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EXXON NUCLEAR METHODOLOGY FOR BOILING WATER REACTORS VOLUME 2C

VERIFICATION AND QUALIFICATION OF EXEM

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VERIFICATION AND QUALIFICATION OF EXEM

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1.0 INTRODUCTION AND SUMMARY

This Document (Volume 2C) summarizes the verification and qualification work to demonstrate that EXEM is an acceptable evaluation model for performing loss-of-coolant accident (LOCA) analyses for jet pump BWR's. EXEM is an extension of the approved Exxon Nuclear non-jet pump BWR evaluation model and consists of three codes: RELAX, FLEX and HUXY. Volume 2 of this Document generally describes the EXEM model, including interaction of the three codes, and discusses its conformity to 10 CFR 50 Appendix K criteria. Volume 2A describes the improvements made to the approved RELAP4-based ENC blowdown code for application to jet pump BWR's; this improved blowdown code has been named RELAX. Volume 2B describes the FLEX code, which was developed from the approved ENC PWR reflood model REFLEX, for analysis of the refill and reflood period of a jet pump BWR LOCA. The approved ENC BWR heatup code HUXY remains unchanged and is to be used for the heatup analyses of jet pump BWR plants.

Section 2.0 of this report presents the verification work in support of the changes made to the approved version of RELAP4 to develop RELAX.

Section 3.0 provides the verification of the FLEX model. This verification is based on the good agreement between the data taken in ENC's Fuel Cooling Test Facility (FCTF) and the FLEX predictions of these spray and reflood tests.

Section 4.0 provides the overall qualification of the EXEM model to the integral TLTA test data. The TLTA (Two-Loop Test Apparatus) facility is a scale model of a jet pump BWR plant in which LOCA tests have been run, both with and without ECCS injection. The good agreement in the trends resulting from ECCS injection, and the conservative prediction of time of reflood and clad temperatures form the basis of the overall qualification of EXEM.

Thus, EXEM is shown on an individual model and overall basis to be an acceptable evaluation model for analyses of LOCA events in jet pump BWR's.

2.0 RELAX VERIFICATION

RELAX is a RELAP4-based LOCA blowdown analysis code. RELAP4-EM/ENC28B[1] is the approved Exxon Nuclear Company blowdown computer program for use in ECCS Evaluation Models on PWR's and NJP-BWR's. RELAX[2] was developed by adding a new jet pump model, a slip flow model, and additional heat transfer regimes to RELAP4-EM/ENC28B. The focus of this section is on these new models, as the RELAP4-EM/ENC28 code has previously been verified and approved for licensing analyses[3].

The RELAX verification described herein is divided into two parts: This section (2.0) describes the verification of the individual models, and Section 4.0 focuses on the qualification of the composite blowdown model.

2.1 RELAX JET PUMP MODEL

The RELAX jet pump model was verified against published experimental data taken on BWR jet pumps. The first set of data consists of single-phase flow data for a 1/6 scale model jet pump built by General Electric but tested at the Idaho National Engineering Laboratory (INEL)[4]. This data covered the full range of flow conditions of interest as shown by typical M-N curves in Figure 2.1. The second set consists of single-phase flow data for the TLTA jet pump[5]. The basic approach used to check the analytical model was to compare predicted and measured pressure differences for the experimental flow conditions.

2.1.1 INEL Jet Pump Test Data

Jet pump tests at the INEL have been performed and reported for a series of tests on a 1/6 scale model jet pump(4). The tests were performed under cold conditions and at the temperature and pressure conditions of a BWR. The tests considered all six possible flow types as shown in Figure 2.1. Data at 282°C were the most extensive and were used to assess the jet pump model behavior.

Principal dimensions for the jet pump are as follows:

Drive Diameter	0.1050 ft (0.0304 m)
Jet Diameter	0.0459 ft (0.0144 m)
Mixing Section Diameter	0.0951 ft (0.0290 m)
Mixing Section Length	1.1877 ft (0.3616 m)
Diffuser Exit Diameter	0.2362 ft (0.0720 m)
Exit Pipe Length	0.1903 ft (0.0580 m)
$R = (D_{jet}/D_{mix})^2$	0.247

These dimensions are geometrically very similar to the TLTA jet pump as indicated by the similarity of the R ratio. One additional difference is the separation of the nozzle exit from the plane of the suction by 0.0014 m (0.055 in).

2.1.2 TLTA Test Data

Reverse flow tests have been performed on one of the TLTA jet pumps(4). These tests considered Type 1- and Type 3+ flow (see Figure 2.1 for flow regime definition). The tests were performed with water at a nominal temperature of 70°F. Principal dimensions for the jet pump are as follows:

Drive Diameter	0.07817 ft (0.0238 m)
Jet Diameter	0.03775 ft (0.0115 m)
Mixing Section Diameter	0.07617 ft (0.0232 m)
Mixing Section Length	0.95075 ft (0.2898 m)
Diffuser Exit Diameter	0.18883 ft (0.0576 m)
Diffuser Length	1.1481 ft (0.3499 m)
$R = (D_{jet}/D_{mix})^2$	0.246

2.2 LEVEL SWELL - COMPARISON TO EXPERIMENTAL DATA

The RELAX drift flux model was verified in the challenging application of a swell test. A swell test is demanding because counter-current conditions exist throughout the mixture so the drift flux model must predict counter-current conditions as well as mixture level phenomena. The data was taken from the G.E. blowdown level swell test series(7). This blowdown experimental series was performed in a cylindrical test vessel 1.0 foot in diameter by 14 feet high. For this experiment, the initial level was at 8.14 feet and the fluid was saturated with a typical BWR reactor pressure of 1042 psia.

The break flow was used as a boundary condition in the analysis to obtain the experimental depressurization rate. The vessel was represented by 13 vertically stacked control volumes and associated heat slabs, 12 flow junctions all with drift flux, and connecting critical flow path to a suppression tank.

2.3 CCFL VERIFICATION

The Counter-current Flow Limiting (CCFL) phenomenon is a basic feature of RELAX hydraulics. To verify the RELAX CCFL model, an upper tie plate CCFL problem was chosen. This problem consisted of a pool of upper plenum water above a 12 foot channel of saturated steam. The flow channel and upper tie plate were prototypic of a BWR fuel assembly. The vapor exiting the bundle limited the water that drained down through the upper tie plate.

2.4 TLTA - BROMLEY HEAT TRANSFER COEFFICIENT

The Bromley correlation is the classic low void fraction film boiling model. A thin vapor film is assumed to prevent the denser mixture from contacting the heated surface. Bromley(8) was the first to analyze the convective contribution to stable film boiling on a heated horizontal cylinder and extend this model to vertical surfaces. His classic theoretical equation has held up well under the scrutiny of investigators through the years. This correlation is a conservative model in that heat is transferred through the film by conduction and radiation with a parabolic velocity profile in the film. Motion of the denser fluid core is not directly involved in the model. Thus, the classic Bromley correlation neglects possible disturbance of the vapor-liquid interface, turbulence within the film, and acceleration of both denser fluid and film. These phenomena would enhance the heat transfer above that predicted by Bromley.

Current literature(9)(10) identifies various refinements of the original Bromley model. The equation version incorporated into RELAX is termed herein the Modified Bromley Correlation(11). A wide range of data is available to verify this Modified Bromley Correlation for low flow film boiling(9)(10)(11). Typical of the experiments conducted, with water as the medium, is the single rod quench tests performed by Leonhard, et.al.(11) as shown in Figure 2.2.

Bundle experiments FLECHT program[12] for transient reflood conditions below the two-phase level confirm the applicability of the Modified Bromley Correlation. Typical FLECHT results are provided in Figure 2.3 which shows the conservatism of the Modified Bromley Correlation in reactor bundle configurations.

Full scale BWR 7x7 bundle data from the TLTA has been shown[11] to establish the conservatism of the "Modified Bromley" correlation for the "window" and "Post-Lower Plenum Flashing" film boiling periods in a JP-BWR blowdown. Figure 2.4 is a typical comparison of the TLTA evidence. It can be seen that the Modified Bromley correlation conservatively predicts the heat transfer under pool film boiling conditions.

Full scale BWR 8x8 bundle data from the TLTA has also been shown[13] to produce the same conclusions deduced for 7x7 bundles[11]. Figures 2.5 through 2.6 provide data[13][14] for the peak power test 6006, Run 3, at the 100 inch elevation. Using the highest thermocouple temperature reading at the 100 inch level within the Bundle[14], the minimum heat transfer coefficient at this elevation was deduced and is presented in Figure 2.5. The Bromley heat transfer coefficient was calculated using the same data for the period the 100 inch level was in pool boiling (covered) (see Figure 2.6) and is superimposed on Figure 2.5. This procedure was used also at the 120 inch elevation of the same test and is displayed in Figure 2.7. It is concluded that the Modified Bromley Correlation conservatively predicts the heat transfer for pool film boiling conditions.

Thus, the Modified Bromley pool boiling correlation has been verified by a wide range of laboratory type experiments as well as in experiments using JP-BWR geometries.

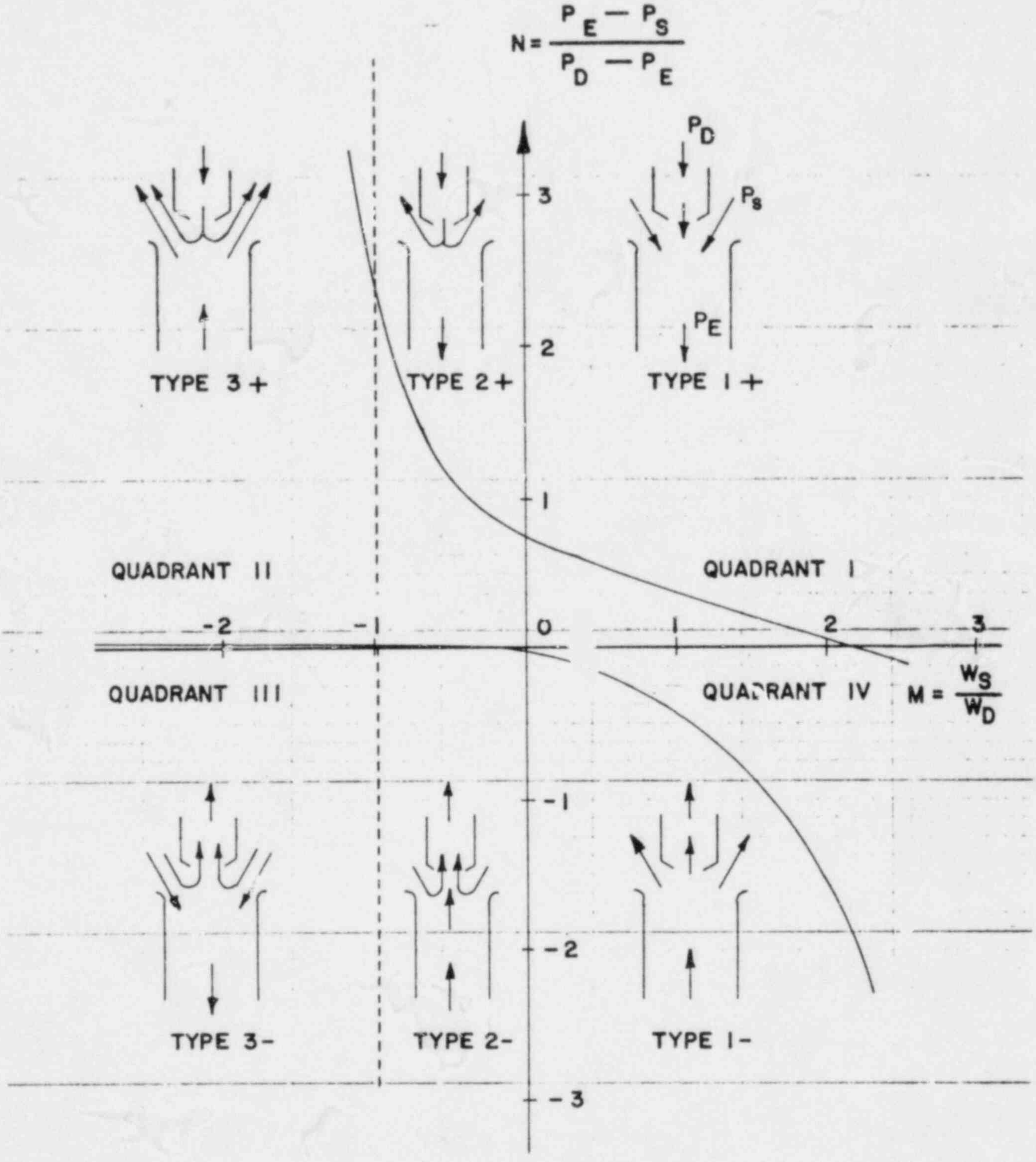


FIGURE 2.1 TYPICAL N-M JET PUMP CURVE WITH CLASSIFICATION OF FLOW TYPES.

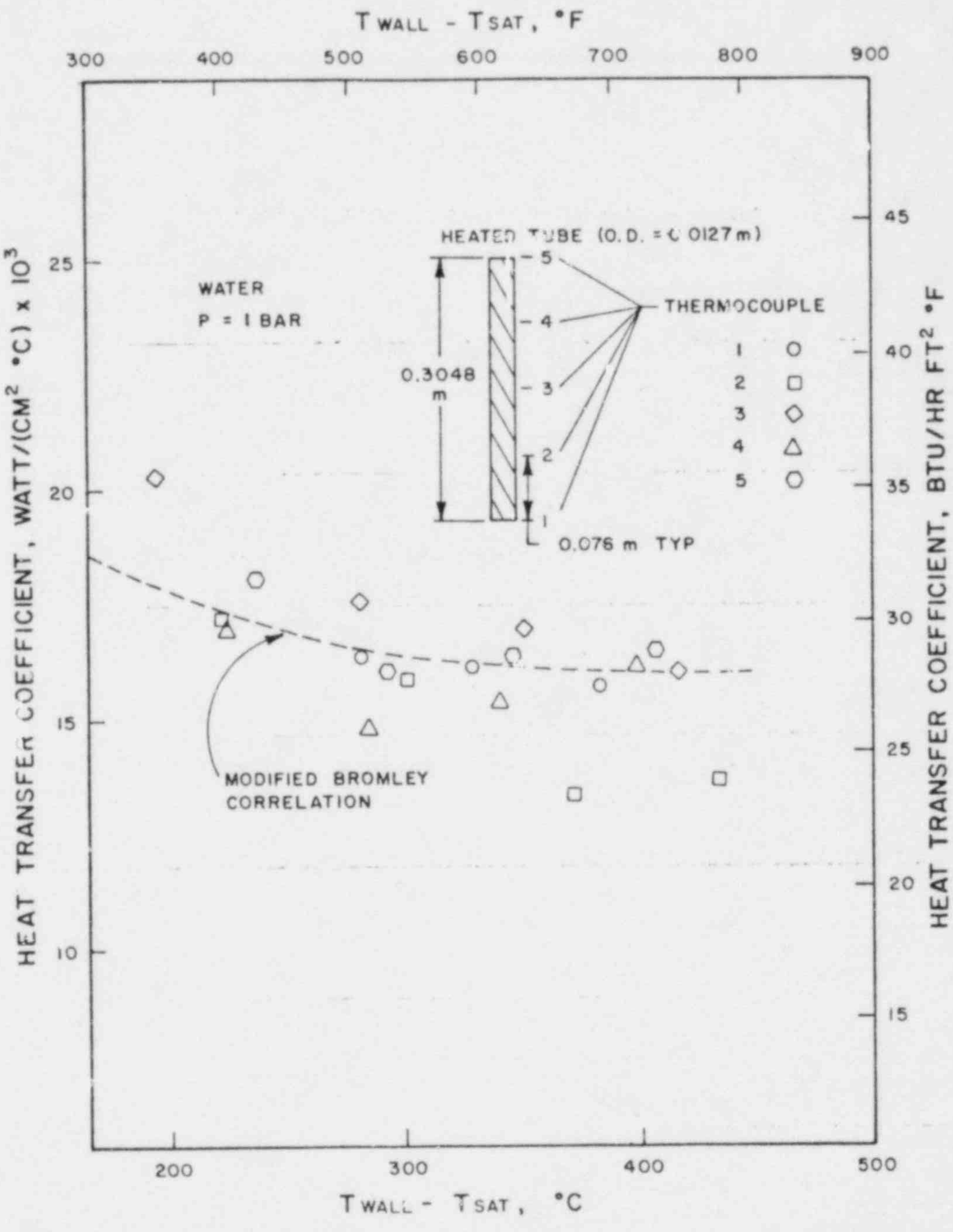


Figure 2.2 Single Rod Data Compared to Modified Bromley Correlation

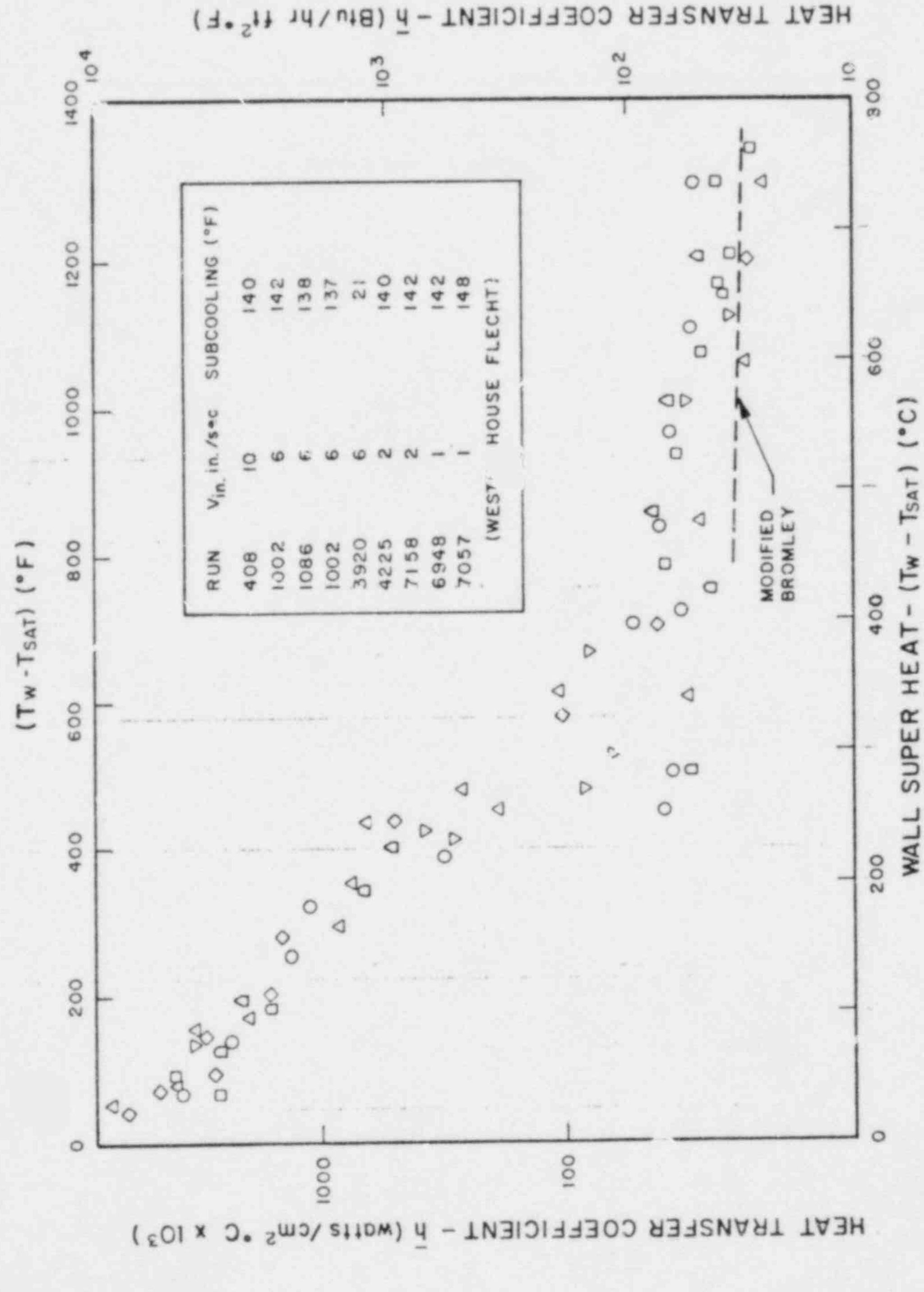


Figure 2.3 Heat Transfer Coefficient for Transition and Film Boiling at 60 psi

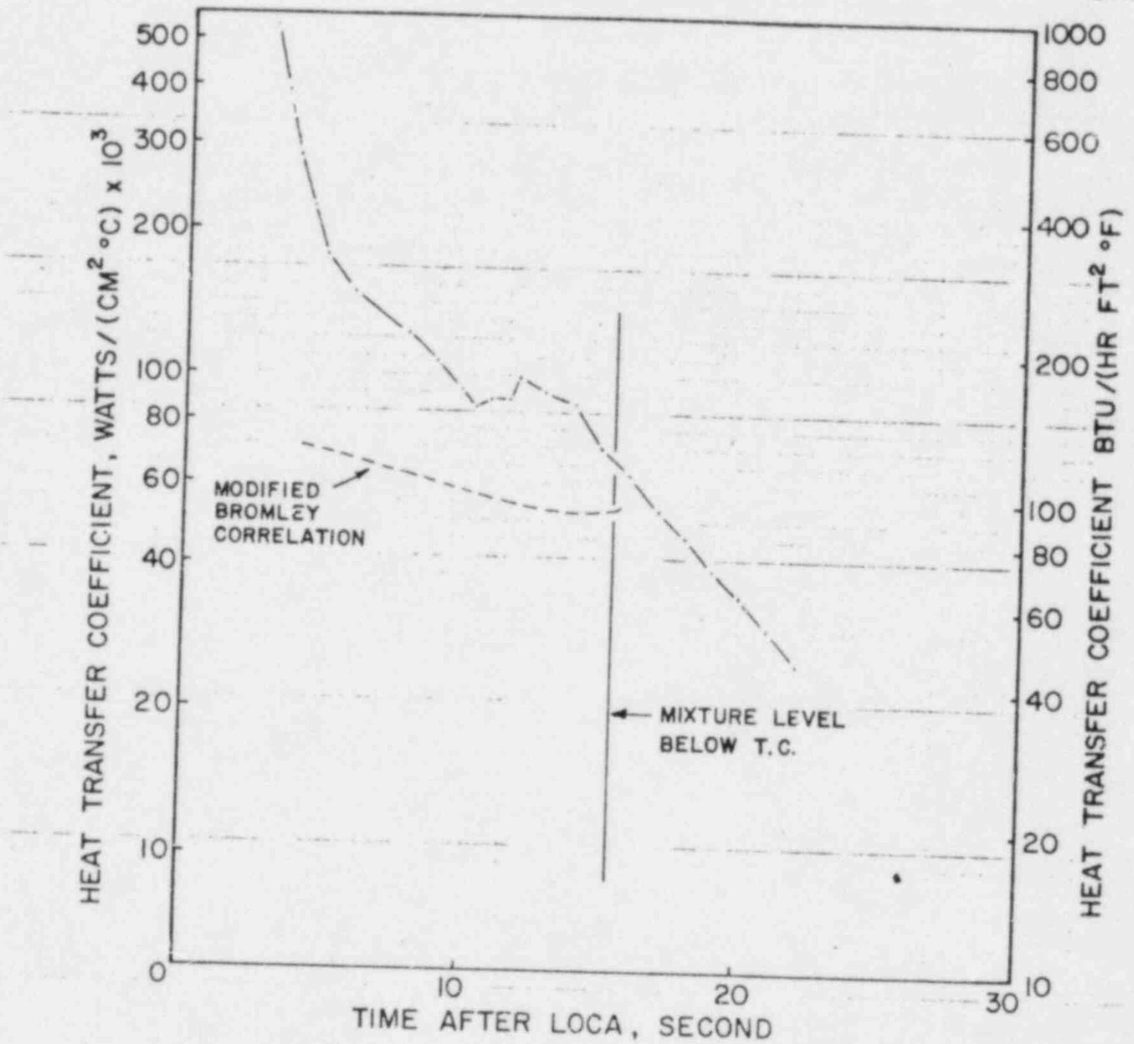


Figure 2.4 Post Boiling Transition Blowdown Heat Transfer for the 7x7 BWR TLTA Peak Power Bundle - Test 4907 Run 10

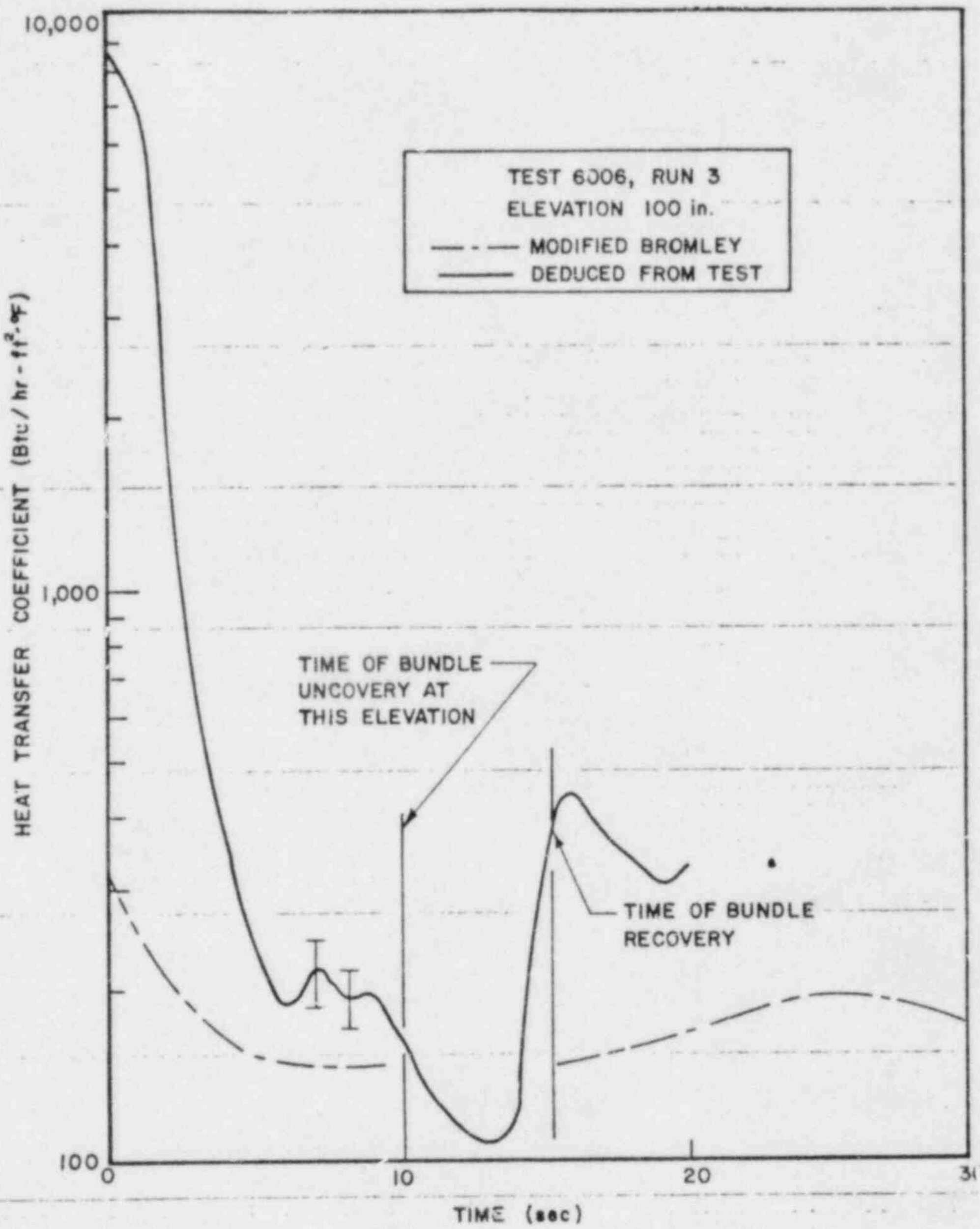


Figure 2.5 Deduced Heat Transfer Coefficient vs Modified Bromley Correlation from the 8x8 BWR TLTA Test 6006, Run 3

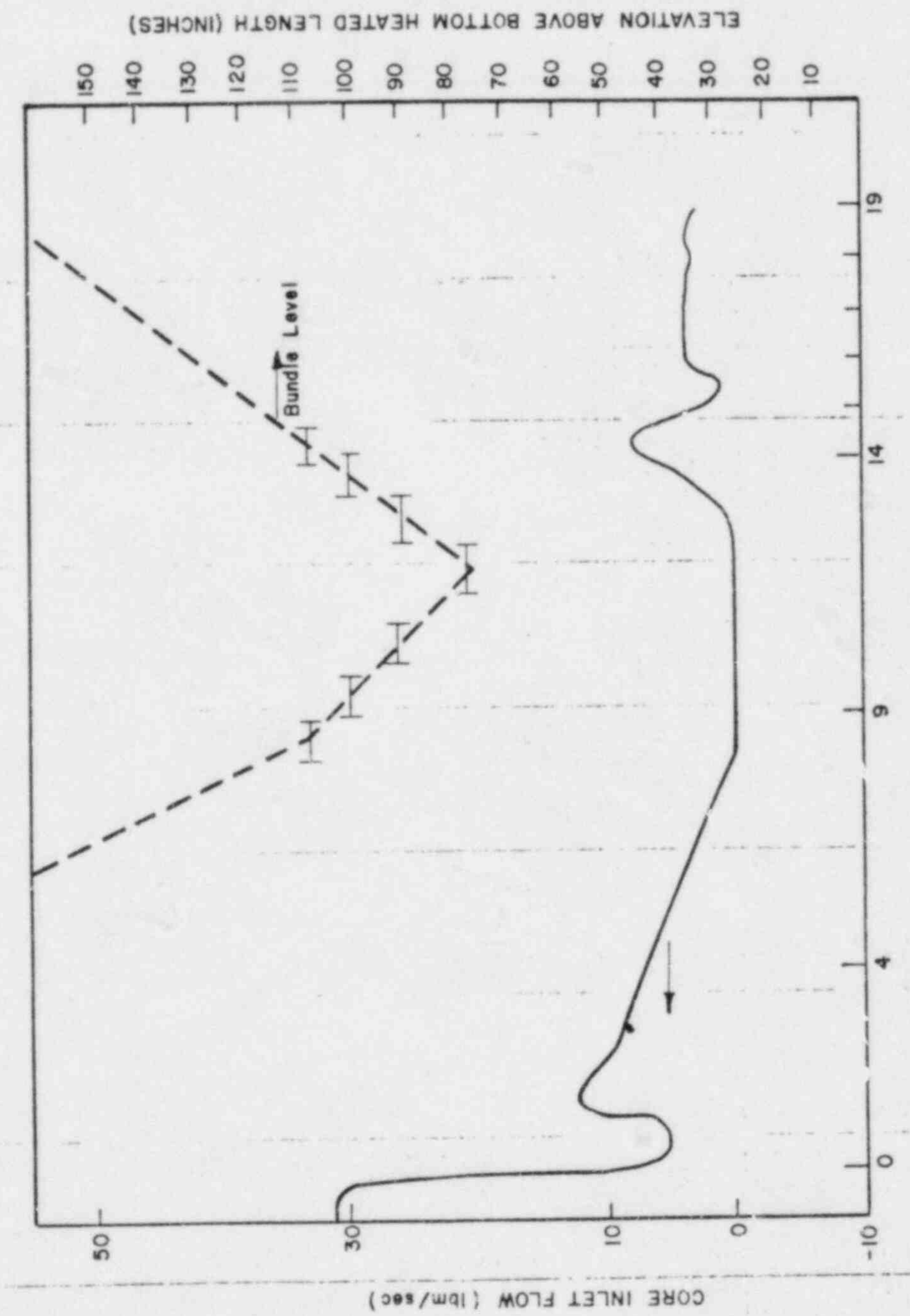


Figure 2.6 Comparison of Core Inlet Flow and Bundle Level for the 8x8 BWR TLTA Test 6006, Run 3

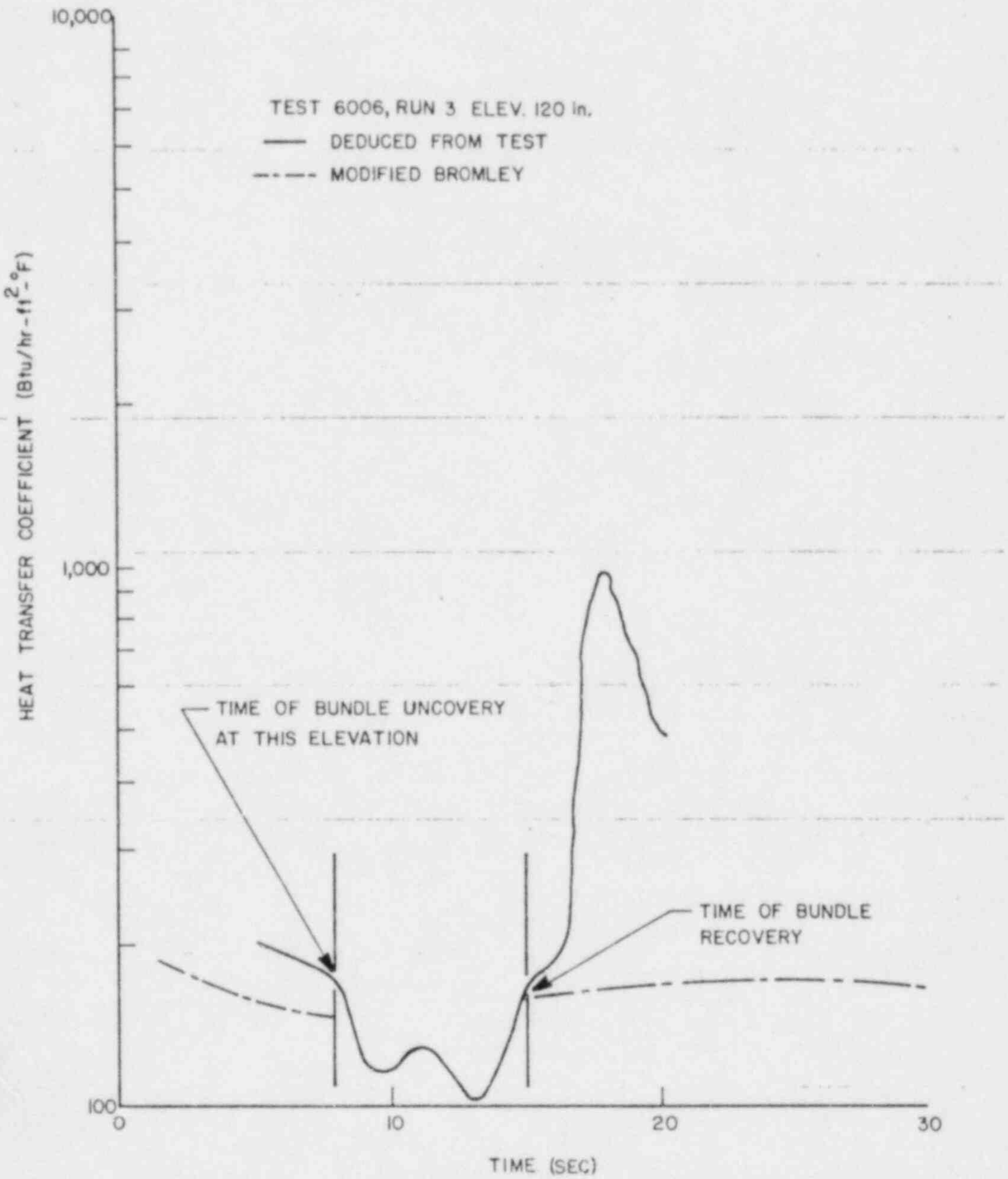


Figure 2.7 Deduced Heat Transfern Coefficient vs Modified Bromlcy Correlation from the 8x8 BWR TLTA Test 6006, Run 3

3.0 FLEX VERIFICATION

The FLEX code as described in Volume 2B of this Document has been developed by Exxon Nuclear Company to perform Evaluation Model licensing analysis during the refill and reflood phases of a hypothetical Loss-of-Coolant Accident (LOCA) in a Jet Pump Boiling Water Reactor (JP-BWR). The FLEX calculation begins at the time of rated Low Pressure Core Spray (LPCS) as calculated by RELAX, and continues until the time of hot node reflood cooling. The time of hot node reflood is used to determine when Appendix K reflood heat transfer coefficients are applied in the heatup analysis as performed by the HUXY code.

FLEX mechanistically calculates the system fluid inventory during the refill and reflood period. Counter-current flow through the fuel assembly and bypass regions is used to determine the spray penetration rate through the core and bypass and ultimately into the lower plenum. Critical flow models provide for the calculation of system mass depletion and depressurization rate.

In order to verify the FLEX code, detailed simulations comparing individual FLEX models and complete system transients to established codes, ENC refill and reflood data, and other test results have been conducted. A set of refill and reflood tests encompassing JP-BWR LOCA conditions has been run in ENC's Fuel Cooling Test Facility, described in Sections 3.1 and 3.2, to provide a data base for evaluation of the complete FLEX code. The results of the FLEX comparisons to this data are presented in Section 3.3.

Phase separation models have been incorporated into FLEX to provide an accurate account of system liquid inventories throughout the transient. The code uses the Wilson bubble rise model[1] or the drift flux phase separation model in calculating the void fraction below the mixture level. In order to verify the implementation of these models in FLEX, detailed comparisons to experimental swell tests[1] have been conducted. These swell tests are discussed in Section 3.4.

The critical flow and depressurization models in FLEX have been verified by comparison of FLEX results to the RELAP4[2] simulation of the same transient. RELAP4 is an NRC approved code which itself has been verified against extensive depressurization and critical flow data. The FLEX and RELAP4 depressurization and critical flow model comparison is presented in Section 3.5.

FLEX uses a model based on a five degree of freedom one-dimensional finite element solution of the heat conduction equation to determine the temperature profile and heat transfer in passive components. In order to establish the validity of the FLEX heat slab model, several transients have been evaluated using both FLEX and SINGLE[3], an established 21 degree of freedom finite element code. The results of this comparison are presented in Section 3.6.

3.1 FUEL COOLING TEST FACILITY DESCRIPTION

The Fuel Cooling Test Facility (FCTF) is designed to simulate the thermal hydraulic phenomena which would occur in a Jet Pump Boiling Water Reactor (JP-BWR) during the refill and reflood stages of a hypothetical Loss-of-Coolant Accient (LOCA). The facility consists of a pressure vessel with an internal steam supply system, a coolant injection system, a water treatment system, a mass balance system, interconnecting piping and valves, an AC power supply, and a data acquisition and process control system. The test vessel contains a full-scale electrically heated simulation of a production ENC JP-BWR 8x8 fuel assembly and channel.

FCTF instrumentation provides for measurement of the following parameters: bundle power; heater rod, channel, thermal shield and upper plenum temperatures; flow temperatures; selected pressure differences; and steam and liquid flow rates sufficient to determine a mass balance, including all injected flow rates, core exit steam flow, bundle and bypass penetration rates, and entrainment flow rate.

3.2 FUEL COOLING TEST PROGRAM RESULTS

A series of JP-BWR ECCS tests designed to encompass the conditions which may exist during the refill and reflood phases of a LOCA have been run in the FCTF loop. The test series included adiabatic counter-current flow limited tests, transient spray tests, and reflood tests. The effect of steam updraft rate, initial rod temperature, initial power, spray flow temperature, spray flow rate, and system configuration was determined on spray penetration into the bundle. This section summarizes the test trends and results. Section 3.3 provides the FLEX verification by comparing FLEX calculational results to this test data.

3.2.1 Counter-current Flow Limiting Tests

Counter-current flow limiting tests, conducted in an adiabatic quasi-steady state manner, were performed in order to determine the behavior of spray flow penetration as a function of steam upflow. All CCFL testing was conducted in a steam-first manner.

3.2.2 Transient Spray Tests

The transient spray test series simulated the refill portion of a LOCA transient in a JP-BWR.

3.2.3 Reflood Tests

All reflood tests were hot assembly tests with an initial bundle power of either 240 kw or 314 kw. The parameters of importance during a reflood test are the hot plane temperature and the time at which this temperature turns over.

3.3 FLEX CODE VERIFICATION AGAINST FCTF TESTS

The FLEX computer code has been developed by ENC to determine the thermal hydraulic response during the refill and reflood phases of a Loss-of-Coolant Accident (LOCA) in a Jet Pump Boiling Water Reactor (BWR). The FLEX code is used during the period of Low Pressure Core Spray (LPCS) injection into the reactor system to the time of hot node reflood. This includes the refilling of lower plenum with liquid and reflooding of the core with liquid sprays.

The FLEX code calculates the time of hot node reflood, allowing for counter-current flow phenomena and carefully accounting for the liquid inventory throughout the reactor vessel. ECCS water that is sprayed into the upper plenum is calculated by counter-current flow to fall through the core or bypass regions into the lower plenum. Phase separation and entrainment models are included to accurately account for liquid inventories in the reactor vessel. Break models calculate the inventory loss and depressurization rate throughout the ECCS spray period.

The FCTF tests were conducted to provide a data base for the evaluation and verification of FLEX models. Detailed simulations of several of the FCTF tests have been performed using the FLEX code for code verification purposes. The tests selected for the verification task are listed in Table 3.1.

For each of the tests simulated, FLEX was initialized from the actual test conditions. Rod and channel temperatures, lower plenum level, and any injected flow rates were specified as initial or boundary conditions as required. All FCTF tests were conducted at atmospheric pressure.

These FLEX-FCTF comparisons verify the applicability of the FLEX code for JP-BWR ECCS refill/reflood licensing analyses. In terms of the most important comparison for the spray tests, the lower plenum liquid level, FLEX predictions are in excellent or conservative agreement. Furthermore, vapor generation rates, rod temperatures and time of channel quench compare favorably with the data for spray tests.

3.4 VERIFICATION OF FLEX PHASE SEPARATION MODELS (WILSON AND DRIFT FLUX) TO THE G.E. LEVEL SWELL TEST

The Wilson and Drift Flux phase separation models incorporated into FLEX have been compared to two experiments from the G.E. level swell test program(1) to provide verification of the models as implemented in FLEX and to serve as a base case comparison between these two models.

Saturated liquid, at approximately 6.9 x 10^6 Pa (1000 psia), partially fills the vessel. The rupture disk is broken at the beginning of the test and the vessel is allowed to blow down. Primary measurements from the experiment are mixture level, pressure and mass as a function of time.

3.5 VERIFICATION OF FLEX DEPRESSURIZATION AND CRITICAL FLOW MODELS

In order to provide verification for the depressurization and critical flow models used in FLEX, a blowdown run was made using the hypothetical system shown in Figure 3.1. These results were then compared to the RELAP simulation of the same system transient. The agreement between the RELAP and FLEX depressurization curves is then an indication of the validity of the depressurization and critical flow models used in FLEX.

RELAP4(2) is a well established code developed by INEL for blowdown analyses. It has been checked against many experiments involving depressurization and critical flow, and ENC's latest version used herein (RELAP4/ENC28) has been approved for EM analyses.

The one-loop, six-volume nodalization used for this task is pictured schematically in Figure 3.1. The same nodalization was used in both the RELAP and FLEX runs, and the codes were initialized identically.

The components and dimensions are typical of those which would be used for a plant licensing analysis. However, this system was simplified as much as possible in order to make a true comparison of depressurization and critical flow models only. Node 1 represents a reactor core, although in both FLEX and RELAP there was assumed to be no heat addition from the decay power. Node 2 represents an upper plenum, and Node 3 the steam dome. Nodes 4 and 5 simulate a downcomer and jet pump volume, respectively, and Node 6 is a lower plenum.

The results of the RELAP and FLEX analyses are presented in Figures 3.2 and 3.3. The steam dome pressures (Node 3) are shown in Figure 3.2. The FLEX results follow the RELAP depressurization closely. At the point of largest deviation, the two curves are apart by less than 5%. The break flow comparison is shown in Figure 3.3. The agreement between the RELAP and FLEX results is excellent again (well within 5%). In both cases, the Moody model, with a discharge coefficient of 0.6, was used to calculate choked flow. Since both the depressurization and critical flow models were dominant in these analyses, this comparison provides a verification of both the FLEX depressurization and critical flow models.

3.6 FLEX HEAT TRANSFER CONDUCTION MODEL VERIFICATION

The slab heat conduction model used in FLEX is a one-dimensional finite element solution of the heat conduction equation based on the Method of Weighted Residuals (MWR) and employs the Galerkin Approach[4]. The model has five degrees of freedom and can treat two different materials. Within each of the two regions, a quadratic temperature profile is used.

The general one-dimensional heat conduction equation solved is

$$\frac{\partial}{\partial x} \left(k \frac{\partial T}{\partial x}\right) + p''' = \rho c \frac{\partial T}{\partial t}$$

which is subject to the following boundary conditions:

$$T\ (t=0,x) = f(x)$$

$$-k \frac{\partial T}{\partial x} (x=0,t) = h\ (x=0,t)(T_\infty\ (x=0,t) - T\ (x=0,t))$$

$$k \frac{\partial T}{\partial x} (x=L,t) = h\ (x=L,t)(T_\infty\ (x=L,t) - T\ (x=L,t)).$$

A comparison study has been performed between FLEX and an established 21 degree of freedom finite element reference code(3) for the purpose of confirming the validity of the conduction heat transfer model in FLEX. The first two cases compare the results of analyses for two TLTA heat slabs, and the third case is a hypothetical situation chosen because it is a difficult transient to simulate with a five degree of freedom model.

Case 1 represents a heat slab used in the FLEX analysis of TLTA Test 6007. This heat slab models a stainless steel wall below the mixture level in the lower plenum. The dimensions and heat transfer parameters used are listed in Table 3.2. The results of this comparison are shown in Figure 3.4. The heat slab temperatures predicted by FLEX correspond exactly to the results of the reference code throughout the 25 second transient.

The heat slab modeled in Case 2 represents the electrically heated rods in the core region in the FLEX simulation of TLTA Test 6007. Case 2 thus differs from Case 1 in that Case 2 has a low thermal conductivity, a volumetric heat source, and a low surface heat transfer coefficient. The actual parameters are listed in Table 3.3. The results of this comparison are presented in Figures 3.5 through 3.8. These heater rods are directly heated (current is supplied directly to clad), so the interior of the heater rods does not exhibit the normal parabolic temperature profile of a nuclear fuel rod. The FLEX results are quite reasonable here, and very good agreement is obtained with the surface temperature. The maximum error in surface heat flux for this problem was found to be less than 2%.

Case 3 modeled the problem shown in Figure 3.9. This is a particularly difficult transient for a five degree of freedom model to approximate since the body sees a step change in the convective environment which tends to decrease the body temperature, while at the same time internal generation is trying to increase the body temperature. The following three figures give a comparison of the FLEX conduction model and the 21 degree of freedom solution for this problem. FLEX results are denoted by a Δ symbol and the reference solution values by a o symbol. No lines are drawn on the figures as none of the results can be represented by linear segments. FLEX results are quadratic in each region.

The first figure (3.10) gives the temperature profile through the slab at t=0.5 seconds. The second figure (3.11) gives the surface temperature (at x=0) as a function of time, and the third figure (3.12) gives the energy content in the slab as a function of time, where

$$\overline{T} = \frac{\int_0^L T\,dx}{L} \quad .$$

As shown, the results of the FLEX conduction model are in good agreement with the reference solution even for this challenging problem. Thus, the FLEX heat slab solution has been verified through comparison to accurate transient solutions for typical BWR inert heat slabs, core heat slabs, and a classic application.

Table 3.1 FCTF Tests Chosen for FLEX Simulation (Saturated Injections)

Table 3.2

HEAT SLAB MODEL VERIFICATION

CASE 1 INPUT

ρ = 8030 Kg/m^3

C = 381. Joules/Kg°C

K = 18.9 watts/M°C

δ = 0.00329 M

h_1 = 5680 watts/M^2°C (1000 Btu/hr ft^2°F)

h_2 = 0.0

T_{sat} varies in time from 194.5 to 157.1°C

Table 3.3

HEAT SLAB MODEL VERIFICATION

CASE 2 INPUT

ρC_p = 3.296×10^6 Joules/M^3°C

K = 1.71 watts/M°C

δ = 0.0613 M

q_1''' = 607274 watts/M^3 (volumetric heat source in heater cladding)

h_1 = 28.4 watts/M^2°C (5 Btu/hr ft^2°F)

h_2 = 0.0

T_{sat} varies in time from 194.5 to 157.1°C

Table 3.4

HEAT SLAB MODEL VERIFICATION

CASE 3 INPUT

ρ = 8016.4 kg/m^3

c = 418.7 j/kg °C

k = 8.65 j/m-sec °C

L = 0.122 m

q_1''' = 103497.0 w/m^3

h_1 = 567.83 j/m^2 sec °C

h_2 = 0.

T_∞ = 21.11 °C

T(x,t=0) = 76.67°C

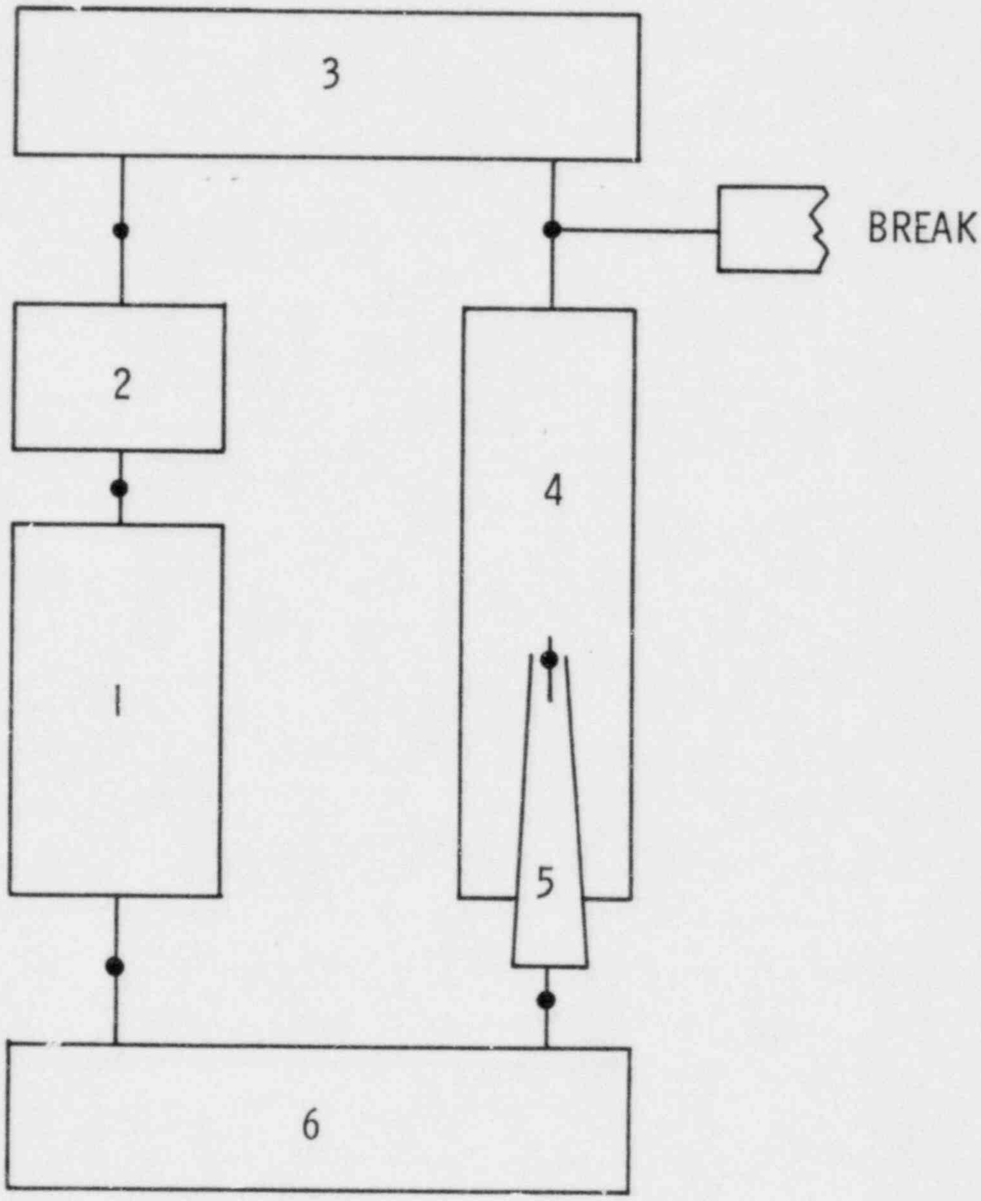


Figure 3.1 System Nodalization Used for FLEX Verification to RELAP4 Depressurization and Critical Flow Models

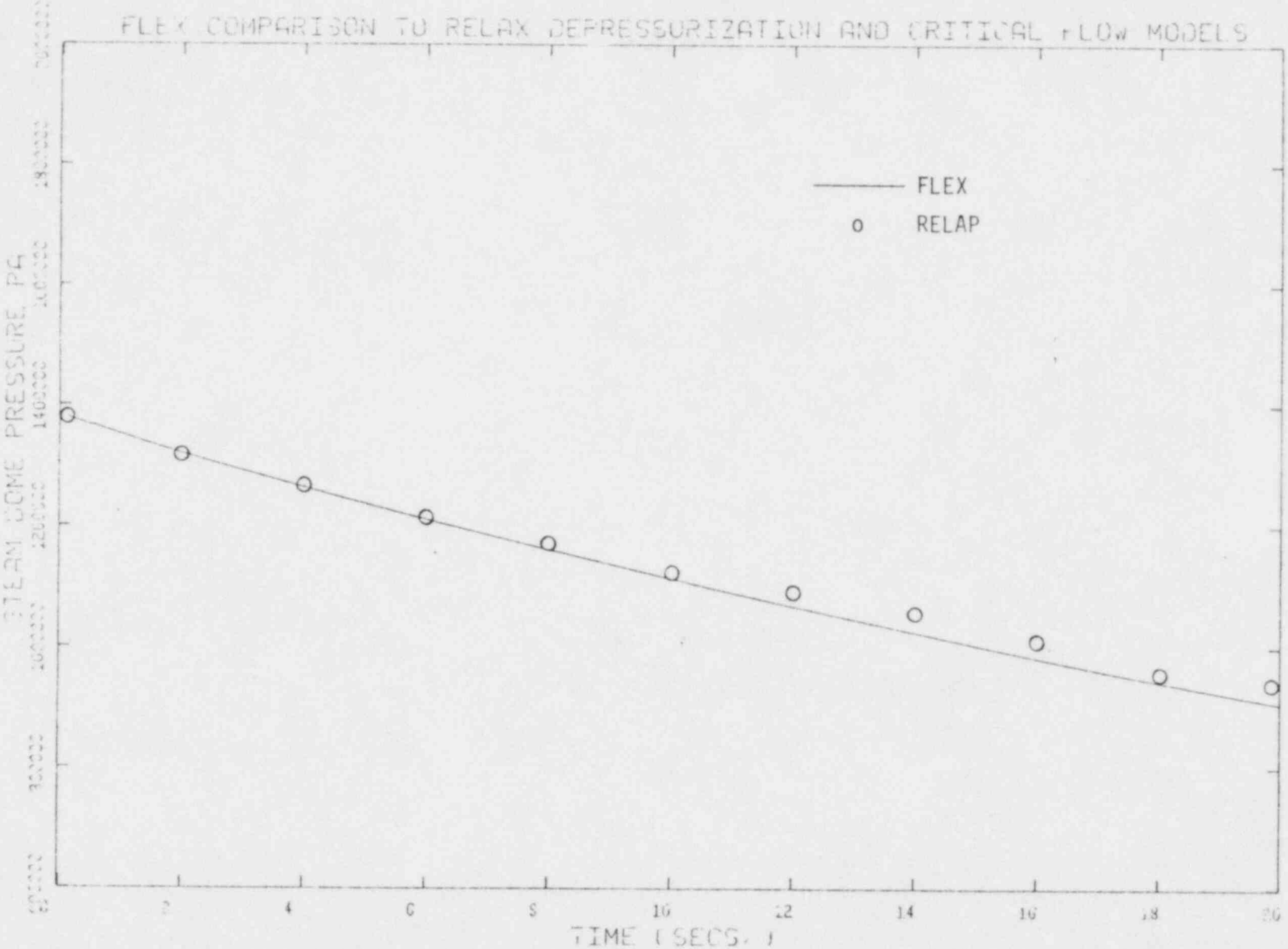


Figure 3.2 FLEX to RELAP4 Steam Dome Pressure Comparison

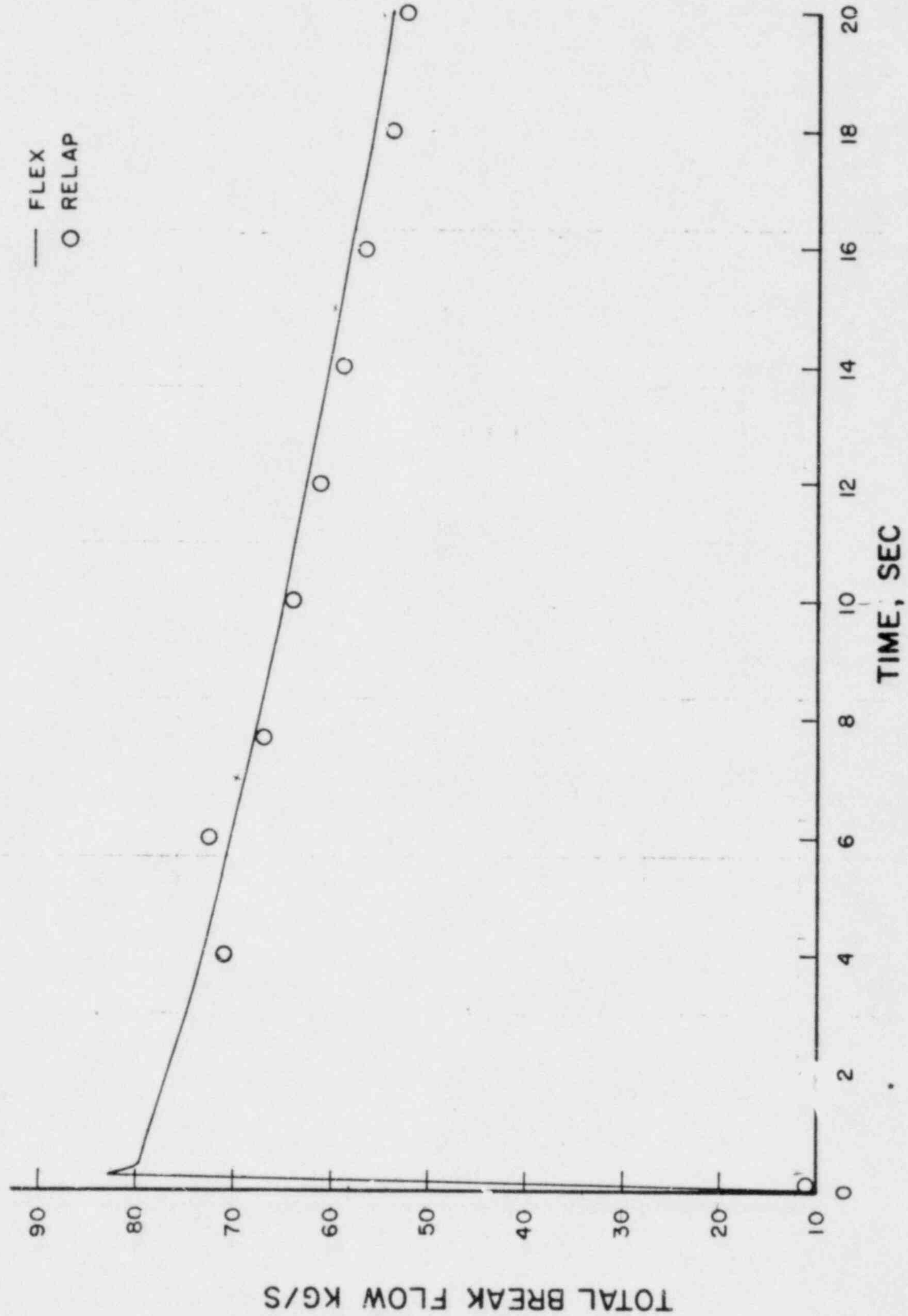


Figure 3.3 FLEX to RELAP4 Break Flow Comparison

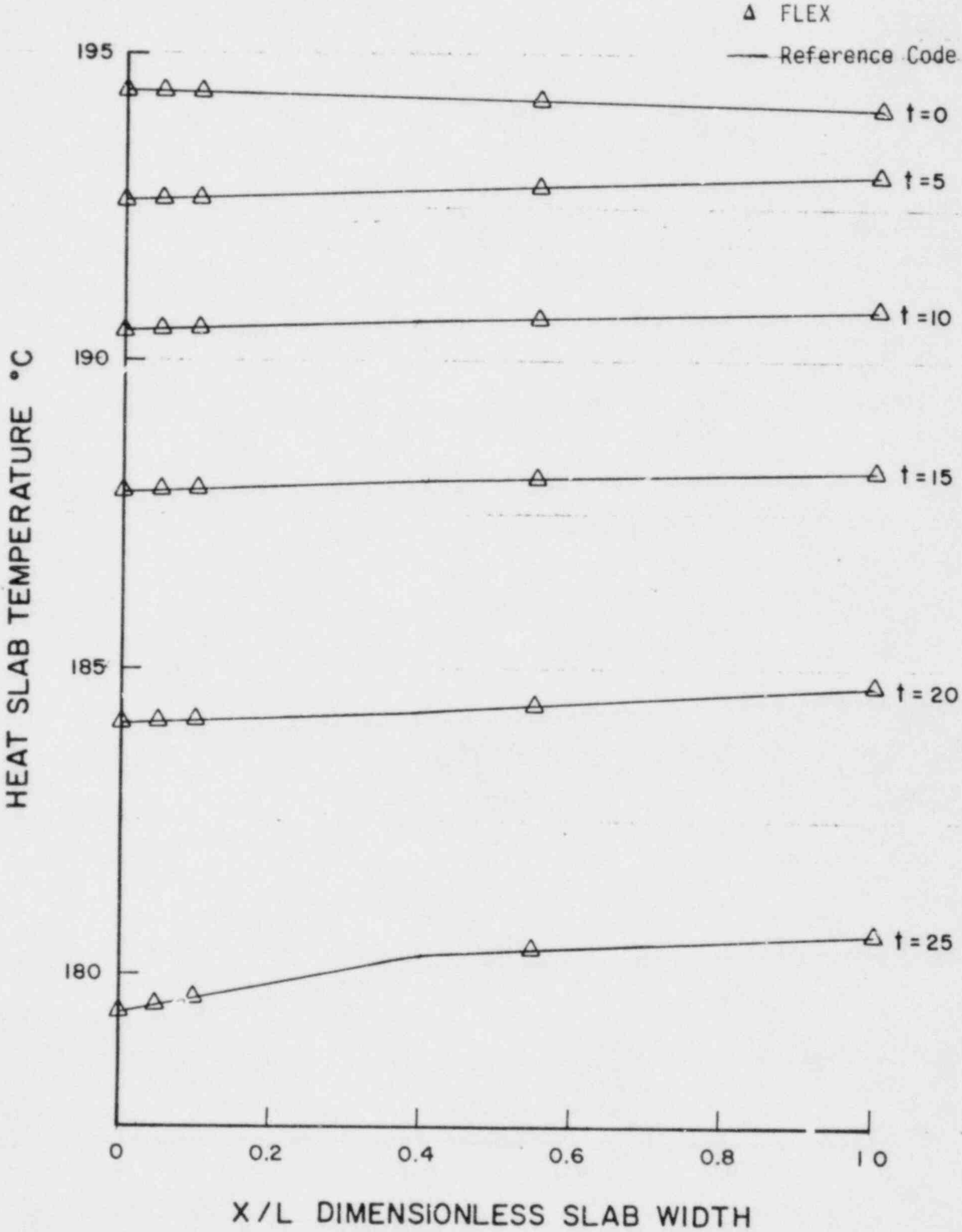


Figure 3.4 Case 1 Heat Slab Temperature Transient

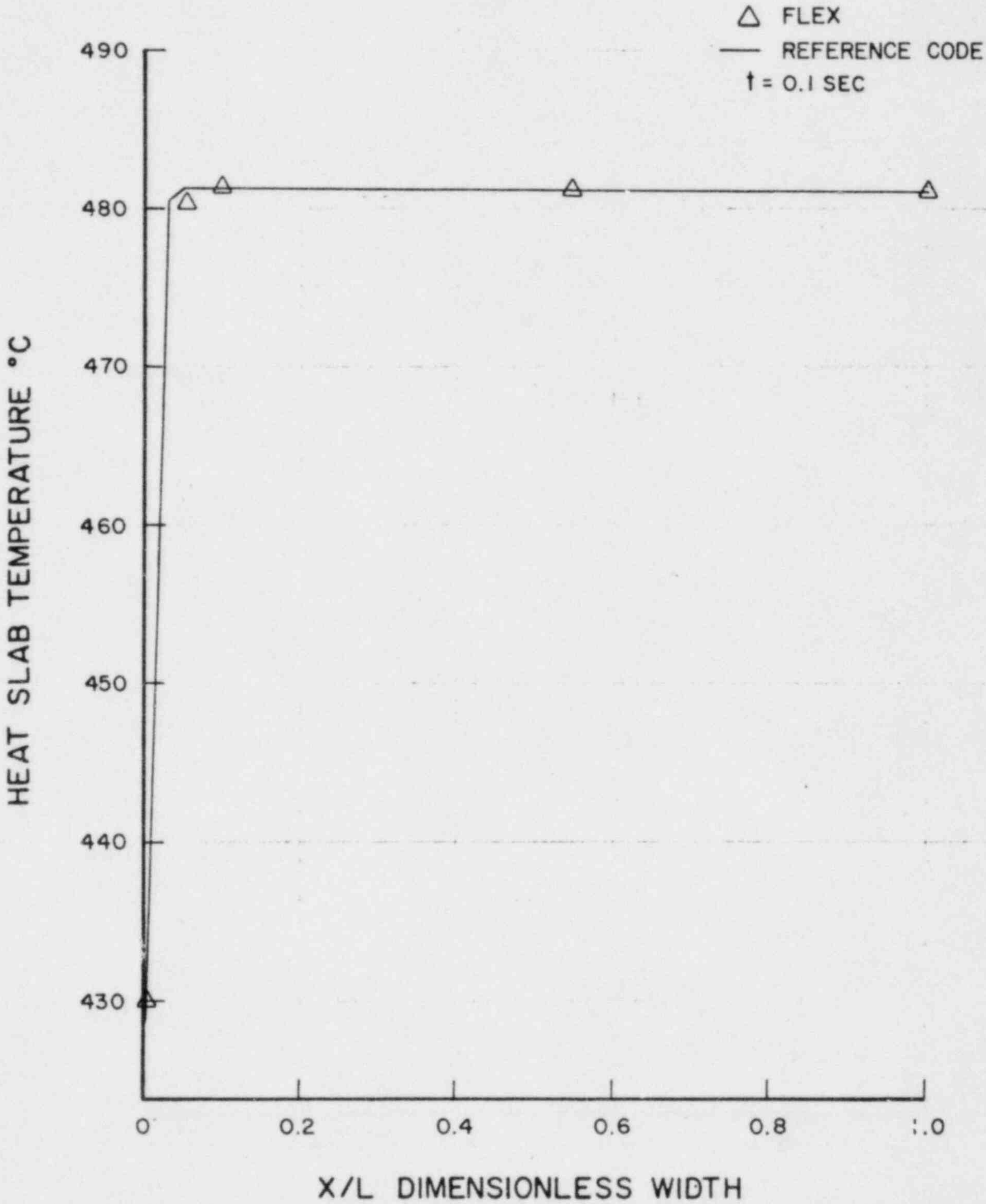


Figure 3.5 Case 2 Heat Slab Temperature Profile at 0.1 Seconds

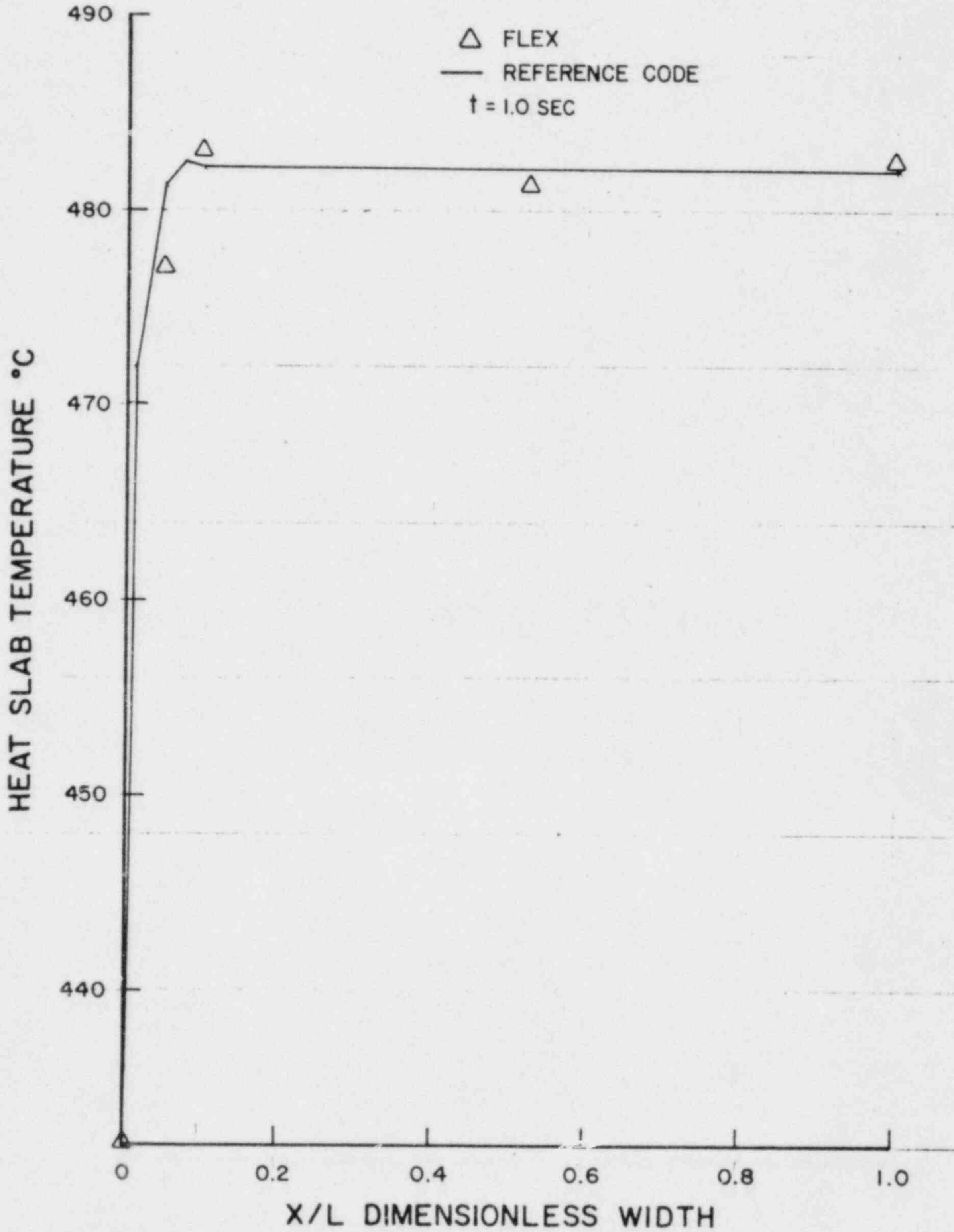


Figure 3.6 Case 2 Heat Slab Temperature Profile at 1.0 Seconds

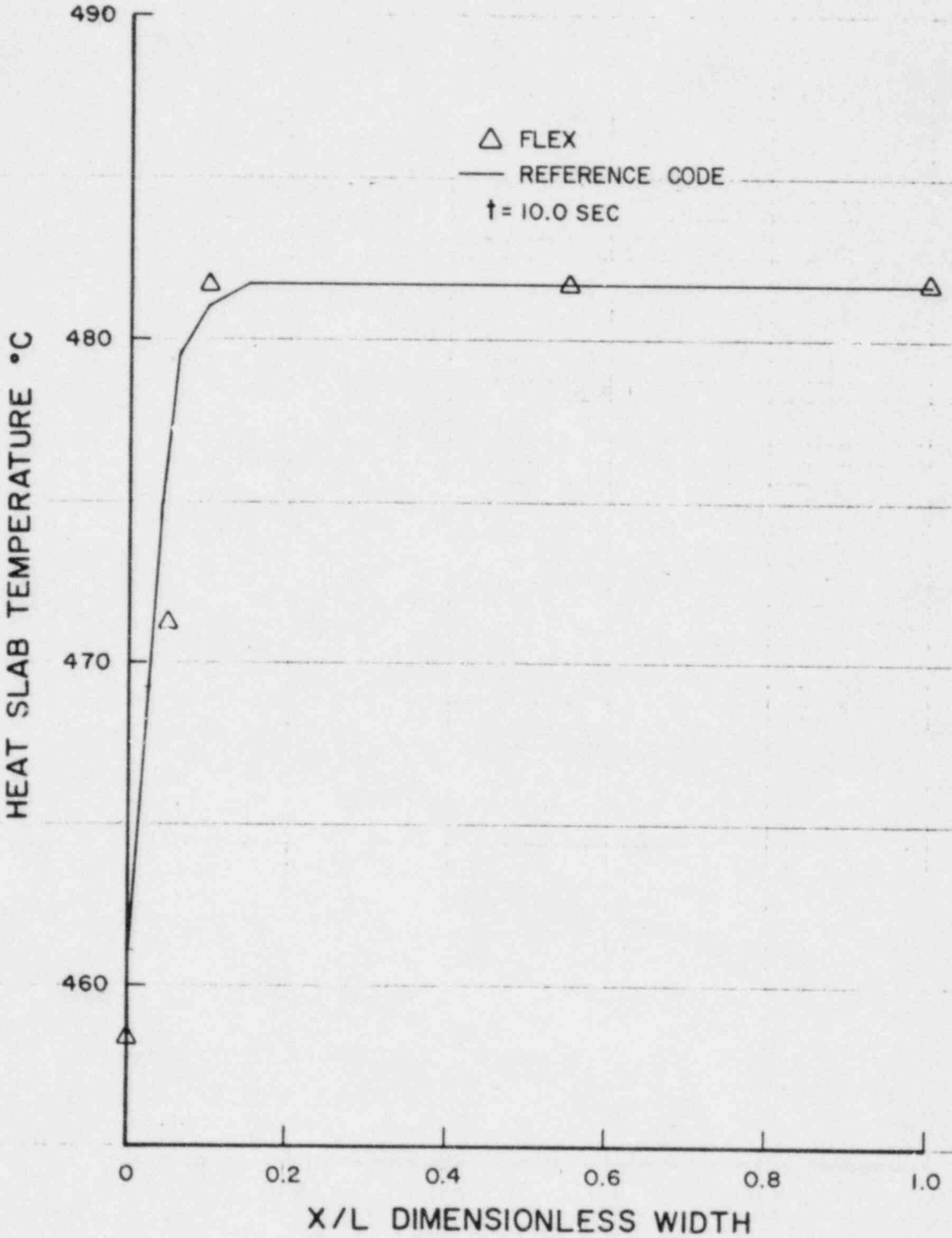


Figure 3.7 Case 2 Heat Slab Temperature Profile at 10.0 Seconds

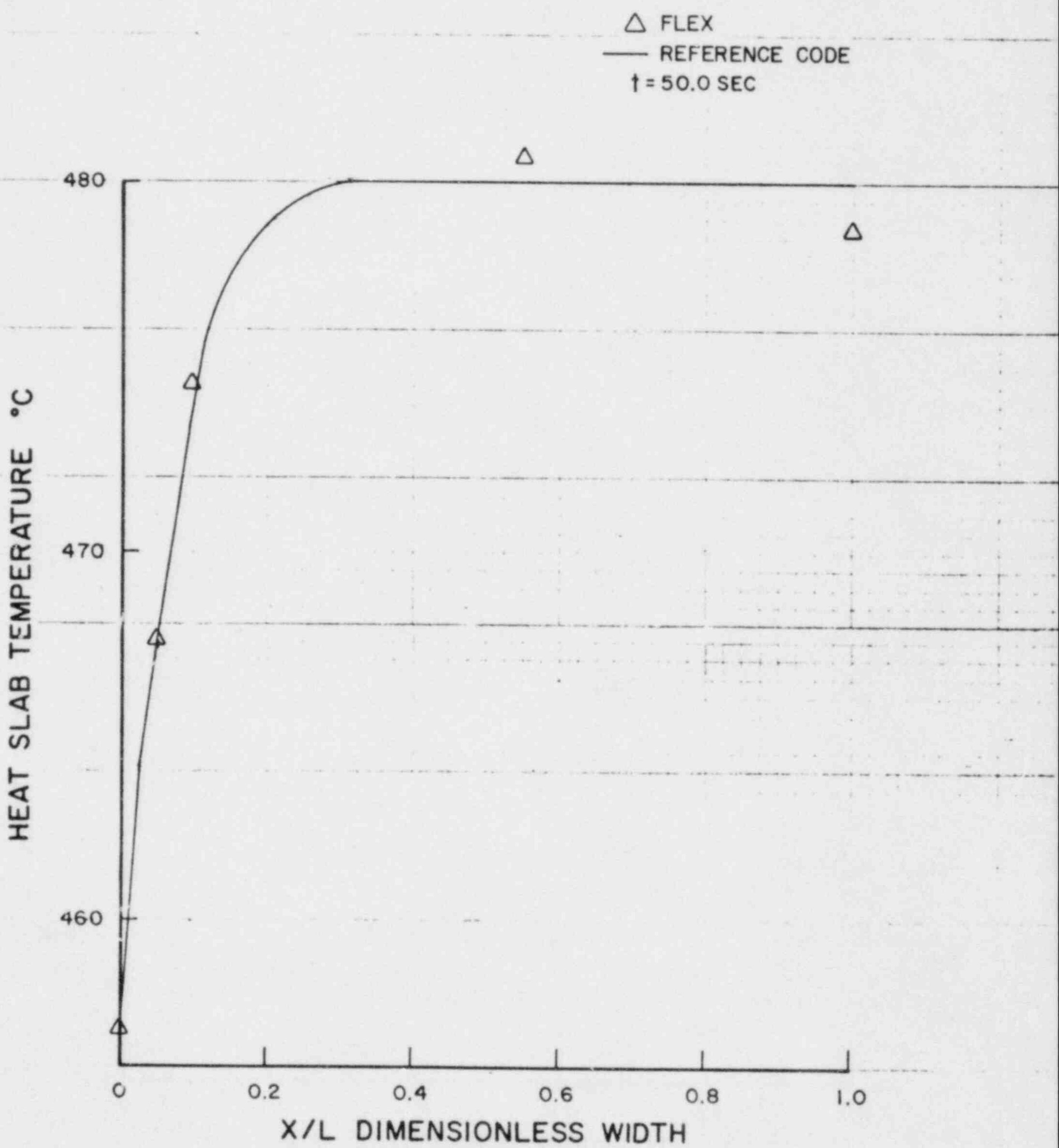


Figure 3.8 Case 2 Heat Slab Temperature Profile at 50.0 Seconds

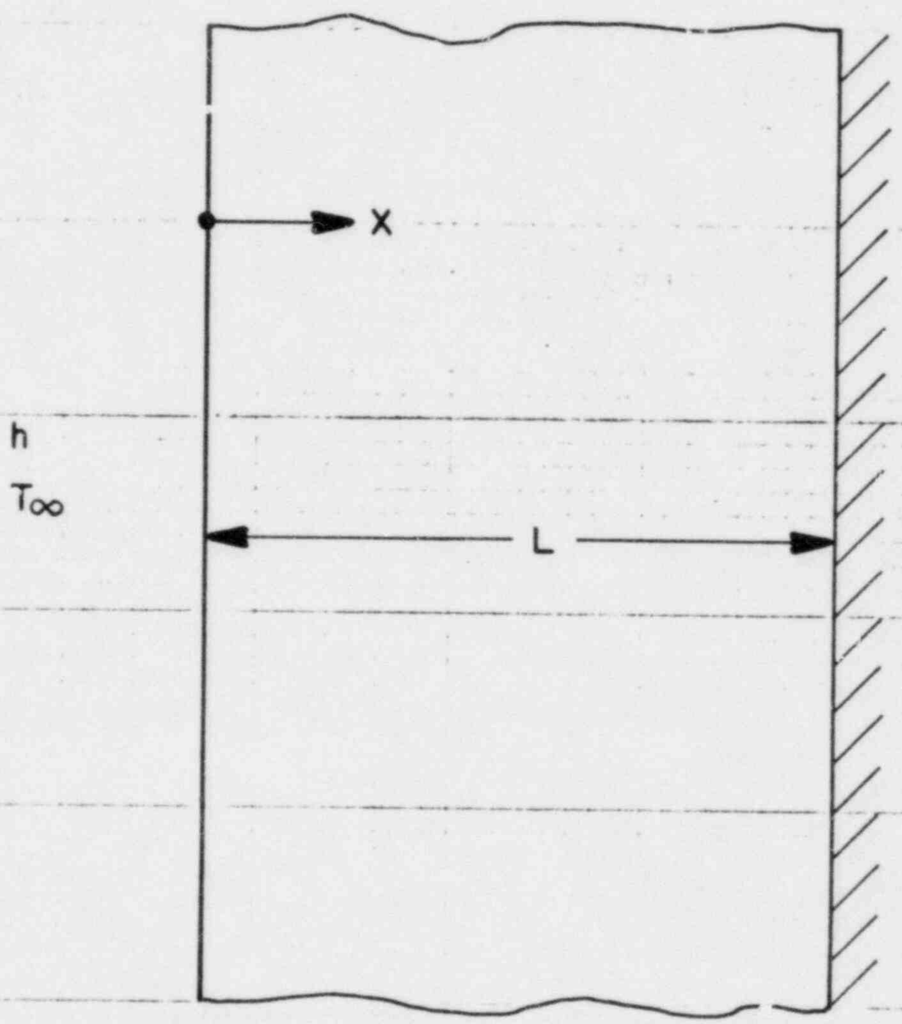


Figure 3.9 Schematic of Heat Slab Used for Case 3

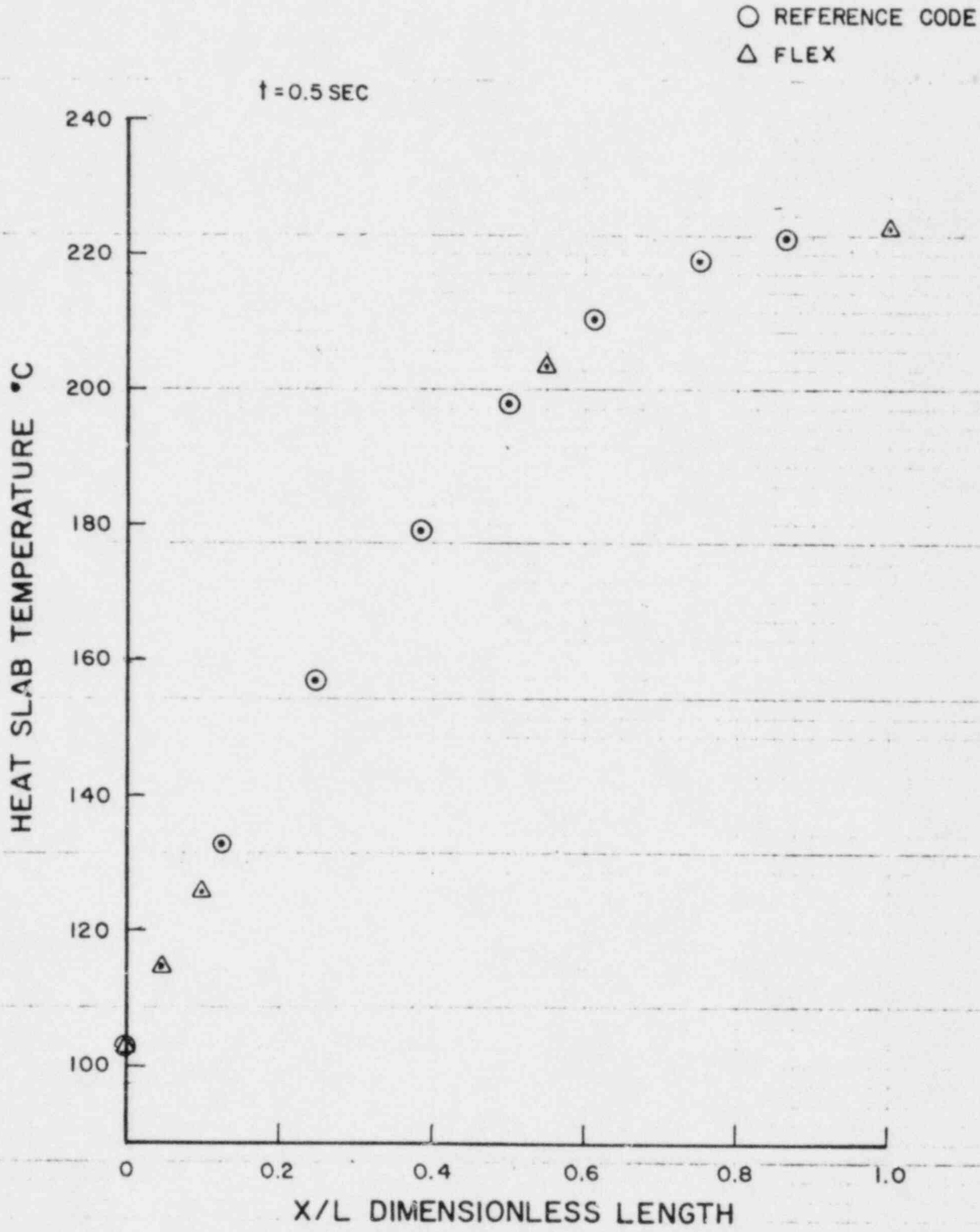


Figure 3.10 Case 3 Heat Slab Temperature Profile at 0.5 Seconds

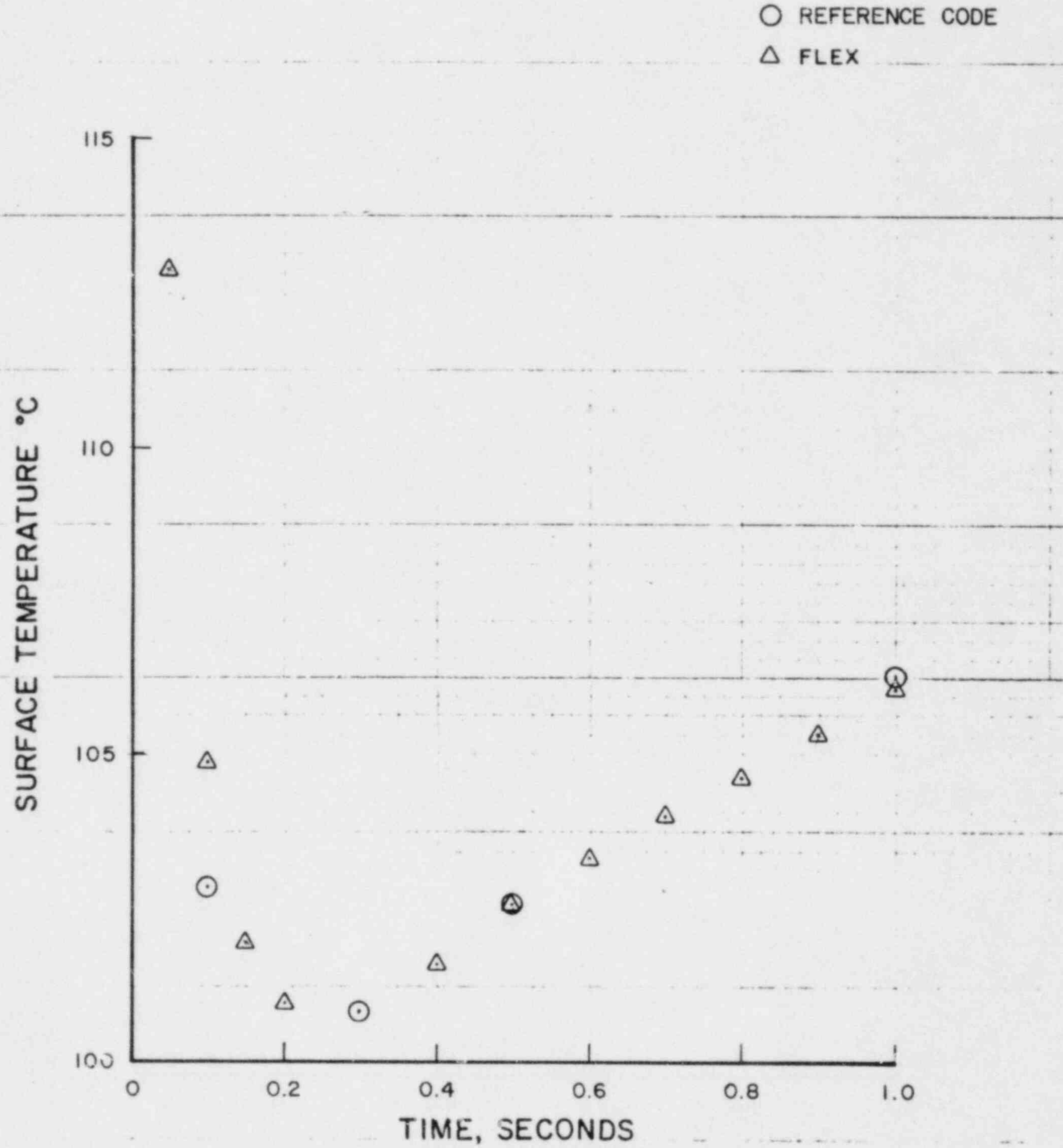


Figure 3.11 Case 3 Heat Slab Surface Temperature Transient

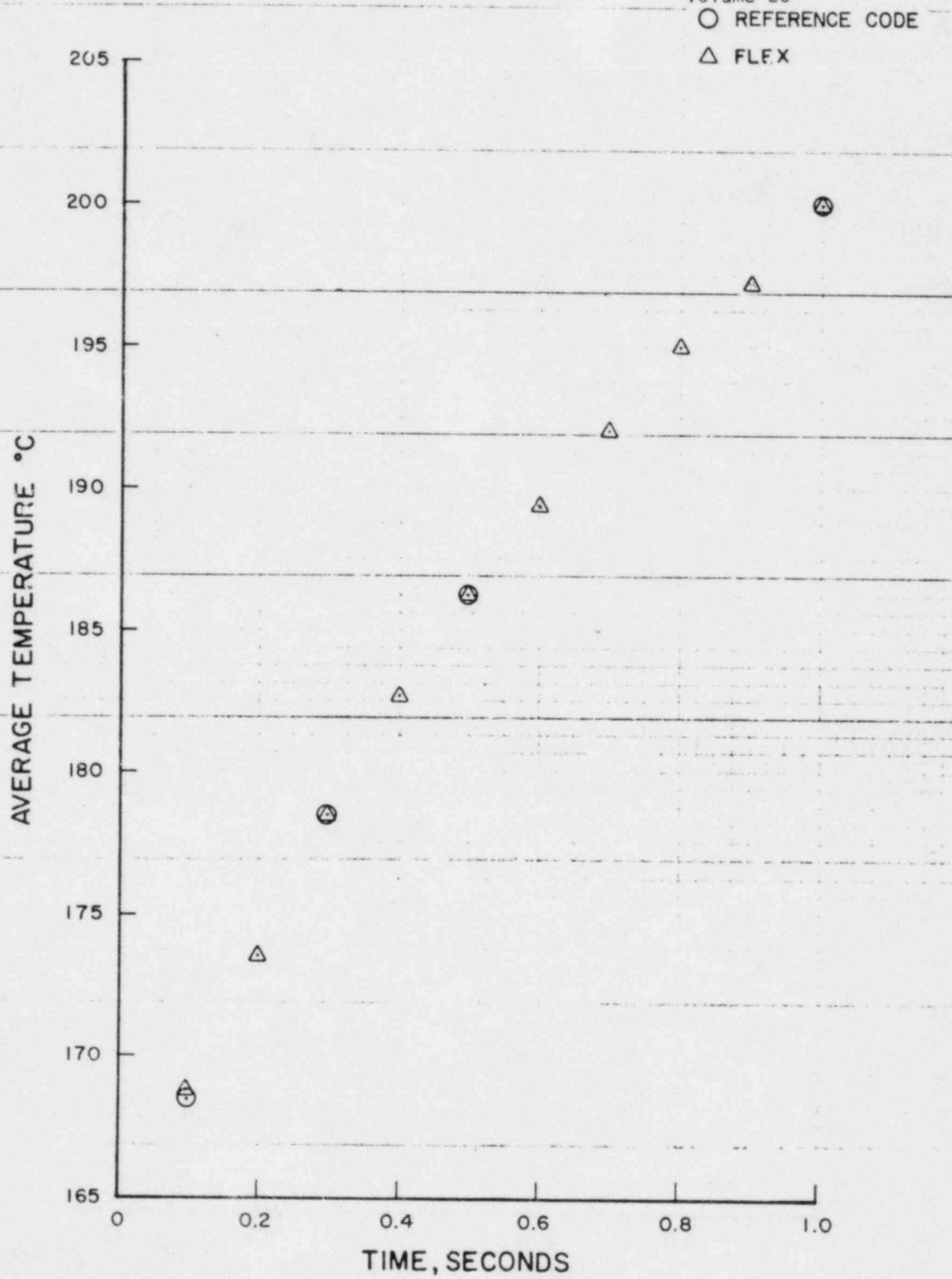


Figure 3.12 Case 3 Heat Slab Average Temperature Transient

4.0 EXEM QUALIFICATION TO EXPERIMENTAL BWR SYSTEM BEHAVIOR (TLTA)

This section presents the results of the qualification effort of the overall EXEM model to the integral test data from the JP-BWR System Facility, referred to as TLTA (Two-Loop Test Apparatus)[1].

Two key tests in the TLTA test series are #6007 and #6406. These tests were run with a facility configuration representative of the latest JP-BWR plants. Test #6007 was run without ECCS injection, and #6406 was run in a similar manner to 6007 but with the ECCS injections. The TLTA jet pumps are significantly shorter than in a plant which prevents reflooding of the core in the TLTA tests; thus, this comparative study extends through the blowdown and refill periods to the start of reflood.

The EXEM calculations were made in the Evaluation Model (EM) mode, except that actual core power which was simulated electrically in the tests was input and a best estimate break flow multiplier was used. Use of actual core power removes one of the major conservatisms in predicted clad temperature during blowdown (the 20% addition to the ANS decay power); thus, only a minor conservatism in the predicted versus measured clad temperatures during blowdown would be anticipated. Since licensing calculations include a break size spectrum, the use of a multiplier other than unity at the break is justified in a qualification calculation.

RELAX was used to time-of-rated LPCS. FLEX was initialized from the RELAX calculation and run about 100 seconds into the spray period to the start of reflood or until the power was tripped in the test. Thus, the qualification calculations were performed consistent with the plant licensing calculations except for the removal of the 1.2 multiplier on decay power.

The TLTA test data plotted in the following figures are those reported in References (2) and (3), as supplemented by data from publicly available tapes[4,5].

RELAX RESULTS

Figure 4.1 shows the RELAX nodalization diagram for TLTA for blowdown calculations. This nodalization was used on both #6007 and #6406 and is the same as is being used on plant analyses except for areas unique to TLTA; i.e., nodalization associated with the shortened TLTA jet pumps and the TLTA break. The simulated break is in the discharge piping of the recirculation loop which is the right hand loop of the figure.

The blowdown (RELAX) results will be discussed in the following order: Test 6007, Test 6406, and finally a relative comparison of these two tests to accent the effect of ECCS injections. Only the most important parameters (system pressure, jet pump flows, core flows, liquid retention, and clad temperatures) are presented. The calculated parameters are presented in the same RELAX output plot format as in plant analyses. For parameters and periods for which TLTA data is reported, the data is included on the figures.

4.1.1 RELAX Simulation of TLTA Test 6007

The calculated and measured system pressure responses in the steam dome are shown to be in good agreement in Figure 4.2. The rapid decrease in pressure, around 10 seconds in the measured and calculated pressure, is a result of the break uncovering when the downcomer water level falls below the suction line with the corresponding increase in volumetric flow out of the break. The calculated pressure tends to be below the measured value near the end of blowdown. This discrepancy is probably a result of break conditions; a slightly smaller break multiplier might have been more appropriate.

Figure 4.3 shows the good agreement in the flow through the jet pump of the broken loop. The calculated and measured flow quickly reverse in the broken loop, allowing a large reverse flow in the associated jet pump.

The flow in the jet pump of the intact loop is shown in Figure 4.4. The measured and calculated flow are also in good agreement; after a small rise in flow due to reversing of flow in the broken loop jet pump, the flow decays slowly during the pump coastdown period (0-7 seconds). At seven seconds the downcomer water level uncovers the jet pump suction and the mass flow drops off rapidly as steam instead of water is sucked through the jet pump.

The measured and predicted core inlet flow are shown in Figure 4.5; the magnitudes of these flows are in good agreement. The core flow rapidly drops in the first second as the flow in the jet pump of the broken loop reverses. The core flow then coasts down with the intact loop jet pump flow and drops to near zero (but positive) at 7 seconds when the jet pump suction is uncovered. The lower plenum starts to flash when it reaches saturation conditions causing a significant upflow through the core. The predicted magnitude (peak) of this flashing flow through the core is in good agreement with the reported data.

4.1.2 RELAX Simulation of TLTA Test 6406

Figures 4.6 through 4.9 provide the same type of comparisons for Test 6406, and show similar agreement between RELAX calculated and measured phenomena to that shown in Figures 4.2 through 4.5 for Test 6007. The HPCI spray starts at 27 seconds and the LPCI and LPCS start at 88 and 67 seconds, respectively.

Figure 4.6 compares the measured and calculated pressure responses. As with 6007, the 6406 predicted pressure decays faster than measured; a multiplier smaller than 0.8 would have given closer agreement as with 6007. The fact that the measured vs calculated pressure responses are similar for 6007 and 6406 indicates the effects of ECCS spray are properly being predicted by RELAX. More comparisons and discussions of the effect of ECCS sprays on pressure are included in the next section.

Figure 4.7 shows good agreement in the reverse flow through the broken loop jet pump and Figure 4.8 shows the calculated and measured flow in the intact loop jet pump which generally agree within experimental uncertainties. The measured and predicted core inlet flow is shown in Figure 4.9. Again, the magnitude of the flashing peak is in good agreement.

4.1.3 Effect of ECCS Injections

Comparisons of 6406 and 6007 show the effect of ECCS sprays. Two effects are of interest: the effect of the ECCS systems was to slightly decrease the depressurization rate and lower clad temperatures. Both trends are predicted by RELAX.

While one might anticipate the ECCS subcooled sprays would increase the depressurization rate due to condensation, compensating phenomena such as more core and inert slab heat transfer and decreased quality at the break combine to decrease the depressurization. The latter phenomena combine to overcome the condensation effect with the net effect of the ECCS systems to decrease the depressurization rate.

Since RELAX calculated slightly higher depressurization rates than measured for both 6406 and 6007, it is not easy to see how well RELAX did in predicting the overall effect of ECCS spray on vessel pressure behavior from absolute pressure plots. Thus, a relative comparison was chosen as shown in Figures 4.10 and 4.11, where RELAX-to-RELAX comparisons and data-to-data comparisons are shown for 6406 and 6007. RELAX follows the same trends as the data for the two tests. As seen in Figure 4.10, the RELAX prediction of Test 6007 steam dome pressure decays faster initially than that of Test 6406. The measured pressure in Figure 4.11 for Test 6007 also decreases faster than Test 6406 data after 10 seconds, possibly due to a difference in initial conditions. As the transient continues and the HPCS injection begins, the pressure in Test 6406 decays slower than 6007. Thus, at the end of the 80 second transient, the pressure is higher in Test 6406 than Test 6007. RELAX predicts this data trend, and at 80 seconds the difference between the RELAX predictions of Tests 6406 and 6007 pressures is in the same direction and is almost as great as the difference in the data.

4.2 FLEX VERIFICATION TO TLTA TESTS

In order to be consistent with plant EM analyses, the FLEX calculations for 6007 and 6406 were initialized using the conditions calculated by RELAX at the end of the blowdown period. All nodal masses, pressures, heat slab temperatures and heater rod temperatures were initialized so as to be consistent with the RELAX results. The FLEX system nodalization used in the simulation of Tests 6007 and 6406 is shown schematically in Figure 4.12. This nodalization is identical to that used in a plant licensing analysis. Figures of FLEX results are presented in such a way as to make them as compatible as possible with the corresponding RELAX plot scales.

The results presented in this section concentrate on the ECCS injection test (6406) since the role of FLEX is to follow the liquid inventory from the ECCS systems. During the FLEX calculational period, the HPCI, LPCI and LPCS were all on in Test 6406 while none of these systems were on in Test 6007.

4.2.1 FLEX Simulation of TLTA Test 6007

The FLEX simulation of Test 6007 begins at 80 seconds of transient time and terminates at 160 seconds, at which time the power was tripped in the test. The comparison of the FLEX and measured steam dome pressures is shown in Figure 4.13. FLEX parallels the actual depressurization rate closely, and the difference between pressures due to the initialization from RELAX decreases slightly as the transient continues.

4.2.2 FLEX Simulation of TLTA Test 6406

The FLEX simulation of TLTA Test 6406 begins at 100 seconds of transient time and ends at 200 seconds when the bundle reflood level has almost reached the extent of its penetration into the core.

The comparison of steam dome pressure is shown in Figure 4.14. The initial pressure difference is maintained throughout the transient. This proper prediction of the measured pressure trend strongly supports the ability of the FLEX EM to properly analyze the refill and reflood period. The pressure response is an integrated result of ECCS interaction with all the FLEX key models: thermodynamics, critical flow, core CCFL model, bypass model, phase separation model, etc.

The overall EXEM RELAX-FLEX calculation is shown to adequately predict the trends of the TLTA data well, yet conservatively. System pressure is well predicted both with and without ECCS sprays. The refilling of the lower plenum and time of reflood are closely, yet conservatively, predicted. The peak bundle temperatures are conservatively predicted by RELAX.

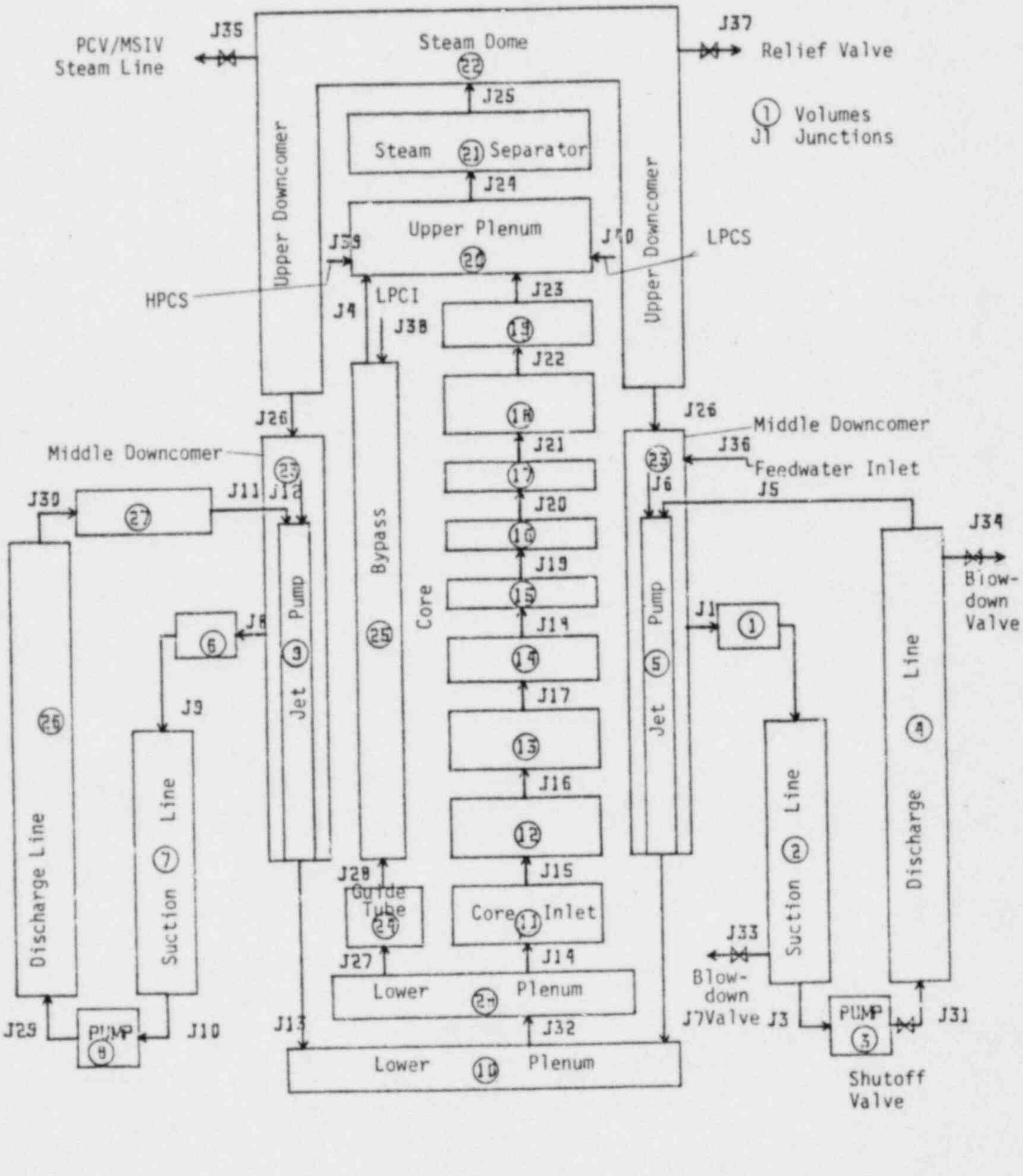


Figure 4.1 RELAX TLTA Nodalization Diagram

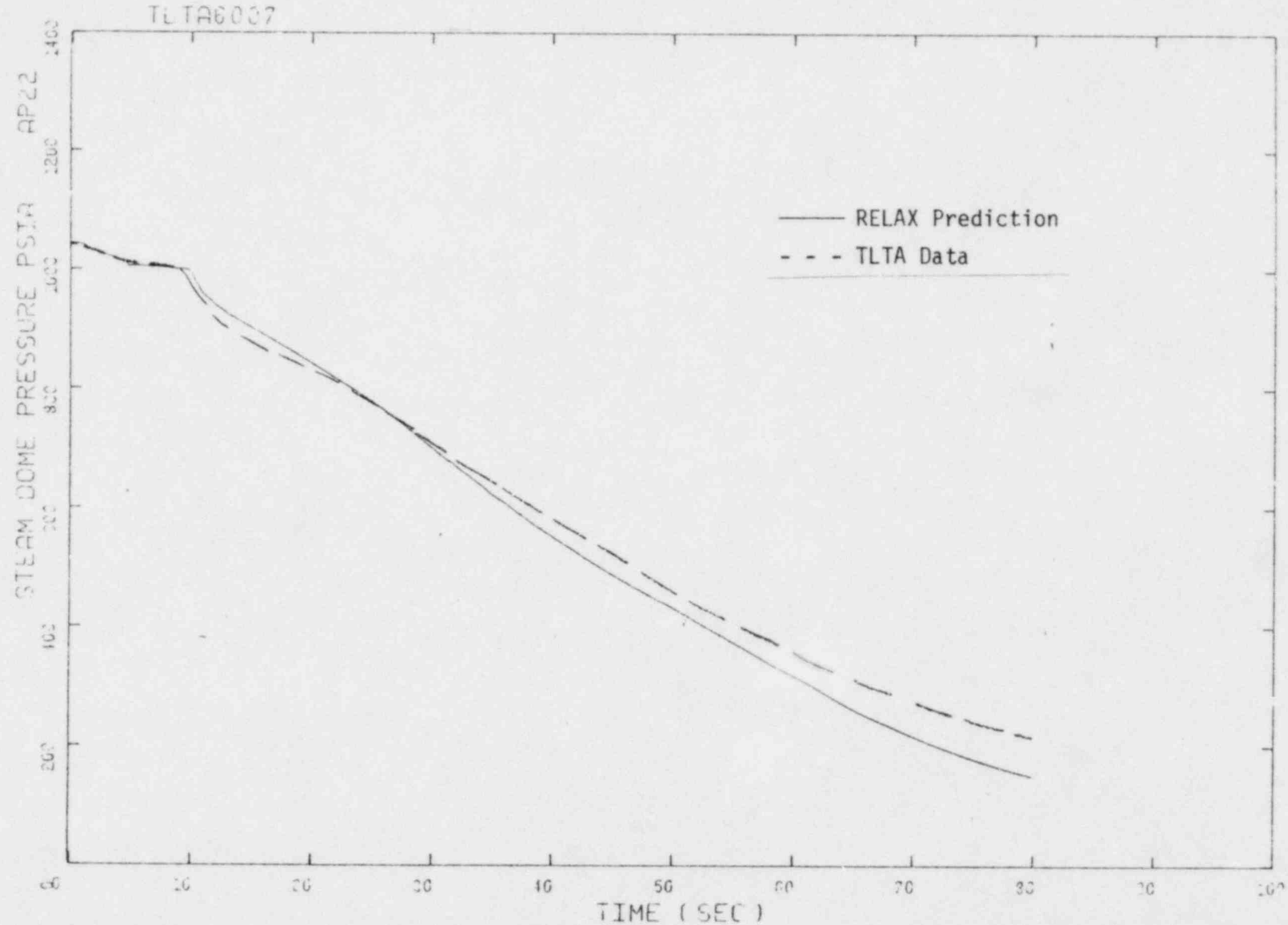


Figure 4.2 RELAX vs Data Steam Dome Pressure Comparison - Test 6007

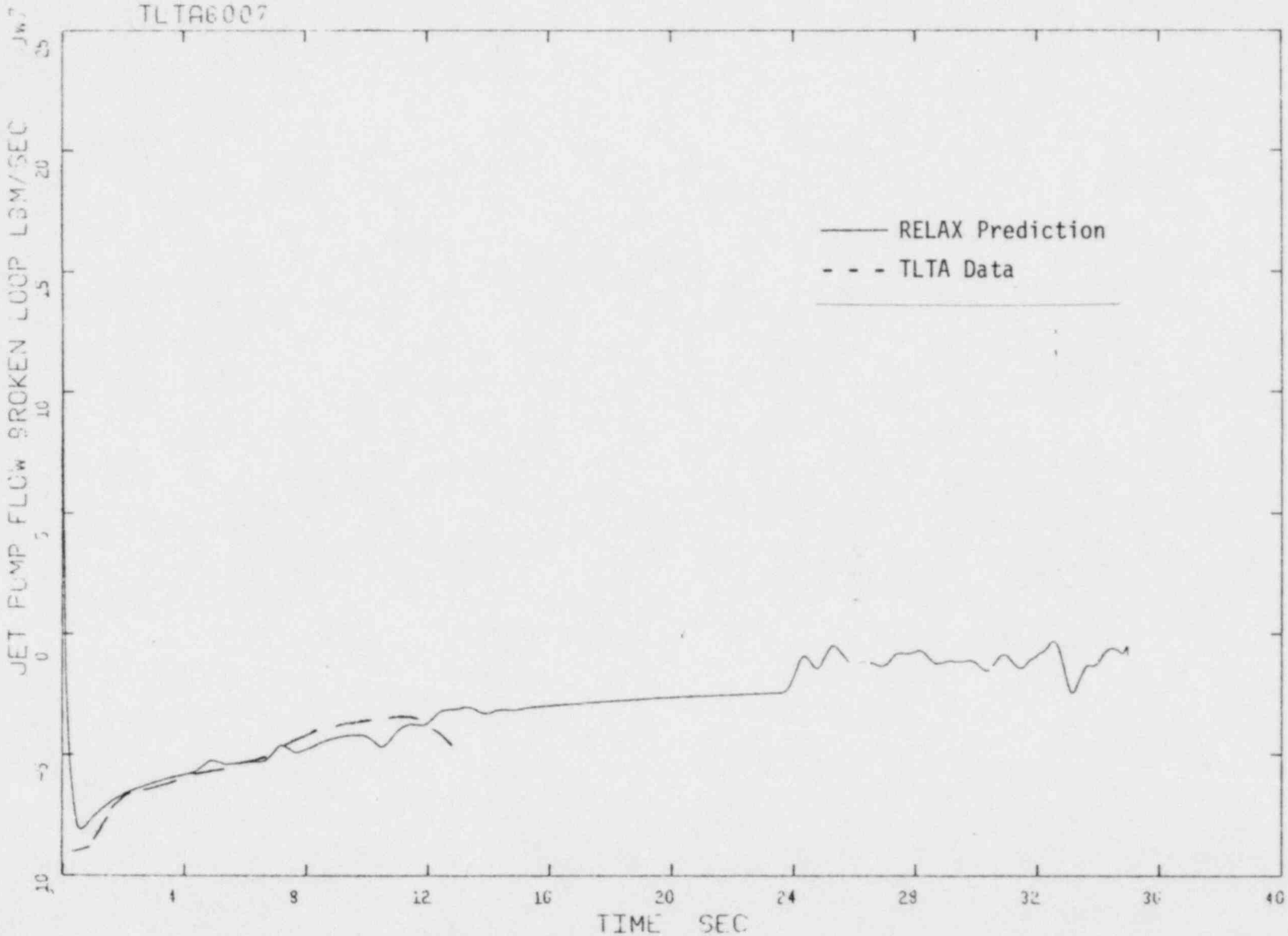


Figure 4.3 RELAX vs Data Broken Loop Jet Pump Flow Comparison - Test 6007

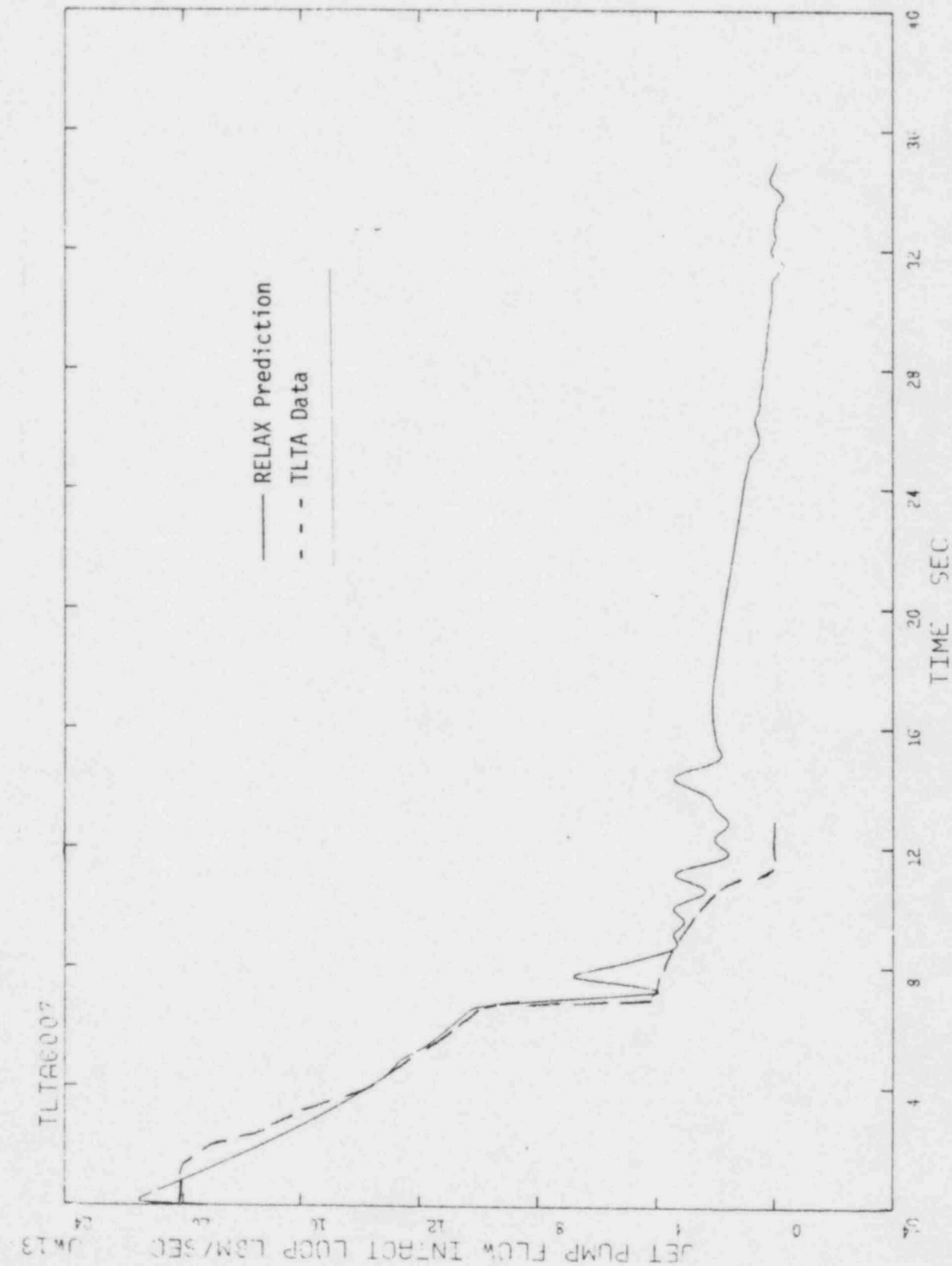


Figure 4.4 RELAX vs Data Intact Loop Jet Pump Flow Comparison - Test 6007

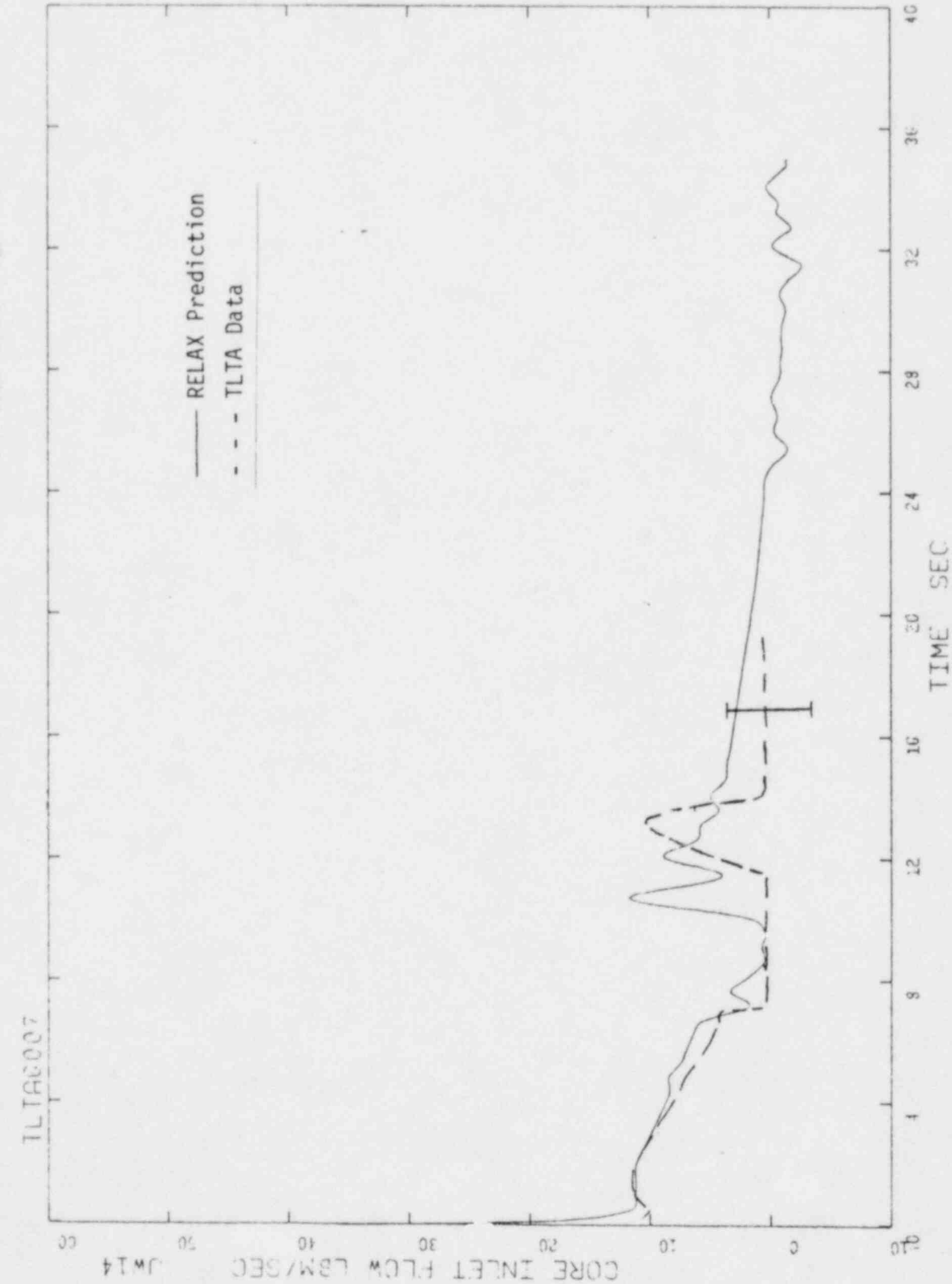
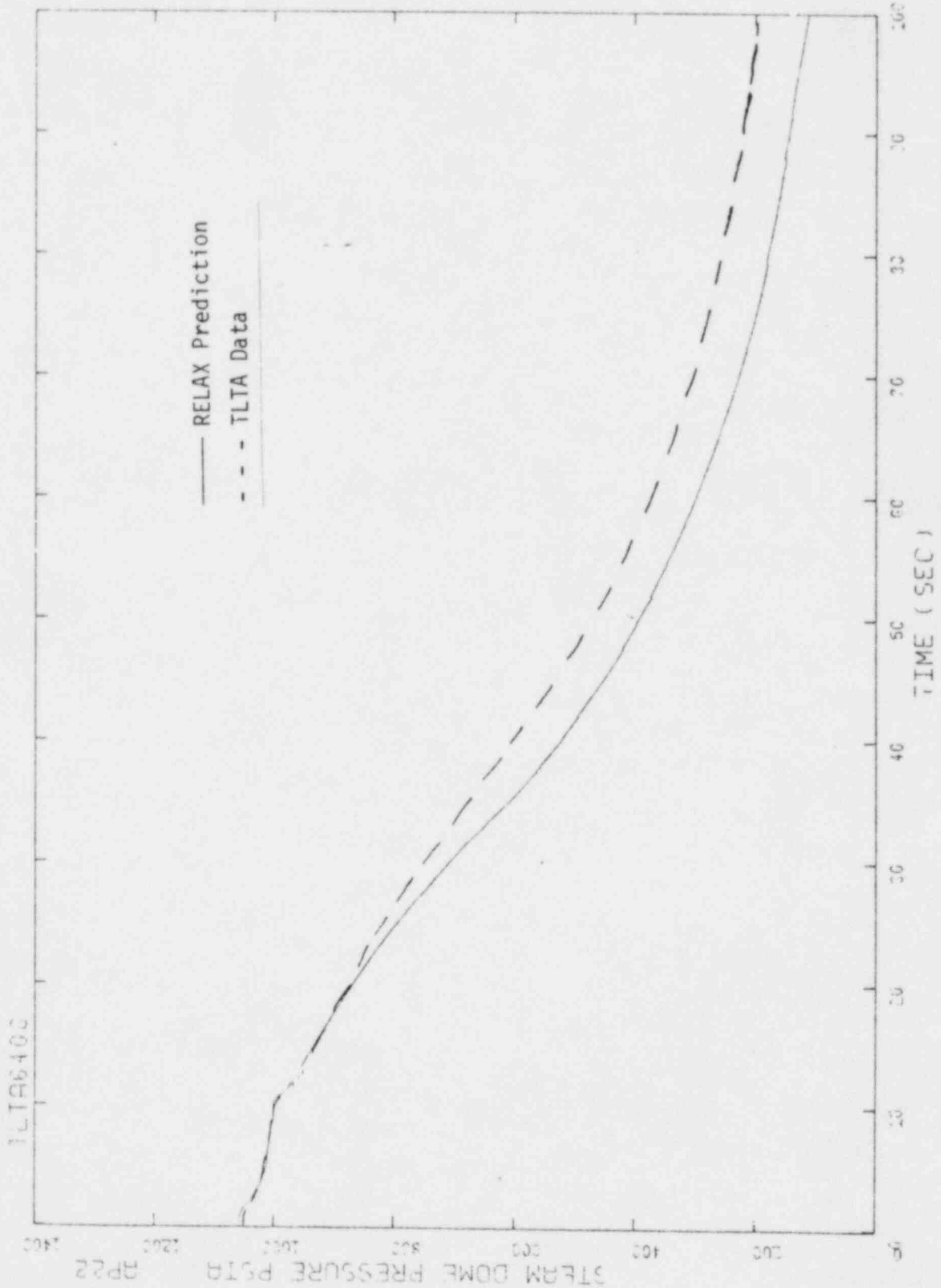


Figure 4.5 RELAX vs Data Core Inlet Flow Comparison - Test 6007



gure 4.6 RELAX vs Data Steam Dome Pressure Comparison - Test 6406

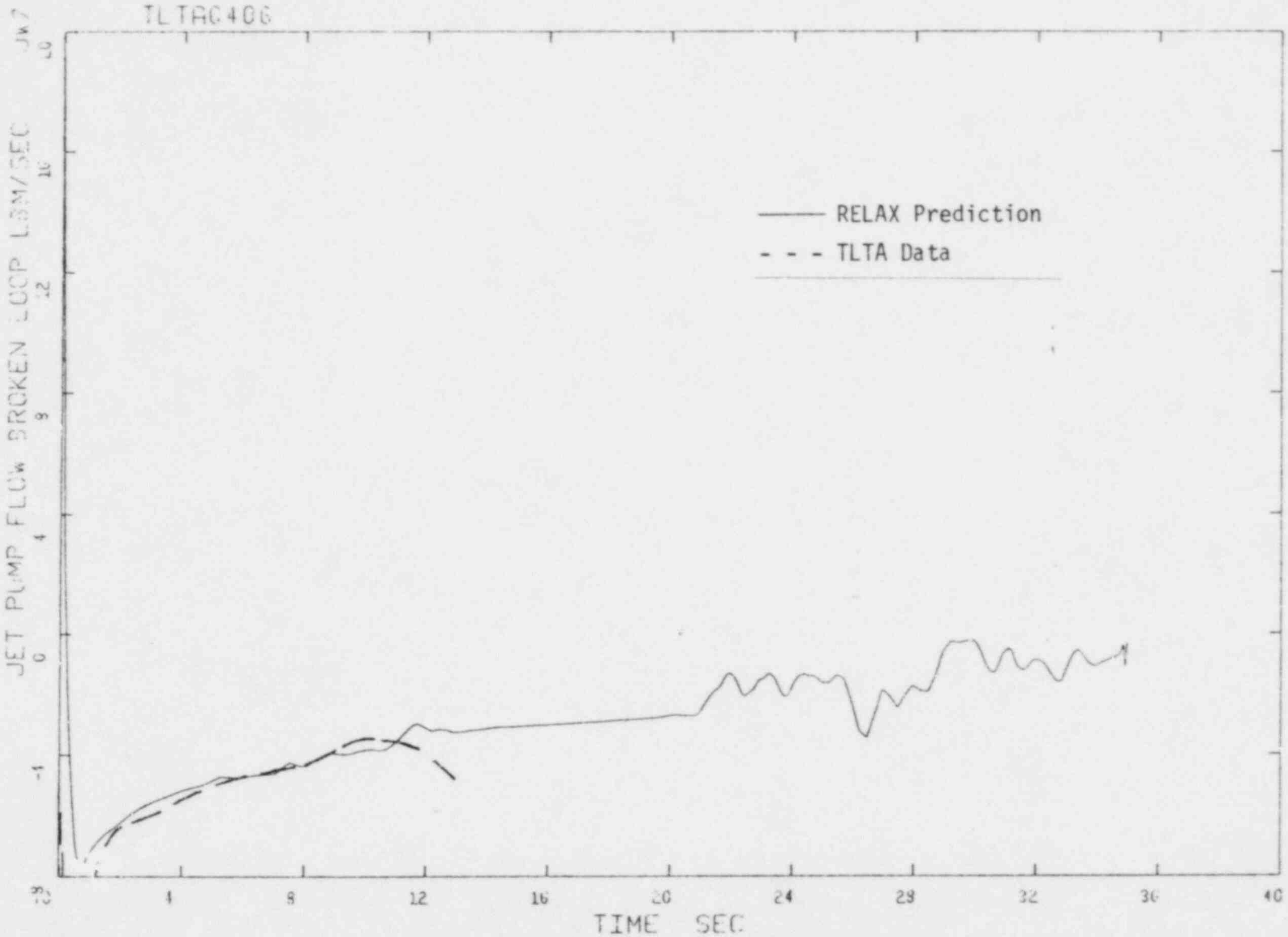


Figure 4.7 RELAX vs Data Broken Loop Jet Pump Flow Comparison - Test 6406

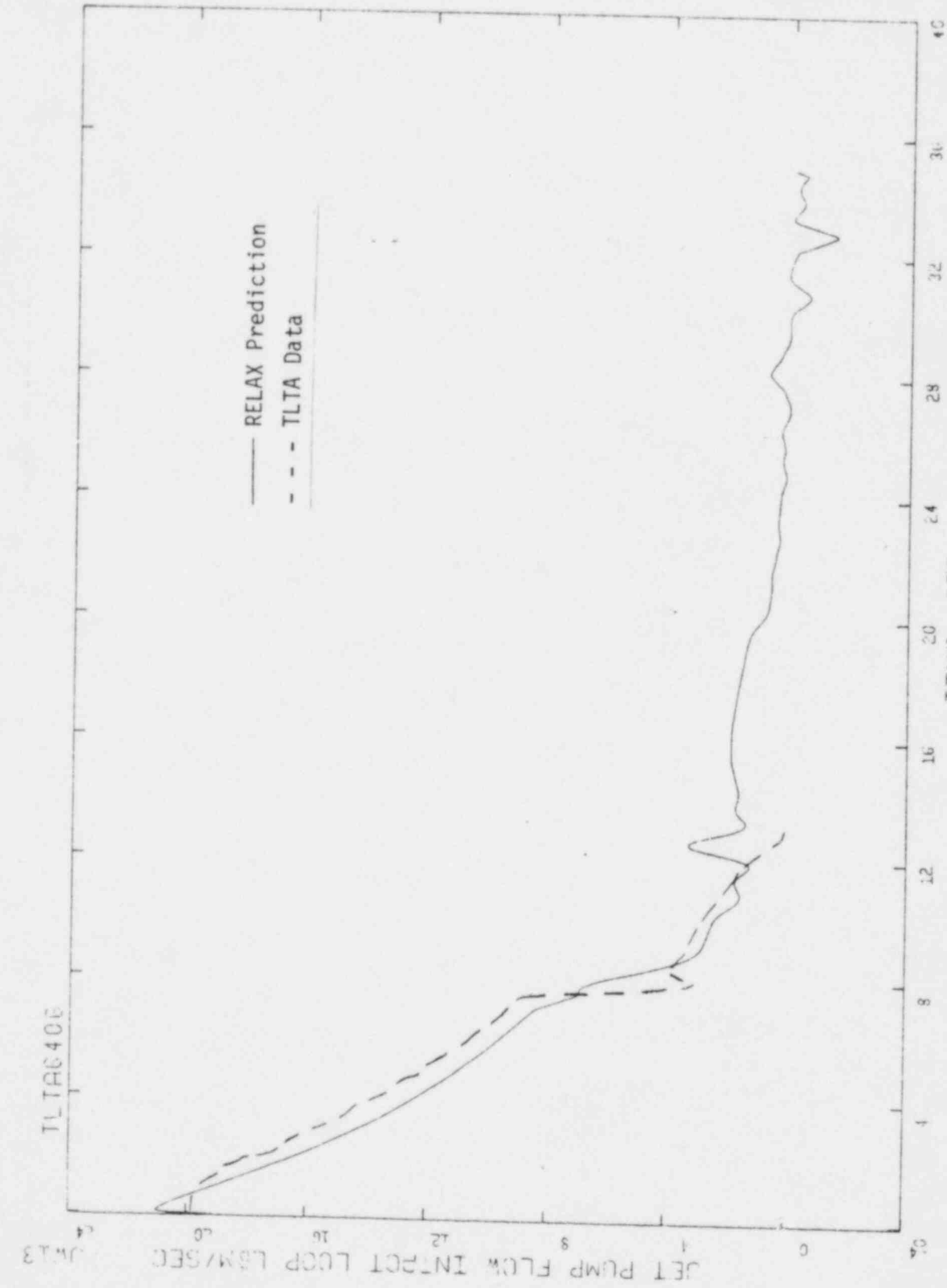


Figure 4.8 RELAX vs Data Intact Loop Jet Pump Flow Comparison - Test 6406

Figure 4.9 RELAX vs Data Core Inlet Flow Comparison - Test 6406

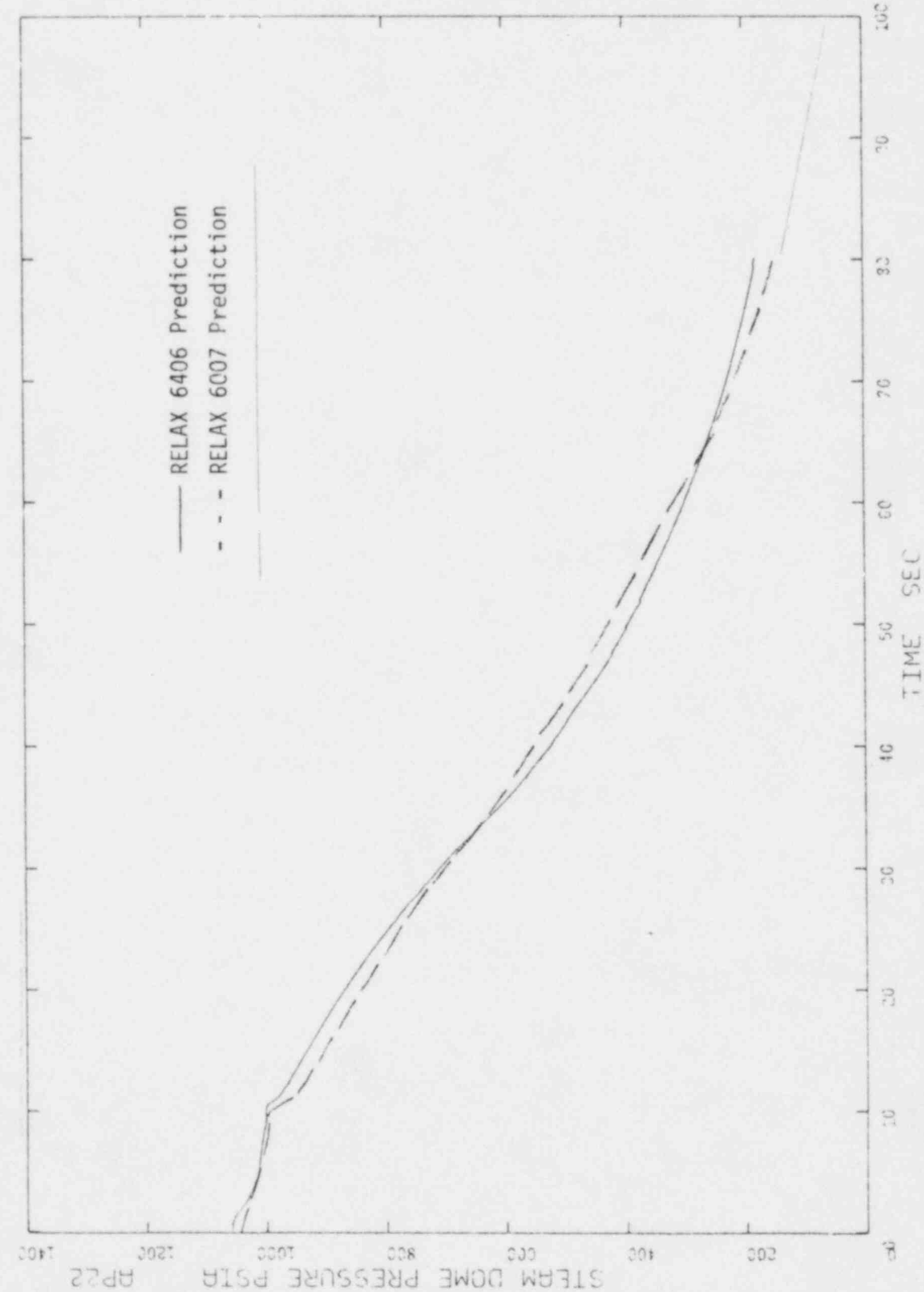


Figure 4.10 RELAX Steam Dome Pressure Comparison - Test 6406 vs Test 6007

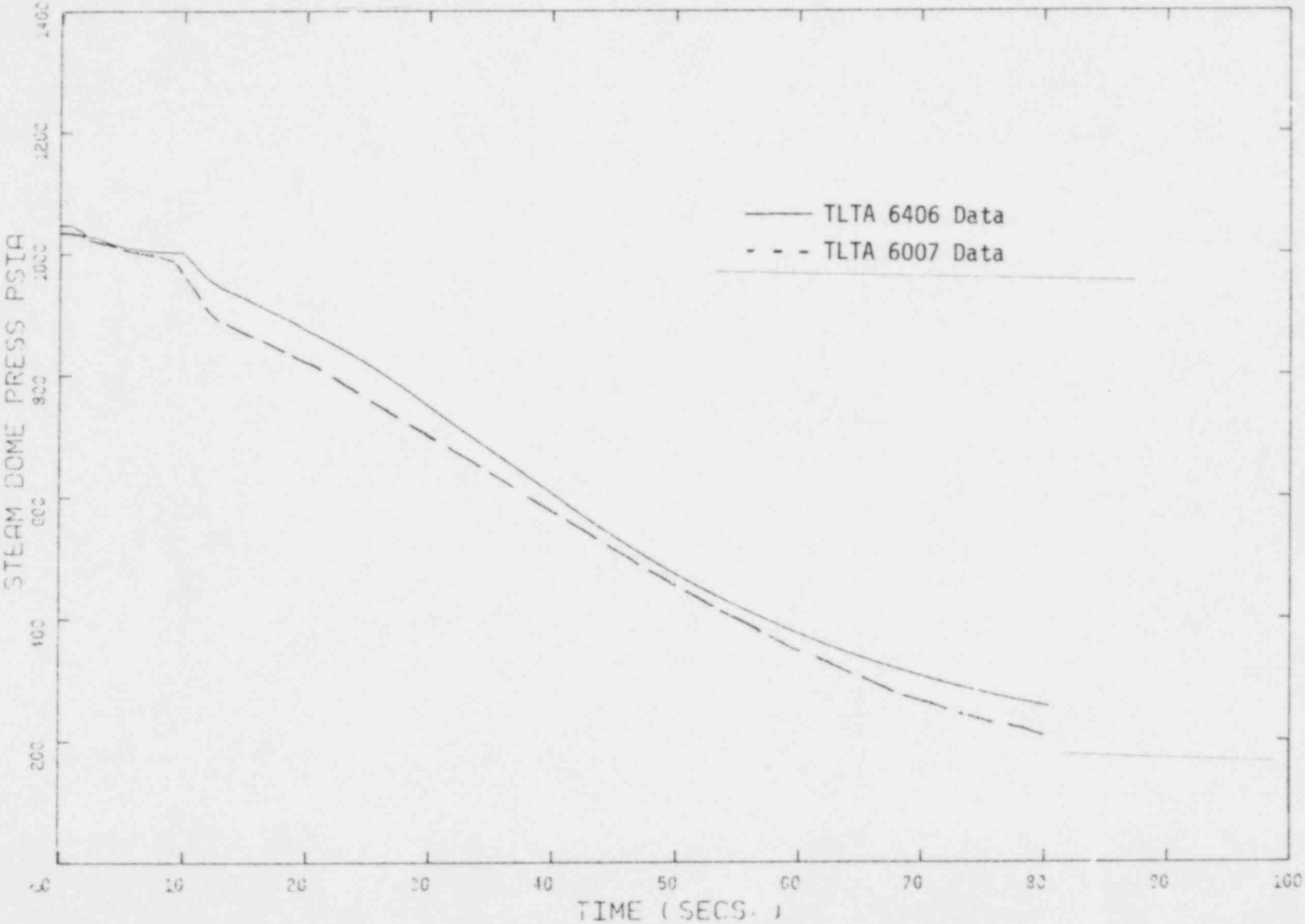


Figure 4.11 TLTA Data Steam Dome Pressure Comparison - Test 6406 vs Test 6007

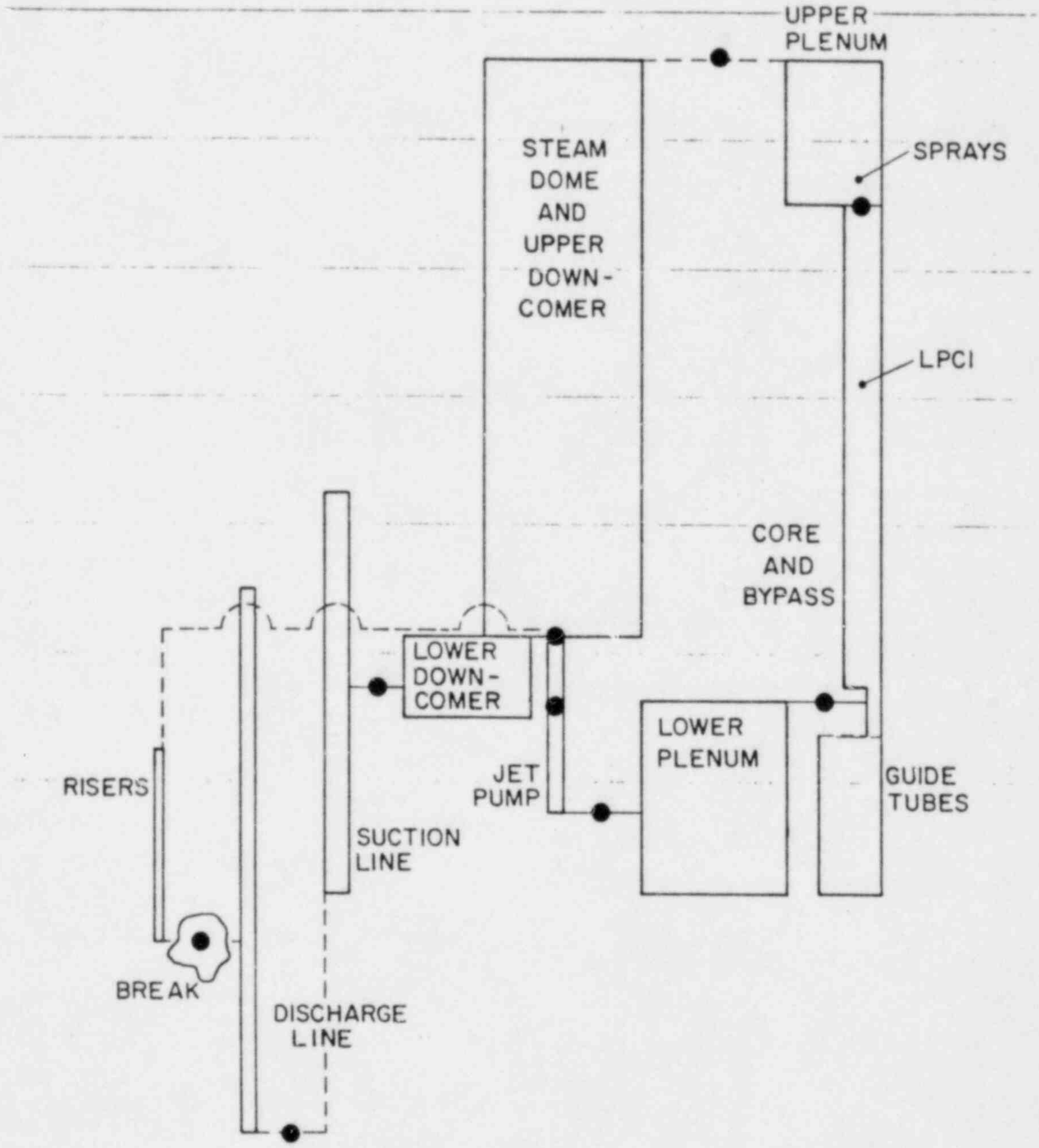


Figure 4.12 FLEX Nodalization Used for TLTA Test Simulation

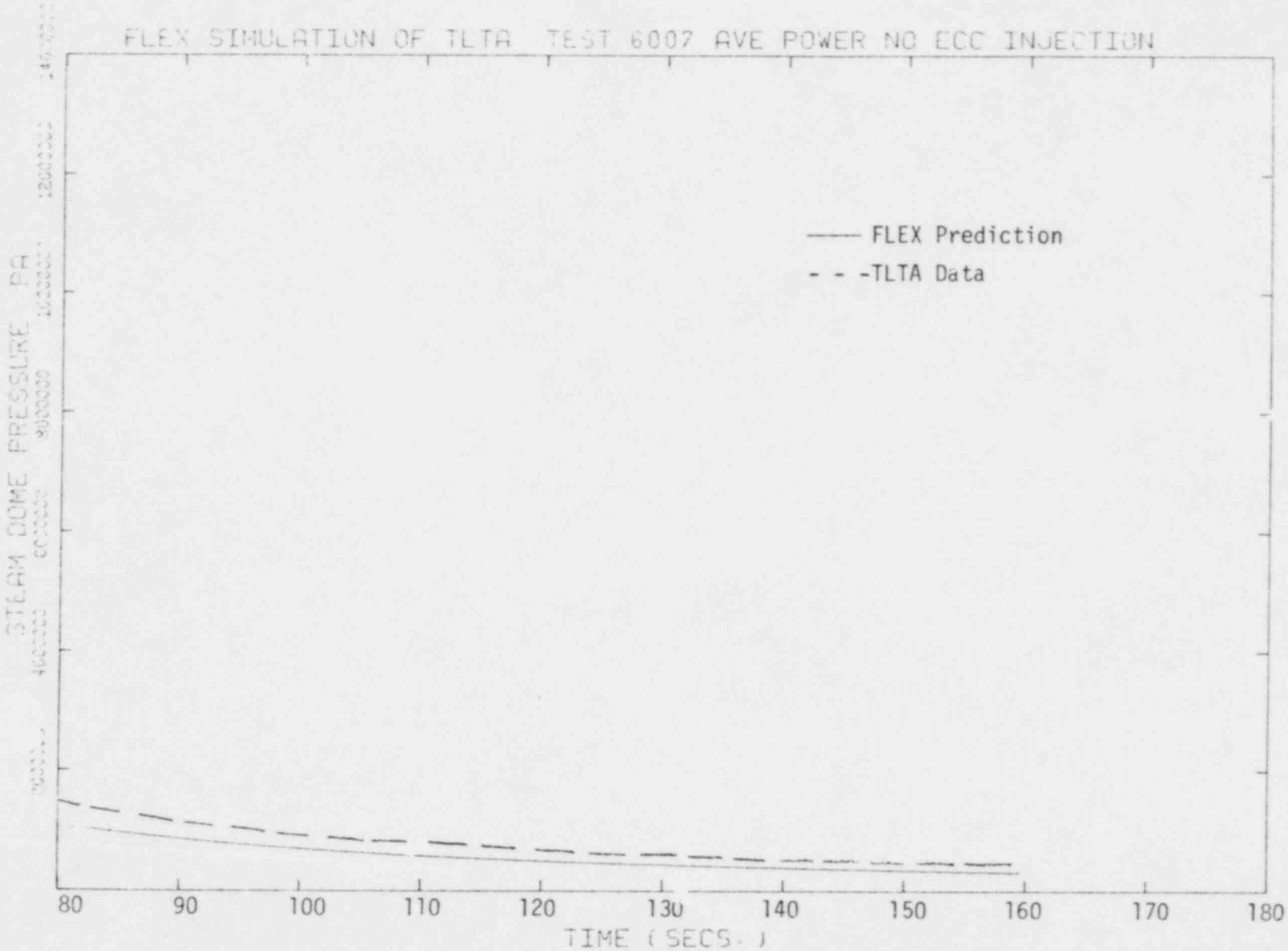


Figure 4.13 FLEX vs Data Steam Dome Pressure Comparison - Test 6007

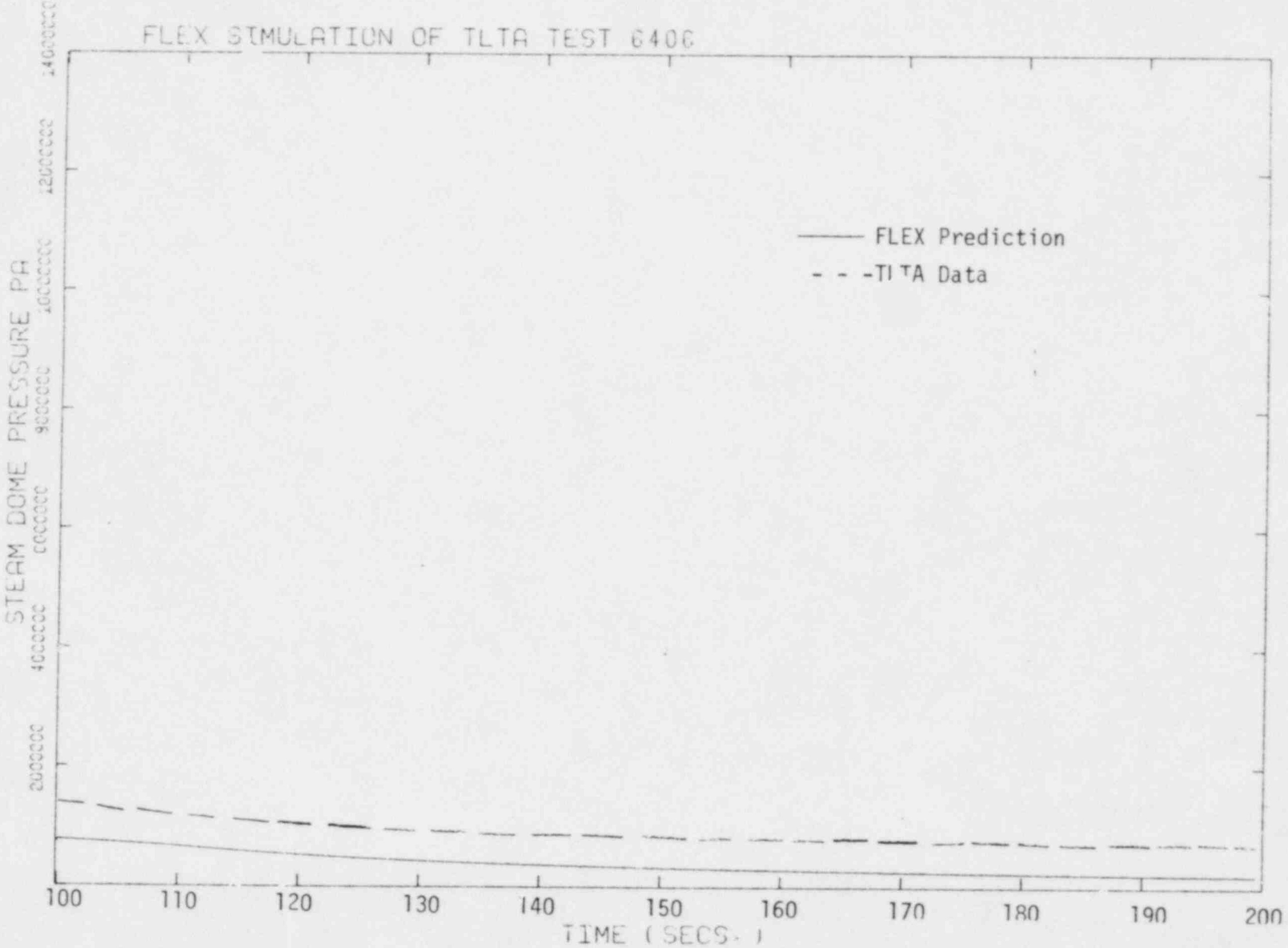


Figure 4.14 FLEX vs Data Steam Dome Pressure Comparison - Test 6406

5.0 REFERENCES

5.1 SECTION 2.0

1) ENC Letter from G. F. Owsley to D. F. Ross (USNRC), "Description of RELAP4/ENC28B," dated October 30, 1978.

2) "Exxon Nuclear Methodology for Boiling Water Reactors - Volume 2A, RELAX: A RELAP4 Based Computer Code for Calculating Blowdown Phenomena," XN-NF-80-19(P), Volume 2A, May 1980.

3) Safety Evaluation by USNRC on RELAP4-EM/ENC28, forwarded by NRC Letter from T. A. Ippolito to W. S. Nechodom (ENC), dated March 30, 1979.

4) G. W. Burnette, Sixth Monthly Report, "BWR Blowdown/Emergency Core Cooling Program," General Electric Co., San Jose, CA., April 1976.

5) H. S. Crapo, "One-Sixth Scale Model BWR Jet Pump Test," EGG-LOFT-5063, LTR 20-105, November 29, 1979.

6) Dresden, Final Safety Analyses Report, Section 4.3.

7) "Loss-of-Coolant Accident & Emergency Core Cooling Models for General Electric Boiling Water Reactors," General Electric Company, April 1971 (NEDO-10329).

8) Bromley, L.A., "Heat Transfer in Stable Film Boiling," Chem. Engr. Progr., 46, 221 (1950).

9) J. M. Delhaye, et al., "Thermohydraulics of Two-Phase Systems for Industrial Design and Nuclear Engineering," McGraw-Hill, 1981.

10) Dwyer, O.E., "Boiling Liquid-Metal Heat Transfer," American Nuclear Society, 1976.

11) J. E. Leonard, et al., "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K - Amendment No. 1 Calculation of Low Flow Film Boiling Heat Transfer for BWR LOCA Analysis, January 1977 (NEDO-20566-1).

12) Hsu, Y.Y., "Tentative Correlations of Reflood Heat Transfer," LOCA Research Highlights, April 1 - June 30, 1975 (July 1975).

13) "BWR BD/ECC Program Fiftieth Monthly Report December 197 , General Electric Company.

14) "BWR Blowdown/Emergency Core Cooling Program 64-Rod Bundle Blowdown Heat Transfer (8x8 BDHT) Final Report," General Electric Company, September 1978 (GEAP-NUREG-23977).

5.2 SECTION 3.0

1) "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, April 1971.

2) "Safety Evaluation by USNRC on RELAP4-EM/ENC28B," (ENC Letter dated October 30, 1978), forwarded by NRC letter from T. A. Ippolito to W. S. Nechodom, March 30, 1979.

3) "SINGLE - A One-Dimensional Finite Element Steady State and Transient Thermal Conduction and Convection Code," RE-A-78-119 May 1978.

4) Finlayson, B. A., The Method of Weighted Residuals and Variational Principles, Academic Press, 1972.

5.3 SECTION 4.0

1) Letzring, W. J., "BWR Blowdown/Emergency Core Cooling Program Preliminary Facility Description Report fo. the BD/ECC1A Test Phase," GEAP-23592, December 1977.

2) Muralidharan, R., "BWR Blowdown Emergency Core Cooling Fifteenth Quarterly Progress Report," GEAP-21304-15, July 1-September 30, 1979.

3) Hwang, W. S. and Schneidman, B. S., "BWR Blowdown/Emergency Core Cooling Program - 64-Rod Bundle Blowdown Heat Transfer (8x8 BDHT) Final Report," GEAP-NUREG-23977, September 1978.

4) "Magnetic Tape of BDHT Test 6406 Run 1 Data," Letter from S. D. Stevens (G.E.) to S. F. Bankert (EG&G Idaho), March 29, 1979.

5) "Magnetic Tape of BDHT Test 6007 Run 26 Data," Letter from S. D. Stevens (G.E.) to S. F. Bankert (EG&G Idaho), November 29, 1978.

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VOLUME 2C

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