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Cofrentes NPP (BWR/6) ATWS (MSIVC) Analysis with TRAC-BF1

1D vs. Point Kinetics and Containment Response

Prepared by
A. Escrivá, M.D. Bovea, G. Verdú, J. L. Muñoz Cobo, UPV,
F. Castrillo, ID

Universidad Politecnica de Valencia (UPV)
Departmento de Ingenieria Quimica y Nuclear
Camino de Vera 14
46021 VALENCIA
SPAIN

IBRDROLA (ID)
Hermosilla, 3
28001 MADRID
SPAIN

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Abstract

Anticipated transient without scram (ATWS) are considered as events which might produce a severe accident in a boiling water reactor (BWR). It has been selected the most unfavourable of the different scenarios that could lead to an ATWS accident: an inadvertent closure of the main steam isolation valves (MSIVs).

To mitigate this accident, a borated water solution is injected. This action is an alternative way to shutdown the reactor quickly and effectively. This event has been analyzed using the TRAC-BF1 computer program.

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1 Introduction

The evaluation of events which might lead to severe damage in nuclear reactors, has been object of study recently [1], [2], [3]. Events, in which enough failures are assumed, such that the reactor is not shutdown, are broadly known as anticipated transients without scram (ATWS). It has been selected the most unfavorable of the different scenarios that could lead to anticipated transient without scram: an inadvertent closure of the main steam isolation valves (MSIVs).

For a boiling water reactor (BWR), a large amount of hot steam generated in the reactor is discharged into the containment pressure suppression pool through the safety-relief valves (SRVs). Temperature of the suppression pool increases and, as a result, the containment may eventually be heated and pressurized.

An increase in the vessel pressure takes place, this causes the activation of the feed-water pump runback and then, the water level in the vessel decreases quickly. To avoid this situation two actions are carried out: the recirculation pumps trips and the coolant injection through the high pressure safety systems. Water level control accomplished by a recirculation pump trip and a proper safety coolant injection can maintain the water level at top of active fuel (TAF), introducing negative void reactivity and thus, reducing the power production.

Since these actions are not enough to mitigate the ATWS, at 300 seconds from the transient beginning, a borated water solution is injected through the HPCS system line. Boron injection at a sufficiently large rate provides an alternative way to quickly and effectively shutdown the reactor. The manually actuated standby boron liquid control (SBLC) system, injects borated water to reduce reactor power. SBLC system contributes to maintain the reactor subcritical. In case of having a severe accident, the failure of the borated solution injection must also be included.

The TRAC-BF1 computer program [4] has been used to perform the ATWS event; the plant model used in this analysis corresponds to Cofrentes Nuclear Power Plant. The scenario assumes that after an MSIV closure an ATWS is initiated, the high pressure safety systems are activated and at 300 seconds from the transient, a borated water solution is injected. This event has been analyzed considering point kinetics and 1D nodal model.

The purpose of this study is the comparison of TRAC-BF1 results for this type of transient, when using point or 1D kinetics. The introduction of an specific model for the containment is also a contribution of this study.

In next sections, we will describe briefly the plant model of Cofrentes N.P.P., the initial conditions, and the characteristics of the analyzed transient, we will also present the results and the main conclusions obtained from the simulation.

2 Plant Model

Cofrentes N.P.P. is a General Electric designed BWR/6 MARK-III plant, with a licensed core thermal power of 3015 MWt. The initial plant model used in this analysis, is Cofrentes N.P.P. developed for TRAC-BF1, to simulate the transient originated by a manual turbine trip, [6] and [7].

The main characteristics of the model (see Figure 1), include a 4 ring-11 vessel levels, two recirculation loops, with two centrifugal pumps, with a flow control valve (FCV) in each loop. Feedwater is supplied by two pumps. One representative steam line supply the main turbine with the steam generated in the reactor. This line is equipped with isolation valves and 16 safety-relief valves. The core consists of 624 fuel elements distributed in three parallel channels, what allows to have a 3-dimensional model.

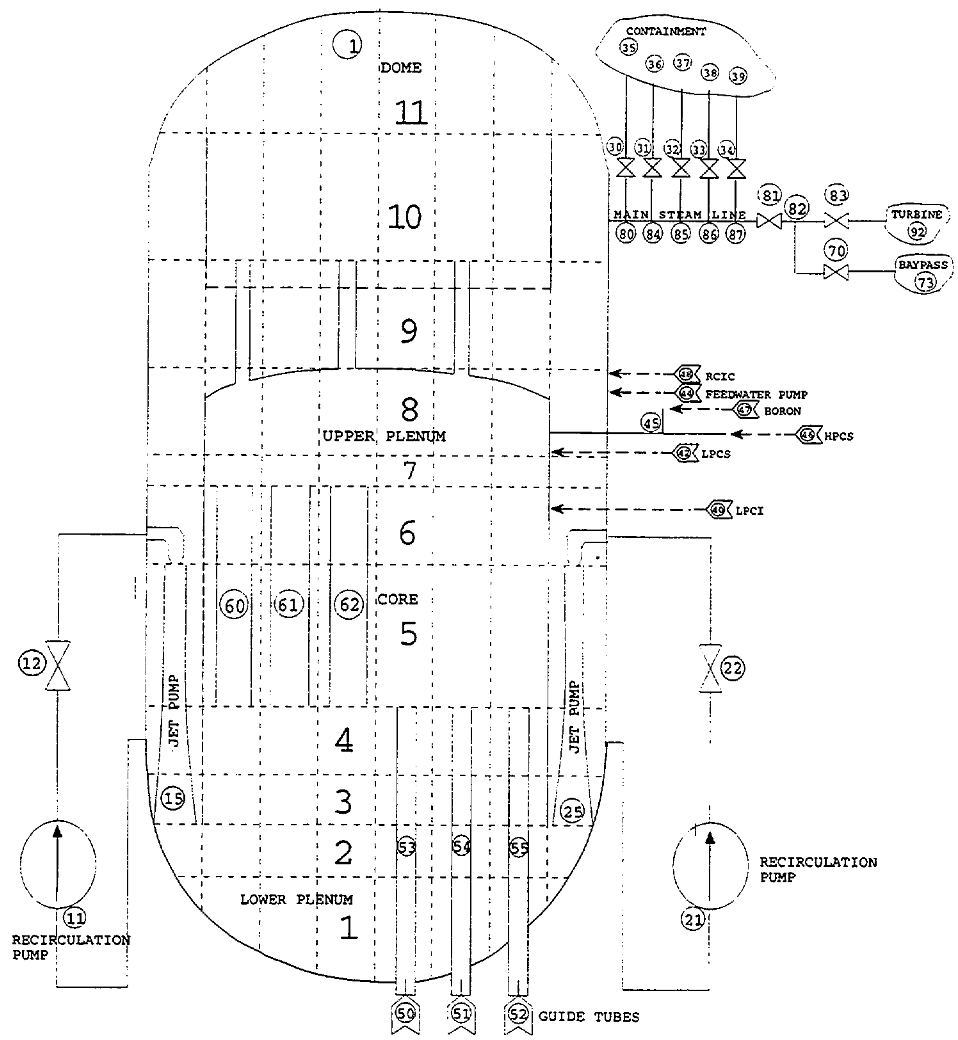

Figure 1: Model of Cofrentes N.P.P.

The steady state conditions for the ATWS analyzed are: core power 3015 MW (104.2 %) and core flow of.8418 kg/s (79 %), which is the minimum flow at licensed power.

In order to obtain an appropriate plant model for the ATWS conditions, the following changes and improvements to the reference model have been performed:

- First, the CHAN component has been replaced by another one, that models the fuel elements corresponding to a full load of GE-11 fuel for Cofrentes N.P.P..
- The contain MARK III model has been connected to the vessel model. In this way, the steam generated in the reactor is discharged into the suppression pool through the safety and/or the relief valves (SRVs).
- The relief valves have been designed as independent valves checked by the control system, instead of a multiple bank used in the reference model.
- The emergency systems have been modeled and connected to the vessel and to the containment.
- A boron model has been included.

All the above mentioned changes will be explained with more detail in the following sections.

2.1 Core Model

The core model considered for Cofrentes N.P.P. corresponds to a full load of GE-11 fuel design. The 624 fuel elements of the core have been distributed in three radial regions: 44 central, 504 middle and 76 peripheral elements. The power radial distribution considered for each of the channels has been the corresponding to a 1.4 peak factor for the fuel elements of the central region, 1.0 for the middle ones and the rest (0.768) for the peripheral ones.

Each region has been modeled with a CHAN component, using 29 axial cells (see Figure 2). The upper and lower core cells belong to the inactive zone and in order to make consistent our calculations with model used in the 3D simulator [8], we will suppose that the active zone is divided into 25 cells of 0.1524 m long, and two cells corresponding to the reflector with the same length. The losses of the spacers have been associated to the shape losses of the closest section.

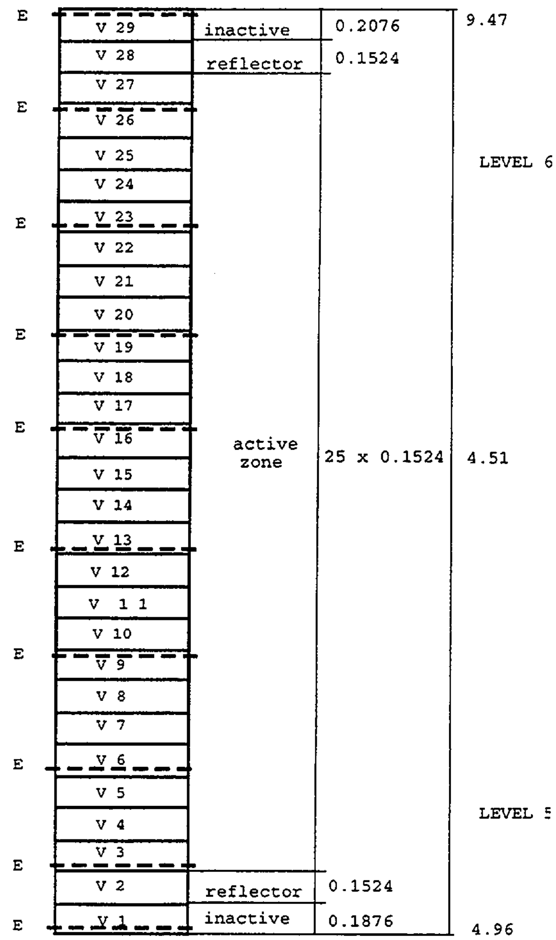

Figure 2: Core Model

To obtain the core characteristics we have taken as a reference the core model of the 9th cycle of Cofrentes N.P.P., [9].

2.2 Safety-Relief Valves

The safety-relief valves modeled in the reference model, were designed with a VALVE component using the option of multiple bank safety relief with ADS trip. Each valve group has the following characteristics:

Group	No. Valves	Area(m^2)
1	1	1.4276E-2
2	1	1.4276E-2
3	3	4.2828E-2
4	4	5.7104E-2
5	7	9.9932E-2

Table 1: Characteristics of the SRVs.

The connection of the multiple bank to the plant model was performed by means of a TEE component to the main steam line, and with a BREAK component to the containment.

In the multiple bank valve option, each valves group opens and closes independently, based on its own opening and closing pressure setpoints. This mechanism is correct if we only have normal setpoints. But for the ATWS analysis, it is necessary to include the low-low setpoints, for groups 1 and 2. These setpoints allow to increase the amount of discharged steam, after the first opening and closing. But the low-low setpoints cannot be included in the multiple bank model. Table 2, shows the normal and low-low setpoints.

	Normal Setpoints		Low-Low Setpoints	
	P_{op1} (MPa)	P_c (MPa)	P_{op2} (MPa)	P_c (MPa)
Group 1	7.9835	7.293	7.5	6.7629
Group 2	8.0525	7.362	7.7766	6.8319
Group 3	8.1214	7.362		6.9
Group 4	8.1214	7.362		
Group 5	8.1214	7.431		

Table 2: Setpoints of the SRVs (Relief Function).

So, the SRV model is not an adequate one for the ATWS analysis, since the pressure drop between the dome and the valve is important for the valves opening, specially in the first seconds of the transient. To solve these two problems it has been necessary to design the SRVs as five independent valves, where each one represents one valve bank group, as it is shown in Figure 3.

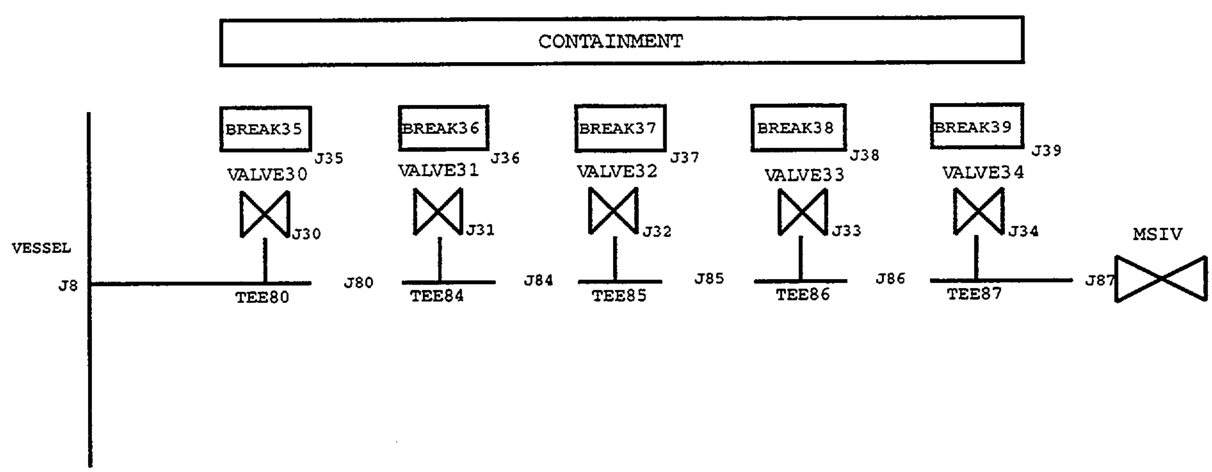

Figure 3: SRVs Model.

The valve function is indicated by the control system. Thus, control blocks have been designed in order to be able us to reach the normal and low-low setpoints. Figure 4, shows the control blocks used by groups 1 and 2, which need normal setpoints (P_{op1}, P_c) and low-low setpoints (P_{op2}, P_c).

This allows to extract higher steam flow from the vessel, since the setpoints of the low-low logic are more relaxed than those of the normal one.

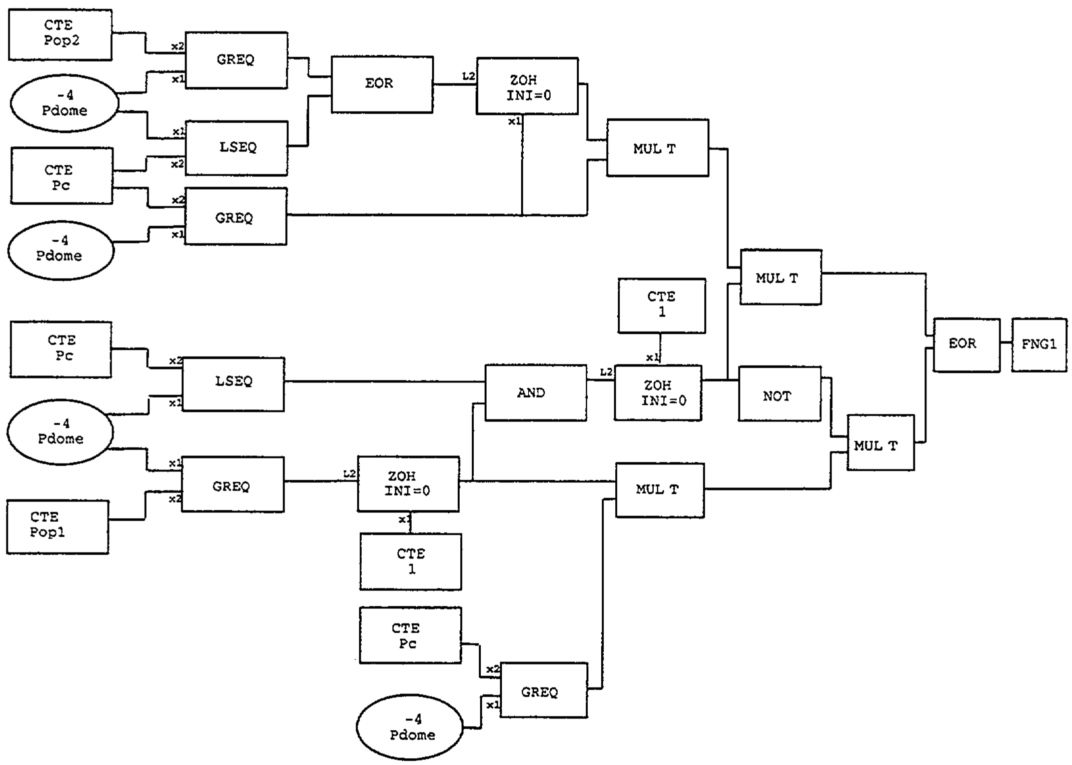

Figure 4: Control Block for Valve Groups 1 and 2.

Figure 5 shows the control blocks for groups 3 and 4, which only use normal setpoints (P_{op}, P_c); that is, the opening pressure remains constant during the successive cycles.

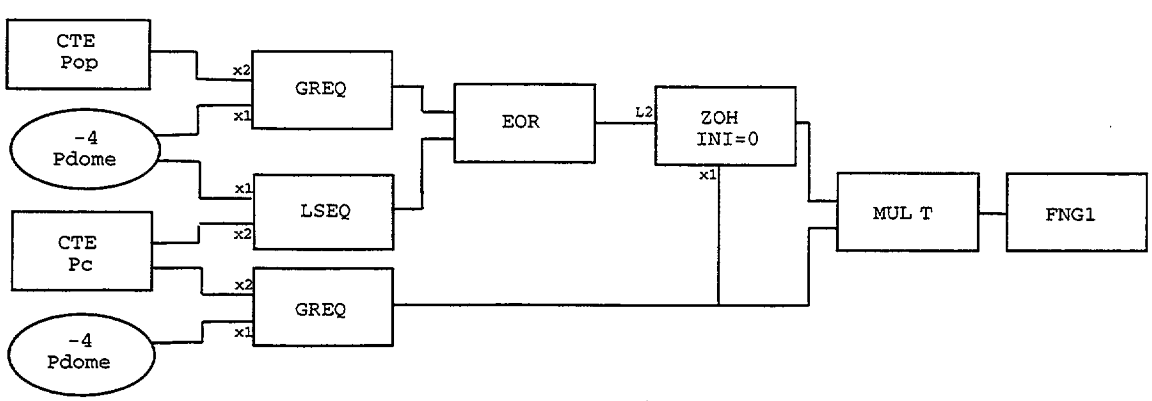

Figure 5: Control Block for Valve Groups 3 and 4.

Finally, Figure 6 shows the operation of the group 5 relief valves. The cycles are regulated by the normal logic, but besides the relief function, this group must include the possibility of core depressurization (ADS), activated by a level trip (Very Very Low Level, L1). So, the implementation of this function differences them from the latter group. The ADS system function needs the opening of seven of the relief valves, what is equivalent to the opening of the modeled group of valves. Once the L1 level trip is activated, this group opens completely, allowing the release of the generated steam and therefore, the pressure in the vessels decreases quickly.

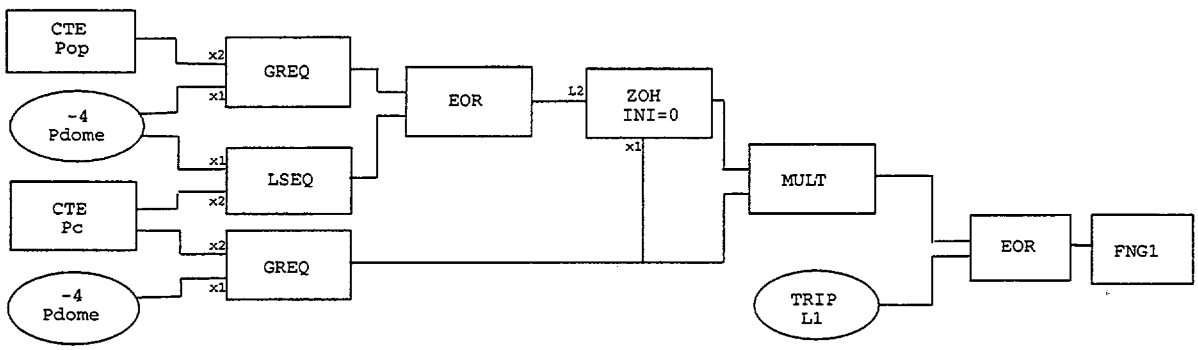

Figure 6: Control Block for Valve Group 5.

2.3 Main Steam Isolation Valve

The ATWS analyzed is initiated by the closure of the main steam isolation valve.

Closure of the MSIV, begins at time zero and the reactor scram fails. The valve is completely closed in three seconds, according to the following time-area table.

Time	Area
0.0	1.0
0.6	1.0
1.7	0.01
3.0	0.0

Table 3: Closure of the MSIV.

2.4 Reactor Emergency Systems

The modeled emergency systems are classified in:

- high pressure safety systems
  - high pressure core spray, HPCS
  - reactor core isolation cooling, RCIC

- low pressure safety systems
  - low pressure coolant injection, LPCI
  - low pressure coolant spray, LPCS

Each of these systems has been modeled by means of a FILL component, and they inject coolant in order to balance the loss of water level in the vessel. The injection flow can be taken from the condensate tank or from the suppression pool, [3]. But according to the increase of the water level in the suppression pool, let us consider that in the analyzed case, the injection flow is taken from the pool, [2].

2.5 High Pressure Safety Systems

The HPCS system injects coolant in the upper plenum, corresponding to the level 8 and to the radial region 3 of the plant model. The RCIC system injects coolant through the same line as the feedwater.

When the water level in the downcomer falls below the L2 level (Very Low Level), these systems are activated with a delay of 37 seconds. The stop signal is the L8 level (High Level). In order to simulate this operation, these systems are controlled by the TRAC control system.

HPCS and RCIC suction water is taken from the suppression pool. But the suppression pool temperature increases as a result of the amount of steam discharged through the SRVs. The HPCS and RCIC pumps cannot operate under an elevated suppression pool temperature. They fail when temperature reaches a fixed value. HPCS system is also used to allow the boron injection.

The injected flow by the HPCS system is a function of core pressure, decreasing when the core pressure increases, and vice versa, as it can be seen in Table 4. The RCIC flow is constant with the core pressure, and its value is 0.03785 m^3/sec.

Pressure (MPa)	Flow (m^3/sec)
1.3792	0.056775
7.9097	0.079485
8.1166	0.2781975

Table 4: Pressure vs. Flow for HPCS.

An additional function of the HPCS system is to allow the injection of a borated water solution through its line. This has been modeled using a TEE component (see Figure 7), that allows the coolant injection through one of its branches and the borated water through the other one.

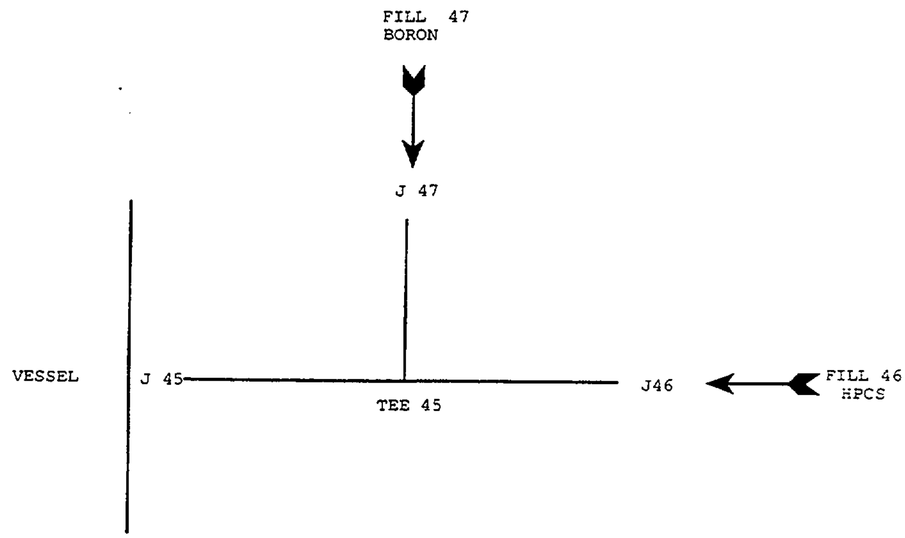

Figure 7: HPCS and Boron Injection Model.

2.6 Low Pressure Safety Systems

LPCI and LPCS systems inject coolant in the radial region 3 of the vessel, but the former injects in the axial level 6 and the latter in the axial level 8.

When the vessel level decreases below level L1 (Very Very Low Level), these systems are activated with a delay of 70 seconds, and they stop working when the vessel level exceeds the L8 level (High Level). This operation mode has been modeled using the TRAC control system.

Provided that the injection pressures are 1,653 MPa for the LPCI and 1.928 MPa for the LPCS, a previous depressurization is not produced and therefore, low pressure safety systems are activated, but coolant is not injected because vessel pressure is never lower than the injection pressure.

The injected flow depends on the pressure, according to Tables 5 and 6.

Pressure (MPa)	Flow (m^3/sec)
0.1013	0.362705
0.5150	0.3153951
0.825275	0.2838553
1.0666	0.2523165
1.27345	0.2207767
1.445825	0.1892368
1.583725	0.157697
1.68715	0.1261582
1.790575	0.0946184
1.82505	0.0630786
1.929	0.0

Table 5: Pressure vs. Flow for LPCS.

Pressure (MPa)	Flow (m^3/sec)
0.1013	0.8044208
0.273675	0.7571019
0.58395	0.6624643
0.894225	0.5678264
1.101075	0.4731887
1.307925	0.378551
1.445825	0.2839132
1.54925	0.1892755
1.653	0.0

Table 6: Pressure vs. Flow for LPCI.

2.7 Containment Model

Another improvement incorporated to the plant model, has been the connection to the vessel, of a MARK III containment model developed by UPV and Iberdrola, [10].

The calculation with TRAC-BF1 is made using the containment component CONTAN, which actually is an independent subprogram based on the CONTEMPT-LT program.

A compartment simulates a volume or room within the containment. composed of a vapor and a liquid region. It is assumed pressure equilibrium, but not temperature equilibrium between the two regions.

The containment (CONTAN) component calculates the time variation of compartment pressures, temperatures in the liquid pool region and in the vapor atmosphere region above the pool, mass and energy inventories, heat structure temperature distributions, and energy exchange with adjacent compartments.

Each compartment may have energy transfer between the pool and vapor regions for the following models:

- Pool boiling, it is an instantaneous mass transfer.

- Evaporation, condensation. During a time step the code does not permit both models. This model is time dependent.

Saturation conditions are presumed at the interface, and the heat transfer between the pool and the vapor region is equal to the heat transfer between the interface and the bulk vapor mixture. The compartments can be communicated using the 'Passive flow junction' component, which simulates pressure-induced convective flow between two compartments. There are three types:

- Single-phase gas flow between the vapor regions of two compartments. Flow may occur in either direction.

- Single-phase gas flow in one direction only between the vapor regions of two compartments.

- Single-phase gas flow in one direction only between the vapor region of the donor compartment and the liquid region of the receiver compartment, when the difference of pressure between compartments reaches a prescribed value.

The BREAK and FILL components may be coupled to the containment as an user option, specifying the containment compartment to which they are connected. Also the VESSEL may be thermally coupled to the containment as another option of the user. By specifying which compartment represents the drywell and the position of the vessel above the drywell floor.

As it has been indicated above, the steam is discharged through the SRVs into the suppression pool, and accumulated into the wetwell, (see Figure 8).

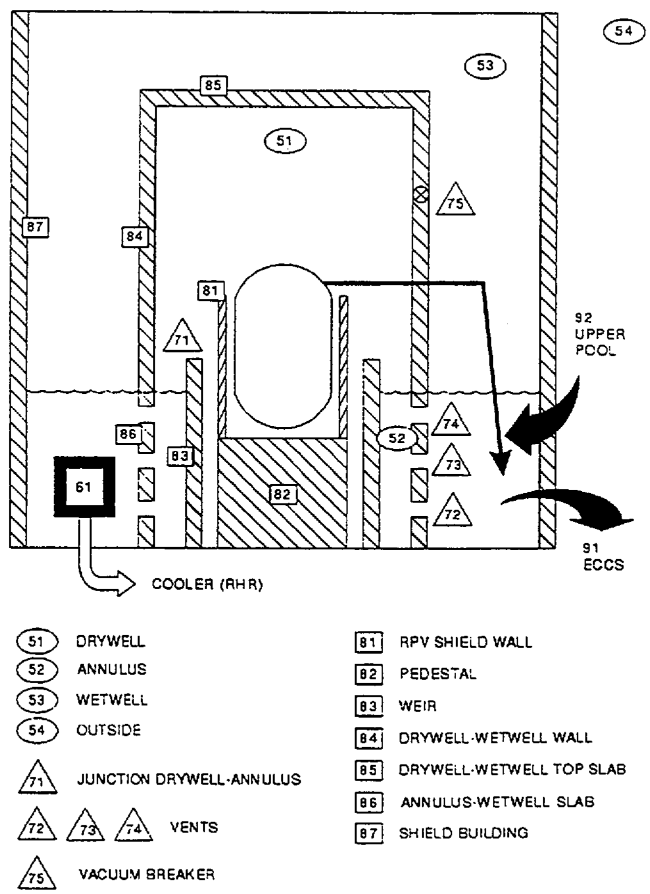

Figure 8: Containment Model.

The main elements in the containment model are: compartments, heat structures, RHR heat exchanger, core emergency system (ECCS), vacuum breaker valves and upper pool.

The steam extracted from the vessel through the SRVs is discharged into the suppression pool. Its initial temperature is 305.36 K, and its initial volume is 3144.59 m^3. Due to the amount of steam discharged, the temperature and water level in the suppression pool increases, and then, a heatup and a pressurization in the containment may occur. But if the water temperature in the suppression pool is over a fixed temperature, the HPCS and RCIC pumps may not work properly. To avoid this problem, the RHR system removes the heat from the suppression pool, working in a cooling mode. Furthermore, if the heat capacity temperature limit (HCTL) for the suppression pool is reached, a depressurization through ADS system must be activated.

2.8 Boron Model

The boron injection from the SBLC provides an alternative method to shutdown the reactor, independent of the control rod system.

The SBLC is manually actuated at 300 seconds from the beginning of the transient, and the borated water is injected through the HPCS system line (as Figure 7 shows), producing a fast reduction of the reactor power. The stop signal is the same for the HPCS one; that is, when the water level in the vessel reaches the high level (L8). The injected flow is constant: 5,44 l/s with a boron concentration of 21698 ppm.

The perturbations due to the incorporation of boron in the system, must be included in the model. So, the absorption cross sections must be modified.

The modification carried out for the point kinetics model has been different from the one done for the 1D model. For the point kinetics model, boron reactivity coefficients has to be calculated following the methodology exposed in [11].

The new cross sections corresponding to boron injection in 1D model, are obtained by adding to the initial ones a new term according to Table 7. This table shows these additive terms for different void concentrations and for each neutron group, [12], [13]. It can be demonstrated that the value of those additive terms does not depend on the burn-up.

Voids %	Group 1	Group 2
0	3.836E-7	1.247E-5
40	2.875E-7	9.667E-6
70	2.160E-7	7.681E-5

Table 7: Additive terms.

3 Initial Conditions

The initial conditions for the starting steady state for the ATWS analysis, are showed in the following table, [14].

Core Thermal Power (MW/%)	3015./104.2
Core Flow (Kg/s/%)	8418.5
By-pass Core Flow	16%
Dome Pressure (MPa)	7.378
Feedwater Flow Rate (Kg/s)	1648
Feedwater Enthalpy (KJ/Kg)	933.44
F.W. Runback Pressure (MPa)	8.0
F.W. Flow Stopping Time (s)	40
Steam Flow Rate (Kg/s)	1648
Recirculation Flow Rate (Kg/s)	1224

Table 8: Initial Conditions for ATWS Analysis.

To adjust the steady state situation with TRAC-BF1 is a hard task to perform, due to the amount of variables and parameters to control.

Once the new stationary situation has been obtained, a null transient from the steady state reached, is run to verify stability of the steady state conditions.

4 Transient

Once the steady state with the initial conditions needed to the ATWS analysis has been achieved, we will study the transient evolution with the TRAC-BF1 code.

The transient starts with the closure of the MSIV at 0 seconds. During the first seconds, the steam flow arriving to the turbine is decreasing, causing the first feedwater pump trip because low flow signal in the main steam line, 549 kg/s. Then, the feedwater pump runback is produced, because high pressure in the vessel. When the dome pressure reaches 8 MP, this trip is activated, with a delay of 40 seconds.

The water level in the downcomer decreases. When this level reaches L4, the recirculation valve runback is activated and it keeps on a fixed opening value of 12.42%.

As a result of the closure of the MSIV, the large amount of generated steam in the reactor, must be discharged into the suppression pool, through the SRVs. When the dome pressure reaches the setpoints corresponding to each valve group, they start opening and closing, taking into account the normal setpoints, and once they have closed, the low-low setpoints are considered. As the suppression pool temperature increases, heat dumped is removed by the operation of the RHR system, in the suppression pool cooling mode.

No credit is taken in this calculation to the recirculation pump trip on high pressure.

The water level in the vessel keeps on decreasing until the low level (L3) is reached. This level activates the recirculation pumps low speed transfer, changing their nominal speed from 155 rad/s to 38.87 rad/s.

The downcomer water level continues decreasing until L2. The recirculation pump trip is produced with a delay of 2 seconds when the L2 level is reached. At the same time, but with a delay of 37 seconds, the coolant injection by the HPCS and RCIC systems are activated. These measures introduce negative void reactivity and, thus, reduce power production.

Despite of these actions, the level reaches the L1 signal. From this situation, the low pressure safety systems actuates with a delay of 70 seconds. But for the coolant injection to be made through the LPCI and LPCS systems it is necessary a previous core depressurization which is not produced in this analysis. So, these systems are activated but there is not coolant injection.

To mitigate the transient consequences, the borated water injection is activated by means of the SBLC system, through the HPCS line. This action is carried out, manually at 300 seconds, and causes a fast drop of the reactor power, decreasing dome pressure below setpoints SRVs. So, these valves close, and do not discharge more steam into the suppression pool, and its temperature is stabilized below the heat capacity temperature limit (HCTL). The boron concentration in the reactor is linearly proportional with time, and it is uniformly distributed in the core. The analysis finishes studying the reactor evolution until the downcomer level reaches the L8 high level signal. This transient has been analyzed using point kinetics and 1D nodal model.

Events	Point Kinetics	1D Kinetics
MSIV Closure	0.0	0.0
Water Level L4	1.520	1.524
1st Feedwater Pump Trip	1.628	1.628
SRV (Group 1) Opening	2.146	2.149
Activation of the Feedwater Pump Runback	2.167	2.170
SRV (Group 2, 3 and 4) Opening	2.244	2.247
SRV (Group 5) Opening	2.473	2.482
MSIV Fully Closed	3.0	3.0
Water Level L3	32.044	31.382
Low Velocity Transfer Recirculation Pumps	32.5	32.0
Feedwater Pump Runback	42.167	42.170
SRV (Group 5) Closure	72.5	70.5
Water Level L2	87.216	87.241
Recirculation Pumps Trip	89.216	89.241
Water Level L1	111.563	111.156
SRV (Group 4) Closure	115.0	119.0
HPCS and RCIC Injection	124.216	124.241
SRV (Group 3) Closure	142.5	140.5
LPCS and LPCI Injection	181.563	181.156
Boron Manual Injection	300.0	300.0
SRV (Group 2) Closure	319.5	332.0
SRV (Group 1) Closure	327.0	341.0
Water Level L8	881.08	908.59

Table 9: Sequence of Events.

4.1 Comparison between Point Kinetics and 1D Nodal Model

The described transient has been analyzed using point kinetics and the 1D nodal model of the TRAC code.

To calculate the kinetics parameters needed in the input file of the TRAC/BF1 code, data from the 3D simulator, used for core design and core following, have been used. Specifically, to obtain the point kinetics parameters the PAPU code [15] has been employed and for the 1D kinetics, the KINPAR methodology, [16], [17], has been applied.

From results showed in Table 9, it can be concluded that the temporal behaviour of the events using both kinetics match completely. However, if the main variables are studied, some differences between this two kinetic models are obtained. The most striking difference was the reactor power evolution since in the analyzed case using the point kinetics model the characteristic spurious peaks appear, but they do not have significant effects on the rest of the studied variables.

If the power profile is studied for different instants in the transient, it is observed that during the first 300 seconds, when the injection of borated water solution has not been produced, the profile is the same as the initial one.

However, for a time longer than 300 seconds, the power profile changes drastically. A deviation of the profile maxima values takes place, initially to the lower axial levels and later to the upper ones. This fact is due to the boron action which is concentrated in the higher cells during the $\approx$ 300-400 seconds range (during this period the void fraction is close to unity and therefore boron solution does not run along the channels). At $\approx$ 400 seconds, boron is delivered homogeneously along the channels and then, the maximum profile values move to upper levels of the core. But this fact is not meaningful because at this time the power reactor is close to residual power.

This fact explains the different behaviour of the point kinetics parameters, since it has been considered for the whole transient, the initial power profile. But the profile evolves in a very different way from the initial one along the time when the transient occurs.

Actually, point kinetics could be applied if it is divided the transient length in temporal intervals, each of them with its appropriate reactivity coefficients. But in practice this is not very easy and consequently a 1D kinetics is needed for this kind of transients.

4.2 Boron Evolution in the Core

This section is focused to study the evolution of the boron concentration along the different core channels: peripheral channels (CHAN60), medium channels (CHAN61) and central channels (CHAN62).

For each of this channels it has been represented the void fraction and the boron concentration along the time for the different core axial levels: level 3 and level 27, corresponding to the first and last active core cells, respectively, and to the levels 11 and 19, corresponding to two intermediate axial cells (see Figures 21 to 26).

The injection of a borated water solution is made manually at 300 seconds in the level 8 of the vessel and radial ring 3, corresponding to the HPCS system injection line. Boron behaves similarly along the three channels in which the core has been modeled.

At the beginning, the boron concentration rises slowly in the lower axial levels, while in the upper ones the concentration reaches very high levels and keep them constant approximately at $\approx$ 80-100 seconds (depending on the channel) as a function of the void fraction values. This behaviour is due, mainly, to the increasing impulse of the existing steam in the vessel levels where the borated water flow injection is produced.

The different behaviour in the boron concentration along the axial levels during the seconds following the injection is mainly due to the no distribution of the boron in the steam. Therefore, when there is a void fraction decrease in the upper levels and the void fraction takes a similar value along the channel, the boron concentration changes homogeneously.

The void fraction decrease in the upper axial level is produced by the slow dilution of the boron in the water, what contributes to reduce the thermal power, and consequently, diminishes the steam production.

Studying the behaviour in each channel during the firsts seconds, sharp oscillations in the boron concentration are observed. These oscillations coincide with the void fraction ones. In this way, when the void fraction decreases, the dilution of boron is favoured, diminishing almost instantaneously the boron concentration in the lower part of the channel.

This fact indicates clearly that boron is not distributed in the steam and therefore, until the void fraction is lower than 0.5, boron can not be distributed homogeneously.

5 Conclusions

The Main Steam Isolation Valves (MSIV) closure transient, with no automatic insertion of control rods (SCRAM), has been analyzed with the TRAC-BF1 code.

To model the core neutron dynamics, point-kinetics and 1D-kinetics options have been used, and the results for both options show no main differences for the overall behaviour of the system.

A model to calculate the response of the containment has been set up, and consistent results have been obtained for the evolution of those parameters that are of primary concern for this transient (suppression pool temperature).

After 150 seconds from the transient initiation, the reactor is isolated and with an almost constant power around 15 %. If no action is taken. the energy released by the SRVs will increase the suppression pool temperature at a constant rate. Therefore, it is necessary to activate the boron injection, which will suppress the core thermal power in around 50 seconds and further discharges to the pool are eliminated.

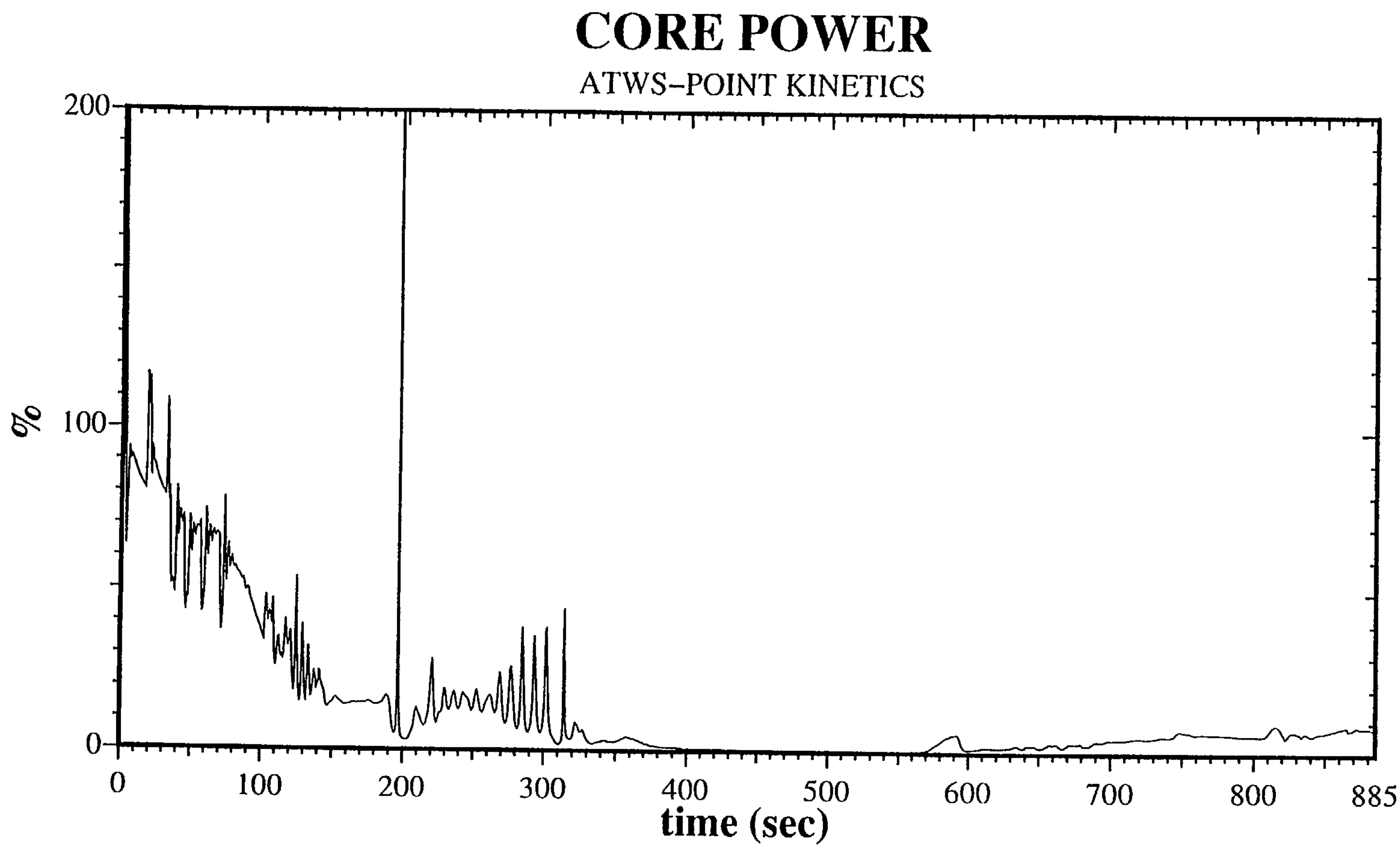

Figure 9: Core Power (Point Kinetics Model)

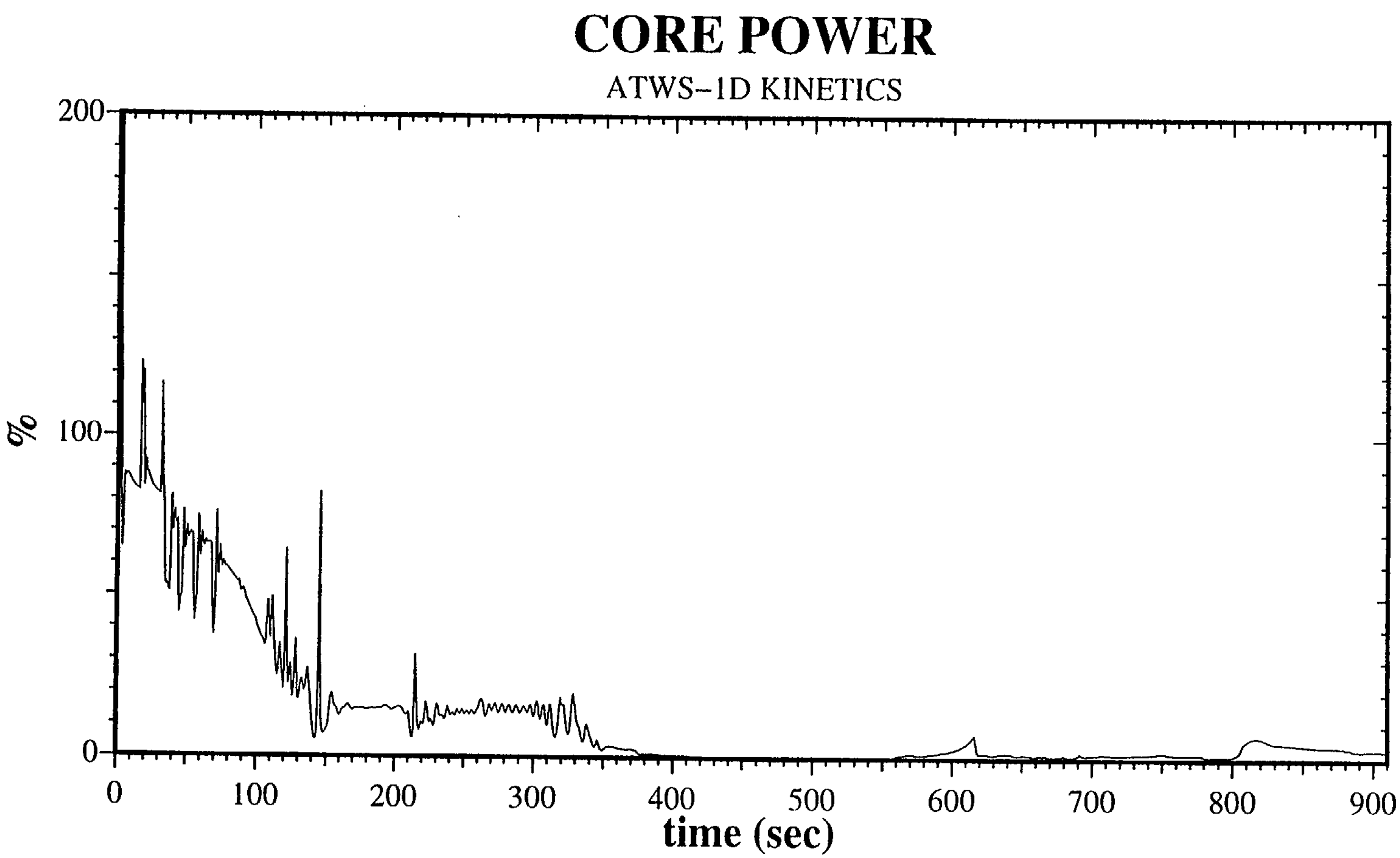

Figure 10: Core Power (1D Kinetics Model)

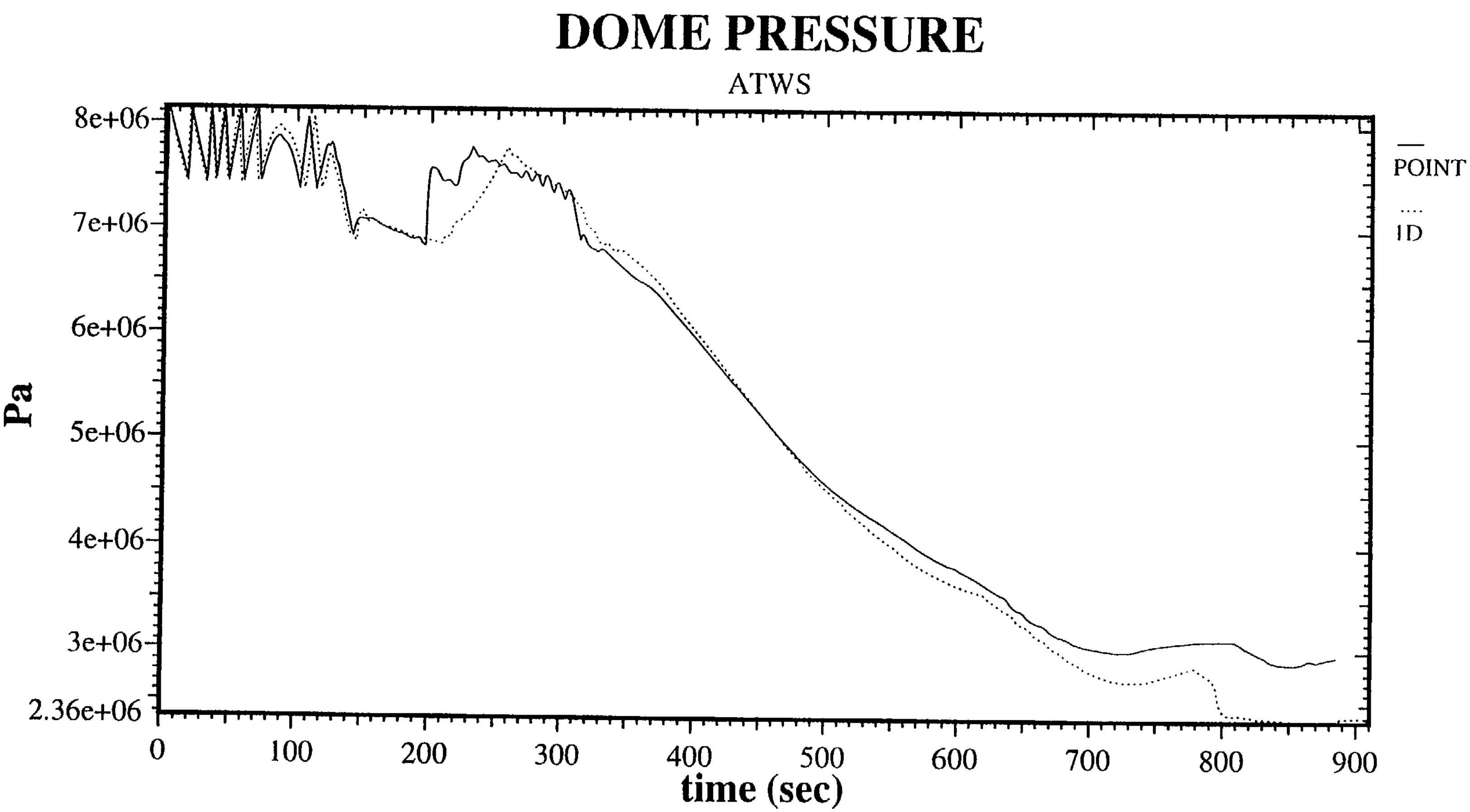

Figure 11: Dome Pressure

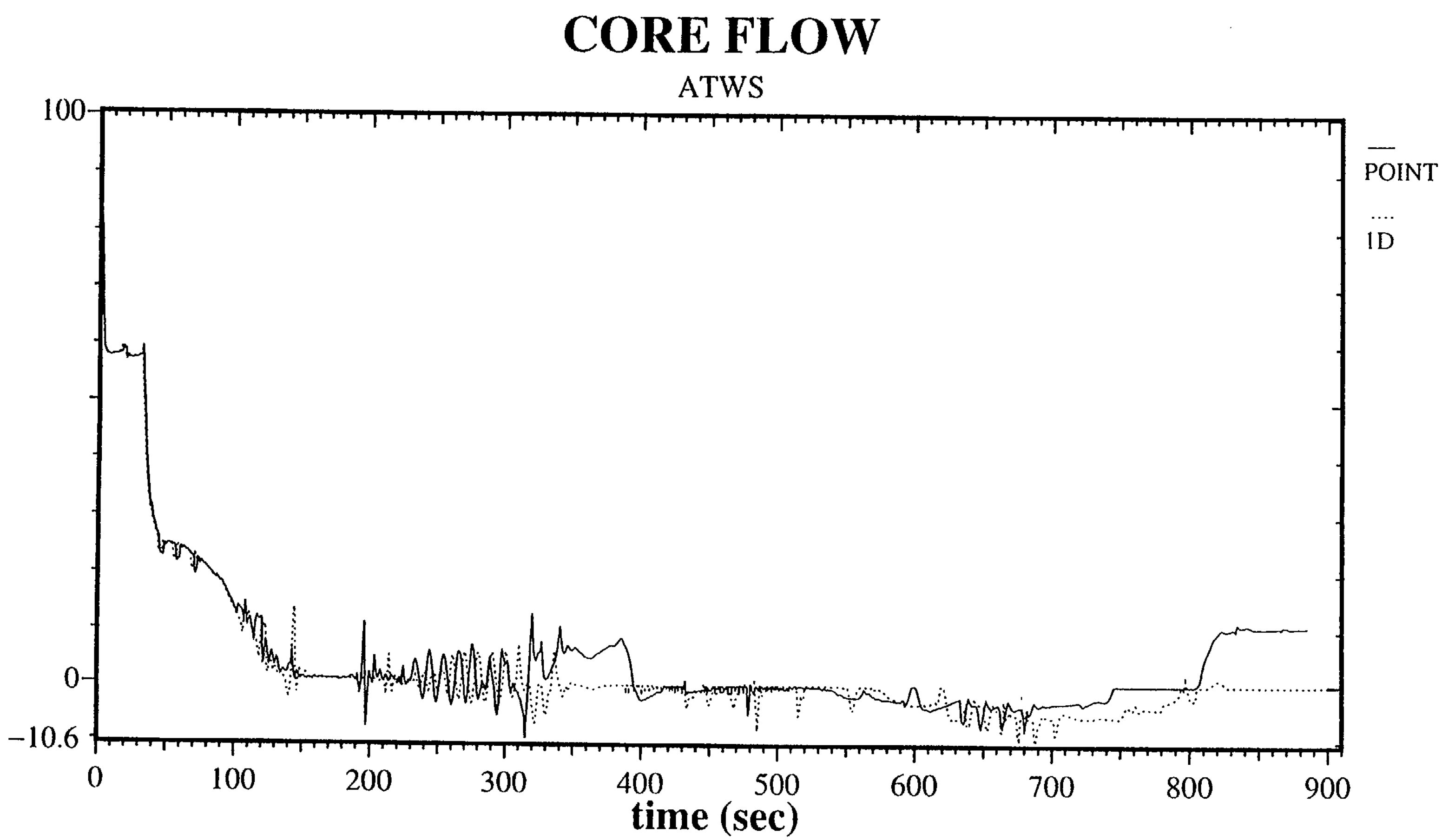

Figure 12: Core Flow

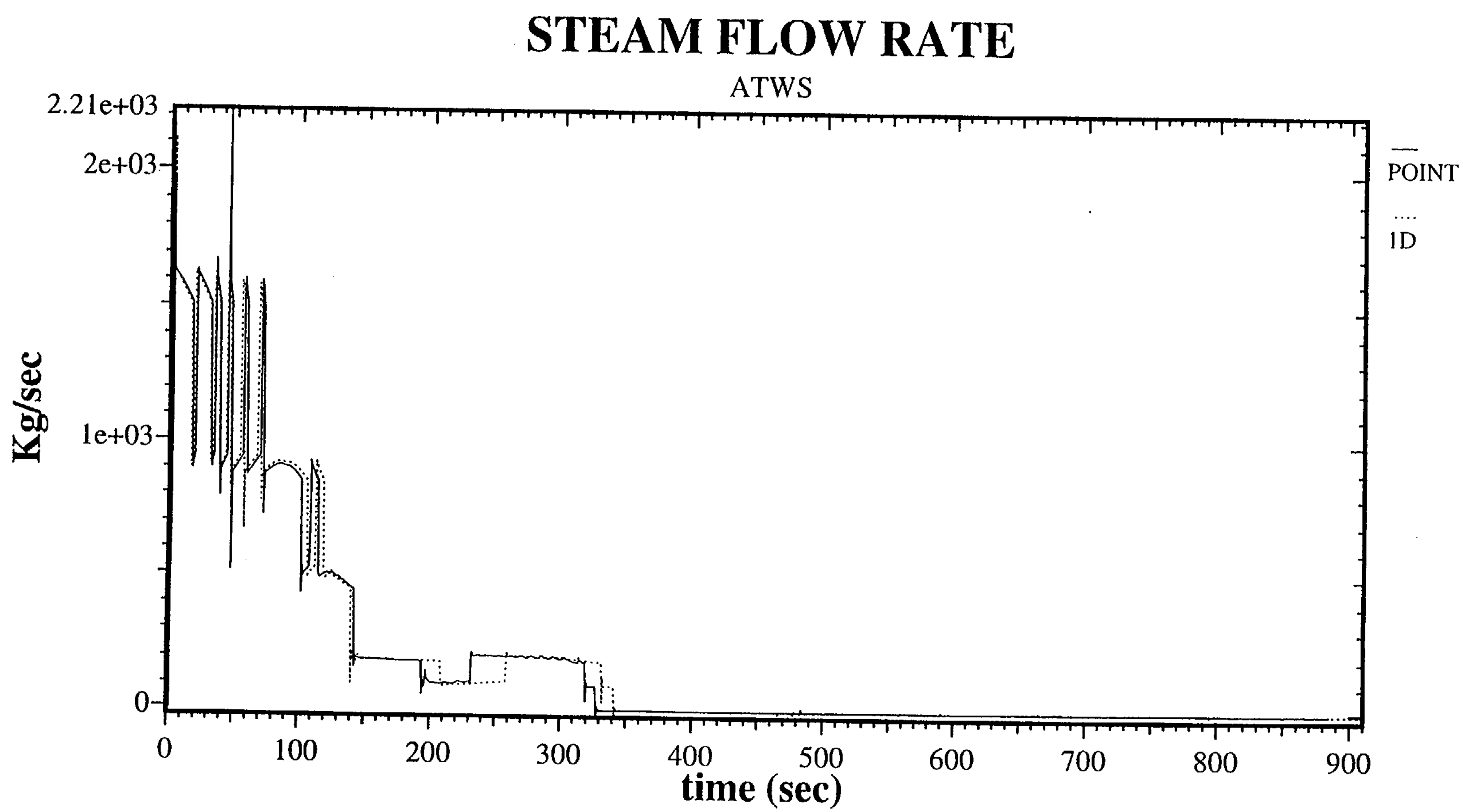

Figure 13: Steam Flow Rate

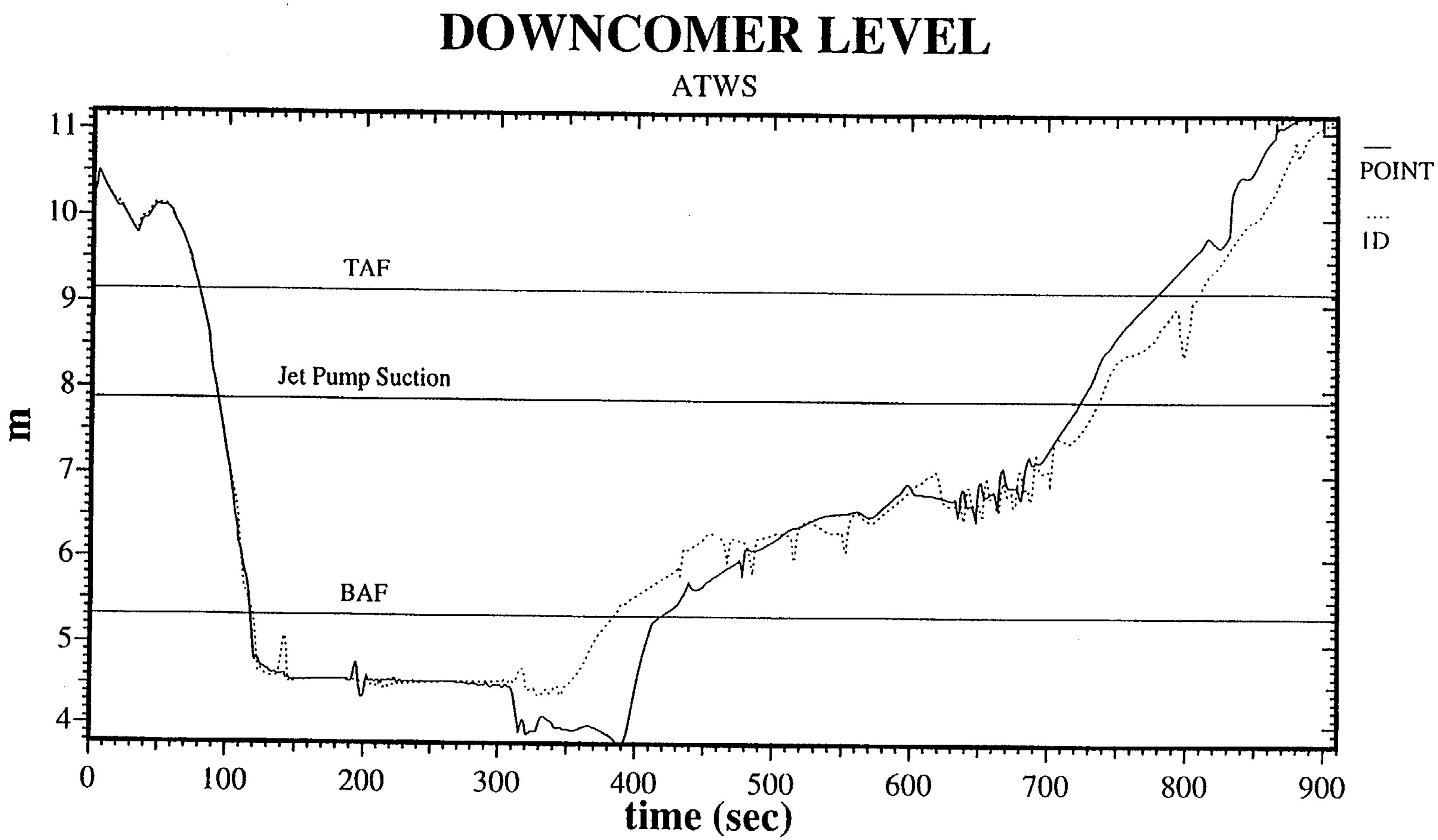

Figure 14: Downcomer Level

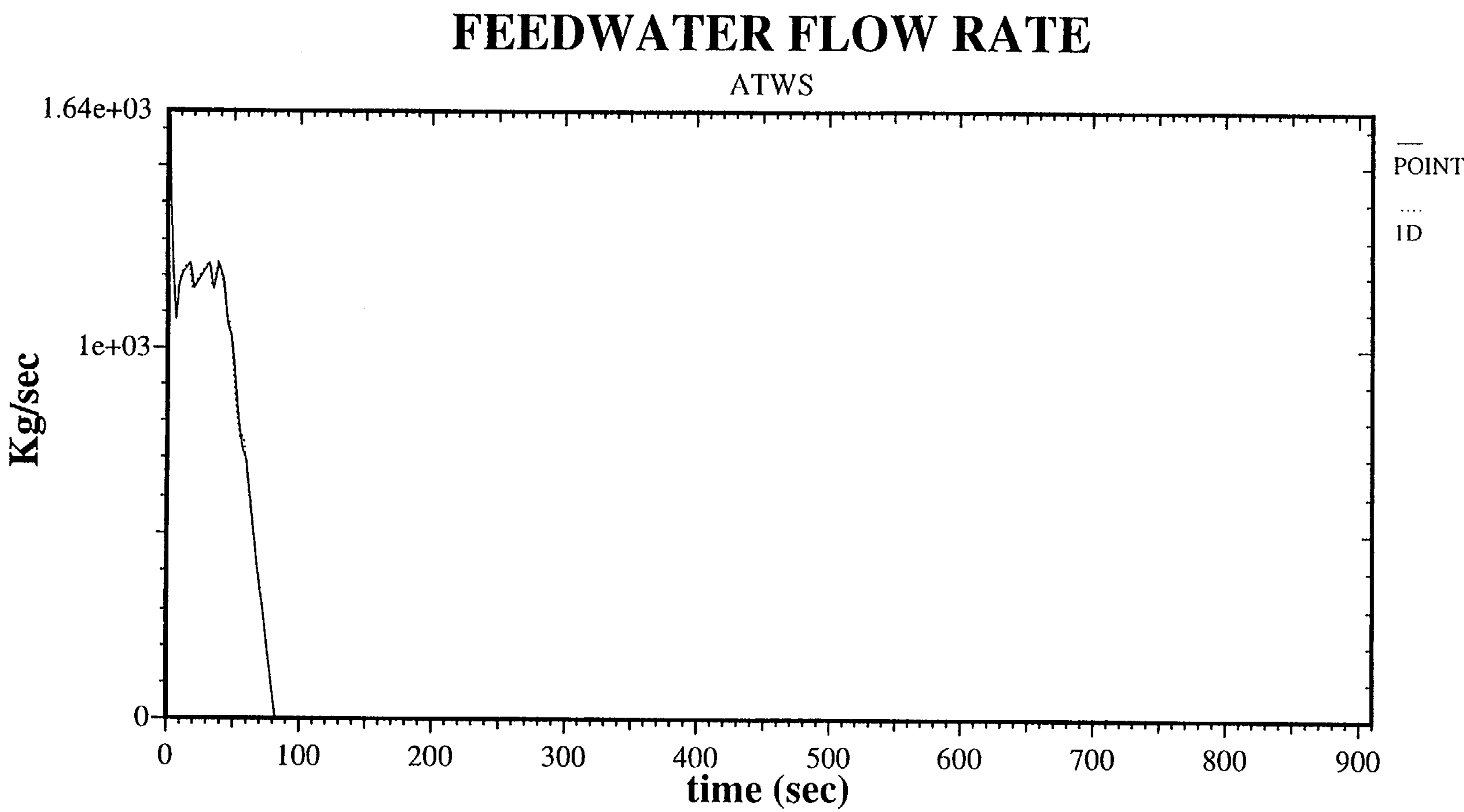

Figure 15: Feedwater Flow Rate

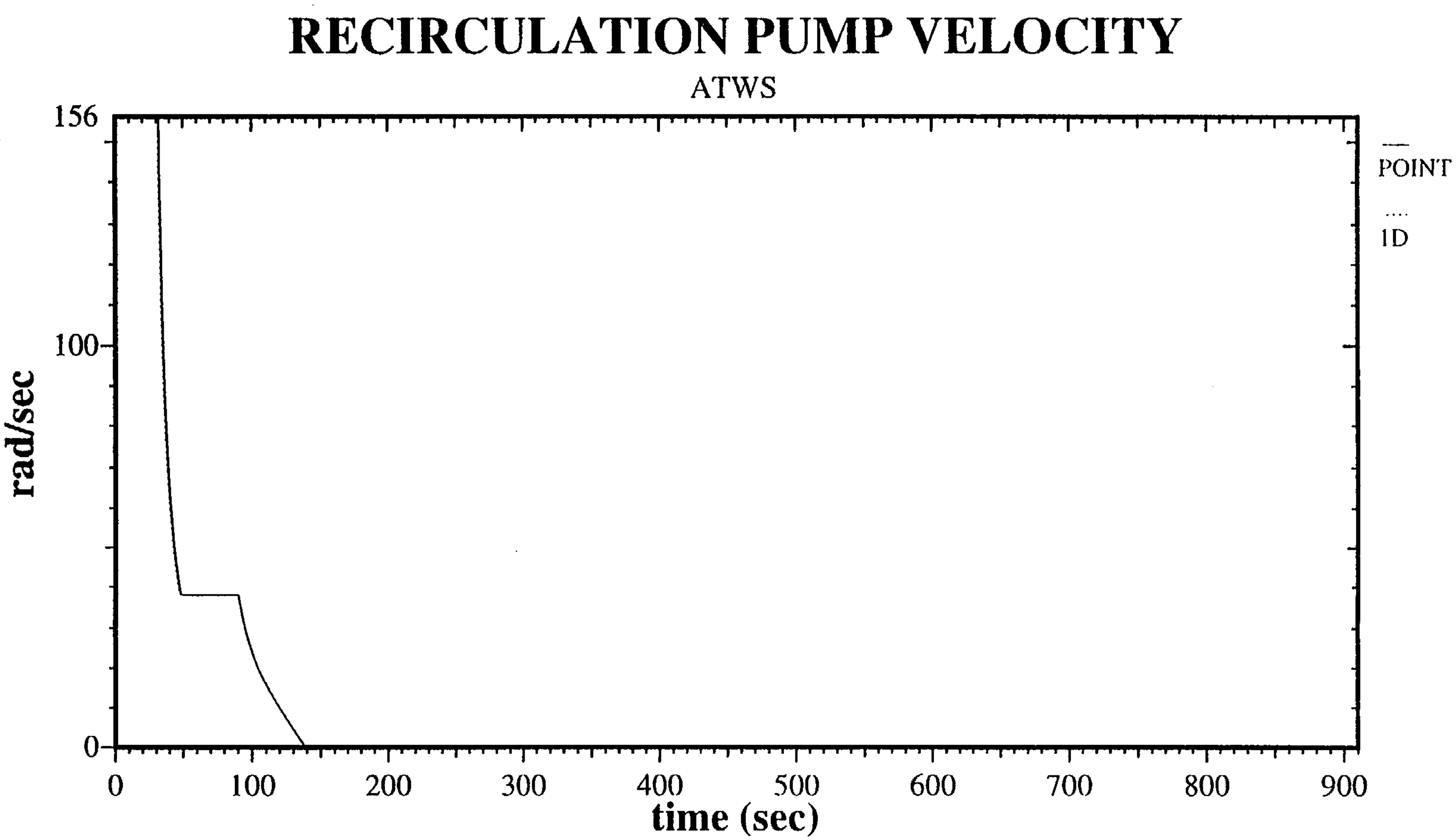

Figure 16: Recirculation Pump Velocity

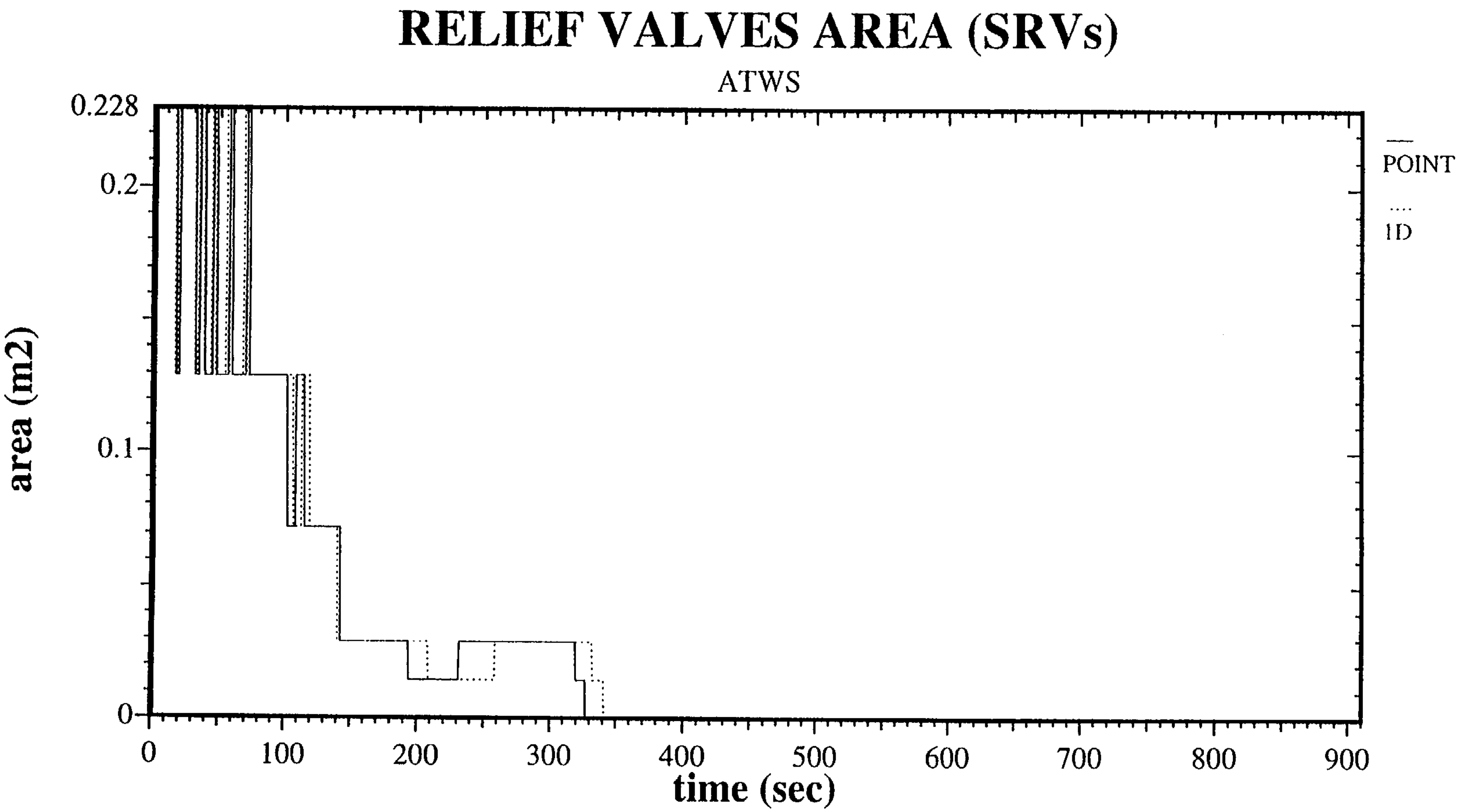

Figure 17: Relief Valves Area

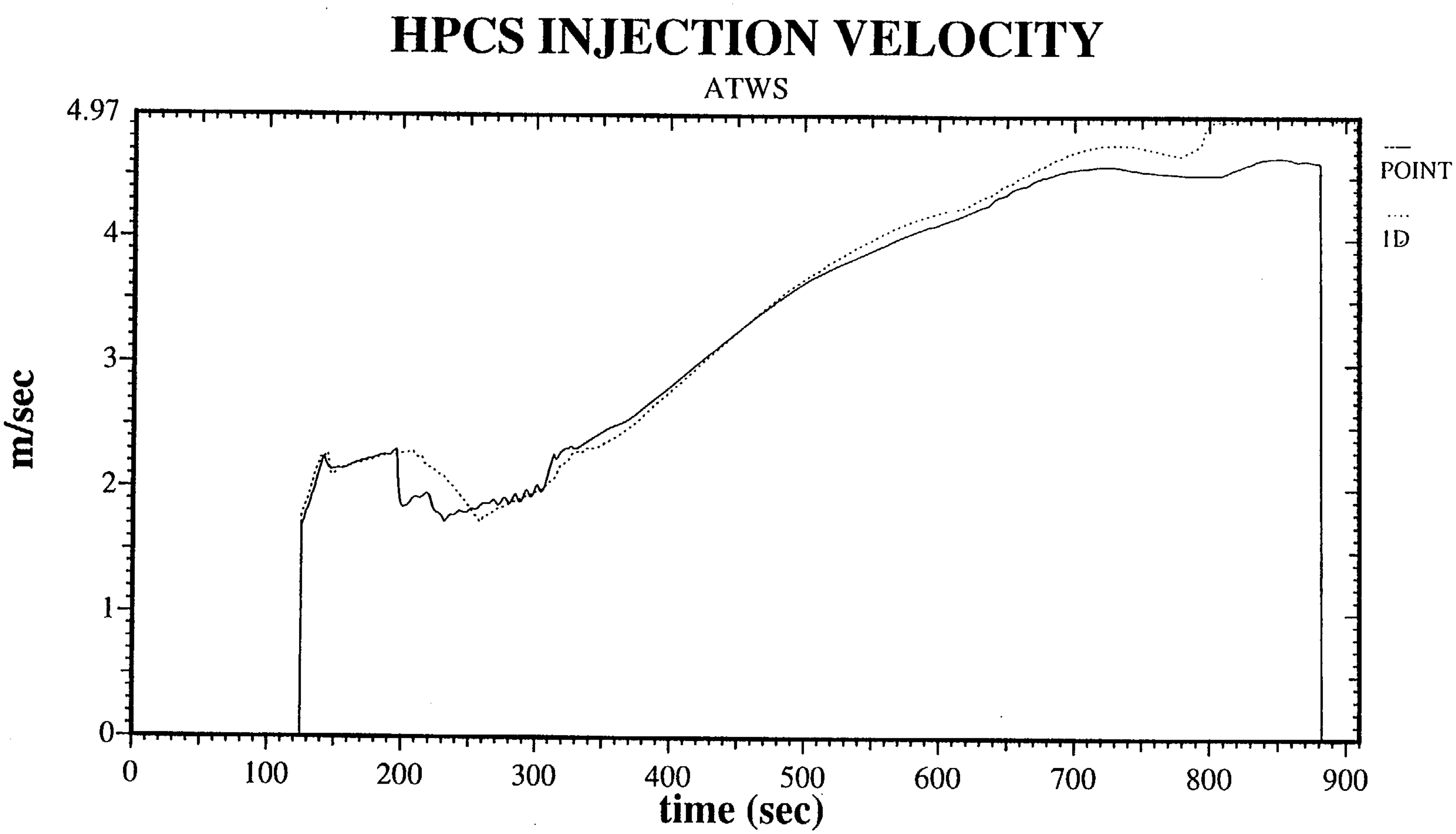

Figure 18: HPCS Injection Velocity

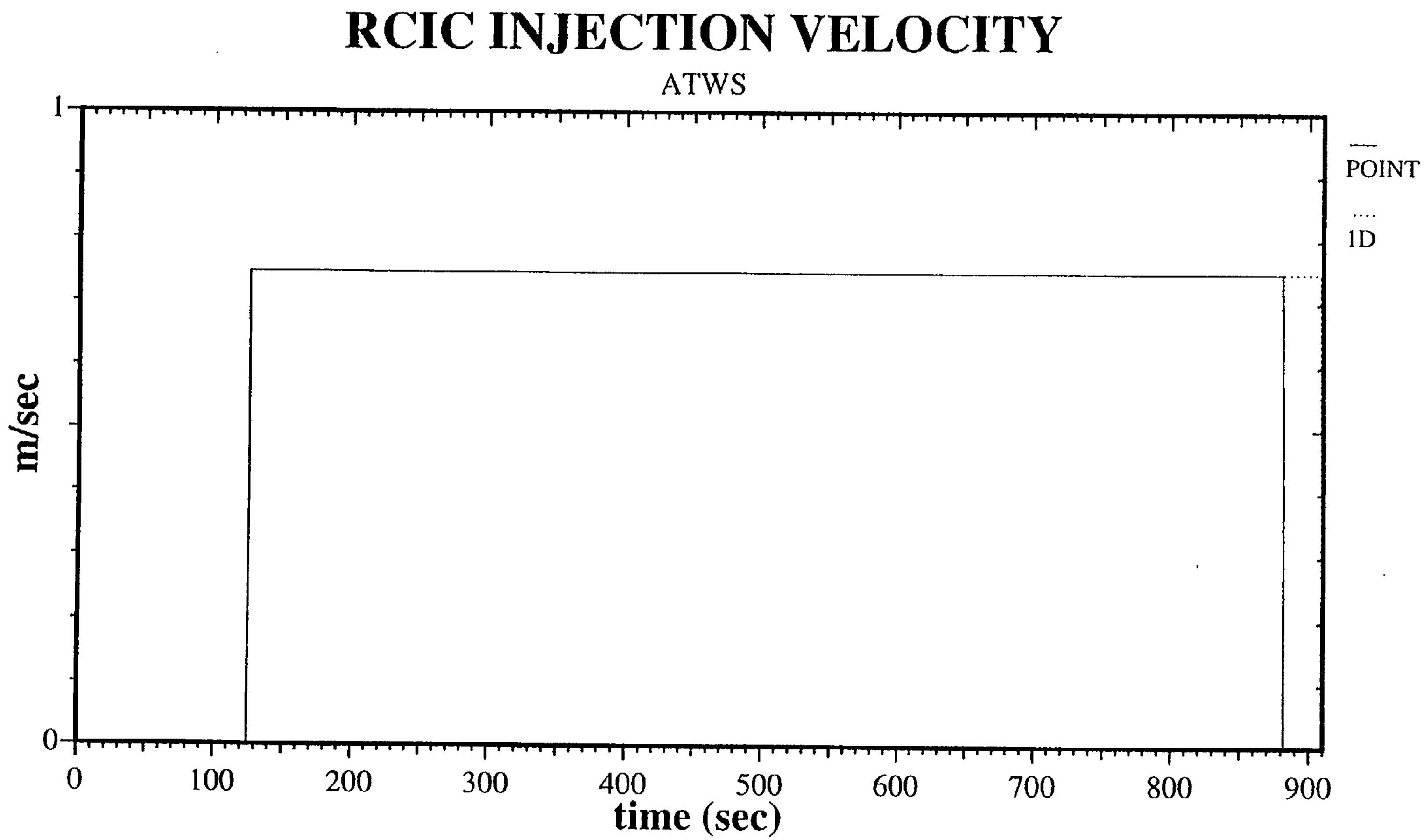

Figure 19: RCIC Injection Velocity

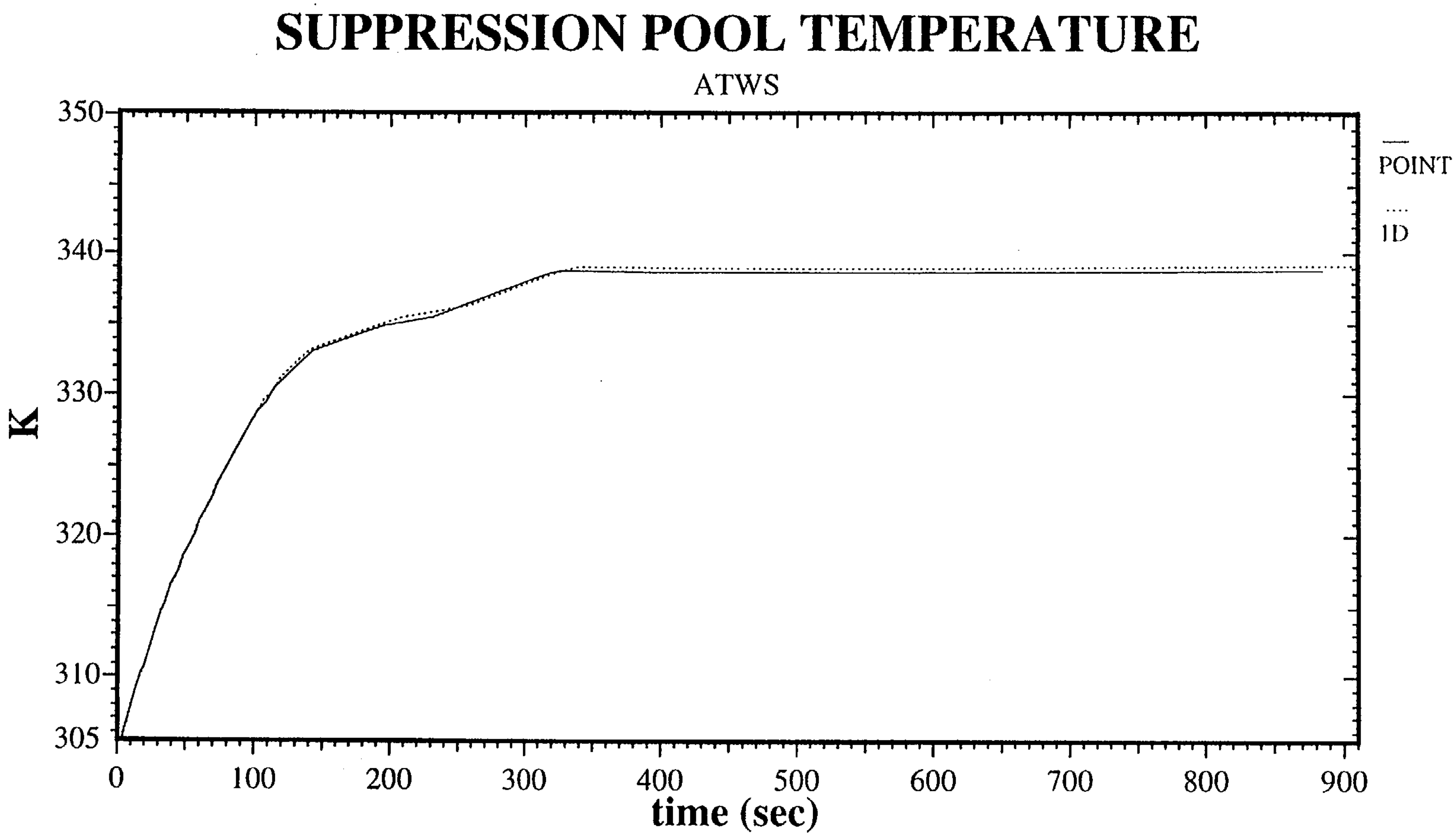

Figure 20: Suppression Pool Temperature

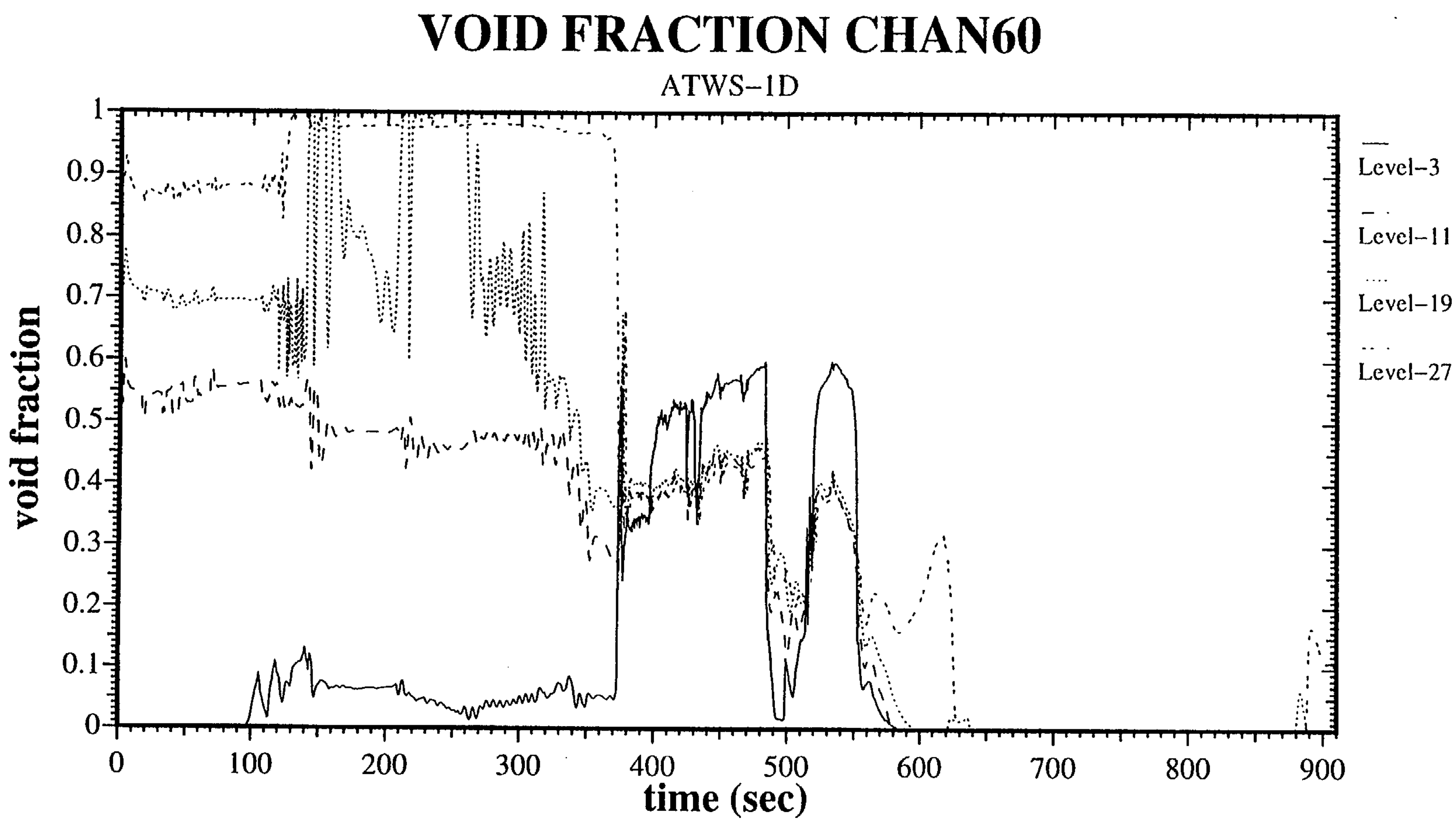

Figure 21: Void Fraction in Peripheral Channels (Chan 60)

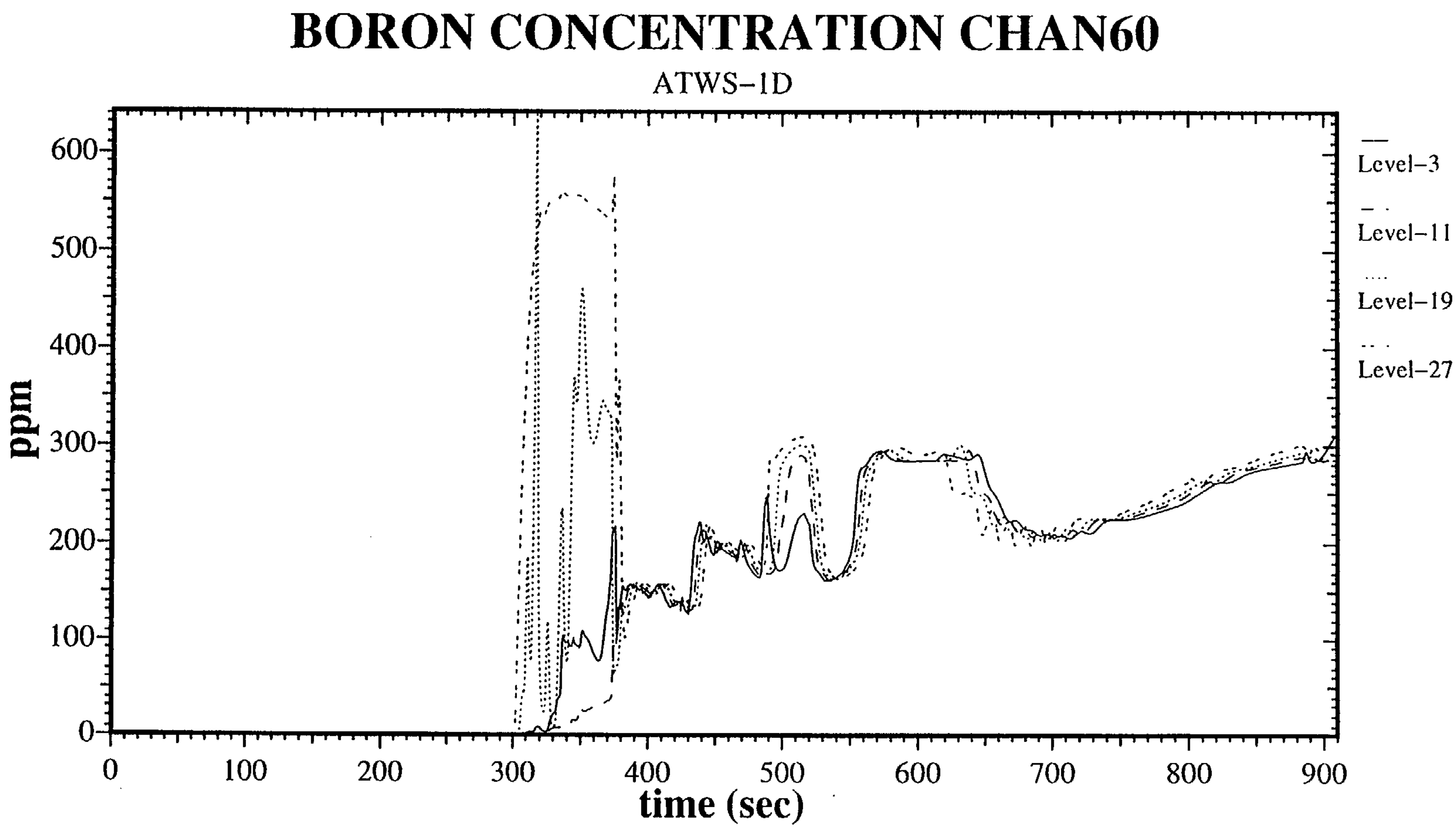

Figure 22: Boron Concentration in Peripheral Channels (Chan 60)

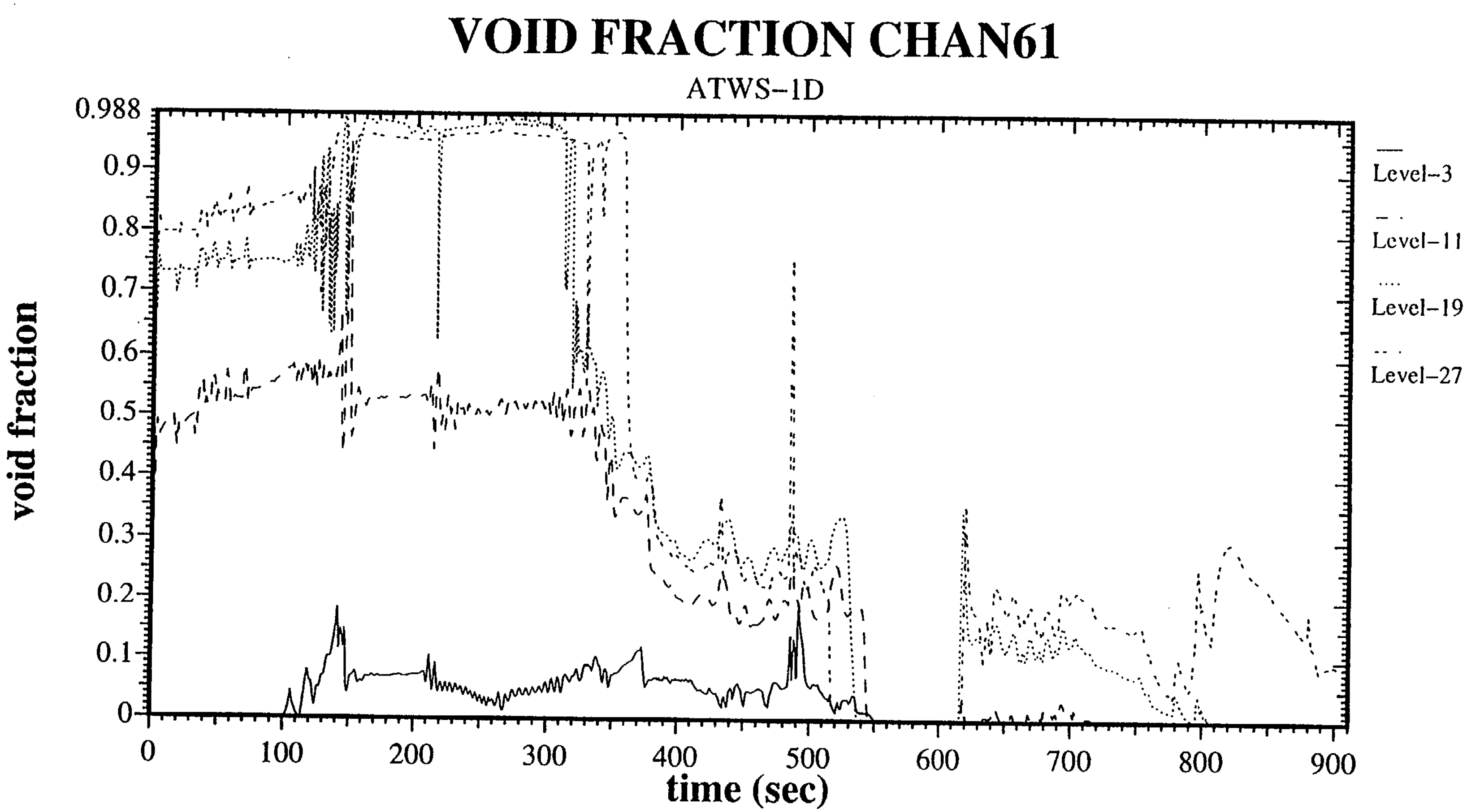

Figure 23: Void Fraction in Middle Channels (Chan 61)

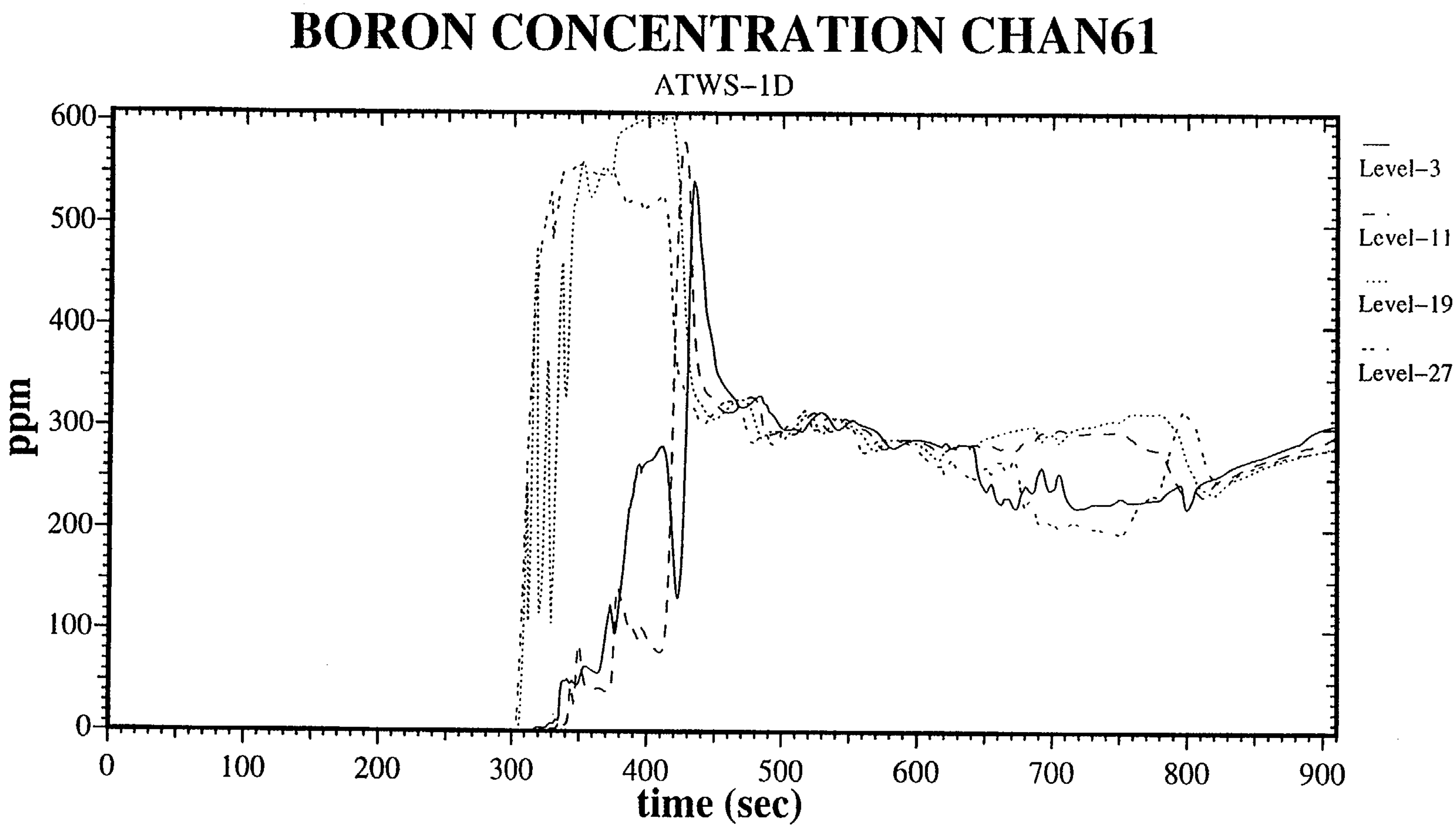

Figure 24: Boron Concentration in Middle Channels (Chan 61)

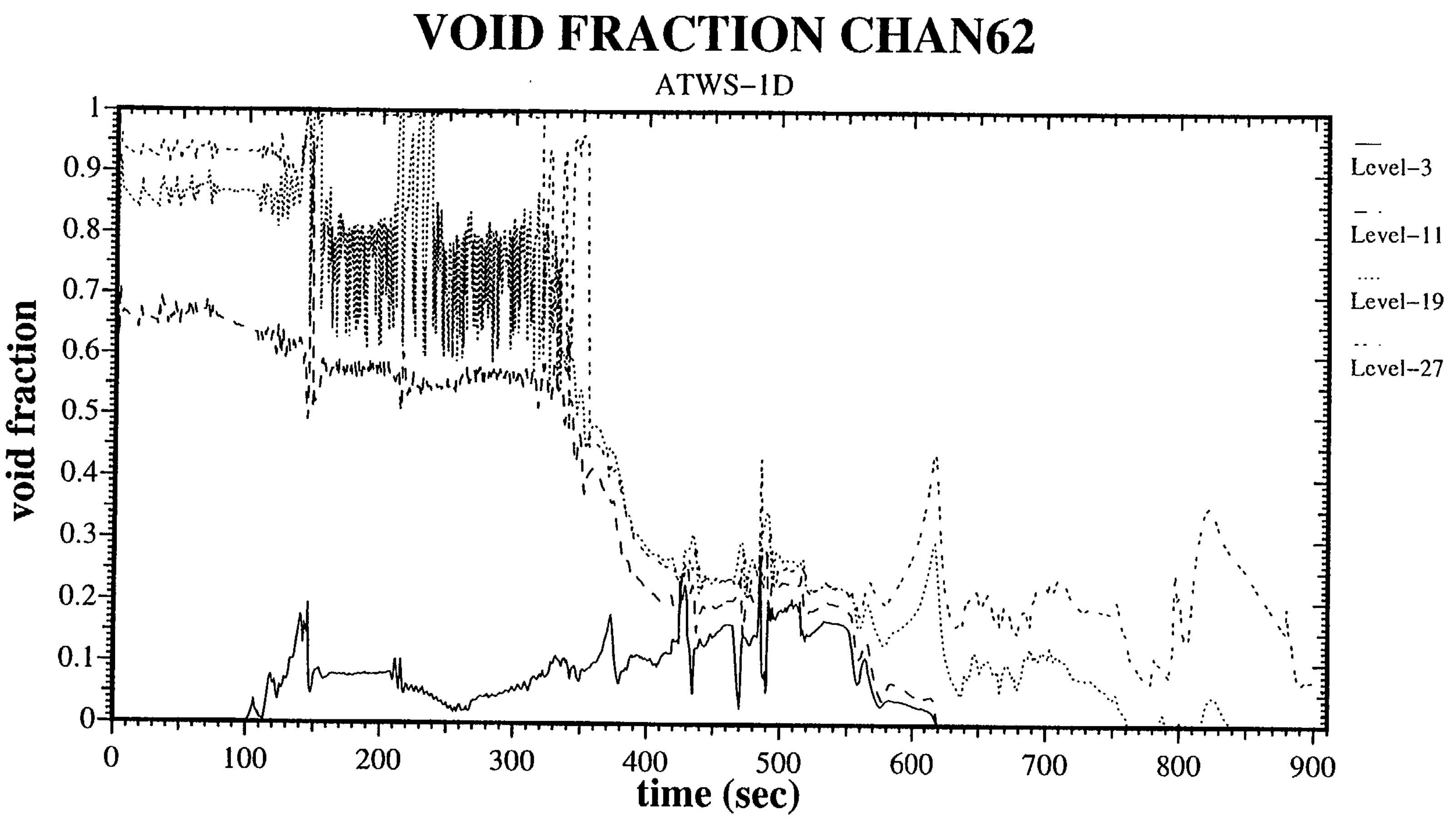

Figure 25: Void Fraction in Center Channels (Chan 62)

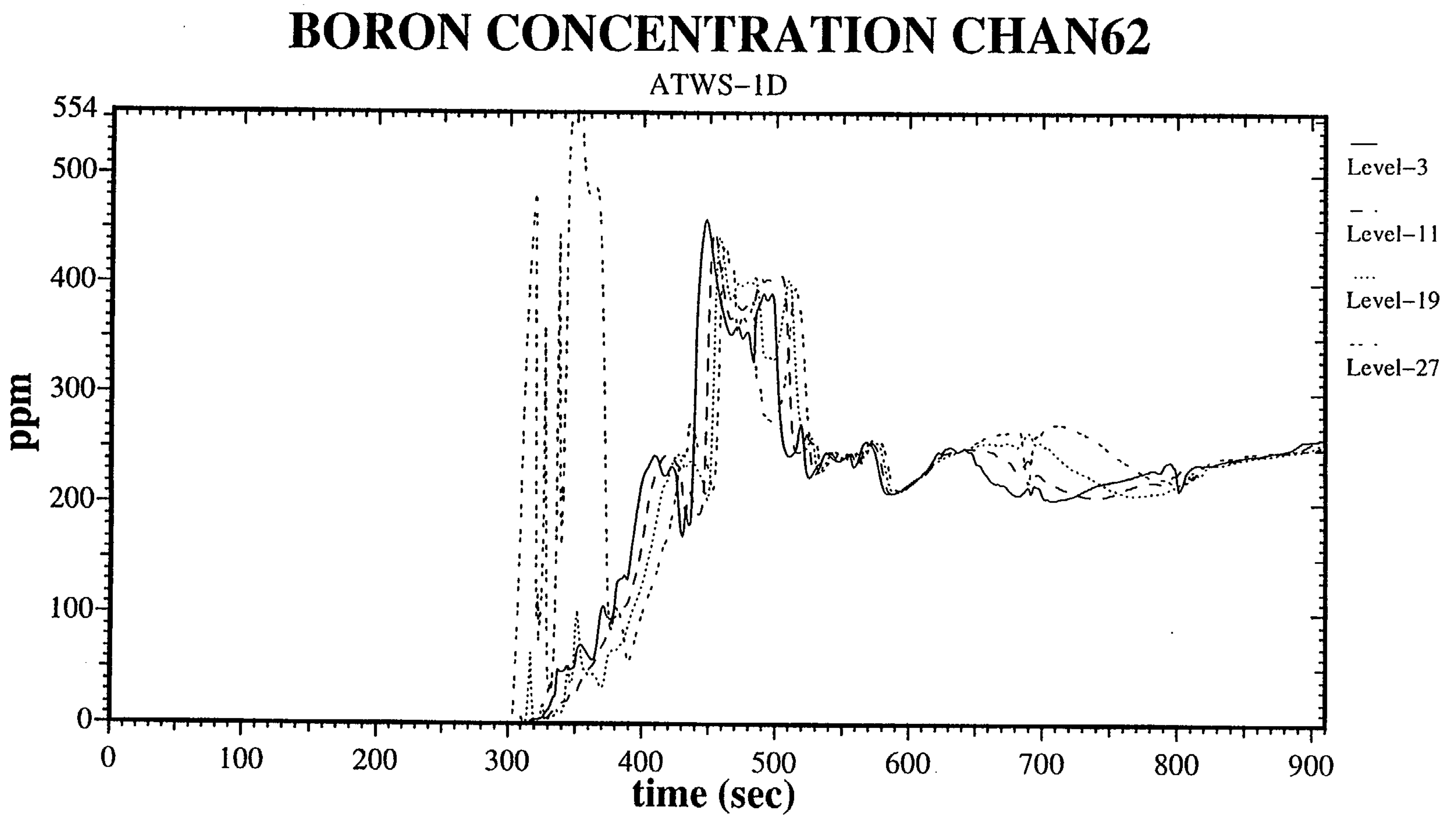

Figure 26: Boron Concentration in Center Channels (Chan 62)

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

Anticipated transient without scram (ATWS) are considered as events which might produce a severe accident in a boiling water reactor (BWR). It has been selected the most favorable of the different scenarios that could lead to an ATWS accident: An inadvertent closure of the main steam isolation valves (MSIVs). To mitigate this accident, a borated water solution is injected. This action is an alternative way to shutdown the reactor quickly and effectively. This event has been analyzed using the TRAC-BF1 computer program.

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