Regional Workshop on Development and Validation of Emergency Operating Procedures/Accident management Guidelines (EOP/AMG) for Effective Prevention/Mitigation of Severe Core Damage

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ANALYSIS OF EMERGENCY OPERATING PROCEDURES EFFECTIVENESS FOR CORE DAMAGE PREVENTION USING COMPUTER CODE RELAP FOR NUCLEAR POWER PLANTS WITH VVER-1000/B-320 IN REFERENCE TO PRIMARY TO SECONDARY CIRCUIT LEAK WITH EXTERNAL POWER LOSS AND BRU-A STUCK OPEN FAILURE

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INTRODUCTION

Kiev Research and Design Institute was created in 1944. During more than 50 years of it's existance many power plants have been designed.

Our institute has thermal hydraulic analysis laboratory on the base of an NPP department. This lab is equipped with modern hard and software for In-Depth Safety Assessment.

Our laboratory develops emergency operating procedures for beyond design basis accidents and performs corresponding safety analyses using different computer codes.

My report presents analysis of developed emergency operating procedures effectiveness for possible accident on nuclear power plant with VVER-1000 reactor type.

Accident initiating event is the primary to secondary circuit leak caused by steam generator primary cover lift-up. In according to conservative assumptions the following additional failures were considered:

- dump valve BRU-A stuck open failure;
- loss of external power.

The results of our work are represented as a comparative analysis of two possible ways of accident evolution:

- according to functioning automatic safety systems responses;
- according to accident management based on developed emergency operating procedures with operator intervention.

1 ACCIDENT DESCRIPTION

Steam generator primary cover lift-up may be caused by the break of corroding cover brads. This process can be provoked by violated chemical composition of feedwater.

Primary to secondary leak results to scram, ECCS initiation and steam generators water level rising. Turbine generators stop valves are tripped. Disbalance between generated power and electric load can lead to transient processes in electric chains. Therefore all turbine generators are tripped. This event can provoke loss of external power. Condensate and feedwater system are not available. Core cooling by means of secondary circuit (BRU-K and bypass condenser) is impossible.

Secondary pressure rising leads to dump valve BRU-A opening. Steam-water mixture is created in affected steam generator. Secondary pressure decreases and BRU-A of ruptured steam generator sticks in open position.

Primary coolant and ECCS water is lost through the break. The leak can not be compensated. Radioactivity release to the environment is considerable and core damage is possible.

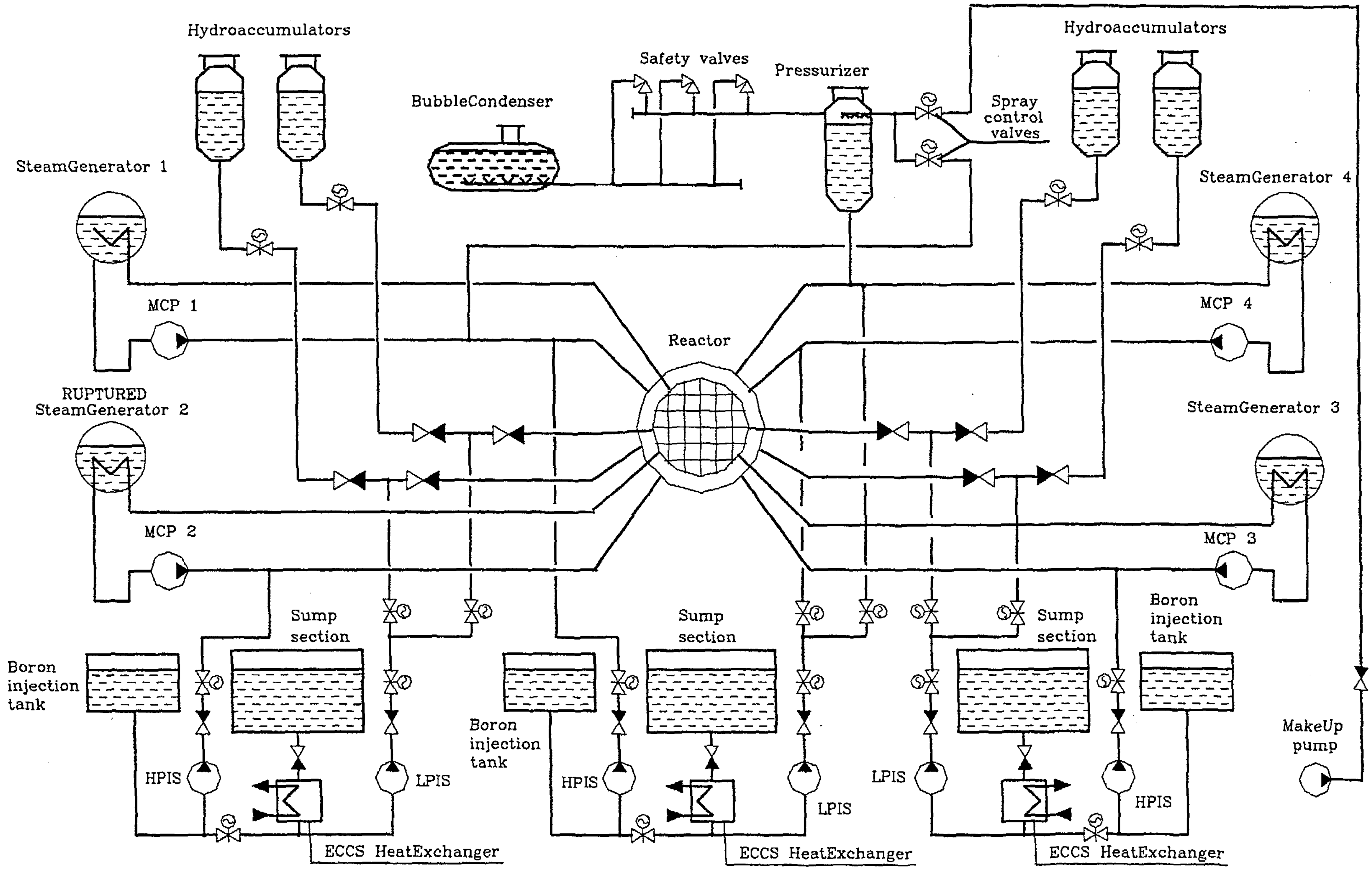


Fig.1 Primary circuit

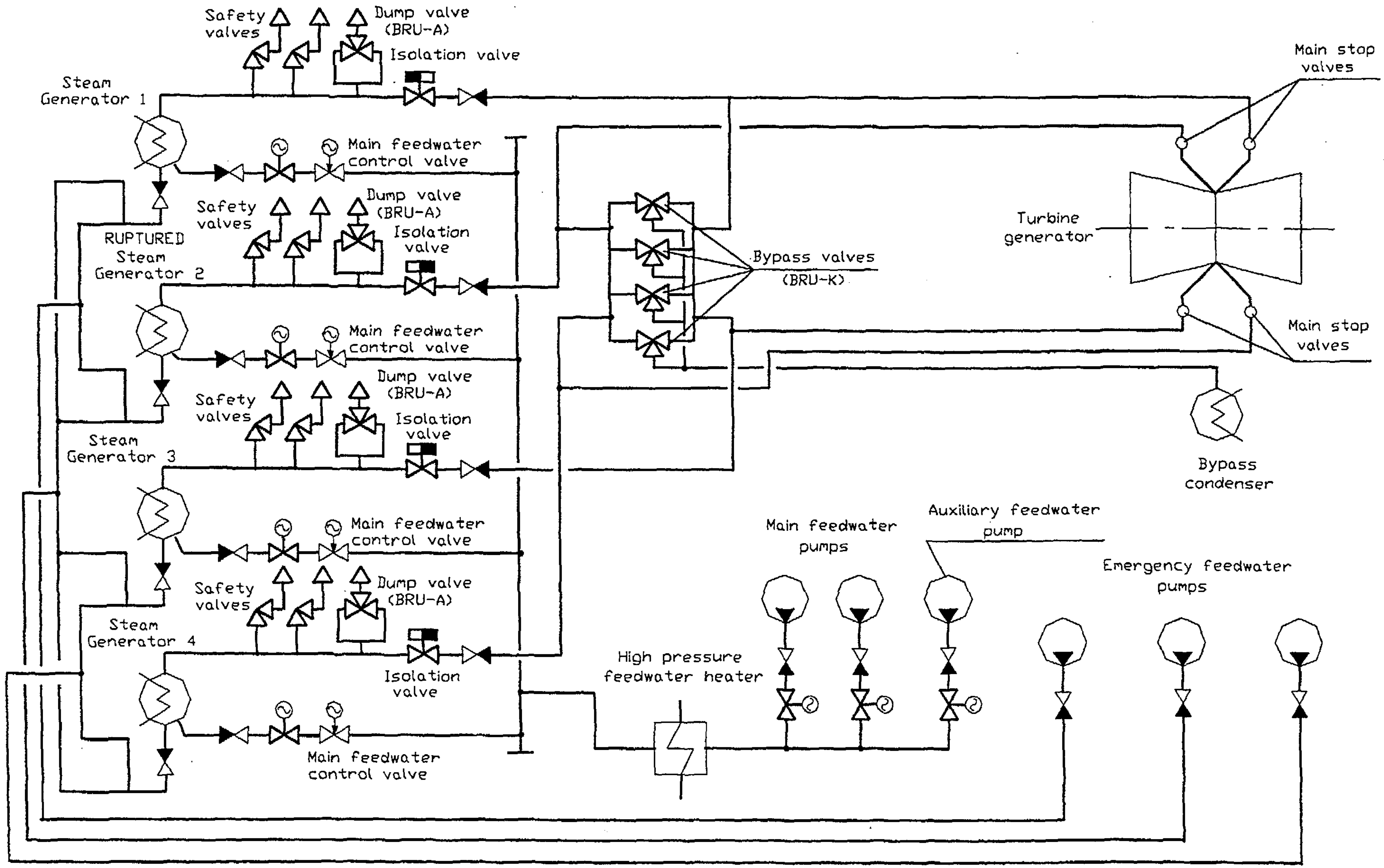


Fig.2 Secondary circuit

2 RELAP5 FEATURES

RELAP5 is a pressurised reactor (PWR) system transient analysis code that can be used for simulation of a wide variety of PWR system transients of interest in light water reactor (LWR) safety. The primary system, secondary system, feed water train, system controls, and core neutronics can be simulated. The code models have been designed to permit simulation of postulated accidents ranging from large break loss-of-coolant accidents to accidents involving the plant controls and fuel system. Transient conditions can be modelled up to the point of fuel damage.

An additional goal of the project has been to provide a more comprehensive and generic modelling of the complete nuclear steam supply system including turbines, generators, condensers, feed systems, and plant controls.

RELAP5 is a generic transient analysis code for thermal-hydraulic systems using a fluid that may be a mixture of steam and water, noncondensible specie, and a non-volatile solute. The fluid and energy flow paths are approximated by one-dimensional stream tube and conduction models. The code contains system component models peculiar to pressurized water reactors. In particular a point neutronics model, pumps, turbines, generator, valves, separator, and controls are included. The code also contains a jet pump component and has been used for modelling boiling water reactor systems.

The PWR applications for which the code is intended include large and small break loss of coolant accidents, operational transients such as anticipated transients with scram, loss of feed, loss of off-site power, loss of flow, and over cooling transients. The system behaviour can be simulated up to the point of fuel damage. neither fuel cladding deformation nor metal-water reaction are modelled.

RELAP5 can also be used for analysis of the transient behaviour of piping systems containing steam/water such as for estimating hydraulic loads on relief valve discharge lines.

The hydrodynamic model and the associated numerical scheme are based on the use of fluid control volumes and junctions to represent trie spatial character of the flow. The control volumes can be viewed as stream tubes having inlet and outlet junctions. The control volume has a direction associated with it that is positive from the inlet to the outlet. The fluid scalar properties such as pressure, energy, density, and void fraction are represented by the average fluid conditions and are viewed as being located at the control volume centre. The fluid vector properties, i. e,, velocities are located at the junctions and are associated with mass and energy flow between control volumes. Control volumes are connected in series using junctions to represent a flow path.

Heat flow paths are also modelled in a one-dimensional sense using a staggered mesh to calculate temperatures and heat flux vectors. The heat conductors can be connected to hydrodynamic volumes to simulate a heat flow path normal to the fluid flow path. The heat conductor or heat structure is thermally connected to the hydrodynamic volume through a heat flux that is calculated using a boiling heat transfer formulation. Electrical or nuclear heating of the heat structure can also be modelled as either a surface heat flux or as a volumetric heat source. The heat structures are used to simulate pipe walls, heater elements, nuclear fuel pins, and heat exchanger surfaces.

The control system model provides a way for simulating any lumped process such as controls or instrumentation in which the process can be defined in terms of system variables through algebraic or logical operations. The control system advancement occurs after the hydrodynamic advancement and uses the same time step as the hydrodynamics so that new time thermal and hydrodynamic information is used in the control model advancement. However, the control variables are not fed back to the thermal and hydrodynamic model until the succeeding time step, i.e., they are explicitly coupled to the hydrodynamic model.

The reactor kinetics model is also advanced in a serially implicit manner after the control system advancement. The kinetics model consists of a system of ordinary differential equations that are integrated using a modified Runge-Kutta technique. The integration time step is regulated by a truncation error control and may be less than the hydrodynamic time step; however, the thermal and fluid boundary conditions are held fixed over each hydrodynamic time interval.

A system code such as RELAP5/MOD3 contains numerous approximations to the behaviour of a real, continuous system. These approximations are necessitated by the finite storage capability of computers, the need to obtain a calculated result in a reasonable amount of computer time, and in many cases because of limited knowledge about the physical behaviour of the components and processes that are modelled.

3 NPP MODEL NODALIZATION FEATURES

Designed model for calculation was based on standartized performances of NPP with VVER-1000 reactor type.

Primary circuit model includs the following elements:

- reactor;
- steamgenerators
- four main circulation pipelines;
- reactor coolant pumps.

3.1 Reactor

The reactor was nodalized to following parts:

- downcomer;
- lower plenum;
- core
  - hot assembly;
  - average assemblies
  - bypass flow
- upper plenum
- top volume.

Downcomer is divided into 8 vertical volumes. Five upper ones model mixture of cold legs coolant and hydroaccumulators water.

Core support columns were considered in lower plenum model.

Core model considers heat transfer between assemblies.

Reactor kinetics represents the point model.

Upper plenum is divided into five parallel channels, that models mixture process. Model also considers unsimmetry of hydroaccumulators inlet nozzles geometry.

3.2 Steam generator

Steam generator model was divided to two independent hydraulic models for primary and secondary circuits.

Primary to secondary leak is modelled as a break valve. Break valve model represents the motor valve component. It's full open area equals 78 cm^2. That's equivalent of 100 mm hydraulic diameter. Leak model is designed as the primary coolant outflow into the feedwater downflow region.

Steamgenerator model divided into the following parts:

- below tube bundles;
- tube bundles;
- below perforated sheet;
- separator volume;
- steam header.

Secondary circulation is modelled as two vertical channels with horizontal cross-connections.

Primary – secondary heat structures model considers heat loss to the environment.

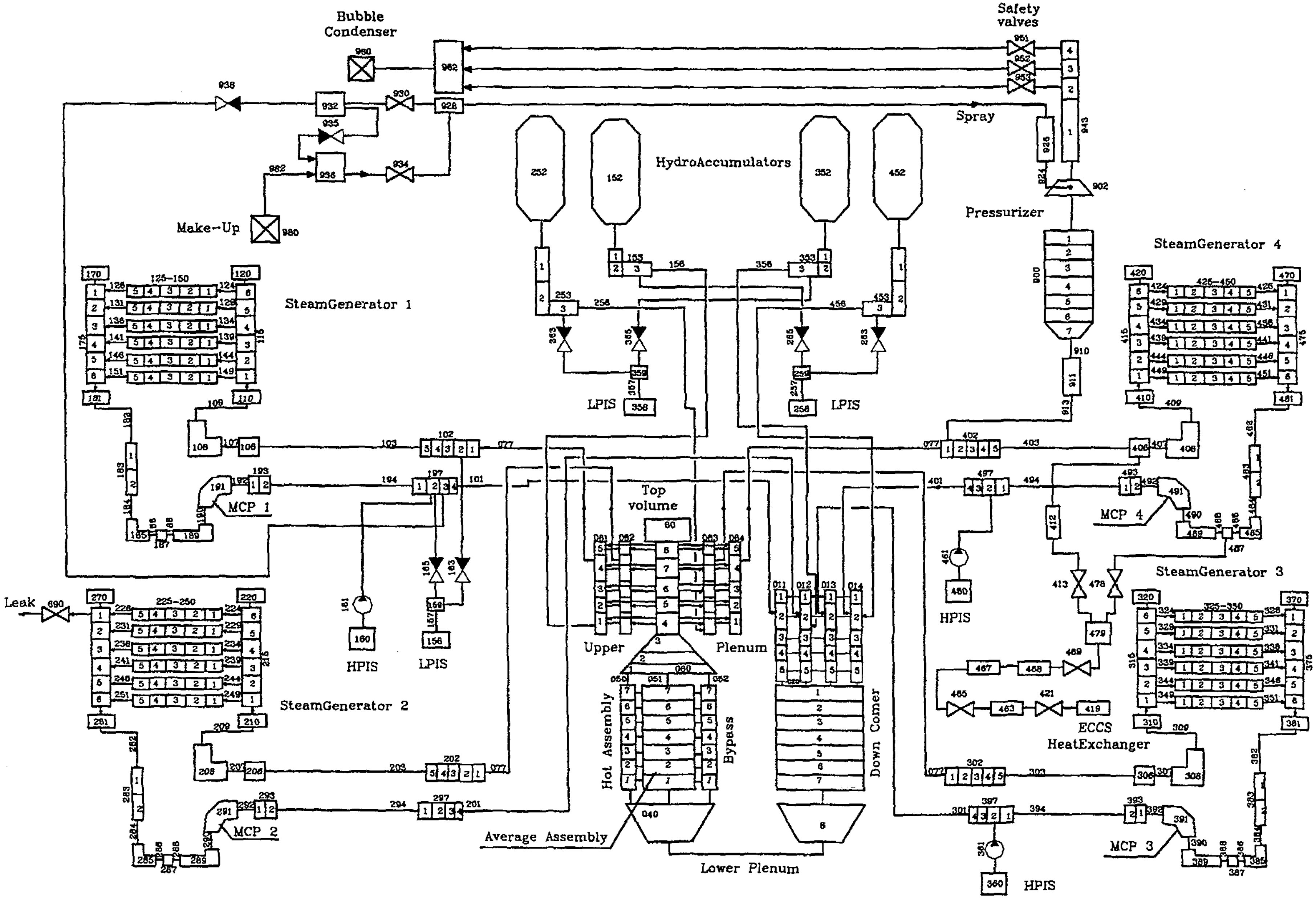


Fig. 3 Primary circuit nodalization scheme

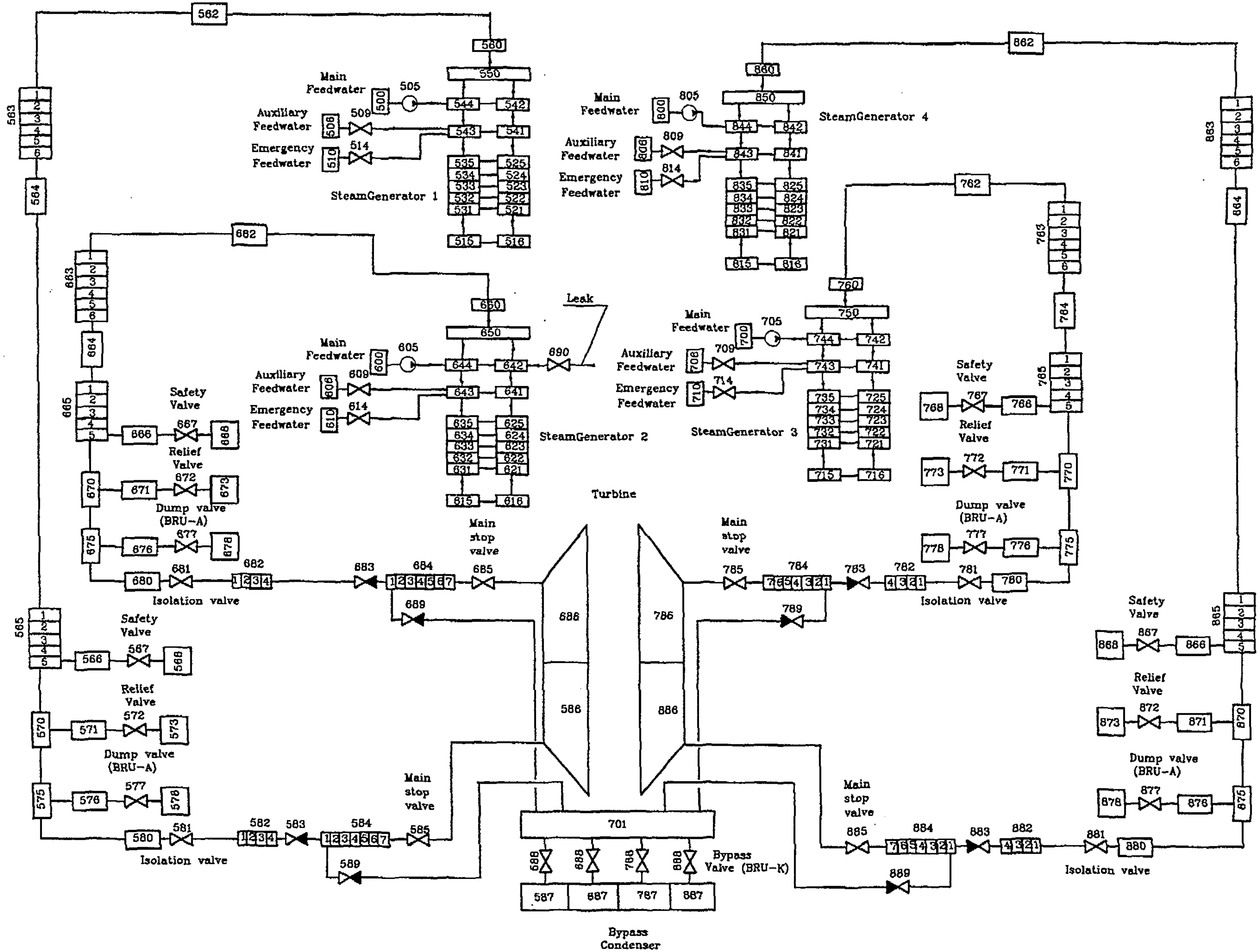


Fig. 4 Secondary circuit nodalization scheme

3.3 Main coolant pump

MCP *model contains:*

- hydrodynamic model;
- pump-water interaction model;
- torque model.

3.4 Main circulation pipelines

Four coolant circulation loops are divided into elementary volumes, considering tees and flow direction changes.

3.5 Auxiliary systems

Primary circuit includes the following auxiliary systems:

- pressure control system;
- hydroaccumulators;
- emergency core cooling system;
- residual heat removal system;
- reactor control and protection system.

3.6 Secondary circuit

Secondary system model contains:

- feedwater system
    - main;
    - auxiliary;
    - emergency;
- main steam system;
- steam dump (BRU-A) system;
- steam bypass (BRU-K) system;
- pressure control system;
- turbine generators.

Feedwater control valves are considered in feedwater system model. Feedwater pumps are represented as a simplified model in order to consider the correlation between flowrate and feedwater header pressure.

Turbine generator model represents boundary conditions and considers steam output and mainsteam header pressure.

Secondary pressure control system contains safety valve, relief valve and it's pipelines.

4 CONSERVATIVE ASSUMPTIONS USED AT ACCIDENT MODELLING

4.1 Signals for affected steamgenerator definition

Two signals are available to the operator define ruptured steamgenerator:

- water level is higher than 3m;
- steam activity rising.

These signals have the shortest time of initiation.
Used computer code does not allow to control steam activity. Therefore we accept this signal has already activated at the moment of steamgenerator water level rising.

4.2 Loss of external power

Loss of external power really happens approximately at the thirtieth second of accident. We assume it happens just after scram activating.

4.3 Turbine stop valves closing

Turbine stop valves closing is accepted at the moment of loss of external power.

4 ACCIDENT EVOLUTION ACCORDING TO FUNCTIONING SAFETY SYSTEMS RESPONSES

Accident initiating event is a steam generator primary cover lift-up.

In accordance with functioning safety systems responses operator is not allowed to interfere in it's action.

In this case we get following accident evolution results:

Table 1

Time, sec.	Accident evolution
0	Steam generator primary cover lift-up. Primary pressure decreasing. Scram initiating when primary pressure equals 14,6 MPa.
5	Loss of external power. Bypass valve BRU-K is disabled to open.
15	Affected SG level above 3m. Turbine stop valves closure.
15-30	Main steam pressure equals 7,14 Mpa. Dump valves BRU-A is opening. MCP are tripped.
20	Turbine generators are tripped.
30	Emergency power system activating. HPIS pumps switch on at recirculation mode. Primary pressure decreasing. Saturation temperature margin of coolant is lower than 10 °C. LPIS pumps are activated at recirculation mode.
45	Primary pressure equals 10,5 Mpa. HPIS pumps switch over to primary circuit.
45-70	SG pressure equals 6,7 Mpa. Dump valves (BRU-A) of non-affected SG are closing. Separator of ruptured SG is flooded. Steam-water mixture is going to steam line. Dump valve (BRU-A) sticks in open position.
440	Primary pressure equals 5,97 Mpa. Hydroaccumulators are activated.
1210	Primary pressure equals 2,37 Mpa. LPIS pumps switch over to primary circuit.

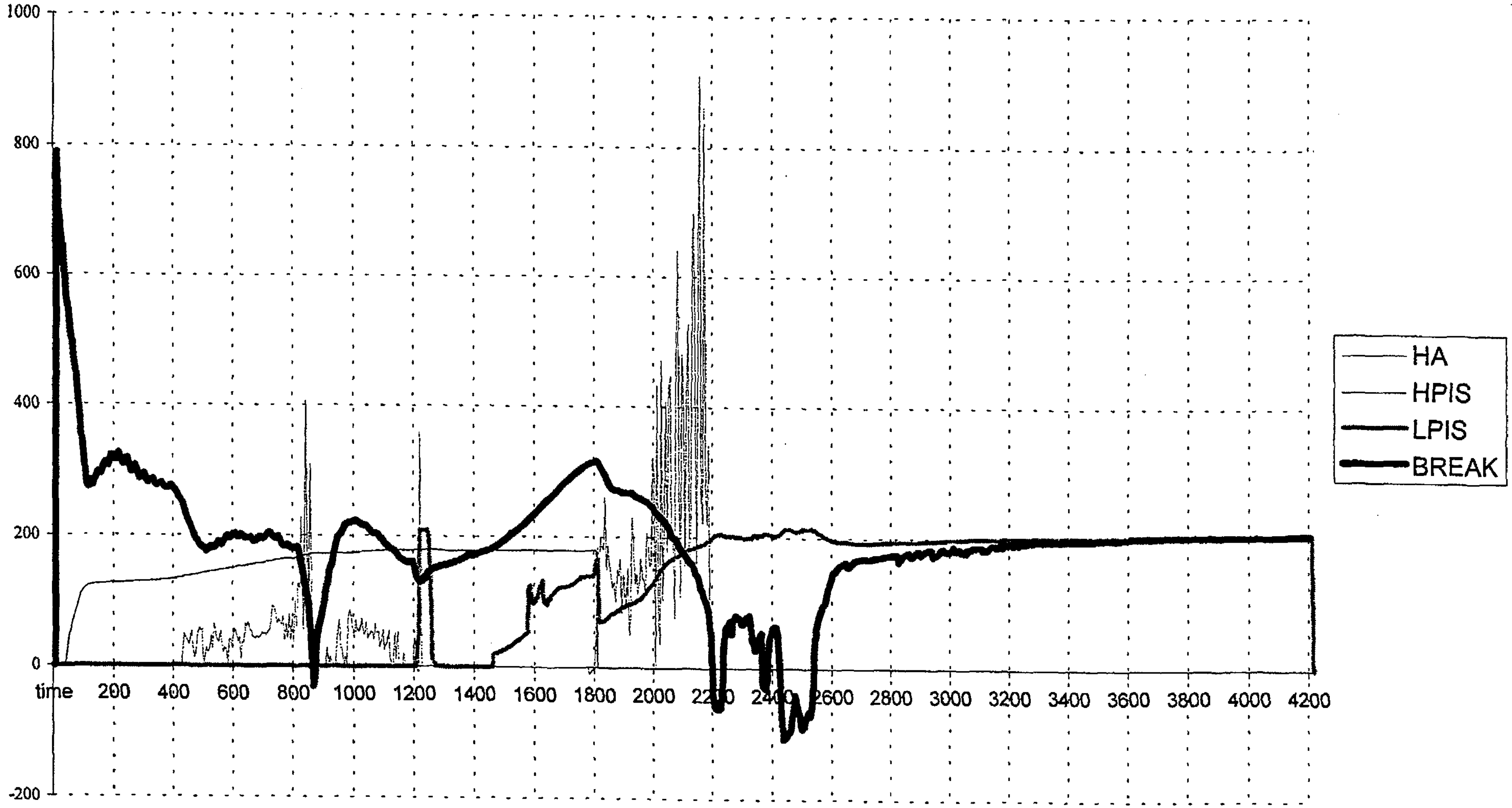
After 30 minutes from accident initiation operator actions are following:

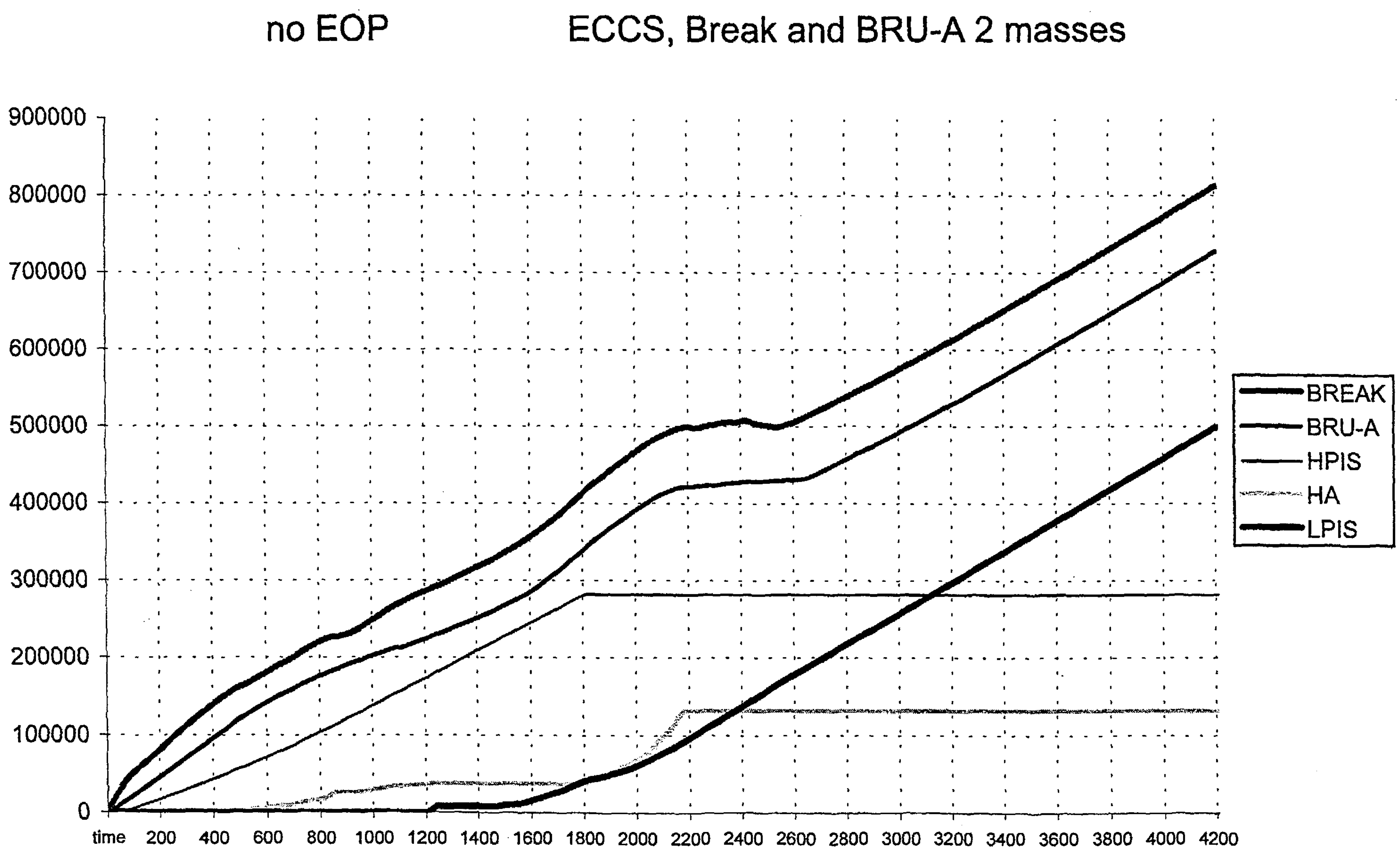
- HPIS switch off;
- LPIS pumps switch into recirculation mode except one acting;
- dump valves BRU-A switch over to 60 °C per hour cooldown mode.

Reactor is cooled by means of LPIS and hydroaccumulators. Heat is removed through the leak. Steamgenerators are full so emergency feedwater system is not activated. Non–affected SGs are isolated. Secondary temperature is higher than primary. Therefore core cooling by means of non-affected SGs is impossible.

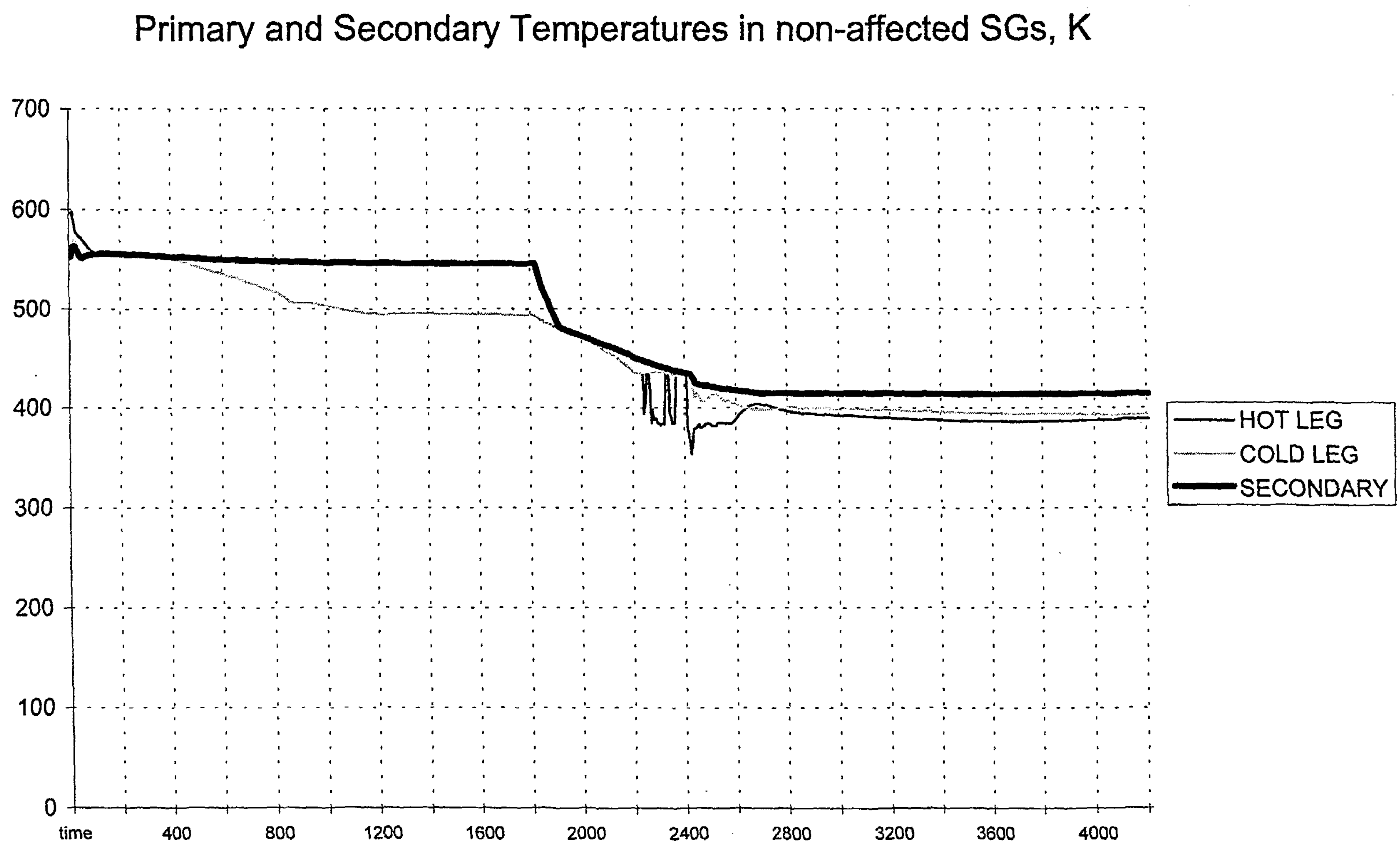
During one hour boron injection tanks, sump and hydroaccumulators are depleted. Core cooling can not be further assured. Reactor coolant level is getting lower than hot legs nozzles elevation. Steam is created in core. Steam thermal conductivity is lower than water one. Heat removal is very small. Fuel elements are dried out. It leads to excessive heat-up and severe core damage after half an hour.

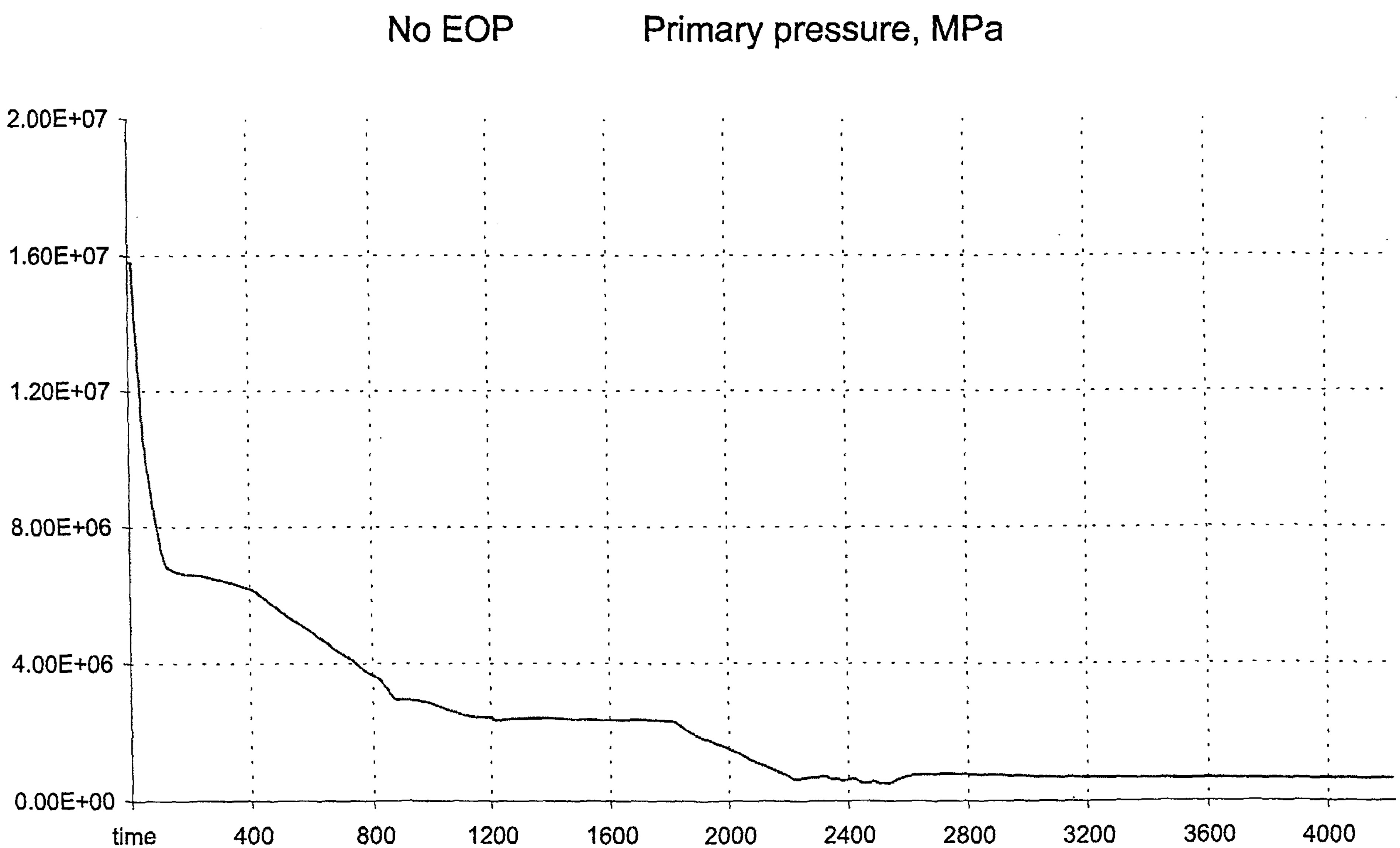
NO EOP Break and ECCS flowrates, kg/s



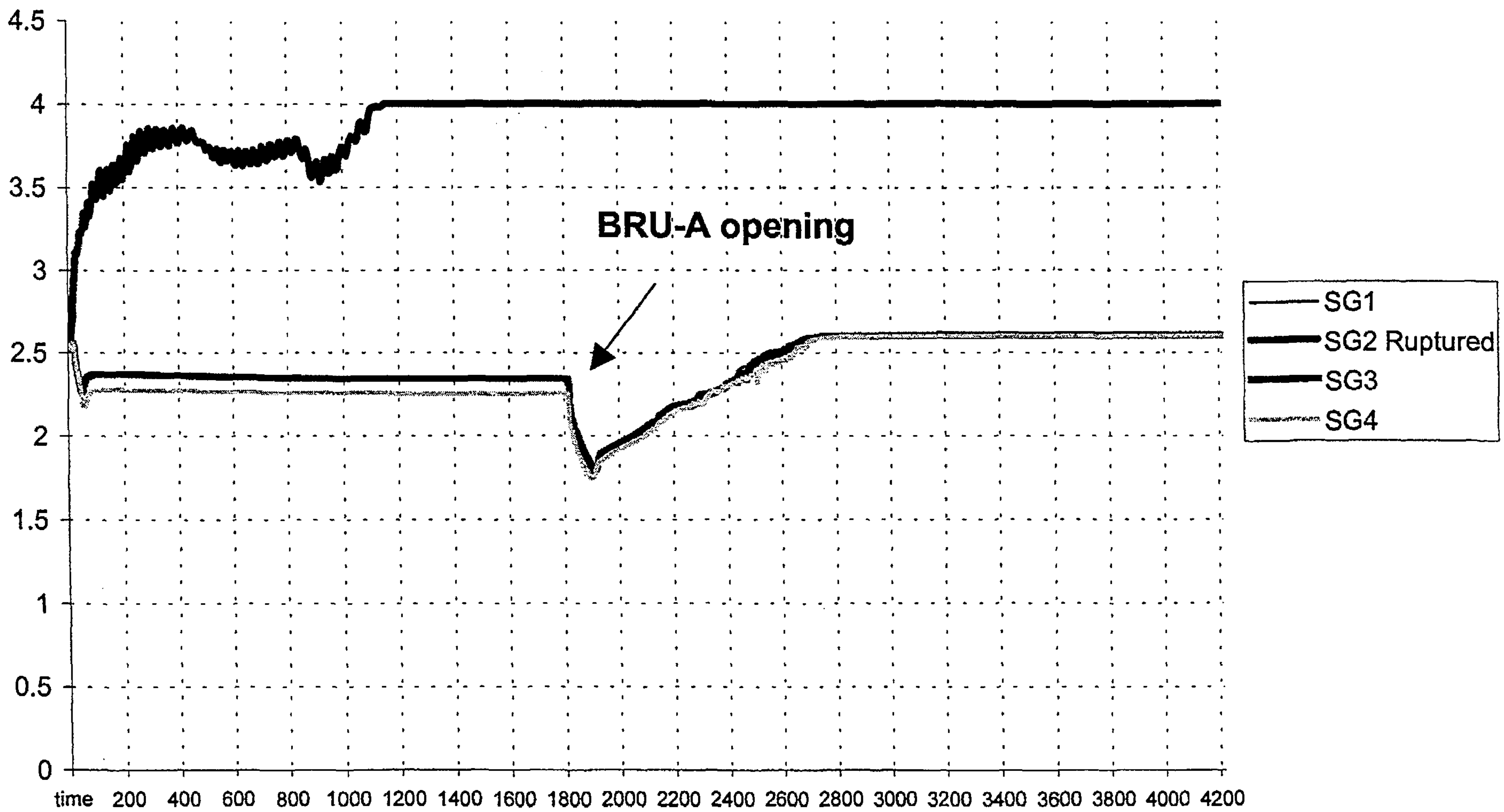


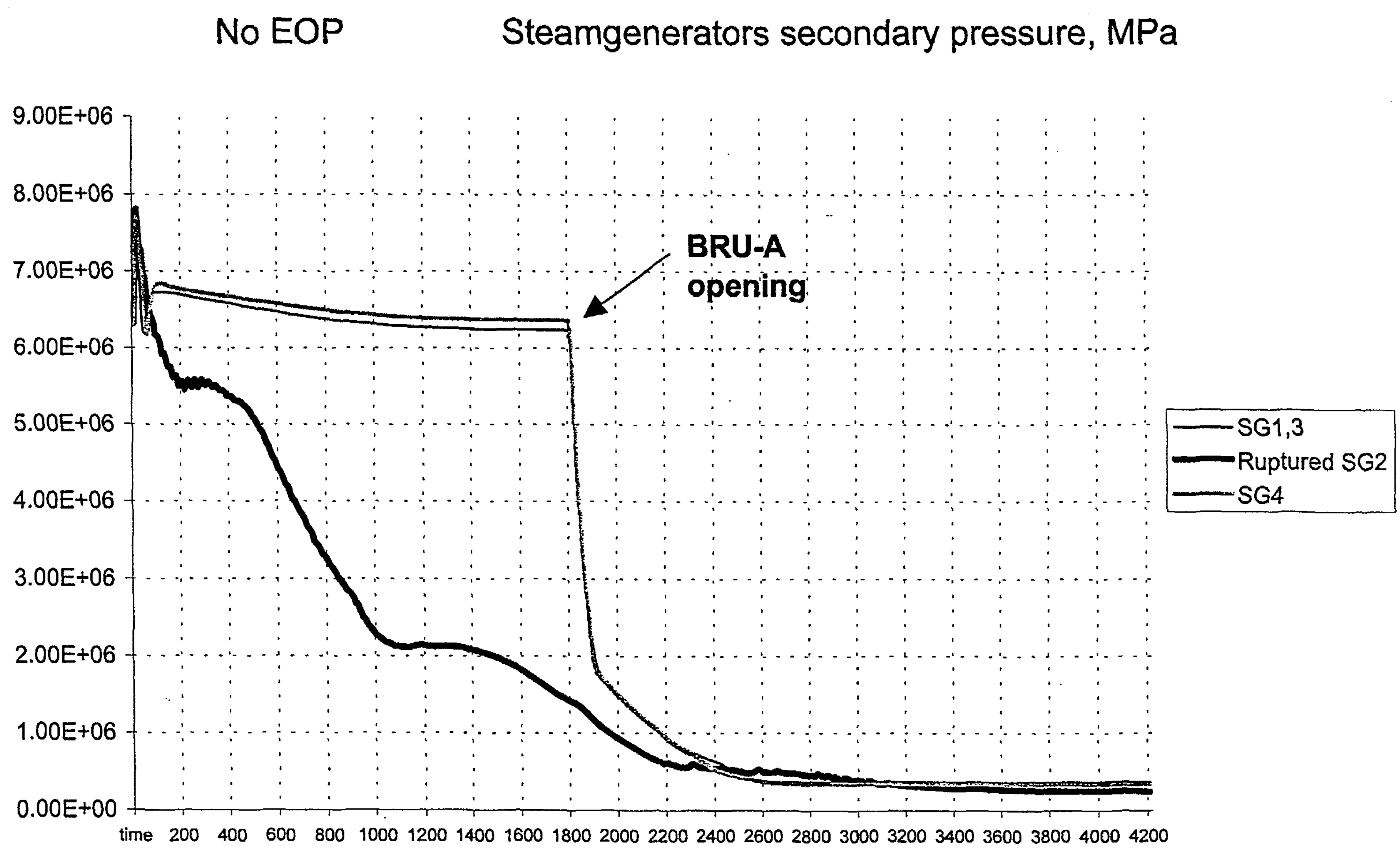
no EOP
ECCS, Break and BRU-A 2 masses
900000
800000
700000
600000
500000
400000
300000
200000
100000
0
time
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
2800
3000
3200
3400
3600
3800
4000
4200
BREAK
BRU-A
HPIS
HA
LPIS


Primary and Secondary Temperatures in non-affected SGs, K
700
600
500
400
300
200
100
0
time
400
800
1200
1600
2000
2400
2800
3200
3600
4000
HOT LEG
COLD LEG
SECONDARY

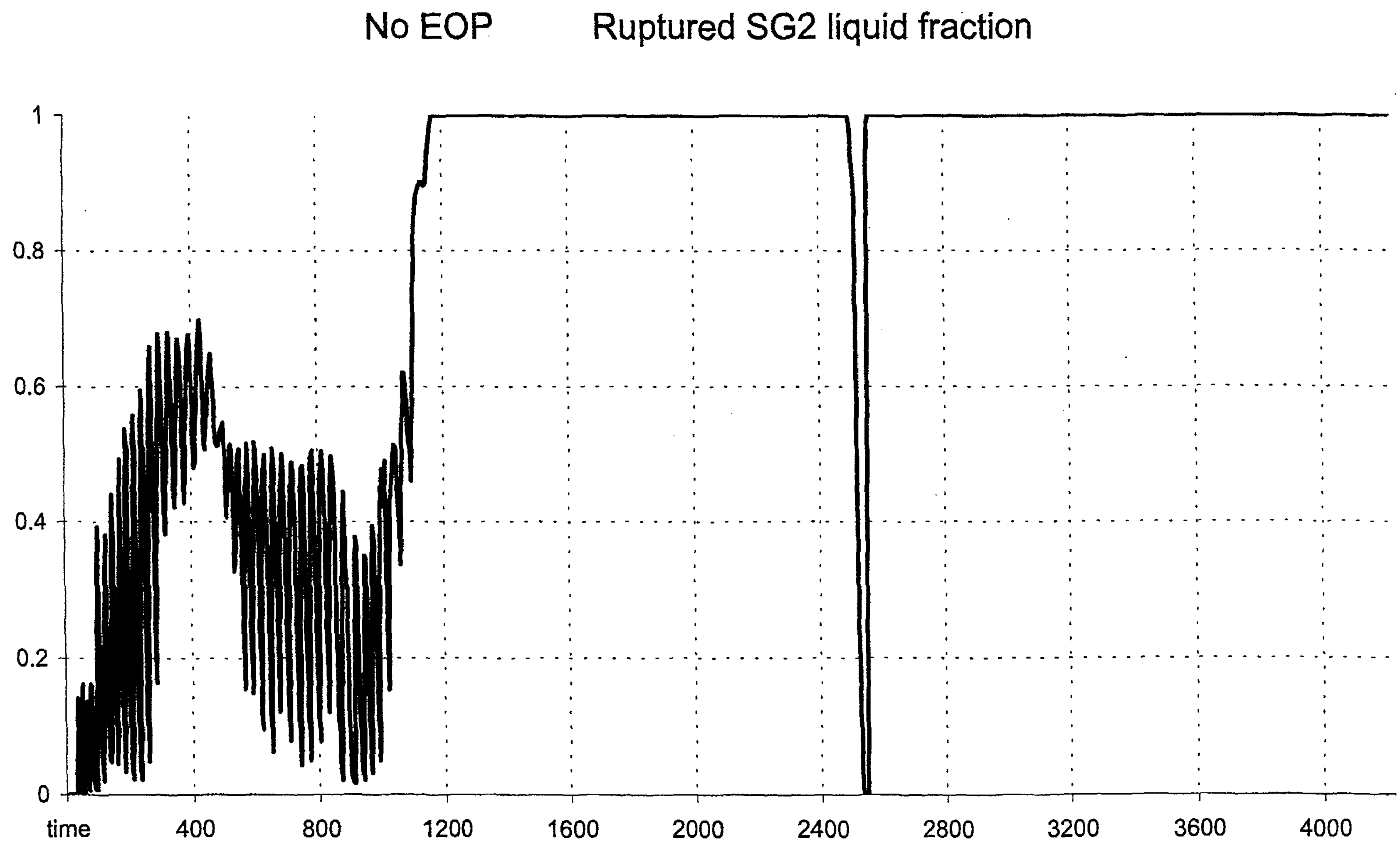

No EOP
Primary pressure, MPa
2.00E+07
1.60E+07
1.20E+07
8.00E+06
4.00E+06
0.00E+00
time
400
800
1200
1600
2000
2400
2800
3200
3600
4000

No EOP Steamgenerators level, m





No EOP
Steamgenerators secondary pressure, MPa
9.00E+06
8.00E+06
7.00E+06
6.00E+06
5.00E+06
4.00E+06
3.00E+06
2.00E+06
1.00E+06
0.00E+00
time 200 400 600 800 1000 1200 1400 1600 1800 2000 2200 2400 2600 2800 3000 3200 3400 3600 3800 4000 4200
BRU-A opening
SG1,3
Ruptured SG2
SG4


No EOP
Ruptured SG2 liquid fraction
1
0.8
0.6
0.4
0.2
0
time
400
800
1200
1600
2000
2400
2800
3200
3600
4000

5 ACCIDENT MANAGEMENT STRATEGY

Main goals of emergency operating procedures are:

- mitigation of radioactivity release to the environment;
- core damage prevention.

Accident management includes following necessary steps:

- immediate HPIS switch off;
- LPIS switch over to recirculation mode except one acting;
- SGs isolation valves closing to mitigate dynamic loads to steam pipe lines and radioactive contamination of secondary circuit equipment;
- dump valve BRU-A full opening to reduce secondary temperature and water level and emergency feedfwater pumps activating.
- closing of nitrogen injection line to pressurizer to reduce primary pressure and leak flowrate.

When primary pressure became lower than 15Mpa and primary circuit parameters were stabilized, following procedures have to be performed:

- one LPIS pump acting into reactor vessel;
- hydroaccumulators closing to reduce leak flowrate and primary pressure;
- normal cooldown pipeline connecting to primary circuit.

Normal cooldown pipeline consists of ECCS heat exchanger, sump and LPIS pump.

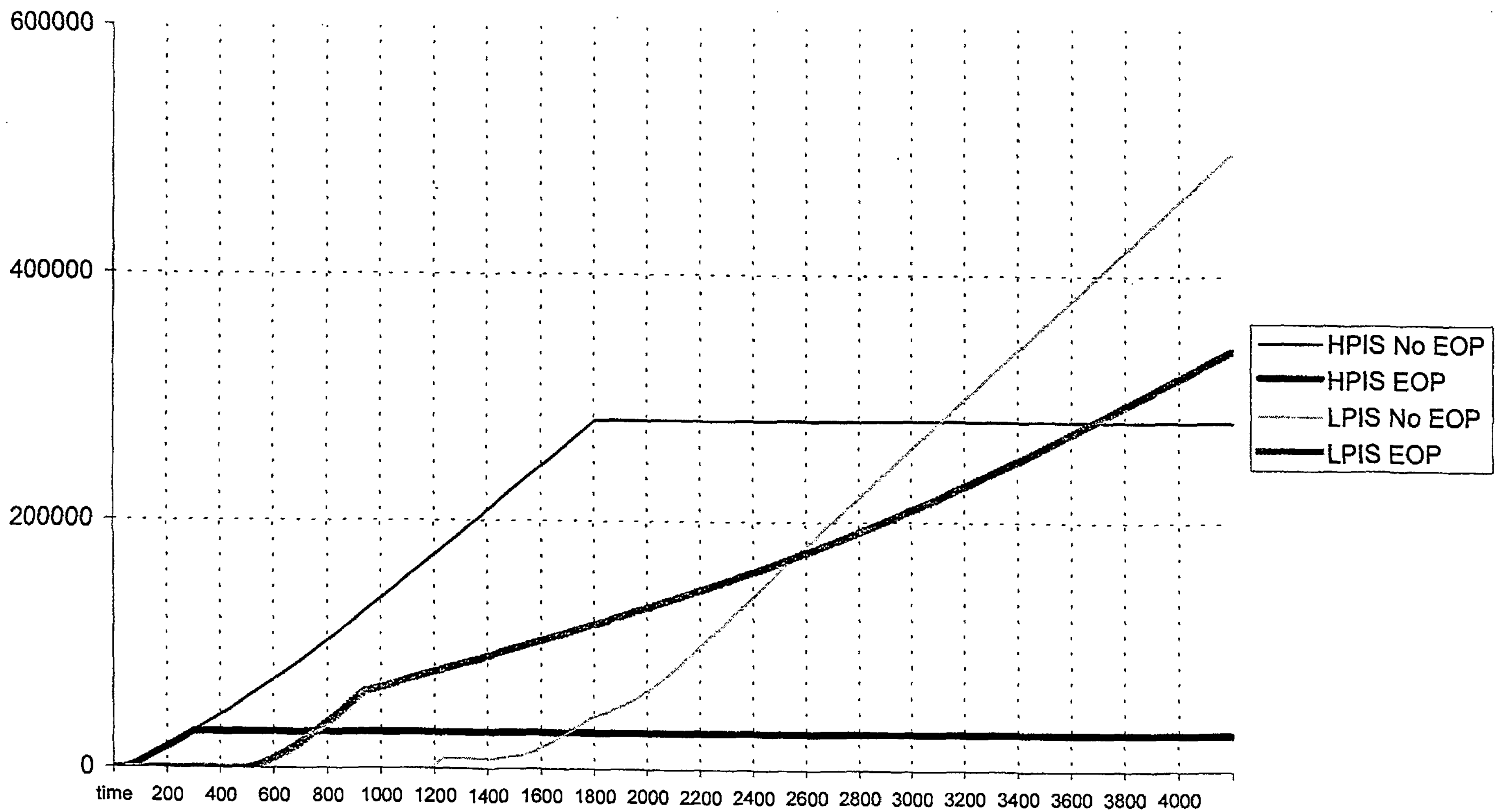
5 EMERGENCY OPERATING PROCEDURES EFFECTIVENESS

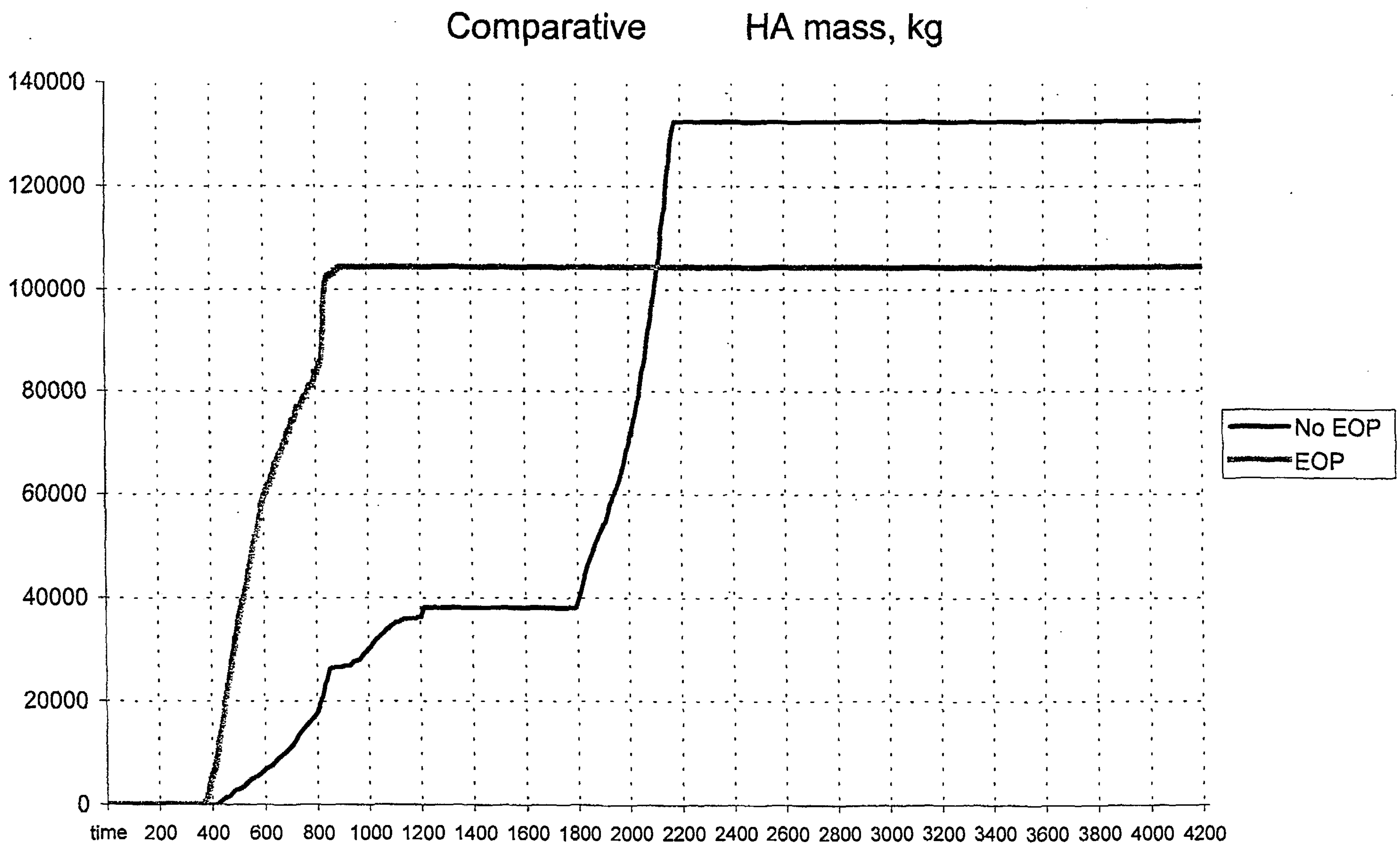
The results of emergency operating procedures application are following:

- the reduction of water loss into the break, including ECCS inventory and hydroaccumulators;
- mitigation of radioactivity release to the environment.

In spite of short-term reversible leak flowrate, primary coolant boron concentration does not fall lower than permissible value.

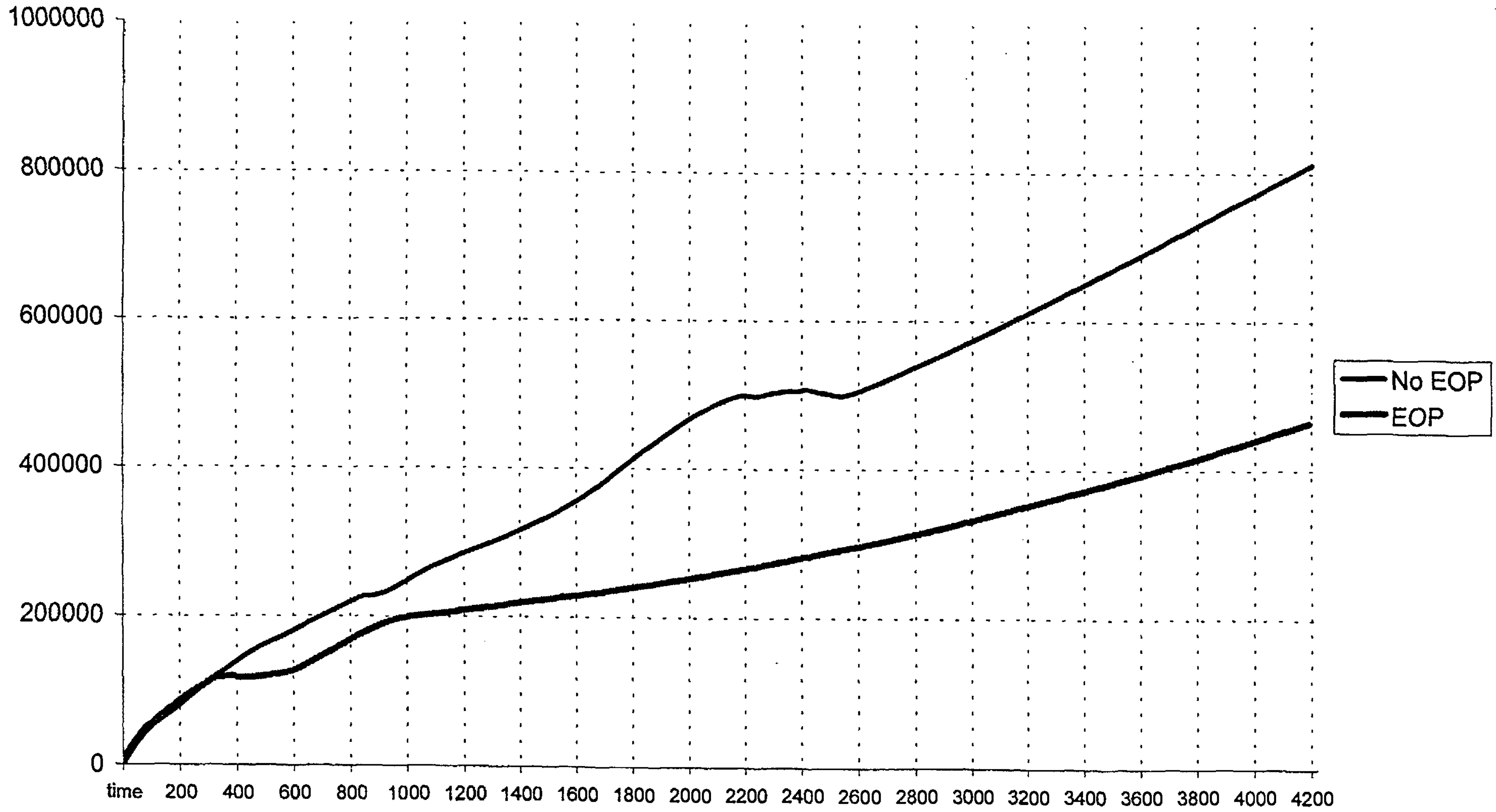
Comparative ECCS masses, kg

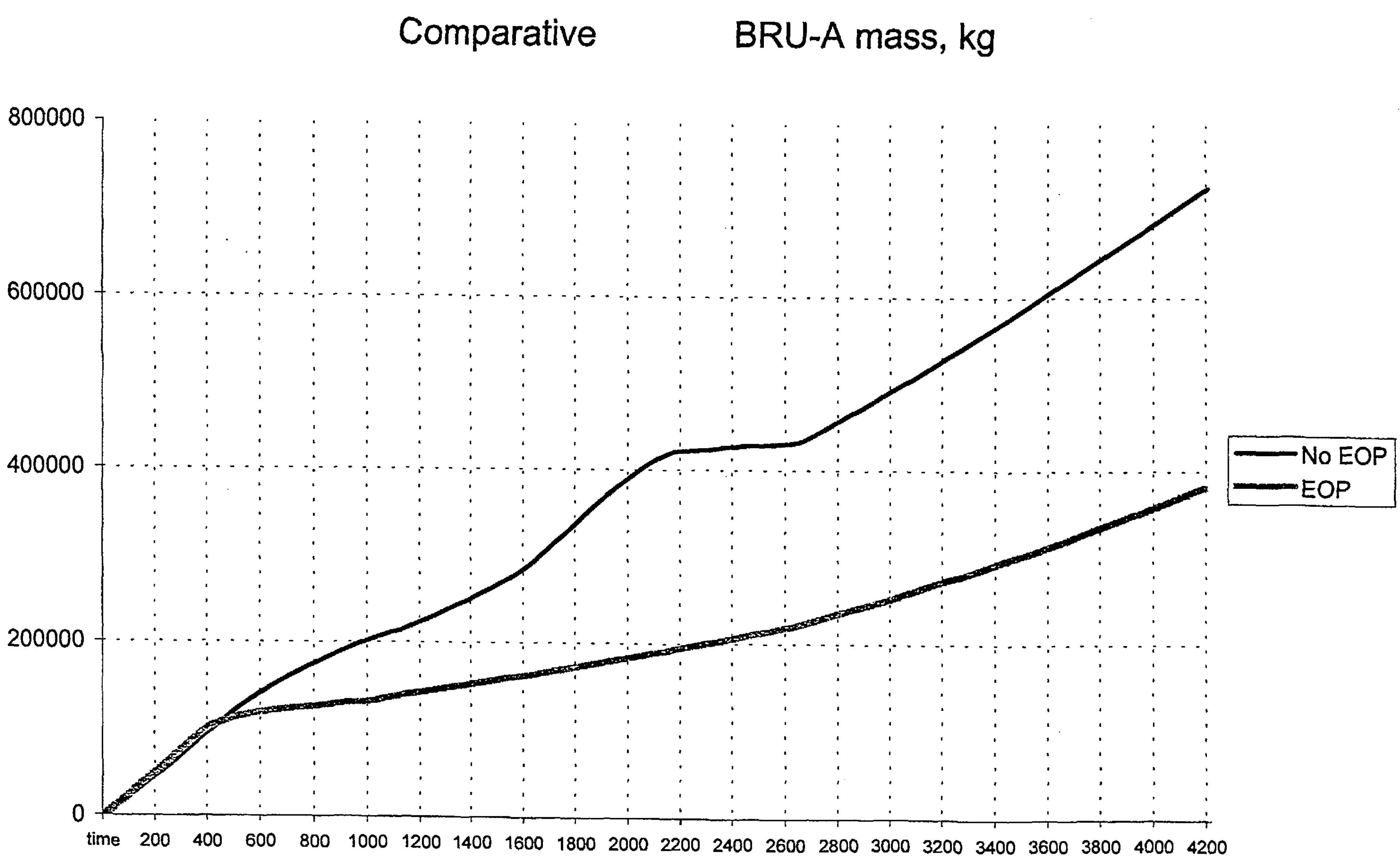




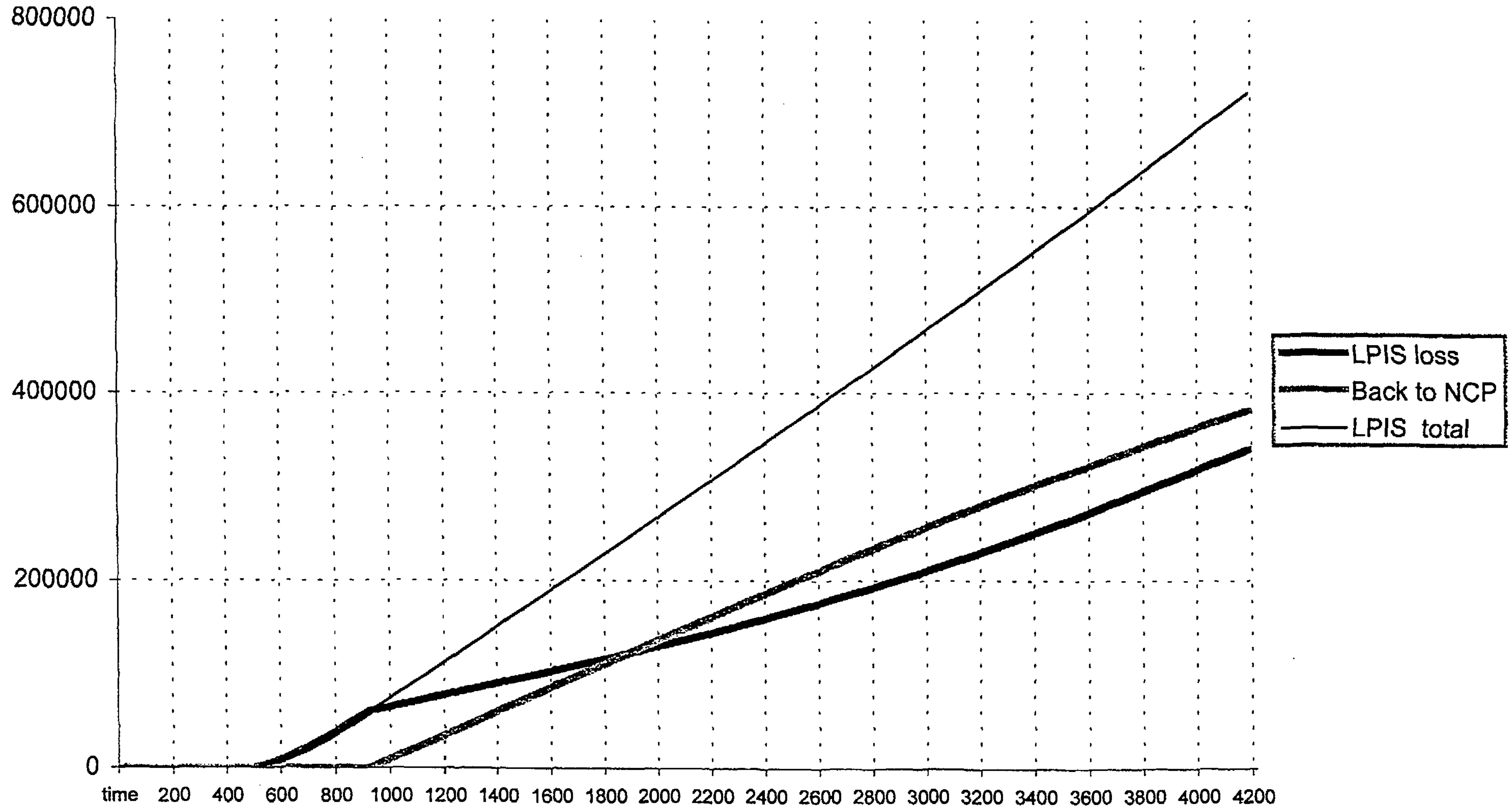
Comparative HA mass, kg
140000
120000
100000
80000
60000
40000
20000
0
time 200 400 600 800 1000 1200 1400 1600 1800 2000 2200 2400 2600 2800 3000 3200 3400 3600 3800 4000 4200
No EOP
EOP

Comparative Break masses, kg

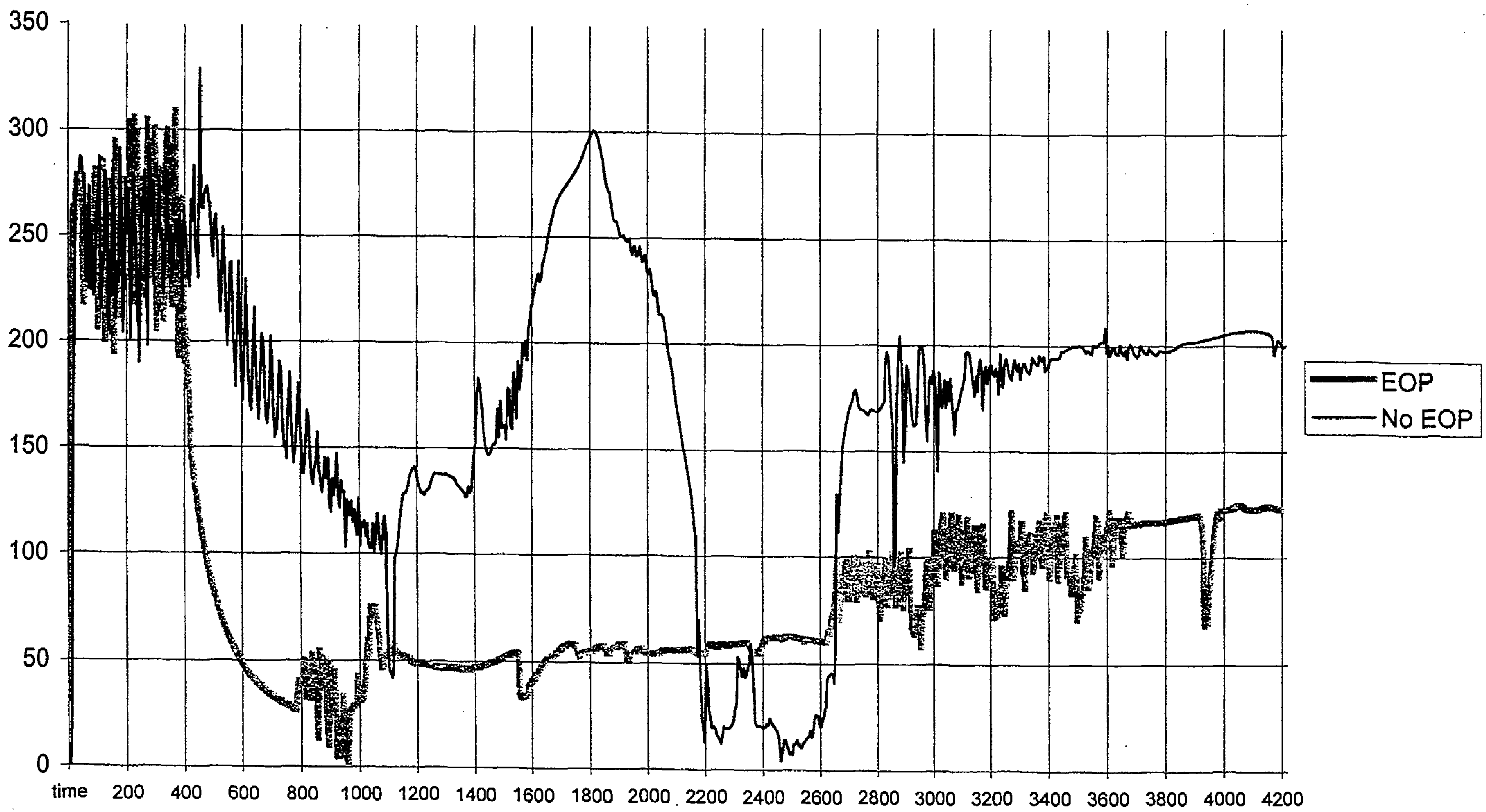


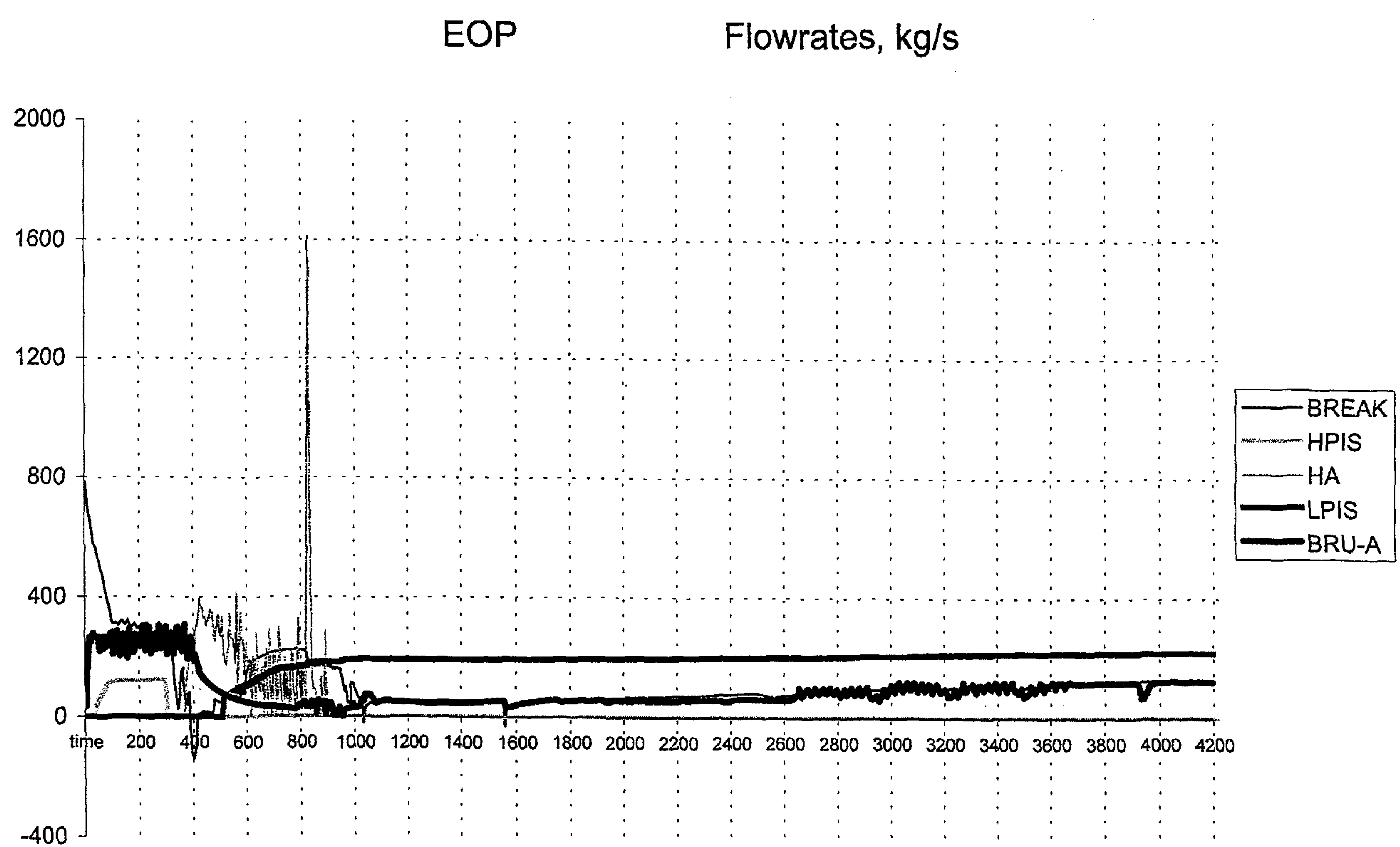

Comparative
BRU-A mass, kg
800000
600000
400000
200000
0
time
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
2800
3000
3200
3400
3600
3800
4000
4200
No EOP
EOP

EOP ECCS Balance, kg

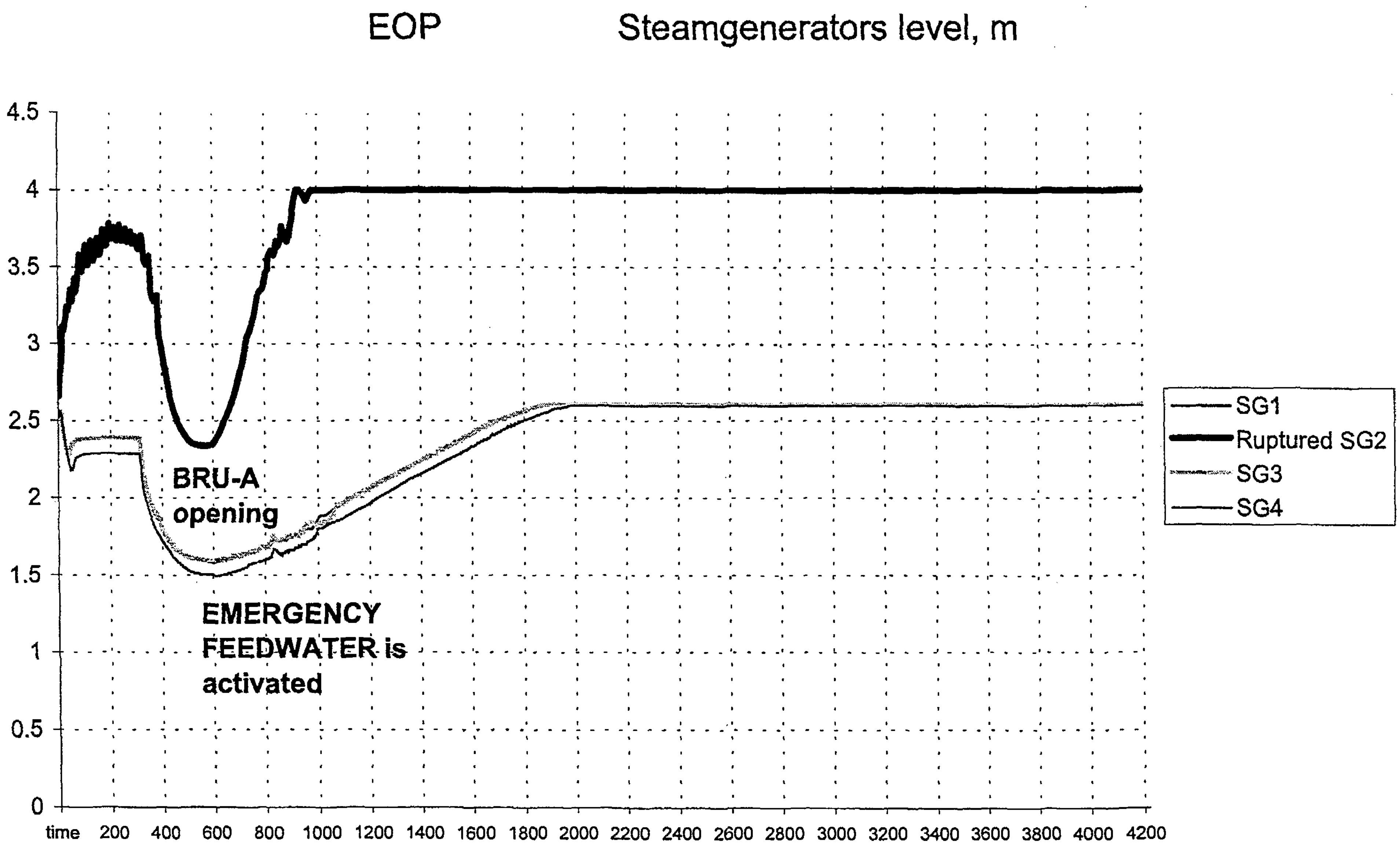


Comparative BRU-A massflow, kg/s

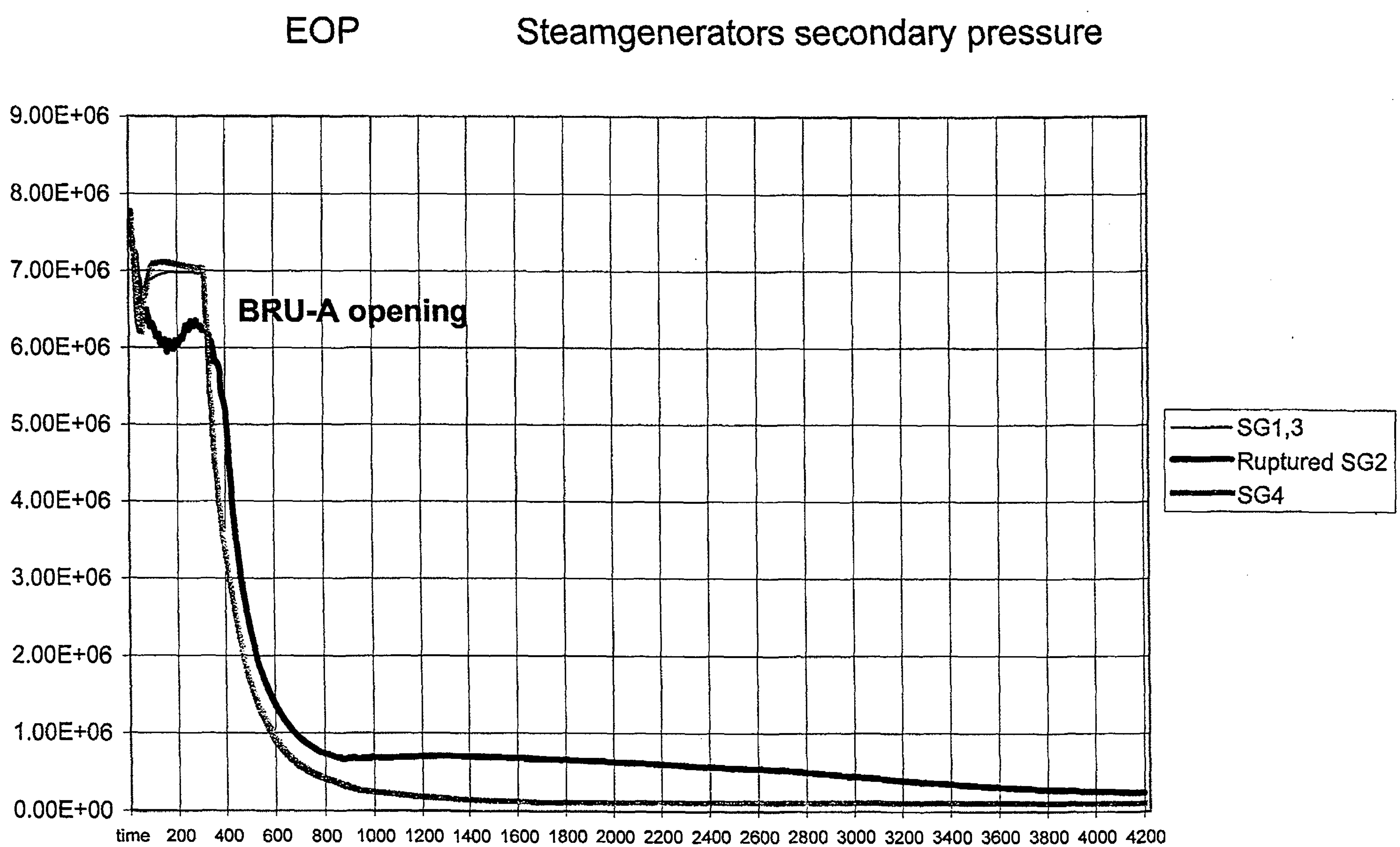




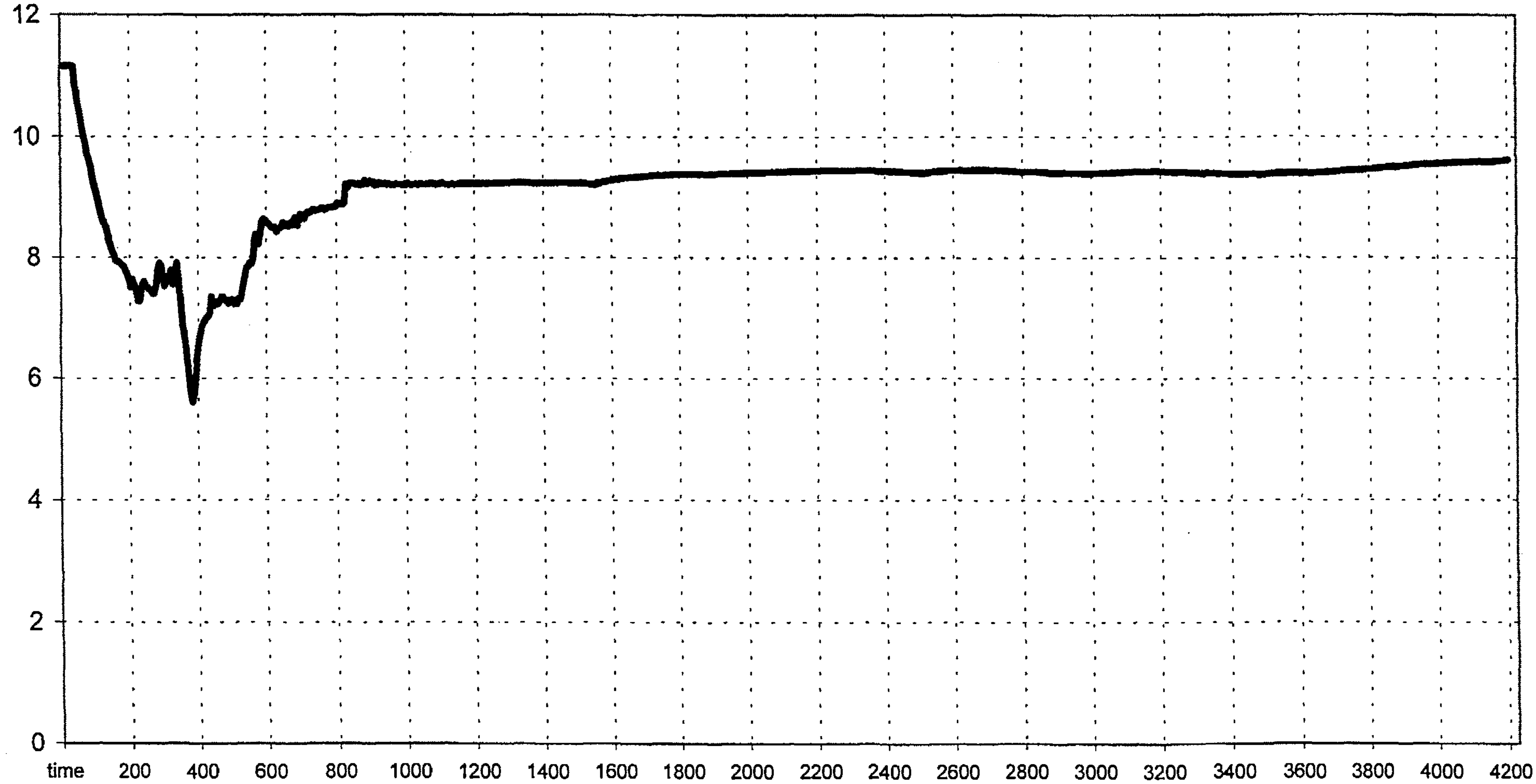
EOP
Flowrates, kg/s
2000
1600
1200
800
400
0
-400
time
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
2800
3000
3200
3400
3600
3800
4000
4200
BREAK
HPIS
HA
LPIS
BRU-A

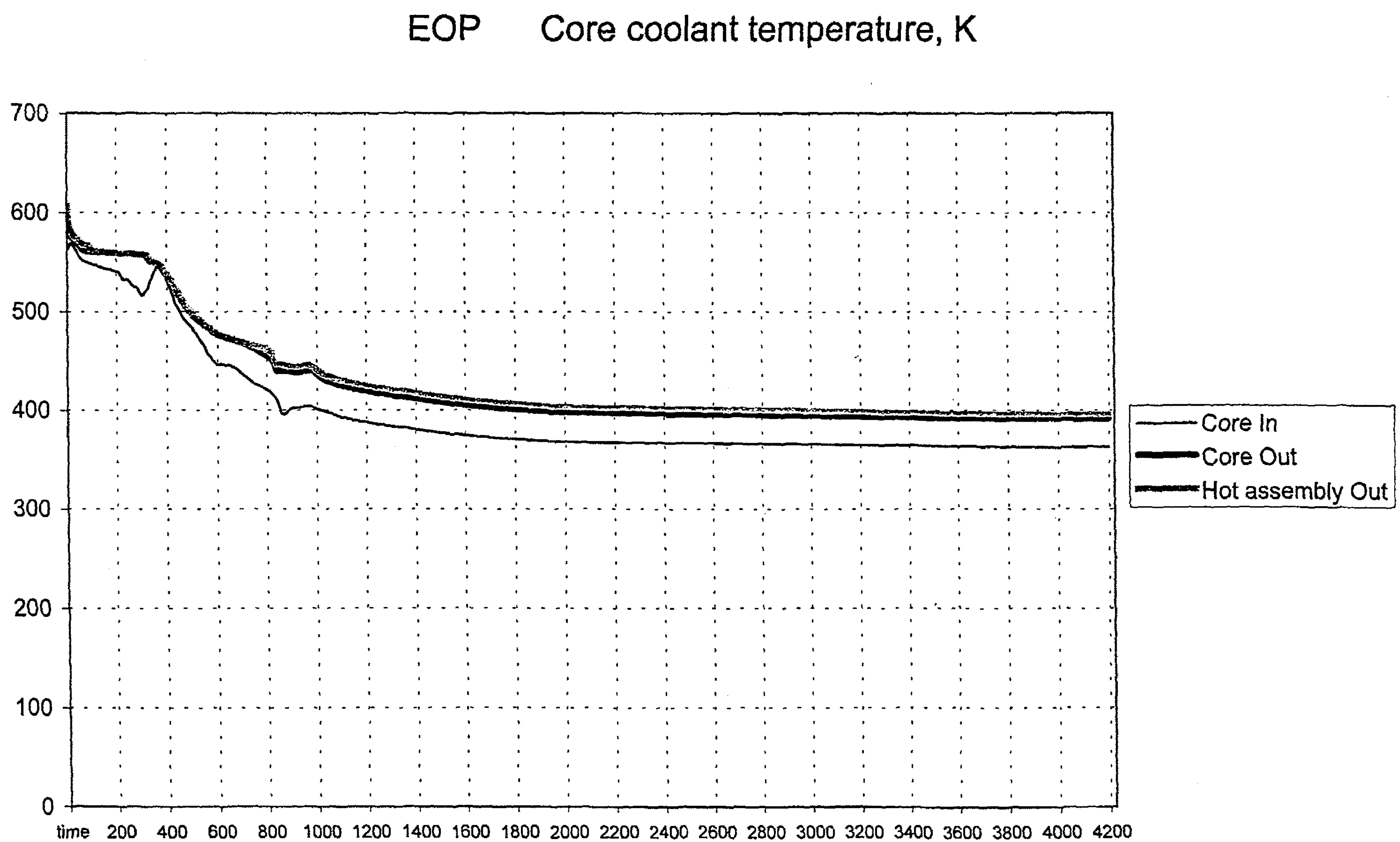


EOP
Steamgenerators level, m
4.5
4
3.5
3
2.5
2
1.5
1
0.5
0
time
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
2800
3000
3200
3400
3600
3800
4000
4200
BRU-A opening
EMERGENCY FEEDWATER is activated
SG1
Ruptured SG2
SG3
SG4

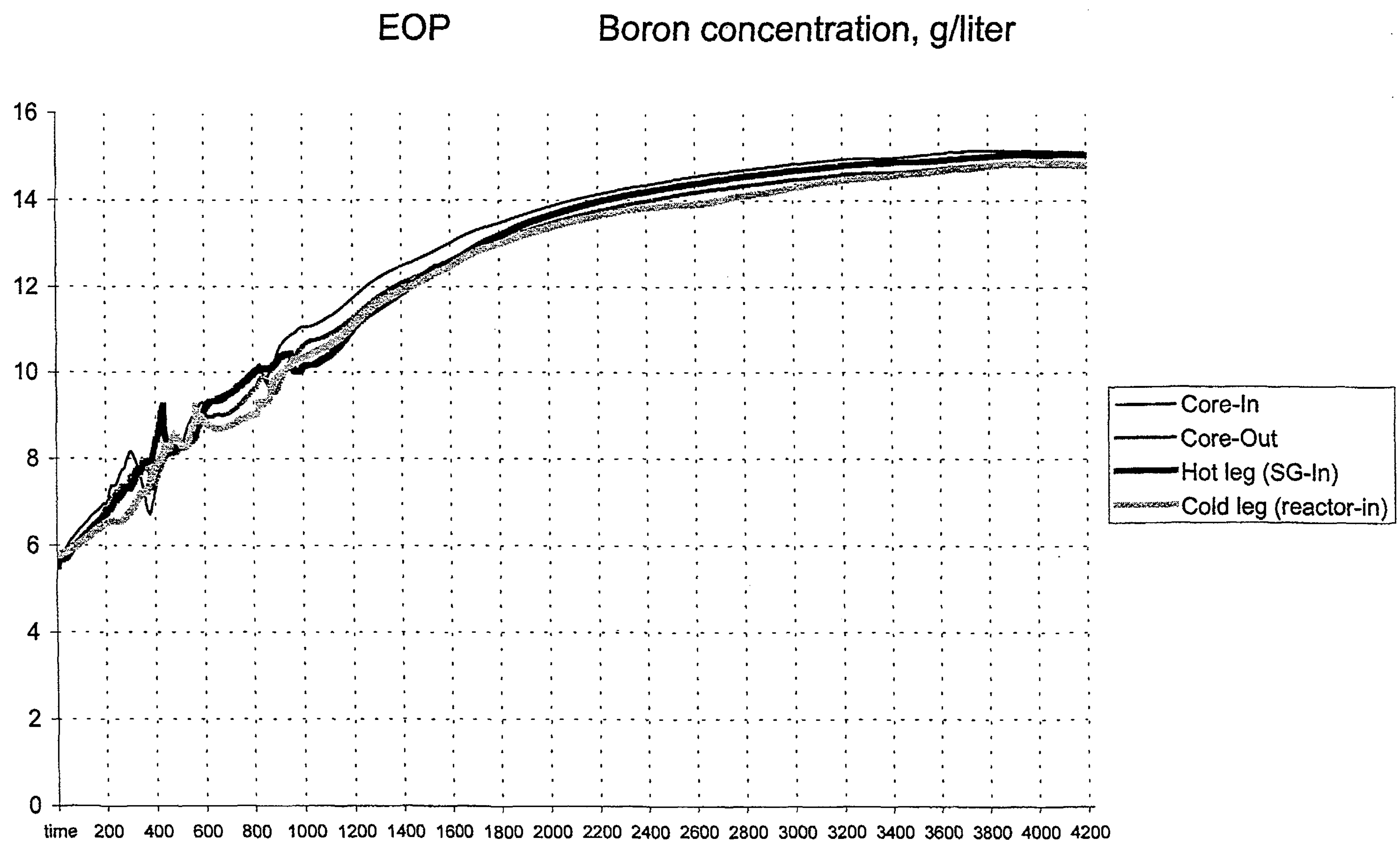

EOP
Steamgenerators secondary pressure
9.00E+06
8.00E+06
7.00E+06
6.00E+06
5.00E+06
4.00E+06
3.00E+06
2.00E+06
1.00E+06
0.00E+00
BRU-A opening
time 200 400 600 800 1000 1200 1400 1600 1800 2000 2200 2400 2600 2800 3000 3200 3400 3600 3800 4000 4200
SG1,3
Ruptured SG2
SG4

EOP Reactor coolant level, m





EOP Core coolant temperature, K
700
600
500
400
300
200
100
0
time 200 400 600 800 1000 1200 1400 1600 1800 2000 2200 2400 2600 2800 3000 3200 3400 3600 3800 4000 4200
Core In
Core Out
Hot assembly Out



EOP
Boron concentration, g/liter
16
14
12
10
8
6
4
2
0
time
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
2800
3000
3200
3400
3600
3800
4000
4200
Core-In
Core-Out
Hot leg (SG-In)
Cold leg (reactor-in)

6 CONCLUSIONS

Developed emergency operating procedures assure the following significant goals to mitigate accident sequences:

- optimal use of ECCS water inventory;
- severe core damage prevention;
- mitigation of environment radioactive contamination.

ABBREVIATIONS

BRU-A	dump valve to the atmosphere
BRU-K	bypass condenser valve
ECCS	emergency core cooling system
HPIS	high pressure injection system
LPIS	low pressure injection system
LWR	light water reactor
PWR	pressurized water reactor
SG	steam generator

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