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Study of Transients Related to AMSAC Actuation, Sensitivity Analysis

Prepared by
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ABSTRACT

The Asociación Nuclear Ascó (ANA) has prepared a model of Ascó NPP using RELAP5/MOD3.2. This model, which includes thermal-hydraulics, kinetics and protection and control systems, has been qualified in previous calculations of several actual plant transients.

Ascó NPP is a two unit station of three loop Pressurized Water Reactors (PWR) of Westinghouse design operated by ANA. ANA is a spanish utility that contributes to the Code Application and Maintenance Project (CAMP) as a member of UNIDAD ELECTRICA S. A. (UNESA).

This report summarizes the results obtained with Ascó NPP model for a loss of normal feedwater ATWS event and presents a sensitivity analysis to kinetic parameters for the same transient.

The phenomenology prediction has been useful from the operation and safety point of view.

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EXECUTIVE SUMMARY

Ascó Nuclear Power Plant is a two unit station of three loop Pressurized Water Reactors (PWR) of Westinghouse design.

The Thermal-hydraulic analysis group of the Asociación Nuclear Ascó (ANA) has prepared a model of the plant using RELAP5/MOD3.2. This model includes thermal-hydraulics, kinetics and protection and control systems.

The aim of the study presented in this report is to fulfill ANA's commitment with the Code Assessment and Maintenance Program (CAMP).

The transient selected to be analyzed for this purpose is the Loss of feedwater ATWS event at Ascó NPP with RELAP5/MOD3.2.

In the following pages the results obtained under two points of view will be presented:

1. Best-estimate results using design assumptions.
2. Sensitivity analysis to kinetic parameters.

The main conclusions are as follows:

- The maximum peak pressure in the reactor coolant system is 19.0 MPa. It is well below the ASME Boiler and Pressure Vessel Code Level for BOL conditions (service limit stress criterion of 3200 psig (22.16 MPa).) /16/.
- The maximum peak pressure in the reactor coolant system depends significantly on the kinetic parameters.
- After the loss of feedwater, the pipe between the feedwater isolation valve and the header of auxiliary and main feedwater, works as a water tank. This stored water mass helps to cool the primary and produces the isolation of the steam generator as well as the start of the High Pressure Injection System (HPIS).
- RELAP5/MOD3.2 is a valuable tool to analyze plant transients.

LIST OF ABREVIATIONS

ANA: Asociación Nuclear Ascó

AMSAC: ATWS Mitigating System Actuation Circuitry

ASME: American Standard for Mechanical Engineers

ATWS: Anticipated Transients Without Scram.

BOL: Begining Of Life

CAMP: Code Application and Maintenance Program

CSN: Consejo Seguridad Nuclear

FW: Feed Water

HPIS: High Pressure Injection System

MOL: Middle Of Life

MSR: Moisture Separator Reheater

NPP: Nuclear Power Plant

PRA: Probabilistic Risk Assessment

PRZ: Pressurizer

PWR: Pressurized Water Reactor

SBLOCA: Small Break Loss Of Coolant Accident

SG: Steam Generator

UNESA: Unidad Eléctrica S.A.

1. *INTRODUCTION*

In 1986 the Asociación Nuclear Ascó (ANA) created a group for plant and core thermal-hydraulic analysis. The objectives of the group are as follows:

1) Create and update core and plant thermal-hydraulic models based on best-estimate criteria.
2) Provide off-line engineering support to the different technical branches of ANA (i.e., technical services, reactor operation):
    a) Analyze operating events that result in event reports.
    b) Assess plant systems and/or equipment modifications as well as plant operating procedures and emergency instructions.
    c) Analyze plant behavior under incident or accident conditions in the above mentioned cases.
    d) Scenarios and core damage evaluation for probabilistic risk assessment.
3) Review final safety analysis report transients and accidents based on best-estimate criteria.
4) In the future and if appropriate, participate in Ascó individual plant examination.

The plant analysis activities developed so far include the following:

1) Implementation of RELAP5/MOD2 and migration to RELAP5/MOD3.2/1/.
2) Preparation of the thermal-hydraulic model of both the primary and secondary system /2/.
3) Preparation of the kinetic model specifically adapted to Ascó.
4) Simulation of control and protection systems /3/.
5) Revision and detailed study of all start-up tests /20/ and every transient that has occurred in each unit along these twelve years (a total of 65 actual cases were studied with the model of plant for RELAP5/MOD2 and RELAP5/MOD3.2).
6) Selection of transients suitable for assesment.
    a) Black-out.

b) Turbine power step.

c) Run-back of turbine because of one feedwater pump trip

d) Turbine trip with all systems available.

e) Reactor Trip because of low-low level in one SG as a result of a closure of the isolation feedwater valve.

f) Loss of load 100-50%

g) Transition mode of the feedwater valves control system

7) Simulation of the above seven transients and adjustment of control parameters /4/, /5/, /6/, /7/, /8/, /9/, /10/.

8) Participation in the International Code assessment and Application Program /11/.

9) Participation in the Code Assessment and Maintenance Program .

10) Analysis of transients such as small-break-loss-of-coolant accident (SBLOCA) /21/, /22/, anticipated transient without scram (ATWS), and others for PRA studies.

At this point, the adjustment and qualification process is considered to be completed. Sufficiently accurate predictions with meaningful sets of measured data provide validation of both the model and the procedures to be used in the future to analyze various transient and accident scenarios of general interest.

The analysis presented in this report has to do with transients related to AMSAC (ATWS Mitigating System Actuation Circuitry) actuation.

Early in 1989, when AMSAC system was installed in Asco NPP, a set of calculations were performed /23/ with the purpose of getting a general knowledge of the phenomenology of related transients. At this time the model of the old configuration of the plant was used (old SGs and old controls).

Today, as the plant has changed (see section 2.1 'Plant Description') and the model has improved its level of qualification, it is interesting to re-study the same subject with a larger scope of analysis.

A transient explanation is given in section 2.2 after a brief description of the plant itself (section 2.1).

Chapter 3 is a summary of what is included in the model and chapter 4 explains the steady state calculation for nominal conditions.

As the transient will be analyzed under two different points of view, chapters 5 and 6 will explain each one of them. The first one consists of a calculation with a best-estimate code (RELAP5/MOD3.2) using the same boundary conditions and setpoints than those defined by the vendor in its calculations /12/. The second one has to do with a sensitivity analysis of the transient under different kinetic parameters (BOL vs MOL).

Run statistics, conclusions and references can be found in chapters 7, 8 and 9 while diagrams showing the nodalization used are in Annex 1 .

2. *PLANT AND TRANSIENT DESCRIPTION*

2.1. PLANT DESCRIPTION

Ascó Nuclear Power Plant is a two unit station of three loop Pressurized Water Reactors (PWR) of Westinghouse design. The first criticality was reached in June 1983 in Ascó I and in September 1985 in Ascó II. Today, unit I is in its twelfth cycle and unit II in its eleventh of normal operation.

The main characteristics of both units are given in Table 1. The core contains 157 fuel assemblies of (17x17 -25) fuel rods.

The steam generators have been recently changed, in October of 1995 for Unit I and in October of 1996 for Unit II. The new ones are SIEMENS-FRAMATOM (61W/D3).

At the same time a deep upgrade of the original control systems has been performed. The new controls are fully digital and allow to establish sets of data more suitable for assessment.

All other major components are standard Westinghouse components.

Main Characteristics of Ascó I and II Nuclear Station	
Thermal reactor power	2686 Mwth
Fuel	UO2
Number of assemblies	157
Fuel rods per fuel assembly	(17x17 – 15) = 264
Active length of fuel rods	3.657 m
Outside diameter of fuel rods	$4.75e^{-3}$ m
Cladding tube material	Zr-4
Cladding tube wall thickness	$0.655e^{-3}$ m
Average linear heat generation rate	17750 W/m
Absorber rods per control assembly	24
Absorber material	Ag – In – Cd
Number of coolant loops	3
Reactor operating pressure (PRZ)	15.51 MPa
Coolant Average Temperature	581.3 K

Coolant flow rate	14287 Kg/s
Steam Generator	
Type	61W/D3
Number	3
Reactor coolant pumps	
Type	Westinghouse 93-DS
Discharge head	86.25 m
Design flow rate	5.928 m^3/s
Speed	155 rad/s
Pressurizer	
Height	12.835 m
Diameter (inner)	2.134 m
Volume	39.64 m^3
Operating saturation pressure	15.51 MPa
Heating power of the heater rods	1.40 Mwe
Steam/Power Conversion Plant	
Feed Water flow rate	497.5 Kg/s (loop)
Main steam flow rate	1492 Kg/s
Steam moisture at steam generator outlet	0.25 %
Feedwater temperature	497.05 K

Table 1. Main Characteristics of Ascó I and II Nuclear Station.

2.2. TRANSIENT DESCRIPTION

After a loss of normal feedwater, the reactor protection system and the average temperature control system are assumed to fail. So there is no reactor trip signal as well as no rod insertion. At this point, the AMSAC (ATWS Mitigating System Actuation Circuitry) generates the turbine. trip signal and starts the auxiliary feedwater pumps to mitigate the transient.

In table 2 a description of the main events are presented for the RELAP study.

Sequence of events of the transient	
Event	Time (seconds)
Loss of main feedwater	0.
Main feedwater flow completely lost	4.
Pressurizer relief valves lift	31.
Low SG level AMSAC setpoint reached	41.
Turbine Trip via AMSAC signal	42.
Auxiliary feedwater pumps start via AMSAC signal	71.
Maximum RCS pressure reached	94.
Primary pumps trip	150.
Main steam isolation signal and HPIS start	150.

Table 2. Sequence of events of the transient.

3. *MODEL DESCRIPTION*

The model corresponds to the plant configuration explained in section 2.1. It includes the new SGs as well as the new controls.

Diagrams 1 y 2 show the model used to simulate the plant. It consists of 152 volumes with its correspondent junctions, 51 valves, 27 time dependent volumes, 8 time dependent junction, 6 heat structures, 466 control variables and 351 trips. The model includes the vessel, the three primary loops, the pressurizer, the three steam generators, the three secondary loops, and the steam lines. The turbine, condenser, moisture separator reheaters, suction lines of main and auxiliary feedwater pumps are modeled as time dependent volumes.

3.1. THERMAL-HYDRAULIC MODEL

a) Primary system

The model of the primary system (diagram 1) includes the main components of the plant. The core is modeled by volume 120 and the proper heat structures. Volume 130 simulates the by-pass region between the core baffle and the core barrel. Volume 140 models the upper plenum and volumes 150 and 160 the vessel upper head. The three hot lines depart from the core upper plenum. Loop 3 (volume 410) is connected with the pressurizer through volume 510. The pressurizer is divided in 2 volumes.

The lower one (volume 520) is divided into 19 nodes. Heat structures, simulating the pressurizer actual heaters, are attached to the first three nodes. Volume 525 is a branch in order to model the junctions connecting the pressurizer with the safety and relief valves and with the spray system.

Volume 420 models the remaining hot leg. Volumes 430 and 440 simulate the water boxes of the steam generator. The primary side of the steam generator is modeled by volume 431 divided into 20 nodes. Volume 460 models the primary coolant pump. Proper homologous curves, given by the vendor, have been used for this purpose. Volumes 268, 368, 468 model the HPIS. Time dependent junctions 267, 367, 467 model the high pressure injection pumps by means of the pump characteristic curves. Accumulators are also modeled.

b) Secondary system

The model of the secondary system (diagram 2) starts with time dependent volumes 662 and 654, that represent the suction lines of the main feedwater pumps. Volumes 679, 693, and 682 represent the suction lines of the auxiliary feedwater pumps.

Volumes 664 and 656 model the main feedwater pumps. Proper homologous curves, given by the vendor, have been used for this purpose.

Time dependent junctions 690, 695, 694 model the auxiliary feedwater pumps by means of the pump characteristic curves.

Volumes 665, 666 and 657, 658 represent lines from pumps to hot-collector (volume 668). The feedwater high pressure heater is modeled by a heat structure between the first node of volumes 666 and 658 with volume 942. This last volume is a time dependent volume with a temperature function of the turbine power.

Volumes 869, 870, 872, 875 (loop 3) model lines of main feedwater to the steam generator. Volumes 888, 852 model lines from auxiliary feedwater pumps to main feedwater connection. Their correspondent check, control and isolation valves are also modeled.

The SG downcomer (diagram 1) is simulated by means of volumes 800, 801, 822 and 825. The steam generator riser is modeled by volume 810. The steam separator with volume 820. Volumes 830, 840 and 850 model the steam dryer and the dome of the steam generator. The steam line starts at volume 880. Safety and relief valves (components 886 and 884) are connected to volume 881. Component valve 885 models the isolation valve. Time dependent volumes 994 and 999 represent the free atmosphere. The steam is conducted throughout volume 883 to the steam collector, volumes 900 and 904. Finally valves from 931 to 938, 903 and 907 model the by-pass to condenser valves, the turbine stop and control valves respectively. And valve 917 represents the steam flow to moisture separator reheater.

Proper heat structures are used to thermally connect the primary side of the steam generator with secondary side. Actual values are used for all the variables. Fouling factor coefficient is introduced in order to achieve de actual heat transfer rate without any change in primary average temperature and secondary pressure.

The data used to model volumes, junctions and valves as well as heat structures were taken form plant design information /13/.

3.2. KINETIC MODEL

The kinetic model /3/ was prepared using the RELAP5/MOD3.2 space-independent reactor kinetic option with data from the ANA Nuclear Analysis Group. The model includes a scram table of reactivity versus time. The total control rod drop time is the actual value measured at plant. This table is activated by reactor trip.

The control model supplies the reactivity of the C and D control rod banks.

This control reactivity is added to feedback reactivities calculated by the kinetic model from the data supplied for the specific burn-up condition of each transient.

3.3. CONTROL AND PROTECTION SYSTEMS MODEL

The protection and control systems are modeled using RELAP5/MOD3.2 control blocks and following specific setpoint studies, logical diagrams and technical specifications of the plant /3/, /14/, /15/.

The model includes the following systems:

a) Reactor trip system.

The reactor can trip in the RELAP5/MOD3.2 model because of the following effects:

- Power range, low range, high level
- Power range, high range, high level
- Overtemperature
- Over power
- High pressurizer pressure
- High pressurizer water level
- Low pressurizer pressure
- Loss of primary coolant flow
- Low-low steam generator water level
- Turbine trip
- Safety injection
- Manual reactor trip

b) High pressure Injection System.

Using the following signals:

- Low pressurizer pressure
- High steam line differential
- High steam line flow coincident with Low steam line pressure or Low-Low Tavg.

c) Turbine Trip and Control System

The position of the turbine control valve is controlled as a function of the difference between the required power and actual Power, with the proper control block to model the actual logic of the plant. Turbine run-back signals have also been modeled.

The Turbine Trip (closure of the turbine stop valve) is modeled. The signals of Safety Injection, very high steam generator level, reactor trip, AMSAC signal and manual signal are used to trip the turbine.

d) Feed Water Control System

The feed water control system of the main and by-pass valves and the pumps have been modeled as shown in diagram 2. The auxiliary feed water system is also included in the model.

e) Pressurizer level and pressure control system

The model of the pressure control system actuates upon heaters, spray valve and charging pump. Pressurizer safety and relief valves and level control systems are also simulated.

f) Average Temperature Control System

The average temperature control system modeled with RELAP5/MOD3.2 controls both the primary average temperature and the primary-secondary power mismatch.

Other systems modeled are:

g) Steam Dump Control System

h) Steam line isolation logic.

i) Safety and Relief valves of the secondary.

j) Primary pumps trip.

k) General Variables.

l) Permissive and interlock circuits.

4. *STEADY STATE CALCULATION*

A steady state calculation was performed with the plant at 100% rated condition. The objective is to obtain a stable condition to start transients.

In Table 3 a comparison between the model results and the plant data is given for the main plant variables.

Steady State Parameters		
VARIABLE	MODEL(RELAP5/MOD3.2)	PLANT(/18/)
Primary mass flow rate (Kg/s)	4760.8	4761.2
Vessel ΔT (K)	32.6	32.7
Primary Pressure (MPa)	15.513	15.516
Primary Avg. Temperature (K)	581.5	581.5
Recirculation ratio	3.68	3.65
SG Narrow Range Level (%)	50.6	50.6
Secondary Pressure (MPa)	6.746	6.733
Steam Mass Flow (Kg/s)	494.5	496.1

Table 3. Steady State Parameters.

5. TRANSIENT CALCULATION AT BOL CONDITIONS.

According to what is establish by the Spanish Nuclear Authority (CSN) ANA must demonstrate that the setpoints implemented in the plant for the AMSAC system are still valid for the new steam generators.

The plant vendor issued a report, analyzing the transient with a design code /12/ and fulfilling the regulatory requirements.

In this section, the results of a base case calculation with ANA's RELAP5/MOD3.2 /1/ model will be presented. The calculation has been performed using the same boundary conditions and setpoints than those defined by the vendor in its report /12/.

The following assumptions are made in order to have the same vendor boundary conditions:

- No credit for automatic reactor trip.
- Initial normal full power operation early in core life.
- No credit for automatic control rod insertion as reactor coolant temperature rises.
- No credit for pressurizer spray in pressure reduction.
- Pressurizer pressure relief through all power operated and spring loaded relief valves available.
- Main feed water falls to 0 % of nominal flow in the first four seconds of the transient.
- The AMSAC signal is actuated when the steam generator water level reaches 12% of narrow range span. Following the signal, a one second delay is assumed for turbine trip.
- Auxiliary feedwater starts 30 seconds after the AMSAC setpoint is reached, at the temperature of 54.4 °C and a total minimum flow of 259 m^3/h, equally delivered to the three steam generators.
- The moderator temperature coefficient is –12.5 pcm/°C corresponding to a conservative value during more than 95% of the reactor time life (Ascó Unit 1 Cycle 12)

The vendor also makes an additional assumption which is the following:

- Primary to secondary heat transfer area is reduced as the steam generator shell-side water inventory falls below the value necessary to cover the tube bundle.

This assumption is not required in RELAP calculations since the code calculates the heat transfer between primary and secondary in all conditions.

After a through analysis of the vendor report, two supplementary assumptions have been made in RELAP5/MOD3.2 study.

- No credit to turbine run-back .
- Secondary relief valves remain closed.

The results of RELAP5/MOD3.2 calculations are given in the following figures:

- Figure 1. Reactor Power.
- Figure 2. Average Coolant Temperature.
- Figure 3. PRZ Water Volume.
- Figure 4. PRZ Pressure.
- Figure 5. Primary relief and safety valves flow.
- Figure 6. SG Pressure.
- Figure 7. Total Steam-dump flow.
- Figure 8. Secondary safety valves flow.
- Figure 9. SG Water Mass.
- Figure 10. SG Narrow Range Level and Turbine Power.
- Figure 11. Heat Transfer from primary to secondary side.
- Figure 12. Auxiliary FW flow.
- Figure 13. Core Flow.
- Figure 14. FW Flow.
- Figure 15. Total Steam Flow.

- Figure 16. Total Reactivity.
- Figure 17. Temperatures. Volume 670.
- Figure 18. Vapor Generation. Volume 670.

At time zero the loss of feedwater takes place and the main feedwater flow is completely lost in 4 seconds (table 2). As a result of this the SG water mass (figure 9) and the SG Narrow Range Level decrease (figure 10).

When two of the three SG Narrow Range Level signals reach the value of 12%, the AMSAC signal is produced and 1 second later the turbine is tripped (figure 10).

With the turbine trip, the secondary pressure increases (figure 6) and the heat transfer from primary to secondary side (figure 11) falls.

Due to heat transfer decrease, the primary average temperature (figure 2) increases and generates a negative feedback reactivity (figure 16) which results in a reactor power decrease (figure 1).

Figure 12 shows auxiliary feedwater mass flows. According to boundary conditions auxiliary feedwater starts with AMSAC signal instead of main feedwater closure or low-low Narrow Range Level signal (17.6 %) (table 2). The same figure also shows a flow decrease (320 s.) due to secondary side pressure increase (figure 6). At that pressure the auxiliary feedwater pumps deliver a flow that corresponds to the boundary condition of 259m^3/h.

The pressurizer pressure predicted by RELAP can be seen in figure 4. When primary pressure reaches the corresponding setpoint relief and safety valves open (figure 5) to control the pressure peak. This pressure peak is smaller than that predicted by the vendor in about 2 MPa. This is due to a larger heat transfer to the secondary side during the first 100 seconds of the transient predicted by RELAP. The calculation considers not only steam-dump capacity (figure 7) and the secondary relief valves (figure 8), but also the steam flow to MSR (Moisture Separator Reheater), auxiliary feedwater pumps and others (see figure 15). This larger heat transfer from primary to secondary is also due to a lower pressure in SG's (figure 6).

As it can be observed in figure 14, RELAP5/MOD3.2 predicts several peaks in feedwater flow after second 150. These are due to flashing in the pipe between the feedwater isolation valve and the header of auxiliary and main feedwater. Once the isolation valve is closed (in the first four seconds of the transient) the water inside this pipe upstream of the check valve remains at the temperature of main feedwater (497 K). So when secondary pressure falls below the saturation pressure for that temperature (figure 17), the amount of vapor generated (figure 18) leads to a peak in feedwater flow (figure 14) and a sudden increase of heat transfer (figure 2). This increase produces two main effects:

- The first one is a rapid fall of the primary pressure (figure 4), which almost reaches the saturation pressure at the main pump suction line. This leads to the main pumps trip due to cavitation induced vibration (figure 13). Pump is assumed to cavitate when cold leg temperature is minus or equal to 6 °F below saturation temperature /16/.

- The second effect consists of a quick evaporation of the water within the steam generator, which activates the high steam line flow coincident with low steam line pressure signal. This produces the closure of the main steam valves and the start of high pressure injection.

The above explained flashing phenomenon has also been predicted for other plants /19/.

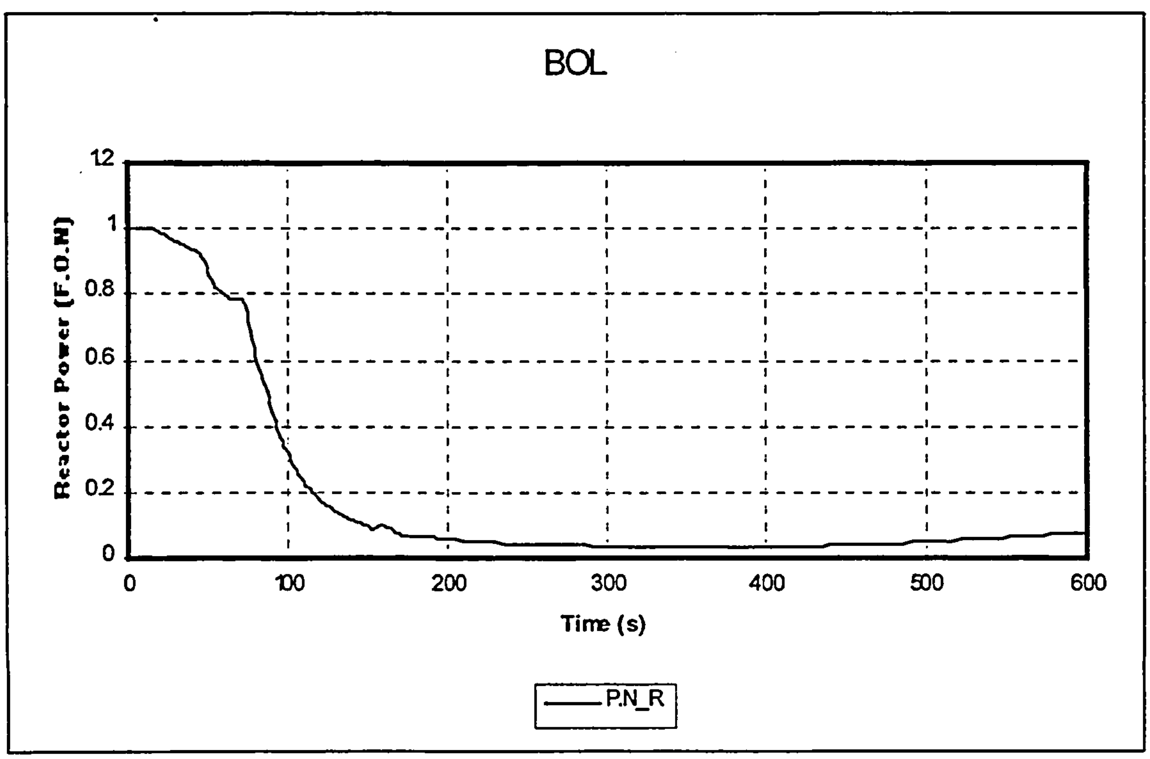

Figure 1. Reactor Power.

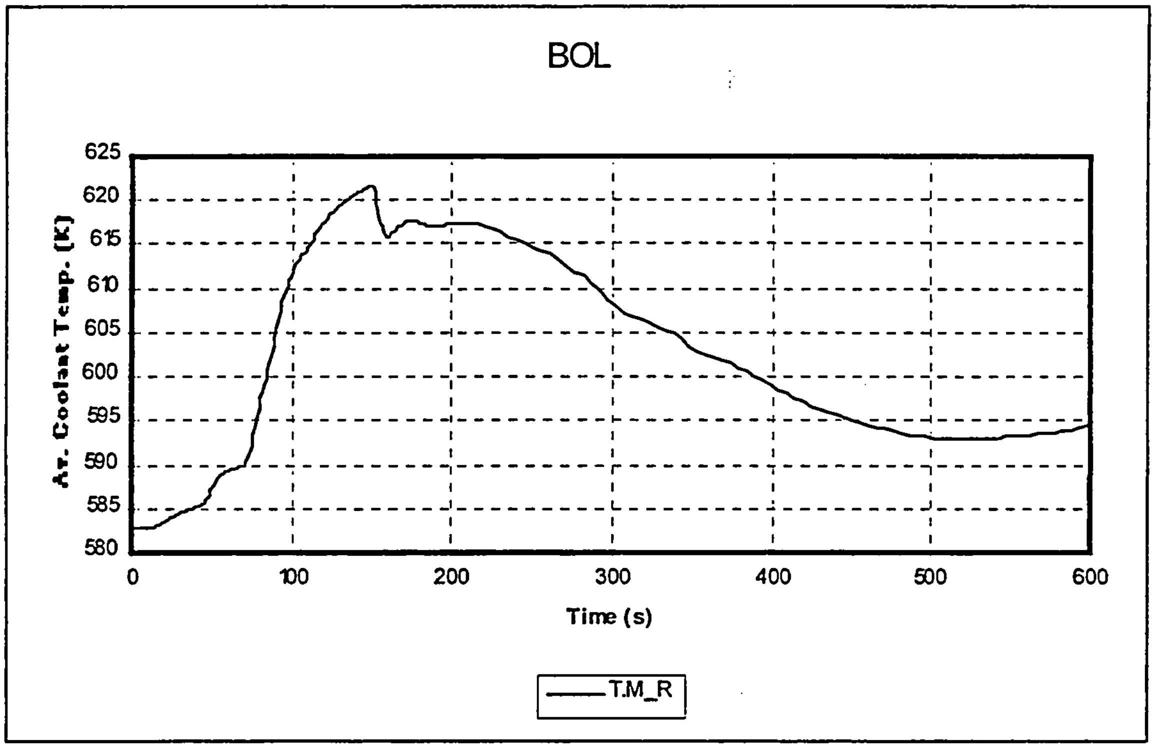

Figure 2. Average Coolant Temperature.

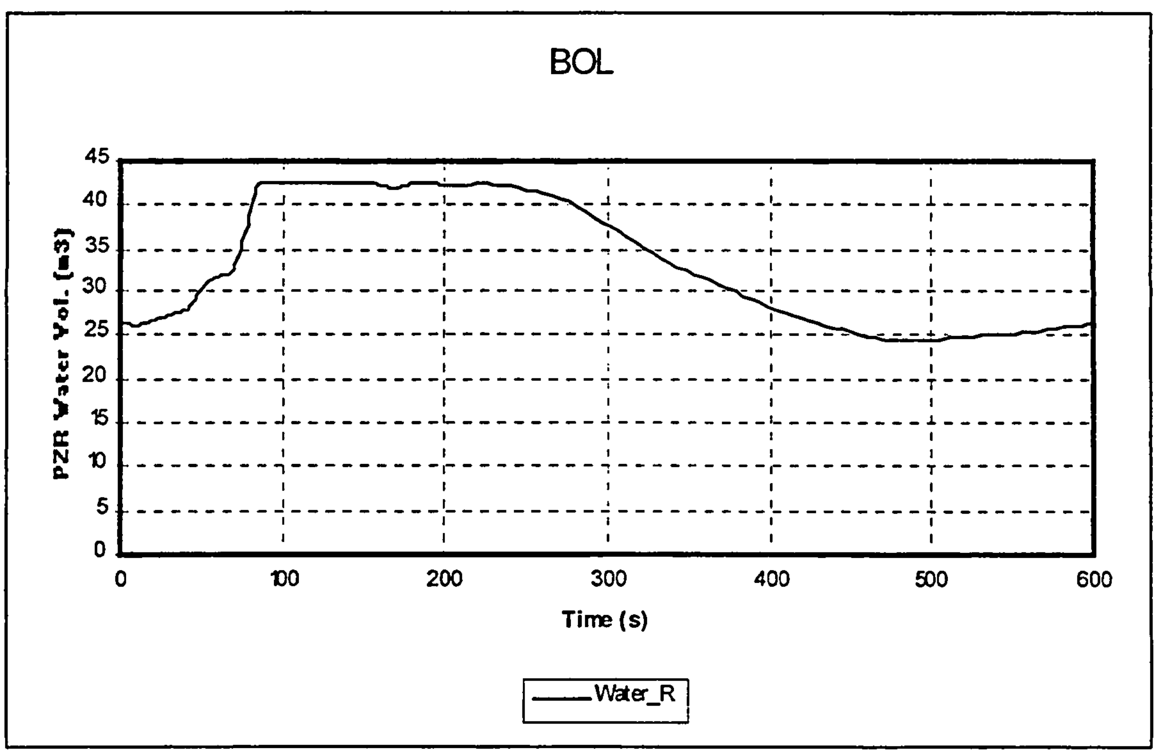

Figure 3. PRZ Water Volume.

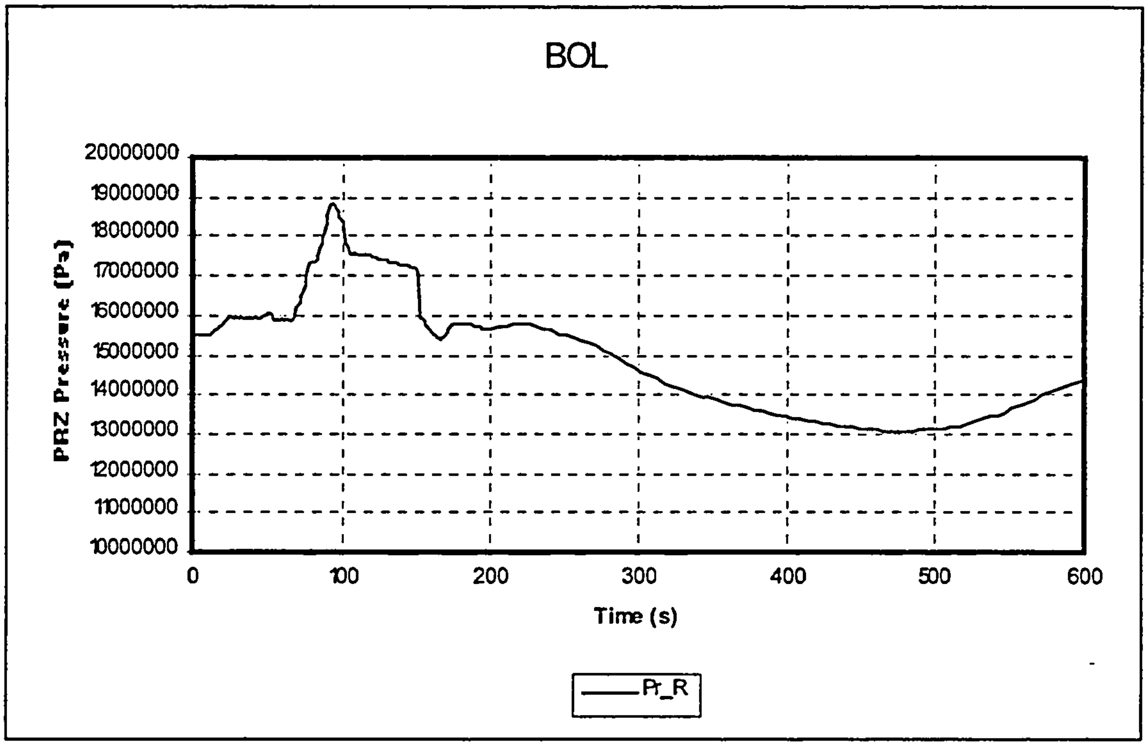

Figure 4. PRZ Pressure.

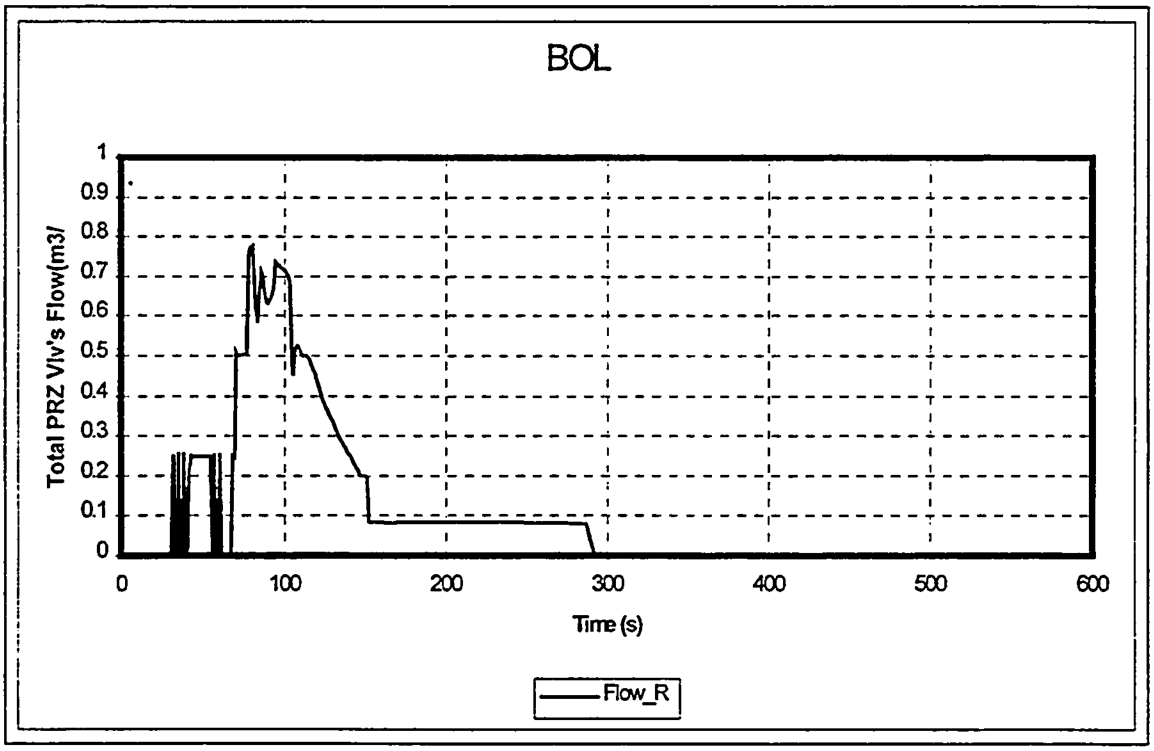

Figure 5. Primary relief and safety valves flow.

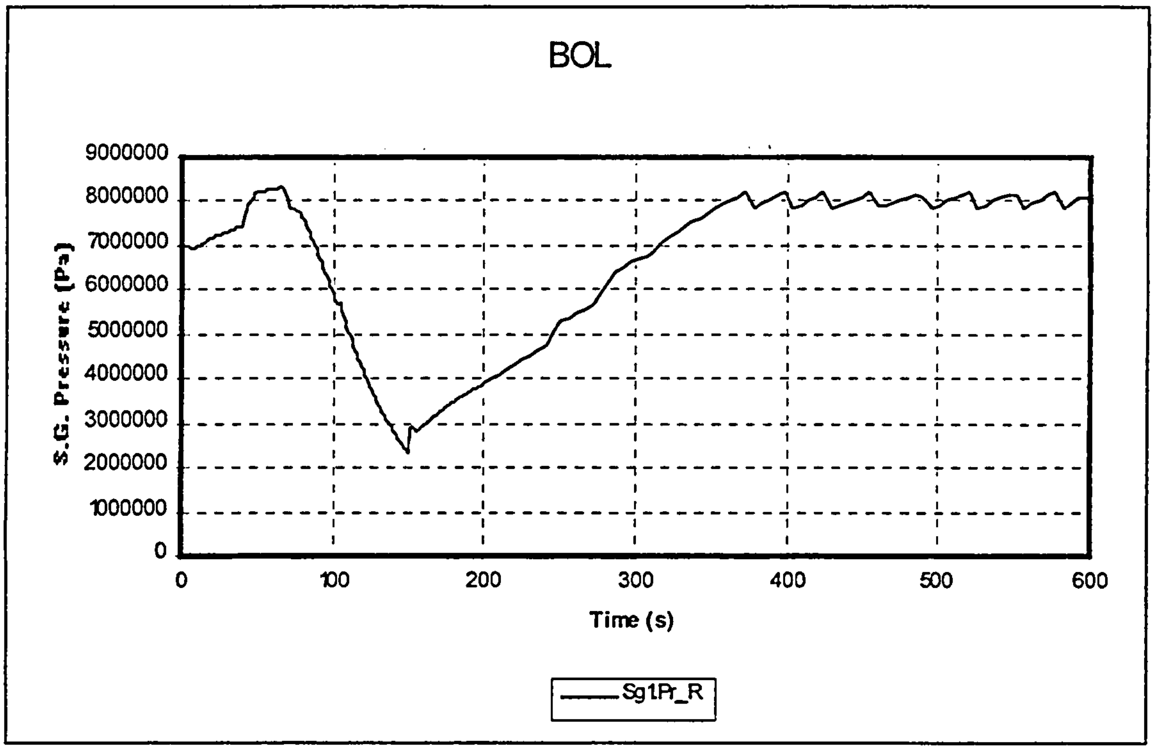

Figure 6. SG Pressure.

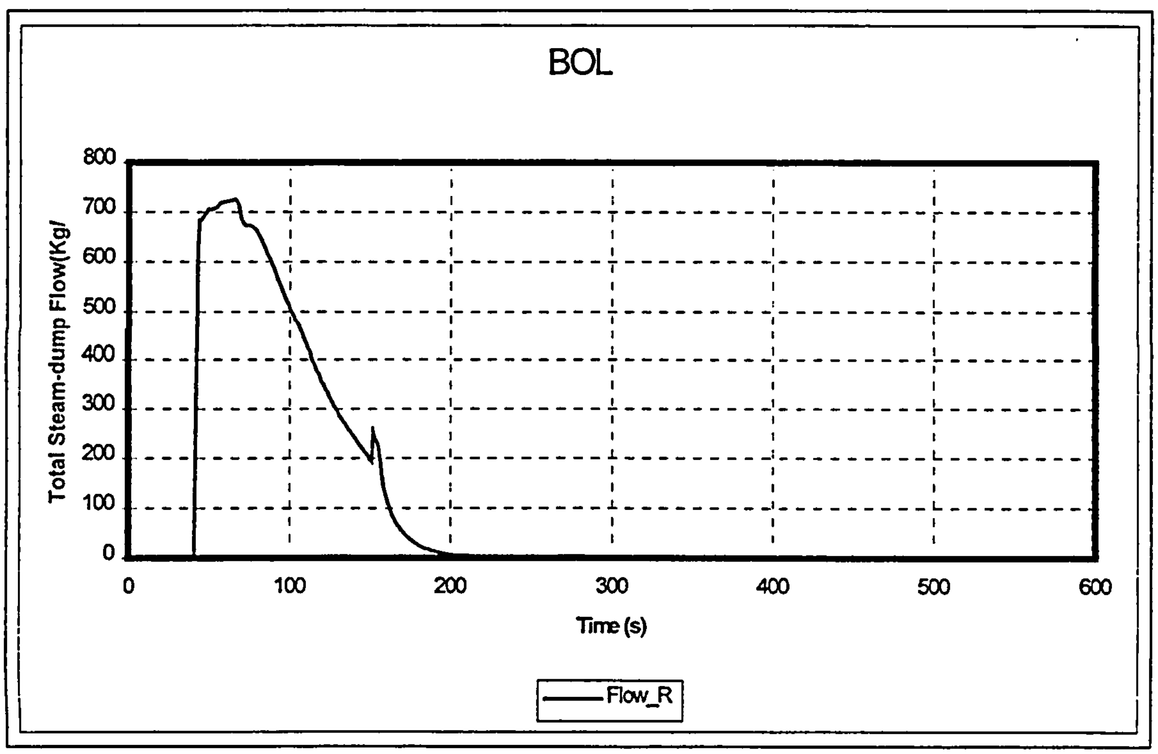

Figure 7. Total Steam-dump flow.

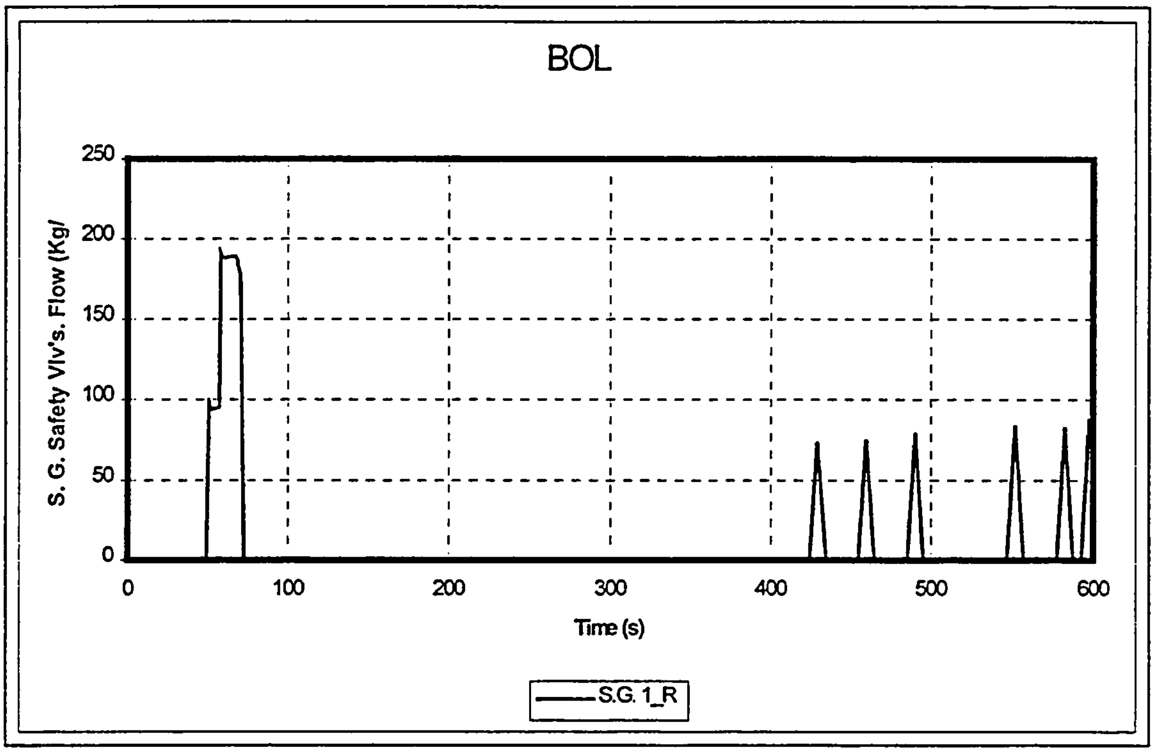

Figure 8. Secondary safety valves flow.

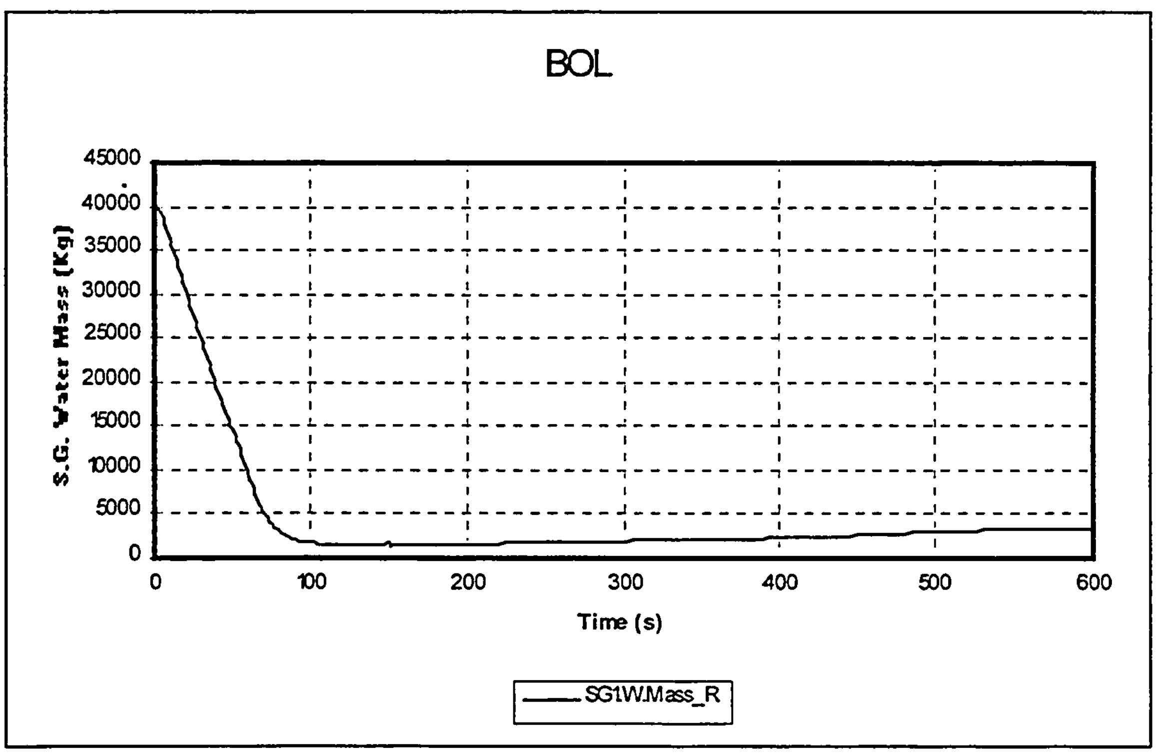

Figure 9. SG Water Mass.

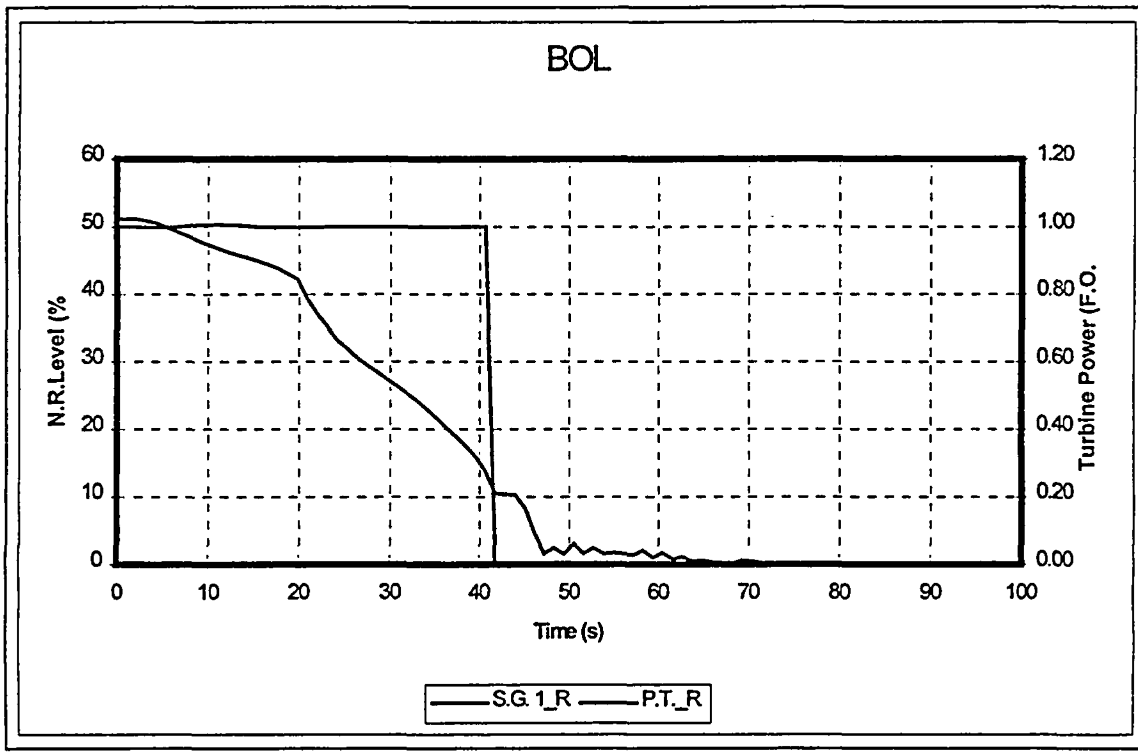

Figure 10. SG Narrow Range Level and Turbine Power.

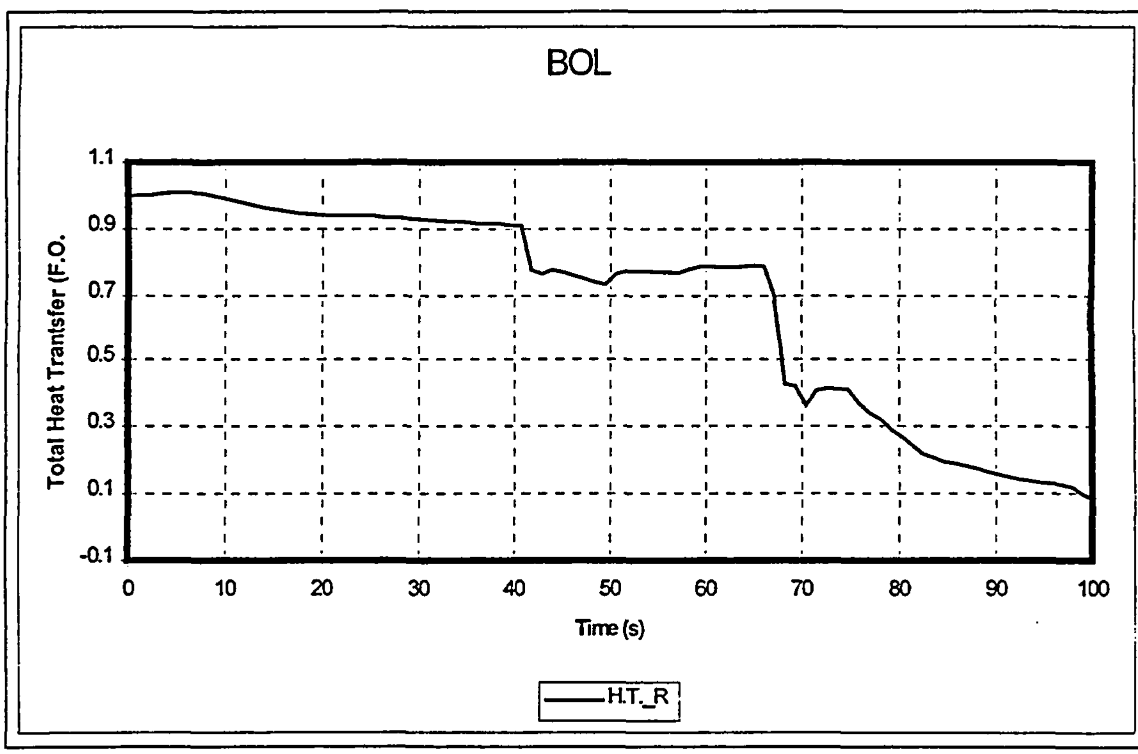

Figure 11. Heat Transfer from primary to secondary side.

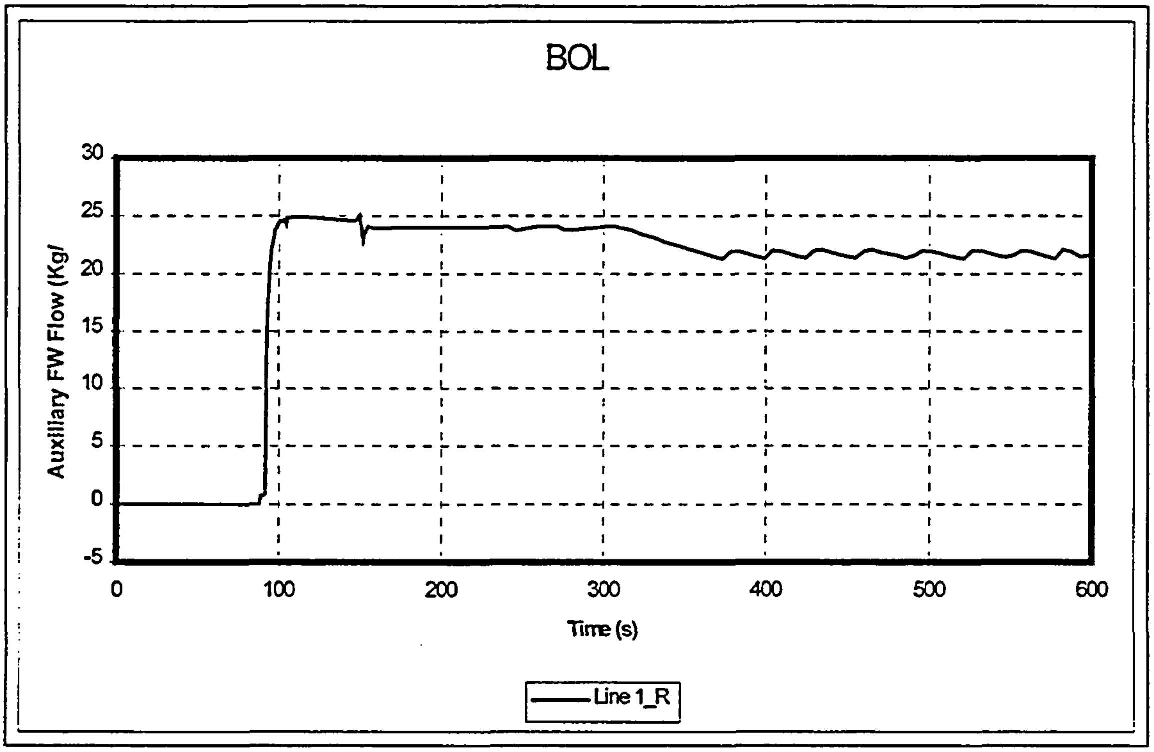

Figure 12. Auxiliary FW flow.

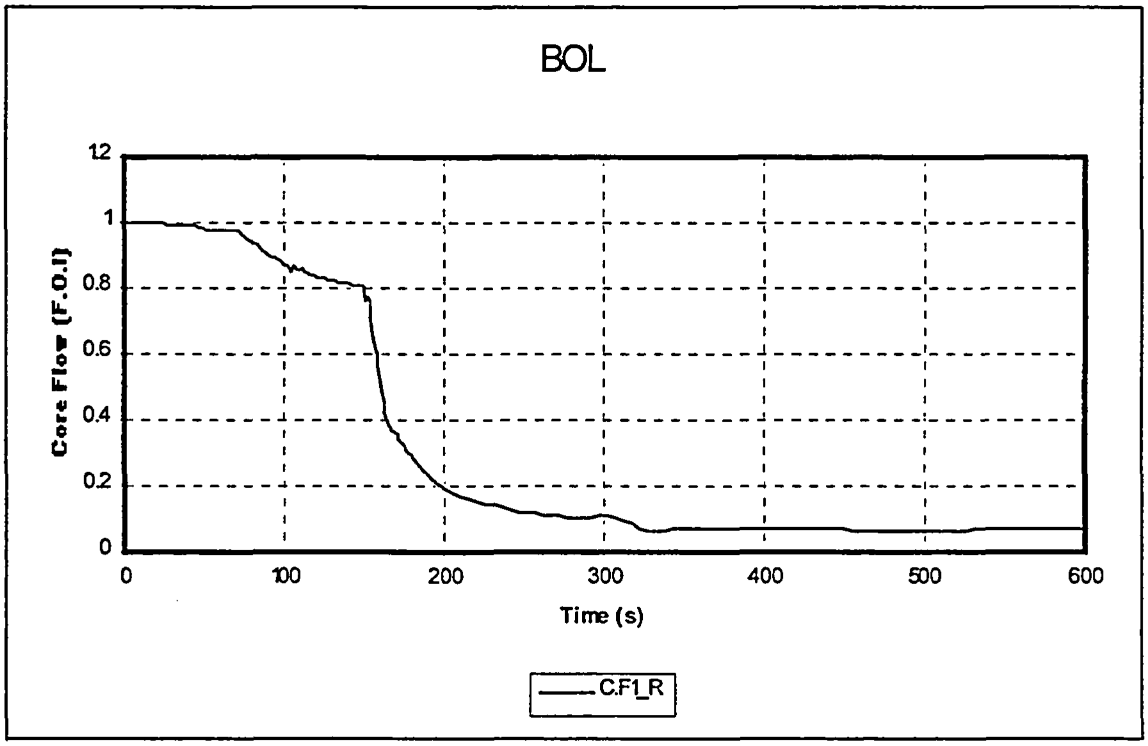

Figure 13. Core Flow.

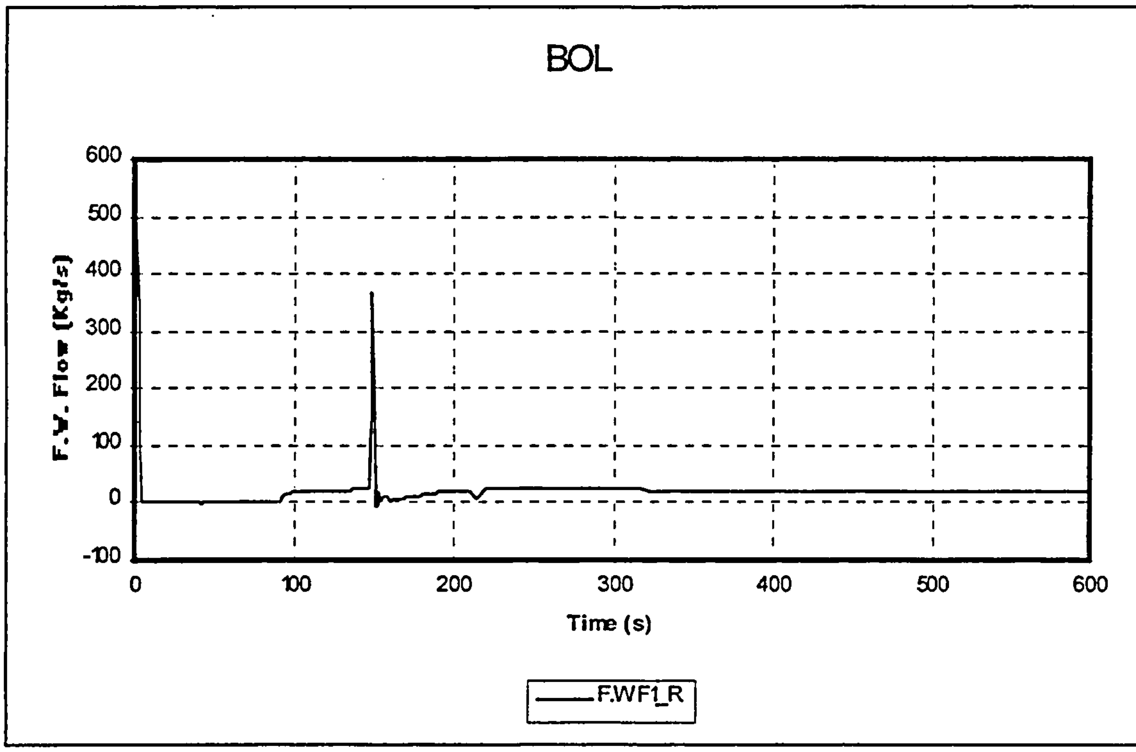

Figure 14. FW Flow.

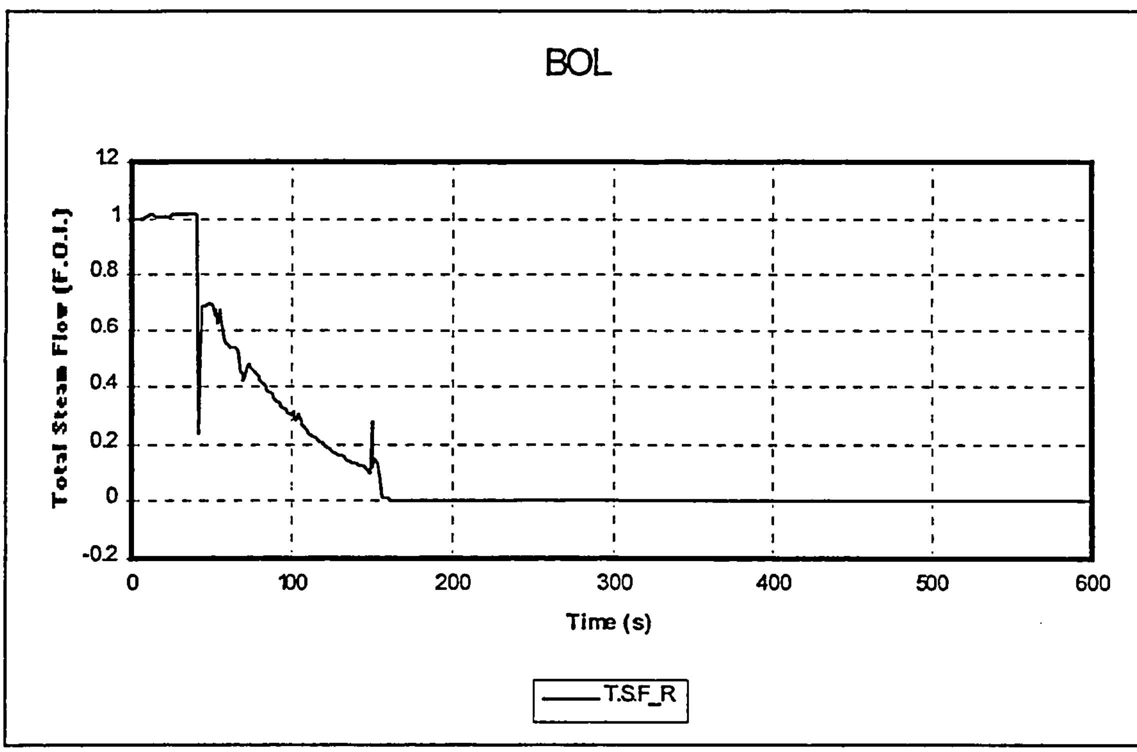

Figure 15. Total Steam Flow.

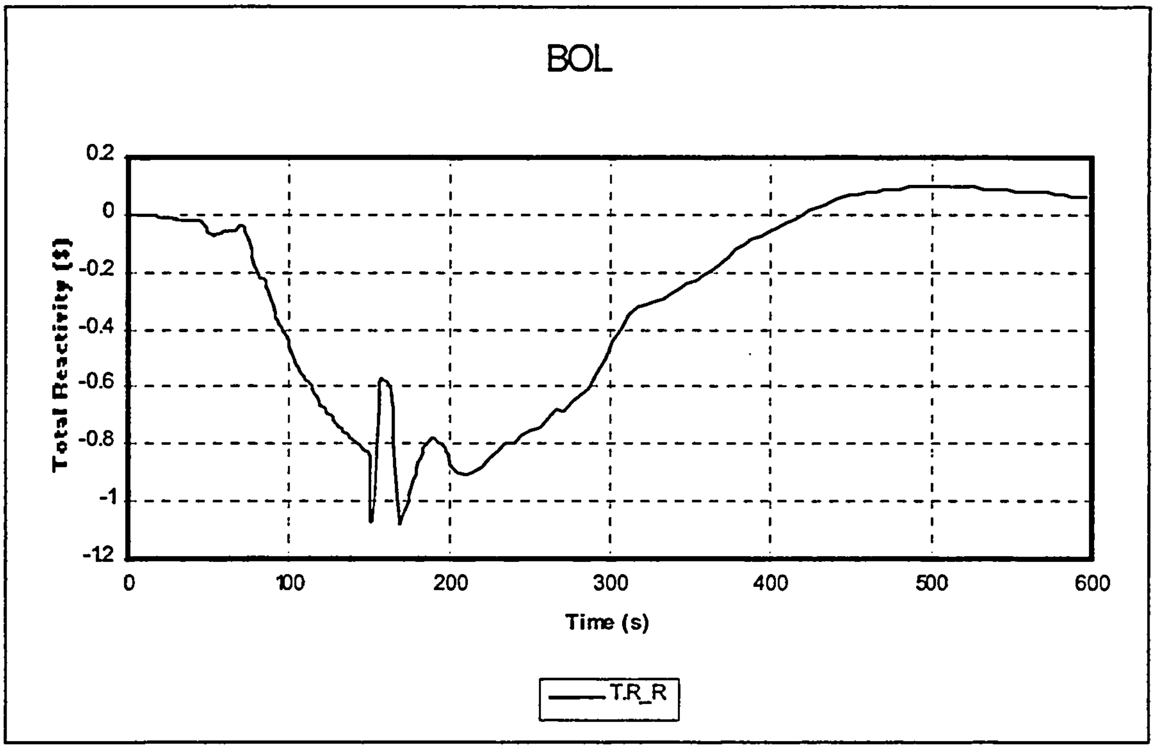

Figure 16. Total Reactivity.

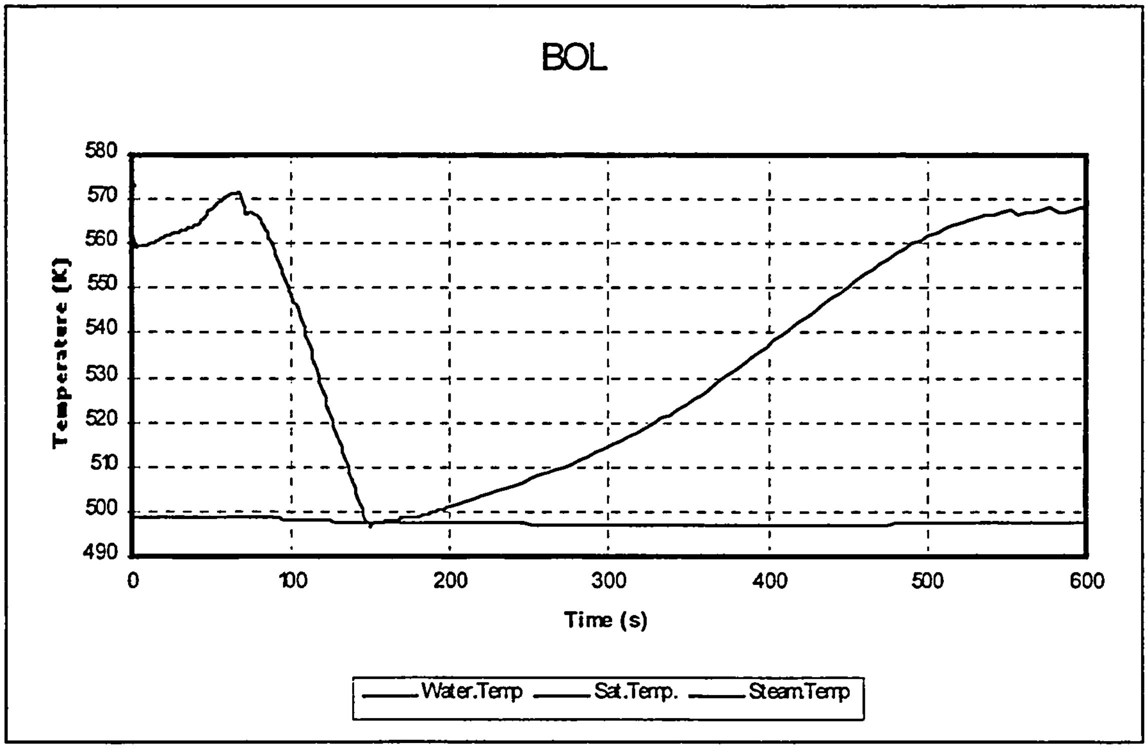

Figure 17. Temperatures. Volume 670.

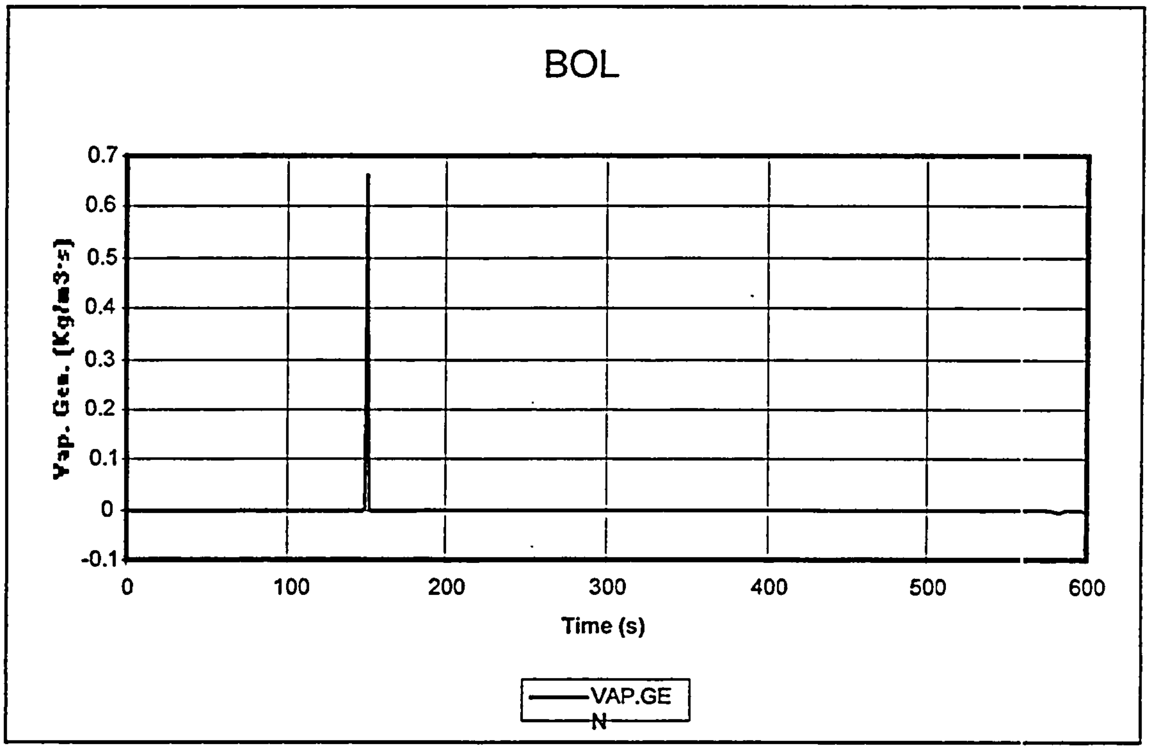

Figure 18. Vapor Generation. Volume 670.

6. *SENSITIVITY ANALYSIS TO KINETIC PARAMETERS.*

In this section, the influence of the kinetic parameters on the peak of pressure for the loss of normal feedwater ATWS event will be studied.

For this purpose the kinetic parameters of the BOL (above section), and the correspondent to MOL /17/ will be used.

All other conditions of the above section remain the same.

The comparison between the results obtained by RELAP5/MOD3.2 on both calculations is given in the following figures:

- Figure 19. Reactor Power.
- Figure 20. Turbine Power.
- Figure 21. Average Coolant Temperature.
- Figure 22. PRZ Water Volume.
- Figure 23. PRZ Pressure.
- Figure 24. SG Pressure.
- Figure 25. SG Water Mass.
- Figure 26. Core Flow.
- Figure 27. FW Flow.
- Figure 28. Total Steam Flow.
- Figure 29. Total Reactivity.

Figure 23 shows that the PRZ pressure predicted in the MOL case is smaller than that predicted in the BOL case. This is due to a more negative feedback reactivity producing a smaller reactor power during the rise of the pressurizer pressure (see figure 19 and 29).

As it can be seen in figure 24 the secondary pressure is the same from the beginning of the transient until the peak pressure in the pressurizer is reached, for both cases. So both cases have a similar heat transfer. The different kinetic parameters used in each case, lead to a different reactor power. This power is lower in the MOL case and produces a lower peak pressure by about 1.0MPa.

As explained in the above section, there is flashing phenomenon that leads to isolate the SGs, to start the HPIS and to trip the primary pumps.

The sudden fall of the average coolant temperature is also produced by the peak of feedwater flow (see figures 21 and 27).

Figure 24 shows how pressure increases until the opening of the SG safety valves.

The evolution of the steam generator water mass is shown in figure 25.

Figure 26 shows that core flow is never zero, meaning that natural circulation keeps cooling the core after the primary pumps trip.

The volume of water in the pressurizer keeps high because of the start of the H.P.I. (figure 22).

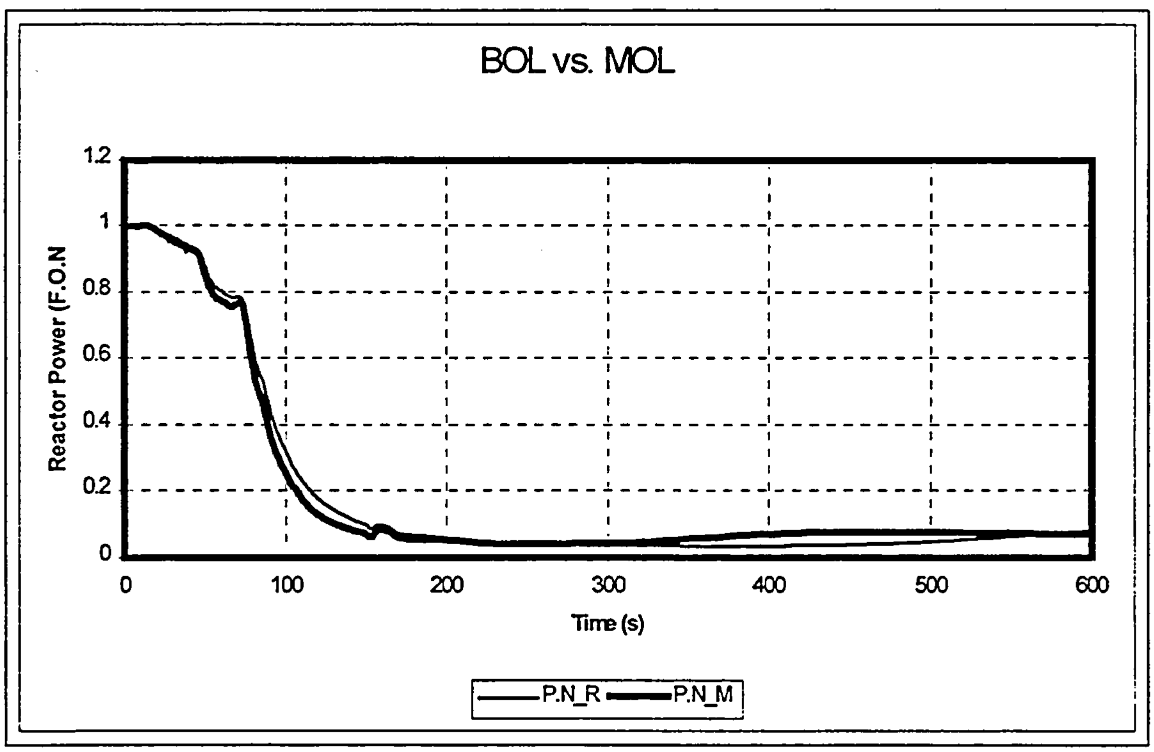

Figure 19. Reactor Power.

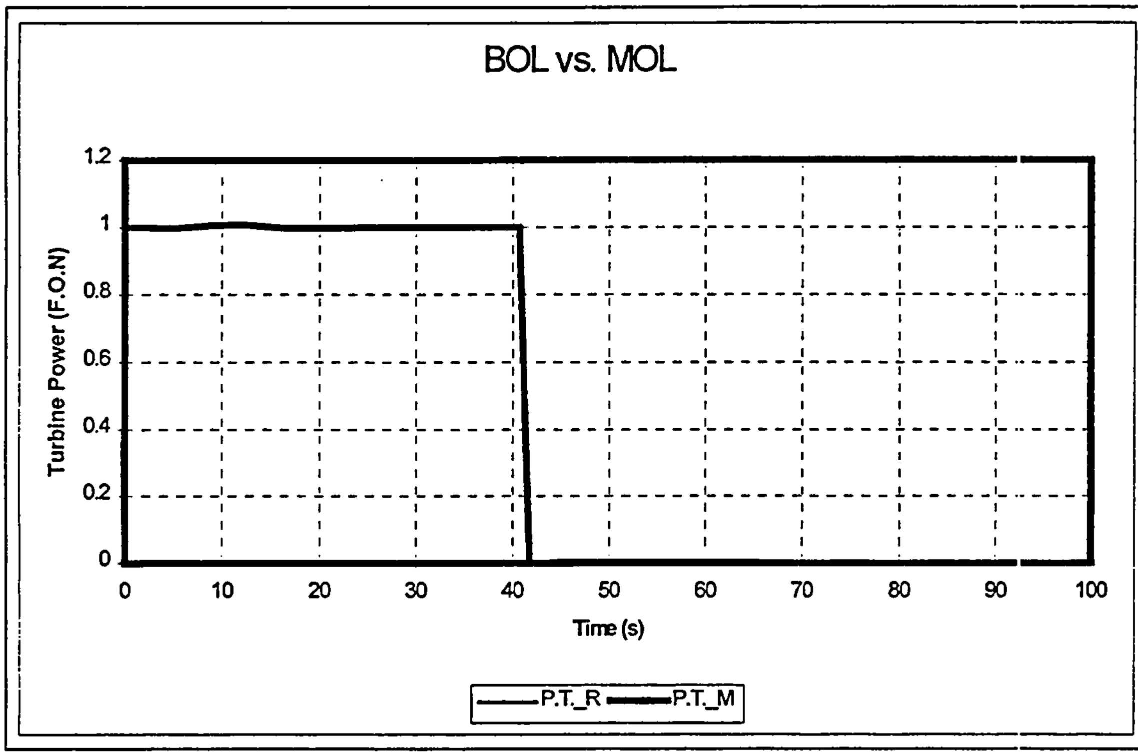

Figure 20. Turbine Power.

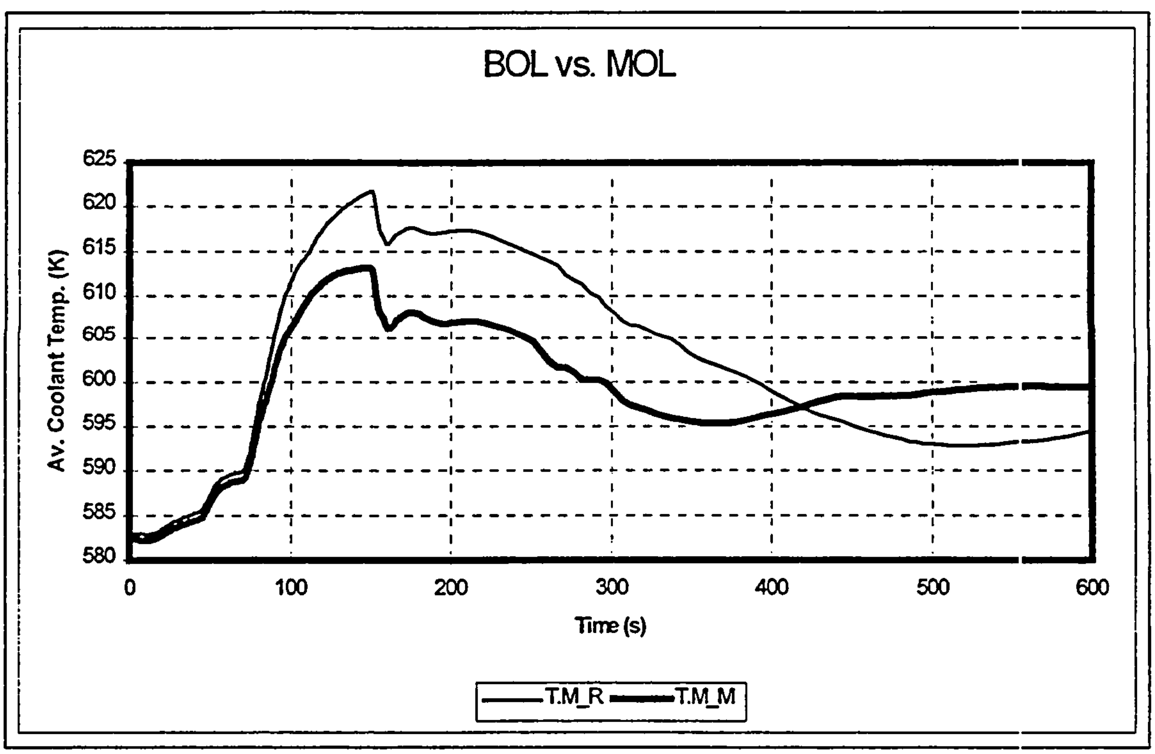

Figure 21. Average Coolant Temperature.

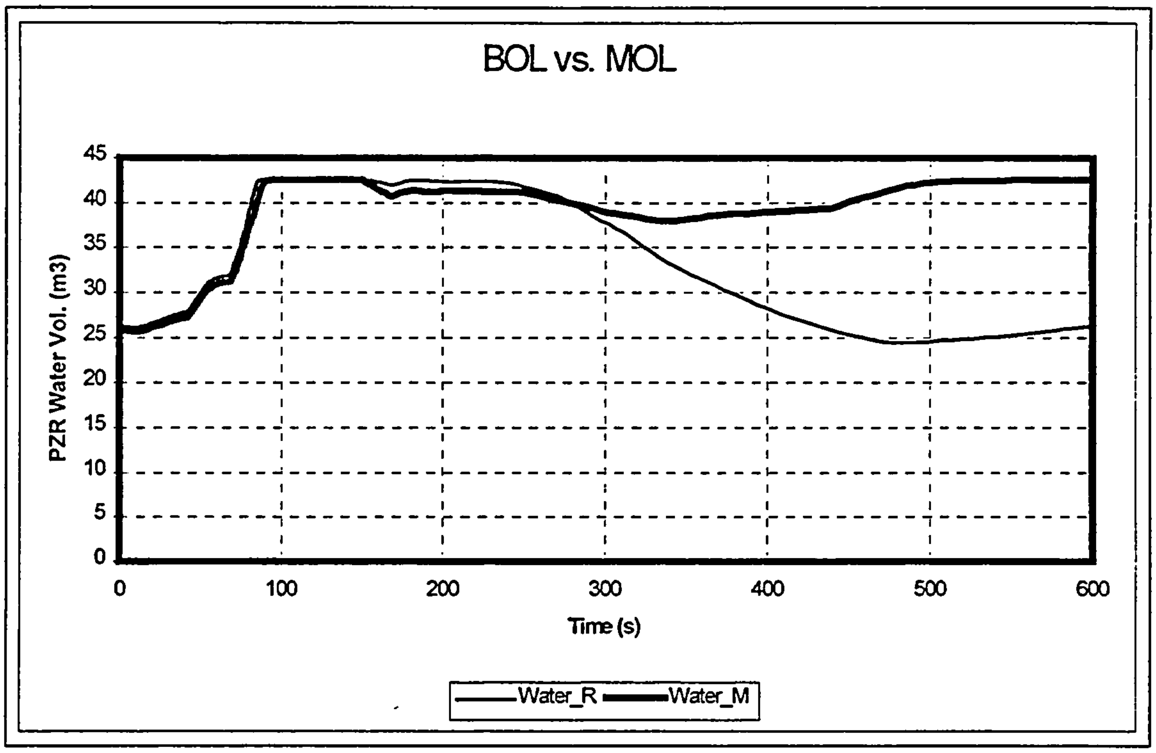

Figure 22. PRZ Water Volume.

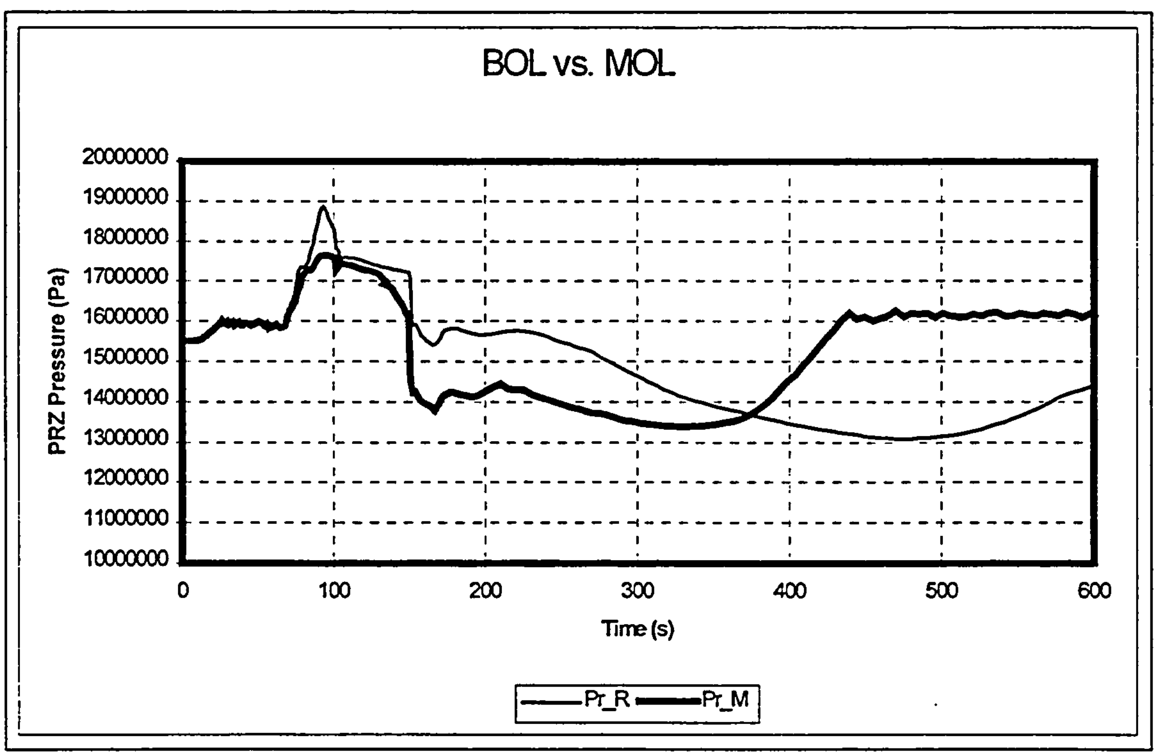

Figure 23. PRZ Pressure.

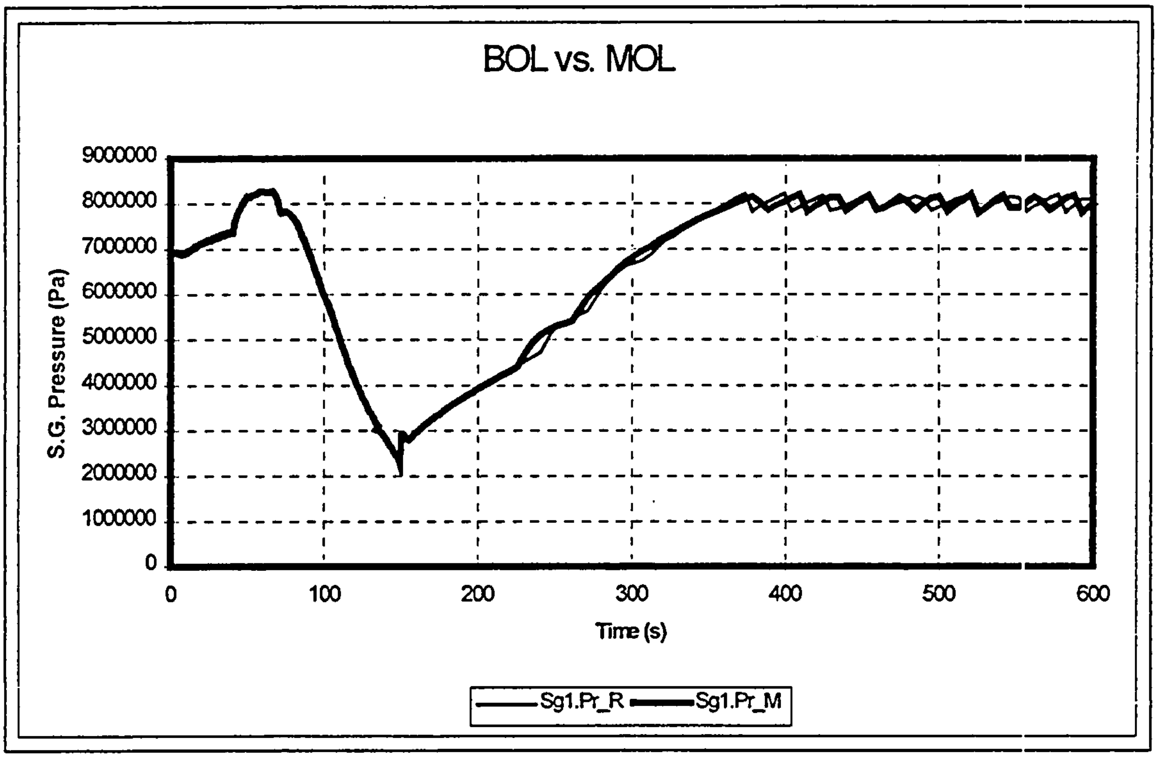

Figure 24. SG Pressure.

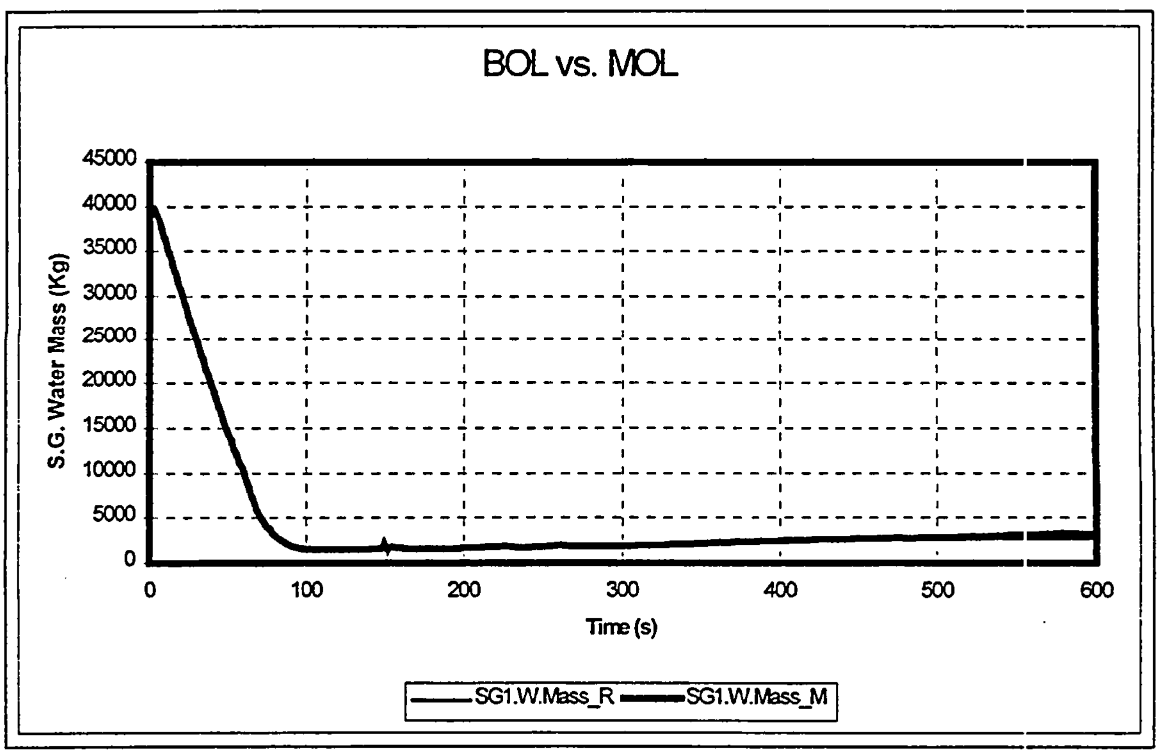

Figure 25. SG Water Mass.

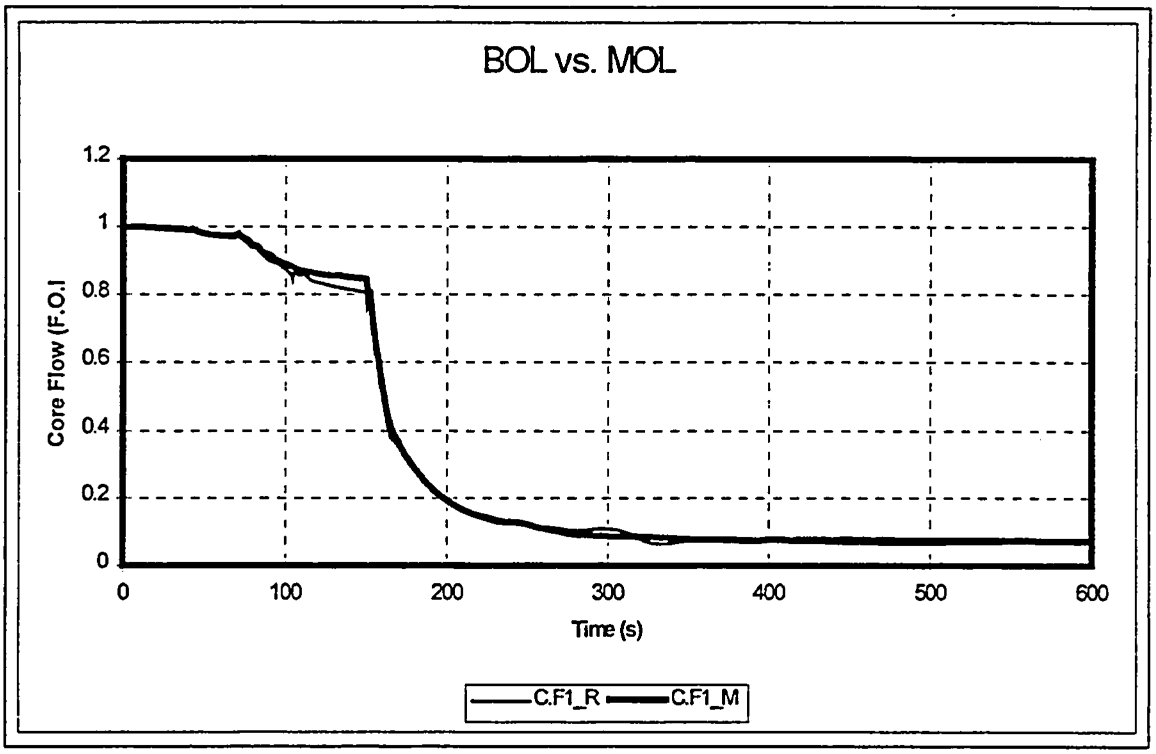

Figure 26. Core Flow.

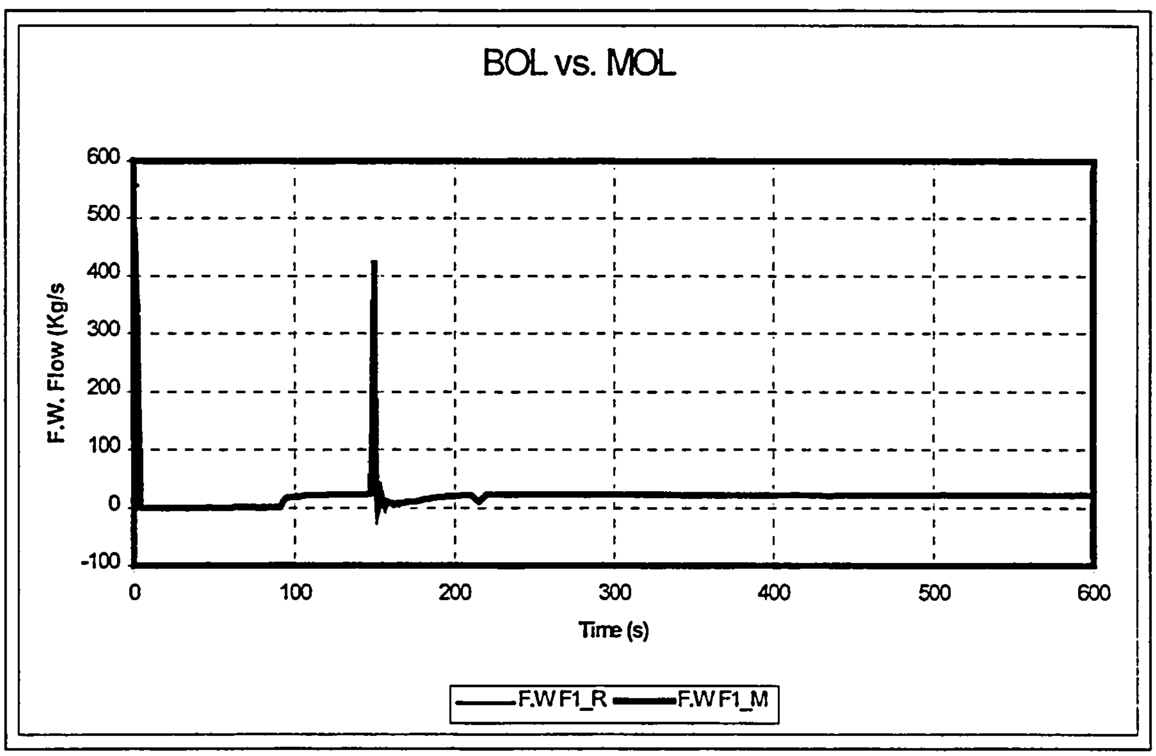

Figure 27. FW Flow.

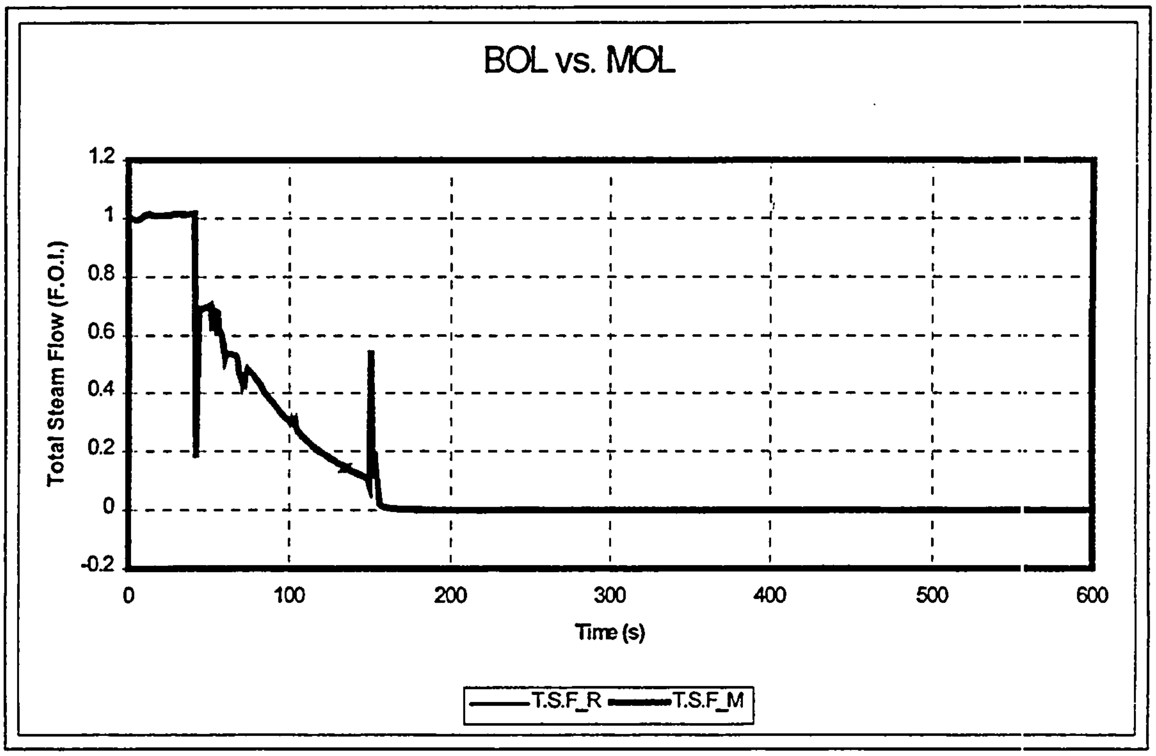

Figure 28. Total Steam Flow.

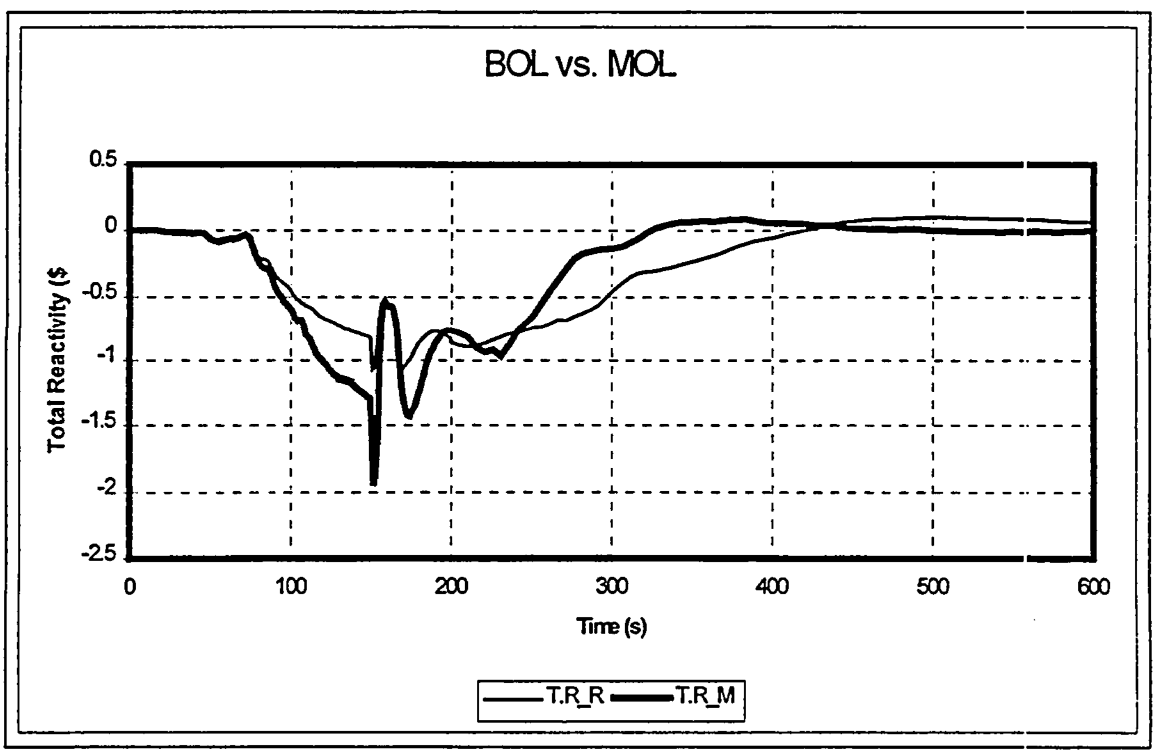

Figure 29. Total Reactivity.

7. *RUN STATISTICS*

Calculations were carried out on a HP 735/125 with UNIX-HP 10.0

RELAP5/MOD3.2 was used in all the calculations.

Run Statistics	
BOL case	
TRANSIENT TIME (s)	600.
CPU TIME (s)	1083.95
Total number of time steps	6049
CPU TIME/TRANSIENT TIME	1.81
MOL case	
TRANSIENT TIME (s)	600.
CPU TIME (s)	1129.19
Total number of time steps	6053
CPU TIME/TRANSIENT TIME	1.88

Table 4. Run Statistics.

8. CONCLUSIONS

A set of best-estimate calculations of transients meant to cause AMSAC actuation has been performed successfully with ANA's model.

The maximum peak pressure in the reactor coolant system is 19.0 MPa. This is well below the ASME Boiler and Pressure Vessel Code Level for BOL conditions (service limit stress criterion of 3200 psig. (22.16MPa).) /16/.

The maximum peak pressure in the reactor coolant system depends significantly on the kinetic parameters.

After the loss of feedwater, the pipe between the feedwater isolation valve and the header of auxiliary and main feedwater, works as a water tank. This stored water mass helps to cool the primary and produces the isolation of the steam generator as well as the start of the High Pressure Injection System.

Complete models of systems like main and auxiliary feedwater, (usually simulated as boundary conditions) help to find out unsuspected phenomena like the previously mentioned.

The results obtained in the study of this transient improve the knowledge of plant dynamics.

The model of Ascó using RELAP5/MOD3.2 is a valuable tool to analyze plant transients and to provide engineering support to plant operation.

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10. ANNEX 1

DIAGRAM 1. Ascó NPP Nodalization Diagram

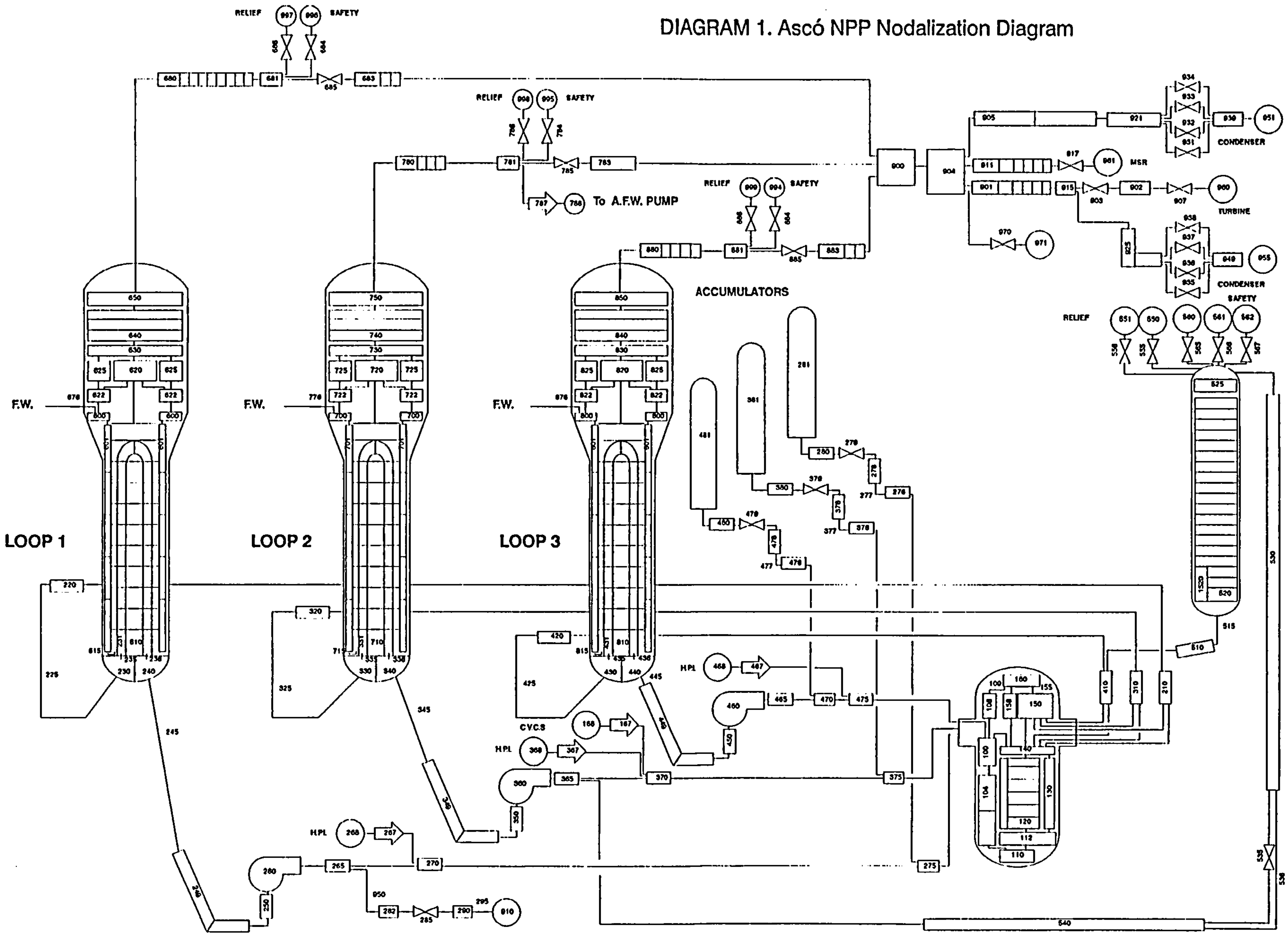
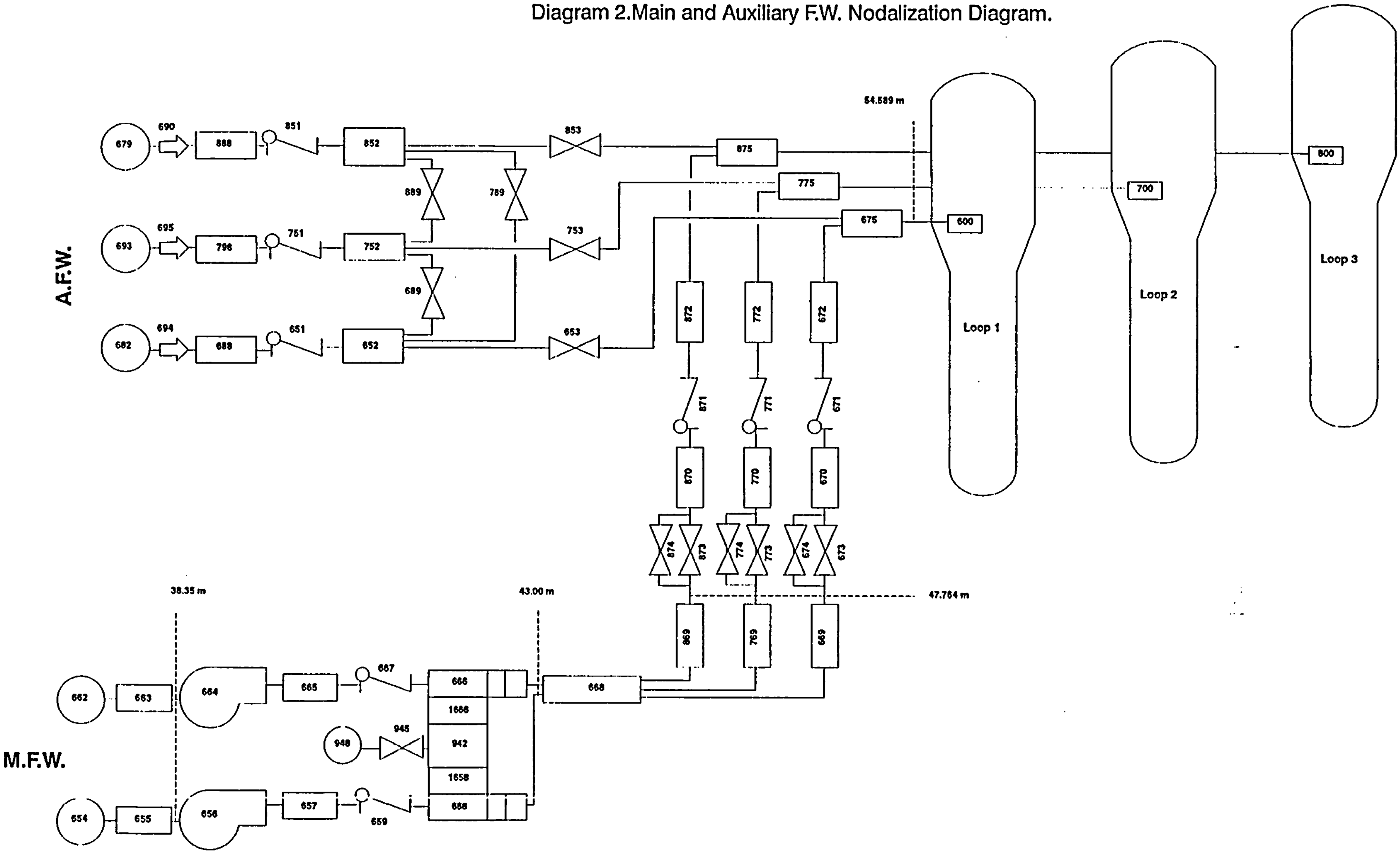


Diagram 2.Main and Auxiliary F.W. Nodalization Diagram.



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10. SUPPLEMENTARY NOTES

J. Uhle, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The Asociacion Nuclear Asco (ANA) has prepared a model of Asco NPP using RELAP5/MOD3.2. This model, which includes thermal-hydraulics, kinetics and protection and control systems, has been qualified in previous calculations of several actual plant transients. Asco NPP is a two unit station of three loop Pressurized Water Reactor (PWR) of Westinghouse design operated by ANA. ANA is a Spanish utility that contributes to the Code Application and Maintenance Project (CAMP) as a member of UNIDAD ELECTRIC A S.A. (UNESA). This report summarizes the results obtained with Asco NPP model for a loss of normal feedwater ATWS event and presents a sensivitity analysis to kinetic parameters for the same transient. The phenomenology prediction has been useful from the operation and safety point of view.

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ATWS

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