

İSTANBUL TECHNICAL UNIVERSITY ★ ENERGY INSTITUTE

LOCA ANALYSIS OF STURE

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STURE REAKTÖRÜ LOCA ANALİZİ

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FOREWORD

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Halil Aslan
Mechanical Engineer

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ABBREVIATIONS

ALARA	: As Low As Reasonably Achievable
BWR	: Boiling Water Reactor
CR	: Control Rod
CRD	: Control Rod Drive
CRS	: Control Rod Stem
DWS	: Demineralized Water System
FP	: Fuel Pin
HMI	: Human-Machine Interface
IRI	: Interfacultair Reactor Instituut
İTU	: İstanbul Technical University
KTH	: Kungliga Tekniska Högskolan
LOCA	: Loss of Coolant Accident
LWR	: Light Water Reactor
MTCS	: Moderator Temperature Control System
PWR	: Pressurized Water Reactor
NRC	: Nuclear Regulatory Commission
RSS	: Reactor Shutdown System
RWPS	: Reactor Water Purification System
SMRSS	: Shutdown Margin of Reactor Shutdown System
SNAP	: The Symbolic Nuclear Analysis Package
STURE	: Svensk Tränings och Undervisningsreaktor
TRACE	: TRAC RELAP Advanced Computational Engine
TRIGA	: Training, Research, Isotopes, General Atomics
UPS	: Uninterruptible Power Supply
WNA	: World Nuclear Association

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LOCA ANALYSIS OF STURE

SUMMARY

Increasing energy demand, plus concerns over climate change and dependence on overseas supplies of fossil fuels are coinciding to make the case for increasing use of nuclear power. Communities in Sweden and Finland have accepted the local construction of permanent disposal sites for nuclear waste.

Today nuclear energy is back on the policy agendas of many countries, with projections for new build similar to or exceeding those of the early years of nuclear power. After licensing time of the Swedish research reactors had passed, students have to go abroad for their studies in reactor experiments in nuclear engineering.

The Nuclear Energy Division of Royal Institute of Technology in Stockholm, Sweden plans to design, find funding for and eventually build a new nuclear research reactor. In order to obtain a license to build a non-commercial research reactor, several safety issues have to be made.

This thesis investigates a Loss of Coolant (LOCA) scenario for a non-commercial research reactor STURE. The reactor will only have a passive cooling system, which means no forced coolant flow. In principle, the thermal-hydraulic of the reactor will be equivalent to that of a heat source submerged in a pool of still standing water.

In the first part of the thesis, a thermal-hydraulic code, TRACE is used to perform steady state calculations. Thermal hydraulics codes are used to analyze LOCA and system transients in light-water nuclear reactors. The TRACE input of the steady state model is created by SNAP tool, which is a graphical user interface that creates text inputs and executes them. In the second part of the thesis some modifications are done and the code is run under transient mode to investigate a LOCA case.

STURE REAKTÖRÜ LOCA ANALİZİ

ÖZET

Artan enerji talebi, iklim değişikliği endişeleri ve fosil yakıtlara olan dışa bağımlılık, nükleer enerji kullanımının artmasına sebep olmuştur. İsveç ve Finlandiya'da hükümetler nükleer atıklar için kalıcı bertaraf alanları inşaatını onaylamışlardır.

Bugün nükleer enerji birçok ülkenin politika gündemine yeni reaktör inşaatı veya daha önceden kurulu olan reaktörlerin ömrünün uzatılması olarak geri dönmüştür. İsveç araştırma reaktörlerinin lisans ömürleri dolmuştur ve bu nedenle öğrenciler reaktör deneyleri yapmaya İsveç dışına gitmek zorunda kalmaktadır.

İsveç, Stockholm'deki Kraliyet Teknoloji Enstitüsü'nün Nükleer Enerji bölümü yeni bir araştırma reaktörü tasarımı için fon bulmayı ve sonunda yeni bir nükleer araştırma reaktörü yapmayı planlamaktadır. Ticari olmayan bir araştırma reaktör inşaatı için lisans almak, çeşitli güvenlik testlerini başarı ile sağlamasına bağlıdır.

Bu tez ticari olmayan bir araştırma reaktörü için bir LOCA senaryosu analizi yapılmasıdır. Bu reaktörün sadece pasif soğutma sistemi vardır, soğutma sisteminde herhangi bir pompa kullanılmamaktadır. Reaktör termal hidrolik olarak durgun suda duran bir ısı kaynağına eşdeğerdir.

Tezin ilk kısmında zamandan bağımsız modda hesaplamalar yapmak için TRACE adı verilen termal hidrolik bir kod kullanılmıştır. Termal hidrolik kodlar hafif su reaktörlerinde LOCA analizi için kullanılmaktadır. SNAP görsel arayüzü ile TRACE kodu için input dosyası hazırlanmış ve bu input dosyaları TRACE kodu ile çalıştırılmıştır. Tezin ikinci kısmında ise çeşitli değişiklikler yapıp kod zamana bağlı modda LOCA senaryosu, STURE reaktörü için incelenmiştir.

1. INTRODUCTION

Increasing energy demand, plus concerns over climate change and dependence on overseas supplies of fossil fuels are coinciding to make the case for increasing use of nuclear power. China is embarking upon a huge increase in nuclear capacity to 70-80 GWe by 2020; India's target is to add 20 to 30 new reactors by 2020. Communities in Sweden and Finland have accepted the local construction of permanent disposal sites for nuclear waste. International cooperation and commerce in the field of nuclear science and technology is growing. A WNA projection shows at least 1100 GWe of nuclear capacity by 2060, and possibly up to 3500 GWe, compared with 373 GWe today. Most of this increase will be in countries which already use nuclear power.

Since about 2001 there has been much talk about an imminent nuclear revival or "renaissance" which implies that the nuclear industry has been dormant or in decline for some time. Whereas this may generally be the case for the Western world, nuclear capacity has been expanding in Eastern Europe and Asia. Globally, the share of nuclear in world electricity has showed slight decline from about 17% to 13.5% since the mid 1980s, though output from nuclear reactors actually increased to match the growth in global electricity consumption.

Today nuclear energy is back on the policy agendas of many countries, with projections for new build similar to or exceeding those of the early years of nuclear power. The signals for a revival in support for nuclear power in the world was diminished by the accidents at Three Mile Island, Chernobyl, Fukushima and also by nuclear power plant construction cost overruns in the 1970s and 1980s, coupled with years of cheap natural gas [1].

Due to the reasons which are told above, nuclear energy was ignored during the last 4 decades but because of the Nuclear Renaissance, the nuclear energy is back on policy agendas of many countries like Sweden. During the last 4 decades, not enough nuclear engineers are brought up and that created a need for nuclear engineers with theoretical and practical knowledge right now. In order to fulfill this demand new master programs in nuclear engineering have started at KTH in 2007 and at Chalmers in 2009.

Nuclear research reactors have played an important role in the development of nuclear science and technology since the first artificial, self-sustaining, nuclear chain reaction was initiated on December 2, 1942 [2]. Research reactors are nuclear reactors which are generally not used for power generation. The thermal power ranges from close to zero power, up to about 100 MW in some larger research facilities. There are about 280 research reactors operating in 56 countries and about 360, have been shut down or decommissioned [3,4].

The first critical research reactor in Sweden, R1 went critical for the first time in July 1954 [5]. Until today, eight research reactors, including two subcritical reactor assemblies, have been built in Sweden, but all of them are now shut down. The last operating research reactors in Sweden, R2 and R2-0 in Studsvik, were decommissioned in June 2005 [6-8].

After the shut down of R2 and R2-0 in Studsvik there is a need for a new research reactor in Sweden. Students of KTH have to go abroad in order to complete their reactor experiments during their education. Many divisions at KTH have been cooperating in designing a new Swedish research and educational reactor. This has resulted in proposing preliminary design parameters for the reactor named STURE, Swedish Training and Education Reactor.

1.1 Purpose of the Thesis

General core design, thermal hydraulics, neutron kinetics and feasibility of the reactor and the radiological consequences of a severe accident have been studied [9-11]. But safety analysis of STURE has not been studied. In order to license any nuclear reactor it has to be shown that the risks are in acceptable limits and the safety of personnel and general public is assured.

This thesis investigates a LOCA scenario for a non-commercial research reactor STURE. The reactor will only have a passive cooling system, which means no forced coolant flow. In principle, the thermal-hydraulic of the reactor will be equivalent to that of a heat source submerged in a pool of still standing water.

In the first part of the thesis, a thermal-hydraulic code, TRACE is used to perform steady state calculations. Thermal hydraulics codes are used to analyze LOCA and system transients in light-water nuclear reactors. First a model of STURE is made by the help of the SNAP application. Then the TRACE input of the model is created by the SNAP tool, which is a graphical user interface that creates text inputs and executes them. The results of the steady state calculation will be shown in the the following part of the thesis. In the second part of the thesis, a LOCA case is investigated for the STURE. If a break occurs in the junction part of the reactor pool after an earthquake, how the reactor behaves is the scenario. After some modifications in the first model transient calculation results will be shown in the following part of the thesis.

1.2 Research and Training Reactors

Many of the world's nuclear reactors are used for research and training, materials testing, or the plutonium production of radioisotopes for medicine and industry. These are much smaller than power reactors or those propelling ships, and many are on university campuses. There are about 241 such reactors operating in 56 countries. Some radioisotope production also uses high-enriched uranium as target material for neutrons, and this is being phase out in favor of low-enriched uranium.

Research reactors comprise a wide range of civil and commercial nuclear reactors, which are generally not use for power generation. They are small relative to power reactors whose primary function is to produce heat to make electricity. The total power of the world's 283 research reactors is little over 3000 MW. As of September 2011, there were 241 operational research reactors (92 of them in developing countries), 3 under construction, 202 shut down (plus 13 temporary) and 211 decommissioned [1].

1.3 Types of Research Reactors

There is a much wider array of designs in use for research reactors than power reactors, where 80% of the world's plants are of just two similar types either pool type or zero power reactors. They also have different operating modes, producing energy, which may be steady or pulsed.

1.3.1 Pool Type Reactors

In pool type reactors (67 units in operation), the core is a cluster of fuel elements located in a large pool of water. There are control rods and empty channels for experimental materials among the fuel elements. Each element comprises several curved aluminum clad fuel plates in a vertical box. The water moderates and cools the reactor, and graphite or beryllium is generally used for the reflector. There are similar types of reactors called tank type reactors (32 units in operation), except that cooling is more active.

1.3.2 Zero Power Reactors

Some reactors are operated at low neutron flux and power level so that no external cooling is needed. These reactors are called zero power reactors and are commonly used for research and education purposes. A large advantage of these is that the fission product activity in the fuel is low enough to permit handling of the fuel, since it is safe for personnel to enter the biological shield within a few hours of shutdown.

1.4 Existing Research and Training Reactors

There are different types of research reactors in operation with different configurations. Some of them have complex designs like nuclear power reactors rather than research reactors and some of them are built in a simpler way. Some of the existing research reactors are described below.

1.4.1 CROCUS

The CROCUS reactor, operated by the Swiss Federal Institute of Technology, Lausanne, is a simple two-zone uranium fuelled, H₂O-moderated critical research facility. Figure 1.1 gives a view of the facility, a so-called zero power reactor, with a

maximum allowed power of 100W. The core is approximately cylindrical in shape with a diameter of about 60 cm and a height of 100 cm.

In 1995, in the CROCUS reactor which has a configuration with a central zone of 1.806 wt.%-enriched UO_2 rods and an outer zone of 0.947 wt.%-enriched uranium metal rods was made critical by raising the water level.

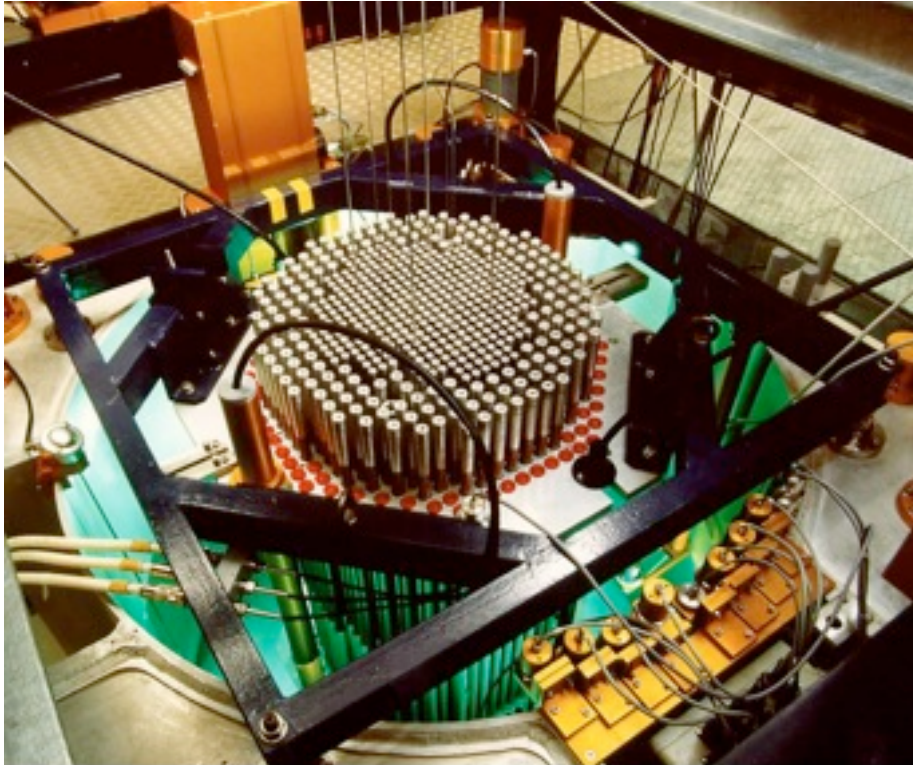


Figure 1.1 : View of the CROCUS reactor of the Swiss Federal Institute of Technology, Lausanne [12].

The two fuel zones in CROCUS consist of a central zone with 336 1.806 wt.%-enriched UO_2 rods in a square lattice with 18.37 mm pitch, and an outer zone with 176 larger 0.974 wt.%-enriched uranium metal rods in a square lattice with 29.17 mm pitch. The fuel diameter of the inner fuel rods is 10.52 mm and the fuel diameter of the outer fuel rods is 17 mm. All fuel rods have an aluminum cladding with a thickness of 0.85 mm for the inner rods and 0.975 mm for the outer rods. They are positioned in a vertical position by two grid plates spaced 100 cm apart [12].

1.4.2 Delphi

The Reactor Physics Department of IRI used to operate a subcritical assembly containing 253 fuel pins made of natural uranium in a hexagonal lattice. Figure 1.2 gives a view of the facility, as a moderator, light water was used. However, for some practical reasons, like the heavy weight of each fuel pin (almost 7 kg), the rather low k_{eff} (0.85) and the fact that some fuel pins stuck to the grid plates, this assembly is not used anymore. Therefore, it was decided to build a new assembly, called Delphi, for both the purposes of training and research.



Figure 1.2 : View of the Delphi subcritical assembly of Delft University of Technology, Netherlands [13].

In principle Delphi consists of two vessels one upon the other. The lower vessel is made of stainless steel and is filled with demineralized water before the start of an experiment. The upper air-filled vessel is used to store 168 fuel pins that can be lowered one after the other using a special handling tool. Below the steel vessel, a shielding box is positioned containing a ^{252}Cf - neutron source that can pneumatically be inserted to its experimental position in the steel vessel.

Each fuel pin contains 43 to 45 pellets of 3.8% enriched UO_2 fuel with aluminum cladding with an outer diameter of 12 mm and a wall thickness of 0.95 mm. The fuel pin length is 66.5 cm, of which 44 cm is fuel, and the UO_2 weight per fuel pin is 365.5 gr, which results in a total fuel weight of 59.9 kg. The fuel is placed in a cylindrical stainless steel vessel with an inner diameter and height of 100 cm and a wall thickness of 3 mm [13].

1.4.3 TRIGA

TRIGA is a pool-type reactor that can be installed without a containment building and is designed for use by scientific institutions and universities for purposes such as undergraduate and graduate education, promote commercial research, non-destructive testing and isotope production. Figure 1.3 and Figure 1.4 give views of the facility.

The TRIGA reactor uses uranium zirconium hydride fuel, which has a large, prompt negative thermal coefficient of reactivity, meaning that as the temperature of the core increases, the reactivity rapidly decreases. So it is impossible for a nuclear meltdown to occur. TRIGA was originally designed to be fueled with highly enriched uranium, but in 1978 the US Department of Energy launched its Reduced Enrichment for Research Test Reactors program, which promoted reactor conversion to low-enriched uranium fuel.

In İTÜ, Turkey there is a TRIGA Mark II reactor at the university which is used for experiments but not for isotope production. The reactor is a typical 250 kW TRIGA Mark II light water reactor using 20% highly enriched uranium fuel with a graphite reflector cooled by natural convection. The core is placed at the bottom of the 6.25 m high open tank with 2 m diameter. In total there are 91 locations in the core, which can be filled either by fuel elements or other components like control rods, a neutron source, irradiation channels.

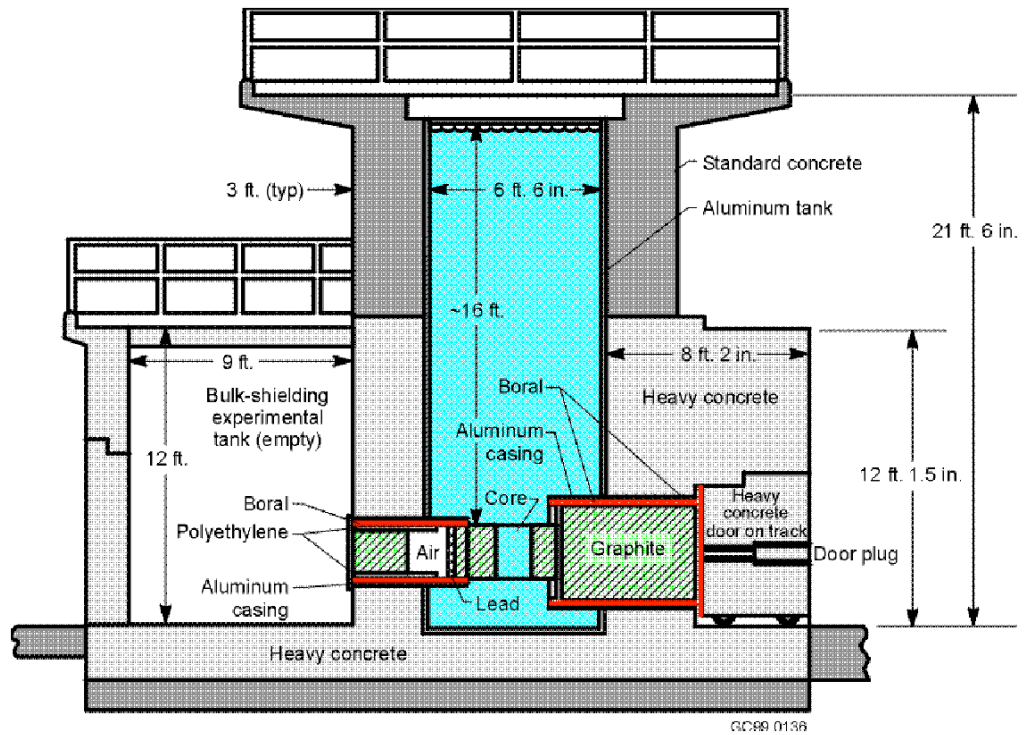


Figure 1.3 : Side view of a TRIGA Reactor [14].

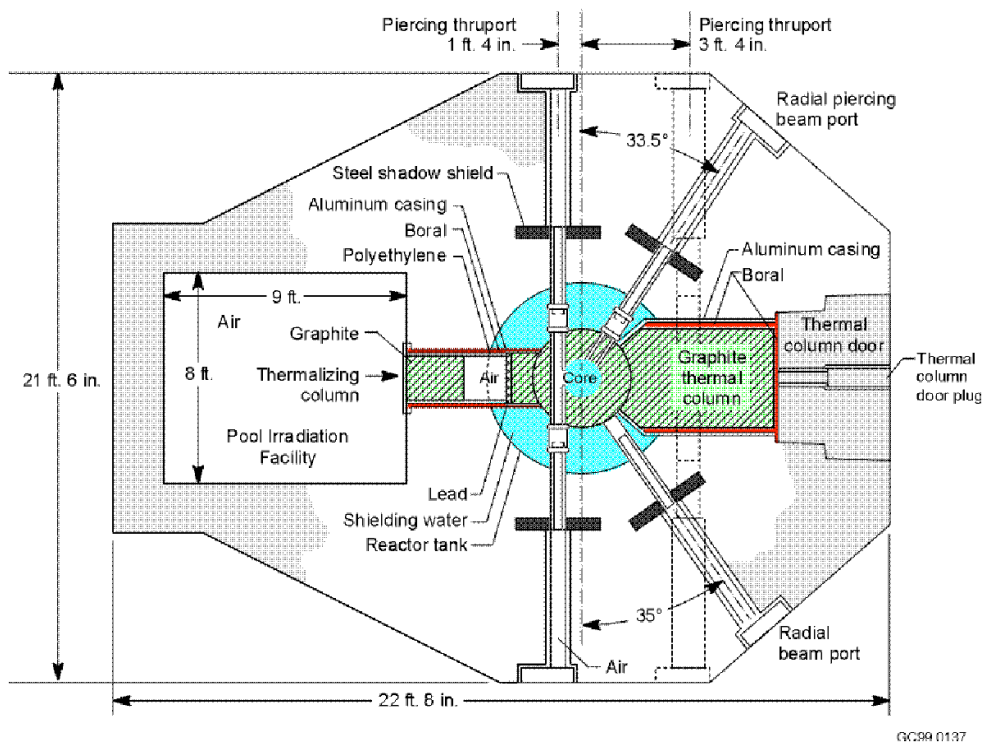


Figure 1.4 : Top view of a TRIGA Reactor [14].

TRIGA reactor fuel consists of metallic uranium alloyed with zirconium hydride with an enrichment of 20%. The active part of the fuel is 38 cm long and has a radius of 1.815 cm. It has either aluminum or stainless steel cladding and a samarium plate at each end that serves as burnable absorber. But the TRIGA reactor at Istanbul Technical University doesn't have the samarium plate [14].

1.4.4 VR-1 Sparrow

VR-1 Sparrow is a zero power pool type reactor with natural cooling. Figure 1.5 gives a view of the facility. Maximum thermal power of the reactor is 5 kW for a short time period in transient mode and for the other time the thermal power is 1 kW. It is a training reactor which is managed by the Faculty of Nuclear Sciences and Physical Engineering, Czech Technical University in Prague. It is used mostly for teaching students and training IAEA inspectors. It was using IRT-3M type of fuel which consists of several square tubes of different dimensions inside each other, with each tube consisting of a fuel layer of 36% enriched UO_2 with aluminum between two layers of aluminum cladding until 2005. Today it is using IRT-4M type of fuel which has the same geometry but with a different enrichment of 19.7%. There is 3 m water above the core [15-17].



Figure 1.5 : View of the VR-1 Sparrow Reactor of the Czech Technical University Prague [18].

2. STURE – SWEDISH TRAINING AND EDUCATION REACTOR

2.1 Why STURE?

Nuclear power production is close to half of the supply of electricity in Sweden and it may continue like this in the future. Today the nuclear power industry needs to employ hundreds of engineers every year. Especially engineers with deep functional knowledge in nuclear physics and nuclear technology are needed. Because of the need, master programs in nuclear energy engineering have started at KTH in 2007 and at Chalmers in 2009 with the support of the industry.

In order to obtain functional knowledge in the operation of reactors, students have to go to training reactors for laboratory experiments such as approach to criticality, control rod calibration, determination of neutron flux and delayed neutron fraction. Similar experiments were made in the R2-0 reactor in Studsvik until 2005. Students have to go abroad in order to do their laboratory experiments after the shut-down of the reactors in Studsvik.

Recently, students of KTH have travelled to Mol in Belgium, Budapest in Hungary, Prague in Czech Republic and Paris in France. Students of Chalmers have been sent to Kyoto in Japan and students of Uppsala University have been sent to Helsinki in Finland for training. This is economically and logistically possible as long as the number of students is limited. More than ten students in a laboratory reduces the quality of the education. The number of the students in Sweden are increasing so the cost of sending students to abroad are also increasing. It increased to a level of considering the construction and operation of a domestic training reactor.

The aim of Swedish universities like KTH, Chalmers and Uppsala University is to train approximately 100 students per year in reactor physics. In order to achieve this aim there is a need for a reactor for training and education in Sweden. The construction of a training reactor in Sweden with 10 kW of power, using standard LWR fuel pins with 43 cm active length and light water moderator, is suggested [19].

2.2 Pre-conceptual design

Training reactors are designed for different purposes. Some of them have multiple purposes such as providing neutrons for neutron science research or material research. For the design of STURE, it is kept;

- as simple as possible, to maximize safety
- as close to industrial design as possible, to emphasize the connection to application.

A standard light water reactor quadratic fuel-pin lattice immersed in light water is selected. The fuel vendor in Sweden already has the low enriched version of this fuel. While performing the experiments, the students can observe the core of the reactor directly due to the low power pool type design. Also, for the heat removal no pumps are required. In order to have no problems with licensing, the fuel is supposed to stay in its initial position for the full life time of the reactor which is 30 years. The main aim of the reactor is to provide training and education. It is not going to be used for producing neutron beams. A single fuel enrichment will avoid the possibility of errors in fuel loading.

The design is made as the core will never become prompt super-critical and the pin pitch is designed to maximize the k-effective. In addition to this, control rod worth will not exceed the beta-effective.

The size of the core is designed to keep an optimum value between the fuel cost and pedagogical utilization. Low enrichment of the fuel will lead to a larger core which will allow more space for measurements. Also high enrichment of the fuel will lead to a smaller core which will reduce the fuel cost. An optimum value of 3.8% enrichment is selected for the reactor with a cost assesment. Also 3.0% and 3.4% of fuel enrichment is also investigated.

In order to have no problems with licensing, the selected cladding is Zircaloy which is available commercially. The fuel will be available from the domestic vendor. The geometrical dimensions of the fuel pellet and pin are given on Table 2.1 [19].

Table 2.1: Dimensions and composition of STURE fuel pellet and cladding [19].

Material	Diameter (mm)	Composition	Density (g/cm ³)
Pellet	8.48	UO ₂	10.6
Cladding	8.63/9.84	Zircalloy-4	6.56

2.3 What is STURE?

STURE is a 10 kW light water moderated pool type reactor with low enriched UO₂ pin-type fuel. STURE has similar composition of fuel rods which are used in Swedish Nuclear Power plants in a dedicated structure immersed into a 3 m deep stainless steel pool.

The reactor pool which has a design that allows full access to the core for training or irradiation services also has enough protection for radiological and nuclear safety concerns. STURE is a low power reactor so that students can directly observe the core while doing experiments. STURE has two sectors which are the Reactor Building and the Auxiliary Building. It creates space for making different activities and deploy the systems [20].

2.3.1 Main Features

The main features of STURE are plant safety and reliability. The core of STURE is located in a pool of light water, having a diameter and a depth of 3 m which provides cooling by natural convection.

The pool has a safe operational environment with controlled accesses which is located in the Reactor Hall Sector of the reactor building. The Control Room, fresh fuel pins storage, laboratory, electrical services room, classrooms and amenities are in the Control Room and Services Sector beside the Reactor Hall Sector. In the Control Room running computers with a dedicated HMI, provides safe operation of the facility. Next to one of the Reactor Hall controlled accesses, the fresh fuel pins are located. During the life time of the reactor, the fuels will remain in the pool. If core configuration is changed, the used fuels will remain in the storage racks in the reactor pool. From the Neutron Activation Lab situated next to the other Reactor

Hall, there will be a direct and straightforward access to the reactor which is provided with units to make the experiments allowing an adequate framework.

The location of the reactor site will be about 30 km north of Oskarshamn at Kalmarsund at Baltic Sea coast. It will share the area with 3 BWR power reactors. It is predicted that STURE won't have an impact on the safety of the existing reactors [20].

2.3.2 Design Concept

The design of the reactor core and the control and safety systems are based on the following criteria:

- The reactor is cooled by natural convection
- It is possible to reach from top of the pool to the reactor core when the reactor is operating at full power, in order to make implementation of experiments.
- Control and safety rod mechanism are located at the top of the reactor tank.
- For the radiation shielding, a high water column is placed over the core.

The design of the systems are based on proven technologies. Also these systems have a flexible design in order to enable modifications if needed in the future. When controlling the moderator parameters for the operational requirement, supporting systems keep the pool water quality to the required levels. The control system ensures safe and reliable operation and makes available accurate readings of relevant parameters for training and academic purposes [20].

2.3.3 Safety

Design is made based on Defense in Depth principle. In Level 1 of the defense in depth prevention of unusual operation and failures takes place. It is aimed confining radioactive material and minimizing deviations from normal operation conditions including transient conditions and plant shutdown.

STURE is a low power reactor, the fuel stays in a stainless steel pool and controlled by systems which have a minimum effect on the staff members, public and the environment. The core will remain safe under water in abnormal conditions and the integrity of the fuel cladding will be guaranteed.

The required protective actions can be done manually or automatically by the Protection System with emergency UPS support. The safe shutdown is going to be made by the control rods which have enough negative reactivity in case the most reactive rod has a failure. The grids which houses the fuel is going to be made from aluminum in order to minimize the activation.

STURE is a low power reactor so that the radioactive waste generation is negligible. The fuel will remain in the reactor during the life time of the facility which 30 years, in order to simplify the licensing procedures. Also, there will be radiation detectors located in contamination risk areas and fire detection systems in order to keep the reactor safety [20].

2.4 Systems and Subsystems

2.4.1 Reactor Core

The reactor core geometry is determined by the core grid plates, which hold the different components: fuel pins (FP), control rods (CR), experimental channels and a pneumatic rig. The reactor core includes two grid plates which are upper and lower grid plates; has 481 positions which houses a variable number of FP, six CR assemblies and a few channels for experiments.

Three of the CR assemblies are used to make the reactor critical, control of the core reactivity during the experiments, irradiation also shutdown the reactor. The other three CR assemblies are called non-regulating rods which are used to shutdown the reactor ensuring an appropriate shutdown margin.

The control rod drives (CRD) have the same design and are a part of the Reactor Shutdown System (RSS). There are 481 positions in the core grid plates which limits the maximum excess reactivity of the core. So that the shutdown margin is calculated from the maximum number of irradiation points.

The reactor core is loaded with FP type of rods and every rod consist of 44 pellets of 3.8% enriched UO_2 fuel with Zircalloy cladding with an outer diameter of 9.84 mm and a wall thickness of 0.605 mm. The pellet diameter is 8.48 mm and UO_2 density is 10.6 g/cm^3 . The active length of the FP is 43 cm.

In an abnormal situation, the RSS gives fast response and shuts the reactor down automatically. RSS has a priority on the CRDs so that when a problem occurs it de-energizes the electromagnet which couples the absorber and the drive motor.

The maximum power generated in the core is removed by a passive method: cooling water in natural convection regime. The temperature of the pool water is regulated by the Reactor Process Plant Systems which remove the core heat, maintain the water quality and the water level in the pool.

There will not be any threat for the integrity of the core due to the large amount of pool water acting as the final heat sink and the low power value of the reactor. The experience gained from similar facilities generating more than one hundred kilowatts which rely on natural convection cooling supports the design of the STURE.

In Figure 2.1, the core and the components located around it can be seen [21].

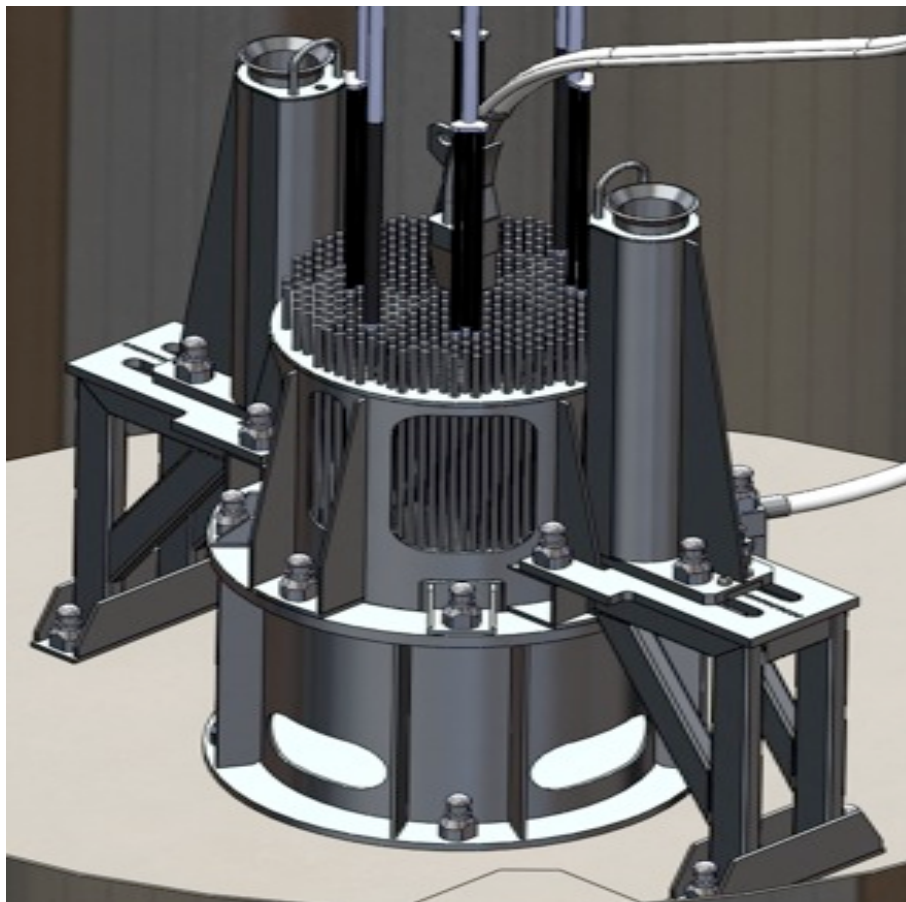


Figure 2.1 : STURE Reactor Core [21].

2.4.2 Reactivity Design Criteria

- For any operating condition, the total power reactivity coefficient must be negative and for all operating states and accident conditions the temperature and void coefficients of reactivity associated with the fuel and core must be negative.

$$\alpha_p < 0 \quad (2.1)$$

- RSS shutdown margin must be at least 3000 pcm.

$$SMRSS \geq 3000 \text{ pcm} \quad (2.2)$$

- The core must be subcritical with a shutdown margin of RSS at least 1000 pcm with any of its control devices out of the core.

$$SMRSS - 1 \geq 1000 \text{ pcm} \quad (2.3)$$

- The core must be subcritical with the non regulating rod out of the core.

$$SMRSS - NR \geq 0 \text{ pcm} \quad (2.4)$$

2.4.3 Preliminary Data Sheet

Configuration of the core is shown in Table 2.2 [21].

Table 2.2: Core Configuration Data [21].

Core Configuration Data	Value
Number of grid positions used for regulating CRs	9
Number of grid positions used for non regulating CRs	9
Maximum Number of FP in the core	$481 - 9 - 9 = 463$
Number of Neutron Detectors	2
Number of Pneumatic Rig	1
Grid positions occupied by the Pneumatic Rig	9

Parameters of core reactivity are shown in Table 2.3 [21].

Table 2.3: Reactivity Core Parameters [21].

Reactivity Parameter	Value
Maximum Reactivity excess (463 FP in Core)	2070 pcm
Total CR Reactivity Worth	7450 pcm
Reactivity worth of a Pneumatic Irradiation Station	1500 pcm
Reactivity worth of a FP located in the core boundary	30 pcm
Shutdown Margin in the most reactivity core (463 FP)	-5400 pcm
Shutdown Margin with single failure (463 FP in core)	-3800 pcm
Coolant temperature coefficient	-10 pcm/K
Fuel Doppler coefficient	-1.4 pcm/K

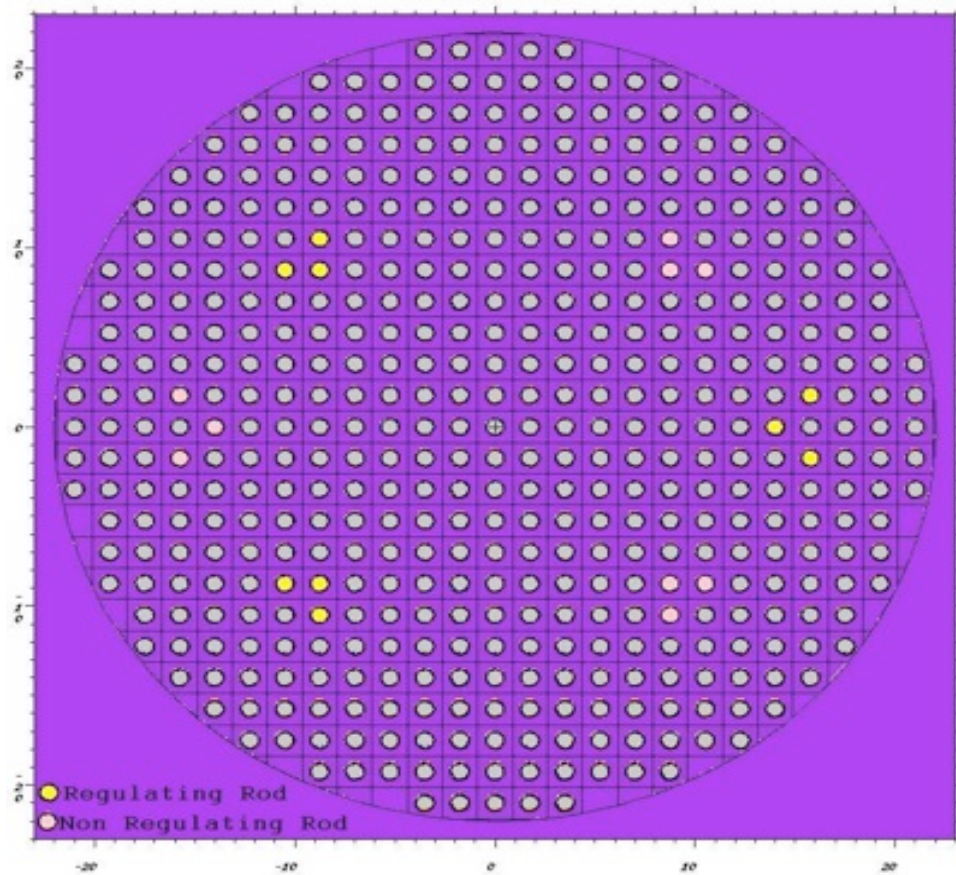


Figure 2.2 : Core Grid without Pneumatic Rig [21].

In the calculations Westinghouse FP specifications are being used. A cone at the lower end of each FP fits into the lower core grid and an underwater fuel handling and management is done by an upper tip. Figure 2.2 shows the Monte Carlo calculation model for the most reactive core (463 FP)

2.4.4 Reactivity Control and Reactor Shutdown System

There are six CR in the reactor and all of them are used to shutdown the reactor. Three of them are used to make the reactor critical and control the core reactivity during the experiments and irradiations. The other three CR's are used only to shutdown the reactor. Each controller has three absorber rod with a shutdown margin of 5400 pcm for the most reactive core (463 FP).

The withdrawal speed of the regulating rods is limited in order to prevent the reactivity insertion accidents and to perform the reactivity adjustments. CRD is located at the top of the reactor pool and controls the CR and CRD and CR connection is made by the Control Rod Stem (CRS).

Figure 2.3 shows an upper view of the STURE reactor core with the six CR's and the detachable central ring grid (3x3 size) which can be replaced by a pneumatic ring.

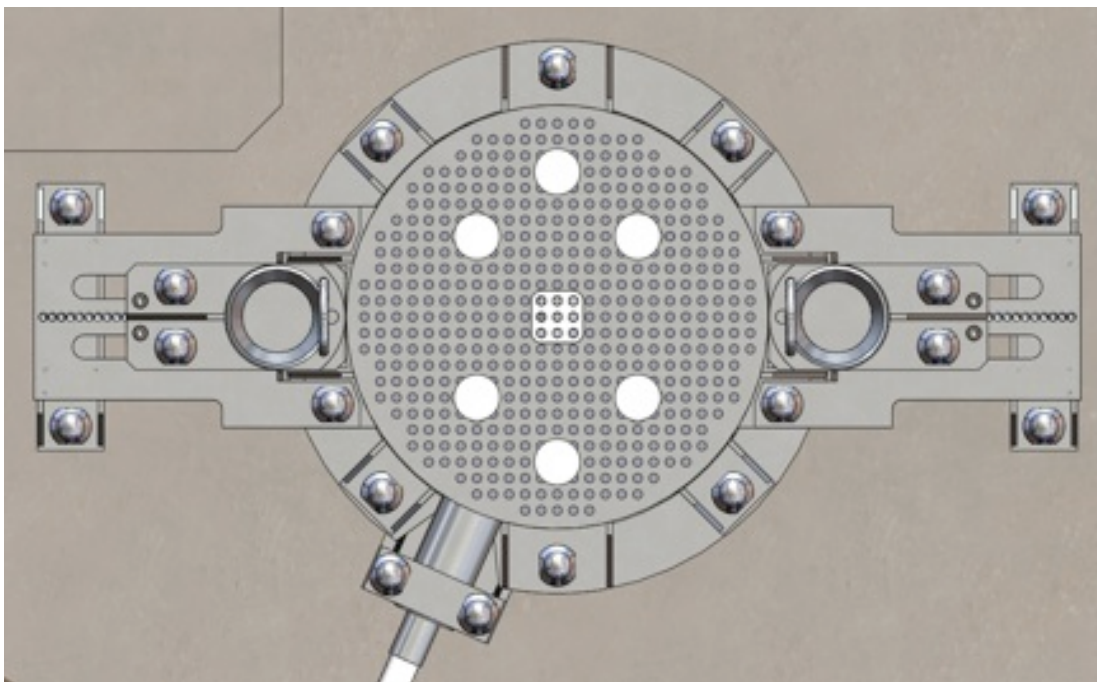


Figure 2.3 : Upper View of the STURE Reactor Core [21].

2.4.4.1 Control Rods

CR's are made of Boron Carbide as absorbing material and enclosed by a Zirconium cladding to prevent contacting with the coolant. CR's are controlled by the CRD's through the CRS.

2.4.4.2 Control Rod Drives

The CRD's has an electromechanical system which is coupled to an electromagnet. When energized CRD holds the CR which is attached to it. When de-energized it falls by gravity. The system is based on a servo motor with a gearbox and a position encoder. The braking system is based on an air damper which smooths the shocks when the rod is tripped, is on the lower part. From top to bottom main components of the system are:

- Encoder
- Motor
- Screw/nut
- Connection bar
- Electromagnet
- Damper

Figure 2.4 shows the elements of the Control Rod Drive.



Figure 2.4 : Control Rod Drive – Section View [21].

2.4.5 Neutron Source and Neutron Source Driving System

An Am-Be source with an activity of 1 Curie which leads to a strength of 2.2×10^6 neutrons/sec, is used to start up the reactor.

The neutron source is located below the bottom grid at the core centerline and is remotely operated. When all the control rods are inserted, the neutron source yield

provides a counting rate of 100 cps in the nucleonic detectors which is enough to make the reactor critical in a safe way.

In a container which runs through a pipeline with one closed and secured to the CRD supporting structure, the neutron source is kept. It is kept in a shielded storage position in the pump room.

2.4.6 Utilization Related Systems

2.4.6.1 Pneumatic System

The pneumatic system has three parts. One is the terminal station in the laboratory, the other is the pneumatic rig in the core and the last one is the tubing connection. The pneumatic rig and the tubing near the core is made of aluminum in order to minimize activation and subsequent wastes. In Table 2.4 the neutron flux values at the pneumatic rig, when the reactor operates at 10 kW, is shown.

Table 2.4: Neutron Flux Values at the Pneumatic Rig [21].

Energy Neutron Flux	Value (n/cm ² s)
Thermal Flux ($E < 0.625$ eV)	2.67E+11
Epithermal Flux (0.625 eV $< E < 1$ MeV)	2.90E+11
Fast Flux ($E > 1$ MeV)	1.14E+11

In Figure 2.5 a section of the core with the pneumatic rig located at the core center is shown.

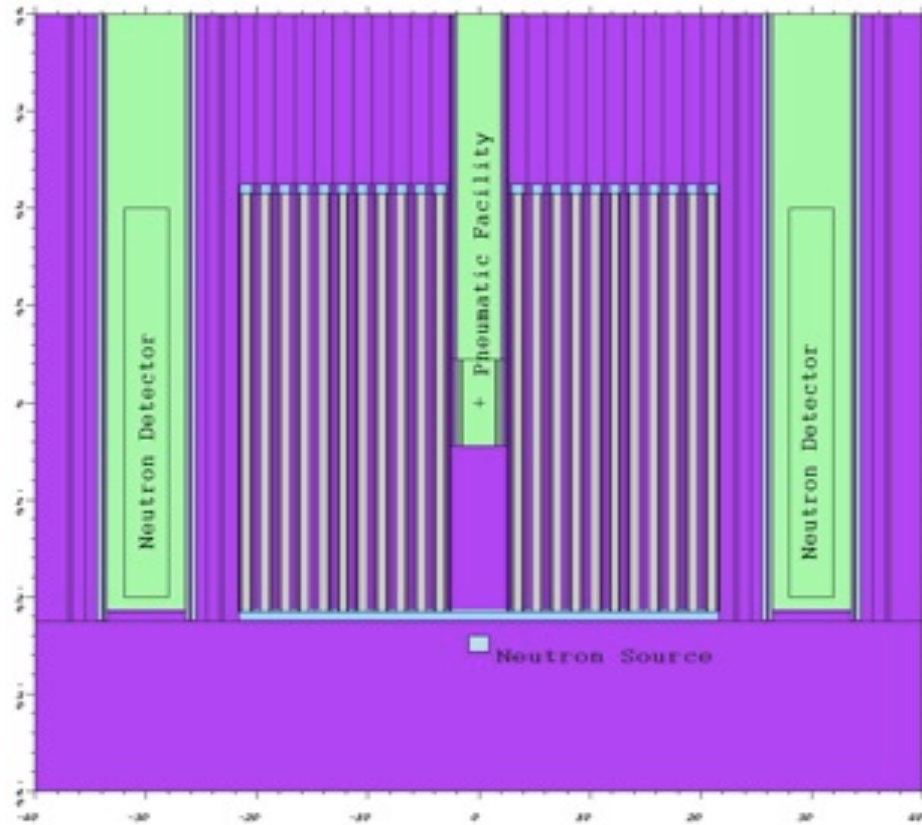


Figure 2.5 : Core Grid, Pneumatic Irradiation Facility, Neutron Detectors & Neutron Source [21].

2.4.6.2 Reactor Experiments

STURE is Swedish Training and Education Reactor so that students are going to do experiments in the reactor. The following experiments can be done with STURE:

- **Reactivity Measurement:** The reactivity can be measured using the Asymptotic Period technique and using the Rod Drop technique. Data are recorded during the experiment and analyzed later.
- **Isothermal Feedback Coefficient:** The pool water temperature can be changed by a few degrees and the reactivity can be measured by tracking the control rod positions.
- **Thermal Neutron Flux Mapping:** The thermal neutron flux can be measured by using flux monitors such as gold and copper foils.

If additional instruments are used more experiments can be done in STURE such as:

- **Neutron Activation Analysis:** Samples can be irradiated in the pneumatic rig and then analyzed in a gamma radiation detector.
- **Neutron Noise Measurement:** The absolute reactor power and the core kinetics parameters can be measured by using the neutron noise technique.
- **Pulse Neutron Source:** A lot of experiments can be done by using a pulsed neutron source.
- **Digital Reactimeter:** The online core reactivity in the facility can be measured with a computer based reactimeter.
- **Thermal Neutron Flux Mapping On Line:** The thermal neutron flux distribution can be measured using a small fission chamber located inside an empty tube positioned in the core grid.
- **Underwater Neutron Radiography:** A submersible device can be placed in the reactor hall and then it can be located close to the core inside the reactor pool. The device contains the sample to be analyzed and the detection system. The detection system can be a film or a converter screen plus a camera.
- **Fuel Rod Gamma Scanning:** The fission product distribution can be determined by the gamma scanning technique. Also by scanning some rods, the power distribution along the core can be determined.
- **Core Lattice Change:** The lattice geometry, fuel pitch and other stuff can be changed by changing the upper and bottom grids.

2.4.7 Radiation Shielding

STURE has all the required radiation shielding structures in order to keep the dose as low as reasonably achievable (ALARA) during normal operation. For the conceptual design of STURE two different reactor states are considered:

- Maximum power operation (10 kW)
- Reactor shutdown operation with reduced water inventory on the pool
- Pneumatic system components and targets
- Start up neutron source

In maximum power operation with the fission process itself there are other radiation sources which require shielding. These other sources are:

- Reactor core, including fission neutrons, fission gamma rays from (n, gamma) reactions and gamma rays from decay of fission products, actinides and activation products.
- Gamma ray source from ^{16}N , due to water activation, through the $^{16}\text{O}(n,p)^{16}\text{N}$ reaction (also ^{17}N and ^{19}O).
- Gamma ray source from impurities, due to the activation of impurities contained in the water, such as ^{41}Ar , ^{24}Na , ^{27}Mg , ^{28}Al and others.

In reactor shutdown, radiation sources are fission products, actinides and activation products from fuel and components in the reactor pool. The shielding which is going to be made for the full power will cover the requirements for the shutdown. But the water level during shutdown could be lowered, however a minimum water column height has to be kept above the core in order to have a biological shielding.

In pneumatic system the activated sources are travelling between the core and the laboratory. In order to keep the radiation dose as low as reasonably achievable for the staff calculations are done.

2.4.8 Reactor Pool and Internals

2.4.8.1 Reactor Pool Overall Design

The reactor pool is made of stainless steel, built in a concrete reactor block. It is an open tank of 3 m in diameter and 3 m height. The water inventory, the core and its structures, sensors and instruments for other purposes, racks for fuel pins and the mechanical supports of these components are kept in the reactor pool. The reactor pool extends 1m above the ground level to create a safe and easy access to the pool top area. All the operative areas of the core can be reached by a movable platform.

On top of the reactor pool there is a platform for CRD which supports the CRD's. In Figure 2.6 reactor pool is shown in general view.

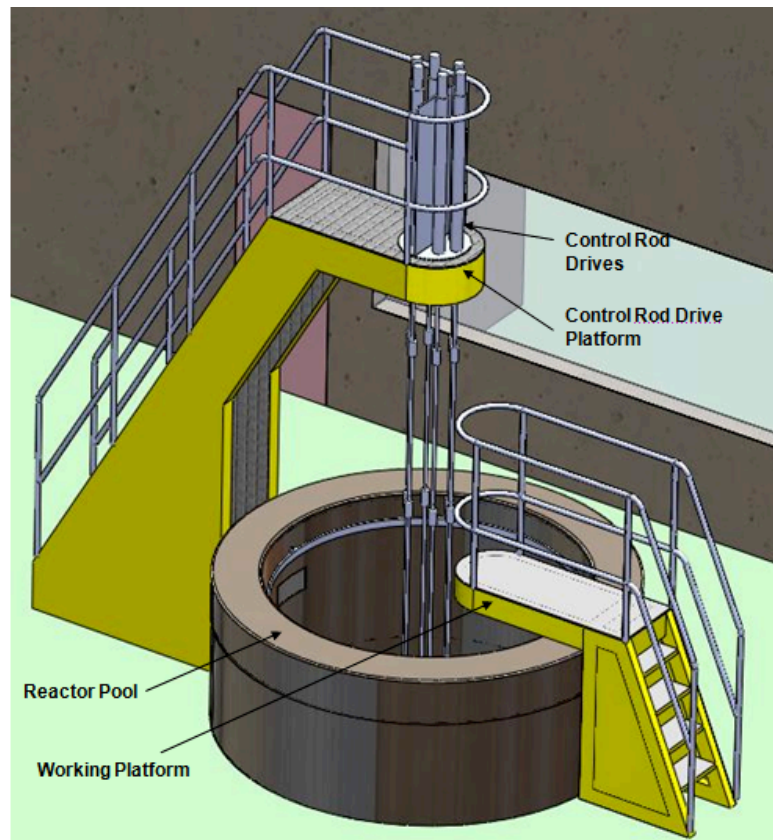


Figure 2.6 : Reactor Pool – General View [21].

2.4.8.2 Reactor Pool Internal Components

The reactor pool internal components are:

- a) Supporting structure
- b) Lower core grid
- c) Upper core grid
- d) Fuel pins
- e) Control rods
- f) Pneumatic system
- g) Neutron source driving system
- h) Fuel pin racks

- i) Nucleonic instrumentation
- j) Pool lights
- k) Control rod storage rack and pneumatic rig storage rack
- l) Internal pipes
- m) Landing area

The reactor pool internal components are shown below. In Figure 2.7 reactor pool components are shown in isometric view and in Figure 2.8 reactor pool components are shown in plan view. In Figure 2.9 reactor core is shown in isometric view.

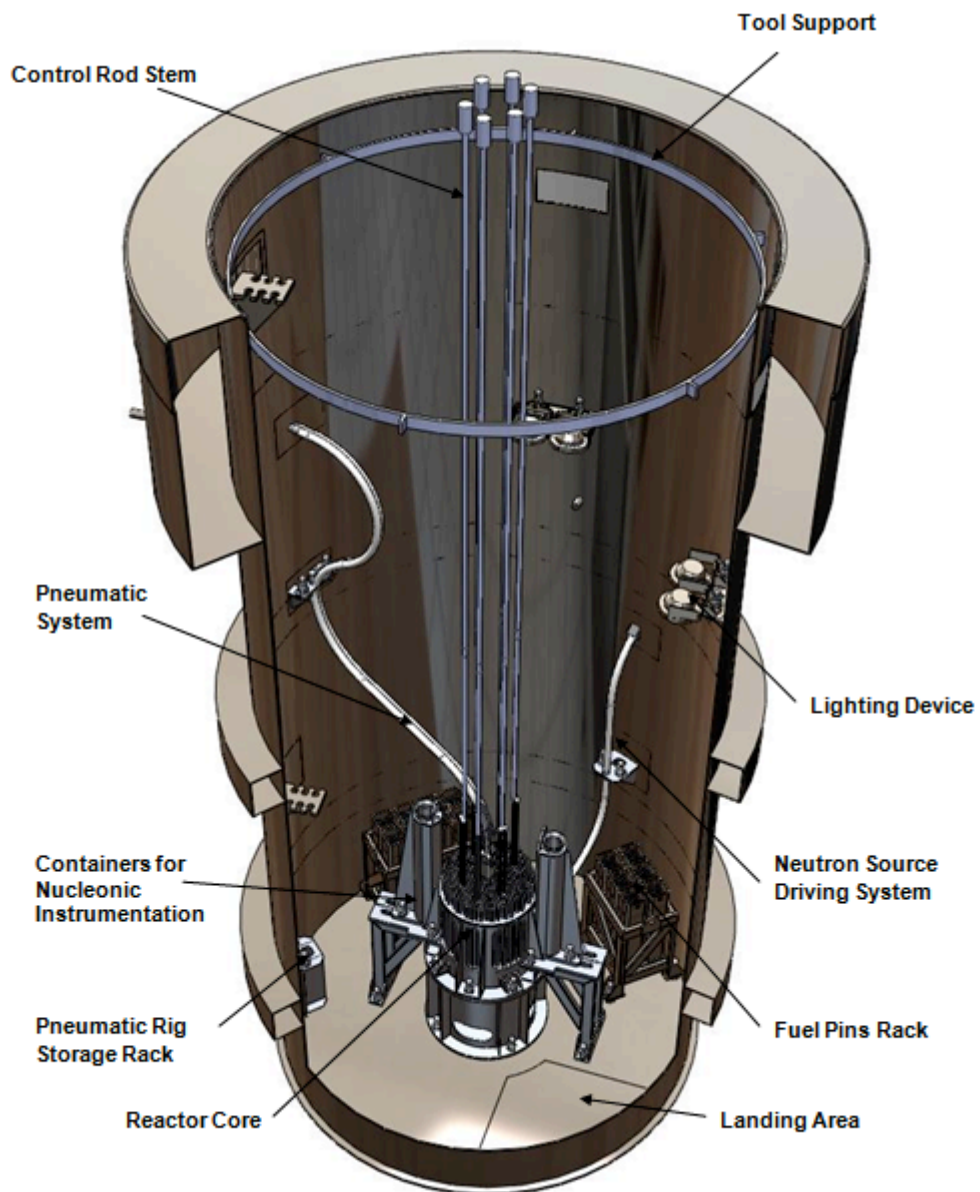


Figure 2.7 : Reactor Pool Internal Components – Isometric View [21]

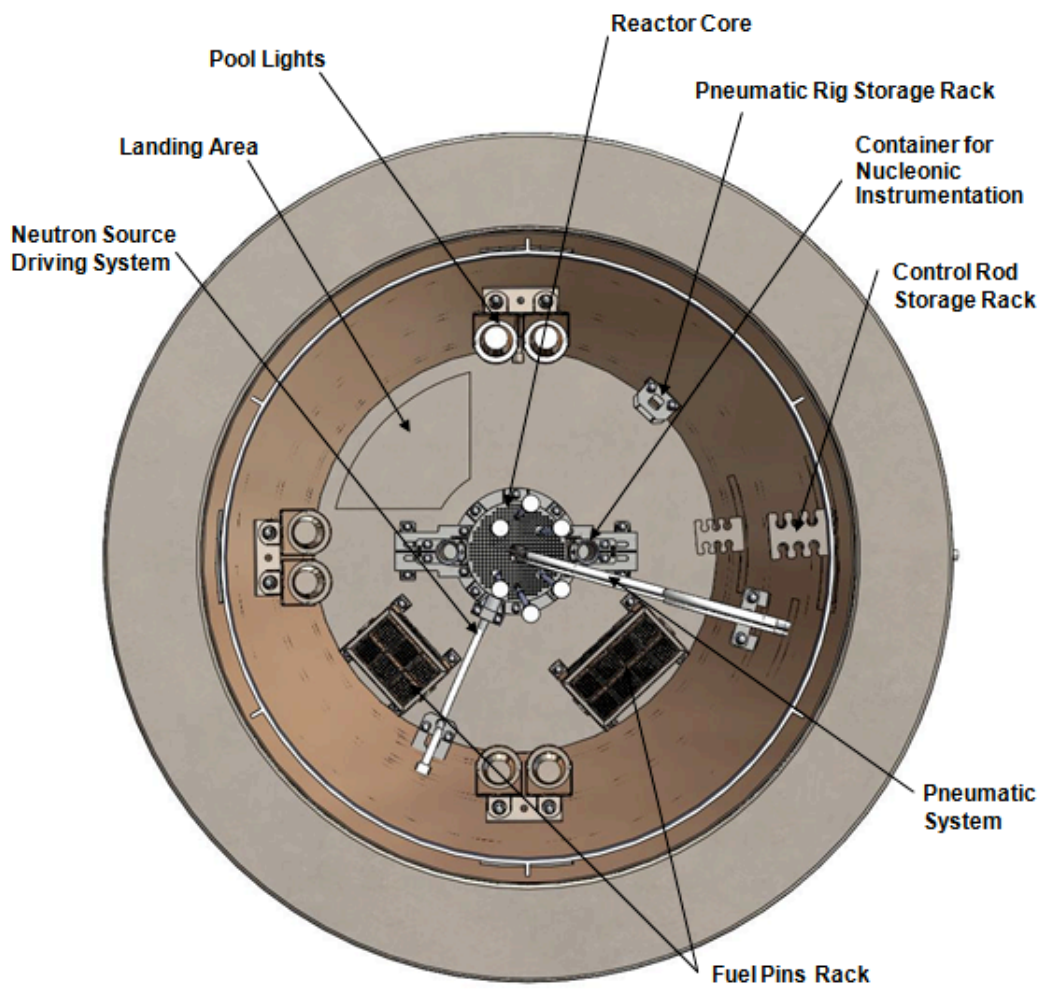


Figure 2.8 : Reactor Pool Internal Components – Plan View [21]

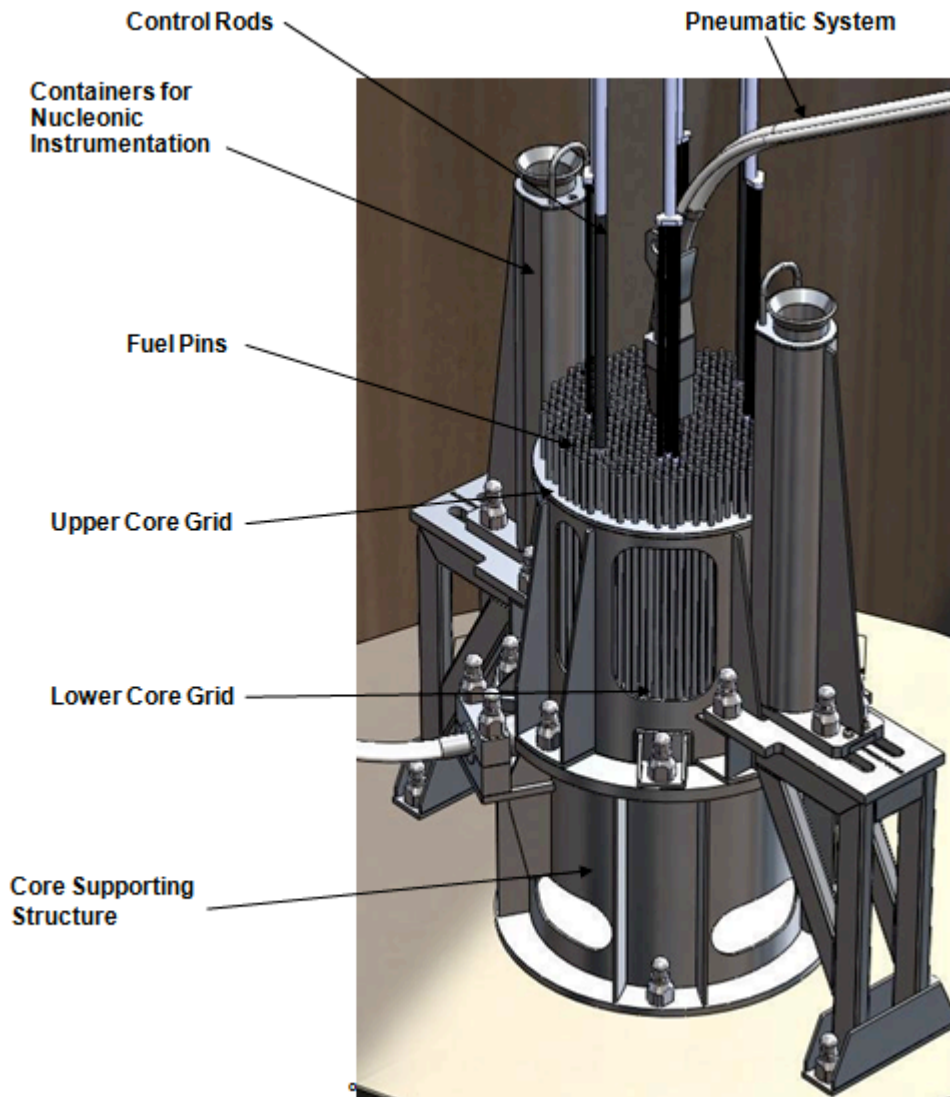


Figure 2.9 : Reactor Core– Isometric View [21].

2.4.9 Reactor Process Plant Systems

The Reactor Process Plant Systems are designed to regulate the temperature of the water, to remove the core and to maintain the water quality and level in the reactor pool.

The Reactor Plant Systems are:

- a) Moderator Temperature Control System (MTCS)
- b) Reactor Water Purification System (RWPS)
- c) Demineralised Water System (DWS)

MTCS is used to remove the heat generated in the core, to heat or cool the pool water, to control the pool level and make up water supply and skimming the water surface of the reactor pool. The system is capable of removing heat from a nominal core thermal load of 10 kW. Also it can cool or heat the moderator up to 2°C/h before startup. The system consist of primary main circuit, skimming circuit and secondary circuit. The primary circuit has a pump and heat exchanger. The skimming circuit has a skimmer, tank and pump. The secondary circuit has a water chilling/heating unit and expansion tank.

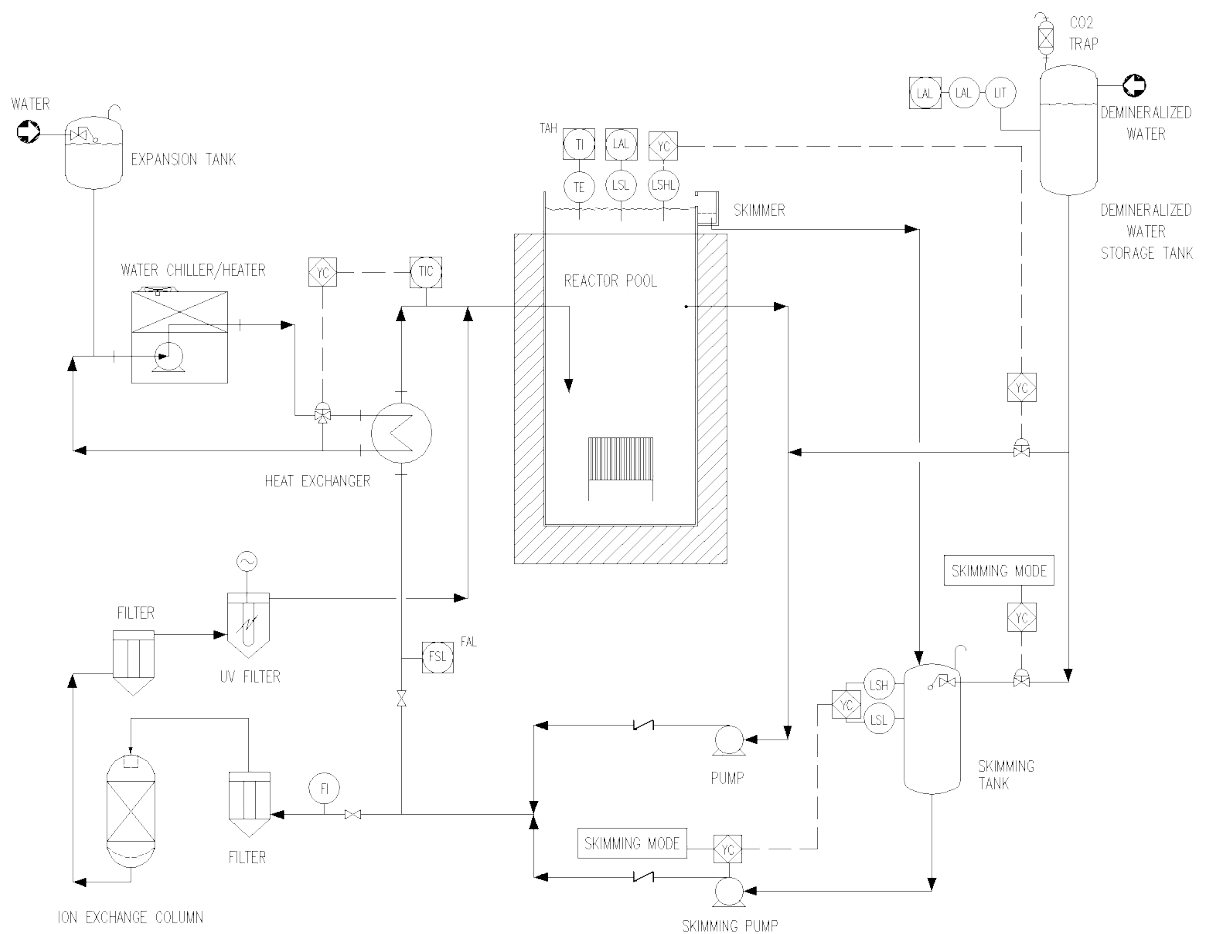


Figure 2.10 : Reactor Process Plant Systems [21].

RWPS is used to control the pool water quality, remove water impurities such as dust, corrosion products and particles. The system consist of a filter, a resin column, a resin trap and a UV sterilizer. It can circulate the whole pool water inventory within 24 hours.

DWS is used to store and distribute the demineralised water. The system consist of a plastic storage tank, a CO₂ trap and connections. In Figure 2.10 all of the systems are shown.

2.4.10 Building and Structure

STURE has two sectors. One of them is the Reactor Hall Sector, the other one is the Control Room and Services. The Reactor Hall Sector is 124 m² and the Control Room and Services Sector is 300 m².

The Reactor Hall Sector has isolated and independent foundations. It has a metallic root supported by a columns and beams structure. Metal panels filled with 10 cm thick injected polyurethane filler covers the outer building. Also concrete walls and blocks provides the biological shielding.

The Control Room and Services Sector has the same style of structure with the Reactor Hall Sector except the outer building is covered in conventional and for drywalls standard inner partitions are used.

The concrete shields are conceptually designed to use standard concrete with a density of 2.2 ton/m³. Standard concrete is used as a biological shield which blocks both neutrons and gamma rays and easy to be shaped. In Figure 2.11 the ground floor of the building is shown.

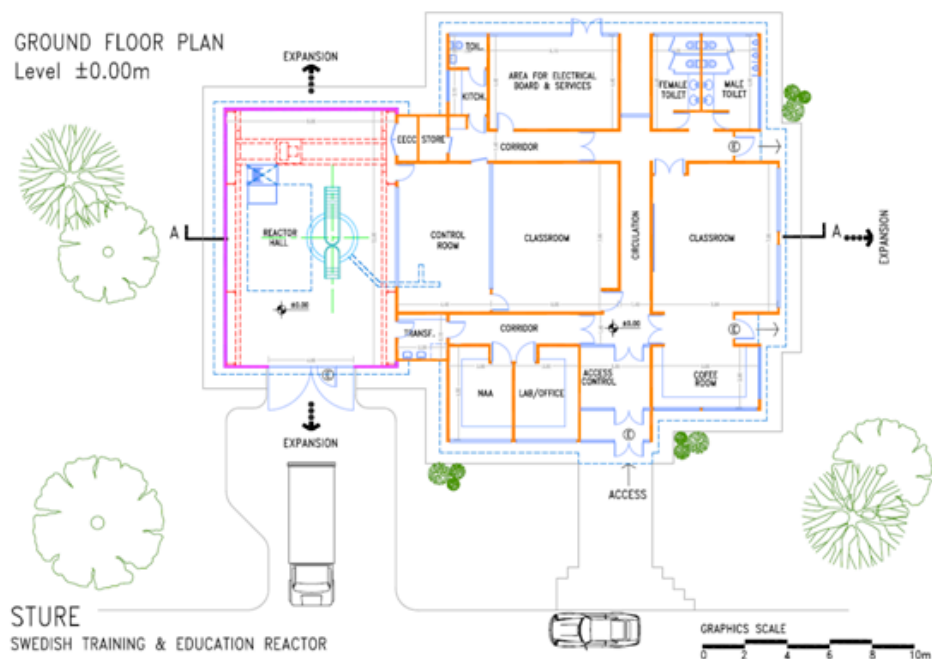


Figure 2.11 : Building Ground Floor [21].

In Figure 2.12 a A-A Section view of the building and Figure 2.13 3D view of the building is shown.

SECTION A-A

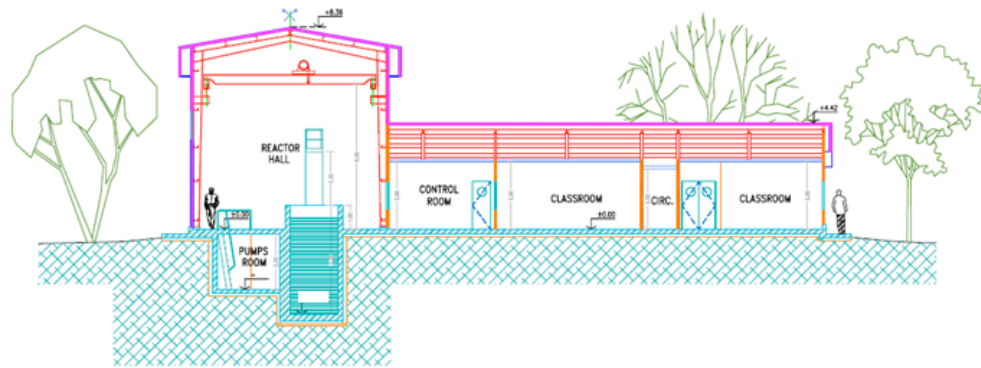


Figure 2.12 : Building Section A-A View [21].



Figure 2.13 : 3D Building View [21].

3. STURE TRACE MODEL

3.1 Task Description

The focus of this chapter is to develop a model of STURE with the TRACE code and learn typical schematics and arrangements of components in nuclear power plants, the function and performance of typical components in nuclear power plants and modern computational tools used for analysis of nuclear power plants. In order to acquaint with these, we use the simulation tools SNAP and TRACE. In this part of the thesis hydraulic components of the circuit (vessel, pipe, break, valve), main heat structure of the circuit, the controlled variables (pressure, flow area, etc), break in the junction of the reactor vessel and how a break is initiated will be identified. By the help of the simulation, tools SNAP and TRACE the main parameters (pressure, temperature and mass flow) in the core, vessel, pipe, valve and breaks for 100% power will be retrieved. Later by making some modifications in the model, a LOCA case will be simulated and the results in the transient mode of the simulation tools will be presented.

3.2 Description of the Simulation Tools

A short description of the simulation tools SNAP and TRACE and the plotting tool Apt Plot will be given.

3.2.1 SNAP (The Symbolic Nuclear Analysis Package)

The Symbolic Nuclear Analysis Package (SNAP) consists of a suite of integrated applications designed to simplify the process of performing engineering analysis. SNAP is built on the Common Application Framework for Engineering Analysis (CAFEAN) which provides a highly flexible framework for creating and editing input for engineering analysis codes as well as extensive functionality for submitting, monitoring, and interacting with the codes.

The SNAP application suite includes the ModelEditor, JobStatus and the Configuration Tool client applications as well as the Calculation Server. In this context, a client application is one that typically is run on the local machine and provides a graphical user interface. A server application is one that runs in the background or on a different computer to provide access to data. The ModelEditor is the primary SNAP client-side user interface. It is responsible for the development and modification of input models for the supported analysis codes. It is also responsible for animating the results of those analyses using the Animation plug-in. The Configuration Tool is used to configure properties for the SNAP client applications as well as to startup, shutdown and configure the Calculation Server. Job Status is used to display the status of jobs on a server as well as to create new jobs by importing local data files. The Calculation Server provides control of and communication with active and completed calculations [22].

3.2.2 TRACE (TRAC/RELAP Advanced Computational Engine)

The TRAC/RELAP Advanced Computational Engine (TRACE - formerly called TRAC-M) is the latest in a series of advanced, best-estimate reactor systems codes developed by the U.S. Nuclear Regulatory Commission for analyzing transient and steady-state neutronic thermal-hydraulic behavior in light water reactors. It is the product of a long term effort to combine the capabilities of the NRC's four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into one modernized computational tool.

TRACE has been designed to perform best-estimate analyses of loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in pressurized light-water reactors (PWRs) and boiling light-water reactors (BWRs). It can also model phenomena occurring in experimental facilities designed to simulate transients in reactor systems. Models used include multidimensional two-phase flow, non-equilibrium thermodynamics, generalized heat transfer, reflood, level tracking, and reactor kinetics. Automatic steady-state and dump/restart capabilities are also provided.

The partial differential equations that describe two-phase flow and heat transfer are solved using finite volume numerical methods. The heat-transfer equations are solved using a semi-implicit time-differencing technique. The fluid-dynamics equations in the spatial one-dimensional (1D), and three-dimensional (3D) components use, by default, a multi-step time-differencing procedure. A more straightforward semi implicit time-differencing method is also available, should the user demand it. The finite difference equations for hydrodynamic phenomena form a system of coupled, nonlinear equations that are solved by the Newton-Raphson iteration method. The resulting linearized equations are solved by direct matrix inversion. For the 1D network matrix, this is done by a direct full-matrix solver; for the multiple-vessel matrix, this is done by the capacitance-matrix method using a direct banded-matrix solver [23].

3.2.3 AptPlot

AptPlot is a 2D plotting tool designed for creating production quality plots of numerical data. AptPlot contains extensive scripting and GUI support for the manipulation and analysis of data sets.

AptPlot is intended as a drop-in replacement for AcGrace: the NRC Analysis Code version of Grace which has been modified to provide direct interfaces to several analysis codes, NRC Databank files, the Symbolic Nuclear Analysis Package (SNAP), and to provide an easier means of performing calculations using data from these files. AcGrace is a descendant of Xmgr5 which was created from software originally developed by Paul Turner, and later maintained by the Xmgr Team, coordinated by Evgeny Stambulchik.

Although Grace has provided extensive capability for plotting and data analysis tools for several years it is also host to several limitations, primarily associated with execution under Microsoft Windows™. It is written as an X-Windows Motif application in a combination of C, LEX, YACC, and IDL coding. The software is specifically designed for Unix machines; users working with operating systems that do not provide Unixlike functionality often found installing and maintaining AcGrace difficult. With this in mind, APT has developed AptPlot as a functional clone of AcGrace using the Java programming language.

In addition to vastly improved portability, authoring the tool in Java eases the addition of a host of other improvements, such as including a plug-in interface. This interface has been used to remove all analysis code specific functionality from AptPlot: those interested in such support can install the associated plug-in [24].

3.3 Model Description

Here in Figure 3.1, a model of STURE is shown. The model consists of a vessel, a pipe and a break component. The vessel describes the reactor pool, which is filled with water that holds the core inside. The break describes the boundary conditions. Since STURE is pool type of reactor, it is an open tank. There is also a pipe in this model because a break cannot be connected directly to a vessel.

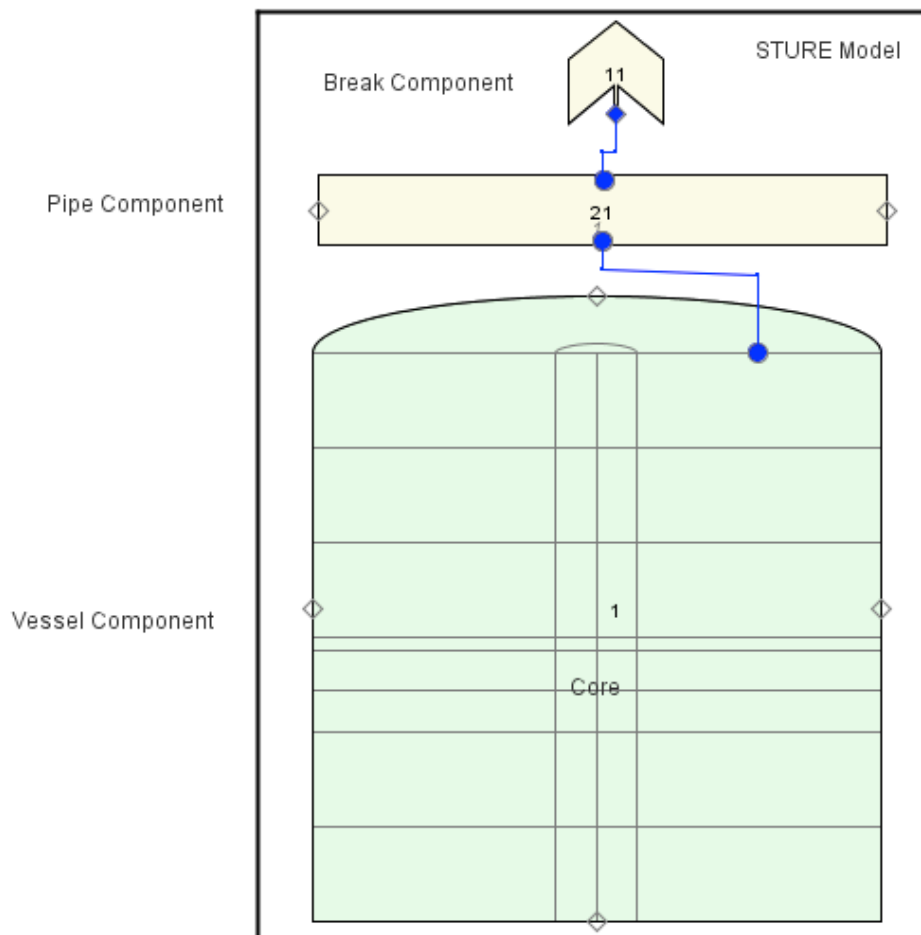


Figure 3.1 : STURE Model.

3.3.1 Vessel Component

In Model Options menu TRACE Version 5.0 Patch 2 and steady state calculation are selected. From the fluid options in this menu light water, H₂O is selected. In Hydraulic Components menu first, a 3D vessel is created and geometric data is entered to the program. There are eight axial levels in the model which are divided from bottom to top as 0.5, 0.5, 0.215, 0.215, 0.07, 0.5, 0.5 and 0.5 m also there are 2 radial rings one symbolizes the core radius 0.217 m and the other is the vessel radius minus core radius, 1.283 m. In addition, the vessel is divided into four azimuthal sectors.

The core is situated in 3rd and 4th levels and in the 1st ring. The sum of the lengths of the axial levels gives us the height of the STURE reactor, which is 3 m. In addition, the sum of the radii gives us the radius of STURE reactor, which is 1.5 m. After the geometry data is entered, then the Volumetric and Edge data in axial, radial and azimuthal directions is calculated by a MATLAB script and entered to the program.

After entering the data on the model than the core is described. In the model, the core is situated in the 3rd and the 4th levels and in the 1st ring. In the Thermal part of the program, four Heat Structures due to four azimuthal sectors are added to the model.

Table 3.1: Heat Structure Radial Mesh

Material Region	Inner Length (m)	Outer Length (m)	Material
1	0.0	8.48E-4	Material 1 (Uranium Oxide)
2	8.48E-4	1.696E-3	Material 1 (Uranium Oxide)
3	1.696E-3	2.544E-3	Material 1 (Uranium Oxide)
4	2.544E-3	3.392E-3	Material 1 (Uranium Oxide)
5	3.392E-3	4.24E-3	Material 1 (Uranium Oxide)
6	4.24E-3	4.315E-3	Material 3 (Gap Gases)
7	4.315E-3	4.92E-3	Material 2 (Zircaloy)

In the program the heat structures represents the fuel rods. On Table 3.1 one fuel rod is described. The clad radius $4.92\text{E-}3$ m, gap radius $4.315\text{E-}3$ m and fuel radius $4.24\text{E-}3$ m is selected due the design. In the beginning, one fuel rod is created in each heat structure. The total number of fuel rods is divided into four, due to the four azimuthal levels and for each heat structure; this number is entered to the Surface Multiplier section in the Heat Structures menu, in order to extend it to the whole amount of fuel rods.

After the fuel rods are described, they are powered by using the Powered Components section in the Power menu. The power is selected as 10.000W , which is selected in the conceptual design. In the power menu, a power shape is added to the model. Radial power density is turned on by selecting 1 for the uranium oxide part because only this part creates power. In Table 3.2, the radial power density distribution is shown.

Table 3.2: Editing Radial Location

Radial Locations (m)	Radial Power Density
0.0	1
$8.48\text{E-}4$	1
$1.696\text{E-}3$	1
$2.544\text{E-}3$	1
$3.392\text{E-}3$	1
$4.24\text{E-}3$	1
$4.315\text{E-}3$	0
$4.92\text{E-}3$	0

3.3.2 Pipe Component

In order to connect the break component to the vessel component, a pipe component need to be used. From the Hydraulic components menu a Pipe component with length of 1 m and a hydraulic diameter of 3 m is created. A flow area data of 7.0685835 m² and a volume data of 7.0685835 m³ is entered to the program. In addition, room temperature 300 K and atmospheric pressure 1 bar, is selected as the boundary conditions. Finally the pipe component is connected to the top of the vessel component.

3.3.3 Break Component

After this, the break component is connected to the vessel component. Break component is used as a boundary condition. Atmospheric pressure 1.0E5 Pa and room temperature 300 K is used for STURE reactor because it is a pool type reactor.

3.4 Simplified Thermal Hydraulic Calculations

In cylindrical fuel rods, it is assumed that the heat is produced at the constant rate q''' throughout the fueled portion of the rod. Accordingly, the temperature does not vary along the length of the rod and it is possible to relate the total heat flow out of the entire rod to the difference in temperature between the center and surface of the rod.

Power density, rate of energy production per unit volume is defined as [26];

$$q''' = q'''_{\max} \cos\left(\frac{\pi Z}{H}\right) \quad (3.1)$$

where, q''' = power density, q'''_{\max} = maximum power density and H = height of the core.

Power of the reactor is defined as [26];

$$P = \int_{-H/2}^{H/2} \int_0^R q'''_{\max} \cos\left(\frac{\pi Z}{H}\right) J_0\left(\frac{2.405r}{R}\right) 2\pi r dr dz \quad (3.2)$$

where R = radius of the core.

If Eq. (3.2) is integrated, q'''_{\max} is calculated;

$$q_{\max}''' = \frac{P}{0.863384HR^2} = \frac{10000}{0.863384 \times 0.43 \times 0.22396} = 5.37 \cdot 10^5 \frac{W}{m^3}$$

where $P = 10000$ W is the power, $H = 0.43$ m is the core height and $R = 0.22396$ m is the core radius.

The temperature drop in the fuel is defined as [27];

$$T_0 - T_f = \frac{q_{\max}''' r_f^2}{4k_f} \quad (3.3)$$

where $r_f = 0.00424$ m is the fuel radius and $k_f = 7.59$ W/mK is thermal conductivity of UO₂ 300 K [28]. Temperature drop in the fuel is calculated;

$$T_0 - T_f = \frac{5,37 \cdot 10^5 \times 0.00424^2}{4 \times 7.59} = 0.318K$$

The temperature drop in the gas gap is defined as [27];

$$T_f - T_g = \frac{q_{\max}''' r_f}{2k_g} l_g \quad (3.4)$$

Where $l_g = 0.000075$ m is the gap thickness and $k_g = 0.142$ W/mK is thermal conductivity of helium at 300 K [29]. Temperature drop in the gas gap is calculated;

$$T_f - T_g = \frac{5,37 \cdot 10^5 \times 0.00424}{2 \times 0.142} \times 0.000075 = 0.601K$$

The temperature drop in the fuel clad is defined as [27];

$$T_g - T_c = \frac{q_{\max}''' r_f^2}{2k_c r_g} l_c \quad (3.5)$$

where $r_g = 0.004315$ m is the gap radius, $l_c = 0.000605$ m is the gap thickness and $k_c = 21.5$ W/mK is thermal conductivity of Zircalloy-4 at 300 K [30]. Temperature drop in the fuel clad is calculated;

$$T_g - T_c = \frac{5,37 \cdot 10^5 \times 0.00424^2}{2 \times 21.5 \times 0.004315} \times 0.000605 = 0.031K$$

The temperature drop between the fuel clad and the coolant is defined as [27];

$$T_c - T_{\infty} = \frac{q_{\max}''' r_f^2}{2r_c h} \quad (3.6)$$

where $r_c = 0.00492$ m is the clad radius, h is the convective heat transfer coefficient for natural convection. For vertical cylinders in natural convection, W.H. McAdams suggested the following correlation [31];

$$h = \frac{0.13kRa^{1/3}}{D_h} \text{ for } 2.10^7 \leq Ra \leq 3.10^{10} \quad (3.7)$$

where k is the thermal conductivity of the coolant at 300 K, D_h is the hydraulic diameter defined as [32];

$$D_h = \sqrt{D^2 - nD_f^2} \quad (3.8)$$

where D is the core diameter, n is the number of fuel pins and D_f is the fuel diameter. Hydraulic diameter D_h is calculated;

$$D_h = \sqrt{D^2 - nD_f^2} = \sqrt{0.44792^2 - 463 \times 0.00424^2} = 0.409 \text{ m}$$

Ra is Rayleigh number defined as [32];

$$Ra = Gr \cdot Pr \quad (3.9)$$

where Gr is Grashof number and Pr is Prandtl number defined as [32];

$$Gr = \frac{D^3 \rho^2 g (T_c - T_\infty) \beta}{\mu^2} \quad (3.10)$$

$$Pr = \frac{\mu C_p}{k} \text{ for } Ra \leq 2.10^{12} \quad (3.11)$$

where $D = 0.44792$ m is the core diameter, $g = 9.81$ m/s² is the gravity and properties of water at 300 K is obtained by making iteration [29];

Density of water : $\rho = 996.45$ kg/m³

Thermal expansion coefficient : $\beta = 0.0002754$ 1/K

Dynamic viscosity : $\mu = 0.0008592$ Ns/m²

Specific heat : $C_p = 4180.2$ J/kgK

Thermal conductivity : $k_w = 0.58$ W/mK

Prandtl number is calculated;

$$Pr = \frac{\mu C_p}{k_w} = \frac{0.0008592 \times 4180.2}{0.58} = 6.1925 \text{ for } Ra \leq 2.10^{12}$$

Grashof number is calculated;

$$Gr = \frac{0.44792^3 \times 996.45^2 \times 9.81 \times 0.0002754 \times \Delta T}{0.0008592^2} = 3.265 \times 10^8 \Delta T$$

Rayleigh number is calculated;

$$Ra = Gr \cdot Pr = 326555342 \Delta T \times 6.1925 = 2.022 \times 10^9 \Delta T$$

Convective heat transfer coefficient for natural convection, h is calculated;

$$h = \frac{0.13 k_w Ra^{1/3}}{D_h} = \frac{0.13 \times 0.58 \times (2.022 \times 10^9 \times \Delta T)^{1/3}}{0.409} = 233.085 \Delta T^{1/3}$$

If this value of h is inserted in Eq. (3.6), it becomes;

$$\Delta T = \frac{q_{\max}'' r_f^2}{2 r_c \cdot 233.085 \Delta T^{1/3}} = \frac{5,37.10^5 \times 0.00424^2}{2 \times 0.00492 \times 233.085 \Delta T^{1/3}} = \frac{4.20929}{\Delta T^{1/3}}$$

If Eq. (3.6) is solved analytically, the temperature drop between the fuel clad and the coolant is calculated;

$$\Delta T^{4/3} = 4.20929 \rightarrow \Delta T = 4.20929^{3/4} \rightarrow T_c - T_\infty = 2.939 \text{ K}$$

Total temperature drop between the center of the fuel and the coolant is;

$$T_0 - T_\infty = 3.889 \text{ K}$$

In this case the temperature at the outer surface of the fuel clad is;

$$T_c - T_\infty = 2.939 \text{ K} \rightarrow T_c = T_\infty + 2.939 = 300 + 2.939 = 302.939 \text{ K},$$

The temperature at the inner surface of the fuel clad is;

$$T_g - T_c = 0.031 \text{ K} \rightarrow T_g = T_c + 0.031 = 302.939 + 0.031 = 302.97 \text{ K},$$

The temperature at the surface of the fuel is;

$$T_f - T_g = 0.601 \text{ K} \rightarrow T_f = T_g + 0.601 = 302.97 + 0.601 = 303.572 \text{ K},$$

The temperature at the center of the fuel is;

$$T_0 - T_f = 0.302 \text{ K} \rightarrow T_0 = T_f + 0.302 = 303.572 + 0.302 = 303.89 \text{ K}.$$

3.5 Steady State Calculation and Results

After STURE model is completed, then the model is run under steady state conditions. The results of the controlled variables (pressure, mass flow, etc.) are shown in Figures 3.2 – 3.13.

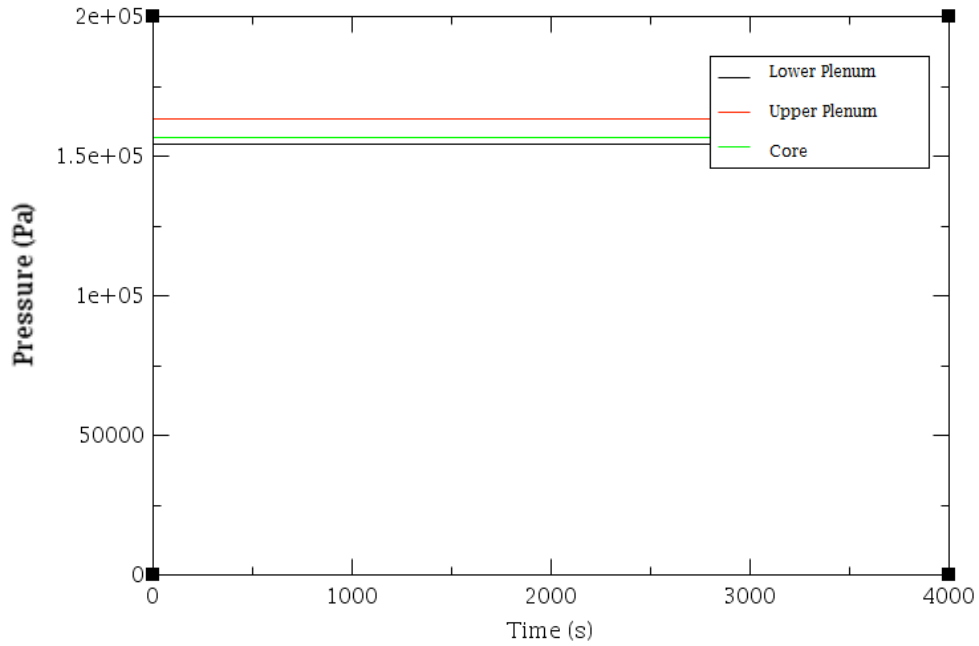


Figure 3.2 : Pressure at the Upper Plenum, Lower Plenum and Core.

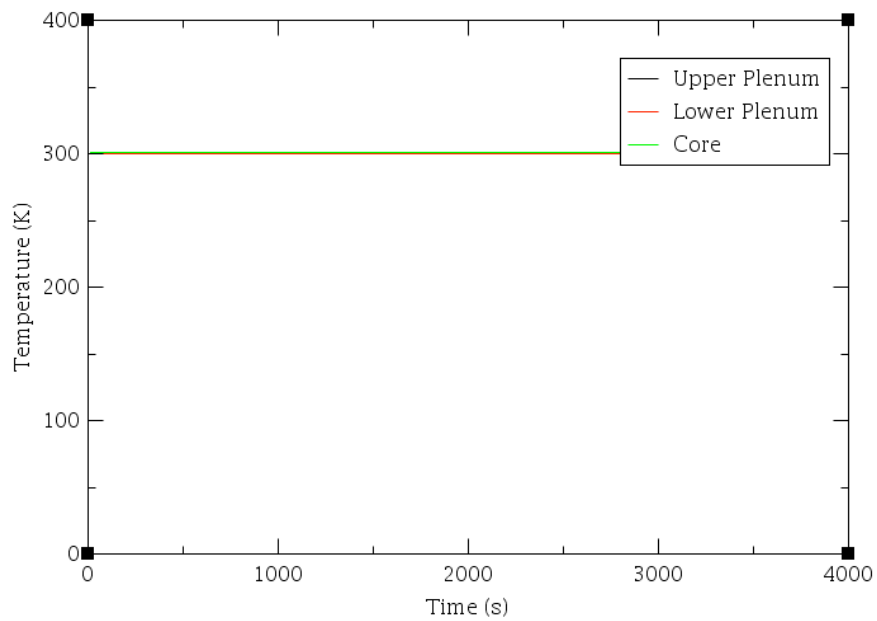


Figure 3.3 : Temperature at the Upper Plenum, Lower Plenum and Core.

In Figure 3.2 the pressures at the core level are a little bit higher than the atmospheric pressure and in Figure 3.3 temperature at the core level is 300 K. In Figure 3.4 liquid mass of the core and Figure 3.5 liquid mass along the vessel are shown.

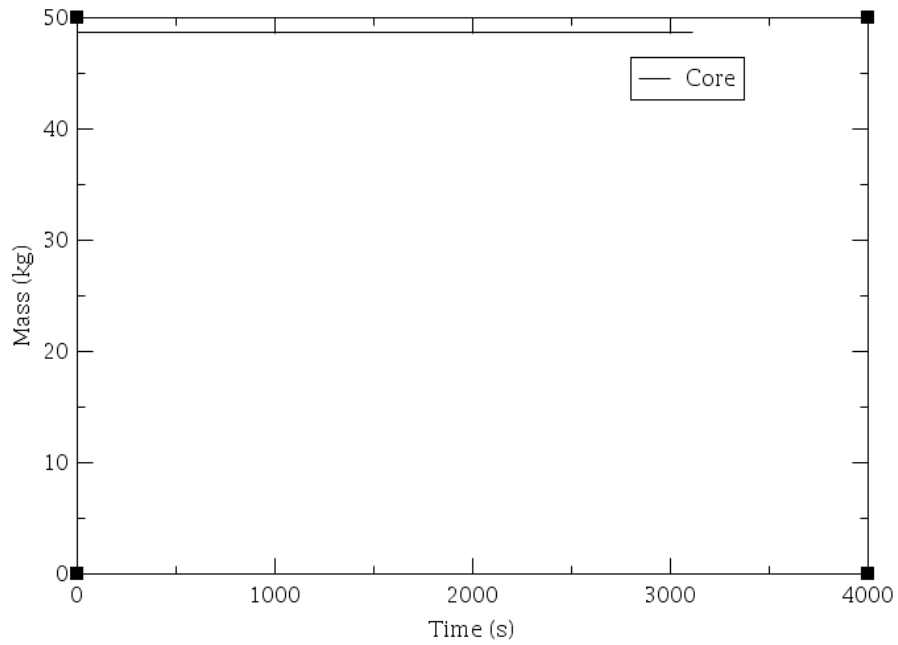


Figure 3.4 : Liquid Mass of the Core.

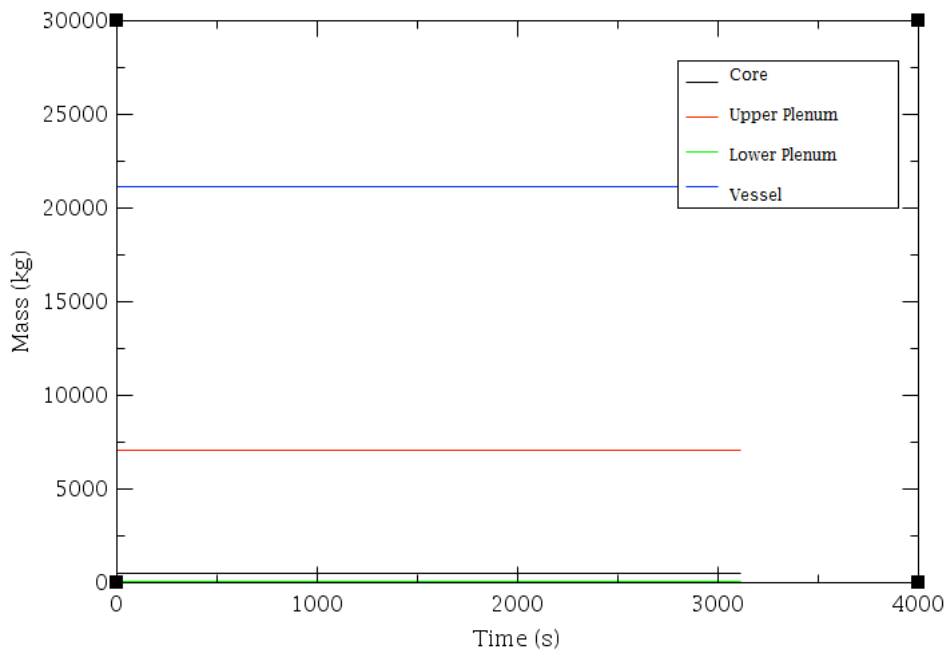


Figure 3.5 : Liquid Mass Along the Vessel Component.

In Figure 3.6 the mass flow of the vessel component is zero which means there is no leakage from the reactor vessel. In Figure 3.7 pressure along the pipe component is shown. At top of the pipe pressure is getting close to the atmospheric pressure level.

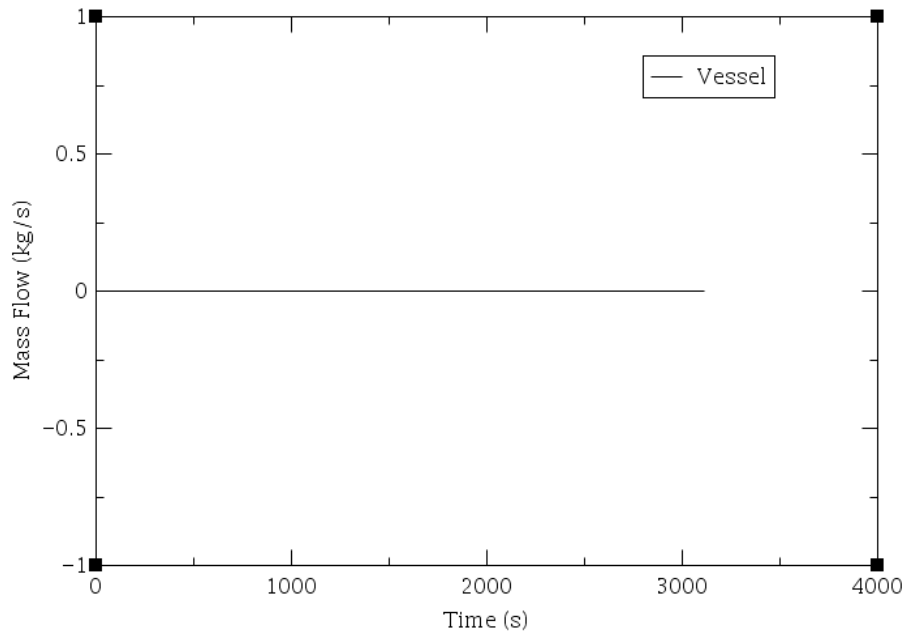


Figure 3.6 : Mass Flow of the Vessel Component.

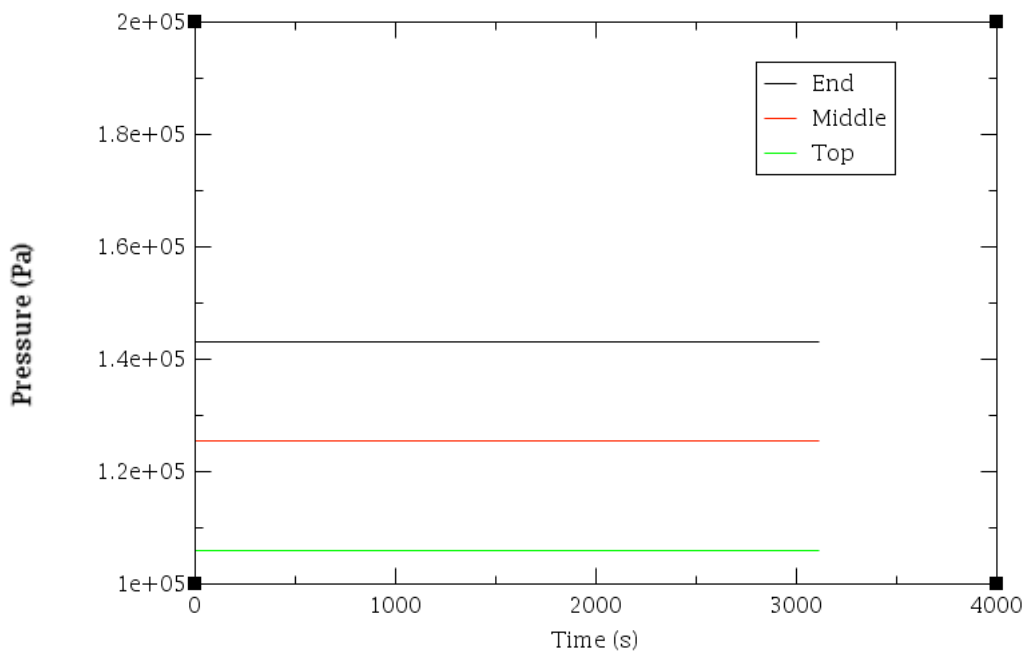


Figure 3.7 : Pressure Along the Pipe Component.

In Figure 3.8 the mass flow of the pipe component is zero and in Figure 3.9 the mass flow rate of the break component is zero which means there is no leakage from the reactor vessel.

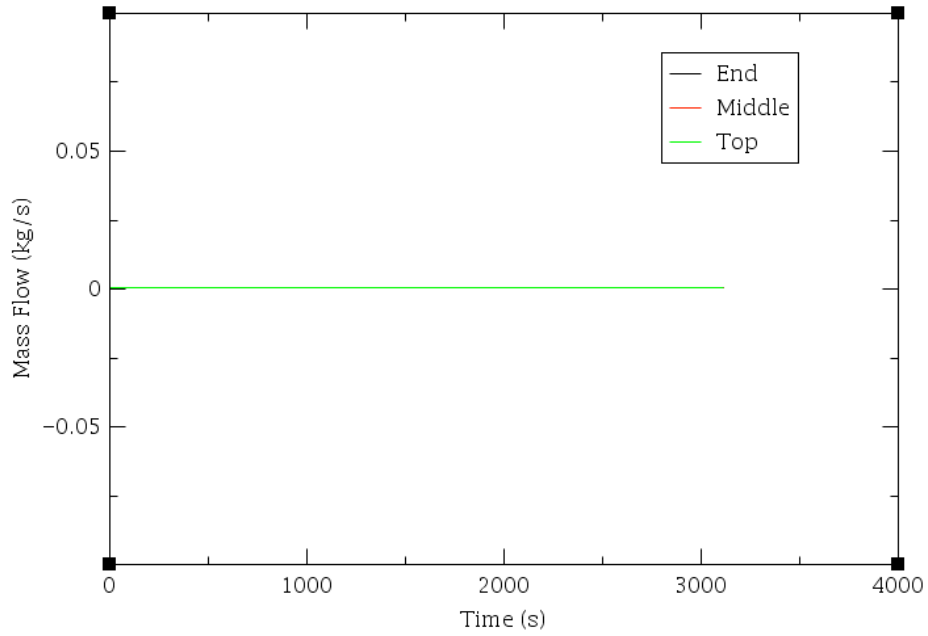


Figure 3.8 : Total Mass Flow Along the Pipe Component.

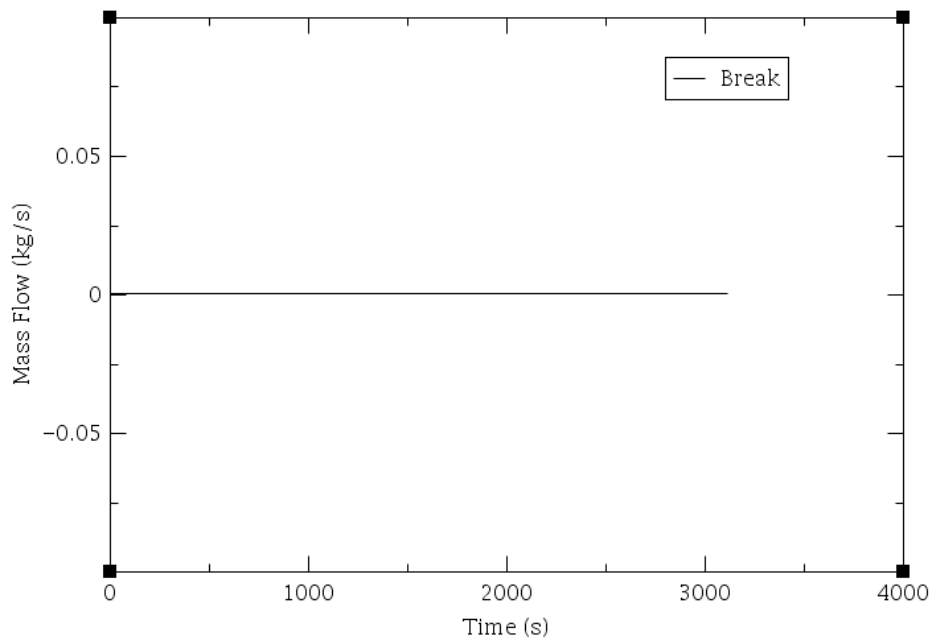


Figure 3.9 : Mass Flow Rate in the Break Component.

In Figure 3.10 the pressure at the break component is at the atmospheric pressure and in Figure 3.11 the temperature at the break component is 300 K which are the boundary conditions of the reactor.

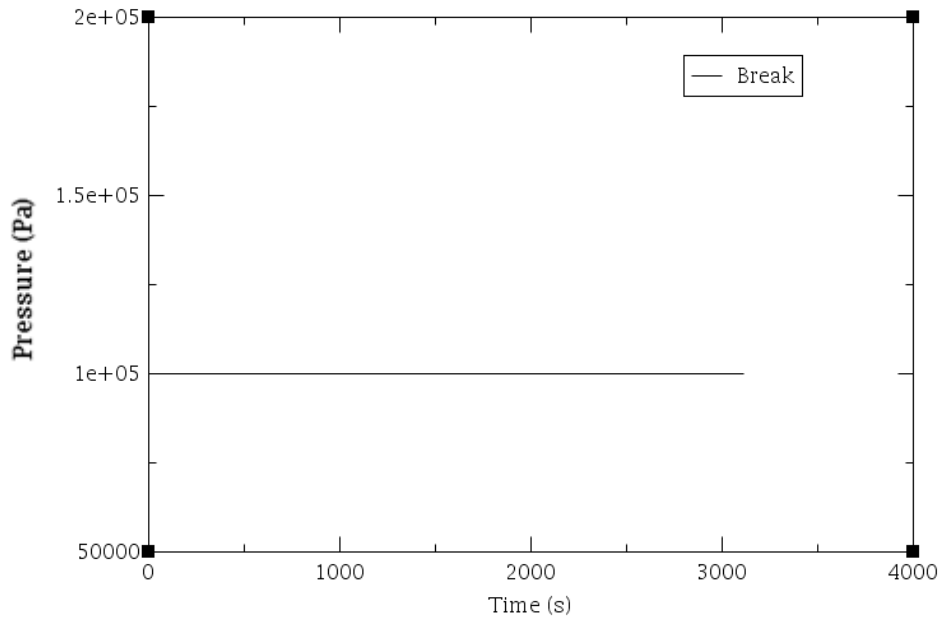


Figure 3.10 : Pressure at the Break Component.

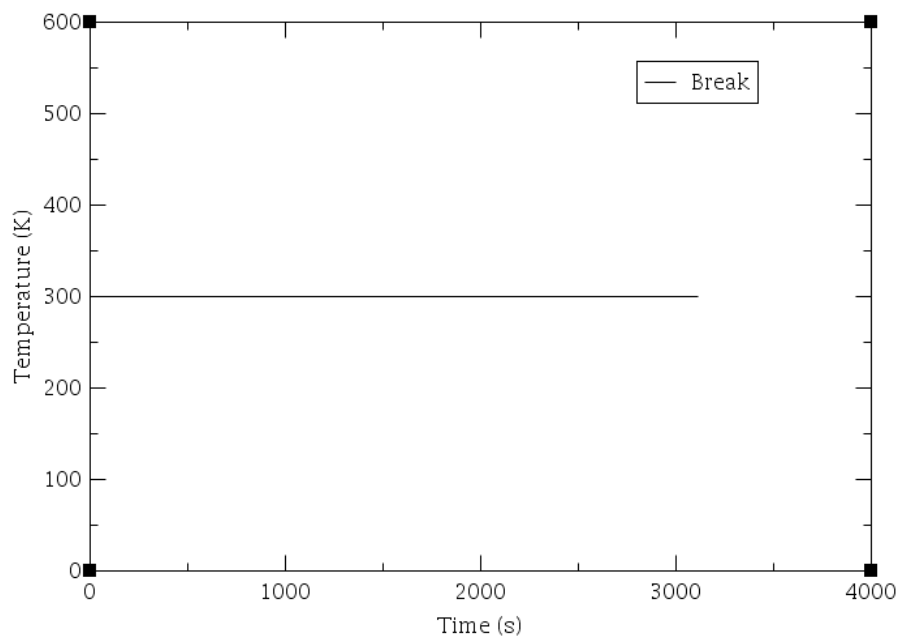


Figure 3.11 : Liquid Temperature at the Break Component.

In Figure 3.12 power of the heat structures are shown. There are 4 heat structures with a power of 2500 W each. In Figure 3.13 the total power of the reactor which is 10000 W is shown.

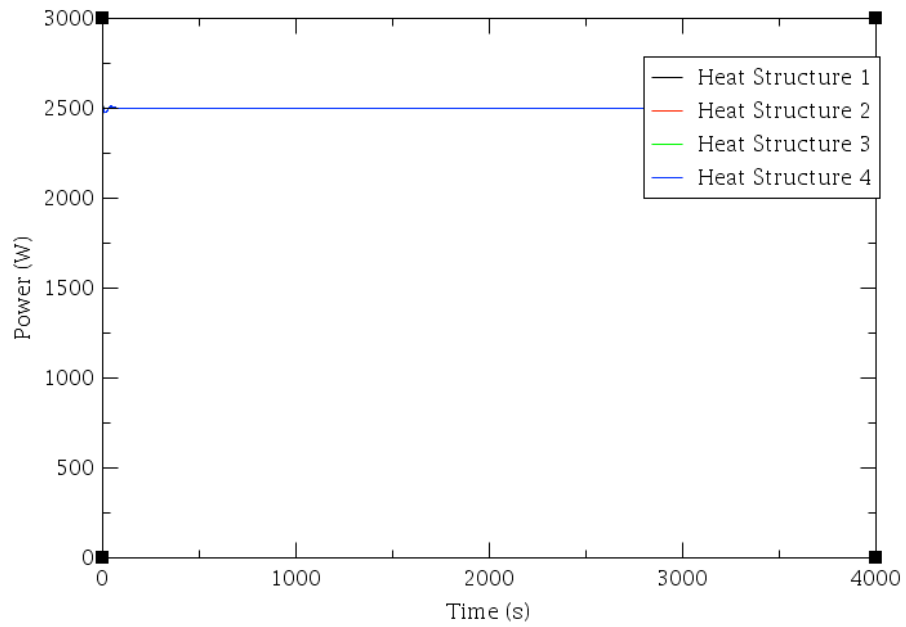


Figure 3.12 : Total Power of the Outer Surface of the Heat Structures.

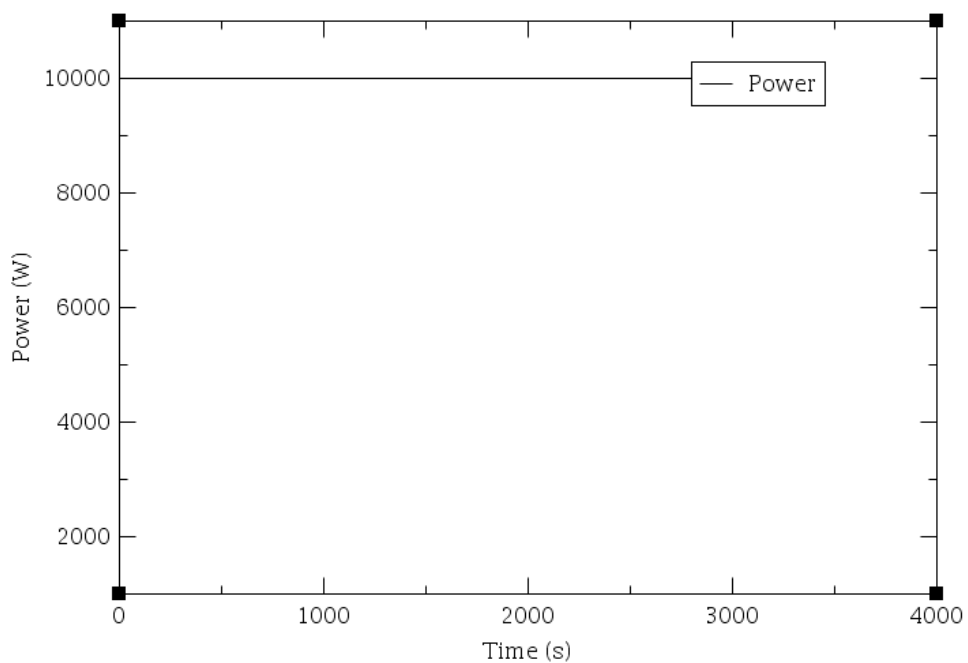


Figure 3.13 : Reactor Power.

3.6 LOCA Analysis

Safety analysis is the study of how the reactor behaves during fault conditions. Safety analysis is a step in the design process and an important part of the safety assessment in the licensing process. Plant safety is continuously monitored during operation and recurrently analyzed in order to maintain and, if needed, raise the level of safety. Safety analysis is carried out in two different ways which complement each other.

Deterministic safety analysis means that the behaviour of the plant after an assumed initial event or malfunction is studied with calculation models, which describe the physical processes in the main reactor systems. The aim of this type of analysis is to verify that permissible values of essential plant variables are not exceeded.

Probabilistic safety analysis concentrates on identifying event sequences, which can lead to core melting, and on studying the reliability of safety systems. The aim of this type of analysis is to indicate weak points in the overall safety design and to provide a basis for improving safety [25].

3.6.1 Type of Events

Events which are important to safety assessment involve primary system variables, such as pressure, temperature, heat flux, coolant flow and coolant density. These events could happen by component failure or by human error. They can also happen by other events such as fire or earthquake. For the aim of analysis, abnormal events are usually grouped into three main categories:

- LOCA (Loss of Coolant Accident), i.e. events caused by a pipe break or leakage in the primary system;
- Transients, a general term for all events (except LOCA) leading to imbalance between the rate of heat release and heat removal in the reactor;
- External events, i.e. earthquake, fire, flooding, explosions, etc.

3.6.2 LOCA (Loss of Coolant Accident)

A LOCA is caused by a pipe break or leak in the primary system of such magnitude that the capacity of the make-up systems is insufficient to replace the lost coolant.

This results in reactor scram, closure of containment isolation valves and initiation of emergency cooling. The course of events is briefly as follows:

1. A break occurs in the primary system and water escapes at high pressure and temperature into the reactor containment.
2. The emergency core cooling systems supply water to keep the core cooled.
3. Radioactive substances, which may be released from the core, are retained within the containment.
4. The containment spray system cools the containment and removes radioactive substances from the containment atmosphere.

A LOCA can be initiated in several ways, e.g. through a pipe break in the primary system, the failure of pressure relief valve to close, or a tube rupture in a steam generator (PWR).

Regarding the size of the break, a distinction is made between large, medium and small LOCA [25].

3.7 LOCA Model Description

In the second part of the thesis a LOCA case for STURE will be investigated. After the Fukushima accident, once again the importance of earthquakes is underlined. The scenario in this part is, if a break happens in the junction part of the reactor pool, the effects of this break will be investigated. In Figure 3.14 the LOCA model is shown.

First the steady state model of STURE which is created in the previous part will be modified by adding a pipe to the bottom part of the reactor pool. From the Hydraulic components menu another pipe component with length of 1 m and a hydraulic diameter of 0.1463 m is created. A flow area data of 0.016810419 m^2 and a volume data of 0.016810419 m^3 is entered to the program.

Between the second pipe and the second break, a valve will be placed. From the Hydraulic components menu a valve component with a flow area data of 0.016810419 m^2 and a hydraulic diameter of 0.1463 m is created.

After this, a second break component is created from the menu of Hydraulic Components. In addition, room temperature 300 K and atmospheric pressure 1 bar, is selected as the boundary conditions.

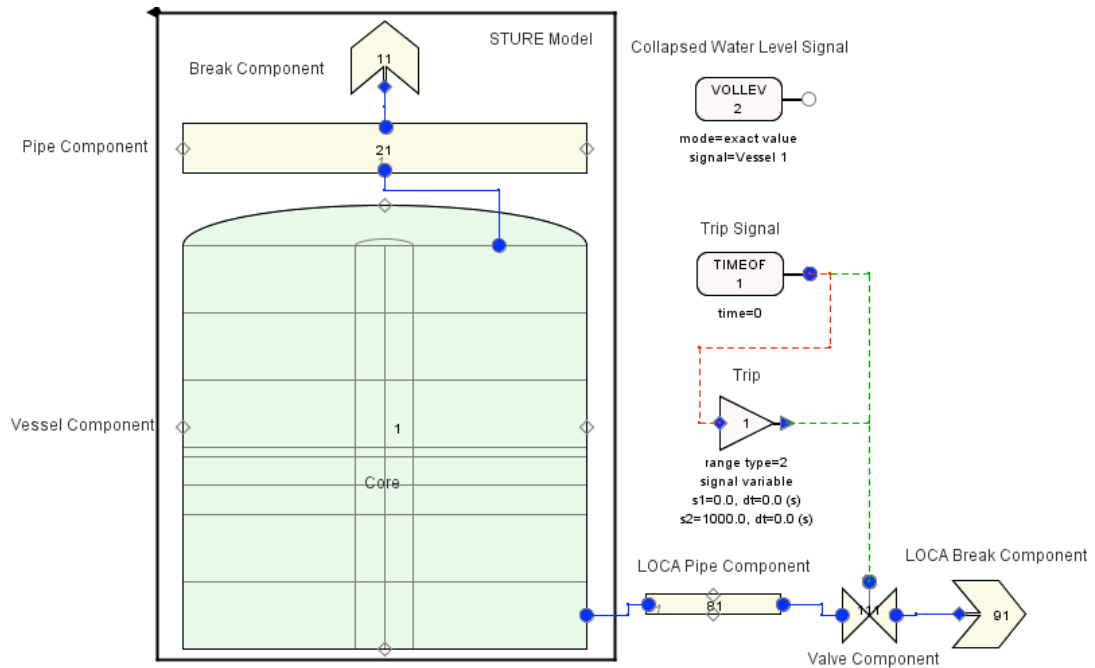


Figure 3.14 : STURE LOCA Model.

From the Control systems menu a time controlled trip is added to the system. The valve will be controlled by a signal which opens the valve after 1000 seconds. After 1000 seconds the trip will receive the signal and opens the valve which will create a leakage in the system. Also another signal will check the water level in the vessel.

3.8 Transient Calculation and Results

After STURE LOCA model is completed, the model is run under transient conditions. The results of the controlled variables (pressure, mass flow, etc.) are shown in Figures 3.15 – 3.28.

In Figure 3.15 the reactor operates in normal conditions and the trip receive the signal at 1000 seconds and opens the valve. In Figure 3.16 the water level in the reactor decreases from 3 m till there is no water in the reactor in 320 seconds. At 1320 seconds there is no water in the reactor vessel and it starts getting cooled by air.

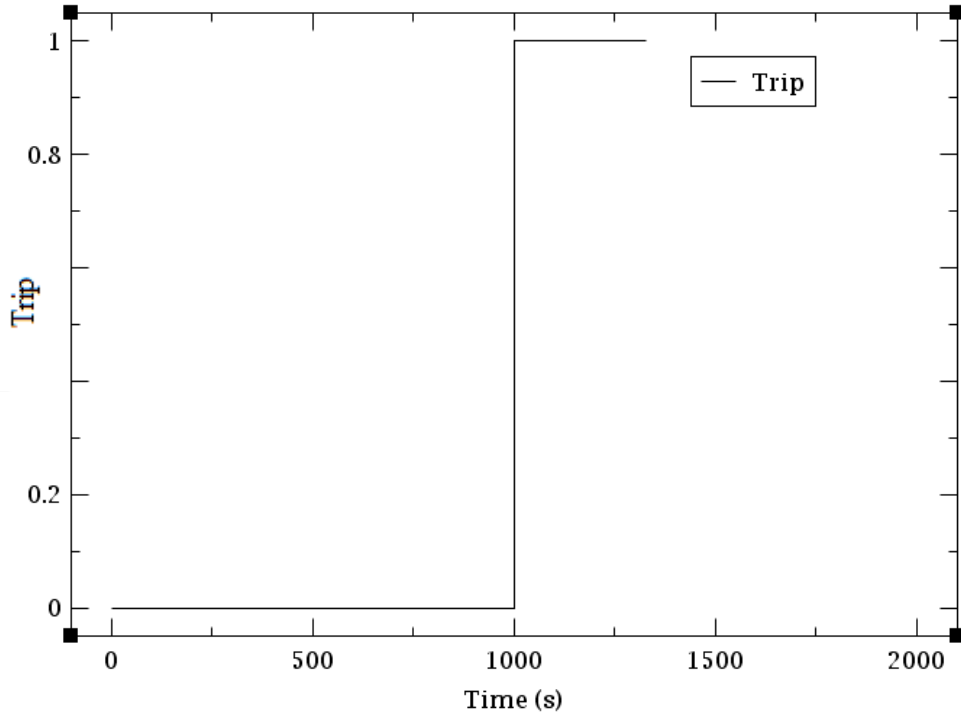


Figure 3.15 : Trip Opens After 1000 seconds.

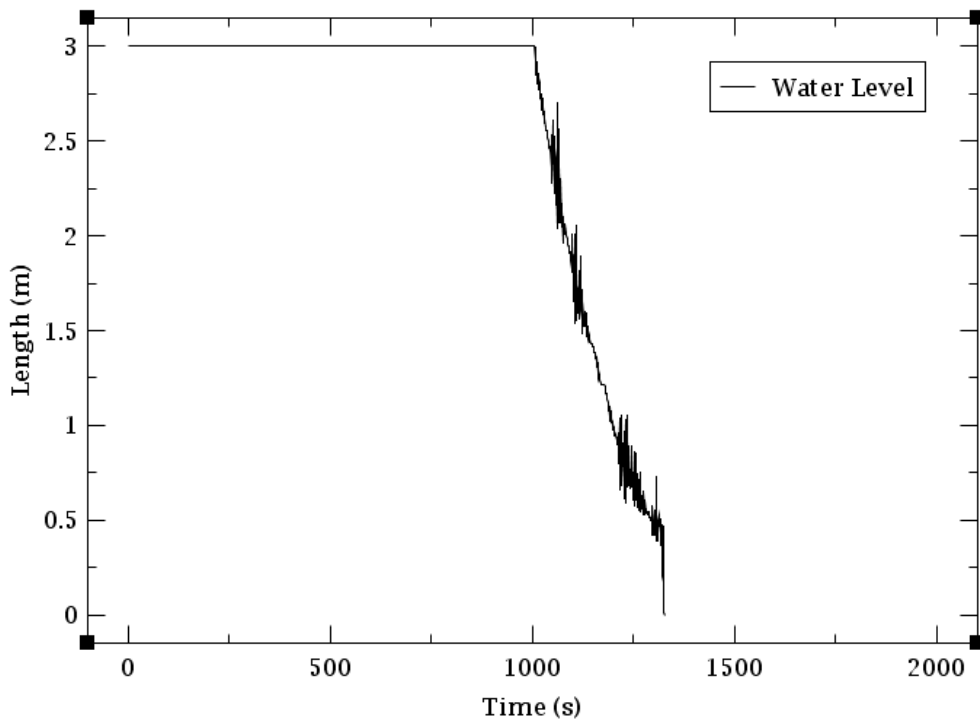


Figure 3.16 : Water Level in the Vessel Component.

In Figure 3.17 the pressure at the core level is shown. After the valve opens the pressure at the core level decreases, then becomes constant at the atmospheric pressure level. In Figure 3.18 the temperature at the core level is shown. After the valve opens the temperature at the core level increases, then becomes constant at 373 K.

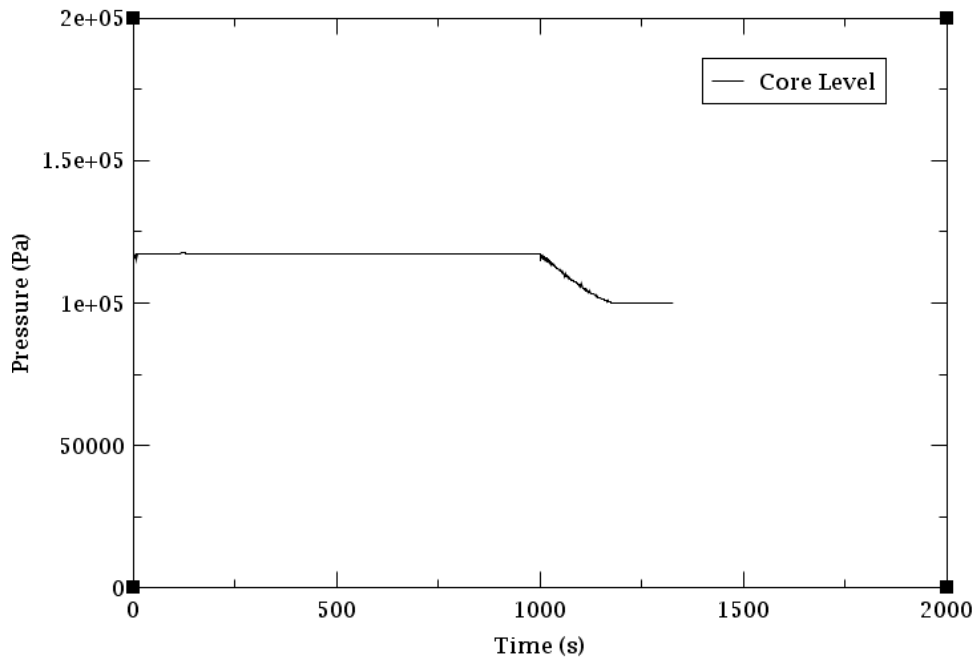


Figure 3.17 : Pressure at the Core.

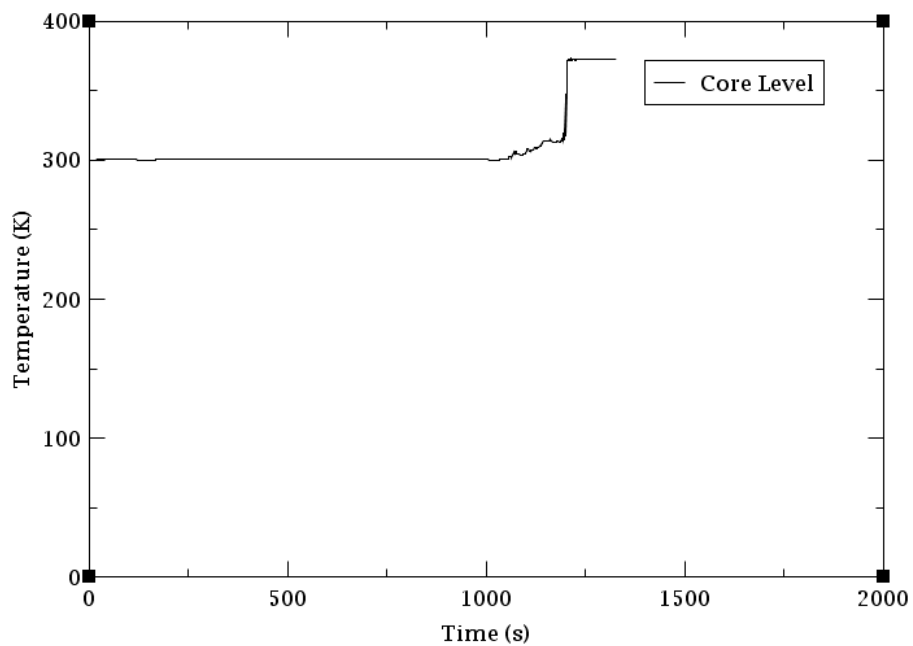


Figure 3.18 : Temperature at the Core Level in Liquid.

In Figure 3.19 the liquid mass of the core is shown. After the valve opens due to the water leakage the liquid mass at the core level decreases till there is no water at the core level. In Figure 3.20 the pressure at the pipe component is shown. The pressure is at the atmospheric pressure level which is the boundary condition for the reactor.

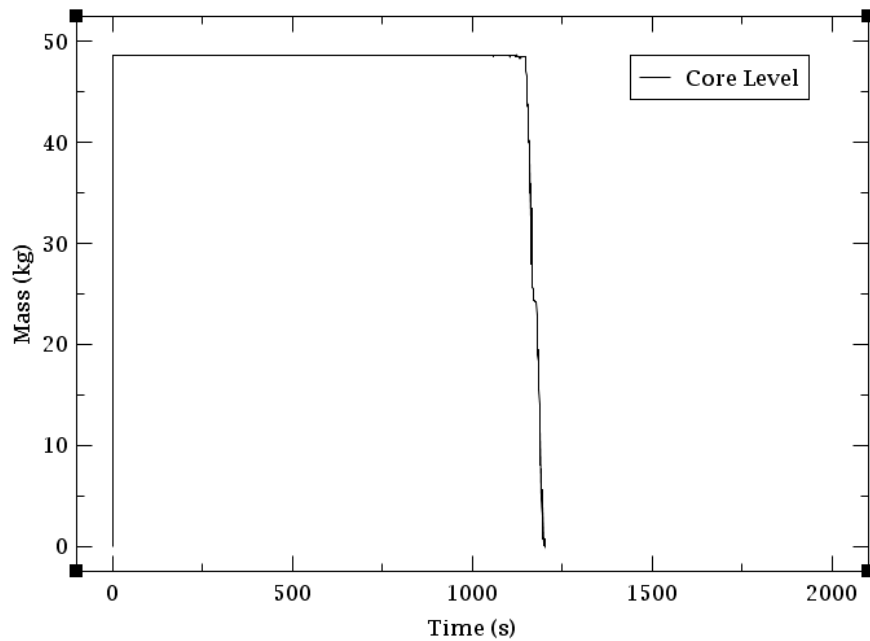


Figure 3.19 : Liquid Mass of the Core.

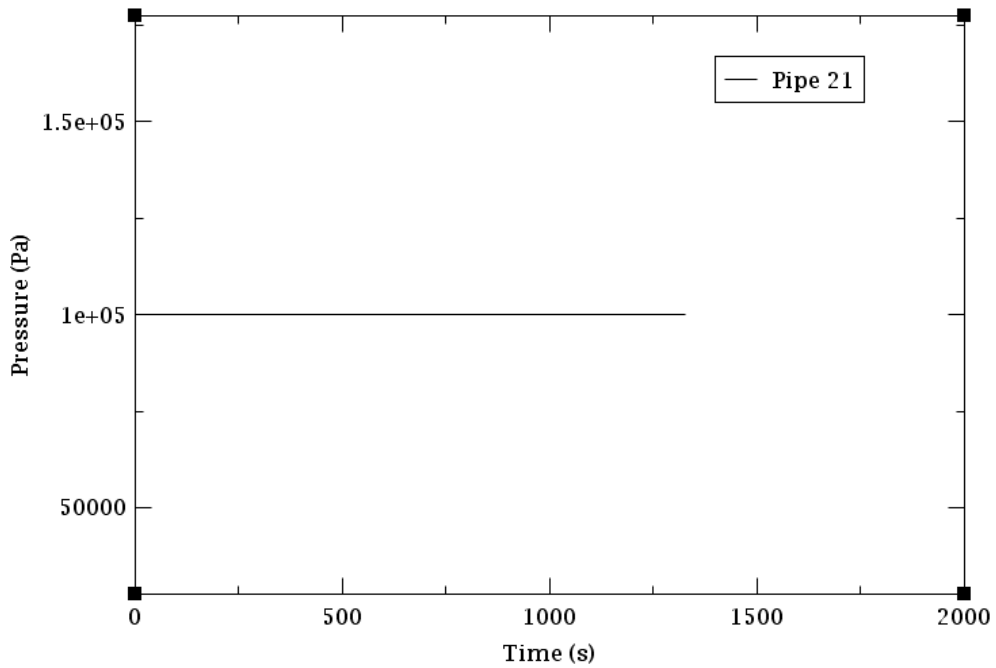


Figure 3.20 : Pressure of the Pipe Component.

In Figure 3.21 the total mass flow at the pipe component is zero which means there is no leakage from the top part of the reactor vessel. In Figure 3.22 the temperature at the pipe component is shown. The temperature is 300 K which is the boundary condition for the reactor.

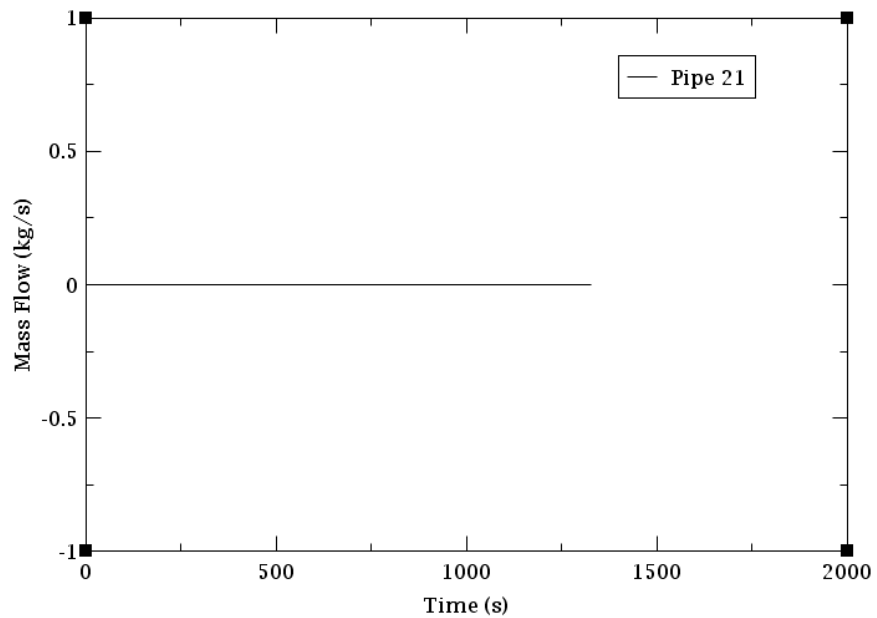


Figure 3.21 : Total Mass Flow at the Pipe Component.

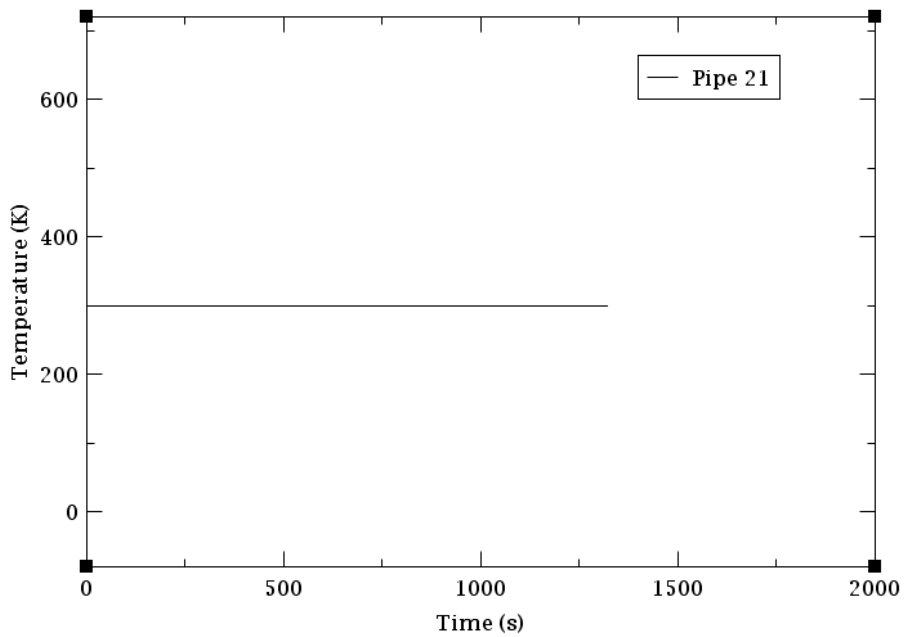


Figure 3.22 : Temperature at the Pipe Component.

In Figure 3.23 the pressure at the break component is at the atmospheric pressure and in Figure 3.24 the temperature at the break component is 300 K which are the boundary conditions of the reactor.

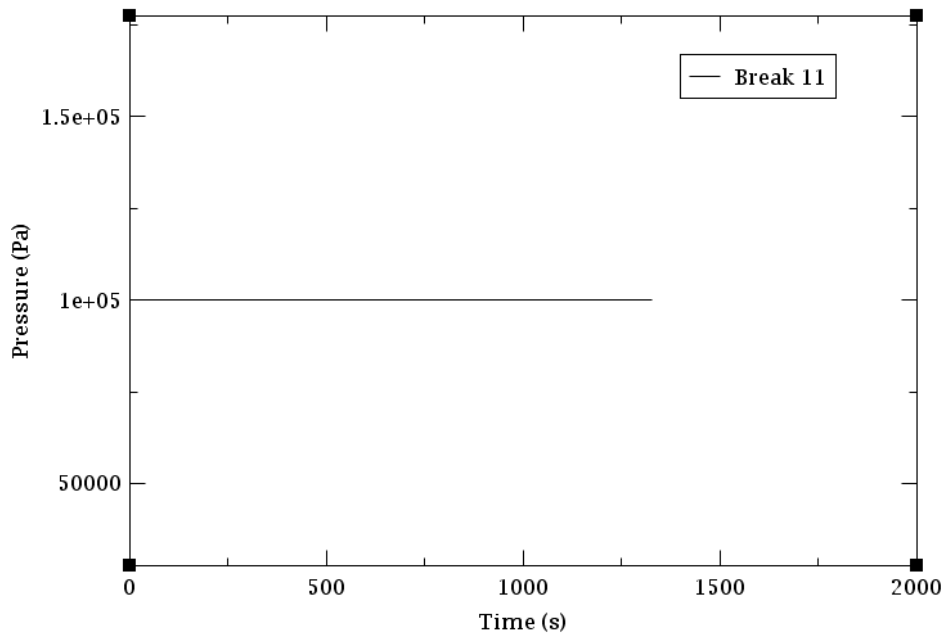


Figure 3.23 : Pressure at the Break Component.

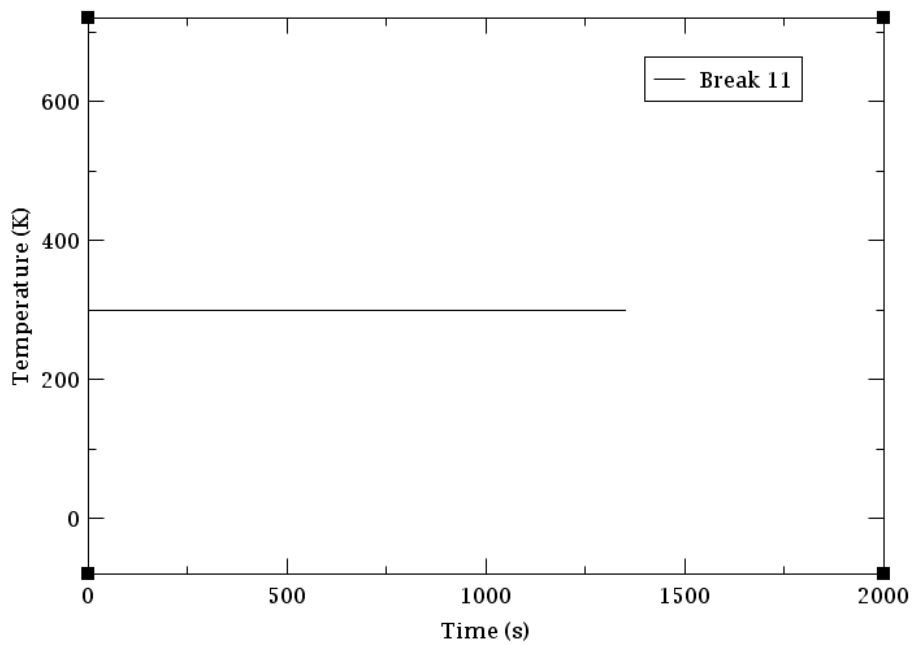


Figure 3.24 : Temperature at the Break Component.

In Figure 3.25 the total power of the reactor which is 10000 W is shown. In Figure 3.26 the pressure at the break component is at the atmospheric pressure level which is the boundary condition of the reactor.

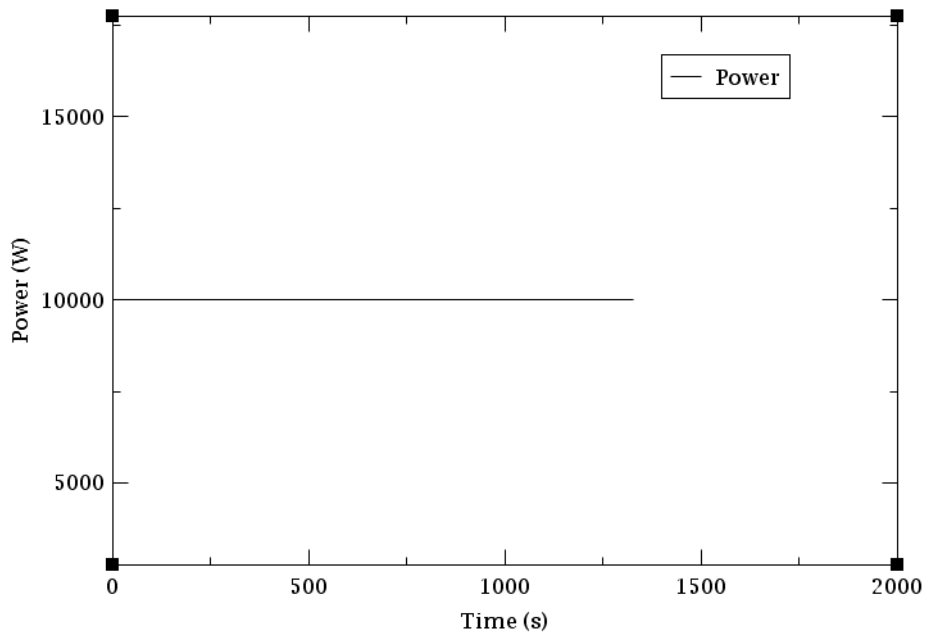


Figure 3.25 : Reactor Power.

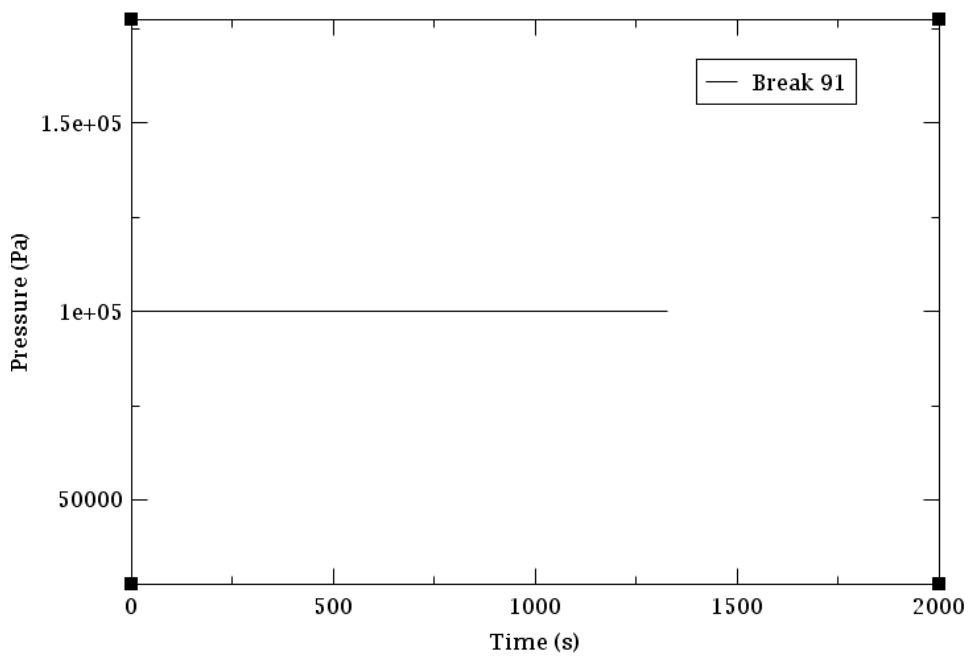


Figure 3.26 : Pressure at the Break Component.

In Figure 3.27 the flow area of the valve component which is 0.016810419 m^2 is shown. In Figure 3.28 the pressure at the valve component is at the atmospheric pressure level which is the boundary condition of the reactor.

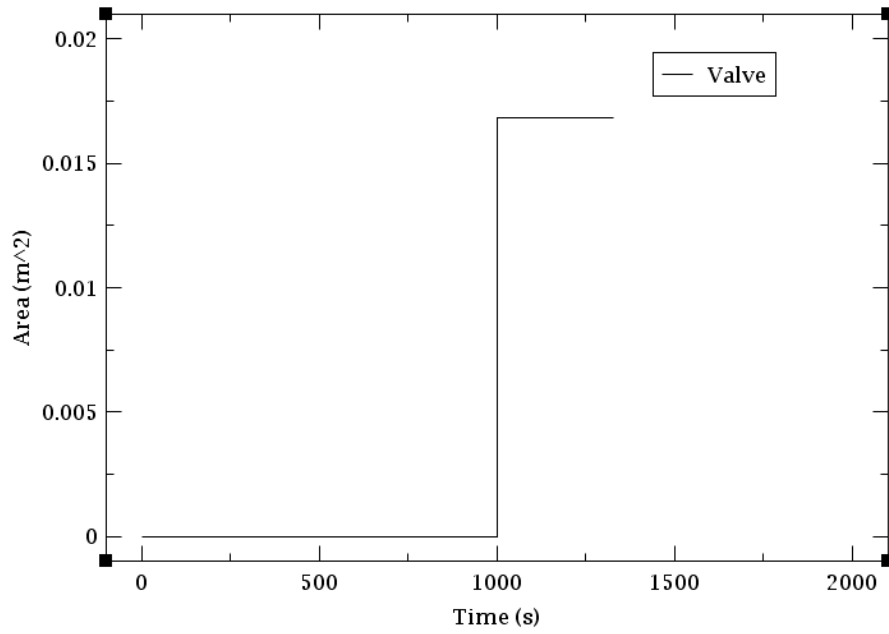


Figure 3.27 : Flow Area of the Valve Component.

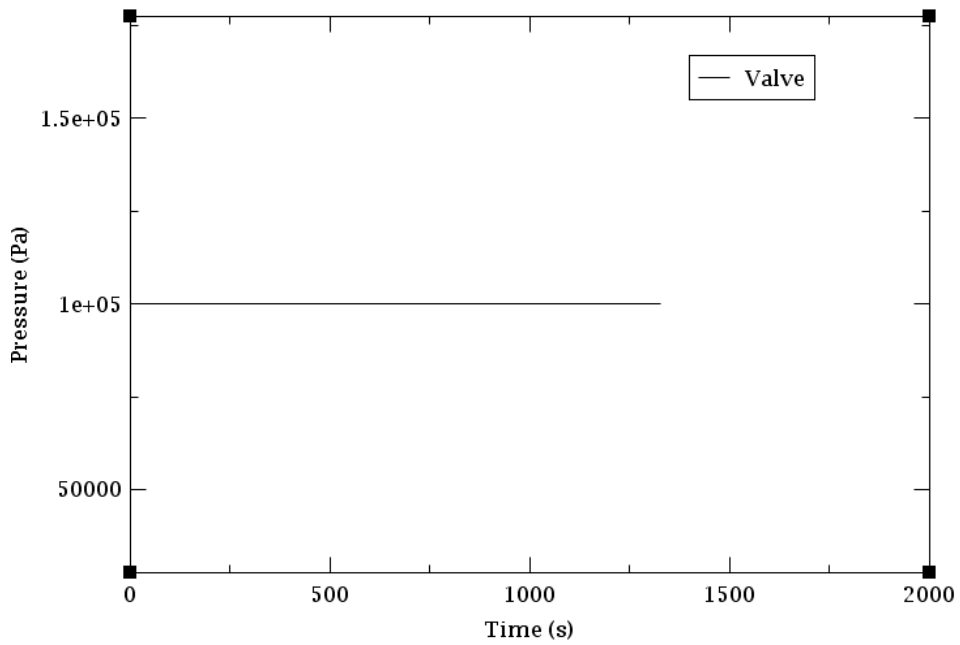


Figure 3.28 : Pressure at the Valve Component.

4. CONCLUSION AND RECOMMENDATIONS

The major purpose of this research was to prove the safety of STURE – Swedish Research and Education Reactor. In order to make the LOCA analysis, a model of STURE is created by the assistance of TRACE code and SNAP interface application. After it was possible to make steady state calculations, the original model is modified to a scenario of a break in the reactor vessel junction due to an earthquake and LOCA case. Due to the break in the junction of the vessel, under a LOCA scenario the behavior of the STURE reactor is being analyzed in this thesis.

In steady state condition the reactor works in normal conditions and the temperatures, pressures and mass flows in the vessel, core and the break acts as normal. It is seen from the graphs that there is no mass flow through the vessel to the break, which means there is no leakage in the system. It acts like a pool of water cooled by natural circulation. The temperature in the reactor vessel is approximately 300 K and the pressure in the vessel is a little bit higher than the atmospheric pressure. It has been proven by the simplified thermal hydraulic calculations.

In transient condition due to an earthquake, a break occurs in the junction of the reactor vessel and LOCA case starts. In 1320 seconds all the water in the reactor pool leaks out from the reactor vessel. While LOCA case happens, the reactor continues its operation at full power. It can be seen from the graphs that the temperature of the core level is at a level of 373 K, which is under the melting temperature of the UO₂ fuel. That means STURE reactor is a safe reactor and under LOCA conditions even though all the water is leaked out of the reactor, the fuel in the reactor core does not melt.

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APPENDICES

APPENDIX A.1 : Matlab Script In Order to Calculate the Volumetric and Edge Data

APPENDIX A.1

Matlab Script In Order to Calculate the Volumetric and Edge Data

```
function[ReactorLength, Area1, Area2, Area3, Area4, Area5, Area6, Vol1, Vol2,  
Dh1, Dh2, Dh3, Dh4, Dh5, Dh6]=Sture(r, r1, r2, n, n_phi, layer1, layer2, layer3,  
layer4, layer5, layer6, layer7, layer8)
```

```
ReactorLength = layer1 + layer2 + layer3 + layer4 + layer5 + layer6 + layer7 +  
layer8;
```

```
Area1 = zeros (1,8);  
Area1(1,1) = 2*r2*pi*layer1/n_phi;  
Area1(1,2) = 2*r2*pi*layer2/n_phi;  
Area1(1,3) = 2*r2*pi*layer3/n_phi;  
Area1(1,4) = 2*r2*pi*layer4/n_phi;  
Area1(1,5) = 2*r2*pi*layer5/n_phi;  
Area1(1,6) = 2*r2*pi*layer6/n_phi;  
Area1(1,7) = 2*r2*pi*layer7/n_phi;  
Area1(1,8) = 2*r2*pi*layer8/n_phi;
```

```
Area2 = zeros (1,8);  
Area2(1,1) = (r2-r1)*layer1;  
Area2(1,2) = (r2-r1)*layer2;  
Area2(1,3) = (r2-r1)*layer3;  
Area2(1,4) = (r2-r1)*layer4;  
Area2(1,5) = (r2-r1)*layer5;  
Area2(1,6) = (r2-r1)*layer6;  
Area2(1,7) = (r2-r1)*layer7;  
Area2(1,8) = (r2-r1)*layer8;
```

```
Area3 = zeros (1,8);  
Area3(1,1) = (r2^2-r1^2)*pi/n_phi;  
Area3(1,2) = (r2^2-r1^2)*pi/n_phi;  
Area3(1,3) = (r2^2-r1^2)*pi/n_phi;  
Area3(1,4) = (r2^2-r1^2)*pi/n_phi;  
Area3(1,5) = (r2^2-r1^2)*pi/n_phi;  
Area3(1,6) = (r2^2-r1^2)*pi/n_phi;  
Area3(1,7) = (r2^2-r1^2)*pi/n_phi;  
Area3(1,8) = (r2^2-r1^2)*pi/n_phi;
```

```
Area4 = zeros (1,8);  
Area4(1,1) = 2*r1*pi*layer1/n_phi;  
Area4(1,2) = 2*r1*pi*layer2/n_phi;  
Area4(1,3) = 2*r1*pi*layer3/n_phi;  
Area4(1,4) = 2*r1*pi*layer4/n_phi;  
Area4(1,5) = 2*r1*pi*layer5/n_phi;  
Area4(1,6) = 2*r1*pi*layer6/n_phi;  
Area4(1,7) = 2*r1*pi*layer7/n_phi;  
Area4(1,8) = 2*r1*pi*layer8/n_phi;
```

```

Area5 = zeros (1,8);
Area5(1,1) = r1*layer1;
Area5(1,2) = r1*layer2;
Area5(1,3) = r1*layer3;
Area5(1,4) = r1*layer4;
Area5(1,5) = r1*layer5;
Area5(1,6) = r1*layer6;
Area5(1,7) = r1*layer7;
Area5(1,8) = r1*layer8;

```

```

Area6 = zeros (1,8);
Area6(1,1) = r1^2*pi/n_phi;
Area6(1,2) = r1^2*pi/n_phi;
Area6(1,3) = (r1^2*pi/n_phi)-(n*pi*r^2);
Area6(1,4) = (r1^2*pi/n_phi)-(n*pi*r^2);
Area6(1,5) = r1^2*pi/n_phi;
Area6(1,6) = r1^2*pi/n_phi;
Area6(1,7) = r1^2*pi/n_phi;
Area6(1,8) = r1^2*pi/n_phi;

```

```

Vol1 = zeros (1,8);
Vol1(1,1) = Area6(1,1)*layer1;
Vol1(1,2) = Area6(1,2)*layer2;
Vol1(1,3) = Area6(1,3)*layer3;
Vol1(1,4) = Area6(1,4)*layer4;
Vol1(1,5) = Area6(1,5)*layer5;
Vol1(1,6) = Area6(1,6)*layer6;
Vol1(1,7) = Area6(1,7)*layer7;
Vol1(1,8) = Area6(1,8)*layer8;

```

```

Vol2 = zeros (1,8);
Vol2(1,1) = Area3(1,1)*layer1;
Vol2(1,2) = Area3(1,2)*layer2;
Vol2(1,3) = Area3(1,3)*layer3;
Vol2(1,4) = Area3(1,4)*layer4;
Vol2(1,5) = Area3(1,5)*layer5;
Vol2(1,6) = Area3(1,6)*layer6;
Vol2(1,7) = Area3(1,7)*layer7;
Vol2(1,8) = Area3(1,8)*layer8;

```

```

Dh1 = zeros (1,8);
Dh1(1,1)=4*Area1(1,1)/(4*r2*pi/n_phi+2*layer1);
Dh1(1,2)=4*Area1(1,2)/(4*r2*pi/n_phi+2*layer2);
Dh1(1,3)=4*Area1(1,3)/(4*r2*pi/n_phi+2*layer3);
Dh1(1,4)=4*Area1(1,4)/(4*r2*pi/n_phi+2*layer4);
Dh1(1,5)=4*Area1(1,5)/(4*r2*pi/n_phi+2*layer5);
Dh1(1,6)=4*Area1(1,6)/(4*r2*pi/n_phi+2*layer6);
Dh1(1,7)=4*Area1(1,7)/(4*r2*pi/n_phi+2*layer7);
Dh1(1,8)=4*Area1(1,8)/(4*r2*pi/n_phi+2*layer8);

```

```

Dh2 = zeros (1,8);
Dh2(1,1)=4*Area2(1,1)/(2*(r2-r1)+2*layer1);
Dh2(1,2)=4*Area2(1,2)/(2*(r2-r1)+2*layer2);
Dh2(1,3)=4*Area2(1,3)/(2*(r2-r1)+2*layer3);
Dh2(1,4)=4*Area2(1,4)/(2*(r2-r1)+2*layer4);
Dh2(1,5)=4*Area2(1,5)/(2*(r2-r1)+2*layer5);
Dh2(1,6)=4*Area2(1,6)/(2*(r2-r1)+2*layer6);
Dh2(1,7)=4*Area2(1,7)/(2*(r2-r1)+2*layer7);
Dh2(1,8)=4*Area2(1,8)/(2*(r2-r1)+2*layer8);

```

```

Dh3 = zeros (1,8);
Dh3(1,1)=4*Area3(1,1)/(2*(r2-r1)+(2*r2*pi/n_phi)+(2*r1*pi/n_phi));
Dh3(1,2)=4*Area3(1,2)/(2*(r2-r1)+(2*r2*pi/n_phi)+(2*r1*pi/n_phi));
Dh3(1,3)=4*Area3(1,3)/(2*(r2-r1)+(2*r2*pi/n_phi)+(2*r1*pi/n_phi));
Dh3(1,4)=4*Area3(1,4)/(2*(r2-r1)+(2*r2*pi/n_phi)+(2*r1*pi/n_phi));
Dh3(1,5)=4*Area3(1,5)/(2*(r2-r1)+(2*r2*pi/n_phi)+(2*r1*pi/n_phi));
Dh3(1,6)=4*Area3(1,6)/(2*(r2-r1)+(2*r2*pi/n_phi)+(2*r1*pi/n_phi));
Dh3(1,7)=4*Area3(1,7)/(2*(r2-r1)+(2*r2*pi/n_phi)+(2*r1*pi/n_phi));
Dh3(1,8)=4*Area3(1,8)/(2*(r2-r1)+(2*r2*pi/n_phi)+(2*r1*pi/n_phi));

```

```

Dh4 = zeros (1,8);
Dh4(1,1)=4*Area4(1,1)/(4*r1*pi/n_phi+2*layer1);
Dh4(1,2)=4*Area4(1,2)/(4*r1*pi/n_phi+2*layer2);
Dh4(1,3)=4*Area4(1,3)/(4*r1*pi/n_phi+2*layer3);
Dh4(1,4)=4*Area4(1,4)/(4*r1*pi/n_phi+2*layer4);
Dh4(1,5)=4*Area4(1,5)/(4*r1*pi/n_phi+2*layer5);
Dh4(1,6)=4*Area4(1,6)/(4*r1*pi/n_phi+2*layer6);
Dh4(1,7)=4*Area4(1,7)/(4*r1*pi/n_phi+2*layer7);
Dh4(1,8)=4*Area4(1,8)/(4*r1*pi/n_phi+2*layer8);

```

```

Dh5 = zeros (1,8);
Dh5(1,1)=4*Area5(1,1)/(2*r1+2*layer1);
Dh5(1,2)=4*Area5(1,2)/(2*r1+2*layer2);
Dh5(1,3)=4*Area5(1,3)/(2*r1+2*layer3);
Dh5(1,4)=4*Area5(1,4)/(2*r1+2*layer4);
Dh5(1,5)=4*Area5(1,5)/(2*r1+2*layer5);
Dh5(1,6)=4*Area5(1,6)/(2*r1+2*layer6);
Dh5(1,7)=4*Area5(1,7)/(2*r1+2*layer7);
Dh5(1,8)=4*Area5(1,8)/(2*r1+2*layer8);

```

```
Dh6 = zeros (1,8);
Dh6(1,1)=4*Area6(1,1)/(2*r1*pi/n_phi);
Dh6(1,2)=4*Area6(1,2)/(2*r1*pi/n_phi);
Dh6(1,3)=4*Area6(1,3)/((2*r1*pi/n_phi)+(n*pi*2*r));
Dh6(1,4)=4*Area6(1,3)/((2*r1*pi/n_phi)+(n*pi*2*r));
Dh6(1,5)=4*Area6(1,5)/(2*r1*pi/n_phi);
Dh6(1,6)=4*Area6(1,6)/(2*r1*pi/n_phi);
Dh6(1,7)=4*Area6(1,7)/(2*r1*pi/n_phi);
Dh6(1,8)=4*Area6(1,8)/(2*r1*pi/n_phi);
```


CURRICULUM VITAE



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