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Michael E. Dunn

Date

July 15, 1994

**Nuclear Criticality Safety Evaluation of the
 ^{233}U Inventory at the Oak Ridge National Laboratory
Using ENDF/B-V Cross Sections**

A Thesis

Presented for the

Master of Science

Degree

The University of Tennessee, Knoxville

Michael E. Dunn

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Dedication

This thesis, which would not be possible without their love and support, is dedicated to my mother, Virginia C. Dunn, and father, Edward Earl Dunn.

The forbearing use of power does not only form a touchstone, but the manner in which an individual enjoys certain advantages over others is a test of a true gentleman.

The power which the strong have over the weak, the employer over the employed, the educated over the unlettered, the experienced over the confiding, even the clever over the silly—the forbearing or inoffensive use of all this power or authority, or a total abstinence from it when the case admits it, will show the gentleman in a plain light.

The gentleman does not needlessly and unnecessarily remind an offender of a wrong he may have committed against him. He cannot only forgive, he can forget; and he strives for that nobleness of self and mildness of character which impart sufficient strength to let the past be but the past. A true man of honor feels humbled himself when he cannot help humbling others.

*— Robert E. Lee
(1807-1870)*

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Abstract

This work presents a nuclear criticality safety evaluation of the ^{233}U inventory which is stored in two different facilities at the Oak Ridge National Laboratory (ORNL). The evaluation employs KENO V.a with ENDF/B-V cross sections. The inventory is housed in retrievable storage wells within Building 3019 and three fuel drain tanks within the Molten Salt Reactor (MSR) Fuel Drain Tank Cell (FDTC) at ORNL. In order to establish subcritical storage limits on the basis of calculations, the calculational method (e.g., Monte Carlo) and ENDF/B-V cross section data were validated against ^{233}U critical experiments. Once validated, KENO V.a was used to evaluate the storage wells located in Building 3019. The calculated k_{eff} for the nominal configuration of the wells is 0.8438 ± 0.0018 which is sufficiently subcritical. Moreover, the calculated storage well loading limits for ^{233}U are 2.45, 1.90, 1.75, 1.30 and 0.95 kg U/ft for H/U atom ratios of 0, 3, 5, 10 and 20, respectively. After examining the storage wells, the FDTC was evaluated using KENO V.a, and the calculated k_{eff} of the nominal configuration of the three drain tanks is 0.8432 ± 0.0028 which is sufficiently subcritical. The FDTC was also evaluated for credible upset conditions using KENO V.a. The results of this evaluation indicate that the facility is sufficiently subcritical during the upset conditions.

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Chapter One

Introduction

1.1 Background

At present, the Oak Ridge National Laboratory (ORNL) has one of the two largest inventories of ^{233}U in the western world.⁴⁰ The ^{233}U is stored in various chemical and physical forms at the Molten Salt Reactor (MSR) site and in Building 3019 at ORNL. If an adequate amount of ^{233}U somehow attains an unsafe geometrical configuration (e.g., due to an earthquake) a nuclear criticality accident might occur. In order to complete a safety analysis report for the above storage locations, a nuclear criticality safety evaluation must be performed to assure that the current storage is effective in maintaining subcriticality during normal and credible abnormal conditions.

Nuclear criticality safety evaluations at ORNL usually utilize KENO V.a, a Monte Carlo code,¹ with various cross section libraries to perform nuclear criticality safety studies. KENO V.a is part of the Standardized Computer Analyses for Licensing Evaluation (SCALE) system.² Included in the SCALE system are the Hansen-Roach 16-energy group and ENDF/B-IV 27- & 218-energy group neutron cross section libraries. Recently, the ENDF/B-V 44- & 238-energy group cross section libraries have become available; however, neither have been validated for ^{233}U systems. In order to use either ENDF/B-V library with the SCALE system, the library must be validated against experimental data.

1.2 Objectives

The purpose of this work is to perform a computational validation and nuclear criticality safety evaluation of the ^{233}U inventory using KENO V.a with the ENDF/B-V based neutron cross sections (henceforth referred to as ENDF/B-V cross sections) to demonstrate subcriticality for both normal and abnormal conditions of ^{233}U storage at ORNL. In order to evaluate the subcriticality of the ^{233}U -uranium inventory in storage, the research was separated into three phases.

The first phase consisted of validating the ENDF/B-V cross section library for ^{233}U and associated materials using KENO V.a. The validation of ENDF/B-V cross sections involved calculating previous critical experiments involving ^{233}U and similar materials as found in the storage facilities with various H/ ^{233}U atom ratios.

Once the cross sections and modules were validated, the goal of the second phase was to use KENO V.a with the validated ENDF/B-V cross sections to perform a nuclear criticality safety evaluation of the inventory of ^{233}U located in the Building 3019 storage wells. In the final phase of the research, a criticality safety evaluation of the MSR fuel salt, which is stored in three drain tanks in Building 7503, was performed.

1.3 Previous Work

The work is useful because criticality safety evaluations are needed to demonstrate the effectiveness and validity of using the ENDF/B-V cross sections for applications at

ORNL. In addition, the validation of the ENDF/B-V ^{233}U neutron cross section library with KENO V.a is needed to document the accuracy of this cross section library with regard to ^{233}U systems. With respect to originality, this work is original in that no criticality safety evaluations of the uranium inventories at both of the above locations have been performed using ENDF/B-V cross sections.

Although the above facilities have not been evaluated with the ENDF/B-V library, it should be noted that a criticality analysis for the proposed storage of mixed oxide waste in Building 3019 has been performed by Primm³ using KENO V.a with the 16 group Hansen-Roach cross section library. In addition, a similar analysis of the current fissile material inventory in Building 3019 was also performed by Primm¹⁵ using the Hansen-Roach library. Also, Hollenbach and Hopper¹⁶ have performed a criticality safety study of the MSR Fuel Drain Tanks using KENO V.a with the 27 group ENDF/B-IV cross section library.

One primary purpose of this work is to perform an independent criticality safety evaluation of the Building 3019 and MSR ^{233}U inventory. This evaluation is intended to support the previous studies and provide new information about the two storage configurations using the ENDF/B-V cross section library.

To accomplish this task, independent models of the two storage locations were developed using KENO V.a. Several calculations were performed with these models to

determine the margin of subcriticality associated with each storage facility. Whenever possible, comparisons were made with the previous studies.

1.4 Scope and Organization

This report is composed of seven chapters. The following chapter presents the validation of the 238 and 44 energy group ENDF/B-V neutron cross section libraries for ^{233}U systems using KENO V.a. Based on the validation study, an acceptable subcritical limit is derived for both libraries. These subcritical limits are used in the criticality safety evaluations of the storage facilities.

Chapter Three presents a comprehensive description of the ^{233}U inventory stored in Building 3019. In addition, Chapter Three presents the development of the facility model and the calculated subcritical margin for the current storage configuration along with the computed ^{233}U subcritical loading limits. Chapter Four compares the results from Chapter Three with the previous analysis¹⁵ of the Building 3019 storage wells and evaluates the sensitivity regarding the derived subcritical loading limits for ^{233}U .

Chapter Five presents a detailed description of the MSR fuel salt stored in the Fuel Drain Tank Cell of Building 7503. Moreover, Chapter Five discusses the development of the facility model and the calculated subcritical margin for the nominal and credible accident storage configurations. A comparison to the previous study¹⁶ is also presented in Chapter Five. Chapter Six presents the sensitivity studies regarding the

KENO V.a model of the MSR facility. Finally, Chapter Seven provides a summary of the results, conclusions and suggestions for future work.

Chapter Two

Validation of KENO V.a and ENDF/B-V Cross Sections

2.1 Introduction

In general, operation and storage limits for ^{233}U material outside of reactors are provided by the American National Standards Institute (ANSI) in ANS-8.1-1983;R88 and ANS-8.7-1982;R87, respectively. Since the limits imposed by the above ANSI standards are generalized limits, they can be overly restrictive for certain operation and storage conditions. In order to circumvent this issue, the standards permit the establishment of subcritical limits based upon applicable experimental data. However, in most cases experimental data which replicates the actual storage configuration of the fissile material is not available. In this case the ANSI standards permit the establishment of subcritical limits based upon calculations which have been shown to be valid when compared with experimental data.

In the criticality safety community, there is a variety of cross section libraries and computer codes based upon particular calculational methods such as Monte Carlo and Discrete Ordinates available to the criticality safety engineer. When subcritical limits are to be established by using a particular code and cross section library, the computer code and library must be validated against experimental data. It is desirable to select experimental data having materials and neutron energy spectra similar to that which exists in the actual fissile storage configuration. However, it is improbable that the experimental database will resemble the actual configuration exactly. Consequently, the

available experimental data will be deficient in some areas. Therefore, caution should be used in extending calculations beyond the scope of the experimental data. Once the calculational method and data libraries are validated against applicable experimental data, subcritical limits for a particular fissile material can be established.

2.2 Calculational Methods

The Monte Carlo method is widely used both in the United States and abroad within the field of nuclear criticality safety to calculate the effective multiplication factor of fissile systems. As noted in Chapter One, KENO V.a is maintained and operated at ORNL as part of the SCALE system. All of the criticality safety calculations described in this work were performed with the multigroup Monte Carlo code KENO V.a as maintained within the SCALE-IV system as part of the criticality safety sequence CSAS25. Furthermore, all of the calculations were performed on an IBM RISC 6000 model 580 workstation. Prior to performing criticality calculations using KENO V.a, resonance processing in the unresolved and resolved energy ranges were performed using the BONAMI and NITAWL-II modules, respectively, in the SCALE system.³⁸ The BONAMI module processes cross section data in the unresolved range using the Bondarenko shielding factor method, and the NITAWL-II module employs the Nordheim Integral Treatment to process cross section data in the resolved range.

Included in the SCALE system are the 238 and 44 energy group ENDF/B-V cross section libraries referred to as 238GROUPNDF5 and 44GROUPNDF5, respectively. The 238 group library was generated from the continuous energy spectrum and consists of 90 thermal and 148 epi-thermal energy groups. The 44 group library was collapsed from the 238 group library by using the cell spectra of a fuel rod from a 17 x 17 Westinghouse LWR fuel assembly.³⁸ The 44 group library consists of 22 thermal and 22 epi-thermal energy groups. A general comparison between the group structures of the ENDF/B-V, ENDF/B-IV, and Hansen-Roach libraries is shown in Table 2.1.1. A comprehensive energy group comparison between the ENDF cross section libraries is presented in Appendix A.

Table-2.1.1 General Group Structure Comparison Between ENDF/B-V, ENDF/B-IV and Hansen-Roach Cross Section Libraries.

Cross Section Library	Number of Thermal Groups	Thermal Group Numbers	Number of Fast Groups	Fast Group Numbers
238 ENDF/B-V	90	149 - 238	148	1 - 148
218 ENDF/B-IV	78	141 - 218	140	1 - 140
44 ENDF/B-V	22	23 - 44	22	1 - 22
27 ENDF/B-IV	13	15 - 27	14	1 - 14
Hansen-Roach	4	13 - 16	12	1 - 12

2.3 ^{233}U Experiments and Calculated Parameters

The data base for critical ^{233}U experiments is very limited in comparison with the experimental data available for ^{235}U and plutonium. The shortage of experimental data for ^{233}U systems is primarily due to the large demands for ^{235}U and plutonium in past years. Another explanation for the shortage of experimental data is due to the handling difficulties associated with the radioactive daughter products from ^{232}U which usually accompanies ^{233}U in trace amounts. A review of the literature⁴⁻⁸ shows there is a shortage of critical experimental data for low moderated systems particularly in the $1 \leq \text{H}/^{233}\text{U} \leq 40$ range. In particular, McNeany and Jenkins⁷ indicate that the most reactive configurations of ^{233}U solutions occur for cylinders and spheres with $\text{H}/^{233}\text{U} \approx 40$ and slabs with $\text{H}/\text{U} \approx 25$. Furthermore, the experimental data is limited to aqueous uranyl nitrate ($\text{UO}_2(\text{NO}_3)_2$), aqueous uranyl fluoride (UO_2F_2) and metal systems. Since the database is relatively small, ^{233}U critical experiments were selected without attempting to determine if there are strong similarities between the experimental data and the configurations of the 3019 and MSR facilities.

Approximately 100 experiments were identified at the start of this work to be included as part of the validation study; however, a few of the experiments are extrapolations to critical based upon subcritical configurations. These extrapolated experiments are not included in the study. In addition, some experiments are not modeled due to redundancies in the $\text{H}/^{233}\text{U}$ ratios between particular experiments. Subsequently, 51 of the 100 original experiments were selected for the validation study.

The critical experiments have H/²³³U ratios ranging from 0 to 1986. A complete description of these experiments is presented in five references⁴⁻⁸. A brief overview of the experiments is presented in Table 2.3.1.

In the sections that follow (i.e. Sections 2.3.1 - 2.3.3), the chemical and geometrical parameters for the 51 critical experiments are presented. Using the chemical parameters, the atomic number densities were calculated and are presented in Appendix B. Moreover, the number densities and geometrical parameters were used to develop KENO V.a input files modeling each of the critical experiments. The effective multiplication factor (k_{eff}) was calculated for each experiment using the 238 and 44 group ENDF/B-V, 218 and 27 group ENDF/B-IV, and 16 group Hansen-Roach cross section libraries. In sections 2.3.1 - 2.3.3, the results obtained with the ENDF libraries are presented while the results obtained with the Hansen-Roach library are presented in Appendix C.

Table 2.3.1: Selected ^{233}U Critical Experiments for Validation.

Chemical Form	Geometrical Description	Number of Experiments
Aqueous $\text{UO}_2(\text{NO}_3)_2$	cylindrical reflected	17
Aqueous $\text{UO}_2(\text{NO}_3)_2$	cylindrical unreflected	2
Aqueous $\text{UO}_2(\text{NO}_3)_2$	spherical unreflected	7
Aqueous $\text{UO}_2(\text{NO}_3)_2$	cylinders in reflected and unreflected cubic arrays	4
Aqueous UO_2F_2	cylindrical reflected	4
Aqueous UO_2F_2	spheres reflected	7
Aqueous UO_2F_2	spheres unreflected	1
^{233}U metal	spheres reflected	7
^{233}U metal	spheres unreflected	2
	Total	51

2.3.1 Aqueous Uranyl Nitrate Experiments

Several critical experiments involving aqueous uranyl nitrate solutions were performed by Fox, Gilley, and Rohrer.⁴ The isotopic percentages of uranium in these solutions are presented in Table 2.3.1.1. Moreover, the critical parameters for cylinders of aqueous $^{233}\text{UO}_2(\text{NO}_3)_2$ are presented in Table 2.3.1.2.

Table 2.3.1.1: Isotopic Percentages of Uranium in $\text{UO}_2(\text{NO}_3)_2$.

Isotope	Wt%
^{233}U	98.7
^{234}U	0.50
^{235}U	0.01
^{238}U	0.79

Table 2.3.1.2: Critical Parameters for Paraffin Reflected* Cylinders of Aqueous $\text{UO}_2(\text{NO}_3)_2$ Solutions (Ref. 4).

Exp #	H/U ²³³	Soln. ht. (cm)	Diam. (cm)	Cyl.** Vol. (cm ³)	Mass of ²³³ U (g)
4.2	67.0	29.0	15.1	5200***	1750
4.9	145.0	18.6	19.1	5300	890
4.11	57.5	14.4	20.5	4750	1810
4.12	84.4	14.7	20.5	4850	1330
4.13	120.0	16.4	20.5	5400	1070
4.15	151	16.7	20.5	5510	880
4.16	193	18.8	20.5	6200	790
4.17	213	19.3	20.5	6370	750
4.19	247	19.4	21.5	7000	700
4.20	297	21.5	21.5	7780	650
4.21	356	21.3	22.9	8750	620
4.22	379	22.9	22.9	9390	620
4.23	394	19.3	25.5	9820	630
4.24	461	22.5	25.5	11450	630
4.27	582	21.1	30.5	15400	680
4.28	630	23.8	30.5	17400	700
4.29	757	30.4	30.5	22200	750

*Reflector thickness = 20.0 cm.

**Aluminum cylinder.

***No reflector on the top surface.

In addition, a series of experiments involving critical aqueous $^{233}\text{UO}_2(\text{NO}_3)_2$ solutions were performed by Gwin and Magnuson⁶ to measure the η of ^{233}U . The spherical critical parameters of uranium in the experiments provided in reference [6] are presented in Table 2.3.1.3. In a previous study of ENDF/B-IV and Hansen-Roach ^{233}U cross sections, twelve KENO IV calculations of critical ^{233}U systems were performed by Jenkins and McNeany.⁷ However, some of the ^{233}U experiments are documented in references [4] and [6] and have been presented in this document. Therefore, the remaining experiments of interest will be addressed at this point. The critical parameters for the experiments provided in reference [7] are presented in Table 2.3.1.4. Paxton and Pruvost⁸ presented an experimental study of $2 \times 2 \times 2$ and $3 \times 3 \times 3$ arrays of uranyl nitrate solution units which was originally performed by J. T. Thomas.¹² The critical parameters for the arrays of aqueous $^{233}\text{U}(97.5\%)\text{O}_2(\text{NO}_3)_2$ units provided by reference [8] are presented in Table 2.3.1.5.

Table 2.3.1.3: Critical Parameters for Spheres of Aqueous $\text{UO}_2(\text{NO}_3)_2$ (Ref. 6).

Exp. #	Diam* (cm)	H/ ^{233}U	Isotopic Percentages (wt%)			
			^{233}U	^{234}U	^{235}U	^{238}U
6.1	69.19	1533	97.7	1.62	0.04	0.64
6.2	69.19	1470	97.7	1.62	0.04	0.64
6.3	69.19	1417	97.7	1.62	0.04	0.64
6.4	69.19	1368	97.7	1.62	0.04	0.64
6.5	69.19	1324	97.7	1.62	0.04	0.64
6.6	122.02	1986	97.67	1.54	0.03	0.76

Table 2.3.1.4: Critical Parameters for Unreflected Aqueous $\text{UO}_2(\text{NO}_3)_2$ Experiments (Ref. 7).

Exp. #	Geometry	H/ ^{233}U	Critical Radius (cm)	Critical Height (cm)
7.2	cylinder	73	25.40	16.9
7.3	cylinder	119	12.70	24.69
7.4	sphere	195	14.58	---

*Container material is Al with a wall thickness = 0.3302 cm.

Table 2.3.1.5: Critical Parameters for Cubic Arrays of Cylindrical Units of Aqueous $^{233}\text{UO}_2(\text{NO}_3)_2$ Solution* (Ref 8).

Exp #	H/ ^{233}U	Number of Units**, Config.	Refl.	Refl. Thickness (cm)	Center Spacing (cm)	
					Horz	Vert
8.10	73	27, 3 x 3 x 3	none	---	25.7	24.6
8.11	73	8, 2 x 2 x 2	poly-ethylene	15.2	32.0	30.4
8.13	119	8, 2 x 2 x 2	none	---	20.2	18.8
8.16	119	27, 3 x 3 x 3	poly-ethylene	15.2	38.0	36.7

Using the critical parameters presented in Table 2.3.1.2 for reflected cylinders of aqueous $\text{UO}_2(\text{NO}_3)_2$, the effective multiplication factor (k_{eff}) was calculated for each experiment. The results are tabulated in Table 2.3.1.6. As shown in Table 2.3.1.6, the average energies in which fissions occur range from 1.04 to 0.1 eV (groups 186 to 216) and 1.02 to 0.1 eV (groups 181 to 208) for the 238 ENDF/B-V and 218 ENDF/B-IV libraries, respectively. Similarly, the average energy range for fissions occur from 0.325 to 0.05 eV (groups 30 to 37) and 0.8 to 0.05 eV (groups 20 to 24) for the 44 ENDF/B-V and 27 ENDF/B-IV libraries, respectively. There is a definite improvement in the agreement between the calculated k_{eff} 's and the experimental results (i.e. $k_{\text{eff}} = 1.0$)

* Uranium isotopic content (wt%): $^{233}\text{U} = 97.54$, $^{234}\text{U} = 1.047$, $^{235}\text{U} = 0.025$, $^{236}\text{U} = 0.001$, $^{238}\text{U} = 1.386$.

**Cylindrical Units of Stainless Steel: O.D. = 18.28 cm., outside height = 17.67 cm., wall thickness = 0.025 cm.

obtained with the newer ENDF/B-V libraries relative to ENDF/B-IV. Regarding the ENDF/B-V results, there appears to be a trend to predict k_{eff} slightly higher than 1.0 for all of the paraffin reflected cases except for the 238 group ENDF/B-V results for experiments 4.20 and 4.22.

Table 2.3.1.6: Calculated Effective Multiplication Factor for Paraffin Reflected Cylinders of Aqueous $\text{UO}_2(\text{NO}_3)_2$.

Exp. #	k_{eff} ENDF/B-V	Avg. Fission Group* (AFG)	k_{eff} ENDF/B-IV	Avg. Fission Group (AFG)
	238 Group 44 Group		218 Group 27 Group	
4.2	1.0025 ± 0.0040 0.9995 ± 0.0044	189.21 31.26	1.0180 ± 0.0047 1.0302 ± 0.0044	181.11 20.72
4.9	1.0087 ± 0.0041 1.0065 ± 0.0045	202.77 34.33	1.0245 ± 0.0043 1.0249 ± 0.0038	194.62 22.46
4.11	1.0157 ± 0.0042 1.0216 ± 0.0042	186.90 30.74	1.0338 ± 0.0049 1.0384 ± 0.0039	178.38 20.45
4.12	1.0222 ± 0.0048 1.0249 ± 0.0046	194.44 32.33	1.0351 ± 0.0040 1.0401 ± 0.0045	186.11 21.36
4.13	1.0243 ± 0.0046 1.0248 ± 0.0039	200.21 33.64	1.0337 ± 0.0047 1.0398 ± 0.0044	191.89 22.10
4.15	1.0022 ± 0.0038 1.0190 ± 0.0044	203.54 34.44	1.0193 ± 0.0038 1.0170 ± 0.0041	195.24 22.54
4.16	1.0158 ± 0.0039 1.0099 ± 0.0037	206.52 35.12	1.0321 ± 0.0042 1.0229 ± 0.0039	198.16 22.93
4.17	1.0112 ± 0.0044 1.0113 ± 0.0044	207.54 35.38	1.0293 ± 0.0047 1.0166 ± 0.0042	199.22 23.09
4.19	1.0008 ± 0.0038 1.0027 ± 0.0038	208.98 35.76	1.0117 ± 0.0042 1.0217 ± 0.0038	200.67 23.30
4.20	0.9916 ± 0.0042 1.0039 ± 0.0039	210.74 36.17	1.0075 ± 0.0037 1.0010 ± 0.0042	202.39 23.54

*Average energy group in which fissions are occurring

Table 2.3.1.6 (Cont'd)

Exp. #	k_{eff} ENDF/B-V	Avg. Fission Group (AFG)	k_{eff} ENDF/B-IV	Avg. Fission Group (AFG)
	238 Group 44 Group		218 Group 27 Group	
4.21	1.0001 ± 0.0040	212.10 36.51	1.0118 ± 0.0036	203.82 23.73
	1.0012 ± 0.0039		1.0161 ± 0.0036	
4.22	0.9970 ± 0.0034	212.50 36.65	1.0058 ± 0.0036	204.30 23.79
	1.0000 ± 0.0034		1.0104 ± 0.0038	
4.23	1.0078 ± 0.0038	212.83 36.69	1.0169 ± 0.0036	204.60 23.83
	1.0013 ± 0.0043		1.0230 ± 0.0039	
4.24	1.0012 ± 0.0031	213.83 36.95	1.0161 ± 0.0038	205.54 23.97
	1.0120 ± 0.0037		1.0183 ± 0.0035	
4.27	1.0129 ± 0.0038	215.14 37.28	1.0227 ± 0.0037	206.87 24.15
	1.0050 ± 0.0031		1.0224 ± 0.0028	
4.28	1.0042 ± 0.0034	215.49 37.37	1.0172 ± 0.0035	207.20 24.21
	1.0096 ± 0.0037		1.0259 ± 0.0035	
4.29	1.0049 ± 0.0030	216.36 37.57	1.0117 ± 0.0034	208.06 24.32
	1.0054 ± 0.0038		1.0120 ± 0.0032	

Likewise, the critical parameters for the unreflected uranyl nitrate experiments presented by Gwin and Magnuson⁶ in Table 2.3.1.3 were modeled with KENO V.a. The results obtained with the ENDF/B-V and ENDF/B-IV libraries are tabulated in Table 2.3.1.7. Unlike the paraffin reflected cylinders in Table 2.3.1.6, a trend exists to predict k_{eff} slightly below (i.e. within 1%) 1.0 for the spheres shown in Table 2.3.1.7. Furthermore, there is no significant improvement in the calculated results obtained with ENDF/B-V relative to ENDF/B-IV. As indicated by the average fission groups in Table 2.3.1.7, the neutron spectrum is very thermalized which is primarily due to the large volumes of dilute uranyl nitrate solution present in the spheres.

Table 2.3.1.7: Calculated Effective Multiplication Factors for Unreflected Spheres of Aqueous $\text{UO}_2(\text{NO}_3)_2$.

Exp. #	k_{eff} ENDF/B-V	Avg. Fission Group (AFG)	k_{eff} ENDF/B-IV	Avg. Fission Group (AFG)
	238 Group 44 Group		218 Group 27 Group	
6.1	0.9943 ± 0.0027 0.9938 ± 0.0025	218.41 38.11	0.9987 ± 0.0023 1.0006 ± 0.0025	210.15 24.63
6.2	0.9944 ± 0.0021 0.9938 ± 0.0024	218.22 38.06	0.9996 ± 0.0023 0.9998 ± 0.0026	209.97 24.60
6.3	0.9970 ± 0.0024 0.9982 ± 0.0023	218.04 38.01	0.9987 ± 0.0024 0.9994 ± 0.0023	209.77 24.57
6.4	0.9950 ± 0.0024 0.9967 ± 0.0024	217.86 37.96	0.9992 ± 0.0023 1.0036 ± 0.0022	209.59 24.55
6.5	0.9929 ± 0.0022 0.9934 ± 0.0025	217.70 37.92	1.0016 ± 0.0023 1.0036 ± 0.0021	209.43 24.52
6.6	0.9967 ± 0.0018 0.9970 ± 0.0017	219.11 38.28	0.9966 ± 0.0016 0.9937 ± 0.0017	210.83 24.73

After modeling the unreflected spheres of uranyl nitrate, the experiments presented by Jenkins and McNeany⁷ in Table-2.3.1.4 were modeled, and the results are tabulated in Table-2.3.1.8. The average fission energies range from 1.09 to 0.5 eV and 1.1 to 0.5 eV for the 238 group ENDF/B-V and 218 group ENDF/B-IV libraries, respectively. In contrast, the average fission energies range from .35 to .15 eV and 0.8 to .225 eV for the 44 group ENDF/B-V and 27 group ENDF/B-IV libraries, respectively. Regarding experiment 7.2, there is a significant improvement ($\sim 3\%$) in the calculated k_{eff} obtained with the ENDF/B-V libraries relative to ENDF/B-IV. As the $\text{H}/^{233}\text{U}$ increases to 119 in experiment 7.3, the agreement between the ENDF libraries

improves. It should be noted that the results obtained with the Hansen-Roach library (see Appendix C) are closer to 1.0 for experiment 7.2 relative to the ENDF libraries, but when the $H/^{233}\text{U}$ increases to 119 in experiment 7.3, the calculated k_{eff} is approximately 3% below 1.0. Similar to the observation in experiment 7.2, the calculated k_{eff} obtained with the ENDF/B-V libraries is significantly closer to 1.0 for the unreflected sphere in experiment 7.4 relative to ENDF/B-IV as shown in Table 2.3.1.8.

Table 2.3.1.8: Calculated Effective Multiplication Factor for Unreflected Aqueous $\text{UO}_2(\text{NO}_3)_2$ Experiments.

Exp #	k_{eff} ENDF/B-V	Avg. Fission Group (AFG)	k_{eff} ENDF/B-IV	Avg. Fission Group (AFG)
	238 Group 44 Group		218 Group 27 Group	
7.2	1.0165 ± 0.0040	180.97	1.0479 ± 0.0048	173.06
	1.0248 ± 0.0049	29.45	1.0511 ± 0.0049	19.67
7.3	0.9904 ± 0.0048	193.26	1.0068 ± 0.0052	185.11
	0.9901 ± 0.0046	32.08	1.0110 ± 0.0044	21.19
7.4	1.0095 ± 0.0043	202.45	1.0312 ± 0.0051	194.32
	1.0084 ± 0.0046	34.21	1.0382 ± 0.0048	22.43

Similarly, the critical parameters for the cubic arrays of uranyl nitrate solution tabulated in Table 2.3.1.5 were used to model each of the experiments. Using the appropriate critical parameters, the effective multiplication factor was calculated for each experiment using KENO V.a. The results obtained with the fine and broad group ENDF/B-V and ENDF/B-IV cross section libraries are presented in Table 2.3.1.9. The average fission energies range from 1.09 to 0.6 eV and 1.1 to 0.65 eV for the 238 group ENDF/B-V and 218 group ENDF/B-IV libraries, respectively. Furthermore, the average

fission energies range from .35 to .15 eV and 0.8 to .225 eV for the 44 group ENDF/B-V and 27 group ENDF/B-IV libraries, respectively. The calculated k_{eff} 's obtained with the ENDF libraries for the lower moderated cubic arrays (i.e. $H/^{233}\text{U} = 73$) are significantly higher than the results for the single unit experiments indicating a positive bias. However, as the moderation increased in experiments 8.13 and 8.16, the calculated multiplication factor is in better agreement with experimental results (i.e. $k_{\text{eff}} = 1.0$). It should be noted that the results obtained with the Hansen-Roach library are in better agreement (i.e. $\pm 2.5\%$) with the critical experiments than either the ENDF/B-V or ENDF/B-IV libraries (see Appendix C).

Table 2.3.1.9: Calculated Effective Multiplication Factors for Cubic Arrays of Cylindrical Units of Aqueous $\text{UO}_2(\text{NO}_3)_2$ Solution* (Ref 8).**

Exp #	k_{eff} ENDF/B-V	Avg. Fission Group (AFG)	k_{eff} ENDF/B-IV	Avg. Fiss. Group (AFG)
	238 Group 44 Group		218 Group 27 Group	
8.10	1.0451 ± 0.0048	180.61	1.0684 ± 0.0052	172.65
	1.0446 ± 0.0040	29.38	1.0707 ± 0.0046	19.64
8.11	1.0420 ± 0.0040	190.63	1.0627 ± 0.0043	182.16
	1.0520 ± 0.0041	31.60	1.0723 ± 0.0043	20.90
8.13	1.0180 ± 0.0045	192.99	1.0427 ± 0.0041	184.93
	1.0297 ± 0.0052	32.02	1.0348 ± 0.0044	21.16
8.16	1.0184 ± 0.0041	199.67	1.0306 ± 0.0045	191.42
	1.0257 ± 0.0039	33.60	1.0370 ± 0.0039	22.05

*** Uranium isotopic content (wt%): $^{233}\text{U} = 97.54$, $^{234}\text{U} = 1.047$, $^{235}\text{U} = 0.025$, $^{236}\text{U} = 0.001$, $^{238}\text{U} = 1.386$.

In order to compare the results for the 26 uranyl nitrate experiments, the calculated k_{eff} 's obtained with the 238 group ENDF/B-V and 218 group ENDF/B-IV cross section libraries are plotted as a function of H/U in Fig. 2.3.1.1. Also included in Fig. 2.3.1.1 are the results obtained with the 16 group Hansen-Roach library. As shown in Fig. 2.3.1.1, there is a significant difference between the calculated k_{eff} 's for a particular cross section library in the lower moderated critical experiments (i.e. lower H/U). As the H/U increases, the agreement between the libraries improves. This is especially noticeable for the thermal systems with $H/U \geq 1000$. In addition, the k_{eff} values obtained with the 218 group library are higher than the values obtained with the newer 238 group library. Except for the systems with $H/U \geq 1000$, the 238 group ENDF/B-V library typically predicts values closer to 1.0 than the 218 group ENDF/B-IV library. The highly thermalized systems with $H/U \geq 1000$ were the only experiments conducted in large vessels, thus minimizing the leakage from the system. For these systems, the 238 group library typically predicts lower k_{eff} values than either the 218 group ENDF/B-IV or the 16 group Hansen-Roach library. As shown in Fig. 2.3.1.1, the agreement between the calculated k_{eff} 's obtained with the 238 group ENDF/B-V library and the experimental results (i.e. $k_{\text{eff}}=1.0$) is quite good for the systems with $250 \leq H/U \leq 500$.

A similar graphical comparison was performed for the KENO V.a results obtained with the 44 group ENDF/B-V, 27 group ENDF/B-IV and 16 group Hansen-Roach libraries. The calculated k_{eff} 's for the uranyl nitrate experiments are presented as a

function of H/U in Fig. 2.3.1.2. Except for subtle differences, the results for the broad group libraries shown in Fig. 2.3.1.2 are similar to the results for the fine group libraries presented in Fig. 2.3.1.1. In general, the agreement between the libraries improves for the more thermalized systems, and the most noticeable improvement in the calculated k_{eff} 's obtained with the 44 group ENDF/B-V library relative to the 27 group ENDF/B-IV library occurs for the lower moderated systems.

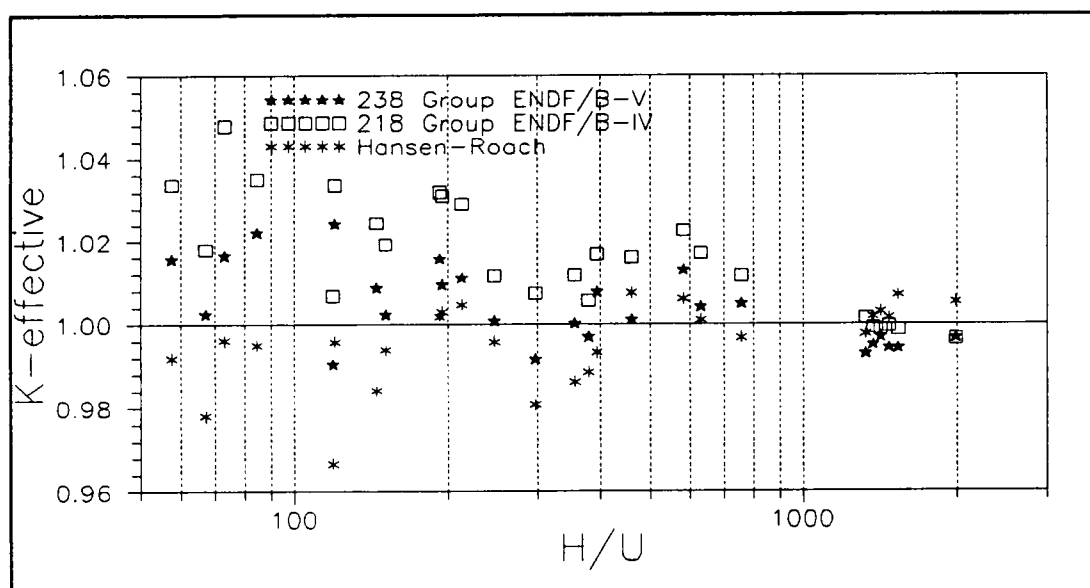


Figure 2.3.1.1: 238 Group ENDF/B-V and 218 Group ENDF/B-IV calculated K_{eff} 's as a Function of H/U for Uranyl Nitrate Systems.

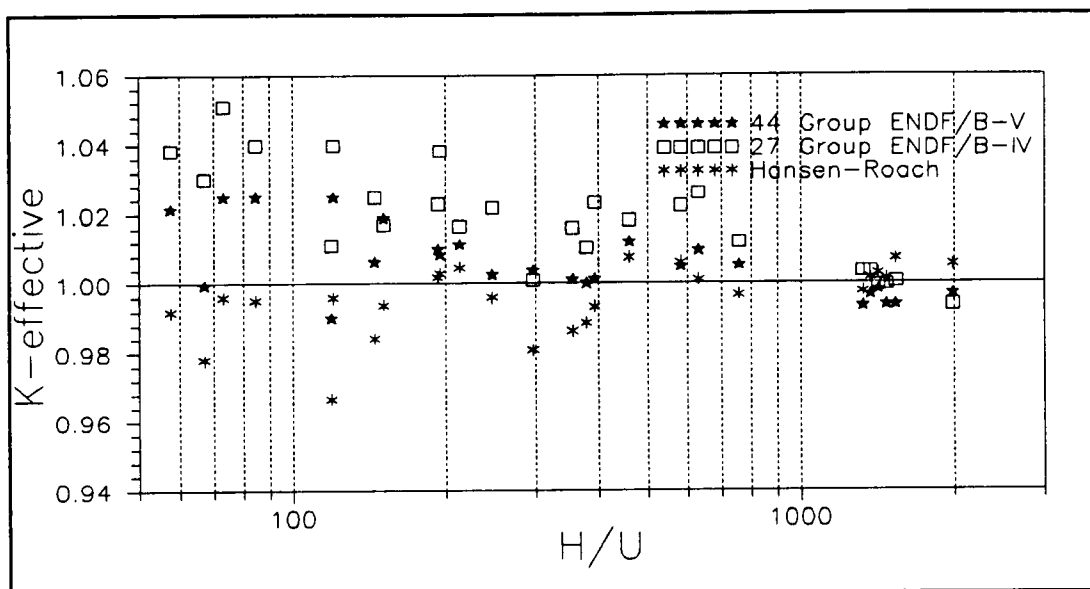


Figure 2.3.1.2: 44 Group ENDF/B-V and 27 Group ENDF/B-IV Calculated K_{eff} 's as a Function of H/U for Uranyl Nitrate Systems.

2.3.2 Aqueous Uranyl Fluoride Experiments

In addition to the uranyl nitrate experiments presented in Section 2.3.1 by Fox, Gilley, and Rohrer⁴, uranyl fluoride experiments in cylindrical and spherical geometry have also been performed. The isotopic percentages for these experiments are presented in Table 2.3.2.1, and the critical parameters for the cylindrical and spherical UO_2F_2 experiments are presented in Tables 2.3.2.2 and 2.3.2.3, respectively. Likewise, critical parameters for water reflected spheres of UO_2F_2 solution which are provided by Callihan, Fox, and Thomas⁵ are presented in Table 2.3.2.4.

Table 2.3.2.1: Isotopic Percentages of Uranium in $\text{UO}_2(\text{NO}_3)_2$ and UO_2F_2

Isotope	Wt%
^{233}U	98.7
^{234}U	0.54
^{235}U	0.04
^{238}U	0.72

Table 2.3.2.2: Critical Parameters for Paraffin Reflected* Cylinders of Aqueous UO_2F_2 Solutions (Ref. 4).

Exp #	H/ U^{233}	Soln. ht. (cm)	Al Cyl. Diam. (cm)	Cyl. Vol. (cm^3)	Mass of ^{233}U (g)
4.35	74.1	24.0	15.1**,***	4310	1430
4.39	45.9	16.9	16.9**	3670	1910
4.43	154	18.4	19.1	5250	870
4.49	775	30.5	30.5	22280	740

*Reflector thickness = 20 cm.

**No reflector on the top surface.

***Containers coated with unichrome; critical masses are about 2% high due to impurities.

Table 2.3.2.3: Critical Parameters for Spheres of Aqueous UO_2F_2 Solutions (Ref. 4).

Exp #	H/ U^{233}	Reflector ⁸	Soln. ht. (cm)	Al Sphere Diam. (cm)	Sphere Vol. (cm^3)	Mass of ^{233}U (g)
4.52	419	water	*	26.6	9620	590
4.54	663	water	Full	31.9	17020	660
4.55	381	none	*	31.9	16980	1140

Table 2.3.2.4: Critical Parameters for Water Reflected Spheres of Aqueous UO_2F_2 Solutions (Ref. 5).

Exp #	H/ U^{233}	Temp. ($^{\circ}\text{C}$)	Al Sphere Diam. (cm)	Soln. Vol. (cm^3)	Mass of ^{233}U (g)
5.2	414.6	39.5	26.4	9675	596.3
5.3	400.5	65.5	26.4	9704	611.0
5.4	389.6	83.2	26.4	9723	623.0
5.7	643.1	56.0	32.0	17042	673.0
5.8	602.8	99.5	32.0	17074	703.0

*There was a 40 cm^3 void above the critical solution.

Using the isotopic percentages from Table-2.3.2.1 with the critical parameters from Tables-2.3.2.2 and 2.3.2.3, the UO_2F_2 experiments were modeled with KENO V.a. The results obtained with both the ENDF/B-V and ENDF/B-IV libraries are tabulated in Table 2.3.2.5. As shown in Table 2.3.2.5, the average energies for fission range between 1.09 and 0.1 eV (groups 181 to 216) and 1.11 to 0.1 eV (groups 172 to 208) for the 238 ENDF/B-V and 218 ENDF/B-IV libraries, respectively. For the 44 and 27 group libraries, the average fission energies range between 0.275 and 0.04 eV (groups 31 to 38) and 1.0 to 0.03 eV (groups 19 to 25), respectively. Using the 238 group library for the paraffin reflected cylinders, the calculated k_{eff} 's are within $\pm 3\%$ of 1.0 for the lower moderated experiments (i.e. exps 4.35, 4.39 and 4.43); however, as the $\text{H}/^{233}\text{U}$ ratio increased for experiment 4.49, the calculated k_{eff} is within $+1\%$ of 1.0 indicating an improvement associated with the increased hydrogen moderation. In a general comparison between the ENDF/B-V and ENDF/B-IV results for the cylinders shown in Table 2.3.2.5, the k_{eff} 's obtained with both the 44 and 238 group libraries are closer to 1.0 than the ENDF/B-IV results except for experiment 4.35. The last three experiments shown in Table 2.3.2.5 correspond to the spheres of UO_2F_2 . The results for these experiments are closer to 1.0 than the paraffin reflected cases for each of the cross section libraries indicating some difficulty in characterizing the return from the paraffin reflector. Also, a noticeable improvement exists in the results obtained with the ENDF/B-V libraries relative to the ENDF/B-IV results for two of the spheres. Regarding these spheres, the highest calculated multiplication factor is obtained for experiment 4.54 (i.e. $+1.8\%$ relative to 1.0), which is the only sphere completely filled

Table 2.3.2.5: Calculated Effective Multiplication Factor for Cylinders and Spheres of Aqueous UO_2F_2 .

Exp. #	k_{eff} ENDF/B-V	Avg. Fission Group (AFG)	k_{eff} ENDF/B-IV	Avg. Fission Group (AFG)
	238 Group 44 Group		218 Group 27 Group	
4.35	0.9799 ± 0.0036	190.91	1.0001 ± 0.0040	182.76
	0.9939 ± 0.0040	31.59	0.9982 ± 0.0037	20.92
4.39	1.0365 ± 0.0041	181.37	1.0505 ± 0.0049	172.94
	1.0390 ± 0.0042	29.67	1.0478 ± 0.0043	19.76
4.43	1.0269 ± 0.0038	203.73	1.0506 ± 0.0037	195.45
	1.0385 ± 0.0043	34.45	1.0435 ± 0.0042	22.56
4.49	1.0073 ± 0.0034	216.43	1.0065 ± 0.0034	208.17
	1.0068 ± 0.0034	37.60	1.0109 ± 0.0031	24.34
4.52	1.0030 ± 0.0034	213.21	1.0234 ± 0.0037	204.95
	1.0189 ± 0.0037	36.79	1.0246 ± 0.0036	23.88
4.54	1.0181 ± 0.0033	215.77	1.0181 ± 0.0034	207.46
	1.0162 ± 0.0034	37.43	1.0233 ± 0.0032	24.25
4.55	1.0036 ± 0.0048	210.68	1.0185 ± 0.0042	202.50
	1.0076 ± 0.0044	36.18	1.0244 ± 0.0043	23.54

with aqueous uranyl fluoride. In contrast, the remaining two spheres have a 40 cm³ void above the critical solution. Interestingly, the results obtained with the ENDF/B-V, ENDF/B-IV and Hansen-Roach libraries (see Appendix C) for experiment 4.54 are in good agreement. Therefore, an error in the measured solution height might exist for experiment 4.54. Nevertheless, the calculated results are in good agreement with the critical experimental results.

Following the same procedure, the critical parameters for the water reflected spheres of UO₂F₂ presented in Table 2.3.2.4 were used to develop KENO V.a input files for each experiment. The results of these calculations are shown in Table 2.3.2.6. As indicated by the average fission group in Table 2.3.2.6, the neutron spectrum for these spherical experiments is very thermalized. As observed for the spheres in Table 2.3.2.5, the calculated k_{eff} 's are slightly higher than 1.0 as shown in Table 2.3.2.6; however, the ENDF/B-V results are approximately 1% closer to critical relative to the ENDF/B-IV results. It should be noted that no significant trends appear to exist due to the increase in temperature for the spherical cases tabulated in Table 2.3.2.6.

Table 2.3.2.6: Calculated Effective Multiplication Factor for Water Reflected Spheres of Aqueous UO_2F_2 .

Exp. #	k_{eff} ENDF/B-V	Avg. Fission Group (AFG)	k_{eff} ENDF/B-IV	Avg. Fission Group (AFG)
	238 Group 44 Group		218 Group 27 Group	
5.2	1.0096 ± 0.0034 1.0078 ± 0.0034	213.20 36.79	1.0260 ± 0.0042 1.0228 ± 0.0035	204.85 23.87
5.3	1.0220 ± 0.0039 1.0207 ± 0.0043	212.65 36.46	1.0311 ± 0.0037 1.0326 ± 0.0038	204.91 23.87
5.4	1.0153 ± 0.0037 1.0102 ± 0.0041	212.34 36.37	1.0219 ± 0.0039 1.0235 ± 0.0035	204.55 23.82
5.7	1.0132 ± 0.0031 1.0125 ± 0.0037	215.12 37.06	1.0219 ± 0.0036 1.0256 ± 0.0036	207.39 24.23
5.8	1.0187 ± 0.0031 1.0195 ± 0.0033	214.82 36.98	1.0381 ± 0.0036 1.0259 ± 0.0034	207.10 24.19

Using the same comparative procedure as for the $\text{UO}_2(\text{NO}_3)_2$ systems, the calculated k_{eff} 's obtained with the 238 group ENDF/B-V and 218 group ENDF/B-IV cross section libraries are shown in Fig. 2.3.2.1 as a function of H/U for the 12 uranyl fluoride critical experiments. Also included in Fig. 2.3.2.1 are the calculated k_{eff} 's obtained with the Hansen-Roach library. As in the uranyl nitrate systems, there is a significant difference between the calculated k_{eff} 's for a particular library in the lower moderated experiments. For example, there is approximately a 5% difference between the results obtained with the 218 group ENDF/B-IV and Hansen-Roach libraries. The agreement between the cross section libraries improves somewhat as the H/U ratio increases. As observed in the results for the uranyl nitrate experiments, the values

obtained for the uranyl fluoride systems with the 218 group ENDF/B-IV library are typically higher than the values obtained with the 238 group ENDF/B-V library.

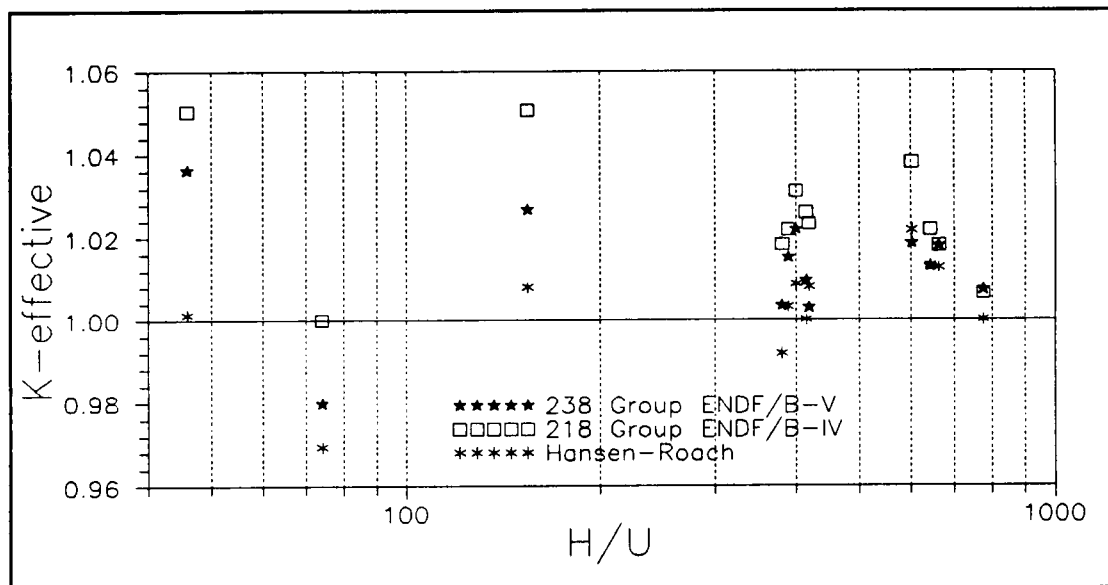


Figure 2.3.2.1: 238 Group ENDF/B-V and 218 Group ENDF/B-IV Calculated K_{eff} 's as a Function of H/U for Uranyl Fluoride Systems.

In addition to the fine group results, the calculated effective multiplication factors obtained with the 44 group ENDF/B-V and 27 group ENDF/B-IV libraries for the UO_2F_2 experiments are presented in Fig. 2.3.2.2. The calculated results shown in Fig. 2.3.2.2 are similar to the general trends for the fine group cross section libraries shown in Fig. 2.3.2.1.

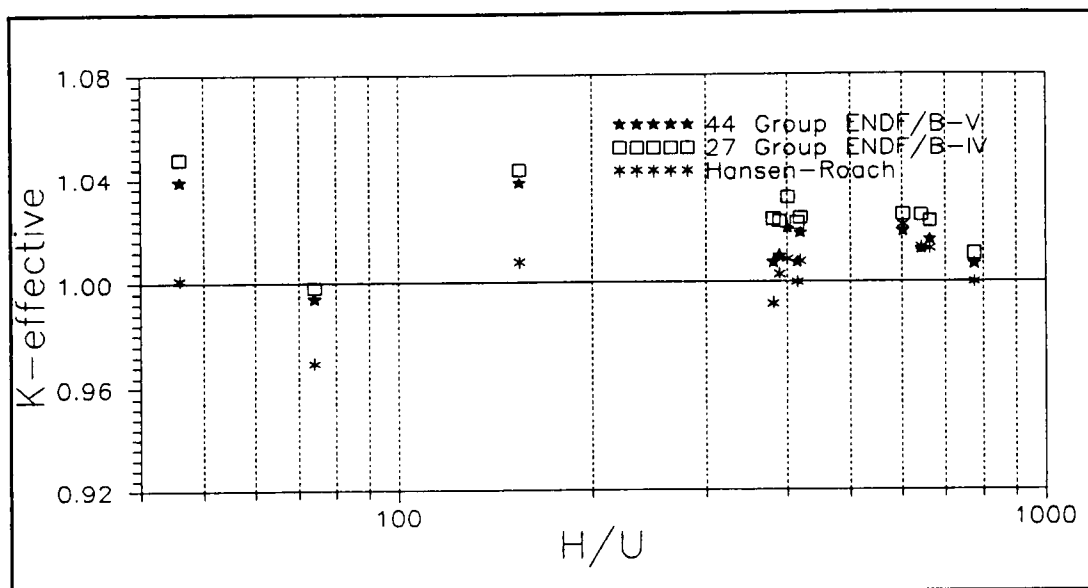


Figure 2.3.2.2: 44 Group ENDF/B-V and 27 Group ENDF/B-IV Calculated K_{eff} 's as a Function of H/U for Uranyl Fluoride Systems.

2.3.3 ^{233}U Metal Experiments

In a brief study of fast systems, Paxton and Pruvost⁸ presented experimental data for ^{233}U metal spheres. The isotopic percentages and critical masses of unmoderated ^{233}U metal spheres documented in reference [8] are presented in Tables 2.3.3.1 and 2.3.3.2, respectively. Also included in Tables 2.3.3.1 and 2.3.3.2 are the isotopic percentages and critical parameters, respectively, for a single metal sphere provided by McNeany and Jenkins.⁷

Table 2.3.3.1: Isotopic Percentages for ^{233}U Metal Spheres (Refs. 7,8).

Exp #	Isotopic Content (wt%)		
	^{233}U	^{234}U	^{238}U
7.1	98.13	1.245	0.60
8.1	98.1	1.25	0.61
8.2	98.1	1.25	0.61
8.3	98.2	1.1	0.7
8.4	98.2	1.1	0.7
8.5	98.2	1.1	0.7
8.6	98.2	1.1	0.7
8.7	98.2	1.1	0.7
8.8	98.2	1.1	0.7

Table 2.3.3.2: Critical Parameters of ^{233}U Metal Spheres (Refs. 7,8).

Exp #	Uranium Density (g/cm ³)	Critical radius (cm)	Reflector		
			Material	Wall Δt (cm)	Density (g/cm ³)
7.1	18.417	5.984	None	---	---
8.1	18.42	5.946	None	---	---
8.2	18.42	4.179	U(N)	19.91	19.0
8.3	18.64	4.60	U(N)	5.309	18.9
8.4	18.62	5.044	U(N)	2.301	18.9
8.5	18.64	4.60	W-alloy*	5.791	17.2
8.6	18.62	5.044	W-alloy*	2.438	17.2
8.7	18.64	4.60	Be (98%)	4.196	1.83
8.8	18.62	5.044	Be (98%)	2.045	1.83

*Composition (wt%): 91.3% W, 5.5% Ni, 2.5% Cu, 0.7% Zr.

Nine different KENO V.a input files were developed based on the critical parameters presented in Table-2.3.3.2 for each of the experiments. The results obtained with the fine and broad group ENDF/B-V and ENDF/B-IV libraries are shown in Table 2.3.3.3. There is a definite improvement for fast systems in the agreement between the calculated k_{eff} 's and the experimental results (i.e. $k_{\text{eff}} = 1.0$) obtained with the newer ENDF/B-V libraries relative to ENDF/B-IV as shown in Table 2.3.3.3. In particular, there is a significant improvement obtained with the ENDF/B-V library for the two tungsten reflected spheres relative to the ENDF/B-IV results. These results for experiments 8.5 and 8.6 indicate an improvement in the treatment of the tungsten cross sections in ENDF/B-V libraries. Despite this improvement, the tungsten reflected spheres were calculated approximately 2% low (i.e. relative to $k_{\text{eff}} = 1.0$) using the ENDF/B-V libraries.

With regard to the Be reflected spheres, there is not much difference between the calculated k_{eff} 's obtained with either the ENDF/B-V or ENDF/B-IV libraries for experiment 8.7 (Be reflector thickness = 4.196 cm.). However, as the Be reflector decreased to 2.045 cm. in experiment 8.8, the ENDF/B-V libraries predict a k_{eff} much closer to 1.0 than the ENDF/B-IV libraries. The improvement associated with the decreased Be content may indicate some inadequacies with the ENDF/B-V cross sections in modeling the room return from the Be reflector. Therefore, some further investigations should be performed for Be reflected systems using the ENDF/B-V cross section library.

Table-2.3.3.3: Calculated Effective Multiplication Factor for ^{233}U Metal Spheres.

Exp. #	k_{eff} ENDF/B-V	Avg. Fission Group (AFG)	k_{eff} ENDF/B-IV	Avg. Fission Group (AFG)
	238 Group 44 Group		218 Group 27 Group	
7.1	0.9920 ± 0.0032 0.9965 ± 0.0032	22.63 7.93	0.9640 ± 0.0031 0.9649 ± 0.0028	16.90 4.75
8.1	0.9905 ± 0.0034 0.9975 ± 0.0032	22.46 7.88	0.9613 ± 0.0032 0.9611 ± 0.0025	17.02 4.75
8.2	1.0005 ± 0.0032 1.0041 ± 0.0032	23.44 7.90	0.9687 ± 0.0028 0.9757 ± 0.0035	17.73 4.73
8.3	0.9988 ± 0.0030 1.0047 ± 0.0034	22.98 7.96	0.9716 ± 0.0033 0.9786 ± 0.0032	17.46 4.77
8.4	1.0054 ± 0.0029 1.0010 ± 0.0032	22.89 7.95	0.9710 ± 0.0031 0.9742 ± 0.0029	17.47 4.79
8.5	0.9805 ± 0.0037 0.9778 ± 0.0033	24.96 8.36	0.9475 ± 0.0029 0.9451 ± 0.0035	19.38 5.06
8.6	0.9881 ± 0.0031 0.9876 ± 0.0030	24.01 8.16	0.9564 ± 0.0029 0.9552 ± 0.0032	18.32 4.95
8.7	0.9859 ± 0.0031 0.9919 ± 0.0035	25.94 8.47	0.9874 ± 0.0034 0.9892 ± 0.0034	20.69 5.27
8.8	0.9955 ± 0.0034 0.9975 ± 0.0034	24.11 8.22	0.9719 ± 0.0035 0.9799 ± 0.0031	18.68 5.01

The improvement obtained with the newer ENDF/B-V library is especially noticeable for the natural uranium reflected and unreflected metal spheres. With regard to experiment 8.2, the k_{eff} calculated with the 238 group ENDF/B-V library is closer to 1.0 and is approximately 5% higher than the 218 group ENDF/B-IV calculated value. Because of the differences observed for experiment 8.2, an investigation of the results

was performed. A plot of the fission percentages as a function of energy is shown in Fig. 2.3.3.1 for the 238 group ENDF/B-V and the 218 group ENDF/B-IV cross section libraries. As shown in Fig. 2.3.3.1, no fissions occur below 10^4 eV, and the majority of fissions occur between 1.5 and 4.0 MeV indicating a fast neutron energy spectrum.

A plot of the ^{233}U fission cross section for experiment 8.2 is shown in Fig. 2.3.3.2 for the ENDF/B-V and ENDF/B-IV fine group libraries. Since the fissions are predominately occurring above 1.5 MeV, a detailed view of the ^{233}U fission cross section above 1.0 MeV is shown in Fig. 2.3.3.3. Except for 6.5 to 8.0 MeV, the 238 group library has a higher fission cross section than the 218 group library above 2.0 MeV. The higher fission probability associated with the newer ENDF/B-V library partly explains the higher calculated k_{eff} relative to the ENDF/B-IV library. Also, an examination of the group structure for both of these libraries (see Appendix A) reveals that the 238 group library has 7 more energy groups from 1.0 to 20.0 MeV, thus indicating a more detailed treatment of the cross sections in the higher energy ranges relative to the 218 ENDF/B-IV library.

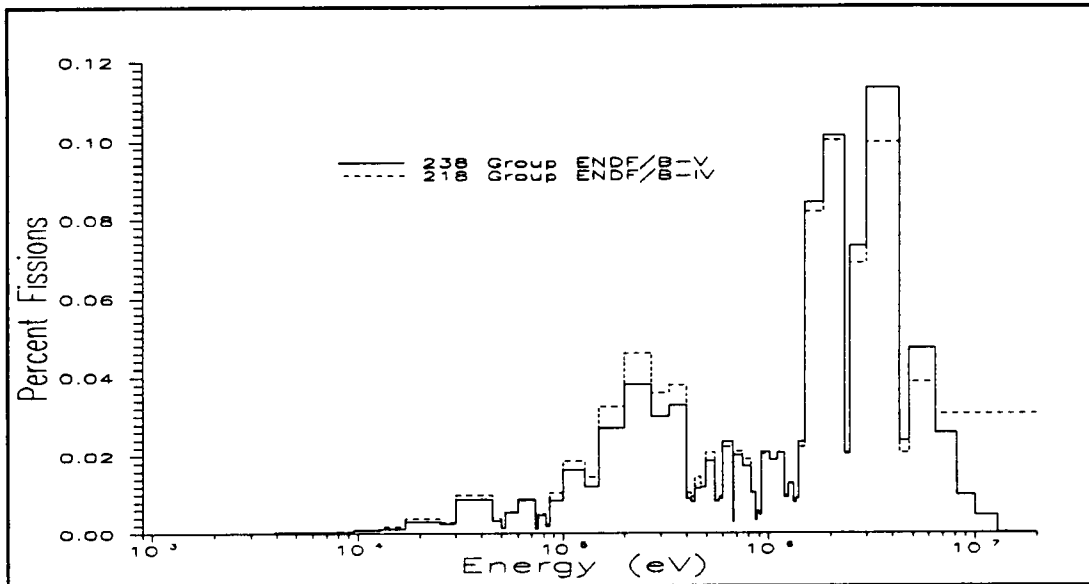


Figure 2.3.3.1: Percent Fissions for a ^{233}U Metal Sphere Reflected with Natural Uranium (Experiment 8.2).

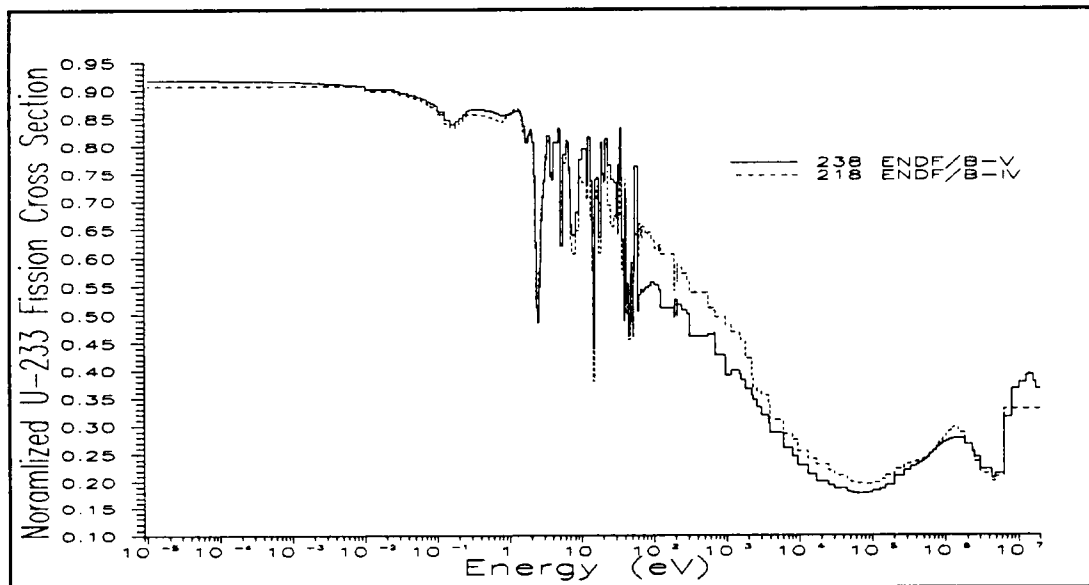


Figure 2.3.3.2: 238 Group ENDF/B-V and 218 Group ENDF/B-IV ^{233}U Fission Cross Section Libraries for Experiment 8.2.

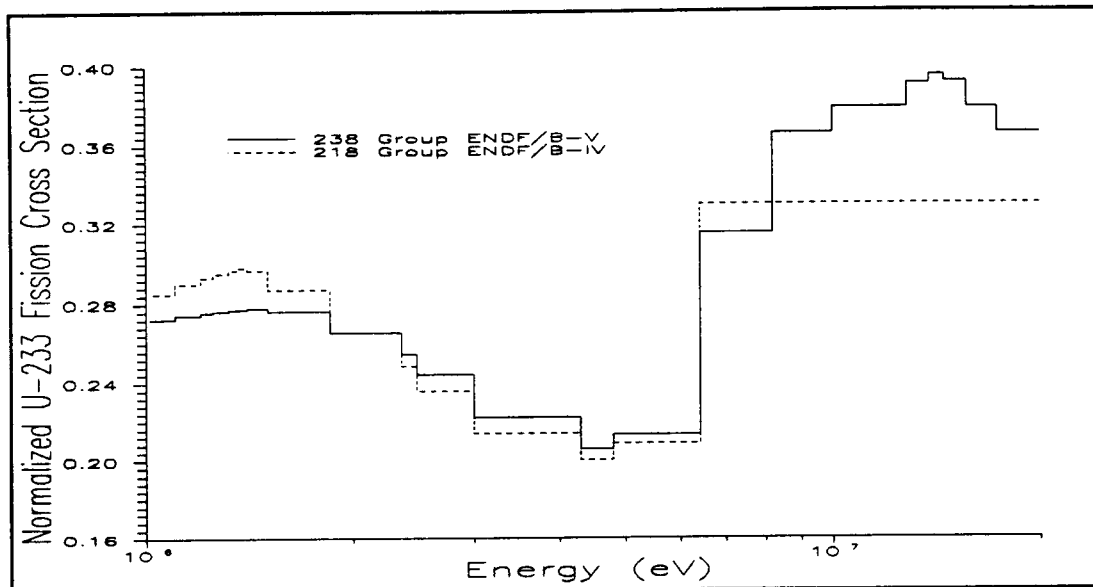


Figure 2.3.3.3: 238 Group ENDF/B-V and 218 Group ENDF/B-IV ^{233}U Fission Cross Section Libraries from 1.0 to 20.0 MeV for Experiment 8.2.

2.3.4 Trends in Calculated K_{eff}

In the previous sections, the calculated multiplication factors for 51 ^{233}U critical experiments are presented. After reviewing the experiments, there appears to be a trend in k_{eff} with respect to the experimental critical parameters. This trend in k_{eff} is investigated in this section.

In order to evaluate the apparent trends, the calculated k_{eff} 's were sorted according to the three variable classifications: geometry, reflector and fuel form. The results of the trend study for the 238 group ENDF/B-V and 218 group ENDF/B-IV cross section libraries are shown in Table 2.3.4.1. Similarly, the results for the 44 group ENDF/B-V, 27 group ENDF/B-IV and 16 group Hansen-Roach cross section libraries are tabulated

in Table 2.3.4.2. In a general comparison, the mean k_{eff} values for the ENDF/B-V libraries are 0.5 to 0.6% lower than the ENDF/B-IV results while the ENDF/B-V values are 0.7% higher than the Hansen-Roach results. Upon further investigation of Tables 2.3.4.1 and 2.3.4.2, there may be a trend for array configurations to calculate higher than single unit configurations. Furthermore, it is noteworthy that the water reflected configurations calculated slightly higher than the bare configurations for each of the cross section libraries. Also, the polyethylene configurations calculated significantly higher than the bare reflected configurations. These results may indicate that reflectors containing hydrogen may have a positive bias on the calculated k_{eff} . As shown in Tables 2.3.4.1 and 2.3.4.2 the tungsten reflected cases calculated significantly lower than the bare reflected cases. Except for the Hansen-Roach library, the Be reflected cases also calculated lower than the bare cases. Regarding the solution fuel form, there is not a significant difference between the mean k_{eff} values obtained with a particular library for the uranyl nitrate and uranyl fluoride. However, $\text{UO}_2(\text{NO}_3)_2$ configurations may calculate slightly lower than UO_2F_2 configurations as indicated in Tables 2.3.4.1 and 2.3.4.2. In contrast, the mean value for the metal configurations obtained with the ENDF/B-IV libraries is significantly lower than the nitrate or fluoride cases indicating a substantial negative bias for undermoderated fast systems. As noted in the previous sections, the significant improvements obtained from using the ENDF/B-V data for the metal systems is apparent from the results presented in Tables 2.3.4.1 and 2.3.4.2.

Table 2.3.4.1: Trend in k_{eff} for 51 ^{233}U Critical Experiments Using KENO V.a with the 238 Group ENDF/B-V and 218 Group ENDF/B-IV Cross Section Libraries.

Parameter (Number of Observations)	Mean Calculated k_{eff}	
	238 ENDF/B-V	218 ENDF/B-IV
Geometry		
Cylinders (23)	1.0079	1.0222
Spheres (24)	1.0009	0.9969
Array (4)	1.0309	1.0511
Reflector		
Bare (14)	1.0026	1.0097
Paraffin (21)	1.0083	1.0217
Water (7)	1.0143	1.0258
Natural U (3)	1.0016	0.9704
W-alloy (2)	0.9843	0.9520
Be (2)	0.9907	0.9797
Polyethylene (2)	1.0302	1.0467
Fuel Form		
UO_2F_2 (12)	1.0128	1.0256
$\text{UO}_2(\text{NO}_3)_2$ (30)	1.0078	1.0246
Metal (9)	0.9930	0.9666
All Exps. (51)	1.0064	1.0125
Overall Bias	0.0066	0.0072

Table 2.3.4.2: Trend in k_{eff} for 51 ^{233}U Critical Experiments Using KENO V.a with the 44 Group ENDF/B-V, 27 Group ENDF/B-IV and 16 Group Hansen-Roach Cross Section Libraries.

Parameter (Number of Observations)	Mean Calculated k_{eff}		
	44 ENDF/B-V	27 ENDF/B-IV	Hansen-Roach
Geometry			
Cylinders (23)	1.0109	1.0236	0.9932
Spheres (24)	1.0022	0.9986	1.0024
Array (4)	1.0380	1.0537	1.0114
Reflector			
Bare (14)	1.0052	1.0112	0.9998
Paraffin (21)	1.0113	1.0229	0.9994
Water (7)	1.0151	1.0255	1.0098
Natural U (3)	1.0033	0.9762	0.9987
W-alloy (2)	0.9827	0.9502	0.9694
Be (2)	0.9947	0.9846	1.0174
Polyethylene (2)	1.0389	1.0547	1.0157
Fuel Form			
UO_2F_2 (12)	1.0160	1.0253	1.0033
$\text{UO}_2(\text{NO}_3)_2$ (30)	1.0102	1.0232	0.9977
Metal (9)	0.9954	0.9693	0.9976
All Exps. (51)	1.0090	1.0142	0.9991
Overall Bias	0.0095	0.0092	0.0012

2.4 Concrete Reflected Experiments

The validation study presented in this chapter is intended to support a criticality safety evaluation of the ^{233}U material stored in the Building 3019 storage wells and the Molten Salt Reactor fuel drain tanks. The ^{233}U material is interstitially and externally reflected by concrete in Building 3019 while only externally reflected in the MSR fuel drain tank cell. Consequently, it is believed that the effective multiplication factor of each storage location is dependent on the concrete reflection present in both facilities. As a result, it is necessary to validate the ENDF/B-V "concrete cross sections" or the constitutive elements present in the concrete prior to evaluating the storage locations. Moreover, it is desirable to use concrete reflected ^{233}U experiments in the validation study; however, as stated in Section 2.3, the ^{233}U critical experiment database is very limited and does not include concrete reflected ^{233}U experiments. In contrast, there is an adequate amount of plutonium and ^{235}U concrete reflected experiments available. To compensate for the lack of ^{233}U experimental data, concrete reflected experiments consisting of fissile material other than ^{233}U is used in the validation study.

A series of four concrete reflected experiments involving Fast Test Reactor (FTR) fuel pin lattices in water was performed in 1979 by Bierman, Durst and Clayton¹⁷ at the Pacific Northwest Laboratory (PNL) in Richland, Washington. These experiments were selected because the concrete composition is similar to the concrete present in the Building 3019 walls housing the in-line wells³ as shown in Table 2.4.1. The FTR fuel pins consisted of a PuO_2 and UO_2 mixture with 19.84 wt% plutonium (11.5 wt% ^{240}Pu)

and natural uranium. A complete description of the chemical composition of the fuel is tabulated in Table 2.4.2. Using a fuel density of $9.830 \pm 0.320 \text{ g/cm}^3$, the atomic number densities of the fuel were calculated and are tabulated in Appendix B.

Table 2.4.1: Comparison of Concrete Composition for Building 3019 In-line Wells and FTR Experiments.

Element	3019 In-line Well Concrete Wall ($\rho = 1.523 \text{ g/cm}^3$) (wt%)	Concrete Reflector FTR Experiment ($\rho = 2.452 \text{ g/cm}^3$) (wt%)
Al	2.26 ± 0.15	2.32 ± 0.01
Ca	7.51 ± 0.28	7.27 ± 0.28
Fe	1.40 ± 0.03	0.28 ± 0.02
Si	38.19 ± 1.21	34.68 ± 0.39
Mg	0.31 ± 0.03	1.45 ± 0.05
K	0.92 ± 0.26	0.75 ± 0.15
Na	0.24 ± 0.03	0.17 ± 0.01
O	48.01 ± 1.56	52.26 ± 0.58
H	0.50	0.82 ± 0.07
C	0.50	—
S	0.08 ± 0.02	—

*Errors are standard deviations based on three analysis.

Table 2.4.2: Chemical Composition of Type 3.2 FTR Fuel Pins.

Chemical Element	(wt%)
^{238}Pu	0.0297 (max)
^{239}Pu	16.9683 ± 0.0964
^{240}Pu	2.2733 (baln)
^{241}Pu	0.4959 ± 0.0028
^{242}Pu	0.0694 (max)
^{235}U	0.4853
^{238}U	67.8638
O	11.8143

The FTR fuel pin experiments have the most complicated geometry of all the experiments presented in this validation study. In order to understand the critical parameters of the FTR experiment, a brief geometrical description of the experiment is in order; however, for a comprehensive description of the experiment, the reader is referred to reference [17]. The fuel region of an FTR fuel pin is 914.4 mm in length and is composed of pellets 4.9403 mm in diameter and 5.2070 mm in length. A schematic diagram of a typical FTR fuel pin illustrating the fuel region is presented in Fig. 2.4.1. The fuel pins were placed in a square lattice configuration within a carbonsteel tank. The tank was partially filled with deionized water and reflected by 406 mm of concrete on five sides excluding the top. A top view of the experimental assembly is shown in Fig. 2.4.2 while an elevation view of the assembly is presented in Fig. 2.4.3. The variable dimensions shown in Fig. 2.4.2 for each of the four experiments are tabulated in Table 2.4.3 while the critical parameters for the fuel

assembly are presented in Table 2.4.4.

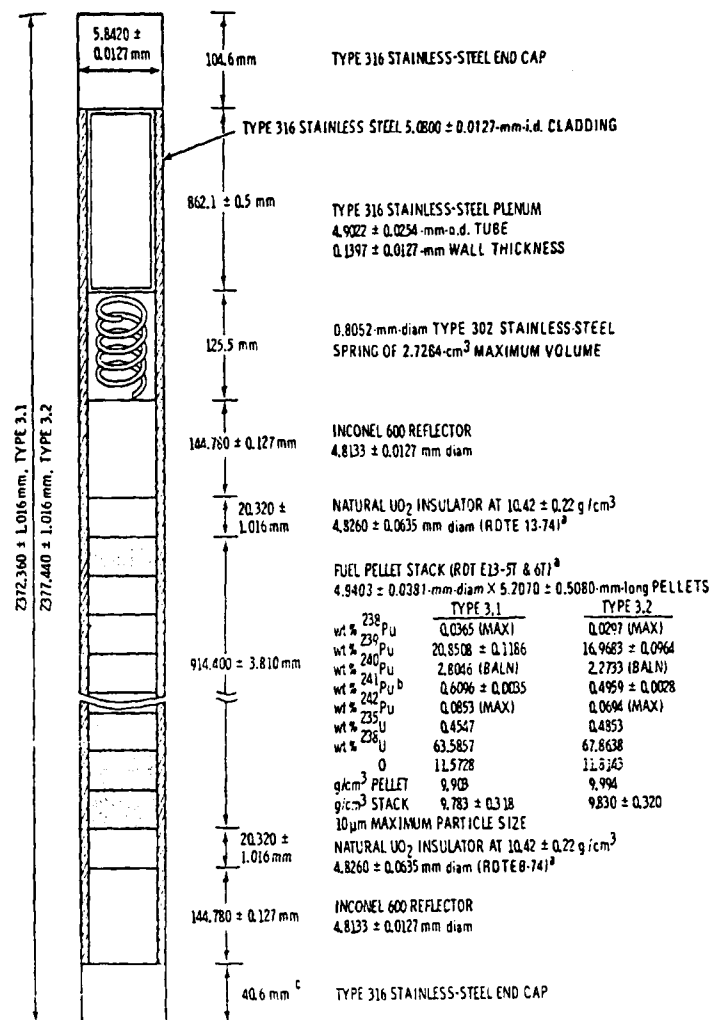
Table 2.4.3: Variable Dimensions for FTR Experiment Assembly.

Exp. #	Dimension i.d. from Fig. 2.4.2 (mm)				
	a	b	c	d	e
010	9.53	20.22	21.48	20.22	20.33
007	12.63	19.09	15.94	18.67	18.78
012	15.41	32.00	22.06	17.28	17.39
011	19.06	23.49	21.48	15.45	15.56

Table 2.4.4: Critical Parameters for Concrete Reflected FTR Experiments.

Exp #	Square Lattice Pitch* (mm)	H ₂ O/Fuel (volume ratio)	Lattice Width (fuel pins)	Crit. Number of Fuel Pins ¹⁴	Crit. Rows of Fuel Pins
010	9.53 ± 0.13	3.34	28	554.2 ± 0.2	19.79
007	12.64 ± 0.13	6.92	18	260.1 ± 1.0	14.45
012	15.41 ± 0.13	10.99	18	191.3 ± 0.1	10.63
011	19.06 ± 0.18	17.55	14	152.0 ± 0.2	10.86

*Error limits shown are standard deviations and are for the indicated measured parameter only.



^a REFERENCED DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY STANDARD.

^b ²⁴¹Am CONTENT OF TOTAL PLUTONIUM WAS MEASURED TO BE 0.13 wt% DURING APRIL AND MAY OF 1972. EXPERIMENTS PERFORMED SEPTEMBER 1978.

^c 35.6 mm FOR TYPE 3.1 FUEL PINS.

Figure 2.4.1: FTR Fuel Pin (from Ref. 17).

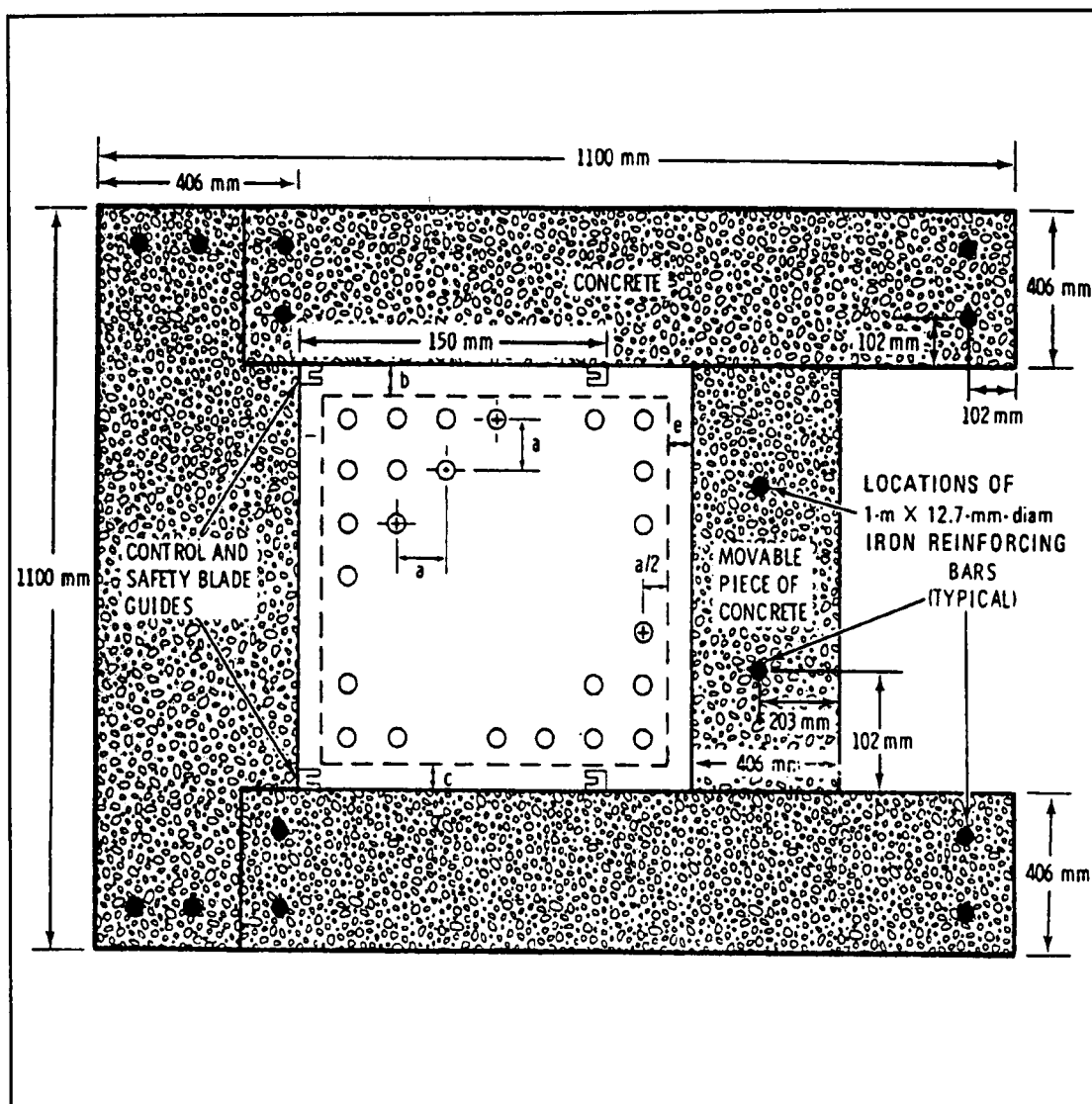


Figure 2.4.2: Top View of FTR Fuel Assembly (from Ref. 17).

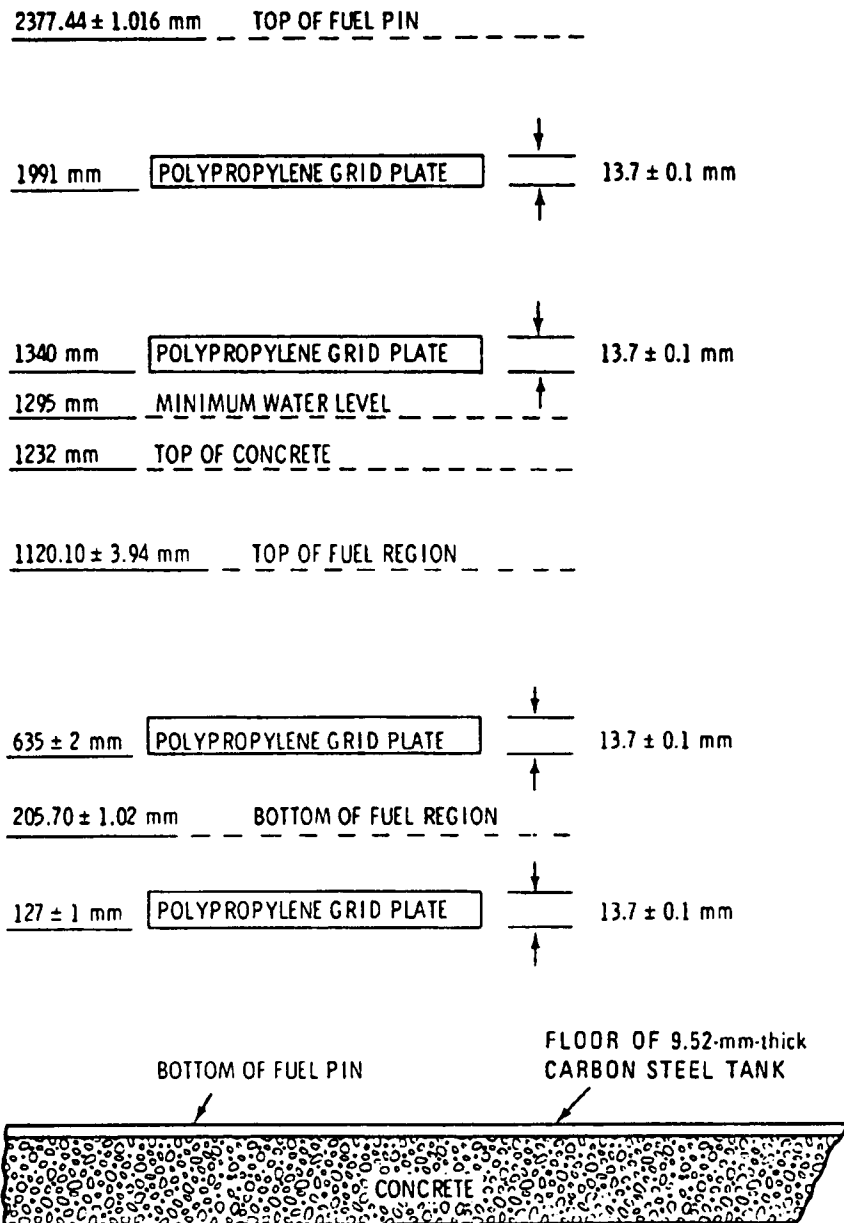


Figure 2.4.3: Elevations for FTR Fuel Pins in Water (from Ref. 17).

Using the atomic number densities along with the geometrical and critical parameters, four separate KENO V.a input files were developed to model each of the four FTR experiments. The 238 and 44 energy group ENDF/B-V, 218 and 27 energy group ENDF/B-IV and 16 energy group Hansen-Roach cross section libraries were used in the KENO V.a calculations. The results obtained with the ENDF/B-V and ENDF/B-IV cross sections are presented in Table 2.4.5 while the Hansen-Roach results are presented in Appendix C. As shown in Table 2.4.5 for the 238 group ENDF/B-V library, the calculated k_{eff} 's for the concrete reflected experiments are within 0.4% of $k_{\text{eff}} = 1.0$ for all of the experiments except experiment 012, which is within 1% of 1.0. In comparison to the 218 group ENDF/B-IV library, there is good agreement in the calculated k_{eff} 's obtained with the 238 group library for each of the experiments. With regard to the broad group cross section libraries, the calculated k_{eff} 's are within $\pm 2.0\%$ of 1.0, and the ENDF/B-V results are in good agreement with the ENDF/B-IV results.

Table 2.4.5: Calculated K_{eff} 's for Concrete Reflected FTR Experiments.

Exp. #	k_{eff} ENDF/B-V	Avg. Fission Group (AFG)	k_{eff} ENDF/B-IV	Avg. Fission Group (AFG)
	238 Group 44 Group		218 Group 27 Group	
010	1.0040 ± 0.0034	184.94	1.0064 ± 0.0030	177.64
	1.0207 ± 0.0030	30.82	1.0112 ± 0.0034	20.67
007	1.0097 ± 0.0033	200.45	1.0119 ± 0.0037	192.38
	1.0133 ± 0.0033	33.65	1.0118 ± 0.0034	22.28
012	0.9904 ± 0.0035	206.12	0.9865 ± 0.0031	198.34
	0.9963 ± 0.0033	34.83	0.9929 ± 0.0032	22.96
011	1.0037 ± 0.0031	210.08	0.9993 ± 0.0036	202.14
	1.0123 ± 0.0033	35.71	1.0070 ± 0.0033	23.45

2.5 Subcritical Limit for ^{233}U Systems

The objective of validating KENO V.a and the ENDF/B-V ^{233}U cross section library with experimental data is to demonstrate their ability to predict the behavior of systems containing ^{233}U material. After completing several benchmark KENO calculations with a particular cross section library, the calculations are used to determine an upper safety limit (USL) for subcriticality. Statistically, the USL is a single-sided, uniform-width, closed-interval, lower tolerance band for calculated k_{eff} values of critical systems.¹⁸ If a calculated k_{eff} plus two standard deviations lies below the USL, then the system is considered acceptably subcritical.

2.5.1 Description of Method

The following discussion is based upon the methodology of determining a USL which is presented in three references.¹⁸⁻²⁰

In order to determine a USL, the calculated k_{eff} 's are determined as a function of some independent variable which is system-dependent (i.e. $k_{\text{eff}}(x)$). In principle, the independent variable should be a parameter which is easily identifiable for the modeled critical experiments and the actual system to be modeled with KENO V.a and the appropriate cross section library. Some examples of independent variables are average fission group (AFG) which is the average energy group causing fissions, hydrogen-to-fissile atom ratio, ratio of thermal neutron absorption to total absorption, fissile material

density, etc. Once an independent variable is selected, a range of applicability is defined based on the selected variable.

After selecting an appropriate independent variable, a linear least squares fit of the calculated k_{eff} values is determined:

$$k_{eff}(x) = \hat{\alpha} + \hat{\beta} (x - \bar{x}) . \quad (2.5.1.1)$$

$\hat{\alpha}$ = mean k_{eff} .

$\bar{x}$ = mean x-value.

$\hat{\beta}$ = slope of regression line.

The slope of the fitted line is determined from the validation data as follows:

$$\hat{\beta} = \frac{\sum_{i=1}^n (x_i - \bar{x}) (k_{eff}^i - \hat{\alpha})}{\sum_{i=1}^n (x_i - \bar{x})^2} . \quad (2.5.1.2)$$

n = total number of calculated experiments.

k_{eff}^i = k_{eff} value corresponding to x_i .

Once the linear least squares fit of the data is determined, a lower one-sided confidence band is determined about the fit. The width of the confidence band is a product of a multiplier, δ , and the standard deviation of the fit, σ , such that there is at least α confidence that a proportion, P , of the population of future calculations will lie

above the lower tolerance band given by the following equation:

$$USL = k_{eff}(x) - \delta \sigma. \quad (2.5.1.3)$$

The α confidence is given by the equation:

$$\alpha = 1 - \left(\frac{\gamma_1}{2} \right) - \gamma_2. \quad (2.5.1.4)$$

$1 - \gamma_1$ = one-sided confidence band about the linear fit.

$1 - \gamma_2$ = confidence on the variance of the fit.

For example, if $\alpha = 95\%$ (0.95) and $P = 99.9\%$ (0.999), then there is at least 95% confidence that 99.9% of future calculations of critical systems will give a k_{eff} value above the lower tolerance band (i.e. USL). For criticality safety concerns, if a calculated k_{eff} value is below the USL, there is a 95% confidence that the probability of the system being critical is less than 10^{-6} .⁴¹

The multiplier, δ , in Eq. (2.5.1.3) is a function of several parameters as shown below:

$$\delta = C^* + z_{(2P-1)} \left[\frac{n-2}{\chi^2_{(n-2)(1-\gamma_2)}} \right]. \quad (2.5.1.5)$$

$z_{(2P-1)}$ = minimum acceptable confidence.

χ^2 = chi square value.

C^* = constant.

The determination of the constant, C^* , in Eq. (2.5.1.5) is accomplished in a step-wise fashion. Bowden and Graybill¹⁹ identify three constant parameters g , h and ρ given by the following equations:

$$g = \left[\frac{1}{n} + \frac{(a - \bar{x})^2}{ns^2} \right]^{1/2}. \quad (2.5.1.6)$$

$$h = \left[\frac{1}{n} + \frac{(b - \bar{x})^2}{ns^2} \right]^{1/2}. \quad (2.5.1.7)$$

$$\rho = \frac{1 + (a - \bar{x})(b - \bar{x})}{s^2 \left[1 + \frac{(a - \bar{x})^2}{s^2} \right]^{1/2} \left[1 + \frac{(b - \bar{x})^2}{s^2} \right]^{1/2}}. \quad (2.5.1.8)$$

where

a = lower bound of x variable.
 b = upper bound of x variable.
 s^2 = sample variance.

The sample variance, s^2 , is given by the following equation:

$$s^2 = \frac{1}{n} \sum_{i=1}^n (x_i - \bar{x})^2. \quad (2.5.1.9)$$

Using n , ρ and the ratio of g to h (i.e., $A = g/h$), Bowden and Graybill provide a Table of values for a constant variable D . The C^* defined in Eq. (2.5.1.5) is calculated as follows:

$$C^* = Dh. \quad (2.5.1.10)$$

For a proportion, P, of the population, the $z_{(2P-1)}$ and χ^2 values in Eq. (2.5.1.5) are tabulated by Ott.³⁹

Once δ is determined, the estimated variance of the fit is calculated by the following equation:

$$\sigma^2 = \frac{1}{(n-2)} \left[\sum_{i=1}^n (k_{eff}^i - \hat{a})^2 - \frac{\left[\sum_{i=1}^n (x_i - \bar{x}) (k_{eff}^i - \hat{a}) \right]^2}{\sum_{i=1}^n (x_i - \bar{x})} \right]. \quad (2.5.1.11)$$

The standard deviation of the fit is obtained by taking the square root of Eq. (2.5.1.11). Furthermore, the upper safety limit is obtained by substituting the standard deviation of the fit and δ into Eq. (2.5.1.3).

2.5.2 Determination of Subcritical Limit

At the start of this research it was anticipated that both the 44 and 238 group ENDF/B-V ^{233}U cross sections would be needed in the evaluations of the building 3019 storage wells and the Molten Salt Reactor fuel drain tanks. Therefore, a USL was derived for each library based on the methodology outlined in Section 2.5.1 and the validation of each cross section library for ^{233}U systems presented in Section 2.3. The following discussion is a summary of the USL calculations.

Prior to determining the USL, an independent variable was selected to base the evaluation of the calculated k_{eff} values presented in Section 2.3. The average energy group in which fissions are occurring (AFG) was selected as the independent variable because it is easily identifiable in the KENO V.a output for a particular system. Also, the AFG permits application to a broad range of problems. In order to determine the USL for the fine and broad group ENDF/B-V libraries, the statistical parameters outlined in Section 2.5.1 were determined for the 44 and 238 group ENDF/B-V calculated k_{eff} values presented in Section 2.3 and are tabulated in Table 2.5.2.1.

Table 2.5.2.1: Statistical Parameters for USL Calculations.

Parameter	44 Group	238 Group
$\bar{x}$	30.373921	173.97392
$\hat{\alpha}$	1.008960	1.006364
n	51	51
a	7.88	22.46
b	38.28	219.11
$\hat{\beta}$	4.13178e-4	7.40111e-5
s^2	116.761788	4946.948
g	0.323383	0.332563
h	0.173506	0.166381
$ \rho $	0.182798	0.135514
C^*	0.353953	0.339418
z_{2P-1}	3.273	3.273
χ^2	86.66	86.66
σ^2	2.124228e-4	1.814607e-4

Based on the parameters in Table 2.5.2.1, a linear least squares fit was determined for the calculated k_{eff} values for the 44 and 238 group ENDF/B-V libraries and is shown in Figs. 2.5.2.1 and 2.5.2.2, respectively. The equation for the fitted line as a function of the average fission group for the 44 and 238 group libraries is given by Eqs. (2.5.2.1) and (2.5.2.2), respectively.

$$k_{eff}(AFG) = 0.99641 + 4.13178e-04 * (AFG) . \quad (2.5.2.1)$$

$$k_{eff}(AFG) = 0.99349 + 7.40111e-05 * (AFG) . \quad (2.5.2.2)$$

Using the parameters in Table 2.5.2.1, a lower one-sided confidence band width of 0.04099 and 0.03772 was calculated for the 44 and 238 group cross section libraries, respectively. An upper safety limit of 0.9554 and 0.9557 was determined by subtracting the appropriate band width for the 44 and 238 group libraries, respectively. The calculated USL represents a lower tolerance band with 95% confidence on 99.9% of the data. Moreover, the equation for the USL as a function of AFG is given for the 44 and 238 group ENDF/B-V cross section libraries by Eqs. (2.5.2.3) and (2.5.2.4), respectively and is represented by the dashed line in Figs. 2.5.2.1 and 2.5.2.2, respectively.

$$USL(AFG) = 0.9554 + 4.1317E-04 * (AFG) . \quad (2.5.2.3)$$

$$USL(AFG) = 0.9557 + 7.4011e-05 * (AFG) . \quad (2.5.2.4)$$

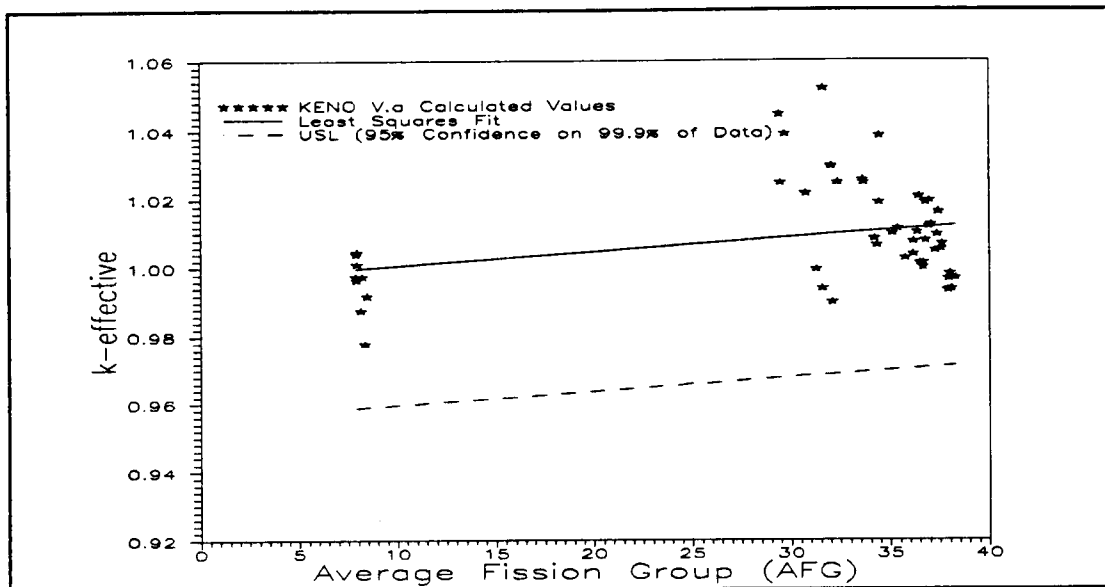


Figure 2.5.2.1: Calculated k_{eff} 's for 51 ^{235}U Critical Experiments Using KENO V.a with 44 Energy Group ENDF/B-V Cross Sections.

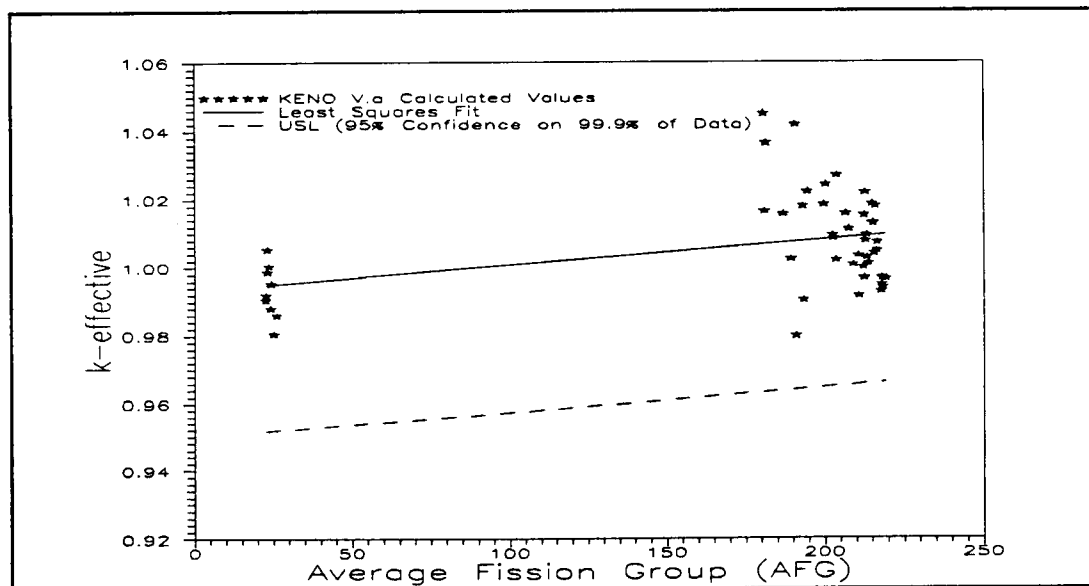


Figure 2.5.2.2: Calculated k_{eff} 's for 51 ^{235}U Critical Experiments Using KENO V.a with 238 Energy Group ENDF/B-V Cross Sections.

Chapter Three

Evaluation of Building 3019

3.1 Facility Description

3.1.1 Configuration of Storage Wells

A substantial amount of ^{233}U is stored in the Radiochemical Processing Plant (RPP) which is identified as Building 3019 at ORNL. The fissile inventory which consists of approximately 1,379 kg of uranium is housed in retrievable storage wells which are encased in concrete within Building 3019. The following facility description is a brief overview of the RPP and is provided in References 15 and 21. A plan view of the Radiochemical Processing Plant is presented in Fig. 3.1.1. As shown in Fig. 3.1.1, Building 3019 consists of 7 shielded hot cells which are located in the southernmost part of the facility. For clarity, plan views of the first and second floors of the hot cells are provided in Fig. 3.1.2. Each hot cell has a floor to ceiling height of ~ 822.96 cm., and all of the hot cells have floor dimensions of ~ 609.6 by ~ 609.6 cm. except for cell 1 which has floor dimensions of ~ 304.8 by ~ 609.6 cm. Moreover, each of the cells is separated by an interior 152.4 cm. reinforced concrete wall except for cells 6 & 7 which are only separated by a partial wall as shown in Fig. 3.1.2. In like manner, all of the cell outer walls are built of 152.4 cm. thick reinforced concrete except for the walls located on the south side which are only 121.92 cm. thick.

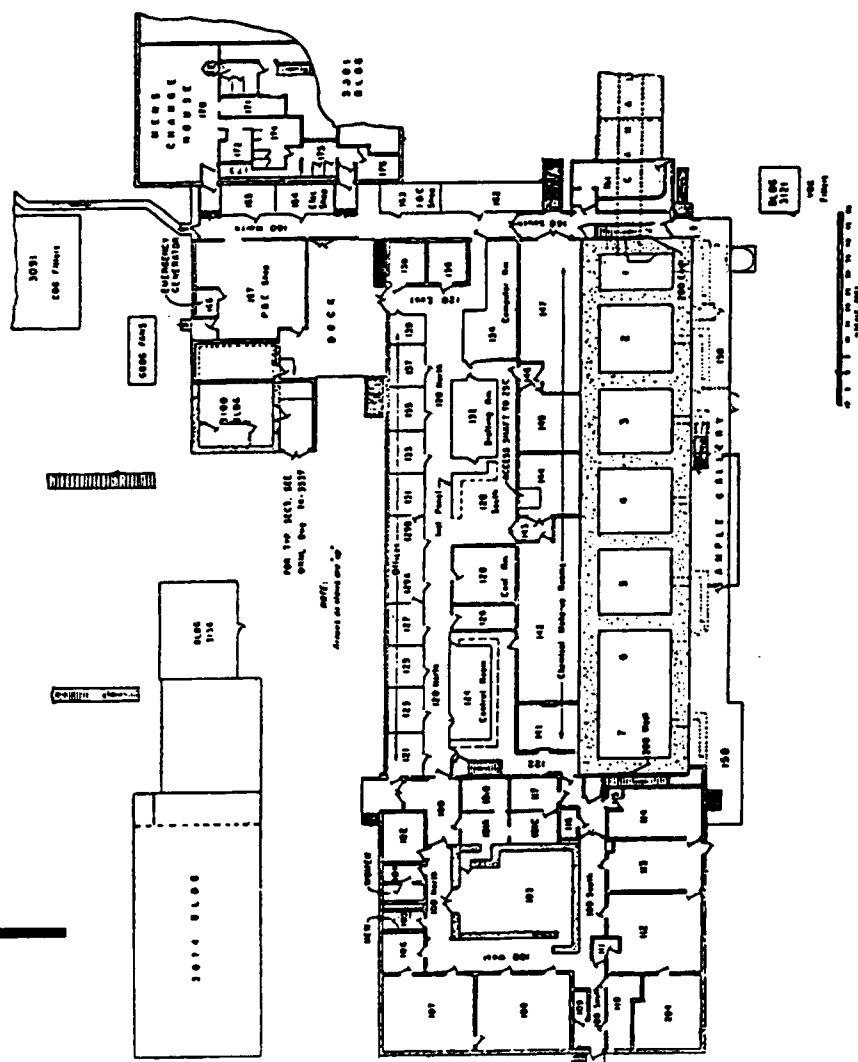


Figure 3.1.1: Plan View of Building 3019 (from Ref. 15).

-2-

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A cross sectional view of cell 1 located at the eastern end of Building 3019 is shown in Fig. 3.1.3. As shown in this cross sectional view, there is a penthouse region located above each of the hot cells. Furthermore, the concrete floor of the penthouse serves as the roof for the hot cells, and located within the southwestern corner of each cell roof is a 274.32 by 274.32 cm. equipment hatch which is accessible from the penthouse. It should be noted that the hatch above cell 4 and the corresponding cell space beneath the hatch have been replaced by an array of storage wells.

The group of storage wells which replaced the equipment hatch in cell 4 consists of an array of 68 wells arranged in a triangular pitch of 32.385 cm. and is located in the southwestern corner of cell 4. To help visualize these wells, an isometric view of the cell 4 wells is presented in Fig. 3.1.4. In addition, an identification pattern for the wells is presented in Fig. 3.1.5. The 68 wells have an overall height of 975.36 cm. and extend from 30.48 cm. above the equipment hatch to the concrete floor of cell 4. Each well is encased in concrete which has a hexagonal structure as depicted in Fig. 3.1.4, and the chemical composition of the concrete³ is tabulated in Table 3.1.1. As shown in Fig. 3.1.5, 45 of the wells are identified as Type A wells while the remaining 23 located in the eastern part of the array are identified as Type B wells. The Type A wells are identified as a mild steel schedule 40 pipe with inside and outside radii of 5.08 and 5.682 cm., respectively. In contrast, the Type B wells are composed of a mild steel and have inside and outside radii of 5.715 and 6.35 cm., respectively. Each of the 68 wells is capped and shielded with a removable 60.96 cm. lead plug.

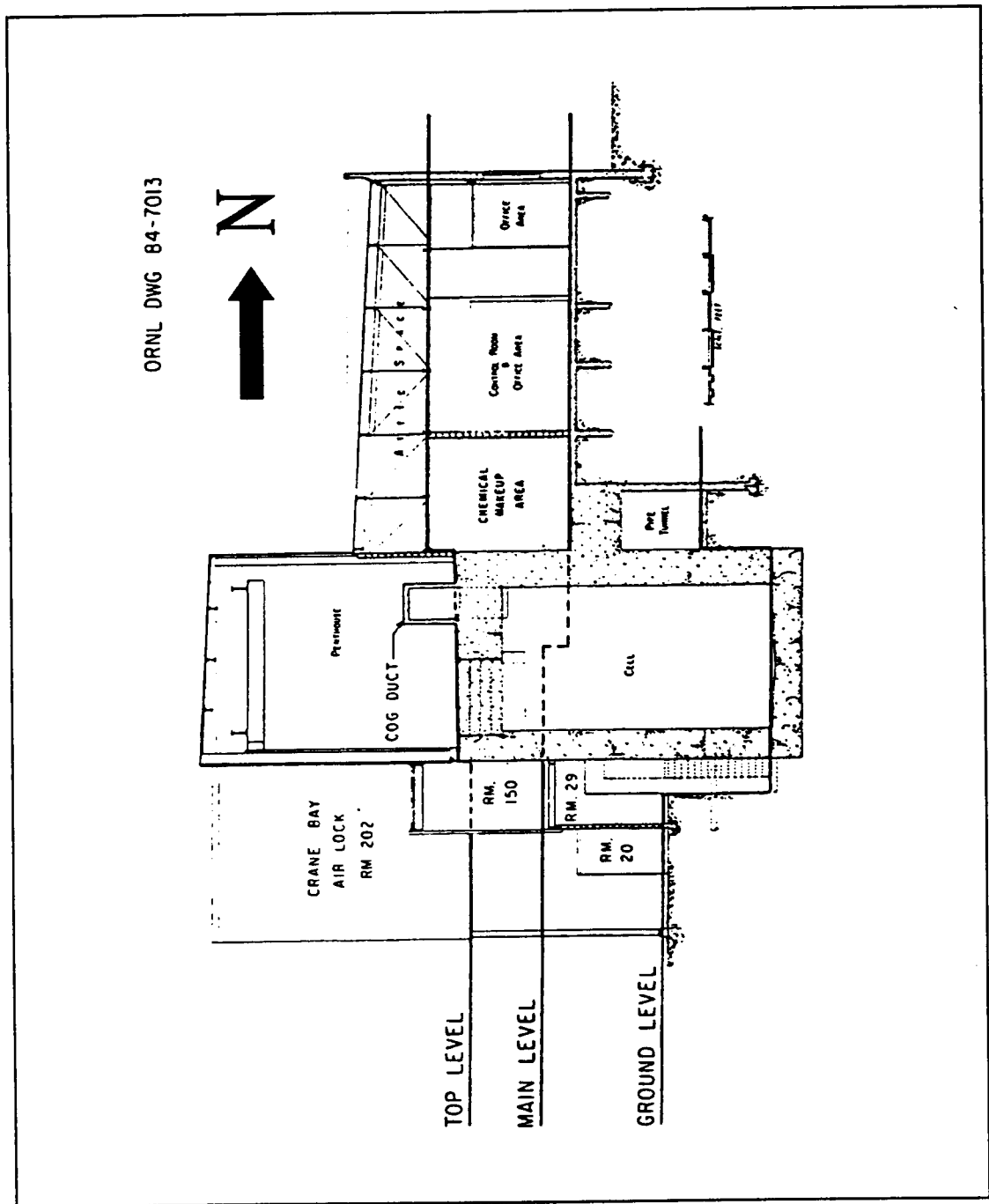


Figure 3.1.3: Cross Sectional View of Cell 1 (from Ref. 15).

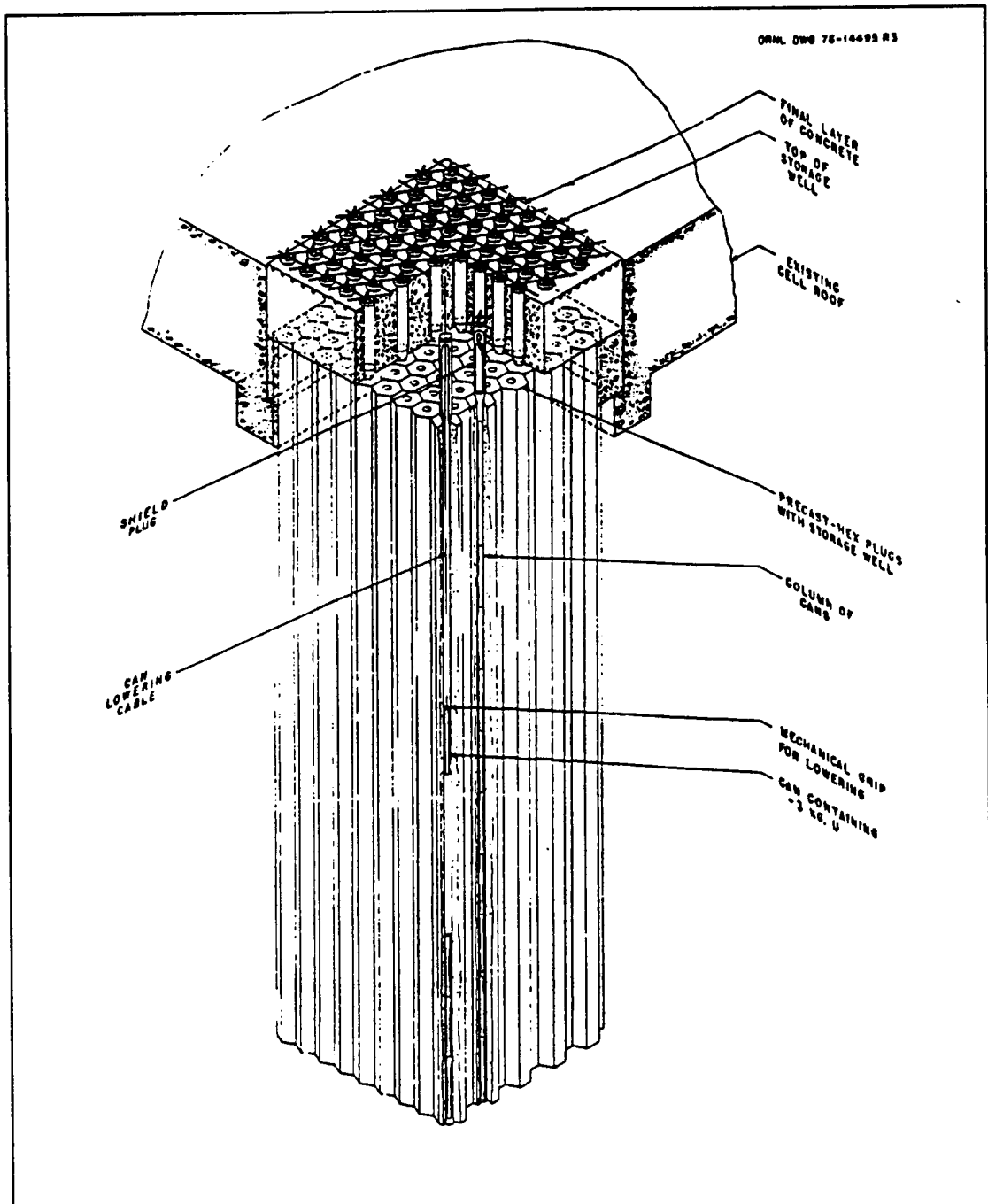


Figure 3.1.4: Isometric View of Cell 4 Wells (from Ref. 15).

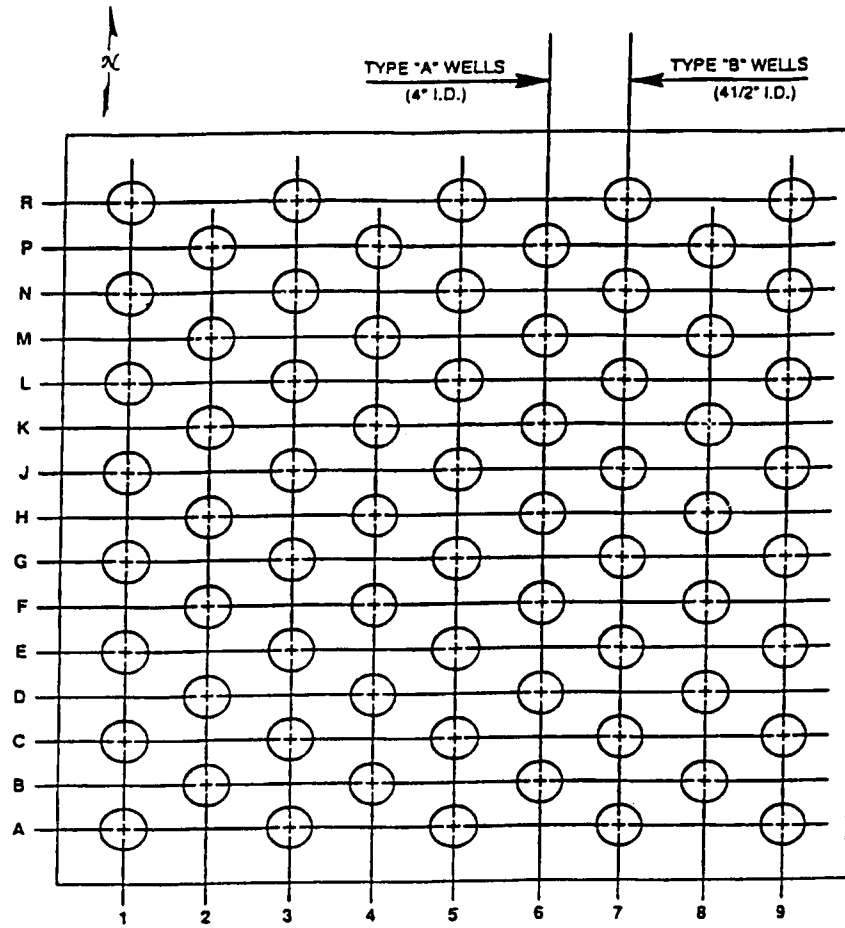


Figure 3.1.5: Cell 4 Wells Identification Pattern (from Ref. 15).

Table 3.1.1: Chemical Composition of Interstitial Cell 4 Concrete (Ref. 3).

Element	Atomic Number Density
H	4.54991e-3
C	3.82155e-4
O	2.75330e-2
Na	9.58077e-5
Mg	1.18155e-4
Al	7.68645e-4
Si	1.24659e-2
S	2.52827e-5
K	2.15730e-4
Ca	1.71935e-3
Fe	2.29130e-4
Zn	9.36739e-6

In addition to the 68 cell 4 storage wells are 26 inter-cell or in-line storage wells located in the concrete walls between cells 2 & 3, 3 & 4 and 4 & 5. An identification pattern for the inter-cell wells is presented in Fig. 3.1.6. To help visualize the configuration of the inter-cell wells, an isometric view of the wells located between cells 2 & 3 is presented in Fig. 3.1.7. As shown in Fig. 3.1.7, inter-cell wells 2 & 3 are arranged in a triangular lattice while the remaining inter-cell wells are arranged in single row. All of the wells are arranged in a 30.48cm. pitch and consist of a mild steel schedule 40 pipe having the same radial dimensions of the Type A wells located in the cell 4 storage array. Furthermore, each well is capped and shielded with a 20.32 cm.

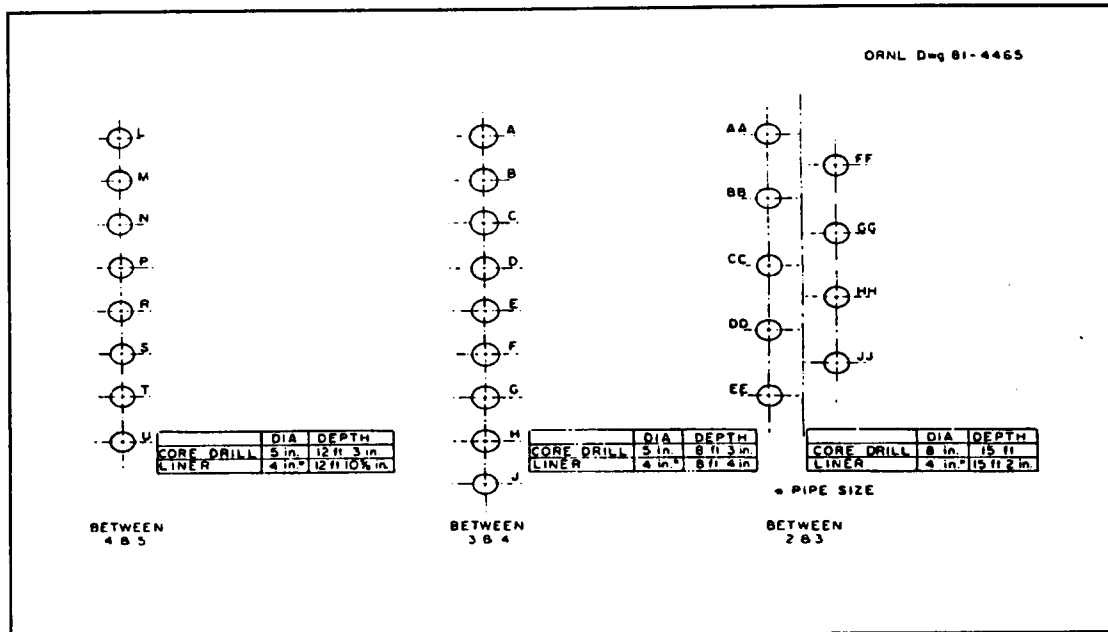


Figure 3.1.6: Inter-cell Wells Identification Pattern (from Ref. 15).

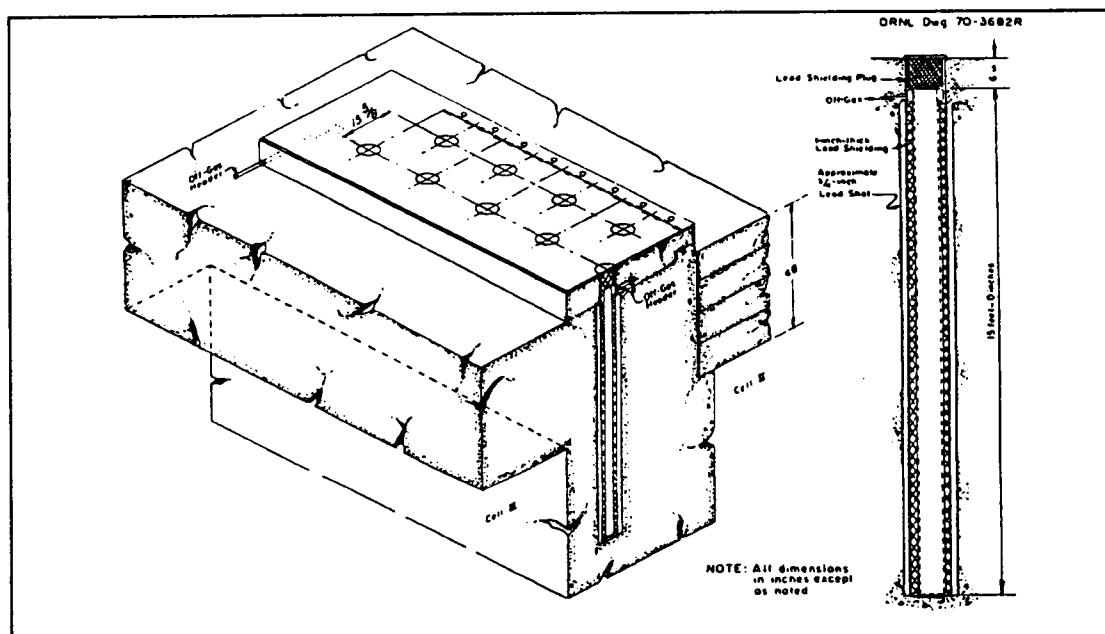


Figure 3.1.7: Isometric View of Inter-cell Wells 2 & 3 (from Ref. 15).

thick removable lead plug and is accessible from the penthouse located above the storage wells. Surrounding each steel pipe of inter-cell wells 2 & 3 is lead shielding followed by concrete. In contrast, in-line wells 3 & 4 and 4 & 5 only have concrete encasing the steel pipes. The composition of the concrete³ encasing the inter-cell wells is tabulated in Table 3.1.2. The chemical analysis of the concrete presented in Reference [3] only provided an upper limit for the water content. Therefore, the lower bound for the water content (i.e. no hydrogen and maximum oxygen content) is also presented in Table 3.1.2. The effect of the water content in the concrete is evaluated in Section 3.2.

Table 3.1.2: Chemical Composition of the Inter-cell Wall Concrete (Ref 3).

Element	Atomic Number Density
H (max.)	4.54991e-3
H (min.)	0.0
C	3.82155e-4
O (min.)	2.75330e-2
O (max.)	2.78196e-2
Na	9.58077e-5
Mg	1.18155e-4
Al	7.68645e-4
Si	1.24659e-2
S	2.52827e-5
K	2.15730e-4
Ca	1.71935e-3
Fe	2.29130e-4
Zn	9.36739e-6

3.1.2 Inventory of Fissile Material

All of the ^{233}U inventory in Building 3019 is stored in the cell 4 wells along with inter-cell wells 3 & 4 and 4 & 5. Currently, the inter-cell 2 & 3 storage wells are empty and are not evaluated in this work. Henceforth, the general referral to inter-cell or in-line wells identifies the 9 wells located between cells 3 & 4 and 8 wells located between cells 4 & 5.

All of the fissile material stored in the Cell 4 and inter-cell wells have a nominal $\text{H/U} \leq 3$.²² Moreover, the chemical form of the fissile material stored in the wells is either oxide, UF_4 or U metal.¹⁵ All of the material stored in the inter-cell wells is UO_2 except for three U-metal hemispheres. The uranium stored in the inter-cell wells is $\geq 95\%$ of ^{233}U by weight. Regarding Cell 4, approximately 75% of the facility inventory is U_3O_8 which is referred to as CEUSP (Consolidated Edison Uranium Solidification Project) material. The isotopic percentages of the CEUSP material¹⁵ is presented in Table 3.1.2.1. The remaining ^{233}U material in Cell 4 is in the form of UO_2 , UF_4 or the

Table 3.1.2.1: Isotopic Percentages of CEUSP Material.

Isotope	Weight Percent
^{233}U	9.7
^{234}U	1.4
^{235}U	76.5
^{236}U	5.6
^{238}U	6.8

disassembled ^{233}U metal sphere Jezebel. The ^{233}U weight percentages for the UO_2 , UF_4 and metal in Cell 4 is ≥ 60 , ≥ 90 and 100^* , respectively. A detailed fissile material inventory for the storage wells is provided in Reference [15] and is presented for the Cell 4 and inter-cell wells in Tables 3.1.2.2 and 3.1.2.3, respectively. It should be noted that several of the storage wells are only partially filled.

Table 3.1.2.2: Fissile Material Inventories for Building 3019 Cell 4 Storage Wells.

Well Identification	Material	Loading (kg U/ft)
1A	UO_2	1.28
1C	UO_2	1.36
1E	Jezebel	1.85
1G	UO_2	1.38
1J - 1R	Empty	—
2B	Empty	—
2D - 2P	CEUSP	0.96 - 1.58
3C - 3R	CEUSP	0.96 - 1.58
4D - 4P	CEUSP	0.96 - 1.58
5C - 5R	CEUSP	0.96 - 1.58
6D - 6P	CEUSP	0.96 - 1.58
7L, 7R	CEUSP	0.96 - 1.58
3A, 4B	Empty	—
5A, 6B	Empty	—
7A	UO_2	1.57

*The ^{233}U wt% for Jezebel is assumed to be 100%.

Table 3.1.2.2 (Cont'd)

Well Identification	Material	Loading (kg U/ft)
7C	UO ₂	1.35
7E	UO ₂	0.93
7G	UO ₂	1.00
7J	UO ₂	0.37
7N	UF ₄	1.34
8D, 8H	UF ₄	1.32, 1.14
8B, 8F	Empty	—
8K, 8M	Empty	—
8P	UO ₂	1.89
9A	UO ₂	0.12
9C - 9G	Empty	—
9J	UO ₂	0.62
9L	UO ₂	0.75
9N	UO ₂	0.75
9R	UO ₂	0.78

Table 3.1.2.3: Fissile Material Inventories for Building 3019 Inter-cell Storage Wells.

Well Identification	Material	Loading (kg U/ft)
Inter-cell 3 & 4 Wells		
A	UO ₂	0.81
B - G	UO ₂	0.81 - 0.84
H - J	UO ₂	0.74, 0.75
Inter-cell 4 & 5 Wells		
L	UO ₂	0.93
M	UO ₂ & 3 metal hemispheres	2.13*
N	UO ₂	1.94
P	Empty	—
R	UO ₂	0.97
S, T	Empty	—
U	UO ₂	0.92

All of the fissile material in the RPP storage facility is contained within steel canisters. According to Reference [15], the composition of the steel is unspecified, and the inside and outside diameters of the canisters are 8.255 and 8.5725 cm., respectively. Moreover, the canister heights range from 7.54 to 61 cm. A schematic diagram of a typical well housing the fissile material canisters is presented in Fig. 3.1.2.1.

*Peak loading is due to the three metal hemispheres which have center to center spacing of 43.18 cm.

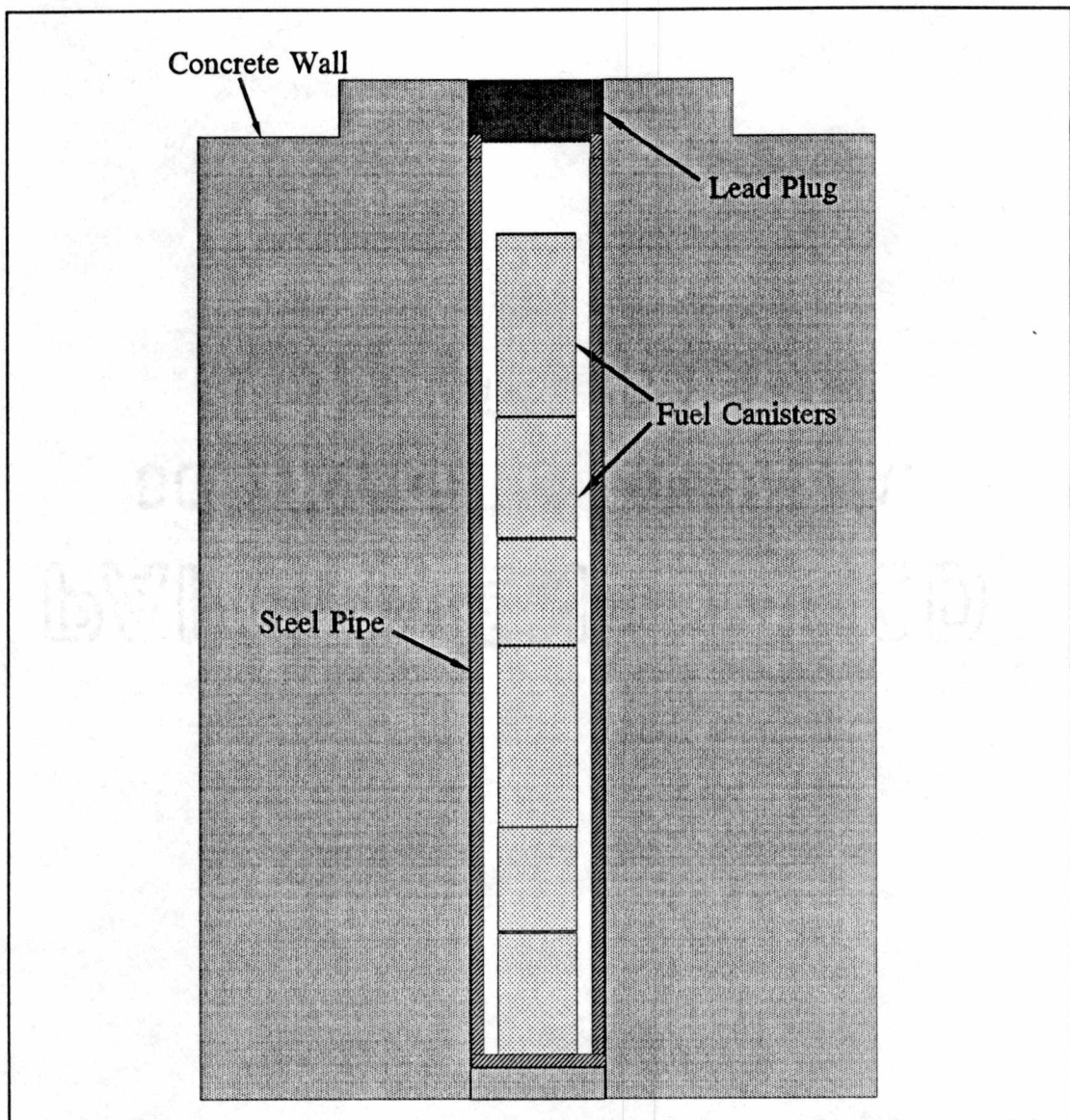


Figure 3.1.2.1: Schematic Diagram of Fissile Material Canisters Located in a Typical Storage Well in Building 3019.

3.1.3 Presence of Neutron Poisons

According to Reference [15], the canisters containing the CEUSP material in the Cell 4 storage wells also contain on average 800 g. of Cd in the form of CdO and 100 g. of Gd in the form of Gd₂O₃. Since Cd and Gd are strong neutron absorbers, it is believed that these absorbers will have a negative effect on the reactivity of the system; thus, Cd and Gd are not included in the criticality safety evaluation of the Cell 4 storage wells. Furthermore, the inclusion of Gd and Cd in the study would require periodic verification of their presence in the CEUSP canisters.

3.1.4 Integrity of the Storage Facility

In the event of a natural disaster such as an earthquake, the structural integrity of the storage facilities in Building 3019 is safe according to Reference [3], which states "that all fuel storage facilities are safe from disruption by design basis earthquake or other natural hazards." It is assumed that the integrity of the containers within the wells remains intact. Likewise, the steel lining and lead plugs are assumed to retain their structural integrity over time. With regard to water flooding within the storage wells, ORNL Procedure RDF-OP-024 specifies that water ingress into the wells is not credible.¹⁵ Therefore, the above natural disasters are not credible accident scenarios and are not addressed in the criticality safety evaluation of the RPP storage wells.

3.2 Facility Model

Using the information presented in Section 3.1, a substantial amount of work was performed in order to develop a "worst case" KENO V.a model of the storage wells in Building 3019. The model was developed in a series of calculational steps in order to alleviate any concerns regarding the sensitivity of system parameters in the model. These calculational steps are summarized in the sections that follow. All of the KENO V.a calculations for Building 3019 were performed with the 238 energy group ENDF/B-V neutron cross section library.

3.2.1 Loading of Fuel

The fissile material loadings are presented in Section 3.1.2 for the Cell 4 and inter-cell storage wells and are in terms of kg U/ft. As noted in the material loadings, several of the wells are only partially filled. The following assumptions regarding the fissile loadings in the KENO V.a model are as follows:

- 1.) Only the maximum fissile material loading for a chemical form (e.g. UO_2 , CEUSP, UF_4 or metal) in a particular set of wells is used in the model.
- 2.) Furthermore, in the model of each well containing fissile material, it is assumed that a well is completely filled with the maximum fissile material specified.

For example, the UO_2 loadings in the Cell 4 storage array range from 0.12 to 1.89 kg U/ft. Despite the wide range of loadings for UO_2 , all of the Cell 4 wells containing UO_2 are assumed to be completely filled with the maximum loading of 1.89 kg U/ft.

Similarly, the UO_2 loadings in the inter-cell wells range from 0.74 to 2.13 kg U/ft. As in the Cell 4 case, all of the inter-cell wells containing UO_2 are assumed to be completely filled with the maximum loading of 2.13 kg U/ft. It should be noted that the specific configuration of In-line 4 & 5 well M, which contains three metal hemispheres, is discussed in greater detail in Section 3.2.4. A summary of the fissile material loadings used in the KENO V.a model of the storage facility is presented in Table 3.2.1.1.

Table 3.2.1.1: Fissile Material Loadings Used in KENO V.a Model of the Building 3019 Storage Wells.

Well Identification	Material	Loading (kg U/ft)
Cell 4 Wells		
1A, 1C, 1G, 7A, 7C, 7E, 7G, 7J, 8P, 9A, 9J, 9L, 9N, 9R	UO_2	1.89
1E	Jezebel	1.85
2D-2P, 3C-3R, 4D-4P, 5C-5R, 6D-6P, 7L, 7R	CEUSP	1.58
7N, 8D, 8H	UF_4	1.34
Inter-cell Wells		
All Inter-cell 3&4 Wells	UO_2	2.13
Inter-cell 4&5 Wells: L, M*, N, R, U		

*The three metal hemispheres are assumed to be at theoretical density (19.05 g/cm^3).

3.2.2 Isotopics of the Fuel

The isotope percentages of the fissile material are presented in Section 3.1.2. All of the uranium present in the storage wells is assumed to be 100 wt%* ^{233}U except for the CEUSP material. The CEUSP material, which contains a substantial amount of ^{235}U , is explicitly modeled according to the isotopic percentages for this material which are repeated for clarity in Table 3.2.2.1.

Table 3.2.2.1: Isotopic Percentages of CEUSP Material.

Isotope	Weight Percent
^{233}U	9.7
^{234}U	1.4
^{235}U	76.5
^{236}U	5.6
^{238}U	6.8

3.2.3 Double Batching of Fuel

Double batching of fissile material within a canister is possible.⁴¹ Based on the modeling of the fissile loadings presented in Section 3.2.1, the double batching of fuel is inherently accounted for in the nominal model of the existing storage inventory. For example, all of the UO_2 in the cell 4 array is loaded at 1.89 kg U/ft, which is ~ 16 times

*The ENDF/B-V cross section library does not have a Chi spectrum for ^{233}U . Therefore, the Chi spectrum for ^{235}U must be used. Consequently, the actual weight percentages used are 99.99% ^{233}U and 0.01% ^{235}U .

higher than the minimum loading (0.12 kg U/ft) in the Cell 4 array. Furthermore, all of the wells are completely filled with fissile material despite the fact that some of the wells are only partially filled.

Currently, nineteen of the storage wells are empty; however, double batching of the fuel in these wells is possible and is addressed in Chapter four.

3.2.4 Model of a Single Well

After establishing the loading of the fissile material within a well, a model of a single well is developed through a series of KENO V.a calculations. One of the first steps is to evaluate the effect of the steel fuel canisters on the effective multiplication factor of an array of wells.⁴¹ An array of 8 Type A wells located on a 30.48 cm. pitch is modeled using KENO V.a. The wells are 392.75 cm. in height and are encased in a 152.4 x 243.84 x 1097.28 cm. concrete wall (inter-cell type composition) and capped with a 20.32 cm. thick lead plug. In the first calculation, each of the wells is completely filled with carbon steel canisters having an inside radius of 4.1275 cm. and a height of 7.54 cm. Each canister is completely filled with UO_2 ($\text{H/U} = 3$) at a linear loading of 2.13 kg U/ft. The calculated k_{eff} for this array of wells is 0.6949 ± 0.0020 . In the second calculation, the cans are removed, and each well is completely filled with a single column of UO_2 having the same radial dimensions as a fuel canister. The H/U and linear loading are the same. The calculated k_{eff} for the second configuration is 0.7181 ± 0.0021 indicating a $\sim 3.3\%$ increase in k_{eff} due to the removal of the fuel canisters. As

a result, the fissile material in a single well is modeled as a single column rather than an array of canisters.

In the previous calculation, the radius of the column of fuel is restricted to the inside radius of a canister (i.e. 4.1275 cm.); however, the inside radius of a storage well is 5.08 cm. In order to evaluate the sensitivity of k_{eff} to the radial dimension of the fuel, the radius of the fuel (i.e. UO_2) is increased to 5.08 cm., thus spreading the fuel over the entire well and decreasing the density (i.e. the linear loading is the same).⁴¹ The resulting calculated k_{eff} is 0.7268 ± 0.0021 which is $\sim 1.2\%$ higher than the single column of fissile material configuration (i.e. $k_{\text{eff}} = 0.7181$). Consequently, the fissile material in a single well is modeled as a single column of fuel spread over the entire storage well as shown in Fig. 3.2.4.1.

In the previous calculations of this section, the fissile material is assumed to be uniformly distributed over the entire canister or storage well; however, it is possible for the fuel to shift and settle to the bottom of a canister over time.⁴¹ Despite the neglect of the fuel canisters in the KENO V.a model, the fuel is actually stored within steel canisters. Therefore, the re-distribution of the fissile material must be evaluated with respect to a single canister.

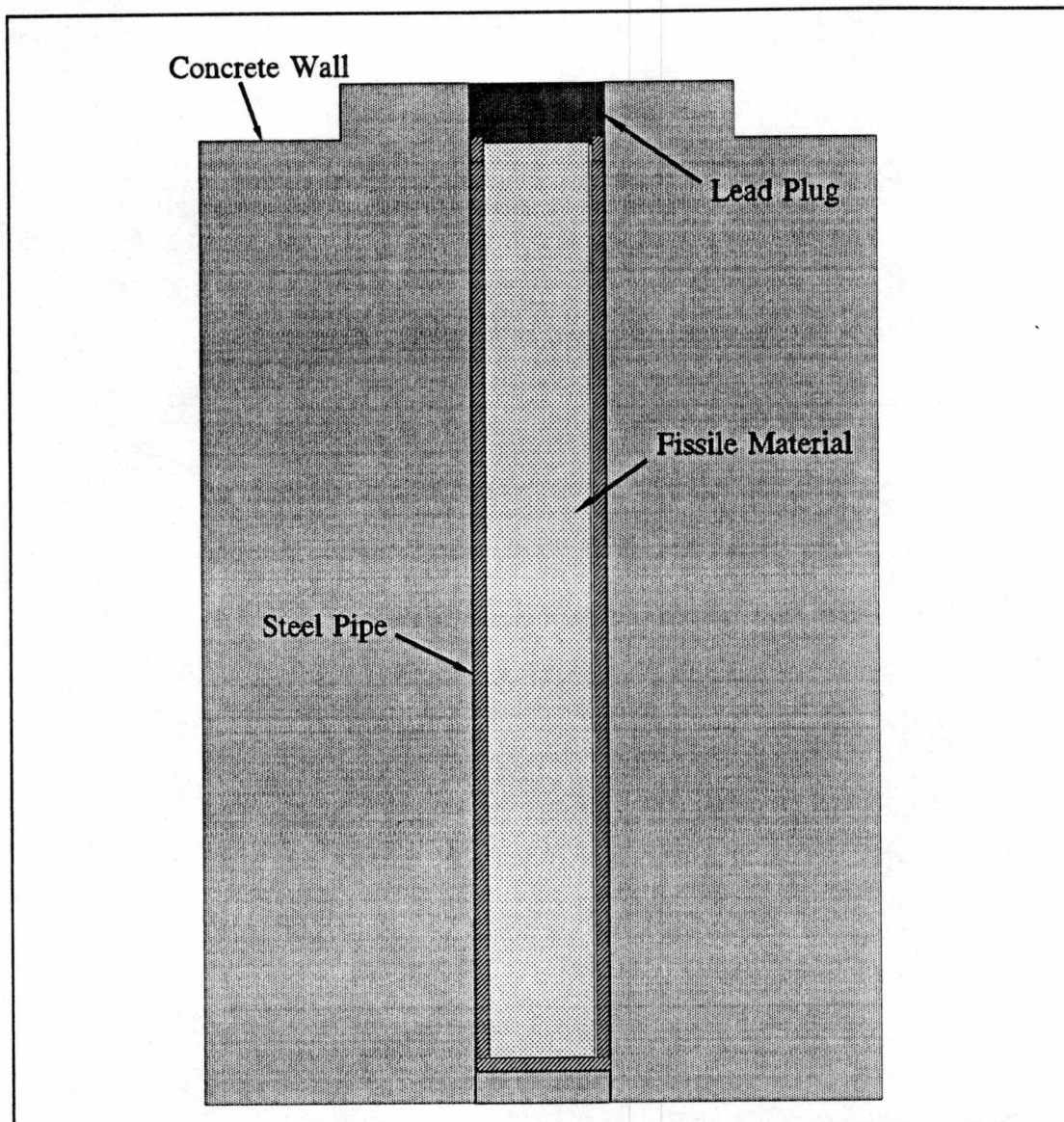


Figure 3.2.4.1: Schematic Diagram of a Typical Storage Well Containing a Single Column of Fissile Material.

The sensitivity of the multiplication factor to the settling of the fissile material is evaluated in a series of three KENO V.a calculations.⁴¹ In each calculation, the UO_2 ($\text{H/U} = 0$) loading is 2.13 kg U/ft, and the fuel radius is equivalent to the radius of a storage well (i.e. 5.08 cm.). The results of the three calculations are tabulated in Table

3.2.4.1. Initially, the UO_2 is distributed over the entire canister with a uranium density of 0.862 g/cm^3 . Each of the wells (i.e. 8) is completely filled with the above configuration of UO_2 . Subsequently, the density of the UO_2 is doubled in the second calculation by decreasing the fuel height from 7.54 cm. to 3.77 cm. within a single canister. In the third calculation, the UO_2 is assumed to be packed down at the bottom of a fuel canister. Given the theoretical density of uranium oxide (i.e. $\text{UO}_2 \rho = 10.96 \text{ g/cm}^3$), the smallest compaction height of UO_2 is 0.674 cm. A schematic diagram summarizing the redistribution of the UO_2 is presented in Figure 3.2.4.2. As shown in Table 3.2.4.1, the largest k_{eff} is obtained for the fuel uniformly distributed over the entire canister. Based on the results presented in Table 3.2.4.1, the settling of the fissile material is determined to be insignificant.

Table 3.2.4.1: Calculated k_{eff} 's for Redistribution of UO_2 in an 8x1x1 Array of In-line Storage Wells.

UO_2 height in a canister (cm.)	ρ^{U} (g/cm^3)	k_{eff}	AFG
7.54	0.862	0.5943 ± 0.0017	89.57
3.77	1.724	0.5868 ± 0.0017	87.0
0.674	9.637	0.5845 ± 0.0017	79.15

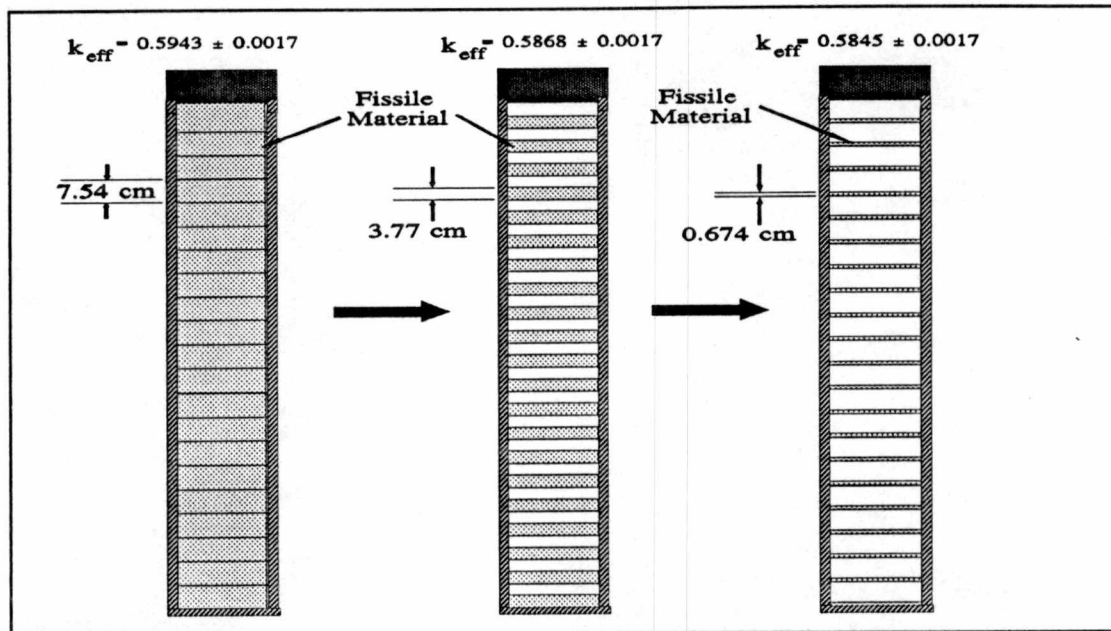


Figure 3.2.4.2: Schematic Diagram of The Settling of UO_2 in a Storage Well.

The chemical composition of the concrete encasing the inter-cell wells is presented in Section 3.1.1. Moreover, the chemical analysis of the concrete only provides an upper bound for the water content (i.e. atom densities: $\text{H}^{\text{max}} = 4.54991\text{e-}03$ and $\text{O} = 2.75330\text{e-}02$ atoms/barn*cm). Since a lower bound is not provided, the lower bound of water content is assumed to be zero (i.e. $\text{H} = 0.0$ and $\text{O} = 2.78196\text{e-}02$ atoms/barn*cm). In the previous calculations of this section, the hydrogen content is assumed to be zero (i.e. dry). In order to evaluate the sensitivity of the multiplication factor to the hydrogen content in the concrete, the upper bound for the water content is included in the KENO V.a model.⁴¹ As in the previous cases, the UO_2 ($\text{H/U} = 3$ and 2.13 kg U/ft) is uniformly distributed over the entire well thereby neglecting the fuel canisters. The fuel is placed in the 8 in-line wells and evaluated with KENO V.a. With

the maximum water content in the concrete, the calculated effective multiplication factor is 0.6744 ± 0.0019 . In contrast, the calculated k_{eff} for the dry concrete case is 0.7268 ± 0.0022 which is substantially higher than the maximum hydrogen case. Based on this result, the presence of hydrogen in the inter-cell wall concrete is shown to have a negative effect on k_{eff} . As a result, the inter-cell concrete wall is modeled without hydrogen.

In the model of the storage wells in Building 3019, each well contains a single chemical form of fissile material except Inter-cell 4&5 well M, which contains 3 metal hemispheres and UO_2 . As stated in Section 3.1.2, the metal is assumed to be 100 wt% ^{233}U at a density of 19.05 g/cm^3 . Moreover, the hemispheres have a center-to-center spacing of 43.18 cm. Well M is modeled with the hemispheres located at the center of the well, and the remaining volume of the well is filled with UO_2 ($\text{H/U} = 3$) at a loading of 2.13 kg U/ft. A schematic diagram of Inter-cell 4&5 well M is presented in Figure 3.2.4.3 It should be noted that this configuration is not possible because the UO_2 and metal are in contact (i.e. no canisters).

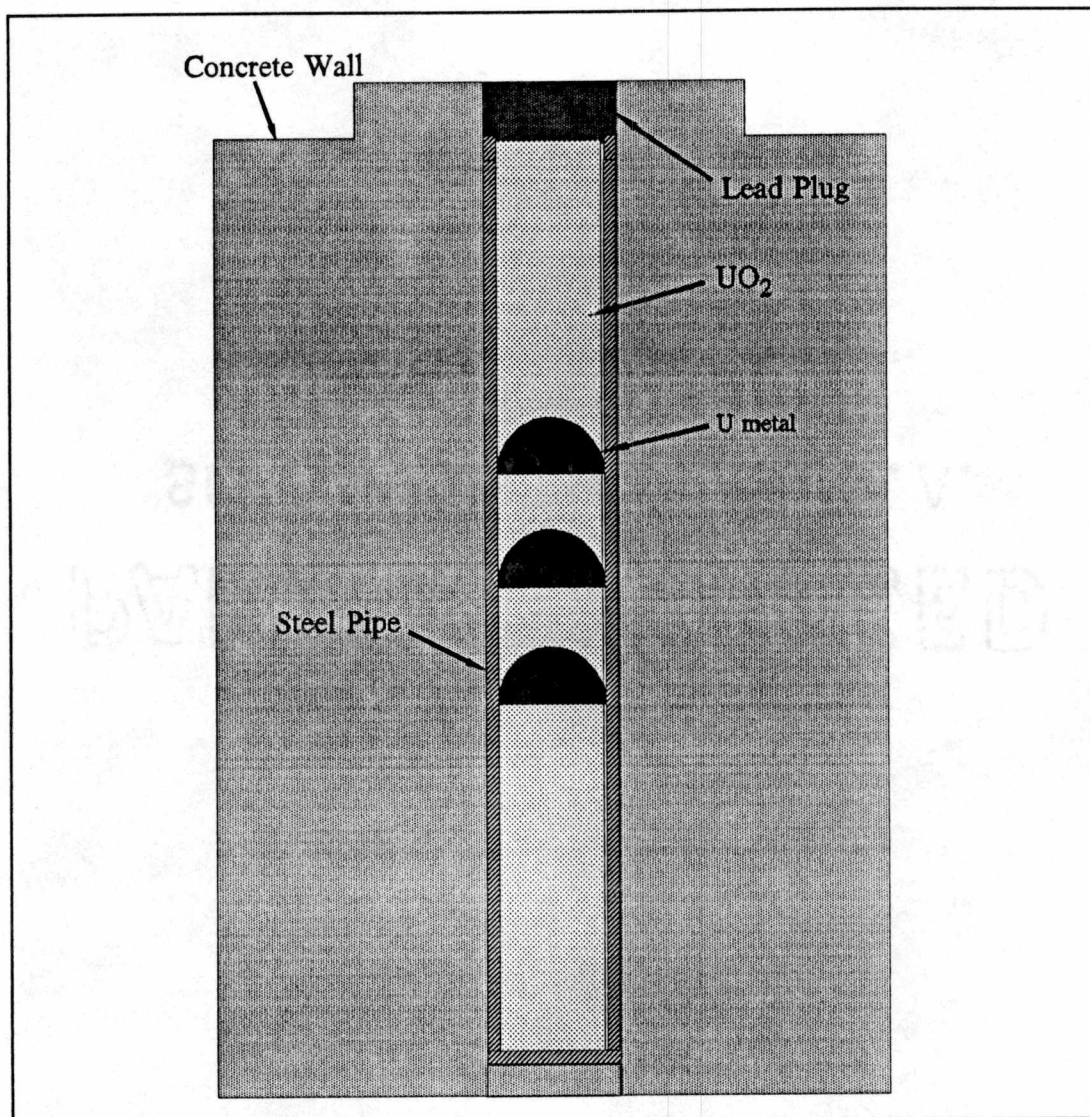


Figure 3.2.4.3: Schematic Diagram of Inter-cell Well 4&5 Well M.

3.2.5 Interaction Between Wells

All of the storage wells in Building 3019 are separated by concrete. The surface-to-surface spacing between the inter-cell wells is 19.116 cm. while the concrete separation distance between the Type A and B cell 4 wells is 21.021 and 19.685 cm., respectively. Before developing a model of the storage facility, the radial interaction between wells must be evaluated.

In order to evaluate the interaction between storage wells, a series of KENO V.a calculations is performed. In the first calculation, an evaluation of the neutron population leaving one well and reaching a neighboring well is evaluated. A single Type A well with an outside height of 392.75 cm. is modeled. The well is completely filled with UO_2 ($\text{H/U} = 3$) at a linear loading of 1.94 kg U/ft and encased in a 152.4 x 40.038 x 1097.28 cm. concrete wall (inter-cell type composition) as shown in Figure 3.2.5.1. The amount of concrete (i.e. 40.038 cm.) is selected to represent the actual thickness of concrete separating two wells (i.e. 19.116 cm.) in the y direction and minimize the neutron leakage in the x and z directions. The neighboring well represented by the dashed line in Figure 3.2.5.1 is not included in the KENO V.a model. The neutron population leaking from the single well encased in concrete is plotted as a function of energy group in Figure 3.2.5.2 along with the resonance corrected fission cross section for ^{233}U . As shown in Figure 3.2.5.2, the leakage spectrum ranges from group 1 to 80 (i.e. $2.0\text{e}+07$ to 186.0 eV) indicating a fast neutron spectrum reaching the neighboring well. Corresponding to this energy range, the ^{233}U fission cross section is much lower relative to the lower energies as shown in Figure 3.2.5.2, indicating a lower fission probability. However, these results only partly explain the interaction between wells.

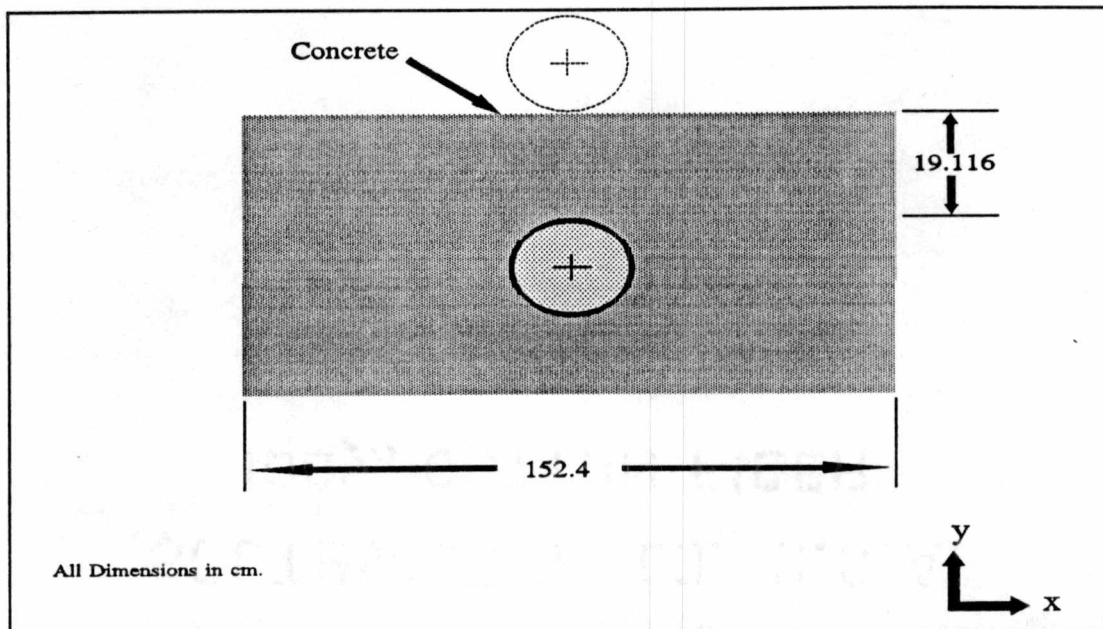


Figure 3.2.5.1: Top View of Single Type A Well Encased in Concrete.

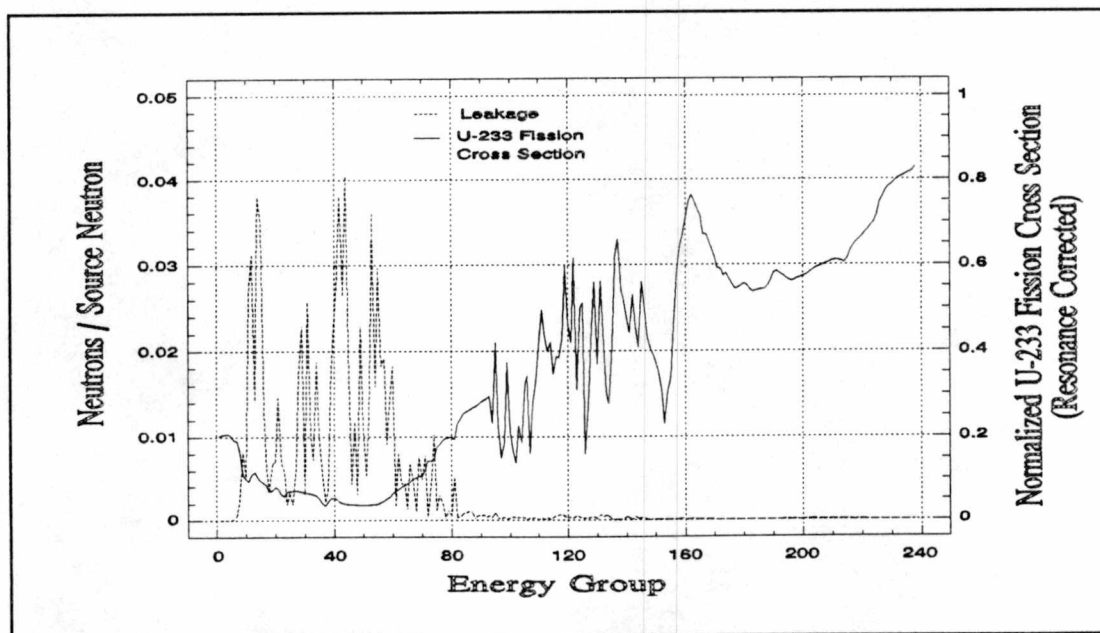


Figure 3.2.5.2: Neutron Leakage from a Single Well Encased in Concrete.

In the second series of calculations, the concrete dimensions are 30.48 x 30.48 x 1097.28 cm., thus representing the unit cell dimensions for an in-line well. The effective multiplication factor is calculated with KENO V.a. Subsequently, seven additional KENO V.a calculations are performed, and a unit cell containing a well is added to the array in each calculation until the last calculation included an array of 8 in-line wells. It should be noted that the eighth KENO V.a calculation included 8 in-line wells; however, the width of the concrete wall is only 30.48 cm., which is smaller than the actual width of the interior wall (i.e. 152.4 cm.). The results for the eight KENO V.a calculations are tabulated in Table 3.2.5.1. As shown in Table 3.2.5.1, the sequential increase in the number of wells leads to an increase in k_{eff} indicating some interaction between wells.

Table 3.2.5.1: Calculated k_{eff} 's for Eight Inter-cell Wells.

Number of Wells	k_{eff}	AFG
1	0.1279 ± 0.0007	37.78
2	0.1422 ± 0.0007	39.91
3	0.1498 ± 0.0007	41.83
4	0.1533 ± 0.0007	42.33
5	0.1555 ± 0.0007	42.98
6	0.1578 ± 0.0008	42.63
7	0.1579 ± 0.0008	43.30
8	0.1564 ± 0.0007	42.69

As stated previously, the entire width of an interior concrete wall is not modeled in the previous calculations. In the following calculation, the concrete dimensions encasing the eight in-line wells are increased to 152.4 x 243.84 x 1097.28 cm. The resulting k_{eff} is 0.6625 ± 0.0019 which is substantially higher than 0.1564. Moreover, seven of the wells are removed from the array leaving a single well centrally located within the concrete. The resulting k_{eff} is 0.3394 ± 0.0014 which is significantly smaller than 0.6625. These calculations were repeated for UF_4 , U metal and the CEUSP material, and results similar to the UO_2 cases were obtained. These results indicate that neutrons leaving a well are substantially moderated by the concrete prior to re-entering the same well or entering another well. As a result, the interaction between wells is determined to be significant. If the interaction between wells had not been significant, a much simpler model (e.g., one dimensional) could have been used to evaluate the storage wells.

All of the above calculations are performed for a series of in-line wells. A detailed evaluation of the Cell 4 array was not performed; however, the interaction between wells should be equally significant because the Cell 4 array has a substantially greater number of wells than the in-line wells. Furthermore, the wells are arranged in a triangular pitch, thus increasing the probability of a neutron reaching fissile material. Therefore, a complete model of the Cell 4 and in-line wells should include a detailed modeling of the all the wells. The interaction between the Cell 4 and in-line wells is discussed in the following section.

3.2.6 Interaction Between Storage Cells

The specific loadings of fissile material in the inter-cell and Cell 4 wells are presented in Section 3.2.1. Moreover, the evaluation of a single well is discussed in Sections 3.2.4 and 3.2.5. Using the above information, a KENO V.a model of the nominal configuration of the storage facility is determined in a series of calculational steps.⁴¹

In the first calculational step, a model of the Cell 4 array is developed. The 68 wells are encased in concrete (Cell 4 type composition) and placed within the Cell 4 room as shown in Figure 3.2.6.1. In addition, the model includes the north and south exterior walls along with the east and west interior walls between cells 3&4 and 4&5, respectively. It should be noted that the inter-cell wells in the east and west walls are not included in the model of the Cell 4 array. The calculated k_{eff} and AFG for the nominal configuration of the Cell 4 array is 0.8460 ± 0.0020 and 170.78, respectively.

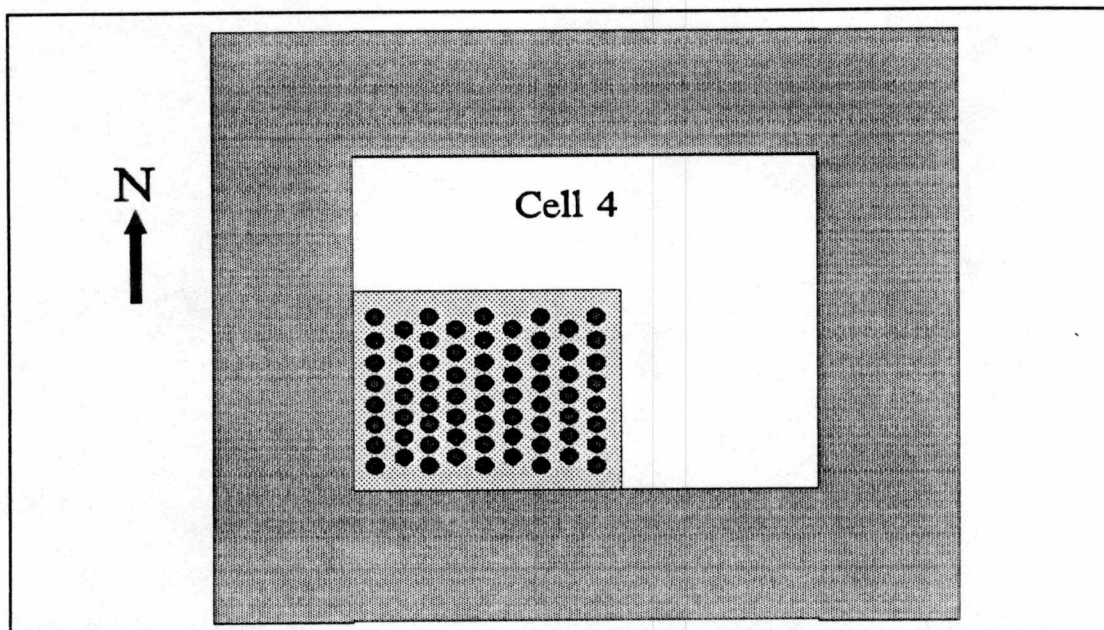


Figure 3.2.6.1: KENO V.a Model of 68 Cell 4 Wells.

The Cell 4 wells are removed from the model in the second calculation leaving the concrete block which encased the Cell 4 storage wells. Moreover, the 8 Inter-cell 4&5 wells are placed in the west interior wall of Cell 4 as indicated in Figure 3.2.6.2. The calculated k_{eff} for the Inter-cell 4&5 wells independent of the Cell 4 and Inter-cell 3&4 wells is 0.6840 ± 0.0020 (AFG = 98.0). When the Cell 4 wells are included with the 4&5 wells, the calculated effective multiplication factor is 0.8427 ± 0.0019 (AFG = 170.33) indicating the 4&5 wells do not contribute significantly to the overall reactivity of the Cell 4 wells.

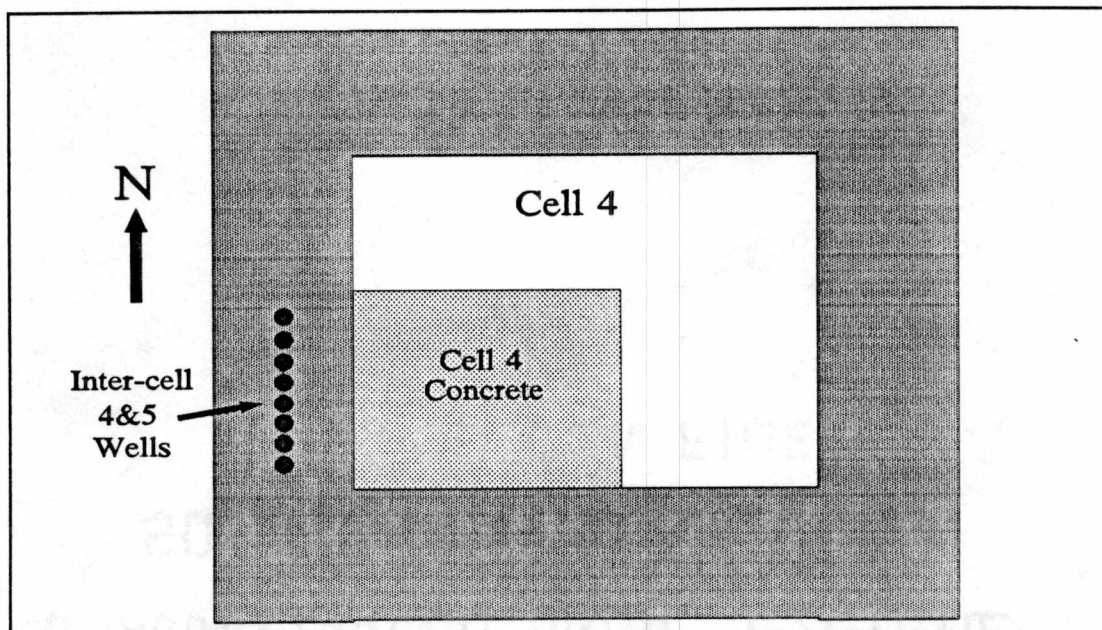


Figure 3.2.6.2: KENO V.a Model of Inter-cell 4&5 Wells Independent from Cell 4 Wells.

In the next calculation, the 9 Inter-cell 3&4 wells are placed in the east interior wall of Cell 4, and the other wells are replaced with concrete. The calculated k_{eff} for the 3&4 wells independent of the other storage wells is 0.7398 ± 0.0021 (AFG = 103.72). In the final calculation the nominal configuration of the storage facility is modeled. Included in the KENO V.a model are the three sets of storage wells (i.e. Inter-cell 4&5, 3&4 and Cell 4), and the calculated effective multiplication factor for the nominal configuration is 0.8438 ± 0.0018 (AFG = 169.74) which is significantly below the upper safety limit of 0.9557 for the 238 group ENDF/B-V cross section library (see Chapter 2). The KENO V.a input file titled *3019facl.input* for the nominal storage configuration of the storage wells is presented in Appendix D.

Based on these results, the nominal configuration of the fissile inventory in Building 3019 is safely subcritical. Furthermore, the overall reactivity of the storage facility is predominately determined by the 68 Cell 4 wells.

3.2.7 Area of Model Applicability

As stated in Section 3.2.6, the average neutron energy fission group (AFG) for the nominal configuration of the Building 3019 storage facility is 169.74. Upon review of the validation study for the 238 group ENDF/B-V library in Chapter 2, there are no critical experiments having the same average fission group. Despite the lack of experimental data, there are seven critical ^{233}U experiments with AFG's ranging from 180 to 191. However, these systems contain aqueous uranyl nitrate and uranyl fluoride which is significantly different than the solid inventory in Building 3019. Furthermore, the H/U for these seven critical systems ranges from 45.9 to 74.1 which is substantially higher than the nominal H/U of 3 for Building 3019. These results further illustrate the need for critical experiments consisting of arrays of ^{233}U interstitially moderated with concrete.

Only two of the seven critical experiments consist of an array of cylindrical units (i.e. Experiments 8.10 and 8.11). Of these two experiments, only Experiment 8.11 is reflected. Despite the dissimilarities between Experiment 8.11 and the RPP facility, it is useful to compare the neutronics of both systems. Figure 3.2.7.1 contains a plot of the percentage of fissions for both systems. As shown in Figure 3.2.7.1, the majority

of fissions occur in the thermal range. A detailed view of the energy groups between 150 and 238 is presented in Figure 3.2.7.2. There are three distinct fission peaks for both systems corresponding to energy groups 216, 223 and 225 as indicated in Figure 3.2.7.2. The presence of three distinct peaks at approximately the same energy level for both systems is encouraging despite the system differences. This result is interesting because the H/U for the RPP facility is 3 which is substantially smaller than Experiment 8.11 (i.e. $H/U = 73$). Despite the difference in H/U, the increased number of thermal fissions in the RPP facility is apparently due to the moderation of neutrons by the concrete encasing the storage wells.

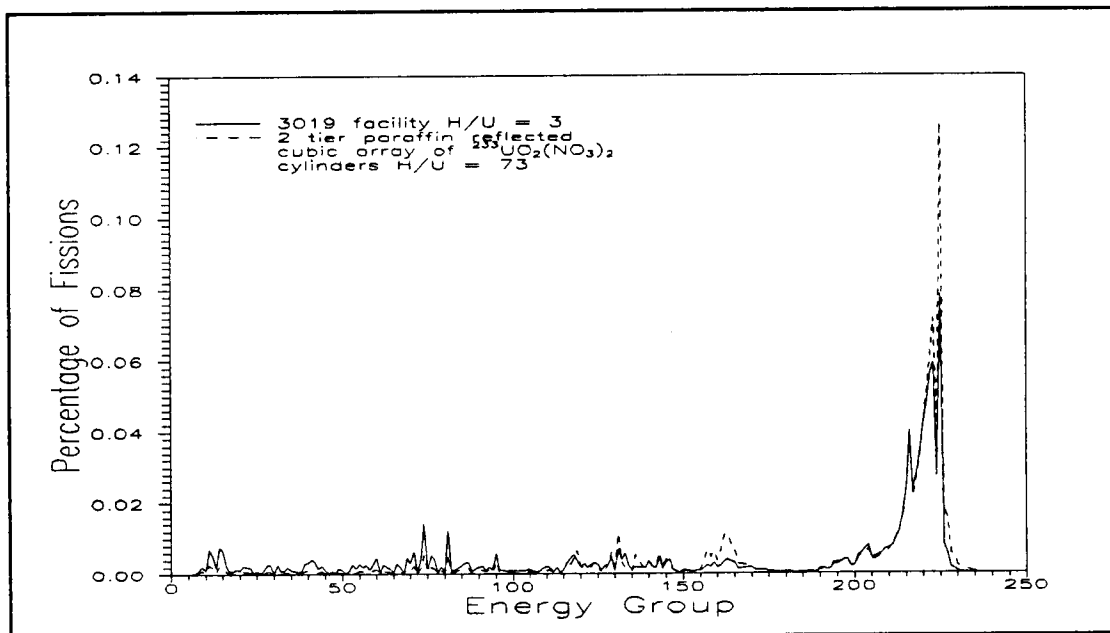


Figure: 3.2.7.1: Comparison of Percentage of Fissions for Building 3019 Fissile Material Inventory and a 2 Tier Cubic Array of Uranyl Nitrate Solution Reflected with Paraffin.

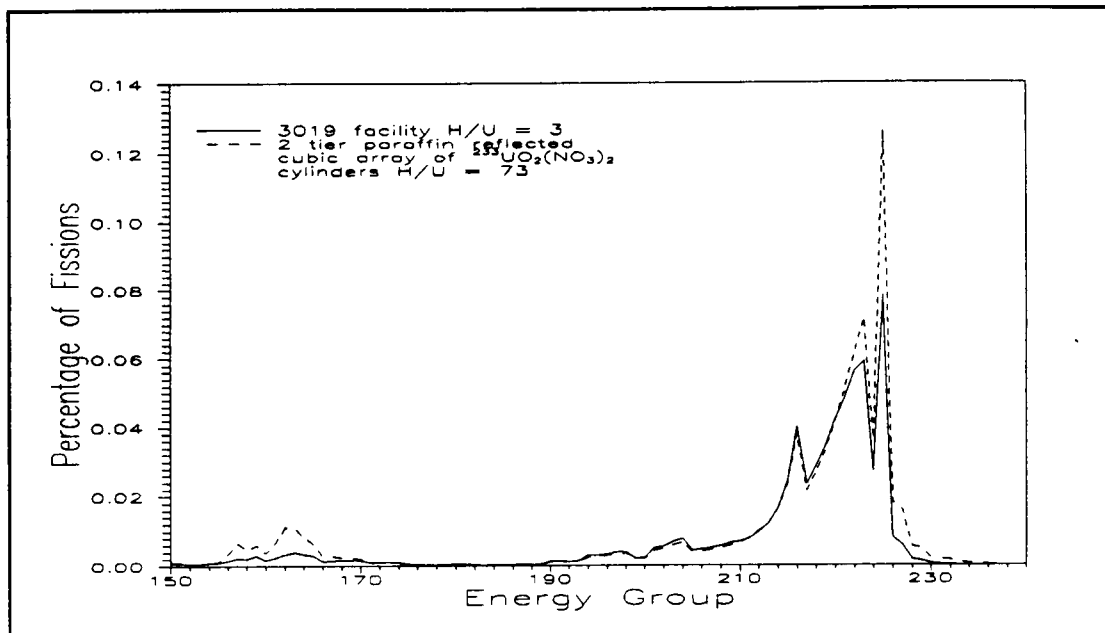


Figure 3.2.7.2: Comparison of Percentage of Fissions for Building 3019 Fissile Material Inventory and a 2 Tier Cubic Array of Uranyl Nitrate Solution Reflected with Paraffin for Energies above Group 150.

Even though the storage configuration of Building 3019 does not fall into the range of applicability defined in Chapter 2, there is evidence that the KENO V.a model is capable of calculating the effective multiplication factor of the stored fissile inventory in Building 3019.

3.3 Facility Limits

Currently, there are nineteen empty wells (excluding Inter-cell 2&3 wells) available for storage of fissile material in the Cell 4 and inter-cell wells. Using the KENO V.a model of the storage facility and the derived subcritical limit for the 238 group ENDF/B-V cross section library (see Chapter Two), maximum loading limits for the available wells can be determined. The loading limits which are presented in the following sections are derived for the storage of ^{233}U .

3.3.1 Double Batching of Available Wells

Prior to evaluating the loading limits for the facility, double batching of the fuel within the available wells must be addressed. As noted in Section 3.2.6, the overall effective multiplication factor of the storage facility is predominately determined by the Cell 4 wells. Moreover, sixteen of the available wells are located in the Cell 4 array as shown in the identification pattern for the wells in Figure 3.3.1.1. Therefore, the double batching of fuel in the available wells is evaluated with respect to the Cell 4 wells.

Double batching of fuel is evaluated with respect to a 100 cm. length centered about the axial midplane of a single well having twice the fissile and H/U loading.¹⁵ Since most of the fuel canisters are less than 20 cm. in height, this scenario corresponds to at least 5 double batched canisters within a single well. The probability of such an event is believed to be exceedingly low. Nevertheless, the determination of this probability is beyond the scope of this work.

Several calculations were performed to determine which well should be double batched. The KENO V.a model of the Cell 4 wells, which is presented in Section 3.2.6, is used to examine the double loading scenario. In addition to the nominal configuration of the wells, the 16 available wells are completely filled with ²³³U at a linear loading of 2.0 kg/ft (i.e. no wells are double batched). The calculated effective multiplication factor for the Cell 4 wells is 0.9377 ± 0.0021 (AFG = 166.70). Using this model, each available well is double batched and the k_{eff} of the wells is calculated. The calculated

k_{eff} 's corresponding to the double loading of each available well are tabulated in Table 3.3.1.1. As shown in Table 3.3.1.1, the differences in the calculated k_{eff} 's are within $\sim 1.3\%$. The double batched well with the highest calculated k_{eff} is Cell 4 well 3-A. Based on these results, Cell 4 well 3-A is double batched in all subsequent loading limit studies.

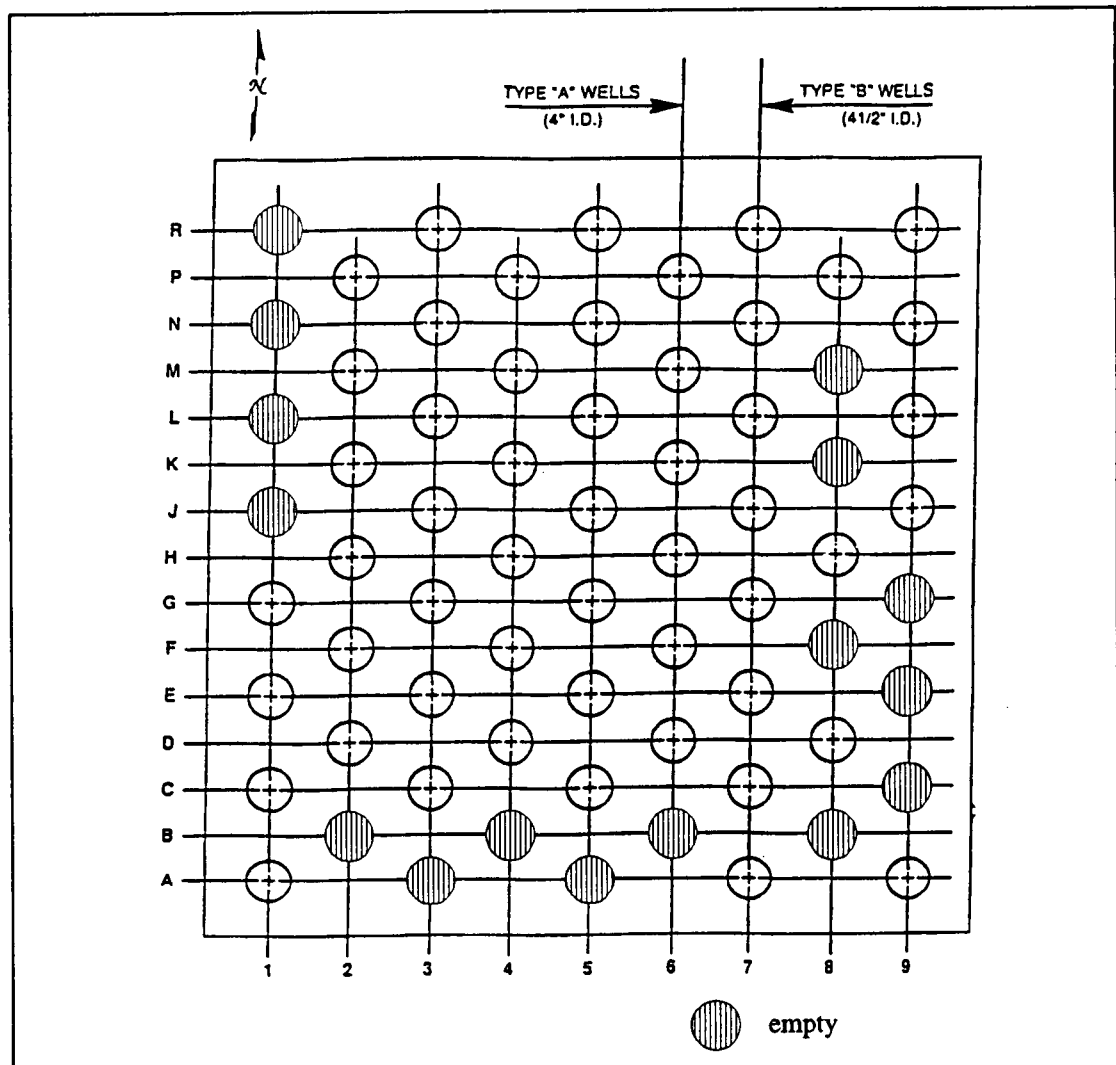


Figure 3.3.1.1: Empty Wells Located in Cell 4 Array.

Table 3.3.1.1: Calculated k_{eff} 's for Double Batching of Available Cell 4 Wells.

Double Batched Well Location	k_{eff}	AFG
1-J	0.9406 ± 0.0023	166.71
1-L	0.9371 ± 0.0024	166.46
1-N	0.9411 ± 0.0022	166.35
1-R	0.9370 ± 0.0027	167.17
2-B	0.9331 ± 0.0023	166.28
3-A	0.9453 ± 0.0024	166.12
4-B	0.9380 ± 0.0025	166.58
5-A	0.9408 ± 0.0024	166.27
6-B	0.9343 ± 0.0024	166.40
8-B	0.9398 ± 0.0027	166.82
8-F	0.9387 ± 0.0025	166.46
8-K	0.9344 ± 0.0024	166.32
8-M	0.9403 ± 0.0019	166.23
9-C	0.9416 ± 0.0019	166.35
9-E	0.9401 ± 0.0019	165.98
9-G	0.9437 ± 0.0018	166.50

3.3.2 Calculation of Loading Limits

In the future it may be desirable to fill the available wells in Building 3019 with ^{233}U . Prior to storing additional fissile material in the wells, subcritical loading limits must be derived to assure subcriticality. In Chapter Two, a USL of 0.9557 was derived based on the validation of KENO V.a and the 238 group ENDF/B-V cross section library for ^{233}U systems. If a calculated k_{eff} plus two standard deviations is less than the USL, the system is considered safely subcritical. Using the computational model of the RPP facility and the USL, numerous calculations were performed to determine safe loading scenarios for the available storage wells in Building 3019. All subsequent loading limits apply to the empty wells which are tabulated in Table 3.3.2.1.

Table 3.3.2.1: Building 3019 Storage Wells Designated for Loading.

Cell 4 Wells	In-line Wells 4&5
1J-1R, 2B, 3A, 4B, 5A, 6B, 8B, 8F, 8K, 8M, 9C-9G	P, S, T

In the first series of calculations, the subcritical loading limits for ^{233}U material having an $\text{H/U} = 0$ are determined. The wells are initially loaded with 3.8 kg U/ft and the effective multiplication factor of the facility is calculated. Subsequently, the linear loadings in the available wells are reduced, and the calculations are repeated. The results of these KENO V.a calculations are presented in Table 3.3.2.2 for the $\text{H/U} = 0$ case. Similarly, the results for H/U 's of 3, 5, 10 and 20 are presented in Tables 3.3.2.3,

3.3.2.4, 3.3.2.5 and 3.3.2.6, respectively. Based on the above KENO V.a calculations and the USL, the ^{233}U subcritical loading limits for the storage facility are presented in Table 3.3.2.7.

Table 3.3.2.2: Calculated k_{eff} 's for the $\text{H}/^{233}\text{U} = 0$ Loading Configurations in the Available Storage Wells in Building 3019.

kg U/ft	k_{eff}	AFG
3.8	0.9712 ± 0.0029	158.28
3.5	0.9652 ± 0.0027	159.82
3.2	0.9659 ± 0.0027	160.99
3.1	0.9607 ± 0.0028	161.51
3.0	0.9548 ± 0.0027	161.52
2.5	0.9547 ± 0.0026	164.65
2.45	0.9393 ± 0.0026	164.63
2.4	0.9428 ± 0.0026	165.37
2.35	0.9447 ± 0.0030	164.58
2.3	0.9485 ± 0.0028	164.84
2.25	0.9417 ± 0.0027	165.39
2.0	0.9319 ± 0.0028	166.29
1.75	0.9288 ± 0.0028	167.39

Table 3.3.2.3: Calculated k_{eff} 's for the $H/^{233}\text{U} = 3$ Loading Configurations in the Available Storage Wells in Building 3019.

kg U/ft	k_{eff}	AFG
2.5	0.9740 ± 0.0026	162.02
2.25	0.9646 ± 0.0029	163.13
2.15	0.9619 ± 0.0027	163.63
2.1	0.9616 ± 0.0028	164.46
2.0	0.9514 ± 0.0025	164.61
1.9	0.9506 ± 0.0027	165.10
1.8	0.9473 ± 0.0028	165.95
1.7	0.9460 ± 0.0027	166.22
1.65	0.9444 ± 0.0028	166.71
1.6	0.9414 ± 0.0029	167.40

Table 3.3.2.4: Calculated k_{eff} 's for the $\text{H}/^{233}\text{U} = 5$ Loading Configurations in the Available Storage Wells in Building 3019.

kg U/ft	k_{eff}	AFG
2.0	0.9647 ± 0.0030	164.21
1.9	0.9546 ± 0.0028	164.92
1.85	0.9623 ± 0.0031	165.01
1.8	0.9552 ± 0.0028	165.62
1.75	0.9492 ± 0.0033	166.04
1.65	0.9463 ± 0.0039	166.42
1.5	0.9484 ± 0.0027	167.38
1.4	0.9461 ± 0.0030	168.01
1.39	0.9371 ± 0.0028	167.91
1.38	0.9390 ± 0.0026	168.22
1.35	0.9372 ± 0.0029	168.00
1.3	0.9362 ± 0.0027	168.22

Table 3.3.2.5: Calculated k_{eff} 's for the $\text{H}/^{233}\text{U} = 10$ Loading Configurations in the Available Storage Wells in Building 3019.

kg U/ft	k_{eff}	AFG
1.75	0.9752 ± 0.0035	164.91
1.6	0.9705 ± 0.0030	166.17
1.55	0.9650 ± 0.0028	166.96
1.5	0.9620 ± 0.0030	166.61
1.45	0.9618 ± 0.0027	166.95
1.3	0.9466 ± 0.0042	167.94
1.27	0.9450 ± 0.0037	168.18
1.25	0.9435 ± 0.0040	168.54
1.22	0.9454 ± 0.0028	168.74
1.2	0.9428 ± 0.0029	168.26
1.15	0.9422 ± 0.0031	169.0
1.0	0.9257 ± 0.0032	170.03
0.25	0.8755 ± 0.0028	172.83

Table 3.3.2.6: Calculated k_{eff} 's for the $H/^{233}U = 20$ Loading Configurations in the Available Storage Wells in Building 3019.

kg U/ft	k_{eff}	AFG
1.75	1.0487 ± 0.0029	164.88
1.5	1.0422 ± 0.0036	165.34
1.25	0.9841 ± 0.0029	169.25
1.15	0.9697 ± 0.0030	169.07
1.10	0.9697 ± 0.0030	170.49
1.05	0.9548 ± 0.0027	169.92
1.0	0.9499 ± 0.0021	170.55
0.96	0.9517 ± 0.0045	170.55
0.95	0.9423 ± 0.0039	171.66
0.94	0.9399 ± 0.0042	170.61
0.75	0.9279 ± 0.0034	171.30
0.5	0.9013 ± 0.0031	172.70
0.25	0.8847 ± 0.0021	172.57

Table 3.3.2.7: ^{233}U Subcritical Loading Limits for the Available Storage in Building 3019.

H/U	kg U/ft
0	2.45
3	1.9
5	1.75
10	1.30
20	0.95

3.3.3 Intermixing of Fissile Material

The subcritical loading limits presented in Section 3.3.2 were derived assuming an entire well is filled with ^{233}U with a single H/U value; however, facility operators may intermix fissile material with differing H/U within a single column. Therefore, the intermixing of fissile material within a well must be evaluated.^{15,41} Since there are an infinite number of possible loading scenarios, the evaluation must be performed in a general fashion.

A schematic diagram of two adjacent Type A wells is presented in Figure 3.3.3.1. Both wells are located on a 30.48 cm. pitch and are encased in a 152.4 x 152.4 x 1158.24 cm. concrete block. Each well is filled with canisters having an inside radius equivalent to the well radius (i.e. 5.08 cm), and the overall height of each canister is 7.54 cm.

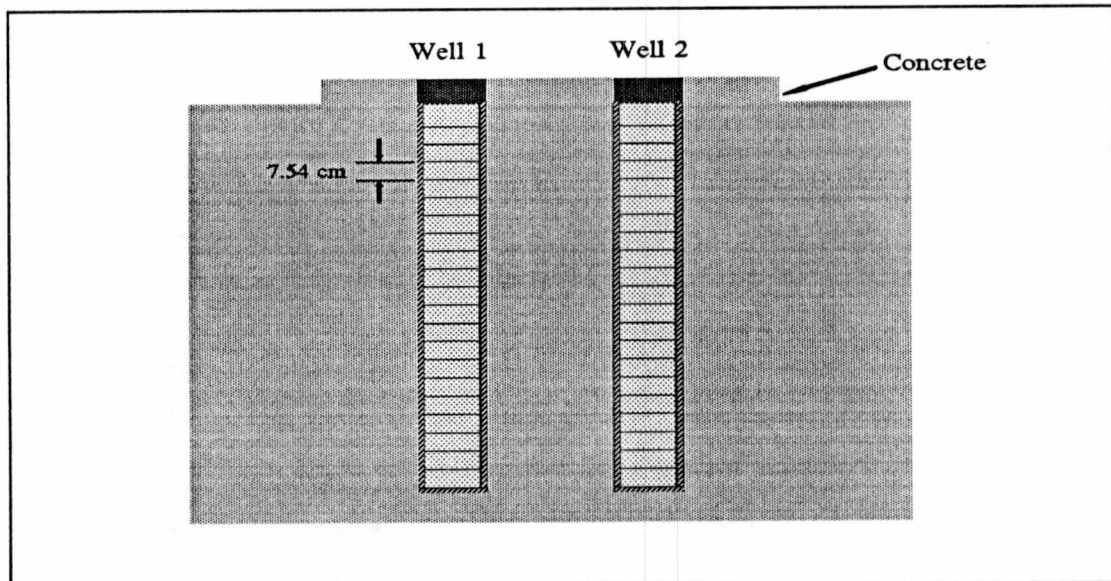


Figure 3.3.3.1: Schematic Diagram of Two Adjacent Type A Wells for Intermixing of Fissile Material Study.

A KENO V.a model of the two wells was developed and five different loading scenarios were evaluated. In the first case, both wells are filled with 2.25 kg $^{233}\text{U}/\text{ft}$ ($\text{H}/\text{U} = 0$). Similarly, both wells are filled with 0.95 kg $^{233}\text{U}/\text{ft}$ ($\text{H}/\text{U} = 20$) in the second case. For the third scenario, well 1 is completely filled with 2.25 kg $^{233}\text{U}/\text{ft}$ ($\text{H}/\text{U} = 0$), and well 2 is filled with 0.95 kg $^{233}\text{U}/\text{ft}$ ($\text{H}/\text{U} = 20$). In the fourth case, each well is filled with alternating canisters of ^{233}U with H/U of 0 and 20. The linear loadings of each canister is the same as the first three cases (i.e. $\text{H}/\text{U} = 0$: 2.25 kg U/ft and $\text{H}/\text{U} = 20$: 0.95 kg U/ft). For the fifth scenario, the middle third of well 1 is filled with 0.95 kg $^{233}\text{U}/\text{ft}$ ($\text{H}/\text{U} = 20$), and the top and bottom thirds are filled with 2.25 kg $^{233}\text{U}/\text{ft}$ ($\text{H}/\text{U} = 0$). Likewise, the middle third of well 2 is filled with 2.25 kg $^{233}\text{U}/\text{ft}$ ($\text{H}/\text{U} = 0$), and the remaining two thirds are filled with 0.95 kg $^{233}\text{U}/\text{ft}$ ($\text{H}/\text{U} = 20$). The results of the five calculations are presented in Table 3.3.3.1. As shown in Table 3.3.3.1, the variation in k_{eff} for the last three cases is insignificant. In case three, there is no intermixing of material, and when the fissile material is intermixed in cases 4 and 5, the increase in k_{eff} is within $\sim 1.06\%$.

Table 3.3.3.1: Calculated k_{eff} 's for Axial Variation of Fissile Material within a Well.

Case	Avg. H/U per Well		k_{eff}	AFG
	Well 1	Well 2		
1	0	0	0.5788 ± 0.0025	136.49
2	20	20	0.6314 ± 0.0028	165.70
3	0	20	0.6089 ± 0.0025	153.25
4	10	10	0.6094 ± 0.0028	150.55
5	6.67	13.33	0.6154 ± 0.0028	153.86

Since the loading limits presented in Section 3.3.2 are derived for a single H/U, the only way to apply these limits uniformly would be to restrict all canisters within a single well to the same H/U. However, if it is necessary to intermix differing H/U canisters within a well, the loading limit for the well should be restricted to the lowest H/U canister.¹⁵ For instance, if the H/U values within a well are 0 and 20, the maximum loading limit for the H/U = 0 canister is 0.95 kg ²³³U/ft (i.e. the loading limit for H/U = 20). Caution should be exercised in the above example because a canister which is acceptably stored in one well may not be acceptable in a different well.⁴¹

When it is desirable to move canisters from a well to a different storage well within the facility, a canister with a low H/U may only be stored with higher H/U canisters if the low H/U canister has a loading equal to or less than the high H/U canisters.⁴¹ Likewise, a high H/U canister can only be relocated to a well with low H/U canisters if the lower H/U canisters are substantially below their respective loading limits. In the latter case, it would be preferable for the lower H/U canisters to have loadings equal to or less than the high H/U canister.

Chapter Four

Investigation of Building 3019 Limits

4.1 Introduction

As stated in Chapter One, the storage wells in Building 3019 have been analyzed previously by Primm¹⁵ using the 16 energy group Hansen-Roach cross section library. The objective of this work is to independently re-evaluate the 3019 storage facility with KENO V.a using the ENDF/B-V cross section library. The results of the independent evaluation of the storage facility are presented in Chapter Three of this work. Upon completion of the evaluation, it is prudent to compare both sets of results and investigate the sensitivity of the results (i.e. ²³³U subcritical loading limits) obtained in this work to the derived USL in Chapter Two.

4.2 Comparison to Previous Work

Prior to comparing the derived ²³³U loading limits of this research with previous work, the interpretation of the loading limits must be understood. It should be noted that the loading limits in this work only apply to the empty storage wells tabulated in Table 4.2.1 (repeated from Chapter 3). Also, presented in Table 4.2.1 are the wells which correspond to the subcritical storage limits derived by Primm.

Table 4.2.1: Building 3019 Storage Wells Designated for Loading.

Cell 4 Wells	Inter-cell Wells
This Work	
1J-1R, 2B, 3A, 4B, 5A, 6B, 8B, 8F, 8K, 8M, 9C-9G	4&5 wells: P, S, T
Ref. 15	
1A-1R, 2B, 3A, 4B, 5A, 6B, 7A-7J, 8B, 8F, 8K-8P, 9A-9R	All inter-cell wells

The ^{233}U subcritical loading limits derived by each author are presented in Table 4.2.2. As shown in Table 4.2.2, the ENDF/B-V limits are lower than the Hansen-Roach limits for each H/U value. In order to understand the differences between the two models of the 3019 wells, KENO V.a calculations were performed using Primm's loading limits with Hansen-Roach cross sections. The results of these calculations are presented in Table 4.2.3. The agreement between the two models is quite good as shown in Table 4.2.3 indicating similar representation of the storage wells. Using the same loading limits with the 238 group ENDF/B-V cross section library in the facility model presented in Chapter Three, the KENO V.a runs were repeated. The results of these calculations are also presented in Table 4.2.3. As shown in Table 4.2.3, the 238 group results are approximately 5% higher than the Hansen-Roach results. Since the 238 group results are consistently higher, one would expect lower loading limits using this library.

Table 4.2.2: Comparison of ^{233}U Subcritical Loading Limits for Building 3019.

H/U	kg $^{233}\text{U}/\text{ft}$	
	238 Group ENDF/B-V (This Work)	16 Group Hansen-Roach (Ref. 15)
0	2.45	2.5
3	1.9	2.0
5	1.75	1.8
10	1.30	1.45
20	0.95	1.0

Table 4.2.3: Effective Multiplication Factor Calculated by KENO V.a for Building 3019 Wells.

kg U/ft	H/U	Chapter 3 Model 16 Group Hansen-Roach 238 Group ENDF/B-V	Ref. 15 Model 16 Group Hansen-Roach
2.5	0	0.9110 ± 0.0025 0.9547 ± 0.0026	0.90466 ± 0.00323
2.0	3	0.9045 ± 0.0026 0.9514 ± 0.0025	0.90052 ± 0.00338
1.8	5	0.9042 ± 0.0027 0.9552 ± 0.0028	0.91393 ± 0.00308
1.45	10	0.9106 ± 0.0026 0.9618 ± 0.0027	0.91326 ± 0.00368
1.0	20	0.9081 ± 0.0029 0.9470 ± 0.0030	0.90776 ± 0.00310

4.3 Sensitivity of Loading Limits

In addition to using a different cross section library, the differences can be attributed to the imposed upper safety limit (USL). In Chapter Two, a USL of 0.9557 was derived based on the validation of KENO V.a and the 238 group ENDF/B-V library for ^{233}U systems. Based on the derivation, a ^{233}U system with a calculated k_{eff} plus two standard deviations being less than the USL is considered safely subcritical. The calculated USL corresponds to a linear function of the average fission group. Moreover, the USL represents a lower tolerance band with 95% confidence on 99.9% of the data and is given by the following equation:

$$USL(AFG) = 7.4011e-05 * (AFG) + 0.9557. \quad (4.3.1)$$

As a result, there are three different ways to use this USL: 0.95, 0.9557, or Eq. (4.3.1). In order to understand the impact of the USL on the loading limits, subcritical loading limits for the three different USL's are presented in Table 4.3.1. Depending on the choice of the USL, the loading limits can vary significantly as shown in Table 4.3.1. The limits based on the 0.95 value are probably too conservative. The limits based on the 0.9557 value are more realistic and are in good agreement with the previous work.

Table 4.3.1: ^{233}U Subcritical Loading Limit Studies for the 3019 Storage Wells.

H/U	kg U / ft USL = 0.95	kg U / ft USL = 0.9557	kg U / ft USL = Eq. (4.3.1)
0	2.25	2.45	3.1
3	1.65	1.9	2.15
5	1.39	1.75	1.8
10	1.20	1.30	1.5
20	0.95	0.95	1.05

Chapter Five

Evaluation of the MSR Drain Tanks

5.1 Introduction

The overall objective of the Molten Salt Reactor (MSR) was to aid in the development of breeder reactors. Moreover, the Molten Salt Reactor Experiment (MSRE) was designed to investigate the feasibility of using a molten salt reactor for central power production applications. Preliminary studies for the MSR began in 1957 and were completed in 1960. Construction began in 1962, and the reactor was brought critical on June 1, 1965.²³

The MSR core was a 139.7 cm. (55 in.) diameter cylinder 162.56 cm. (64 in.) in height and consisted of a graphite-moderator core structure. This assembly consisted of vertical graphite bars arranged in a close-packed array. The graphite bars were 5.08 cm. (2 in.) square by 170.18 cm. (67 in.) in length. The MSR core was encased in a 152.4 cm. (5 ft.) diameter tank with a height of 243.84 cm. (8 ft.) The MSR fuel contained ~36 kg of U (~ 84 wt% ^{233}U) and ~0.690 kg of Pu (~91 wt% ^{239}Pu) in a LiF, BeF₂, ZrF₄, UF₄ and PuF₃ salt mixture. During operation, the fuel salt was circulated through the core at 1200 gal/min. The inlet temperature and pressure of the fuel salt was 1175°F and ~20 psi, respectively. After circulating through the core, the salt exited the vessel at 1225°F and ~7 psi. The reactor was operated successfully until it was shut down on December 12, 1969 due to a decrease in funding.^{23,16}

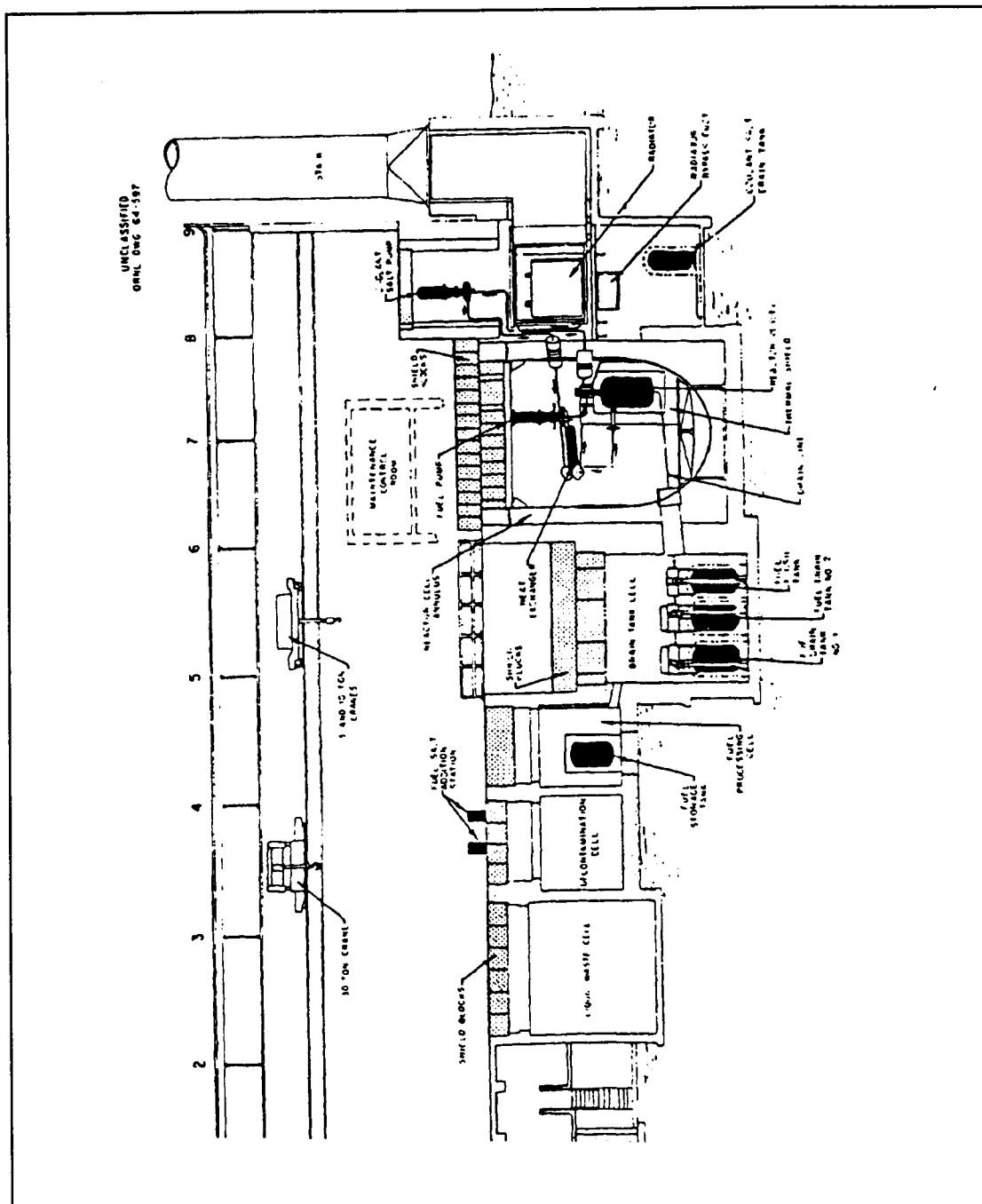
Following shutdown, the molten salt was drained from the reactor into two Fuel Drain Tanks (FDT). After draining the fuel, the reactor was flushed with a clean salt which was drained to a Fuel Flush Tank (FFT). When the molten salt was drained to the tanks, the salt was cooled and solidified very rapidly, thus preserving the homogeneity of the salt mixture.²⁴ Due to removal cost constraints, the fuel has been stored in the tanks since the reactor was shut down.

Consequently, the long term storage of the fuel salt has raised some criticality safety concerns regarding the storage configuration. In order to provide assurance that the salt inventory is subcritical, a criticality safety evaluation of the drain tanks must be performed. The objective of this study is to evaluate the storage configuration of the salt under nominal and abnormal conditions using KENO V.a with ENDF/B-V cross sections.

5.2 Facility Description

5.2.1 Nominal Storage Configuration

As noted in the Introduction, the fuel salt and flush salt is currently stored in three tanks (i.e. 2 FDT's and 1 FFT) within the MSR Facility (Building 7503) at ORNL. Moreover, these tanks are contained in the Fuel Drain Tank Cell (FDTC), which is located in the lower level of Building 7503 as shown in the elevation view of the facility (see Figure 5.2.1.1). To further illustrate the location of the FDTC, a plan view of the MSR facility is provided in Figure 5.2.1.2. The FDTC is a rectangular room with interior dimensions of 646.01 x 535.94 x 624.84 cm (21.19 x 17.58 x 20.5 ft.). The



interior walls of the FDTC are lined with a ~10.16 cm. (4in.) thick stainless steel (SS347) liner. SS347 is not available in the SCALE library, thus the chemical composition must be used to define the material. The chemical composition of SS347²⁵ is presented in Table 5.2.1.1. Moreover, the interior walls of the FDTC are 91.44 cm. (3 ft.) thick and are composed of ordinary concrete. A plan view of the FDTC is presented in Figure 5.2.1.3.

Table 5.2.1.1: Chemical Composition of SS347 (Ref. 25).

Element	Weight Percent
C	0.08
Mn	2.0
P	0.045
S	0.030
Si	1.0
Cr	19.0
Ni	13.0
Fe	64.845

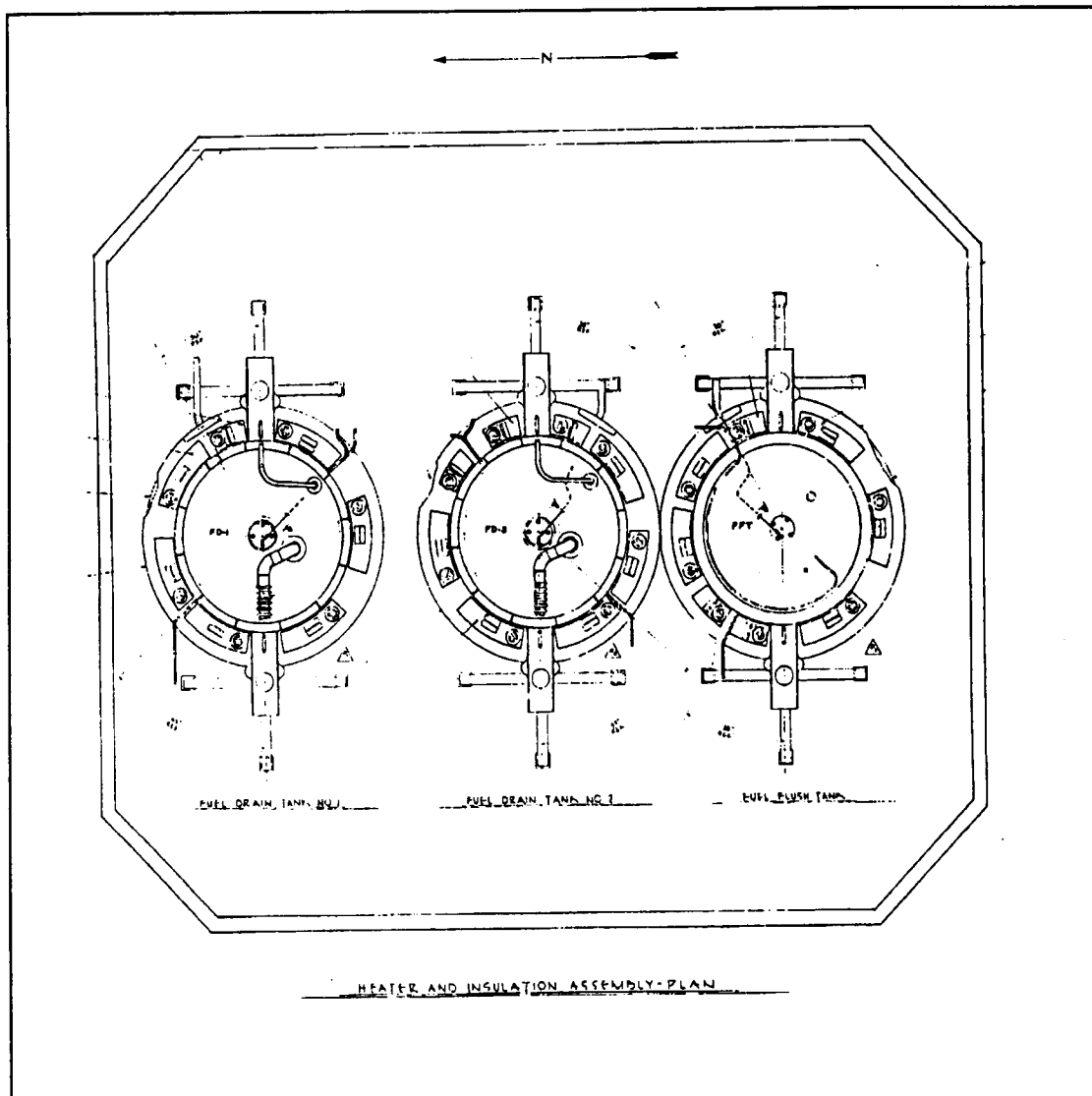


Figure 5.2.1.3: Plan View of Fuel Drain Tank Cell (from MSR Drawing: E-MM-B-51684).

The two FDT's, which contain the salt mixture, are nominally 127 cm. (50 in.) in diameter and 218.44 cm. (86 in.) in height. Each FDT has a top and bottom domed lid as shown in Figure 5.2.1.4. Since the tanks were designed to store high temperature molten salt, the FDT's are equipped with a cooling system. The cooling system inside each tank consists of 32 thimbles which extend from the top of the tank to ~38 cm. above the bottom of the tank. On the top of each tank is a steam dome and 32 bayonet tubes which are inserted in the thimbles. An elevation view of a FDT showing the cooling system is presented in Figure 5.2.1.5. Moreover, a cross sectional view illustrating the thimble configuration is presented in Figure 5.2.1.6. The inner bank of 12 thimbles is arranged on a 30.48 cm. radius while the outer bank of 20 thimbles is arranged on a 46.99 cm. radius. During operation the thimbles serve as a heat exchanger to cool the salt; however, all water connections to the cooling system are sealed and capped. Consequently the cooling system does not (and cannot) contain water.

The FFT has the same nominal dimensions as both FDT's; however, the height of the FFT is only 213.36 cm. (84 in.). In addition to the height difference, the FFT does not have a cooling system. As a result, the internal volume of the FFT is larger than each FDT. All structural components which contact the fuel and flush salt are constructed of Hastelloy-N. The chemical composition of Hastelloy-N²⁶ is presented in Table 5.2.1.2.

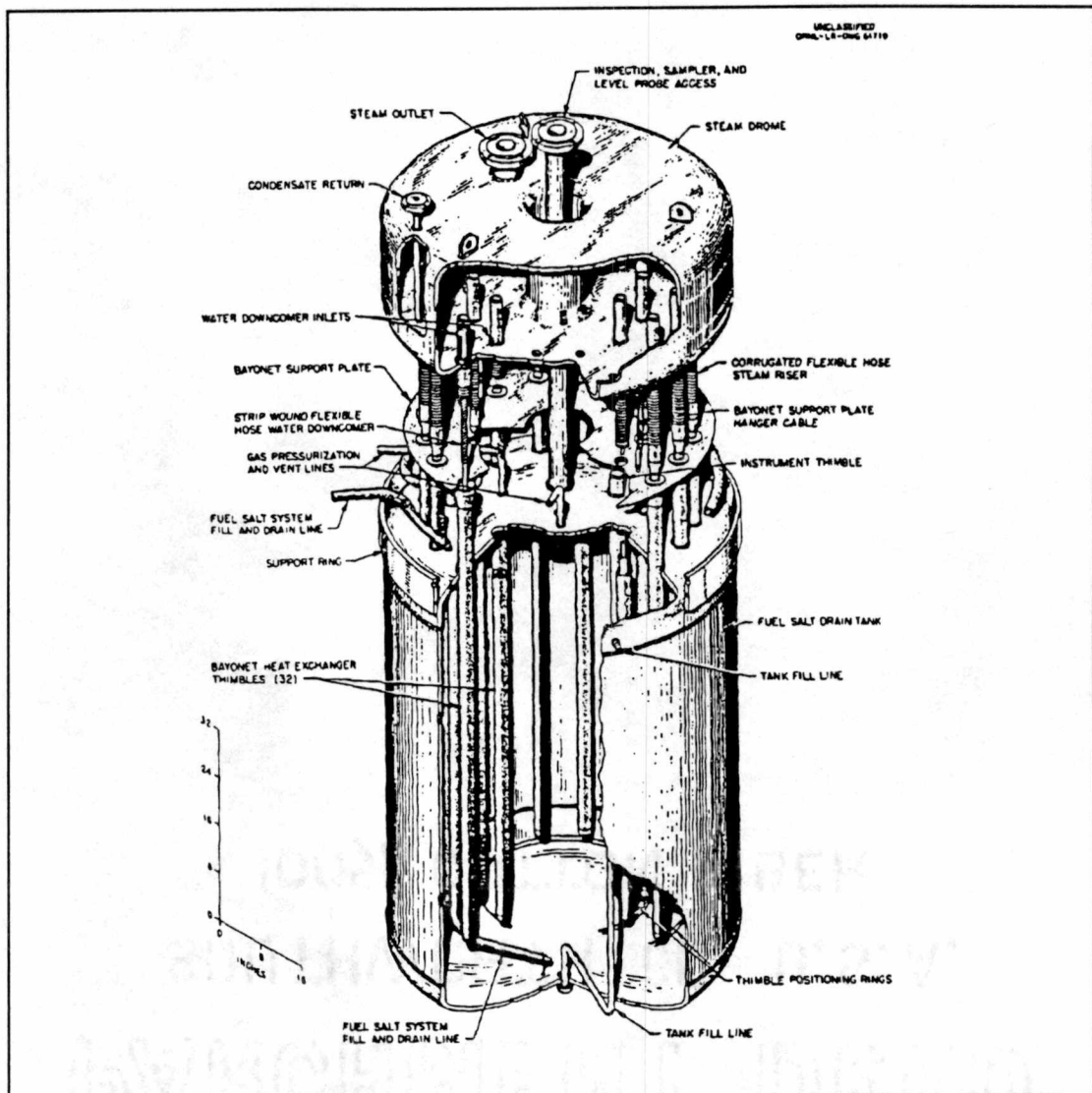


Figure 5.2.1.4: Isometric View of a Fuel Salt Drain Tank (from Ref. 16).

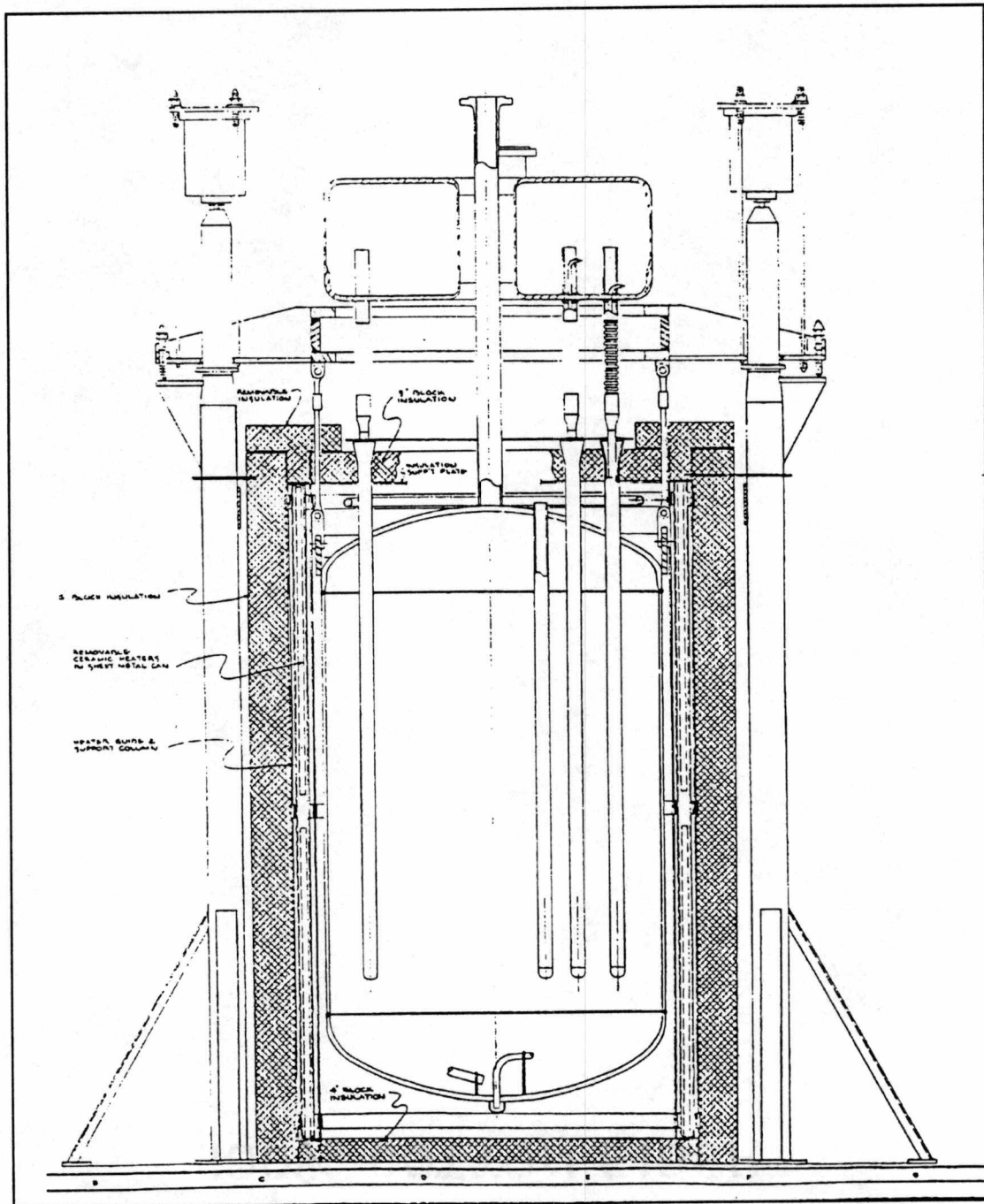


Figure 5.2.1.5: Elevation of a Fuel Salt Drain Tank (from MSR Drawing: E-MM-B-51652).

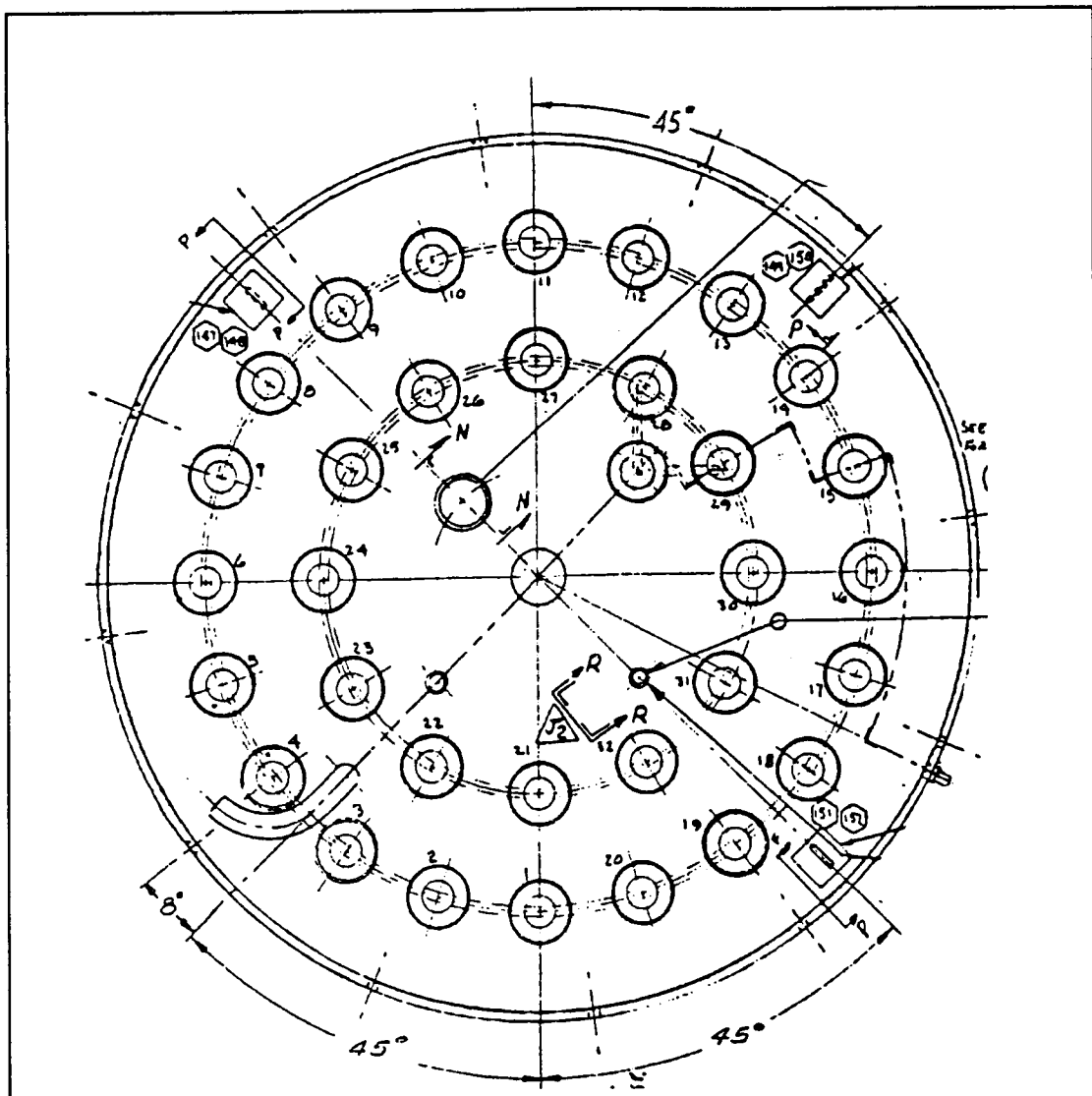


Figure 5.2.1.6: Cross Sectional View of Fuel Salt Drain Tank (from MSR Drawing: D-FF-A-40458).

Table 5.2.1.2: Composition of Hastelloy-N (Ref. 26).

Element	Atomic Number Density (atoms/barn*cm)
Ni	6.45e-02
Mo	9.11e-03
Cr	7.13e-03
Fe	4.74e-03

Each tank is surrounded by a bank of heaters which are constructed of stainless-steel (SS347). The top of each tank is insulated with ~10.795 cm. (4.25 in.) of Fiberfrax (type 907-JH). According to Dillon,²⁷ Fiberfrax is composed of 51.3% Al₂O₃ and 47.2% SiO₂ at a density of 3.247 g/cm³. The insulation on the bottom and surrounding each tank is Careytemp 1600 expanded silica. A search of the literature did not reveal any documentation for Careytemp 1600. However, a common form of expanded silica is vermiculite¹⁶ which is well documented in the literature. The chemical composition of vermiculite^{28,25} is specified as Ca_{0.7}(FeAl)₃Si₈O₂₀(OH)₄(H₂O)₈ at a density of 0.072 g/cm³.

Each tank is suspended by a support structure approximately 30.48 cm. (12 in.) above the concrete floor. The support structure is constructed of low density stainless-steel (SS316). There is additional equipment located in the FDTC, but this equipment is not considered in the evaluation.

5.2.2 Fissile Material Inventory

There are some uncertainties regarding the reported parameters of the fuel and flush salt. Reference [29] addresses these uncertainties by providing the maximum and minimum reported values. Whenever possible, the "maximum" value is used in the evaluation of the tanks. The following sections discuss the fissile material inventory in the two FDT's and FFT.

5.2.2.1 FDT Inventory

The nominal mole percentages of the salt inventory provided by Reference [30] are presented in Table 5.2.2.1.1. In addition, the total mass and density of the fuel salt²⁹ is 4845.7 kg and 2.48 g/cm³, respectively. Using these values, the calculated fuel salt volume is 1,953,911.3 cm³.

Table 5.2.2.1.1: Mole Percentages for FDT Salt Inventory.

Chemical Compound	Mole Percentage
⁷ LiF	64.50
BeF ₂	30.18
ZrF ₄	5.19
²³³ UF ₄	0.132
PuF ₃	2.38e-03

²⁹The word maximum defines the expected parameter that would have the largest positive nuclear reactivity effect for the system.

Based on the mole percentages, the masses of the salt constituents were calculated and are presented in Table 5.2.2.1.2. As shown in Table 5.2.2.1.2, the mass of UF_4 is 49.409 kg which is $\sim 75.44\%$ U by weight. This corresponds to 37.274 kg of uranium. Similarly, the mass of PuF_3 is 0.853 kg which is composed of ~ 80.75 wt% Pu. This corresponds to 0.689 kg of plutonium. Moreover, the isotopic percentages for the fissile material³⁰ are presented in Table 5.2.2.1.3. The amount of fuel salt stored in FDT 1 and 2 is 53.30 and 46.70 wt%, respectively.²⁹ Using the above parameters, the number densities of the fuel salt constituents were calculated and are presented in Appendix E.

Table 5.2.2.1.2: FDT Fuel Salt Composition.

Chemical Compound	Mass (kg)
^7LiF	2029.604
BeF_2	1716.081
ZrF_4	1049.753
$^{233}\text{UF}_4$	49.409
PuF_3	0.853

Table 5.2.2.1.3: Isotopic Percentages of FDT Fissile Material.

Isotope	Weight Percent
UF ₄	
²³³ U	83.918
²³⁴ U	7.478
²³⁵ U	2.558
²³⁶ U	0.105
²³⁸ U	5.941
PuF ₃	
²³⁹ Pu	90.13
²⁴⁰ Pu	9.52
²⁴¹ Pu	0.35

5.2.2.2 FFT Inventory

The total mass of the flush salt is 4300 kg.³¹ Moreover, the density of the FDT inventory is assumed for the flush salt (i.e. 2.48 g/cm³). The flush salt is composed of 66 mole% LiF and 34 mole% BeF₂.³² According to Reference [29], the flush salt also contains 0.49 kg and 0.069 kg of uranium and plutonium, respectively. The uranium and plutonium is assumed to be UF₄ and PuF₃, respectively. The isotopic percentages of the fissile material are presented in Table 5.2.2.2.1. Based on the mole percentages of LiF and BeF₂ and the masses of UF₄ and PuF₃, the total flush salt mass was calculated to be 4300.733 kg. The masses of the flush salt constituents are tabulated in Table 5.2.2.2.2. Based on the total mass, the calculated volume of the flush salt is 1,734,166.8 cm³. The calculated atomic number densities for the flush salt are presented in Appendix E.

Table 5.2.2.2.1: Isotopic Percentages of FFT Fissile Material.

Isotope	Weight Percent
UF₄	
²³³ U	40.55
²³⁴ U	3.73
²³⁵ U	17.67
²³⁶ U	0.25
²³⁸ U	37.80
PuF₃	
²³⁹ Pu	94.74
²⁴⁰ Pu	4.78
²⁴¹ Pu	0.44
²⁴² Pu	0.04

Table 5.2.2.2.2: Flush Salt Composition.

Chemical Compound	Mass (kg)
⁷ LiF	2226.943
BeF ₂	2073.057
²³³ UF ₄	0.648
PuF ₃	0.085

5.2.3 Integrity of the Fuel Drain Tank Cell

According to Reference [33], the flooding of the Fuel Drain Tank Cell is "extremely improbable." Consequently, flooding of the FDTC is not a credible accident scenario. In contrast, no documentation is available regarding the integrity of the FDTC during an earthquake. Therefore, an earthquake is assumed to be a credible accident scenario and is examined in this work.

5.3 Facility Model

The objective of this evaluation is to determine the degree of subcriticality of the solidified fuel salt which is stored in three drain tanks of Building 7503. In order to accomplish this objective, a KENO V.a model of the FDTC must be developed. The development of this model is summarized in the following sections.

5.3.1 Fuel Drain Tank Model

The actual configuration of the FDT's is presented in Section 5.2. This information was used to develop the model of the fuel drain tanks. The dimensions of the FDT's were obtained from mechanical drawings of the fuel drain tank facility which are tabulated in the Appendix G.

The nominal dimensions of the FDT are tabulated in Table 5.3.1.1. Moreover, the thimbles, which are located in each FDT, consist of three regions: a downcomer, steam riser and an outer region. The dimensions of a thimble are tabulated in Table 5.3.1.2, and a schematic diagram of a thimble is presented in 5.3.1.1. It should be noted

that each tank contains salt lines and an additional thimble housing a thermocouple and level probes. The extra thimble and salt lines are not included in the KENO V.a model. Each tank has a top and bottom domed lid with an outside height of 31.909 cm. These lids are approximated in the KENO V.a model using the hemisphere option. Due to this approximation the outside height of the lids is 30.916 cm. Consequently, the overall height of the tank in the model is 216.772 cm. which is slightly shorter than the actual height of the FDT (i.e. 218.758 cm.). A cross sectional and elevational view of the FDT modeled with KENO V.a is presented in Figures 5.3.1.2 and 5.3.1.3, respectively.

Table 5.3.1.1: Nominal FDT Dimensions.

FDT Parameter	Dimension (cm)
Outside diameter	127.0
Outside height	218.758
Wall thickness	1.27
Top & bottom lid wall thickness	1.905

Table 5.3.1.2: Thimble Dimensions.

Thimble Parameter	Dimension (cm.)
Downcomer	
Inner radius	0.528
Outer radius	0.635
Steam Riser	
Inner radius	1.332
Outer radius	1.670
Outer Region	
Inner radius	2.045
Outer radius	2.413

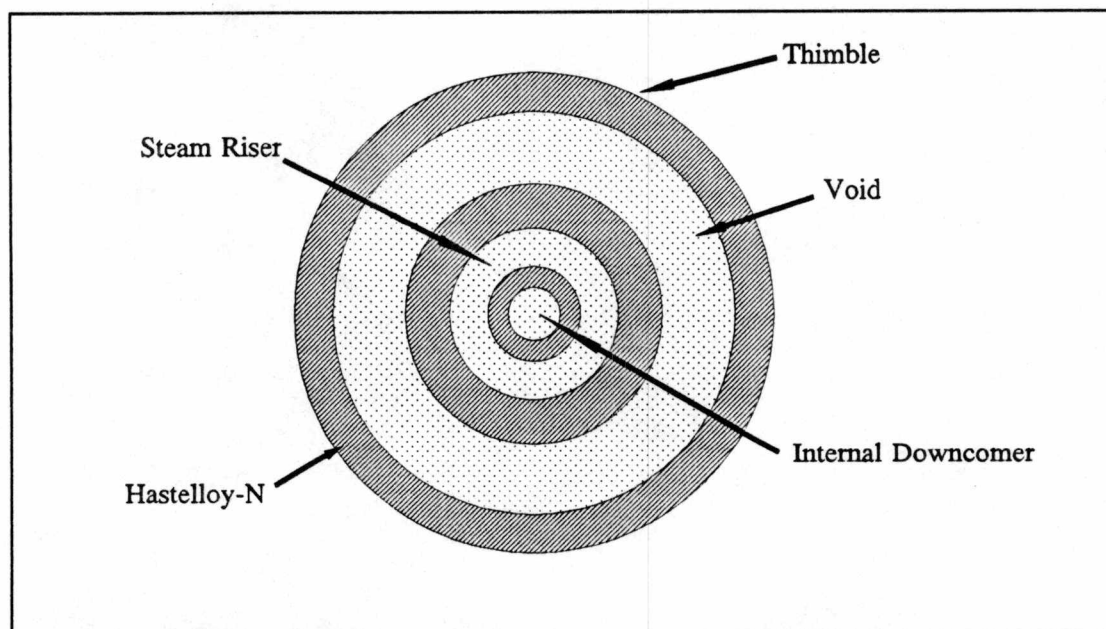


Figure 5.3.1.1: Schematic Diagram of FDT Thimble.

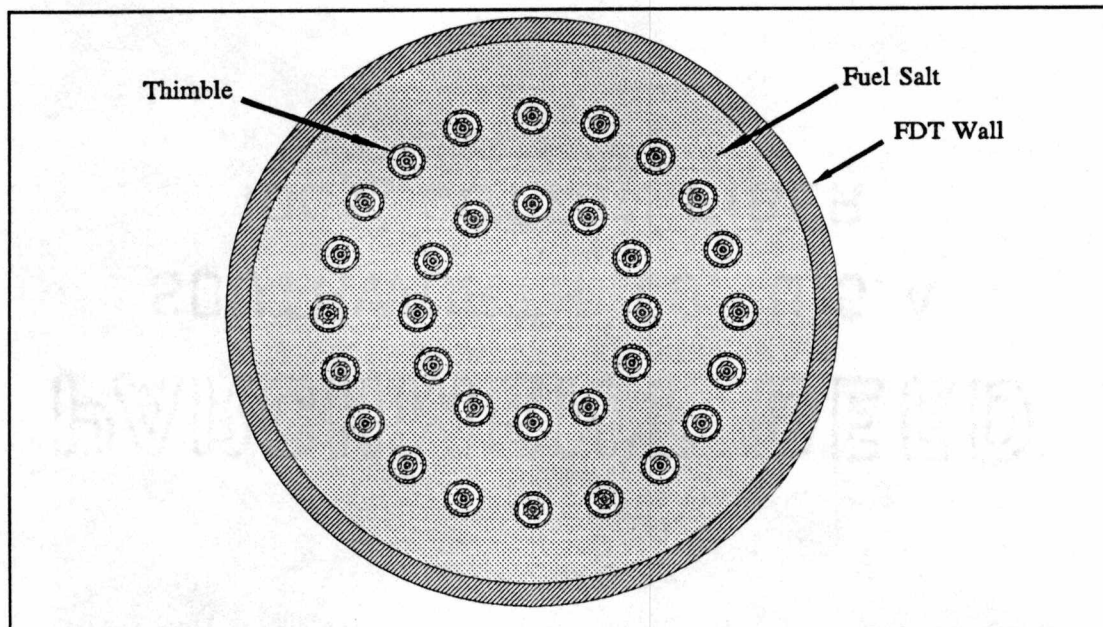


Figure 5.3.1.2: Cross Sectional View of Fuel Region of a FDT.

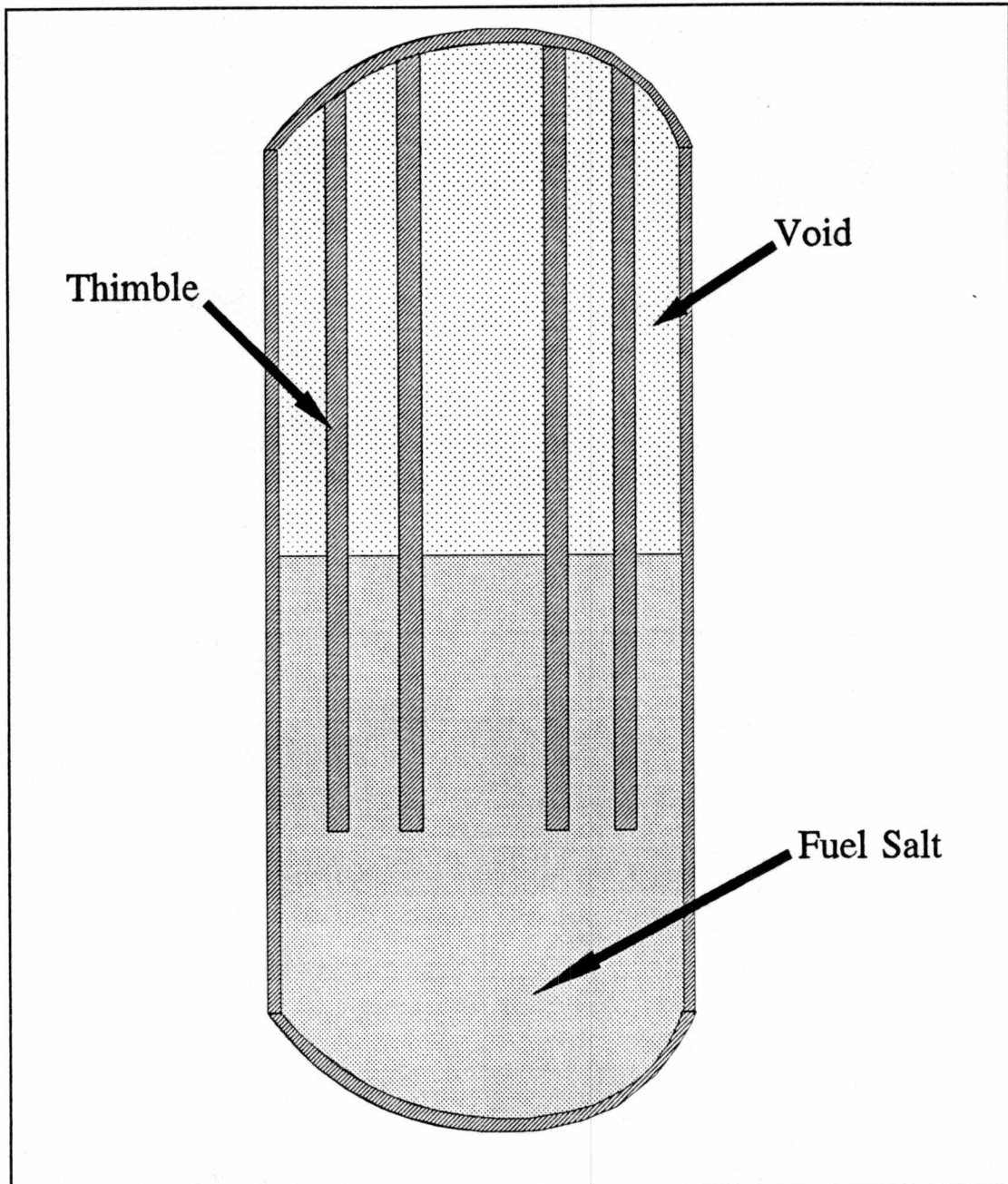


Figure 5.3.1.3: Schematic Diagram of an FDT without Heaters and Insulation.

As noted in Section 5.2.2, FDT 1 and 2 contains 2582.7 and 2263.0 kg of fuel salt, respectively. Based on a density of 2.48 g/cm^3 , the salt occupies 1,041,411.30 and 913,500.02 cm^3 in FDT 1 and 2, respectively. Using the available volume in the FDT model, the calculated salt height in FDT 1 and 2 is ~ 93.77 and 82.64 cm . (36.92 and 32.53 in.), respectively. The calculated salt height may be lower than the actual height due to differences between the model and the actual tank (e.g. neglecting the additional thimble and salt lines).

Surrounding each tank is a bank of heaters and insulation. As stated in Section 5.2.1, the heaters are constructed of SS347. Instead of modeling the heaters explicitly, the heaters are approximated in the KENO V.a model by surrounding the tank with a ring of low density* SS347 as shown in the top view in Figure 5.3.1.4. Also shown in Figure 5.3.1.4 is the vermiculite insulation surrounding the FDT.

*The density was calculated by determining the separate volume occupied by the heaters and air. Based on their respective volumes and theoretical densities, the mass of the heaters and air was determined. The reduced density was calculated by dividing the total mass of the air and steel by the total volume of the heaters and air.

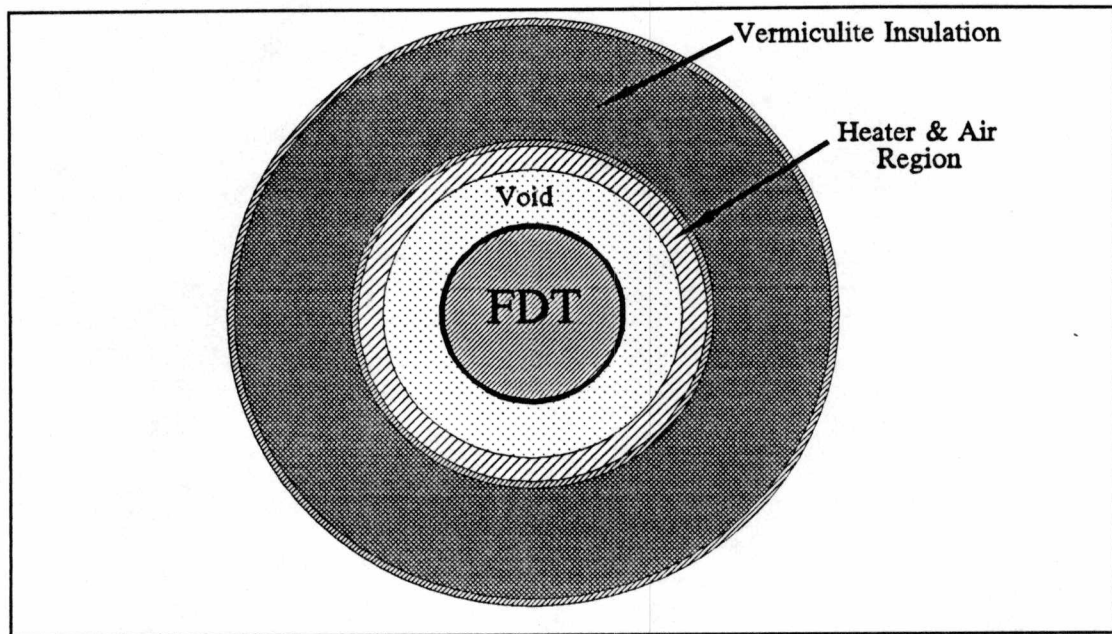


Figure 5.3.1.4: Top View of a Fuel Drain Tank without Top Layer of Fiberfrax Insulation.

As stated in Section 5.2.1, each tank is suspended above the floor with a support structure. In addition, a steam dome is located above each FDT. The steam dome is modeled explicitly with KENO V.a; however, the entire support structure is not modeled. Only the support posts and support ring are included in the model. A diagram of the entire FDT as modeled in KENO V.a is presented in Figure 5.3.1.5.

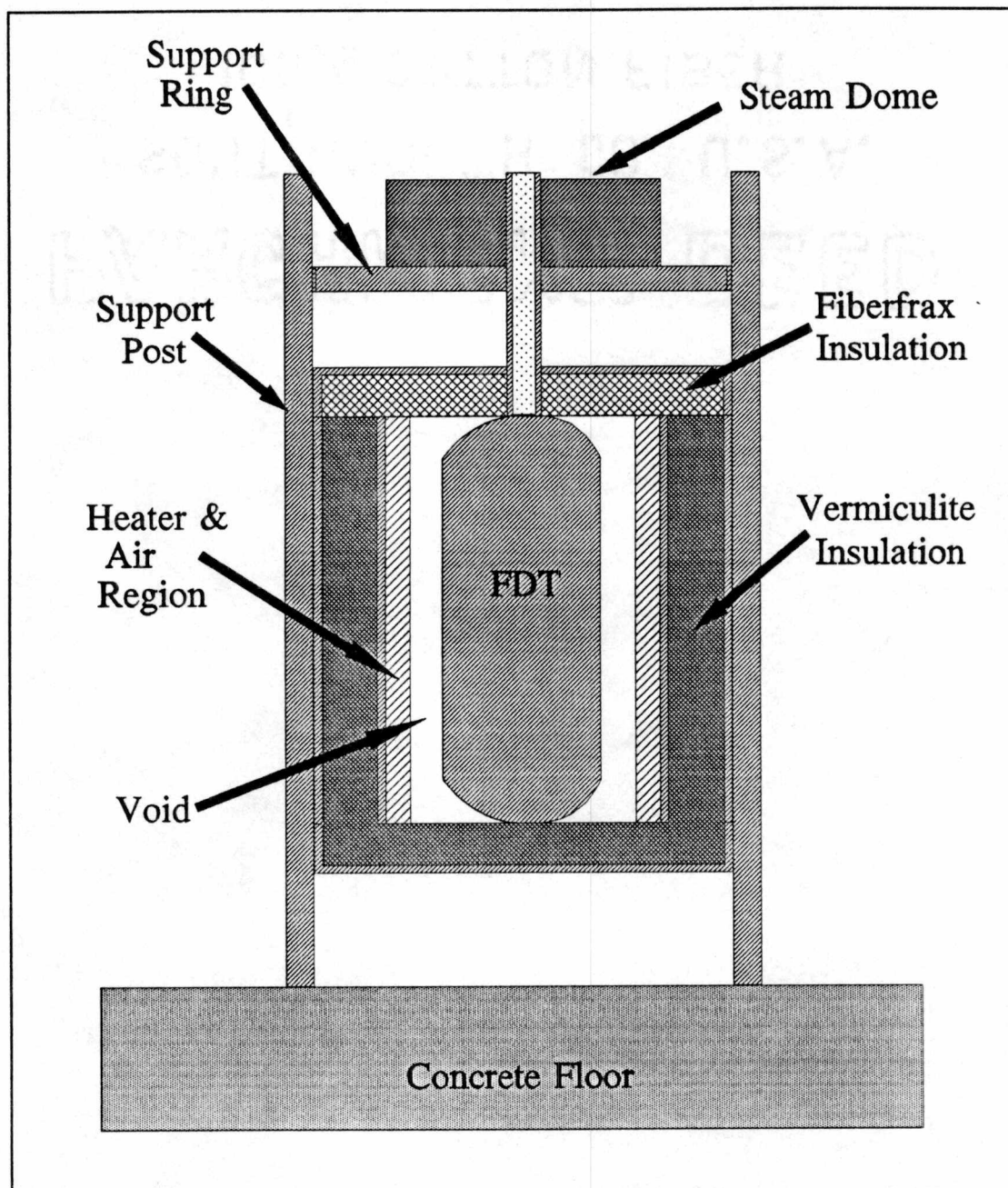


Figure 5.3.1.5: Diagram of a Fuel Drain Tank Including Heaters & Insulation as Modeled with KENO V.a.

5.3.2 Fuel Flush Tank Model

The dimensions of the FFT are the same as the FDT's, except the nominal height of the FFT is 213.678 cm. (84.125 in.). As in the FDT model, the top and bottom lids of the FFT are approximated using the hemisphere option in KENO V.a. As a result, the modeled height of the tank is 211.692 cm. (83.343 in.), which is shorter than the actual height. In contrast with the FDT, the FFT does not have the cooling system; however, the FFT is surrounded by a bank of heaters and insulation. Regarding the flush salt, the FFT contains 4300.733 kg of salt which occupies a volume of 1,734,166.8 cm³. Using the available volume in the FFT model, the calculated salt height is 155.69 cm. (61.30 in.). Consequently, the FFT is similar to the FDT except for the noted differences.

5.3.3 Fuel Drain Tank Cell Model

The nominal dimensions of the FDTC are presented in Section 5.2.1. The relative location of the FDT's and FFT in the FDTC are documented in the mechanical drawings of the MSR facility which are tabulated in the Appendix G.

The KENO V.a representation (top view) of the FDTC is presented in Figure 5.3.3.1. Also shown in Figure 5.3.3.1 is the location of FDT-1, FDT-2, and the FFT within the cell. The interior wall of the FDTC is lined with a 10.16 cm. (4 in.) layer of stainless steel (SS347). Moreover, the FDTC walls are modeled as ordinary concrete and are 91.44 cm. (3 ft.) thick. The KENO V.a model of the Fuel Drain Tank Cell is

entitled *msre.input* and is presented in Appendix F.

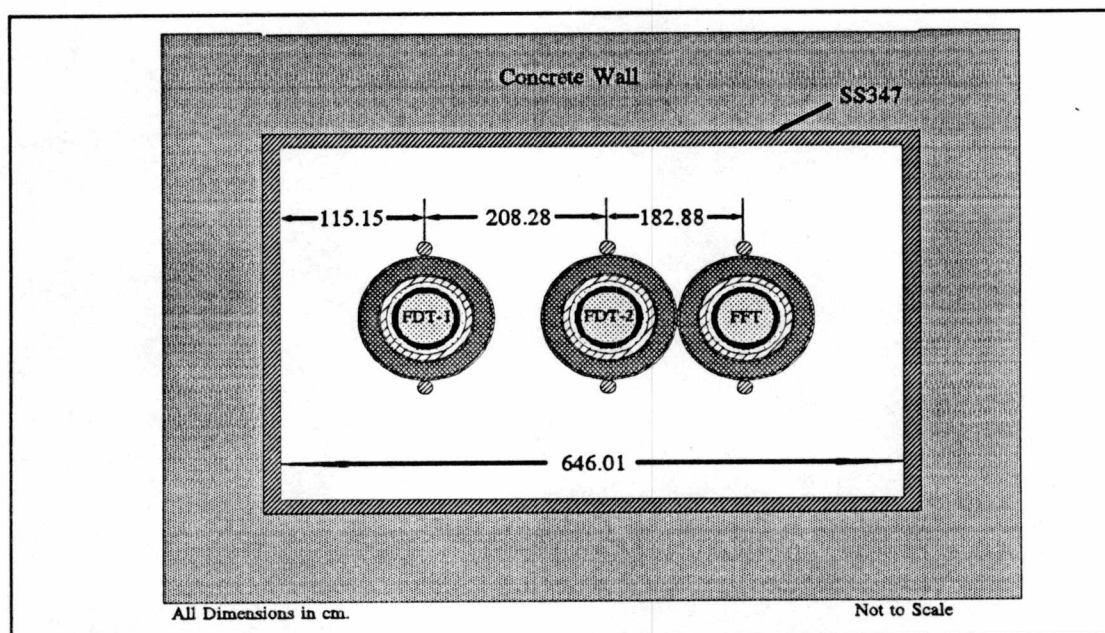


Figure 5.3.3.1: Top View of Fuel Drain Tank Cell as Modeled with KENO V.a.

5.4 Calculated Parameters

5.4.1 Fuel Drain Tank Cell

Using the KENO V.a model developed in Section 5.3, a criticality safety evaluation of the nominal configuration of the 2 FDT's and FFT in the Fuel Drain Tank Cell was performed. The results of these calculations are presented in Table 5.4.1.1 with the associated cross section library. As shown in Table 5.4.1.1, there is approximately a 12% difference between the highest (0.8524 ± 0.0028) and lowest (0.7517 ± 0.0032) calculated k_{eff} . Since KENO V.a is a multigroup Monte Carlo code, the differences in the calculated k_{eff} 's may be attributed to the use of group-wise cross sections. Therefore, the Fuel Drain Tank Cell should be re-evaluated with MCNP which uses continuous energy cross sections.

Table 5.4.1.1: Calculated k_{eff} 's for the Nominal Storage of the MSR Fuel Salt in the FDTC.

Cross Section Library	k_{eff}	AFG
238 ENDF/B-V	0.8059 ± 0.0029	152.07
44 ENDF/B-V	0.8432 ± 0.0028	23.91
218 ENDF/B-IV	0.8291 ± 0.0031	148.62
27 ENDF/B-IV	0.8524 ± 0.0028	16.66
Hansen-Roach	0.7517 ± 0.0032	12.54

In order to understand the differences in the calculated k_{eff} 's, a review of the validation study was performed (see Chapter Two). Based on the average fission groups, the MSR storage configuration does not fall into the range of AFG's presented in the

validation study. Moreover, the available critical experiment data for fluoride is in the form of aqueous UO_2F_2 which is significantly different than the MSR fuel salt. Despite the above differences, some insight can be gained by examining the experimental data.

As documented in Chapter Two, the two lowest moderated UO_2F_2 experiments are Experiments 4.35 and 4.39 with H/U's of 74.1 and 45.9, respectively. For the MSR configuration, the 44 group ENDF/B-V calculated k_{eff} is $\sim 1.0\%$ lower than the 27 group ENDF/B-IV value (see Table 5.3.1.1). This result is consistent with the differences in the calculated k_{eff} 's for the above critical experiments. Similarly, the 238 group calculated k_{eff} for the MSR fuel salt is $\sim 2.8\%$ lower than the value obtained with the 218 group library. In comparison, the 238 group calculated k_{eff} 's for Experiments 4.35 and 4.39 is $\sim 2.1\%$ lower than the corresponding 218 group value. As noted previously in Table 5.4.1.1, the largest difference in the calculated k_{eff} 's exists between the Hansen-Roach and 27 group ENDF/B-V cross section libraries (i.e. $\sim 12.0\%$). Regarding Experiments 4.35 and 4.39, the calculated effective multiplication factors obtained with the Hansen-Roach library are $\sim 5\%$ below the ENDF/B-IV values which is not as substantial as the differences observed for the MSR study. However, as noted in the validation study for the 5 cross section libraries, the differences between the calculated k_{eff} 's increases significantly as H/U decreases. Consequently, this observed trend for the uranyl fluoride experiments is consistent with the spread in the calculated k_{eff} 's for the low moderated MSR fuel.

Based on the observed trend, the ENDF/B-IV results are the most conservative for fluoride systems (i.e. highest calculated k_{eff}). However, the study of the MSR is intended to be performed with ENDF/B-V cross sections. Therefore, the 44 group library is used for the remainder of the evaluation due to the higher value of k_{eff} obtained relative to the 238 group library. As a result, the calculated k_{eff} for the nominal storage of the MSR fuel salt is 0.8432 ± 0.0028 .

5.4.2 Interaction Between Tanks

In order to evaluate the interaction between the drain tanks, each tank is examined independently. Using the KENO V.a model of the FDTC (i.e. filename: *msre.input*) without FDT-2 and the FFT, the effective multiplication factor of FDT-1 in the cell was calculated. Similarly, FDT-1 and the FFT were removed from the FDTC model, and the k_{eff} of FDT-2 was calculated. In the third case, the above calculation was repeated for the FFT without the FDT's in the storage cell. The results of these calculations are presented in Table 5.4.2.1.

As shown in Table 5.4.2.1, the k_{eff} of the FFT is substantially less than the two FDT's, and does not contribute significantly to the overall reactivity of the system. Furthermore, the highest calculated k_{eff} is obtained for FDT-1 which is 0.67% lower than that of the FDT Cell (i.e. 0.8432 ± 0.0028). In contrast, the k_{eff} for FDT-2 alone is 6.3% lower than FDT-1. The differences between the FDT's is attributed to the differences in the amount of salt in each tank. As documented in Section 5.2.2.1, FDT-1

contains 6.6% more salt than FDT-2. These results indicate that each tank behaves independently of the other tanks, and FDT-1, which has the most salt, is responsible for the overall reactivity of the Fuel Drain Tank Cell.

Table 5.4.2.1: Calculated k_{eff} 's for a Single MSR Fuel Drain Tank.

Description	k_{eff}	AFG
FDT-1	0.8357 ± 0.0031	23.90
FDT-2	0.7832 ± 0.0028	23.80
FFT	0.1109 ± 0.0004	36.25

5.5 Area of Model Applicability

A brief comparison of the nominal FDT Cell configuration and the experimental database is presented in Section 5.4.1. As noted previously, the average neutron energy group for fission (i.e. AFG) for the FDT Cell is 23.91 which is not in the range of AFG's presented in the validation study. In addition, the only experimental data involving fluoride is a series of aqueous UO_2F_2 critical experiments which are significantly different than the fluoride salt present in the fuel drain tanks. These results illustrate the need for critical experiments involving $^{233}\text{UF}_4$ and PuF_3 reflected with concrete. Despite the differences, the KENO V.a results can be compared with the available experimental data.

As documented previously, Experiments 4.35 and 4.39 are the lowest moderated $^{233}\text{UO}_2\text{F}_2$ experiments with H/U's of 74.1 and 45.9, respectively. Both experiments consist of aluminum cylinders partially filled with aqueous uranyl fluoride and reflected with paraffin. Although the two experiments are different from the FDT Cell, it is useful to compare the neutronics of the drain tanks with the two experiments. The percent fissions for Experiment 4.39 (H/U = 45.9) and the FDT Cell is presented in Figure 5.5.1. Likewise, the percent fissions for Experiment 4.35 (H/U = 74.1) and the FDT Cell is presented in Figure 5.5.2. As shown in both figures, Experiments 4.35 and 4.39 have similar fission percentages which is to be expected. However, there is a difference in the fission percentages between the MSR drain tanks and the two experiments as shown in Figures 5.5.1 and 5.5.2. The average energy group for fission is 31.59 and 29.67 for Experiments 4.35 and 4.39, respectively. As shown in both figures, there is a greater number of fissions occurring at the lower energy levels (i.e. higher fission groups) for the UO_2F_2 experiments relative to the FDTC which is indicated by the two distinct peaks above the 30th energy group. The larger amount of thermal fissions in the two experiments is due to the presence of hydrogen in both cylinders. In contrast, the FDT Cell has two distinct peaks at the 17th and 23rd energy groups indicating a somewhat harder spectrum than the UO_2F_2 experiments. In the fuel drain tanks the only moderator is the fluoride salt which explains the greater amount of fissions at the higher energy levels. Despite the differences between the UO_2F_2 experiments and the FDT Cell, the peaks although different in magnitude occur at approximately the same energy level.

Although the storage configuration of the MSR Fuel Drain Tank Cell does not fall into the range of applicability defined in the ^{233}U validation study (Chapter Two), there is evidence that the KENO V.a model is capable of calculating the effective multiplication factor of the fuel salt stored in the drain tanks.

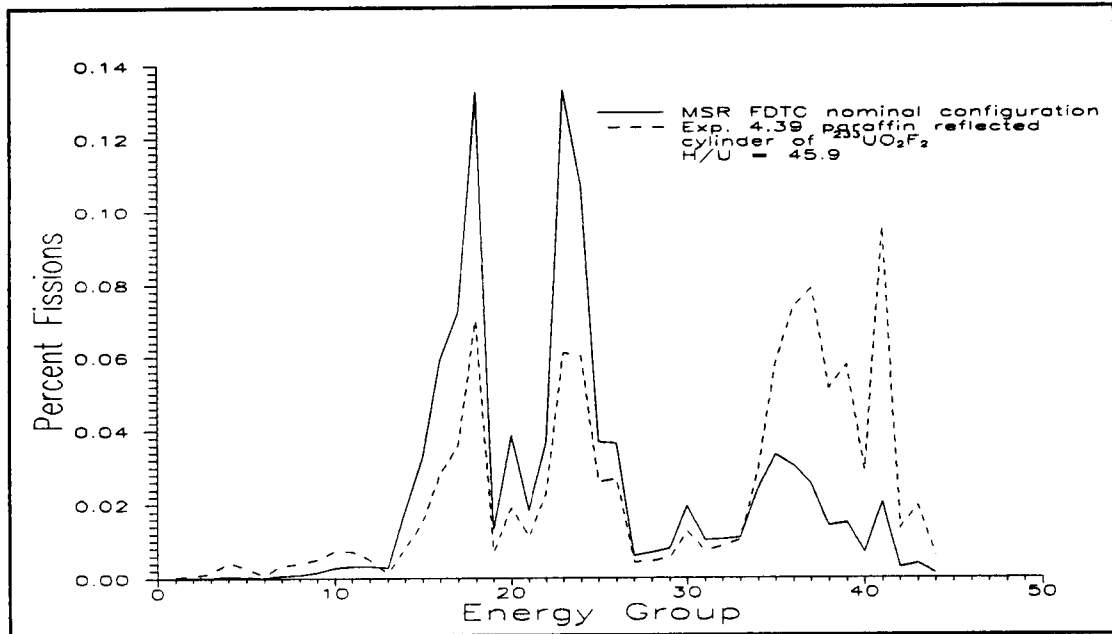


Figure 5.5.1: Comparison of Percent Fissions for the MSR Fuel Drain Tank Cell and a Paraffin Reflected Cylinder of Aqueous $^{233}\text{UO}_2\text{F}_2$ with H/U = 45.9.

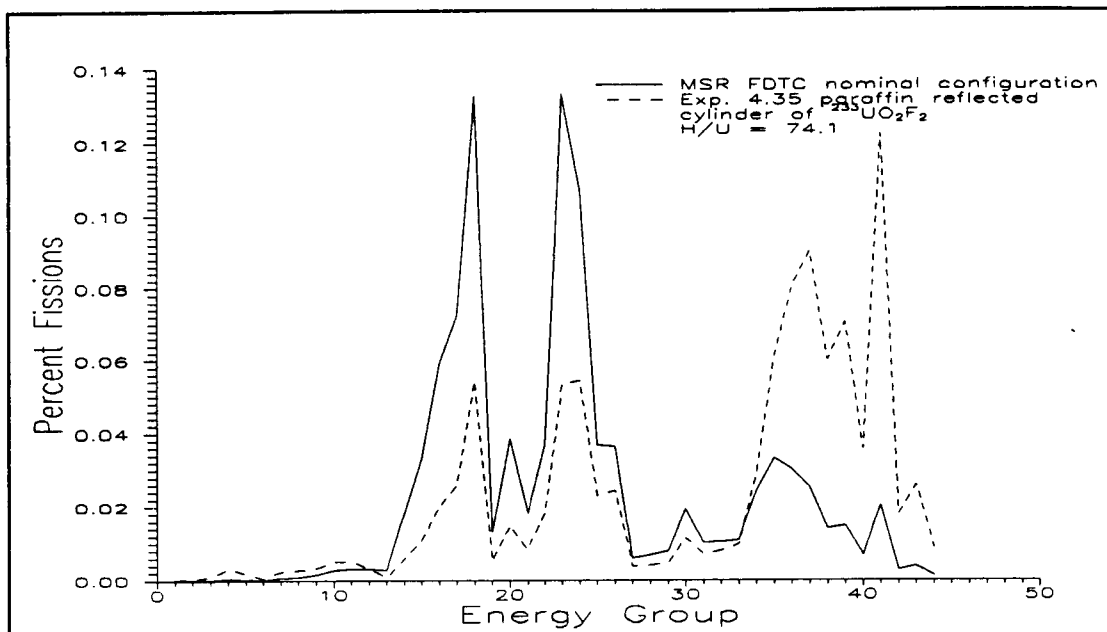


Figure 5.5.2: Comparison of Percent Fissions for the MSR Fuel Drain Tank Cell and a Paraffin Reflected Cylinder of Aqueous $^{233}\text{UO}_2\text{F}_2$ with $\text{H/U} = 74.1$.

5.6 Accident Modeling

After examining the nominal configuration, two abnormal storage configurations are evaluated. As stated in Section 5.2.3, the flooding of the FDTC is not a credible accident scenario; however, the integrity of the storage configuration during a natural disaster (e.g. earthquake) is a credible concern. Consequently, both accident scenarios examine the consequences of such a disaster on the FDTC storage facility.

In the first scenario, the structure of the facility and drain tanks are assumed to remain intact, but the heaters, insulation, and canning around the tanks are assumed to be removed thus allowing the tanks to come into contact.¹⁶ Furthermore, the tanks are

shifted to a corner of the cell increasing the neutron reflection around the tanks. A schematic diagram of the first accident scenario is presented in Figure 5.6.1.

In the second scenario, the assumptions in the first configuration are repeated, but the cell walls and ceiling are assumed to collapse around the storage tanks forming a tight reflector on all faces of the tanks as shown in Figure 5.6.2.

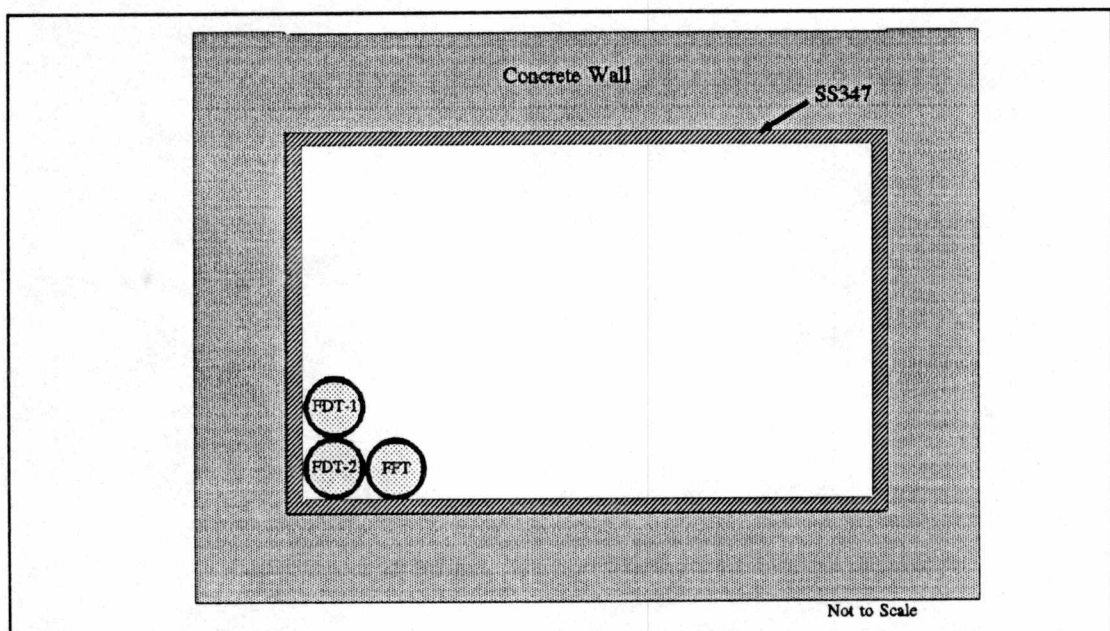


Figure 5.6.1: Diagram of the First Accident Scenario as Modeled with KENO V.a.

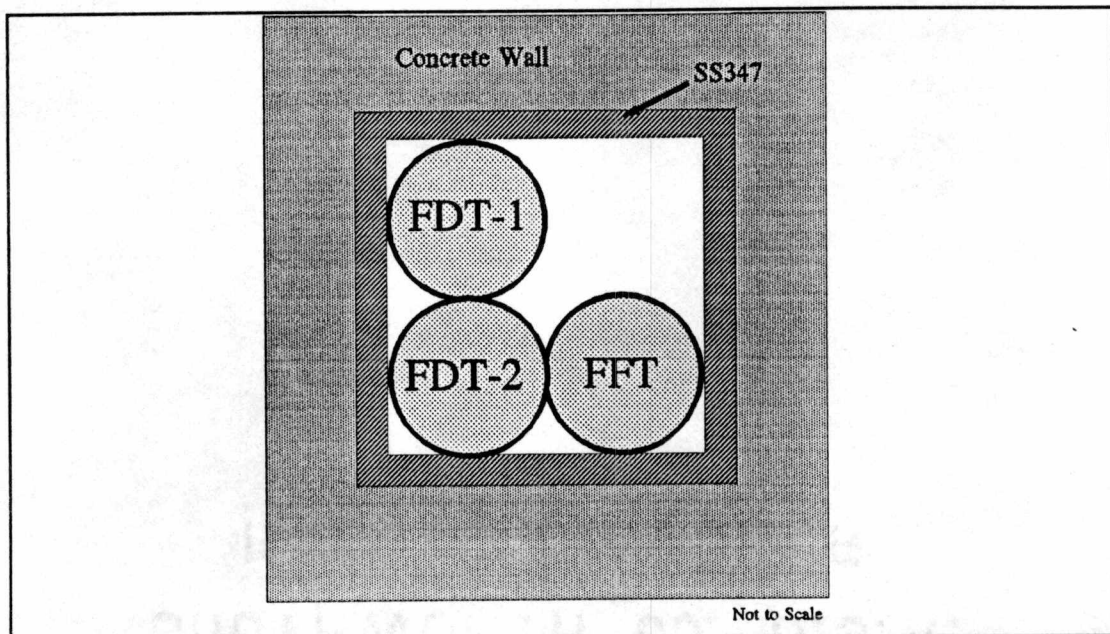


Figure 5.6.2: Diagram of the Second Accident Scenario as Modeled with KENO V.a.

The results of both accident scenarios are presented in Table 5.6.1. As shown in Table 5.6.1, the differences between the cross section libraries is substantial in both accident scenarios. These differences correspond to the results for the nominal configuration which are discussed in Section 5.4.1. Despite the differences, there is not a large increase from the nominal value of k_{eff} for a particular cross section library in either abnormal configuration. Regarding the results obtained with the 44 group ENDF/B-V library, the calculated k_{eff} is significantly below the USL of 0.9554 for this library. This result indicates that the MSR storage configuration is acceptably subcritical during an accident which results in the disruption of the FDT Cell. Furthermore, it is believed that the probability of these accident scenarios is exceedingly low; however, the determination of this probability is outside the scope of this research.

Table 5.6.1: KENO V.a Calculations of the Abnormal Configurations of the FDTC.

Cross Section Library	k_{eff}	AFG
Abnormal Configuration 1		
238 ENDF/B-V	0.8580 ± 0.0028	153.41
44 ENDF/B-V	0.8976 ± 0.0029	24.12
218 ENDF/B-IV	0.8934 ± 0.0030	149.74
27 ENDF/B-IV	0.9078 ± 0.0026	16.75
Hansen-Roach	0.8042 ± 0.0029	12.60
Abnormal Configuration 2		
238 ENDF/B-V	0.8717 ± 0.0029	153.65
44 ENDF/B-V	0.9112 ± 0.0030	24.14
218 ENDF/B-IV	0.9040 ± 0.0029	150.0
27 ENDF/B-IV	0.9245 ± 0.0030	16.79
Hansen-Roach	0.8148 ± 0.0029	12.60

5.7 Comparison to Previous Work

As stated in Chapter One, a criticality safety study of the MSR Fuel Drain Tank Cell using KENO V.a with the 27 group ENDF/B-IV cross section library was performed by Hollenbach and Hopper.¹⁶ Moreover, the work contained herein is intended to serve as an independent criticality safety evaluation of the same storage facility using KENO V.a with the 44 group ENDF/B-V cross section library. Therefore, it is appropriate to compare the results obtained in this work with the previous study.

The calculated k_{eff} 's of reference [16] for the nominal configuration of the Fuel Drain Tank Cell are presented in Table 5.7.1. As shown in Table 5.7.1, the calculated k_{eff} 's obtained in this work are within a 1.0% difference of the previous study. Despite the differences, both models show that the MSR FDT Cell is acceptably subcritical.

Table 5.7.1: Comparison of KENO V.a Calculations for the Nominal Configuration of the MSR FDT Cell.

Cross Section Library	k_{eff}	
	Dunn	Hollenbach
238 ENDF/B-V	0.8059 ± 0.0029	0.8145 ± 0.0018
44 ENDF/B-V	0.8432 ± 0.0028	0.8468 ± 0.0016
218 ENDF/B-IV	0.8291 ± 0.0031	—
27 ENDF/B-IV	0.8524 ± 0.0028	0.8602 ± 0.0021
16 Hansen-Roach	0.7517 ± 0.0032	0.7514 ± 0.0015

Chapter Six

MSR Sensitivity Studies

6.1 Introduction

A review of the literature describing the MSR fuel drain tanks indicated that there is some uncertainty related to the fuel salt parameters. In order to quantify the amount of uncertainty, a sensitivity study is performed. In each case a single parameter is varied in the model, and the k_{eff} of the systems is calculated using KENO V.a with the 44 group ENDF/B-V library. All of the results are compared with the calculated k_{eff} for the nominal storage configuration (i.e. 0.8432 ± 0.0028).

6.2 Sensitivity of Salt Mass

The salt mass used in the nominal model of the drain tanks is 4845.7 kg. As stated in Section 5.7, this mass is $\sim 3\%$ higher than the mass used in the previous work.¹⁶ In order to evaluate the effect of the mass on the system, the overall salt mass is varied $+5\%$ and -5% , respectively.¹⁶ Furthermore, the salt stored in FDT-1 is increased 10% by removing the corresponding salt from FDT-2 in the third calculation.¹⁶ In the fourth scenario, the entire salt inventory is placed in FDT-1.

The results of the above calculations are presented in Table 6.2.1. As shown in Table 6.2.1, a 5% increase in the overall salt inventory results in a 1.25% increase in k_{eff} . Similarly, decreasing the salt inventory by 5% corresponds to a 1.94 decrease in reactivity. In the third case, the shifting of 10% of the fuel from FDT-2 to FDT-1

increases k_{eff} by 4.67%. However, when all of the salt from FDT-2 is transferred to FDT-1 in the fourth scenario, the system is critical. These results show that the tank with the largest fuel salt mass accounts for the overall reactivity of the system, and the salt inventory must be divided and stored in separate tanks.

Table 6.2.1: Calculated k_{eff} 's of the MSR FDT Cell with Variations in the Fuel Salt Mass.

Description	k_{eff}	Percent Change from Nominal Case
5% increase in salt mass	0.8537 ± 0.0029	+1.25
5% decrease in salt mass	0.8268 ± 0.0031	-1.94
10% increase in FDT-1 mass by removing 10% of FDT-2 mass	0.8826 ± 0.0030	+4.67
Total salt inventory stored in FDT-1	1.0280 ± 0.0032	+21.92

6.3 Sensitivity of Density

As documented in Chapter Five, the nominal fuel and flush salt density used in the evaluation is 2.48 g/cm^3 , which is the maximum density reported in the literature. However, the effects of density variations in the system should be investigated.

In the first two calculations, the fuel salt density is varied $\pm 10\%$. Moreover, in the last two cases, the uranium and plutonium density is varied $\pm 10\%$.¹⁶ The results of these calculations are presented in Table 6.3.1. The largest reactivity change occurs with the variation of the density of the fuel salt as shown in Table 6.3.1. Moreover, the

increase and decrease in the fissile material density has a corresponding increase and decrease in the calculated k_{eff} of the system, respectively. Despite the large variations in density, the calculated k_{eff} for the FDT Cell is below the USL of 0.9554.

Table 6.3.1: Calculated k_{eff} 's of the MSR FDT Cell with Variations in the Density.

Description	k_{eff}	Percent Change from Nominal Case
10% increase in salt density	0.8971 ± 0.0030	+6.39
10% decrease in salt density	0.7679 ± 0.0030	-8.93
10% increase in U and Pu density	0.8663 ± 0.0029	+2.74
10% decrease in U and Pu density	0.8223 ± 0.0029	-2.48

6.4 Removal of Tank Thimbles

The 32 bayonet tubes and thimbles located in FDT-1 and -2 are constructed of Hastelloy-N. Each thimble and bayonet tube extends from the top of the tank into the fuel salt. The reactivity effect of these components is investigated in this section.¹⁶ With the bayonet tubes and thimbles removed from the tanks, the calculated effective multiplication factor of the system is 0.8588 ± 0.0032 which is 1.85% higher than the nominal k_{eff} of the drain tanks. This result indicates that the bayonet tubes and thimbles act as a neutron absorber and are significant in determining the overall reactivity of the system.

6.5 Sensitivity of Reflectors

Each tank is surrounded with radial heaters which are constructed of SS347. In addition, the periphery and bottom of each tank is surrounded with expanded silica (Careytemp 1600 modeled as vermiculite) insulation, and the top of each tank is covered with fiberfrax insulation. The three tanks are also reflected by ordinary concrete walls and a steel liner at the cell boundary. The reactivity effect of the different reflectors is investigated in this section.

The first three calculations evaluate FDT-1 independently. In the first case, the heaters are removed, and FDT-1 is placed in the FDT Cell. Secondly, the previous calculation is repeated with the insulation also removed.¹⁶ In the third scenario, the second calculation is repeated, except the steel liner and concrete walls of the FDT Cell are also removed. The previous calculations are repeated for FDT-2, and the results of the six calculations are presented in Table 6.5.1. Also included in Table 6.5.1 are the reference configurations that were used for comparison.

The removal of the heaters from each FDT decreases the calculated $k_{\text{eff}} \sim 1.5\text{-}2\%$ as shown in Table 6.5.1. Furthermore, the removal of the insulation decreases the k_{eff} an additional 1.5%. When the concrete wall and steel liner are removed from the model, k_{eff} decreases an additional 1.0% relative to the reference case as shown in Table 6.5.1.

Table 6.5.1: Calculated k_{eff} 's of the MSR FDT Cell with Variations in the Reflector Conditions.

Description	k_{eff}	Percent Change from Reference Case
FDT-1 Reference Case: FDT-1 with insulation & heaters modeled independently in FDT Cell $k_{eff} = 0.8357 \pm 0.0031$		
Heaters removed	0.8236 ± 0.0032	-1.45
Insulation & heaters removed	0.8107 ± 0.0028	-3.00
Insulation, heaters, concrete wall and steel liner removed	0.8049 ± 0.0031	-3.69
FDT-2 Reference Case: FDT-2 with insulation & heaters modeled independently in FDT Cell $k_{eff} = 0.7832 \pm 0.0028$		
Heaters removed	0.7656 ± 0.0031	-2.25
Insulation and heaters removed	0.7543 ± 0.0029	-3.69
Insulation, heaters, concrete wall and steel liner removed	0.7451 ± 0.0032	-4.86
FDT Cell Reference Case: Nominal Cell Configuration $k_{eff} = 0.8432 \pm 0.0028$		
Water Filled FDT Cell, includes all 3 tanks	0.8494 ± 0.0029	+0.74

As noted in Chapter 5, flooding of the FDT Cell is not considered a credible accident scenario and is not evaluated. For sensitivity purposes only, the flooding of the cell with water is evaluated in the seventh case, and the results of this scenario are also presented in Table 6.5.1. Flooding of the FDT Cell results in a 0.74% increase in the calculated k_{eff} relative to the nominal case as shown in Table 6.5.1. Consequently, if flooding was a credible accident scenario, the additional water reflection would not increase the system reactivity significantly.

Chapter Seven

Conclusions and Recommendations

7.1 Summary of Results

A nuclear criticality safety evaluation of the stored ^{233}U inventory in the wells of Building 3019 and the MSR FDT Cell at ORNL has been performed using KENO V.a with ENDF/B-V cross sections. Prior to evaluating the storage configurations, KENO V.a and the 44 & 238 energy group ENDF/B-V cross section libraries were validated for ^{233}U systems.

A series of 51 ^{233}U critical experiments with H/ ^{233}U ratios ranging from 0 to 1986 were selected to validate the ENDF/B-V libraries. Thirty of the experiments consisted of uranyl nitrate solution in either cylindrical or spherical geometry, and four of the uranyl nitrate experiments involved 2 and 3 tier cubic arrays of cylindrical units. Likewise, twelve experiments consisted of uranyl fluoride solution in cylindrical or spherical geometry while the remaining nine experiments involved spheres of uranium metal. To compensate for the lack of concrete reflected ^{233}U critical experiments, a series of four concrete reflected experiments involving mixed PuO_2 and UO_2 (Fast Test Reactor FTR) were also included in the validation study.

All experiments were modeled using the 44 and 238 group ENDF/B-V, the 27 and 218 group ENDF/B-IV, and the 16 group Hansen-Roach libraries. The mean calculated k_{eff} for all ^{233}U experiments using the 44 and 238 group libraries is $1.0090 \pm$

0.0021 and 1.0064 ± 0.0020 , respectively. In comparison, the mean calculated k_{eff} using the 27, 218 and 16 group libraries is 1.0142 ± 0.0038 , 1.0125 ± 0.0038 , and 0.9991 ± 0.0019 , respectively. In a general comparison of the ^{233}U experiments, the mean k_{eff} values for the ENDF/B-V libraries are 0.5-0.6% lower than the ENDF/B-IV results while the ENDF/B-V values are 0.7-0.98% higher than the Hansen-Roach results. Regarding the FTR experiments, the mean k_{eff} for the four experiments using the 44 and 238 group libraries is 1.0107 ± 0.0032 and 1.0005 ± 0.0033 , respectively. Likewise, the mean calculated k_{eff} using the 27, 218 and 16 group libraries is 1.0057 ± 0.0033 , 1.0010 ± 0.0034 and 1.0311 ± 0.0034 , respectively.

There is a definite improvement in the agreement between the calculated k_{eff} 's and the experimental results (i.e. $k_{\text{eff}} = 1.0$) obtained with the newer ENDF/B-V libraries relative to ENDF/B-IV. One of the more noticeable improvements obtained with the ENDF/B-V data is for uranium metal systems and is attributed to the treatment of the cross sections in the higher energy ranges.

After modeling the experiments, an Upper Safety Limit (USL) was determined based on the validation of KENO V.a and both ENDF/B-V libraries. A calculated k_{eff} plus two standard deviations lying below the USL is considered acceptably subcritical. The USL for the 44 & 238 group libraries is 0.9554 and 0.9557, respectively. In summary, the use of KENO V.a with either the 44 or 238 group ENDF/B-V libraries is considered to be validated for ^{233}U systems of the type considered in this work.

Once KENO V.a and the ENDF/B-V libraries were validated, the ^{233}U inventory in Building 3019 was evaluated. Currently, the fissile material is stored in three different arrays of wells within the facility (i.e. Cell 4, Inter-cell 3&4 and 4&5). The chemical form of the material is either U_3O_8 , UO_2 , UF_4 or U-metal. Regarding abnormal configurations, natural disasters (e.g. flooding, earthquake, etc.) are not credible accident scenarios for the Building 3019 storage wells. Therefore, the evaluation of the facility during the above natural disasters is not addressed in the criticality safety evaluation of the storage wells.

A KENO V.a model of the storage facility was developed and used to evaluate the subcritical margin of the nominal storage configuration. Initially, both ENDF/B-V libraries were used to evaluate the wells. Moreover, the differences in the results obtained with the 238 & 44 group libraries are not significant. Since the 238 group library has a finer group structure relative to the 44 group library, the 238 group ENDF/B-V library was used in the evaluation of the storage wells. The calculated k_{eff} for the nominal configuration is 0.8438 ± 0.0018 which is significantly below the USL of 0.9557. Therefore, the current storage configuration of ^{233}U in Building 3019 is acceptably subcritical.

Currently, there are nineteen empty wells (excluding the Inter-cell 2&3 wells) available for storage of ^{233}U . Using the KENO V.a model of the storage facility and the derived subcritical limit for the 238 group library, maximum loading limits for the empty

wells were determined. The loading limits for ^{233}U are 2.45, 1.90, 1.75, 1.30 and 0.95 kg U/ft for H/U atom ratios of 0, 3, 5, 10 and 20, respectively.

In the last phase of the research, the storage of the MSR fuel salt in the Fuel Drain Tank Cell was evaluated. Currently, the fuel salt, which is a mixture of LiF , BeF_2 , ZrF_4 , UF_4 and PuF_3 , is stored in 2 drain tanks (FDT -1 & -2), and the reactor flush salt is stored in a single tank (FFT). The three tanks are located in the Fuel Drain Tank Cell which has ordinary concrete walls lined with stainless steel (SS347).

A KENO V.a model of the FDT Cell was developed and used to evaluate the subcritical margin of the nominal storage configuration. Initially, both ENDF/B-V libraries were used to evaluate the FDT Cell; however, the results obtained with the 44 group library are $\sim 4.4\%$ higher relative to the 238 group library. Consequently, the 44 group library is used to evaluate the drain tanks. The calculated k_{eff} for the nominal configuration is 0.8432 ± 0.0028 which is significantly below the USL of 0.9554. Therefore, the current storage configuration of ^{233}U in the MSR Fuel Drain Tank Cell is acceptably subcritical.

After examining the nominal configuration, two abnormal storage configurations were evaluated. Flooding of the FDT Cell is not a credible accident scenario; however, the integrity of the tanks during a natural disaster such as an earthquake is a credible

concern. As a result, both accident scenarios, examined the consequences of such a disaster on the FDT Cell.

In the first scenario, the structure of the cell and drain tanks are assumed to remain intact, but the heaters, insulation, and canning around the tanks are assumed to be removed allowing the tanks to come into contact. In addition, the three tanks are shifted to a corner of the cell increasing the neutron reflection. The calculated k_{eff} for the first abnormal configuration is 0.8976 ± 0.0029 which is acceptably below the USL (i.e. 0.9554). In the second scenario, the assumptions in the previous calculations are repeated, but the cell walls and ceiling are assumed to collapse around the storage tanks forming a tight neutron reflector on all faces of the tanks. The calculated k_{eff} for the second abnormal configuration is 0.9112 ± 0.0030 which is acceptably below 0.9554.

7.2 Recommendations for Future Work

Prior to the completion of this research, revisions were being made to the ENDF/B-V library. Consequently, it would be worthwhile and informative to re-evaluate the storage wells in Building 3019 and the MSR fuel drain tanks with the revised ENDF/B-V library.

In addition, the original Molten Salt Reactor experiment could be used to validate KENO V.a and the ENDF/B-V cross section libraries.¹⁶ Moreover, there is a need for ^{233}U critical experiments which are low moderated (i.e. $H/U \leq 3$) and reflected with

concrete. Similar experiments are also needed which are interstitially moderated with concrete. Furthermore, there is a need for mixed ^{233}U & ^{235}U systems as well as fluoride salt (i.e. UF_4 and PuF_3) systems. If the above experimental data becomes available, the ENDF/B-V library and KENO V.a should be validated with these critical systems, and the ^{233}U inventory in Building 3019 and the MSR FDT Cell should be re-evaluated.

With regard to the evaluation of the MSR FDT Cell using KENO V.a, there are significant differences in the calculated k_{eff} 's obtained with the 44 & 238 group ENDF/B-V, 27 & 218 group ENDF/B-IV and 16 group Hansen-Roach cross section libraries. The differences may be attributed to the group-wise treatment of the resonance structure in the cross sections. Therefore, the FDT Cell should be re-evaluated with the Monte Carlo code MCNP which uses continuous energy cross sections.

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Appendix A

ENDF/B-IV and ENDF/B-V Energy Group Structure

Table A.1: Energy Group Comparison Between the 44 Group ENDF/B-V and 27 Group ENDF/B-IV Cross Section Libraries.

44 Group ENDF/B-V		27 Group ENDF/B-IV	
Group	Upper Energy Bound (eV)	Group	Upper Energy Bound (eV)
1	2.0000e+07	1	2.0000e+07
2	8.1873e+06		
3	6.4340e+06	2	6.4340e+06
4	4.8000e+06		
5	3.0000e+06	3	3.0000e+06
6	2.4790e+06		
7	2.3540e+06		
8	1.8500e+06	4	1.8500e+06
9	1.4000e+06	5	1.4000e+06
10	9.0000e+05	6	9.0000e+05
11	4.0000e+05	7	4.0000e+05
12	1.0000e+05	8	1.0000e+05
13	2.5000e+04		
14	1.7000e+04	9	1.7000e+04
15	3.0000e+03	10	3.0000e+03
16	5.5000e+02	11	5.5000e+02
17	1.0000e+02	12	1.0000e+02
18	3.0000e+01	13	3.0000e+01
19	1.0000e+01	14	1.0000e+01
20	8.1000e+00		
21	6.0000e+00		
22	4.7500e+00		
First Thermal Group 23	3.0000e+00	First Thermal Group 15	3.0500e+00
24	1.7700e+00	16	1.7700e+00
		17	1.3000e+00
		18	1.1300e+00
25	1.0000e+00	19	1.0000e+00
26	6.2500e-01	20	8.0000e-01
27	4.0000e-01	21	4.0000e-01
28	3.7500e-01		

44 Group ENDF/B-V		27 Group ENDF/B-IV	
Group	Upper Energy Bound (eV)	Group	Upper Energy Bound (eV)
29	3.5000e-01	22	3.2500e-01
30	3.2500e-01		
31	2.7500e-01		
32	2.5000e-01	23	2.2500e-01
33	2.2500e-01		
34	2.0000e-01		
35	1.5000e-01	24	1.0000e-01
36	1.0000e-01		
37	7.0000e-02		
38	5.0000e-02	25	5.0000e-02
39	4.0000e-02		
40	3.0000e-02		
41	2.5300e-02	26	3.0000e-02
42	1.0000e-02		
43	7.5000e-03		
44	3.0000e-03	27	1.0000e-02
	1.0000e-05		
			1.0000e-05

Table A.2: Energy Group Comparison Between the 238 Group ENDF/B-V and 218 Group ENDF/B-IV Cross Section Libraries.

238 Group ENDF/B-V		218 Group ENDF/B-IV	
Group	Upper Energy Bound (eV)	Group	Upper Energy Bound (eV)
1	2.0000e+07	1	2.0000e+07
2	1.7333e+07		
3	1.5683e+07		
4	1.4550e+07		
5	1.3840e+07		
6	1.2840e+07		
7	1.0000e+07		
8	8.1873e+06		
9	6.4340e+06	2	6.4340e+06
10	4.8000e+06	3	4.8000e+06
11	4.3040e+06	4	4.3040e+06
12	3.0000e+06	5	3.0000e+06
13	2.4790e+06	6	2.4790e+06
14	2.3540e+06	7	2.3540e+06
15	1.8500e+06	8	1.8500e+06
16	1.5000e+06	9	1.5000e+06
17	1.4000e+06	10	1.4000e+06
18	1.3560e+06	11	1.3560e+06
19	1.3170e+06	12	1.3170e+06
20	1.2500e+06	13	1.2500e+06
21	1.2000e+06	14	1.2000e+06
22	1.1000e+06	15	1.1000e+06
23	1.0100e+06	16	1.0100e+06
24	9.2000e+05	17	9.2000e+05
25	9.0000e+05	18	9.0000e+05
26	8.7500e+05	19	8.7500e+05
27	8.6110e+05	20	8.6110e+05
28	8.2000e+05	21	8.2000e+05
29	7.5000e+05	22	7.5000e+05
30	6.7900e+05	23	6.7900e+05

238 Group ENDF/B-V		218 Group ENDF/B-IV	
Group	Upper Energy Bound (eV)	Group	Upper Energy Bound (eV)
31	6.7000e+05	24	6.7000e+05
32	6.0000e+05	25	6.0000e+05
33	5.7300e+05	26	5.7300e+05
34	5.5000e+05	27	5.5000e+05
35	4.9952e+05	28	4.9952e+05
36	4.7000e+05	29	4.7000e+05
37	4.4000e+05	30	4.4000e+05
38	4.2000e+05	31	4.2000e+05
39	4.0000e+05	32	4.0000e+05
40	3.3000e+05	33	3.3000e+05
41	2.7000e+05	34	2.7000e+05
42	2.0000e+05	35	2.0000e+05
43	1.5000e+05	36	1.5000e+05
44	1.2830e+05	37	1.2830e+05
45	1.0000e+05	38	1.0000e+05
46	8.5000e+04	39	8.5000e+04
47	8.2000e+04	40	8.2000e+04
48	7.5000e+04	41	7.5000e+04
49	7.3000e+04	42	7.3000e+04
50	6.0000e+04	43	6.0000e+04
51	5.2000e+04	44	5.2000e+04
52	5.0000e+04	45	5.0000e+04
53	4.5000e+04	46	4.5000e+04
54	3.0000e+04	47	3.0000e+04
55	2.5000e+04	48	2.5000e+04
56	1.7000e+04	49	1.7000e+04
57	1.3000e+04	50	1.3000e+04
58	9.5000e+03	51	9.5000e+03
59	8.0300e+03	52	8.0300e+03
60	6.0000e+03	53	6.0000e+03
61	3.9000e+03	54	3.9000e+03

238 Group ENDF/B-V		218 Group ENDF/B-IV	
Group	Upper Energy Bound (eV)	Group	Upper Energy Bound (eV)
62	3.7400e+03	55	3.7400e+03
63	3.0000e+03	56	3.0000e+03
64	2.5800e+03	57	2.5800e+03
65	2.2900e+03	58	2.2900e+03
66	2.2000e+03	59	2.2000e+03
67	1.8000e+03	60	1.8000e+03
68	1.5500e+03	61	1.5500e+03
69	1.5000e+03	62	1.5000e+03
70	1.1500e+03	63	1.1500e+03
71	9.5000e+02	64	9.5000e+02
72	6.8300e+02	65	6.8300e+02
73	6.7000e+02	66	6.7000e+02
74	5.5000e+02	67	5.5000e+02
75	3.0500e+02	68	3.0500e+02
76	2.8500e+02	69	2.8500e+02
77	2.4000e+02	70	2.4000e+02
78	2.1000e+02	71	2.1000e+02
79	2.0750e+02	72	2.0750e+02
80	1.9250e+02	73	1.9250e+02
81	1.8600e+02	74	1.8600e+02
82	1.2200e+02	75	1.2200e+02
83	1.1900e+02	76	1.1900e+02
84	1.1500e+02	77	1.1500e+02
85	1.0800e+02	78	1.0800e+02
86	1.0000e+02	79	1.0000e+02
87	9.0000e+01	80	9.0000e+01
88	8.2000e+01	81	8.2000e+01
89	8.0000e+01	82	8.0000e+01
90	7.6000e+01	83	7.6000e+01
91	7.2000e+01	84	7.2000e+01
92	6.7500e+01	85	6.7500e+01

238 Group ENDF/B-V		218 Group ENDF/B-IV	
Group	Upper Energy Bound (eV)	Group	Upper Energy Bound (eV)
93	6.5000e+01	86	6.5000e+01
94	6.1000e+01	87	6.1000e+01
95	5.9000e+01	88	5.9000e+01
96	5.3400e+01	89	5.3400e+01
97	5.2000e+01	90	5.2000e+01
98	5.0600e+01	91	5.0600e+01
99	4.9200e+01	92	4.9200e+01
100	4.8300e+01	93	4.8300e+01
101	4.7000e+01	94	4.7000e+01
102	4.5200e+01	95	4.5200e+01
103	4.4000e+01	96	4.4000e+01
104	4.2400e+01	97	4.2400e+01
105	4.1000e+01	98	4.1000e+01
106	3.9600e+01	99	3.9600e+01
107	3.9100e+01	100	3.9100e+01
108	3.8000e+01	101	3.8000e+01
109	3.7000e+01	102	3.7000e+01
110	3.5500e+01	103	3.5500e+01
111	3.4600e+01	104	3.4600e+01
112	3.3750e+01	105	3.3750e+01
113	3.3250e+01	106	3.3250e+01
114	3.1750e+01	107	3.1750e+01
115	3.1250e+01	108	3.1250e+01
116	3.0000e+01	109	3.0000e+01
117	2.7500e+01	110	2.7500e+01
118	2.5000e+01	111	2.5000e+01
119	2.2500e+01	112	2.2500e+01
120	2.1000e+01	113	2.1000e+01
121	2.0000e+01	114	2.0000e+01
122	1.9000e+01	115	1.9000e+01
123	1.8500e+01	116	1.8500e+01

238 Group ENDF/B-V		218 Group ENDF/B-IV	
Group	Upper Energy Bound (eV)	Group	Upper Energy Bound (eV)
124	1.7000e+01	117	1.7000e+01
125	1.6000e+01	118	1.6000e+01
126	1.5100e+01	119	1.5100e+01
127	1.4400e+01	120	1.4400e+01
128	1.3750e+01	121	1.3750e+01
129	1.2900e+01	122	1.2900e+01
130	1.1900e+01	123	1.1900e+01
131	1.1500e+01	124	1.1500e+01
132	1.0000e+01	125	1.0000e+01
133	9.1000e+00	126	9.1000e+00
134	8.1000e+00	127	8.1000e+00
135	7.1500e+00	128	7.1500e+00
136	7.0000e+00	129	7.0000e+00
137	6.7500e+00	130	6.7500e+00
138	6.5000e+00	131	6.5000e+00
139	6.2500e+00	132	6.2500e+00
140	6.0000e+00	133	6.0000e+00
141	5.4000e+00	134	5.4000e+00
142	5.0000e+00	135	5.0000e+00
143	4.7500e+00	136	4.7500e+00
144	4.0000e+00	137	4.0000e+00
145	3.7300e+00	138	3.7300e+00
146	3.5000e+00	139	3.5000e+00
147	3.1500e+00	140	3.1500e+00
148	3.0500e+00	First Thermal Group 141	3.0500e+00
First Thermal Group 149	3.0000e+00	142	3.0000e+00
150	2.9700e+00	143	2.9700e+00
151	2.8700e+00	144	2.8700e+00
152	2.7700e+00	145	2.7700e+00

238 Group ENDF/B-V		218 Group ENDF/B-IV	
Group	Upper Energy Bound (eV)	Group	Upper Energy Bound (eV)
153	2.6700e+00	146	2.6700e+00
154	2.5700e+00	147	2.5700e+00
155	2.4700e+00	148	2.4700e+00
156	2.3800e+00	149	2.3800e+00
157	2.3000e+00	150	2.3000e+00
158	2.2100e+00	151	2.2100e+00
159	2.1200e+00	152	2.1200e+00
160	2.0000e+00	153	2.0000e+00
161	1.9400e+00	154	1.9400e+00
162	1.8600e+00	155	1.8600e+00
163	1.7700e+00	156	1.7700e+00
164	1.6800e+00	157	1.6800e+00
165	1.5900e+00	158	1.5900e+00
166	1.5000e+00	159	1.5000e+00
167	1.4500e+00	160	1.4500e+00
168	1.4000e+00	161	1.4000e+00
169	1.3500e+00	162	1.3500e+00
170	1.3000e+00	163	1.3000e+00
171	1.2500e+00	164	1.2500e+00
172	1.2250e+00	165	1.2250e+00
173	1.2000e+00	166	1.2000e+00
174	1.1750e+00	167	1.1750e+00
175	1.1500e+00	168	1.1500e+00
176	1.1400e+00	169	1.1400e+00
177	1.1300e+00	170	1.1300e+00
178	1.1200e+00	171	1.1200e+00
179	1.1100e+00	172	1.1100e+00
180	1.1000e+00	173	1.1000e+00
181	1.0900e+00	174	1.0900e+00
182	1.0800e+00	175	1.0800e+00
183	1.0700e+00	176	1.0700e+00

238 Group ENDF/B-V		218 Group ENDF/B-IV	
Group	Upper Energy Bound (eV)	Group	Upper Energy Bound (eV)
184	1.0600e+00	177	1.0600e+00
185	1.0500e+00	178	1.0500e+00
186	1.0400e+00	179	1.0400e+00
187	1.0300e+00	180	1.0300e+00
188	1.0200e+00	181	1.0200e+00
189	1.0100e+00	182	1.0100e+00
190	1.0000e+00	183	1.0000e+00
191	9.7500e-01	184	9.7500e-01
192	9.5000e-01	185	9.5000e-01
193	9.2500e-01	186	9.2500e-01
194	9.0000e-01	187	9.0000e-01
195	8.5000e-01	188	8.5000e-01
196	8.0000e-01	189	8.0000e-01
197	7.5000e-01	190	7.5000e-01
198	7.0000e-01	191	7.0000e-01
199	6.5000e-01	192	6.5000e-01
200	6.2500e-01		
201	6.0000e-01	193	6.0000e-01
202	5.5000e-01	194	5.5000e-01
203	5.0000e-01	195	5.0000e-01
204	4.5000e-01	196	4.5000e-01
205	4.0000e-01	197	4.0000e-01
206	3.7500e-01	198	3.7500e-01
207	3.5000e-01	199	3.5000e-01
208	3.2500e-01	200	3.2500e-01
209	3.0000e-01	201	3.0000e-01
210	2.7500e-01	202	2.7500e-01
211	2.5000e-01	203	2.5000e-01
212	2.2500e-01	204	2.2500e-01
213	2.0000e-01	205	2.0000e-01
214	1.7500e-01	206	1.7500e-01

238 Group ENDF/B-V		218 Group ENDF/B-IV	
Group	Upper Energy Bound (eV)	Group	Upper Energy Bound (eV)
215	1.5000e-01	207	1.5000e-01
216	1.2500e-01	208	1.2500e-01
217	1.0000e-01	209	1.0000e-01
218	9.0000e-02	210	9.0000e-02
219	8.0000e-02	211	8.0000e-02
220	7.0000e-02	212	7.0000e-02
221	6.0000e-02	213	6.0000e-02
222	5.0000e-02	214	5.0000e-02
223	4.0000e-02	215	4.0000e-02
224	3.0000e-02	216	3.0000e-02
225	2.5300e-02	217	2.5300e-02
226	1.0000e-02	218	1.0000e-02
227	7.5000e-03		
228	5.0000e-03		
229	4.0000e-03		
230	3.0000e-03		
231	2.5000e-03		
232	2.0000e-03		
233	1.5000e-03		
234	1.2000e-03		
235	1.0000e-03		
236	7.5000e-04		
237	5.0000e-04		
238	1.0000e-04		
	1.0000e-05		1.0000e-05

Appendix B
Atomic Number Densities for Critical Experiments

Table B.1: Atomic Number Densities (Atoms/b*cm) for Paraffin Reflected Cylinders of Aqueous $\text{UO}_2(\text{NO}_3)_2$ Solution.

Exp #	U^{233}	U^{234}	U^{235}	U^{238}	H	O	N
4.2	8.698e-04	4.387e-06	8.738e-08	6.816e-06	5.828e-02	3.619e-02	1.762e-03
4.9	4.340e-04	2.189e-06	4.360e-08	3.400e-06	6.293e-02	3.498e-02	8.793e-04
4.11	9.848e-04	4.968e-06	9.893e-08	7.717e-06	5.663e-02	3.630e-02	1.995e-03
4.12	7.087e-04	3.575e-06	7.119e-08	5.553e-06	5.981e-02	3.565e-02	1.429e-03
4.13	5.122e-04	2.584e-06	5.146e-08	4.014e-06	6.146e-02	3.488e-02	1.038e-03
4.15	4.127e-04	2.082e-06	4.146e-08	3.234e-06	6.232e-02	3.450e-02	8.361e-04
4.16	3.293e-04	1.661e-06	3.308e-08	2.581e-06	6.356e-02	3.445e-02	6.672e-04
4.17	3.043e-04	1.535e-06	3.057e-08	2.385e-06	6.482e-02	3.488e-02	6.165e-04
4.19	2.584e-04	1.304e-06	2.596e-08	2.025e-06	6.383e-02	3.400e-02	5.235e-04
4.20	2.159e-04	1.089e-06	2.169e-08	1.692e-06	6.412e-02	3.381e-02	4.374e-04
4.21	1.831e-04	9.238e-07	1.940e-08	1.435e-06	6.518e-02	3.407e-02	3.710e-04
4.22	1.706e-04	8.607e-07	1.714e-08	1.337e-06	6.466e-02	3.371e-02	3.456e-04
4.23	1.658e-04	8.364e-07	1.666e-08	1.299e-06	6.533e-02	3.400e-02	3.359e-04
4.24	1.422e-04	7.173e-07	1.429e-08	1.114e-06	6.555e-02	3.393e-02	2.881e-04
4.27	1.141e-04	5.757e-07	1.146e-08	8.943e-07	6.641e-02	3.413e-02	2.312e-04
4.28	1.040e-04	5.245e-07	1.045e-08	8.148e-07	6.552e-02	3.360e-02	2.109e-04
4.29	8.730e-05	4.404e-07	8.771e-09	6.842e-07	6.609e-02	3.375e-02	1.769e-04

All number densities are in units of atoms/barn-cm.

Table B.2: Atomic Number Densities for Unreflected Spheres of Aqueous $\text{UO}_2(\text{NO}_3)_2$ Solution.

Exp #	^{233}U	^{234}U	^{235}U	^{238}U	N	H	B
6.1	4.33E-5	7.16E-7	1.8E-8	2.81E-7	1.18E-4	6.64E-2	0.0
6.2	4.51E-5	7.44E-7	1.8E-8	2.91E-7	1.22E-4	6.63E-2	1.33E-6
6.3	4.68E-5	7.72E-7	1.8E-8	3.01E-7	1.27E-4	6.63E-2	2.59E-6
6.4	4.85E-5	8.01E-7	2.1E-8	3.11E-7	1.32E-4	6.63E-2	3.83E-6
6.5	5.01E-5	8.27E-7	2.1E-8	3.27E-7	1.36E-4	6.63E-2	5.08E-6
6.6	3.35E-5	5.25E-7	1.0E-8	2.56E-7	7.53E-5	6.65E-2	0.0

Table B.3: Atomic Number Densities for Unreflected Aqueous $\text{UO}_2(\text{NO}_3)_2$ Experiments.

Exp #	^{233}U	^{234}U	^{235}U	^{238}U	N	H	O
7.2	8.39e-4	9.0e-6	2.56e-7	1.17e-5	1.72e-3	6.13e-2	3.75e-2
7.3	5.14e-4	5.52e-6	1.37e-7	7.31e-6	1.3e-3	6.14e-2	3.55e-2
7.4	3.30e-4	3.54e-6	1.01e-7	4.61e-6	6.77e-4	6.44e-2	3.49e-2

Table B.4: Atomic Number Densities for Cubic Arrays of Cylindrical Units of Aqueous $\text{UO}_2(\text{NO}_3)_2$ Solution.

Exp #	^{233}U	^{234}U	^{235}U	^{236}U	^{238}U	H	O	N
8.10	8.40e-4	8.98e-6	2.14e-7	8.50e-9	1.17e-5	6.13e-2	3.76e-2	1.72e-3
8.11	8.40e-4	8.98e-6	2.14e-7	8.50e-9	1.17e-5	6.13e-2	3.76e-2	1.72e-3
8.13	5.14e-4	5.50e-6	1.31e-7	5.21e-9	7.15e-6	6.12e-2	3.48e-2	1.05e-3
8.16	5.14e-4	5.50e-6	1.31e-7	5.21e-9	7.15e-6	6.12e-2	3.48e-2	1.05e-3

Table B.5: Atomic Number Densities for Paraffin Reflected Cylinders of Aqueous UO_2F_2 Solution.

Exp #	^{233}U	^{234}U	^{235}U	^{238}U	H	O	F
4.35	8.576-4	4.672-6	3.446-7	6.125-6	6.355-2	3.351-2	1.738-3
4.39	1.345-3	7.328-6	5.405-7	9.606-6	6.174-2	3.360-2	2.725-3
4.43	4.283-4	2.333-6	1.721-7	3.059-6	6.596-2	3.385-2	8.677-4
4.49	8.580-5	4.676-7	3.449-8	6.130-7	6.650-2	3.342-2	1.738-4

Table B.6: Atomic Number Densities for Spheres of Aqueous UO_2F_2 Solution.

Exp #	^{233}U	^{234}U	^{235}U	^{238}U	H	O	F
4.52	1.5852e-4	8.6355e-7	6.3695e-8	1.1321e-6	6.642e-2	3.3531e-2	3.321e-4
4.54	1.0023e-4	5.4601e-7	4.0273e-8	7.1578e-7	6.6452e-2	3.3429e-2	2.030e-4
4.55	1.7352e-4	9.4527e-7	6.9722e-8	1.2392e-6	6.6111e-2	3.3407e-2	3.515e-4

Table B.7: Atomic Number Densities for Water Reflected Spheres of Aqueous UO_2F_2 Solution.

Exp #	^{233}U	^{234}U	^{235}U	^{238}U	H	O	F
5.2	1.5928E-4	8.6772e-7	6.4002e-8	1.1375e-6	6.6037e-2	3.3341e-2	3.227e-4
5.3	1.6273E-4	8.8648e-7	6.5386e-8	1.1621e-6	6.6976e-2	3.3818e-2	3.297e-4
5.4	1.6561E-4	9.0217e-7	6.6543e-8	1.1827e-6	6.4522e-2	3.2597e-2	3.355e-4
5.7	1.0207e-4	5.5603e-7	4.1012e-8	7.2891e-7	6.5641e-2	3.2027e-2	2.068e-4
5.8	1.0643e-4	5.7978e-7	4.2764e-8	7.6005e-7	6.4156e-2	3.2294e-2	2.156e-4

Table B.8: Atomic Number Densities for ^{233}U Metal Spheres.

Exp #	^{233}U	^{234}U	^{235}U	^{238}U
7.1	4.670e-2	5.925e-4	1.416e-5	2.843e-4
8.1	4.6695e-2	5.9235e-4	1.8878e-5	2.8397e-4
8.2	4.6695e-2	5.9235e-4	1.8878e-5	2.8397e-4
8.3	4.7301e-2	5.2758e-4	0.0	3.3002e-4
8.4	4.7250e-2	5.2701e-4	0.0	3.2966e-4
8.5	4.7301e-2	5.2758e-4	0.0	3.3002e-4
8.6	4.7250e-2	5.2701e-4	0.0	3.2966e-4
8.7	4.7301e-2	5.2758e-4	0.0	3.3002e-4
8.8	4.7250e-2	5.2701e-4	0.0	3.2966e-4

Table B.9: Atomic Number Densities for FTR Concrete Reflected Experiments (Type 3.2 Fuel).

Isotope	Atomic Number Density
^{238}Pu	7.3856e-6
^{239}Pu	4.2019e-3
^{240}Pu	5.6059e-4
^{241}Pu	1.2178e-4
^{242}Pu	1.6972e-5
^{235}U	1.2222e-4
^{238}U	1.6876e-2
O	4.371e-2

Appendix C

KENO V.a Results for Critical Experiments Using the Hansen- Roach Neutron Cross Section Library

Table C.1: Calculated Effective Multiplication Factor for Paraffin Reflected Cylinders of Aqueous $\text{UO}_2(\text{NO}_3)_2$.

Exp. #	k_{eff} Hansen-Roach	Avg. Fission Group (AFG)
4.2	0.9780 ± 0.0042	14.09
4.9	0.9842 ± 0.0041	14.84
4.11	0.9919 ± 0.0041	13.97
4.12	0.9951 ± 0.0048	14.38
4.13	0.9959 ± 0.0043	14.70
4.15	0.9939 ± 0.0047	14.88
4.16	1.0019 ± 0.0045	15.05
4.17	1.0048 ± 0.0042	15.11
4.19	0.9959 ± 0.0040	15.20
4.20	0.9807 ± 0.0036	15.29
4.21	0.9862 ± 0.0036	15.37
4.22	0.9886 ± 0.0040	15.39
4.23	0.9933 ± 0.0035	15.41
4.24	1.0076 ± 0.0039	15.46
4.27	1.0062 ± 0.0033	15.54
4.28	1.0011 ± 0.0037	15.56
4.29	0.9968 ± 0.0031	15.61

Table C.2: Calculated Effective Multiplication Factors for Unreflected Spheres of Aqueous $\text{UO}_2(\text{NO}_3)_2$.

Exp. #	k_{eff} Hansen-Roach	Avg. Fission Group (AFG)
6.1	1.0071 ± 0.0022	15.73
6.2	1.0015 ± 0.0025	15.71
6.3	1.0030 ± 0.0025	15.71
6.4	1.0017 ± 0.0024	15.69
6.5	0.9977 ± 0.0026	15.68
6.6	1.0054 ± 0.0017	15.77

Table C.3: Calculated Effective Multiplication Factor for Unreflected Aqueous $\text{UO}_2(\text{NO}_3)_2$ Experiments.

Exp. #	k_{eff} Hansen-Roach	Avg. Fission Group (AFG)
7.2	0.9961 ± 0.0049	13.61
7.3	0.9667 ± 0.0049	14.29
7.4	1.0030 ± 0.0046	14.81

Table C.4: Calculated Effective Multiplication Factors for Cubic Arrays of Cylindrical Units of Aqueous $\text{UO}_2(\text{NO}_3)_2$ Solution (Ref 8).

Exp. #	k_{eff} Hansen-Roach	Avg. Fission Group (AFG)
8.10	1.0259 ± 0.0042	13.61
8.11	1.0248 ± 0.0044	14.18
8.13	0.9884 ± 0.0041	14.27
8.16	1.0066 ± 0.0043	14.68

Table C.5: Calculated Effective Multiplication Factor for Cylinders and Spheres of Aqueous UO_2F_2 .

Exp. #	k_{eff} Hansen-Roach	Avg. Fission Group (AFG)
4.35	0.9695 ± 0.0038	14.18
4.39	1.0014 ± 0.0041	13.66
4.43	1.0080 ± 0.0042	14.90
4.49	1.0001 ± 0.0028	15.61
4.52	1.0081 ± 0.0038	15.43
4.54	1.0128 ± 0.0033	15.57
4.55	0.9919 ± 0.0040	15.28

Table C.6: Calculated Effective Multiplication Factor for Water Reflected Spheres of Aqueous UO_2F_2 .

Exp. #	k_{eff} Hansen-Roach	Avg. Fission Group (AFG)
5.2	1.0001 ± 0.0036	15.42
5.3	1.0088 ± 0.0035	15.42
5.4	1.0033 ± 0.0038	15.40
5.7	1.0133 ± 0.0036	15.57
5.8	1.0219 ± 0.0038	15.55

Table C.7: Calculated Effective Multiplication Factor for ^{233}U Metal Spheres.

Exp. #	k_{eff} Hansen-Roach	Avg. Fission Group (AFG)
7.1	1.0081 ± 0.0048	3.10
8.1	1.0010 ± 0.0037	3.09
8.2	0.9880 ± 0.0033	3.11
8.3	1.0043 ± 0.0039	3.13
8.4	1.0037 ± 0.0038	3.13
8.5	0.9625 ± 0.0041	3.35
8.6	0.9763 ± 0.0032	3.25
8.7	1.0218 ± 0.0040	3.42
8.8	1.0129 ± 0.0037	3.27

Table C.8: Calculated K_{eff} 's for Concrete Reflected FTR Experiments.

Exp. #	k_{eff} Hansen-Roach	Avg. Fission Group (AFG)
010	1.0338 ± 0.0033	14.65
007	1.0483 ± 0.0035	13.82
012	1.0298 ± 0.0031	15.24
011	1.0124 ± 0.0035	15.0

Appendix D

**KENO V.a Model of Cell 4 and Inter-cell 4&5 and 3&4 Storage
Wells Located in Building 3019 at ORNL
filename: 3019faci.input**

#csas25

Building 3019 cell 4 and inter-cell 4&5 and 3&4 wells: file=3019faci.input

238group infhommedium

u-233 1 0 1.9247e-3 end
u-235 1 0 3.8948e-5 end
o 1 0 6.8726e-3 end
h 1 0 5.8908e-3 end
u-233 2 0 1.8959e-3 end
u-235 2 0 3.8367e-5 end
u-233 3 0 1.6025e-4 end
u-234 3 0 2.3026e-5 end
u-235 3 0 1.2532e-3 end
u-236 3 0 9.1334e-5 end
u-238 3 0 1.1001e-4 end
o 3 0 6.8241e-3 end
h 3 0 4.9134e-3 end
u-233 4 0 2.1779e-3 end
u-235 4 0 4.4073e-5 end
o 4 0 7.777e-3 end
h 4 0 6.666e-3 end
u-233 5 0 1.5301e-3 end
u-235 5 0 3.0964e-5 end
o 5 0 5.4639e-3 end
h 5 0 4.6833e-3 end
u-233 6 0 1.085e-3 end
u-235 6 0 2.1957e-5 end
f 6 0 4.4278e-3 end
h 6 0 3.3209e-3 end
o 6 0 1.6604e-3 end
u-233 7 0 2.5876e-2 end
u-235 7 0 5.2359e-4 end
u-233 8 0 1.2661e-4 end
u-234 8 0 1.8193e-5 end
u-235 8 0 9.9018e-4 end
u-236 8 0 7.2164e-5 end
u-238 8 0 8.6919e-5 end
o 8 0 5.3921e-3 end
h 8 0 3.8823e-3 end
carbonsteel 9 1 end
pb 10 1 end
h 11 0 7.25655e-3 end
c 11 0 1.05173e-2 end
o 11 0 4.49751e-2 end
na 11 0 1.28310e-5 end

```

mg      11 0 5.09498e-3 end
al      11 0 4.16956e-4 end
si      11 0 2.67500e-3 end
s       11 0 9.61505e-5 end
k       11 0 1.27795e-5 end
ca      11 0 8.7644e-3 end
fe      11 0 1.39779e-4 end
c       12 0 3.82155e-4 end
o       12 0 2.75330e-2 end
na      12 0 9.58077e-5 end
mg      12 0 1.18155e-4 end
al      12 0 7.68645e-4 end
si      12 0 1.24659e-2 end
s       12 0 2.52827e-5 end
k       12 0 2.15730e-4 end
ca      12 0 1.71935e-3 end
fe      12 0 2.29130e-4 end
end comp
cell4 parameters
read parm tme=200 tba=1 gen=300 npg=600 run=yes plt=yes nub=yes end parm
read geom
unit 1
com='lead plug for type a well'
cylinder 10 1 5.682 1035.718 974.758
cuboid   11 1 2p14.023117 2p16.1925 1035.718 974.758
cuboid    0 1 2p14.023117 2p16.1925 1035.718 974.758
unit 2
com='lead plug for type b well'
cylinder 10 1 6.35 1035.718 974.758
cuboid   11 1 2p14.023117 2p16.1925 1035.718 974.758
cuboid    0 1 2p14.023117 2p16.1925 1035.718 974.758
unit 3
com='type a well: uo2 matl 1'
cylinder 1 1 5.08 974.758 0
cylinder 9 1 5.682 974.758 -.602
cuboid   11 1 2p14.023117 2p16.1925 974.758 -.602
cuboid    0 1 2p14.023117 2p16.1925 974.758 -.602
unit 4
com='type a well 1-e: jezebel matl 2'
cylinder 2 1 5.08 974.758 0
cylinder 9 1 5.682 974.758 -.602
cuboid   11 1 2p14.023117 2p16.1925 974.758 -.602
cuboid    0 1 2p14.023117 2p16.1925 974.758 -.602
unit 5

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com='type a well: ceusp u3o8 matl 3'
cylinder 3 1 5.08 974.758 0
cylinder 9 1 5.682 974.758 -.602
cuboid 11 1 2p14.023117 2p16.1925 974.758 -.602
cuboid 0 1 2p14.023117 2p16.1925 974.758 -.602
unit 6
com='type a well: uf4 matl 4 not used'
cylinder 4 1 5.08 974.758 0
cylinder 9 1 5.682 974.758 -.602
cuboid 11 1 2p14.023117 2p16.1925 974.758 -.602
cuboid 0 1 2p14.023117 2p16.1925 974.758 -.602
unit 7
com='type b well: uo2 matl 5'
cylinder 5 1 5.715 974.758 0.033
cylinder 9 1 6.35 974.758 -.602
cuboid 11 1 2p14.023117 2p16.1925 974.758 -.602
cuboid 0 1 2p14.023117 2p16.1925 974.758 -.602
unit 8
com='type b well: uf4 matl 6'
cylinder 6 1 5.715 974.758 0.033
cylinder 9 1 6.35 974.758 -.602
cuboid 11 1 2p14.023117 2p16.1925 974.758 -.602
cuboid 0 1 2p14.023117 2p16.1925 974.758 -.602
unit 9
com='type b well: ceusp u3o8 matl 8'
cylinder 8 1 5.715 974.758 0.033
cylinder 9 1 6.35 974.758 -.602
cuboid 11 1 2p14.023117 2p16.1925 974.758 -.602
cuboid 0 1 2p14.023117 2p16.1925 974.758 -.602
unit 10
com='empty type a well'
cylinder 0 1 5.08 974.758 0
cylinder 9 1 5.682 974.758 -.602
cuboid 11 1 2p14.023117 2p16.1925 974.758 -.602
cuboid 0 1 2p14.023117 2p16.1925 974.758 -.602
unit 11
com='empty type b well'
cylinder 0 1 5.715 974.758 0.033
cylinder 9 1 6.35 974.758 -.602
cuboid 11 1 2p14.023117 2p16.1925 974.758 -.602
cuboid 0 1 2p14.023117 2p16.1925 974.758 -.602
unit 12
com='array of lead plugs for type a wells 1, 3, & 5'
array 1 -14.023117 -129.54 974.758

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```

unit 13
com='array of lead plugs for type a wells 2, 4, & 6'
array      2  -14.023117 -113.3475 974.758
cuboid     11 1  2p14.023117 2p129.54 1035.718 974.758
cuboid      0 1  2p14.023117 2p129.54 1035.718 974.758
unit 14
com='array of lead plugs for wells 7 & 9'
array      3  -14.023117 -129.54 974.758
unit 15
com='array of lead plugs for wells 8'
array      4  -14.023117 -113.3475 974.758
cuboid     11 1  2p14.023117 2p129.54 1035.718 974.758
cuboid      0 1  2p14.023117 2p129.54 1035.718 974.758
unit 16
com='wells 1a-1r'
array      5  -14.023117 -129.54 -.602
unit 17
com='wells 1a-1r with lead plugs'
array      6  -14.023117 -129.54 -.602
unit 18
com='wells 2b-2p'
array      7  -14.023117 -113.3475 -.602
cuboid     11 1  2p14.023117 2p129.54 974.758 -.602
cuboid      0 1  2p14.023117 2p129.54 974.758 -.602
unit 19
com='wells 2b-2p with lead plugs'
array      8  -14.023117 -129.54 -.602
unit 20
com='wells 3a-3r'
array      9  -14.023117 -129.54 -.602
unit 21
com='wells 3a-3r with lead plugs'
array     10  -14.023117 -129.54 -.602
unit 22
com='wells 4b-4p'
array     11  -14.023117 -113.3475 -.602
cuboid     11 1  2p14.023117 2p129.54 974.758 -.602
cuboid      0 1  2p14.023117 2p129.54 974.758 -.602
unit 23
com='wells 4b-4p with lead plugs'
array     12  -14.023117 -129.54 -.602
unit 24
com='wells 5a-5r'
array     13  -14.023117 -129.54 -.602

```

```

unit 25
com='wells 5a-5r with lead plugs'
array    14    -14.023117 -129.54 -.602
unit 26
com='wells 6b-6p'
array    15    -14.023117 -113.3475 -.602
cuboid   11 1   2p14.023117 2p129.54 974.758 -.602
cuboid   0 1   2p14.023117 2p129.54 974.758 -.602
unit 27
com='wells 6b-6p with lead plugs'
array    16    -14.023117 -129.54 -.602
unit 28
com='wells 7a-7r'
array    17    -14.023117 -129.54 -.602
unit 29
com='wells 7a-7r with lead plugs'
array    18    -14.023117 -129.54 -.602
unit 30
com='wells 8b-8p'
array    19    -14.023117 -113.3475 -.602
cuboid   11 1   2p14.023117 2p129.54 974.758 -.602
cuboid   0 1   2p14.023117 2p129.54 974.758 -.602
unit 31
com='wells 8b-8p with lead plugs'
array    20    -14.023117 -129.54 -0.602
unit 32
com='wells 9a-9r'
array    21    -14.023117 -129.54 -.602
unit 33
com='wells 9a-9r with lead plugs'
array    22    -14.023117 -129.54 -.602
unit 34
com='cell 4 storage wells'
array    23    -126.20805 -129.54 -.602
cuboid   11 1   2p126.20805 129.54 -251.46 1035.718 -122.522
cuboid   0 1   2p126.20805 129.54 -251.46 1035.718 -122.522 unit 35
com='void section of cell 4 room on north side of storage wells'
cuboid   0 1   2p126.20805 480.06 129.54 822.358 -.602
cuboid   12 1   2p126.20805 632.46 129.54 944.278 -122.522
cuboid   0 1   2p126.20805 632.46 129.54 944.278 -122.522
unit 36
com='ceiling of cell 4 & penthouse region on north side of storage wells'
cuboid   0 1   2p126.20805 480.06 129.54 1035.718 944.278
cuboid   12 1   2p126.20805 632.46 129.54 1035.718 944.278

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```

cuboid    0 1 2p126.20805 632.46 129.54 1035.718 944.278
unit 37
com='void section & ceiling on northside of wells'
array    24  -126.20805 129.54 -122.522
unit 38
com='cell 4 wells with void region & floor of penthouse on north side'
array    25  -126.20805 -251.46 -122.522
unit 39
com='east section of cell 4 room'
cuboid    0 1 483.39195 126.20805 480.06 -129.54 822.358 -.602
cuboid    12 1 483.39195 126.20805 632.46 -251.46 944.278 -122.522
cuboid    0 1 483.39195 126.20805 632.46 -251.46 1035.718 -122.522
unit 40
com='cell 4 of building 3019 with lower 3ft section of penthouse region'
array    26  -126.20805 -251.46 -122.522
unit 41
com='lead plug for type a well--intercell wells 4&5, 3&4'
cylinder  10 1 5.682 20.32 0
cuboid    12 1 2p76.2 2p15.24 20.32 0
cuboid    0 1 2p76.2 2p15.24 20.32 0
unit 42
com='array of lead plugs for intercell wells 4&5'
array    27  -76.2 -121.92 392.7475
cuboid    12 1 2p76.2 754.38 -129.54 413.0675 392.7475
cuboid    0 1 2p76.2 754.38 -129.54 413.0675 392.7475
unit 43
com='array of lead plugs for intercell wells 3&4'
array    28  -76.2 -137.16 254.0
cuboid    12 1 2p76.2 746.76 -137.16 274.32 254.0
cuboid    0 1 2p76.2 746.76 -137.16 274.32 254.0
unit 44
com='4&5 type a well: uo2 matl 1 without lead plug'
cylinder  4 1 5.08 392.7475 0
cylinder  9 1 5.682 392.7475 -.602
cuboid    12 1 2p76.2 2p15.24 392.7475 -653.7325
cuboid    0 1 2p76.2 2p15.24 392.7475 -653.7325
unit 45
com='metal hemisphere for intercell 4&5 well m'
hemisphere 7 1 5.08
cylinder  9 1 5.682 5.08 0
cuboid    0 1 4p5.682 5.08 0
unit 46
com='150.65375 cm section of uo2 matl 1'
cylinder  4 1 5.08 150.65375 0

```

```

cylinder  9 1  5.682 150.65375 0
cuboid    0 1  4p5.682 150.65375 0
unit 47
com='38.1 cm section of uo2 matl 1'
cylinder  4 1  5.08 38.1 0
cylinder  9 1  5.682 38.1 0
cuboid    0 1  4p5.682 38.1 0
unit 48
com='4&5 well m with 3 metal hemispheres'
array     29  -5.682 -5.682 0
cuboid    12 1  2p76.2 2p15.24 392.7475 -653.7325
cuboid    0 1  2p76.2 2p15.24 392.7475 -653.7325
unit 49
com='empty intercell 4&5 type a well'
cylinder  0 1  5.08 392.7475 0
cylinder  9 1  5.682 392.7475 -.602
cuboid    12 1  2p76.2 2p15.24 392.7475 -653.7325
cuboid    0 1  2p76.2 2p15.24 392.7475 -653.7325
unit 50
com='intercell 4&5 wells u-l without lead plugs'
array     30  -76.2 -121.92 -653.7325
cuboid    12 1  2p76.2 754.38 -129.54 392.7475 -653.7325
cuboid    0 1  2p76.2 754.38 -129.54 392.7475 -653.7325
unit 51
com='intercell 4&5 wells with lead plugs--west wall'
array     31  -76.2 -129.54 -653.7325
cuboid    0 1  2p76.2 754.38 -129.54 504.5075 -653.7325
unit 52
com='3&4 type a well: uo2 matl 1 without lead plug'
cylinder  4 1  5.08 254.0 0
cylinder  9 1  5.682 254.0 -.602
cuboid    12 1  2p76.2 2p15.24 254.0 -792.48
cuboid    0 1  2p76.2 2p15.24 254.0 -792.48
unit 53
com='intercell 3&4 wells j-a without lead plugs'
array     32  -76.2 -137.16 -792.48
cuboid    12 1  2p76.2 746.76 -137.16 254.0 -792.48
cuboid    0 1  2p76.2 746.76 -137.16 254.0 -792.48
unit 54
com='intercell 3&4 wells with lead plugs--east wall'
array     33  -76.2 -137.16 -792.48
cuboid    0 1  2p76.2 746.76 -137.16 365.76 -792.48
global unit 55
com='cell 4 wells with all walls'

```

```

array    34    -278.60805 -251.46 -122.52
end geom
read array
ara=1 nux=1 nuy=8 nuz=1
com='array of lead plugs for wells 1,3,& 5'
fill 8r1 end fill
ara=2 nux=1 nuy=7 nuz=1
com='lead plugs for wells 2,4,& 6'
fill 7r1 end fill
ara=3 nux=1 nuy=8 nuz=1
com='lead plugs for wells 7 & 9'
fill 8r2 end fill
ara=4 nux=1 nuy=7 nuz=1
com='lead plugs for well 8'
fill 7r2 end fill
ara=5 nux=1 nuy=8 nuz=1
com='wells 1a-1r'
fill 2r3 4 3 4r10 end fill
ara=6 nux=1 nuy=1 nuz=2
com='wells 1a-1r with lead plugs'
fill 16 12 end fill
ara=7 nux=1 nuy=7 nuz=1
com='2d-2p wells'
fill 10 6r5 end fill
ara=8 nux=1 nuy=1 nuz=2
com='wells 2b-2p with plugs'
fill 18 13 end fill
ara=9 nux=1 nuy=8 nuz=1
com='wells 3a-3r'
fill 10 7r5 end fill
ara=10 nux=1 nuy=1 nuz=2
com='wells 3a-3r with lead plugs'
fill 20 12 end fill
ara=11 nux=1 nuy=7 nuz=1
com='wells 4b-4p'
fill 10 6r5 end fill
ara=12 nux=1 nuy=1 nuz=2
com='wells 4b-4p with lead plugs'
fill 22 13 end fill
ara=13 nux=1 nuy=8 nuz=1
com='wells 5a-5r'
fill 10 7r5 end fill
ara=14 nux=1 nuy=1 nuz=2
com='wells 5a-5r with lead plugs'

```

```

fill 24 12 end fill
ara=15 nux=1 nuy=7 nuz=1
com='wells 6b-6p'
fill 10 6r5 end fill
ara=16 nux=1 nuy=1 nuz=2
com='wells 6b-6p with lead plugs'
fill 26 13 end fill
ara=17 nux=1 nuy=8 nuz=1
com='wells 7a-7r'
fill 5r7 9 8 9 end fill
ara=18 nux=1 nuy=1 nuz=2
com='wells 7a-7r with lead plugs'
fill 28 14 end fill
ara=19 nux=1 nuy=7 nuz=1
com='wells 8b-8p'
fill 11 8 11 8 2r11 7 end fill
ara=20 nux=1 nuy=1 nuz=2
com='wells 8b-8p with lead plugs'
fill 30 15 end fill
ara=21 nux=1 nuy=8 nuz=1
com='wells 9a-9r'
fill 7 3r11 4r7 end fill
ara=22 nux=1 nuy=1 nuz=2
com='wells 9a-9r with lead plugs'
fill 32 14 end fill
ara=23 nux=9 nuy=1 nuz=1
com='cell 4 storage wells'
fill 17 19 21 23 25 27 29 31 33 end fill
ara=24 nux=1 nuy=1 nuz=2
com='void & ceiling on north side of wells'
fill 35 36 end fill
ara=25 nux=1 nuy=2 nuz=1
com='cell 4 wells with north side of room'
fill 34 37 end fill
ara=26 nux=2 nuy=1 nuz=1
com='cell 4 of building 3019'
fill 38 39 end fill
ara=27 nux=1 nuy=8 nuz=1
com='lead plugs for wells 4&5'
fill 8r41 end fill
ara=28 nux=1 nuy=9 nuz=1
com='lead plugs for wells 3&4'
fill 9r41 end fill
ara=29 nux=1 nuy=1 nuz=7

```

```
com='4&5 well m with 3 metal hemispheres'
fill 46 45 47 45 47 45 46 end fill
ara=30 nux=1 nuy=8 nuz=1
com='4&5 wells without plugs'
fill 44 2r49 44 49 44 48 44 end fill
ara=31 nux=1 nuy=1 nuz=2
com='4&5 wells with lead plugs'
fill 50 42 end fill
ara=32 nux=1 nuy=9 nuz=1
com='3&4 wells j-a without lead plugs'
fill 9r52 end fill
ara=33 nux=1 nuy=1 nuz=2
com='3&4 wells j-a with plugs'
fill 53 43 end fill
ara=34 nux=3 nuy=1 nuz=1
com='cell 4 wells with all walls'
fill 51 40 54 end fill
end array
end data
end
```

Appendix E

MSR Fuel and Flush Salt Atomic Number Densities

Table E.1: MSR Fuel Salt Atomic Number Densities.

Isotope	Atom Density (atoms/barn*cm)
^{233}U	4.1370e-5
^{234}U	3.6708e-6
^{235}U	1.2503e-6
^{236}U	5.1104e-8
^{238}U	2.8672e-6
^{239}Pu	8.0004e-7
^{240}Pu	8.4152e-8
^{241}Pu	3.0810e-9
F	5.4487e-2
^7Li	2.4046e-2
Be	1.1251e-2
Zr	1.9348e-3

Table E.2: MSR Flush Salt Atomic Number Densities.

Isotope	Atom Density (atoms/barn*cm)
^{233}U	2.9603e-7
^{234}U	2.7114e-8
^{235}U	1.2790e-7
^{236}U	1.8019e-9
^{238}U	2.7014e-7
^{239}Pu	9.5113e-8
^{240}Pu	4.7788e-9
^{241}Pu	4.3807e-10
^{242}Pu	3.9656e-11
F	6.0358e-2
^7Li	2.9727e-2
Be	1.5314e-2

Appendix F

**KENO V.a Model of the Flush Tank and Two Fuel Drain Tanks
Located in the MSR Fuel Drain Tank Cell**

filename: msre.input

```

#csas25
msr fdt cell file=msre.input
44group          infhommedium
u-233  1 0 4.137e-5 end
u-234  1 0 3.6708e-6 end
u-235  1 0 1.2503e-6 end
u-236  1 0 5.1104e-8 end
u-238  1 0 2.8672e-6 end
pu-239 1 0 8.0004e-7 end
pu-240 1 0 8.4152e-8 end
pu-241 1 0 3.081e-9 end
f      1 0 5.4487e-2 end
li-7   1 0 2.4046e-2 end
be     1 0 1.1251e-2 end
zr     1 0 1.9348e-3 end
ni     2 0 6.45e-3 end
mo     2 0 9.11e-3 end
cr     2 0 7.13e-3 end
fe     2 0 4.74e-3 end
arbm-c 5.0018 1 0 0 0 6012 100 3 0.0008 293 end
arbm-mn 5.0018 1 0 0 1 25055 100 3 0.02 293 end
arbm-p 5.0018 1 0 0 0 15031 100 3 0.00045 293 end
arbm-s 5.0018 1 0 0 0 16000 100 3 0.00030 293 end
arbm-si 5.0018 1 0 0 0 14000 100 3 0.01 293 end
arbm-cr 5.0018 1 0 0 1 24000 100 3 0.19 293 end
arbm-ni 5.0018 1 0 0 1 28000 100 3 0.13 293 end
arbm-fe 5.0018 1 0 0 1 26000 100 3 .64845 293 end
arbm-sio2 0.072 2 0 1 0 14000 1 8016 2 4 .35 293 end
arbm-al2o3 0.072 2 0 1 0 13027 2 8016 3 4 .15 293 end
arbm-mgo 0.072 2 0 1 0 12000 1 8016 1 4 0.50 293 end
arbm-al2o3 3.247 2 0 1 0 13027 2 8016 3 5 0.5205 293 end
arbm-sio2 3.247 2 0 1 0 14000 1 8016 2 5 .4795 293 end
ss316  6 1 293 end
u-233  7 0 2.9603-7 end
u-234  7 0 2.7114e-8 end
u-235  7 0 1.279e-7 end
u-236  7 0 1.8019e-9 end
u-238  7 0 2.7014e-7 end
pu-239 7 0 9.5113e-8 end
pu-240 7 0 4.7788e-9 end
pu-241 7 0 4.3807-10 end
pu-242 7 0 3.9656e-11 end
f      7 0 6.0358e-2 end
li-7   7 0 2.9727e-2 end

```

```

be      7 0 1.5314e-2 end
arbm-c 8.02 1 0 0 0 6012 100 8 0.0008 293 end
arbm-mn 8.02 1 0 0 1 25055 100 8 0.02 293 end
arbm-p 8.02 1 0 0 0 15031 100 8 0.00045 293 end
arbm-s 8.02 1 0 0 0 16000 100 8 0.00030 293 end
arbm-si 8.02 1 0 0 0 14000 100 8 0.01 293 end
arbm-cr 8.02 1 0 0 1 24000 100 8 0.19 293 end
arbm-ni 8.02 1 0 0 1 28000 100 8 0.13 293 end
arbm-fe 8.02 1 0 0 1 26000 100 8 .64845 293 end
orconcrete 9 1 293 end
carbonsteel 10 1 293 end
end comp
msre.input 44 group
read parm tme=200 tba=1 gen=300 npg=300 run=yes plt=yes nub=yes end parm
read geom
unit 1
com='salt part of fdt-1 cylinder--no thimbles'
cylinder 1 1 62.23 8.09625 0
cylinder 2 1 63.5 8.09625 0
cylinder 0 1 76.835 8.09625 0
cylinder 3 1 79.375 8.09625 0
cylinder 8 1 79.6925 8.09625 0
cylinder 4 1 91.1225 8.09625 0
cylinder 10 1 91.44 8.09625 0
cuboid 0 1 4p91.44 8.09625 0
unit 2
com='thimble in salt on x-axis of fdt-1 region 3'
cylinder 0 1 0.52832 64.75963 0
cylinder 2 1 0.635 64.75963 0
cylinder 0 1 1.33223 64.75963 0
cylinder 2 1 1.67005 64.75963 0
cylinder 0 1 2.0447 64.75963 0
cylinder 2 1 2.413 64.75963 0
cuboid 1 1 4p8.255 64.75963 0
cuboid 0 1 4p8.255 64.75963 0
unit 3
com='cuboid of salt on x-axis'
cuboid 1 1 2p22.225 2p8.255 64.75963 0
cuboid 0 1 2p22.225 2p8.255 64.75963 0
unit 4
com='thimble for a hole in region 3 of fdt-1'
cylinder 0 1 0.52832 64.75963 0
cylinder 2 1 0.635 64.75963 0
cylinder 0 1 1.33223 64.75963 0

```

```

cylinder 2 1 1.67005 64.75963 0
cylinder 0 1 2.0447 64.75963 0
cylinder 2 1 2.413 64.75963 0
unit 5
com='region 3 of fdt-1 tank including thimbles & salt'
array 1 -55.245 -8.255 0
cylinder 1 1 62.23 64.75963 0
hole 4 26.396454 -15.24 0
hole 4 15.24 -26.396454 0
hole 4 0 -30.48 0
hole 4 -15.24 -26.396454 0
hole 4 -26.396454 -15.24 0
hole 4 -26.396454 15.24 0
hole 4 -15.24 26.396454 0
hole 4 0 30.48 0
hole 4 15.24 26.396454 0
hole 4 26.396454 15.24 0
hole 4 44.690146 -14.520709 0
hole 4 38.015709 -27.620029 0
hole 4 27.6720029 -38.015709 0
hole 4 14.520709 -44.690146 0
hole 4 0 -46.99 0
hole 4 -14.520709 -44.690146 0
hole 4 -27.620029 -38.015709 0
hole 4 -38.015709 -27.620029 0
hole 4 -44.690146 -14.520709 0
hole 4 -44.690146 14.520709 0
hole 4 -38.015709 27.620029 0
hole 4 -27.620029 38.015709 0
hole 4 -14.520709 44.690146 0
hole 4 0 46.99 0
hole 4 14.520709 44.690146 0
hole 4 27.620029 38.015709 0
hole 4 38.015709 27.620029 0
hole 4 44.690146 14.520709 0
cylinder 2 1 63.5 64.75963 0
cylinder 0 1 76.835 64.75963 0
cylinder 3 1 79.375 64.75963 0
cylinder 8 1 79.6925 64.75963 0
cylinder 4 1 91.1225 64.75963 0
cylinder 10 1 91.44 64.75963 0
cuboid 0 1 4p91.44 64.75963 0
unit 6
com='cuboid of void on x-axis in region 4 of tank'

```

```

cuboid    0 1 2p22.225 2p8.255 82.08412 0
unit 7
com='thimble on x-axis of fdt-1 region 4'
cylinder  0 1 0.52832 82.08412 0
cylinder  2 1 0.635 82.08412 0
cylinder  0 1 1.33223 82.08412 0
cylinder  2 1 1.67005 82.08412 0
cylinder  0 1 2.0447 82.08412 0
cylinder  2 1 2.413 82.08412 0
cuboid    0 1 4p8.255 82.08412 0
unit 8
com='thimble for a hole in region 4 of fdt-1'
cylinder  0 1 0.52832 82.08412 0
cylinder  2 1 0.635 82.08412 0
cylinder  0 1 1.33223 82.08412 0
cylinder  2 1 1.67005 82.08412 0
cylinder  0 1 2.0447 82.08412 0
cylinder  2 1 2.413 82.08412 0
unit 9
com='region 4 of fdt-1 including thimbles and void'
array      2 -55.245 -8.255 0
cylinder  0 1 62.23 82.08412 0
hole       8 26.396454 -15.24 0
hole       8 15.24 -26.396454 0
hole       8 0 -30.48 0
hole       8 -15.24 -26.396454 0
hole       8 -26.396454 -15.24 0
hole       8 -26.396454 15.24 0
hole       8 -15.24 26.396454 0
hole       8 0 30.48 0
hole       8 15.24 26.396454 0
hole       8 26.396454 15.24 0
hole       8 44.690146 -14.520709 0
hole       8 38.015709 -27.620029 0
hole       8 27.6720029 -38.015709 0
hole       8 14.520709 -44.690146 0
hole       8 0 -46.99 0
hole       8 -14.520709 -44.690146 0
hole       8 -27.620029 -38.015709 0
hole       8 -38.015709 -27.620029 0
hole       8 -44.690146 -14.520709 0
hole       8 -44.690146 14.520709 0
hole       8 -38.015709 27.620029 0
hole       8 -27.620029 38.015709 0

```

hole 8 -14.520709 44.690146 0
hole 8 0 46.99 0
hole 8 14.520709 44.690146 0
hole 8 27.620029 38.015709 0
hole 8 38.015709 27.620029 0
hole 8 44.690146 14.520709 0

cylinder 2 1 63.5 82.08412 0
cylinder 0 1 76.835 82.08412 0
cylinder 3 1 79.375 82.08412 0
cylinder 8 1 79.6925 82.08412 0
cylinder 4 1 91.1225 82.08412 0
cylinder 10 1 91.44 82.08412 0
cuboid 0 1 4p91.44 82.08412 0

unit 10

com='bottom domed lid of fdt-1 tank'

hemisphe-z 1 1 78.226372 chord -48.222622
hemisphe-z 2 1 79.138455 chord -48.222622
cylinder 0 1 76.835 -48.222622 -79.138455
cylinder 3 1 79.375 -48.222622 -79.138455
cylinder 8 1 79.6925 -48.222622 -79.138455
cylinder 4 1 91.1225 -48.222622 -79.138455
cylinder 10 1 91.44 -48.222622 -79.138455
cuboid 0 1 4p91.44 -48.222622 -79.138455

unit 11

com='inner thimble for lid max ht = 14.118982 cm'

cylinder 0 1 0.52832 14.118982 0
cylinder 2 1 0.635 14.118982 0
cylinder 0 1 1.33223 14.118982 0
cylinder 2 1 1.67005 14.118982 0
cylinder 0 1 2.0447 14.118982 0
cylinder 2 1 2.413 14.118982 0

unit 12

com='outer thimble for lid max ht = 6.0767397 cm'

cylinder 0 1 0.52832 6.0767397 0
cylinder 2 1 0.635 6.0767397 0
cylinder 0 1 1.33223 6.0767397 0
cylinder 2 1 1.67005 6.0767397 0
cylinder 0 1 2.0447 6.0767397 0
cylinder 2 1 2.413 6.0767397 0

unit 13

com='top domed lid with void region of fdt-1 tank'

hemisphe+z 0 1 78.226372 chord -48.222622
hole 11 30.48 0 48.222622
hole 11 26.396454 -15.24 48.222622

```

hole      11    15.24 -26.396454 48.222622
hole      11    0 -30.48 48.222622
hole      11   -15.24 -26.396454 48.222622
hole      11   -26.396454 -15.24 48.222622
hole      11   -30.48 0 48.222622
hole      11   -26.396454 15.24 48.222622
hole      11   -15.24 26.396454 48.222622
hole      11    0 30.48 48.222622
hole      11   15.24 26.396454 48.222622
hole      11   26.396454 15.24 48.222622
hole      12   46.99 0 48.222622
hole      12   44.690146 -14.520709 48.222622
hole      12   38.015709 -27.620029 48.222622
hole      12   27.6720029 -38.015709 48.222622
hole      12   14.520709 -44.690146 48.222622
hole      12    0 -46.99 48.222622
hole      12   -14.520709 -44.690146 48.222622
hole      12   -27.620029 -38.015709 48.222622
hole      12   -38.015709 -27.620029 48.222622
hole      12   -44.690146 -14.520709 48.222622
hole      12   -46.99 0 48.222622
hole      12   -44.690146 14.520709 48.222622
hole      12   -38.015709 27.620029 48.222622
hole      12   -27.620029 38.015709 48.222622
hole      12   -14.520709 44.690146 48.222622
hole      12    0 46.99 48.222622
hole      12   14.520709 44.690146 48.222622
hole      12   27.620029 38.015709 48.222622
hole      12   38.015709 27.620029 48.222622
hole      12   44.690146 14.520709 48.222622
hemisphe+z  2 1 79.138455 chord -48.222622
cylinder   0 1 76.835 79.138455 48.222622
cylinder   3 1 79.375 79.138455 48.222622
cylinder   8 1 79.6925 79.138455 48.222622
cylinder   4 1 91.1225 79.138455 48.222622
cylinder  10 1 91.44 79.138455 48.222622
cuboid     0 1 4p91.44 79.138455 48.222622
unit 14
com='insulation on top of tank'
cylinder   0 1 2.38125 10.795 0
cylinder   6 1 2.69875 10.795 0
cylinder   5 1 91.44 10.795 0
cuboid     0 1 4p91.44 10.795 0
unit 15

```

```

com='insulation on bottom of tank'
cylinder  4 1  91.44 7.62 0
cuboid    0 1  4p91.44 7.62 0
unit 16
com='6 in. space between top of insulation and support ring'
cylinder  0 1  2.38125 15.24 0
cylinder  6 1  2.69875 15.24 0
cuboid    0 1  4p91.44 15.24 0
unit 17
com='support ring below steam dome'
cylinder  0 1  2.38125 20.32 0
cylinder  6 1  2.69875 20.32 0
cylinder  0 1  57.15 20.32 0
cylinder  6 1  60.96 20.32 0
cuboid    0 1  2p91.44 2p60.96 20.32 0
unit 18
com='connector from support ring to support posts'
cuboid    0 1  2p17.145 2p13.97 20.32 0
cuboid    6 1  2p18.415 2p15.24 20.32 0
cuboid    0 1  2p91.44 2p15.24 20.32 0
unit 19
com='support ring below steam dome with post connectors'
array     3  -91.44 -91.44 0
unit 20
com='space between support ring and steam dome'
cylinder  0 1  2.38125 2.2225 0
cylinder  6 1  2.69875 2.2225 0
cuboid    0 1  4p91.44 2.2225 0
unit 21
com='steam dome for fdt-1'
cylinder  0 1  2.38125 45.72 0
cylinder  6 1  2.69875 45.72 0
cylinder  0 1  10.16 45.72 0
cylinder  6 1  10.795 45.72 0
cylinder  0 1  60.325 45.72 0
cylinder  6 1  60.96 45.72 0
cuboid    0 1  4p91.44 45.72 0
unit 22
com='fdt-1 without support posts'
array     4  -91.44 -91.44 -159.34458
unit 23
com='support post for fdt'
cylinder  0 1  5.715 318.68916 0
cylinder  6 1  6.35 318.68916 0

```

```

cuboid    0 1 2p91.44 2p8.255 318.68916 0
unit 24
com='fdt-1 with support posts'
array     5  -91.44 -107.95 -159.34458
unit 25
com='thimble in salt on x-axis of fdt-2 region 3'
cylinder  0 1 0.52832 53.62807 0
cylinder  2 1 0.635 53.62807 0
cylinder  0 1 1.33223 53.62807 0
cylinder  2 1 1.67005 53.62807 0
cylinder  0 1 2.0447 53.62807 0
cylinder  2 1 2.413 53.62807 0
cuboid     1 1 4p8.255 53.62807 0
cuboid     0 1 4p8.255 53.62807 0
unit 26
com='cuboid of salt on x-axis'
cuboid     1 1 2p22.225 2p8.255 53.62807 0
cuboid     0 1 2p22.225 2p8.255 53.62807 0
unit 27
com='thimble for a hole in region 3 of fdt-2'
cylinder  0 1 0.52832 53.62807 0
cylinder  2 1 0.635 53.62807 0
cylinder  0 1 1.33223 53.62807 0
cylinder  2 1 1.67005 53.62807 0
cylinder  0 1 2.0447 53.62807 0
cylinder  2 1 2.413 53.62807 0
unit 28
com='region 3 of fdt-2 tank including thimbles & salt'
array      6  -55.245 -8.255 0
cylinder   1 1 62.23 53.62807 0
hole       27 26.396454 -15.24 0
hole       27 15.24 -26.396454 0
hole       27 0 -30.48 0
hole       27 -15.24 -26.396454 0
hole       27 -26.396454 -15.24 0
hole       27 -26.396454 15.24 0
hole       27 -15.24 26.396454 0
hole       27 0 30.48 0
hole       27 15.24 26.396454 0
hole       27 26.396454 15.24 0
hole       27 44.690146 -14.520709 0
hole       27 38.015709 -27.620029 0
hole       27 27.6720029 -38.015709 0
hole       27 14.520709 -44.690146 0

```

```

hole      27  0 -46.99 0
hole      27 -14.520709 -44.690146 0
hole      27 -27.620029 -38.015709 0
hole      27 -38.015709 -27.620029 0
hole      27 -44.690146 -14.520709 0
hole      27 -44.690146 14.520709 0
hole      27 -38.015709 27.620029 0
hole      27 -27.620029 38.015709 0
hole      27 -14.520709 44.690146 0
hole      27  0 46.99 0
hole      27 14.520709 44.690146 0
hole      27 27.620029 38.015709 0
hole      27 38.015709 27.620029 0
hole      27 44.690146 14.520709 0
cylinder  2 1 63.5 53.62807 0
cylinder  0 1 76.835 53.62807 0
cylinder  3 1 79.375 53.62807 0
cylinder  8 1 79.6925 53.62807 0
cylinder  4 1 91.1225 53.62807 0
cylinder 10 1 91.44 53.62807 0
cuboid    0 1 4p91.44 53.62807 0
unit 29
com='cuboid of void on x-axis in region 4 of tank'
cuboid    0 1 2p22.225 2p8.255 93.215685 0
unit 30
com='thimble on x-axis of fdt-2 region 4'
cylinder  0 1 0.52832 93.215685 0
cylinder  2 1 0.635 93.215685 0
cylinder  0 1 1.33223 93.215685 0
cylinder  2 1 1.67005 93.215685 0
cylinder  0 1 2.0447 93.215685 0
cylinder  2 1 2.413 93.215685 0
cuboid    0 1 4p8.255 93.215685 0
unit 31
com='thimble for a hole in region 4 of fdt-2'
cylinder  0 1 0.52832 93.215685 0
cylinder  2 1 0.635 93.215685 0
cylinder  0 1 1.33223 93.215685 0
cylinder  2 1 1.67005 93.215685 0
cylinder  0 1 2.0447 93.215685 0
cylinder  2 1 2.413 93.215685 0
unit 32
com='region 4 of fdt-2 including thimbles and void'
array     7 -55.245 -8.255 0

```

```

cylinder 0 1 62.23 93.215685 0
hole 31 26.396454 -15.24 0
hole 31 15.24 -26.396454 0
hole 31 0 -30.48 0
hole 31 -15.24 -26.396454 0
hole 31 -26.396454 -15.24 0
hole 31 -26.396454 15.24 0
hole 31 -15.24 26.396454 0
hole 31 0 30.48 0
hole 31 15.24 26.396454 0
hole 31 26.396454 15.24 0
hole 31 44.690146 -14.520709 0
hole 31 38.015709 -27.620029 0
hole 31 27.6720029 -38.015709 0
hole 31 14.520709 -44.690146 0
hole 31 0 -46.99 0
hole 31 -14.520709 -44.690146 0
hole 31 -27.620029 -38.015709 0
hole 31 -38.015709 -27.620029 0
hole 31 -44.690146 -14.520709 0
hole 31 -44.690146 14.520709 0
hole 31 -38.015709 27.620029 0
hole 31 -27.620029 38.015709 0
hole 31 -14.520709 44.690146 0
hole 31 0 46.99 0
hole 31 14.520709 44.690146 0
hole 31 27.620029 38.015709 0
hole 31 38.015709 27.620029 0
hole 31 44.690146 14.520709 0
cylinder 2 1 63.5 93.215685 0
cylinder 0 1 76.835 93.215685 0
cylinder 3 1 79.375 93.215685 0
cylinder 8 1 79.6925 93.215685 0
cylinder 4 1 91.1225 93.215685 0
cylinder 10 1 91.44 93.215685 0
cuboid 0 1 4p91.44 93.215685 0
unit 33
com='fdt-2 without support posts'
array 8 -91.44 -91.44 -159.34458
unit 34
com='fdt-2 with support posts'
array 9 -91.44 -107.95 -159.34458
unit 35
com='10 in space between fdt-1 & 2'

```

```

cuboid    0 1 2p12.7 2p107.95 2p159.34458
unit 36
com='fdts 1 and 2'
array    10  -195.58 -107.95 -159.34458
unit 37
com='cylindrical salt region of flush tank'
cylinder  7 1 62.23 126.68179 0
cylinder  2 1 63.5 126.68179 0
cylinder  0 1 76.835 126.68179 0
cylinder  3 1 79.375 126.68179 0
cylinder  8 1 79.6925 126.68179 0
cylinder  4 1 91.1225 126.68179 0
cylinder 10 1 91.44 126.68179 0
cuboid    0 1 4p91.44 126.68179 0
unit 38
com='cylindrical void region of flush tank'
cylinder  0 1 62.23 23.17821 0
cylinder  2 1 63.5 23.17821 0
cylinder  0 1 76.835 23.17821 0
cylinder  3 1 79.375 23.17821 0
cylinder  8 1 79.6925 23.17821 0
cylinder  4 1 91.1225 23.17821 0
cylinder 10 1 91.44 23.17821 0
cuboid    0 1 4p91.44 23.17821 0
unit 39
com='top domed lid with void region of flush tank'
hemisphe+z  0 1 78.226372 chord -48.222622
hemisphe+z  2 1 79.138455 chord -48.222622
cylinder  0 1 76.835 79.138455 48.222622
cylinder  3 1 79.375 79.138455 48.222622
cylinder  8 1 79.6925 79.138455 48.222622
cylinder  4 1 91.1225 79.138455 48.222622
cylinder 10 1 91.44 79.138455 48.222622
cuboid      0 1 4p91.44 79.138455 48.222622
unit 40
com='space above support ring'
cylinder  0 1 2.38125 53.022494 0
cylinder  6 1 2.69875 53.022494 0
cuboid    0 1 4p91.44 53.022494 0
unit 41
com='bottom domed lid of flush tank'
hemisphe-z  7 1 78.226372 chord -48.222622
hemisphe-z  2 1 79.138455 chord -48.222622
cylinder  0 1 76.835 -48.222622 -79.138455

```

```

cylinder 3 1 79.375 -48.222622 -79.138455
cylinder 8 1 79.6925 -48.222622 -79.138455
cylinder 4 1 91.1225 -48.222622 -79.138455
cylinder 10 1 91.44 -48.222622 -79.138455
cuboid 0 1 4p91.44 -48.222622 -79.138455
unit 42
com='flush tank without support posts'
array 11 -91.44 -91.44 -159.34458
unit 43
com='flush tank with support posts'
array 12 -91.44 -107.95 -159.34458
global unit 44
com='fdts 1 & 2 and flush tank'
array 13 -287.02 -107.95 -159.34458
cuboid 0 1 335.28 -310.72667 271.78 -264.16 442.63542 -182.20458
cuboid 8 1 345.44 -320.88667 281.94 -274.32 452.79542 -192.36458
cuboid 9 1 436.88 -412.32667 373.38 -365.76 544.23542 -283.80458
cuboid 0 1 436.88 -412.32667 373.38 -365.76 544.23542 -283.80458
end geom
read array
ara=1 nux=5 nuy=1 nuz=1
com='4 thimbles on x-axis in region 3 of fdt-1'
fill 2r2 3 2r2 end fill
ara=2 nux=5 nuy=1 nuz=1
com='4 thimbles on x-axis in region 4 of fdt-1'
fill 2r7 6 2r7 end fill
ara=3 nux=1 nuy=3 nuz=1
com='support ring'
fill 18 17 18 end fill
ara=4 nux=1 nuy=1 nuz=11
com='fdt-1 without support posts'
fill 15 10 1 5 9 13 14 16 19 20 21 end fill
ara=5 nux=1 nuy=3 nuz=1
com='fdt-1 with support posts'
fill 23 22 23 end fill
ara=6 nux=5 nuy=1 nuz=1
com='4 thimbles on x-axis in region 3 of fdt-2'
fill 2r25 26 2r25 end fill
ara=7 nux=5 nuy=1 nuz=1
com='4 thimbles on x-axis in region 4 of fdt-2'
fill 2r30 29 2r30 end fill
ara=8 nux=1 nuy=1 nuz=11
com='fdt-2 without support posts'
fill 15 10 1 28 32 13 14 16 19 20 21 end fill

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```

ara=9 nux=1 nuy=3 nuz=1
com='fdt-2 with support posts'
fill 23 33 23 end fill
ara=10 nux=3 nuy=1 nuz=1
com='fdts 1 and 2'
fill 24 35 34 end fill
ara=11 nux=1 nuy=1 nuz=9
com='flush tank without support posts'
fill 15 41 37 38 39 14 16 19 40 end fill
ara=12 nux=1 nuy=3 nuz=1
com='flush tank with support posts'
fill 23 42 23 end fill
ara=13 nux=2 nuy=1 nuz=1
com='fdts 1 and 2'
fill 36 43 end fill
end array
read plot
ttl='!x-y plot of fdts 1 & 2 and fft at z = -20 cm!'
plt=yes pic=mixture xul=-500 yul=500 zul=-20 xlr=500 ylr=-500 zlr=-20 uax=1
vax=0 wax=0 udn=0 vdn=-1 wdn=0 nax=130 lpi=6 end
end plot
end data
end

```

Appendix G
MSR Fuel Drain Tank Cell Drawings

D-FF-A-40456	MSRE FDT Cooling System Assembly & Details
D-FF-A-40457	MSRE Fuel Drain Tank Assembly & Details FDT-1
D-FF-A-40458	MSRE Fuel Drain Tank Upper Head Assembly & Details FDT-1
D-FF-A-40459	MSRE Fuel Drain Tank Details
D-FF-A-40460	MSRE Fuel Drain Tank Shell & Lower Head Assembly & Details
D-FF-A-40462	Fuel Salt System Flush Tank Assembly & Details
D-FF-A-40463	MSRE FDT Cooling System Steam Dome Upper Head Assembly & Details
D-FF-A-40464	MSRE FDT Cooling System Steam Dome Upper Head Details
D-FF-A-40465	MSRE FDT Cooling System Steam Dome Lower Head Assembly & Details
E-GG-D-40709	Piping in Reactor Fuel Flush & Drain Tank Cell - Elevation "A-A"
D-AA-A-40881	Molten Salt Reactor Experiment Coolant System Process Flow Sheet
D-AA-A-40882	Fuel Drain Tank System Process Flow Sheet MSRE
E-FF-D-41500	MSRE Fuel Drain Tank Supporting Structure and Weigh Assembly
E-FF-D-41501	MSRE Fuel Drain Tank Support Ring Assembly
E-FF-D-41502	MSRE Fuel Drain Tank Support Pedestal and Weigh Cell Mount
E-FF-D-41503	MSRE Coolant Salt Drain Tank Support Assembly
E-FF-D-41504	MSRE Coolant Salt Drain Tank Support Details
E-GG-D-41512	MSRE Piping in Reactor Fuel Flush & Drain Tank Cell - Plan - Sheet 2
E-MM-B-51610	MSRE Fuel Drain & Flush Tanks 8 Unit Heater Assembly Sheet - 1

E-MM-B-51611	MSRE Fuel Drain & Flush Tanks 8 Unit Heater Assembly Sheet - 2
E-MM-B-51612	MSRE Fuel Drain & Flush Tanks 8 Unit Heater Details
E-MM-B-51613	MSRE Fuel Drain & Flush Tanks 4 Unit Heater Assembly Sheet - 1
E-MM-B-51614	MSRE Fuel Drain & Flush Tanks 4 Unit Heater Assembly Sheet - 2
E-MM-B-51615	MSRE Fuel Drain & Flush Tanks 4 Unit Heater Details
E-MM-B-51616	Fuel Drain & Flush Tanks Ceramic Heater Unit MSRE
E-MM-B-51652	Heater & Insulation Assembly Fuel Drain & Flush Tanks MSRE
E-MM-B-51684	MSRE Fuel Drain Tank Nos 1 & 2 and Fuel Flush Tank Heater and insulation Assembly Sheet 1
E-MM-B-51685	MSRE Drain and Flush Tanks - Typ Elevation Heater & Insulation Assembly - Sheet 2
E-MM-B-51686	MSRE Fuel Drain Tank Nos 1 & 2 and Fuel flush Tank Heater and Insulation Can Assemblies
E-MM-B-51687	MSRE Fuel Drain Tank No. 1 Heater and Insulation Can Details
E-MM-B-51688	MSRE Fuel Drain Tank No. 1 Heater and Insulation Can Details
E-MM-B-51689	MSRE Fuel Drain Tank No. 1 Heater and Insulation Cover Assembly
E-MM-B-51690	MSRE Fuel Drain Tank No. 1 Heater and Insulation Cover Details
E-MM-B-51691	MSRE Fuel Drain Tank No. 1 Heater and Insulation Cover Details
E-MM-B-51692	MSRE Fuel Drain Tank No. 2 Heater and Insulation Can Details
E-MM-B-51693	MSRE Fuel Salt Flush Tank Heater and Insulation Can Details
E-MM-B-51694	MSRE Fuel Drain Tank No. 2 Heater and Insulation Cover Assembly

E-MM-B-51695	MSRE Fuel Drain Tank No. 2 Heater and Insulation Cover Details
E-MM-B-51696	MSRE Fuel Salt Flush Tank Heater and Insulation Cover Assembly
E-MM-B-51697	MSRE Fuel Salt Flush Tank Heater and Insulation Cover - Details

Vita

Michael E. Dunn was born in Knoxville, Tennessee on February 22, 1970. He graduated from Central High School in June 1988. The following August he entered the University of Tennessee, Knoxville, where he received a Bachelor of Science degree in Nuclear Engineering in August 1992.

In August 1992 he accepted a Department of Energy (DOE) Nuclear Engineering & Health Physics Fellowship and began study toward a Master of Science degree in Nuclear Engineering at the University of Tennessee, Knoxville. This degree was awarded in August 1994. Currently, he is pursuing a Doctor of Philosophy degree in Nuclear Engineering at the University of Tennessee.