## Implicit Local Void Model for Cross Section Generation in the APOLLO2-A - ARTEMIS<sup>TM</sup> BWR Core Simulator Methodology

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**Abstract** - In this paper, we present a new approach to account for the local heterogeneity in the moderator density for Boiling Water Reactor cross section generation. The so-called Implicit Local Void Model (ILVM) functionalizes the water density distribution as a function of lattice type, average void fraction, and burnup. The reference water density distributions that are used in the ILVM are calculated from the spectral code APOLLO2-A coupled with the subchannel code F-COBRA-TF<sup>TM</sup>. Then the resulting local variations are implicitly captured in the neutronics data (cross sections, pin form factors, etc.) generated by APOLLO2-A. The ILVM is tested on the newly developed APOLLO2-A/HERMES/ARTEMIS<sup>TM</sup> [1] BWR computational chain for two different equilibrium cycles (12 months and 24 months, respectively) based upon AREVA's ATRIUM<sup>TM</sup> 11 assembly. The effects of the ILVM are found to be very dependent on the assembly neutronic design. For the first equilibrium cycle, eigenvalue trends, peaking factors, and projected thermal limits during the cycle are found to be mostly unaffected by the ILVM, while the second one had a decrease in most of the thermal limits and peaking factors. The change in the neutron flux spectrum introduced by the functionalization of the radial moderator density in the ILVM model causes an increase in burnable poison worth, and as a result, the impact of the model is stronger on assembly designs with high gadolinium content.

### I. INTRODUCTION

In most Light Water Reactor (LWR) core design methodologies, nodal diffusion methods are used together with cross sections that are both homogenized in space and condensed to a few energy groups. During the cross section generation process, the water density distribution is usually assumed to be uniform at the 2D lattice level. However, in Boiling Water Reactors (BWR) we know this assumption to be incorrect (see chapter 6.4 of [2]). Depending on the flow regime, the vapor will migrate to the open areas and the liquid will surround the structures. This vapor effect causes non-uniformities in the water density distribution. Non-uniformities in the water density distribution are also reinforced by non-uniformities in the rod powers. The non-uniformities in the rod powers are due to variations in the uranium enrichment, gadolinium rods, partial length rods, water rods/channels, and presence of the control blades.

There has been many articles that try to estimate the effect of this assumption. In [3], the maximum effect of the water density distribution was  $0.05\%\Delta k/k$  on the multiplication factor and 4.3% on the pin power peaking factor. This analysis was done without modeling any gadolinium or a control blade. Similar research was done here at AREVA on the ATRIUM<sup>TM</sup> 11 assembly in [4]. It was shown that the main effect of the water density distribution was that it increased the worth of the gadolinium and the control blade. Using a high gadolinium concentration, the maximum difference in the multiplication factor was about ±500 pcm depending on the axial location and the maximum difference in the radial peaking factor was about ±2.5%. Modeling the control blade increased these differences further. In [5], the maximum difference

ference in the multiplication factor due to the water density distribution was found to be about 500 pcm and about 4.5% in the difference in pin powers. This analysis was done with gadolinium pins but the presence of a control blade was not studied.

All the previous research on this topic has been done at the assembly level. The main interest of this topic is the effect at the core simulator level on global quantities such as the eigenvalue trend, flux maps, detector responses and thermal limit predictions. While the research done so far gives a good idea that the water density distribution is not negligible; it does not give the differences at the full core level since in reality there will be a range of axial power profiles, assembly powers, inlet thermal-hydraulic (TH) conditions, and radial boundary conditions. To fully get the effect of the water density distribution, the water density distribution must be taken into account in the core simulator. In this paper we propose the Implicit Local Void Model (ILVM) which is a model that functionalizes the water density distribution that is used during cross section generation. While the core simulator will still rely on lattice homogenized neutronic data (such as macroscopic cross sections, discontinuity factors, pin form factors, etc.) the water density distribution will then be inherently captured in the neutronic data, and thus represented at the core simulator level.

The reference water density distributions used to create the ILVM comes from a dedicated coupled calculation relying on the spectral code APOLLO2-A and the thermal-hydraulic subchannel code F-COBRA-TF<sup>TM</sup>. The coupling scheme is described in Section II. The ILVM is tested using the BWR computational chain APOLLO2-A/HERMES/ARTEMIS<sup>TM</sup> which is described in Section III. The description of the Im-



Fig. 1: Flow Chart for the Coupled Code

plicit Local Void Model is given in Section IV. The geometry that is used to create the ILVM is the AREVA ATRIUM<sup>TM</sup> 11 assembly and is described in Section V. The results of taking into account the ILVM model is given in Section VI.

# II. APOLLO2-A / F-COBRA-TF<sup>TM</sup> COUPLING

The reference water density distributions used to create the ILVM are calculated using a coupled APOLLO2-A/F-COBRA-TF<sup>TM</sup> code. APOLLO2-A [6] is a deterministic multi-group spectral code that uses the Method of Characteristics. F-COBRA-TF<sup>TM</sup> [7] is a thermal hydraulic subchannel code. F-COBRA-TF<sup>TM</sup> is a three phase, two fluid code that solves the mass, momentum, and energy conservation equations. These codes are externally coupled using Python.

The flow chart for the coupled calculation is shown in Fig. 1. In the coupling, APOLLO2-A calculates the eigenvalue and the pin power distribution. The pin power distribution is the feedback parameter that is passed to F-COBRA-TF<sup>TM</sup>. F-COBRA-TF<sup>TM</sup> calculates the water density and temperature distributions which are feedback parameters for APOLLO2-A. The convergence of the iterative scheme between APOLLO2-A and F-COBRA-TF<sup>TM</sup> is monitored through the relative difference in the eigenvalue and the absolute difference in the channel average density.

The geometry mesh of both codes are different and a mapping is created to provide accurate passing of feedback parameters. APOLLO2-A calculates the pin powers for each axial node while F-COBRA-TF<sup>TM</sup> requires an average pin power. The axial pin powers are collapsed to an average pin power using:

$$f_{i} = max(N) \frac{\sum_{j=1}^{N} f_{i,j} p_{j}}{\sum_{i=1}^{N} N_{i} p_{i}}$$
(1)

where f is the pin power, i is the rod number, j is the axial node, p is the axial power profile, and N is the number of rods in a lattice. F-COBRA-TF<sup>TM</sup> calculates subchannel centered water densities while APOLLO2-A requires rod centered subchannel water densities. The water densities are area averaged to calculate the rod centered subchannel water densities as:

$$\rho_{i,j} = \frac{\sum_{k \in subchannels} w_k a_{k,j} \rho_{k,j}}{\sum_{k \in subchannels} w_k a_{k,j}}$$
(2)

where  $\rho$  is the water density, k is the subchannel number, a is the area of a subchannel centered subchannel, and w is the

weighting factor. For a corner subchannel w = 1.0, for a side subchannel w = 0.5, and for a normal subchannel w = 0.25.

## **III. BWR COMPUTATIONAL CHAIN**

The ILVM is tested using the BWR computational chain APOLLO2-A/HERMES/ARTEMIS<sup>TM</sup>. APOLLO2-A is the spectral code used to generate the cross sections. To account for historical effects, depletions paths are modelled at different void fractions. These depletion paths are then perturbed on moderator density, fuel temperature, and the presence or not of the control blade. A multiparameter database is then created, where the parameters are:

- burnup, usually from 0 to 80 GWD/T
- coolant void history, for instance at 0, 40, 80%
- instantaneous coolant void, for instance at 0, 40, 70, 90%
- fuel temperature, for instance at 560, 900, 1500 K
- moderator temperature (e.g. 293, 365, 460 K) only needed for cold conditions.
- presence or not of the control blade, instrumentation / guide tube (TIP/LPRM), etc.

The neutronic data is then be interpolated based upon the local quantities obtained in the core simulator. In the APOLLO2-A/HERMES/ARTEMIS<sup>TM</sup> chain, the user can select and easily change the mathematical order and the functional basis used for the interpolation process, as long as the APOLLO2-A calculation incorporates the required sources and/or branches. Adding and removing a parameter is also done simply via the input.

HERMES is the interface code that processes the multiparameter database and computes the coefficients needed for the interpolation step. For the microscopic cross sections, the functional expansion is given as a N-parameter tuple  $(p_1, p_2, \dots p_N)$ :

$$\sigma_{r,g}(p_1, p_2, \cdots, p_N) = \sum_{j=1}^{N_{cross-terms}} f_j(\sigma_{r,g,j}(p_1, p_2, \cdots, p_N))$$
(3)

where:

- *f<sub>j</sub>* is a functional representation based on a tensor product of 1D B-splines and/or 1D polynomials.
- $\sigma_{r,g,i}$  are the fitting coefficients, computed by HERMES.
- $N_{cross-terms}$  is the total number of cross terms used in the mathematical representation. Cross terms are introduced optionally to reduce the dimension of the multiparameter database. If  $N_{cross-terms} = 1$ , then a full blown cross section representation is selected.

The interpolation of the cross sections is done on the fly. The local conditions are obtained by iterating on the neutron flux, thermal hydraulic, fuel rod, and cross section solvers in the ARTEMIS<sup>TM</sup> core simulator.



Fig. 2: Water Density Distribution

#### **IV. IMPLICIT LOCAL VOID MODEL**

The ILVM first requires setting up a F-COBRA-TF<sup>TM</sup> input for the whole fuel assembly using representative thermal hydraulic (TH) boundary conditions. The TH boundary conditions that are needed include: representative axial power profile and mass flow rate for the assembly design, and the range of linear heat rates for each lattice type. The output of the model includes the water density distributions that are used during the cross section generation. A sample water density distribution is given in Fig. 2. These water density distributions are created as a function of the lattice type, average void fraction, and burnup.

To take into account the variations of the water density distribution, the water density distribution is generated as a function of the average void fraction. The water density distribution is calculated every 10% void up to a maximum lattice-dependent limit. The different void fractions are obtained by changing the linear heat rate in F-COBRA-TF<sup>TM</sup>. The maximum void fraction is obtained by giving F-COBRA-TF<sup>TM</sup> the largest possible linear heat rate for that assembly type.

To take into account the burnup dependence, the water density distributions are calculated at beginning of life (BOL) and approximated at gadolinium burnout. The water density distributions at BOL are directly available after running the coupled APOLLO2-A/F-COBRA-TF<sup>TM</sup> calculation. For obtaining an estimate of the water distribution at gadolinium burnout, it is possible to run the APOLLO2-A/F-COBRA-TF<sup>TM</sup> coupled calculation with depletion enabled. However this procedure is computationally very expensive, and thus an approximation is required. The water density distributions at gadolinium burnout are approximated by running the coupled calculation with the gadolinium removed. To measure the validity of this assumption, the sum of square difference of the water density distribution summed over all the subchannels:

$$\sum_{i} (\rho_{i,nogad} - \rho_{i,ref})^2 \tag{4}$$

is used to compare the results of the gadolinium removed cal-



culation (no gad) with reference results. The reference results come from running the coupled calculation from BOL to end of life (EOL). The sum of square difference is shown in Fig. 3 and each line corresponds to a different axial node. It can be observed that removing the gadolinium is a good approximation for gadolinium burnout due to the small difference in the water density distributions. It is important to note that the burnup value at which the minimum difference occurs is different for each axial node and is different for each lattice type.

The location of gadolinium burnout is dependent on both the lattice type and the average void fraction. Again, running the coupled calculation with different linear heat rates to find the location of gadolinium burnout for each lattice type and average void fraction would be very computationally expensive. Instead a standalone APOLLO2-A calculation is run for each lattice type and average void fraction and the location of the maximum k-infinity is calculated. The plot of k-infinity as a function of burnup from the standalone APOLLO2-A calculation is shown in Fig. 4. It can be observed that the location of maximum k-infinity is similar to the location of minimum sum of square error in Fig. 3, although there can be a slight deviation. For example, in axial node 26 there is about 60% void and the maximum k-infinity occurs at 14.0 GWd/T, where as the minimum sum of square difference between the water density distributions occurs at 13.5 GWd/T. The difference in gadolinium burnout prediction between the two methods is 1.0 GWd/T or less for most cases.

The water density distribution is approximated with a linear dependence in burnup from BOL to gadolinium burnout and it is approximated to be constant after gadolinium burnout. A plot of the reference water density and the approximated water density for 10 representative subchannels is shown in Fig. 5. The representative subchannels are a result of grouping similar subchannels together. In the ILVM, the full 60 subchannels are used to generate the cross sections. From Fig. 5, it is observed that the water density distribution is approximated well early in the cycle but deviates later in the cycle. It is most important to capture the water density



Fig. 4: Standalone APOLLO2-A Calculation of k-infinity



Fig. 5: Actual vs Approximated Water Density

distribution accurately when the gadolinium is present. In [4], it was observed that after gadolinium burnout, the effects of using the water density distribution were small. While it would be more accurate to try to accurately model the water density distribution later in the cycle as well, it would be computationally expensive to model more burnup points.

The end result of the ILVM is functionalizing the water density distribution as a function of lattice type, average void fraction, and burnup. A sample result of the IVLM for a single lattice type, at 40% void is represented at 3 burnup points is shown in Fig. 6. For this lattice type, gadolinium burnout occurs at 14 GWd/T, so the density distribution shown at 14 GWd/T is also used for all burnup points greater than 14 GWd/T.

#### V. GEOMETRY

The geometry that is used to create the ILVM is the AREVA ATRIUM<sup>TM</sup> 11 assembly and is shown in Fig. 7. The assembly is modeled using 50 axial nodes in both



Fig. 6: Water Density Distribution as a function of Burnup



Fig. 7: ATRIUM<sup>TM</sup> 11 Geometry

APOLLO2-A and F-COBRA-TF<sup>TM</sup>. The axial variations in the ATRIUM<sup>TM</sup> 11 assembly are due to the presence of the partial length fuel rods, fuel enrichment variations, gadolinium enrichment variations and the location of spacer grids. The TH boundary conditions for this geometry are obtained by running the core simulator using cross sections that have uniform void distributions (UVM) and extracting the BOC axial power profile, BOC flow rates, and the range of linear heat rates for the assembly of interest. To analyze the effects of the ILVM, two different equilibrium cycles are modeled, both based on the ATRIUM<sup>TM</sup> 11 assembly, but for two different reactors and cycle lengths. As a result the neutronic design (U235 content, Gadolinium poisoning, pin placement, etc.) differ strongly between the two cases.

The first equilibrium cycle that is analyzed is Core A. This design has 12 month cycles with only one type of feed assembly. Each cycle, 1/6 of the assemblies in the core are replaced. The feed assembly has little variation in uranium enrichment radially or axially. There is also little variation in the gadolinium content. A plot of the uranium enrichment and gadolinium content per pin for a single lattice is shown in Fig. 8.

The second equilibrium cycle that is analyzed is Core B. This design has 24 month cycles with three types of feed assemblies. Each cycle, 1/3 of the assemblies in the core are replaced. These feed assemblies have more variation in uranium enrichment radially and axially than the feed assembly in Core A. Each of the feed assemblies has axially and radially varying amounts of gadolinium. The plot of the uranium enrichment and gadolinium per pin and for a single lattice is shown in Fig. 9.



Fig. 8: Lattice in Core A



Fig. 9: Lattice in Core B

## VI. RESULTS

To measure the effect of the ILVM, two different cross section sets are generated to be run in the core simulator:

- UVM
- ILVM

In the UVM, the water density distribution is set to uniform and the cross sections are created at 0% Void, 40% Void, and 80% Void. The UVM case is the traditional method for generating cross sections and is considered the reference calculation for all the following comparisons. The UVM case was also tested using 0%, 10%, 20%, ..., max(Void Fraction). These results are similar to the traditional UVM case, which also indicates that the historical effects due to different void levels are accurately captured using only these three points, so they are not presented in this paper. In the ILVM, the cross sections are generated with the water density distributions created using the algorithm given in Section IV.

The results of interest are the eigenvalue trend, peaking factors and thermal limit margins. The ARTEMIS<sup>TM</sup> results are based upon an equilibrium cycle, where the same loading pattern, control rod position, etc. is repeated over and over until convergence to the same quantities from cycle to cycle. The eigenvalue is shown at the lattice level (k-infinity) from APOLLO2-A and at the core level (k-eff) from ARTEMIS<sup>TM</sup>. All of the other results are shown at the core level from ARTEMIS<sup>TM</sup>. The CMRPF is the Core Maximum Radial Peaking Factor and the CMPF is the Core Maximum Nodal Peaking Factor. The CMFLHGR is the Core Maximum Fraction Limiting Heat Generation Rate which is a safety margin for the maximum LHGR. The maximum LHGR is thermal mechanical safety limit that is limited by the melting point of the fuel. Lower values of CMFLHGR represent higher safety margins. The CLCPR is the Core Limiting Critical Power Ratio which is a safety margin for the minimum CPR. The minimum CPR (critical power ratio) is a safety limit for dryout (for BWR). Similar to CMFLHGR, lower values of CLCPR represent higher safety margins.

#### 1. Core A

The differences in k-infinity in the spectral code for a particular lattice are given in Fig. 10. The differences in k-infinity at the lattice level are within 150 pcm. There is no systematic trend, as some of the calculations show an increase in k-infinity while other show a decrease. The plot of k-eff from the core simulator is given in Fig. 11. The ILVM causes a decrease in k-eff at BOC and an increase in k-eff at the end of cycle (EOC). The maximum difference is 25 pcm. The plot of the CMRPF is given in Fig. 12. The ILVM causes the radial peaking factor increases over the entire cycle but the magnitude of the difference is small. The plot of the CMPF is given in Fig. 13. The ILVM causes the nodal peaking factor decreases over most of the cycle but similar to the CMRPF the difference is small. The plot of the CM-FLHGR is given in Fig. 14. The ILVM causes a decrease in the CMFLHGR over the entire cycle. Similar to the peaking factors, the difference is small. The CMFLHGR tends to



Fig. 10: Lattice from Core A: APOLLO2-A k-infinity Difference



change as a function of the nodal peaking factor which is observed by comparing the difference in the CMPF and the difference in the CMFLHGR. The plot of the CLCPR is given in Fig. 15. The ILVM causes an increase in the CLCPR over the entire cycle but once again, the difference is small. The CLCPR tends to change as a function of the radial peaking factor which is observed by comparing the difference in the CMRPF and the difference in the CLCPR.

## 2. Core B

The differences in k-infinity in the spectral code for a particular lattice are given in Fig. 16. The differences for this equilibrium cycle go up to 400 pcm. The differences for all of the average void percent follow a similar trend. The ILVM causes a decrease in k-infinity at BOC and then an increase at gadolinium burnout. This is caused by the increased rate of gadolinium depletion. The plot of k-eff from the core sim-



Fig. 12: Core A: Core Maximum Radial Peaking Factor



Fig. 14: Core A: Core Maximum Fraction Limiting Heat Generation Rate



Fig. 13: Core A: Core Maximum Nodal Peaking Factor



Fig. 15: Core A: Core Limiting Critical Power Ratio



Fig. 16: Lattice from Core B: APOLLO2-A k-infinity Difference

ulator is given in Fig. 17. The ILVM causes an increase in k-eff over most of the cycle, except at EOC. The maximum difference is about 200 pcm. The plot of the CMRPF is given in Fig. 18. The ILVM causes the radial peaking factor to increase and decrease over the cycle. The plot of the CMPF is given in Fig. 19. The ILVM causes the nodal peaking factor to also increase and decreases over the cycle but the nodal peaking factor is decreasing more than it is increasing. The plot of the CMFLHGR is given in Fig. 20. Similar to Core A, the difference in CMFLHGR follows the difference in the nodal peaking factor. The ILVM causes the CMFLHGR to decrease over most of the cycle, but it does cause increases in some areas. The plot of the CLCPR is given in Fig. 21. The ILVM causes the CLCPR to increase and decrease over the cycle depending on the burnup. Similar to Core A, the difference in CLCPR follows the difference in the radial peaking factor. It is important to note that the differences in magnitude for this equilibrium cycle are larger compared to the previous one.

#### 3. Most Limiting Results

An important result is the effect of the ILVM on the most limiting results from the cycle. The most limiting results are shown for the peaking factors and thermal limits described before (CMRPF, CMPF, CMFLHGR, CLCPR), as well as for the thermal limit CMAPRT which is the Maximum Fraction of Limiting Average Planar Linear Heat Generation Rate. CMAPRT is thermal limit that protects against fuel damage during a loss of coolant accident. The most limiting results for Core A are given in Table I. For Core A, there are decreases in the CMPF and the CMFLHGR and slight increases to the CMRPF and the CLCPR. The effect of the ILVM on Core A is small. The most limiting results for Core B are given in Table II. For Core B, there are decreases in the



Fig. 18: Core B: Core Maximum Radial Peaking Factor



Fig. 19: Core B: Core Maximum Nodal Peaking Factor

|      |       | •     |         |       |  |  |
|------|-------|-------|---------|-------|--|--|
|      | CMPF  | CMRPF | CMFLHGR | CLCPR |  |  |
| UVM  | 3.018 | 1.563 | 0.924   | 0.961 |  |  |
| ILVM | 2.968 | 1.566 | 0.920   | 0.962 |  |  |

TABLE I: Core A: Most Limiting Results

| TА | BLE | II: | Core | B: | Most | Limiting | Resu | lts |
|----|-----|-----|------|----|------|----------|------|-----|
|----|-----|-----|------|----|------|----------|------|-----|

|      | CMPF  | CMRPF | CMFLHGR | CLCPR | CMAPRT |
|------|-------|-------|---------|-------|--------|
| UVM  | 2.884 | 1.481 | 0.857   | 0.943 | 0.852  |
| ILVM | 2.892 | 1.474 | 0.858   | 0.935 | 0.844  |



Fig. 20: Core B: Core Maximum Fraction Limiting Heat Generation Rate



Fig. 21: Core B: Core Limiting Critical Power Ratio

CMRPF, the CLCPR, and the CMAPRT as well as slight increases to the CMPF and the CMFLHGR. This core is a lot more effected by the ILVM. Most of the thermal limits and peaking factors are decreased due to the ILVM.

## VII. CONCLUSION

In this paper, the assumption of using a uniform void distribution at the spectral code level is analyzed. The ILVM is created to functionalize the water density distribution so that the water density distribution could be incorporated in the spectral code calculations. The water density distribution is then inherent in the cross sections. The ILVM is applied to the APOLLO2-A/HERMES/ARTEMIS<sup>TM</sup> BWR computational chain and the results for two different equilibrium cycles are calculated. The impact of the ILVM on key parameters needed for core design are found to be very dependent on the assembly design. Assembly designs that are more effected by the ILVM have high gadolinium concentrations and have radial or axial variations due to the uranium enrichment or gadolinium concentration. Core A, which has low gadolinium and small variations, was mostly unaffected due to the ILVM. Core B, which has high gadolinium and many radial and axial variations, had most of the peaking factors and thermal limits decrease due to the ILVM.

Future work will include a generalization of the methodology to include local feedback of the coolant density (i.e. from the core simulator's thermal hydraulic solver). Also validation of the ILVM, where ARTEMIS<sup>TM</sup> results obtained with the ILVM will be compared toward actual detector measurements and flux maps.

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