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Complete only applicable items.

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Table of Contents

1. PURPOSE

The purpose of this analysis is to document the Commercial Reactor Critical (CRC) benchmark evaluation performed for the Quad Cities Unit 1 boiling water reactor (BWR). The CRC benchmark is performed at a beginning of life (BOL) statepoint representing reactor start-up critical conditions. The objective of this CRC benchmark analysis is to provide a validation benchmark for the MCNP 4A analytic tool for use in the disposal criticality analysis methodology.

2. QUALITY ASSURANCE

The Quality Assurance (QA) program applies to this analysis. The work reported in this document is part of the criticality disposal methodology development that will eventually support the License Application Design phase. This activity, when appropriately confirmed, can impact the proper functioning of the Mined Geologic Disposal System (MGDS) waste package; the waste package has been identified as an MGDS *Q-List* item important to safety and waste isolation (Reference 5.2, pp. 4, 15). The waste package is on the *Q-List* by direct inclusion by the Department of Energy (DOE), without conducting a QAP-2-3 evaluation. The Waste Package Development Department (WPDD) responsible manager has evaluated this activity in accordance with QAP-2-0, *Conduct of Activities*. The *Perform Criticality, Thermal, Structural, and Shielding Analyses* evaluation (Reference 5.1) has determined that the preparation and review of this design analysis is subject to the *Quality Assurance Requirements and Description* (QARD; Reference 5.3) requirements. As specified in NLP-3-18, *Documentation of QA Controls On Drawings, Specifications, Design Analyses, and Technical Documents*, the development of this analysis is subject to QA controls.

The analysis described in this document supports development of the disposal criticality analysis methodology. No designs were analyzed in this document. This document will not directly support any construction, fabrication, or procurement activity and therefore is not required to be procedurally controlled as TBV (to be verified). The calculation design inputs or information used in this document come from technical documents and reports that are accepted by the scientific and engineering community as established fact. The specific references are listed in Section 5. The information is therefore not treated as unqualified data.

3. METHOD

The CRC reactivity calculations documented in this analysis are performed with the MCNP 4A code system (Reference 5.7). The MCNP 4A code system utilizes a Monte-Carlo calculational technique to perform particle transport calculations. MCNP 4A is capable of simulating particle transport and associated physics within complex three-dimensional geometries. MCNP 4A uses pseudo-continuous-energy cross section libraries for each element or isotope to determine necessary parameters at particle interaction sites during transport calculations. The effective multiplication factor (k_{eff}) resulting from the CRC benchmark reactivity calculation may be used to validate the MCNP 4A code system for predicting reactivity worth in similar configurations.

4. DESIGN INPUTS

This analysis does not contain or utilize input data for any repository component design. Rather, the calculation documented in this analysis utilizes CRC benchmark input data. The CRC benchmark is used to validate the MCNP 4A code as an analytic tool for use in subsequent repository component design analyses. The CRC benchmark reactivity calculation input data utilized in this analysis is listed in Section 4.1.

4.1 Design Parameters

The CRC benchmark reactivity calculation input parameters utilized in this analysis are listed in this section. All of the input parameters listed in this section were obtained from References 5.11 and 5.12. Tables 4.1-1 through 4.1-13 list all of the design input parameters used in this reactivity calculation. The location page numbers for input parameters obtained from these references are provided with each listed input table.

Table 4.1-2 shows effective axial dimensions of the Quad Cities Unit 1 core that are used in this calculation. These effective axial dimensions are determined from actual core structures that are selected for inclusion in homogenized regions for the 1/4 core model. Top axial section I includes the assembly upper tie plate and the upper core grid as a homogenized region. Top axial section II is a region above the core that consists of moderator. Bottom axial section I includes the assembly bottom tie plate and core support plate homogenized region. Bottom axial section II consists of a 50 weight percent Type 304 stainless steel and 50 weight percent water homogenized region that accounts for fuel support casting region of the lower core. The bottom axial section III consists of a moderator region. Additional explanation for the axial regions is provided in Section 7.2.1.

Information for assembly identification, core loading pattern, and control blade pattern as provided in Reference 5.11, pp. A-9, -10, -11, -12, -13, C-35 is shown in Section 7.2.2, Figure 7.2.2-4. Fresh fuel material compositions provided in Reference 5.11, pp. A-1, -3, -4, -5 are shown in Section 7.3.4, Table 7.3.4-1. Fuel material compositions for rods incorporating integral burnable absorber, listed in Reference 5.11, pp. A-1, -3, -4, -5, are shown in Section 7.3.4, Table 7.3.4-2.

Table 4.1-1. Quad Cities Unit 1 Radial Dimensions

Description	Thickness (cm)	Outer Dimension (cm)
Core Center	---	0.0
Core Shroud	5.08	263.048
Pump Region Water	55.721	318.77
Pressure Vessel Wall	15.557	334.327

(Reference 5.11, p. A-9; Reference 5.12, pp. 3.6.5, 4.1-1)

Table 4.1-2. Quad Cities Unit 1 Effective Axial Dimensions

Description	Height (cm)
Top Axial Section I	54.20
Top Axial Section II	45.72
Bottom Axial Section I	27.33
Bottom Axial Section II	28.55
Bottom Axial Section III	45.72
Active Fuel Height	365.76

(Reference 5.11, pp. C-10, -26, -27)

Table 4.1-3. Fuel Rod Types

Rod Type	Fuel Rod Description
1	2.47 wt% U-235, undished
1d[a]	2.47 wt% U-235, dished
1s[s]	2.47 wt% U-235, undished
2	1.70 wt% U-235, undished
2d	1.70 wt% U-235, dished
3	1.20 wt% U-235, dished
3d	1.20 wt% U-235, dished
4	2.47 wt% U-235, 3.00 wt% Gd_2O_3, undished
5	2.47 wt% U-235, 0.50 wt% Gd_2O_3, undished

[a] d-dished pellet
[s] s-spacer capture rod
(Reference 5.11, pp. A-3, -4, -5)

Table 4.1-4. Fuel and Clad Dimensions

Rod Type	Pellet Outer Diameter (cm)	Clad Outer Diameter (cm)	Clad Thickness (cm)
dished, fuel only	1.23952	1.43002	0.08128
dished, w/gadolinia	1.23698	1.43002	0.08128
undished, fuel only	1.23952	1.43002	0.08128
undished, w/gadolinia	1.23698	1.43002	0.08128

(Reference 5.11, pp. A-3, -4, -5; C-1, -2, -3)

Table 4.1-5. Control Blade Dimensions

Parameter	Value
Blade Pitch	30.48 cm
Stroke	365.76 cm
Poison Length	363.22 cm
Absorber Tubes per Wing	21
Wing Span	12.38 cm
Wing Thickness	0.793 cm
Sheath Thickness	0.142 cm
Half-thickness of Tie Rod	1.985 cm

(Reference 5.11, pp. A-9, C-15)

Table 4.1-6. Upper and Lower Region Structural Dimensions

Parameter	Value (cm)
Upper Top Guide Beam Separation	30.48
Upper Top Guide Beam Thickness	0.914
Core Support Plate Thickness	5.08

(Reference 5.11, pp. C-26, -27)

Table 4.1-7. Assembly Component Masses

Component	Quantity	Mass (g)
Zircaloy-4 Spacers	7	1704.15
Zircaloy-2 End Plugs	98	1617.06
Type 304 stainless steel Lower Tie Plate	1	4360.84
Type 304 stainless steel Upper Tie Plate	1	2047.52
Type 304 stainless steel Plenum Spring	49	1543.12
Type 304 stainless steel Plenum Getter	49	440.89

(Reference 5.11, p. A-8)

Table 4.1-8. Channel Dimensions and Material

Parameter	Value
Outside Dimensions	13.81 cm
Height	411.88 cm
Thickness	0.2032 cm
Material	Zircaloy-4

(Reference 5.11, pp. A-1, C-14)

Table 4.1-9. Core Layout Dimensions

Parameter	Value
Assembly Pitch	15.24 cm
Control Blade Pitch	30.48 cm
Fuel Rod Pitch	1.875 cm

(Reference 5.11, p. A-3)

Table 4.1-10. Type 304 Stainless Steel Composition

Ele./Iso. [a]	MCNP ZAID	Wt. % [b]	Ele./Iso.	MCNP ZAID	Wt. %
C-nat	6000.50c	0.080	Fe-54	26054.60c	3.918
N-14	7014.50c	0.100	Fe-56	26056.60c	63.156
Si-nat	14000.50c	0.750	Fe-57	26057.60c	1.472
P-31	15031.50c	0.045	Fe-58	26058.60c	0.200
S-nat	16032.50c	0.030	Ni-58	28058.60c	6.234
Cr-50	24050.60c	0.793	Ni-60	28060.60c	2.465
Cr-52	24052.60c	15.903	Ni-61	28061.60c	0.109
Cr-53	24053.60c	1.838	Ni-62	28062.60c	0.350
Cr-54	24054.60c	0.466	Ni-64	28064.60c	0.092
Mn-55	25055.50c	2.000	Density = 7.9 g/cc		

[a] element or isotope
[b] weight percentage
(Reference 5.10, p. 12)

Table 4.1-11. Zircaloy-2 Composition

Ele./Iso.	MCNP ZAID	Wt. %	Ele./Iso.	MCNP ZAID	Wt. %
Cr-50	24050.60c	0.004	Ni-60	28060.60c	0.013
Cr-52	24052.60c	0.084	Ni-61	28061.60c	0.001
Cr-53	24053.60c	0.010	Ni-62	28062.60c	0.002
Cr-54	24054.60c	0.002	Ni-64	28064.60c	0.0005
Fe-54	26054.60c	0.006	O-16	8016.50c	0.120
Fe-56	26056.60c	0.092	Zr-nat	40000.60c	98.23
Fe-57	26057.60c	0.002	Sn-nat	50000.35c	1.400
Fe-58	26058.60c	0.0003	Density = 6.56 g/cc		
Ni-58	28058.60c	0.034			

(Reference 5.10, p. 21)

Table 4.1-12. Zircaloy-4 Composition

Ele./Iso.	MCNP ZAID	Wt. %	Ele./Iso.	MCNP ZAID	Wt. %
Cr-50	24050.60c	0.004	Fe-57	26057.60c	0.004
Cr-52	24052.60c	0.084	Fe-58	26058.60c	0.001
Cr-53	24053.60c	0.010	O-16	8016.50c	0.120
Cr-54	24054.60c	0.002	Zr-nat	40000.60c	98.180
Fe-54	26054.60c	0.011	Sn-nat	50000.35c	1.400
Fe-56	26056.60c	0.184	Density = 6.56 g/cc		

(Reference 5.10, p. 21)

Table 4.1-13. Fuel Rod Density And Mass Loading

Rod Type	Stack Density (g/cc)	Mass UO_2 Per Rod (g)	Mass Gd_2O_3 Per Rod (g)
1	10.34	4566	0
1d	9.94	4377	0
1s	10.34	4186	0
2	10.34	4566	0
2d	9.94	4377	0
3	10.34	4566	0
3d	9.94	4377	0
4	10.26	4380	130
5	10.34	4550	9

(Reference 5.11, pp. A-3, -4, -5)

4.2 Criteria

The design of the waste package will depend on waste package configuration criticality analyses performed, using an acceptable disposal criticality analysis methodology. Criteria that relate to the development and design of repository and engineered barrier components are derived from the applicable requirements and planning documents. The Engineered Barrier Design Requirements Document (EBDRD, Reference 5.5) provides requirements for engineered barrier segment design. The Repository Design Requirements Document (RDRD, Reference 5.6) provides requirements for repository design. The Controlled Design Assumptions Document (Reference 5.14) provides guidance for requirements listed in the EBDRD and RDRD which have unqualified or unconfirmed data associated with the requirement.

This analysis supports the disposal criticality analysis methodology by providing input, in the form of effective multiplication factor results, to benchmark calculations which address the prediction of both spent fuel isotopic compositions and their associated reactivity. These benchmark calculations will contribute to the determination of bias values in the method of critical multiplication factor calculation that is implemented by the analytic tools to be used in the disposal criticality methodology. The requirements for utilizing the bias in the method of calculation of the critical multiplication factor for disposal configurations containing spent nuclear fuel are located in Section 3.2.2.5 of the RDRD and Section 3.2.2.6 of the EBDRD. This analysis does not satisfy these requirements, but the results from this analysis will be used as input to subsequent analyses which will satisfy these requirements.

4.3 Assumptions

4.3.1 The moderator density at the time of the cold critical calculation is unknown and assumed to be 0.9813 g/cc. The basis for this assumption is that this is the calculated density using steam tables and a reactor system pressure of 1 atmosphere and moderator temperature of 147°F. This assumption is used in Sections 7.1, 7.2 and 7.3.

4.3.2 The specific control blade axial positions for the cold critical condition were assumed to be both fully inserted into the core and fully withdrawn from the core (Reference 5.11, p. C-35). This control blade configuration is indicated in Figure 7.2.2-4. The basis for this assumption is the control blade configuration data that is provided in Reference 5.11. This assumption is used in Section 7.2.

4.4 Codes and Standards

Not applicable.

5. REFERENCES

5.1 *QAP-2-0 Activity Evaluations.* ID #WP-20 *Perform Criticality, Thermal, Structural, & Shielding Analyses*, Civilian Radioactive Waste Management System (CRWMS) Management and Operating Contractor (M&O), August 3, 1997.

5.2 *Yucca Mountain Site Characterization Project Q-List.* YMP/90-55Q, REV 04, Yucca Mountain Site Characterization Project (YMP).

5.3 *Quality Assurance Requirements and Description.* DOE/RW-0333P REV 07, U.S. Department of Energy (DOE), Office of Civilian Radioactive Waste Management (OCRWM).

5.4 *Introduction To Nuclear Engineering.* J. R. Lamarsh, Polytechnic Institute of New York, 2nd Ed., ISBN 0-201-14200-7, 1983.

5.5 *Engineered Barrier Design Requirements Document.* YMP/CM-0024, REV 0, ICN 1, U. S. DOE OCRWM.

5.6 *Repository Design Requirements Document.* YMP/CM-0023, REV 0, ICN 1, U. S. DOE OCRWM.

5.7 *MCNP 4A: Monte Carlo N-Particle Transport Code System.* User's Manual. Los Alamos National Laboratory, Los Alamos, NM. Distributed by Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, TN. Document Number: CCC-200.

5.8 *Software Qualification Report for MCNP 4A, A General Monte Carlo N-Particle Transport Code.* DI#: 30006-2003 REV 02, CRWMS M&O.

5.9 *Nuclide and Isotopes, Chart of the Nuclides, Fourteenth Edition.* General Electric Company, 1989.

5.10 *Material Compositions and Number Densities for Neutronics Calculations.* DI#: B00000000-01717-0200-00002 REV 00, CRWMS M&O.

5.11 *Core Design And Operating Data For Cycles 1 & 2 for Quad Cities 1.* EPRI NP-240, Project 497-1, Topical Report, 1976.

5.12 *Quad Cities Station, Units 1 And 2, Final Safety Analysis Report.* Vol. 1, DOCKET-50254-18, Amendment No. 11, Commonwealth Edison Co., Chicago, IL, 1970.

5.13 *Scale-4 Analysis of Pressurized Water Reactor Critical Configurations: Volume 2-Sequoyah Unit 2 Cycle 3.* S.M. Bowman, O.W. Hermann, and M.C. Brady, Oak Ridge National Laboratory, Oak Ridge, TN, ORNL/TM-12294/V2.

5.14 *Controlled Design Assumptions Document*. DI#: B00000000-01717-4600-00032 REV 04, ICN 01, CRWMS M&O.

5.15 Attachment for B00000000-01717-0200-00146 REV 00 - MCNP CRC Reactivity Calculation for Quad Cities BWR. Batch Number: MOY-970828-03.

6. USE OF COMPUTER SOFTWARE

A. The MCNP 4A code system was used in this analysis to calculate the effective multiplication factor (k_{eff}) of CRC benchmarks. The MCNP 4A code system is subject to the requirements of the QARD (Reference 5.3). The MCNP 4A code system was obtained from Software Configuration Management in accordance with appropriate procedures. The CSCI number for MCNP 4A is 30006 V4A. The MI number for MCNP 4A is 30006-M03-002. The MCNP 4A calculations documented in this analysis were performed on Hewlett Packard (HP) 9000 series workstations. The MCNP 4A code system utilizes a Monte-Carlo calculational technique to perform particle transport calculations. MCNP 4A is capable of simulating particle transport and associated physics within complex three dimensional geometries. MCNP 4A uses pseudo-continuous-energy cross-section libraries for each element or isotope to determine necessary parameters at particle interaction sites during transport calculations. Section 7.3.1 contains a detailed listing of all MCNP 4A cross-section libraries utilized in the CRC reactivity calculations documented in this analysis. A detailed description of the MCNP 4A code system is provided Reference 5.7. The MCNP 4A code system is applicable to the engineering application within this analysis and is used within the range of verification and validation as documented in Reference 5.8.

B. The Excel, Version 7.0 spreadsheet package is utilized in this analysis. The user-defined formulas, inputs, and results for all calculations performed with this spreadsheet package are documented, where applicable, throughout this analysis.

7. DESIGN ANALYSIS

The Quad Cities Unit 1 CRC benchmark reactivity calculation is performed to contribute to the validation of MCNP 4A for use in repository component design (specifically waste package design). The MCNP 4A CRC reactivity calculation is a detailed critical benchmark calculation which represents the actual reactor configuration. This analysis provides the geometry, material, core loading, and calculational control descriptions for the CRC benchmark calculation performed with MCNP 4A. The k_{eff} result for the CRC benchmark is also presented and discussed.

7.1 CRC Benchmark Reactivity Calculations

This analysis documents the CRC benchmark MCNP 4A reactivity calculation for Cycle 1 startup of Quad Cities Unit 1. Quad Cities Unit 1 is a Type 3 BWR with core lattice Type D. This reactivity calculation models the BOL cold critical calculation performed on the reactor on the date April 5, 1972.

The cold critical temperature was 337 K (147°F) with a period of 230 seconds (Reference 5.11, p. C-35). The reactor system pressure is assumed to be 1 atmosphere.

The approximated reactivity effect of a positive or negative period of 230 seconds is 4.0E-4 (Reference 5.4, p. 285). This positive or negative change in the measured k_{eff} of the core falls within the 2σ statistical uncertainty (95% confidence level) of the k_{eff} result calculated in this CRC analysis.

7.2 CRC Benchmark MCNP 4A Geometrical Descriptions

The MCNP 4A CRC benchmark model for the Quad Cities Unit 1 BWR incorporates detailed and explicit representations of the fuel assemblies and reactor core components in 3-dimensions (3-D). The MCNP 4A CRC benchmark model takes advantage of the core symmetry found in the startup fresh fuel configuration and uses a 1/4 core model of the actual reactor. Fuel assembly and core structure modeling is incorporated for regions beyond the extent of the active fuel in the axial direction to ensure that neutron leakage is correctly simulated. Actual core loading patterns are utilized in the models. Reactor control blade patterns are accurately modeled as well as the details of the actual control blade. Sections 7.2.1 through 7.2.5 discuss the MCNP 4A 3-D modeling details for the various components of the Quad Cities Unit 1 CRC configurations. Sketches of modeled reactor components are provided throughout this document to aid in explanation of modeling technique. The sketches provided are not to scale and are for illustrative purposes only.

7.2.1 Quad Cities Unit 1 Core Description

The Quad Cities Unit 1 BWR is a General Electric reactor core design consisting of 724 fuel assemblies, 177 control blades, 41 in-core instrument assemblies, and 7 in-core neutron sources. The 1/4 core MCNP 4A core model consists of 181 assemblies, 15 in-core instrument tubes, 28 complete control blades, and 11 partial control blades. A core shroud surrounds the periphery fuel assemblies in the core. The periphery of the reactor consists of the core shroud, the pump region, and the pressure vessel. Each of these components are separated by a region of water. A radial view of the modeled reactor internals is shown in Figure 7.2.1-1. The height of the active fuel region in the core is 365.76 cm. The assembly pitch in the core is 15.24 cm. The core lattice model is filled with control cell quadrant regions that contain the correctly oriented assembly, channel, moderator, and control blade portions. Table 4.1-1 provides modeled dimensions from the center of the core to the outside surface of the pressure vessel as shown in Figure 7.2.1-1.

An axial view of the modeled reactor core internals is shown in Figure 7.2.1-2. The axial reactor dimensions are provided in Table 4.1-2. The reactor structure above the upper end of the active fuel region in the fuel assemblies is divided into two regions: 1) the upper assembly plenum region, tie plate, and upper core grid; and 2) the open space above the upper core grid. The reactor structure below the lower end of the active fuel is divided into three regions: 1) the lower connection of the assembly to the lower core tie plate; 2) the fuel support casting region; and 3) the region between the fuel support casting and the core plate. Homogenized material weight percentages are calculated for the regions directly above and below the active fuel region. The technique for determining these material weight percentages is to first determine the different material volume fractions by calculating the total volume for the 1/4 core model in each specific region (based on the dimensions given in Table 4.1-2) and then, approximating the volume of the core structure in that region using structural dimensions and/or given masses and densities (Reference 5.11, p. A-8). These volume fractions are then used to determine the material weight percentages for all materials found in that region. Additionally, the second lower region below the active fuel is calculated using 50 weight percent Type 304 stainless steel and 50 weight percent water. The regions above and below the fuel assemblies in the reactor core have a limited effect on system reactivity. However, the regions closest to the active fuel are modeled with appropriately homogenized material composition using specific material volume fractions to evaluate leakage characteristics.

The remaining axial regions are modeled with moderator where appropriate. In the radial direction the core water regions on both sides of the shroud are modeled with moderator. The pressure vessel wall composition is modeled as Type 304 stainless steel.

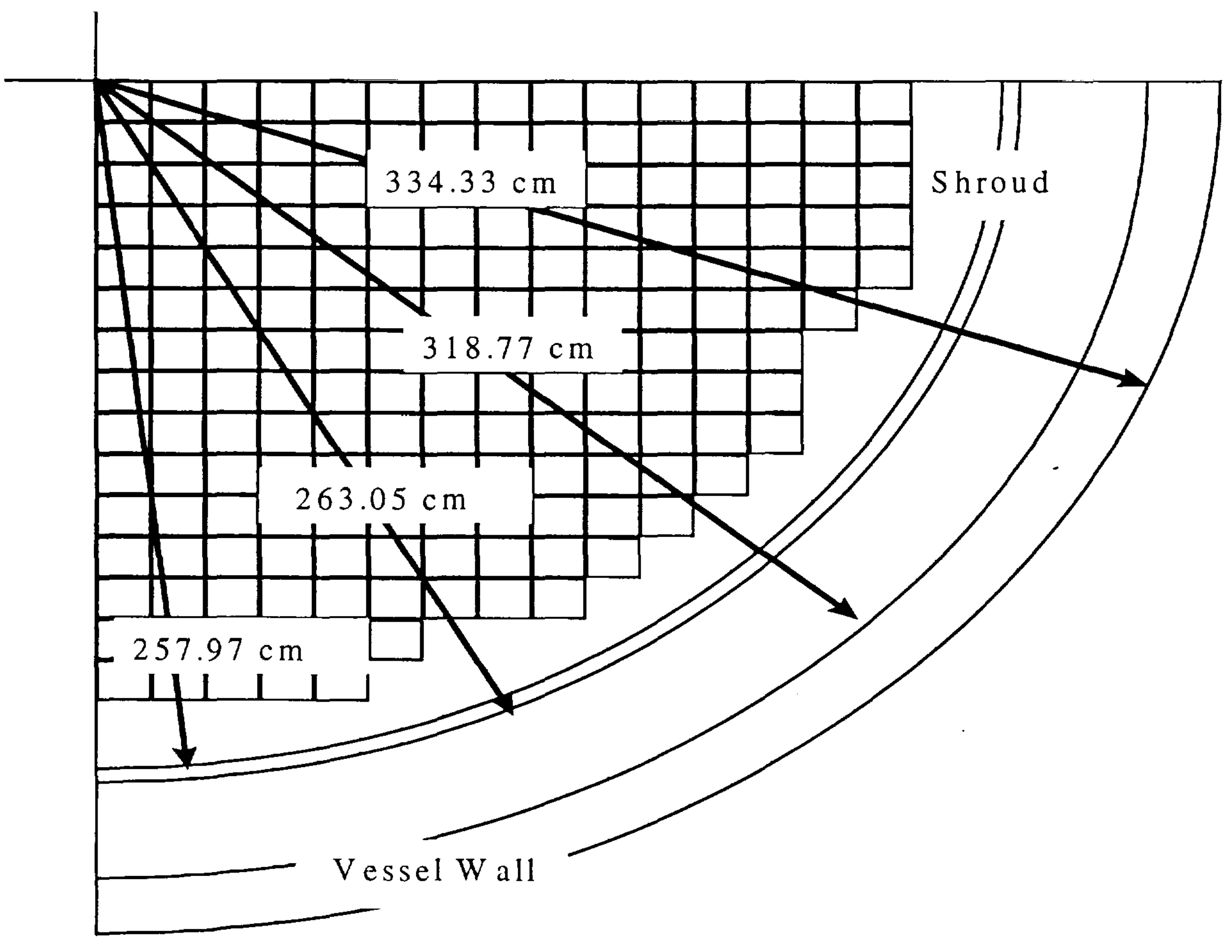
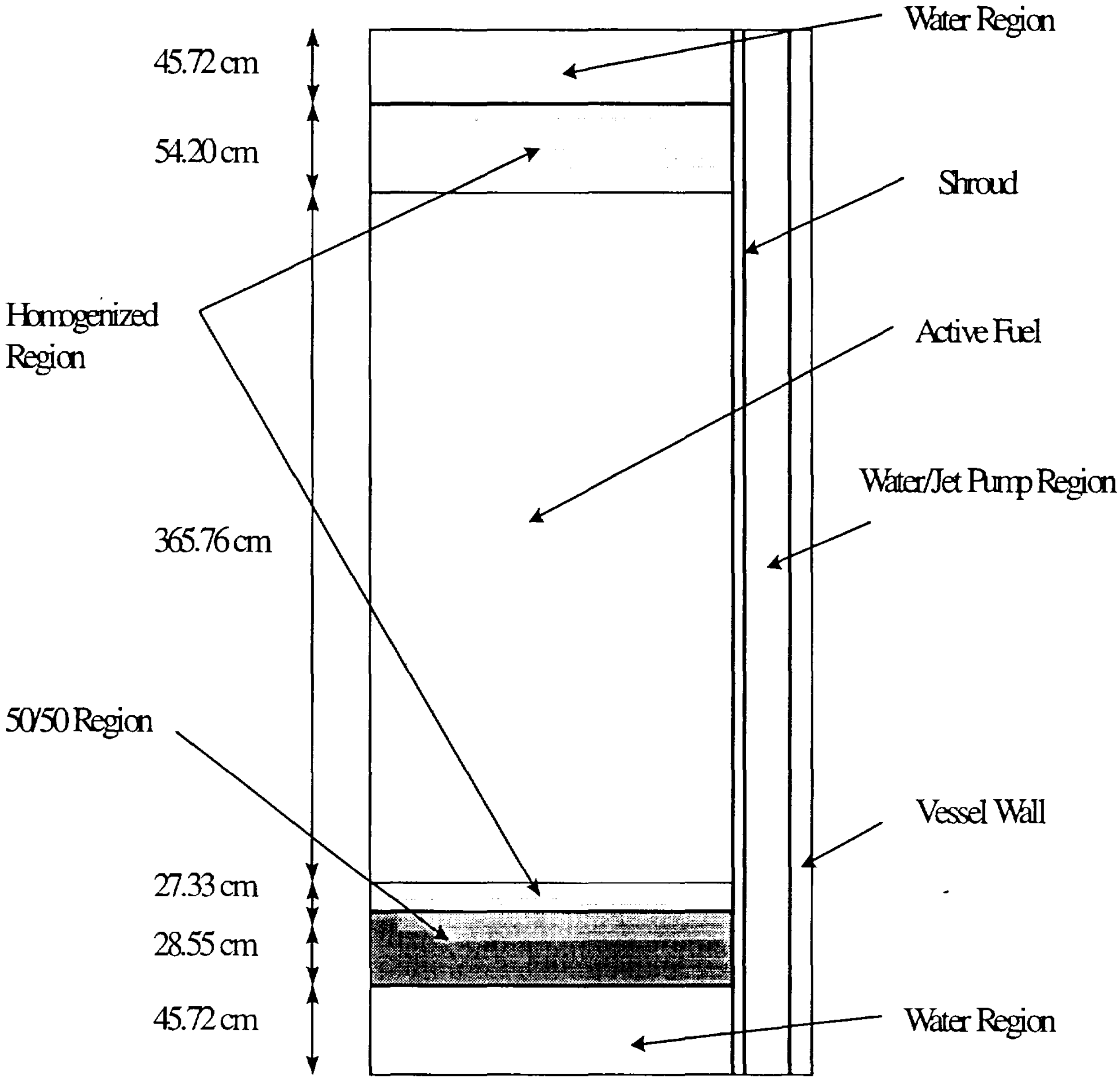


Figure 7.2.1-1. Radial View of Quad Cities Unit 1 - 1/4 Core



7.2.1-2. Axial View of Reactor Regions

7.2.2 Fuel Assembly Description

Cycle 1 of Quad Cities Unit 1 contains only fresh fuel assemblies. These fresh fuel assemblies consist of radially enriched fuel with two types of rods containing integral burnable absorber (i.e., gadolinia). Specific fuel rods in both assembly types may be dished or undished. These variations in enrichment, poisons and dishing require four basic assembly models that are used in the 1/4 core model. Descriptions of these 4 assembly types are provided in Table 7.2.2-1. All of the fuel assemblies are General Electric 7x7, Type D cell lattice designs. The fuel rod pitch is 1.87452 cm in each assembly design. The assembly lattice is surrounded by a 0.2032 cm thick, Zircaloy-4 channel. The 0.9652 cm radius, rounded corner is omitted from the channel model for simplification purposes. The fuel assembly design contains 1 center-located spacer capture rod that alternates axially between fuel regions and spacer capture regions which contain no fuel material. The modeled fuel assembly lattice configuration, channel, and fuel pin enrichments are shown in Figures 7.2.2-1 and 7.2.2-2. The fuel rod descriptions for each different type of rod are provided in Table 4.1-3. The upper region of the assembly that includes the fuel rod plenum and assembly upper tie plate is modeled as a homogenous distribution of moderator, Zircaloy-2, and Type 304 stainless steel. The lower region of the assembly that includes the lower tie plate connection is modeled as a homogenous distribution of moderator, Zircaloy-2, and Type 304 stainless steel. Model details for the upper and lower assembly regions are shown in Figure 7.2.1-2. The control cell view of four assemblies, a control blade, and an instrument tube is shown in Figure 7.2.2-3.

The overall core map for the Quad Cities 1/4 core model is shown in Figure 7.2.2-4. This figure depicts actual control blade and instrument tube locations as well as assembly type. The core map is representative of core conditions at the time of the cold critical calculation. The control blade positions for the cold critical condition were assumed to be both fully inserted into the core and fully withdrawn from the core (Reference 5.11, p. C-35). Figure 7.2.2-4 indicates the locations of the fully inserted and the fully withdrawn control blades.

Table 7.2.2-1. Basic Assembly Types

Assembly Identifier	Assembly Description
1a	dished, 2 - 3.0 wt% Gd_2O_3 burnable poison rods, 1 - 0.5 wt% Gd_2O_3 burnable poison rod, UO_2 average enrichment 2.12 wt%
1b	undished, 2 - 3.0 wt% Gd_2O_3 burnable poison rods, 1- 0.5 wt% Gd_2O_3 burnable poison rod, UO_2 average enrichment 2.12 wt%
2a	dished, 2 - 3.0 wt% Gd_2O_3 burnable poison rods, UO_2 average enrichment 2.12 wt%
2b	undished, 2 - 3.0 wt% Gd_2O_3 burnable poison rods, UO_2 average enrichment 2.12 wt%

3d	3d	2d	2d	2d	2d	3d
3d	2	2	1	1	1d	2d
2d	2	5z	1	1	1	1d
2d	1	1	1s	1	4y	1d
2d	1	1	1	1	1	1d
2d	1d	1	4y	1	1	1d
3d	2d	1d	1d	1d	1d	2d

Figure 7.2.2-1. Assembly Lattice Map Type 1

3d	3d	2d	2d	2d	2d	3d
3d	2	2	1	1	1d	2d
2d	2	1	1	1	1	1d
2d	1	1	1s	1	4y	1d
2d	1	1	1	1	1	1d
2d	1d	1	4y	1	1	1d
3d	2d	1d	1d	1d	1d	2d

Figure 7.2.2-2. Assembly Lattice Map Type 2

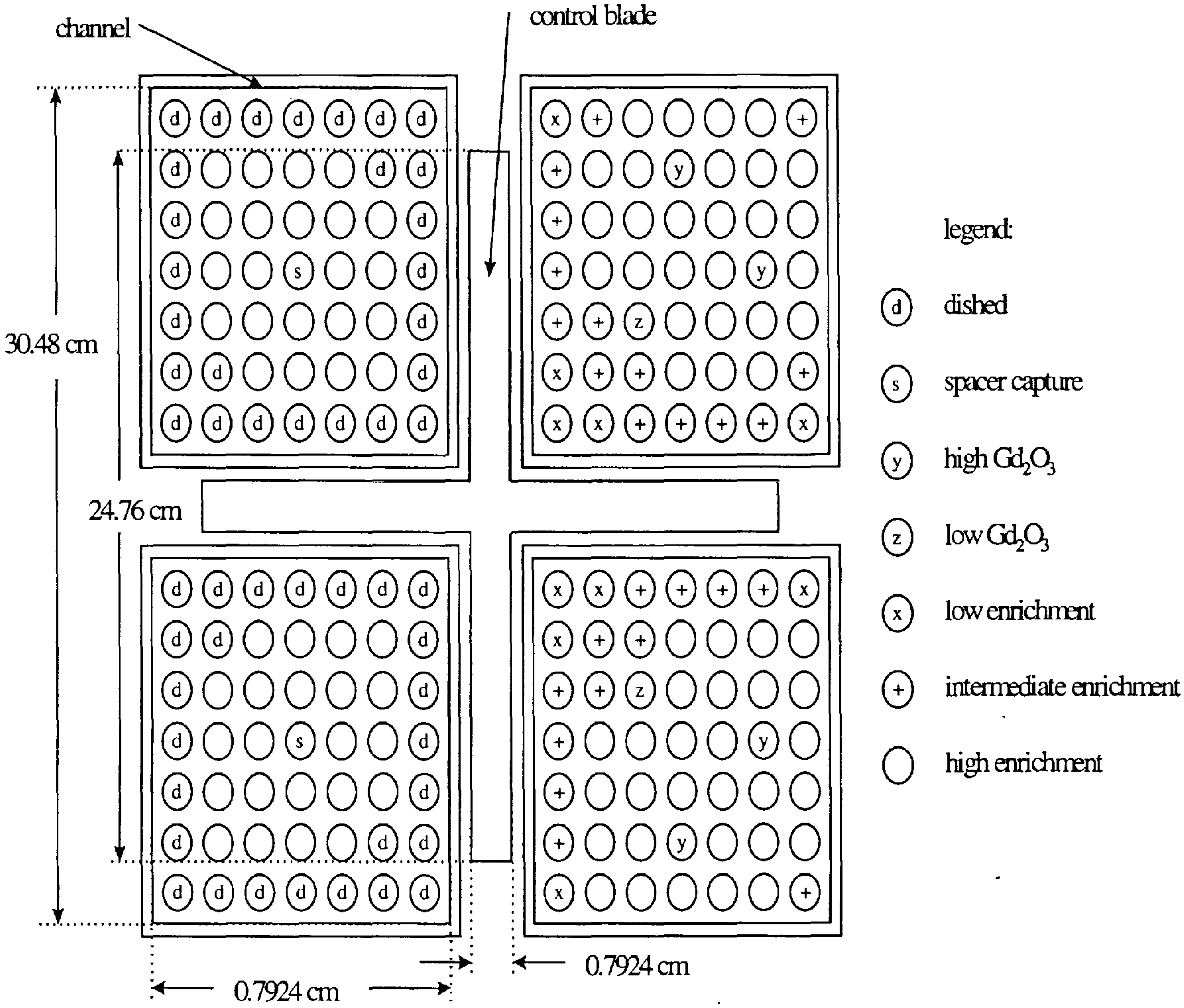


Figure 7.2.2-3. Control Cell Radial View

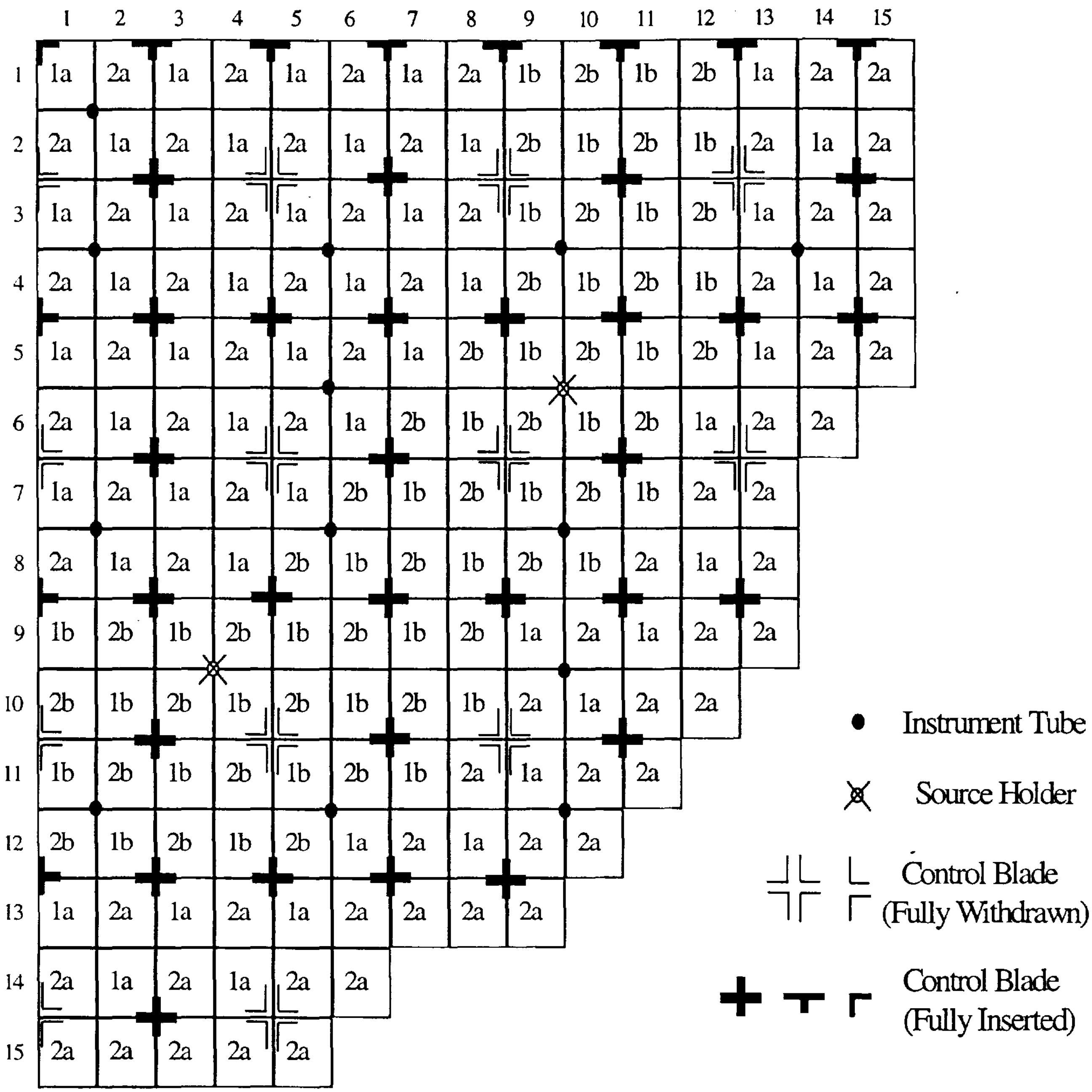


Figure 7.2.2-4. Core Map

7.2.3 Fuel Rod and Burnable Poison Rod Descriptions

The fuel rods in each assembly design are modeled explicitly with the fuel rod radial enrichment depending upon position in the fuel assembly. The fuel pellet outer diameter is 1.23952 cm for un-poisoned fuel rods and 1.23698 cm for fuel rods containing an integral burnable absorber. The clad outer diameter is 1.43002 cm with a clad thickness of 0.08128 cm. The active fuel height for all fuel rods is 365.76 cm. Individual fuel rod enrichment locations are shown in Figures 7.2.2-1 and 7.2.2-2. Depending upon the rod and assembly type used, certain fuel pellets are dished. This change in pellet volume is modeled with an appropriate change in stack density of the rod. The total number of different fuel rod types and enrichments per assembly type is shown in Table 7.2.3-1(a-d). The radial model specifications for fuel rods with gadolinia are the same as for unpoisoned fuel rods. Axial zoning for rod Types 4 and 5 are shown in Figure 7.2.3-1. Axial views for Type 1 and Type 2 assembly models are shown in Figures 7.2.3-2 and 7.2.3-3.

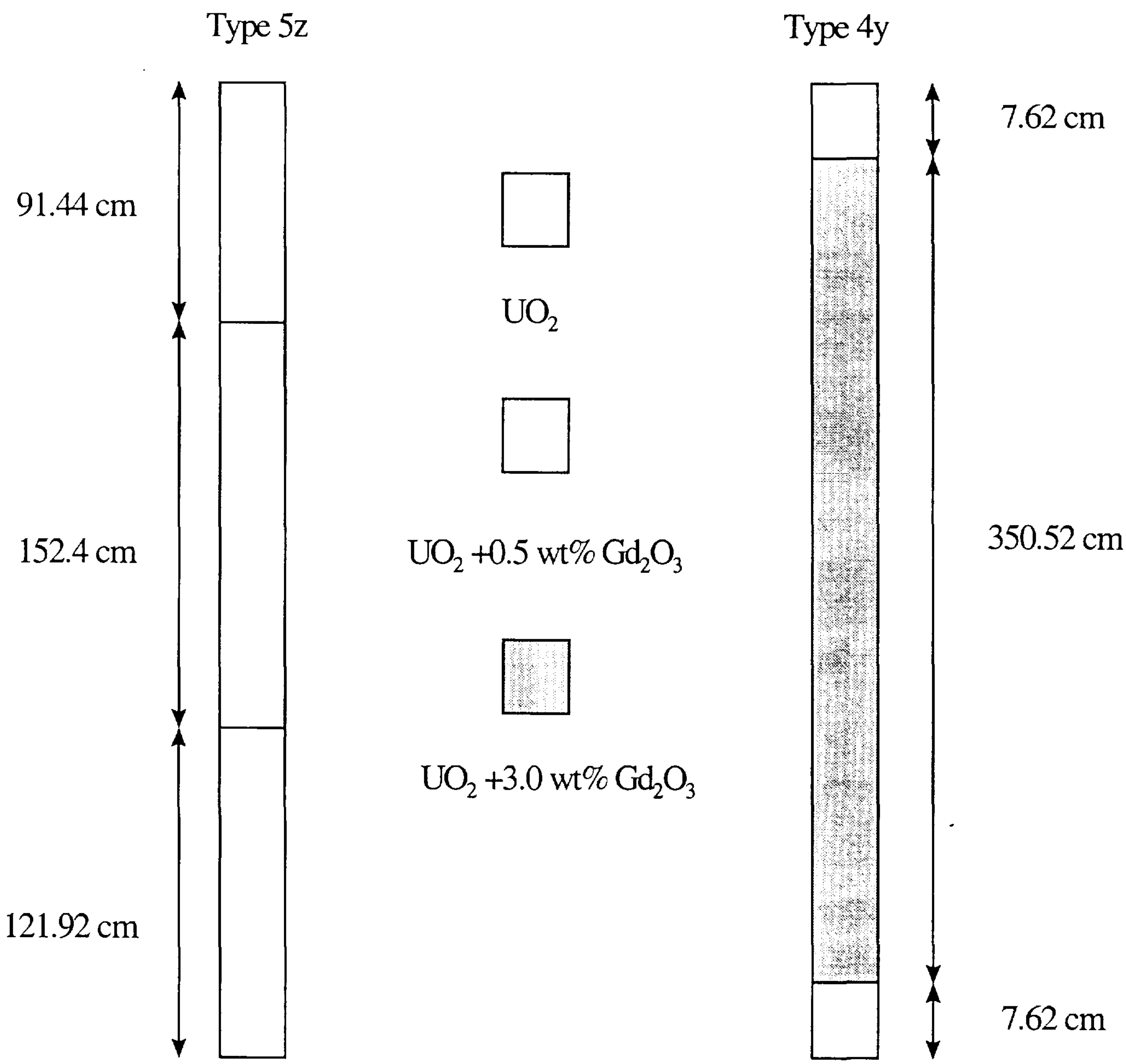


Figure 7.2.3-1. Axial View of Gadolinia-Bearing Rods

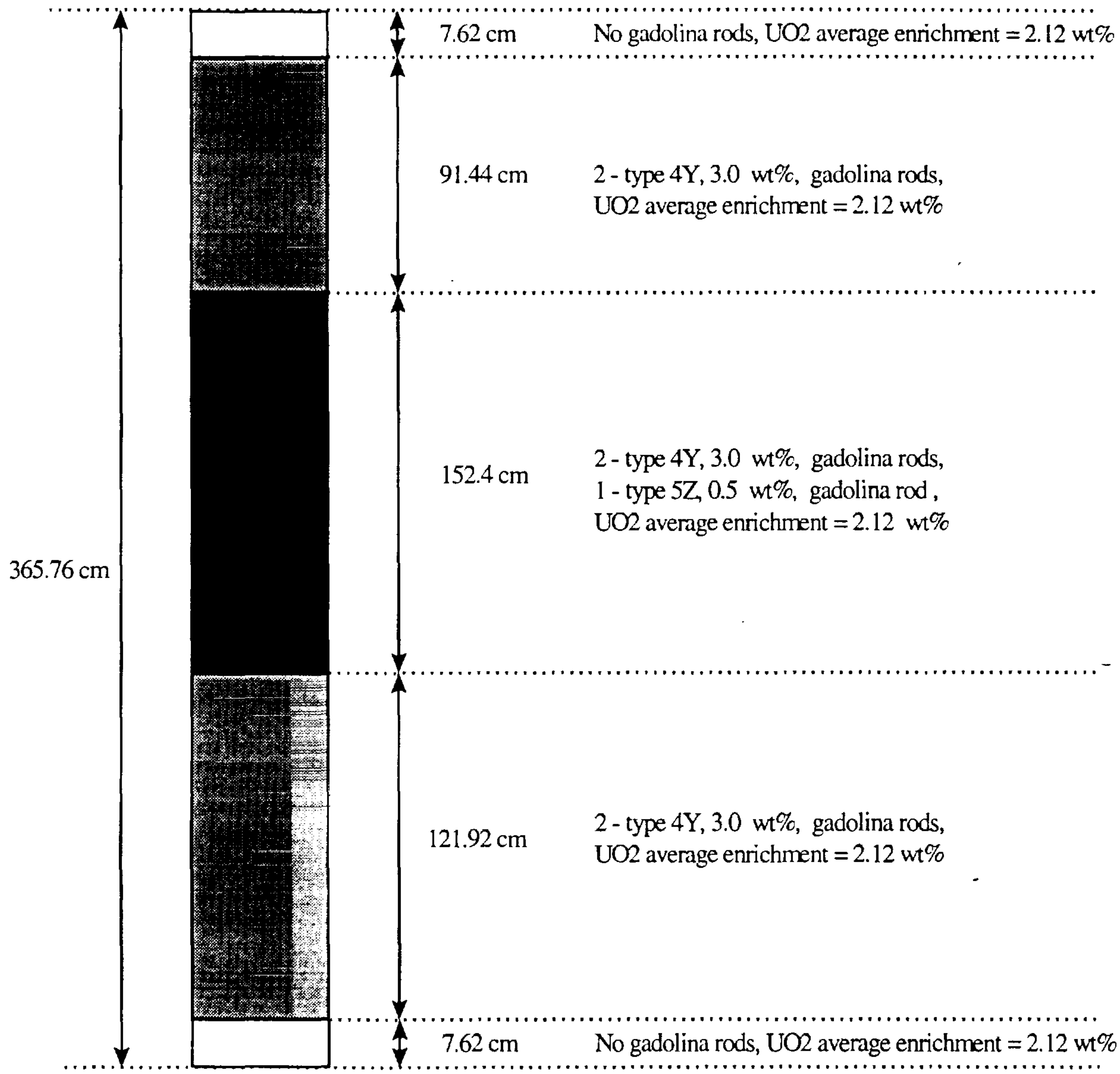


Figure 7.2.3-2. Axial View Type 1 Assembly

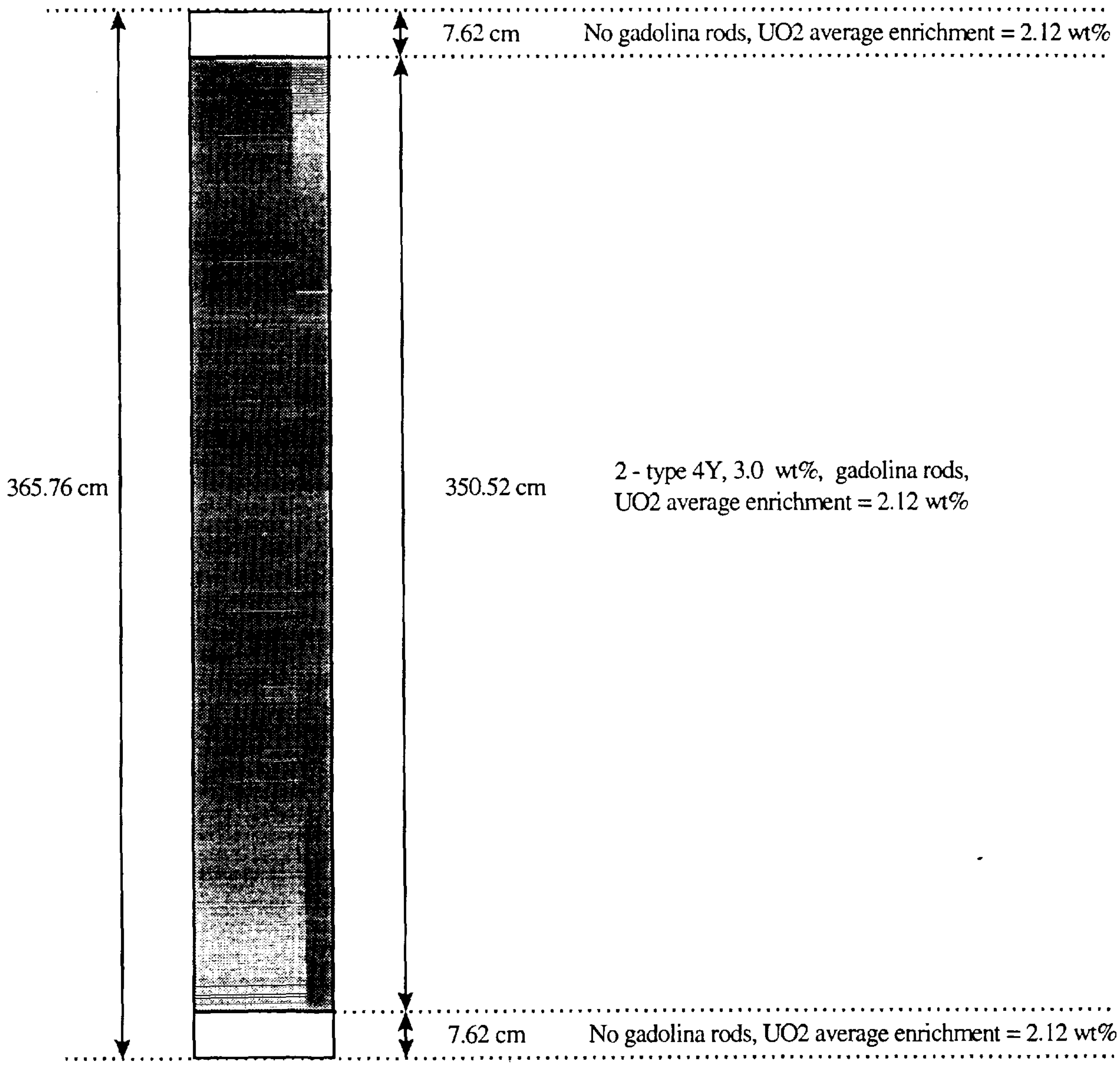


Figure 7.2.3-3. Axial View Type 2 Assembly

Table 7.2.3-1a. Assembly Type 1a

No. of Fuel Rods	Fuel Rod Description
16	rod Type 1, 2.47 wt% U-235, undished
10	rod Type 1d[a], 2.47 wt% U-235, dished
1	rod Type 1s[s], 2.47 wt% U-235, undished
3	rod Type 2, 1.70 wt% U-235, undished
11	rod Type 2d, 1.70 wt% U-235, dished
5	rod Type 3d, 1.20 wt% U-235, dished
2	rod Type 4, 2.47 wt% U-235, 3.00 wt% Gd_2O_3, undished, axially zoned
1	rod Type 5, 2.47 wt% U-235, 0.50 wt% Gd_2O_3, undished, axially zoned

[a] d-dished pellet.
[s] s-spacer capture rod.

Table 7.2.3-1b. Assembly Type 1b

No. of Fuel Rods	Fuel Rod Description
26	rod Type 1, 2.47 wt% U-235, undished
1	rod Type 1s, 2.47 wt% U-235, undished
14	rod Type 2, 1.70 wt% U-235, undished
5	rod Type 3, 1.20 wt% U-235, undished
2	rod Type 4, 2.47 wt% U-235, 3.00 wt% Gd_2O_3, undished, axially zoned
1	rod Type 5, 2.47 wt% U-235, 0.50 wt% Gd_2O_3, undished, axially zoned

Table 7.2.3-1c. Assembly Type 2a

No. of Fuel Rods	Fuel Rod Description
17	rod Type 1, 2.47 wt% U-235, undished
10	rod Type 1d, 2.47 wt% U-235, dished
1	rod Type 1s, 2.47 wt% U-235, undished
3	rod Type 2, 1.70 wt% U-235, undished
11	rod Type 2d, 1.70 wt% U-235, dished
5	rod Type 3d, 1.20 wt% U-235, undished
2	rod Type 4, 2.47 wt% U-235, 3.00 wt% Gd_2O_3, undished, axially zoned

Table 7.2.3-1d. Assembly Type 2b

No. of Fuel Rods	Fuel Rod Description
27	rod Type 1, 2.47 wt% U-235, undished
1	rod Type 1s, 2.47 wt% U-235, undished
14	rod Type 2, 1.70 wt% U-235, undished
5	rod Type 3, 1.20 wt% U-235, undished
2	rod Type 4, 2.47 wt% U-235, 3.00 wt% Gd_2O_3, undished, axially zoned

7.2.4 Control Blade Description

The cruciform control blade model cross section is shown in Figure 7.2.4-1. The blade model represents the absorber tube, sheath, and tie rod design of the all-boron carbide GE control blade originally deployed with domestic BWRs. The absorber tubes are filled with vibratory compacted boron carbide powder of multiple sieve sizes to obtain a loading corresponding to 70% of theoretical density. Each control blade consists of 84 separate Type 304 stainless steel tubes containing the boron carbide. The tubes are contained within an outer sheath that connects to a control blade tie rod. The sheath constrains the absorber tubes in the desired locations and the tie rod, along with the handle and velocity limiter, forms the primary structure. The control blades are moved in and out of the core from the bottom by individual drive mechanisms. The control blade pitch in the reactor is 30.48 cm. The effective stroke of the control blade is 365.76 cm. The span of the control blade wing is 12.38 cm. The control blades are either fully inserted or fully removed for the startup cold critical calculation. The MCNP 4A 1/4 core model of the Quad Cities Unit 1 control blades consists of partial control blade models fully inserted around the bordering assembly locations and full control blade models inserted at specific locations in the model. The nominal dimensions are shown in Table 4.1-5. The control blade locations are shown in Figure 7.2.2-4.

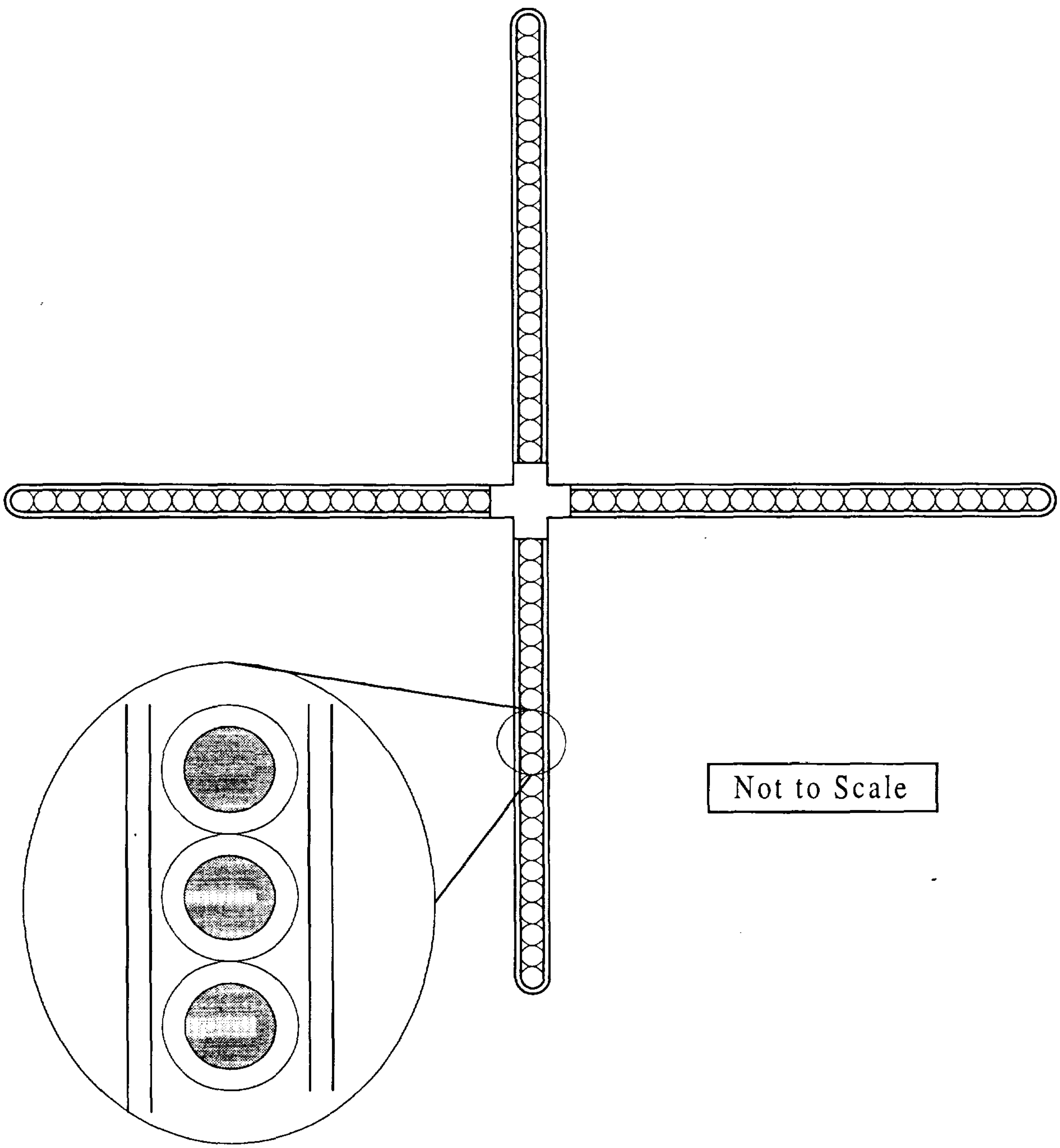


Figure 7.2.4-1. Control Blade Model

7.2.5 Instrument Tube Description

Instrument tubes have limited effect on reactivity calculations. The most significant effect is from moderator displacement in the bypass regions of the core. All instrument tubes were modeled slightly smaller than actual to fit within the square channels. Additionally, the inner regions of the instrument tubes were modeled as voids to characterize the bypass moderator displacement. The instrument tube material was modeled as Type 304 stainless steel with an outer diameter of 1.178 cm and a clad thickness of 0.0762 cm.

7.3 CRC Benchmark MCNP 4A Materials Description

The material descriptions used in the Quad Cities Unit 1 MCNP 4A CRC BOL reactivity calculation correspond to the actual reactor component materials. Some component materials in the upper and lower regions of the core having geometrical features that were not required to be explicitly defined were homogenized with surrounding materials. The homogenization of these materials preserves the average neutron interaction rate such that the reactivity worth of these materials in the system is preserved. Identification of MCNP 4A cross section libraries used in the reactivity calculations are listed in section 7.3.1. The materials for structural components of the 1/4 core model are described in Sections 7.3.2 and 7.3.3. Fuel rod materials are described in Section 7.3.4

7.3.1 MCNP 4A Cross-Section Libraries

The MCNP 4A cross-section libraries utilized in the reactivity calculations are one of the primary components of the calculation which determine whether or not the neutronic behavior of the system is simulated correctly. The k_{eff} result is a direct indicator of the validity of the MCNP 4A cross-section library data utilized in the calculation. Additionally, it should be noted that the sources of the cross-section libraries utilized in this reactivity calculation contribute to the definition of the range of applicability for any bias value determined using the associated benchmark k_{eff} result. Table 7.3.1-1 lists all of the MCNP 4A cross-section library identifiers (ZAIDs) utilized in the CRC benchmark calculation documented in this analysis. The MCNP 4A ZAIDs are used to identify the cross-section libraries. The ZAID consists of a 5 integer element and isotope identifier followed by a cross-section library designation suffix. The first one or two integers in the ZAID refer to the atomic number of the corresponding element. The three integers preceding the decimal always refer to the mass number of the element. The ZAID suffixes presented in Table 7.3.1-1, correspond to libraries compiled from either ENDF/B-V, ENDF/B-VI, LANL/T-2, or LLNL evaluated cross-section data sets. The atom percent in nature of the various isotopes presented in Table 7.3.1-1 are obtained from Reference 5.9. The atomic weight ratios, temperatures, library names, and data sources are obtained from Attachment IV of Reference 5.8.

Table 7.3.1-1. MCNP 4A Cross-Section Libraries Used in the CRC Reactivity Calculations

Element / Isotope	MCNP ZAID	Atom % in Nature	Atomic Wt. Ratio [1]	Temp. (K)	Library Name	Data Source
H-1	1001.50c	99.985	0.999167	294.0	rmccs	ENDF/B-V.0
He-4	2004.50c	99.999	3.968219	294.0	rmccs	ENDF/B-V.0
B-10	5010.50c	19.400	9.926922	294.0	rmccs	ENDF/B-V.0
B-11	5011.56c	80.600	10.914730	294.0	newxs	LANL/T-2
C-nat	6000.50c	100.0	11.907856	294.0	rmccs	ENDF/B-V.0
N-14	7014.50c	99.630	13.882780	294.0	rmccs	ENDF/B-V.0
O-16	8016.50c	99.760	15.857510	294.0	rmccs	ENDF/B-V.0
Si-nat	14000.50c	100.0	27.844241	294.0	endf5p	ENDF/B-V.0
P-31	15031.50c	100.0	30.707682	294.0	endf5u	ENDF/B-V.0
S-32	16032.50c	100.0	31.788939	294.0	endf5u	ENDF/B-V.0
Cr-50	24050.60c	4.345	49.516983	294.0	endf60	ENDF/B-VI.1
Cr-52	24052.60c	83.790	51.494313	294.0	endf60	ENDF/B-VI.1
Cr-53	24053.60c	9.500	52.485863	294.0	endf60	ENDF/B-VI.1
Cr-54	24054.60c	2.365	53.475519	294.0	endf60	ENDF/B-VI.1
Mn-55	25055.50c	100.0	54.466099	294.0	endf5u	ENDF/B-V.0
Fe-54	26054.60c	5.900	53.476242	294.0	endf60	ENDF/B-VI.1
Fe-56	26056.60c	91.720	55.454429	294.0	endf60	ENDF/B-VI.1
Fe-57	26057.60c	2.100	56.446290	294.0	endf60	ENDF/B-VI.1
Fe-58	26058.60c	0.280	57.435600	294.0	endf60	ENDF/B-VI.1
Ni-58	28058.60c	68.270	57.437652	294.0	endf60	ENDF/B-VI.1
Ni-60	28060.60c	26.100	59.415952	294.0	endf60	ENDF/B-VI.1
Ni-61	28061.60c	1.130	60.407628	294.0	endf60	ENDF/B-VI.1
Ni-62	28062.60c	3.590	61.396349	294.0	endf60	ENDF/B-VI.1
Ni-64	28064.60c	0.910	63.378793	294.0	endf60	ENDF/B-VI.1
Zr-nat	40000.60c	100.0	90.439990	294.0	endf60	ENDF/B-VI.1
Zr-93	40093.50c	0.0	92.108361	294.0	kidman	ENDF/B-V.0
Sn-nat	50000.35c	100.0	117.690428	0.0	endl85	LLNL
Gd-152	64152.50c	0.20	150.614731	294.0	endf5u	ENDF/B-V.0
Gd-154	64154.50c	2.18	152.598614	294.0	endf5u	ENDF/B-V.0

Element / Isotope	MCNP ZAID	Atom % in Nature	Atomic Wt. Ratio [1]	Temp. (K)	Library Name	Data Source
Gd-155	64155.50c	14.80	153.591761	294.0	endf5u	ENDF/B-V.0
Gd-156	64156.50c	20.47	154.582676	294.0	endf5u	ENDF/B-V.0
Gd-157	64157.50c	15.65	155.575907	294.0	endf5u	ENDF/B-V.0
Gd-158	64158.50c	24.84	156.567459	294.0	endf5u	ENDF/B-V.0
Gd-160	64160.50c	21.86	158.553203	294.0	endf5u	ENDF/B-V.0
U-233	92233.50c	0.0	231.037695	294.0	rmccs	ENDF/B-V.0
U-234	92234.50c	0.0055	232.030412	294.0	endf5p	ENDF/B-V.0
U-235	92235.50c	0.7200	233.024773	294.0	rmccs	ENDF/B-V.0
U-236	92236.50c	0.0	234.017806	294.0	endf5p	ENDF/B-V.0
U-238	92238.50c	99.2745	236.005803	294.0	Rmccs	ENDF/B-V.0

[1] The atomic weight ratio presented for each isotope/element is the ratio of the isotope/element mass to the mass of a neutron. The mass of a neutron is 1.008664904 amu (Reference 5.9).

7.3.2 Reactor Materials

The reactor components modeled in the Quad Cities Unit 1 MCNP 4A CRC reactivity calculation include the following: core shroud, pressure vessel, moderator, reactor upper plenum region, core grid, core support plate, upper tie plate region, lower tie plate region, fuel support casting region, and lower core plenum. The material compositions are described in terms of elemental or isotopic weight percents with an overall material density.

The core shroud, pressure vessel, and upper and lower structural components are modeled with Type 304 stainless steel. The Type 304 stainless steel composition is shown in Table 4.1-10.

The core moderator is water modeled at 147°F and 1 atmosphere. The Quad Cities Unit 1 BOL CRC reactivity calculation occurs at the point of a cold critical calculation where the core is flooded with moderator and no steam or voids are present.

For the Quad Cities Unit 1 BOL CRC reactivity calculation the reactor model contains two regions above the active fuel and three regions below the active fuel. The reactor upper plenum has no structure and is modeled as a flooded moderator region. The lower fuel support casting region is modeled as a homogenous composition of 50% by volume Type 304 stainless steel and 50% by volume moderator. Below the fuel support casting region the core is modeled as a flooded moderator region. The upper and lower tie plate regions of the assembly are modeled as a homogenous composition of Type 304 stainless steel, Zircaloy-2, and moderator. The upper tie plate region includes the upper assembly fuel rod plenum volume, the Zircaloy-2 fuel rod end plugs, the Type 304 stainless steel rod plenum spring and getter, and the Type 304 stainless steel assembly handle and the upper tie plate. The lower tie plate region model consists of the Zircaloy-2 fuel rod end plugs, the Type 304 stainless steel lower tie plate assembly connection, and the lower tie plate. Three axial regions in the Quad Cities Unit 1 reactivity calculation model are homogenized. These regions are the upper tie plate region, the lower tie plate region, and the fuel support casting region. Dimensions and material masses for the axial regions of the reactor are provided in Tables 4.1-6 and 4.1-7. The individual material volume fractions and homogenous material compositions in these specific reactor regions are shown in Tables 7.3.2-1, 7.3.2-2, and 7.3.2-3. The Zircaloy-2 composition is presented in Table 4.1-11. These component material compositions are used in conjunction with their volume fractions in each reactor region to obtain a homogenized material composition and density to be specified in the MCNP 4A CRC reactivity calculation. Where reactor component masses were specified in references, theoretical densities where used to determine volumes. Otherwise, volumes of components of different materials including moderator were taken as fractions of the total volume in a region to determine volume fractions. These volume fractions were then used to determine a homogenized density for all the materials in the specified region. Then, the weight fraction for each material composition was developed using this homogenized density. This homogenized weight fraction was then applied to the isotopic ZAIDs for each material as available in MCNP 4A.

Table 7.3.2-1. Material Volume Fractions for Reactor Regions Above and Below Tie Plates

Upper Reactor Region	Material Volume Fractions	
	Type 304 Stainless Steel	Moderator
Upper Core Plenum	0.0	1.0
Fuel Support Casting Region	0.50	0.50
Lower Core Plenum	0.0	1.0

Table 7.3.2-2. Material Volume Fractions for Reactor Tie Plate Regions

Upper Reactor Region	Material Volume Fractions		
	Type 304 Stainless Steel	Moderator	Zircaloy-2
Upper Tie Plate Region	0.1073 [a]	0.7384 [a]	0.0376 [a]
Lower Tie Plate Region	0.2580	0.7257	0.0161

[a] This region includes the upper assembly gas plenum volume and the instrument tube volume as a part of the calculation. These volumes are only used for the contribution to moderator displacement.

Table 7.3.2-3. Reactor Upper and Lower Region Material Compositions

MCNP ZAID	Wt. % of Element/Isotope in Material Composition				
	Reactor Upper Plenum	Reactor Upper Tie Plate Region	Reactor Lower Tie Plate Region	Fuel Support Casting Region	Reactor Lower Plenum
6000.50c	0	0.037	0.057	0.071	0
7014.50c	0	0.047	0.071	0.089	0
14000.50c	0	0.349	0.535	0.667	0
15031.50c	0	0.021	0.032	0.040	0
16032.50c	0	0.014	0.021	0.027	0
24050.60c	0	0.370	0.566	0.705	0
24052.60c	0	7.422	11.353	14.146	0
24053.60c	0	0.858	1.312	1.635	0
24054.60c	0	0.218	0.333	0.415	0
25055.50c	0	0.932	1.427	1.779	0
26054.60c	0	1.827	2.796	3.485	0
26056.60c	0	29.447	45.077	56.178	0
26057.60c	0	0.686	1.051	1.309	0
26058.60c	0	0.093	0.143	0.178	0
28058.60c	0	2.905	4.449	5.545	0
28060.60c	0	1.149	1.759	2.193	0
28061.60c	0	0.051	0.077	0.097	0
28062.60c	0	0.163	0.250	0.312	0
28064.60c	0	0.043	0.065	0.082	0
40000.60c	0	13.337	3.637	0	0
50000.35c	0	0.190	0.052	0	0
1001.50c	11.111	4.426	2.77	1.237	11.111
8016.50c	88.889	35.425	22.167	9.813	88.889
Density (g/cc)	0.98135	1.8192	2.8562	4.4407	0.98135

7.3.3 Fuel Assembly and Instrument Tube Materials

In the Quad Cities Unit 1 MCNP 4A BOL CRC criticality calculation the only other components modeled in the assembly other than the fuel rods are the assembly channel and the moderator. Dimensions for the assembly channel are provided in Table 4.1-8. The assembly channel is composed of Zircaloy-4. The material composition for Zircaloy-4 is provided in Table 4.1-12. Both in-channel and bypass moderator are modeled at density of 0.98135 g/cc.

The moderator region in the vicinity of the spacer grids is composed of homogenized Zircaloy-4 and moderator. The spacer grid material composition is Zircaloy-4. Homogenized material weight percentages for Zircaloy-4 and water are determined for each spacer grid region within the assembly channel. Volume fractions for the homogenized materials are calculated by determining total volumes of spacer grid and moderator for a specified spacer grid region. The volume fraction of spacer grid and in-channel moderator is determined to be 0.1042 and 0.8958, respectively. Figure 7.3.3-1 shows the locations of spacer grids and the spacer capture rod. The instrument tube cladding is composed of Type 304 stainless steel. Table 4.1-9 provides dimensions for the assembly, control blade and pin pitch. In the MCNP 4A criticality calculation the interior regions of the instrument tubes are treated as voids to accurately represent the displaced moderator. Table 7.3.3-1 provides material weight percentages and densities for the fuel assembly channel, instrument tube, moderator, and homogenized spacer grid.

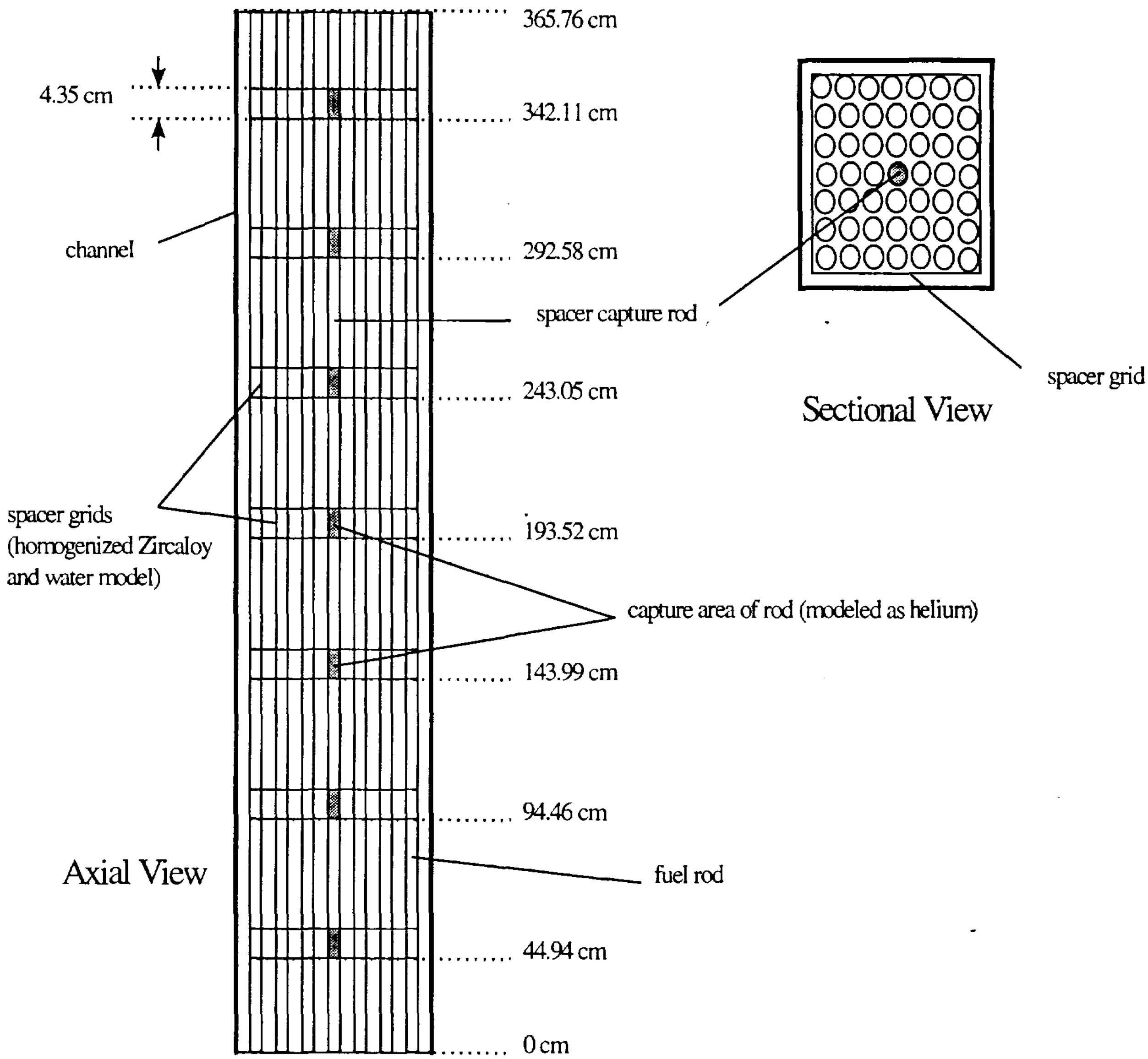


Figure 7.3.3-1. Spacer and Capture Rod Details In Fuel Assembly

Table 7.3.3-1. Fuel Assembly Component Material Compositions

MCNP ZAID	Wt. % of Element/Isotope in Material Composition			
	Assembly Channel	Instrument Tube Clad	Moderator	Homogenized Spacer Grid
6000.50c	0	0.080	0	0
7014.50c	0	0.10	0	0
14000.50c	0	0.75	0	0
15031.50c	0	0.045	0	0
16032.50c	0	0.03	0	0
24050.60c	0.004	0.793	0	0.002
24052.60c	0.084	15.903	0	0.037
24053.60c	0.01	1.838	0	0.004
24054.60c	0.002	0.466	0	0.001
25055.50c	0	2.0	0	0
26054.60c	0.011	3.918	0	0.005
26056.60c	0.184	63.156	0	0.080
26057.60c	0.004	1.472	0	0.002
26058.60c	0.001	0.2	0	0
28058.60c	0	6.234	0	0
28060.60c	0	2.45	0	0
28061.60c	0	0.109	0	0
28062.60c	0	0.350	0	0
28064.60c	0	0.092	0	0
40000.60c	98.18	0	0	42.947
50000.35c	1.40	0	0	0.612
1001.50c	0	0	11.111	6.296
8016.50c	0	0	88.889	50.013
Density (g/cc)	6.56	7.90	0.98135	1.5626

7.3.4 Fuel Rod and Burnable Poison Rod Materials

The fuel rod components include the fuel rod cladding, helium gap, and the fuel. This section describes the materials in the fuel rod below the top of the active fuel and above the bottom of the active fuel. The non-poisoned fresh fuel enrichment and composition does not vary axially. Two different types of burnable poison rods are used in the assemblies. The burnable poison rods vary in Gd_2O_3 enrichment axially. The fuel rod cladding is modeled as Zircaloy-2. The gap model between the fuel rod cladding and the fuel is filled with helium. There is a spacer capture rod that holds the spacer grids in place for the 7x7 assembly. This rod is made up of alternating sections of fuel rod and spacer capture hardware built into the rod. The mass loading of uranium, gadolinium, and fuel rod stack density for each different rod is also presented in Table 4.1-13. The weight percent of the fresh fuel in the varying rod enrichments for the uranium of the fabricated UO_2 is presented in Table 7.3.4-1. The compositions for rods with integral burnable absorber are presented in Table 7.3.4-2. The isotopic weight percentages in the fresh fuel composition are calculated using the following equations.

The weight percentages of the various isotopes in the enriched uranium of the UO_2 fuel composition are calculated using the following equations (Reference 5.13).

$$U^{234}\ wt\% = (0.007731)*(U^{235}\ wt\%)^{1.0837}$$

$$U^{236}\ wt\% = (0.0046)*(U^{235}\ wt\%)$$

$$U^{238}\ wt\% = 100 - U^{234}\ wt\% - U^{235}\ wt\% - U^{236}\ wt\%$$

The following equations are used to calculate the mass of oxygen in the fuel and gadolinium. The weight percentages of the uranium isotopes (U-234, U-235, U-236, and U-238) in uranium are calculated using the equations previously presented.

$$Oxygen\ Mass\ in\ UO_2 = \frac{[(Mass\ of\ Uranium\ in\ UO_2)*(2)*(15.994915)]}{\left[\begin{array}{c}(wt\%\ U^{235})*(235.043915) + (wt\%\ U^{234})*(234.040904) + \\ (wt\%\ U^{236})*(236.045637) + (wt\%\ U^{238})*(238.05077)\end{array}\right]}$$

$$Oxygen\ Mass\ in\ Gd_2O_3 = \frac{[(Mass\ of\ Gadolinium)*(3)*(15.994915)]}{\left[\begin{array}{c}(wt\%\ Gd^{152})*(151.9197) + (wt\%\ Gd^{154})*(153.9208) + \\ (wt\%\ Gd^{155})*(154.9226) + (wt\%\ Gd^{156})*(155.9221) + \\ (wt\%\ Gd^{157})*(156.9239) + (wt\%\ Gd^{158})*(157.9241) + \\ (wt\%\ Gd^{160})*(159.927)\end{array}\right]}$$

The weight percent of each uranium isotope in the fresh UO_2 composition is determined by multiplying the weight percent of each uranium isotope in the enriched uranium by the weight fraction of uranium in the UO_2. The weight percent of oxygen in the UO_2 is the weight fraction of oxygen in UO_2 multiplied by 100. The weight percent of oxygen in the Gd_2O_3 is the weight fraction of oxygen in Gd_2O_3 multiplied by 100.

The total weight percent of oxygen is combined with the fuel and gadolinium weight percent obtained from the previously mentioned equations to determine the various fuel and BPR isotopic weight percents. The Quad Cities Unit 1 BOL MCNP 4A reactivity calculation output file is contained in Attachment I (moved to Reference 5.15). This output file contains an echo of the MCNP 4A input file for the reactivity calculation. The fuel isotopic compositions are listed in the input file in terms of ZAIDs, weight percents, and density (g/cc). The MCNP 4A ZAIDs used for fuel composition are at a temperature of 294 K (69.53 °F). The actual fuel temperature at cold critical is that of the moderator or 337 K (147 °F). The fresh fuel density is modeled as the stack density of 10.34 g/cc and 9.94 g/cc for undished pellets and dished pellets, respectively. The BPR pellets are not dished and have densities for the Type 4 rod and Type 5 rod of 10.26 g/cc and 10.39 g/cc, respectively.

Table 7.3.4-1. Fresh Fuel Material Compositions

MCNP ZAID	Wt. % of Element/Isotope in Material Composition		
	Rod Type 1	Rod Type 2	Rod Type 3
8016.50c	13.4425	13.4412	13.4403
92234.50c	0.017828	0.011893	0.008154
92235.50c	2.137970	1.471500	1.038716
92236.50c	0.009835	0.003769	0.004778
92238.50c	84.391880	85.06867	85.50805

Table 7.3.4-2. Fresh Fuel With Gadolinia Material Compositions

MCNP ZAID	Wt. % of Element/Isotope in Material Composition	
	Rod Type 4	Rod Type 5
8016.50c	13.956330	13.523910
92234.50c	0.017292	0.017743
92235.50c	2.073621	2.127775
92236.50c	0.009539	0.009788
92238.50c	81.851810	83.989420
64152.50c	0.0041830	0.0006330
64154.50c	0.0455930	0.0007224
64155.50c	0.309528	0.0490420
64156.50c	0.428110	0.678310
64157.50c	0.327305	0.0518590
64158.50c	0.519505	0.0823110
64160.50c	0.4571810	0.072436

7.4 Core Loading Descriptions

The core loading description for the Quad Cities BOL CRC reactivity calculation includes the specification of the various fuel assembly locations, control blade locations, and instrument tube locations. A core loading description is provided in Figure 7.2.2-4.

7.5 Calculation Control Description

The Quad Cities BOL CRC reactivity calculation followed the histories of 250 cycles of 10,000 neutrons. The initial cycle of each calculation utilized cylindrical neutron volume sources located in the fuel region of the 1/4 core model. The location of the neutron volume sources is shown in Figure 7.5-1.

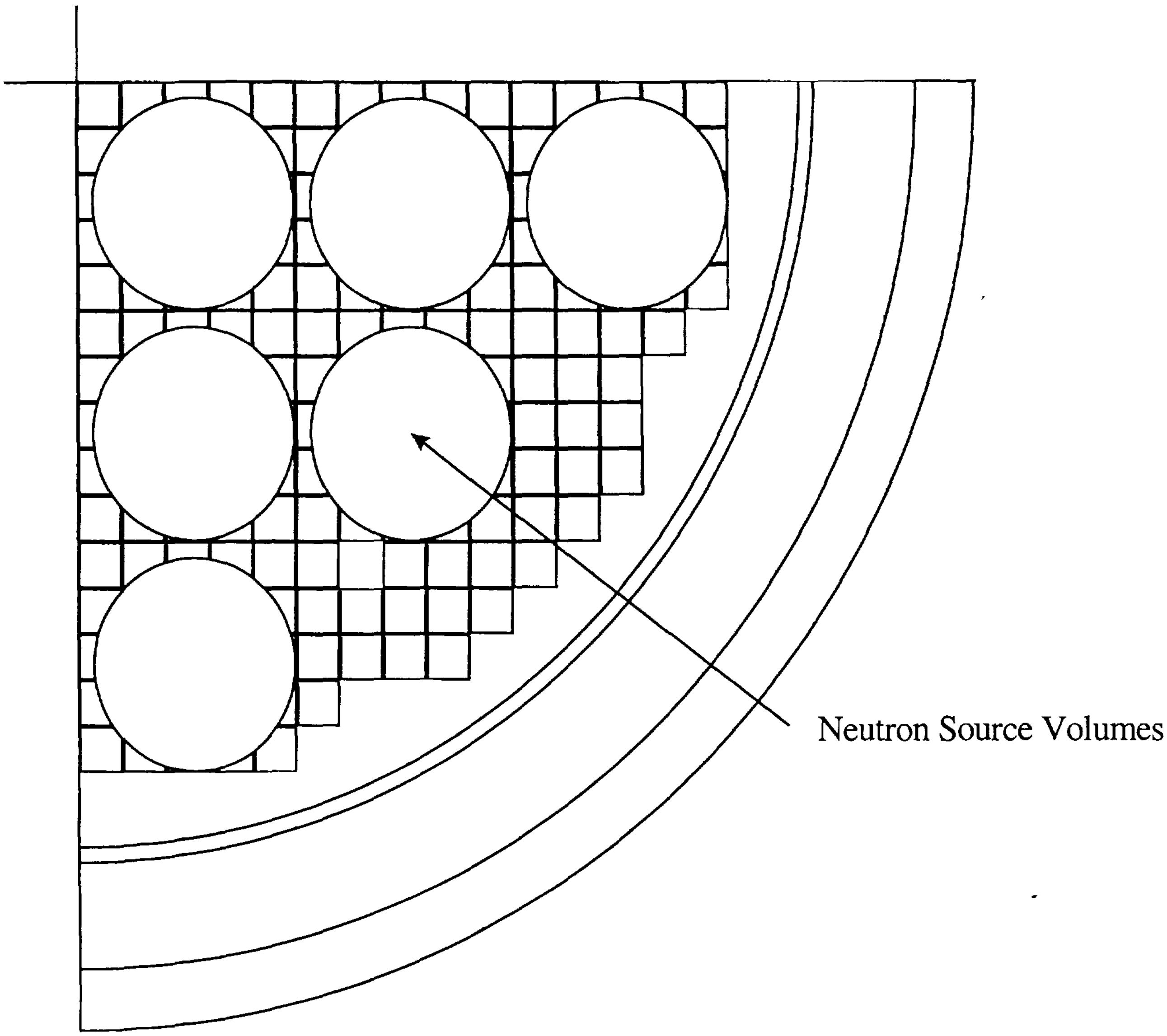


Figure 7.5-1. Radial View of Neutron Source Cylinders

7.6 CRC k_{eff} Results

The CRC k_{eff} calculated result and standard deviation for the Quad Cities BOL MCNP 4A CRC reactivity calculation are 1.00435 and 0.0004, respectively.

8. CONCLUSIONS

The k_{eff} result presented in Section 7.6 demonstrates that the MCNP 4A reactivity calculation performed using the cross-section libraries presented in Table 7.3.1-1 provides an acceptable prediction of fresh fuel reactivity for the Quad Cities Unit 1 BWR in a cold critical condition.

9. ATTACHMENTS

Attachment I which contains the MCNP 4A output file for the Quad Cities Unit 1 BOL CRC criticality calculation, has been submitted as a reference (see Reference 5.15).

This file contains an echo of the original input used in the MCNP 4A run. Additionally, the output file contains the k_{eff} for the Quad Cities Unit 1 BOL CRC criticality calculation.

The output file information follows:

Filename:	qc1bolo
Pages:	210
Size:	1177188 bytes