ESCUELA TÉCNICA SUPERIOR DE INGENIERÍA (ICAI)

INGENIERO INDUSTRIAL

ANÁLISIS TERMOHIDRÁULICO DE DIVERSOS ACCIDENTES CON PÉRDIDA DE REFRIGERANTE EN UN REACTOR DE AGUA A PRESIÓN GENÉRICO

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Madrid

Junio 2015

UNIVERSIDAD PONTIFICIA ICAI ICADE COMILLAS MADRID

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Autora: Osorio Larraz, Cristina

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RESUMEN DEL PROYECTO

Introducción

La necesidad del estudio de la seguridad en todas las obras de ingeniería que presentan de forma innata algún elemento de riesgo para los operarios, el público y el entorno ha crecido a lo largo de los años. Es necesario cumplir una serie de criterios de seguridad en el diseño, con el fin de evitar cualquier daño y mejorar la fiabilidad estructural.

El objetivo prioritario en el diseño de una planta nuclear es asegurar que los productos de fisión permanezcan seguros dentro del sistema, impidiendo la liberación de radioactividad. Por esta razón, la Comisión Reguladora Americana ha adoptado el concepto 'defensa en profundidad'. Esta filosofía requiere la implantación de numerosas medidas para asegurar que no se superen las barreras físicas que impiden el escape de radioactividad. Las bases de esta filosofía son:

1. un cuidado diseño, construcción y vigilancia de la planta para la prever accidentes;
2. el uso de diversos sistemas de seguridad y dispositivos para minimizar los daños en caso de producirse un accidente;
3. la provisión de medidas de seguridad adicionales como sistemas de refrigeración de emergencia del núcleo o fuentes de energía independientes del reactor para operar los sistemas de seguridad.

Una serie de análisis determinísticos definen la eficacia del segundo y tercer precepto. Mediante el uso de un código computacional, se postulan los accidentes severos producidos por el fallo simultáneo de diferentes componentes y se modelan las consecuencias. Tales eventos se llaman accidentes base de diseño, y su análisis es la base del diseño y se requiere antes de que una planta pueda obtener una licencia de la Comisión Reguladora.

Accidentes base de diseño tales como una pequeña rotura en la tubería de vapor o un accidente con pérdida de refrigerante (en inglés: loss-of-coolant accident, LOCA) debido a una pequeña rotura en alguna de las tuberías, se consideran eventos improbables pero no imposibles. Existen cientos de kilómetros de tuberías en una planta nuclear y multitud de tubos de los generadores de que transfieren calor del lado primario al secundario. Se pueden producir fugas en cualquiera de los sistemas operativos, funcionales o de seguridad.

La gama de posibles ubicaciones de un LOCA por rotura pequeña (en inglés: small break loss of coolant accident, SBLOCA) es mayor que para un LOCA por rotura grande (en inglés: large break loss of coolant accident, LBLOCA), lo que lo convierte en un análisis de seguridad más complejo. Debido a los bruscos cambios termohidráulicos producidos en un LBLOCA, las consecuencias pueden ser muy graves. Sin embargo, las largas escalas de tiempo que presentan los SBLOCAs (que conducen a largos periodos de descubrimiento del núcleo y mayores cantidades de H_2 si se produce la oxidación del Zircaloy), también constituyen un reto para la seguridad. Por esta razón, el análisis de esta tesis se centra en estos desafíos.

Metodología

El análisis termohidráulico de los SBLOCAs se lleva a cabo utilizando un modelo ATHLET de un reactor de agua a presión (en inglés: Pressurized Water Reactor, PWR) alemán genérico (tipo vor-Konvoi). Este código computacional se compone de un método de Euler de integración numérica y diversos módulos que se utilizan para el análisis de los diferentes fenómenos implicados en el funcionamiento de un reactor de agua ligera. Mediante la conexión de elementos básicos, es posible simular la configuración deseada del sistema. La red hidráulica está representada por una sucesión de nodos interconectados. Es imprescindible una buena nodalización para simular los parámetros de flujo correctamente.

En este modelo, el lado primario del PWR consta de cuatro circuitos de refrigeración independientes conectados al reactor. Cada lazo se compone de una bomba de refrigerante, un generador de vapor, y las tuberías que conectan estos componentes a la vasija del reactor, es decir, una rama fría y una rama caliente. El presionador está conectado a un único lazo a través de la tubería de compensación. Además, el núcleo se divide en siete regiones: tres en el centro y cuatro en la periferia. En el lado secundario, hay cuatro circuitos independientes de agua de alimentación para el suministro de los generadores de vapor.

Para cada tamaño de rotura, se simulan el estado estacionario y el estado transitorio. Las roturas constituyen el 10%, 5% y 1% del área de la sección transversal de la rama fría, y una rotura de 20 cm2 en la tubería de compensación Además, para el caso de rotura del 10% del área, se tienen en cuenta diferentes orientaciones de fugas, así como la desactivación del sistema de inyección de alta presión. Todas las roturas se producen en el lazo 1, donde se encuentra el presionador, por lo que es probable que el agua que contiene fluya a través de la fuga y no sea capaz de enfriar el reactor.

Las condiciones iniciales son las mismas para los casos de ruptura en la rama fría, siendo la potencia del reactor de 4240 MW. Las condiciones después del transitorio también son iguales con respecto a los sistemas de inyección de baja presión de alta presión (disponibles únicamente en los bucles 1 y 2), las bombas de agua de alimentación de emergencia y la inyección adicional de agua borada (disponible en bucles 2, 3 y 4). Sin embargo, los acumuladores sólo están disponibles para los casos de ruptura del 5% y 1% del área de la sección transversal.

Para el caso de 20 cm2 de ruptura, la potencia del reactor es de 4000 MW. La señal de SCRAM se activa manualmente y 60 s después, tiene lugar la fuga en la tubería de compensación y se pierde el suministro eléctrico desde el exterior de la central. Los sistemas de inyección de alta y baja presión están desactivados, y los acumuladores están disponibles únicamente en la rama caliente. Las bombas de agua de alimentación de emergencia dejan de funciona en el momento en que se produce la pérdida de suministro del exterior (en inglés: Loss-of-Offsite Power, LOOP) y hasta que los generadores son capaces de proporcionar energía eléctrica.

Resultados

Las siguientes siguientes tablas muestran los resultados obtenidos en las simulaciones de los estados estacionario y transitorio.

Parámetro	Valor Estado Estacionario
Potencia del Reactor	4240 MW
Potencia Térmica Extraída por los Generadores de Vapor	4263 MW
Presión en el Circuito Primario	15.72 MPa
Presión del Vapor	6.86 MPa
Presión Total Perdida en el Circuito primario	0.42 MPa
Presión Perdida en el Reactor	0.198 MPa
Temperatura de entrada al Generador de Vapor	327.3 ºC
Temperatura de salida del Generador de Vapor	290.6 ºC
Caudal Másico en cada Lazo	5068 kg/s
Caudal Másico Neto en el Núcleo del Reactor	20094 kg/s
Nivel Colapsado en el Presionador	7.663 m
Nivel Colapsado en el Generador de Vapor	12.2 m
Concentración de Boro en el Circuito Primario	900 ppm

Tabla 1 Verificación Estado Estacionario de las Rupturas del 10%, 5% y 1% del Área Transversal

Parámetro	Valor Estado Estacionario
Potencia del Reactor	4000 MW
Potencia Térmica Extraída por los Generadores de Vapor	4023 MW
Presión en el Circuito Primario	15.7 MPa
Presión del Vapor	6.66 MPa
Presión Total Perdida en el Circuito primario	0.43 MPa
Presión Perdida en el Reactor	0.21 MPa
Temperatura de entrada al Generador de Vapor	330.7 ºC
Temperatura de salida del Generador de Vapor	296.3 ºC
Caudal Másico en cada Lazo	4976 kg/s
Caudal Másico Neto en el Núcleo del Reactor	19459 kg/s
Nivel Colapsado en el Presionador	8.21 m
Nivel Colapsado en el Generador de Vapor	12.2 m
Concentración de Boro en el Circuito Primario	900 ppm

Tabla 2 Verificación Estado Estacionario de la Ruptura de 20 cm^2

Acontecimiento	Tiempo [s]
Ruptura	0
Señal SCRAM (Baja Presión del Primario)	3,1
Disparo del Reactor y la Turbina, LOOP	3,1
Disparo de los Generadores de Vapor (curva 100k/h)	3,1
Inicio Inyección Agua Borada	3,6
Alcance Máxima Presión PermitidaSecundario	7,2
Alcance Valor de Actuación Sist. de Inyección de Alta Presión	9,6
Cambio Flujo de Fuga de Monofásico a Bifásico	20,4
Inicio Inyección de Alta Presión	41,1
Presión Primario Menor Presión Secundario	67,6
Nivel Mínimo de Refrigerante en el Núcleo	186,4
Cambio Flujo de Fuga a Monofásico Vapor	253,3
Inicio Inyección de Baja Presión	392,3
Vaciado de Tanques Alta Presión	4964,9
Planta Bajo Control	5500,0

Tabla 2 Estado Transitorio de la Ruptura del 10% del Área Transversal

Acontecimiento	Tiempo [s]
Ruptura	0
Señal SCRAM (Baja Presión del Primario)	4,2
Disparo del Reactor y la Turbina, LOOP	4,2
Disparo de los Generadores de Vapor (curva 100k/h)	4,2
Inicio Inyección Agua Borada	4,8
Alcance Máxima Presión Permitida Secundario	8,7
Alcance Valor de Actuación Sist. de Inyección de Alta Presión	13,1
Cambio Flujo de Fuga de Monofásico a Bifásico	36,3
Inicio Inyección de Alta Presión	42,3
Presión Primario Menor Presión Secundario	171,3
Nivel Mínimo de Refrigerante en el Núcleo	289,6
Inicio Inyección de Acumuladores	503,2
Inicio Inyección de Baja Presión	784,9
Inicio Inyección Agua de Alimentación de Emergencia	4705,1
Vaciado de Tanques Alta Presión	5116,3
Planta Bajo Control	5500,0

Tabla 4 Estado Transitorio de la Ruptura del 5% del Área Transversal

Acontecimiento	Tiempo [s]
Ruptura	0
Señal SCRAM (Baja Presión del Primario)	29,5
Disparo del Reactor y la Turbina, LOOP	29,5
Disparo de los Generadores de Vapor (curva 100k/h)	29,5
Inicio Inyección Agua Borada	30,0
Alcance Máxima Presión Permitida Secundario	44,9
Alcance Valor de Actuación Sist. de Inyección de Alta	54,7
Inicio Inyección de Alta Presión	67,6
Cambio Flujo de Fuga de Monofásico a Bifásico	221.8
Nivel Mínimo de Refrigerante en el Núcleo	551,6
Inicio Inyección Agua de Alimentación de Emergencia	686,8
Inicio Inyección de Acumuladores	3047,7
Presión Primario Menor Presión Secundario	4679,0
Vaciado de Tanques Alta Presión	5942,6
Planta Bajo Control	6000,0

Tabla 5 Estado Transitorio de la Ruptura del 1% del Área Transversal

Acontecimiento	Tiempo [s]
SCRAM Activado Manualmente	0
Disparo de los Generadores de Vapor (curva 100k/h)	0
Disparo del Reactor	0
Inicio Inyección Agua Borada	0,6
Alcance Máxima Presión Secundario	6,6
Inicio Inyección Agua de Alimentación de Emergencia	10,0
Alcance Mínima Presión Permitida Primario	23,5
Ruptura	60
LOOP	62,5
Cambio Flujo de Fuga de Monofásico a Bifásico	912,7
Cambio Flujo de Fuga a Monofásico Vapor	1733,3
Inicio Inyección de Acumuladores	3054,1
Nivel Mínimo de Refrigerante en el Núcleo	4531,0
Planta Bajo Control	8000,0

Tabla 6 Estado Transitorio de la Ruptura de 20 cm²

Conclusión

Los resultados de la simulación muestran que el núcleo del reactor no sufre daños graves durante el transitorio si el tamaño de la rotura es del 10%, 5% o 1% del área transversal de la rama fría, o si se encuentra en la tubería de compensación con una superficie de 20 cm2. La temperatura puede aumentar varios cientos de grados, estando todavía por debajo del punto de que el material de revestimiento (1200 ºC), lo que significa que el reactor permanecerá protegido de forma segura.

Los principales procesos físicos esperados durante este tipo de accidentes se han hecho visibles en los resultados, lo que revela que el código ATHLET implementado proporciona una buena base de confianza para el estudio de SBLOCAs en un PWR.

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Semesterarbeit

von cand. Ing. CRISTINA OSORIO LARRAZ
Matrikelnummer 03656834

Analysis of the Thermal-Hydraulics of several Loss of Coolant Accidents in a generic Pressurized Water Reactor

Betreuer TUM: Prof. Dr. Rafael Macian-Juan
Betreuer: Dipl.-Ing. Clotaire Geffray
Ausgegeben: 09.03.2015
Abgegeben: 01.07.2015

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München, den 01.07.2015.

Cristina Osorio Larraz

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Abstract

The need to ensure the safety in nuclear power plants for operators, public and environment make essential to take special care in design in order to avoid any hazard, improve structural reliability or minimize the damages in case of accident.

The purpose of the present thesis is to analyse the thermal-hydraulics behaviour in a pressurized water reactor during a small break loss-of-coolant accident (SBLOCA), so as to implement different modelling options and parameters in ATHLET code. Several scenarios are considered based on different break sizes, leak orientations and hypotheses on the availability of emergency core cooling systems.

In this model, the PWR consists of four independent coolant loops connected to the reactor on the primary side, and four independent feedwater loops on the secondary side to supply the steam generators. Furthermore, the core is divided into seven regions: three at the centre and four around the periphery.

The results obtained in the validation of the steady state and the transient calculation show the physical phenomena expected in this kind of accidents, which reveals that the ATHLET code provides a good confidence basis for the study of SBLOCAs in a PWR.

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List of Acronyms

CL Cold Leg

CSA Cross Sectional Area

CV Control Volume

DBA Design Basis Accident

DNB Departure Nucleate Boiling

ECC Emergency Core Cooling

ECCS Emergency Core Cooling System

FEBE Forward-Euler, Backward-Euler

GCSM General Control Simulation Module

GRS Gesellschaft für Anlagen- und Reaktorsicherheit (gGmbH)

HCO Heat Conduction Object

HECU Heat Transfer and Heat Conduction

HL Hot Leg

HPIS High Pressure Injection System

LBLOCA Large Break Loss-of-Coolant Accident

LOCA Loss-of-Coolant Accident

LOOP Loss of Offsite Power

LP Lower Plenum

LPIS Low Pressure Injection System

NEUKIN Neutron Kinetics

NRC Nuclear Regulatory Commission

ODE Ordinary differential equation

PCT Peak Cladding Temperature

PORV Power-Operated Relief Valve

PWR Pressurized Water Reactor

RCS Reactor Coolant System

RPV Reactor Pressure Vessel

SBLOCA Small Break Loss-of-Coolant Accident

SG Steam Generator

TFD Thermo-fluiddynamics

TFO Thermo-fluiddynamic object

UH Upper Head

UP Upper Plenum

Chapter 1

Introduction

The need for safety analysis in all the engineered systems that innately present some element of risk to the operators, the public or the environment has grown over the years. It is essential to meet a series of safety criteria in design in order to avoid any damages and improve structural reliability.

The central safety objective in the design of a nuclear plant is to assure that the fission products remain safely enclosed, preventing the release of radioactivity. It is for this reason that the Nuclear Regulatory Commission (NRC) has adopted the defense-in-depth concept. This approach requires the implementation of several measures in order to assure that the physical barriers used to prevent the escape of radioactivity are not overcome. As described in [TODR90], these standards are:

- (1) a careful design, construction and surveillance of the plant for the prevention of accidents;

- (2) the use of various safety systems and devices to minimize damage if an accident occurs;

- (3) the provision of additional safety features like emergency core cooling systems or power sources independent of the reactor to operate the safety systems.

A series of deterministic analysis define the effectiveness of the second and third precepts. Severe incidents emerging from the simultaneous failure of different components are postulated and the consequences are modeled using a computer code. Such events are called design basis accidents (DBAs), and their analysis is indispensable for the design and licensing of a nuclear power plant.

The NRC has classified the spectrum of possible accidents into nine categories, in ascending order of severity. Class 1 refers to trivial accidents that release levels of radioactivity that are not substantially different from those involved in normal operation. Classes 2 to 7 deal with fuel accidents and the release of higher amounts of radioactivity. Class 8 includes DBAs, like loss-of-coolant accidents (LOCA), rod ejection accidents or steam-line breaks. Severe accidents or accidents that involve a sequence of unlikely events are present in class 9.

The basic principle of safety design is that the possible injury or damage should be significantly less for accidents that are likely to occur than for those which are unlikely. The range of potential occurrences is categorized into three major groups:

1. Events of moderate frequency (several times per year) leading to insignificant radioactive releases from the facility.

2. Events of low probability (10^{-1} to 10^{-2} per year) with the potential for small radioactivity releases.

3. Highly unlikely (10^{-3} to 10^{-4} per year) for which safety features are provided to mitigate potentially severe consequences.

Design basis accidents like a small size steam-line break or a LOCA due to a small size piping rupture are considered infrequent events and placed in the second group. Moreover, a break with rod ejection or a LOCA due to a large size break would be placed in the third group.

The fact that the loss-of-coolant accidents are unlikely to occur does not mean that they are impossible. There are hundreds of kilometers of pipes in a nuclear power plant and thousands of steam generator tubes in the primary side from which the heat is transferred to the secondary circuit. Leaks can appear in any of the operational, functional or safety systems.

1.1 Background

Many accidents caused by loss-of-coolant have occurred at nuclear power plants since the 1950s, though very few resulted in high damage to plant operators or the general public. The three most serious events caused by loss-of-coolant are the accidents at Lucens (1969), Three Mile Island (1979) and Mihama-2 (1991).

1.1.1 The Lucens Accident (Vaud, Switzerland)

The small Swiss-designed Lucens reactor was built in 1962 in an underground cavern. It was moderated with heavy water, cooled with carbon dioxide gas and fueled with enriched uranium. The reactor produced 30 MW of heat, used to generate 8 MW of electricity.

On January 21, 1969, the reactor suffered a LOCA during a startup, leading to a partial core meltdown and massive contamination of the cavern. The cavern was then sealed and the Swiss government ordered an investigation.

The high humidity of the cavern caused the corrosion of the fuel element components, leading to an accident. The corroded products obstructed one of the fuel channels and prevented the flow of carbon dioxide coolant, so that the cladding melted and further blocked the channel. The temperature increase caused the split of the pressure tube surrounding the fuel channel, and the coolant leaked out of the reactor.

Though the cavern containing the reactor was substantially contaminated, neither plant workers nor the public suffered irradiation [LELI12]

1.1.2 The Three Mile Island Accident (Pennsylvania, United States)

The worst accident in the United States occurred at the Three Mile Island Unit 2 reactor on March 28, 1979. The reactor was a Babcock & Wilcox pressurized water reactor (PWR) with an electrical capacity of 961 MW.

A combination of element malfunctions, design-related problems and worker errors, initiated a chain of events that led to a partial meltdown of the core and small releases of radioactivity.

The accident began when a condensate pump moving water from the condensers in the turbine building stopped, preventing the main feedwater pumps from injecting water into the steam generators that extract heat from the core. This lead to the trip of the turbine.

The primary pressure started to rise, which caused the opening of the power-operated relief valve (PORV) on top of the pressurizer. However, in order to reduce the pressure immediately, this was insufficient and the control rods were automatically driven into the core. This immediately stopped the fission reaction.

The decreasing coolant pressure reached the set point for automatic closure of the PORV, but the valve failed to close and primary coolant water started to leak through the PORV, which stuck open. Instruments in the control room indicated that the valve was closed, so the operators did not realize that a leak was occurring.

The reactor pressure dropped significantly and the emergency core cooling systems (ECCS) triggered automatically, injecting cold borated water into the RPV. The liquid level in the pressurizer continued rising rapidly. As there was no instrumentation showing how much water covered the core, the operators assumed that as long as the pressurizer water level was high, the core was being well covered.

The cooling water escaping through the PORV reduced the primary system pressure significantly, as a result the operators turned off the coolant pumps to prevent dangerous vibrations. The amount of emergency cooling water that was being pumped to the primary circuit was also reduced in order to prevent the pressurizer from filling up. These actions deprived the reactor from coolant, causing it to overheat. The nuclear fuel overheated to a point which burst the cladding and began to melt the fuel pellets.

Although the core suffered a severe meltdown, the consequences outside the plant were minimal and the small radioactive releases caused no significant harm to the workers, population or the environment, as the NRC state in [ROGO80]

Many lessons were learned from the Three Mile Island accident, the improvement in instrumentation and procedures significantly increase the likelihood of withstanding a LOCA without major core damage.

1.1.3 The Mihama-2 Accident (Fukui Prefecture, Japan)

The accident that occurred at Mihama Unit 2 on February 9, 1991, was the first incident in Japan where the ECCS was actuated. The Mihama-2 is a PWR that can generate 500 MW of electricity.

While Unit 2 of the Mihama Power Plant Station, was in operation, the personnel received an increasing radioactivity signal from one of the steam generators. The pressure in the primary circuit and the water level in the pressurizer started to decrease and the operators activated the charging pumps, thus reducing the reactor power. The reactor, turbine and generator shut down automatically because of the low pressurizer water level and the ECCS was activated.

The pressurizer relief valves were opened in order to equalize the pressures on the primary and secondary side, but this operation was not successful. Finally, the depressurization of the primary side was undertaken by the pressurizer spray system and the high pressure injection system (HPIS) pumps were stopped, terminating the leakage from the primary to the secondary circuit.

Subsequent analysis determined that the cause of the accident was a rupture on the steam generator tube, which allowed 55 tons of coolant to pass from the primary to the secondary systems and 1.3 tons of radioactive steam were released into the turbine hall. The experts established that the small amount of radioactivity released to the environment, was far below the safety hazard levels.

Further details regarding these accidents can be found in [HEWI00]

1.2 Motivation

Loss-of-coolant accidents in a nuclear power plant have to be thoroughly investigated since the failure to extract the decay heat from the reactor core can endanger the integrity of the fuel. Therefore, they are paramount to the safety of the power plant and of its surroundings.

The LOCAs experienced at Lucens, Three Mile Island and Mihama-2, highlight that such accidents can really take place, and that adequate prevention and mitigation measures of these accidents is essential for avoiding grave consequences.

The loss of-coolant can be caused by a stuck-open relief valve or by a small or a large break in any of the pipes. The main difference between a small break LOCA (SBLOCA) and a large break LOCA (LBLOCA) lies in the rates of coolant discharge and pressure variations with time. In a LBLOCA, the reactor will immediately go subcritical due to the rapid loss of reactor coolant; while the SBLOCAs are characterized by an extended period after the break, in which the primary side remains at a relatively high pressure and the core remains covered.

The range of possible locations of a SBLOCA is wider than for LBLOCAs, which makes for a more complex safety analysis. As in the LBLOCAs the reactor is put under rapid changes of thermal-hydraulic conditions as a result the consequences can be very severe if not mitigated properly and fast enough. However, the longer time scales presented by the SBLOCAs are also challenging for safety leading to long uncover and possibly larger amounts of H_2 if oxidation of the Zircaloy occurs. For this reason the analysis of this thesis will be focused on these challenges.

1.3 Objectives

The present thesis is aimed at the analysis of the thermal-hydraulics of SBLOCAs using an ATHLET model of a typical German PWR (vor-Konvoi type). Different scenarios are considered based on different break sizes, leak orientations and hypotheses on the availability of emergency core cooling systems.

The break size is expressed relatively to the cross sectional area (CSA) of the pipe. In this case, the ruptures are of 1%, 5% and 10 % of the CSA of the cold leg (CL), and of 20 cm^2.

Different hypothetical situations such as the loss of offsite power (LOOP) or the blockage of the accumulators are also taken into account.

Chapter 2

ATHLET Description

The thermal-hydraulic computer code ATHLET (Analysis of THermal-Hydraulics of LEaks and Transients) has been developed by the Gesellschaft für Anlagen- und Reaktorsicherheit gGmbH (GRS), and allows an easy implementation of different physical models.

The ATHLET structure is defined in [LERC09]. This code is composed of a numerical integration method FEBE (Forward-Euler, Backward Euler) and diverse modules which are used for the analysis of the different phenomena involved in the operation of a light water reactor:

- Thermo-fluiddynamics (TFD)
- Heat Transfer and Heat Conduction (HECU)
- Neutron Kinetics (NEUKIN)
- General Control Simulation Module (GCSM)

Connecting basic fluiddynamic elements, called thermo-fluiddynamic objects (TFOs), it is possible to simulate the desired system configuration. There are several TFO types which are classified into three fundamental categories: pipe objects, branch objects and special objects. Even though most of the TFOs are pipes, applied for a one-dimensional TFD-Model with partial differential equations describing the transport of fluid.

The TFOs are composed of two elements: the control volumes (CVs) and the interconnections between them, called junctions. The junction centres are displaced at the mid-length between two adjacent CV centres, as shown in Figure 2.1 [AUST09].

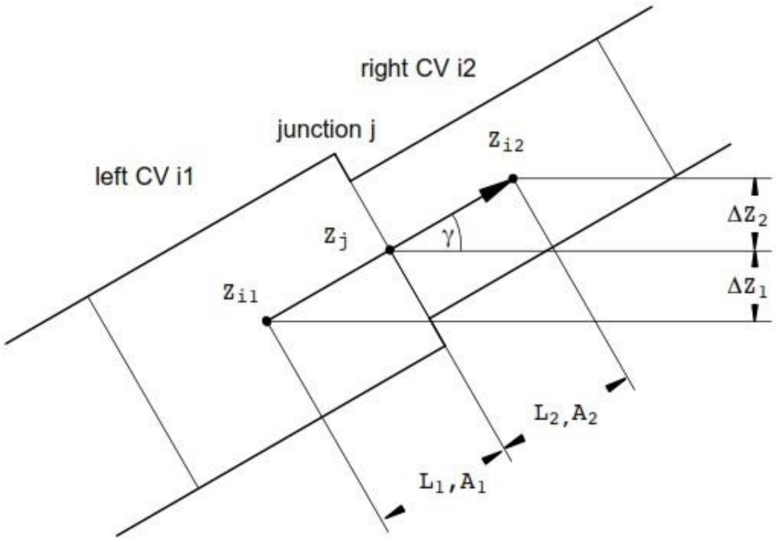

Figure 2.1 Junction connecting two adjacent control volumes with different areas

This object structure allows the coupling of different physical models and spatial discretization techniques. For each thermal-fluid object, a subset of the entire ordinary differential equation (ODE) system is applied. In the case of the pipe objects, a spatial approximation method can be applied to obtain the ODEs from the partial differential equations.

ATHLET offers different models for the simulation of fluiddynamics, like the two-fluid model, the 4- , 5- or 6-equation model, or the drift-flux model. The two-fluid model uses separate conservation equations for liquid and vapour mass, energy and momentum, while the equation model uses a mixture equation for the momentum and includes a mixture level tracking capability. The solution variables are the pressure, liquid temperature, vapour temperature and mass quality, as well as mass flow rate (5-equation model), the phase mass velocities (6-equation model) or the relative velocity between phases (drift-flux model). The 4-equation model is only applied at the steady state calculation, having as solution variables the pressure, the mass quality, the enthalpy and the mass flow rate.

Heat conduction in structures, fuel rods and electrical heaters, can be simulated with heat conduction objects (HCOs). These objects are applied for a one-dimensional HECU module, which provides a simulation of the temperature profile and the energy transport in solid materials.

The NEUKIN module models the nuclear heat generation. The power generated in the nuclear reactor consists of the prompt power from fission of short-lived fission products and the decay heat power from the long-lived fission products. A GCSM signal provides the steady state part of the decay heat. For the computation of the time-dependent behaviour of the prompt power generation either a point-kinetics model or a one-dimensional neutron dynamics model can be used.

Chapter 3

Nodalization

The hydraulic network is represented by a succession of interconnected nodes that form a node-link diagram. A fine nodalization is necessary for a correct simulation of the flow parameters.

The reactor modelled in the context of this work[1] is a generic german PWR with four independent coolant loops connected to the reactor on the primary side, and four independent feedwater loops on the secondary side to supply the steam generators.

The water is heated in the reactor pressure vessel (RPV) in contact with the fuel rods and flows through the hot legs (HLs) towards the steam generators (SGs). This heat when transferred generates steam at the secondary circuit which drives a turbine coupled to a generator, converting the thermal energy into mechanical energy and then into electrical energy. The steam leaving the turbine is condensed into water in a condenser before being pumped to the SGs again. The cooled

[1] The nodalization of the reactor has not been designed in this thesis.
The existing model developed at the Department of Nuclear Engineering at TUM has been extended to realize the analysis.

water is routed again through the cold legs (CLs) to the core. Figure 3.1 [USNR12] shows a schematic representation of the operation of a PWR.

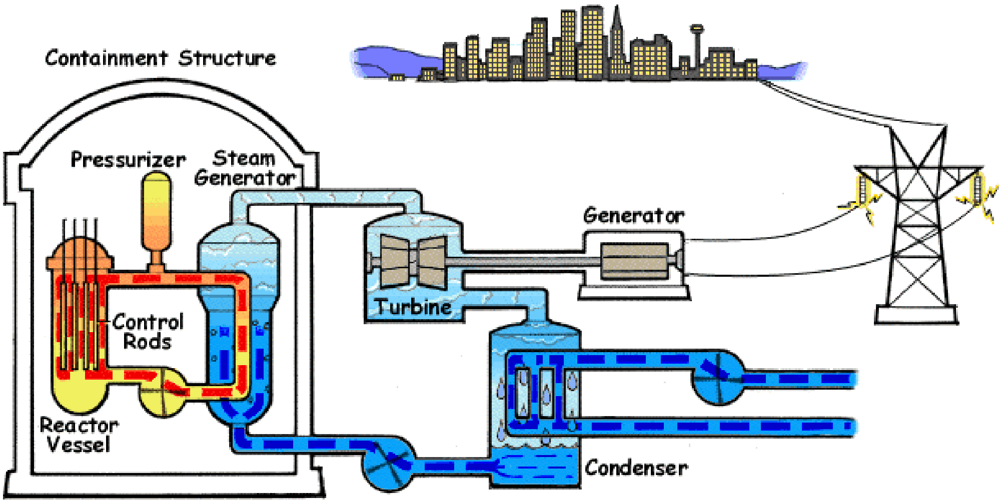

Figure 3.1 Schematic representation of a PWR

3.1 Primary side

The primary side, also known as the reactor coolant system (RCS), consists of the reactor pressure vessel, the main coolant pumps, the steam generators, a pressurizer and the connecting piping (surge line, cold leg and hot leg).

Each loop is composed of a reactor coolant pump, a steam generator, and the pipes that connect these components to the reactor vessel i.e. a cold leg and a hot leg. Additionally, the pressurizer is connected to one loop via the surge line, as shown in Figure 3.2 [LOBN90]

The main functions of the RCS are to transfer the heat from the fuel rods to the steam generator and to contain any fission products that escape the cladding.

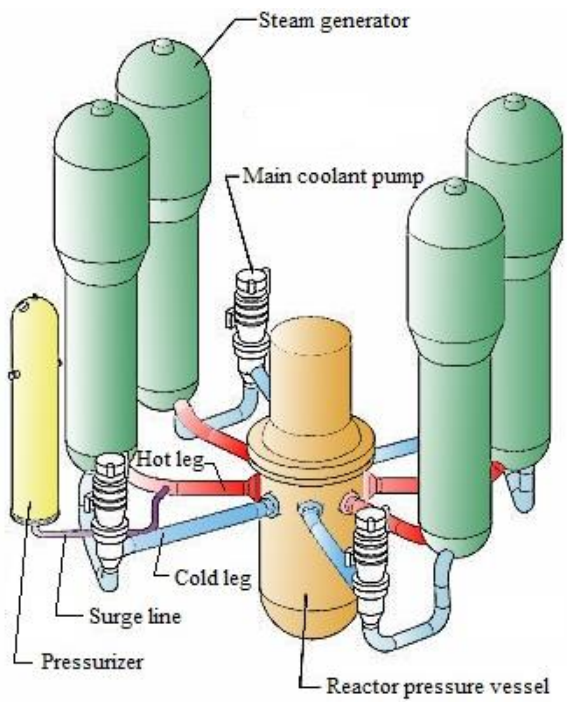

Figure 3.2 Reactor Cooling System of a PWR

Figure 3.3 shows the coolant loop modelled with ATHLET with a leak located in the cold leg.

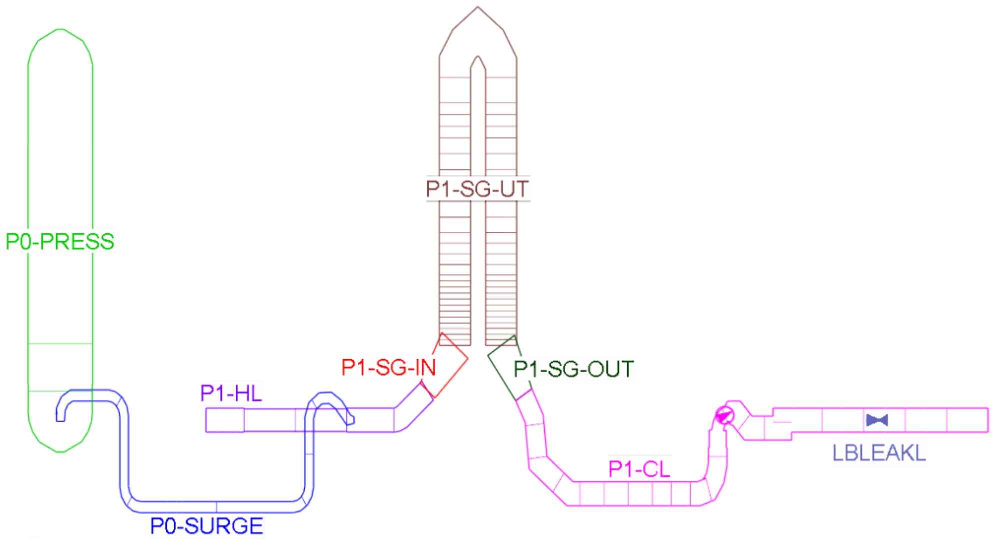

Figure 3.3 Coolant loop with a leak at the CL modelled with ATHLET

The reactor core and all the devices tasked with support and alignment are lodged within the RPV.

The configuration of the core used for the simulation is shown in Figure 3.4. The core is divided into seven regions: three at the centre and four around the periphery. Each region is modeled with one TFO, which is coupled to a HCO representing the fuel rods.

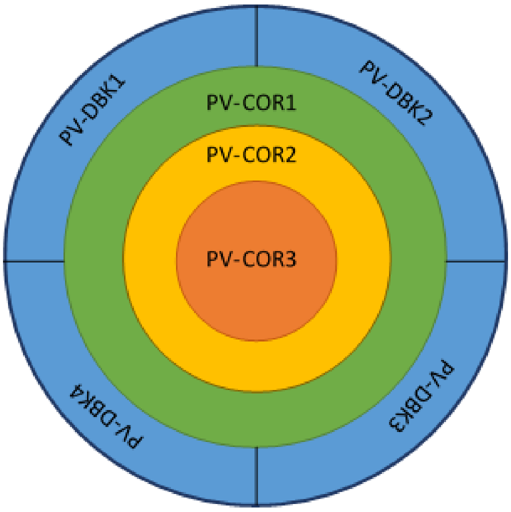

Figure 3.4 Schematic representation of the reactor core

The power is highest in the centre and decreases as approaching the periphery. An additional HCO is coupled to the PV-COR3 and represents the hottest fuel rod of the core (HPV-COR3 represents the hottest fuel assembly, HPV-COR4 the hottest fuel rod in this assembly).

The downcomer, lower plenum (LP), reflector, upper plenum and upper head (UH) are also included in the model. The downcomer is divided into four parts, each one connected to one HL and one CL. Figure 3.5 shows the RPV nodalization picture created with help of ATHLET.

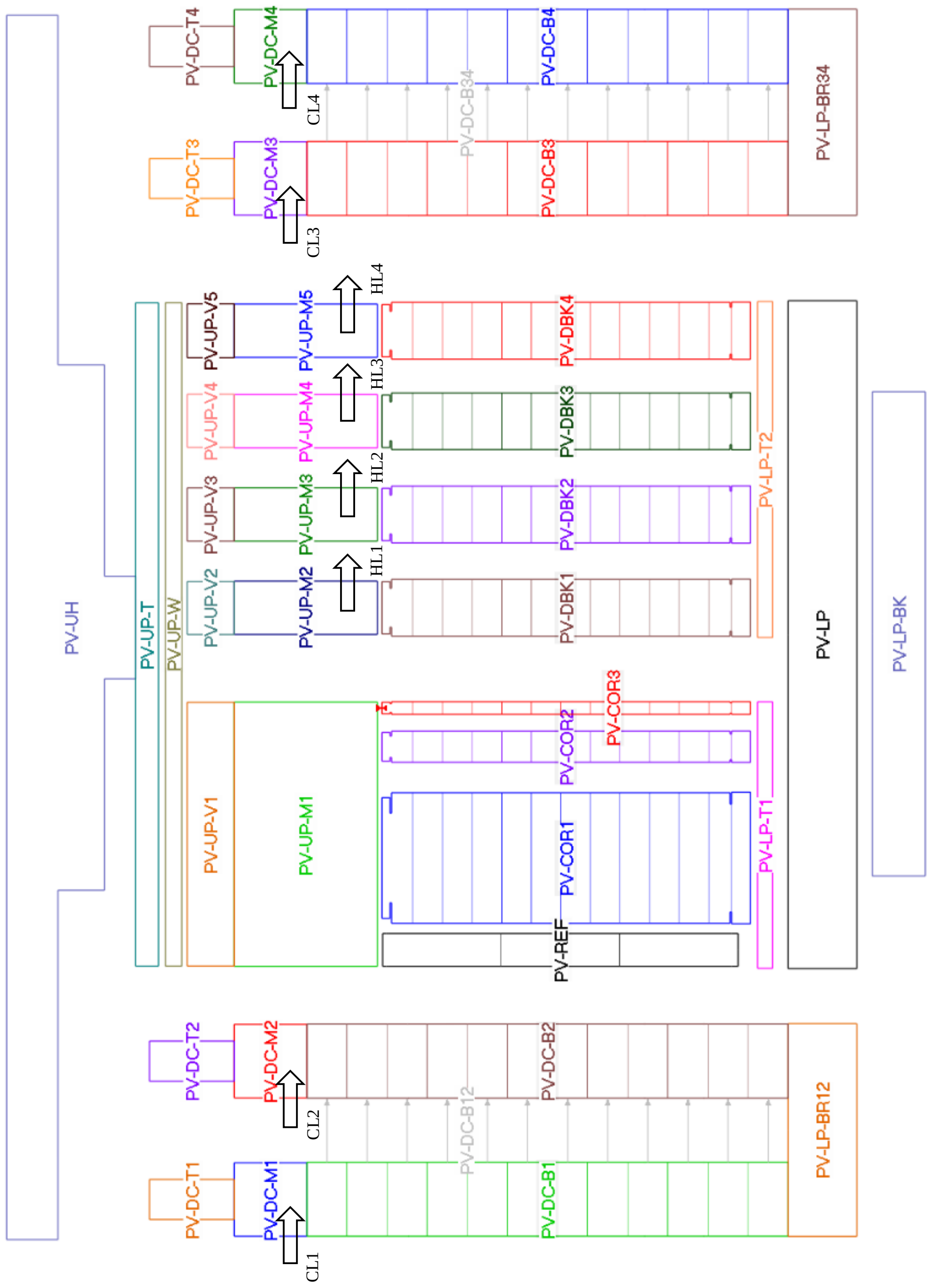

Figure 3.5 Schematic representation of the RPV nodalization

Chapter 4

Model Description

4.1Point Kinetics Model

The point kinetics is used to simulate the reactor neutron kinetics. This model assumes that the power of a nuclear fuel is proportional to the average neutron flux. The neutron flux is measured by the kinetics equation for one group of prompt neutrons and six groups of delayed neutrons:

$$\frac{dn}{dt} = \frac{R(t) - \beta}{L} \, . \, n + \sum_{i=1}^{NG} (\lambda_i \, . \, C_i) \tag{4.1}$$

$$\frac{dC_i}{dt} = \frac{\beta_i}{L} \, . \, n - \lambda_i \, . \, C_i \qquad i = 1, \ldots, NG \tag{4.2}$$

where:

n : neutron flux

R : total reactivity

β_i : effective fraction of delayed neutron group i

β : effective delayed neutron fraction, $\beta = \sum_{i=1}^{NG} \beta_i$

L : neutron generation time

λ_i : decay constant of the delayed neutron group i

C_i : concentration of delayed neutron group i

NG : number of delayed neutron groups

The reactivity R(t) includes the external reactivity R_{ext}, which accounts for the control rod positions, and the feedback reactivity R_{FB}:

$$R(t) = R_{ext}(t) + R_{FB}(t) \tag{4.3}$$

The feedback reactivity consists of the reactivity contribution of the fuel temperature R_{Tf}, of the coolant temperature R_{Tc}, of the coolant density $R_{\rho c}$, and of the boron concentration R_B:

$$R_{FB}(t) = R_{Tf}(t) + R_{Tc}(t) + R_{\rho c}(t) + R_B(t) \tag{4.4}$$

4.2 Quench Front Model

The quench front model is used to calculate the position and the movement of the different quench regions when rewetting the core. The model is only applied during the transient calculation and for fluid pressures below 3 MPa. Otherwise, a spontaneous quench model is activated.

The calculation of upper and lower quench front position takes into account axial heat conduction in the cladding and pre-cooling of the dry rod surface near the quench front, as well as reducing the influence of nodalization on the calculated results.

As shown in Figure 4.1 [TODR90], the boiling curve divides the physical region into 4 heat transfer levels at certain temperatures: convection, nucleate boiling, transition boiling and film boiling. The calculated quench front position determines the change in the heat transfer regime from nucleate boiling to film boiling (Leidenfrost effect).

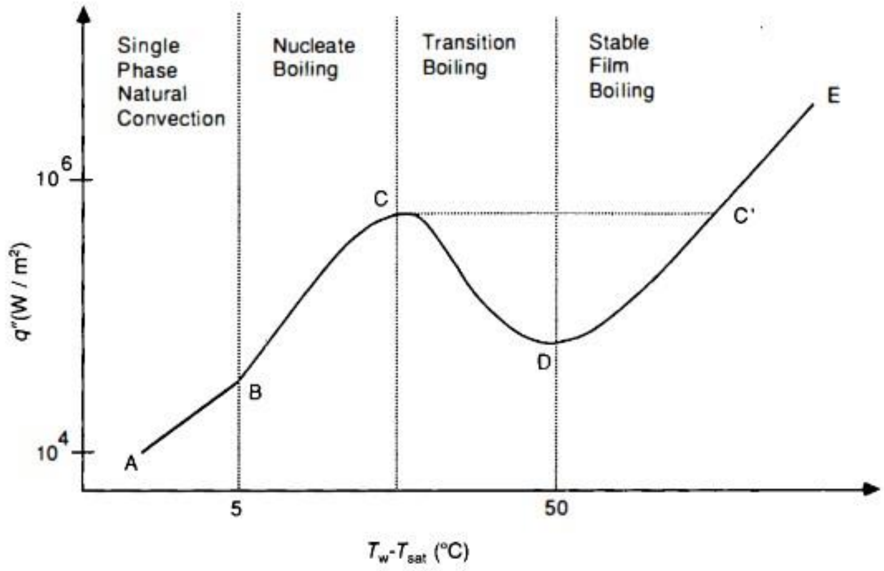

Figure 4.1 Typical pool boiling curve for water under atmospheric pressure

4.3 Critical Discharge Model

The critical discharge model (CDR1D model) is applied for the representation of the region where a leak takes place. This region, show in Figures 4.2 and 4.3 [LERC09], is characterized by an anomalous behaviour, with thermal non-equilibrium of the fluid, substantial variances of the thermal-hydraulic quantities and critical mass flow at the leak.

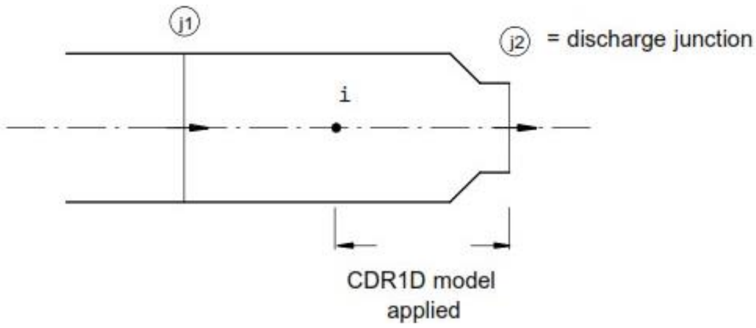

Figure 4.2 Application of the CDR1D model at the end of a pipe

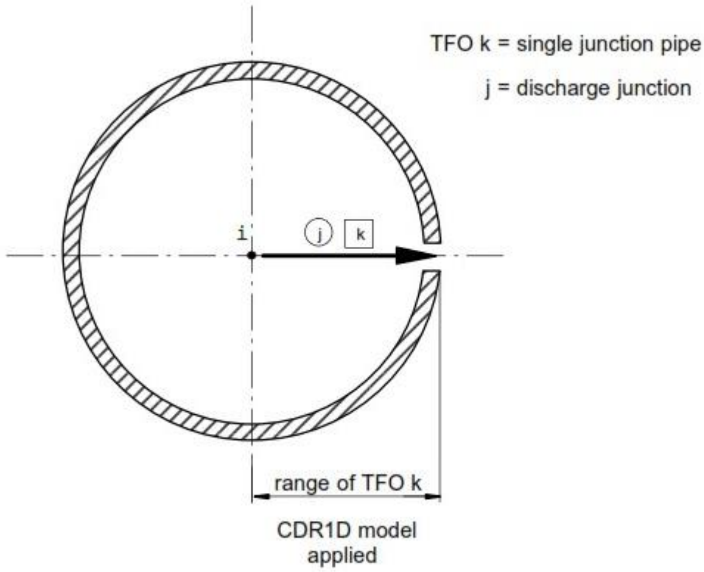

Figure 4.3 Application of the CDR1D model within a single junction pipe to simulate a discharge from a leak in a main coolant pipe

The critical mass flow at the discharge region is calculated automatically in this model by the multiplication of the current critical mass flux (determined by interpolation) and the current discharge cross sectional area. For this purpose, the model only uses the break geometry and the consideration of non-equilibrium of the fluid. The following figures 4.4, 4.5 and 4.6 [LERC09] show the applicable types of geometry: with constant CSA, with a thin orifice plate and with a flow limiter of finite length.

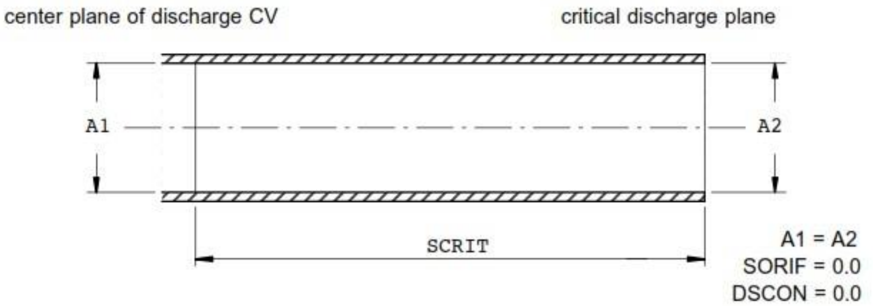

Figure 4.4 Discharge geometry with constant CSA

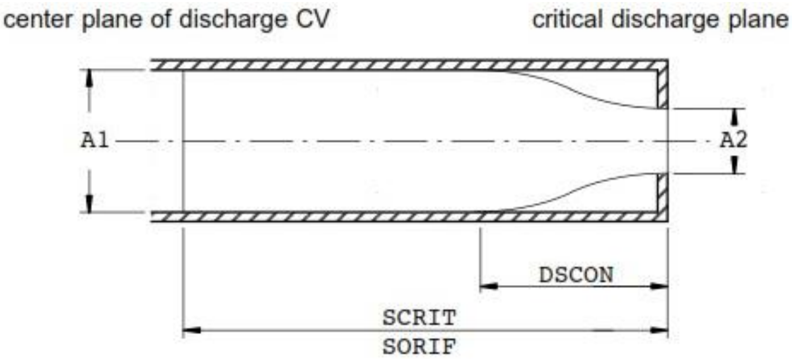

Figure 4.5 Discharge geometry with a thin orifice plate

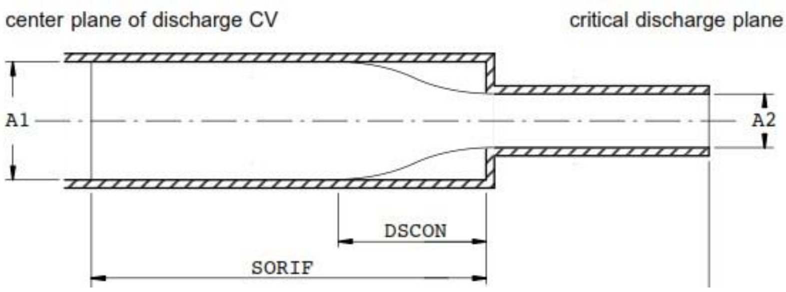

Figure 4.6 Discharge geometry with a flow limiter of finite length

4.4 SCOOP Injection Model

The scoop model is applied on the injection of emergency core cooling water in the hot leg via a scoop flow device that guides the liquid towards the upper plenum (UP). A thermal stratification will be established between the saturated liquid originating from the UP and the sub-cooled ECC liquid, as shown in Figure 4.7 from [AUST09].

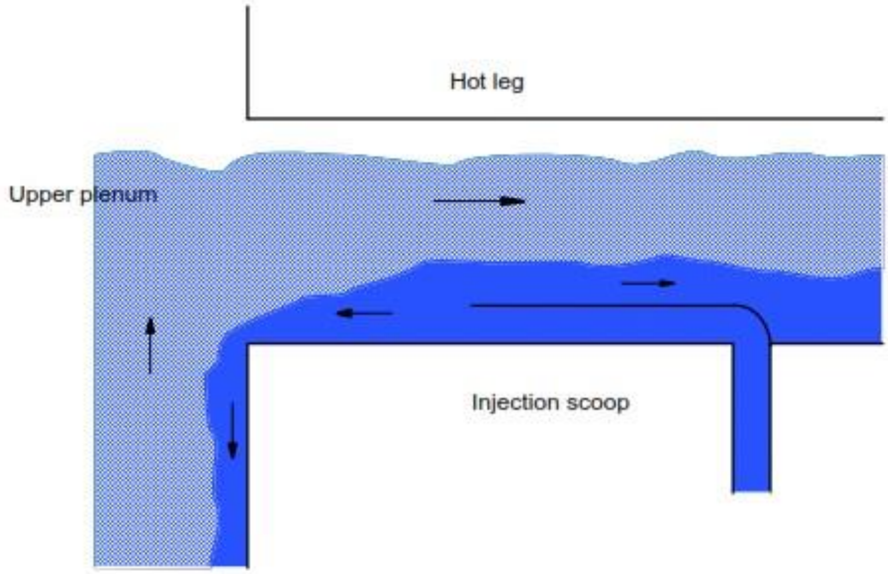

Figure 4.7 Thermal stratification of saturated and sub-cooled liquid caused by ECC scoop injection

The cold liquid injected flows counter-currently to the hot leg flow towards the UP and descends in an 'ECC injection channel' due to the gravity force. The hot leg diameter, the void fraction, the velocity of the saturated liquid in the hot leg and the densities of both the saturated and the sub-cooled liquids influence the amount of liquid directed to the upper plenum.

4.5 ECCMIX Model

The EECMIX model is applied on the injection of cold ECC water into a cold leg largely filled with hot coolant. A thermal stratification will be established and the mixture will flow at the bottom of the pipe towards the reactor vessel annulus, as shown in Figure 4.8 [LERC09]. With high injection rates, the cold leg will be filled with cold ECC water; whereas with low injection rates, it will be filled with

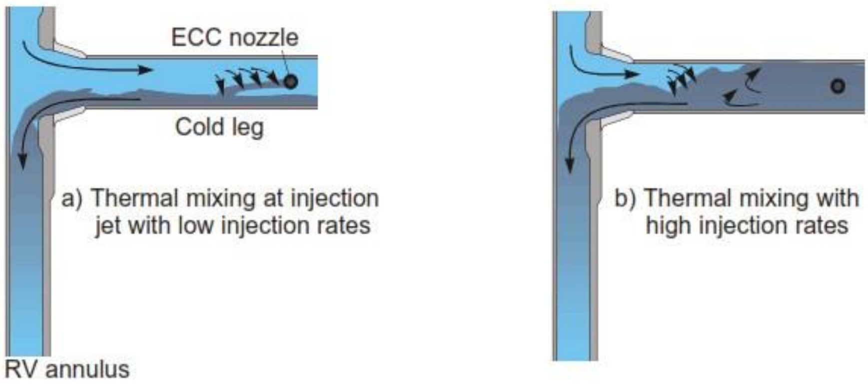

Figure 4.8 Thermal mixing in the cold leg during ECC injection

The efficiency of this process is contingent on the cold leg geometry, the injection mass flow, the liquid level on the core leg and the density difference between the coolant and the ECC water.

Chapter 5

Small Break Loss-of-Coolant Accident

The SBLOCA is characterized by having a break flow area of less than 460 cm2. The accident can be caused by the rupture of a SG tube, a stuck-open relief valve, a small break in the piping, or the rupture of the housing of the control rod housing. The worst effects are expected when the break is in the cold leg, as the entry of ECC water to the downcomer is restricted because of the rapid upflow of steam.

5.1 General Features

The following general features characterize a SBLOCA:

1. The rapid loss of coolant in the affected pipe can cause a power increase inside the reactor.

2. The depressurization is much slower than in LBLOCAs.

3. Reactor shut down systems may not detect the loss immediately so that the reactor continues operating within the safety margins during some time because of the longer time scales.

4. The rate coolant discharge that occurs from the break into reactor subsystems is lower compared to the discharge rate in the LBLOCAs.

5. Fuel failures occur in the case of flow blockages in the affected pipe or if the temperature reaches the melting point of the cladding.

5.2 Main Phases

Blowdown

When a break occurs, the pressure starts decreasing and the reactor is tripped automatically. As the pressure reaches the set point of the HPIS, ECC water starts to be injected. The pressure continues decreasing and the water reaches saturation point in the hotter parts of the primary circuit. Steam forms and collects in the UH of the reactor.

Reflux condensation

The water moves down to the level of the inlet and outlet pipes and the pressure remains high, impeding the actuation of the accumulators and LPIS. The original water content of the SGs is lost through the leak, which allows for the steam from the reactor to flow through the SGs tubes to be condensed and flow back into the core.

First core uncovery

As the ECC water does not compensate for the leaking water, the level keeps dropping and the core begins to uncover from the top downward. The system is still pressurized due to the steam that cannot escape through the break, being blocked by the water in the vessel and the pump. The pump has a U-bend (the loop seal) that remains full of water and prevents the steam from flowing out from the vessel. This water can reach the core.

Loop seal clearance and second uncovery

When the water level reaches the bottom of the U-bend, the steam can pass the loop seal. This involves a rapid depressurization and thus the core may be dried out again.

Long term cooling

The depressurization permits the actuation of the LPIS and the accumulators, reflooding the core and taking it to a cold condition.

The main phases of SBLOCAs are shown in detail in [TONG96].

Chapter 6

Model Simulation

The model developed at the Department of Nuclear Engineering [GEFF12] has been used as a basis for the work reported here. It was adapted in order to improve the model and conform to the desired hypotheses.

Different break sizes have been simulated: 10%, 5%, and 1% of the CSA of the CL, and 20 cm^2 in the surge line. Additionally, for the 10% case, different leak orientations have been taken into account, as well as the deactivation of the high pressure injection system. All the breaks occur in Loop 1, where the pressurizer is located so the water it contains is likely to flow through the leak and is not amenable to cool down the reactor.

Conservative initial and boundary conditions are specified. In this sense, the power of the core during the steady state is greater than the nominal power (4000 MW) and some of the ECCS components are assumed not to be available as the result of maintenance or of failure as they are required to start.

Furthermore, a preliminary analysis has been performed to verify that the model reaches a stable steady state before the leak opens[2].

The reactor trip occurs by the rapid insertion of the control rods, either automatically or manually. The SCRAM signal is implemented to initiate the shutdown of the reactor. The automatic SCRAM is triggered when one of the following criteria is met:

- The primary side pressure is lower than 13.2 MPa.
- The secondary pressure is higher than 8.83 MPa.
- The pressurizer pressure is higher than 16.92 MPa.
- The hot leg temperature is higher than 335ºC.
- The pressurizer water level is lower than 2.85 m, or higher than 11 m.
- The steam generator water level is lower than 8.85 m.
- The reactor power is higher than 108% of the nominal power.

The ECCS used to inject borated water into the primary side consists of:

- the high pressure injection system (HPIS), which injects the water through the hot legs;

[2] Only one steady state verification has been conducted for 10%, 5% and 1% cases, as the same model, boundary and initial conditions are used before the break opening.

- the accumulators, which are passive delivery systems (do not need booster pumps) located in the hot and the cold legs;

- the low pressure injection system (LPIS), also present on the hot leg and the cold leg.

Additional borated water is injected shortly after the SCRAM with a constant mass flow rate. Although this mass flow rate is small compared to the others ECCS injections.

6.1 10% CSA Break

6.1.1 Case Study

Break Localization	Cold Leg (Loop1)
Leak Area	441cm^2
Reactor Power	4240 MW (106% of Nominal Power)
HPIS Set point	Loop 1, Loop 2: Active Loop 3, Loop 4: Disable Primary Pressure < 11.1 MPa
LPIS Set point	Loop 1, Loop 2: Active Loop 3, Loop 4: Disable Primary Pressure < 1.0 MPa
Accumulators Set point	Hot Leg Side: Disable Cold Leg Side: Disable Primary Pressure < 2.5 MPa
Emergency Feedwater Pumps Set point	Active SG Water Level < 5 m
Additional Borated Water Injection (2kg/s)	Loop 1: Deactivated Loop 2, Loop 3, Loop 4: Active
Quench Front Model Set point	Primary Pressure < 3.0 MPa

Table 6.1 10% CSA Break. Case Study

6.1.2 Steady State Verification

Parameter	Steady State Value
Power	
Reactor Power	4240 MW
Power Extracted by the Steam Generators	4263 MW
Pressure	
Primary Pressure	15.72 MPa
Steam Pressure	6.86 MPa
Total Pressure Loss along the Primary Side	0.42 MPa
Pressure Loss along the Reactor Core	0.198 MPa
Temperatures	
Steam Generator Inlet Temperature	327.3 ºC
Steam Generator Outlet Temperature	290.6 ºC
Primary Side Mass Flow Rates	
Mass Flow Rate per Coolant Loop	5068 kg/s
Net Mass Flow Rate through the Reactor Core	20094 kg/s
Level	
Pressurizer Collapsed Level	7.663 m
Steam Generator Collapsed Level	12.2 m
Primary Side Boron Concentration	900 ppm

Table 6.2 10% CSA Break. Steady State Verification

The values of the key parameters obtained show good agreement with the data obtained from [KRAF84].

It is observed that the loop 1 behaves differently than the others, as the pressurizer, which is located in this loop, provides an additional pressure loss. The pressure in the RPV does not stabilize perfectly because the control system used for the sprays and heaters of the pressurizer work on a simplified “all or nothing” basis.

The following figures show the parameters above-mentioned.

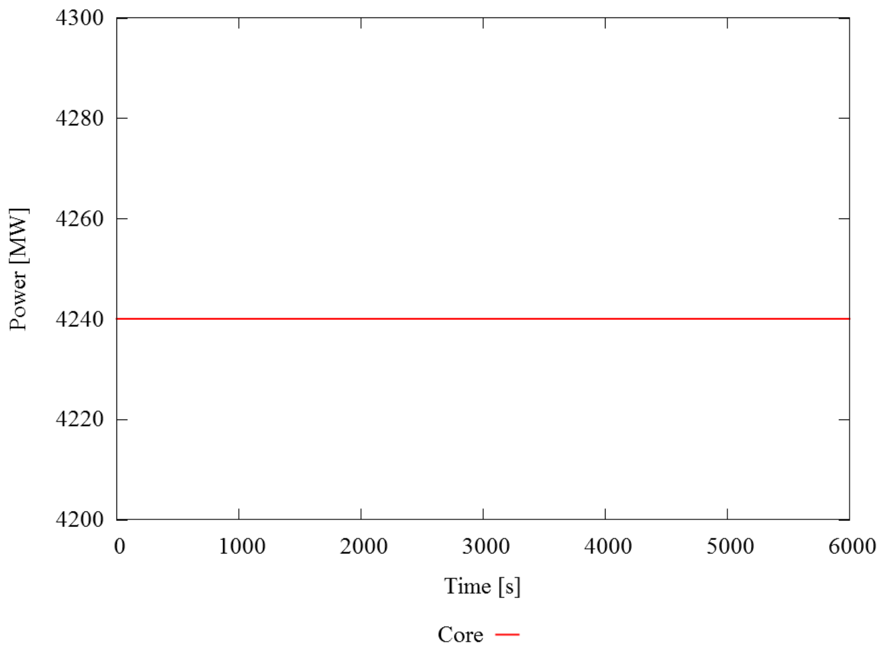

Figure 6.1 Reactor Core Power

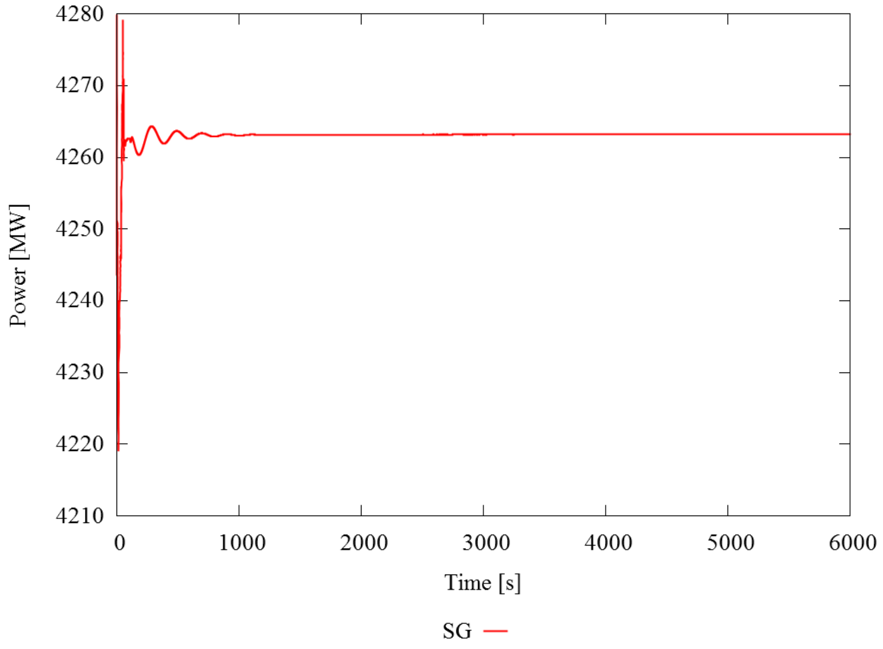

Figure 6.2 Heat Removed from the Primary Side by the Steam Generators

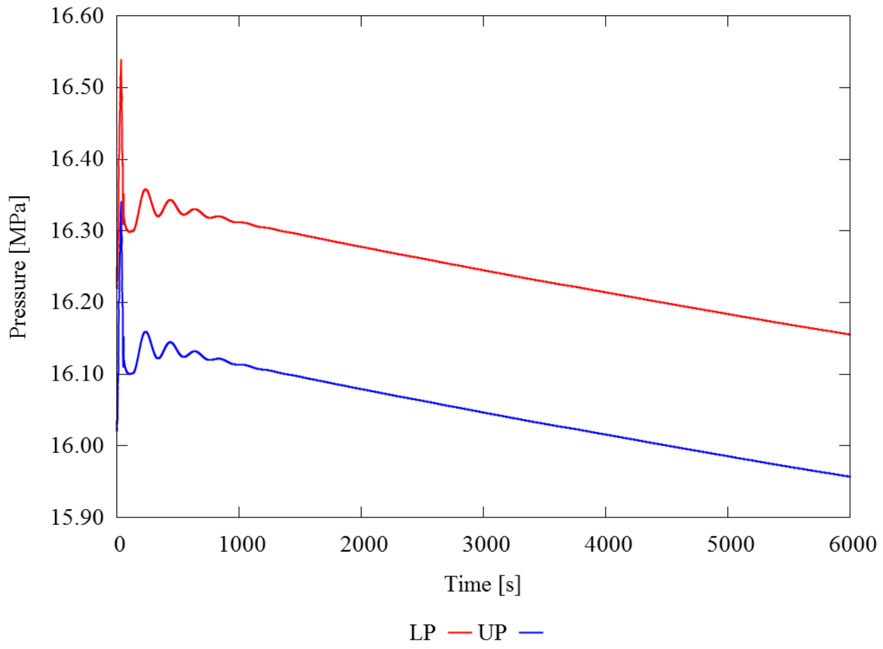

Figure 6.3 Pressure in the Lower Plenum and the Upper Plenum

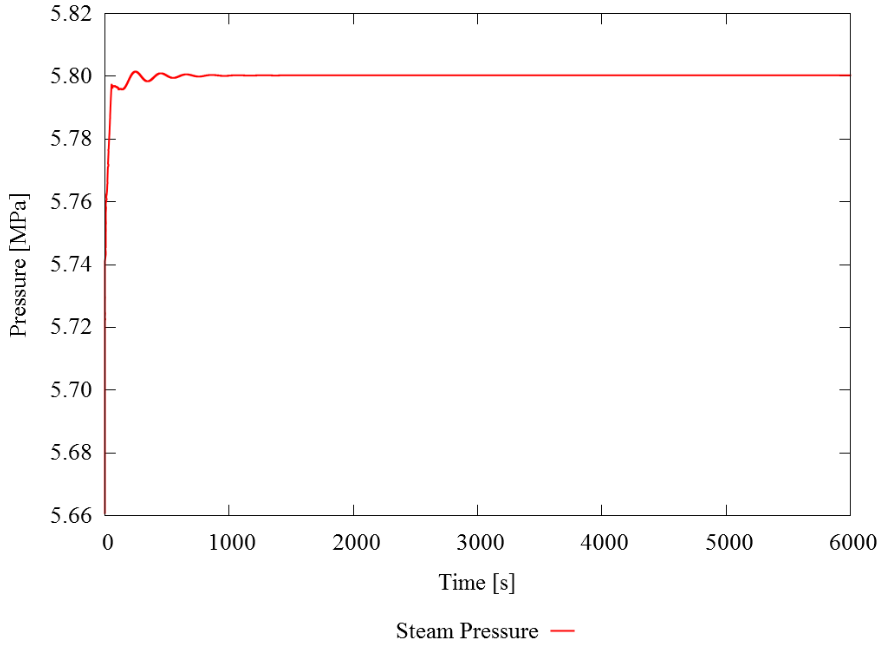

Figure 6.4 Steam Pressure

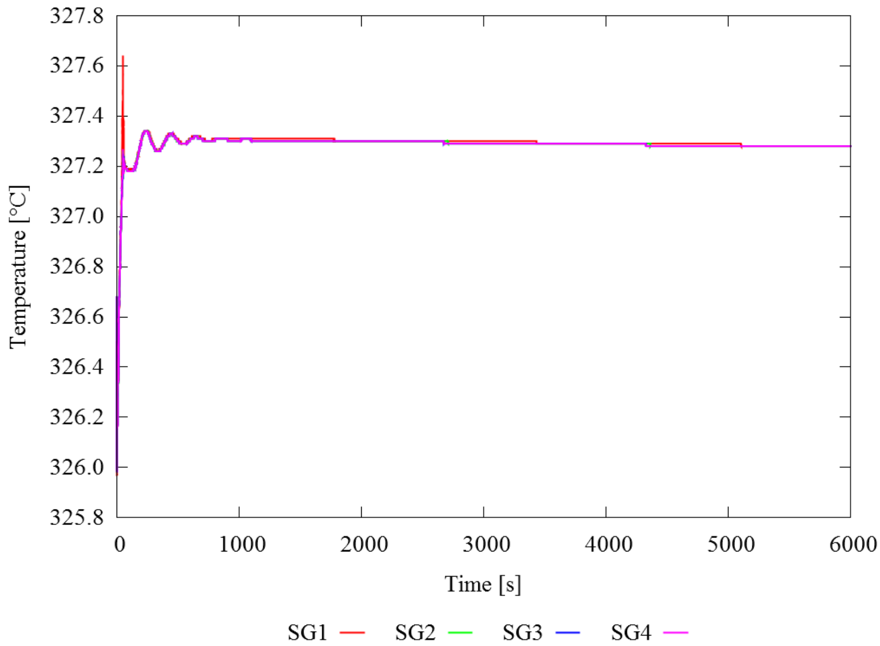

Figure 6.5 Steam Generators Inlet Temperature

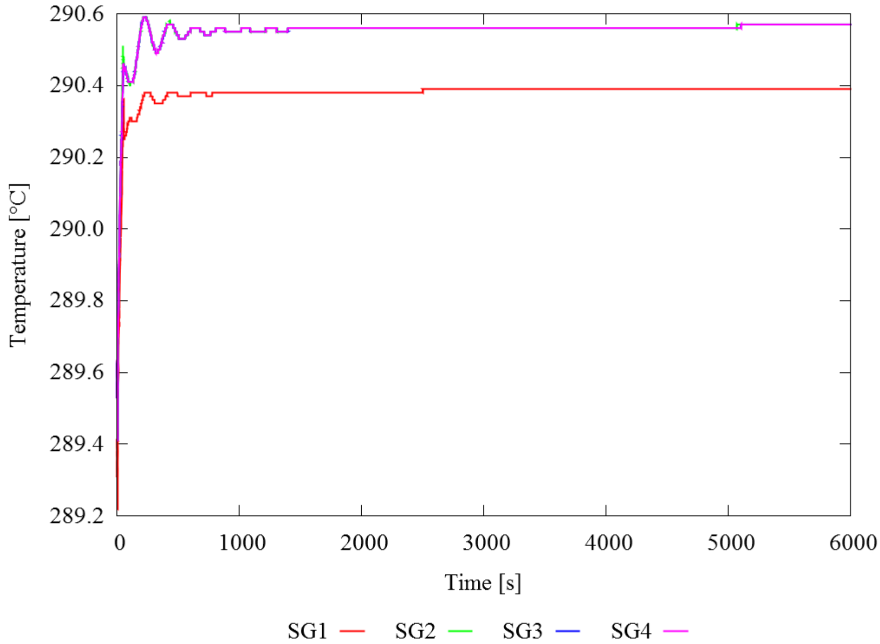

Figure 6.6 Steam Generator Outlet Temperature

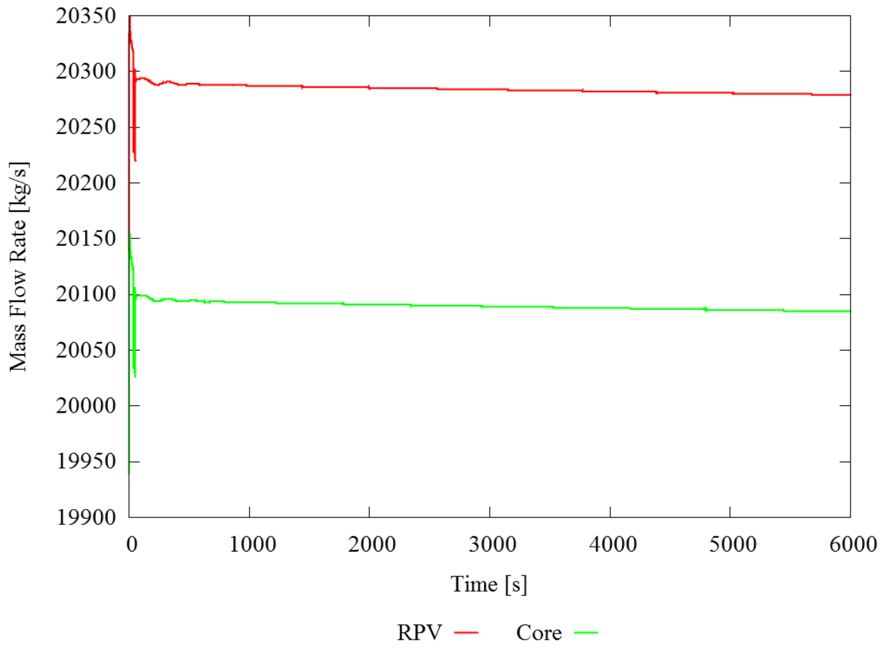

Figure 6.7 Mass Flow Rate through the RPV

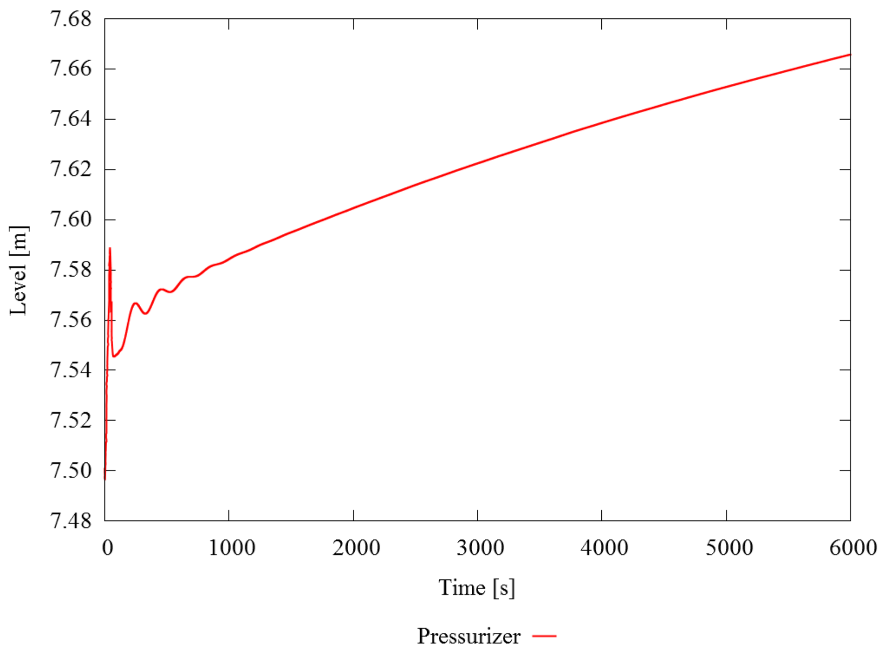

Figure 6.8 Water Level within the Pressurizer

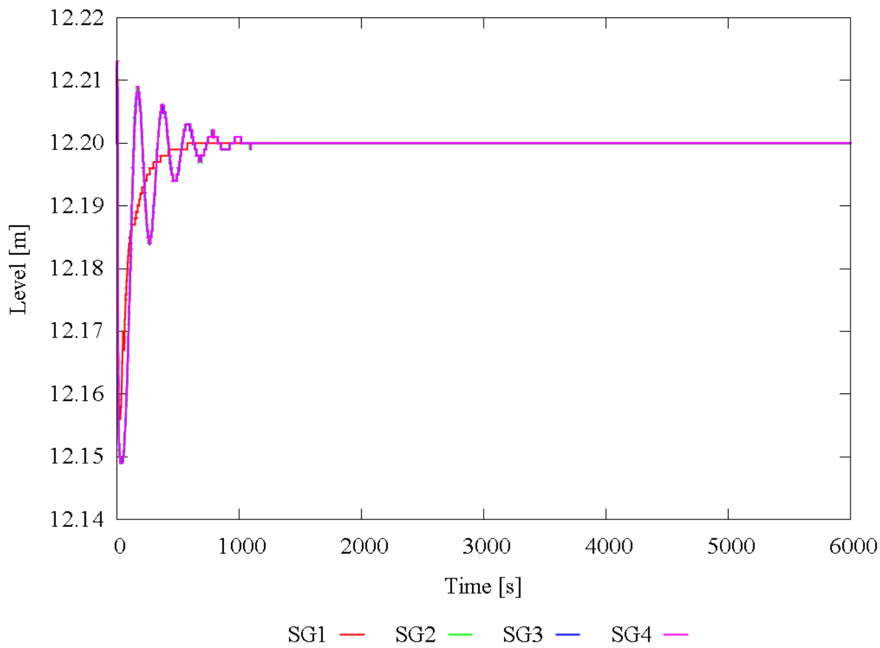

Figure 6.9 Water Level within the Steam Generators

6.1.3 Transient Calculation

Event	Event Time [s]
Leak Opening	0
SCRAM signal (primary pressure low)	3,1
Reactor and Turbine Trip, LOOP	3,1
Steam Generators Trip (100k/h gradient)	3,1
Extra Borated Water Injection Start	3,6
Maximum Pressure Secondary Side Set point Reached	7,2
HPIS Actuation Set point Reached	9,6
Break Flow from Single-Phase to Two-Phase	20,4
HPIS Injection Start	41,1
Primary Pressure lower than Secondary Pressure	67,6
Minimum Core Liquid Coolant Inventory	186,4
Break Flow to Single-Phase Vapour	253,3
LPIS Actuation Set point Reached	392,3
LPIS Injection Start	392,3
HPIS Tanks Drained	4964,9
Plant Under Control[3]	5500,0
End of Simulation	6000,0

Table 6.3 10% CSA Break. Transient Calculation

[3] A nuclear power plant is considered under control when the core is covered, the water level within the steam generators remains constant, the ECCS injects water enough to surpass the leak mass flow rate and the cladding temperature is equal to the water saturation temperature

6.1.4 Analysis of the results

At the moment of the break, the pressure in the primary side drops rapidly and reaches the SCRAM set point. The reactor core, the turbine and the steam generators are tripped. The core power decreases rapidly (Figure 6.10) due to the emergency injection of the control rods into the core which stop the chain reaction. However, the core power is not zero exactly after the shutdown because of the decay heat (about 7% of the power before SCRAM immediately after the shutdown of the reactor).

As the SGs are isolated, the pressure within them side increases rapidly until the maximum allowed value (8.83 MPa) is reached and starts decreasing by the action of the SGs relief valves (Figure 6.11). Meanwhile, the primary pressure keeps decreasing (Figure 6.12) and the HPIS actuation set point is reached after 9 s. The injection is however further delayed because the emergency diesel motors, which have to be started due to the LOOP, are not available yet. (Figure 6.13).

Because the pressure in the primary side continues falling, the saturation point is reached 20 s after the leak opening (Figure 6.14). Steam forms leading to a partial core uncovery (Figure 6.15). The temperature and the pressure remain constant during the phase change, this is why pressure keeps dropping but more smoothly. The minimum primary side pressure observed is 0.5 MPa. Eventually it gets close to the atmospheric pressure.

The primary pressure remains higher than the secondary pressure during the early stages of the transient (Figure 6.16), which means that the secondary side cools down the primary. However, 67 s after the break opened, the secondary pressure (controlled by the 100K/h shut-down curve) drops below the primary pressure and starts to heat up the primary side. This prevents the reflux condensation of the vapour located in the primary side.

The water contained in the U-tubes of the steam generators flows through the leak due to the large pressure difference with the atmosphere (Figure 6.17). The water level within the steam generator decreases rapidly (Figure 6.18). The space left by the liquid in the SGs, is occupied by the steam from the core, which flows through the SG U-tubes (Figure 6.19) and then through the loop seals (Figure 6.20). The loop seals of loops 1 and 2 clear at 170 s and provide cooling water for the core. This leads to a drop of the peak cladding temperature (Figure 6.21).

The break flow switches to two-phase flow. This two-phase flow stabilizes the leak mass flow for 105 s. The break is completely uncovered 230 s after the break opening (Figure 6.22). This situation remains for 400 s, when the leak mass flow rate increases again due to the recovery of the coolant level in the core following the injection of important high amounts of borated water. The observed fluctuations that appear after 600 s are caused of the vapour production and condensation due to the coolant injection by the ECCS.

As the accumulators are blocked, the injection mass flow is carried out through the HPIS and the LPIS (Figure 6.23). The LPIS helps substantially to re-establish the reactor core, as the high pressure injection, due to the relatively high primary pressure, does not exceed the leak mass flow and provides no massive cooling of the core. Actually, only a small amount of ECCS mass flow rate from the SCOOP injection reaches the core (Figure 6.24).

The situation is under control when the core is quenched and the injected water mass flow rate exceeds the leak mass flow rate. In the present, case, the situation is back under control 400 s after the break opening. The core reactor suffers no severe damages during this transient, even though a rapid temperature increase of several hundred degrees is observed. The PCT remains the whole time below 1200 ºC (melting point of the cladding material).

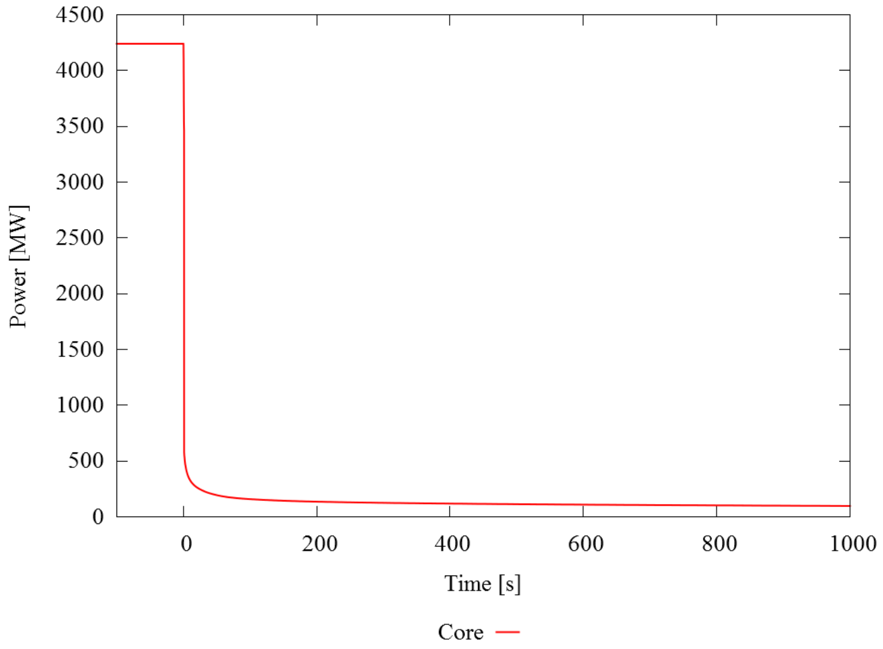

Figure 6.10 Reactor Core Power

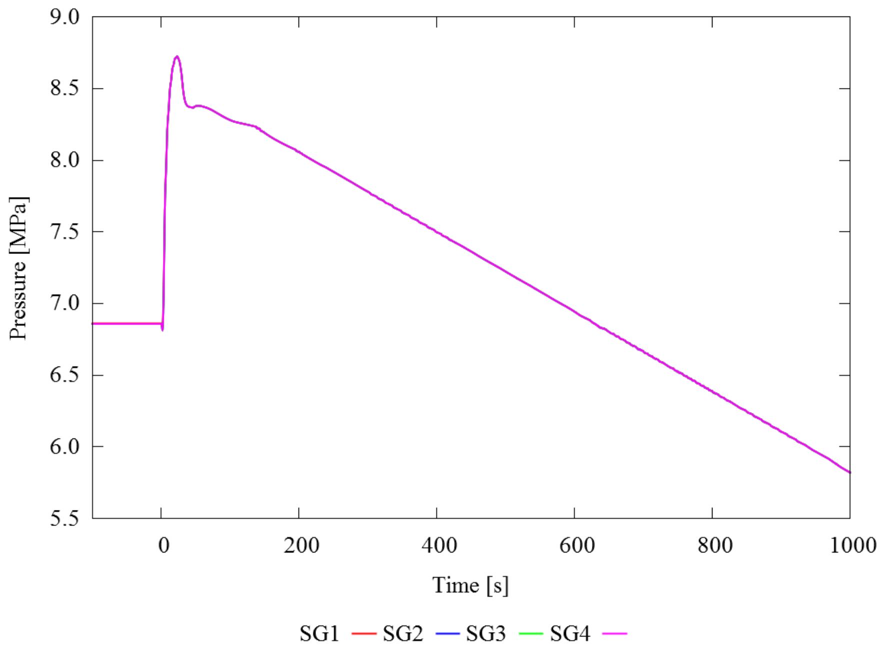

Figure 6.11 Pressure Secondary Side

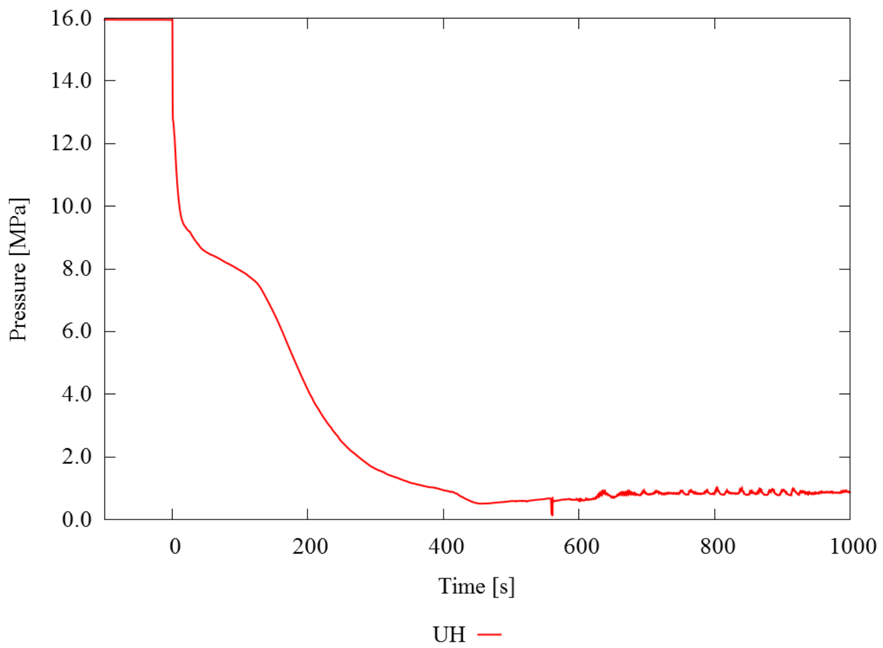

Figure 6.12 Pressure Upper Head of the RPV

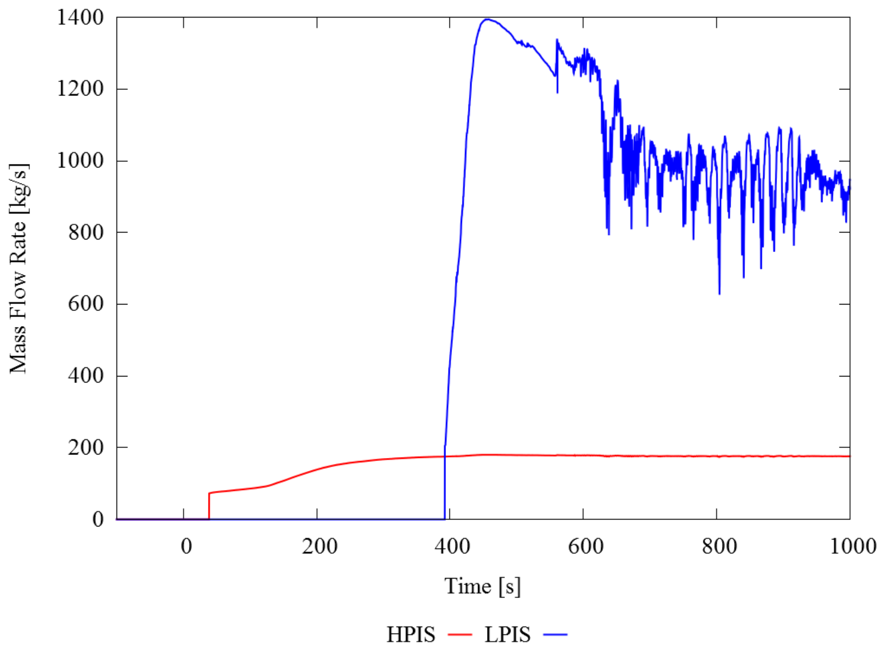

Figure 6.13 HPIS and LPIS Mass Flow Rate

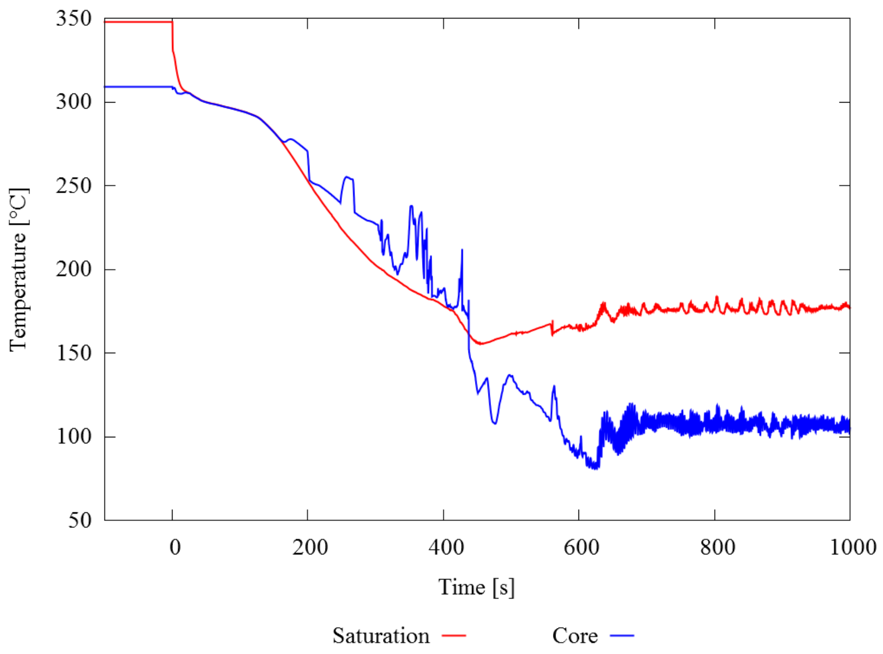

Figure 6.14 Comparison Mean Coolant Temperature and Saturation Temperature within PV-COR3

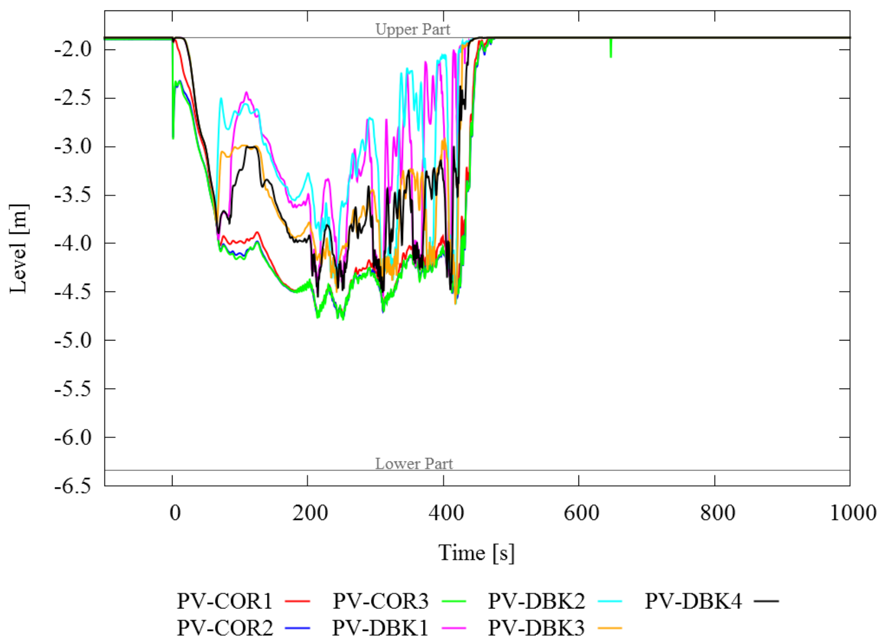

Figure 6.15 Comparison of Water Collapsed Level within the Core

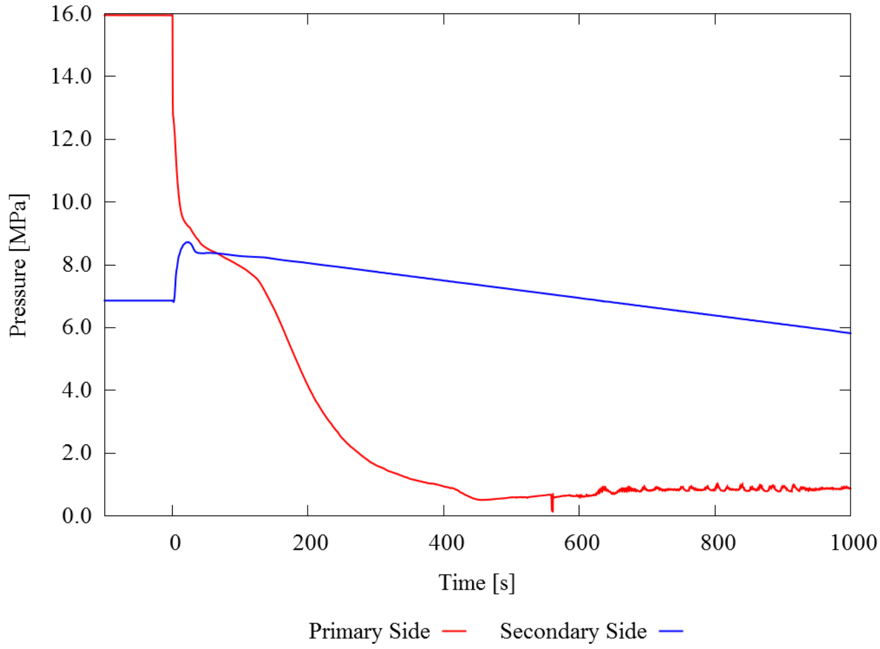

Figure 6.16 Pressure Primary and Secondary Side

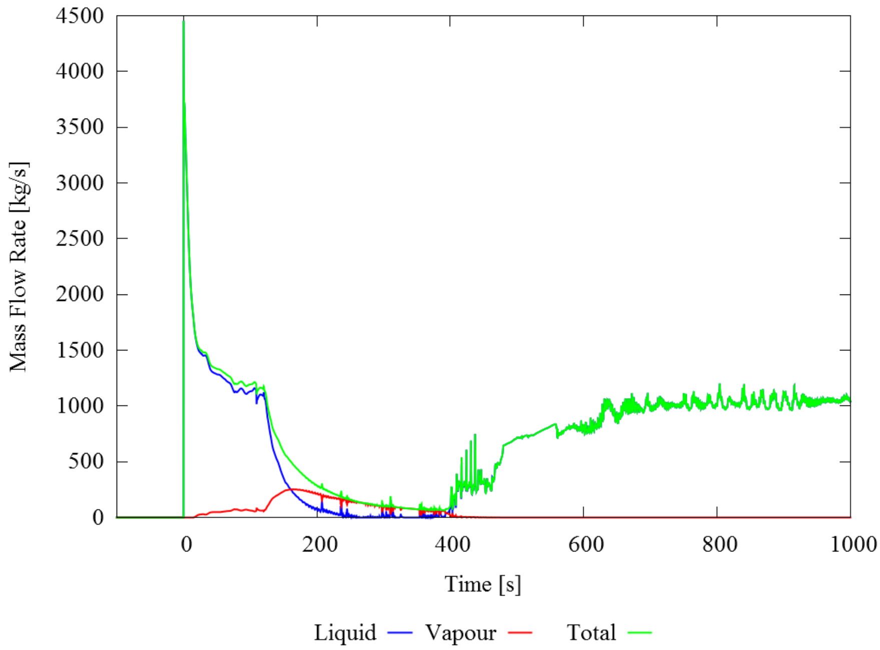

Figure 6.17 Comparison Mass Flow Rate through the Leak

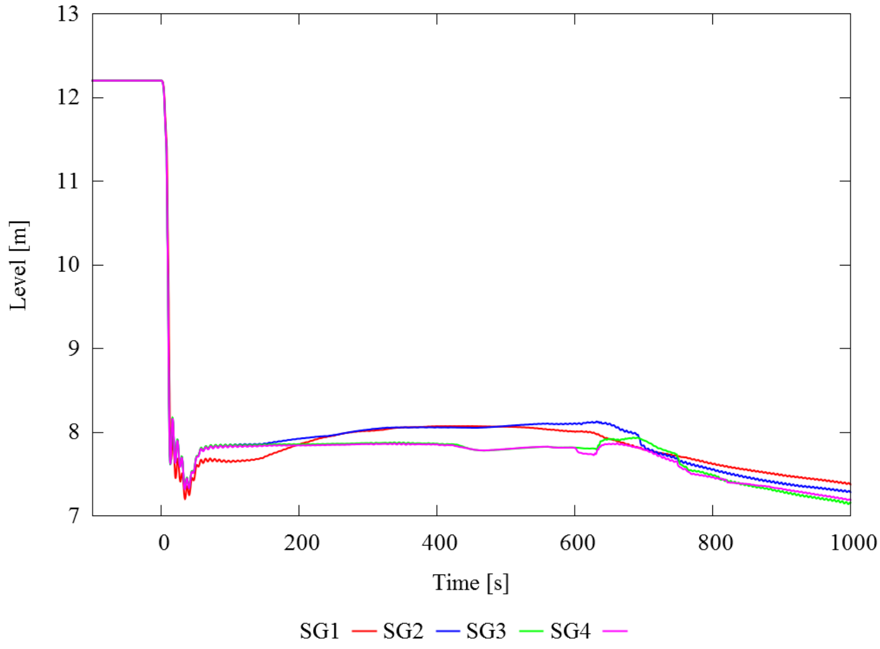

Figure 6.18 Water Level within the Steam Generators

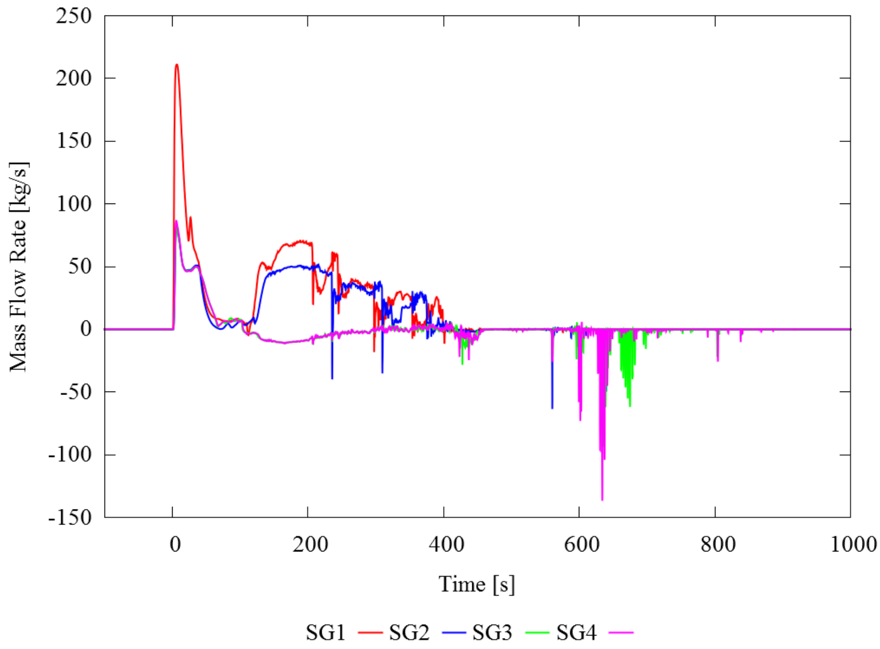

Figure 6.19 Vapour Mass Flow Rate within the U-Tubes of Steam Generators

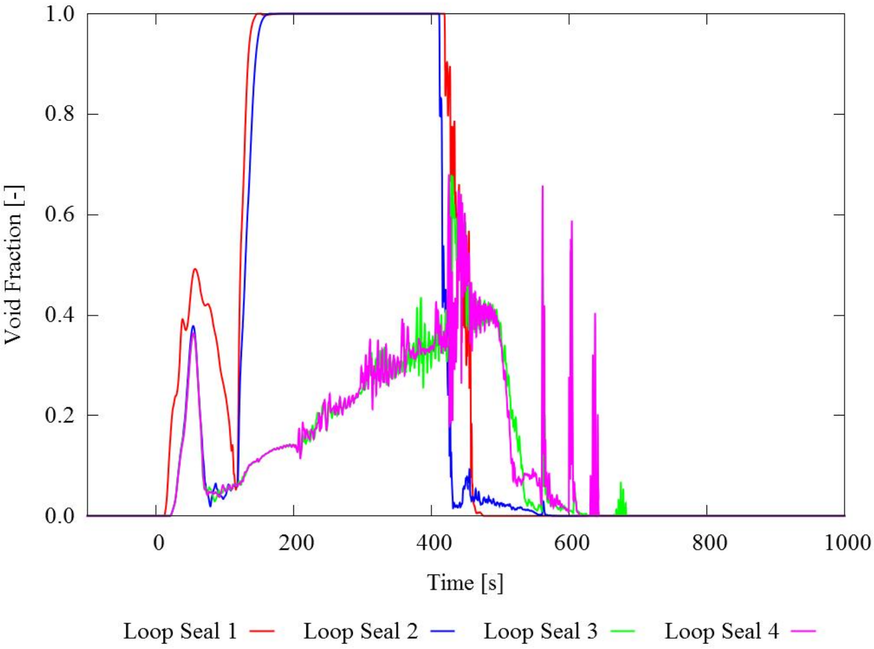

Figure 6.20 Void Fraction at the Loop Seals

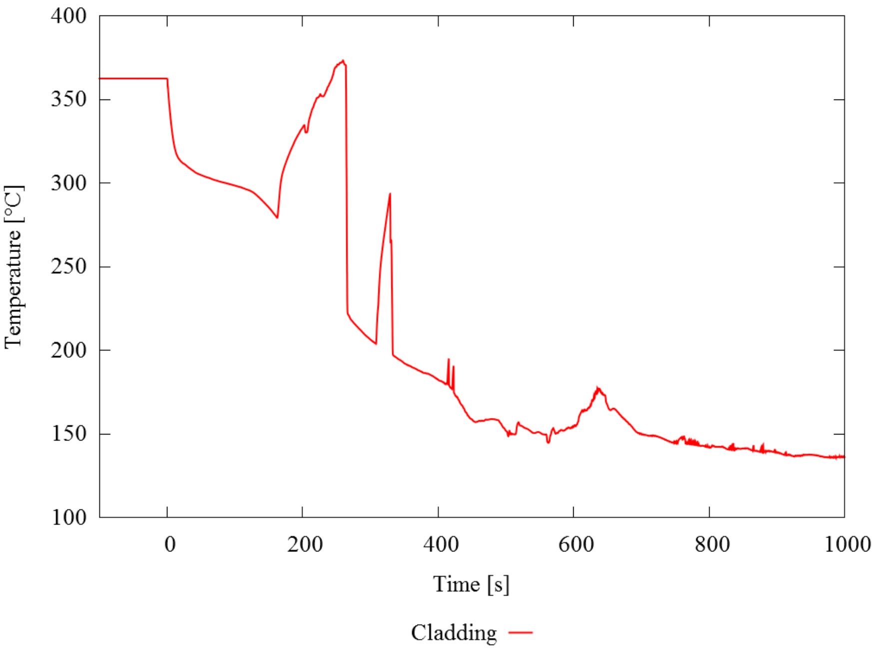

Figure 6.21 Peak Cladding Temperature

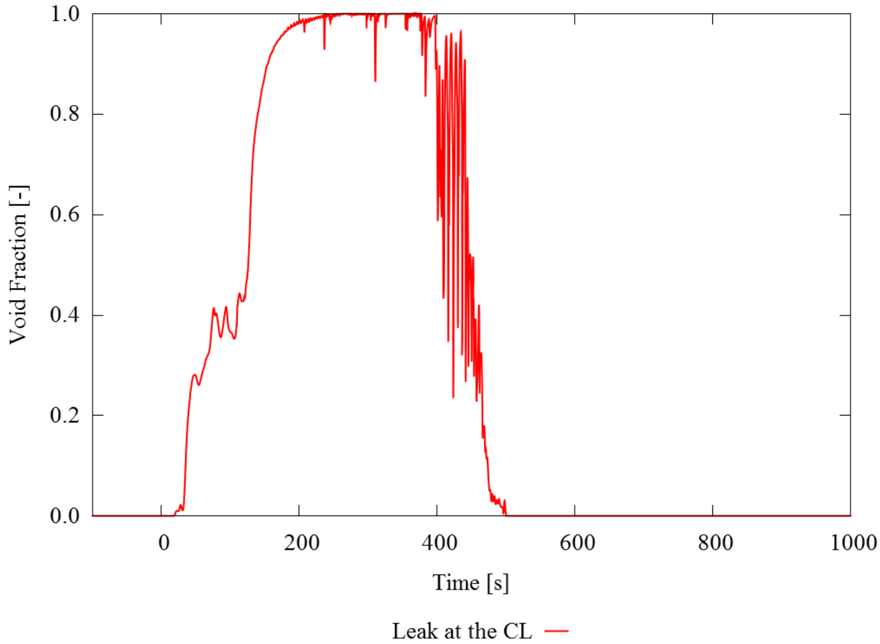

Figure 6.22 Void Fraction at the Leak in the CL

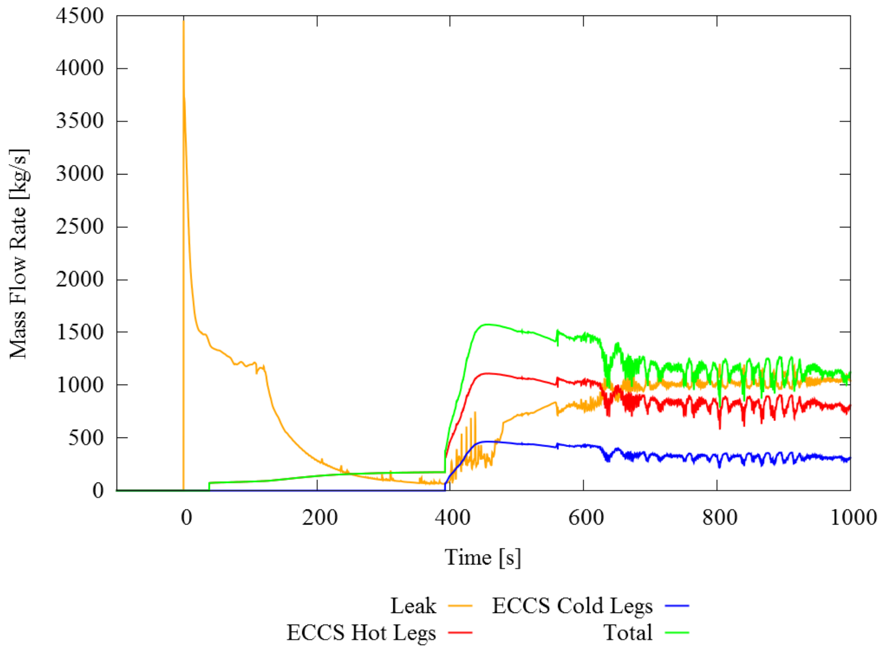

Figure 6.23 Comparison Mass Flow Rate of the Leak and Injections

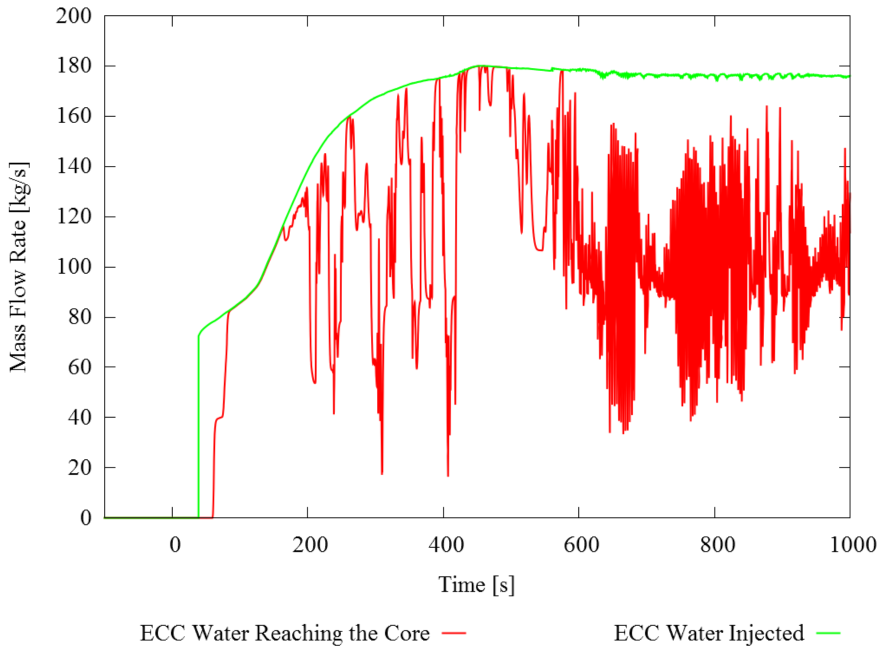

Figure 6.24 Hot Leg SCOOP Injection Mass Flow Rate

6.1.5 Influence of the Leak Orientation and Availability of HPIS

The objective is to investigate the influence of different modelling options. In this sense, the effects of the leak orientation and of the availability of the HPIS are studied.

The different orientations taken into account are:

- Vertical upward: the break is located at the top of the cold leg.
- Vertical downward: the break is located at the bottom of the cold leg.
- Horizontal: the break is located at the middle elevation of the cold leg.

When the HPIS are considered available, only the ones of Loop 1 and 2 are active. If they are not available, none of them are.

The following plots show the results obtained. Although the simulation time was also 6000 s after the break, a shorter time frame has been chosen to gain a better view. The break occurs at 0 s.

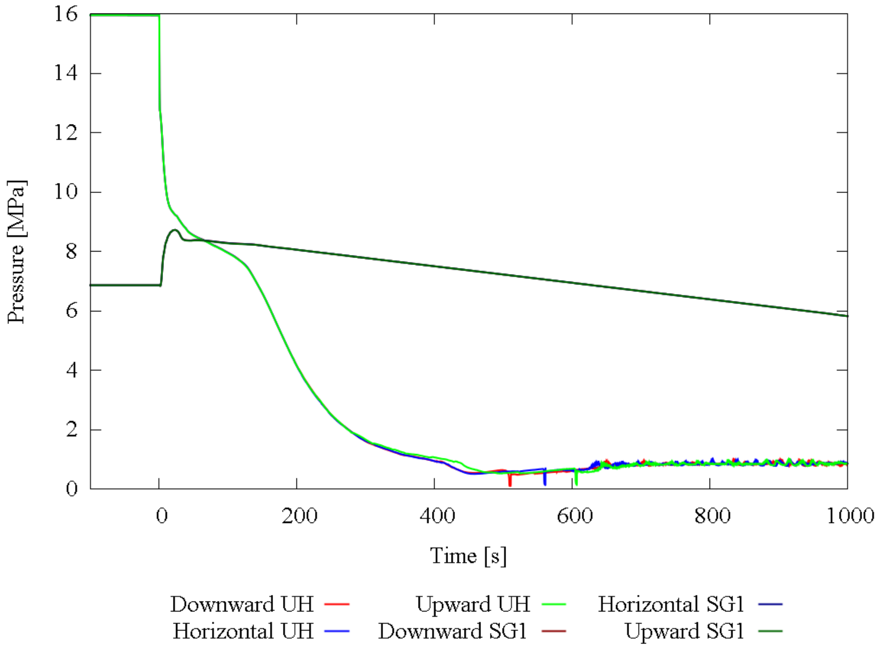

Figure 6.25 Comparison Primary and Secondary Side Pressures

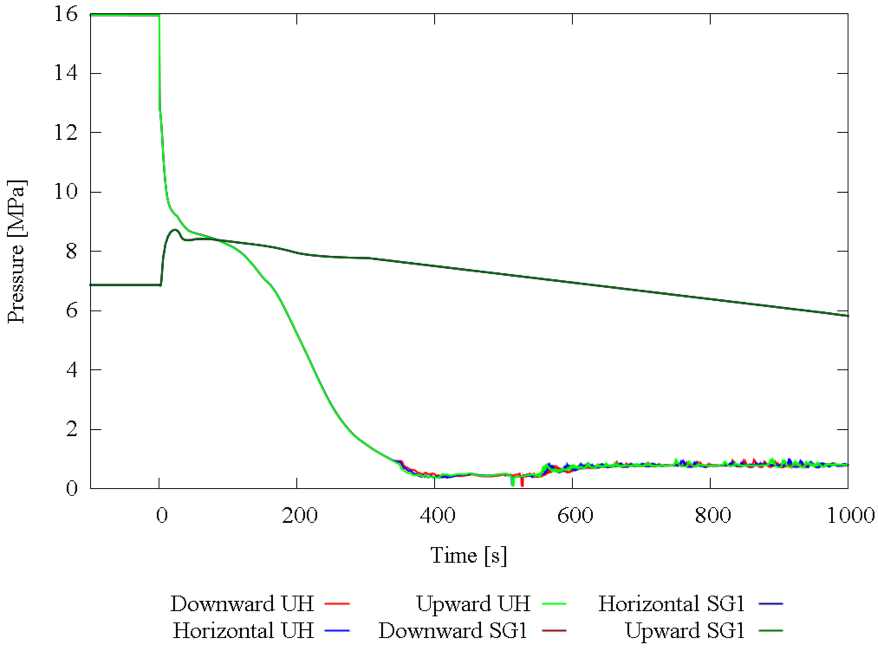

Figure 6.26 Comparison Primary and Secondary Side Pressures with HPIS Deactivated

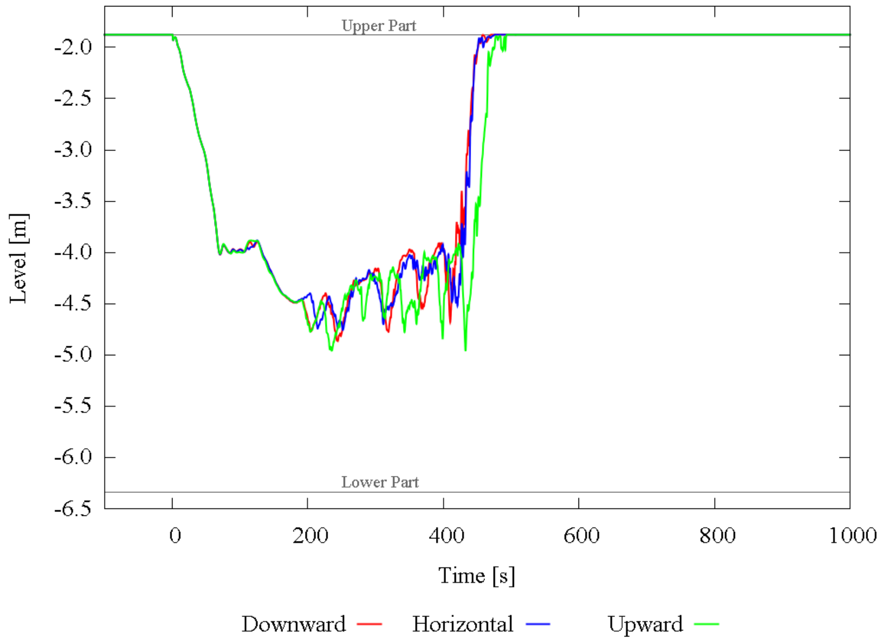

Figure 6.27 Comparison Water Collapsed Level within PV-COR1

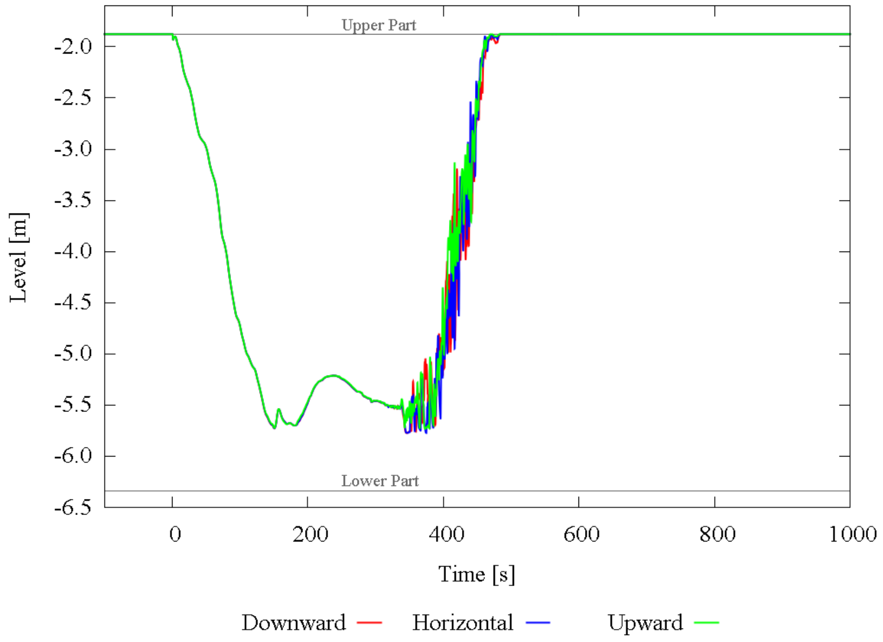

Figure 6.28 Comparison Water Collapsed Level within PV-COR1 with HPIS Deactivated

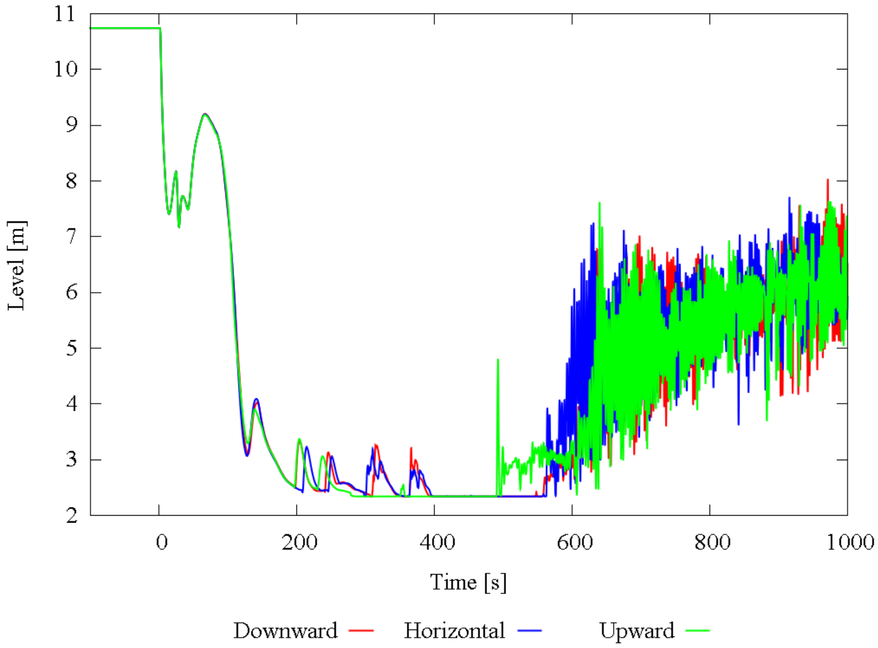

Figure 6.29 Comparison Water Collapsed Level in the U-tubes of Loop 1

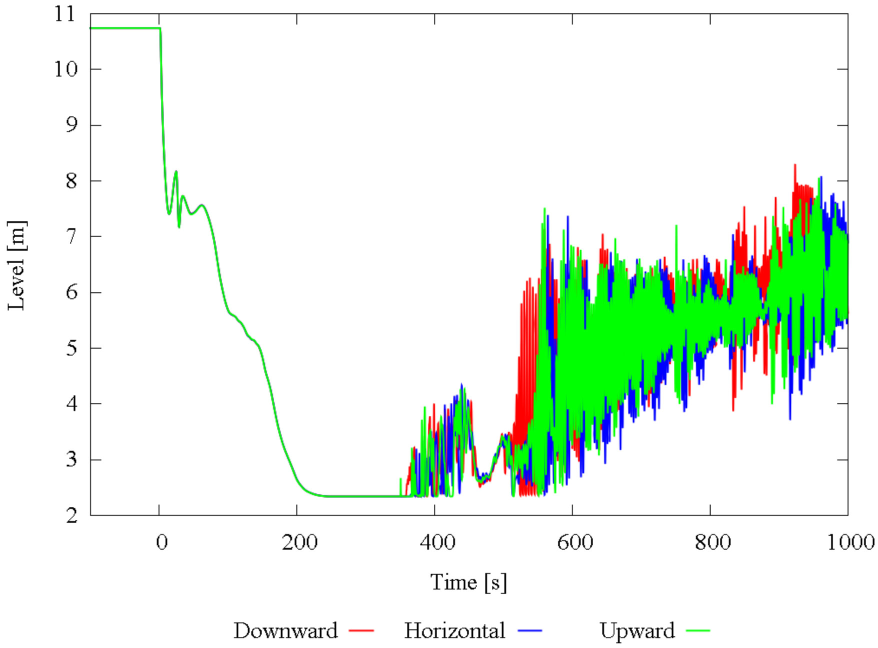

Figure 6.30 Comparison Water Collapsed Level in the U-tubes of Loop 1 with HPIS Deactivated

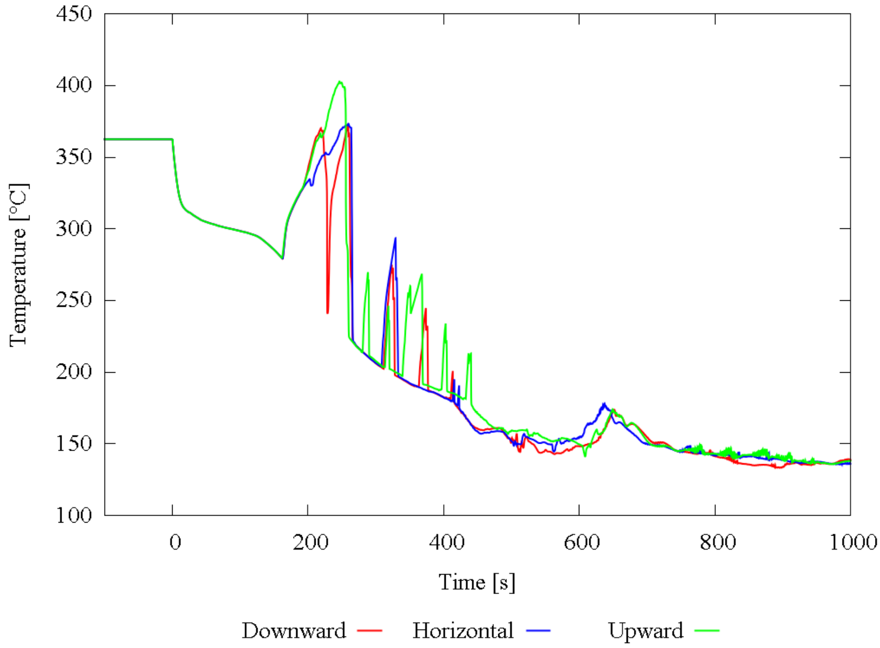

Figure 6.31 Comparison Peak Cladding Temperature

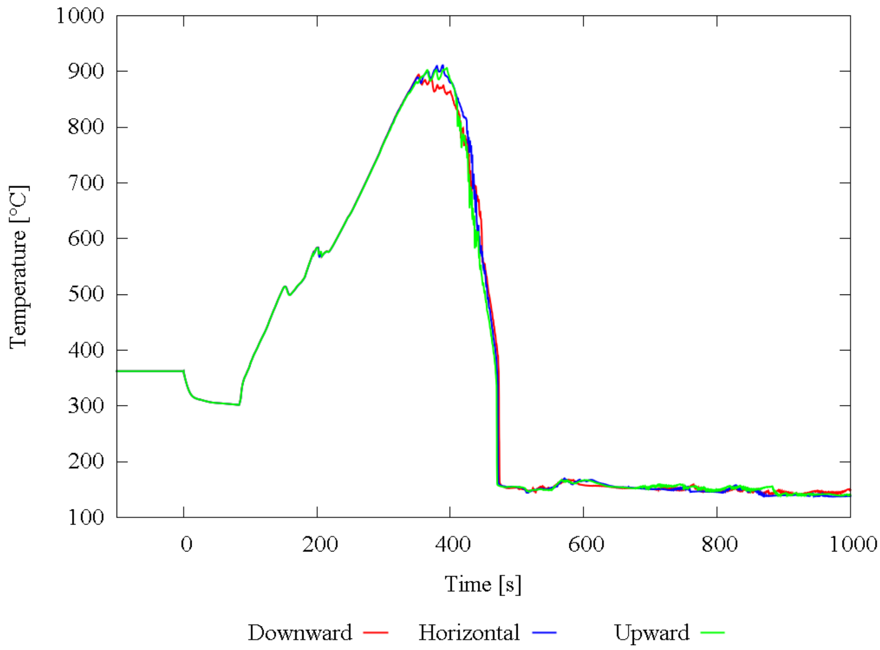

Figure 6.32 Comparison Peak Cladding Temperature with HPIS Deactivated

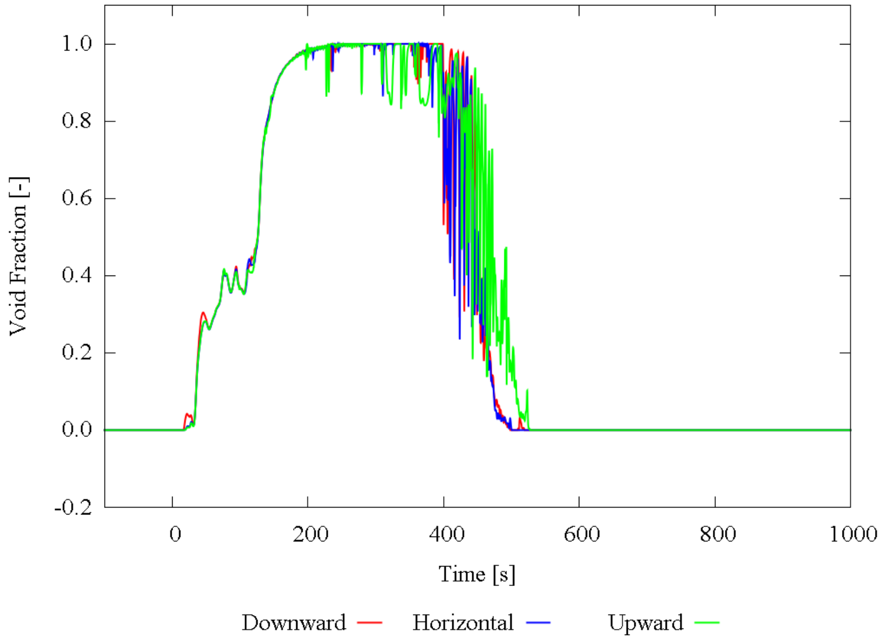

Figure 6.33 Comparison Void Fraction at the Leak in the CL

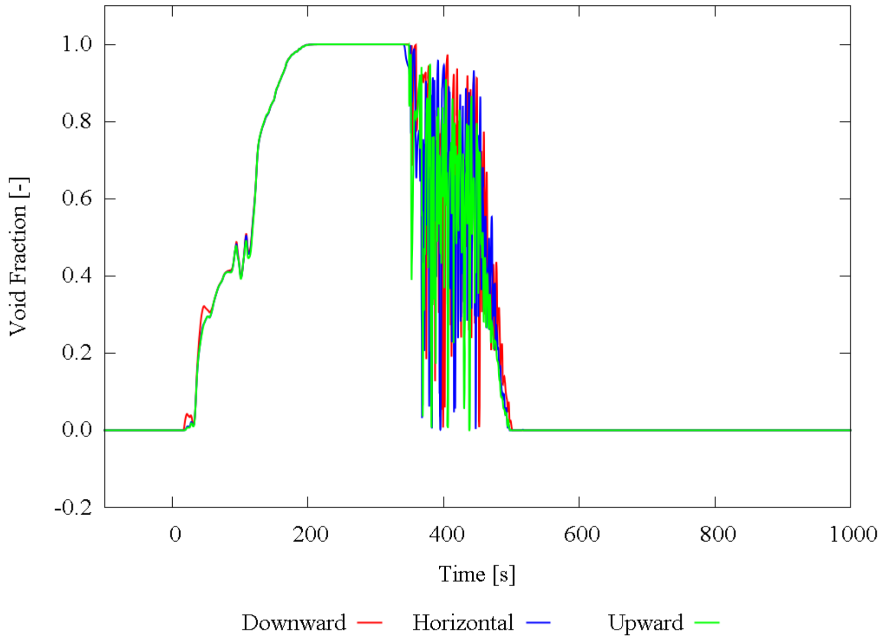

Figure 6.34 Comparison Void Fraction at the Leak in the CL with HPIS Deactivated

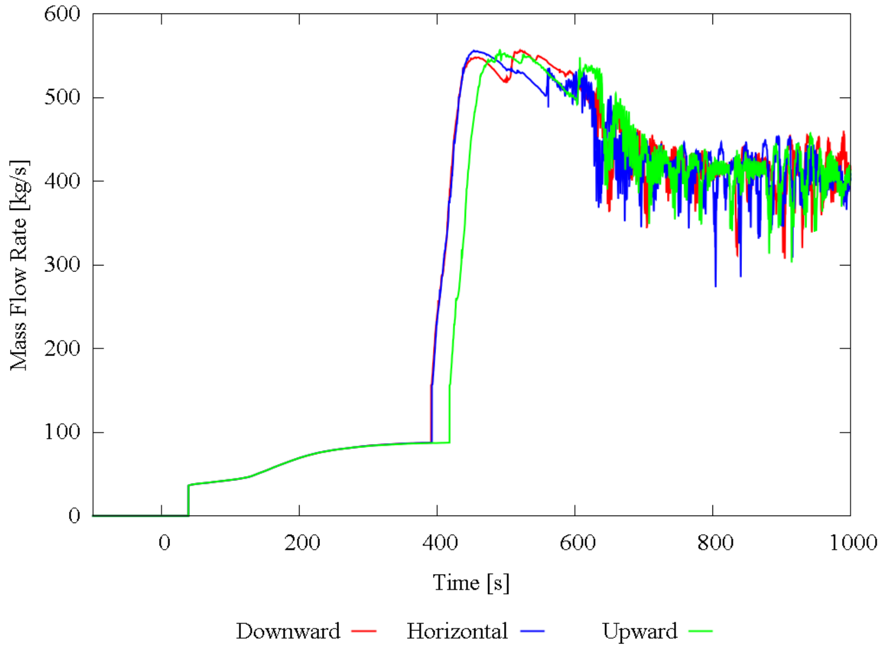

Figure 6.35 Comparison ECCS Injection Mass Flow Rate Loop 1 Hot Side

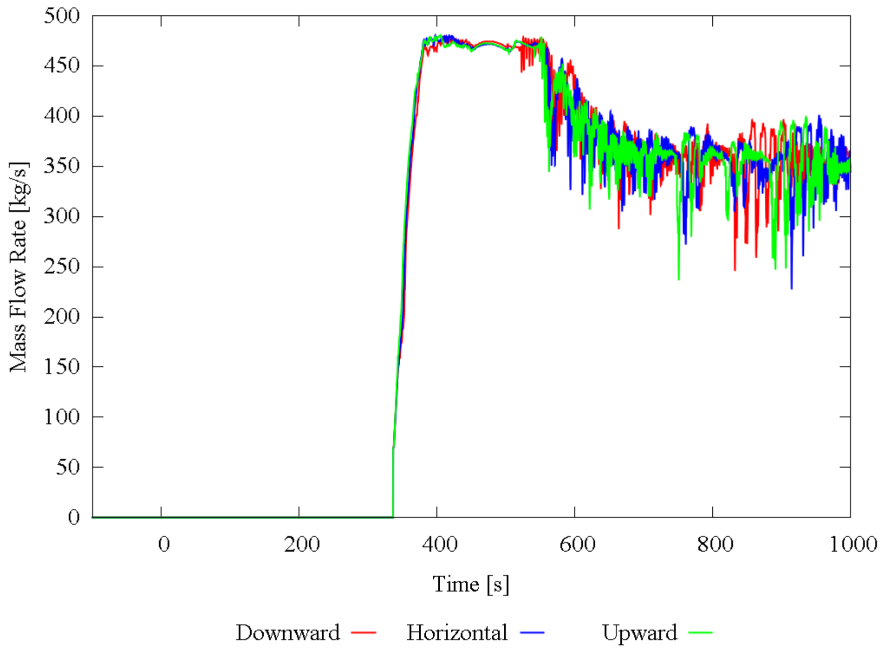

Figure 6.36 Comparison ECCS Injection Mass Flow Rate Loop 1 Hot Side with HPIS Deactivated

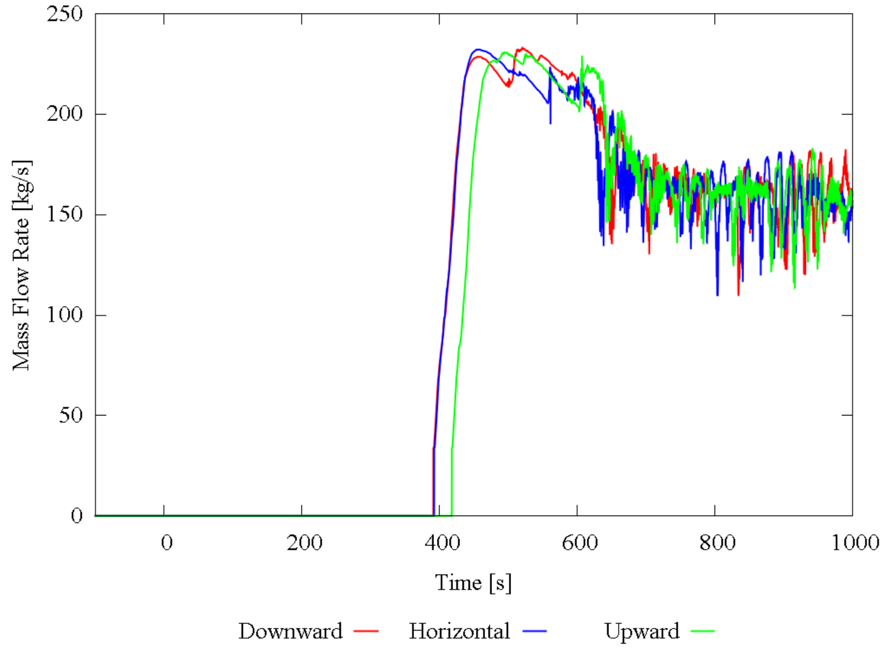

Figure 6.37 Comparison ECCS Injection Mass Flow Rate Loop 1 Cold Side

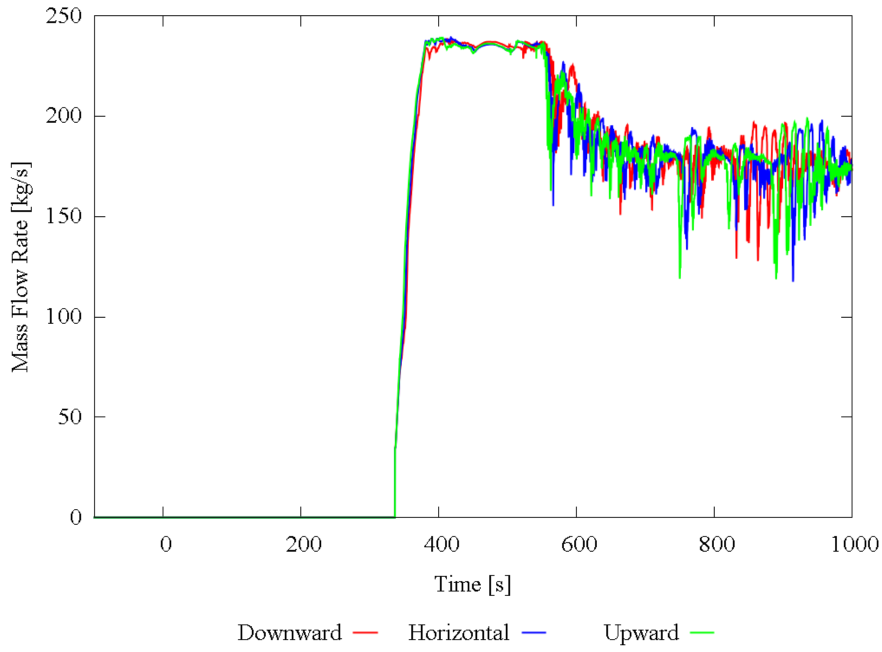

Figure 6.38 Comparison ECCS Injection Mass Flow Rate Loop 1 Cold Side with HPIS Deactivated

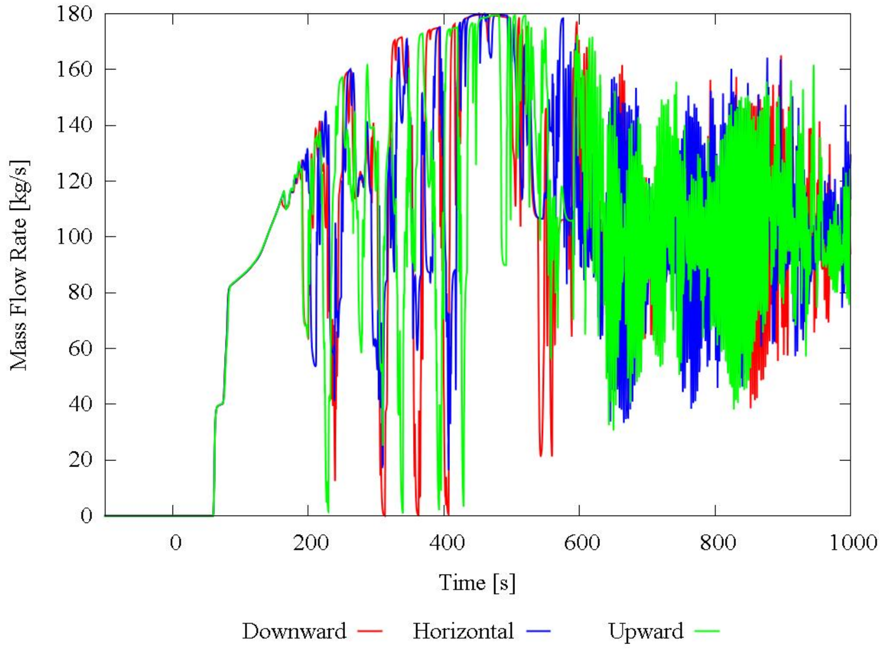

Figure 6.39 Comparison Hot Leg SCOOP Injection Mass Flow Rate

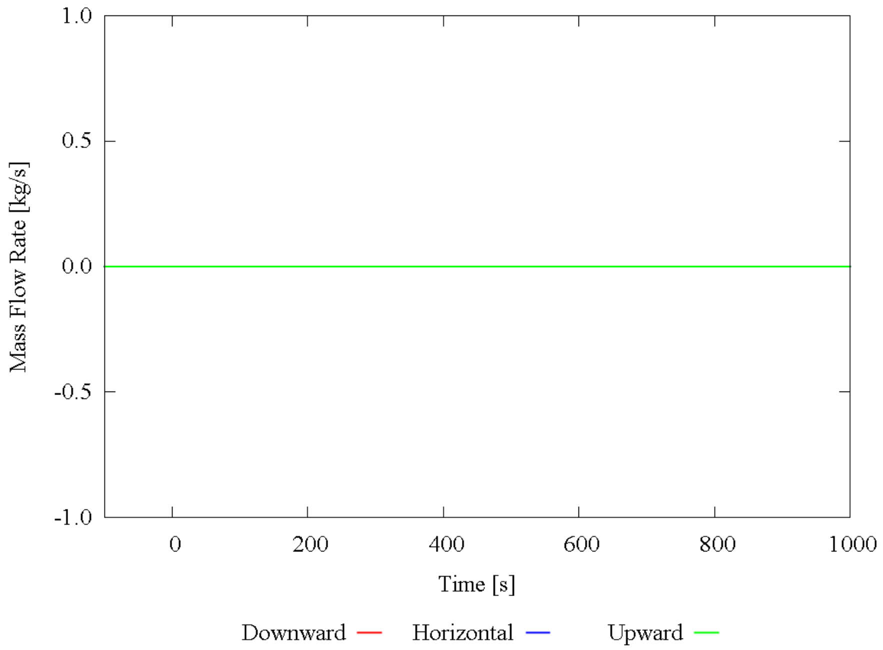

Figure 6.40 Comparison Hot Leg SCOOP Injection Mass Flow Rate with HPIS Deactivated

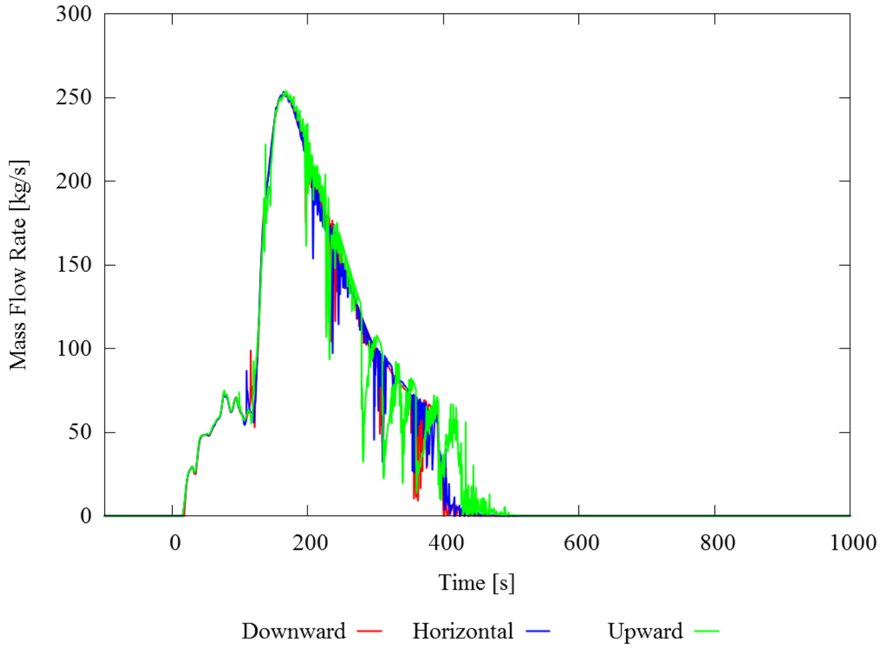

Figure 6.41 Comparison Vapour Mass Flow Rate through the Leak

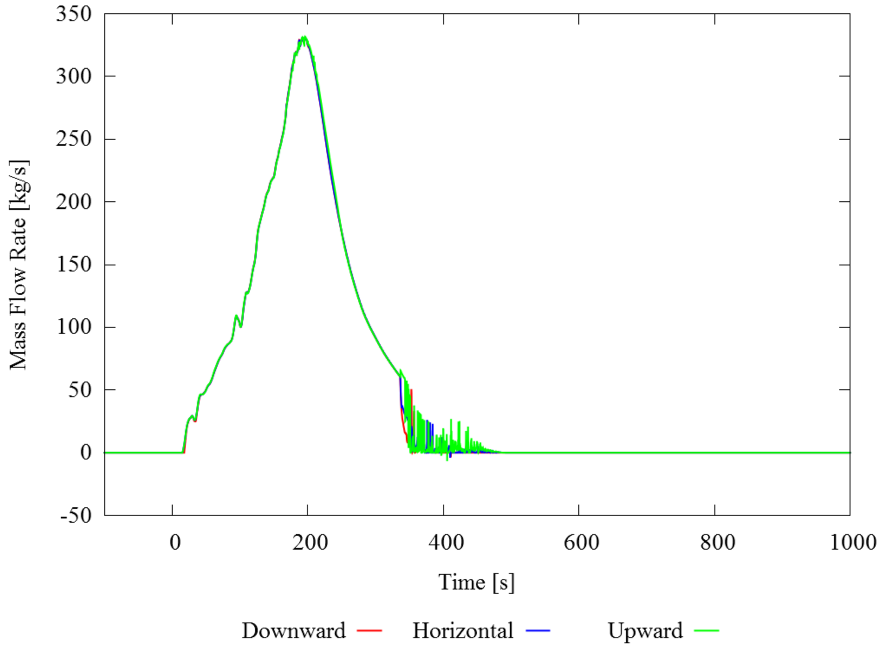

Figure 6.42 Comparison Vapour Mass Flow Rate through the Leak with HPIS Deactivated

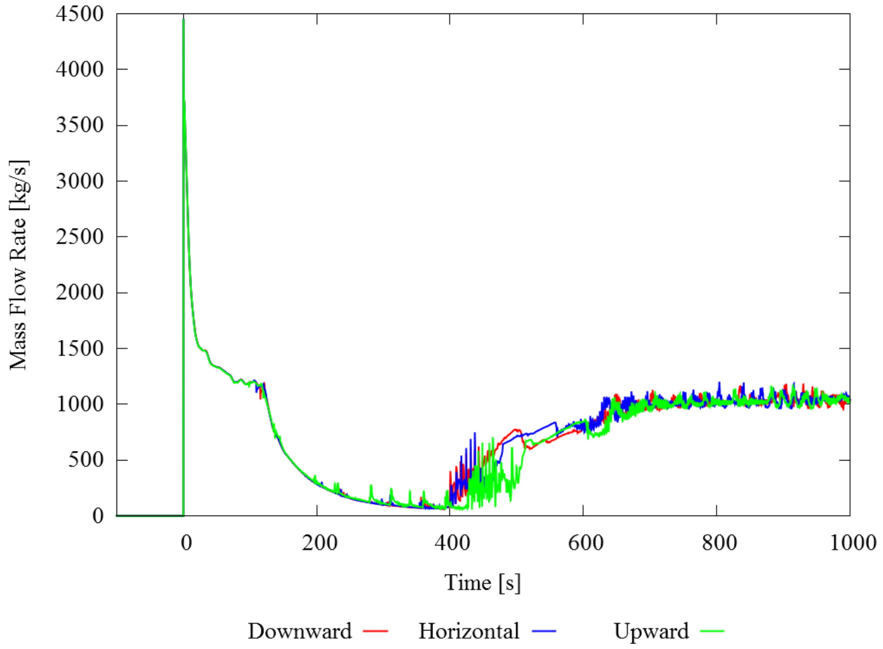

Figure 6.43   Comparison Total Mass Flow Rate through the Leak

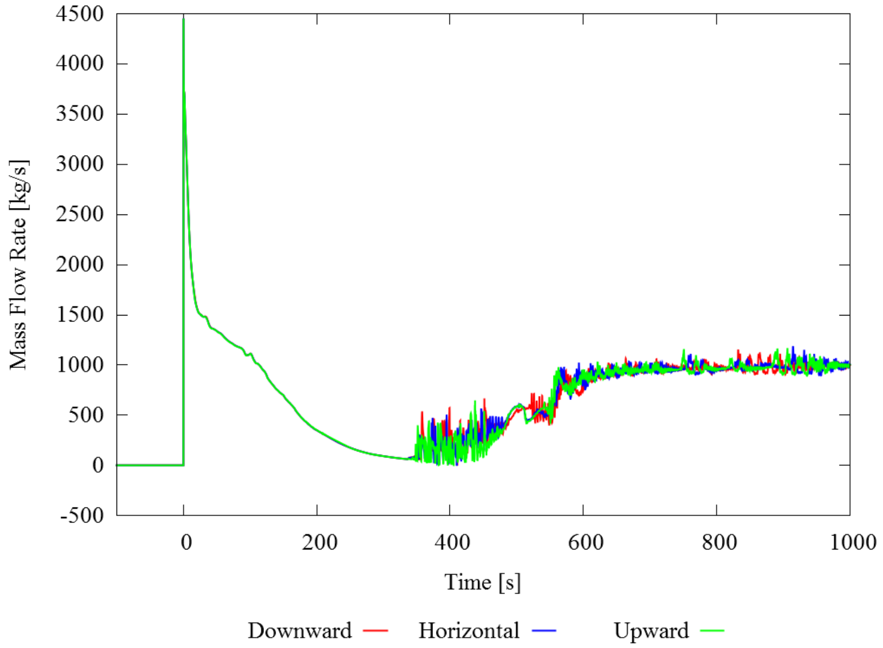

Figure 6.44   Comparison Total Mass Flow Rate through the Leak with HPIS Deactivated

6.1.6 Conclusions

The reactor core is not completely uncovered at any time. Nevertheless, if the HPIS is deactivated, the water collapsed level is much lower (Figure 6.27 and Figure 6.28). It also takes longer time to refill the upper part of the core if the break is located at the top of the cold leg.

If the HPIS is active, the steam remains a longer period of time in the U-tubes of the steam generator (Figure 6.29 and Figure 6.30). This happens also for the leak. The leak remains uncovered for 210 s if the HPIS works, while if the HIPS is deactivated, the single vapour phase flow in the break last for 170 s (Figure 6.33 and Figure 6.34). If the leak has a vertical upward orientation, more steam is vented into the containment.

The maximum PCT is significantly higher if the HPIS is deactivated (Figure 6.31 and Figure 6.32). If the HPIS is active, the difference between the maximum values within the different leak orientations is higher.

The ECCS starts to inject water through the hot leg via the HPIS earlier if the break is located in the middle or at the bottom of the cold leg (Figure 6.35 and Figure 6.37). If this is not active, the borated water injection starts later, when the set point of LPIS is reached (Figure 6.36 and Figure 6.38). Although, this set point is achieved later if the HPIS is active.

The steam mass flow rate through the leak is higher if the HPIS is deactivated (Figure 6.41 and 6.42); while the total leak mass flow rate is almost equal if the HPIS is available or if it is not (Figure 6.43 and Figure 6.44).

For this size of break, if the HPIS is available, the worst case is the vertical upward orientation, since it takes longer to recover the core and the peak cladding temperature reaches the highest value. If the HPIS is not available, there are no noticeable differences between the leak orientations.

6.2 5% CSA Break

6.2.1 Case Study

Break Localization	Cold Leg (Loop 1)
Leak Area	220cm^2
Reactor Power	4240 MW (106% of Nominal Power)
HPIS Set point	Loop 1, Loop 2: Active Loop 3, Loop 4: Disable Primary Pressure < 11.1 MPa
LPIS Set point	Loop 1, Loop 2: Active Loop 3, Loop 4: Disable Primary Pressure < 1.0 MPa
Accumulators Set point	Hot Leg Side: Active Cold Leg Side: Active Primary Pressure < 2.5 MPa
Emergency Feedwater Pumps Set point	Active SG Water Level < 5 m
Additional Borated Water Injection (2kg/s)	Loop 1: Deactivated Loop 2, Loop 3, Loop 4: Active
Quench Front Model Set point	Primary Pressure < 3.0 MPa

Table 6.4 5% CSA Break. Case Study

6.2.2 Transient Calculation

Event	Event Time [s]
Leak Opening	0
SCRAM signal (primary pressure low)	4,2
Reactor and Turbine Trip, LOOP	4,2
Steam Generators Trip (100k/h gradient)	4,2
Extra Borated Water Injection Start	4,8
Maximum Pressure Secondary Side Set point Reached	8,7
HPIS Actuation Set point Reached	13,1
Break Flow from Single-Phase to Two-Phase	36,3
HPIS Injection Start	42,3
Primary Pressure lower than Secondary Pressure	171,3
Minimum Core Liquid Coolant Inventory	289,6
Accumulators Set point Reached	503,2
Accumulators Injection Start	503,2
LPIS Actuation Set point Reached	784,9
LPIS Injection Start	784,9
Emergency Feedwater Injection Start	4705,1
HPIS Tanks Drained	5116,3
Plant Under Control	5500,0
End of Simulation	6000,0

Table 6.5 5% CSA Break. Transient Calculation

6.2.3 Analysis of the results

The primary pressure reaches the SCRAM set point 4 s after the leak opening, which leads in the emergency shutdown of the core (Figure 6.45). The secondary pressure increases rapidly at this point but 4 s later it starts decreasing by action of the SGs relief valves (Figure 6.46).

As the primary pressure keeps dropping (Figure 6.47), 13 s after the break, the HPIS actuation set point is reached, although it starts injecting borated water after 40 s (Figure 6.48) because of the LOOP. Meanwhile, the temperature reaches the saturation point (Figure 6.49), therefore steam forms. Consequently the water collapsed level in the core starts to decrease (Figure 6.50) and the upper head is uncovered. The mixing of the cold water injected with the water already present in the core and the possibility of remove the energy without interruption due to the sufficient amount of liquid water around the fuel rods, explain the smooth shape of the PCT (Figure 6.51).

The steam produced in the core flows through the U-tubes of the steam generators (Figure 6.52), which cannot condense it since the pressure in the primary side is lower than the pressure in the secondary side (Figure 6.53). On the secondary side, the water level within the SGs decreases rapidly, but the value required for the start of injection of the emergency feedwater pumps is not reached before 4700 s (Figure 6.54).

The break flow remains in two-phase flow for 900 s (Figure 6.55). Although the upper part of the active region of the core is nearly to be uncovered, there is always some amount of liquid flowing through it. The ECCS and the accumulators inject enough borated water to exceed the break mass flow rate

(Figure 6.56) and to recover the water inventory in the core (Figure 6.57), preventing the core from being severely damaged.

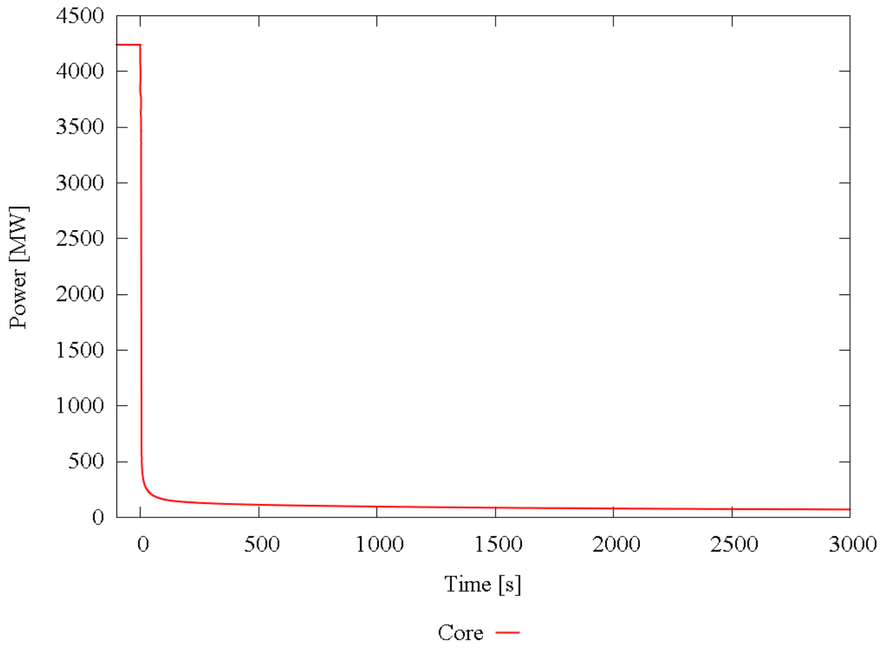

Figure 6.45 Reactor Core Power

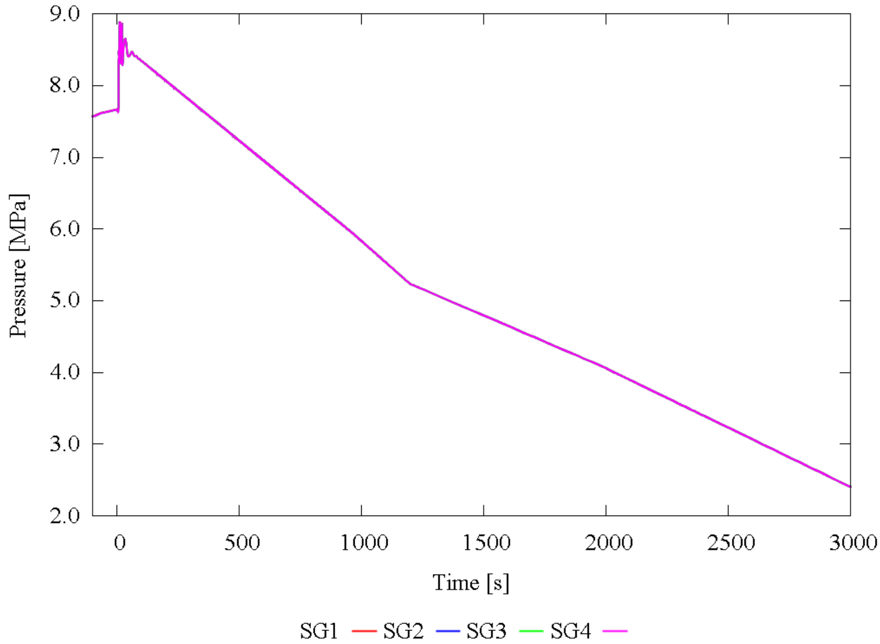

Figure 6.46 Pressure Secondary Side

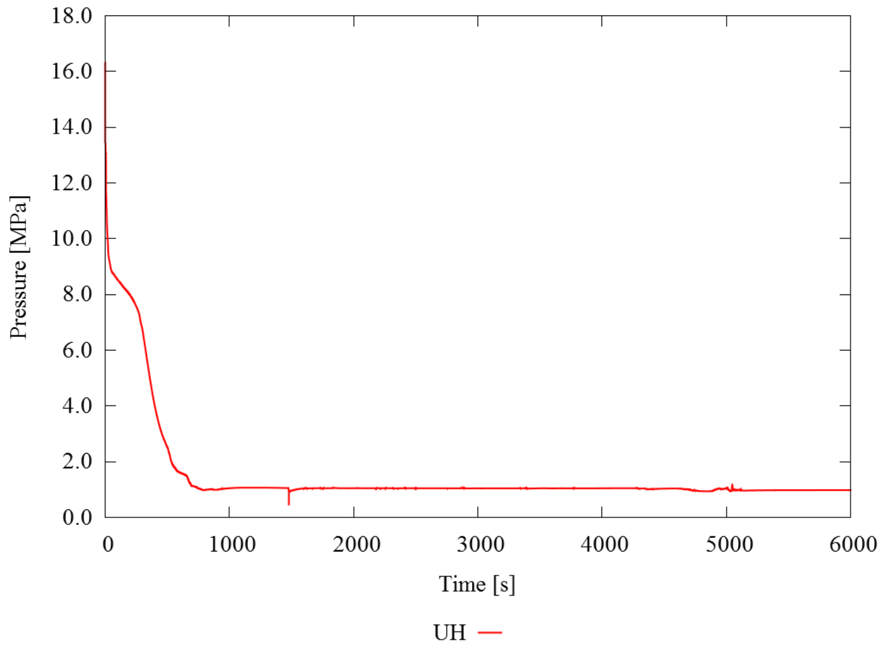

Figure 6.47 Pressure within the Upper Head of the Reactor Core

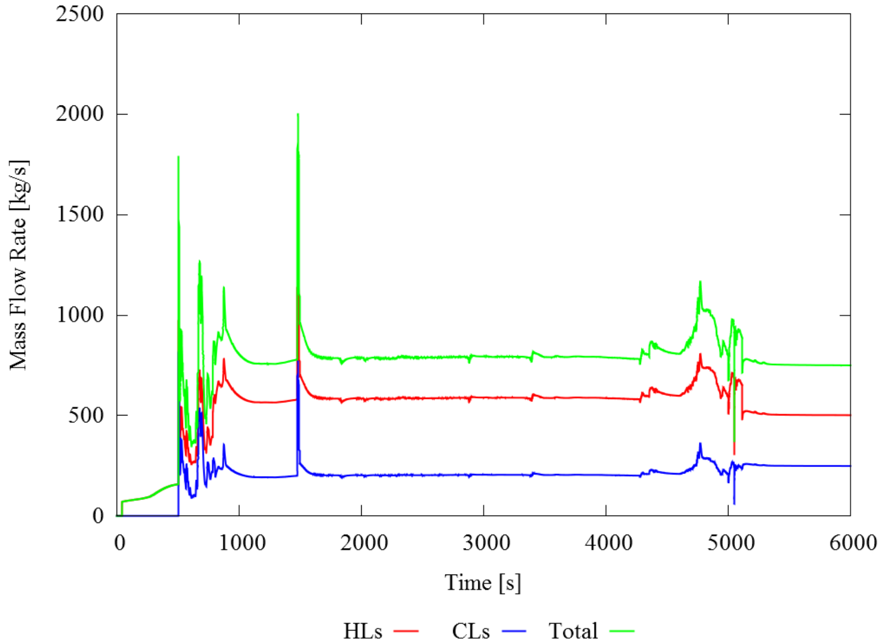

Figure 6.48 ECCS Injection Mass Flow Rate

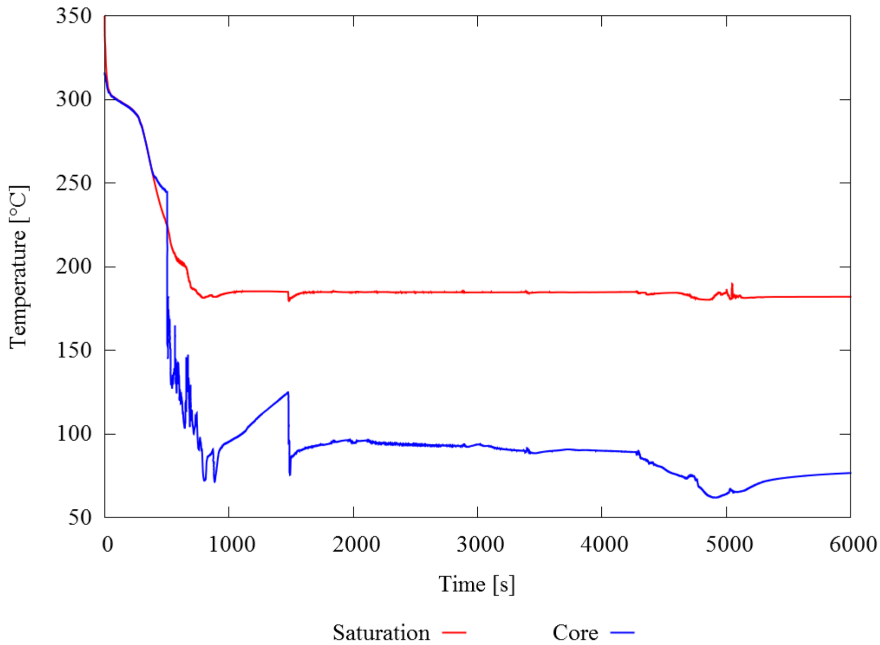

Figure 6.49 Comparison PV-COR3 Temperature and Saturation Temperature

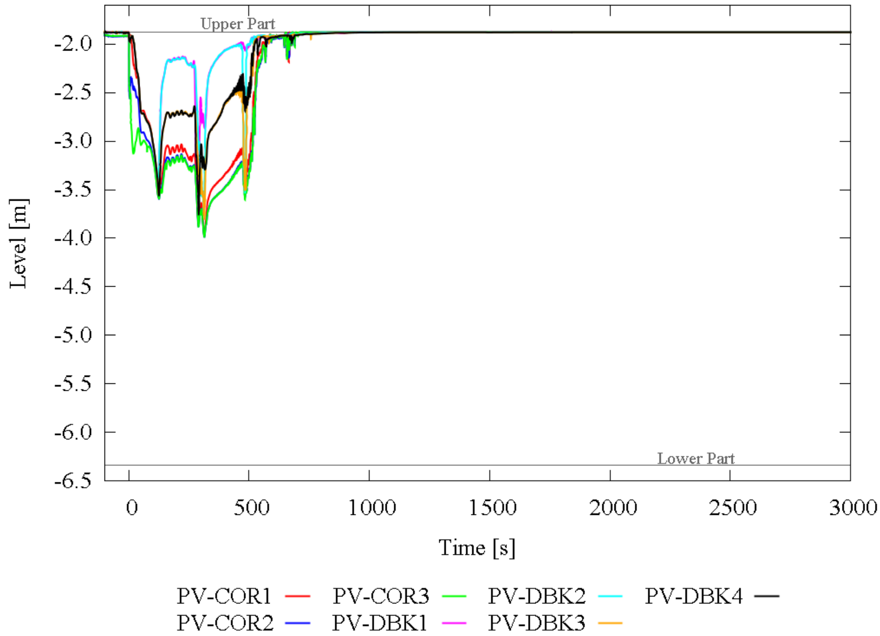

Figure 6.50   Comparison of Water Collapsed Level within the Core

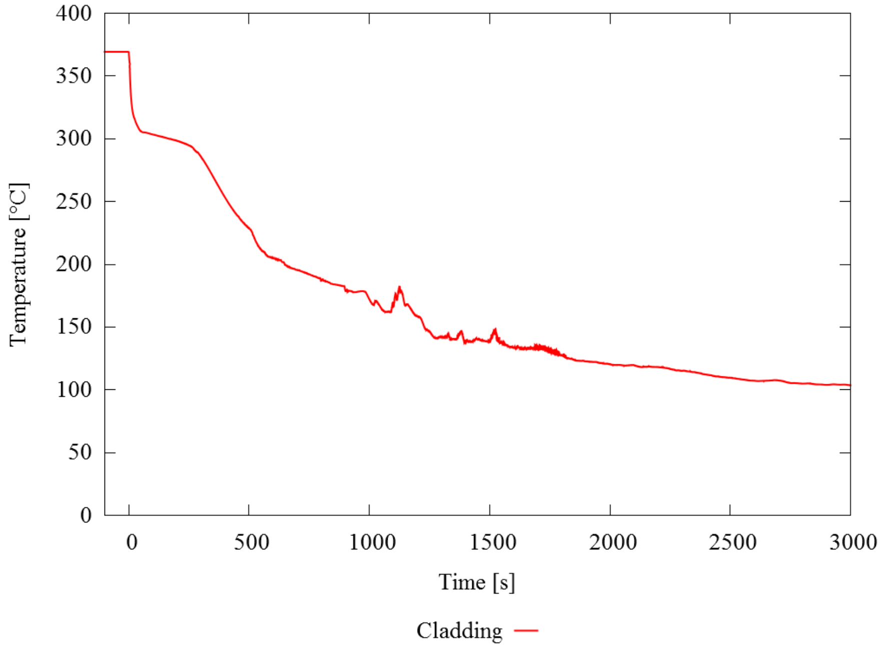

Figure 6.51   Peak Cladding Temperature

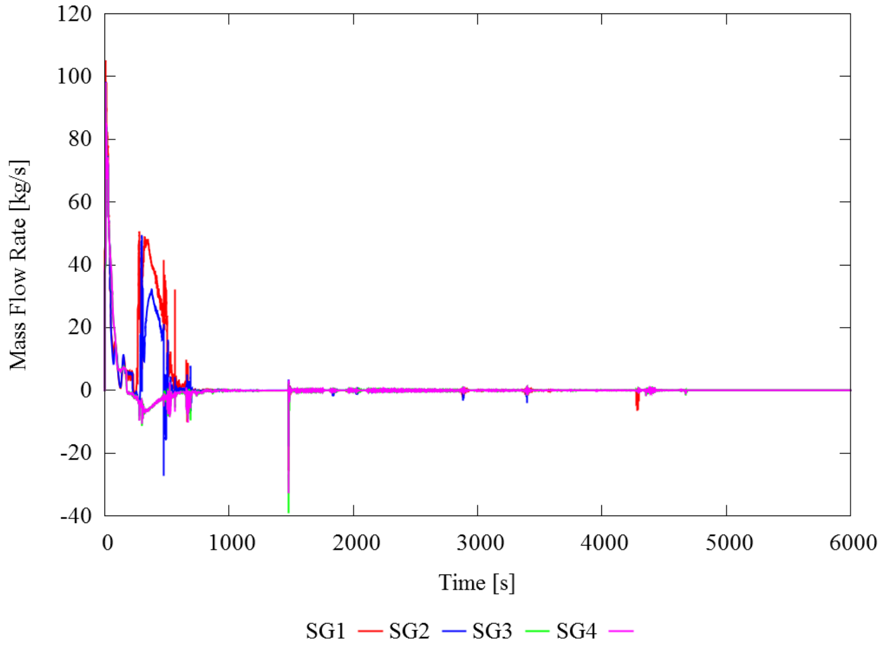

Figure 6.52 Vapour Mass Flow Rate within the U-Tubes of Steam Generators

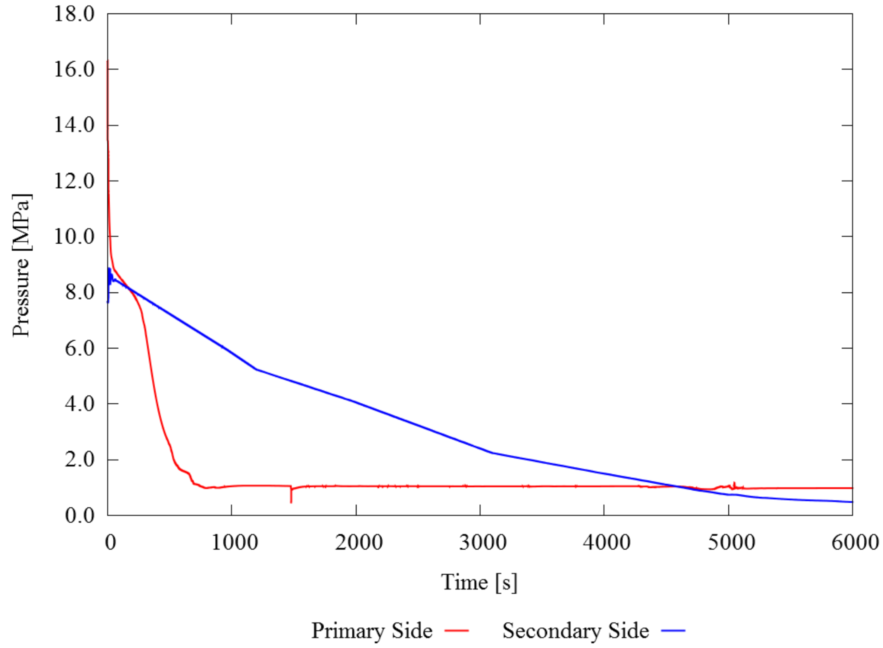

Figure 6.53 Pressure Primary and Secondary Side

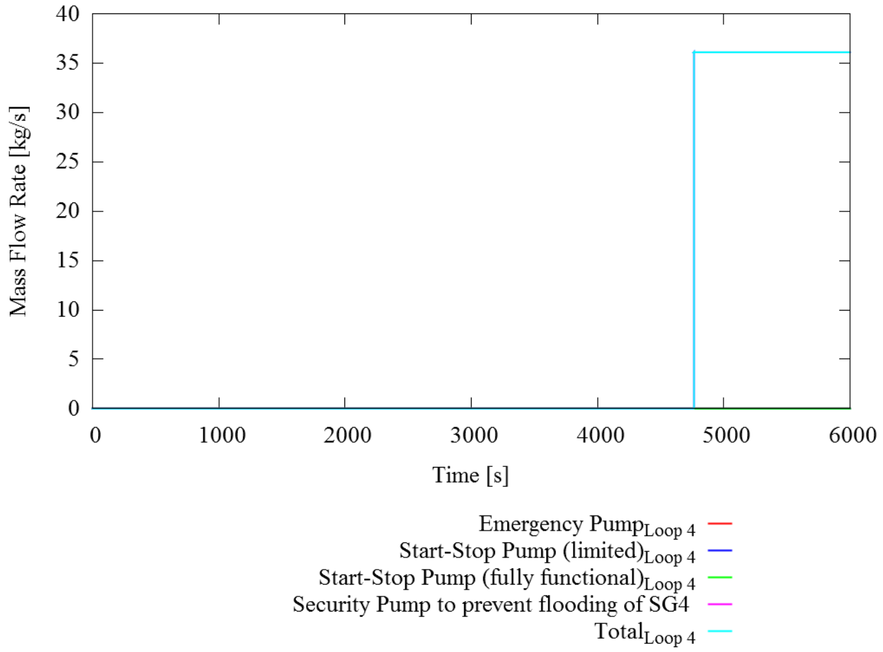

Figure 6.54 Emergency Feedwater Pumps Mass Flow Rate

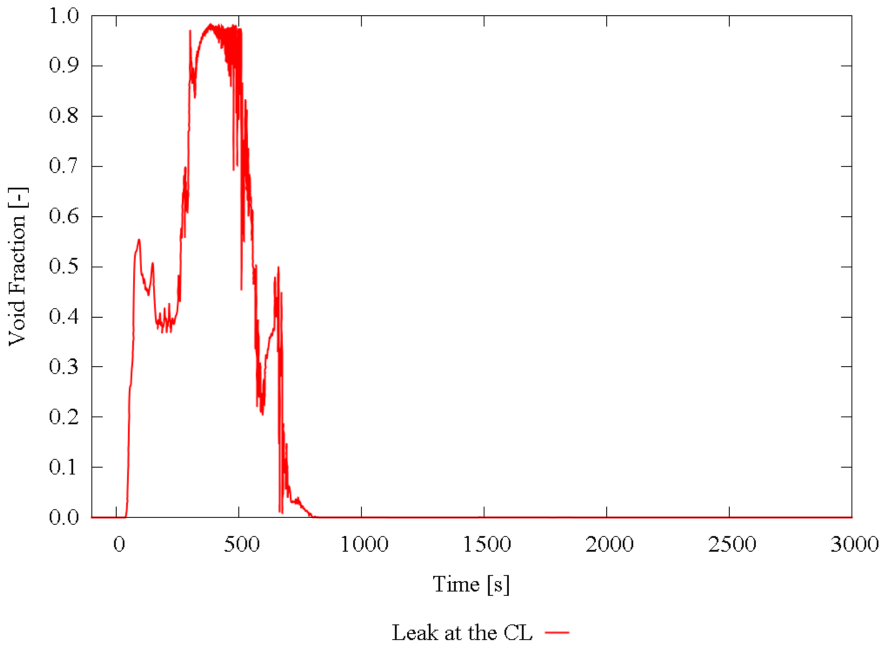

Figure 6.55 Void Fraction at the Leak in the Cold Leg

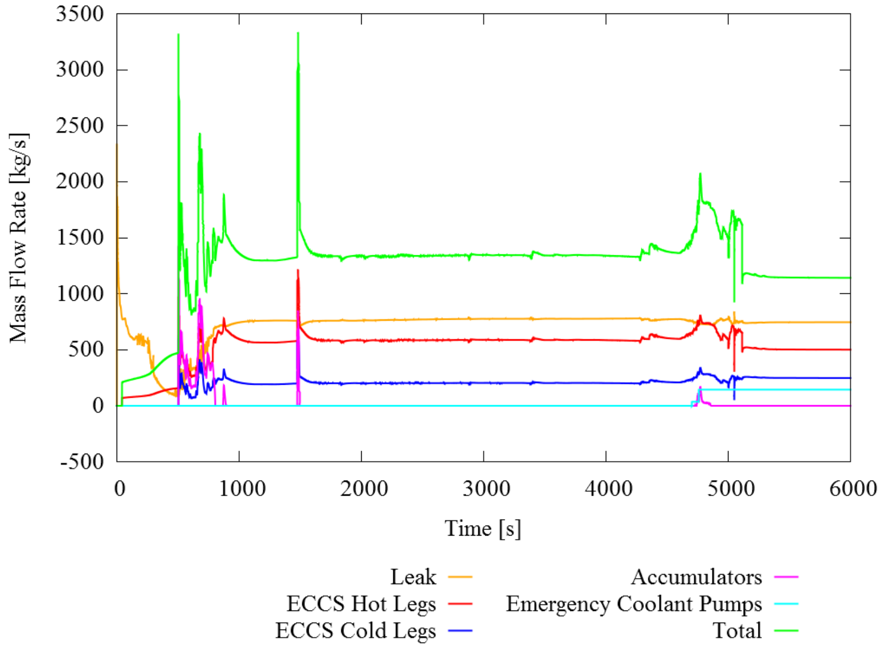

Figure 6.56 Comparison Mass Flow Rate of the Leak and Water Injections

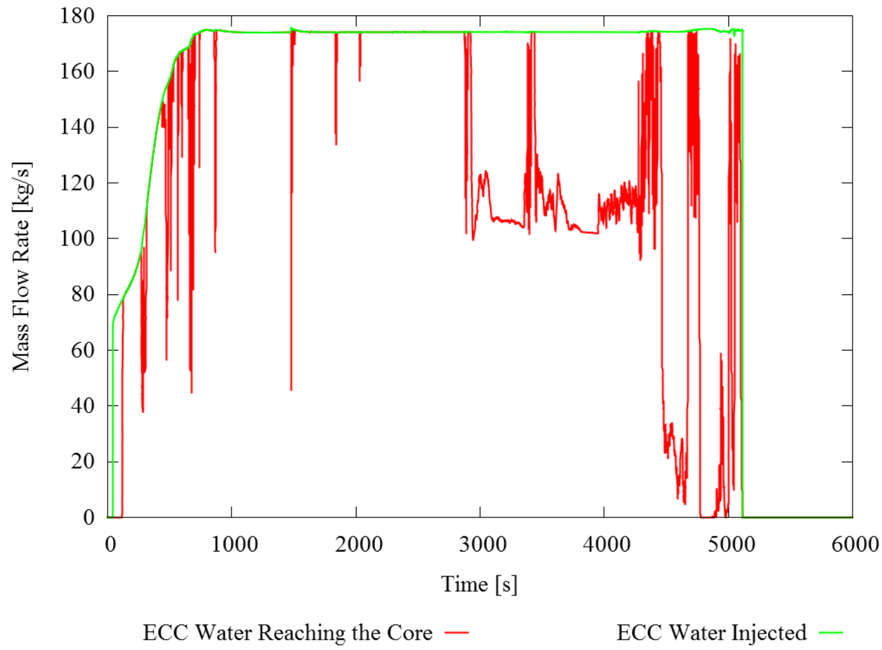

Figure 6.57 Hot Leg SCOOP Injection Mass Flow Rate

6.3 1% CSA Break

6.3.1 Case Study

Break Localization	Cold Leg (Loop 1)
Leak Area	44 cm^2
Reactor Power	4240 MW (106% of Nominal Power)
HPIS Set point	Loop 1, Loop 2: Active Loop 3, Loop 4: Disable Primary Pressure < 11.1 MPa
LPIS Set point	Loop 1, Loop 2: Active Loop 3, Loop 4: Disable Primary Pressure < 1.0 MPa
Accumulators Set point	Hot Leg Side: Active Cold Leg Side: Active Primary Pressure < 2.5 MPa
Emergency Feedwater Pumps Set point	Active SG Water Level < 5 m
Additional Borated Water Injection (2kg/s)	Loop 1: Deactivated Loop 2, Loop 3, Loop 4: Active
Quench Front Model Set point	Primary Pressure < 3.0 MPa

Table 6.6 1% CSA Break. Case Study

6.3.2 Transient Calculation

Event	Event Time [s]
Leak Opening	0
SCRAM signal (primary pressure low)	29,5
Reactor and Turbine Trip, LOOP	29,5
Steam Generators Trip (100k/h gradient)	29,5
Extra Borated Water Injection Start	30,0
Maximum Pressure Secondary Side Set point Reached	44,9
HPIS Actuation Set point Reached	54,7
HPIS Injection Start	67,6
Break Flow from Single-Phase to Two-Phase	221.8
Minimum Core Liquid Coolant Inventory	551,6
Emergency Feedwater Injection Start	686,8
Accumulators Set point Reached	3047,7
Accumulators Injection Start	3047,7
Primary Pressure lower than Secondary Pressure	4679,0
HPIS Tanks Drained	5942,6
Plant Under Control	6000,0
End of Simulation	6000,0

Table 6.7 1% CSA Break. Transient Calculation

6.3.3 Analysis of the results

In this case, the depressurization rate is smaller due to the small C.S.A. of the break (Figure 6.58). As a result, the reactor operates normally during 29 s after the break opening before reaching the pressure lower threshold for the SCRAM actuation. However, the power starts decreasing before the SCRAM (Figure 6.59) as the consequence of the implementation of the point kinetics model, which assumes that the power is proportional to the average neutron flux. If coolant is lost through the leak (this is moderator lost through the leak), the speed of fast neutrons increases, thus decreasing the fission probability. Thanks to this slow depressurization, the secondary side cools down on the primary side during the whole transient (Figure 6.60).

The core reflooding is achieved by the HPIS (Figure 6.61). The leak mass flow rate is not excessively high (Figure 6.62), compared to the 10 and 5% cases, which permits the core water inventory to be completely recovered at about 1700 s (Figure 6.63).

On the secondary side, the water collapsed level decreases (Figure 6.64) until the set point of the emergency feedwater pumps is reached and they start injecting liquid to the SGs (Figure 6.65). Loop 1 and 2 have different behaviour than loops 3 and 4, since the HPIS and the LPIS are only available in these loops.

On the primary side, the temperature and the pressure continue dropping and steam forms at about 200 s (Figure 6.66). This steam flows through the U-tubes of the SGs (Figure 6.67) and through the leak, although the leak is never completely uncovered (Figure 6.68).

The PCT drops rapidly after the SCRAM (Figure 6.67), since, despite the leak, the core can be continuously cooled down. Afterwards, the mixing of the cold injected water with the water present in the core, lead to a slow and smooth temperature decrease. Since the core is never completely uncovered, the safety of the reactor is not seriously at risk during this transient.

The accumulator injection starts at about 3000 s, which helps the ECCS mass flow rate to exceed the leak mass flow rate (Figure 6.68).

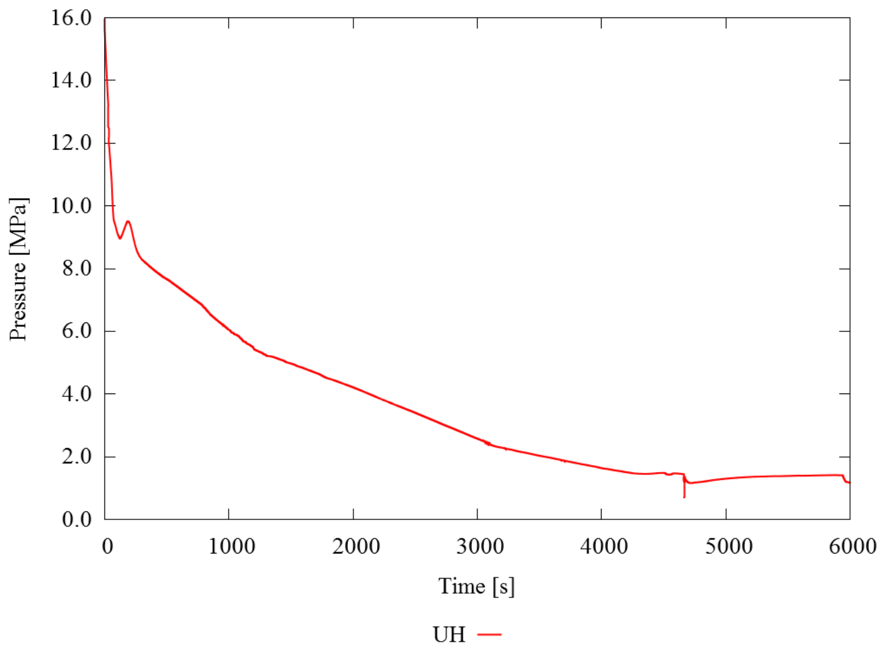

Figure 6.58 Pressure Upper Head of the Core

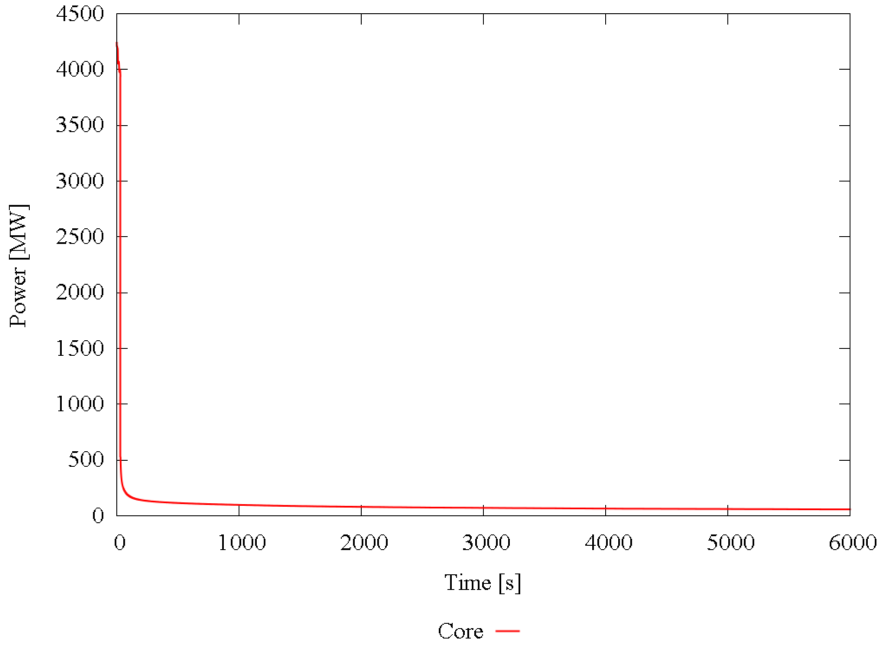

Figure 6.59 Reactor Core Power

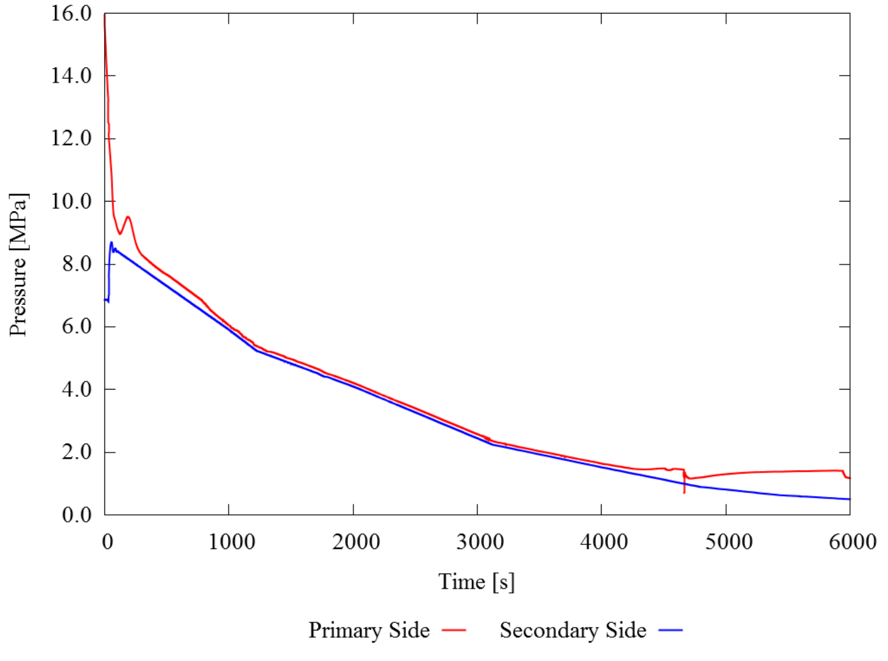

Figure 6.60 Pressure Primary and Secondary Side

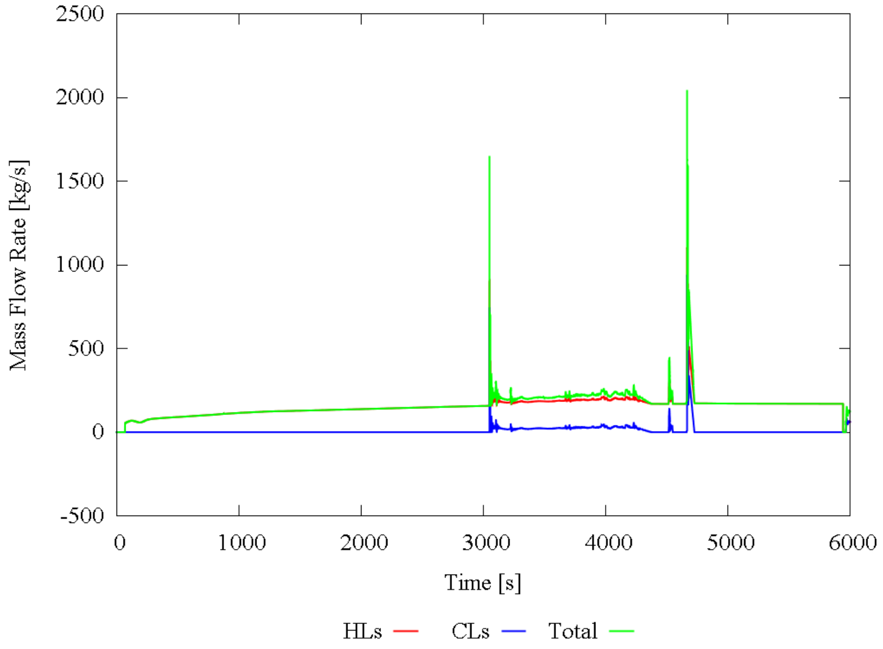

Figure 6.61 Hot Leg and Cold Leg Water Injection

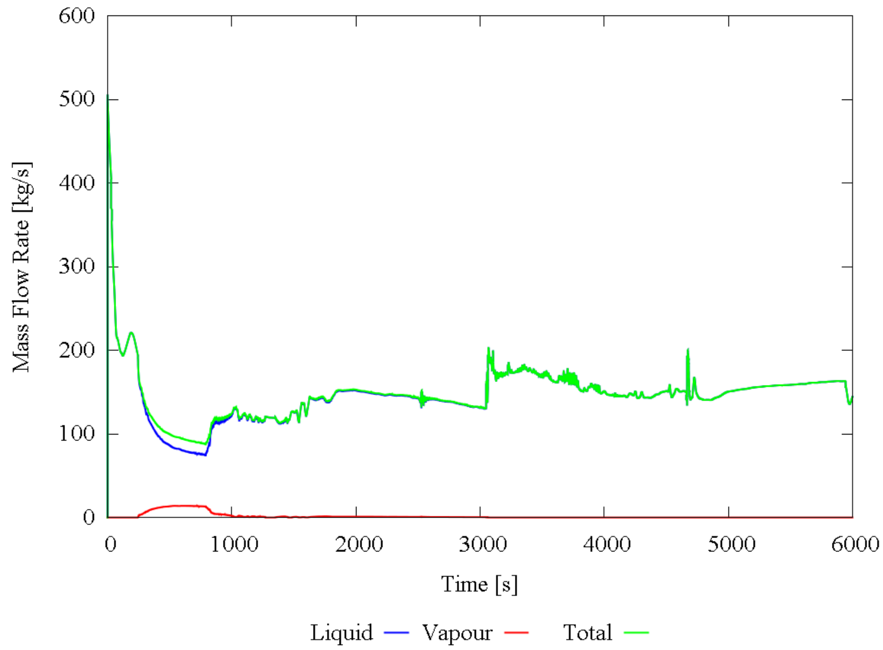

Figure 6.62 Comparison Mass Flow Rate through the Leak

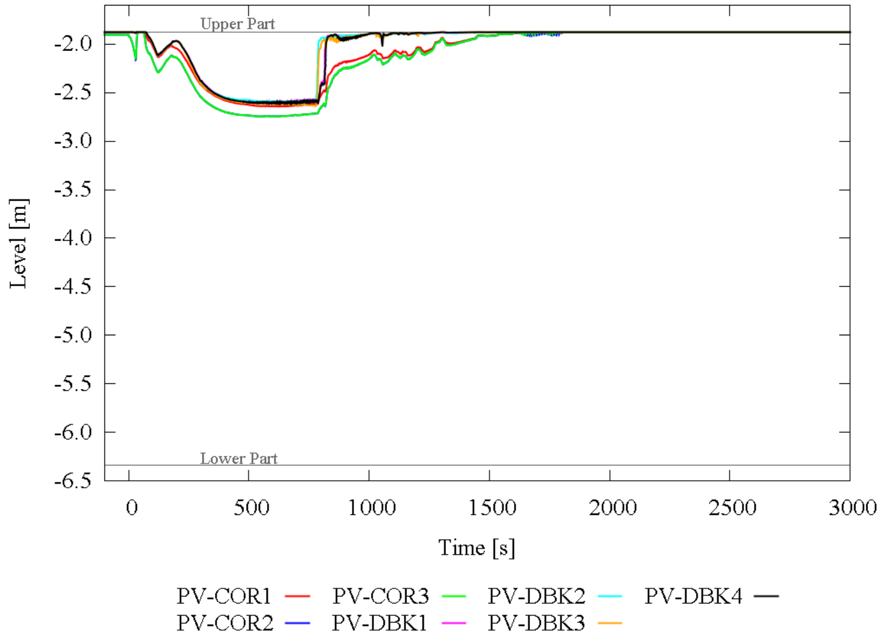

Figure 6.63 Comparison of Water Collapsed Level within the Core

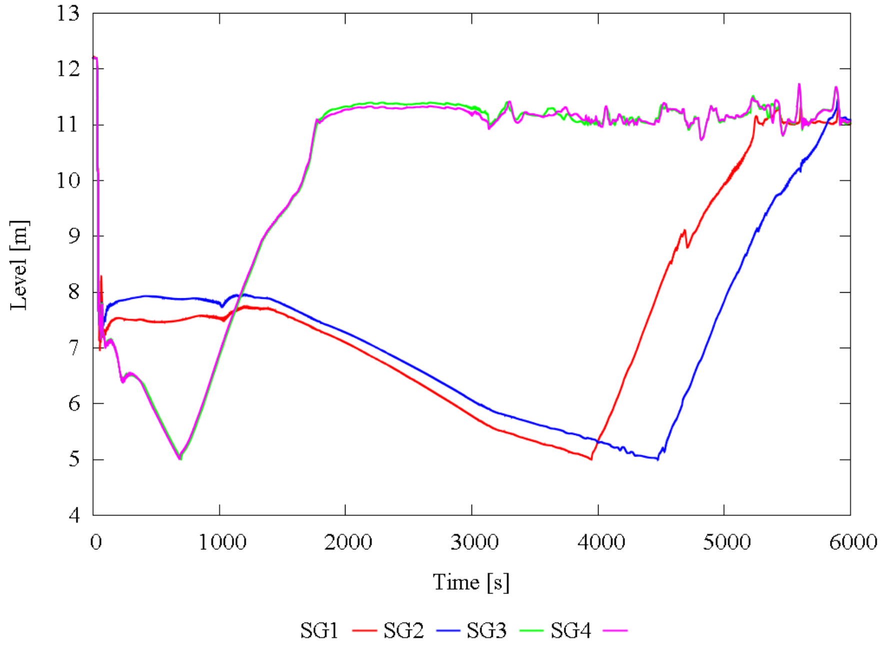

Figure 6.64 Water Level in the Steam Generators

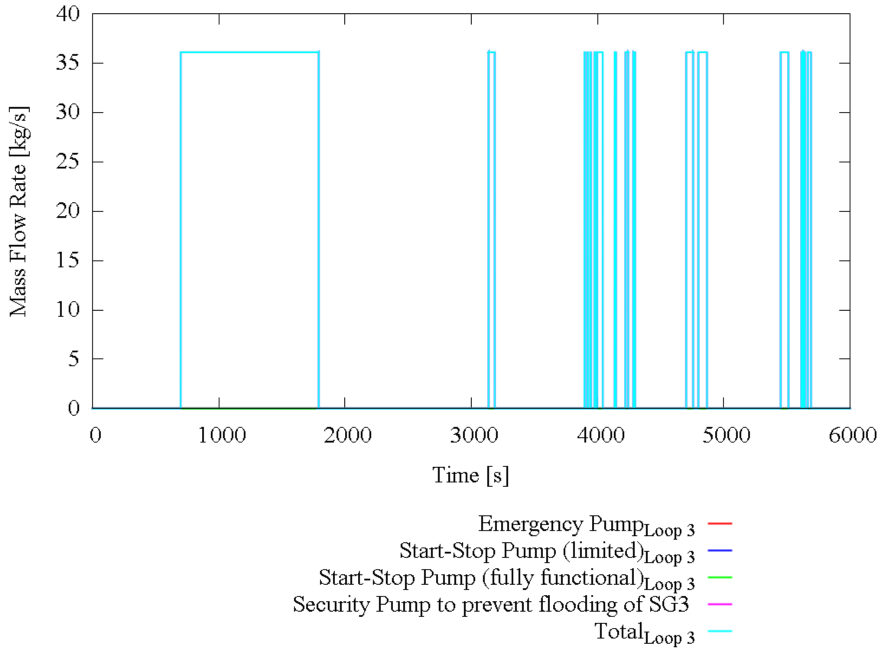

Figure 6.65 Emergency Feed Water Mass Flow Rate Loop 3

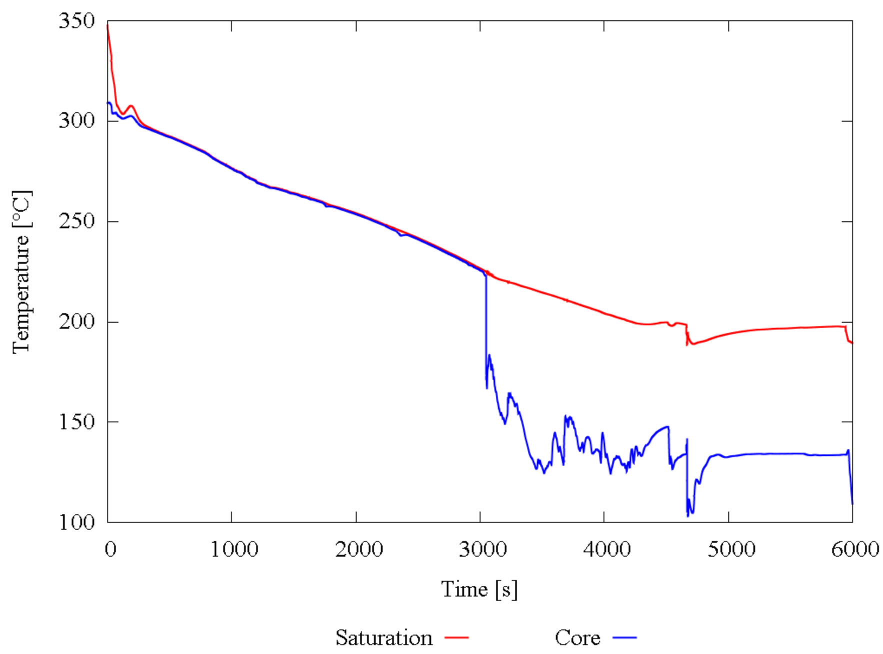

Figure 6.66 Comparison Mean Coolant Temperature and Saturation Temperature in PV-COR·

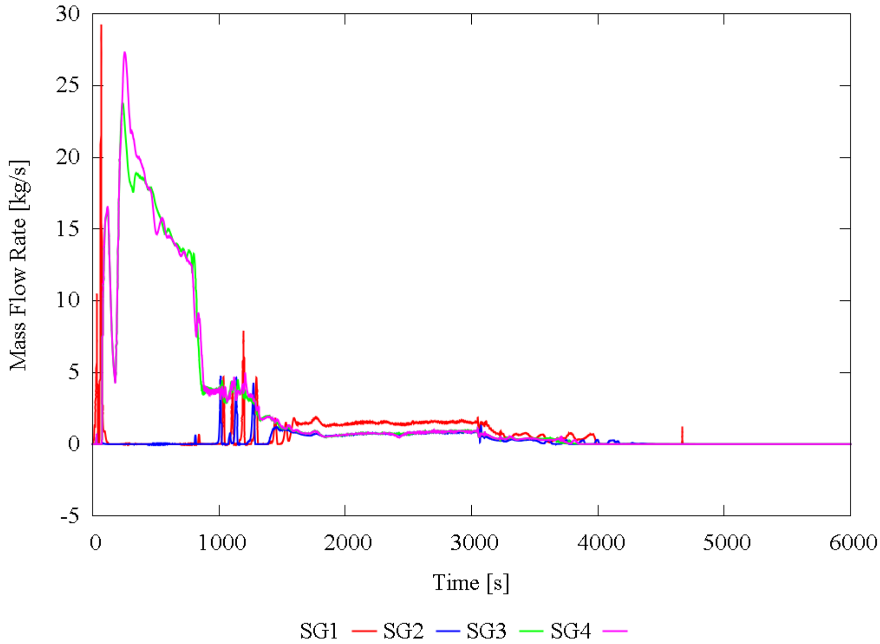

Figure 6.67 Vapour Mass Flow Rate within the U-Tubes of Steam Generators

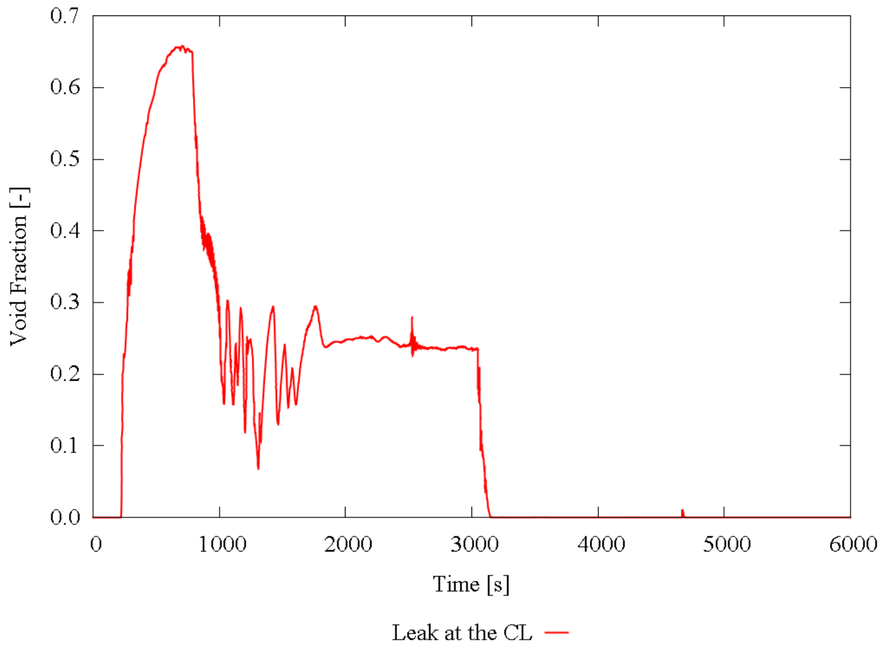

Figure 6.68 Void Fraction at the Leak in the CL

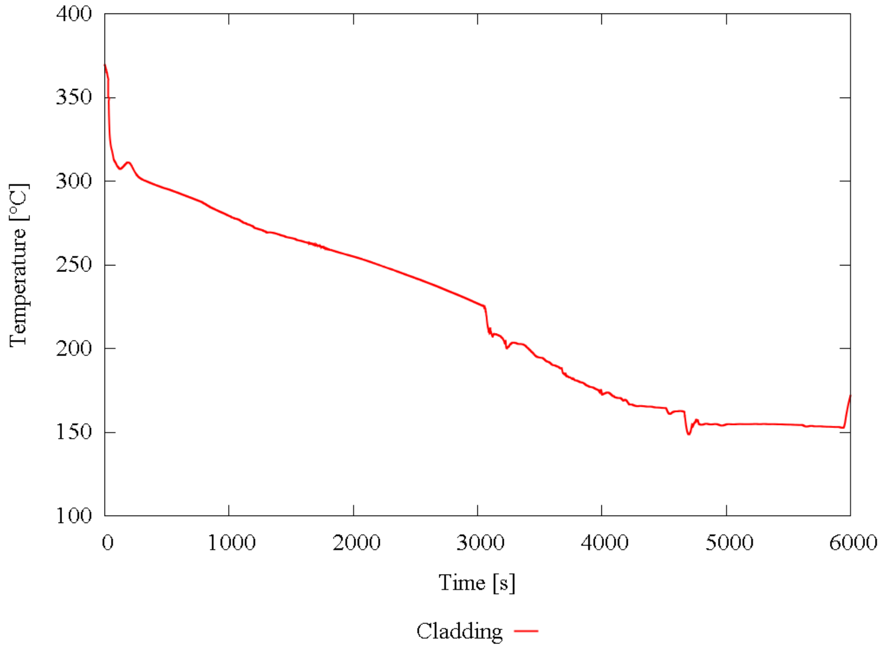

Figure 6.69 Peak Cladding Temperature

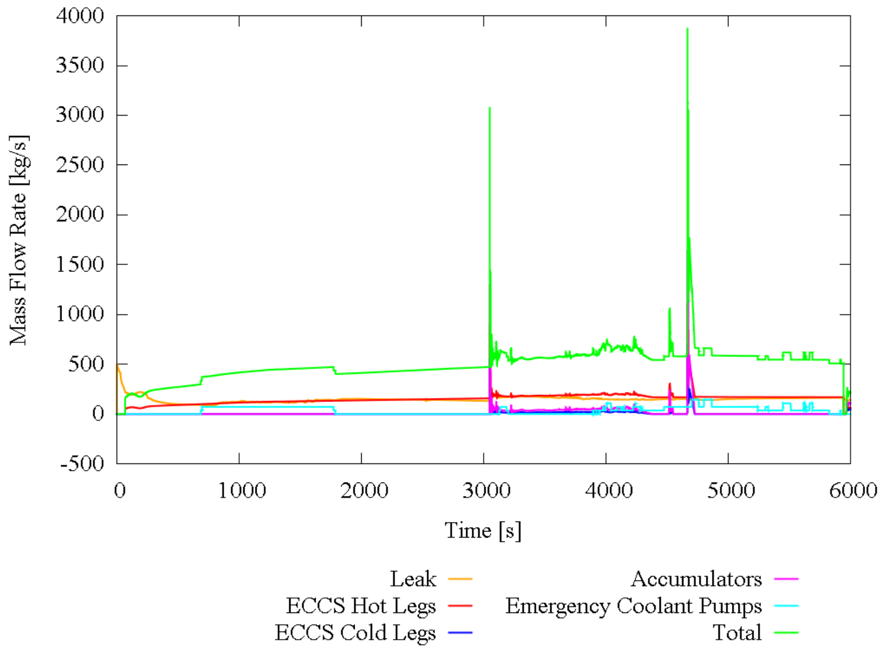

Figure 6.70 Comparison Mass Flow Rate of the Leak and Injections

6.4 20 cm^2 Break

6.4.1 Case Study

Break Localization	Surge Line
Leak Area	220 cm^2
Break Orientation	45° to the Top
Reactor Power	4000 MW (100% of Nominal Power)
Scram Set point	Triggered Manually
HPIS	Disable
LPIS	Disable
Accumulators Set point	Hot Leg Side: Active Cold Leg Side: Disable Primary Pressure < 2.5 MPa
Loss of Offsite Power (LOOP)	60 s After the Scram
Diesel Generators	40s After the LOOP
Emergency Feedwater Pumps Set point	Active Before the LOOP Disable at the LOOP Active at the Diesel Generators Start SG Water Level < 8 m
Pressurizer	Disable at the LOOP
Additional Borated Water Injection Start (2kg/s)	Loop 1: Deactivated Loop 2, Loop 3, Loop 4: Active
Quench Front Model Set point	Core Refill Point

Table 6.8 20 cm^2 Break. Case Study

6.4.2 Steady State Verification

Parameter	Steady State Value
Power	
Reactor Power	4000 MW
Power Extracted by the Steam Generators	4023 MW
Pressure	
Primary Pressure	15.7 MPa
Steam Pressure	6.66 MPa
Total Pressure Loss along the Primary Side	0.43 MPa
Pressure Loss along the Reactor Core	0.21 MPa
Temperatures	
Steam Generator Inlet Temperature	330.7 ºC
Steam Generator Outlet Temperature	296.3 ºC
Primary Side Mass Flow Rates	
Mass Flow Rate per Coolant Loop	4976 kg/s
Net Mass Flow Rate through the Reactor Core	19459 kg/s
Level	
Pressurizer Collapsed Level	8.21 m
Steam Generator Collapsed Level	12.2 m
Primary Side Boron Concentration	900 ppm

Table 6.9 20 cm2 Break. Steady State Verification

6.4.3 Transient Calculation

Event	Event Time [s]
Manual Scram	0
Steam Generators Trip (100k/h gradient)	0
Reactor Trip	0
Extra Borated Water Injection Start	0,6
Maximum Pressure Secondary Side Set point Reached	6,6
Emergency Feedwater Injection Start	10,0
Minimum Primary Pressure Set point Reached	23,5
Leak Opening	60
LOOP	62,5
Break Flow from Single-Phase to Two-Phase	912,7
Break Flow to Single-Phase Vapour	1733,3
Accumulators Set point Reached	3054,1
Accumulators Injection Start	3054,1
Minimum Core Liquid Coolant Inventory	4531,0
Plant Under Control	8000,0
End of Simulation	9800,0

Table 6.10 20 cm2 Break. Transient Calculation

6.1.4 Analysis of the results

The transient begins with a SCRAM activated manually because of the presence of intruders on the power plant site. The reactor (Figure 6.71) and the steam generators (Figure 6.72) are shut down immediately.

The primary side starts depressurizing very slow (Figure 6.73). At the moment of the break, the gradient of the pressure curve changes, although, the pressure continues decreasing in a very smooth way. At the end of the simulation reaches a constant value of 6.32 MPa.

The secondary pressure experiences a rapid increase due to the closure of the turbine isolation valve. After that, the secondary pressure decreases also in a very smooth way (Figure 6.74). The primary pressure does not drop at any moment below the secondary pressure (Figure 6.75): the secondary side is continuously cooling the primary side.

The liquid level within the steam generators experiences a rapid decrease after the SCRAM (Figure 6.76) and keeps decreasing until the emergency feedwater pumps setpoint is reached (Figure 6.77). These pumps work before the LOOP occurs, then stop working until the diesel generators are able to provide them electrical power. The injection is interrupted during 40 s. With help of the emergency water pumps, the steam generators make possible the decreasing of the PCT (Figure 6.78).

60 s after the SCRAM, the break opens. The higher amount of mass escaping through the leak is liquid (Figure 6.79). The leak is uncovered during most of the transient (Figure 6.80), since the pressurizer is drained (Figure 6.81).

The core remains fully covered during 500 s. After that time, steam locates in the upper part of the core and the water level starts decreasing (Figure 6.82). The steam flows through the steam generator U-tubes (Figure 6.83) and through the loop seals (Figure 6.84). Although, the amount of steam is very small, since most of it can be condensed.

As the HPIS and LPIS are disabled, the accumulators are the only safety injection system capable to inject mass flow to the core (Figure 6.85). 3000 s after the break, the accumulators start to inject borated water to the system, beginning thus the recovery of the core.

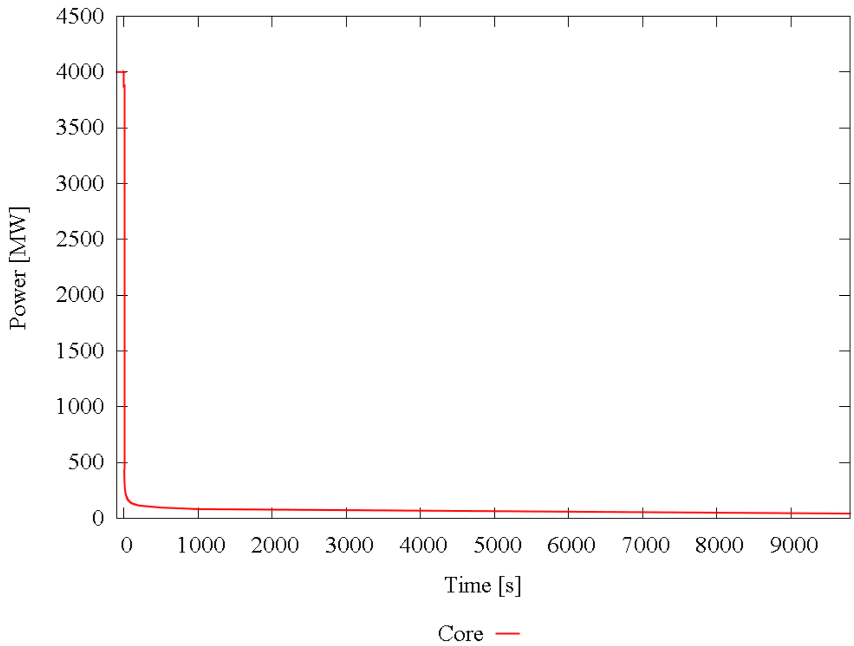

Figure 6.71 Reactor Core Power

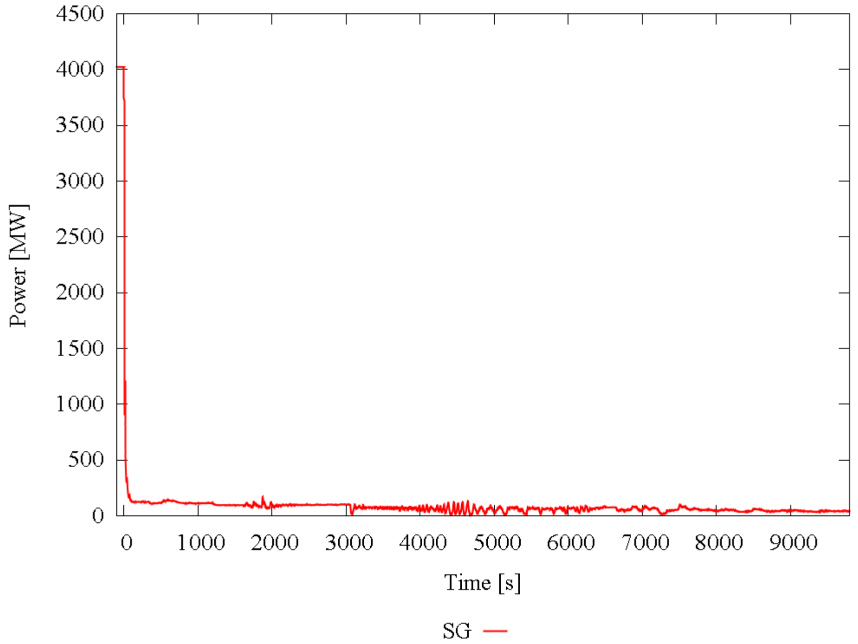

Figure 6.72 Heat Removed from the Primary Side by the Steam Generators

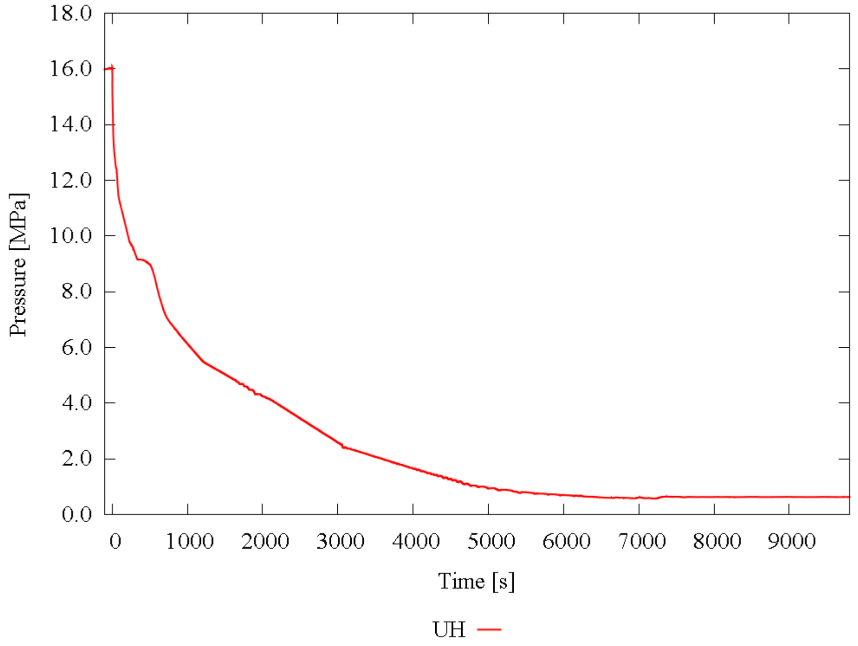

Figure 6.73 Pressure Upper Head of the RPV

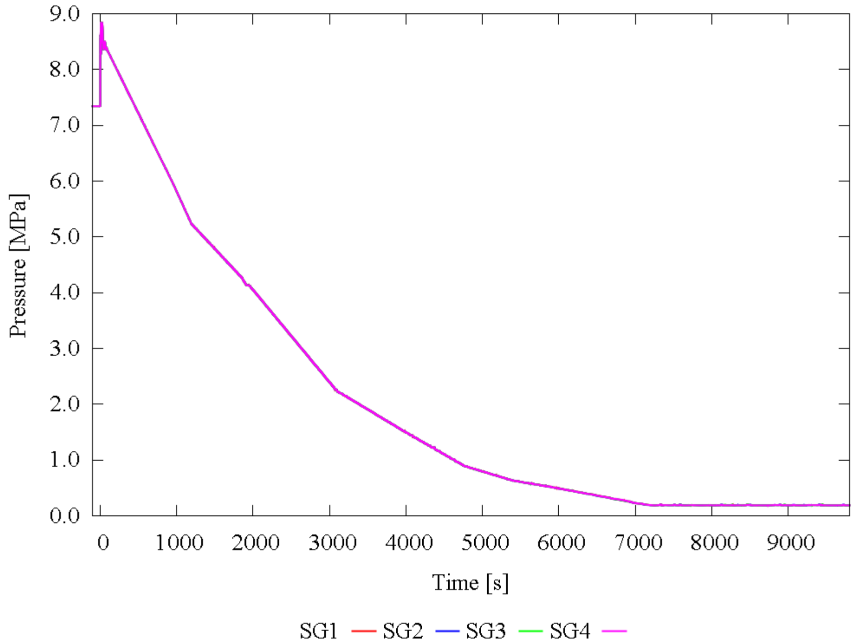

Figure 6.74 Secondary Side Pressure

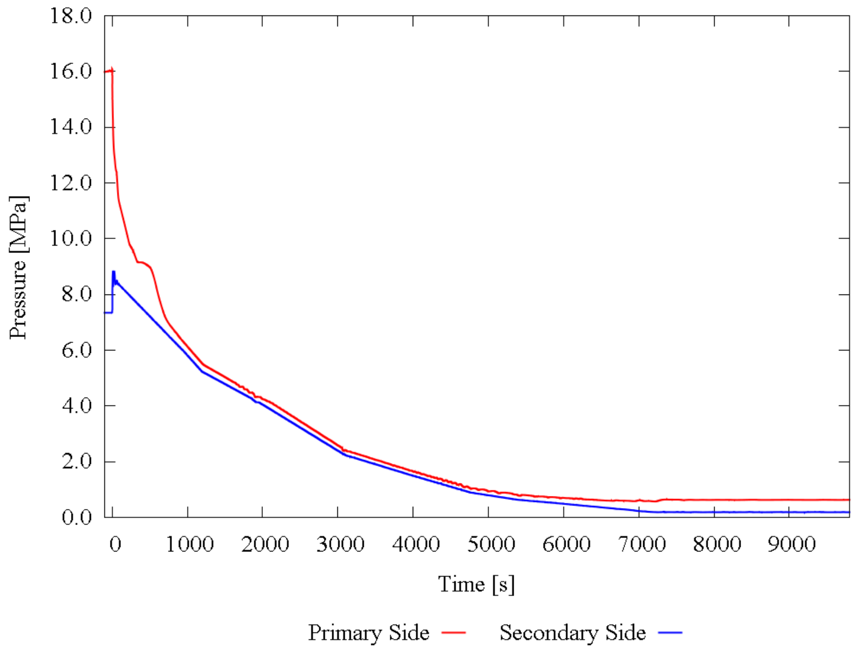

Figure 6.75 Comparison Primary and Secondary Side Pressures

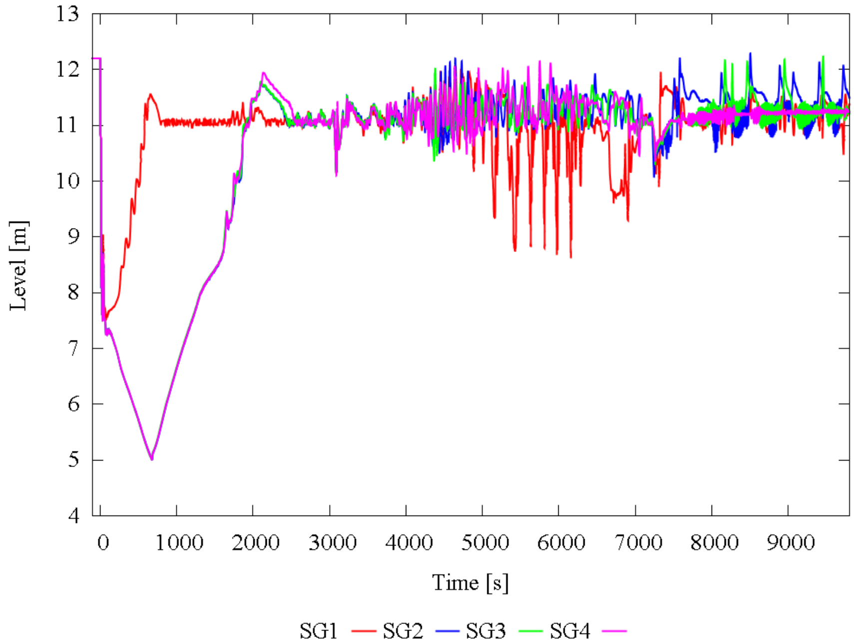

Figure 6.76 Water Collapsed Level within the Steam Generators

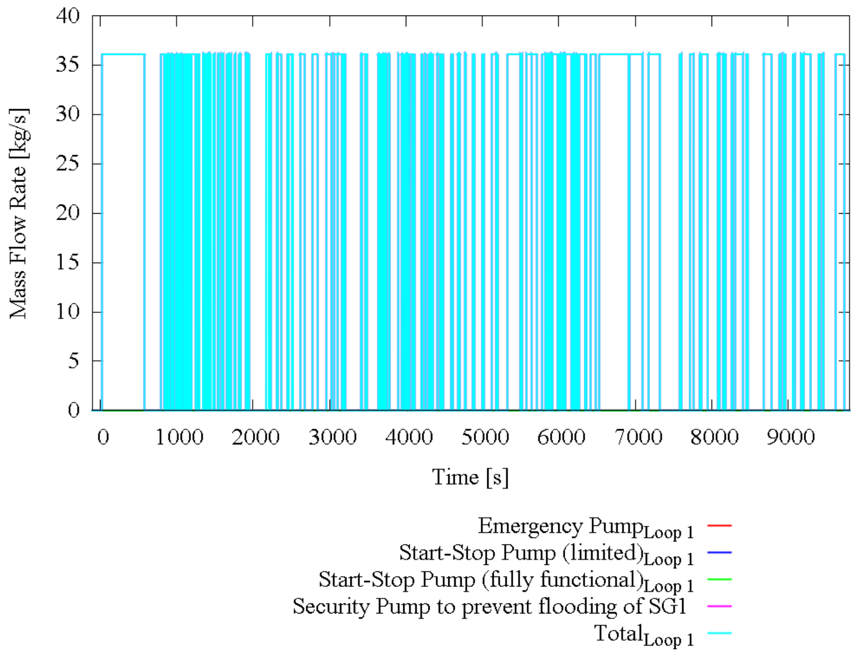

Figure 6.77 Emergency Feed Water Mass Flow Rate Loop 1

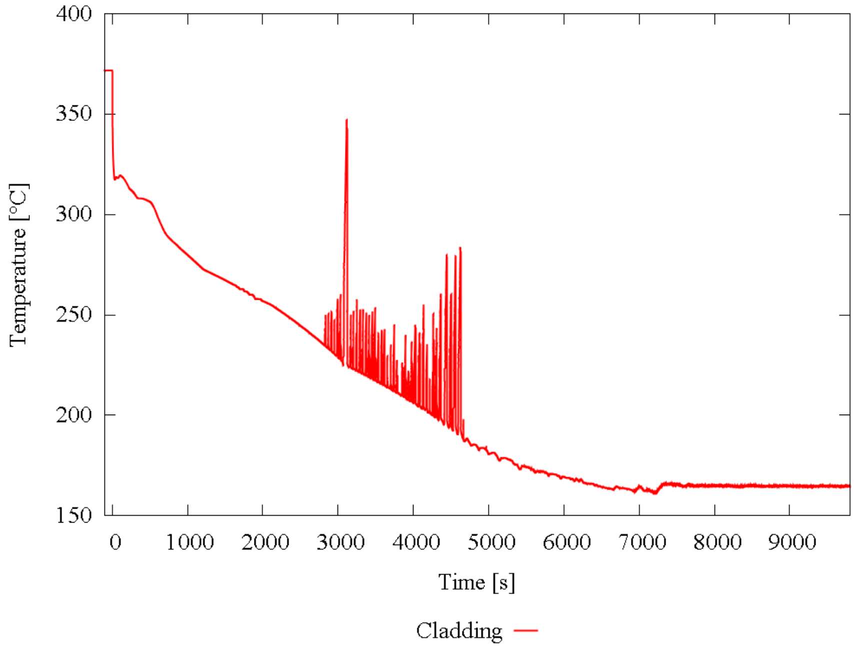

Figure 6.78 Peak Cladding Temperature

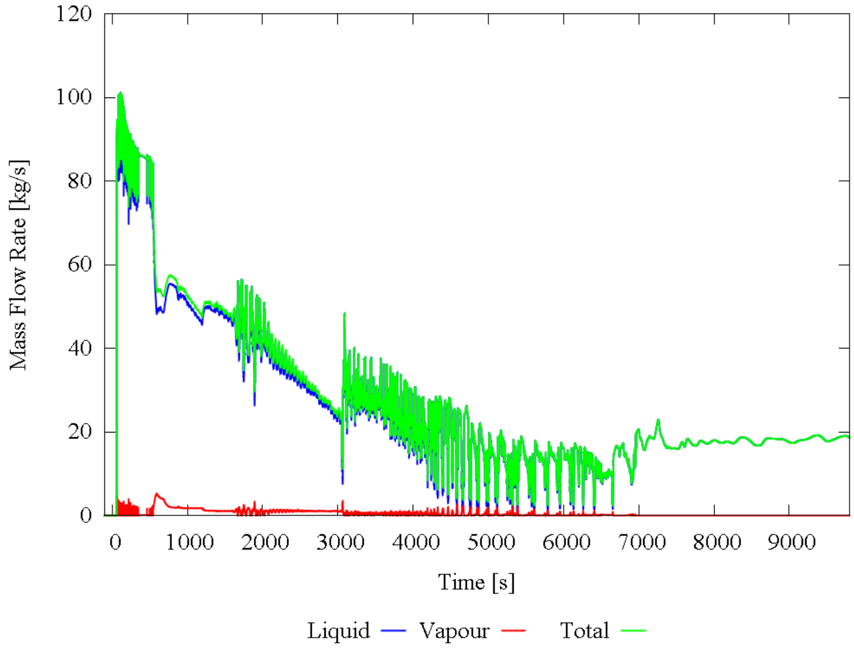

Figure 6.79 Comparison Mass Flow Rate through the Leak

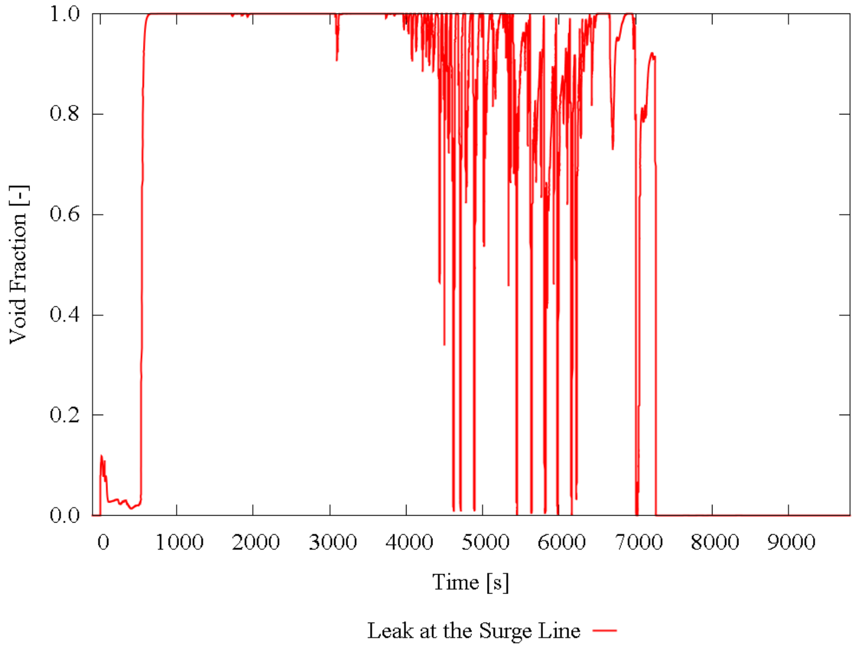

Figure 6.80 Void Fraction at the Leak in the Surge Line

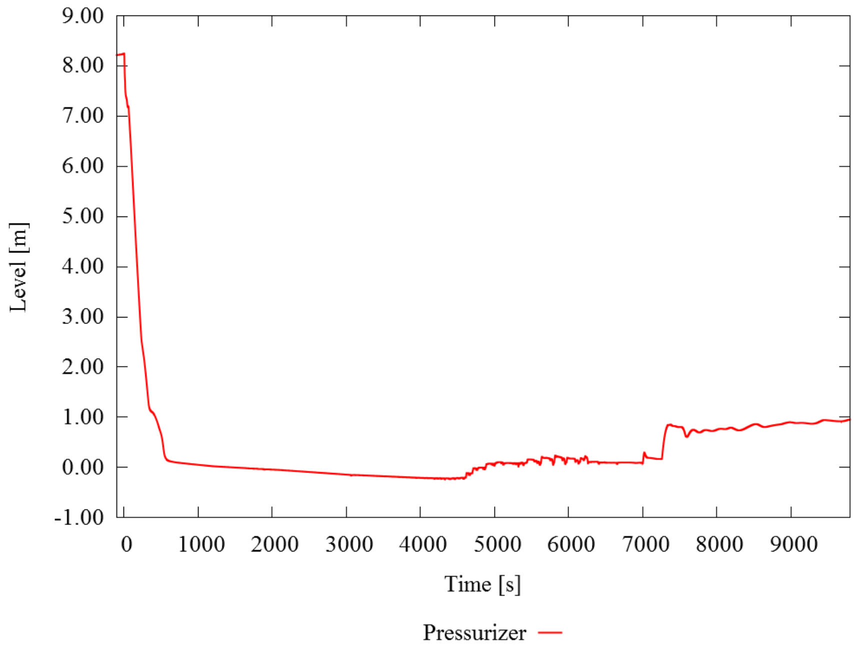

Figure 6.81 Water Level within the Pressurizer

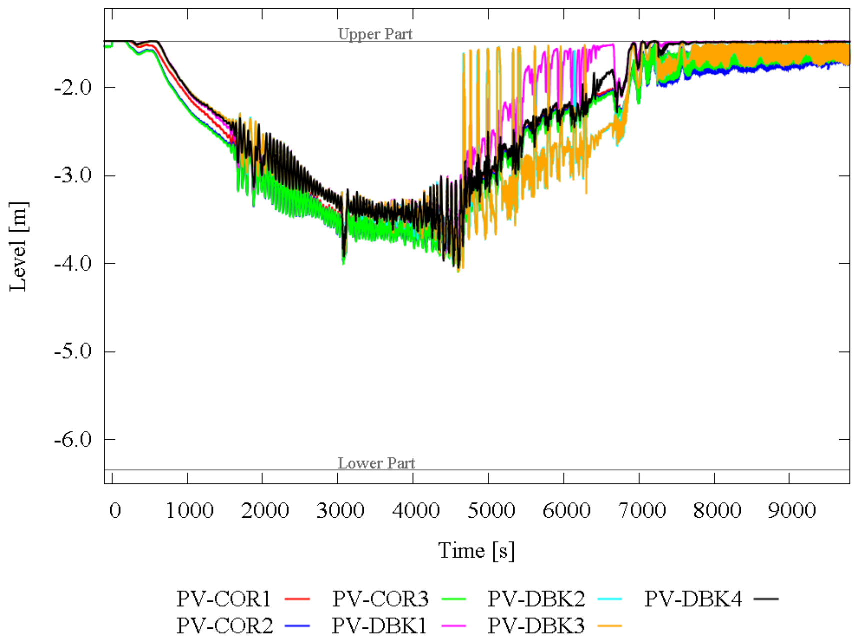

Figure 6.82 Comparison of Water Collapsed Level within the Core

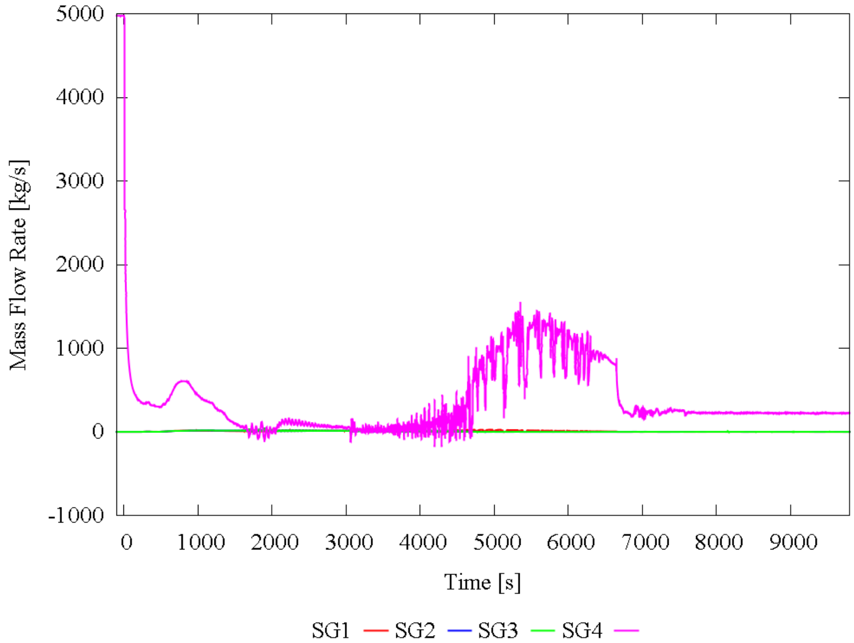

Figure 6.83 Vapour Mass Flow Rate within the U-Tubes of Steam Generators

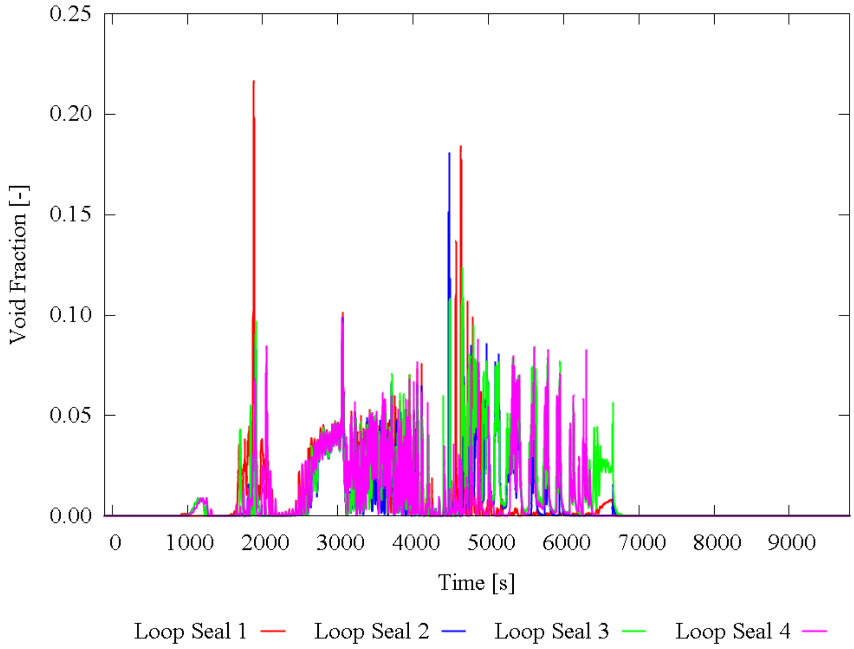

Figure 6.84 Void Fraction at the Loop Seal

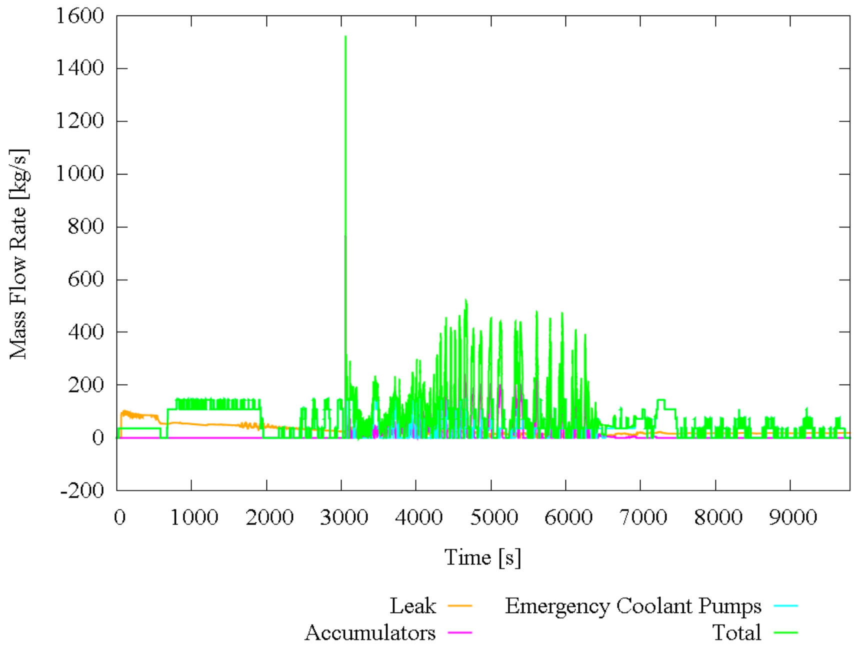

Figure 6.85 Comparison Mass Flow Rate of the Leak and Injections

Chapter 7

Conclusion

Motivated by the importance of safety in nuclear power plants, a computational model is developed to simulate the SBLOCAs by using ATHLET code.

The steady state and transient state are simulated and thermal-hydraulically analysed in the present thesis. The steady state shows the initial conditions, while the transient presents the behaviour of the reactor during the different SBLOCAs cases.

The simulation results show that the reactor core suffers no severe damages during the transient if the break size is 10%, 5% or 1% of the CSA of the cold leg, or if it is located in the surge line with an area of 20 cm^2. The temperature can increase several hundred degrees and still being below the melting point of the cladding material (1200 ºC), which means that the reactor will remain safely protected.

The main physical expected processes during this kind of accidents have been highlighted in the results, which reveals that the implemented ATHLET code provides a good confidence basis for the study of SBLOCAs in a PWR.

Appendix A

Equation Model

The different equation models used in ATHLET for the simulation of fluiddynamics are described hereafter.

4.1 6-Equation Model

The 6-Equation, is the most general equation system used in ATHLET. It solves the mass and energy balances in the control volumes separately for liquid and vapour, and a mixture momentum balance at the junctions [AUST09].

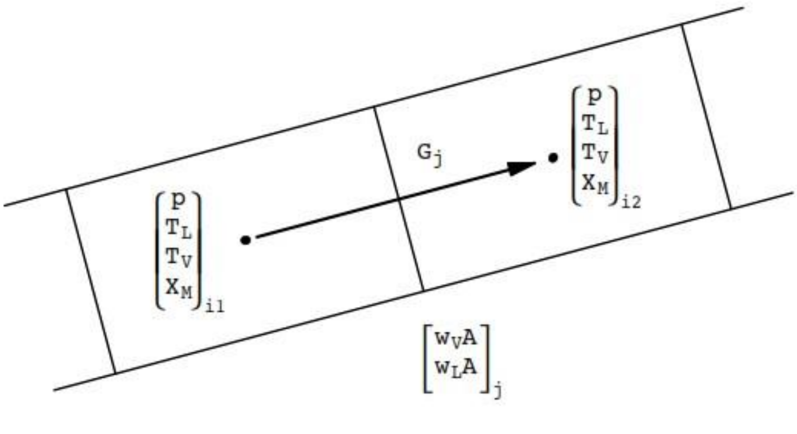

Figure A.1 Displaced grid of CVs and junctions

The solution variables of the system of differential equations are:

- The mass quality x_M:

$$\frac{dx_M}{dt} = \frac{d\left(\frac{M_V}{M_L + M_V}\right)}{dt} = \frac{M_L \frac{dM_V}{dt} - M_V \frac{dM_L}{dt}}{(M_L + M_V)^2} \tag{A.1}$$

where:

$\frac{dM_V}{dt}$: vapour mass balance

$\frac{dM_L}{dt}$: liquid mass balance

- The phase temperatures T_L and T_V:

$$\frac{dT_k}{dt} = \frac{1}{c_{p,k}} \frac{E_k}{M_k} + \frac{1}{c_{p,k}} \left(T_k - \frac{\partial h_k}{\partial p}|_{T_k}\right) \frac{dp}{dt} \quad , k = L, V \tag{A.2}$$

where:

$c_{p,k}$: specific heat capacity at constant pressure of the liquid/vapour

E_k : liquid/vapour energy

h_k : liquid/vapour enthalpy

p : pressure

- The pressure p:

$$\frac{\mathrm{dp}}{dt} = -\frac{Z_1}{Z_2} \tag{A.3}$$

with

$$Z_1 = \frac{dM_V}{dt} v_V + \frac{\partial v_V}{\partial T_V}|_{\mathrm{p}} \frac{E_V}{c_{p,V}} + \frac{dM_L}{dt} v_L + \frac{\partial v_L}{\partial T_L}|_{\mathrm{p}} \frac{E_L}{c_{p,L}} \frac{1}{c_{p,V}} \left(T_V - \frac{\partial h_L}{\partial p}|_{T_V}\right) \frac{dp}{dt} \tag{A.4}$$

and

$$Z_2 = M_V \left[\frac{\partial v_V}{\partial p}|_{T_V} + \frac{1}{c_{p,V}} \frac{\partial v_V}{\partial T_V}|_{\mathrm{p}} \left(v_V - \frac{\partial h_V}{\partial p}|_{T_V}\right)\right] + M_L \left[\frac{\partial v_L}{\partial p}|_{T_L} + \frac{1}{c_{p,L}} \frac{\partial v_L}{\partial T_L}|_{\mathrm{p}} \left(v_L - \frac{\partial h_L}{\partial p}|_{T_L}\right)\right] \tag{A.5}$$

where:

v_V : vapour volume

v_L : liquid volume

- The phase velocities w_L and w_V

The phase velocities are obtained by the integration over a junction *j* connecting two control volumes i_1 and i_2.

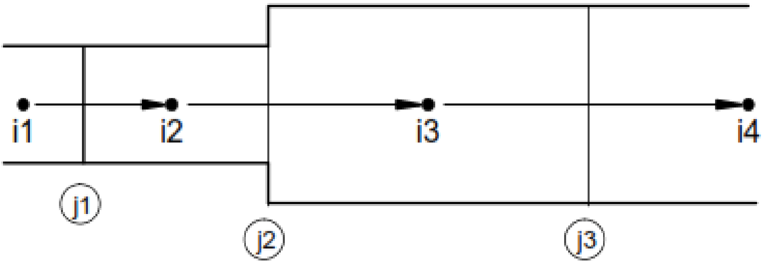

Figure A.2 Flow channel with area change

$$\int -\frac{\rho_k(j) w_k}{A}\frac{\partial A}{\partial t} ds = 0 \qquad , \qquad k = L, V \qquad (A.6)$$

with

$$\rho_k(j) = \frac{L_1 \rho_k(i_1) + L_2 \rho_k(i_2)}{L_1 + L_2} \qquad , \qquad k = L, V \qquad (A.7)$$

where:

A : flow area

ρ_k : liquid / vapour density

L_1 : i_1 length

4.2 5-Equation Model

The 5-equation model solves the system in the same way as the 6-equation model, obtaining the same results for the mass quality, the phase temperatures and the pressure within the control volumes. The main difference lies in the achievement of the mixture mass flow rate at the junctions instead of the phase velocities.

- The mixture mass flow rate G

Once again, the mass flow rate is obtained by the integration over a junction *j* connecting two control volumes i_1 and i_2.

$$\frac{dG}{dt} = \left(\int \frac{1}{A} ds\right)^{-1} \left[\Delta p_s + \Delta p_{MF} + \Delta p_{grav} + \Delta p_{fric} + \Delta p_\rho + \Delta p_I\right] \quad (A.8)$$

with

$$\Delta p_s = -\int \frac{\partial p}{\partial s} ds = p(i_2) - p(i_1)$$

$$\Delta p_{MF} = -\int \rho_m \vec{w}_m \frac{\partial \vec{w}_m}{\partial s} ds$$ momentum flux term

$$\Delta p_{WR} = -\int \frac{\partial \left(\alpha\,(1-\alpha)\frac{\rho_L \rho_V}{\rho_m} \vec{w}_R \vec{w}_R\right)}{\partial s} ds$$ relative velocity term

$$\Delta p_{grav} = \int \rho_m \vec{g} ds$$ elevation term

$\Delta p_{fric} = \int f_{W,m} ds$ friction and form loss pressure drop

$\Delta p_{\rho} = \int \vec{w}_m \frac{\partial \rho_m}{\partial t} ds$ density derivative term

$\Delta p_I = \int S_{I,m} ds$ source term, e.g. pump differential pressure

and

$$\rho_m = \alpha\, \rho_V + (1-\alpha)\, \rho_L$$

$$\vec{w}_m = \frac{1}{\rho_m}(\alpha\, \rho_V \vec{w}_V + (1-\alpha)\, \rho_L \vec{w}_L)$$

$$\vec{w}_R = \vec{w}_V - \vec{w}_L$$

where:

s : length coordinate

α : steam void fraction

$S_{I,m}$: momentum source

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