Sensitivity of a Linear Cross Section Model in 3-Dimensional Core Transient Calculations

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1. Introduction

In 3-dimensional core transient calculations, the cross sections are required for various reactor conditions, which are continuously changed during event. However, it is not easy to numerically generate cross sections for thermal-hydraulic conditions with fuel and moderator temperatures, density, boron, void, etc. The approximate formula that cross sections are linearly varied for these feedback variables is generally used in the 3-dimensional core dynamics code.

For example of PARCS code using macroscopic nodal cross sections, a set of macroscopic cross sections generated at a reference thermal-hydraulic condition is assigned for each fuel composition, and the sets of derivative cross sections are provided to describe the feedback variables as follows [1]:

$$\sum(B,T_f,T_m,\alpha,\xi)=\Sigma_0+a_1(B-B_0)+a_2(\sqrt{T_f}-\sqrt{T_{f0}})+a_3(T_m-T_{m0})$$
$$+a_4(D_m-D_{m0})+a_5(D_m-D_{m0})^2+a_6\alpha+a_7\alpha^2+\xi\Delta\Sigma_{cr}$$

where a_1~a_4 and a_6 are 1st order derivative constant, and a_5 and a_7 are 2nd order derivative constant. Also, ξ is the effective rodded fraction. These constants can be obtained from their derivatives for two reference values of the feedback variable to the corresponding macroscopic cross sections; e.g., $a_1=(\Sigma_1-\Sigma_0)/(B_1-B_0)$.

The previous studies indicate that the above linear approximation may be uncertain for the broad transients such as the rod ejection accident(REA) or steam line break(SLB) [2,3]. These studies show that nodal powers during REA and SLB may be largely dependent on how to select two reference values of these feedback variables for determination of the 1st and 2nd derivative constants.

This study clarifies the uncertainties of macroscopic cross sections as the selection of two reference values of boron concentration.

2. Uncertainty Analysis

Three types of fuel compositions of the following are selected for their analysis:

(1) Fuel Type 1 : Burnup 0, 3.798 w/o U^{235}
(2) Fuel Type 2 : Burnup 18200MWD/MTU, 3.798 w/o U^{235}
(3) Fuel Type 3 : Burnup 39500MWD/MTU, 3.798 w/o U^{235}

For case studies to examine the effect of two reference boron concentrations on cross sections, the 3 sets are selected as follows:

(1) Case 1 : (B_0=1000ppm, 2000ppm)
(2) Case 2 : (B_0=500ppm, 1000ppm)
(3) Case 3 : (B_0=1400ppm, 2000ppm)

Also, their uncertainties for fuel temperature in this study are examined, and thus other feedback variables except T_f and B are fixed to a base condition; ($T_{f0}=596.85^0C$, $T_m=T_{m0}=301.85^0C$, $D_m=D_{m0}=0.7225g/cm^3$, $\alpha=0$).

This study calculates the relative uncertainties of each macroscopic cross section for Case 2 and 3, compared with a basis value of Case 1. Two group macroscopic cross sections are calculated by CASMO-3.

2.1 *Example Results for Transport Cross Section of Group 1 (Σ_{tr}^1)*

<u>*Fuel Type 1*</u>

Transport cross sections for fuel type 1 are calculated for three cases of boron concentration. The following equations are obtained for each case, as shown in Figure 1:

(i) Case 1

$$\sum_{tr}^1=0.230543-0.822286\times10^{-4}(\sqrt{T_f}-\sqrt{T_{f0}})$$

(ii) Case 2

$$\sum_{tr}^1=0.230540-0.822296\times10^{-4}(\sqrt{T_f}-\sqrt{T_{f0}})$$

(iii) Case 3

$$\sum_{tr}^1=0.230549-0.819608\times10^{-4}(\sqrt{T_f}-\sqrt{T_{f0}})$$

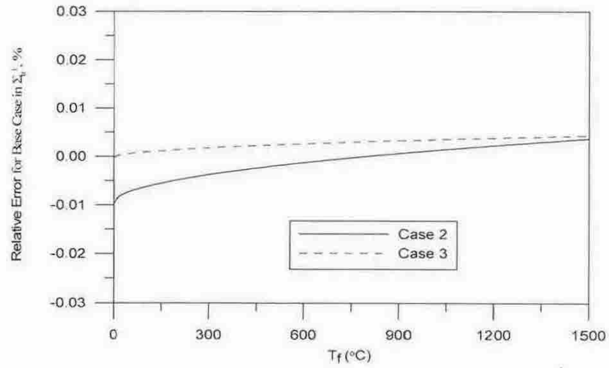

Figure 1 Relative Error of Σ_{tr}^1 for Fuel Temperature (Fuel Type 1)

<u>*Fuel Type 2*</u>

Transport cross sections for fuel type 2 are obtained in the same way as fuel type 1 before. Figure 2 shows their relative errors in Case 2 and Case 3 to Case 1.

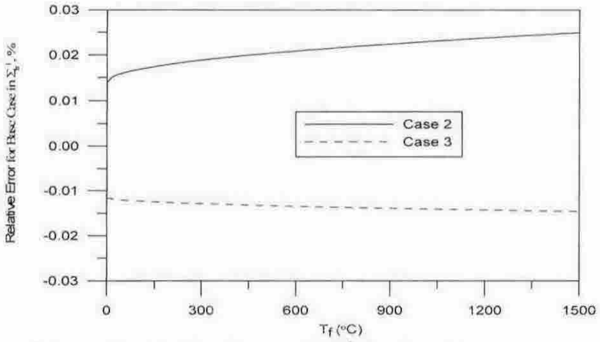

Figure 2 Relative Error of Σ_{tr}^{1} for Fuel Temperature (Fuel Type 2)

Fuel Type 3

Transport cross sections for fuel type 3 are obtained in the same way as fuel type 1 before. Figure 3 shows their relative errors in Case 2 and Case 3 to Case 1.

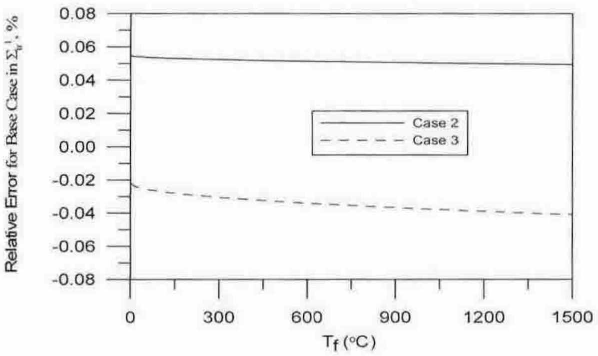

Figure 3 Relative Error of Σ_{tr}^{1} for Fuel Temperature (Fuel Type 3)

2.2 Uncertainty Results

In the case of two energy groups, macroscopic cross sections considered in PARCS are transport, absorption, fission, down scattering. Their uncertainties are verified in the same way as Section 2.1, for 9 macroscopic cross sections of the following:

- Group 1 : Σ_{tr}^{1}, Σ_{a}^{1}, $\nu\Sigma_{f}^{1}$, $\kappa\Sigma_{f}^{1}$, Σ_{12}^{1}
- Group 2 : Σ_{tr}^{2}, Σ_{a}^{2}, $\nu\Sigma_{f}^{2}$, $\kappa\Sigma_{f}^{2}$

Table 1 shows the uncertainties for these cross sections. For the sake of convenience, the relative errors for only $T_f=0\,^oC$ and $T_f=1500^oC$ are given in Table 1.

Table 1 Relative Errors of Macroscopic Cross Sections for Fuel Temperature

(unit: %)

		Fuel Type 1		Fuel Type 2		Fuel Type 3	
$T_f(^oC)$		0	1500	0	1500	0	1500
Σ_{tr}^{1}	Case 2	-0.009	0.003	0.014	0.024	0.055	0.049
	Case 3	0.000	0.004	-0.011	-0.015	-0.022	-0.041
Σ_{a}^{1}	Case 2	0.027	-0.021	-0.220	-0.220	-0.290	-0.250
	Case 3	-0.017	0.018	0.190	0.180	0.270	0.200
$\nu\Sigma_{f}^{1}$	Case 2	-0.006	-0.001	-0.680	-0.660	-1.570	-1.560
	Case 3	0.007	0.007	0.540	0.540	1.490	1.480
$\kappa\Sigma_{f}^{1}$	Case 2	-0.006	-0.001	-0.670	-0.650	-0.190	-0.185
	Case 3	0.023	0.019	0.530	0.510	1.500	1.450
Σ_{12}^{1}	Case 2	-0.025	0.005	0.151	0.201	0.400	0.430
	Case 3	0.040	0.015	-0.057	-0.130	-0.252	-0.301
Σ_{tr}^{2}	Case 2	0.028	-0.007	-0.072	-0.100	-0.114	-0.158
	Case 3	-0.037	-0.019	0.048	0.077	0.080	0.119
Σ_{a}^{2}	Case 2	-0.168	-0.117	-1.120	-1.080	-2.300	-2.300
	Case 3	0.208	0.170	0.880	0.890	1.800	1.800
$\nu\Sigma_{f}^{2}$	Case 2	-0.030	-0.024	-1.330	-1.400	-3.400	-3.400
	Case 3	0.178	0.160	1..050	1.100	2.600	2.600
$\kappa\Sigma_{f}^{2}$	Case 2	-0.030	-0.020	-1.225	-1.300	-3.300	-3.300
	Case 3	0.175	0.165	0.420	0.440	2.490	2.510

3. Conclusion

The uncertainties of macroscopic cross sections in the linear cross section model used in the 3-dimensional core dynamics methodology were examined for some cases. This study shows that relative errors become large for fuel type 3 and energy group 2, where fuel type 3 represents high-burnup fuels. As a result, it is concluded that absorption and fission cross sections of thermal neutron of the high-burnup fuels were relatively sensitive to the selection of two reference values of boron concentration.

Various uncertainty analyses can give the guideline to select reference values of feedback variables. Moreover, it is possible to find out the best way to select these values.

REFERENCES

[1] "User Manual for the PARCS-v2.3Beta Kinetics Core Simulator Module," Chapter 9, US NRC, Mar. 2003.

[2] C. Yang, et. al, "Coupled 3-D Reactor Kinetics and Thermal-Hydraulics Analysis for a SLB Accident of an Operating Nuclear Power Plant by using RELAP5/PARCS Code," International Meeting on Mathematical Methods for Nuclear Applications, Salt Lake City, USA, Sep. 2001.

[3] C. Yang, et. al, "Development of Regulatory Technology on a 3-D Core Analysis of Rod Ejection Accident," KINS/RR-184, Mar. 2003.