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RELAP5 CODE DEVELOPMENT AND ASSESSMENT AT THE SAVANNAH RIVER SITE (U)

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A paper proposed for presentation at the
1991 RELAP5/TRAC-B International User Seminar
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ABSTRACT

Over the past year, the focus of RELAP5 use at the Savannah River Site has been on code applications to reactor accidents having a direct bearing on setting power limits, with a lesser emphasis on code development. In the applications task, RELAP5/MOD2.5 has been used to predict the thermal-hydraulic system response to large break loss of coolant accidents and to provide boundary conditions for a detailed fuel assembly code. This paper describes the significant phenomena affecting the ability of RELAP5 to perform the system calculations, the benchmarking work completed to validate the application of RELAP5 to Savannah River Site reactors, and the results of the system calculations. This paper will also describe the code and model development effort and will describe briefly certain significant gains.

INTRODUCTION

Over the past year, the Code Development Group at the Savannah River Laboratory (SRL) has pursued an aggressive program of calculating the thermal-hydraulic system response of the Savannah River Site (SRS) K-reactor to the emergency cooling system (ECS) phase of a large break loss-of-coolant accident (LOCA). The purpose of the calculations was to provide thermal-hydraulic boundary conditions, notably fuel assembly inlet temperature and liquid flow rate, to a detailed fuel assembly code for the purpose of calculating the limiting fuel assembly power for the large break LOCA. This paper describes the application of RELAP5 to this task, including a description of the reactor, the accident assumptions, benchmarking the code against an appropriate data base, identifying key phenomena that could not be adequately represented with the current version of RELAP5/MOD2.5, and the model and code development approach used to obtain meaningful results while recognizing the code limitations.

The K-reactor is a six-loop, low pressure, low temperature, heavy water reactor. A sketch of the reactor system is shown in Figure 1. The reactor utilizes two horizontal heat exchangers in each of the six process water loops. Cool water enters an inlet plenum which distributes primary coolant to each of several hundred fuel assemblies. Water flows downward through the fuel assemblies and exits at the bottom into the moderator tank. Under normal operating conditions, the top of the moderator tank is maintained at a slight overpressure by a blanket gas system. Under LOCA conditions, the loss of primary system inventory causes the moderator liquid level to drop, vent paths from the top of the tank to atmosphere to clear, and the top of the moderator tank to be maintained at atmospheric pressure.

Coolant exits the moderator at the bottom through nozzles to each of the six process water loops. Process water pumps are located in the hot leg at the low point of the loops. Water circulates from the pump discharge to the heat exchangers, and then to the inlet plenum.

The Code Development Group uses RELAP5/MOD2.5 as the base code for the thermal-hydraulics system analysis for K-reactor power limits calculations. Modifications to the code used to represent phenomena unique to the SRS reactors will be described later. The RELAP5 model of the K-reactor represents all six process water loops, including the pumps and heat exchangers in a fashion fairly typical of normal RELAP5 modeling practice. The six loops feed a plenum model that is set up in a manner characteristic of a three-dimensional grid in polar coordinates. The model is called the r-θ model and is meant to represent three concentric rings, six azimuthal sectors, and a single vertical elevation. The fuel assembly models are lumped models representing the equivalent number of fuel assemblies communicating with each of the inner twelve plenum nodes. All the plenum nodes are connected to each other using the crossflow option, and the assemblies are connected to the plenum cells using the crossflow option. This option removes the momentum flux contribution from the solution of the momentum equation for those junctions where it is selected.

The moderator tank is nodalized in a manner similar to the plenum, except that three levels are used to represent the vertical dimension. The fuel assemblies connect the inlet plenum to the lowermost node ring of cells in the tank.

The scenarios selected for analysis require that two distinct tank nodalizations be employed. Each of the two decks was run to a steady state. Transient calculations were performed by renodalizing on restart to include the break geometry, then restarting from the steady state condition.

The remainder of this paper will address the accident scenarios that have been addressed with RELAP5, and will mention the key features of reactor response. The focus will be on those aspects of the calculation with which the code has difficulty, for these are the areas requiring further development to improve the calculational methods. The discussion will also address the data bases used to verify code performance. Phenomenological model development to address phenomena observed in the experiments and thought to be important to calculating system response will be described, although some of the models are still in their development stages. Lastly, the plans for continued code development at SRL will be described. A brief summary will draw all the ideas together.

CODE APPLICATIONS

Accident Scenario Selection

The six accident scenarios analyzed were determined outside the Code Development Group as the six large break scenarios representing the limiting conditions for the LOCA. They included three separate break locations and two modes of process water pump operation. The three locations were plenum inlet, pump discharge, and pump suction. The two pump modes are AC motors continuing to run, and AC motors tripped at two seconds, leaving only the DC pump motors running. All six scenarios were analyzed, and the plenum inlet with AC motors continuing to run (PIAC) was identified as the limiting transient.

The analysis covers the ECS phase of the LOCA. This phase begins approximately ten seconds after the break occurs, and extends to about 200 s, by which time the reactor has reached a relatively stable configuration. The initial phase of the LOCA is designated FI for the flow instability phase, that period during which the initial depressurization and flow reduction from the break precedes a reactor power reduction. During this phase, a flow instability in the fuel assemblies is possible, and a limiting fuel assembly power is determined to prevent safety criteria from being exceeded. Although RELAP5 must calculate the

LOCA through this initial phase of the accident, RELAP5 is not used to analyze this phase. The ECS phase is the natural progression of the accident once the power has been reduced from the reactor scram.

The general nature of the LOCA for each of the scenarios is summarized here by observing the tank level response. Figure 2 shows the level response for each of the calculated scenarios. Five of the six scenarios, including the PIAC case, show similar behavior. The tank level drops rapidly as system inventory is lost through the break. After about 40-60 s, the tank level equilibrates near 2 ft. This steady level is maintained as a mass balance is established between ECS injection flow and a reduced break flow resulting from the low tank level. The exception to this behavior is the PIDC transient. In this case, the break is near the high point in the reactor loop, but the head developed by pump operation with the DC motors is not sufficient to maintain a high break flow. Therefore, the system gradually refills as ECS water is supplied in excess of the break flow.

The key features of the PIAC transient are shown in Figures 3 and 4. These include minimum assembly flow and assembly inlet temperature, which is shown here as the hot plenum sector temperature. The minimum assembly flow is taken from that average assembly showing the least average liquid flow for a 10-second period during the transient. In this case, this occurs in the innermost ring of nodes in the break sector.

The temperature associated with the minimum assembly flow was taken as the hottest of the plenum sectors, rather than the temperature associated with the minimum flow sector. The reason for this was to accommodate the uncertainty associated with nonphysical mixing in the plenum and to assure a conservative result.

Problems were noted with several phenomena: air flow throughout the LOCA calculations was higher than data would support; pump degradation behavior caused nonphysical oscillations in the system performance; calculational dependence on tank nodalization remains troublesome; our ability to calculate the delayed AC trip scenario, one in which the reactor operator trips the AC pump motors according to operational guidelines at some time after the two seconds assumed for the DC analysis, remains limited.

These problems were generally addressed by recognizing their impact on the calculations and either forcing a conservative bounding condition or adding a margin to the final result to account for the effect. The most significant problem was air flow. Based on benchmark calculations discussed below, the minimum assembly liquid flow was found to be nearly independent of the calculated air flow, although assembly void fraction and pressure drop were impacted.

The tank nodalization dependence was addressed by using different input models for the AC and DC cases. This solution was acceptable for the six base cases, but was not suitable for analyzing the delayed AC trip scenario.

Benchmark Calculations

Code validation was extended beyond that already provided by existing RELAP5 developmental assessment and documentation by completing benchmark calculations against data from the 1989 L-Area tests. These included experiments to test the reactor system response to a range of moderator tank liquid levels under conditions of both symmetrical and asymmetrical pump operation with either AC and DC pump motors operating, or with just DC pump motors operating. Although LOCA conditions could not be represented exactly, backflow conditions at the plenum characteristic of a break in one of the loops could be simulated by turning off the pumps in a single loop.

The experiments used to validate RELAP5 were chosen for their similarity to LOCA conditions under both AC and DC operating conditions. They included AC10, AC4M, DC9, and DC10. In addition, the delayed AC trip scenario could be estimated with transient data taken following a pump trip in one of the loops, test AC-11.

The L-area tests were designed to provide detailed measurements of phasic flow rates in a single reactor loop, as well as pressures throughout the loop and in the inlet plenum. An array of pressure and level measurements in the inlet plenum provided profile information in the plenum. Flow rate measurements at three locations in the instrumented loop, as well as void fraction measurements at the pump suction and the plenum inlet provided a measure of loop flow rates as a function of tank liquid level, a traditional parameter to which other reactor characteristics, such as loop flow or plenum pressure, are related.

The results of a typical RELAP5 calculation of a steady state hold point are shown in Figure 5, plenum level distribution in test AC-10 at a tank level of 1.279 ft. In general, the results of the benchmark calculations showed that RELAP5 could represent key phenomena such as plenum liquid level and plenum pressure distribution fairly well, even in an asymmetric condition such as test AC-10. This finding had a key impact on the minimum fuel assembly liquid flow calculation, since flow into the assembly under conditions of a stratified level in the plenum is governed by a weir relationship. With only a slight modification, RELAP5 represented the weir relationship well, so an accurate level calculation established confidence in the assembly liquid flow calculation.

The benchmark calculations also showed that RELAP5 did a poor job calculating air flow. Under two phase conditions in which the gas component is principally air, RELAP5/MOD2.5 tends to calculate far too much air flow for a given liquid flow. This result impacts the plenum pressure calculation, but under the conditions mentioned above, a stratified liquid level in the plenum and weir flow into the fuel assemblies, the air flow rate does not have a significant impact on the liquid flow rate. For this reason, the calculation can still be employed. The air flow problem remains troublesome, though, and further applications of RELAP5/MOD2.5 to Savannah River reactors are hampered by this aspect of the calculation.

Results

The key LOCA results are shown in Table 1. The worst case scenario was determined by the minimum assembly liquid flow. Based on the results shown in Table 1, the PIAC scenario was identified as the worst case. The pump discharge break with the AC motors tripped at two seconds (PDDC) was nominally worse, but some of the conservatisms taken in the DC run were not employed in the AC run. Therefore, the PIAC proved to be the more limiting case after all the appropriate uncertainties were applied to the nominal result.

CODE DEVELOPMENT

Code Limitations

Based on the benchmark calculations, it was found that the code had difficulty representing:

- pump degradation, especially DC performance
- air flow, both in the assemblies and in the loops
- friction in annuli, which affects heat exchanger dp

- loop void fractions, which are probably closely tied to the air flow problem
- air entrainment from the tank free surface as a function of pump speed
- siphon behavior in the cold leg, depending on nodalization

It was also found that the code calculated plenum levels were representative of a largely unvented plenum. The L-area tests showed that the plenum levels were a sensitive function of the air flow to the plenum, and that loop behavior strongly affected the point at which the plenum changed from unvented to vented behavior. The RELAP5 calculations have not yet been extended into the region where this transition is expected to occur, though the PDDC calculations are not far removed. It is not clear that this effect is a code deficiency or limitation, but the sensitivity needs to be explored.

Table 1. Break Spectrum Calculation Results

	PIAC	PIDC	PSAC	PSDC	PDAC	PDDC
Time of minimum flow (s)	196	149	194	196	120	186
Minimum assembly liquid flow (gpm)	51.2	53.2	128.7	77.8	67.9	48.7
Assembly air flow (gpm)	90.4	18.6	193.4	17.6	220.3	32.3
Plenum pressure (psia)	15.8	14.8	36.3	13.3	21.8	12.8
Minimum tank level (ft)	1.7	5.5	2.8	2.3	2.1	1.7
Core flow (gpm)	30793	30793	64979	35758	46909	27808

KEY

PI = Plenum Inlet
PS = Pump Suction
PD = Pump Discharge

AC = AC & DC pump motors on
DC = DC pump motors on, AC pump motors tripped off at 2 seconds after the break

Model Development

Certain special models have been incorporated in the SRS version of RELAP5/MOD2.5. One of these is a weir flow model to represent the flow of water from the inlet plenum to the fuel assemblies under conditions of a stratified liquid level in the plenum. The model was essentially to force the code to use a small interfacial drag coefficient when the liquid level was determined to be low enough that stratified conditions were present.

Other special models addressed the single phase liquid friction factor calculation, both in pipes in the region of Re=4000, and in annuli, in which the RELAP5 laminar friction factor was too low. In fact, RELAP5 did not use a specific friction factor for laminar flow in annuli.

The air flow problem impacts several aspects of the RELAP5 system calculation. Although different methods have been used to work around this limitation, none represents a satisfactory long term solution. The three aspects of the air flow problem receiving the most aggressive treatment are:

1. Air and water flow in the fuel assemblies.
2. Tank nodalization sensitivity.
3. Plenum flow behavior.

The assembly air and water flow problem is being addressed in a manner similar to that used in the FLOWTRAN-TF code.[2] This work is described in another paper presented at this conference.[3]

The tank nodalization issue has been treated by addressing both the interfacial drag and the specific air-water behavior at the muff. The most fruitful approach to date has been the use of Kataoka-Ishii[4] interfacial drag in the loops, with crossflow junctions in the tank, and a specific stratification and entrainment model applied at the muff. This model modifies the interfacial drag coefficient depending on the flow rate and void fraction at the muff, and gives a reasonable representation of air entrainment from the tank free surface to the muff. Its use allows us to calculate both AC and DC reactor transient scenarios with a single tank nodalization, an inherently more satisfying approach that using separate nodalizations for different reactor conditions. In addition, it allows us to calculate the delayed AC trip scenario, one in which the reactor operator properly trips the AC pump motors after diagnosing that a LOCA has occurred. The calculation is more challenging than either of the other pump scenarios, because the pumps are tripped with a significant amount of air already contained in the reactor loops. This affects the tank level response, as well as the pump degradation behavior.

The problem of flow behavior in the plenum is difficult to treat because the independent variables determining the flow regime, in particular those which indicate the occurrence of stratified flow in the presence of a forest of tubes and a high nozzle inlet flow rate are not well known. The presence of a stratified liquid level is important to determining assembly flow rate, since the weir relation depends on stratification. A plenum nozzle experiment to help characterize this behavior over a range of possible plenum and nozzle flow conditions is currently under construction at SRL. This facility will be used to provide data for model development and benchmarking. It will also be used to help assess a newly developed three-dimensional version of RELAP5.

FUTURE WORK

The work at SRL will continue to be divided between applications and development, with support of experimental facilities and benchmarking occupying a position in between. Continued applications will address such topics as minimum acceptable ECS flow rates, small break analysis, and extension of the existing models to cover such applications as LOCA-FI and Loss-of-Pumping Accident (LOPA). A significant input model development effort is associated with the extension to LOCA-FI and LOPA, including development of a refined top shield model, a five-ring tank and plenum model, and possibly a new fuel assembly model. Further work to support a power ascension program will also be required, but the application of RELAP5 to that task will likely require code development.

The code development tasks will address those limitations discussed earlier, air-water behavior in the assemblies and in the loops, plenum flow regimes, tank-muff behavior, and nodalization sensitivity. Experiments at the INEL and SRL will be supported through both pretest calculations to help set test matrix parameters, and posttest analysis to develop or validate phenomenological models in the code.

SUMMARY

SRL has made extensive use of RELAP5/MOD2.5 over the past year, principally in support of Safety Analysis Report analyses. The code was successfully applied to accident scenarios to provide fuel assembly boundary conditions for a determination of power limits for the K-reactor startup. In the course of performing the SAR analyses, several code deficiencies were noted, mainly in the area of air-water behavior. Some work has been accomplished to address and rectify these code deficiencies, but further work is continuing to adequately resolve them. A brief description of the experimental and code development work at SRL to improve the RELAP5/MOD2.5 code was given.

REFERENCES

1. R. A. Dimenna and C. B. Davis, "RELAP5 Modeling of a Savannah River Site Reactor (U)", WSRC-MS-90-206, 1990 Joint RELAP5 and TRAC-BWR International User Seminar, Chicago, IL, September 17-21, 1990.

2. G. P. Flach et al., *FLOWTRAN-TF Code Description (U)*, WSRC-TR-90-413, December, 1990.

3. D. L. Caraher, "Air-Water Hydraulics Modeling for a Mark-22 Fuel Assembly with RELAP5 - Part 2 (U)", to be presented at the 1991 RELAP5/TRAC-B International User Seminar, Baton Rouge, LA, November 4-8, 1991.

4. I. Kataoka and M. Ishii, "Drift flux model for large diameter pipe and new correlation for pool void fraction", *Int. J. Heat Mass Transfer*, Vol. 30, No. 9, pp. 1927-1939, 1987.

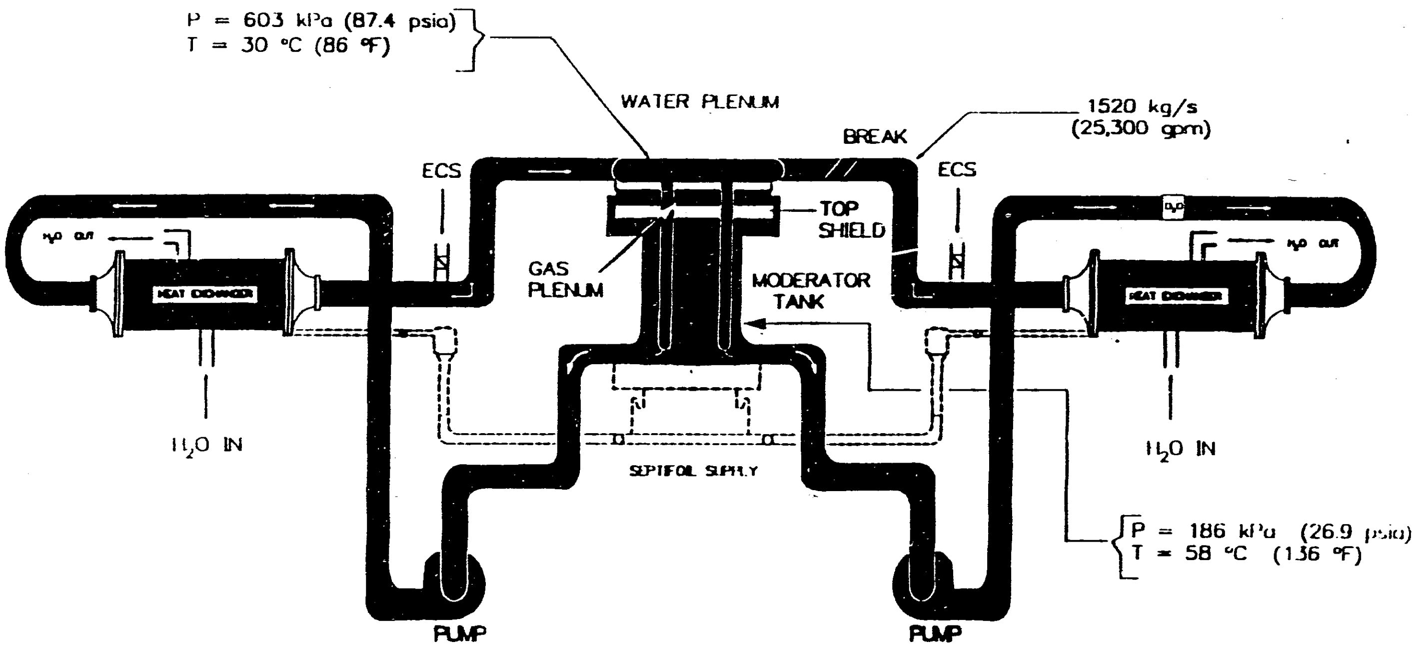


Figure 1. Sketch of the SRS Reactor Loop

Figure 2

Tank levels

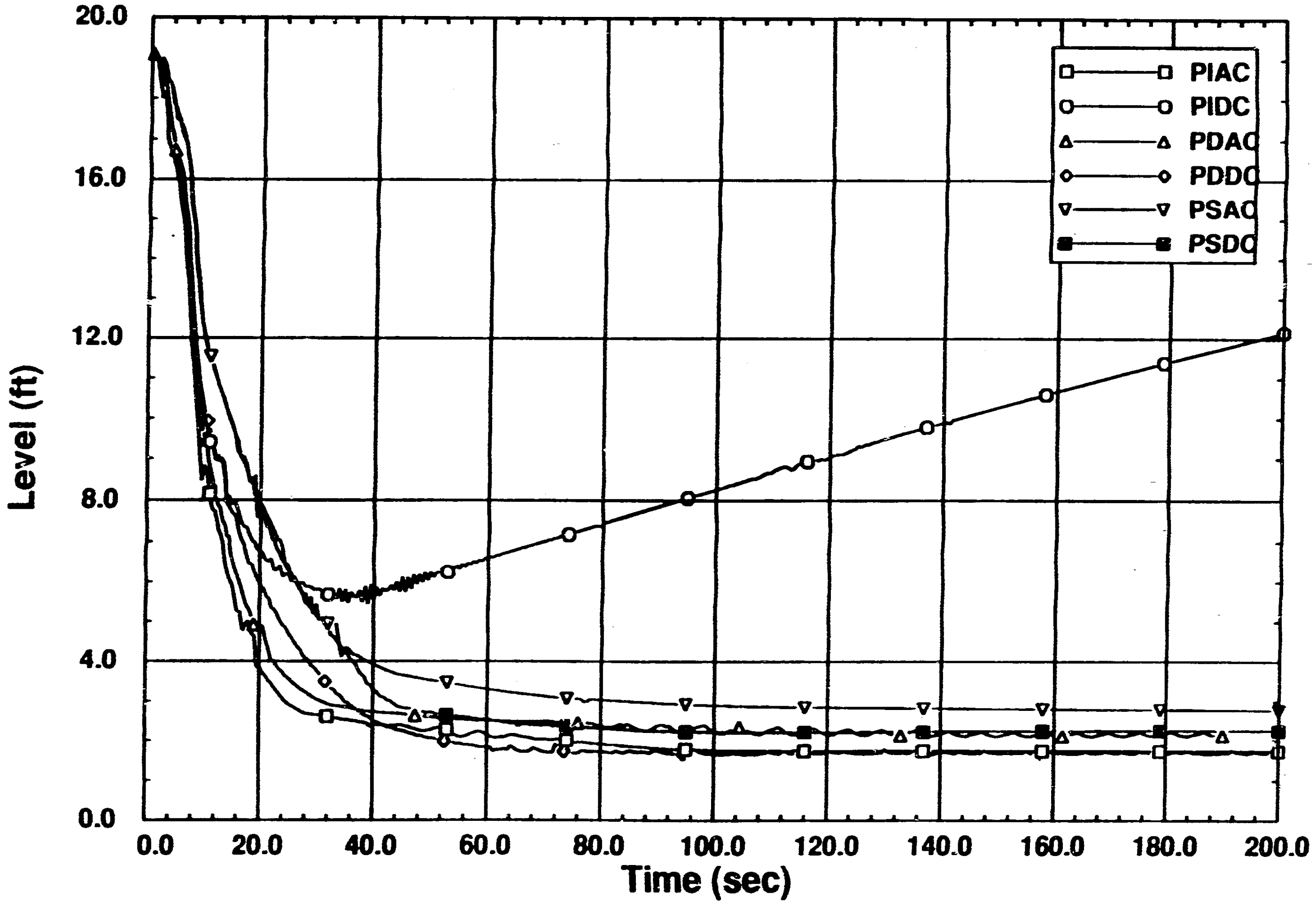


Figure 3

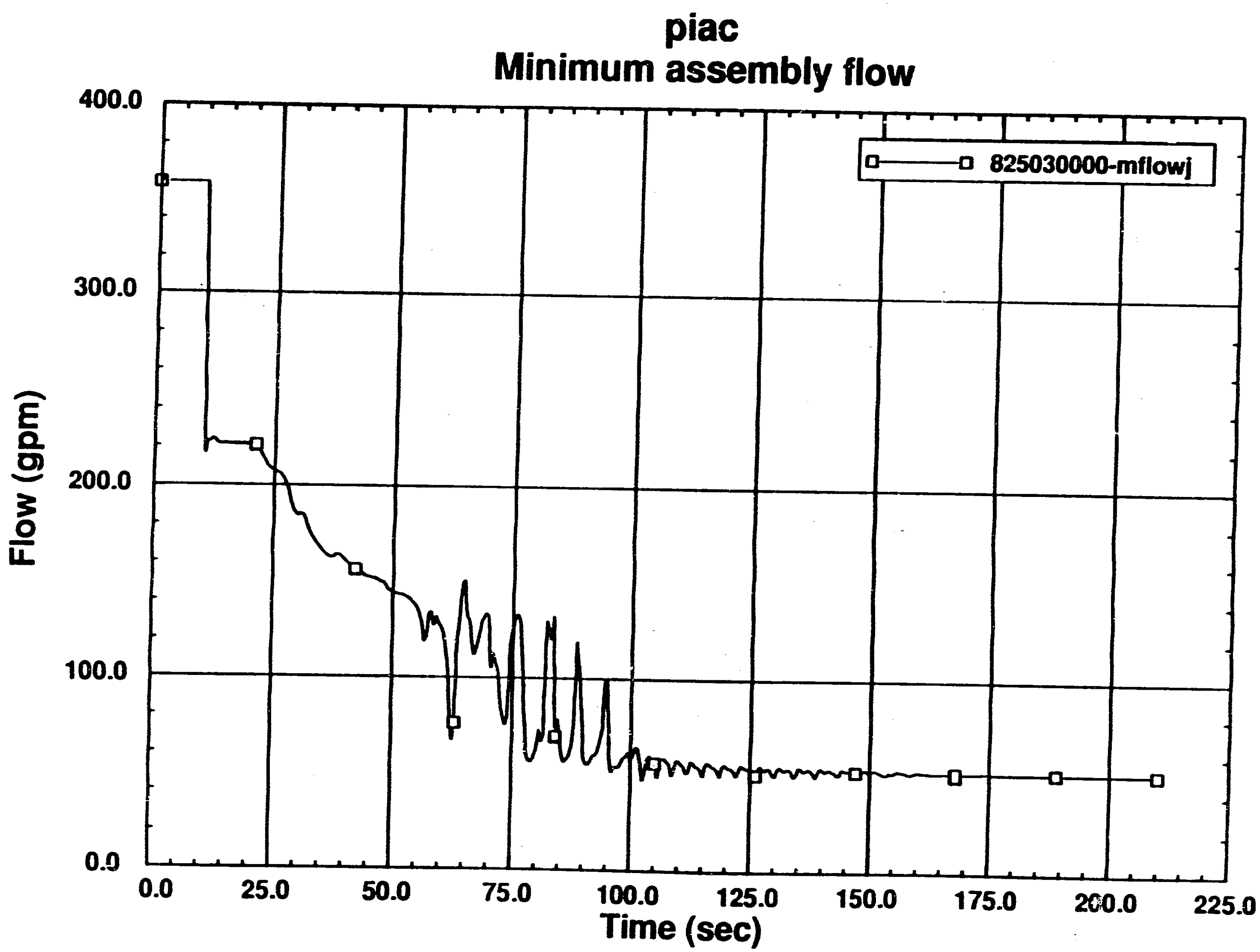
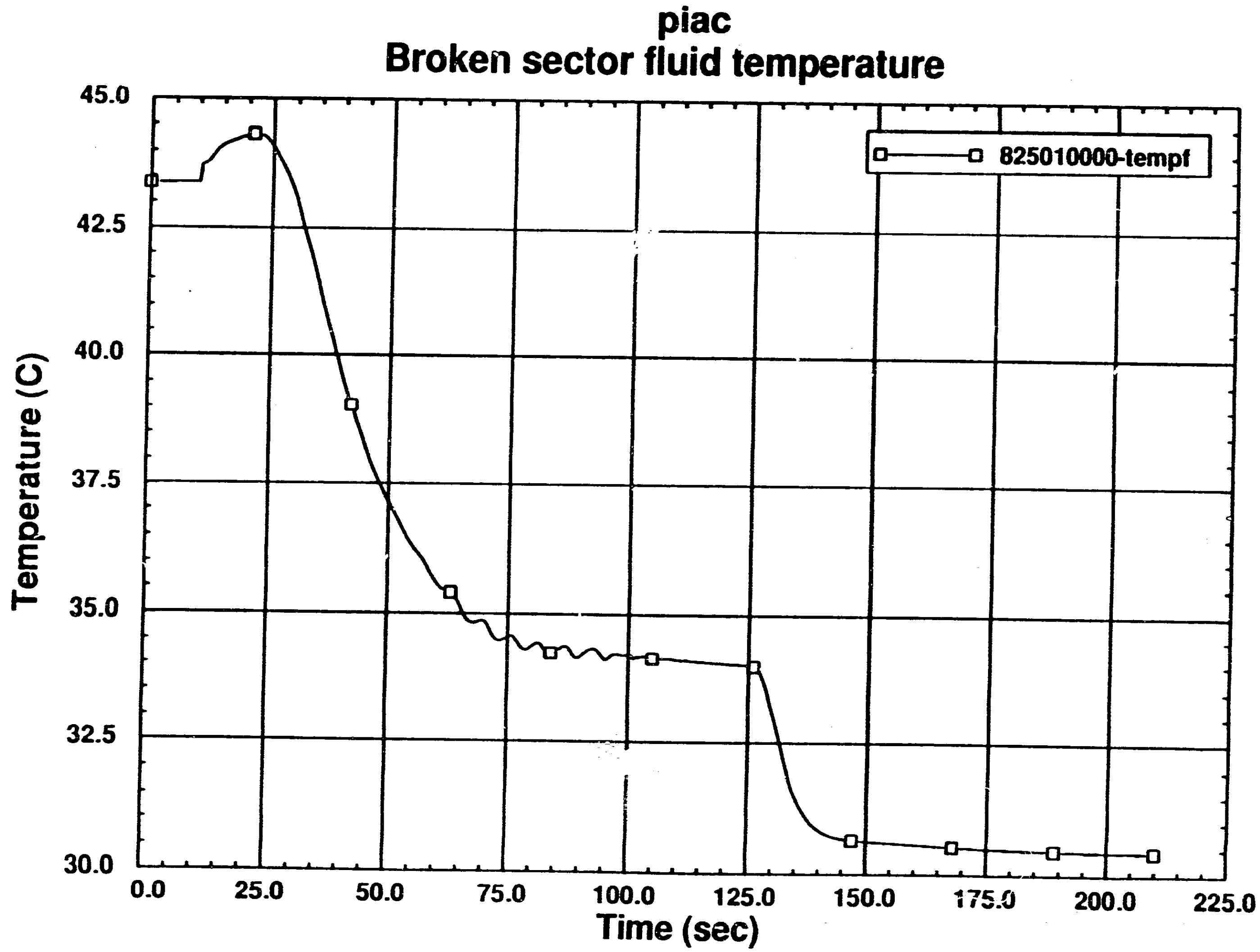

piac
Minimum assembly flow
825030000-mflowj
Flow (gpm)
400.0
300.0
200.0
100.0
0.0
0.0
25.0
50.0
75.0
100.0
125.0
150.0
175.0
200.0
225.0
Time (sec)

Figure 4


piac
Broken sector fluid temperature
45.0
42.5
40.0
37.5
35.0
32.5
30.0
Temperature (C)
0.0
25.0
50.0
75.0
100.0
125.0
150.0
175.0
200.0
225.0
Time (sec)
825010000-tempf

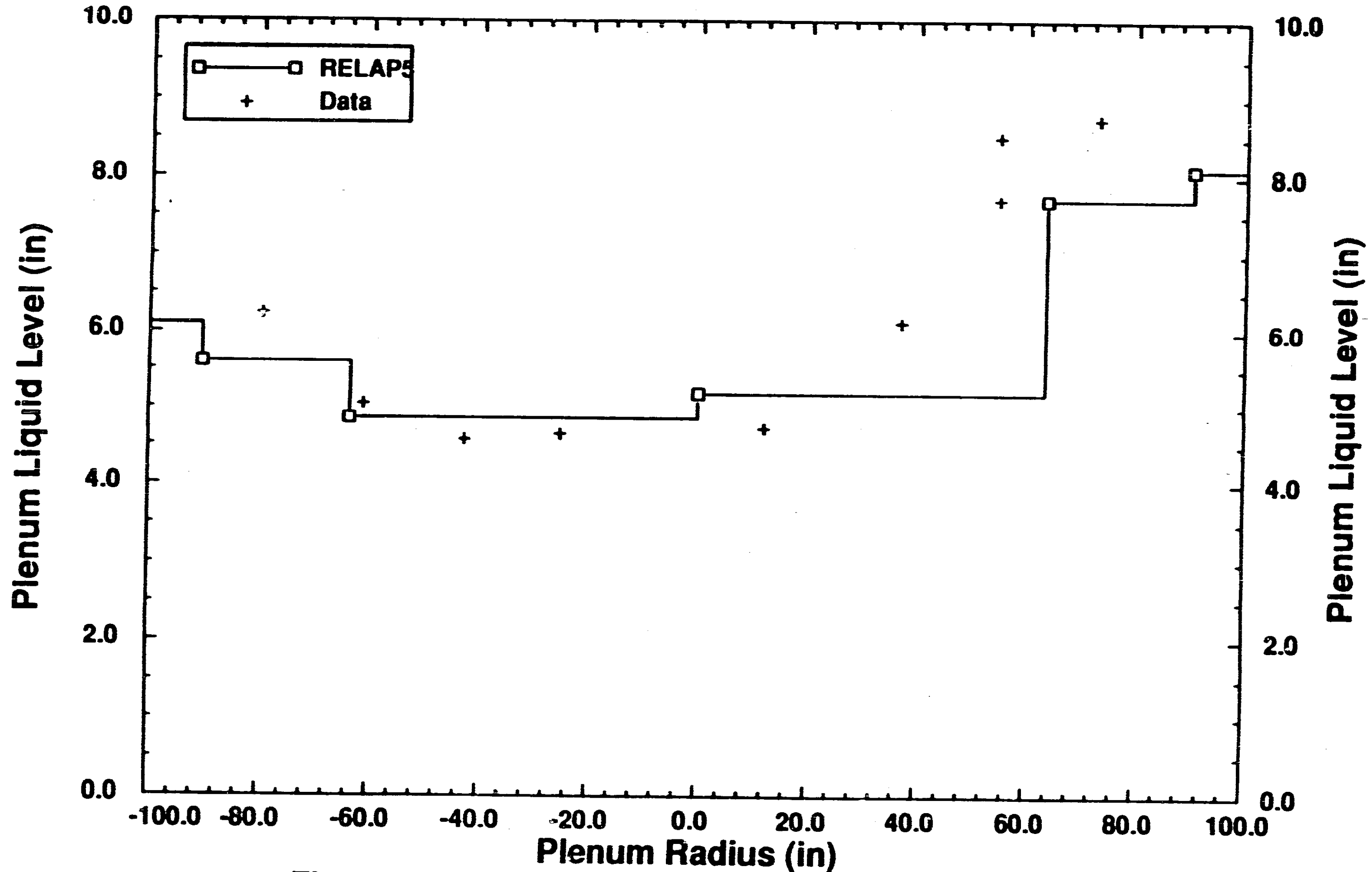


Figure 5. Test AC-10 Plenum Level Distribution Comparison at a Moderator Tank Level of 1.279 ft.

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