ANALYSIS OF REVERSE FLOW RESTRICTION DEVICE TO PREVENT FUEL DRYOUT DURING LOSS OF COOLANT AND INSTABILITY Accidents of Boiling Water Reactors

BY

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THESIS

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Abstract

This work introduces a new method to increase the safety of Boiling Water Reactors (BWRs) during the BWR instability and Loss of Coolant Accidents (LOCA). The method is based on a device called Reverse Flow Restriction Device (RFRD) and its purpose is to allow the flow in the forward direction, but prevent the flow in reverse direction which occurs in multiple accident scenarios. In this thesis, detailed TRACE/PARCS simulations have been used to investigate the effect of RFRD on the peak clad temperature during BWR instability and LOCA. The device is simulated in TRACE by using high friction coefficients for the reverse flow to ensure that only forward flow is allowed. The results demonstrate that by adding the RFRD device, flow reversal in fuel bundles could be substantially blocked and so the inlet flow reversal is thus prevented. The RFRD device also showed a modest impact on reducing the power oscillations. The use of RFRD device could prevent fuel dryout damage by preventing excessive high clad temperatures due to sustained dryout without timely rewetting. For LOCA, the device is capable of containing the coolant inside the core during the blowdown and when activating the emergency systems which keep the peak clad temperature at lower levels. Moreover, the RFRD achieved the reflood phase (when the saturation temperature of the clad is restored) earlier than without the RFRD. Sensitivity results demonstrated that for LOCA, high reverse flow friction coefficient is needed and hence the RFRD should be well-fitted to the lower tie plate to be able to sustain the high pressure caused by the large coolant flow during the blowdown phase of LOCA.
Dedication

To the almighty god “ALLAH” who guides and helps me in this work and throughout my whole life.

To my beloved parents who spent their life in growing me and my brothers, I could not accomplish anything in my life without your unlimited support and motivation.

To the soul of my science school teacher “Ahmad Qawasmeh” who encouraged me to seek a career in science and engineering and praised my talent. You and your words will be forever in my heart.
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I want to thank my close friends Ibrahim Jarrah, Rami Saeed, Zaid Taher, Mohammed Radaideh, Motaz Bani-Hani and many others for supporting me throughout the last years. I also want to thank my graduate colleagues Travis Mui, Guojun Hu, Guanfeng Gao, and Xu Wu for their help and support when I arrived to the department one year ago. I want to recognize the faculty and staff at Nuclear, Plasma and Radiological Engineering who taught and guided me since I came to this university.

I want to thank my brothers Mohammed and Ahmad Radaideh and my lovely sister Reem Radaideh for keeping my spirit high during this work. Finally, special thanks to my JUST professors: Professor Ziad Kodah, Professor Mahmoud Elgohari, and Professor Salaheddin Malkawi for their efforts in teaching and guiding me during the undergraduate years.
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Nomenclature

Variables

\(\alpha\) Gas volume fraction
\(\Gamma\) Interfacial mass-transfer rate
\(\rho\) Physical density
\(\Pi\) Full Stress tensor
\(\Delta T\) Temperature change
\(E_i\) Rate of energy transfer per unit volume across phase interfaces
\(e\) Internal energy
\(\vec{g}\) Gravity vector
\(M_i\) Rate of momentum transfer per unit volume across phase interfaces
\(q_d\) Direct heating (e.g. radioactive decay)
\(q'\) Heat flux
\(T\) Temperature without using the RFRD device
\(t\) Time
\(\vec{V}\) Velocity vector

Subscripts

\(Base\) Original case without using the RFRD device
\(g\) Gas
\(i\) Interface
\(l\) Liquid
\(RFRD\) Modified case with using the RFRD device
<table>
<thead>
<tr>
<th>Abbreviation</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>APRM</td>
<td>Average Power Range Monitor</td>
</tr>
<tr>
<td>ATWS</td>
<td>Anticipated Transient Without Scram</td>
</tr>
<tr>
<td>ATWSI</td>
<td>Anticipated Transient Without Scram with Instability</td>
</tr>
<tr>
<td>BWR</td>
<td>Boiling Water Reactor</td>
</tr>
<tr>
<td>CCFL</td>
<td>Counter Current Flow Limitation</td>
</tr>
<tr>
<td>DWO</td>
<td>Density Wave Oscillations</td>
</tr>
<tr>
<td>ECCS</td>
<td>Emergency Core Cooling System</td>
</tr>
<tr>
<td>HPCI</td>
<td>High Pressure Core Injection</td>
</tr>
<tr>
<td>KFAC</td>
<td>K-factor for forward flow (TRACE input)</td>
</tr>
<tr>
<td>KFACR</td>
<td>K-factor for reverse flow (TRACE input)</td>
</tr>
<tr>
<td>LBLOCA</td>
<td>Large Break Loss of Coolant Accident</td>
</tr>
<tr>
<td>LOCA</td>
<td>Loss of Coolant Accident</td>
</tr>
<tr>
<td>LPCI</td>
<td>Low Pressure Core Injection</td>
</tr>
<tr>
<td>LWR</td>
<td>Light Water Reactors</td>
</tr>
<tr>
<td>PARCS</td>
<td>Purdue Advanced Reactor Core Simulator</td>
</tr>
<tr>
<td>PCT</td>
<td>Peak Clad Temperature</td>
</tr>
<tr>
<td>PWR</td>
<td>Pressurized Water Reactor</td>
</tr>
<tr>
<td>RFRD</td>
<td>Reverse Flow Restriction Device</td>
</tr>
<tr>
<td>SBLOCA</td>
<td>Small break Loss of Coolant Accident</td>
</tr>
<tr>
<td>SNAP</td>
<td>Symbolic Nuclear Analysis Package</td>
</tr>
<tr>
<td>TRACE</td>
<td>TRAC/RELAP Advanced Computational Engine</td>
</tr>
<tr>
<td>U.S.NRC</td>
<td>United States Nuclear Regulatory Commission</td>
</tr>
</tbody>
</table>
CHAPTER 1. INTRODUCTION

Stability of Boiling Water Reactors (BWRs) is a phenomenon that continues to attract great interest since BWRs become unstable under certain conditions. Power oscillations become dangerous if the automatic reactor scram is lost, like in Anticipated Transient without Scram (ATWS) accidents where the reactor scram can be done only by the operator action. During the instability period, the clad temperature would increase to dangerous levels that might lead to fuel melt. BWRs are also prone to Loss of Coolant Accident (LOCA) when a break occurs in the recirculation loop. The break forces the reactor to scram and start depressurization. Moreover, the coolant is lost from the core, leaving the fuel uncovered with increased clad temperatures. Thus, it is obvious that instability event and loss of coolant accidents in BWRs could damage the fuel if the power oscillations are left without power suppression like reactor scram or the reactor is left without cooling during LOCA. Consequently, this study aims to analyze the concept of preventing the flow in the reverse (downward) direction in the fuel bundle to investigate the capability of mitigating the power oscillations without reactor scram during the instability event and containing the coolant inside the core during LOCA. Farawila [1] recommended a device to restrict the flow in the reverse direction to minimize the oscillations magnitude in BWRs instability events. In this thesis, a TRACE/PARCS model for Ringhals-1 power plant has been used to validate the potential effect of restricting the flow in the downward direction on mitigation of power and flow oscillations. In addition to that, a TRACE/PARCS LOCA model based on Oskarshamn-2 power plant design has been used to validate the potential effect of reducing the clad temperature during LOCA when restricting the flow in the reverse direction by using this
kind of device. This study could be valuable if it proves that oscillation magnitude could be minimized if the flow in the reverse direction is prevented since it will increase the safety of BWRs. If the clad temperature can be reduced during progression of the LOCA accident, this will increase the cooling efficiency of the safety systems because the fuel will be kept in low temperatures. The following sections provide description about BWR instability, LOCA, and system codes.

1.1 BWR and Flow Instabilities

The main reasons behind the instability in BWRs could be due to the two-phase flow, neutronics feedback, thermal-hydraulics feedback, and plant control system. Among all of these, when both neutronics and thermal-hydraulics feedbacks are added, coupled neutronics-thermal-hydraulics instabilities can occur and these instabilities are considered the dominant type of instabilities in BWRs, especially when the core is subjected to high power-low flow conditions [2]. The change in the void fraction as voids travel upward inside fuel bundles causes perturbations in the reactivity which affect the neutronics feedback in addition to the thermal-hydraulics feedback. Three main modes of instability might happen due to coupled neutronics-thermal-hydraulics feedback [2]:

- **Single channel instability**: occurs in a single fuel bundle in the core due to density-wave oscillations, pressure drop oscillations, or power disturbance in a single channel.

- **Core-wide oscillation (In-phase or global mode)**: it can be described as a single channel where both the power and the inlet flow rate have in-phase oscillations, and these oscillations occur coherently in all fuel bundles.
- **Regional oscillations (Out-of-phase or local mode):** the core behaves like two parallel channels where the core is divided into two regions and each region oscillates out-of-phase with respect to the other region. The flow of the first region channels oscillates out-of-phase with respect to the channels of the second region, and vice versa.

To prevent the oscillations in BWR or at least mitigate their effect, the parameters that affect the stability of BWR should be measured. Many parameters that can destabilize the reactor like distribution of pressure drop inside the core, axial power distribution, reactivity coefficients and subcooling at the core inlet [3]. Many instability events have been reported over the years, for example, Oskarshamn-2 power plant suffered from flow instabilities in 1999 due to a load rejection signal resulted in a turbine trip and loss of feedwater preheaters which left the core unstable due to the high power-low flow conditions. Kozlowski et al. [4] provided detailed description of this event in their paper. The high power conditions were detected by the Average Power Range Monitors (APRM) and the reactor was automatically scrammed without any fuel damage. The power oscillations during this accident have been recorded and the plot of the oscillations is shown in Figure 1.1 where the oscillations have been terminated by reactor scram at 252s.

The issue of BWRs attracted researchers in the previous years to understand, model and simulate this phenomenon. Kozlowski et al. [4] used TRACE/PACRS code to model the Oskarshamn-2 event and validate with the measured data from the plant. Their study demonstrated good agreement with the measured data including oscillation growth,
beginning of instability, and oscillation frequency. Validation of measured power oscillations with TRACE/PARCS (which will be used in this work) is shown in Figure 1.2.

Figure 1.1. Power oscillations in Oskarshamn-2 instability event [4]

Figure 1.2. Validation of measured power oscillations with TRACE/PARCS solution [4]

Since BWR instability is dominated by Density Wave Oscillations (DWO), this topic has to be described in this context. DWO are undesirable in boiling channel systems because sustained oscillations could cause mechanical vibration and system control problems.
Consider the heated channel that is shown in Figure 1.3 which shows a channel with constant pressure drop (boundary condition) where the coolant enters as subcooled water at the core inlet, and then flows upward through the channel [5]. The heat is applied to the channel to boil the flowing liquid. Water density varies as water moves up because of the two-phase flow. If a positive perturbation occurs in the inlet velocity (flow rate), a high density wave will develop and travel to the channel exit, making the pressure drop at exit (ΔP₂) to increase. However, in order to keep the pressure drop constant across the channel (boundary condition), pressure drop at the inlet (ΔP₁) decreases with same amount but with opposite sign. Now, the process is reversed as this decrease in ΔP₁ will cause the inlet velocity to drop and a low density wave will develop and travel to the channel exit. This will make ΔP₂ to decrease resulting in an increase in the inlet velocity and the cycle is starting over again leaving the channel in unstable flow behavior[2] [6].

![Figure 1.3. Schematic of the flow in a single channel in BWRs](image-url)

Figure 1.3. Schematic of the flow in a single channel in BWRs [5]
1.2 Loss of Coolant Accident (LOCA)

BWR is characterized by two-phase flow, where the steam-water mixture that exits from the core goes to steam separators and dryers which are located on the top of the core to separate steam and water (See Figure 1.4). The steam goes to turbine while the separated water flows downward to mix with the feedwater that comes from the turbine and the both of them return to the core. Figure 1.4 shows a sketch of BWR-6 and the flow path inside that reactor. Now, if a break occurs in the suction side of the recirculation pump, a Loss of Coolant Accident (LOCA) begins and it is one of the most challenging accidents for nuclear reactors because the coolant is lost from the core due to the break, resulting in increased fuel temperature which could lead to fuel damage and core melt. Once the break occurs, the reactor scrams and the core starts depressurization. Reverse flow in the broken loop occurs, and this flow will be lost through the break as well. In general, there are three phases for LOCA accident, the three phases are described in Table 1.1 [7], [8].

![Figure 1.4: Structure and recirculation flow path in general electric BWR-6](image)

Figure 1.4: Structure and recirculation flow path in general electric BWR-6 [8]
Table 1.1: Major events during LOCA phases [7], [8]

<table>
<thead>
<tr>
<th>Phase</th>
<th>Events</th>
</tr>
</thead>
<tbody>
<tr>
<td>Blowdown (0-30)</td>
<td>1- Reactor pressure and coolant inventory decreased rapidly, resulting in increase in fuel cladding temperature.</td>
</tr>
<tr>
<td></td>
<td>2- Core becomes fully uncovered.</td>
</tr>
<tr>
<td></td>
<td>3- During the early phase of the depressurization, the exiting coolant provides core cooling.</td>
</tr>
<tr>
<td></td>
<td>4- The High Pressure Core Injection (HPCI) injecting water but with small flow rate due to the high pressure to provide some heat removal.</td>
</tr>
<tr>
<td></td>
<td>5- The end of blowdown is defined to occur when the core spray and Low Pressure Core Injection (LPCI) reach rated flow.</td>
</tr>
<tr>
<td>Refill (30-40s)</td>
<td>1- LPCI part of Emergency Core Cooling System (ECCS) and core sprays on the core top are functioning, to provide a high flow rate of coolant.</td>
</tr>
<tr>
<td></td>
<td>2- During this phase, the core sprays and LPCI provide core cooling and supply liquid to refill the lower plenum of the reactor vessel.</td>
</tr>
<tr>
<td>Reflood (40-250s)</td>
<td>1- This phase begins when the lower plenum is refilled and the fuel assemblies are start to cool from bottom to top.</td>
</tr>
<tr>
<td></td>
<td>2- The clad retains its saturation temperature as the cladding quenches.</td>
</tr>
<tr>
<td></td>
<td>3- The LPCI and core sprays continue to reflood the core until all heat is removed.</td>
</tr>
</tbody>
</table>

LOCA is characterized by loss of coolant and core heat-up, as the temperature of the fuel increases, different physical phenomena come to the picture as listed in Table 1.2. Each phenomenon results in a release of different types of actinides and fission products depending on the temperature reached during the accident. According to the United States Nuclear Regulatory Commission (U.S. NRC), the clad temperature should not exceed the limit of 2200 °F (~1480 K) during LOCA in order to reuse the fuel again in operation.

Table 1.2: Physical phenomena that would occur during LOCA [7]

<table>
<thead>
<tr>
<th>Temperature (°C)</th>
<th>Physical Phenomena</th>
</tr>
</thead>
<tbody>
<tr>
<td>350</td>
<td>Approximate cladding temperature during normal operation</td>
</tr>
<tr>
<td>800-1450</td>
<td>Clad swelling due to internal gas pressure, some fission gases release, solid reaction between Zircaloy and stainless steel, clad swelling would block flow path.</td>
</tr>
<tr>
<td>1450-1500</td>
<td>Cladding-steam reaction produce excess energy, cladding become brittle, H₂ formed, steel alloy melts</td>
</tr>
</tbody>
</table>
Table 1.2 (cont.)

<table>
<thead>
<tr>
<th>Time</th>
<th>Event Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1550-1650</td>
<td>Zircaloy-steam reaction would become autocatalytic (i.e. feeding upon itself) unless Zircaloy is quenched by immersion.</td>
</tr>
<tr>
<td>1900</td>
<td>Zircaloy cladding melts</td>
</tr>
<tr>
<td>2150</td>
<td>Significant release of fission product from UO$_2$</td>
</tr>
<tr>
<td>2700</td>
<td>UO$_2$ and ZrO$_2$ both melt</td>
</tr>
</tbody>
</table>

As LOCA is one of the challenging accident in nuclear industry, significant research has been conducted in this area to investigate the core behavior during LOCA as well as the consequences of such accidents. Computational codes have been developed to analyze LOCA such as TRACE [9]. TRACE has been used to investigate the Counter-Current Flow Limitation (CCFL) which is a two-phase flow phenomenon occurs during LOCA where two phases (e.g. liquid water and steam) flow in opposite directions. The steam generated in the core flows upward preventing liquid coolant from reaching the fuel in the core. In general, CCFL affects the ability to re-introduce liquid coolant into a reactor in an LOCA accident [10], [11]. Experiments have been conducted in small scaled facilities to investigate the effect Small Break LOCA (SBLOCA) as well as Large Break LOCA (LBLOCA) on various parameters inside the core, and these experiments have been used to validate TRACE [12].

1.3 System Codes

In general, system codes are classified into two main categories: Frequency Domain Codes and Time Domain Codes. Frequency domain codes are designed to be simple to get faster computational time with acceptable accuracy by employing reduced order model like 1D thermal-hydraulics model or point reactor kinetics to simplify the phenomena [13]. The main concept behind these codes is linearization of the governing equations and using the Laplace transformation in frequency domain. Frequency domain codes are preferred when analyzing the linear stability behavior of BWRs and for steady-state problems [14].
However, time domain codes evolved to overcome the shortcomings of frequency domain codes and now are widely used to analyze the non-linear stability behavior of BWRs and for transient problems. These codes started evolving during the 80s when the computer computational power was improving such that the simplifications done by frequency domain codes can be avoided to obtain accurate nuclear power plant models. Consequently, time domain codes provide a sophisticated modeling like 3D spatial reactor kinetics, 3D parallel channel modeling, and reactor components modeling [14]. A comparison between the main characteristics of frequency and time domain codes are given in Table 1.3.

<table>
<thead>
<tr>
<th>Item</th>
<th>Frequency Domain Code</th>
<th>Time Domain Code</th>
</tr>
</thead>
<tbody>
<tr>
<td>Governing solution method</td>
<td>Reduced Order Model</td>
<td>Numerical scheme</td>
</tr>
<tr>
<td>Computational Time</td>
<td>Fast</td>
<td>Slow</td>
</tr>
<tr>
<td>Linear Capabilities</td>
<td>Only linear</td>
<td>Linear and non-linear</td>
</tr>
<tr>
<td>Usage</td>
<td>For linear stability and steady state problems</td>
<td>For non-linear stability and transient problems</td>
</tr>
<tr>
<td>Examples</td>
<td>LAPUR-5, ODYSY, HIBLE</td>
<td>RELAP5, RAMONA, TRACE</td>
</tr>
</tbody>
</table>

**1.4 Thesis Outline**

This thesis is divided into six chapters including this introductory chapter. The remaining Chapters of this work are organized as follows: Chapter 2 provides an overview of the codes that have been used in the thesis. Chapter 3 provides a description of the flow restriction device proposed for this study. TRACE/PARCS models for Ringhals-1 and Oskarshamn-2 and the modeling details used in this work are described in details in Chapter 4. The results obtained from this study along with discussion of the results are presented in Chapter 5. Finally, the conclusions from this work and any possible future work that can be built on this study are given in Chapter 6.
CHAPTER 2. CODES AND TOOLS DESCRIPTION

2.1 TRACE (TRAC/RELAP Advanced Computational Engine)

TRACE is the latest best-estimate reactor systems code developed by the U.S. Nuclear Regulatory Commission (U.S.NRC) for analyzing steady state and transient thermal-hydraulics systems for light water reactors. U.S.NRC combined its main four codes (TRAC-P, TRAC-B, RELAP5, and RAMONA) into one modernized and advanced computational code. Originally, TRACE has been designed to perform best-estimate analyses of LOCAs, but it can simulate other phenomena in Light Water Reactors (LWRs) like operational transients, ATWS, two-phase flow, heat transfer problems, and others. Strictly speaking, TRACE can be seen as a two-phase, two-fluid solver where the two-fluid six conservation equations (i.e. 2 continuity, 2 momentum, and 2 energy equations) are solved for liquid and vapor phases of water. If the user wants tracking of non-condensable gases and dissolved solute in liquids, two additional equations are solved. To solve these conservation equations, additional constitutive relations and jump conditions are required. The six field conservation equations coupled with interface jump conditions can be expressed as the following [9]:

Continuity Equations

\[
\frac{\partial [(1 - \alpha) \bar{\rho}_l]}{\partial t} + \nabla \cdot [(1 - \alpha) \bar{\rho}_l \bar{V}_l] = -\bar{F} \tag{2.1}
\]

\[
\frac{\partial [\alpha \bar{\rho}_g]}{\partial t} + \nabla \cdot [\alpha \bar{\rho}_g \bar{V}_g] = \bar{F} \tag{2.2}
\]

Momentum Equations
\[
\frac{\partial[(1-\alpha)\bar{\rho}_l \bar{V}_l]}{\partial t} + \nabla \cdot (1 - \alpha) \bar{\rho}_l \bar{V}_l \bar{V}_l = \nabla \cdot [(1 - \alpha) \bar{\Pi}_l] + (1 - \alpha) \bar{\rho}_l \bar{g} - \bar{M}_l
\]  

(2.3)

\[
\frac{\partial[\alpha \bar{\rho}_g \bar{V}_g]}{\partial t} + \nabla \cdot \alpha \bar{\rho}_g \bar{V}_g \bar{V}_g = \nabla \cdot [\alpha \bar{\Pi}_g] + \alpha \bar{\rho}_g \bar{g} + \bar{M}_l
\]  

(2.4)

**Energy Equations**

\[
\frac{\partial[(1-\alpha)\bar{\rho}_l (e_l + \frac{\bar{V}_l^2}{2})]}{\partial t} + \nabla \cdot [(1 - \alpha) \bar{\rho}_l (e_l + \frac{\bar{V}_l^2}{2}) \bar{V}_l] = -\nabla \cdot [(1 - \alpha) \bar{q}_l] + \nabla \cdot [(1 - \alpha) \bar{\Pi}_l, \bar{V}_l] + (1 - \alpha) \bar{\rho}_l \bar{g} \cdot \bar{V}_l - \bar{E}_l + \bar{q}_{dl}
\]  

(2.5)

\[
\frac{\partial[\alpha \bar{\rho}_g (e_g + \frac{\bar{V}_g^2}{2})]}{\partial t} + \nabla \cdot [\alpha \bar{\rho}_g (e_g + \frac{\bar{V}_g^2}{2}) \bar{V}_g] = -\nabla \cdot [\alpha \bar{q}_g] + \nabla \cdot [\alpha \bar{\Pi}_g, \bar{V}_g] + \alpha \bar{\rho}_g \bar{g} \cdot \bar{V}_g + \bar{E}_l + \bar{q}_{dg}
\]  

(2.6)

where the over bar represents a time average, and the subscripts of “g” and “l” represent gas and liquid term, respectively. \(\bar{V}\) is the velocity vector, \(\bar{\rho}\) is the physical density, \(\alpha\) is the gas volume fraction, \(e\) is the internal energy, \(\bar{g}\) is the gravity vector, and \(\bar{\Pi}, \bar{E}_l\), and \(\bar{M}_l\) represent the contribution of time averaged jump conditions to transfer of mass, energy and momentum, respectively. In addition, \(q'\) is conductive heat flux, \(q_i\) is the heat flux by direct heating and \(\Pi\) is the full stress tensor. The other variables that include two or more quantities with a bar on the top (e.g. \(\bar{V}_l \bar{V}_l, (e_g + \frac{\bar{V}_g^2}{2}) \bar{V}_g\)) can be resolved into their constituent variables using numerical methods.

At this point it is possible to mention that TRACE is a 1D code which means that the above conservation equations (Eq. 2.1 – Eq. 2.6) need to be averaged on area to get the 1D
conservation equations. TRACE tackles the differential equations of the flow and heat transfer as follows:

- Finite volume scheme is used to solve the partial differential equations of two-phase flow and heat transfer.
- Semi-implicit time difference scheme is used to solve heat transfer equations.
- Multi-step time-differencing procedure is used to solve the fluid dynamics equations in the 1-D and 3-D.
- Newton-Raphson iteration method is used to tackle the coupled, nonlinear equations of the hydrodynamic phenomena.
- Direct matrix inversion is used to solve the linearized form of equations for the hydrodynamic phenomena.

TRACE is a time domain code which means that the code usually takes a long time to run (See Table 1.3) depending on the complexity of the problem. Other factors like number of mesh cells and the timestep size also play a major role in affecting the execution time. A powerful technique called SETS (Stability-Enhancing Two-Step) is used by TRACE to adjust the time step size, so that a large time step could be used in slow transients in which the phenomena becomes easy to evaluate. This leads to a significant reduction in the computational cost for slow-developing accident. Automatic restart capabilities are available in TRACE along with coupling with reactor kinetics code PARCS (See Section 2.3).

In summary, TRACE has been selected here because it has many features that fulfill the requirements of this study including:

- TRACE is already designed for LOCA modeling, and it can also simulate the
instability phenomena in BWRs with good accuracy since it can be coupled with reactor physics code PARCS.

- Effect of oscillations on power, flow rate, and clad temperature can be studied through TRACE.
- TRACE is capable both the forward and reverse flow with the option of controlling the friction coefficient for both directions.
- TRACE is accurate in predicting clad temperature excursion in case of exceeding critical heat flux conditions which means that the effect of using the device can be demonstrated.

2.2 PARCS (Purdue Advanced Reactor Core Simulator)

PARCS is a 3D reactor core simulator used to solve both multi-group neutron diffusion and SP3 transport equations. PARCS is capable to solve steady-state and transient problems for orthogonal and non-orthogonal geometries. PARCS needs cross-section data to solve the differential equations which can be imported from lattice physics codes like TRITON or CASMO. A separate module called GenPMAXS is used to process the cross-sections generated by lattice physics codes and convert it into PMAXS format that can be read by PARCS. PARCS has many reactor analysis features including (1) eigenvalue calculations which can be used as initial state for transient calculations, (2) reactor kinetics, (3) Xenon transient, (4) decay heat, (5) pin power, and (6) depletion calculations. PARCS code capabilities include not only LWRs but also Heavy Water Reactors and High Temperature Gas Reactors. PARCS can be run in standalone mode (without coupling) or coupled with
thermal-hydraulic codes like TRACE or RELAP5 to provide the flow field and temperature for neutronics feedback. Further details about PARCS can be found in the code manual [15].

2.3 TRACE/PARCS

TRACE deals with the neutronics on a core-wide basis by solving the point reactor kinetics. This makes TRACE limited when dealing with transients where large power excursions could happen like in BWR instability or control rod ejection accidents. This limitation can be overcome by coupling TRACE with PARCS where the spatially local neutronic response can be modeled. For coupling, the user has to prepare two input files, one for TRACE and one for PARCS, with a coupling file to couple neutronic nodes to thermal hydraulic channels. Coupled TRACE/PARCS can be executed through the TRACE executable file (See Figure 2.1).

![Figure 2.1. Stages of execution of coupled TRACE/PARCS](image-url)
A common approach is used when running a coupled TRACE/PARCS and it includes three stages, (1) TRACE stand-alone steady-state, (2) TRACE/PARCS coupled steady-state, and (3) TRACE/PARCS coupled transient. Figure 2.1 summarizes the stages for coupling between TRACE and PARCS codes. The three stages are connected by restart files for both TRACE and PARCS. After steady-state calculations, two restart files along with two input files for TRACE and PARCS could be used to initiate coupled TRACE/PARCS transient calculations.
CHAPTER 3. REVERSE FLOW RESTRICTION DEVICE (RFRD)

The current suppression of instability event in BWR depends on the ability of the operator or automatic protection systems like reactor scram to interfere and terminate the event. However, in the case of Anticipated Transient without Scram with Instability (ATWSI) where the ability of automatic scram is lost by definition, the only way to terminate the event depends on operator recognition of the accident and correct response to reduce feedwater flow to lower the water level in the reactor vessel, and ultimately boron injection. Therefore, this solution is highly dependent on operator response and hence subjected to human error. For LBLOCA, when the break happens, the core starts depressurization, and once the core pressure drops below a certain level (about 0.2 MPa), the core spray and ECCS begin working. The activation of the ECCS pumps to start reflooding the core takes time. In case of delay of activation of the pumps, the core is left without cooling and the fuel bundles become hotter. It is clear that the above problems would benefit from practical solutions to keep the reactor safe during these types of accidents. In this thesis, a hardware device is proposed to overcome the problems that would occur during LOCA and ATWSI, and an introduction to this device is presented in this Chapter.

3.1 Device Configuration

The device should preserve the stability characteristics of the host fuel where the inception of instability or LOCA accidents are not prevented by this RFRD device. The device may mitigate the adverse effects during these two accidents by limiting the growth
of the oscillations during the instability accident and keeping more coolant in the core during LOCA. The device is called Reverse Flow Restriction Device (RFRD) [16], and it is introduced to the lower tie plate of the fuel assembly to act as a check valve. When the flow is going in the (forward) direction, the valve opens and the flow is allowed. Figure 3.1 shows an example of flow in a pipe where the forward flow is unobstructed. However, when the flow goes in the reverse direction, the valve will close resulting in blocking, or substantially reducing the flow in the reverse direction. Figure 3.1 also demonstrates that when the flow goes reversal, the screen closed to substantially reduce the flow. In both cases, the device should not affect the hydraulic characteristics of the core to assure proper utilization of the device.

![Figure 3.1](image1.png)  (a)  
Figure 3.1. Flow in a pipe controlled by a check valve where (a) forward flow is allowed and (b) reverse flow is prevented [17].

RFRD concept is similar to that shown in Figure 3.1 where the limiting reverse flow magnitude can be achieved by equipping the fuel bundle inlet with RFRD. Farawila [1], [16] in his paper was first to recommend this device. Figure 3.2 demonstrates the lower tie plate of a fuel assembly for BWR without the device on the left and it shows one- half of the lower
tie plate but with the proposed device equipped on the right. RFRD consists of a grid of check valves for each fuel channel inside the fuel assembly. The sketch shows that RFRD consists of two parallel plates where each plate has holes forming a cavity inside, and the screen is free to move between the plates.

![Figure 3.2](image.jpg)

(a) Figure 3.2. Lower tie plate of the fuel bundle: (a) without RFRD and (b) with RFRD inserted to prevent the reverse flow [1]

Figure 3.3 shows isometric and top view of the screen that moves between the plates during the flow direction changes. The screen has a grid structure and it consists of array of disks aligned with the holes in the plates. The holes in the grid have tabs to keep the forward flow unobstructed, while during the reverse flow the screen goes down blocking the holes beneath it. The disks should be well-fitted to the screen to assure high friction factor in the reverse direction to block the flow and to avoid releasing of loose parts during large
blowdown of water (e.g. LBLOCA). The device shown in Figure 3.3 is designed for 9x9 type of BWR fuel bundles.

During the forward flow the coolant flow exerts a force to lift the screen into the upper plate. In the reverse flow, the pressure exerted by the coolant disappears and hence the screen drops to the down position and rests against the lower plate (See Figure 3.4). The floating screen switches between the up (open) and down (closed) position based on the flow direction and this distance is very small to eliminate high speed movement which means the opening and closure of the flow path are not abrupt but rather smooth. Figure 3.4 shows a vertical cut drawing of the lower tie plate structure including the floating screen between two parallel plates. The screen on the left figure is in the up position which is the normal
position with the flow in the upward direction, while the figure on the right shows the screen in the down position to prevent the reverse flow which happens during accidents conditions.

![Diagram showing Forward Flow is Allowed and Reverse Flow is Prevented](image)

**Figure 3.4.** Position of the RFRD against the upward and downward (reverse) flow

There are some constraints that should be considered when designing this device to assure proper utilization of the device during reactor operation [1]:

1- The probability of failure of this device in the blocked position should be negligible.
2- The floating screen should be rigid and well-designed to avoid release of loose parts which might cause further problems like blocking the forward flow in the fuel bundles, or damaging the cladding.
3- Flow tests under various flow conditions should be carried out to ensure that the flow-induced vibrations caused by the device are negligible.
The introduction of the RFRD should not adversely affect the flow in the bundles without RFRD (i.e. if the device is used only in specific bundles) or any other aspects of the plant operation.

3.2 RFRD for Instability Events

During the BWR instability accident, the reactor enters the unstable state and the inlet mass flow rate of the coolant in each fuel bundle will start to oscillate about its average value. If the inlet mass flow oscillation amplitude is large enough, then flow reversal occurs in this fuel bundle. At the beginning of the oscillations, the waves take a sinusoidal shape and the flow oscillates as a function of time in the positive region where the peaks and valleys are both positive (see Figure 3.5 in the period 110-120 s). As time approaches, the oscillation magnitude grows and the peak value increases. In this case, the screen remains in the up position as long as the minimum flow value remains positive (i.e. flow remains in the upward direction) and exerts small pressure to lift the screen into the upper plate and maintain the flow through the open holes. When the oscillation magnitude of the inlet flow increases such that the flow is insufficient to maintain the screen in the up position, i.e. flow reverses in direction (see Figure 3.5), the screen shifts to the lower plate and the disk blocks the holes beneath them. After that, the flow oscillation will again grow to positive value where the hydraulic forces and the forward flow lift the screen to the up position and the holes will open leaving the upward flow unobstructed. The flow cycle above keeps repeating and the screen switches between up and down positions depending on the flow direction. In summary, the main goal of the device in this type of accidents is allowing forward flow and preventing reverse flow during the oscillatory cycle.
3.3 RFRD for LOCA

During LBLOCA (e.g. large break in the recirculation loop), a massive amount of coolant leaves the core through the break in a short time. This makes the fuel to uncover quickly and become hotter due to the considerable loss of cooling (i.e. after occurrence of the critical heat flux). Once the break happens, the coolant direction reverses, and the upward flow becomes downward allowing the coolant to flow out from the inlet of the fuel assembly to the lower plenum, then leaving the vessel out to the containment.

After the break during LOCA, the core pressure decreases continuously, and the low pressure cooling systems take a long time to activate because it can operate only at relatively small pressure (about 0.2 MPa), while the BWR core pressure is approximately 7.0 MPa.
During the time of depressurization, the fuel is subjected to overheat and possibly melt down until the core sprays and LPCI are activated to provide adequate heat transfer to remove the heat deposited in the core region. If RFRD is used at the fuel bundle inlet, the depletion of coolant from the core bottom both during blowdown and when ECCS is activated may be prevented, which leads to reduction of clad temperature rise and significant improvements in LOCA performance. Some plants that are constrained by LOCA, the regulatory bodies force these plants to operate at reduced power level to minimize the effects of LOCA. If this device proves to be effective during LOCA, regulatory constraints can be avoided which would result in higher power operation, better fuel utilization, and improvement in economics and safety of the plant [18].
CHAPTER 4. METHODOLOGY AND MODELING

TRACE is a component-based code where the user can represent the physical system by a combination of pre-defined components. Each component can be nodalized by the user by specifying how many nodes are in that component. A brief list of the TRACE components that are relevant to this study is listed in Table 4.1. The VESSEL component is used to represent the pressure vessel, and CHAN component is used to model the fuel assemblies inside the core region. 1-D PIPE component is used to represent the piping system in the reactor. VALVE is used to control the flow in the system through a user-defined hydraulic diameter and flow area. FILL and BREAK components are used to impose boundary conditions for mass flow rate and pressure, respectively. SEPD component is used to represent the steam separators and dryers in BWRs.

Table 4.1: Brief list of TRACE components that are used in this study

<table>
<thead>
<tr>
<th>TRACE notation</th>
<th>SNAP representation</th>
<th>TRACE notation</th>
<th>SNAP representation</th>
</tr>
</thead>
<tbody>
<tr>
<td>VESSEL (Core vessel)</td>
<td><img src="image1" alt="SNAP representation of VESSEL" /></td>
<td>PUMP (Pump)</td>
<td><img src="image2" alt="SNAP representation of PUMP" /></td>
</tr>
<tr>
<td>PIPE (Pipe)</td>
<td><img src="image3" alt="SNAP representation of PIPE" /></td>
<td>VALVE (Valve)</td>
<td><img src="image4" alt="SNAP representation of VALVE" /></td>
</tr>
<tr>
<td>BREAK (Pressure boundary condition)</td>
<td><img src="image5" alt="SNAP representation of BREAK" /></td>
<td>FILL (Flow boundary condition)</td>
<td><img src="image6" alt="SNAP representation of FILL" /></td>
</tr>
<tr>
<td>SEPD (Steam Separator)</td>
<td><img src="image7" alt="SNAP representation of SEPD" /></td>
<td>CHAN (BWR fuel assembly)</td>
<td><img src="image8" alt="SNAP representation of CHAN" /></td>
</tr>
</tbody>
</table>
4.1 RFRD Implementation

In TRACE, two different approaches can be used to implement the RFRD device which was described in Chapter 3. First is applying a check valve at the bundle inlet of the fuel bundle. The issues for this approach are: (1) the device will keep changing its position from up to down through instability accident and (2) the RFRD device will be introduced to all fuel bundles (~ 400 – 700 assemblies) and both of these issues will increase the complexity of the model and make it computationally expensive for TRACE. Therefore, this approach will not be used here. The second method to apply RFRD can be done by taking advantage of the capability of TRACE to simulate both the forward and reverse flow. If a large friction coefficient is used at the bundle inlet for the reverse flow, and leaving the forward flow friction coefficient as it is (with considering the increase in friction in the forward direction due to the device), the RFRD effect may be simulated in TRACE. In this case, the forward flow is allowed and once the flow reverses in direction, the high friction coefficient will prevent the reverse flow. In TRACE, two different friction coefficients in the form of additive loss coefficients or k-factor can be used as follows (see Figure 4.1):

1- FRIC or KFAC: friction coefficient or k-factor in the forward direction.

2- FRICR or KFACR: friction coefficient or k-factor in the reverse direction.

The fuel channel is divided into 28 axial nodes (See Figure 4.1). The RFRD device is applied by increasing the value of KFACR at the first inlet node and this is sufficient to ensure blocking the reverse flow. The activation of the reverse flow friction coefficient in TRACE can be done by editing the appropriate NAMELIST variables in TRACE.
Figure 4.1. Fuel channel nodalisation in TRACE and location of inlet friction factor coefficient

4.2 Ringhals-1 TRACE Model for Instability

Ringhals-1, a Swedish BWR reactor, has been selected to simulate the BWR instability accident since the reactor is large and it has large number of fuel bundles, and the flow rate reversal could be observed during instability events. General plant data and operating conditions are listed in Table 4.2. The TRACE/PARCS model is based on OECD/NEA Ringhals-1 Stability Benchmark [19], [20]. However, the model in the benchmark is modified in this thesis to induce in-phase instability, and increase the oscillation amplitude to investigate the effect of preventing flow in the reverse direction on the core stability.
Specific channels have been selected where the flow rate oscillates from positive to negative values (i.e. flow rate reversal) to clearly demonstrate the effectiveness of the device.

Table 4.2: Ringhals-1 vessel geometry and operating conditions

<table>
<thead>
<tr>
<th>Item</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rated thermal power (MWth)</td>
<td>1700</td>
</tr>
<tr>
<td>Dome pressure (MPa)</td>
<td>7.0</td>
</tr>
<tr>
<td>Steam flow rate (kg/s)</td>
<td>855</td>
</tr>
<tr>
<td>Rated recirculation flow (kg/s)</td>
<td>7800</td>
</tr>
<tr>
<td>Vessel height (m)</td>
<td>20</td>
</tr>
<tr>
<td>Wall thickness (mm)</td>
<td>134</td>
</tr>
<tr>
<td>Equivalent core diameter (mm)</td>
<td>2975</td>
</tr>
<tr>
<td>Equivalent core height (mm)</td>
<td>4398</td>
</tr>
<tr>
<td>Number of fuel bundles</td>
<td>648</td>
</tr>
</tbody>
</table>

The nodalization scheme of Ringhals-1 TRACE model is shown in Figure 4.2. The reactor model consists of the following parts:

1- Reactor vessel represented by VESSEL component.

2- 648 fuel bundles represented by 325 BWR CHAN components with one-half core symmetry.

3- One recirculation loop consists of two pipes and one recirculation pump. Each loop has an outtake pipe connected to the downcomer, through which the coolant is redirected to a recirculation pump.

4- Turbine system consists of a PIPE, a VALVE, and a BREAK to impose a pressure boundary condition.

5- Feedwater system consists of a PIPE component and a FILL to impose a flow rate boundary condition.

6- One steam separator represented by SEPD component.
Figure 4.2. Ringhals-1 TRACE model nodalization to simulate BWR instability accident.

The reactor vessel is modeled by 3D VESSEL component with 11 axial cells, 2 radial cells, and 1 azimuthal sector. Each CHAN component is modeled with 28 uniform axial nodes. Both forward and reverse flow friction coefficients have been added to each channel component to model RFRD device. PARCS is supplemented with cross-section, geometry and coupling files. TRACE/PARCS Ringhals-1 model was benchmark against the measured data [20].

TRACE is coupled with a 3D PARCS, a neutronics code. PARCS is capable of calculating the kinetic behavior of the core and other neutronics calculations to predict the response of the reactor in steady-state and transient conditions. Control rod movements are controlled with PARCS while flow rate is controlled by TRACE. The reactor instability
condition is induced by the low flow-high power condition. First, the pump is tripped after 10 seconds and the natural pump coastdown begins and hence the flow rate drops. To increase the core power to large values, two control rod banks are withdrawn from the core 70 seconds after the pump trip. After that, the core becomes unstable since the flow rate is low and the power is high which makes all thermal-hydraulic parameters (e.g. flow rate, power, temperature, pressure) oscillate. The results from the instability are calculated with using the RFRD at the bundle inlet and they are compared with the cases without using the RFRD. The accident scenario parameters used during TRACE simulation are listed in Table 4.3.

<table>
<thead>
<tr>
<th>Item</th>
<th>value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pump trip Time (s)</td>
<td>10</td>
</tr>
<tr>
<td>Control Rod Movement Time (s)</td>
<td>80</td>
</tr>
<tr>
<td>Reverse flow Friction Coefficient (KFACR)</td>
<td>500</td>
</tr>
<tr>
<td>Device axial position</td>
<td>Axial node 1</td>
</tr>
<tr>
<td>Time step (Min-Max) (s)</td>
<td>10^{-6} - 0.1</td>
</tr>
<tr>
<td>Initial relative power</td>
<td>75%</td>
</tr>
</tbody>
</table>

4.3 Oskarshamn-2 TRACE Model for LOCA

A TRACE model based on Oskarshamn-2 benchmark [4] has been developed in this study to investigate the effect of using RFRD to improve the safety of BWR during LOCA. The benchmark is based on transient measurements of the February 25, 1999 event at the Oskarshamn-2 Nuclear Power Plant. In this study Oskarshamn-2 model is used to simulate LOCA. The reason for selecting Oskarshamn-2 is that the core size is smaller than Ringhals-1, the number of fuel bundles in Oskarshamn-2 reactor is 444 compared to the 648 bundles in Ringhals-1. In general, small cores are less complicated and less computationally
expensive while the improvement achieved by RFRD can still be observed. The core thermal-hydraulics and geometric data are given in Table 4.4, and the core vessel geometry is shown in Figure 4.3.

![Figure 4.3. Sketch of Oskarshamn-2 vessel](image)

**Table 4.4: Oskarshamn-2 vessel geometry and operating conditions**

<table>
<thead>
<tr>
<th>Item</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rated thermal power (MWth)</td>
<td>1700</td>
</tr>
<tr>
<td>Dome pressure (MPa)</td>
<td>7.0</td>
</tr>
<tr>
<td>Steam flow rate (kg/s)</td>
<td>900</td>
</tr>
<tr>
<td>Rated recirc. Flow (kg/s)</td>
<td>7700</td>
</tr>
<tr>
<td>Internal height (m)</td>
<td>20</td>
</tr>
</tbody>
</table>
Table 4.4 (cont.)

<p>| | |</p>
<table>
<thead>
<tr>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Weight (kg)</td>
<td>530,000</td>
</tr>
<tr>
<td>Wall thickness (mm)</td>
<td>134</td>
</tr>
<tr>
<td>Equivalent core diameter (mm)</td>
<td>3672</td>
</tr>
<tr>
<td>Equivalent core height (mm)</td>
<td>3712</td>
</tr>
<tr>
<td>Number of fuel bundles</td>
<td>444</td>
</tr>
</tbody>
</table>

The nodalization scheme of Oskarshamn-2 TRACE model is shown in Figure 4.4. The reactor model consists of the following components:

1- Reactor vessel represented by VESSEL component.

2- 444 fuel bundles represented by 222 BWR CHAN components in one-half core symmetry.

3- One recirculation loop has an outtake pipe connected to the downcomer, through which the coolant is redirected to a recirculation pump. The coolant is pumped via the intake pipe to the lower plenum in the vessel.

4- LOCA break is connected to the recirculation pipe that connected to the pump discharge. The break size is controlled by area fraction of a valve connected directly to the break.

5- Core spray and LPCI consist of a FILL and a PIPE component.

6- Turbine system consists of a PIPE, a VALVE, and a BREAK to impose a pressure boundary condition.

7- Feedwater system consists of a PIPE component and a FILL to impose a flow rate boundary condition.

8- One steam separator represented by SEPD component.
Figure 4.4. Oskarshamn-2 TRACE model nodalization to simulate BWR LOCA.

The reactor vessel is modeled by 3D vessel component with 15 axial cells, 2 radial cells, and 1 azimuthal sector. Each CHAN component is modeled with 28 uniform axial nodes. Both forward and reverse flow friction coefficients have been added to each channel component to model RFRD device. PARCS is supplemented with cross-section, geometry and coupling files. LBLOCA is selected as a base case to test the model, the results for this case including the pressure, power, break flow and other variables are reported in Appendix A. The results with and without the RFRD have been calculated using this TRACE/PARCS model.
The device effect for LOCA is determined through the Peak Clad Temperature (PCT) change calculated through:

\[ \Delta T = T_{\text{Base}} - T_{\text{RFRD}} \]  

A positive \( \Delta T \) means that the RFRD device is effective and PCT is reduced.

### 4.2.1 LOCA Transients

LOCA transient model includes all the changes in operation that occur after the break as appropriate systems have to be activated in order to mitigate adverse effects on the fuel. The LOCA is initiated by opening the BREAK component which is connected to the pipe in the recirculation loop through a valve. The initiation of LOCA is set at \( t = 15 \text{s} \). After 0.5s, control rod banks are inserted to shutdown the reactor and the pump is tripped. Three other events after that include the closure of the feedwater, closure of the turbine valve, and the activation of the emergency cooling systems at low pressures. Table 4.5 lists the main events following the LOCA break with the time for each event.

<table>
<thead>
<tr>
<th>Event</th>
<th>Time</th>
</tr>
</thead>
<tbody>
<tr>
<td>Break valve open time</td>
<td>15.0 s</td>
</tr>
<tr>
<td>Pump trip</td>
<td>15.5 s</td>
</tr>
<tr>
<td>CR bank insertion</td>
<td>15.5 s</td>
</tr>
<tr>
<td>Closure of feedwater flow</td>
<td>16 s</td>
</tr>
<tr>
<td>Closure of turbine valve</td>
<td>17 s</td>
</tr>
<tr>
<td>ECCS Activation</td>
<td>( t @ P \leq 0.2 \text{ MPa} )</td>
</tr>
</tbody>
</table>

The timing for activating and closing different systems during LOCA such as feedwater, turbine, and core spray systems should be described. For turbine valve, it is assumed that the turbine valve receives the signal in 0.2s followed by an additional 0.8s to start the procedure of closing the valve. 0.5s is needed after that to close the turbine valve itself. Closure of
feedwater is done in 1s where the feedwater flow is dropped from rated flow to zero. The activation of the emergency systems to cool the reactor fuel is determined by means of the pressure value in the core instead of time. When the pressure inside the core drops to a value of about 2 bar (0.2 MPa), core sprays and LPCI are activated. Timetables for the closure of turbine valve and feedwater, and the activation of ECCS are summarized in Table 4.6.

Table 4.6: Timetable for closure and activation of various systems during LOCA

<table>
<thead>
<tr>
<th>Event</th>
<th>Time</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>Turbine valve closure</td>
<td>15.0 s</td>
<td>Open</td>
</tr>
<tr>
<td></td>
<td>16.0 s</td>
<td>Open</td>
</tr>
<tr>
<td></td>
<td>16.5 s</td>
<td>Closed</td>
</tr>
<tr>
<td>Feedwater closure</td>
<td>15.0 s</td>
<td>Open</td>
</tr>
<tr>
<td></td>
<td>16.0 s</td>
<td>Closed</td>
</tr>
<tr>
<td>ECCS activation</td>
<td>15.0 s</td>
<td>Closed</td>
</tr>
<tr>
<td></td>
<td>t @ P≤0.2 MPa</td>
<td>Open</td>
</tr>
</tbody>
</table>

The purpose of the main recirculation system is to provide the core with adequate coolant flow at all power levels. A single recirculation loop is used to provide the core flow (See Figure 4.4). The pump is able to provide the core with a 2.55 m$^3$/sec flow rate at a temperature of 274 °C, and elevate the water to 55 m height. The pump curves including pump head and relative hydrodynamic torque that are used in TRACE simulation are obtained from the validated model of Oskarshamn-2 [4].

Choked or critical flow is occurring in cases where fluid moves from higher pressure volume at speed limited only by speed of sound for fluid. This situation occurs in LOCA as the break mass flow depends on the condition of the main system not on the pressure outside which is the containment pressure. TRACE is able to predict choked flow, and so in this thesis the choked flow model is activated with default TRACE parameters only at the BREAK component.
During normal operation the reactor is operating at full power and this value stays constant during steady-state calculations. In transient calculations, the power stays constant for the first 15s, but when the break occurs, the reactor scrams and the power decreases abruptly as it will decrease by 90% in about 2s. In this thesis, because the power changes in LOCA are not as critical as in BWR instability, PARCS is not used for power calculations. Instead, a power profile describes how the power is changing with time after scram should be provided for TRACE to do the transient calculations. Table 4.7 shows the time behavior of the decay power after scram based on modified ANS standards which is reported in this thesis [21]. It is clear from Table 4.7 that most of the core power is disappeared in 2s after scram, and the remaining heat source inside the core comes from the decay of the fission products.

Table 4.7: Decay power variation after shutdown [21]

<table>
<thead>
<tr>
<th>Time (sec)</th>
<th>Fraction</th>
<th>Power (MW)</th>
</tr>
</thead>
<tbody>
<tr>
<td>15</td>
<td>1</td>
<td>1.705E+09</td>
</tr>
<tr>
<td>15.1</td>
<td>0.525</td>
<td>8.951E+08</td>
</tr>
<tr>
<td>16</td>
<td>0.134</td>
<td>2.285E+08</td>
</tr>
<tr>
<td>17</td>
<td>0.103</td>
<td>1.756E+08</td>
</tr>
<tr>
<td>19</td>
<td>0.077</td>
<td>1.313E+08</td>
</tr>
<tr>
<td>21</td>
<td>0.074</td>
<td>1.262E+08</td>
</tr>
<tr>
<td>23</td>
<td>0.07</td>
<td>1.194E+08</td>
</tr>
<tr>
<td>25</td>
<td>0.068</td>
<td>1.159E+08</td>
</tr>
<tr>
<td>35</td>
<td>0.061</td>
<td>1.040E+08</td>
</tr>
<tr>
<td>55</td>
<td>0.0526</td>
<td>8.968E+07</td>
</tr>
<tr>
<td>75</td>
<td>0.0485</td>
<td>8.269E+07</td>
</tr>
<tr>
<td>95</td>
<td>0.046</td>
<td>7.843E+07</td>
</tr>
<tr>
<td>115</td>
<td>0.044</td>
<td>7.502E+07</td>
</tr>
<tr>
<td>215</td>
<td>0.0366</td>
<td>6.240E+07</td>
</tr>
<tr>
<td>415</td>
<td>0.0315</td>
<td>5.37E+07</td>
</tr>
<tr>
<td>615</td>
<td>0.0284</td>
<td>4.84E+07</td>
</tr>
</tbody>
</table>
4.2.2 Break Size

The critical factor that determines LOCA scenario and severity is the break size. In general, as the break size increases, more coolant leaks, and the core uncovers more rapidly resulting in a higher clad temperature. The terminology that will be used in this study is listed in Table 4.8, as four categories will be used to specify the LOCA size based on the break area as a percentage of the recirculation loop flow area. For example, 50% LOCA means that the break area is 50% of the recirculation loop flow area. Double-ended guillotine break (also known as 200% LOCA) is a hypothetical accident that occurs when the recirculation loop pipe attached to the pressure vessel is totally broken into two separate flow paths.

Table 4.8: Classification of LOCA based on the break size

<table>
<thead>
<tr>
<th>Type</th>
<th>Break area (% of flow area)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Small Break LOCA (SBLOCA)</td>
<td>1%-30%</td>
</tr>
<tr>
<td>Intermediate Break LOCA (IBLOCA)</td>
<td>30%-60%</td>
</tr>
<tr>
<td>Large Break LOCA (LBLOCA)</td>
<td>60%-100%</td>
</tr>
<tr>
<td>Double-Ended Guillotine (DEG)</td>
<td>200%</td>
</tr>
</tbody>
</table>

4.4 Simulation Procedures

For BWR instability scenarios, the TRACE/PARCS simulation consists of three different simulation steps connected through restart files as follows:

1. Steady-state standalone TRACE thermal-hydraulics simulation for user-defined flow and power conditions.
2. Coupled TRACE/PARCS steady-state calculations to initialize neutronics and thermal-hydraulics conditions for transient calculations.
3. Coupled TRACE/PARCS transient calculations using a pump trip as initiating event and control rod movements to induce instability.

For LOCA scenarios, PARCS is deactivated since the power profile is provided to TRACE and so the simulation consists of two steps connected through a restart file as follows:

1. Steady-state TRACE thermal-hydraulics simulation for user-defined flow and power conditions.

2. Transient TRACE calculations using pipe break as initiating event.

Steady-state results are used as initial conditions for the transient calculations. It is worth to mention that all of the simulation steps are important for accurate simulation, and this is one of the reasons that makes TRACE/PARCS calculations to be time-consuming.
CHAPTER 5. RESULTS AND DISCUSSION

This Chapter is divided into two main sections: the first one presents the results that show the effect of the RFRD device on BWR instability accident while the second one demonstrates RFRD effect on LOCA. For BWR instability accident, three different cases have been analyzed to demonstrate the effectiveness of using RFRD on the unstable behavior of BWRs:

1- The original Ringhals-1 model where RFRD is not applied at any bundle inlet. This case is designated as “No RFRD”.

2- RFRD is implemented in a single channel (two fuel bundles) where flow rate reversal occurs. This case is designated as “RFRD in a single channel”.

3- RFRD is implemented in all core channels. This case is designated as “RFRD in all channels”.

The above three cases have been used to study the effect of RFRD on local parameters that are related to a single fuel bundle, namely bundle flow rate and PCT, and global parameters like core flow rate, power, and pressure.

For LOCA accident, three different LOCA scenarios with different break size have been simulated, namely, (1) 25% SBLOCA (2) 100% LBLOCA (Base Case) (3) 200% Double Ended Guillotine. The bundle flow rate and PCT are mainly studied for LOCA scenarios and the temperature reduction achieved when using the RFRD device is also demonstrated and compared for the three scenarios. Friction factor sensitivity on flow rate is also investigated to determine the value that would be adequate to prevent the reverse flow for both BWR instability and LOCA accidents.
5.1 BWR Instability Accident

Ringhals-1 TRACE model is used to test the proposed device as RFRD is used in one channel (two fuel bundles) during BWR instability to investigate its effect on the bundle flow rate. Although knowing the fuel channel with flow reversal before the accident is not possible experimentally, this can be done in simulation. Therefore, a channel with flow reversal is selected to study the effect of RFRD on local parameters such as flow rate and PCT. Simulation results demonstrate that without restricting the flow in the downward direction, inlet mass flow rate in the fuel channel oscillates about its average value and it oscillates from positive values to negative values as shown in Figure 5.1. When the flow reversal occurs, the device drops down to block the holes beneath it and the flow is prevented in the reverse direction. Only insignificant flow rate reversal is allowed and the oscillation magnitude is thus limited. Moreover, a phase shift occurs in the oscillation waves after the addition of the device and the frequency increases by negligible amount (see Figure 5.1).

![Figure 5.1. Inlet flow rate oscillation in the fuel bundle with and without using the RFRD](image)

Since determining the bundles with flow reversal is not possible before the accident, the RFRD device should be implemented in all fuel bundles as in this case the simulation will
be more realistic. Consequently, the simulation is repeated with the RFRD in all channels to investigate the cumulative effect of the device. Figure 5.2 demonstrates that if RFRD is used in all bundles, small improvement of bundle flow oscillations can be achieved compared to the single channel results (See Figure 5.1). Flow rate oscillations start growing after 120s where the oscillation magnitude further decreases, and the frequency increases slightly.

![Graph showing mass flow rate versus time with and without RFRD in all bundles.](image)

**Figure 5.2.** Inlet flow rate oscillations in a single bundle with flow reversal behavior

Large power oscillations and reduced flow rate can cause excessive clad temperature which could lead to sustained clad dryout and hence fuel damage. Therefore, the clad temperature of the fuel should not exceed the safety limit during the core instability event. Since the flow reversal in a single channel has been mitigated by the RFRD device, the clad temperature should be reduced. Figure 5.3 shows that PCT increases sharply due to dryout
for the case without RFRD. However, the reduction in the PCT is significant either when RFRD is applied in one or all fuel bundles.

According to U.S.NRC regulations, cladding temperature has to be less than about 2200°F (~1480 K) in order to reuse fuel rods without any restriction, and the RFRD device ensures the PCT is well below that limit. To obtain lower clad temperatures, the cumulative effect of using the RFRD device in all fuel bundles can reduce the PCT to even level lowers than those achieved in a single bundle RFRD, even though single bundle RFRD resulted in adequate temperature. In other words, dryout-rewet cycle that could happen in this type of accidents could be eliminated with this hardware device.

Similar to flow rate and clad temperature, power oscillation magnitude has been also affected by the RFRD device. Figure 5.4 shows the plot of the core power versus time with
and without using the RFRD. Without the RFRD device, core power oscillates between 20% and 230%. If the device is applied to only one channel, it has a very small effect on reducing the core power oscillations. This is due to the fact that the effect of a single channel on a global parameter like power is small. However, the effect of RFRD is significant if the device is used in multiple channels. Figure 5.4 demonstrates that if RFRD is used in all fuel bundles, it will reduce the power oscillation amplitude by about 50%. However, it should be noted that the reduction in power oscillation amplitude is rather modest, and so, the oscillatory dryout behavior is more sensitive to flow than power oscillation.

![Graph showing power oscillations](image)

**Figure 5.4.** Core power oscillations after using RFRD in one channel and all fuel channels

It is important to mention that the simulation with RFRD device on all bundles has been done for two reasons. First, to demonstrate the best possible result from the device to guarantee the presence of liquid coolant in all bundles and ensure fast rewetting of the fuel
cladding surface. Thus, the cladding temperature remains below the safety limit and fuel damage is avoided when the RFRD is used. Second, the device should be used in all fuel bundles in practice since the bundles with flow reversal could not be known before the accident.

The effect of using the RFRD in a single bundle and in all core bundles has been investigated to find its effect on the core flow rate and PCT. As only a limited number of bundles have a flow reversal behavior in the considered scenario, total core flow rate is always positive (oscillates approximately between 2350 kg/s and 2800 kg/s) as shown in Figure 5.5.

![Core flow rate with and without RFRD in all fuel channels](image)

Figure 5.5. Core flow rate with and without RFRD in all fuel channels

As expected the effect of the RFRD in a single bundle on the core flow rate is small. On the other hand, it is clear that the core flow rate oscillation amplitude is decreased when
using the RFRD device in all fuel bundles. Therefore, RFRD reduces the flow rate oscillation magnitude locally in a single channel as well as globally in the core. In addition, it is worth to mention that the coupling effect between the power and flow rate can be observed here as the application of the device reduced the core power oscillation amplitude (see Figure 5.4), and due to the coupling, the flow rate oscillation amplitude should be reduced also.

As the device proved to be effective to reduce the oscillations in flow, power, and PCT, other variables have been investigated to determine if the device can mitigate other thermal-hydraulics oscillations (see Figure 5.6 and Figure 5.7).

Since the effect of a single bundle on global parameters is generally small, the results are presented when the device is applied in all fuel bundles. It seems that the device can mitigate the oscillations in other parameters like steam and feedwater flow rate by reducing the
oscillation amplitude by about 30%. In addition to that, the core pressure and water level oscillations can be reduced further even though the oscillation amplitude is already small even for the case without RFRD device (e.g. pressure oscillates from 7.03-7.07 MPa). However, in general the RFRD device can even reduce the oscillations amplitude in these variables and hence bring some improvements in the safety of the plant during this type of accidents.

As the study is based on the value of the reverse flow friction coefficient (KFACR) that the device can achieve at the channel inlet, a sensitivity study for the value of this parameter is investigated. Figure 5.8 shows the plot of the bundle flow rate for increasing values of KFACR. When KFACR is 0, this indicates that there is no device at all and it refers to the original case. The oscillation magnitude starts to decrease as KFACR increases, and the flow reduction increases as KFACR becomes bigger. A significant reduction can be seen when
KFACR increases from 0 to 10 compared to the one when KFACR increases from 500 to 1E+4. KFACR value of 500.0 is sufficient to achieve a restricted flow for instability accident, with taking into account that some leakage would still be possible.

![Diagram showing flow rate oscillations for different values of KFACR during BWR instability](image)

Figure 5.8. Flow rate oscillations for different values of KFACR during BWR instability

### 5.2 LOCA Accident

100% LBLOCA (Base Case)

Simulation results for Oskarshamn-2 100% LOCA model shows that the RFRD device can successfully prevent reverse flow that leads to coolant leakage from the fuel bundle inlet during LBLOCA. The results of 100% LOCA without RFRD including the pressure, power, break flow and other important variables are reported in Appendix A. As mentioned before, the RFRD is applied to all core bundles for LOCA case. Figure 5.9 shows that the flow rate behavior for a hot channel and an average channel during LBLOCA is nearly the same.
Negative flow rate occurs directly after the break as it reaches about -10 kg/s in hot bundle and -5 kg/s in average bundle due to the blowdown phase of LOCA as the coolant leaks from the break and from the bundle inlet out to the containment. After that the bundle is left dry with only steam flow for some time until the activation of the core sprays and LPCI. The reverse flow occurs again when the emergency water starts to flow inside the core. However, the RFRD device prevents flow reversal when it is applied to all bundles except at the beginning of the blowdown phase where insignificant flow reversal is still observed (see Figure 5.9).

![Figure 5.9. Bundle flow rate during LBLOCA for (a) hot bundle (b) normal bundle](image)

The reverse flow has been eliminated after using RDRD for both stages during blowdown and after the activation of safety systems. Therefore, it is clear that the effectiveness of the device depends on how much flow reversal happens during the LOCA accident (See Figure 5.9). Therefore, by preventing the leakage by reverse flow, more coolant will stay within the core during blowdown and from the emergency systems which will help to cool the fuel.
The success of the RFRD device is measured by its effect on the PCT as additional coolant is preserved inside the core. Figure 5.10 shows that PCT without using RFRD reaches a value of 1100 K and could lead to fuel damage as it is close to U.S.NRC limit of 2200 °F (~1480 K). On the other hand, RFRD can reduce the PCT of hot and average bundles to safer levels by increasing the amount of available coolant (see Figure 5.10). The device has another great advantage as we can see the fuel reaches the reflood period faster when using the RFRD device as the fuel quenches in shorter time than without using the RFRD. This is because as the cladding quenches, the surface becomes wetted, and saturation temperature of the clad decreases rapidly. The RFRD effect on LOCA is:

1- Before the break, the device is already in the up position, no effect is observed ($\Delta T=0$).

2- Directly after the break, the $\Delta T$ (difference in PCT between LOCA with and without RFRD) increases sharply since the coolant inventory decreases rapidly (blowdown) without RFRD.

3- After that, $\Delta T$ starts to decrease until the activation of the safety systems when RFRD can keep more water inside the core and hence $\Delta T$ rises again.

4- RFRD results in earlier quenching for the clad. In this case, $\Delta T$ reaches its maximum value since the PCT without RFRD is still high.

The temperature difference between the two cases demonstrates that the device is capable of achieving a reduction up to 600 K for hot channel and up to 250 K for average channel during LBLOCA.
On the other hand, RFRD has negligible effect on other quantities inside the core like the core pressure and the break flow. Figure 5.11 shows that the core pressure behavior is similar with and without using the RFRD device and this means that the device will not affect the core depressurization and hence the safety systems are activated at the same time. Similarly, the break flow is practically unchanged when using the RFRD since the break is in the recirculation loop and the device has no effect on the flow through the break.

25% SBLOCA

A SBLOCA with break size of 25% is analyzed here. The results including the flow rate of the bundle as well as the PCT are shown in Figure 5.12. The flow rate of the bundle for small break is quite different from that of large break. The coolant blowdown is slower and the reverse flow is not as large as in the large break. However, the RFRD device benefit can be seen when the emergency systems are activated and the reverse flow would occur at that time due to the leakage of the ECCS water from the bundle inlet. Therefore, thanks to RFRD device as the cladding quenching occurs earlier and the PCT is decreased (See Figure 5.12).
The PCT trend begins by temperature reduction to low temperature of 400 K after the break followed by gradual increase to the maximum value of about 950K. After that the cladding quenching occurs after activation of the core spray. The figure shows that the device has negligible effect on PCT for the first 400s as the $\Delta T$ (difference in PCT between LOCA with and without RFRD) is approximately zero during this period.
However, when the spray water starts to refill the core, the maximum PCT is reduced to 900 K by the RFRD device and the ΔT increases sharply to 500 K. After that, the cladding quenches and the saturation temperature is restored. Therefore, the effect of the RFRD device in SBLOCA is limited to the reflood phase of the transient as quenching begins in about 50s earlier when using the RFRD.

**Double Ended Break**

The scenario for double ended break that has been simulated is 200% double ended guillotine break. The double ended break for BWRs occurs when a guillotine break occurs in the recirculation loop pipe that is attached to the pressure vessel so that the pipe is broken into two separate flow paths. Figure 5.13 shows the flow rate as well as PCT during the transient. The 200% LOCA is characterized by fast depressurization and large blowdown of flow rate to levels even higher than LBLOCA as flow rate drops to -35 kg/s directly after the break. This is because cutting the recirculation pipe into two flow paths will increase the leakage of coolant to large values compared to the other LOCA breaks, and this makes double ended guillotine break to be the most severe type of LOCA. For PCT, it seems that RFRD is not effective for the first 50s as the PCT with RFRD becomes slightly higher than without RFRD. This result means that the RFRD device would increase the PCT during the double ended break and this is undesirable effect. However, it is clear that the negative ΔT (difference in PCT between LOCA with and without RFRD) values are considered small and span for short time after the break. After that, the RFRD device reduced the temperature as ΔT starts to grow from negative values to large positive values. RFRD device can achieve a positive ΔT up to 550 K during the double-ended break LOCA. Similar to SBLOCA and
LBLOCA cases, the cladding saturation temperature is restored faster than the case without using the RFRD device (see Figure 5.13).

![Figure 5.13](image_url)

Figure 5.13. 200% double-ended guillotine: (a) bundle flow rate and (b) bundle PCT

Therefore, based on the previous analysis, the effectiveness of the RFRD device is seen in three areas:

1- The RFRD device can achieve a significant reduction in PCT for all three cases: LBLOCA, SBLOCA, and double-ended guillotine break.

2- The RFRD device can achieve a faster quenching of the cladding for all three cases: LBLOCA, SBLOCA, and double-ended guillotine break.

3- Duration of effectiveness is different for the three cases. For instance, RFRD is active for the whole period of LBLOCA. However, for SBLOCA RFRD is nearly inactive for the first 400s of the scenario.

Friction Factor Sensitivity

The value of the reverse flow friction coefficient (KFACR) that the device can provide during LOCA to ensure blocking the reverse flow is investigated. Since the blowdown phase
is the most critical moment as a large amount of water leaks from the core and carries significant momentum, the device should be able to prevent that leakage. Consequently, if the device is able to prevent the flow leakage during blowdown, it should be able to prevent any other smaller reverse flows. Figure 5.14 shows the plot of the bundle flow rate with increasing values of KFACR during LOCA. Increasing the value of KFACR from 0 to 500 reduces the reverse flow but it seems to be inadequate, which means that the device should be able to sustain even larger pressure created when the water flows downward. The amount of coolant leakage decreases as KFACR increases, but the relative reduction becomes smaller as KFACR becomes bigger. For example, Figure 5.14 shows that a relatively small reduction is seen when increasing the KFACR from 5000 to 1E+4. Consequently, the value of 5000 for KFACR would be sufficient to achieve a well-restricted flow during LOCA.

![Figure 5.14. Bundle flow rate for different values of KFACR during LOCA](image-url)
CHAPTER 6. CONCLUSIONS

In this study, TRACE/PARCS code package has been used to study the effect of reverse flow restriction on mitigation of the power and flow rate oscillations to prevent the sustained fuel dryout and so increase the safety of BWRs. The presented results clearly demonstrate the potential of the previously described hardware device [1] to prevent core flow reversal and fuel dryout during a BWR instability event allowed to progress without scram. According to the post-dryout modeling in TRACE, the RFRD device prevents the failure in rewetting of the clad by reducing the time under dryout which reduces the PCT and maintains it within safe limits. The results also demonstrated that the device is suitable for core-wide mode instability events when RFRD is implemented in multiple fuel channels which reduces core power oscillations amplitude. For bundles that oscillate only in forward flow direction (i.e. no flow reversal), the device did not affect the flow rate oscillations of these bundles directly, but flow oscillation reduction was observed due to the neutronics coupling. This device will ensure presence of sufficient coolant to cause rewetting of the fuel cladding. The RFRD device demonstrates a great capability to keep the fuel safe during LOCA through two stages:

1- Containing the coolant that leaves from the bundle inlet during the blowdown phase when massive amount of coolant leaves through the break at the beginning of LOCA.

2- Containing the emergency cooling injection inside the core when the emergency systems are activated.

RFRD is capable to reduce the PCT to safer levels through maintaining additional coolant inside the core during LOCA. RFRD is also able to reduce the time needed to reach
the reflood phase and quenching when the saturation temperature of the clad is restored. The device demonstrates beneficial capabilities during large break and small breaks LOCA. Therefore, if the coolant leakage in LOCA accident is reduced by applying flow reversal restriction, there potentially can be a longer coping time during LOCA accidents and the cooling effectiveness of the ECCS will increase.

However, the main limitation of RFRD is that it could not prevent the power and flow oscillations themselves from growing during BWR instability. It can only mitigate dryout-rewet cycle that occurs due to power and flow rate oscillations in BWR instability scenarios, and reduce the clad temperature and restore the saturation temperature faster during LOCA accidents. RFRD could increase the PCT compared to the case without RFRD as in the case of double ended break even though that increase is small and spans only for short time. This issue needs more investigation to determine the factors that made the PCT to increase more in RFRD case. Furthermore, the problem of loose parts that might occur during the device operation puts another challenge for proper design of the device since these parts could block the flow or damage the cladding.

As a future work, the minimum fraction of bundles in a fresh fuel reload that is needed to achieve universal core protection against instability clad temperature excursions will be investigated. Furthermore, the device can be investigated experimentally in facilities for LOCA and flow instability to validate the simulation results. Additional studies should be performed to see if RFRD can be useful for other types of accidents either in BWR or PWR and determine its capability to improve the safety of the reactor during these accidents.
APPENDIX A. Base LBLOCA (100%) Results

Figure A.1. Core flow during LOCA

Figure A.2. Core pressure during LOCA
Figure A.3. Core power during LOCA

Figure A.4. Break Flow during LOCA
Figure A.5. Feedwater flow rate

Figure A.6. Core spray flow rate (activated when the core pressure < 0.2 MPa)
REFERENCES


