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DEPARTMENT OF MECHANICAL AND NUCLEAR ENGINEERING

EFFECT OF CORE SIZE ON ROD WORTH DURING A CONTROL ROD DROP ACCIDENT

KELLY MCCLURE
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Reviewed and approved* by the following:

Kostadin Ivanov
Distinguished Professor of Nuclear Engineering
Thesis Supervisor

Seungjin Kim
Associate Professor of Mechanical and Nuclear Engineering
Honors Adviser

* Signatures are on file in the Schreyer Honors College.

ABSTRACT

A control rod drop accident (CRDA) is a reactivity-initiated transient event analyzed by engineers for boiling water reactors (BWRs). During a CRDA, a control rod drops from the core at a high speed, causing a reactivity spike and risking damage to the fuel and cladding. Engineers must prove that a reactor's core design will remain safe in the event of such an accident. Thus, to gain a full understanding of the effect and severity of a CRDA, it is beneficial to explore the effect of reactor core size on the transient results.

Beginning with the largest available BWR reactor core, five common BWR core sizes were created and a CRDA was simulated at each size. The initial core re-sizing methodology was developed at GE Hitachi Nuclear Energy, then the process was repeated and the CRDA simulation was performed at Penn State. The goal of the core reduction process was to create core that could be considered "similar" for the CRDA analysis, as defined by adherence to a ±0.005 eigenvalue limit between the calculated and expected leakage-adjusted values.

A CRDA was simulated in the five core sizes to compare the severity of the transient event – as defined by the rod worth of a particular rod of interest. The accident simulation indicated that the rod worth, and thus fuel enthalpy increase, decreases as the core size increases. The same trend was found when comparing power peaking factor (F_Q) in the various core sizes. These results suggest that smaller cores yield a more limiting control rod drop accident.

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Chapter 1

INTRODUCTION

When analyzing a Boiling Water Reactor (BWR), one of the transient events engineers must anticipate is a control rod drop accident (CRDA). Within a BWR, control rods are inserted to absorb excess neutrons and keep the nuclear reaction operating at criticality (i.e. 1:1 ratio of neutrons being absorbed and emitted by the uranium oxide fuel). These rods are inserted from beneath the core using a control rod drive mechanism. This mechanism uses pressurized nitrogen gas to insert and withdraw the control rods from the reactor core, and can quickly insert all control rods to shut down, or SCRAM, the reactor in the event of an emergency [1].

During a control rod drop accident, the control blade becomes decoupled from the drive mechanism, sticks in the core, and later falls at a rapid pace to the position of the drive. This drop causes a localized spike in reactivity in the area surrounding the dropped rod equal to the worth of rod. If the reactivity spikes enough, the core risks overheating, causing cracks or melting within the fuel and cladding [2].

To prevent such overpower transients, engineers must simulate the CRDA and determine the control rod location and cycle exposure at which the most limiting (most severe) CRDA would occur. The core must be designed so that the worst case CRDA satisfies the acceptance criteria stipulated by the NRC. The allowed prompt fuel enthalpy rise during a CRDA is specified for BWRs as a function of hydrogen concentrations in the fuel cladding, as per NUREG-0800 Chapter 4.2 [3]. Hydrogen concentration in the cladding increases with exposure and causes zircalloy cladding to become more brittle. Thus, as the cycle proceeds and the hydrogen content increases, the maximum allowed change in pellet enthalpy decreases, as shown in Figure 1. According to the NRC's Standard Review Plan 4.2, four interim coolability criteria must be maintained: the peak radial average fuel enthalpy must remain below 230 cal/g, maximum fuel temperature must remain below the melting point, mechanical energy generated due to fuel-to-coolant interaction and fuel rod burst must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity, and there must be no loss of coolable geometry due to pellet and cladding fragmentation [3]. To satisfy these conditions, the core must be designed and control blade withdrawals must be specified to prevent excessive control blade worths – at the expense of utility operation flexibility and economics.

Due to this detriment, it becomes beneficial to study the severity of control rod drop accidents at various core sizes. This thesis aims to compare the static control blade worth at common reactor core sizes to determine any correlation between the two. The initial core-resizing analysis was performed at GE Hitachi using PANACEA [4], but was re-created and completed at Penn State using SIMULATE [5]. Chapter two outlines the model creation at both GE Hitachi and Penn State and the simulation of the CRDA, and chapter three presents the results found from the developed models.

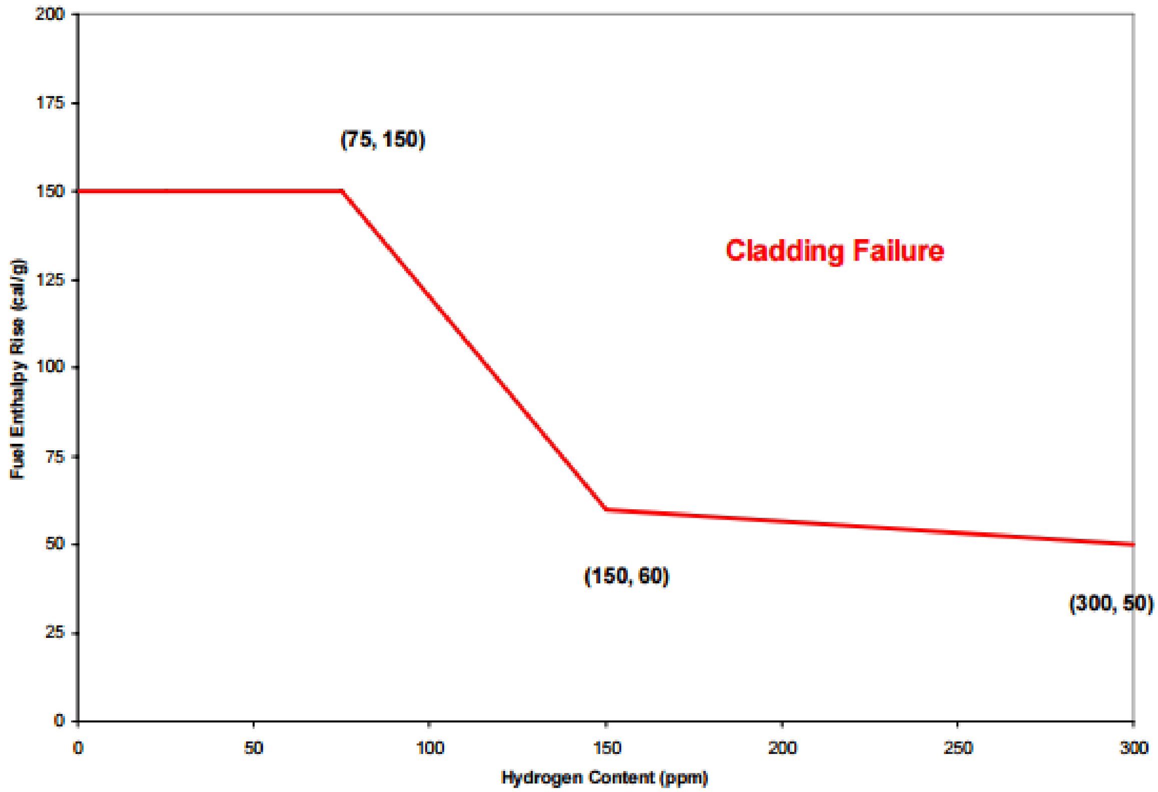

Figure 1: Cladding failure limits as hydrogen concentration increases [3].

Chapter 2

MODEL DESCRIPTIONS

Process to Reduce Core Size

This section describes the creation of the core sizes that were studied. To create the six cores for the CRDA analysis, the advanced BWR (ABWR) core was systematically reduced to the sizes of interest – 872, 764, 724, 560, 400 and 368 fuel bundles. These values represent the full range of currently operating BWR core sizes.

The initial ABWR core consists of 872 fuel bundles of five different assembly types, labeled one through five. Fifty four percent of the core (not including the periphery) is made up of type two bundles, twenty percent type three, fifteen percent type four, and ten percent type five. Type one bundles comprise the periphery of the core. The core is a fresh core, so none of the bundles have any exposure prior to the analysis. Figure 2 shows the color-coded core map for the ABWR core. Each number represents a unique bundle type, which corresponds with pre-made bundles in the GEH databank. The ABWR core is ideal for the reduction process as it is the largest forced circulation core size. Reducing the core size proves significantly easier than increasing a smaller core, because reducing only requires removing already available information, as opposed to supplying new information for a small core. The core chosen represents an initial ABWR core, eliminating the necessity to manipulate the bundle exposures as the core is reduced.

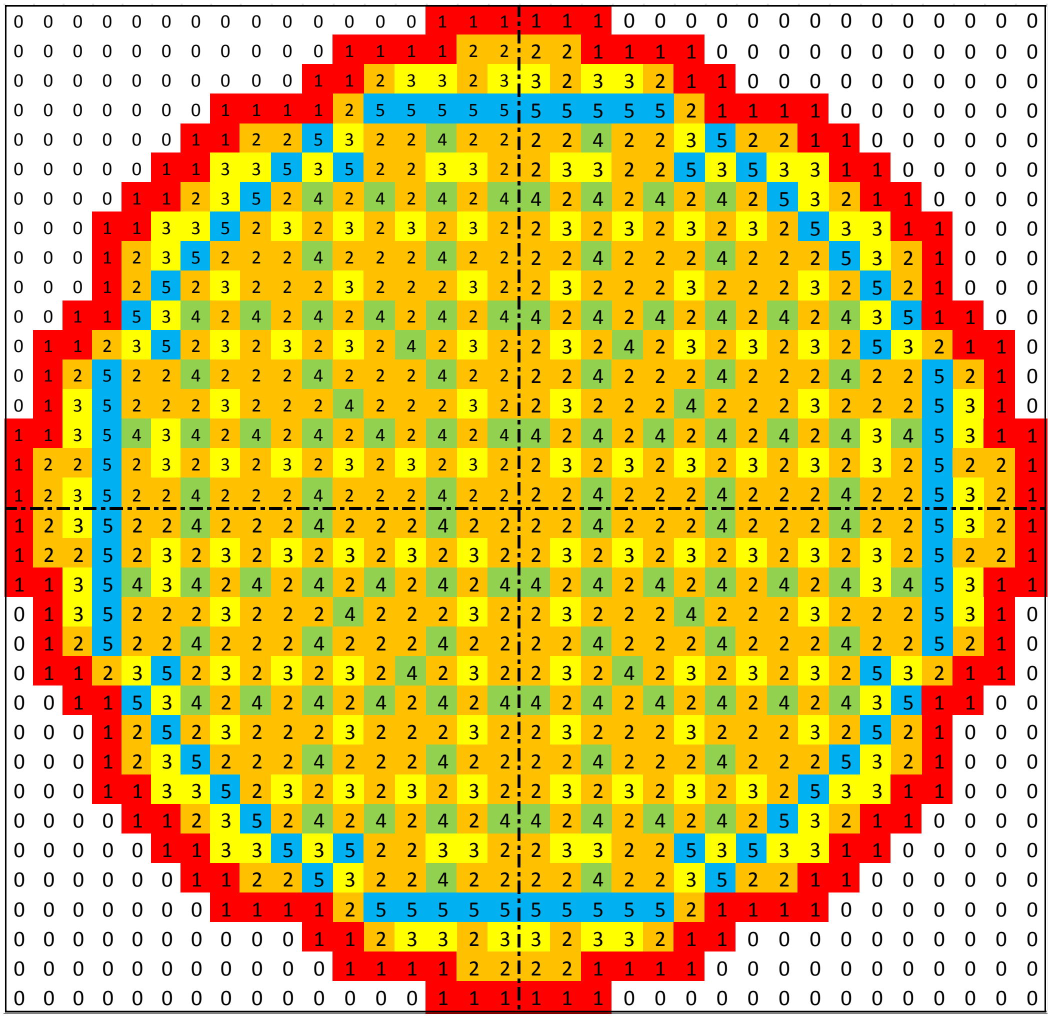

Figure 2: Example of a ABWR core, including various bundle types.

Several assumptions were made when performing the core re-size on the ABWR core. First, all cores were assumed to be quadrant-symmetric and thus, only one quarter of the core had to be altered. Additionally, the maximum array dimensions remained constant, which eliminates the need to further alter any thermodynamic qualities within the core. Finally, the inconsistencies in bundle type near the periphery were assumed to have negligible impact on the worth of the control blades in the center of the core.

The overall goal of the core-resizing process was to create six cores that remained similar enough to justify an accurate comparison between CRDA analysis results. This similarity was proven via the eigenvalue for each core size; factoring in leakage, the values should be within 0.005 of each other to be considered sufficiently similar. Thus, the goal of the re-sizing process was to maintain the same eigenvalue as the size was reduced. To accomplish this goal, peripheral bundles were removed while preserving the bundle configuration near the most-limiting, highest-worth control rod. Additionally, the ratio of each bundle type (1-5) was held approximately constant, to maintain the same power shape across the different sizes.

For the first reduction – 764 bundles – the periphery was defined using the core map for an existing 764 bundle core. To maximize the similarity between the initial and smaller core, the area within the white outline remained identical to the ABWR core. This duplicated area was defined as the largest number of interior bundles that still allowed for one or two rows/columns between the replicated bundles and peripheral bundles. Once the peripheral and internal areas were defined, bundles were added to the remaining rows one type at a time until the core consisted of 764 bundles with approximately the same percentage of bundle types 2-5 as the initial core. The fraction of each bundle type was continuously tracked using an Excel if/else function. In the case of the 764 core size, bundle type 4 could not be reduced, as all of the type 4 bundles were located within the unchanged interior section. Figure 3 shows the quarter core reduction for the 764-bundle core, and Table 1 shows the number and relative percentage of each bundle type.

0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	1	1	1	1	1	1
0	0	0	0	0	0	0	0	0	1	1	1	3	5	2	5	5
0	0	0	0	0	0	0	0	1	3	5	3	2	2	4	2	2
0	0	0	0	0	0	0	1	3	5	3	5	2	2	3	3	2
0	0	0	0	0	0	1	2	5	2	4	2	4	2	4	2	4
0	0	0	0	0	1	2	5	2	3	2	3	2	3	2	3	2
0	0	0	0	1	3	5	2	2	2	4	2	2	2	4	2	2
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0	0	1	5	2	3	2	3	2	3	2	3	2	3	2	3	2
0	0	1	5	2	2	4	2	2	2	4	2	2	2	4	2	2

Figure 3: 764-bundle core loading.

Table 1: Total number and percentage of each bundle type in the reduced core.

5	68	10.1190
4	112	16.6667
3	128	19.0476
2	364	54.1667
	672	100.000

This reduction process was repeated for the remaining four desired core sizes. Figures 4-7 show the quarter-core maps for the 724, 560, 400, and 368-bundle cores, and Table 2 shows the bundle type percentages for all six sizes. As with the 764-bundle core, the area within the white line is identical to the ABWR core, and bundles were removed from the outer layers by trial and error to maintain the bundle-type ratios. The peripheral bundle locations for each size were again determined from available core maps of operating reactors at the desired sizes within GEH's databank.

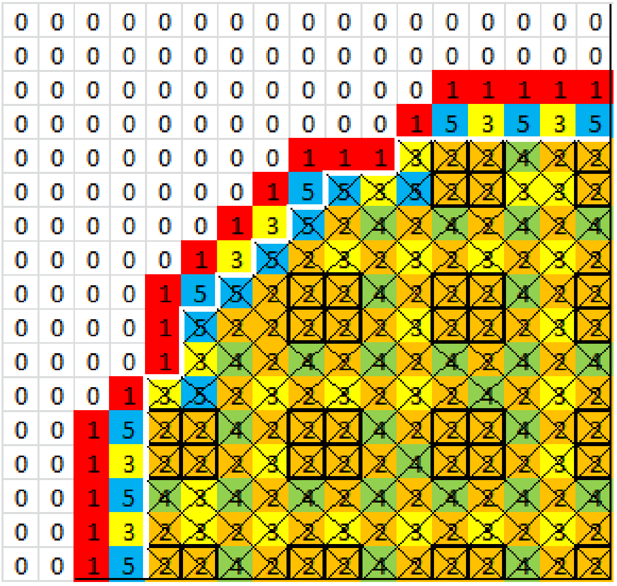

Figure 4: 724-bundle core loading.

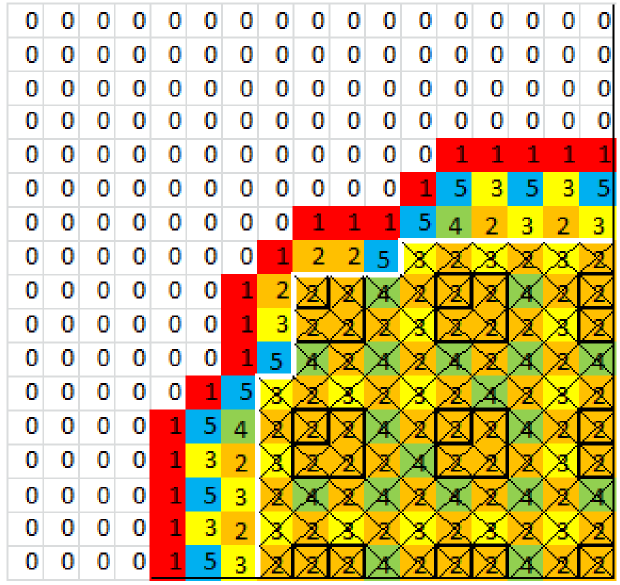

Figure 5: 560-bundle core loading.

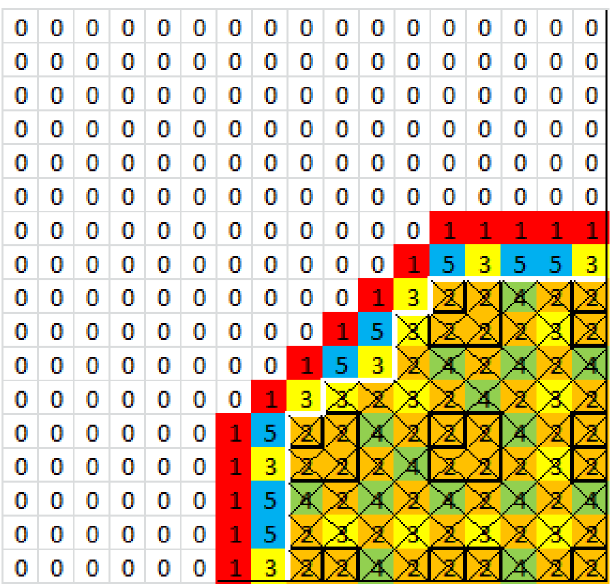

Figure 6: 400-bundle core loading.

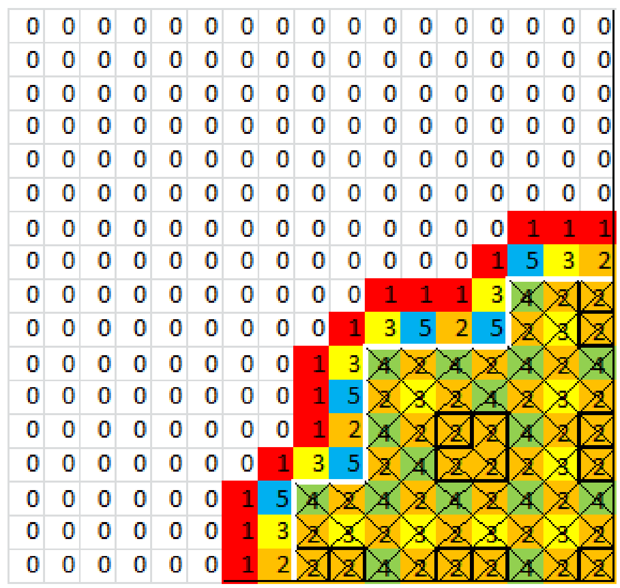

Figure 7: 368-bundle core loading.

Table 2: Percentage of each bundle types for all six core sizes.

Core		Number	Percentage	Core		Number	Percentage
872	5	76	10.27%	560	5	40	8.26%
	4	112	15.14%		4	80	16.53%
	3	152	20.54%		3	100	20.66%
	2	400	54.05%		2	264	54.55%
764	5	68	10.12%	400	5	32	9.41%
	4	112	16.67%		4	60	17.65%
	3	128	19.05%		3	68	20.0%
	2	364	54.17%		2	180	52.94%
724	5	60	9.38%	368	5	24	7.79%
	4	112	17.5%		4	64	20.78%
	3	124	19.38%		3	56	18.18%
	2	344	53.75%		2	164	53.25%

Model Description – SIMULATE

Due to a lack of access to GE's codes at Penn State, the core-resizing process was repeated at Penn State using the steady-state neutronics code, SIMULATE, to replicate the control rod drop accident. As with the previous re-sizing, the largest BWR core available was used as a starting point – in this case, a generic 764-bundle BWR. The core size was again reduced using the same systematic approach to create five sizes of interest – 764, 724, 560, 400, and 368 fuel bundles. Despite the lack of the 872-bundle core, the remaining sizes still effectively represent the range of operating BWRs, since the ABWR and ESBWR are the only two cores larger than 800 assemblies.

A generic 764-bundle BWR core consisting of five assembly types, labeled two through six was used. Type two bundles comprise the periphery, followed by a ring of type 3 bundles and a center interspersed with types 3 through 6. Figure 8 shows the color-coded core map for the generic core and Table 3 lists the relative percentage of each bundle type in the core. As with the ABWR core, the core is fresh, so no bundles have any prior exposure.

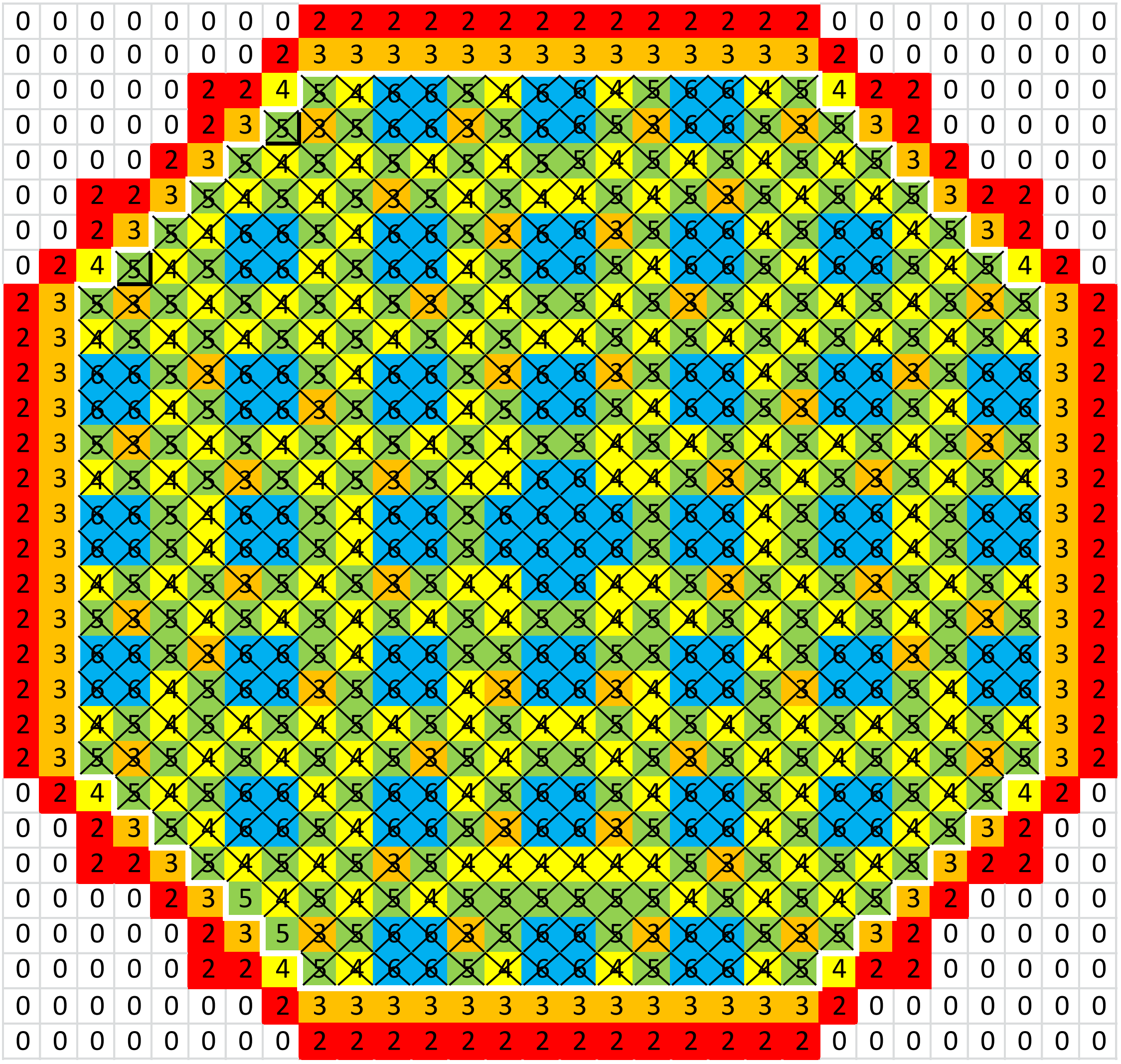

Figure 8: Generic BWR core, including various bundle types.

Table 3: Total number and percentage of internal bundle types in original core.

Core		Number	Percentage
764	6	156	23.21%
	5	224	33.33%
	4	172	25.60%
	3	120	17.86%

In order to reduce the core size, the same Core Re-sizing Process, outlined above, was used. Beginning from the original 764-bundle core, assemblies were removed one by one while maintaining the same core configuration at the center, along with the same approximate ratio of bundle types. Again, the goal was to create core sizes that remained similar for the CRDA analysis, as measured by the eigenvalue. Figures 9 – 12 show the quarter-core maps for the 724, 560, 400, and 368-bundle cores, and Table 4 shows the bundle type percentages for all six sizes. As with the ABWR reduced cores, the area within the white outline remained the same as the original generic BWR input and bundles were removed from the outer layers by trial and error. The periphery locations for each size were determined based on the original GE analysis.

0	0	0	0	0	0	0	0	0	0	2	2	2	2	2
0	0	0	0	0	0	0	0	0	2	3	3	3	3	3
0	0	0	0	0	0	2	2	2	4	6	6	5	4	6
0	0	0	0	0	2	3	5	3	5	6	6	3	5	6
0	0	0	0	2	3	5	4	5	4	5	4	5	4	5
0	0	0	2	3	5	4	5	4	5	3	5	4	5	4
0	0	2	3	5	4	6	6	5	4	6	6	5	3	6
0	0	2	5	4	5	6	6	4	5	6	6	4	5	6
0	0	2	3	5	4	5	4	5	4	5	3	5	4	5
0	2	4	5	4	5	4	5	4	5	4	5	4	5	4
2	3	6	6	5	3	6	6	5	4	6	6	5	3	6
2	3	6	6	4	5	6	6	3	5	6	6	4	5	6
2	3	5	3	5	4	5	4	5	4	5	4	5	4	5
2	3	4	5	4	5	3	5	4	5	3	5	4	4	6
2	3	6	6	5	4	6	6	5	4	6	6	5	6	6

Figure 9: 724-bundle core loading.

0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	2	2	2	2	2
0	0	0	0	0	0	0	0	0	2	3	3	3	3	3
0	0	0	0	0	0	2	2	2	3	5	4	5	4	5
0	0	0	0	0	2	3	5	4	5	3	5	4	5	4
0	0	0	0	2	3	6	6	5	4	6	6	5	3	6
0	0	0	0	2	5	6	6	4	5	6	6	4	5	6
0	0	0	0	2	4	5	4	5	4	5	3	5	4	5
0	0	0	2	3	5	4	5	4	5	4	5	4	5	4
0	0	2	3	5	3	6	6	5	4	6	6	5	3	6
0	0	2	3	4	5	6	6	3	5	6	6	4	5	6
0	0	2	3	5	4	5	4	5	4	5	4	5	4	5
0	0	2	3	4	5	3	5	4	5	3	5	4	4	6
0	0	2	3	5	4	6	6	5	4	6	6	5	6	6

Figure 10: 560-bundle core loading.

0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	2	2	2	2	2
0	0	0	0	0	0	0	0	0	2	3	5	3	5	3
0	0	0	0	0	0	0	0	2	4	6	6	5	3	6
0	0	0	0	0	0	0	2	3	5	6	6	4	5	6
0	0	0	0	0	0	2	3	5	4	5	3	5	4	5
0	0	0	0	0	2	4	5	4	5	4	5	4	5	4
0	0	0	0	2	3	6	6	5	4	6	6	5	3	6
0	0	0	0	2	5	6	6	3	5	6	6	4	5	6
0	0	0	0	2	3	5	4	5	4	5	4	5	4	5
0	0	0	0	2	5	3	5	4	5	3	5	4	4	6
0	0	0	0	2	3	6	6	5	4	6	6	5	6	6

Figure 11 : 400-bundle core loading.

0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	2	2	2
0	0	0	0	0	0	0	0	0	0	0	2	3	5	3
0	0	0	0	0	0	0	0	2	2	2	4	5	3	6
0	0	0	0	0	0	0	2	3	6	6	5	4	5	6
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0	0	0	0	0	0	2	6	5	4	6	6	5	3	6
0	0	0	0	0	2	4	5	3	5	6	6	4	5	6
0	0	0	0	2	3	5	4	5	4	5	4	5	4	5
0	0	0	0	2	5	3	5	4	5	3	5	4	4	6
0	0	0	0	2	3	6	6	5	4	6	6	5	6	6

Figure 12: 368-bundle core loading.

Table 4: Percentage of each bundle type for all five core sizes.

Core		Number	Percentage	Core		Number	Percentage
764	6	156	23.21%	400	6	92	27.06%
	5	224	33.33%		5	116	34.12%
	4	172	25.60%		4	76	22.35%
	3	120	17.86%		3	56	16.47%
724	6	156	24.38%	368	6	76	24.68%
	5	216	33.75%		5	108	35.06%
	4	164	25.63%		4	76	24.68%
	3	104	16.25%		3	48	15.58%
560	6	108	22.31%				
	5	164	33.88%				
	4	124	25.62%				
	3	88	18.18%				

The cores were inputted into SIMULATE as quarter-core inputs with the assumption of quarter-core symmetry. Additionally, several other inputs were changed to alter the core size, including the control rod locations, periphery location, and inlet orifice locations. The power and flow were set at 100%, indicating hot full power conditions with all of the control rods withdrawn.

Model Description – CRDA

After completing the core re-sizing process, a control rod drop accident was simulated, using SIMULATE. The accident concerns the highest worth control rod, which would cause the greatest spike in reactivity during an accident. SIMULATE models the rod drop as a steady-state phenomenon, meaning that data is only available for directly before the accident and after the rod has dropped fully out of the core. This section describes the development of the control rod drop accident simulation using SIMULATE.

The generic BWR modeled for the CRDA comes from the example inputs included with the SIMULATE code and utilizes assemblies stored in the fuel library databank. The core is 3.83 meters high with a power density of 50 kW/L and is set to run at 100% rated power and flow. There are five bundle types loaded into the core with enrichments varying from .7% (natural uranium) to 2.96% U-235. The bundle configuration, along with the reflector locations can be manually inputted. Control rods are located between each of the bundles and can be inserted or withdrawn in one node (7.62 centimeter) intervals.

Before performing the rod drop, the highest worth rod had to be determined. Since the rod of interest must exist in all five core sizes, the smallest (368-bundle) core was used to find this control rod. Using the manual control rod input, every rod was individually inserted and then withdrawn, with the remaining control rods at their fully withdrawn positions. The rod pull which resulted in the greatest change in eigenvalue was taken as the highest worth rod and considered the rod of interest for remaining calculations. After inserting and withdrawing all the rods, the highest worth rod was found to be at location (7,7) in the 368-bundle core. This rod was subsequently used in all core sizes to perform the CRDA.

After determining the highest-worth rod, the CRDA was performed by fully inserting the rod of interest, then withdrawing it completely out of the core. The rod block mechanism was not credited, as to give the largest possible reactivity spike at the location of the drop. Two control rod location inputs were added to the generic BWR input in SIMULATE to define the control blade locations at the desired set points. Figure 13 shows the position of all of the control rods in the 764-bundle core before and after the rod drop. Zero indicates a control rod that is fully inserted and 48 indicates fully withdrawn, with the rod of interest is outlined in red. The procedure was repeated for the four primary desired core sizes – 764, 560, 400, and 368 bundles – by changing the size of the control rod input to match the specified dimensions. As with the core-resizing, the rod drop was performed at hot full power. Additionally, a pin edit was turned on to perform pin reconstruction and determine the location of the power peak during the rod drop. Appendix B includes the full input file for the 764-bundle core.

Before:

48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
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48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
48	48	48	48	48	48	48	48	48	48	48	48	48	48	48

After:

48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
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48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
48	48	48	48	48	48	48	48	48	48	48	48	48	48	48

Figure 13: Control rod positions before and after the rod drop.

Chapter 3

RESULTS

Core Re-size Process

Based on the core-resize process, eigenvalue was found to vary with core size – decreasing as the size decreased. This decrease can be attributed to increased core leakage as the size decreases. In order to prove similarity for the sake of CRDA comparison, the calculated eigenvalues must lie within ±0.005 of the original core eigenvalue. However, due to the leakage, this “base case” eigenvalue must be corrected at each individual core size using the core geometry and assumed leakage value.

The leakage for the reference cores (the ABWR and generic 764-bundle core) was assumed to be 0.0275, which yields a k_∞ and diffusion area of 1.034 and 164.6 cm^2, respectively, for the 764-bundle core. These two values were assumed constant since they are independent of core size, and were used to calculate leakage fraction values for the remaining three core sizes. From there, the following equation was used to calculate the desired eigenvalue for each size:

$$k = \frac{k_\infty}{1+L^2B_g^2} \quad [6]$$

The geometric buckling was calculated using the known fuel dimensions and number of bundles. The height of the core was assumed to be 381 cm, with a pitch of 15.5 cm. These new k_{eff} values were used as the standard for determining similarity amongst the cores, with an allowed tolerance of ±0.005. Appendix A gives a detailed account of the calculations used to determine the corrected eigenvalues.

SIMULATE Core Re-Size

Using the core-resizing and eigenvalue comparison methodology developed at GE, the five core sizes of interest were run to determine the change in eigenvalue over the various sizes. Table 5 shows the SIMULATE output eigenvalues for the five sizes run – 764, 724, 560, 400, and 368 assemblies.

Table 5: Eigenvalue results for various core sizes.

	764	724	560	400	368
Eigenvalue	1.00613	1.00463	1.00121	0.99067	0.99010

These values were corrected to account for the additional leakage as the core shrank using the methodology described in the Core Re-Size Process Results and in Appendix A. The geometric buckling value remained the same as the ABWR, based on an assumed core height of 3.81 meters and bundle area of 250 square centimeters. Table 6 compares the expected eigenvalue to that determined by SIMULATE, and Figure 14 shows a plot of the two values. The error bars on the graph represent the ±0.005 limit. Using the corrected values, all results still fall within the allowed tolerance, indicating that the core reduction methodology provides similar core sizes to analyze using SIMULATE.

Table 6: Comparison of estimated and calculated eigenvalues.

	764	724	560	400	368
Calculated Eigenvalue	1.006013	1.00463	1.00121	0.99067	0.99010
SIMULATE Eigenvalue	1.006013	1.00513	1.00023	0.99169	0.98912
Difference	0.00000	0.00050	-0.00098	0.00102	-0.00098

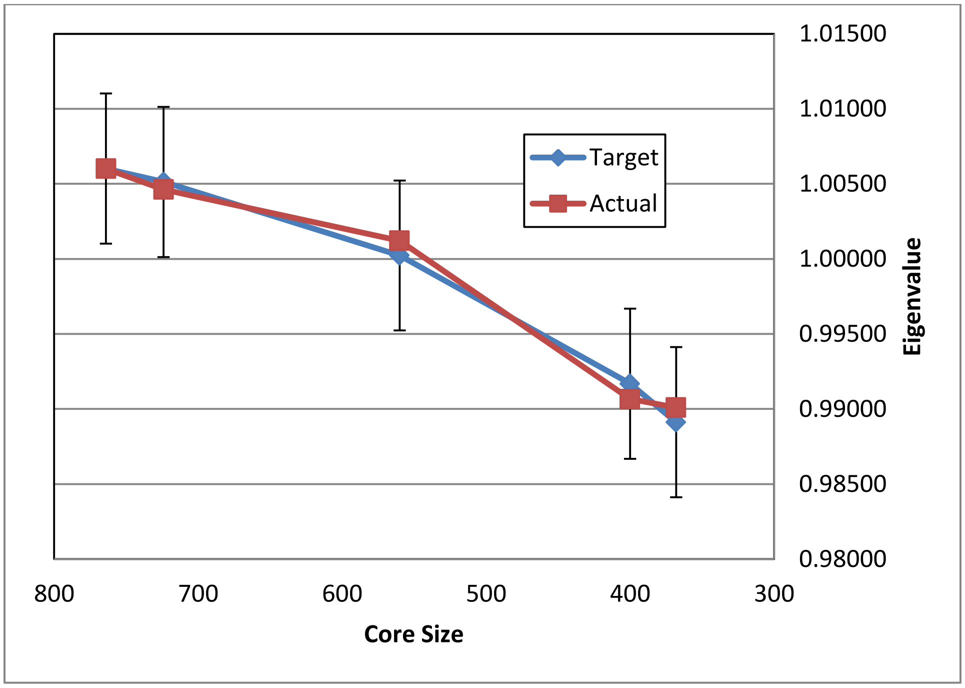

Figure 14: Calculated versus expected eigenvalues for the five core sizes.

Control Rod Drop Accident

After proving similarity between the five core sizes of interest, the highest worth control rod was used to perform the control rod drop accident and compare the severity of the accident over the array of sizes. The severity was measured by comparing the blade worth of the rod of interest and the power peaking. The control blade worth was calculated using the following equation:

$$\rho = \frac{k_2 - k_1}{k_1 k_2} \quad [6]$$

That value was converted into dollars by dividing by an approximate beta fraction (0.007) [6]. Figure 15 shows a graph of the control blade worths over the range of sizes.

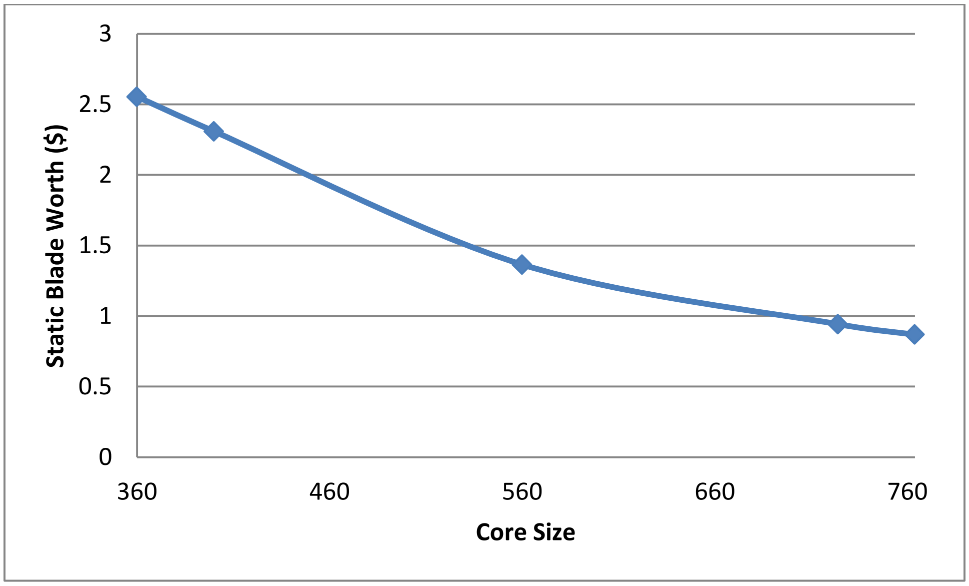

Figure 15: Blade worth of the rod of interest as a function of core size.

As expected, the worth of the blade of interest decreases as the core size increases. This trend can be attributed to the difference in area over which the core power is distributed at each size. In a larger core, there are more control blades and the power is spread over a larger area. Thus, the relative reactivity insertion of one rod is fairly small. In a smaller core, each control blade has a larger effect on the core power, causing a larger insert of reactivity in the case of a CRDA.

Similar results were seen when comparing the peaking factors at each core size. The peaking factor was determined from the "3-PIN" pin reconstruction edit, included in the output file summary. While the power peak did not occur adjacent to the blade of interest, as with the blade worth, the change in F_Q also increased as the core size decreased. Figure 16 shows the change in peaking factor as the core size decreases. This decrease is caused by the same phenomenon as the trend in blade worth; the power is more concentrated in smaller cores, causing a more severe power spike during an accident scenario than in larger cores where the power is spread over a wider area.

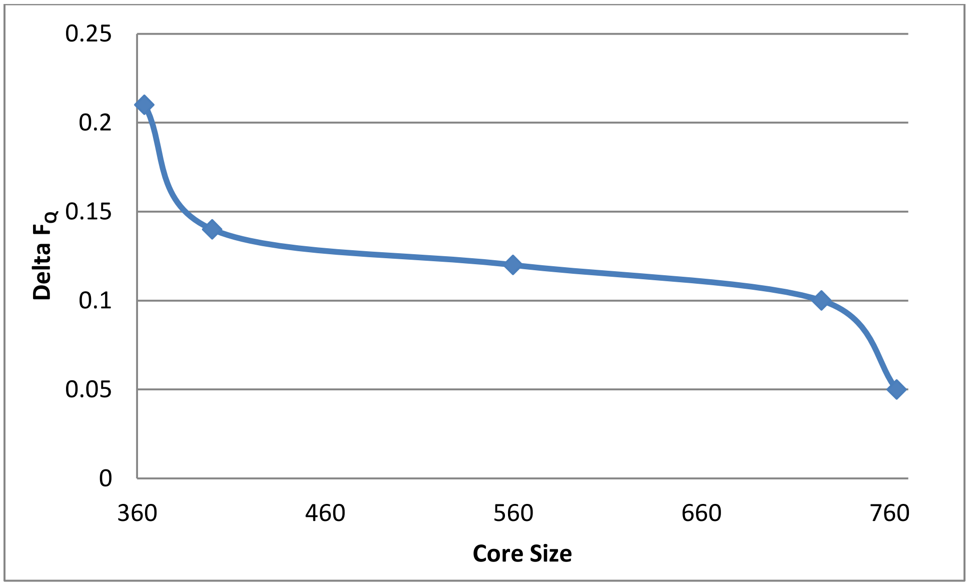

Figure 16: Change in peaking factor as a function of core size.

Together, these results suggest that the severity of a CRDA increases as reactor core size decreases. The static blade worth can be correlated with the hydrogen content in the core and subsequently the fuel enthalpy. Thus, a higher static blade worth equates to a larger increase in fuel enthalpy and a greater chance of causing fuel melt or damage.

Chapter 4

SUMMARY/CONCLUSIONS

Over each fuel cycle at a nuclear reactor, engineers must simulate accident conditions in addition to normal operation conditions to assure the safety of the reactor in the event of an accident. These transients, such as loss of coolant accidents, over-pressurization events, and reactivity insertion accidents, dictate the limits at which the reactor can operate. Thus, utilities consistently search for ways to maximize power while maintaining compliance with nuclear regulatory commission limits and maintaining safe operation of the reactor. In the case of a control rod drop accident, reactor size has a significant effect on accident severity. Understanding this effect provides a better understanding of CRDA limitations, benefiting both vendors and utilities.

Control rod drop accidents are simulated by using a steady-state neutronics code to determine the power spike caused by the highest-worth rod in the core falling from its fully-inserted to fully-withdrawn position. Using the highest worth rod and dropping the rod from its fully-inserted position gives the most conservative result and is thus used as the basis for meeting the necessary NRC criteria. The severity of the CRDA depends on the worth of the dropped control blade, which is determined via the eigenvalue before and after the rod drop.

In order to effectively compare accident severity at different core sizes, five similar core sizes were created by systematically removing fuel assemblies. The similarity between the cores was verified by comparing the initial eigenvalue of each to a calculated value based on the expected leakage. If the calculated value falls within ±0.005 of the expected value, the core can be considered similar and valid for comparison of CRDA results. The core reduction process used produced cores within this limit, indicating that the five offered a valid comparison for CRDA simulation.

After establishing similarity, the highest worth rod was found by inserting and withdrawing every control rod in the smallest core size of interest. This rod was subsequently "dropped" from its full-in to full-out position to simulate the accident, with the remaining control blades fully-withdrawn from the core. Based on the eigenvalues before and after the rod drop, the magnitude of the rod worth was found to vary inversely with the core size – decreasing as the core size increased. The same trend was observed in the change in peaking factor before and after the accident. These results stem primarily from the greater concentration of power in smaller cores, as opposed to larger cores. Because of this increased power concentration, an accident like a CRDA that involves a sudden sharp increase in power produces a larger prompt fuel enthalpy increase in a smaller core compared to a larger core of similar design.

Currently, the NRC offers one limit for fuel enthalpy, temperature, and mechanical strain increase during said accidents that is consistent across all core sizes. Based on the demonstrated increase in accident severity as core size decreases, it is hypothesized that the acceptance criteria will be slightly more restrictive for a smaller core than a larger core of similar design. Determination of exactly how much more restrictive would require more extensive analysis CRDAs at varying core sizes, including calculation of the hydrogen concentration and subsequently the fuel enthalpy based on the computed static blade worth values.

Moving forward, it would be beneficial to repeat this experiment at various cycle burnups and using an equilibrium core as opposed a fresh core. Including analysis at additional burnup ensures that the most severe case is evaluated when considering future CRDA limit values. Using an equilibrium core would give more representative blade worth values, as the fuel assemblies will behave differently after being exposed for one or two cycles than they will with all fresh fuel. Additionally, more accurate results could be determined if the dynamic blade worth is considered instead of the static blade worth. Dynamic blade worth is determined during a CRDA as the net peak reactivity during the transient minus the initial reactivity at steady state before the blade began to move. Determination of this value requires the use of a time-dependent neutronics code, which proves significantly more time-consuming and costly and, thus, was not included in the scope of this project.

With further research, successful demonstration of the effect of core size of CRDA severity could offer the nuclear industry another means of optimizing core design to increase efficiency without compromising safety. While more in-depth analysis will be required to prove the full extent of this effect, this thesis serves as an overview of the general trend and a starting point for future investigation.

Appendix A – Method to Calculate Eigenvalue

Neutron leakage probability for a bare homogeneous reactor with one neutron energy group:

$$P_L = 1 + L^2 B_g^2 \tag{1}$$

$$L\ (cm) = \sqrt{\frac{D}{\Sigma_a}} \equiv neturon\ leakage \tag{2}$$

$$B_g^2(cm^{-2}) = \left(\frac{v_o}{\tilde{R}}\right)^2 + \left(\frac{\pi}{\tilde{H}}\right)^2 \equiv geometric\ buckling\ (cyclinder) \tag{3}$$

D (cm) = diffusion coefficient

Σ_a (cm^{-1}) = macroscopic absorption cross section

Vo = 2.405; smallest zero of J0 Bessel function

$\tilde{R}$ (cm) = reactor radius

$\tilde{H}$ (cm) = reactor height

To find the expected eigenvalue:

$$k = k_\infty P_L^{-1} = \frac{k_\infty}{1+L^2 B_g^2} \tag{4}$$

Core radius approximated by number of fuel assemblies.

$$A_{core} = \tilde{R}^2 \pi = N A_{bundle} \tag{5}$$

$$A_{bundle} = P_{bundle}^2\ (P = pitch) \tag{6}$$

$$\widetilde{R^2} = \frac{N P_{bundle}^2}{\pi} \tag{7}$$

Substituted to solve for buckling:

$$B_g^2\ (cm^{-2}) = \frac{\pi}{N}\left(\frac{2.405}{P_{bundle}}\right)^2 + \left(\frac{\pi}{\tilde{H}}\right)^2 \tag{8}$$

Need to find the initial k_∞ value to determine the leakage for the smaller cores.

$$k_\infty = (1 + L_{Fr})k \tag{9}$$

$$L_{Fr} = L_R^2 B_g^2\ \equiv reference\ leakage \tag{10}$$

k = eigenvalue for initial reference core

L_{Fr} = [0.02,0.03]

Values for k_∞ and L_R from the reference core are applied to the reduced core since they depend only on materials. For the reduced core sizes, Bg is recalculated (equation 8) and used to solve for a new leakage. That leakage value is then used to find the expected eigenvalue using equation 4.

Appendix B – 764-Bundle Input File

```
'COM'  * -----------------------------------------------------* /
'COM'  *                                                      * /
'COM'  *                GENERIC BWR MODEL                     * /
'COM'  *                                                      * /
'COM'  *                 C Y C L E    1                       * /
'COM'  *                                                      * /
'COM'  * -----------------------------------------------------* /

'DIM.BWR' 30 15 /                * 30 fuel assemblies, 15 c-rods & 16 det. loc.
'DIM.CAL' 25 2 1 1 /            * 25 axial nodes, Quarter core
'DIM.DEP' 'EXP' 'XEN' 'SAM' 'HVOI' 'HCRD' 'PIN' /   * Select depletion arguments

'TIT.CAS' '764-bundle'/   * Title for each case in run
'TIT.PRO' 'Generic BWR model'/                * Project name
'TIT.RUN' 'BaseDepl'/                         * Run name for this stack of cases

'COR.DAT' 15.24 381.0 50.0 266.3 /                * Core data given above
'COR.OPE' 100. 100. 1035. /         * Core operating data

'COM' Build axially zoned fuel types and assign axial top and bottom reflectors
'COM' As a matter of practice always specify reflector segments first
'COM' since they will always be required - in all cases

'REF.LIB'  01 'RADREF'/       * Link 'RADREF'in the CMSLINK library to SIMULATE
segment number 01
'REF.LIB'  02 'BOTREF'/
'REF.LIB'  03 'TOPREF'/
'SEG.LIB'  04 'BWRU071G00'/   * Fuel composition: same as for reflector case
'SEG.DAT'  04  0.711/
'SEG.LIB'  05 'BWRU096G00'/
'SEG.DAT'  05  0.96/
'SEG.LIB'  06 'BWRU296G43'/
'SEG.DAT'  06  2.96/
'SEG.LIB'  07 'BWRU261G33'/
'SEG.DAT'  07  2.61/
'SEG.LIB'  08 'BWRU171G22'/
'SEG.DAT'  08  1.71/

'PIN.EDT' 'ON'/

'COM' *--------------------- Specify the fuel assembly axial zones    ----------------
---------*

'FUE.ZON'  01 1 'RADIAL'   02 0.00  01                       381.0 03 /
'FUE.ZON'  02 1 'E071'     02 0.00  04                       381.0 03 /
'FUE.ZON'  03 1 'E296-4G3' 02 0.00  04 15.24  06  365.76  04 381.0 03 /
'FUE.ZON'  04 1 'E261-2G3' 02 0.00  04 15.24  07  365.76  04 381.0 03 /
'FUE.ZON'  05 1 'E171-2G2' 02 0.00  04 15.24  08  365.76  04 381.0 03 /
'FUE.ZON'  06 1 'E096'     02 0.00  04 15.24  05  365.76  04 381.0 03 /


'FUE.TYP'  1,  6  6  5  6  6  4  5  6  6  4  5  6  6  3  2  1/
'FUE.TYP'  2,  6  4  4  5  3  5  4  5  3  5  4  5  4  3  2  1/
'FUE.TYP'  3,  5  4  5  4  5  4  5  4  5  4  5  3  5  3  2  1/
'FUE.TYP'  4,  6  5  4  6  6  5  3  6  6  5  4  6  6  3  2  1/
'FUE.TYP'  5,  6  3  5  6  6  4  5  6  6  3  5  6  6  3  2  1/
'FUE.TYP'  6,  4  5  4  5  4  5  4  5  4  5  4  5  4  3  2  1/
'FUE.TYP'  7,  5  4  5  3  5  4  5  4  5  4  5  3  5  3  2  1/
```

```
'FUE.TYP'  8,  6  5  4  6  6  5  4  6  6  5  4  5  4  2  1  1/
'FUE.TYP'  9,  6  3  5  6  6  4  5  6  6  4  5  3  2  1  1  0/
'FUE.TYP' 10,  4  5  4  5  3  5  4  5  4  5  3  2  2  1  0  0/
'FUE.TYP' 11,  5  4  5  4  5  4  5  4  5  3  2  1  1  1  0  0/
'FUE.TYP' 12,  6  5  3  6  6  5  3  5  3  2  1  1  0  0  0  0/
'FUE.TYP' 13,  6  4  5  6  6  4  5  4  2  2  1  0  0  0  0  0/
'FUE.TYP' 14,  3  3  3  3  3  3  3  2  1  1  1  0  0  0  0  0/
'FUE.TYP' 15,  2  2  2  2  2  2  2  1  1  0  0  0  0  0  0  0/
'FUE.TYP' 16,  1  1  1  1  1  1  1  1  0  0  0  0  0  0  0  0/

'FUE.BAT'  1,  5  5  4  5  5  3  4  5  5  3  4  5  5  2  1/
'FUE.BAT'  2,  5  3  3  4  2  4  3  4  2  4  3  4  3  2  1/
'FUE.BAT'  3,  4  3  4  3  4  3  4  3  4  3  4  2  4  2  1/
'FUE.BAT'  4,  5  4  3  5  5  4  2  5  5  4  3  5  5  2  1/
'FUE.BAT'  5,  5  2  4  5  5  3  4  5  5  2  4  5  5  2  1/
'FUE.BAT'  6,  3  4  3  4  3  4  3  4  3  4  3  4  3  2  1/
'FUE.BAT'  7,  4  3  4  2  4  3  4  3  4  3  4  2  4  2  1/
'FUE.BAT'  8,  5  4  3  5  5  4  3  5  5  4  3  4  3  1/
'FUE.BAT'  9,  5  2  4  5  5  3  4  5  5  3  4  2  1/
'FUE.BAT' 10,  3  4  3  4  2  4  3  4  3  4  2  1  1/
'FUE.BAT' 11,  4  3  4  3  4  3  4  3  4  2  1/
'FUE.BAT' 12,  5  4  2  5  5  4  2  4  2  1/
'FUE.BAT' 13,  5  3  4  5  5  3  4  3  1  1/
'FUE.BAT' 14,  2  2  2  2  2  2  2  1/
'FUE.BAT' 15,  1  1  1  1  1  1  1/

'LIB'  'cl.bwr.lib' /    * Link to the library generated by CMS-LINK
'DEP.CYC' 'CYCLE01' 0.0  01 /
'DEP.FPD' 3/
'FUE.INI' 'JILAB'/
'BAT.LAB' 1,'C1NAT' 2,'C1U296' 3,'C1U261' 4,'C1U171' 5,'C1U096'/
'BAT.EDT' 'ON' '2RPF' '3RPF' '2EXP' '3EXP' '2KIN' '3KIN'/

'COM' *---------------    Control rod data  --------------*

'CRD.DAT' 48 7.62 /                              * Number of steps and a single step
in cm
'CRD.ZON' 1  2  'CRUCIFORM' 10  365.76  0  381.0 /
'CRD.SYM' 4  0 /
'CRD.GRP' 1
          4*0            1  1  1  1  1  1  1             4*0
          3*0         1  1  1  1  1  1  1  1  1          3*0
          2*0      1  1  1  1  1  1  1  1  1  1  1       2*0
            0   1  1  1  1  1  1  1  1  1  1  1  1  1      0
             1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
             1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
             1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
             1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
             1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
             1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
             1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
            0   1  1  1  1  1  1  1  1  1  1  1  1  1      0
          2*0      1  1  1  1  1  1  1  1  1  1  1       2*0
          3*0         1  1  1  1  1  1  1  1  1          3*0
          4*0            1  1  1  1  1  1  1             4*0  /

'COM' *-----------------  Fuel Temperature data  -----------------*

'SEG.TFU', 0,    0.0,  206.5,  17.2 / * Segment fuel temperature fit calculated by
INTERPIN
'COM' EXPOSURE IN GWD/MT
'TAB.TFU'   1, 0, 'EXP', 8, 'POW',      1
```

```
           0.0    5.0   10.0   15.0   20.0   30.0   40.0   50.0
   1.0,   83.3   46.7   22.0    0.0   -2.7   -0.7    1.1    1.7 /

'COM' *------------- BWR Heat Balance data  -----------------*

'BWR.BAL' 'ON' 0.001, 120, 3.32E-3, 6.197E-4, 1.859E-3/ DEFAULTS FROM MANUAL
'BWR.CTP' 40., 60.,   80.,   90., 100./                 DEFAULTS FROM MANUAL
'BWR.TFW' 363, 409, 422.0, 430.0, 434.0/
'BWR.CWT'      40,      60,      80,      90,     100/  DEFAULTS FROM MANUAL
'BWR.QRP' 0.10E-4, 0.30E-4, 0.80E-3, 1.25E-3, 3.00E-3/
'BWR.FCU' 5*0.0025/                                     DEFAULTS FROM MANUAL

'COM' *------------- Essential BWR Hydraulic data  -----------------*

'BWR.ELE' -33. 0.0 381. 418. 525. 799.99 880. / * Core elevation data
'BWR.SEP' 225 12.84 /                            * No of separators and the pressure
loss coeff.
'BWR.TYP' 1
          1 1 1 1 1 1 1 1 1 1 1 1 1 1 2
          1 1 1 1 1 1 1 1 1 1 1 1 1 1 2
          1 1 1 1 1 1 1 1 1 1 1 1 1 1 2
          1 1 1 1 1 1 1 1 1 1 1 1 1 1 2
          1 1 1 1 1 1 1 1 1 1 1 1 1 1 2
          1 1 1 1 1 1 1 1 1 1 1 1 1 1 2
          1 1 1 1 1 1 1 1 1 1 1 1 1 1 2
          1 1 1 1 1 1 1 1 1 1 1 1 1 2   1*0
          1 1 1 1 1 1 1 1 1 1 1 1 2     2*0
          1 1 1 1 1 1 1 1 1 1 1 2 2     2*0
          1 1 1 1 1 1 1 1 1 1 2         4*0
          1 1 1 1 1 1 1 1 1 2           5*0
          1 1 1 1 1 1 1 1 2 2           5*0
          1 1 1 1 1 1 1 2               7*0
          2 2 2 2 2 2 2                 8*0  / * Location of inlet orifices

'BWR.KOR' 18.8 180.4 / * Loss coeff. for the two types of inlet orifices
'HYD.ITE'        / * Perform Hydraulic iteration
'BWR.DLP'  'BYP' / * Activate the core flow distribution model
'BWR.WLT'  5   5
               20.    40.    60.    80.    100.
          20.  0.011  0.063  0.081  0.088  0.090
          40.  0.028  0.075  0.093  0.097  0.098
          60.  0.050  0.091  0.106  0.108  0.106
          80.  0.064  0.107  0.118  0.117  0.115
         100.  0.075  0.120  0.129  0.126  0.122 /

'BWR.SUP' 185  3774.  0.0  0.0  -10.0
           55   114.  0.0  0.0  -10.0
            1 26496.  0.0  0.0  -10.0 / * Core support plate and shroud leakage path
data


'COM' *------------- BWR Flow distribution Hydraulic data  -----------------*

'BWR.WTR' 1 2 0.0 381.0 1.32 1.5 132. 3.99 / * Flow characteristics of water tod
'BWR.DIM' 1 1.23 1.5 13.4 0.25 0.97  / * Fuel design of GE8X8 with two water rods
'BWR.SPA' 1 48.28 99.12 149.96 201.90 252.74 303.58 355.52 / * Locations of axial
spacer grids (for T-H modeling)
'BWR.LOS' 1  8.07  1.15  1.36 /    * Tie plate and spacer loss coeff.
'BWR.LKG' 1 1850. 0.5  -10. 0
             702. 0.71 -10. 0 / * Assembly leakage flow paths to the interstitial
bypass region
```

```
'PRI.INP' 'CMAP'/
'CMS.EDT' 'ON'  '3RPF' '3EXP'  '3KIN'/
'WRE' 's3.bwr.01.res' 0.0 /
'STA'/

'DEP.STA' 'AVG' 0.0/
'STA'/

'CRD.POS' 1
48 48 48 48 48 48 48 48 48 48 48 48 48 48 48
48 48 48 48 48 48 48 48 48 48 48 48 48 48 48
48 48 48 48 48 48 48 48 48 48 48 48 48 48 48
48 48 48 48 48 48 48 48 48 48 48 48 48 48 48
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'STA' /

'CRD.POS' 1
48 48 48 48 48 48 48 48 48 48 48 48 48 48 48
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48 48 48 48 48 48 48 48 48 48 48 48 48 48 48/
'STA'  /

'END'/
```

BIBLIOGRAPHY

1] "Boiling Water Reactor Power Plant," September 2007. [Online]. Available: http://atominfo.ru/. [Accessed April 2015].

2] D. K. Ivanov, "Fuel Behavior under Transient and LOCA," [Online]. Available: https://www.engr.psu.edu/. [Accessed 12 April 2015].

3] "Fuel System Design," U.S. Nuclear Regulatory Commission, 2007.

4] *PANACEA, [Computer Code]* GE Hitachi Nuclear Energy.

5] *SIMULATE, [Computer Code]* Studsvik.

6] J. Lamarsh and A. Baratta, Introduction to Nuclear Engineering, Upper Saddle River: Prentice Hall, 2001.

ACADEMIC VITA

Kelly McClure
krm5386@psu.edu

EDUCATION

Bachelor of Science in Nuclear Engineering
Schreyer Honors College — May 2015
The Pennsylvania State University, University Park, PA

WORK EXPERIENCE

GE Hitachi Nuclear Energy — Wilmington, NC
Intern – BWR Transient Analysis — Summer 2014

- Performed sensitivity studies to develop a procedure for rod withdrawal error (RWE) analysis for Standard plants/BWR2s.
- Presented results to initiate a formal design change for the Standard RWE procedure.
- Established a method to run RWE events using TRACG and benchmarked outcomes against PANACEA results.

Exelon Corporation — Kennett Square, PA
Intern – BWR Core Design; nuclear fuels — Summer 2013

- Developed a cost-benefit analysis of changing the top natural uranium lattice from 6" to 12" in Clinton Power Station (CPS).
- Compared Zr-4 and NSF as fuel channel materials in CPS.
- Created contingency rod patterns for Clinton's upcoming cycle.
- Utilized PANACEA to model reactor cycles and create core designs.
- Designed fuel bundles and rod pattern designs for future fuel cycles.

Pennsylvania Commercial Management Services — Pittsburgh, PA
Administrative Assistant — Summer 2012

- Organized and monitored property files and office filing systems.
- Created health, safety, and information manuals for residing tenants.
- Communicated with tenants and property owners to maintain current information.

SOFTWARE

PANACEA TRACG SolidWorks C++ Excel

LEADERSHIP & ACTIVITIES

Elected Secretary, American Nuclear Society – Penn State Section (2014-2015)

- Maintain club communications, including meeting minutes and website updates.
- Organize presentations for monthly meetings.

Elected Vice-President, Women in Nuclear (WIN) (2013-2014)

- Worked to gain University recognition as a start-up club.

Events Lead, ANS Student Conference Planning Committee (2013-2014)

- Worked with vendors to organize dinners and social events

HONORS

The President's Freshman Award
Bernadetta and Warren Witzig Nuclear Engineering Merit Scholarship Recipient
National Academy for Nuclear Training Scholarship Recipient
Alpha Nu Sigma Honor Society
Dean's list – all semesters