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Civilian Radioactive Waste Management System
Management & Operating Contractor

Summary Report of Commercial Reactor Criticality Data for McGuire Unit 1

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1.0 INTRODUCTION

The "Summary Report of Commercial Reactor Criticality Data for McGuire Unit 1" contains the detailed information necessary to perform commercial reactor criticality (CRC) analyses for the McGuire Unit 1 reactor.

1.1 Background

The United States Department of Energy (DOE) Office of Civilian Radioactive Waste Management (OCRWM) is developing a methodology for criticality analysis to support disposal of commercial spent nuclear fuel in a geologic repository. A topical report on the disposal criticality analysis methodology is scheduled to be submitted to the United States Nuclear Regulatory Commission (NRC) for formal review in September 1998. This technical report provides data that will be used in analyses that will support the development of parts of the disposal criticality analysis methodology.

1.2 Objective

The objective of this report is to present the data required for performing analytical CRC evaluations for the McGuire Unit 1 reactor. Results from the CRC evaluations will support the development and validation of the neutronics models used for criticality analyses involving commercial spent nuclear fuel. These models and their validation will be discussed in the Disposal Criticality Analysis Methodology Topical Report.

1.3 Scope

The scope of this Summary Report is the data required to perform the first of 6 statepoint calculations from cycles 1, 6 and 7 of McGuire Unit 1 (revision will include the data for the remaining 5 statepoints). The only interface for the development of the information in this document is with Framatome Cogema Fuels (FCF). FCF is one of the teammates of the Civilian Radioactive Waste Management System Management and Operating Contractor (M&O). FCF independently requested and received permission from Duke Power Company, the owner/operator of McGuire Unit 1, to publish the information related to statepoint measurements that is recorded in this document. All the information contained in this report is documented in an FCF calculational file (Reference 5). The data provided in reference 5 was obtained from various other reports, calculations, and drawings developed under an NRC accepted quality assurance program (Reference 1) and the data has supported prior licensing submittals. The data therefore will be considered acceptable for quality affecting activities and for use in analyses affecting procurement, construction, or fabrication.

1.4 Quality Assurance

The Quality Assurance program applies to the development of this report. The data provided in this report will indirectly be used to develop the methodology for evaluating the Mined Geologic Disposal System (MGDS) waste package and engineered barrier segment; the waste package and engineered barrier segment have been identified as MGDS Q-List items important to safety and waste isolation (Reference 2). The waste package is on the Q-List by direct inclusion by the DOE; a QAP-2-3 evaluation has yet to be conducted. There are no determination of importance evaluations developed in accordance with Nevada Line Procedure, NLP-2-0, since the report does not involve any field activity.

The Waste Package Development Department responsible manager has evaluated the technical document development activity in accordance with QAP-2-0 Conduct of Activities. The "Develop Technical Documents" (Reference 3) evaluation has determined the preparation and review of this technical document is subject to *Quality Assurance Requirements and Description* (Reference 4) controls. No scientific and engineering software or computational software was used in the development of this report.

2.0 REACTOR DESIGN INFORMATION

This section provides general material and geometry data for modeling the McGuire Unit 1 reactor. Figures 2-1 through 2-10 provide pictorial representations of various components that must be modeled. A horizontal view of the vessel internals is presented in Figure 2-1. This includes the 193 fuel assemblies (FA) in the reactor core region. All dimensions in this figure are measured from the center of the reactor core. A radial view of the fuel assembly layout (along the core flat) and extending through the core liner is provided in Figure 2-2. The core liner, core barrel, neutron pad, and vessel weld liner are represented as stainless steel (SS304). The pressure vessel is carbon steel (CS508). Table 2-1 provides dimensions from the center of the core (along the core flat) to the outside surface of the pressure vessel.

Table 2-1. Dimensions from Core Center to Outside Surface of Pressure Vessel

Description	Thickness, cm	Outer Radius, cm
Core Center	-	00.00000
½ FA-1	10.70102	10.70102
Water	0.10160	10.80262
FA-2	21.40204	32.20466
Water	0.10160	32.30626
FA-3	21.40204	53.70830
Water	0.10160	53.80990
FA-4	21.40204	75.21194
Water	0.10160	75.31354
FA-5	21.40204	96.71558
Water	0.10160	96.81718
FA-6	21.40204	118.21922
Water	0.10160	118.32082
FA-7	21.40204	139.72286
Water	0.10160	139.82446
FA-8	21.40204	161.22650
Water	0.21350	161.44
Core Liner	2.85000	164.29
Water	23.67	187.96
Core Barrel	5.72	193.68
Water	25.47	219.15
Vessel Liner	0.56	219.71
Pressure Vessel	21.99	241.70

For Figure 2-1, the axial dimensions of the four symmetric neutron pads can be represented as the same as the active height of the fuel in the core.

Table 2-2 summarizes fuel assembly and reactor core data used for modeling the McGuire Unit 1 reactor for cycle 1. Additional fuel cycle design, core operations, and reactor criticality statepoint information will be provided in Sections 3 and 4.

Figure 2-1. Horizontal View of Vessel Internals Along Core Midplane

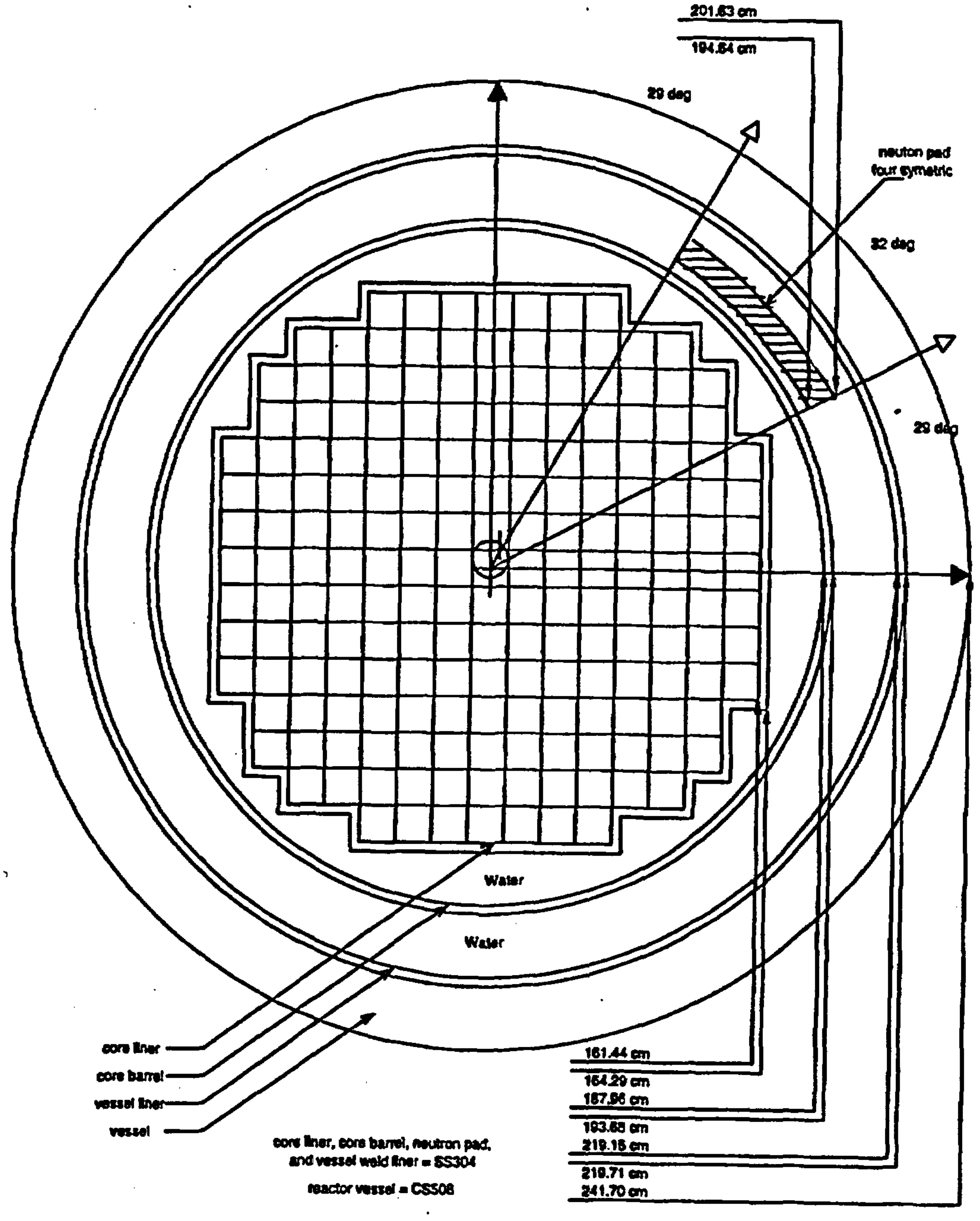


Figure 2-2. Radial View of Fuel Assembly Layout Along the Core Flat

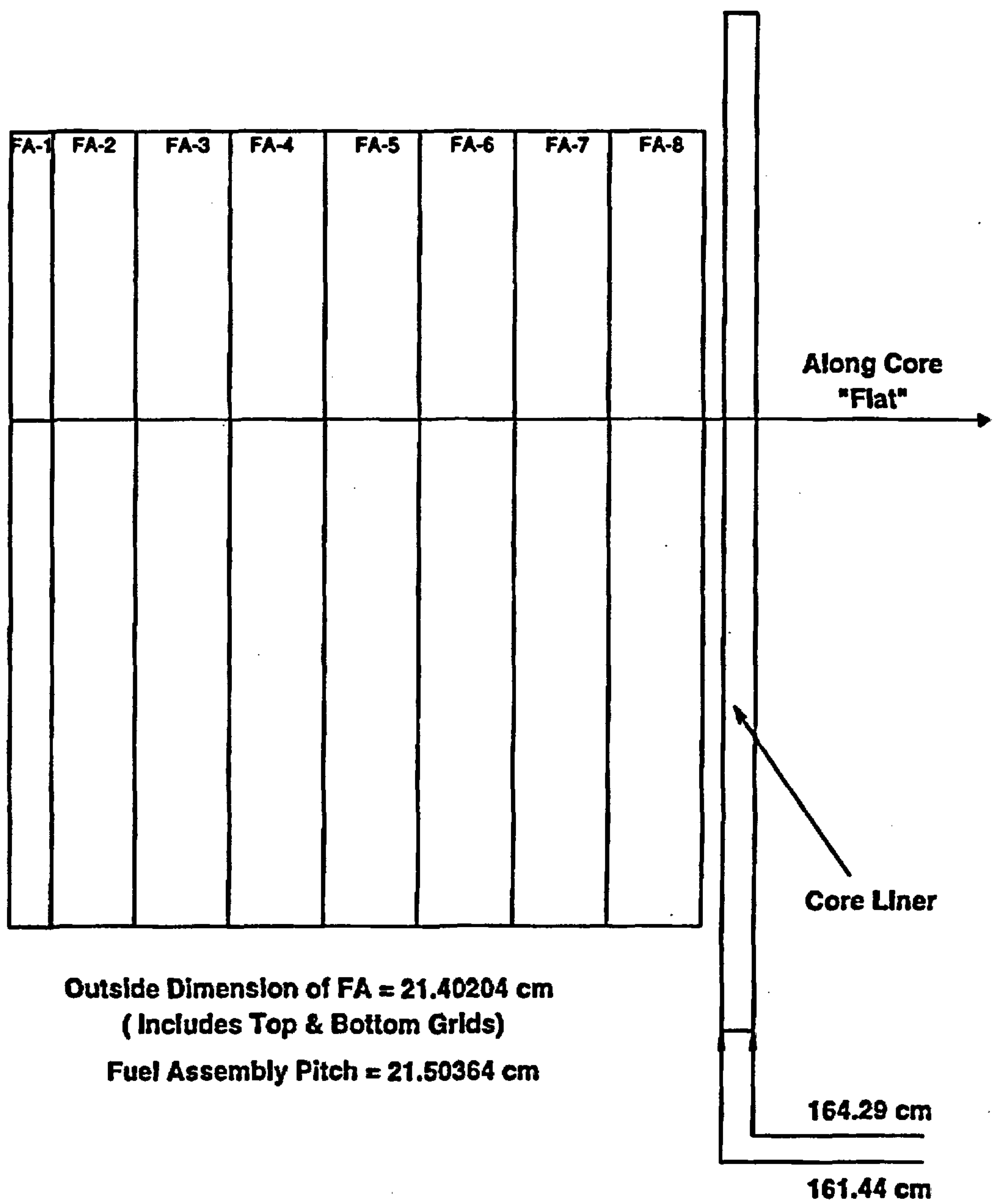


Table 2-2. McGuire 1 Cycle 1 Fuel Assembly/Core Data

Fuel Assembly Array Size and Type	17 x 17 Std
Number of Fuel Pins (N_R) / Assembly	264
Number of Guide Tubes (N_{GT}) / Assembly	24
Number of Instrument Tubes (N_{IT}) / Assembly	1
Number of Assemblies in Core	193
System Pressure	2280 psia/1.572 x 10^7 Pa
Core Height (H)	365.76 cm
Pin Pitch	1.25984 cm
Fuel Pin Cladding OD (outer diameter - OD_C)	0.94996 cm
Fuel Cladding Material	Zircaloy
Guide Tube Upper Region	
Length in Active Fuel Region (H_1)	308.4703 cm
Guide Tube OD (OD_{GT-U})	1.22428 cm
Guide Tube Lower Region	
Length in Active Fuel Region (H_2)	57.2897 cm
Guide Tube OD (OD_{GT-L})	1.08966 cm
Guide Tube Material	Zircaloy
Instrument Tube OD (OD_{IT})	1.22428 cm
Instrument Tube Material	Zircaloy
Assembly Pitch (P)	21.50364 cm
Inconel Spacer Grid Height	3.35788 cm

Grid Volume for Active Fuel Region in Single Assembly:

Volume of Inconel Grid = V_{IG} = 666.6352 cm^3

V_{M+G} = Volume of Moderator plus Grid in Fuel Assembly (excluding inside guide tubes and instrument tube)

$$= P^2 \cdot H - H \cdot \frac{\pi}{4}[N_R \cdot OD_C^2 + N_{IT} \cdot OD_{IT}^2] - N_{GT} \cdot \frac{\pi}{4}[H_1 \cdot OD_{GT-U}^2 + H_2 \cdot OD_{GT-L}^2]$$

$$= 90{,}263.3285 \text{ cm}^3$$

Assembly Volume Fraction of Inconel Grid = V_{IG}/V_{M+G} = 0.0073854

Figure 2-3 presents a radial view of a single fuel assembly showing the locations of the guide tubes, instrument tube, and fuel pins. Figure 2-4 provides axial dimensions, by region, for the Westinghouse 17 x 17 standard fuel assembly (17 x 17 Std). This assembly contains 6 inconel intermediate spacer grids and two inconel end spacer grids. The upper end spacer grid is above the active fuel region, whereas the lower end spacer grid and the 6 intermediate spacer grids are inside the active fuel region.

Figure 2-3. Radial View of a Single 17 x 17 Fuel Assembly

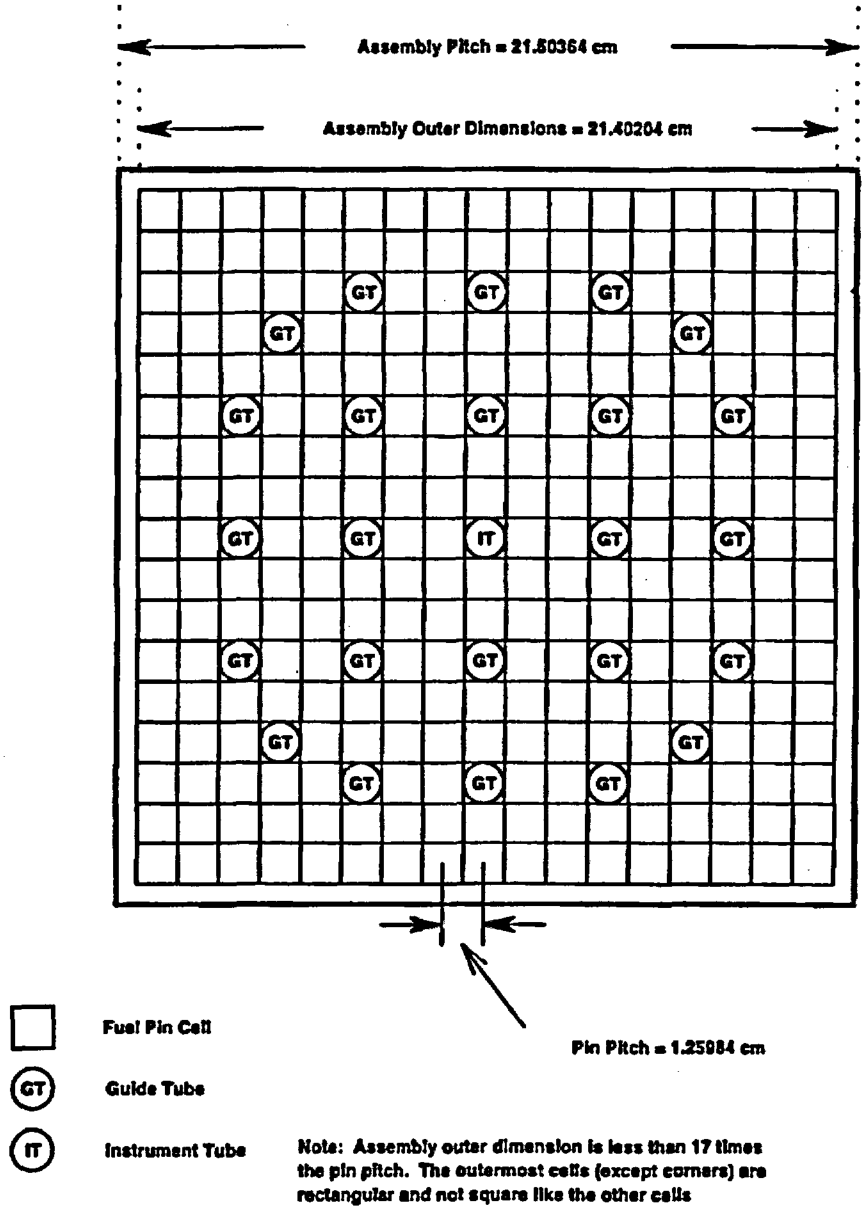
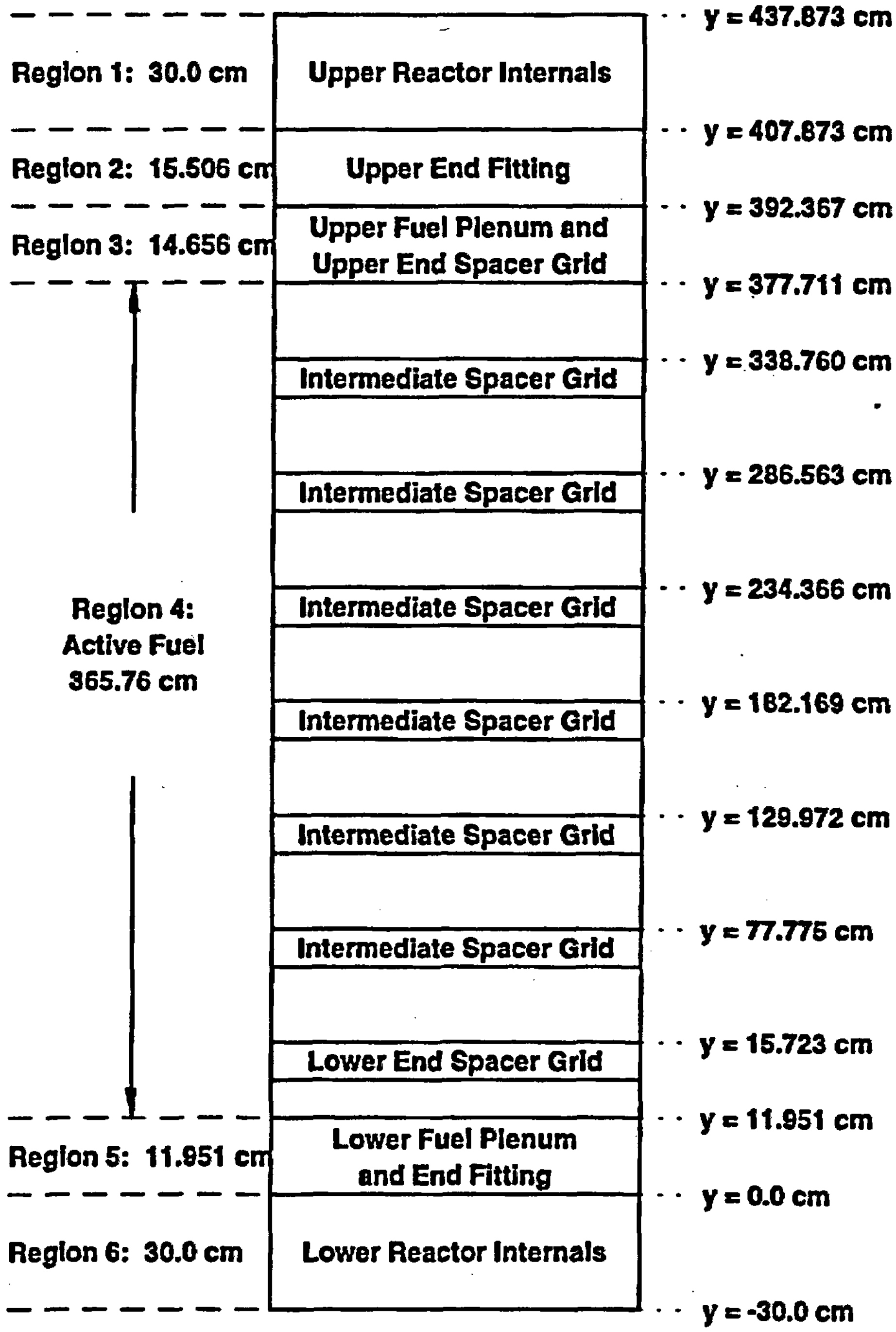


Figure 2-4. Axial Dimensions by Region for Westinghouse 17 x 17 Standard Fuel Assembly



Figures 2-5 through 2-7 provide axial dimensions for the guide tubes, instrument tube, and fuel pins shown in Figure 2-3. Figures 2-8 through 2-10 provide axial dimensions for rod cluster control assemblies (RCCAs) with rods at 0% withdrawn, pyrex burnable poison rod assemblies (BPRAs), and thimble plugs that are attached to BPRAs at empty locations.

Regions 1 and 6, in Figures 2-4 through 2-10, are represented as homogenized regions of stainless steel and water. Regions 2, 3, and 5 contain various combinations of guide tubes, instrument tube, and fuel rod assemblies (no fuel pellets), as well as other materials (stainless steel, inconel, and water). The fraction of guide tubes, instrument tube, and fuel rod assemblies will be represented explicitly in these regions. (Note: the fuel rod assemblies do not extend to region 2.) The other materials will be homogenized within the remaining portions of the regions. The water inside the guide tubes and instrument tube will be represented explicitly within the respective tubes. The volume fractions of other materials, by region, for the Westinghouse 17 x 17 standard fuel assembly are presented in Table 2-3.

Table 2-3. Volume Fractions for Non-Fuel Regions for Non-Control Assemblies

	Volume Fractions*		
Region	SS	Inc	Water
1	0.1770	0.0	0.8230
2	0.1243	0.0168	0.8589
3	0.0031	0.0264	0.9705
5	0.1625	0.0	0.8375
6	0.1720	0.0	0.8280

* The volume fractions presented exclude the guide tubes, instrument tube, and fuel rod assembly portions of these regions.

Note: Inc = Inconel
SS = Stainless Steel

The fuel rods are contained in regions 3, 4, and 5. Region 4 is modeled explicitly. Regions 3 and 5 contain various amounts of stainless steel and zircaloy in the fuel rod assembly which represent plenum springs and end caps. In addition, these regions also contain helium and fission gases, as well as the zircaloy cladding. The fuel rod assembly volume fractions for materials in these regions for the Westinghouse 17 x 17 standard fuel assembly are as follows:

Table 2-4. Fuel Rod Assembly Volume Fractions for Regions 3 and 5

	Fuel Rod Assembly Volume Fractions			
Region	SS	Zr	Cladding*	Gas
3	0.0764	0.0513	0.2173	0.6550
5	0.1241	0.1685	0.1898	0.5176

* The zircaloy (Zr) cladding extends from Y = 8.278 cm to Y = 391.615 cm. For all 264 rods, 13.904 cm length of fuel cladding is included in region 3 and 3.673 cm length of fuel cladding is included in region 5.

Figure 2-5. Axial Dimensions for Guide Tubes for Westinghouse 17×17 Standard Fuel Assembly

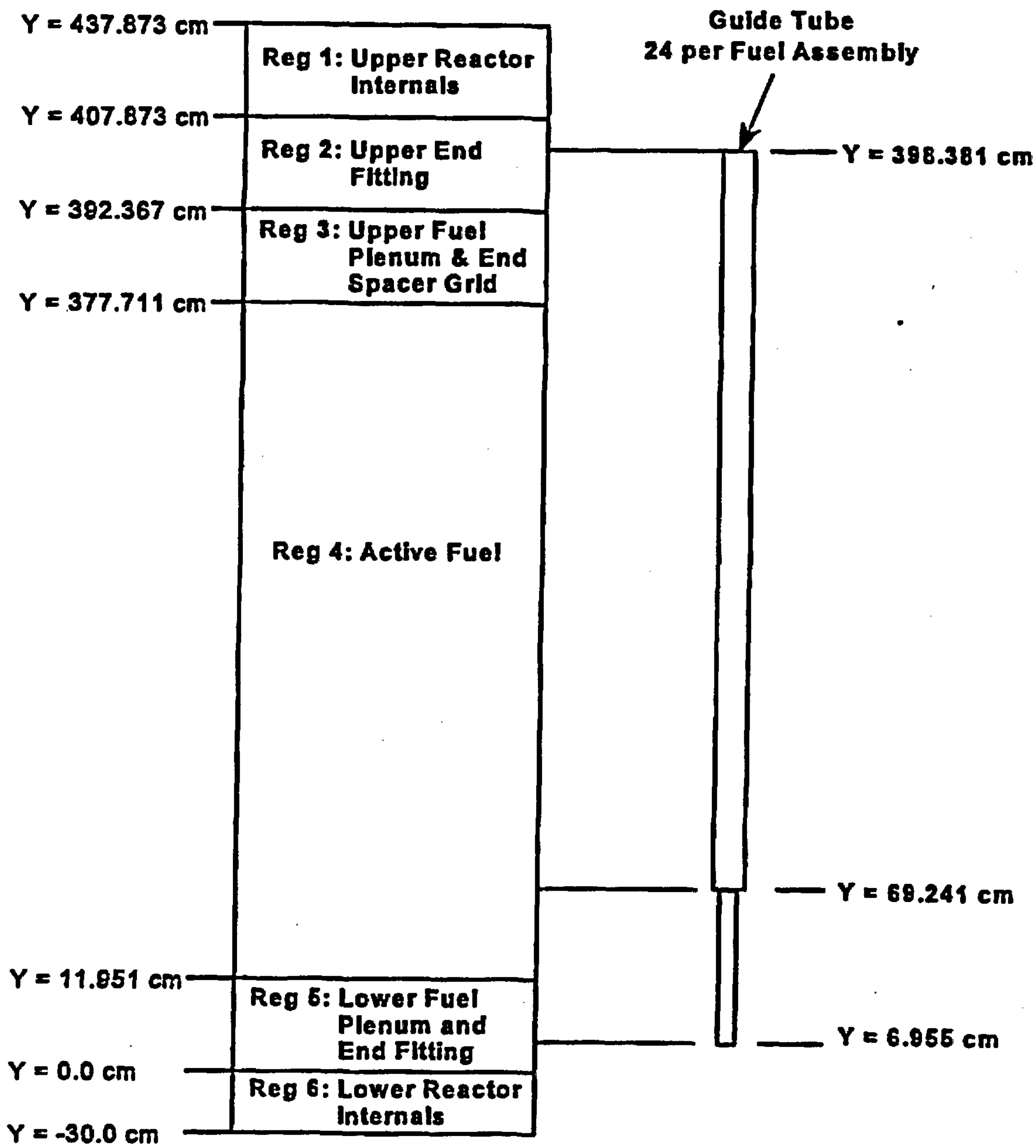


Figure 2-6. Axial Dimensions for Instrument Tube for Westinghouse 17×17 Standard Fuel Assembly

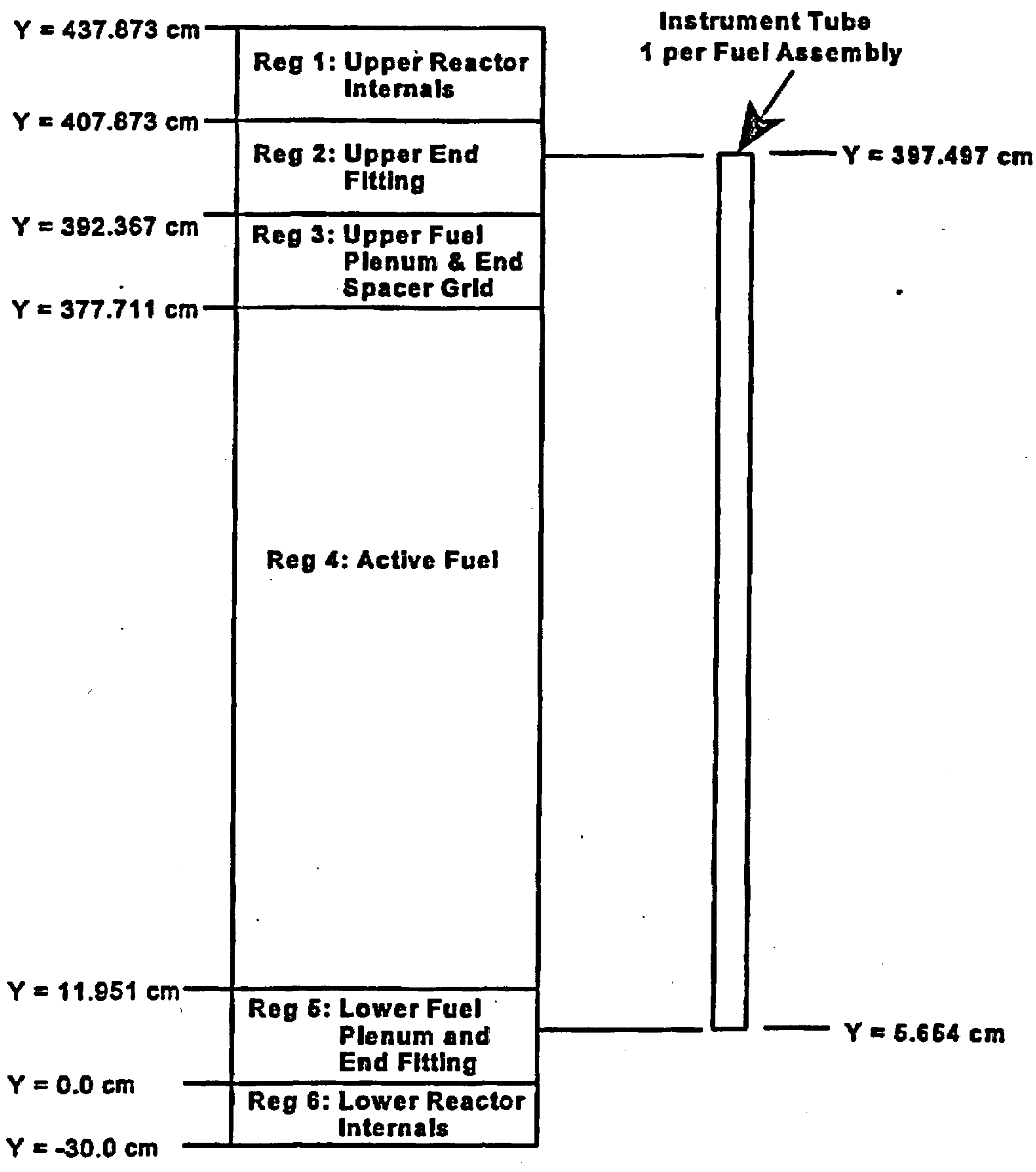


Figure 2-7. Axial Dimensions for Fuel Rod Assembly for Westinghouse 17×17 Standard Fuel Assembly

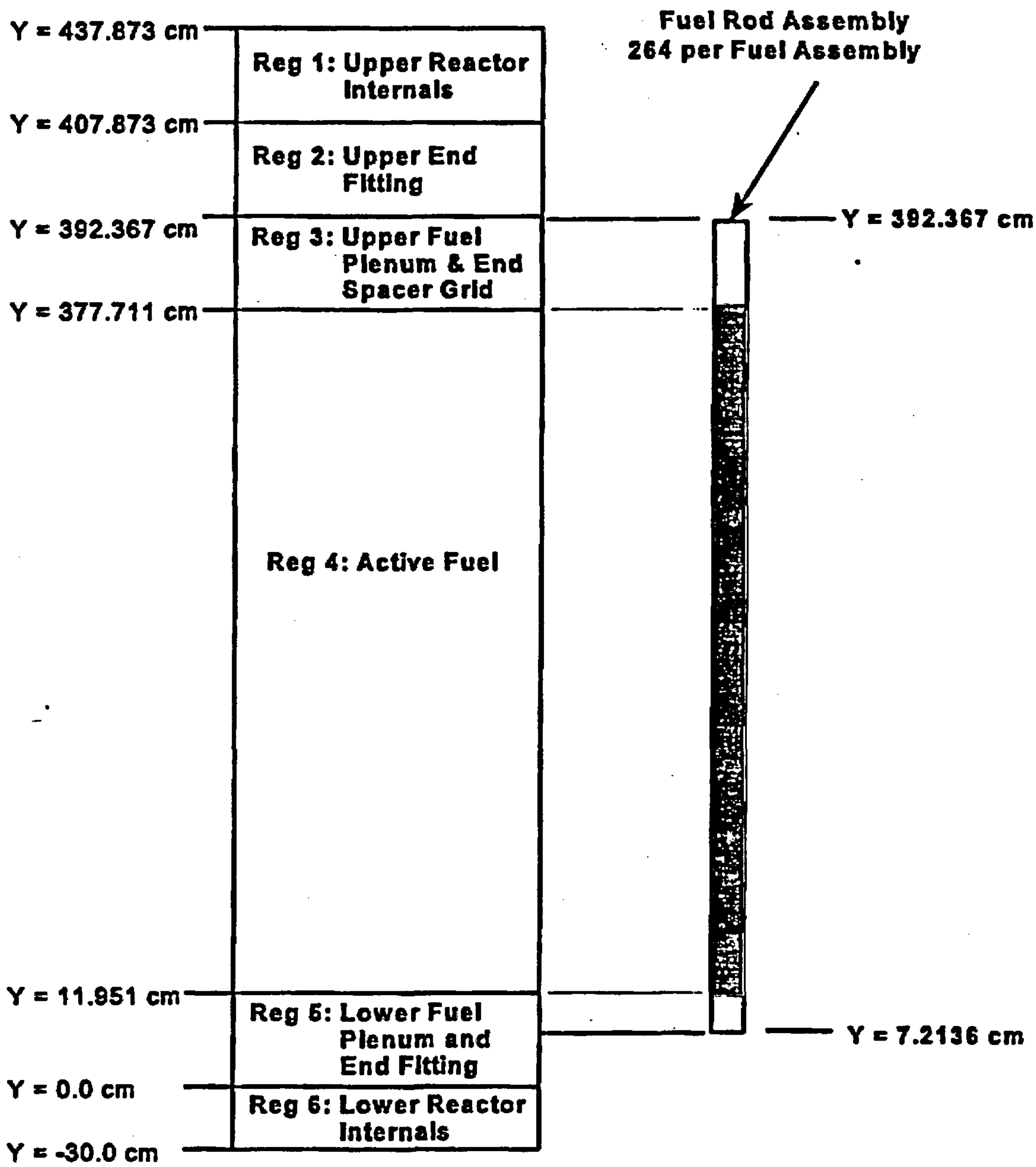


Figure 2-8 provides axial dimensions for a fully inserted (0% withdrawn) control rod for a Westinghouse 17×17 standard fuel assembly.

RCCA Materials/Dimensions:

Lower cap - stainless steel (diameter = 0.96774 cm)

Cladding - stainless steel (Clad OD = 0.96774 cm, Clad ID (inner diameter) = 0.87376 cm)

Absorber - Ag-In-Cd (diameter = 0.86614 cm)

Spacer - stainless steel (diameter = 0.8585 cm)

Upper plenum/spring area - Volume Fractions:	Clad - Stainless Steel	= 0.1848
	Spring - Inconel	= 0.2784
	Gas	= 0.5368

Upper cap - stainless steel (diameter = 0.96774 cm)

Upper stem - stainless steel (diameter = 0.5563 cm)

RCCA Volume Fractions:

The control rods are represented explicitly in regions 2, 3, and 4. The remainder of materials (excluding fuel rods, instrument tube, and guide tubes) are homogenized in regions 1, 2, and 3. The volume fractions of these materials (including non-RCCA materials) for RCCAs with rods at 0 % withdrawn are given in Table 2-5.

Table 2-5. Volume Fractions for Assemblies with RCCAs (0% Withdrawn) - Regions 1 - 3

	Volume Fractions (Rods 0% WD)		
Region	SS	Inc	Water
1	0.1907	0.0035	0.8058
2	0.1444	0.0218	0.8338
3*	0.0031	0.0264	0.9705

* Region 3 volume fractions are the same as for non-control assemblies.

For fully withdrawn control rods (100% withdrawn) the volume fractions presented in Table 2-3 (for non-control assemblies) should be used.

Figure 2-8. Axial Dimensions for RCCAs (Rods 0% Withdrawn) for Westinghouse 17×17 Standard Fuel Assembly

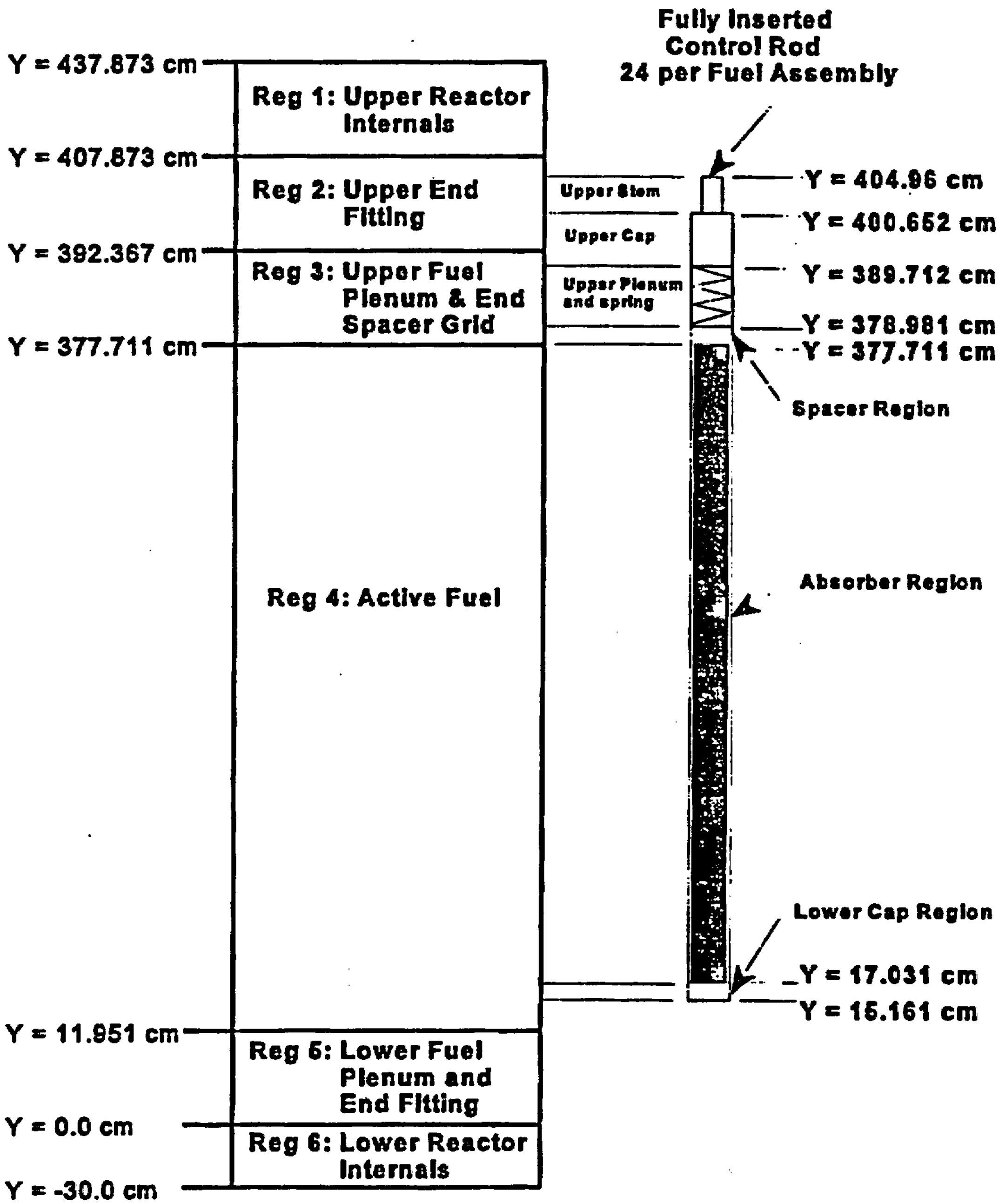


Figure 2-9 provides axial dimensions for the pyrex burnable absorber rod assembly for a Westinghouse 17×17 standard fuel assembly.

BPRA Materials/Dimensions:

Lower cap - stainless steel (diameter = 0.96774 cm)

Cladding - stainless steel   Outer tube - OD = 0.96774 cm, ID = 0.87376 cm
Inner tube - OD = 0.46101 cm, ID = 0.42799 cm

Absorber - B_2O_3-SiO_2 Pyrex tube - OD = 0.85344 cm, ID = 0.48260 cm

Upper plenum region - stainless steel clad (outer tube), helium gas in annulus

Upper cap - stainless steel (diameter = 0.96774 cm)

Upper stem - stainless steel (diameter = 0.54356 cm)

BPRA Volume Fractions:

The burnable poison and other materials inside the guide tubes are represented explicitly through region 3 and into region 2. This includes most of the upper end cap. The BPRA upper structure (beyond the end cap) is homogenized with the other assembly components within region 2. The volume fractions of these materials (including non-BPRA materials) are given in Table 2-6. There are 24 locations (guide tubes) for rod insertion in the fuel assembly. The number of burnable poison rods varies from 9 to 20 among the BPRAs for cycle 1 of McGuire 1. A thimble plug (Figure 2-10) is used for any empty location where a burnable poison rod is not installed.

Table 2-6. Volume Fractions for Assemblies with BPRAs - Regions 2 - 3

	Volume Fractions BPRAs		
Region	SS	Inc	Water
2	0.1649	0.0228	0.8123
3*	0.0031	0.0264	0.9705

* Region 3 volume fractions are the same as for non-control assemblies.

Figure 2-9. Axial Dimensions for Pyrex BPRAs for Westinghouse 17×17 Standard Fuel Assembly

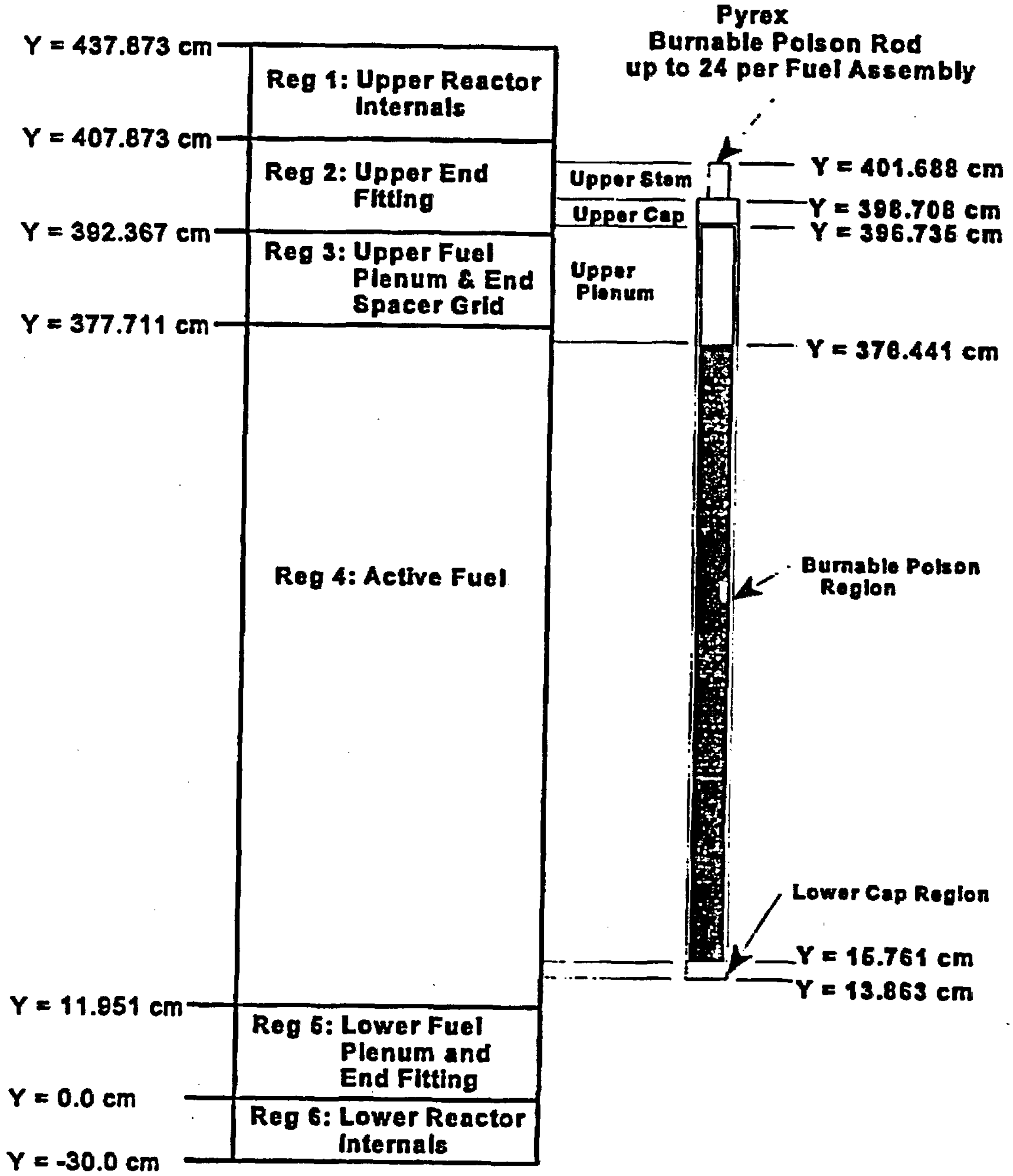
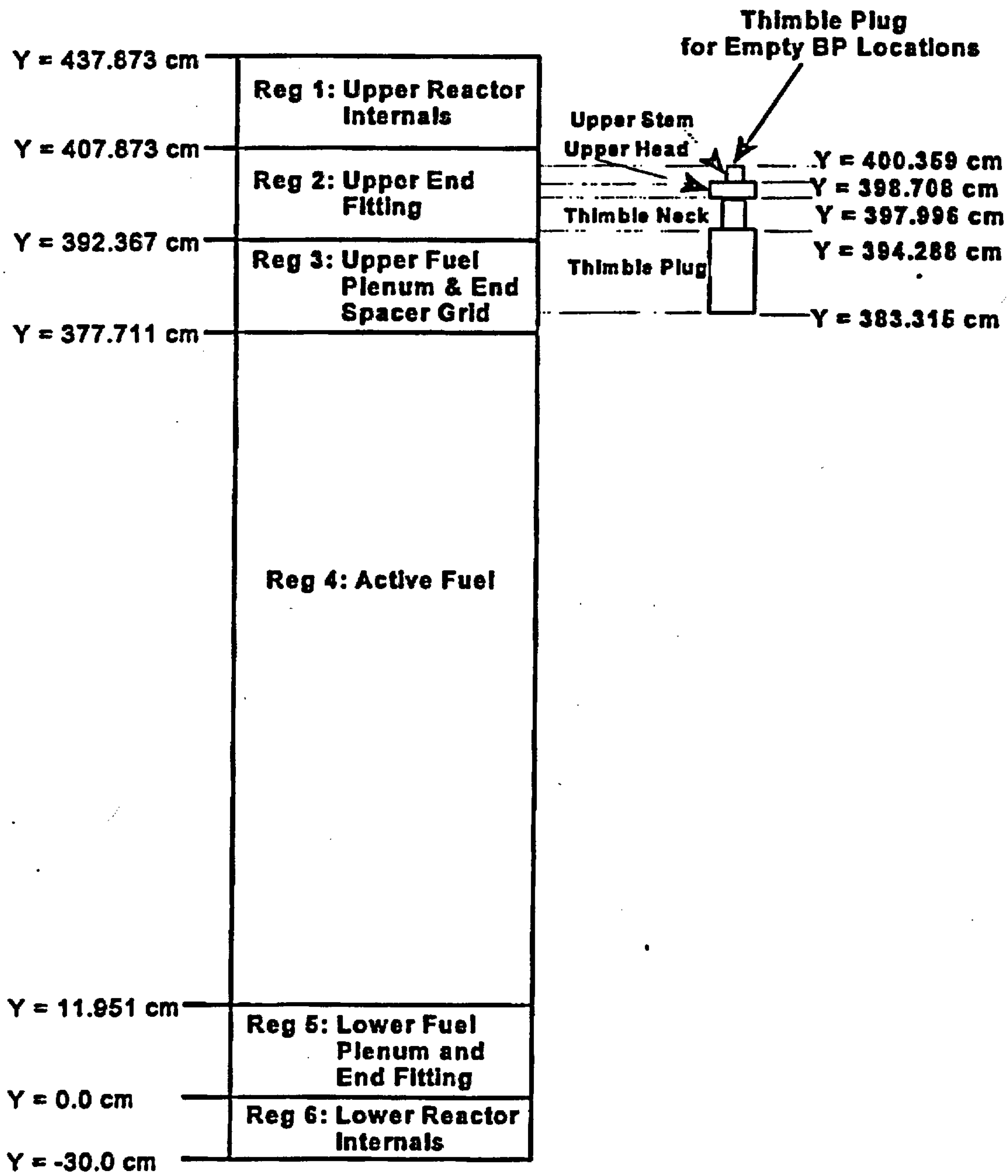


Figure 2-10. Axial Dimensions for Thimble Plug for Westinghouse 17×17 Standard Fuel Assembly



Thimble Plug Materials/Dimensions:

Thimble plug - stainless steel (diameter = 1.08204 cm)

Thimble neck - stainless steel (diameter = 0.4826 cm)

Upper head - stainless steel (diameter = 0.96774 cm)

Upper stem - stainless steel (diameter = 0.54356 cm)

3.0 FUEL CYCLE DESIGN INFORMATION

This section provides fuel assembly design data for cycle 1 of the McGuire Unit 1 reactor. Fuel assembly design data for later cycles will be provided in a revision to this report. Material and geometry data for the fuel assembly components are presented in Section 3.1. The fuel assembly locations for cycle 1, fuel enrichments and number of burnable absorber rods for each assembly, and control rod bank locations are presented in Section 3.2.

3.1 Fuel Batch Data

Material and geometry data for each fresh fuel batch present in cycle 1 are given in Table 3-1. This includes the fuel assembly type, the enrichment and kilograms of uranium in each fuel assembly (by batch), the diameter of the fuel pellets, and the type of fuel assembly grid material. The radial dimensions of the fuel clad, instrument tube, and guide tube are also presented. In addition, material and radial dimensions for RCCAs and BPRAs are provided. This data should be used in modeling each fuel assembly type for burnup calculations and the reactor criticality calculations for the statepoints defined in Table 3-2.

The length of each fuel cycle, expressed as effective-full-power-days (EFPD), is provided in Table 3-2. The time during each cycle where statepoint criticality data was measured is also presented.

Table 3-1. Fuel Assembly/Pin/Cycle Description for Cycle 1

Cycle	Fresh Fuel Batch	Assembly Type	wt % U235	kgU/ Assembly	FP Pellet OD (cm)	FP Clad OD (cm)	FP Clad ID (cm)	FA Grid Material	BPRA Type
1	1	Std	2.108	458.93	0.81915	0.94996	0.83566	Inconel	Pyrex
	2	Std	2.601	458.97	0.81915	0.94996	0.83566	Inconel	Pyrex
	3	Std	3.106	460.39	0.81915	0.94996	0.83566	Inconel	Pyrex

FP - Fuel Pin; FA - Fuel Assembly; BP - Burnable Poison

Description	Material	OD (cm)	ID (cm)
Instrument Tube	Zircaloy	1.22428	1.143
Guide Tube (Upper Region)	Zircaloy	1.22428	1.143
(Lower Region)	Zircaloy	1.08966	1.00838

RCCAs

Pellet Material	Ag-In-Cd
Fraction of Pellet Materials	Ag(80%), In(15.0%), Cd(5.0%)
Pellet Density	10.16 g/cc
Pellet OD	0.86614 cm
Clad Material	SS304
Clad OD	0.96774 cm
Clad ID	0.87376 cm

BPRAS (Annular - Pyrex tubes)

Material	B_2O_3-SiO_2
Density	2.25 g/cc
Pyrex OD	0.85344 cm
Pyrex ID	0.4826 cm
Clad Material	SS304
Outer Clad OD	0.96774 cm
Outer Clad ID	0.87376 cm
Inner Clad OD	0.46101 cm
Inner Clad ID	0.42799 cm

Table 3-2. Cycle Length and Time During Cycle Statepoint Data Measured for Cycles 1-7

Cycle	End-of-Cycle EFPD	Statepoint Number*	Time of Measurement EFPD
1	401.4	SP46	0.0
2	268.0	-	-
3	288.5	-	-
4	300.0	-	-
5	316.3	-	-
6	298.0	SP47	0.0
		SP48	62.4
7	408.0	SP49	0.0
		SP50	129.0
		SP51	282.3

* The unique statepoint numbers SP46, SP47, SP48, SP49, SP50, and SP51 are assigned to McGuire Unit 1 data.

3.2 Fuel Assembly Data

The fuel assembly loadings for cycle 1 are presented in Figure 3-1. (Fuel assembly loadings for later cycles will be presented in the next revision to this report.) A one-eighth core representation is used, where the fuel assembly at the center of the core is in location H8. The enrichment of U-235 (by batch), the locations of BPRAs, and number of burnable poison (BP) rods in each, and the location of the various control rod banks are also presented. The fuel assemblies with BPRAs may contain different number of BP rods (i.e., 9, 10, 12, 16, or 20 BP rods). The location of these BP rods in a fuel assembly along with the orientation of the assembly in the reactor core is presented in Figure 3-2.

Figure 3-1. Cycle 1 Core Loading

1/8 CORE LOADING FOR MCGUIRE UNIT 1, CYCLE 1
Westinghouse 193FA Core (17 x 17 Array)

	H	G	F	E	D	C	B	A
8	F(1) 1	F(1) 2	F(1) 1	F(1) 2	F(1) 1	F(1) 2	F(1) 1	F(1) 3
	9	F(1) 1	F(1) 2	F(1) 1	F(1) 2	F(1) 1	F(1) 3	F(1) 3
		10	F(1) 1	F(1) 2	F(1) 1	F(1) 2	F(1) 1	F(1) 3
			11	F(1) 1	F(1) 2	F(1) 1	F(1) 3	F(1) 3
				12	F(1) 2	F(1) 2	F(1) 3	
					13	F(1) 3	F(1) 3	

Batches 1, 2, & 3 are STD assy

CR	= Previous FA position Column/Row (C/R) - 1/8 Core
F	= Cycle FA was Fresh (F)
B	= Fuel Batch (B)

Cycle	Batch	WT%	KgU/FA
1	1	2.108	458.93
	2	2.601	458.97
	3	3.106	460.39

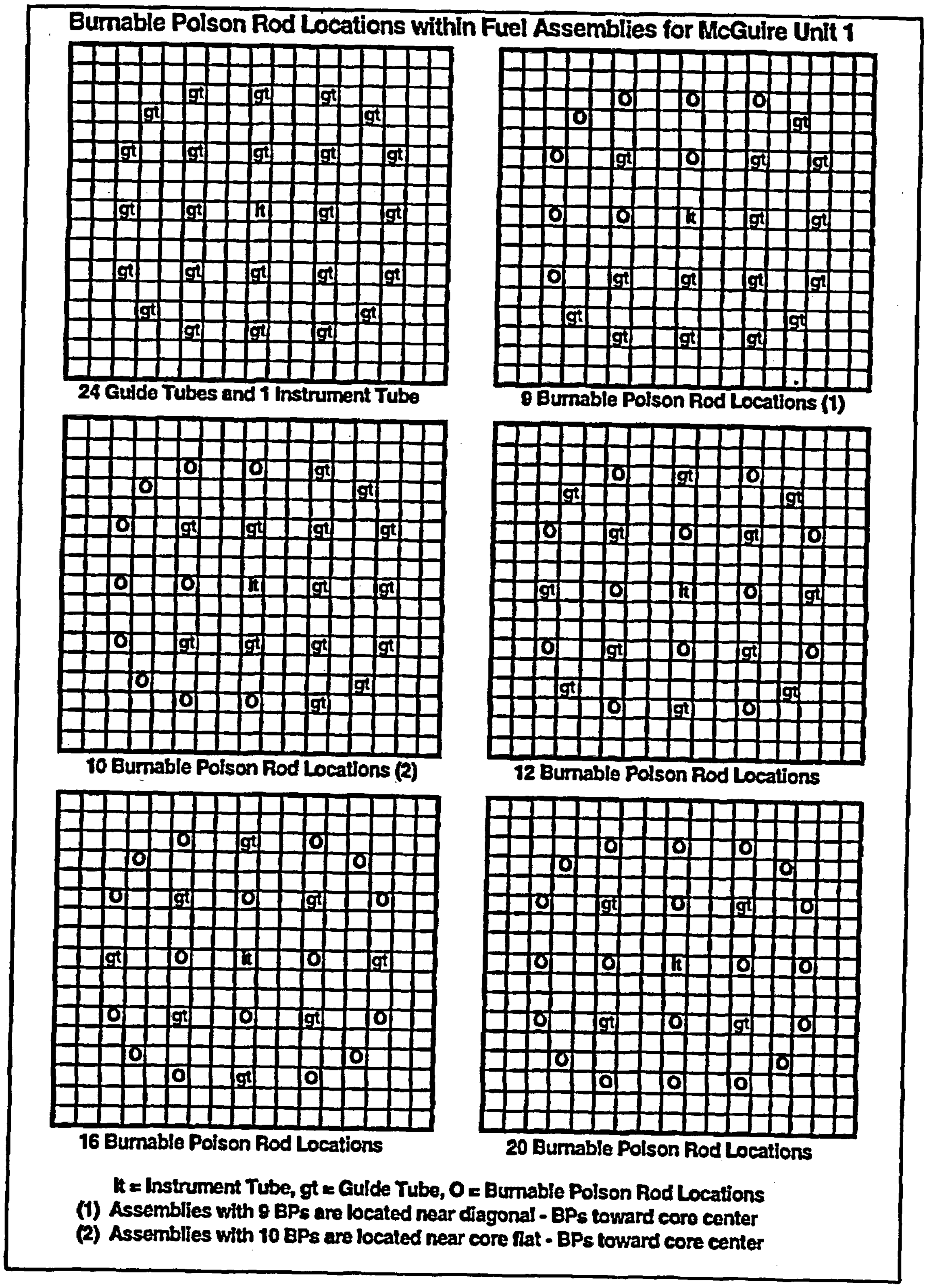
SA = Shutdown Bank A; B12
SB = Shutdown Bank B; C09
SC/SD = Shutdown Bank C/D; C11
SE = Shutdown Bank E; D08
CA = Control Bank A; F08
CB = Control Bank B; B10
CC = Control Bank C; B08, F10
CD = Control Bank D; H08, D12

BPRA Locations
(X)BP = BP Rods at X number of locations
9BP; B13
10BP; A8, A10
12BP; B11
16BP; E8, C8, D9, E10, C10
20BP; G8, F9, B9, D11, C12

All BP Rods have Boron loading of 12.5 wt % B2O3 in glass rod. Weight of Boron is 0.000419 lb/ft (0.006236 g/cm)

Note: All burnable poison rod assemblies (BPRA) assemblies were removed during steam generator repair outage (some BP holdown springs broken) at 191.4 EFPD.

Figure 3-2. Burnable Poison Rod Locations



4.0 CORE OPERATIONS AND STATEPOINT INFORMATION

This section provides core operations data for the burnup calculations required to generate isotopic concentrations for the statepoint evaluations. The measured critical conditions for the statepoints evaluated are also contained in this section.

4.1 Core Follow Data

The use of commercial reactor criticality data for model validation requires detailed knowledge of how the reactor was operated for the lifetime of every fuel assembly contributing to the criticality database. This is necessary in order to adequately model the conditions for burnup calculations at each axial location of each fuel assembly represented in the reactor core for each statepoint evaluation. Thus, core follow calculations based on core operation data are used to provide local conditions as a function of time to be used for all burnup calculations performed in support of the statepoint evaluations. In addition, measured global data such as rod insertions and boron letdown data are also provided.

The core follow calculations provide three-dimensional thermal-hydraulic (TH) feedback and burnup data. (This data will be provided in the next revision to this report.)

4.2 Statepoint Critical Condition Measurements

Measured critical conditions for 6 reactor startups (or statepoints) are provided in Table 4-1. The data includes the initial startup of the reactor or beginning-of-life (BOL), the beginning-of-cycle (BOC) of reload cycles 6 and 7, and three reactor restarts during cycles 6 and 7 of McGuire Unit 1. The cycle and statepoint number, along with the EFPDs during the cycle for which the startup occurred, is provided. The elapsed time (in hours) since the reactor was shutdown (downtime) prior to the startup is also given for each statepoint. In addition, Table 4-1 provides the measured soluble boron concentration (ppmB), rod bank positions, and temperature of the moderator or coolant in the reactor (for each statepoint) when criticality was achieved.

Table 4-2 provides shutdown and startup dates for each cycle and statepoint. The cycle shutdown and startup dates can be used in determining the downtime for fuel assemblies that are out of the reactor for one or more cycles and are then reinserted in a later cycle.

Table 4-1. Statepoint Data for McGuire Unit 1 - Measured Critical Conditions

Cycle(SP)	EFPD	Downtime (hours)	ppmB	Rod Positions, cm above bottom of fuel* Bk CA	Bk CB	Bk CC	Bk CD	T(coolant) (F)
1(SP46)	0.0	0	1279	WD	WD	313	129	558.9
6(SP47)	0.0	1872	1538	WD	WD	358	174	558.1
6(SP48)	62.4	1505	1320	WD	WD	315	131	557.9
7(SP49)	0.0	3120	1689	WD	WD	313	129	558.8
7(SP50)	129.0	711	1335	WD	WD	278	94	558.2
7(SP51)	282.3	451	931	WD	WD	WD	283	557

WD = Rod Withdrawn

* Measured from the bottom of active fuel region to bottom of control rod absorber region (See Figure 2-8).

Table 4-2. Statepoint Data for McGuire Unit 1 - Shutdown and Startup Dates

Cycle(SP)	EFPD	Shutdown Date	Startup Date
1(SP46)	0.0	-	08 Aug 1981
2(-)*	0.0	24 Feb 1984	28 Apr 1984
3(-)*	0.0	19 Apr 1985	24 Jun 1985
4(-)*	0.0	16 May 1986	07 Sep 1986
5(-)*	0.0	04 Sep 1987	12 Nov 1987
6(SP47)*	0.0	12 Oct 1988	29 Dec 1988
6(SP48)	62.4	07 Mar 1989	09 May 1989
7(SP49)*	0.0	08 Jan 1990	18 May 1990
7(SP50)	129.0	15 Oct 1990	14 Nov 1990
7(SP51)	282.3	25 Apr 1991	14 May 1991
	408.0 (EOC)	20 Sep 1991	

EOC = end-of-cycle

* Shutdown date is for previous cycle.

5.0 REFERENCES

1. *Quality Assurance Program for Framatome Cogema Fuels*, Document Number: 56-1177617-04, FCF, August 5, 1996.

2. Q-List, YMP/90-55Q, REV 04, Yucca Mountain Site Characterization Project.

3. QAP-2-0 Activity Evaluations, ID No. WP-06, *Develop Technical Documents*, Civilian Radioactive Waste Management System M&O, August 3, 1997.

4. Quality Assurance Requirements and Description, DOE/RW-0333P, REV 7, DOE OCRWM.

5. *McGuire 1 NEMO Depletion and Statepoints (HLW)*, Document Number: 32-1267111-00, FCF.