# RESPONSE KERNELS FOR EX-CORE IONIZATION CHAMBERS

Frej Wasastjerna
VTT Energy
P.O.Box 1604
FIN-02044 VTT
FINLAND
e-mail: frej.wasastjerna@vtt.fi

Pertti Siltanen
Fortum Engineering Ltd
P.O. Box 10
FIN-00048 FORTUM
FINLAND
e-mail: pertti.siltanen@fortum.com

## ABSTRACT

The objective of the work reported here was to help resolve a long-standing discrepancy between measured and calculated results of rod drop experiments for the VVER-440 reactors at Loviisa in Finland. Due to changes in the flux and power distribution, the response of the ionization chambers outside the core is not strictly proportional to the source of fission neutrons in the core. Instead of assuming proportionality, one needs to calculate the time dependence of the ionization chamber signal when simulating rod drop experiments with the 3-dimensional dynamic core simulator code HEXTRAN. To this end, a kernel library giving the contribution of each node in the core to the ionization chamber response was calculated.

The calculations were performed using MCNP4B in adjoint mode. This required only six calculations, for three detector locations and two core configurations, whereas roughly a hundred calculations would have been required in forward mode. Kernel values were calculated only for nodes near the periphery of the core in a 150-degree sector, since the contributions from the interior of the core and from peripheral nodes far from the ionization chamber are small. A small auxiliary program was written to rotate the kernel to correspond to any of the 24 possible ionization chamber locations and to correct the kernel values of the outermost nodes for the highly skewed source distributions in these nodes.

# 1. INTRODUCTION

There has long been a discrepancy between calculated control rod efficiencies and reactivity measurements taken during rod drop experiments for the Loviisa reactors. Such measurements are based on signals obtained from ex-core ionization chambers (ICs) and they include reactor scram from the hot critical state. The observed discrepancy is partly due to the fact that moving the control rods changes the neutron flux and power distribution in the core. As a consequence, the signals from the ICs do not remain proportional to the total source of fission neutrons in the core. Taking this spatial effect into account requires one to estimate the dependence of the ionization chamber signals on the 3-dimensional power distribution.

The 3-dimensional dynamic core simulator code HEXTRAN[1] has been upgraded for the simulation of rod drop experiments, including the signals of ICs. The IC signals are simulated by taking the fission neutron source in each node and multiplying it by a 3-dimensional detector response kernel, which gives the contribution to the detector signal from each node.

The purpose of this work was to calculate such detector response kernels. The normalization of this kernel is immaterial, since only relative responses are required. Only relative values are needed, and calculating them with an accuracy of a few percent was considered sufficient for this application.

# 2. METHOD

The required kernel could be calculated as a forward calculation, postulating a source in a node and then calculating the resulting flux at each ionization chamber position, repeating the calculation for each node but taking into account the symmetry of the core. Alternatively it could be done as an adjoint calculation,[2] in effect following the neutrons backwards from an ionization chamber to the core and obtaining the contribution from each node as the adjoint flux in that node. The ionization chambers are located at 15-degree intervals around the reactor vessel, which, accounting for the 30-degree symmetry of the core, means that only 3 IC positions need be considered. Thus only 3 adjoint calculations were required for each of the 2 core geometries considered (full core, with 349 fuel assemblies, and reduced core, with 313 fuel assemblies and 36 dummy assemblies). In a forward calculation the number of nodes to be considered, and thus the number of calculations required for each geometry, would have depended on which nodes were considered to contribute significantly to the signal, but it would have been at least of the order of a hundred. Thus, in this case adjoint calculations were preferred.

The system to be modeled in the calculations was definitely three-dimensional, comprising the reactor core, the regions above and below it and the surroundings out to some distance beyond the ICs. In addition, diffusion theory would not have been satisfactory, since the strong flux gradients and the cavity made transport effects important. Thus some kind of 3-D transport program had to be used. Two alternatives were available to us: the $S_N$ program TORT[3] and the Monte Carlo program MCNP4B.[4]

Some disadvantages of these two programs were as follows:

TORT:

- The program was unfamiliar, not having been used previously at VTT Energy.

- The geometry, consisting of hexagonal prisms and cylinders, both concentric and non-concentric, could not be modeled as such. The concentric cylinders (such as the basket, barrel, pressure vessel and surrounding shielding) could be modeled in $(r,\theta,z)$ geometry, but the hexagonal fuel assemblies and the IC channels would have had to be approximated.

- For adjoint calculations, the cross sections used by TORT must be ordered differently, which would have required the introduction of a new module in the cross section preparation program TOPICS to execute this reordering. (Admittedly this was not a big task, thanks to the modular design of TOPICS. It was actually done later, requiring a few days.)

MCNP4B:

- Adjoint calculations require the use of a multigroup library, which is less accurate than the pointwise cross sections normally used by MCNP.

- The neutrons should not be followed farther back than the last fission in which they were born. One should not pursue their ancestry back to previous fissions. Thus something like the NONU option in MCNP (which turns fission off in forward calculations) was needed. This option does not work in adjoint calculations, instead the $\nu$ values in the cross section library had to be set to 0.

- Monte Carlo calculations are ill suited for calculating distributions, which is what needs to be done in this work. They are more appropriate for calculation of integral values.

It was decided to try MCNP4B, with TORT available as a backup.

## 3. MODEL

### 3.1 GEOMETRY

The fuel assemblies in the core were homogenized, using water densities corresponding to full power, although simplified to the extent that only a single density was used in the core. Only fast neutrons were of concern in the core, since only they had any significant probability of penetrating to the ICs. For fast neutrons the detailed structure of the fuel assemblies was not important. Although modeling the actual heterogeneous geometry would have been quite feasible in a Monte Carlo program, it would have slowed down the tracking of particles and thus increased the calculation time, probably quite considerably. Since this was a deep penetration problem, which required a large number of histories, and since the requirement to calculate a distribution rather than an integral value further increased the number of histories required, it was considered better to concentrate on getting as many histories as practicable rather than to model unimportant details correctly.

For similar reasons, the dummy assemblies were also homogenized. They were considered to consist of 68 % stainless steel and 32 % water. The regions above and below the core were homogenized into axial layers. This homogenization had the additional benefit of making it possible to use the material compositions previously prepared for activity inventory calculations.

The importance of the nodes diminished rapidly, going towards the interior of the core from the part nearest the ICs. For this reason it was considered sufficient to include only the three outermost layers of fuel assemblies in the reduced core. In the full core the same assemblies were retained, and in addition those assemblies that in the reduced core are replaced by dummies were included.

Similarly it was not considered necessary to include the whole periphery of the core. Only a 150-degree sector was included, in such a way that for all IC positions a sector of at least 60 degrees was present on both sides of the direction from the core axis to the IC. The restriction to 150 degrees also applied to the regions outside the core, except in the insulation and biological shield.

Thus particles exiting this 150-degree sector were killed. The same applied to particles in the interior of the core, or above or below the core and less than 90 cm from the axis. Since this was the fate even of particles that might have been reflected back into the regions of interest, there was a source of error here, leading to an underestimation of the contribution from the innermost included assemblies and the assemblies at the edges of the included sector. However, it appears unlikely that their contribution was underestimated by more than a factor of 2. Since their calculated contribution was about $10^{-3}$ times that of the assemblies nearest the IC, this error had no real significance.

The core was surrounded (in the 150 degree sector) by water, with the baffle ignored and with a density corresponding to 260 degrees C and 125 bar. Included in the model were the basket, barrel, pressure vessel, thermal insulation and concrete shield. The steel liner of the serpentinite concrete constituting the inner part of the biological shield was modeled, both on the inner and the outer sides of the serpentinite. The model was cut off at the outer surface of the outer liner. The neglect of particles backscattered from the concrete beyond this affects mainly the absolute values of the adjoint flux and is thus of little significance.

Three IC channels were included at a distance of 255 cm from the core axis and at 15-degree intervals. The channel walls and the ICs themselves were not modeled, since they would not have had any significant effect on the relative contributions from different nodes.

The geometry is shown in Fig. 1.

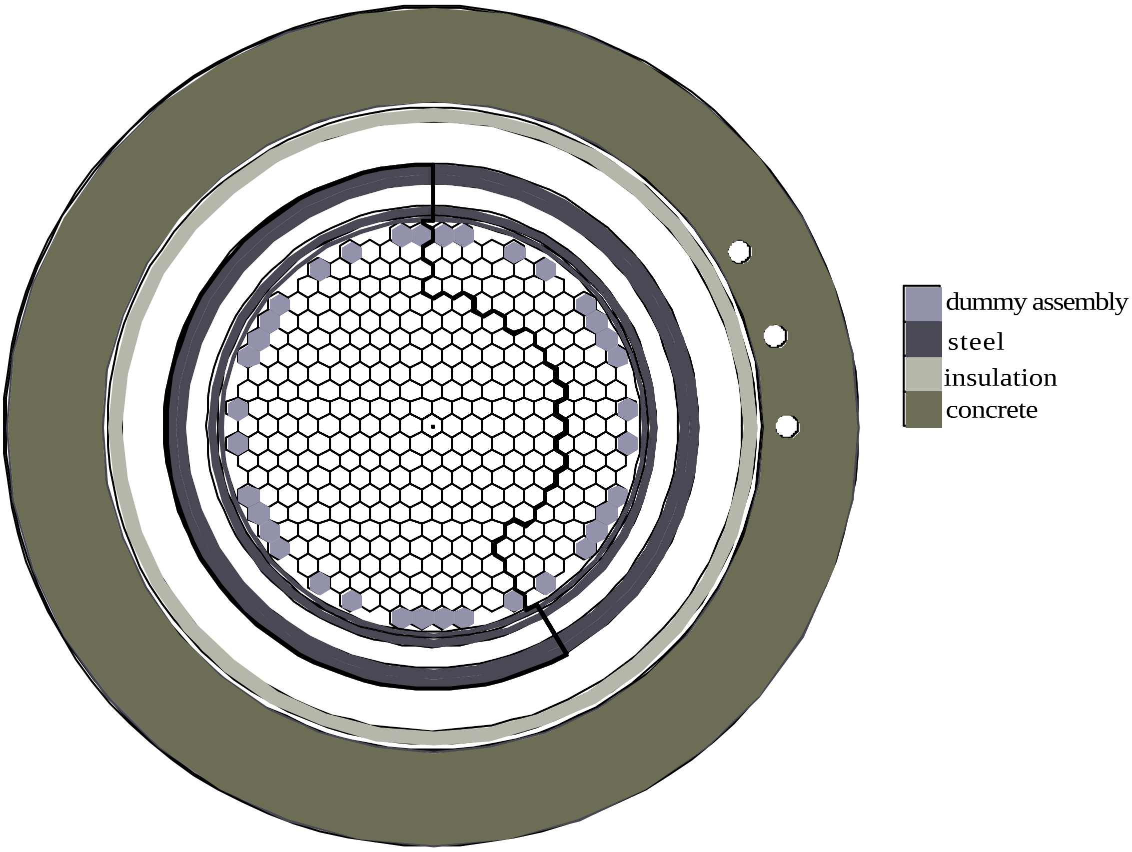


Fig. 1. Overall Transverse Geometry. The Thick Line is the Boundary of the Excluded Part of the Reactor. Not all Cell Boundaries are Shown.

## 3.2 PHYSICS

The source of the adjoint particles was located in the IC channels, filling the channels radially and extending to 25 cm above and below the core midplane, corresponding to an IC height of 50 cm. (The actual active height of the ICs is less, but this would not change the results much.) In terms of energy, this source was flat between 0.15 meV (the lower boundary of the cross section library) and 0.35 eV, 0 elsewhere. This should be a sufficient approximation for the energy dependence of the IC response, since the details will be smeared out very effectively by the reverse thermalization, and the multigroup calculation does not permit an accurate modeling of energy dependence anyway.

To tally the adjoint flux, each fuel assembly was divided into 10 segments, corresponding to the usual 10-node axial subdivision (22 cm at the ends, 25 cm otherwise). The adjoint flux in each segment was folded with a prompt fission spectrum for U-235 to get the contribution from a fission source to the IC signal.

In an adjoint calculation, the energy cutoff should be set above the energy range of interest. The energy range covered by the multigroup library mgxsnp1 extends to 17 MeV, with the uppermost group extending from 15 to 17 MeV. In a multigroup adjoint calculation particles entering the topmost group have no way of leaving it except weight cutoff (e.g., from absorption) or escape from the system. This leads to a pile-up of particles in this group, where the fission spectrum is near zero. Therefore it was decided to set the cutoff to 15 MeV to avoid wasting time on numerous but unimportant particles in the topmost group. Again this leads to a slight underestimation of the contributions to the signal, but this error is likely to affect mainly the absolute values, which are not of interest. The effect on the relative values for different nodes should be very small.

### 3.3 OTHER

In deep penetration calculations with Monte Carlo, some kind of variance reduction is necessary to get useful results in reasonable time. The method used here was weight windows calculated from user-specified importances. 6 250 000 histories were run for each of the 6 cases, giving good statistics at a cost of about 30-120 hours CPU time per case on a combination of HP 9000/780 and DEC Alpha machines.

## 4. QUALITY OF THE RESULTS

The fractional standard deviations (fsd's) were very small in the nodes nearest the IC, indicating good statistics. However, farther from the IC the fsd's were larger, and in the farthest nodes included in the calculation, they exceeded 0.1, which in MCNP is regarded as the limit for good statistics. In some cases the fsd even exceeded 0.3. For these large fsd values the stochastic scatter in the kernel values obscured the true spatial dependence of the kernel, see Figures 2 and 3.

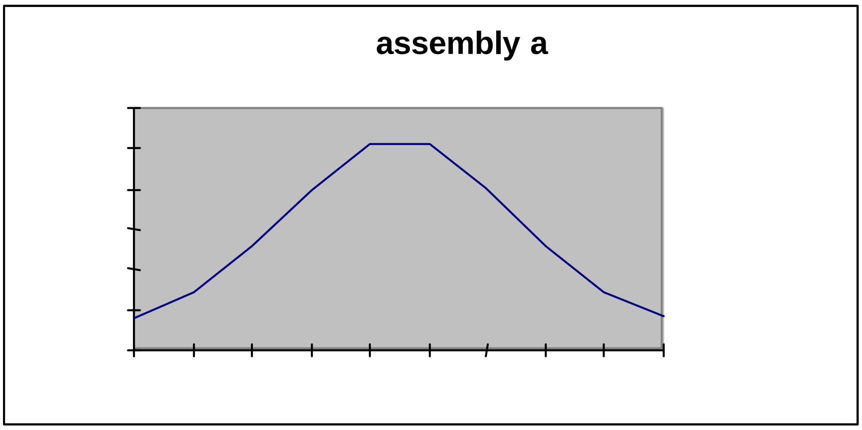


Fig. 2. Axial Distribution of Kernel Values in a Fuel Assembly Near the Ionization Chamber. Note the Relatively Smooth Appearance When the Stochastic Errors are Negligible.

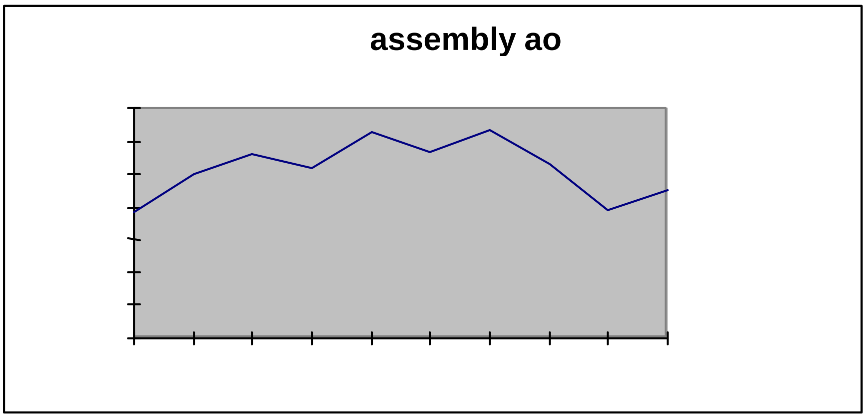


Fig. 3. Axial Distribution of Kernel Values in a Fuel Assembly Far From the Ionization Chamber. Note the Jagged Appearance Caused by Significant Stochastic Errors.

The kernel values decrease rapidly as one goes inward from the core periphery, almost by one order of magnitude per assembly row. They also decrease with increasing azimuthal distance from the detector direction, but much more slowly, see Fig. 4. This shows that streaming in the cavity smears out most of the azimuthal dependence that could otherwise be expected.

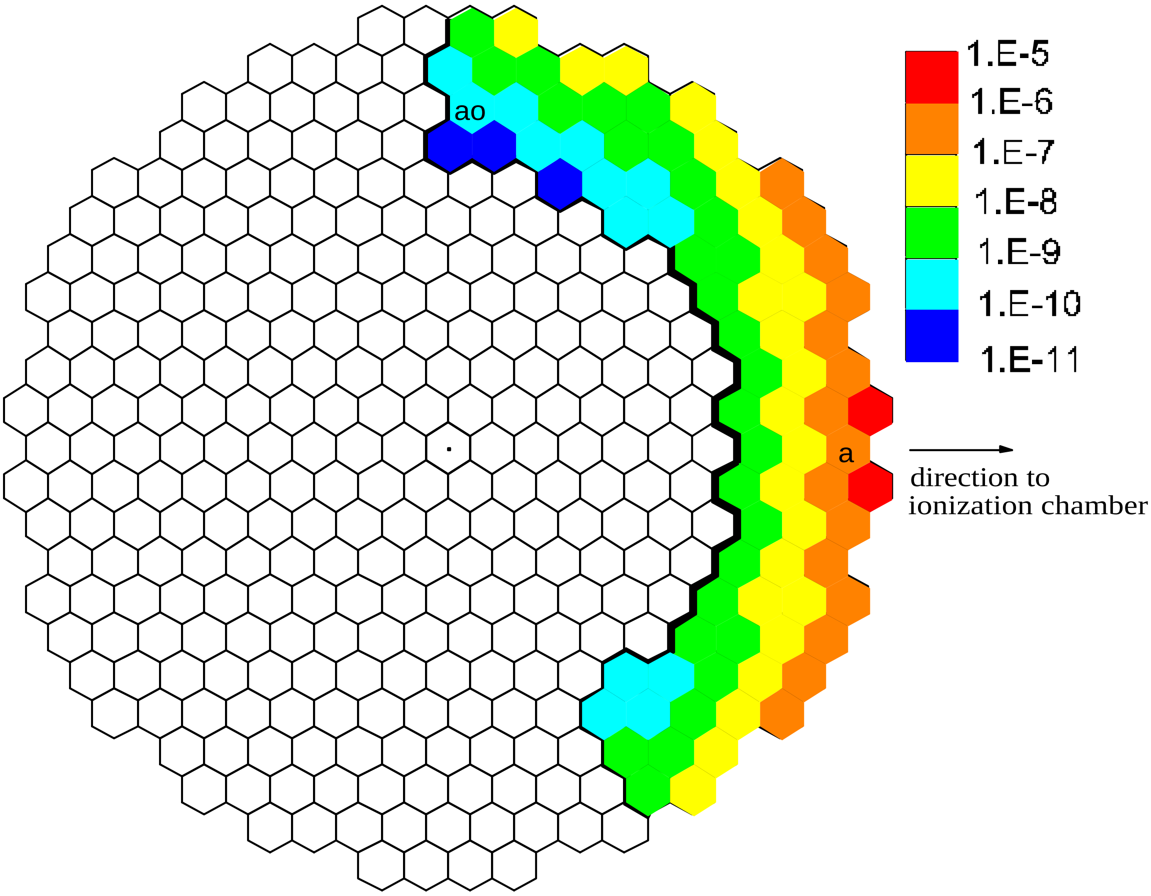


Fig. 4. Approximate Spatial Dependence of Kernel Values (Assembly Averages).

A comparison of results for the full and reduced cores shows that the dummy assemblies in the reduced core are significantly more effective as shielding than fuel assemblies, even apart from the fact that there is no source in the dummy assemblies. The kernel value for a node shadowed by a dummy assembly rather than a fuel assembly is some 20 % lower.

The kernel values are not symmetric with respect to the reactor midplane, although the core itself and its radial surroundings were modeled as vertically symmetric. This is due to the fact that the region above the core is different from that below. The former is apparently less effective as shielding, with the result that the kernel values near the top of the core are somewhat greater than those near the bottom. As one could expect, this effect is more pronounced towards the interior of the core, where transport paths through the regions above and below the core are longer.

Much of the axial asymmetry derives from the fact that the original modeling was done for full power conditions, so the water density above the core was lower than below the core. This difference would vanish in the hot zero power case. However, even for the hot zero power case the asymmetry persisted, though it was much reduced. There is somewhat more steel below than above the core, which makes the former region more effective at attenuating fast neutrons.

As was pointed out above, in the more remote nodes the stochastic uncertainty of the results becomes considerable. In the graphs this shows up as a scatter of the kernel values, which no longer follow a smooth curve. The kernel values themselves are small in these nodes, so the stochastic uncertainty is not important in practice. It was decided to accept the scatter of the kernel values in the remote nodes, which is, after all, merely an esthetic problem. It is of essentially no practical importance for the integrated IC response.

# 5. CORRECTIONS

After the initial calculations had been performed, it was decided to check the effect of three possible sources of inaccuracy in these calculations.

- The calculations had been done using material parameters, especially water densities, that corresponded to full power. Since rod drop experiments are generally conducted at hot zero power, it was necessary to check whether the higher water density in such a case would have given different results.

- It had been assumed that the delayed neutrons could be neglected, even though they may in a strongly subcritical situation constitute a significant fraction of all neutrons. This assumption was based on the fact that the delayed neutron source spectrum is substantially softer than the prompt spectrum, which means that a delayed neutron has a lower probability of reaching a ionization chamber. However, it was considered necessary to check whether this probability is so much lower that the neglect of delayed neutrons was justified.

- Simply integrating the adjoint flux over a node is equivalent to assuming a flat source distribution in the node. In reality there are strong gradients in the outermost fuel assemblies, and the effect of this had to be investigated.

The first two of these questions were elucidated with an additional MCNP4B run, otherwise identical with one of the cases already calculated but with the water density everywhere corresponding to hot zero power conditions. To study the effect of delayed neutrons, two additional tallies were included. One gave the adjoint spectrum in five assemblies. The other weighted the adjoint flux with a fission spectrum similarly to the main tally, but with the difference that the fission spectrum was that for delayed neutrons rather than prompt neutrons. This delayed neutron spectrum, for U-235, was processed into a 30-group weighting function using an ad hoc program.

## 5.1 COOLANT DENSITY DEPENDENCE

The effects of the density dependence were investigated by calculating the ratio of the kernel values for hot zero power to the full power values. These values stay close to the average value 0.900, and there is no obvious trend.

However, a close investigation reveals that there actually is a trend. As one goes inward from the core boundary, the ratio does decrease slightly. This is exactly what can be expected on physical grounds, given the somewhat higher water density in the hot zero power state. This trend is nonetheless weak. The ratio decreases by a few percent from the core periphery to the third row of assemblies. This is negligible compared with the drop in kernel values by a factor close to 100, so one can consider the ratio essentially constant. Thus a kernel calculated for full power conditions can legitimately be used at hot zero power as well, since we are interested only in the relative kernel values, not the absolute ones.

## 5.2 DELAYED NEUTRONS

The relative importance for the ionization chamber signal of delayed neutrons, compared with prompt neutrons, is obtained as the ratio of the tally weighted with a delayed fission spectrum to that weighted with a prompt fission spectrum.

The largest value found was 0.077, but this appears to be a statistical fluke. This is supported by the large fractional standard deviations for the kernels in this remote node. A more realistic upper limit is obtained by observing that in the outermost assemblies the ratio lies near or below 0.03. In the inner assemblies it's usually lower, since the neutrons have farther to travel and the softness of the delayed neutron spectrum decreases the contribution of these neutrons even more.

Thus we can regard 0.03 as a slightly conservative estimate of the relative importance of delayed neutrons. Considering that in a deeply subcritical situation about 10 % of all emitted neutrons will be delayed ones, this means that the total contribution from delayed neutrons will even in this case be responsible for only 0.003 of the signal. Consequently we can conclude that this contribution can safely be ignored.

## 5.3 SOURCE GRADIENT

To get the contribution to the ionization chamber signal from the source in a node, one should multiply the adjoint flux calculated here by the space and energy distribution of the source and integrate over the node. However, there is no known way of including a space-dependent weighting function, corresponding to the spatial distribution of the source, in a single cell in MCNP4B. It would have been possible to split each node into multiple cells and tally the adjoint flux in each cell separately, then multiplying by an assumed source distribution, but this would have been awkward. Consequently only the energy spectrum of the fission source was used to multiply the adjoint flux. This means that the results are valid for a spatially flat source distribution within each node.

However, in reality the source distribution is strongly slanted in the outermost fuel assemblies, less so as one goes deeper into the core. The adjoint flux is also highly space-dependent, in such a way that it is largest where the source is smallest. This means that the integral of the product of the adjoint flux and source is smaller in reality than the value obtained without a space dependence for the flux. For the outermost assemblies, this effect can be significant, so some way of correcting for it is desirable.

Such a correction was devised as follows: Estimates of the source distribution (pin power distribution) in node 5 (just below the midplane) of each of the outermost assemblies under normal operating conditions were provided by Fortum Engineering Ltd.[5] Different values were provided for a full core (Loviisa 1 cycle 1), an out-in reduced core (Loviisa 1 cycle 5) and a low-leakage loading pattern (Loviisa 1 cycle 17) and for the beginning and end of each such cycle.

The adjoint flux was found, on the basis of the calculations described above, to depend much more strongly on the radius (the distance from the core axis) than on the azimuth relative to the IC. Consequently it was approximated by an exponential function of the radius alone, with a relaxation length of 7.2124 cm, derived from a comparison of the average kernel values for 3 radially adjacent assemblies in one of the cases. This approximate adjoint flux was then used in an ad hoc program to multiply on one hand the actual pin power distribution, on the other hand a flat distribution with the same average. These products were summed over the pins, and their ratio was taken as the correction factor.

These correction factors ranged from 0.872 to 0.974, with a marked dependence on both the position in the core, the cycle (loading pattern) and the number of effective full power days since the beginning of the cycle.

## 6. KERNEL ROTATION

As stated in Section 2, there are 24 ICs at 15-degree intervals. While these can be reduced on symmetry grounds to 3 positions for the purpose of evaluating the kernels, it is desirable when using HEXTRAN to be able to locate the IC whose response is calculated in any of the 24 possible positions. While it would in principle be possible to choose the coordinate system defining the positions of the fuel assemblies so that the IC of interest is always located in one of 3 positions, that would be inconvenient in practice. Therefore an auxiliary program was written to rotate the calculated kernels into any desired orientation. This program also applies the source gradient corrections described in the previous section.

## 7. CONCLUSIONS

Detector response kernels were calculated which, when folded with the nodewise prompt fission neutron source distribution, will give the signals from the ionization chambers apart from an unknown but constant normalization factor. If the source distribution itself is unavailable, the fission power distribution can be used as a substitute.

Although the Monte Carlo method is poorly suited for calculation of distributions, the use of a fairly large number of histories and heavy variance reduction gave good statistics and consequently good distributions in the important nodes, those nearest the detectors. In the more remote nodes the stochastic errors are considerable, but these nodes make only a minor contribution to the detector response. In addition, adding the contributions from many nodes smears out the uncertainty of the kernel values for individual nodes. Thus the stochastic uncertainty does not have any significant effect on the detector response.

There are also systematic uncertainties, but it is believed that they will not significantly distort the changes in the detector responses, which are the quantities of interest when these kernels are applied in HEXTRAN.

## ACKNOWLEDGEMENTS

This work was funded by Fortum Engineering Ltd. The assistance of Aapo Tanskanen of VTT Energy and Jari Kuusisto of Fortum is gratefully acknowledged.

## REFERENCES

[1] R. Kyrki-Rajamäki, *Three-Dimensional Reactor Dynamics Code for VVER Type Nuclear Reactors.* VTT Publication 246, Espoo, Finland (1995).

[2] G.I. Bell and S. Glasstone, *Nuclear Reactor Theory* (Chapter 6). Van Nostrand Reinhold Company, New York, USA (1970).

[3] W.A. Rhoades, The TORT Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code. ORNL-6268 (1987). TORT is available as part of RSIC Computer Code Collection package CCC-543, comprising TORT and DORT.

[4] J.F. Briesmeister, Editor, MCNP—A General Monte Carlo N-Particle Transport Code. Version 4B. LA-12625-M, Version 4B, March 1997. Available at http://www-xdiv.lanl.gov/XCI/PROJECTS/MCNP/manual.html.

[5] J. Kuusisto, memo LO1-K919-514, Fortum Engineering Oy.