NUREG/IA-0057

International Agreement Report

Assessment Study of RELAP5/MOD2 Cycle 36.04 Based on the Commissioning Test Reactor Trip at Full Load at the Philippsburg 2 Nuclear Power Plant

Prepared by
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Office of Nuclear Regulatory Research
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Abstract

The commissioning test "Reactor Trip at Full Load", which was performed at the nuclear power plant Philippsburg 2 (KKP 2), was recalculated with RELAP5/MOD2. The comparison of the results with the commissioning test results shows very good agreement between measurement and calculation. Difficulties arised attempting to adjust the RPV inlet temperature, which depends on the steam generator pressure, to the initial test condition. It is assumed that the heat transfer correlations in RELAP5/MOD2 are not optimized for this problem. The deviation of SG water level during the transient between calculation and measurement is assumed to be caused by the separator model in RELAP5/MOD2.

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Ececutive Summary

The commissioning test "Reactor Trip at Full Load" which was performed at the nuclear power plant Philippsburg 2 (PWR constructed by SIEMENS/KWU) on February 15, 1985 was recalculated with RELAP5/MOD2 Cycle 36.04.

The comparison of the results of the RELAP5/MOD2 calculation with the commissioning test results shows very good agreement between measurement and calculation, but in general a critical review of the measured data is needed before comparing them with calculation.

There is one significant deviation between calculated and measured RPV inlet temperature coming from heat transfer correlations in RELAP5/MOD2 which are not optimized for this problem. Furthermore for a short period a deviation exists between the calculated and measured value of the SG water level about 15 s after beginning of the transient, which is assumed to be due to the separator model in the SG modelling.

1 Introduction

As agreed between the government of the Federal Republic of Germany and SIEMENS-KWU among other post test calculations of PWR commissioning tests are to be carried out by SIEMENS-KWU with the computer code RELAP5/MOD2. These calculations are a contribution to the

"Agreement on Research Participation and Technical Exchange between the United States Nuclear Regulatory Commission (USNRC) and the Federal Minister of Research and Technology of the Federal Republic of Germany in USNRC Thermal Hydraulic Research Programs and BMFT Thermal Hydraulic Research Programs covering a five-year period" (called as "Code Assessment Agreement").

The commissioning test "Reactor Trip at Full Load" (D-100-301) of the NPP Philippsburg 2 was selected for the first post test calculation with the computer code RELAP5/MOD2.

In this report the results of the calculation are discussed and compared with the commissioning test results.

A general problem of code assessment is to decide whether the measured values are suitable for comparison with calculated data because the measurement can be faulty or falsified by instrumentational effects (for instance damped with known or unknown time constants). Besides, each measurement has a permissible uncertainty range usually being in the order of magnitude of 1 % - 2 % of the measuring range depending from design specifications. This has to be borne in mind when performing such comparisons.

Therefore a critical review of the measured values is needed before comparing them with the calculation.

2 Description of Philippsburg 2 Nuclear Power Plant

The 1300 MWe Philippsburg 2 nuclear power plant being constructed by SIEMENS-KWU has been equipped with a pressurized water reactor as a nuclear steam generator with a thermal power output of 3765 MW (design value). A more detailled descirption can be seen from the Data Sheets (Appendix A).

3 Description of the Commissioning Test

3.1 Transient Behaviour of the Plant

In the scope of the commissioning of the plant the reactor trip signal has to be initiated manually to verify the transient behaviour (as predicted by computer codes) and to show that the design of the relevant systems is proper to reach conditions which are within the designed values.

As a result of the reactor scram being initiated (manually), the control rods drop in so that the negative reactivity insertion reduces the reactor power very rapidly to decay heat level. The RESA (reactor scram) signal switches measurement from the ex-core to the in-core instrumentation. The steam generator power (power transferred from the primary to the secondary side) follows the decrease in reactor power with some delay because of the energy stored in the primary circuit.

The decrease in the reactor power also produces a decrease in the temperature rise which is a direct measure of thermal power.

The RESA signal also initiates TUSA (turbine trip); as a result the main steam letdown is very rapidly reduced to zero so that the main steam pressure rises until, when the maximum pressure set point is reached, the main steam bypass station (FDU) opens and limits the pressure rise.

The rise in main steam pressure causes a rise in the primary-side steam generator outlet temperature and, consequently, in the reactor pressure vessel inlet temperature.

The reactor outlet temperature is dependent on the reactor pressure vessel inlet temperature, which initially rises, and on the decreasing temperature rise in the core. There is an overall decrease in both reactor outlet temperature and average coolant temperature.

The decrease in the (average) coolant temperature produces a contraction of the coolant which results in a corresponding outsurge from the pressurizer and, consequently, in a drop in the pressurizer water level which leads to a corresponding decrease in coolant pressure. In the long term, the coolant pressure is again increased to its set point by the pressurizer heating elements.

The RESA signal activated by the TUSA signal abruptly push up the maximum pressure set point to 77 bar (design value) and is then raised at 20 bar/min to 80 bar.

As already mentioned above, as a result of the reduction in the main steam letdown, the pressure in the steam generators rises until the pressure is held at the set point value by the main steam bypass control.

The decrease in power results in the collapse of the steam bubbles in the steam generator riser chamber and consequently in a drop in the steam generator water level.

In order to avoid increasing the feed to the steam generator, the full-load feedwater supply valves are closed. Further feed takes place via the low-load control valves. The feed via the full-load control valves remains blocked for 180 s.

3.2 Available Test Data

There is a difference between tests performed on test facilities to verify computer models and commissioning tests which are performed to check the overall behaviour of the plant. On a plant the instrumentation, both the measuring technique and the measuring positions, are not optimized for a test, and furthermore instead only the standard operating instrumentation is available. Therefore only a few selected values are available for the evaluation of the tests.

From the commissioning tests for checking the dynamic behaviour of the plant Philippsburg 2 at postulated transients the following information is available for evaluation of the tests:

- "HP 1000 Band"
  Magnetic tape, written by the data acquisition system HP 1000. There are 240 measured values selected for storage on magnetic tape with a time interval of 0.04 s. From this tape plots can be made.

- "Zyklisches Meßwertprotokoll"
  About 140 measured values extracted from the process computer systems are printed in an interval of 5 s. A part, but not all of these values are also stored on the "HP 1000 Band".

- "Anlagenzustandsprotokoll"
  The actual values of all analog measuring points which are available from the process computer system are printed before and after the test (about 940 measured values).

Only a few of these data are usable for the postcalculation of the test D-100-301 "Reactor Trip at Full Load".

3.3 Review and Discussion of the Measured Data

As mentioned above, a critical review and discussion of the available data is necessary before comparing them with computed data or/and before taking them as steady-state operation values for the initial conditions of the computation.

As mentioned in the introduction the measurement uncertainty is in the order of magnitude of 1 % - 2 % of the measuring range. The maximal allowed uncertainty of each measurement equipment is specified by the responsible design groups. Some examples are given by the responsible design groups (Table 3.1).

Measuring Point Identification	Measuring Range	Measuring Accuracy	
		% of Range	Absolut
JEC30 CT832	250 - 325 °C	± 0.715 %	± 2.32 K
JEC10 CT811	0 - 40 K	± 1.376 %	± 0.55 K
JEC20 CP821	100 - 180 bar	± 1.376 %	± 2.48 bar
JEA10 CP851	0 - 150 bar	± 1.01 %	± 1.515 bar
JEA20 CL871	8.02 - 15.68 m	± 0.92 %	± 0.144 m

Table 3.1: Examples for Measuring Accuracy

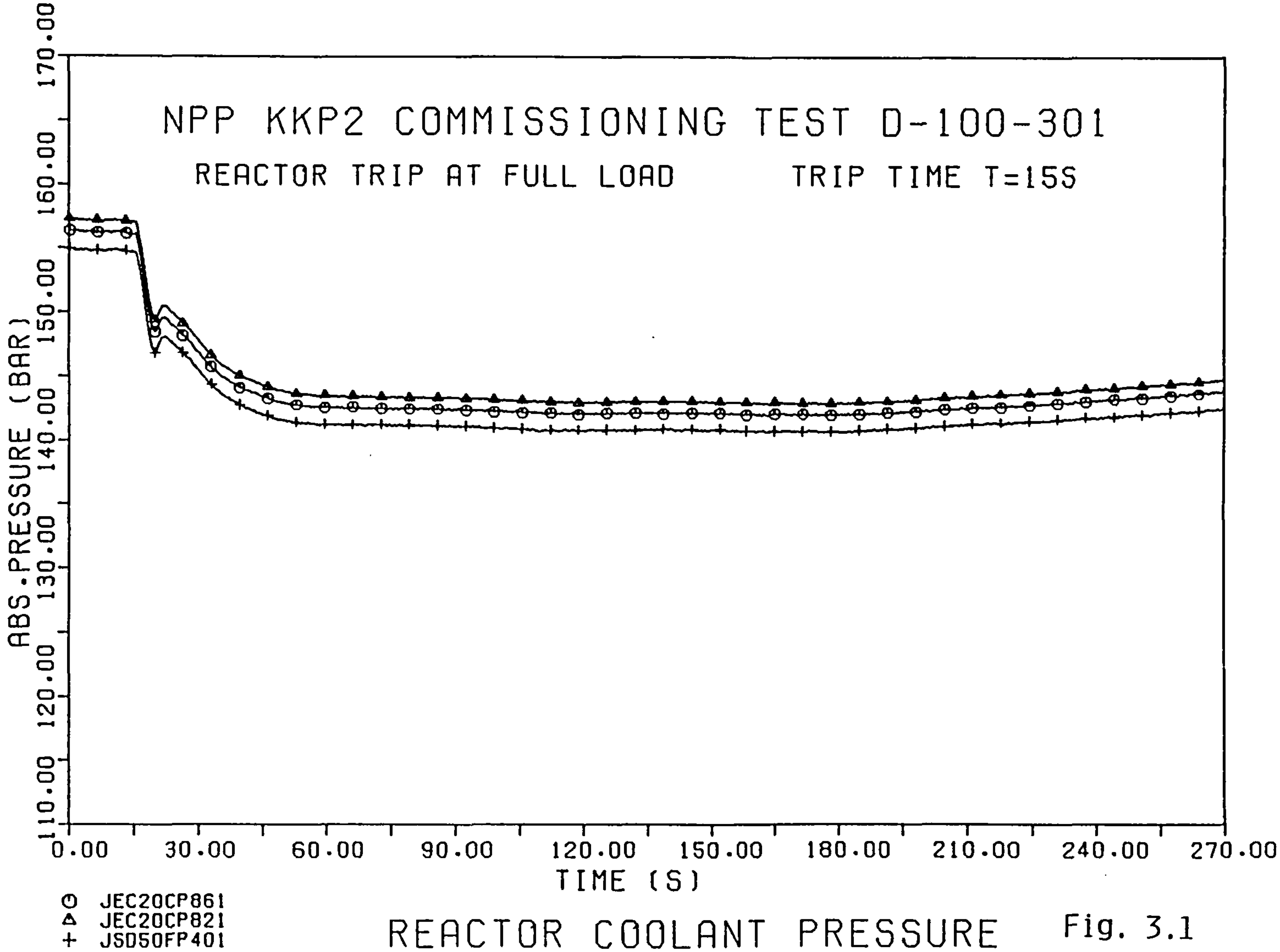
3.3.1 Reactor Coolant Pressure

For the reactor coolant pressure three measured values are available (Fig. 3.1):

- JEC20 CP861 measuring range 0 - 180 bar (wide range)
- JEC20 CP821 measuring range 100 - 180 bar (narrow range)
- JSD50 FP401 measuring range 0 - 180 bar (actual value)

For measurement of the first two values the same nozzle is used; it is located in Loop 20 upstream of the surge line. The small range value is about 1.0 bar higher than the wide range value (about 0.5 % of the measuring range of the wide range) in steady-state operation as well as during the transient.

The third value is a selected value (single channel) for the analog signal processing and control system. This value is always about 1.5 bar lower than the wide range value due to a calibration effect in the analog signal processing. This value is not suitable for comparison with calculation.



REACTOR COOLANT PRESSURE Fig. 3.1

3.3.2 Pressurizer Water Level

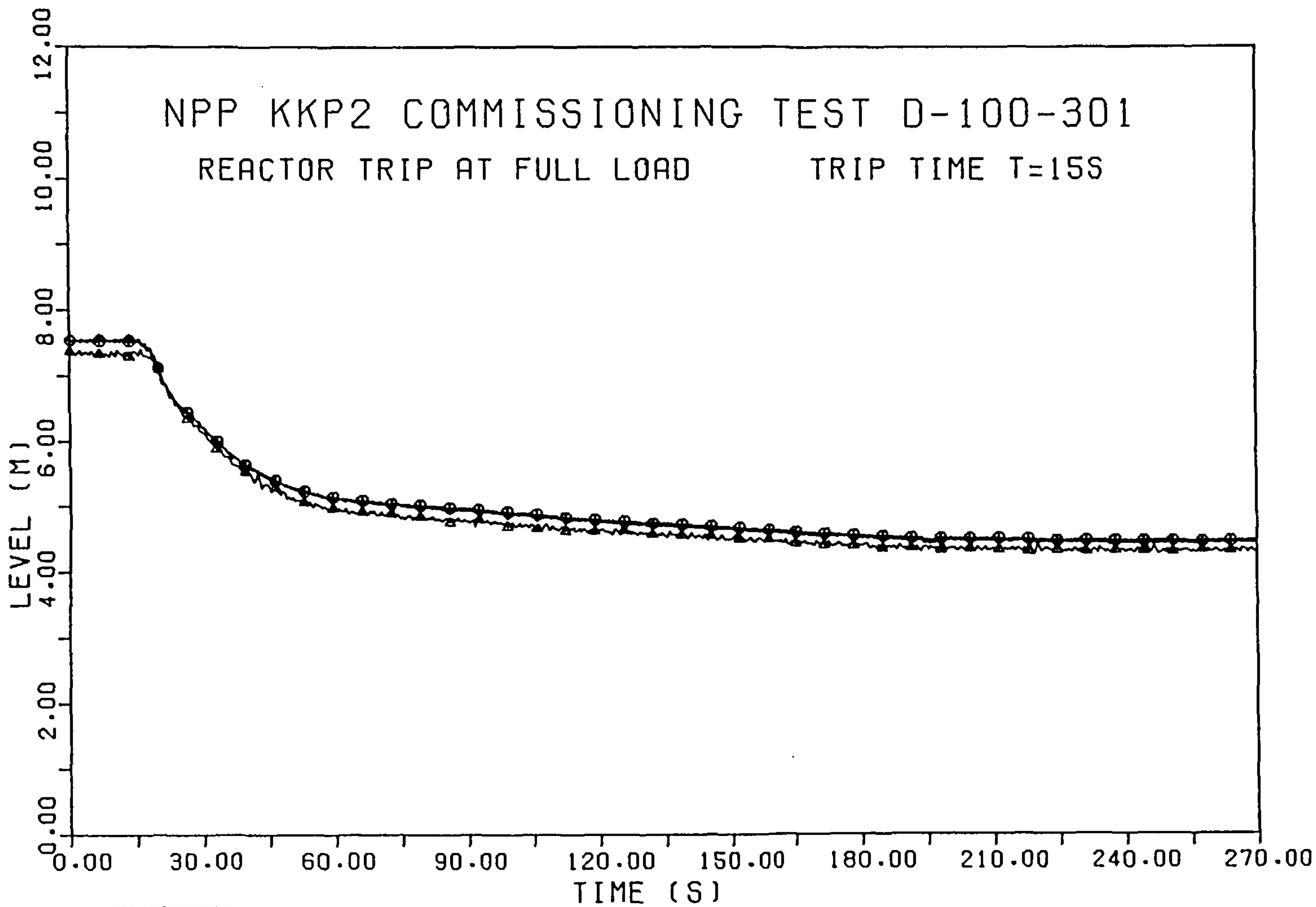
Three measured values for the pressurizer (PRZ) water level are available (Fig. 3.2).

- JEF10 FL001 measuring range 1.75 - 11.45 m
- JEF10 CL871 measuring range 1.75 - 11.45 m
- JTK00 FL501 "actual value" measuring range 1.75 - 11.45 m

The measured value JTK00 FL501 is the second highest signal value (2. Max) used in the limitation system.

The value JEF10 FL001 is always lower than the other values. The deviation of about 0.2 m (about 2 % of the measuring range) is considered as allowable within the tolerance range.

The measuring principle for the pressurizer water level is the measurement of the differential pressure between the instrumentation nozzles at the levels 1.75 m and 11.45 m, respectively.



JTK00FL501
JEF10FL001
JEF10CL871

PRESSURIZER WATERLEVEL

Fig. 3.2

3.3.3 Reactor Coolant Temperature (RPV Inlet)

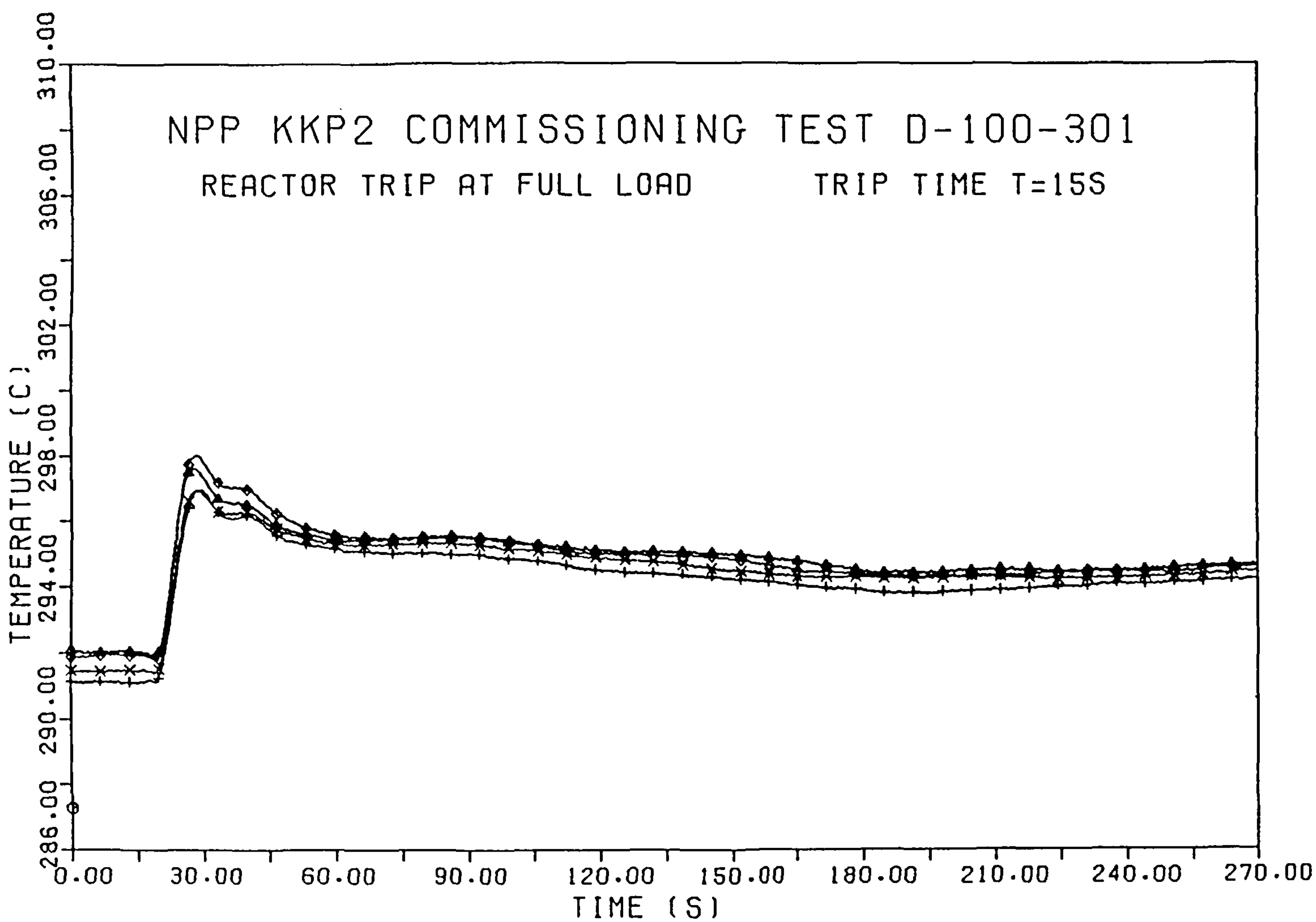
One measured value of the reactor coolant temperature in the cold leg between reactor pressure vessel and reactor coolant pump is available on the "HP 1000 Band" for each loop (Fig. 3.3).

- JEC10 CT812	measuring range
- JEC20 CT822	250 - 325 °C
- JEC30 CT832	(narrow range)
- JEC40 CT842	

At steady-state operation the values of Loop 10 and Loop 40 are nearly the same (≈292 °C): the equivalent values of Loop 20 and Loop 30 are about 0.9 K and 0.5 K, respectively, lower. During the transient these values are always lower than the values of Loop 10 and Loop 40.

From the "Anlagenzustandsprotokoll" is to be seen, that the other measured value (narrow range JEC20 CT733) of Loop 20 is about 1.6 K higher, whereas in the other three loops the two narrow range measured values have only an uncertainty range of maximal 0.2 K.

Therefore the measured value JEC20 CT822 should not be applied to comparison with calculation.



△ JEC10CT812
+ JEC20CT822
× JEC30CT832
◇ JEC40CT842

RPV INLET TEMPERATURE Fig. 3.3

3.3.4 Temperature Rise

The temperature rise is an important measurement for the determination of the thermal reactor power. Therefore all 16 measuring positions are printed on the "Zyklisches Meßwertprotokoll". One of the four measured values of each loop is stored on the "HP 1000 Band" (Fig. 3.4).

- JEC10 CT811	
- JEC20 CT811	measuring range
- JEC30 CT811	0 - 40 K
- JEC40 CT811	

As to be seen in Fig. 3.4 the values of Loop 10 and Loop 40 are nearly the same at steady-state operation (about 33 K). The values of Loop 20 (30.7 K) and Loop 30 (31.5 K) are about 2.3 K and 1.5 K, resp. lower than these of Loop 10 and Loop 40.

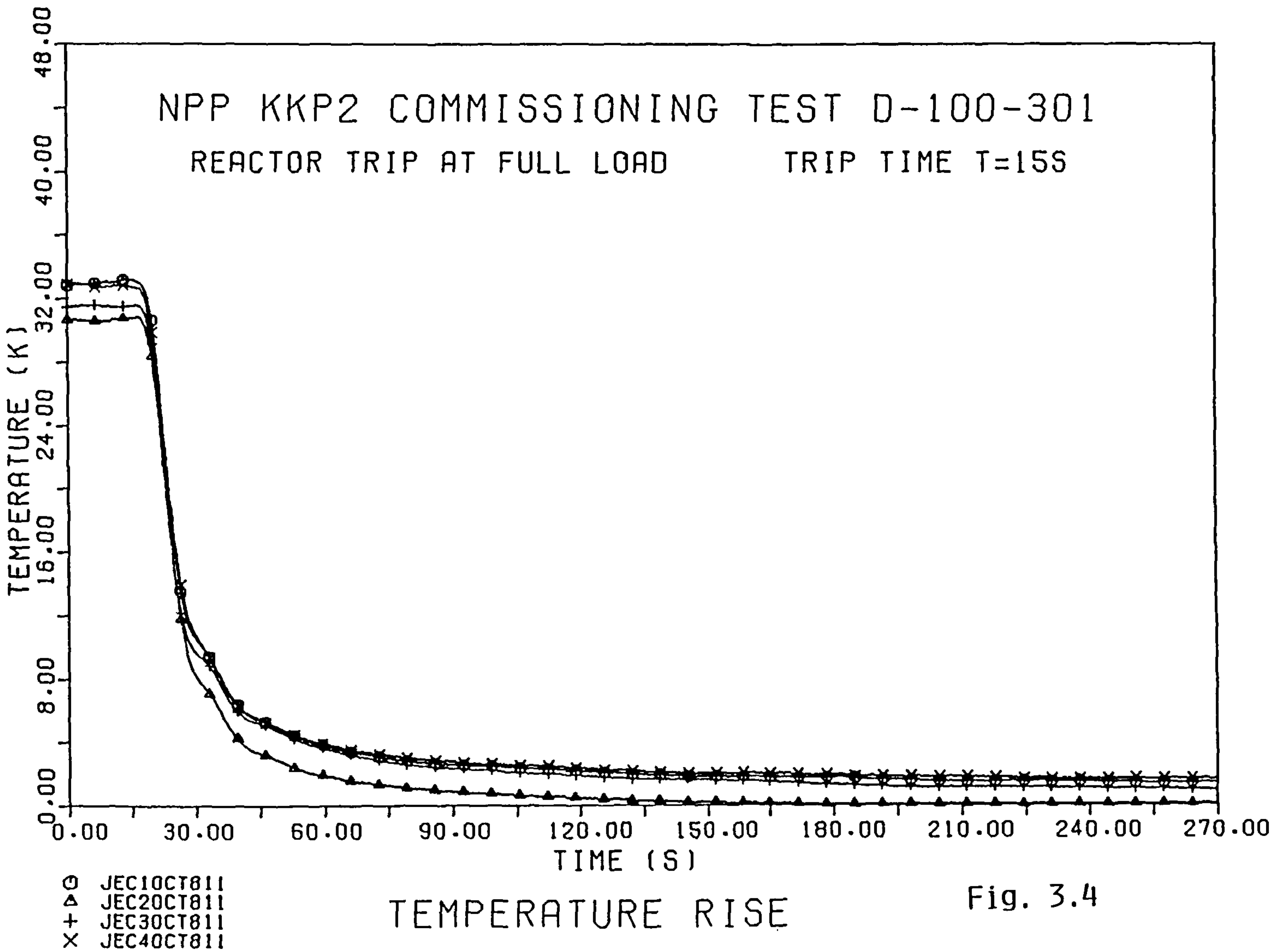
During the transient the temperature rise in Loop 20 (JEC20 CT811) deviates from the values of the other loops and equals to zero about 150 s after the beginning of the transient, in spite of a decay power of about 3 %. This indicates that this measured value is systematically too low and is therefore not suitable for comparison with calculation.

The evaluation of the "Zyklisches Meßwertprotokoll" yields an uncertainty range of about 1.2 K, 2.5 K, 1.0 K and 1.2 K in the Loops 10, 20, 30 and 40, respectively.

The average of the four measured values of each loop is

- 32.4 K in Loop 10
- 31.8 K in Loop 20
- 32.0 K in Loop 30
- 32.1 K in Loop 40

From this results an average of about 32 K for the temperature rise at full power.



TEMPERATURE RISE

Fig. 3.4

3.3.5 Reactor Power

The "Corrected Thermal Reactor Power" signal is formed by the neutron-flux (measured with the excore instrumentation) and the thermal reactor power calculated by means of the temperature rise. Thus the signal of the neutron-flux is corrected toward the thermal power with 10 %/min if the neutron-flux is lower than the thermal power and the signal of the neutron flux is corrected toward the thermal power with 1 %/min if the neutron flux is greater than the thermal power.

The following measured values of the "Corrected Thermal Power" are available on the "HP 1000 Band".

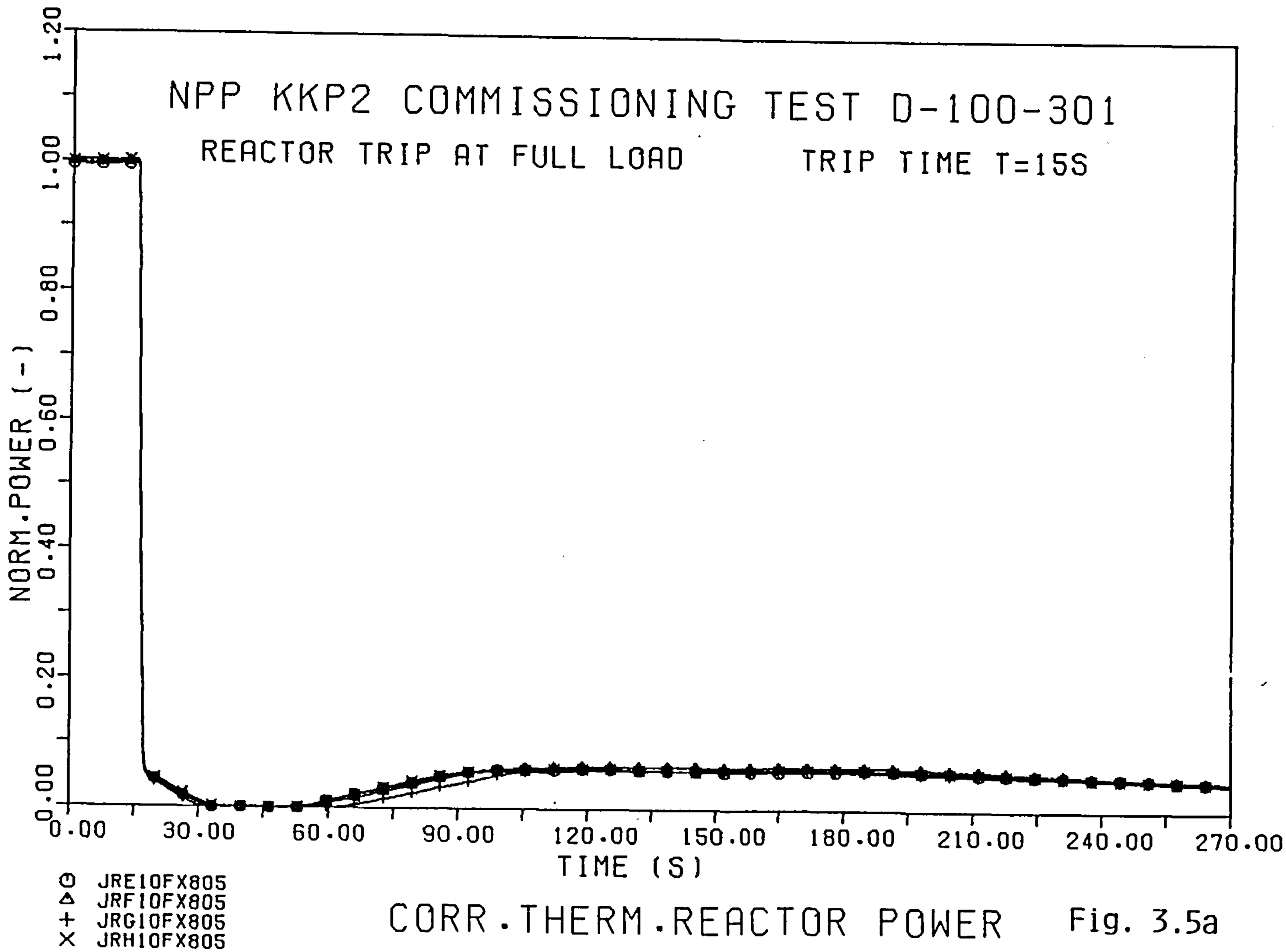
- JRE10 FX805	
- JRF10 FX805	measuring range
- JRG10 FX805	0 - 125 %
- JRH10 FX805	

These values are applied to the reactor protection system.

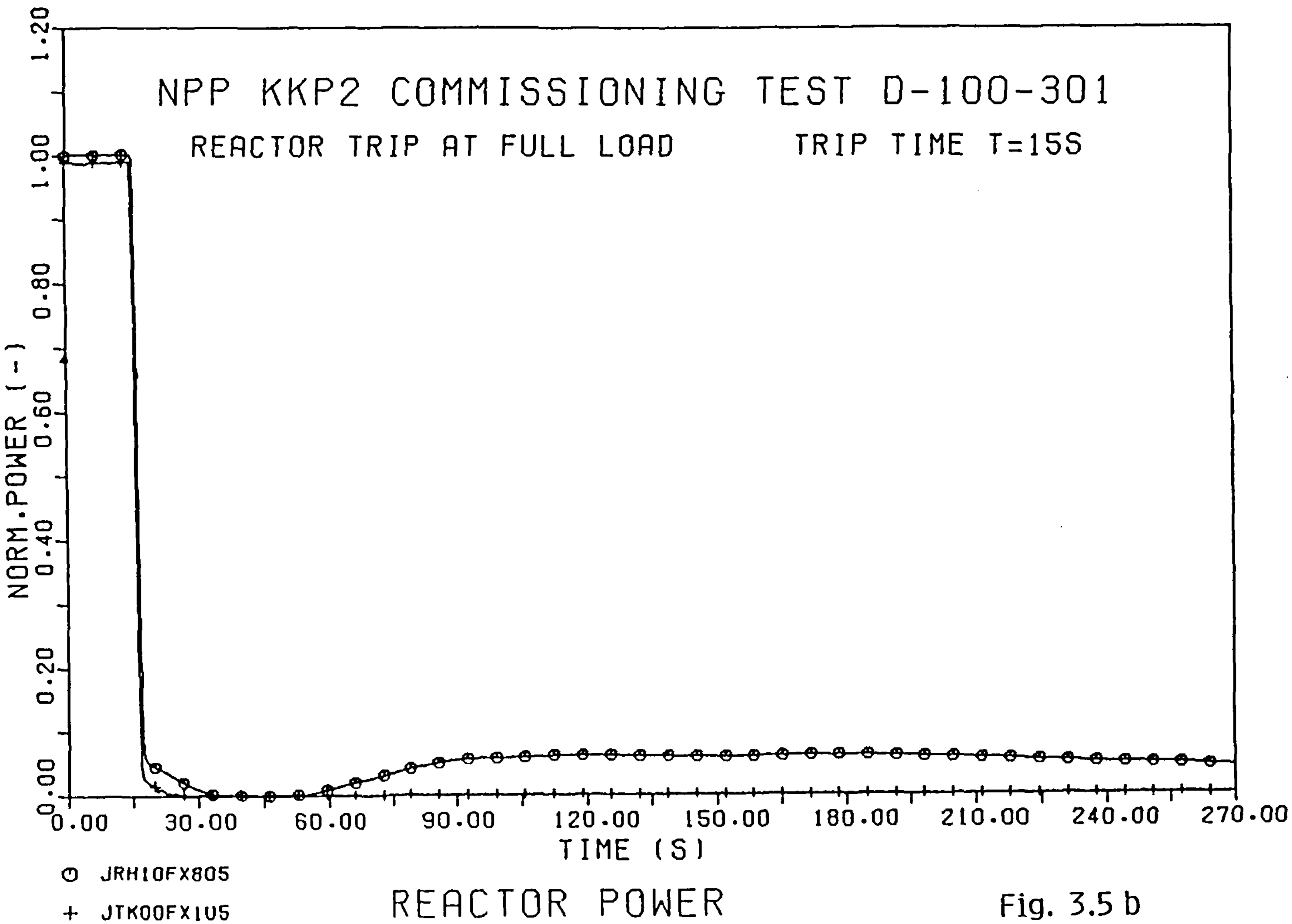
The signal (Fig. 3.6b)

- JTK00 FX105 measuring range 0 - 125 %

is the measured signal of the n-flux, but it is switched over from excore to incore instrumentation after reactor trip. This signal is applied to the reactor limitation system. It appears to be the best suited for comparison with calculated values.



CORR.THERM.REACTOR POWER Fig. 3.5a



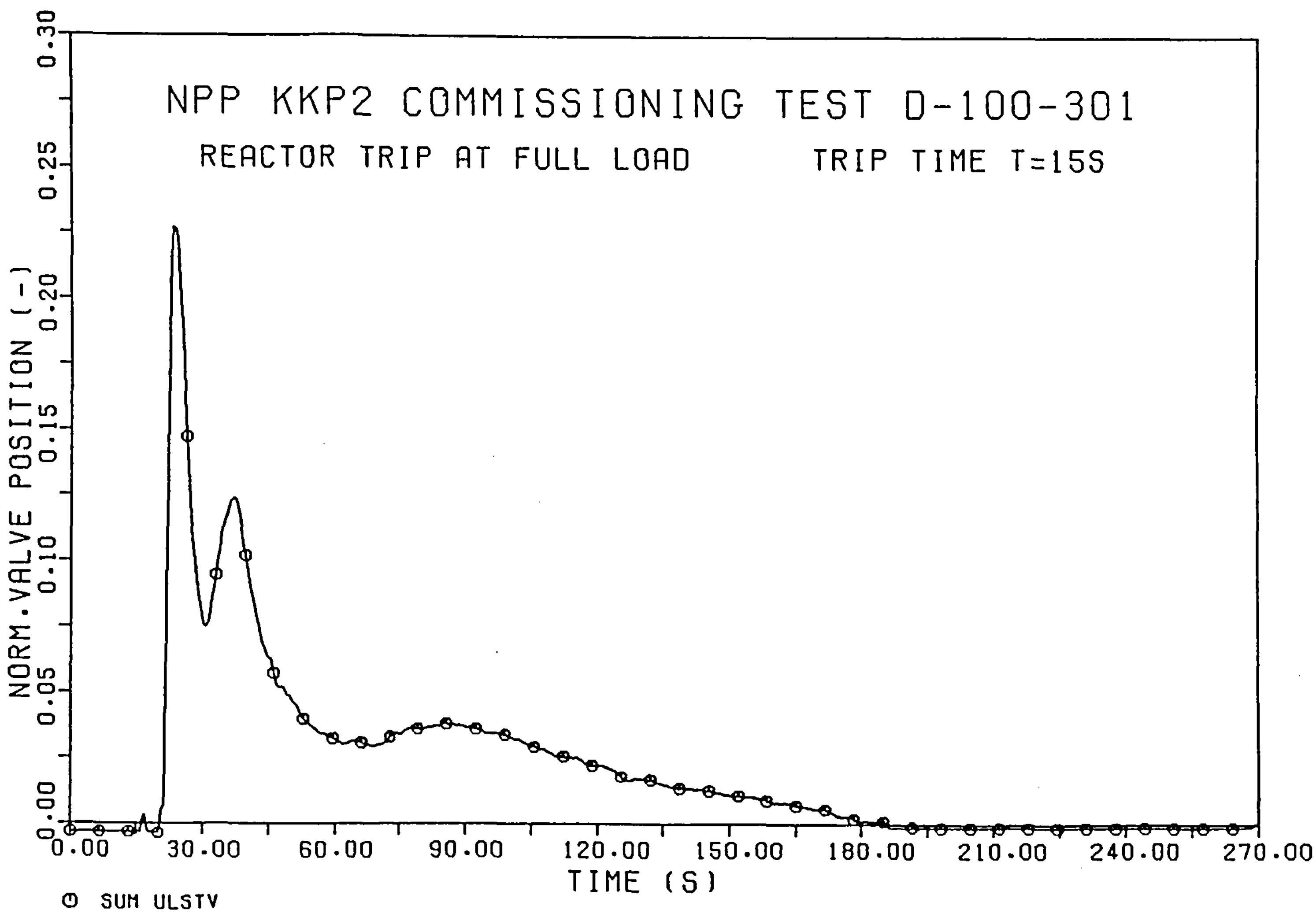
REACTOR POWER Fig. 3.5 b

3.3.6 Main Steam Bypass Valve Position

From the position of the six MS-Bypass valves only one measured value is available, namely the normalized sum of all six positions (Fig. 3.6):

- SUM ULSTV   measuring range 0 - 100 %

This value can be used for comparison with the calculation.



SUM OF BYPASS VALVES Fig. 3.6

3.3.7 Main Steam Pressure

From the measurement of the main steam (MS) pressure the measured values at several positions are available:

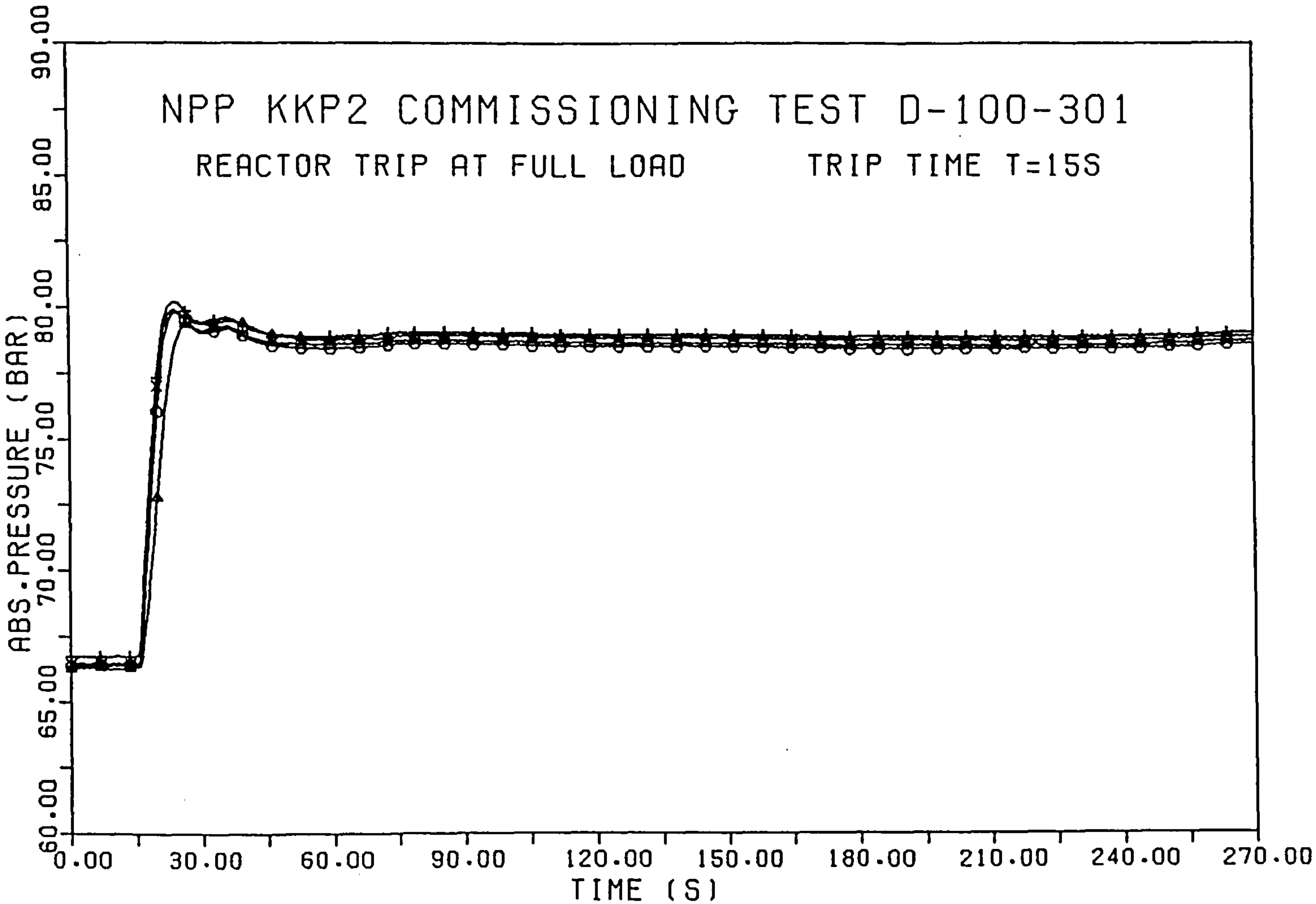
a) One measured value in each steam generator stored on the "HP 1000 Band" (Fig. 3.7a)

JEA10 CP851	
JEA20 CP861	measuring range
JEA30 CP871	0 - 150 bar
JEA40 CP851	

b) One measured value in each MS piping between steam generator and MS isolation valve printed on the "Zyklisches Meßwertprotokoll" (Fig. 3.7b)

LBA10 CP001	
LBA20 CP001	measuring range
LBA30 CP001	0 - 100 bar
LBA40 CP001	

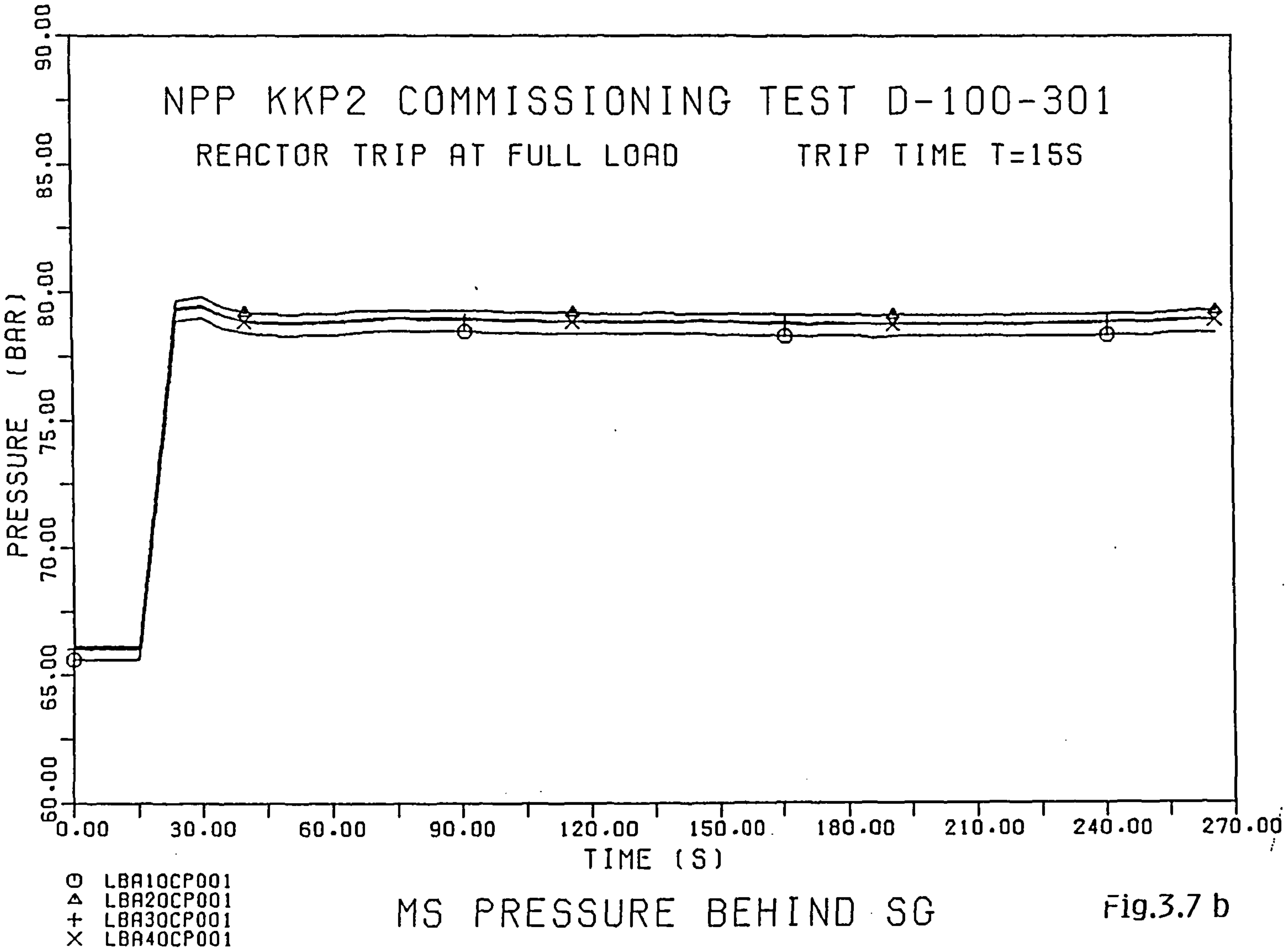
From Fig. 3.7a and Fig. 3.7b it can be concluded, that the increase of the pressure in SG20 (JEA20 CP861) is somewhat slower as in the other loops and in the MS piping (LBA20 CP001), indicating that this signal is damped, the time constant is estimated to be about 3 s.



ʘ JEA10CP851
Δ JEA20CP861
+ JEA30CP871
× JEA40CP851

MS PRESSURE IN SG

Fig. 3.7a



MS PRESSURE BEHIND SG

Fig.3.7 b

3.3.8 Feedwater Flow Rate

The transient behaviour of the plant is strongly affected by the feedwater flow rate. In particular, the SG water level and the pressure in the SG is influenced by the feedwater flow rate.

In the RELAP5/MOD2 calculation the feeding of the steam generator is simulated by time dependent input data (time dependent junction) in order to avoid the modelling of the feedwater control. Therefore it is necessary to get the "most exact" time dependent feedwater flow rate.

There exists a lot of direct and indirect informations about the feedwater flow rate:

a) the valve positions of the feedwater full load control valves (on the "HP 1000 Band") (Fig. 3.8a)

- LAB60 CG002	
- LAB70 CG002	measuring range
- LAB80 CG002	0 - 100 %
- LAB90 CG002	

b) The valve positions of the feedwater low load control valves stored on the "HP 1000 Band" (Fig. 3.8b).

- LAB64 CG003	
- LAB74 CG003	measuring range
- LAB84 CG003	0 - 100 %
- LAB94 CG003	

From this valve positions and the valve characteristics the feedwater flow rate can be estimated, but the real characteristic is not exactly known.

c) One measured value of the feedwater flow rate, measured in each loop upstream of the Steam Generator (Fig. 3.8c).

- LAB60 CF711	
- LAB70 CF721	measuring range
- LAB80 CF731	0 - 600 kg/s
- LAB90 CF741	

These values are strongly damped; a time constant of about 20 s is assumed. Therefore these values are not directly suitable for input data for the calculation.

d) The 2. Max from the measurements of each loop (Fig. 3.8d), stored on the "HP 1000 Band"

- LAB60 FF901	
- LAB70 FF901	measuring range
- LAB80 FF901	0 - 600 kg/s
- LAB90 FF901	

e) One measurement for each loop printed on the "Zyklisches Meßwertprotokoll" (Fig. 3.8e)

- LAB60 CF001A	
- LAB70 CF001A	measuring range
- LAB80 CF001A	0 - 600 kg/s
- LAB90 CF001A	

By comparing Fig. 3.8e and Fig. 3.8d it can be assumed that the measured values LAB60 - 90 FF901 (Fig. 3.8d) are slightly damped.

The most reliable input for calculation seem to be the measured values LAB60 - 90 CF001, which are in good agreement with the estimation from the valve positions.

From all these information the real feedwater flow rate can be deduced.

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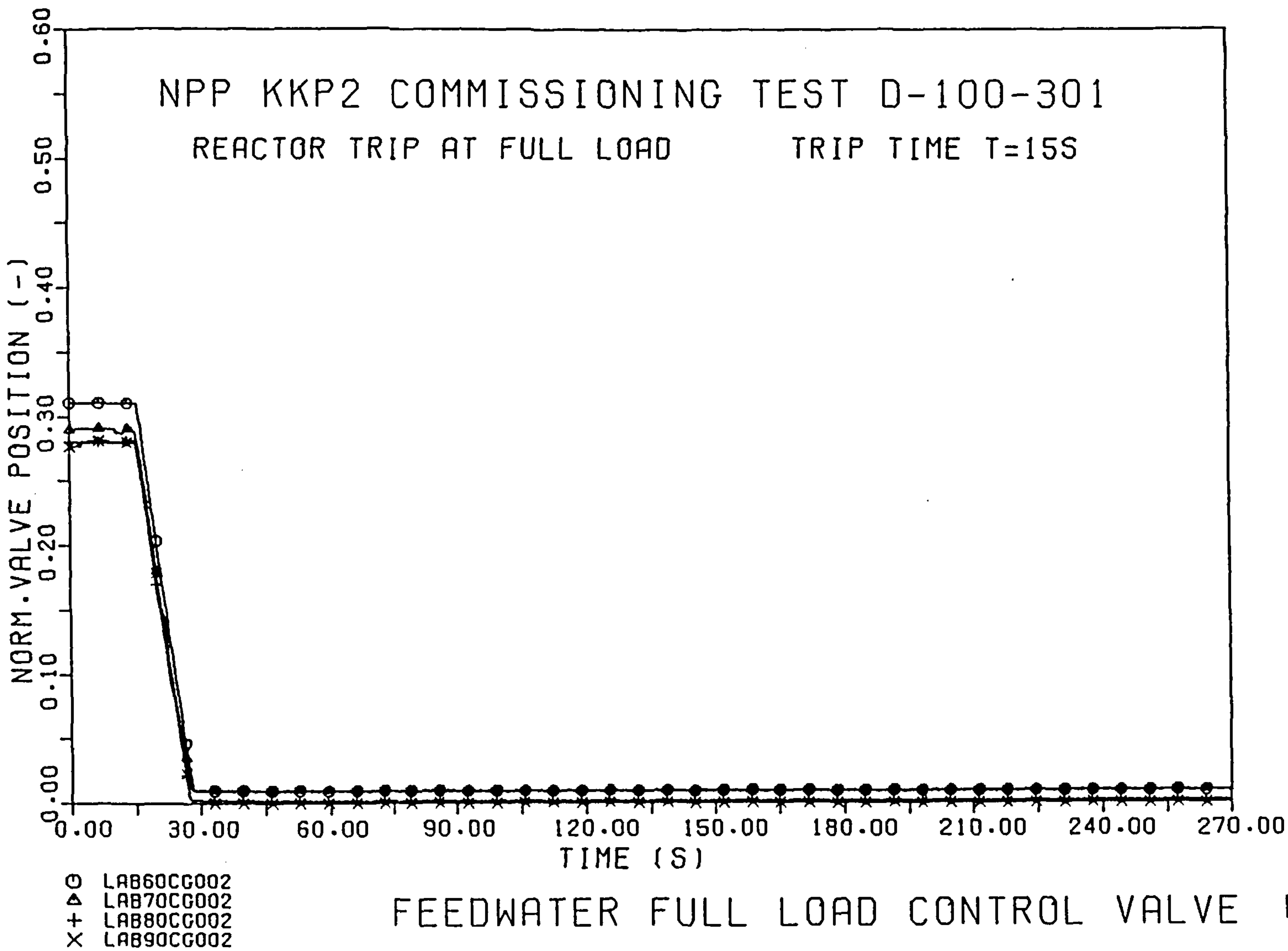
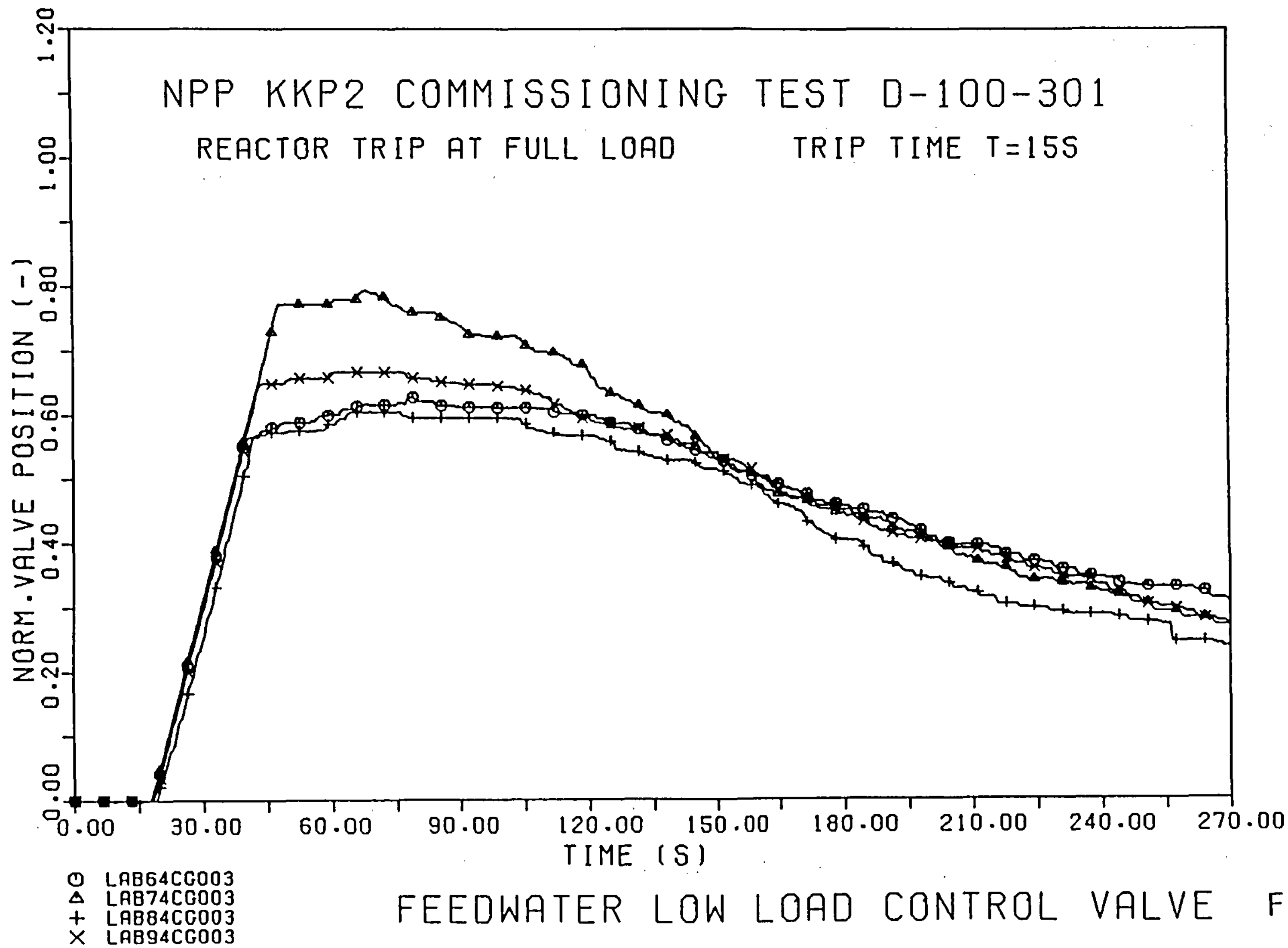
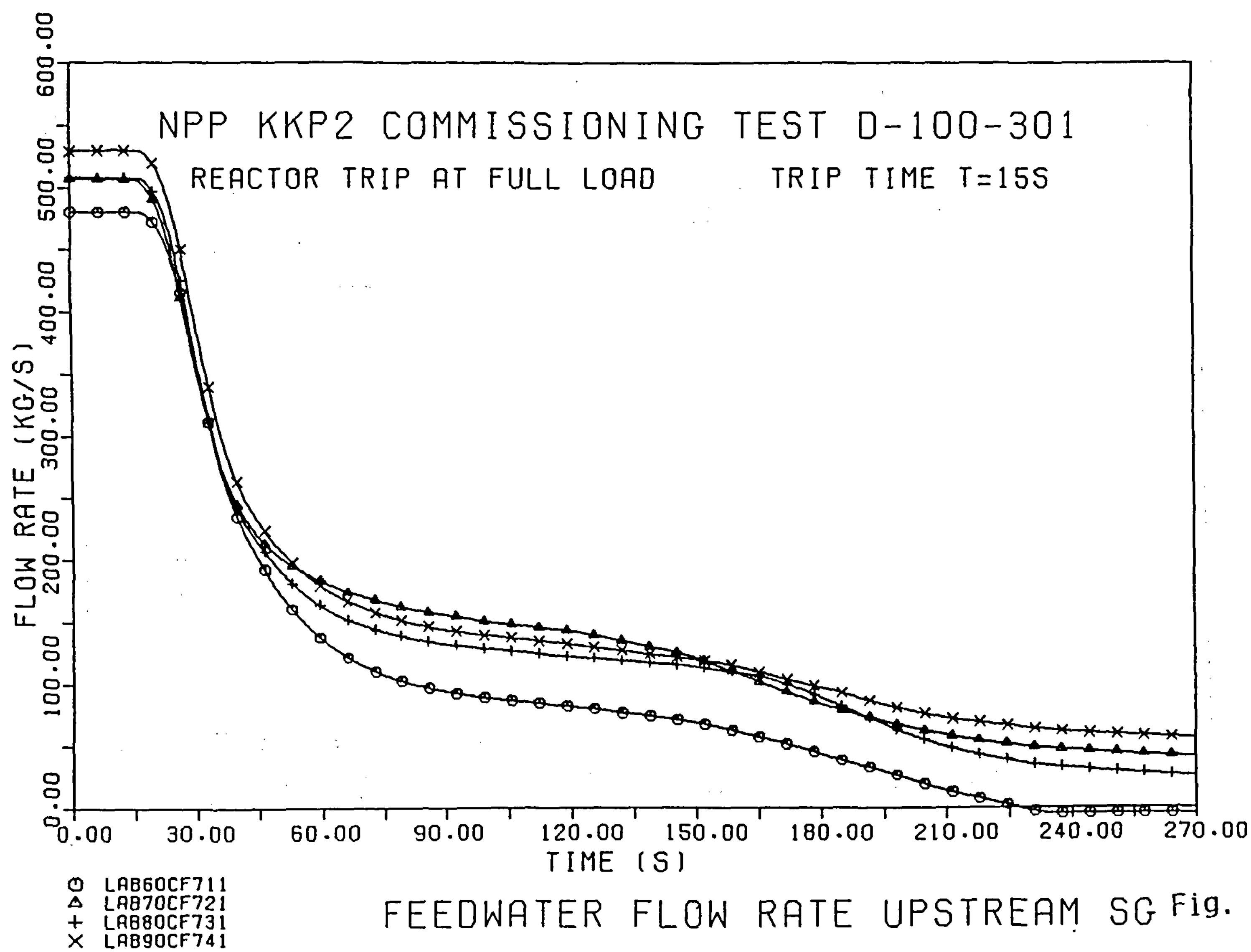


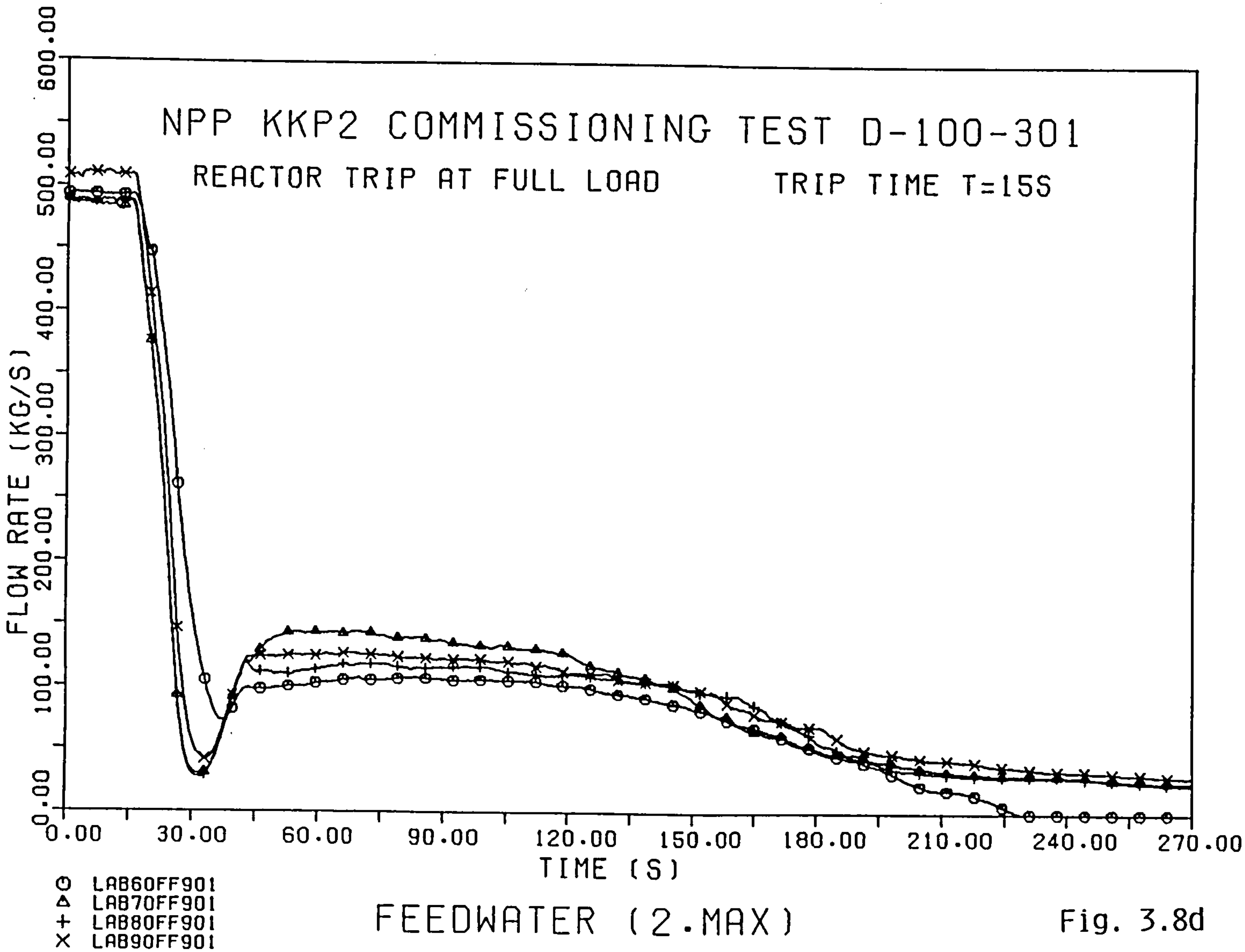
FEEDWATER FULL LOAD CONTROL VALVE Fig. 3.8a



FEEDWATER LOW LOAD CONTROL VALVE Fig. 3.8b

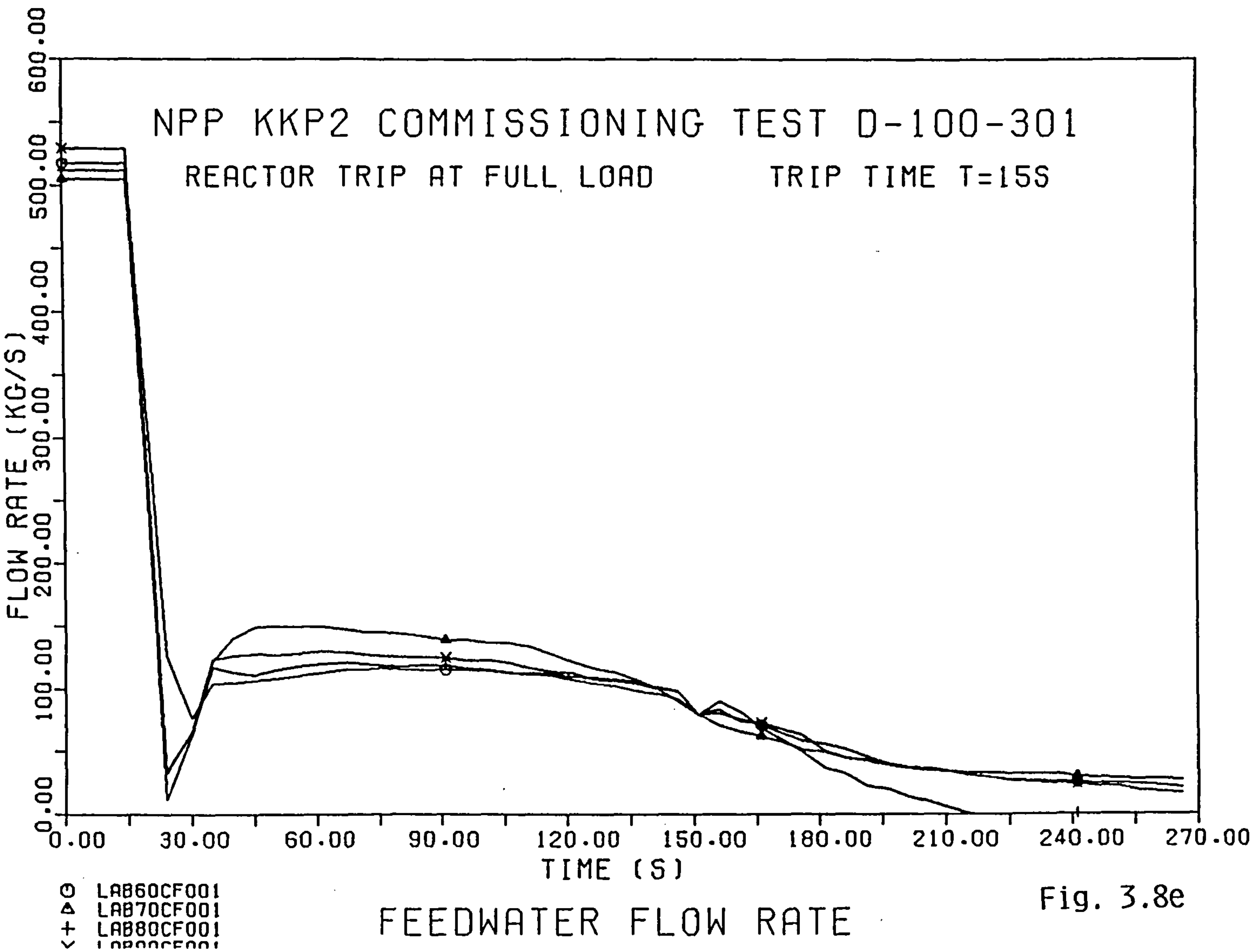


FEEDWATER FLOW RATE UPSTREAM SG Fig. 3.8c



FEEDWATER (2.MAX)

Fig. 3.8d



FEEDWATER FLOW RATE

Fig. 3.8e

3.3.9 Steam Generator Water Level

The measurement of the SG water level is based on the measuring of the differential pressure between the measuring nozzles. There are two measuring ranges:

narrow range:	upper nozzle	15.68 m
	lower nozzle	8.02 m
wide range:	upper nozzle	15.68 m
	lower nozzle	4.31 m

From the differential pressure the water level is calculated; however, no correction of the pressure and temperature dependent density is made. The calibration point is not at full load conditions but at a pressure of about 43 bar. Thus the real water level at full load operation is higher than the measured value.

Besides, this measuring principle assumes static pressure between the nozzles. The pressure drop in the downcomer due to the circulation flow rated about four times the feedwater rate (at full load) effects a measured level lower than the real level. Necessarily, the wide range value is lower than the narrow range value, because the influence of the pressure drop in the downcomer is essentially greater than in the narrow range.

One measurement from each steam generator is available for the wide range and the narrow range, respectively:

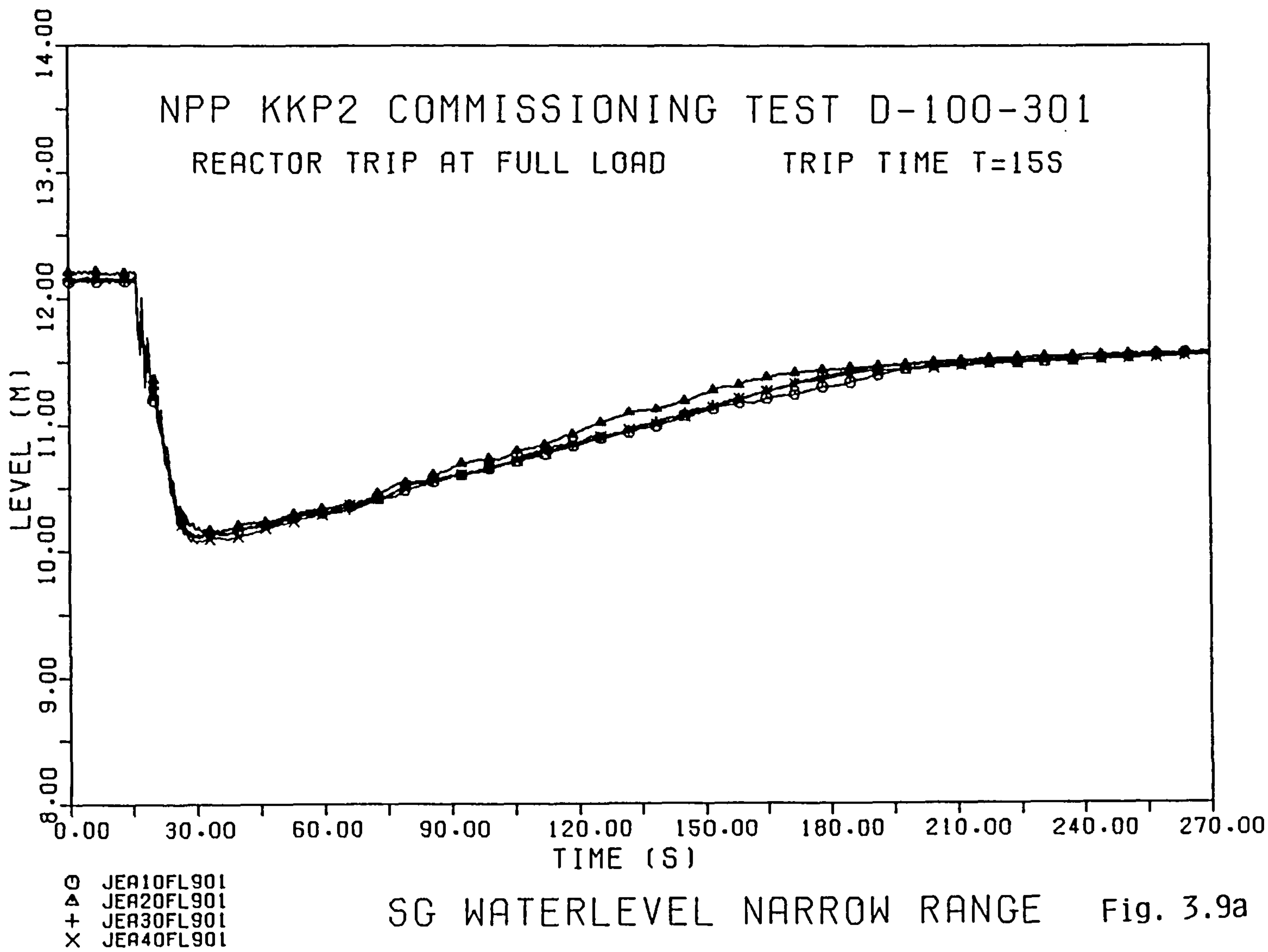
- JEA10 FL901	narrow range
- JEA20 FL901	measuring range
- JEA30 FL901	8.02 - 15.68 m
- JEA40 FL901	

respectively

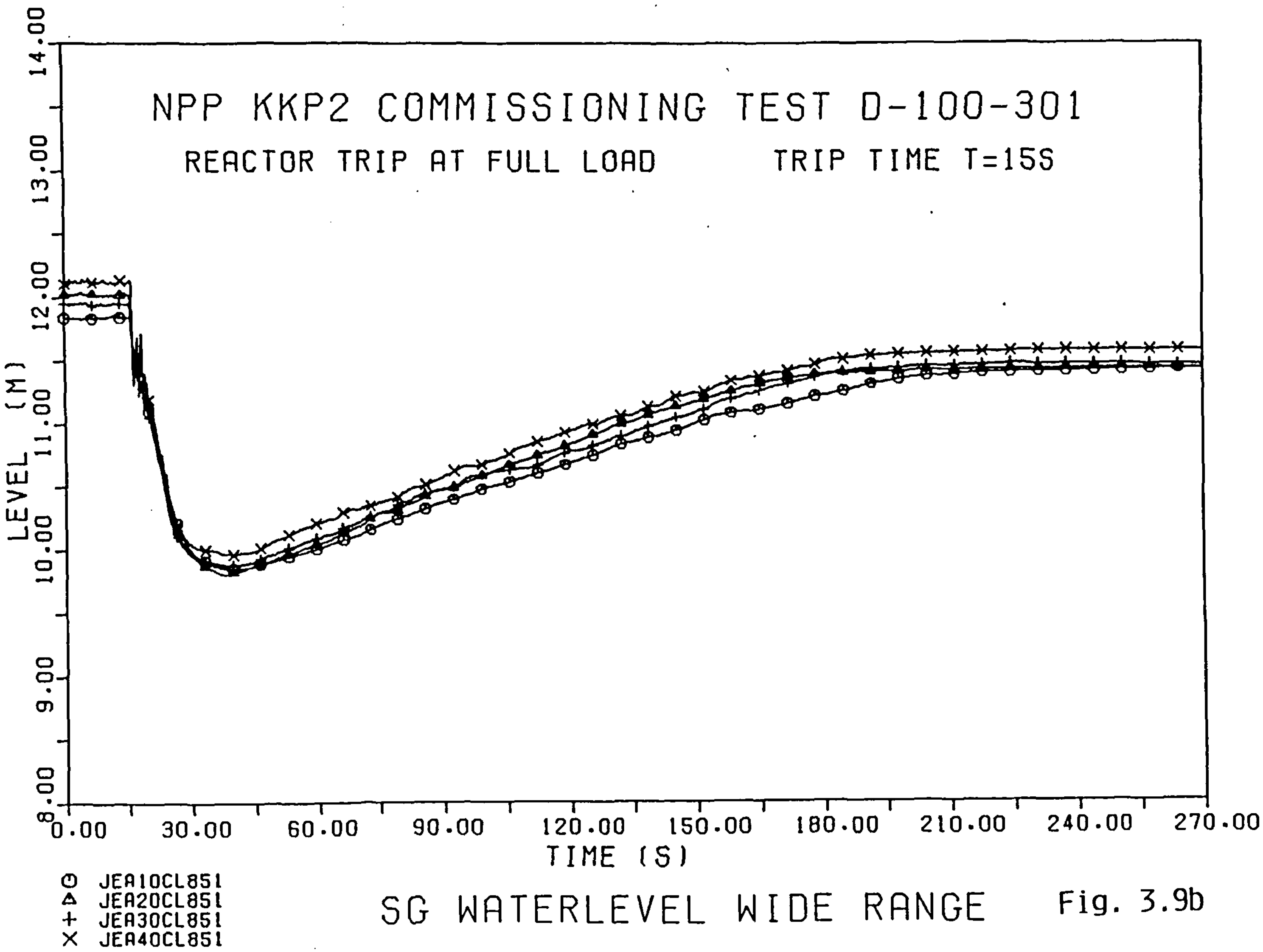
- JEA10 CL851	wide range
- JEA20 CL851	measuring range
- JEA30 CL851	4.31 - 15.68 m
- JEA40 CL851	

At steady-state operation the measured values of the narrow range are nearly the same, but the values of the wide range measurement have an uncertainty range of about 0.3 m.

During the transient the measured values are different due to the different feed-water flow (Fig. 3.9a and 3.9b).



SG WATERLEVEL NARROW RANGE Fig. 3.9a



SG WATERLEVEL WIDE RANGE Fig. 3.9b

3.3.10 Valve Position for Steam Extraction for Feedwater Heating

The steam for feedwater heating is extracted from the main steam header. After reactor trip the amount of steam extraction influences the rate of pressure increase in the header. Therefore the flow rate of extracted steam is needed as input for the calculation.

Only the normalized valve positions of the four valves

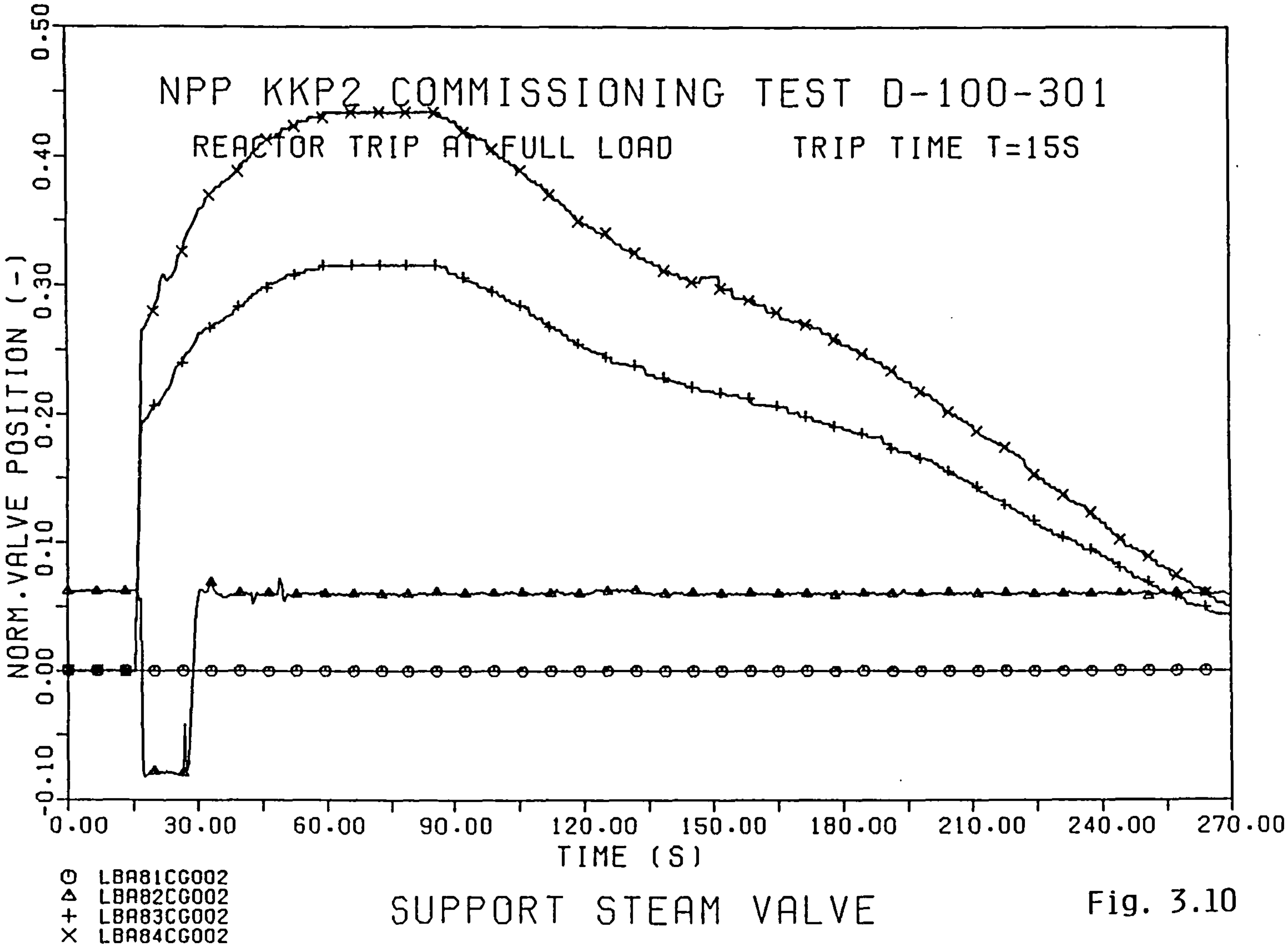
LBA81 and LBA82 steam supply for the feedwater tank

and

LBA83 and LBA84 steam supply of the feedwater heater A5

are available.

From these signals the flowrate of steam extraction can be deduced using the valve characteristic (design values). This flowrate is used in the RELAP5/MOD2 input.



SUPPORT STEAM VALVE

Fig. 3.10

4 Code and Model Description

4.1 Code Version

The calculations were performed with RELAP5/MOD2 Cycle 36.04 without modifications.

The Code was implemented on a CDC Computer (Cyber 176) operating under NOS/BE.

4.2 Simulation of the Plant

The nodalisation of the plant is depicted in Fig. 1 and Fig. 2.

The four loops of the plant are simulated by a "Single Loop" and a "Threefold Loop". The "Single Loop" represents one loop of the plant, whereas the other three loops are condensed to the "Threefold Loop" with threefold cross-section and therefore threefold volume and threefold flow rates. The pressurizer is connected to the "Single Loop".

The core is simulated by eight BRANCH components which represents the normal channel only, i. e. neither hot channel nor hot rod are simulated.

The steam generator riser is simulated by a PIPE-component with 5 volumes (zones) in the region of the tube bundle and one volume above the bundle below the water steam separator. This pipe is connected with a SEPARATOR component. The steam outlet of the separator is connected with a single volume component SNGLVOL (steam expansion tank). The liquid return junction is connected with a BRANCH component which simulates the zone of feedwater injection. This BRANCH is connected downward with the downcomer which is simulated by an ANNULUS component with 6 volumes. Above the feedwater injection branch an ANNULUS component is arranged which represents the volume beside the separator and steam dryer. This ANNULUS, divided in three volumes is connected with the steam expansion tank at the other end. Thus a Bypass to the SEPARATOR component is simulated.

The two main steam pipings, each simulated by a BRANCH component, are connected with another BRANCH component simulating the branching to the turbine valves and the main steam header. This branching to the turbine valves is connected with a servo valve component (SRVVLV), which represents both the turbine stop and control valves.

The turbine is simulated by a time dependent volume (TMDPVOL) component. Test calculations yield good results for the operating of the servo valve as long as the pressure in the "turbine" (time dependent volume) is in such an order of magnitude (about 40 bar) that no critical flow conditions are reached.

The BRANCH component, which simulates the main steam header, is connected with two time dependent volume (TMDPVOL) components representing the main heat sink (condenser) and the heat sink for the steam supplied feedwater heating, respectively. The junction between condenser and header is a servo valve (SRVVLV) component representing the six main steam Bypass valves; the other junction from the header is a time dependent junction (TMDPJUN) component with the time dependent flow rate of the steam extraction

During steady-state calculation the main steam Bypass valve is closed. Test calculations revealed that in regard to time step, consideration of the steam extraction in the nodalisation is recommendable.

The 6 main steam Bypass valves are condensed to one servo valve (SRVVLV) component. This servo valve is controlled by a proportional-integral controller. Thus the slightly different behaviour of the six valves cannot be simulated. As specified one valve opens earlier than the other five valves due to different calibration. This fact has a great impact in the event of small control deviations as in this case.

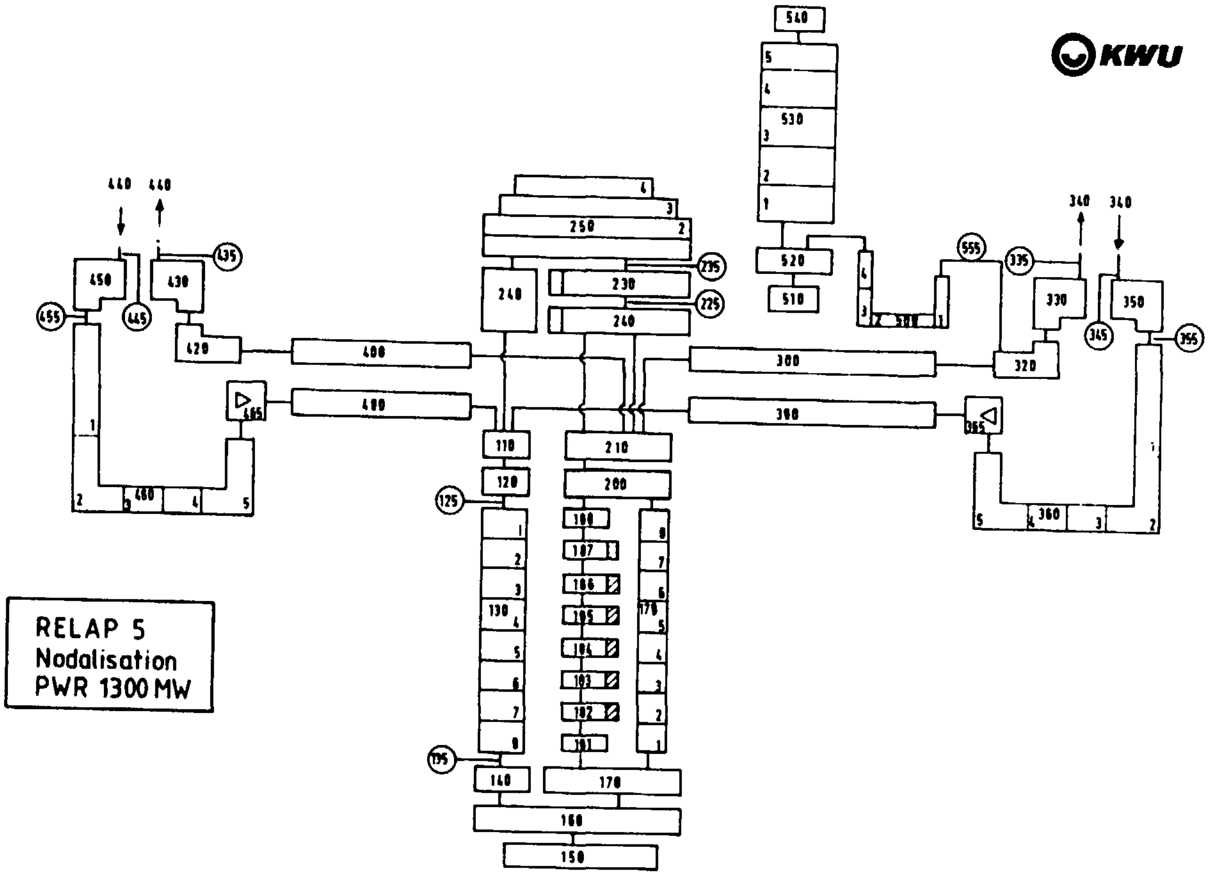
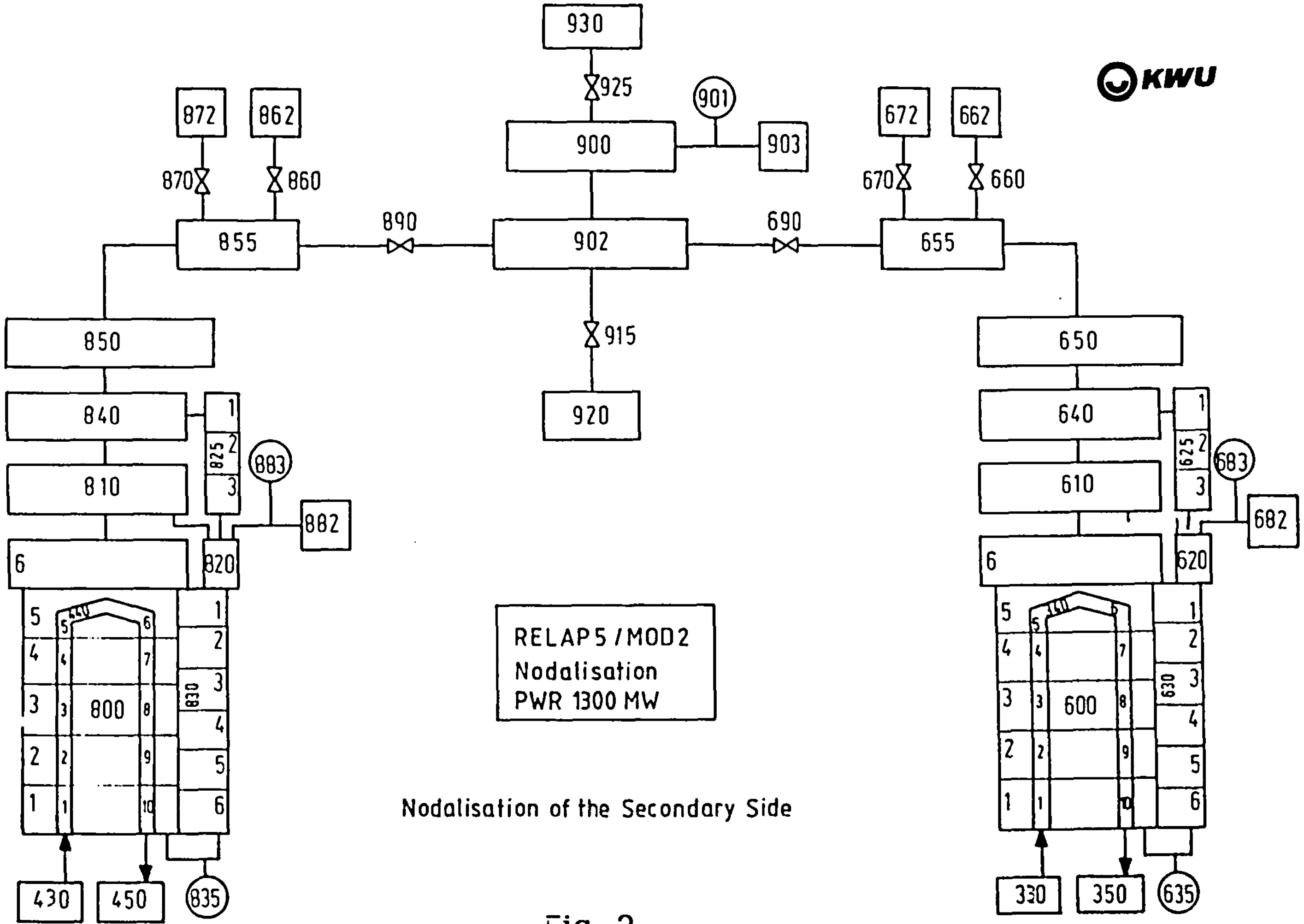


Fig. 1 Nodalisation of the Primary Side



Nodalisation of the Secondary Side

Fig. 2

5 Discussion of the Results

5.1 Results of Steady-State Calculation

For steady-state calculation the run option STDY-ST was chosen. Test calculations, performed with the run option TRANSNT yielded no significantly different results.

Great efforts were made to obtain the desired initial conditions although the nearly exact values were taken as input.

To adjust the primary coolant pressure to the setpoint, a time dependent volume component has to be connected with the "Threefold Loop" for a period of about 30 s. When connecting this time dependent volume component with the pressurizer, as seen in some input data listings, the pressurizer water level could not be adjusted to the setpoint.

The MS pressure was adjusted to setpoint by the turbine control valve.

Many difficulties arose when we tried to reach a balance between the power generated on the primary side and the power transferred to the secondary side. An iterative process was necessary to establish this level.

However, an exact balance could not be obtained within the desired setpoints. The remaining discrepancy of about 0.2 % was balanced by a time dependent volume (TMDPVOL) component connected with the pressurizer. A flow rate of about 0.6 kg/s was needed to render the energy balance stable.

The following boundary and initial conditions result from the steady-state calculation or are input data.

The reactor coolant temperature at SG outlet and consequently also at RPV inlet is influenced by the heat transfer correlations in the steam generator. The RPV inlet temperature could not be adjusted to the measured value. Due to the heat transfer package in RELAP5/MOD2 a value of about 294.6 °C results from steady-state calculations which is about 2.5 - 3.5 K higher than the measured values.	294.6 °C
No efforts were made to change the heat transfer correlations. A test run showed that the demanded temperature could be reached with an unrealistic high value of the heat conductivity of Incoloy (input data).	
The pressure in the steam generator was adjusted to about 65.8 bar, somewhat lower than the measured value JEA20 CP861.	65.8 bar
The feedwater flow rate amounted to 505 kg/s for the "Single Loop" and 1515 kg/s for the "Threefold Loop", respectively.	505 kg/s
The control of the main steam Bypass valve was simulated by a proportional-integral controller with LAG TERM (response time of the control valve), gain factor and follow-up time in accordance with the actual setting.	15 %/bar 15 s
The actual MS maximum pressure setpoint was set to 78.8 bar. With reactor trip the setpoint is lowered by 3 bar and subsequently increases with 20 bar/min until the setpoint is reached (X_d-Speicher).	78.8 bar 3 bar

The flow rate in the SG downcomer relative to the full load
feedwater flow rate (circulation rate) should be in the order 5.0
of 4.2, but with the best-estimated pressure drop coefficients
ascertained in the process of SG design a value of about 5.0
results from the steady-state calculation.

The steam generator water level (narrow range) was adjusted 12.13 m
to 12.13 m, somewaht lower than the measured value JEA20
FL901. This results in a wide range level of 12.0 m and a col- 12.0 m
lapsed level of nearly 12.6 m. 12.6 m

It was necessary to start with a lower quality in the upper zone of the downcomer than expected from the steady-state results to adjust the desired water level. Test runs performed to adjust the water level with a time-dependent feedwater flow rate controlled by the water level produced no acceptable results.

5.2 Results of the Transient Calculation

Starting from the steady-state results, the transient run was performed. It was assumed that the reactor trip occurs 15 s after beginning of the run. The transient run was terminated when a real time of 330 s was reached. Furthermore it was assumed that the trip was not triggered by the protection system but manually.

In the following discussion generally the measured values of the plant Loop 20 are compared with the values of the "Single Loop" obtained by computation. Sometimes, however, it is necessary to compare with measured values of Loop 10 or Loop 40 as discussed in Chapter 3.

5.2.1 Description of the Dynamic Behaviour of the Plant after Reactor Trip (Comparison of Calculation with Experiment)

From the reactor trip signal (input data) all absorber elements of the trip system fall into the core. Therefore the reactor power is reduced very rapidely to the decay power. With the trip signal the measured value of the ractor power JTK00 FX105 represents the fission power alone (s. Chap. 3.3.5). This fact is taken into consideration by calculating the CNTRLVAR 855 for adequate comparison with measurement. As to be seen from Fig. 4.1 good agreement between measurement and calculation is obtained.

The steam generator power follows the decrease of the reactor power somewhat delayed due to the stored heat in the primary system.

The decrease of the reactor power also causes a decrease of the temperature rise (see Chapter 3.3.4) because the reactor coolant pumps are still in operation. This decrease is in good agreement with the measured value of Loop 10 (Fig. 4.2). In the calculation the gradient is somewhat steeper than in the measurement, which may be due to a slight damping of the measured value. The measured value of Loop 20 is not suitable for comparison.

The coolant temperature at SG outlet (primary side) and with this also the temperature at RPV inlet approach the saturation temperature associated to the pressure in the steam generator. The deviation of the steady-state values between measurement and calculation is due to the heat transfer correlations implemented in RELAP5 which are obviously not optimal for this problem.

As mentioned in Chapter 3.3.3 the measured value of Loop 20 is always lower than the corresponding ones of Loop 10 and Loop 40. Therefore the calculated results are compared with the measured values of Loop 10.

Apart from the different initial values the calculation is in good agreement with the measured value JEC10 CT812 (Fig. 4.3).

Due to the decrease of the reactor power and the temperature rise, respectively, the RPV outlet temperature decreases, too. The sum of RPV inlet temperature and temperature rise of Loop 10 is compared with the calculated RPV outlet temperature (Fig. 4.4).

With respect to the deviation of the RPV inlet temperature from measurement as mentioned above good agreement exists between measurement and calculation.

The decrease of the (mean) coolant temperature results in a contraction of the coolant; thus a flow rate out of the pressurizer (outsurge) occurs causing a decrease of the coolant pressure and the pressurizer water level (see Chapter 3.3.1 and 3.3.2).

Fig. 4.5 and 4.6 reveal that the calculated results are in good agreement with the measurement. At the beginning of the transient the measured values of the coolant pressure are somewhat lower and the measured values of the pressurizer water level are somewhat higher. This may be due to that fact that the pressurizer water level control system is not simulated by the RELAP5 input procedure but was in action during the experiment.

From the reactor trip signal the turbine trip signal is triggered; thus the turbine stop valves close. In the following no energy is removed from the secondary side so that the MS pressure increases. This increase is in good agreement with the measured value of Loop 10 JEA10 CP851 (Fig. 4.7). The measurement in Loop 20 seems to be damped as discussed in Chapter 3.3.7.

If the pressure reaches the maximum pressure setpoint the Bypass valves open. Thus the MS pressure is limited and kept near the setpoint. The calculated MS pressure is in good agreement with the measured pressure (Fig. 4.7 and Fig. 4.8).

The deviation between measurement and calculation of the main steam Bypass valve position (see Chapter 3.3.6), however, is evident (Fig. 4.9). It is suspected, that this discrepancy is due to the fact that the Bypass valves are not correctly modelled in the calculation as mentioned in Chapter 4.2 (one valve opens earlier than the others).

The reduction of the steam generator power causes a collapse of the steam bubbles in the steam generator, resulting in a decrease of the steam generator water level. Later on, the SG is filled up to the setpoint by the feedwater control.

The calculated decrease of the water level (see Chapter 3.3.9) at the beginning of the transient is in good agreement with measurement; this applies to both narrow (Fig. 4.10) and wide range (4.11). After this decrease the calculation predicts a sharp increase of about 0.6 m to a maximum which does not appear in the measurement. In the calculation both the collapsed level as well as the narrow and wide range levels show this intermediate maximum.

No explanation has been found for this deviation between calculation and measurement. Due to the good resolution at the beginning of the transient a damping of the measurement can be excluded. It is assumed that this effect results from deficiencies in the modelling of the separator in RELAP5/MOD2.

Triggered by the reactor trip signal the full load feedwater control valves close and the low load control valves open. In the calculation the feedwater flow rate is given by input data. Fig. 4.12 demonstrates that the flow rate for the input is properly choosen, because the input data, which are derived indirectly from various measurements (see Chapter 3.3.8), are in good agreement with the measured value if the time dependent data are damped by a LAG term with a time constant of 20 s.

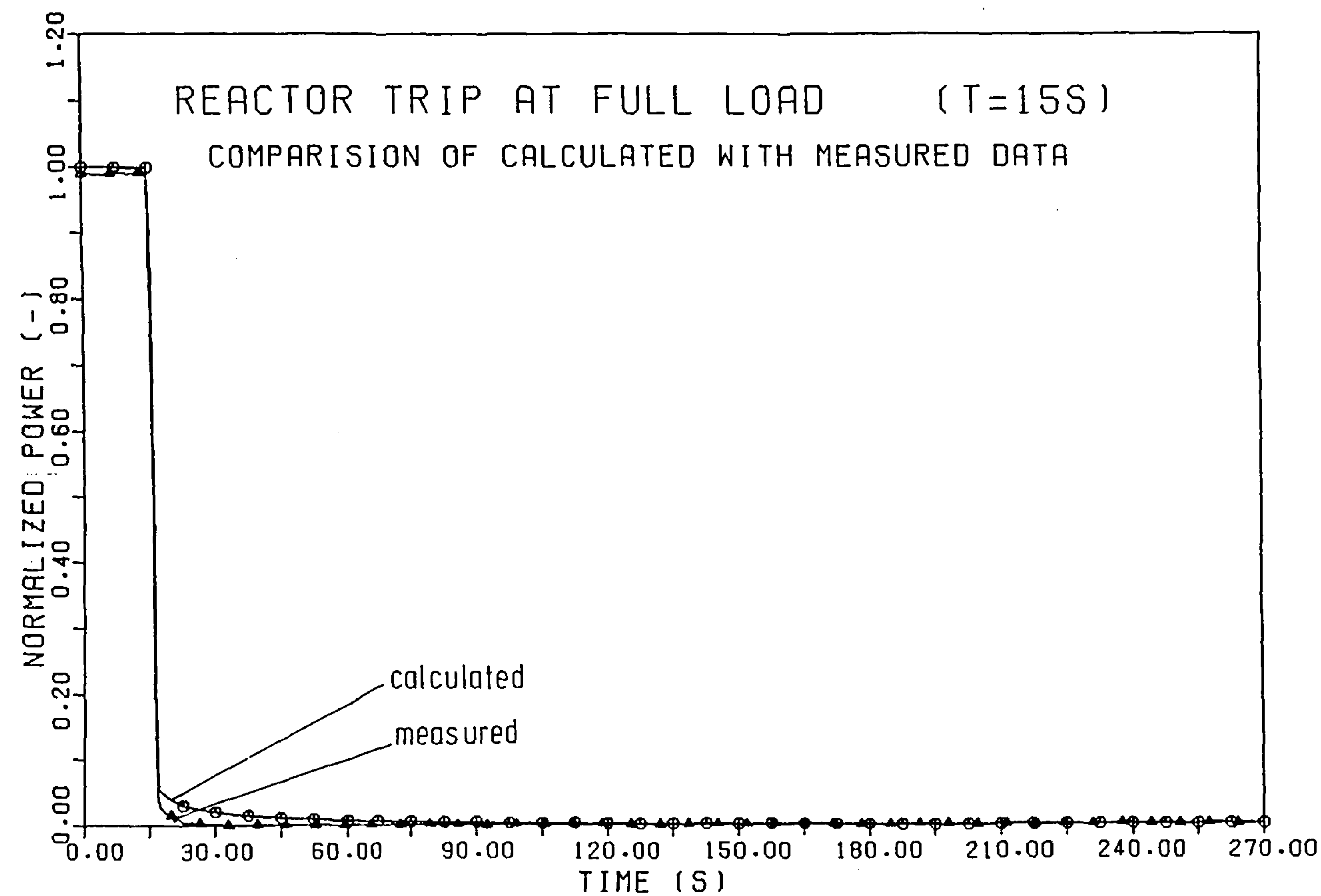


FIG. 4.1 REACTOR POWER

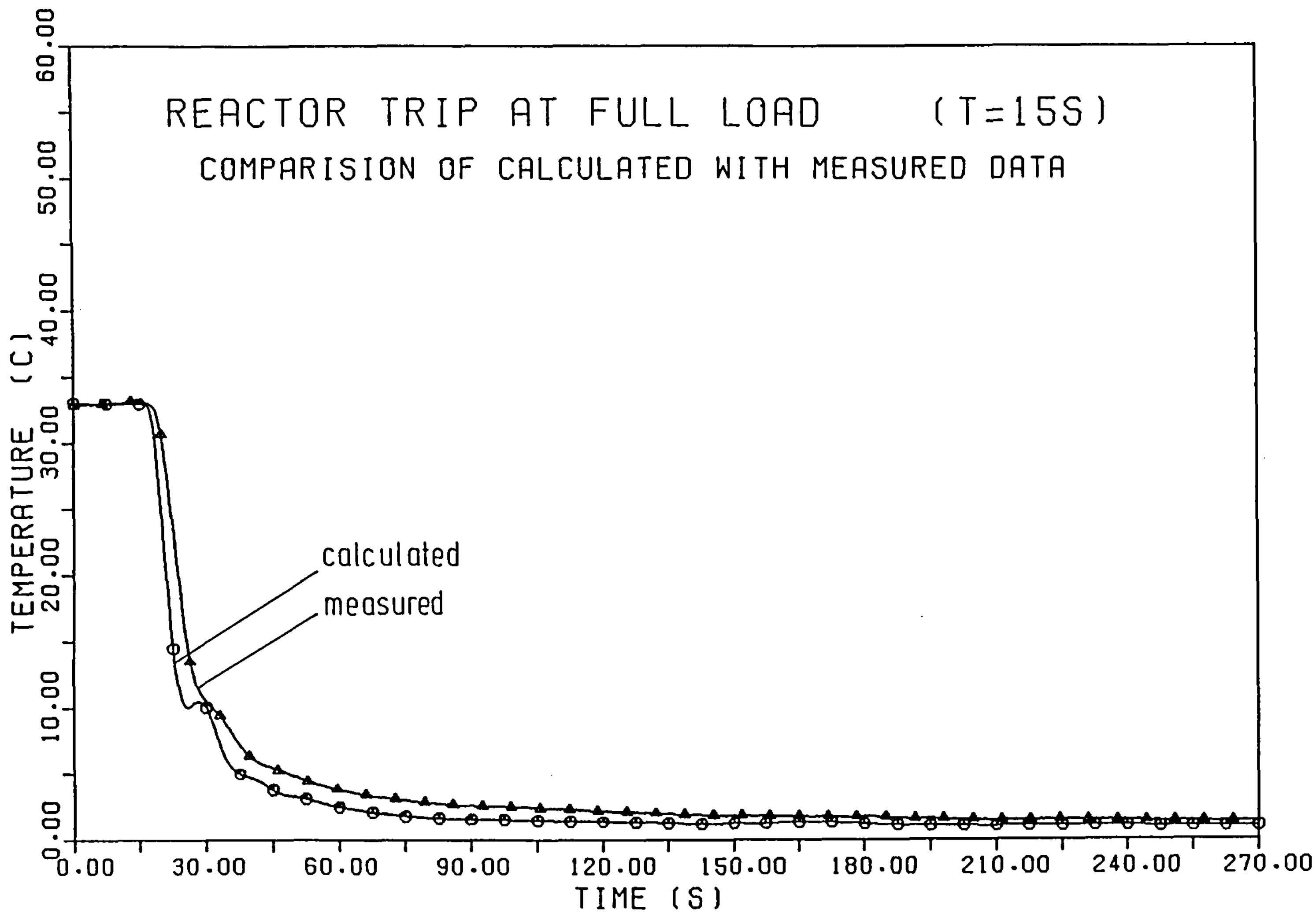


FIG. 4.2 TEMPERATURE RISE

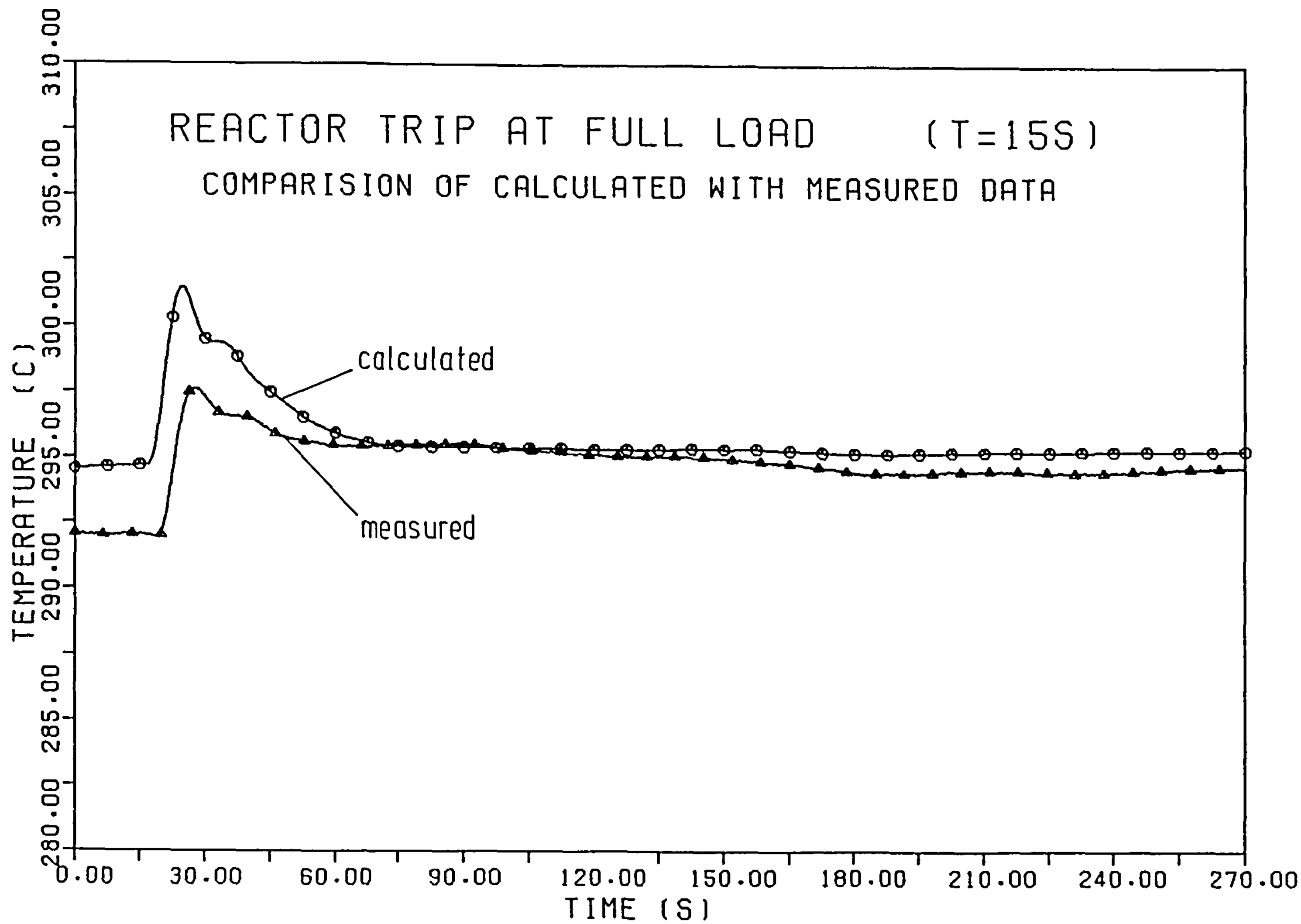


FIG. 4.3 RPV INLET TEMPERATURE

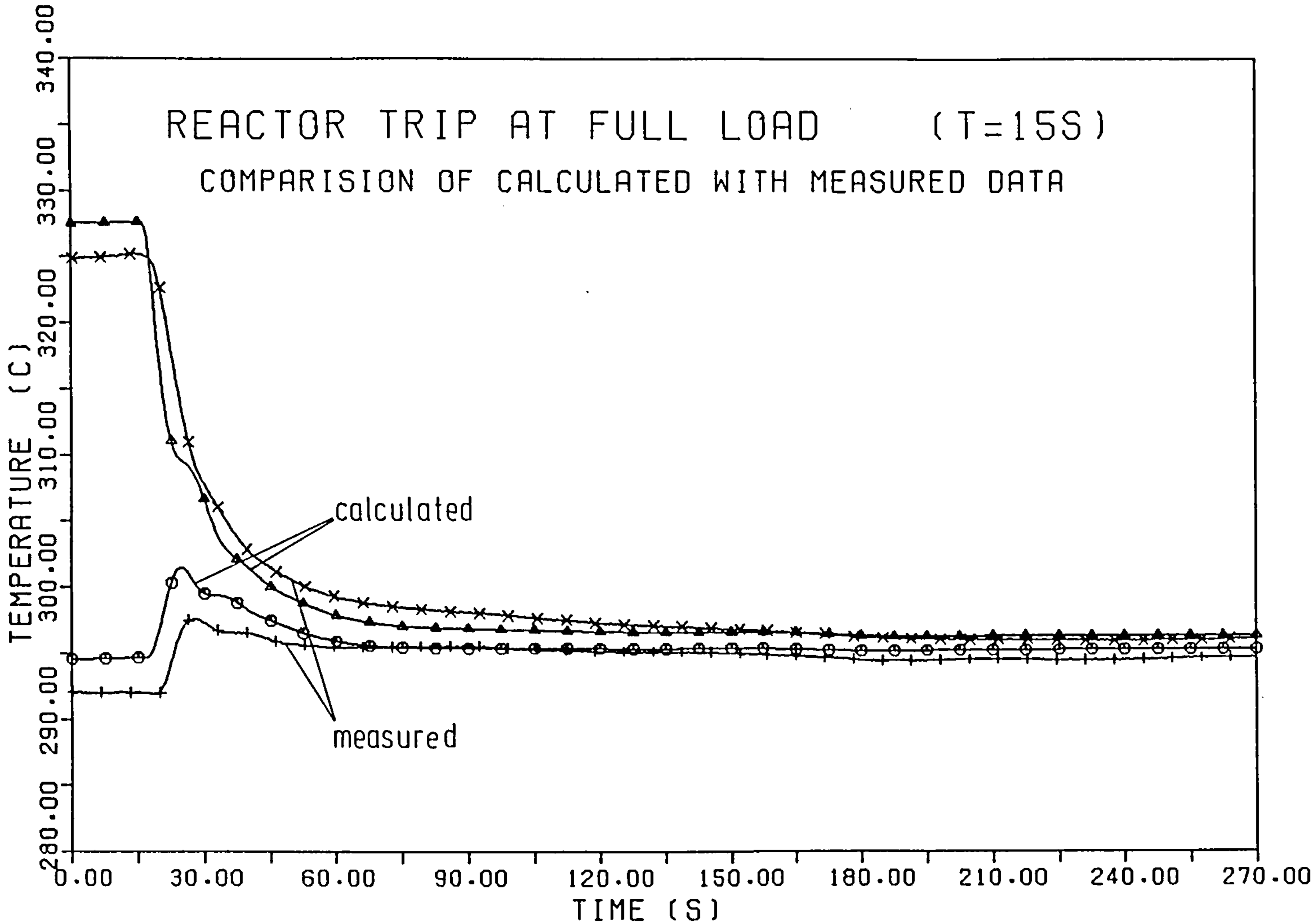


FIG. 4.4 RPV INLET AND OUTLET TEMPERATURE

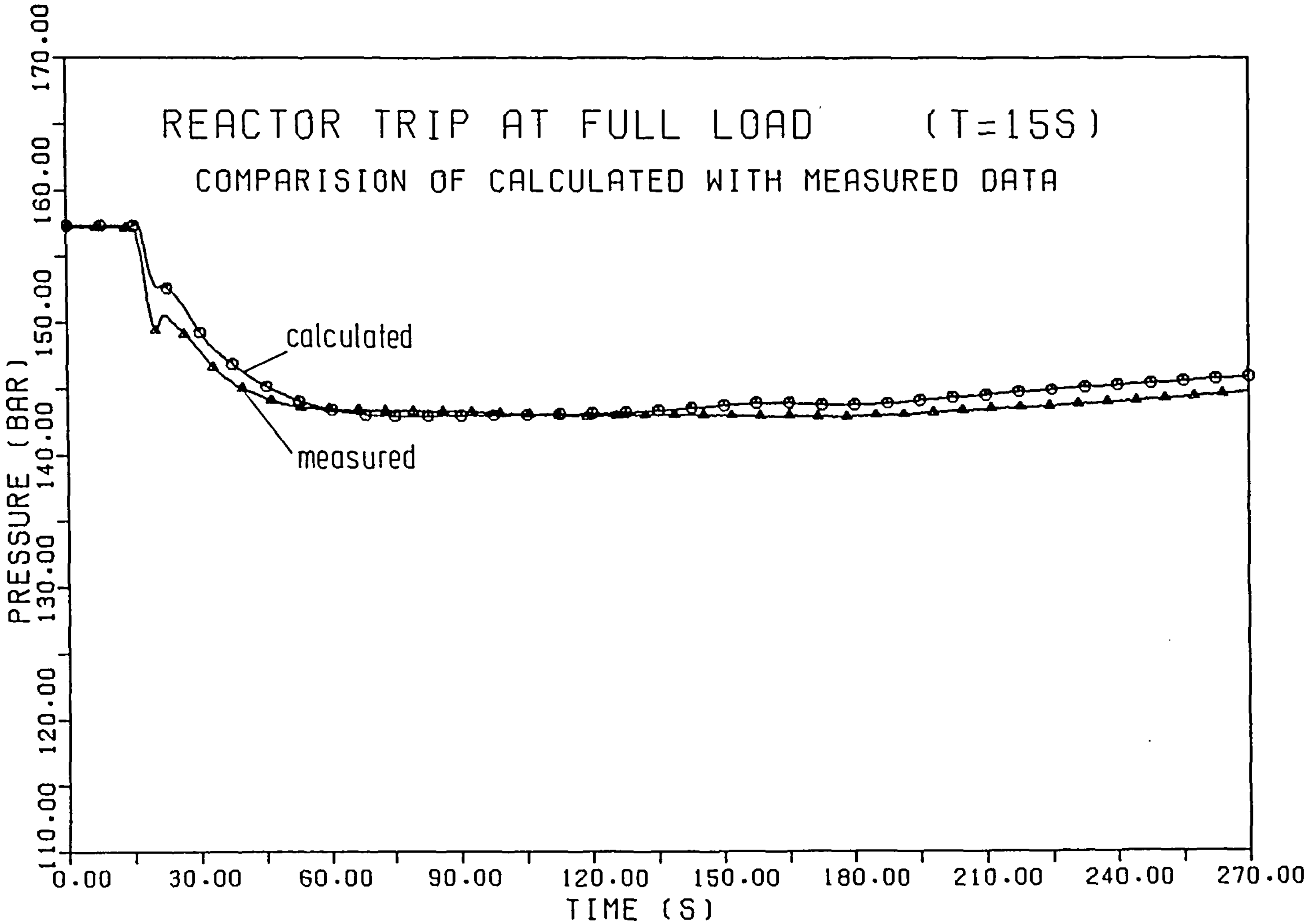


FIG. 4.5 REACTOR COOLANT PRESSURE (HOT LEG)

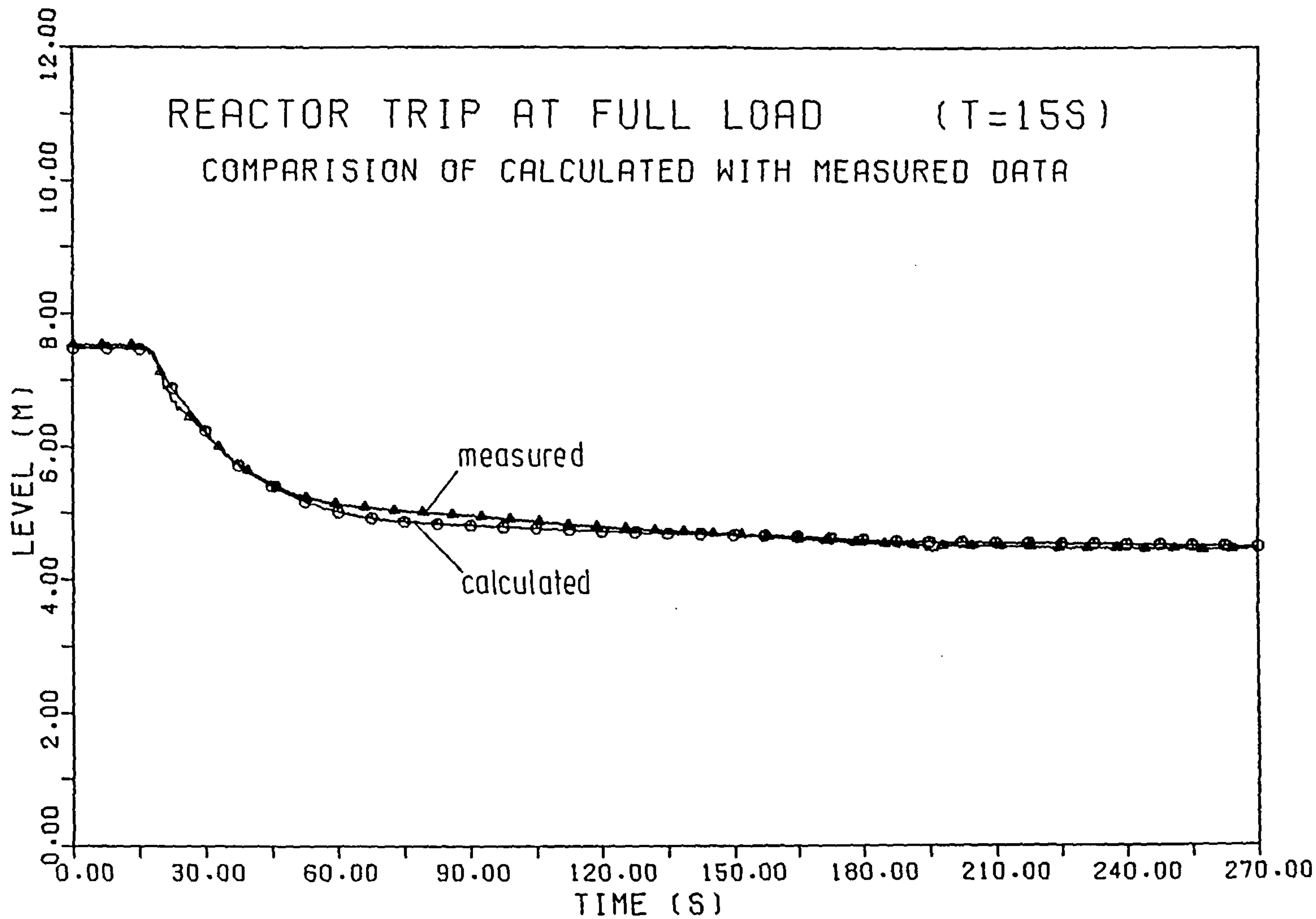


FIG. 4.6 PRESSURIZER WATER LEVEL

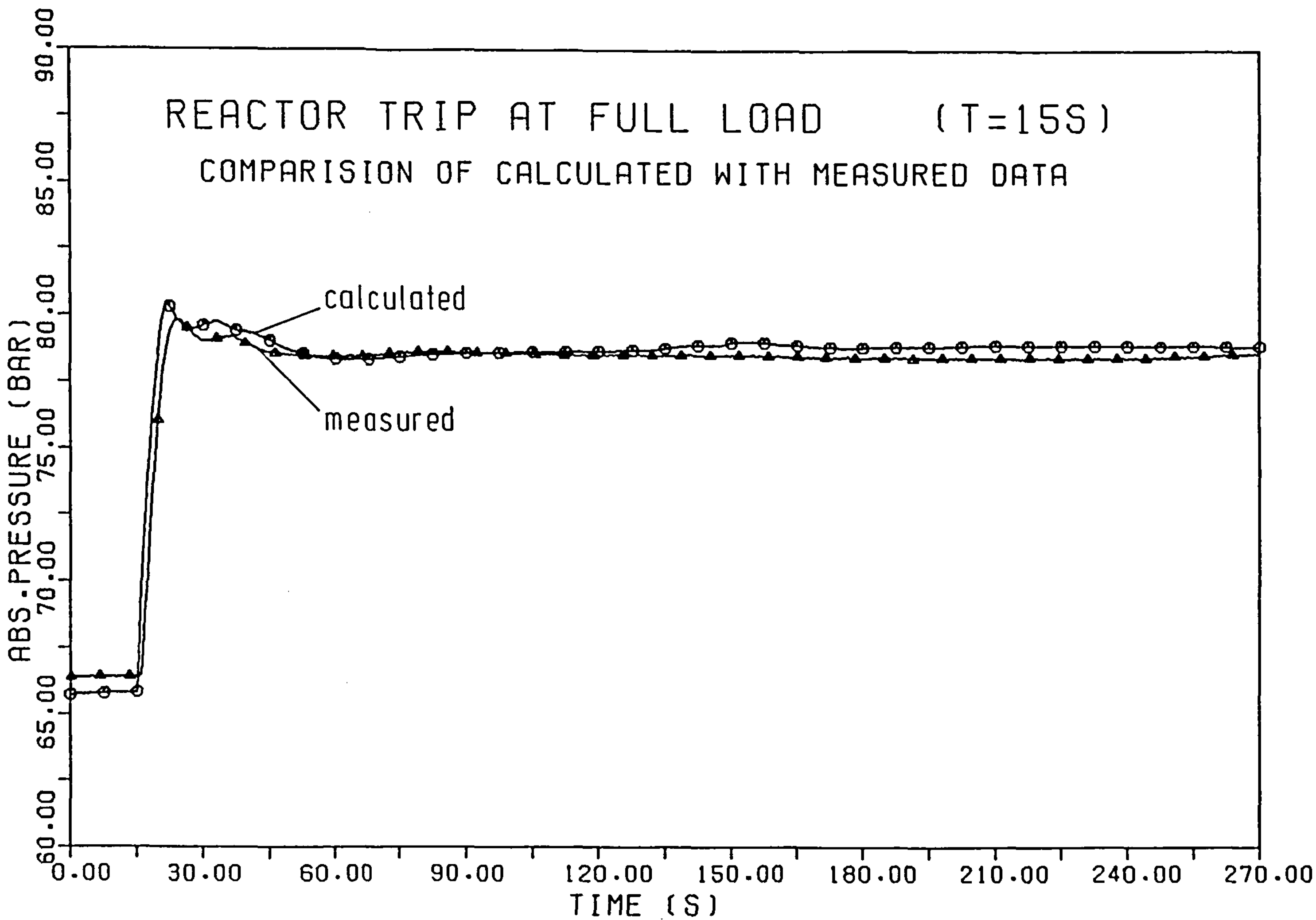


FIG. 4.7 MS PRESSURE IN SG 10

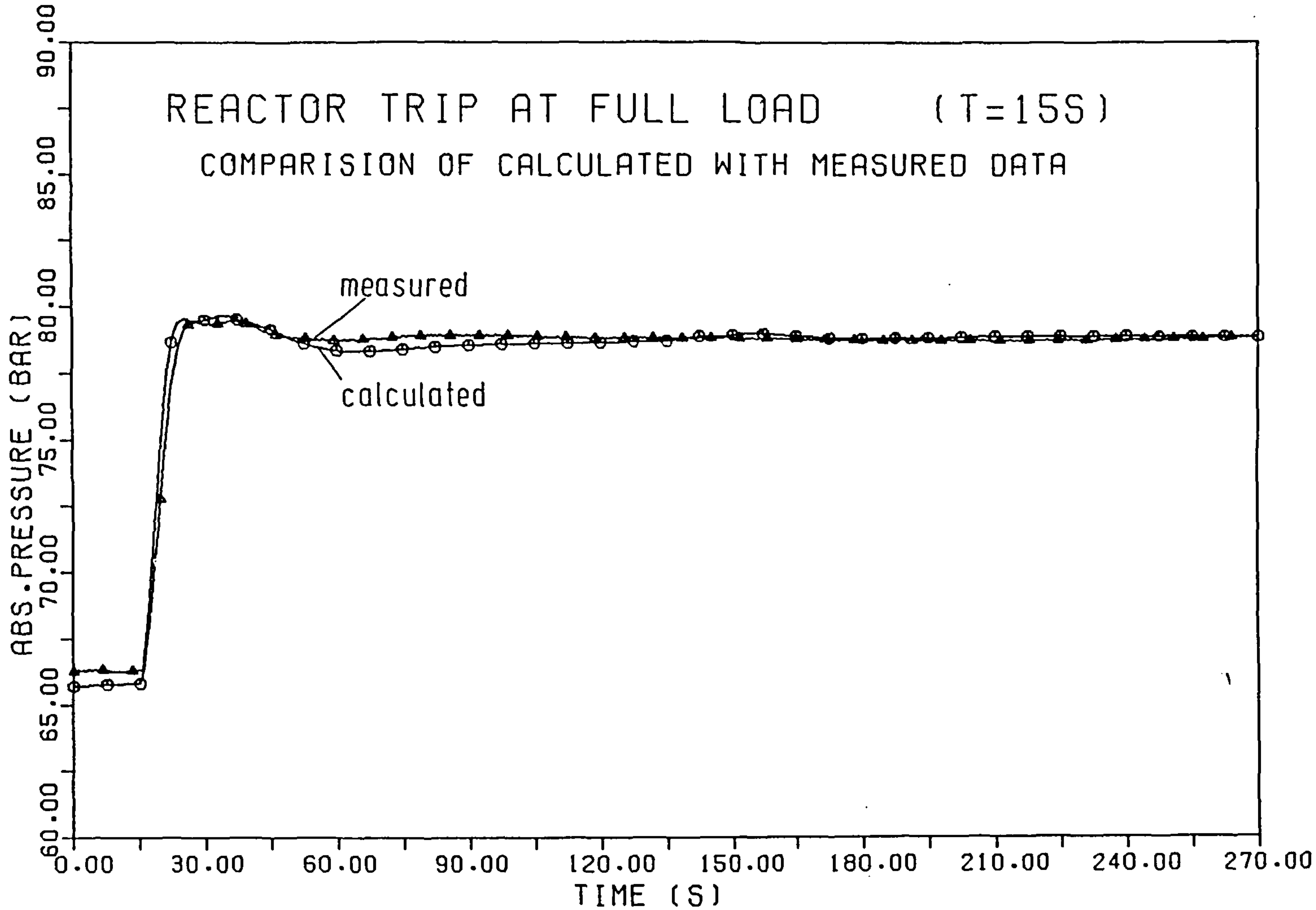


FIG. 4.8 MS PRESSURE IN SG 20 (DAMPED)

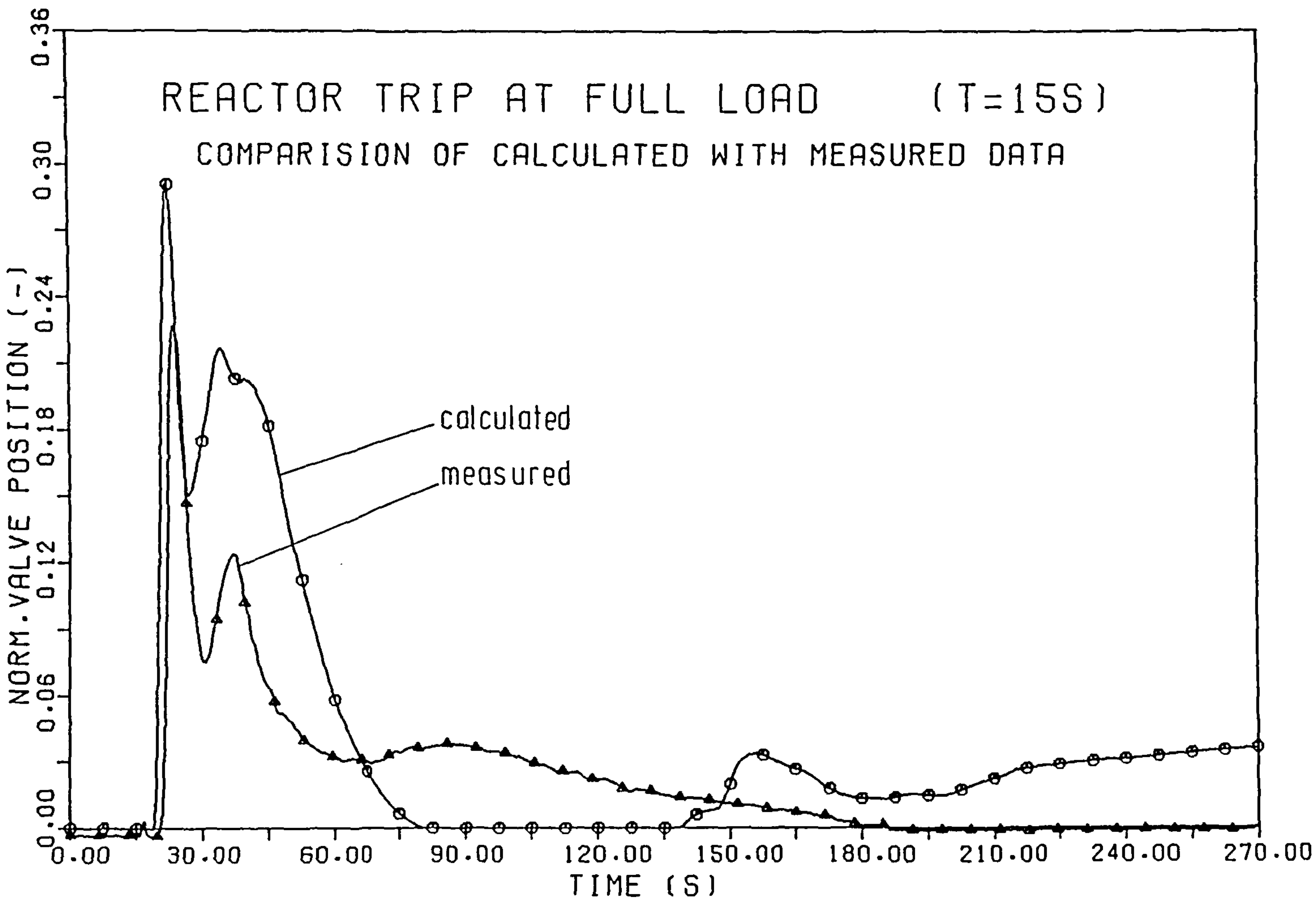


FIG. 4.9 BYPASS VALVES (SUM)

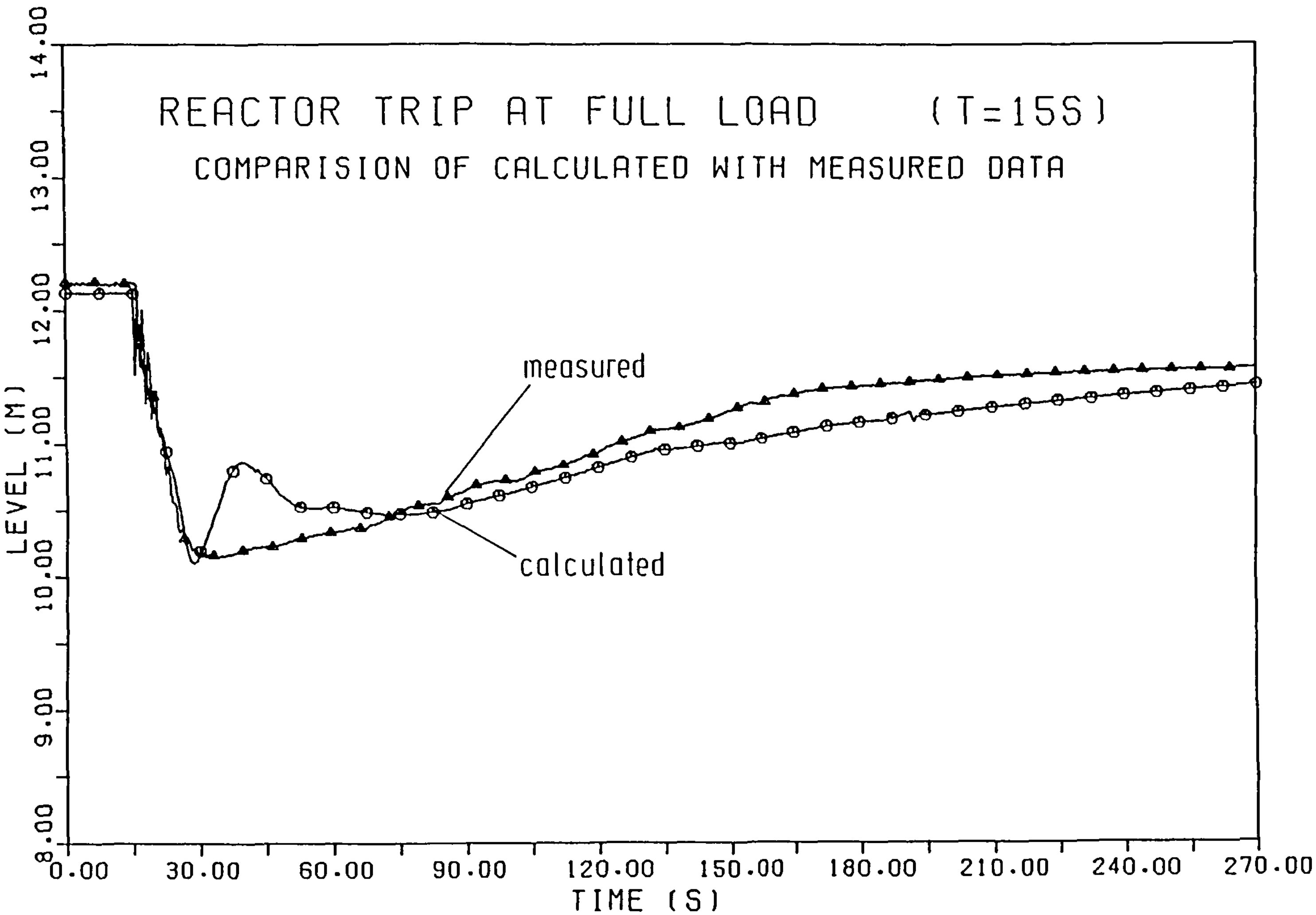


FIG. 4.10 SG WATER LEVEL (NARROW RANGE)

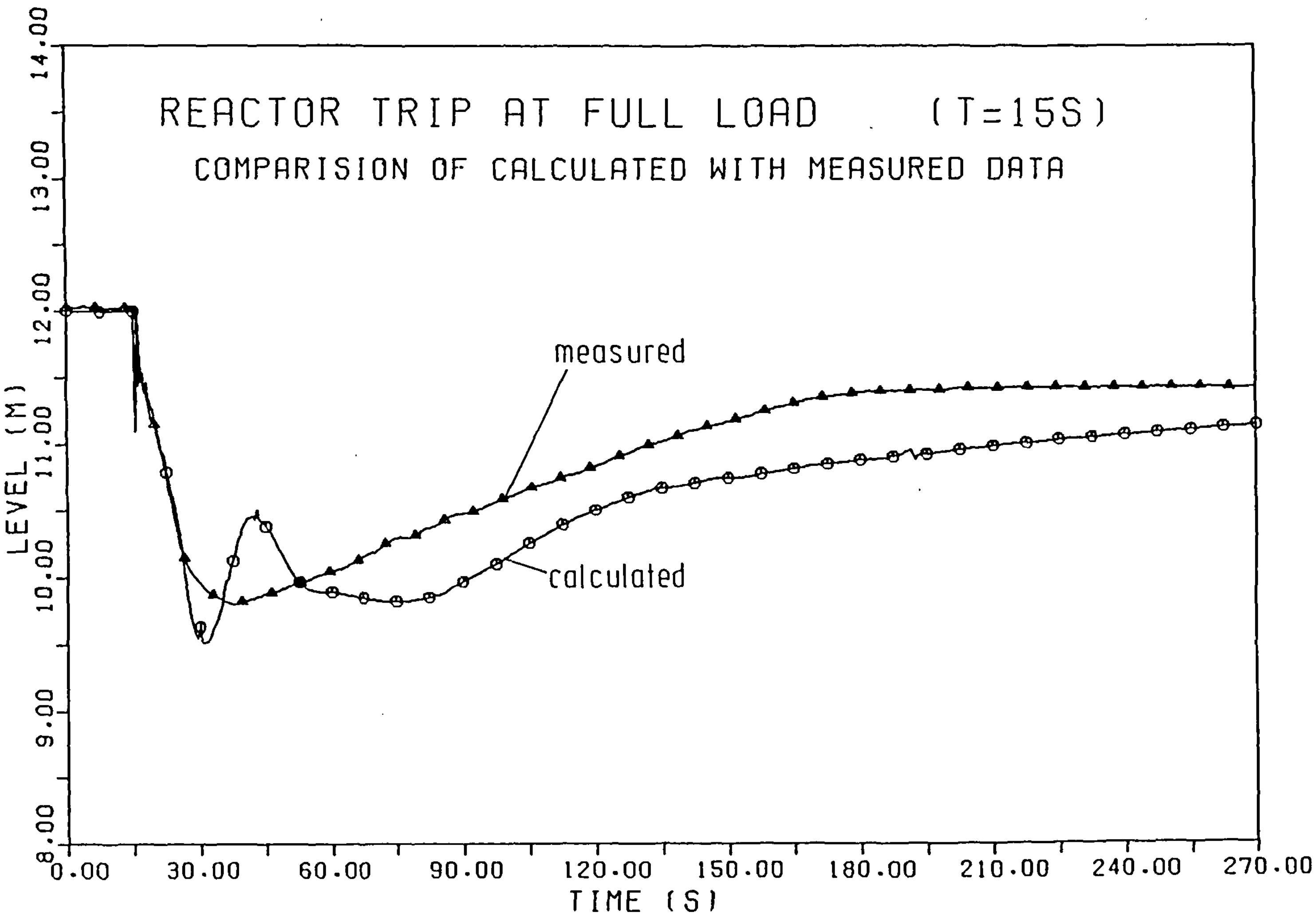


FIG. 4.11 SG WATER LEVEL (WIDE RANGE)

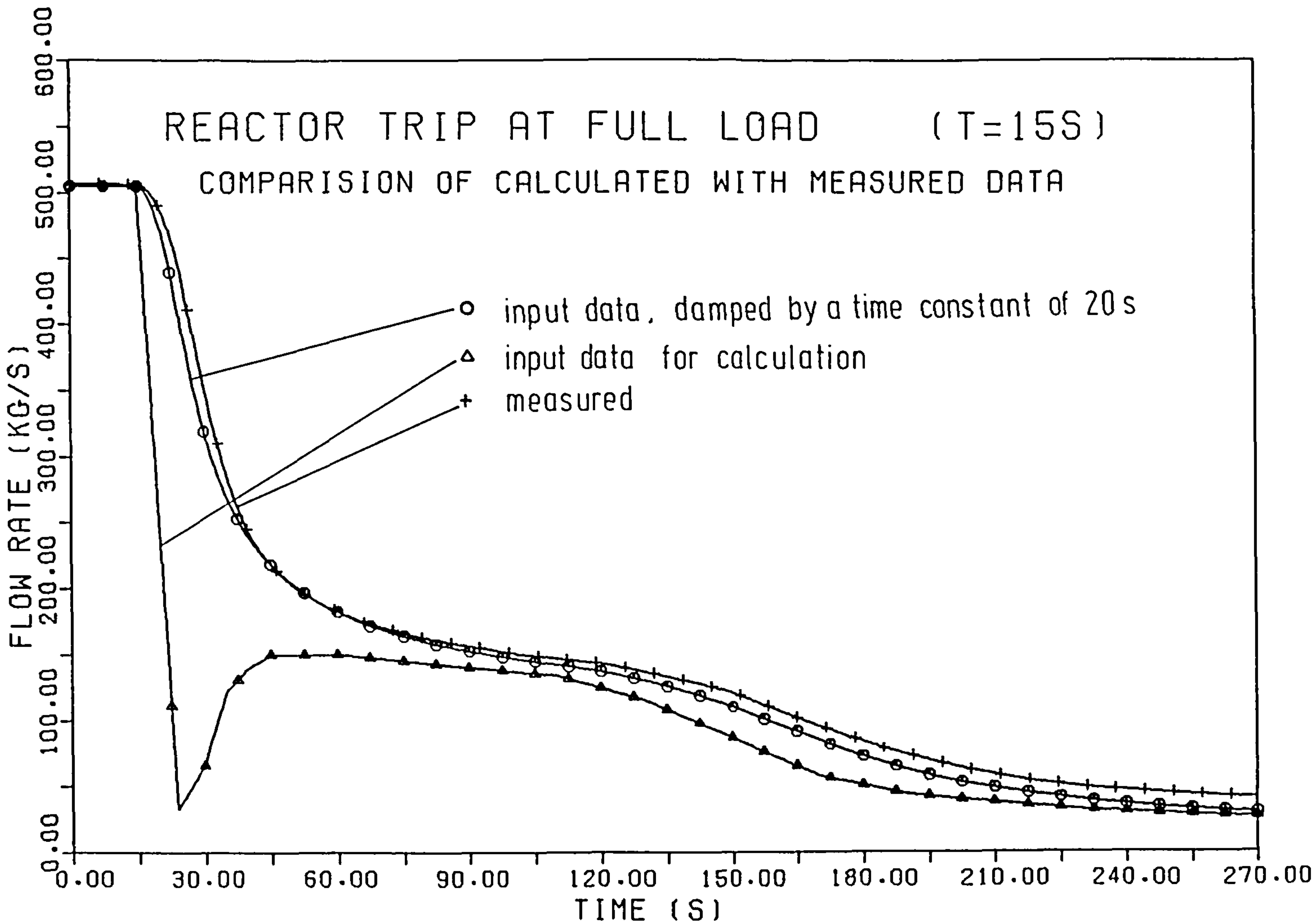


FIG. 4.12 FEEDWATER FLOW RATE

6 Runs Statistics

a) Steady-State Run

Time of Transient	50	s
CP Time	192	s
Maximal time step	$5.0 \cdot 10^{-2}$	s
Minimal time step	$1.25 \cdot 10^{-2}$	s

b) Transient Run

Time of Transient	330	s
CP Time	1448	s
Minimal time step	$3.9 \cdot 10^{-3}$	s
Maximal time step (controlled by courant limit)	$6.25 \cdot 10^{-2}$	s
Attempted ADV	7907	
Repeated ADV	87	

Time of Transient	CPU
0.	3.0
50.	261.0
100.	787.2
150.	934.2
200.	1078.6
250.	1219.8
300.	1361.4
330.	1448.0

c) General Statistical Data

Volumes	146
Junctions	152
Heat Structures	29
Mesh Points	204

7 Summary and Conclusions

The comparison of the results of RELAP5/MOD2 calculations with the commissioning test results shows very good agreement between measurement and calculation in general.

The only significant deviation between calculated and measured values is the RPV inlet temperature coming from heat transfer correlations in RELAP5/MOD2 not optimized for this problem. Furthermore, for a short period a deviation exists between the calculated and measured value of the SG water level about 15 s after beginning of the transient, which is assumed to be due to the separator model in the SG model.

In general, a critical review of the measured values is needed before comparing them with calculated results. A deviation of calculated values from measured data is not necessarily resulting from a bad calculation or not optimized modelling, but perhaps due to a faulty measurement or the measurement is affected by instrumentational short-comings and characteristics, resp. (for instance a damping of the measurement) or the measured value is only at the upper or lower limit of the allowed uncertainty range.

8 Comparison with Recalculations with another Transient Code (NLOOP)

8.1 Brief Description of NLOOP

The multiple-loop computer code NLOOP is employed to investigate the dynamic behaviour of PWR power plants in the event of in-service upsets and accidents. In the field of transient analysis the NLOOP code is the standard design and analysis tool of SIEMENS-KWU in the German licensing procedure. The range of application can be seen from Table 8.1

The code contains models for major components of the primary and secondary side, for important auxiliary systems and for the essential control, limitation, protection and interlock logic.

The range of components and systems normally considered for transient analysis of a KWU PWR plant is presented in Table 8.2.

The fluid in the primary coolant system is treated as homogeneous. Temperature non-equilibrium is allowed in the pressurizer, steam generators, feedwater tank and in the reactor pressure vessel head. The one-dimensional conservation equations of mass and energy are integrated by an explicit numerical method. Node/flow path networks are used to model the flow rates in the primary coolant loops and in the main steam and feedwater systems.

Adaptation to different plants is by modification of the input data and replacement of some modules, especially in the field of instrumentation and control system.

In the following, the main characteristics of the NLOOP code are listed:

- Point kinetics model
- Average coolant channel
- All kinds of reactivity effects can be simulated including recriticality after shut-down
- reactor coolant loop, incl. piping, RCP, SG, RPV, reactor pressure vessel head
- pressurizer
- heat transfer in SG
- natural circulation mode
- backflow in loops

- piping system on the secondary side (feedwater/main steam) incl. valves
- feedwater supply system
- turbine/bypass
- all important I & C
- relevant auxiliary systems including valves and controls
- reactor protection system
- safety systems

Two main restrictions to be mentioned are:

1. The coolant within the primary coolant circuit is treated as homogeneous. Therefore, LOCA accidents for which phase separation effects are determining factors should not be calculated by NLOOP.

2. As a point kinetics model for the fission power is used, the core power distribution preset by input data remains unchanged during the calculation. Consequently, any reactor power distribution control and limitation system actions are not considered.

Operation	Normal Operation	Load ramps/Load steps
	Upset Operation	Turbine trip/Load rejection/Reactor scram/Failure of reactor coolant pumps/Malfunction of auxiliary systems
Disturbances and Accidents	Reactiviy Disturbance	Start-up fault/Rod insertion/Rod drop/Rod withdrawal/Rod ejection/Boron dilution
	Disturbance of Heat Removal	Inadvertent opening of bypass station/Turbine trip without rod drop or with bypass station unavailable/Loos of auxiliary power supply/Inadvertent closing of main steam isolation valves/Malfunction of feedwater supply
	Loss of Secondary-Side Coolant	Inadvertent opening of MS safety or relief valves/ MS line break/Feedwater line break
	Loss of Primary Coolant	Steam generator tube rupture/Inadvertent opening of pressurizer valves
Specials	ATWS	Loss of heat sink/Loss of heat sink and auxiliary power supply/Max. increase in MS flow/Max. reduction in feedwater supply/Max. reduction in reactor coolant flow/Max. reactivity insertion/Inadvertent opening of primary safety valve/Max. subcooling caused by feedwater supply

Table 8.1 Application of NLOOP

Thermohydraulic Systems			Instrumentation and Control		
Primary	Secondary	Control	Limitation	Protection	Interlocking
Coolant circuit - RPV with core - coolant lines - coolant pumps - pressurizer - relief tank Volume control system Extra borating system Chemical feeding system Emergency core cooling system	MS line system - turbine/generator - bypass station - isolation valves - relief/safety valves Feedwater system (condenser) - feedwater tank - feedheaters - pumps, valves Emergency feedwater pumps	Coolant temp. Coolant pressure Pressurizer water - level turbine/ generator Feedwater	Reactor power Rod movement Rod insertion Coolant inventary/ - presssure/temp. grad.	Reactor scram Safegurads	of all simulated active compo-nents
Steam generator					

Table 8.2 Scope of Simulation in NLOOP

8.2 Comparison of Analysis Results between NLOOP and RELAP5 on the Basis of a Reactor Trip Transient

Within the terms of reference of

"Code Assessment Agreement"

between the United States Nuclear Regulatory Commission (USNRC) and the Federal Minister of Research and Technology (BMFT) of the Federal Republic of Germany, a post-test calculation was performed at SIEMENS-KWU using the computer code

RELAP5/MOD2

for the test

"Reactor Trip at Full Load (D-100-301)"

performed during commissioning of the Philippsburg 2 nuclear power plant. A post-test calculation was also performed for the same test using the SIEMENS-KWU developed code

NLOOP

The results obtained from use of each of these codes were compared with the results of the commissioning tests.

The results of this comparison can be summarized as follows:

The two computer codes NLOOP and RELAP5 yield almost identical results, which match up very well with the results of actual testing.

In RELAP5, the main steam pressure rises somewhat more steeply than in NLOOP, and thus matches up very well with the pressure measurements during the test (pressure rise from about 66 bar to 80 bar in 6 s in RELAP5 and in 8 s in NLOOP).

The initial value for the RPV inlet temperature in RELAP5 shows a distinct discrepancy from the test reading; it was not possible to match the RELAP5 result with the test result under the terms of the steady-state computation. This discrepancy likewise influences the initial values of the RPV outlet temperature and the average coolant temperature. This in turn affects the transient history of the coolant pressure, which shows good agreement with the test results only by comparison, sinc e the error due to the high value for the average coolant temperature is compensated by the fact that the coolant flow extraction by the pressurizer level control system is not modelled by RELAP5/MOD2 input.

The RELAP5/MOD2 computation shows for steam generator water level a minimum followed by a pronounced maximum which was not observed either during the commissioning test or in the NLOOP calculation. This maximum appears unrealistic. Both programs agree well on the minimum. In the long-term history, both computations diverge from the actual test results, although RELAP5 remain closer to the results measured.

To sum up, it can be concluded that both codes are equally suitable for post-test calculation of this transient, as can be seen from the figures 5.1 - 5.4 (transient history of coolant pressure, RPV inlet temperature, MS pressure, SG water level).

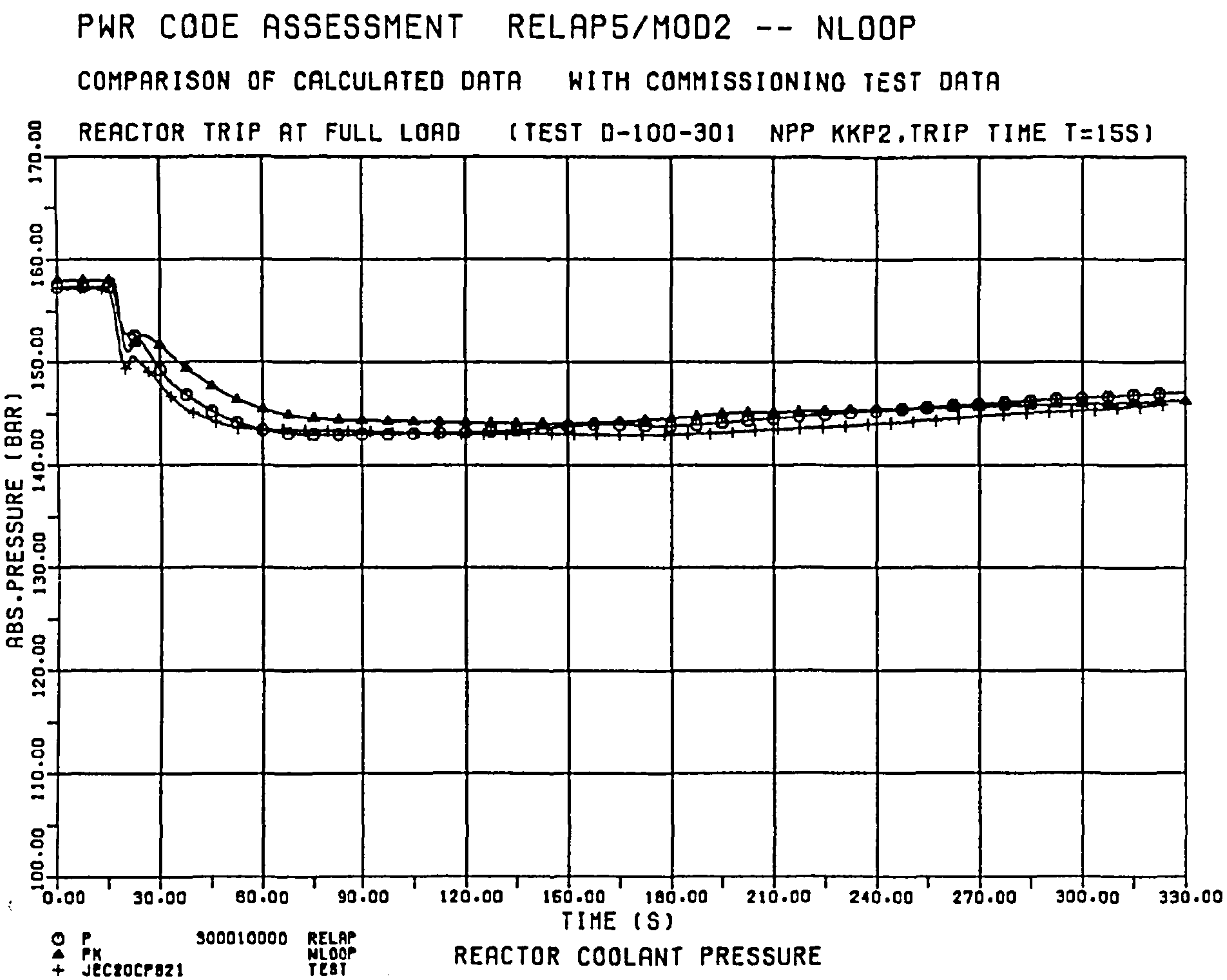
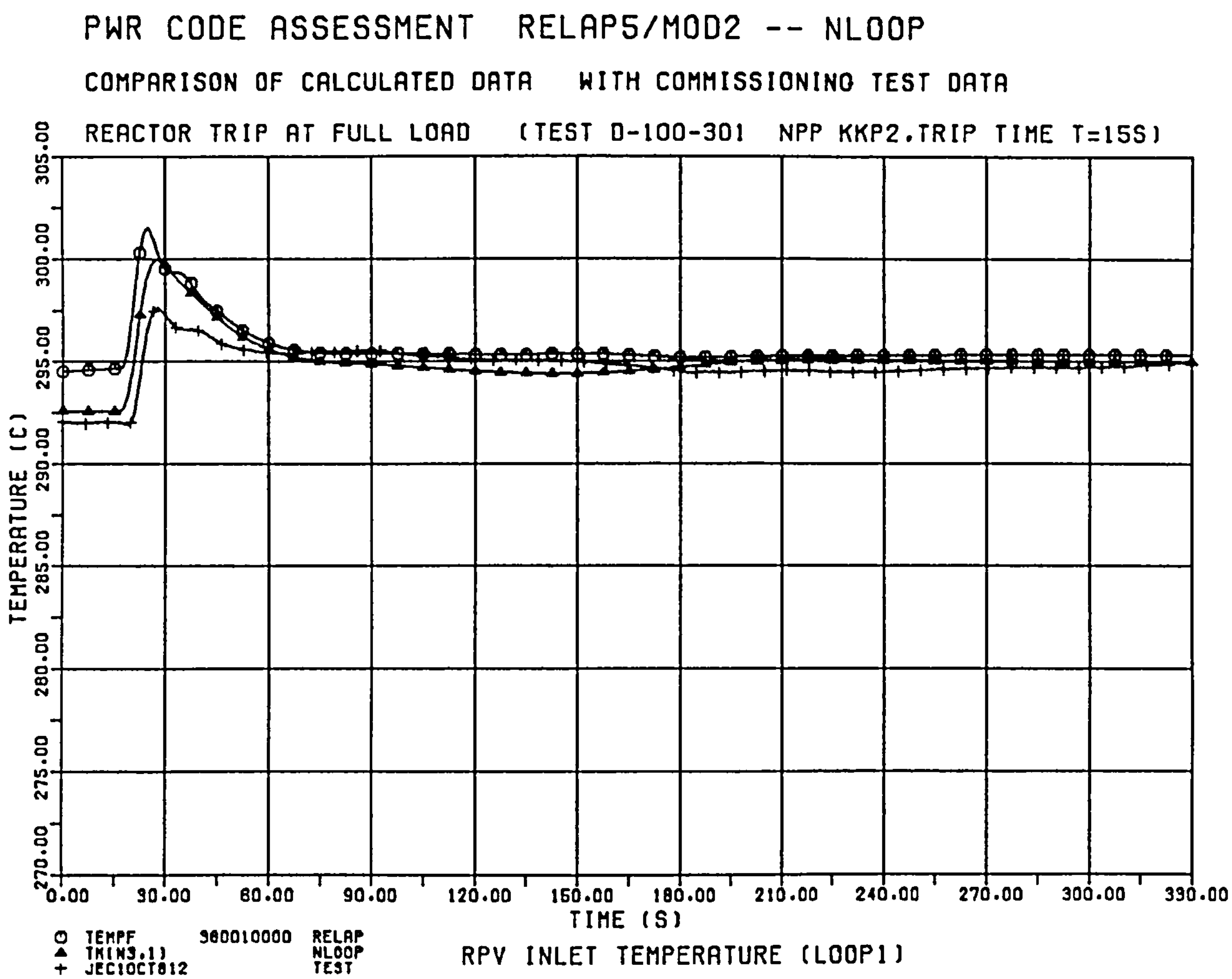

PWR CODE ASSESSMENT RELAP5/MOD2 -- NLOOP
COMPARISON OF CALCULATED DATA WITH COMMISSIONING TEST DATA
REACTOR TRIP AT FULL LOAD (TEST D-100-301 NPP KKP2,TRIP TIME T=15S)
ABS.PRESSURE (BAR)
100.00
110.00
120.00
130.00
140.00
150.00
160.00
170.00
0.00
30.00
60.00
90.00
120.00
150.00
180.00
210.00
240.00
270.00
300.00
330.00
TIME (S)
P 300010000 RELAP
PK NLOOP
JEC20CP821 TEST
REACTOR COOLANT PRESSURE
RUN 77/85 AND 125/86/GE

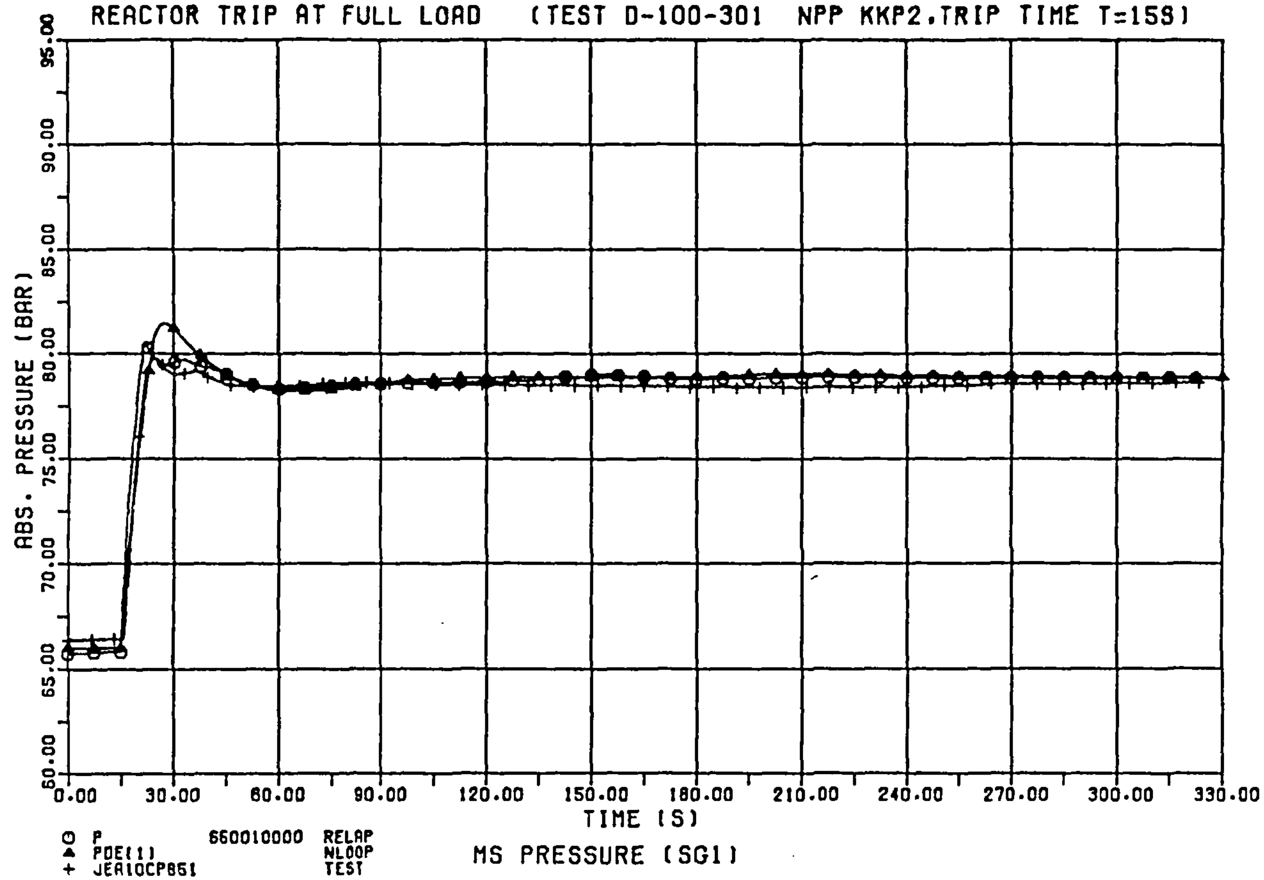
Fig. 5.1


PWR CODE ASSESSMENT RELAP5/MOD2 -- NLOOP
COMPARISON OF CALCULATED DATA WITH COMMISSIONING TEST DATA
REACTOR TRIP AT FULL LOAD (TEST D-100-301 NPP KKP2.TRIP TIME T=15S)
TEMPERATURE (C)
305.00
300.00
295.00
290.00
285.00
280.00
275.00
270.00
0.00
30.00
60.00
90.00
120.00
150.00
180.00
210.00
240.00
270.00
300.00
330.00
TIME (S)
TEMPF 360010000 RELAP
TK(N3.1) NLOOP
JEC10CT812 TEST
RPV INLET TEMPERATURE (LOOP1)
RUN 77/85 AND 125/86/GE
Fig. 5.2

PWR CODE ASSESSMENT RELAP5/MOD2 -- NLOOP

COMPARISON OF CALCULATED DATA WITH COMMISSIONING TEST DATA

REACTOR TRIP AT FULL LOAD (TEST D-100-301 NPP KKP2,TRIP TIME T=15S)



MS PRESSURE (SG1)

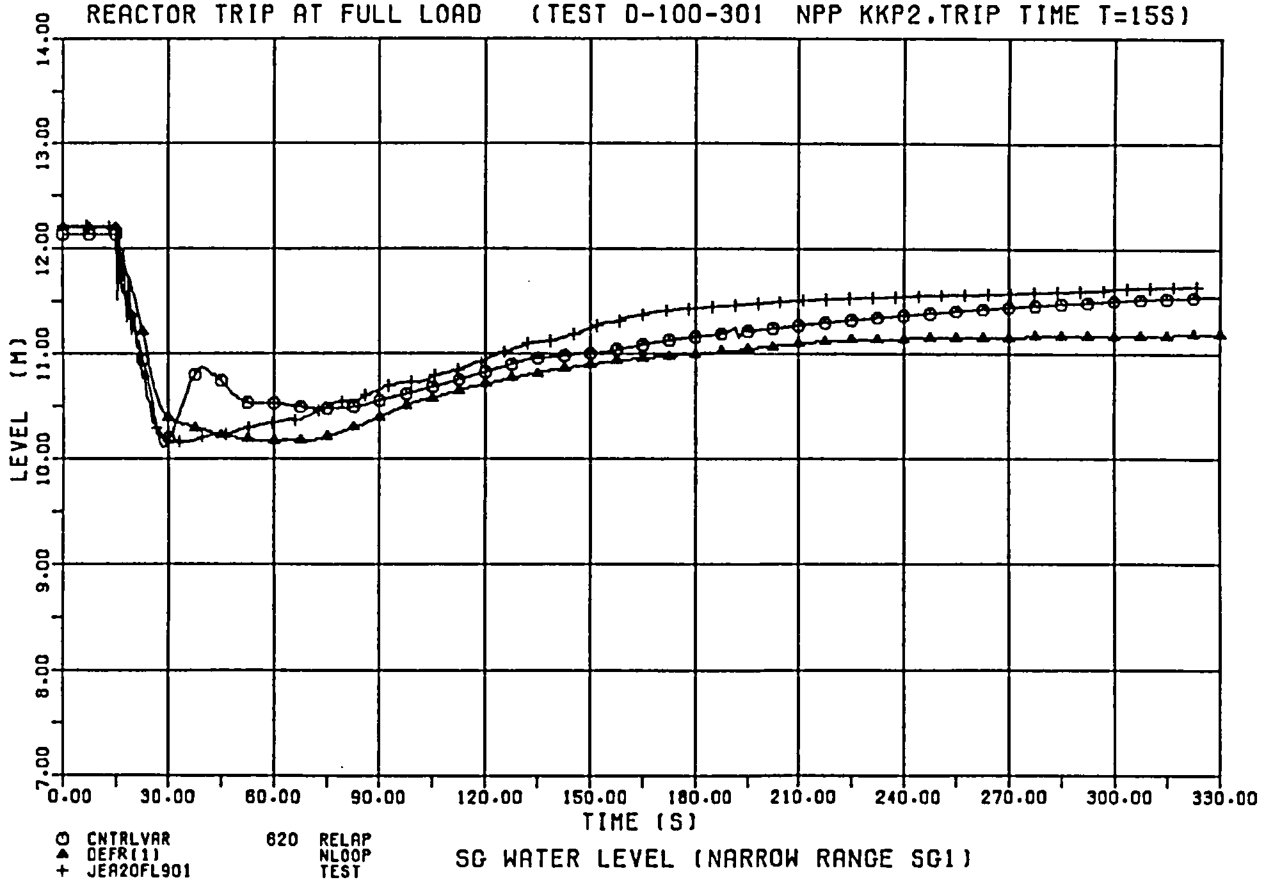
RUN 77/85 AND 125/86/GE

Fig. 5.3

PWR CODE ASSESSMENT RELAP5/MOD2 -- NLOOP

COMPARISON OF CALCULATED DATA WITH COMMISSIONING TEST DATA

REACTOR TRIP AT FULL LOAD (TEST D-100-301 NPP KKP2.TRIP TIME T=15S)



SG WATER LEVEL (NARROW RANGE SG1)

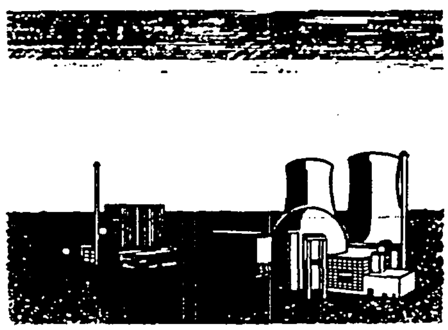
RUN 77/85 AND 125/86/GE

Fig. 5.4

Appendix A

Data Sheet of Philippsburg 2 Nuclear Power Plant

Philippsburg 2 Nuclear Power Station



The Rheinschanz Island, which is situated between the old and new beds of the Rhine near Philippsburg, about 30 km north of Karlsruhe, is the site of two nuclear power plants, Philippsburg 1 and Philippsburg 2, being constructed by Kraftwerk Union AG (KWU) for Kernkraftwerk Philippsburg GmbH (KKP).

The 1300 MW Philippsburg 2 nuclear power plant has been equipped with a pressurized water reactor as a nuclear steam generator, and is due to go into operation towards the end of 1982.

KWU Pressurized Water Reactor

The reactor, loaded with 103 tonnes of enriched uranium as the nuclear fuel, is designed for a thermal output of 3765 MW. 18,800 kg of water per second flow through the core of the reactor, where the uranium is arranged in 193 fuel assemblies. This water is heated in the process from 291 °C to 326 °C, but does not evaporate because it is maintained at a pressure of 158 bar. The reactor coolant transfers heat to the secondary steam cycle via four identical loops, each with its own circulating pump and heat exchanger - the so-called "steam generator". It is a characteristic of the pressurized water reactor that both loops of the nuclear steam generating system are completely separated from each other, and are both pressure-tight, so that it is impossible for radioactive substances to escape to the steam cycle.

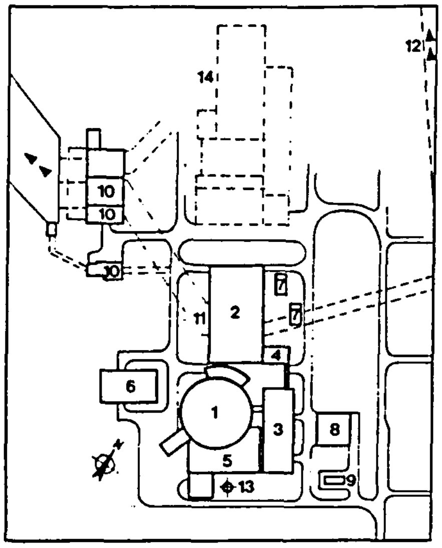
1362 MW from One Turbo-generator Set

On the secondary side of the four steam generators a total of 2061 kg of saturated steam at a temperature of 284 °C and a pressure of 68 bar are produced per second at full load. This flows into the turbine building and on to the single-shaft condensing turbine, which, along with the three-phase synchronous generator coupled to it, is currently one of the largest units manufactured in the world.

The turbine, comprising a high pressure section and a three-cylinder low pressure section with interposed moisture separator and superheater, has been specially developed by KWU for operation on saturated steam. The 1560 MVA-generator generates a terminal voltage of 27 kV; the windings of both stator and rotor are water-cooled.

Condenser Cooling Using Water from the Rhine and a Cooling Tower

The steam condenses in the condensers after leaving the turbine, and the condensate is then pumped back to the steam generators being simultaneously heated. The condensers in the Philippsburg 2 nuclear power plant are cooled either with Rhine water or with water that releases heat to the atmosphere via a cooling tower (height approximately 150 m, diameter at base 120 m). Temperature, oxygen content and level of the Rhine, and also prevailing weather conditions determine which of the possible methods should be used, or whether, in the event of fresh water cooling, the cooling water should be returned to the river via the cooling tower.



1 Reactor building
2 Turbine building
3 Switchgear building
4 Administration and amenities building
5 Reactor auxiliary systems
6 Emergency feed building
7 Unit transformer
8 Emergency power generator
9 Gas supply system
10 Cooling water pumps
11 Cooling water supply canal
12 Cooling water discharge canal
13 Vent stack
14 Power plant, unit 1

Safety Has Priority

All measures to guarantee environmental protection and technical safety have been taken for the Philippsburg 2 nuclear power plant. Thus, for example, the nuclear steam generating system together with the reactor auxiliary systems that are necessary for the reliable operation of the plant are housed in the reactor building. In accordance with KWU philosophy, this building has two shells positioned one inside the other. The inner shell, a steel sphere 56 m in diameter, is pressure-resistant and gas-tight in order to limit the effects of maximum credible accidents to the reactor to its interior. The outer reinforced concrete shell is resistant to all impacts, including that of a crashing aircraft.

Technical Data

Overall Plant

Thermal output of reactor	3765 MW
Output at generator terminals	1362 MW
Net electrical output	1281 MW

Reactor (pressurized water, enriched uranium)

Coolant and moderator	H_2O
-- No. of reactor coolant loops	4
-- Flow rate	18,800 kg/s
-- Operating pressure	158 bar
-- Inlet/outlet temperature	291.3 °C/326.1 °C
No. of fuel assemblies	193
Weight of uranium	103 t
No. of control rods	61

Fuel Assemblies (193, Fuel UO_2 - Pellets)

Fuel rods per assembly	236
-- Diameter	10.75 mm
-- Active length	3900 mm
-- Specific output	36.5 kW/kg U
Average enrichment (subsequent cores)	3.3 % by weight U-235
Burn-up of equilibrium core	34,000 MWd/t

Reactor Coolant Pumps (4)

Mass flow per pump	4700 kg/s
Discharge head	8.63 bar
Rating of each pump	7350 kW

Steam Generators (4)

Steam generation per unit	515 kg/s
Pressure/temp. of steam at outlet	68.6 bar/284.5 °C
Max. steam wetness	0.25 % by weight

Turbo-generator Set (four-cylinder single-shaft saturated steam condensing turbine)

Output/speed	1362 MW/25 s^{-1}
Main steam mass flow	2061 kg/s
Main steam temperature	284.5 °C
Reheating to	240 °C
No. of extraction points	6
Condenser pressure	38 mbar
--Volumetric circulating water flow	58.3 m^3/s
-- Circulating water temp.	10 °C

Generator (three-phase synchronous generator)

Apparent power	1560 MVA
Power factor	0.87
Terminal voltage	27 kV ± 5 %
Stator/rotor cooling medium/Laminated core	$H_2O/H_2O/H_2$
Exciter type	rotating diode rectifiers

Unit Transformer (2 three-phase transformer)

Unit 1 rating	750 MVA
Transformation ratio	27/420 kV ± 11 %
Unit 2 rating	1020 MVA
Transformation ratio	27/415 kV ± 12 %

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Appendix B: Input Deck

RELAP5MOD2

RESA KKP2

"Appendix B contains Proprietary Material - contents removed at author's request. Questions concerning Appendix B model should be addressed to author".

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The commissioning test "Reactor Trip at Full Load", which was performed at the nuclear power plant Philippsburg 2 (KKP-2), was recalculated with RELAP5/MOD2. The comparison of the results with the commissioning test results shows very good agreement between measurement and calculation. Difficulties arised attempting to adjust the RPV inlet temperature, which depends on steam generator pressure, to the initial test condition. It is assumed that the heat transferred correlations in RELAP5/ MOD2 are not optimized for this problem. The deviation of SG water level during the transient between calculation and measurement is assumed to be caused by the separator model in RELAP5/MOD2.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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