CALCULATIONS IN SUPPORT OF THE MNR CORE CONVERSION

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ABSTRACT

Calculations and results in support of the HEU to LEU fuel conversion for the McMaster Nuclear Reactor are described. Static reactor physics studies were used to determine local and global power distributions; facilitating the definition of a Reference Core configuration for mixed HEU-LEU and complete LEU loadings. Fission product inventory calculations were used to compare the two fuel enrichments from a radiological hazard point of view. Thermalhydraulic models were created and analyzed to determine steady-state temperature distributions and safety margins, and used as a scoping tool the in development of a full core thermalhydraulic model. The behaviour of the two enrichment fuels was investigated in the context of a protected startup transient. The simulation results support the conclusion that the LEU fuel behaves in much the same way as the HEU fuel, which it is replacing. The conversion results in no new safety issues or significant changes in safety parameters.

INTRODUCTION

The McMaster Nuclear Reactor (MNR) reached first criticality on April 4, 1959. MNR was granted approval to switch from HEU (93% U-235 enrichment) to LEU (19.75% U-235 enrichment) in 1998. The first LEU fuel assembly was installed in January 1999. The conversion is a gradual process with replacement of spent HEU assemblies with fresh LEU assemblies. Presently (October 2002) MNR is just past the halfway point for the core conversion. Full LEU loading is expected by 2006. Two similar LEU assemblies, limited to restricted use in MNR, were introduced in 1990. The first reached exit burnup and was removed in July 2000. The second will be removed shortly.

This document describes simulation methods, models and results used in support of this conversion.

General Facility Description

MNR is a light-water-cooled and moderated, pool-type reactor. It is currently licensed to operate at thermal power levels not to exceed 5 megawatts (5 MW_{th}), using plate-type fuel. Below 110 kW cooling may be accomplished by natural convection. For higher power, cooling is by gravity flow through the fuel locations to a plenum and then to a holdup tank. Water is pumped from the hold-up tank through the heat exchanger and back to the pool. Energy is removed from the primary coolant through a heat exchanger to the secondary cooling system. The secondary system delivers heat to the atmosphere through two cooling towers. A schematic of the MNR primary system is shown in Figure I.

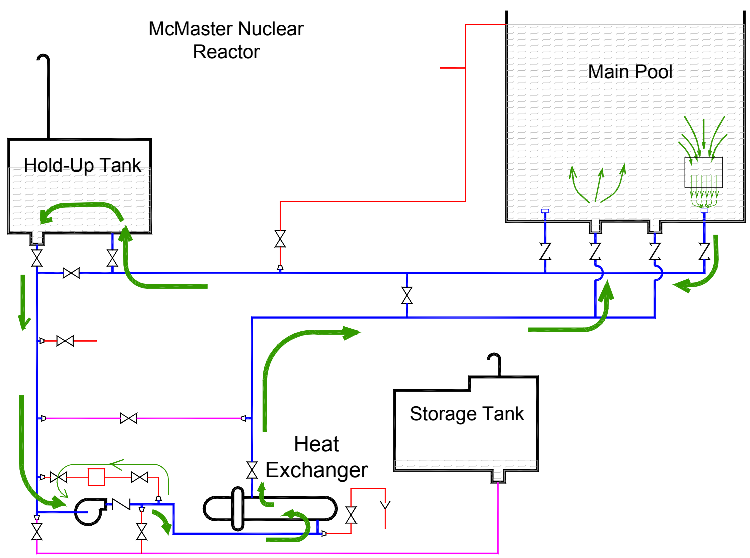

Figure I: Schematic of MNR Primary Side

Fuel Assembly Description

MNR uses typical MTR-type fuel assemblies. Two designs have been used for HEU fuel. 18-plate fuel assemblies are composed of 16 fuelled plates and two, outer, solid aluminum, "dummy" plates. 10-plate assemblies contain fuel in all plates. The 10-plate fuel plates are thicker as are the coolant channels. Each plate is a multi-layer construction of fuel meat, clad on both sides with aluminum. The plates have a curvature of 5.5" in radius and are supported by aluminum side-plates.

Control-fuel assemblies contain nine fuelled plates. A central space accommodates an absorber rod. The plate dimensions of the control-fuel are identical to those of the 18-plate assemblies.

LEU 18-plate standard-fuel assemblies are geometrically identical to HEU 18-plate standard-fuel assemblies. Similarly, LEU and HEU control-fuel assemblies are geometrically identical.

Cross-sectional views of the 18-plate standard-fuel and the 9-plate control-fuel assemblies are shown in Figure II.

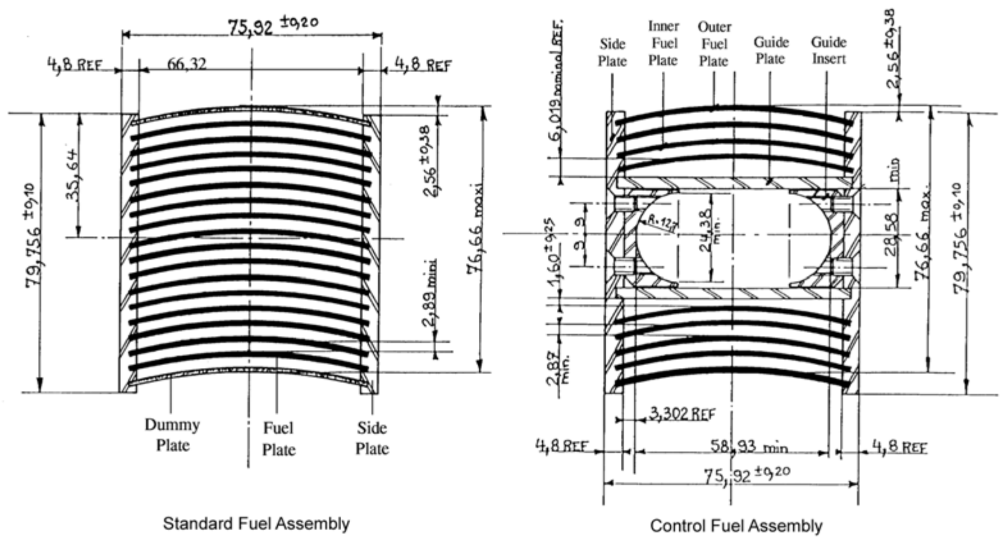

Figure II: MNR Standard- and Control-Fuel Assemblies (dimensions in mm)

Table 1: Fuel properties, Ref. [1]

	HEU 18-Plate	**HEU 10-Plate**	**LEU 18-Plate (new)**	**LEU 18-Plate (old)**	**LEU 9-Plate**	**HEU 9-Plate**
Plate (cm) Thickness Width Height (inner plate) (outer plate)	 0.127 7.140 62.55 	 0.152 7.348 62.55 71.44	 0.127 7.140 62.55 	 0.127 7.140 62.55 	 0.127 7.140 62.55 71.44	 0.127 7.140 62.55 71.44
Meat (cm) Thickness Width Height	 0.051 6.230 60.00	 0.076 6.350 60.00	 0.051 6.230 60.00	 0.051 6.230 60.00	 0.051 6.230 60.00	 0.051 6.230 60.00
Meat Composition	UAl_x-Al or U_3O_8-Al	UAl_x-Al or U_3O_8-Al	U_3Si_2-Al	U_3Si_2-Al	UAl_x-Al or U_3O_8-Al	U_3Si_2-Al
Coolant Channel Thickness (cm)	0.300	0.631	0.300	0.300	0.300	0.300
Number of Plates (fuelled/total)	16/18	10/10	16/18	16/18	9/9	9/9
Initial U-235 loading (g/plate)	12.25	16.0	14.1	17.75	12.25	12.5

HEU fuel is nominally 93% enriched and the fuel meat is UAl_x-Al or U_3O_8-Al (depending on the manufacturer) with initial per-plate U-235 loadings of 12.25 grams for both 18-plate and control-fuel assemblies, and 16.0 grams for the 10-plate assemblies. LEU fuel is nominally 19.75% enriched and the fuel meat is U_3Si_2-Al. The initial per-plate U-235 loadings are 17.75 grams, 14.1 grams and 12.5 grams for the older 18-plate, new 18-plate, and control-fuel respectively. Geometry and material specifications for the various assembly types are given in Table 1.

Core Configuration

The MNR core is defined by a 9-by-6 grid plate of which a 7-by-6 region contains fuel assemblies. Any grid site may house a fuel, reflector or experimental assembly or can be left vacant. Typically, the MNR core contains about 30 standard-fuel assemblies and six control-fuel assemblies. Exit burnup is nominally 50% and 35% U-235 depletion for the standard- and control-fuel respectively.

The core is reflected on one side by a row of graphite assemblies. On the opposite side of the core a beryllium reflector assembly is included as part of a startup source. The core configuration also includes several irradiation sites including one central high-flux position. Peripheral structures include six beam tubes and a lead block.

Coarse control and emergency shutdown function are achieved with five shim-safety absorber rods consisting of a silver-indium-cadmium alloy. Fine control is accomplished with a single stainless steel regulating rod. All rods move vertically within a control-fuel assembly. The rods are oval shell design.

REACTOR PHYSICS

A series of static reactor physics calculations were performed to compare the HEU and LEU fuel. This included a detailed power-peaking analysis, the results of which are used to define the MNR Reference Core for both mixed HEU-LEU and complete LEU loading patterns. In addition, the fission product inventories of the two fuel enrichments were investigated.

Codes

The WIMS-AECL/3DDT code package was used in this analysis. WIMS-AECL [2][3] is a transport theory cell code with 1-D slab geometry capability. The cross section library contains 89 energy groups and was compiled from ENDF/B-V. This code was used to model fuel assemblies and other core components.

3DDT [4] is a diffusion theory code, which references WIMS-AECL homogenized and condensed cross-section data. MAPDDT [5] is a companion input preparation code for 3DDT, which greatly simplifies the input file construction. The MAPDDT/3DDT code was used to create 3-D Cartesian models of the MNR core.

The WIMS-AECL/3DDT code package was validated against the IAEA 10 MW static benchmark problem [6] for both HEU and LEU fuel. In addition, simulation results have been compared to experimental measurements in MNR (*e.g.*, flux wires, control-rod calibration, and excess reactivity). The benchmark analysis is reported in Reference [7].

The SCALE suite of codes (version 4.3) was used to determine fission product inventories in an average fuel plate, for both HEU and LEU fuel. This primarily means the driver module SAS2H [8]. The modules BONAMI and NITAWL perform resonance self-shielding calculations; their output and user-specified geometry is input to the one-dimensional transport code XSEDRNPM, which produces homogenized cross-sections. These are input to ORIGEN with power history data for depletion calculations. Cross-sections are updated during the calculation cycles to reflect changes in nuclide densities and neutron spectrum.

The basic data library for the calculations is called "27BURNLIB". It contains 27-energy group neutron and gamma data for common materials as well as a large amount of information on fission products. It is reported to be the most suitable database for general depletion problems [9].

Power Peaking

The power-peaking analysis involved quantifying power density distributions both within a given fuel assembly and over the entire core. The distributions are expressed in terms of a set of peak-to-average power density ratios called "power-peaking factors" (PPFs).

Plate-to-plate local power-peaking factors (LPPFs) describe power variations between individual fuel plates in a given assembly. Vertical power-peaking factors (VPPFs) describe the vertical power distribution in a given fuel assembly. Horizontal power-peaking factors (HPPFs) describe the horizontal (*i.e.*, x-y) power distribution between different assemblies, *i.e.*, in a given core. The product of all of these factors gives a total peak-to-average power density ratio for the core and is referred to as the overall power-peaking factor (OPPF). Schematics of local, vertical and horizontal power density distributions are shown in Figure III.

Multi-plate, *i.e.*, complete-assembly, cell models were constructed to calculate power variations between the plates of a single fuel assembly. Plate-averaged power densities were used to determine the LPPFs for each fuel type at various stages of burn-up and in different core environments.

Local power peaking was found to be greatest in the outer fuel plates of 18-plate assemblies. In a fuel lattice environment the outer plates are adjacent to proportionally more moderator than the inner fuel plates due to the presence of the outer dummy plates. This results in increased moderation and higher power densities. The effect is magnified for assemblies near large water gaps such as the Central Irradiation Facility (CIF) site in MNR.

For the control-fuel assembly the power density was found to be greatest near the vacant central absorber slot, again because of the water gap. The power peaking was found to be significantly smaller in the 10-plate fuel assemblies due to the more even distribution of light water moderator in the larger coolant channels. The LPPF was also found to strongly depend on burn-up, with the power density distribution flattening as the fuel burns.

The limiting local power peaking was found to be associated with a fresh 18-plate standard-fuel assembly.

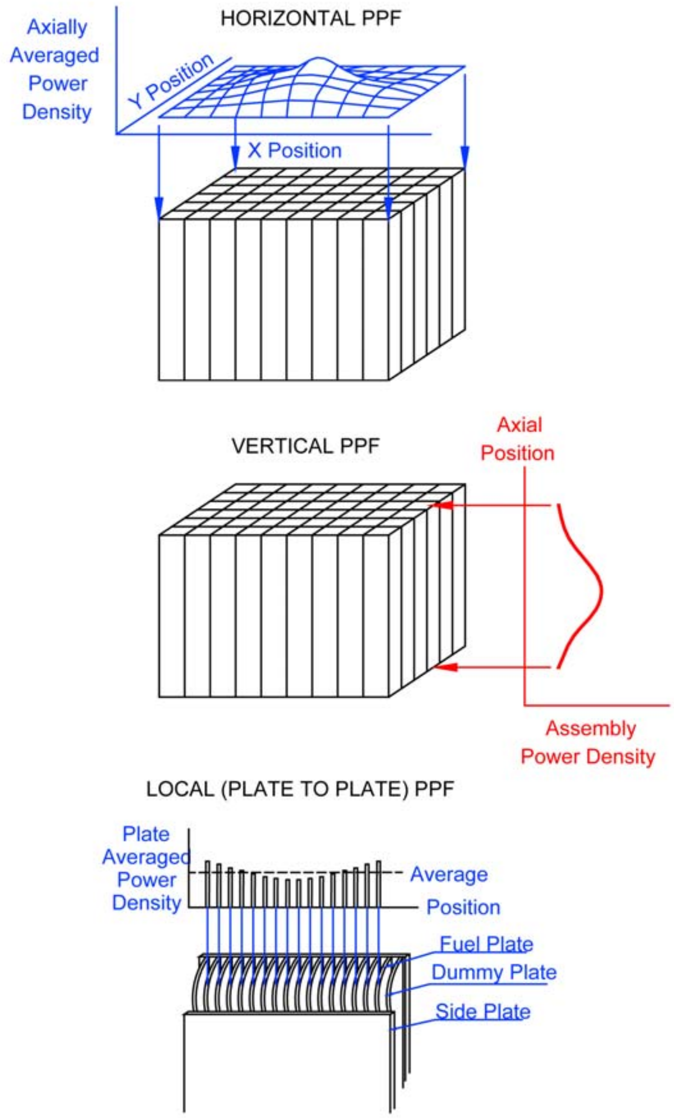

Figure III: Schematics of Horizontal, Vertical and Local Power Distributions in MNR

The LEU 18-plate standard-fuel was found to have slightly larger LPPFs compared to the HEU 18-plate standard-fuel. This is due to the increased U-235 and U-238 loading in the LEU fuel relative to the HEU type resulting in increased shielding of the inner fuel plates. Power-peaking factor results for fresh fuel assemblies in various environments typical of MNR are given in Table 2.

Table 2: Local Plate-to-Plate Power-Peaking Factors for MNR Fuel Assemblies in Typical Core Environments (note: Be = Beryllium, G = Graphite)

Fuel Type	Fuel Environment						
	Fuel	**Be**	**G**	**H_2O**	**Void (in-core)**	**Void (edge)**	**Lead**
HEU 18-plate	1.14	1.32	1.38	1.39	1.12	1.33	1.32
LEU 18-plate (new)	1.16	1.37	1.43	1.44	1.14	1.39	1.37
LEU 18-plate (old)	1.19	1.44	1.50	1.52	--	--	--
HEU 10-plate	1.01	1.16	1.21	1.22	--	--	--
HEU 9-plate	1.17	1.14	1.18	1.14	1.17	1.13	1.15
LEU 9-plate	1.17	1.14	1.18	1.14	1.17	1.14	1.15

The power density solution from a 3-D core model was averaged over the horizontal mesh of a given grid site for each axial mesh region. These horizontal averages were used to compute the vertical and horizontal PPFs. A detailed sensitivity analysis included burn-up loading patterns, reflector material and size, irradiation and control-fuel positions, as well as absorber vertical position.

The LEU 18-plate standard-fuel behaves similarly to the HEU 18-plate standard-fuel in terms of VPPFs and HPPFs. Likewise, LEU and HEU control-fuel have similar global power peaking. At the same burn-up and in the same environment in the same grid position, the LEU 18-plate standard-fuel is associated with slightly higher HPPFs than the HEU 18-plate standard-fuel, due to the higher U-235 loading.

The results of the sensitivity analysis helped define reference core configurations for both mixed HEU/LEU and complete LEU loadings. The Reference Core defines a limiting configuration in terms of maximum power density while conserving realistic core features. The Mixed HEU/LEU Reference Core loading pattern is shown in Figure IV. The complete LEU Reference Core loading pattern is based on an identical assembly burn-up distribution as that for the mixed HEU/LEU configuration. Characteristics of the Reference Core are discussed in Reference [10].

The Reference Core has been defined without inclusion of the 10-plate fuel assemblies as it is designed for analysis of future operation. The remaining 10-plate fuel assemblies are all beyond mid-life burnup and once they reach exit burnup, will be replaced by 18-plate standard-fuel. Current licensing does not allow operation above 2 MW_{th} for a core containing 10-plate fuel.

By combining appropriate local, vertical and horizontal power-peaking factors, the OPPFs for each assembly in the core were calculated. The maximum PPFs for an MNR core containing a single central flux trap are shown in Table 3. The PPF analysis is described in more detail in Reference [11].

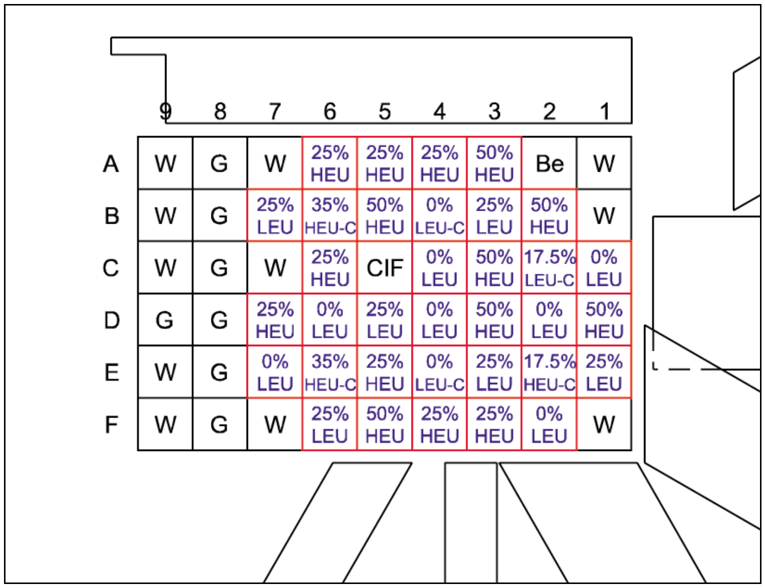

Figure IV: MNR Mixed HEU/LEU Reference Core (note: % denotes percent U-235 depletion, HEU = HEU Standard-Fuel Assembly, HEU-C = HEU Control-Fuel Assembly, LEU = LEU Standard-Fuel Assembly, LEU-C = LEU Control-Fuel Assembly, Be = Beryllium Reflector, CIF = Central Irradiation Facility, G = Graphite Reflector, W = Vacant Water Site)

Table 3: Limiting Overall Power Peaking Factors for MNR

Fuel Type	LPPF	VPPF	HPPF	OPPF
HEU 18-Plate	1.39	1.6	1.6	3.56
LEU 18-Plate (new)	1.44	1.6	1.85	4.26
HEU 10-Plate	1.23	1.6	2.7	5.31

Fission Product Inventory

The fission product inventory in both HEU and LEU 18-plate fuel was examined for the same irradiation and decay conditions. Details are given in Reference [12].

These calculations used a single-plate model with power ratings representative of average and maximum plate power for nominal core powers of 2 MW_{th} and 5 MW_{th}. The calculations for decay power were compared to the Untermeyer-Weills [13] empirical curve and yielded satisfactory agreement. Full core fission product inventories based on the ORIGEN results were compared favourably to previous MNR SAR results [14]. For the same exposure and decay conditions there was no significant difference in the total fission product inventory for HEU and LEU fuel. The LEU fuel activity was found to be 2-3% higher than the corresponding HEU fuel.

As a further indication of the similarity of the two fuel types, the distribution of isotopes in the inventory was examined. This is illustrated in Figure V, which shows the total fission product inventory as a function of time after operation. The fact that the two curves do not diverge suggests that the inventory distributions are similar.

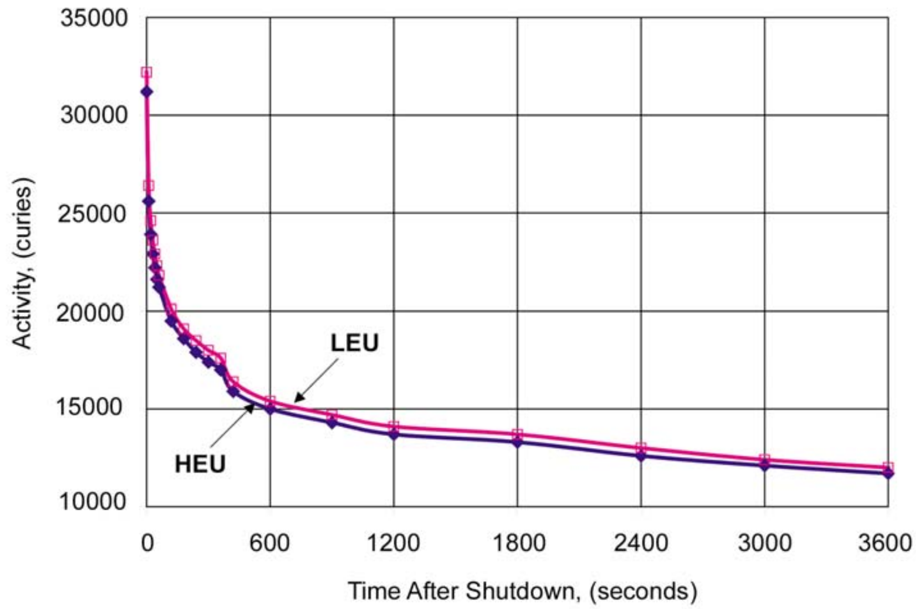

Figure V: Fission Product Activity in HEU and LEU Fuel for maximum plate power, 50% HEU Burnup Exposure After Shutdown

Table 4 summarizes the actinide concentrations in (nominal 2 MW_{th} core power) HEU and LEU plate irradiated at average power. As expected, the relative amount of plutonium produced is much greater for the LEU assembly. However, the quantity produced is still small (2-3% of U-235 inventory) in terms of quantity, power production and reactivity effects.

Table 4: Per Plate Actinide Concentrations (grams)

Isotope	HEU		LEU		
	Initial	**1169 days**	**Initial**	**1169 days**	**1342 days**
U-234	--	0.00005	--	0.00006	0.00007
U-235	12.25	6.34	14.06	8.23	7.4
U-236	--	0.886	--	0.879	1
U-237	--	0.00018	--	0.00016	0.00019
U-238	0.919	0.906	57.1	56.8	56.8
U-239	--	0	--	0.00001	0.00001
Np-237	--	0.00891	--	0.00839	0.0112
Np-238	--	0.00001	--	< 0.00001	0.00001
Np-239	--	0.00005	--	0.00104	0.00111
Pu-238	--	0.00055	--	0.00046	0.00072
Pu-239	--	0.00724	--	0.185	0.201
Pu-240	--	0.00135	--	0.0288	0.037
Pu-241	--	0.00025	--	0.00488	0.00714
Pu-242	--	0.00003	--	0.00045	0.0008

Am-241	--	0.00001	--	0.00017	0.00027
Am-242m	--	--	--	< 0.00001	< 0.00001
Am-243	--	0	--	< 0.00001	0.00001
Cm-242	--	--	--	< 0.00001	0.00002
Total grams	13.17	8.15	71.19	66.14	65.46

Note: 1169 and 1342 days are the times of operation at average plate power for a 2 MW core to achieve a nominal 50% burn-up of HEU and LEU assemblies, respectively.

The results presented here indicate that for a given set of irradiation and decay conditions, there is no significant difference between HEU and LEU assemblies, *i.e.*, the HEU and LEU fuel assemblies present the same or a similar radiological hazard.

THERMALHYDRAULICS

Models were created to investigate differences in thermalhydraulic behaviour between HEU and LEU fuel and also for assembly and core model development. The node-link code CATHENA [15] was used in this analysis. This code is industry-standard in Canada for power reactor analysis. The MNR thermalhydraulic model is described in Reference [16]. In addition to providing a comparison of the HEU and LEU fuels, the CATHENA modelling was also a first attempt at building a full-core thermalhydraulic model of MNR.

Single-plate models were constructed for average and high power 18-plate HEU and LEU assemblies. LEU and HEU standard-fuel assemblies are geometrically identical. The only differences in fuel assemblies of different enrichments, which affect thermalhydraulic performance, are the heat capacity and thermal conductivity associated with the different fuel meat materials.

The different fuel types were examined at nominal 2 MW_{th}, 5 MW_{th} and 12 MW_{th} steady state conditions for a range of nominal flow rates.

Modelling

CATHENA is limited to annular geometry. For the models used in this analysis, the fuel plate and coolant channel geometry was converted to shell and annulus models while conserving the flow area to wetted perimeter ratio in an equivalent diameter, *i.e.*, $D_e \equiv 4A/P$ where A is area of the coolant and P is the wetted perimeter.

A pipe correlation was used for turbulent flow (*e.g.*, the Dittus-Boelter correlation is part of the modified Chen correlation in CATHENA) in channels as the width to thickness ratios are ~ 23 and ~ 10 for 18 and 10-plate fuel respectively [17]. For laminar flow, the heat transfer is via conduction through the boundary layer. This is consistent with the approach used in CATHENA.

Steady State Simulation

The aluminum cladding and the aluminum in the fuel meat offer excellent thermal conductivity, resulting in little or no temperature rise through the clad and fuel meat for steady-state operation. The largest resistance to heat transfer is at the clad/coolant interface. Tables 5 and 6 show the simulation results for 2 MW_{th} and 5 MW_{th} cases respectively. There is no discernible difference in thermalhydraulic response of HEU and LEU 18-plate standard-fuel.

Table 5: Flow and Temperature Results for the 2 MW_{th} Case

Fuel Type	Coolant Velocity (m/s)	Coolant Outlet Temperature (°C)	Maximum Sheath Temperature (°C)	Maximum Fuel Temperature (°C)
Average HEU 18-plate	0.69	35	43	43
Hot HEU 18-plate	0.69	43	60	60
Hot LEU 18-plate (new)	0.69	43	60	60
Hot LEU 18-plate (old)	0.69	43	60	61

Table 6: Flow and Temperature Results for the 5 MW_{th} Case

Fuel Type	Coolant Velocity (m/s)	Coolant Outlet Temperature (°C)	Maximum Sheath Temperature (°C)	Maximum Fuel Temperature (°C)
Average HEU 18-plate	0.94	40	53	54
Hot HEU 18-plate	0.95	54	85	86
Hot LEU 18-plate (new)	0.95	54	85	86
Hot LEU 18-plate (old)	0.95	54	85	86

For the overpower case (12 MW_{th}) at nominal 2 MW_{th} flow, the simulations indicate that the exit coolant temperature for the hot assemblies is approaching saturation and a heat transfer crisis. The heat transfer crisis occurs at the clad/coolant interface, not within the fuel meat. Again, the performance of LEU fuel is virtually identical to that of HEU fuel. Therefore, there are no thermalhydraulic concerns regarding operations over the licensed power range with the LEU fuel and a large margin is maintained. Simulation details are in Reference [18].

There are some 10-plate HEU assemblies in current use. They are of lower hydraulic resistance than the 18-plate assemblies and therefore, "steal" some flow from the 18-plate assemblies. It is thus worth noting that by switching to LEU fuel, the 10-plate fuel will eventually be displaced. This will result in additional cooling to the remaining 18-plate HEU and LEU assemblies, thereby increasing the safety margins for a given core flow-rate [19].

Core Modelling

It was also found that the core flow distribution is sensitive to the modelling of the non-fuel assemblies and the fuel assembly by-pass holes. Comparison of assembly flow distributions from CATHENA core models with experimental measurements [20][21] on mock-up fuel assemblies showed significant discrepancies. As a result, development of CATHENA full assembly and full core models were not continued for SAR analysis. Absolute numbers presented herein should be used with caution.

TRANSIENT BEHAVIOUR

The reactor physics and thermalhydraulic response of the HEU and LEU fuel for a postulated start-up accident described in the 1972 Safety Report [14] was considered. This scenario involves multiple system failures without operator intervention. From a shutdown state the shim-safety rod bank is withdrawn at nominal motor speed (26”/minute). Rod withdrawal continues until the power excursion is terminated by the scram shutdown on reactor overpower, all other scram systems and operator intervention are assumed unavailable.

The analysis used a point kinetics model, which includes shim rod movement, shim rod worth axial importance, and temperature feedback. The influence of key parameters was examined for both HEU and LEU fuel [22].

Modelling

A point kinetics FORTRAN code was written which incorporates the total reactivity of the shim-safety rods, the withdrawal rate based on the nominal motor performance, the known characteristics of the high-power trip, and rod drop due to gravity acceleration, reduced by 50% to account for water resistance. Measurements [23] indicate that a 25% reduction is more realistic. Thus, results herein are conservative in this regard. The code includes a one-dimensional time-dependent heat transfer calculation, but does not account for any flow effects (*i.e.*, heat transfer is by conduction only). Boiling was not modelled, however, no case studied involved high enough temperatures for this to be an issue.

Kinetics parameters were taken from Reference [24], except for the HEU fuel prompt neutron lifetime; this value is from Reference [14]. Temperature feedback coefficients were calculated from a series of static cases using the WIMS-AECL/3DDT code package [25].

Maximum power, time of peak power, total energy release, and maximum fuel and outlet water temperatures were calculated. Results were determined for realistic variation of kinetics parameters as well as for a bounding calculation.

Results

The results of the calculations are summarized in Table 7 and shown in Figure VI.

Table 7: Results for HEU and LEU fuel for the Postulated Start-up Accident

Parameter	HEU	LEU	Units
Prompt neutron lifetime	51	44	µsec
Effective beta ratio	1.1	1.05	--

Moderator temperature coefficient	-5.50×10^{-2}	-2.60×10^{-2}	mk/°C
Fuel temperature coefficient	-2.70×10^{-3}	-1.17×10^{-2}	mk/°C
Thermal conductivity	178	95	W/m-°C
Volumetric heat capacity	3.688×10^{6}	2.648×10^{6}	W/m^3-°C
Criticality occurred at	55940	55940	msec
Prompt criticality occurred at	73821	73022	msec
High power trip at	74347	73533	msec
Rod drop began at	74397	73583	msec
Peak power	9.61	9.31	MW
Peak at	74418	73599	msec
Accumulated energy	1.77	1.79	MW-sec
Maximum fuel temperature	54.5	57.6	°C
Final water temperature	40.8	41.4	°C

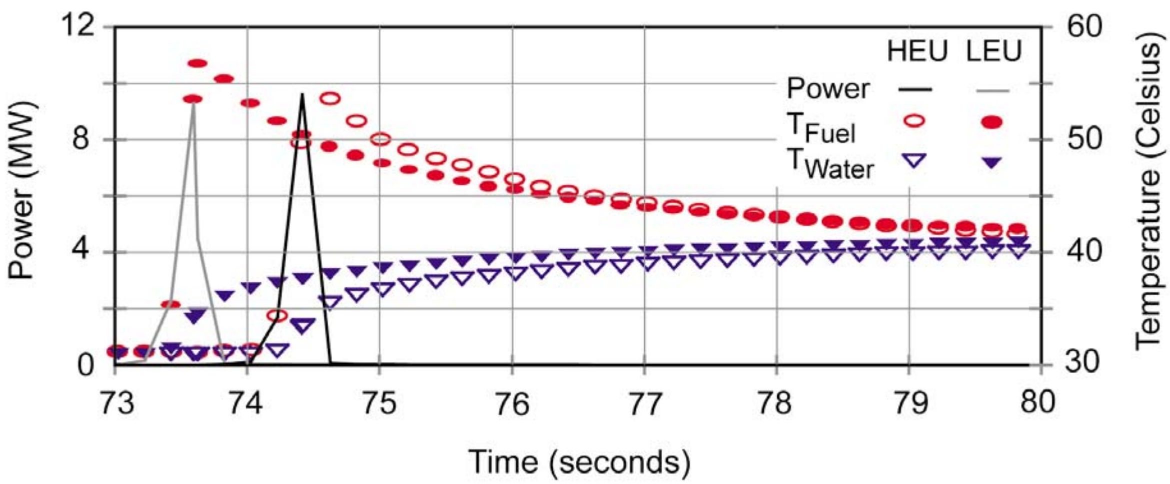

Figure VI: Results for HEU and LEU Fuel for the Postulated Start-up Accident

The smaller value of β_{eff} for LEU fuel means that events happen sooner; this is particularly important if prompt criticality occurs. The shorter prompt neutron lifetime for LEU fuel means that events happen faster. The lower conductivity and heat capacity of the LEU fuel meat results in higher fuel temperatures for a given energy release. The much larger LEU fuel temperature coefficient more than compensates for the smaller LEU moderator temperature coefficient as a self-limiting effect for any power excursion transient event.

The results show that in the context of this event, no new or different hazard arises from the use of LEU fuel. These results are consistent with previously reported analyses on similar systems (for example see References [26][27]).

Sensitivity Analysis

Accurate values have not been measured or calculated for some of the parameters used in these calculations. Therefore, reasonable variations, over a range including both the HEU and LEU values, were examined to demonstrate that any uncertainty arising does not significantly affect any conclusions with respect to safety hazards.

The parameters varied include:

- β_{eff} (0.00690 to 0.00761)
- Λ, prompt neutron lifetime (44 to 57 μsec)
- delay time before shim rod release (25 to 500 msec)
- moderator & fuel temperature coefficients (HEU value to LEU value)
- total shim rod worth (92 to 140 mk)
- axial shim worth weighting (flat and cosine)

Cases involving multiple parameter variations were also studied. The results of this sensitivity analysis show that uncertainty in kinetics parameters is not a significant factor. Specifically, variation of the value of β_{eff} and also the prompt neutron lifetime over reasonable ranges alter the timing of events but, for the protected transient, do not result in significant changes in peak power or temperatures.

The time between when the ion chamber registers a high power condition and the time at which the rod release occurs is very important, in that the power excursion can evolve significantly in this time. For delay times up to 100 msec the transient is benign (*i.e.*, no fuel damage or coolant boiling).

Variation of the feedback coefficients over the HEU to LEU range shows no significant effect on results. Eliminating feedback entirely significantly changes peak power and total energy release but has little effect on peak temperatures, and there is still no expected fuel damage or coolant boiling. The transient with no feedback is still benign.

The total shim worth and axial weighting defines the reactivity ramp size and insertion rate, which affects the transient response. As a result of the initial shim bank position (*i.e.*, 45% withdrawn), the cosine weighting results in a faster power increase and higher peak power due to a larger and faster reactivity insertion. However, the transient is still benign.

HEU and LEU fuel respond similarly. There are no new safety issues with regards to the enrichment conversion.

Bounding Analysis

A bounding calculation was performed using the transient power history generated by the MNR point kinetics code.

An upper limit on the plate temperatures was calculated by assuming all energy generated is deposited into the fuel plate. Based on a 1.88 MW-sec energy release in an 18-plate high power (125 kW) assembly, as expected, significantly higher temperatures are found compared to the simulation results, however, the temperatures are still much lower than those associated with fuel damage. These results are summarized in Table 8. Thus, any uncertainty in thermal-hydraulics does not mask a possible hazard.

Table 8: Maximum possible temperature increases for a 1.88 MJ energy release.

Region of Energy Deposition	Maximum Temperature Increase, (°C)	
	HEU	LEU
Fuel Meat Only	104	163
Fuel Meat and Clad	52	64

CONCLUSIONS

Physics and thermal-hydraulic simulation for the McMaster Nuclear Reactor support the argument that the performance of LEU fuel is equivalent to that of the original HEU fuel. No new hazards, different in nature or greater in magnitude, are expected as a result of the fuel conversion. Both static and transient simulation results for HEU and LEU fuel have been found to be consistent with those reported in the literature.

The power peaking results presented herein are typical of an MTR-type core containing a central water trap.

This work identifies sensitivities as well as modelling approaches directly applicable to safety analysis work on converted LEU or converting MTR-type facilities.

Areas requiring further investigation include development and validation of more sophisticated models and techniques and improving thermalhydraulic models. In particular, difficulties with full core thermal-hydraulic modelling, *i.e.*, flow distributions, have been identified. These are long-term projects; however, it is not expected that the conclusions presented here will change.

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