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THE CROSS SECTIONS OBTAINED BY THE SERPENT CODE AND FORMATTING THE INPUT DATA FOR THE PARCS CODE USING THE GenPMAXS CODE

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ABSTRACT

The Purdue Advanced Reactor Core Simulator (PARCS) is a computer code that solves the time-dependent two-group neutron diffusion equation in three-dimensional Cartesian geometry using nodal methods to obtain the transient neutron flux distribution. The code is used in the analysis of reactivity-initiated accidents in light-water reactors where spatial effects may be important. It may be run in the stand-alone mode or coupled to other NRC thermal-hydraulic codes such as RELAP5. The PARCS neutron code accepts libraries from HELIOS, TRITON, WIMS, SERPENT, etc., codes, but for some libraries is required special formatting. In the case of the SERPENT code, the GenPMAXS code must be used for the PARCS code to be able to read the cross sections library correctly. This work is part of a study on the PARCS/RELAP5 coupling for analyzing the control rod ejection of the Angra 2 reactor core. For this case, the core cross sections were obtained for 6 different branches varying the fuel temperature, moderator temperature, moderator density, boron concentration and considering rods removed and inserted. After obtaining the cross sections with the code SERPENT 2.1.26, these data were passed by a special formatting realized with the code GenPMAXS v6.2. Since GenPMAXS has several options controlling how to process the cross-sections generated by Serpent, a several doubts arose about the correct use of the code. When the doubts are answered, the file with the input data that will be used for the PARCS / RELAP coupling can be built.

1. INTRODUCTION

For safety analysis of potential power reactor core-related accidents, the use of modern and reliable computational tools accepted by the responsible government agencies for building and operating license for nuclear installations is required. These tools are not always user friendly and easy to understand and use.

One of the tools used for control rod ejection accident analysis is the Purdue Advanced Reactor Core Simulator (PARCS) code [1] which is a three-dimensional (3D) reactor core simulator that solves steady and transient state (such as such as control rod movement, boron ejection, etc.), multi-group neutron diffusion and transport equations in orthogonal and non-orthogonal geometries. The program can be coupled to the RELAP5 code [2] using the PVM interface,

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which provides PARCS temperature and flow information during transient calculations by generating cross sections in a few groups. PARCS is available as a standalone code for performing non-coupling calculations.

A separate module, GenPMAXS [3], is used to process cross sections generated by codes such as TRITON, HELIOS, CASMO or SERPENT for PMAXS [3] format that can be read by PARCS. The use of the GenPMAXS module differs for each code used in generating the cross-section library. Thus, this work aims to verify the operation of this module to receive the cross sections obtained through the SERPENT code 2.1.26 [4].

2. METODOLOGY

The final objective of this project is to analyze Control Rod Ejection Accident (CRE) in Angra 2 NPP using coupled PARCS/RELAP. One of the key steps is to generated Angra 2 cross-section library dependent of TH conditions suitable to be used in CRE accident conditions. This library shall be readable by PARCS code. To perform lattice calculation and obtain cross sections as function of TH conditions and burnup SERPENT 2,1,26 Monte Carlo code has been chosen. SERPENT calculations output file can be processed, can be processed by GenPMAXS, in some cases, some minor editions are needed to adapt the output to GenPMAXS. In the following section the methodology used are presented.

2.1. SERPENT Code

Serpent [4] is a multi-purpose three-dimensional continuous-energy Monte Carlo particle transport code, developed at VTT Technical Research Centre of Finland, Ltd. The development started in 2004, and the code has been publicly distributed by the OECD/NEA Data Bank and RSICC since 2009. Serpent started out as a simplified reactor physics code, but the capabilities of the current development version, Serpent 2, extend well beyond reactor modeling. The applications can be roughly divided into three categories:

1) Traditional reactor physics applications, including spatial homogenization, criticality calculations, fuel cycle studies, research reactor modeling, validation of deterministic transport codes, etc.
2) Multi-physics simulations, i.e. coupled calculations with thermal hydraulics, CFD and fuel performance codes
3) Neutron and photon transport simulations for radiation dose rate calculations, shielding, fusion research and medical physics

Serpent has the capability to perform branch calculations, changing TH conditions or materials for each burnup step. As first tentative, Serpent input as asked to calculated cross-sections for 750 branches, varying the branches in fuel and moderator temperatures, boron dilution, moderator density and fuel burning, for the inserted control rod and removed control rod cases. However, it was difficult for a GenPMAXS input build to receive such a large file, and the time taken to resolve the difficulties proved unfeasible. Thus, it was decided to generate a six fuel temperature branches for six different burnups to verify the feasibility of using the SERPENT file in GenPMAXS. The input data for the SERPENT code has been selected:

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- Control rod inserted;
- No boron dilution;
- Moderator density = 1 g/cm^3;
- Moderator temperature = 300 K;
- Fuel temperature = 873.00 K; 895.15 K; 573 K; 1073 K e 1473 K; and
- Burnup = 0; 0.10; 1.5; 5; 15 e 25 MWd/kgU.

2.2. GenPMAXS Code

PARCS [1] is a neutronics code for the prediction of nuclear reactor core steady-state and transient behavior at a specific core burnup state. PARCS solves the steady-state and time-dependent neutron diffusion and SP3 transport equations in three-dimensional geometry to obtain the neutron flux distribution. Because PARCS is capable of performing core eigenvalue calculations, as well as such functions as analyzing the control rod movement and searching for the critical boron concentrations, it has the functionality necessary to analyze both short term (kinetic) and longer term (depletion) core behavior. In order to provide depletion capability to the PARCS code, a depletion module was added to PARCS, as well as a new cross section module for retrieving node-wise cross section for its burnup history and current thermal-hydraulic state from PMAXS. The overview of PARCS package for core depletion analysis is shown in Figure 1. The depletion module generates new burnup and other history state information corresponding to the PARCS neutron flux solution. The cross-section module calculates cross section based on burnup and other history state information, as well as on the current thermal-hydraulic state. The PARCS neutronic module then calculates the neutron flux with the cross sections generated by the cross-section module [3].

Currently, PARCS employs a macroscopic depletion method in which the microscopic cross sections and the fuel number densities are not tracked individually during core depletion. Only the macroscopic cross sections are determined by the depletion module with burnup and history indices. The macroscopic cross sections at the appropriate fuel conditions are prepared using a lattice physics code such as HELIOS, CASMO, TRITON, SERPENT etc. The PMAXS file is constructed using the output of the lattice code and provides the cross-section data in the specific format that can be read by PARCS. Because the output format of each lattice code is unique, the cross-section processing program, GENPMAXS [3], was written to process the output of the lattice codes and prepare the PMAXS [3] formatted cross section. The GENPMAXS [3] (Generation of the Purdue XS set) code is therefore the interface between the lattice code and the depletion code. It processes the output of the lattice code and generates the PMAXS formatted files.

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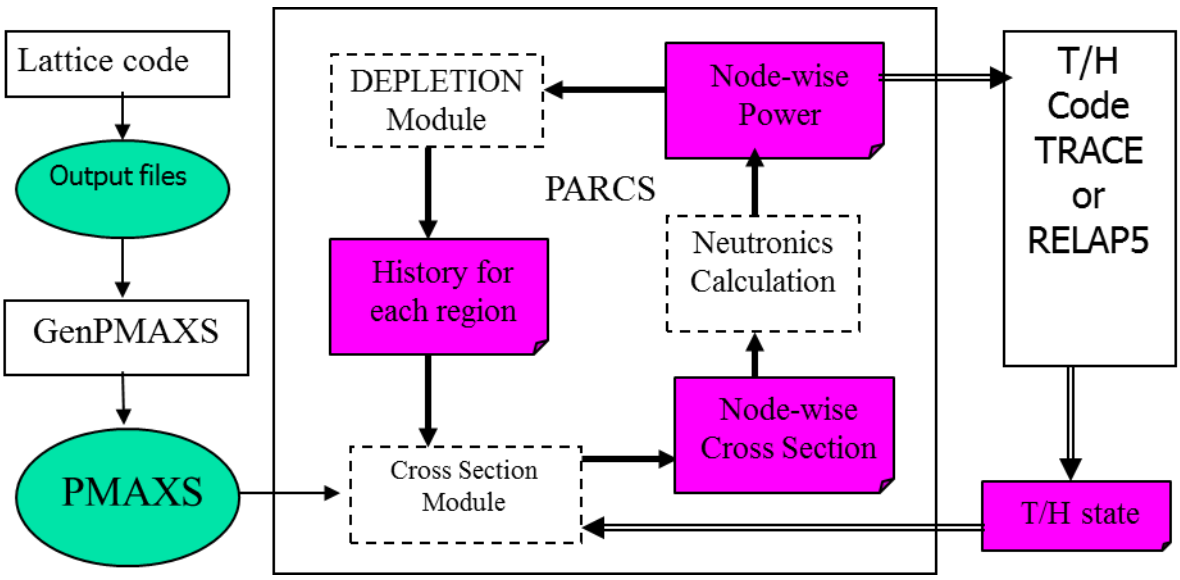

Figure 1: Overview of PARCS Code System for Core Depletion Analysis.

Thus, the cross sections generated by the SERPENT code were read by the GenPMAXS module for correct formatting for use by the PARCS code.

The SERPENT code generates a cross section file for the data presented in section 2.1 with extension .m that should be read from the GenPMAXS input.

The first card of GenPMAXS input contains several logical flags signalizing to GenPMAXS to include or not data from lattice code output to PMAXS file, for example the user can choose if neutron kinetic parameters will be read from lattice code output or a default value will be used.

Other cards specified the lattice code utilized (SERPENT2 in this case), the number of lattice code output files, type of fuel assembly, etc. All the branches variable and values shall be specified in GenPMAXS input as well as the burnup steps considered. The branch variables are fraction of insertion of control rods, moderator density, fuel temperature, etc.

The generated PMAXS file has the following form:

```
GLOBAL_V      1 2 6 0 4 0  0 0 T F F F T F T F F F F T T F T

 Contents of T/H Invariant Variables(TIV) block and Cross Sections(XS) block
      TIV:Chi,inV/Bet/Lam/
       XS:tr,ab,nf,kf/sct/ADF/

 2  Group value of each variable are put together in a line.
 Some variables(separated by ",") share a line,"/" means change line
 Generated by GenPMAXS-V6.2co
```

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```
STA_VAR    5 CR DC PC TF TC

BRANCHES   1   0   0   0   5   0
   RE   1     1.00000     1.00000     0.00000   873.00000   300.00000
   TF   1     1.00000     1.00000     0.00000   573.00000   300.00000
   TF   2     1.00000     1.00000     0.00000   895.15000   300.00000
   TF   3     1.00000     1.00000     0.00000  1073.00000   300.00000
   TF   4     1.00000     1.00000     0.00000  1473.00000   300.00000
   TF   5     1.00000     1.00000     0.00000  2173.00000   300.00000

BURNUPS    1
   1   6 0.00000 0.10000 1.50000 5.00000 15.0000 25.0000

XS_SET  00000001  1 4 0  0  0 0 0.00000 0.00000 0.00000 1.41178 0.73580 0.00000 0.00000
0.00000

HISTORYC   1     1.00000     1.00000     0.00000   873.00000   300.00000

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3. CONCLUSIONS

GenPMAXS version 6.2 proved to be suitable for formatting the SERPENT cross sections file by correctly obtaining the PMAXS file for a few branches in a very short computation time. The cross sections were verified to those from SERPENT, and are correct.

The next step in the work is to be able to group the cross sections for various fuel and moderator temperature, moderator density, boron concentration, burnup, and control rod inserts branches into a single cross section file. This will be done for the actual information from the Angra II reactor.

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