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Fuel Rod Dimension Sensitivity Analysis for DUPIC Fuel Design

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ABSTRACT

Sensitivity of fuel rod dimension for DUPIC fuel design has been analyzed. Analysis was done using two-dimensional transport theory based computer code HELIOS. For lattice calculations, 190-group adjusted library was used. DUPIC fuel material was used in these calculations. It is found that system reactivity increase in this case is about 8.24 mk. The discharge burnup decrease is about 80 MWD/T compared with the standard DUPIC lattice. In this case amount of heavy metal is also reduced up to about 32%, which will ultimate reduce the radioactive waste and this will enhance the major attraction of DUPIC fuel. The effects on Doppler, Moderator temperature and Void coefficient have also been studied.

I. INTRODUCTION

During the last decade in Canada deuterium uranium (CANDU) the option of Direct Use of Spent Pressurized Water Reactor (PWR) fuel In CANDU reactors (DUPIC) fuel has been critically analyzed. This option is based on the utilization of the spent fuel

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from Pressurized Water Reactor (PWR) fuel discharged at 35000 MWD/T heavy element, which typically contains 0.9% ^{235}U and 0.6% fissile Pu, which compares favorably with the fissile content of fresh natural uranium of 0.71% ^{235}U (Ref. 1). In the DUPIC fuel cycle, the spent PWR fuel powder is produced through an oxidation and reduction (OREOX) process and is refabricated as DUPIC fuel to be burned again in a CANDU reactor that was originally designed for natural uranium fuel. Such a synergistic fuel cycle between PWRs and CANDUs provides excellent resource utilization, spent fuel reduction, and safeguard ability. However, because there is no separation of isotopes from spent PWR fuel during the refabrication process, DUPIC fuel contains all the actinides and fission products, which results in a higher fissile content and isotopic composition change, depending on the initial discharge conditions of PWR fuels. Also due to higher fissile contents in DUPIC fuel, it is possible to achieve a discharge burnup of the DUPIC fuel that is more that of natural uranium fuel in the CANDU reactor which reduced the discharged rate as well. Consequently DUPIC fuel cycle enhances the potential of reducing the spent PWR fuel. This is the major attraction of DUPIC fuel cycle is that it offers a high degree of proliferation resistance.

In this study, the possibility of fuel rod dimension change for enhancing the fuel utilization has been investigated. Also, the effect of these changes on the Doppler coefficient, moderator temperature coefficient, void coefficient, and burnup has been analyzed. As HELIOS has the capability of modeling the two-dimensional fuel cluster geometry and benchmark calculations shows that this code has the potential to be used for CANDU fuel lattice[2,3]. A comparison has also been made to investigate the effect of the parameter's sensitivity.

II. HELIOS

HELIOS is a neutron and gamma transport code that uses the current collision probability method and has the capability of modeling any general two-dimensional geometry[2]. The input and output data are processed separately by ARURORA and ZENITH codes, respectively. Currently HELIOS system version 1.5 has been released,

which contains three energy groups libraries based on ENDF/B-VI: the master library with 190/48 neutron/gamma groups, a library with 90/18 neutron/gamma/groups, and a production library with 35/18 neutron/gamma groups. The HELIOS library consists of two sub libraries, which are typically called the adjusted and unadjusted libraries. The only difference between these two libraries is that a 3.4% reduction of the capture cross-section of the ^{238}U isotope is applied for the adjusted library in the resolved resonance range to improve the discrepancy of r^{28} in the TRX-1 pin-cell lattice problem.

AURORA reads, process and then saves the user' s input. These output is written into a HERMES data base for retrieval by HELIOS and/or ZENITH. For each case HELIOS retrieves the input from this data base and executes the calculations specified therein. The output is again written into the same data base for retrieval and further processing by ZENITH. The input except for the basic nuclear data in the library, it consists of the following data types:

- The nuclear-data library with the basic nuclear data, which also defines the energy discretizations (group structures) of the particle transport calculations;
- Data that define the (initial) number densities of materials, and the elements of the albedo matrix elements;
- Data that defines the geometry of the system, including the spatial and angular discretizations to be used in the transport calculations.
- Data that assign one or more property sets to the geometric system, thus defining one or more "states" of the system.
- Data that define the execution sequence of the calculations;
- Data that define what output will be saved in the data base.
- Optional data that define an experimental buckling, change default iteration parameters, accuracies and methods, and produce output.

The methods used by the HELIOS can be described in three areas as;

The geometrical buildup of the system and its properties: The physical methods used to obtain flux, currents, and number densities, which are basically consist of five parts.

Calculating resonance-shielded microscopic XSs: Calculating fluxes and currents by current coupling collision-probabilities (CCCP) method for particle transport.

Evaluating first-flight probabilities: Evaluating, with the B1 method, the criticality spectrum which is used to rebalance the spectrum of CCCP solution.

Solving the burnup chains to get new number densities: The output processing to restart calculations based on ZENITH arrays with number densities and other information.

III. ANALYSIS PROCEDURE

The DUPIC fuel bundle utilizes a 43-element cluster geometry that has been developed for the advanced CANDU fuels as shown in Fig. 1. The fissile content of the reference DUPIC fuel is 1.0 and 0.45 wt% for ^{235}U and ^{239}Pu, respectively. In addition, the center rod of the DUPIC fuel bundle contains 4.3 wt% natural dysprosium[4]. Under normal operating condition, the temperatures of the fuel, clad, coolant, and moderator for both fuel lattices are 960.16, 561.16, 561.16, and 342.16 ^{o}K respectively. The fuel gap is smeared into the clad material and endcap is not considered. The fuel composition, for fresh, equilibrium and discharged stage are the same as employed in Ref. 3.

For HELIOS lattice calculations, 190-group adjusted library was used. One half symmetry of the core model was designed two dimensionally, and the specular reflective boundary condition was used to all external surfaces of the cell.

For fuel dimension change, the standard lattice pitch was fixed for the DUPIC fuel and a change was made in the fuel in such a way that it provides us the actual pitch as that of standard DUPIC lattice. The radius of the fuel rods was decreased keeping all fuel ring radius size conserved and system K_{inf} was noted against each change. These changes were performed for DUPIC fuel as well as for natural uranium fuel in 43-cluster geometry. Figure 2 shows the system multiplication factor with the decreased fuel rod radius for DUPIC and natural uranium fuel. Based on these results it was selected to reduce the fuel dimensions by 1.0 mm in radius keeping the burnup of the fuel at discharge. With this dimension, lattice pitch versus system reactivity is shown in Fig. 3,

which clearly indicates that, almost the same dimension of lattice pitch as for DUPIC standard fuel. Based on these dimensions, cell geometry for this case is shown in Fig. 1

IV. COMPARATIVE ANALYSIS

In first case, standard DUPIC lattice was used. In second case the fuel dimensions were changed. Next we will refer these cases as case-1, case-2, respectively. Firstly effect on the burnup was studied. Then the important parameters such as void coefficient, Doppler coefficient, and moderator temperature coefficient have been calculated and compared for different fuel burnup conditions. Three stages were taken as fresh, equilibrium and discharge. The change in reactivity was calculated as:

$$a\,(mk) = 1000 \times \left[\frac{1}{k_{normal}} - \frac{1}{k_{perturb}} \right]. \qquad (1)$$

The temperature coefficient of reactivity was calculated as:

$$a_T\,(pcm/^oK) = \frac{k_1 - k_2}{T_1 - T_2} \times 10^5. \qquad (2)$$

where;
k_1, k_2 are k_{inf} for normal and perturbed, respectively.
T_1, T_2 are temperature (^oK) for normal and perturbed, respectively.

IV.A. Effect on Burnup

For both cases, the lattice calculations were done from zero to 20000 MWD/T burnup. The system reactivity vs burnup for these cases is shown in Fig. 4. From this result, it is known that discharge burnup for case-1, and case-2 is 15000, and 14920 MWD/T, respectively.

IV.B. Void Coefficient of Reactivity

The coolant density has been changed from 0.8% to 0.0001% to investigate the effect of void. The results for fresh condition in both cases are shown in Fig. 5. The values of change in the reactivity at 0.0001% void condition with change in burnup are given in Table I. The change in void coefficient of reactivity for case-1 and case-2 is 9.50694 and 8.67019 mk, respectively.

IV.C. Doppler Coefficient of Reactivity

For this calculation, the fuel temperature was changed from 293.16 to 1473.16 ^{o}K. At fresh condition, the changes in Doppler coefficient of reactivity are shown in Fig. 6. From these results, it is known that the Doppler coefficient of reactivity for case-1 is 1.087 pcm / ^{o}K during temperature decreasing from operating condition and –0.528 pcm / ^{o}K in case of temperature increasing while for case-2 this parameter became 0.823 pcm / ^{o}K in temperature increasing and –0.359 pcm / ^{o}K for temperature increasing. The Doppler coefficients for both decreasing and increasing side are given in Table II.

]IV.D. Moderator Temperature Coefficient of reactivity

Moderator temperature coefficient of reactivity has also been analyzed and the temperature change was made from 283.16 to 373.16 ^{o}K. For fresh condition the moderator temperature coefficients for both cases are shown in Fig. 7. In case-1, the moderator coefficient of reactivity is 3.93 pcm / ^{o}K during temperature decrease from operating condition and –5.065 pcm / ^{o}K for temperature increase while for case-2 this parameter became in case of temperature decrease is 6.237 pcm / ^{o}K and –6.968 pcm / ^{o}K for temperature increase. The effect of burnup on this parameter is given in Table. III.

V. SUMMARY

Lattice sensitivity due to fuel rod change has been carried out for DUPIC fuel using the two-dimensional transport theory based computer code HELIOS. Fuel rod diameter was modified with optimum diameter with existing lattice pitch. Standard fuel rod and rod change cases were studied for the physics parameter and time dependent behavior and then compared with each other.

For standard fuel rod dimension case (case-1), we have calculated the discharge burnup as 15000 MWD/T. Under fresh condition, the change in void coefficient of reactivity is about 9.5 mk. The Doppler coefficient of reactivity for this case is 1.087 pcm / ^{o}K during temperature decreasing from operating condition and –0.528 pcm / ^{o}K in case of temperature increasing. The moderator coefficient of reactivity is 3.93 pcm / ^{o}K during temperature decrease from operating condition and –5.065 pcm / ^{o}K for temperature increase.

In case-2 at fresh condition the system reactivity is increased by 8 mk as compared to case-1. The discharge burnup for this case is 14920 MWD/T, which is 80 MWD/T less than case-1. Although burnup decreased a little bit but there is an increase in system reactivity, and also decrease of heavy element by about 32% that will ultimate reduce the radioactive waste. For case-2, at fresh condition, the change in void coefficient of reactivity is 8.67019 mk which is about 8% less as compared to case-1. The Doppler coefficient of reactivity for this case is 0.823 pcm / ^{o}K in temperature decreasing and –0.359 pcm / ^{o}K for temperature increasing. This decrease is about 21%, in case of decrease in temperature and 36 % for increase in temperature as compared to case-1. The moderator temperature coefficient of reactivity in case of temperature decrease is 6.237 pcm / ^{o}K and –6.968 pcm / ^{o}K for temperature increase. Comparing to case-1, this effect is about 58% for temperature decreasing and 38% for temperature increasing. In case-2, the values of physics parameters are more favorable as compared to case-1. Because, these results are based on the reactor physics aspect only, it is recommended that the thermal hydraulic, mechanical and economic analyses should be performed in future.

ACKNOWLEDGEMENT

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Table I

Comparison of Change in Void Coefficient of Reactivity (from 80% - 0.0001%)

Burnup	Case-1 (mk)	Case-2 (mk)
Fresh	9.50694	8.67019
Equilibrium	12.4296	10.30691
Discharge	14.97878	10.02141

Void = $(1/k_{(0.8\%)}-1/k_{(0.0001\%)})*1000$

Table II

Comparison of Fuel Temperature Coefficient of Reactivity (α_T)

Burnup	Case-1 (pcm / ^oK)	Case-2 (pcm / ^oK)
Fresh	1.087 (-0.528)*	0.823 (-0.359)
Equilibrium	0.753 (-0.191)	0.5292 (0.060)
Discharged	0.195 (0.347)	0.0315 (0.421)

* In brackets the reactivity effect due to temperature increase.

$\alpha_T = ((k_{(Tsteady)}-k_{(Tperturbed)})/(T_{steady}-T_{perturbed})) * 10^5$

Table III

Comparison of Moderator Temperature Coefficient of Reactivity (α_m).

Burnup	Case-1 (pcm / ^oK)	Case-2 (pcm / ^oK)
Fresh	3.93 (-5.065)*	6.237 (-6.968)
Equilibrium	7.407 (-8.774)	9.492 (-10.516)
Discharged	13.136 (-14.65)	14.711 (-15.87)

*In brackets the reactivity effect due to temperature increase.

$\alpha_m = ((k_{(Tsteady)}-k_{(Tperturbed)})/(T_{steady}-T_{perturbed})) * 10^5$

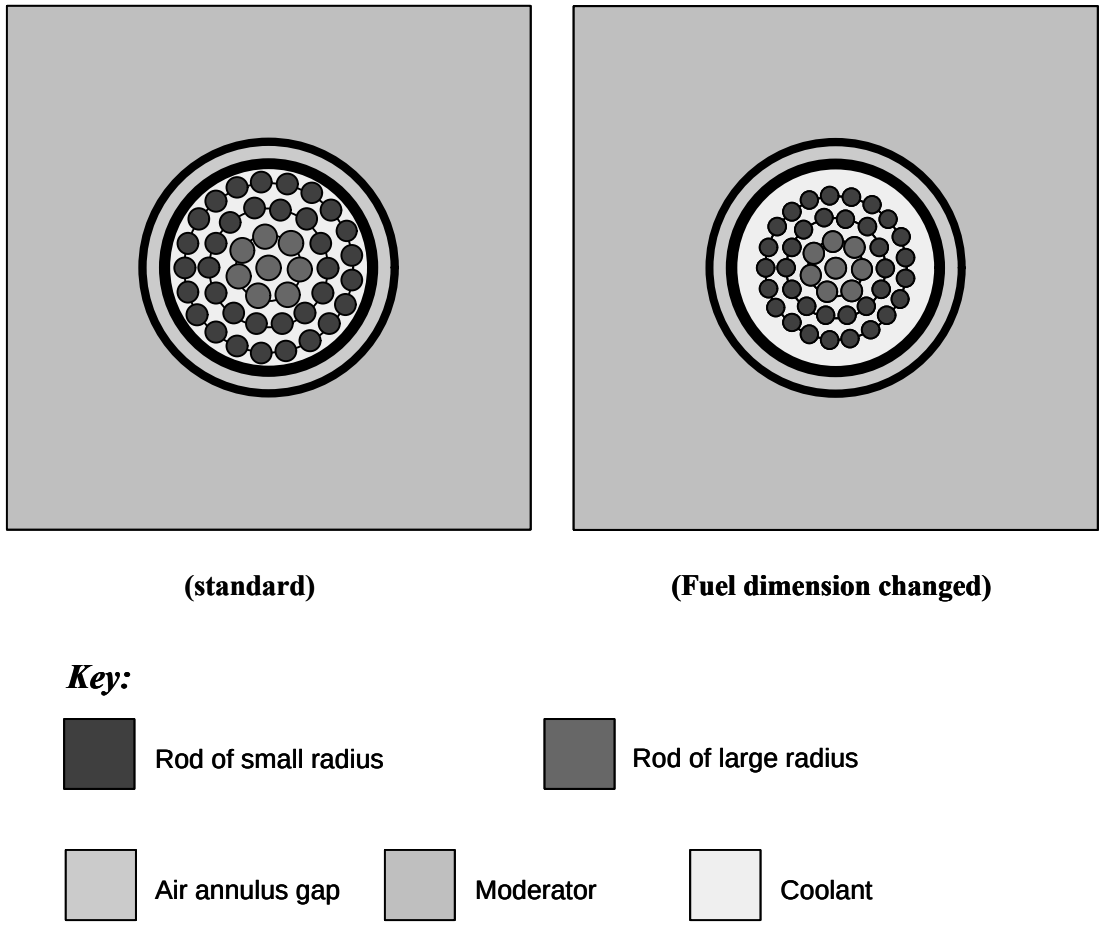

Figure 1 Configuration of the DUPIC fuel lattice with 43-element fuel bundle presenting both cases (not scale).

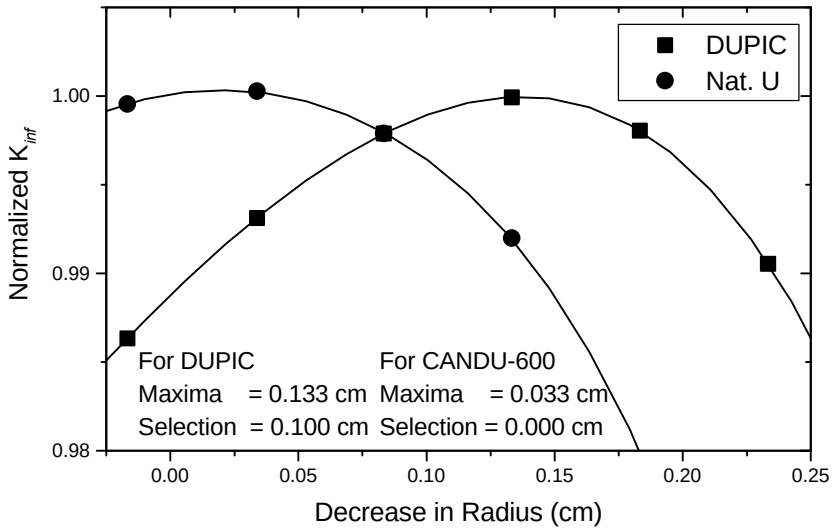

Figure 2. Decrease in fuel rod radius vs normalized K_{inf} for natural uranium and DUPIC fuel.

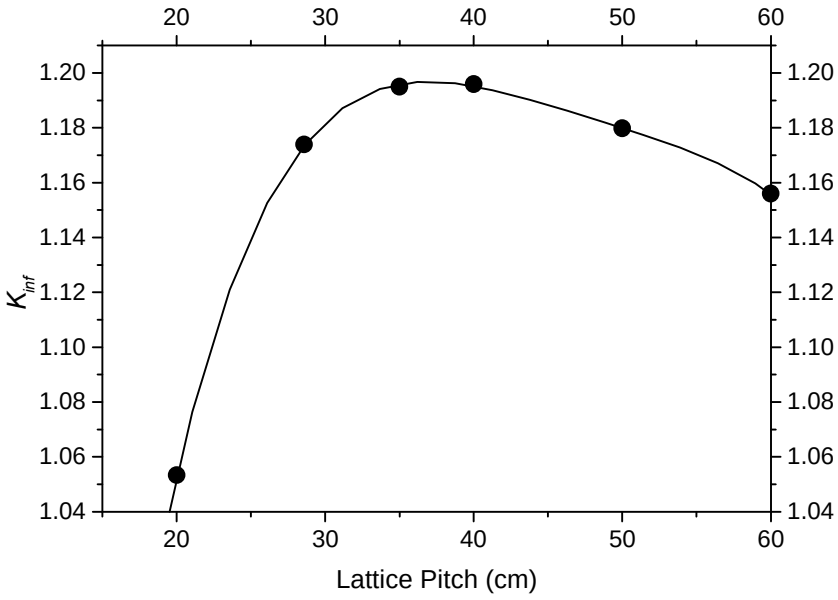

Figure 3. Lattice pitch vs system K_{inf} after fuel dimension changed in standard 43-elemens fuel bundle.

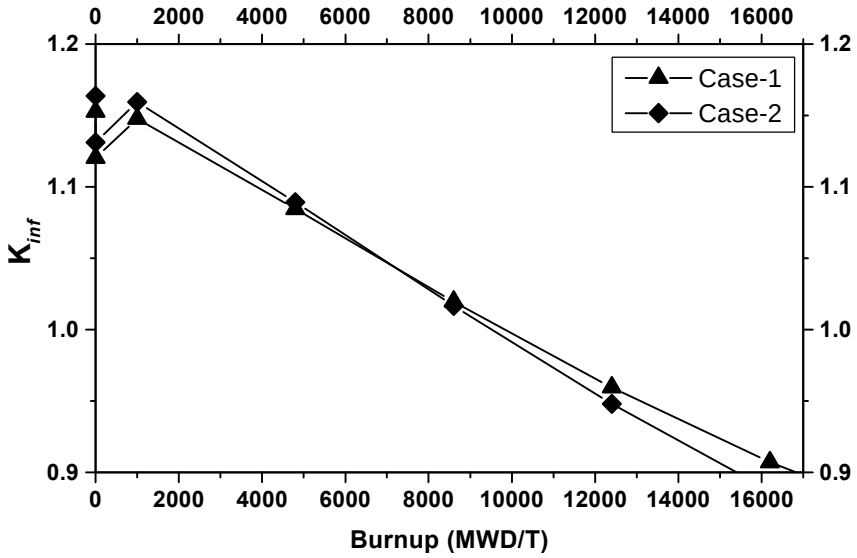

Figure 4. System K_{inf} change on burnup for standard, and fuel dimension change cases.

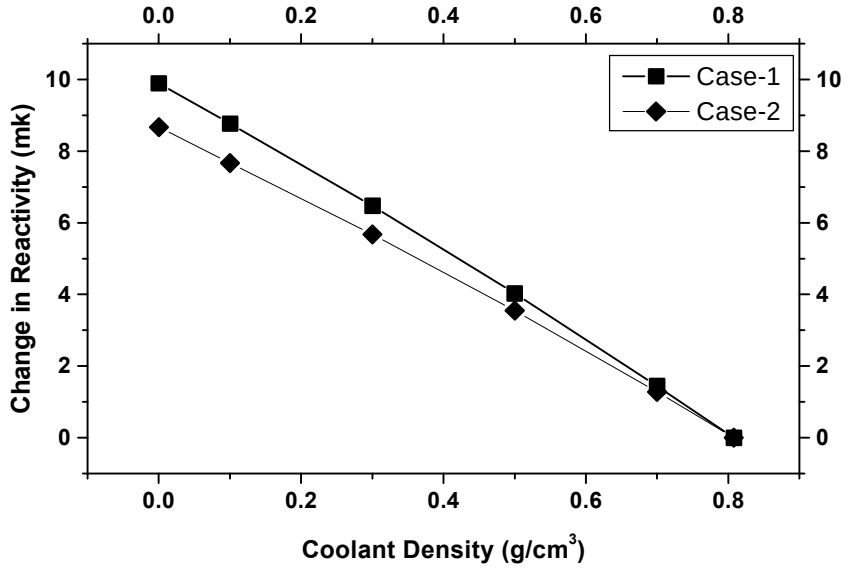

Figure 5 Coolant density vs change in system reactivity for standard, and fuel dimension change cases.

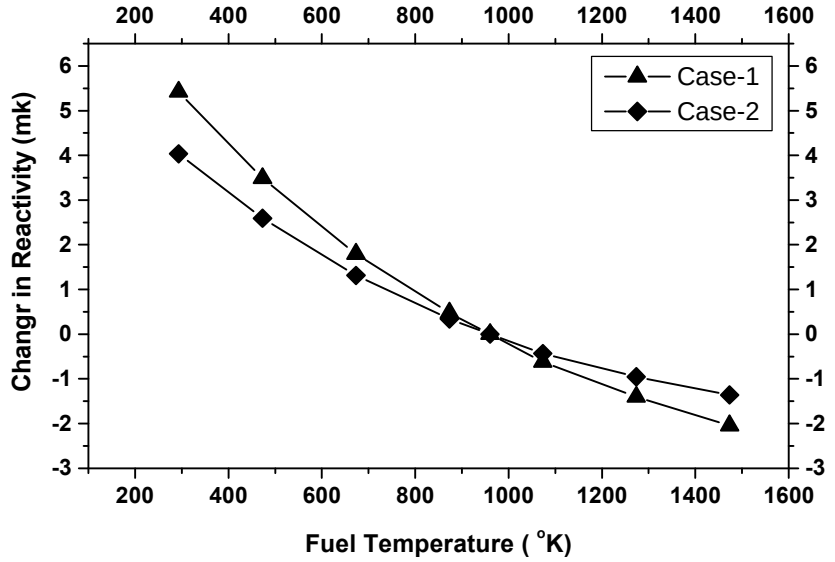

Figure 6 Fuel temperature vs change in system reactivity for standard, and fuel dimension change cases.

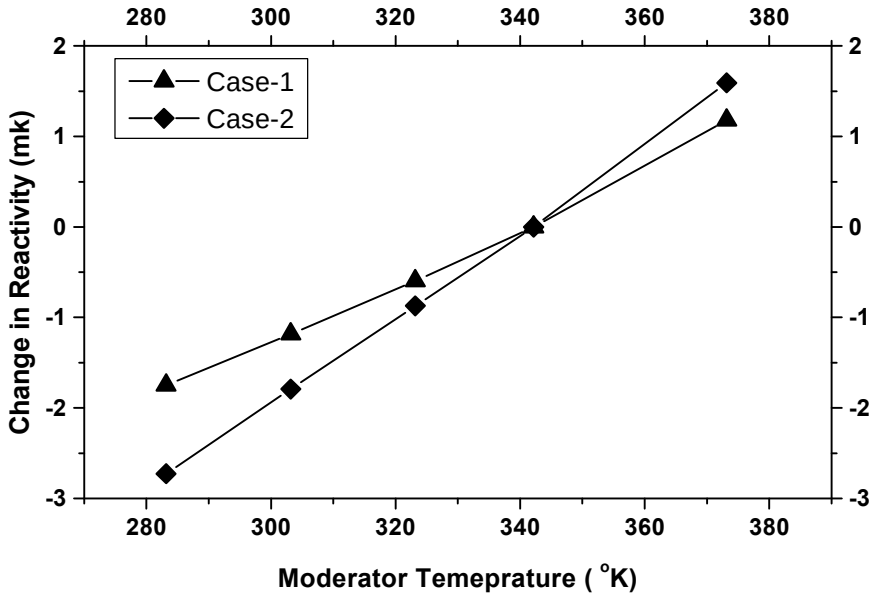

Figure 7 Moderator temperature vs change in system reactivity for standard, and fuel dimension change cases.