NE 571: Project 6

Small Modular Reactors

NuScale and mPower

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# Abstract

***Inset short blurb about results here. Revisit later.***

# Introduction

Diffusion theory is a crucial approximation to simulating reactors. It is an approximation to the neutron transport theory, that makes several key assumptions and completely breaks down in some circumstances. In most scenarios, it is accurate enough to obtain a reasonable estimation of the neutron flux. While an analytical solution cannot be found for all but the simplest cases, discretizing the differential equation will allow you to obtain a valid solution, so long as the step sizes are reasonably small. Going further, you can normalize the flux to actual values if you know the thermal power generated by the reactor.

Current commercial reactors are large-scale baseload type plants, requiring billions of dollars in initial investments, meaning that only large, well-funded utilities can even consider taking the risk of constructing them. Newer, smaller reactors are being developed, with initial investments coming in at several million dollars. This means that even medium-sized towns and utilities could consider purchasing a small-modular reactor (SMR). They have long life-spans, operating cycles, and can even be used for load-following. Some SMRs, such as the B&W mPower and NuScale Power’s Nuscale SMR, even use existing pressurized water reactor fuel rods, just cut to shorter length. While no full-scale plants have been constructed yet, their finalized characteristics have been published, allowing interested parties to simulate the reactors.

Finally, SCALE is an extremely powerful software used for detailed simulations of reactor cores. Using properties described on “input cards,” we can calculate various properties required for other software, such as the absorption coefficient in the simulation used for this project. We used SCALE to determine the properties of our fuel at various points throughout its life, simulating a reactor with fresh, once-burned, and twice-burned fuel rods. We also simulated the fuel with and without its control rods. As such, no individual piece of software can fully simulate nuclear reactors, even small ones, and many independently developed codes must be used in conjunction. Reinforcing this point was the overarching goal of this project.

# Methods

We will not go into many details regarding discretizing the diffusion equation, since it has been repeated ad nauseam, but our derivations will be attached in Appendix A. We built upon a code written earlier in the course in Python, allowing it to select different properties based on the region of the core; for example, a twice-burned fuel rod will have a completely different fission cross-section than a fresh one. A further goal of this project is, instead of using pre-generated cross sections, to generate our own using SCALE based on the properties of our two reactors, an mPower and a NuScale SMRs. A table of pertinent values and our obtained macroscopic cross-sections we obtained will be provided in Appendix B. Finally, once we have obtained our fractional flux from each of a series of simulations, we will normalize the flux to the power of the reactor, using Equation 1. P is the thermal power, υ is the average number of neutrons produced per fission, Σf is the macroscopic fission cross-section, and V is the reactor volume.

[Equation 1]

Turning our attention to the values we obtained in SCALE, *insert short section of how we generated those values, and for what conditions here.*

# Results

First, we have the results of our B&W mPower reactor simulations, using the macroscopic cross-sections provided in Appendix B. They are shown in Figure y, again, with details of that particular simulation in the caption. Note that the flux has been normalized to the power of a single mPower reactor, which produces 530 MWth.

|  |  |  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- | --- | --- |
| mPower Cross Sections | | | | | | | |
| Burnup (GWd/MT) | Group | Total | Transport | Removal | Eff Abs | Nu-Fission | Scatter Down |
| 0 | 1 | 5.32E-01 | 2.26E-01 | 2.53E-02 | 9.94E-03 | 8.73E-03 | 1.55E-02 |
| 2 | 1.38E+00 | 8.89E-01 | 1.11E-01 | 1.10E-01 | 1.98E-01 | 1.71E-03 |
| 15 | 1 | 5.39E-01 | 2.28E-01 | 2.63E-02 | 1.07E-02 | 7.46E-03 | 1.56E-02 |
| 2 | 1.40E+00 | 9.09E-01 | 1.21E-01 | 1.19E-02 | 1.86E-01 | 1.82E-03 |
| 30 | 1 | 5.44E-01 | 2.29E-01 | 2.69E-02 | 1.12E-02 | 6.41E-03 | 1.56E-02 |
| 2 | 1.40E+00 | 9.19E-01 | 1.17E-01 | 1.15E-01 | 1.69E-01 | 1.75E-03 |
| 45 | 1 | 5.48E-01 | 2.31E-01 | 2.74E-02 | 1.17E-02 | 5.50E-03 | 1.57E-02 |
| 2 | 1.40E+00 | 9.23E-01 | 1.09E-01 | 1.08E-01 | 1.46E-01 | 1.63E-03 |
|  |  |  |  |  |  |  |  |

Finally, we have the results for the NuScale Power NuScale reactor simulations, using the properties obtained from SCALE, also in Appendix B. As before, the flux is shown in Figure 1, and has been normalized to the 160 MWth power of the reactor.

|  |  |  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- | --- | --- |
| NuScale Cross Sections | | | | | | | |
| Burnup (GWd/MT) | Group | Total | Transport | Removal | Eff Abs | Nu-Fission | Scatter Down |
| 0 | 1 | 5.35E-01 | 2.27E-01 | 2.61E-02 | 1.05E-02 | 8.80E-03 | 1.56E-02 |
| 2 | 1.37E+00 | 8.81E-01 | 1.22E-01 | 1.20E-01 | 1.93E-01 | 2.22E-03 |
| 15 | 1 | 5.43E-01 | 2.29E-01 | 2.65E-02 | 9.95E-03 | 5.50E-03 | 1.66E-02 |
| 2 | 1.38E+00 | 9.08E-01 | 9.27E-02 | 9.13E-02 | 1.32E-01 | 1.43E-03 |
| 30 | 1 | 5.48E-01 | 2.31E-01 | 2.72E-02 | 1.07E-02 | 4.73E-03 | 1.65E-02 |
| 2 | 1.39E+00 | 9.17E-01 | 8.97E-02 | 8.82E-02 | 1.17E-01 | 1.37E-03 |
| 45 | 1 | 5.52E+00 | 2.32E-01 | 2.78E-02 | 1.13E-02 | 4.19E-03 | 1.65E-02 |
| 2 | 1.38E+00 | 9.20E-01 | 8.49E-02 | 8.36E-02 | 1.01E-01 | 1.28E-03 |

Initial Conditions

# System properties.

# k = 1.0873

AsigTr\_ = [0.0269]

Asiga\_ = [0.1152]

Avsig\_ = [0.00641,0.169]

AsigS\_ = [0.015561] # 0.00155

AD\_ = [1.2627,0.3543]

# k = 0.977

BsigTr\_ = [0.0274]

Bsiga\_ = [0.108]

Bvsig\_ = [0.0055,0.146]

BsigS\_ = [0.0157]

BD\_ = [1.2427,0.3543]

# k = 1.21

CsigTr\_ = [0.0263]

Csiga\_ = [0.119]

Cvsig\_ = [0.00746,0.186]

CsigS\_ = [0.0156]

CD\_ = [1.2627,0.3543]

# k = 1.44

DsigTr\_ = [0.0254]

Dsiga\_ = [0.1097]

Dvsig\_ = [0.00873,0.198]

DsigS\_ = [0.0155]

DD\_ = [1.2627,0.3543]

WsigTr\_ = [0.0494]

Wsiga\_ = [0.0197]

Wvsig\_ = [0.0,0.0]

WsigS\_ = [0.0494]

WD\_ = [1.13,0.16]

Z = 2000cm

D of fuel element = 20cm

D of water = 15cm

|  |  |
| --- | --- |
| ../../../../Figure_1-1.png | ../../../../Figure_1.png |
| Figure 1: 5 mesh per element -- 10 mesh per In z --LoadingPattern = [FuelA,FuelC,FuelB,FuelD,Water]  K = 1.07494266252 | |

|  |  |
| --- | --- |
| ../../../../Figure_1-1.png | ../../../../Figure_1-2.png |
| Figure 2: 5 mesh per element -- 10 mesh per In z -- LoadingPattern = [FuelB,FuelA,FuelB,FuelC,Water] K = 1.00823848115 | |

|  |  |
| --- | --- |
| ../../../../Figure_1-1.png | ../../../../Figure_1.png |
| Figure 3: 5 mesh per element -- 10 mesh per In z -- LoadingPattern = [FuelB,FuelA,FuelB,FuelA,Water]  K = 0.996174846504 | |

|  |  |
| --- | --- |
| ../../../../Figure_1-2.png | ../../../../Figure_1.png |
| Figure 4: 5 mesh per element -- 10 mesh per In z -- LoadingPattern = [FuelD,FuelD,FuelD,FuelD,Water] All fresh -- K = 1.20790060525 | |

|  |  |
| --- | --- |
| ../../../../Figure_1-1.png | ../../../../Figure_1.png |
| Figure 5: 5 mesh per element -- 10 mesh per In z -- LoadingPattern = [FuelB,FuelB,FuelB,FuelB,Water] Most burnt -- K = 0.967264281277 | |

|  |  |
| --- | --- |
| ../../../../Figure_1-2.png | ../../../../Figure_1.png |
| Figure 6: Z = 2000cm -- D of fuel element = 18cm -- D of water = 15cm -- 5 mesh per element  10 mesh per In z -- LoadingPattern = [FuelB,FuelA,FuelB,FuelC,Water] -- K = 1.00253673468 | |

# Conclusion

Simulations are a valuable tool in nuclear reactor development. Due to the large initial costs of testing even small-scale reactors, as well as the time necessary to construct them, simulations are used as much as possible. However, simulations are not simple; as a matter of fact, they are extremely complex. As evidenced by our relatively simple project, which focuses on only the core, and only at steady-state. There are further variations that build in a time-dependence, the remainder of the plant, and even accident scenarios. The basis for all of these simulations are the material properties of the reactor, which we obtained through the use of SCALE. Once these were obtained, we built upon our two-group diffusion code to include various regions, as opposed to a homogeneous core. Finally, we evaluated our reactors under several different circumstances, to see how our changes affected the k-value of the configuration and the flux. Overall, this project was effective in showing the overall complexity of reactor simulations, as well as reinforcing how key these simulations are in the industry.

# References

https://www.nrc.gov/docs/ML1700/ML17007A001.pdf

https://www.iaea.org/NuclearPower/Downloadable/SMR/files/IAEA\_SMR\_Booklet\_2014.pdf

# Appendix A: Derivations

# Appendix B: Reactor Properties and Macroscopic Cross-Sections

**Reactor Properties**

|  |  |
| --- | --- |
| B&W mPower | |
| Length | 2.4 m |
| # of Assemblies | 69 |
| Rods per Assembly | 17x17 |
| Max Enrcihment | 5% |
| Burnup | >40 GWd/MT |
| Cycle Length | 48 Months |
| Thermal Capacity | 560 MW |
| Coolant Inlet | 295 °C |
| Coolant Outlet | 319 °C |

|  |  |
| --- | --- |
| NuScale NuScale | |
| Length | 2 m |
| # of Assemblies | 37 |
| Rods per Assembly | 17x17 |
| Max Enrcihment | 4.95% |
| Burnup | TBD |
| Cycle Length | 24 Months |
| Thermal Capacity | 160 MW |
| Coolant Inlet | 497 °F |
| Coolant Outlet | 543 °F |