ABWR ATWS STABILITY STUDY

February 19, 1993

1. Scope

This report provides the results of the transient study for ABWR ATWS events with the 3-D TRACG code. These analyses cover the most limiting ATWS events to demonstrate the design of mitigation for ATWS is effective and to show that the SSAR study conducted with the REDYA code is bounding.

2. Introduction

Under certain transient events, the steamflow for feedwater heater is shut off and the core inlet temperature drops substantially. In the event of ATWS, the resulting core inlet enthalpy drop reduces the core void fraction and the power increases. This creates the condition for a temporarily unstable condition and the core power and flow start to oscillate. The fuel surface may experience boiling transition. The ATWS mitigation which provides the alternate means to achieve hot shutdown, eventually reduces the core inlet subcooling and, consequently, suppresses the oscillation and maintains the reactor fuel integrity.

The events analyzed are mainsteam line isolation and turbine trip with bypass operable. Upon receiving ATWS signals, the recirculation pumps either trip or start to runback. This will reduce the core power. Because the steam flow to the feedwater heaters are cut off, the feedwater temperature drops to that condensed from main condenser. However, if the control rods fail to insert in response to scram, ARI, FMCRD run-in and manual signals, the introduction of the highly subcooled water to the core will increase the reactor power. This will push the reactor into the unstable operating region. In ABWR, the automatic feedwater runback is designed to mitigate the effect of the increase of the subcooling in the core flow. To assure the integrity of the fuel, an analysis was conducted to calculate the core response through the transient.

As a comparison between ABWR and LaSalle stability performance, core decay ratio calculations with the frequency domain code FABLE for both plants were conducted at the same operation point in the power/flow map.

3. Reactor Modeling and Code Application

The 3-D TRACG code was used to analyze the transient event. The reactor core is divided into three (3) rings in the radial direction and a fourth ring was set for the downcomer region. The reactor vessel is divided into 15 levels with all the major components included. The fuel bundles are grouped into 14 channel groups and standard nodalization for stability study is employed, i.e., the non-boiling region of the fuel channel are finely divided into shorter nodes to better simulate the onset of the nuclear boiling. A quarter core simulation for the 3-D kinetics and an EOEC nuclear

condition was assumed for the calculation. This modeling procedure is consistent with that used for BWROG's evaluation of ATWS stability for operating BWRs.

The ATWS events investigated were main steamline isolation valve (MSIV) closure and turbine trip with bypass valve open. For the MSIV closure case, three cases with different mitigation actions were studied: alternate rod insertion (ARI), fine motion control rod drive (FMCRD) run-in, and standby liquid control system (SLCS).These cases were intended to confirm that the results from REDYA calculation are bounding and also to observe the stability of ABWR during ATWS events. Only SLCS activation was studied for the turbine trip to compare the ABWR stability performance.

The main steamline isolation valves closed in 3.0 seconds and the turbine stop valves closed in 0.1 seconds. Upon high pressure ATWS signal, all four (4) reactor internal pumps (RIP) not connected to M/G sets were tripped and the remaining six (6) pumps started to runback at 10% per second to the minimum speed. The combined effect of the trip and runback reduces the core flow and increases core void, thereby reducing power generation which limits pressure increase. The safety/relief valves ensure the reactor vessel pressure stays within the design criteria by steam discharge to the suppression pool.. After a two(2) minutes delay, automatic feedwater runback is initiated. The SLCS is initiated on three (3) minutes of delay. These actions are not initiated if the APRM is down scale, which indicates that either ARI or FMCRD run-in is successful and the reactor is shut down properly.

The basedeck was initialized at rated power and flow conditions and the dome pressure at 7.17 MPa (1040 Psia). The initial feedwater temperature was also set at the rated condition. Table 3.1 summarizes the initial conditions for the study.

Table 3.1. **INITIAL OPERATING CONDITIONS**

Parameter	Value
Power (MWt)/(%NBR)	3926/100
Core Flow (Kg/sec)/(%NBR)	14580/100
Steamline Flow (Kg/sec)/(%NBR)	2123/100
Feedwater Flow (Kg/sec)/(%NBR)	2123/100
Feedwater Temperature (°C)	215.6
Dome Pressure (MPa)/(Psia)	7.17/1040
Number of Fuel Bundles	872
S/R Valves Steam Capacity at 1st Setpoint (%NBR)	91.3
First Relief Valve Opening Setpoint (MPa)/(Psia)	8.00/1160

4. Results and Evaluation

4.1 REDYA Result Comparison

4.1.1 ARI

In the event of MSIV closure, ARI is initiated upon receiving the high pressure ATWS signal. It is conservatively assumed that there is 15 seconds delay and all rods are inserted in 25 seconds after the initiation signal. The reactor achieves the hot shutdown in less than 40 seconds after the onset of the transient event. Figures 4.1.1.1 to 4.1.1.4 show the results from TRACG calculation. Figures 4.1.1.1.5 to 4.1.7 are REDYA results for the comparison. REDYA predicts a higher peak pressure and neutron flux and a wider water level swing.

Table 4.1.1 **Comparison of Key Plant Parameters between REDYA and TRACG**

	REDYA	TRACG
Peak Vessel Pressure (MPa)	8.95	8.[illegible]
Peak Neutron Flux (%NBR)	451	248
Minimum Water Level (m)	11.54	12.15

4.1.2 FMCRD Run-in

For the FMCRD run-in case which assumes a failure of ARI, the control rod drives need 135 seconds to be fully inserted electrically. The results from TRACG are presented as Figures 4.1.2.1 to 4.1.2.4 while Figures 4.1.2.5 to 4.1.2.7 represent the REDYA results. Again, REDYA predicts a higher dome pressure and larger water level swing. However, the trend for both codes have the similar direction.

Table 4.1.2 **Comparison of Key Plant Parameters between REDYA and TRACG**

	REDYA	TRACG
Peak Vessel Pressure (MPa)	8.95	8.68
Peak Neutron Flux (%NBR)	451	248
Minimum Water Level (m)	12.06	12.98

4.1.3 Boron Injection

REDYA again predicts higher pressure, larger water level swing and more cycling of S/RV. It under-predicts the buildup of boron concentration in the reactor core.

Results for TRACG are shown in Figures 4.1.3.1 to 4.1.3.6 and REDYA in Figures 4.1.3.7 to 4.1.3.10. It takes about 2,000 seconds to achieve a concentration of 350 ppm based on REDYA predictions while it takes about 600 seconds for TRACG to show that the boron concentration is close to that level. This is caused by the extremely conservative boron mixing model in REDYA which assumes instantaneous dispersion of boron through the vessel. On the other hand, REDYA can not predict the onset of power oscillations caused by the increase of core inlet subcooling. This is mainly the result of using a point kinetics model for the neutronic calculation. Otherwise, REDYA yields similar trends as TRACG and envelops the TRACG calculation results as shown in Table 4.1.3.

Table 4.1.3 **Comparison of Key Plant Parameters between REDYA and TRACG**

	REDYA	TRACG
Peak Vessel Pressure (MPa)	8.95	8.68
Peak Neutron Flux (%NBR)	451	470
Minimum Water Level (m)	9.03	9.09

4.2 ATWS Stability

In the event of MSIV closure, the reactor power was brought down by the rapid core flow reduction resulting from the pump trip and runback. However, the reduction of feedwater temperature by the loss of the steam supply to the heaters starts to increase the core coolant inlet subcooling with a time constant of 60 seconds. The rising of core power from the cooler water slowly pushs the core into the unstable condition as the core flow rate remains relatively constant. At about 90 seconds, the reactor reachs about 75% rated power and 30% rated core flow, and oscillation of the reactor power was observed. The fuel starts to experience boiling transition. However, the peak power only reached 470% of rated and the maximum peak cladding temperature was less than 500°C.

The automatic feedwater runback was initiated at about 124 seconds. This action cut off the cold water supply, the core inlet subcooling is subsequently reduced and the power oscillation starts to recede. Makeup water from ECCS is activated when water level drop below Level 2 and Level 1.5 which keeps the core covered at all times. The boron injection is initiated at about 184 seconds. The boron reaches the core at about another 60 seconds because of the pump startup delay and filling the pipe space. With a 100 GPM flow rate, the core average boron concentration is close to 300 ppm at 475 seconds and steadily builds up. The reactor power should reach hot shutdown condition when the boron concentration reaches 355 ppm at about 600 seconds after the transient started. Continuous monitoring the reactor power and total reactivity confirms this conclusion as shown in Figures 4.2.1 and 4.2.2.

For the turbine trip with bypass open ATWS event, a similar trend was observed. Because of the similarity of long term transient behavior as that of MSIV closure case, boron injection and ECCS initiation were not simulated for the turbine trip case. Especially, their presence does not significantly affect the overall reactor behavior before 200 seconds as far as the most severe stability problem is concerned. Results for this transient event are shown in Figures 4.2.3 to 4.2.7. Again, automatic feedwater runback successfully mitigates the power oscillation induced by the injection of cold water and no criteria for fuel integrity is violated.

4.3 FABLE Results

To confirm the validity of ABWR study, the results are compared with a similar study for LaSalle-2 plant. In the TRACG simulation for LaSalle ATWS events, peak reactor power reached up to 2,000% of rated condition (Reference 1). A FABLE calculation based on 75% power and 30% core flow is conducted for the respective plant. The core decay ratio and hot channel decay ratio are listed below:

	Core Decay Ratio	Hot Channel Decay Ratio
ABWR	1.4	0.9
LaSalle-2	3.8	4.4

Both plants are operating in the unstable region. However, judging from the magnitude of the core decay ratios, LaSalle should experience much worse power oscillation and, indeed, this is observed from the TRACG calculations.

5. Conclusion

Based on the above analysis, the proposed ATWS design for ABWR is able to mitigate the consequence of an ATWS for the most limiting event. Temporary core instability is mitigated with no ATWS design criteria being violated. The comparison of REDYA/TRACG results also indicate that the conclusions drawn from REDYA results are valid.

6. Reference

1. NEDO-32164, "Mitigation of BWR Core Thermal-Hydraulic Instability in ATWS", December 1992.

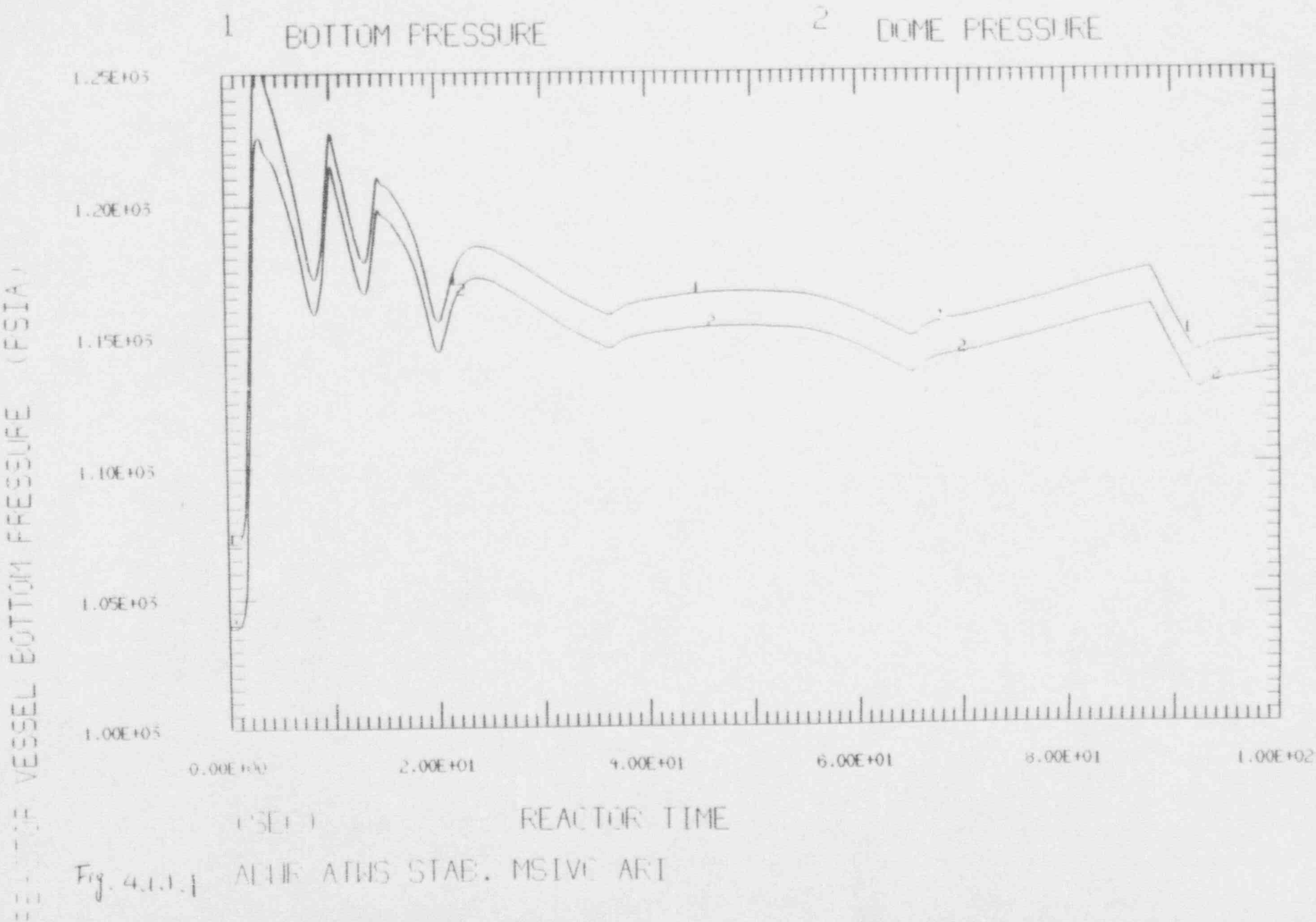

1 BOTTOM PRESSURE
2 DOME PRESSURE
1.25E+03
1.20E+03
1.15E+03
1.10E+03
1.05E+03
1.00E+03
VESSEL BOTTOM PRESSURE (PSIA)
0.00E+00
2.00E+01
4.00E+01
6.00E+01
8.00E+01
1.00E+02
(SEC) REACTOR TIME
ALHR ATWS STAB. MSIVC ARI

Fig. 4.1.1.1

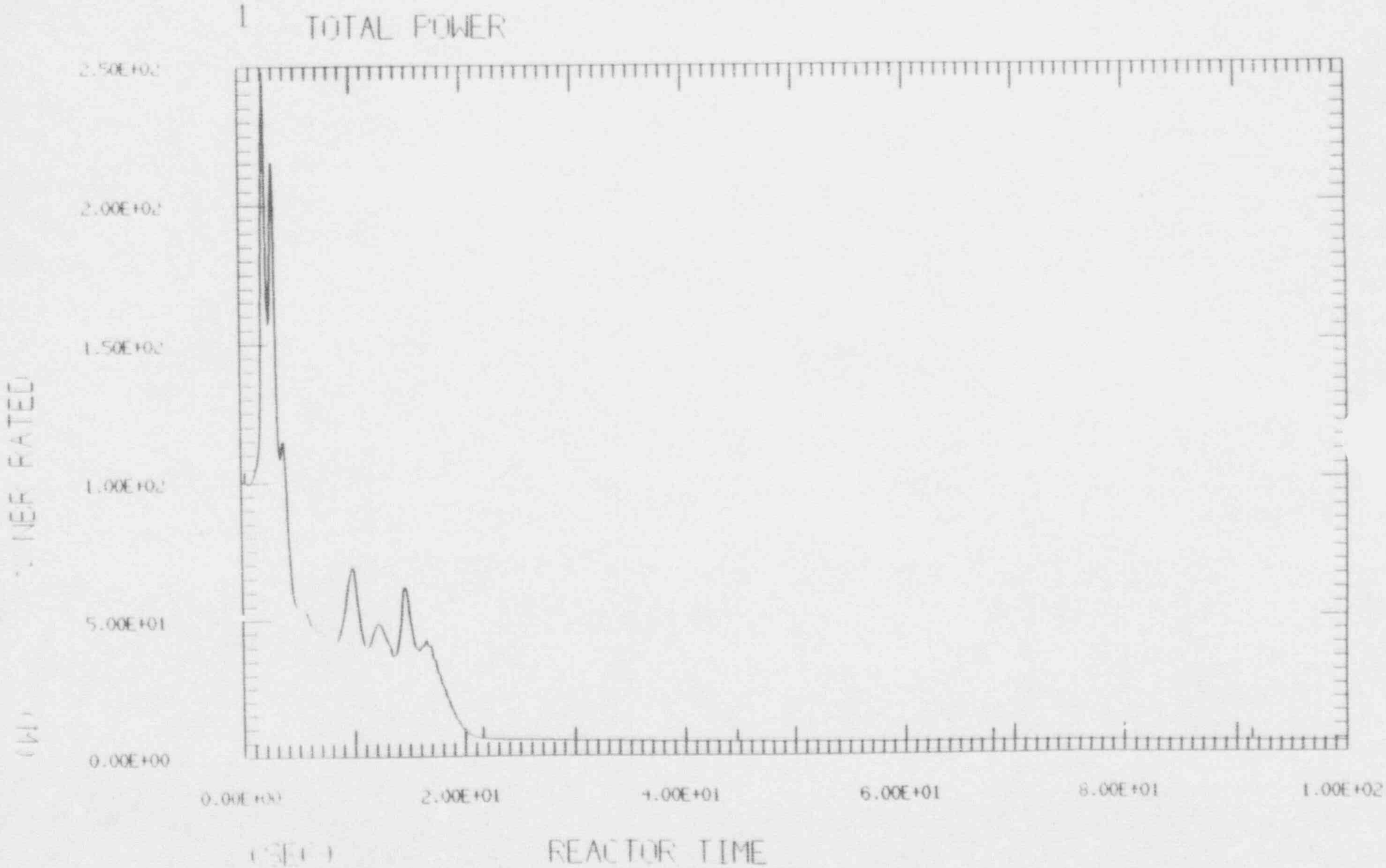


Fig. 4.1.1.2 AENF ATWS STAB. MSIVC ARI

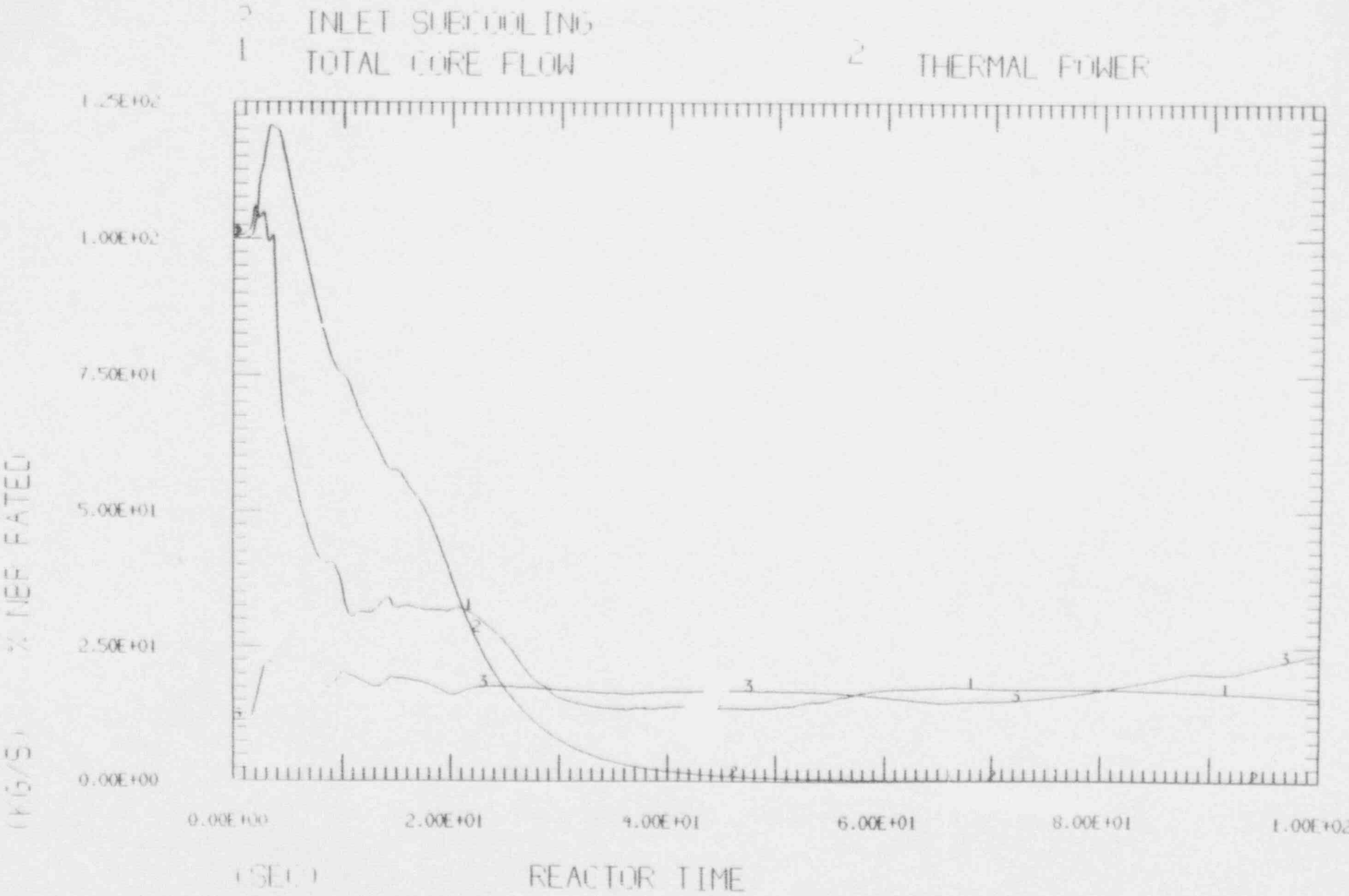


Fig. 4.1.1.3 AEWR ATWS STAB. MSIVC ARI

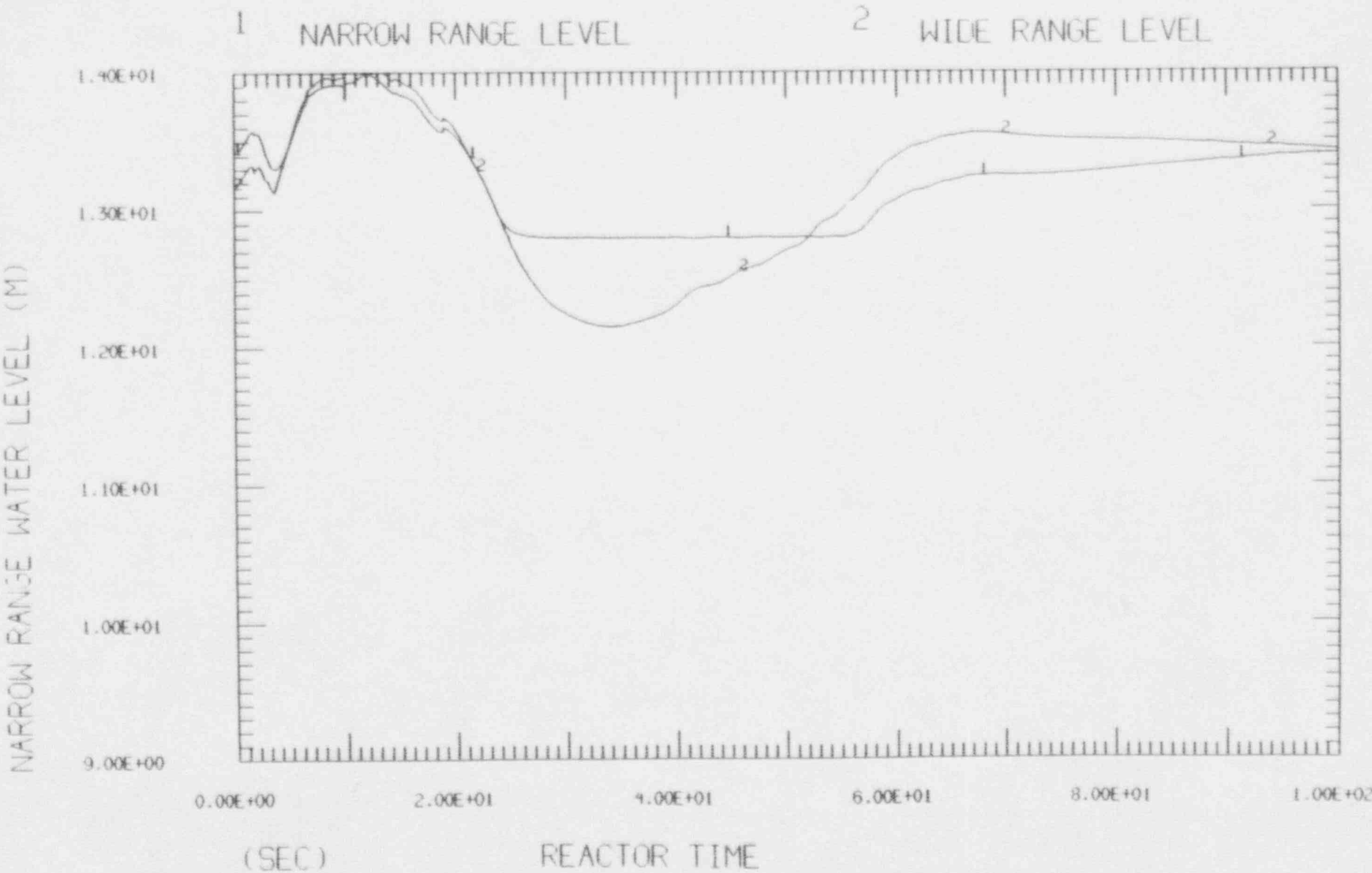


Fig. 4.1.1.4 ABWR ATWS STAB. MSIVC ARI

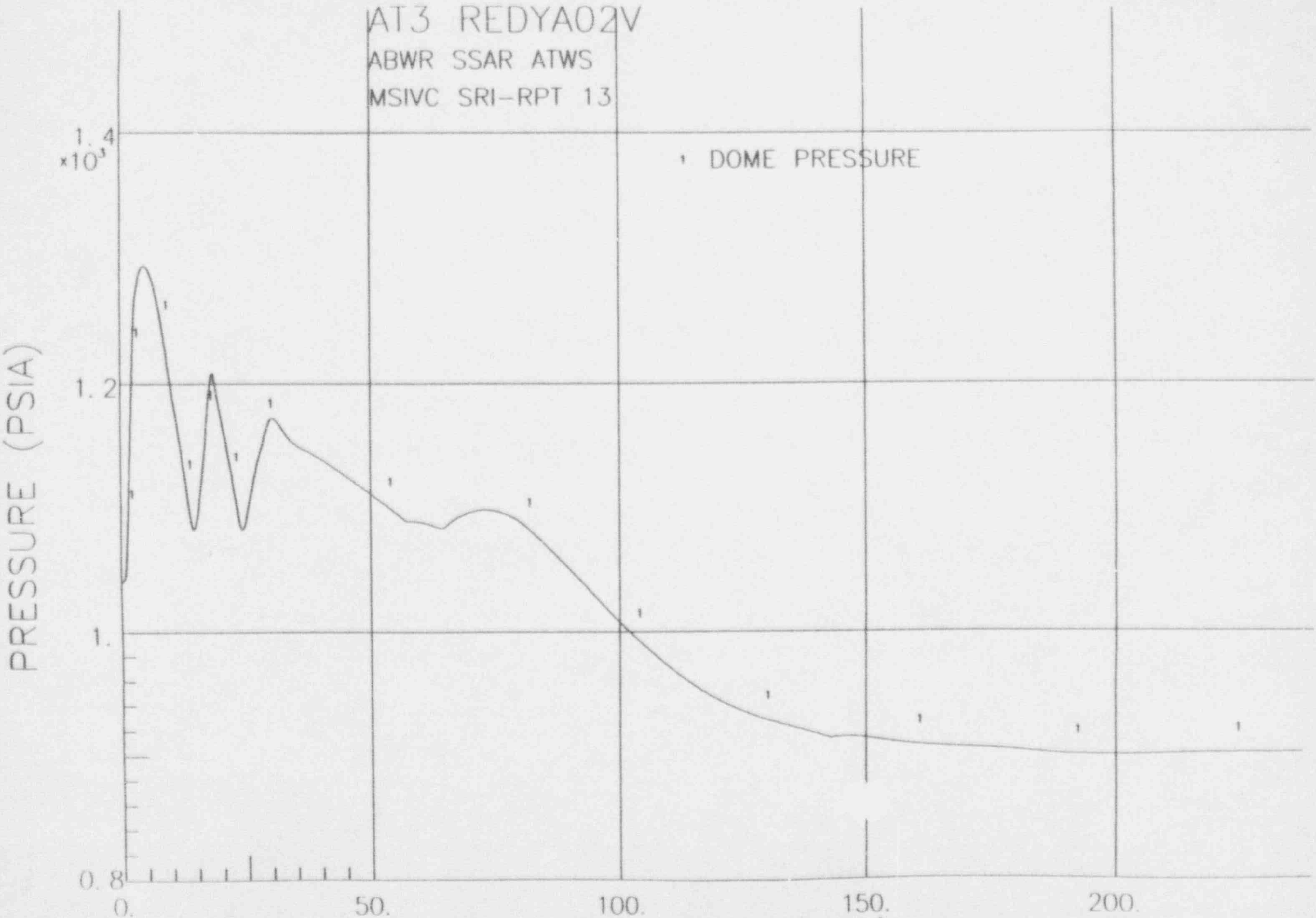
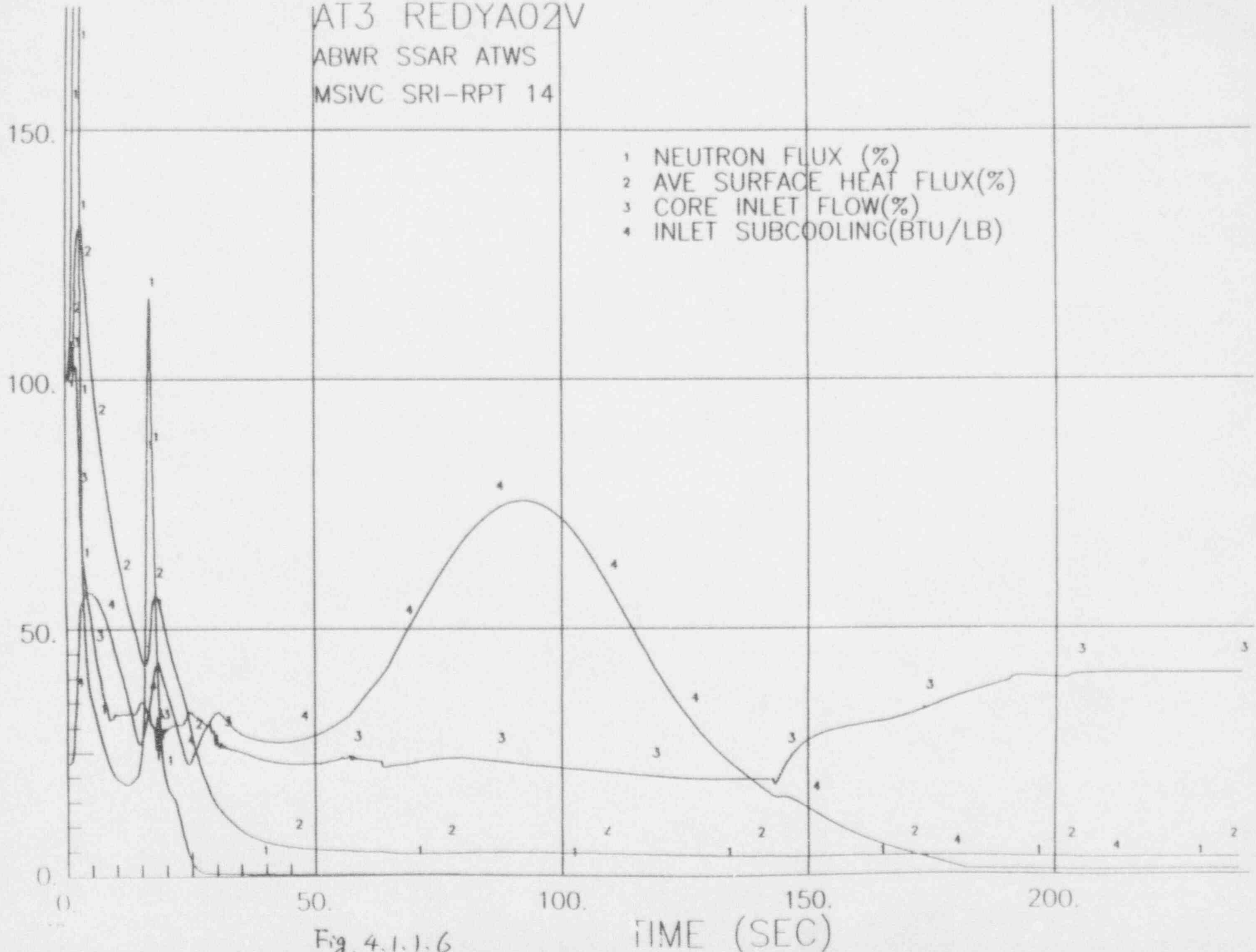
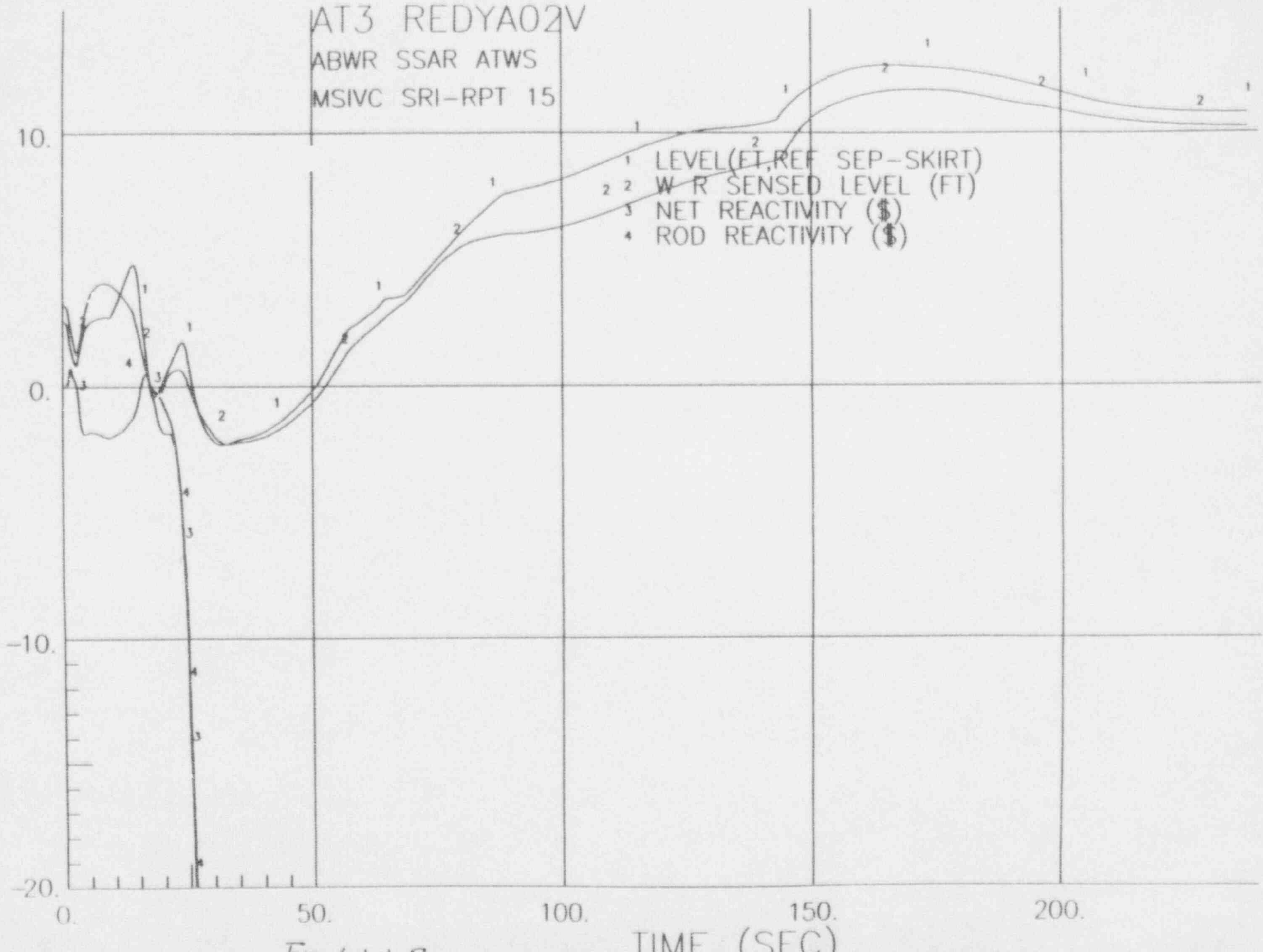

AT3 REDYA02V
ABWR SSAR ATWS
MSIVC SRI-RPT 13
1 DOME PRESSURE
1.4
×10³
1.2
1.
0.8
0.
50.
100.
150.
200.
PRESSURE (PSIA)
TIME (SEC)

Fig 4.1.1 C


AT3 REDYA02V
ABWR SSAR ATWS
MSIVC SRI-RPT 14
1 NEUTRON FLUX (%)
2 AVE SURFACE HEAT FLUX(%)
3 CORE INLET FLOW(%)
4 INLET SUBCOOLING(BTU/LB)
150.
100.
50.
0.
0.
50.
100.
150.
200.
TIME (SEC)
Fig. 4.1.1.6


AT3 REDYA02V
ABWR SSAR ATWS
MSIVC SRI-RPT 15
1 LEVEL(FT,REF SEP-SKIRT)
2 W R SENSED LEVEL (FT)
3 NET REACTIVITY ($)
4 ROD REACTIVITY ($)
10.
0.
-10.
-20.
0.
50.
100.
150.
200.
TIME (SEC)
Fig. 4.1.1.7

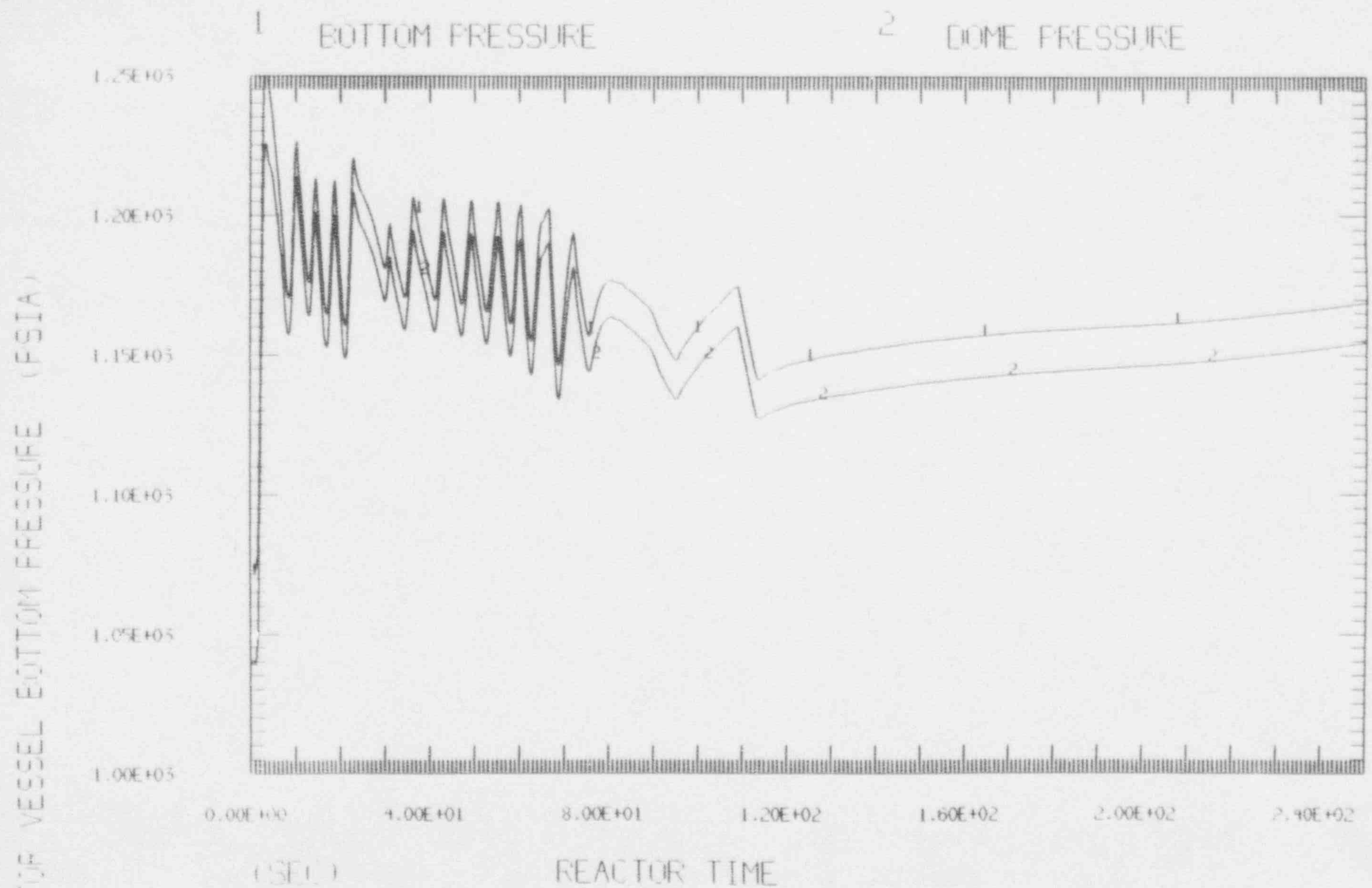


Fig 4.1.2.1 ABWR ATWS STAB. MSIVC FMCRD RUN-IN

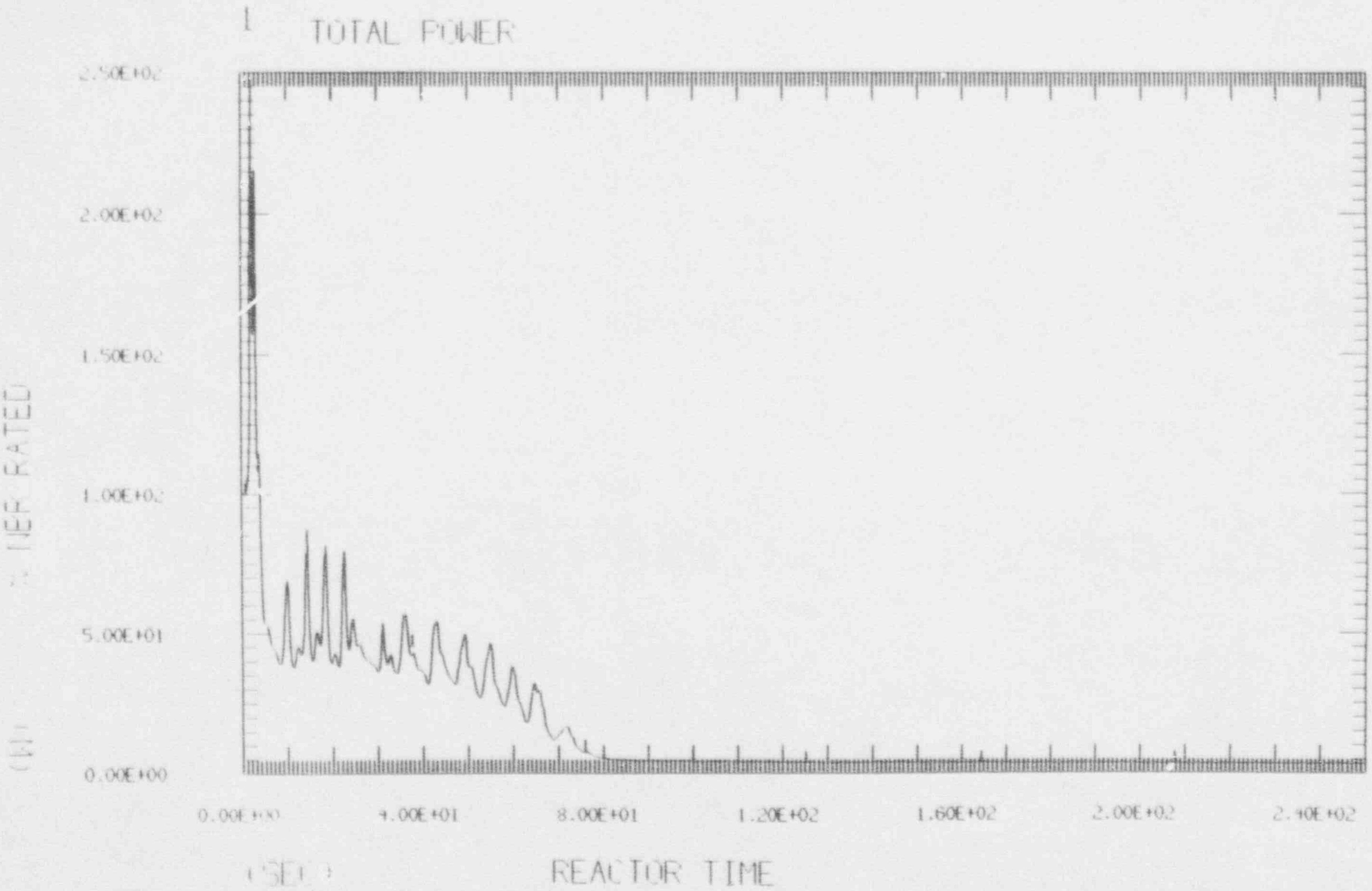


Fig. 4.1.2.2 ABWR ATWS STAB. MSIVC FMCRD RUN-IN

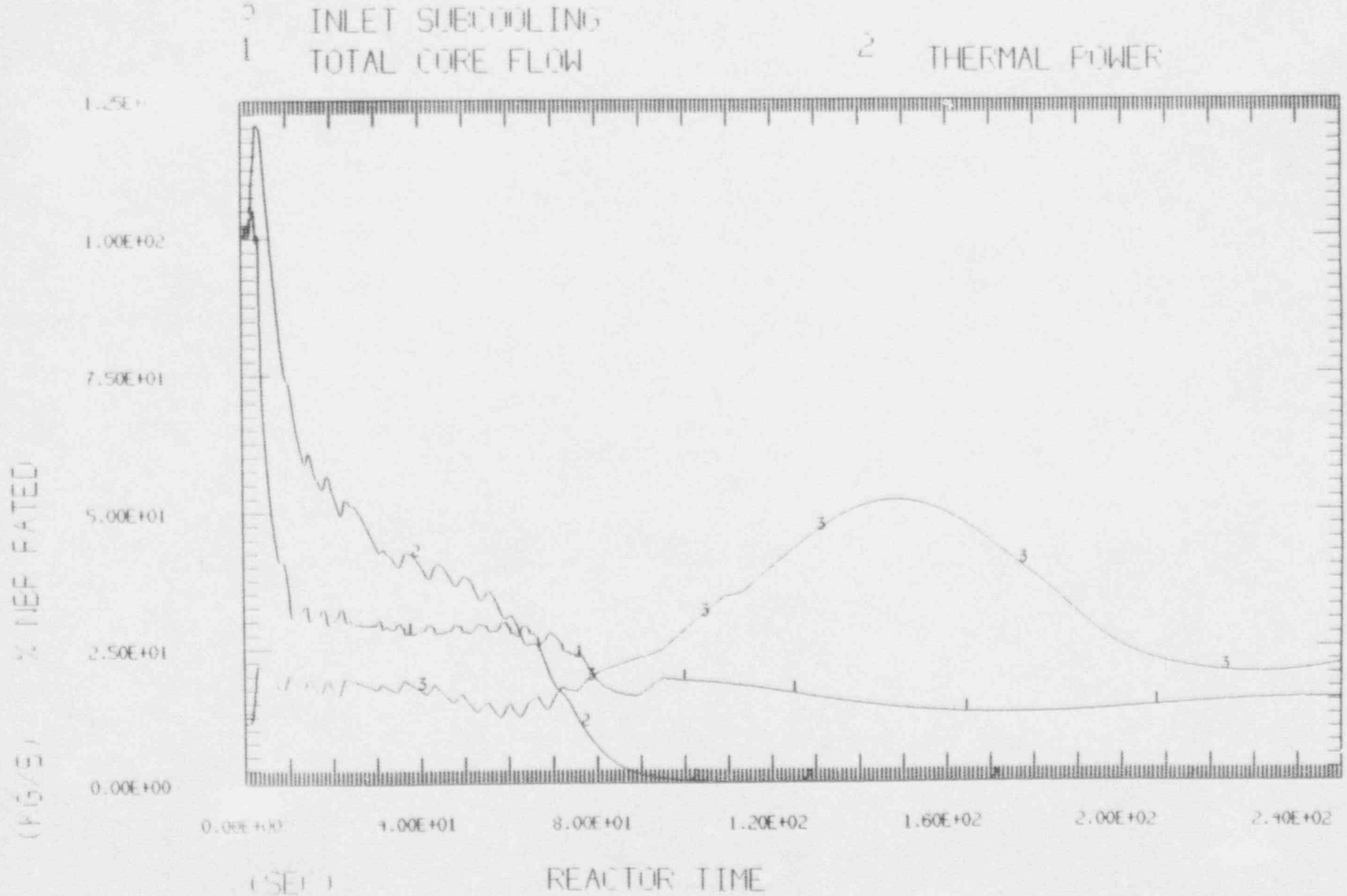


Fig. 4.1.2.3 ABWR ATWS STAB. MSIVC FMCRD RUN-IN

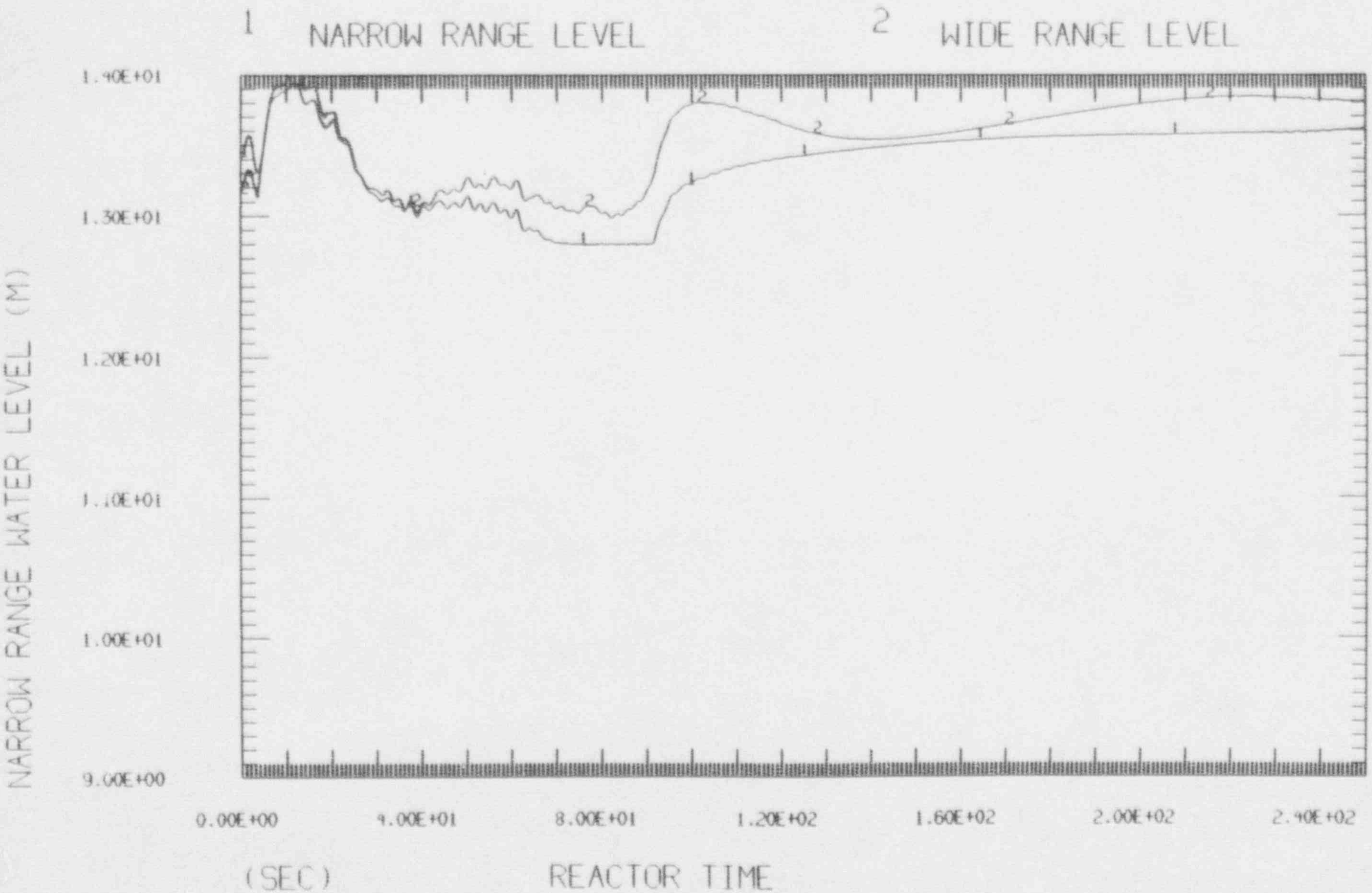


Fig. 4.1.2.4 ABWR ATWS STAB. MSIVC FMCRD RUN-IN

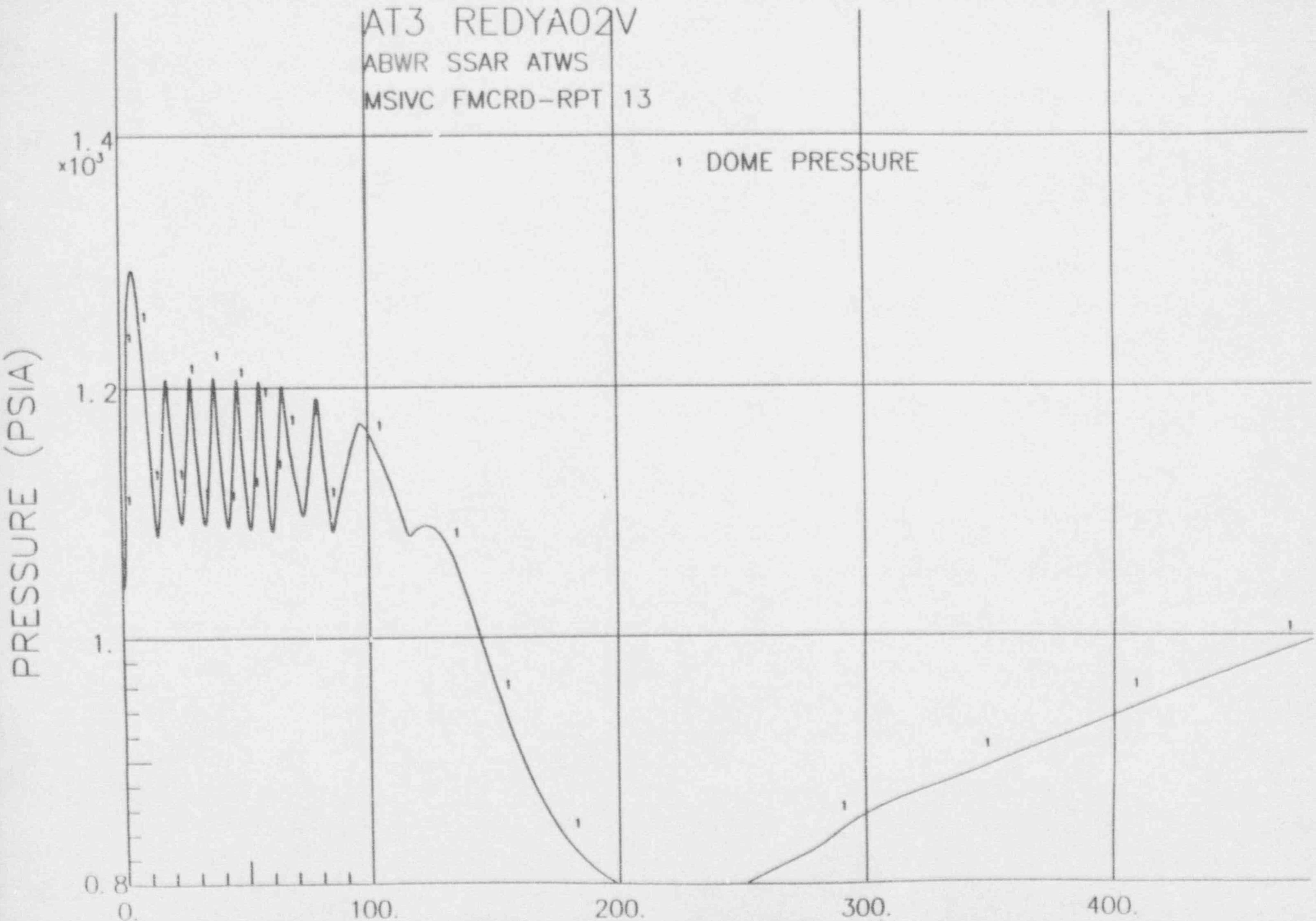
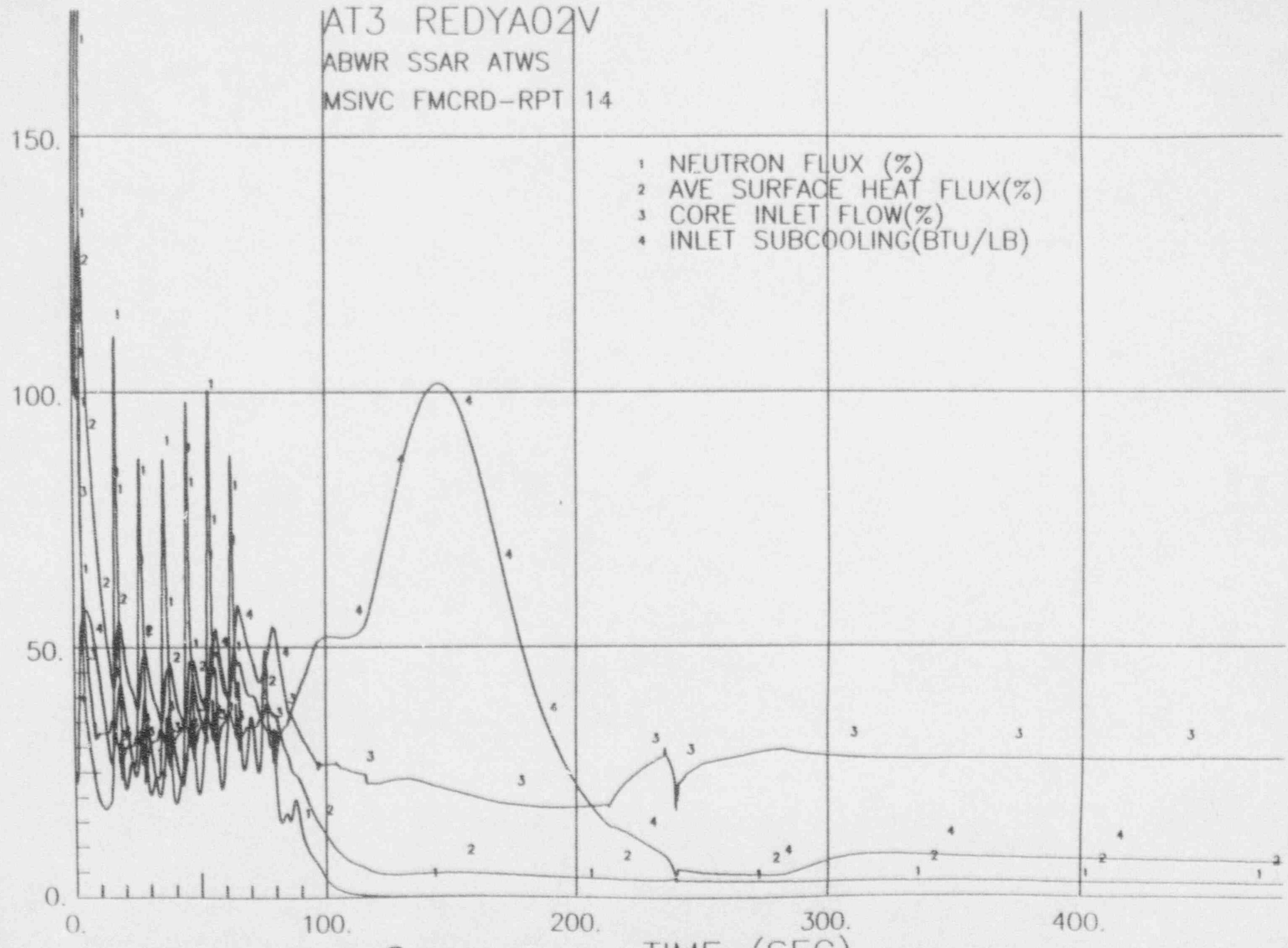
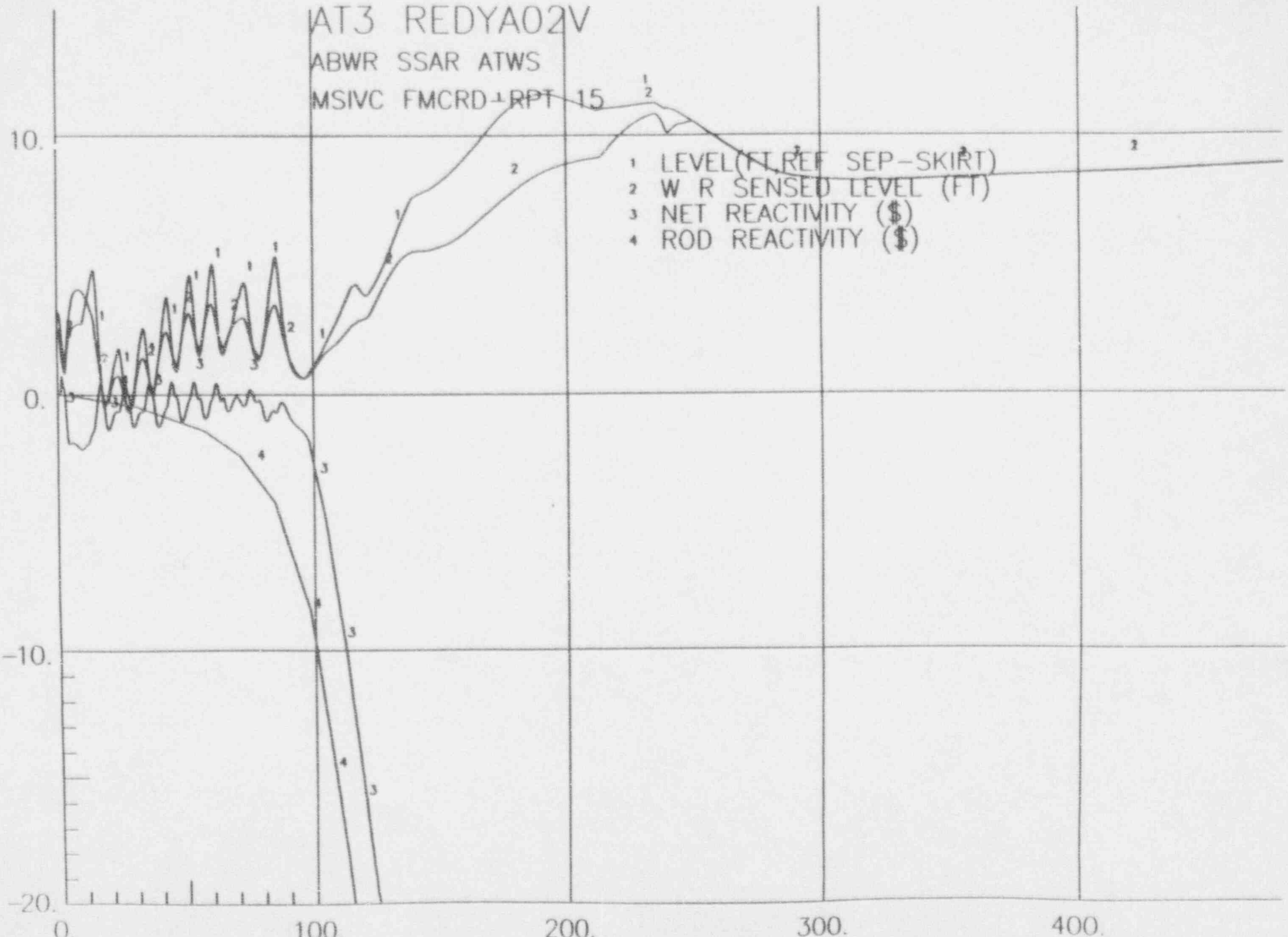

AT3 REDYA02V
ABWR SSAR ATWS
MSIVC FMCRD-RPT 13
DOME PRESSURE
PRESSURE (PSIA)
1.4
×10³
1.2
1.
0.8
0.
100.
200.
300.
400.
TIME (SEC)

Fig. 4.1.2.5



AT3 REDYA02V
ABWR SSAR ATWS
MSIVC FMCRD-RPT 14
1 NEUTRON FLUX (%)
2 AVE SURFACE HEAT FLUX(%)
3 CORE INLET FLOW(%)
4 INLET SUBCOOLING(BTU/LB)
150.
100.
50.
0.
0.
100.
200.
300.
400.
TIME (SEC)

Fig. 4.1.2.6



AT3 REDYA02V
ABWR SSAR ATWS
MSIVC FMCRD+RPT 15
1 LEVEL(FT,REF SEP-SKIRT)
2 W R SENSED LEVEL (FT)
3 NET REACTIVITY ($)
4 ROD REACTIVITY ($)
10.
0.
-10.
-20.
0.
100.
200.
300.
400.
TIME (SEC)

Fig. 4.1.2.7

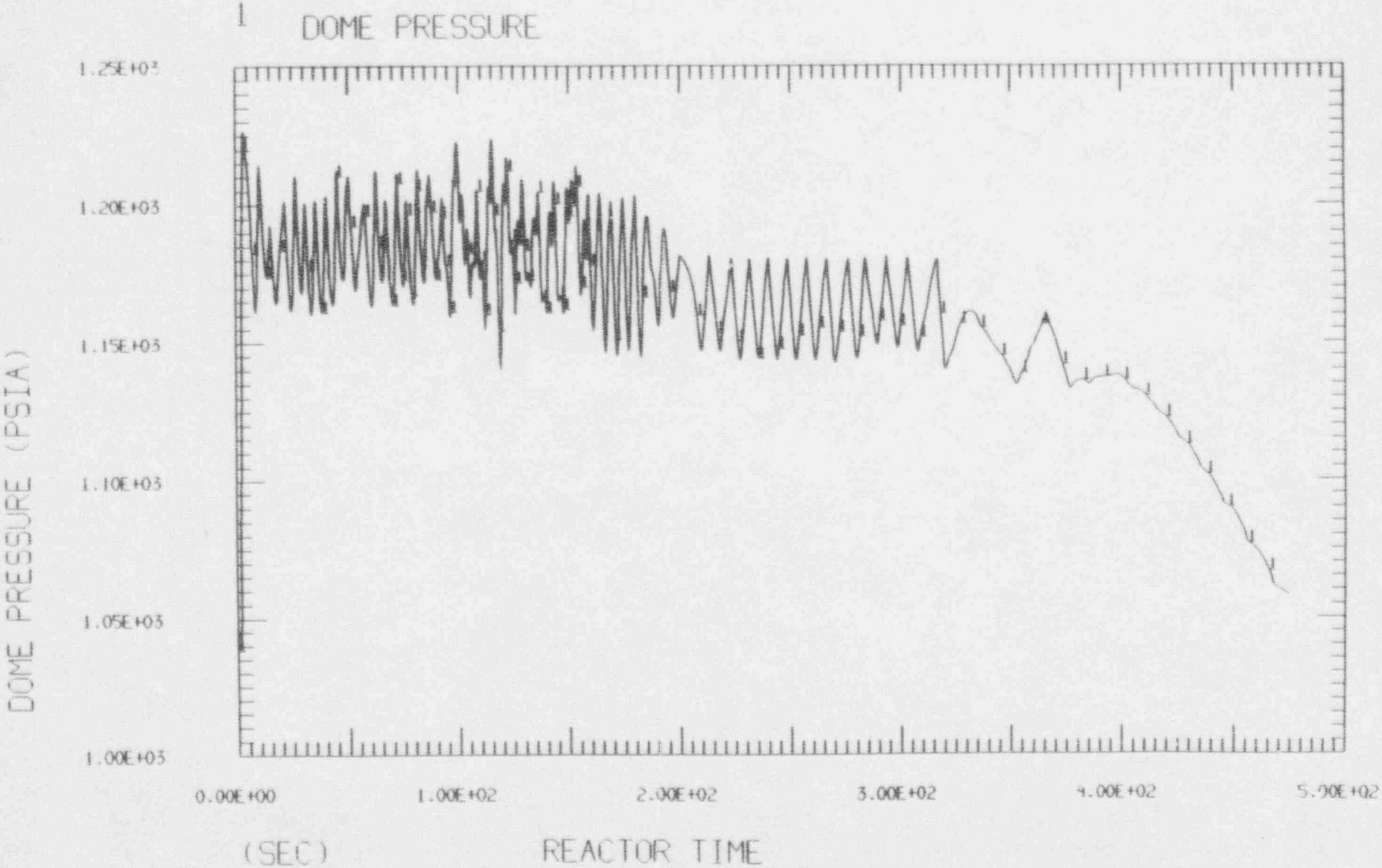


Fig. 4.1.3.1 ABWR ATWS STABILITY, MSIVC SLCS

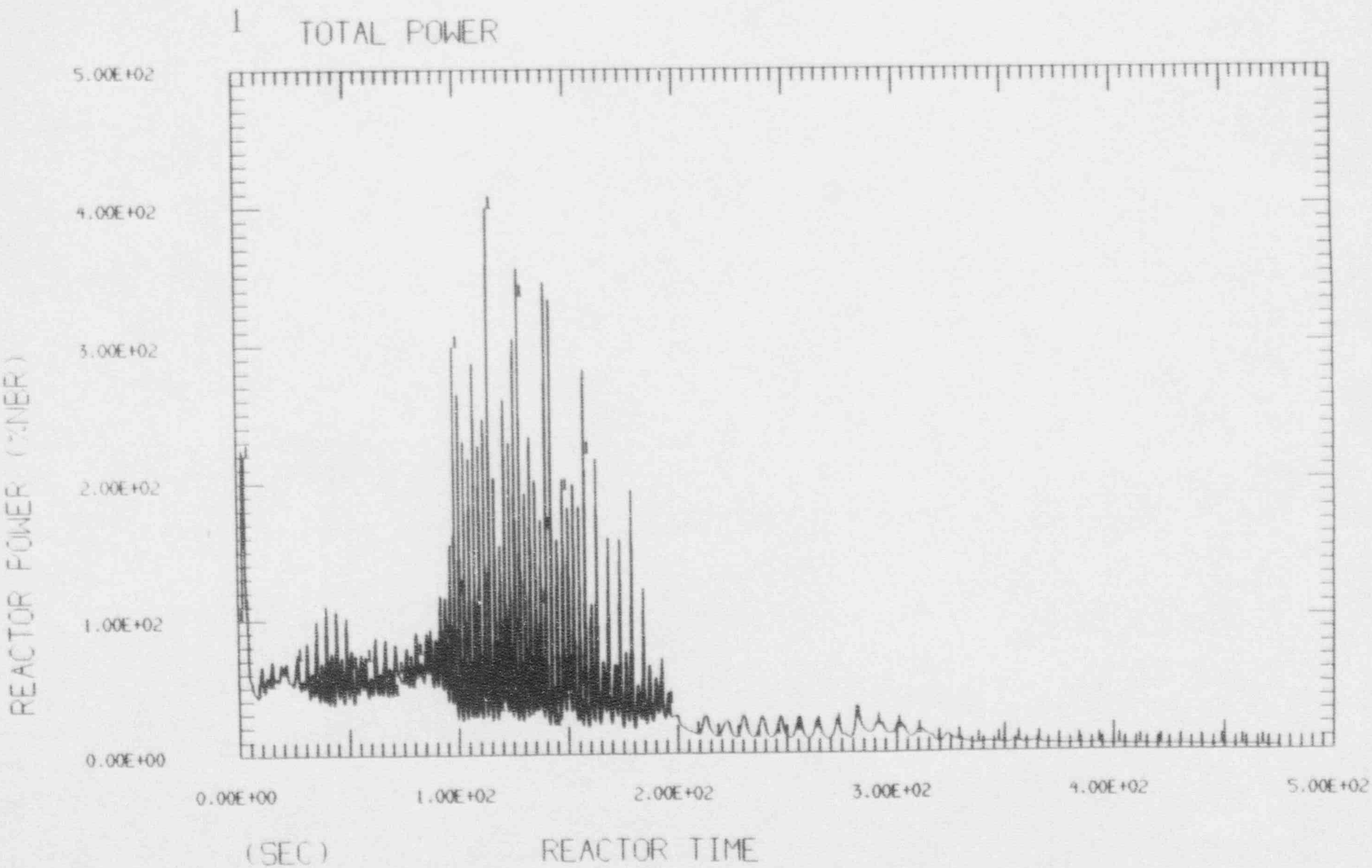


Fig 4.1.3.2 ABWR ATWS STABILITY, MSIVC SLCS

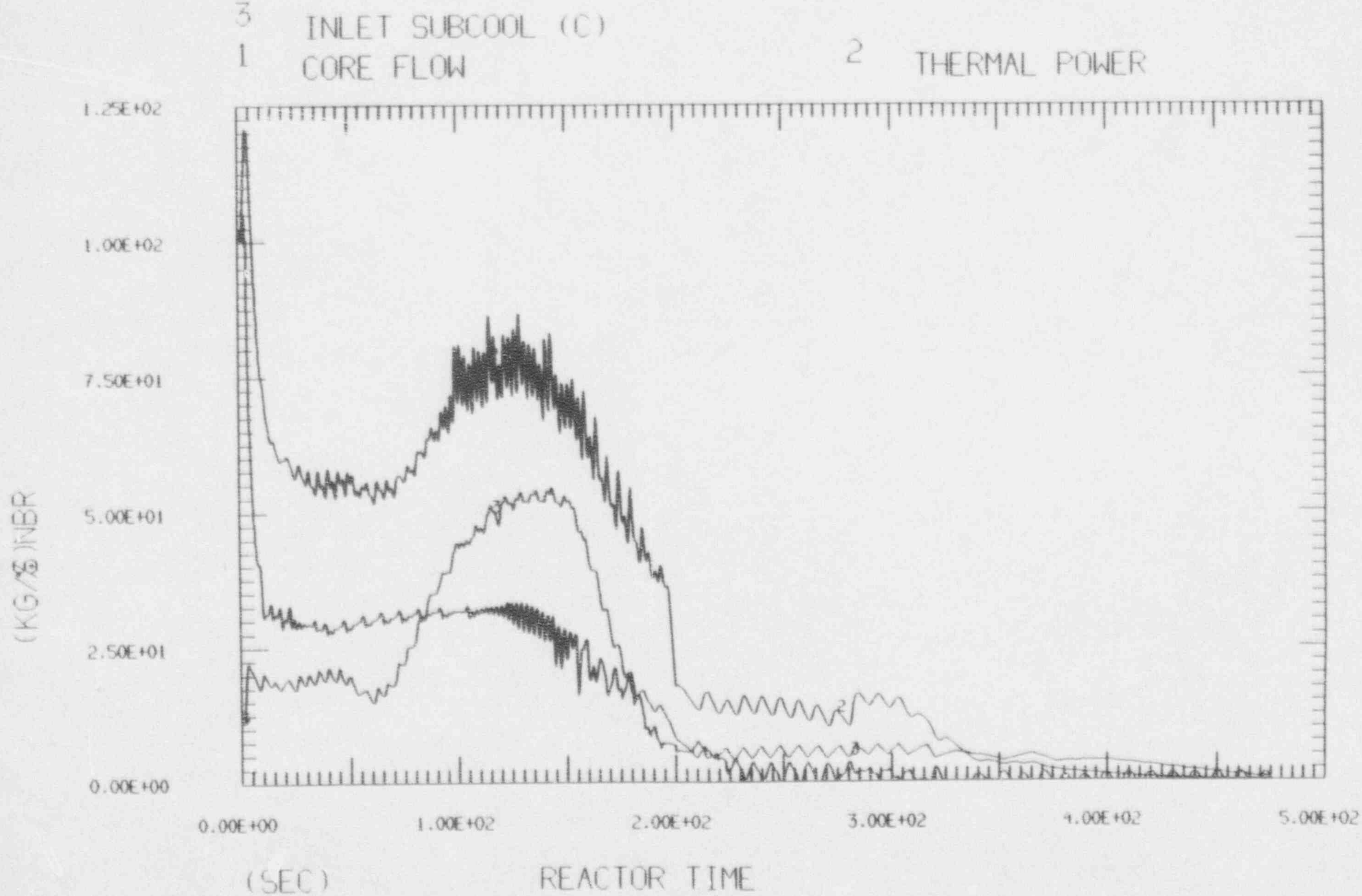


Fig. 4.1.3.3 ABWR ATWS STABILITY. MSIVC SLCS

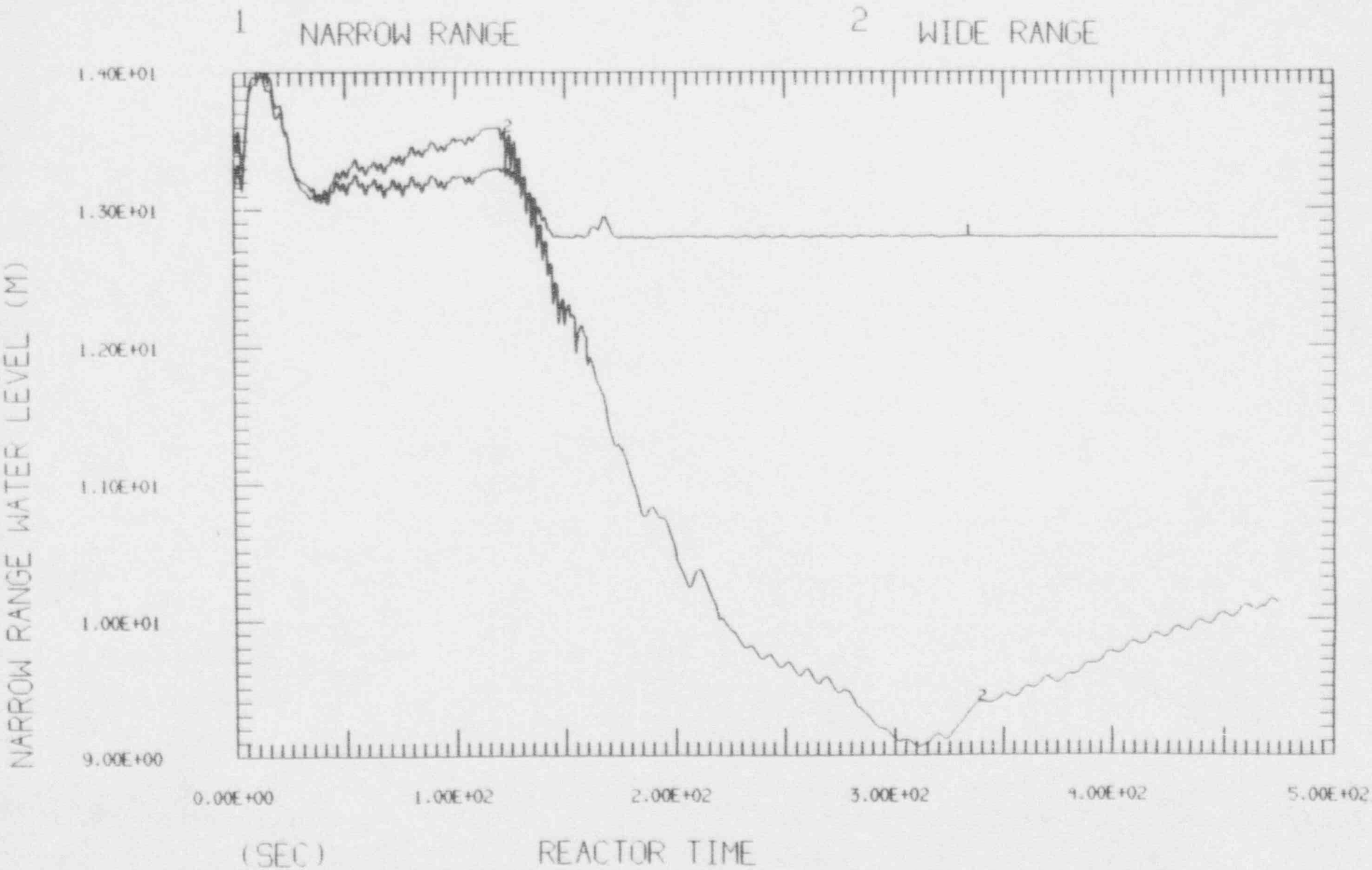


Fig. 4.1.3.4 ABWR ATWS STABILITY. MSIVC SLCS

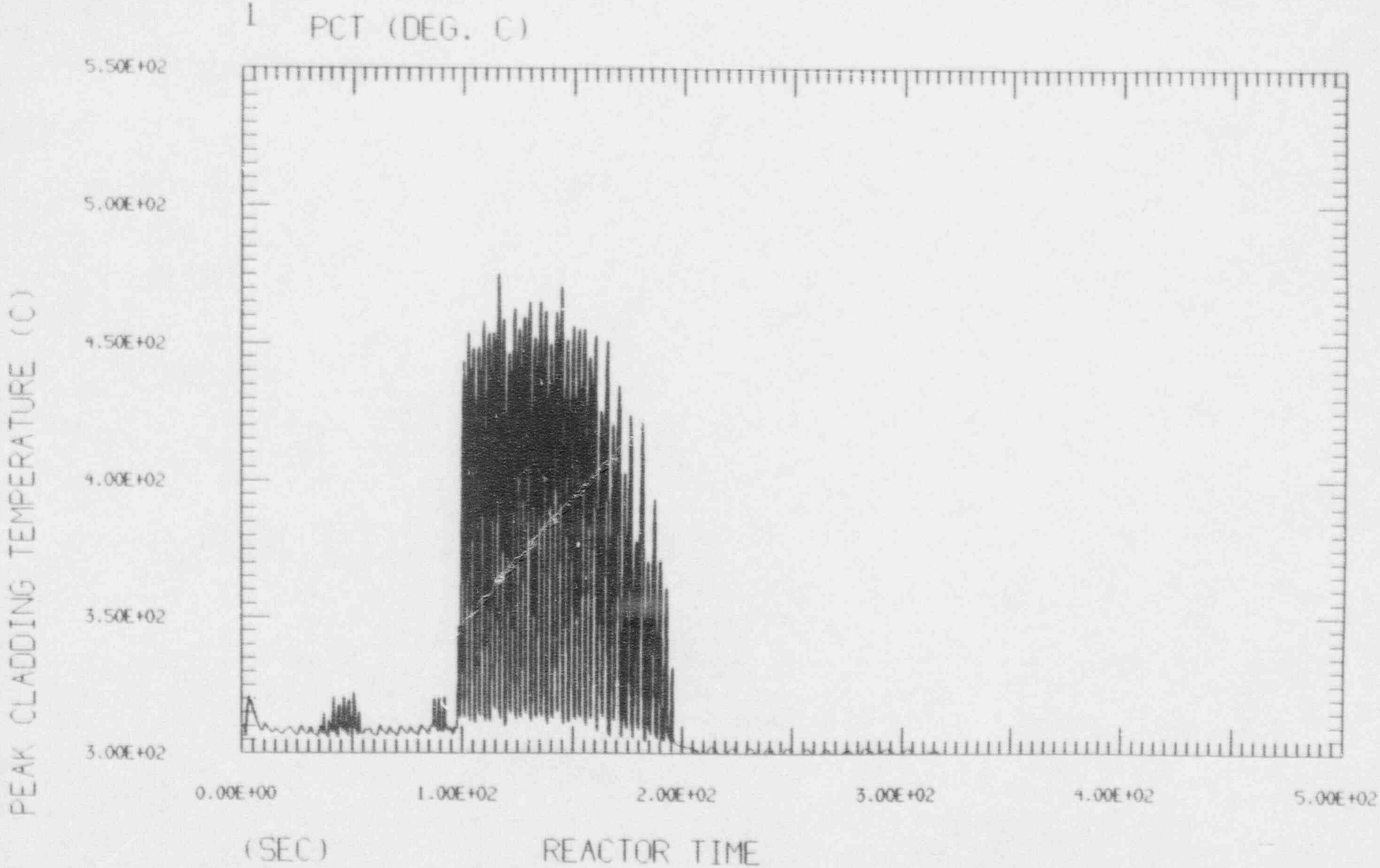


Fig. 4.1.3.5 ABWR ATWS STABILITY. MSIVC SLCS

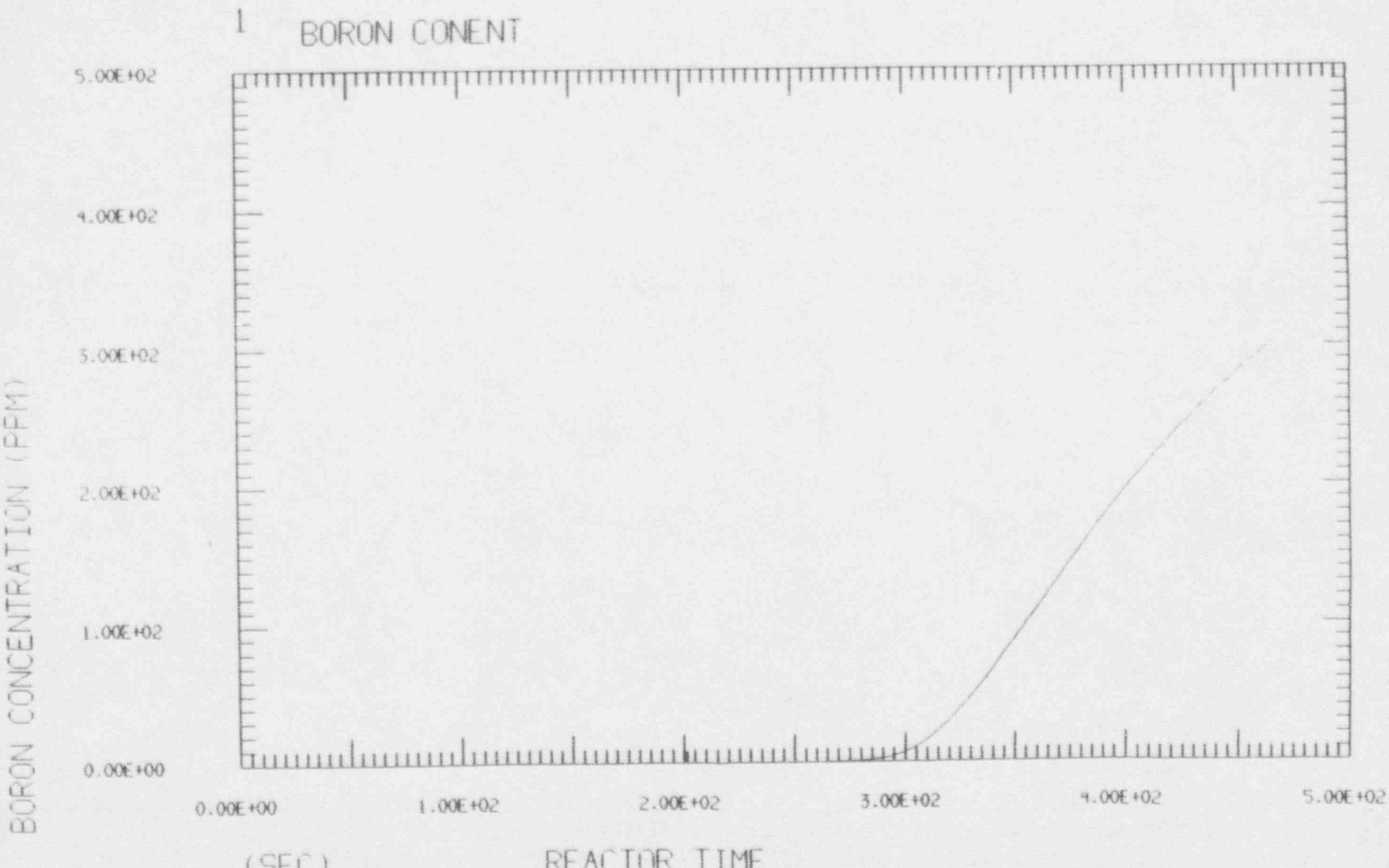
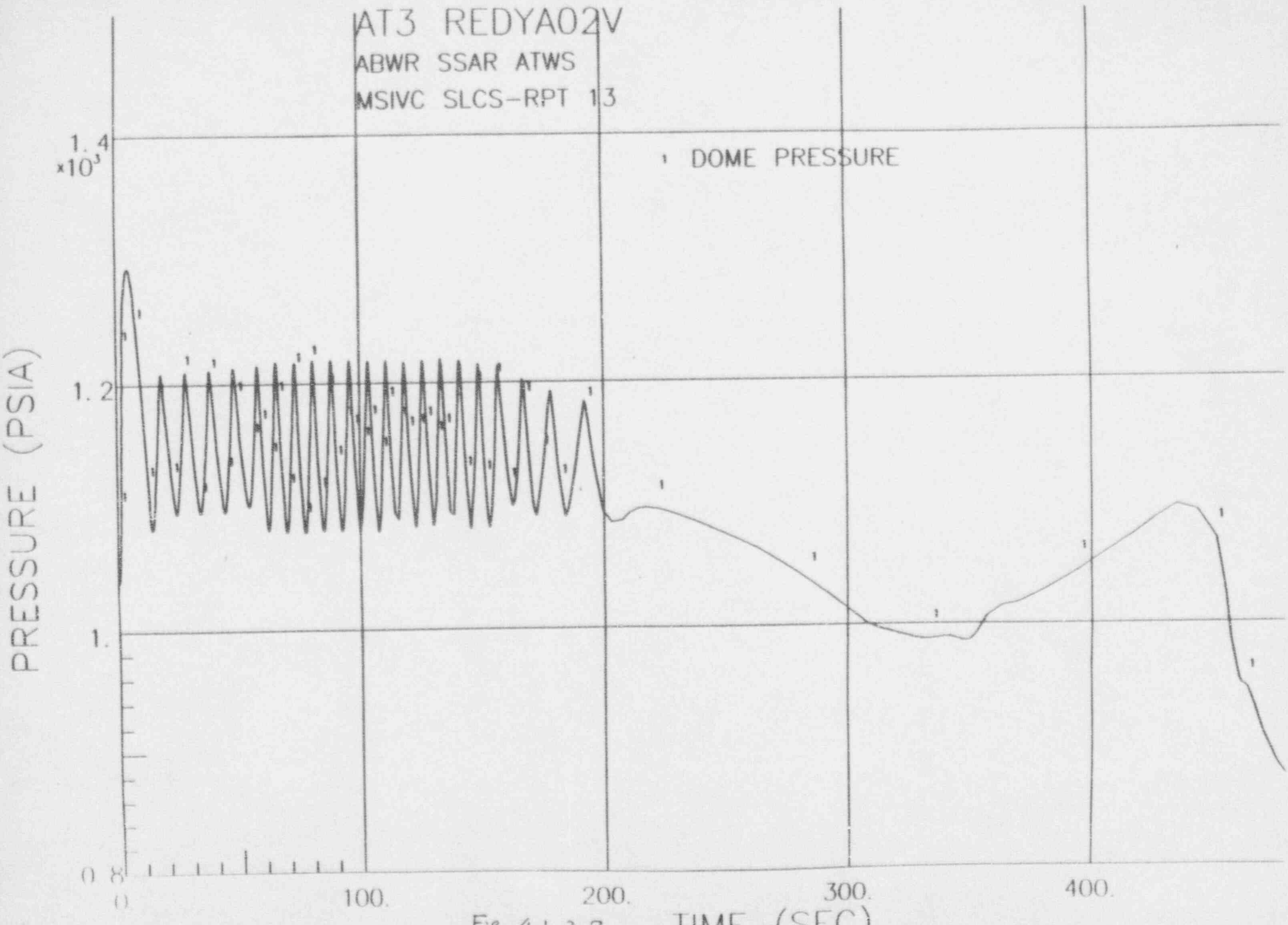
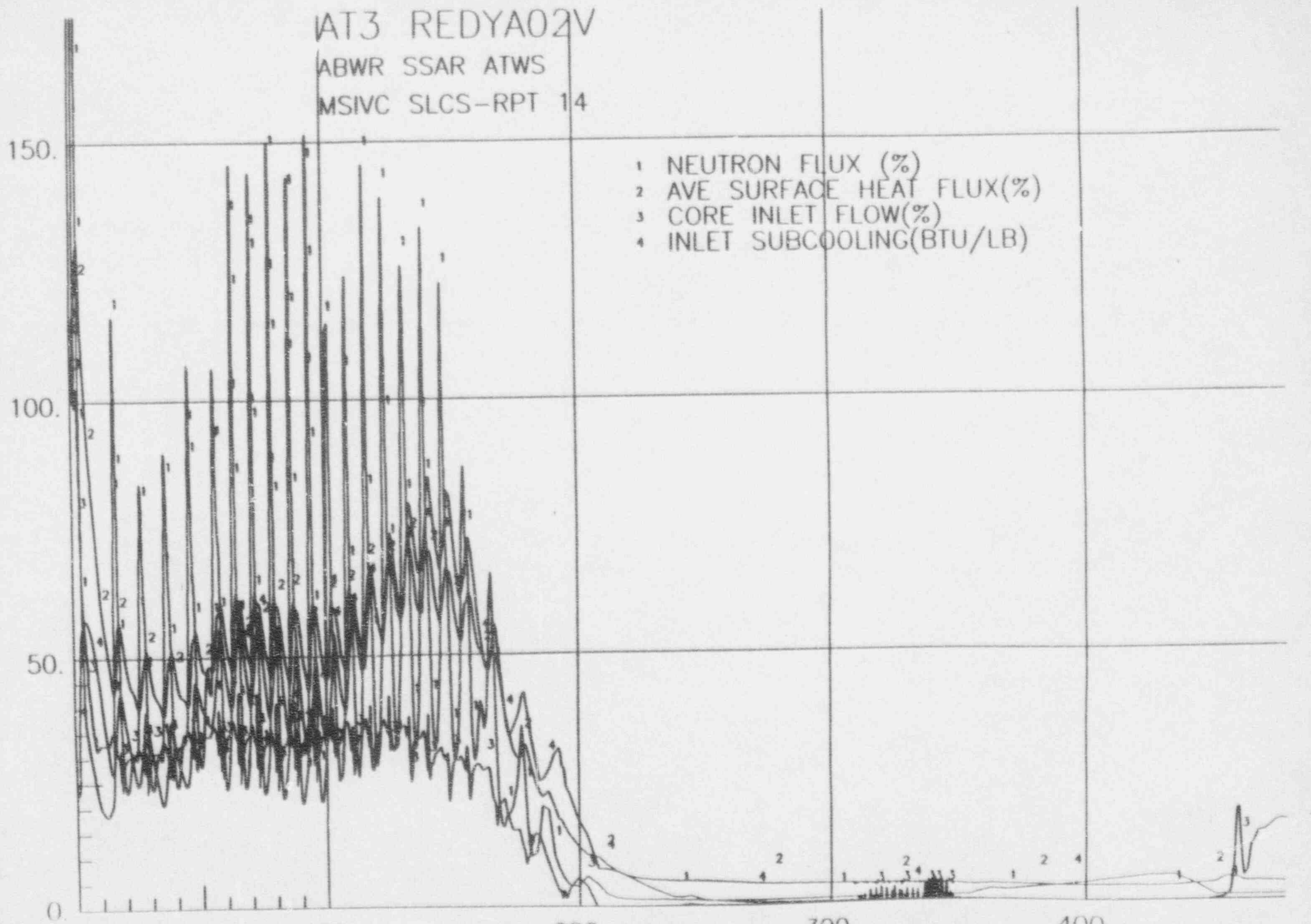


Fig. 4.1.3.6 AEWR ATWS STABILITY. MSIVC SLCS



AT3 REDYA02V
ABWR SSAR ATWS
MSIVC SLCS-RPT 13
1 DOME PRESSURE
PRESSURE (PSIA)
1.4
×10³
1.2
1.
0.8
0.
100.
200.
300.
400.
TIME (SEC)

Fig. 4.1-3.7


AT3 REDYA02V
ABWR SSAR ATWS
MSIVC SLCS-RPT 14
1 NEUTRON FLUX (%)
2 AVE SURFACE HEAT FLUX(%)
3 CORE INLET FLOW(%)
4 INLET SUBCOOLING(BTU/LB)
150.
100.
50.
0.
0.
100.
200.
300.
400.
TIME (SEC)
Fig. 4.1.3.8

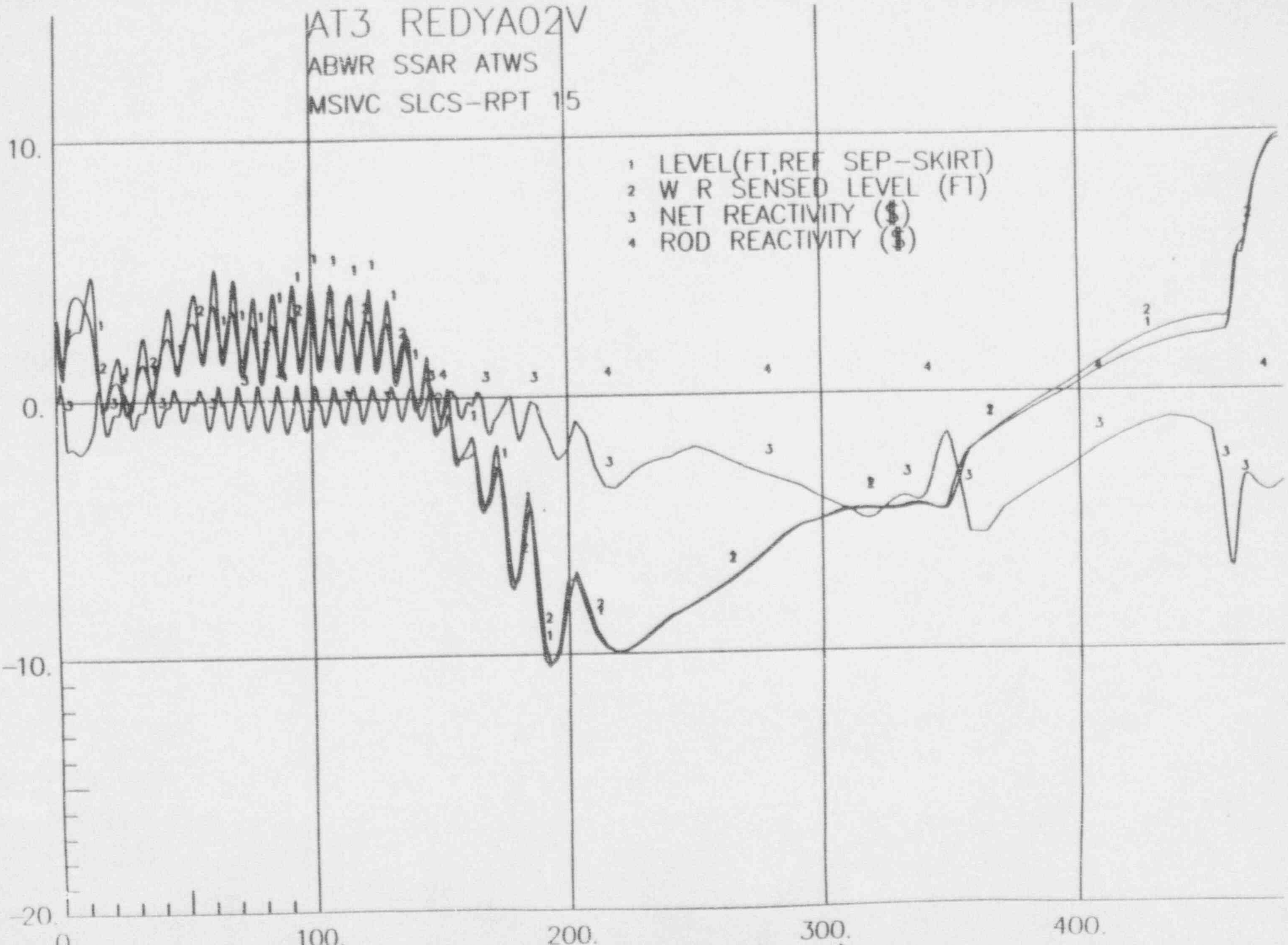
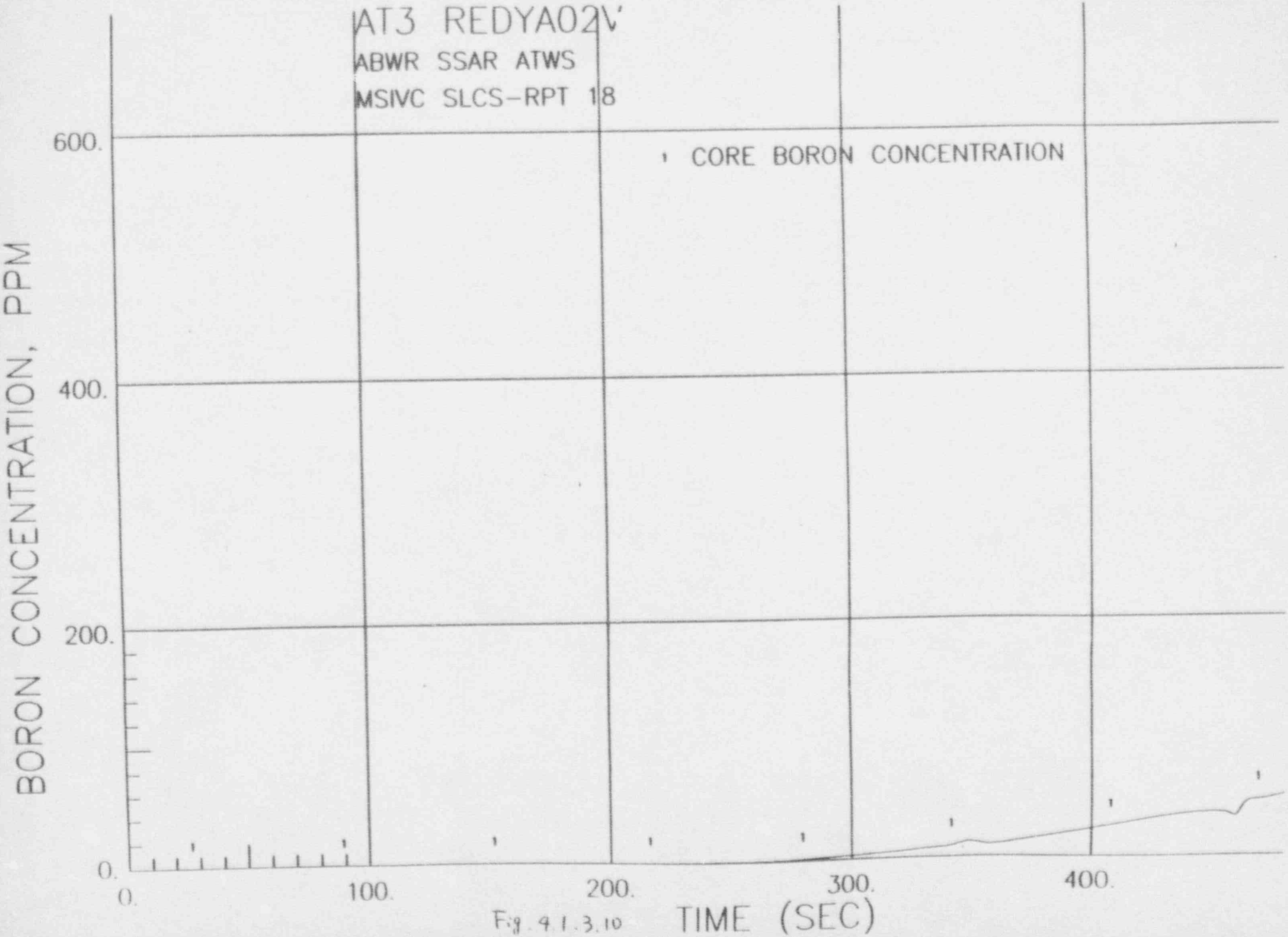

AT3 REDYA02V
ABWR SSAR ATWS
MSIVC SLCS-RPT 15
1 LEVEL(FT,REF SEP-SKIRT)
2 W R SENSED LEVEL (FT)
3 NET REACTIVITY ($)
4 ROD REACTIVITY ($)
10.
0.
-10.
-20.
0.
100.
200.
300.
400.
TIME (SEC)

Fig. 4.1.3.9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ABWR SSAR ATWS
MSIVC SLCS-RPT 18
CORE BORON CONCENTRATION
BORON CONCENTRATION, PPM
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400.
200.
0.
0.
100.
200.
300.
400.
TIME (SEC)

Fig. 4.1.3.10

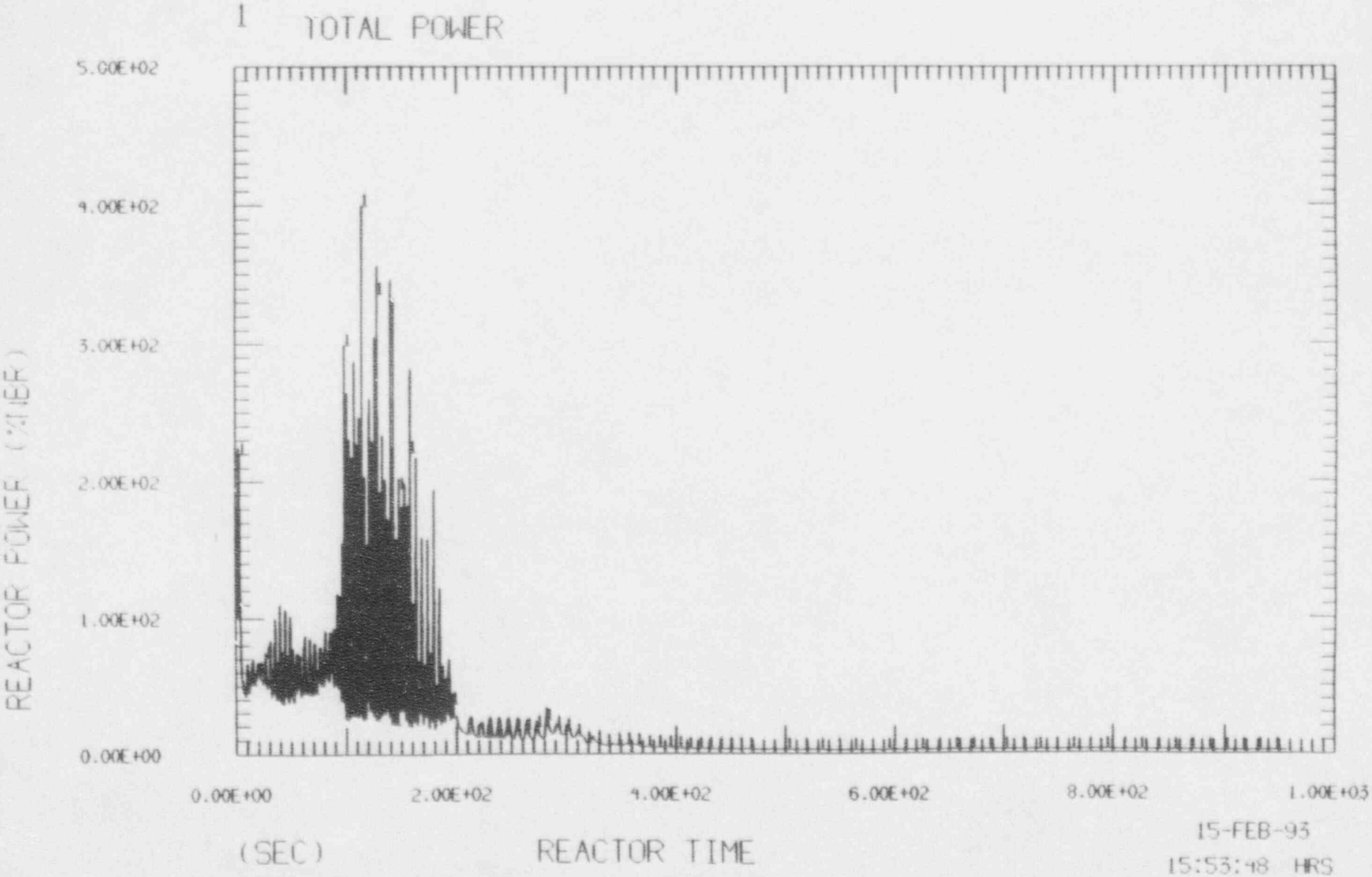

Fig. 4.2.1 ABWR ATWS STABILITY, MSIVC SLCS

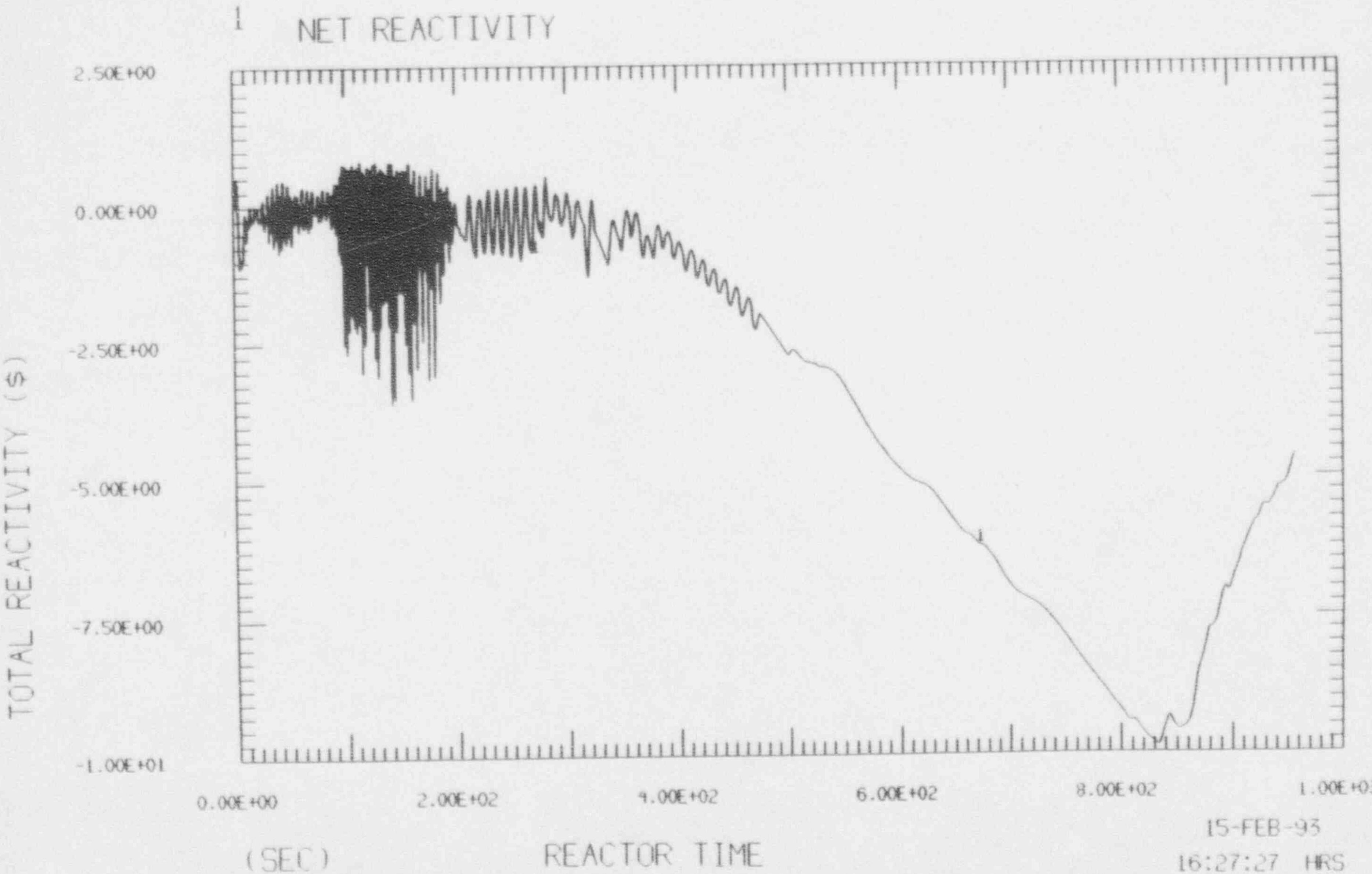

Fig. 4.2.2 ABWR ATWS STABILITY. MSIVC SLCS

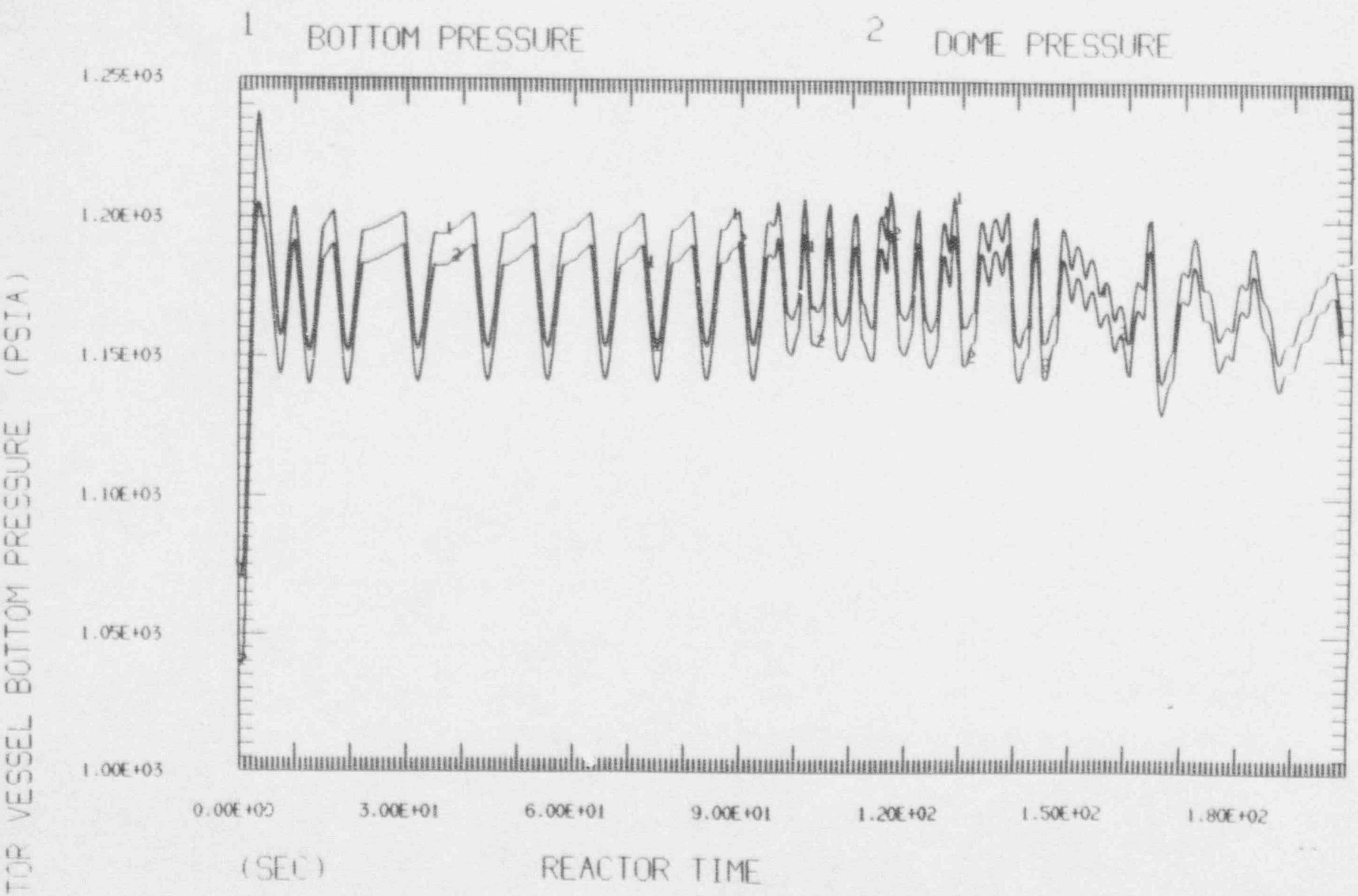

Fig. 4.23 ABWR ATWS STABILITY. TURBINE TRIP W/ BYP

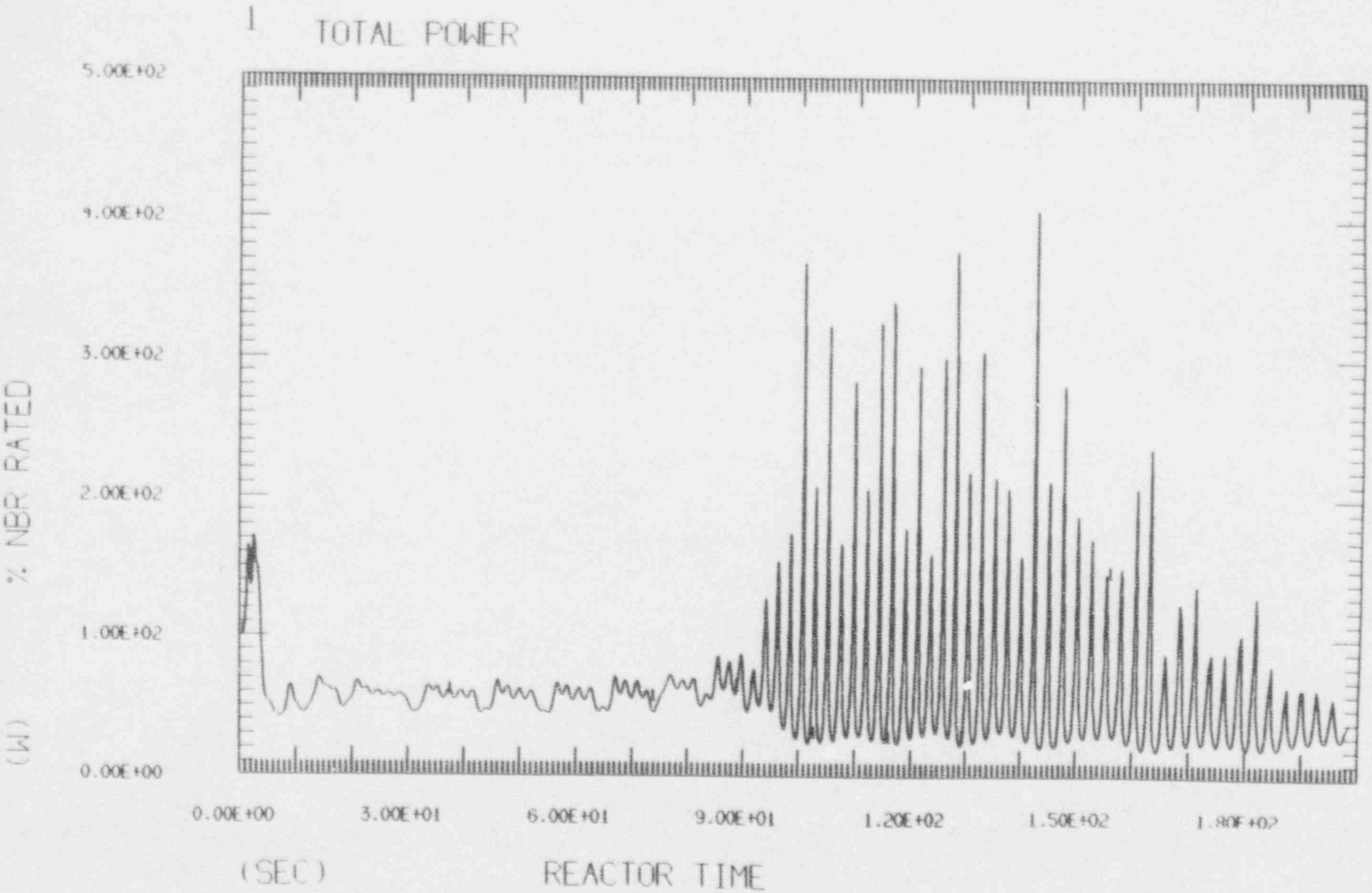

Fig. 4.2.4 AFWR ATWS STABILITY. TURBINE TRIP W/ BYP

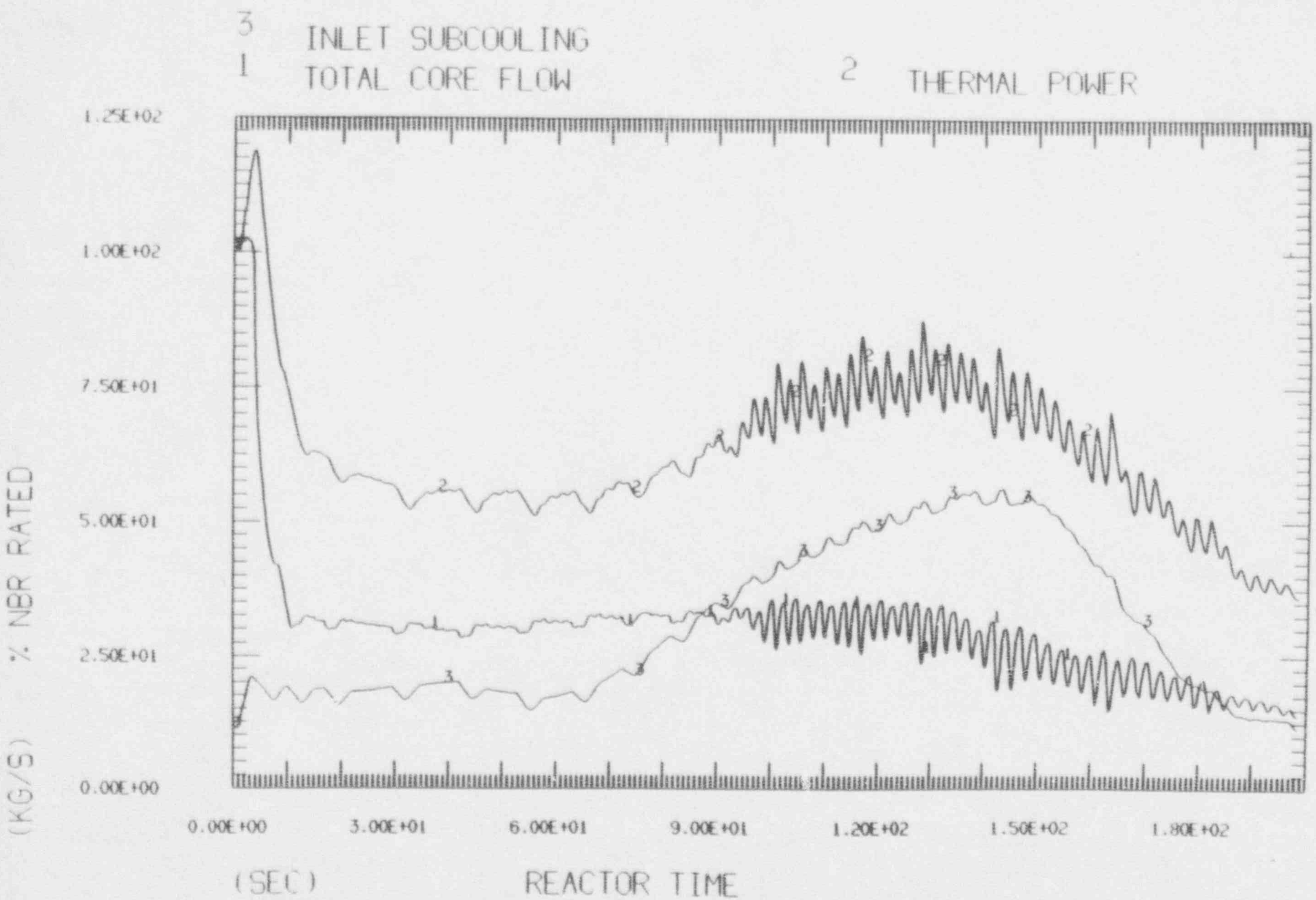

Fig 4.2.5 ABWR ATWS STABILITY, TURBINE TRIP W/ BYP

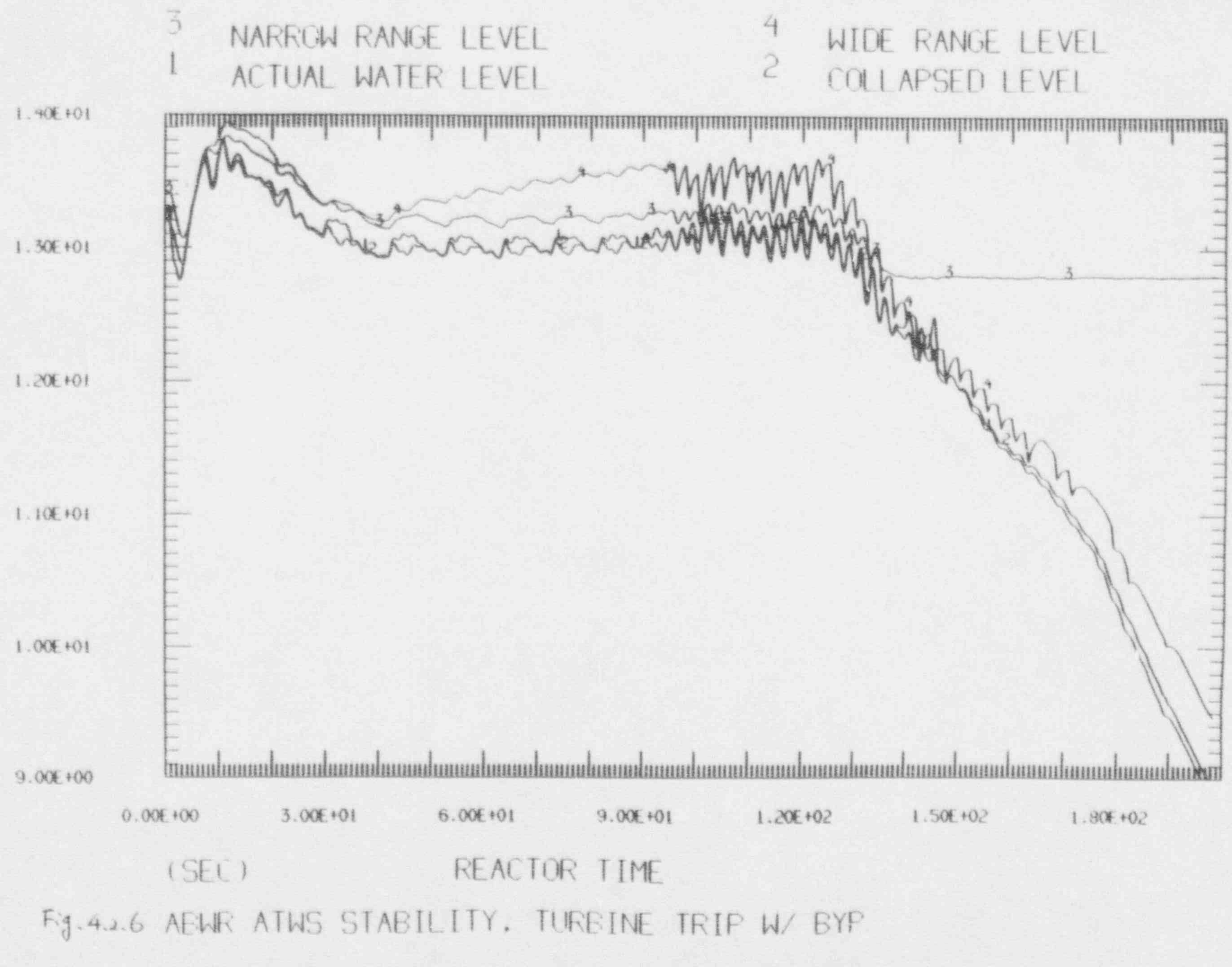

Fig.4.2.6 ABWR ATWS STABILITY, TURBINE TRIP W/ BYP

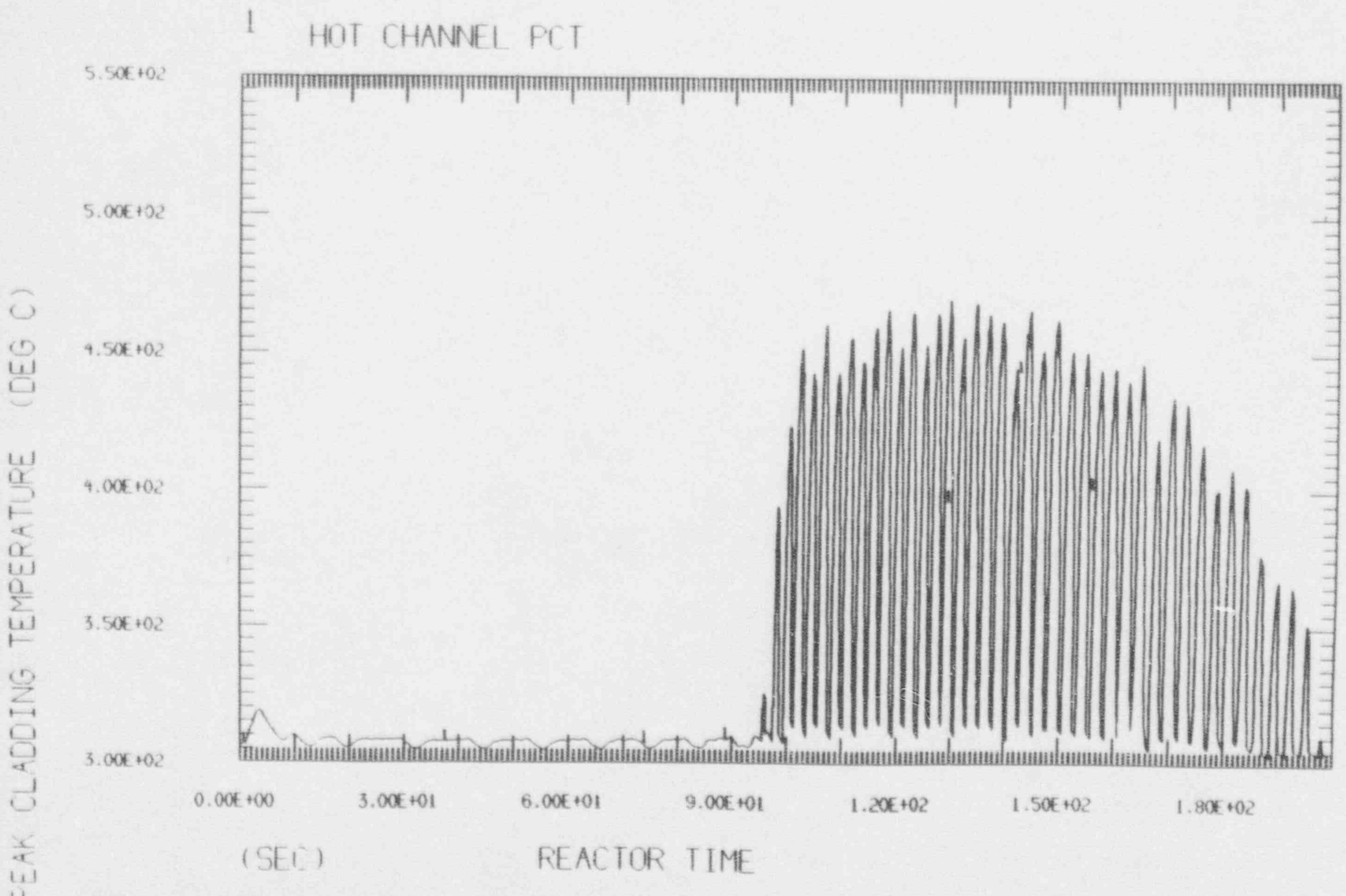

Fig. 4.2.7 ABWR ATWS STABILITY, TURBINE TRIP W/ BYP