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Boil-Off Experiments with the EIR-NEPTUN Facility: Analysis and Code Assessment Overview Report

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Summary

The NEPTUN data discussed in this report are from core uncovery (boil-off) experiments designed to investigate the mixture level decrease and the heat up of the fuel rod simulators above the mixture level for conditions simulating core boil-off for a nuclear reactor under small break loss-of-coolant accident conditions.

The first series of experiments performed in the NEPTUN test facility consisted of ten boil-off (uncovery) and one adiabatic heat-up tests. In these tests three parameters were varied: rod power, system pressure and initial coolant subcooling. The repeatability of the experiments was also demonstrated.

The NEPTUN experiments showed that the external surface thermocouples do not cause a significant cooling influence in the rods to which they are attached under boil-off conditions. The reflooding tests performed later on indicated that the external surface thermocouples have some effect during reflooding for NEPTUN electrically heated rod bundle. Peak cladding temperatures are reduced by about 30–40°C and quench times occur 20–70 seconds earlier than rods with embedded thermocouples[1]. Additionally, the external surface-thermocouples give readings up to 20 K lower than those obtained with internal surface thermocouples (in the absence of external thermocouples) in the peak cladding temperature zone.

Some of the boil-off data obtained from the NEPTUN test facility are used for the assessment of the thermal-hydraulic transient computer codes. These calculations were performed extensively using the frozen version of TRAC-BD1/MOD1 (version 22). A limited number of assessment calculations were also done with RELAP5/MOD2 (version 36.02). In this report the main results and conclusions of these calculations are presented with the identification of problem areas in relation to the models relevant to boil-off phenomena. On the basis of further analysis and calculations done, changing some of the models such as the bubbly/slug flow interfacial friction correlation which eliminate some of the problems are recommended.

1 Introduction

The drying out of a nuclear reactor core by boiling off the coolant inventory is known as a core uncovery event. The accident that happened at the Three Mile Island Unit-2 (TMI-2) Plant on March 29, 1979 can be classified as a small break loss-of-coolant accident (SBLOCA) followed by core uncovery. The flow of the core was blocked and the coolant evaporated through the stuck-open pressure relief valve causing gradual depletion of the core coolant inventory. Part of the core was uncovered and extensive damage to the fuel rods was done. After the occurence of this accident, increased attention has been paid in Light Water Reactor (LWR) Safety analysis to low flow and intermediate or low pressure transients which, if no remedial measures are taken, may sequentially lead to uncovery, overheating and damage of the core. When the reactor power is at decay heat levels and the coolant entering the core is subcooled, uncovery of the core is likely to occur only at relatively low coolant mass flow rates. Thus, hydraulic and heat transfer mechanisms associated with cooling of fuel rods in a pool of water without external circulation need to be better understood. In particular, there is an obvious need for assessing thermal hydraulic safety codes as far as their ability to correctly predict the level swell (or expansion of the boiling pool) and the fuel rod temperatures in the uncovered region. Consequently, considerable effort is being spent not only in improving and extending the available transient thermal hydraulic codes, but also in performing carefully controled experiments in test facilities; the results of these simulations can be directly utilized for assessing the predicting capabilities of these codes or even developing new models, hence giving a direct feed-back to the code-developers.

At the Swiss Federal Institute for Reactor Research (EIR), the NEPTUN facility [2] and rod bundle were originally designed for low pressure ($\leq$ 5 bars) reflood experiments whose aim was to study the heat transfer characteristics between the rods and the coolant. Additionally, a number of core-uncovery (boil-off) experiments have been performed to investigate the mixture level decrease and resulting fuel rod heat-up above that level that may occur in a PWR during small and intermediate break LOCAs. These boil-off tests have been performed using a variety of initial parameters (eg. rod power, system pressure, coolant subcooling etc.).

In these experiments it was also possible to evaluate the accuracy and cooling influence of the external surface thermocouples (similar to those used in the LOFT facility at the Idaho National Engineering Laboratory) over a range of power and system pressures.

This report summarizes the NEPTUN boil-off experiments as well as comparisons between experimental data obtained from the tests with corresponding predictions obtained using the best-estimate thermal hydraulic code TRAC-BD1[8] .

The NEPTUN system configuration and boil-off tests and related test matrix are discussed in Sections 2 and 3, respectively. Experimental results and system response are presented in Section 4. Section 5 summarizes the results of NEPTUN boil-off experiments with regard to external thermocouple response. Section 6 discusses the results of code assesssment calculations. Conclusions related to NEPTUN boil-off experiments are presented in Section 7.

2 Description of the NEPTUN Test Facility

2.1 General Description

A simplified flow diagram of the NEPTUN facility is shown in Figure 1.
NEPTUN consists of three main parts:

- The test section, including
  - a measuring system for the liquid carry-over rate and for the steam expelled during the experiment,
    and
  - a backpressure control system.
- A water loop to bring the water to the test conditions.
- Several desired inlet or initial auxiliary systems to maintain normal operating conditions.

2.2 Test Section

2.2.1 The Test Bundle

The test bundle consists of 33 electrically heated rods and 4 unheated guide-tubes placed in an octagonal housing as shown in Figure 2. The NEPTUN bundle arrangement corresponds to a section of the LOFT nuclear fuel bundle.

The length of the bundle is subdivided into 8 equally spaced measurement levels (Fig. 3). At these levels local fluid temperatures, pressures and pressure differences within the test section are measured.

A detailed description of the components in the test bundle follows:

The heater rods have a heated length of 1680 mm. Figure 4 shows the axial chopped cosine-type power distribution used. The axial power distribution for the LOFT nuclear fuel and the SEMI-SCALE heater elements are also given in this figure for comparison. Details of the NEPTUN electrical heater rods and their main specifications are shown in Figure 5.

The cosine power distribution is obtained by a kanthal heater rod of axially changing diameter in the center of the heater element. The electric current, flowing from the bottom to the top of the element, is led back to the bottom by a noncentric copper tube. The kanthal rod is electrically insulated by a boron nitride layer; the copper tube which is placed between two thin inconel tubes is also insulated by an Al_2O_3 plasma

coating. The canning consists of 2 thin-walled inconel tubes. The heater rods are equipped with 4 or 8 thermocouples which are situated in small notches between the two outer inconel tubes. Several drawing processes during the assembly of the heater element assure good heat transfer from one layer to the next.
Axial heat conduction calculations did show that the axial heat profile is not seriously disturbed by the good heat conductivity of the copper tube [4].

Guide tubes
In the NEPTUN bundle there are 4 unheated guide-tubes. Their positions are shown in Figure 2. The outer diameter of the guide-tubes is 13.87 mm and is slightly larger than that of the heater rods.
All of the 4 guide-tubes are equipped with 8 thermocouples to measure the temperature of the guide-tube wall.

Spacer grids
The spacer grids are of the same design as in the LOFT nuclear core. There are 5 spacer grids, equally spaced, within the NEPTUN test section. Their axial positions are shown in Figure 3.

2.2.2 Octagonal Housing and Pressure Vessel

In order to obtain a low heat capacity, the housing has been designed with a wall thickness of only 2 mm. Because the thin wall cannot stand too high pressure differences at high temperatures, the housing is surrounded by a pressure vessel containing pressurized nitrogen. The space between the housing and the pressure vessel is also filled with an insulating (low heat capacity) material. The pressure in this place is controlled by a pressure regulating system and is held automatically at approximately the same pressure as inside the housing.

The shape of the housing (Figure 2) is chosen in such a way that the parasitic flow area does not exceed 9 percent of the actual flow area in a square pitched 37 rod bundle. The housing is fabricated from inconel 600 tubing which is formed into the required shape by several drawing processes.

Two tubes, displaced by 180 degrees, are welded radially onto the housing at each measurement level (Figure 2). Each one of these tubes is connected with the test space by a boring of 2 mm diameter through the wall of the housing. The tubes can be used to measure pressures, pressure differences or fluid-temperatures.

The axial and azimuthal temperature distributions of the outer surface of the housing and axial and radial temperature distributions in the insulating material between the housing and the pressure vessel are measured by up to 31 thermocouples. Details are shown in Figure 6.

2.2.3 Steam Flow and Carry-over Measurement System

The two-phase mixture which is expelled from the test section is separated into steam and water. The steam flow is measured with a turbine flow meter in the steam exhaust line, while the entrained water is collected in the carry-over tank.

Steam-water separator and carry-over tank (Figure 7)
Directly above the upper end of the test section, there is a diverting plate which deflects the steam-water mixture expelled from the test section into horizontal direction. Due to the large cross sectional area of the steam-separator, the steam moves slowly towards the steam exhaust line. The water falls down to the bottom of the separator and flows then into the carry-over tank in which the carry-over rate is measured.

Exhaust steam line(Figure 1)
After leaving the separator, the steam flows through a flow measuring system consisting of:

- an electrically heated tube to dry and superheat the steam for avoiding any condensation (item 12),
- a turbulence promoting plate for equalizing the temperature profile in the flow (not shown),
- a flow straightener to suppress any rotational component of the flow (not shown),
- a tap for pressure measurement,
- a turbine flow meter (item 8),
- sensors to determine the steam temperature.

2.2.4 Regulation of the Test Pressure

The experiments should be run at a test section pressure as constant as possible, in spite of time dependent steam production during the boil-off phase. Therefore a regulating system is necessary. It consists of a V-ball valve (Figure 1, item 10) and a regulating unit, which actuates the valve in such a way, that the desired pressure level is kept constant. The actual pressure signal is supplied to the regulating unit from the steam water separator chamber.

2.3 Water Loop

The water is brought up to the required inlet or initial conditions (temperature, pressure, flow rate) in a closed water loop (fig 1). The following components are available to obtain the desired water conditions:

- a water supply tank with an electric heater (item 4),
- a cooler (item 5),
- a pump (item 6),
- a regulating valve (item 13),
- a motor valve (item 14).

To avoid oxidation of the heater rods as much as possible and to avoid any deposition on hot surfaces, demineralized water with very low oxygen content is used. This last condition is obtained by boiling and degassing the water in the water supply tank at atmospheric pressure.

2.4 Auxiliary Systems

2.4.1 Fresh steam Supply System

A steam boiler (fig.1, item 7) produces steam by boiling demineralized, oxygen free water. Prior to heating-up the heater rods, steam is fed into the steam-water separator:

- to purge the whole test circuit,
- to build up and maintain the desired test pressure.

2.4.2 Data Acquisition System

The data acquisition system is the primary data collecting system and consists of a HP-2100 computer and associated equipment. The system can record 300 channels of analog input data representing bundle and system temperatures, bundle power, flows, and absolute and differential pressures. Each data channel is recorded at least once every 2 seconds. The digitized data are stored on magnetic tape. The data reduction and processing is carried out at the computer center of EIR.

2.4.3 Nitrogen Supply System

Nitrogen is used:

- to adjust the pressure in the pressure vessel according to the pressure in the test section, in order to avoid an overloading of the housing,
- to maintain a slight overpressure in the main test loop during shut down. This is a preventive action for keeping the oxygen content in the test section as low as possible.

3 Experimental Procedure

This chapter is a summary of significant instructions which were followed during experiments 5000–5011.

In each experiment the following 3 parameters defined the test matrix:

- the test section pressure — p
  and the corresponding saturation temperature — $T_{sat,p}$
- the initial temperature of the water in the — T_ℓ
  test section and corresponding saturation pressure — $p_{sat,\ell}$
- the heater-rod power.

In order to establish the desired initial test conditions, a series of operations were conducted prior to the start of the experiment. The main operating instructions are described below:

1. 1-2 Hours prior to the start of a boil-off experiment:
   Degas the demineralized water in the water supply tank by boiling at atmospheric pressure.
   Circulate the water in the water loop and adjust the parameters to the desired values:
   (a) The pressure in the water supply tank to a value approximately 0.5 bar higher than $p_{sat,\ell}$ by boiling water at a small power rate and bleeding an appropriate amount of vapour.
   (b) the temperature near the test section inlet valve to T_ℓ by adjusting the flow-rate (and therefore the heat losses in the piping system) to an appropriate value.
   (c) Heat up the fresh steam generator to a temperature corresponding to a pressure of approximately 8 bars.
   (d) Heat up the main components in the contact with steam by means of electric strip heaters:

       the steam-water separator,

       the entrainment tank, to $T_{sat,p}$

       the exhaust steamline,
       the superheater to 250–300 °C

2. Purge all pressure transmission lines of the differential pressure measuring system with cold, degassed water. Vent the differential pressure cells.

3. a) Determine the calibration coefficients of the test section differential pressure transducers by means of a two point method [a] assuming a linear characteristic,

   b) Set the zero of the absolute pressure transducers at atmospheric pressure.

   All other pressure transducers are calibrated prior to experiment 5000 only.

4. Set the pressure control system to maintain a constant pressure $p_{sat,\ell}$ in the test section.

5. Increase the test section temperature (housing, heater rods, guide tubes) to the initial test wall temperature T_w by feeding steam from the steam generator into the steam-water separator until the pressure in the test-section reaches $p_{sat,\ell}$.

   After reaching $p_{sat,\ell}$, purge the test section during a few minutes with a small steam flow.

   As soon as temperatures of the heater rod and housing have stabilized at T_w: fill the test section with water of temperature T_ℓ from the water loop. The test section pressure is held constant by the pressure control system.

6. Set the pressure control system to maintain a constant pressure p in the test section.

7. Increase the pressure in the test section to p by feeding again steam into the steam-water separator.

   Maintain a continuous steam flow from the steam generator through the steam-water separator and the exhaust steam line.

   After reaching p purge the entrainment tank several times with steam to stabilize the wall temperature at $T_{sat,p}$.

8. As soon as the wall temperatures have stabilized at $T_{sat,p}$, increase the steam flow to a value at which both the fresh steam turbine and the exhaust steam turbine are operating.

9. Switch the data acquisition computer into the fast-scanning mode.

10. When all of the specified initial conditions are established, start the experiment by applying the desired electric power to the heater rods.

[a] completely flooded test section and empty test section

11. Keep the rod temperatures under observation. (10 digital displays were available for monitoring rod temperatures of special interest). As soon as the maximum rod temperatures reach ~850°C shut off the electric power to the heater rods.

12. Shut down NEPTUN by:
    - flooding the test-section with water from the water loop,
    - draining the test section and the entrainment tank,
    - switching off all heaters, pumps, etc.

    During the cooldown of NEPTUN nitrogen is automatically fed into the main test loop to maintain a pressure of approximately 1.1 bar.

4 Experimental Results and System Response

The ten core boil-off experiments performed are summarized in Table 1. Three parameters were varied-rod power, system pressure, and initial coolant subcooling. Rod power levels were chosen to represent the nuclear decay power. System pressure was varied over the 1–5 bar range possible in the NEPTUN facility.

Experiment number 5007 (see Table 1) was chosen as the "base case" because it was conducted at higher pressure and intermediate power. However, several experiments will be discussed to show the difference in the NEPTUN system response due to varying rod power, system pressure, initial water subcooling, and finally, to demonstrate experiment repeatability.

4.1 System Response–Base Case, Experiment 5007

Figure 8 presents an overlay[b] of the core power history, core fluid level as measured by the core total Δp measurement, and typical responses of the heater rod thermocouples for the base case. Notice that the power is increased at about 50 s, and that a rapid drop in the core fluid level occurs at about 100 s. This delay time between initial power and the rapid initial liquid level decrease is a result of heating the subcooled water to the saturation temperature. Shortly after the water reaches the saturation temperature (between 100 and 110 s), large voids are formed due to vapor generation, expelling some of the liquid from the core. After this initial liquid swell, the core liquid continues to be slowly boiled off as shown by the decreasing core Δp in Figure 8. The cladding thermocouples dry out and heat up with differing heat-up rates for each axial elevation. At ~800 s the power is shut off and the rods are allowed to cool down before the system is reflooded.

[b]The system response overlays are presented to compare selected general system response data

4.2 System Response–Effect of Core Power

The effects of increasing the core power (experiment 5006) and decreasing core power (experiment 5008) on the boil-off response are presented in Figures 9 and 10, respectively, which can be compared directly to the base case (5007) response shown in Figure 8. The system responded as expected in each case; increased core power resulted in more rapid boil-off and cladding heatup, and decreased power resulted in the opposite trends.

4.3 System Response–Effect of System Pressure

The effect of lowering the system pressure from 5 to 1 bar (experiment 5002) on the core liquid level and cladding temperatures is shown in Figure 11. The lower system pressures result in more rapidly decreasing liquid levels and rod dryouts ranging from 100–200 s earlier than experienced for the base case (Figure 8).

4.4 System Response–Experiment Repeatability

The experiment was very repeatable for both high- and low-pressure conditions. Experiments 5001 and 5002 were repeat experiments at low pressures. Figure 12 shows the system response for the repeat conditions (experiment 5001) for comparison to Figure 11, discussed in the previous section. Experiments 5006 and 5009 are repeat experiments at high pressure and comparison is shown in Figures 9 and 13.

5 Evaluation of the External Surface Thermocouple Response

One of the objectives of the boil-off experiments in the NEPTUN facility was to obtain experimental data for assessing any perturbing effects of the external surface, thermocouples used in LOFT, during simulated small-break core uncovery conditions. Prior to the NEPTUN Experiments, it was hypohtesized that the external surface thermocouples might cause additional selective cooling of the rods, which would result in delayed dryout for a slow core uncovery experiment in LOFT. Also, the added increase in surface area for heat transfer (fin effect) might result in additional atypicalities.

The first series of experiments performed in NEPTUN consisted of eleven tests, ten boil-off and one adiabatic heat-up test. In these tests, three parameters were varied: rod power, system pressure and initial coolant subcooling as already described in chapter 4. The effects of the external surface thermocouples were determined by comparing the cladding temperatures (as measured by the external LOFT-type surface thermocouples) to cladding temperatures from thermocouples within the cladding of the NEPTUN heater rods[1]. Figure 2 shows a schematic of both the external surface thermocouples (LOFT-type) and NEPTUN embedded thermocouple configuration used for the experiments discussed in this report. Note that there is only one active external surface thermocouple on each of the five LOFT-type rods, the other external surface thermocouples are replaced by dummy elements.

Overlay plots of the thermocouple responses for many different thermocouples at each axial elevation are contained in the appendices of reference 5 for each experiment. These plots indicate that the readings of the external surface thermocouple is well within the response spread of the internal thermocouples. In this report, as bounding cases, experiments 5007 (base case) and 5011 corresponding to small and intermediate break LOCA decay heat levels, respectively, are discussed.

Overlay plots of the thermocouple responses at level 4 corresponding to the maximum linear heat generation position on the heaters are shown in figures 14 and 15 for experiment numbers 5007 and 5011 respectively. Notice that a systematic lower temperature is measured by the external surface thermocouples. This temperature difference can be taken as an estimate of the cooling effect of the external surface thermocouples and is less than 20 K. Notice also in figures 14 and 15 that there is less than 5–10 s difference in the initial dry-out times for all level 4 thermocouples, both embedded and external.

6 Code Assessment

The NEPTUN boil-off experimental data were used for assessing the thermal-hydraulic transient computer codes TRAC-BD1/MOD1 and RELAP5/MOD2. The assessment calculations performed with TRAC-BD1/MOD1 will be given in more detail in the next subsections. During the early phase of the assessment of the RELAP5/MOD2 code, some simulation and calculational difficulties were encountered for boil-off cases e.g. very large discrepancy in calculating the amount of expelled water out of the test section as shown in figure 16 [11]. Further calculations were not performed with RELAP5/MOD2, until the reasons for such discrepancies were identified. Two other attempts using RELAP4/MOD6 and RELAP5/MOD1 were also not very successful [13]. The RELAP4/MOD6 calculations could be performed until the onset of nucleate boiling and as soon as vapor was produced, very large pressure spikes were observed, resulting in time consuming and costly calculations. RELAP5/MOD1 which employes five-equation hybrid model calculated about 100 seconds earlier critical heat flux occurance with respect to RELAP5/MOD2 and the amount of expelled water out of test section was even more over-predicted.

6.1 Assessment of the frozen version of TRAC-BD1/MOD1 and problem areas

A number of NEPTUN boil-off experiments have been utilized for assessing the predicting capabilities of the thermal-hydraulics transient analysis code TRAC-BD1. Originally, the experiments were analyzed with version 12 of TRAC-BD1 [3] and the problem areas were identified [6,7]. Subsequently, five of these experiments were re-analyzed by using a frozen version of the code (version 22) commonly known as MOD1 [8].

Since the models related to the dominant physical phenomena in these core uncovery experiments are the same in both versions of the code and the problem areas identified with MOD1 were almost the same with the ones of version 12, we shall concentrate on the results obtained by employing TRAC-BD1/MOD1. For more details, the interested reader is referred to a series of reports [6,7]; here we shall outline our findings related to the problem areas as well as the model improvements already implemented in MOD1. One of these improvements is already included in the BF1 version of the code recently released.

Four boil-off experiments at 5 bar and one at 1 bar were analysed using the frozen version of TRAC-BD1/MOD1 and experiments are separately summarized in Table 2. A number of numerical problems have been revealed in the course of the analysis of these experiments with TRAC and have been extensively analysed and reported elsewhere [6,7]; here, we shall restrict our attention on the problem areas of the code related to the actual physical modeling of the phenomena taking place.

Comparison of measured and calculated collapsed liquid level (CLL) and peak axial power level rod surface temperature histories for the five experiments are shown in

Figures 18 to 22. One can readily draw the following conclusions regarding the predicting capabilities of the code:

(a) TRAC-BD1/MOD1 underpredicts the CLL histories and hence, predicts an earlier CHF than the measurements show. Clearly, the code overpredicts the amount of water expelled from the test section. These differences are more pronounced for the 1 bar experiment.

(b) Generally, TRAC-BD1/MOD1 predicts an earlier CHF than the measurements shown; hence, the sudden expulsion of water from the test section is predicted to occur earlier [6,7]. Also, the predicted rod surface temperatures during nucleate boiling are 8–15 K below the measured ones.

(c) It was noticed that after the rod power was turned off, the slopes of the predicted and measured rod surface temperatures were different, indicating that the calculated heat transfer coefficient in this region was overpredicted. This can be seen in Fig. 20 for Exp. 5007. This was changed as we shall discuss in due course; all the boil-off runs to be reported in this report were made with the code version incorporating this change.

6.2 Model Improvements

As Figs. 18–22 show, the main problem of TRAC-BD1 is that it overpredicts the amount of water expelled in the boil-off tests. The origin of this was traced back to the rather high interfacial shear calculated by the interfacial friction correlation used for bubbly/slug flow. This correlation although appropriate for tubes, has recently been shown not to be suitable for rod bundles [9]. The interfacial shear force per unit volume f_i in TRAC-BD1 for the bubbly/slug flow regime is based on the following vapor drift velocity correlation

$$V_d = \sqrt{2}\left\{\frac{\sigma g(\rho_\ell - \rho_g)}{\rho_\ell^2}\right\}^{1/4} \tag{1}$$

Starting from this expression, it can readily be shown [8,10] that

$$f_i = \frac{\rho_\ell^2 \alpha(1-\alpha)^5}{4\sigma} \mid C_1 V_g - C_0 V_\ell \mid^3 \left(C_1 V_g - C_0 V_\ell\right) \tag{2}$$

where σ is the surface tension, g the gravity constant,

$$C_1 = \frac{1-\alpha C_0}{1-\alpha} \tag{3}$$

and $C_0 \simeq 1.3$. Based on the work of Bestion [9], we implemented in TRAC-BD1 a new bubbly/slug f_i correlation suitable for rod bundles; it is based on the following vapor drift velocity correlation

$$V_d = 0.124\left\{\frac{g(\rho_\ell - \rho_g)D_H}{\rho_g}\right\}^{1/2} \tag{4}$$

where D_H is the channel hydraulic diameter; this results in the following expression for f_i:

$$f_i = \frac{65\alpha(1-\alpha)^3\rho_g}{D_H} \mid C_1V_g - C_0V_\ell \mid \left(C_1V_g - C_0V_\ell\right) \tag{5}$$

Figures 23–27 show the comparison of the measured and calculated CLL's (with the new TRAC-BD1/MOD1 version) and peak axial power level rod surface temperature histories for the five boil-off experiments. Except in the 1 bar case, for which the code still underpredicts the CLL history, there is now excellent agreement between measurements and predictions. The code developers have already implemented this new f_i correlation in the new code version TRAC-BF1.

We could not trace the origin of the earlier transition to nucleate boiling predicted by the code; though, recent work [11] has shown that since the thermocouples are a little below the rod surface, an oxide layer having a thickness of 25 μm on the surface could in fact increase the thermocouple readings by as much as 12 K. This is a plausible explanation of the differences between measured and predicted rod surface temperatures during nucleate boiling.

Finally, the problem of overprediction of heat transfer after the power was turned off was traced back to the steam cooling logic of the code and since this problem was not encountered when analyzing the boil-off experiments with version 12 of the code [6,7], the steam cooling heat transfer logic of version 12 was re-introduced in MOD1.

Specifically, for steam cooling, in MOD1 the following heat transfer coefficients are defined [8]

$$h_{v,lam} = \frac{4k_g}{D_H} \tag{6}$$

$$h_{v,turb} = 0.023(Re_\ell)^{0.8}(Pr)^{0.33}K_g/D_H \tag{7}$$

$$h_{v,nc} = 0.13(Gr.Pr)_g^{0.333}/D_H \tag{8}$$

where all the symbols have their usual meaning and

$$Gr_v = \frac{\rho_g g \mid T_w - T_g \mid}{\mu_g^2 T_g} D_H^3 \tag{9}$$

In MOD1, the following selection logic exists for h_v if $\alpha > 0.999$:

$$h_v = h_{v,tur} \tag{10}$$

$$If \quad h_v < h_{v,lam}, h_v = h_{v,lam} \tag{11}$$

$$If \quad h_v < h_{v,nc}, h_v = h_{v,nc} \tag{12}$$

This was modified as follows (for $\alpha > 0.999$):

$$h_v = \mathrm{MAX}\{h_{v,nc}, h_{v,tur}\} \tag{13}$$

Comparison between measured and predicted peak axial power rod surface temperature histories for Exp. 5006 are shown in Fig. 28a (standard MOD1) and 28b (modified as above).

Based on the experience gained from TRAC-BD1 modifications, the bubbly and slug flow regime interfacial friction correlation used in CATHARE code for bundle geometries was implemented into RELAP5/MOD2. As it can be seen from figures 16 and 17, the results of the calculated entrained water and cladding surface temperature are very well comparable with the experimental data of experiment 5007.

7 Conclusions

The NEPTUN experiments have provided thermal-hydraulic data simulating nuclear reactor core boil-off conditions at low pressure (1–5 bar). The data obtained from these tests proved to be useful in assessing the modeling capability of available computer codes.

Analysis of the experimental boil-off data indicate that:

- increasing core power resulted in more rapid boil-off and cladding heat-up, while decreasing power resulted in the opposite trends, as expected
- the lower system pressures resulted in more rapid decrease of liquid levels and faster rod dry-outs relative to the base case
- dry-out times of the internal and external surface thermacouples were within 10 seconds of each other at any axial elevation for all rods in the bundle. The cladding external surface thermocouples measure the cladding temperatures that would have been measured in their absence within 0 to -20 K.

Analysis of a number of NEPTUN boil-off experiments and comparisons with TRAC-BD1/MOD1 predictions showed that:

- the collapsed liquid level history is underpredicted and consequently, CHF occurs earlier than in the experiments. Clearly, TRAC overpredicts the amount of water expelled from the test section.

- Generally, earlier incipience of nucleate boiling is predicted; and recent investigations indicate that differences between measured and predicted rod surface temperatures during nucleate boiling can be due to the formation of an oxide layer around the electrical heater rods.

- After the rod power was turned off, the slopes of the predicted and measured rod surface temperatures were different, indicating that the calculated steam cooling heat transfer coefficient was overpredicted.

To improve the prediction capability of TRAC-BD1/MOD1 the following main modifications were introduced:

- An alternative bubby/slug interfacial shear correlation, more appropriate for bundles and used in the CATHARE code, is implemented in the code. As a result of this change, the collapsed liquid level histories are correctly predicted by decreasing the interfacial friction in this flow regime.

- The steam cooling heat transfer logic used in version 12 is re-introduced in MOD1, specifically to eliminate the differences during the steam cooling phase after the power was turned off.

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Table 1: Summary of NEPTUN Boil-Off Experiments

Experiment Number	Bundle Power (kW)	Rod Peak Power (kW/m)	System Pressure (bar)	Initial Coolant Temperature (°C)	Initial Subcooling (K)	Comments
5000	24.6	0.744-4.6	1	100	0	Power was too high for the first test; the data were not evaluated from this experiment
5001	24.6	0.744	1	100	0	Repeat experiment at low pressure
5002	24.6	0.744	1	100	0	Repeat experiment at low pressure
5004	24.6	0.744	5	120	32	Effects of changing rod power and initial coolant subcooling
5005	42.1	1.276	5	120	32	Effects of changing rod power and initial coolant subcooling
5006	42.1	1.276	5	140	12	Effects of chaning rod power at high system pressure
5007	24.6	0.744	5	140	12	Effects of changing rod power at high system pressure
5008	10.5	0.319	5	140	12	Effects of changing rod power at high system pressure
5009	42.1	1.276	5	140		Repeat of high-power, high-pressure experiment 5006 with higher data-scanning rate
5011	75.1	2.276	5	112	38	High power test
5012	42.1	1.276	5	steam	steam	Adiabatic heat-up test and, from 345 to 470 s flooding with flooding velocity of 15 cm/s and coolant temperature of 74°C

	EXPERIMENT NUMBER				
	5002	5006	5007	5008	5011
PRESSURE (BAR)	1	5	5	5	5
SUBCOOLING (^{o}K)	0	12	12	12	39
BUNDLE POWER (kW)	24.6	42.1	24.6	10.5	75.1

TABLE 2

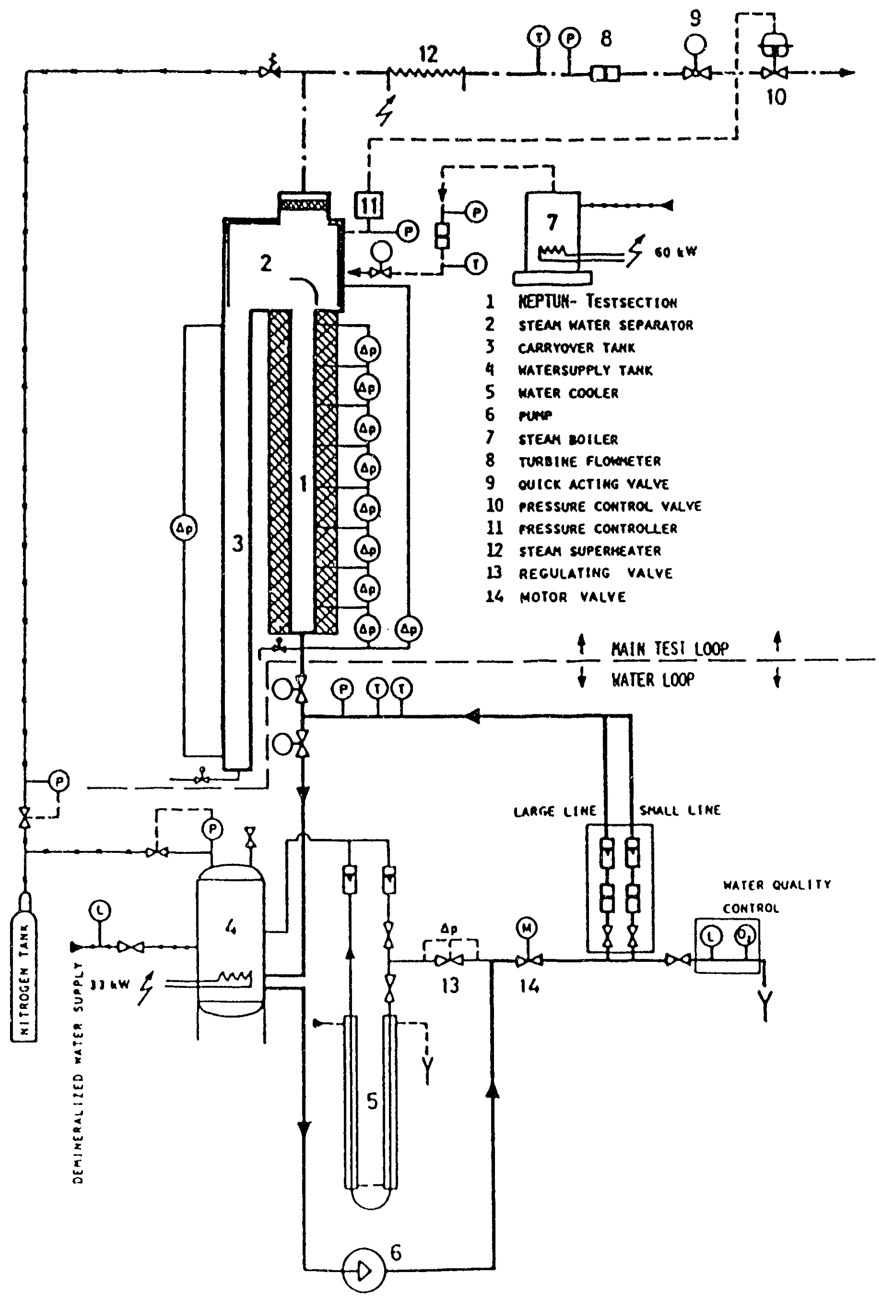


Figure 1: Simplified NEPTUN Flow Diagram

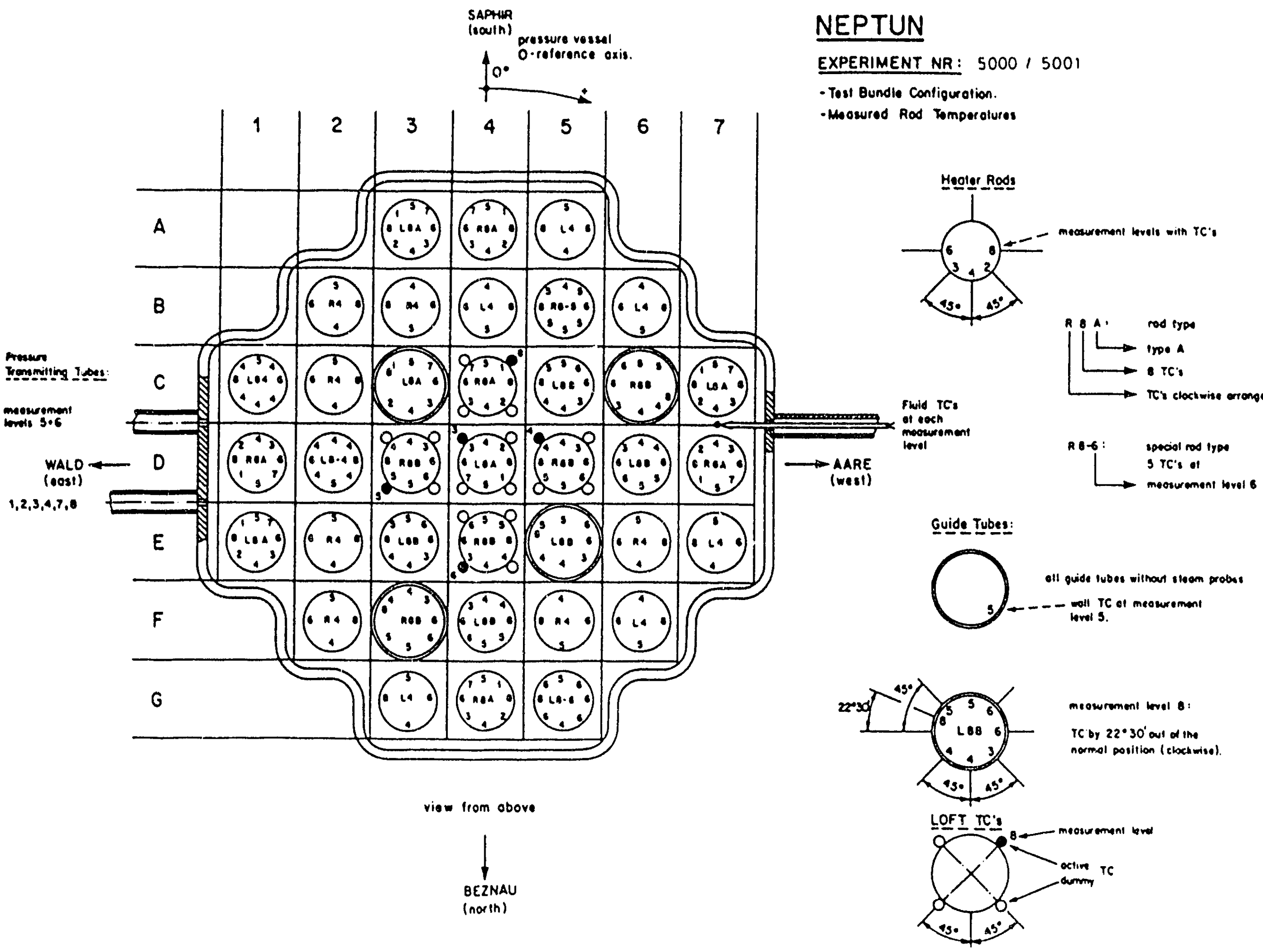


Figure 2: NEPTUN Bundle Cross-Section

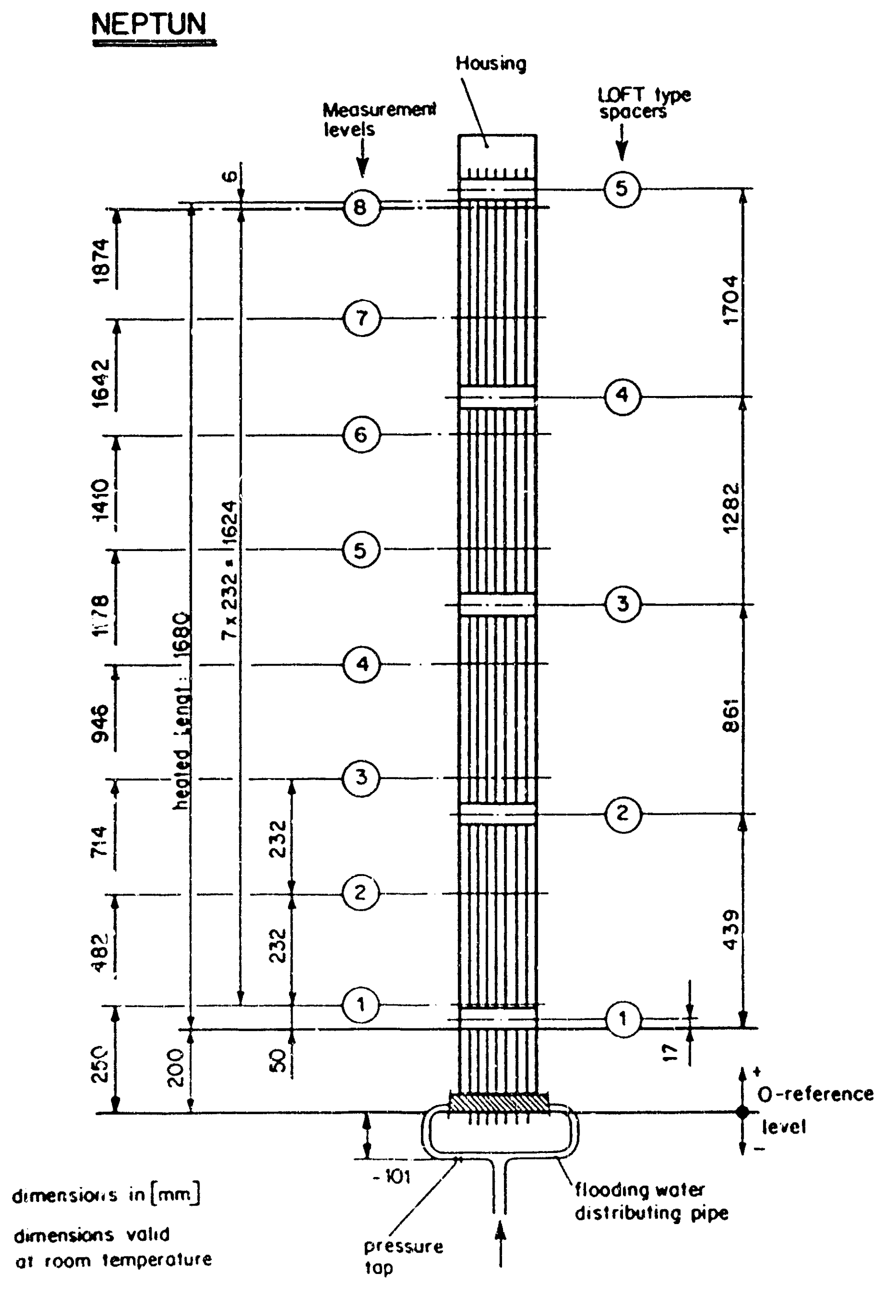


Figure 3: Axial Distribution of the Measurement Levels and the Spacers

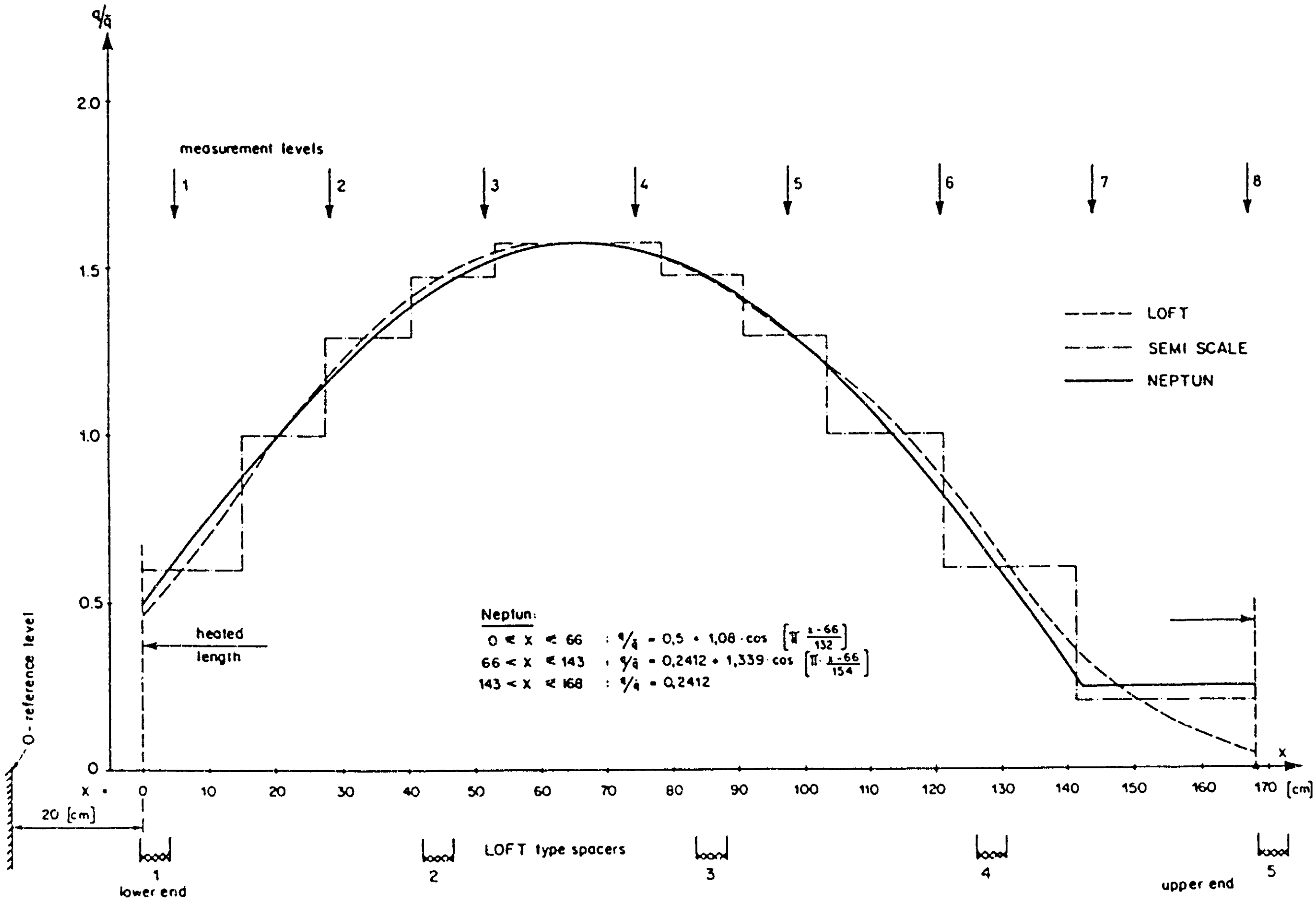


Figure 4: NEPTUN: Axial Power Distribution of the Heater Rods

- HEATED LENGTH 1680 MM
- O. DIAMETER 10.72 MM
- POWER DISTRIBUTION CHOPPED COSINE
- ROD POWER 4.5 KW (MAX)
- AVERAGE HEAT FLUX 7.86 W/CM^2
- PEAK HEAT FLUX 12.4 W/CM^2
- PEAK LINEAR HEAT RATING 41.9 W/CM
- AXIAL PEAKING FACTOR 1.58

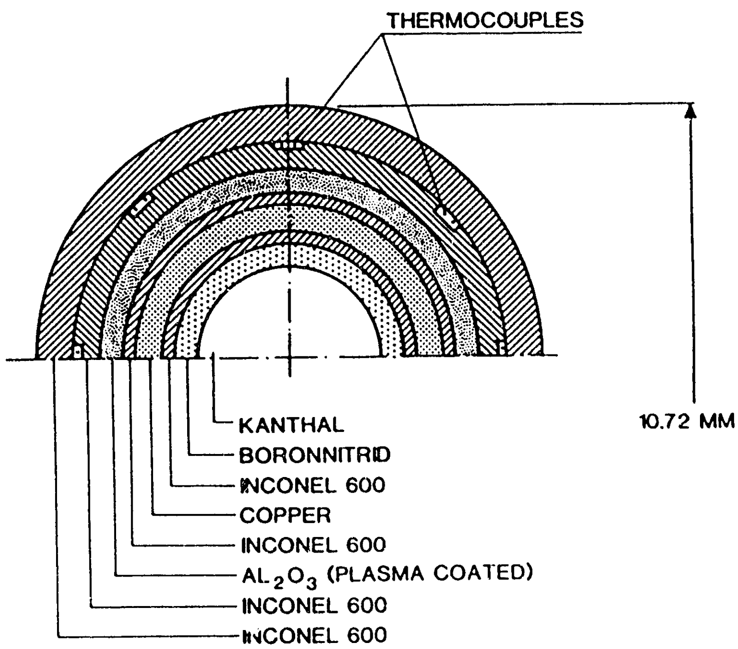


Figure 5: NEPTUN: Electrical Heater Rods

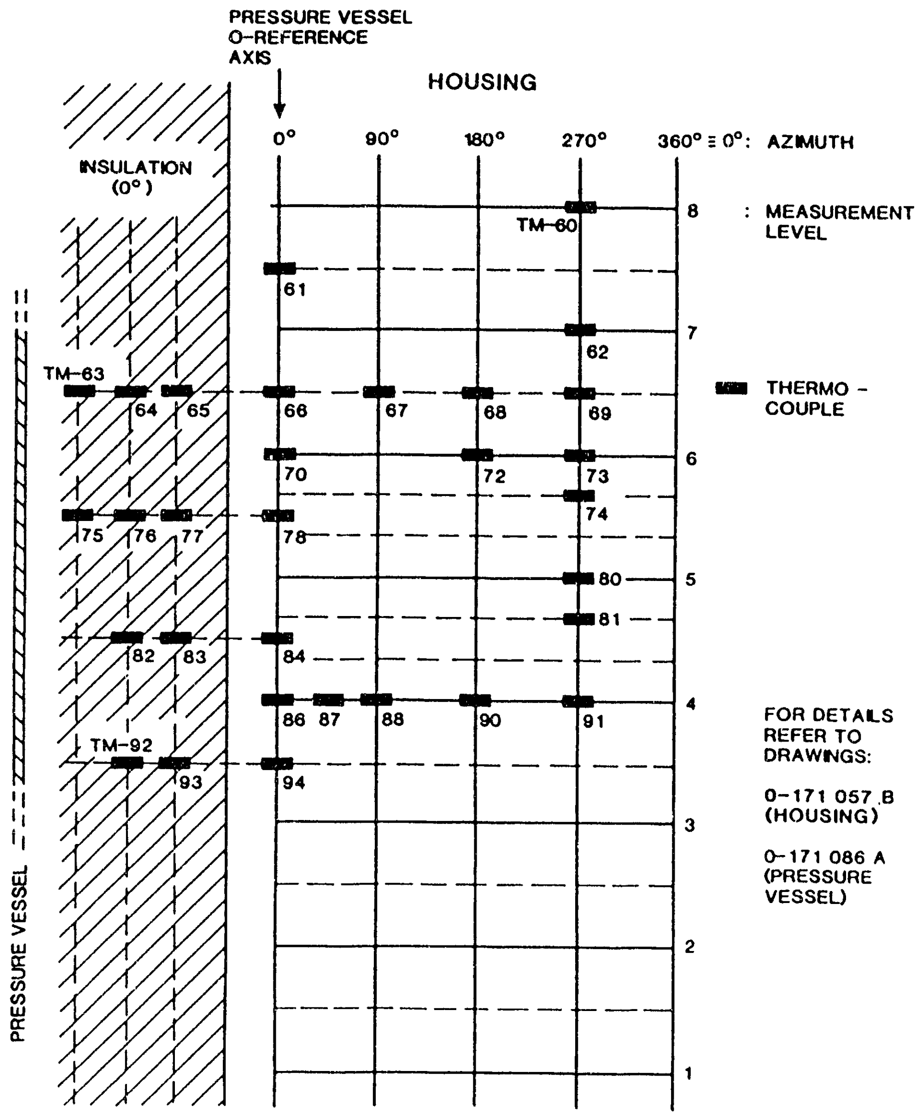


Figure 6: NEPTUN (Experiment no. 5000–5007): Measured temperatures at the outer surface of the housing and within the insulation

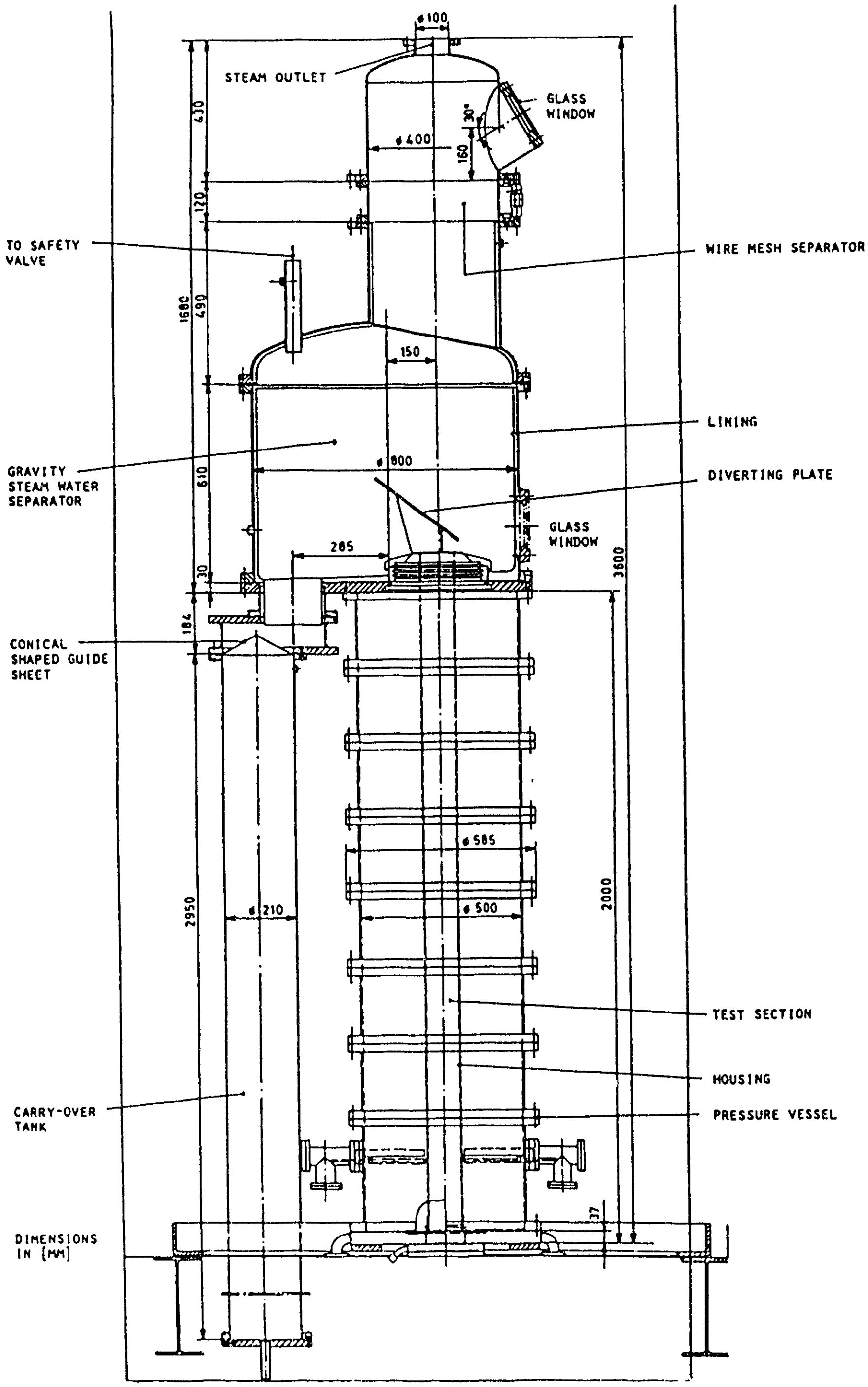


Figure 7: NEPTUN: Steam Water Separator and Carry-Over Tank

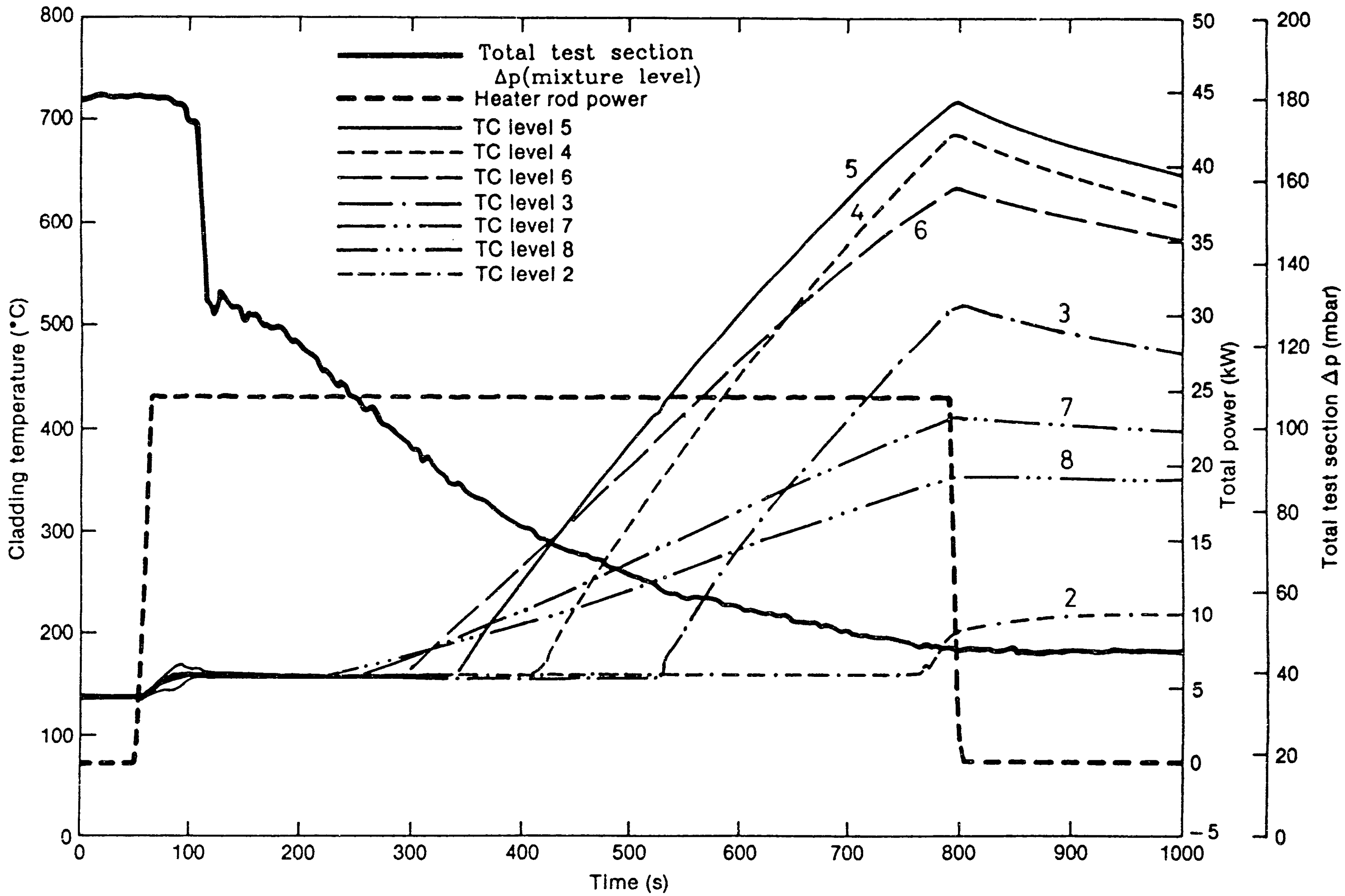


Figure 8. NEPTUN system response--base case (test 5007--high system pressure, intermediate rod power).

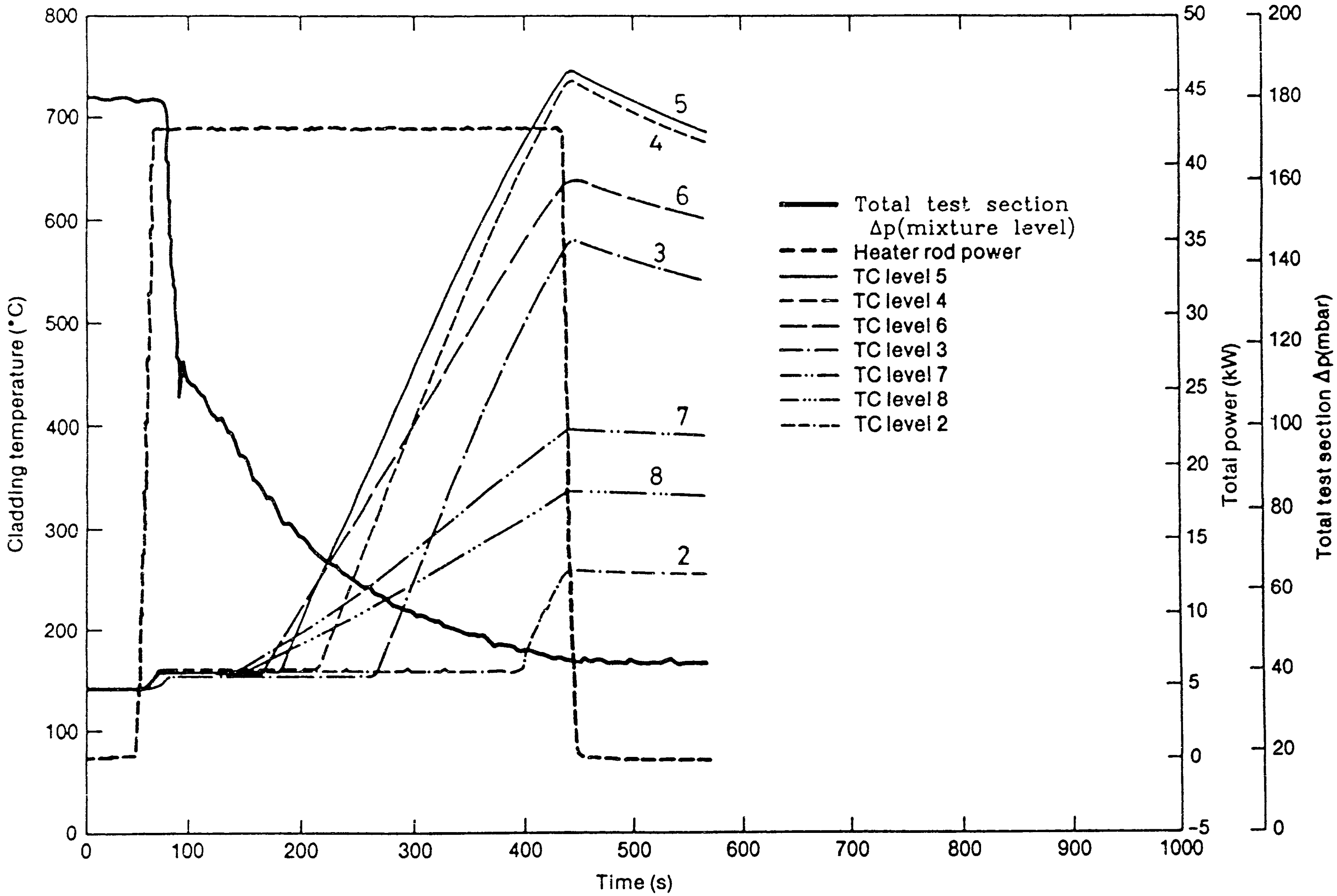


Figure 9. NEPTUN system response--test 5006--high system pressure, high rod power.

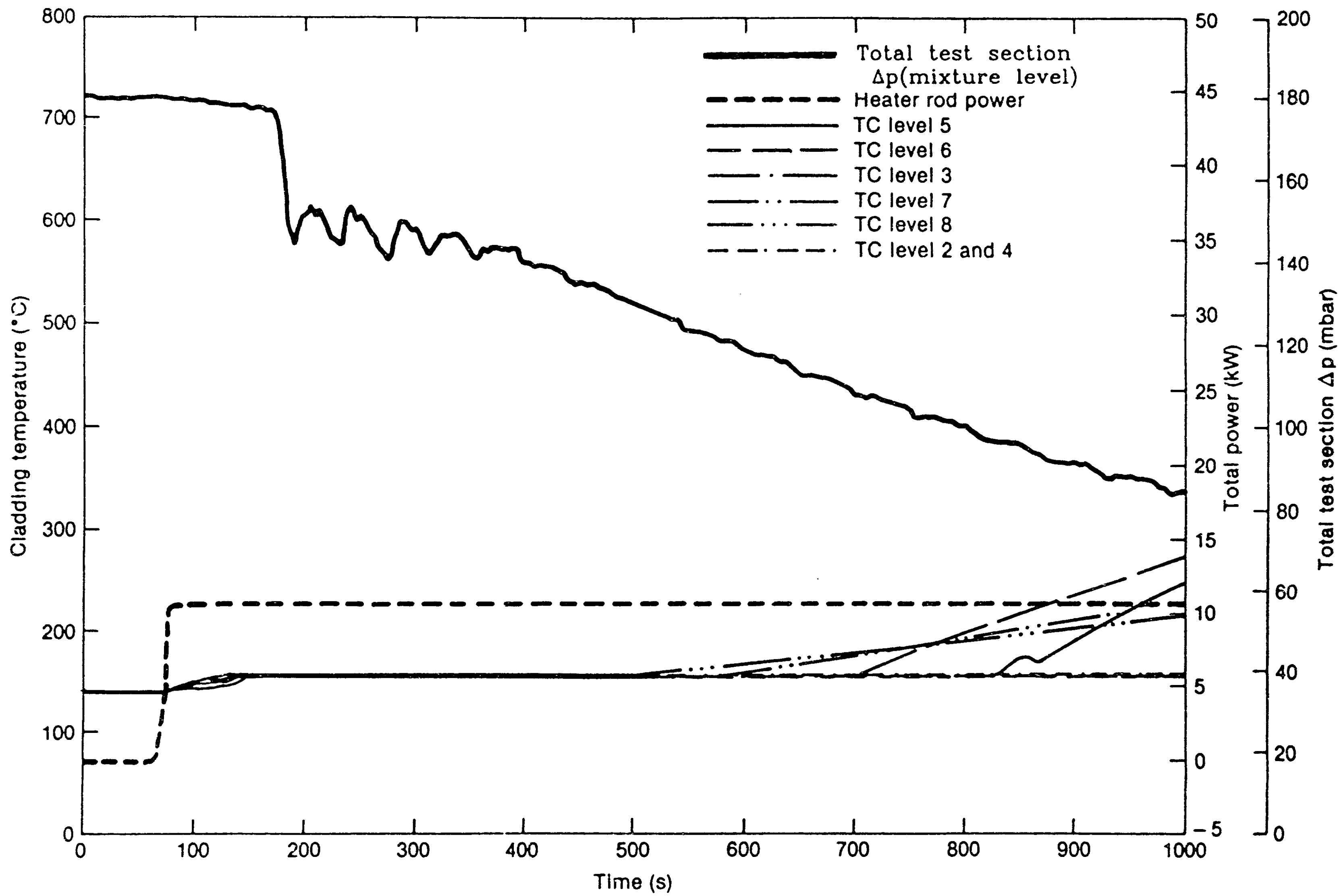


Figure 10. NEPTUN system response--test 5008--high system pressure, low rod power.(Figure continued)

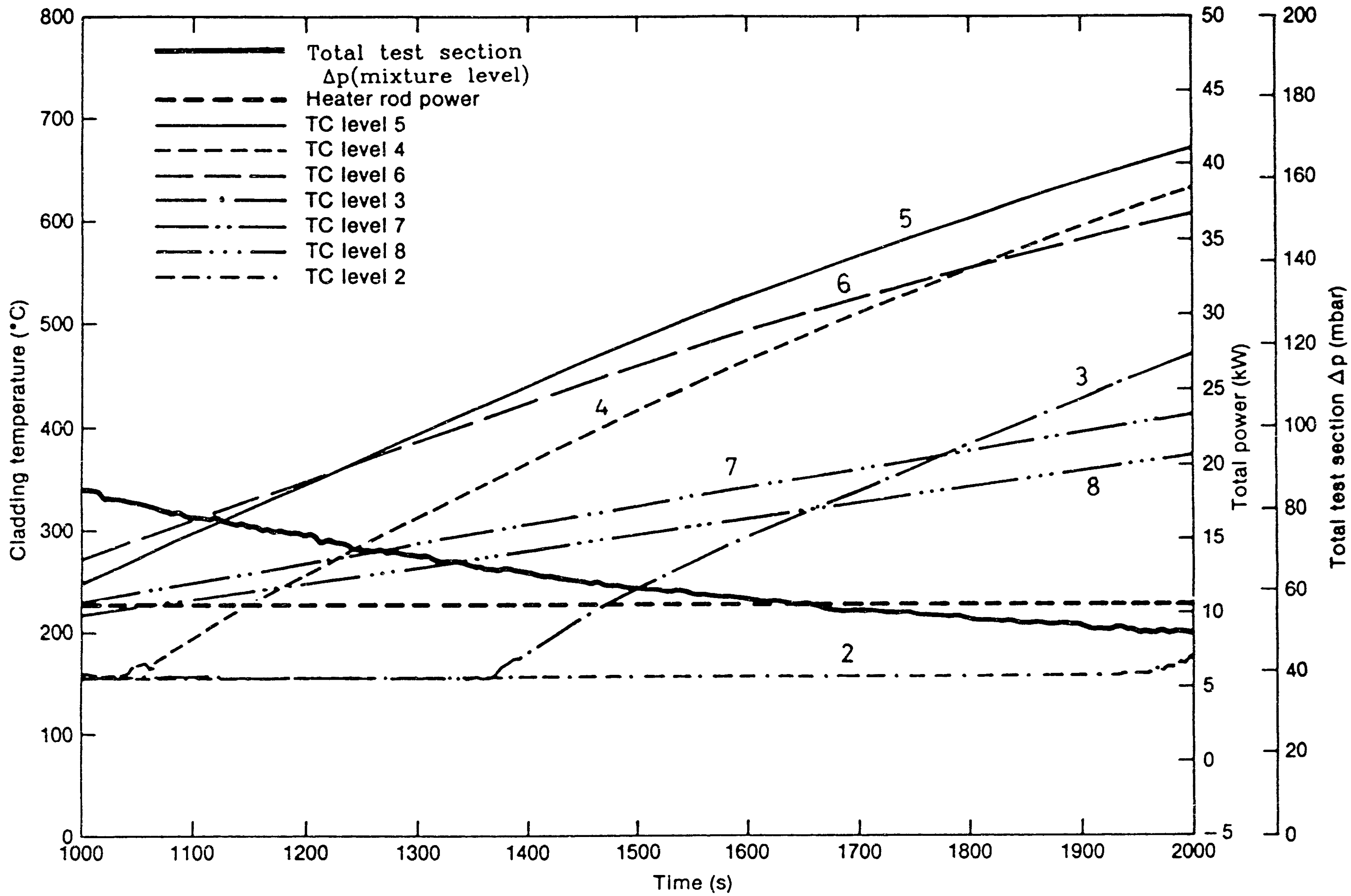


Figure 10. (continued).

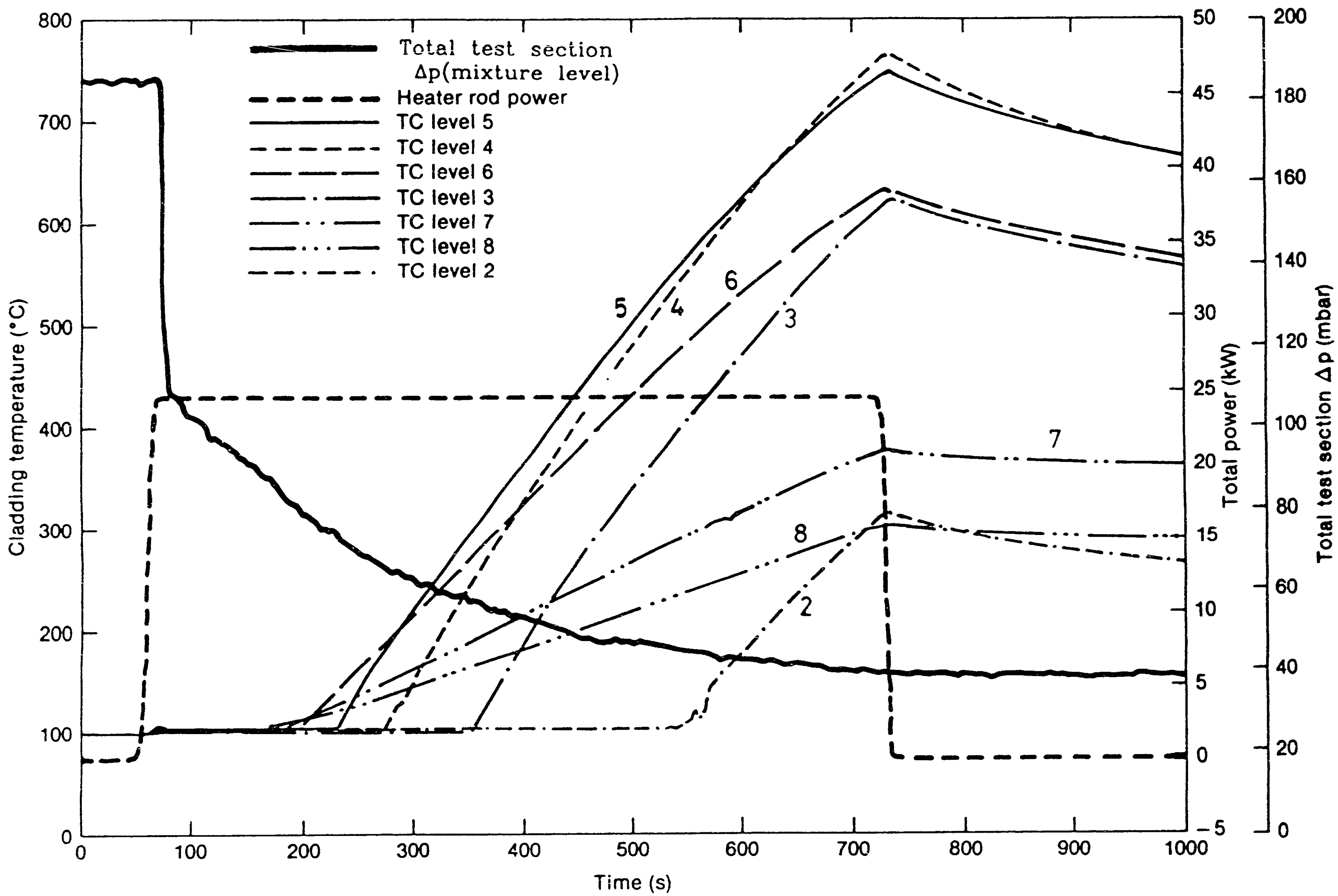


Figure 11. NEPTUN system response--test 5002--low system pressure, intermediate rod power.

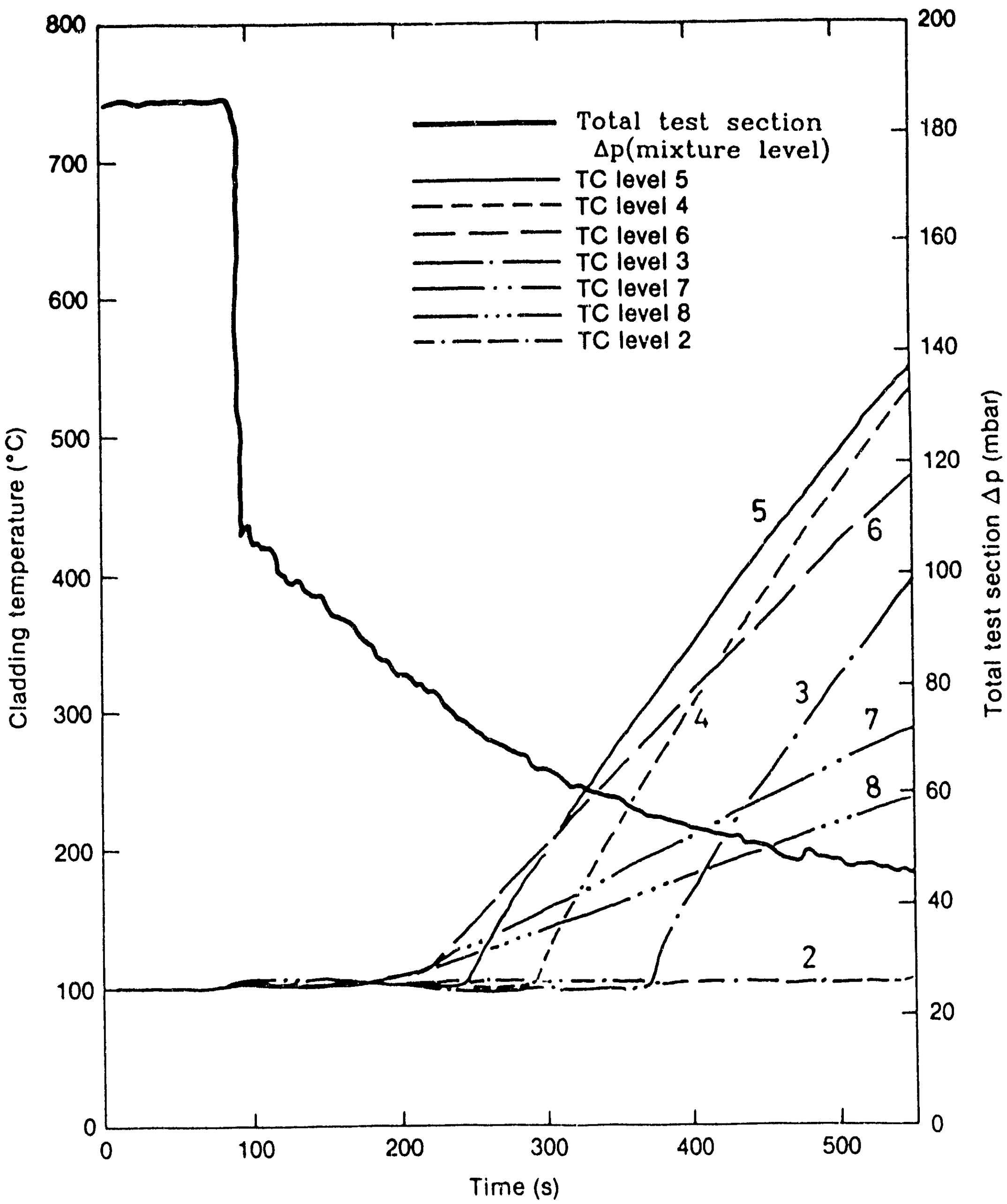


Figure 12. NEPTUN system response--test 5001--low system pressure, intermediate rod power.

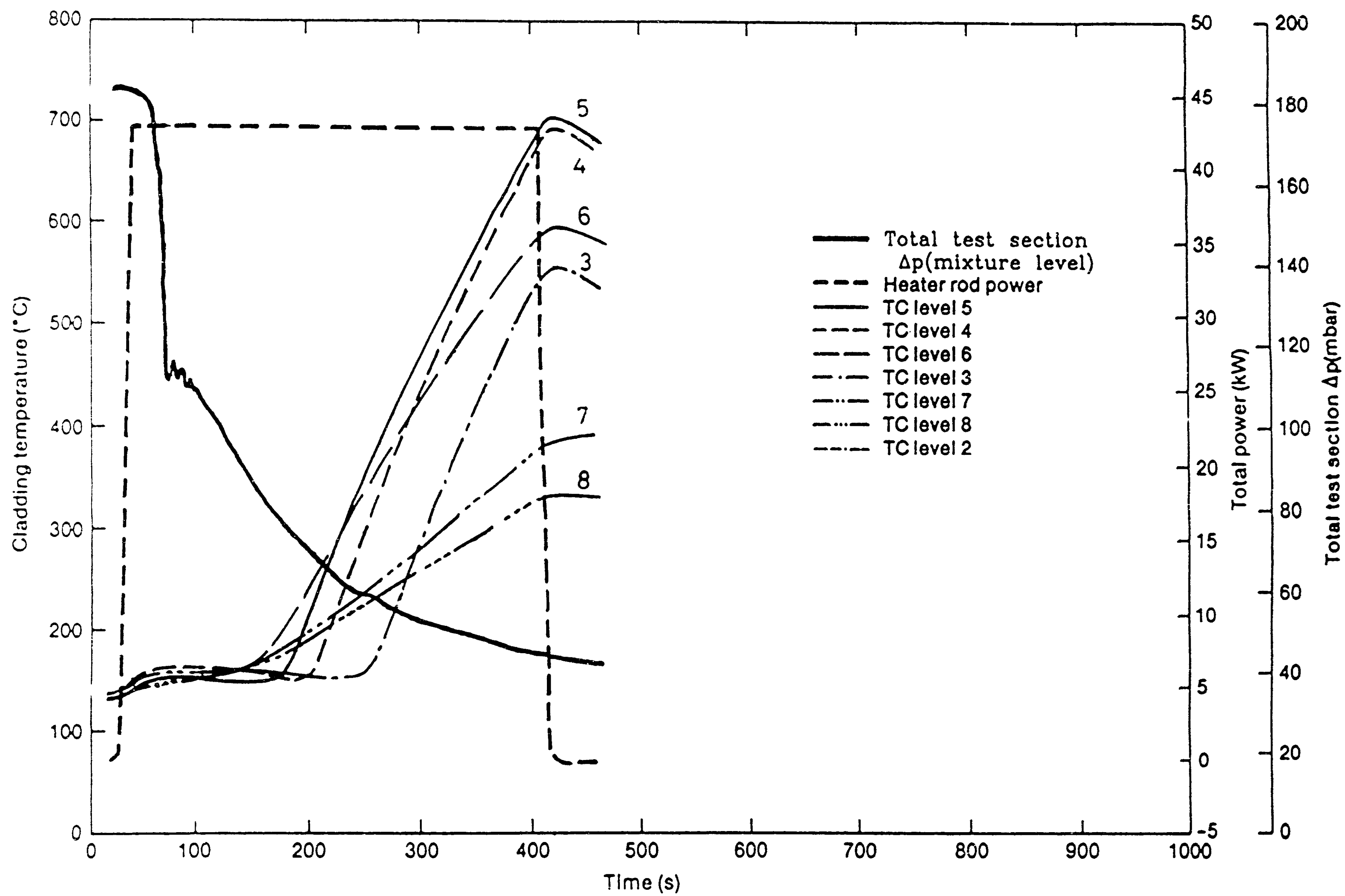


Figure 13 : NEPTUN system response - test 5009 - High system pressure, high rod power, repeat test

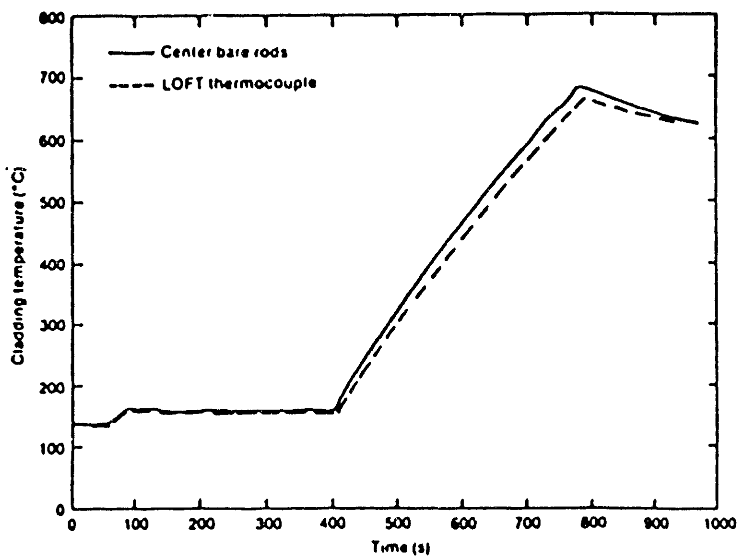


Figure 14. Comparison of center rod internal and LOFT thermocouples (test 5007, axial elevation, 946 mm -level 4).

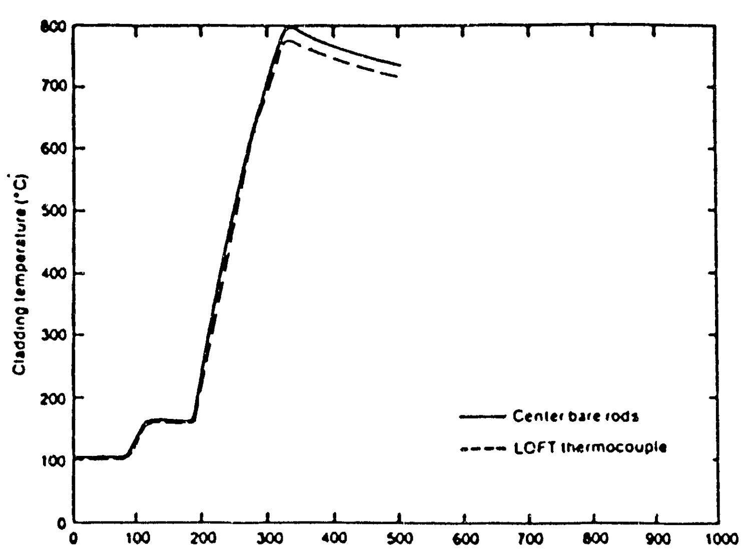


Figure 15. Comparison of center rod internal and LOFT thermocouples (test 5011, axial elevation, 946 mm -level 4).

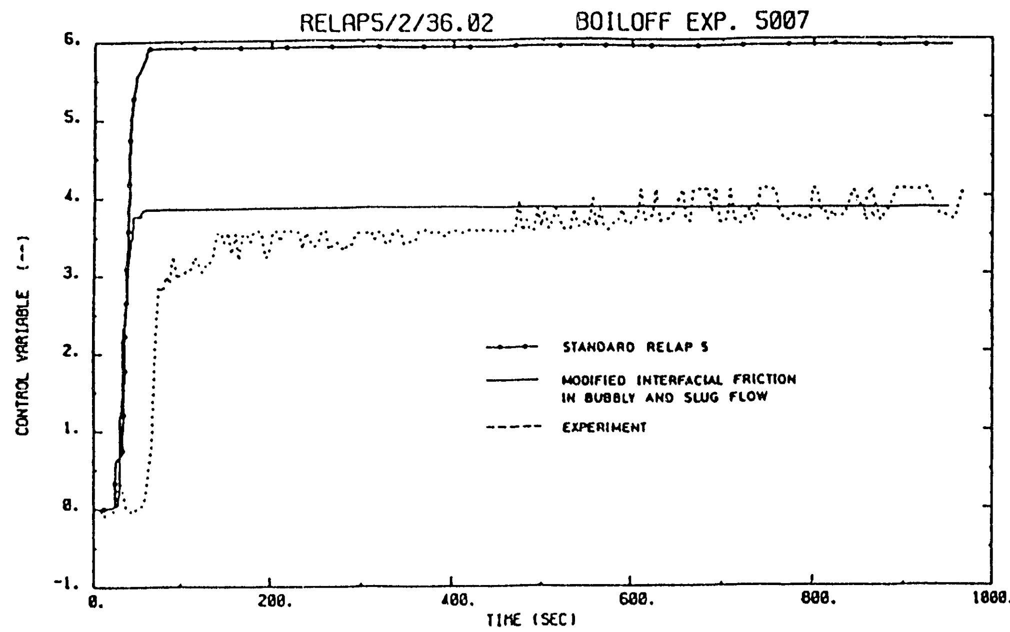


Figure 16: Water entrainment in NEPTUN boil-off experiment 5007, calculated by RELAP5/MOD2, with and without new correlation for the interfacial friction in bubbly and slug flow.

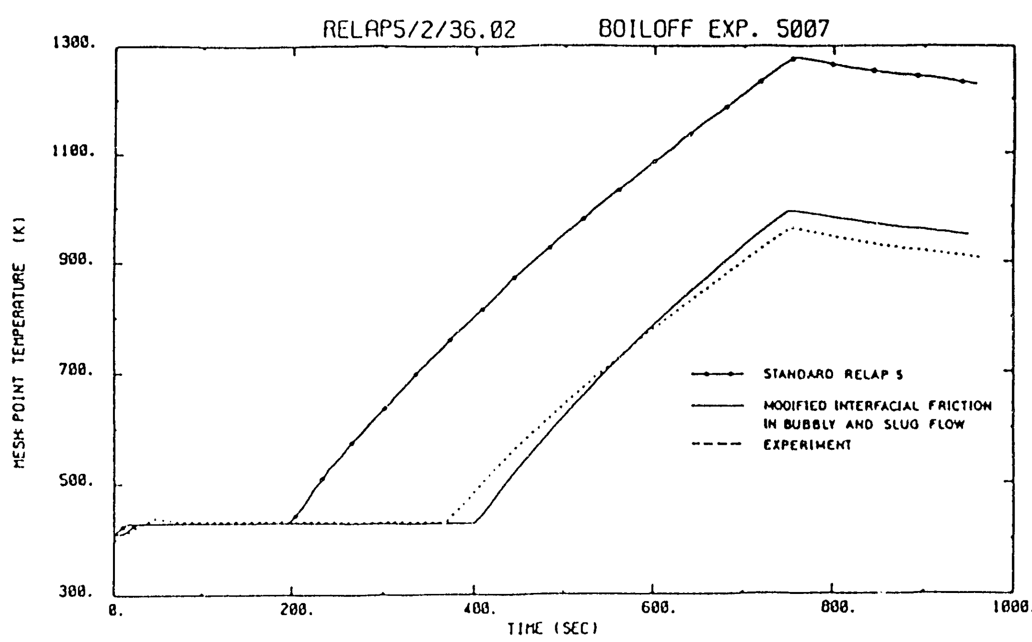


Figure 17: Rod cladding temperature at measurement level 4 in NEPTUN boil-off experiment 5007, calculated by RELAP5/MOD2, with and without new correlation for the interfacial friction in bubbly and slug flow.

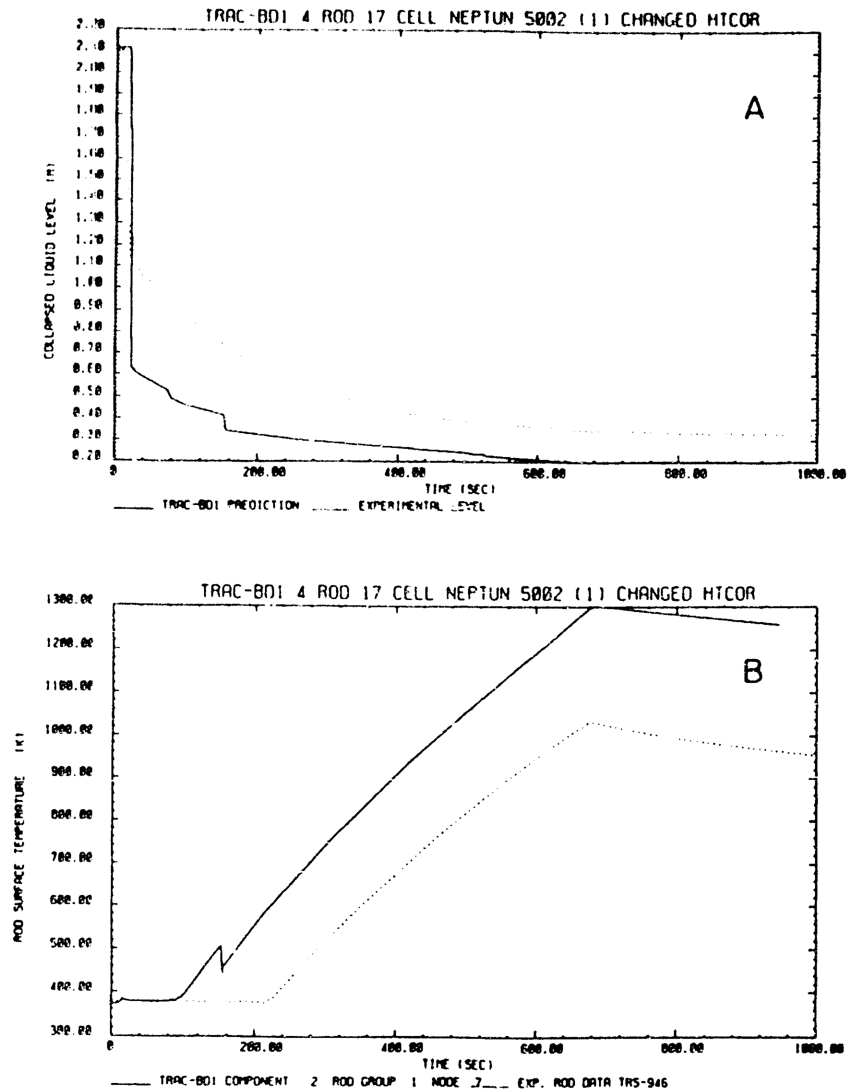


Fig. 18 : Comparison of measured and calculated collapsed liquid level and peak axial power level rod surface temperature histories, using frozen version of TRAC-BD1 for NEPTUN boil-off experiment 5002.

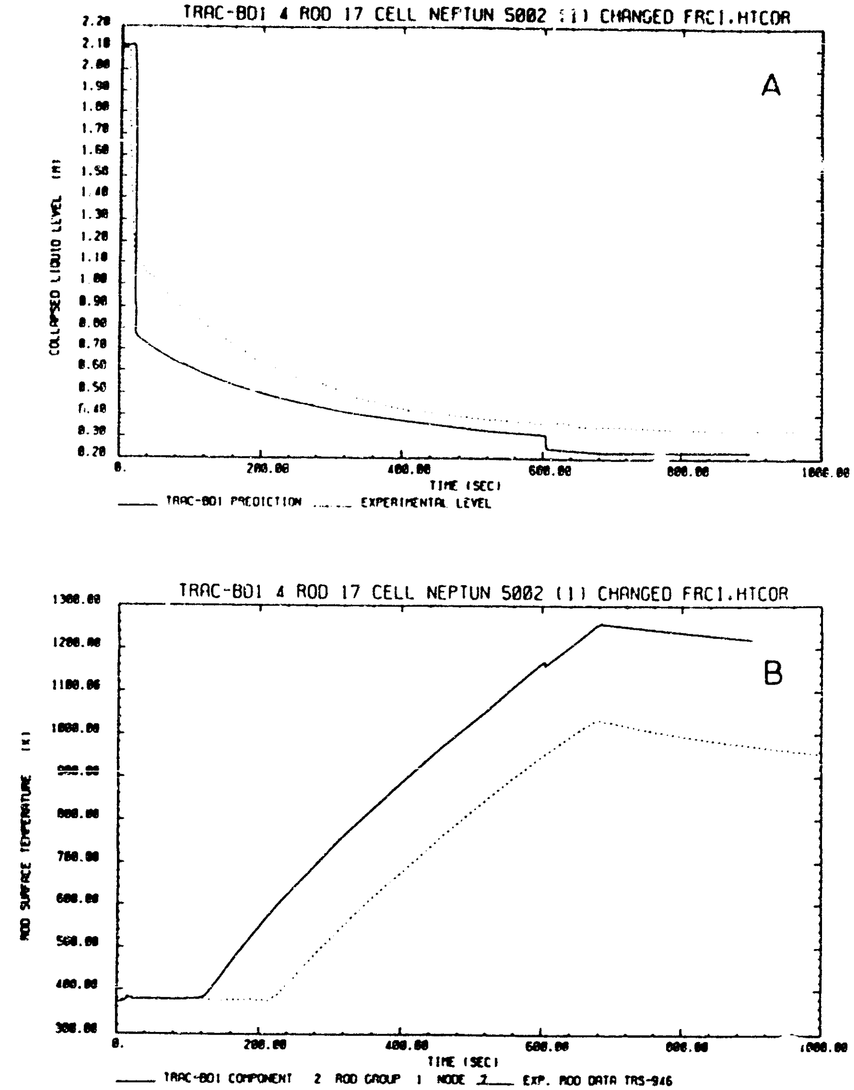


Fig. 23 : Comparison of measured and calculated collapsed liquid level and peak axial power level rod surface temperature histories, using frozen version of TRAC-BD1 for NEPTUN boil-off experiment 5002.

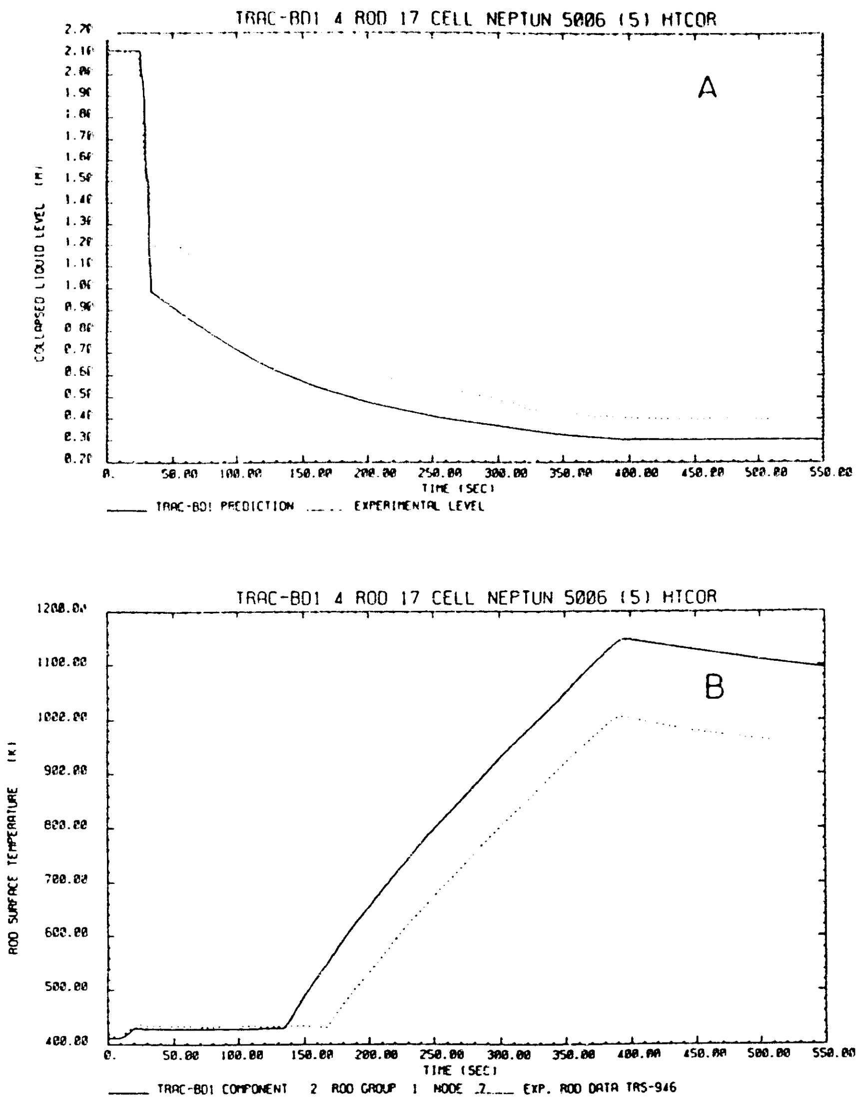


Fig. 19: Comparison of measured and calculated collapsed liquid level and peak axial power level rod surface temperature histories, using frozen version of TRAC-BD1 for NEPTUN boil-off experiment 5006.

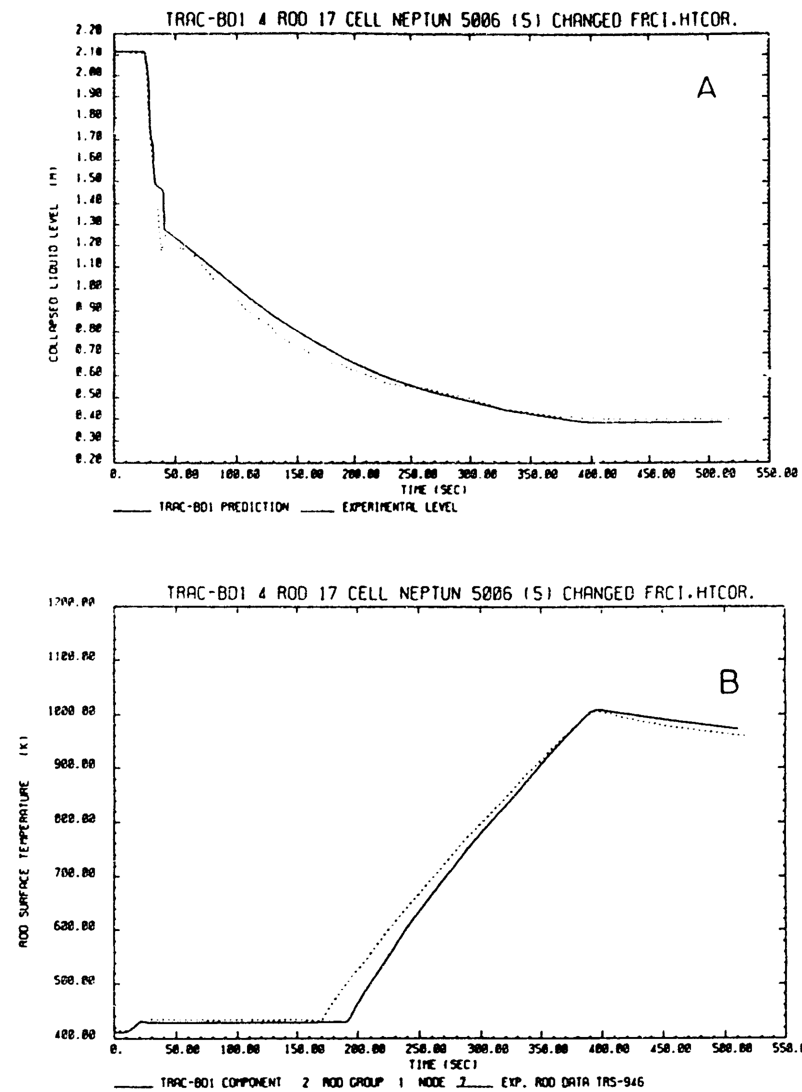


Fig. 24: Comparison of measured and calculated collapsed liquid level and peak axial power level rod surface temperature histories, using frozen version of TRAC-BD1 for NEPTUN boil-off experiment 5006.

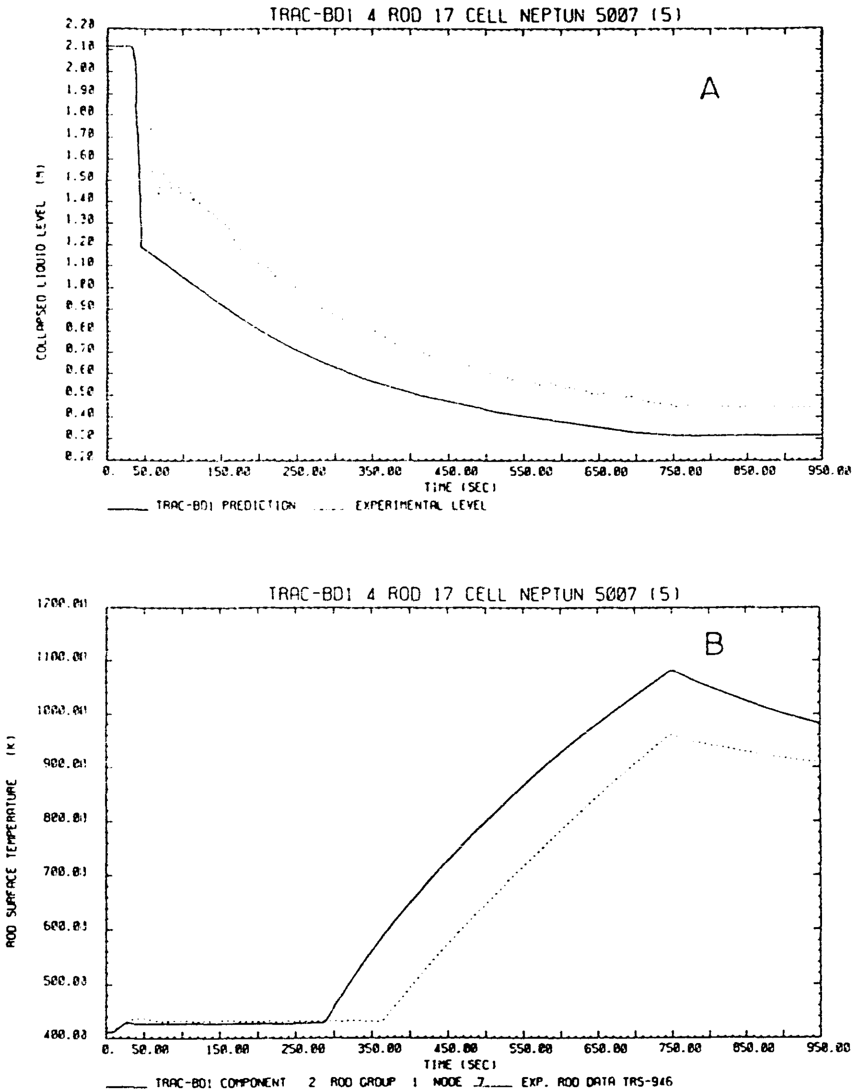


Fig. 20: Comparison of measured and calculated collapsed liquid level and peak axial power level rod surface temperature histories, using frozen version of TRAC-BD1 for NEPTUN boil-off experiment 5007.

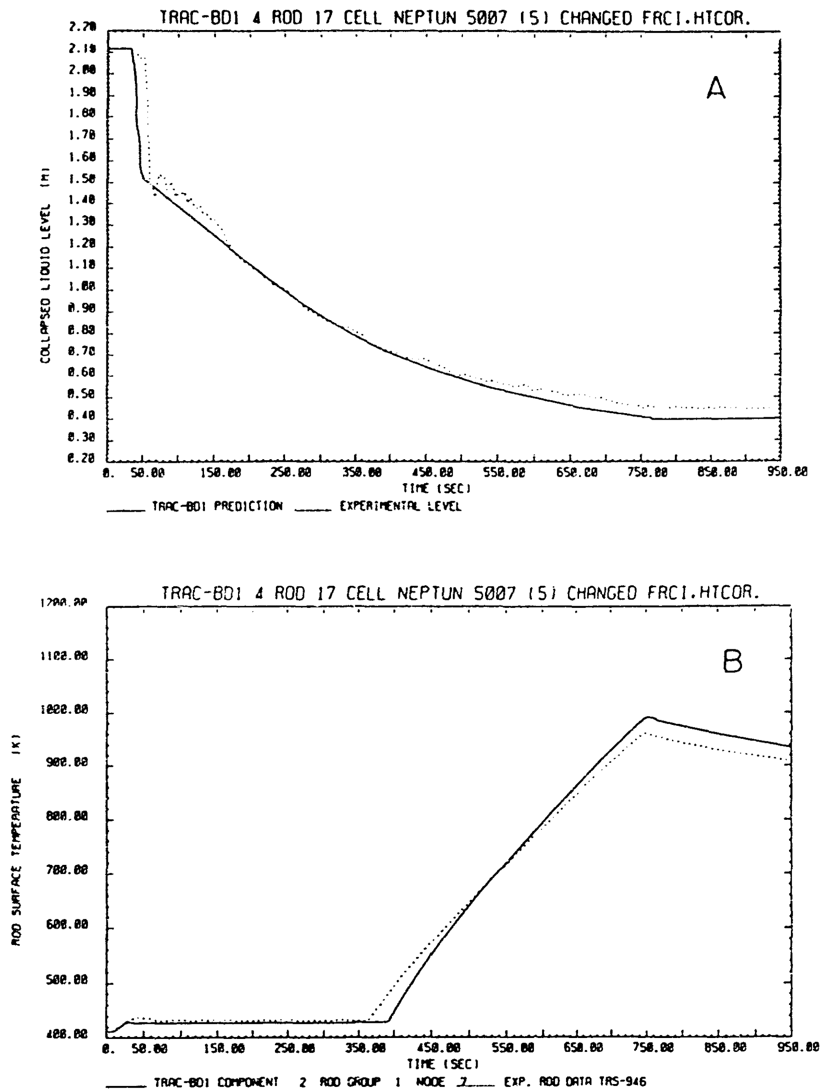


Fig. 25: Comparison of measured and calculated collapsed liquid level and peak axial power level rod surface temperature histories, using frozen version of TRAC-BD1 for NEPTUN boil-off experiment 5007.

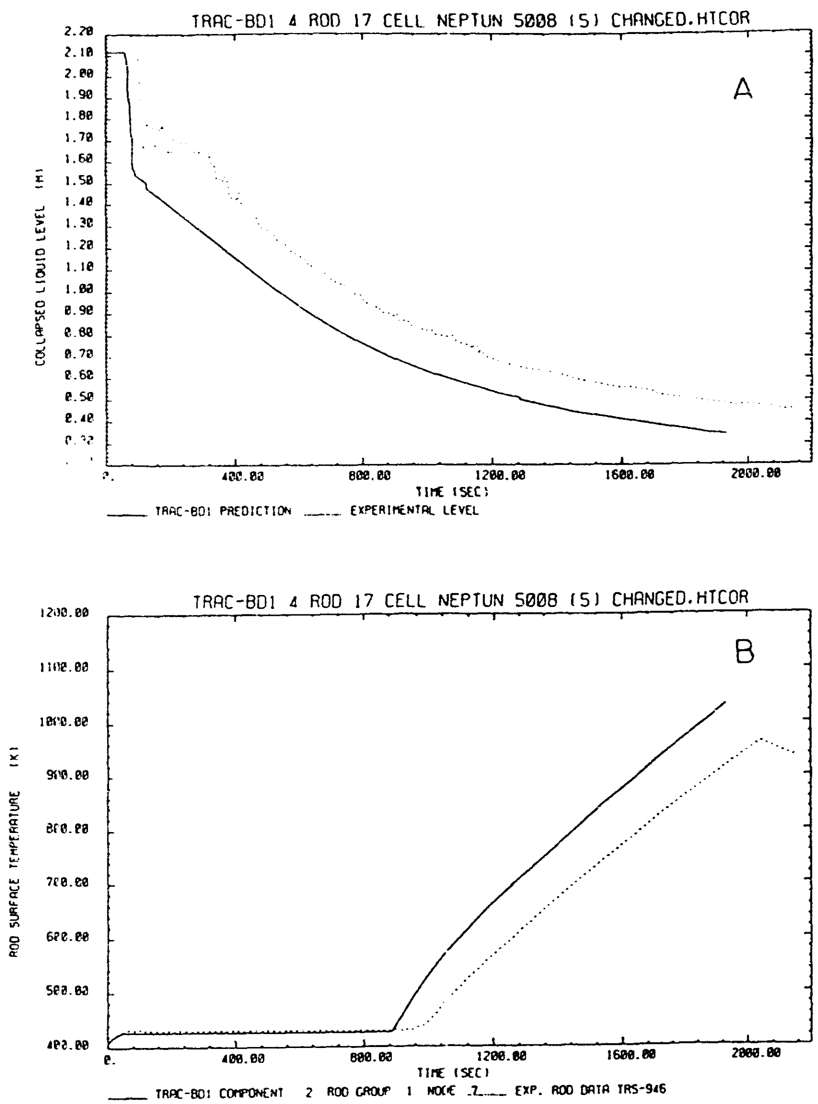


Fig. 21: Comparison of measured and calculated collapsed liquid level and peak axial power level rod surface temperature histories, using frozen version of TRAC-BD1 for NEPTUN boil-off experiment 5008.

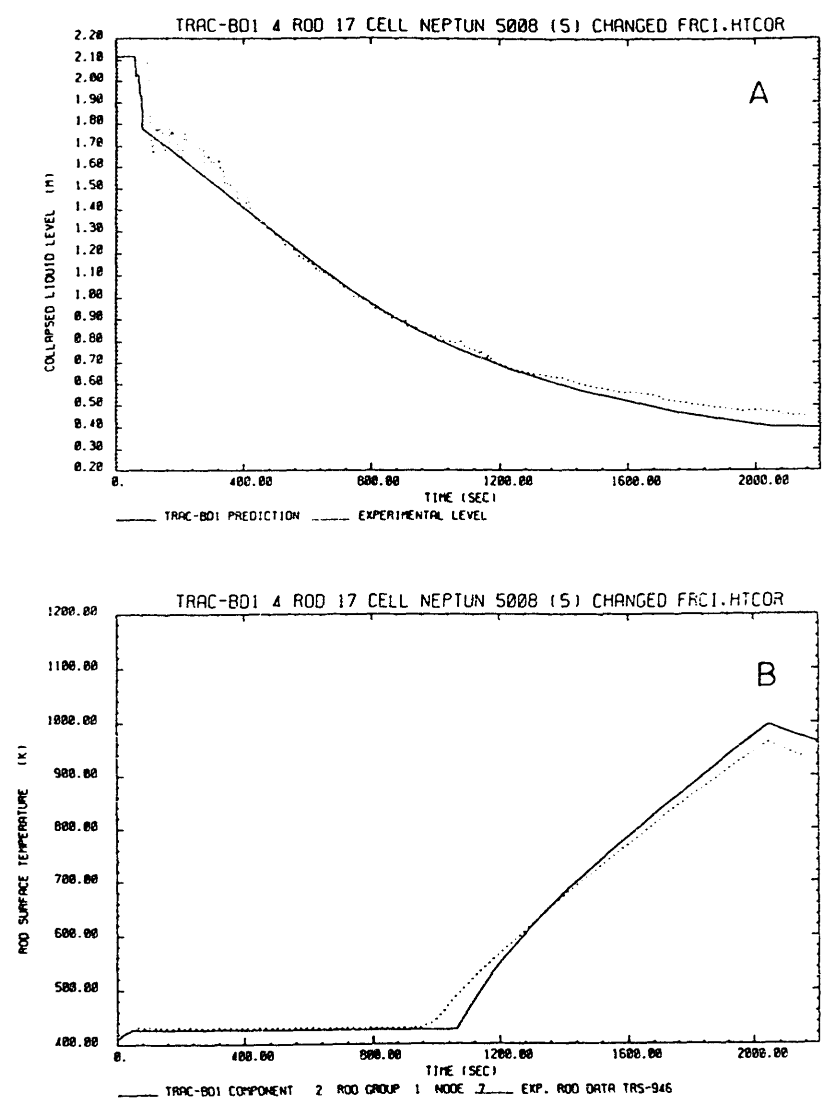


Fig. 26: Comparison of measured and calculated collapsed liquid level and peak axial power level rod surface temperature histories, using frozen version of TRAC-BD1 for NEPTUN boil-off experiment 5008.

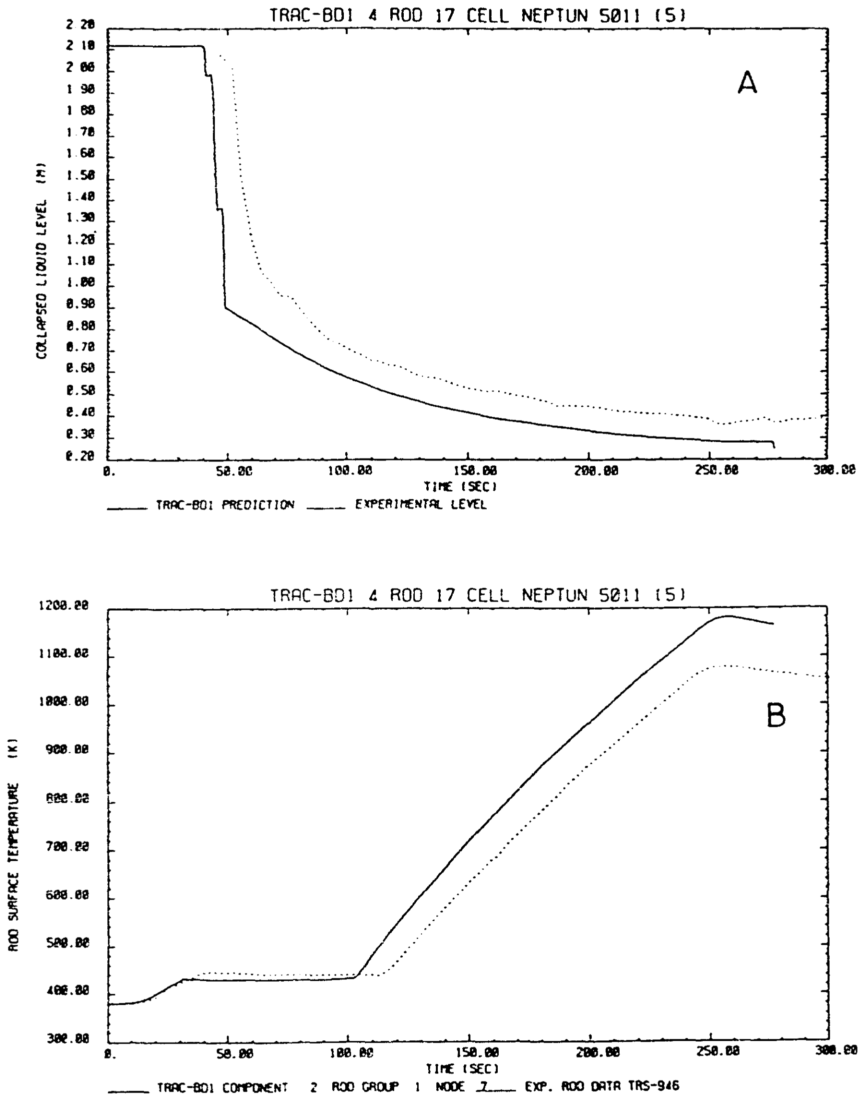


Fig. 22 : Comparison of measured and calculated collapsed liquid level and peak axial power level rod surface temperature histories, using frozen version of TRAC-BD1 for NEPTUN boil-off experiment 5011.

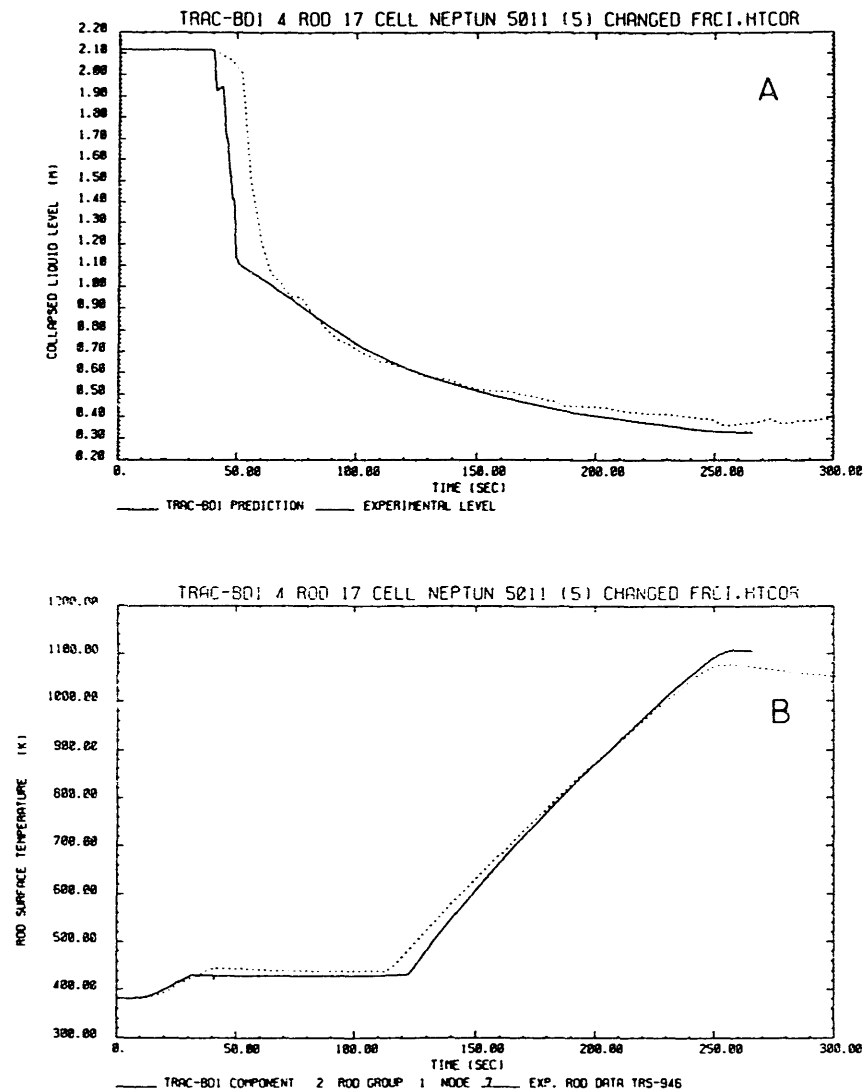
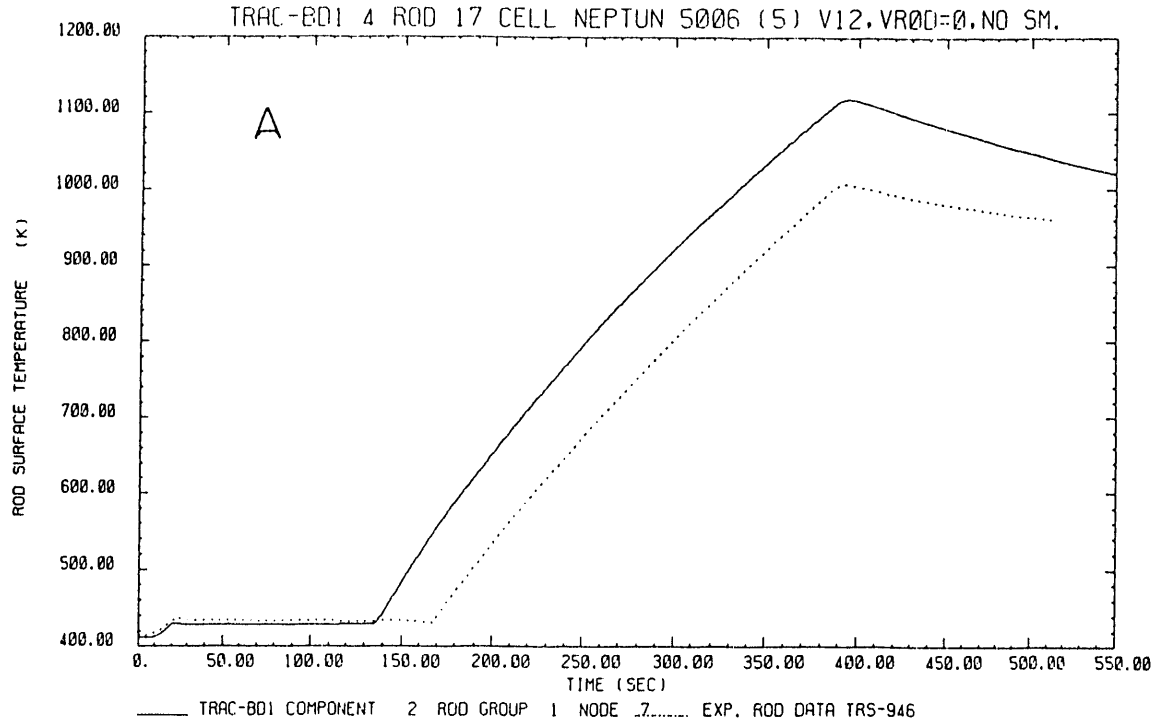


Fig. 27 : Comparison of measured and calculated collapsed liquid level and peak axial power level rod surface temperature histories, using frozen version of TRAC-BD1 for NEPTUN boil-off experiment 5011.



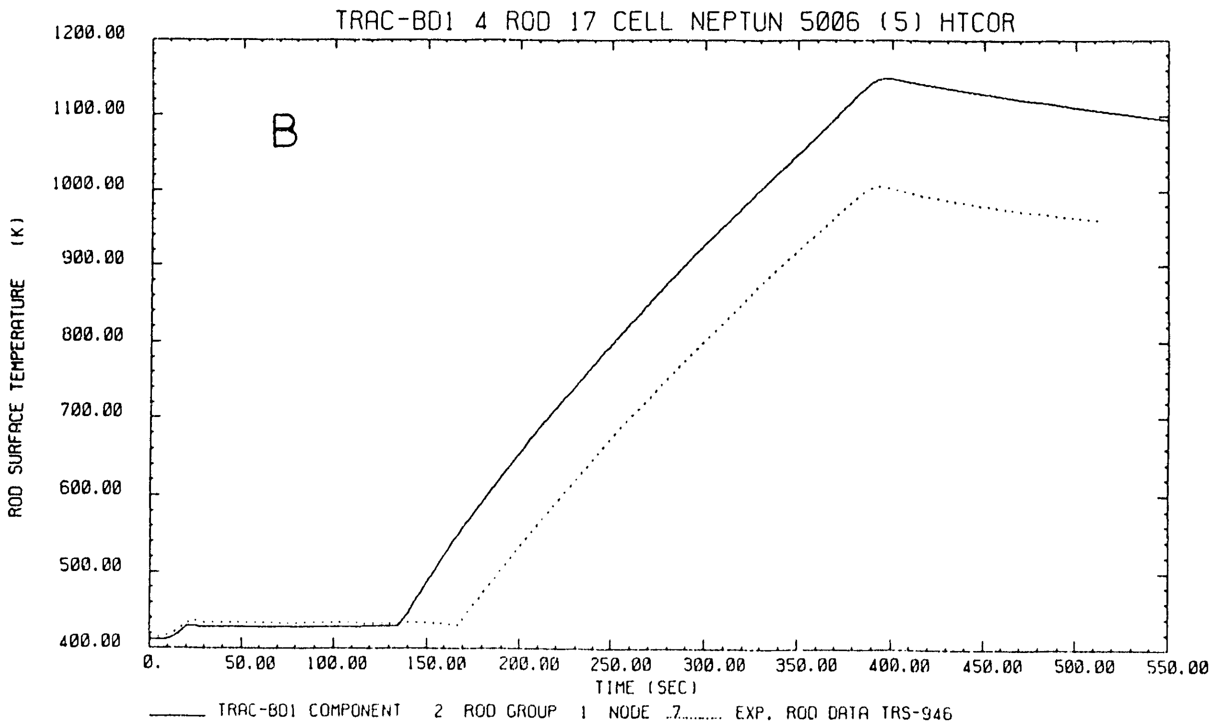


Fig. 28: Rod surface temperature histories at peak axial power level in NEPTUN boil-off experiment 5006 calculated by TRAC-BD1, (a) frozen version, (b) modified for steam cooling.

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