20th Annual Conference of the Canadian Nuclear Society
Montreal, Quebec, Canada / May 30 - June 2, 1999

MCNP Analysis of a D_2O-Filled Fuel-Channel Penetration Through the End Shield

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ABSTRACT

Using a CANDU 9 reactor fuel channel as the model, a Monte Carlo analysis was performed which showed that, when the stainless-steel end-shield plug in the fuel channel was replaced with heavy-water coolant, the radiation dose rate from the fission product decay gammas was 2.4 mrem/h at the end plane of the end-fittings 24 hours after reactor shutdown.

INTRODUCTION

In a CANDU™ reactor, the end shield was designed to stop radiation leaving the ends of the reactor and allow personnel access to the fuelling machine vaults with the reactor shut down. Fuel-channel penetrations through the end shields allow coolant to flow through, and permit refuelling operations.

The fuel-channel penetrations are formed by thin-walled lattice tubes, and are plugged by stainless-steel end fittings that can be thought of as thick-walled tubes and shield-plug assemblies. Except for a narrow gas annulus between the lattice tube and the end fitting, the remaining volumes are filled with heavy-water coolant. The shield plugs were designed to reduce radiation streaming from the fuel channels at the ends of the reactor core. The fuel-channel assembly, showing the end fitting, liner tube and shield plug, is shown in Figure 1. The end fittings project out of the end shield and personnel access is possible only to the end plane of the end fittings.

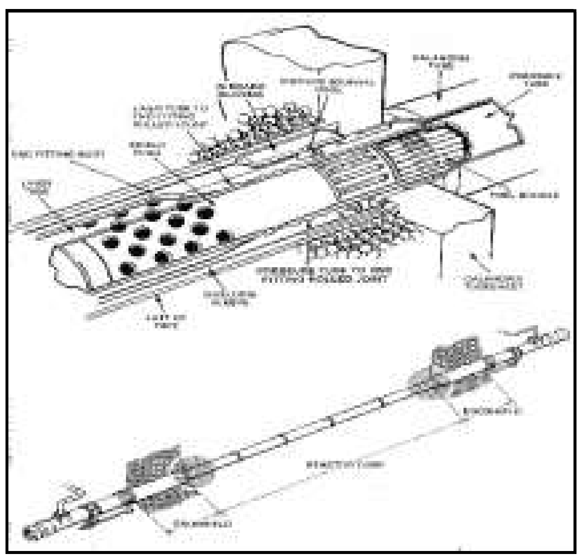

Figure 1 End Shield of the CANDU 9 Fuel Channel

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If one ignores the deposited activity from corrosion products in the feeder piping, the principal source of radiation—with the shield plug present—at the end plane of the end fittings 24 hours after shutdown is ^{60}Co generated by neutron activation of cobalt impurities in the fuelling-machine-side tube sheet (FMTS). As a follow up to the recommendation in an earlier paper [1], this paper investigated the reactor shutdown condition—with and without the shield plug in the fuel channel—to demonstrate that the ^{60}Co-activation and the fission-product—24 hours after reactor shutdown—decay-gamma dose rates remain below the design value of 25 mrem/h. That is, from the perspective of dose rates after reactor shutdown, the shield plugs in CANDU 9 fuel channels penetrations could be eliminated.

BACKGROUND

From a physics viewpoint, replacing the stainless-steel shield plug with D_2O reduces the number of epithermal neutrons penetrating through the end shield, on account of the greater neutron attenuation provided by D_2O. This change increases the number of thermal neutrons scattered from the D_2O-filled regions into the FMTS. Consequently, the thermal-neutron contribution to ^{60}Co activity (2200 m/s absorption cross section of ^{60}Co = 37 barns) in the FMTS increases, but the epithermal neutron contribution (resonance integral above 0.5 eV = 74 barns) decreases. Decay gamma rays from activation of other end-shield components are less important in intensity compared to the ^{60}Co-decay gamma rays from the FMTS.

Another important contribution to the shutdown dose rates at the end fittings is the fission-product decay gammas. When the end-shield plug in the fuel channel is replaced by D_2O, the transmission of fission-product decay gamma rays through the fuel channel increases because of the smaller shielding effect of the heavy water.

In this paper, the dose rates from fission-product decay gammas were calculated. Activation dose rates due to decay gammas from ^{60}Co-activity in the FMTS were also calculated. However the results to date, obtained using Monte Carlo methods, include a large relative error, making it difficult to draw any definitive conclusions from the results at this point. To achieve any significant reduction in the relative error would demand a substantial increase in execution time.

METHODOLOGY

The analysis was based on the end-shield design of the CANDU 9 reactor. The end shield is made up of a region of carbon-steel balls and light water sandwiched between the calandria-side tube sheet and the fuelling-machine-side tube sheet. The tube sheets are essentially two stainless-steel discs located at the axial ends of the end shield.

There are 480 fuel-channel penetrations in the end shields. The penetrations allow D_2O coolant to flow through the end shields during refuelling operations The penetrations consist of thin-walled lattice tubes and are arranged in a square lattice of pitch 28.575 cm. The penetrations are plugged by the stainless-steel end fittings that can be thought of as thick-walled tubes and shield-plug assemblies. Except for a narrow gas annulus between the lattice tube and the end-fitting, the remaining volumes are filled with heavy-water coolant. The fuel-channel assembly showing the end fitting, liner tube and shield plug is illustrated in Figure 1. The feeder pipes were ignored in the analysis.

The analysis modelled the end-shield assembly in square geometry, using the three-dimensional Monte Carlo transport-analysis code MCNP-4B [2]. In addition to the end shield, the model simulated 6 fuel bundles in the fuel channel. Each bundle was modelled with 37 individual elements, arranged in four rings. The 6 fuel bundles were included in the model to simulate the reactor core power during operation. This model generated the axial and radial neutron-source distribution in the core during operation at the start of the neutron-transport calculation through the end shield, to establish the ^{59}Co activation rates in the FMTS.

Material compositions used in the analysis for the different regions are the same as reported in the earlier paper [1] with the ^{59}Co content repeated here: 700 μg/g in the calandria-side tube sheet and the lattice tube, 500 μg/g in the FMTS, 250 μg/g in the end fitting, shield plug and liner tube, 150 μg/g in the homogenised region consisting of carbon steel balls and water.

The focus of the analysis was to evaluate the shutdown dose rates at the end plane of the end fittings. To generate the dose rates, the analysis was divided in two parts, to calculate the decay-gamma contribution from ^{60}Co and the fission-product decay-gamma contribution.

1) **Cobalt Activation in the FMTS**

To calculate cobalt activation rates in the FMTS, the MCNP model was run in criticality mode. This simulated the neutron source spectrum in the core during normal operation and, consequently, the transport of the neutrons through the end shield to the FMTS.

The attenuation of neutrons through the end shield is in the order of 10^7. This presents a difficult transport problem using the Monte Carlo method. To circumvent this difficulty, the problem was handled in two parts. In the first part the criticality calculation established the axial and radial (among fuel rings) distribution of fission neutrons and tracked the transport of these neutrons to a plane source just beyond the calandria-side tube sheet. The angular neutron-flux spectrum in this plane was calculated to have an error of less than ±5%. Based on the particle information for this plane source, 20 subsequent MCNP runs were performed, starting with a different random-number seed, but using the same source distribution. This approach is different than running a single MCNP run with 20 times more histories. A single MCNP run with 20 times more histories would use all particles generated from the source region (at the fuel bundle) as opposed to have 20 shorter MCNP runs, that use all particles generated from the plane source This approach was adopted to save computer time and to improve the statistical precision of the results.

The final activation rates in the FMTS were combined using a statistical formula with a weighting of $1/\sigma^2$, where σ is the standard deviation of the MCNP-calculated results.

The calculated cobalt activation rates at the FMTS, averaged over 20 runs, were then translated to the equivalent ^{60}Co-decay gamma rays; they were input in a separate MCNP model as a fixed source to calculate the dose rates at the end plane of the end fittings.

The above process for calculating the dose rates at the end fittings was repeated twice - the first analysis modelled the shield plug inside the fuel channel, and the second analysis modelled the shield plug replaced by heavy-water coolant. The absolute dose rates were calculated by normalising the MCNP calculation to the CANDU 9 fission power of 2827 MW [4]. And, the calculations were done for a condition in which the ^{60}Co activation reaction rate is equal to the decay rate. This situation is achieved towards the end of life of the CANDU 9 reactor.

2) **Fission-Product Decay Gammas**

To evaluate the impact of removing the shield plug, the fission-product decay-gamma source was first calculated with ORIGEN-S [3] and then input in MCNP as a fixed source. Only the last fuel bundle in the channel was included in the MCNP model.

The decay fission-product spectrum was calculated using ORIGEN-S, an isotope generation and depletion code. The spectrum, in 12 energy groups, was taken from an ORIGEN-S output at 24 hours after shutdown for a 490-kW CANDU 9 bundle irradiated for 150 days. The spectrum is listed in Table 1.

Table 1 Fission-Product Decay-Gamma Spectrum 24 hours after Shutdown

E-min (MeV)	E-max (MeV)	gammas/secon	MeV/second
0.2	0.4	4.68E+15	1.41E+15
0.4	0.9	6.07E+15	3.95E+15
0.9	1.35	4.46E+14	5.02E+14
1.35	1.8	9.71E+14	1.53E+15
1.8	2.2	3.46E+13	6.91E+13
2.2	2.6	4.57E+13	1.10E+14
2.6	3.0	6.94E+11	1.94E+12
3.	3.5	2.71E+11	8.81E+11
3.5	4.	1.67E+09	6.27E+09
4.	4.5	2.34E+08	9.96E+08
4.5	5.	1.74E+09	8.26E+09
5.	10.	8.16E+06	6.12E+07
	totals	1.23E+16	7.57E+15

The spectrum used as the input delayed-gamma source was the same for all fuel rings; no radial variations were considered, because their effect is negligible for our calculations.

Since MCNP calculates the delayed-gamma flux per particle (photon) emitted by the source, this gamma flux was normalized by the product of 3 factors:

- the total number of particles (photons) released from a CANDU 9 fuel bundle irradiated at 490 kW for 150 days after 24 hours decay
- the ratio between the power of the last fuel bundle in the channel (188 kW) and the core-average bundle power (490 kW)
- a reduction factor of 5, which accounts for the axial power shape in the last fuel bundle.

COMBINATION OF DIFFERENT MONTE CARLO OUTPUTS

The transport of neutrons from the reactor through the CANDU-9 end shields proved to be intractable. To calculate dose rates at the end plane of the end fitting, various variance-reduction techniques were investigated before the methodology of following particles from the surface-dump source was adopted.

Two surface-dump source files were generated to calculate the cobalt activation rates for the normal operating conditions, i.e., with and without the end-shield plug present. At the beginning, numerous short MCNP "inactive" cycles were run until the multiplication factor k_{eff} of the lattice converged; the value at convergence was 1.077. Subsequently, "active" Monte Carlo cycles were run to obtain a statistically converged neutron distribution at the surface-source-dump plane. In MCNP terminology, "active cycles" indicates that particle tracks across the regions of the model are tallied, whereas "inactive cycles" keep track of the neutron population in the problem, but not of the particle tracks.

The case with the end-shield plug present in the fuel channel was run for 2300 cycles with 10,000 histories per cycles. The final relative error on the 69 neutron energy groups was less than 5%. For the case with the end-shield plug replaced with heavy-water coolant, the total number of cycles run was 1600, with 10,000 histories per cycle. The final relative error on the 69 neutron energy groups was less than 8%.

Based on these two surface-dump sources, 20 MCNP runs were executed for each source. Each of the 20 cases was started by a different random-number seed, using the DBCN card in MCNP. The results, i.e., the cobalt activation rates at the FMTS, calculated in each of the 20 cases, were then combined statistically using the following weighted formulae.

The mean value of the activation rates is

$$< x > \quad = \quad \Sigma_i \ (x_i \ / \ \sigma_i^2) \ / \ \Sigma_i \ (1/ \ \sigma_i^2)$$

the combined standard deviation of the mean is

$$< \sigma >_{<x>} \quad = \quad 1 \ / \ \Sigma_i \ (1/ \ \sigma_i^2)$$

and the relative error is

$$\varepsilon_{<x>} \quad = \quad < \sigma >_{<x>} \ / \ < x >$$

RESULTS

The total neutron activation rates of ^{59}Co in the FMTS for the scenario with and without the end-shield plug in the fuel channel are listed in Table 2. The reaction rates are given for the entire FMTS per fission neutron.

Table 2. Total ^{59}Co Reaction Rates in FMTS

Activation in the tube sheet	(number of ^{59}Co reactions/n source)	ε
with shield plug	$1.36*10^{-15}$	5 %
no shield plug	$5.61*10^{-16}$	6 %

Both values are given per initial particle in the source, which, for each set (with and without shield plug) of 20 cases is a different neutron source over the surface S. If the dose rates outside the end shield were directly proportional to the ^{59}Co neutron activity in the tube sheet, the gamma dose rate at shutdown would be 59% lower when the shield plug inside the end shield is replaced with D_2O. However, because the tube sheet itself acts as a gamma shield, the analysis also examined the effect of the distribution of ^{60}Co activity in the FMTS. To study this effect, the ^{60}Co activities were scored in 16 different regions of the tube sheet with and without the shield plug. This relative change is given in Table 3 .

Table 3. Relative Change in Activation in the Tube Sheet

	93.54<z<96	96.<z<98.069	98.069<z<100.	100.<z<102.44
r = 14. cm	59%	39%	61%	-55%
r = 12. cm	68%	95%	10%	-189%
r = 10.15 cm	88%	40%	7%	21%
r = 9.525 cm	5%	70%	59%	-165%

The relative differences due to the removal of the shield plug that correspond to the regions of the tube sheet furthest from the fuel bundle (between z = 100 cm and z = 102.44 cm) indicate an increase in activation which is not significant, because the activation values are of the same order of magnitude.

In Table 4, we show the actual values of the ^{59}Co reaction rates in the tube sheet (in reactions per barn-cm) for both cases, i.e. with and without the shield plug.

Table 4. ^{59}Co Reaction Rates (in reactions/barn-cm) in the Tube Sheet

Shield plug inserted

	93.54<z<96	96.<z<98.069	98.069<z<100.	100.<z<102.44
r = 14. cm	288	32	15	22
r = 12. cm	505	253	56	27
r = 10.15 cm	465	213	66	37
r = 9.525 cm	337	145	109	23

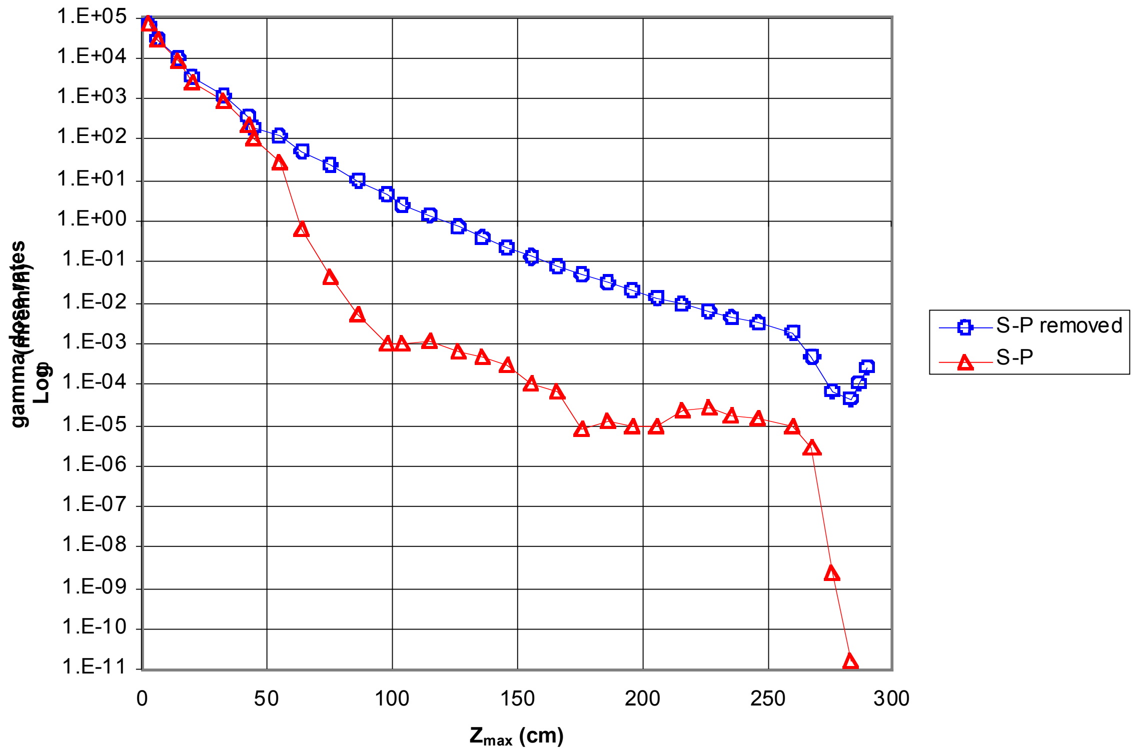
Shield plug removed.

	93.54<z<96	96.<z<98.069	98.069<z<100.	100.<z<102.44
r = 14. cm	117	20	6	33
r = 12. cm	160	12	51	78
r = 10.15 cm	55	128	61	29
r = 9.525 cm	320	44	44	60

These values of the ^{59}Co activation in the tube sheet were used to calculate the ^{60}Co activation gamma dose rates in air, half way between end fittings, at the end plane of the end fittings (290 cm from the fuel bundle). The values obtained were 0.11 mrem/h with the shield plug inserted, and 0.06 mrem/h with the shield plug removed.

For the delayed gammas produced by the fission products 24 hours after shutdown, we calculated the dose rates produced for both cases, with and without the shield plug. The results are shown in Figure 2.

Figure 2. Axial Distribution of Fission-Product Decay-Gamma Dose Rates.



Although the statistics are not as good at the end fittings as at the source (fuel bundle), since the number of photons is many orders of magnitude smaller (see Table 1), the figure shows that the delayed- gamma dose rates at the axis between the end fittings are of the order of 100 times higher when the shield plug is removed.

We also calculated the delayed gamma dose rates across the air region between end fittings at the end plane of the end fittings. When the shield plug is inserted, the delayed-gamma dose is close to 0.02 mrem/h. When the shield plug is removed, the dose rate is 2.4 mrem/h. This value compares well with the dose rate obtained as a related work of Reference [1]. We can see that the ratio between the two cases is again about 100.

Note that the dose rates calculated with MCNP were obtained using a heavy-water density of 0.8 g/cm^3 (normal operating conditions). But, 24 hours after shutdown, the heavy-water density is 1.1 g/cm^3; in practice, therefore, the dose rates will be lower than the values presented here.

CONCLUSIONS

Although the dose rates from fission-product decay-gammas when the shield plug is removed (2.4 mrem/h) become dominant over the dose rates from ^{60}Co decay gammas 24 hours after shutdown (approximately 0.1 mrem/h when the shield plug is removed), they remain significantly below the design target (25 mrem/h). Further refining of the statistics for calculation of the activation dose rates is needed.

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