THERMAL HYDRAULIC ANALYSIS FOR THE OREGON STATE TRIGA REACTOR USING RELAP5-3D

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ABSTRACT

Thermal hydraulic analyses have being conducted at Oregon State University (OSU) in support of the conversion of the OSU TRIGA reactor (OSTR) core from high-enriched uranium (HEU) to low-enriched uranium (LEU) fuel as part of the Reduced Enrichment for Research and Test Reactors program. The goals of the thermal hydraulic analyses were to calculate natural circulation flow rates, coolant temperatures and fuel temperatures as a function of core power for both the HEU and LEU cores; calculate peak values of fuel temperature, cladding temperature, surface heat flux as well as departure from nuclear boiling ratio (DNBR) for steady state and pulse operation; and perform accident analyses for the accident scenarios identified in the OSTR safety analysis report. RELAP5-3D Version 2.4.2 was implemented to develop a model for the thermal hydraulic study. The OSTR core conversion is planned to take place in late 2008.

1. Introduction

Oregon State University has recently completed its core conversion analysis as part of the Reduced Enrichment for Research and Test Reactors (RERTR) program. As part of the core conversion analysis, the Nuclear Regulatory Commission (NRC) has required that complete neutronic and thermal hydraulic analyses be conducted on the existing OSTR (HEU) and potential (LEU) core. The goals of the thermal hydraulic analyses were to:

- Calculate natural circulation flow rates, coolant temperatures and fuel temperatures as a function of core power for both the HEU and LEU cores.
- For steady state and pulsed operation, calculate peak values of fuel temperature, cladding temperature, surface heat flux as well as departure from nucleate boiling ratio (DNBR) and temperature profiles in hot channel for both the HEU and LEU cores.
- Perform accident analyses for the accident scenarios identified in the OSTR safety analysis report.

2. OSTR Description

The OSTR is a TRIGA Mark II design having a circular lattice configuration and is solely dependant on natural convection as is method of heat removal. The top of the core is located ~ 5 meters from the surface of the ~ 6 meter deep pool as seen in Figure 1. Figure 2 and Figure 3 present the HEU Normal Core and LEU ICIT Core. The HEU Normal Core was used in conjunction with operational data for the benchmark comparisons during this study, while the LEU ICIT Core provides the bounding thermal conditions found in all potential (LEU) core configurations.

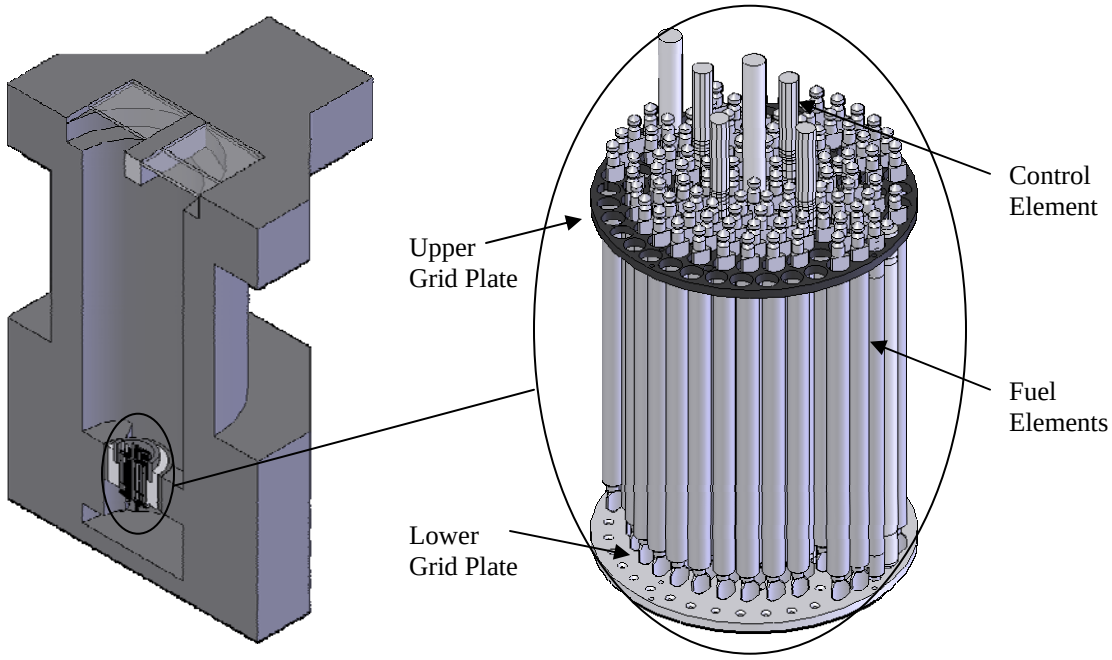

Figure 1: Sectional Isometric View of the OSTR and Exploded View of the OSTR Core

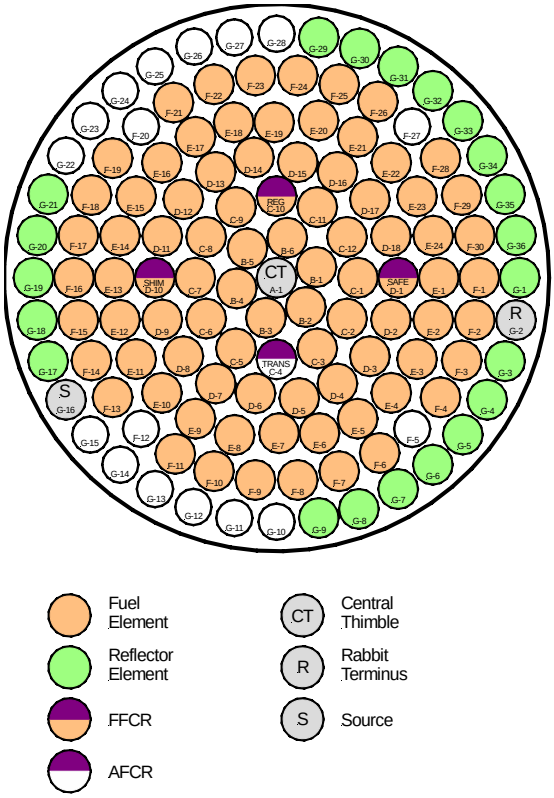

Figure 2: HEU-Normal Core Configuration

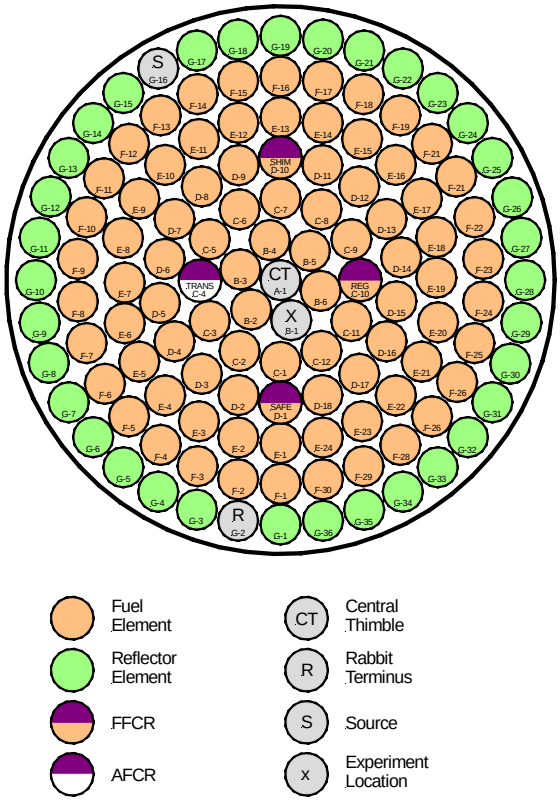

Figure 3: LEU-ICIT Core Configuration

Table 1: Comparison of HEU FLIP and LEU 30/20 Fuel Designs

Fuel Components	HEU FLIP	LEU 30/20
Uranium content [mass %]	8.5	30
U-235 enrichment [mass % U]	70	19.75
Erbium content [mass %]	1.6	1.1
Fuel alloy inner diameter [mm]	6.35	6.35
Fuel alloy outer diameter [mm]	36.449	36.449
Fuel alloy length [mm]	381	381
Cladding material	Type 304 SS	Type 304 SS
Cladding thickness [mm]	0.508	0.508
Cladding outer diameter	37.465	37.465

A summary of the fuel components of the HEU FLIP and LEU 30/20 fuel is provided in Table 1. Table 2 provides the associated power peak factors for each core analysed during this study. It can be seen that the HEU Beginning of Life (BOL) Normal Core configuration holds an effective peak factor greater than both the HEU Middle of Life (MOL) and HEU End of Life (EOL) Normal Cores; while the LEU MOL In Core Irradiation Tube (ICIT) Core holds the bounding effective peak factor over all other LEU core configurations. It is for this reason that the HEU BOL Normal Core and LEU MOL ICIT Core are referenced in this paper.

Table 2: Hot-Rod Power Summary

	Hot Channel Location	Hot Channel Thermal Power [kW]	Relative Error Associated to Thermal Power [%]	Hot Channel Peak Factor [P_{max}/P_{avg}]	Hot Channel Fuel Axial Peak Factor [P_{max}/P_{avg}]	Hot Channel Fuel Radial Peak Factor [P_{max}/P_{avg}]
HEU-BOL Normal Core	B3	18.02	0.0017	1.442	1.236	1.907
HEU-MOL Normal Core	B6	18.37	0.0009	1.420	1.209	1.518
HEU-EOL Normal Core	B6	16.48	0.0011	1.273	1.234	1.708
LEU-BOL ICIT Core	B6	18.47	0.0018	1.477	1.221	1.963
LEU-BOL CLICIT Core	B3	17.03	0.0017	1.362	1.221	1.943
LEU-BOL Normal Core	B3	17.77	0.0017	1.422	1.219	1.945
LEU-MOL ICIT Core	B6	18.52	0.002	1.482	1.225	1.846
LEU-MOL CLICIT Core	B3	17.03	0.0031	1.363	1.225	1.821
LEU-MOL Normal Core	B3	17.80	0.0033	1.424	1.222	1.823
LEU-EOL ICIT Core	B6	17.61	0.0032	1.409	1.181	1.699
LEU-EOL CLICIT Core	C7	16.35	0.00105	1.308	1.212	1.732
LEU-EOL Normal Core	B3	17.02	0.0010	1.362	1.178	1.707

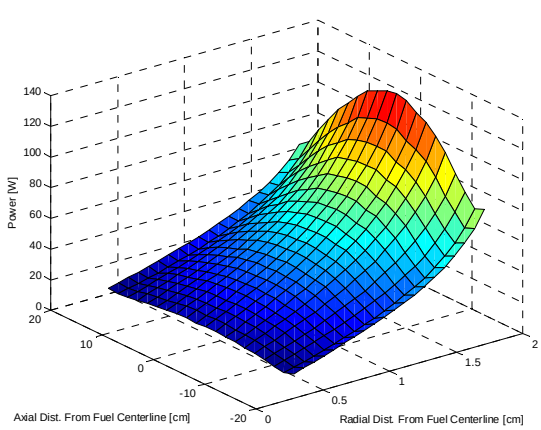

Figure 4: Fuel Rod Power (HEU-BOL Normal Core)

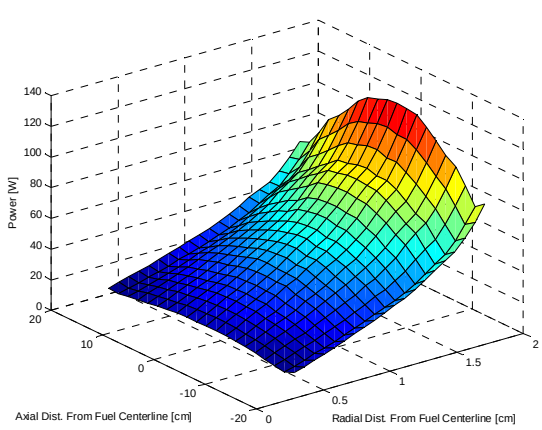

Figure 5: Fuel Rod Power (LEU-MOL ICIT Core)

Figure 4 and Figure 5 provide the hot rod fuel element power distribution for the reference core configurations. These power distributions were normalized to their power factors and broken into two vectors defining an axial and radial power shape for each hot rod fuel element as seen in Figure 6 and Figure 7. These two vectors were implemented in the RELAP5-3D model as their heat source distribution.

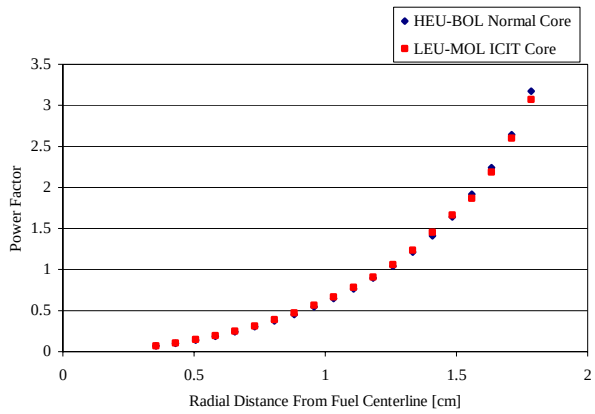

Figure 6: Radial Power Factor

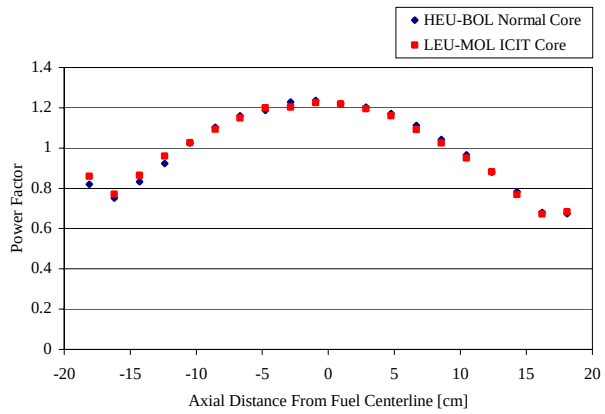

Figure 7: Axial Power Factor

3. RELAP5-3D Model

The RELAP5-3D model seen in Figure 8 consists of a Coolant Source, Cold Leg, Horizontal Connector, Hot Channel, and Coolant Sink. This model is representative of a single OSTR core subchannel, assumed to be the hot channel.

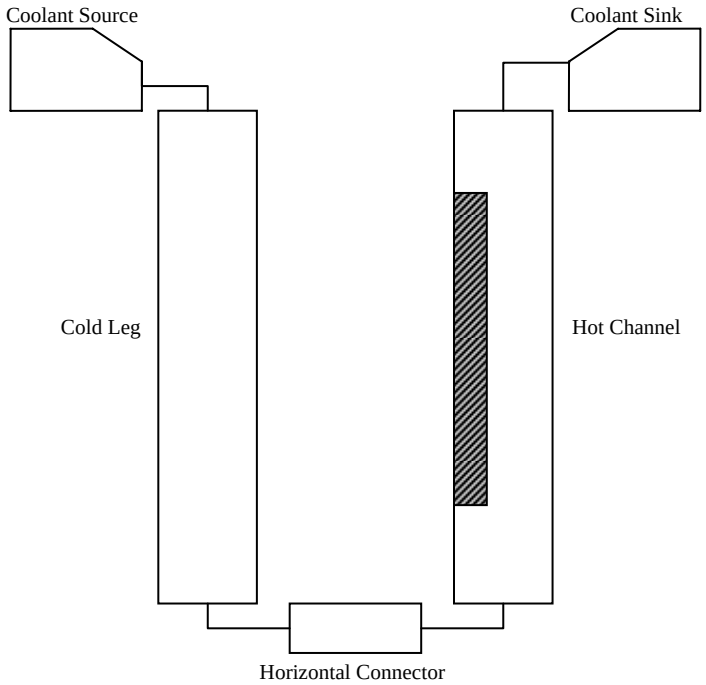

Figure 8: RELAP5-3D Model Schematic

The analysis was performed using a single flow channel divided into axial segments that represent a single fuel element. The natural convection system for the OSTR was based on a triangular rod lattice configuration (equivalent to that found in subchannels adjacent to the A and B Ring). The natural convection flow in the RELAP5-3D model is obtained by the equilibrium of static head in the coolant source and sink; while the enthalpy created by the heat source allows for a dynamic head gradient between the hot channel and cold leg providing a buoyant upward drive in the hot channel. The reactor geometric and hydraulic inputs that were implemented into the RELAP5-3D model are provided in Table 3.

Table 3: Geometric/Hydraulic Hot Channel Inputs Implemented in RELAP5-3D

Geometric/Hydraulic Description	Value
Unheated core length at inlet [m]	0.1655
Unheated core length at outlet [m]	0.1647
Inlet pressure loss coefficient	1.29
Exit pressure loss coefficient	0.574
Absolute pressure at the top of the core [Pa]	1.49E5
Inlet coolant temperature [°C]	49.0
Flow area [m^2]	3.304E-04
Wetted perimeter [m]	0.1177
Hydraulic diameter [m]	2.051E-02
Fuel element heated length [m]	0.381
Fuel element surface area [m^2]	3.810E-1
Fuel element surface roughness [m]	2.134E-06

4. Results

4.1. HEU

The heat generation in the fuel element is distributed axially in a piece-wise fashion as represented by the curve in Figure 7. A radial heat source distribution was applied as well; this piece-wise profile is given in Figure 6. There are 85 fuel Elements in the initial core, all are FLIP HEU fuel. The hot-rod peak factor is 1.442, producing a power of 18.02 kW (Table 2) at 1.1 MW_{th} core steady state power.

4.1.1. Steady State

As stated earlier the driving force is supplied by the buoyancy of the heated water in the core. Countering this force are the contraction and expansion losses at the entrance and exits to the channel [1], and friction losses due to coolant to fuel element interfacial contact [2]. A summary of the RELAP5-3D results for the OSTR HEU beginning of life normal core configuration is given in Figure 9 through Figure 12.

Figure 9 and Figure 10 present the hot channel (HC) thermal hydraulic parameters as a function of steady state hot rod power. Using the Groeneveld look up tables [3] as the bounding MDNBR value, it can be seen in Figure 10 that the MDNBR does not reach a value of 2.0 until the hot rod produces a power of ~30.0 kW. This is approximately 166 % greater than the 18.02 kW required for the hot rods contribution to the equivalent cores 1.1 MW_{th} output.

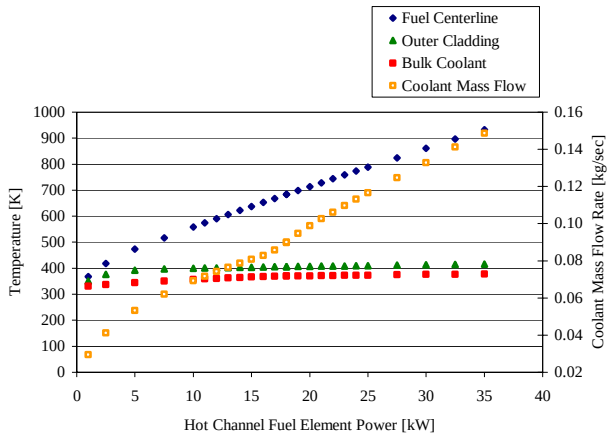

Figure 9: HC- Properties (HEU BOL Normal Core)

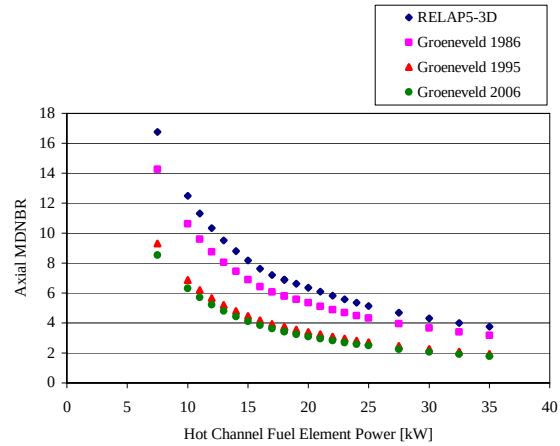

Figure 10: HC- MDNBR (HEU-BOL Normal Core)

Figure 11 presents the axial DNBR when the hot channel is at 18.02 kW. The methods for calculating DNBR shown use the results produced from RELAP5-3D to apply the appropriate correction factors used in the Groeneveld 1986 [4], 1995 [5], and 2006 [3] AECL-UO look-up tables. The MDNBR value produced from the bounding DNBR method, Groeneveld 2006, is 3.420.

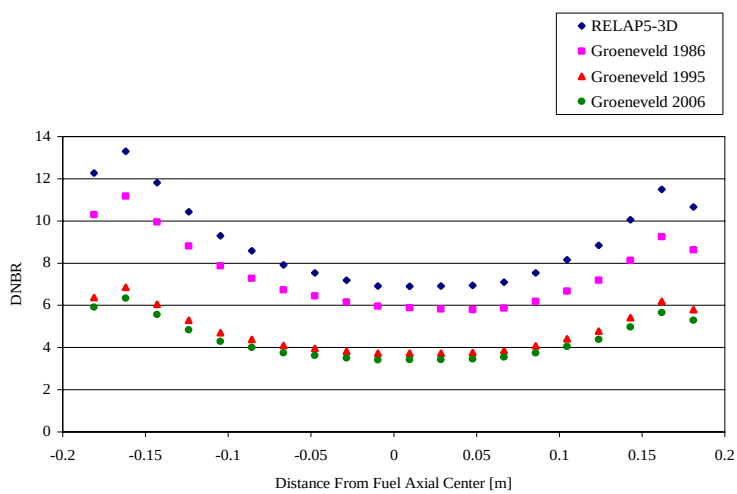

Figure 11: Hot Channel Axial DNBR at 18.02 kW (HEU-BOL Normal Core)

The OSU HEU TRIGA fuel is composed of a radial centralized zirconium pin with a diameter of 6.35 [mm]. This pin is surrounded by the U-ZrH fuel meat with an outer diameter of 36.449 [mm]. A contact gap is produced between the fuel meat and the cladding; this fuel gap is found to vary from 0.05 to 2.0 mils. The stainless steel clad thickness is 0.508 [mm] (Table 1).

A radial temperature profile at 1.1 MW_{th} integral core steady state power was mapped while varying the fuel to clad gap from 0.05 to 0.20 mils, the corresponding temperature was compared to that found in the Instrumented Fuel Element (IFE) during the original 1976 core configuration. These trends are given in Figure 12, as a result of this figure a clad gap of 0.1 mils was used in all core configurations.

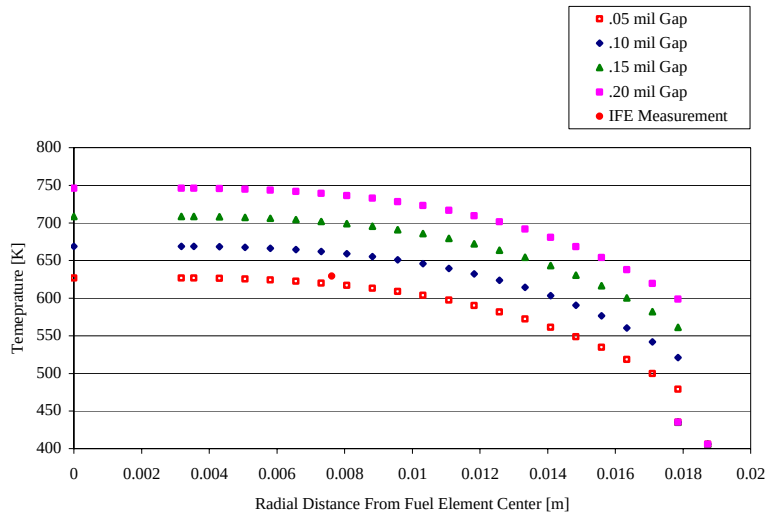

Figure 12: Fuel Element Radial Temperature Distribution at 1.1 MW_{th}

4.1.2. Pulse Operation

The pulsing performance of the reactor was analysed using a point reactor kinetics model. Using this methodology and the fissile fuel characteristics produced from the MCNP5 model during this study a pulse power trace was developed for given reactivity insertions (Figure 13 provides three example power traces) [6]. Applying power traces similarly as given in Figure 13, Figure 14 was produced. During the pulse analysis the fuel thermo-physical properties inherently dominate the tendency for peak fuel temperature over heat removal capability of the system, therefore the thermal conductivity and volumetric heat capacity [7] are as follows:

$$K(T)_{HEU,LEU-FUEL} = .18 \pm 0.009 \qquad \text{[W/cm-°C]} \qquad (4.1)$$

$$\rho C_P(T)_{HEU,LEU-FUEL} = 2.04 + 4.17 \cdot 10^{-3}(T) \qquad \text{[W-sec/cm}^3\text{-°C]} \qquad (4.2)$$

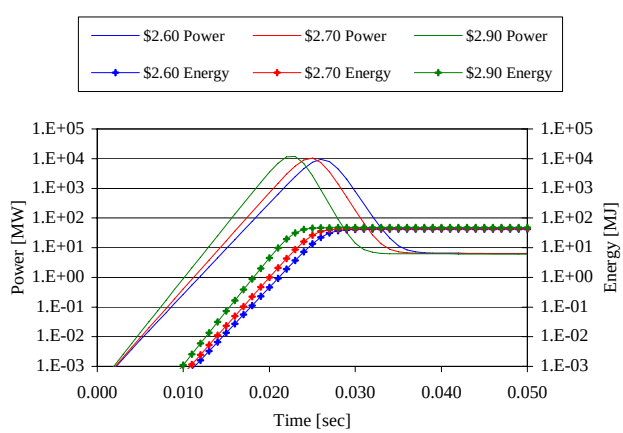

Figure 13: HEU EOL Normal Core Pulse Trace

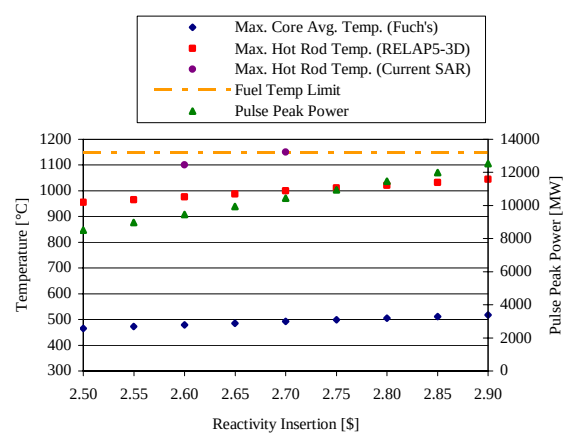

Figure 14: Pulse Results (HEU EOL Normal Core)

The maximum fuel temperature in the current SAR is identified at 1100 [°C] for a reactivity insertion of $2.60 and 1150 [°C] for a reactivity insertion of $2.70. These temperatures are 123.3 [°C] and 149.8 [°C], respectively, greater than that produced during this study. The effective hot rod peak factor for the current SAR in the HEU EOL fuel is 3.41, while the effective hot rod peak factor for the HEU EOL during this study is 2.68. This difference in peak factor complements the lower values for maximum fuel temperature produced during this study.

4.2. LEU

As stated earlier, the heat generation in the fuel element is distributed axially and radially in a piece-wise fashion as provided in Figure 6 and Figure 7. There are 88 fuel Elements in the potential core, all are LEU 30/20 fuel elements. The hot-rod peak factor is 1.482, producing a power of 18.52 kW (Table 2) at 1.1 MW_{th} core steady state power.

4.2.1. Steady State

Because the HEU and LEU core have no geometric alterations toward the fuel elements or integral core, all hydraulic/geometric parameters used in the HEU core were preserved in the LEU RELAP5-3D model (Table 3).

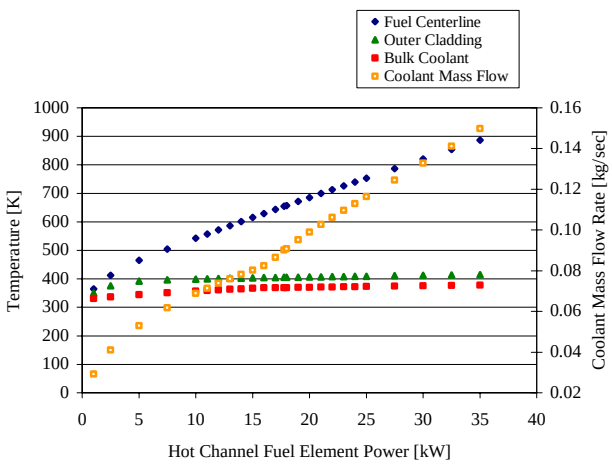

Figure 15: HC- Properties (LEU MOL ICIT Core)

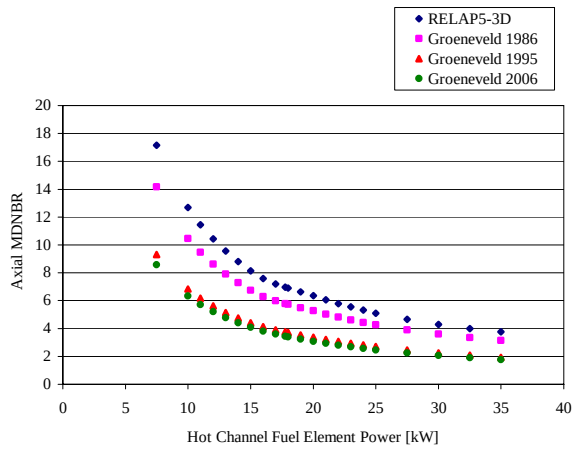

Figure 16: HC- MDNBR (LEU MOL ICIT Core)

Figure 16 provides the axial MDNBR as a function of hot channel rod power. This figure shows that the hot channel MDNBR does not reach a value of 2.0 until the hot rod has a power production of ~30 kW, nearly 162% of that found in the hot rod to maintain an integral core power of 1.1 MW_{th}.

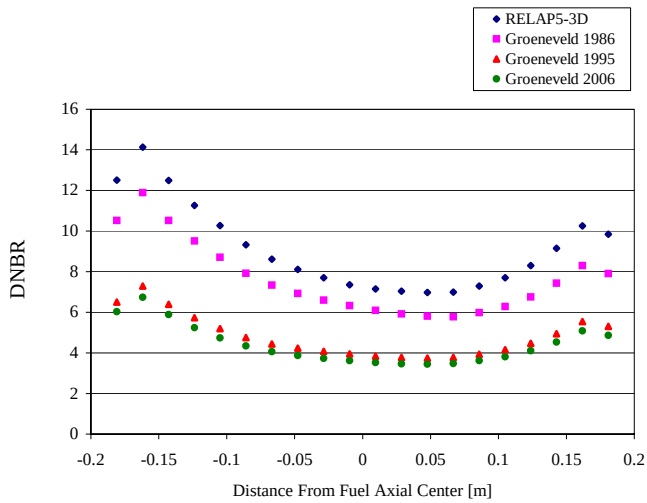

Figure 17: Hot Channel Axial DNBR at 18.52 kW, 0.1 mil gap (LEU MOL ICIT Core)

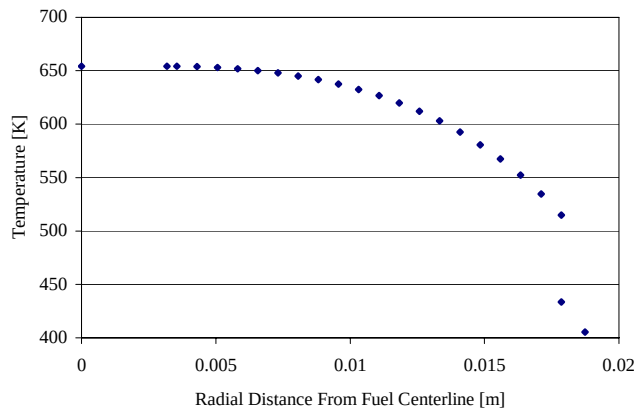

Figure 18: Radial Temperature Distribution at 18.52 kW, 0.1 mil gap (LEU MOL ICIT Core)

The maximum fuel temperature is ~379 [°C] at 18.52 kW hot rod power during steady state operation; while the bulk exit coolant temperature is 96.8 [°C], ~15 [°C] below saturation temperature at the operating pressure.

4.2.2. Pulse Operation

The LEU ICIT Core contains the bounding accumulative peak factor over all other LEU Core configurations (Table 2). For this reason the ICIT Core was used to conduct the LEU pulse analysis. The intra-fuel power factor profiles referenced are given in Figure 6 and Figure 7.

Figure 19 represents the integral core power trace for the LEU ICIT Core given the BOL, MOL, and EOL fuel fissile characteristics produced from this study at a reactivity insertion of $2.65. Accordingly, a reduced power profile was developed as the hot rod power trace that was used to calculate peak fuel power, after applying the hot rod peak factors respectively.

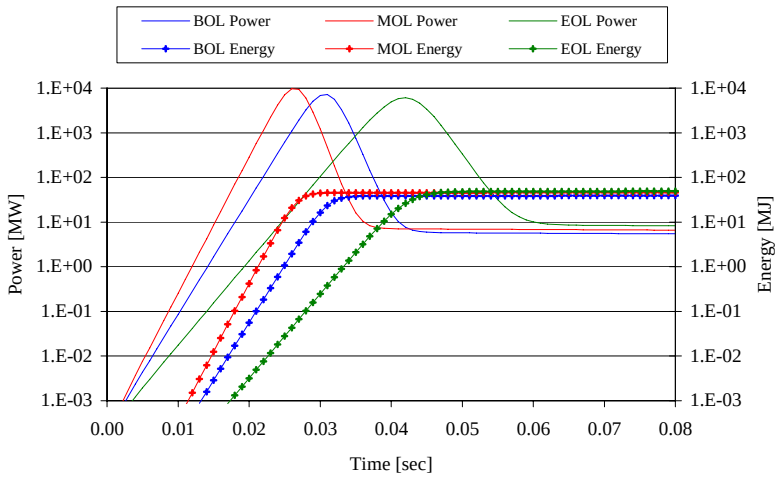

Figure 19: Integral Core Pulse Power Trace ($2.65 Pulse, LEU ICIT Core)

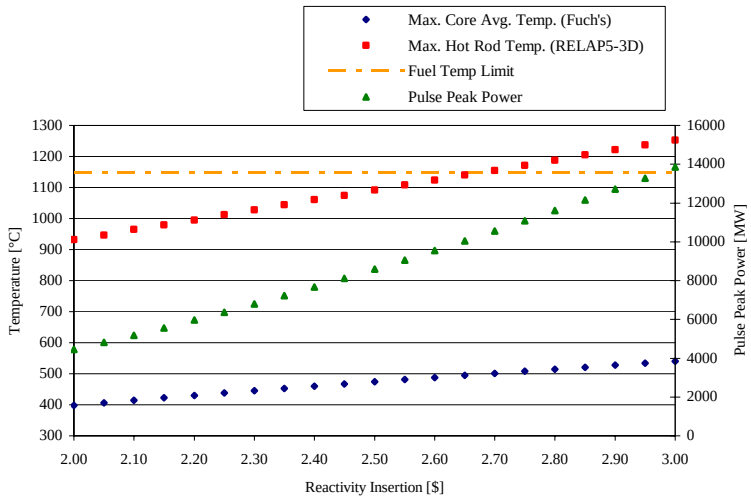

Figure 20: Pulse Summary (LEU MOL ICIT Core)

Referring to Figure 20 the LEU MOL ICIT Core has the bounding maximum fuel temperature as a function of reactivity insertion. At a pulse of $2.70 the maximum fuel temperature is 1154.4 [°C], exceeding the licensed maximum fuel temperature limit by 4.4 [°C]. As a conclusion of this the limiting pulse for the OSTR LEU core is $2.65 for all core life.

5. Conclusions

As a result of this thermal hydraulic analysis for the OSTR during the HEU to LEU core conversion project it is shown that the RELAP5-3D model is validated within relative error margin to operational data; while all thermal hydraulic safety limits are met for the most bounding LEU core configuration.

6. References

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4. Groeneveld, D.C., S.C.Cheng, and T. Doan, *1986 AECL-UO Critical Heat Flux Lookup Table.* Heat Transfer Engineering, 1986. **7**(1-2): p. 46-62.
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