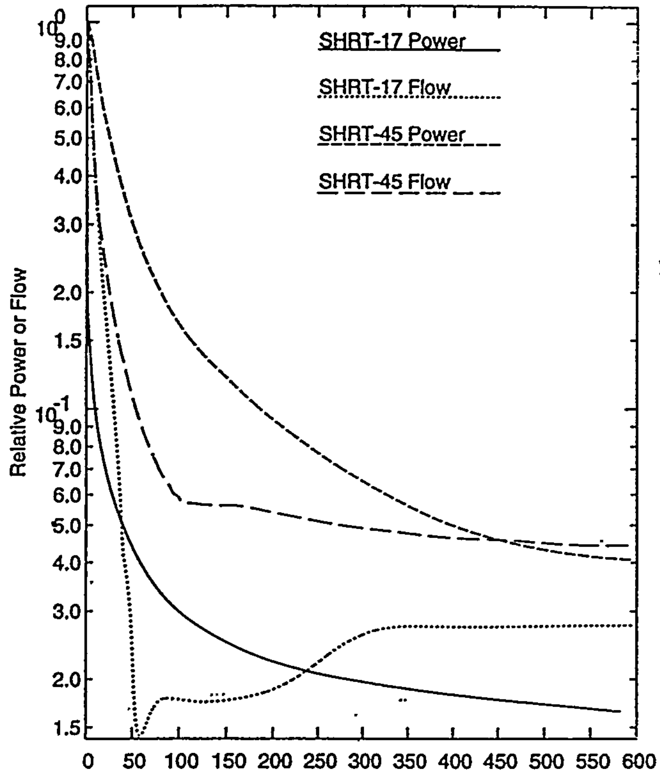

Fig. 1. SHRT-17 and SHRT-45 Normalized Total Power and Flow

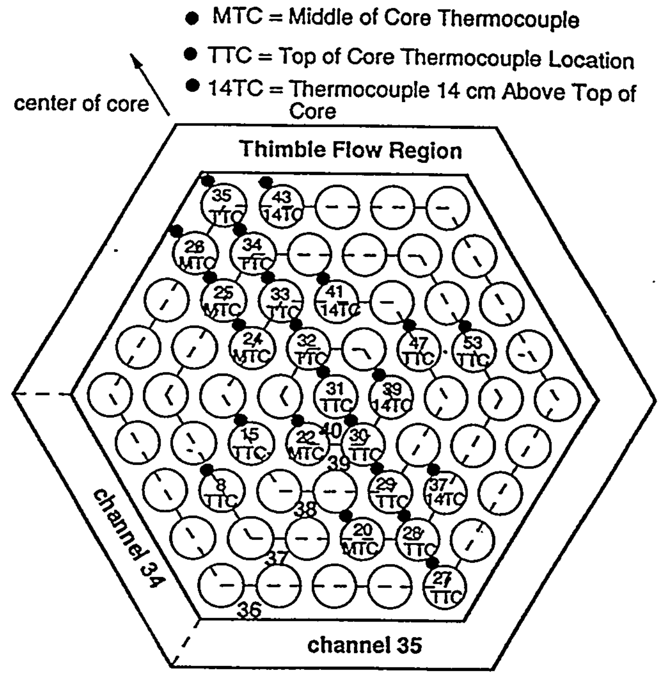

Fig. 2. SASSYS-1 Multi-Channel Representation and Thermocouple Locations for the EBR-II XX09 Instrumented Subassembly

A. SASSYS-1 Analysis of the SHRT Tests

For the analysis of the SHRT-17 and SHRT-45 tests, SASSYS-1 models were used for the core, the rest of the primary system, and the intermediate sodium loop. Detailed multiple pin models for XX09 and its immediate neighbors were used. Single pin models were used for the channels representing the rest of the core, the radial blankets, and the radial shield subassemblies. Nothing beyond the steam generators was modeled, and even the steam generator had little influence on the core during the times of interest for these tests. On the other hand, the behavior of the primary pumps as they coasted down had a large impact on the early parts of the transients, and primary loop natural circulation heads were dominant in the later parts of the transients after the pumps had stopped. The centrifugal pump model used for this analysis was the homologous pump model described in Ref. 13, based on the work of Wylie and Streeter.[14,15]

Figure 2 shows the XX09 subassembly and the multi-channel representation used to model this subassembly. XX09 is almost identical to a MK-IIA driver subassembly, except that in XX09 the outer row of pins is replaced by an extra hex can wall and the thimble flow region. The thimble flow region contains sodium with a small flow rate. Also, below the pin section in XX09 are two identical electromagnetic flowmeters. A number of the pin wrapper wires in XX09 are replaced with thermocouple leads for thermocouples in the coolant. The locations of some of these thermocouples are shown in Figure 2. Five channels, channels 36-40, are used in SASSYS-1 to represent concentric rows of coolant channels and pins in the pin section of XX09. Two channels, channels 34 and 35, are used to represent the thimble flow region. In both the SHRT-17 test and the SHRT-45 test the six subassemblies adjacent to XX09 included one cool-running experimental subassembly and five hotter driver subassemblies. One thimble flow channel represents the region next to the cool-running subassembly, and the other thimble flow channel represents the region next to the drivers.

B. Results

Comparisons between computed and measured coolant temperatures at the top of the fuel in XX09 for the SHRT-17 test are given in Figure 3. The internal average temperatures in these figures are averages over all coolant subchannels except the edge and corner subchannels, weighted by the number of pins in the corresponding SASSYS-1 channel. There are no thermocouples in edge subchannels at the top of the core in XX09. The experimental peak temperatures are the average of the TTC-30 and TTC-31 values. Agreement between calculated and measured temperatures is good in the initial part of the transient up to the peak, but from 100 seconds to about 400 seconds the calculated temperatures are lower than the measured values. The calculated coolant flow rates are also high in relation to the measured values in this range. Even though the EBR-II outlet plenum is small, temperature

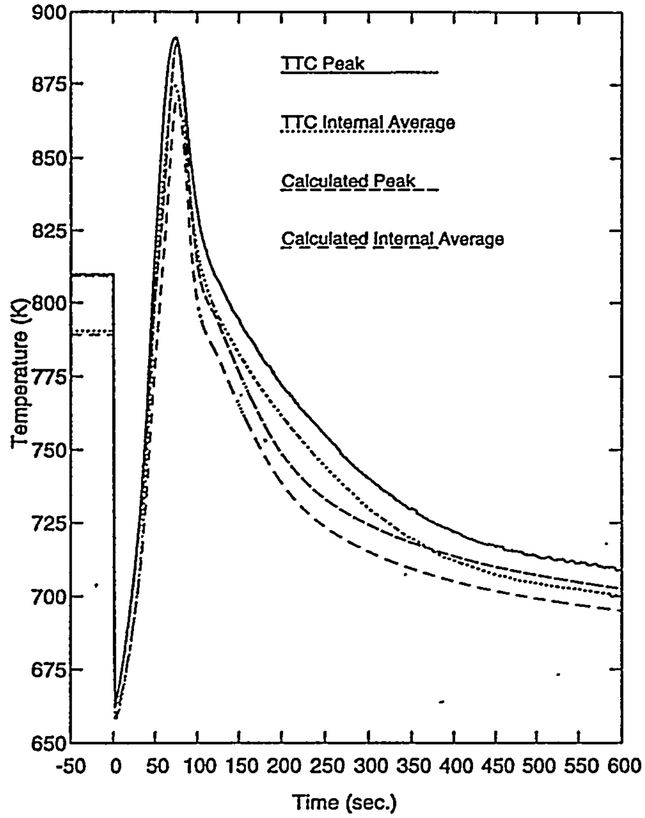

Fig. 3. SHRT-17 Top of Core Temperatures, Perfect Mixing Model for the Outlet Plenum

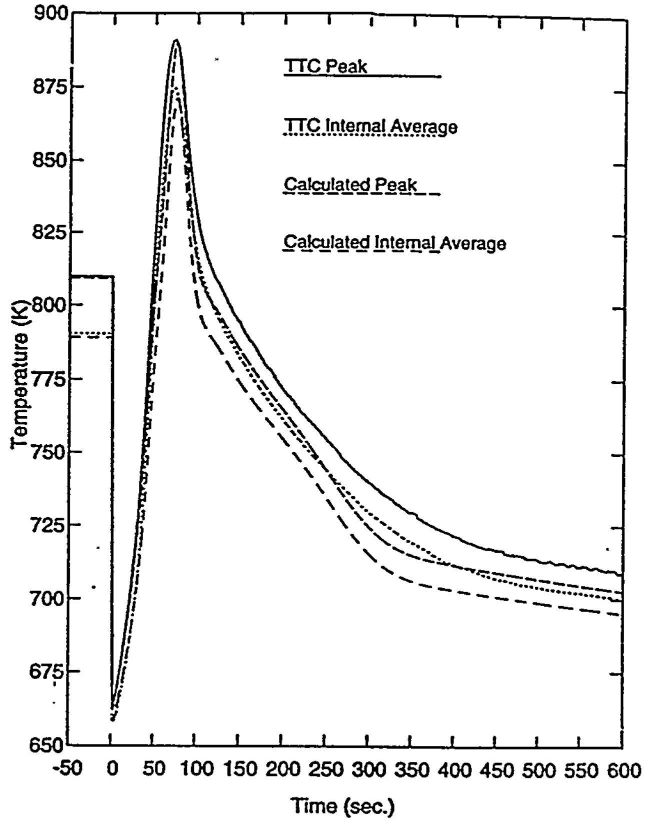

Fig. 4. SHRT-17 Top of Core Temperatures, Stratified Temperature Model for the Outlet Plenum

stratification in the outlet plenum seems to cause this discrepancy by influencing the primary loop gravity head. The calculated results shown in Fig. 3 were obtained using a one node perfect mixing model for the outlet plenum. As shown in Fig. 3, the core outlet temperature drops by more than 100 K immediately after the start of the transient, sending cool liquid into the outlet plenum. The core outlet temperature then rises. The warmer liquid rises to the top of the outlet plenum; and the coolant leaving the outlet plenum through the exit, which is in the middle of the side, is cooler than a perfect mixing model would predict during the period from 100 seconds to 400 seconds in the transient. Much of the primary loop natural circulation head in EBR-II is generated in the Z pipe connecting the outlet plenum and the intermediate heat exchanger. In order to address this situation, a one-dimensional outlet plenum temperature stratification model, based partly on the work of Lorenz and Howard,[16] was added to SASSYS-1. Figure 4 shows the top of core coolant temperature comparisons when the stratified outlet plenum model was used for SHRT-17. The discrepancy in the 100 to 400 second time range is greatly reduced by using the stratified model, but the discrepancy does not disappear entirely. The actual temperature stratification pattern in the outlet plenum is probably more complex than what a one-dimensional model can handle accurately. All of the computed results in this paper, other than those in Fig. 3, were obtained with the stratified outlet plenum model, although for SHRT-45 there is a negligible difference between the results from the stratified model and those from the perfect mixing model.

Figure 5 shows the top of core coolant temperature comparisons for SHRT-45. In this case, the agreement between computed and measured values is quite good. Comparisons for individual TTC thermocouples in the row from TTC-27 to TTC-35 are shown in Figures 6 and 7. There was a power gradient across XX09, with the highest power toward the center of the core. This power gradient is reflected in the individual TTC readings. There is also some scatter in the individual readings, especially before the start of the transient. This scatter probably corresponds to real fluctuations in coolant temperatures, rather than to thermocouple calibration error. When the reactor goes up to power with temperature gradients across the subassemblies, bowing of the duct walls and pins occurs, causing the fuel pins to shift around. If the bowing and shifting is not completely uniform, then local non-uniformities in the pin spacing and resulting coolant temperatures will result. The transient TTC results show less scatter, probably because at the very low flow rates that occurred during these transients, thermal conduction between adjacent coolant subchannels tends to smooth out local temperature variations. In order to partly account for this non-uniform spacing, the central XX09 channel, channel 40, was modified slightly in the calculations

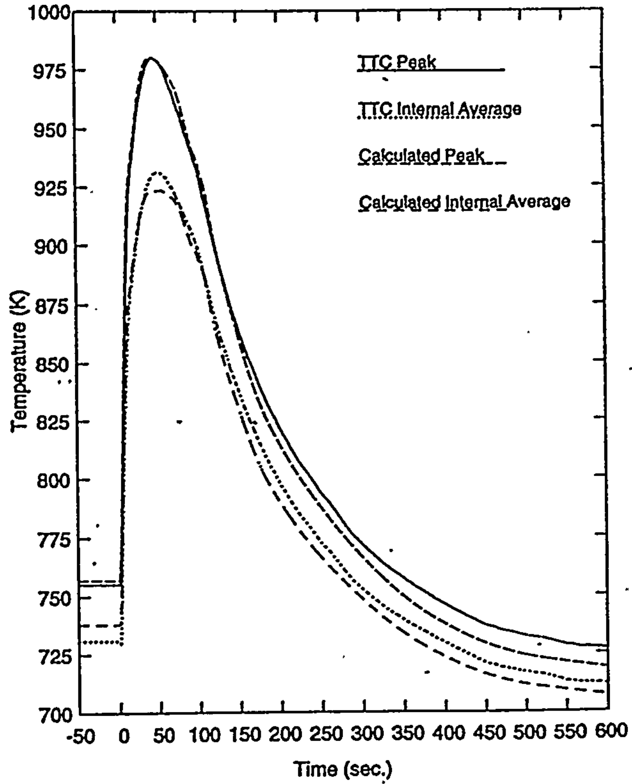

Fig. 5. SHRT-45 Top of Core Temperatures

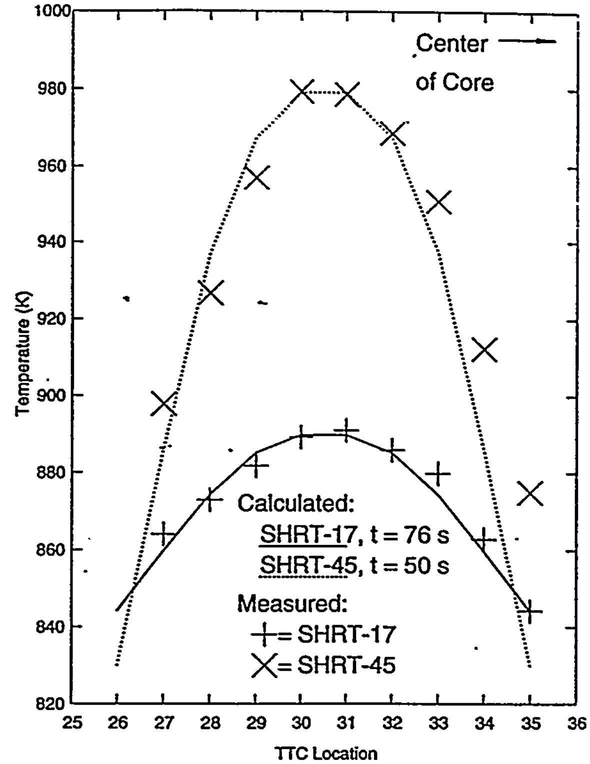

Fig. 7. Detailed Transient XX09 Thermocouple Comparisons

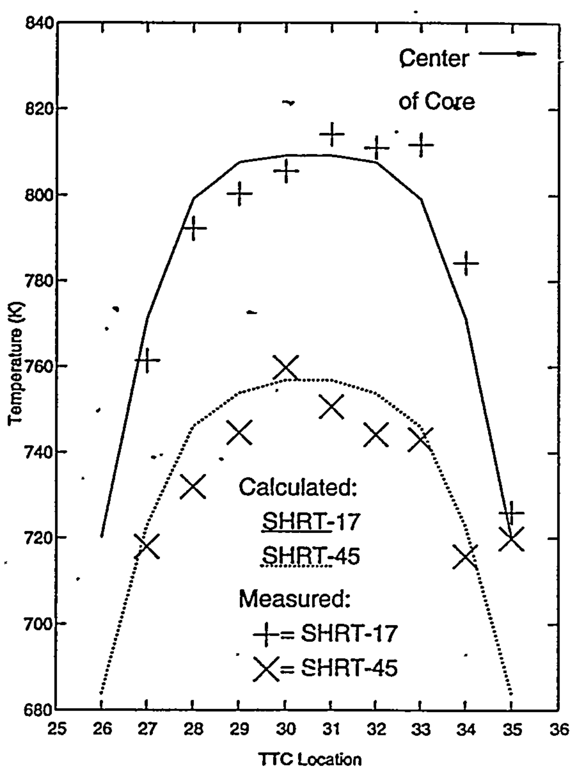

Fig. 6. Detailed Steady-State XX09 TTC Thermocouple Comparisons

for SHRT-45. The coolant flow area and hydraulic diameter used for this channel were calculated using the size of the wrapper wire for the spacing between pins, rather than using the slightly larger uniform spacing value. The extra coolant flow area was put in the edge channel, channel 36. This change slightly increased the peak-to-average temperature differences across the subassembly.

The thermocouple readings in XX09 in these tests were probably more accurate and reliable than some of the XX09 flow meter readings. There are two identical electromagnetic flow meters in XX09, located below the fuel pins. In the SHRT-17 test, run soon after XX09 was first put into the core, the two flow meters gave almost identical readings before the start of the transient. These steady-state flow meter readings were about 9% lower than what one would calculate on the basis of the total core flow and the measured pressure drop vs. flow characteristics for XX09, but the measurements were made before XX09 was put into the core, and the measurements did not include an upper adapter that was added when the subassembly was put into the core. The pressure drop in the upper adapter accounts for at least part of the 9% discrepancy, so the flow meter readings before the start of the test were probably quite accurate. As shown in Fig. 8 though, during the SHRT-17 test the readings of the two X09 flow meters differed significantly at low flows, as well as showing a significant amount of noise. The differences between

the calculated flows and the measured flows during this transient are comparable to the differences between the two flow meter readings. At the low flows encountered in this test, the flow meter signal was very small, so a small offset in flow meter readings could account for the differences between the two flow meters, as well as most of the differences between measured and calculated flows. By the time that SHRT-45 was run, XX09 had been in the core much longer, and the flow meters had suffered. Before the start of the SHRT-45 transient the readings of the two XX09 flow meters differed by a factor of four. The lower flow meter readings were obviously wrong, and the calibration of the upper flow meter may have changed a bit. The pre-transient reading of the upper flow meter was about 18% lower than the calculated value, rather than the 9% difference for SHRT-17. For consistency with the SHRT-17 analysis, the steady-state flow rate used in the SASSYS-1 calculations for SHRT-45 was 9% less than the calculated value. As shown in Fig. 8, the agreement between calculated and measured normalized XX09 flow rates for SHRT-45 was relatively good. The flow meter readings for SHRT-45 show some noise during the transient, but the signal to noise ratio is better than in SHRT-17, probably just because the signal is higher in SHRT-45.

In addition to XX09 instrumentation, there were some primary system measurements for the SHRT tests. Fig. 9 shows the measured and calculated normalized flow rates from pump 2 to the high pressure inlet plenum which supplies the coolant flow to the core subassemblies. Agreement between measured and calculated flows is fairly good; although as in the case of the XX09 flows, the calculated flows tend to be a bit higher than the measured flows.

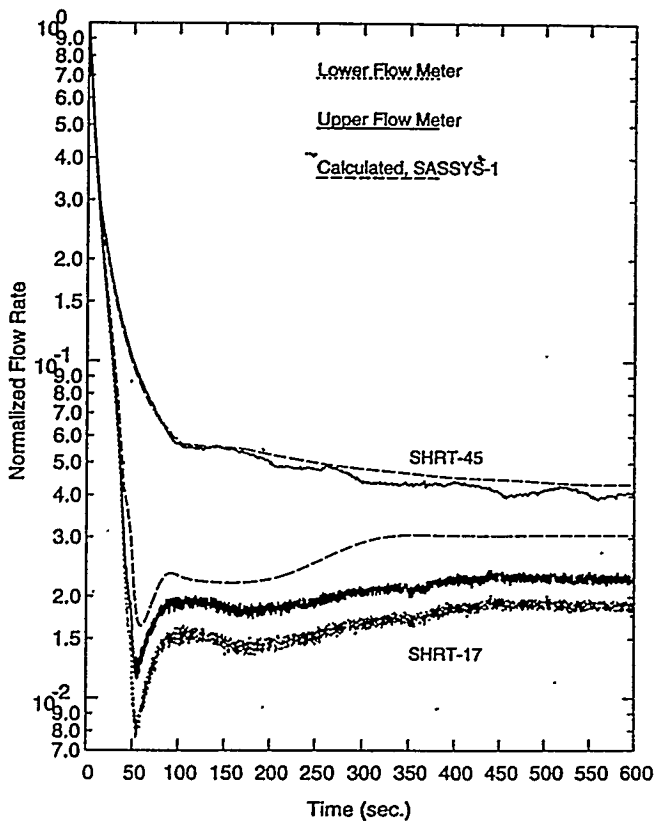

Fig. 8. XX09 Normalized Coolant Flow Rates, SHRT-17 and SHRT-45

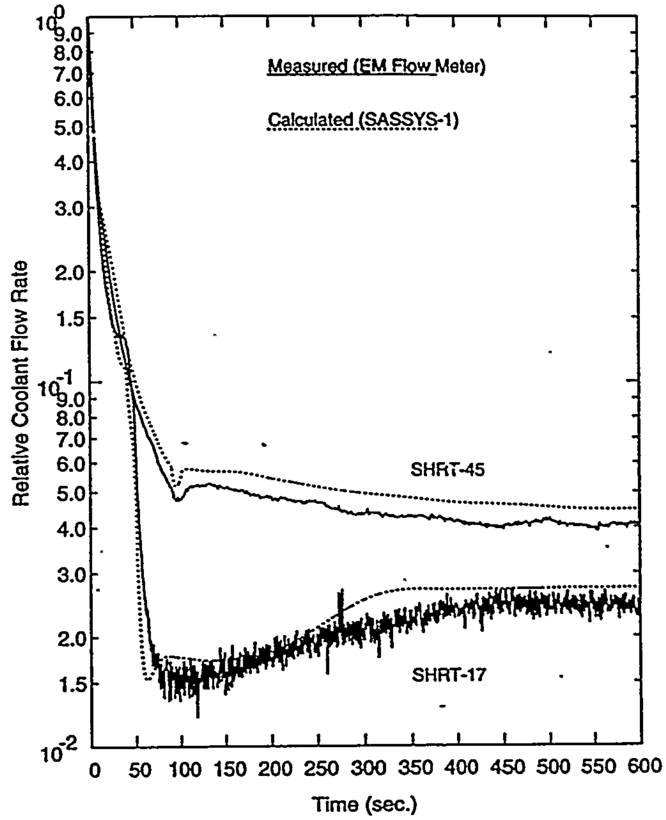

Fig. 9. Normalized Pump 2 to High Pressure Inlet Plenum Flow Rates, SHRT-17 and SHRT-45

In addition to the TTC thermocouples located near the top of the fuel, there were thermocouples at other elevations in XX09. Fig. 10 shows some computed and measured temperatures at the middle of the core, and Fig. 11 shows temperatures 14 cm above the top of the fuel. As shown in Fig. 2, the MTC-20, MTC-24, 14TC-37 and 14TC-41 thermocouples are in the region represented by channel 38. The effects of the higher powers toward the center of the core show up in the thermocouple measurements. There were also two thermocouples, OTC-1 and OTC-2, at the outlet of XX09. Figure 12 shows a comparison between the measured and calculated coolant outlet temperatures in XX09. The agreement between measured and calculated temperatures at these elevations is comparable to the agreement at the top of the fuel.

C. Evaluation of Results

Agreement between measured and calculated temperatures and flow rates for the SHRT-17 and SHRT-45 tests is remarkably good, especially where the measurements

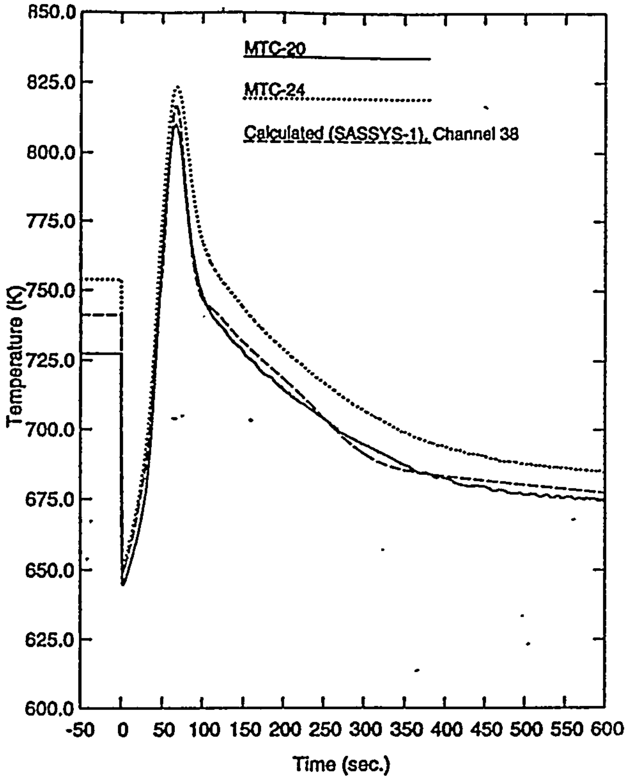

Fig. 10. SHRT-17 XX09 Mid-Core Temperatures

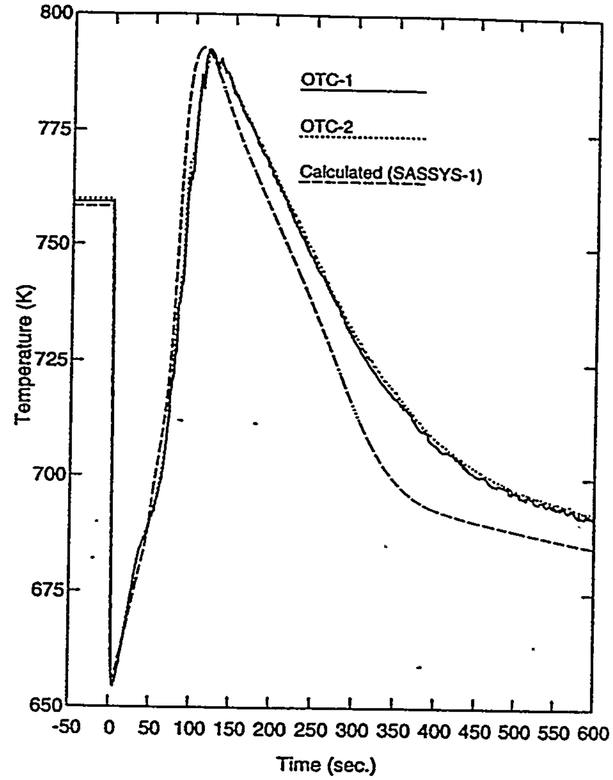

Fig. 12. SHRT-17 XX09 Subassembly Outlet Temperatures

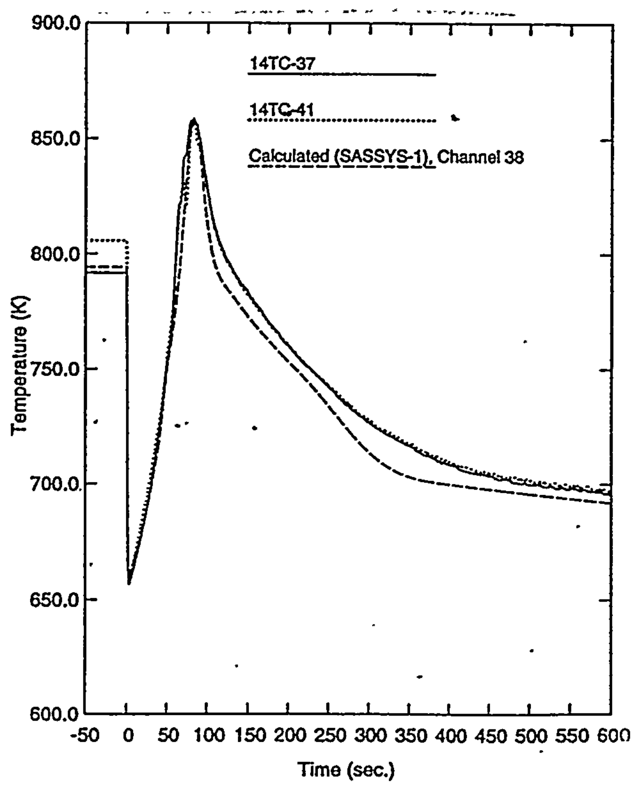

Fig. 11. SHRT-17 Coolant Temperatures In the Gas Plenum Region

appear to be reliable; and the main sources of disagreement can be identified. One source of disagreement is that there is probably a very small bias, in the range of .0025 to .005 of nominal flow, in the XX09 flow meter readings. This bias shows up at the very low flows encountered in these tests. Also, the calibration of one XX09 flow meter may have shifted a few per cent by the time of the SHRT-45 test. The other main source of disagreement is effects that the SASSYS-1 model did not account for accurately. The simple one-dimensional model in SASSYS-1 only gives an approximation to the temperature stratification in the outlet plenum in SHRT-17. This temperature stratification was apparently negligible in SHRT-45. Bowing and the somewhat random non-uniform spacing of the fuel pins was not accounted for in the calculations except in the central XX09 channel for the SHRT-45 case. This non-uniform spacing can change the readings of individual thermocouples by up to 10 K or more before the start of the transient. Also, the power gradient across the XX09 subassembly was not accounted for in the calculations; although in principle, the multiple pin model could account for it. This power gradient shows up in the individual thermocouple results, but it largely cancels out in the average TTC comparisons of Figures 3-5.

IV. SUMMARY AND CONCLUSIONS

The multiple pin model in the SASSYS-1 LMR systems analysis code provides for the rapid calculation of detailed steady-state and transient temperatures within a subassembly. Data from the SHRT-17 and SHRT-45 tests in the EBR-II reactor was used to validate the multiple pin model over a wide range of transient conditions. Agreement between measured and calculated temperatures and flow rates for these tests is remarkably good, and the main sources of disagreement can be identified.

ACKNOWLEDGMENTS

This work was supported by the U. S. Government under contract No. W-31-109-ENG-38.

REFERENCES

1. F. E. Dunn, "Integrated Intra-Subassembly Treatment in the SASSYS-1 LMR Systems Analysis Code," Proceedings of the Fifth International Topical Meeting on Reactor Thermal Hydraulics, NURETH-5, Salt Lake City, Utah, American Nuclear Society, pp. 408-415 (September 21-24. 1992).

2. F. E. Dunn, F. G. Prohammer, D. P. Weber and R. B. Vilim, "The SASSYS-1 LMFBR Systems Analysis Code," ANS International Topical Meeting on Fast Reactor Safety, Knoxville, TN, pp. 999-1006 (April 1985).

3. H. P. Planchon, R. M. Singer, D. Mohr, E. E. Feldman, L. K. Chang and P. R. Betten, "The Experimental Breeder Reactor II Inherent Shutdown and Heat Removal Tests - Results and Analysis," Nucl. Eng. and Design, 91,287 (1986).

4. D. Mohr, L. K. Chang, E. E. Feldman, P. R. Betten and P. R. Planchon, "Loss-of-Primary-Flow-Without-Scram Tests: Pretest Predictions and Preliminary Results," Nucl. Eng. and Design, 101, 45, (1987).

5. G. H. Golden, H. P. Planchon, J. J. Sackett and R. M. Singer, "Evolution of Thermal Hydraulics Testing in EBR-II," Nucl. Eng. and Design 101, 3 (1987).

6. J. Poloncsik, E. C. Filewicz, G. J. Kamis and J. T. Natoce, "The Experimental Breeder Reactor II (EBR-II) Instrumented Subassemblies, Inset XX09 and XX10," Proceedings of the Conference on Fast, Thermal, and Fusion Reactor Experiments, Salt Lake City, Utah, American Nuclear Society, pp. 1-276 to 1-287, (April 12-15, 1982).

7. Floyd E, Dunn, "Verification and Implications of the Multiple Pin Treatment in the SASSYS-1 LMR Systems Analysis Code", Proceedings of ARS'94 International Topical Meeting on Advanced Reactors Safety, Pittsburg, PA, pp. 58-65 (April 17-21, 1994).

8. D. S. Rowe, "COBRA-IIIC: A Digital Computer Program for the Steady State and Transient Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements," BNWL-B-82 (1971).

9. C. L. Wheeler and C. W. Stewart et al., "COBRA-IV-I: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores," BNWL-1962, Batelle-Pacific Northwest Laboratories (Mar. 1976).

10. T. L. George, R. E. Masterson, and K. L. Basehore, "A Modified Version of COBRA for Whole-Core LMFBR Transient Analysis," Trans. Am. Nucl. Soc., 32, 531 (1979).

11. K. L. Basehore and N. E. Todreas, "SUPERENERGY-2: A Multiassembly Steady-State Computer Code for LMFBR Core Thermal Hydraulics Analysis," PNL-3379, Pacific Northwest Laboratory (Aug. 1980).

12. Decay Heat Power in Light Water Reactors, American Nuclear Society, ANSI/ANS-5.1-1979.

13. F. E. Dunn and D. J. Malloy, "LMR Centrifugal Pump Coastdowns", Proceedings of an International Conference on Anticipated and Abnormal Transients in Nuclear Power Plants, Atlanta, Ga., American Nuclear Society, pp. VIII-21 to VIII-29 (April 12-15,1987).

14. E. B. Wylie and V. L. Streeter, Hydraulic Transients, Chapter Nine, McGraw-Hill, New York (1967).

15. E. B. Wylie and V. L. Streeter, Fluid Transients, Chapter Six, McGraw-Hill, New York (1978).

16. J. J. Lorenz and P. A. Howard, "Entrainment by a Jet at a Density Interface in a Thermally Stratified Vessel," Trans. of the ASME, 101, 538 (1979).