ISTANBUL TECHNICAL UNIVERSITY ★ ENERGY INSTITUTE

ANALYSES OF CONTROL ROD WORTH AND REACTIVITY INITIATED ACCIDENT (RIA) OF ITU TRIGA MARK II RESEARCH REACTOR

M.Sc. THESIS

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Nuclear Researches Division

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<u>İSTANBUL TEKNİK ÜNİVERSİTESİ ★ ENERJİ ENSTİTÜSÜ</u>

İTÜ TRIGA MARK II ARAŞTIRMA REAKTÖRÜNÜN KONTROL ÇUBUĞU DEĞERİ VE REAKTİVİTE NEDENLİ KAZA ANALİZİ

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Fadime Özge ÖZKAN, a M.Sc. student of ITU Institute of Energy student ID 302161016, successfully defended the thesis entitled "ANALYSES OF CONTROL ROD WORTH AND REACTIVITY INITIATED ACCIDENT (RIA) OF ITU TRIGA MARK II RESEARCH REACTOR", which she prepared after fulfilling the requirements specified in the associated legislations, before the jury whose signatures are below.

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Date of Submission : 15 November 2019
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To my decedent grandfather Fazlı ÖZKAN and my family,

FOREWORD

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December 2019

Fadime Özge ÖZKAN
(Nuclear Engineer)

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ABBREVIATIONS

CRW	**:** Control Rod Worth
DNBR	**:** Departure from Nucleate Boiling Ratio
FSAR	**:** Final Safety Analysis Report
HEU	**:** Highly Enriched Uranium
IAEA	**:** International Atomic Energy Agency
LEU	**:** Low Enriched Uranium
LOCA	**:** Loss of Coolant Accident
LOFA	**:** Loss of Flow Accident
MCNP	**:** Monte Carlo N-Particle Transport Code
MNSR	**:** Miniature Neutron Source Reactor
MTR	**:** Material Testing Reactor
NEA	**:** Nuclear Energy Agency
PARET	**:** Program for the Analysis of Reactor Transients
PRIDE	**:** Program for Reactor In-core Analysis using Diffusion Equation
PWR	**:** Pressurized Water Reactor
RELAP	**:** Reactor Excursion and Leak Analysis Program
RIA	**:** Reactivity Initiated Accident
RSICC	**:** Radiation Safety Information Computational Center
SPERT	**:** Special Power Excursion Reactor Test
TRIGA	**:** Training, Research, Isotopes, General Atomics
VVER	**:** Water-Water Energetic Reactor
WIMS	**:** Winfrith Improved Multigroup Scheme

NOMENCLATURE

$\boldsymbol{\rho}$	: reactivity ($)
$\boldsymbol{k}$	: multiplication factor
$\boldsymbol{t}$	: time (s)
$\boldsymbol{n}$	: neutron density (neutrons/cm^3)
$\boldsymbol{\beta}$	: effective delayed neutron fraction
$\boldsymbol{\Lambda}$	: prompt neutron generation time (s)
$\boldsymbol{\lambda_i}$	: decay constant of group i (s^{-1})
$\boldsymbol{C_i}$	: concentration of delayed neutron precursors of group i
$\boldsymbol{\beta_i}$	: delayed neutron fraction for group i
$\boldsymbol{G}$	: mass flux (kg/m^{2}s)
$\boldsymbol{P}$	: pressure (Pa)
$\boldsymbol{g}$	: gravitational acceleration (m/s^2)
$\boldsymbol{H}$	: specific enthalpy (J/kg)
$\boldsymbol{f}$	: friction factor
$\boldsymbol{D_h}$	: hydraulic diameter of the coolant channel (m)
$\boldsymbol{D_i}$	: diameter of the heat source (m)
$\boldsymbol{\bar{\rho}}$	: average density of two phases (kg/m^3)
$\boldsymbol{T}$	: reactor period (s)
$\boldsymbol{\alpha_f}$	: temperature reactivity coefficient of fuel ($/°C)
$\boldsymbol{c_p}$	: specific heat capacity of the fuel (J/kg°C)
$\boldsymbol{l}$	: prompt neutron lifetime (s)
$\boldsymbol{h_{sp}}$	: heat transfer coefficient for single phase (W/m^{2}K)
$\boldsymbol{h_{crit}}$	: critical heat transfer coefficient (W/m^{2}K)
$\boldsymbol{G}$	: mass flux (kg/m^{2}s)
$\boldsymbol{\mu}$	: dynamic viscosity of the fluid (kg/ms)
$\boldsymbol{q_{fd}}$	: heat flux for subcooled nucleate boiling region (W/m^2)
$\boldsymbol{T_{crit}}$	: critical clad surface temperature (°C)
$\boldsymbol{T_f}$	: bulk temperature (°C)
$\boldsymbol{u}$	: fluid velocity (m/s)

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ANALYSES OF CONTROL ROD WORTH AND REACTIVITY INITIATED ACCIDENT (RIA) OF ITU TRIGA MARK II RESEARCH REACTOR

SUMMARY

The control rod worth and RIA pulse analyses of ITU TRIGA Mark II research reactor have been done in the scope of this thesis study. ITU TRIGA Mark II is a research reactor in ITU Energy Institute and it reached the first criticality in 1979. Control rod worth is a very important concept in terms of safety of a nuclear reactor. Control rods are basically used to make changes on the power level of the reactor. Therefore, the accuracy in calculating the control rod worth value is crucial. Having a reliable computational model rather than doing experiments is a great advantage in terms of time efficiency and being able to analyze different cases without being dependent on experimental procedures. Analyses on research reactors provide opportunity to validate numerical models using the experimental data done on research reactors. One of the aims of this thesis is to estimate control rod worth values numerically and compare the results with experimental ones. ITU TRIGA Mark II research reactor has three control rods; transient, safety, and regulating rods. Transient rod is used mainly for pulse transients and regulating and safety rods are used to safely change the reactor power with transient rod. The research reactor is shut down with the control rods when it is necessary. Control rod worth is expressed in two ways; integral and differential rod worth. Integral rod worth curves are obtained using positive period method experimentally. The differential rod worth means the reactivity change per unit movement of the control rod, so the differential rod worth curve is obtained using the slope of integral rod worth curve. 3D full core MCNP model ,which is generated at the Energy Institute of ITU by Dr. Lecturer Senem Şentürk Lüle (thanks to Dr. Türkmen and Mr. Allaf for their contributions) for various calculations on ITU TRIGA Mark II research reactor, is used to create integral and differential rod worth values numerically and to compare the numerical results with experimental data. Rod insertion method is used for numerical analysis rather than positive period method as in experiment. Rod insertion method is applied for the concerned rod when the other control rods are fully withdrawn. This method is applied in two ways as source recorded and no source recorded rod insertion methods for this thesis study. The results of these two methods are so close to each other, but the shapes of integral rod worth curves are little bit closer to the experimental ones for source recorded case than no source recorded case. The reason for this is that the source output of previous step is used for source recorded analysis rather than giving an initial source for criticality calculations for MCNP. Relative error between total control rod worth values of numerical and experimental methods is less than 5% which is very low. Control rod worth analysis is carried out for fresh fuel configuration of the reactor core based on experimental data. In addition, it is observed that excess reactivity can be compensated by the control rods since it is lower than the total worth of all control rods in the reactor.

RIA (Reactivity Initiated Accident) analysis is significant for the safety of nuclear reactors. It creates changes in fission rate, so in power in the reactor. Analyzing power and temperature values after RIA provides to see if safety limits for peak values are exceeded or not. PARET/ANL code couples the thermal hydraulic and point kinetics equations and it is used for transient analysis of research reactors. RIA pulse analysis has been done for \$1.5, \$1.81 and \$2 reactivity insertions, based on experimental data, using PARET/ANL code for ITU TRIGA Mark II research reactor. Initial power is 50 W for \$1.5 and \$1.81 reactivity insertions and 200 W for \$2 reactivity insertion. The power, fuel centerline temperature, clad surface temperature and coolant temperature versus time behaviors are analyzed after aforementioned pulse scenarios. The peak power limit for ITU TRIGA Mark II research reactor is 1200 MW for pulse. In addition, safety limits for fuel and clad temperatures are 1150°C and 760°C, respectively. It has been seen that the peak power values for \$1.5 and \$1.81 pulse reactivity insertions are so close to the experimental values. The power is over predicted by almost 11% for \$2 pulse reactivity insertion. However, peak power values after all pulse scenarios are over predicted by PARET/ANL code which is good in terms of safety. This shows that PARET/ANL is a conservative code for the pulse analyses. If the peak temperature values do not exceed the safety limits after pulse for PARET/ANL analysis, they will not exceed safety limits in real pulse situation anyway. The peak power values are 69 MW, 180 MW and 275 MW for \$1.5, \$1.81 and \$2 pulse reactivity insertions respectively according to PARET/ANL analysis. The peak fuel centerline temperature values are 290°C, 332°C and 376°C respectively. In addition, peak clad surface temperature values are 102°C for \$1.5 reactivity insertion, 125°C for \$1.81 and \$2 reactivity insertions. It can be seen from the results that peak power and peak temperature values are in safety limits. The reactivity insertions higher than \$2 could not be modeled using PARET/ANL, the code developers recommend that PARET/ANL should not be used in case of high reactivity insertions for natural convection models. Because of void formation in the core for high reactivity insertions, the code cannot simulate the whole transient time. Finally, all analyses for control rod worth and RIA show that the reactor will continue operating safely in case of aforementioned reactivity insertions and the worth of control rods are enough to compensate the excess reactivity and carry out the shutdown of reactor when it is needed.

İTÜ TRIGA MARK II ARAŞTIRMA REAKTÖRÜNÜN KONTROL ÇUBUĞU DEĞERİ VE REAKTİVİTE NEDENLİ KAZA ANALİZİ

ÖZET

Bu tez çalışması kapsamında İTÜ TRIGA Mark II eğitim ve araştırma reaktörü için kontrol çubuğu değeri ve reaktivite nedenli kaza analizi (RIA) yapılmıştır. İTÜ TRIGA Mark II reaktörü İTÜ Enerji Enstitüsü'nde yer almakla beraber ilk kritikliğine 1979 yılında ulaşan bir araştırma reaktörüdür. Reaktör güvenliği konsepti açısından kontrol çubuğu değeri çok önemlidir. Kısaca, kontrol çubukları reaktörün güç seviyesini istenen seviyede tutmak için kullanılır. Bu nedenle kontrol çubuğu değerinin doğru bir şekilde hesaplanması büyük önem taşımaktadır. Teknolojinin de gelişmesiyle hesaplamalı yöntemler ile yapılan analizlerin önemi günümüzde her alanda olduğu gibi nükleer alanda da geçmişe nazaran artmıştır. Her analiz için deneysel yöntemleri kullanmaktansa güvenilir bir hesaplamalı yöntem kullanmak çok daha avantajlı hale gelmiştir. Deneysel prosedürlere bağlı kalmadan, güvenilirliği sağlanmış hesaplamalı bir yöntemle farklı senaryolar için analiz yapabilmek zamanı verimli kullanmak açısından büyük fayda sağlamaktadır. Araştırma reaktörlerinde yapılan deneyler, geliştirilen nümerik metodların doğruluğunu test etme olanağı sağlamaktadır. Bu tez çalışmasının amaçlarından biri de kontrol çubuğu değerlerinin nümerik analiz sonuçlarını deneysel sonuçlarla karşılaştırıp nümerik modelin doğruluğunu değerlendirmektir. İTÜ TRIGA Mark II araştırma reaktöründe darbe, güvenlik ve ayar çubukları olmak üzere üç adet kontrol çubuğu bulunmaktadır. Darbe çubuğu asıl olarak geçici darbe durumunda, genel olarak ise tüm kontrol çubukları reaktördeki güç seviyesi değişimlerinin güvenli bir şekilde gerçekleştirilmesinde kullanılmaktadır. Ayrıca, gerekli durumlarda reaktörün çalışmasının durdurulmasında da kontrol çubukları kullanılmaktadır. Deneysel olarak integral çubuk değeri grafiklerini elde etmede pozitif periyot metodu kullanılmıştır. Diferansiyel çubuk değeri kontrol çubuğunun birim hareketi sonucu oluşan reaktivite değişimini ifade etmektedir. Buna istinaden, kontrol çubuğunun toplam değerini ifade eden integral çubuk değeri grafiğinin eğimi ile diferansiyel çubuk değeri grafiği elde edilmektedir. Integral ve diferansiyel değer grafiklerini elde etmek için 3B tam kalp MCNP modeli kullanılmıştır ve bu model ile nümerik olarak elde edilen sonuçlar deneysel sonuçlarla karşılaştırılmıştır. 3B MCNP modeli Dr. Senem Şentürk Lüle tarafından (Dr. Mehmet Türkmen'in ve Mohammad Allaf'ın katkılarıyla) İTÜ TRIGA Mark II eğitim ve araştırma reaktörü için çeşitli analizlerde kullanılmak üzere İTÜ Enerji Enstitüsünde üretilmiştir. İTÜ Nümerik analizler için deneysel yöntem olan pozitif periyot metodun aksine çubuk ekleme metodu kullanılmıştır. Diğer kontrol çubukları reaktörün dışındayken, analizi yapılacak olan kontrol çubuğu belirlenen adımlarla reaktörün içine sokulur ve her adımda elde edilen reaktivite değişimi ile integral ve diferansiyel çubuk değeri elde edilir.

Bu metot nümerik olarak tez çalışmasında iki şekilde uygulanmıştır; MCNP modelinde, adımlar arasında nötron kaynağı kaydederek ve bir de nötron kaynağı kaydedilmeden yapılan çubuk ekleme analizi. Normalde MCNP'de KSRC kartı ile bir başlangıç nötron kaynağı atanarak yapılan kritiklik hesaplama analizi, bir de KSRC kartı kullanılmadan, bir önceki adım sonucu çıkan nötron kaynağı gelecek adımın kritiklik hesabına eklenerek yapılmıştır. Bu iki yöntemin sonuçları birbirine çok yakın çıkmıştır, bunun yanında nötron kaynağı kaydedilerek yapılan analiz sonucu elde edilen integral çubuk değeri grafiği deneysel grafiğe şekil olarak biraz daha yakın davranış göstermiştir. Bunun nedeni, başlangıç nötron kaynağı olarak KSRC kartı kullanmak yerine bir önceki adım sonucu elde edilen nötronları sonraki adımın nötron kaynağı olarak kullanarak kritiklik hesapları için MCNP ile daha gerçekçi bir yaklaşım yapılmasıdır. Toplam çubuk değerleri için nümerik ve deneysel sonuçlar arasındaki bağıl hata %5'ten az çıkmaktadır. Bu da nümerik modelin yeterince doğru olduğunu ve ileriki analizlerde kullanılabileceğini göstermektedir. Deneysel verilere dayanarak kontrol çubuğu değeri analizleri reaktör kalbinin ilk konfigürasyonuna göre yapılmıştır. Bununla birlikte, MCNP ile kritiklik analizi yapılarak tüm kontrol çubukları reaktörün dışındayken fazlalık reaktivite hesabı yapılmıştır. Fazlalık reaktivite değeri darbe çubuğunun değerinden daha az, dolayısıyla tüm kontrol tubuklarının toplam değerinden de daha az çıkmıştır. Bu da gösteriyor ki kontrol çubukları fazlalık reaktiviteyi dengeleyebilecek durumdadır. Bu durumun sağlanması, güvenlik açısından reaktör için önemlidir.

RIA analizi nükleer reaktörlerin güvenliği açısından önemli konseptlerden biridir. RIA durumunda reaktörde fisyon oranı değişmektedir, dolayısıyla güç de değişmektedir. Bu nedenle RIA sonucu güç ve sıcaklık değişimlerini incelemek reaktör için güvenlik limitlerinin aşılıp aşılmadığını kontrol etmek için gereklidir. RIA darbe analizleri \$1.5, \$1.81 ve \$2'lık reaktivite girişleri ile İTÜ TRIGA Mark II araştırma reaktörü için PARET/ANL kodu kullanılarak yapılmıştır. PARET/ANL kodu nokta kinetik denklemleri ile termal hidrolik denklemleri kuplaj ederek analiz yapmaktadır ve araştırma reaktörleri için süreksiz durum analizlerinde kullanılmaktadır. Deneysel verilere istinaden \$1.5 ve \$1.81'lık reaktivite girişleri için başlangıç gücü 50 W, \$2'lık reaktivite girişi için ise 200 W olarak alınmıştır. Reaktör gücünün, yakıt merkez sıcaklığının, zarf yüzeyi sıcaklığının ve soğutucu sıcaklığının darbe sonrası zamana göre değişimi gözlemlenmiştir. İTÜ TRIGA Mark II araştırma reaktörü için darbe durumunda maksimum gücün güvenlik limiti 1200 MW'dır. Bununla birlikte yakıt ve zarf sıcaklıkları için güvenlik limitleri sırasıyla 1150°C ve 760°C'dir.

PARET/ANL analizi sonuçlarına göre \$1.5 ve \$1.81'lık darbe reaktivite girişleri sonucu elde edilen maksimum güç deneysel verilere çok yakındır. \$2'lık darbe reaktivite girişi için ise maksimum güç deneysel verilere göre %11 farkla daha fazla çıkmıştır. Sonuç olarak, PARET/ANL analizleri sonucu elde edilen maksimum güç değerleri deneysel verilere nazaran daha fazladır. Bu da güvenlik açısından olumlu birşeydir, çünkü PARET/ANL sonucu yapılan güç ve sıcaklık analizlerinde eğer güvenlik limitleri aşılmamışsa gerçek darbe durumda zaten aşılmayacaktır. PARET/ANL sonuçlarına göre \$1.5, \$1.81 ve \$2'lık darbe reaktivite girişleri için maksimum güç değerleri sırasıyla 69 MW, 180 MW ve 275 MW olarak elde edilmiştir. Maksimum yakıt merkez sıcaklıkları ise bahsedilen darbe reaktivite girişleri için sırasıyla, yaklaşık 290°C, 332°C ve 376°C olarak elde edilmiştir.

Bununla birlikte \$1.5'lık darbe reaktivite girişi için maksimum zarf yüzeyi sıcaklığı 102°C, \$1.81 ve \$2'lık darbe reaktivite girişleri için ise 125°C olarak elde edilmiştir. Elde edilen maksimum güç ve sıcaklık değerleri gösteriyor ki reaktör için güvenlik limitleri aşılmamaktadır. Ayrıca, \$2'lık darbeden daha fazla olan reaktivite girişleri için PARET/ANL ile verilen tüm zaman boyunca analiz yapılamamıştır. PARET/ANL kodu geliştiricilerine göre doğal taşınım modelleri için yüksek reaktivite girişi analizi yapılmaması önerilmektedir. Çünkü yüksek reaktivite girişi esnasında oluşan boşluk formlarından dolayı PARET/ANL istenilen tüm zaman dilimi boyunca geçici durum analizi yapamamaktadır. Sonuç olarak, yapılan kontrol çubuğu değeri ve RIA analizlerine göre bahsedilen reaktivite girişleri esnasında reaktör güvenli bir şekilde çalışmaya devam edecektir, kontrol çubuğu değerleri fazlalık reaktiviteyi dengelemede ve gerektiğinde reaktörün çalışmasını durdurmada yeterlidir.

1. INTRODUCTION

Two analyses are carried out in the scope of this thesis; the control rod worth analysis using a Monte Carlo radiation transport code and the RIA analysis using PARET/ANL of ITU TRIGA Mark II research reactor. The scope of the analyses is given in this section. The studies, related to aforementioned analyses, done in past are given under the section of Literature Review. In addition, the theory and methodology of the analyses are explained in the following sections of Control Rod Worth Analysis and PARET/ANL Analysis. Finally, the results are given and discussions are made in Section 5 and Section 6.

Control rods are mainly used to keep the power at required level in nuclear reactors. As it can be understood from their function, control rods play a very important role in terms of safety. Therefore, one of the most important issues is control rod worth analysis for safe operation of a nuclear reactor. The accuracy in the measurement of control rod worth is crucial. Control rod worth analysis provides the information about safety margin of a nuclear reactor. In addition, control rod worth analysis is very important in terms of controlling the reactivity of a nuclear reactor. There are two kinds of control rod worth; integral control rod worth and differential control rod worth. Integral control rod worth refers to the total reactivity worth of the control rod at the point of withdrawal. Differential control rod worth refers to the reactivity change per unit movement; the unit of differential control rod worth is generally expressed with ρ/in $(or\ pcm/in)$ and measured by calculating $\Delta k/k$ per inch [1]. In this study, the integral and differential control rod worth of ITU TRIGA Mark II research reactor is aimed to be analyzed. A Monte Carlo 3D Model of ITU TRIGA Mark II Research Reactor created with MCNP code is used for the analyses. One of the aims of the study is to compare computational simulation (by using 3D Monte Carlo model) results and experimental results for control rod worth calculations. There are multiple studies done in past on the analysis of control rod worth for research and power reactors both experimentally and numerically.

Since computational analysis has taken an important place in any engineering field nowadays for many advantages like saving time, it is important to validate numerical results with experimental ones to examine computational capacity.

Thermal hydraulic safety analysis is always important for nuclear industry. Nuclear power plants have a significant place in our current world in terms of producing electricity. Moreover, nuclear research practices are carried out widely other than producing electricity using nuclear energy. Therefore, carrying out safety analyses is crucial and necessary for the nuclear systems. These safety analyses are done to investigate accidents like LOFA, LOCA, RIA and others. Nowadays, as technology develops, the analyses mostly are carried out by computer codes. Using validated computer codes saves time and provide the opportunity to make more analyses with different scenarios than experiments. Research reactors play a very important role in validating this kind of codes. The data taken from the experiments made on research reactors can be used to validate developed computer codes.

PARET/ANL code is used for this thesis study to make $1.5, $1.81 and $2 pulse reactivity insertion analyses based on experimental data. The original PARET code was developed for transient and thermal hydraulic analysis for research and test reactors. PARET is a computer code that has the capability of coupling thermal hydraulics , and point kinetics equations [2]. The original version of the code was adjusted by Reduced Enrichment Research and Test Reactor (RERTR) Program. It is applicable both for plate and cylindrical fuel type reactors. PARET/ANL 2001 revised version of the code is used for the pulse analysis of ITU TRIGA Mark II research reactor in the scope of this thesis. It has been observed in the past that this code shows good agreement for pulse analysis of TRIGA reactors in pin geometry. The comparisons for PARET/ANL code are made with SPERT experiments that will be explained in detail later [3].

2. LITERATURE REVIEW

In this section, several important control rod worth analysis and PARET/ANL analysis studies done in the literature are given. Firstly, some of the significant studies in the literature are explained for control rod worth and PARET/ANL analyses. After that, the history of TRIGA reactors is given and the introduction of ITU TRIGA Mark II Research Reactor is made. In addition, the studies done on aforementioned analyses in the literature for TRIGA Mark II reactors are given in this section.

2.1 Control Rod Worth Analysis

The computer simulation of VVER1000 reactor is done using WIMS and CITATION codes by Fadaei and Setayeshi. Both research and power reactors can be modeled by WIMS code, which solves the transport equation. CITATION code solves the diffusion equation for the core. In this study, neutronic calculation for the core of VVER1000 is done by the mentioned two codes and control rod worth of the reactor is estimated. As it is indicated in this study, the worth of an absorber rod is proportional to the neutron flux of the core before the rod inserted. After required input such as geometry, material, burnup and buckling values are entered into WIMS code, macroscopic cross sections of fuel assemblies are obtained and used in CITATION code for next step. This code solves diffusion theory by finite-difference numerical method.

For calculation of the rod worth, control rod is inserted into the core step by step and then reactivity change is investigated by using CITATION code. Since CITATION does neutronic calculations in steady-state, a FORTRAN script was written for dynamic coupling of WIMS and CITATION codes. The authors observed that the results of simulation for control rod worth are similar with the ones in FSAR of the reactor [4].

The impact of background noise to the measured reactivity is analyzed for a PWR by Huo et al. from China Institute of Atomic Energy. Because of the gamma radiation, electronic devices in the reactor are affected and this effect changes the measured values for reactivity. In addition, a method is proposed for control rod worth calibration in an iterative way in the scope of this thesis. The results of the calculations are compared with the theoretical values and the results of other methods. Finally, it is seen that they are all in good agreement [5].

Monte Carlo technique and perturbation theory are combined to have a better approximation for experimental method to estimate control rod worth of BR2 (Belgian Reactor 2) reactor by Kalcheva and Koonen from SCK-CEN, BR2 Reactor Department, Belgium. Perturbation method is applied to provide the equation for relative efficiency of control rod insertion. In the scope of this study, a series of coefficients that represent the axial absorption profile are used. These coefficients are needed to be determined to adapt the equation for a composite rod. In this study, concerned coefficients are obtained from macroscopic absorption cross-sections. And these macroscopic cross sections are obtained using MCNPX 2.6.F code. The integral control rod worth values are obtained using MCNP model for fresh and depleted absorbing materials. These control rod values are investigated both for cadmium and hafnium rods. As a result of this study, it is observed that the accuracy of the hybrid Monte Carlo-perturbation method is higher when it is compared to mixed experimental-perturbation method. Because the axial absorption profile in any composite rod with a complex burn-up history can be obtained by hybrid Monte Carlo-perturbation method rather than mixed experimental-perturbation method [6].

Two methods, which are deterministic and stochastic, are used to calculate the integral and differential control rod worth of Greek Research Reactor (GRR-1) theoretically by Varvayanni et al. from NCSR “DEMOKRITOS”, Greece. The methods are applied for two control rods. SCALE and CITATION, which are neutronic codes, are used for deterministic; TRIPOLI which is a Monte Carlo code is used for stochastic analysis. Both deterministic and stochastic approaches are used for rod insertion method to estimate rod worth and the results are compared with the experimental ones. It was seen that the Monte Carlo calculations are in good agreement with the experimental results.

In addition, it was observed that deterministic approach is affected by the presence of other control rods. Therefore, deterministic analysis is repeated for the integral rod worth with the rod movement away from other rods. In this study, it is stated that, since the integral rod worth is not a value that is calculated directly, some error was expected following experimental results. These errors may be caused by instrument accuracy and the way of using in-hour equation for the experiments. On the other hand, while Monte Carlo analysis took several weeks, analysis using deterministic approach took only a few hours. Finally, in terms of time efficiency, deterministic approach is much better if the proper modeling is used for increasing the accuracy of the results [7].

Monte Carlo code is used for control rod worth estimation of two research reactors; the IRT MEPhI reactor at National Research Nuclear University, MEPhI and the IRT-T reactor at the National Research Tomsk Polytechnic University in Russia by M.V. Shchurovskaya et al. Calculations are carried out using the Monte Carlo code MCU-PTR. 3D models of the aforementioned reactors are created using the code. Calculation results of the code are compared with the experimental results for the control rod worth. For the reactor in MEPhI, the discrepancy is observed to be less than 10% between the measured and calculated integral rod worth for the shim rod. In addition, the discrepancy is observed to be less than 15% for the regulating rod. For IRT-T, the discrepancy is observed to be less than 10% only for two of the three shim rod groups between the measured and calculated integral rod worth. As a conclusion, they came up with that the results are in good agreement for IRT MEPhI but further work is required for IRT-T reactor [8].

The worth of depleted control rod is estimated by M. Varvayanni et al. Identifying the position of depleted control rod for new core configuration is very important in terms of safety concerns. In this study, a methodology is proposed to estimate absorbing capacity along a depleted control rod. This method is based on previous control rod worth measurements carried out for the former core configuration. The calculations are carried out depending on the absorber concentration before the control rod is irradiated.

Multi-group theory is used for formulation of the methodology. The formulation is tested for one-group approximation, for the depleted control rod of Greek Research Reactor, and it was seen that the results are in good agreement. In addition, the total macroscopic cross section of the control rod is calculated using the proposed methodology [9].

2.2 PARET/ANL Analysis

The PARET code couples the thermal hydraulics and point kinetics equations and is used for transient analysis of research reactors. It is originally developed in Idaho National Laboratory in 1960s. Since 1980s, PARET is maintained and developed in Argonne National Laboratory, that is why it is named as PARET/ANL since then [10]. SPERT experiments are used to adapt PARET/ANL code as it is mentioned before in this chapter. The code is also used for benchmarking analysis of the IAEA 10 MW research reactor for several transients.

The original PARET code was developed for SPERT III experiments at Idaho National Engineering Laboratory. SPERT III E-Core experiments are explained in detail in the report by Arne P. Olson. from ANL (Argonne National Laboratory) [11]. SPERT III reactor core has 60 assemblies. There are 8 control rods of the reactor. Control rods are located as pairs in the core and one is in the center as transient rod. The fuel is made of uranium oxide composition and the clad material is made of stainless steel. Experiments are made in two phases as high initial power and low initial power phases. In addition, these two phases are investigated under hot standby and operating power conditions. Initial reactor power phases are analyzed under 50 W reactor power with 21°C coolant inlet temperature for cold start-up and 127°C, 260°C coolant inlet temperatures for hot start-up conditions. Hot standby power is taken as 1 MW and operational power is taken as 20 MW. Initial test conditions are taken as commercial PWR operating conditions in terms of specific core power, coolant pressure, velocity, and subcooling except for cold start-up conditions. Cold start-up test was under atmospheric pressure and no flow condition. MCNP5 code is used to obtain reactivity, kinetic parameters of delayed neutrons, and power shape by static calculation.

They observed that the temperature feedback effect of fuel heat up (Doppler Effect) was the most important phenomenon that controls the progress of each transient case. The predicted peak power value by PARET was higher than expected for the operational power case, but the trend of power versus reactivity insertion was observed as expected from PARET results.

Operating power conditions for measurements were; $0.42 reactivity insertion with 1.2 seconds, $0.87 reactivity insertion with 1 second and $1.17 reactivity insertion with 0.5 seconds. Moreover, according to measurements peak power values are 39 MW, 130 MW and 610 MW respectively for each reactivity insertion cases.

The reactivity insertions were $0.9, $0.77, $1.13, $1.17, $1.21 for cold start-up conditions. For $1.21 reactivity insertion case, the measured power was 280 MW and this value is obtained as 278.7 MW from PARET which is very close to the result of measurement. The temperature feedback coefficient derived from MCNP code is not used in PARET as it is (- 0.003427 $/K), it was fitted to -0.0109 $/K. The power was obtained as 914 MW without this fitting, which is much higher than expected. Both the measured power value and the power from PARET are 280 MW after fitting the temperature feedback coefficient value.

As a result of this experiment, it is observed that PARET results are always conservative, the results show too high peak power. Besides this, the trend of power vs. reactivity insertion obtained from PARET results is as expected. It is also concluded that PARET and experiment results are similar for reactivity feedback at peak power. Moreover, especially for the cold start-up tests with no initial flow, the peak power is over-predicted by PARET. The reason is that axial power shape and peak power is very sensitive to the position of the control rod.

Comparison between another set of SPERT experiment results and computational results is also done. These experiments are made on SPERT IV reactor [12]. Experimental results are compared with RELAP5/MOD3 and PARET/ANL code results by W. L. Woodruff et al. from ANL. RELAP is another coupled code that thermal hydraulics safety analysis can be done with. The experiments are reactivity insertion transients. In this study, it is observed that RELAP5/MOD3 code diverges from experimental results when boiling occurs and also it shows similar results with PARET code for midrange transients.

It is understood from this study that RELAP5 code should be limited for no boiling cases when it is used for research reactors. The difference of RELAP/MOD3 between other RELAP codes is that it can be applicable for research reactors too with additional options in it. Original RELAP code was developed for pressurized water reactors with pin type fuel. Earlier studies was done with SPERT I core that has 25 fuel assemblies with 12 fuel plates in each assembly. SPERT IV core is similar to SPERT I core but SPERT IV core analysis is done with forced flow and there is 18-foot (548.64 cm) head of water over the core. SPERT I core analysis is done with no forced flow and 2-foot (60 cm) head of water over the core. Two channels were created for PARET/ANL; one channel was modelled as the hottest in the core, the rest of the core was represented as second channel.

There are 21 axial nodes in each channel. As a result, the comparison of PARET/ANL and RELAP5/MOD3 codes is made for the behavior of peak power, energy to the time of peak power, and temperature of the clad at the time of the peak power versus step reactivity insertion. It is known that $1.2 reactivity insertion is the threshold value for SPERT IV reactor for boiling. Therefore, RELAP5 results diverged significantly from experimental results for the higher reactivity insertion values than $1.2. In addition, it was seen that PARET/ANL and RELAP5 results were similar for lower reactivity insertion values but both codes still over predicted the results as compared to experimental ones. These mentioned results were for no forced flow. RELAP5 results again diverged from experimental ones for the forced flow case when more than $1.2 reactivity is inserted.

Another comparison between PARET/ANL and RELAP5/MOD3 codes is made by W. L. Woodruff et al. In this study, benchmark transient analyses for loss of flow and reactivity insertion cases with scram are done for IAEA research reactor. According to this study, the results of benchmark matched very well [13]. The original RELAP5 series was developed in Idaho National Engineering Laboratory like PARET code. The series were developed for US Nuclear Regulatory Commission for the safety analysis of pressurized water reactors. It is possible to model all of the components of the system with RELAP5/MOD3 rather than PARET.

In this study, to be able to make comparisons with PARET, the active core is represented by two channels; one of them represents the hottest channel and the other one represents the rest of the core. Scram scenario for each case is -$10 linear reactivity insertion in 0.5 seconds. Four different benchmark analyses are done for low enriched uranium fuel. These are fast and slow LOFA accidents and fast and slow reactivity insertion incidents. Fast LOFA transient is analyzed where the reactor scram is started at 85% of nominal flow and control insertion system begins with a 200 ms delay.

Slow LOFA has the same conditions with fast LOFA, but the flow begins to be reduced at T= 1 s for fast and T=25 s for slow LOFA. $0.09 / s ramp insertion case is analyzed for slow reactivity insertion transient and $1.5 ramp insertion in 0.5 seconds case is analyzed for fast reactivity insertion transient. The scram condition for reactivity insertion transients is 12 MW over power trip point with 25 ms delay of control blade system. In addition, initial power where the reactor is critical is 1 Watt.

As a result, it is observed that the peak temperatures of clad and fuel show a few degrees of difference between PARET and RELAP analyses results for fast LOFA transient. RELAP predictions are a little higher for these temperature values. Besides this, PARET predictions are a little higher for coolant temperature. For slow LOFA, similar behaviors are observed between two codes. The general results for fast and slow reactivity insertions are identical for both codes as LOFA transients. The largest difference between the results of two codes for slow reactivity insertion transient was observed for coolant temperature, but even that was less than a degree. In the case of fast reactivity insertion transient, the largest difference was seen for peak clad temperature. RELAP predicted the peak clad temperature 12°C higher than PARET for fast reactivity insertion transient.

Benchmark analysis for SPERT transients is done for heavy water application of the PARET code by W. L. Woodruff. The analysis is done for SPERT II transient series [14]. Subroutines that the library of the heavy water properties code uses are provided by KFA-Jülich which is one of the largest research center in Europe [15]. SPERT II core has 24 MTR type fuel elements which each has 22 fuel plates. Big differences between heavy water and light water analyses come from kinetic parameters which the prompt neutron generation time is the biggest.

The void coefficient of coolant for heavy water is about one-fifth of light water and the temperature coefficient for heavy water is about one-third of light water. The cause of all these differences is the large slowing down rate in the heavy water.

No forced coolant flow at ambient temperature condition is taken for the analyses. There are experimental data for the reactivity insertions between \$0.3 and \$3. Corresponding analyses are done with PARET code. According to the results, agreement is generally quite good for the peak power, peak clad temperature at the time of the peak power and energy released to the time of the peak power. The best agreement is observed for the peak clad temperature. The mentioned values for comparison are taken as the function of inverse initial period. It is also observed that the contribution of peak coolant temperature and the feedback is lower for higher reactivity insertion as expected. Because there will be less time to transfer the heat to the coolant for higher reactivity insertions.

Analytical verification of the safety analysis done for 10 MW IAEA research reactor can be seen in the 3rd volume of Research Reactor Core Conversion Guidebook of IAEA [16]. The guidebook was prepared to provide a guide for research reactor operators in safety and licensing for conversion of the reactor cores from highly enriched uranium fuel use to low enriched uranium fuel use. The appendices of the 3rd volume of the book includes the benchmark analysis and the comparisons of calculation with measurements. Safety related benchmark studies are done to calculate several safety parameters such as the control rod worth, power peaking factors, temperature and void coefficients, and kinetic parameters. These calculations are done for a typical light water, pool type reactor. In addition, the analyses on self-limiting transients for heavy water moderated reactors are performed. Transient analysis of the HEU and LEU benchmark cores is performed by using PARET code in the scope of this study. The most severe HEU case is defined as \$1.5 reactivity insertion in 0.5 seconds for this analysis. The analysis of PARET was done for different number of channels and it was seen that peak power and energy release values are almost identical for one, two and four channel analyses. This shows that radial dependence of the source and reactivity coefficient is not important and multi-channel analyses are not necessary to reach an accurate solution. Two-channel model is used for PARET analysis of the reactor. One of the channel represents the hottest channel and the second channel represents the rest of the core.

The channels are divided into 21 nodes for axial source distribution. 1.2 MW over-power trip is used for fast loss of flow transient for PARET analysis. According to the results, the peak for fuel and clad temperatures was seen after 1.4 seconds for both HEU and LEU cores. 89.2°C, 87.5°C, and 60.3°C are the peak temperatures for fuel centerline, clad surface and coolant exit respectively for fast loss of flow transient in HEU core. These values are 90.3°C, 87.5°C, and 60.3°C respectively for LEU core. When 15% of the nominal flow was reached, the transient was stopped. Since the conductivity of fuel for LEU core is lower, the fuel centerline temperature is higher than HEU core. In the case of slow loss of flow transient, the analysis was done in the same way with fast loss of flow transient but time constant was different. Peak fuel centerline, clad surface, and coolant exit temperatures are 85.8°C, 83.9°C, and 58.9°C respectively for the HEU core. These values are 86.8°C, 83.7°C, and 58.8°C, respectively for the LEU core.

Peak clad surface temperature for LEU core is higher than the HEU core for slow reactivity insertion. Peak clad surface temperatures are 77.7°C, and 69.0°C respectively for LEU core and HEU core. It was observed that these temperatures are well below the critical values and no boiling occurred. Peak power reaches up to 14.1 MW for HEU core and 12.4 MW for LEU core. $1.5 reactivity was inserted in 0.5 seconds for fast reactivity insertion case. It was seen that the HEU and LEU cores showed similar behavior. Peak fuel centerline temperature is about 13°C and peak clad surface temperature is about 1°C higher in LEU core than HEU core, because thermal conductivity in LEU core is lower comparing to the HEU core as mentioned before. In addition, the coolant exit temperature is 2°C lower for LEU core.

2.3 ITU TRIGA Mark II Research Reactor

TRIGA (Training, Research, Isotopes, General Atomics) research reactors were started to be installed in 1950s. US President Eisenhower made an announcement of “Atoms for Peace” at the United Nations and following this, TRIGA reactors were constructed from late 1950s to early 1980s by 23 countries. Many of these TRIGA reactors were operated for about 60 years.

TRIGA reactors are open pool, water moderated research reactors. General Atomics group aimed to have inherently safe characteristics for TRIGA research reactors. In other words, they aimed to design a reactor so that when all control rods are fully out rapidly, the fuel in the reactor core should keep its integrity without melting or any damage. It took 2 years to create a working reactor for the group. The prototype TRIGA Mark I was first critical in May 1958. Two more TRIGA reactors were commissioned in the same year. TRIGA Mark I and TRIGA Mark II designs were both offered to the customers.

TRIGA Mark I has an underground design. TRIGA Mark II is above ground and has beam ports additionally. First above ground designed TRIGA reactors were taken critical in ENEA Center, Rome and the University of Illinois. In Figure 2.1, the original 10 kW TRIGA Mark I core can be seen. Furthermore, General Atomics revealed two more TRIGA designs namely TRIGA Mark F and TRIGA Mark III. The Mark F had 1.5 MW steady state power level and it was operated from 1960 to 1993. TRIGA Mark III had 2 MW steady state power level and it was built to test the fuel for space power applications. The cores of both reactors had moveable design.

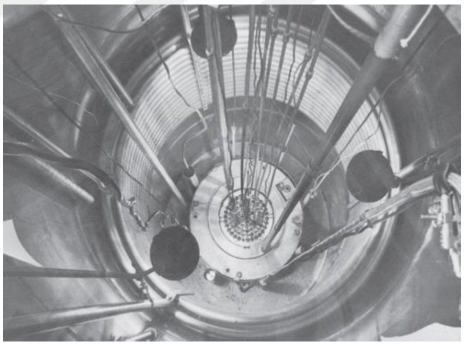

Figure 2.1 : Original TRIGA Mark I Core (10 kW) [17].

One of the benefits of this technology is that it can be used for educational, research, industrial and medical purposes widely since it is an inherently safe reactor. The fuel of the reactor is uranium-zirconium hydride (UZrH) which is the main contributor to keep reactor inherently safe in case of any reactivity insertion incident. Hydrogen in the fuel is the moderator material and zirconium is chosen as the absorber of hydrogen like “sponge” to create proper hydrogeneous medium.

In addition, further researches on TRIGA fuels were decided to be focused on ZrH because it does not require high critical mass, also it has low cost and very good thermal conductivity characteristics.

The primary feature of TRIGA reactors was to perform irradiation experiments with them. Therefore, General Atomics decided to carry this feature to its all TRIGA Mark II reactors and added beam ports to TRIGA Mark II reactor design. In Figure 2.2, installation of typical TRIGA Mark II reactor can be seen [17].

Figure 2.2 : Typical TRIGA Mark II Installation (Johannes Gutenberg University, Germany) [17].

ITU TRIGA Mark II research reactor is located in Energy Institute at İstanbul Technical University campus. The reactor was built for educational and research purposes. It was first critical in 1979. Since then, experiments on radiography, neutron activation and many other research activities are done more than 30 years with the research reactor. As it can be seen from Figure 2.3, reactor core has 5 outer rings around central thimble. There are total 91 places for elements in the core. Total 6 rings are named as A, B, C, D, E and F respectively from the inside out. The fuel rods are located in 69 of these 91 positions and the two of the fuel rods have thermocouples in them. There are 16 positions for graphite dummies and 3 positions for control rods. There are also one neutron source, one pneumatic system and one central thimble positions in the core. The reflector of the core is graphite and there are three beam ports, which are piercing, radial and tangential. ITU TRIGA Mark II research reactor operates under both steady state and pulse modes.

Steady state power of the reactor is 250 kW. Maximum power that the reactor can reach is limited to 1200 MW in case of the pulse. Reactor is brought to critical state at a power under 1 kW and then reactivity inserted to make a pulse. Cooling is provided by natural convection for the reactor. There is an external circuit to cool and purify the coolant water of the reactor.

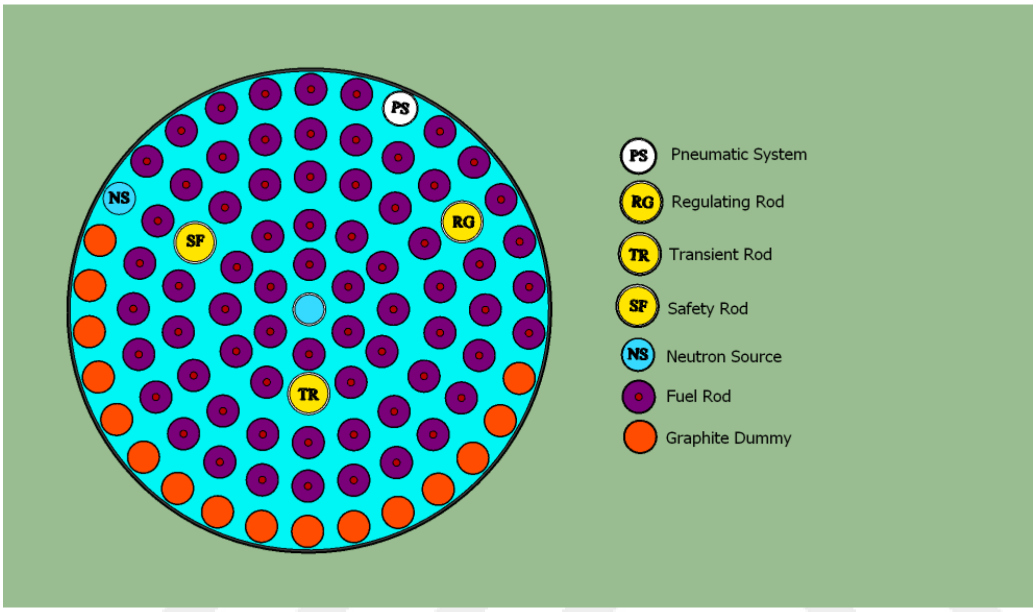

Figure 2.3 : Radial view of ITU TRIGA Mark II research reactor.

The fuel is a homogeneous composition of zirconium hydride ($ZrH_{1.6}$) with no more than 20% enriched uranium. The weight percent of the uranium in the fuel is 8.5%. The cladding material of the fuel is stainless steel (SS304). Zirconium rods are placed in the center of each fuel element. These zirconium rods increase the mechanical strength of the stainless steel clad and it has low neutron absorption cross-section. Boron carbide rods are used as control rods. There are three control rods in the core namely; transient rod, safety rod and regulating rod. Transient rod's role in the core is to carry out pulse transients. Regulating rod is used for fine control during the reactor operation. All control rods are used to safely change the reactor power by reactivity effect or shut down the reactor when it is necessary. There are some major concepts that are important to define TRIGA reactor system design bases; fuel temperature, control rod worth, pool water temperature, power of the reactor and prompt temperature coefficient [18].

The most dominant factor among these that defines safety limit is fuel temperature. Negative temperature feedback coefficient is the main phenomenon that makes TRIGA reactors inherently safe. Main characteristics of ITU TRIGA Mark II research reactor are used for the analyses in the scope of this thesis. Integral and differential control rod worth curves of ITU TRIGA Mark II research reactor are obtained with different methods using MCNP code and the main parameters of the reactor are observed for different reactivity insertion transients (pulse) using PARET/ANL code for this thesis study.

There are several studies done in literature for the control rod worth analysis of TRIGA Mark research reactors. Some of them are given below.

A Monte Carlo model of Moroccan 2 MW TRIGA Mark II research reactor is created by MCNP code. Integral rod worth of control rods is compared between experimental and simulation results by B. El Bakkari et al. from Reactor Operating Unit (UCR), National Centre of Sciences in Morocco [19].

In addition, control rod worth is calculated by positive period method experimentally. In simulation, effective multiplication factor was calculated when all control rods were arranged to make reactor critical. Then, one of the control rods were withdrawn step by step and for each step difference in reactivity worth was recorded to obtain integral reactivity worth curve. Integral reactivity worth curves are given for Shim I, Shim II, Shim III, Shim IV, and Regulating rod. The results are consistent for Shim I, Shim II, and Shim III rods. Calculated reactivity worth of Shim IV and Regulating rods was slightly less than the experimental results.

Some neutronic safety parameters are calculated for 3 MW TRIGA Mark II research reactor by M. A. Salam et al. from Reactor Operation & Maintenance Unit, Bangladesh Atomic Energy Commission (BAEC) [20]. These safety parameters are control rod worth, core excess reactivity, loss of reactivity with power increases, power defect, reactivity coefficients, cooling effect on fuel temperature and xenon poisoning. Neutronic safety parameters are measured by the digital Instrumentation and Control (I&C) system of the reactor. Experimental results of safety parameters are obtained from safety analysis report (SAR).

This research reactor is controlled by 6 control rods (Shim-1, Shim-2, Shim-3, Shim-4, Transient and Regulating). Worth measurements are performed at low reactor power (40 W) to avoid the temperature effect. All control rod worth values are measured by the positive period method. The integral and differential worth curves are drawn for all six control rods. Total core excess reactivity was found as \$7.839 at 50 W. Therefore, measured core excess reactivity was in safety limits (≤ \$10.27). Integral worth curves of rods were not in the traditional S-shape, this shows that rod worth is negligible for last one inch rod withdrawal.

The Monte Carlo model of ITU TRIGA Mark II research reactor is created by MCNP5 code and integral reactivity worth curves of control rods via rod position are given by Türkmen and Çolak from Department of Nuclear Engineering, Hacettepe University and Energy Institute of ITU respectively [21]. The purpose was to compare the experimental integral reactivity worth curves of control rods with the simulation results. Fresh fuel configuration was used when benchmark was done between experiment and simulation. According to simulation results, reactivity worth of the control rods match with the experimental ones. Reactivity worth curves were obtained by positive period method in experiment, but the control rod worth was calculated by a different method with the simulation. In the method that was used for simulation, first reactor core was modelled when all control rods are out. Then, after stabilizing the criticality value and the source distribution of the model, concerned control rod was inserted gradually into the core. The reactivity change between steps are used to obtain integral rod worth values [22].

An additional irradiation channel (IC) is created for Moroccan 2 MW TRIGA Mark II research reactor to improve the capacity of radioisotope production in the reactor by E. Chham et al. [23]. Radioisotope production capacity is one of the most important concepts for a research reactor. In the scope of this study, new core configurations are offered which IC positions are different for each of them. For these core configurations, a Monte Carlo model is created with MCNP code for the research reactor. Then, thermal neutron flux distribution and some neutronic safety parameters such as power peaking factors, excess reactivity and control rod worth are calculated by using the created model.

ENDF/B-VII is used for cross-section library. It is observed that, thermal flux distribution in the Central Thimble (CT) for the reference core is similar with the new core configurations which has new IC positions. But, the thermal flux distribution in ICs differs from the reference core for new core configurations due to the position of IC in the core. Positive period method as in experiment is used to simulate control rod worth calculation with MCNP code. Therefore, the calculations are done by withdrawing only one control rod while providing criticality by other control rods. The difference in reactivity caused by withdrawal of concerned control rod is recorded and integral reactivity worth is obtained. It is observed that the total worth of five control rods are close to each other for all core configurations and it is almost $16. In addition, to demonstrate that Moroccan TRIGA Mark II reactor can operate safely at 2 MW with new core configurations due to new added ICs, different thermal hydraulic parameters were analyzed by using PARET code. The results for this analysis showed that, reactor can operate safely with new core configurations. Because, the obtained thermal hydraulic parameters met with safety limits of the reactor.

Rehman and Ahmad have considered two different core configurations of TRIGA Mark II research reactor in this study [24]. These core configurations were named as core 133 and core 134. Core 133 was used for criticality safety calculations and core 134 was a rearrangement of core 133 with additional fuel elements to increase the excess reactivity. In this study PRIDE code was used for control rod worth calculations. Group constants to be used in diffusion theory with PRIDE code were generated by using 1-D transport theory code WIMS-D/4. Excess reactivity was calculated when all control rods are out.

When the control rod worth was calculated; one control rod was selected, insertion length was taken as 5 cm, group constants for the selected rod were used for “z” cm from the top of the core while other control rods are fully out, the effective multiplication factor was calculated and by using this multiplication factor reactivity worth is calculated. This was done for all control rods. As a result of their study; core 134 has slightly more excess reactivity than core 133, so core 134 was chosen for the analysis of control rod worth. Then comparison was made between experimental and Monte Carlo simulation results.

It was seen that for all control rods, the diffusion calculation overestimated the control rod worth with group constants of a single control rod. Therefore, instead of using same group constants, homogenized group constants of the control rods and the adjacent fuel elements were calculated and given as input for PRIDE. In this case, diffusion theory code gave better results with reduced deviations.

A model of TRIGA Mark II reactor of Pavia (Italy) is created with Serpent code by Castagna et al. from Department of Physics “G. Occhialini”, University of Milano-Bicocca [25]. Simulation of integral and differential control rod worth curves is done by the Serpent code. Fresh fuel configuration is used for the benchmark between the experiment and simulation results. The experimental procedure used to calculate the reactivity worth values is stable period method based on in-hour equation. Reactivity change is calculated measuring the period “T” of power increase when the control rods are small step extracted and then this period value is put in the in-hour equation to obtain criticality, this criticality is used to obtain reactivity. For the Serpent model, control rods are simply withdrawn by small steps and for each step, reactivity change is calculated to draw the integral and differential reactivity worth curves. As a result, it was seen that curves for regulating and transient control rods matched with the experimental results. For the shim rod, reactivity worth is calculated within 10% accuracy.

Neutronic analysis of TRIGA Mark II research reactor of Joseph Stefan Institute (JSI) is done by Wanbin Tan et al. from Chinese Academy of Sciences in this study [26]. This reactor has 250 kW thermal power like ITU TRIGA Mark II research reactor. Computational codes are used for reactor design and safety analysis, so validation of these codes by benchmarking is important. The scope of study in this paper is investigating the neutronic characteristics of JSI TRIGA Mark II research reactor. These neutronic characteristics are the effective multiplication factor and two safety parameters (control rod worth and fuel temperature reactivity coefficient). The investigations are done by SuperMC program.

Analyses are done by using different cross-section libraries. It was seen that effective multiplication factor showed changes for different cross-section libraries. As a result, it was also seen that control rod worth values showed better results with Monte Carlo codes.

Experimental temperature reactivity coefficient is smaller than computational one because temperature change is not considered in computational analysis. Eventually, it is seen that benchmark results are consistent with experimental ones.

PARET-ANL code is designed for analyzing the consequences of nondestructive accidents in research and test reactor cores. There are several researches done using PARET for TRIGA Mark II research reactors. Some of them can be seen below.

B. Nacir et al. created a PARET model for Moroccan TRIGA Mark II research reactor in order to calculate the 3-D temperature distribution in the core and all the most important parameters like axial distribution of DNBR across the hottest channel. The most important conclusion is that 12% of the fuel element utilization will have no influence on the safety of the research reactor while working around 2 MW power [27].

MCNP5 and PARET/ANL codes are used to create 3D Model of 2 MW Moroccan TRIGA Mark II research reactor for thermal-hydraulic and safety analysis by Y. Boulaich et al. [28]. There are two instrumented fuel elements near the center of the core. The temperature validations for PARET/ANL are done by using the data taken from these instrumented fuel elements. These validations are done at several power levels and also after shut down (SCRAM). Besides temperature validations, power validations are also done. The power factors needed to be described for PARET/ANL analysis are created by MCNP5 model of the reactor. The safety analysis for reactivity insertion cases is done and it is seen that safety design limits are not exceeded for the clad integrity by analyzing the behavior of thermal hydraulic parameters versus time. 2200 kW for maximum power and 750°C for maximum fuel temperature are the safety limits of this reactor.

Thermal-hydraulic parameters such as fuel centerline, surface and coolant temperatures of the 3 MW TRIGA Mark II research reactor under steady-state conditions are analyzed by Rahman et al. from Atomic Energy Research Establishment, Bangladesh [29]. NCTRIGA, PARET and COOLOD-N2 computer codes are used for these analyses.

Neutronic analysis is performed with MCNP4C code and the results are used in NCTRIGA. 3-D diffusion code CITATION and 3-D Monte Carlo code MCNP4B2 are used to couple the output of neutronic analysis. And the coupled output is used in PARET to investigate the thermal-hydraulic behavior of the reactor under steady state conditions. The experimental and operational data of TRIGA reactor are used in benchmarking to test NCTRIGA, PARET and COOLOD-N2 model calculations. As a result, it is seen that these models can be used in investigating thermal-hydraulic behavior of the reactor under steady-state operation for both natural and forced convection of coolant flow. The codes are used to obtain the safety margins. In this study, authors indicate that the results can be used to obtain a better core configuration of the reactor.

A thesis study on GSTR (Geological Survey TRIGA Reactor) is reported by Nicolas Shugart. The US GSTR has 1 MW steady state power and it is located in Lakewood, Colorado [30]. Some analyses are done in the scope of the thesis under the relicensing of the reactor. The 3-D Model of the reactor is created with MCNP5 code. This model is used to obtain; temperature and void coefficients, power distribution and excess reactivity to be used in thermal-hydraulic analysis later. According to the result of MCNP5 code, the excess reactivity of the core is \$6.48 and the peak rod power is 22.2 kW. In addition, the fuel and void reactivity coefficients are negative while the water temperature coefficient is slightly positive as observed from other TRIGA analyses. In addition, RELAP5 and PARET-ANL models of the reactor are created for hot-rod fuel channel under steady state and transient conditions. A worst case scenario for GSTR is analyzed as a part of this thesis using the mentioned codes. Peak fuel temperature is 829 K and the DNBR is 2.16 for the hot rod during steady state operation. A \$3 pulse reactivity insertion is applied for the core and it was seen that the peak fuel temperature reached up to 1070 K. RELAP5 mod 3.3 code is used for steady-state and transient analysis and PARET/ANL version 7.5 is used to make comparison with RELAP for low reactivity insertions. Comparisons show that RELAP and PARET results are in agreement with each other generally. The hot-rod is analyzed at normal operation with 1.1 MW reactor power.

The pulse analyses are done from $2 to $3 with RELAP code. It was seen that some amount of void occurred for all pulse reactivity insertions. As a result, after the pulse, it was seen that there was flow instabilities caused by weaknesses in the two-channel model. Experimental parameters such as peak power, pulse energy and FWHM (**F**ull **W**idth at **H**alf **M**aximum) are analyzed in terms of dependence of inserted reactivity in case of a pulse by Anže Pungerčič et al. These parameters analyzed for JSI (Jozef Stefan Institute) TRIGA Mark II research reactor in Slovenia [31]. The analyses are done using Fuchs-Hansen model and it was seen that results were in good agreement with the experimental ones. Sensitivity analyses are carried out to investigate the uncertainties of Fuchs-Hansen model. It was seen that uncertainties were relatively high for the reactivity insertions below $1.5. Experimental database used in this study for pulse is publicly available at the website of JSI. The experimental database can be reached from [32]. Inserted reactivity, peak power, energy of a pulse, power and temperature signals, control rod calibration and position, maximum fuel temperature, the scheme of core configuration, FWHM and isotopic composition of the fuel are all available at the website.

3. CONTROL ROD WORTH ANALYSIS

Control rods are used for two main purposes in nuclear reactors; arranging the criticality of the reactor to bring the power at a preferred level and keeping the reactor critical through the changes like power over reactor's lifetime. Therefore, analyses on control rods for safety of nuclear reactors is crucial. Keeping the reactor period stable is important and understanding the rod worth concept is the first manner for this purpose. The rod worth is the amount of reactivity given to the system to obtain the observed period. The worth of a rod is the change in multiplication factor of the reactor that the rod can compensate to keep the reactor at critical state [33]. Here, multiplication factor is the ratio of number of neutrons in one generation to number of neutrons in preceding generation. Since the number of neutrons generated depends on the number of fission events, multiplication factor is directly related with the amount of fission in the reactor. Multiplication factor is shown with "*k*" and is defined as eq. (2.1) [34]:

$$k = \frac{number\ of\ neutrons\ in\ one\ generation}{number\ of\ neutrons\ in\ preceding\ generation} \tag{2.1}$$

As it can be understood from the equation that when $k=1$, number of neutrons generated in consecutive generations are the same, and thus, the reactor is said to be critical. Moreover, when $k<1$, the number of neutrons generated in previous generation is more, so the reactor is said to be subcritical and when $k>1$, the number of neutrons generated in previous generation is less, so the reactor is said to be supercritical.

As indicated before, the change in the multiplication factor can be used to express the term of rod worth. The rod worth is defined as eq. (2.2) [33]:

$$\rho_w = |\rho| = \frac{k-1}{k} = \frac{k-k_0}{k} \tag{2.2}$$

In eq. (2.2), ρ_w represents the reactivity of rod worth, k and k_0 represent the multiplication factors of two consecutive generations. One of the safety parameters for a nuclear reactor is control rod worth. Therefore, the accuracy in control rod worth analysis is very important. Correct simulation of control rods used in the core model to make numerical analysis is very important too. Even small deviations in the model can create significant systematic errors of calculated multiplication factor values [35]. Making benchmark analysis is important to validate the numerical model used. Therefore, there is sufficient number of studies on control rod worth calculation that the results of numerical model analysis are compared with experimental ones. Analyzing the results of numerical model of ITU TRIGA Mark II research reactor created with MCNP for control rod worth and making comparison with the experimental results is in the scope of this thesis study.

Control rod worth analysis of ITU TRIGA Mark II research reactor was done using positive period method experimentally. For one of the studies in the literature, the positive period method is used to determine integral worth curve of MNSR control rod and the theory behind this method is explained [36]. Reactor period is the time required for the power to change by a factor of “e”. Here, “e” is the base of natural logarithm and the value of it is 2.718. The unit of the reactor period is usually in seconds. As it can be interpreted from the definition, the reactor period characterizes the reactor power. Relation between reactor power and reactor period can be seen in eq. (2.3);

$$P = P_0 e^{t/T} \tag{2.3}$$

Here, P is the transient reactor power, P_0 is the initial reactor power, t is the time during the reactor transient and T is the reactor period. Relationship between the reactivity and the reactor power can be seen in eq. (2.4);

$$\rho = \frac{l}{T} + \sum_{i=1}^{6} \frac{\beta_i}{1+\lambda_i T} \tag{2.4}$$

Where, β_i is the delayed neutron fraction for group i, λ_i is the decay constant of delayed neutron group i (s^{-1}), l is the prompt neutron lifetime (s), and T is the reactor period.

Eq. (2.4) is defined as “reactivity equation” or “in-hour equation”. When the parameters of β_i, λ_i and l are known, reactivity can be obtained by inserting the reactor period into the equation or vice versa.

When calculating the control rod worth of a reactor, period is used to obtain the reactivity inserted to the reactor by the control rod movement.

Positive period method steps used in calculating the control rod worth of ITU TRIGA Mark II research reactor experimentally is as follows;

1. The reactor is brought to a critical point and control rod positions are recorded.
2. The corresponding control rod is withdrawn to a specific level from the critical position. The step sizes for control rod to withdrawn are specified in the experiment.
3. The power doubling time, which is period, is measured after each withdrawal step and it is recorded.
4. The reactivity for each rod withdrawal position is obtained by using the measured period value and the available “in-hour” data.
5. It is ensured that the corresponding control rod is brought to a level to make reactor critical after 4^{th} step, before additional withdrawing.
6. All steps are repeated till the corresponding control rod is fully withdrawn.
7. The reactor is brought back to original critical state at low power level.
8. Finally, the integral rod worth curve is obtained for the fresh core configuration.

Steps explained above show the experimental method to obtain integral and differential control rod worth curves of ITU TRIGA Mark II research reactor. MCNP is Monte Carlo N-particle code that can be used for neutron, photon and electron transport problems [37]. The axial and radial view of the MCNP model for ITU TRIGA Mark II research reactor can be seen in Figure 3.1.

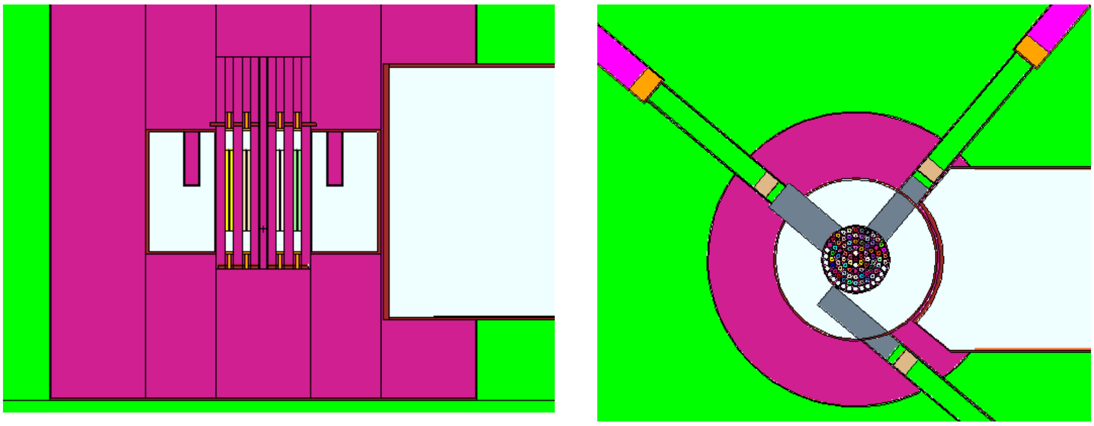

Figure 3.1 : The axial (left) and radial (right) views of MCNP model of ITU TRIGA Mark II research reactor.

Control rod worth can be expressed in two ways as integral and differential control rod worth. Typical curves of integral and differential control rod worth can be seen in Figure 3.2. Integral rod worth is defined as the total reactivity of the rod at the level of withdrawal that is the rod is brought. In addition, it has the highest value when it is calculated at fully out position. Slope of the integral rod worth curve ($\Delta\rho/\Delta x$), which is the amount of reactivity inserted per unit step, has the highest value in the middle way of the graph. The reason for this is that highest neutron flux distribution is at the center of the core. When the slope of the integral rod worth curve is drawn as a graph, differential control rod worth curve is obtained. Change in rod worth in the differential rod worth curve is highest in the middle of the graph because rod movement has the biggest effect there. As it can be interpreted from the curve, the differential control rod worth is the change in reactivity per unit movement of the rod. Area under the differential rod worth gives the value of integral rod worth [38].

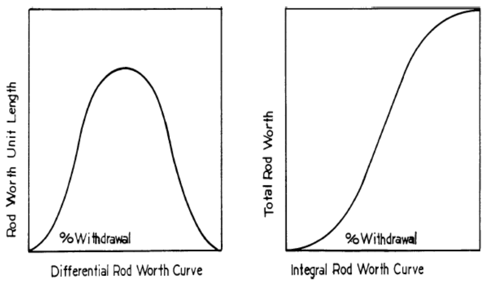

Figure 3.2 : Typical differential and integral rod worth curves [39].

As it is mentioned above, positive period method is used to obtain integral and differential control rod worth curves experimentally for the fresh core configuration of ITU TRIGA Mark II research reactor. Rod insertion method is used to obtain the control rod worth curves along with the experimental method by using 3D MCNP model of the reactor. Active fuel region has been divided to steps as done in experiment and these step sizes are used for rod insertion method. In addition, rod worth estimations are done for fresh fuel configuration since experimental results are for the fresh fuel configuration of ITU TRIGA Mark II research reactor. Rod insertion method is applied in two ways numerically for this thesis study; one is with recorded neutron source and the other one is without recorded neutron source for MCNP model. Source recorded case states that the neutron source output of the previous step is used for each step instead of “KSRC” card in MCNP. KSRC card is used to specify source for KCODE calculations for MCNP [40]. In short, an initial source is not stated for the steps after first run, the source of previous step’s output is used. In addition, MCNP input is run with 45000 particles and 2200 cycles in total. The standard deviation in the estimation of k_{eff} (effective multiplication factor) is 10^{-4} for MCNP results, which is very low. Reactor is brought to critical state at a low power level and then only one control rod is inserted to the core step by step while other control rods are fully out. Reactivity change by the movement of control rod is recorded for each step and the worth is obtained [41]. As one of the aims of this thesis study, all numerical results are compared with experimental ones. Aforementioned analyses are done for transient, safety and regulating control rods of ITU TRIGA Mark II research reactor for the fresh fuel configuration. Eq. (2.2) is used to calculate the reactivity difference between steps for the numerical simulation.

Reactivity difference between each step represents the reactivity inserted per unit movement of the rod. As explained above, total inserted reactivity gives the total worth of the control rod which is the integral control rod worth. Reactivity differences per unit step are used to create differential control rod worth curve.

Reactivity can be calculated in the unit of dollars using delayed neutron fraction (β_{eff}). The delayed neutron fraction (β) is the ratio of delayed neutrons to the all fission neutrons. Effective delayed neutron fraction (β_{eff}) is the ratio of delayed neutrons born in thermal energies to all fission neutrons. Effective delayed neutron fraction represents the one dollar of reactivity [42].

Relation between delayed neutron fraction and the reactivity in dollars can be seen from eq. (2.5) and eq. (2.6) [43]. The delayed neutron fraction is taken as 0.0073 for ITU TRIGA Mark II research reactor for all rod worth analyses.

$$\$1 = \frac{\rho}{\beta} \tag{2.5}$$

$$1cent = \$\frac{1}{100} = \frac{\rho}{100\beta} \tag{2.6}$$

Another concept that is important for nuclear reactors is excess reactivity. Excess reactivity is calculated when the all poison is removed from the nuclear reactor and control rods are fully out. Large excess reactivity is unwanted since the larger it is, the more poison is needed to compensate it [44]. Excess reactivity of ITU TRIGA Mark II research reactor for fresh fuel configuration is obtained in the scope of this study.

4. PARET/ANL ANALYSIS

In this section, PARET/ANL code structure is described in a general manner. In addition, examples of PARET/ANL analyses done before for GSTR (Geological Survey TRIGA Reactor) and 10 MW IAEA research reactor are repeated using the PARET/ANL 2001 revised version, that is used for this thesis study, and the results are compared with reference documents. Finally, the method of RIA pulse analysis done for ITU TRIGA Mark II research reactor using PARET/ANL is explained.

4.1 PARET/ANL Code

Reactivity initiated accident (RIA) causes changes in fission rate, and therefore, in power. In case of RIA, reactor power increases rapidly and this causes to temperature rise in the fuel. Heat transfer from the fuel cladding to the coolant is important to keep fuel temperature in defined limits. In addition, there will be a fast heating of clad surface, and the coolant may be heated at some locations on the clad surface so that it may cause bubble formations. Therefore, making thermal hydraulics analysis of RIA for reactors is essential for safety concerns.

RIA incidents require multiphysics analysis. Computational analysis of RIA requires two models in a code; one is for analyzing neutron kinetics of the reactor core and the other one is for analyzing thermal-hydraulic behavior in the core [10].

As mentioned before, PARET/ANL code, that RIA analysis can be done with, couples the thermal hydraulics and point kinetics equations. The coupling scheme of the PARET/ANL code can be seen in Figure 4.1. As it can be seen from the figure, the output of neutronics equations is used as input for thermal diffusion equations in fuel elements and hydrodynamics equations in coolant channels. All feedback effects are taken into account for neutronics equations.

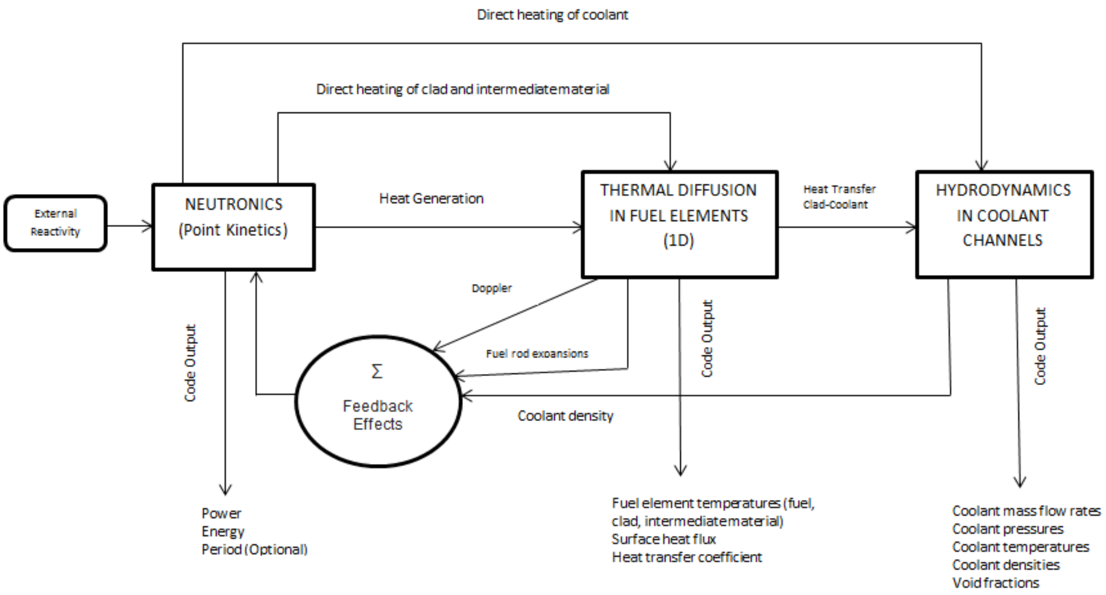

Figure 4.1 : Block Diagram of PARET/ANL [10].

Point kinetic equations, which define the power behavior of the reactor, solved numerically in PARET code can be seen from eq. (4.1) and eq. (4.2) [45];

$$\frac{dn(t)}{dt} = \frac{\left[\rho(t) - \beta\right]}{\Lambda} n(t) + \sum_{i=1}^{I} \lambda_i C_i(t) \tag{4.1}$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} n(t) - \lambda_i C_i(t), \quad i = 1, 2, \ldots, I \tag{4.2}$$

Here,

t = time

n = neutron density

ρ = system reactivity

β = effective delayed neutron fraction

Λ = prompt neutron generation time

λ_i = decay constant of group i

C_i = concentration of delayed neutron precursors of group i

β_i = delayed neutron fraction of group i

Runge-Kutta method of Cohen is used in PARET to solve point kinetics equations. Reactivity feedback from time zero to time under consideration is used in these equations. Externally inserted reactivity is specified as a function of time for the reactivity specified problems which is considered for PARET analysis in this thesis study.

All feedback effects such as fuel temperature and coolant density effects are summed up to calculate the feedback reactivity. These feedback effects are calculated pointwise region by region in the reactor.

The pointwise contributions are summed up and total reactivity feedback is estimated. Total compensated reactivity calculated at time t^m in PARET/ANL can be seen from eq. (4.3);

$$r_c^m = r_{Rod}^m + r_{MD}^m + r_{Dop}^m + r_{Cool}^m \tag{4.3}$$

Where, r_c^m is the total compensated reactivity and r_{Rod}^m is the reactivity feedback due to fuel rod expansion, r_{MD}^m is the reactivity feedback due to coolant density changes, r_{Dop}^m is the reactivity feedback due to fuel temperature changes, and r_{Cool}^m is the reactivity feedback due to coolant temperature changes. Eq. (4.4) is used for solving the r_c^m in PARET;

$$r\left(t^m\right) = r_{in}\left(t^m\right) - r_c^m \tag{4.4}$$

Where, $r_{in}(t^m)$ represents the externally inserted reactivity, the exact value of compensated reactivity is calculated at the end of the time step. Therefore, the reactivity between time steps is estimated by extrapolation using the value of previous steps to insert the required reactivity in point kinetic equations [45].

One dimensional heat conduction equation is used to calculate the heat transfer from fuel for each segment divided axially in PARET/ANL. While modelling the core, maximum 4 channels and 21 axial nodes can be used in PARET/ANL code. Figure 4.2 shows the radial regions and axial segments used in PARET/ANL model.

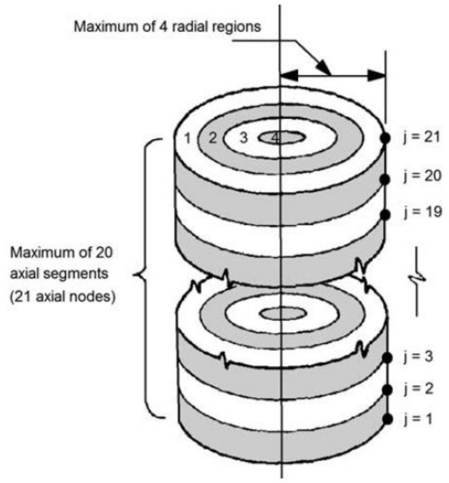

Figure 4.2 : Radial regions and axial segments in PARET/ANL model [10].

Properties of the coolant are taken as average for all axial nodes. Coolant is not divided radially for each axial node. The thermal diffusion equation that is used for each fuel element is as eq. (4.5) [45];

$$\frac{\partial}{dt}\left[g(u,r)u(r,t)\right]=\nabla k(u,r)\nabla u(r,t)+S(r,t) \tag{4.5}$$

In eq. 4.5; *u(r,t)* represents the temperature, *g(u,r)* represents the volumetric heat capacity and *k(u,r)* represents the thermal conductivity. *S(r,t)* is the heat source per unit volume. The value of *S(r,t)* is defined by average core power and pre-defined axial and radial weighting factors. The *r* and *t* represent the radial position and time respectively. Radial subdivision used in PARET/ANL for heat transfer calculations can be seen in Figure 4.3;

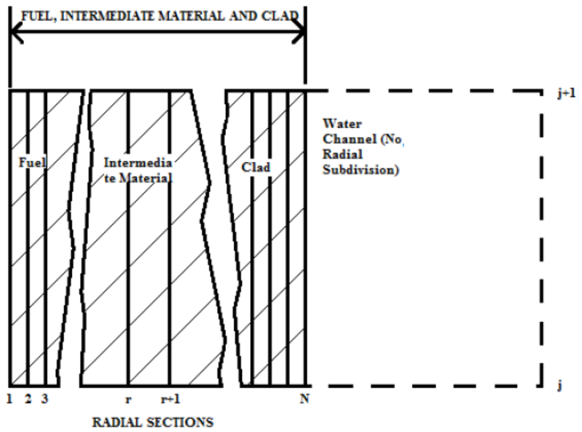

Figure 4.3 : Radial subdivision used in PARET for heat transfer calculations [45].

Mass, energy, and momentum conservation equations in one dimension are used in hydrodynamic model of PARET/ANL code. These one dimensional mass, momentum and energy conservation equations are as follows respectively [10]:

$$\frac{\partial \bar{\rho}}{\partial t} + \frac{\partial G}{\partial z} = 0 \tag{4.6}$$

$$\frac{\partial G}{\partial t} + \frac{\partial}{\partial z}\left(\frac{G^2}{\rho^{'}}\right) = -\frac{\partial p}{\partial z} - \frac{f\left|G\right|G}{2\bar{\rho}D_h} - \bar{\rho}g \tag{4.7}$$

$$\rho^{''}\frac{\partial H}{\partial t} + G\frac{\partial H}{\partial z} = q^{'''} \tag{4.8}$$

In these equations, t and z represents the time and position. G is the mass flux (kg/m^{2}s), $\rho^{'}$ and $\rho^{''}$ are the effective slip-flow densities (kg/m^3) for momentum and energy conservation equations respectively, p is the pressure (Pa), g is the gravitational acceleration (m/s^2), H is the specific enthalpy (J/kg), f is the friction factor, D_h is the hydraulic diameter (m) of the coolant channel, $q^{'''}$ is the volumetric heat source (MW/m^3). In addition, $\bar{\rho}$ is the average density (kg/m^3) of vapor and liquid phases.

Modified momentum integral method is used to solve one-dimensional hydrodynamic equations in PARET code. Momentum integral method solves the equations by evaluating all water/steam properties at a specified pressure that is given as input.

Lagrangian extrapolation is used to calculate local pressure values for each specific time step. The extrapolated pressure values are used to calculate fluid properties. Iteration is applied till extrapolated values are in agreement for local fluid pressure values. Only in case of estimating density, both local temperature and local pressure are used. Other fluid properties are estimated according to local temperatures at a given specific pressure.

Mass, energy and momentum conservation equations are solved using finite difference numerical method in space for each coolant channel in PARET/ANL model. Spatially discretized equations are integrated by time to incrementally improve the solution [10].

4.2 Examples of PARET/ANL Analyses

In this section, previous RIA studies done with PARET/ANL on GSTR and a 10 MW IAEA research reactor are repeated and the results are compared with the reference documents.

$1.5 pulse reactivity insertion analysis of 1 MW US Geological Survey TRIGA Reactor is done with PARET/ANL besides RELAP analysis as a part of thesis study [30]. A single average rod channel of the reactor is modelled for PARET analysis. Point kinetic parameters of the reactor used for PARET can be seen from Table 4.1 and Table 4.2.

Table 4.1 : Prompt fuel temperature reactivity for GSTR core [30].

Temperature (K)	Prompt Fuel Reactivity ($)
293.6	0.00
400	-0.99
600	-4.34
800	-7.55
1200	-13.0

Table 4.2 : 6 group delayed neutron fractions used in PARET [30].

Temperature (K)	Decay constant (1/s)	Fraction of delayed neutrons
1	0.0124	0.0323
2	0.0301	0.2185
3	0.1118	0.1969
4	0.3013	0.3954
5	1.1361	0.1154
6	3.0130	0.0415

In case of a reactivity insertion for pulse analysis, transient rod of the reactor is set to a height through its pneumatic system. The rod is held at this height for 1.5 seconds before fully dropping in the core. Thus, the reactor is at the same reactivity condition that is before pulse, after the rod is fully dropped.

As a final step, after 15 seconds from the initiation of the pulse, the remaining control rods are fully dropped into the core. The steps of $1.5 pulse reactivity insertion procedure for GSTR can be seen in Table 4.3;

Table 4.3 : Reactivity insertion sequence for $1.5 pulse reactivity insertion of GSTR [30].

Time (s)	Inserted Reactivity ($)
0	0
1	0
1.2	1.5
2.5	1.5
4.5	0
16	0
17	-5
3600	-5

This pulse procedure is applied for both PARET/ANL and RELAP analysis, then the results are compared. For this thesis study, to be able to see that created code input is validated for the pulse analysis, the same procedure for PARET/ANL is applied and results are compared with reference thesis study.

As it can be seen from Figure 4.4, Figure 4.5, Figure 4.6, Figure 4.7, Figure 4.8 and Figure 4.9; the results of the reference thesis study and PARET/ANL model created for validation are similar for power, peak fuel temperature and externally inserted reactivity behavior vs time. Peak power is 204 MW and the peak fuel temperature is 209.85°C according to PARET/ANL result of reference thesis study. These values are 214.77 MW and 216.39°C respectively for the PARET/ANL model created for validation.

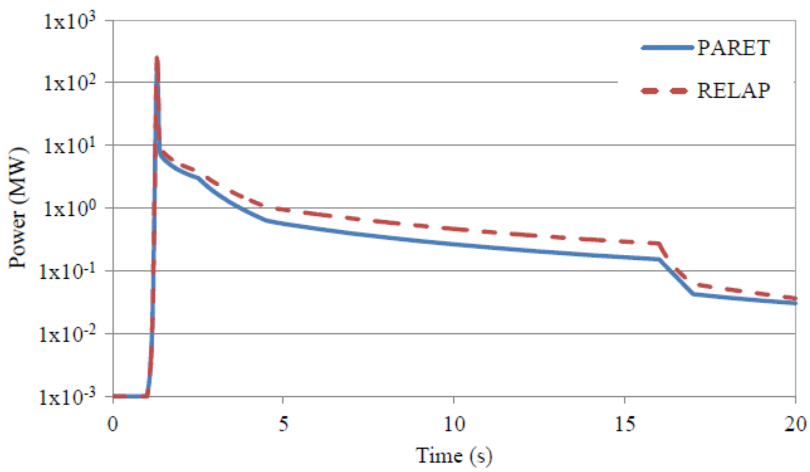

Figure 4.4 : Variation of power with time for PARET and RELAP models during $1.5 pulse [30].

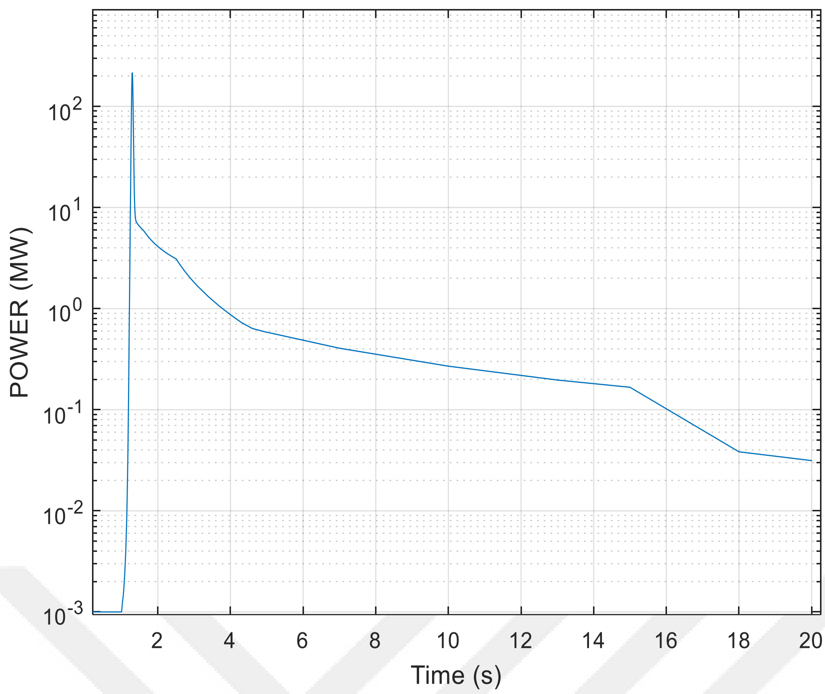

Figure 4.5 : Variation of power with time for PARET model created for validation during $1.5 pulse.

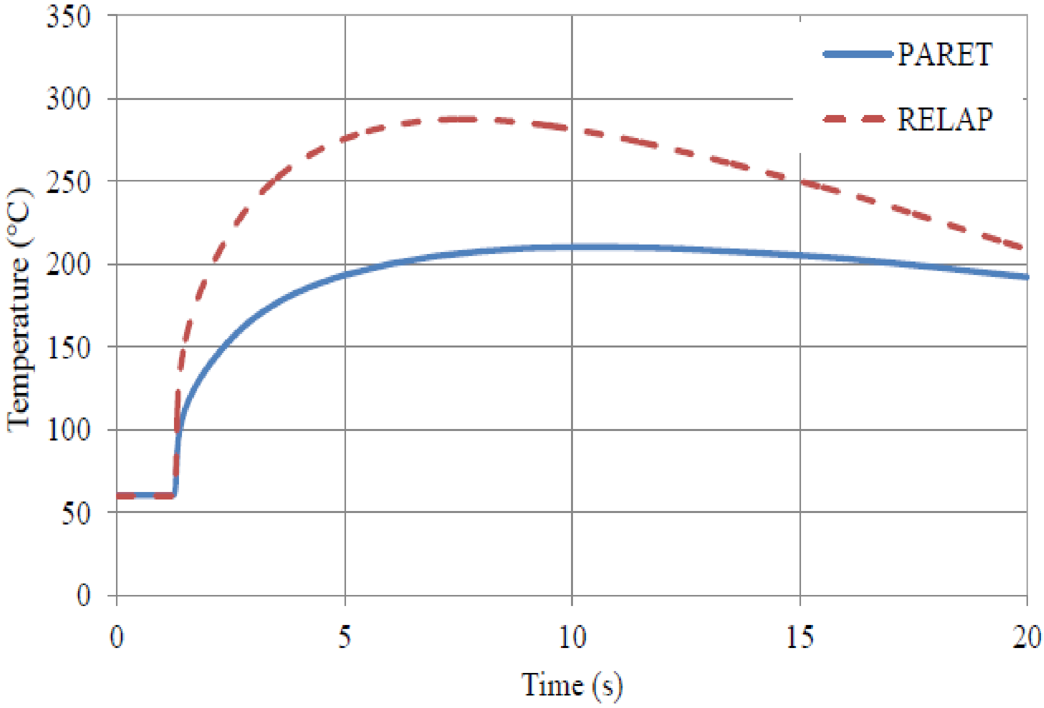

Figure 4.6 : Variation of fuel temperature with time for both PARET and RELAP models during $1.5 pulse [30].

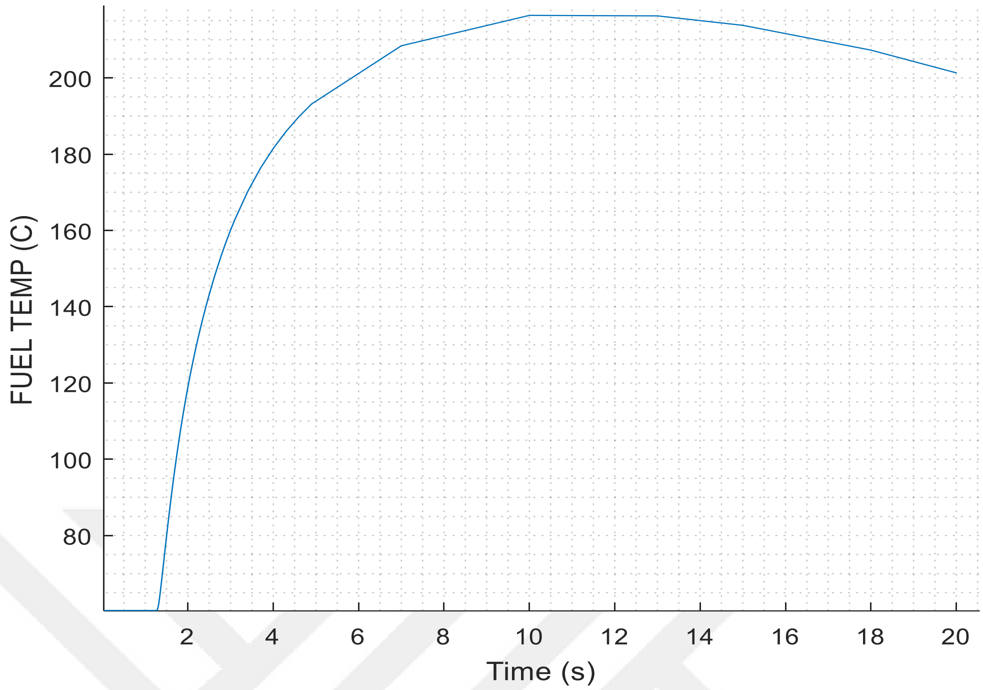

Figure 4.7 : Variation of fuel temperature with time for PARET model created for validation during $1.5 pulse.

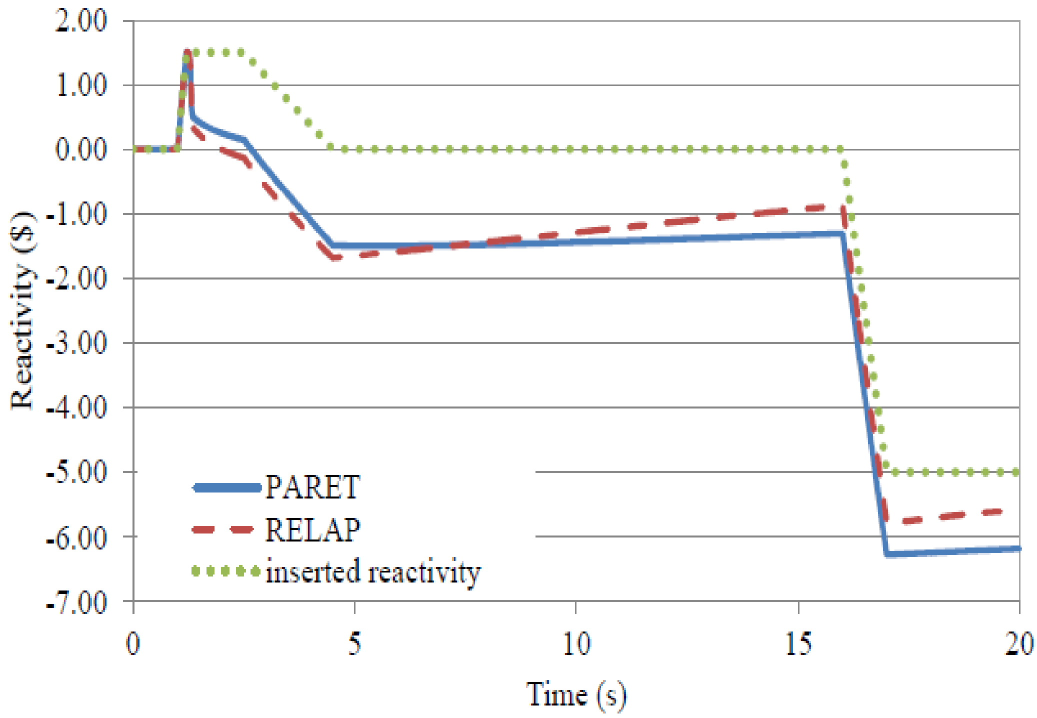

Figure 4.8 : Variation of inserted and total reactivity with time for both PARET and RELAP models during $1.5 pulse [30].

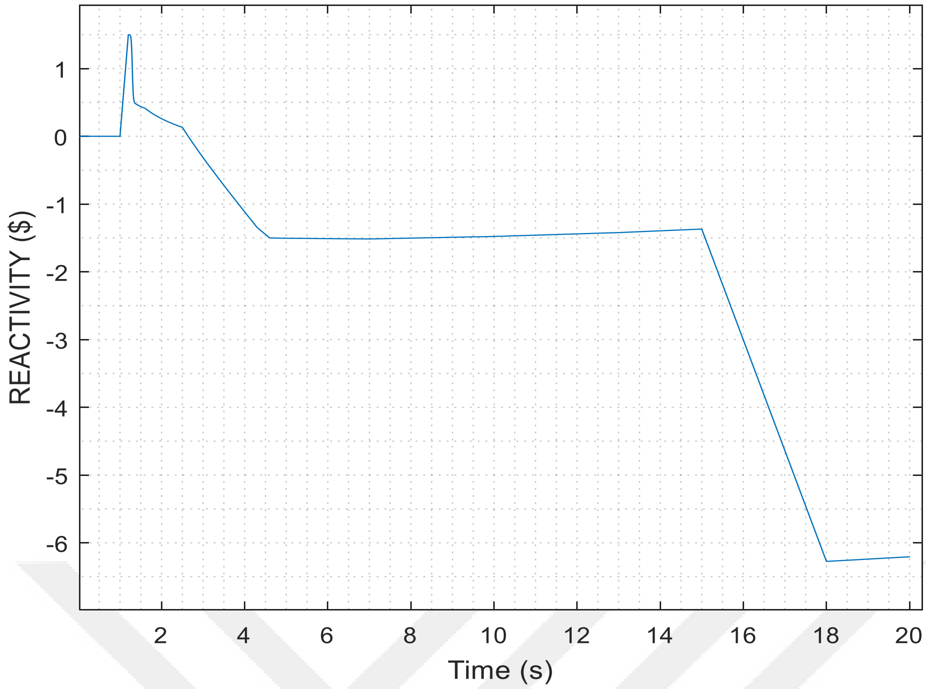

Figure 4.9 : Variation of total reactivity with time for PARET model created for validation during $1.5 pulse.

Another validation is done for a reference PARET/ANL analysis of a 10 MW IAEA MTR type, HEU core research reactor. A fast reactivity insertion case is analyzed. $1.5 reactivity is inserted in 0.5 seconds, then peak power and peak temperature values are compared. As it can be seen from Table 4.4, peak power of PARET/ANL model created for validation is closer to measured value of IAEA than power of PARET/ANL model of reference study.

The peak power is 132 MW for IAEA and 132.04 MW for mentioned PARET/ANL model. There is almost 7% difference between the results of time to peak power and overpower trip time values of IAEA measurements and the PARET/ANL model created for validation.

Time values to reach peak temperatures are very close to each other for three results. The PARET/ANL model created for validation predicted temperature values lower than expected but not so much. The causes of the difference between two PARET/ANL models of the same reactor may be computational.

The computational performance of the machine and software used to run the codes can create differences between results. In addition, the version of the PARET/ANL can cause differences. The model created is run in Linux with the version that is revised in 2001 while the version of the PARET/ANL used in reference study is 7.6.

Table 4.4 : Results of IAEA measurements and PARET/ANL analysis for \$1.50/0.5s ramp insertion.

	IAEA Measurement Results [16]	PARET/ANL results of reference study [16]	Results of PARET/ANL model created for validation
Time to Peak Power	0.609	0.609 s	0.656 s
Overpower (12.0 MW) Trip Time	0.660	0.655 s	0.610 s
Energy to Time of Peak Power	3.260 MWs	3.157 MWs	3.194 MWs
Peak Power	132.0 MW	129.7 MW	132.04 MW
Peak Fuel Temperature	170.9°C (0.670 s)	170.5°C (0.670 s)	167.65°C (0.673 s)
Peak Clad Temperature	155.9°C (0.672 s)	155.7°C (0.672 s)	154.54°C (0.675 s)
Peak Coolant Outlet Temperature	83.8°C (0.780 s)	83.25°C (0.755 s)	75.99°C (0.8 s)

4.3 RIA Analysis of ITU TRIGA Mark II Research Reactor

As aforementioned, RIA (Reactivity Initiated Accident) causes unwanted power increase, so it causes also temperature increases. It is important to do RIA analysis for making sure that safety parameters are under the limits to maintain the safety of the reactor. Pulse is a kind of RIA that is used to produce intensive power and radiation impacts by inserting reactivity in a short time interval in research reactors.

Neutron flux density is important for research reactors to do experiments such as irradiating the samples to make analyses on them. Neutron flux density is much higher for pulsed reactor than the reactor under steady state operation condition. Therefore, pulse is a commonly applied procedure for research reactors [46].

One of the TRIGA Mark II research reactors in the world is located in Jozef Stefan Institute, Slovenia. Reactor Infrastructure Center in the institute is responsible of operating the research reactor [47]. Reactor physics of the pulse procedure for TRIGA reactors can be explained by Fuchs-Hansen adiabatic model. Systematically, after pulse, the power increases, fuel temperature increases, and the reactivity decreases since TRIGA Mark II research reactor has negative temperature reactivity feedback. Therefore, power decreases and temperature increases after pulse.

Finally, reactivity reaches almost to zero value, sometimes a negative value, and then power is stabilized at a low level. Transient rod is withdrawn by its pneumatic system in case of a pulse experiment. Energy of the pulse depends on the fuel temperature reactivity effects (Doppler Effect).

Pulse procedure of TRIGA Mark II research reactor is as follows [48]:

- Reactor is made critical at low power level, generally less than 1 kW, when the transient rod is fully inserted.
- Vertical position of the transient rod is preset by adjusting the piston stopper to be able to insert the concerned reactivity.
- Other control rods are fully withdrawn.
- SCRAM is preset almost after 5 seconds from pulse.
- The signal is fired and pulse is initiated. Transient rod is withdrawn from fully in position to the preset upper position until upper stopper is reached.
- Pulse is created from 0.1 sec to 1 sec after signal is fired. This time depends on the inserted reactivity.
- SCRAM is carried out after preset time (usually 5 sec).
- Cooling of the reactor is carried out, it takes approximately 15 mins.
- Then the procedure is repeated for another pulse.

Inserted reactivity depends on the upper position of the transient rod and the feedback reactivity depends on the fuel temperature and temperature reactivity coefficient.

Feedback reactivity can be explained by eq. (4.9);

$$reactivity\ feedback \approx \alpha_f * \Delta T = \frac{\partial \rho}{\partial T} * \Delta T \tag{4.9}$$

Here, α_f is the temperature reactivity coefficient of fuel ($/°C), ΔT is the temperature difference in the fuel (°C), $\partial \rho$ is the reactivity change ($) caused by the reactivity insertion. When applying the aforementioned Fuchs-Hansen adiabatic model, some assumptions are made.

In the model, point kinetic approximation is used contribution of delayed neutrons during pulse is negligible, transient rod is withdrawn before temperature reactivity feedback takes place. Finally, eq. (4.10) and eq. (4.11) are taken into account to observe the power for pulse [48].

$$\frac{dP(t)}{dt} = P(t)\left[\alpha_0 - b\int_0^t P(t')\,dt'\right] \tag{4.10}$$

$$for\, t = 0, \frac{dP(t)}{dt} = \alpha_0 P(t) \rightarrow P(t) = P_0 e^{\alpha_0 t} \tag{4.11}$$

Here,

$$\alpha_0 = \frac{\rho'}{l}\,,\; b = \frac{\gamma}{l}\,,\; \gamma = -\frac{\alpha_f}{mc_p}$$

c_p is the specific heat capacity of the fuel (J/kg/°C), α_0 is the initial inverse period, *l* is the prompt neutron lifetime (s) and *m* is the total mass of fuel (kg) in the reactor.

Some pulse experiments of ITU TRIGA Mark II research reactor recorded in the past can be seen in Table 4.5. Pulse experiments are simulated using PARET/ANL code and the results are compared with experimental ones.

The analyses are done by dividing the core into 2 channels for PARET/ANL analysis and each channel has 19 axial nodes. One of the channels represents the hottest channel in the core and the other one represents the rest of the core. So the input values are averaged over rest of the core for second channel. Power ratios used in PARET/ANL input are obtained from 3D MCNP model of the reactor.

Table 4.5 : Experimental pulse records of ITU TRIGA Mark II Research Reactor.

Initial Power (W)	Inserted Reactivity ($)	Coolant Bulk Temperature (°C)	Peak Power After Pulse (MW)
50	1.5	28	68
50	1.81	28	175
200	2	28	246.5

Correlations used for thermal-hydraulic analysis of ITU TRIGA Mark II research reactor after aforementioned pulse scenarios are stated in the PARET/ANL code. Dittus-Boelter correlation is used in the code to estimate heat transfer coefficient (h_{sp}) for single-phase region analysis [49]. The correlation can be seen from eq. (4.12).

$$h_{sp} = 0.023\frac{k}{D_h}\left(\frac{GD_h}{\mu}\right)^{0.8}\left(\frac{c_p\mu}{k}\right)^{0.4} \tag{4.12}$$

h_{sp} is the heat transfer coefficient (W/m^{2}K) for the single-phase region, D_h (m) is the hydraulic diameter of the channel, G is mass flux (kg/m^{2}s), c_p is the specific heat (J/kgK), and μ is dynamic viscosity of the fluid (kg/ms).

The clad surface is covered by a bubble layer for high heat fluxes. In this case, heat is removed from the clad surface by subcooled nucleate boiling. Therefore, McAdam's correlation is used for the heat transfer at two-phase regions [50]. The correlation can be seen from eq. (4.13).

$$q_{fd} = 0.074\Delta T_s^{3.86} \tag{4.13}$$

In eq. (4.13), q_{fd} is the heat flux for subcooled nucleate boiling region (W/m^2), and ΔT_s (°C) is the difference between the saturation temperature of the coolant and the clad surface.

Critical heat flux mainly depends on the coolant velocity, pressure, and the degree of subcooling. Bernath's correlation is used specifically for TRIGA reactors by GA (General Atomics) and Argonne National Laboratory for DNBR (Departure from Nucleate Boiling Ratio) [18]. When the heat flux is over a specified value, bubble formation occurs on the surface of the clad. Therefore, the clad surface is covered by vapor film after the nucleate boiling. Since heat transfer decreases fast, clad surface temperature increases. This situation is expressed with the term of DNBR. DNBR is the ratio of the critical heat flux to the local heat flux in the core [50]. There are several correlation options in PARET/ANL code but, since Bernath's correlation gives the minimum DNBR it is more conservative. Therefore, it is better to use it for analysis of TRIGA reactors in terms of safety.

Bernath's correlation can be seen from eq. (4.14) [51].

$$CHF = h_{crit}\left(T_{crit} - T_f\right) \tag{4.14}$$

$$h_{crit} = 61.84\left(\frac{D_h}{D_h + D_i}\right) + \left(0.01863\left(\frac{23.53}{D_h^{0.6}}\right)u\right) \tag{4.15}$$

$$T_{crit} = 57\ln p - \left(\frac{54P}{P + 0.1034}\right) + 283.7 - \frac{u}{1.219} \tag{4.16}$$

Where, *CHF* is Critical Heat Flux (W/m^2), h_{crit} is critical heat transfer coefficient (W/m^{2}K), T_{crit} is the critical surface temperature (°C), T_f is the bulk temperature, p is the pressure (MPa), u is the fluid velocity (m/s), D_h is the hydraulic diameter (m), and D_i is the diameter of the heat source (m). The main parameters used in PARET/ANL code for ITU TRIGA Mark II research reactor can be seen from Table 4.6.

Table 4.6 : Characteristics of a Typical TRIGA Mark II Research Reactor [52].

Parameters	Values
Steady state power	250 kW
Peak power at pulse mode	1200 MW
Zirconium Rod Radius	0.3175 cm
Fuel Rod Radius	1.82 cm
Fuel + Cladding Rod Radius	1.88 cm
Active Fuel Height	38.1 cm
Delayed Neutron Fraction (β)	0.0073
Prompt neutron lifetime [53]	60 μs
Fuel temperature coefficient [53]	-1.0x10^{-4} $\delta k/k$°C
Pressure [27]	1.7 bar
Total flow area	~ 0.042 m^2
Hydraulic diameter [27]	0.0183 m
Coolant inlet temperature	~ 28 ℃
Coolant mass flux (below boiling)	~ 100 kg/m^{2}s
Coolant mass flux (below or possibly at boiling) [54]	~ 300 kg/m^{2}s

5. RESULTS AND DISCUSSIONS

In this section, results of the control rod worth and RIA analyses for ITU TRIGA Mark II research reactor are given and discussions are made. Control rod worth analysis using 3D Monte Carlo model of the reactor is done with two different methods; positive period method and rod insertion method. In addition, some pulse scenarios of the reactor are simulated using coupled PARET/ANL thermal-hydraulic code. Finally, all results are compared with the experimental ones.

5.1 Control Rod Worth Analysis of ITU TRIGA Mark II Research Reactor

3D full core MCNP model of ITU TRIGA Mark II research reactor is used for the control rod worth analysis. The methodology used is explained in chapter 3 of this thesis study.

Rod insertion method is used to estimate control rod worth using MCNP. This method is applied in two ways which are no source recorded and source recorded methods. Total control rod worth of transient, safety and regulating rods can be seen from Table 5.1.

As it can be seen from the table, experimental and numerical method results are in good agreement. Standard deviation for criticality (k) values obtained from MCNP results is about 10^{-4}, which is in acceptable level.

Numerical method results are based on a probabilistic code, which is MCNP, and the computational characteristics play an important role for the numerical results. Therefore, small differences between experimental and numerical results are in the expected limits (<10%) although the error is little bit different for source recorded case. Since an initial source guess is not given, instead the source output of previous step is used, worth values differ in small amount for source recorded case.

Table 5.1 : Control rod worth values from experimental and numerical methods.

Control Rod	Rod Worth Values from Experiment	Rod Worth Values from Rod Insertion Method (no source recorded)	Rod Worth Values from Rod Insertion Method (source-recorded)	Relative Error for no source recorded case (%)	Relative Error for source recorded case (%)
Transient	$3.16	$3.075	$3.089	2.69	2.25
Safety	$2.18	$2.219	$2.227	1.79	2.16
Regulating	$1.84	$1.826	$1.858	0.76	0.98

Integral rod worth curves of transient, safety and regulating control rods can be seen from Figure 5.1, Figure 5.2 and Figure 5.3 respectively. As it can be seen from the graphs, the “S” shape for integral rod worth curves are obtained for MCNP analysis as in the experiment. Integral worth curves are in very good agreement for safety and regulating control rods. Besides, total integral rod worth value for transient rod is in good agreement with the experimental value but the shape of the curve differ, specially in the middle region of the core. The amount of reactivity inserted per unit movement of the rods is the highest in the middle region of the core. This is caused by neutron flux distribution. Neutron flux area is the highest in the middle of the core [55]. Therefore, change in absorption rate is also the highest in the middle region of the core. Transient rod has the highest worth in the core since it is closer to the center of the core than other control rods. Its effect on absorption is the highest among the control rods. Therefore, eventhough the total worth estimation with MCNP is in good agreement with the experimental one, the integral worth curve differ mostly for transient rod. Besides all, starting point of rod insertion method and the step size for movement also have effects when it comes to compare the shape of the curves [8]. As it is indicated before, rod insertion method is applied in two ways as no source recorded and source recorded methods for MCNP simulations. Results are almost same for integral rod worth curves, where the shape of the curves for safety and regulating rods is closer to experimental ones for source recorded case. Since the source output of previous step is used instead of giving an initial source for each step in MCNP, source recorded case shows more realistic results. It shows almost same behavior with no source recorded case.

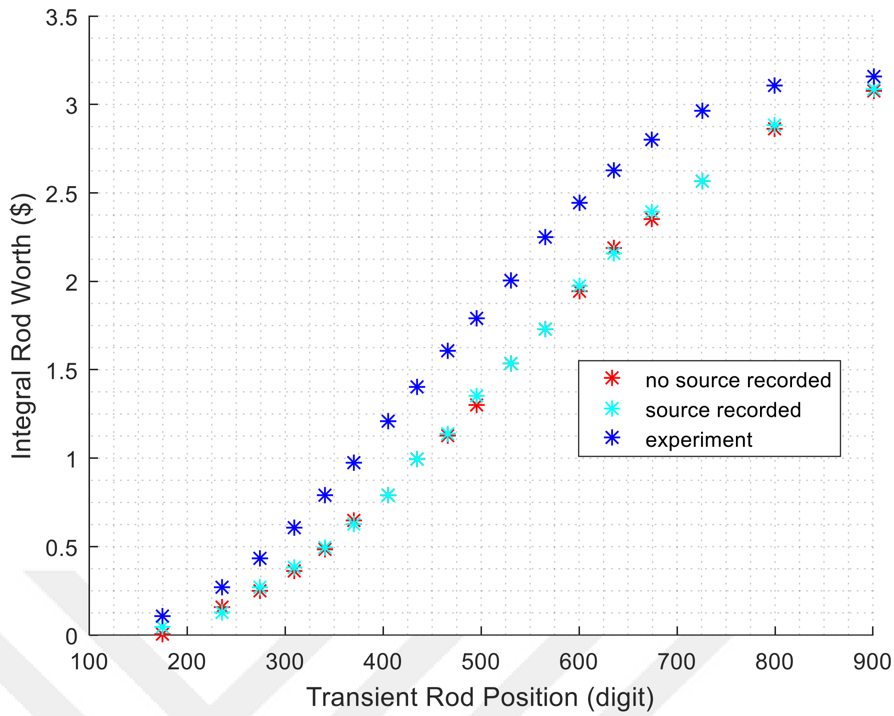

Figure 5.1 : Integral rod worth curve of transient control rod for MCNP and experimental results.

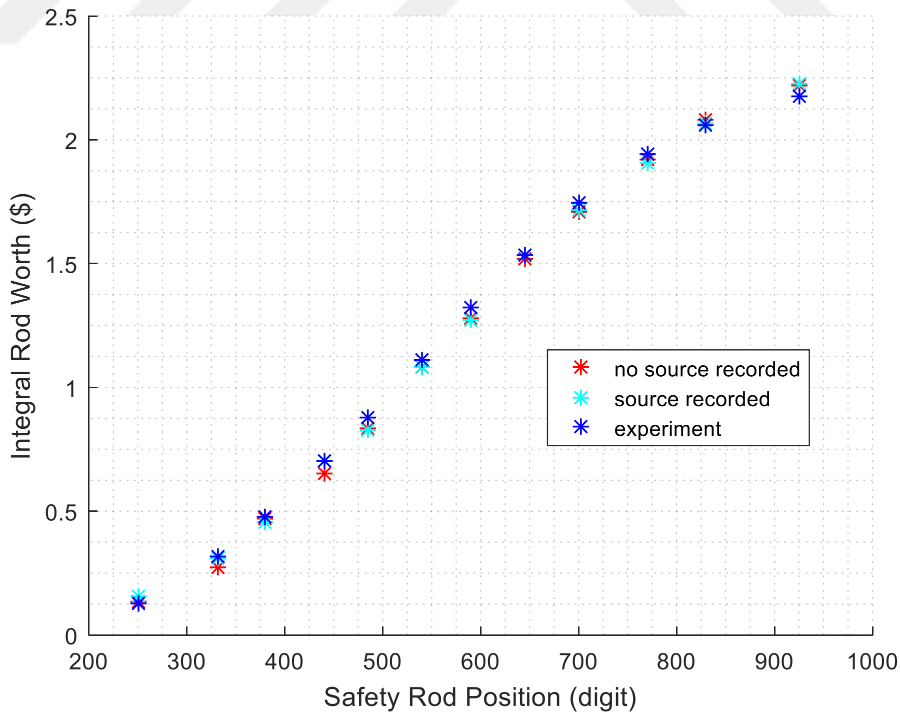

Figure 5.2 : Integral rod worth curve of safety control rod for MCNP and experimental results.

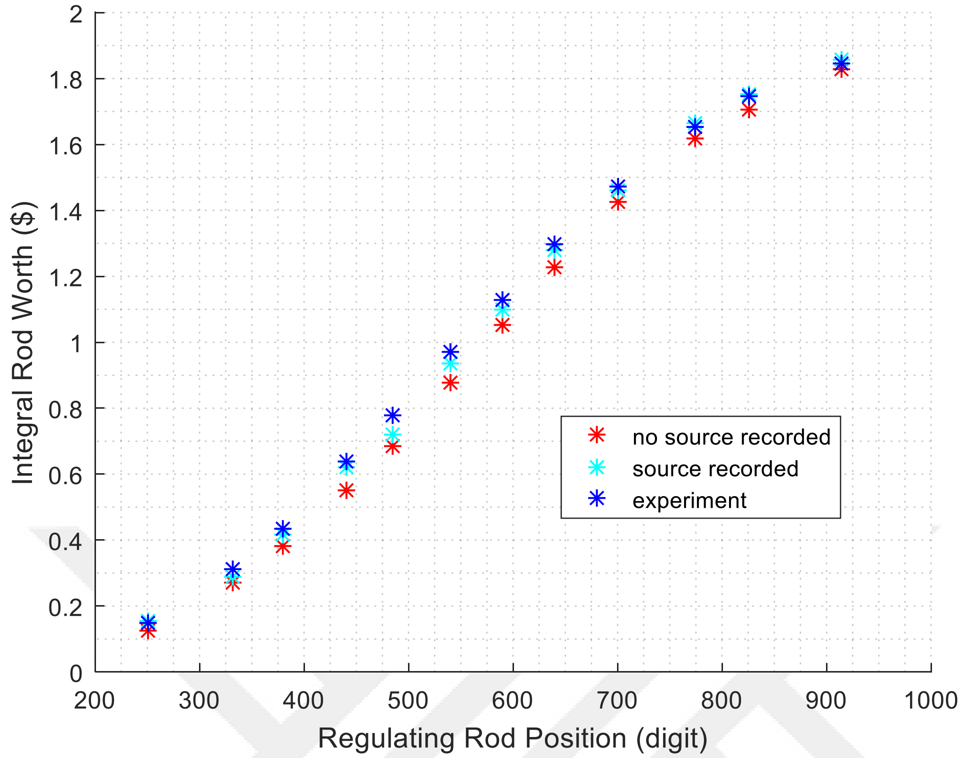

Figure 5.3 : Integral rod worth curve of regulating control rod for MCNP and experimental results.

Differential rod worth curves can be seen from Figure 5.4, Figure 5.5, and Figure 5.6. These graphs are obtained by drawing the slope of integral rod worth curve for each step. So the differential rod worth curves show the change in reactivity per unit step as explained in chapter 3 of this thesis study. As it can be seen from the figures, the curves are not exactly symmetrical about the midpoint of the core, this is due to the neutron flux distribution in the core.

Source recorded, and no source recorded cases of rod insertion method for transient and regulating rods show similar behavior for differential rod worth curves. The peak of the curves tends to be shifted to upper region of the middle point of the core. The difference between the behavior of the curves of MCNP simulation and experiment is the highest for transient rod. Since it has the highest worth among the control rods and it is closer to the center of the core, the deviation is expected to be most for transient rod as it is indicated before in this section. The reason is that the absorption effect is the highest for transient rod.

However, the step size for the movement of the rod and the starting point of the rod insertion method have important effects in the shape of the curves as it is stated before. Because, if they are not stated correctly, they may cause shifting in the behavior of the curves. Besides all, all step sizes and starting points are applied as in the experimental data.

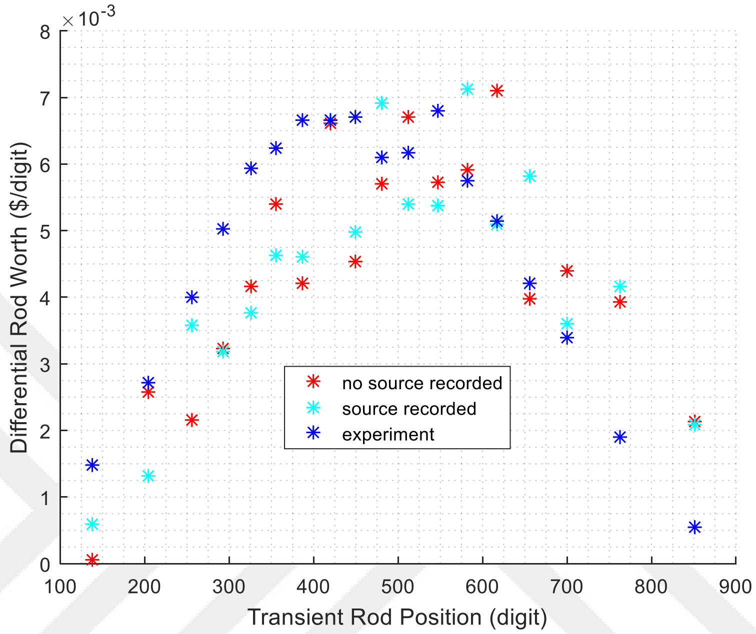

Figure 5.4 : Differential rod worth curve of transient control rod for MCNP and experimental results.

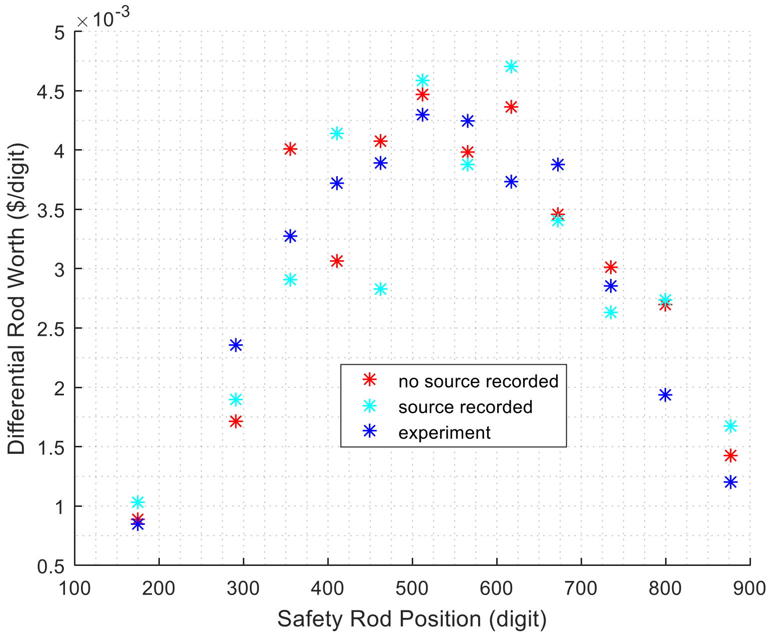

Figure 5.5 : Differential rod worth curve of safety control rod for MCNP and experimental results.

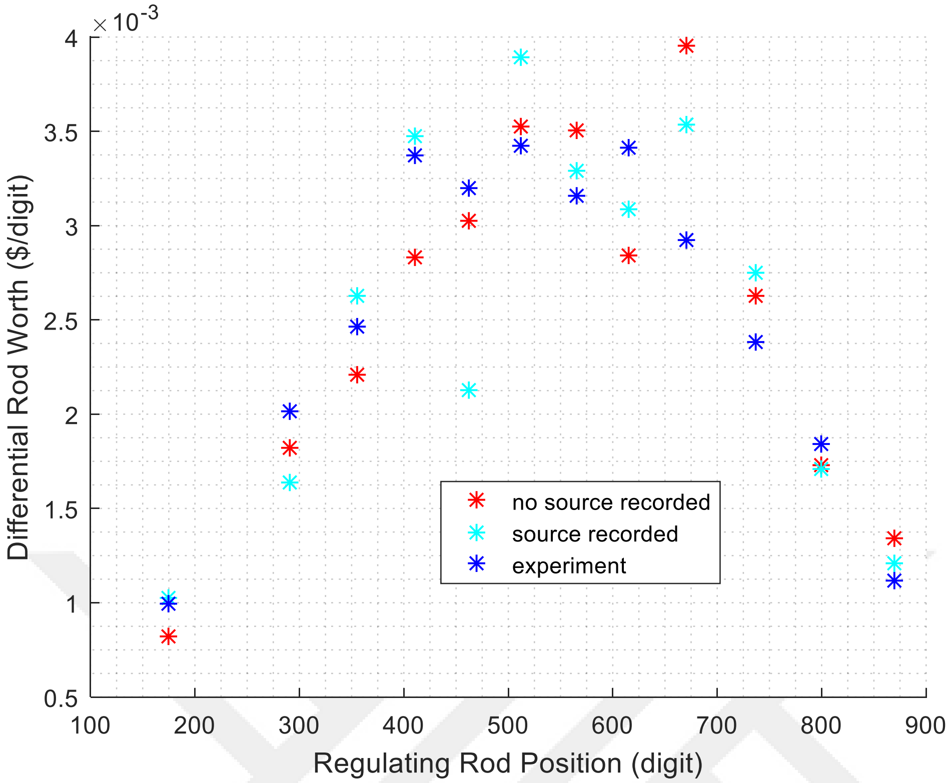

Figure 5.6 : Differential rod worth curve of regulating control rod for MCNP and experimental results.

One of the important parameters for the safety of nuclear reactors is excess reactivity. Criticality and excess reactivity values are calculated using MCNP when all control rods are withdrawn. These values can be seen from Table 5.2.

The total worth of control rods is about $7.18 for fresh core configuration of the reactor. Therefore, it can be seen that the total worth of control rods is enough to compensate the excess reactivity which is a desired situation for safety of the research reactor.

It is seen that the transient rod itself is enough to compensate the excess reactivity for the fresh fuel core configuration of ITU TRIGA Mark research reactor.

Table 5.2 : Criticality and excess reactivity for fresh fuel configuration when all control rods are withdrawn.

k_{eff}	ρ_{ex} ($)
1.02308 ± 0.0001	~ 3.09

5.2 RIA Analysis of ITU TRIGA Mark II Research Reactor Using PARET/ANL

Pulse analysis simulation of ITU TRIGA Mark II research reactor is done using PARET/ANL code based on the experimental pulse records. The results of the simulation are compared with the experimental ones for main parameters in this section. Ratios of local fission power density to the core average fission power density for axial nodes are taken from MCNP calculations and since it is observed that the ratios are almost same for the power values of 50 W and 250 kW, same power ratios are used for the different pulse analyses. Core of TRIGA Mark II research reactor is modelled as two channels; one represents the hottest channel in the core and the other one represents the rest of the core. Therefore, all analyses are done on these two channels and the temperature values are averaged for these two channels. Some of the experimental pulse records are given in Table 4.5. These pulse scenarios are simulated using PARET/ANL code.

Results of the simulations can be seen from

Table 5.3. As it can be seen from the table, peak power values are over predicted by PARET/ANL. Peak power values obtained from PARET/ANL simulations are so close to the experimental ones for the initial power of 50 W.

In case of initial power of 200 W and $2 pulse reactivity insertion, peak power is over predicted more, which is acceptable, because higher reactivity insertion cases cause more peak power output for PARET/ANL. This shows that the code is conservative in terms of safety concerns. This means that if the fuel and clad temperature values are in the desired limits for PARET/ANL results, they will be in the limits for real pulse case too.

Pulse scenarios higher than $2 could not be simulated using PARET/ANL. Because of the void formation after higher pulse reactivity insertions, PARET has difficulties to analyze the pulse. Therefore, the developers of PARET code suggests that PARET code shoud not be used for high reactivity insertion cases for natural convection model [30]. If the source code of the PARET/ANL can be improved, higher reactivity insertion pulse scenarios can be simulated in detail using the code.

Table 5.3 : Results of experiment and PARET/ANL for peak power values after pulse.

Initial Power (W)	Inserted Reactivity ($)	Inlet Coolant Temperature (°C)	Peak Power After Pulse (MW) Experiment	Peak Power After Pulse (MW) PARET/ANL	Relative Error (%)
50	1.5	28	68	69	1.4
50	1.81	28	175	180	2.8
200	2	28	246.5	275	11.6

Fuel temperature is one of the design limits for TRIGA reactors. This limit is for preventing the out-gassing of hydrogen from fuel composition. Out-gassing of hydrogen from the fuel produces stress in the clad material and since keeping the integrity of clad material is very important, this is an unwanted situation. Therefore, 1150°C is stated as upper limit of the fuel temperature to maintain integrity of the clad. In addition 760°C is calculated as upper limit of the clad temperature when the fuel temperature reaches up to 1000°C [56]. Maximum fuel, clad and coolant temperature values from PARET/ANL results can be seen from Table 5.4. As it can be seen from the table, the clad surface and fuel centerline tempertaures are in safety limits. In addition, minimum DNBR (Departure from Nucleate Boiling Ratio) values are stated in the table. DNBR is the ratio of the critical heat flux to the local heat flux in the core. So, minimum DNBR is one of the important concepts that should be considered in analyses in terms of safety. Minimum DNBR values, shown in below table, are for the hottest channel that is modelled for PARET/ANL. As it is expected, minimum DNBR has the lowest value for the highest reactivity insertion since the local heat flux is more and closer to the critical heat flux than other reactivity insertion cases. Nevertheless, they are higher than the value of 1.0, which shows that critical heat flux value is not reached and safety limits are satisfied.

Table 5.4 : Maximum fuel centerline, clad surface and coolant temperatures after pulse obtained from PARET/ANL.

Initial Power (W)	Inserted Reactivity ($)	Inlet Coolant Temperature (°C)	Peak Power After Pulse (MW)	Maximum Fuel Centerline Temperature (°C)	Maximum Clad Surface Temperature (°C)	Maximum Coolant Temperature (°C)	Minimum DNBR
50	1.5	28	69	290	102	95	2.65
50	1.81	28	180	332	125	115	1.48
200	2	28	275	376	125	115	1.31

Behavior of power, fuel centerline temperature, clad surface temperature, and coolant temperature versus time for $1.5 pulse reactivity insertion with initial power of 50 W can be seen from Figure 5.7, Figure 5.8 and Figure 5.9 respectively. In these figures, channel 1 represents the hottest channel and channel 2 represents the channel for rest of the core.

Peak power of 69 MW can be seen from Figure 5.7. Temperature values for channel 1 and channel 2 increases due to the pulse and with the effect of negative fuel temperature coefficient, temperature values decrease for both channels.

Maximum temperature values are observed for fuel centerline as expected. In addition, it can be seen also from the figures that the fuel centerline and clad surface temperatures remain in safety limits. Temperature feedback of the fuel has an important role in decreasing the power and temperature after the pulse as it can be observed from the results.

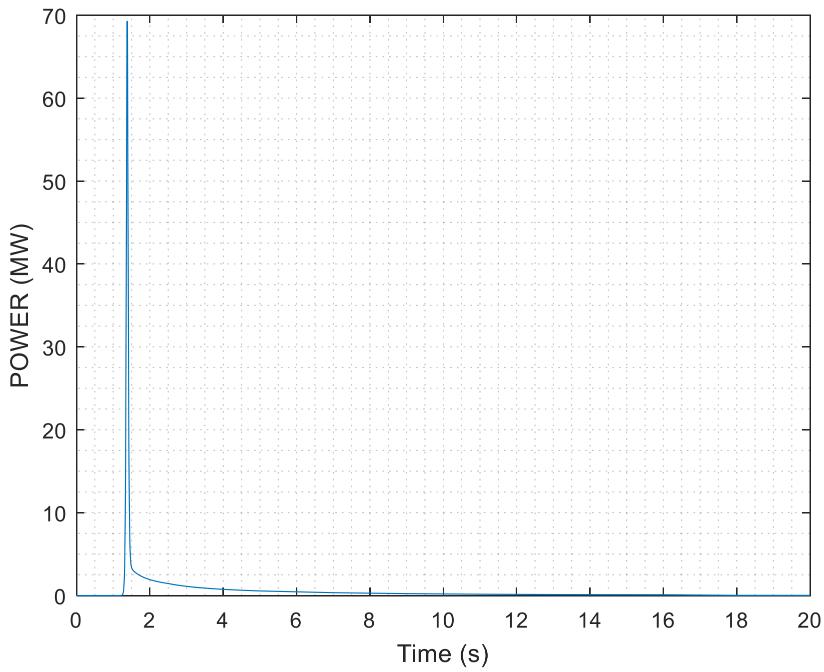

Figure 5.7 : Variation of power with time for $1.5 pulse reactivity insertion for initial power of 50 W.

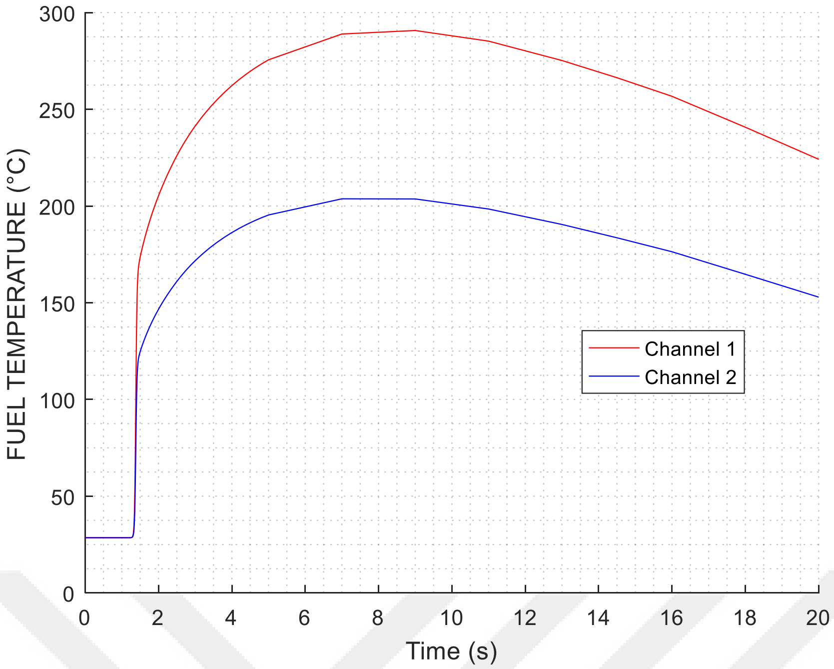

Figure 5.8 : Variation of fuel centerline temperature with time for $1.5 pulse reactivity insertion for initial power of 50 W.

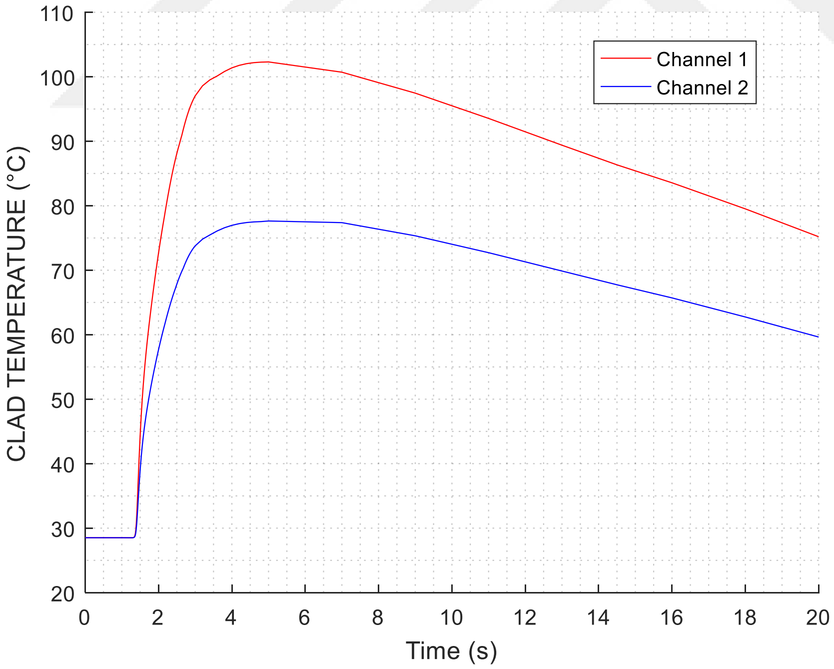

Figure 5.9 : Variation of clad surface temperature with time for $1.5 pulse reactivity insertion for initial power of 50 W.

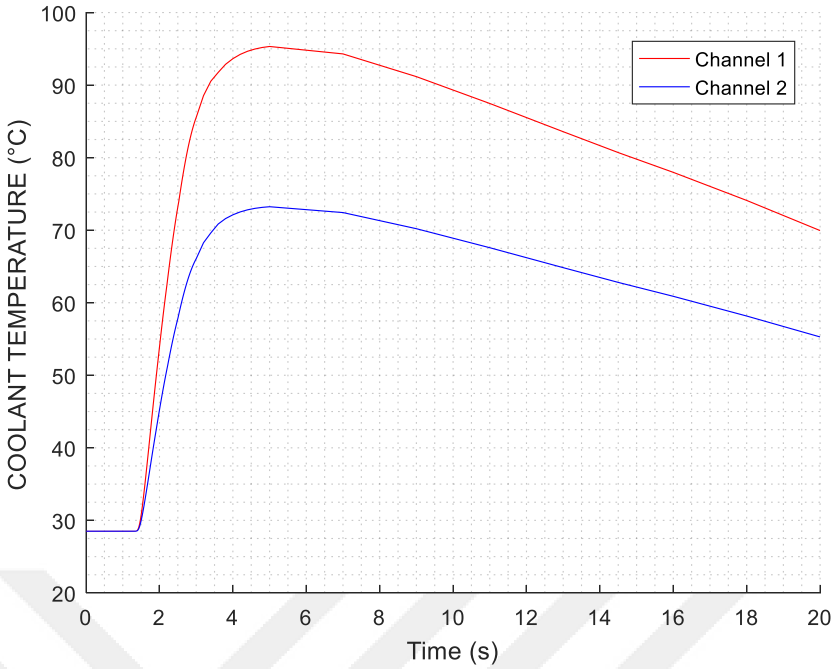

Figure 5.10 : Variation of coolant temperature with time for \$1.5 pulse reactivity insertion for initial power of 50 W.

Results for \$1.81 pulse reactivity insertion with 50 W initial power and \$2 pulse reactivity insertion with 200 W initial power can be seen in Appendix A respectively. As it is shown in figures, fuel centerline temperatures and the clad surface temperatures are within safety limits. In addition, due to the temperature feedback effects, the power, and the temperatures of fuel, clad and coolant decrease after the pulse as expected. Due to the void formation, PARET/ANL has difficulties to analyze high reactivity insertion pulse scenarios for natural convection mode. Therefore, when the temperature exceeds about 115°C, coolant temperature shows a constant behavior for a short while for the graphs of \$1.81 and \$2 reactivity insertion pulses. All these results show that, in case of given pulse reactivity insertions, the reactor will continue to operate safely.

6. CONCLUSIONS AND RECOMMENDATIONS

Integral and differential rod worth curves are obtained using 3D MCNP full core model of ITU TRIGA Mark II research reactor for fresh core configuration and the results are compared with experimental data. Positive period method is applied to create integral and differential rod worth curves experimentally. However, rod insertion method is used to obtain integral and differential rod worth values using MCNP. This method is applied in two ways as source recorded and no source recorded methods. It has been observed that the numerical and experimental results are in very good agreement. The errors, between numerical and experimental results, for total rod worth values of transient, safety and regulating rods are less than 5%. In addition, source recorded rod insertion method shows very similar behavior with no source recorded method. Besides, source recorded method shows closer behavior for integral rod worth curves of safety and regulating control rods. Differential rod worth curves are obtained taking the slope of integral rod worth curve for each step. In addition, the deviation in integral and differential rod worth curves has been observed for transient rod at most. This was expected because it has the highest worth and since it is closer to the middle region of the core where the neutron flux has highest distribution; its effect on absorption is high among the control rods.

Some experimental pulse scenarios are simulated using PARET/ANL code for RIA (Reactivity Initiated Accident) analysis of ITU TRIGA Mark II research reactor. The power density ratios are calculated using MCNP model of the reactor for PARET/ANL analysis. The pulse reactivity insertions more than \$2 could not be analyzed throughout the whole transient time by PARET/ANL due to the void formations, the code has difficulties in analyzing high reactivity insertion cases for natural convection models. Three pulse scenarios are analyzed using the code; \$1.5 and \$1.81 pulse reactivity insertions with an initial power of 50 W and \$2 pulse reactivity insertion with an initial power of 200 W.

Peak power values for $1.5 and $1.81 reactivity insertions are in very good agreement for numerical and experimental results. Peak power values are 69 MW, 180 MW, and 275 MW for $1.5, $1.81 and $2 reactivity insertions respectively. Peak power is over predicted by almost 11% for $2 reactivity insertion with the initial power of 200 W. Higher peak power value prediction for higher reactivity insertion is expected. Besides all, PARET/ANL over predicts all peak power values, compared to experimental results, which is a good outcome in terms of safety concerns. This shows that PARET/ANL gives conservative results. The power, fuel centerline, clad surface and coolant temperature behaviors versus time are also observed. It has been seen that the peak power, fuel centerline temperature, and clad surface temperature are in safety limits for all pulse reactivity insertion cases even though the peak power values are over predicted. The safety limit for peak power at pulse mode is 1200 MW, it is 1150°C for fuel centerline temperature and it is almost 670°C for clad surface temperature. This shows that the research reactor remains in safe operation limits after these pulses.

Experimental data is based on fresh fuel configuration of the reactor, therefore the control rod worth analysis is carried out for fresh fuel configuration. Since the accuracy of the numerical model is confirmed by this study, same core model can be used to estimate the control rod worth values for burnt fuel configuration as a future work. In addition, further work can be done for developing PARET/ANL code to be able to analyze high reactivity insertion cases for natural convection models.

The source code of the PARET/ANL is open to the public through the OECD NEA and US RSICC for the version of 5.0. However, since there is no documentation about the implementation of the numerical methods in the code, development and modification is difficult on the code [10]. Moreover, benchmark analysis between PARET/ANL and another code used for same analyses on research reactors (such as RELAP5/MOD3) can be beneficial to compare the results and understanding the behavior of the system in a better way.

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APPENDICES

APPENDIX A: The variation of power, fuel centerline temperature, clad surface temperature and coolant temperature with time for \$1.81 and \$2 pulse reactivity insertions with initial powers of 50 W and 200 W respectively.

APPENDIX A

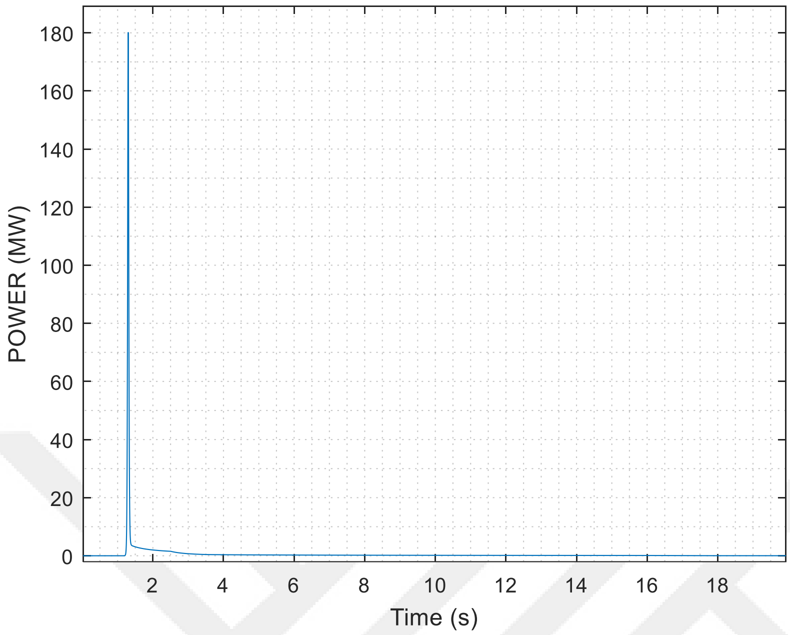


Figure A.1 : Variation of power with time for \$1.81 pulse reactivity insertion for initial power of 50 W.

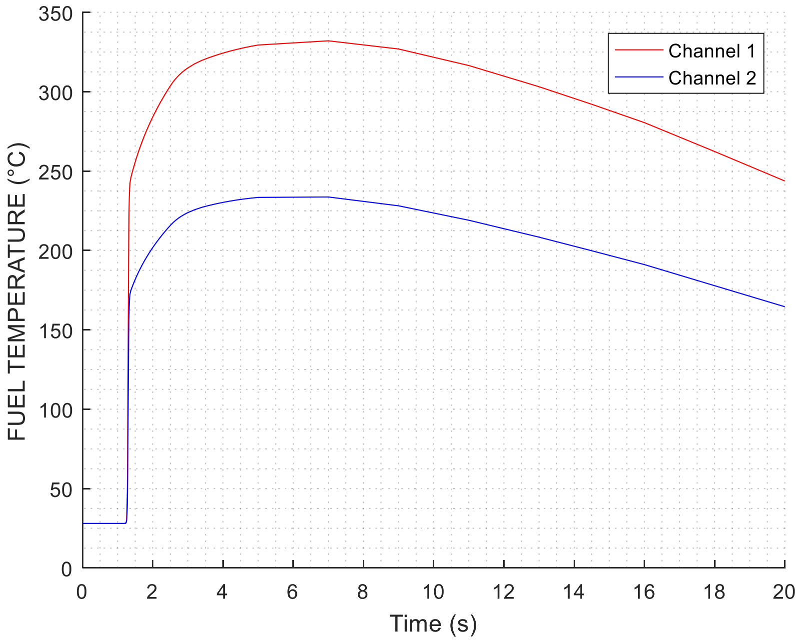
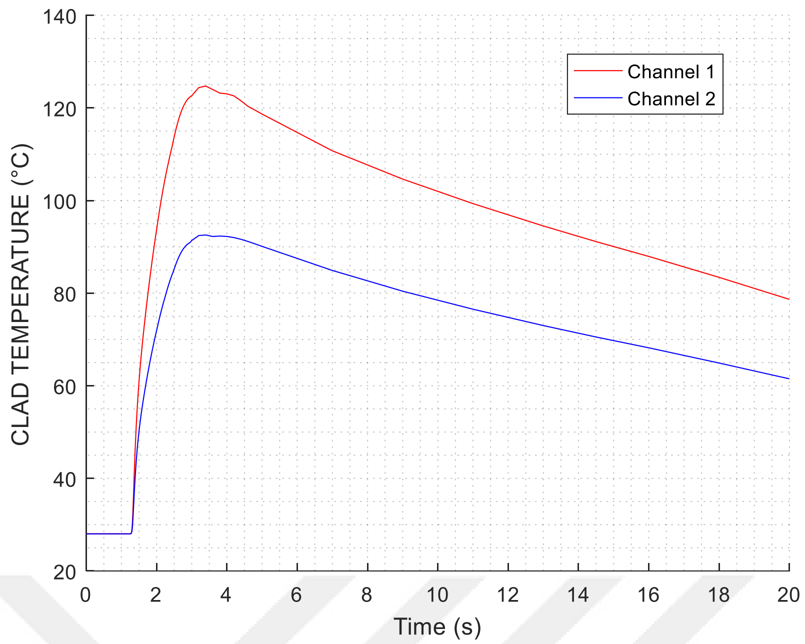


Figure A.2 : Variation of fuel centerline temperature with time behavior for \$1.81 pulse reactivity insertion for initial power of 50 W.



FigureA.3 : Variation of clad surface temperature for $1.81 pulse reactivity insertion for initial power of 50 W.

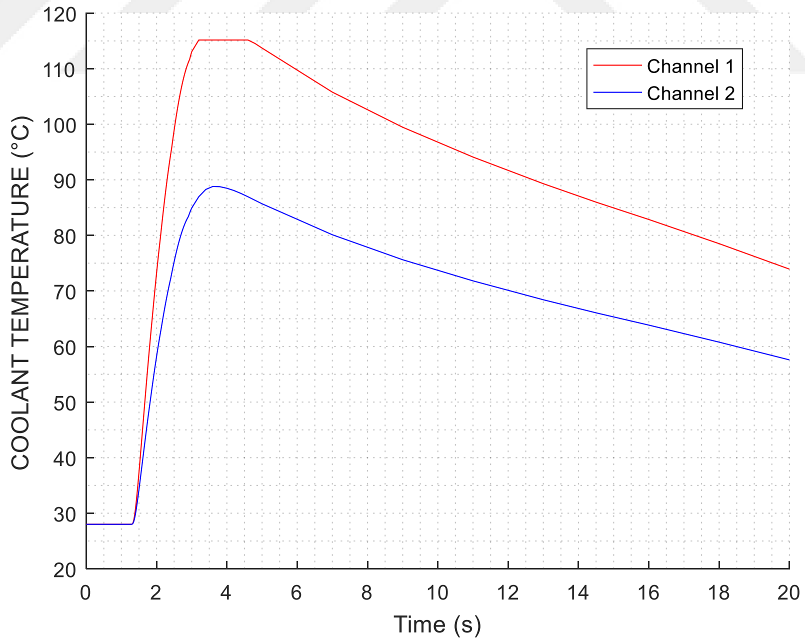


Figure A.4 : Variation of coolant temperature with time for $1.81 pulse reactivity insertion for initial power of 50 W.

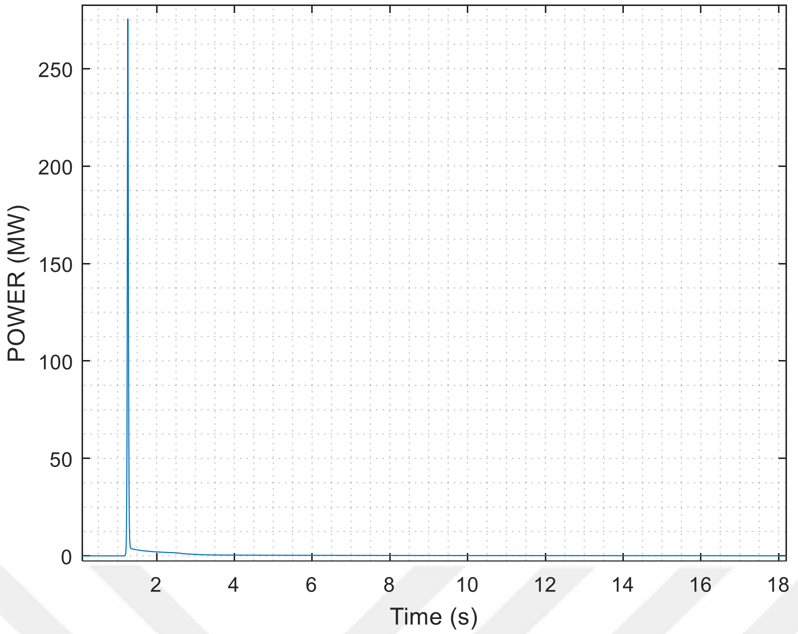


Figure A.5 : Variation of power with time for $2 pulse reactivity insertion for initial power of 200 W.

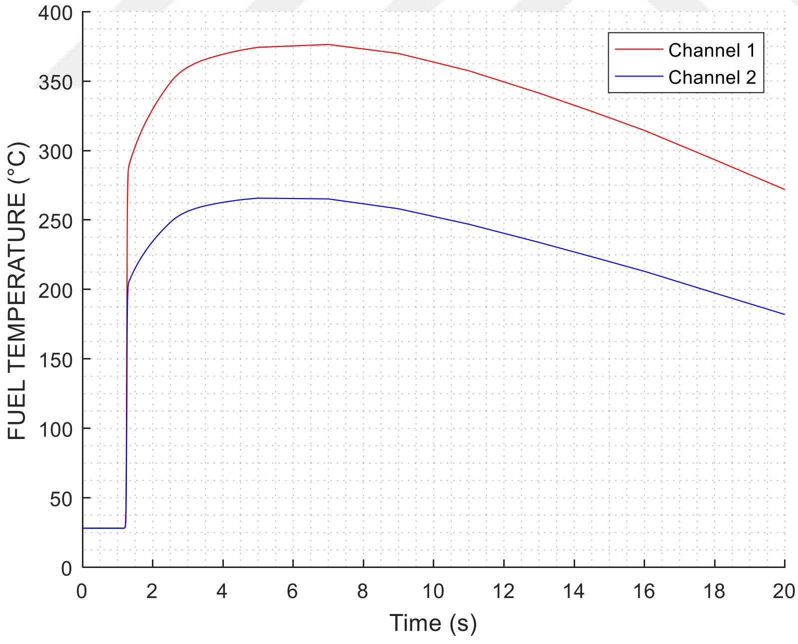


Figure A.6 : Variation of fuel centerline temperature with time or $2 pulse reactivity insertion for initial power of 200 W.

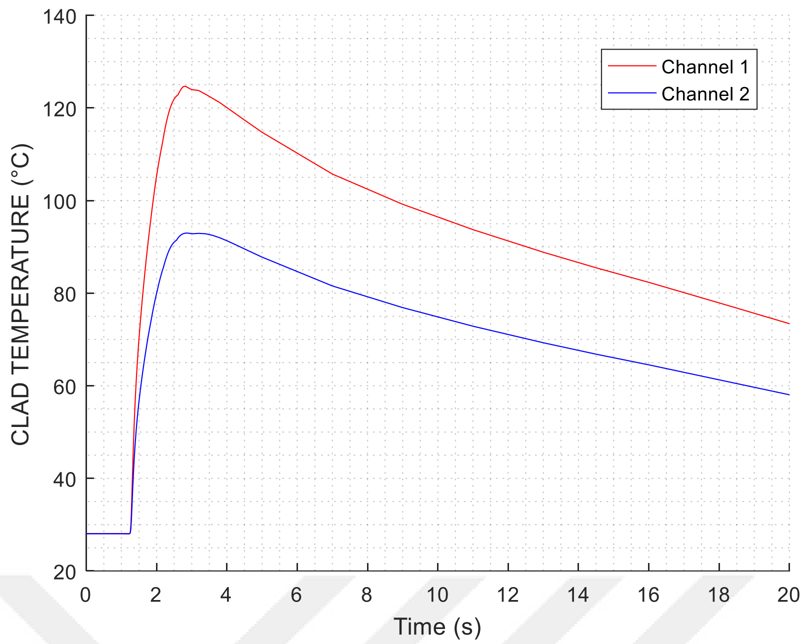


Figure A.7 : Variation of clad surface temperature with time for $2 pulse reactivity insertion for initial power of 200 W.

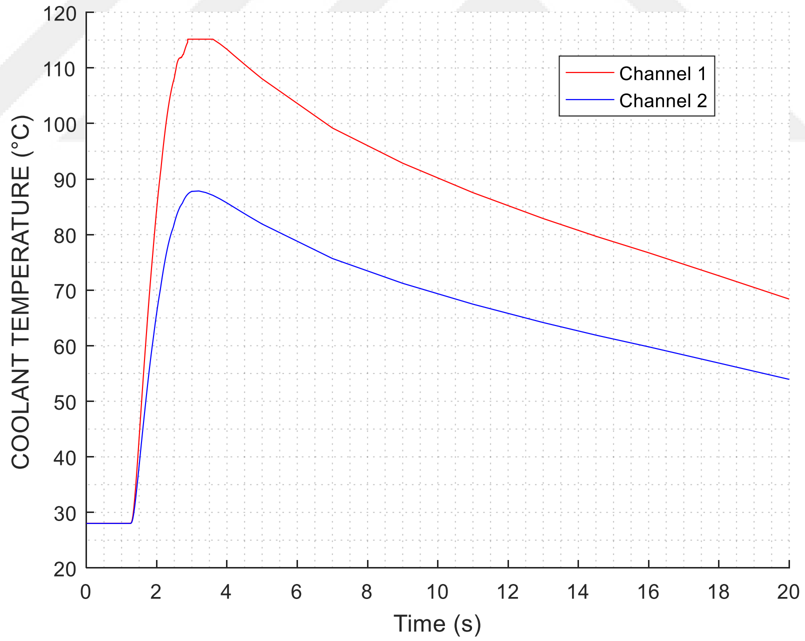


Figure A.8 : Variation of coolant temperature with time for $2 pulse reactivity insertion for initial power of 200 W.

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