Removing Fuelling Transient Using Neutron Absorbers

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Summary

Preliminary results indicate that by selecting a small amount of neutron absorber the fuelling transient, currently occurring in a CANDU 37-element fuel bundle, can be successfully removed. A parametric study using the Los Alamos National Laboratories'MCNP 5 code and Atomic Energy of Canada Limited's WIMS-AECL 3.1 is presented in this paper.

1. Introduction

The reactor size, geometry and the distribution of fluence control the flux and power distribution across a reactor core. From a fuel management point of view, it is well known that fuel with high fluence has low reactivity. The opposite is also true. The neutron flux (and power density) is higher in regions where fuel has low irradiation. The on-site fuelling engineer and reactor physicist use these principles to shape the radial flux and power distributions throughout the reactor core. On-power refuelling is the primary means of maintaining a CANDU reactor in a critical condition [1]. Replacing irradiated fuel with fresh fuel in a specific channel has immediate consequences on the local power density distribution and reactivity. The ratio of the peak transient channel power relative to the steady state channel power can often be as high as 6-7%. These power ripples (also known as refuelling ripples) often bring channel operating parameters very close to compliance values and require immediate zone reactivity control system actions or a slight lowering in power if there is not enough safety (and/or operating) margin.

A typical fuel burnup attained in CANDU reactors using the 37-element fuel bundle is 7500 MWd $Tonne^{-1}$ of U [2]. However, the "actual" burnup attained depends on the operating parameters of the core and its reactivity (taking into account the "actual" neutron leakage). When a channel is refuelled with fresh fuel, its reactivity is high. This higher local reactivity tends to promote a power increase in the newly fuelled channel, as well as in its neighboring channels. The saturating fission products reaching saturation in the clean fuel cause a large drop of reactivity and bundle power after a few days of operation. The production of plutonium from neutron absorption in U^{238} causes the reactivity (and power) to slowly increase again with irradiation over the next 40 to 50 Full Power Days (FPD). Following the plutonium peak, the plutonium production can no longer compensate for the depletion of U^{235} and the buildup of fission products. The excess reactivity and power density in the refuelled channel, as well as the power in its neighboring channels, slowly decrease. Eventually, the channel becomes sub-critical and a refuelling is required.

1.1 Objectives

A neutronic analysis on improving operating margins of a CANDU 37-element fuel bundle is described in the following sections. Each of the 37 fuel rods consists of a thin zircaloy tube, the fuel sheath, filled with high density natural UO_2 fuel pellets. A 6 μm thick graphite-based coating, known as CANLUB, is located between the pellets and the sheath. This CANLUB graphite coating was developed to mitigate stress-corrosion cracking of the sheath during, or following, power ramps [3].

In this parametric study, Gadolinium Oxide (Gd_2O_3), a burnable neutron absorber, is uniformly added in the CANLUB for the purpose of mitigating the fission products saturation and xenon free effects in fresh bundles loaded into the reactor core during refuelling. Gadolinium is commonly used for this application as its depletion rate matches closely the xenon build-up rate. It acts as a temporary high absorption cross section surrogate for fission product that would burn-out as fission products build-up. The main objective is to determine the amount of Gd_2O_3 required to suppress power peaking and to flatten axial power peaking at low burnup.

2. Analysis Tools

The CANDU nuclear reactor has been modeled using the MCNP5 Monte Carlo code [4] and the Winfrith Improved Multigroup Scheme-Atomic Energy of Canada Limited (WIMS-AECL Version 3.1) lattice-depletion code [5].

MCNP 5 is a general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled neutron/photon/electron Monte Carlo transport code with the capability to calculate k_{eff} eigenvalues for fissile systems. The code simulates individual neutrons in a 3D fissile environment based on transport data and records their average behaviours. It follows the nuclear particles from the source throughout their entire life. The outcome at each step of their life is determined based on probabilistic events (based on random probability generators). MCNP5 provides the effective multiplication factor of a system in the form of a statistical answer to the transport equation. A 3 Dimension Graphical User Interface continually provides the user with an updated view of the model. It also ensures that gaps in the geometry and overlapping of cells are avoided.

The computer code WIMS-AECL, Version 3.1, is a two-dimensional multi-group neutron transport code capable of applying leakage corrections and of performing fuel depletion for multi-cell lattices. The code is well suited for flux distribution calculations and determines the reactivity of a nuclear reactor by solving the integral form of the neutron transport equation. This is why WIMS-AECL is called a "deterministic" code, whereas MCNP5 is referred to as a probabilistic code since it uses a random probability generator [6].

3. Gd_2O_3 into CANLUB

The CANLUB coating is mixed (doped) with various amounts of Gd_2O_3. The intent is to observe the effects of Gd_2O_3 on a typical CANDU nuclear reactor reactivity (power) and burnup over time. It is assumed that the internal radius of each fuel element is initially coated with a uniform 6 μm thick layer of graphite (CANLUB) over its entire length. Considering a density of 1.7 g cm^{-3} [7] for nuclear grade graphite, the total mass of CANLUB is estimated at 7.3 g per 37-element fuel bundle. Amounts of Gd_2O_3 added to CANLUB is in the 100-300 mg range, which indeed, represents only very small mass fractions (1.3% to 4%).

4. Simulation Results and Discussions

The first step consists of modeling a 480 fuel channels CANDU nuclear reactor (Bruce A configuration) with MCNP and WIMS. The initial excess of reactivity (k_{eff}) can be estimated. Geometrical buckling values (radial and axial) reflecting the CANDU reactor core radius and height are provided to WIMS, allowing for the code to account for neutron leakage and evaluate k_{eff}. At this point, results from MCNP and WIMS are very close.

Criticality calculations are performed with both codes for various amounts of Gd_2O_3 added to CANLUB. Then, burnup calculations are carried out with WIMS for each scenario. The intent is to observe the effects that Gd_2O_3 may have on reactor reactivity while maintaining a channel output of 6.5 MW_{th}, the designed thermal power. The reactor is freshly fuelled and burns until it becomes critical (k_{eff} equals 1), as demonstrated in Figures 1 and 2.

It is observed that a reactor loaded with fresh fuel bundles free of Gd_2O_3 has an initial excess of reactivity in the vicinity of 95 mk (k_{eff} of 1.105). The reactor experiences, as expected, a large drop in reactivity (and power) over the first 2 days of operation mainly due to fission products saturation, resulting in a reactivity value in the vicinity of 60 mk (k_{eff} of 1.065). Following this large drop (-35mk), the reactivity increases slowly for approximately 50 FPD, at which point the plutonium production reaches a maximum. The plutonium peak occurs near a burnup of 1000

MWd Tonne^{-1} of U [7]. The excess of reactivity at this point is estimated at 67 mk (k_{eff} of 1.072), resulting in a small increase of 7 mk. Following the plutonium peak, the reactor reactivity slowly decreases (monotonically) until criticality is reached. Based on WIMS results, an average thermal neutron flux of 2.11 x 10^{14} n cm^{-2} s^{-1} is required to maintain a channel power of 6.5 MW_{th}. As the amount of Gd_2O_3 added to a fuel bundle increases, the initial reactor excess of reactivity decreases. Furthermore, one can also observe that the reactivity drop over the first 2 days of operation significantly decreases as the amount of Gd_2O_3 added to a fuel bundle increases.

Preliminary results reveal that with 190 mg of Gd_2O_3 per fuel bundle, the initial reactor reactivity is in the vicinity of 50 mk (k_{eff} of 1.054). This reactivity value remains constant for the first day of operation and slightly increases, by approximately 2 mk, over the second day. Then, it gradually increases over the next 40-50 FPD until the plutonium peak is reached. Considering the main objective of the present work, this is quite an achievement when compared to a reactor fuelled with bundles free of Gd_2O_3. It can be predicted that the gadolinium isotopes referred to as useful absorbers, Gd-155 and Gd-157, burn out within the first 3-4 FPD following a refuelling. According to the burnup evaluations, the location of the plutonium peak and the useful life of a fuel bundle (burnup) remain unchanged, regardless of the amount of Gd_2O_3 initially added into the CANLUB of a fuel bundle.

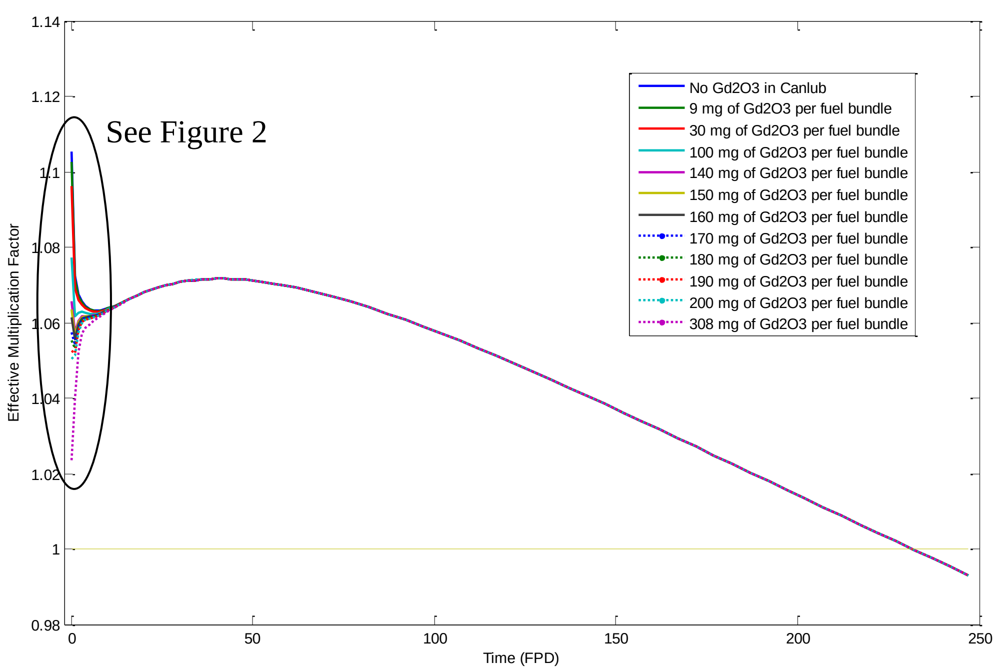

Figure 1: k_{eff} vs Time for a CANDU Reactor

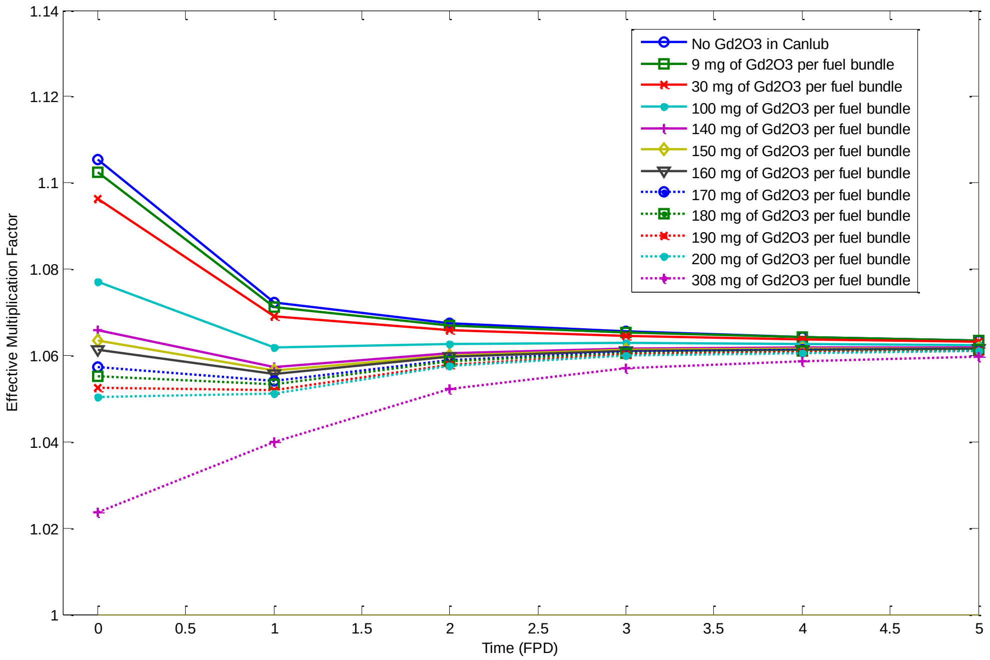

Figure 2: k_{eff} vs Time for a CANDU Reactor

5. Closing Remarks

The neutronic analysis results presented in the previous sections reveal that 190 mg of Gd_2O_3 per fuel bundle meet the objectives of the present work. 190 mg of Gd_2O_3 per fuel bundle clearly mitigate the fission products saturation and xenon free effects in fresh bundles loaded into the reactor core. From a safety aspect, it is anticipated that the ratio of the peak transient channel power relative to the steady state channel power will be reduced. Furthermore, the required actions from the zone reactivity control systems will also be less demanding.

From a fuel management perspective, the refuelling scheme and/or Standard Operating Procedures (SOPs) could be modified. By reducing the amount of reactivity introduced by fresh fuel bundles into a specific channel, the amplitude of the refuelling ripples, dampened by the irradiated fuel of the surrounding channels, will not be as important as the ones currently occurring in operating CANDU reactor cores. In other words, inserting several fresh fuel bundles with Gd absorber in a given channel, will not cause a power ramp as high for the fresh bundles in the adjacent channels, thus permitting them to meet the power ramp criterion more easily. In the end, eliminating or reducing the amount of heavily irradiated fuel bundles in the reactor core translates into an increase in neutron economy although this may reduce the average discharged burnup of the fuel bundles.

6. Future Work

In the second phase of this project, neutronic analysis using various combinations of neutron absorbers will be carried out. The intent is to minimize power peaking and flatten axial power distribution at high burnup without compromising the useful life of a fuel bundle. Potential benefits could be translated as a gain in neutrons over power trip margin, as well as mitigation for reactor derating.

7. References

[1] B. Rouben, “CANDU Fuel Management Course”, Atomic Energy of Canada Limited, 2002, page 37.

[2] Ibid., page 10.

[3] P.K. Chan, K.J. Kaddatz, “How Does CANLUB Work?”, Atomic Energy of Canada Limited Report, 1996.

[4] Brown, Forrest B., “MCNP – A General Monte Carlo N-Particle Transport Code, Version 5”, Volume 1: Overview and Theory, Los Alamos National Laboratory, 24 April 2003.

[5] G. Jonkmans, “WIMS-AECL User’s Manual, Version 3.1”, Reactor and Radiation Physics Branch, Chalk River Laboratories, August 2006.

[6] S. Paquette, “The Super Near Boiling Reactor – Conceptual Design of a Small Nuclear Reactor to Supply District Heating and Electrical Energy to Canadian Forces Bases Located in the Arctic or Northern Remote Communities”, Master’s Degree Thesis, Royal Military College of Canada, April 2010.

[7] S. Glasstone, A. Sesonske, “Nuclear Reactor Engineering, Third Edition”, Van Nostrand Reinhold Company, New York, N.Y., 1981, page 788.

[8] Ref. 5, page 31.