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Nuclear Systems

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DCP/NRC1491
Project 711

November 9, 2001

Document Control Desk
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

ATTENTION: Mr. Andrzej Drozd, NRC, MS 10H1

SUBJECT: Westinghouse Responses to NRC Requests for Additional Information

Dear Mr. Drozd:

Attached please find the Westinghouse responses to the following Requests for Additional Information (RAIs) related to the pre-certification review of the AP1000:

P69	P72
P71	P73

The RAIs enclosed as Attachment 1 are considered non-proprietary to Westinghouse. Please contact me if you have questions on this issue at 412-374-5355.

Very truly yours,

M. M. Corletti
Passive Plant Projects & Development

/Attachments

1. "Westinghouse Response to NRC Requests for Additional Information," Westinghouse Non-Proprietary, dated November 2001

cc: H. A. Sepp	Westinghouse, Monroeville, PA	1L	
C. B. Brinkman	Westinghouse, Rockville, MD	1L	1A

D063

DCP/NRC1491
Project 711

November 9, 2001

Attachment 1

Westinghouse Non-Proprietary Response

to Requests for Additional Information

Regarding the AP1000 Pre-Certification Review

REQUEST FOR ADDITIONAL INFORMATION

RAI: P69

Question:

Provide justification that the noding detail in NOTRUMP is adequate to predict core uncovery at elevated voided conditions and at low pressures such as might occur during small break LOCA at AP1000. Provide justifications that heat transfer and critical heat flux correlations within NOTRUMP are adequate to conservatively calculate core cooling at these conditions both for the average core and for the hottest fuel pin in the core. Provide comparisons with the void fractions and core heat fluxes for the ACHILLES and G-2 tests that were evaluated using NOTRUMP as described in WCAP-14807. Justify that the conditions for these tests encompass those calculated for AP1000.

Westinghouse Response:

The noding utilized in the NOTRUMP model was developed based on the two-phase level swell simulations documented in Section 4 of the NOTRUMP Final Validation Report (RAI P69-1). Sensitivity studies were performed in which the noding was varied until significant changes in results were no longer observed. The final noding selected, representing 1-foot core node heights, represents this point. This noding size was subsequently utilized on all Integral test facility simulations (OSU and SPES) as well as the final AP600 plant simulations. For the AP1000 plant simulations, this noding size was carried forward resulting in the addition of two core nodes to reflect the added core length of the AP1000 plant design.

The NOTRUMP rod-to-coolant heat transfer logic is described in Appendix-T of Reference RAI P69-2. The Macbeth correlation (Reference RAI P69-5), utilized for the calculation of critical heat flux, is valid for the pressure range of 14.7 psia to 2750 psia. The AP1000 plant simulations performed in Reference RAI P69-3 showed that DNB does not occur. When DNB does not occur, the NOTRUMP heat transfer package utilizes the maximum of the forced convection, natural convection and Thom nucleate boiling correlation (Reference P69-4). NOTRUMP uses this logic for the range of void fractions approaching 100% assuming DNB has not occurred. NOTRUMP comparisons to test data using this approach show that NOTRUMP conservatively under-predicts core mixture level.

For AP600, the NOTRUMP code predictions of core void fraction did not exceed 85%. For the transients simulated with NOTRUMP for the AP1000 plant design, only the DVI line break and Inadvertent ADS actuation cases result in core node void fractions which exceed 85% for any significant period of time. In these cases, only the top core node (1-foot) was observed to be above 85% (see response to RAI P68). Westinghouse believes the NOTRUMP treatment of high core void fraction is acceptably conservative for AP1000 for the following reasons:

- The heat transfer package in the NOTRUMP code is only responsible for determining the thermal-hydraulic response of the Reactor Coolant System, not fuel rod cladding heat-up. The cladding heat-up calculations are performed by the SBLOCTA code, which assumes

single-phase convection to vapor. The NOTRUMP code only considers the average channel conditions during the thermal-hydraulic simulation with the detailed rod heat-up calculation being performed by the SBLOCTA code. Should fuel uncovery be predicted for the average channel, boundary conditions are transferred to the rod heat-up code (SBLOCTA) which calculates the hot rod response following an SBLOCA event. In the absence of average rod uncovery conditions, SBLOCTA calculations are not performed. Since the NOTRUMP simulations of the core uncovery/level swell simulations, as documented in Section 4 of Reference RAI P69-1, indicate that NOTRUMP conservatively predicts core mixture level for a wide range of test conditions, perturbations to heat transfer, under high void fraction conditions, are expected to have a negligible effect on the predicted core mixture levels.

- For the preliminary analyses performed, only the top core node (top 1-foot of the core) was shown to have void fractions in excess of 85% for any significant period of time. Since the upper plenum void fractions were observed to reach a maximum of 66% for the simulations performed, if degraded heat transfer were to be assumed in the top core node, the core would still remain covered with only a minor perturbation in overall mixture level expected.
- The AP1000 preliminary analyses were performed utilizing conservative Appendix-K modeling features/assumptions (primarily ANS 1971 Decay heat + 20%) while neglecting the potential benefit of containment pressurization, as a result of the transient. Including the effect of containment back-pressure would reduce the calculated void fractions.

Figures P69-1 through P69-4 present void fraction comparisons between test data and the NOTRUMP simulations for the ACHILLES test series. Heat flux test data was not readily available for the test series to provide comparisons with the available NOTRUMP simulations. The range of conditions, simulated in this test facility, encompasses those expected of the AP1000 plant design.

The G2 test void fractions and heat flux distributions can be found in Figures P69-5 through P69-114. A constant uncertainty of +/- 0.076 psid was applied to the test data for all test simulations for generation of the void fraction information. The void fraction comparisons between the NOTRUMP and G2 test data are presented with the NOTRUMP simulations compared to the high and low uncertainty data ranges. No uncertainties are applied to the G2 heat-flux data comparisons with the NOTRUMP simulation results. The G2 test data is applicable to the range of conditions predicted to occur for the AP1000 plant design and the tests simulated as part of the NOTRUMP Final Validation Report (Reference RAI P69-1).

References:

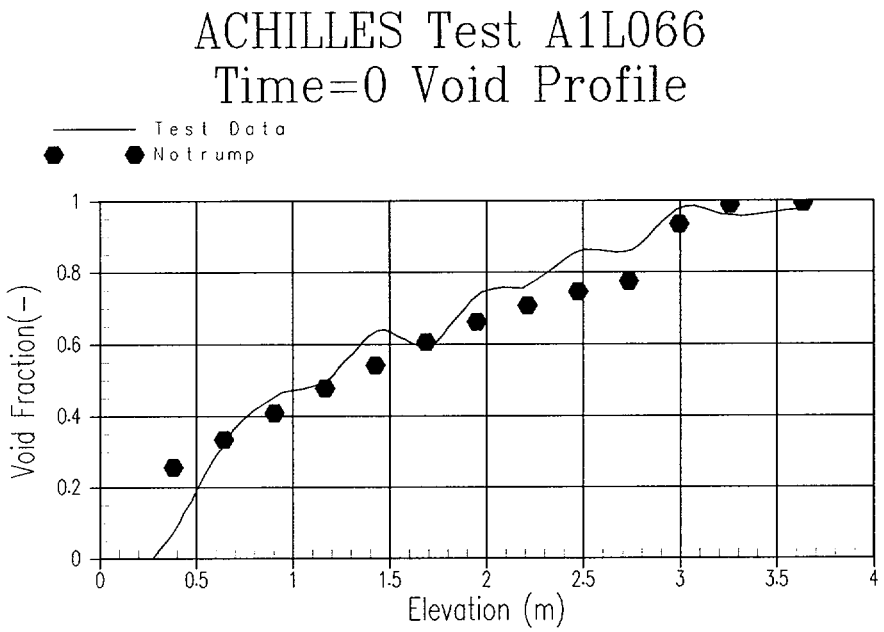
RAI P69-1 WCAP-14807, Revision 5, "NOTRUMP Final Validation Report For AP600," R. L. Fittante, et al, August 1998.

RAI P69-2 WCAP-10079-P-A, "NOTRUMP: A Nodal Transient Small Break And General Network Code," P. E. Meyer, August 1985.

RAI P69-3 WCAP-15612, "AP1000 Plant Description and "Analysis Report," December, 2000.

RAI P69-4 Proceedings of the Institute of Mechanical Engineers, 180, pp. 226-246, London, 1965-1966, "Boiling in Sub-cooled Water During Flow Up Heated Tubes of Annuli," Thom, J. R. S., et al.

RAI P69-5 Proceedings of the Institute of Mechanical Engineers, 1965-1966, "An Appraisal of Forced Convection Burnout Data," Macbeth, R. V.



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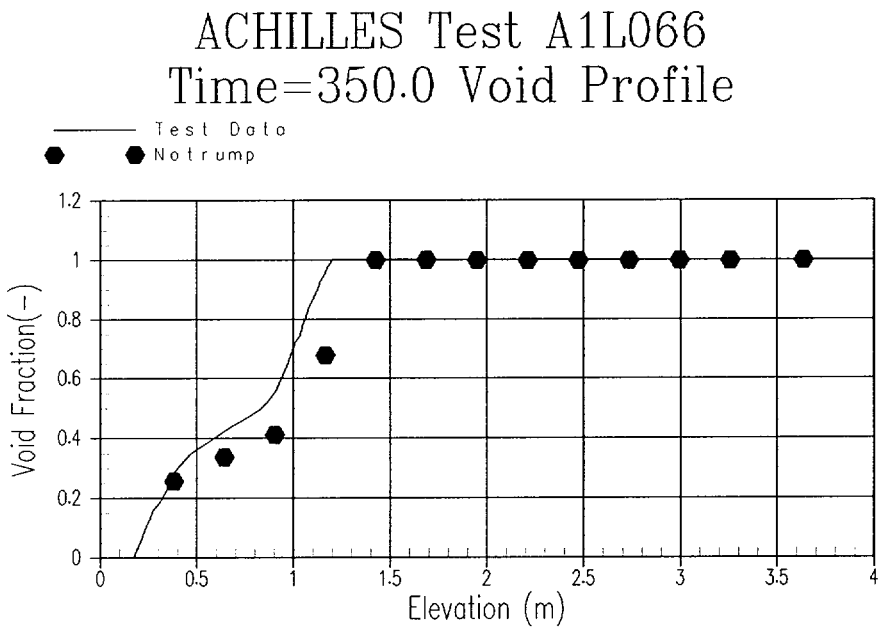

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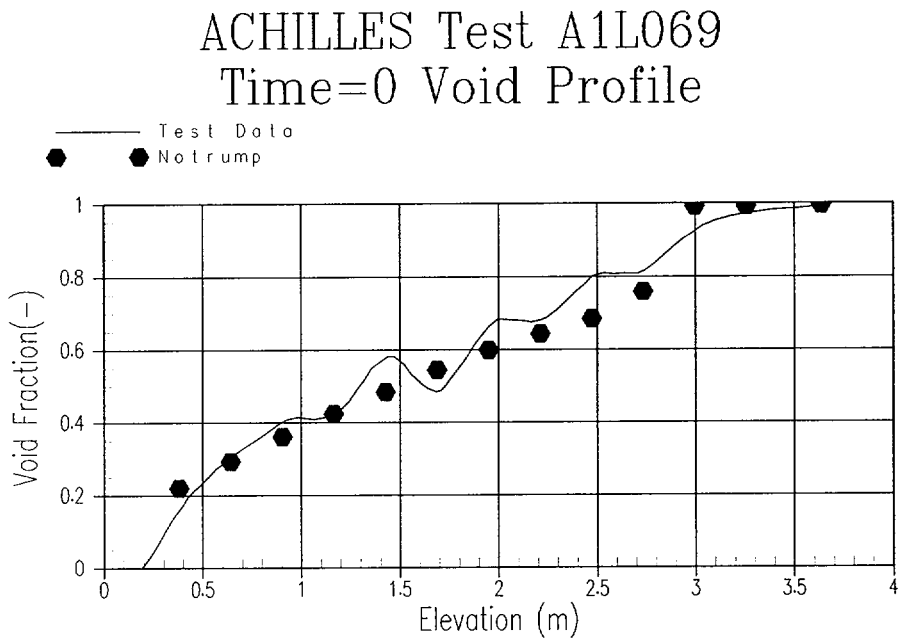

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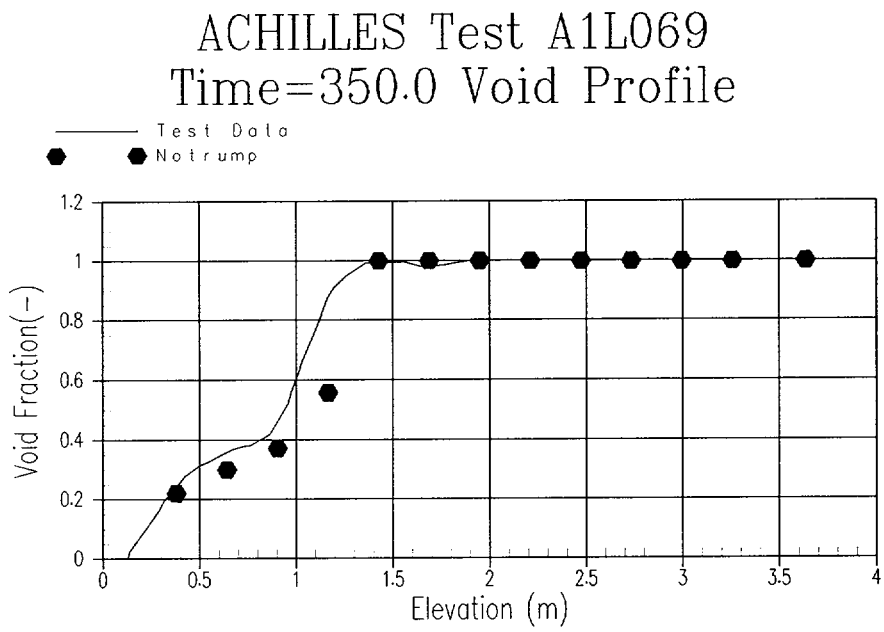

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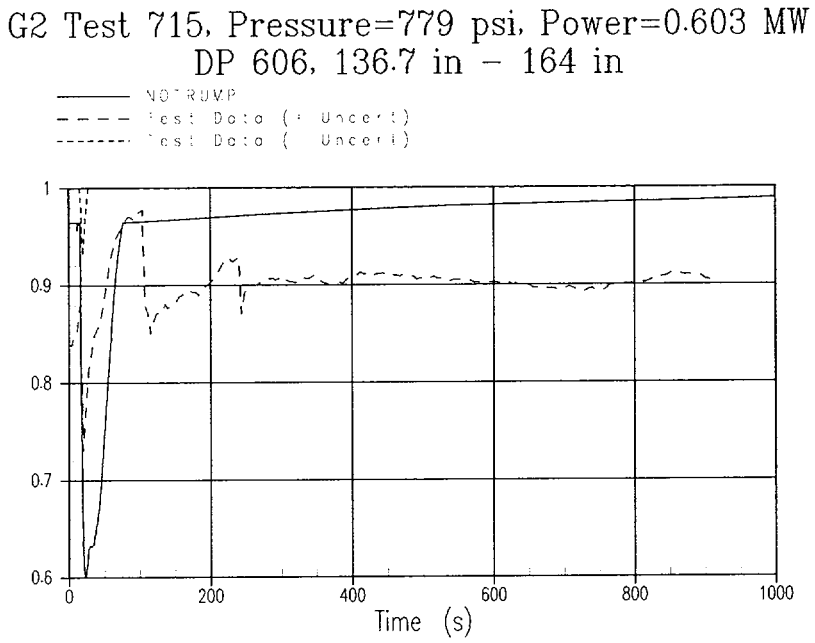

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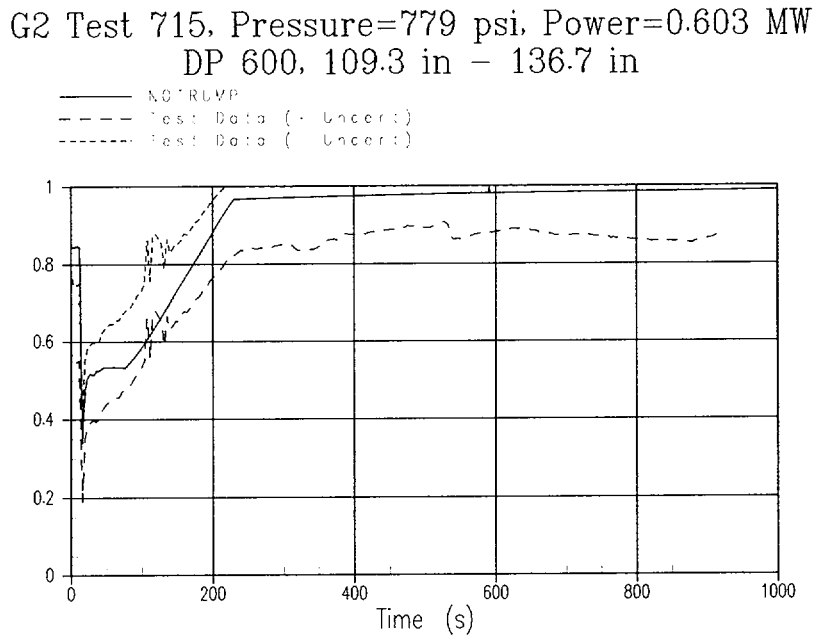

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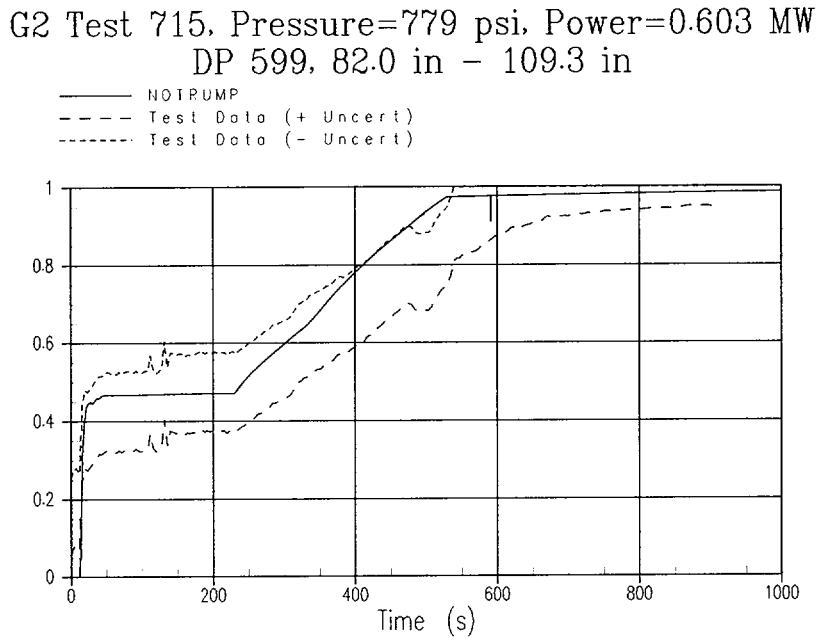

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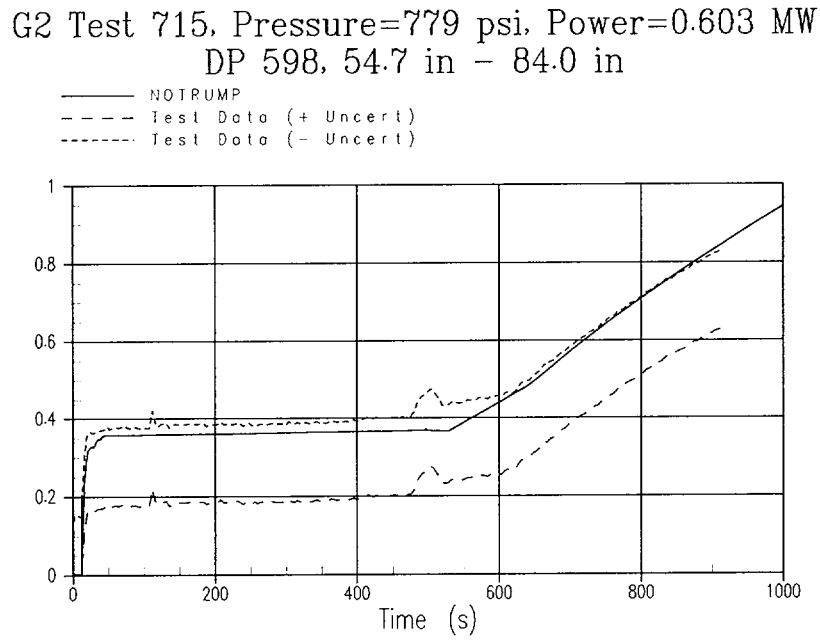

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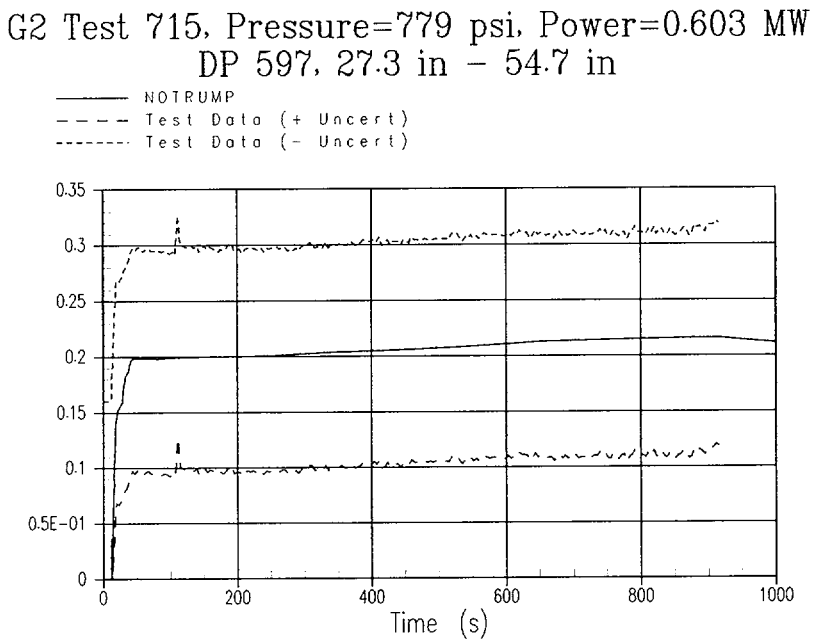

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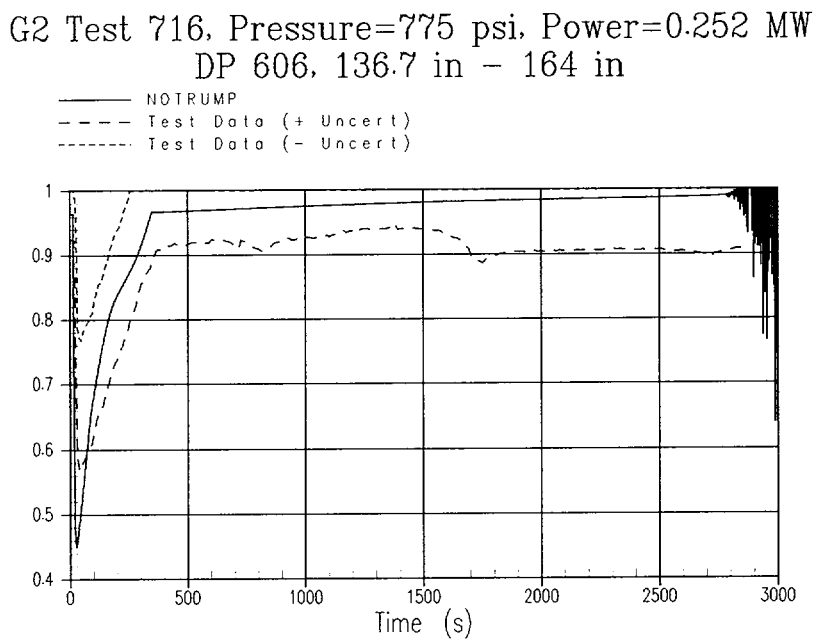

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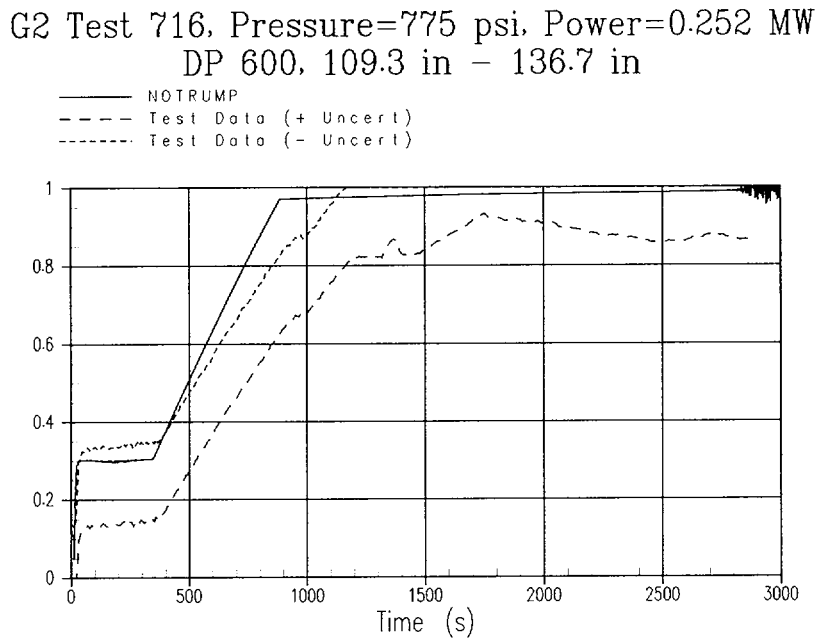

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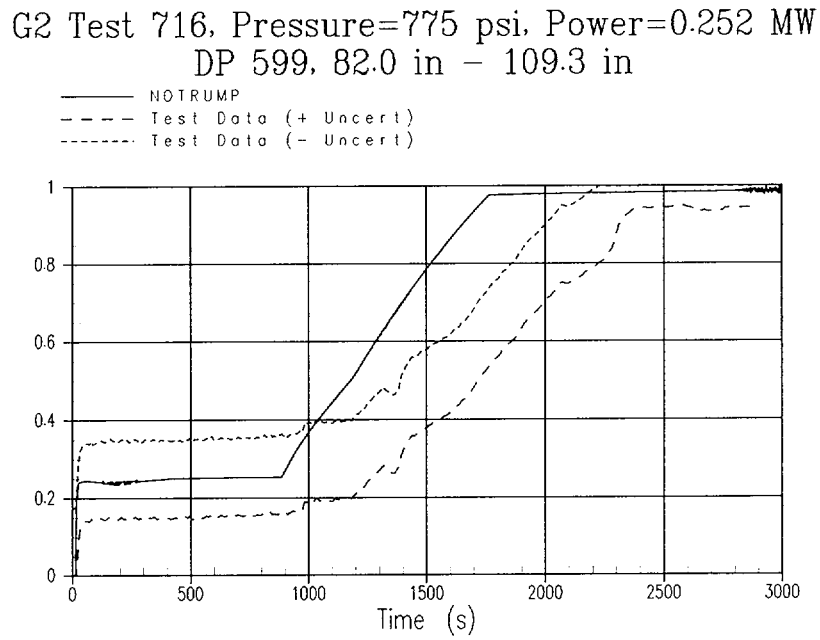

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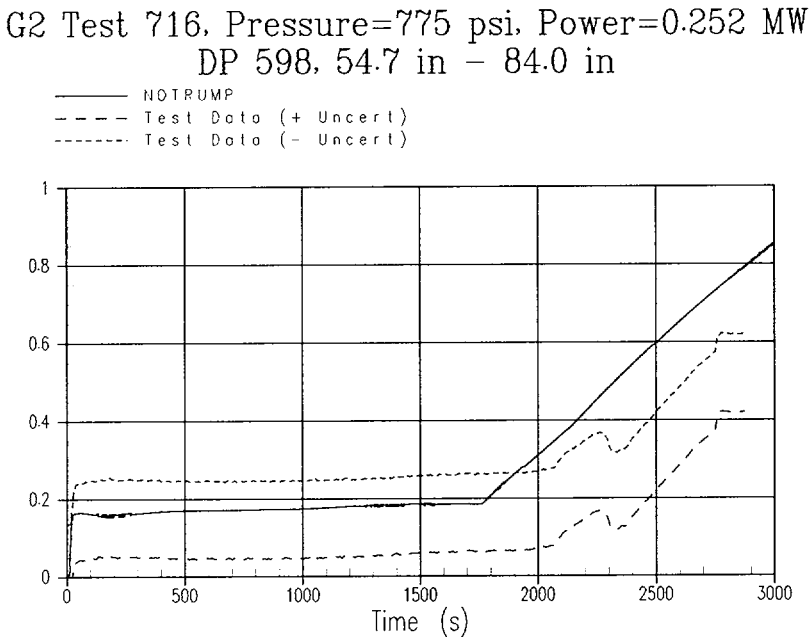

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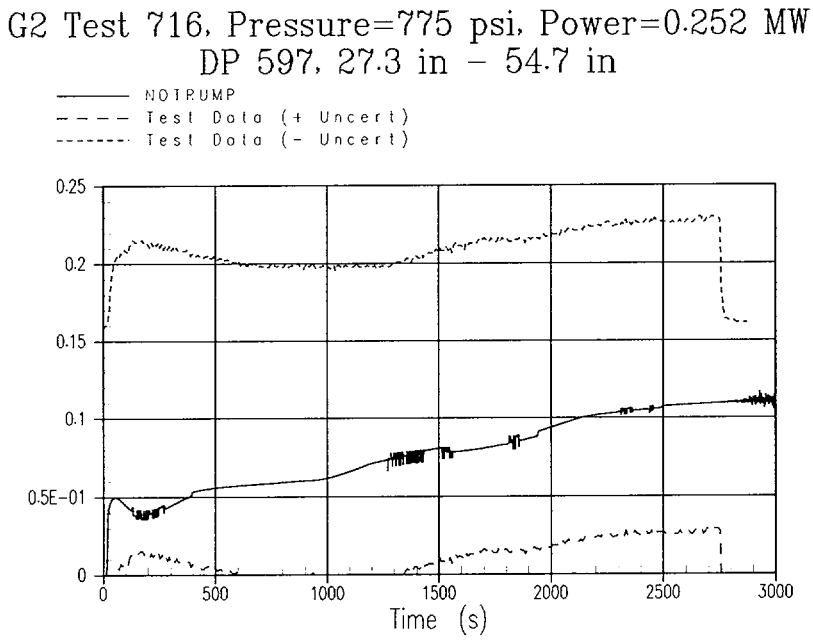

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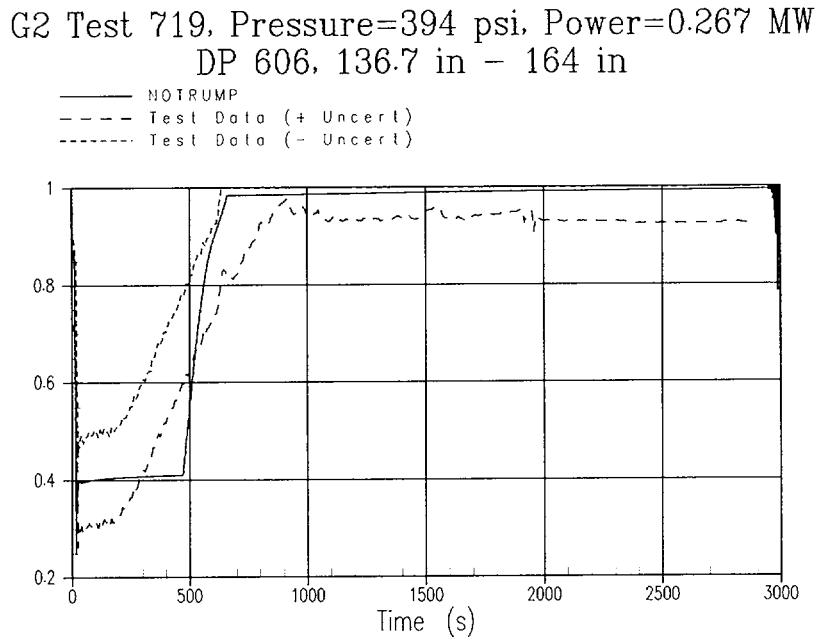

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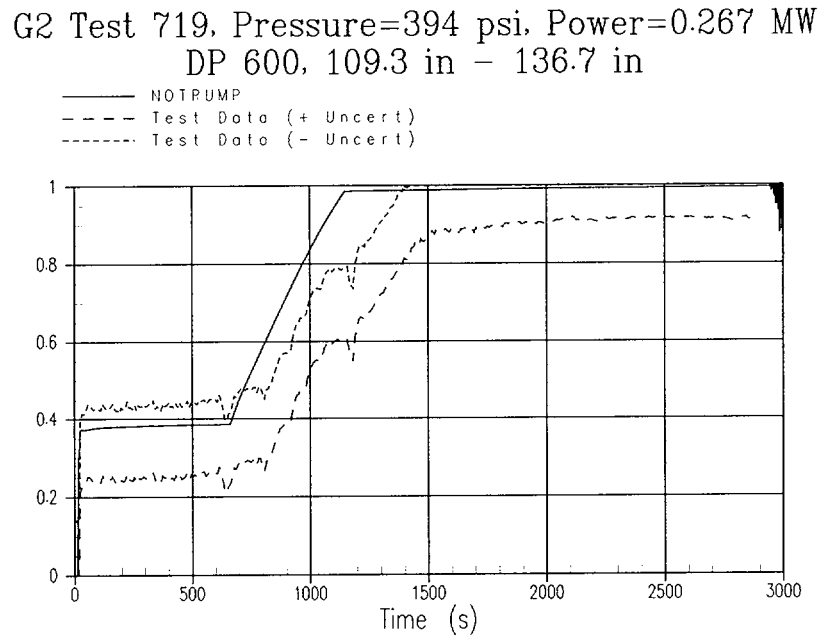

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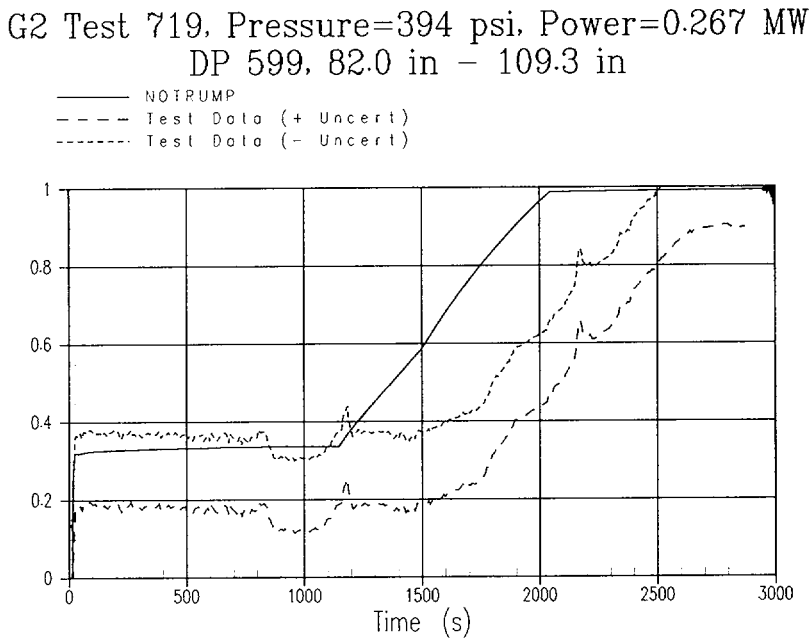

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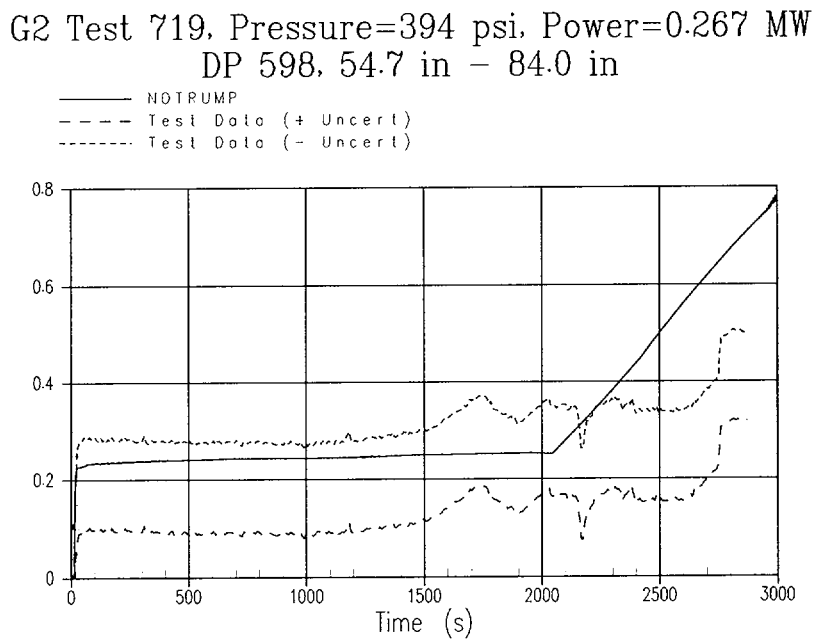

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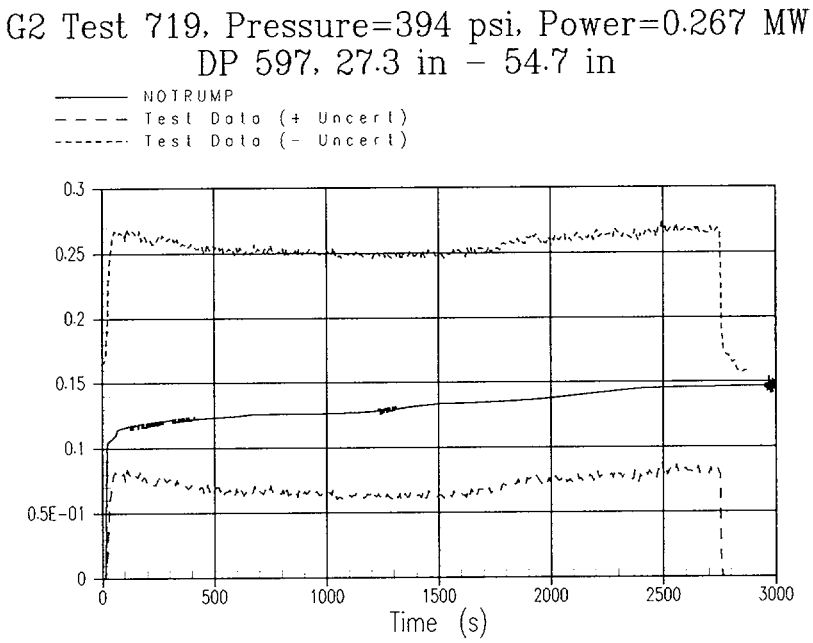

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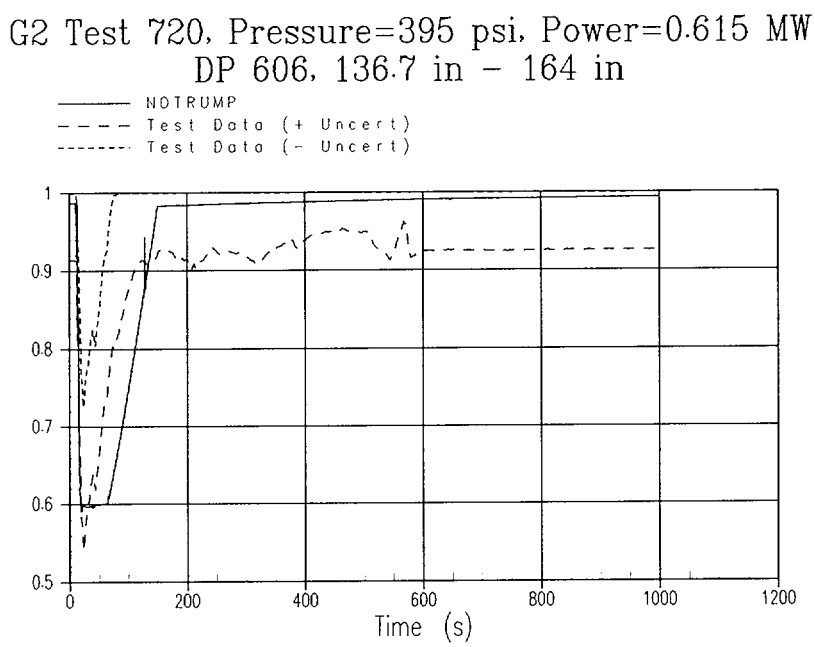

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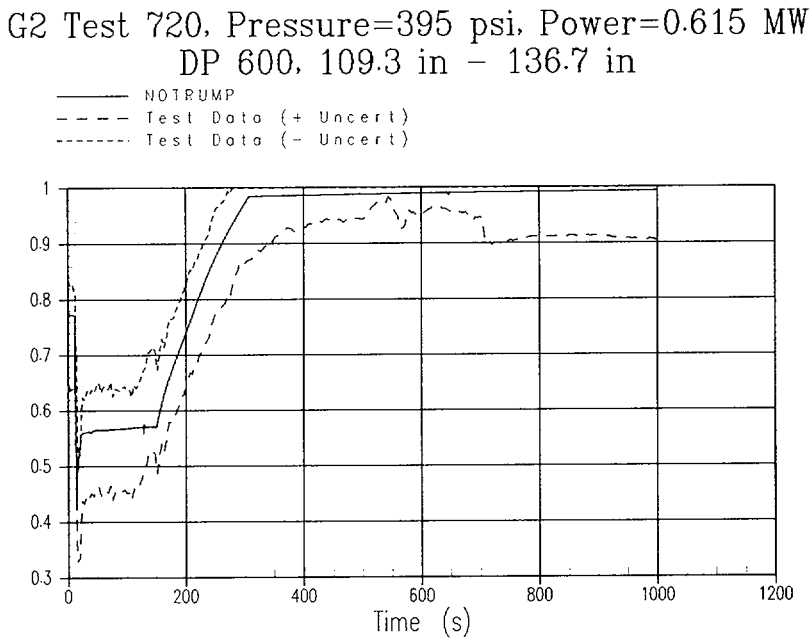

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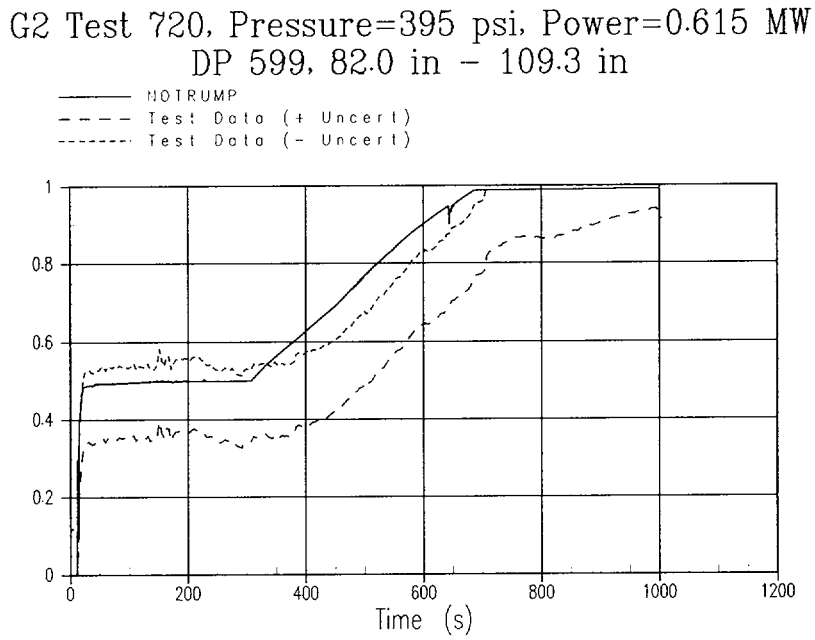

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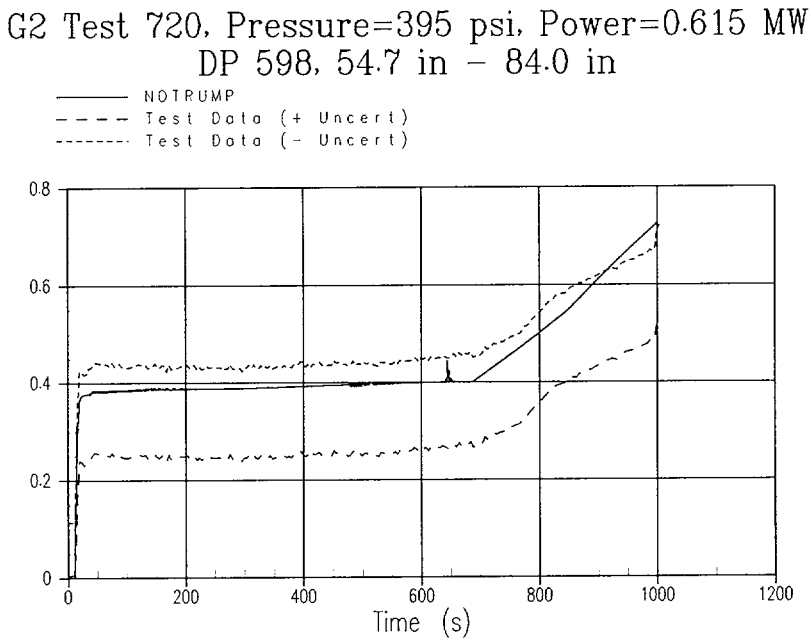

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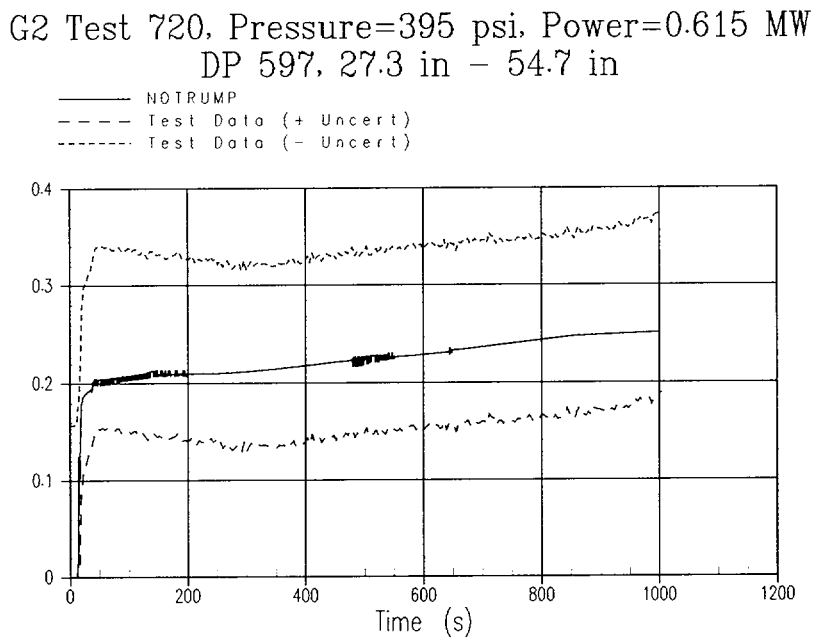

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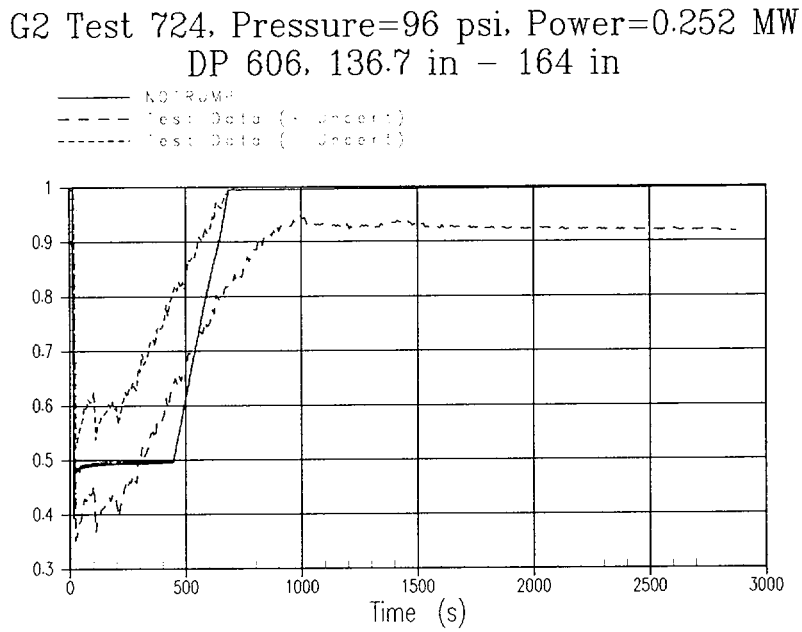

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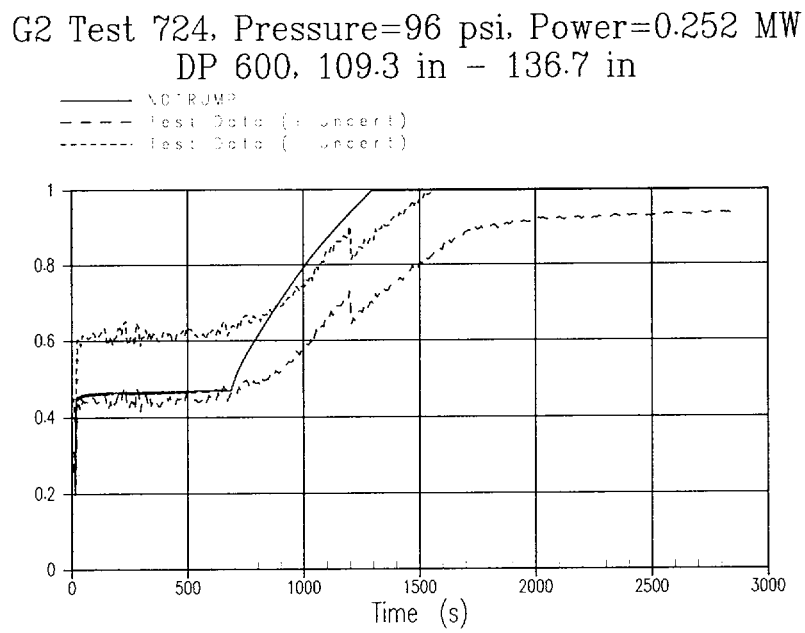

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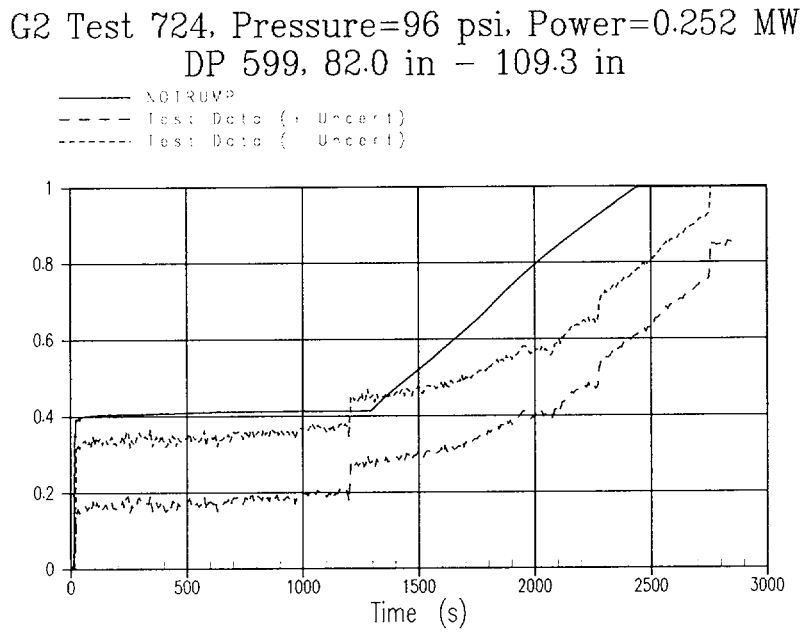

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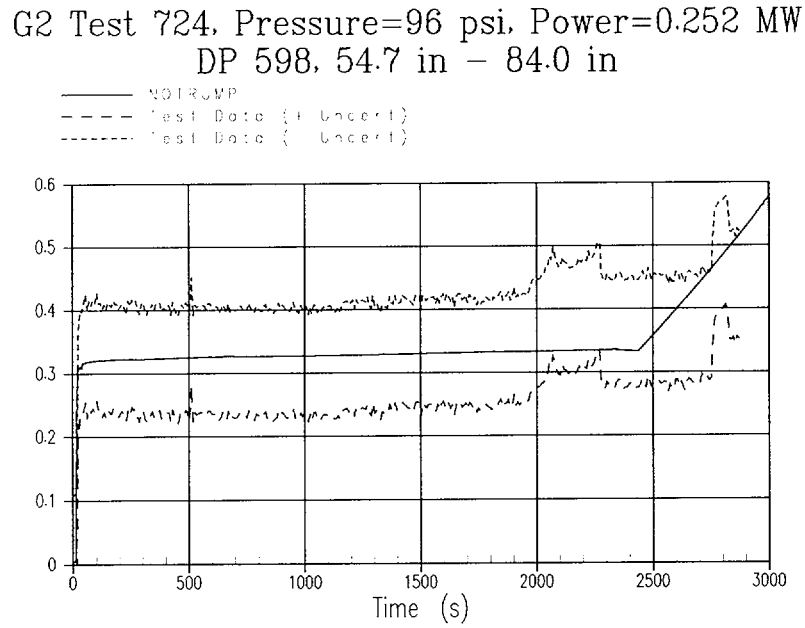

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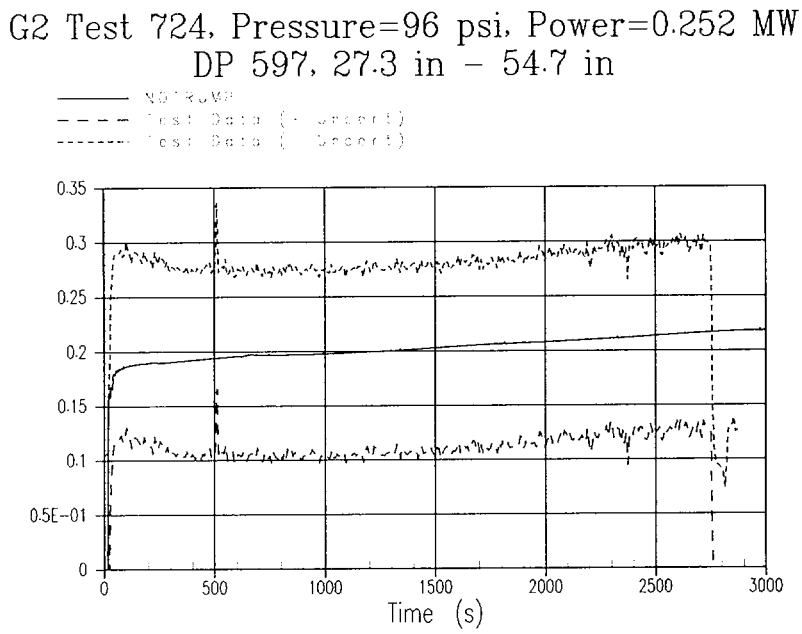

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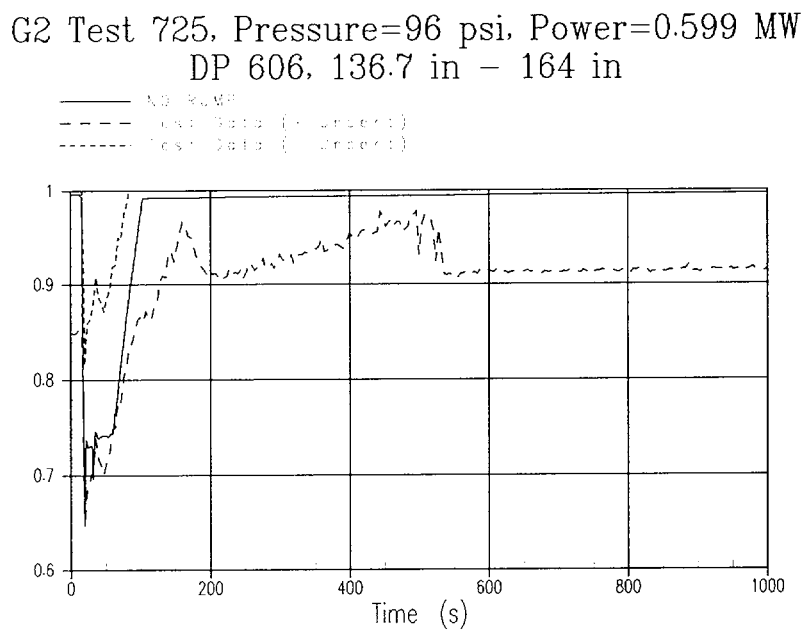

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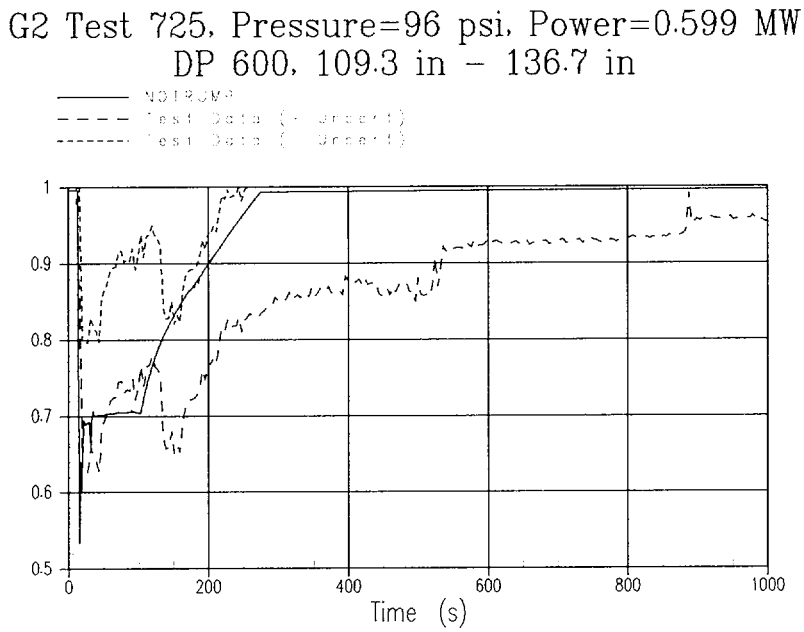

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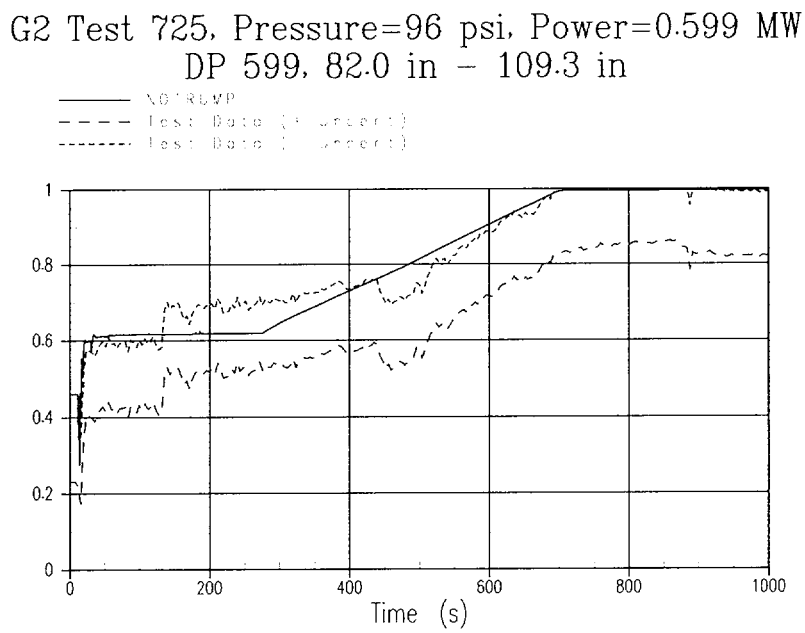

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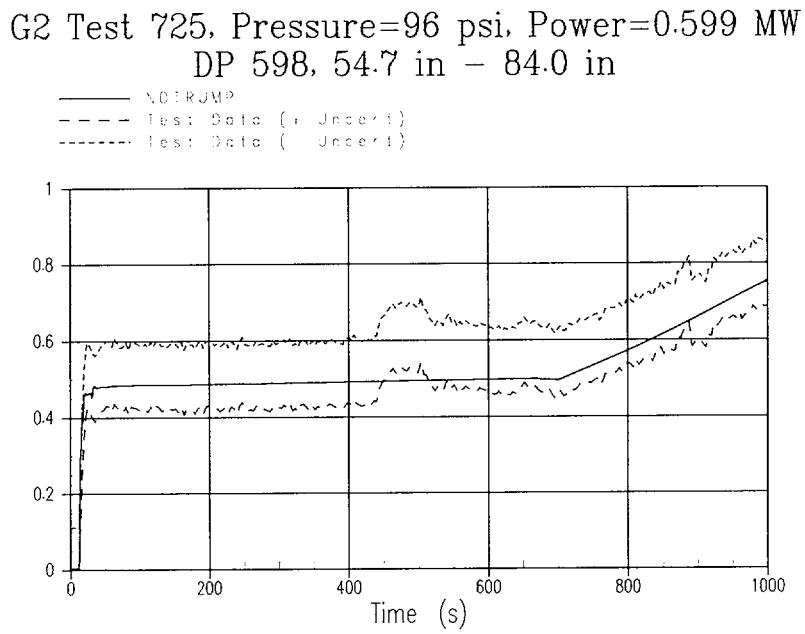

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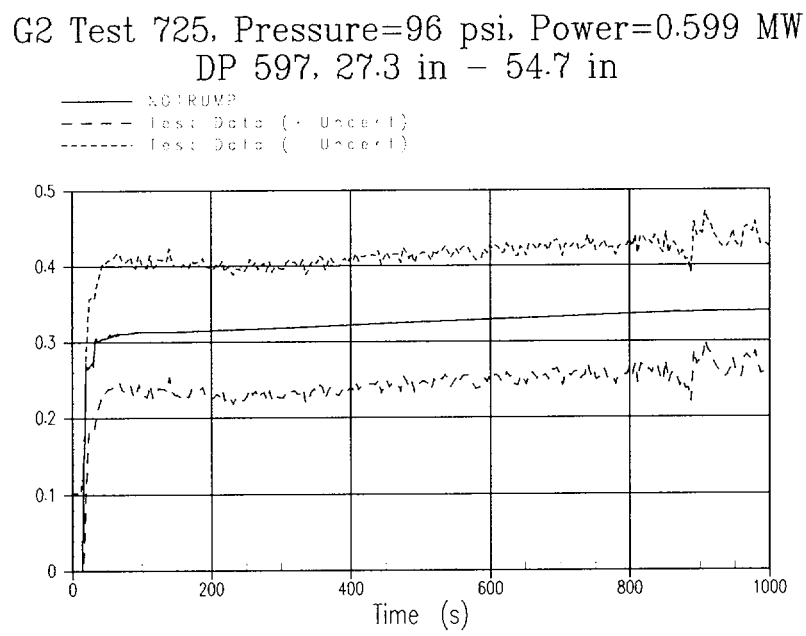

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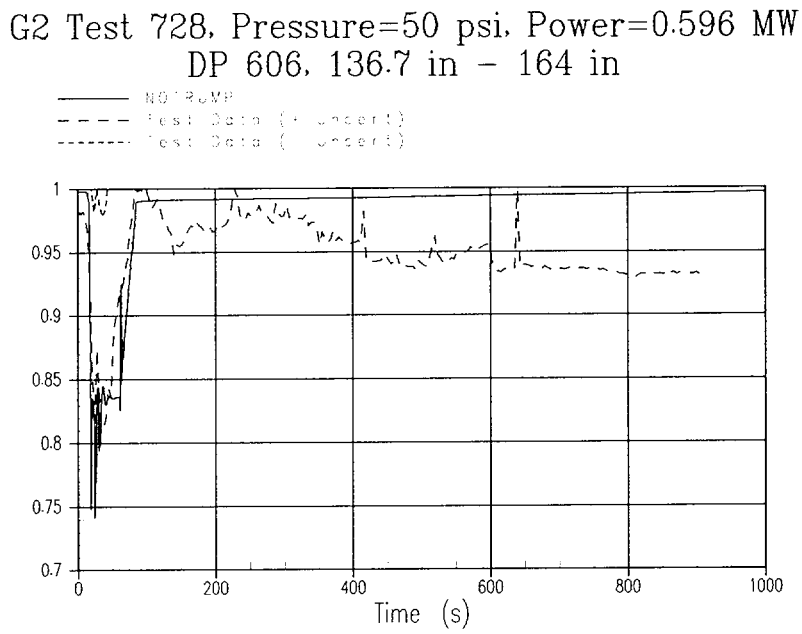

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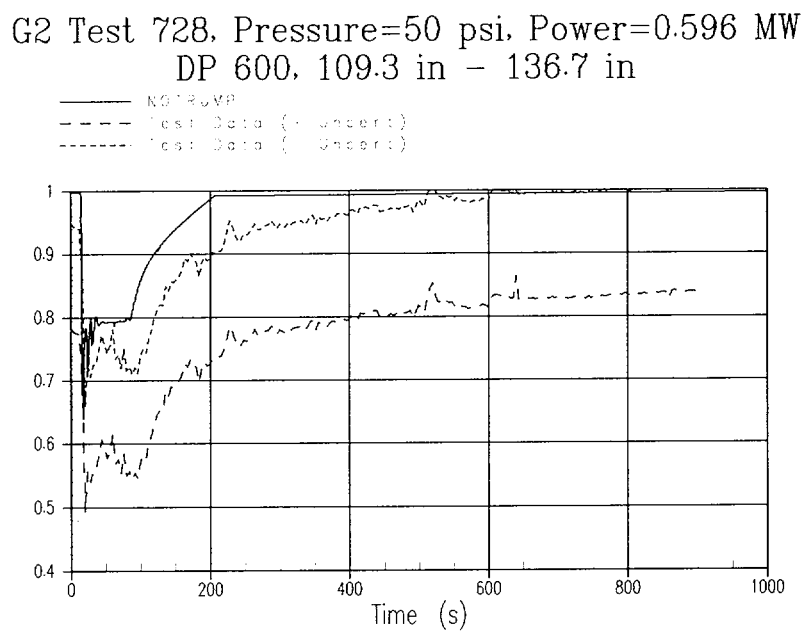

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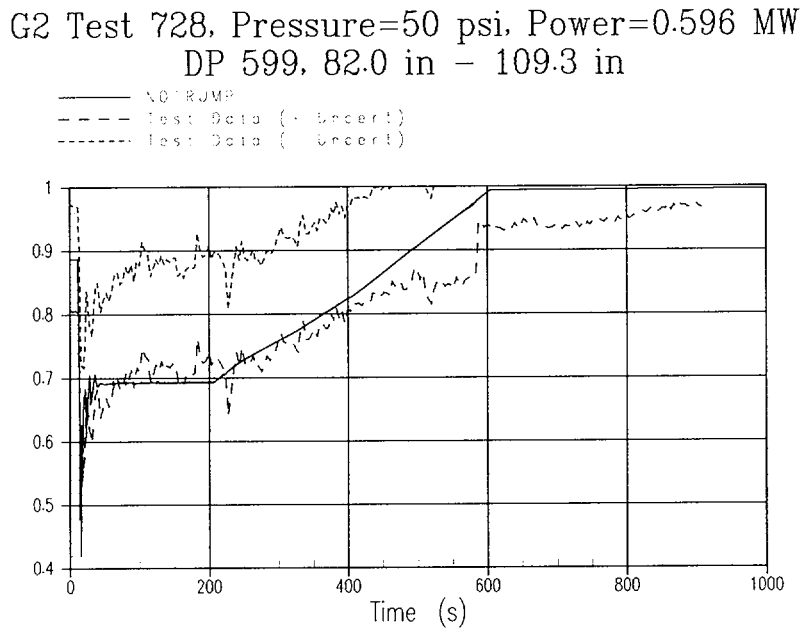

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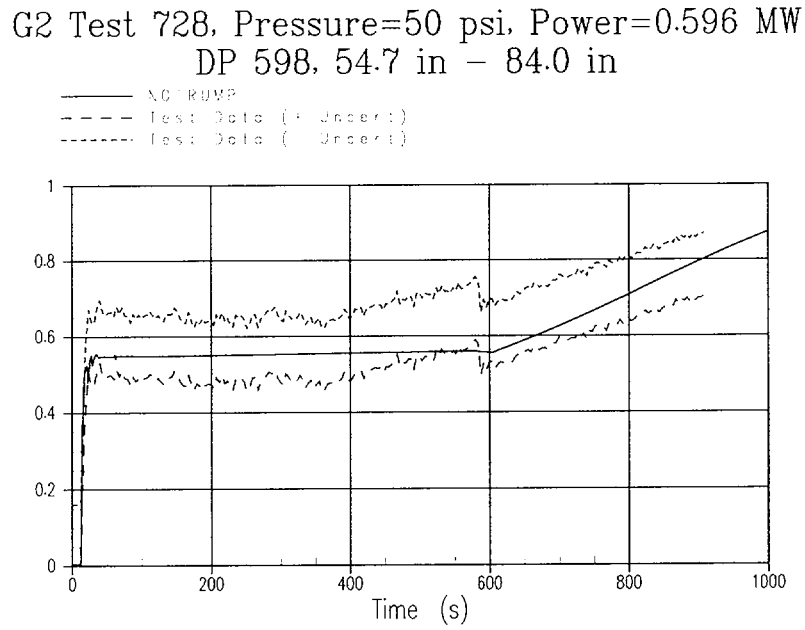

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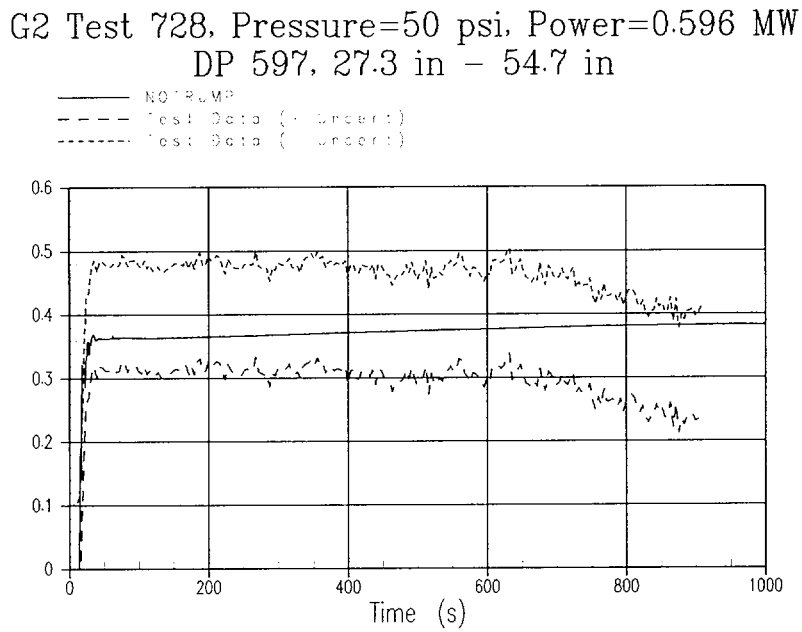

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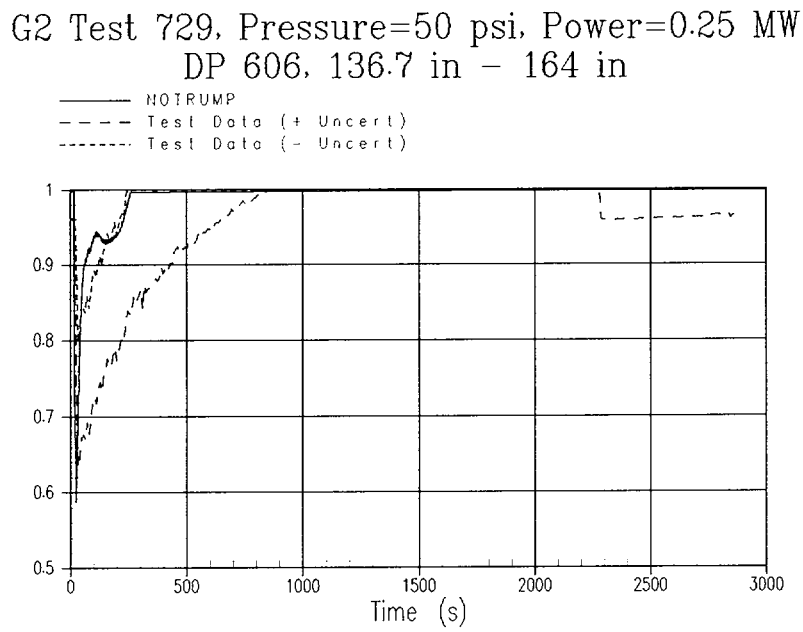

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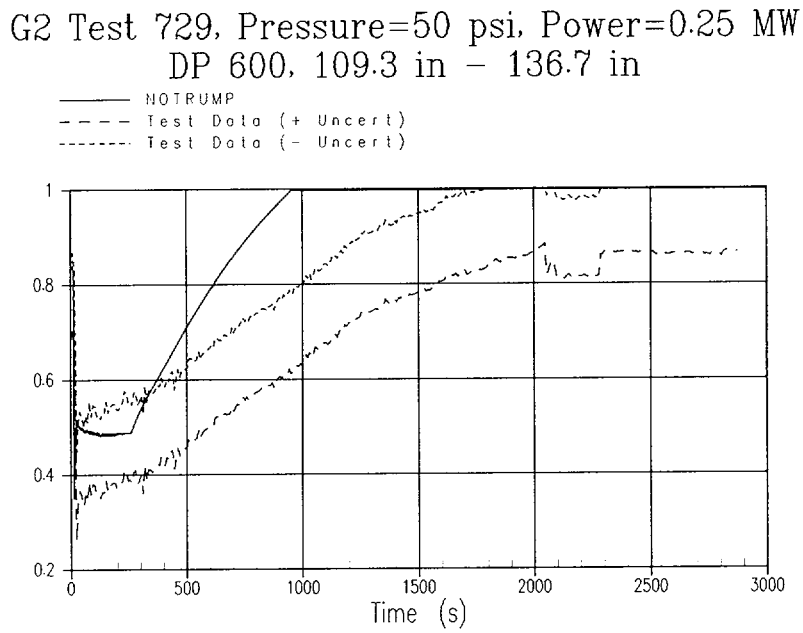

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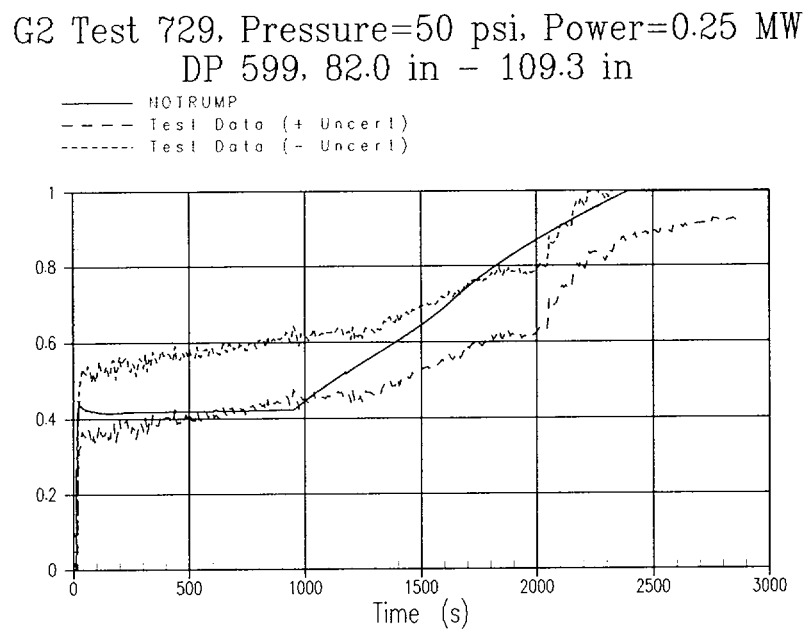

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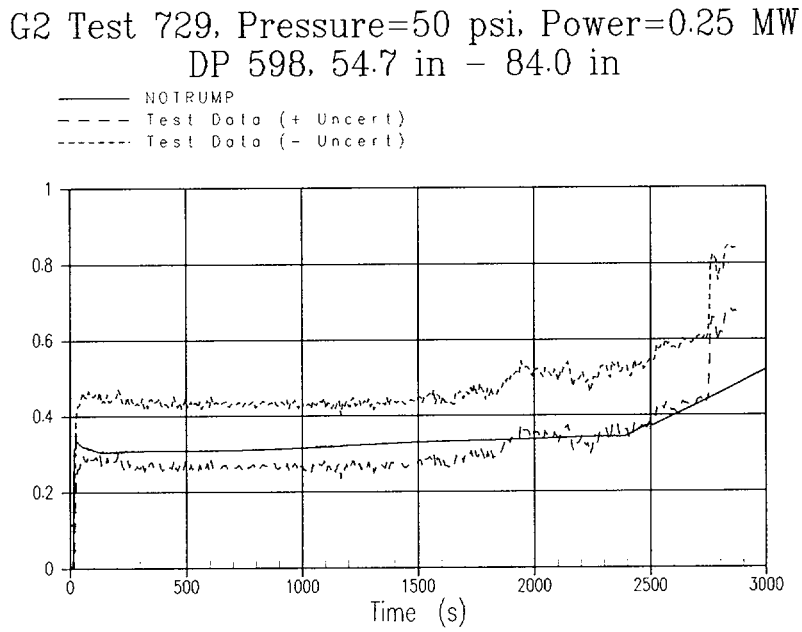

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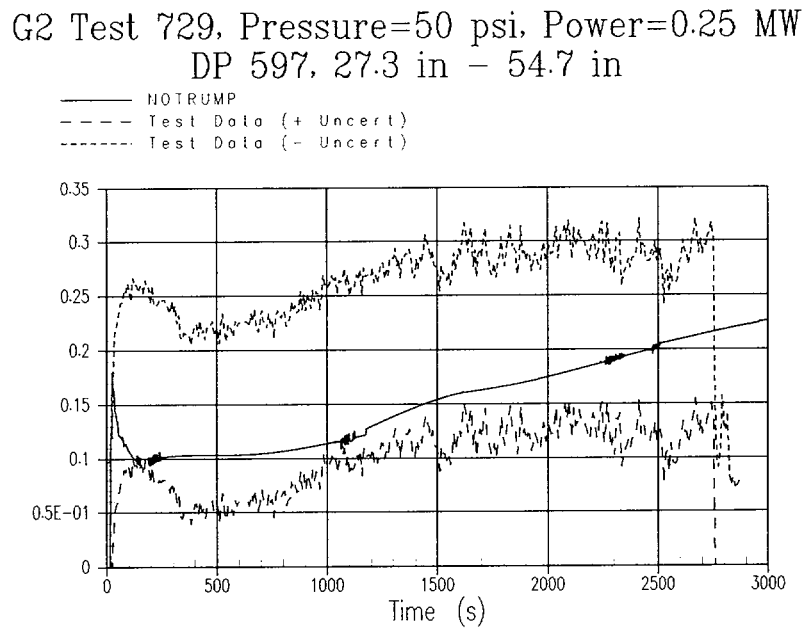

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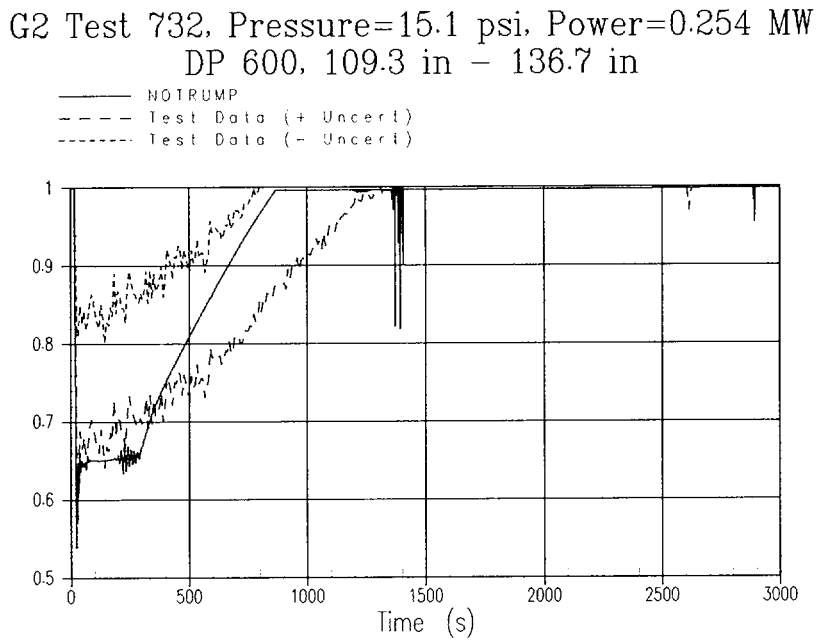

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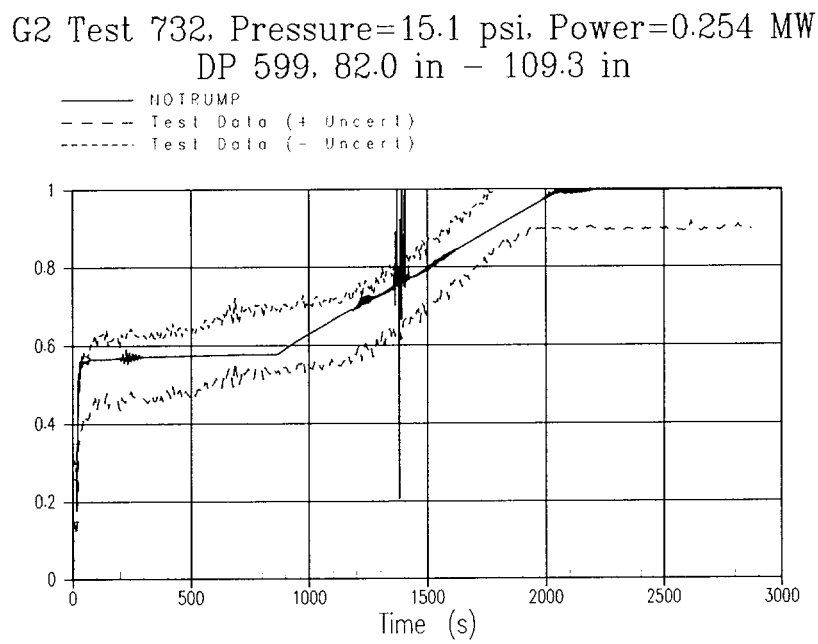


Figure P69-46

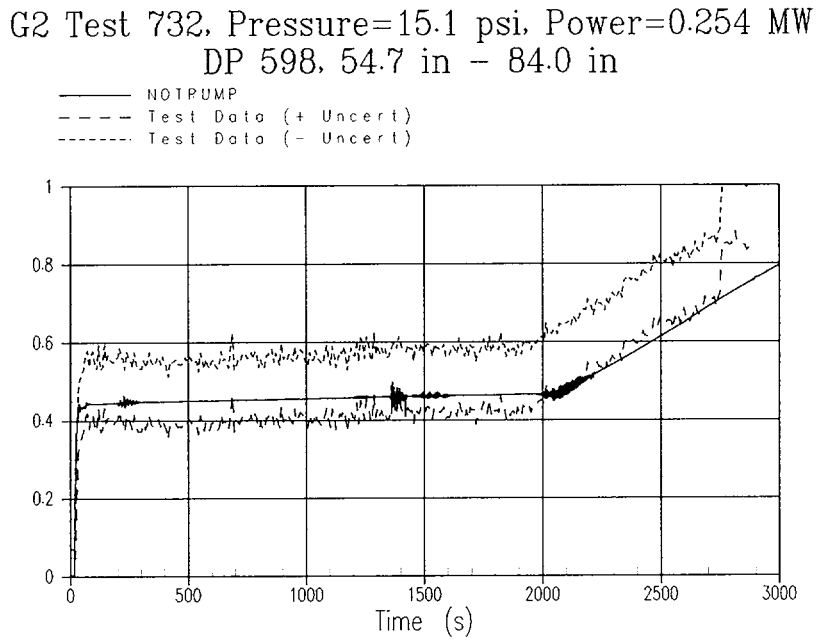
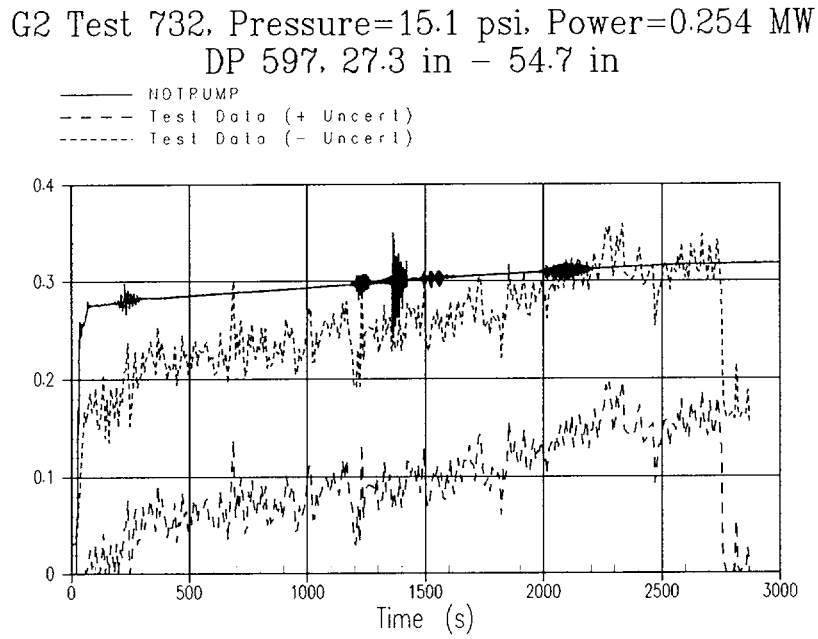


Figure P69-47



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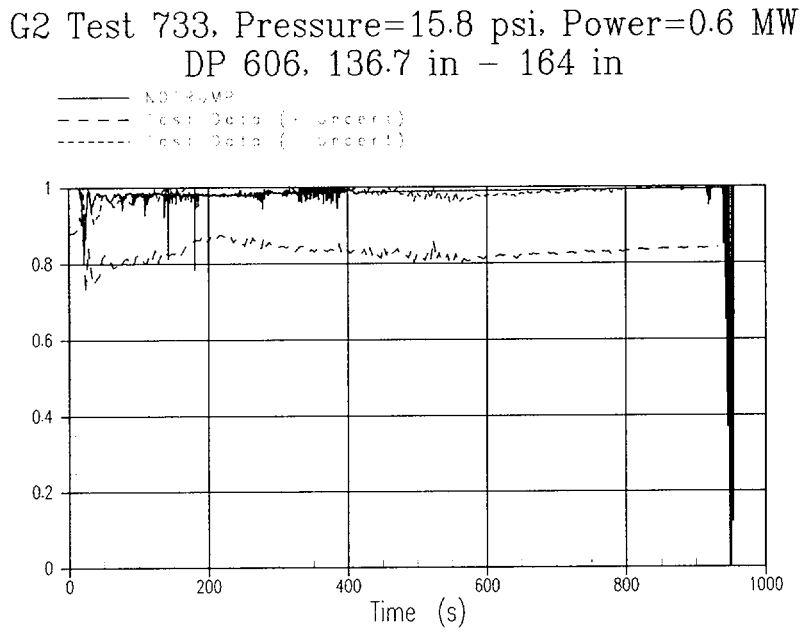

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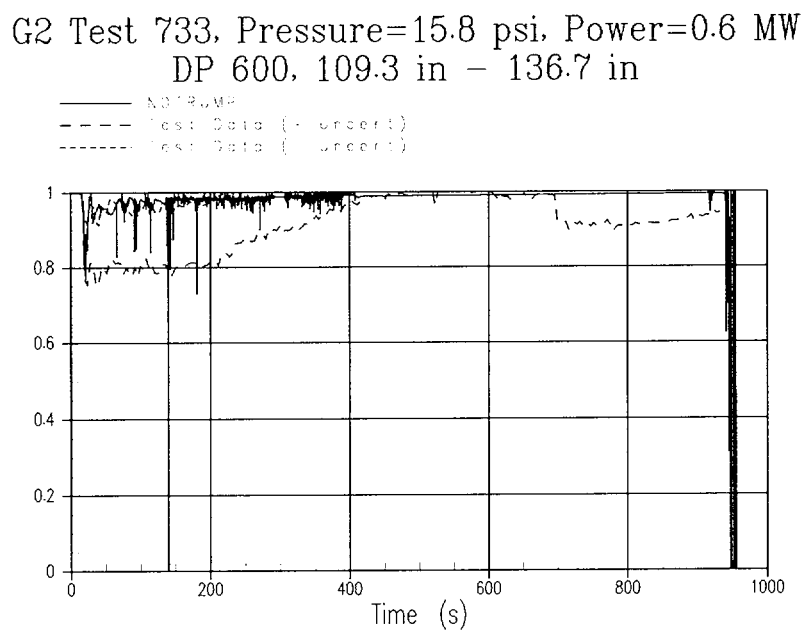

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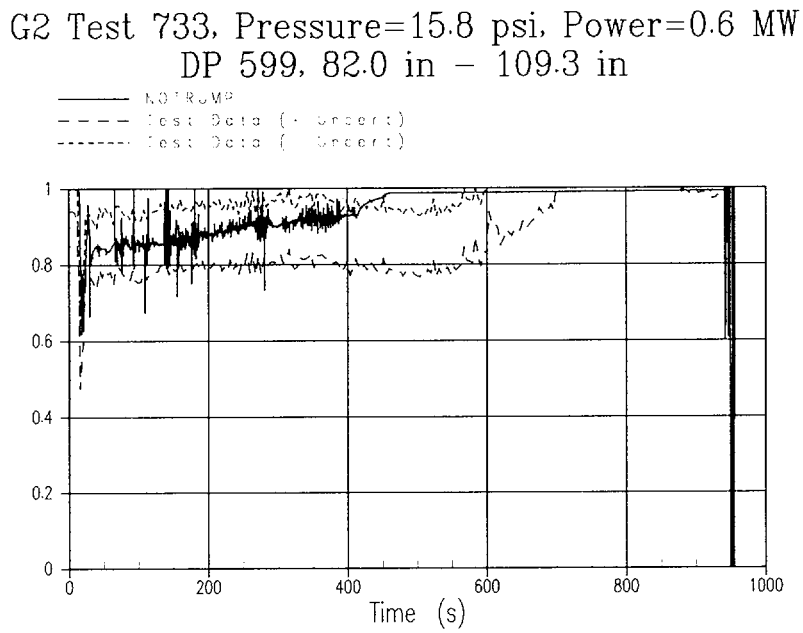

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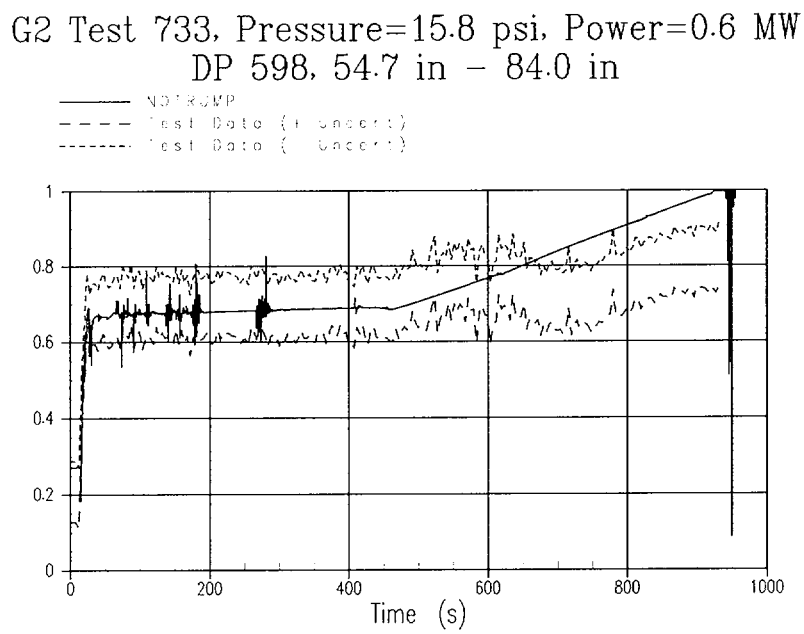

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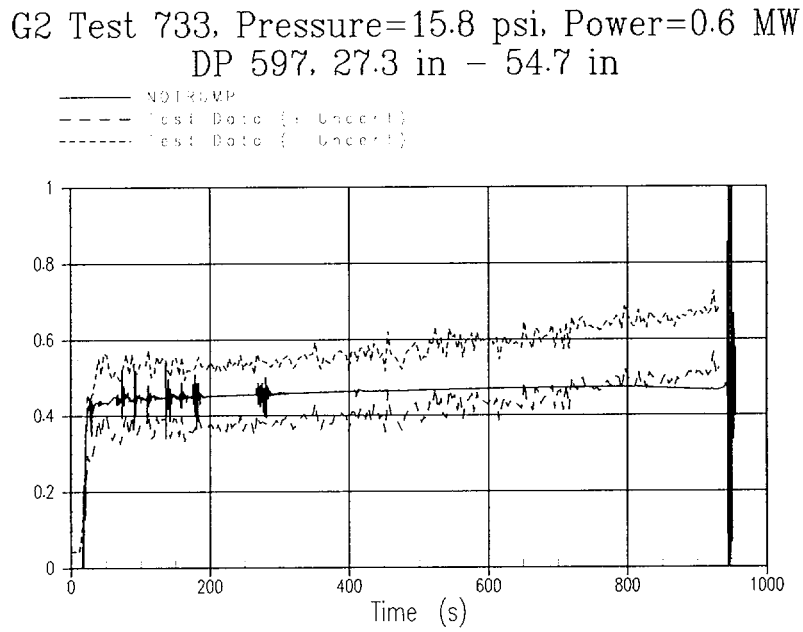

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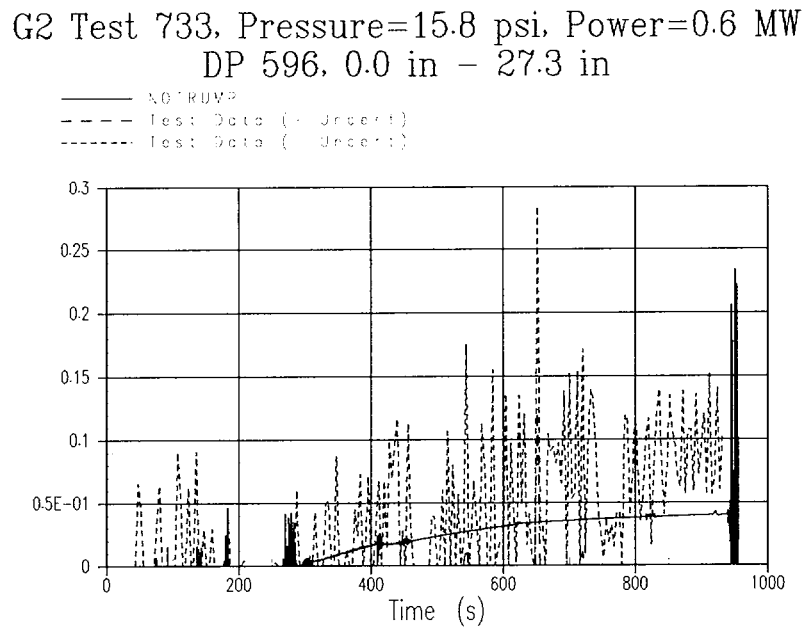

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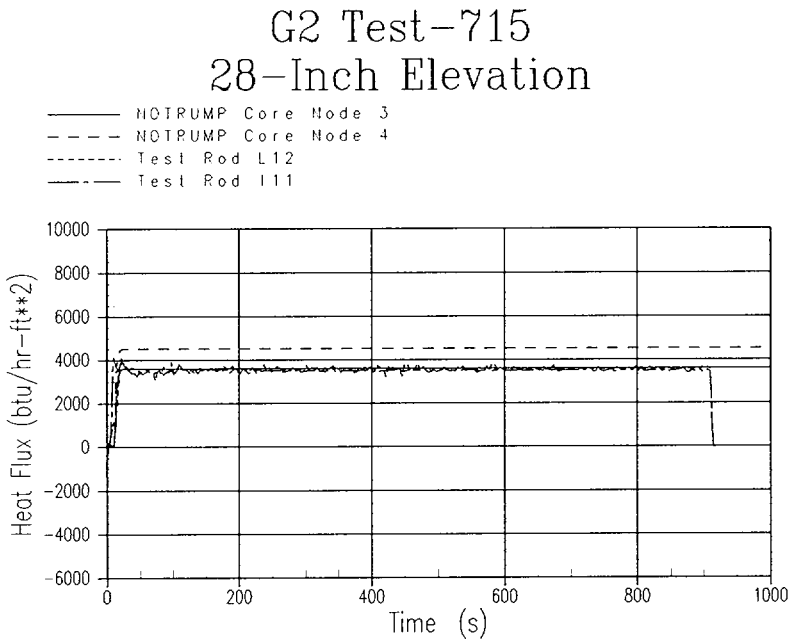

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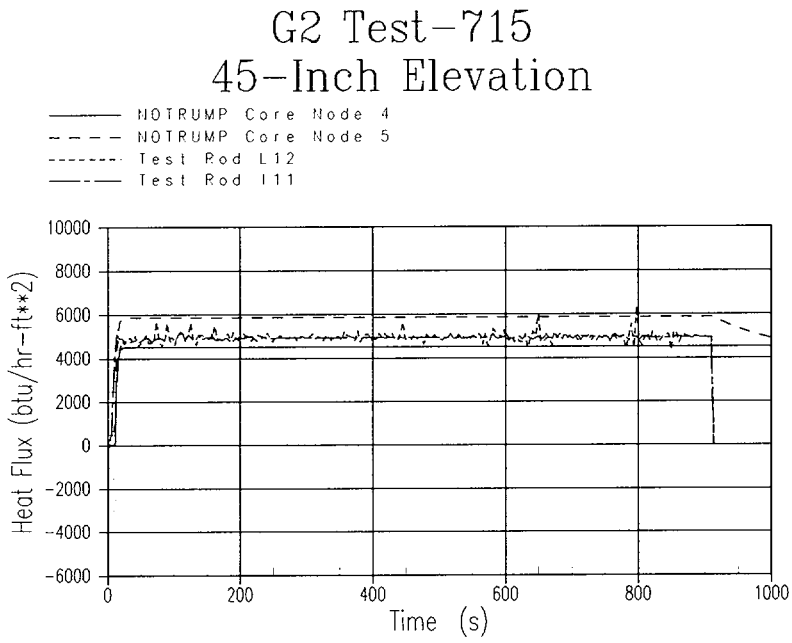

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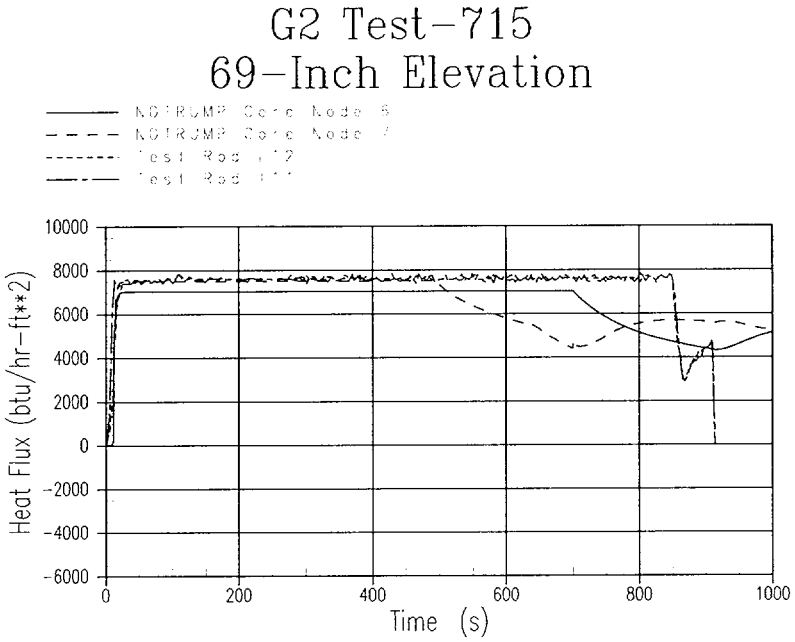

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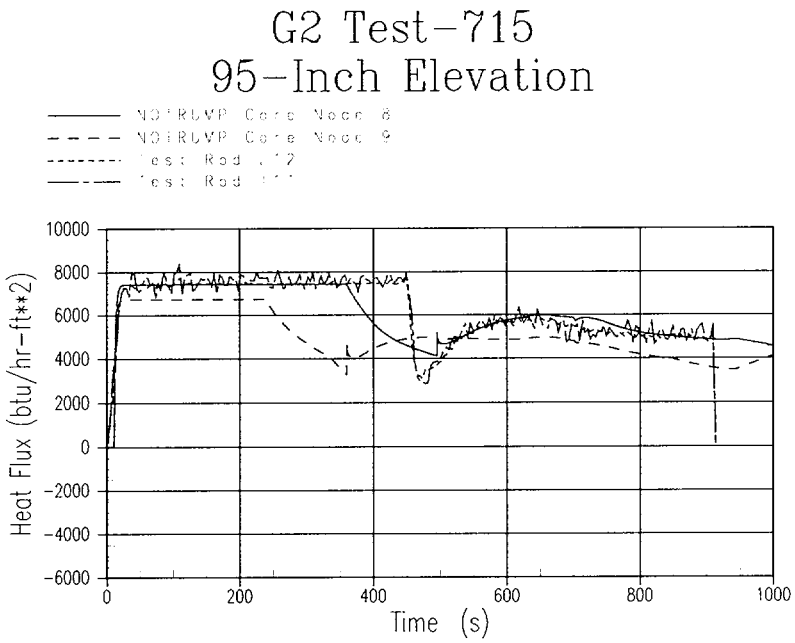

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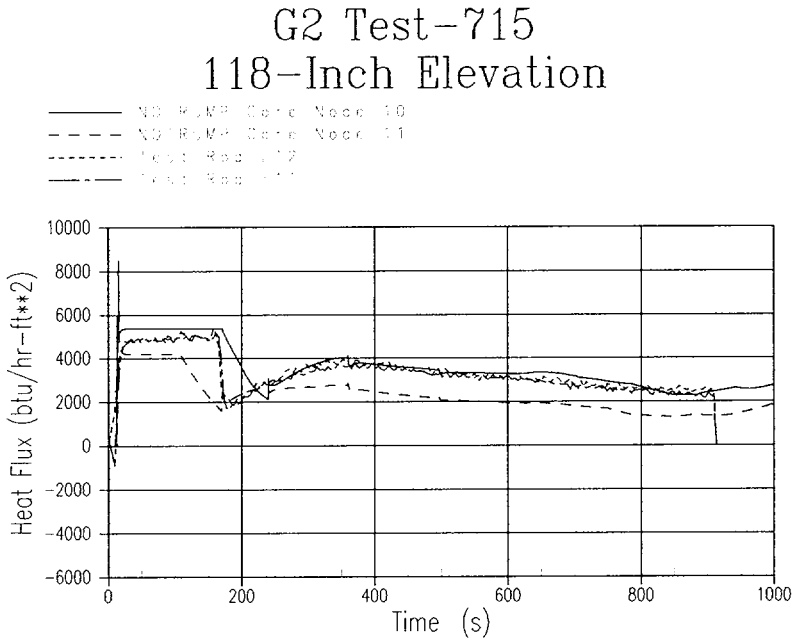

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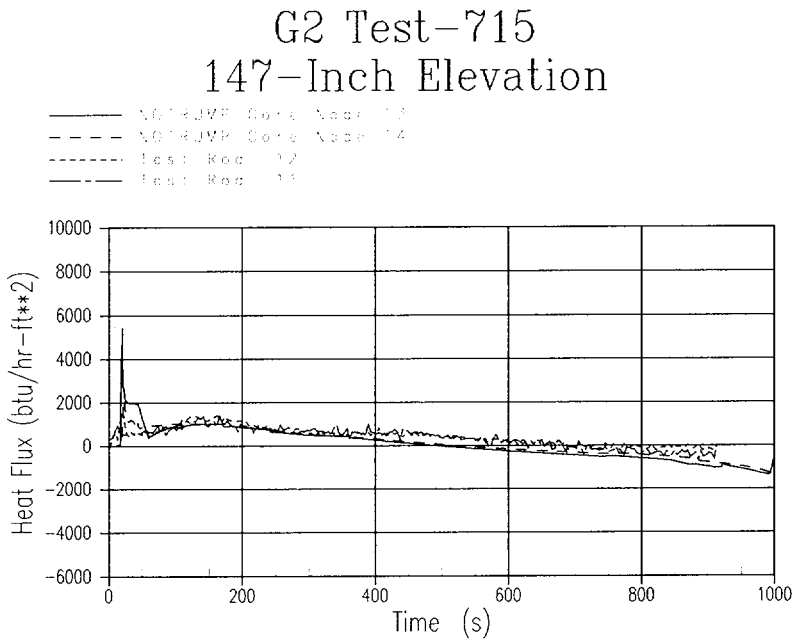

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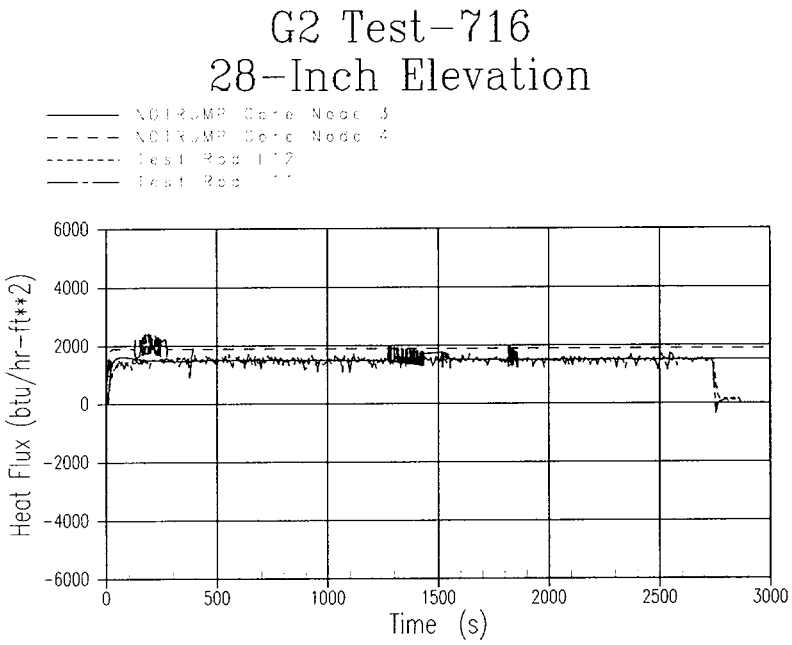

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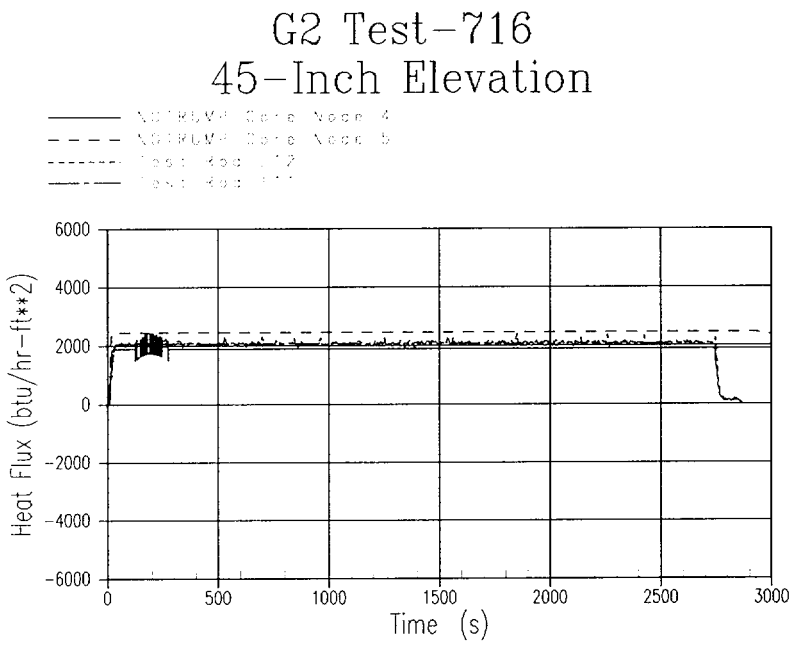

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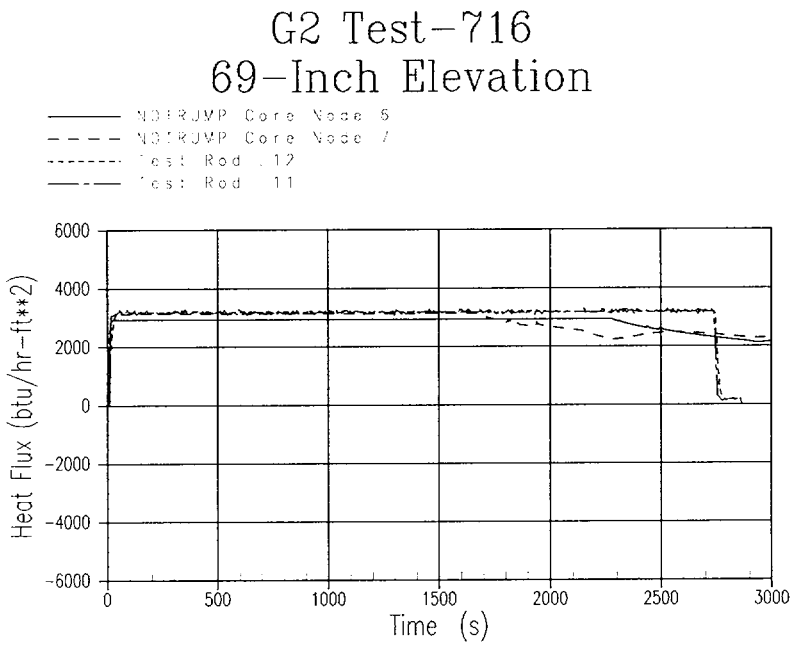

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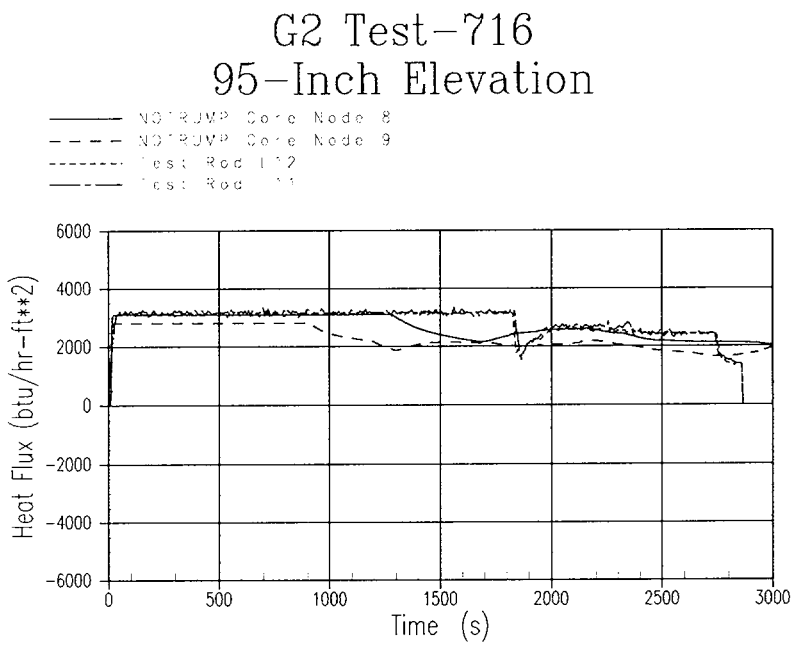

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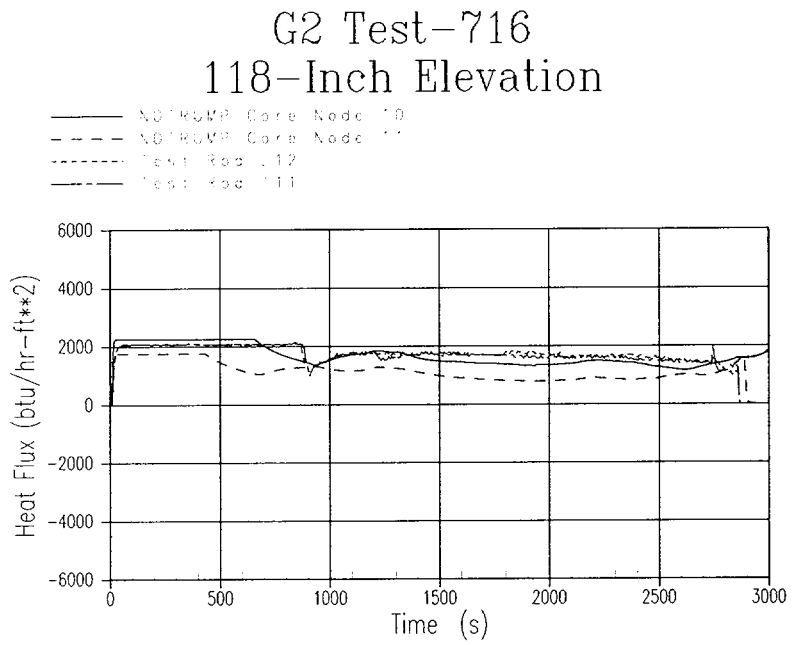

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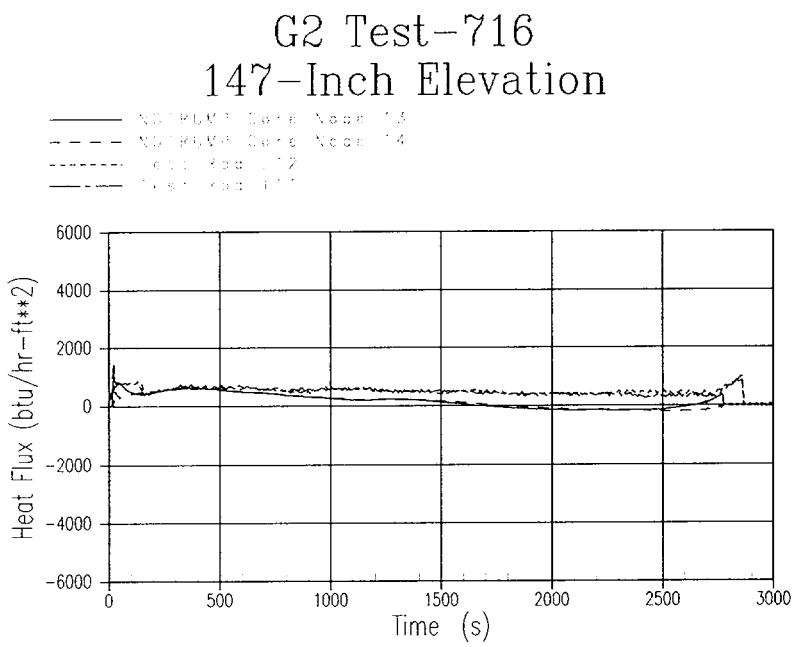

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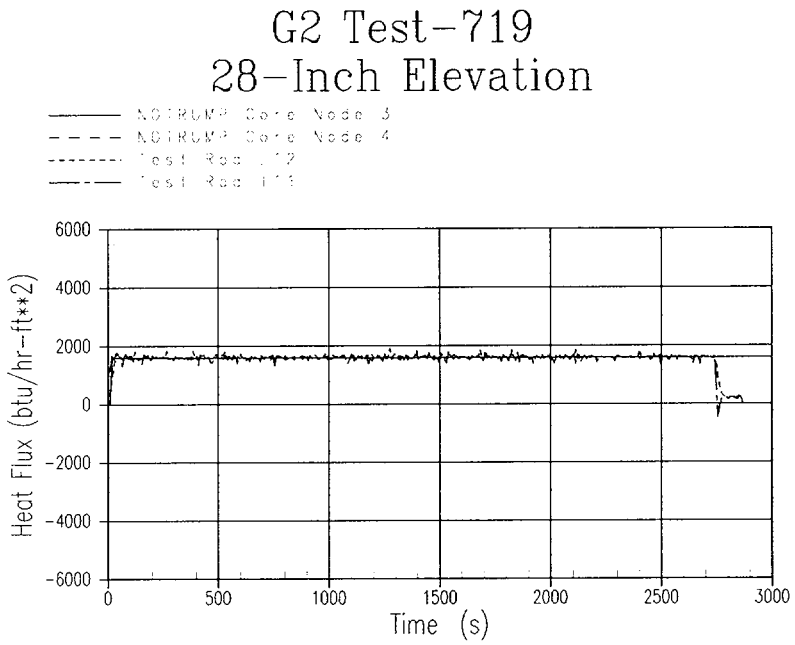

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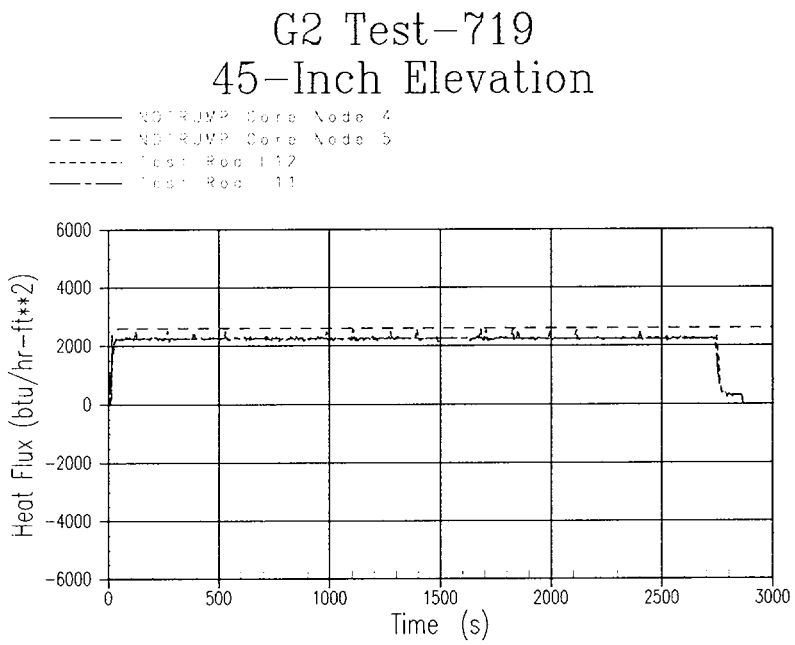

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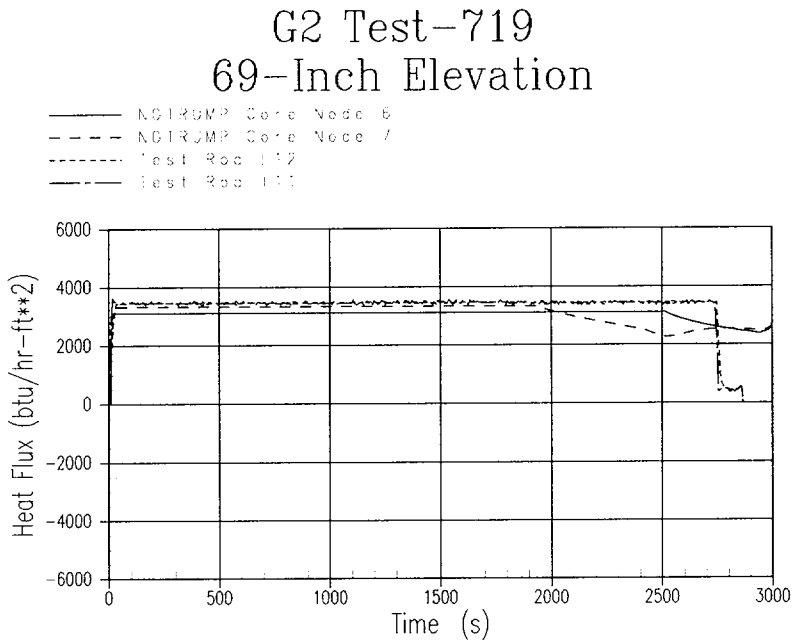
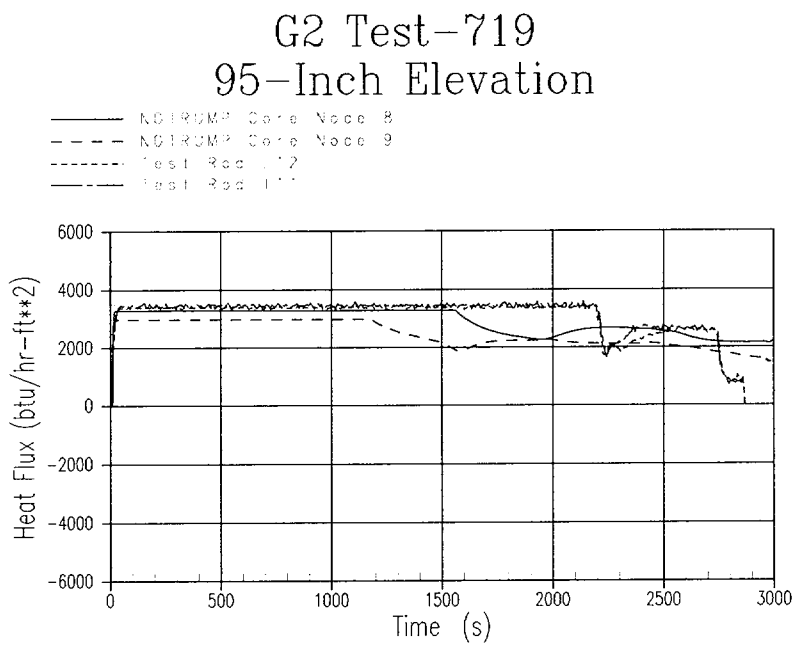


Figure P69-69



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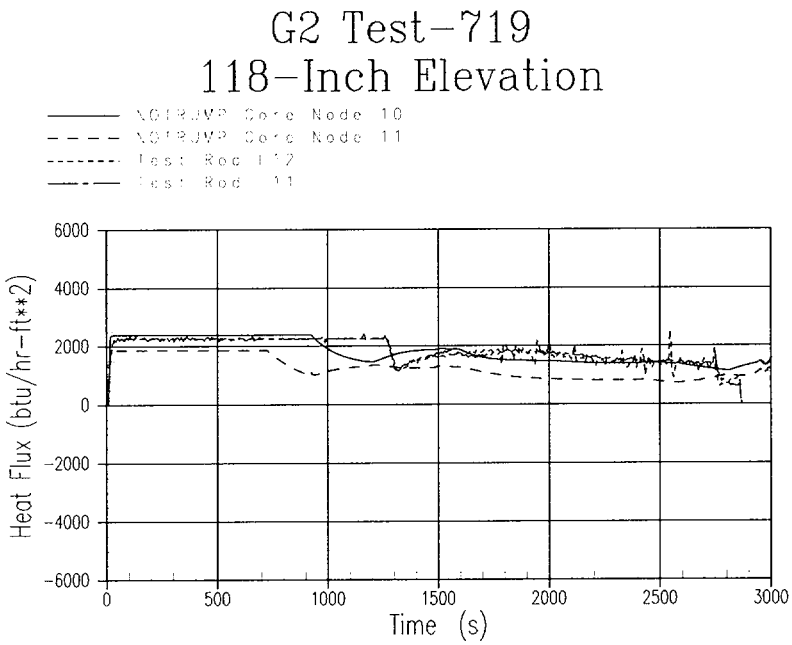

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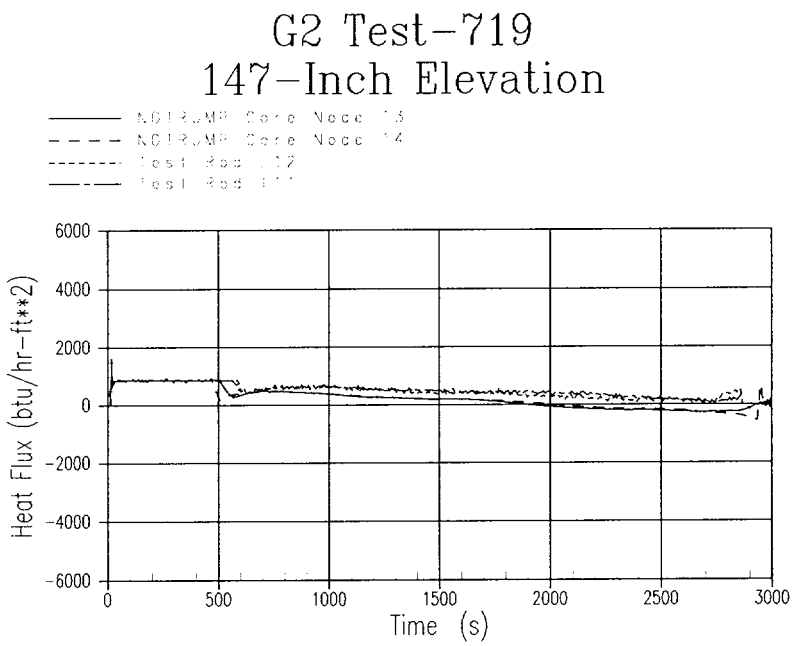

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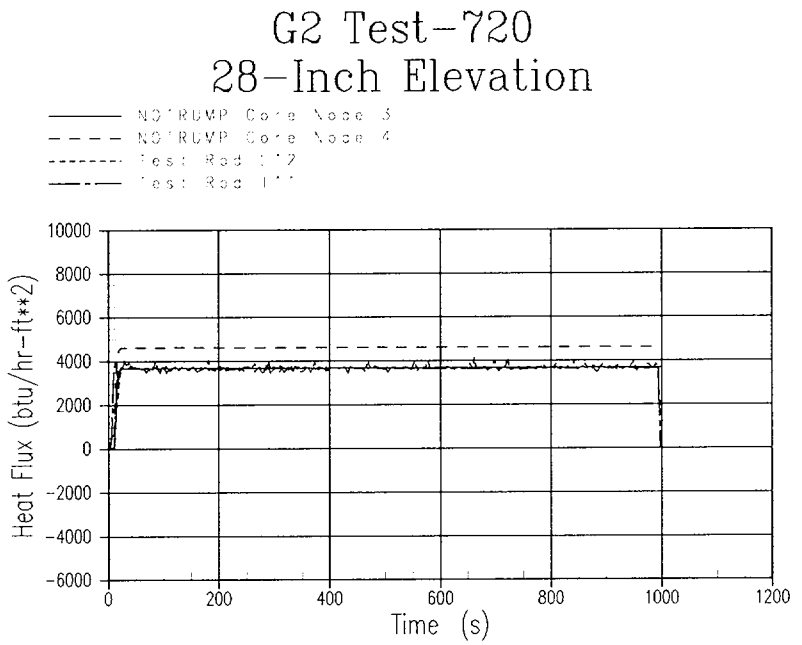

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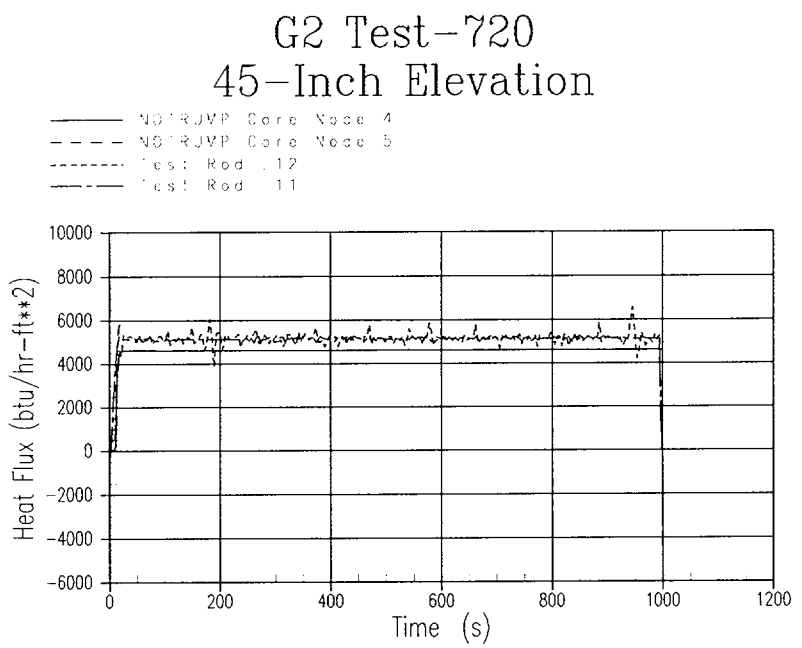

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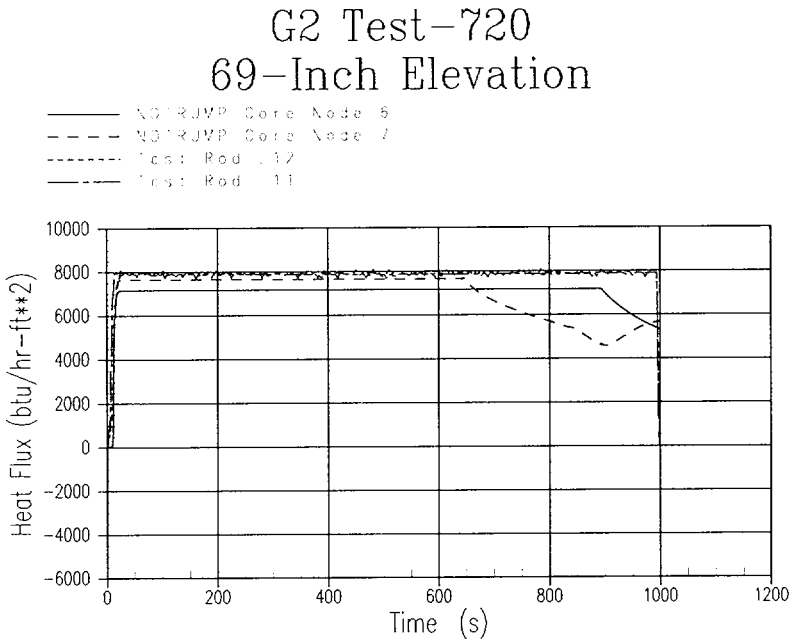

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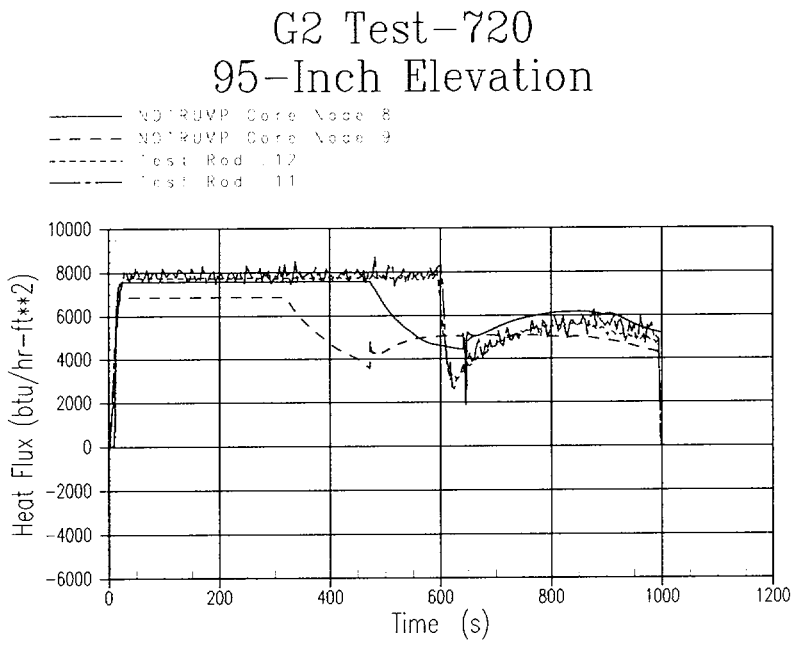

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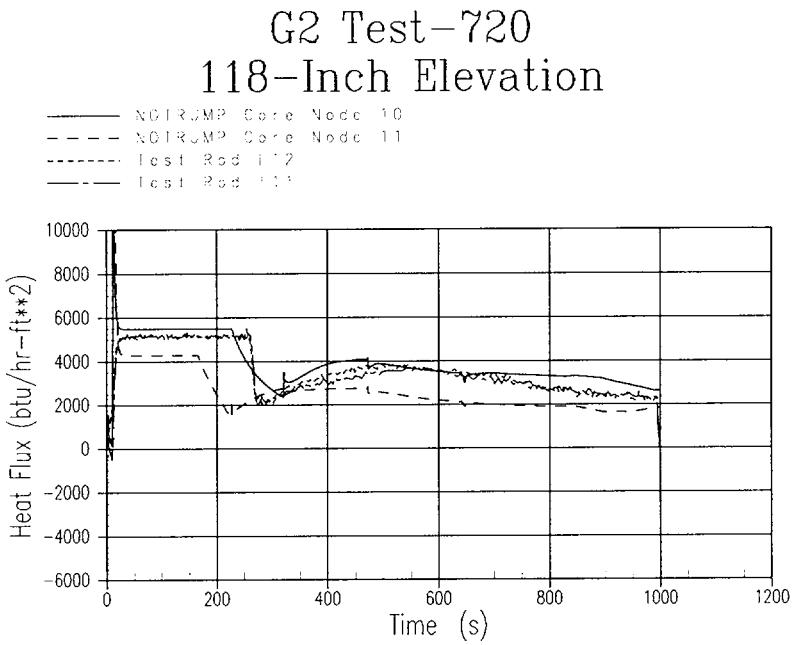

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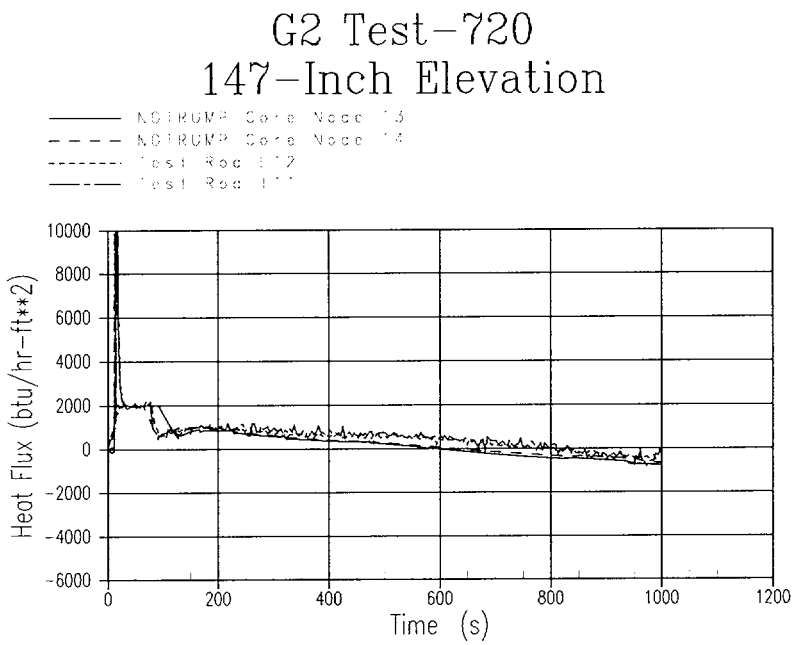

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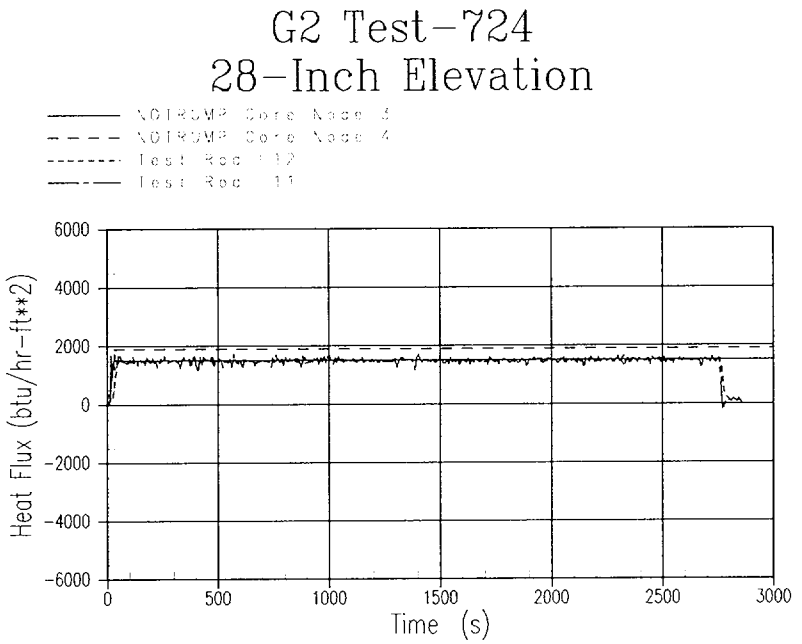

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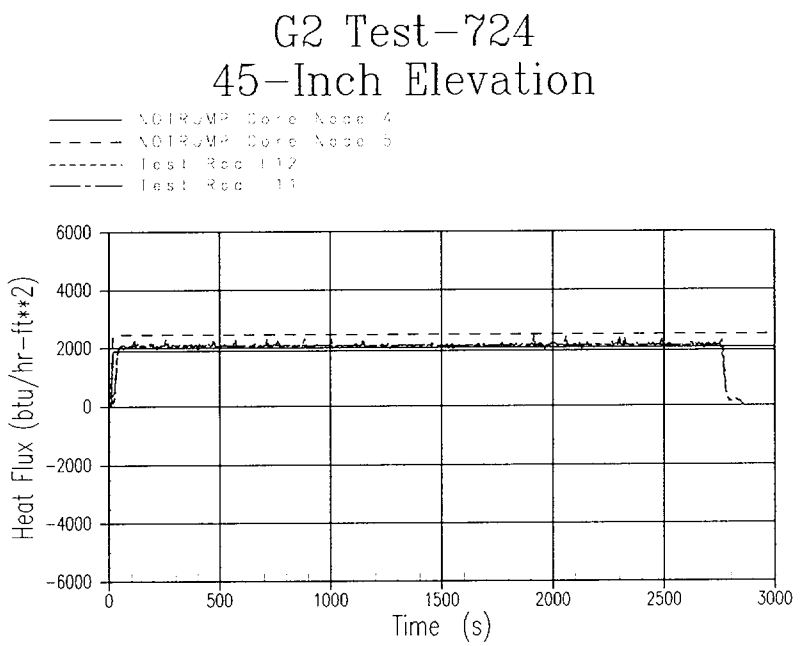
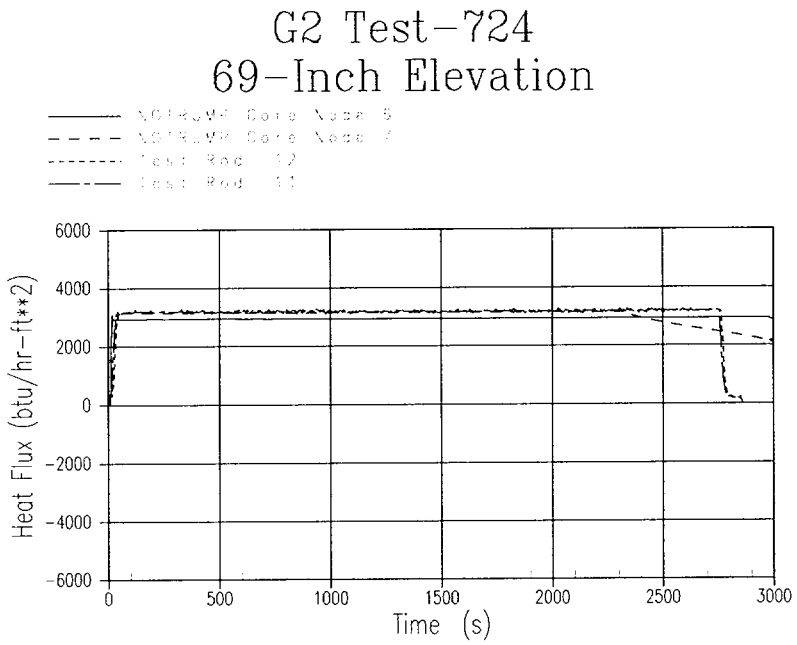


Figure P69-80



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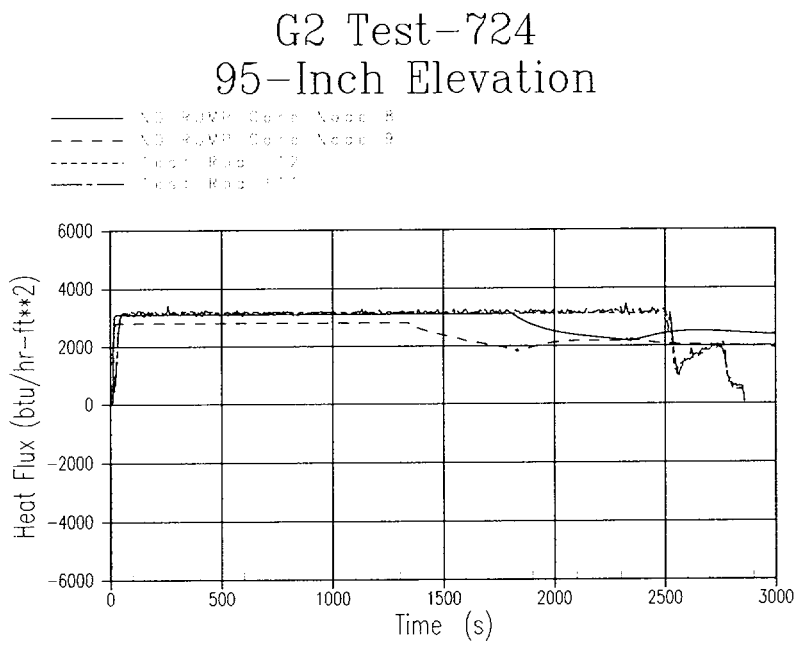


Figure P69-82

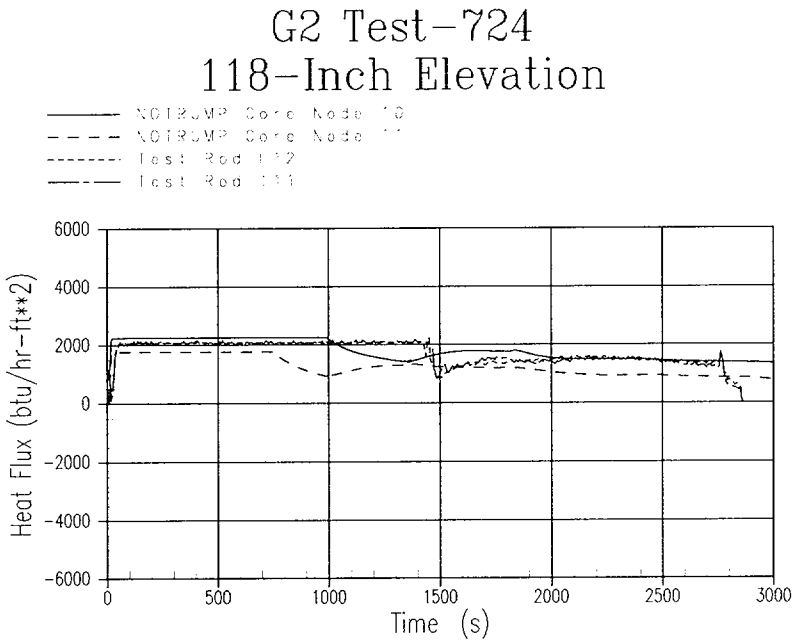
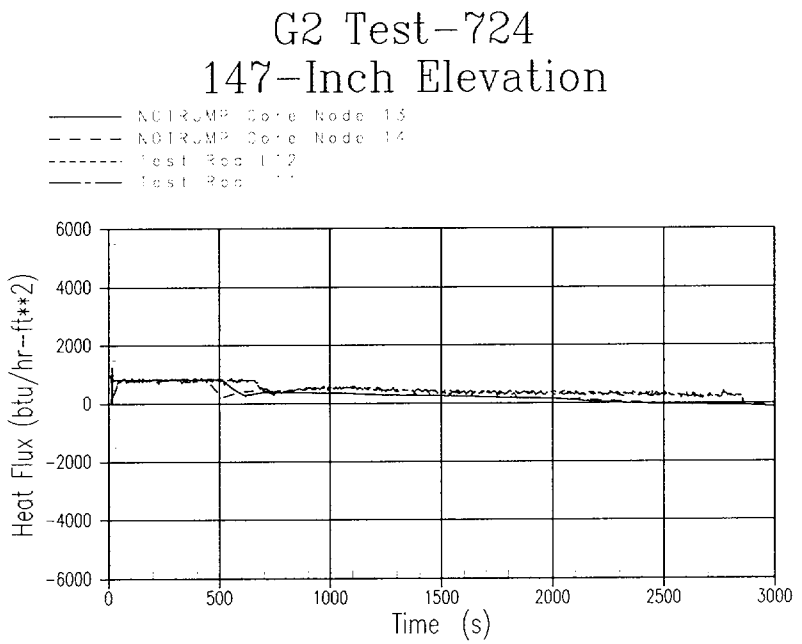


Figure P69-83



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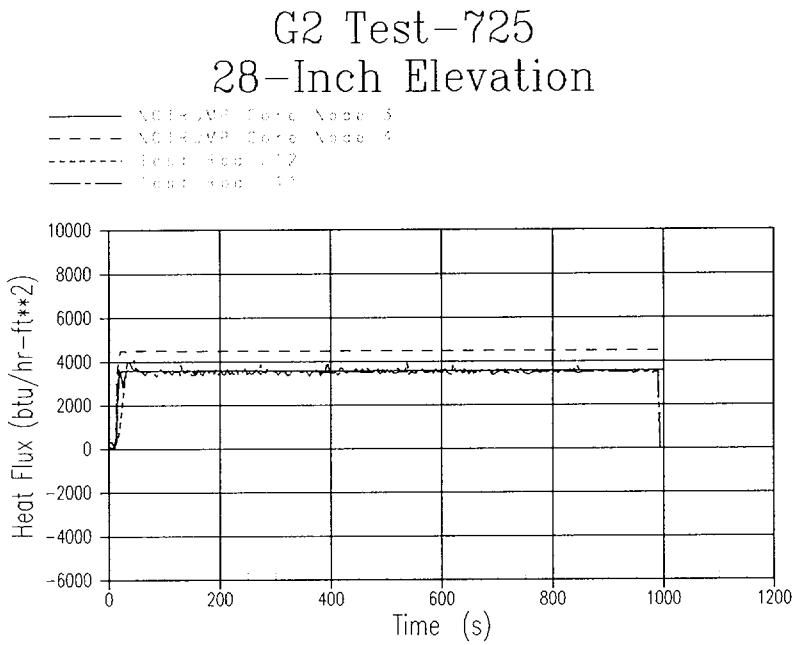

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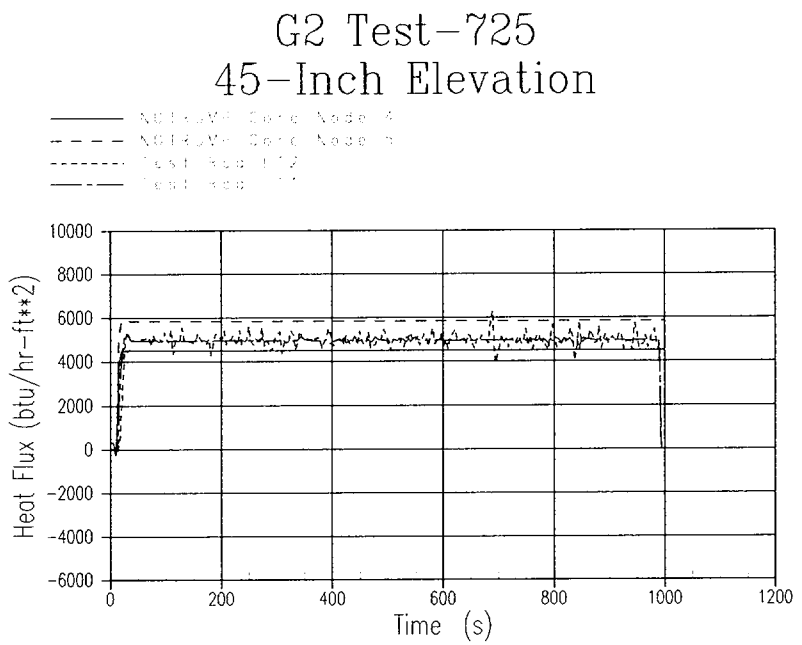

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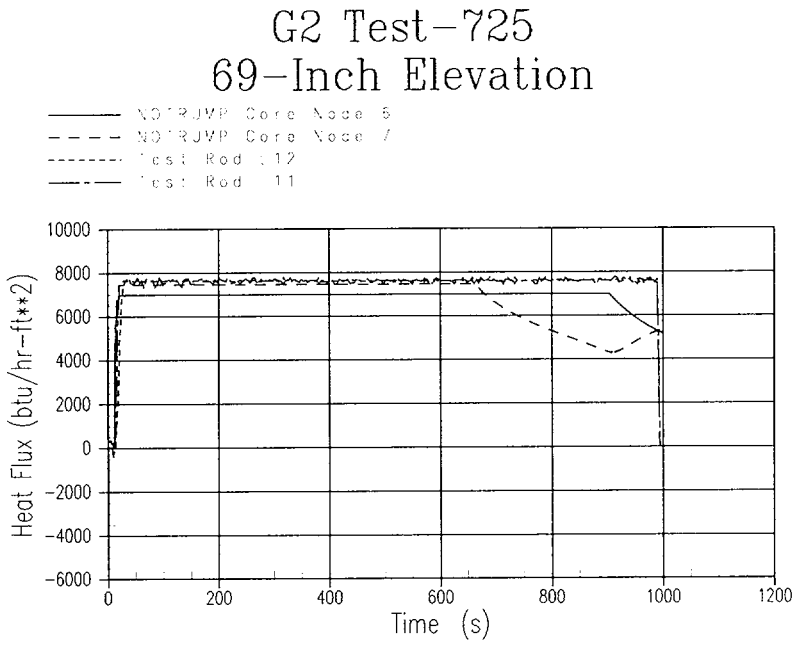


Figure P69-87

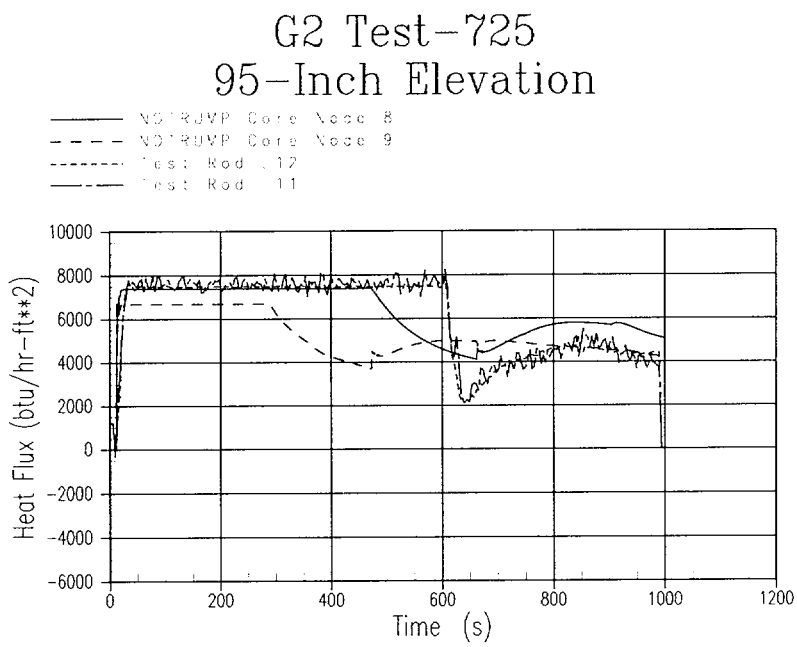
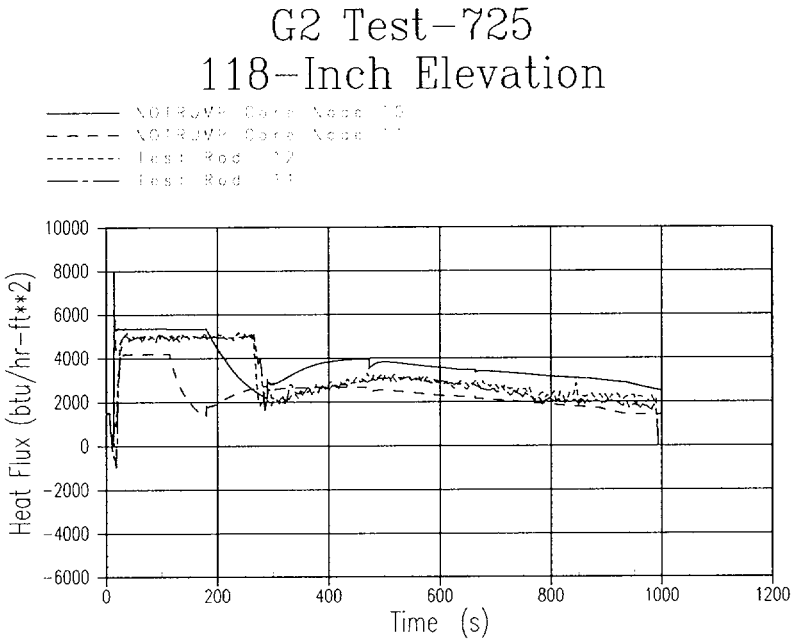


Figure P69-88



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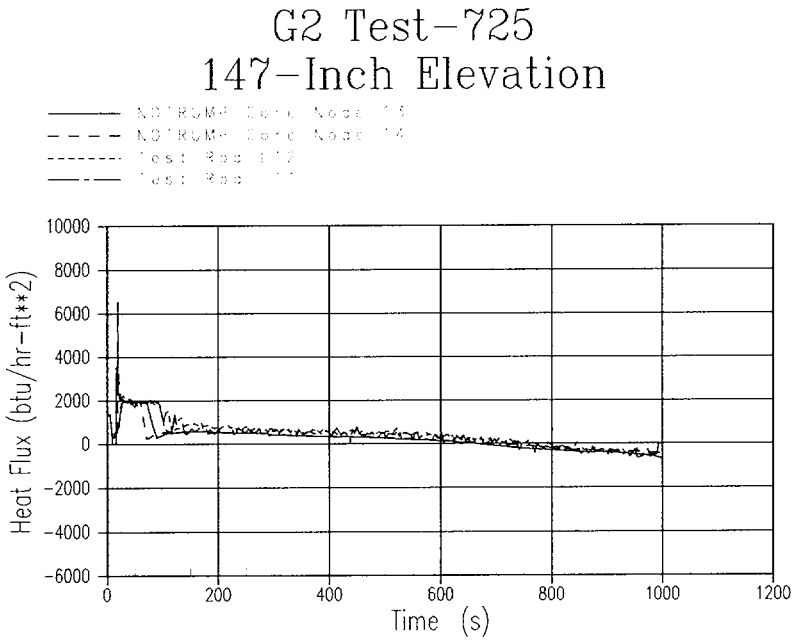

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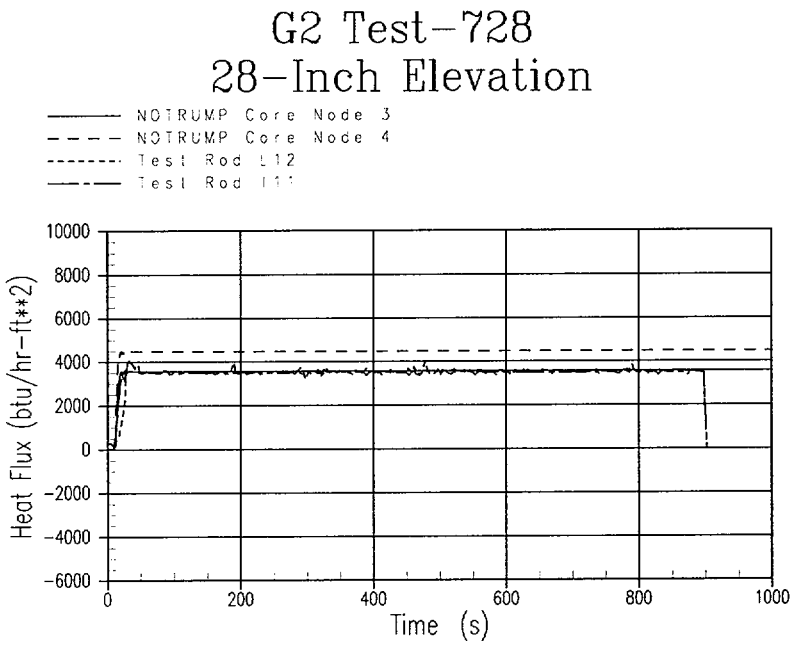

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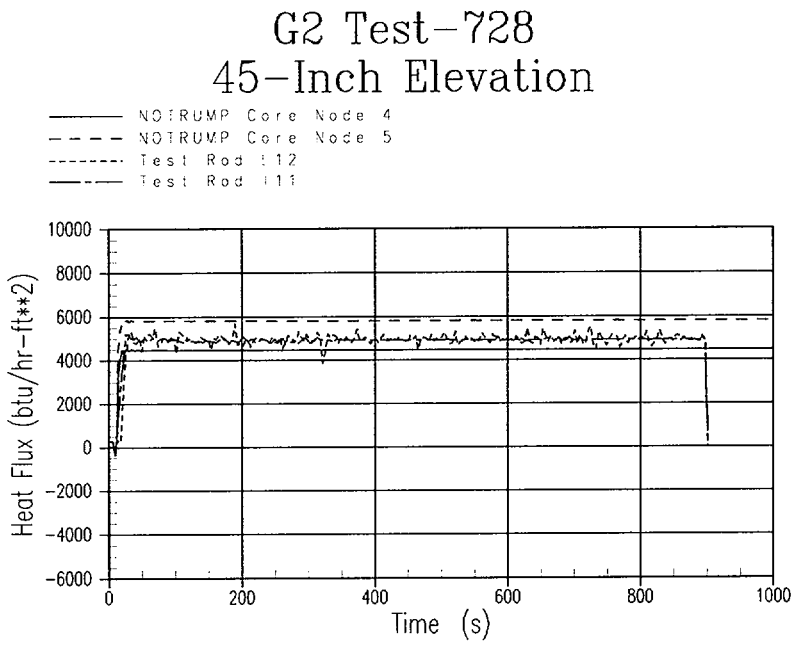

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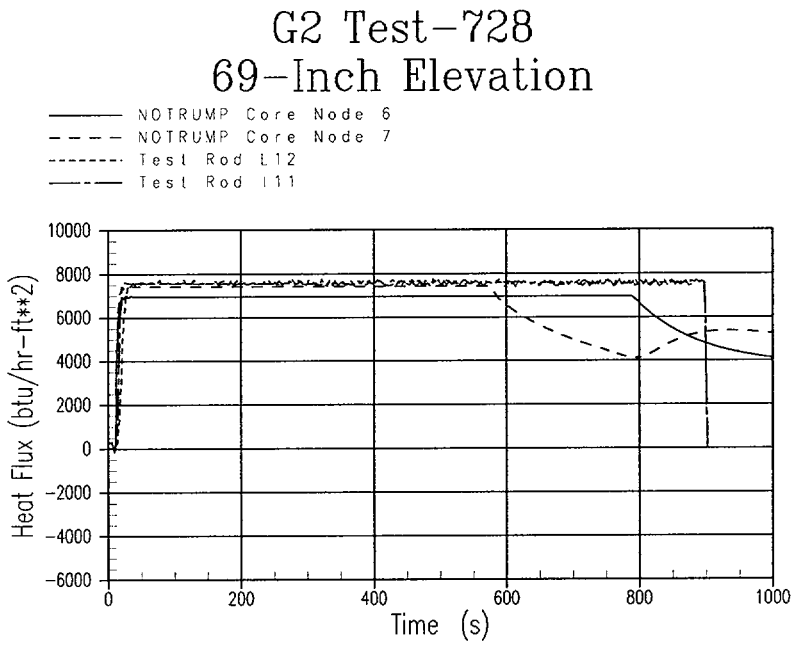

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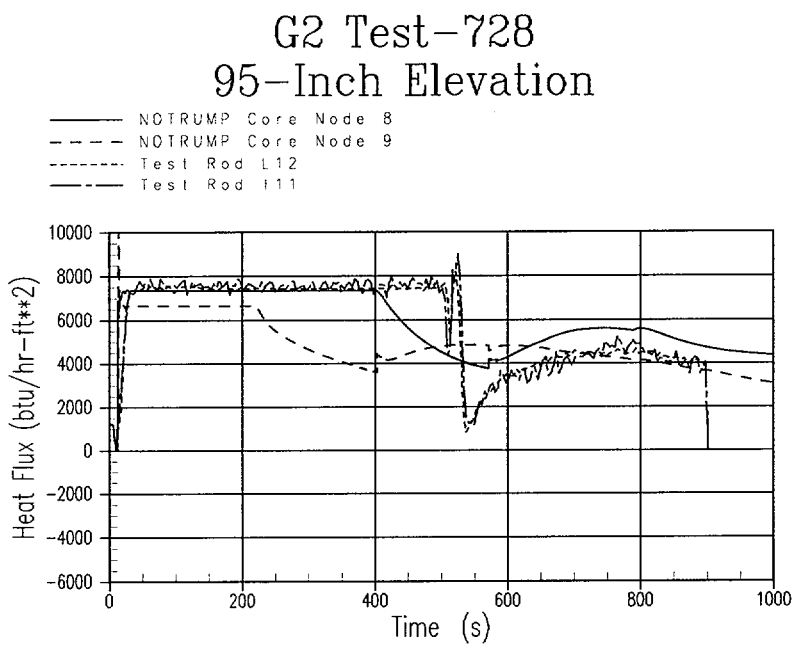

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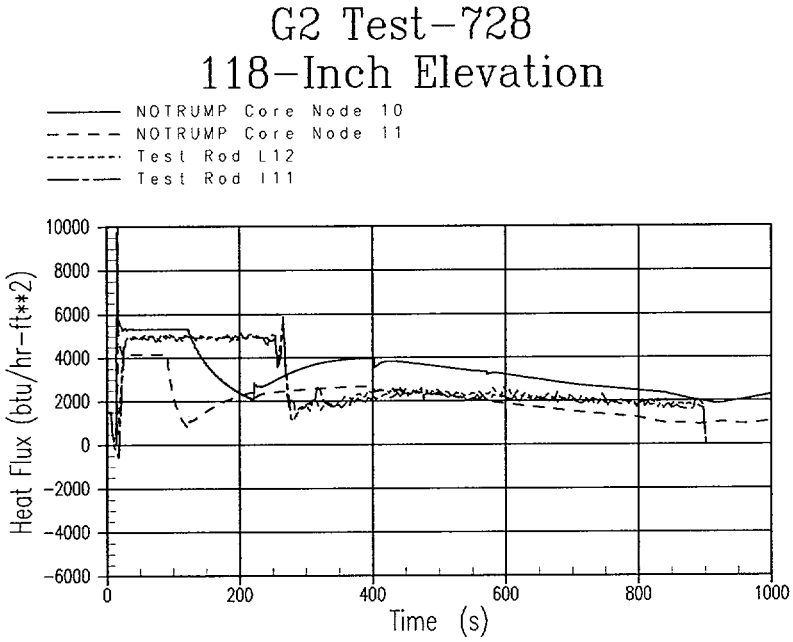

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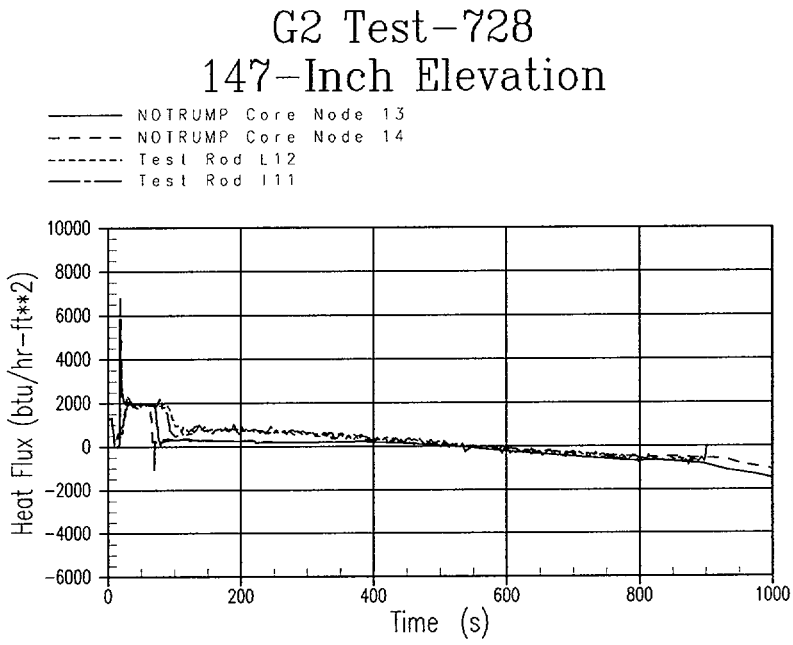

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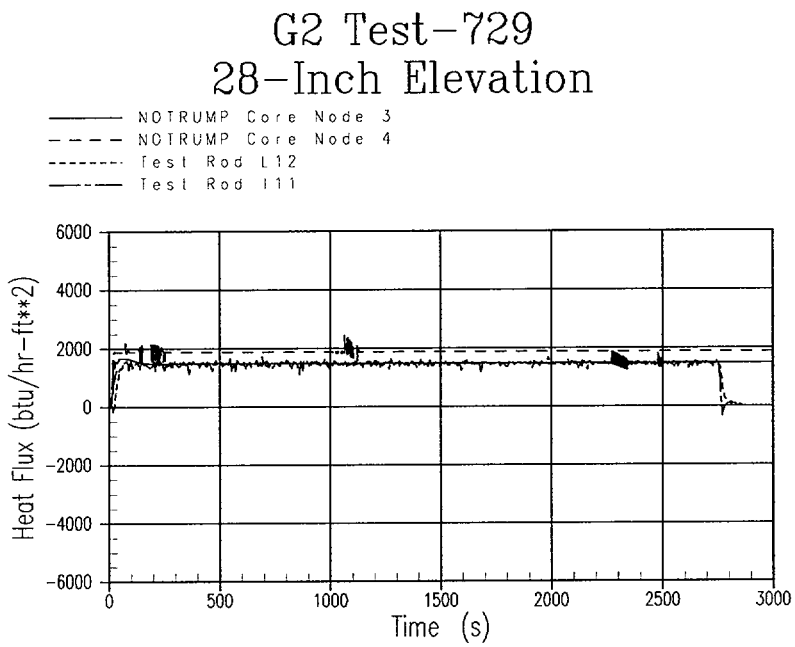

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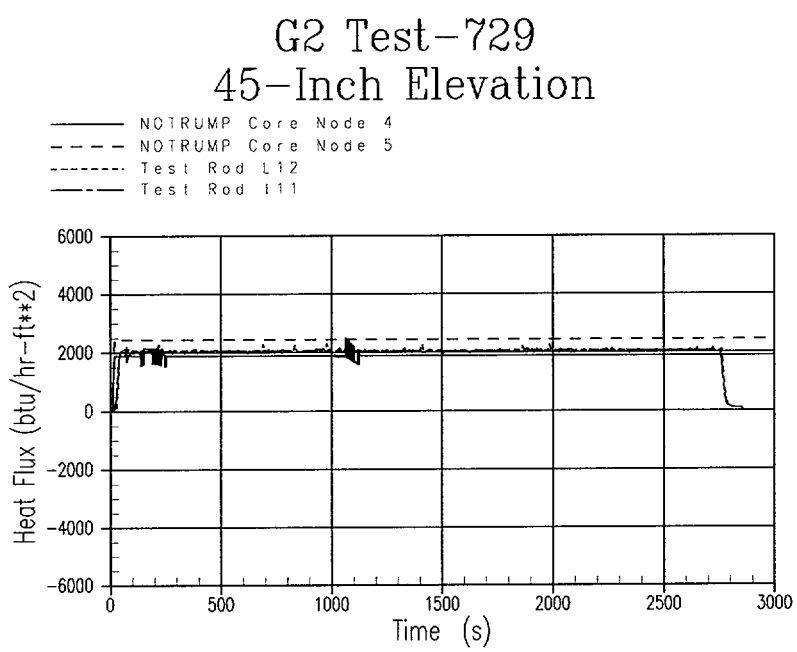

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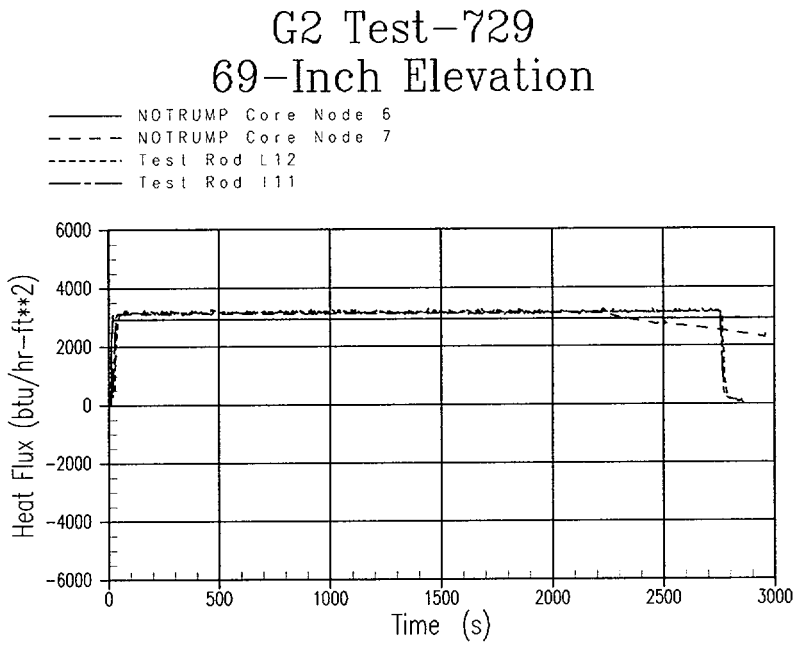

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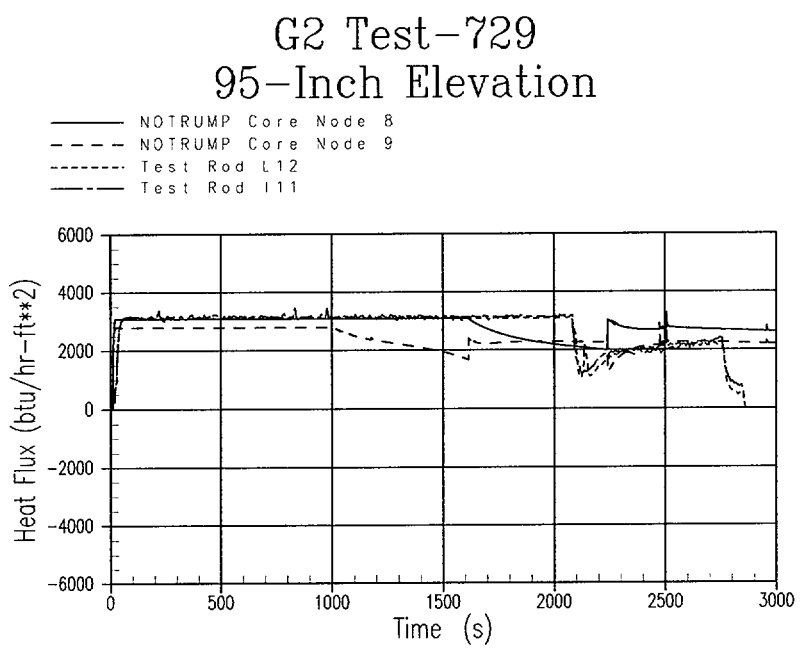
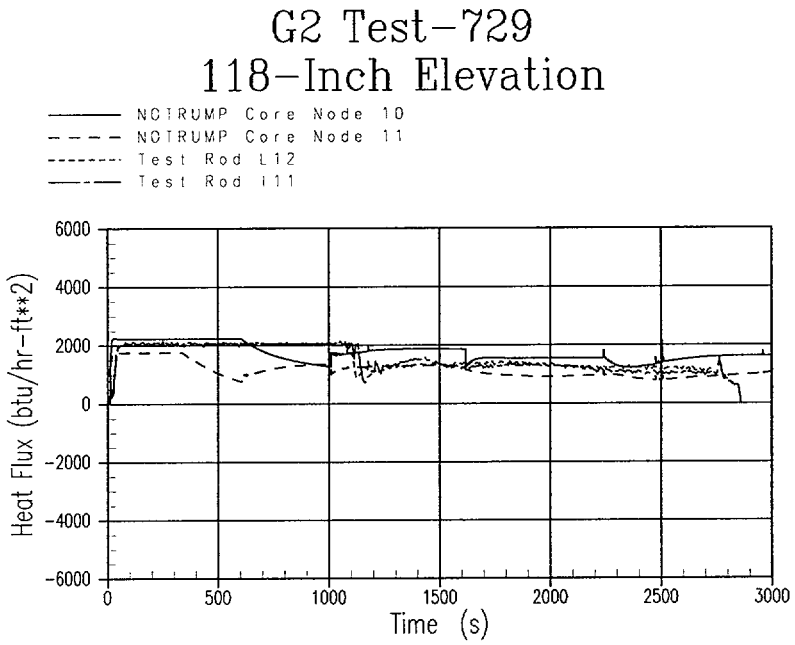


Figure P69-100



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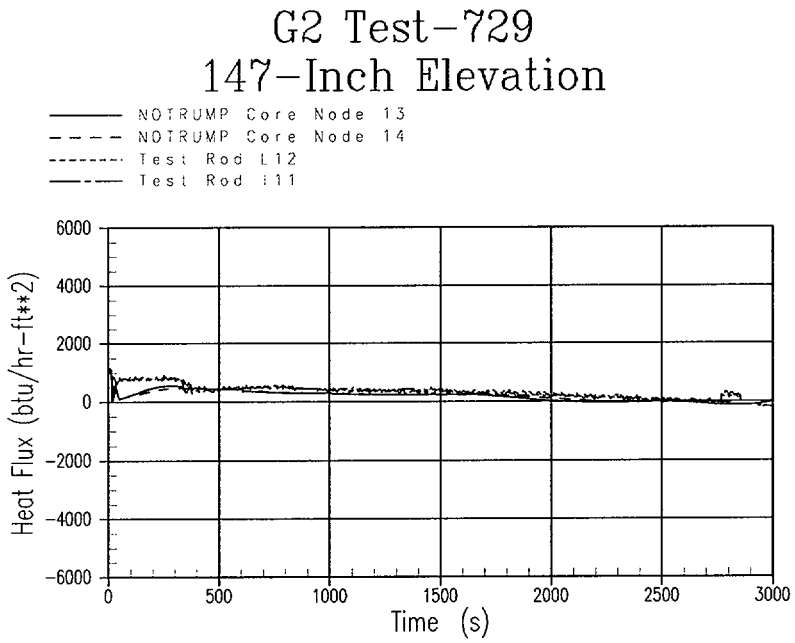

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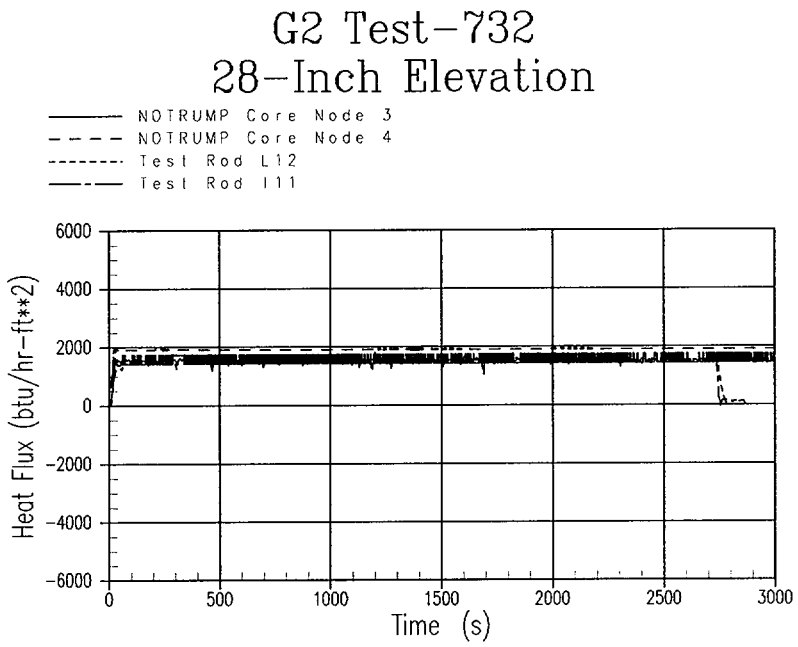
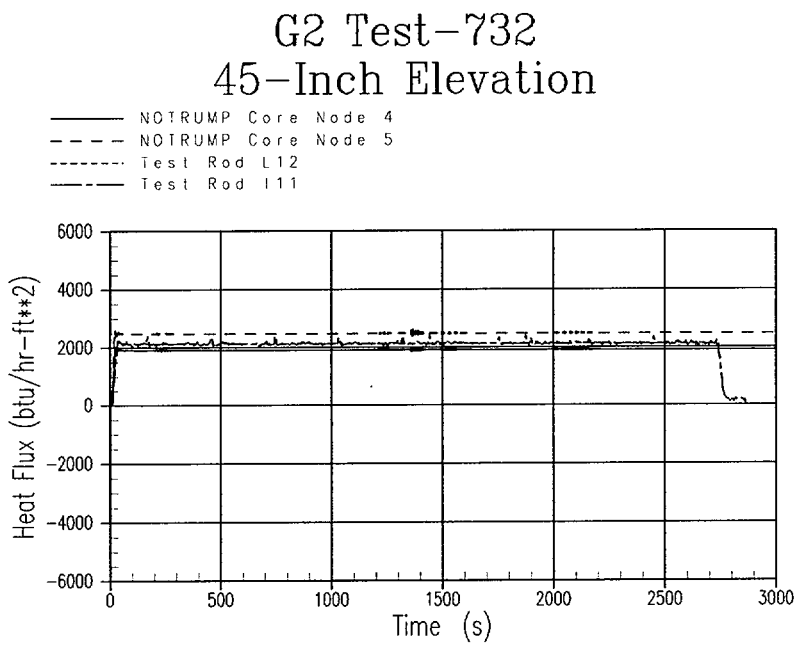


Figure P69-103



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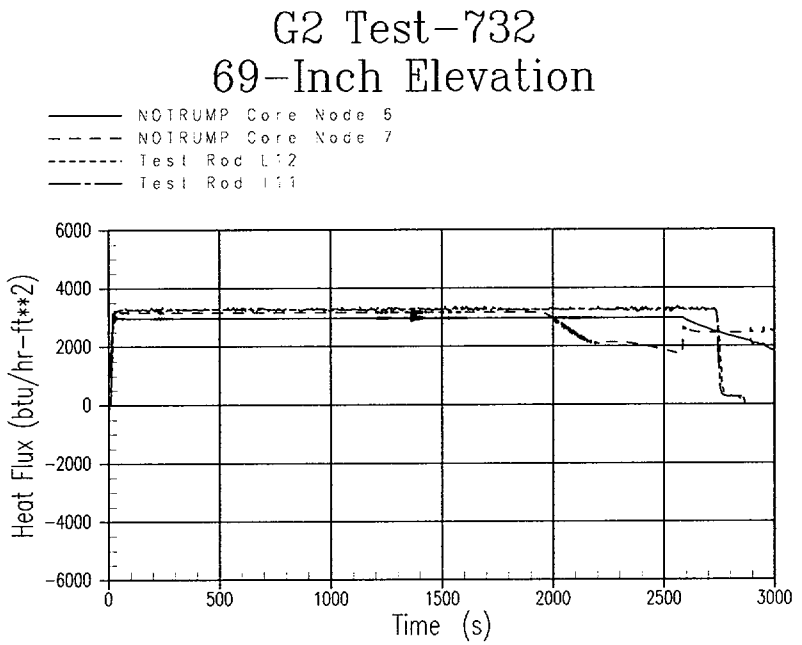


Figure P69-105

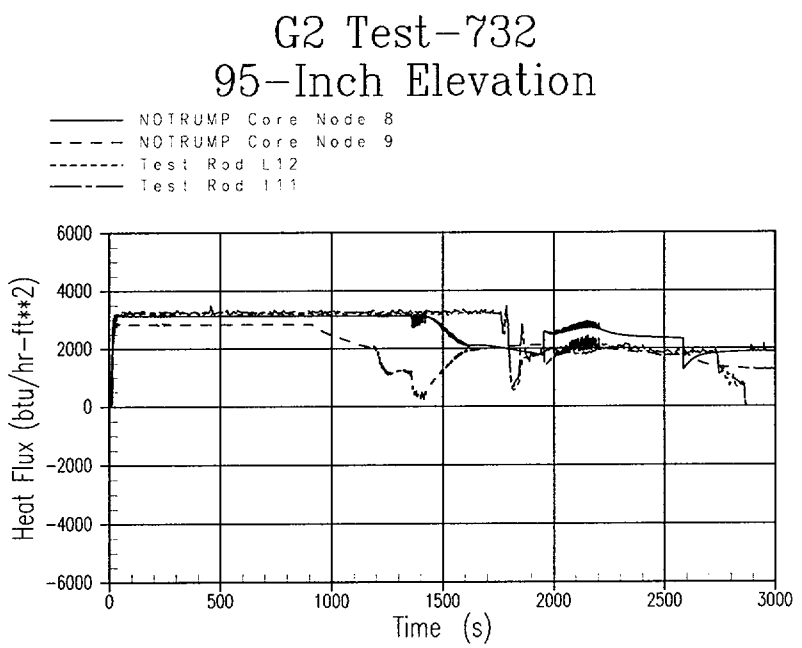


Figure P69-106

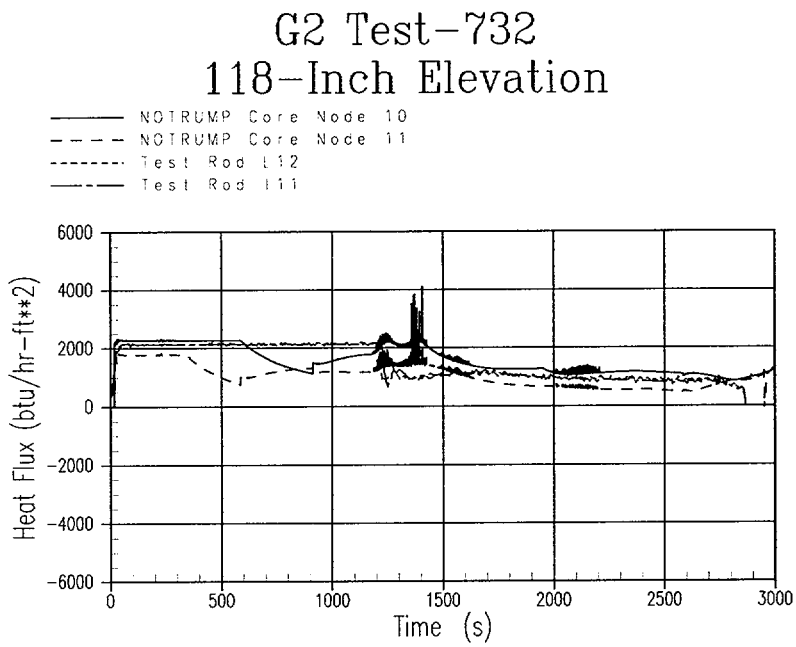
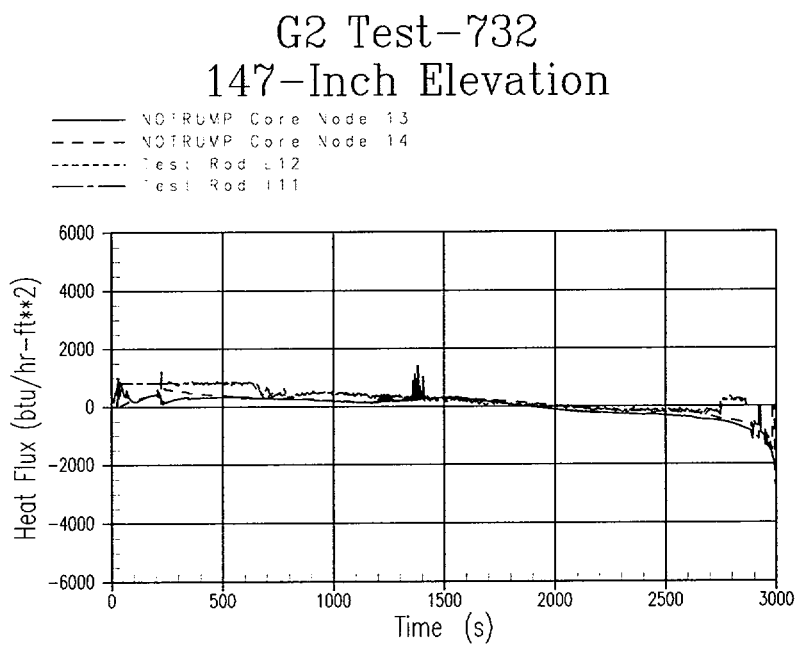


Figure P69-107



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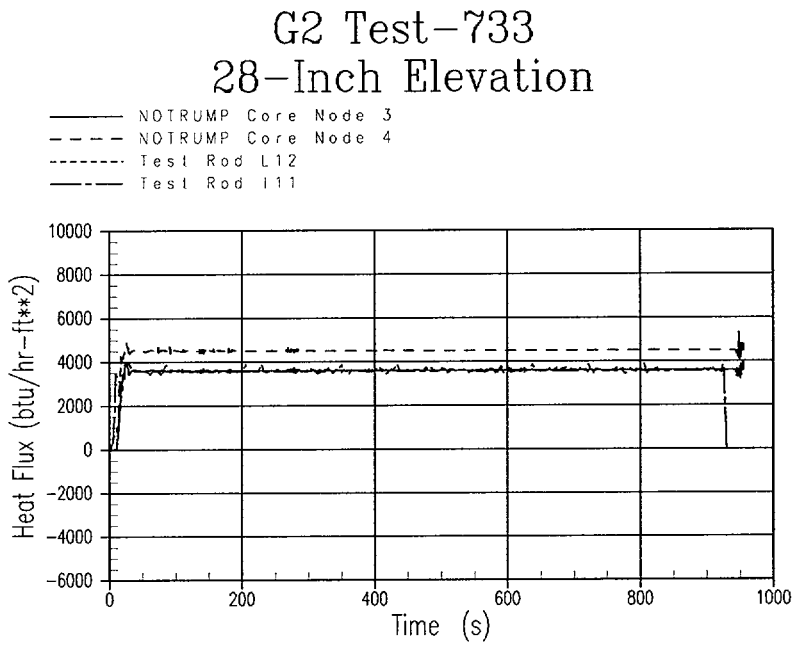
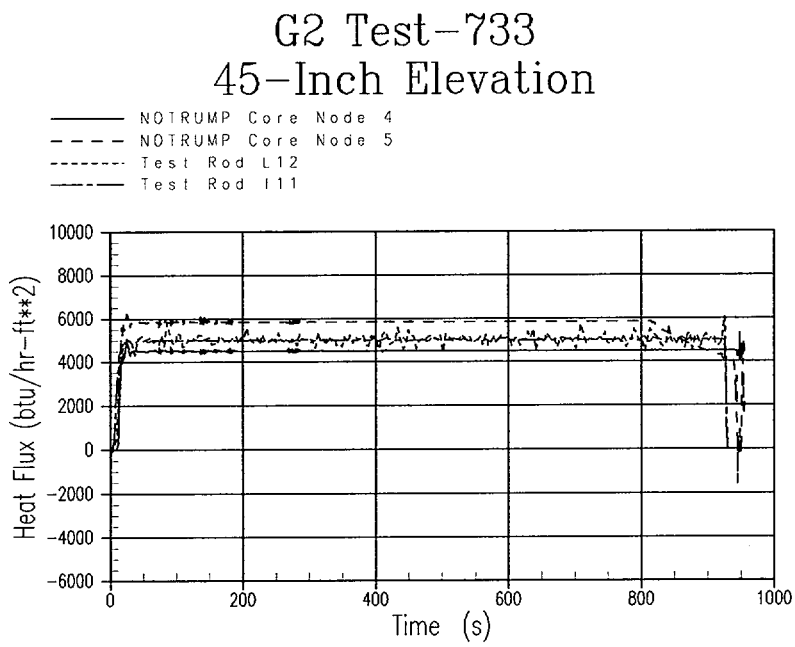


Figure P69-109



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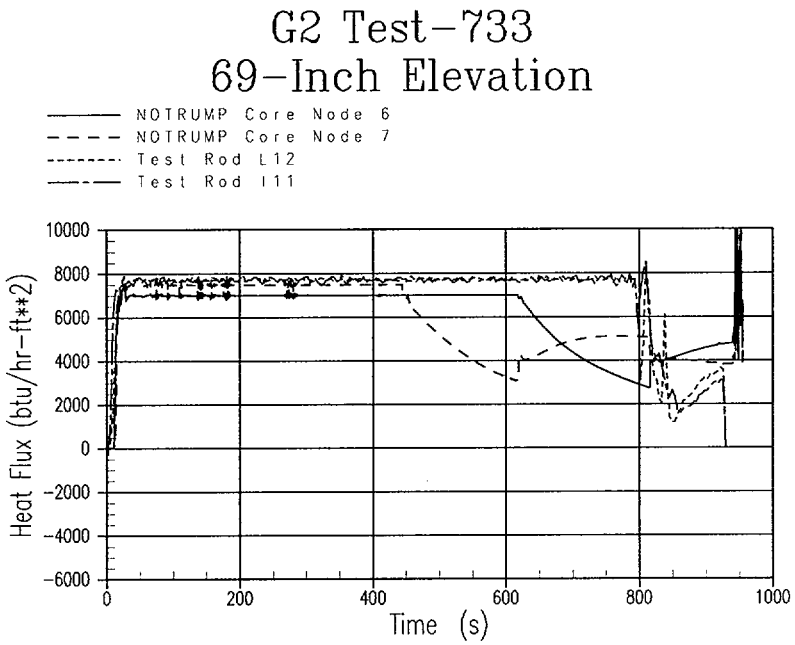

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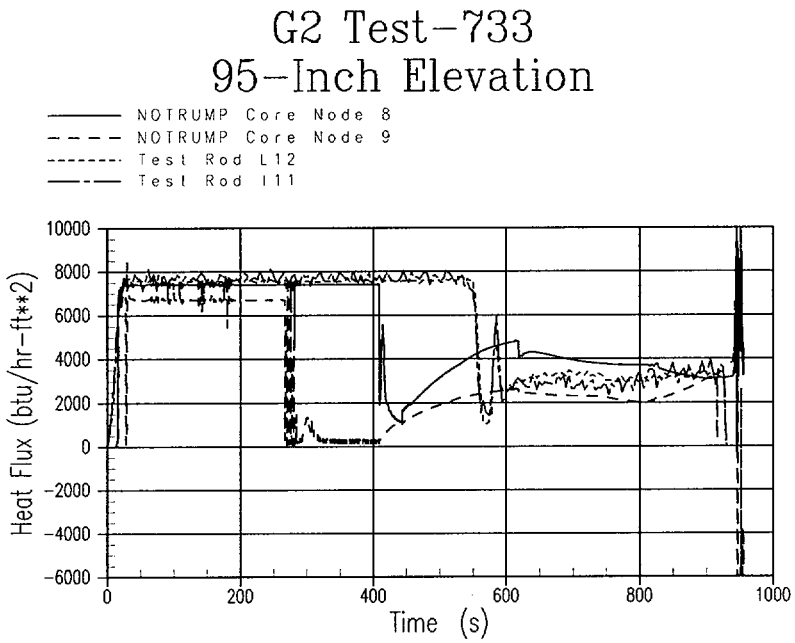

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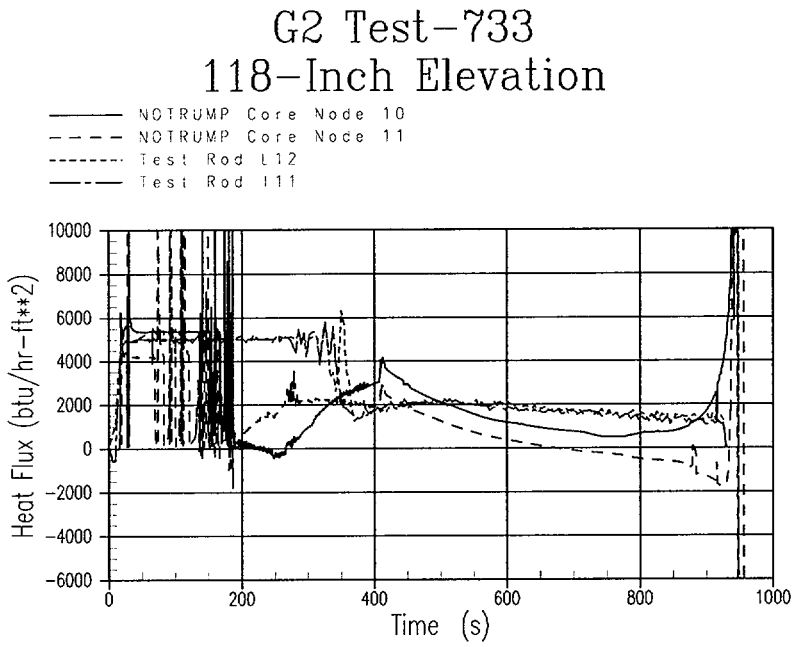

Figure P69-113

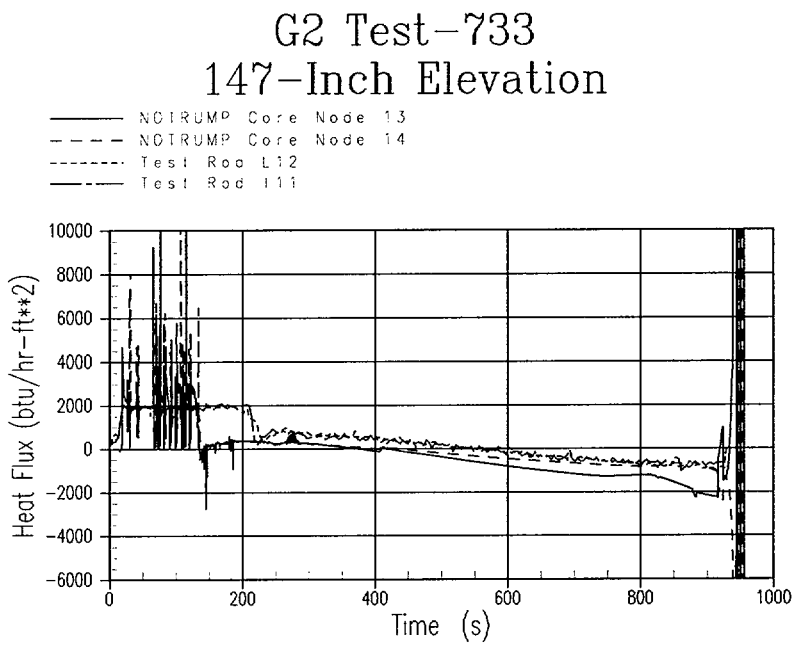

Figure P69-114

REQUEST FOR ADDITIONAL INFORMATION

RAI: P71

Question:

In Section 4.1.2.4, "ADS-IRWST Transition," of WCAP-15613, Westinghouse concluded, based on the top-down scaling analysis of CMT and in-containment refueling water storage tank (IRWST) injection (Sec. 4.1.2.4.1), that (1) the OSU [Oregon State University] and SPES [Simulatore per Esperienze di Sicurezza] facilities are acceptably scaled to AP600 and AP1000 for ADS-4 critical flow and with respect to core power, and therefore can be used for code validation while ADS-4 flow is choked, and (2) when ADS-4 flow is non-critical, only OSU test data can be used for code validation as SPES is distorted due to the oversized ADS-4 vent path. In the bottom-up scaling analysis (Sec. 4.1.2.4.2), Westinghouse concluded that for hot leg flow pattern, OSU is well scaled to AP600 and AP1000, and that SPES is distorted.

However, the staff's preliminary top-down scaling analysis indicated that both Rig of Safety Assessment (ROSA) and SPES are appropriately scaled for AP1000, but not the OSU's Advanced Plant Experiment (APEX). Since the AP1000 has higher core power density than the AP600, it would have higher steam flow in the upper plenum and hot legs, thus higher liquid entrainment in the hot legs and ADS-4, during the ADS-4 blowdown. Since the AP600 and APEX did not show significant entrainment during the ADS-4 blowdown, the AP600 code validation using the APEX results during this phase of transient would be unacceptable for the AP1000 application.

A. Will Westinghouse perform the AP1000 code validation using appropriate SPES or ROSA test data of ADS-4 blowdown? Will this code validation be performed as a part of the Phase 3 design certification application? If not, what is the basis for the acceptability of the AP600 code validation for application to the AP1000?

B. Will Westinghouse obtain additional hot leg entrainment data during ADS-4 blowdown for the AP1000 code validation? If yes, describe how, where and when the additional test data will be obtained. If not, describe the basis for not requiring additional data.

Westinghouse Response:

A. Westinghouse has performed code validation of ADS-4 blowdown behavior using both the SPES and the OSU test facilities. Specifically, the NOTRUMP validation report (Reference 1) included predictions of the ADS-4 behavior for both SPES and APEX (OSU). Therefore, given the staff's preliminary assessment of the acceptability of SPES for ADS-4 behavior, one can conclude that the validation of NOTRUMP against SPES that was performed for AP600 is acceptable for AP1000.

Westinghouse does not agree that the APEX facility is not suitably scaled for this phase of a SBLOCA. The conclusions reached by Westinghouse in WCAP-15613 regarding the scaling of the SPES and APEX test facilities for the ADS-IRWST transition phase for AP1000 are based upon scaling ADS-4 flow when it is choked and when it is non-critical

as both modes are possible. The scaling analysis in WCAP-15613 showed that SPES and APEX are sufficiently scaled to AP1000 when the ADS-4 flow is choked. When the ADS-4 flow becomes sub-critical, SPES becomes distorted while OSU remains acceptable.

Review of the scaling analysis performed by Idaho National Engineering Laboratory (INEL) for the U.S. NRC as presented in Reference 2 reveals that ADS-4 flow is treated in the choked mode only. Subcritical ADS-4 flow is not considered until the IRWST or sump injection phases. Based upon this, it would seem that definitive conclusions regarding the scaling of the ADS-4 flow based upon the treatment in the INEL scaling report could not be made until subcritical flow mode is addressed, especially for a low-pressure test facility such as APEX. The scaling studies performed for AP1000 in Reference 3 addresses scaling of both the critical and sub-critical flow regimes.

B. Westinghouse has concluded that additional hot leg entrainment data during ADS-4 blowdown is not needed for AP1000 code validation. During the ADS-4 blowdown phase, it is expected that higher liquid entrainment may occur in AP1000 relative to AP600. However, this does not render AP600 code validation during the ADS-IRWST transition phase unacceptable for AP1000. The APEX test facility showed significant entrainment during the ADS-4 blowdown period.

In a typical SBLOCA in APEX, notable entrainment occurs in the hot legs and ADS-4 vent paths when the ADS-4 valves are actuated. This can be most readily seen for example in Figures 58 and 64 of Reference 3 for APEX matrix test SB09. Figure 58 shows noticable increase in liquid mass in Hot Leg 1 (non-pressurizer side) coincident with ADS-4 actuation while figure 64 shows ADS-4 liquid being discharged from the system. Some of this entrainment can be attributed to the liquid draining from the pressurizer toward the reactor vessel while some is entrained from the reactor vessel. Based upon the existence of entrainment in the APEX tests and the hot leg scaling analysis of APEX to AP1000, code validation based upon the APEX test facility is adequate for AP1000.

As part of Design Certification, Westinghouse will perform a supplemental calculation using WCOBRA/TRAC of the transition to IRWST injection phase, where Hot leg / ADS-4 entrainment may have an effect on the overall system performance. This supplemental calculation will include suitable validation against the APEX tests.

References:

1. WCAP-14807 Rev. 5, "NOTRUMP Final Validation report for AP600," August 1998.
2. INEL-96/0040, "Top-Down Scaling Analyses Methodology for AP600 Integral Tests," May 1997.
3. WCAP-15613, "AP1000 PIRT and Scaling Assessment," dated February 2001.
4. WCAP-14292, "AP600 Low Pressure Integral Systems Test at Oregon State University Test Analysis Report," September 1995.

REQUEST FOR ADDITIONAL INFORMATION

RAI: P72

Question:

On page 1.11-2 of WCAP-14807 it was concluded that the NOTRUMP passive residual heat removal (PRHR) model contains a model deficiency that needs to be monitored to assure that excess PRHR heat transfer is not calculated. This is because the NOTRUMP code does not model the thermal plume in the IRWST that would occur from extended operation of the PRHR heat exchanger (HX). For AP1000 the heat flux from the PRHR HX will be greater than for AP600 and therefore more likely to produce a thermal plume in the IRWST. Page 1.13-2 of WCAP-14807 states that for PRHR HX flow velocities greater than about 1.5 ft/sec, the heat transfer from the PRHR will be overpredicted because the secondary-side heat transfer coefficient is overpredicted. The flow velocity within the PRHR HX tubes of AP1000 may exceed 1.5 ft/sec under natural circulation conditions since the resistance to flow in the inlet and exit lines to the PRHR has been reduced. The NRC staff's SER for AP600 (NUREG-1512) concludes that the NOTRUMP PRHR model is acceptable because the PRHR HX heat transfer calculated by NOTRUMP for SPES and OSU tests is lower than that measured in the experiments. The PRHRs for the SPES and OSU test facilities were scaled for AP600 which has lower flow velocities and heat fluxes than does AP1000. Justify that the PRHR HX heat transfer model is conservative for modeling AP1000 by comparing the NOTRUMP predictions for AP1000 with appropriate experimental data.

Westinghouse Response:

For LOCA events, the PRHR provides decay heat removal from the RCS during the period between the S-signal and ADS actuation. The length of this period is dependent on the break size. Unlike non-LOCA events where the PRHR flow is single phase, the flow transitions from single-phase to two-phase for LOCA events when the RCS inventory is reduced to the level of the hot legs. When the PRHR flow transitions from single-phase to two-phase, the buoyancy driving head, (the density difference between the PRHR supply and return lines) increases. However the pressure drop increases more dramatically due to the two-phase multiplier, and the net result is lower flow during two-phase operation.

In the response to AP600 RAI 440.339, Westinghouse presented results of noding sensitivity studies that indicated that the overall prediction of PRHR heat transfer performance was not significantly impacted by more detailed noding of the PRHR and IRWST. The nucleate boiling heat transfer correlation used in NOTRUMP is dependent on the IRWST saturation water temperature, and is not dependent on the local IRWST water temperature and the nucleate boiling heat transfer predictions are unaffected by local plume temperatures. Therefore, the NOTRUMP noding of the PRHR and IRWST are sufficient to adequately predict PRHR heat transfer, and no modifications are necessary to model the thermal plume in the IRWST that develops during extended operation of the PRHR.

REQUEST FOR ADDITIONAL INFORMATION

WCAP-14807 determined that when the exterior of the tubes are in the nucleate boiling regime and the interior of the tubes are in single-phase forced convection, the heat transfer could be overpredicted. It was also determined that the fluid velocity inside the tubes had a large effect on the overall heat transfer. When the tube velocity was less than 1.5 ft/s, the thermal resistance due to the film loss inside the tubes controlled the overall resistance, and for velocities greater than 1.5 ft/s, a degradation of the overall heat transfer could be considered. In WCAP-14807, Westinghouse recommended that a 50% reduction in the PRHR area be applied when the single-phase velocity inside the tubes exceeded 1.5 ft/s for a considerable period of time during the transient. For AP1000 LOCA analyses using NOTRUMP, Westinghouse will continue to follow this recommendation. Westinghouse will apply a 50% reduction to the AP1000 PRHR heat transfer area when the single-phase velocity inside the tubes exceeds the 1.5 ft/s limit for a significant period of time during transient analyzed with NOTRUMP.

For the integral tests (SPES-2 and OSU), NOTRUMP was shown to underpredict the flow in the tubes after the onset of two-phase flow. This is shown if Figure 1 for SPES-2. Lower predicted flow results in conservatively lower heat removal. In addition, nodalization sensitivity studies were completed for AP600 in response to NRC request for additional information, RAI-440.399. These studies showed that dividing the IRWST into several nodes did not significantly affect the results. Therefore, NOTRUMP conservatively predicts PRHR heat transfer for portions of the event when the PRHR flow is two-phase. No reduction in the PRHR heat transfer area would be applicable to the two-phase portion of the event.

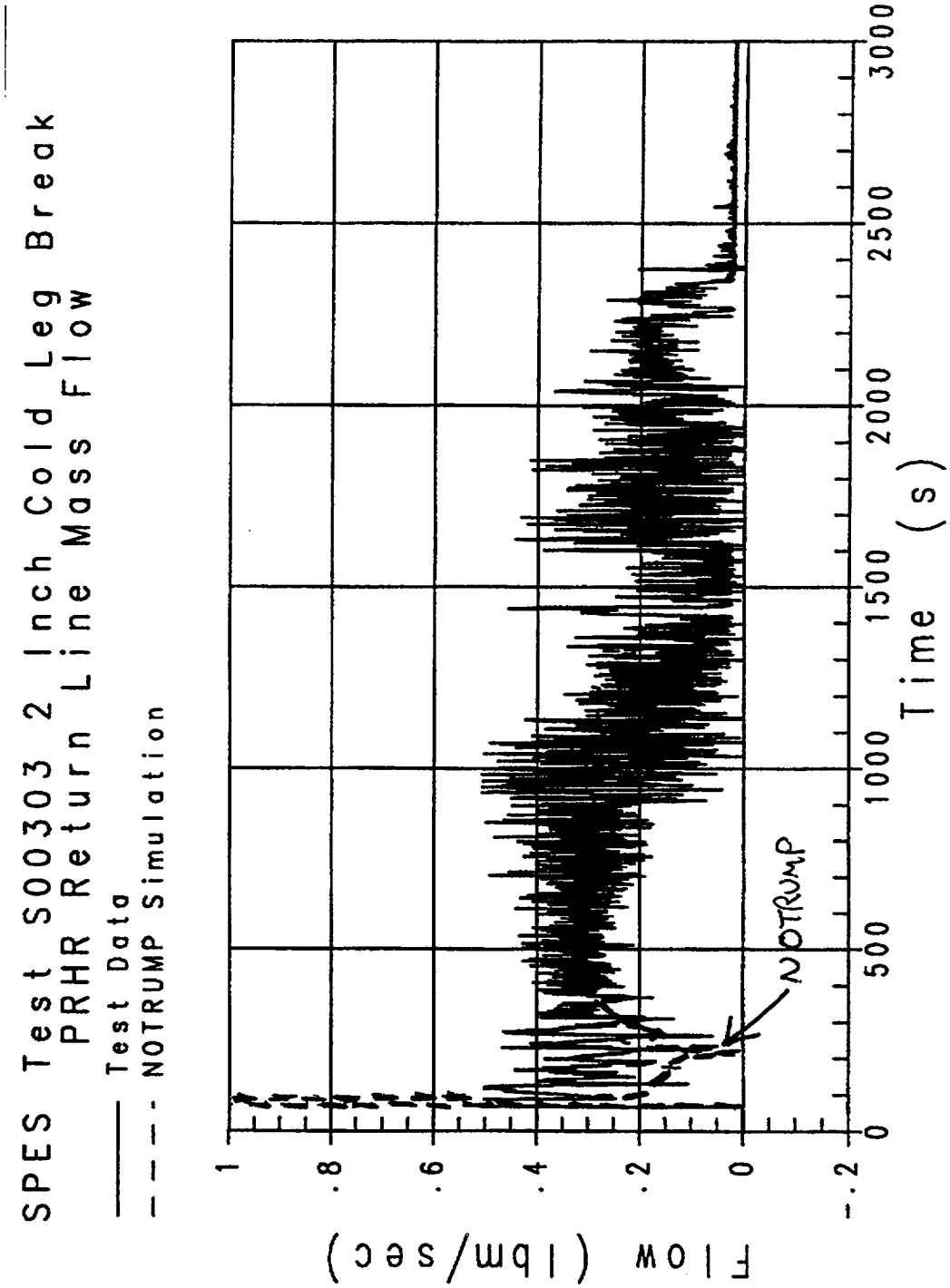

Figure 1: NOTRUMP Prediction of SPES-2 2-inch Cold Leg LOCA PRHR Flow

REQUEST FOR ADDITIONAL INFORMATION

RAI: P73

Question:

Similar to NOTRUMP, the LOFTRAN code also models the IRWST as one node. LOFTRAN also does not model the thermal plume that would occur in the IRWST from extended operation of the PRHR HX. LOFTRAN uses the average IRWST temperature to determine heat flow between the PRHR HX and the IRWST. Considering that elevated IRWST local temperatures could occur in the region of the PRHR HX, justify that LOFTRAN will conservatively calculate PRHR heat flow for the AP1000 transients and accidents listed in Chapter 2.1 of WCAP14307.

Westinghouse Response:

The heat transfer correlations used in LOFTRAN to model PRHR heat transfer performance are not strongly dependent on the IRWST sub-cooled water temperature. Westinghouse performed LOFTRAN calculations comparing the PRHR HX performance with a stratified water temperature in the IRWST (based on data from the SPES2 and PRHR HX test program data), with results that assumed a homogeneous IRWST water temperature. This study indicated that LOFTRAN provides conservative results using the homogeneous IRWST water temperature model for calculating the PRHR HX heat transfer. This study was provided to the NRC in Reference P73-1.

Westinghouse also performed calculations using the PRHR HX heat transfer correlation developed from the AP600 PRHR HX testing program that favorably compare to the actual heat transfer results obtained in the ROSA test facility. These calculations utilized inlet fluid conditions obtained from the actual test conditions but used averaged IRWST water temperatures. The calculation results predicted the actual ROSA HX outlet and tube wall temperatures well, with calculated temperatures within a few degrees of the measured temperatures. This information was provided to the NRC via Reference P73-2.

The above studies and test data comparison conclusions are valid for the AP1000 PRHR HX performance following non-LOCA initiating events, where the primary fluid temperature stays elevated and where the PRHR HX performance is important. In LOFTRAN, PRHR heat transfer is calculated using either one of two correlations, based on whether nucleate boiling is predicted (modified Rohsenow correlation), or whether natural convection is predicted (McAdams). The boiling heat transfer correlation is dependent on the local IRWST water saturation temperature, and is not dependent of IRWST local fluid temperature, while the McAdams correlation is dependent on the fluid temperature. Following non-LOCA events in the AP600, the dominant heat transfer mode for the PRHR HX is the nucleate boiling mode. This nucleate boiling mode is even more predominant for the AP1000 PRHR HX, due to increased primary fluid flow rates and higher local heat fluxes that are achieved.. The increased heat flux results in a higher delta-T between the tube outside wall and the IRWST fluid. This delta-T (T_{OWE} - T^*) is sufficiently high, to result in nucleate boiling conditions even when the IRWST water is cold. For

these conditions, the modified Rohsenow nucleate boiling correlation applies, and the heat transfer rate is dependent on the IRWST saturation water temperature. For these non-LOCA conditions, the primary fluid temperature does not decrease until all the IRWST water has been heated to, or near, saturation temperature. Thus, the AP1000 PRHR HX operating conditions makes the heat transfer to the IRWST insensitive to the local (plume) temperature distribution in the vicinity of the PRHR HX.

T^* = The sub-cooled IRWST water temperature for natural convection, low heat flux conditions, or = The saturation temperature of the IRWST water for nucleate boiling conditions (Reference P73-3).

References:

1) Westinghouse Letter DCP/NRC-1063 dated October 2, 1997.
2) Westinghouse Letter DCP/NRC-1032 dated September 18, 1997.
3) WCAP-12980, Revision 2 "AP600 Passive Residual Heat Removal Heat Exchanger Test Final Report" dated September 1996.