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Justification for the Use of RAPTOR-M3G for the Catawba Unit 1 Measurement Uncertainty Recapture (MUR) Power Uprate Fluence Evaluations

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RECORD OF REVISIONS

Revision	Description	Completed
0-A	Original Draft Issue	3/27/2015
0-B	Revision 0-B issued to incorporate 1st round of comments from Duke Energy, and it going to be sent to Duke for review.	See EDMS

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1 INTRODUCTION

As part of the Catawba Unit 1 Measurement Uncertainty Recapture (MUR) power uprate project, Westinghouse performed an applicability assessment of the Catawba Unit 1 Pressure-Temperature (P-T) limit curves, and this information was then submitted to the U. S. Nuclear Regulatory Commission (NRC) for review and approval. The evaluations performed by Westinghouse are described in detail in Reference 1. These evaluations included a determination of neutron fluence for several reactor vessel materials.

Reactor vessel neutron fluence has traditionally been quantified with discrete ordinates radiation transport calculations. Codes used to perform early calculations include TWOTRAN (Reference 9) and DOT (Reference 10). With the limitations on computing power at the time, both TWOTRAN and DOT were only capable of analyzing one-dimensional (1D) and two-dimensional (2D) models. In the 1980s, Oak Ridge National Laboratory developed the DORT and TORT codes, and these codes remain in widespread use today. Both the DORT and TORT codes are described in Reference 3. The DORT code is a descendent of the DOT code, and the TORT code is a three-dimensional successor to the two-dimensional DORT and DOT codes.

The Westinghouse fluence methodology (described in Reference 2) discusses radiation transport calculations to be performed with either the two-dimensional DORT code or the three-dimensional TORT code. This methodology has been approved by NRC per Reference 8. Both the DORT and TORT codes are described in Reference 3. Both codes suffer from limitations that often require analysts to make overly-conservative, simplifying assumptions. TORT is better at accounting for and tracking high energy particles axially above and below the fuel than 2D methods thus decreasing the need for assumptions and overly-conservative fluence values.

However, 3D radiation transport calculations are complex and computer resource intensive. The TORT code is only capable of performing its calculations on a single workstation, and it is therefore limited by the available memory and hard drive storage space on that workstation. This translates into limitations in the scope of the calculations that TORT can perform.

Westinghouse developed RAPTOR-M3G in order to overcome limitations associated with the TORT code. RAPTOR-M3G is a three-dimensional, parallel-processing discrete ordinates (S_N) radiation transport code that follows essentially the same calculational methodology as TORT. The parallel-processing feature of RAPTOR-M3G allows large, 3D radiation transport calculations to be divided across networks of workstations and solved simultaneously. This allows RAPTOR-M3G to perform calculations that would be prohibitively time consuming or impossible with TORT.

The methodology employed by RAPTOR-M3G is essentially the same as the methodology employed by the TORT code. The input format of RAPTOR-M3G is also compatible with TORT. The table below compares the TORT and RAPTOR-M3G codes:

Feature	TORT	RAPTOR-M3G
Solves the linear Boltzmann radiation transport equation in 3D	✓	✓
Applies the method of discrete ordinates (the S_N method) to treat the directional variable	✓	✓
Applies weighted finite-difference methods to treat spatial variables	✓	✓
Applies a multigroup formulation to treat energy dependence	✓	✓
DOORS Package (DORT/TORT) input format	✓	✓
Executes on a one-workstation platform	✓	✓
Executes simultaneously ("in parallel") on a network of workstations		✓

The radiation transport calculations in Reference 1 were performed with RAPTOR-M3G. Reference 4, which accompanied the delivery of Reference 1, describes extensive benchmark testing performed with RAPTOR-M3G, demonstrating very close agreement between calculated results of RAPTOR-M3G and TORT.

The NRC recently issued SRXB-RAI 8:

> *The RAPTOR-M3G code used to calculate fluence for MUR conditions does not appear to be approved by the NRC for use in this scenario. The NRC staff requests that the licensee provide justification for the use of RAPTOR-M3G for fluence calculations for MUR conditions, or provide an alternative fluence calculation using an NRC approved method.*

This document provides justification for the use of RAPTOR-M3G for the Catawba Unit 1 MUR power uprate application. Included in this justification is a comparison of fluence results obtained with RAPTOR-M3G to alternate calculations performed with the TORT code. In addition, the justification presented herein includes a discussion of how the existing RAPTOR-M3G calculations, as performed for the MUR power uprate, comply with Regulatory Guide 1.190 (Reference 5).

2 COMPARISON WITH ALTERNATE CALCULATIONS

2.1 OVERVIEW

The Catawba Unit 1 MUR power uprate application submittal evaluated neutron fluence for the following materials:

- Upper Shell to Intermediate Shell Circumferential Weld W06
- Intermediate Shell to Lower Shell Circumferential Weld W05
- Lower Shell to Bottom Head Ring Circumferential Weld W04

In order to evaluate these materials using the TORT code, reduced-size models were derived from the RAPTOR-M3G models used for the MUR power uprate application. The reduced-size models are solvable by the TORT code, whereas the original RAPTOR-M3G model is beyond the capabilities of TORT on the Westinghouse computing platform.

As part of this comparison, the following power distributions are analyzed with each reduced-size model:

- Cycle 3, representative of Out-In (High-Leakage) fuel design strategies
- Cycle 21, representative of Low-Leakage fuel design strategies
- A time-weighted average of power distributions through 54 Effective Full-Power Years (EFPY)

2.2 DEVELOPMENT OF REDUCED-SIZE MODELS

The reduced-size models are derived from the RAPTOR-M3G models described in Reference 1. An effort was made to preserve the full radial, azimuthal, and axial level of detail included in the original RAPTOR-M3G mesh structure. The reduced-size models are described in detail below:

- **Upper Reactor Environment (URE) model** – This model extends axially from the active fuel midplane to an elevation 160.6 cm above the top of the fuel. A reflective boundary condition is included at the fuel midplane to simulate the presence of the bottom half of the reactor core. The radial, azimuthal, and axial mesh boundaries are unchanged from the original RAPTOR-M3G model documented in Reference 1.

  The URE model contains 209 radial, 195 azimuthal, and 89 axial mesh intervals, and is plotted in Figure 2-1.

- **Midplane Reactor Environment (MRE) model** – This model extends axially from an elevation 8.3 cm below the bottom of the fuel to an elevation 7.4 cm above the top of the fuel. Void boundary conditions are included at the top and bottom of the problem. The radial and azimuthal mesh boundaries (and boundary conditions) are unchanged from the original RAPTOR-M3G model documented in Reference 1. It was necessary to remove axial mesh boundaries near the top

and bottom of this model in order to reduce the size of the problem, such that the problem would run successfully in TORT.

The MRE model contains 209 radial, 195 azimuthal, and 85 axial mesh intervals, and is plotted in Figure 2-2.

- **Lower Reactor Environment (LRE) model** – This model extends axially from an elevation 180.4 cm below the bottom of the fuel to the active fuel midplane. A reflective boundary condition is included at the fuel midplane to simulate the presence of the top half of the reactor core. The radial, azimuthal, and axial mesh boundaries are unchanged from the original RAPTOR-M3G model documented in Reference 1.

  The LRE model contains 209 radial, 195 azimuthal, and 91 axial mesh intervals, and is plotted in Figure 2-3.

All three reduced-size models employ the same radial and azimuthal mesh boundaries. The approximate numbers of radial intervals per inch in several regions of interest are summarized below:

- Peripheral fuel assemblies: 6 intervals per inch
- Bypass water: 6 intervals per inch
- Core barrel (stainless steel): 3.6 intervals per inch
- Downcomer water: 3.3 intervals per inch
- Reactor vessel (carbon steel): 1.7 intervals per inch

This mesh spacing complies with the guidance outlined in Reference 5.

The reduced-size models employ the same geometry and BUGLE-96 cross-section data that was used in Reference 1.

2.3 INPUT POWER DISTRIBUTIONS

Reference 1 performed fuel cycle-specific radiation transport calculations to determine the neutron fluence at materials of interest for Catawba Unit 1. This study considers a simplified subset of these calculations for the purpose of demonstrating the validity of the RAPTOR-M3G results.

To this end, the following power distributions are analyzed with each reduced-size model:

- Cycle 3, representative of Out-In (High-Leakage) fuel design strategies
- Cycle 21, representative of Low-Leakage fuel design strategies
- A time-weighted average of power distributions through 54 EFPY

The energy distributions considered for the Cycle 3 and Cycle 21 evaluations are based upon the same cycle-specific power distribution data described in Reference 1. The energy distribution is determined using the methodology described in Reference 2.

The time-weighted average power distribution was developed to be broadly representative of many different loading patterns employed over the life of Catawba Unit 1. In this representative cycle, the radial power distributions, axial power distributions, core power density, and the reactor coolant densities are all determined using time-weighted average over the life of Catawba Unit 1. This treatment is sufficient to demonstrate the validity of the projected fast neutron fluence at 54 EFPY in Reference 1.

2.4 DISCUSSION OF BOUNDARY CONDITIONS AND MODEL APPLICABILITY

Boundary conditions represent assumptions about the physics outside the boundaries of the model. This is true for reflective boundary conditions (used in the URE and LRE models), as well as void boundary conditions (used in the MRE model). The degree of uncertainty imparted by any boundary condition is highest near the boundaries themselves.

In the case of the URE and LRE models, a reflective boundary condition is placed at the axial midplane of the active fuel. The reflective boundary condition assumes that the power distribution is symmetric about the fuel midplane.

Figure 2-4 plots the axial power distributions averaged over the irradiation periods considered for this comparison. While none of the power distributions are exactly symmetric about the fuel midplane, the degree of asymmetry is small and does not affect the conclusions of the comparisons.

In the case of the MRE model, void boundary conditions are placed on the top and bottom boundary surfaces, immediately outside the active fuel region. This imparts significant uncertainties to calculated results near the top and bottom of the core, and for this reason, the use of the MRE model is confined to materials located within 75 cm of the fuel midplane.

A summary of the axial boundary conditions used for the reduced-size models appears in Table 2-1, along with a listing of the reactor vessel materials evaluated using each reduced-size model.

2.5 SOLUTION PARAMETERS

The neutron fluence calculations in Reference 1 were performed using a directional theta-weighted (DTW) spatial differencing scheme combined with a level-symmetric S_8 quadrature set. The DTW spatial differencing scheme is discussed and endorsed in Regulatory Guide 1.190. (See Reference 34 of Regulatory Guide 1.190 for more information about the DTW scheme.) The DTW scheme generally produces improved results, as compared to traditional theta-weighted (TW) schemes. The TORT code does not include the DTW scheme, so the reduced-size models discussed in this document are analyzed with a traditional, TW scheme in both TORT and RAPTOR-M3G. Also, the RAPTOR-M3G models in Reference 1 have been rerun with TW scheme.

The reduced-size models analyzed in RAPTOR-M3G and TORT are nearly identical. The mesh structures, cross-section data, and core source data are all completely identical. The solution parameters are almost identical, with one minor difference relating to the "initiating directions" in the directional quadrature sets. See Page 3-33 of the TORT manual in Reference 3 for a discussion of this difference. RAPTOR-M3G uses the technique of Lathrop and Brinkley, which has been established to provide good results. The results from rerunning RAPTOR-M3G models in Reference 1 with TW scheme are also identical to the reduced-size models analyzed in RAPTOR-M3G and TORT, which demonstrates that the reduced-size models have no impact on the results. The conservatism for the most limiting material (upper circ weld 06) is due to RAPTOR-M3G DTW scheme.

2.6 RESULTS

Calculated neutron (E > 1.0 MeV) fluence rate and fluence results, comparing RAPTOR-M3G and TORT, are shown in Table 2-2 through Table 2-4. In all cases, the Reference 1 results are in good agreement with the reduced-size models described previously. Further, the results obtained from the RAPTOR-M3G and TORT reduced-size models provide clear demonstration that their methods of evaluation are essentially the same.

Table 2-1 Boundary Conditions and Extent of Applicability for the Reduced-Size Models

Parameter	Reduced-Size Model		
	URE	MRE	LRE
Bottom of Model[1]	0.0 cm	-191.206 cm	-363.296 cm
Bottom Boundary Condition	Reflective	Void	Void
Top of Model[1]	343.46 cm	190.289 cm	0.0 cm
Top Boundary Condition	Void	Void	Reflective
Bottom Extent of Model Applicability[1]	75.0 cm	-75.0 cm	-330.0 cm
Top Extent of Model Applicability[1]	300.0 cm	75.0 cm	-75.0 cm
Materials Analyzed in Model	Weld W06	Weld W05	Weld W04

Note:

1. Dimensions are given relative to the active fuel midplane.

Table 2-2 Calculated Neutron Fluence Rates for Catawba Unit 1 Cycle 3

Model	Calculated Neutron (E > 1.0 MeV) Fluence Rate [n/cm^2-s]		
	Weld W06	**Weld W05**	**Weld W04**
RAPTOR-M3G Model in Reference 1	1.14E+09	2.33E+10	1.98E+09
RAPTOR-M3G Model in Reference 1 with TW	1.06E+09	2.36E+10	1.90E+09
Reduced-Size Models (RAPTOR-M3G) with TW	1.06E+09	2.36E+10	1.90E+09
Reduced-Size Models (TORT) with TW	1.06E+09	2.36E+10	1.91E+09

Table 2-3 Calculated Neutron Fluence Rates for Catawba Unit 1 Cycle 21

Model	Calculated Neutron (E > 1.0 MeV) Fluence Rate [n/cm^2-s]		
	Weld W06	**Weld W05**	**Weld W04**
RAPTOR-M3G Model in Reference 1	6.98E+08	1.54E+10	1.26E+09
RAPTOR-M3G Model in Reference 1 with TW	6.40E+08	1.54E+10	1.20E+09
Reduced-Size Models (RAPTOR-M3G) with TW	6.40E+08	1.54E+10	1.20E+09
Reduced-Size Models (TORT) with TW	6.41E+08	1.54E+10	1.20E+09

Table 2-4 Calculated Neutron Fluence Following 54 EFPY at Catawba Unit 1 (Reduced-Size Models Calculated Using the Time-Weighted Average Power Distributions)

Model	Calculated Neutron (E > 1.0 MeV) Fluence [n/cm^2]		
	Weld W06	**Weld W05**	**Weld W04**
RAPTOR-M3G Model in Reference 1	1.16E+18	2.60E+19	1.95E+18
RAPTOR-M3G Model in Reference 1 with TW	1.05E+18 (1.07E+18)*	2.66E+19 (2.63E+19)*	1.83E+18 (1.86E+18)*
Reduced-Size Models (RAPTOR-M3G) with TW	1.05E+18	2.66E+19	1.83E+18
Reduced-Size Models (TORT) with TW	1.05E+18	2.66E+19	1.83E+18

* The projected 54 EFPY fluence value in the parenthesis is calculated by accumulating cycle-specific fluence for Cycles 1 through 22, and assuming Cycle 22 at MUR power for cycles beyond Cycle 22, the same approach used in Reference 1.

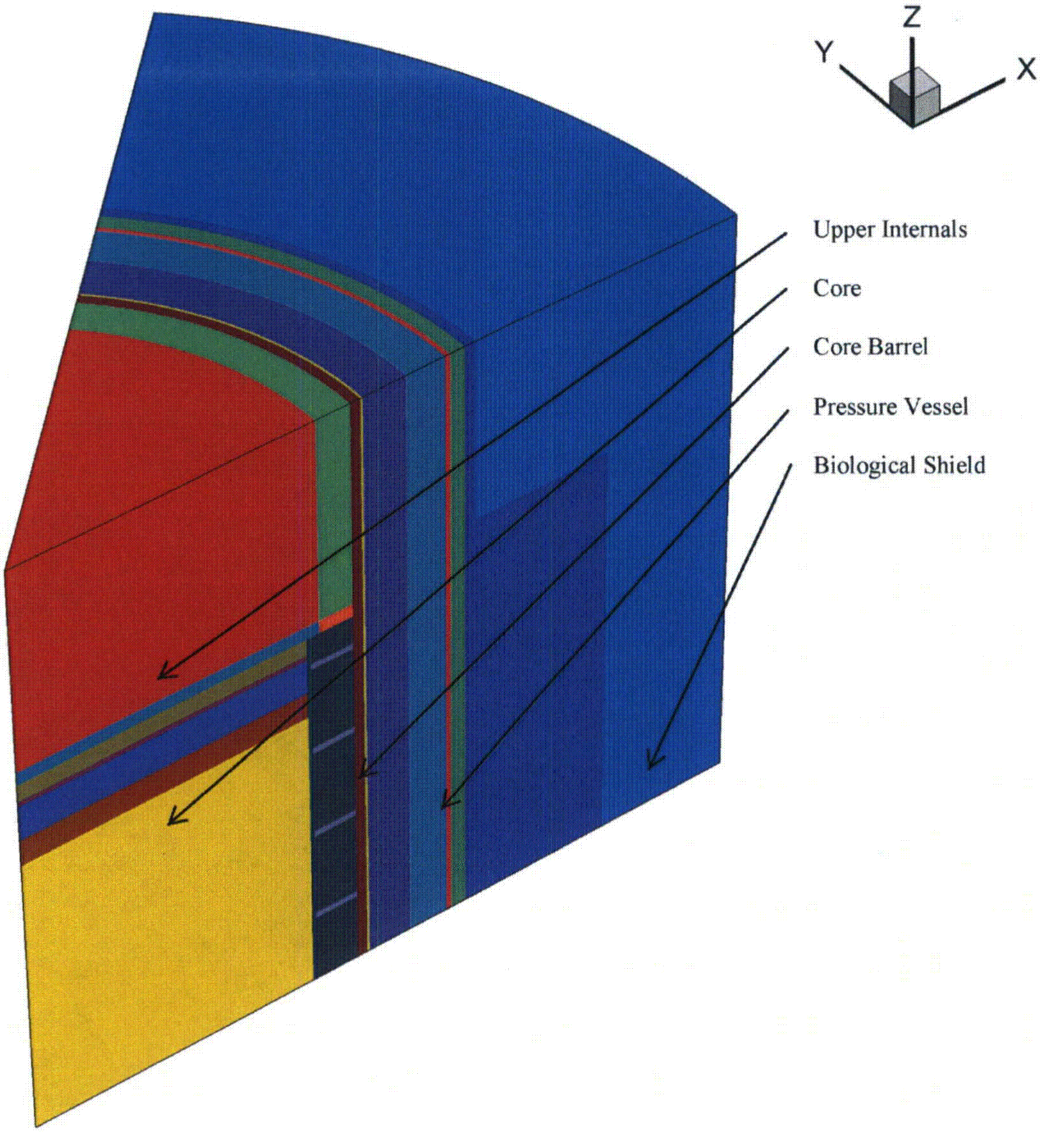

Figure 2-1 Catawba Unit 1 URE Reduced-Size Model

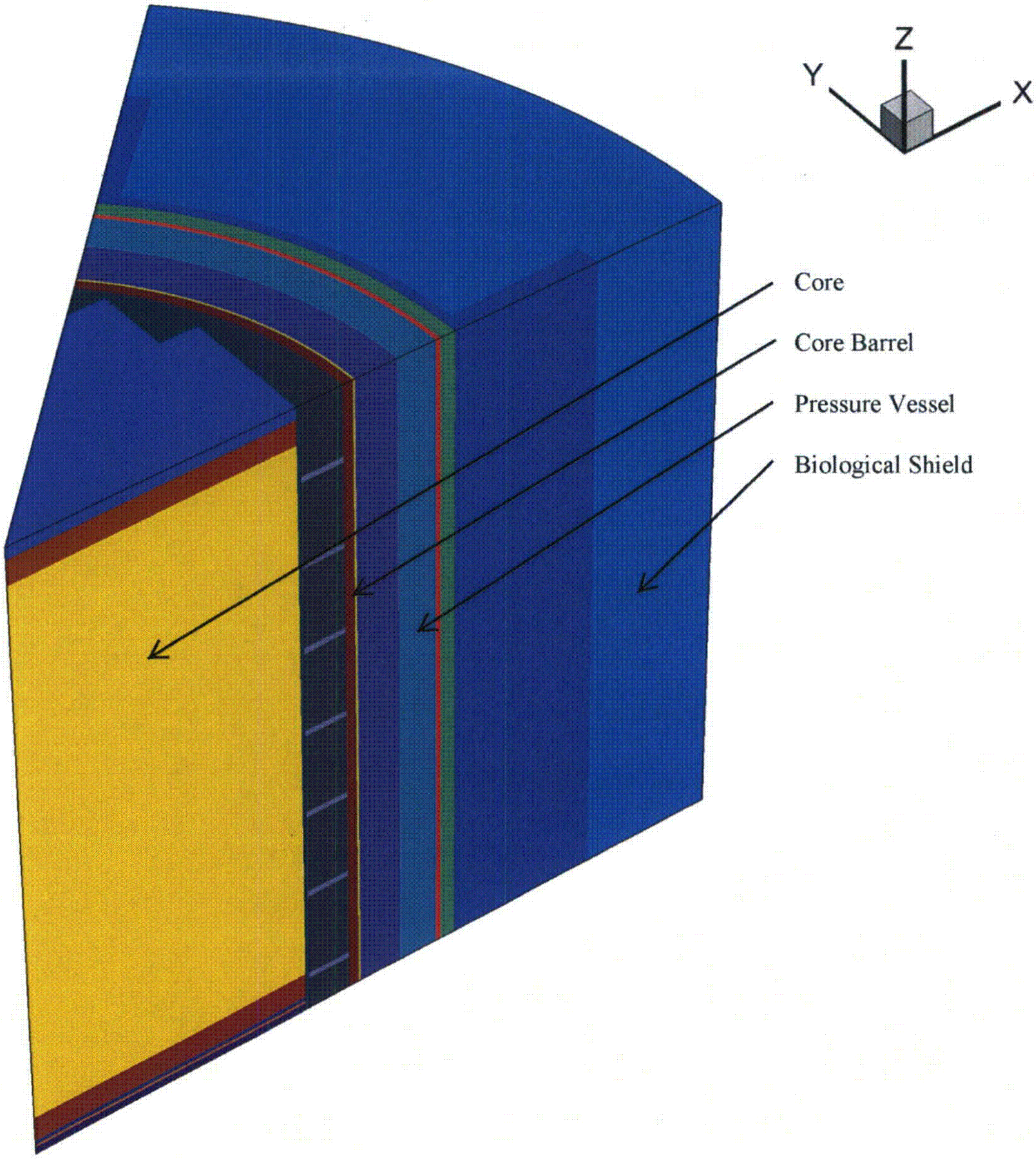

Figure 2-2 Catawba Unit 1 MRE Reduced-Size Model

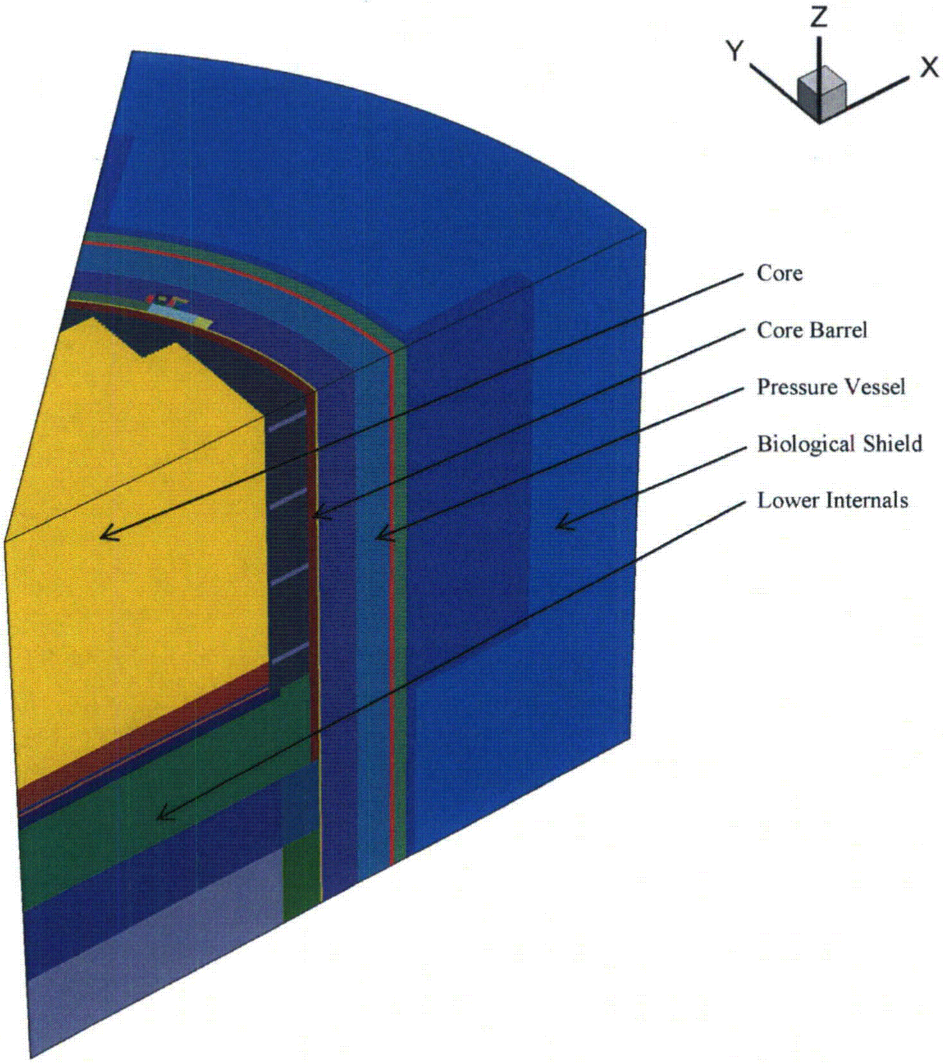

Figure 2-3 Catawba Unit 1 LRE Reduced-Size Model

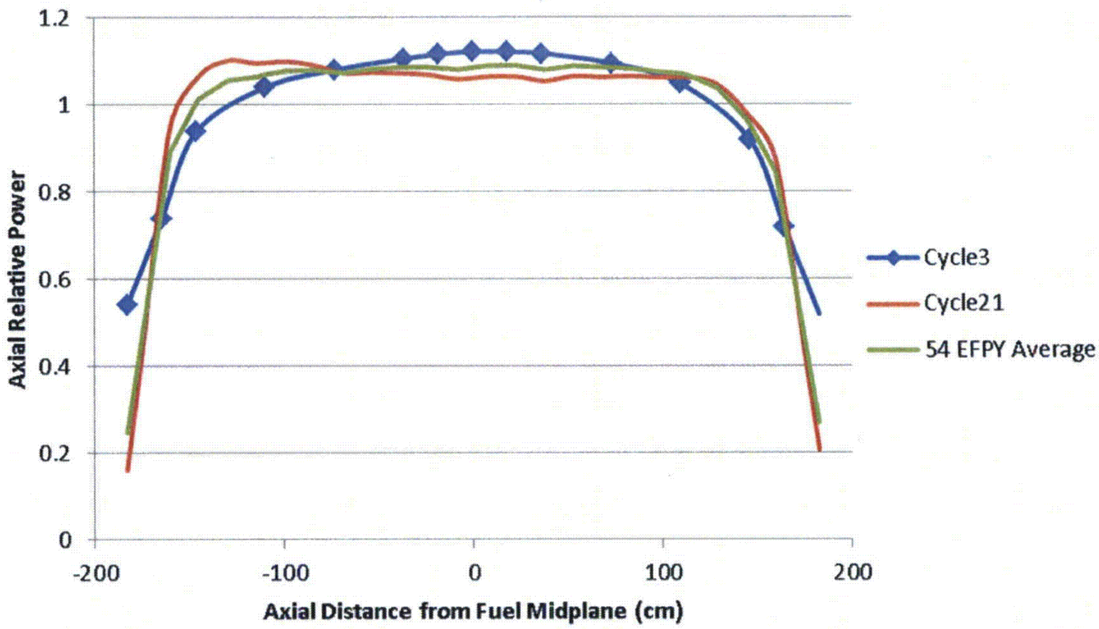

Figure 2-4 Relative Axial Power Distributions for the RAPTOR-M3G and TORT Comparisons

3 COMPLIANCE WITH REGULATORY GUIDE 1.190

The NRC has determined that the methodology outlined in Reference 2 adheres to Regulatory Guide 1.190. The Catawba Unit 1 MUR power uprate application used the methodology documented in Reference 2 for neutron fluence determination. In this application, the RAPTOR-M3G code was used to perform radiation transport calculations instead of the TORT code.

In Regulatory Guide 1.187 (Reference 6), the NRC has determined that NEI 96-07 (Reference 7) provides acceptable methods for complying with the provisions of 10 CFR 50.59. Reference 7 allows licensees to change one or more methods of evaluation without prior NRC approval, provided the results are "essentially the same". The criterion given for "essentially the same" is that the new results must be "within the margin of error for the type of analysis being performed."

Regulatory Guide 1.190 states that a vessel fluence uncertainty of 20% (1σ) is acceptable for RT_{PTS} and RT_{NDT} determination. Reference 2 establishes a net calculational uncertainty at the 1σ level, obtained by combining four uncertainty components in quadrature:

PCA Benchmark Comparisons	3%
H. B. Robinson Benchmark Comparisons	3%
Analytical Sensitivity Studies	11%
Other Factors	5%
Reference 2 Net Calculational Uncertainty	**13%**

Section 2 of this document performs Catawba Unit 1-specific comparisons of the RAPTOR-M3G and TORT codes. After reviewing the results of these comparisons, the following conclusions can be drawn:

- When the RAPTOR-M3G and TORT codes are run with matching solution parameters (i.e., when comparing the reduced-size model results), the codes demonstrate agreement within a threshold of 1% when comparing values of neutron (E > 1.0 MeV) fluence.
- The results obtained with RAPTOR-M3G that are reported in Reference 1 are consistent with results obtained using TORT. The observed differences are due to the use of the DTW differencing scheme in RAPTOR-M3G versus the TW differencing scheme option in TORT. The neutron (E > 1.0 MeV) fluence results generated by TORT in this study show relative differences that range from –9% to +1% as compared to the RAPTOR-M3G results reported in Reference 1. That is: the TORT-generated values are generally less-conservative than the RAPTOR-M3G-generated values. Further, the only RAPTOR-M3G result reported in Reference 1 that is less conservative than TORT result is at Weld W05, which is not the limiting material for Catawba Unit 1 reactor pressure vessel.
- The results from RAPTOR-M3G and TORT agree better than the 13% uncertainty assigned to the calculational methodology and well within the 20% uncertainty deemed acceptable for RT_{PTS} and RT_{NDT} determination.

Therefore, methodology for neutron fluence determination used for the Catawba Unit 1 MUR power uprate application complies with Regulatory Guide 1.190.

4 REFERENCES

1. Westinghouse Report WCAP-17669-NP, Rev. 0, "Catawba Unit 1 Measurement Uncertainty Recapture (MUR) Power Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations," June 2013.

2. Westinghouse Report WCAP-16083-NP-A, Rev. 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," May 2006.

3. RSICC Computer Code Collection, CCC-650, "DOORS 3.2: One-, Two-, and Three Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Oak Ridge National Laboratory, Oak Ridge, Tennessee, April 1998.

4. Westinghouse Report WCAP-16083-NP, Rev. 1, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," April 2013.

5. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

6. Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Test, and Experiments," U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, November 2000.

7. NEI Report NEI 96-07, Rev. 1, "Guidelines for 10 CFR 50.59 Evaluations," Nuclear Energy Institute, February 2000.

8. NRC Agencywide Documents Access and Management System (ADAMS) ML053550466, "Final Safety Evaluation for Westinghouse Owners Group Topical Report WCAP-16083-NP, Revision 0, "Benchmark Testing of the Ferret Code for Least Squares Evaluation of Light Water Reactor Dosimetry" (TAC No. MC3974)," January 2006.

9. Gulf General Atomic Report GA-8747, "TWOTRAN, a FORTAN Program for Two Dimensional Transport," July 1968.

10. WANL-PR-(LL)-034, "Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation. Vol. 5 – Two-Dimensional Discrete Ordinates Transport Technique," August 1970.