

FEASIBILITY STUDY OF THE UNIVERSITY OF UTAH TRIGA REACTOR POWER UPGRADE

Part II: Thermohydraulics and Heat Transfer Study in Respect to Cooling System Requirements and Design

by

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The thermodynamic conditions of the University of Utah's TRIGA Reactor were simulated using SolidWorks Flow Simulation, Ansys, Fluent and PARET-ANL. The models are developed for the reactor's currently maximum operating power of 90 kW, and a few higher power levels to analyze thermohydraulics and heat transfer aspects in determining a design basis for higher power including the cost estimate. It was found that the natural convection current becomes much more pronounced at higher power levels with vortex shedding also occurring. A departure from nucleate boiling analysis showed that while nucleate boiling begins near 210 kW it remains in this state and does not approach the critical heat flux at powers up to 500 kW. Based on these studies, two upgrades are proposed for extended operation and possibly higher reactor power level. Together with the findings from Part I studies, we conclude that increase of the reactor power is highly feasible yet dependable on its purpose and associated investments.

Key words: TRIGA, research reactor, heat transfer, FLUENT, SolidWorks, PARET

INTRODUCTION

The university of Utah TRIGA reactor (UUTR) has been operating since 1975 without incident. The UUTR is a modified TRIGA Mark I pool-type reactor that is operated at a maximum of 90 kW, although licensed to operate at a maximum power of 100 kW. However, the fuel core design has the potential of increasing the overall power up to 1 MW. The UUTR is operated by trained and NRC licensed staff and students of the University of Utah Nuclear Engineering Program. The UUTR is utilized in many ways: to train students on reactor operation and nuclear principles, it provides a neutron and gamma source for research and is used for neutron activation analysis. The reactor is as well a major research and community outreach tool; tours of the facility are conducted educating the public and younger students about nuclear engineering.

This paper is a continuation from Part I [1] which focused on optimization of the higher power levels for UUTR based on neutronics parameters; in this Part II, we present thermodynamic calculations and computational fluid dynamics (CFD) simulations as performed to optimize heat transfer parameters for higher UUTR power levels. These analysis when integrated, deter-

mined a theoretical, range of higher power levels for UUTR. The methodologies used to assess thermal-hydraulics and heat transfer aspects of increased power of the UUTR are based on the methodologies as used for re-licensing its operation (in 2011). These information are easily found in the UUTR Safety Analysis Report published at the U. S. Nuclear Regulatory Commission [2].

COMPARISON OF MAIN OPERATIONAL PARAMETERS IN CURRENT RESEARCH REACTORS

As of 2012, there are 30 research and test reactors operating in the United States [3] as shown in fig. 1. Among TRIGA type reactors, UUTR is one of the reactors with the lower power. From this group of research reactors powered by 1 kW and greater, only three, including our UUTR, rely primarily on natural, ambient heat transfer for cooling and heat transfer. All other reactors are equipped with forced convection systems of equal or greater cooling load to the reactor's licensed power level. The most common type of system used is a dual loop heat exchanger linked to a cooling tower located at outside environment.

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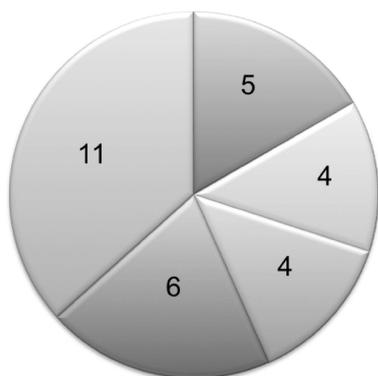


Figure 1. Operating research and test reactors in the USA grouped per their licensed power [3]

HEAT TRANSFER PHENOMENA IN UUTR

During reactor operation energy is released through the fission process starting the chain of heat transfer. While there exists an extensive research on heat transfer inside fuel and on fuel rods for TRIGA reactors [4, 5], this study focuses on the system as a whole. The fuel rods in the UUTR core heat the surrounding pool water, which flows up and over them in a natural convection loop. The heated water then can either evaporate from the top of the reactor pool or transfer the heat to the environment. Transferring the heat to the environment can be either through the top surface, which is open to the air or through the aluminum tank wall, sand barrier and outer steel tank wall. Currently, this is the only way the pool water is cooled back to ambient temperature.

In order to determine the heat transfer into the reactor pool from the operating core, the reactor is first considered to be a closed system with a control volume surrounding the core and water. The energy equation is then

$$Q_{in} = (m_t C_p \Delta T) + m_e (\Delta H_V + C_p \Delta T) \quad (1)$$

where Q_{in} [J] is the heat, m_t [kg] – the mass of water in the reactor tank, C_p – the specific heat of water (4,183.2 J/kgK, at 20 °C) [6], ΔT [°C] – the temperature difference from the starting and ending conditions, m_e [kg] – the mass of water evaporated, ΔH_V – the latent heat of vaporization for water at 20 °C (2,453.5 10³ J/kg) [6]. Using this equation it is possible to estimate the overall average temperature increase of the reactor tank water. From historical reactor run data as found in ref. [7] it was recorded that average water evaporation is 2.16 kg/h when operating at 90 kW. This leads to a temperature increase per hour of 2.58 °C. Table 1 shows calculated values for higher theoretical UUTR power levels. Because these power levels are theoretical values, they do not include the evaporative portion of the equation. This amount is variable and dependent on current atmospheric and starting

Table 1. Temperature increases per hour for higher UUTR power levels

Power level [kW]	Temperature rise per hour [°C]
100	2.87
200	5.74
300	8.60
400	11.47
500	14.34

conditions, but judging from the prior data from ref. [7] it would lower the temperatures by 2-4% with a greater amount of water being evaporated.

In addition to the system losing energy through evaporation, it is also lost through conduction. The heat generated in the fuel is transferred to the water through natural convection, from the water to the aluminum, sand, and steel enclosures by conduction and is finally cooled by the ambient air through convection. This process is calculated using a radial thermal circuit

$$R = \frac{1}{2\pi r_{core} L_{core} h} + \frac{T_{core} - T_{SS,2}}{R} + \frac{\ln \frac{\gamma_{Al,2}}{\gamma_{Al,1}}}{2\pi k_{Al} L_{Al}} + \frac{\ln \frac{\gamma_{sand,2}}{\gamma_{sand,1}}}{2\pi k_{sand} L_{sand}} + \frac{\ln \frac{\gamma_{SS,2}}{\gamma_{SS,1}}}{2\pi k_{SS} L_{SS}} \quad (2)$$

where q_r [W] is the heat rate, k [Wm⁻¹K⁻¹] – the thermal conductivity, r [m] – the radial distance, L [m] – the cylindrical length, T [K] – the temperature, γ [KW⁻¹] – the thermal resistance, and h [Wm⁻²K⁻¹] – the heat transfer coefficient. In this form the heat transfer to the outside surface of the steel tank is given by knowing the temperature of the core (contacting the water) and temperature of the outer stainless steel tank.

In order to use eq. (2) and find the energy loss through the tank walls the convective heat transfer coefficient (h) of the tank water must first be calculated. The first step requires the calculation of the dimensionless Grashof number

$$Gr = \frac{g\beta(T_{core} - T_{water})L_{core}^3}{\nu^2} \quad (3)$$

where Gr is the Grashof number, g [ms⁻²] – the gravitational acceleration, β – the volumetric thermal expansion coefficient of water (207.71 10⁻⁶ K⁻¹, at 20 °C) [8], T [K] – the temperature, L [m] – the length, ν – the kinematic viscosity of water (1.0058 10⁻⁶ m²s⁻¹, at 20 °C) [8]. This result is multiplied by the Prandtl number for water at 20 °C, 6.97 [6], to give the Rayleigh number.

The heated, upward-facing, flat plate correlation is used to model the reactor core. This correlation calculates the ratio of conductive to convective heat

Table 2. Variables affecting UUTR heat conduction

Variable	Description	Value
T_{core}	Fuel temperature contacting water at 90 kW	53.7 °C [3]
$T_{\text{SS},2}$	Temperature of outer SS wall	23.0 °C
r_{core}	Core radius	0.29 m
L_{core}	Core height	0.67 m
h	Heat transfer coefficient of water	704.84 W/m ² K
$r_{\text{Al},1}$	Inner radius of Al tank	1.17 m
$r_{\text{Al},2}$	Outer radius of Al tank	1.18 m
k_{Al}	Thermal conductivity of Al	177 W/mK [6]
L_{Al}	Height of Al tank	7.32 m
$r_{\text{sand},1}$	Inner radius of sand layer	1.18 m
$r_{\text{sand},2}$	Outer radius of sand layer	1.48 m
k_{sand}	Thermal conductivity of sand	0.27 W/mK [6]
L_{sand}	Height of sand layer	7.32 m
$r_{\text{SS},1}$	Inner radius of SS tank	1.48 m
$r_{\text{SS},2}$	Outer radius of SS tank	1.485 m
k_{SS}	Thermal conductivity of SS	14.9 W/mK [6]
L_{SS}	Height of SS tank	7.32 m

transfer known as the Nusselt number (Nu). Based on the calculated Rayleigh number and the chosen correlation the following relation is used [8]

$$\text{Nu} = \sqrt[3]{0.15\text{Ra}} \quad (4)$$

From knowing the Nusselt number, the heat transfer coefficient (h) can be known through their relationship derived from Newton's law of cooling

$$\text{Nu} = \frac{hL}{k} \quad (5)$$

where L [m] is the heated length and k [Wm⁻¹K⁻¹] – the thermal conductivity. This value is calculated to be 704.84 Wm⁻²K⁻¹. Now that it is known, the calculation can proceed to obtain the overall thermal resistance of the reactor system using eq. (2). Table 2 defines the other variables needed for the equation. The total thermal resistance (R) calculated for UUTR is 0.019 KW⁻¹ resulting in a loss of 1590 W of heat through conduction to the outer wall while operating at 90 kW.

When UUTR is at 90 kW it was found that the typical range for evaporative energy was 1.8-3.6 kW while only 1.59 kW were transferred through conduction. These methods only account for a maximum 5.77% of the total power generated. The remaining continues to heat the tank water through natural convection until it is eventually removed passively to the environment or actively by a heat exchanger.

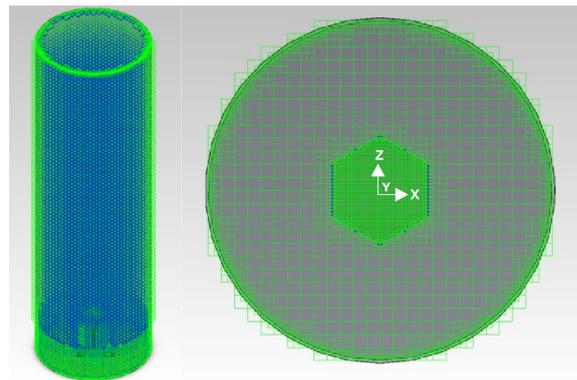
SOLIDWORKS BASED ASSESSMENT OF UUTR HEAT TRANSFER AND THERMOHYDRAULICS PHENOMENA

SolidWorks Flow Simulation is a fluid dynamics and thermal simulation program [9]. The software is based on the finite volume solution method to solve

the time-averaged Navier-Stokes and energy equations over the computational mesh. Because the equations are discretized over each volume, values for each surface can be known making this method conservative.

The basic elements of the UUTR were modeled in SolidWorks with their corresponding dimensions and materials. In the Flow Simulation software the computational domain was applied up to the outer wall edge and a 3-D rectangular mesh was selected. Using mesh refinement a minimum gap size of 0.79 inches (0.02 meters) was specified. The mesh near the walls and core has a finer resolution for better modeling of the thermal and velocity boundary layers whereas in the center of the tank the mesh is coarse. Figure 2 shows views of the meshed model and tab. 3 shows the mesh statistics.

The analysis was carried out using an internal simulation with time dependency and gravity enabled (-9.81 ms⁻² in the y-direction). Water was chosen as the working fluid to fill the tank. The variable properties of density, dynamic viscosity, specific heat (C_p), and thermal conductivity were defined over the range of 20 °C-70 °C with the software using a linear interpolation for intermediate values. Starting temperature for the simulations was set at 20 °C or 24 °C and air pressure at 1 atm (1 atm = 101325 Pa). The interior walls of the aluminum tank were defined to act as real walls with the heat transfer coefficient set at the previously calculated 704.84 Wm⁻²K⁻¹ (negative sign convention). The core was defined as a surface heat source with a constant, overall heat flux equal to the reactor's power level. Simulation time was set to run for a total of 6 hours

**Figure 2. Isometric and bottom views of flow simulation mesh****Table 3. Flow simulation mesh statistics**

Cell type	Number
Fluid cells	70 800
Solid cells	17 080
Partial cells	27 840
Total	115 720

*ft = 30.48 cm

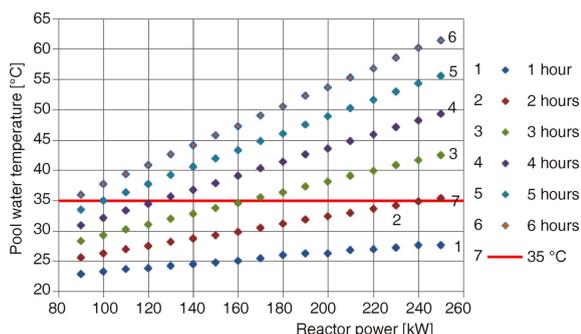


Figure 3. UUTR pool water temperature as a function of UUTR power as obtained with SolidWorks flow simulation

(21,600 seconds) with results being recorded at each hour.

Power levels from 90 kW to 250 kW were simulated in 10 kW increments. Each was started from an initial temperature of 20 °C. Additionally, a single 90 kW, 1 hour simulation was run with a starting temperature of 24 °C. Temperature measurements were taken at an arbitrary location 4.5 m (14.76 ft*) from the base of the reactor tank along the center axis. This location was chosen as it proved to be an area of even temperature distribution to represent the overall heating of the reactor tank water. The results from the 20 °C simulation are shown plotted in fig. 3.

FLUENT MODEL OF THE UUTR

Fluent is a computational fluid dynamics code based on the finite element method [10]. A computational mesh is generated for the model being analyzed with the defining equations applied over each mesh element. Boundary conditions are defined at all mesh edges (walls) creating a conservative system where individual element values can be known. For all simulations Fluent solves the continuity equations of mass and momentum and if heat transfer is involved the energy equation is also added.

The tank, walls, core, and water of UUTR were modeled to their original dimensions in the DesignModeler program included with the simulation package. During this process the UUTR core was further discretized into zones so that the pin power distribution could be mapped to the surface. For the meshing portion the created geometry was opened and a tetrahedral shaped mesh and CFD physics preference were chosen. A coarse relevance center was used with medium smoothing and a slow transition area around the core for increased flow detail in that region. The generated mesh is shown in fig. 4 and its statistics are presented in tab. 4.

The mesh is loaded into the Fluent solver and is set to perform a pressure-based, absolute, transient simulation with gravity enabled (-9.81 ms^{-2} in the



Figure 4. External surfaces of fluent mesh

Table 4. Fluent mesh statistics

Type	Number
Cells	329 024
Faces	668 208
Nodes	60 094
Face zones	22

y-direction). For this simulation the energy equation is enabled and the laminar model is used. In the materials section the working fluid is set to liquid water and the solid materials are defined to be aluminum. The properties of all the materials are defined at the starting temperature of 20 °C.

Additionally, water density was defined to follow a Boussinesq approximation. Using the Boussinesq model for natural convection flows faster convergence is obtained than having the fluid density as only a function of temperature. The model assumes that fluid density stays constant in all solved equations except in the momentum equation where it is replaced by the approximation as

$$\rho = \rho_0(1 - \beta \Delta T) \quad (5)$$

where ρ [kgm^{-3}] is the new density, ρ_0 [kgm^{-3}] – the constant density, β – the thermal expansion coefficient of water ($207 \cdot 10^{-6} \text{ K}^{-1}$). The Boussinesq approximation is accurate as long as $\beta(T - T_0) \ll 1$ which applies for all cases during these simulations [11].

Next, the cell zones and boundary conditions are defined. Under the Cell Zone Conditions section the interior zone is changed to a fluid and edited to contain the water defined in the above steps. In the Boundary Conditions section all named wall sections created in the meshing process appear. The side and bottom tank walls are defined as convection/conduction boundaries between the water and aluminum while the top is de-

fined as a convection boundary open to the atmosphere. Both use the heat transfer coefficient previously defined and specify a room temperature of 22 °C. The core surfaces are defined as thermal boundaries with a heat flux. The core was discretized into sixths for entry into the software as shown in fig. 5 to facilitate the power mapping onto the thermal boundaries in the model.

To increase simulation speed, aid in modeling and in mapping the power distribution to the thermal boundaries, the top and bottom surfaces were divided into ring sections corresponding to the fuel element rings. The pin power distributions [1, 12] and core surface area were used to determine the heat flux as

$$\frac{\text{Power distribution [kW]}}{\text{Surface area [m}^2\text{]}} = \text{Heat flux } \frac{\text{kW}}{\text{m}^2} \quad (6)$$

The process of dividing the core up into sections and then using the pin power distributions to calculate the heat flux for each thermal boundary was repeated for each power level level that was simulated. The heat flux values varied based on the total reactor power and core layout as calculated in Part I [1].

The Fluent analysis was carried out using the pressure implicit with splitting of operators (PISO) solution algorithm with the spatial discretization of pressure set to second order and the remainder of the solution settings left at the default values. Under-relaxation factors for the solution controls were also left at the default values. The solution was initialized to start at a temperature of either 20 °C or 24 °C and calculation activities were set up to record fluid temperature and velocity at 5 minutes intervals. After this setup the solver was run with a 1 s time step until a maximum simulation time of 3 600 s was reached.

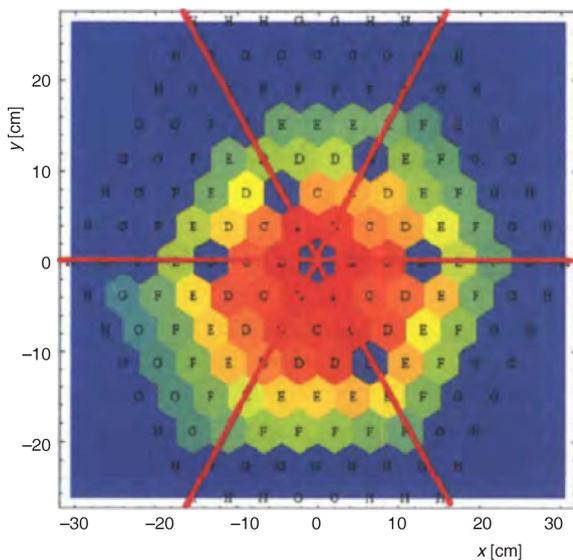


Figure 5. Discretizing the core sides prior to simulation (90 kW core) [12]

Power levels of 90, 100, 150, 300, 400, and 500 kW were simulated with the 20 °C starting temperature while only 90 kW was run using the 24 °C initial temperature. Temperatures were calculated at the same 4.5 m (14.76 ft) distance from the bottom of the reactor pool and at predetermined radial distances. The results from the 20 °C simulations are shown plotted in fig. 6, while fig. 7 shows the locations for which the temperatures were calculated.

COMPARISON OF RESULTS AND MEASUREMENTS

Three power levels were the same in these two simulations: 90, 100, and 150 kW. In fig. 8, Fluent and SolidWorks comparison of the center temperatures at the 4.5 m plane is shown.

It can be observed that SolidWorks always produced a higher temperature value. The highest temperature difference is obtained at 90 kW with a difference of 0.36 °C or 1.6%. The difference at the 100 kW level

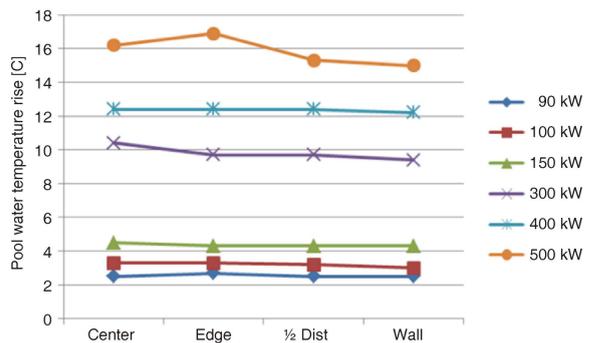


Figure 6. UUTR pool water temperature rise over 1 hour as a function of UUTR power and pool location as obtained with Ansys Fluent

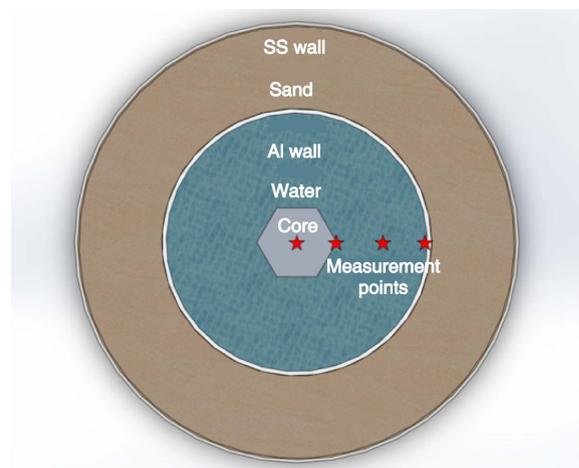


Figure 7. Radial locations of temperature measurements calculation in the UUTR

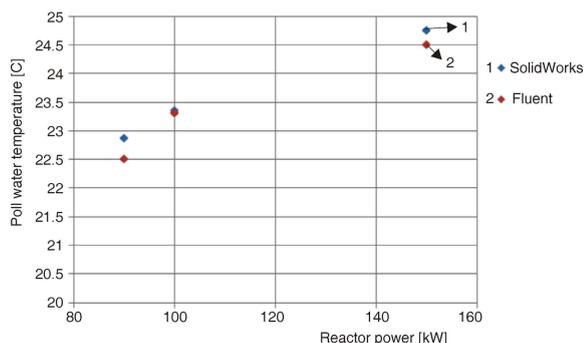


Figure 8. Comparison of SolidWorks and Fluent centerline temperatures on the 4.5 m plane

and overall variability between the two data sets is attributed to the location of the natural convection current. If the point where the results are taken is outside the convection current, the temperature is near the average fluid temperature.

Both models show the near constant upward slope of increasing temperature vs. reactor power, as expected. The Fluent model shows more variation since the core was discretized according to fuel element levels. The asymmetrical layout of the core leads to a more unstable and mobile convection column causing more variation in the temperature results. The SolidWorks model used a constant, volume type power source and thus had less water movement with a more stable, uniform convection column. The Fluent model more accurately models the actual water flow while the SolidWorks model better shows the overall average temperature.

The UUTR pool water temperature is measured with the readings taken at 4.5 m (14.76 ft) distance from the bottom of the reactor pool and at the predetermined radial distances previously shown in fig. 7. During normal 90 kW reactor operations three temperature measurements were taken: during startup, after 30 minute, runtime, and after one hour runtime. A type K thermocouple attached to an Omega TrueRMS Super Meter was used for collecting all the data. The thermocouple was lowered into the desired position, allowed to acclimate for one minute and then the detected temperature range was recorded.

The results from the pool water measurements over one hour are summarized in tab. 5 (Note: the first trial was only conducted in the central location, C. L.) and are compared with the corresponding temperatures from both the 20 °C and the 24 °C 90 kW simulations.

All of the measurements show some variation between readings. This variation can be attributed to four reasons. First, and affecting both the simulations and UUTR is the location of the convection current. The flow around the core begins to exhibit vortex shedding once it has developed. This moves the convection current of heated water back and forth around the middle area of the reactor pool. If the thermal column is away from the point of measurement at the time readings are taken the temperature will be lower since the surroundings were measured and not the heated water directly from the core.

Table 5. Comparison of simulation and temperature measurements at 90 kW

	C. L. point temperature rise [°C]	Core edge temperature rise [°C]	1/2 distance temperature rise [°C]	Tank wall temperature rise [°C]
UUTR 3/29/12	2.5 0.5	–	–	–
UUTR 4/5/12	2.9 0.1	2.4 0.2	2.8 0.1	2.7 0.1
UUTR 5/29/12	2.8 0.3	3.0 0.2	2.7 0.1	2.7 0.1
UUTR 7/28/12	3.2 0.4	3.1 0.2	2.8 0.1	2.7 0.1
UUTR 8/8/12	3.0 0.3	2.9 0.2	2.8 0.1	2.8 0.1
Fluent [20 °C]	2.54	2.73	2.51	2.50
SolidWorks [20 °C]	2.86	2.90	2.82	2.80
Fluent [24 °C]	3.11	2.98	2.97	2.73
SolidWorks [24 °C]	2.96	2.95	2.90	2.68

The temperature is also affected by the operators measuring and operating the UUTR. At UUTR the reactor power is manually controlled by the operator. While each trial was conducted at 90 kW it is normal to expect a variation of the power as 90 ± 1 kW. Over long periods of time small changes in the power can create slight differences in the temperature between the measurements. The operator also controls the rate of the reactor power is increased increase before reaching the level of 90 kW. Ideally the power would be ramped to the desired level instantaneously. However, this operation can take longer depending on current conditions and experiments conducted. Time spent ramping up to power increases tank water temperature and has an effect on the temperature measurements.

Finally, the temperature is affected by the ambient starting conditions. During the summer the reactor room temperature is higher causing the starting pool water temperature to also be higher. From freezing to 35 °C the water's heat capacity (C_p) slightly decreases, making it easier to heat. This is evidenced in the two starting temperatures (20 °C and 24 °C) of both the SolidWorks and Fluent simulations with the 90 kW, where at 24 °C simulations more closely match the measurements taken.

The simulations of UUTR at higher power levels show that the reactor undergoes the same natural convective and vortex shedding processes only the effects become more pronounced. In the 90 kW simulation the open water fluid velocity peaked at 0.0988 m/s while in the 500 kW simulation the maximum velocity has increased to 0.156 m/s. These results are similar, only more conservative, to those reported in the UUTR SAR which reports 0.115 m/s for 90 kW and 0.130 m/s for 100 kW [12]. The slower velocities can be attributed to the simpler reactor core models used in both simulations. These models did not include the coolant channels through the center of the reactor and

around each element. If these channels were included additional convection heating would occur increasing the velocities while also decreasing the blunt body effects that lead to the vortex shedding.

DEPARTURE FROM NUCLEATE BOILING RATIO

The departure from nucleate boiling ratio (DNBR) is a ratio of the critical heat flux (CHF) needed to cause departure from nucleate boiling to the actual heat flux on the fuel element. The DNBR is dependent on the coolant velocity, the pressure and the extent the fluid is below the saturation temperature. For fuel safety it is recommended that the DNBR for TRIGA reactors should not be below 1.0 [12, 13].

The first step in finding the DNBR is the calculation of the CHF. For TRIGA reactors the accepted, traditional method of estimating the CHF is using the Bernath Correlation [5, 14]

$$CHF = h_{crit} (T_{crit} - T_f) \tag{7}$$

$$h_{crit} = 6184 \frac{D_w}{D_w D_i} - 0.01863 \frac{23.53}{D_w^{0.6}} u \tag{8}$$

$$T_{crit} = 57 \ln p \frac{54P}{P} - 0.1034 - 283.7 \frac{u}{1.219} \tag{9}$$

where CHF [Wm⁻²] is the critical heat flux, h_{crit} [Wm⁻²K⁻¹] – the critical coefficient of heat transfer, T_{crit} [°C] – the critical surface temperature, T_f [°C] – the bulk fluid temperature, p [MPa] – the pressure, u [ms⁻¹] – the fluid velocity, D_w [m] – the wet hydraulic diameter, D_i [m] – the diameter of the heat source. From MCNP5 performed in part one of the feasibility study simulation it is found that the fuel element with the highest power is located in the B-ring [1]. For calculating the CHF the most conservative fuel pin sub-geometry for this location is used [12]. All the values used to calculate the CHF are presented in tab. 6 and are shown for a starting fluid temperature of 20 °C. The CHF as given by Bernath's Correlation is displayed in fig. 9 for reactor powers of 90, 100, 150, 300,

Table 6. Values used in calculating the critical heat flux

	90 kW	100 kW	150 kW	300 kW	400 kW	500 kW
h_{crit} [Wm ⁻² K ⁻¹]	3.847	4.001	4.329	4.467	4.720	4.935
T_{crit} [°C]	150.49	150.48	150.45	150.44	150.42	150.40
T_f [°C]	20.0					
p [MPa]	0.176					
u [ms ⁻¹]	0.115	0.130	0.161	0.174	0.198	0.218
D_w [m]	0.00499					
D_i [m]	0.112					

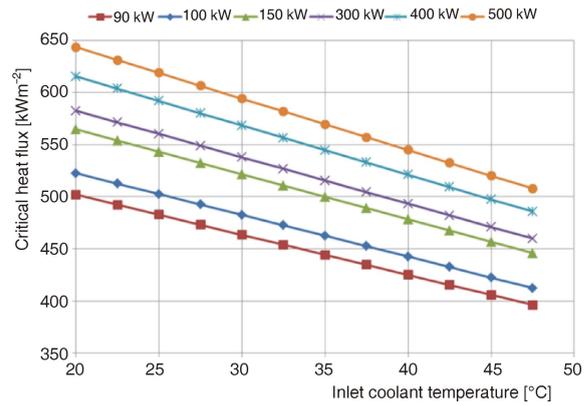


Figure 9. CHF calculated using Bernath's correlation as a function of coolant inlet temperature for various reactor powers

400, and 500 kW. The CHF values shown are based on inlet temperature and correspond to the moment when DNBR is expected to occur.

The actual heat flux under the same conditions is then required. The ratio of these two values forms the departure from nucleate boiling ratio. The program for the analysis of reactor transients (PARET) code from Argonne National Laboratory is used to model the heat flux for UUTR for both the hottest and nominal fuel elements in steady-state operation. Table 7 shows the main input parameters used in the PARET code.

For the DNBR the maximum surface heat flux from the PARET calculation is used along with the CHF found using Bernath's correlation. The UUTR Technical Specifications require that the pool water

Table 7. Comparison of simulation and temperature measurements at 90 kW

Variable	90 kW	100 kW	150 kW	300 kW	400 kW	500 kW
Inlet moderator temperature [°C]	20.6					
Fuel pin radius [m]	1.792 10 ⁻²					
Thermal conductivity UZrH [Wm ⁻¹ K ⁻¹]	18.0					
Thermal conductivity air gap [Wm ⁻¹ K ⁻¹]	0.199					
Thermal conductivity S.S. clad [Wm ⁻¹ K ⁻¹]	16.8					
Axial neutron flux ratio	Data for each power level provided from [1, 12]					
Moderator inlet mass velocity [kgs ⁻¹ m ⁻²]	115	130	158	179	202	218

Table 8. UUTR heat fluxes and DNBR at 35 °C

Power level	Maximum heat flux [Wm ⁻²]	Critical heat flux [kWm ⁻²]	DNBR
90 kW	44 419	444.2	1000
100 kW	49 057	462.5	943
150 kW	74 032	499.8	675
300 kW	148 064	515.6	348
400 kW	197 419	544.7	276
500 kW	246 774	569.5	231

temperature be no higher than 35 °C [12] so the CHF is read at this temperature by finding the point at which each power level intersects the 35 °C line in fig. 9. The resulting heat fluxes and DNBR are shown in tab. 8.

The DNBR calculation shows that using Bernath's Correlation and maximum heat flux data from the PARET code that a boiling crisis will not occur even if the UUTR's power is upgraded to 500 kW. Currently, during 90 kW operations there is no boiling on any of the reactor fuel elements. From the PARET calculations it is also shown that boiling would begin at 131.9 °C or when UUTR is at 210 kW. However, even though boiling has started it remains in a nucleate regime and does not approach the CHF, remaining safe up through 500 kW.

HIGHER POWER UUTR COOLING SYSTEM DESIGN

Without any core or reactor modifications it is found based on neutronics analysis only that the maximum UUTR power level is 150 kW [1]. UUTR does not operate at this level as only 2.1 hours of runtime would be available if started at 20 °C before the technical specification regarding the temperature limit would be reached. This limit is set to protect the integrity of the deionizing resin beds. At higher temperatures the resin integrity begins to degrade leaching the removed contaminants back into the water.

It is suggested to extend the operating time and increase the operating power of UUTR for minimal cost new deionizing beds be installed and the Technical Specifications for UUTR (TS) be amended to have a higher water temperature limit. Recently, higher grade, more temperature resistant deionizing resins have become widely available for less cost. The ResinTech® MBD-10 nuclear grade, mixed bed resin is one example. It functions up to 60 °C if rechargeable and up to 80 °C if single use and meets all other water quality requirements [15, 16]. The increase in temperature limit from using the new deionizers would allow 3.5 additional hours of operating time at 90 kW or 2.17 hours additional time if upgraded and operating at 150 kW. Figure 10, based off of the previous simulations conducted, illustrates the theoretical runtimes for various power levels when starting at 20 °C.

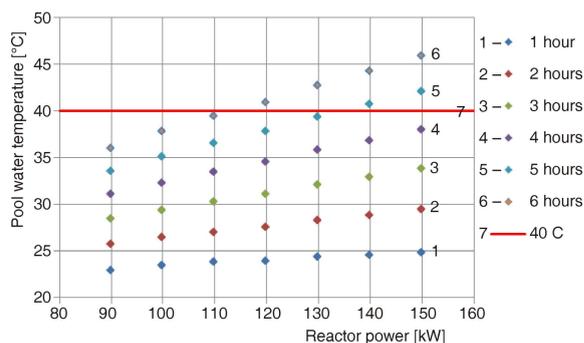


Figure 10. UUTR pool water temperature as a function of UUTR power after TS temperature increase as obtained with SolidWorks flow simulation

This upgrade only retrofits UUTR for higher temperature operations and does not provide any additional cooling capabilities. UUTR would still be cooled through natural heat transfer to the surroundings. Because of this, an extra shutdown time would be required following higher temperature runs to allow for ambient cooling.

To enable longer, more frequent UUTR runtimes with an increase in power level a new cooling system needs to be considered. A dual-loop system connected through a heat exchanger and routed to an outdoor cooling tower is commonly used to cool research reactors and would be best suited for UUTR. For the sole purpose of estimating the costs of a cooling system UUTR is assumed to be operating at 250 kW and the system components are sized appropriately. Only the size of the heat exchanger, cooling tower, pumps and pipes change as needed. Based only off of a 20 °C temperature difference in the cooling water from the heat exchanger and assuming no losses, a flow rate of 47.4 gram per meter in the primary loop is required to cool a 250 kW reactor. To supply this flow rate and provide extra head for the losses occurring in the pipes, resin beds and heat exchanger a 5 horse-power pump is needed. Stainless steel, schedule 40 pipe is strong, durable and recommended for all piping connections. A plate heat exchanger is recommended for heat transfer between the two cooling loops together.

The secondary loop supplies cooled water to the heat exchanger from the cooling tower. A 300 kW counter-flow cooling tower is recommended as it allows for maximum heat transfer from the heat exchanger, has a more freeze resistant design for winter use and is commonly available in preassembled units. This tower requires 110 gram per meter of water flow which could be provided by a second 5 horse-power pump. Stainless steel piping would be used in secondary loop as well. Because the cooling tower is open to the atmosphere, a water treatment system is recommended for the secondary loop to prevent fouling.

CONCLUSIONS

The results from both feasibility studies, one based on neutronics (Part I of this paper), and the other based on thermal-hydraulics and heat transfer, form a basis for a power upgrade to UUTR. Figure 11 summarizes the relationship between UUTR power, neutron flux, runtime and cost assuming the UUTR Technical Specifications will be modified to allow for a higher pool water temperature limit. While the low-cost upgrade allows higher temperatures to be reached, the current UUTR core configuration is only capable of 150 kW [1]. To operate at values beyond 150 kW the additional core upgrade outlined in [1] would be needed. Figure 12 shows an estimated cost (as of 2012) of both of the upgrades, the increased power level they provide and the corresponding total neutron flux. The estimated cost is based on the projected costs of the UUTR core upgrade described in Part I [1] and from the costs of the cooling system components. The total neutron flux represents the highest value obtained from MCNP5 simulations [1]. After 250 kW the price continues to rise because components such as the cooling tower, heat exchanger, pipes and pumps increase proportionally to the power level.

The thermodynamic calculations and CFD simulations presented help to gain a better understanding of heat flow in UUTR. These analyses, along with the

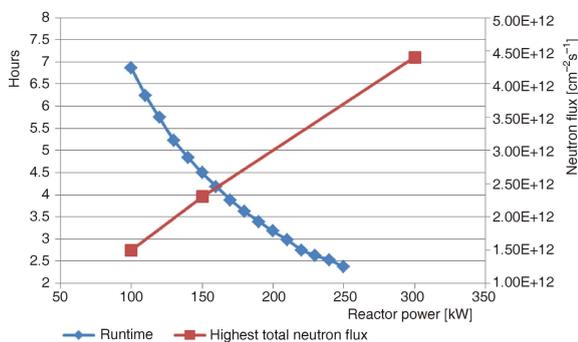


Figure 11. UUTR simulated runtime and total neutron flux at varying power levels with pool temperature above the current technical specification limit

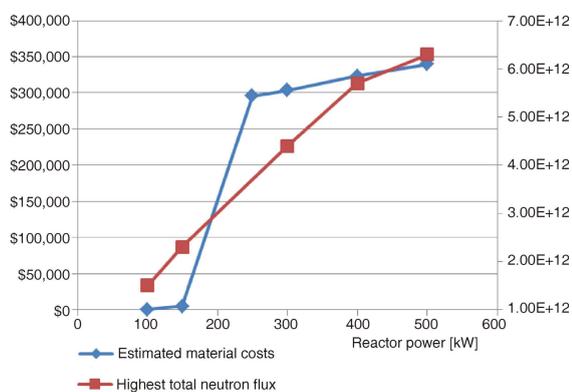


Figure 12. UUTR estimated power upgrade costs and total neutron flux at varying power levels

neutronics simulations from Part I of this study [1] and in [17], were necessary to gauge the feasibility of a power upgrade for UUTR. The general conclusion is that the UUTR can be upgraded to a higher power level should such upgrade be required for the use of a reactor beyond current use for training, education and yet various and numerous experiments performed weekly.

AUTHOR CONTRIBUTIONS

P. Babitz developed all numerical models and performed integrated analysis and cost estimates. D. Choe assisted P. Babitz in reviewing the simulation results and validity of data interpretation. T. Jevremovic supervised the research that resulted in this paper.

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**ПОВЕЋАЊЕ СНАГЕ ИСТРАЖИВАЧКОГ РЕКТОРА НА
УНИВЕРЗИТЕТУ У ЈУТИ
Део II: Анализа термохидраулике и преноса топлоте према захтевима
расхладног система и пројекта**

Термохидраулички параметри истраживачког TRIGA реактора на Универзитету у Јути прорачунати су коришћењем нумеричких програма SolidWorks Flow, Ansys, Fluent и PARET-ANL. Развијени су нумерички модели за рад реактора на максималној снази од 90 kW, а и за снаге вишег нивоа, да би се анализирали термохидраулика и пренос топлоте у циљу добијања подлога за пројектовање рада реактора на вишим снагама и процену трошкова. Прорачуни су потврдили да природна конвекција постаје значајна на вишим снагама реактора када се уочава и појава вртложног изливања. Анализа појаве кључања показала је да се она јавља на снази реактора од око 210 kW, али да не постаје критична све до снаге реактора од 500 kW. На основу овога закључено је да су могућа два начина повећања снаге. Имајући у виду и налазе неутронских анализа реактора, који су приказани у првом делу овог рада, закључује се да је повећање снаге реактора веома изгледно, мада условљено потребом за таквим повећањем и пратећим инвестицијама.

Кључне речи: TRIGA, истраживачки реактор, пренос топлоте, FLUENT, SolidWorks, PARET