10. On p. 446, L&T state, "These tallies failed to . . . , either as the result of the arrival of an isolated and heavily weighted particle from an undersampled region . . . or due to a folding of high-uncertainty portion . . . flux bin with moderately high. . . ." If L&T do not know/understand why their calculations give improper results, they should not publish their work because, after all, it is their, not the reader's, responsibility to explain and resolve the issues of their work—especially when they are claiming that they have developed "an *alternative* process that produces accurate results."

11. On p. 446, L&T state, "The method offers the possibility of an automated procedure for downloading core instrumentation data directly into the MCNP input for generation of computed flux spectra at multiple locations about the PV and ex-vessel cavity, within a moderate computational time." L&T do not give any new technique that facilitates downloading core instrumentation data. Further, it is ironic that they have not explicitly modeled regions such as cavity dosimetry and somewhat defeated the purpose of three-dimensional combinatorial geometry, which is offered by the MCNP code.

I realize that this is a rather long and detailed critique of Ref. 1, but I am very concerned with its inaccurate and misleading results and conclusions—especially since they can be used to project the life of commercial nuclear power plants. Also, I am concerned with the failure of the review process. Obviously, this paper does not comply with the following: (a) the work is correct and complete and (b) the authors give adequate credit to earlier work. (Note that none of the papers I referred to in my comments were referenced by L&T.)

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Reply to "On 'Neutron Fluence at the Pressure Vessel of a Pressurized Water Reactor Determined by the MCNP Code'"

We share with Dr. Haghighat a sincere and serious desire to further the application of Monte Carlo transport methods for the study of pressure vessel (PV) fluence and dosimetry with the important long-term goal of supporting reactor PV life-extension studies. We were very happy to read his detailed review of our work. We would like, however, to offer the following specific comments in response.

1. Like Wagner, Haghighat, and Petrovic's work in Ref. 1, we compared MCNP results using one set of cross sections to DOT calculations made with a different cross-section library. In our case, we employed the results of an earlier study[2] for the same reactor and PV performed using DOT and ENDF/B-IV cross sections. Obviously, our present study[3] could not have used the same cross sections since with MCNP, one uses the continuous-energy model, whereas DOT requires multigroup cross-section libraries.

2. Results in Ref. 9 of Ref. 4 and particularly Ref. 10 of Ref. 4 [published about 6 months after Ref. 3 was submitted to *Nuclear Science and Engineering* (NSE)] indicate indeed that the source fission spectrum has an effect on high-energy threshold reactions, such as $^{63}Cu(n,\alpha)^{60}Co$, but an insignificant effect on lower energy threshold reactions.

3. Our failure to distinguish clearly between calculation to experiment and the MCNP tally errors is admitted, but our conclusions still stand correct as stated. Our statement that the methodology presented is practical and easily accessible for adoption by utility engineers and other researchers is valid, leaving out the details of CPU time and machine type.

4. Five of the 14 references offered in Ref. 4 as evidence that the review process failed were published well after our paper was submitted to NSE, and seven of the remaining nine were conference papers. Three of these were from the *Proceedings of the 8th International Conference on Radiation Shielding*, appearing only 11 days before NSE received our manuscript. Clearly, we did not omit referencing the work of Dr. Haghighat and his students by design. All this activity makes us feel fortunate to be participants in a surge of research and publication in this very important area.

5. Consistent with the standard practice for any technical paper, we reported clearly the source of the cross sections used for all our calculations. The point made about the Radiation Shielding Information Center's not compiling but just distributing MCNP cross sections is correct. We regret the misunderstanding caused by the words used in the text, although the persons who use these cross sections know what we mean, including Dr. Haghighat.

Dr. Haghighat is correct in pointing out that the default weighting of the cross sections, as compiled and distributed, is not ideal for power reactor ex-vessel dosimetry. This observation notwithstanding, many of our results have improved (as compared with measurements) with the use of ENDF/B-VI threshold and dosimetry cross sections.

6. We feel it is a mistake to dismiss a priori the technique of forward, continuous-energy Monte Carlo calculation simply because the tally volume used was larger than the actual foil region. This technique has been used successfully in other benchmarking studies.[5] If the tally volume is larger than the actual volume by 10%, the error is not necessarily 10%; the error depends on the space variation of the flux in the area of the tally; that variation was checked and was found to be <10% over any direction of the tally volume.

7. We believe that at the time we submitted our manuscript, it represented a significant accomplishment in the use of Monte Carlo neutron transport for the computation of PV fluence verified by the agreement between calculated and ex-vessel measured foil reaction rates. Today, with modest computing resources and as-distributed compilations of cross sections (ENDF/B-VI and T2) and MCNP-4A, utility engineers and other researchers can perform very useful calculations using the methodology of our paper.

We are sincerely grateful for the constructive comments offered by Dr. Haghighat and recognize his and John Wagner's contributions in this field.

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