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Passive Safety Testing at the Fast Flux Test Facility

D. M. Lucoff

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Westinghouse Hanford Company
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PASSIVE SAFETY TESTING AT THE FAST FLUX TEST FACILITY

D. M. Lucoff
Westinghouse Hanford Company

INTRODUCTION

Accidents at Unit 4 of the Chernobyl Station and Unit 2 at Three Mile Island have changed the safety paradigm of the nuclear power industry. New emphasis has been placed on assured safety that protects not only the public but investment in the plant as well with less reliance on mechanical systems and human performance. This new approach to safety uses the intrinsic characteristics of nuclear reactors to assure nuclear shutdown and heat removal for any event even if the reactor scram system fails to operate and human error is made.

Reactor design upgrades to exploit intrinsic system features for passive safety have been considered for the Light Water Reactor (LWR), the High Temperature Gas-Cooled Reactor (HTGR), and the Liquid Metal Reactor (LMR). Each has the potential to incorporate passive safety features to improve safety margins and ensure system integrity in the event of an accident. The liquid metal reactor offers exceptional passive safety potential due to its low system pressure, high thermal conductivity and high boiling point of its liquid metal coolant, and its capacity to be cooled by natural convection. Because an LMR operates at a low system pressure, it has a low level of stored energy that could do damage in an accident. In addition, LMR fuel and structural reactivity feedback mechanisms return the reactor to a safe reactivity status in the event of over-temperature operation.

The design and licensing of a passively safe LMR, requires that reactor feedback mechanisms be thoroughly understood over the entire range of core conditions relevant to any transient event. While experience with LMRs has provided a solid basis for understanding core reactivity feedback coefficients over the normal range of core conditions, few experiments have been conducted to explore the full range of thermal-hydraulic conditions associated with unprotected transients or so-called ATWS (Anticipated Transients Without Scram) events. The RAPSODIE LOFWOS (Loss of Flow Without

Scram) test conducted in 1983, was the first attempt to show that LMR inherent feedbacks and natural convection cooling were sufficient to preserve core integrity. More recently EBR-II has successfully completed a series of Shutdown Heat Removal Tests (SHRT) that demonstrated an LMR's passive safety behavior for both an unprotected Station Blackout and Loss of Heat Sink events.

The next logical progression is to perform similar testing at the Fast Flux Test Facility (FFTF). The FFTF is a full size core operating at a power of 400 MWt with provisions for in-bundle temperature measurements. The FFTF is operated for the Department of Energy by the Westinghouse Hanford Company. The FFTF Passive Safety Testing Program began two years ago. The program has two near-term objectives: (1) to extending passive safety testing experience to a large size LMR to improve design analysis computer codes, e.g., SASSYS,[1] and (2) to develop and test passive safety enhancements that could be used for future LMRs. In addition, the FFTF program has a third, follow-on, objective to develop Technical Specification Surveillance to confirm that actual core reactivity feedback behavior assures predicted performance in the event of an off-normal transient.

Elements of the FFTF Passive Safety Testing Program

The FFTF Passive Safety Testing Program tries to quantify, to the extent possible, the component reactivity feedback in an LMR. Both static and dynamic tests are used to examine reactivity feedback effects to determine magnitude and time constant. In theory, different feedback mechanisms can be stimulated or emphasized by changing core power, flow, and inlet temperature. The resultant changes in fuel and structure temperatures alter core reactivity which can be measured by compensating movement of a calibrated control rod. The interpretation of these reactivity changes is complex and two different approaches to resolve measured data into specific thermal-hydraulic feedback mechanisms have been identified for use in the FFTF program. The first method, known as group-by-group separation, was suggested independently by Professor Karl Ott of Purdue University and by WHC scientific staff. This approach recognizes that it may be "impossible" to separate two feedback mechanisms that both depend on the same system temperature indicator, e.g., Doppler effect and fuel axial expansion which depend on the fuel temperature. However, it is entirely possible to measure the "group" effect of both feedbacks by carefully specifying the reactor parameters between two static reactivity states. Each group reactivity feedback is measured directly. Separation of each component feedback is not possible without making an assumption about at least one of the sub-group components.

The second method, known as the statistical approach, utilizes regression fitting and statistical inference methods to "unfold" individual coefficients of reactivity feedback from the aggregate data without preference to the specific core operating parameters that are varied or held constant. This approach can yield an unbiased evaluation of component reactivity effects which may not be possible with the "group-by-group" approach. On the other hand, the statistical approach may produce correlative results that lack causative links needed to understand the mechanism for feedback. The two approaches complement each other, in that, the group-by-group method relies on physical insight to determine "group" reactivity coefficients while the statistical approach is an unbiased evaluation of the amount of real information inherent in a set of reactivity feedback measurements. When the two approaches agree, we can confidently use the inferred coefficient. When the two approaches disagree, it indicates that we may be trying to force an understanding into the data that is beyond the measurements.

To the degree that FFTF static and dynamic test data can be separated in component feedback mechanisms, verification of models used in system codes like SASSYS and improved estimates of associated model uncertainties can be obtained. This is a key objective of the FFTF Passive Safety Testing Program. A specific reactor's test data can be applied to other reactors through the validation of design codes. A test of an off-the-shelf design computer program is whether it can predict a real reactor without having to resort to fudged up models. Indeed, off-the-shelf system analysis codes like SASSYS are being used to analyze PRISM and SAFR and to verify their passive safety characteristics. The credibility of these system models can be greatly enhanced if they are shown to make reasonably good predictions of

LMR reactivity feedback effects. The rationale driving the FFTF program is to provide a complete set of accurate static and dynamic measurements of reactivity feedbacks so that LMR system analysis code models can be confirmed.

The final element that completes the FFTF Passive Safety Testing Program is the development of passive safety enhancement features for an LMR. Activities centered around the development and proof-testing of the GEM (Gas Expansion Module). The GEM increases core neutron leakage when forced cooling to the core is lost. The amount of shutdown reactivity available with GEM depends on the number of GEMs used and the size of the core. A GEM is an empty pipe sealed at the top, filled with an inert gas, and inserted vertically into a core radial reflector or radial blanket position. The bottom of the GEM is open to the high pressure inlet plenum of the reactor. Figure 1 shows in simple pictorial form how GEM works. When the pumps are on and there is full flow through the core,the pressure in the high pressure plenum raises the sodium level in the GEM so that it stands above the active core height. If forced convection cooling is lost, the compressed gas trapped in the GEM will force the sodium level in the GEM to fall causing an increase in neutron leakage from the core and a corresponding reduction in core reactivity. GEMs are passive because they respond directly to loss of pumping power without need of flow instrumentation and actuation circuitry. GEMs are also a diverse shutdown mechanism, being fundamentally different from absorber "scram" rods and inherent feedback mechanisms.

In summary, the FFTF Passive Safety Testing Program contains three important elements: (1) the static and dynamic measurement of component reactivity feedback effects, (2) the application of these data to improve mechanistic models in LMR system analysis codes, and (3) the development of LMR passive safety enhancement features. These elements were pursued in the 1986-1987 testing program which is discussed in the next section.

The 1986-1987 Testing Program at FFTF

The 1986-1987 testing program looked at three fundamental areas of LMR passive safety. Two natural circulation cooling tests were performed to passive safety. Two natural circulation cooling tests were performed to demonstrate inherent core cooling capability from a refueling condition where there is no thermal driving head and in steady state operating conditions. A series of static state point measurements of reactivity feedback at a variety of power, flow, and inlet temperature conditions and a dynamic reactivity feedback measurement of rapid change in core flow rate were performed to provide the first assessment of reactivity feedback components in FFTF. The GEMs were built and proof-tested by conducting 13 "unprotected loss of flow" tests with GEMs. In addition to performing these tests, the SASSYS system model was used to make posttest calculations of the GEM loss-of-flow-without-scram (LOFWOS) tests. A summary of each test and the key results is presented below.

Natural Circulation Cooling Tests

Prior to performing the LOFWOS with GEMs tests, two natural convection cooling tests were performed in order to verify adequate natural circulation performance at high heat load and to demonstrate satisfactory response of special instrumentation installed in the reactor in support of the LOFWOS tests. In addition to providing the necessary performance verification, these two tests also provide additional data on the natural circulation performance of the FFTF primary system; these data can be used to validate systems analysis computer codes.

The Delayed Pony Motor Trip (DPMT) test, the first test, demonstrated the ability of an LMR to transition to natural circulation from nearly isothermal conditions without experiencing excessive core temperatures. This event was analyzed as part of the FFTF safety analysis. Calculations indicated that a delayed loss of pony motor flow was less limiting than a transition to natural circulation immediately following scram from full power. However, there was some concern about the ability to predict the development of natural circulation driving head and flow from initially isothermal conditions. The DPMT test also provided an opportunity to verify the dynamic response characteristics of some special fast response thermocouples which were installed in the reactor specifically for the 1986 Passive Safety Tests. These thermocouples, located directly above the outlet nozzle of a fuel assembly, were to be used to gather sodium outlet temperatures during rapid changes in flow rate and to provide additional plant protection.

The outlet temperature from most core assemblies is monitored via thermocouples located in the above core instrument trees about 0.889 meters above the outlet of the pin bundle (2.24 meters above the top of fuel). These thermocouples are located in gas filled thermowells and thus have very slow time response (time constant of approximately four minutes). There are eight FFTF core positions into which "open test assemblies" may be inserted. Two of these core positions, in Row 2 and Row 6, were provided with a special fast response thermocouple package, Figure 2. Instead of having the thermocouples located inside of a gas filled thermowell, a special plug was installed which allowed the thermocouple tips (five per assembly) to protrude out into the sodium stream. In addition, the fast response PIOTA thermocouples were located closer to the top of the fuel pins, (approximately 0.254 meters) in order to reduce the fluid transport delay time. The fast response PIOTA thermocouple time constant is estimated to be three seconds.

The second test, the Steady State Natural Circulation (SSNC) test, consisted of establishing primary loop natural circulation on decay heat and then taking the reactor to power with only natural circulation cooling of the core. The major purpose of this test was to better understand FFTF natural circulation performance at high heat loads since this is a potential final condition for a Loss of Flow Without Scram from high power.

The major conclusions drawn from these tests are:

1. Measured overall system response generally agreed quite well with predictions for both tests.

2. Modeling of radial heat transfer is required to accurately predict the response of the outer row of fuel assemblies.

3. Reactivity feedback correlations which were developed from data near

response under natural circulation conditions.

4. There is a significant time lag in the thermal response of the outer regions of the reactor vessel. This affects the expansion of the reactor vessel and its associated reactivity feedback effect.

5. Fast response PIOTA analytic models were confirmed as acceptable.

<u>FFTF Reactivity Feedback Components</u>

During February and March of 1986 (Cycle 8A) an extensive series of static reactivity state-point measurements were conducted at the FFTF. The Cycle 8A tests consisted of almost 200 measurements of control rod positions at selected power and coolant conditions. The reactor power was varied between 10% and 100%, coolant flow rate was varied between 67% to 100%, and core inlet temperature was varied between 303°C to 369°C. All reactor plant conditions during the test series remained within current reactor operational limits. The magnitude of the associated temperature reactivity feedbacks between test states was determined by converting rod movements to reactivity. Additional physics measurements were conducted before and during the test series to provide accurate rod worth information for post test rod movement reactivity conversions.

State-point changes may be characterized as one of seven "types" depending on the combination of reactor operating parameters varied. A description of each of these seven types is given below.

Type 1 The reactor inlet temperature and outlet temperature were held constant while the power and flow rate were varied. The inlet temperature was controlled by adjustments to the heat removal rates of the secondary coolant loops. The outlet temperature was maintained by keeping the power-to-flow ratio constant. To first order, the temperatures of the coolant and structural materials in the reactor did not change. Observed reactivity changes were attributed to changes in the temperature of the fuel material.

Type 2 The temperature of the fuel pin columns was held constant by keeping both the reactor power level and axially averaged coolant temperature constant while the flow rate was increased. The change in power-to-flow ratio was accommodated by increasing the core coolant inlet temperature. This type of state-point change emphasizes subassembly bowing feedback while minimizing fuel temperature reactivity feedback.

Type 3 This type of measurement was very similar to Type 2 in that an attempt was made to eliminate any fuel feedbacks. As the flow rate was increased (or decreased) the core outlet temperature was held constant. Reactor power was adjusted as needed to maintain constant fuel temperature. The core outlet temperature was held constant to eliminate expansion of the control rod driveline and any radial expansion at the top of the core.

Type 4 The temperature of the coolant entering the reactor core was varied in this measurement while holding the reactor power level and flow rate constant. For this change all components in the reactor experience a uniform temperature increase. The major feedbacks come from uniform radial expansion. The contribution from bowing of the subassemblies are small because the temperature gradients across the ducts will remain constant. The fuel temperature effects should be small because the fuel temperatures are most sensitive to changes in reactor power.

Type 5 The flow rate of the coolant entering the reactor core was varied in this measurement while the reactor power level and coolant inlet temperature were held constant.

The major reactivity feedbacks in this test come from subassembly bowing because the power-to-flow ratio changes.

Type 6 This measurement statically simulated the flow change transient which was conducted in Cycle 8B in which the coolant flow rate was rapidly reduced without any control rod movements. The reactor power was allowed to seek a new level to compensate for the reactivity change caused by the associated change in coolant temperatures. This compensation was obtained by changing the fuel temperatures.

Type 7 The reactor power coefficient was measured for each series of state-point measurements as an overall indicator of reactivity feedback.

Measurement Method

Transition between test states was made by moving a single test rod together with changes to reactor control parameters such as primary and secondary pump flow rates or heat rejection to ambient through the dump heat exchangers (DHXs). The entire control rod system, consisting of six rods, was rebanked periodically to allow positioning the test rod such that its total movement between sequential test steps relating to particular feedback effect measurement would be about the average bank position. Zero power differential test rod worth measurements were performed to accurately characterize the worth of the test rod in its movement about a fixed bank position.

After equilibrium reactor conditions were established for a specific state-point, an automated data collection routine was started. This routine collected primary and secondary flow and temperature conditions and unfiltered thermal and nuclear power values for six minutes at a rate of once per second. The routine then averaged the data and calculated calibration factors for the nuclear power signals, based on thermal power, and for primary flow rate signals, based on a primary flow rate calculation using a heat balance across the primary and secondary loops with accurate secondary flow rates measured by venturi meters. These calibration factors were then applied by the program to the raw nuclear power and primary flow sensor values. If these calibrated power and flow values, along with additional coolant temperature values, fell within specified test state variable tolerance ranges, then the operators recorded data displayed on control room terminal screens manually onto data sheets.

control room terminal screens manually onto data sheets.

State-point data have been validated; these data will be used to verify models used in the SASSYS computer code.

In June of 1986 a flow transient test was performed at the FFTF. The purpose of this test was to investigate dynamic reactivity feedback of a rapid decrease in core flowrate. For this test, control rods were not adjusted; thus, reactivity changes due to a flow rate perturbation were compensated by changes in reactor power level and inlet temperature. The test was originated from a steady state condition of 75% power, 96% flow rate and a core inlet coolant temperature of 343°C. A rapid decrease in the flow rate was induced by manually driving the flow controller electrodes apart. This resulted in an average flow rate decrease of 9%. The reactor power level decreased to 72% and then returned to near its initial value as the core inlet coolant temperature dropped. The drop in inlet temperature was the result of automatic controllers which hold the secondary coolant loop cold leg temperatures constant. Eventually the power again decreased as changes to the secondary cold leg temperature were made to increase the primary cold leg temperature to its initial value. The final reactor state was 71% power, and 87% flow with a coolant inlet temperature of 343°C.

End-point results of the flow transient test are very consistent with the static reactivity data measured in February. An evaluation of dynamic features of this test remains to be done.

GEM LOFWOS Tests

Testing of the GEM (Gas Expansion Module) was divided into two parts, (1) subcritical verification of GEM worth and response to a flow coastdown and (2) a LOFWOS proof-test demonstration. The first part of the GEM testing program was conducted in May of 1986 and had as its measurement objectives the following:

1) GEM reactivity worth (pumps on to pumps off).
2) GEM reactivity versus time during a primary pump coastdown.

3) Sensitivity of GEM response to system temperature.

4) GEM replacement reactivity worth versus a reflector assembly.

Nine GEMs were fabricated and loaded into symmetric Row 7 core locations. All of the above measurements were performed with the reactor near critical, but subcritical.

Figure 3 shows GEM reactivity as a function of time after pump trip for two different primary system temperatures. The shift to the left of this curve for the higher primary system temperature is due to a shift in the pre-trip sodium level in GEM towards the top of the active fuel column as the temperature of the GEM gas plenum becomes hotter. At 227°C, the total worth for the pumps-on to pumps-off condition of nine GEMs in FFTF was measured to be -1.31$ or -14.5¢ per GEM.

A three dimensional diffusion theory calculation gave a predicted GEM worth for pump-on to pumps-off of -19.2¢. In addition, the worth of replacing an inconel radial reflector with a GEM (pumps-on) was measured to be -13¢.

In July, 1986, the nine GEM devices were returned to the core periphery for prototypic LOFWOS testing. To support this test series, major safety analysis and engineering packages were prepared to eliminate the automatic scram following primary pump trip. These changes were directed at providing comparable levels of PPS protection, to that specified in the FSAR, for all test conditions.

The first LOFWOS test series was conducted by leaving the primary pump pony motors on throughout the transient so that the minimum flow reached in each test was 9%. The reactor was taken to the target power level (with the revised PPS in place), the primary pump main motors were tripped, and the resulting thermal transient was observed for 15 minutes. The tests were run from 10, 20, 30, 40 and 50% of 400 MWt. Peak temperature for the test series was approximately 493°C. The tests were then repeated with the same initial conditions, except the primary pony motors were left off, so that a transition was made directly to natural circulation flow in the primary system. Tests were run from 10, 20, 30, 40, 45, and 50% power. The peak assembly outlet temperature for these two types of tests as measured by the fast response thermocouples is shown in Figure 4.

In addition to measuring core power, core flowrate, and the outlet temperature of the highest power fuel assembly, the core reactivity was measured during the LOFWOS transient. The measured and predicted values for all four of the measured parameters for the LOFWOS test from 50% power (200 MW) to natural circulation flow conditions are compared in Figure 5. Predictions were calculated with the IANUS code, a system model developed for the FFTF that uses only one thermal hydraulics channel to model the core. A comparison of predicted and measured reactivity as a function of time after pump trip suggests that the actual GEM worth was larger than the assumed value of -1.46$ used in the calculation. The assumed GEM worth of-1.46$ was based on the subcritical worth measurements; however, apparent GEM worth increased as initial test power increased to an inferred value of about -1.6$ at 50% power.

The overprediction of "PIOTA Temperature" shown in Figure 5 is due to (1) an overprediction of core power because the assumed GEM worth was too small and (2) the inability of the single channel IANUS model to adequately handle flow redistribution that occurs during natural circulation flow. Subsequent calculations with a six channel core model in the SASSYS code showed substantially improved agreement. Figure 6 shows the "posttest" SASSYS calculation of the hot assembly PIOTA temperature compared with measurement and the "pretest" IANUS calculation. The multichannel SASSYS calculation predicts a peak temperature of about 532°C which is only 22°C above the observed value. However, the SASSYS model still requires some refinement in that the timing of the peak is delayed by about 40 seconds and the fine structure of the core reactivity is not consistent with measurement.

Future Activities

The Passive Safety Testing Program at FFTF has only just begun. Additional unprotected transient testing should be performed without GEMs. GEM reactivity feedback overwhelms structural feedbacks which are of interest. In order to fully understand structural reactivity feedback during an unprotected, transient, additional static-feedback measurements should be done at very low power levels (minimize fuel feedback). Finally, the FFTF should begin to define and demonstrate Technical Specification surveillance methods that verify core passive safety feedback coefficients without requiring that unprotected transient testing be performed on a regular basis.

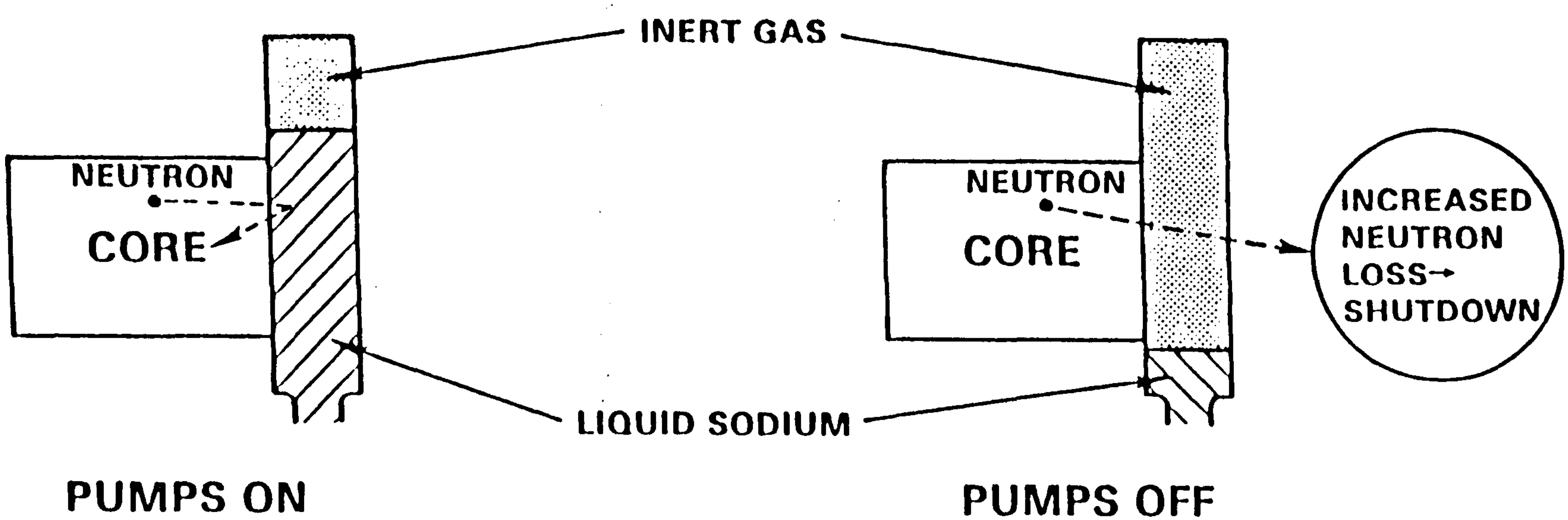
References

1. F. E. Dunn, et al, The SASSYS-1 LMFBR Systems Analysis Code, ANL/RAS 84-14, June 1984.

PASSIVELY SAFE LMR

GEM (GAS EXPANSION MODULE)

ASSEMBLY IN FIRST RADIAL BLANKET ROW



HEDL 8701-068.8

FIGURE 1.

FIGURE 1.1

FAST RESPONSE PIOTA/FUEL HANDLING SOCKET

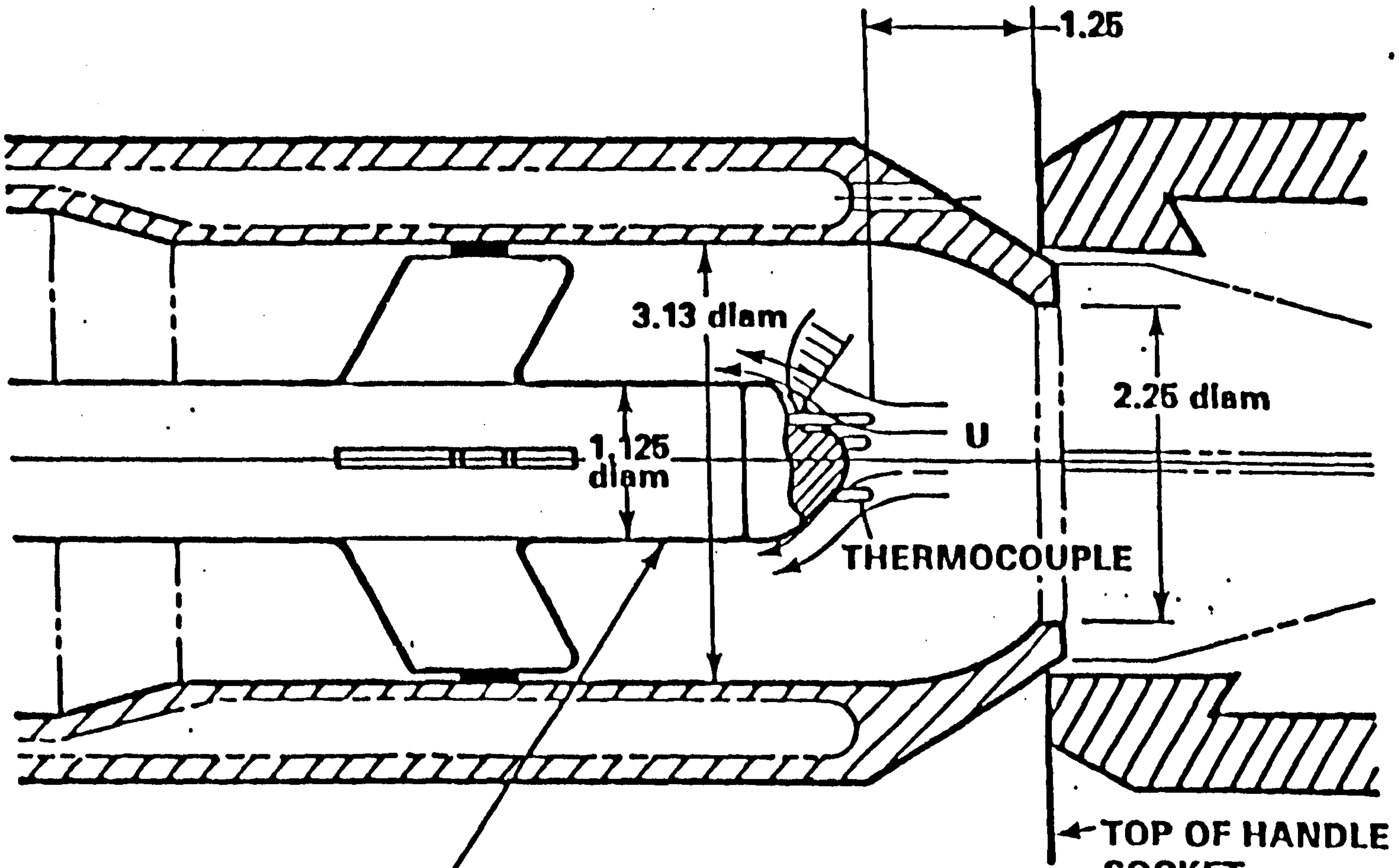


FIGURE 2.

HEDL 8604 006 84

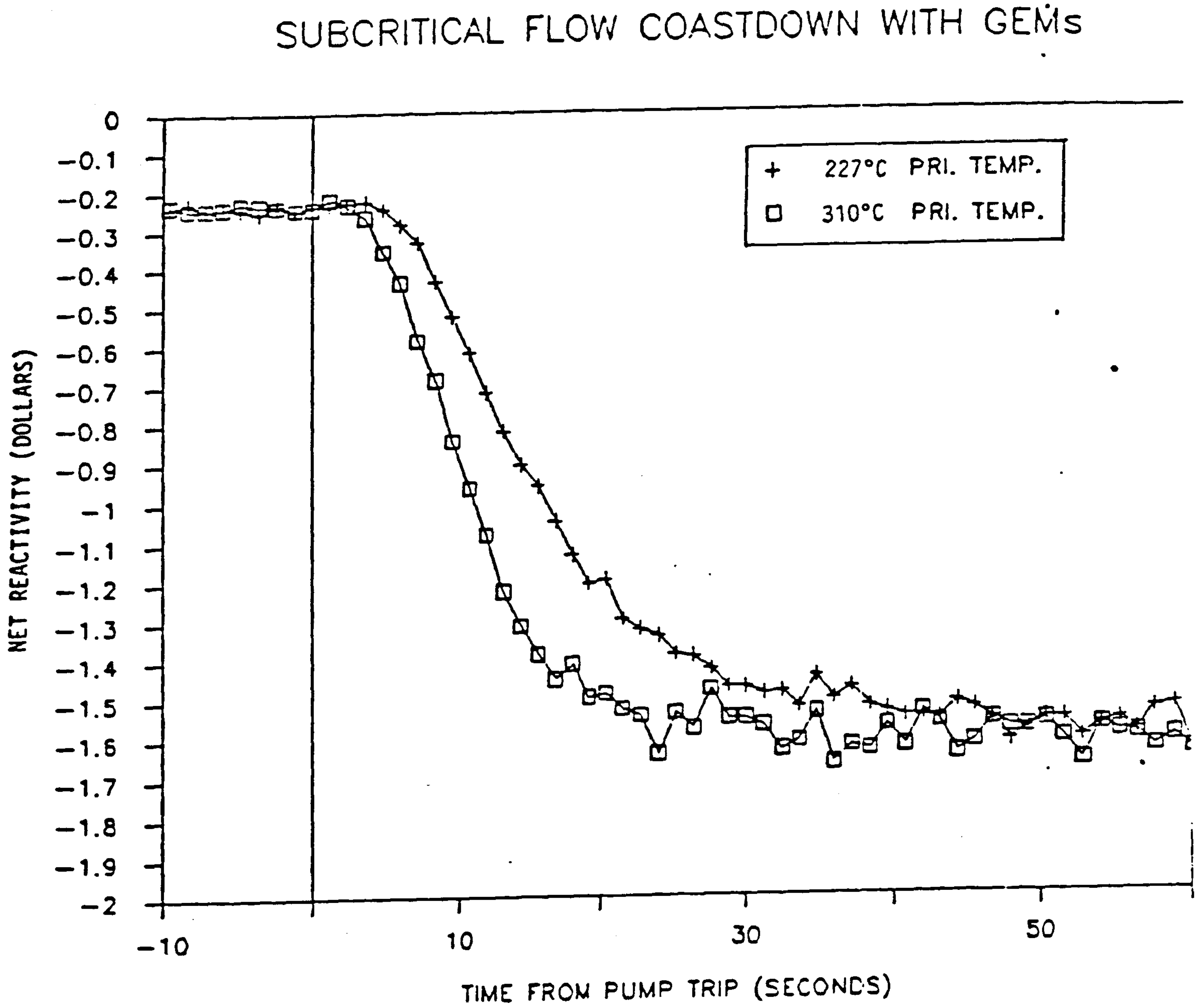
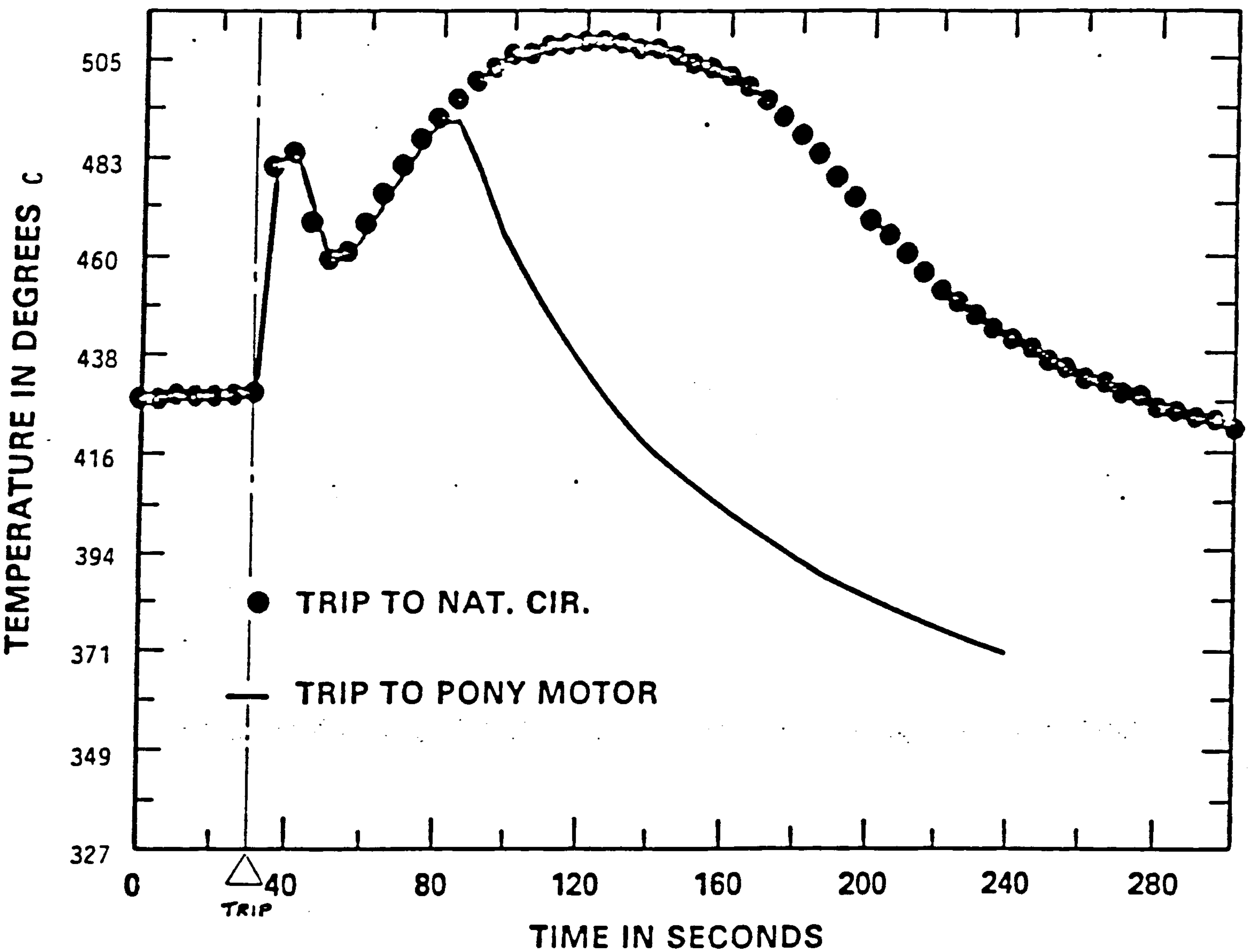

SUBCRITICAL FLOW COASTDOWN WITH GEMs
227°C PRI. TEMP.
310°C PRI. TEMP.
NET REACTIVITY (DOLLARS)
TIME FROM PUMP TRIP (SECONDS)
0
−0.1
−0.2
−0.3
−0.4
−0.5
−0.6
−0.7
−0.8
−0.9
−1
−1.1
−1.2
−1.3
−1.4
−1.5
−1.6
−1.7
−1.8
−1.9
−2
−10
10
30
50

FIGURE 3.

COMPARISON OF ROW 2 PIOTA TEMPERATURE

FOR 50% PM AND NC TESTS



HEDL 8705-046.12

FIGURE 4.

LOFWOS (W/GEMs) FROM 50% POWER/100% FLOW TRIPPED TO NATURAL CIRCULATION

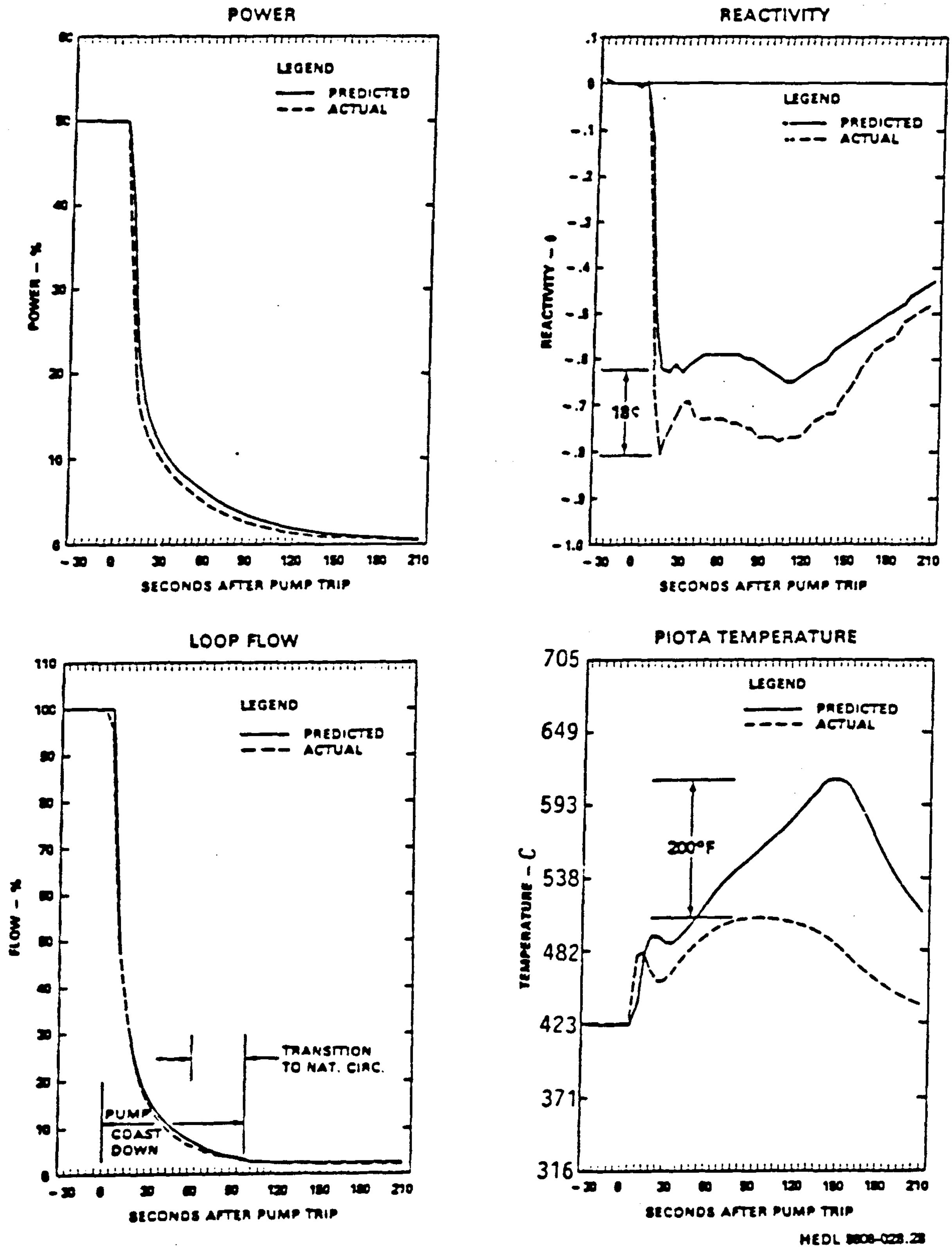


FIGURE 5.

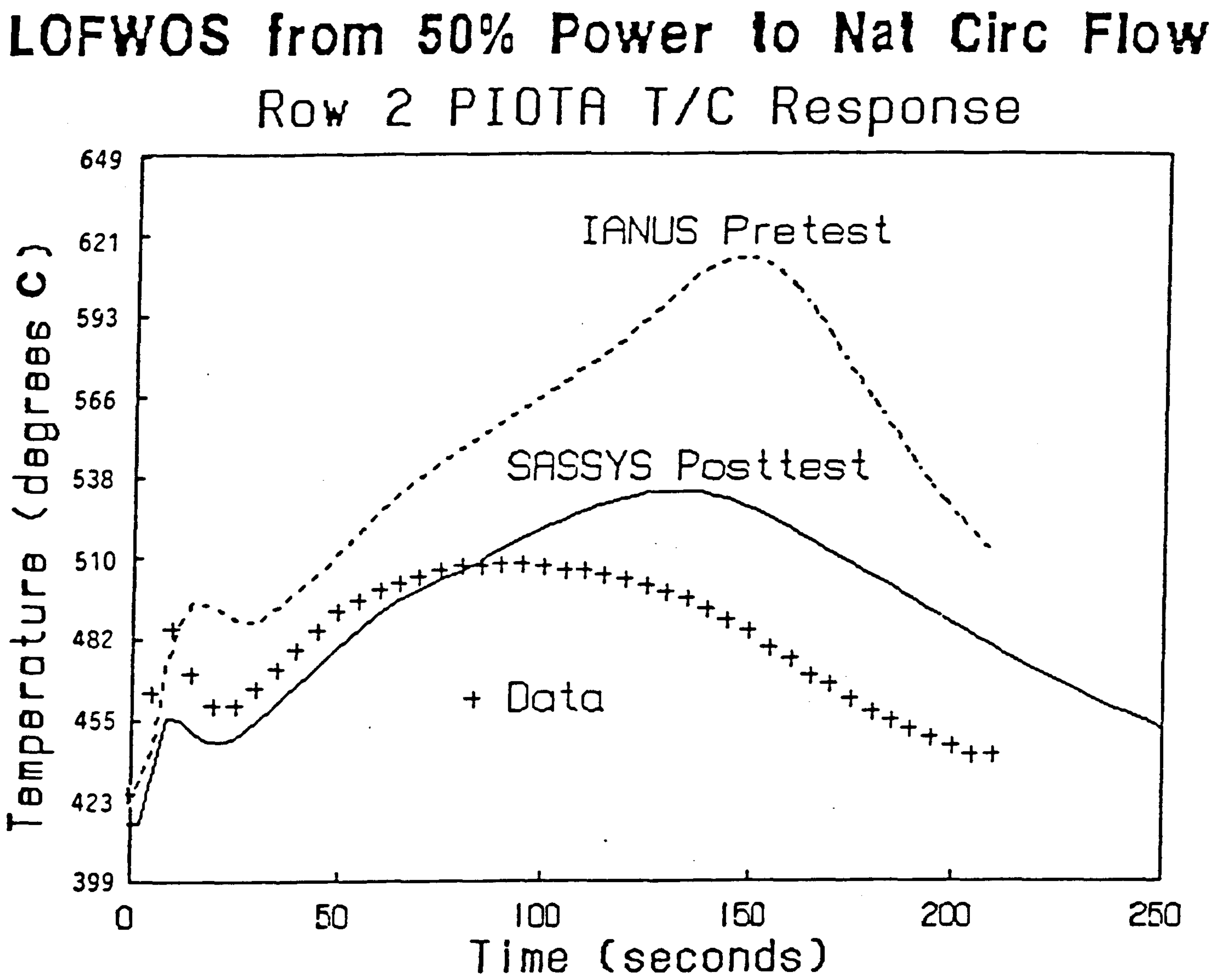

LOFWOS from 50% Power to Nat Circ Flow
Row 2 PIOTA T/C Response
IANUS Pretest
SASSYS Posttest
+ Data
Temperature (degrees C)
Time (seconds)
649
621
593
566
538
510
482
455
423
399
0
50
100
150
200
250

FIGURE 6.