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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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NRC FIN No. A-3350

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BNL-NUREG -31417
INFORMAL REPORT

LIMITED DISTRIBUTION

ANALYSIS OF A PARTIAL SCRAM EVENT IN A TYPICAL BWR/4

M.S. LU, W.G. SHIER, M.M. LEVINE AND R.J. CERBONE

JANUARY 1982

PLANT ANALYSIS TRANSIENT GROUP

DEPARTMENT OF NUCLEAR ENERGY BROOKHAVEN NATIONAL LABORATORY
UPTON, NEW YORK 11973

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Upton, New York 11973

January 1982

This project was conducted under the direction of NRC's Division of Systems Integration, Director: Roger Mattson, Assistant Director for Reactor Safety: Themis Speis; Branch Chief for Reactor Systems: Brian Sheron; Project Manager: M.W. Hodges and Technical Monitor: Charles Graves
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ABSTRACT

An accident with the potential for serious consequences occurs if there is an MSIV closure followed by a partial scram failure. One of the criteria to mitigate the effect of this accident is that the pressure suppression pool temperature remain below a specified limit. RELAP3B was used to obtain the pool temperature assuming the validity of GE calculations of post-MSIV closure power level and boron reactivity worth. The peak temperature was 148°F; below the safety limit and in good agreement with GE results.

1. INTRODUCTION

Anticipated boiling water reactor (BWR) transients in which there is no reactor trip (i.e. no scram) have been, and continue to be, an issue in the licensing process (1). BNL has, in the past, provided technical assistance to the NRC staff by doing independent calculations of these "ATWS" events (2). In 1980 there was a partial scram failure at the Browns Ferry 3 plant. As a result, it was of interest to the NRC staff to have BNL analyze an anticipated transient in which some of the control rods failed to insert after a reactor trip.

The present study focused on two aspects of this accident: the temperature of the pressure-suppression pool and the amount of boron required to shut down the reactor. Comparisons were made between results calculated at BNL and those quoted by General Electric (3). This report discusses the results for pool temperature. A future report will document the comparison of boron requirements.

The pool temperature is determined using the RELAP3B code (4) for a transient initiated by the inadvertent closure of the main steam line isolation valves (MSIV) in a BWR/4 at (approximately) rated core power and flow conditi[illegible] The MSIV closure event is one of the most limiting for suppression po[illegible] emperature (and peak vessel pressure). Upon reactor trip it is assumed that only some of the control rods insert. Other safety systems are assumed operable and hence, during the transient, the recirculation pumps trip (on a high pressure signal), pressure relief valves operate, the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems are activated (on a low water level signal) and the standby liquid control system (SLCS) is turned on (by operator action). The RELAP3B calculation assumes the validity of GE calculations for the power in the relatively quiescent period after 20 seconds of the transient.

In Section 2 the calculational methods used to obtain the pool temperature are explained. Results of the calculations are given in Section 3. Section 4 discusses conclusions and recommendations.

2. COMPUTATIONAL MODEL

2.1 RELAP3B Reactor Model

The RELAP3B reactor model represents the Peach Bottom 2 reactor (5,6). This model has been shown to give excellent results for the analysis of the Peach Bottom 2 turbine trip tests (7,8). It has also been applied successfully to the analysis of turbine trip events both with (8) and without (2) scram.

The initial reactor state corresponds to operation at 104.5% of rated power and 100% of rated flow. Table 1 lists some of the initial conditions.

The reactor protection and control system as assumed for the calculation is summarized in Table 2. The reactor trip assumes that only some of the control rods are inserted into the core (see Sect 2.2 and 2.3) The feedwater control system was not modelled in RELAP3B. The flow rate was assumed known (6) and was imposed as a boundary condition. It is based on a trip due to the MSIV closure followed by activation at 35 seconds (due to a low water level) and then shutoff again at about 135 seconds (see Fig. 9). The low level signal for HPCI/RCIC actuation is Level 2 which is 5.8 feet below the normal water level outside the steam separator skirt (see Sect. 2.5). The high level signal for shutoff is Level 8 which is 3.8 feet about the normal level.

Table 1 Reactor initial conditions

Power, MW	3440
Core flow rate, Mlb/hr	92.8
Dome pressure, psia	1020
Steam/Feed flow, lb/sec	3900
Feedwater temperature, °F	380

The trip levels given in Table 2 for the safety and relief valves (S/RVs) are for closing and opening respectively. The four S/RV banks represent 4,4,3 and 2 valves respectively and have a total capacity of 88% of nominal steam flow. The first three banks are relief valves and the fourth is safety valves. The SLCS was activated on a time signal at 11 min. to represent operation action.

2.2 RELAP3B Nodalization

The RELAP3B representation of the Peach Bottom 2 plant used for this analysis consists of 20 control volumes (or nodes) and 27 flow paths. Figure 1 shows the arrangement. The core is represented by four fuel regions: two represent the region where the control rods are fully inserted and two represent the region where only part of the control rods are inserted. The hot channel representation has been removed from the PB-2 model described in Reference 8.

The steam line from the reactor vessel up to the main steam isolation valve is represented by three nodes. Previous analyses (7) have shown that this representation for the steam line is sufficient to provide a reasonable calculation of the pressure waves that enter the steam dome following the closure of the main steam isolation valve.

Table 2 Reactor protection and control system

	Trip Signal	Note
MSIV closure	time zero	4 sec for closure
Reactor Trip	MSIV 10% closure	0.27 sec delay
Recirculation pump trip	1165 psia	0.53 sec delay
Feedwater	--	Modelled as a boundary condition
HPCI	water level	663 lb/s, 90°F
RCIC	water level	79 lb/s, 90°F
S/RV		All banks have 0.4 s delay and open in 0.1 s
Bank 1	1086, 1106 psia	871.8 lb/s capacity
Bank 2	1096, 1116 psia	871.8 lb/s capacity
Bank 3	1106, 1126 psia	653.8 lb/s capacity
Bank 4	1237, 1257 psia	518.5 lb/s capacity
SLCS	11 min	43 gpm with 23800 ppm boron

2.3 RELAP3B Reactivity Model

The analysis of overpressurization transients by others (e.g. Reference 8) has shown the importance of spatial neutron kinetics during the early phase of the transient. However, a recent study (2) has demonstrated that for the long-term transient, an appropriate point kinetics model would be adequate. The reactivity parameters for this model are derived from a spatial kinetics model (BNL-TWIGL) coupled to the plant transient model (RELAP3B) during the early phase of the transient. This point kinetics model was found to be adequate even when the power was oscillating due to pressure oscillations caused by the rapid opening and closing of the relief valves. Since in the present study we are interested in the long-term effects, the point kinetics model of RELAP3B is acceptable.

The reactivity ρ is assumed to be separable into components:

$$\rho = \rho_c + \rho_\alpha + \rho_f + \rho_m + \rho_b$$

where ρ_c is the result of control rod movement, ρ_α, ρ_f and ρ_m are the result of changes in void fraction (α), fuel temperature (T_f) and moderator temperature (T_m), respectively, and ρ_b is the result of adding soluble boron.

The total reactivity due to partial control rod insertion was not calculated directly. Instead, the assumption was made that the reactor power level in the relatively quiescent period (after 20 sec) after rod insertion was 10%

of the initial power - as had been calculated by GE (3). An iteration between the power (calculated with RELAP3B) and the scram reactivity was then done to obtain the worth of the inserted control rods. The 10% power level was the sum for the fully rodded regions in the core where there was only decay heat and the partially rodded regions. The time dependence of the control rod reactivity was taken from GE (6).

The reactivity parameters to describe the thermal-hydraulic feedback were those obtained previously with the coupled BNL-TWIGL and RELAP3B codes (2). The resulting reactivity components in units of dollars are

$$\rho_\alpha = -45.44\ (\alpha-\alpha_0) - 23.47\ (\alpha-\alpha_0)^2$$

$$\rho_f = -0.3553\ (T_f^{1/2} - T_{fo}^{1/2})$$

$$\rho_m = -0.06\ (T_m - T_{mo})$$

where α_0, T_{fo} and T_{mo} are the initial values. These formulae are applied to each core node to get the contribution from each node. The nodal reactivities are weighted by a power fraction, which is based on an assumed initial power distribution, and then summed to get the total reactivity.

The RELAP3B calculation uses a very coarse mesh (6 ft.) along the flow path in the core in order to minimize computer costs. Since this increases the error in the nodal-average void fraction, a correction factor is applied to the calculated void fraction before it is used to calculate ρ_α. This correction factor is based on the initial void distribution obtained using 10 axial nodes (2,8). It ensures that the coarse mesh nodal void fraction is equal to the void fraction averaged over the corresponding 5 nodes in the fine mesh calculation at time zero.

The boron reactivity worth used is 0.689$/lb (obtained from GE). Power fraction weighting is also applied to this component.

2.4 RELAP3B Phase Separation Model

In a BWR plant there are standpipes, separators and dryers between the core exit plenum and the steam dome. The standpipes guide the steam-liquid mixture out of the core exit plenum to the centrifugal separators where most of the liquid-vapor separation is achieved via centrifugal forces. The slightly wet mixture leaving the separator enters the dryer region where more phase separation is achieved.

RELAP3B solves conservation equations for the mixture mass, momentum and energy. In order to represent phase separation effects, a bubble rise model is usually used. For the present analysis, bubble rise models are employed in the separator, vessel dome and the downcomer region. In order to achieve more effective phase separation so that high quality steam enters the steam line (as designed) and in order to prevent excessive amounts of bubbles from being carried (circulated) into the downcomer and reentering the core region, a large bubble rise parameter (10^5 ft/sec) has been used. At steady state conditions, the phase separation in the separator is nearly complete (99.8% quality). The pressure drop across the separator has also been set equal to the GE design value. Throughout the transient (0-1500 seconds), the quality in the steamline remains above 99.7%, and no appreciable bubbles have been recirculated back into the core region via the downcomer region. The artificial phase separation is a limitation of the RELAP3B modeling. The effects can be evaluated by using more advanced computer codes with better phase separation modeling such as RAMONA3B, RELAP5, and TRAC-BD1.

2.5 RELAP3B Water Level Estimate

In a BWR the HPCI and RCIC systems trip on and off based on the water level in the reactor vessel. Usually, with a homogeneous two-phase model, as is in RELAP3B, the water level calculation is based on a bubble rise model. The bubble rise model in the RETRAN code has been shown (9) to be deficient in predicting the water level transients in turbine trip tests. Since the basic thermal-hydraulic model in the RELAP3B code is not much different from that of the RETRAN code, the water level calculation for RELAP3B was based on a different method. Based on the information available from the Peach Bottom 2 plant (6), the coolant mass in Volumes 10, 11 and 12 (Fig. 1) corresponding to the water levels for HPCI and RCIC trips (on and off) was first calculated. The RELAP3B code was then modified so that during the transient, if the sum of the coolant mass in Volumes 10, 11 and 12 falls below the trip-on value (about 14000 lb less than steady state conditions), the HPCI and RCIC would be turned on, and would stay on until the coolant mass rises to more than the trip-off value (about 65000 lb more than the steady state value). This modeling corresponds to a collapsed water level. However, since the water level measurement in a BWR also corresponds to the collapsed level, this modeling should be adequate.

2.6 Calculation of Pool Temperature

Containment integrity is one of the more important safety considerations for an accident in which only some of the control rods are available to shut down the reactor and there is MSIV closure. During such an accident steam is dumped into the suppression pool via the safety/relief valves. This removes energy from the core and increases the temperature of the suppression pool. The residual heat removal system (RHR) is capable of removing some of this energy from the suppression pool. In order to maintain the integrity of the containment system the pool temperature should not be excessive. For a BWR/4 it is necessary to limit this temperature to 160°F.

The pool average temperature T is calculated by applying conservation of mass and energy:

$$\frac{d}{dt}(mcT) = Wh - UA(T-T_o)$$

$$\frac{dm}{dt} = W$$

where m is the mass of water in the pool, c is the specific heat of the pool water, UA is the cooling capacity of the RHR and T_0 is the service water (RHR) temperature. The mass flow rate W and the enthalpy h of the steam flow through the S/RVs into the pool are calculated by RELAP3B. However, after the reactor is shut down (~1500 sec) and the only power generated is due to decay heat, the RELAP3B calculation is not used. Instead it is asumed that the rate of energy addition to the pool is equal to the decay heat generation rate Q (10) and the flow rate into the pool is Q divided by the saturated steam enthalpy. Some of the data used for this calculation is listed in Table 3.

Table 3 Pressure suppression pool and RHR characteristics

Volume, ft^3	17237
Initial temperature, °F	90
RHR service water temperature, °F	85
RHR cooling capacity, BTU/sec-°F	578.8

3. RESULTS

The major events during the transient as calculated by RELAP3B are summarized in Table IV. Figures 2-11 show power, reactivity components and important pressures and flows for the first 3 minutes of the transient. Figures 12-21 show the same results up to 25 minutes. The specific parameters plotted are:

Figures 2, 12	Normalized power level
Figures 3, 13	Total reactivity
Figures 4, 14	Void reactivity
Figures 5, 15	Doppler reactivity
Figures 6, 16	Coolant temperature reactivity
Figures 7, 17	Control rod reactivity
Figures 8, 18	Safety/relief valve flow
Figures 9, 19	Feedwater, HPCI/RCIC flow
Figures 10, 20	Steam line pressure
Figures 11, 21	Vessel dome pressure

The integrated safety/relief flow into the suppression pool is shown in Figure 22. The pool temperature transient is given in Figure 23.

The transient begins with the closure of the MSIV in 4 seconds. With MSIV closure, the pressure pulse so generated quickly reaches the reactor vessel, resulting in a void collapse and power increase (cf Figs. 10, 11, 4 and 2). The scram signal is initiated by the high power level (120%). After a 0.27 second delay, the control rods on one side of the reactor are fully inserted, and the control rods on the other side of the reactor are assumed to be partially stuck (cf. Fig. 7). The transient reactor power reaches the peak value of 195% at 3.9 seconds and decreases rapidly to about 10% due to the inserted control rod worth, Doppler and void feedback (cf. Figs. 2-7). At 4.3 seconds, the safety/relief valves open to arrest the pressure rise (cf Fig. 8). The vessel dome pressure reaches the recirculation pump trip set point pressure of 1165 psia at 5.5 seconds. After a delay of 0.53 second, both recirculation pumps trip to reduce long-term reactor power level. The vessel dome pressure continues to increase until about 8.3 seconds, when it peaks at 1232 psia (cf. Fig. 10).

The feedwater was first run back to zero at 18 seconds and then reactuated at 35 seconds due to low water level in the vessel. This relatively cold water causes a gradual increase in reactor power (cf. Figs. 2 and 9). The feedwater is finally terminated around 135 seconds. Following this, the reactor vessel water level keeps decreasing. At about 10 minutes, the level reaches Level 2 - the setpoint of the High Pressure Coolant Injection and the Reactor Core Isolation Cooling system (cf. Fig. 19). These two systems are designed to provide cooling for the reactor core, but they also introduce some positive reactivity (due to the increased core coolant flow and the reduced core coolant temperature), and hence slightly increase the power (cf. Fig. 12). Throughout the transient, the reactor power follows the cycling of safety/relief valves (cf. Figs. 12 and 18).

At 10 minutes, the Residual Heat Removal (RHR) system is also assumed to be turned on to remove heat from the suppression pool. The Standby Liquid Control System (SLCS) is also assumed to be initiated by the operator and boron flow reaches the jet pump nozzles at 11 minutes to provide negative reactivity for reactor shutdown (cf. Fig. 13). At 14 minutes, the reactor vessel water level reaches Level 8 - the level to turn off HPCI and RCIC systems (cf. Fig. 19). The negative reactivities introduced by the SLCS finally brings the reactor to hot shutdown condition at about 20 minutes (cf Fig. 12).

The flow rate (cf. Fig. 22) and the enthalpy out of the safety/relief valves are used to calculate the pool temperature. The results are shown in Figure 23. Up to 1500 seconds, the calculated pool temperature agrees very well with GE's results (3). At 1500 seconds, the pool temperature is 2°F below that obtained by GE. In terms of the temperature increment from the steady state condition, the difference is 4%. After 1500 seconds the calculation is done differently (see Sect. 2.6) and the difference between the GE and BNL results increases. The pool temperature peaks at about 5100 seconds at a value of 148°F calculated by BNL and 153° calculated by GE.

Table 4 Sequence of events

1.	MSIV closure	0 - 4 sec
2.	Scram initiated (120% power)	3.2 sec
3.	Reactor power peak (193%)	3.9 sec
4.	S/R valve opens	4.3 sec
5.	Recirculation pump trip on high pressure	5.5 sec
6.	Reactor vessel peak pressure (1232 psi)	8.3 sec
7.	Feedwater stopped	18 sec
8.	Feedwater reactuated by low level	35 sec
9.	Feedwater terminated	135 sec
10.	HPCI and RCIC flow starts	10 min
11.	RHR flow begins	10 min
12.	SLCS initiated at jet pump	11 min
13.	HPCI and RCIC stops	14 min
14.	Hot shutdown achieved	20 min

4. SUMMARY AND CONCLUSIONS

An MSIV trip should produce a reactor scram so that power is quickly reduced to the decay heat level. This heat is carried by the steam flow through the safety/relief valves to the suppression pool, whence it is removed by the residual heat removal system. In the present study it is assumed that some scram rods fail to insert into the core so that heat continues to be produced by fission. This energy production exceeds the heat removal capacity of the residual heat removal system so that the pool heats up when the fission power is reduced below the decay heat levels by the introduction of soluble boron, the pool temperature decreases.

The pool temperature has been calculated along with the thermal-hydraulic behavior of the core and recirculation system in a BWR/4. System pressure, safety/relief valve mass and energy flow rates, and reactor water level were monitored to understand the sequence of events during the accident.

The calculations were done with the RELAP3B code which has a homogeneous equilibrium model. A collapsed water level was calculated for trip setpoints. The scram reactivity was set so that the power level was ~10% of its initial value after the MSIV closure, scram and recirculation pump trip.

The peak system pressure was 1232 psia and the peak pool temperature was 148°F. Both of these values are below the safety limits for the system. The calculated pool temperature during the transient was in good agreement with that calculated by GE. Since the BNL calculations relied on GE results to obtain the scram reactivity these calculations constitute an audit of the vendor's thermal-hydraulic methods and not of the neutronic methods.

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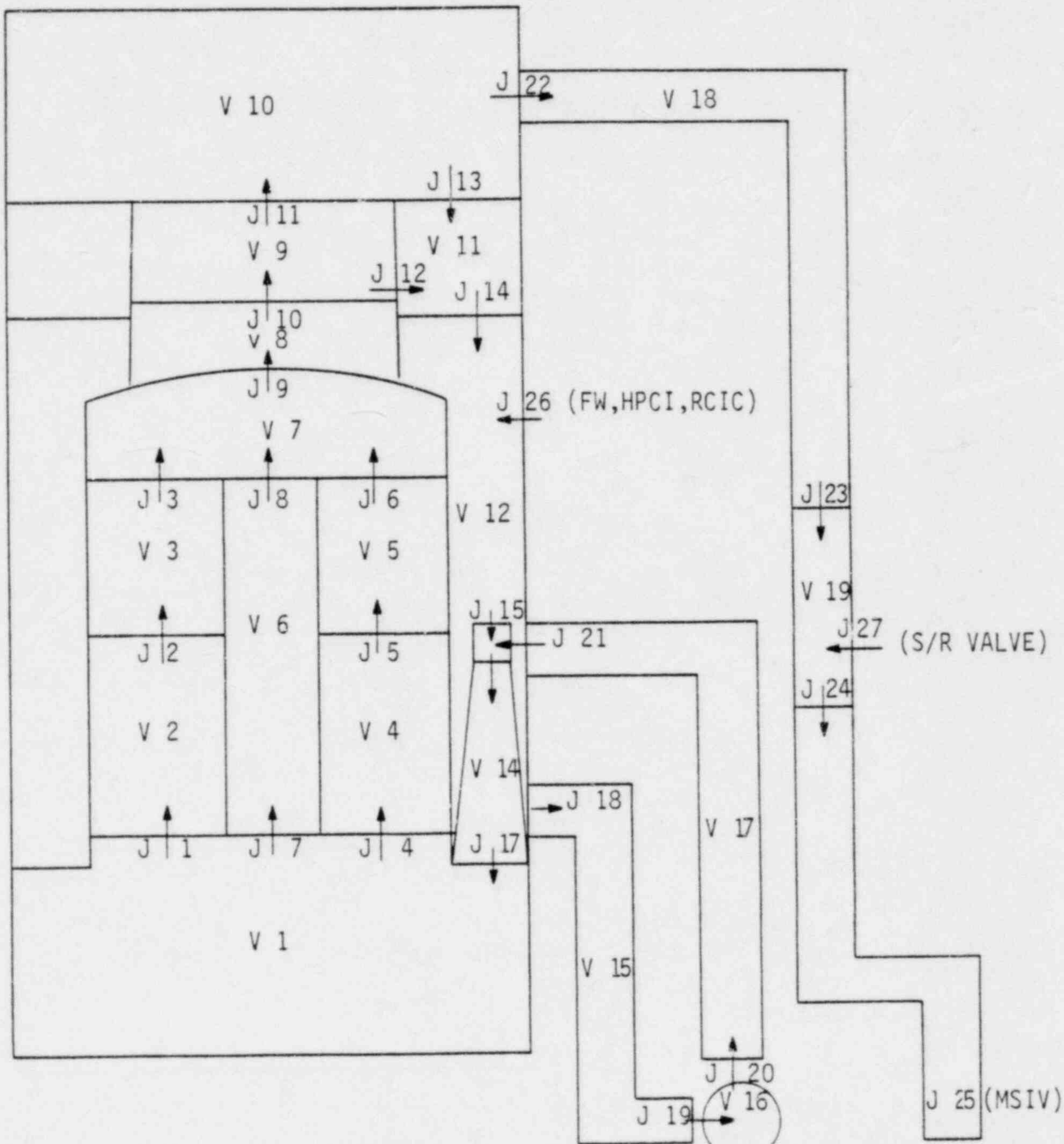
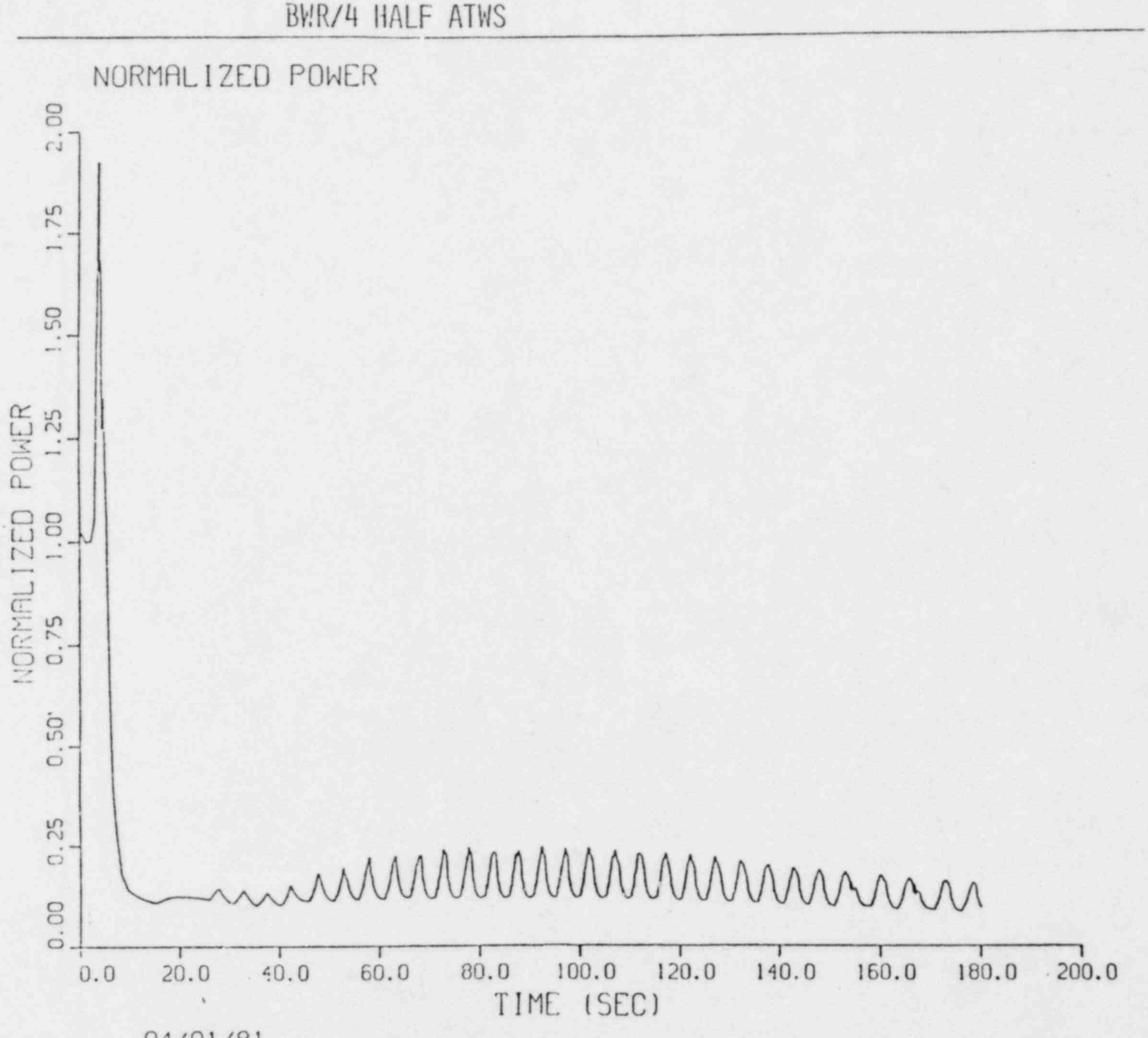


Figure 1. BWR/4 Nodal Representation.


BWR/4 HALF ATWS
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NORMALIZED POWER
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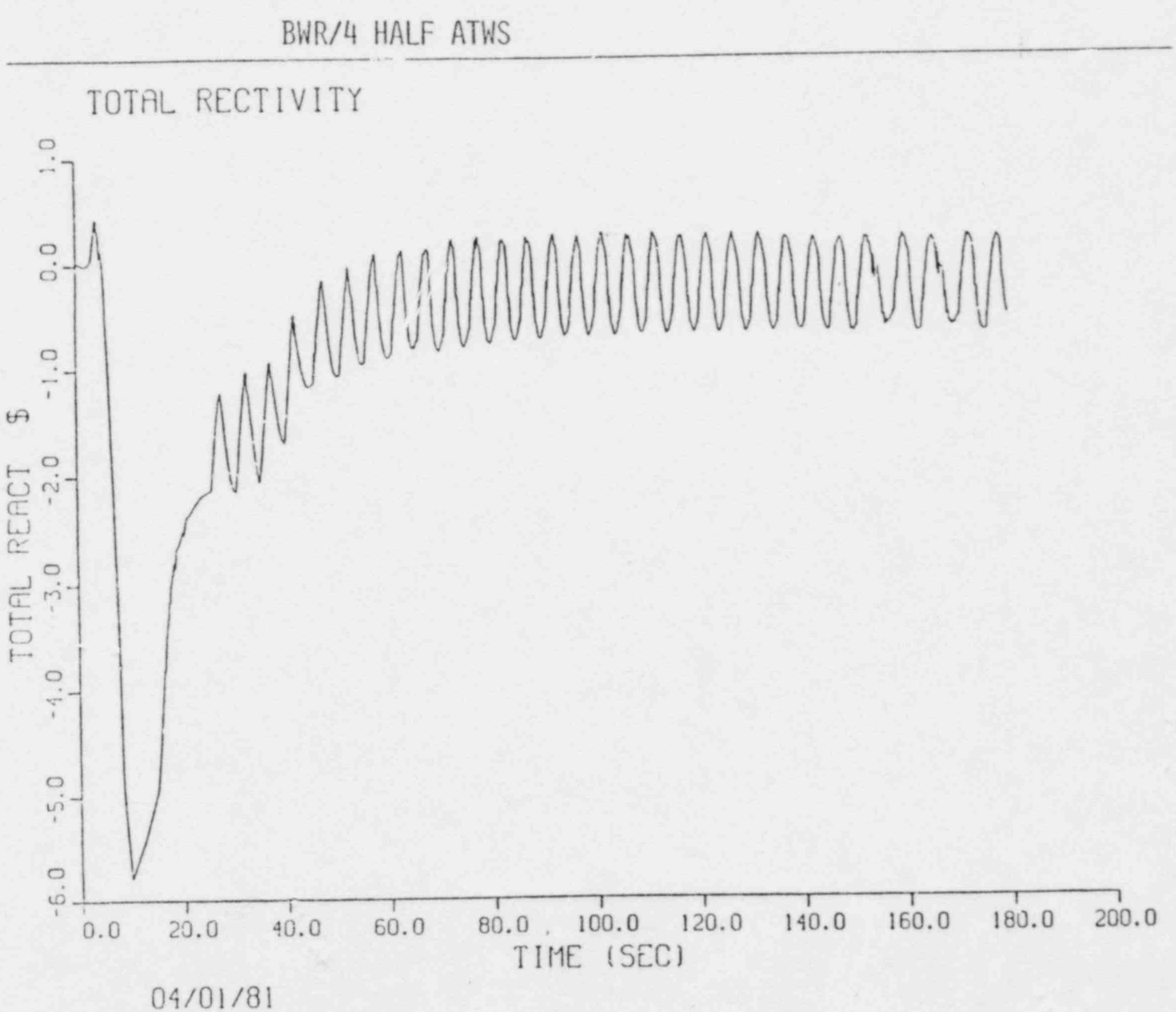
Figure 2


BWR/4 HALF ATWS
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140.0
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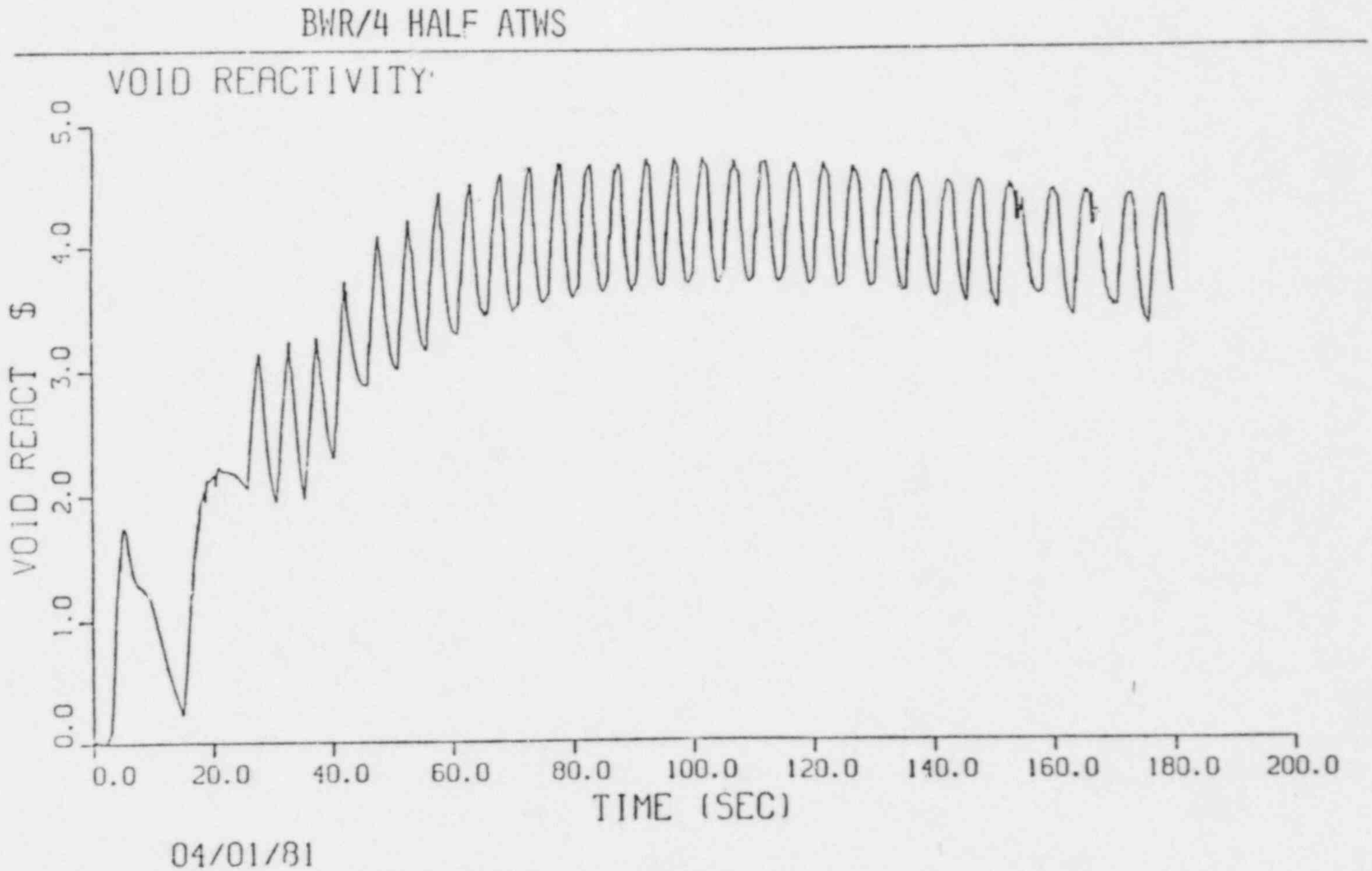
TIME (SEC)
04/01/81

Figure 3


BWR/4 HALF ATWS
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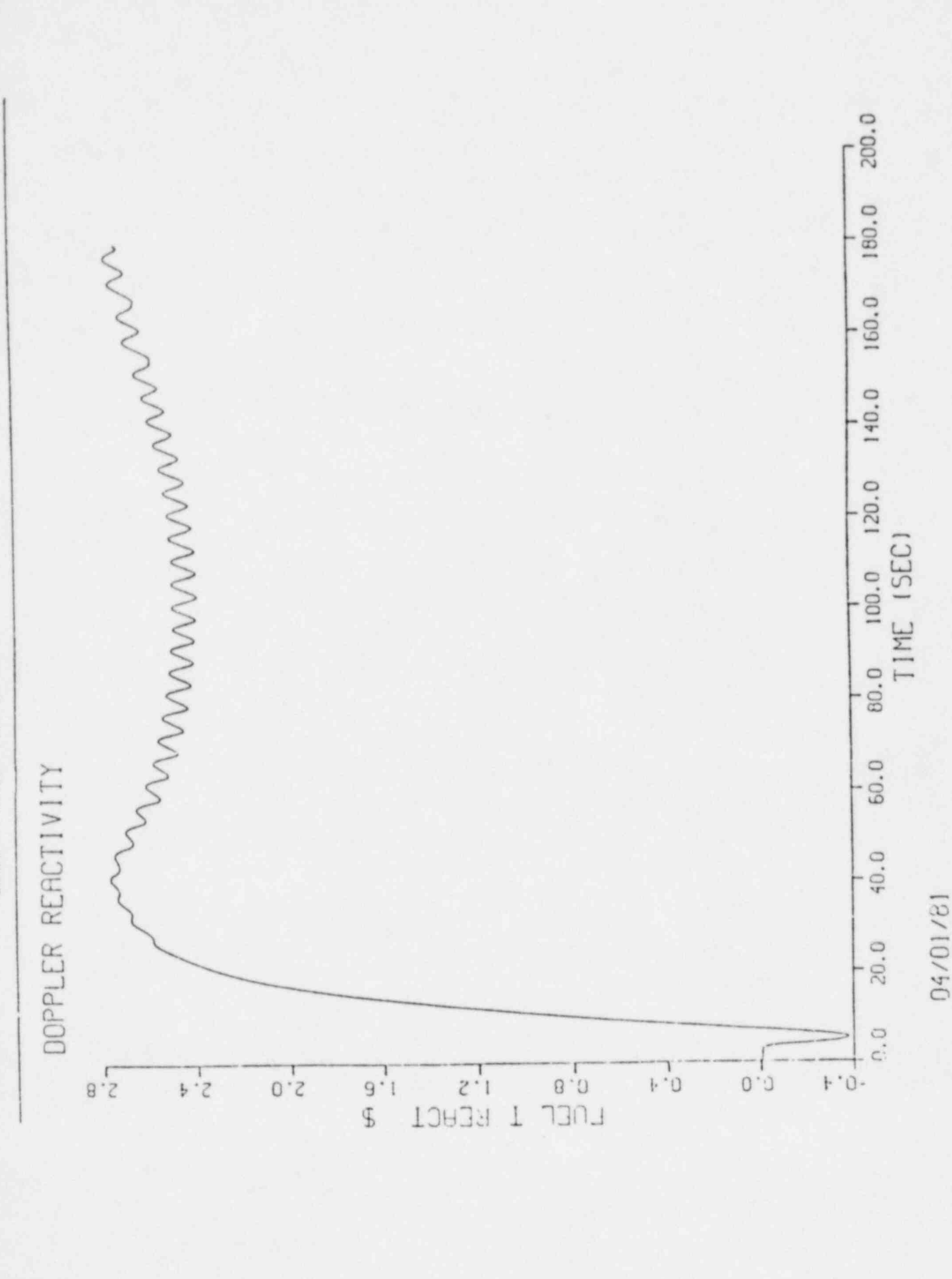
TIME (SEC)
04/01/81

Figure 4


BWR/4 HALF ATWS
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100.0
120.0
140.0
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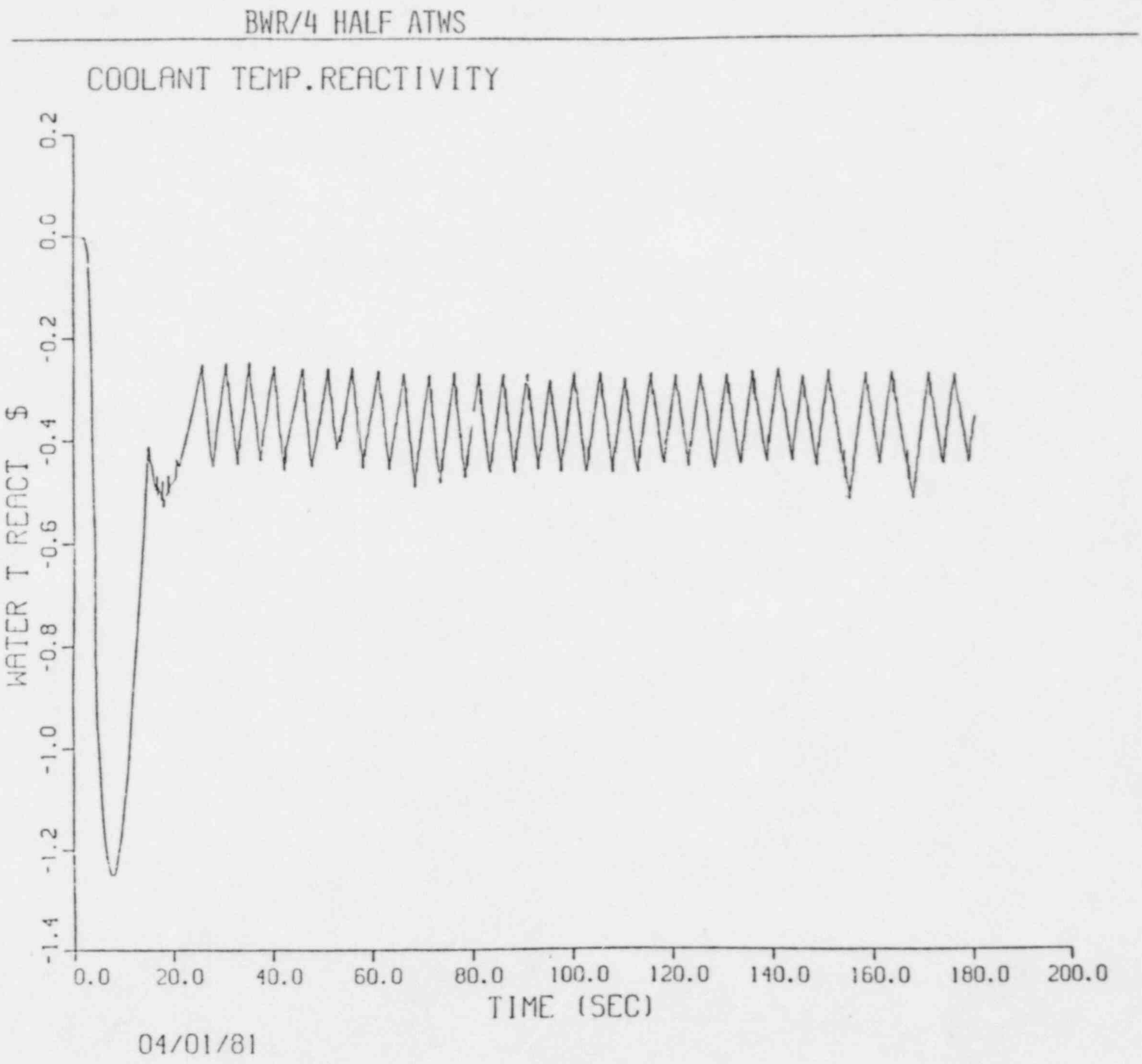
180.0
200.0
TIME (SEC)
04/01/81

Figure 5



BWR/4 HALF ATWS
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WATER T REACT $
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0.0
-0.2
-0.4
-0.6
-0.8
-1.0
-1.2
-1.4
0.0
20.0
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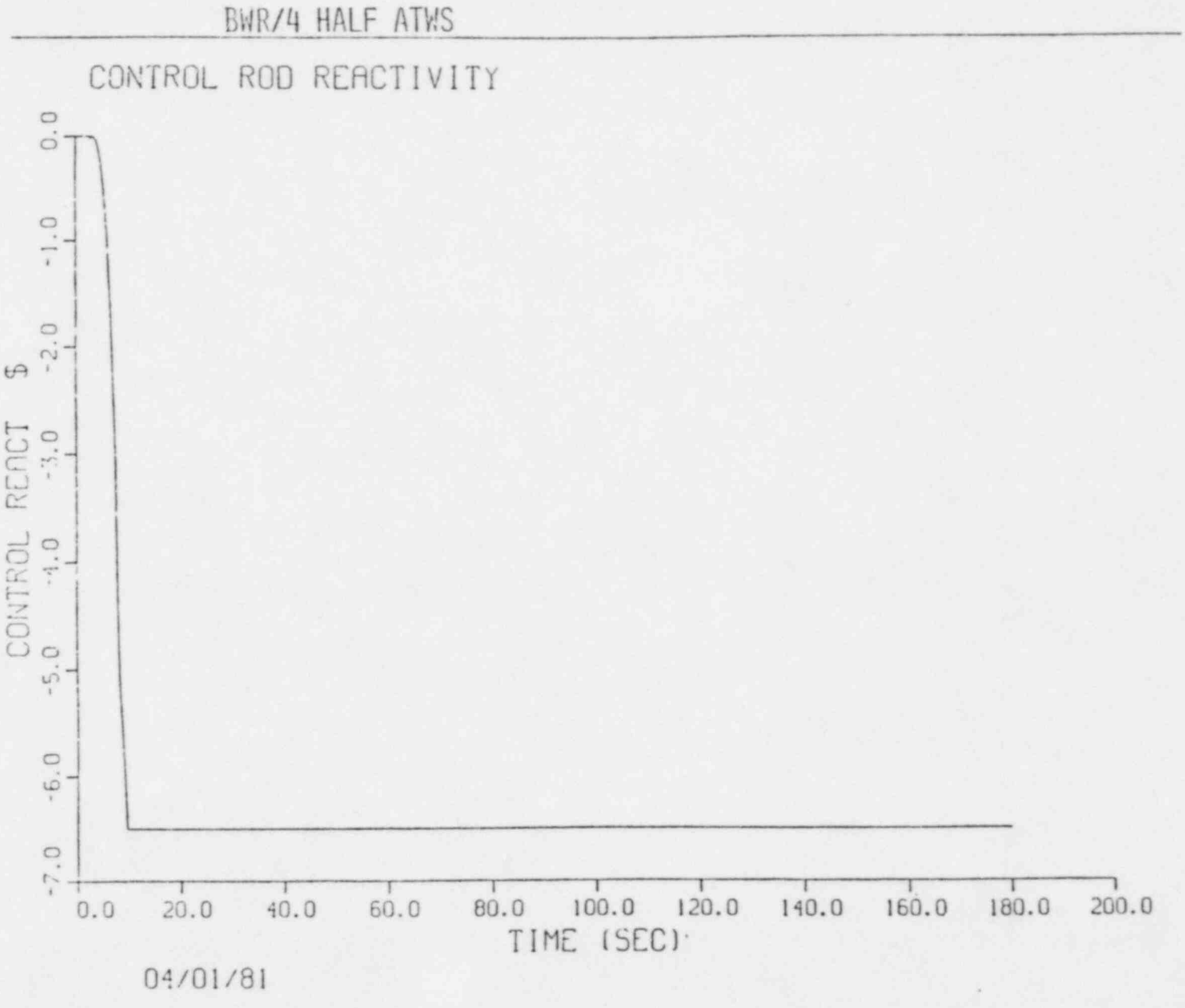
100.0
120.0
140.0
160.0
180.0
200.0
TIME (SEC)
04/01/81

Figure 6


BWR/4 HALF ATWS
CONTROL ROD REACTIVITY
CONTROL REACT $
0.0
-1.0
-2.0
-3.0
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-5.0
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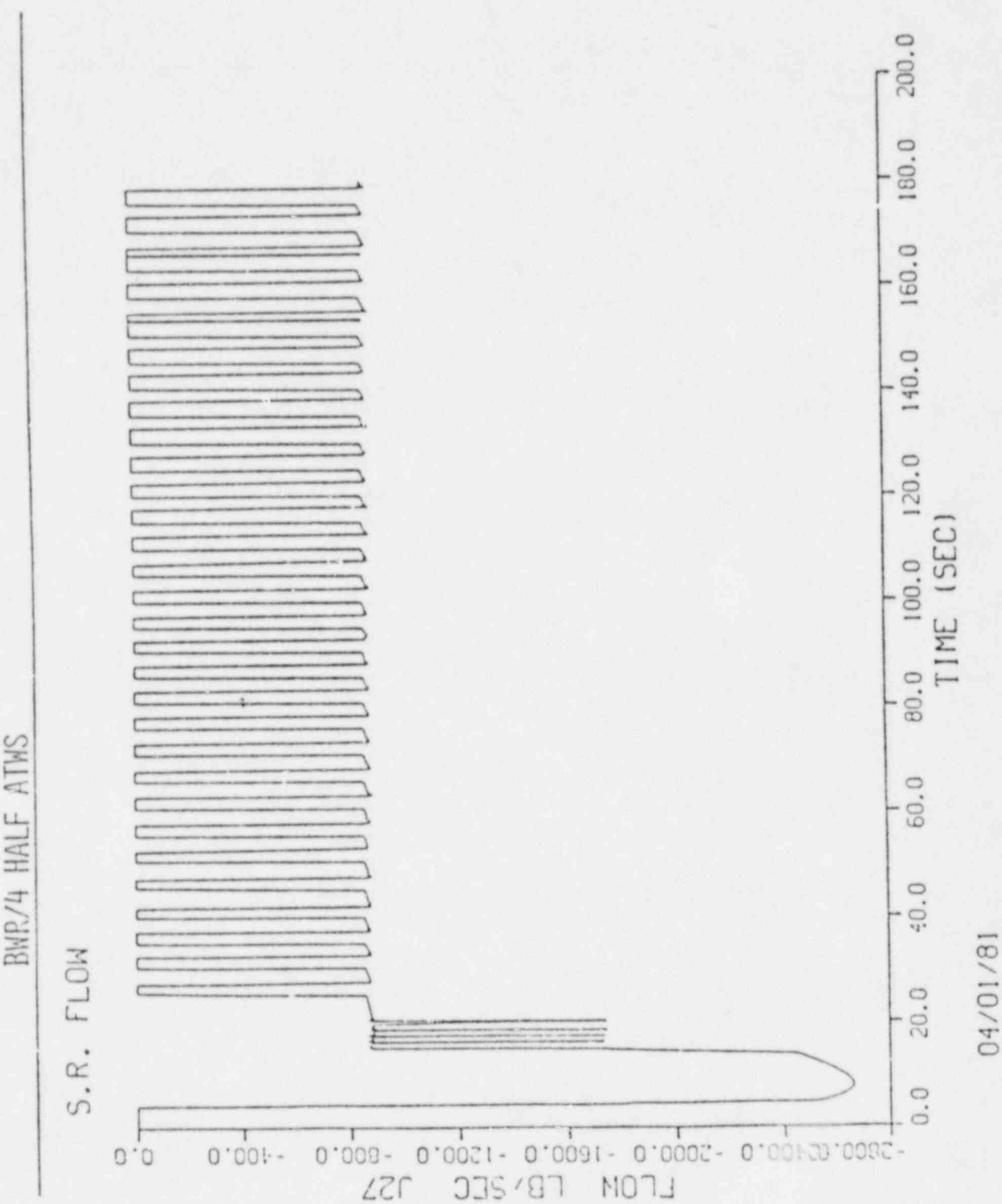
TIME (SEC)
04/01/81

Figure 7



BWR/4 HALF ATWS
S.R. FLOW
FLOW LB/SEC J27
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-800.0
-1200.0
-1600.0
-2000.0
-2400.0
-2800.0
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TIME (SEC)
04/01/81

Figure 8

BWR/4 HALF ATWS

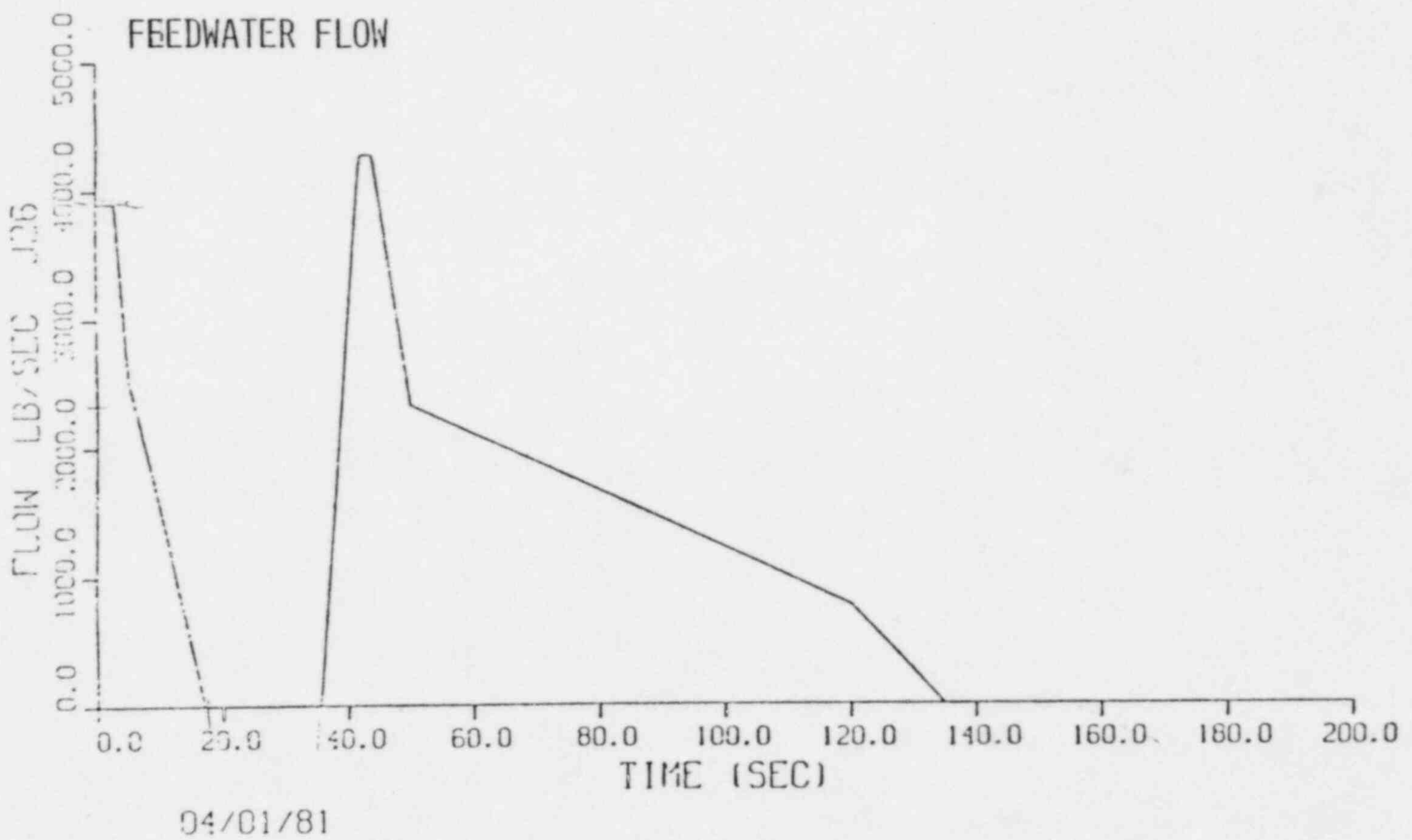
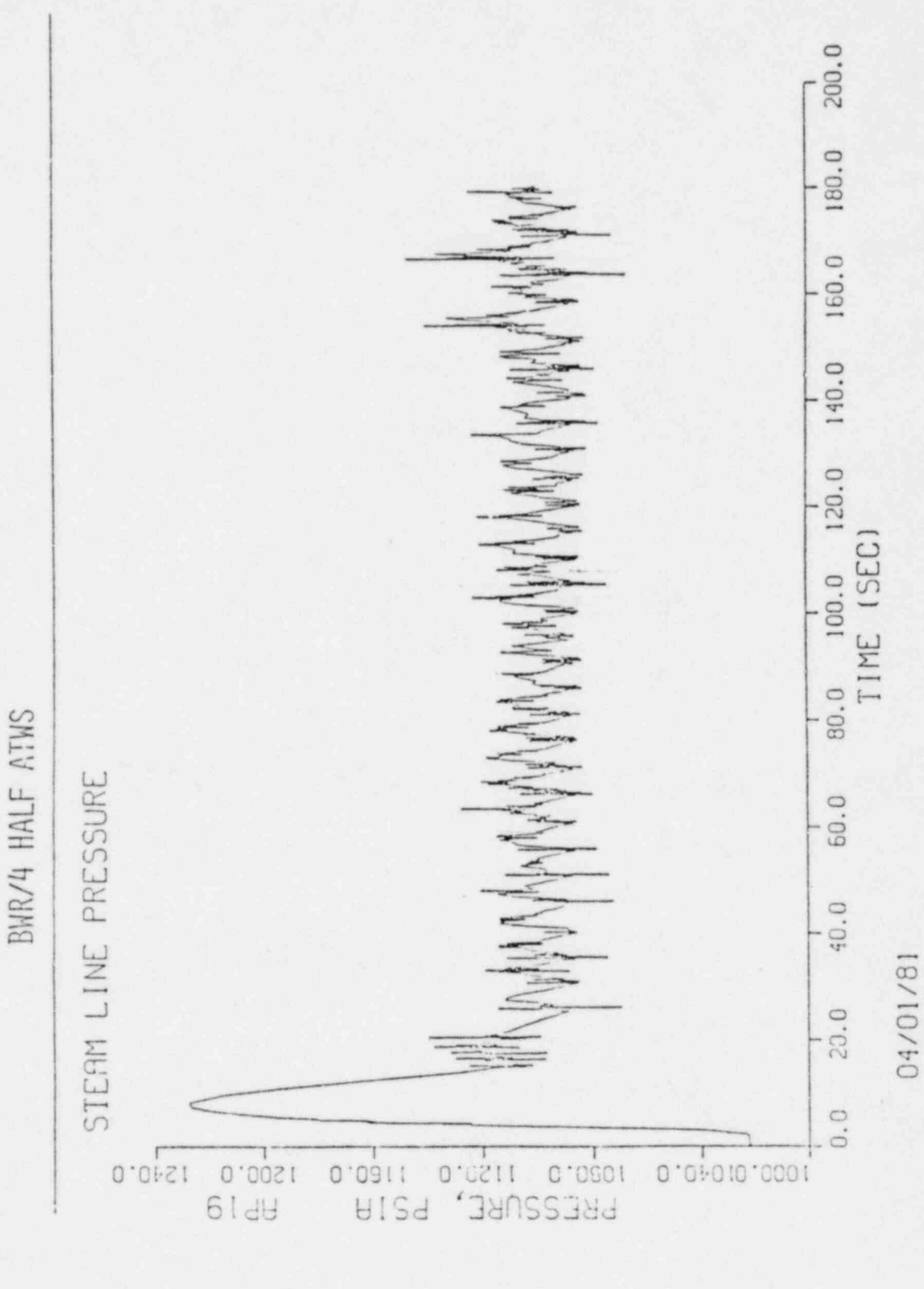
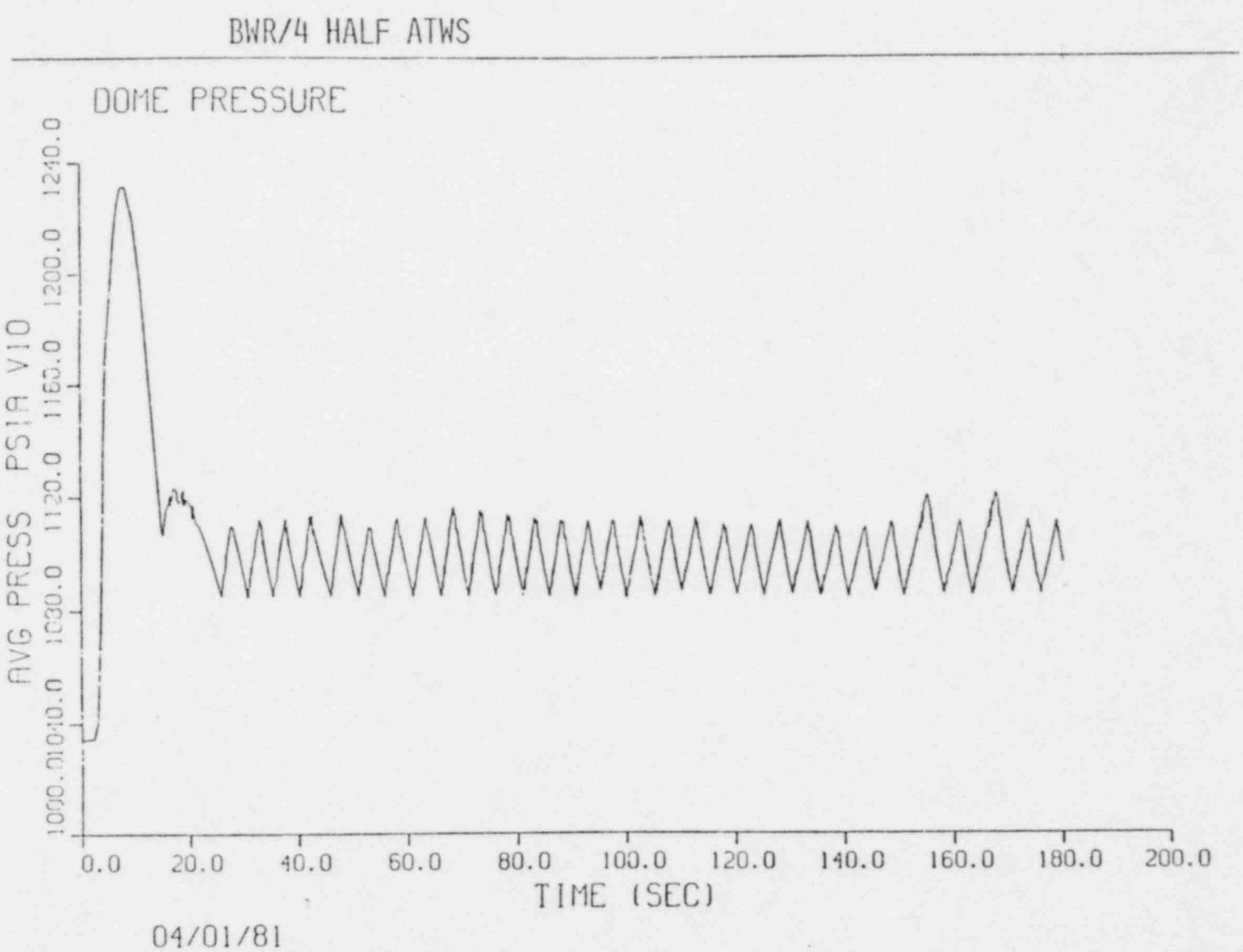


Figure 9



BWR/4 HALF ATWS
STEAM LINE PRESSURE
04/01/81
TIME (SEC)
PRESSURE, PSIA
AP19
0.0
20.0
40.0
60.0
80.0
100.0
120.0
140.0
160.0
180.0
200.0
1000.0
1040.0
1080.0
1120.0
1160.0
1200.0
1240.0

Figure 10



BWR/4 HALF ATWS
DOME PRESSURE
AVG PRESS PSIA V10
1000.0
1030.0
1120.0
1200.0
1240.0
0.0
20.0
40.0
60.0
80.0
100.0
120.0
140.0
160.0
180.0
200.0
TIME (SEC)
04/01/81

Figure 11

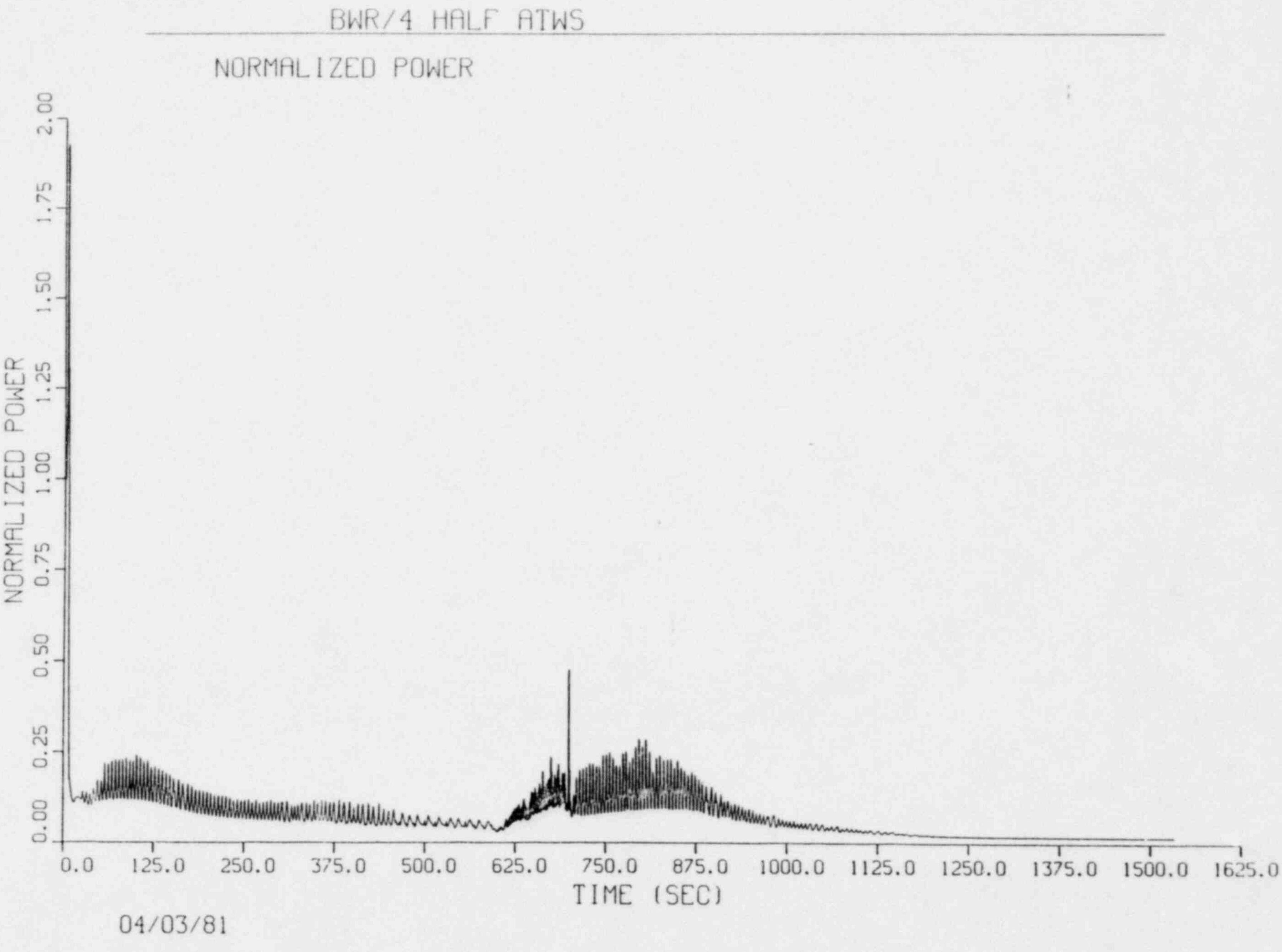

BWR/4 HALF ATWS
NORMALIZED POWER
NORMALIZED POWER
0.00
0.25
0.50
0.75
1.00
1.25
1.50
1.75
2.00
0.0
125.0
250.0
375.0
500.0
625.0
750.0
875.0
1000.0
1125.0
1250.0
1375.0
1500.0
1625.0
TIME (SEC)
04/03/81

Figure 12

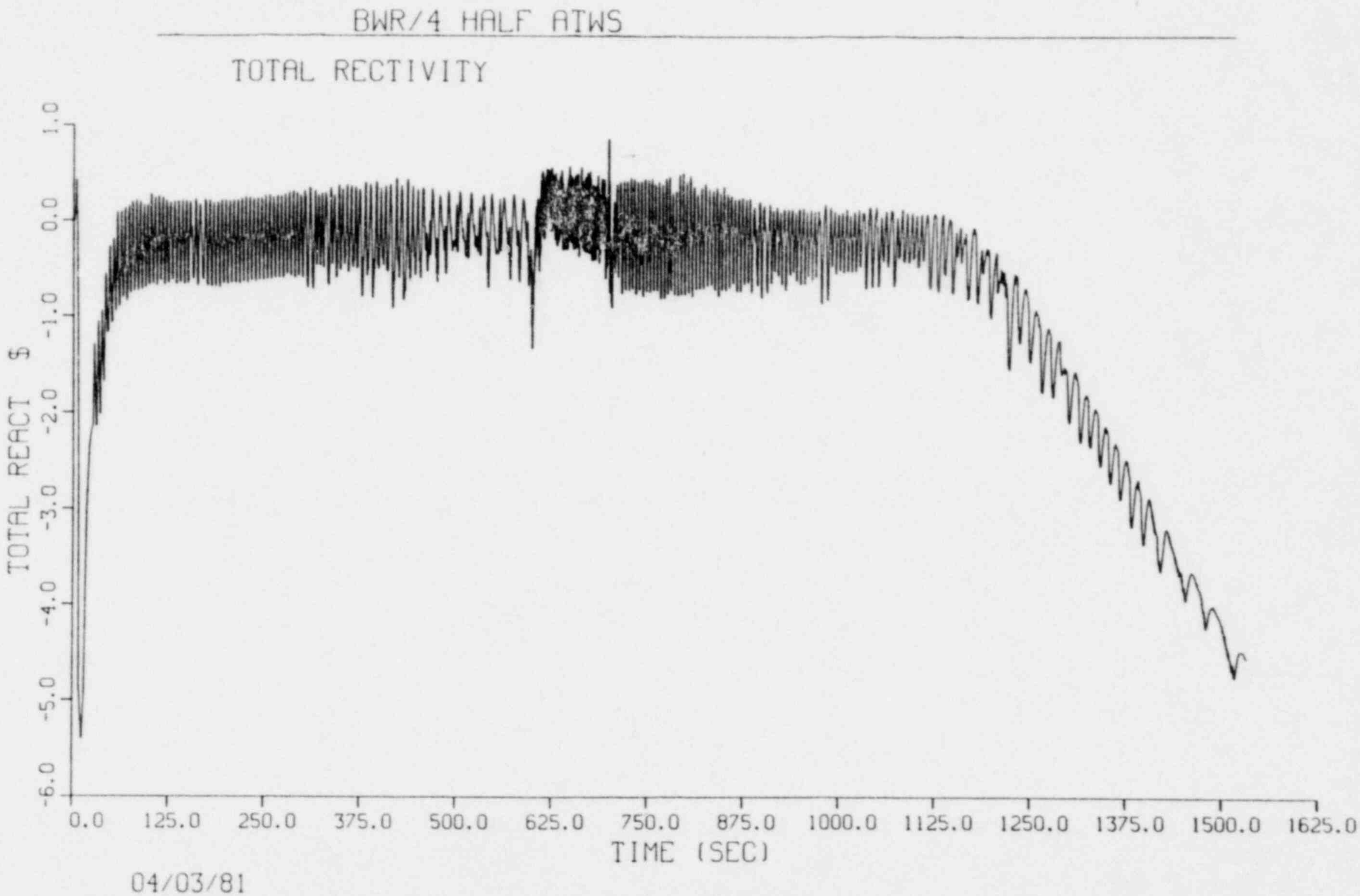

BWR/4 HALF ATWS
TOTAL RECTIVITY
TOTAL REACT $
1.0
0.0
-1.0
-2.0
-3.0
-4.0
-5.0
-6.0
0.0
125.0
250.0
375.0
500.0
625.0
750.0
875.0
1000.0
1125.0
1250.0
1375.0
1500.0
1625.0
TIME (SEC)
04/03/81

Figure 13

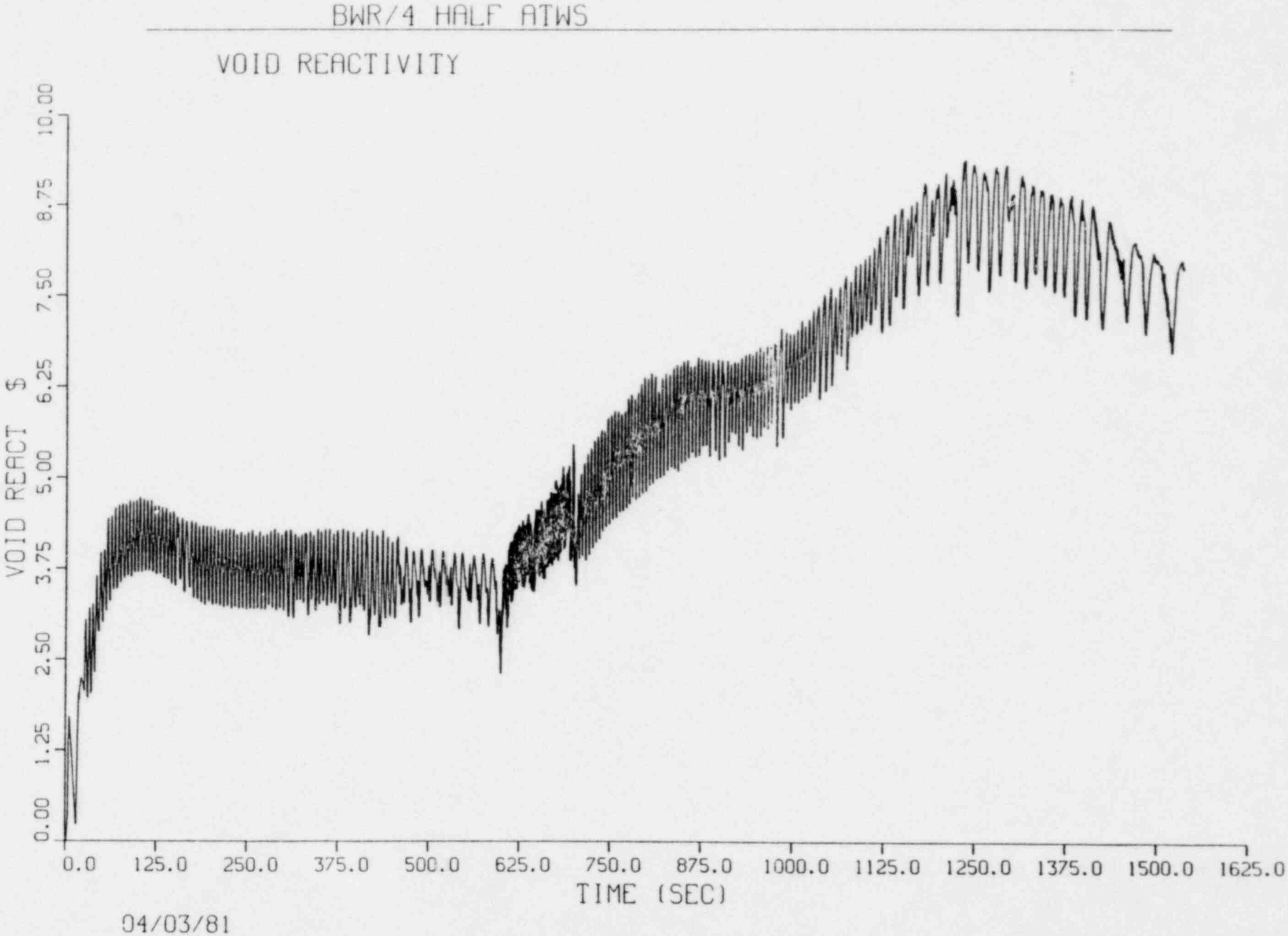

BWR/4 HALF ATWS
VOID REACTIVITY
VOID REACT $
0.00
1.25
2.50
3.75
5.00
6.25
7.50
8.75
10.00
0.0
125.0
250.0
375.0
500.0
625.0
750.0
875.0
1000.0
1125.0
1250.0
1375.0
1500.0
1625.0
TIME (SEC)
04/03/81

Figure 14

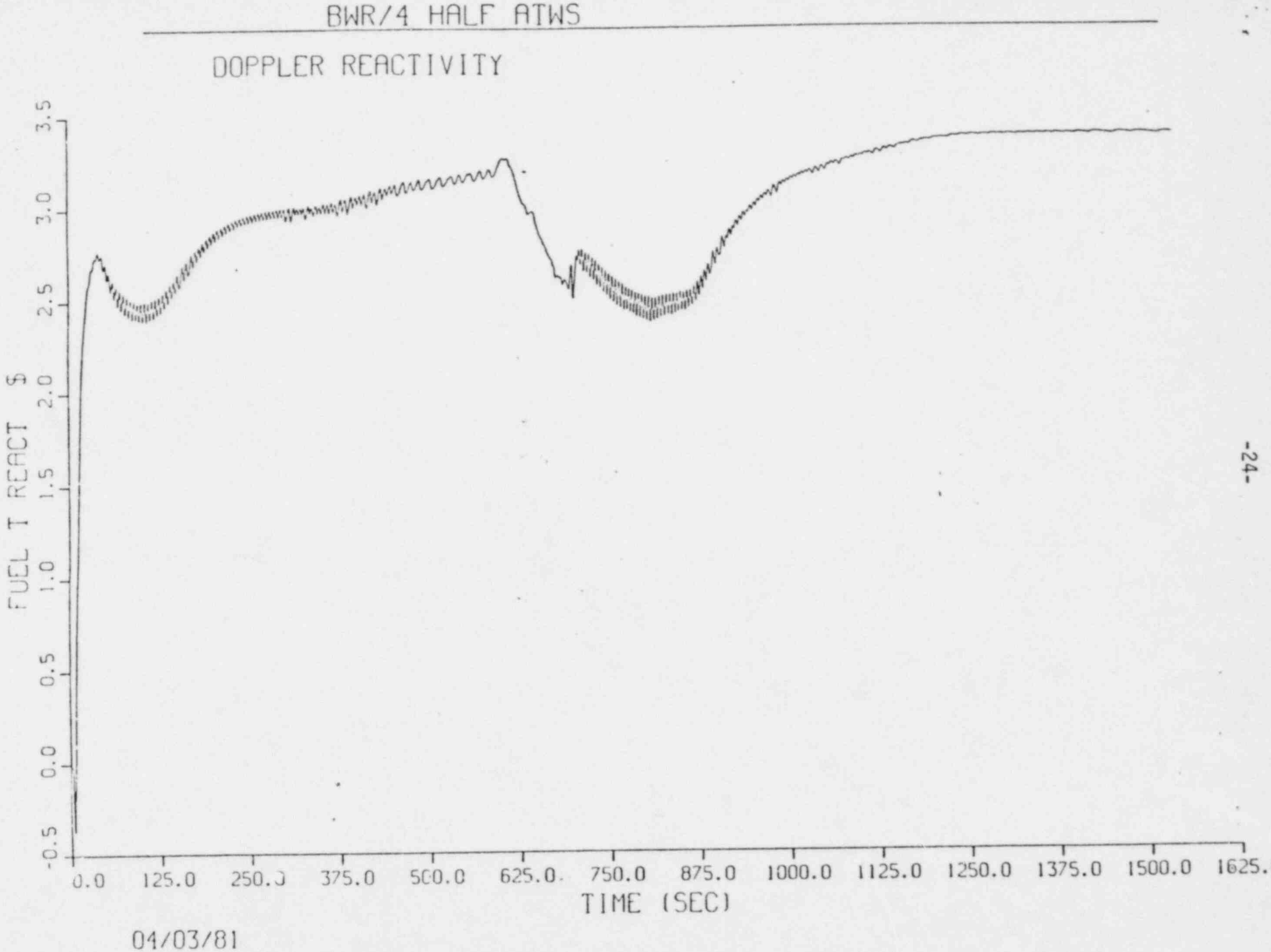

BWR/4 HALF ATWS
DOPPLER REACTIVITY
FUEL T REACT $
3.5
3.0
2.5
2.0
1.5
1.0
0.5
0.0
-0.5
0.0
125.0
250.0
375.0
500.0
625.0
750.0
875.0
1000.0
1125.0
1250.0
1375.0
1500.0
1625.0
TIME (SEC)
04/03/81

Figure 15

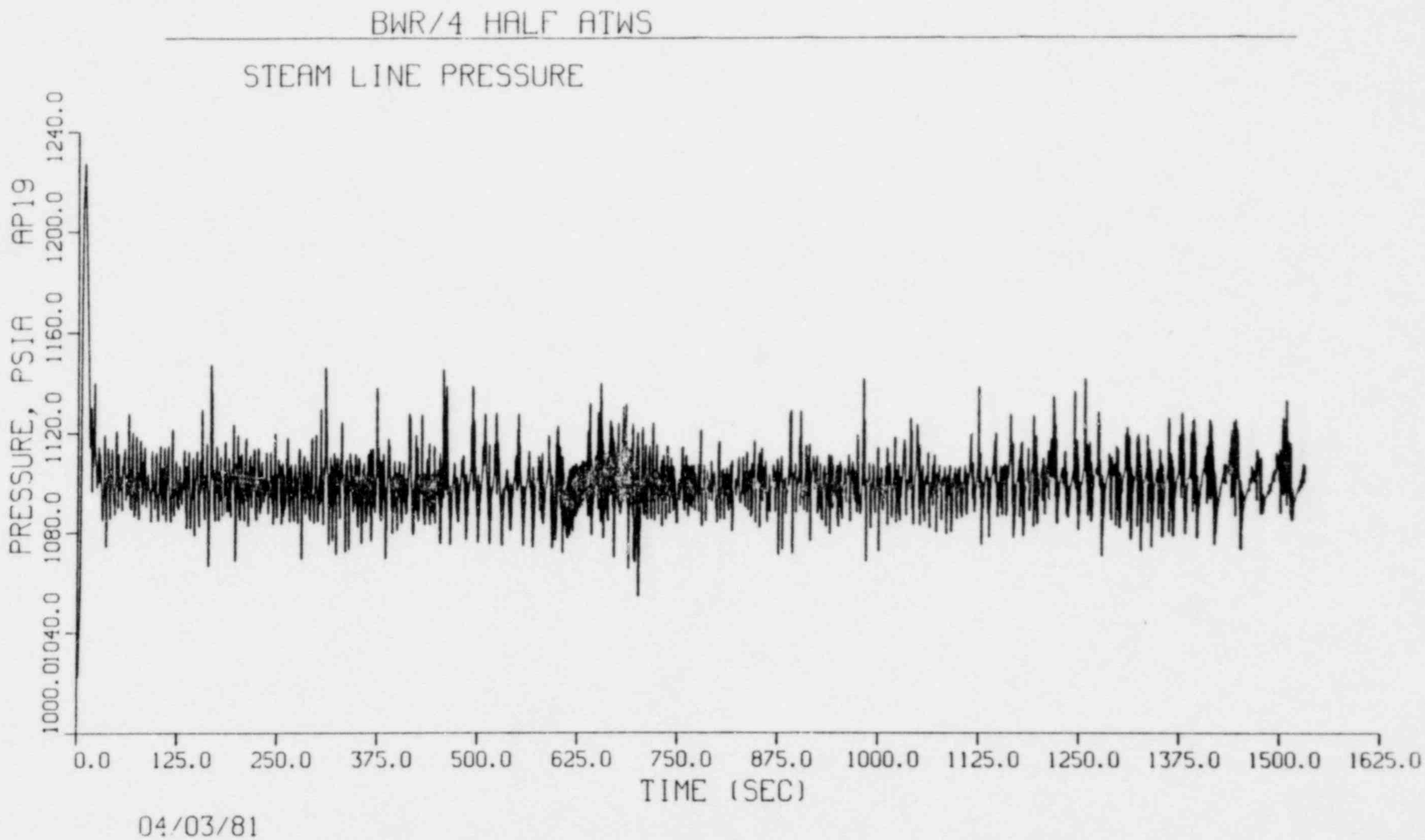

BWR/4 HALF ATWS
STEAM LINE PRESSURE
PRESSURE, PSIA AP19
1000.0
1040.0
1080.0
1120.0
1160.0
1200.0
1240.0
0.0
125.0
250.0
375.0
500.0
625.0
750.0
875.0
1000.0
1125.0
1250.0
1375.0
1500.0
1625.0
TIME (SEC)
04/03/81

Figure 16

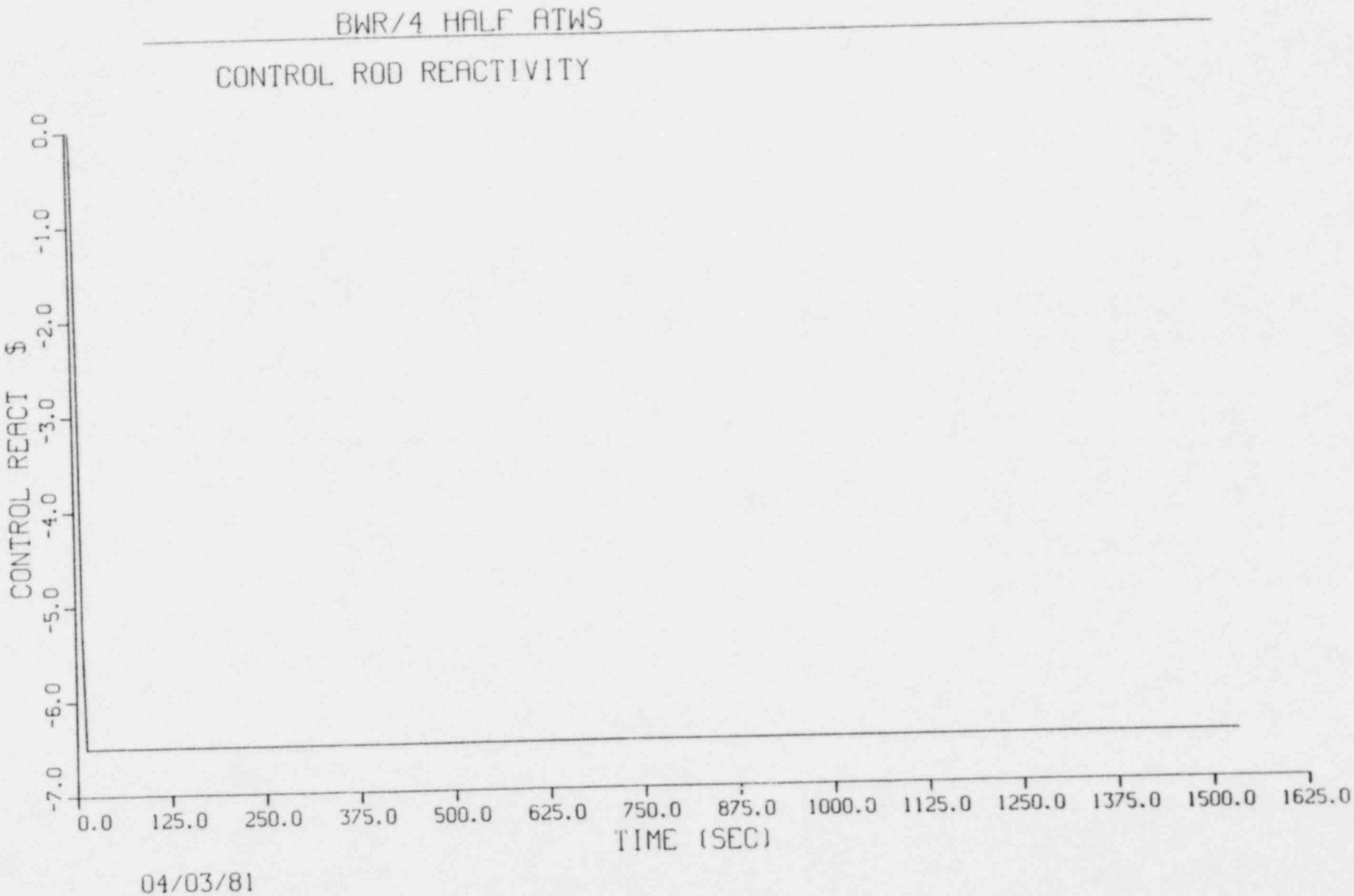

BWR/4 HALF ATWS
CONTROL ROD REACTIVITY
CONTROL REACT $
0.0
-1.0
-2.0
-3.0
-4.0
-5.0
-6.0
-7.0
0.0
125.0
250.0
375.0
500.0
625.0
750.0
875.0
1000.0
1125.0
1250.0
1375.0
1500.0
1625.0
TIME (SEC)
04/03/81

Figure 17

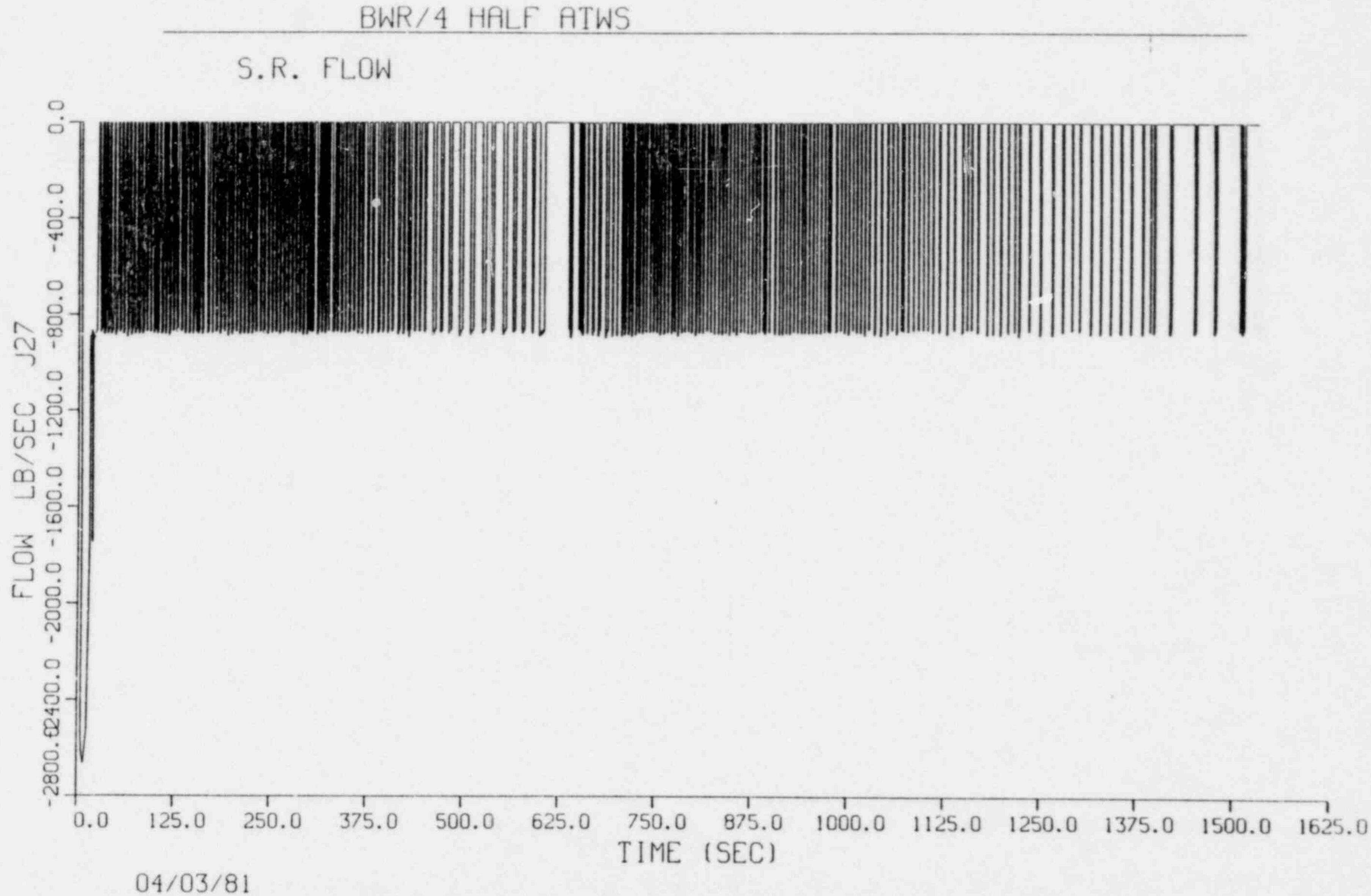

BWR/4 HALF ATWS
S.R. FLOW
FLOW LB/SEC J27
0.0
-400.0
-800.0
-1200.0
-1600.0
-2000.0
-2400.0
-2800.0
0.0
125.0
250.0
375.0
500.0
625.0
750.0
875.0
1000.0
1125.0
1250.0
1375.0
1500.0
1625.0
TIME (SEC)
04/03/81

Figure 18

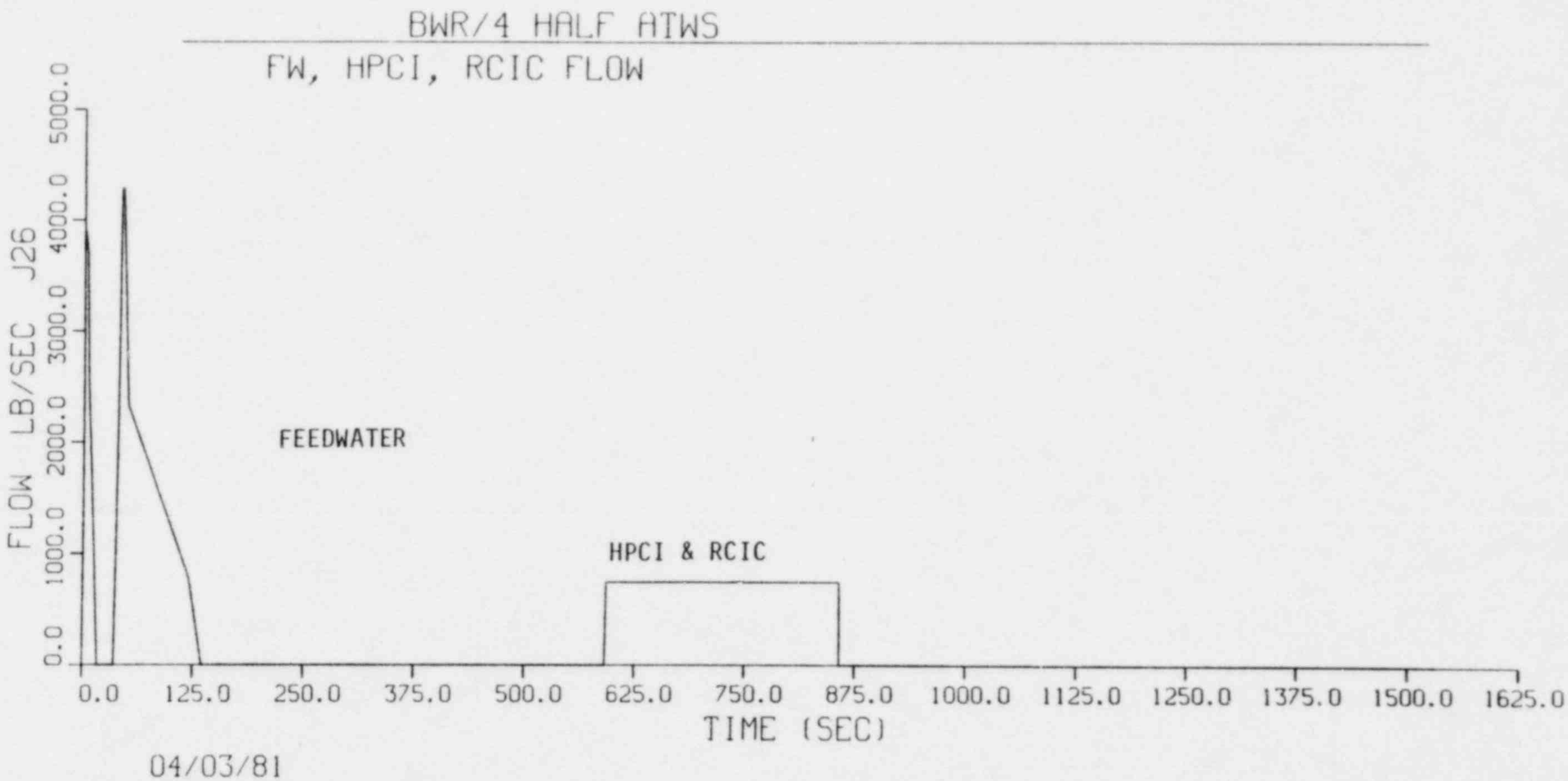

BWR/4 HALF ATWS
FW, HPCI, RCIC FLOW
FLOW LB/SEC J26
0.0
1000.0
2000.0
3000.0
4000.0
5000.0
FEEDWATER
HPCI & RCIC
0.0
125.0
250.0
375.0
500.0
625.0
750.0
875.0
1000.0
1125.0
1250.0
1375.0
1500.0
1625.0
TIME (SEC)
04/03/81

Figure 19

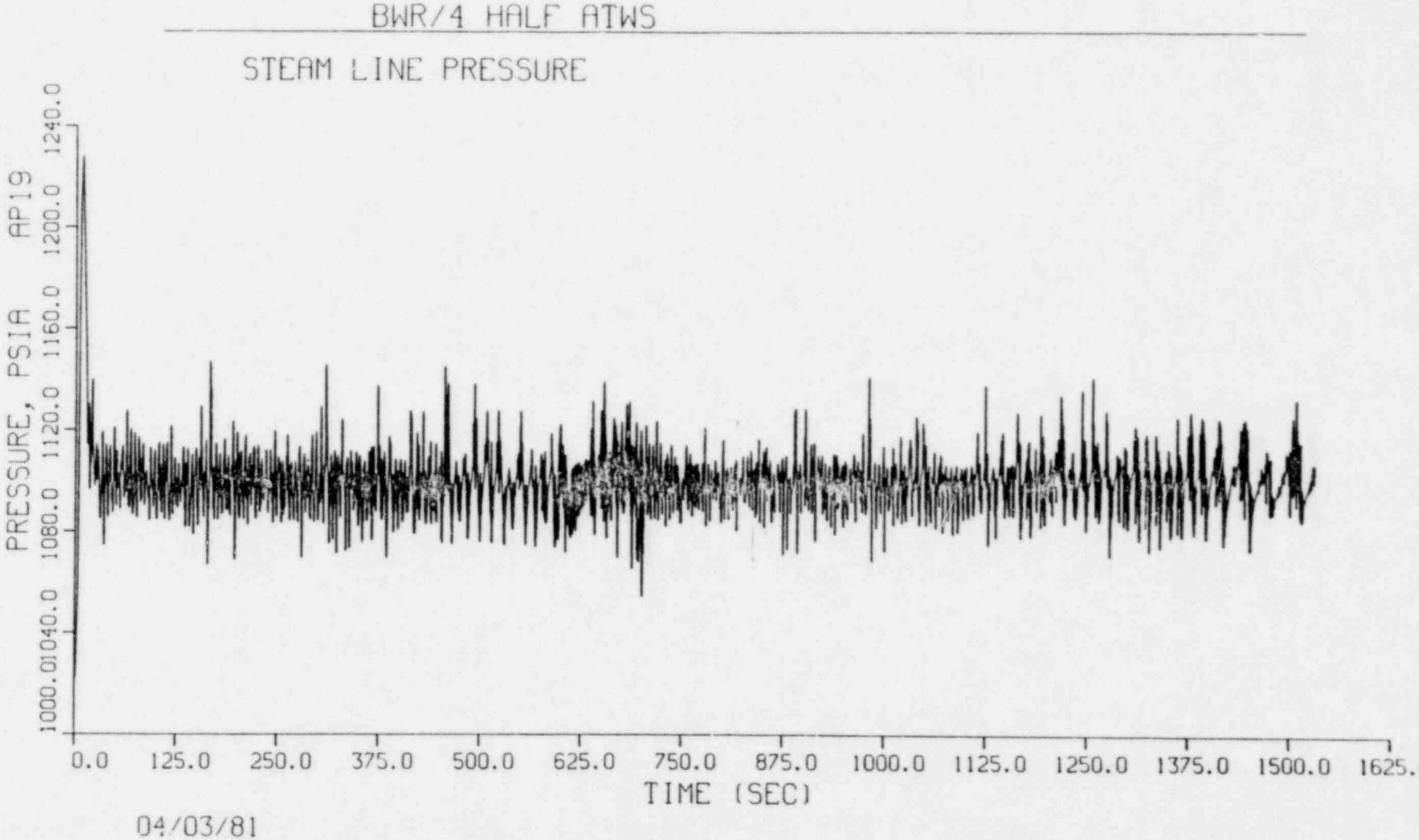

BWR/4 HALF ATWS
STEAM LINE PRESSURE
PRESSURE, PSIA AP19
1000.0
1040.0
1080.0
1120.0
1160.0
1200.0
1240.0
0.0
125.0
250.0
375.0
500.0
625.0
750.0
875.0
1000.0
1125.0
1250.0
1375.0
1500.0
1625.0
TIME (SEC)
04/03/81

Figure 20

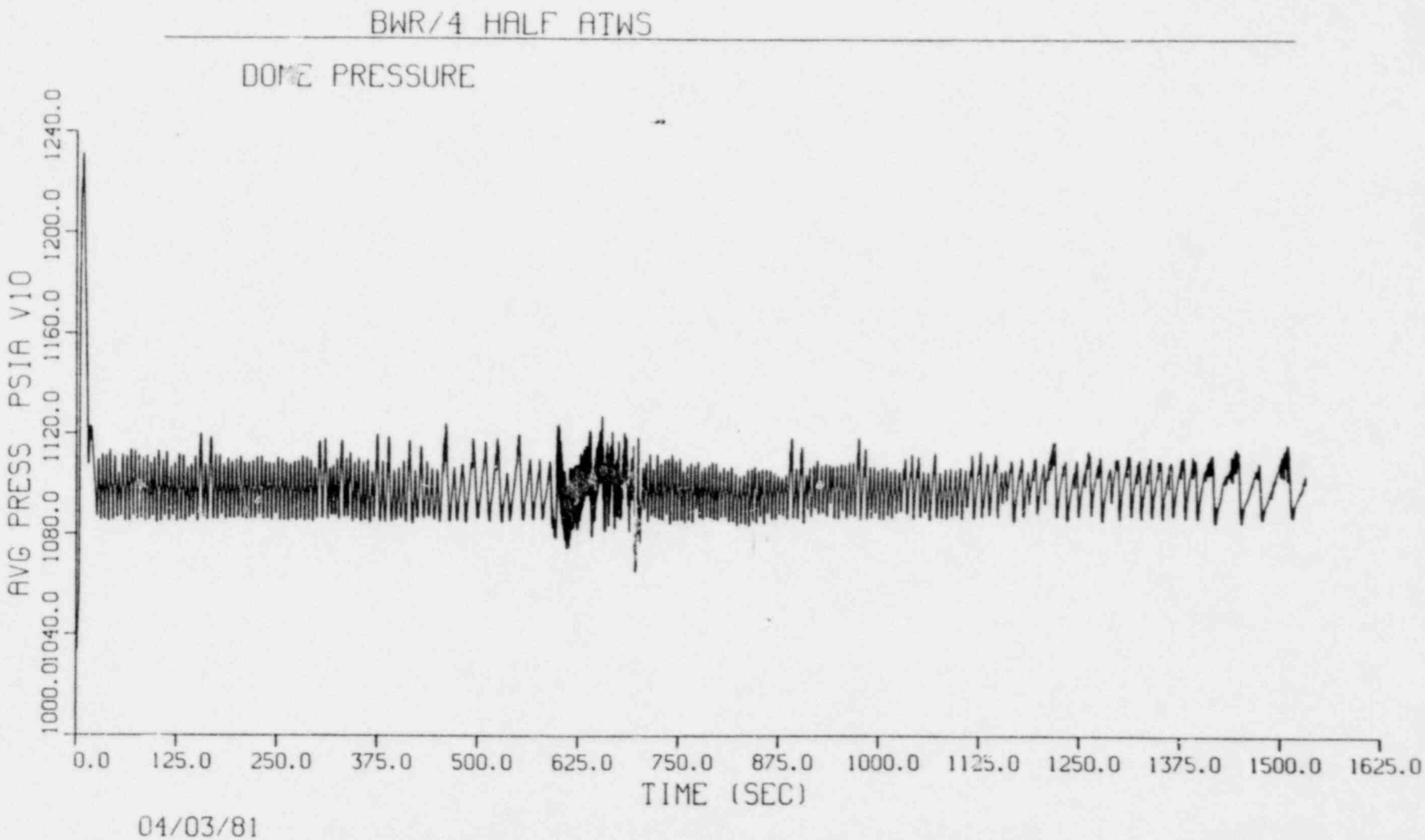

BWR/4 HALF ATWS
DOME PRESSURE
AVG PRESS PSIA V10
1000.0
1040.0
1080.0
1120.0
1160.0
1200.0
1240.0
0.0
125.0
250.0
375.0
500.0
625.0
750.0
875.0
1000.0
1125.0
1250.0
1375.0
1500.0
1625.0
TIME (SEC)
04/03/81

Figure 23

Integrated Safety/Relief Valve Flow

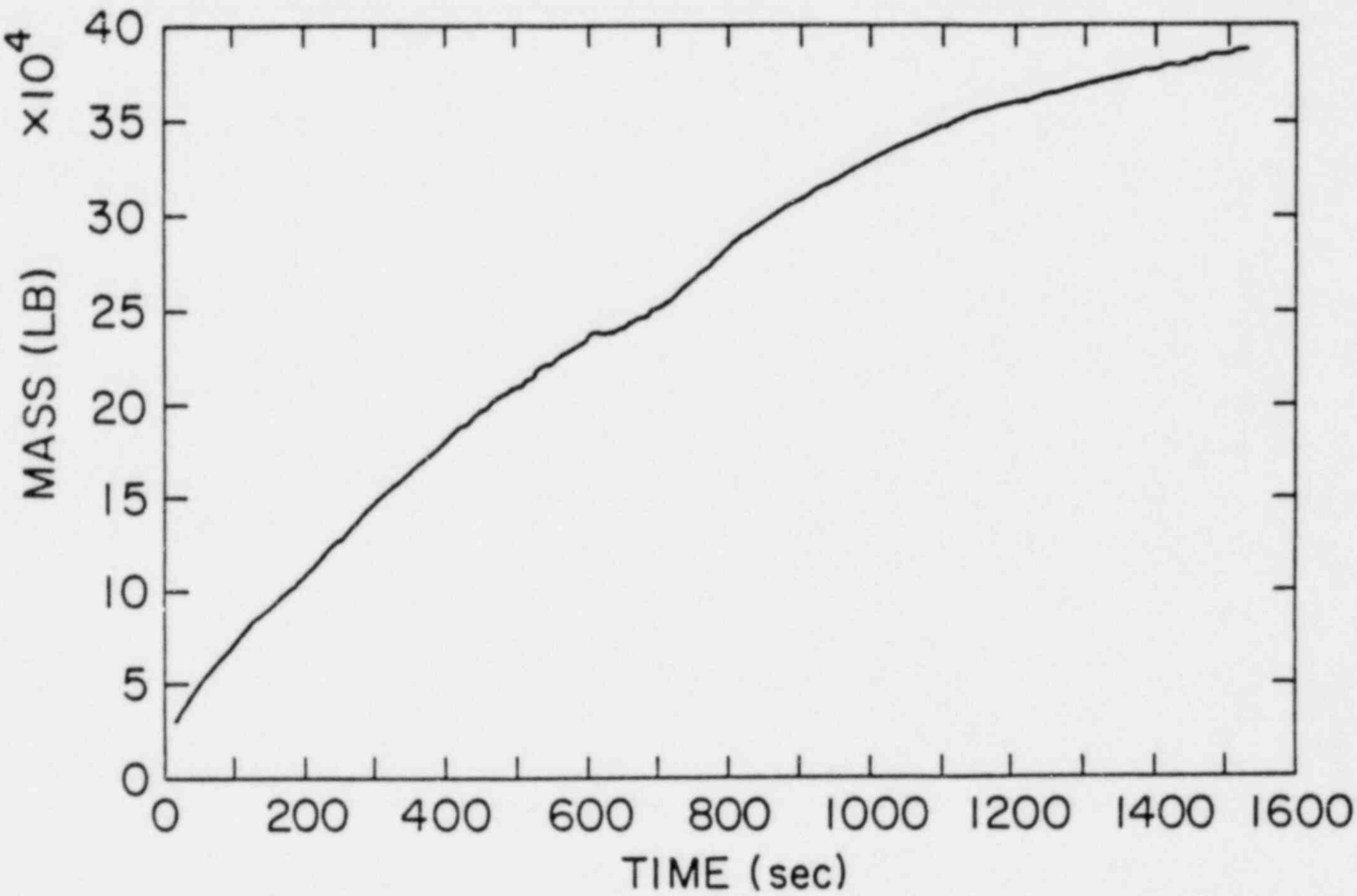


Figure 22

Pressure Suppression Pool Temperature

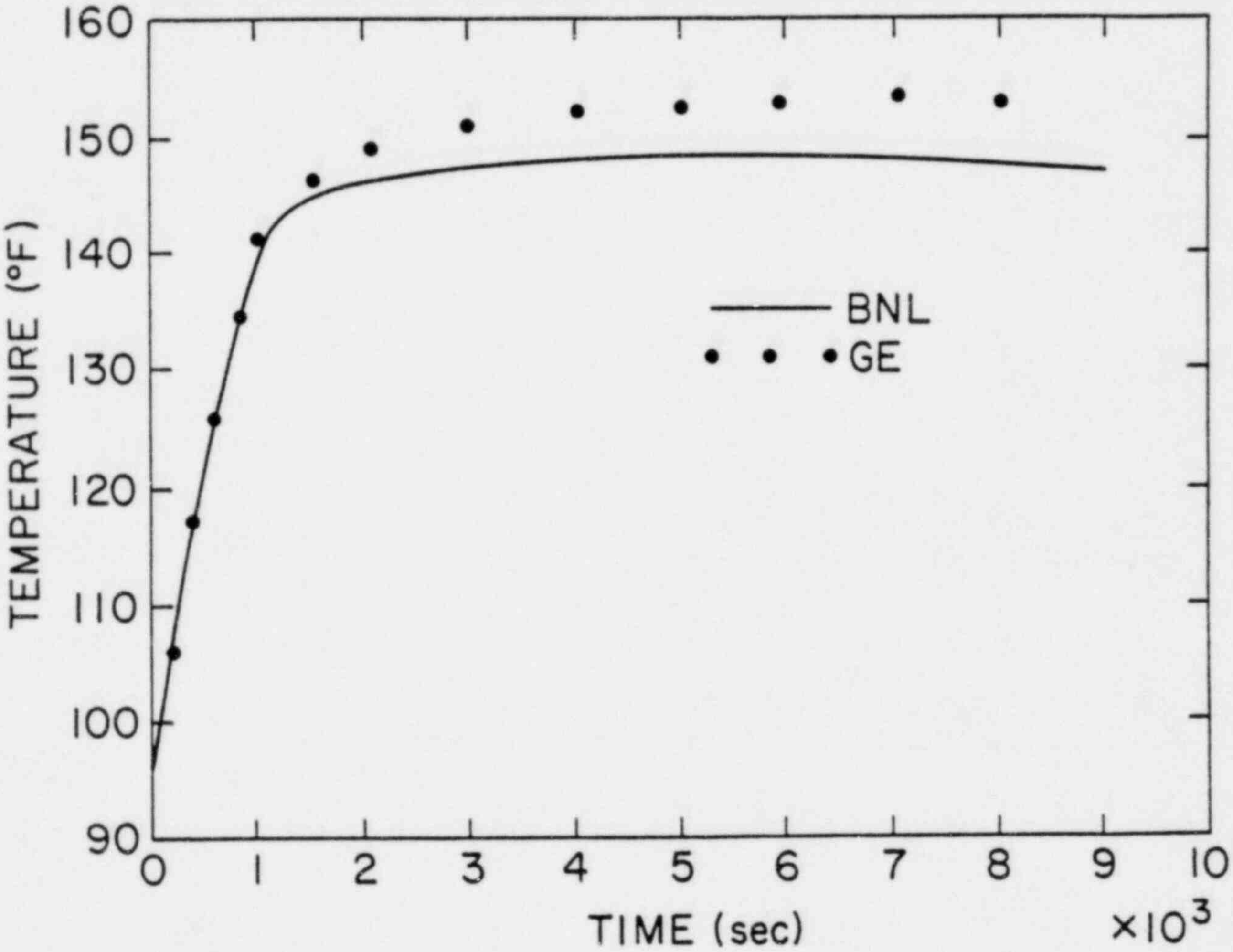


Figure 23

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