SENSITIVITY OF LWR FUEL CYCLE COSTS TO UNCERTAINTIES IN DETAILED THERMAL CROSS SECTIONS

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ABSTRACT

Cross sections averaged over the thermal energy (<1 or 2 eV) group have been shown to have an important economic role for light-water reactors. Cost implications of thermal cross section uncertainties at the few-group level were reported earlier. When it has been determined that costs are sensitive to a specific thermal-group cross section, it becomes desirable to determine how specific energy-dependent cross sections influence fuel cycle costs. Multigroup cross-section sensitivity coefficients vary with fuel exposure. By changing the shape of a cross section displayed on a view-tube through an interactive graphics system, one can compute the change in few-group cross section using the exposure dependent sensitivity coefficients. With the changed exposure dependent few-group cross section, a new fuel cycle cost is computed by a sequence of batch depletion, core analysis, and fuel batch cost code modules. Fuel cycle costs are generally most sensitive to cross section uncertainties near the peak of the hardened Maxwellian flux.

INTRODUCTION

An interactive system has been developed at RPI to analyze the sensitivity of water reactor fuel cycle parameters and costs to uncertainties in nuclear data. The sensitivities of few-group cross sections to uncertainties in detailed thermal nuclear data (referred to as Path A) are determined by methods presented in this paper. A sequence of batch depletion, core analysis, and fuel cycle cost codes (referred to as Path B) determines the changes in fuel cycle parameters and costs for changes in few-group microscopic cross sections, in fission yields, and decay data. These Path B methods and results are discussed in detail in our other paper presented at this workshop.[1]

Cross sections averaged over the thermal energy (<1 or 2 eV) group have been shown to have an important economic role for light-water reactors.[2,3] When it has been determined that fuel cycle parameters and costs are sensitive to a specific thermal-group cross section, it becomes desirable to determine how specific energy-dependent cross sections influence fuel cycle parameters and costs. A cross section can be

modified and viewed easily, along with the base cross section, flux, adjoint, and sensitivity coefficient, with the aid of Rensselaer's Interactive Graphics Analysis System (RIGAS)[4]. After computing the changed exposure-dependent thermal-group cross section, a new fuel cycle cost is computed by the sequence of Path B code modules. We can therefore obtain the change in fuel cycle cost for different fuel cycle options induced by a change in the shape of a detailed thermal cross section. This type of analysis on thermal cross section uncertainties can be used to indicate directions for worthwhile improvements in thermal nuclear data and methods.

METHODS FOR COMPUTING THERMAL REGION SENSITIVITY COEFFICIENTS

Path A analyses examine the sensitivities of few-group microscopic cross sections to changes in basic nuclear data. These sensitivities have two components, one associated directly with the nuclear data change and one associated with the change in flux resulting from the change in nuclear data. This change in flux affects few-group microscopic cross sections for other reactions as well as for that originally changed. Sensitivities of thermal group cross sections to uncertainties in multigroup cross sections can be calculated by direct and by perturbation methods. Each method has advantages and disadvantages.

In the direct method the change in thermal-group cross section σ^{i}_{rG} of nuclide i and reaction type r contingent on a change $\delta\sigma^{i'}_{r'g}$ in the multigroup cross section $\sigma^{i'}_{r'g}$ is computed directly by subtracting the changed case calculation from the base case calculation. The flux calculation is performed once with base multigroup cross sections. This flux is used to collapse the multigroup cross-sections σ^{i}_{rg} to the few-group cross-section σ^{i}_{rG}. The flux is computed again with a changed multigroup cross section $\sigma^{i'}_{r'g} + \delta\sigma^{i'}_{r'g}$, where $\delta\sigma^{i'}_{r'g}$ is the uncertainty. The few-group cross section σ^{i}_{rG} is then computed again with this changed flux. By subtracting the changed value from the base value, one determines the microscopic cross section sensitivity coefficients

$$S^{i,i'}_{gr,r'} = \frac{\sigma^{i'}_{r'g} \quad \delta\sigma^{i}_{rG}}{\sigma^{i}_{rG} \quad \delta\sigma^{i'}_{r'g}} \quad . \tag{1}$$

Direct methods are conceptually simple but require a new flux calculation for each data change, which can be prohibitively expensive when several sensitivities must be computed. Direct methods, however, are not limited to small cross section uncertainties.

Perturbation methods permit calculation of many sensitivity coefficients from a practicable number of forward and adjoint calculations.

Industry codes could be used for this purpose if they were capable of computing adjoint solutions; however, they do not have this capability. The FASTT code, to be described later, was written to perform the flux and adjoint calculations in the thermal energy region and combine these to yield the required sensitivity coefficients. Perturbation methods are limited to small changes, however, yielding partial derivatives $\partial\sigma^i_{rG}/\partial\sigma^{i'}_{r'g}$ instead of $\delta\sigma^i_{rG}/\delta\sigma^{i'}_{r'g}$.

Few-group cross-sections are usually determined in LWR design by solving inhomogeneous equations,

$$L\phi = S \quad , \tag{2}$$

for the position, direction, and energy dependent neutron flux $\phi(\underline{r},E,\underline{\Omega})$ in a unit cell or other region of a reactor. The source S in the thermal energy region has the shape of neutrons slowing down from higher energies. If thermal nuclear data are changed, the transport operator L will change, and the flux will change as required by the transport equation. To first order, the change in flux is given by the equation

$$\delta L\phi + L\delta\phi = 0 \quad . \tag{3}$$

It is customary to compute few-group microscopic cross sections, σ^i_{rG}, by reaction rate averaging,

$$\sigma^i_{rG} = \frac{\langle \sigma^i_r C^i \phi \rangle_G}{\bar{C}^i \langle \phi \rangle_G} \quad . \tag{4}$$

Here the symbol $\langle f \rangle_G$ represents the integral of $f(\underline{r},E,\underline{\Omega})$ over position in the region, over the energy band G, and over all directions. $\bar{C}^i$ is the cell averaged concentration for nuclide i.

The change in few-group cross section σ^i_{rG} can be computed directly by calculations using Eqs. 2 and 4 both with and without the change in multigroup cross section $\sigma^{i'}_{r'g}$,

$$\delta\sigma^i_{rG} = \frac{\langle (\sigma^i_r + \delta_{ii'}\delta_{rr'}\delta\sigma^{i'}_{r'}) C^i(\phi+\delta\phi) \rangle_G}{\bar{C}^i \langle \phi + \delta\phi \rangle_G} - \frac{\langle \sigma^i_r C^i \phi \rangle_G}{\bar{C}^i \langle \phi \rangle_G} \quad . \tag{5}$$

The Kronecker delta $\delta_{ii'}$ is equal to 1 when i = i', and $\delta_{rr'}$ is equal to 1 when r = r'; both are equal to zero otherwise. This is referred to as the direct technique. Using Eq. 4, after the terms in Eq. 5 have been placed on the common denominator and expanded, gives to first order,

$$\delta\sigma^i_{rG} = \frac{\langle \delta_{ii'}\delta_{rr'}\delta\sigma^{i'}_{r'} C^{i'} \phi \rangle_G}{\bar{C}^i \langle \phi \rangle_G} + \frac{\langle (\sigma^i_r C^i - \sigma^i_G \bar{C}^i)\delta\phi \rangle_G}{\bar{C}^i \langle \phi \rangle_G} \quad . \tag{6}$$

From this expression we can see that if $i \neq i'$ or $r \neq r'$, the first term is zero and the change in few-group cross-section results from the change in flux alone.

Perturbation theory is applied as follows; the equation adjoint to Eq. 2 is

$$L^{+}\phi^{+} = S^{+} \quad , \tag{7}$$

where the operator L^{+} has the usual linear adjoint property

$$\langle \chi^{+}, L\psi \rangle_G = \langle L^{+}\chi^{+}, \psi \rangle_G \quad . \tag{8}$$

Commas are introduced at times in these equations to delimit operations, e.g., the operator L^{+} operates on χ^{+} but not on ψ. Using the last two equations we find

$$\langle S^{+}, \delta\phi \rangle_G = \langle L^{+}\phi^{+}, \delta\phi \rangle_G = \langle \phi^{+}, L\delta\phi \rangle_G \quad . \tag{9}$$

By comparing the first term of this expression with the last term in Eq. 6, we choose the adjoint source,

$$S^{+}_{ir} = \sigma^{i}_{r}(E)C^{i}(\underline{r}) - \sigma^{i}_{rG}\bar{C}^{i} \quad . \tag{10}$$

From Eqs. 3, 9 and 10 we obtain,

$$\langle (\sigma^{i}_{r}C^{i} - \sigma^{i}_{rG}\bar{C}^{i}), \delta\phi \rangle_G = -\langle \varphi^{+}_{ir}, \delta L\phi \rangle_G \quad . \tag{11}$$

Using the adjoint we have elminated the need to compute the change in flux. Using Eq. 11, Eq. 6 becomes

$$\delta\sigma^{i}_{rG} = \frac{\langle \delta_{ii'}, \delta_{rr'}, \delta\sigma^{i'}_{r'}, C^{i'}\phi \rangle_G}{\bar{C}^{i}\langle \phi \rangle_G} - \frac{\langle \phi^{+}_{ir}, \delta L\phi \rangle_G}{\bar{C}^{i}\langle \phi \rangle_G} \quad . \tag{12}$$

The change δL is just $C^{i'}\delta\sigma^{i'}_{r'}(E)$ for a reaction cross-section change.

One forward flux calculation and one adjoint flux calculation suffice to yield the dependence of the group cross-section σ^{i}_{rG} on all small changes $\delta\sigma^{i}_{r'}(E)$ as arbitrary functions of energy. For a reaction cross section change, we are actually interested in the sensitivity coefficient defined by the partial derivative,

$$S^{i,i'}_{gr,r'} = \frac{\sigma^{i'}_{r'g}}{\sigma^{i}_{rG}} \frac{\partial\sigma^{i}_{rG}}{\partial\sigma^{i'}_{r'g}} = \frac{\langle (\delta_{ii'}, \delta_{rr'}, -\phi^{+}_{ir})\sigma^{i'}_{r'}, C^{i'}\phi \rangle_g}{\langle \sigma^{i}_{r}C^{i}\phi \rangle_G} \quad . \tag{13}$$

In arriving at this equation it was assumed that the cross-section was perturbed slightly only in group g.

The sensitivities of thermal-group average cross sections to changes in detailed energy-dependent cross sections in the thermal range are calculated by the code FASTT. This code combines selected features of the thermal neutron treatments in LASER[5] and LEOPARD[6] to compute both the multigroup neutron flux and adjoint, then combines these to yield the required sensitivities. Sensitivity coefficients computed by both the direct and perturbation methods are shown in Table 1. The results agree with reasonable accuracy, thus confirming both methods.

THERMAL REGION CALCULATIONS: THE FASTT CODE

Cross sections averaged over the thermal energy (<1 or 2 eV) group play an important role in computational methods for light-water reactors. The thermal-group cross sections are computed in reactor cell programs by code modules that can be classified as being like SOFOCATE[7] or like THERMOS[8]. In the former case the emphasis is on the neutron energy spectrum with simplified spatial treatments when these are included at all. In the latter case there is a more balanced treatment of spatial and spectral effects, but there is a corresponding penalty in running time. Because a large part of the power reactor industry utilizes SOFOCATE in the form included as a module in cell codes like LEOPARD[6] and CHEETAH[9], and because this approach is simpler, less costly, and hence, suited for sensitivity analysis, the FASTT code to be described utilizes this approach.

Neutrons slowing down in the moderator from energies above 1 or 2 eV provide the transport into the thermal range. This equation is solved in LWR design for the order of 10 or 10^2 energy points or groups using spatial calculations of greater or lesser accuracy. Codes solving this problem are reasonably complex, and their adjoint versions could be complex also. Fortunately, we have developed a technique for computation of adjoints using forward codes. The adjoint source, $\sigma(E)C(\underline{r}) - \sigma_G\bar{C}$, appropriate to the present application requires a particular transformation for solution with conventional methods.

The reactor parameters calculated by FASTT and the procedures used to determine them are similar to those in other industry type codes. The microscopic cross sections read by FASTT must be Doppler broadened to the proper temperature. D. E. Cullen et al.[10] have presented a comparison of Doppler broadening methods in the thermal energy range. We have chosen to use the LASER[5] 35 group structure with an upper cutoff of 1.855 eV in order to include the effects of the Pu-240 resonance. The FASTT code is quite general and can run with any group structure.

Table 1. Comparison Between Direct and Perturbation Methods for the Green County PWR at Middle of Exposure (1% changes for Direct Method)

Energy Midpoint of Group(eV)	SENSITIVITY COEFFICIENTS U-235(n,f) Direct	U-235(n,f) Pert.	Boron(n,γ) Direct	Boron(n,γ) Pert.	Pu-240(n,γ) Direct	Pu-240(n,γ) Pert.
.001139	.003081	.003091	.005888	.005885	.000487	.000491
.036685	.086202	.086267	.088536	.088541	.015800	.015799
.165080	.058685	.058711	.063540	.063545	.015564	.015566
.295940	.003922	.003934	.003631	.003635	.001028	.001031
.564090	.011168	.011176	.015874	.015877	.013222	.013224
1.047700	.000140	.000147	.000309	.000310	.042377	.042540
1.237200	.004546	.004546	.005722	.005723	.034977	.034980
1.790500	.000916	.000923	.003094	.003089	.001170	.001171

The simplest model of neutron thermalization yielding results of sufficient accuracy is that which treats the reactor core as a free proton gas, neglecting chemical binding. In this Wigner-Wilkins[11] model the protons are assumed to be in thermal equilibrium with a Maxwell-Boltzman energy distribution. FASTT can compute scattering cross-sections using the free proton model or the Nelkin[12] model, in which the chemical binding is accounted for by a set of discreet molecular frequencies. In FASTT only the downscattering part of the kernel is calculated and stored. The upscattering is calculated as needed from the condition of detailed balance,

$$M(E)\sigma_s(E\rightarrow E') = M(E')\sigma_s(E'\rightarrow E) \quad . \tag{14}$$

where M is the Maxwellian flux

$$M(E) = \frac{E}{(kT)^2} e^{-E/kT} \quad . \tag{15}$$

The FASTT code has the option to read in any scattering kernel in the proper group structure.

The unit cell consists of the cylindrical fuel pellet, a cylindrical annular clad region, and a coolant region between fuel rods. A modified Amouyal-Benoist-Horowitz method defined by Strawbridge is used to calculate the spatial variation of flux within the pin cell. The fuel-to-moderator flux ratio and the clad-to-moderator flux ratio are computed by FASTT at each energy group. The macroscopic cross sections are then computed with respect to the moderator.

In FASTT we have included the spatial flux dependence with the use of flux ratios and are therefore able to simplify the linear transport equation (Eq. 2) to

$$\Sigma_t(E)\phi(E) - \int \Sigma_s(E'\rightarrow E)\phi(E')dE' = S(E) \quad , \tag{16}$$

where the slowing down source S(E) from energies above the thermal range is uniform into the effective pin cell. This forward balance equation can be solved by iteration,

$$(\Sigma_{ag} + \Sigma_{sg})\phi_g^{(m+1)} = \sum_{g'} \Sigma_{g'\rightarrow g}\phi_{g'}^{(m)} + S_g \quad . \tag{17}$$

The initial flux $\phi_g^{(o)}$ is assumed to be a Maxwellian distribution,

$$\phi_g^{(o)} = M_g \sum_{g'} S_{g'} / \sum_{g'} M_{g'}\Sigma_{ag'} \quad , \tag{18}$$

which is normalized so that the spectrum averaged absorption equals the total source. FASTT renormalizes each iterate flux so that $\sum_g \Sigma_{ag}\phi_g^{(m)}$ equals the source $\sum_g S_g$ into the thermal group. Convergence is very

rapid; in fact, even the first iterate may be adequate for some sensitivity studies.

Earlier it was shown that few-group sensitivities are readily computed in terms of solutions to the forward and adjoint transport problems. The difficulty is that industry codes for the thermal energy range do not have adjoint capability. Fortunately, a method has been developed for computation of the thermal adjoint using forward codes[14]. The adjoint equation corresponding to the forward equation for the effective uniform cell, Eq. 16, is

$$\Sigma_t(E)\phi^+(E) - \int \Sigma_s(E\to E')\phi^+(E')dE' = S^+(E) \quad . \tag{19}$$

Multiply this by M(E), and replace $M(E)\Sigma_s(E\to E')$ by its detailed balance equivalent from Eq. 14. Then

$$\Sigma_t(E)\psi(E) - \int \Sigma_s(E'\to E)\psi(E')dE' = M(E)S^+(E) \quad , \tag{20}$$

where the auxiliary function $\psi(E)$ is defined by

$$\psi(E) = \phi^+(E)M(E) \quad . \tag{21}$$

Eq. 20 has the structure of a forward equation, so the adjoint ϕ^+ can be computed from any code that solves this forward equation. Here a source $M(E)S^+(E)$ is used, and the auxiliary function $\psi(E)$ is divided by M(E) to yield the adjoint $\phi^+(E)$.

It is required that the scattering function $\Sigma_s(E'\to E)$ actually obeys detailed balance. In LWR codes this is so for hydrogen scattering. Scattering of thermal neutrons from other nuclides usually is taken to be monoenergetic, in which case the detailed balance equation is satisfied automatically.

As was noted earlier, the adjoint source appropriate for the group averaged cross-section σ_G^i for isotope i in region j is $\sigma_g^i C_j^i - \sigma_G^i \bar{C}^i$. This source is positive for some groups and negative for others. It is convenient to further transform the adjoint equation as follows:

$$(\Sigma_{ag} + \Sigma_{sg})\phi_g^+ = \sum_{g'} \Sigma_{g'\to g}\phi_{g'}^+ + \sigma_g^i C_j^i - \sigma_G^i \bar{C}^i \quad . \tag{22}$$

Subtract $\Sigma_{ag} + \Sigma_{sg}$ from both sides, then multiply by -1.

$$(\Sigma_{ag}+\Sigma_{sg})(1-\phi_g^+) = \sum_{g'} \Sigma_{g\to g'}(1-\phi_{g'}^+) + \sigma_G^i \bar{C}^i + (\Sigma_{ag} - \sigma_g^i C_j^i) \quad . \tag{23}$$

The source to this balance equation is non-negative. Now convert this to a forward equation as developed previously,

$$(\Sigma_{ag}+\Sigma_{sg})\psi_g = \sum \Sigma_{g'\to g}\psi_{g'} + M_g\left[\sigma_G^i \bar{C}^i + \Sigma_{ag} - \sigma_g^i C_j^i\right] \quad , \tag{24}$$

where the auxiliary function ψ_g is

$$\psi_g = M_g(1-\phi_g^+) \quad . \tag{25}$$

FASTT calculates the auxiliary function ψ_g by the iteration technique described earlier, and since the source approximates the Maxwellian, convergence is even faster for the auxiliary function than for the forward flux.

The sensitivities of thermal group average cross sections to changes in detailed energy-dependent cross sections in the thermal range can be calculated from the flux and proper adjoint. The macroscopic sensitivity coefficient is computed from

$$D_g = \frac{\Sigma_{ag}}{\Sigma_{aG}} \frac{\partial \Sigma_{aG}}{\partial \Sigma_{ag}} = \frac{(1-\phi_g^+)\Sigma_{ag}\phi_g}{\sum_{g'} \Sigma_{ag'}\phi_{g'}} \quad . \tag{26}$$

The microscopic sensitivity coefficients for a change in a few group cross section for nuclide i and reaction type r, induced by a change in multigroup cross section for nuclide i' and reaction type r', are calculated by

$$S^{i,i'}_{gr,r'} = \frac{\sigma^{i'}_{r'g}}{\sigma^{i}_{rG}} \frac{\partial \sigma^{i}_{rG}}{\partial \sigma^{i'}_{r'g}} = \frac{\sum_j (\delta_{ii'}\delta_{rr'} - \phi^{+i}_{rjg}) C^{i'}_j V_j \sigma^{i'}_{r'g} (\frac{\phi_j}{\phi_M})_g \phi_{Mg}}{\sum_g \phi_{Mg} \sum_j C^{i}_j V_j \sigma^{i}_{rg} (\frac{\phi_j}{\phi_M})_g} \quad . \tag{27}$$

Here V_j refers to the volume of region j, and M refers to the moderator.

We are also interested in determining the change in a few-group cross section for a change in the shape in the detailed cross section. Using the definition of microscopic sensitivity coefficient, we find,

$$\frac{\delta \sigma^{i}_{rG}}{\sigma^{i}_{rG}} = \sum_g S^{i,i'}_{gr,r'} \frac{\delta \sigma^{i'}_{r'g}}{\sigma^{i'}_{r'g}} \quad , \tag{28}$$

where the fractional changes in $\sigma^{i'}_{r'g}$ are input quantities. For example, we may wish to investigate the effects of an uncertainty in the 1/v component of a cross section or in a resonance parameter. The cross section can be modified and viewed easily, along with the base cross section, flux, adjoint, and sensitivity coefficient, with the aid of Rensselaer's Interactive Graphics Analysis System (RIGAS)[15].

Some sensitivity coefficients have large variations with exposure due to self-shielding and the hardening of the flux spectrum as

plutonium builds up. Even 1/v cross section sensitivity coefficients have large variations with exposure in certain energy regions, primarily due to flux dipping at plutonium resonances. Therefore, the fractional changes in σ^i_{rG} from Eq. 28 will vary with exposure. Once these changes are computed at each exposure time step, they can be used in the Path B code sequence as input to the fuel depletion module FASTCELL[16]. We can therefore obtain the change in fuel cycle cost for different fuel cycle options induced by a change in the shape of a detailed thermal cross section. Thus we have the necessary tools to investigate fuel cycle cost implications to improvements in ENDF data sets.

COMPUTED SENSITIVITY COEFFICIENTS

The sensitivity of fuel cycle costs to uniform thermal-group cross section changes with exposure were determined for the 40 nuclides included in the FASTCELL fuel depletion module. Path B sensitivity coefficients for 1985 are shown in Table 2 for both pressurized and boiling water reactors and for two back end fuel cycle alternatives. Nuclides with sensitivities less than .006 are not listed. Pu-239 and U-235 have the largest sensitivities, followed by H, Pu-241, and U-238. Fission products and structural materials have much smaller sensitivities. Sensitivity to the Pu-240 thermal capture cross section turns out to be small because of the high worth of the Pu-241 produced from neutron capture in Pu-240.

The sensitivities of thermal-group average cross sections to changes in detailed energy-dependent cross sections in the thermal range are calculated by the FASTT code. Figure 1 shows the energy dependent Pu-240 capture sensitivity coefficient per unit energy at three different exposures. This figure illustrates the importance of self-shielding effects with exposure. (Analogous effects are observed with Pu-239). The region of the 1.058 eV resonance is much more important at beginning of exposure. The Maxwellian region, however, becomes more important towards the end of exposure. This shifting of importance occurs because as the Pu-240 builds up in the pin cell, the flux dips at the resonance. Since there is less flux, a change in a resonance parameter will perturb the few-group cross-section less than it would have at beginning of exposure. This strong exposure dependence of sensitivity coefficients illustrates the importance of considering the fuel cycle as a whole, rather than considering any single time.

Figure 1 of our other paper in this proceedings illustrates the Pu-239 fission cross-section, adjoint, and sensitivity coefficient as a function of energy. The cross-section is more than five times as large at its 0.3 eV resonance than at the Maxwellian peak; yet its detailed sensitivity is about 50% larger at the Maxwellian peak than at the resonance. This illustrates the importance of the hardened

Table 2. Fuel Cycle Cost Sensitivity to Thermal Group Cross-Sections for 1985

	BWR		PWR	
	Recycle	Throwaway	Recycle	Throwaway
Pu-239 (n,f)	-.522	-.688	-.560	-.753
Pu-239 (n,γ)	.439	.578	.478	.619
U-235 (n,f)	-.387	-.578	-.384	-.568
U-235 (n,γ)	.210	.232	.208	.232
H (n,γ)	.125	.169	.095	.135
Pu-241 (n,f)	-.094	-.123	-.099	-.130
Pu-241 (n,γ)	.068	.080	.073	.089
U-238 (n,γ)	.033	.151	.037	.147
Fission Product Lump (n,γ)	.037	.052	.035	.049
B (n,γ)	—	—	.024	.045
Xe-135 (n,γ)	.017	.024	.020	.029
Zr (n,γ)	.019	.025	.011	.015
Nd-143 (n,γ)	.018	.024	.019	.027
Rh-103 (n,γ)	.008	.011	.008	.011
Pu-240 (n,γ)	.005	.012	.009	.020
Eu-153 (n,γ)	.004	.005	.004	.006

Maxwellian energy range for thermal-group cross sections.

SUMMARY

Cross sections averaged over the thermal energy group have been shown to have an important economic role for light-water reactors. Pu-239 and U-235 have the largest sensitivities (Path B), followed by H, Pu-241, and U-238. Fission products and structural materials have much smaller sensitivities.

The methods used to compute the sensitivity of few-group cross sections to uncertainties in basic nuclear data have been presented

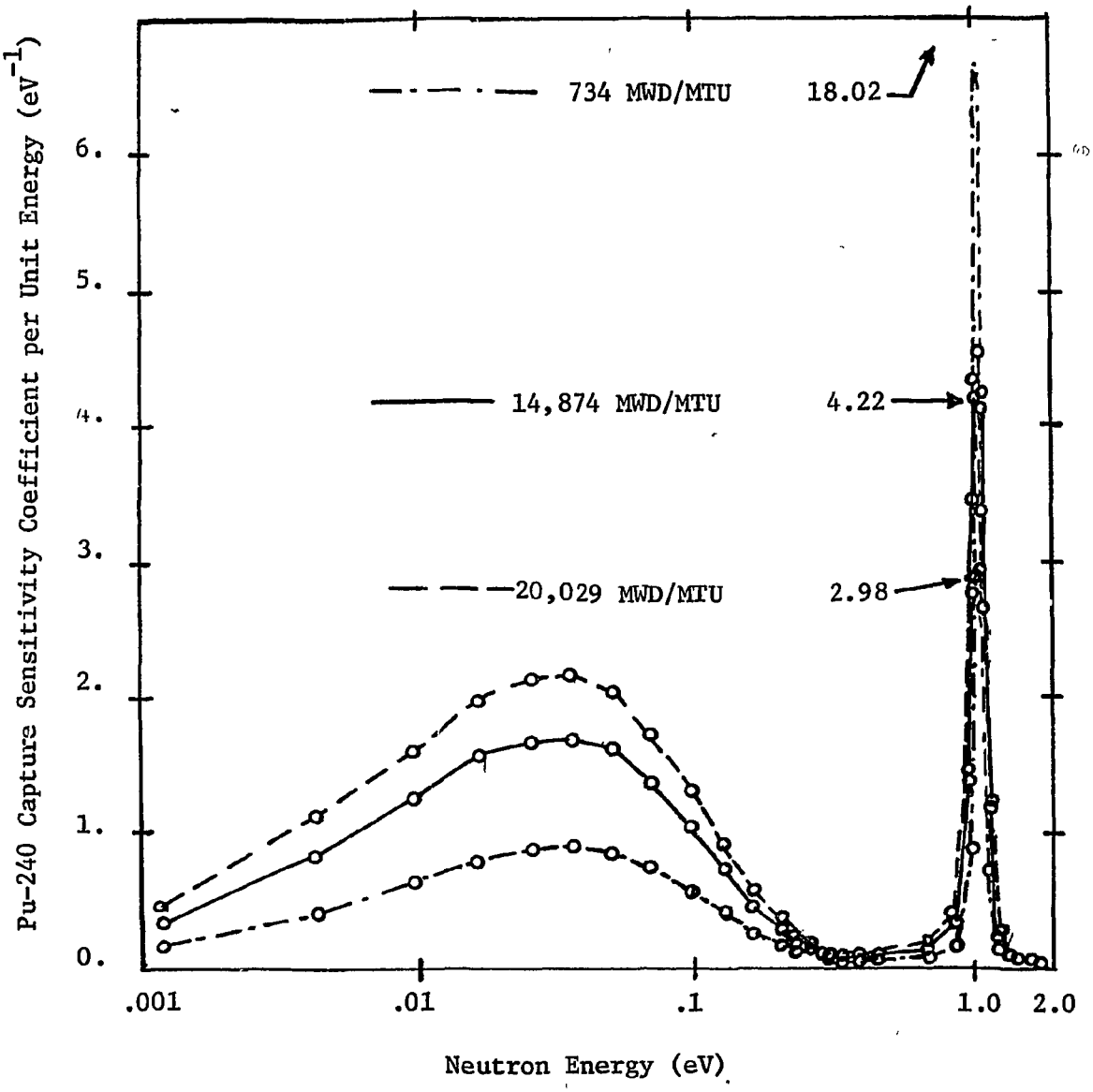

Fig. 1. Detailed Pu-240 Sensitivity Coefficient for the Montague BWR at Different Exposures.

and have been implemented in the FASTT code. Multigroup cross section sensitivity coefficients (Path A) vary with fuel exposure. A striking feature of our thermal analyses has been the (usually) overwhelming importance of the hardened Maxwellian energy region. Both Path A and Path B analyses indicate that it is very important to analyze the fuel cycle as a whole, rather than to consider one time only.

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REFERENCES

1. M. Becker, D. R. Harris, B. Quan, and J. M. Ryskamp, "Sensitivity of Water Reactor Fuel Cycle Parameters and Costs to Nuclear Data," these proceedings.

2. M. Becker, D. R. Harris, J. Parillo, A. Parvez, and J.M. Ryskamp, "Fuel Cycle Sensitivity Analysis System for Light-Water Reactors," Trans. Am. Nucl. Soc., 24, 216 (1976).

3. A. Parvez, M. Becker, D. Boguslawski, D. R. Harris and J. M. Ryskamp, "Fuel Cycle Cost Sensitivity Analysis for Boiling and Pressurized Water Reactors," Trans. Am. Nucl. Soc., 27, 468 (1977).

4. M. Danchak, W. R. Moyer and M. Becker, "The Rensselaer Interactive Graphics Analysis System," Trans. Am. Nucl. Soc., 18, 159 (1974).

5. C. F. Poncelet, "LASER, A Depletion Program for Lattice Calculations Based on MUFT and THERMOS," WCAP-6073 (1966).

6. R. F. Barry, "LEOPARD-A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," Westinghouse Electric Corporation report, WCAP-3269 (Sept. 1963).

7. H. Amster and R. Suarey, "The Calculation of Thermal Constants Averaged Over a Wigner-Wilkins Flux Spectrum: Description of the SOFOCATE Code," Westinghouse Electric Corporation report, WAPD-TM-39 (1957).

8. H. C. Honeck, "THERMOS, A Thermalization Transport Theory Code for Reactor Lattice Calculations," Brookhaven National Laboratory report, BNL-5826 (1961).

9. "CHEETAH-P," Nuclear Associates International Corporation report, (1974).

10. D. E. Cullen, C. R. Weisbin, R. Q. Wright, and J. E. White, "Comparison of Doppler Broadening Methods," NBS Special Publication 425, Vol. 1, 392 (1975)

11. E. P. Wigner and J. E. Wilkins, Jr., "Effect of the Temperature of the Moderator on the Velocity Distribution of Neutrons with Numerical Calculation for Hydrogen as Moderator," U. S. AEC report, AECD-2275 (1944).

12. M. Nelkin, "The Scattering of Slow Neutrons by Water," Phys. Rev., 119, 741 (1960).

13. L. E. Strawbridge, "Calculation of Lattice Parameters and Criticality for Uniform Water Moderated Lattices," Westinghouse Electric Corporation report, WCAP-3269-25 (1963).

14. J. M. Ryskamp, D. R. Harris and M. Becker, "Sensitivity of Light-Water Reactor Parameters to Uncertainties in Thermal Nuclear Data," Trans. Am. Nucl. Soc., 26, 581 (1977).

15. M. Danchak, W. R. Moyer, and M. Becker, "The Rensselaer Interactive Graphics Analysis System," Trans. Am. Nucl. Soc., 18, 159 (1974).

16. J. M. Ryskamp, "FASTCELL, A Code for Fast Computation of Reactor Cell Characteristics Over a Fuel Cycle," Master's Thesis, Rensselaer Polytechnic Institute (1976).