BURN-UP PHYSICS IN A COUPLED HAMMER-TECHNION/CINDER-2 SYSTEM AND ENDF/B-V AGGREGATE FISSION PRODUCT THERMAL CROSS SECTION VALIDATION

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ABSTRACT

The effect of the fission product poisoning in a thermal reactor has been explicitly addressed. The proposed scheme is based in a coupled HAMMER-TECHNION/CINDER-2 system. The fission product chain treatment considers nearly 99% of all original CINDER-2 fission product neutron absorption.The calculational methodology and the ENDF/B-V fission product nuclear data performances have been investigated by comparing the calculated aggregate fission product neutron absorption against the available experimental data. Good agreement has been found for the fission product quantity σ_{2200} for U235 and U233. The Pu239 calculated values are not in good agreement.

INTRODUCTION

Fission product poisoning in nuclear reactors is one of the most important aspects in reactor technology .The accurate prediction of neutron absorption by fission products is extremely important in the fuel cycle strategy of a nuclear reactor. In the past, the lack of adequate nuclear data sets and the unknown fission product transmutation behavior have imposed severe restrictions in the predictions of fission product poisoning in nuclear reactors.However, since the development of the CINDER [1] computer code in 1962 and after the release of ENDF/B-IV [2-6] and ENDF/B-V [7] libraries there has been a significant progress in these areas. Furthermore the computer technology development has contributed for the use of more sophisticated approaches in reactor physics analysis such as an explicit fission product treatment that was made available by the development of the CINDER computer code.

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Finally,from the point of view of cross section pre–processing code qualification, the IAEA Verification Project on Pre–Processing Code [8] has significantly contributed for error reduction in several widely used pre–processing codes so that in the comparison between calculated and measured integral parameters, the error due to the pre–processing code has been significantly reduced.

The present status in fission product poisoning is that one can predict it quite accurately starting from the present time knowledge together with the computational and processing code resources, without any further approximation like in the lumped absorber approach.

Based on the above explanations, the present work has the following purposes:

a) To implement a burn–up capability into the HAMMER–TECHNION [9] computer code by using the CINDER–2 [10] computer code to perform the transmutation analysis for the actinides and fission products.

b) To implement a reduced version of the CINDER–2 fission product chain structure to treat explicitly nearly 99% of all original CINDER–2 fission product neutron absorption in a typical PWR unit cell.

c) To treat the effect of the fission product neutron absorption in an unit cell in a multigroup basis.

d) To develop a tentative validation procedure for the ENDF/B–V stable and long–lived fission product nuclear data based on the available experimental data [11-14]. The analysis will be performed by using the reduced chain in the coupled system HAMMER–TECHNION/CINDER–2 to generate the time dependent effective four group cross sections for actinides and fission products and CINDER–2 to perform the complete transmutation analysis with its built–in chain structure.

PREVIOUS STUDIES

The study of the fission product neutron absorption started in 1950 [15-18] with the development of the first nuclear power reactor. However, it was only after 1960 that the field of fission product neutron absorption has experienced a significant progress.

In early 1960 the chain structure to analyze transmutation analysis of fission product was very simplified. Walker [19];and Greenhow and Hansen [20] used 2 to 3 coupled nuclides to analyze fission product buildup. England and Eckert [21] extended the analysis to allow 5 nuclides per chain. With the advent of CINDER computer code the transmutation analysis of the fission product was generalized. The version of the CINDER computer code published by England, in 1962, used 163 fission products in 67 linear chains. Acomparison with a integral experimental was performed. At this time it was recognized the tremendous amount of nuclides and nuclear data needed to treat explicitly the fission product poisoning during a nuclear reactor life.

This computer program has experienced progress through the years. In 1965 [22] it incorporated 179 fission products in 69 linear chains. After the release of the ENDF/B–IV evaluated fission product data , it was made a version [23] based on this file which incorporated 186 fission product in 84 linear chains and finally with the ENDF/B–V the version denominated CINDER–2 incorporated 211 fission products in 102 linear chains.

Parallel to that, a number of simplified models was developed to analyze fission product poisoning, Xe135 and Sm149 classically known as heavy absorbers were treated explicitly and the remainder was treated in one or more groups.

Garrison and Roos [24] model treated the fission poisoning in three distinct groups according to the fission products behavior: slowly saturating, rapidly saturating and non-saturating. Nephew [25] treated the fission products according to their cross-sections. The fission products with high cross sections were treated explicitly and the remainder were included in pseudo fission product. Other models [26-31] basically are an adaptation of these two models.

The simplified models mentioned in this review has been incorporated in a variety of cell computer code [32-33].

From the point of view of nuclear data, the ENDF/B-IV fission product file was the first file to be made in a sistematic and organized way so that every single data could be referenced to an unique source. The ENDF/B-V fission product file has even more increased the knowledge of the fission product nuclear data.

FISSION PRODUCT CROSS SECTION GENERATION

The computational strategy starts with the NJOY [34] processing code system and with the ENDF/B-V fission product library. Here the multigroup libraries in 54 fast groups and 30 thermal groups for the fission products of interest are elaborated. The only exceptions were Gd-155 and Gd-157 which are not available in the ENDF/B-V fission products files. These two fission products were taken from the original ENDF/B-IV HAMMER-TECHNION library. The fission product multigroup libraries were generated at infinite dilution and at 300K .

The output of module GROUPR of NJOY is transformed into the Master library of AMPX-II [35] by using the interface program (AMPXR) [36] developed at IPEN/CNEN-SP. The objective here is to check the consistency of the group cross sections generated by NJOY through the use of the module RADE of AMPX-II . The final binary fission product libraries for the HAMMER-TECHNION are subsequently prepared by two special versions of the NITAWL module and by the HELP [9] and LITHE [9] programs. The LITHE program has been also modified to handle the thermal scattering matrices produced by GROUPR.

HAMMER-TECHNION/CINDER-2 COUPLING

Basically the unit cell coupled set of transmutation and transport equations is solved by the quasi-static approach [37]. The transport spectral calculations are performed at the beginning of each time-step by the HAMMER-TECHNION and the transmutation equations for the fission products and actinides are solved by the CINDER-2 every time step, assuming constant values for the collapsed four group cross sections, the asymptotic flux ratios and the power density.

The HAMMER-TECHNION spectral calculations are performed taking into account the fission product contribution to the multigroup macroscopic absorption cross sections (fast and thermal) , in the $\bar{\mu}\sigma_s$, in the $\xi\sigma_s$, in the inelastic scattering matrix and even in the thermal scattering matrices.

TRANSMUTATION CHAIN

The elaboration of the reduced fission product chains was the hardest part of this work .

Firstly, it was necessary to find out the main fission product neutron absorbers and the principal chain branches of these fission products. Also, some original linear chains of CINDER–2 have been broken down when the coupling between fission products was found to be weak. The proposed set contains 98 fission products in 51 linear chains. The fission products choice criteria was to include all fission products such that the reduced chain neutron absorption is about 99% of that of the original CINDER–2 set.

Besides that, the original CINDER–2 actinide linear chains were reduced to include only the main actinides of interest in LWR burn–up physics: 32 actinides were included in 38 linear chains. Finally the whole ENDF/B–V CINDER–2 20 set fission product yields were kept throughout the analysis.

FISSION PRODUCT METHODOLOGY VALIDATION

The available experimental data for fission product neutron absorption in the literature are scarce, have complex nature and do not have the simplicity for benchmark testing to be reproduced by model calculations .Basically, these experiments consist of three distinct phases: i) an irradiation period followed by a cooling period, ii) reactivity measurement of the irradiated sample by a pile oscillator technique [38] and iii) the obtention of the aggregate fission product absorption parameters from the measured reactivity by using an analytical model based on perturbation theory[13 39]. Provisions are made for not considering the actinide and structural material contributions.

There are four available experiments [11-14] in the literature.The only experiment analyzed in this work is the experiment of Okazaki and Sokolowski [14], due to its good quality and also due to its applicability to the LWR environment.

In the experiments of Okazaki and Sokolowski, samples of U–235, U–233 and Pu–239 were irradiated in the Canadian NRU reactor from April 1960 to May 1961. The reactivity of the samples was measured in Swedish pile oscillator facility in the RO reactor at Studvisk. The experiment of Okazaki and Sokolowski was analyzed by Walker [14] with the FISSPROD [40-41] computer code. His analysis was based on the Wescott formalism [42].

The quantities reported in the experiment of Okazaki and Sokolowski are 2200m/s barns/fission effective cross sections for the aggregate fission products of each sample. No resonance integral data for the fission products have been reported. The neutron spectrum used in this work in the collapsation of the fission product and actinides cross sections was calculated by the coupled system HAMMER–TECHNION/CINDER–2 using a three region model: sample , aluminum sheath , and the heavy water moderator. The heavy water region was considered wide enough such that the sample can be considered isolated. The coupled system, HAMMER–TECHNION/CINDER–2 calculations were performed at constant neutron flux.

The HAMMER–TECHNION/CINDER–2 calculations were also performed during the cooling time. The reason was to calculate the collapsed cross sections of the fission products at the end of the cooling period so that the collapsed fission product cross sections were averaged in the proper spectrum; i.e, without the short–lived fission products and actinides and also with the proper temperature.

The fission product and actinide time–dependent four group microscopic cross section set, four group flux ratios (including the cooling period) calculated by the coupled system HAMMER–TECHNION/CINDER–2 were then transferred to CINDER–2. For the fission products and actinides not included in the reduced chain set of HAMMER–TECHNION/CINDER–2 coupled system the procedure was to use the CINDER–2 cross sections modified by the usual $<\sigma_{1/v}>$ cross section [23].

The resulting values of the aggregate fission product thermal cross sections in units of barns/fission calculated by CINDER–2 at the end of the cooling period were then transformed to aggregate effective 2200 m/s thermal cross sections by dividing them by the $<\sigma_{1/v}>$ cross section evaluated at the Studvisk experiment heavy water hole spectrum of each sample.

The irradiation history was assumed to be at uniform flux and the burn–up of each sample was determined by adjusting the magnitude of the neutron flux to match the published depletion of the fissile nuclide (U–233, U–235, or Pu–239) since it was not possible to use the Nd–148 technique [43] due to the unavailability of the concentration of this nuclide.

The final results are shown in the Table I along with the corresponding results obtained by Walker [14].

TABLE I
Calculated and Measured Values of σ_{2200}

Fissile Isotope/ Sample (*)	BARNS / FISSION (2200 m/s)		
	calculated/measured		measured /14/
	This work	Walker /14/	
U233/A3B	1,0111	1,05	41,9 +– 2,5
U233/A3F	1,0340	1,10	38,9 +– 1,8
U235/A5B	1,0156	1,07	48,7 +– 2,2
U235/A5F	1,0075	1,08	46,5 +– 1,6
Pu239/A9A	0,8747	1,07	62,3 +– 3,3
Pu239/A9D	0,8047	0,96	71,1 +– 3,8

(*) Sample description and experiment details can be found in Ref.14.

CONCLUDING REMARKS

The new methodology developed in this work to treat fission product neutron absorption shows that the explicit multigroup treatment in a thermal reactor lattice code can be achieved without too much computational effort.
The proposed fission product treatment considers nearly 99% of all CINDER-2 fission product neutron absorption, therefore apart from nuclear data uncertainties the procedure meets the required accuracy for a thermal reactor lattice code [44]. Besides that, the procedure can be extended to include all fission products treated by CINDER-2 chains in a straightforward fashion.

The advantage of this explicit treatment is that the thermal reactor lattice code becomes independent of the application and can even be used to produce lumped fission product cross sections to be used in other codes.

In all calculations the values obtained for (σ_{2200}) for U-235 and U-233 are in excellent agreement with the experimental results reported by Okazaki and Sokolowski. The reported values of σ_{2200} measured in Okazaki and Sokolowski experiments are consistent with the values measured by Gunst et al for U-233 [13], and by Nisle for U-235 [11]. This consistency makes the validation procedure of (σ_{2200}) for U-233 and U-235 reliable.

However for Pu-239 samples there is no much experimental results available and we must rely on the consistency of Okazaki and Sokolowski results. The Pu-239 results obtained using ENDF/B-V are not in good agreement with the experimental results. The four group microscopic cross sections of the fission products utilized in the CINDER-2 calculations were a few percent different from sample to sample. Since σ_{2200} is primarily a function of the fission product yield and fission product cross sections, part of the difference might be credited to the fission product yield of Pu-239 and Pu-241. Partly due to the lack of any other similar experiment for the Pu-239, partly due to the complex nature of the experiment of Okazaki and Sokolowski and partly due to the lack of any other similar comparison besides the one performed by Walker , very little can be said about the discrepancies found in the samples of Pu-239.

The field of fission product neutron absorption has reached a state of the art that may be considered mature. Today, there is a plenty of nuclear data and scientific knowledge of the fission product transmutation behavior. Besides that the computer technology development together with the processing code resources allows one to perform Burn-up Physics calculations taking into account the effect of the fission product neutron absorption in a straightforward fashion. The missing link is the validation procedure for the fission product nuclear data in order to be able to access their applicability in a Burn-up Physics code. Very few work have been done on the validation of the aggregate fission product nuclear data so that the present work has definitively contributed to access the applicability of the ENDF/B-V fission product data in the LWR environment.

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