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Application of RELAP5/MOD3.1 to ATWS Analysis of Control Rod Withdrawal From 1% Power Level

Prepared by

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ABSTRACT

The RELAP5/MOD3.1 computer code has been applied in the analysis of a regulating control rod group withdrawal from 1 % power level. The analysis has been carried out as an Anticipated Transient Without Scram (ATWS) event.

The analysis is related to an extensive inherent boron dilution study which was performed to Loviisa Nuclear Power Plant (NPP) in the mid 1990's. The main goal of the analysis was to study if during the accident so called boiling-condensing mode develops and thereafter a water plug with low boron concentration is formed in the cold leg loop seal.

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EXECUTIVE SUMMARY

An extensive study for Loviisa NPP concerning inherent boron dilution was carried out in the mid 1990's. One of the main concerns was that during a transient or an accident liquid level in the pressure vessel decreases below the hot leg elevation and so called boiling-condensing mode begins. Boiling-condensing mode means that coolant boils in the pressure vessel and steam condenses in the steam generators forming a low boron concentration water plug in the cold leg loop seal. In case of Loviisa, condensing in the steam generator is obvious, since primary side cooldown is done by cooling down the secondary side.

One of the accidents which was analyzed during the study using RELAP5/MOD3.1 computer code was a regulating control rod group withdrawal from 1 % power level as an ATWS case. Concerning this kind of analysis the point core kinetics model in RELAP5/MOD3.1 is not a very good model to calculate power behavior since it does not take into consideration the change in the axial power profile when the control rod is moving upward. However, from the boron dilution point of view this shortcoming was not considered significant because the main goal was to find out events during which boiling-condensing mode develops. Therefore RELAP5/MOD3.1 code was used in the preliminary analysis to get qualitative data, which was later utilized when the sequences were considered for detailed analyses using the coupled thermal-hydraulic and 3-D reactor dynamics code HEXTRAN [1].

The RELAP5/MOD3.1 analyses results showed that continuous long-term boiling-condensing mode did not develop. Instead it was typical that one or sometimes more boiling-condensing mode phases took place during an accident but they lasted so short time that a serious boron dilution problem was not discovered. The maximum boron dilution was only 300 ppm, which corresponds to 2.3 % reactivity. This kind of dilution would probably not cause a serious core power transient during natural circulation even if all the control rods would be above the core.

1 INTRODUCTION

The first scenarios of local boron dilution accidents concerning Loviisa NPP were analyzed in 1990. It was quite soon recognized that the loop thermal-hydraulics in the two-phase mode had a strong effect on the development of inherent boron dilution conditions. Therefore, scoping studies of different ATWS sequences were also made using RELAP5 code with the point kinetics core model.

The inherent boron dilution arises in the primary circuit when boiling-condensing cooling mode prevails. Boiling-condensing mode occurs when coolant boils in the pressure vessel and steam condenses in the steam generators forming a low boron concentration water plug in the cold leg loop seal. Very detailed description of the thermal hydraulics and core neutronics is necessary to reliably analyze the event. It was found that only 3-D modeling of neutronics is capable to predict the core reactivity level reliably in all conditions because axial and even radial density distributions in the core strongly changed during the transient due to different flow conditions: nominal forced flow, main circulation pump rundown, and different natural circulation phases. Using fewer dimensions in the core modeling in some cases changed the whole accident scenario with respect to the occurrence of inherent boron dilution.

The amount of the primary coolant mass lost during the pressure increase and after stop of the reactor coolant pumps (RCP) depends crucially on the fission power behavior of the core. In the beginning of cycle (BOC) conditions, when the reactivity feedback of coolant density is very weak, there must be boiling in order to decrease the fission power, and the coolant mass inventory can decrease to such low level that boiling-condensing mode occurs. In the middle of cycle (MOC) conditions, threat of inherent boron dilution during ATWS cases is over because of clearly stronger feedback of coolant density. The studies with the different core models indicated that reliable fission power prediction during the transient was only achieved with a 3-D core model describing neutronics and flow channels of the fuel assemblies individually. Using less dimensions in the core modeling did in some cases change the whole accident scenario with respect to the occurrence of inherent boron dilution, even though the model was separately validated both in nominal and natural circulation flow conditions.

Mixing provides an effective natural defense against severe reactivity excursions. Mixing occurs already in the steam generator in the build up phase of the diluted water plug and later in the reactor pressure vessel downcomer and in the lower plenum, when the diluted plug starts to move towards the core. Consequences of a reasonably conservative 50 % diluted water plug reaching the core with natural circulation would be limited to heat transfer crisis and consequent temporary fuel cladding overheating. High fuel temperatures are not to be expected with natural circulation.

It should also be emphasized that a common cause failure causing mechanical jamming of all control rods in the VVER-440 reactor is considered to be highly improbable. Drop of at least some of the rods help the shutdown of the core to such an extent that natural circulation is not disturbed. In the case of electrical failure even delayed scram actuated by the operator effectively eliminates the consequences of possible diluted water plugs.

In 1996 RELAP5/MOD3.1 computer code was applied in the ATWS analysis of a regulating control rod group withdrawal from 1 % power level. When RCPs are not running and the coolant inventory is reduced in the primary circuit, it is possible that two-phase natural circulation is interrupted when liquid level decreases below hot leg elevation. The main goal of the analysis was to investigate if during the accident this phenomenon takes place and a boiling-condensing mode develops. If it does, there is a possibility that steam generated in the core flows to the steam generators, condenses and forms a diluted water plug in the cold leg loop seal. The analysis concentrated on the formation of the diluted plug and less attention was paid on the reactivity effect when the diluted plug flows into the core after restart of the two-phase natural circulation.

The shortcoming of the core model in the RELAP5/MOD3.1 code in this kind of event is that it cannot calculate the change in axial power profile when a control rod moves upwards and when a diluted water plug flows into the core. On the other hand, the effect of the control rod movement can be taken into account in the point kinetics model in such a way that the total power generation in the core is tuned to correspond the real power generation. Control rod group withdrawal is given in the code input so that the insertion of reactivity is based on the elevation where the withdrawal starts and it changes linearly as a function of control rod position. Additionally, also reactivity feedback from coolant density, fuel temperature and boron concentration is needed.

Several parameter variations were carried out and the main interest was focused on the reactivity worth of the control rod group withdrawal, which mainly depends on the initial position of the control rod. Therefore, because of strong reactivity variations from the control rod group withdrawal, the feedback inaccuracy of point kinetics model from coolant density and fuel temperature at least partly disappeared.

The selection of RELAP5/MOD3.1 code to these analyses can be justified since the main goal of the analyses was to get qualitative data, which was later utilized when the sequences for detailed analyses using the 3-D reactor dynamics code HEXTRAN [1] were considered.

2 NODALIZATION MODEL

There are two VVER-440 units at the Loviisa site and Unit 1 was chosen as a reference unit. Basically both units are similar but there are some differences like a little higher (approx. 4 %) primary coolant mass flow rate in Unit 2.

The input model included a detailed description of the primary circuit with all six loops modeled separately. In Figures 2-1 and 2-2 the nodalization model of pressure vessel and one of the six loops is depicted, respectively.

Concerning pressure vessel nodalization one important feature can be mentioned. The volume in the upper plenum at the hot leg elevation (volume # 5) should locate at the hot leg elevation and should have the height of the hot leg pipe diameter. The reason for this is when the liquid level in the pressure vessel decreases below the hot leg elevation, the flow at the same time changes from two-phase to single-phase steam flow. Higher volume was found to have a distinct effect on the results.

As Figure 2-2 shows, Loviisa has a unique feature since there are two loop seals in the primary loop. One is in the hot leg and the other in the cold leg. Also reactor coolant pump is different from a typical one. Suction takes place from the side and discharge is downwards.

Figure 2-3 shows the nodalization model of the pressurizer. As one can see pressurizer is connected to two hot legs (YA13 and YA14). Pressurizer pressure vessel was divided into 15 volume.

High-pressure injection (HPI), make-up injection, letdown and also primary coolant purification systems were modeled as shown in figure 2-4.

The secondary side included a detailed modeling of each steam generator and steam lines to the turbines. Also all the main, emergency and auxiliary emergency feed water systems were modeled as shown in Figure 2-5. Secondary side was divided only into three stacked volumes on the heat transfer tube area. The height of the bottom volume was 0.75 m, the middle volume 0.69 m and the top volume 0.549 m. During the accident, however, liquid level in steam generators decreased and the number of steam generator secondary side volumes can be considered too small. This is due to the fact that RELAP5 tends to overestimate heat transfer in steam generator when liquid level decreases. A good heat transfer rate is predicted from primary to secondary side in the volumes until practically all the water has boiled off and void fraction in the volume becomes 1.0. This kind of behavior results in unrealistic coolant temperature behavior in the cold leg. When secondary side volume dries out, heat transfer to the secondary side stepwise decreases and cold leg coolant temperature stepwise increases. This phenomenon was also seen in these analyses, especially in the cases where steam generators liquid level decreased slowly.

3 ASSUMPTIONS AND ANALYZED CASES

As a reference case an earlier calculation, which was carried out using 3-D reactor dynamics code HEXTRAN, was used. In that case control rod group withdrawal took place from 1 % power level assuming 2.2 % reactivity worth of the control rod and the initial position of the rod 2 m below the top of the core. This assumption made it possible to supply maximum reactivity during withdrawal since, according to the Technical Specifications, control rods are not allowed to locate deeper during a hot standby situation.

In order to get the result of RELAP5/MOD3.1 calculation correspond the reference result, some tuning calculations were carried out. During these calculations reactivity feedback data was tuned to get approximately equal total reactor power at the end of withdrawal as in the reference case.

In RELAP5/MOD3.1 calculation the reactivity insertion as a result of the control rod group withdrawal was modeled, e.g. in the base case, in such a way that reactivity increased linearly from 0 to 3.6 $ in 100 s. This was due to the fact that the initial position of the control rod was the lowest allowed and withdrawal speed was 2 cm/s. The calculation was continued in each case at least up to 30 min. Reactor scram was not assumed to take place from any reactor protection signal during this period. However, it is realistic to assume that after some time (30 min) the operators are able to insert the control rods into the core e.g. by disconnecting the electricity from them. Then at least the rod, which was withdrawn, drops down into the core because it could not be mechanically stuck.

In one case analysis was continued up to 60 min without reactor scram since the accident had developed to a boiling-condensing mode and the length of the mode was examined. After the reactor trip starting of the boiling-condensing mode is not very probable since there is not any leakage in the primary circuit. When the control rods are in the core the reactor can also tolerate more dilution than in the case when the control rods are above the core.

To maximize decay heat power initial power history was assumed to be such that fuel had been in the reactor for two years in average. About two days earlier the reactor had been scrammed and the operators were returning it back to the power so that the reactor had been 15 min at 1 % power level. This kind of power history results to the state where both fission power and decay heat power are 0.5 % of the nominal full power. On the other hand, to get the weakest reactivity feedback and thus to maximize boiling in the core, BOC reactivity feedback was assumed. Reactivity dependency from coolant density, fuel temperature and coolant boron concentration is presented in the Appendix in RELAP5 input format.

In the analysis it was assumed that one out two emergency feedwater pumps is available and secondary side pressure was approximately at the opening pressure of the turbine bypass valves (4.7 MPa). It was also assumed that one out of two high head boron injection pumps is available and both HPI pumps from one redundancy. These assumption were not, however, considered to have any greater significance to the results since the actuation signal was not expected to turn on, at least not in the early phase of the accident. Moreover, due to the shut-off head of the HPI pumps, the HPI system is capable to inject ECC water into primary circuit only when pressure is less than 12.5 MPa. All other ECC and normal operating systems were assumed to operate as designed including the letdown system.

The fuel gap width was adjusted to get the mean fuel temperature of 530 °C as in HEXTRAN run. Initial boron concentration was assumed to be 1646 ppm. Main primary and secondary side parameter values are presented in Table 2-1.

The effect of reactivity worth was studied in such a way that the withdrawal started from different elevations. Altogether six different variations were analyzed. The initial control rod position and reactivity worth in each case is presented in Table 2-2. The difference between case 1 and 2 was due to the fact that the case 2 was calculated in order to find out the effect of unrealistic high (4 $) control rod reactivity worth. When withdrawal begins from 2 m, 3.6 $ reactivity worth can be considered a conservative value.

In addition to 6 cases presented in Table 2-2, two parameter variations were carried out for the case 6 studying the effect of secondary side pressure and feedwater capacity.

4 ANALYSIS RESULTS

4.1 Base case, reactivity of withdrawal 3.6 $

As already mentioned an event, where control rod is locating 2 m below the top of the core and withdrawal velocity is 2 cm/s, was chosen as a base case. The case was used to check that the reactor power at the end of withdrawal was close to the power calculated using HEXTRAN. In Figures 4.1-1...4.1-12 the behavior of the main parameters during calculation is presented.

As a result of the control rod group withdrawal reactor power begins to increase reaching the maximum power of 2342 MW at the point of time when control rod movement stops (Figure 4.1-1). After that feedback from coolant density or core void fraction (Figure 4.1-4) makes core power to decrease first slowly but after 140 s faster, because as a result of low level (< 1.6 m) in steam generators RCP stops in the corresponding loop. In this case all RCPs stop at the same time since in the early phase of the accident the steam generator liquid level behaves in the same way in each steam generator (Figure 4.1-9). Also primary pressure increases in the early phase reaching maximum 16.6 MPa at 205 s (Figure 4.1-7). Thereafter two-phase natural circulation phase begins. Hot leg void fraction maintains at approximately 0.6 (Figure 4.1-5) and the mass flow rate into the core at about 400 kg/s (Figure 4.1-6). Steam condenses totally in the steam generators and coolant is approximately 10 °C subcooled when entering the core (Figure 4.1-2). From 500 s onward the situation remains rather constant till the end of the calculation.

Boron concentration does not decrease in the core volumes from the initial value (Figure 4.1-10). Instead it slightly increases in the early phase of the accident because a lot of steam is discharged through the pressurizer safety valves. At certain point of time during the calculation boron concentration in each cold leg loop seal decreases below the initial value (Figure 4.1-11) as a result of steam condensation in the steam generators. Degradation is, however, rather small (< 100 ppm) and therefore it does not cause any safety problem.

In the beginning of the accident reactivity increases very strongly (Figure 4.1-12) reaching the maximum value of 0.58 $ at 20 s. After that point of time feedback, especially from coolant density, becomes effective and reactivity begins to decrease.

At the moment when maximum core power is reached reactivity is already less than 0.1 $.

4.2 Base case, reactivity of withdrawal 4.0 $

Since previous case did not result in boiling-condensing mode, it was decided to increase reactivity from the withdrawal up to 4.0 $. This value can be considered very conservative because the initial position of the control rods and the withdrawal time was maintained the same.

The increased reactivity results, of course, in greater maximum power (2588 MW) at 100 s (Figure 4.2-1). After that the behavior of power does not essentially differ from the power behavior in the previous case. RCPs stop some time earlier (130 s) due to faster degradation of the steam generators liquid level (Figure 4.2-9). This also results in earlier power degradation due to the reactivity feedback. Just before the calculation was stopped fission power fades away and the situation develops to a boiling-condensing mode in which the two-phase natural circulation stops. The analysis was not, however, continued since at 30 min the operators can be assumed to be able to get at least the withdrawn control rod into the core.

In spite of the greater maximum power in the early phase of the accident primary pressure does not rise as high as in the previous case (Figure 4.2-7). This results from the fact that power degradation begins 10 s earlier due to stop of RCPs.

As in the previous case boron concentration in the core volumes does not decrease below the initial value (Figure 4.2-10). In the later phase boron concentration strongly increases because boiling in the core. Boron concentration in the cold leg loop seals behaves also like in the previous case and not any low boron concentration water plugs are predicted to form (Figure 4.2-11).

Also the behavior of reactivity is similar if the last 100 s is not taken into account which indicates the beginning of the boiling-condensing mode. The behavior of void fraction in the core (Figure 4.2-4) and in hot legs (Figure 4.2-5) as well as the core inlet mass flow rate (Figure 4.2-6) at the end of calculation also indicates the beginning of the boiling-condensing mode.

As a conclusion it can be mentioned that the increase of reactivity worth from 3.6 $ to 4.0 $ has only a rather mild effect on the results during the first 30 min.

4.3 Reactivity of withdrawal 2.2 $

In this case withdrawal reactivity was decreased down to 2.2 $. This meant that the initial position of the control rod was 1.22 m below the top of the core and withdrawal time 61 s.

When less reactivity is inserted the reactor maximum power does not rise as high as in the previous cases; only up to 1707 MW (Figure 4.3-1). Smaller reactor power also causes slower degradation of the steam generators liquid level (Figure 4.3-9) and later stop (175 s) of RCPs. At about 800 s boiling in the core begins continuing almost continuously to the end of the calculation. This results in strong reactivity feedback (Figure 4.3-12) and at about 15 min fission power fades away. Steam is not, however, able to flow into the steam generators because of water in the hot leg loop seals blocks the steam flow. At 20 min steam pushes the hot leg loop seals open and flows into the steam generators condensing there. As a result primary pressure decreases rapidly from 13.6 MPa down to 12.2 MPa (Figure 4.3-7). This makes two-phase natural circulation to restart but only for a short period (Figure 4.3-6). When the hot leg loop seals fill with water the earlier boiling mode is restored.

Boron concentration in the core volumes behaves as in two previous cases; i.e. it does not decrease below the initial value during the calculation (Figure 4.3-10). Instead the situation in the cold leg loop seals is different. In three loops boron concentration decreases below initial concentration (Figure 4.3-11). In YA13 and YA14 loops boron concentration decreases less than 100 ppm but in YA11 loop minimum concentration is 300 ppm below initial value. No problem is expected if the control rods are in the core but if they are above some kind of power transient can be expected. The power transient depends on the size of diluted water plug, entrance velocity and core reactivity level when the water plug enters the core.

It can be concluded that a continuous boiling-condensing mode was not predicted but occasional boiling-condensing mode was developed during the event. This mode is characterized by sporadic steam flow through hot leg loop seals and it can be called a bubbling mode.

4.4 Reactivity of withdrawal 1.4 $

This case dealt with the event in which initial control rod position was 0.78 m below the top of the core. Withdrawal time was 39 s and reactivity worth 1.4 $.

As a result of withdrawal maximum power reaches "only" 1083 MW (Figure 4.4-1). Power behavior shows that this case does not develop to a bubbling mode like the previous case. This can also be seen from the behavior of the core entrance mass flow rate (Figure 4.4-6), which never decreases as small as in bubbling mode. Also pressure vessel liquid level maintains above hot leg elevation during the calculation (Figure 4.4-3).

As for boron concentration the situation also looks good since both in the core volumes (Figure 4.4-10) and in the cold leg loop seals (Figure 4.4-11) boron concentration never decreases below the initial value.

1.4 $ reactivity worth as a result of control rod group withdrawal is not enough to remove so much coolant from the primary circuit through the safety and relief valves that the liquid level in the pressure vessel would decrease below hot leg elevation. The event can be considered as a "traditional" ATWS where reactor power finally settles down to a certain low level.

4.5 Reactivity of withdrawal 1.8 $

Since in the previous case reactivity effect caused by the withdrawal was too weak, it was decided to increase reactivity assuming that the initial location of the control rod was 1 m below the top of the core. Thus withdrawal time was 50 s and the reactivity effect 1.8 $ which was 50 % of the base case reactivity.

The peak power in this case reaches 1438 MW (Figure 4.5-1). Power behavior shows that fission power does not fade away during the calculation, which is a clear indication, that two-phase natural circulation continues throughout the analyzed period. This can also be seen from the behavior of the pressure vessel liquid level (Figure 4.5-3), void fraction in the core (Figure 4.5-4) and the core entrance mass flow rate (Figure 4.5-6).

Boron concentration behavior is also favorable due to the primary circuit behavior. Boron concentration in the core volumes (Figure 4.5-10) and in the cold leg loop seal (Figure 4.5-11) does not decrease below the initial boron concentration except temporarily in the loop YA13 (<50 ppm).

As a conclusion it can be mentioned that a withdrawal, starting from 1 m depth, is not able to decrease primary coolant mass inventory so much that liquid level in the pressure vessel would drop below the hot leg elevation and two-phase natural circulation would stop.

4.6 Reactivity of withdrawal 1.9 $

Since the previous case did not result in boiling-condensing mode, it was decided still increase the withdrawal depth; this time, however, only a little bit from 1 m to 1.06 m. This increased reactivity worth from 1.8 $ to 1.9 $. The reason for such a small increment was that there was a feeling that the "edge" was quite close.

It can be easily seen from the reactor power behavior that the situation changes drastically compared with the previous one (Figure 4.6-1). When reactor remains on decay power it means that two-phase natural circulation stops due to the pressure vessel liquid level dropping below hot leg elevation (Figure 4.6-3). This time it was also decided to find out how long time it takes until two-phase natural circulation restarts and how the reactor behaves during that period. Therefore calculation was continued up to 1 h point of time.

The reactor is under boiling-condensing mode altogether for 35 min (Figure 4.6-6). At 1500 s two-phase natural circulation restarts for a short period but it is not until at 3100 s when the two-phase natural circulation restarts and continues at least up to the point of time when calculation is stopped. The reason for the second restart is that large amount of steam enters the steam generators and condenses there and as a result pressurizer level decreases below the HPI pumps set point. At the same time primary pressure decreases below the HPI pumps shut-off head and water draining from the pressurizer together with high-pressure injection increases the pressure vessel coolant mass inventory so much (Figure 4.6-8) that two-phase natural circulation can restart.

Like in all other cases boron concentration in the core volumes does not decrease below the initial value (Figure 4.6-10). Instead in three cold leg loop seals boron concentration decreases below the initial concentration after 1500 s (Figure 4.6-11). The drop is, however, rather small since minimum concentration in the loop YA16 is only approximately 200 ppm below the initial concentration.

4.6.1 Reactivity of withdrawal 1.9 $ and turbine bypass not available

In the all calculated cases it was assumed that turbine bypass valves are operating as designed. The valves start to open at 4.7 MPa and are fully open at 5.1 MPa. This resulted in situation in which heat transfer to the secondary side was better than in the case if only secondary side relief valves would be available. These vales start to open at 5.2 MPa and are fully open at 5.4 MPa. Therefore it was decided to calculate a case where only secondary side relief valves were assumed to be available and thus the initial secondary side pressure was raised from 4.7 MPa up to 5.2 MPa. As a result of this assumption ΔT between primary and secondary side became smaller which decreased heat transfer between primary and secondary side.

Reactor power behavior in this case is very much the same as in the previous case (Figure 4.6.1-1). Maximum power is a little bit smaller and fission power fades away somewhat earlier. The former is due to the fact that reactivity feedback from coolant density is slightly stronger and the latter results from earlier stop of two-phase natural circulation (Figure 4.6.1-6) since pressure vessel liquid level decreases more rapidly below the hot leg elevation (Figure 4.6.1-3). Core entrance mass flow rate behavior shows that like in the previous case two-phase natural circulation restarts at 1500 s. However, now the two-phase natural circulation does not quickly stop again but after a short degradation period it seems to start again. The reason for this is that as a result of condensation in the steam generators, liquid in the pressurizer is discharged to the pressure vessel raising the liquid level (Figure 4.6.1-3). Another reason is that the primary coolant mass inventory does not decrease as much as in the previous case (Figure 4.6.1-8).

Boron concentration behavior in the core volumes (Figure 4.6.1-10) and in the cold leg loop seals (Figure 4.6.1-11) is almost similar in both cases.

Weaker heat transfer does not seem to have any major effect on the results. The situation seems even more favorable during the first 30 min if the criterion is the generation of the boiling-condensing mode.

4.6.2 Reactivity of withdrawal 1.9 $ and one main feedwater pump available

In the second variation one out of four feedwater (FW) pumps was assumed to be available in addition to one emergency feedwater (EFW) pump which was available in all other cases.

During the first 500 s reactor power behavior is quite close to the behavior in the two previous cases (Figure 4.6.2-1). Fast fission power degradation as a result of RCPs stop begins this time a little bit later because liquid level in the steam generators decreases more slowly (Figure 4.6.2-9). After 500 s power behavior is, however, completely different. This is due to the fact that as a result of better heat transfer to the secondary side primary coolant mass inventory drops clearly less than in the reference cases (Figure 4.6.2-8). Therefore pressure vessel liquid level never decreases below the hot leg elevation (Figure 4.6.2-3) and boiling-condensing mode cannot start. Two-phase natural circulation maintains good during the whole calculation (Figure 4.6.2-6) and the system reaches a stationary state where heat generated in the core is removed from the primary coolant by the steam generators. The only disturbing factor is the letdown system, which decreases the primary coolant mass inventory as a result of high pressurizer liquid level. If the operators do not do anything the degradation of the mass inventory later ends the stationary state.

As a concluding remark it can be mentioned that one main FW pump during the accident improves essentially the situation and the beginning of boiling-condensing mode is not predicted during 30 min.

5 RUN STATISTICS

The analyses were run on DEC 3000 AXP Model 700 workstation. The input model included 587 volume, 680 junctions and 376 heat structures with 1476 mesh points. Each calculation was carried out using a maximum time step size of 0.04 s from 0 s to 300 s. After that the maximum time step size was increased to 0.1 s and it was used during rest of the calculation. Run time statistics from 0 s to 300 s and from 300 s to 1800 s is presented in Table 5-1. The average grind time

CPU-time / [(number of vol's) · (number of time steps)]

was about 0.33 ms.

6 CONCLUSIONS

This report deals with the regulating control rod group withdrawal from an initial 1 % power level. The analysis was carried out using RELAP5/MOD3.1 computer code and it focused on studying the effect of the initial control rod location to the primary coolant boron dilution.

The weakness of RELAP5/MOD3.1 code in the analysis of reactivity accidents is the fact that it only has the point kinetics calculation capability, which does not take into account the change in the axial power profile during the control rod group withdrawal. One should, however, bear in mind that from the boron dilution point of view this shortcoming is not significant. The aim of the analysis was find out situations where boiling-condensing mode begins and as a result a diluted water plug may develop in the cold leg loop seal. The analysis covers quite many variations of the initial control rod location. Thereby the feedback inaccuracy of the coolant density and the fuel temperature can be considered to have only minor effect as compared to the broad variation of the control rod reactivity worth.

The results indicate that the continuous boiling-condensing mode does not develop in the analyzed cases. In some cases the occasional boiling-condensing or bubbling mode was predicted. This kind of situation does not result in a serious boron dilution problem, because steam condensation followed by dilution takes place only temporarily when steam can flow through the hot leg loop seals and the steam generator primary side is filled with water. Between bubbling periods primary coolant is getting "power" to be able to bubble again. The maximum dilution in the cold leg loop seal was only about 300 ppm, which corresponds approximately 2.3 % reactivity. During natural circulation this kind of dilution will not probably cause any serious power transient though the control rods are above the core. Moreover, the diluted water plug hardly flows into the core without any mixing on the way.

It should be pointed out that due to the numerical diffusion, the boron concentration behavior in different parts of the primary circuit is not predicted correctly. Therefore, more attention should be paid to phenomena like bubbling of the loop seals and fading away of the fission power than to the value of the boron concentration. Also from the operators point of view almost the only indication of the boiling-condensing mode is the information received from the fission power measurement since there is not any pressure vessel liquid level measurement in Loviisa. It is a clear indication of boiling-condensing if fission power fades away in case where control rods are above the core and no boron is injected into the primary circuit.

Another important feature is also the initial position of the control rod since the reactivity worth was found to be a sensitive parameter. When initial position was changed only from 1 m to 1.05 m (1.8 $ → 1.9 $), thermal-hydraulic behavior changed considerably. Good two-phase natural circulation ended and the bubbling mode began.

An important finding was also that even if one MFW pump is available, the situation is considerably improved because one MFW pump is capable to maintain steam generators liquid level except in the early phase of the accident. A good heat transfer to the secondary side ensures that the primary coolant mass inventory does not decrease so much that the pressure vessel liquid level would drop below the hot leg elevation.

REFERENCE

[1] R. Kyrki-Rajamäki, HEXTRAN: Three Dimensional Reactor Dynamics Code for VVER-reactor Cores. Proceedings of the International Topical Meeting on Advances in Mathematics, Computations and Reactor Physics. Pittsburgh, PA, USA, 28.4. - 2.5.1991

Parameter	Initial value
Pressure	**[MPa]**
Upper part of pressurizer	12.30
Hot leg	12.33
Cold leg	12.69
Steam generators	4.71
Steam header	4.71
Pressure difference	**[MPa]**
Pressure vessel inlet - exit	0.35
Across core	0.23
Across reactor coolant pump	0.46
Temperature	**[°C]**
Hot leg	261.1
Cold leg	260.9
Mass flow rate	**[kg/s]**
Downcomer	8709
Core	7710
Core bypass	11.5 %
Steam flow	8.9
Miscellaneous	
Pressurizer liquid level	3.88 m
Steam generator liquid level	2.22 m
Core power	13.84 MW
Steam generators power	16.46 MW
Average linear power	1.44 W/cm
Maximum linear power	3.25 W/cm

Table 2-1 Initial value of the main parameters

Case	Position of the control rod [cm]	Reactivity worth [$]
1	200	3.6
2	200	4.0
3	122	2.2
4	78	1.4
5	100	1.8
6	105	1.9

Table 2-2 Initial control rod position and reactivity worth

Reactivity worth [$]	CPU time [s] $0 \rightarrow 300$	Average time step size [ms]	CPU time [s] $300 \rightarrow 1800$	Average time step size [ms]
3.6	1893	30.33	2913	99.86
2.2	2149	27.95	3073	99.91
1.8	2049	28.70	3066	99.97
1.4	1442	40.00	2976	100.00
1.9	1969	28.72	2961	99.97
1.9 Var. 1	2009	29.65	3086	100.00
1.9 Var. 2	2110	27.25	3006	99.95

Table 5-1 Run time statistics

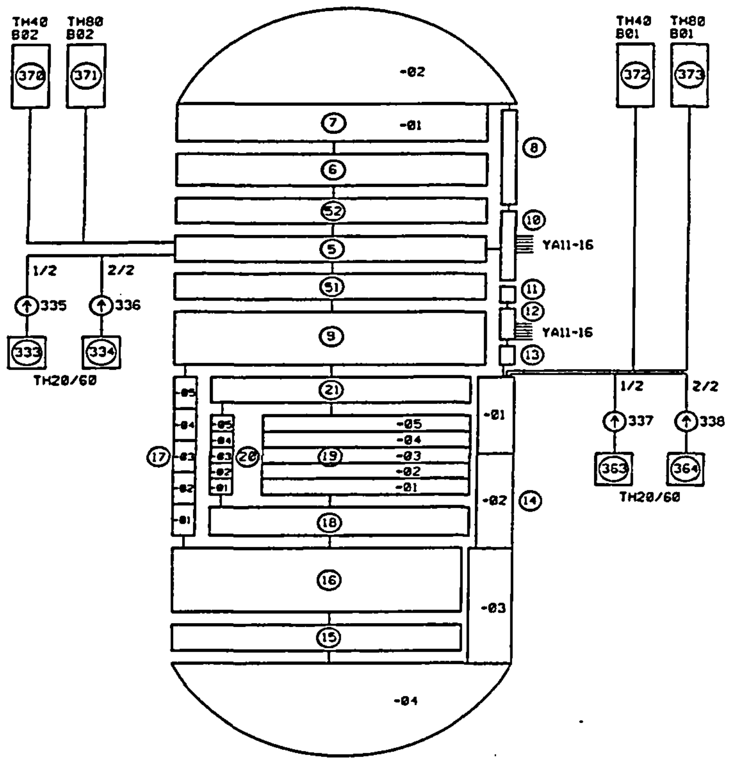


Figure 2-1   Pressure vessel nodalization

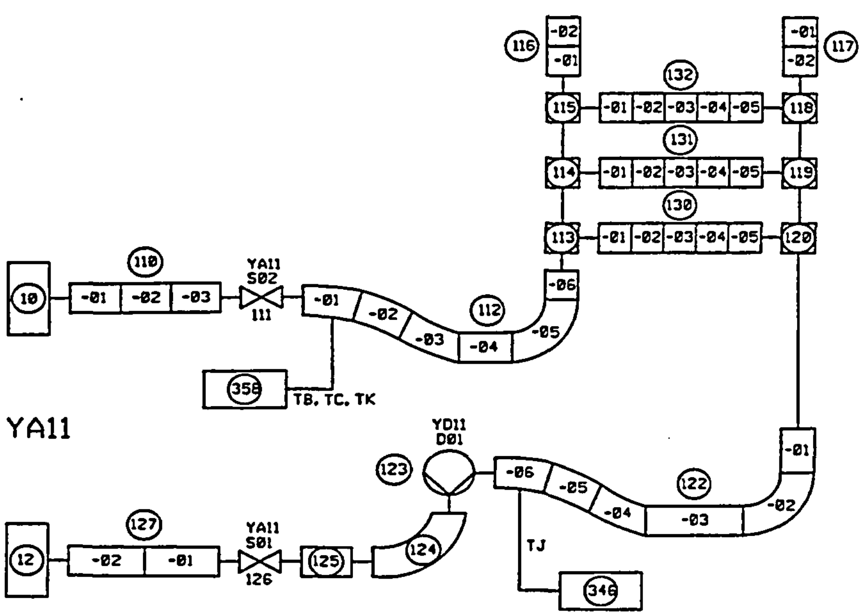


Figure 2-2   Loop YA11 nodalization

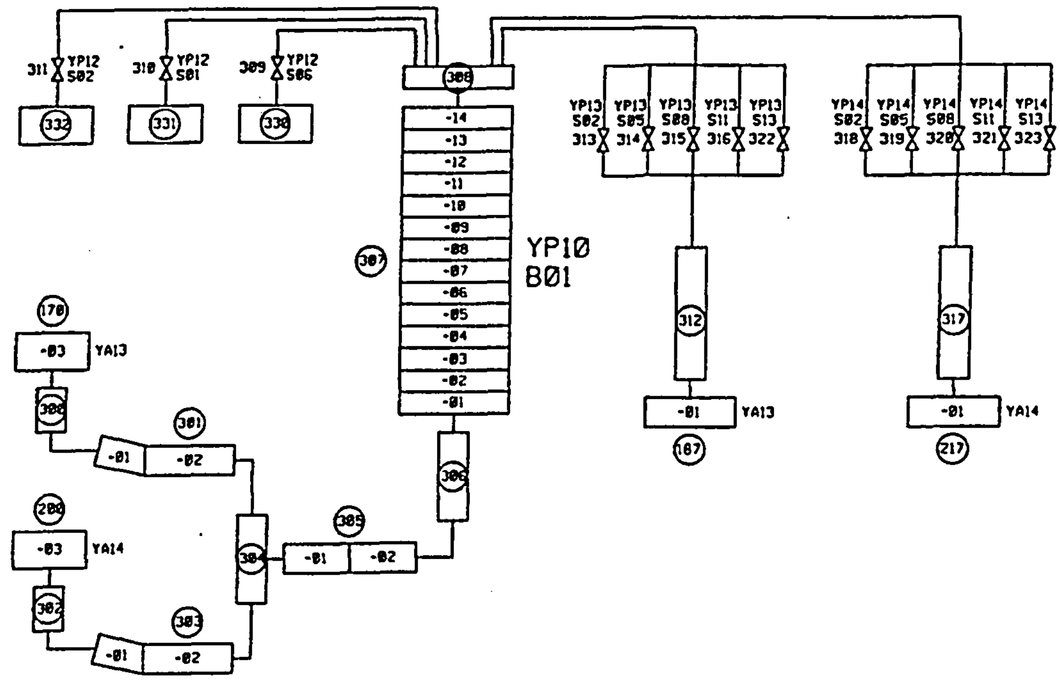


Figure 2-3 Pressurizer nodalization

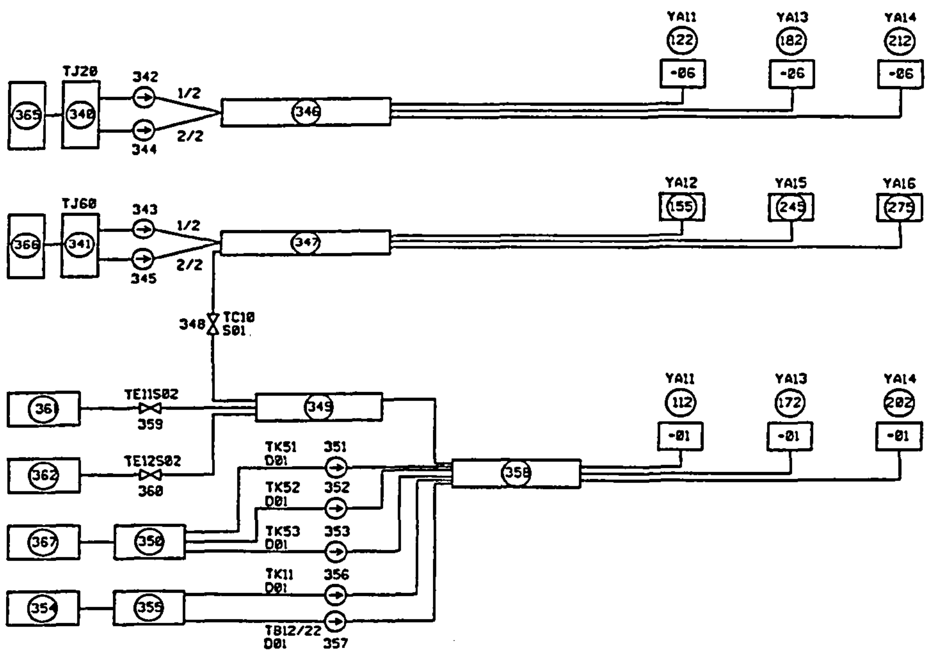


Figure 2-4 HPI, make-up, letdown and purification systems nodalization

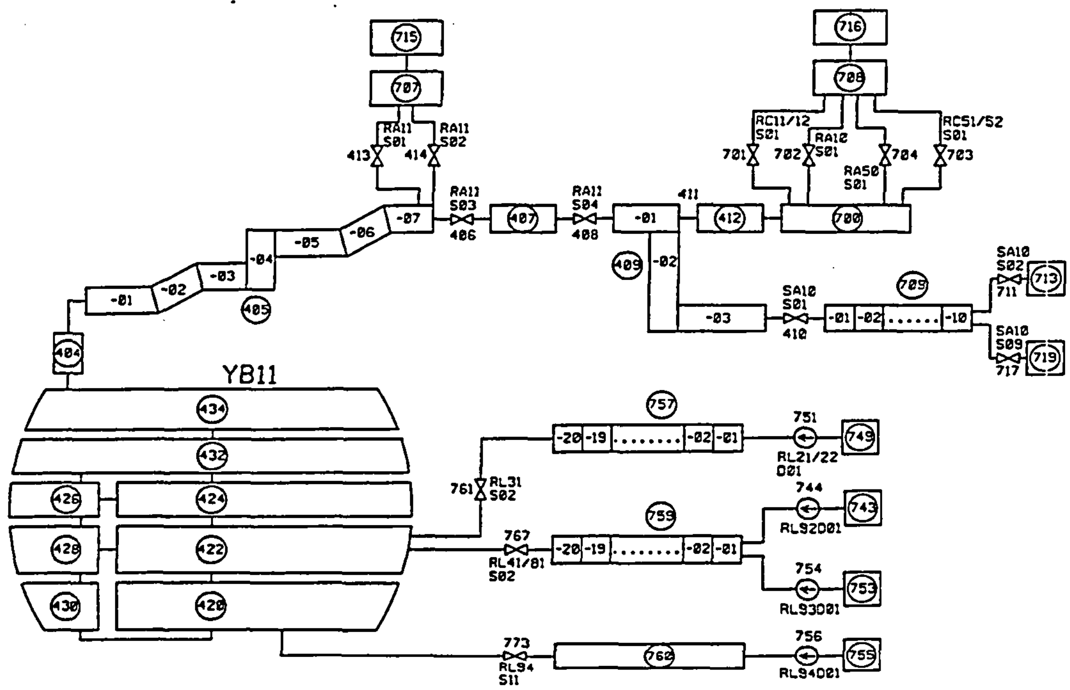
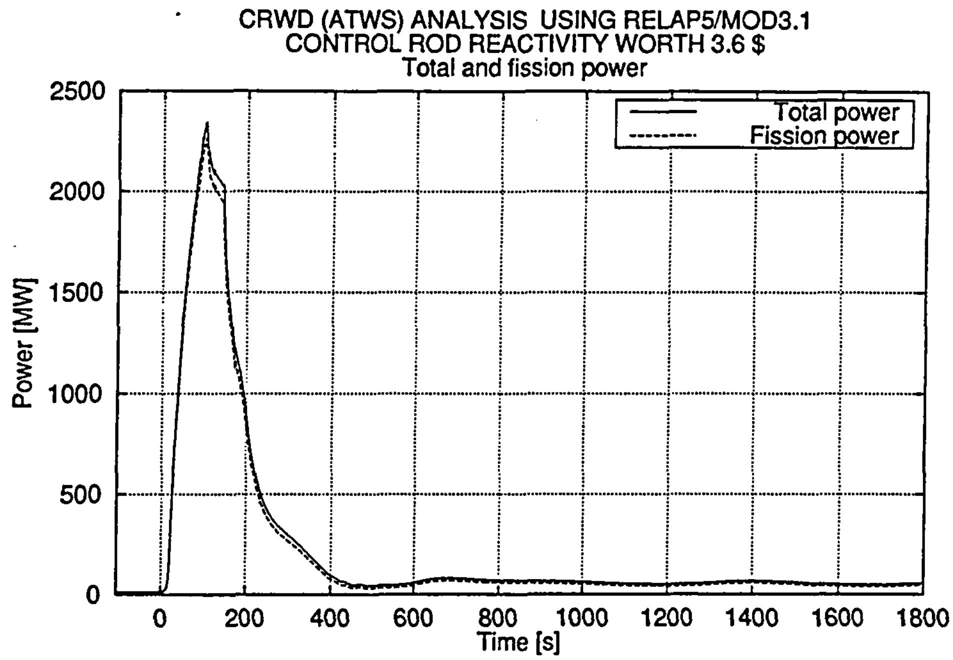


Figure 2-5 Steam generator YB11, steam line and feedwater injection system nodalization



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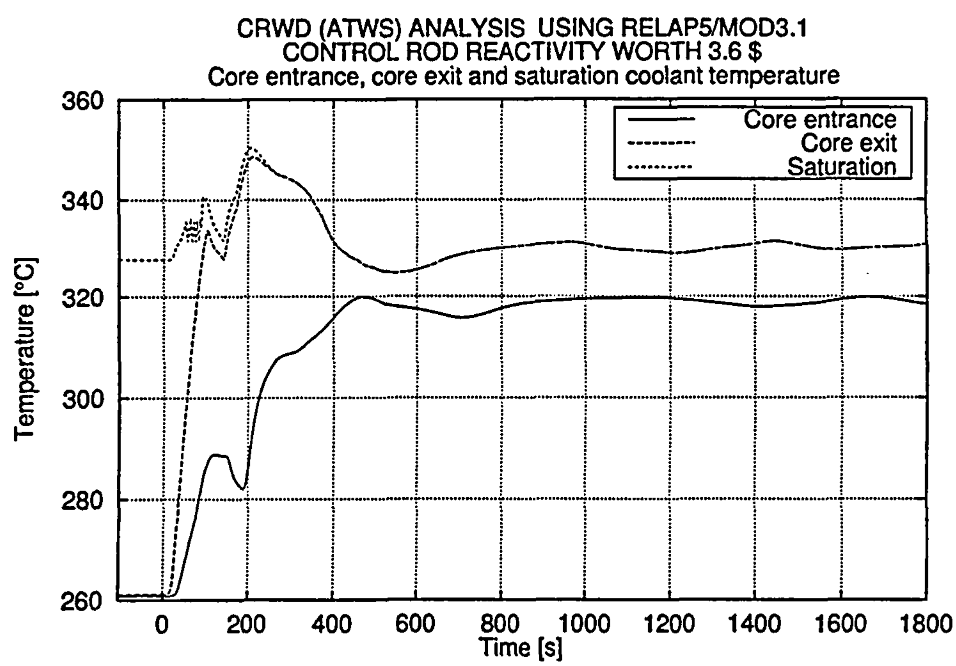
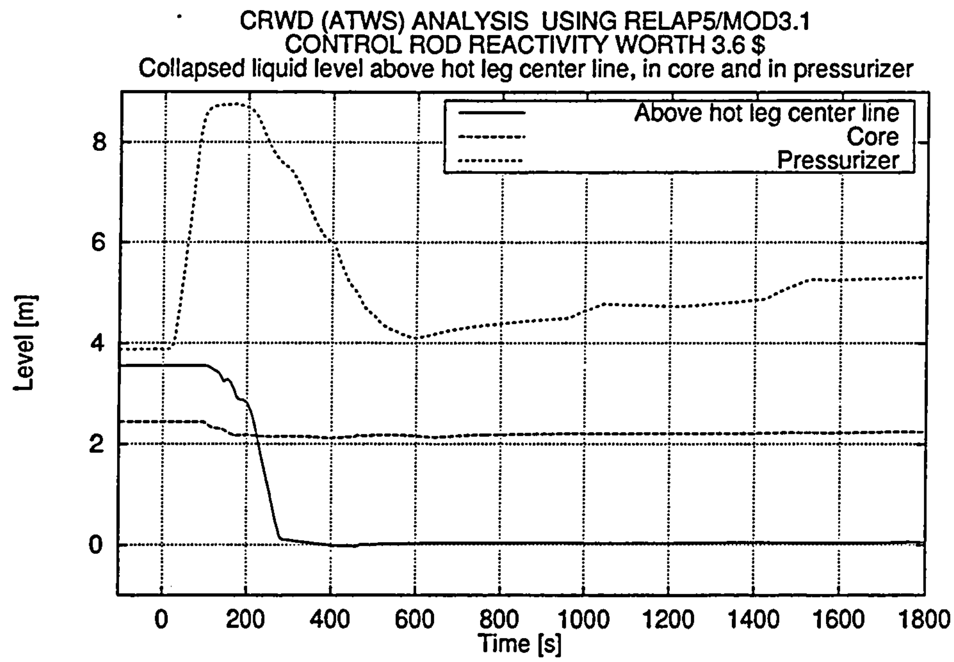


Figure 4.1-2



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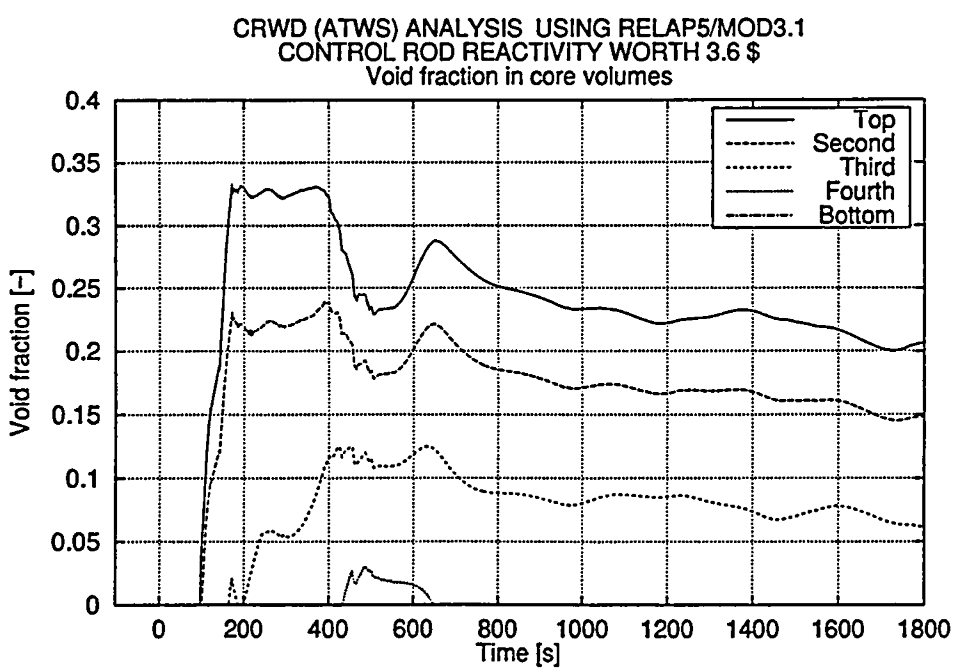

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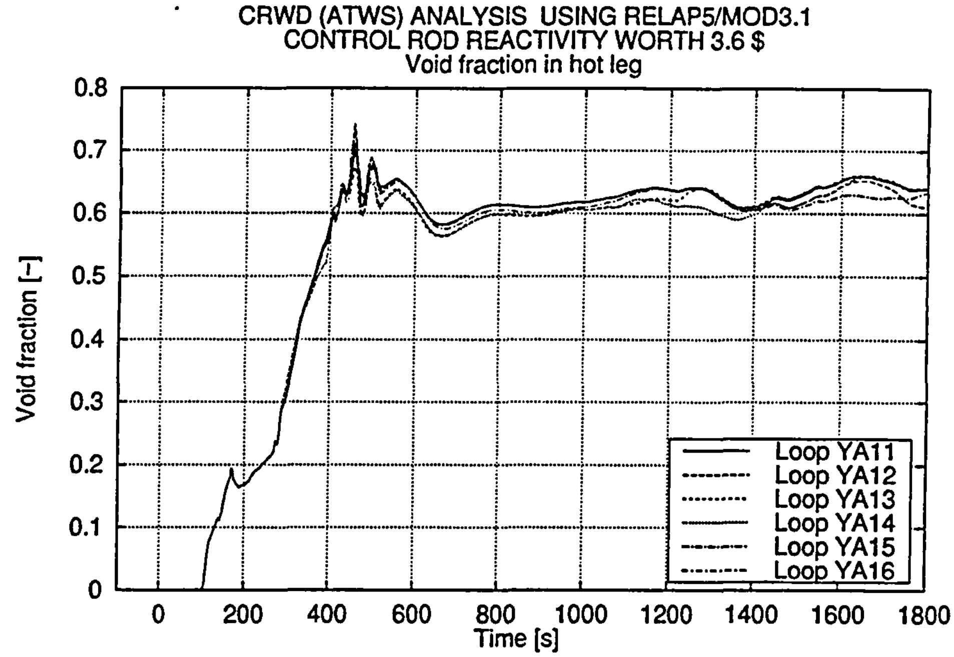

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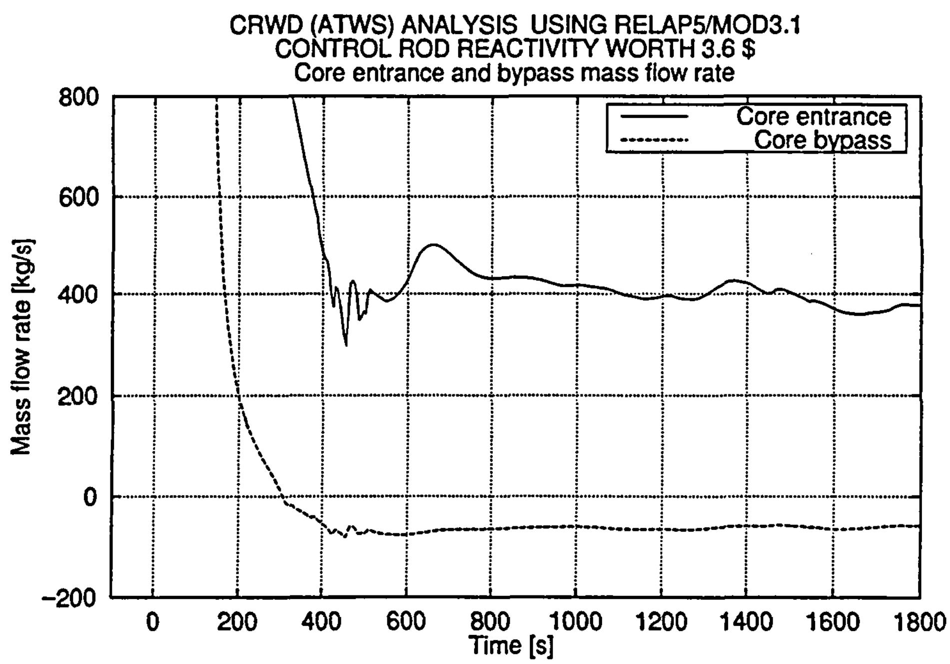

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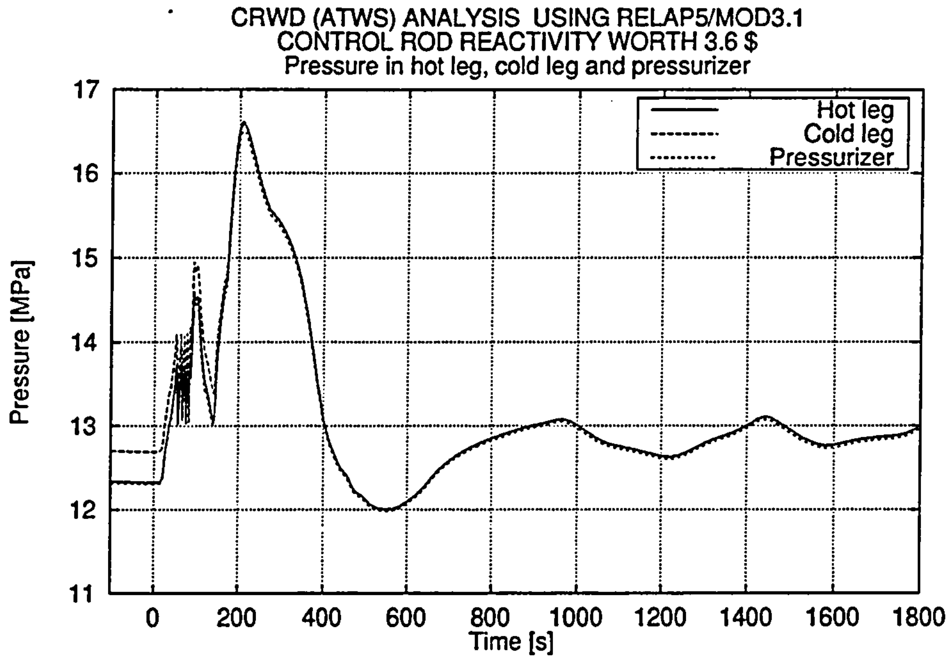

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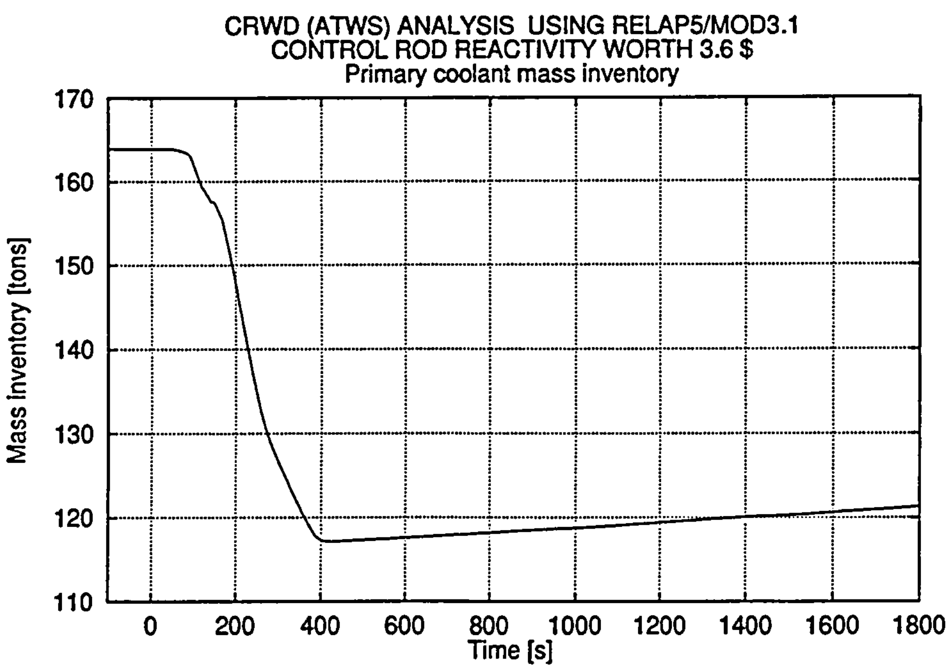

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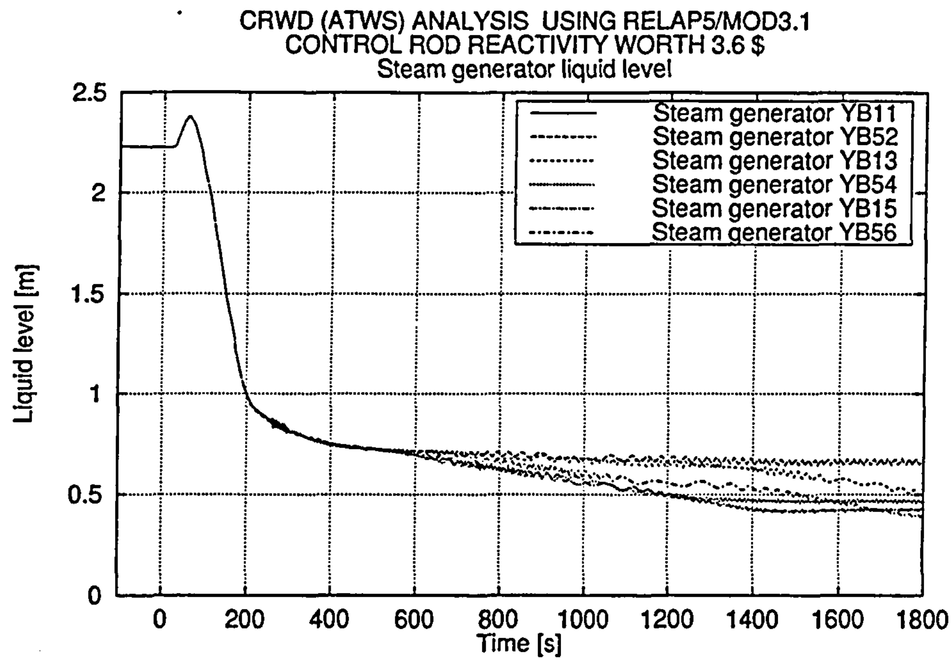

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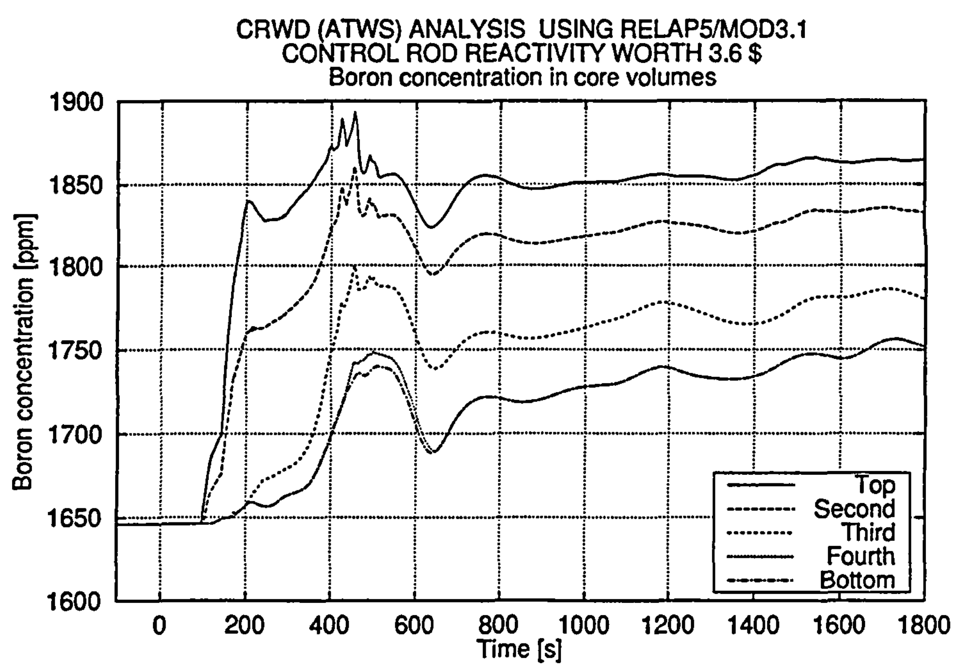

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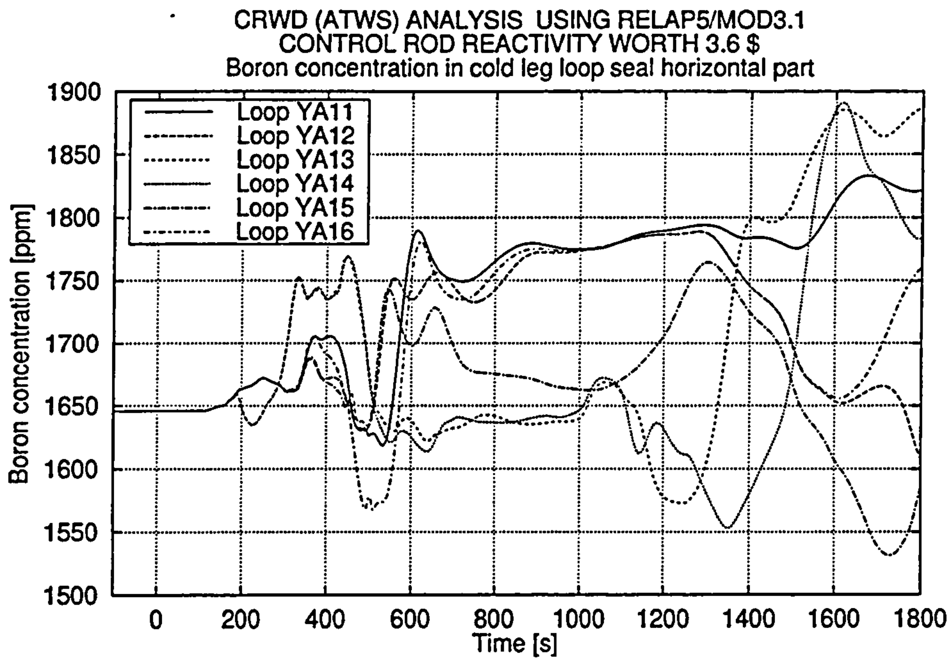

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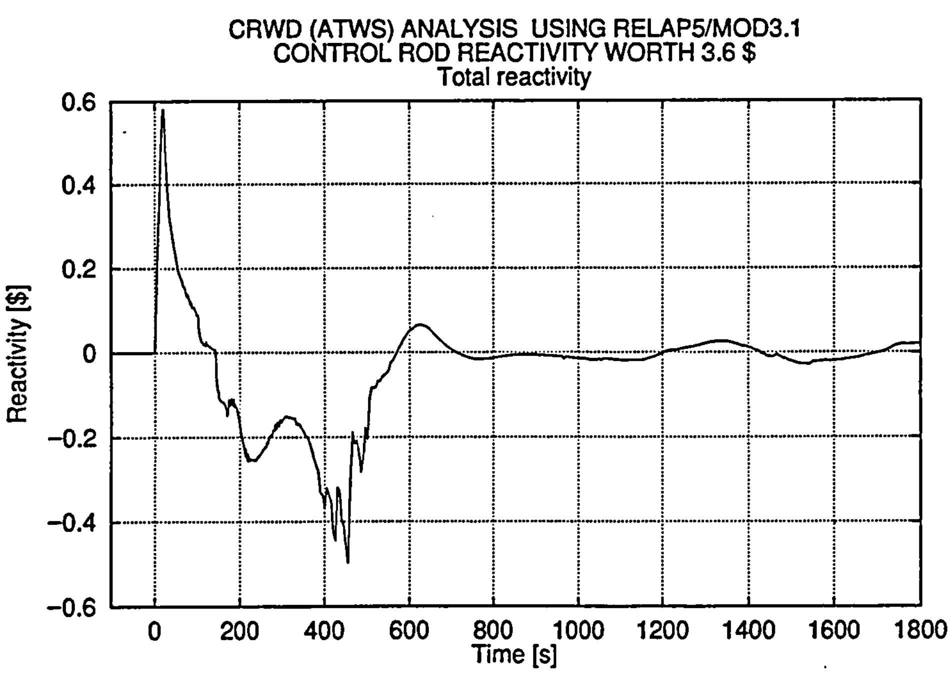

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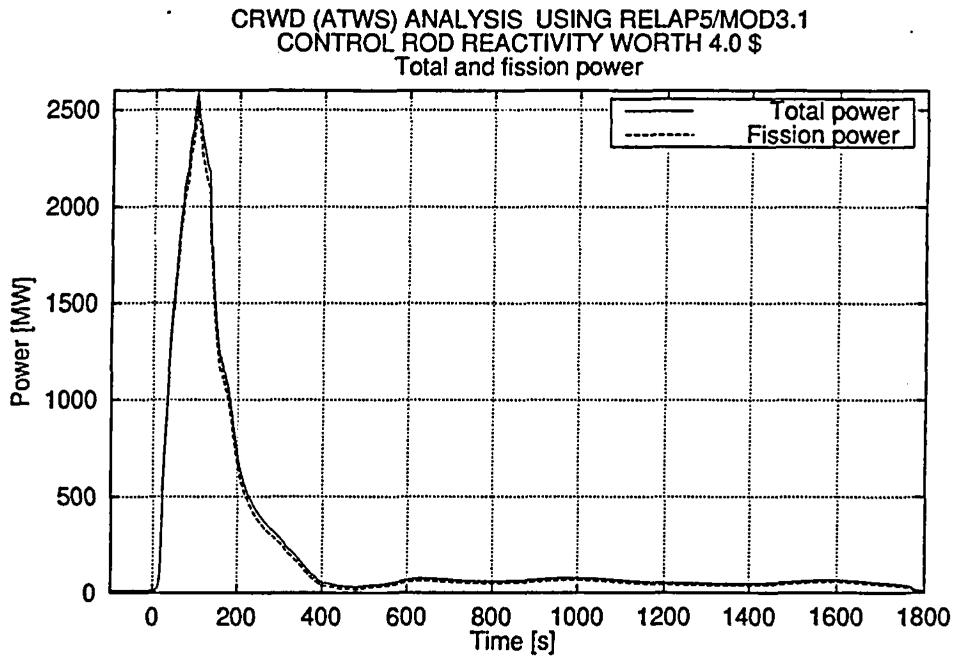

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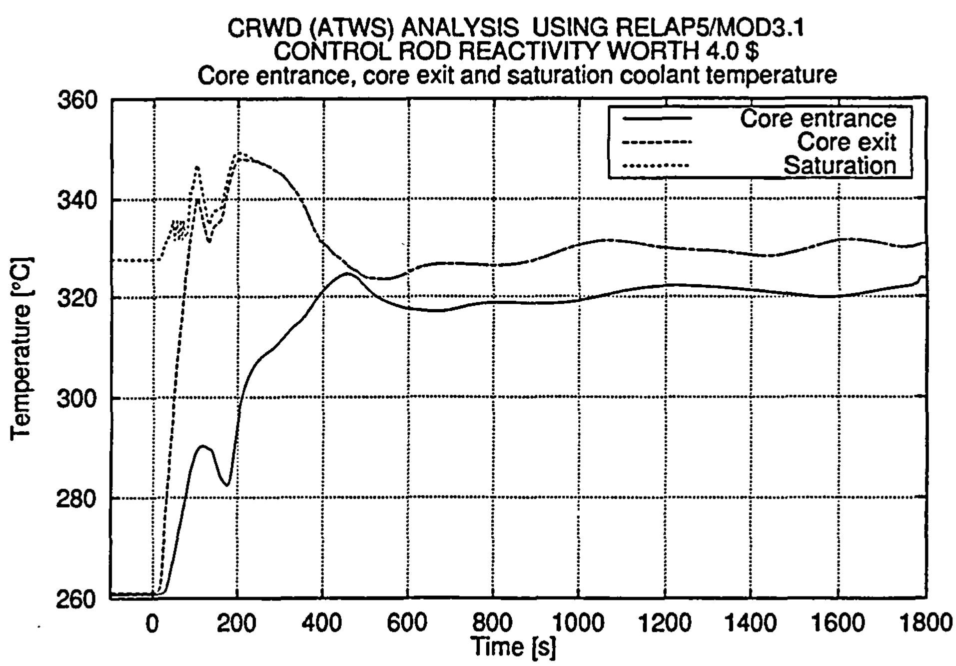

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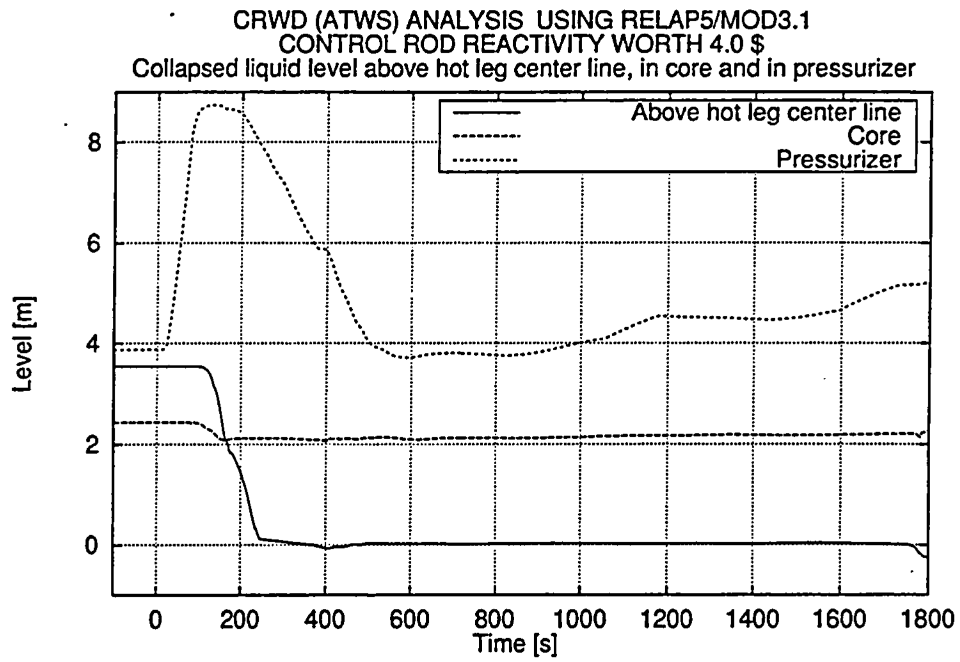

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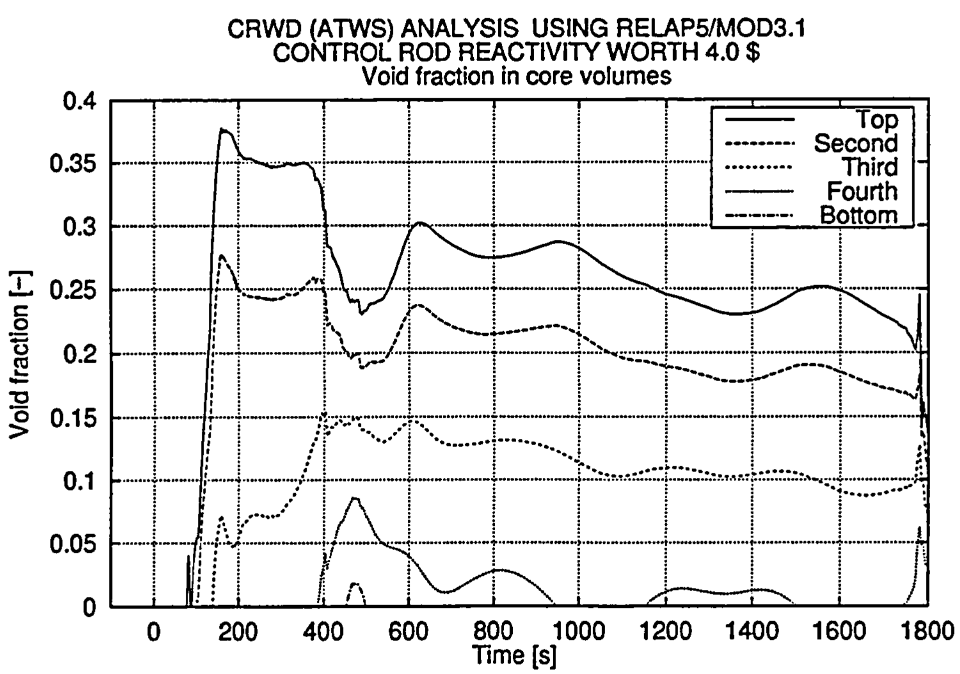

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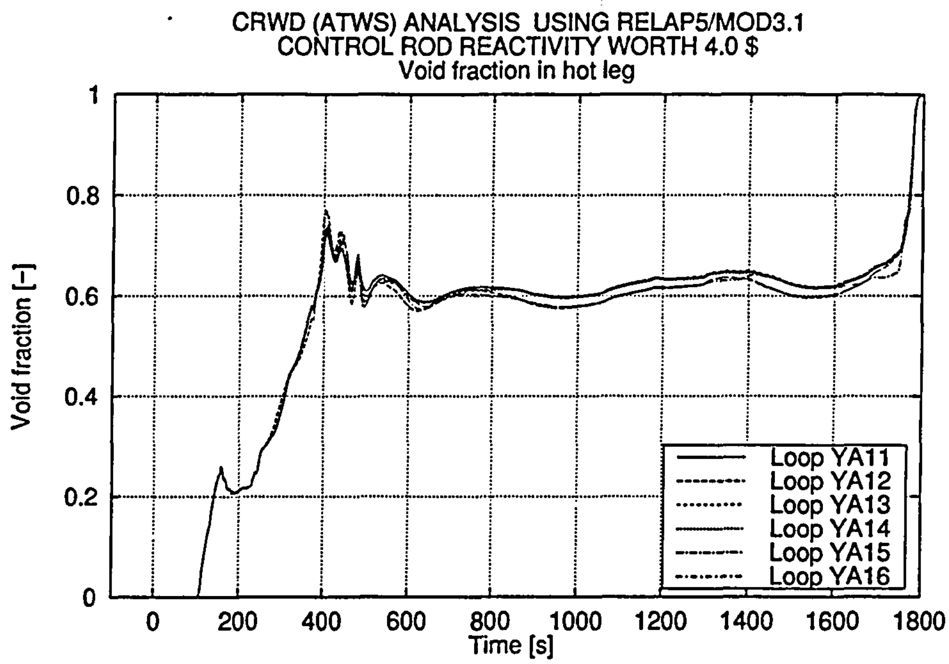

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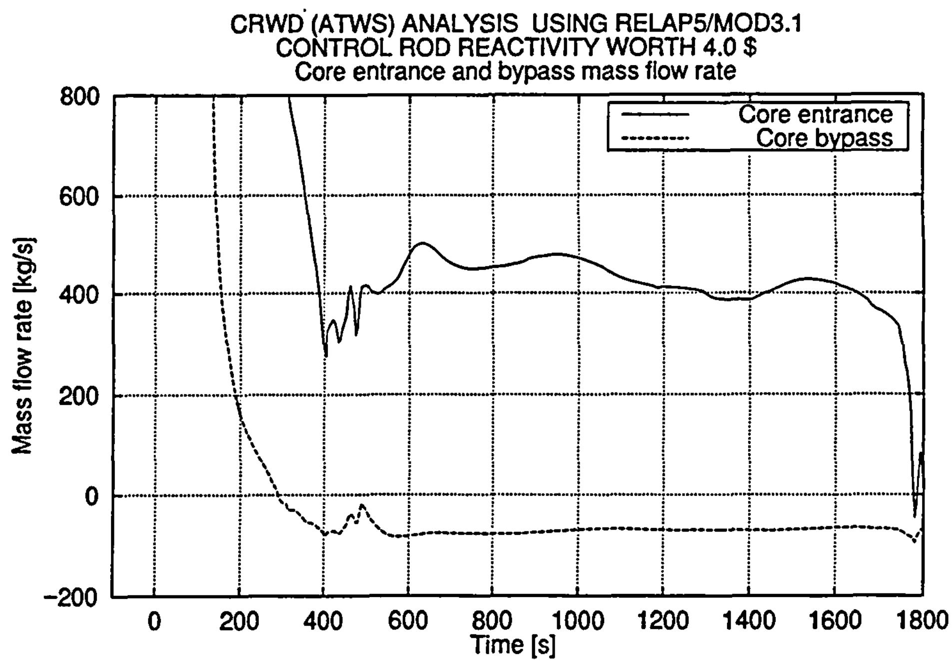

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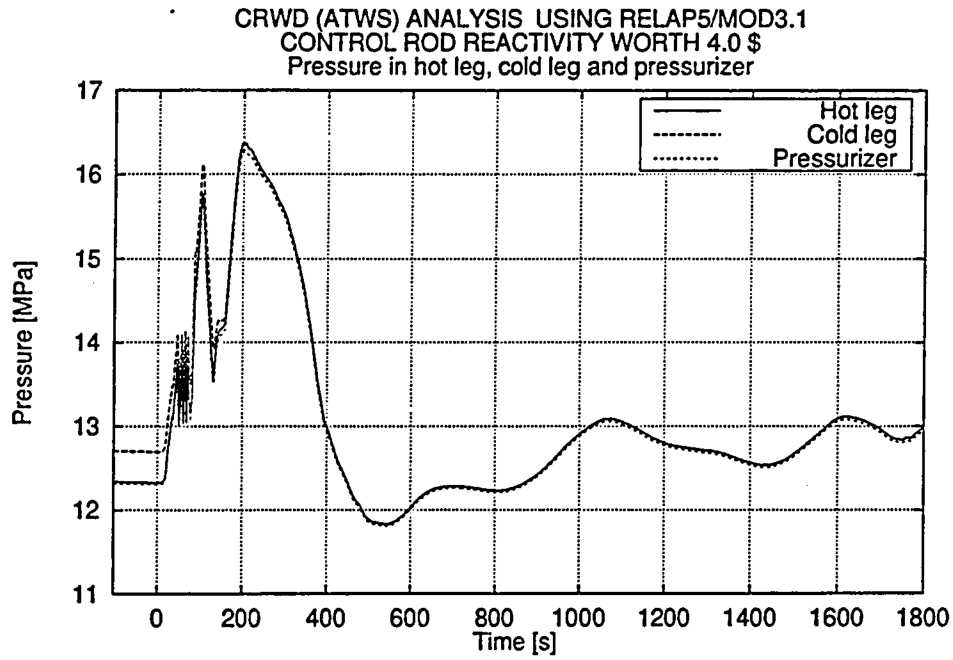

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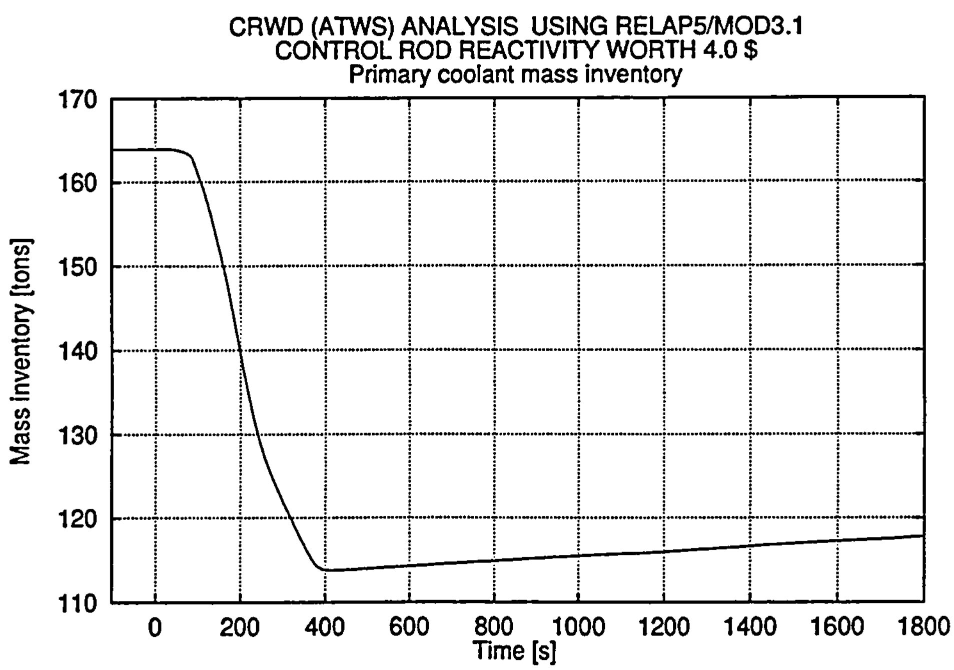

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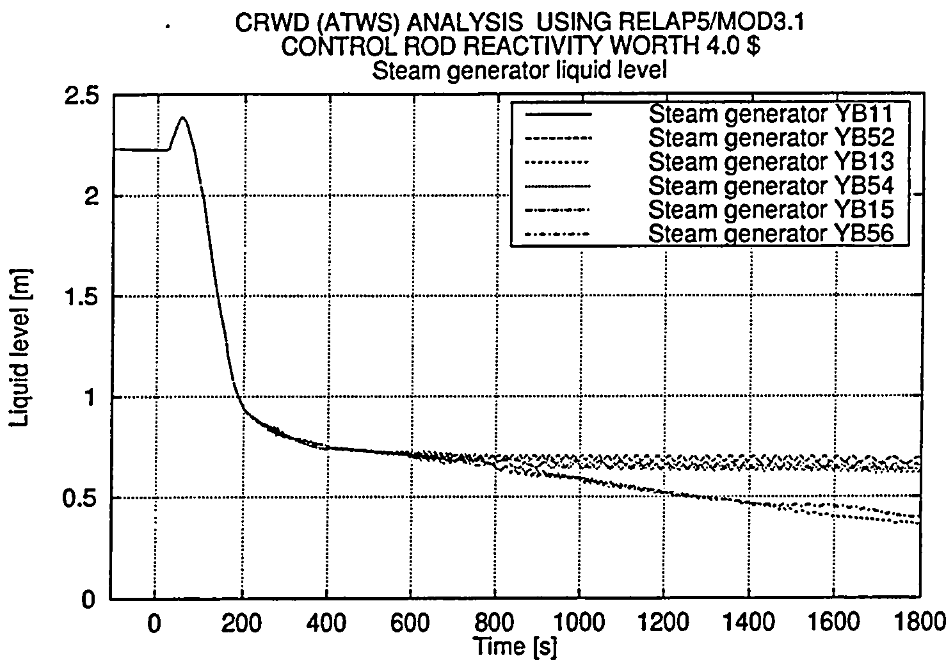

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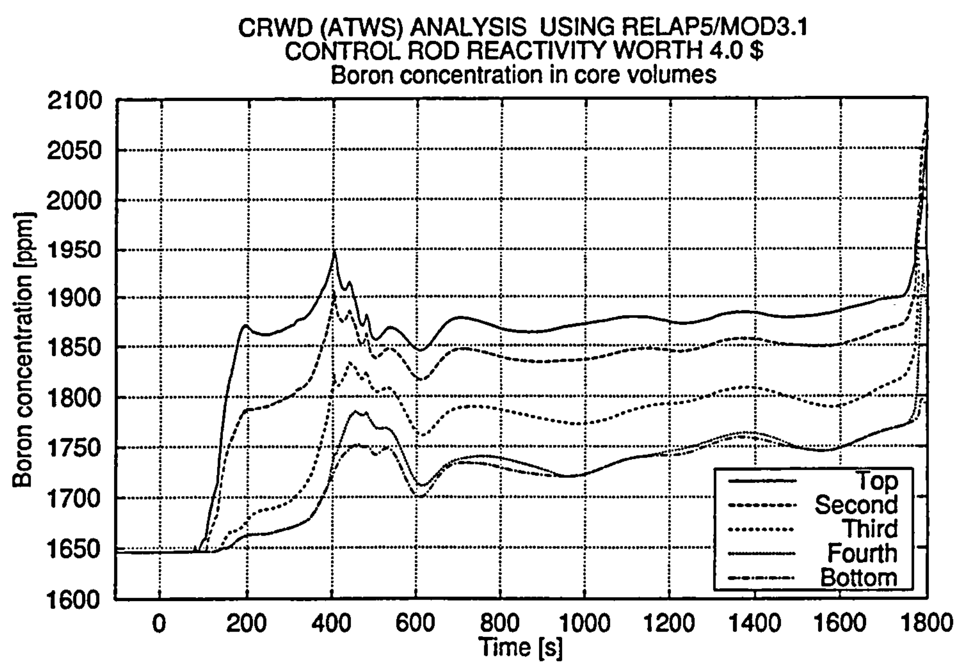

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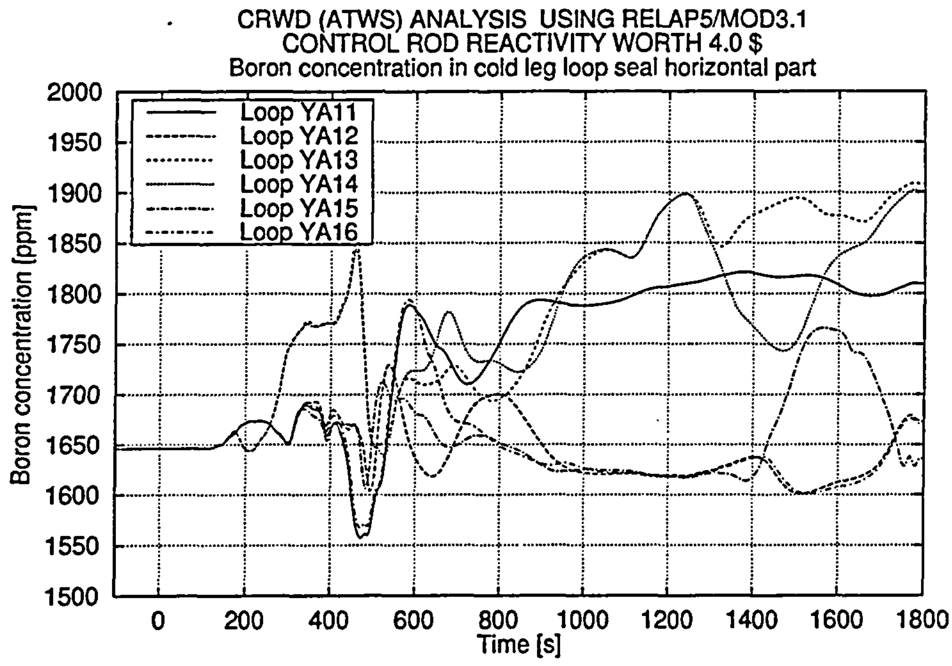

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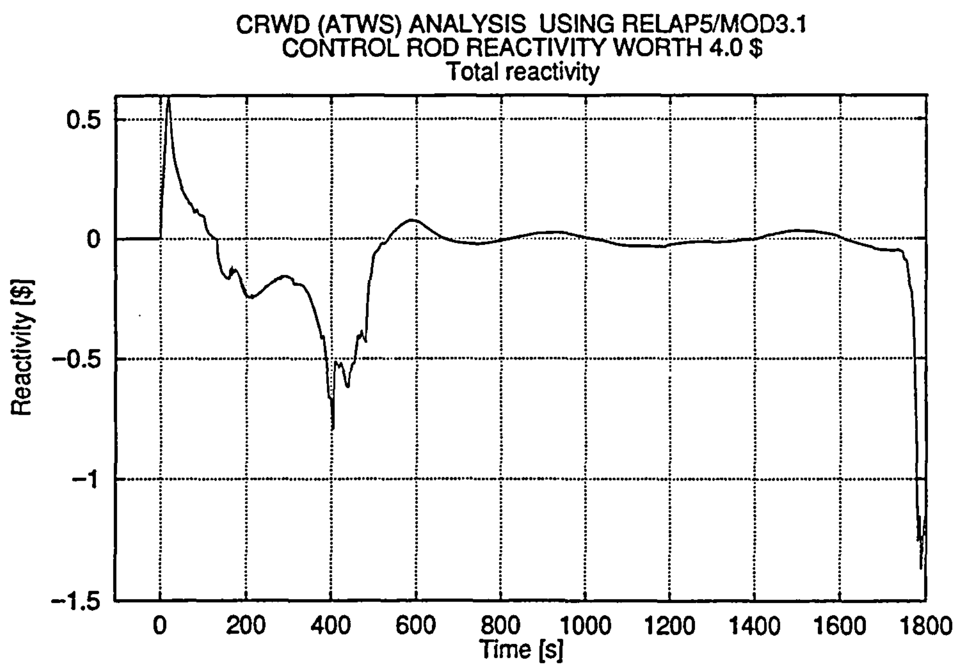

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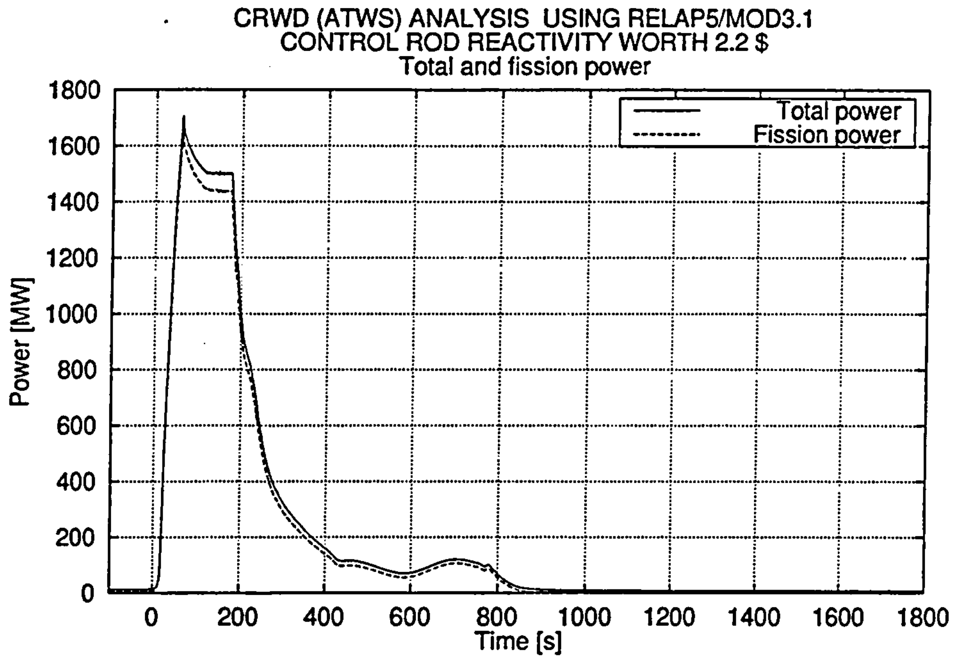

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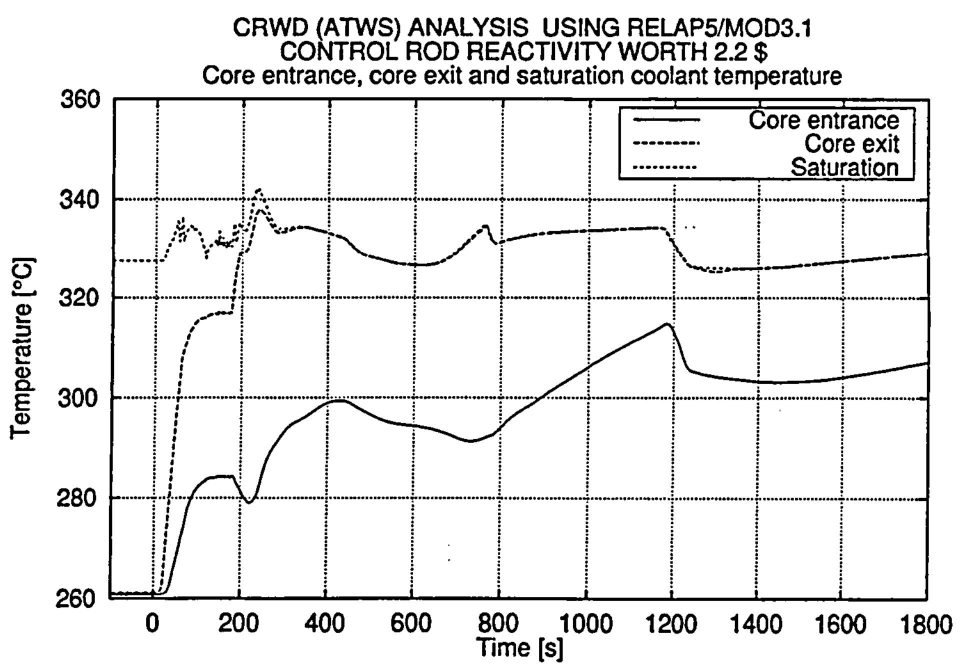

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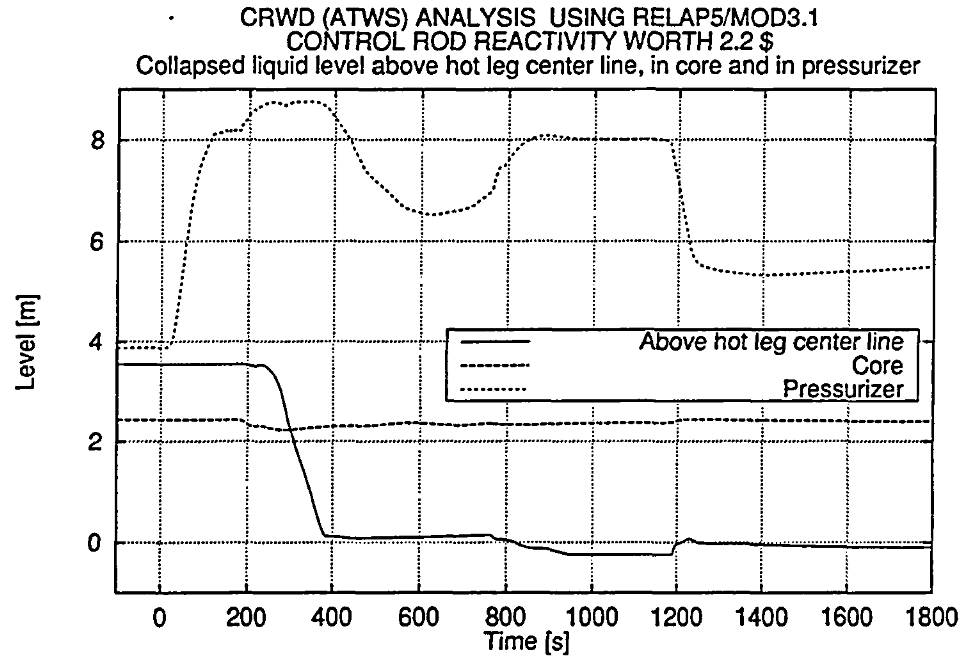

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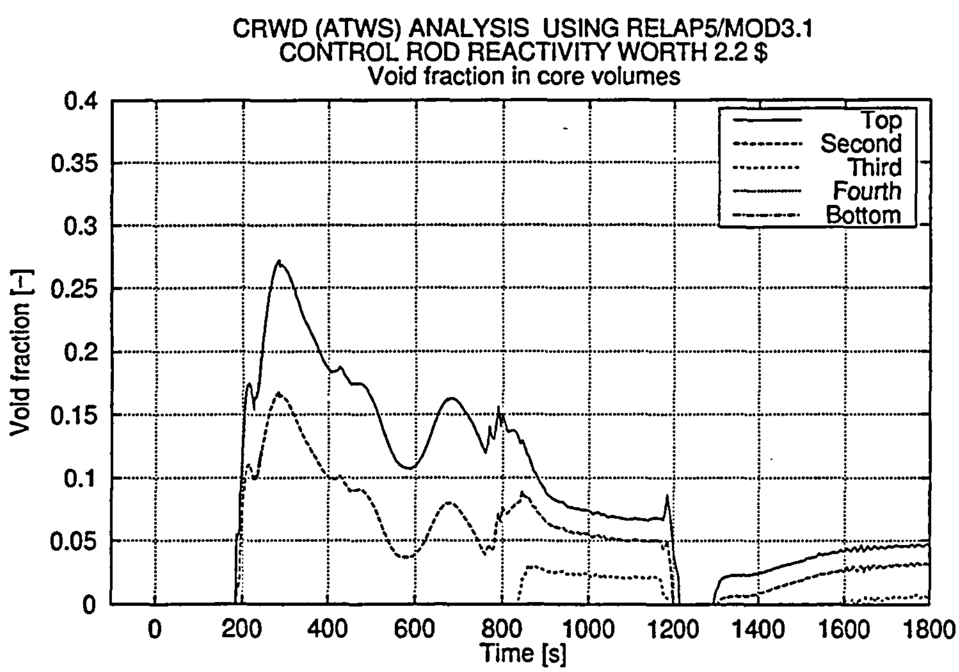

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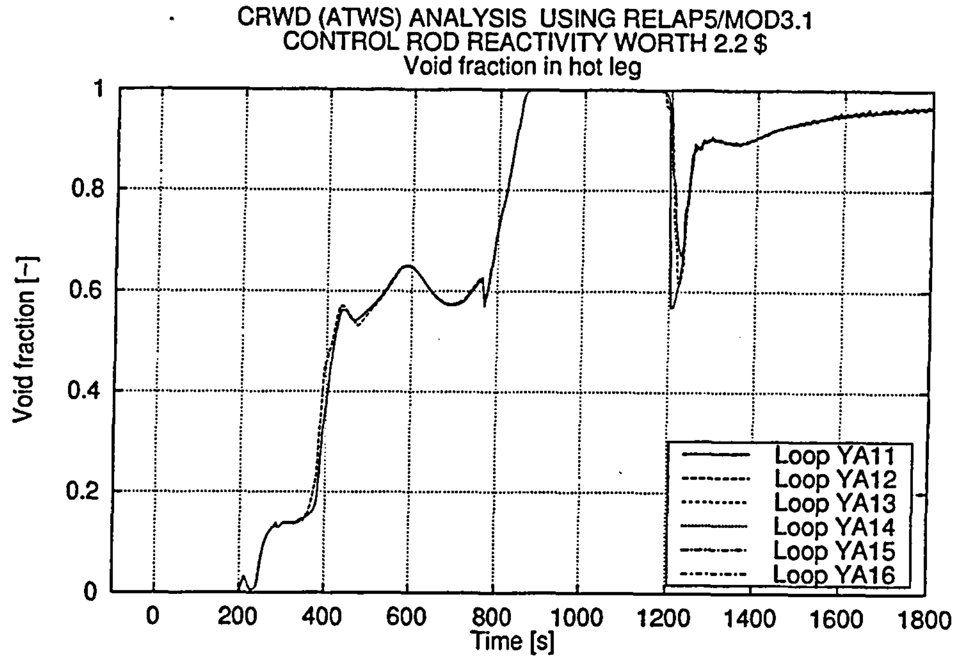

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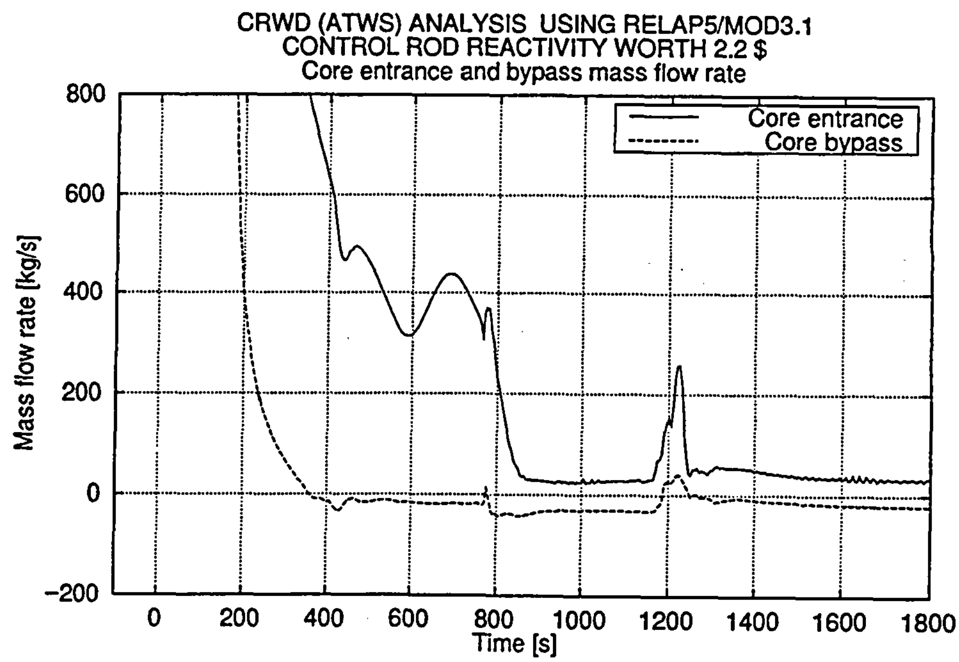

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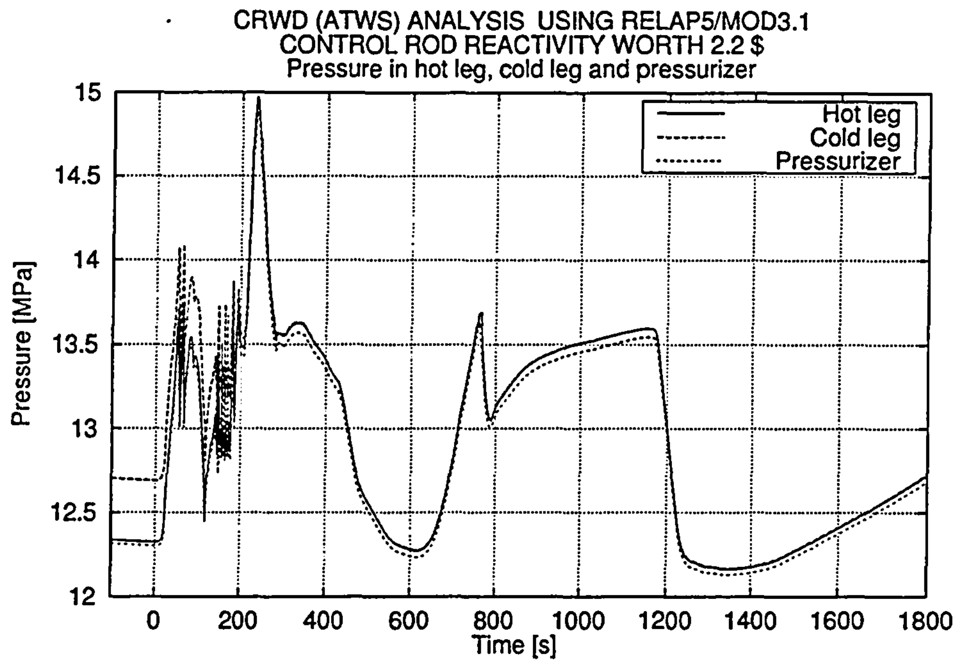

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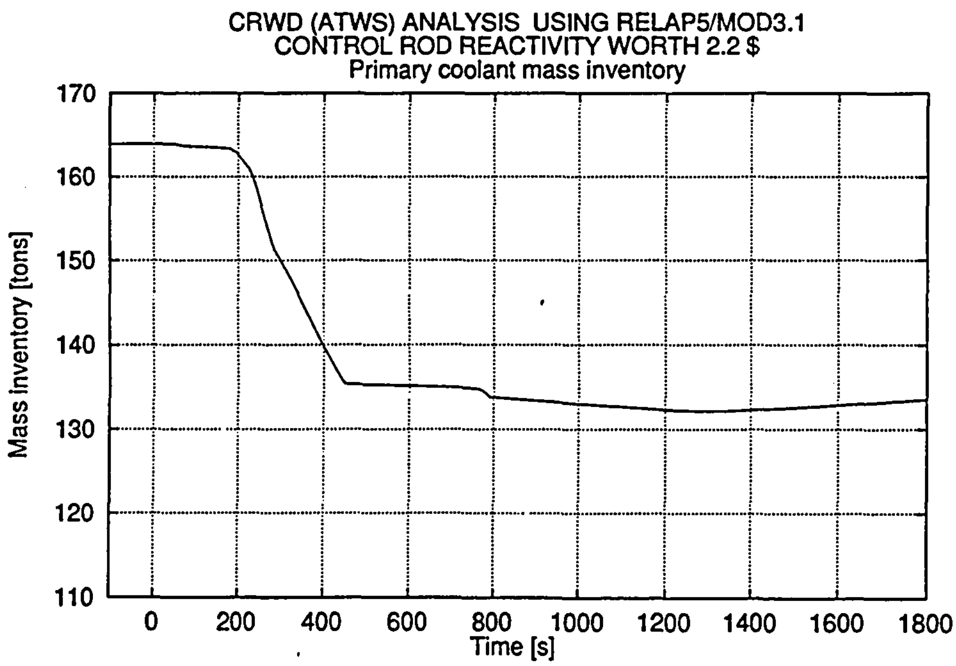

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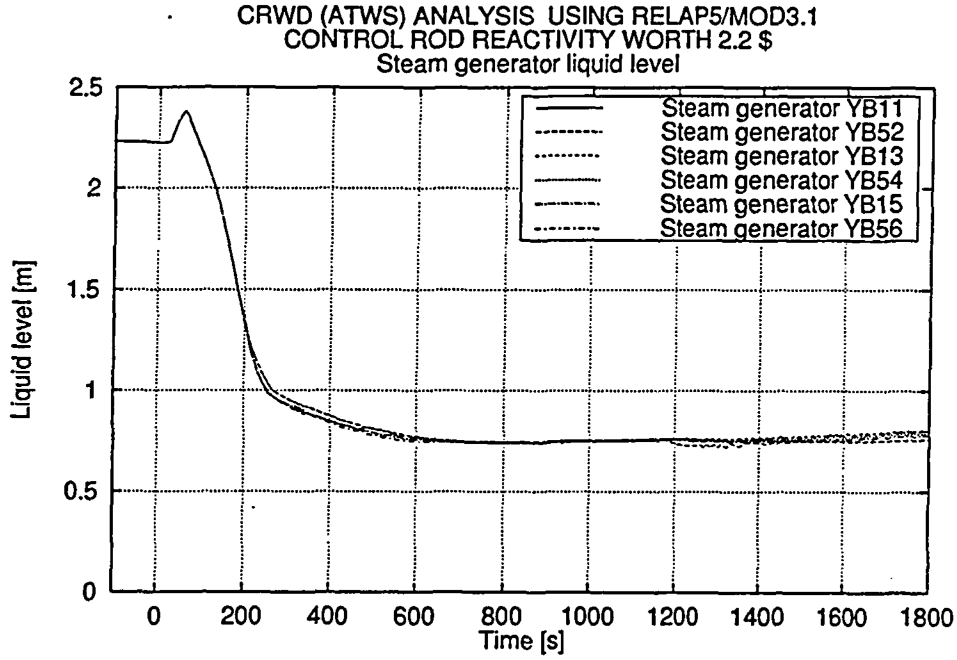

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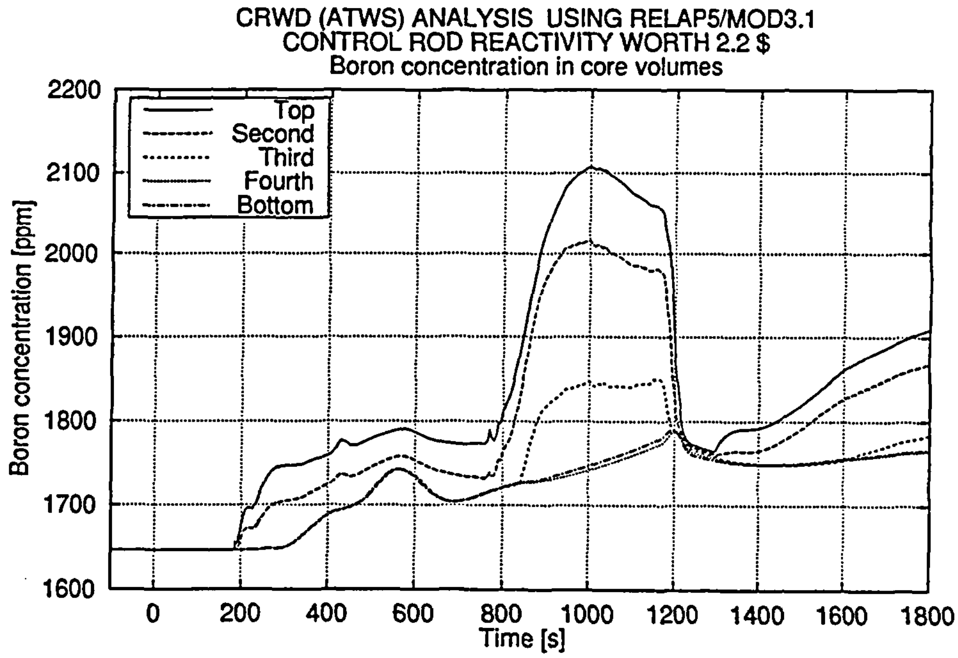

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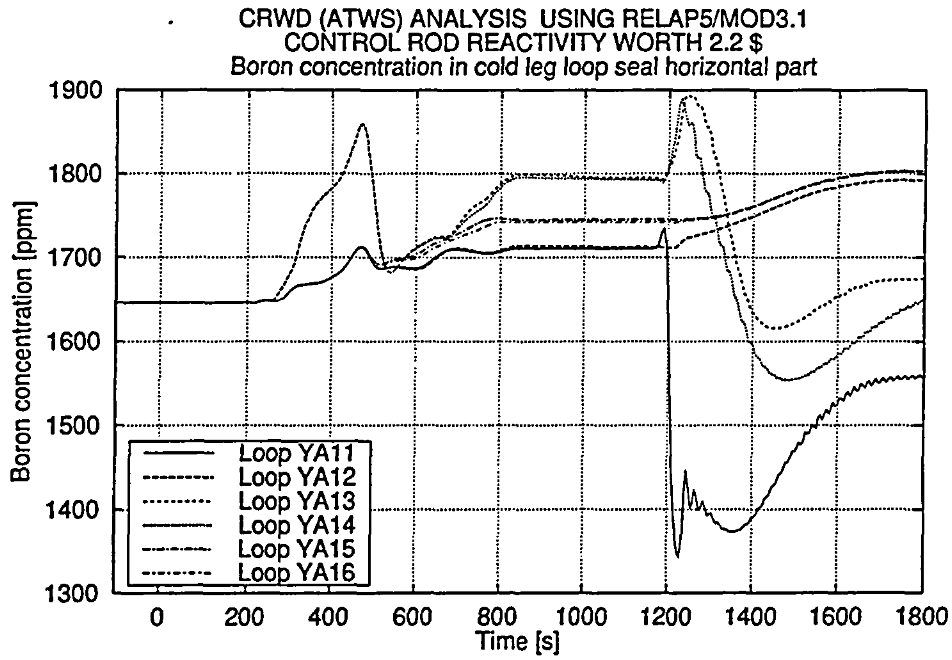

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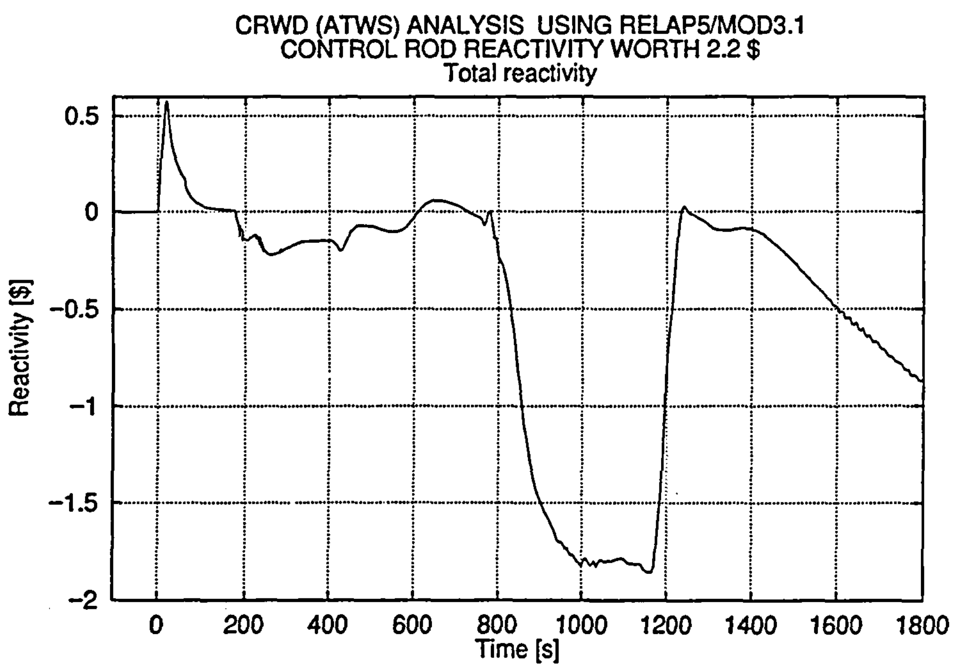

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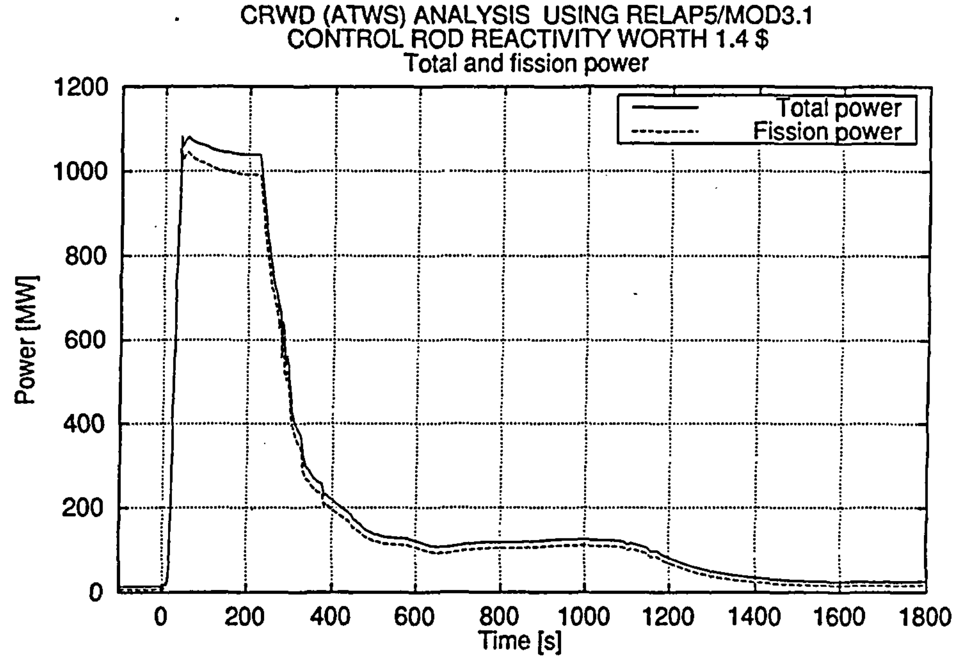

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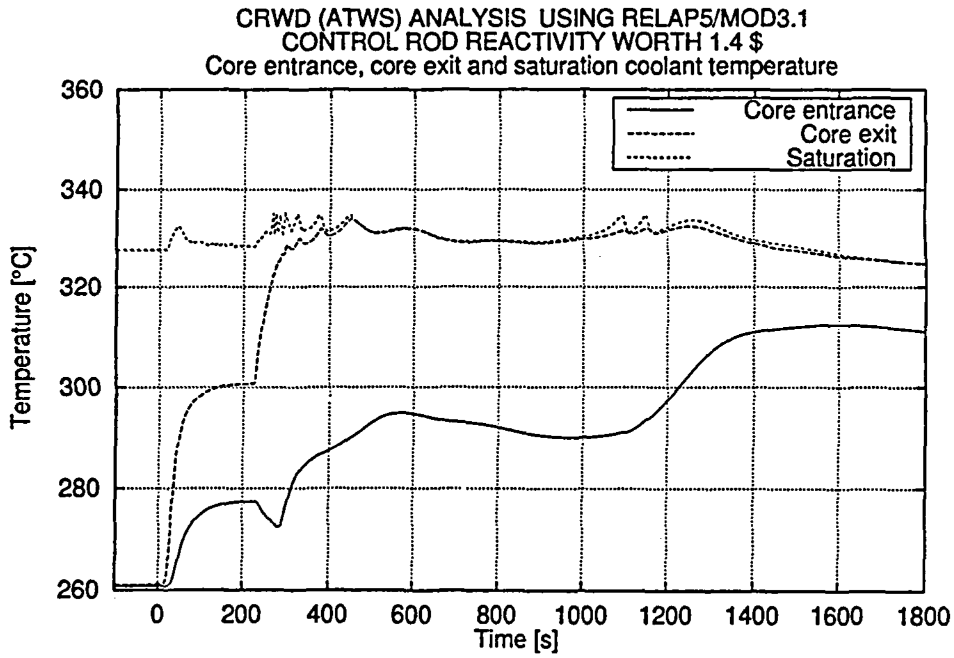

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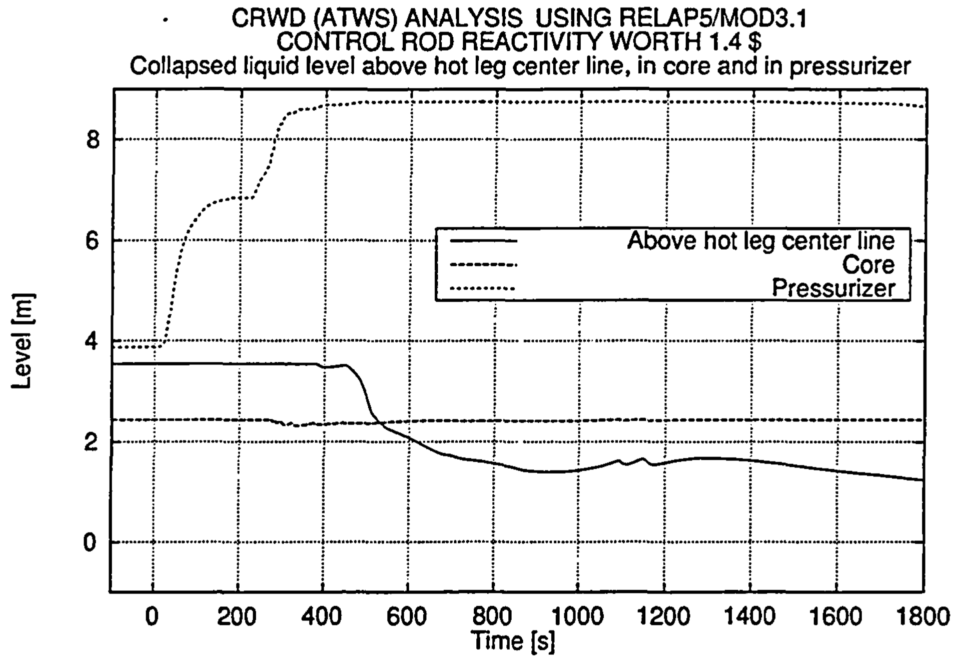

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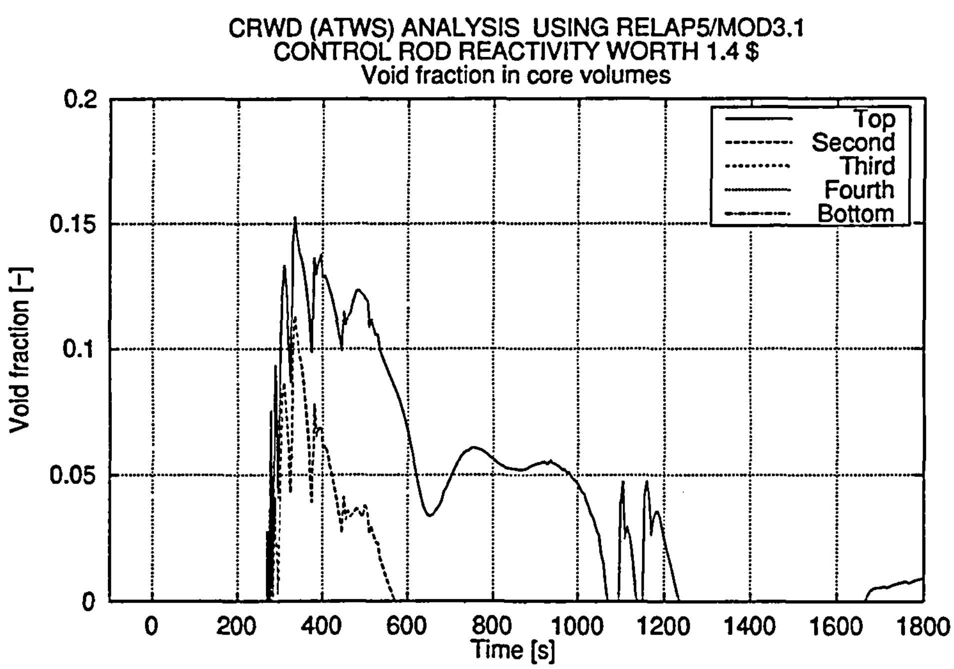
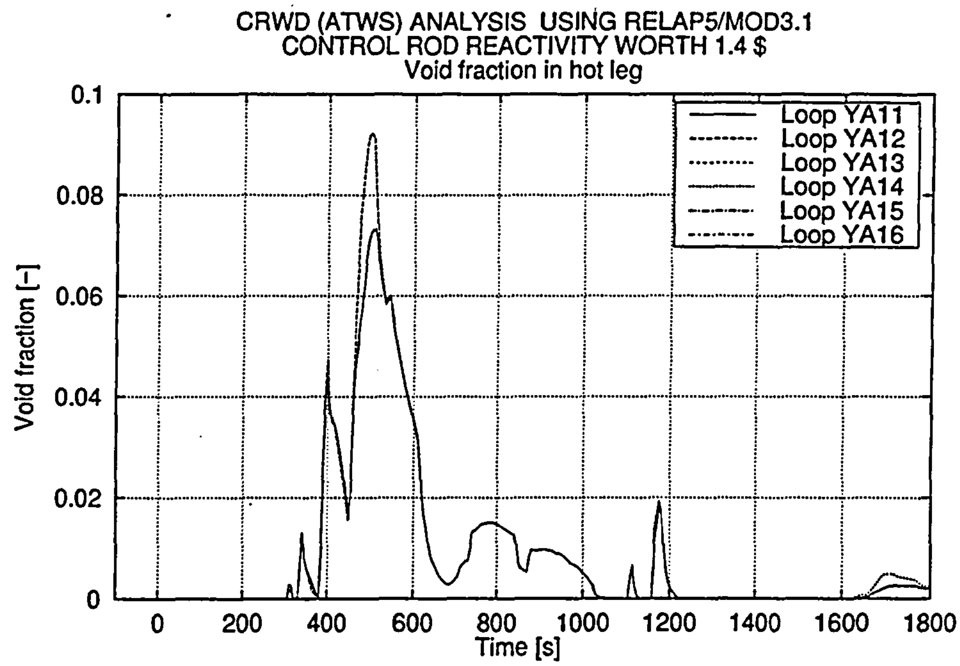


Figure 4.4-4



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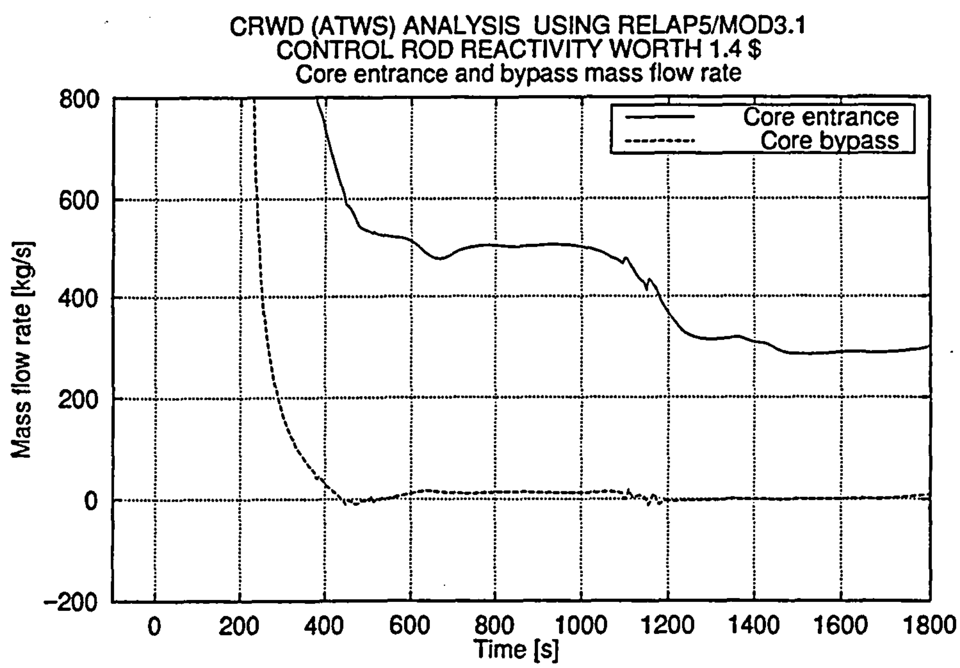

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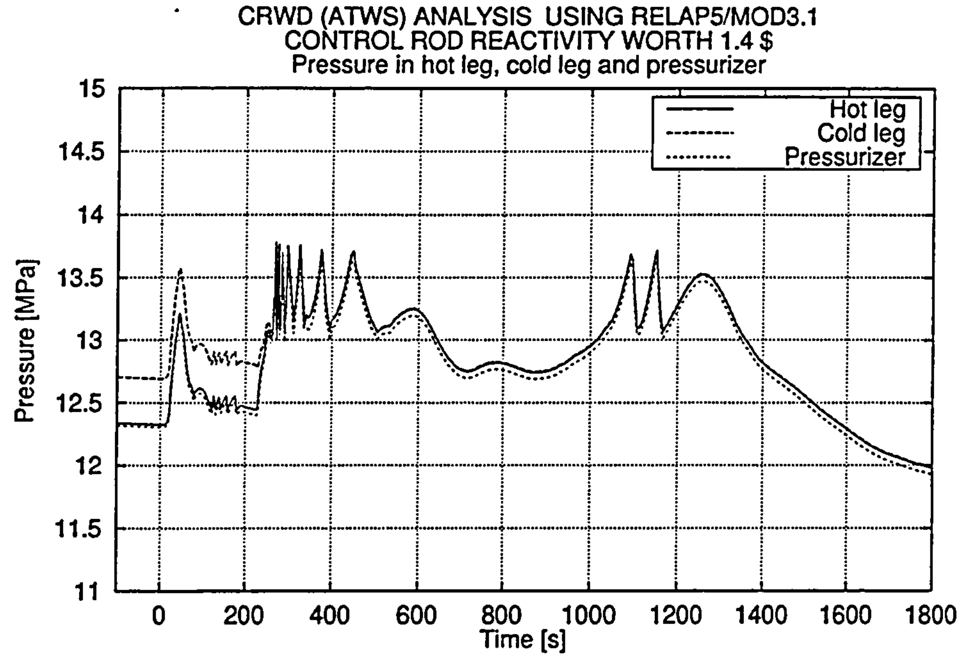

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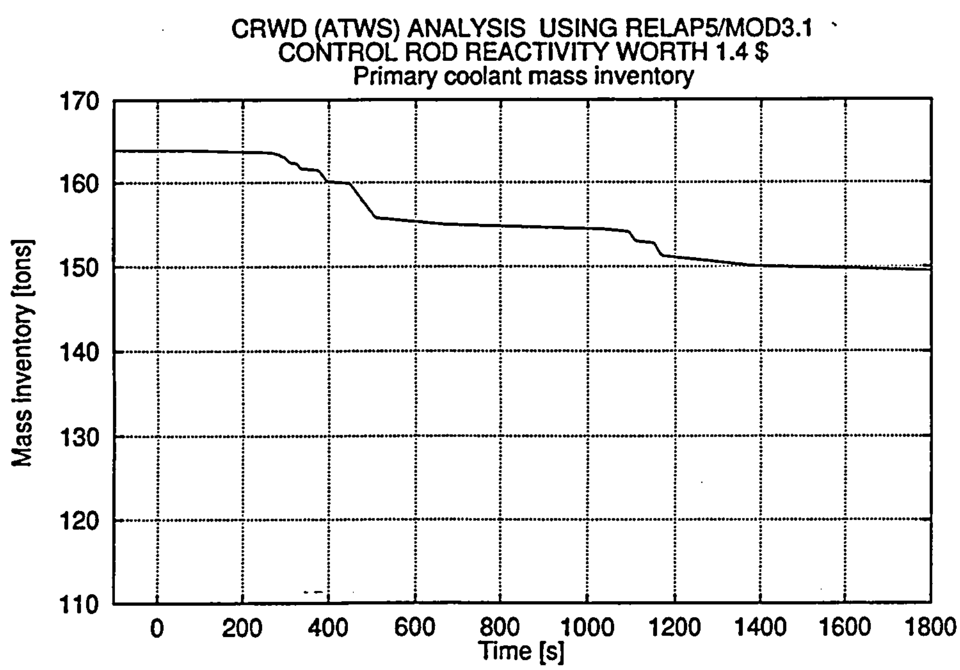

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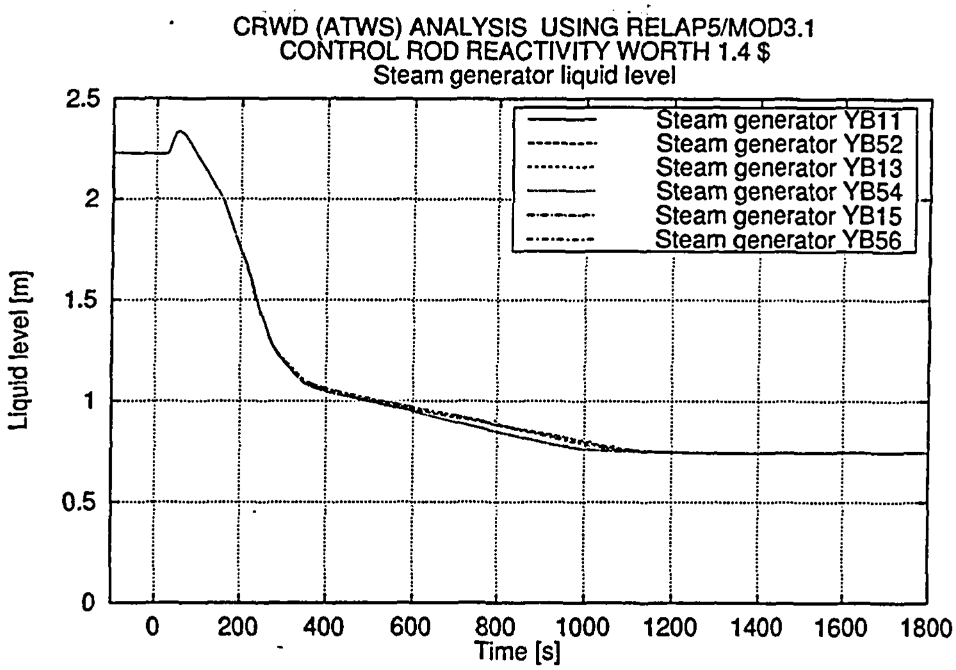

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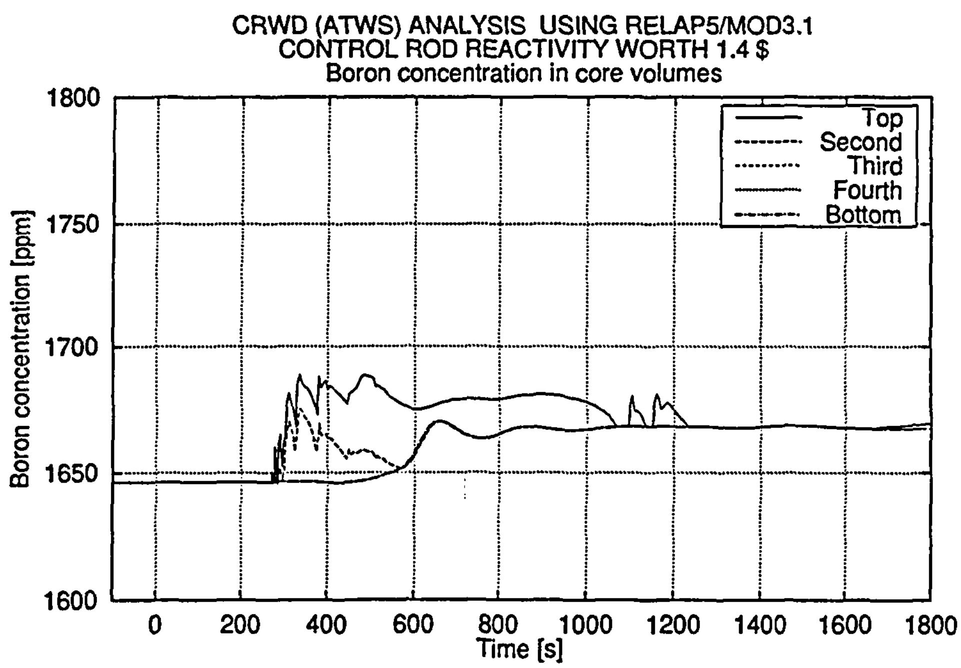

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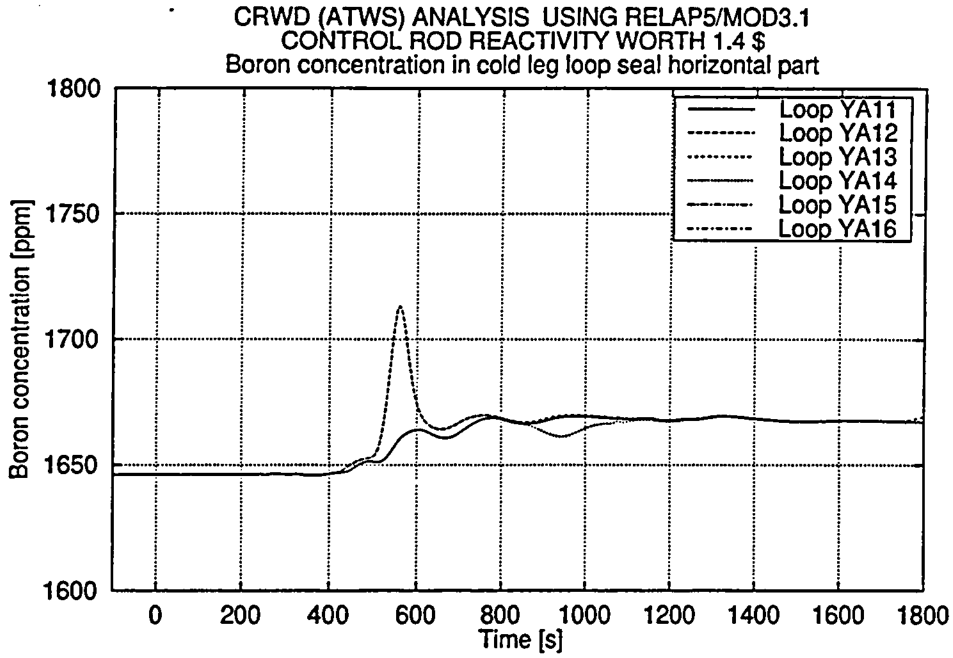

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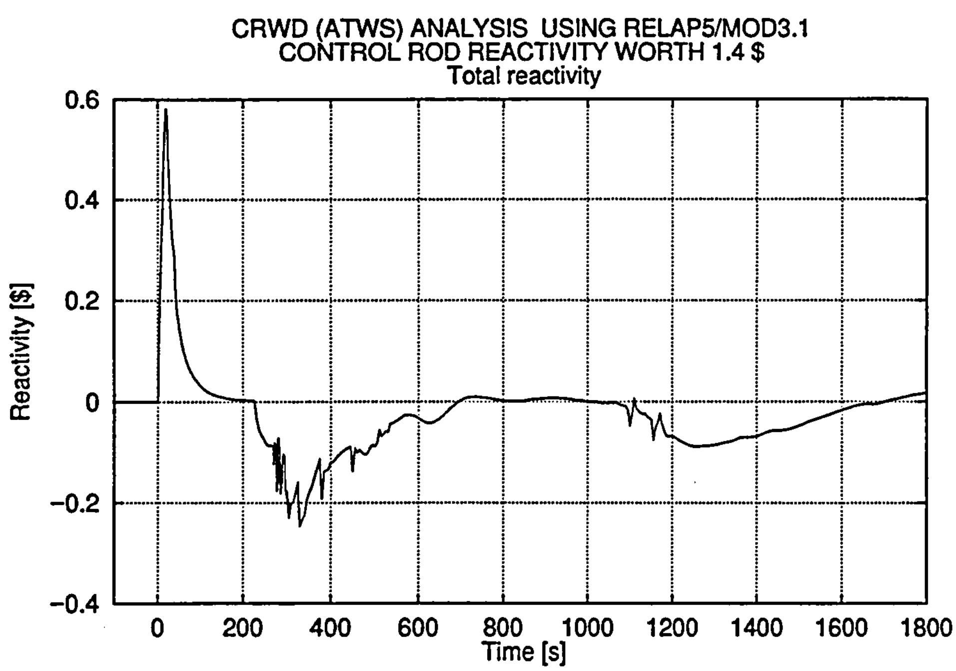

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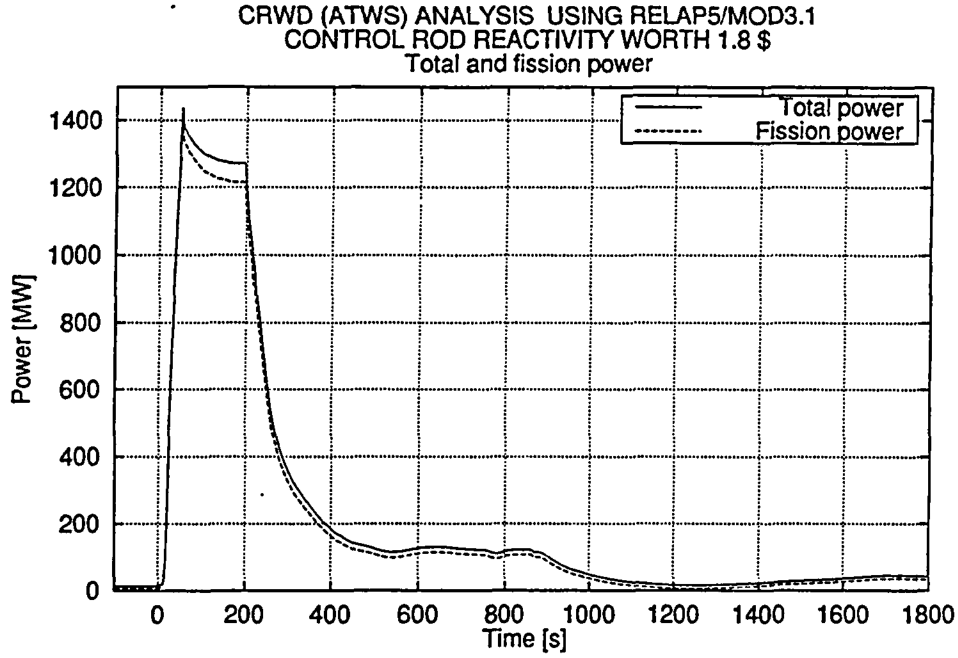

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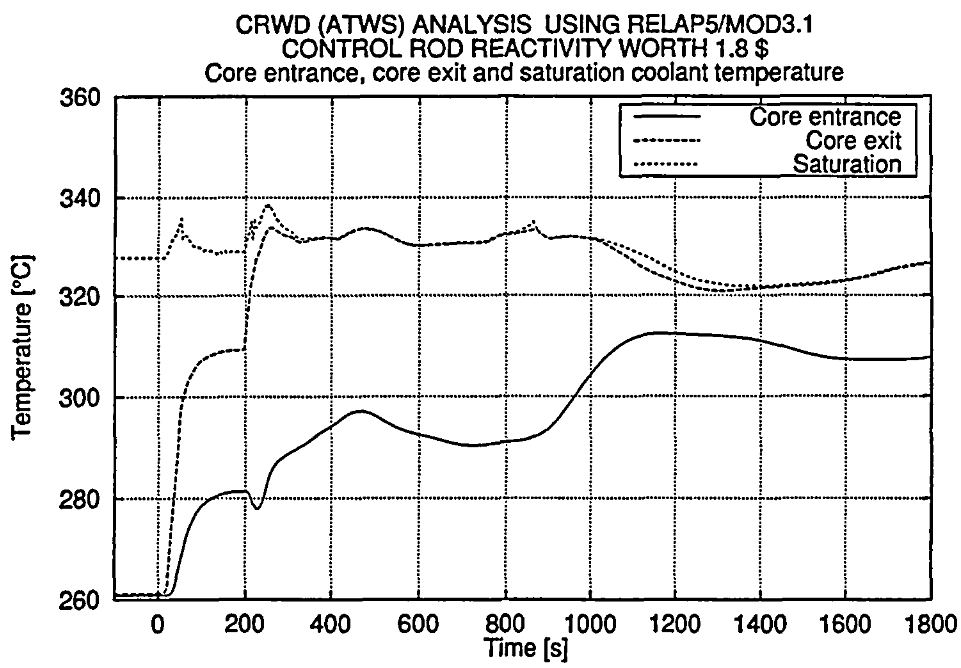

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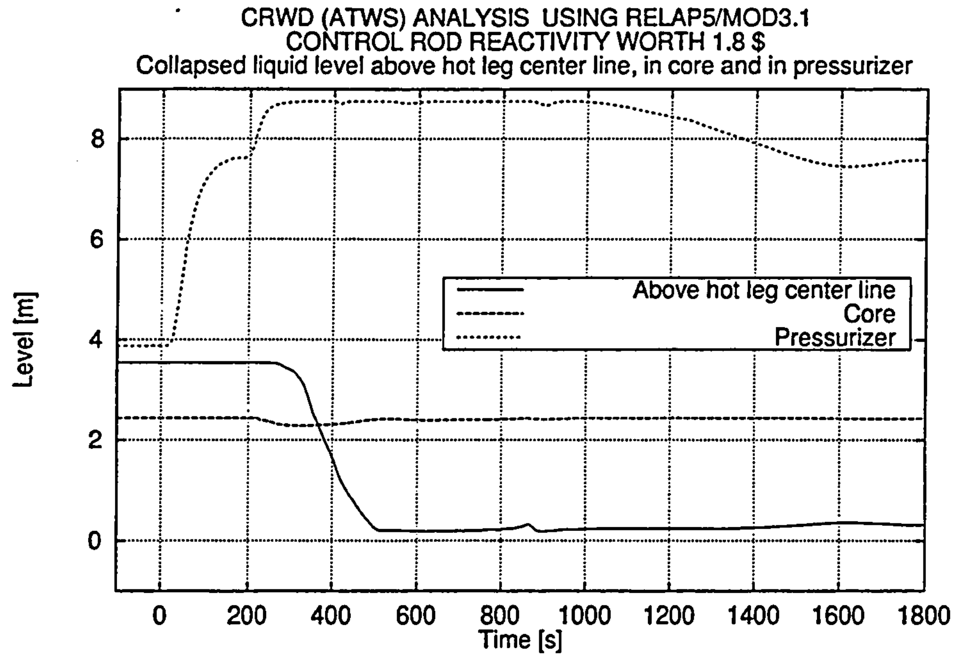

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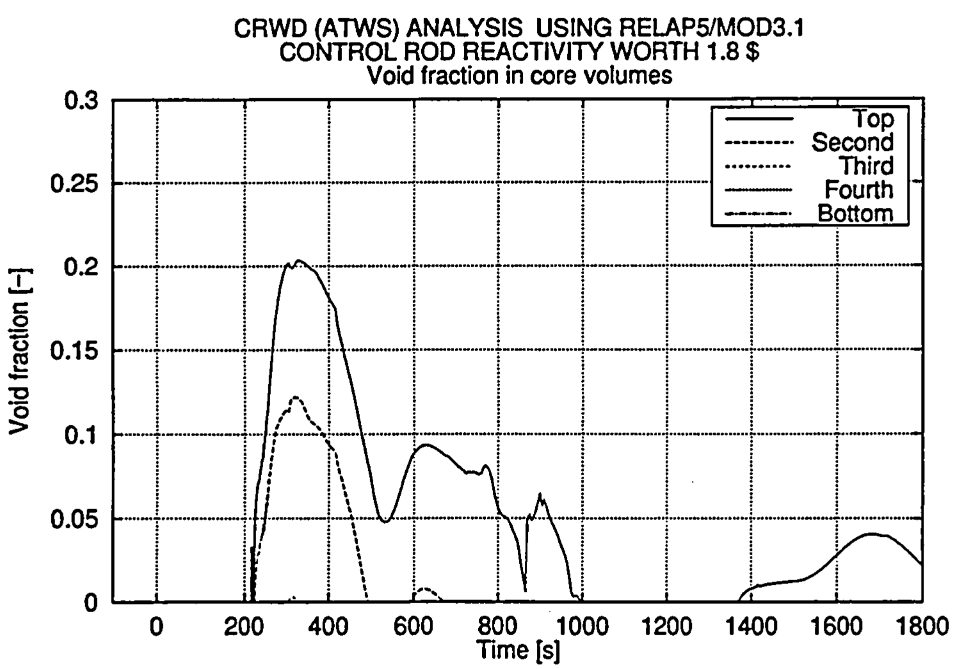

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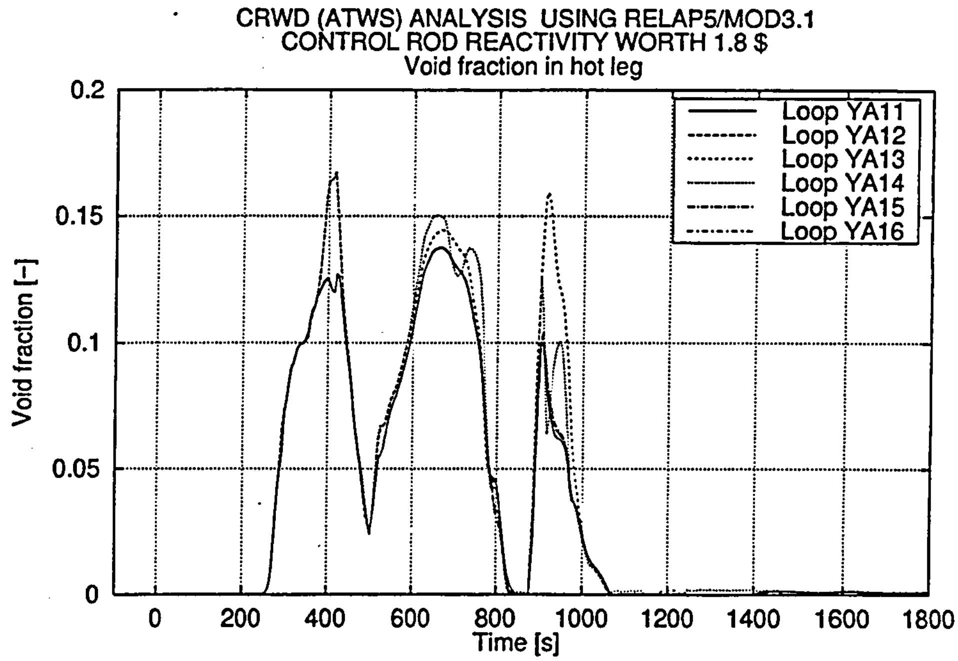

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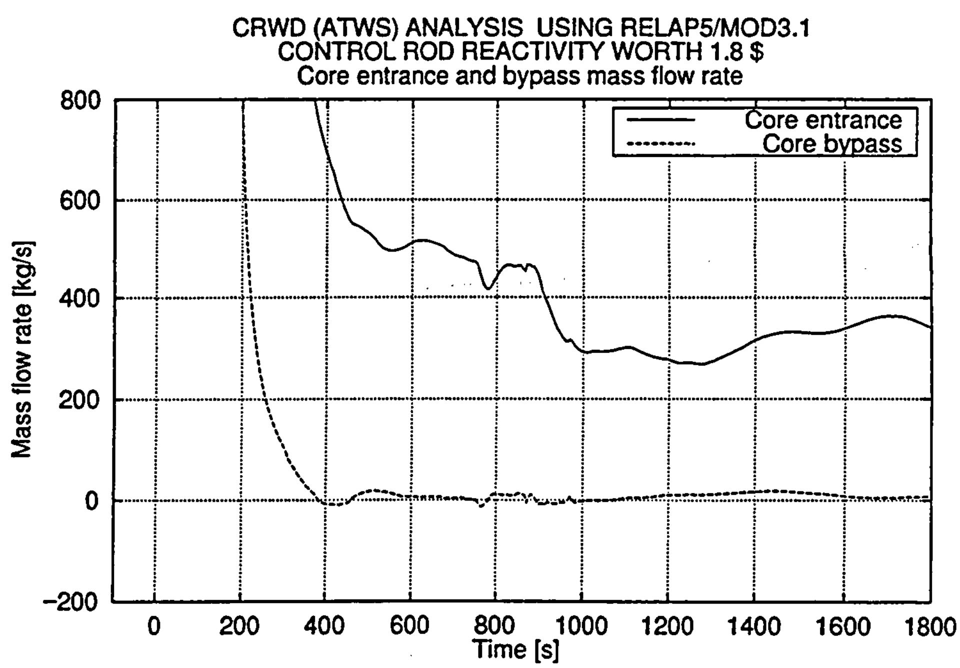

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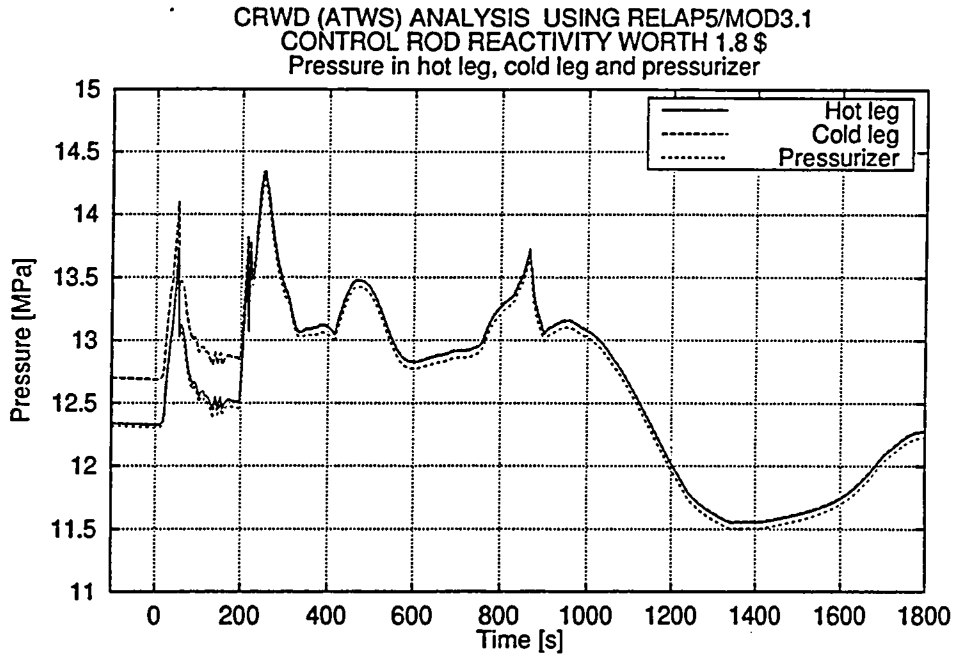

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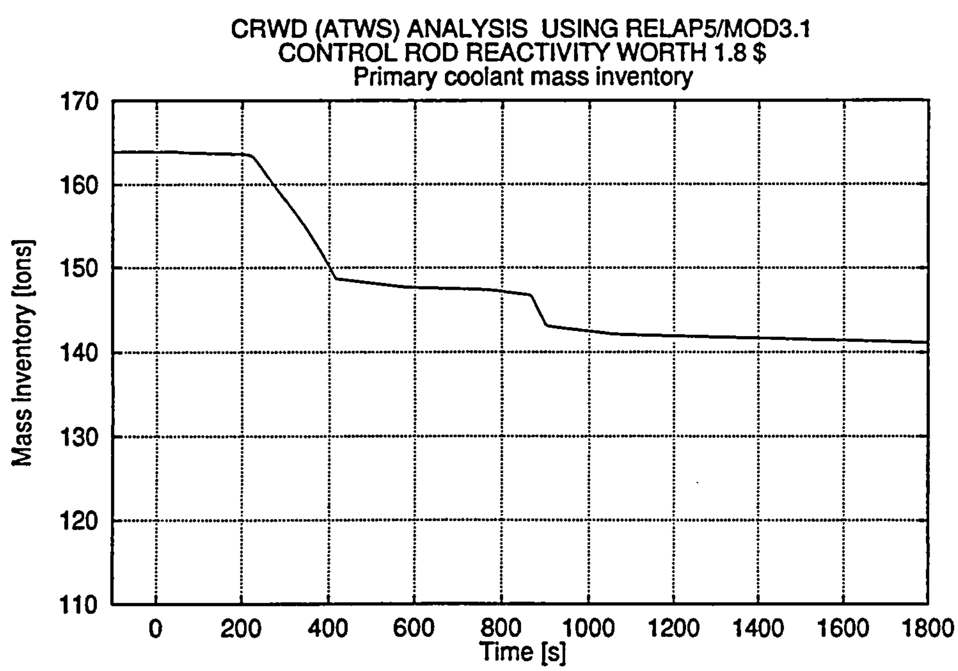

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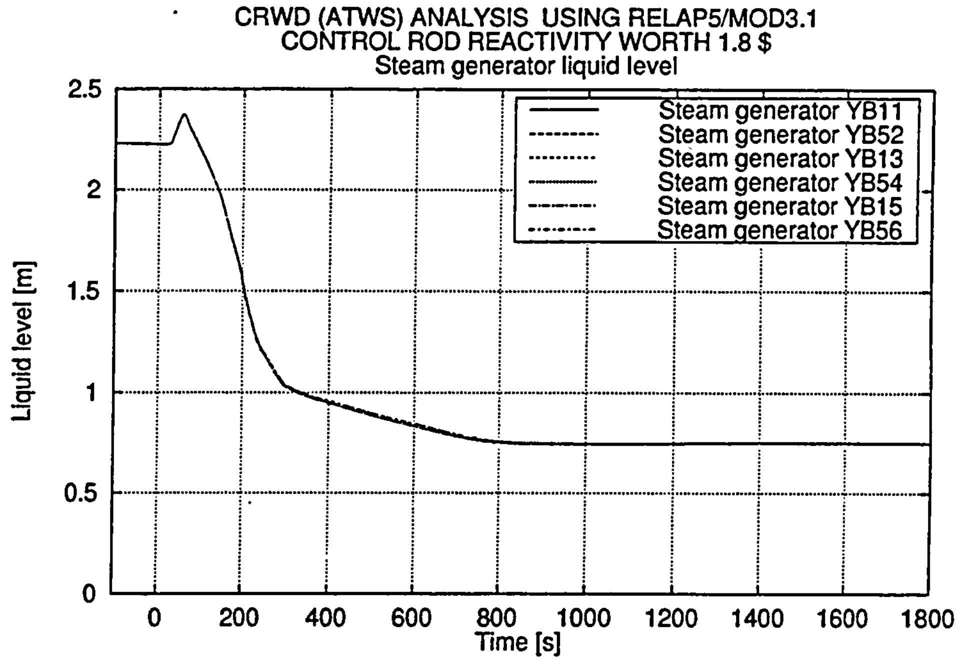

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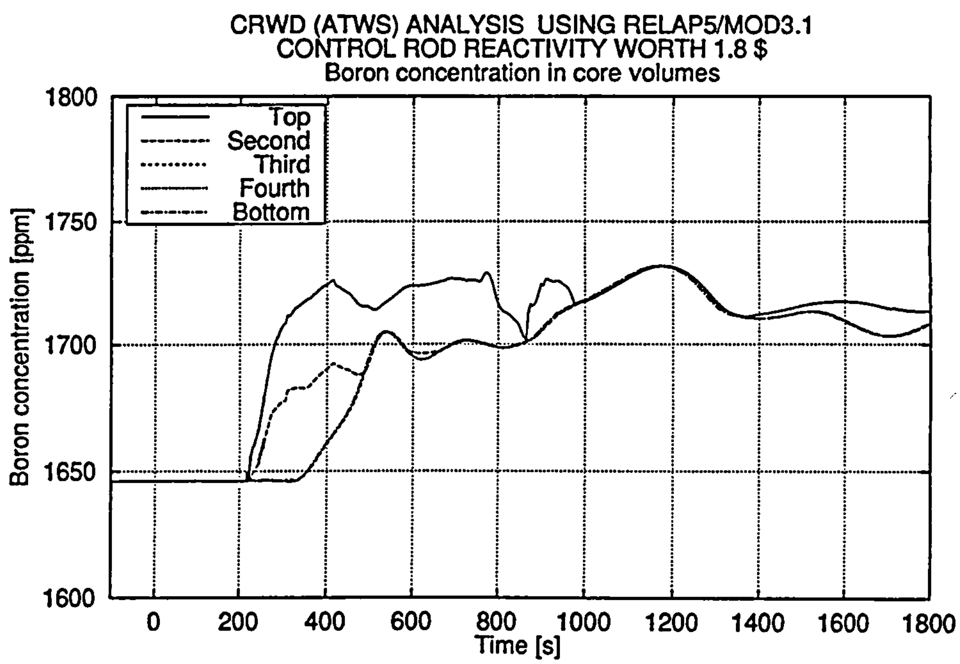

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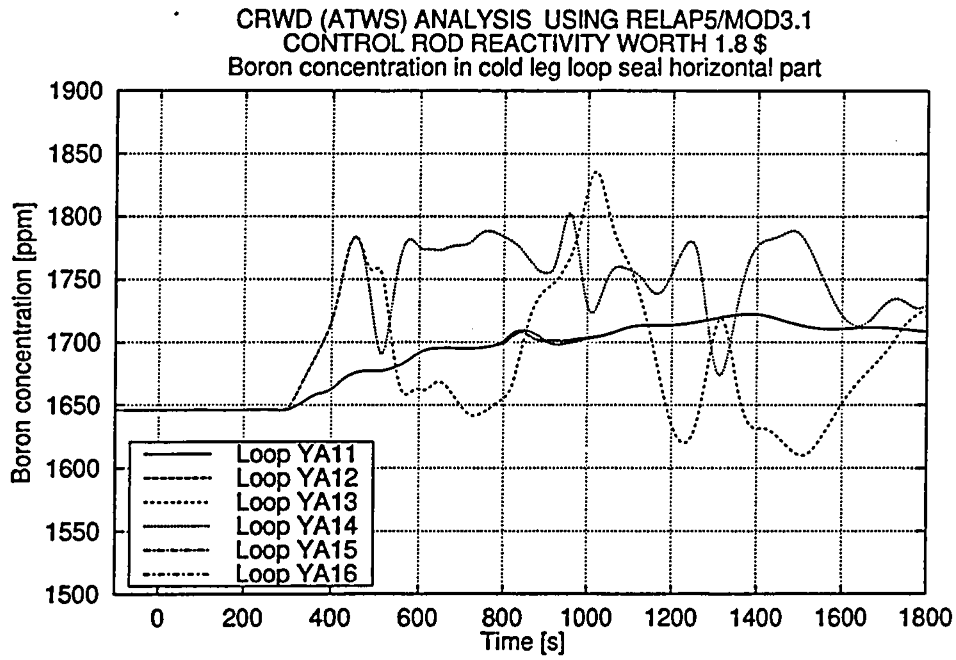

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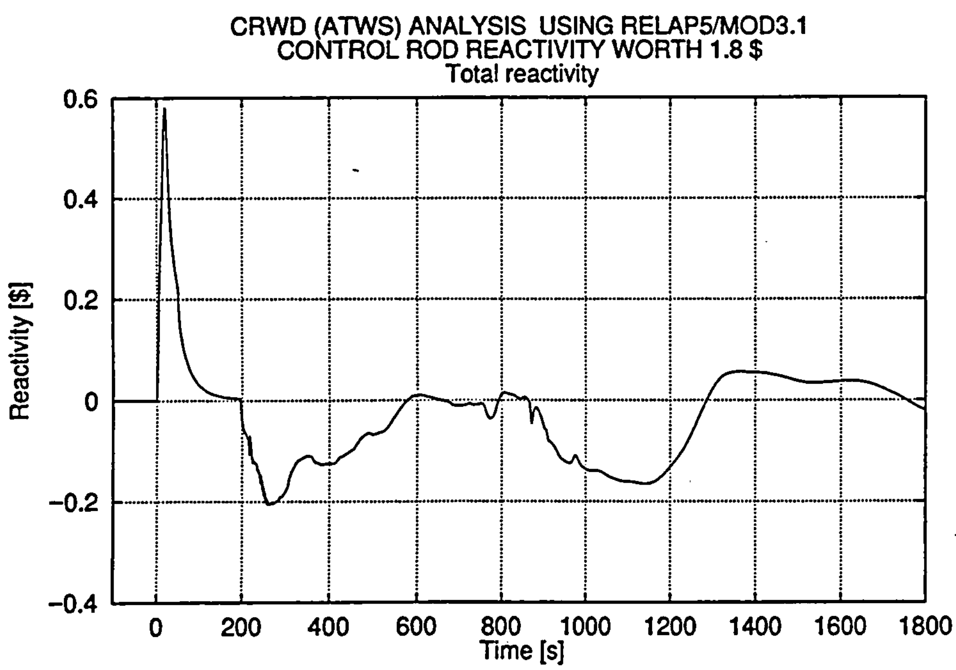

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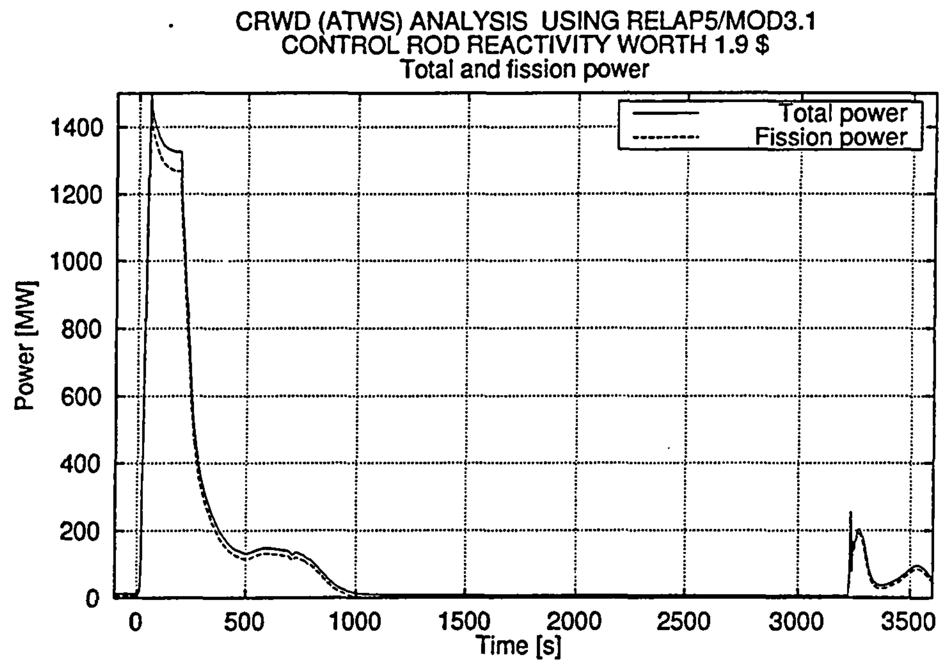

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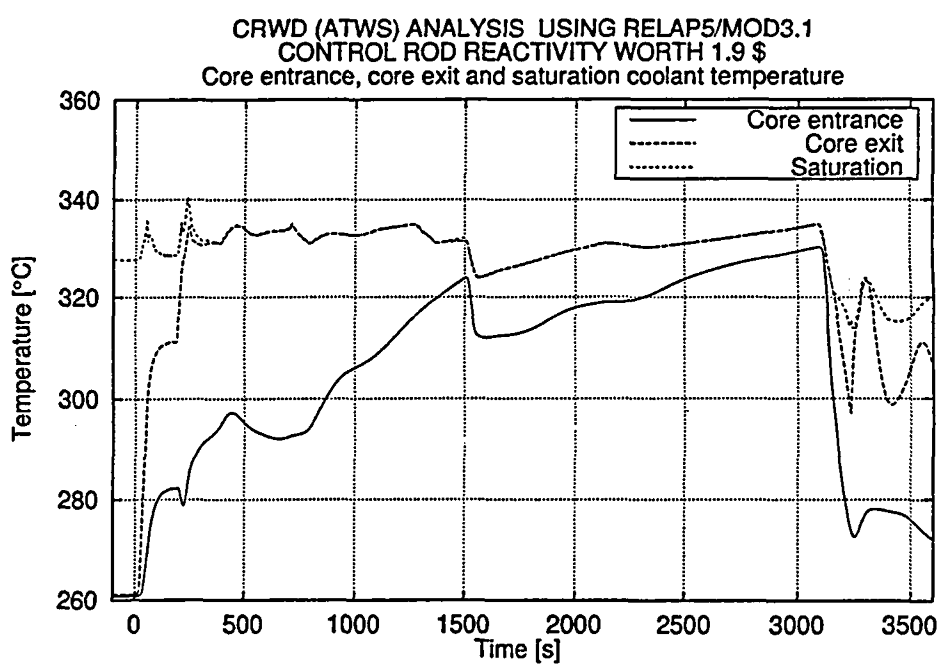

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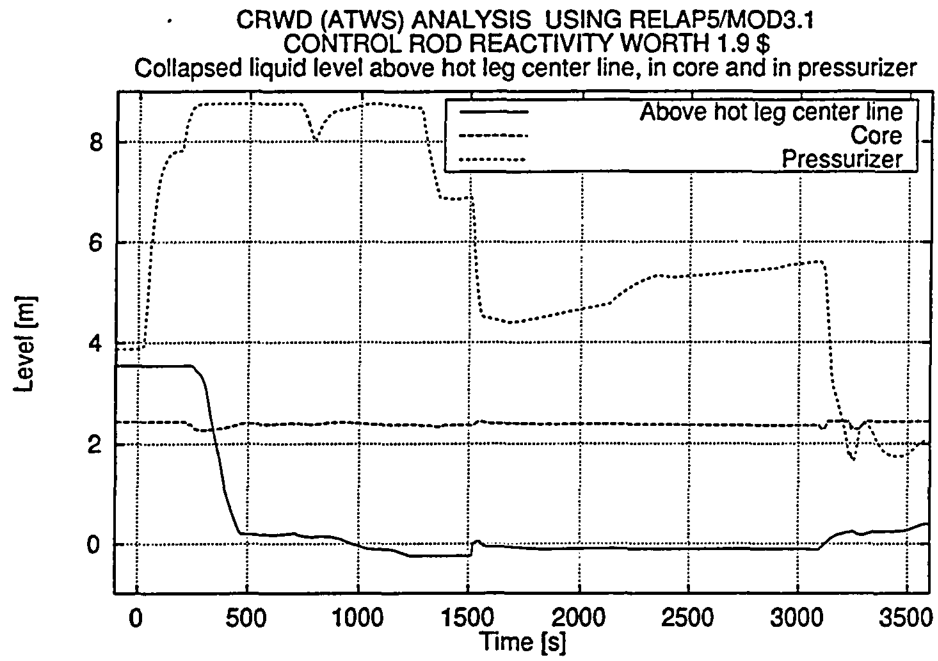

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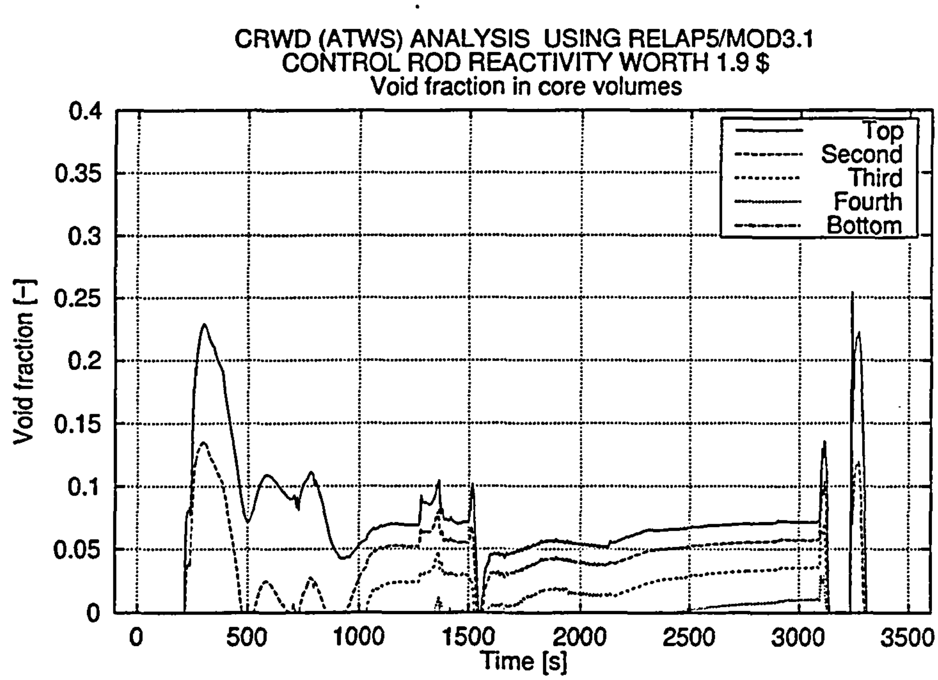

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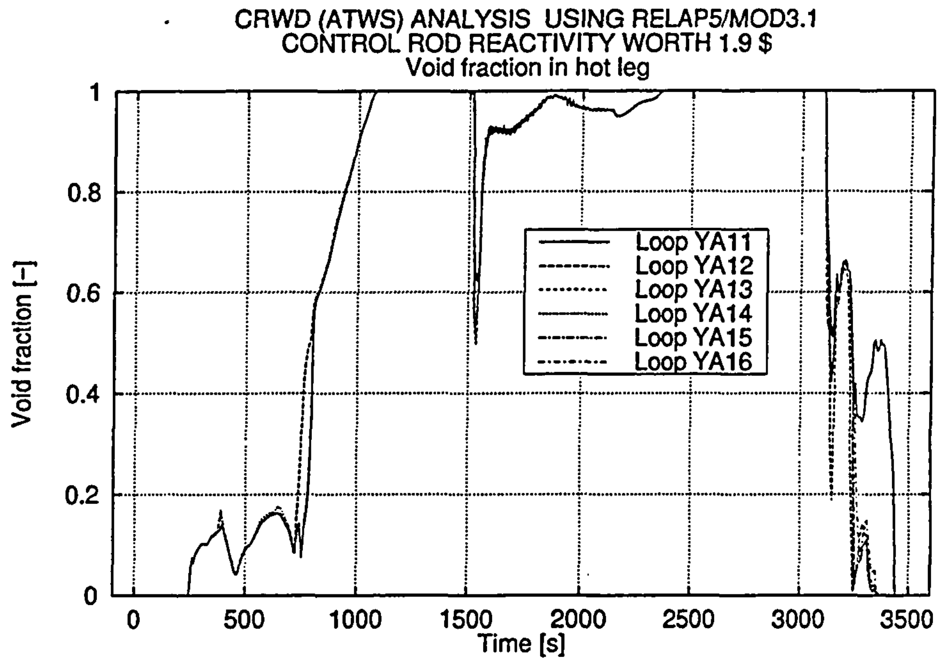

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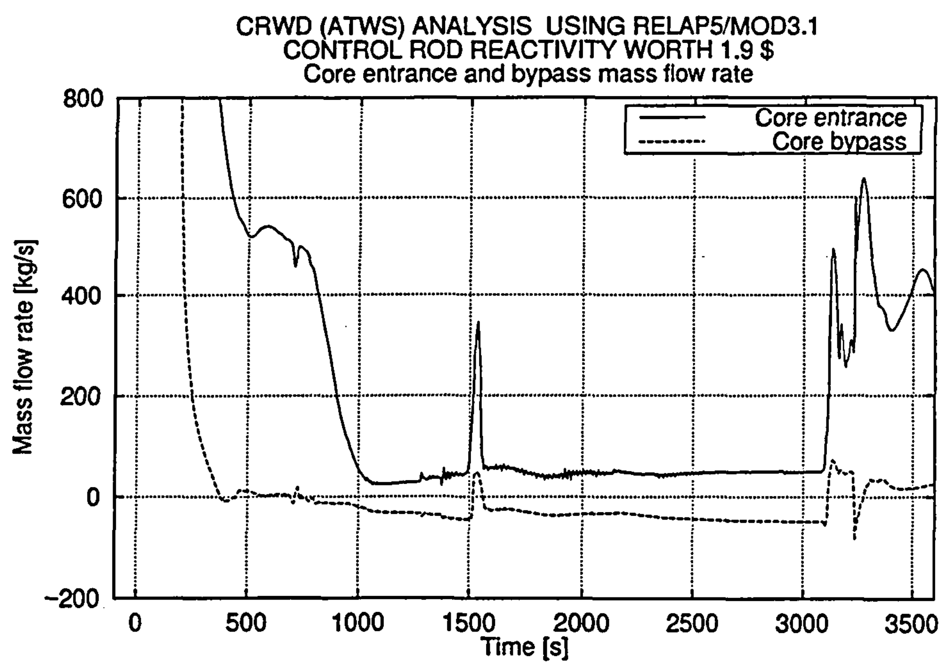

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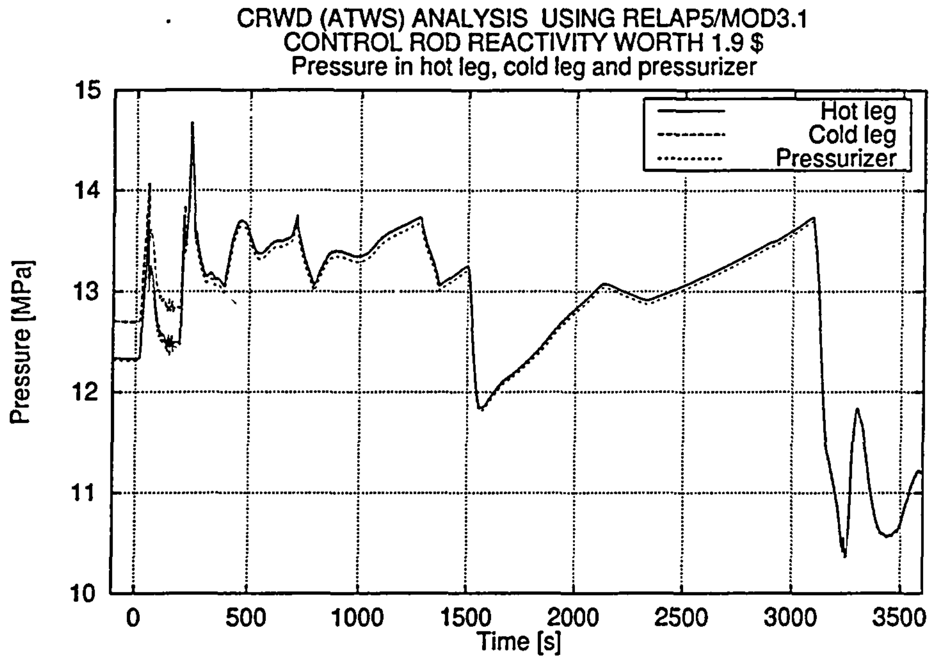

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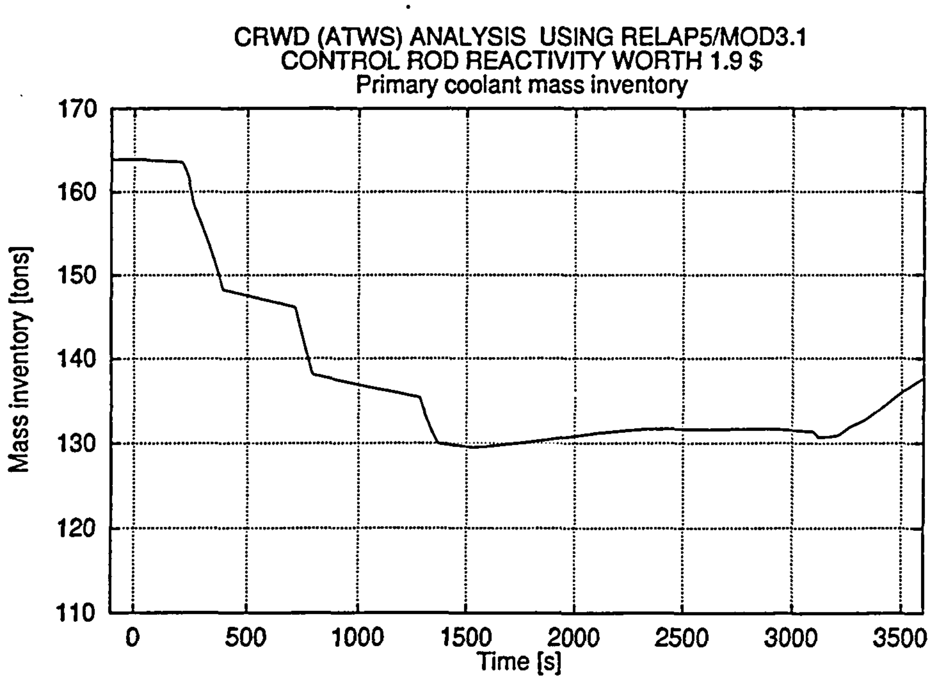

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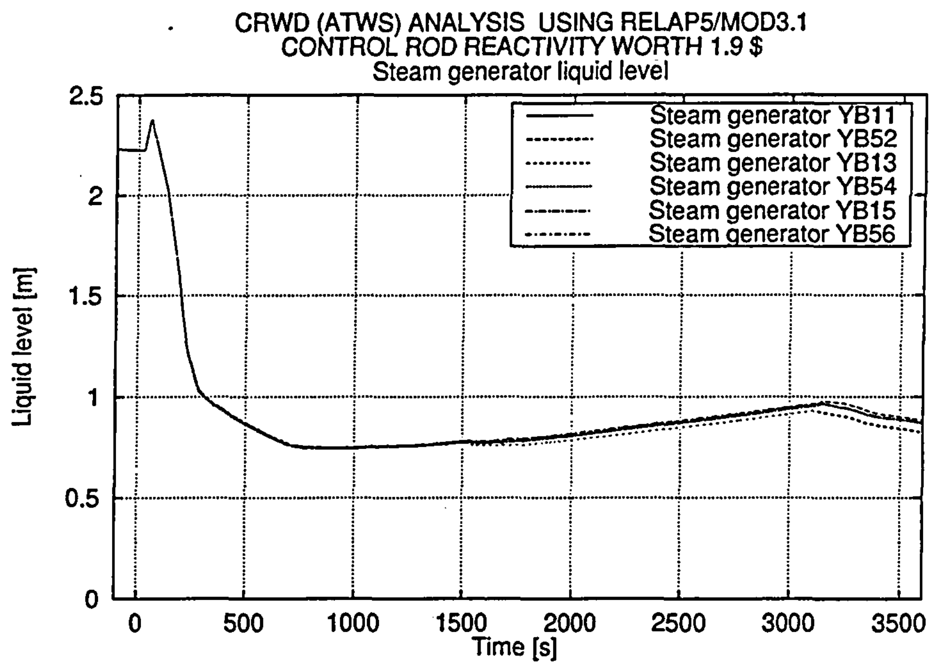

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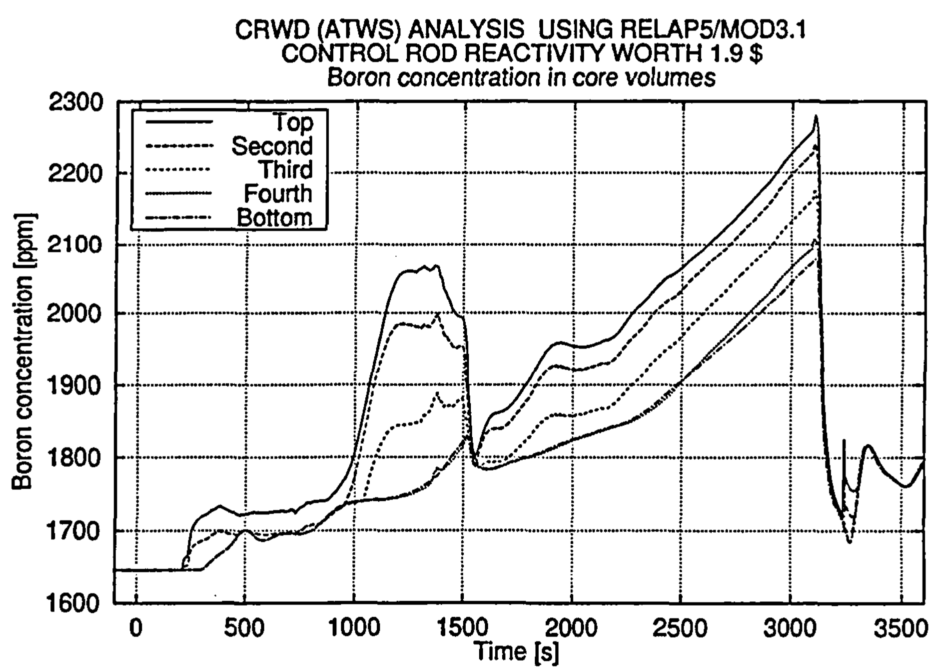

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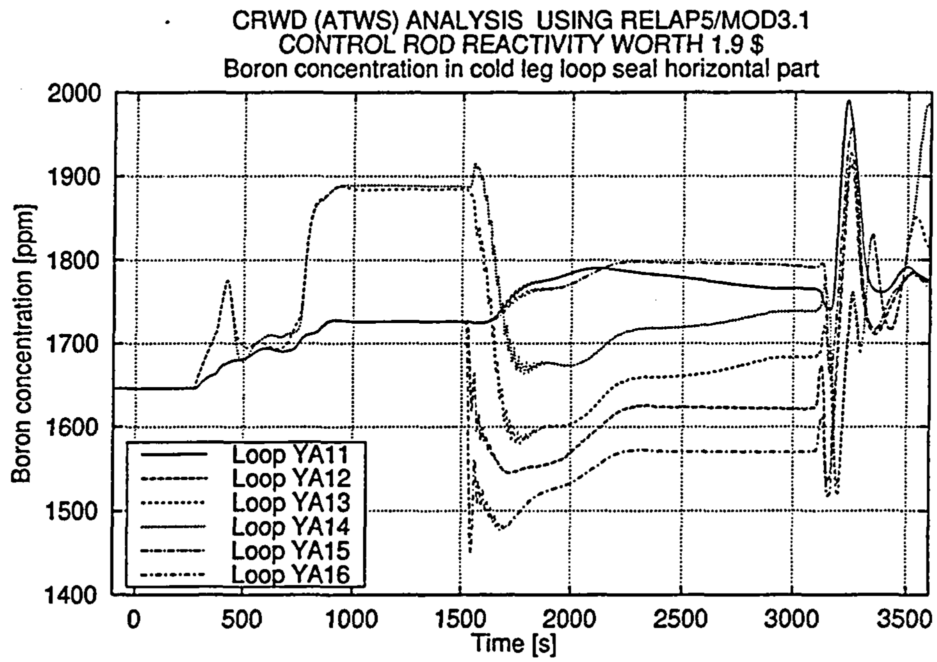

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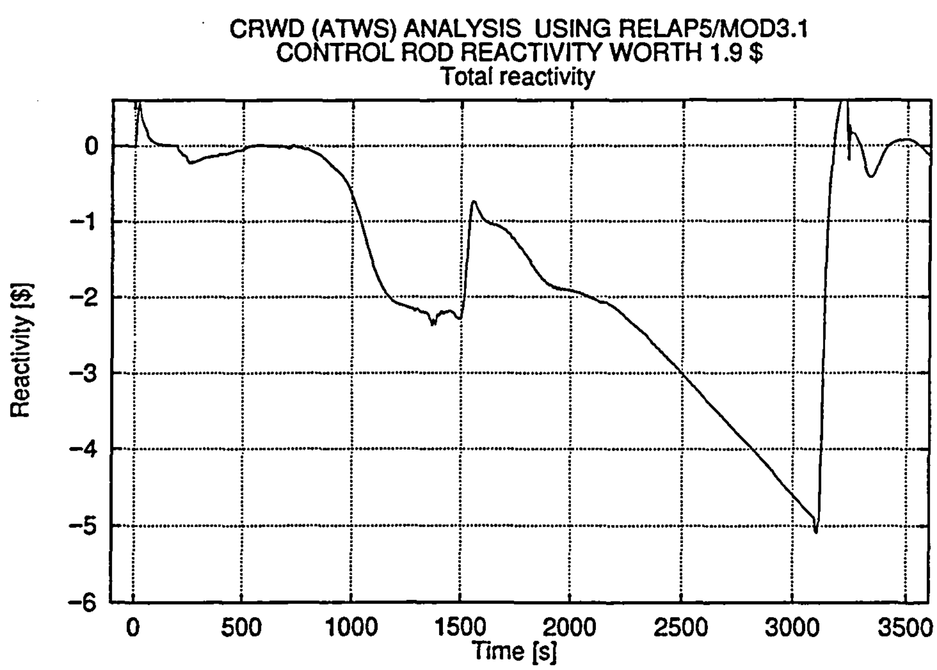

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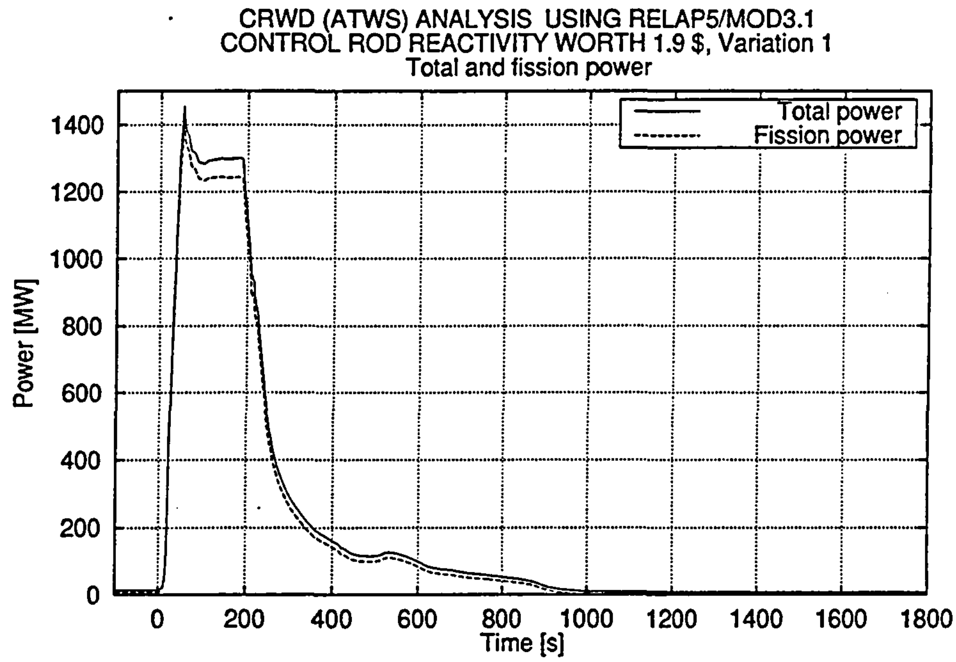

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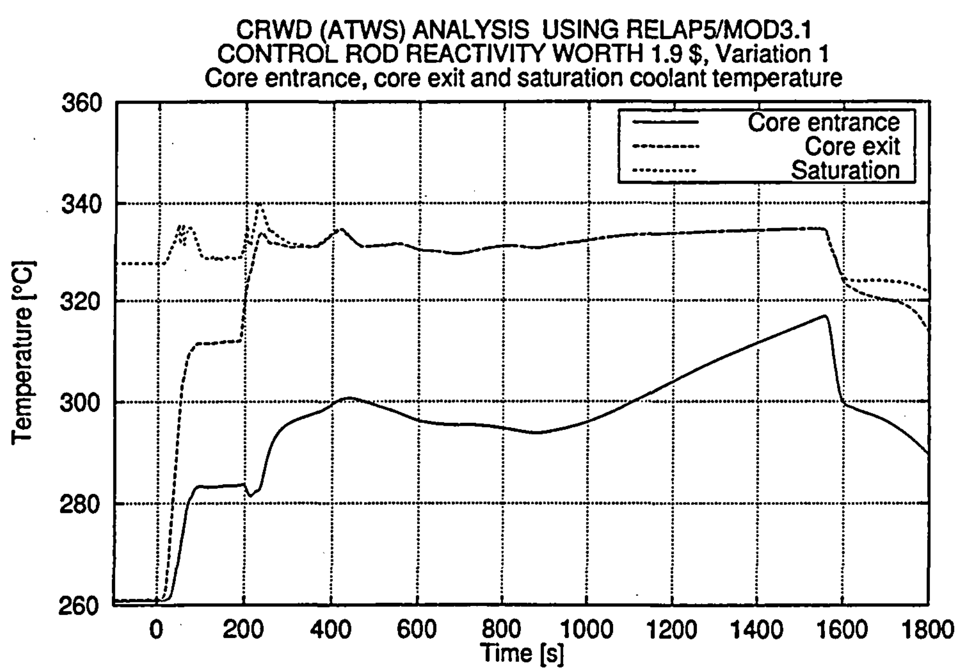

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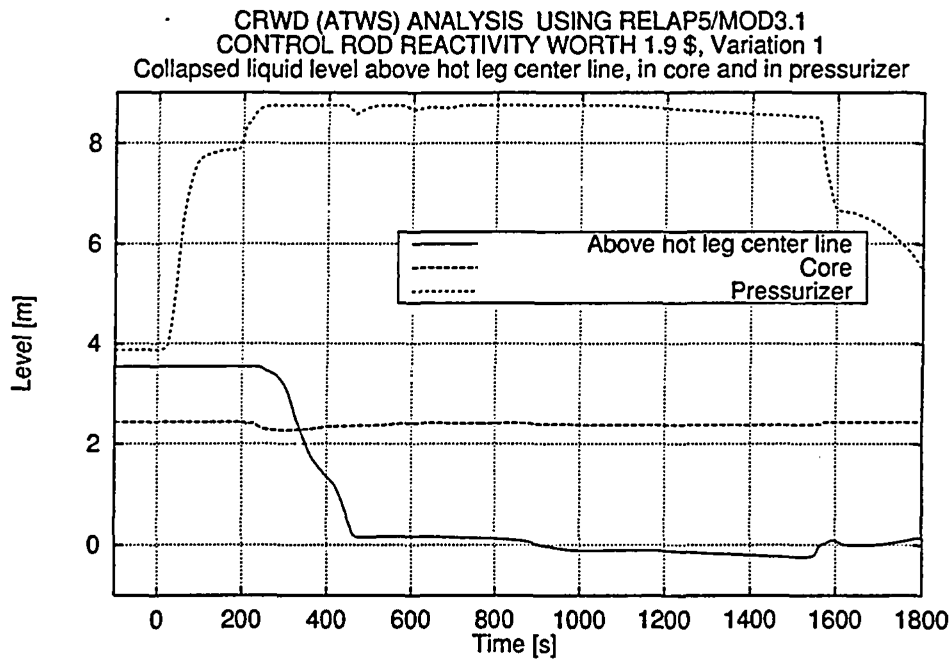

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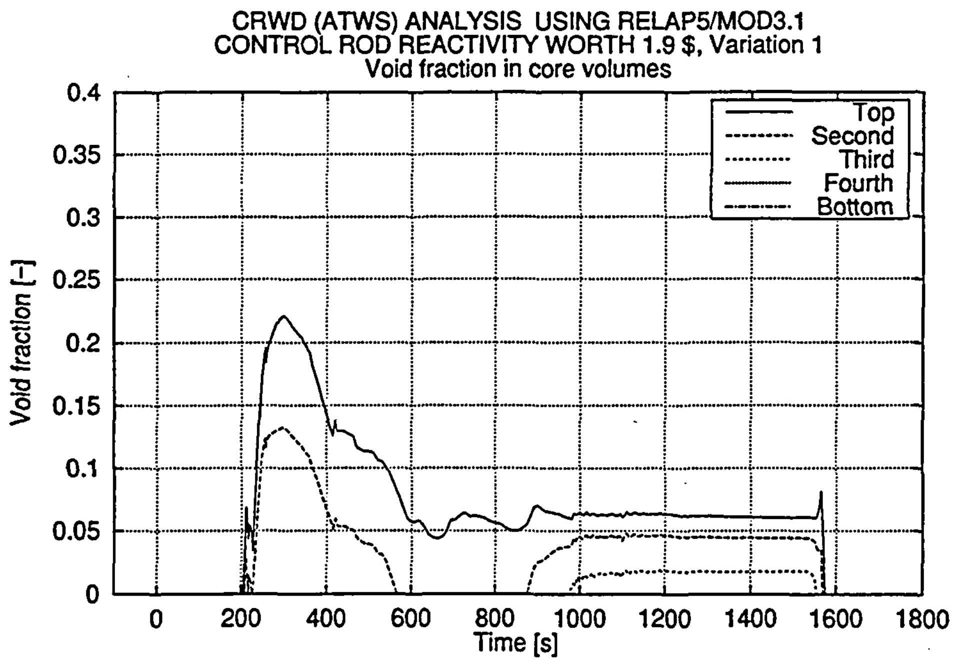

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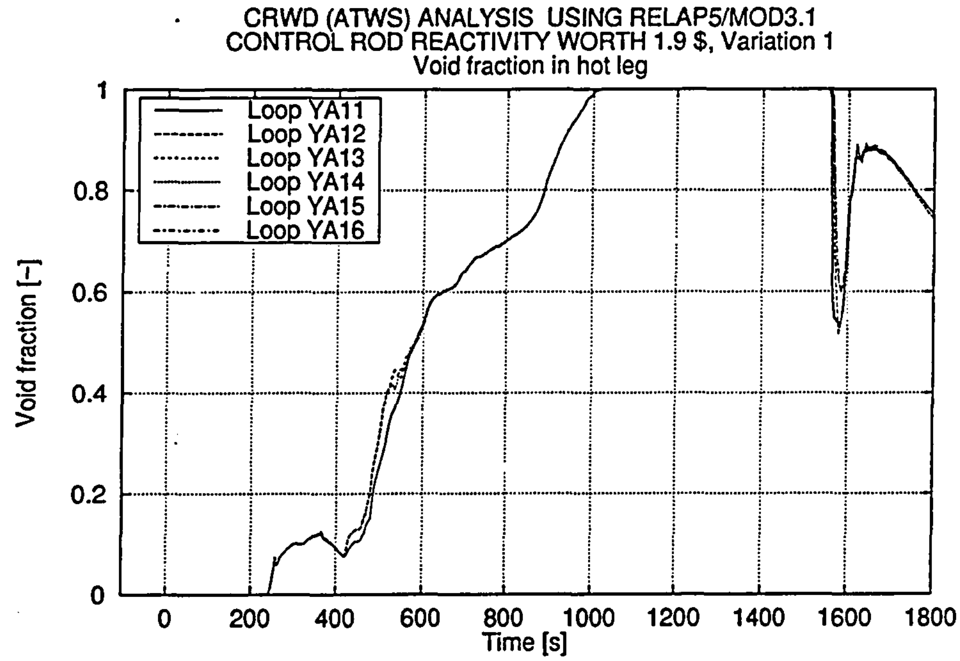

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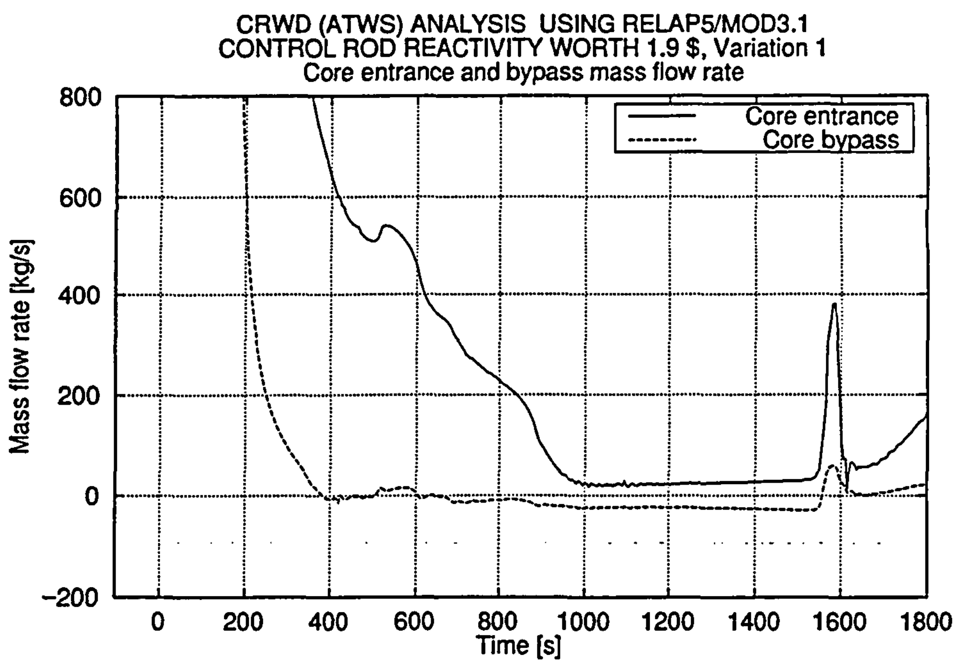

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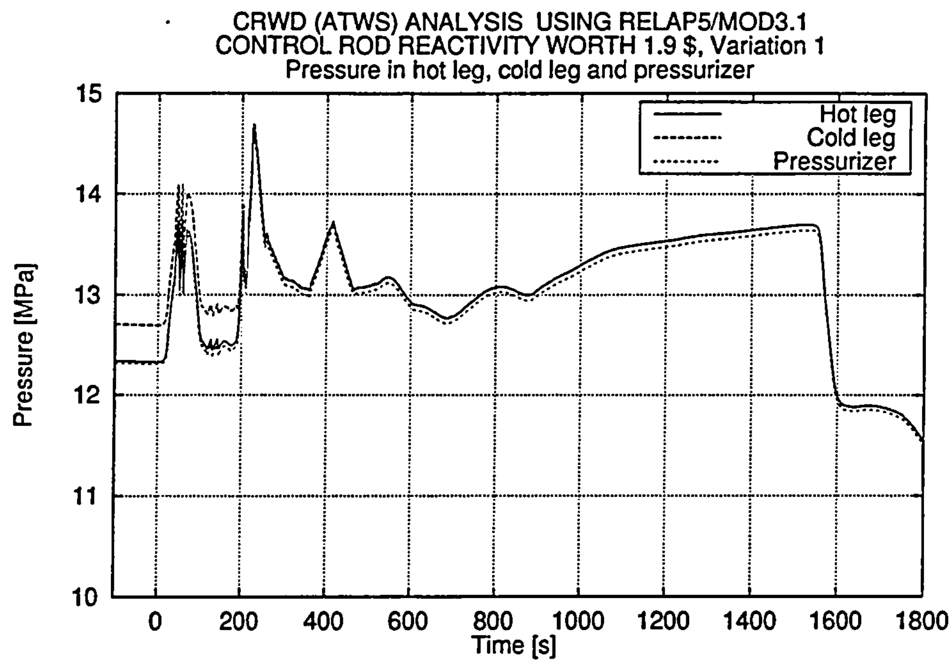

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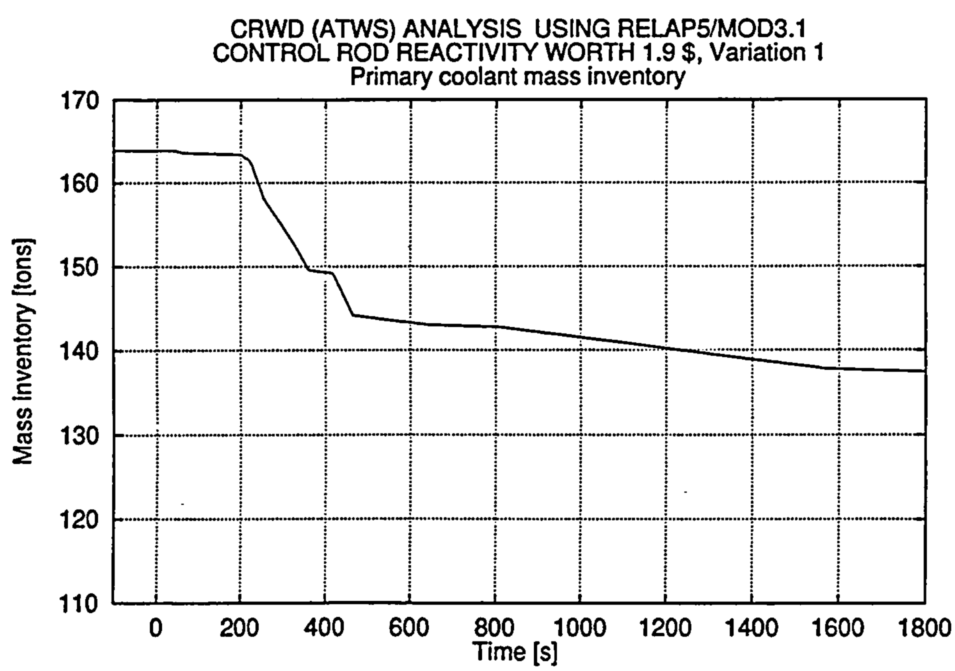

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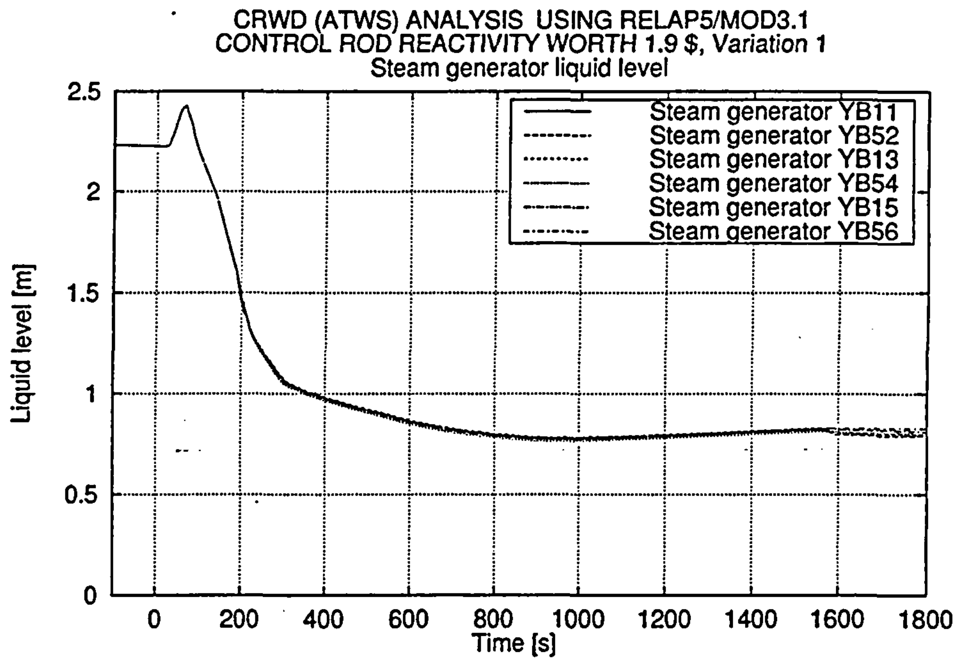

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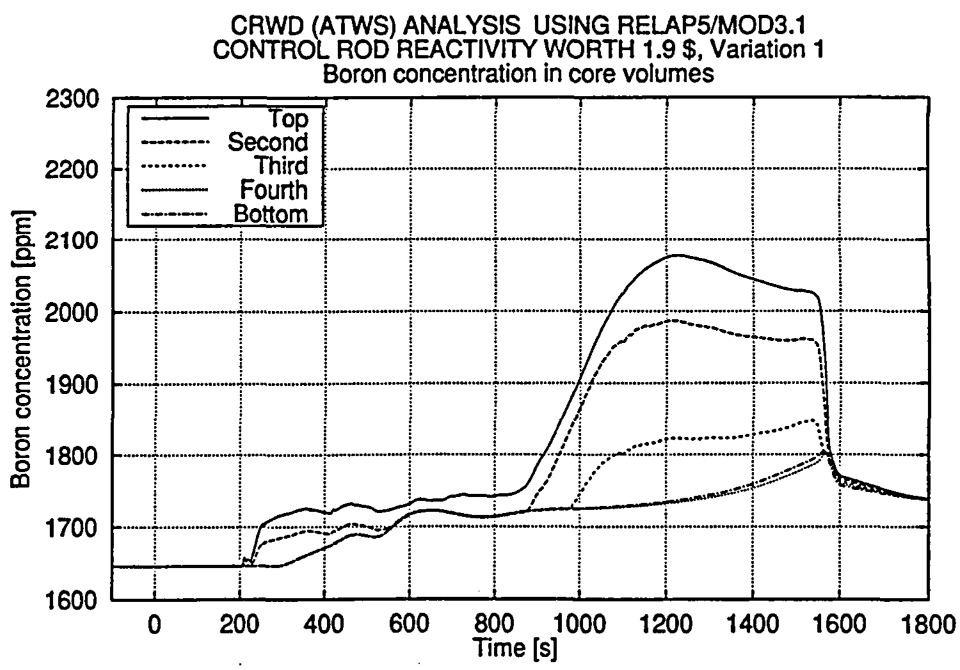

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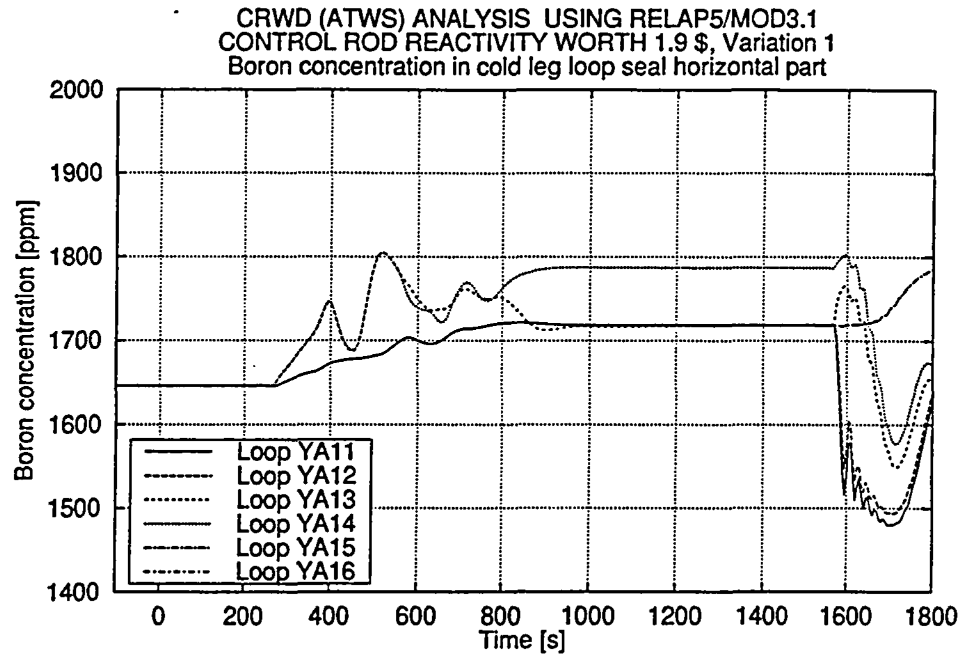

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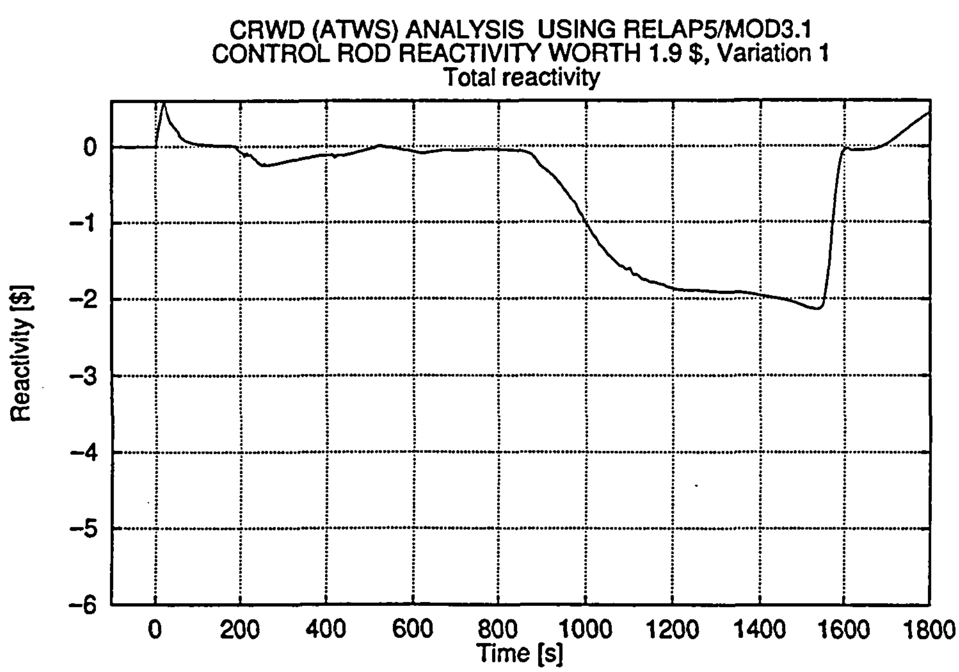

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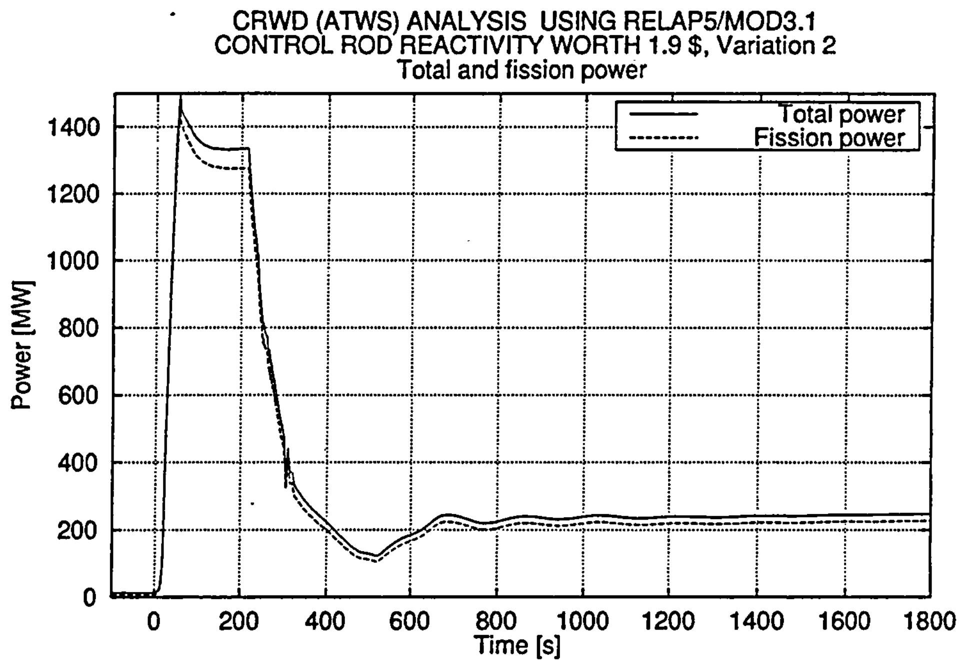

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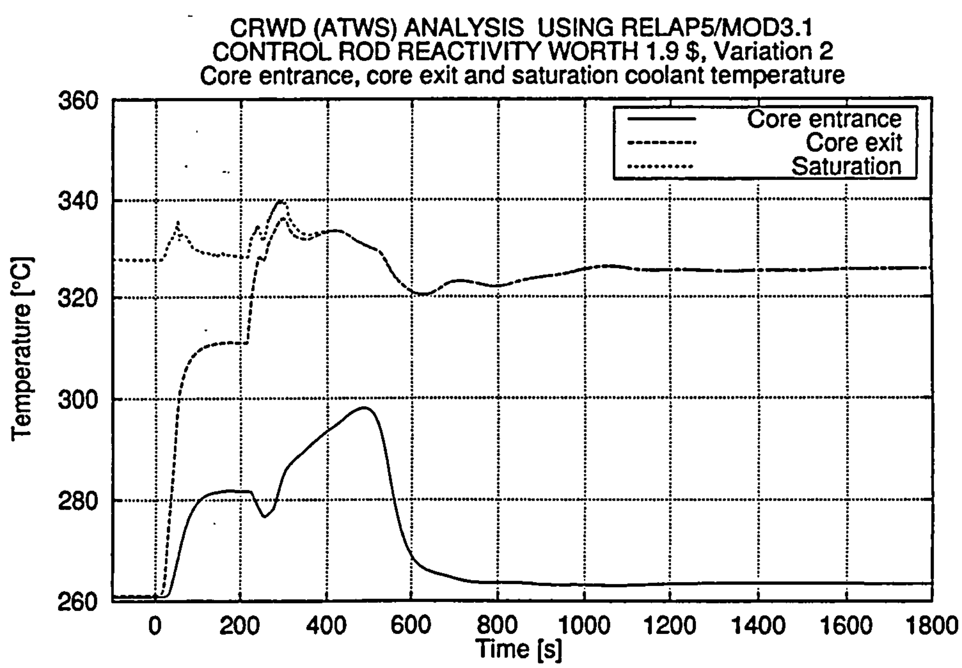
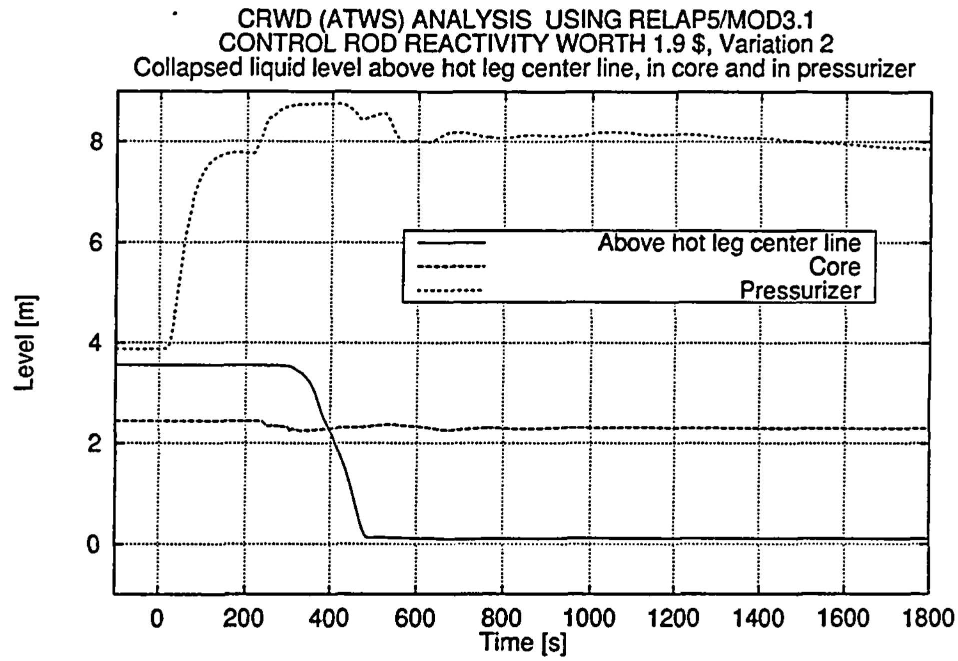


Figure 4.6.2-2



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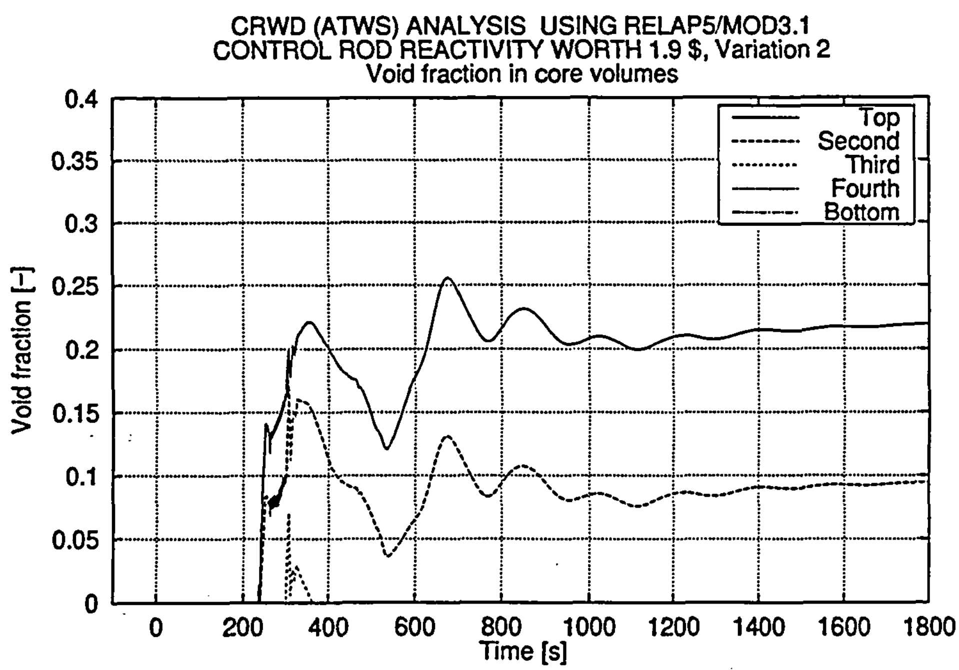

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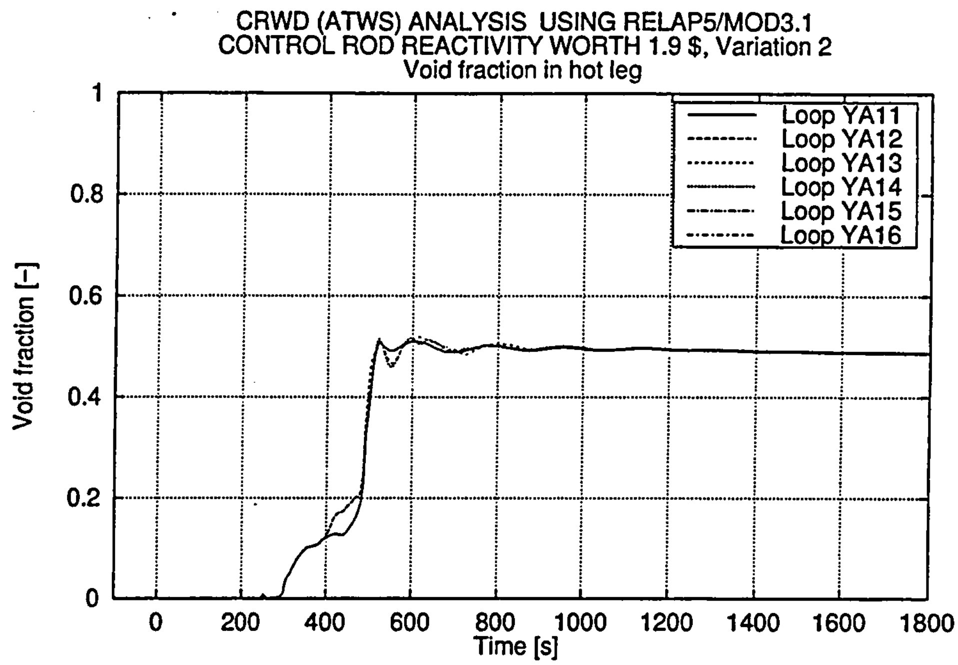

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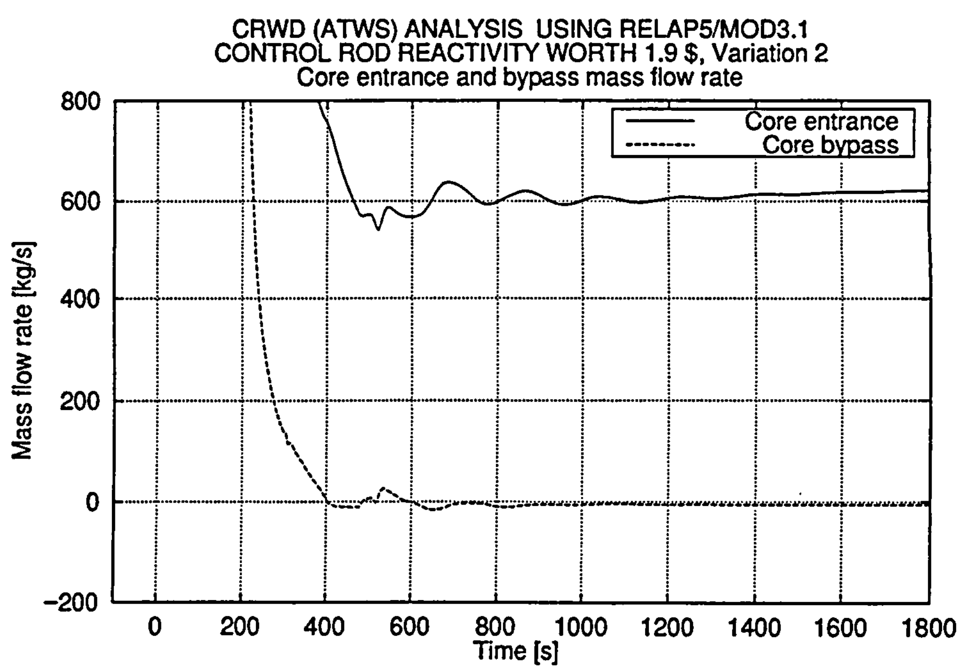

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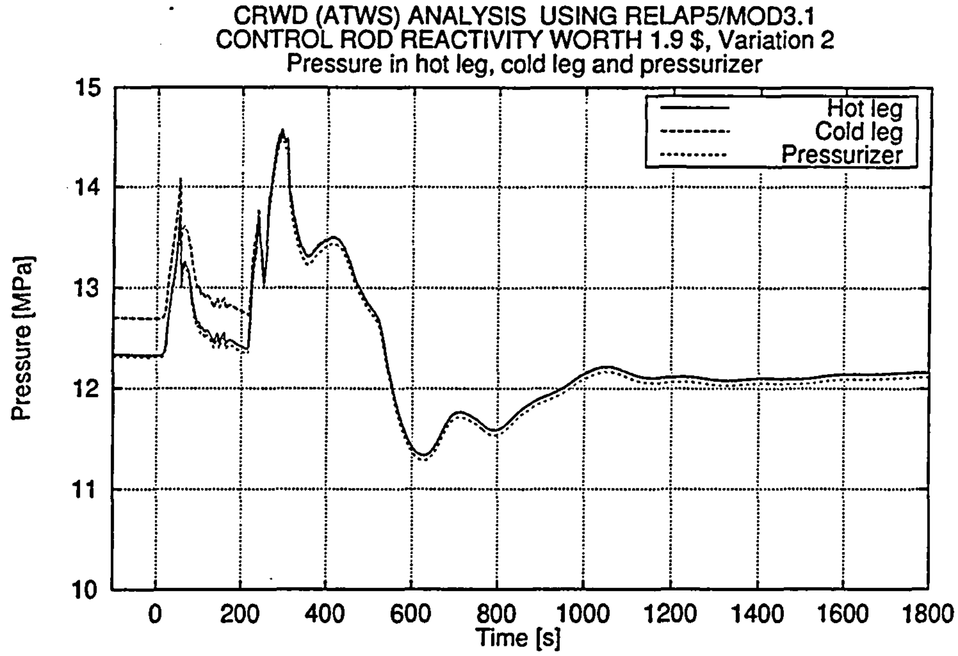

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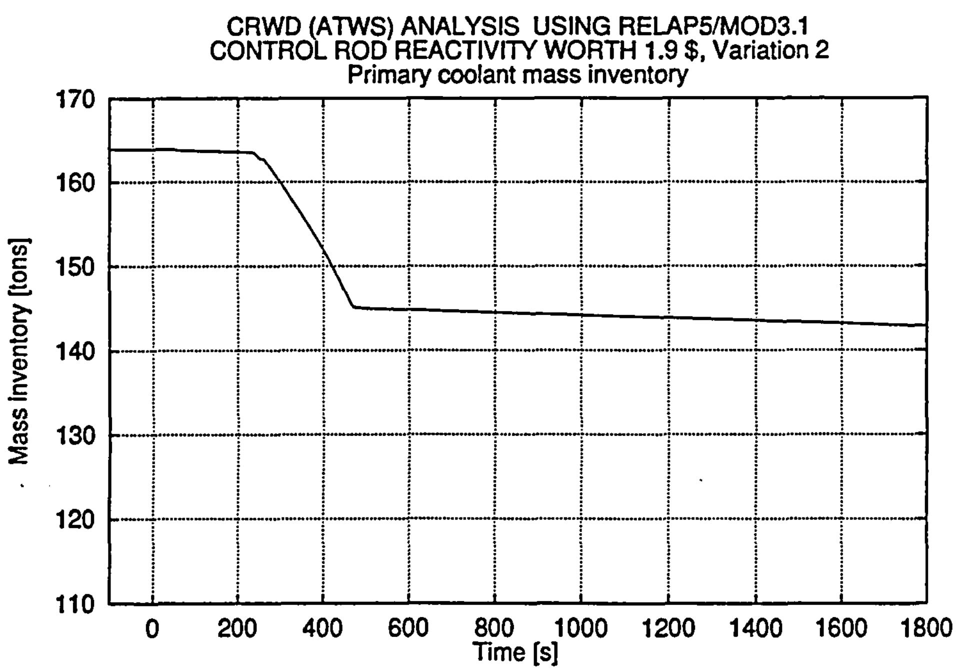

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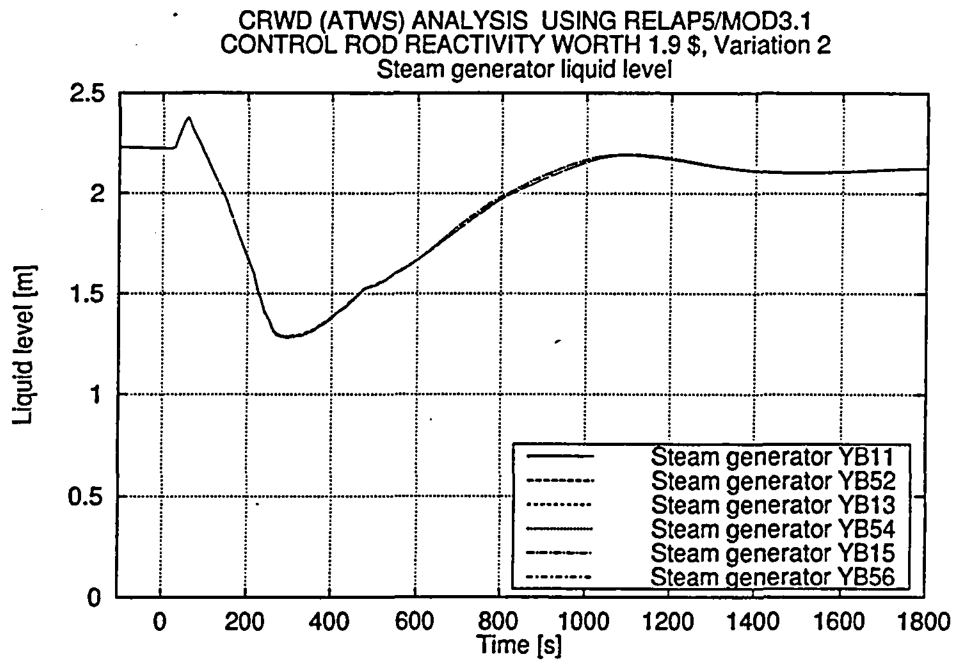

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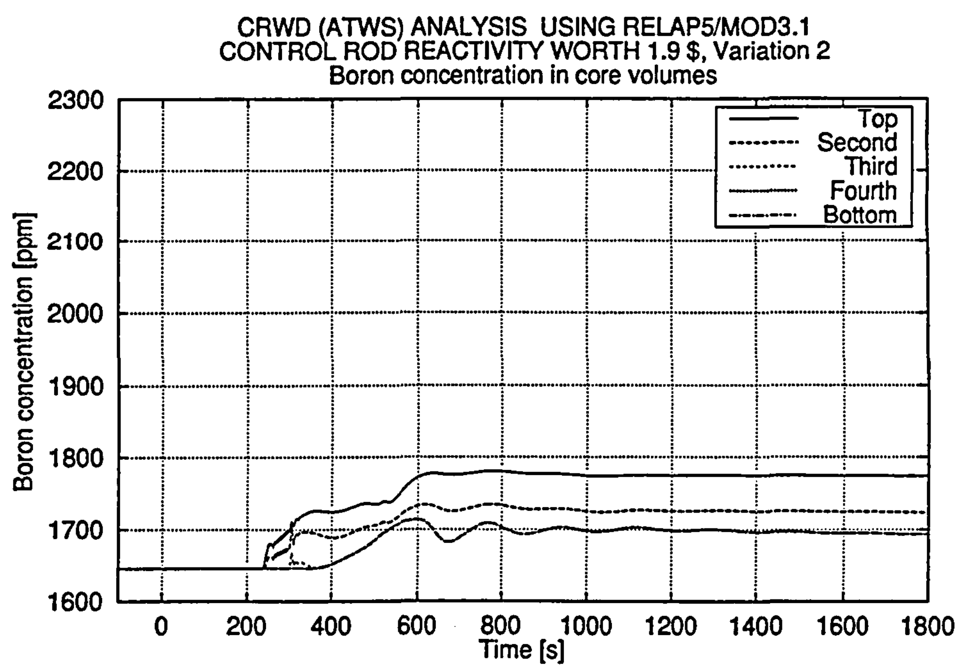

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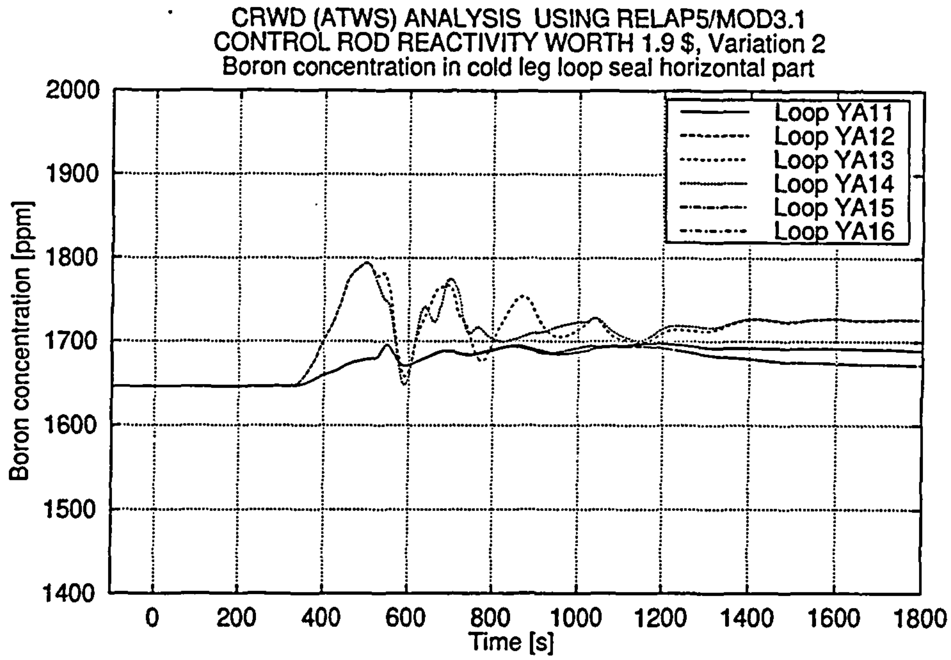

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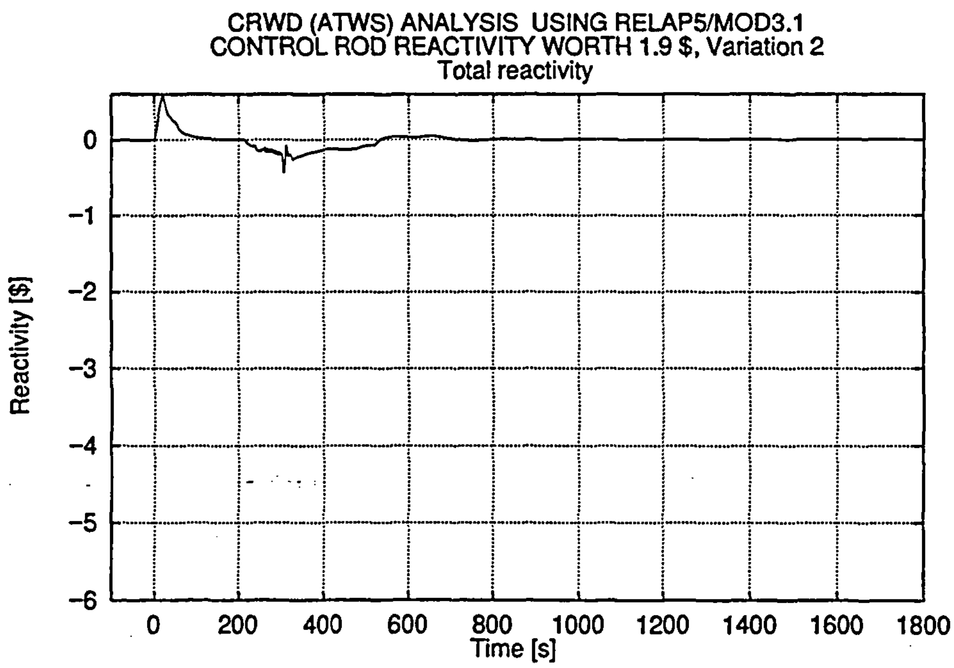

Figure 4.6.2-12

APPENDIX

Reactor kinetics input

```
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*   Reactor kinetics
*---- ----1---- ----2---- ----3---- ----4---- ----5---- ----6---- ----7
30000000  point     table4
*   Reactor kinetics information card
30000001  gamma-ac  13.8378+6 0.0       274.0     1.0       0.631
*   Fission product decay information
30000002  ans79-1
*   Power history data
30000401  1375.0+6  90.0      wk
30000402  0.0       35.0      hr
30000403  13.75+6   1000.0    sec
*   Reactivity curve
30000011  900
*mmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmm
*   Kinetics feedback table coordinate data
*---- ----1---- ----2---- ----3---- ----4---- ----5---- ----6---- ----7
30001911  300.0     400.0     500.0     600.0     650.0     * Density
30001912  700.0     725.0     750.0     800.0     1000.0
*
30001921  473.0     633.0     * Moderator T, Feedback = ZERO
*
30001931  500.0     700.0     803.0     900.0     1100.0    * Doppler
*
30001941  0.000     0.375     0.750     1.050     1.500     * Boron
30001942  1.875
*mmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmmm
*   Kinetics feedback table
*---- ----1---- ----2---- ----3---- ----4---- ----5---- ----6---- ----7
30002001  00000000    -6.258            30002002  00000001   -11.836
30002003  00000002   -16.930            30002004  00000003   -20.697
30002005  00000004   -25.895            30002006  00000005   -29.860
30002007  00000100    -8.216            30002008  00000101   -13.706
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30002011  00000104   -27.540            30002012  00000105   -31.442
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30002015  00000202   -19.534            30002016  00000203   -23.213
30002017  00000204   -28.291            30002018  00000205   -32.164
30002019  00000300    -9.900            30002020  00000301   -15.313
30002021  00000302   -20.255            30002022  00000303   -23.911
30002023  00000304   -28.955            30002024  00000305   -32.803
30002025  00000400   -11.434            30002026  00000401   -16.777
30002027  00000402   -21.657            30002028  00000403   -25.265
30002029  00000404   -30.244            30002030  00000405   -34.042
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30002033  00010002   -16.930            30002034  00010003   -20.697
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30002041  00010104   -27.540            30002042  00010105   -31.442
30002043  00010200    -9.109            30002044  00010201   -14.558
30002045  00010202   -19.534            30002046  00010203   -23.213
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30002051	00010302	-20.255	30002052	00010303	-23.911
30002053	00010304	-28.955	30002054	00010305	-32.803
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30002057	00010402	-21.657	30002058	00010403	-25.265
30002059	00010404	-30.244	30002060	00010405	-34.042
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30002065	01000004	-17.140	30002066	01000005	-21.555
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30002075	01000202	-9.514	30002076	01000203	-13.626
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30002079	01000300	1.414	30002080	01000301	-4.637
30002081	01000302	-10.165	30002082	01000303	-14.255
30002083	01000304	-19.902	30002084	01000305	-24.211
30002085	01000400	0.031	30002086	01000401	-5.958
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30002107	01010204	-19.302	30002108	01010205	-23.635
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30002125	02000004	-10.571	30002126	02000005	-15.329
30002127	02000100	11.323	30002128	02000101	4.728
30002129	02000102	-1.300	30002130	02000103	-5.762
30002131	02000104	-11.925	30002132	02000105	-16.631
30002133	02000200	10.590	30002134	02000201	4.028
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30002137	02000204	-12.543	30002138	02000205	-17.226
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30002143	02000304	-13.089	30002144	02000305	-17.751
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30002155	02010004	-10.571	30002156	02010005	-15.329
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30002173	02010304	-13.089	30002174	02010305	-17.751
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30002183	03000002	5.740	30002184	03000003	0.983
30002185	03000004	-5.590	30002186	03000005	-10.612
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30002189	03000102	4.388	30002190	03000103	-0.324
30002191	03000104	-6.836	30002192	03000105	-11.810
30002193	03000200	17.035	30002194	03000201	10.107
30002195	03000202	3.771	30002196	03000203	-0.920
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30002517	08010100	26.430	30002518	08010101	18.910
30002519	08010102	12.031	30002520	08010103	6.937
30002521	08010104	-0.104	30002522	08010105	-5.484
30002523	08010200	25.848	30002524	08010201	18.354
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30002539	08010404	-1.872	30002540	08010405	-7.185
30002541	09000000	33.209	30002542	09000001	25.029
30002543	09000002	17.565	30002544	09000003	12.049
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30002553	09000200	31.574	30002554	09000201	23.469
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30002589	09010300	31.122	30002590	09010301	23.037
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H. Kantee

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The RELAP5/MOD3.1 computer code has been applied in the analysis of a regulating control rod group withdrawal from 1% power level. The analysis has been carried out as an Anticipated Transient Without Scram (ATWS) event.

The analysis is related to an extensive inherent boron dilution study which was performed to Loviisa Nuclear Power Plant (NPP) in the mid 1990's. The main goal of the analysis was to study if during the accident so called boiling-condensing mode develops and thereafter a water plug with low boron concentration is formed in the cold leg loop seal.

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

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