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Thermal-Hydraulic Analysis of the UCI TRIGA Reactor

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911201
Revision 1

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Thermal-Hydraulic Analysis of the University of California – Irvine TRIGA® Reactor

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Prepared under
Contract No. 00096970
for the U.S. Department of Energy

GA PROJECT 39364

GENERAL ATOMICS

GENERAL ATOMICS

GA 1485 (REV. 08/06E)

ISSUE/RELEASE SUMMARY

☒ R & D ☐ DV&S ☐ DESIGN ☐ T&E ☐ NA	APPVL LEVEL	DISC	QA LEVEL	SYS	DOC. TYPE	PROJECT	DOCUMENT NO.	REV
	2	N	IB	N/A	RGE	39364	911201	0

TITLE:

Thermal-Hydraulic Analysis of the University of California – Irvine TRIGA® Reactor

CM APPROVAL/ DATE	REV	PREPARED BY	APPROVAL(S) ENGINEERING	QA	PROJECT	REVISION DESCRIPTION/ W.O. NO.
	0	J. Crozier	J. Bolin	K. Partain	T. Veca	Initial Issue A39364.0310

CONTINUE ON GA FORM 1485-1

NEXT INDENTURED DOCUMENT(S)

COMPUTER PROGRAM

PIN(S)

☐ GA PROPRIETARY INFORMATION

☒ NO GA PROPRIETARY INFORMATION

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LIST OF ABBREVIATIONS/ACRONYMS

APF	Axial Peaking Factor
ARI	All Rods In
ARO	All Rods Out
BOL	Beginning of Life
CHF	Critical Heat Flux
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
GA	General Atomics
kW	kilowatt
LEU	Low Enriched Uranium
MNRC	McClellan Nuclear Radiation Center
MW	Megawatt
PTS	Pneumatic Transfer System
RPF	Radial Peaking Factor
SAR	Safety Analysis Report
STAT	A GA proprietary code for heat transfer with natural convection
TAC2D	A general purpose 2D heat transfer code
TRIGA®	Training Research Isotope General Atomics
UCI	University of California- Irvine

1 INTRODUCTION

This report provides an overview of the thermal-hydraulic characteristics of the University of California – Irvine (UCI) TRIGA® Reactor.

2 REACTOR DESCRIPTION

2.1 Reactor Facility

As shown in Table 2-1, the table provides a comparison of the key thermal-hydraulic features of the [redacted], along with a comparison of these key features for various power levels. As discussed in Sections 2.3, the maximum fuel temperature is safely below the limiting fuel temperature; hence, it shows that the UCI reactor facility can be operated safely with [redacted].

The computations produced results for the hot element maximum fuel temperature, hot element maximum clad temperature, core average temperature, hot element outlet flow temperature, average element outlet flow temperature, hot element flow rate, and average element flow rate.

Table 2-1 LEU Design Data and Thermal-Hydraulic Reactor Parameters

DESIGN DATA	
Number of Fuel Rods	[redacted]
Fuel Type	$UZrH_x$
Uranium Enrichment, %	19.79
Zirconium Rod Outer Diameter, mm	[redacted]
Fuel Meat Outer Diameter, mm	[redacted]
Fuel Meat Length, mm	[redacted]
Clad Thickness, mm	[redacted]
Clad Material	[redacted]
THERMAL-HYDRAULIC REACTOR PARAMETERS	
Reactor Steady State Operation, kW	250
Limited Safety System Setting, kW	275
Number of fuel elements	[redacted]
Diameter, mm (in.)	[redacted]
Length (heated), mm (in.)	[redacted]
Core total flow area, mm^2 (ft^2)	36,362 (0.3914)
Core total wetted perimeter, mm (ft.)	9416 (30.89)
Flow channel hydraulic diameter, mm (ft.)	15.45 (0.05068)
Core total heat transfer surface, m^2 (ft^2)	3.587 (38.61)
"Radial" peaking factor (rpf)	1.446
Axial peaking factor (apf)	1.352
Hot rod factor (rpf x apf)	1.955
Inlet coolant temperature, °C (°F)	25. (77)
Coolant saturation temperature, °C (°F)	114. (237)
Peak fuel temperature in average fuel element, °C (°F) [1]	214 (418)
Maximum wall temperature in hottest element, °C (°F) [1]	123 (254)
Peak fuel temperature in hottest fuel element, °C (°F) [1]	253 (488)
Core average fuel temperature, °C (°F) [1]	164 (327)
Minimum DNB ratio at 0.275 MW	7.27
Minimum DNB ratio at 0.30 MW	6.67

2.2 Reactor Core

This section provides a detailed description of the components and structures in the reactor core. The UCI reactor is a primarily homogeneous, light water moderated and cooled, tank-type reactor fueled with a full core of LEU (19.79% enriched $UZrH_x$) TRIGA® fuel in a cylindrical lattice configuration. The fuel elements are supported by a 19.05mm thick aluminum grid plate.

[1] The thermal-hydraulic parameters shown are for the reactor operating at 300 kW.

The UCI core configuration is a circular arrangement of core elements consisting of fuel, control, reflector, and testing elements, and it is shown in Figure 2-1. The grid-plate consists of 7 circular rings of elements, A for the central ring and G for the outer ring. The individual locations for the elements are numerically ordered in a clockwise direction. The design of the grid-plate will allow it to accept up to a total of [redacted] A total of [redacted] reflector elements, located on the periphery of the grid plate, are used for reflection of neutrons.

[redacted]

Figure 2-1 UCI Core Configuration

2.2.1 Fuel Elements

The LEU (8.5% wt) TRIGA® fuel installed in the UCI core consists of [redacted]. Figure 2-2 shows a detailed illustration of the fuel, graphite, and zirconium rod regions. The zirconium rod is manufactured to a diameter of [redacted], but the hole for the zirconium rod has a diameter of [redacted].

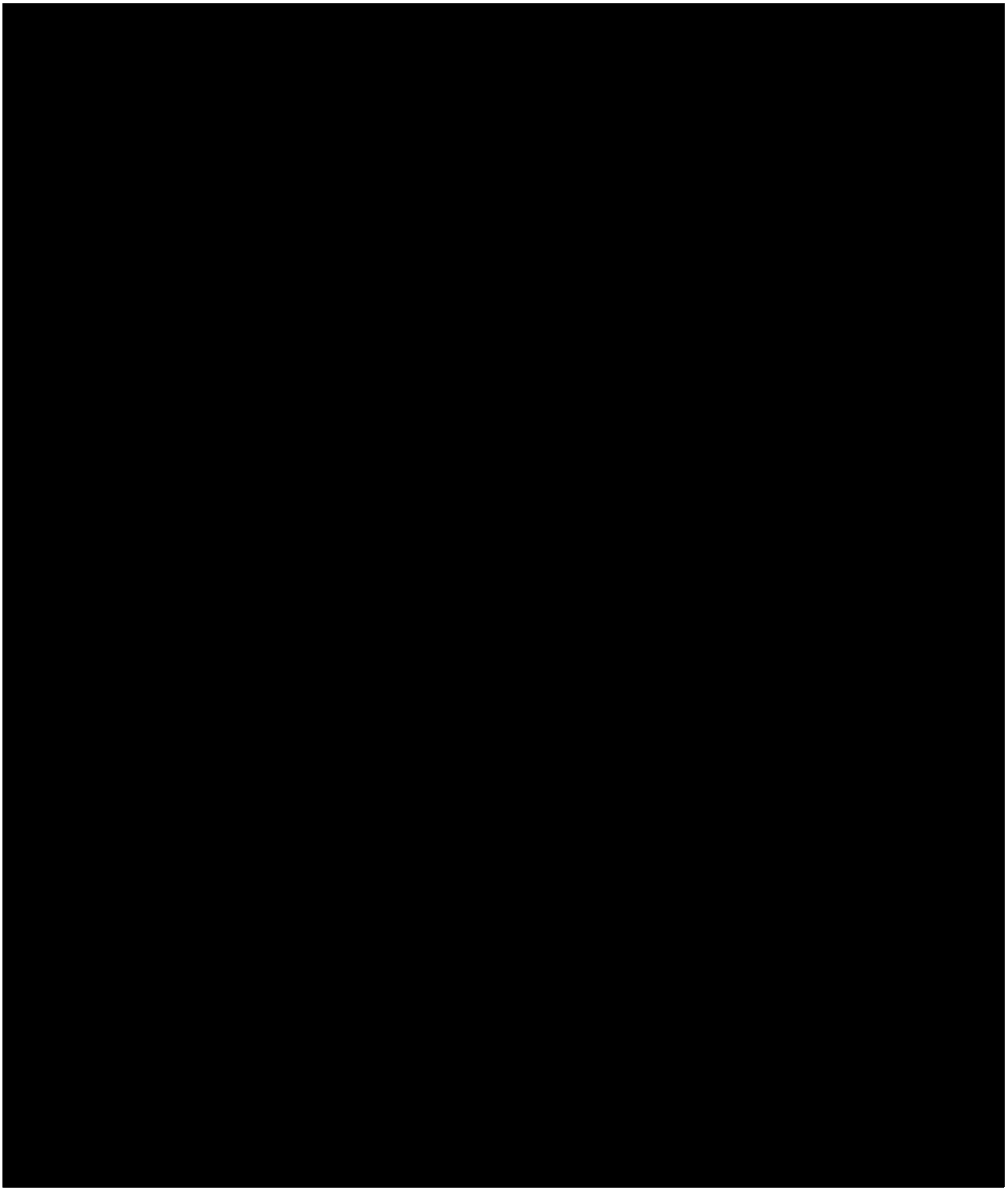

Figure 2-2 Fuel Element Details

An aluminum grid plate (19.1 mm thick) is used to support the fuel elements on the bottom of the reactor. In addition, an aluminum grid plate (16.1 mm thick) is located at the top of the core to provide lateral restraint for the core components. The reactor is controlled by poison rods supported by a bridge mounted at the top of the biological shield.

The core is positioned [redacted] above the bottom of the reactor tank and is supported by an aluminum core support structure, which is bolted to the floor of the reactor tank.

The geometries, materials, and fissile loadings of the current fuel elements are summarized in Table 2-2.

Table 2-2 Description of the Averaged UCI Fuel Elements

Design Data	
Number of Fuel Elements Full Load	[redacted]
Fuel Type	$UZrH_x$
Enrichment, %	19.79
Uranium Density, g/cm^3 Wt-%	 1.70 8.5
^{235}U per Fuel Element, g	[redacted]
Zirconium Rod Outer Diameter, mm.	[redacted]
Fuel Meat Outer Diameter, mm.	[redacted]
Fuel Meat Length, mm.	[redacted]
Cladding Thickness, mm.	[redacted]
Cladding Material	[redacted]

2.2.2 Control Rods

The UCI reactivity control system consists of four standard TRIGA® control rods; one fuel-followed shim rod, one fuel-followed regulating rod, one air-followed adjustable transient rod, and one air-followed fast transient rod as shown in Figure 2-1. All four control rods are supported from the bridge structure at the top of the biological shield.

2.2.3 Neutron Reflector

The primary reflector for UCI reactor consists of nuclear-grade graphite designed in a ring shaped block around the core. The graphite block is placed in a leak-tight, welded, aluminum container. It is [redacted] thick radially with an inside diameter of [redacted] and [redacted] high.

2.2.4 Reactor Materials

Table 2-3 presents the material composition of components other than the fuel used in the computational models.

Table 2-3 Material Composition Used in Non-Fuel Regions

Material	Nuclide[2]	Nuc. Den. (atoms/b-cm)	Physical Density (g/cc)
[redacted] (clad)	Cr-50 Cr-52 Cr-53 Fe-56 Ni-58 Mn-55	0.000778 0.015003 0.001701 0.056730 0.007939 0.001697	7.98
Graphite (reflector in fuel) (reflector blocks)	C	 0.087745 0.078719	 1.75 1.57
Zirconium (rod w/ 60 ppm Hf)	Zr	0.034790	5.27
6061 Al (upper grid plate, lower grid plate, and control rod clad)	Al-27 Fe-56	0.058693 0.000502	2.70
90% B_4C (control rod)	B-10 B-11 C	0.020950 0.084310 0.026320	2.30
Water			1.0
Air			0.000123

2.2.5 Power Peaking Results

Power peaking in the BOL core is analyzed on the basis of the following component values:

1. $\overline{P}_{rod} / \overline{P}_{core}$: rod power factor, the power generation in a fuel rod (element) relative to the core averaged rod power generation.

2. $(\hat{P} / \overline{P})_{axial}$: axial peak-to-average power ratio within a fuel rod (element).

3. $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$: rod peaking factor, the peak-to-average power in a radial plane within a fuel rod (element).

Since maximum fuel temperature is the limiting operational parameter for the core, the peaking factor of greatest importance for steady-state operation is $\overline{P}_{rod} / \overline{P}_{core}$. The maximum value of this factor for the hottest rod, the hot-rod factor, [$(\overline{P}_{rod} / \overline{P}_{core})_{max}$ = hot-rod factor], determines the power generation in the hottest fuel element. The hot rod power factor is calculated to be 1.446, which can be found in fuel element C6. When combined with the axial power distribution, the hot-rod factor is used in the thermal analysis for determination of the maximum fuel

[2] Other isotopes were reviewed in the process, but they were determined to have an insignificant impact on the results.

temperature. The axial power peaking factor, $(\hat{P}/\overline{P})_{axial}$, is calculated to be 1.352, and at BOL it is relatively independent of fuel temperatures or radial position in the core. The radial power distribution within the element has only a small effect on the peak temperature, but it is also used in the steady-state thermal analysis.

The rod peaking factor, $(\hat{P}_{rod}/\overline{P}_{rod})_{\text{radial}}$, is of importance in the transient analysis for calculating maximum fuel temperatures in the time range where the heat transfer is not yet significant, and was calculated to not exceed 1.717. It is used in the safety analysis to calculate the peak fuel temperature under adiabatic conditions, where temperature distribution is the same as power distribution.

The reactor radial power peaking map for the UCI core is shown in Figure 2-3 (Ref. 1).

[redacted]

Figure 2-3 Hot Critical - Core Power Map

2.3 Thermal-Hydraulic Analysis

2.3.1 Analysis of Steady State Operation – UCI Core

A thermal evaluation was made for the UCI core operating with cooling from natural convection water flow through the core. In this study, the predicted steady state thermal-hydraulic performance of the UCI core with [redacted] was determined for the reactor operating at 0.1, 0.25, 0.275, and 0.3 MWs and a reactor pool temperature of 25°C.

The RELAP5-Mod 3.3 (RELAP5) computer code calculates the thermal hydraulics and fuel element temperatures in nuclear reactors (Ref. 2). In this application it is used to calculate the steady state natural convection flow through a vertical water coolant channel adjacent to fuel element heat source. The code also calculates radial heat fluxes through and from the fuel element to the natural convection flow at the axial discretized points of the model. Accordingly, the code determines the clad, fuel and zirconium center rod temperatures within the fuel element and the axial temperature distribution of the natural convection flow.

The RELAP5 thermal analysis for the UCI TRIGA is based on a G-ring configuration. The RELAP5 model contains two separate fuel elements and their corresponding flow channels. An "average rod" represents the [redacted] of the entire core. A "maximum rod" represents the single hottest fuel element. Since the model for the maximum rod is a single isolated element, there is neither thermal or flow interaction with adjacent elements. However, the static pressures of the flow for the maximum rod and average rod are equal at the core inlet and at the core outlet. This static pressure is also equal to the respective pressures of the cold water column outside of the core.

The flow area for the hottest fuel element is taken as twice the flow area formed by the elements with one of those elements being in the "B" ring. For the "average rod" the flow area, the wetted perimeter and the heat source in the model equal the values for all fuel elements [redacted] in the core. In this manner, the RELAP5 output gives the total flow rate, the total power, and the average temperature for the entire core. For the "maximum rod" the flow area, the wetted perimeter, and the heat source in the model are for a single fuel element.

The model of a single element contains the heated section and the unheated sections below and above the heated section. A single inlet flow loss coefficient represents the flow losses associated with the bottom grid plate. This inlet loss coefficient is estimated by first calculating the individual contraction and expansion losses from the pool as it enters the bottom area of the core. The sum of these losses is then converted to the single inlet loss coefficient based on the element flow area. A similar calculation is performed for the exit loss coefficient that represents the losses of the flow expansions and contractions from the top end of the element, through the top grid plate and into the pool region. See Reference 3, Appendix A, for details of this type of calculation.

Figure 2-4 illustrates the RELAP5 model used for the UCI thermal-hydraulics analyses. The figure shows the separate average and maximum fuel rod heated sections ("heat structures") and associated flow channels ("pipes"). Branches represent the lower and upper unheated sections of the fuel elements. "Junctions" prior to the inlet of the lower branches and following the outlet of the upper branches provide the respective inlet and outlet flow losses. A "pipe" volume parallel to the average and maximum flow channels ("pipes") models the cold water column adjacent to the core. A "pipe" above these components represents the tank water volume above the core. A time-dependent-volume ("tmdpvol") with an associated "junction" at the top of the diagram fixes the ambient pressure above the pool, and serves as a flow sink for the reactor water flows.

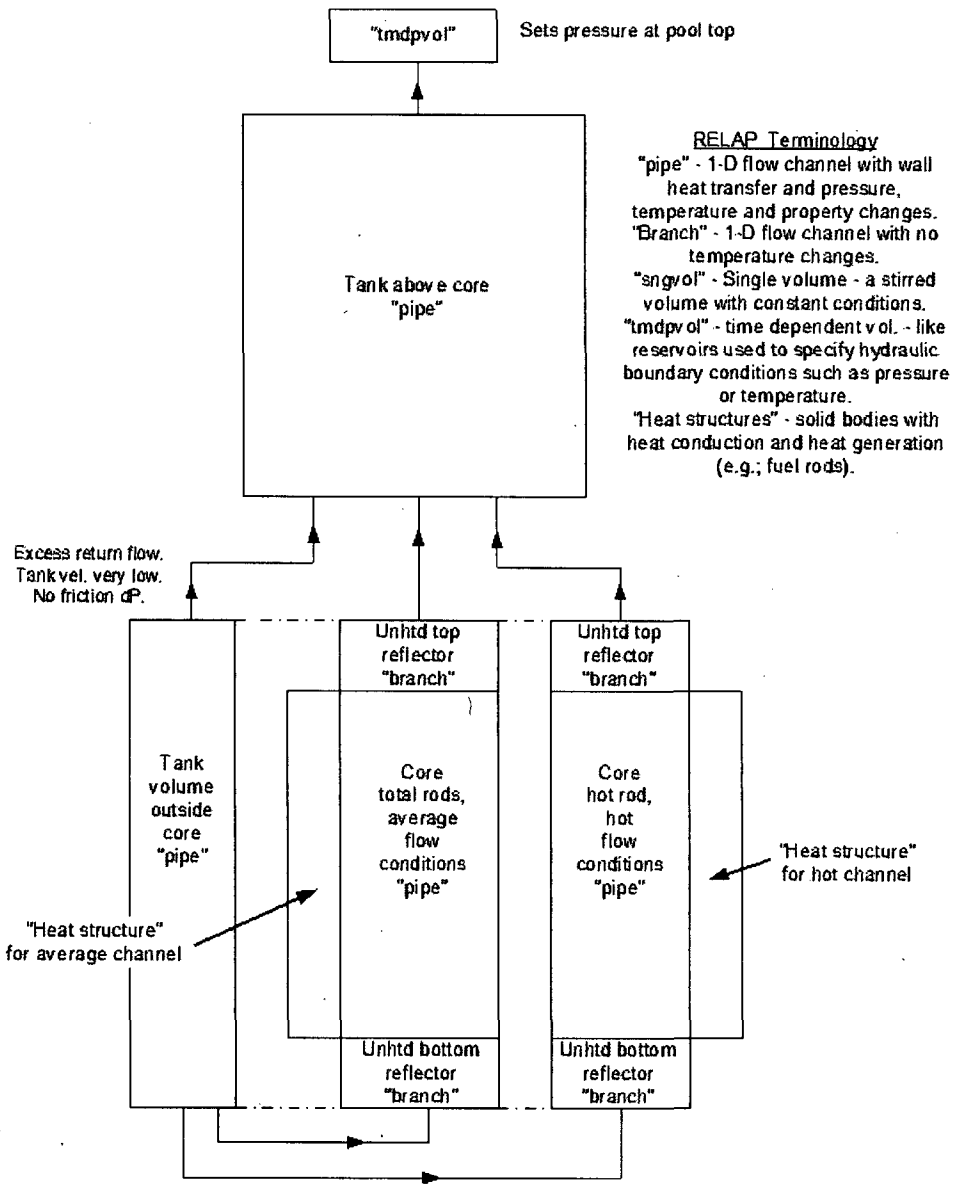

Figure 2-4 RELAP5 Natural Convection (No Primary Coolant Flow) Block Diagram

In the RELAP5 model, an assumption is made for the hot rod that there is no cross flow between the flow channel of analysis and any adjacent channels. (The average rod implicitly assumes that it is surrounded by similar rods.) A pressure difference between the hot rod channel and a colder channel axially along the channels would provide a cross flow. However, the pressures at the core inlet and at the core outlet are equal for the two channels. Thus, there is likely little difference in pressure between the two channels traversing from the bottom of the core to the top of the core. Hence, if any, only a small cross flow would be expected. A look at the overall buoyancy/friction pressure changes in channels adjacent to the hot channel indicates the cross flow would be from the cold to the hot channel. The hot channel flow rate would increase. A cross flow in this direction would decrease the hot channel density (buoyancy) and diminish the effect of increased hot channel flow rate.

In a RELAP5 analysis of the McClellan Nuclear Radiation Center (MNRC) 2 MW reactor, Jensen and Newell (Ref. 4) state the following. "The RELAP5 code provides a means for estimating the effects of cross flow between the hot and average channels. The cross flow effect is expected to be very small, and it is impossible to assess the accuracy of computed cross flows. Scoping calculations with RELAP5 showed cross flow to have no effect on fuel temperature and to increase slightly the critical heat flux ratio. Thus, cross flow is conservatively neglected in this analysis."

These conclusions are supported in a STAT-RELAP5 comparison study (Ref. 3). That study shows that fuel temperatures are little affected by the channel flow rate because the channel coolant is in a condition of sub-cooled nucleate boiling. The bulk flow saturation pressure mostly determines the bulk flow temperature.

There will be some change in the DNBR because there is a velocity (mass flux) effect in the DNB correlations. Calculations have shown that a 20% increase in the channel flow rate for 1 MW reactors produces approximately a 2% increase in the wall heat flux (reactor power) to re-achieve a DNBR = 1.0.

As noted in a Sandia report by Rao (Ref. 6), various experiments have revealed that cross flow is negligible for tightly packed geometries such as UCI. The references for these experiments are Becker, et al., 1969 (Ref. 7), Silvestri, et al., 1966 (Ref. 8), and Gaspari, et al., 1974 (Ref. 9). The Sandia report further states that the ultimate effect of cross flow is to increase the DNBR, so neglecting it in a sub-channel approach would result in a conservative estimate of the DNBR. As a result, and to be more conservative, no cross flow is considered in this analysis.

The reactor power for predicting the critical heat flux (CHF), or departure from nucleate boiling (DNB), is calculated using the Bernath CHF correlation (Ref. 10). The Bernath correlation has historically been used for TRIGA reactor DNB predictions. In the current UCI DNB analysis, the RELAP5 code provides the thermal-hydraulics conditions needed for input to the Bernath correlation. A recent Groeneveld, et al., 2006 CHF correlation (Ref. 11) also appears applicable

for TRIGA DNB calculations. However, the Bernath correlation gives a lower (more conservative) DNB reactor power than the Groeneveld correlation. The Bernath correlation is used in this analysis.

2.3.1.1 RELAP5 Code Analysis

Input to the RELAP5 program includes the following:

1. Full geometry of the selected fuel elements and flow channel;
2. Radial and axial heat source distribution within the fuel;
3. Discretized axial spacing of the flow channel and the fuel element, and radial spacing in the fuel element;
4. Pool height above the core;
5. Inlet and exit pressure loss coefficients;
6. Inlet water temperature.

The fuel element geometry and hydraulic data for the UCI core RELAP5 model are given in Table 2-4.

Table 2-4 RELAP5 Input for Reactor and Core Geometry and Heat Transfer, UCI Core

Core and Reactor Geometry	
Unheated core length at inlet, mm	[redacted]
Unheated core length at outlet, mm	[redacted]
Distance from top of pool surface to top of core, mm	[redacted]
Hydraulic Data	
Inlet pressure loss coefficient	2.26
Exit pressure loss coefficient	0.63
Ambient pressure at pool surface, MPa	0.101

As indicated above in Section 2.3.1, the UCI core RELAP5 thermal-hydraulic model contains an average rod and its flow channel and a maximum powered rod and its channel. Table 2-5 provides the RELAP5 hydraulic characteristics for the maximum powered fuel element and its associated flow channel. The flow area and wetted perimeter for the average channel is for the entire core of fuel elements (80).

The maximum powered fuel element is located in a triangular arrangement of fuel elements. The equilateral triangle joins the centers of three fuel elements with a pitch of [redacted] and a gap between fuel elements o[redacted]. The area between these three fuel elements is doubled to represent a full fuel element because the triangular arrangement only represents the heated surface and heat generation of ½ of a fuel element. This flow area is used for the UCI maximum powered fuel element.

Table 2-5 Hydraulic Flow Parameters for Maximum Element, UCI

Flow area (mm^2/element)	408.68
Wetted perimeter (mm/element)	[redacted]
Hydraulic diameter (mm)	13.89
Fuel element heated length (mm)	[redacted]
Fuel element diameter (mm)	[redacted]
Fuel element heated surface area (mm^2/element)	[redacted]

The axial power distribution in the fuel section of the fuel element is has an axial peaking factor (apf) of 1.352. The radial profile of the power generated within the fuel region is shown in Figure 2-5, which is a plot of the intra-rod factor. This factor is also normalized to unity over the radial distance of the fuel region.

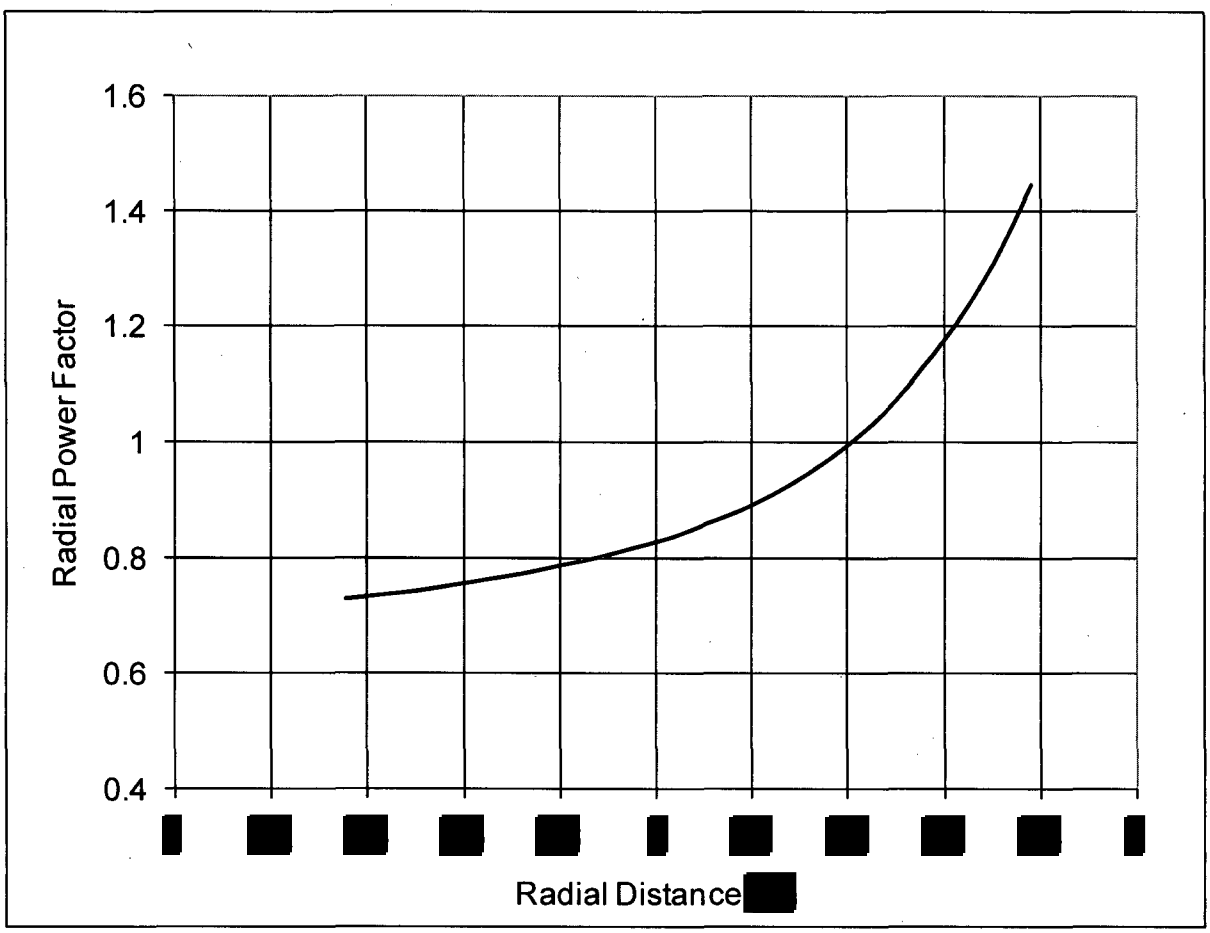

Figure 2-5 Radial Intra-Rod Power Factor versus Fuel Element Radial Distance – UCI Core

The heat generation in the fuel element is distributed axially in a piece-wise fashion of 25 intervals to represent the curve in Figure 2-5. This heat generation is distributed radially over 20 uniform intervals within the fuel meat.

A fuel-to-clad gap is also included in the RELAP5 model. The fuel-to-clad gaps change as the temperatures of the fuel meat and the stainless steel clad increase with reactor power. The relative thermal expansion between these two components decreases the gap width. Table 2-6 shows the effect of a closing gap on fuel temperatures.

Table 2-6 Fuel-to-Clad Radial Gap Widths for RELAP5 Model – UCI Core

MW	Average Element	Hot Element
0.1	0.0193 mm (0.76 mils)	0.0180 mm (0.71 mils)
0.25	0.0155 mm (0.61 mils)	0.0130 mm (0.51 mils)
0.3	0.0142 mm (0.56 mils)	0.0117 mm (0.46 mils)

The RELAP5 UCI Core model has built into it a nominal gap of 1.0 mils (0.0254 mm). As the model is run for each of the three reactor powers, the physical geometry input cards are not changed. Rather the gap gas conductivity is adjusted so that the conductance (gap gas conductivity/gap width) in the RELAP5 model equals the true conductance (k_{gas}/gap width from Table 2-6). The gap gas thermal conductivity in both cases is taken as a linear function of gap temperature.

Since the RELAP5 model has a gap width of 1.0 mils, the fuel element has the following radial dimensions:

- Fuel meat center zirconium rod hole diameter [redacted]
- Fuel meat outer diameter [redacted]
- Fuel-to-clad gap 0.0254 mm (0.001 in.)
- Clad outer diameter cold [redacted]

2.3.1.2 RELAP5 Code Results

As shown in Table 2-7, the table summarizes the thermal results for the UCI Core RELAP5 analysis for a 300 kW core. The table also includes several RELAP5 input data for completeness.

[3] The zirconium rod (diameter of 0.225 in) is homogenized over the entire volume of the hole (diameter of 0.25 in) for the zirconium rod in MCNP, so to be consistent RELAP modeled the fuel element in the same manner.

Table 2-7 Thermal Results Summary for a 300 kW Core – UCI Core

Parameter	Initial Core
Number of fuel elements	[redacted]
Diameter, mm (in.)	[redacted]
Length (heated), mm (in.)	[redacted]
Core total flow area, mm^2 (ft^2)	36,362 (0.3914)
Core total wetted perimeter, mm (ft.)	9416 (30.89)
Flow channel hydraulic diameter, mm (ft.)	15.45 (0.05068)
Core total heat transfer surface, m^2 (ft^2)	3.587 (38.61)
"Radial" peaking factor (rpf)	1.446
Axial peaking factor (apf)	1.352
Hot rod factor (rpf x apf)	1.955
Reactor pool temperature, °C (°F)	25. (77)
Coolant saturation temperature, °C (°F)	114. (237)
Exit coolant temperature (average), °C (°F)	46.33 (115.4)
Exit coolant temperature (maximum), °C (°F)	53.56 (128.4)
Coolant total mass flow, kg/sec (lb/hr)	3.37 (26,677)
Average flow velocity, mm/sec (ft/sec)	93 (0.305)
Peak fuel temperature in average fuel element, °C (°F)	214 (418)
Maximum wall temperature in hottest element, °C (°F)	123 (254)
Peak fuel temperature in hottest fuel element, °C (°F)	253 (488)
Core average fuel temperature, °C (°F)	164 (327)
Average heat flux, W/cm^2 ($BTU/hr\text{-}ft^2$)	8.36 (26,500)
Maximum heat flux in hottest element, W/cm^2 ($BTU/hr\text{-}ft^2$)	16.34 (51,800)
Minimum DNB ratio at 0.30 MW	6.67
Minimum DNB ratio at 0.275 MW	7.27

Figure 2-6 shows axial temperature profiles for the hot rod centerline, the clad mid-radius, and the bulk flow. Figure 2-7 show the fuel element radial temperature profile at the axial location of the maximum fuel temperature. Both plots are for 300 kW.

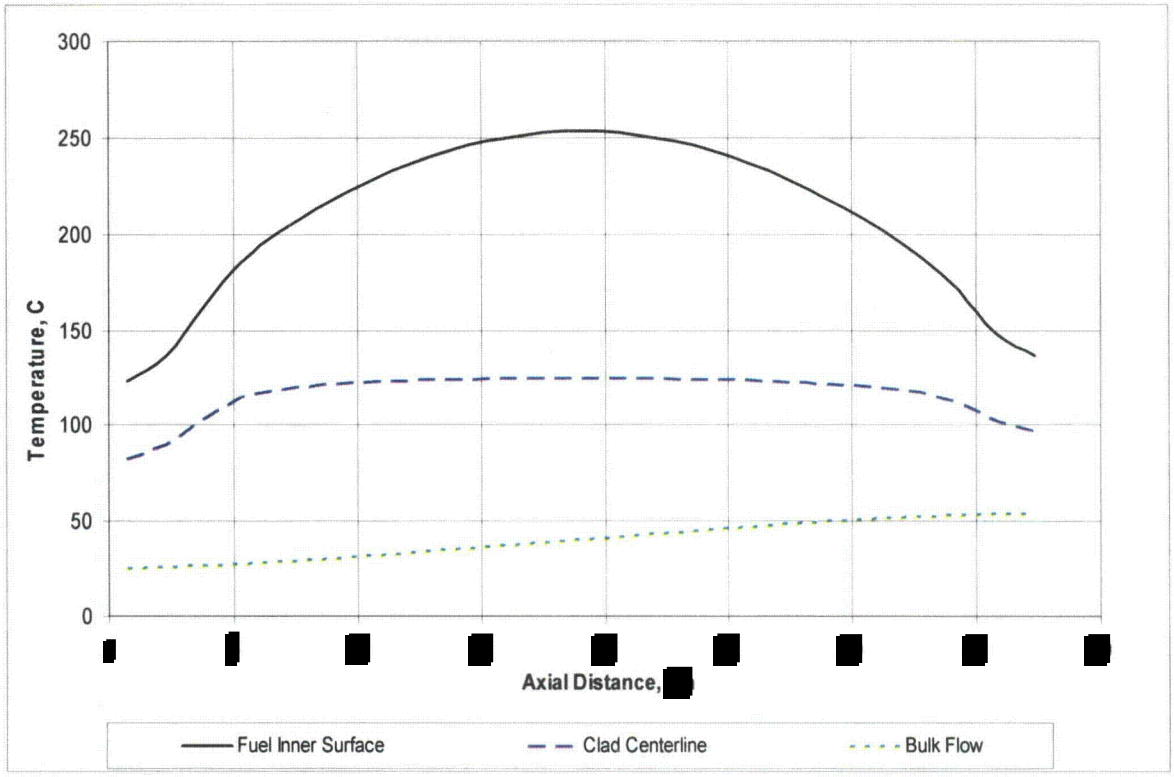

Figure 2-6 300 kW Axial Temperature Profile – UCI Core

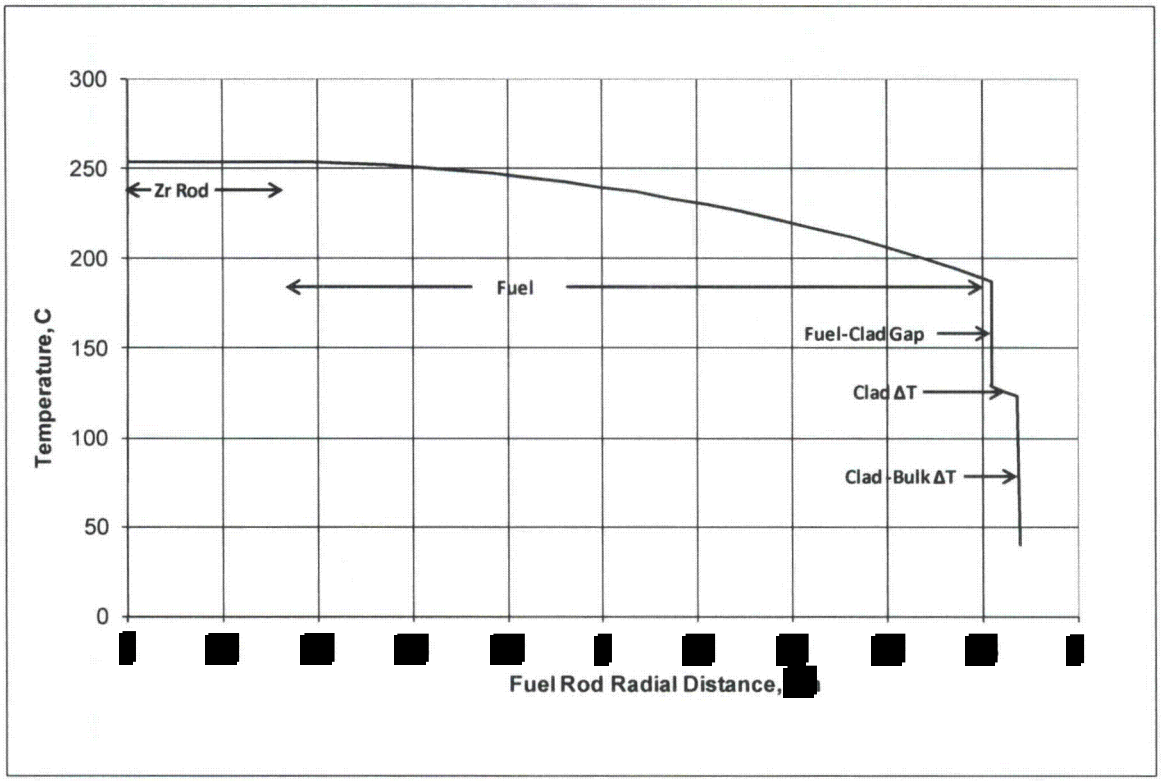

Figure 2-7 300 kW Radial Temperature Profile – UCI Core

Table 2-8 shows calculated quantities from RELAP5 for four UCI Core reactor powers – 100, 250, 275, and 300 kW.

Table 2-8 Calculated Thermo-Hydraulic Values – UCI Core

kW	Hot Element Max Temp (°C)	Hot Element Max Clad (°C)	Core Avg. (°C)	Hot Element Flow Out, (°C)	Avg Element Flow Out, (°C)	Hot Element Flow (kg/s)	Avg Element Flow (kg/s)
100	146	73	90	40	36	0.029	2.122
250	236	121	150	50	44	0.043	3.143
275	245	122	157	52	45	0.044	3.261
300	253	123	164	54	46	0.046	3.368

The combination of RELAP5 thermal hydraulics and the Bernath critical heat flux correlation was used to determine the maximum reactor power at which the departure from nucleate boiling, DNB, would occur. The reactor power for DNB is obtained using a reactor pool temperature of 25°C and by systematically increasing the reactor power until a local wall heat flux in the hot rod equals the DNB flux as predicted by the Bernath correlation. At this power the DNBR equals 1.0, that is, the ratio of the predicted DNB flux divided by the wall heat flux equals 1.0.

When the reactor power is systematically increased in the model, the highest theoretical achievable power is 2.00 MW, for which the DNBR is 1.15. The UCI pool depth to the core top is 20 ft, which yields a higher operating pressure within the core and thus a higher saturation temperature. At 2.00 MW, the hot rod generates a fair amount of vapor along the flow channel. Based on the RELAP model, the maximum theoretical achievable power of the UCI reactor would be 2.00 MW corresponding to a hot rod power of 36.2 kW/element, at which point DNB could occur.

The UCI licensed operating limit is 275 kW, which is based on the reactor cooling capacity. With the UCI Core operating at 275 kW with a maximum allowable core inlet temperature of 25°C, the DNBR would be 7.27 (2.0 MW/0.275 MW), respectively. This analysis demonstrates that there is a large margin to safety when the UCI Core is operating at the maximum power and core inlet temperature.

3 SUMMARY

Based on the thermal-hydraulic analysis performed the UCI TRIGA® reactor is considered to be safely operating under steady-state conditions.

4 REFERENCES

1. Crozier, J., "Nuclear Analysis of the University of California – Irvine TRIGA® Reactor," GA 911196, General Atomics, December 2010

2. Shieh, A. S., Ransom, V. H., and Krishnamurthy, R., "Validation of Numerical Techniques in RELAP5/ MOD3.0," Vol. 6, Nuclear Regulatory Commission, March 2009

3. Safeguards Summary Report for the TRIGA Reactor at the Puerto Rico Nuclear Center, Mayaquez, Puerto Rico, PRNC 123, April 1969.

4. Jensen, R. T., and D. L. Newell, "Thermal Hydraulic Calculations to Support Increase in Operating Power in McClellan Nuclear Radiation Center (MNRC) TRIGA Reactor," 1998 RELAP5 International User's Seminar, College Station, Texas, May 1998.

5. Bolin, J., "TRIGA Reactor Thermal-Hydraulic Study," General Atomics – Electronic Systems, Inc., TRD 070.01006.04, Rev. A, April 2008.

6. Rao, D. V., and M. S. El-Genk, "Critical Heat Flux Predictions for the Sandia Annular Core Research Reactor," Sandia report SAND 90-7089, August 1994.

7. Becker, K. M., G. Hernborn, M. Brodl, and G. Erikson, "Burnout Data for Flow of Boiling Water in Vertical Round Ducts, Annuli and Rod Clusters, AE-177, AB Atomenergi, Sweden, 1969.

8. Silvestri, M., "On the Burnout Equation and on the Location of Burnout Points," Energia Nucleare, Vol. 13, No. 9, pp. 469–479, 1966.

9. Gaspari, G. P., A. Hassid, and F. Lucchini, "A Rod Centered Subchannel Analysis with Turbulent Mixing for Critical Heat Flux Prediction in Rod Clusters Cooled by Boiling Water," Paper B6.12, Proc. Fifth Int. Heat Transfer Conf., 1974.

10. Bernath, L. B., "A Theory of Local Boiling Burnout and Its Application to Existing Data," Chem. Eng. Progress Symposium Series No. 30, Vol. 56, pp 95–116.

11. Groenveld D. C., et al, "The 2006 CHF Look–up Table," Nuclear Engineering and Design 237, pp 1909–1922, 2007.

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