AECL

EFFECT OF PRESSURE-TUBE CREEP ON THE VOID EFFECT OF A CANDU FUEL BUNDLE

by

M.S. Milgram*

ABSTRACT

The results of a study of the effect of an increase in pressure tube diameter due to radiation induced creep on the total void effect of a CANDU 37-element fuel bundle are reported here. The calculation presupposes that the pressure tube diameter increases by ~6%, and considers how the lattice reactivity and nuclide components of reactivity change between the two extremes in both voided and cooled conditions. MCNP4A is used as the basis for predictions. Fresh fuel nuclide compositions are used throughout.

For the model considered, MCNP4A predicts that the void effect increases by about 3 mk, due to a decrease in lattice reactivity for the case of the cooled channel as the pressure tube diameter increases by 6% and the flux distribution varies. This is due to the fact that the yield rate from U235 decreases because the fuel is in a lower flux region. No significant reactivity differences are predicted to occur for the voided channel in the presence of diametral creep. An experimental test of the predictions is proposed.

Reactor and Radiation Physics Branch
Chalk River Laboratories
Chalk River, Ontario

1997 August

---

* milgramm@aecl.ca

1. INTRODUCTION

The phenomenon of fuel-channel creep in CANDU reactors is a well-known occurrence that appears as a gradual lengthening of the fuel channel over the life-span of the reactor. Another manifestation of this effect comes in the form of a diametral increase of the pressure tube. While the axial elongation should have a negligible effect on the physics of the nuclear processes taking place inside the reactor at the lattice cell level, since visible changes occur well outside the active core and are homogeneous in the axial direction, there is the possibility that pressure-tube diametral expansion might induce slight changes in the transport of neutrons within the reactor, since it will induce variations in important parameters such as the fuel/coolant material density ratio, and the effective distance of the fuel from the moderator. Although diametral creep appears to be small (~2.5–5% is the design value[1] for pressure tubes), it is likely that resulting alterations to operating flow regimes and control margins may be important. In particular, changes to reactivity coefficients may be significant, since these coefficients typically arise from a delicate balance among an aggregation of competing processes, any one of which might be transformed by diametral creep, thereby inducing a significant change in the coefficient.

The purpose of this paper is to report on a high-accuracy MCNP[2] (Monte-Carlo) calculation of the infinite lattice void reactivity coefficient of standard 37-element CANDU fuel in the presence of diametral creep with the intent to:

- provide a reasonably accurate estimate of the magnitude of the change in the total void effect; and
- indicate how the various important components of the total void effect change as the pressure-tube diameter increases.

2. THE MODEL

A standard model of a 37-element CANDU fuel bundle used in previous MCNP investigations was modified for this study. The data library was based entirely on ENDFB-6, with one check using ENDFB-5 to make sure no significant effects could be attributed to this choice of nuclear data. The modified model employs two-fold symmetry about a vertical half-plane with explicit representation of 18.5 fuel pins within a standard CANDU pressure tube, to model a lattice cell (Figure 1(a)). The hexahedral cell is bounded by reflective planes representative of the standard CANDU lattice pitch parallel to the pressure-tube axis, and axially positioned 25 cm apart for convenience. With this choice of symmetry axes, the model becomes equivalent to that of an infinite lattice, so the results to be quoted only shed light on the various components of k_∞. By using two-fold symmetry, it is possible to translate the interior of the pressure tube in the radial direction by an amount equal to approximately one-half of the diametral increase, and thus to also investigate the effect of offsetting the fuel from its normal position at the centre of symmetry. Figure 1(b) illustrates this representation. Unless stated otherwise, the geometry of Figure 1(a) is used in most of the analyses that follow.

The material content of the lattice cell is a reasonable facsimile of a true lattice cell, with moderator D_2O purity equal to 99.83 a/o, coolant D_2O purity of 99.72 a/o, fuelled with natural uranium (at room temperature, since this is not expected to significantly affect the total void effect). Thermal neutron scattering kernels for water at 300°C and 100°C were included in the coolant and moderator, respectively, where respective operating densities of 0.804 g/cm^3 and 1.084 g/cm^3 were used to define the ***cooled*** cases. The ***voided*** cases were defined by changing the operating coolant densities to 0.001 g/cm^3.

To model the diametral creep, the pressure tube, normally an annular region of 5.17 and 5.62 cm inner and outer radius, was split into three annuli, with inner and outer radii 5.17, 5.47, 5.62 and 5.90 cm, respectively. The outermost of these annuli extends into the annular gap. To define the ***regular*** cases, the inner two annuli were filled with pressure-tube material (Zr-2.5% Nb) and the outermost was voided (annular gap material). In the ***crept*** cases, the inner annulus was filled with coolant and the outermost two annuli were filled with a pressure-tube material. For the ***crept*** cases, these radii represent a 5.8% increase in pressure-tube diameter with conservation of pressure-tube material. The ***offset*** cases were defined by translating the fuel in the ***crept*** cases (***cooled*** and ***voided***) by 0.45 cm downwards, so that the bottom fuel element just touches the revised inner radius of the crept pressure tube (Figure 1(b)).

Defining the geometry in this way allows the perturbation option embedded in the new version of MCNP (MCNP4XQ[2]) to be investigated. As will be shown later, this capability cannot be used for this problem.

The investigation proceeded by predicting the values of k_∞ for each of three geometrical arrangements (***regular***, ***crept*** and ***offset***) with two coolant densities for each (***cooled*** and ***voided***). This was done using the KCODE procedure[2] for each case by initially imposing an arbitrary neutron source distribution on the problem (a single neutron emanating from a single point in the innermost fuel pin) and progressively iterating each computed distribution 50 times with a fixed number of source points, and increasing the number of source points through 7 generations of 50 iterations each (100, 1000, 2000, 5000, 10 000, 20 000 and 50 000) until a final stable source distribution was found for each of the six cases. As will be seen later, it is important that the source distribution for each case be found independently, since it is the distinctiveness in source distribution that appears to be driving the differences that will be reported here. This is in reasonable accord with physical reality, where the source distribution for each of the cooled cases would inevitably establish itself over the years during which the creep process occurs. In the voided cases, it is reasonable to assume that the source distribution will establish itself within a time period characterized by neutron lifetimes (milli-seconds), and thus the source distribution belonging to a voided case should be the one chosen to characterize a voided channel (this also happens to coincide with the choice made in deterministic codes where alternatives are not usually available).

The results quoted here for the six ***regular*** and ***crept*** cases are all calculated, using source distributions obtained in the manner described above, to start a new iterative procedure which consists of 410 iterations of 20 000 neutrons each, omitting the first 10 iterations from the final results. They each represent the cumulative tally of 8 million histories (8 Mh). For k_∞, this is more than enough information to obtain accurate estimates, but for four cases an attempt was also made to isolate changes in the components of k_∞ and for this purpose this large number of histories is seen to marginally suffice. For the ***offset*** cases, only estimates of k_∞ are quoted, since the results are similar to the ***crept*** cases.

3(a). K_∞ RESULTS

Estimates of k_∞ for each of the six cases and total void effect ρ defined by

$$\rho = k_\infty(\text{voided}) - k_\infty(\text{cooled})$$

are listed in Table 1. It is immediately obvious that the value of ρ in the ***crept*** cases increases by about 3 mk compared to the ***regular*** cases, corresponding to a 13% increase, due to the 5.8% pressure-tube diametral creep. The value of ρ predicted for the ***regular*** cases is consistent with results that have been obtained elsewhere[3], verifying that the methodology is reasonable. Strikingly, a change in reactivity with creep is seen only in the ***cooled*** cases, where the value of k_∞ decreases by about 3 mk between the two extremes (***regular*** and ***crept***); in the ***voided*** cases no similar decrease is seen. This behaviour is evaluated in Table 1 by proffering a new 'creep parameter' ζ defined by

$$\zeta = k_\infty(\text{regular}) - k_\infty(\text{crept})_{|\text{constant coolant conditions}},$$

Table 1. Reactivity and reactivity coefficients for cooled and voided cells with and without diametral creep. All errors, reported in mk, represent 1 standard deviation (sd)

	regular		***crept***			
			centred		***offset***	
	cooled	***voided***	***cooled***	***voided***	***cooled***	***voided***
k_∞	1.1276±0.1 1.1286±0.1*	1.1491±0.1	1.1246±0.1	1.1489±0.1	1.1248±0.1	1.1487±0.1
ρ (mk)	**21.5±0.2**			**24.4±0.2**		
ζ (mk)		**3.0±0.2**	**0.1±0.2**			

* ENDFB-5 library

introduced to enumerate these differences more thoroughly. It is also clear from the table that the predictions for the ***offset*** cases are virtually indistinguishable from those for the ***crept*** cases. Therefore, only the former four cases were investigated in detail.

It is important to emphasize that only reactivity effects are considered here, in contrast to other analyses where reactivity differences are normalized to the reactivity (k_∞) itself. The rationale for the latter approach is that normalized reactivity coefficients are required for a time-dependent calculation of reactor reactivity change after a voiding perturbation. Here, in addition to predicting reactivity effects, one of the goals is to isolate differences between estimates of components of reactivity and reactivity itself, for each of the scenarios. The simplest way to focus on those differences is to study the basic quantities generated by a lattice code, which are the material reaction rates normalized to a fixed number of absorptions in the lattice cell.

3(b). COMPONENTS OF REACTIVITY - CREEP

In this section, I investigate the reasons for the striking result that the creep parameter is important only for ***cooled*** cases, by extracting the components of reactivity by material and searching for significant differences among them. Consider Table 2(a), which separates each of the main reaction rates "R" as a fractional contribution to a fixed number of absorptions "A" , where absorption is defined as the sum of capture + fission + (n,2n).

In this table, the total reactions of each of the major materials that constitute a standard CANDU bundle have been apportioned to that material as (δR/A); they are presented in units of thousandths (.001), which, for convenience, will be referred to as one mk, although *most of the time this unit does not refer to a reactivity scale*. Within the statistical accuracy of the calculation, the proportions must sum to 1000 mk. The yield

Table 2(a). The contribution (δR/A) of each of the main materials to the total absorption A of the ***cooled, regular*** lattice cell. All units are normalized to 1000 Absorptions; 1 sd statistical errors are indicated by brackets [ ].

	captures	fissions	(n,2n)	fission yield[1]
U238	395.0 ± [0.1]	20.38 ± [0.01]	0.717 ± [0.003]	56.57 ± [0.03]
U235	77.16 ± [0.02]	438.1 ± [0.1]		1065.6 ± [0.3]
sheath	9.139 ± [0.005]			
coolant	0.811 ± [0.001]			
pressure tube	29.34 ± [0.01]			
calandria tube	11.93 ± [0.01]			
moderator	15.21 ± [0.01]		2.250 ± [0.004]	
TOTALS/1000	1.0000 ± [0.0002]			1.1280 ± [0.0003]
best k_∞ estimate				1.1276 ± [0.0001]

columns (fission yield† + 2 (n,2n)) represent the predicted multiplication for this lattice cell, and sum to an independent estimate of the cell reactivity. For consistency, this can be compared to the best MCNP k_∞ estimate obtained independently, which is presented in the last row. The agreement is within the statistical bounds (although the yield estimate is usually higher than the k_∞ estimate in the cases to be presented), and demonstrates that the assignment of reactivity components is consistent.

Table 2(b) gives the same information for the case of a ***cooled, crept*** lattice cell. Table 2(c) lists the differences (***regular*** - ***crept***) between Tables 2(a) and 2(b), and therefore represents the material contributions to the creep parameter ζ for a cooled lattice cell.

Table 2(b). The contribution ($\delta R/A$) of each of the main materials to the total absorption of the ***cooled, crept*** lattice cell. All units are normalized to 1000 Absorptions; 1 sd statistical errors are indicated by brackets [ ].

	captures	**fissions**	**(n,2n)**	**fission yield**
U238	394.80 ± [0.1]	20.33 ± [0.01]	0.713 ± [0.003]	56.42 ± [0.03]
U235	76.94 ± [0.02]	436.97 ± [0.1]		1062.87 ± [0.3]
sheath	9.16± [0.005]			
coolant	1.03 ± [0.001]			
pressure tube	30.36 ± [0.01]			
calandria tube	12.16 ± [0.01]			
moderator	15.40 ± [0.01]		2.15 ± [0.004]	
TOTALS/1000	1.0000 ± [0.0002]			1.1250 ± [0.0003]
best k_∞ estimate				1.1246 ± [0.0001]

From a study of Table 2(c), a number of things can be inferred. First, and most significant, the contribution to ζ from changes in the proportion of captures ascribed to U238 reactions is statistically consistent with zero. This suggests that the creep parameter owes its existence in only a minor way to changes in U238 capture rate as the pressure tube expands. A statistically significant decrease in the U235 capture rate with diametral creep, and a concomitant decrease in the contribution of the fission and yield components from U235, shows why the overall reactivity decreases with creep. U235 and U238 occupy the same physical location in the bundle, and show changes of similar small magnitude in their contribution to ($\delta A/A$), suggesting that the flux of thermal neutrons in the bundle is redistributing itself spatially as the pressure-tube creeps, producing less overall U235 reactions, particularly fissions. This is consistent with the observation in Table 2(c) that the decreased proportion of absorptions assigned to fuel materials U235

† fission yield $\equiv \nu\Sigma_f$

Table 2(c). Differences between the corresponding elements of Tables 2(a) and 2(b) and 1 sd errors, all expressed in mk for the cooled cases.

	captures	fissions	(n,2n)	fission yield
U238	0.2 ± [0.2]	0.05 ± [0.02]		0.15 ± [0.04]
U235	0.22 ± [0.03]	1.1 ± [0.2]		2.7 ± [0.4]
sheath	-0.021 ± [0.005]			
coolant	-0.220 ± [0.002]			
pressure tube	-1.02 ± [0.02]			
calandria tube	-0.23 ± [0.01]			
moderator	-0.19 ± [0.01]		0.08 ± [0.006]	
TOTALS	-0.03 ± [0.4]			3.0 ± [0.4]

and U238 is compensated by an increase in the proportion of reactions in the structural elements of the lattice cell, notably the pressure-tube. This too agrees with the expectation that the crept pressure tube, now in a region of slightly higher thermal flux, will absorb a slightly higher proportion of thermal neutrons. The fact that the other structural elements outside the pressure tube also increase their proportionate contribution to the change in absorption rate suggests that the thermal flux distribution across the cell is also changing in the presence of diametral creep. The remaining challenge is to verify that this effect is not seen for the case of voided channels.

To investigate this further, consider Tables 3(a-c), which present the same information as their counterparts in Tables 2(a-c), but for the corresponding ***voided*** cases. Perusal of Table 3(c) shows the requisite cancellation of changes in fractional absorption, which are unaccompanied by any large change in the fission component in the presence of creep. In contrast to the ***cooled*** case, there is evidence for a minor amount of ***decreased*** absorption

Table 3(a). The contribution (δR/A) of each of the main materials to the total absorption of the ***voided, regular*** lattice cell. All units are normalized to 1000 absorptions; 1 sd statistical errors are indicated by brackets [ ].

	captures	fissions	(n,2n)	fission yield
U238	388.7 ± [0.1]	22.55 ± [0.01]	0.780 ± [0.003]	62.53 ± [0.03]
U235	78.13 ± [0.02]	443.8 ± [0.1]		1079.5 ± [0.3]
sheath	8.912 ± [0.004]			
coolant	—			
pressure tube	28.58 ± [0.01]			
calandria tube	11.20 ± [0.01]			
moderator	14.88 ± [0.01]		2.515 ± [0.005]	
TOTALS/1000	1.0000 ± [0.0002]			1.1486 ± [0.0004]
best k_∞ estimate				1.1491 ± [0.0001]

from the pressure tube after creep, and the added absorptions in the moderator and calandria tube that occur in the presence of cooling are not significant in the voided case. This suggests that the change in radial flux distribution, so noticeable in the cooled case, is small for the voided case. Any additional absorption in the pressure tube that was present with coolant is missing with voiding, and works the other way to a small extent. A small mixture of opposing effects from uranium isotopes makes up the difference, hinting that spectral changes come into play in this instance.

Table 3(b). The contribution ($\delta A/A$) of each of the main materials to the total absorption of the ***voided, crept*** lattice cell. All units are in mk; 1 sd statistical errors are indicated by brackets [ ].

	captures	**fissions**	**(n,2n)**	**fission yield**
U238	389.1 ± [0.1]	22.52 ± [0.01]	0.781 ± [0.003]	62.47 ± [0.03]
U235	78.21 ± [0.02]	444.2 ± [0.1]		1080.6 ± [0.3]
sheath	8.939 ± [0.004]			
coolant	—			
pressure tube	27.67 ± [0.01]			
calandria tube	11.15 ± [0.01]			
moderator	14.89 ± [0.01]		2.537 ± [0.005]	
TOTALS/1000	1.0000 ± [0.0002]			1.1497 ± [0.0003]
best k_∞ estimate				1.1489 ± [0.0001]

Table 3(c). Differences between the corresponding elements of Tables 3(a) and 3(b) and 1 sd errors, all expressed in mk for the ***voided*** cases.

	captures	**fissions**	**(n,2n)**	**yield**
U238	-0.4 ± [0.2]	0.03 ± [0.02]	—	0.06 ± [0.05]
U235	-0.08 ± [0.03]	-0.4 ± [0.2]		-1.1 ± [0.4]
sheath	-.037 ± [0.006]			
coolant	—			
pressure tube	0.91 ± [0.01]			
calandria tube	.05 ± [0.01]			
moderator	0.00 ± [0.01]		-0.02 ± [0.007]	
TOTALS	0.01 ± [0.3]			-1.0 ± [0.4]

4. MONTE-CARLO PERTURBATION METHODS

The newest version of MCNP, identified as MCNP4XQ, was used for this study; this version contains a new option that allows the investigation of the effects of small changes

in a problem without the necessity of running two different cases and studying differences, as has been done here. The method works by introducing alternative paths for particles being tracked, and adding the alternatives as first- and second-order perturbations to reference tally quantities. Unfortunately, in order for this method to work, a standard reference case (unperturbed configuration) must be used for the main transport calculation, and the perturbations must not affect the reference transport problem to a significant degree. As has been seen in previous sections, this is not the case here, since the redistribution of material that constitutes creep induces a change in the flux distribution which cannot be modelled by tracking the reference case. In other words, the change in neutron flux distribution affects the source distribution calculated by multiple iterations in the KCODE method (see Section 2), so if the perturbation of the source in the presence of creep is not included in the KCODE iterations, then MCNP will essentially be calculating the effect of perturbations in an unperturbed source distribution. If the perturbation of the spectrum is the genesis of the effect, as claimed here, nothing will be seen. This was observed when the MCNP4XQ perturbation method was used to analyze this problem. As the number of KCODE iterations increased, the reactivity difference between ***regular*** and ***crept*** cases vanished. This behaviour provides support for the thesis that a redistribution of flux is at the root of the non-zero value of ζ. It also emphasizes that MCNP Monte-Carlo perturbation techniques are only well-suited for fixed source problems, and should be used with KCODE reactivity iterations only after careful thought.

5. SUMMARY

MCNP predictions of the effect of coolant channel voiding in the presence of diametral creep have been presented resulting in an estimate of a ~3 mk change in reactivity for a ~6% change in pressure-tube radius, for fresh fuel. The effect is entirely due to a decrease in reactivity of the cooled channel as the pressure-tube diameter increases, and the radial thermal flux distribution varies in concert. This decreases the U235 yield rate relative to the absorption components of the fuel, since the outer ring of the fuel is in a lower thermal flux and the structural components are in a higher thermal flux. The effect is not due to a large variation in the U238 capture rate.

- For an infinite CANDU lattice of fresh fuel with creep limits of about 3½%, the cooled channel reactivity loss as defined in this report should be about 1.5 mk, assuming linear interpolation can be used. For channels with 5% creep limits, the estimate is 2.5 mk.

- For an operating reactor, the change in void reactivity with creep may be considerably different than the values quoted here, since the effect of fuel burnup has been omitted from this work, as has the change in leakage on voiding.

- Since the effect on reactivity appears only as a decrease in reactivity of the channel subject to cooled and normal operating conditions, the predictions contained here can be tested by inspecting the histories of operating reactors. A 1 mk overall decrease in reactivity should result in a decrease in fuel burnup of ~200 MWd/Mg; this is large

enough that it might be visible as a trend over the reactor's operating lifetime if fuelling statistics for an operating reactor with significant diametral creep were to be correlated with the amount of that creep, after allowing for changes in other factors such as moderator and coolant purity, moderator poison, etc.

6. REFERENCES

1. A. Villamagna, private communication.

2. J. Briesmeister, (Ed.), "MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A", LA-12625 (1988); unpublished revision for MCNP4XQ (1996).

3. M. Milgram, "Void Reactivity Predictions for an Infinite Lattice of 37-Element CANDU Fuel", presented at the CNS Fourth International Conference on Simulation Methods in Nuclear Engineering, Montreal (1993), Session 4c, Volume 2.

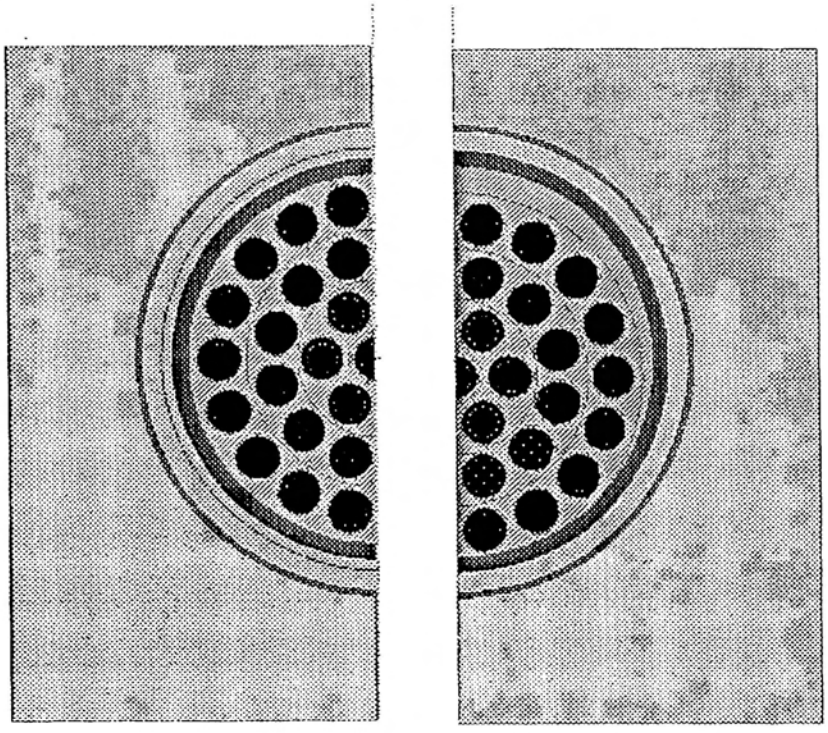

Figure 1(a). Cross section of 37-element ***regular*** CANDU fuel bundle, no creep.

Figure 1(b). Cross section of 37-element standard CANDU fuel bundle, with ***creep***, and geometry ***offset.***