PWR CONTROL ROD EJECTION ANALYSIS WITH THE MOC CODE DECART

Mathieu Hursin* and Thomas Downar
University of California
Berkeley, USA
mhursin@nuc.berkeley.edu, downar@nuc.berkeley.edu

ABSTRACT

During the past several years, a comprehensive high fidelity reactor core modeling capability has been developed called the Numerical Nuclear Reactor (NNR) for detailed analysis of Light Water Reactors. The project was initially sponsored as part of a US-ROK collaborative DOE I-NERI project and the past two years has been sponsored by EPRI to address specific fuel performance issues as part of EPRI BWR Fuel Reliability Program. The NNR achieves high fidelity by integrating whole-core neutron transport solution and ultra-fine-mesh computational fluid dynamics/heat transfer solution. During the past several months there has been interest in taking advantage of the NNR to improve the fidelity for LWR transient analysis. The work described in this paper is a preliminary demonstration of the ability of the whole core neutron transport code, DeCART, to provide an accurate core average power during a control rod ejection. The current state of the art in analysis of this event is to rely upon the assembly averaged power from a whole core nodal neutronics to provide a conservative analysis.

Key Words: DeCART, Control Rod Ejection, Numerical Reactor

1. INTRODUCTION

The Numerical Nuclear Reactor (NNR) is a software system for high-fidelity multi-physics LWR core simulations [1,2]. During the past few years in support of the EPRI BWR Crud Deposition Analysis program, the principal focus of NNR applications has been to steady-state LWR problems, with the most recent emphasis on depletion analysis of BWR fuel assemblies.[3,4] As part of that effort, the neutronic analysis module, DeCART, was extended to include consideration of geometric heterogeneities of BWR fuel assemblies and a depletion analysis capabilities was implemented and tested. A new CFD-based boiling heat transfer model was also incorporated in thermal-hydraulic analysis module, STAR-CD, capable of simulating high void fraction boiling regimes encountered in BWRs and the predicting the critical heat flux in PWRs.

During the past several months, there has been interest in taking advantage of the NNR to improve the fidelity for LWR transient analysis. In response, the ability of the whole core neutron transport code, DeCART, to provide a accurate average power during a control rod ejection has recently been investigated. The state-of-the art in analysis of this event is to rely upon the assembly averaged power from a whole core nodal neutronics simulator to provide a conservative analysis. The benefits of the NNR is a full physics simulation of the coupled fields

* Corresponding author
during the rod ejection transient, which could then provide the detailed intrapin power temperature distributions during the transient. The primary goal of the work summarized in this paper is to demonstrate the capability of the NNR neutronic module DeCART to accurately predict the core average power during a rod ejection accident, the reference is PARCS, the core simulator used by the US-NRC to study this kind of accident.

2. MODEL DESCRIPTION

The geometry used to study a rod ejection accident is a “mini-core” consisting of a 3x3 array of typical PWR fuel assemblies. A control rod is partially inserted into the central assembly which is fresh fuel and is adjacent to once and twice burned fuel as shown in Figure 1.

A reflective boundary condition is assumed at the exterior of the 3x3 array. The "mini core" is assumed to be at a hot-zero power state: the fuel and coolant temperatures are at their nominal

Figure 1. 3D Geometry of the "mini core"

A reflective boundary condition is assumed at the exterior of the 3x3 array. The "mini core" is assumed to be at a hot-zero power state: the fuel and coolant temperatures are at their nominal
values but the power generated is set to zero. It is typical of a reactor in standby position (control rods in) and its most sensitive state with respect to reactivity initiated accidents (RIA) (rod ejection).

### 3. METHODOLOGY

The current methodology to study a rod ejection accident, is to use U.S. NRC core simulation code PARCS. PARCS uses the standard nodal methods with assembly homogenized cross sections. For the purpose of having a consistent comparison with the DeCART solution, the nuclear data used by PARCS (macroscopic cross sections, kinetic parameters) are generated with DeCART.

#### 2.1. Generation of the nuclear data for PARCS

In order to perform transient calculations, PARCS needs two energy group macroscopic cross sections, assembly discontinuity factors, six group precursor's delayed neutron fraction, and two group neutron velocities. DeCART was used as a lattice code (e.g. CASMO) to generate the cross section data for PARCS. For each materially different assembly, fresh unrodded, fresh rodded, once burned and twice burned, a 2D single assembly model is build with DeCART and run to extract the nuclear data needed by PARCS.

#### 2.1.1. Verification of the nuclear data

In order to check the consistency of the nuclear data generated with DeCART for PARCS, a set of steady state calculations are performed with PARCS and their results is compared to DeCART. The first set of calculations is done for each of the materially different assembly. The infinite multiplication coefficient are recorded. The results are shown in the Table 1.

<table>
<thead>
<tr>
<th>Assembly</th>
<th>$k_{\text{inf}}$ (DeCART)</th>
<th>$k_{\text{inf}}$ (PARCS)</th>
<th>$\Delta k$ in pcm</th>
</tr>
</thead>
<tbody>
<tr>
<td>fresh unrodded</td>
<td>1.11891</td>
<td>1.11891</td>
<td>0</td>
</tr>
<tr>
<td>fresh rodded</td>
<td>0.75454</td>
<td>0.75454</td>
<td>0</td>
</tr>
<tr>
<td>once unrodded</td>
<td>1.07594</td>
<td>1.07594</td>
<td>0</td>
</tr>
<tr>
<td>twice unrodded</td>
<td>0.95760</td>
<td>0.95761</td>
<td>-1</td>
</tr>
</tbody>
</table>

The good agreement of PARCS with DeCART shows that the two group macroscopic cross sections generated with DeCART are correct. The next step is to check the quality of the assembly discontinuity factors by comparing the effective multiplication coefficient of PARCS and DeCART for the 3D "mini core" in steady state. Two cases are considered, one with the control rod inserted (control rod in) and the other one with the control rod withdrawn (control...
Comparing the k-effective of both calculations give access to the reactivity inserted upon withdrawal of the control rod. The results are shown in the Table 2.

The agreement between PARCS and DeCART is not as good as in the previous single assembly calculations. This is expected since the nuclear data, and particularly the assembly discontinuity factors are generated at the single assembly level assuming reflective boundary condition. In the real case, the assembly is next to a materially different assembly and the reflective boundary condition assumption made during the cross section generation stage doesn't hold anymore. This assumption is typical of the current generation of methods. Its effect is not important in this case. The other important finding of this set of calculations is the control rod worth which is similar in DeCART and PARCS and slightly over a dollar of reactivity, around 1.15 dollar. The transient resulting of a rod ejection is then a super prompt critical transient.

Table 2. Steady State Comparison of DeCART and PARCS for 3D "mini core"

<table>
<thead>
<tr>
<th>Case</th>
<th>k effective</th>
<th>Δk in pcm</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>DeCART</td>
<td>PARCS</td>
</tr>
<tr>
<td>Control Rod In</td>
<td>1.15936</td>
<td>1.15880</td>
</tr>
<tr>
<td>Control Rod Out</td>
<td>1.16840</td>
<td>1.16813</td>
</tr>
<tr>
<td>Rod Worth in pcm</td>
<td>667</td>
<td>689</td>
</tr>
</tbody>
</table>

Since the nuclear data generated by DeCART have been verified to be accurate, the next step is to look at the transient calculation.

4. TRANSIENT CALCULATION RESULTS

The control rod is ejected from the central assembly and the transient calculation is performed with DeCART and PARCS. Two sets of calculations are done with each code. In the first one (blue curves), the thermo hydraulic feedbacks are turned off. The purpose of this calculation is to evaluate the kinetic parameters generated with DeCART for PARCS. In the second set of calculations (red curves), the thermo hydraulic feedbacks are turned on. Both DeCART and PARCS used a similar simple internal thermo hydraulic solver to simulate the fuel and moderator temperature response during the course of the transient. The only difference between both solvers is that DeCART provides temperature feedback at the fuel rod level whereas PARCS does the same but at the assembly level.

The average power of the mini-core obtained by each code for both calculation is shown in Figure 2. The agreement between PARCS and DeCART during the set of transient calculation without feedbacks is very good showing that the procedure to generate kinetics parameters with DeCART is correct. The agreement between PARCS and DeCART for the set of transient calculations with thermo hydraulic feedback is good at the beginning of the transient and tends to degrade toward the end of the transient due to difference in the way the thermo hydraulic feedback are obtained. Nonetheless the goal of this study, the assessment of the transient capability of the DeCART code has been reached: the agreement of the DeCART results with the
US-NRC code PARCS is good. The DeCART approach to perform transient calculations is validated.

![Graph showing comparison of core power during a rod ejection accident](image)

**Figure 2. Evolution of the core power during a rod ejection accident**

### 3. CONCLUSIONS

A control rod ejection transient has been analyzed with both DeCART and with the U.S. NRC core simulation code PARCS. The cross sections are generated with DeCART in order to provide a consistent comparison of the calculation here with the current state of the art in analysis of the control rod ejection accident. The agreement in terms of core average power between DeCART and PARCS is very good and the small differences can be attributed to the different thermo hydraulic solvers and to the assumptions used to generate nuclear data for PARCS. The benefit of the DeCART approach is to have access to very detailed data like intrapin power density throughout the transient. This kind of data is not available with the current methodology and could be used to reduce some of the conservatism in the fuel design limit for reactivity initiated accident.

### REFERENCES

