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DOE STANDARD

GUIDELINES FOR PREPARING CRITICALITY SAFETY EVALUATIONS AT DEPARTMENT OF ENERGY NON-REACTOR NUCLEAR FACILITIES

U.S. Department of Energy
Washington, D.C. 20585

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Guidelines for Preparing Criticality Safety Evaluations at Department of Energy Non-Reactor Nuclear Facilities

Page/Section	**Change**
p. iii / Foreword to the Revised Edition	This section was added.
p. 1 / Overview / first paragraph	The reference to DOE O 420.1, Section 4.3 was added.
p. 2 / Definitions / Note section	The note on DOE O 420.1, Section 4.3 was added.
p. 4 / Section 3.0 / first paragraph	The reference to DOE O 420.1, Section 4.3 was added.
p. 6 / Section 5.0 / item 2	The reference to DOE O 420.1, Section 4.3 was added.
p. 5-1 / Example 5	The example was added.
p. 6-1 / Example 6	The example was added.
Concluding Material	The Preparing Activity was changed from DOE-DP-62 to DOE-EH-34.

FOREWORD TO THE REVISED EDITION

This revision to the Standard was prepared in accordance with the DOE Implementation Plan for Defense Nuclear Facility Safety Board (DNFSB) Recommendation 97-2. The Department committed to revise and reissue DOE-STD-3007-93 to include specific annotated examples of criticality safety evaluations that rely upon comparative analysis to existing data and calculations to emphasize the acceptability of this approach. Two examples (five and six) have been added to the appendix of the standard fulfilling this commitment. The examples are actual criticality safety evaluations from DOE sites. These two were selected from approximately a dozen candidates submitted for consideration by the DOE Criticality Safety Support Group (CSSG) formed by the DOE Implementation Plan. The DOE Office of Environment, Safety and Health, Office of Nuclear and Facility Safety, prepared the revision. The CSSG members were:

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FOREWORD

This Department of Energy (DOE) standard is approved for use by all components of DOE. It contains guidelines that should be followed when preparing Criticality Safety Evaluations that will be used to demonstrate the safety of operations performed at DOE Non-Reactor Nuclear Facilities. Adherence with these guidelines will provide consistency and uniformity in Criticality Safety Evaluations (CSEs) across the complex and will document compliance with DOE Order 5480.24 requirements as they pertain to CSEs.

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I. OVERVIEW

Contained in this document are guidelines that should be followed when preparing Criticality Safety Evaluations that will be used to demonstrate the safety of operations performed at Department of Energy (DOE) Non-Reactor Nuclear Facilities. Adherence with these guidelines will provide consistency and uniformity in Criticality Safety Evaluations (CSEs) across the complex and will document compliance with the requirements of DOE Orders 420.1, Section 4.3, and 5480.24 that pertain to CSEs as given in the following documents.

ANS-8.1	4.1.2	- Process Analysis
	4.2	- Technical Practices
	4.3	- Validation of Calculational Method
	5.	- Single Parameter Limits for Fissile Nuclides
	6.	- Multiparameter Control
ANS-8.19	8. (8.1-8.3)	- Process Evaluation For Nuclear Criticality Safety
5480.24	7.a.(2)(a)	- Double Contingency
	7.c (First Paragraph) -	"... contractors shall perform detailed nuclear criticality safety analyses for specific operations, storage arrangements, and the handling and transportation of fissionable material."
420.1	4.3.2	- General Requirements
	4.3.3.h	- Application of DOE-STD-3007-93

These guidelines become effective 180 days after issuance by DOE-HQ. Rewriting existing evaluations solely to comply with these guidelines is not required.

In accordance with Section 3.2 of ANS-8.1-1983, the word "shall" is used in these guidelines to denote a requirement, the word "should" to denote a recommendation, and the word "may" to denote permission, neither a requirement nor a recommendation.

II. DEFINITIONS

BIAS - A measure of the systematic disagreement between the results calculated by a method and experimental data. The uncertainty in the bias is a measure of both the precision of the calculations and the accuracy of the experimental data.

CALCULATIONAL METHOD - The mathematical equations, approximations, assumptions, associated numerical parameters (e.g., neutron cross sections), and calculational procedures that yield the calculated results.

CONTINGENCY - A possible but unlikely change in a condition/control important to the nuclear criticality safety of a fissionable material operation that would, if it occurred, reduce the number of barriers (either administrative or physical) that are intended to prevent an accidental nuclear criticality.

CRITICALITY SAFETY EVALUATION (CSE) - A documented process that establishes the technical basis for nuclear criticality safety and provides subcritical operating values.

CREDIBLE - Offering reasonable grounds for being believed on the basis of commonly acceptable engineering judgment. Assigning a numerical probability that defines credible is not required when demonstrating compliance with the double contingency principle.

DESIGN FEATURES - Passive or active features that are necessary to prevent or reduce the probability of a criticality accident.

DOUBLE-CONTINGENCY PRINCIPLE [Per DOE 5480.24 7a(2)(a)*] - Process designs shall incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. Protection shall be provided by either:

1. The control of two independent process parameters (which is the preferred approach, if practical) or

2. A system of multiple (at least two) controls on a single parameter.

In all cases, no (credible) single failure shall result in the potential for a criticality accident.

*NOTE: DOE 5480.24 has been replaced by DOE O 420.1, Section 4.3.

PROCESS PARAMETER - A physical property whose value affects the nuclear reactivity of a system. Parameters include the mass, density, and isotopic enrichment of fissionable material; the geometry, reflection, and interaction conditions of the system; and the moderation, composition and neutron absorption characteristics of the fissionable material mixture and other system materials.

III. FORMAT AND CONTENT OF CRITICALITY SAFETY EVALUATIONS

Criticality safety evaluations that are prepared for Department of Energy (DOE) Non-Reactor Nuclear Facilities should conform to the format and content guidelines provided in this section. An outline of an evaluation is given below. All evaluations used to demonstrate facility safety shall contain, at an appropriate level of detail, the information embodied in these major sections. These sections should appear in the specified order whenever practical. [Criticality safety evaluations for transportation packages that are required to follow a pre-specified Nuclear Regulatory Commission (NRC) format are exceptions.] Subsections may be used at the author's discretion to provide order and clarity. A description of each section of an evaluation follows the outline.

1.0 INTRODUCTION

2.0 DESCRIPTION

3.0 REQUIREMENTS DOCUMENTATION

4.0 METHODOLOGY

5.0 DISCUSSION OF CONTINGENCIES

6.0 EVALUATION & RESULTS

7.0 DESIGN FEATURES (PASSIVE & ACTIVE) AND ADMINISTRATIVELY CONTROLLED LIMITS & REQUIREMENTS

8.0 SUMMARY & CONCLUSIONS

9.0 REFERENCES

APPENDIX A: MATERIALS & COMPOSITIONS

APPENDIX B: TYPICAL INPUT LISTINGS

APPENDIX C, D,... ETC.: OPTIONAL APPENDICES PROVIDING SUPPLEMENTAL INFORMATION

1.0 INTRODUCTION

The purpose and scope of the evaluation shall be stated in this section. Relevant background information should also be provided.

2.0 DESCRIPTION

The system or process should be described in Section 2.0. The description should contain sufficient detail, clarity, and lack of ambiguity to allow a peer reviewer either to independently evaluate the system/process or to independently assess the adequacy and accuracy of the existing evaluation. (In most cases, firsthand knowledge obtained through facility visits is required to supplement the written documentation.) Drawings and/or sketches should be provided as needed to provide clarity.

References, including drawings, should be provided to allow a reviewer the opportunity to further research the system being evaluated and to verify the accuracy of the descriptive information provided. References should be specific and should include page numbers if possible. Multiple levels of references should be avoided. (i.e., References should be original documents and not documents that reference other documents.)

It may not be practical to provide this level of detail in evaluations of very complicated systems or processes. For such evaluations, documents that do provide this level of detail should be incorporated by reference. Sufficient information should be provided for the system or process to allow adequate support of the assumptions used in the evaluation.

3.0 REQUIREMENTS DOCUMENTATION

Indicate the specific DOE Orders or Guides, ANS standards, Code of Federal Regulations, or internal requirement documents that are uniquely applicable to the evaluation. (NOTE: General requirement documentation such as DOE Order 5480.24, DOE O 420.1, Section 4.3, ANS-8.1, or ANS-8.19 apply to all evaluations. These types of documents need not be referenced or discussed in every evaluation unless certain sections require emphasis.)

Examples of specific requirements documents are: ANS-8.5 (applicable to evaluations that involve Raschig rings) and 10 CFR Part 71 (applicable to evaluations of shipping containers that require NRC licensing). Where specific requirements are involved, they should be identified (e.g., “For a Fissile Class I package, Section 71.55 of 10 CFR Part 71 requires ...”).

Exemptions to DOE Orders should also be discussed in this section.

4.0 METHODOLOGY

In order to establish that a proposed system or process will be subcritical under the normal conditions and under postulated process upset conditions, it is first necessary to establish acceptable subcritical values for the operation and then show that the proposed operation will not exceed those values. There are currently four acceptable methods for the establishment of these values. They are:

1. By reference to national standards that present Subcritical Limits.

2. By reference to widely accepted handbooks of Subcritical Limits.

3. By reference to experiments with appropriate adjustments to ensure subcriticality when the uncertainties of parameters reported in the experiment documentation are considered.

4. By validated calculational techniques.

The method used in a given evaluation must be supported in the text of this section.

When employing methods 1 or 2, simply provide the reference giving the Subcritical Limit. When employing method 3, provide the references giving the critical parameters, and fully explain your consideration of uncertainty in the reported critical parameters when determining limits.

When employing method 4, indicate the specific methods that were used in the assessment of subcriticality. References should be provided to allow a reviewer the opportunity to further research the methods used in the evaluation. It is not necessary to describe the theory behind any calculational methods used. (This should be done by reference.) It is only necessary to indicate what methods were used.

Examples of calculational methods are: the three-dimensional Monte Carlo code, KENO-V.a or MCNP; the one-dimensional S_n discrete-ordinates transport theory code, ANISN; and hand calculation methods such as limited surface density, density analog, or solid angle methods.

When applicable, identify the nuclear cross section data that were used and the methods used to process these data.

Compliance with Section 4.3, “Validation of a Calculational Method,” of ANS-8.1 should also be demonstrated in this section. Reference can be made to more detailed validation reports; however, the results of these validation reports should be summarized, the method used to determine the bias (if any) should be described, and the overall conclusion(s) of validation efforts and how the conclusions were applied to the evaluation results should be stated.

When computer neutronics calculations were used, the type of computing platform should be stated along with relevant code configuration control information. This information may be provided by reference.

For Example: All calculations documented in this evaluation were performed on an IBM Model 320 RISC 6000 workstation operating under AIX Version 3.1 with Version 1.1 of the Fortran Compiler. Configuration Release 1.10 of the KENO-V.a code with Hansen-Roach cross section data were used for this evaluation. More detailed configuration information is given in Reference _.

5.0 DISCUSSION OF CONTINGENCIES

Compliance with the double contingency principle, as stated in DOE Order 5480.24 and DOE O 420.1, Section 4.3, and the requirements of Section 4.1.2 of ANS-8.1 should be demonstrated in this section. This may be done by several different methods. Two possible methods are listed below.

1. List the independent unlikely events, at least two of which must occur concurrently before a criticality accident could be possible. Describe the controls, conditions, etc. that ensure that no credible single failure will result in a criticality. An illustration of this method is given below (format shown is not a requirement).

Contingency		
No.	Description	Barriers
1	Unlikely Event # 1	Controls, conditions, etc. that make Event # 1 unlikely
2	Unlikely Event # 2	Controls, conditions, etc. that make Event # 2 unlikely

2. If control of two independent parameters is not practical, DOE 5480.24 and DOE O 420.1, Section 4.3, allow a system of multiple controls (at least two) on a single parameter. The overall likelihood of concurrent failure of these controls should be comparable to two unlikely events as discussed in Method 1. The basis for controlling a single parameter as opposed to the preferred method of controlling two independent process parameters should also be documented in this section. An illustration of this method is given below (format shown is not a requirement).

	Controls	
Control Parameter	No.	Description
Parameter	1	Describe Control # 1
	.	
	.	
	.	
	N	Describe Control # N

[Note: Examples of each method are given in Section IV of this document.]

Contingency analysis, in general, is not required for incredible upset conditions.

Some systems will remain subcritical during any combination of credible upset conditions. For such systems, simply state the fact in this section. (e.g., "As shown in Section 6.0, this system will remain subcritical under any combination of credible upset conditions.")

6.0 EVALUATION & RESULTS

After the establishment of appropriate and sufficient subcritical limits (see Section 4.0), it must be shown that both the normal and worst-case postulated upset conditions (may be limited to credible conditions only) will be equal to or less than those values. If the actual conditions differ from those used to establish the subcritical limits, it must be shown that the comparison is conservative.

If calculational techniques are used, a detailed description of the models should be presented in this section. The level of detail should be sufficient to allow an independent reviewer to easily reconstruct the computational model, compare the model with the descriptive information in Section 2.0, and determine if the overall model is conservative. Sketches showing relevant model dimensions should be included. (Very simple modeling that can clearly be described in a few sentences is an exception.) Significant assumptions and simplifications should be stated. Mesh spacing, quadrature order, the order of the Legendre scattering expansion, and other parameters pertinent to numerical techniques should be specified or incorporated by reference.

All material compositions used in the evaluation should be specified in Appendix A. Typical input listings should be given in Appendix B. Any of this information may be incorporated into the text or by reference.

All calculational results should be reported in this section. Estimated uncertainties in the methodology (e.g., statistical uncertainties associated with Monte Carlo calculations) and sensitivities to modeling simplifications (e.g., effects of homogenization, dimension or geometry modifications, etc.) should be stated.

7.0 DESIGN FEATURES (PASSIVE & ACTIVE) AND ADMINISTRATIVELY CONTROLLED LIMITS & REQUIREMENTS

Design features (passive and active) and administratively controlled limits and requirements for the purpose of preventing or reducing the probability of a criticality accident should be stated in this section or should be incorporated by reference.

8.0 SUMMARY & CONCLUSIONS

The overall criticality safety assessment of the system being analyzed should be summarized in this section. The range of applicability and special limitations in the evaluation should be stated.

A statement about the application of Double Contingency should also be provided. (e.g., "As indicated in Section 6.0, this operation adheres to the Double Contingency Principle;" or "as discussed in Section 6.0, this system will remain subcritical under all credible upset conditions.")

If unique requirements must be satisfied (those discussed in Section 3.0), a statement of compliance with these requirements should be given.

When applicable, reference to normal and abnormal ranges of operational parameters may also be made. Portions of the evaluation that have been deferred to other efforts or documents should be restated.

9.0 REFERENCES

To the extent practical, a criticality safety evaluation should stand on its own. However, references may be used liberally so all external technical information (information from handbooks or information from other reports that is beyond the scope of the evaluation) and relevant descriptive information can easily be verified. Private communications, as references, should be avoided, and crucial conclusions of the evaluation shall not depend on private communications.

APPENDIX A
MATERIALS & COMPOSITIONS

All materials used in computations should be identified and atom densities should be given. When applicable, nuclide library IDs should be included. All other information, such as assumed bulk or theoretical densities, that is required to duplicate the atom densities used in the evaluation should also be given. This information may be incorporated by reference.

APPENDIX B
TYPICAL INPUT LISTINGS

Typical input listings for all major perturbations in modeling should be included in this appendix. If cross section preprocessing is performed, typical input listings for all cross section codes (e.g., NITAWL, BONAMI, XSDRNPM, COMBINE/PC) that are used in the evaluation should also be included. This information may be incorporated by reference.

APPENDIX C, D,... ETC.
OPTIONAL APPENDICES PROVIDING SUPPLEMENTAL INFORMATION

Other appendices providing supplemental information may be provided at the author's discretion.

IV. EXAMPLES OF CONTINGENCY ANALYSES

EXAMPLE # 1 Contingency analysis for a dry storage cask that is used to store irradiated PWR or BWR fuel assemblies.

Contingency		Barriers
No.	Description	
1	Flood Cask Interior[a]	Cask is sealed; lid is well above flood plain[b]
2	Loss of Neutron Poison from Insert[a]	Insert is in a noncorrosive environment[b]

a. The degree of flooding or the magnitude of poison loss required to have a criticality accident is discussed in Section 6.0.

b. Surveillance is required to ensure the integrity of this barrier. Details of surveillance procedures are documented in Reference _.

EXAMPLE # 2 Contingency analysis for a large dissolver vessel. Criticality safety margins are maintained by controlling the mass of fissile material and the concentration of neutron poison inside a large dissolver vessel. Inadvertent criticality can occur by exceeding the fissile mass, dilution or improper makeup of soluble neutron poison, or by a combination of the two.

	Controls[a]	
Control Parameter	No.	Description
Fissile Mass (U-235)	1.	Mass Balance System # 1
	2.	Mass Balance System # 2
	3.	Visual Inspection
OR		
Concentration of Neutron Poison	1.	Poison Monitor # 1
	2.	Poison Monitor # 2
	3.	Sampling

a. A more detailed description of these controls is given in Reference __. At least two controls are required; no further significance should be placed on the number of controls listed in this example.

EXAMPLE # 3 Contingency analysis for a dry fuel element staging area.

Contingency		
No.	Description	Barriers
1	Flooding[a]	Only water source is fire suppression system - fire or malfunction is required to actuate. Flow alarms on the fire water, heat and smoke detectors in the staging area. Doors do not impede reasonable water flow rates. Water will not build up on the floor.
2	Item Limit Violation[a]	Administratively controlled limits and written procedures. Operator training and certification. Independent observation

a. The degree of flooding or the magnitude of item limit violation required to have a criticality accident is discussed in Section 6.0.

EXAMPLE # 4 Contingency analysis for fuel handling in a storage pool.

	Controls[a]	
Control Parameter	No.	Description
Item Limit (Number of fuel handling units out of storage)	1.	Administratively controlled limits & procedures
	2.	Operator training & certification
	3.	Pre-job briefing
	4.	Independent observation

a. A more detailed description of these controls is given in Reference __. At least two controls are required; no further significance should be placed on the number of controls listed in this example.

EXAMPLE # 5 Contingency analysis for a hypothetical process area operation involving loading of fissile parts into a container.

Contingency		Barriers
No.	Description	
1	Loss of Mass Control By:	Administratively controlled limits and requirements[a]
1a	Exceeding Number of Parts in a Container	
OR		
1b	Exceeding Mass of Part	
2	Loss of Moderator Control By:	Administratively controlled limits and requirements[a] (These events must occur during loading operation, or an operator must fail to close to the container.)
2a	Failure of Sprinkler Head	
OR		
2b	Sprinkler Activation Because of Fire	
OR		
2c	Roof Leak During Rainfall	

a. A detailed description of these administratively controlled limits and requirements are given in Reference__.

EXAMPLE # 6 Contingency analysis for a hypothetical process area operation involving loading of fissile parts into a container where flooding is incredible (i.e., there are no liquid sources and container cannot hold liquid).

Control Parameter	Controls No.	Controls Description
Number of Parts		Administratively controlled limits & requirements
	1.	Limit on number of parts in loading area.
	2.	Limit on number of parts in container.
OR		
Mass of Parts		Administratively controlled limits & requirements
	1.	Limit on mass of parts
	2.	Mass verification of parts (Weighing Requirements)

V. EXAMPLES OF CRITICALITY SAFETY EVALUATIONS THAT MEET THE INTENT OF THE DEPARTMENT OF ENERGY NON-REACTOR NUCLEAR FACILITY GUIDELINES

Examples of criticality safety evaluations that have been adapted to follow the Department of Energy Non-Reactor Nuclear Facility Guidelines for preparing Criticality Safety Evaluations are included in this section. These evaluations were prepared at various DOE facilities. They are presented here for illustration purposes only, with no endorsement of fissile material limits or the methods of derivation. Use of these examples for other purposes (i.e., quoting results to demonstrate the safety of other systems or use of requirements that may apply only to specific facilities) is not authorized. Document Control Numbers, review and approval signatures, implementation documentation, etc., are not addressed in these guidelines and are not illustrated in these examples.

Notes contained in brackets [ ] either relate information contained within an example to the guidelines or highlight illustrative weaknesses in an example.

EXAMPLE 1

Provided by

Oak Ridge Y-12 Plant

CRITICALITY SAFETY EVALUATION FOR DISTRIBUTION TRAP F-1202 FOR SOLUTION TRANSFER TO D-WING

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1.0 INTRODUCTION

The following analysis has been performed to document the technical basis for the approval of distribution trap F-1202, which is used to transfer product solution from Room 1021 scrap recovery operation to the D-1 wing or to safe bottles.

2.0 DESCRIPTION

<u>Operation</u>

To transfer solution into F-1202, valves are set as appropriate to receive solution from either F-1000 (the accountability tank) or F-1202 A-D (west wall storage tanks). Wet vacuum is applied to deliver solution to F-1202, using intermediate vacuum trap F-1201. F-1201 has a high level probe interlocked with a block valve in the vacuum line to stop overflow from F-1202. Vacuum to F-1202 is supplied by the final process area wet vacuum trap located in the northwest corner of Room 1022; this trap has an audible alarm.

Solution in F-1202 may be drained to a safe bottle or the F-400 tanks in D-Wing. As each safe bottle is filled, it is transferred to an approved storage rack or dolly. For transfer to D-Wing, valves in D-Wing are set to deliver solution to any tanks F-400-1 through F-400-3, or F-400-6 through F-400-10.

In Room 1021, the transfer line to D-Wing passes over the Room 1021 fissile storage array. Over this array, the stainless steel transfer line is encased in a PVC pipe that is sealed at either end and provided with a drain on the lower end. This drain is routed to a single safe bottle mounted near the stairway at the scrubber system.

The transfer line to D-Wing is also double-walled from the Room 1022 wall penetration all the way to the entrance to D-Wing. Any transfer line leaks in this section will drain to a safe bottle located on the north wall of D-Wing.

The entire floor area underneath the Room 1021 west wall tanks and accountability tank, and the area in front of the tray dissolver hoods has a stainless steel floor bordered with a 0.75” high dike. This floor area is intended to contain leaks or spills and to aid in contamination control.

Equipment

F-1202 is a standard 4” Pyrex pipe, 3’ tall. F-1202 is located 2’ south of tank F-1200-A.

Reference Drawings (not necessarily as-built)

J2E-XXXXX,P2E-XXXXX.

Reference Operation Procedures

OP-XXXXX.

3.0 REQUIREMENTS DOCUMENTATION

There are no unique requirements for this analysis.

4.0 METHODOLOGY

J. T. Thomas, ed., “Nuclear Safety Guide, TID-7016, Revision 2,” NUREG/CR-0095 (ORNL/NUREG/CSD-6), June 1978 - Figure 2.3.

5.0 DISCUSSION OF CONTINGENCIES AND EVALUATION

Parameters

Criticality safety of the distribution trap and transfer operations depends on controlling geometry.

NOTE: Equipment approved by this CSA is limited to (a) Trap F-1202 and various connecting lines for feed and vacuum, and (b) the transfer line to D-Wing and the safe bottle near the Room 1021 scrubber.

Final wet vacuum traps (i.e., northwest corner of Room 1022) and the wet vacuum equipment in the Headhouse fan room, other Room 1021 tanks, D-Wing tanks, and the safe bottle used in D-Wing as a drain collection for the second double wall section are currently approved by other CSAs.

The operational steps specifically analyzed here are transfers into F-1202 and transfers from F-1202 to D-Wing tanks. Safe bottle handling and storage, dolly handling and storage, area cleanup and mopping, etc. are standard area activities that are not unique to this operation. These are addressed in Y/DD-395, “Basic Nuclear Criticality Safety Guidelines.”

Controls

The following controls are identified for the above parameter.

- The dimensions of the trap and its location relative to other tanks are fixed by design. All transfer lines are 1.5” diameter or less. The double-walled pipe sections have sleeve diameters of either 3” (steel jacket in D-1 Wing and Room 1010 plenum areas or 4” (CPVC pipe over the Room 1021 fissile storage array).

- A spill or a leak would result in relocation of uranium solution to another safe geometric configuration. Near the west wall of Room 1021, the solution would spill into the diked area underneath the storage tanks and other equipment. Over the Room 1021 fissile storage array, the liquid would drain into the 4” CPVC pipe jacket and drain into a safe bottle in the diked scrubber area. In the Room 1010 plenum or D-1 Wing, the liquid would enter the 3” stainless steel pipe jacket and drain into a safe bottle in D-Wing. Overfill of this safe bottle, D-Wing tank overfill, or tank or line leakage in D-Wing would result in a simple solution spill onto the stainless steel floor of the D-Wing solution storage area.

- Except for the double-walled line section Room 1010 plenum and D-1 Wing, all areas near transfer lines, the trap or D-Wing storage tanks are identified as unsafe container exclusion areas: open, large geometry containers are strictly prohibited without specific Criticality Safety approval.

- Administrative control is exercised for use of the wet vacuum for transfers. Procedurally, the operator is to observe F-1202 to avoid overfill. Minor carryover (splatter, etc.) to F-1201 would be routinely expected. Occasionally, overfill of F-1202 might briefly occur. In such case, engineered operational controls (level alarm plus interlocked block valve on vacuum line) are provided for F-1201. These operational controls are not considered to be required from a criticality safety standpoint - the primary point of criticality safety control is that the final wet vacuum trap in the northwest section of Room 1022 must not be allowed to overfill. As required by other CSAs, any alarm of this trap requires that use of wet vacuum immediately stop. (Manual block valves are present at the trap.) The solution source must be identified and corrected before use of the wet vacuum may resume. Weekly testing is required for the alarm system on this final trap.

- Selection of the correct container (safe bottle) for drainage of solution is an administrative control. The use of proper equipment for spill cleanup (4-liter beakers or process area mop buckets) is also an administrative control.

NOTE: Detailed interaction considerations are not warranted here. The analyses of other CSAs provide interaction considerations for all fixed process equipment in the west area of Room 1021, including F-1202.

Likewise, the elevation of the small volume trap (well above manually handled containers near floor level) avoids the need to consider portable process containers in contact with the trap. The trap is fabricated of standard 4” diameter x 36” tall Corning conical Pyrex glass. Based on Corning vendor data[1], the design outside diameter is 4 17/32” with a 17/64” wall, giving an inside diameter of 4”. The trap volume is thus only 7.4 liters. The trap base is elevated about 11’8” above the Room 1021 floor.

There is no ready means for operator access to the trap. There is no operational incentive to have a process container in close proximity to the trap.

CONTINGENCIES

1. Loss of geometry control

   (a) Transfer line leak, valve leak, trap breakage, etc., near the west wall of Room 1021
   (b) Leak in transfer line to D-Wing
   (c) Overfill of D-Wing storage tanks, line leak, etc., in D-Wing
   (d) Loss of liquid to the wet vacuum system
   (e) Use of an improper container for solution drainage from the trap

ANALYSIS

Contingency 1a

Any fissile solution release from the process equipment along the west wall of Room 1021 will result in a simple solution spill into a safe geometry slab.

The other analyses demonstrate that this diked area could retain the entire solution inventory of the Room 1201 west wall solution process equipment without exceeding a solution depth of 0.75”. Room 1021 is identified as an unsafe container exclusion area; large, open containers are prohibited without specific Criticality Safety approval. Posting is present regarding this limitation.

Contingency 1b

Any leak of the transfer line to D-Wing will result in solution drainage into safe bottles or a spill into a safe slab configuration.

The jacket diameters of the double walled sections (3” and 4”) are less than the subcritical fissile solution diameter limits given on Figure 2.3 of TID-7016[2]. The jackets (like the transfer line) are continuously sloped to avoid solution holdup, and are designed of material compatible with the nitric-acid based fissile solution. Minor leakage would drain to the safe bottles; monthly inventory efforts include checking such bottles for material. Should the liquid not be detected in a partially loaded bottle, or if the leak were at a greater rate, either bottle could overfill and result in a simple solution spill onto the floor in areas designed for safe spill retention.

Any leak from a non-jacketed section of the transfer line (from the west wall area of Room 1021 to the west edge of the storage array, or from the east edge of the storage array to the Room 1022 wall penetration) would result in a simple solution leak in a process area that excludes unsafe geometry containers. Established practices exist and would be strictly followed for material cleanup (use 4-liter beakers or process area mop buckets).

Contingency 1c

Overfill of the D-Wing solution storage tanks will result in a simple spill of fissile solution into a safe slab geometry.

D-Wing is designed as a dedicated fissile solution storage area, therefore, such a spill should be of no consequence for a criticality safety standpoint. Mop buckets or 4-liter beakers would be used for material cleanup.

Contingency 1d

Adequate controls are in place to preclude criticality due to misuse of the wet vacuum system.

A reasonable level of operational control is present to avoid routine transfer of solution to the final Special Process area trap in the northwest area of Room 1022.

Significant solution bypass of the final trap would require gross system misoperation for the transfer efforts to F-1202, coupled with either alarm failure or lack of personnel response to the alarm. In the Headhouse, the final wet vacuum equipment provides a significant safe geometry surge volume (multiple 6-inch diameter steel traps in parallel), as discussed in CSA HH-123, followed by two alarmed 6-inch glass traps in series. Upon liquid detection by these final alarms, automatic block valves in the main vacuum header close to prevent fissile solution transfer to the jet vacuum producers. This alarm system and interlocked block valve is formally designated as a safety system. The surge volume plus safety system serve to positively preclude loss of geometry control fissile solution, by preventing loss of fissile material to the storm sewer drain system.

Contingency 1e

Use of an improper container for drainage of solution from F-1202 is precluded by operator training and supervisory oversight.

A situation whereby an operator intentionally transfers fissile solution from F-1202 into a large geometry container is precluded by the strict nature of operator training and supervisory oversight for this process area. Operator training emphasizes the immediate danger of such action. Such an event would be expected only as the result of either sabotage or suicidal action, which are outside the bounds of the analysis.

6.0 RESULTS

The distribution trap is approved for use as described.

7.0 DESIGN FEATURES AND ADMINISTRATIVELY CONTROLLED LIMITS AND REQUIREMENTS

Criticality safety of this operation depends on maintaining all solution in safe geometry.

1. Proper spacing (at least 12-inches edge-to-edge) must be maintained between multiple process containers, or between process containers, safe tanks, and vacuum traps.

2. Spills or leaks must be collected in approved solution containers such as safe bottles or 4-liter beakers.

3. If the final wet vacuum trap is in the northwest corner of Room 1022 alarms, vacuum transfer must be immediately stopped. The source of solution must be determined and corrected before further use of wet vacuum.

4. Large geometry containers or equipment (internal volume available for solution holdup exceeding 4 liters) are not permitted near this process equipment without Criticality Safety approval.

8.0 SUMMARY AND CONCLUSIONS

It has been shown in section 5.0 that the proposed trap is a safe geometry and meets the requirements of the double contingency principle.

9.0 REFERENCES

1. Corning Process Systems, “Process Fittings and Hardware,” July 1985.

2. J. T. Thomas, ed., “Nuclear Safety Guide, TID-7016, Revision 2, “NUREG/CR-0095 (ORNL/NUREG/CSD-6), June 1978.

EXAMPLE 2

Provided By

Idaho Chemical Processing Plant (ICPP)

CRITICALITY SAFETY EVALUATION FOR STORAGE OF BORAX V ON CPP-603 NORTH AND MIDDLE BASIN FLOOR

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CONTENTS

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FIGURES

TABLES

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Criticality Safety Evaluation for Storage of BORAX V on CPP-603 North and Middle Basin Floor

1.0 INTRODUCTION

The purpose of this evaluation is to provide a basis to store BORAX V Superheater fuel in the Idaho Chemical Processing Plant (ICCP) Building 603 north and middle basins, with no spacing restrictions between fuel storage buckets within a single row. As a result of the age and degradation of the Building 603 fuel storage facility, failure of a BORAX V fuel storage bucket hanger occurred in July 1992, thus causing the loaded storage bucket to fall to the floor. Due to the lower reactivity of BORAX V Superheater fuel, there exists a possibility that the spacing provided at one time by the basin monorail fuel storage system might not be necessary. If this spacing is deemed unnecessary, the definition of approved storage for BORAX V fuel elements could possibly be changed to eliminate the requirement for spacing within a given row. The redefinition of approved storage would entail the placement of storage buckets, containing three or fewer BORAX V Superheater fuel elements, onto the basin floor. This redefinition would apply only to rows exclusively containing BORAX V Superheater fuel elements. With this redefinition all spacing restrictions between storage buckets could be removed.

Considered in this evaluation are various arrangements of BORAX V Superheater fuel assemblies within a single storage row. Only BORAX V Superheater fuels are considered in this evaluation.

2.0 DESCRIPTION

The Building 603 fuel storage facility consists of three underwater storage basins, the north, middle, and south. (See Figure 1).[1] The south storage basin consists of a series of storage racks. The two basins of concern are the north and middle basins, since those basins contain the monorail fuel storage system. The north and middle basins are 40 feet (12.2 m) by 60 feet (18.3 m), with a water depth of 21 feet (6.4 m). The middle and north basins consist of a series of channels, 12 inches in width (30.48 cm), through which the monorail system runs. Each channel is separated by a concrete divider that is 12 inches in width (30.48 cm) and 30 inches high (76.2 cm), as shown in Figure 2.[1] The current age of the facility and degradation due to corrosion problems have led to failures of the monorail fuel storage system, such that the loss of spacing provided by the monorail system can no longer be considered a contingency.

Fuel is currently stored in the north and middle basins by one of two methods. The first method, common to most of the navy fuels, consists of fuel hanging from a single hook arrangement. A standard is placed through the center of the fuel assembly and hung directly on the hook. In the second method, which is used for the BORAX V fuel, the fuel assemblies are placed into a bucket that is then hung from the monorail system.

An example of a monorail hanger system and a typical bucket is given in Figure 3.[1] The buckets are constructed of either aluminum or stainless steel, depending upon fuel types stored in the bucket, thus assuring materials compatibility.

The failure of the monorail storage system is directly related to the corrosion problems in the north and middle basins. Earlier corrosion problems led to modification of some monorail hangers. Some of these modifications included the cutting of carbon steel hanger systems and the replacement of the carbon steel hooks with stainless steel hooks.

In the process of attaching the stainless steel hooks to the carbon steel hanger assemblies, a galvanic couple was created. With existing corrosion problems and loss of structural integrity due to the galvanic couple, the load present caused a failure to occur.

Due to the loss of the engineered safety feature (spacing) provided by this system, evaluation of a new fuel configuration was required. Considered in this evaluation are various arrangements of BORAX V fuel elements without the spacing that was once maintained by the monorail storage system. In addition to the loss of spacing between storage buckets, the reactivity effects due to reflection from the concrete storage channel walls was also evaluated. A more detailed description of the 603 Underwater Fuel Storage Facility can be found in the Plant Safety Document (PSD) Section 4.6.[1] A more detailed description of the BORAX V Superheater fuel and the computational models used in this evaluation can be found in the "Evaluation and Results" section of this report and References 3, 4, 5, 6 and 7.

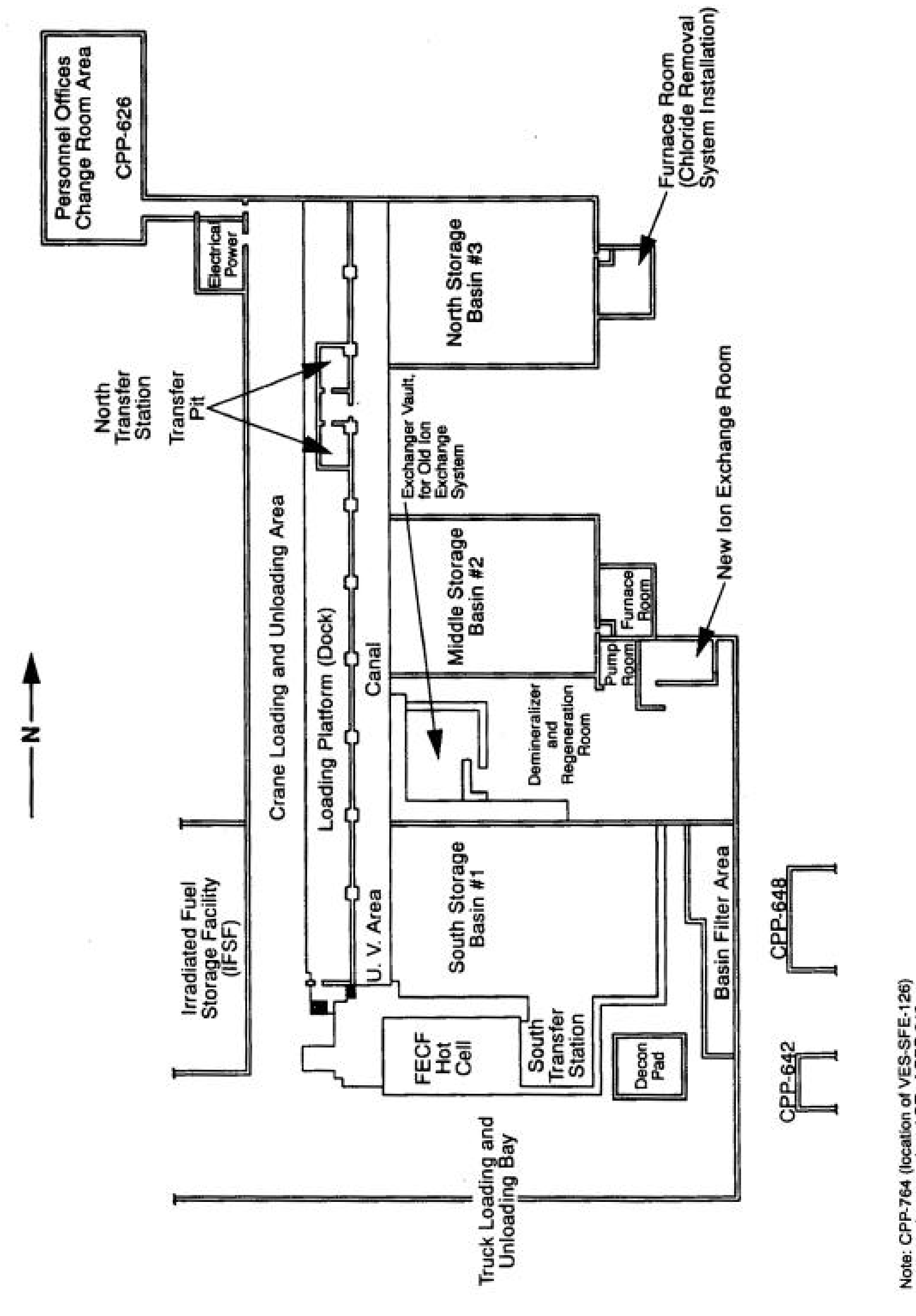

Figure 1. CPP 603 Fuel Storage Facility Layout.

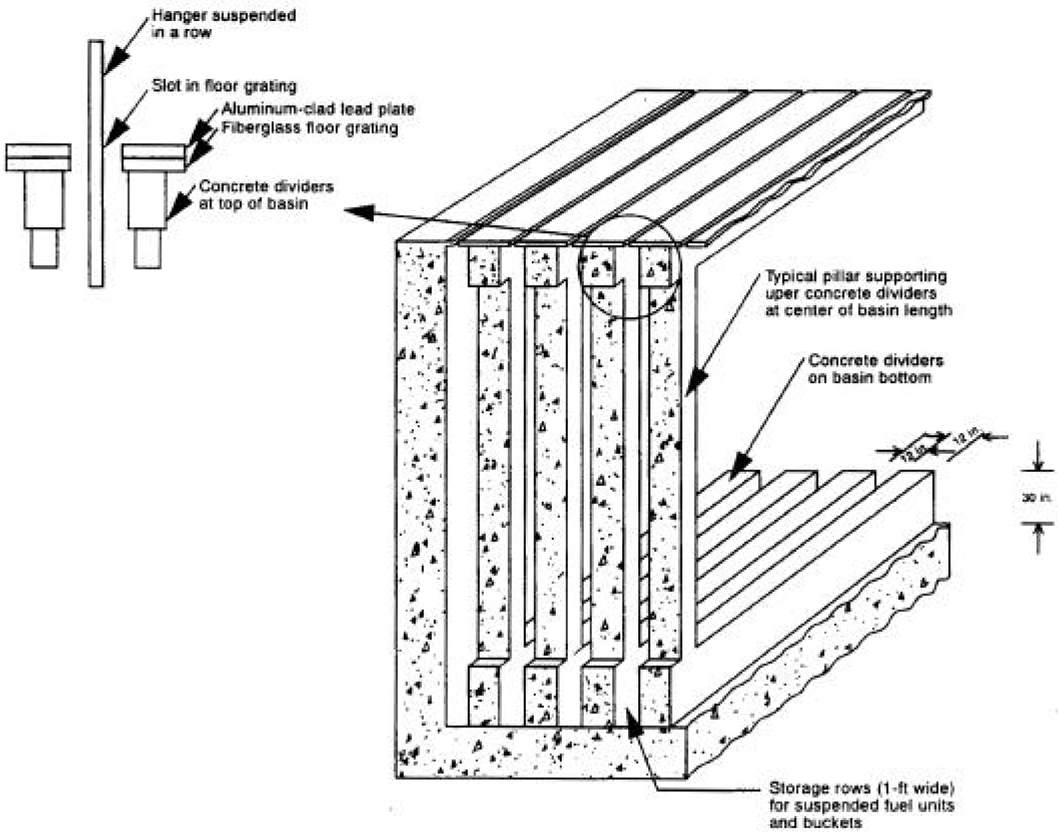

Figure 2. North/Middle Basin Concrete Dividers.

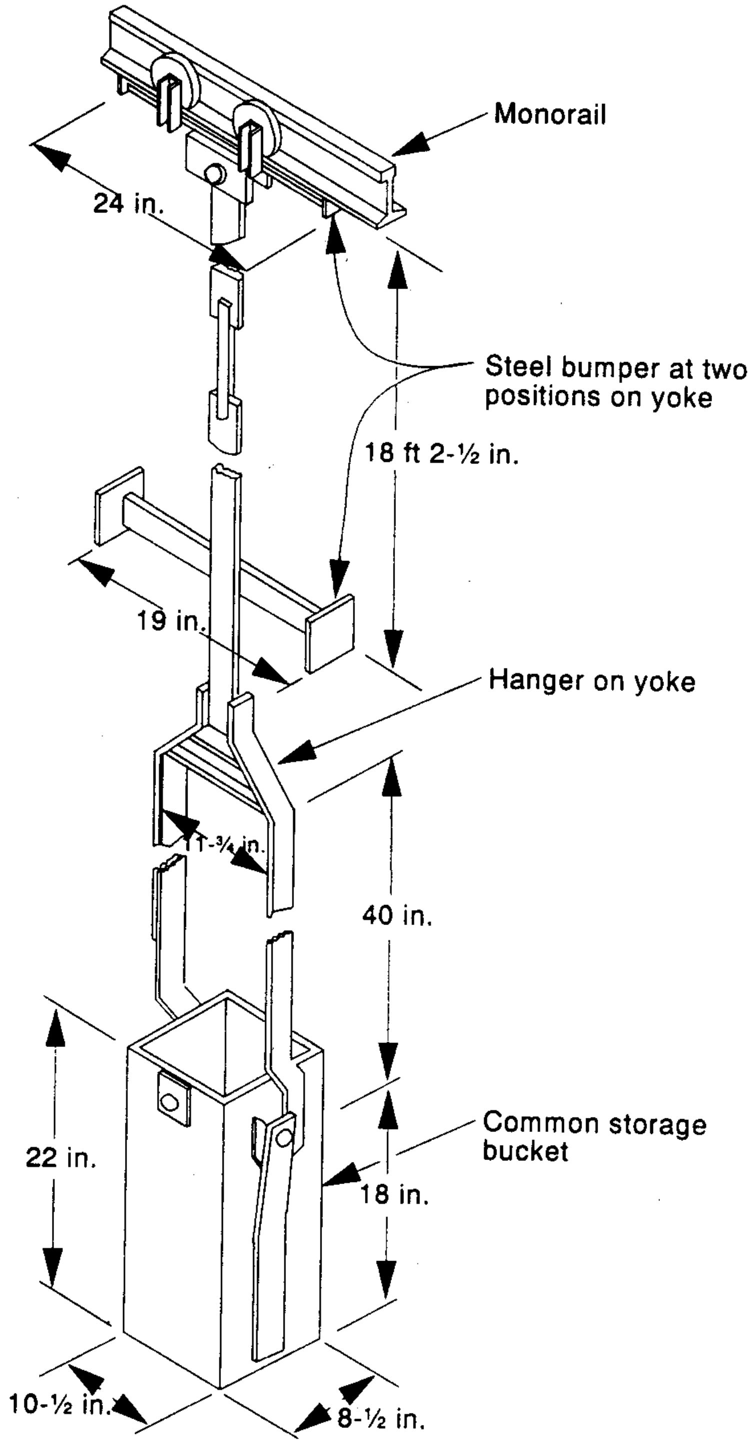

Figure 3. Example of Monorail Hangar – STR with Typical Storage Bucket.

3.0 REQUIREMENTS DOCUMENTATION

No unique requirements are applicable to this evaluation.

4.0 METHODOLOGY

The calculations listed in this evaluation were performed using the CSAS25 module of the SCALE4 package. The CSAS25 module contains three programs that were utilized, BONAMI-II, NITAWL-II, and KENO V.a.[2]

BONAMI-II and NITAWL-II are cross section processing codes that are utilized to create resonance corrected problem dependent cross sections for use in the KENO V.a program.

The system was modeled using the 3-dimensional neutron transport code KENO V.a, as found in the SCALE4 package. KENO V.a is a Monte Carlo based computational method that can be used in the determination of effective multiplication factors, k_{eff}, of various systems. KENO V.a can utilize the problem dependent cross sections, as mentioned previously, in the determination of k_{eff}. Various geometrical configurations can be modeled in KENO V.a.

The 27 group ENDF/B-IV cross section library, used for this evaluation, was collapsed from the 218 group ENDF/B-IV library. This library contains 13 thermal energy groups (below 3ev) and 14 epithermal and fast groups (above 3 ev).

All calculations listed in this evaluation were performed on a Data General Avion Series 500 or Series 400 workstation operating under Data General Unix Version 5.4. The SCALE4 package was compiled using the Green Hills Fortran compiler, Version 1.8.5.

As part of the validation for this report a calculation was performed, using the above described computer platform and code package, on an actual critical experiment performed in the Borax V reactor.[3] As outlined in “BORAX V Neutronics Report,” (ANL-6964)[3], an attempt to create a critical configuration with the peripheral BORAX V Superheater fuel assemblies was made. At the time of the attempt only sixteen peripheral assemblies existed. The sixteen peripheral assemblies were configured in a 4 x 4 array as shown in Figure 5. As shown by the results given in that report this array was slightly subcritical. Estimation by the inverse count rate curves showed that seventeen assemblies would be critical and eighteen assemblies would be supercritical.

The resulting multiplication factor for the 4 x 4 array was calculated to be $k_{eff} \pm 1s = 1.0032 \pm 0.0034$. This array was modeled using KENO V.a along with 27 group ENDF/B-IV cross section library. As discussed previously, this array was shown by experiment to be slightly subcritical, but was shown in the KENO V.a model to be critical thus indicating no code bias compensation is needed for this case.

The second part of the validation for this report involved the modeling of two critical experiments involving arrays of SPERT-D fuel elements. The SPERT-D fuel is a uranium-aluminum plate type fuel. Each fuel element contained about 300 g of U^{235} in twenty two aluminum-clad flat plates. Each fuel plate consisted of a uranium-aluminum alloy core (93.17 weight % U - Al), containing 23.8 w/o uranium, completely clad with 0.02 inches (0.0508 cm) thick type 6061 aluminum. Each fuel plate slid into a set of grooved side plates contained inside a 3 inch (7.62 cm) square, type 6060-T6 aluminum tubing, 27.625 inches (70.1675 cm) long. An example of the SPERT-D fuel is given in Figure 6. A more detailed description of this fuel can be found in Reference 8.

The critical arrays assembled consisted of a combination of full and partial fuel assemblies. Each of the critical arrays were moderated and reflected with demineralized water. Two arrays were modeled for the purposes of validation. The first of the arrays modeled was listed as the 4 x 3.16 x 1 array with the 0.25 inches (0.635 cm) water gap. (The 3.16 indicates that only four of the possible twenty two fuel plates were present in four of the fuel assemblies present in the array.) The effective multiplication factor was calculated to be $k_{eff} \pm 1s = 1.0016 \pm 0.0033$.

The second of the arrays modeled was listed as the 4 x 3.09 x 1 array with the 0.5 inches (1.27 cm) water gap. (The 3.09 indicates that only two of the possible twenty two fuel plates were present in four of the fuel assemblies present in the array.) The effective multiplication factor was calculated to be $k_{eff} \pm 1s = 1.0055 \pm 0.0031$.

As shown by the results listed in these two examples both critical systems were shown to be critical in the KENO V.a models. These results along with the BORAX V Superheater results lead to the conclusion that no bias compensation is needed in this evaluation.

The configuration of the critical systems modeled for the purposes of validation and applicability were similar to the modeled configurations used in this evaluation. The actual critical configurations consisted of arrays of full and partial BORAX-V and SPERT-D fuel elements, as previously described. The models in the evaluation consisted of arrays of BORAX-V fuel elements as described in the text of this report. Also, the SPERT-D fuel elements consisted of a uranium stainless steel fuel matrix similar to the BORAX-V elements.

In addition to the described critical experiments, various other critical experiments[9,10,11] were modeled in an attempt to validate the applicability bounds of the previously described code and cross section package. The critical experiments consisted of various uranium forms (metal, oxide, and liquid), moderator ratios, reflector materials, and single units and various array sizes.

Concrete reflection existed in some of the critical examples which, although are not exact comparisons, can be used to validate the applicability of the code and cross section package for reflected systems as modeled in this evaluation.

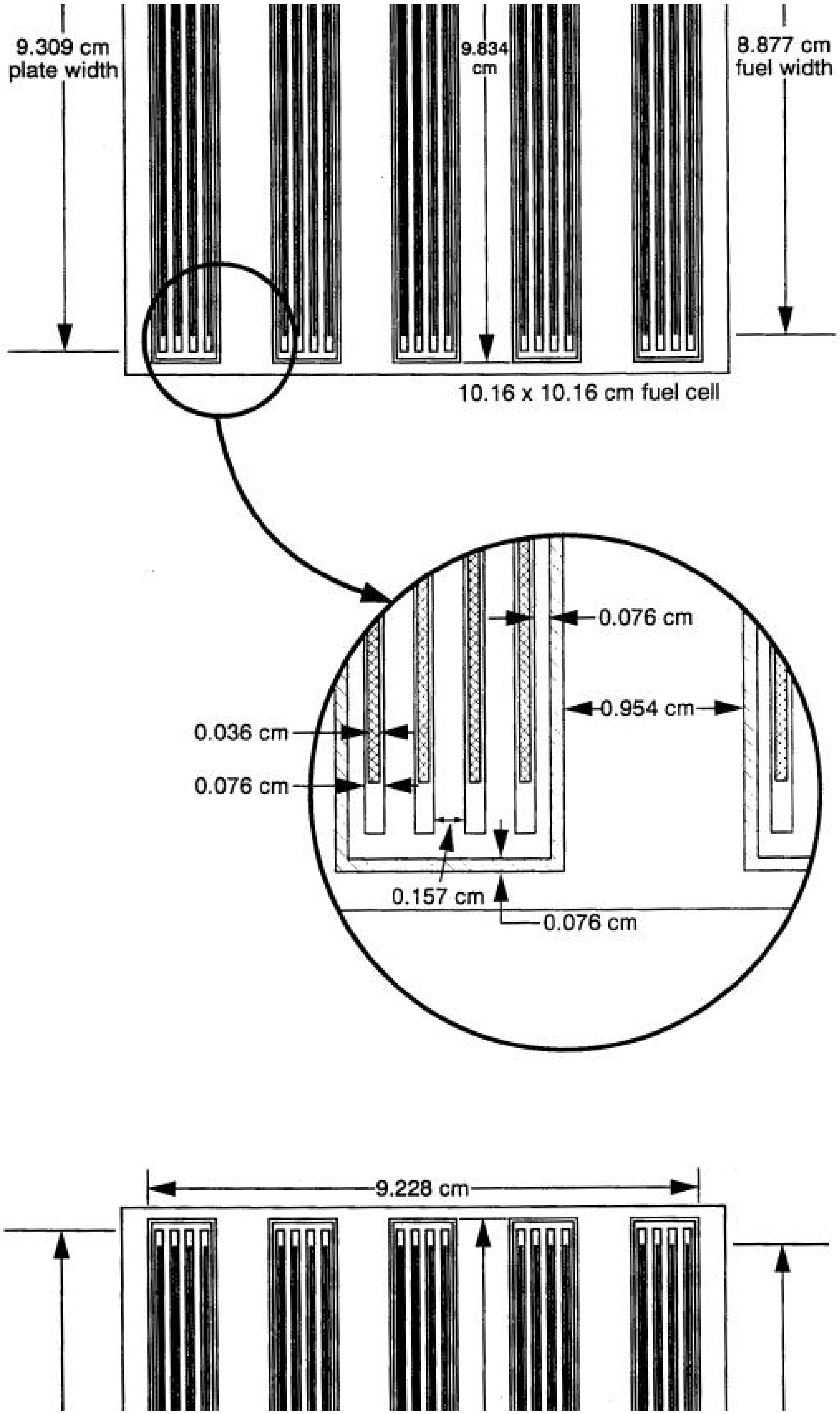

Figure 4. BORAX V Superheater Fuel Assembly.

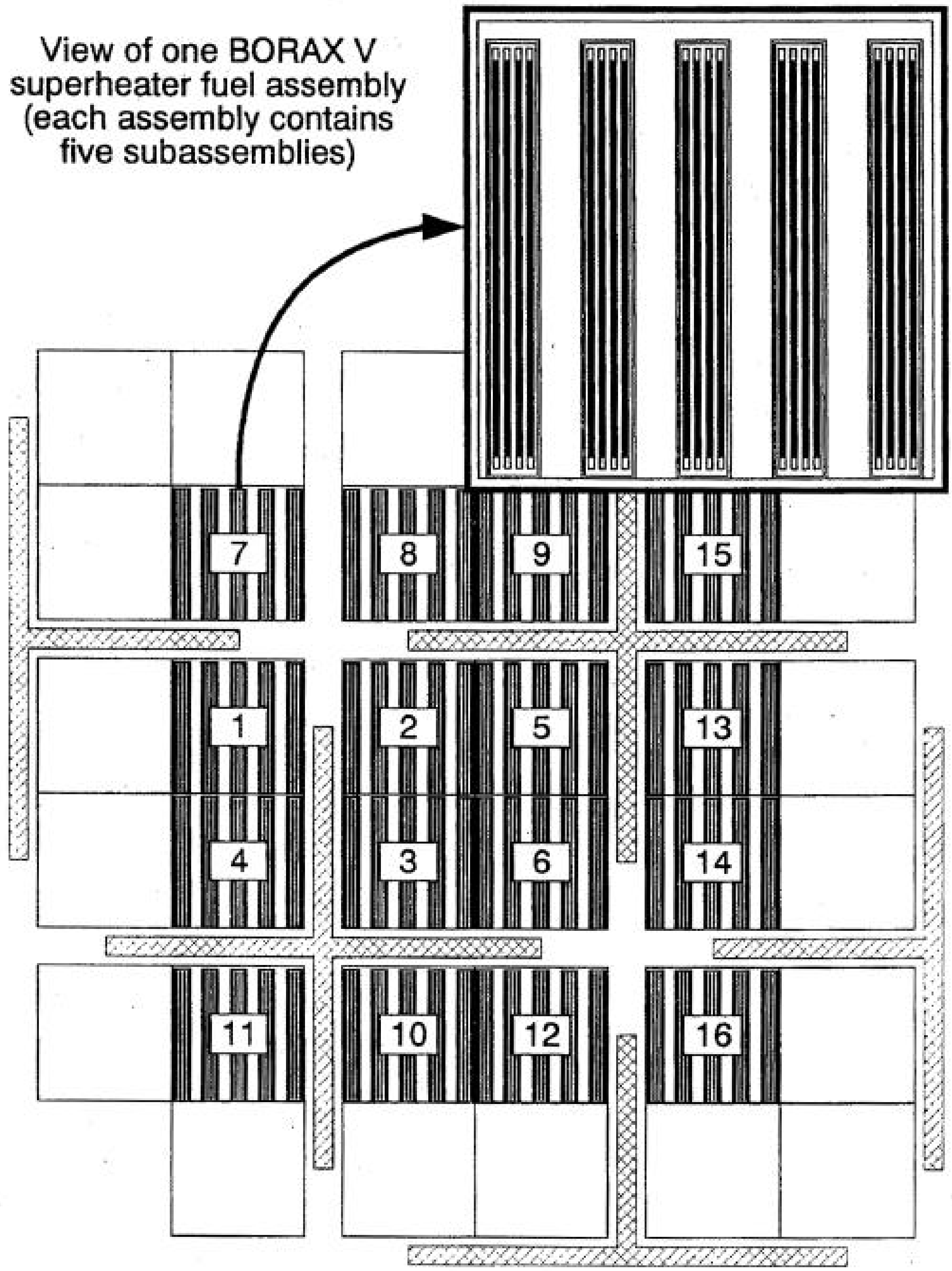

Figure 5. 4 x 4 Array of BORAX V Superheater Fuel Assemblies[3].

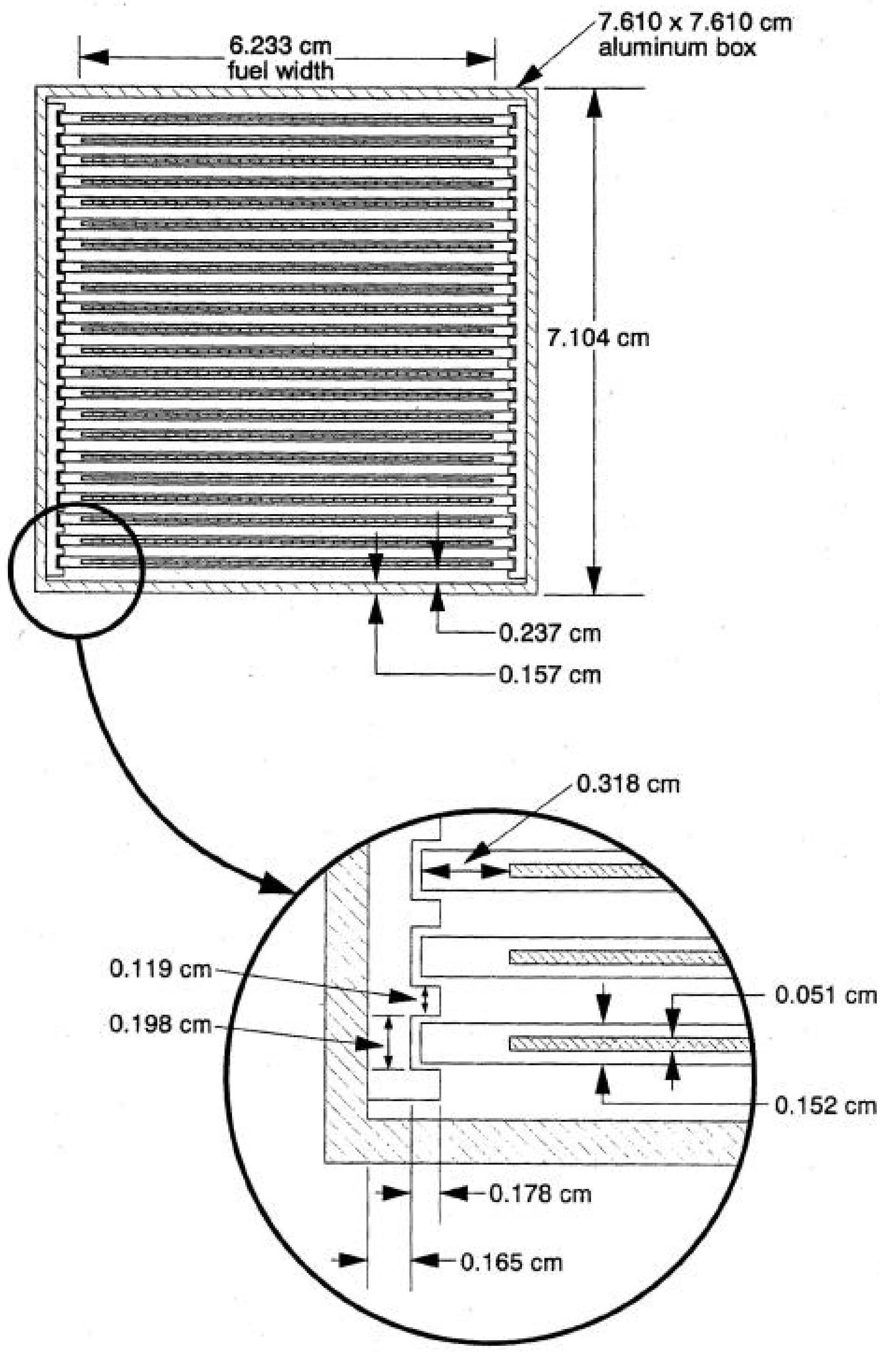

Figure 6. SPERT-D Fuel Element.

5.0 DISCUSSION OF CONTINGENCIES

A detailed outline of the various contingencies supporting criticality safety can be found in the Plant Safety Document (PSD), Section 4.6, “Underwater Fuel Storage”.[1] In this section various contingencies are specifically called out along with the results from, and references to, criticality safety analyses supporting these contingencies.

6.0 EVALUATION AND RESULTS

As discussed previously, the purpose of this evaluation was to determine the reactivity of a single row of storage buckets placed on the floor of a storage channel in the middle or north basin of CPP 603. The middle and north basins consist of a series of channels, 12 inches in width (30.48 cm), through which the monorail system runs. As shown in Figure 2[1], each channel is separated by a concrete divider that is 12 inches in width (30.48 cm) and 30 inches high (76.2 cm). Only buckets containing BORAX V Superheater fuel assemblies are considered in this evaluation.

The BORAX V superheater fuel is a UO_2-stainless steel cermet fuel, consisting of highly enriched uranium (93 weight % U^{235}) dispersed in a stainless steel matrix. A superheater fuel element is made of five subassemblies of fuel plates with intervening water gaps. Each subassembly contains four fuel plates. The dimension of each fuel plate is listed as 3.665 inches x 0.030 inches x 25.25 inches (9.309 cm x 0.0762 cm x 64.14 cm), with the fuel meat thickness of 0.014 inches (0.03566 cm) and a stainless steel (304L) cladding thickness of 0.008 inches (0.02032 cm). Each fuel plate is spaced 0.062 inches (0.1575 cm) apart. The water volume fraction of a fuel assembly is given as 64.5%. A main underlying assumption of this evaluation is that the BORAX V Superheater fuel has maintained its structural integrity such that its physical form does not grossly deviate from that listed in this evaluation. Past basin history indicates that the stainless steel material which has been stored in the CPP-603 basin environment has good structural integrity.

As stated previously, each subassembly consisted of four fuel plates. Each subassembly has an insulating air gap of 0.030 inches (0.0762 cm) between the outer fuel plates and the outer cladding. The dimensions of a subassembly are given as 3.665 inches x 0.426 inches x 25.25 inches (9.309 cm x 1.082 cm x 64.140 cm). A full BORAX V Superheater fuel assembly consists of five subassemblies and has dimensions given as 3.633 inches x 3.875 inches x 25.462 inches (9.228 cm x 9.843 cm x 64.674 cm). The superheater fuel assembly was modeled as previously described and shown in Figures 4, 5, & 7.

Currently 34 BORAX V Superheater Fuel Elements are in inventory at the ICPP. This inventory consists of a combination of central and peripheral fuel elements, each at a U^{235} enrichment of approximately 93%. The maximum U^{235} loading per assembly is listed as 679 grams for a peripheral assembly and 429 grams per central assembly. Several series of calculations were performed to determine reactivity effects. The calculational models considered in this evaluation consisted only of the highly loaded peripheral assemblies.

Evaluated in the first series of cases were various arrays surrounded only by water. This was done to determine the minimum number of assemblies needed to achieve a critical system. A pictorial representation of reactivity as a function of array size is given in Figure 8. Each block represents one BORAX V Superheater fuel assembly. As shown in Figure 8, a 3 x 3 x 1 array of fuel assemblies exceeded the k_{eff}+ 2s > 0.95 safety limit criteria and a 3 x 4 x 1 array is shown to be critical with k_{eff}+ 2s > 1.0.

Considered in the next series of calculations were arrays that were reflected not only by water but also by concrete. Due to the physical dimensions of the fuel assemblies and the width of the concrete channel walls, it was only possible to store fuels in arrays that were either one or two assembly units wide. In this series of calculations, in which the concrete[12] channel walls and basin floors were modeled, various arrangements of fuel assemblies were evaluated. These arrangements are shown in Figure 9. As shown in Figure 9, a 2 x infinite x 1 array of fuel elements placed in a water filled concrete storage channel gives a k_{eff}+ 2s = 1.0386. Current technical specifications (TS/S 4.6A1) limit each storage bucket to four BORAX V Superheater fuel assemblies. Current inventory fuel storage maps indicate that no more than three assemblies are stored per bucket. Since only three fuel assemblies are stored per bucket, a more realistic configuration is given at the bottom of Figure 9. As shown in this case, the fuel assemblies are arranged in a staggered configuration. The result calculated in this case yields an effective multiplication factor of k_{eff}+ 2s = 0.9340, which is within the ICCP acceptance criteria.

A revision to the current technical specification, TS/S 4.6A1, will allow only three assemblies per bucket when stored on the basin floor. This revision will redefine approved storage to allow these buckets to be stored on the floor of a storage channel with no spacing restriction between buckets.

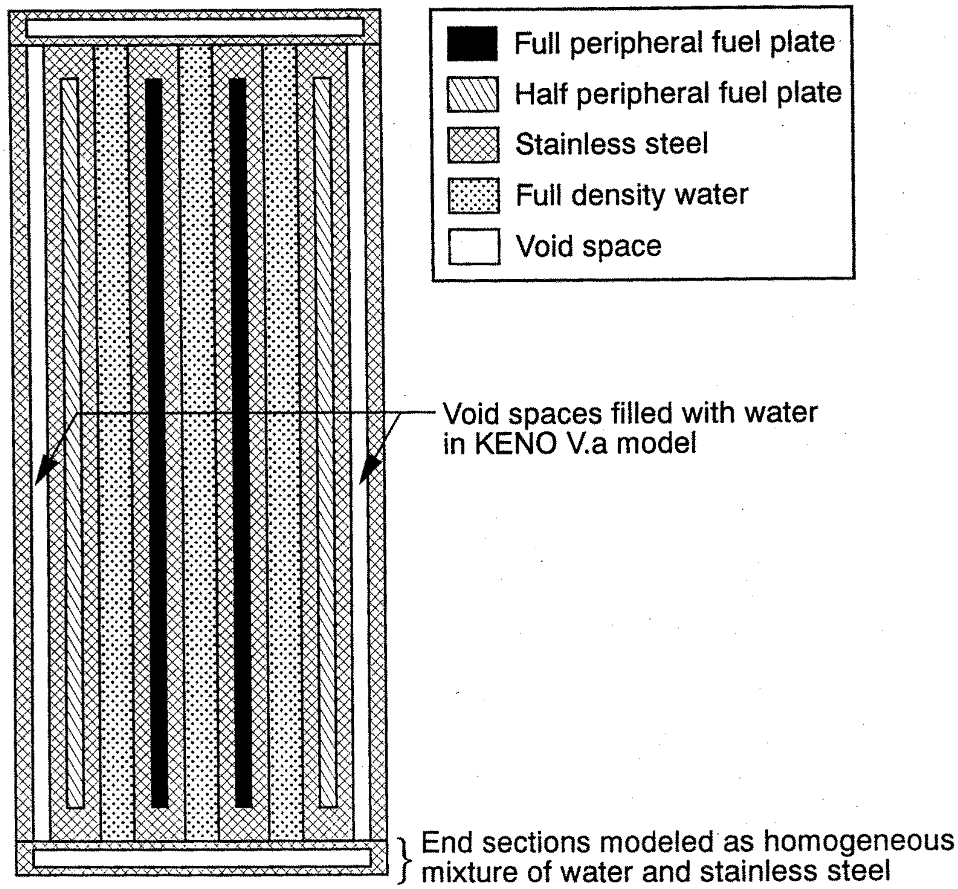

Figure 7. BORAX V Superheater Fuel Subassembly.

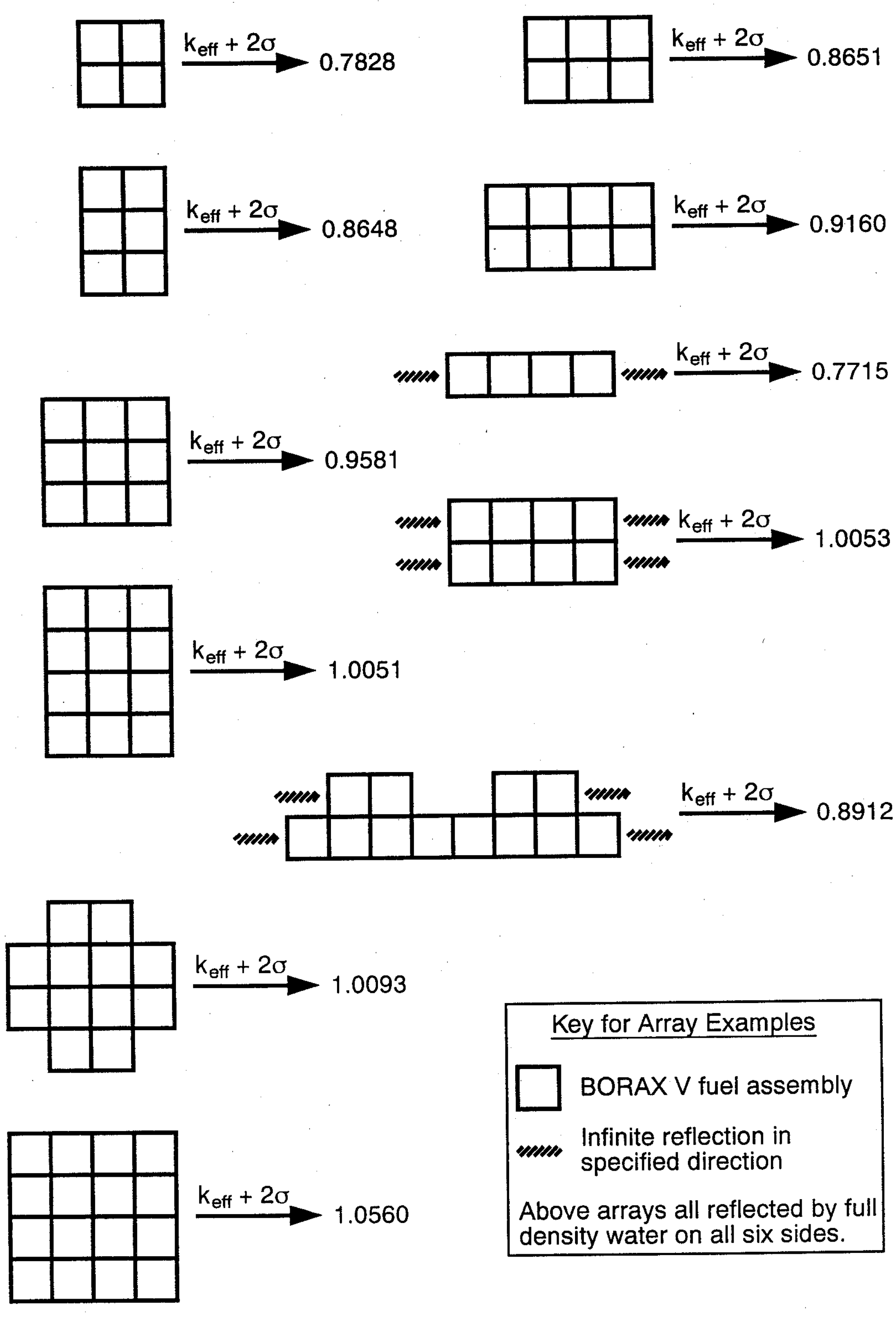

Figure 8. Various Water Reflected BORAX V Fuel Assembly Arrays.

In all calculations, the actual storage buckets were ignored, thus allowing the assemblies to be placed side by side. In reality, a spacing device exists that is connected to the monorail hanger, which provides a small amount of spacing between the assemblies. For the purposes of this evaluation, this spacing was ignored.

The last configurations to be evaluated were those in which the spacing between the fuel assemblies was varied to determine reactivity effects. For this set of calculations the 3 x 3 x 1 water reflected array was modeled. The distance between the fuel assemblies was varied from 0 cm to 1.0 cm by 0.1 cm increments and from 1.0 cm to 4.0 cm by 1.0 cm increments. An example of this configuration is shown in Figure 10, with X being the distance between fuel assemblies. The results of these calculations are given in Table 1. As shown by these results, the increase in spacing between fuel assemblies had an indistinguishable effect on reactivity. This result was expected due to the large water volume fraction of a BORAX V Superheater fuel assembly and already present moderator gaps inherent to the design.

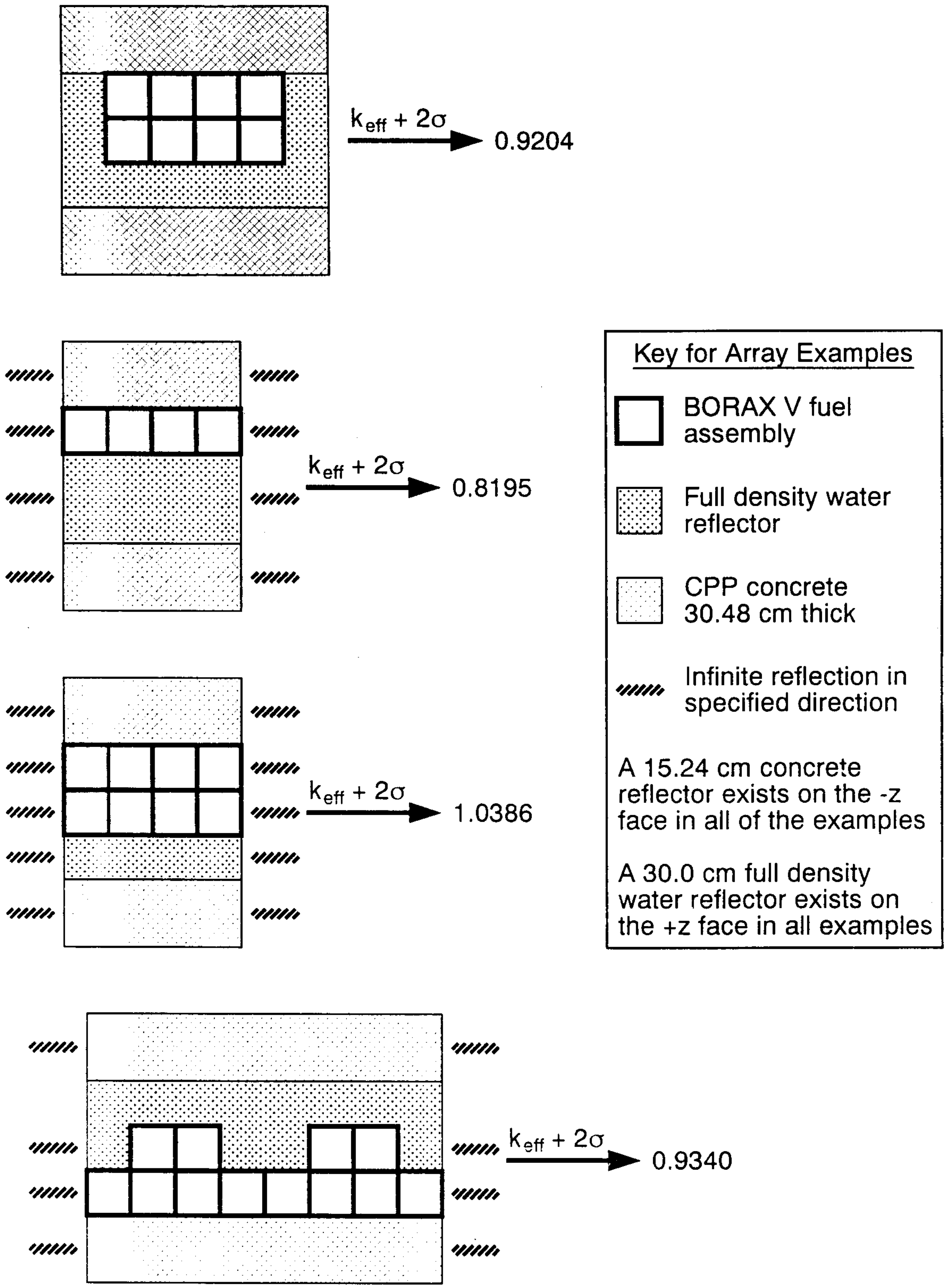

Figure 9. X-Y View of Various Water & Concrete Reflected BORAX V Assembly Arrays.

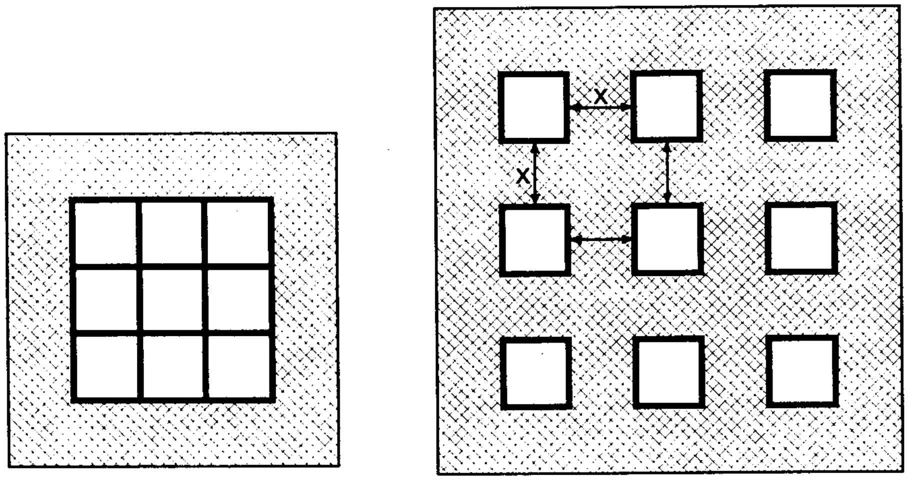

Figure 10. 3 x 3 Array of BORAX V Superheater Fuel Assemblies with Varied Spacing Between Assemblies.

Table 1. Reactivity as a Function of distance Between BORAX V Fuel Assemblies.

Distance Between Fuel Assemblies	K_{eff} + 2s
0.0 cm	0.9581
0.1 cm	0.9572
0.2 cm	0.9605
0.3 cm	0.9601
0.4 cm	0.9603
0.5 cm	0.9498
0.6 cm	0.9649
0.7 cm	0.9541
0.8 cm	0.9590
0.9 cm	0.9499
1.0 cm	0.9508
2.0 cm	0.9222
3.0 cm	0.8699
4.0 cm	0.8248

7.0 DESIGN FEATURES AND ADMINISTRATIVE LIMITS

The following design features and administratively controlled limits are necessary to ensure the criticality safety of the system. Operation safety limits are set in the Plant Safety Document (PSD) and the Technical Specifications (TS/S) associated with that operation.

7.1 Design Features

1. The BORAX V Superheater fuel has uranium gram loadings and enrichments at or below those listed in this evaluation.

2. The BORAX V Superheater fuel has maintained its structural integrity, due to its material compositions, such that its physical form does not grossly deviate from that listed in this evaluation.

7.2 Administrative Limits

1. Each storage bucket stored on the basin floor will be limited to three BORAX V Superheater fuel assemblies.

2. Storage buckets suspended from the monorail system will be limited to four Borax V Superheater assemblies provided the buckets are preventively rigged to avoid dropping to the basin floor. If the remaining hanging buckets are not preventively rigged, the limit per bucket will be set at three BORAX V assemblies.

3. A row containing storage buckets placed on the floor will be limited to buckets containing only BORAX V Superheater fuel assemblies. No other fuel types will be allowed in this row.

8.0 SUMMARY AND CONCLUSIONS

The summary and conclusions from this evaluation are given below:

- Storage buckets that contain three (3) BORAX V Superheater fuel assemblies may be stored on the floor of the north and middle basins without any restrictions upon spacing between the storage buckets.

- Only BORAX V Superheater peripheral and central assemblies were considered in the evaluation. The storage of a single row of buckets containing BORAX V Superheater fuel assemblies was evaluated. Interaction with other fuel types within the row was not considered since other fuel types in a single row will be restricted through administrative controls.

9.0 REFERENCES

1. Idaho Chemical Processing Plant Safety Document, Underwater Fuel Storage, Section 4.6.

2. Staff of Technical Applications, Computing and Telecommunications Division at ORNL, SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluations, Vols. 1-3, CCC-545.

3. J. I. Hagen and R. W. Goin (Argonne National Laboratory), BORAX V Neutronics, ANL-6964, December 1965.

4. W. C. Kramer and C. H. Bean (Argonne National Laboratory), Fabrication of UO2-Stainless Steel Dispersion Fuel For BORAX V Nuclear Superheat, ANL-6649, December 1963.

5. (Argonne National Laboratory), Design and Hazards Summary Report Boiling Reactor Experiment V (BORAX V), ANL-6302, May 1961.

6. BORAX-V Project Staff (Argonne National Laboratory), Experiments with Central Superheater Core CSH-1, ANL-6961, January 1965.

7. K. E. Plumlee, Q. L. Baird, G. S. Stanford, and P. I. Amundson (Argonne National Laboratory), Critical Experiment with BORAX V Internal Superheater, ANL-6691, December 1965.

8. E. B. Johnson and R. K. Reedy (Oak Ridge National Laboratory), Critical Experiments With SPERT-D Fuel Elements, ORNL-TM-1207, July 14, 1965.

9. B. M. Palmer, C. Crawford, Validation of the CSAS25 Module of Scale, Part I: Highly Enriched Uranium Solutions, CSS-91-005, Westinghouse Idaho Nuclear Company, July 1991.

10. B. M. Palmer, C. Crawford, Validation of the CSAS25 Module of Scale, Part II: Highly Enriched Uranium Solutions, CSS-91-013, Westinghouse Idaho Nuclear Company, September 1991.

11. B. M. Palmer, C. Crawford, Validation of the CSAS25 Module of Scale, Part III: Highly Enriched Uranium Solutions, CSS-91-015, Westinghouse Idaho Nuclear Company, October 1991.

12. Letter from J. E. Tanner to G. T. Paulson, Subject: “Best CPP Concrete Formula,” JET-03-91, September 9, 1991.

APPENDICES

A. Number Density Calculations

UO2 - Stainless Steel Fuel Matrix - Full Peripheral Fuel Plates

Nuclide	Atom Density	Nuclide I.D.
U^{235}	5.8990×10^{-3}	92235
U^{238}	4.3840×10^{-4}	92238
O	1.2967×10^{-2}	8016
Cr	1.2918×10^{-3}	24304
Mn	4.4162×10^{-2}	25055
Fe	5.7442×10^{-3}	26304
Ni	1.2670×10^{-2}	28304

\--
UO2 - Stainless Steel Fuel Matrix - Half Peripheral Fuel Plates

Nuclide	Atom Density	Nuclide I.D.
U^{235}	3.1430×10^{-3}	92235
U^{238}	2.3360×10^{-4}	92238
O	1.5036×10^{-2}	8016
Cr	1.4979×10^{-3}	24304
Mn	5.1208×10^{-2}	25055
Fe	6.6607×10^{-3}	26304
Ni	6.7530×10^{-2}	28304

304 Stainless Steel (Density = 7.92 g/cc)

Nuclide	Atom Density	Nuclide I.D.
Cr	1.7428×10^{-2}	24304
Mn	1.7363×10^{-3}	25055
Fe	5.9358×10^{-2}	26304
Ni	7.7207×10^{-3}	28304

Water (Density = 0.9982 g/cc)

Nuclide	Atom Density	Nuclide I.D.
H	6.6751×10^{-2}	1001
O	3.3375×10^{-2}	8016

B. Sample KENO V.a Inputs

Sample Input 1- Example of Single BORAX V Assembly

```
=CSAS25
BORAX
27G  INF
U-235       1 0 5.899-03 END
U-238       1 0 4.384-04 END
SS304       1 0.7440     END
O           1 0 1.267-02 END
U-235       2 0 3.143-03 END
U-238       2 0 2.336-04 END
SS304       2 0.8627     END
O           2 0 6.753-03 END
SS304       3 1.0        END
H2O         4 1.0        END
SS304       5 0.6        END
H2O         5 0.4        END
END COMP
KENOV.A
READ PARAM
  GEN=153   NPG=300
END PARAM
READ GEOM
  UNIT 1
COM=*0.03 IN INSIDE FUEL PLATE*
    CUBOID  1 1 0.03556 0.0 8.8773 0.0 60.96 0.0
    CUBOID  3 1 0.05588 -0.02032 9.09315 -0.21585 62.55 -1.59
  UNIT 2
COM=*0.03 IN OUTSIDE FUEL PLATE*
    CUBOID  2 1 0.03556 0.0 8.8773 0.0 60.96 0.0
    CUBOID  3 1 0.05588 -0.02032 9.09315 -0.21585 62.55 -1.59
  UNIT 3
COM=*0.062 IN SPACE BETWEEN PLATES*
    CUBOID  4 1 0.15748 0.0 9.09315 -0.21585 62.55 -1.59
  UNIT 4
COM=*0.03 IN INSULATING GAP*
    CUBOID  4 1 0.07620 0.0 9.09315 -0.21585 62.55 -1.59
  UNIT 5
COM=*0.03 IN SIDE PLATES*
    CUBOID  3 1 0.07620 0.0 9.09315 -0.21585 62.55 -1.59
  UNIT 6
COM=*0.37575 IN MODERATOR GAP*
    CUBOID  4 1 0.95441 0.0 9.35990 -0.48260 62.55 -1.59
  UNIT 7
ARRAY 1 3*0.0
REFLECTOR 5 1 2*0.0 2*0.26675 2*0.0 1
  UNIT 8
ARRAY 2 3*0.0
  GLOBAL
ARRAY 3 3*0.0
REFLECTOR 4 2 6*3.0 10
END GEOM
READ BIAS ID=500 2 11 END BIAS
READ ARRAY ARA=1 NUX=11 NUY=1 NUZ=1
FILL
5 4 2 3 1 3 1 3 2 4 5
END FILL
ARA=2 NUX=9 NUY=1 NUZ=1
FILL
7 6 7 6 7 6 7 6 7
END FILL
ARA=3 NUX=1 NUY=1 NUZ=1
FILL
8
END FILL
END ARRAY
END DATA
END
```

Sample Input 2 - Example of BORAX V Array With Concrete Reflector Present

```
=CSAS25
BORAX
27G  INF
U-235     1 0 5.899-03   END
U-238     1 0 4.384-04   END
SS304     1 0.7440       END
O         1 0 1.267-02   END
U-235     2 0 3.143-03   END
U-238     2 0 2.336-04   END
SS304     2 0.8627       END
O         2 0 6.753-03   END
SS304     3 1.0          END
H2O       4 1.0          END
SS304     5 0.6          END
H2O       5 0.4          END
C         6 0  1.6000-3  END
O         6 0  3.9700-2  END
NA        6 0  5.5000-4  END
AL        6 0  1.6000-3  END
SI        6 0  1.5200-2  END
S         6 0  5.0000-5  END
CA        6 0  3.1000-3  END
FE        6 0  3.8000-4  END
H2O       6  DEN=0.114   END
END COMP
KENOV.A
READ PARAM
  GEN=153   NPG=300
END PARAM
READ GEOM
  UNIT 1
COM=*0.03 IN INSIDE FUEL PLATE*
    CUBOID  1 1 0.03556 0.0 8.8773 0.0 60.96 0.0
    CUBOID  3 1 0.05588 -0.02032 9.09315 -0.21585 62.55 -1.59
  UNIT 2
COM=*0.03 IN OUTSIDE FUEL PLATE*
    CUBOID  2 1 0.03556 0.0 8.8773 0.0 60.96 0.0
    CUBOID  3 1 0.05588 -0.02032 9.09315 -0.21585 62.55 -1.59
  UNIT 3
COM=*0.062 IN SPACE BETWEEN PLATES*
    CUBOID  4 1 0.15748 0.0 9.09315 -0.21585 62.55 -1.59
  UNIT 4
COM=*0.03 IN INSULATING GAP*
    CUBOID  4 1 0.07620 0.0 9.09315 -0.21585 62.55 -1.59
  UNIT 5
COM=*0.03 IN SIDE PLATES*
    CUBOID  3 1 0.07620 0.0 9.09315 -0.21585 62.55 -1.59
  UNIT 6
COM=*0.37575 IN MODERATOR GAP*
    CUBOID  4 1 0.95441 0.0 9.35990 -0.48260 62.55 -1.59
  UNIT 7
ARRAY 1 3*0.0
REFLECTOR 5 1 2*0.0 2*0.26675 2*0.0 1
  UNIT 8
```

```
ARRAY 2 3*0.0
  UNIT 9
    CUBOID  4 1 9.2278 0.0 9.8425 0.0 64.14 0.0
  GLOBAL
ARRAY 3 3*0.0
REFLECTOR 6 1 0.0 0.0    30.48  0.0      0.0  0.0   1
REFLECTOR 4 1 0.0 0.0      0.0  10.7900   0.0  0.0   1
REFLECTOR 6 1 0.0 0.0      0.0  30.48     0.0 15.24  1
REFLECTOR 4 2  2*0.0  3*3.0  0.0   10
END GEOM
READ BOUNDS
XFC=MIRROR
END BOUNDS
READ BIAS ID=500 2 11 END BIAS
READ ARRAY ARA=1 NUX=11 NUY=1 NUZ=1
FILL
5 4 2 3 1 3 1 3 2 4 5
END FILL
ARA=2 NUX=9 NUY=1 NUZ=1
FILL
7 6 7 6 7 6 7 6 7
END FILL
ARA=3 NUX=2 NUY=2 NUZ=1
FILL
9 8
8 8
END FILL
END ARRAY
END DATA
END
```

Sample Input 3 - Example of BORAX V Superheater Fuel Critical Experiment

```
=CSAS25
BORAX
27G  INF
U-235        1 0 5.899-03  END
U-238        1 0 4.384-04  END
SS304        1 0.7440      END
O            1 0 1.267-02  END
U-235        2 0 3.143-03  END
U-238        2 0 2.336-04  END
SS304        2 0.8627      END
O            2 0 6.753-03  END
SS304        3 1.0         END
H2O          4 1.0         END
SS304        5 0.6         END
H2O          5 0.4         END
AL           6 1.0         END
END COMP
KENOV.A
READ PARAM
  GEN=153   NPG=300
END PARAM
READ GEOM
  UNIT 1
COM=*0.03 IN INSIDE FUEL PLATE*
    CUBOID  1 1 0.03556 0.0 8.8773 0.0 60.96 0.0
    CUBOID  3 1 0.05588 -0.02032 9.09315 -0.21585 62.55 -1.59
  UNIT 2
COM=*0.03 IN OUTSIDE FUEL PLATE*
    CUBOID  2 1 0.03556 0.0 8.8773 0.0 60.96 0.0
    CUBOID  3 1 0.05588 -0.02032 9.09315 -0.21585 62.55 -1.59
  UNIT 3
COM=*0.062 IN SPACE BETWEEN PLATES*
    CUBOID  4 1 0.15748 0.0 9.09315 -0.21585 62.55 -1.59
  UNIT 4
COM=*0.03 IN INSULATING GAP*
    CUBOID  0 1 0.07620 0.0 9.09315 -0.21585 62.55 -1.59
  UNIT 5
COM=*0.03 IN SIDE PLATES*
    CUBOID  3 1 0.07620 0.0 9.09315 -0.21585 62.55 -1.59
  UNIT 6
COM=*0.37575 IN MODERATOR GAP*
    CUBOID  4 1 0.95441 0.0 9.35990 -0.48260 62.55 -1.59
  UNIT 7
ARRAY 1 3*0.0
REFLECTOR 5 1 2*0.0 2*0.26675 2*0.0 1
  UNIT 8
ARRAY 2 3*0.0
REFLECTOR 4 1 2*0.46609 2*0.15875 2*0.0 1
  UNIT 9
COM=*WATER INSTEAD OF FUEL ASSEMBLY*
    CUBOID  4 1 10.16002 0.0 9.67740 -0.48260 62.55 -1.59
  UNIT 10
COM=*CENTER FOUR*
ARRAY 3 3*0.0
REFLECTOR 6 1 4*0.396875 2*0.0 1
REFLECTOR 4 1 4*0.238125 2*0.0 1
REFLECTOR 6 1 4*0.396875 2*0.0 1
  UNIT 11
ARRAY 4 3*0.0
REFLECTOR 6 1 4*0.396875 2*0.0 1
REFLECTOR 4 1 4*0.238125 2*0.0 1
REFLECTOR 6 1 4*0.396875 2*0.0 1
  UNIT 12
ARRAY 5 3*0.0
REFLECTOR 6 1 4*0.396875 2*0.0 1
REFLECTOR 4 1 4*0.238125 2*0.0 1
REFLECTOR 6 1 4*0.396875 2*0.0 1
  UNIT 13
ARRAY 6 3*0.0
REFLECTOR 6 1 4*0.396875 2*0.0 1
REFLECTOR 4 1 4*0.238125 2*0.0 1
REFLECTOR 6 1 4*0.396875 2*0.0 1
  UNIT 14
ARRAY 7 3*0.0
REFLECTOR 6 1 4*0.396875 2*0.0 1
REFLECTOR 4 1 4*0.238125 2*0.0 1
REFLECTOR 6 1 4*0.396875 2*0.0 1
  UNIT 15
ARRAY 8 3*0.0
REFLECTOR 6 1 4*0.396875 2*0.0 1
REFLECTOR 4 1 4*0.238125 2*0.0 1
REFLECTOR 6 1 4*0.396875 2*0.0 1
  UNIT 16
ARRAY 9 3*0.0
REFLECTOR 6 1 4*0.396875 2*0.0 1
REFLECTOR 4 1 4*0.238125 2*0.0 1
REFLECTOR 6 1 4*0.396875 2*0.0 1
  UNIT 17
ARRAY 10 3*0.0
REFLECTOR 6 1 4*0.396875 2*0.0 1
REFLECTOR 4 1 4*0.238125 2*0.0 1
REFLECTOR 6 1 4*0.396875 2*0.0 1
  UNIT 18
ARRAY 11 3*0.0
REFLECTOR 6 1 4*0.396875 2*0.0 1
REFLECTOR 4 1 4*0.238125 2*0.0 1
REFLECTOR 6 1 4*0.396875 2*0.0 1
GLOBAL
ARRAY 12 3*0.0
REFLECTOR 4 2 6*3.0 10
END GEOM
READ BIAS ID=500 2 11 END BIAS
READ ARRAY ARA=1 NUX=11 NUY=1 NUZ=1
FILL
5 4 2 3 1 3 1 3 2 4 5
END FILL
ARA=2 NUX=9 NUY=1 NUZ=1
FILL
7 6 7 6 7 6 7 6 7
END FILL
ARA=3 NUX=2 NUY=2 NUZ=1
FILL
8 8
8 8
END FILL
ARA=4 NUX=2 NUY=2 NUZ=1
FILL
9 9
9 8
END FILL
ARA=5 NUX=2 NUY=2 NUZ=1
FILL
9 9
8 8
END FILL
ARA=6 NUX=2 NUY=2 NUZ=1
FILL
9 9
8 9
END FILL
ARA=7 NUX=2 NUY=2 NUZ=1
FILL
8 9
8 9
END FILL
ARA=8 NUX=2 NUY=2 NUZ=1
FILL
8 9
9 9
END FILL
ARA=9 NUX=2 NUY=2 NUZ=1
FILL
8 8
9 9
END FILL
ARA=10 NUX=2 NUY=2 NUZ=1
FILL
```

```
9 8
9 9
END FILL
ARA=11 NUX=2 NUY=2 NUZ=1
FILL
9 8
9 8
END FILL
ARA=12 NUX=3 NUY=3 NUZ=1
FILL
11 12 13
18 10 14
17 16 15
END FILL
END ARRAY
END DATA
END
```

Sample Input 4 - Example of SPERT-D Fuel Critical Experiment

```
=CSAS25
SPERT #2
27G  INF
U-235        1 0 1.849-03 END
U-238        1 0 1.339-04 END
AL           1 0.9189    END
AL           2 1.0  END
H2O          3 1.0  END
END COMP
KENOV.A
READ PARAM
  GEN=153   NPG=500
END PARAM
READ GEOM
  UNIT 1
COM=*0.06 IN FUEL PLATE*
    CUBOID  1 1 6.23316 0.0 0.0508 0.0 60.96 0.0
    CUBOID  2 1 6.41096 -0.17780 0.10160 -0.0508 62.3888
-1.42875
  UNIT 2
COM=*0.0645714 IN GAP BETWEEN PLATES*
    CUBOID  3 1 6.41096 -0.17780 0.16401 0.0 62.3888 -1.42875
  UNIT 3
COM=*0.06 IN HUNK OF WATER*
    CUBOID  3 1 6.41096 -0.17780 0.10160 -0.0508 62.3888
-1.42875
  UNIT 4
COM=*COMPLETE ELEMENT*
ARRAY 1 3*0.0
REFLECTOR 3 1 2*0.0 2*0.24892 2*0.0 1
REFLECTOR 2 1 2*0.34290 4*0.0 1
REFLECTOR 2 1 4*0.15748 2*0.0 1
REFLECTOR 3 1 4*0.31750 2*0.0 1
  UNIT 5
COM=*INCOMPLETE ELEMENT*
ARRAY 2 3*0.0
REFLECTOR 3 1 2*0.0 2*0.24892 2*0.0 1
REFLECTOR 2 1 2*0.34290 4*0.0 1
REFLECTOR 2 1 4*0.15748 2*0.0 1
REFLECTOR 3 1 4*0.31750 2*0.0 1
GLOBAL
ARRAY 3 3*0.0
COM=*SURROUNDING WATER*
REFLECTOR 3 2 6*3.0 10
END GEOM
READ BIAS ID=500 2 11 END BIAS
READ ARRAY ARA=1 NUX=1 NUY=43 NUZ=1
LOOP
1 1 1 1 1 43 2 1 1 1
2 1 1 1 2 42 2 1 1 1
END LOOP
ARA=2 NUX=1 NUY=43 NUZ=1
LOOP
1 1 1 1 1 43 1 1 1 1
2 1 1 1 2 42 2 1 1 1
3 1 1 1 9 43 2 1 1 1
END LOOP
ARA=3 NUX=4 NUY=4 NUZ=1
FILL
4 4 4 4
4 4 4 4
4 4 4 4
5 5 5 5
END FILL
END ARRAY
END DATA
END
```

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EXAMPLE 3

Provided by

Los Alamos National Laboratory

CRITICALITY SAFETY EVALUATION OF THE DEVICE FOR THE IMAGINARY EVENT, PIT SHIPMENT AND NEVADA TEST SITE HANDLING

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1.0 INTRODUCTION

The purpose of this evaluation is to provide a documented basis for demonstrating subcriticality during shipment and Nevada Test Site (NTS) handling of the device for the Imaginary Event.

2.0 DESCRIPTION

The device for the Imaginary Event is shown on drawings 18Y-123456 and 18Y-123457. These drawings also contain mass and dimension information sufficient to readily demonstrate that the device poses no unusual criticality concerns and in fact will remain subcritical even in the presence of water internal to and surrounding the device.

For shipment of the pit for this device, the AL-R8[1] packaging is planned.

3.0 REQUIREMENTS DOCUMENTATION

There are no requirements that are unique to this evaluation; however, shipment of the device for the Imaginary Event will be in compliance with the Fissile Class I requirements of 10 CFR Part 71.

4.0 METHODOLOGY

Evaluation of this device will be done by comparison with a previously evaluated and approved device.

5.0 CONTINGENCY ANALYSIS

Assembly and handling of components and the complete device at the NTS do not pose any unusual criticality concerns. Stated differently, no upset conditions or abnormal environments can be foreseen whereby criticality is a credible event.

6.0 EVALUATION AND RESULTS

The most reactive pit condition during normal, planned operations is when it is enclosed in High Explosives (HE). Since water is a better neutron reflector than HE, regardless of the thickness, thick water reflection is a bounding condition for normal operations that permits ease of analysis. The fissile mass in this pit is substantially less than the water reflected critical mass as published in LA-10860-MS[2], Table XYZ.

Internal pit flooding is arguably an incredible upset condition for planned operations at the NTS. However, even this condition, combined with thick water reflection, is easily demonstrated to be far subcritical based on comparisons to critical mass data in Figures XX and YY of LA-10860-MS[2].

For the pit shipment in an AL-R8 container, a comparative analysis demonstrates that a transport index of zero is appropriate for criticality control. By comparing fissile masses and dimensions, it is obvious that the Imaginary Pit is less reactive than the type ABC pit that has been previously analyzed, shown to have a transport index of zero for criticality control, and approved for shipment in the AL-R8[1,3].

7.0 DESIGN FEATURES (PASSIVE & ACTIVE) AND ADMINISTRATIVELY CONTROLLED LIMITS AND REQUIREMENTS

Beyond the design features inherent to the AL-R8, there are no design features or administratively controlled limits that are required to ensure criticality safety during shipment of the device for the Imaginary Event. For operations at the NTS, assembly procedures[4] limit the number of fissile components present at any location such that no credible criticality threat is foreseeable.

8.0 SUMMARY & CONCLUSIONS

For transportation of the pit for the Imaginary Event in the AL-R8 packaging, a transport index of zero is appropriate for criticality control. Criticality during transport or handling of this pit is considered to be incredible.

9.0 REFERENCES

1. F.E. Adcock, Rocky Flats Container, Model AL-R8, SARP, RFP-8801, April 1988.

2. H. C. Paxton and N. L. Pruvost, Critical Dimensions of Systems Containing ^{235}U, ^{239}Pu, and ^{233}U, 1986 Revision, LA-10860-MS, July 1987.

3. T. P. McLaughlin, Memo HS-6-86-51, March 12, 1986.

4. WX-3, Standard Operation Procedures for Handling Fissile Items, Most Current Revision.

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EXAMPLE 4

Provided By

Idaho National Engineering Laboratory

CRITICALITY SAFETY EVALUATION FOR SHIPMENT OF RADIOACTIVE WASTE WITHIN THE BOUNDARIES OF THE INEL

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FIGURES

TABLES

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**Criticality Safety Evaluation for
Shipment of Radioactive
Waste within the Boundaries of the INEL**

1.0 INTRODUCTION

Shipments of radioactive waste within the boundaries of the INEL are made in a variety of shipping casks. The purpose of this report is to establish appropriate limits and demonstrate that these limits are in compliance with the criticality safety requirements stated in DOE-ID 5480.5A[1] and 10 CFR Part 71[2], Sections 71.55 and 71.61, for a Fissile Class III shipment.

2.0 DESCRIPTION

Shipments of radioactive waste may be made within the boundaries of the INEL in several different types of shipping casks. One such cask, TRUPACT-II, is licensed for Fissile Class I shipments with up to 325 Fissile Gram Equivalent (FGE) of Pu-239[3]. Any number of TRUPACT-II packages may be simultaneously transported.

Other shipping casks that do not have formally established fissile material limits may also be used to make waste shipments, one-cask-at-a-time, with no more than 160 FGE of Pu-239 per shipment. Loading of a second cask will not take place until the first cask is removed from the area. The casks involved are typically constructed of concentric regions of lead and steel or depleted uranium and steel. Limits for shipments involving these casks will be conservatively based on generic minimum critical mass data. Therefore, details of the various cask designs are not pertinent. However, if the limits identified in this evaluation are to be exceeded, an evaluation specific to the cask shall be required.

[Note: The description section of this example evaluation does not include the level of detail (i.e., cask dimensions and sketches) that would normally be found in an evaluation due to the generic nature of the evaluation.]

3.0 REQUIREMENTS DOCUMENTATION

Shipments of radioactive waste in casks for which this evaluation is used as the basis for criticality safety will be limited to one cask per shipment with no more than 160 FGE Pu239 per cask. Therefore, criticality safety margins will be maintained for these shipments by controlling the number of casks per shipment and by controlling the amount (mass) of fissile material per cask. Compliance with the following DOE-ID and Code of Federal Regulations (CFR) requirements will be demonstrated.

1. DOE-ID 5480.5A "For a system to be considered critically safe by mass alone, it shall not have more than 75 percent of the critical mass assuming the system is in its most reactive credible state. If over batching is credible, 45 percent of the critical mass shall be used." (Acceptable k_{eff} values must also be # 0.95.)

2. 10 CFR Part 71 Section 71.55(b) "...a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system...so that, under the following conditions, maximum reactivity of fissile material would be attained:

   a. The most reactive credible configuration consistent with chemical and physical form of the material;

   b. Moderation by water to the most reactive credible extent; and,

   c. Close reflection by water on all sides."

3. 10 CFR Part 71 Section 71.61 " A package for Fissile Class III shipment must be so designed and constructed and its contents so limited, and the number of packages in a Fissile Class III shipment must be so limited, that:

   a. Twice this number of undamaged packages would be subcritical if stacked together in any arrangement, assuming close reflection on all sides of the stack by water; and

   b. This number of packages would be subcritical if stacked together in any arrangement, closely reflected on all sides of the stack by water, and with optimum interspersed hydrogenous moderation....Each package must be considered to have been subjected to the tests specified in 10 CFR 71.73 (Hypothetical Accident Conditions)."

This analysis is designated as Quality Level A (according to the EG&G Idaho Quality Manual) and is intended to comply with NQA-1. As required by the Nuclear Engineering Standard Practice, NE-SP-2[4], any operational limits or procedures that result from or are based on information contained in this report must be reviewed by Reactor and Radiation Physics personnel to ensure that the technical information given in this Criticality Safety Evaluation has been properly incorporated into the limits or procedures.

[Note: Many of the requirements discussed in this evaluation are site specific and company specific].

4.0 METHODOLOGY

All calculations were performed with the one-dimensional S_n (n = 8) discrete-ordinates transport theory code, ANISN[5]. Sixteen-energy-group Hansen-Roach[6] cross section data were used as input to the ANISN code. The calculations were performed on an IBM PS/2 Model 60 with a version of the ANISN code that was developed at the INEL, ANISN/PC.

Six thermal reactor benchmarks, PNL-1 through PNL-5 and PNL-7, were evaluated with the ANISN code and the 16-energy-group Hansen-Roach data set. These experiments are documented in Reference 7. PNL-1 through PNL-5 are experiments with unreflected spheres of plutonium nitrate solutions with H/Pu-239 ratios ranging from 131 to 1204. PNL-7 is an experiment with a water reflected (30-cm-thick) stainless steel sphere of plutonium nitrate with an H/Pu ratio of 980. These PNL experiments provide adequate validation of the methods and cross sections used to derive the limits established in this report for the following reasons:

1. The geometric configurations of the fuel regions are spherical in both the benchmark experiments and in this analysis.

2. Both the benchmark experiments and the systems evaluated are hydrogen moderated. The benchmarks are moderated with nitrate solution. A polyethylene (60 vol.%) - water (40 vol.%) composition was assumed for this analysis.

3. H/Pu ratios range from 131 to 1204 for the benchmarks. H/Pu ratios range from 700 to 1100 for this analysis. Optimum values occurred between 800 and 900 for reflected systems. The predominant cross section data for Pu-239 at optimum conditions are associated with Hansen-Roach Library ID 94913. Hansen-Roach Library ID 94913 is used for analysis of experiments PNL-1, PNL-3, PNL-4, and PNL-7.

All ANISN benchmark calculations were performed with a quadrature order of 8 and 100 mesh intervals in the fuel region. For PNL-7, 100 mesh intervals were also represented in the reflector region with 10 mesh intervals in the thin stainless steel wall.

The results of the ANISN benchmark calculations with Hansen-Roach 16-energy-group cross section data are summarized in Table 1.

The results of the ANISN calculations agree well with the experimental values(1.0). The largest deviation occurs for PNL-3 with an H/Pu ratio of 1204 and agreement in this case is within 0.8%. Results for PNL-1, PNL-4, and PNL-7 indicate little (less than 0.15%) or no bias for H/Pu ratio between 700 and 900 (enveloping optimal values of this evaluation). Inclusion of a bias of this magnitude is inconsequential [Note: A more detailed bias determination is often required]. There are no statistical uncertainties associated with S_n discrete ordinates methods.

Table 1. Results of ANISN benchmark calculations with Hansen-Roach 16-energy-group cross section data.

Benchmark ID	Description	H/Pu-239	ANISN Calculated k-effective
PNL-1	Bare sphere	700	1.0051
PNL-2	Bare sphere	131	1.0069
PNL-3	Bare sphere	1204	0.9918
PNL-4	Bare sphere	911	0.9985
PNL-5	Bare sphere	578	0.9979
PNL-7	Reflected sphere	980	1.0023

All calculations documented in this evaluation were performed on an IBM PS/2 Model 60 operating under DOS Version 3.10 with Version 1.0 of the IBM PRO FORTRAN Compiler. Configuration Release 3.0 of the ANISN/PC code was used for the evaluation and was verified for proper operation on the IBM PS/2 Model 60 computational platform by repeating the 21 sample problems as detailed in the Software Quality Assurance Package. A more detailed description of this verification effort is given in Reference 8.

5.0 DISCUSSION OF CONTINGENCIES

As required by DOE[1], "The double contingency principle shall be used as a minimum to ensure that a criticality accident is an extremely unlikely event. Compliance with the double contingency principle requires that two unlikely, independent, and concurrent changes in process or system conditions occur before a criticality accident is possible."

As shown in Table 2 and the results in Section 6.0, the requirements of the double contingency principle are met for the proposed shipment of radioactive waste within the boundaries of the INEL. The barriers associated with these contingencies (i.e., accident prevention and fissile inventory verification measures) are yet to be determined.

Table 2. Contingency Analysis.

Contingencies (unlikely events)		
No.	Description	Barriers
1	An accident that causes all plutonium in a shipment to form into a homogenous mixture of polyethylene (60 vol.%), water (40 vol.%), and Pu-239 in optimally moderated and reflected spherical geometry. (This scenario is considered incredible by itself.)	Accident prevention measures (to be determined)
2	Exceeding the operating mass limit (160 FGE of Pu-239) by a factor of at least 2.5 (405 FGE of Pu-239 can result in a k-effective value of 1.0 under optimally moderated, reflected, and configured conditions). Waste shipments will typically contain only a few grams Pu per shipment.	Fissile mass verification measures (to be determined)

6.0 EVALUATION AND RESULTS

Calculations for this evaluation were taken directly from the Safety Analysis Report for the TRUPACT-II[4] and are reported in this section for completeness. The calculations were performed to establish limits for the shipment of radioactive waste in casks covered by this evaluation.

A summary of materials and compositions used for this analysis is given in Appendix A. Typical ANISN input listings are given in Appendix B.

Calculations to determine the minimum mass of Pu-239 required to obtain k-effective values of 0.95 and 1.0 were performed with the one-dimensional S_n discrete-ordinates transport theory code, ANISN. The model consisted of a sphere of Pu-239, polyethylene, and water surrounded by a 30-cm-thick (12-in.-thick) polyethylene-water reflector. A 60 vol.% polyethylene - 40 vol.% water composition was used in the centrally located fuel sphere and in the external reflector. This composition was used as an upper bound on mechanically compacted polyethylene [Note: Information such as this should be referenced; however, a reference is not available in this instance] and, for relatively small quantities of plutonium, conservatively represents transuranic waste. Displacement by plutonium was conservatively neglected (the actual effect of this assumption is insignificant). The radius of the fuel sphere was determined from the mass of plutonium in the sphere and the desired H/Pu ratio, both of which were varied parametrically. All plutonium was represented as Pu-239.

A quadrature order of 8 and a relative convergence criterion of 0.0001 was used in all calculations. One hundred mesh intervals were used in both the fuel sphere and in the reflector region. The width of each mesh interval is, therefore, 0.30 cm in the reflector and less than 0.20 cm in the fuel. The 16-energy-group Hansen-Roach cross section data have a Legendre scattering expansion order of 1.

With this model, the minimum masses of Pu-239 required to obtain k-effective values of 0.95 and 1.0 in a polyethylene (60 vol.%) - water (40 vol.%) environment were determined from calculations in which the mass of Pu-239 in a polyethylene - water reflected and moderated sphere was varied parametrically between 300 and 425 g. For each Pu-239 mass value, the H/Pu ratio was varied parametrically between 700 and 1100. The results of these calculations are summarized in Table 3 and are shown graphically in Figure 1. As indicated from the results in Figure 1, maximum calculated k-effective values occur for H/Pu ratios between 800 and 900. The maximum calculated k-effective values from Figure 1 were used to plot maximum k-effective versus Pu-239 mass and the results are shown in Figure 2. From Figure 2, the minimum masses of Pu-239 that are required to obtain k-effective values of 0.95 and 1.0 are shown to be about 325 g and 405 g, respectively. Therefore, a value of 325 g was selected as the safety limit ($k = 0.95$) and 160 g was very conservatively selected as the operating limit in order to be in compliance with the requirements of both DOE-ID Order 5480.5A [mass limit < 45% of the failure mass (k=1.0)] and 10 CFR Part 71 Section 71.61 (twice the number of allowed undamaged packages are safely subcritical when optimally configured with close water reflection on all sides). As indicated by the results of benchmark calculations reported in Section 5.0 and the conservative nature of the evaluation, a bias is not warranted.

Thick steel and lead reflectors, when positioned tightly around fissile regions, are often more effective than water reflectors. However, in a cask, these types of reflectors would not fit tightly around the fissile material as does the polyethylene-water reflector used in the evaluation. The thick, tight-fitting polyethylene-water reflector conservatively represents actual reflector conditions.

The materials to be transported in these casks is “waste” that will not normally contain superior reflector materials such as beryllium. Waste containing more than trace amounts of beryllium shall not be transported in these casks without additional review and approval.

Table 3. Results of ANISN calculations for spheres of Pu-239 reflected and moderated with a polyethylene-water composition.

Case ID	Pu Mass (g)	H/Pu	Sphere Radius (cm)	k-effective
1	300	700	11.93127	0.9257
		800	12.47435	0.9311
		900	12.97377	0.9332
		1000	13.43755	0.9324
		1100	13.87125	0.9298
2	325	700	12.25389	0.9438
		800	12.81164	0.9489
		900	13.32464	0.9506
		1000	13.80092	0.9494
		1100	14.24642	0.9464
3	350	700	12.56036	0.9606
		800	13.13208	0.9653
		900	13.65783	0.9665
		1000	14.14067	0.9651
		1100	14.60264	0.9616
4	400	700	13.13206	0.9904
		800	13.72979	0.9944
		900	14.27948	0.9950
		1000	14.78994	0.9928
		1100	15.26729	0.9887
5	425	700	13.40014	1.0039
		800	14.01007	1.0076
		900	14.57098	1.0078
		1000	15.09186	1.0053
		1100	15.57895	1.0008

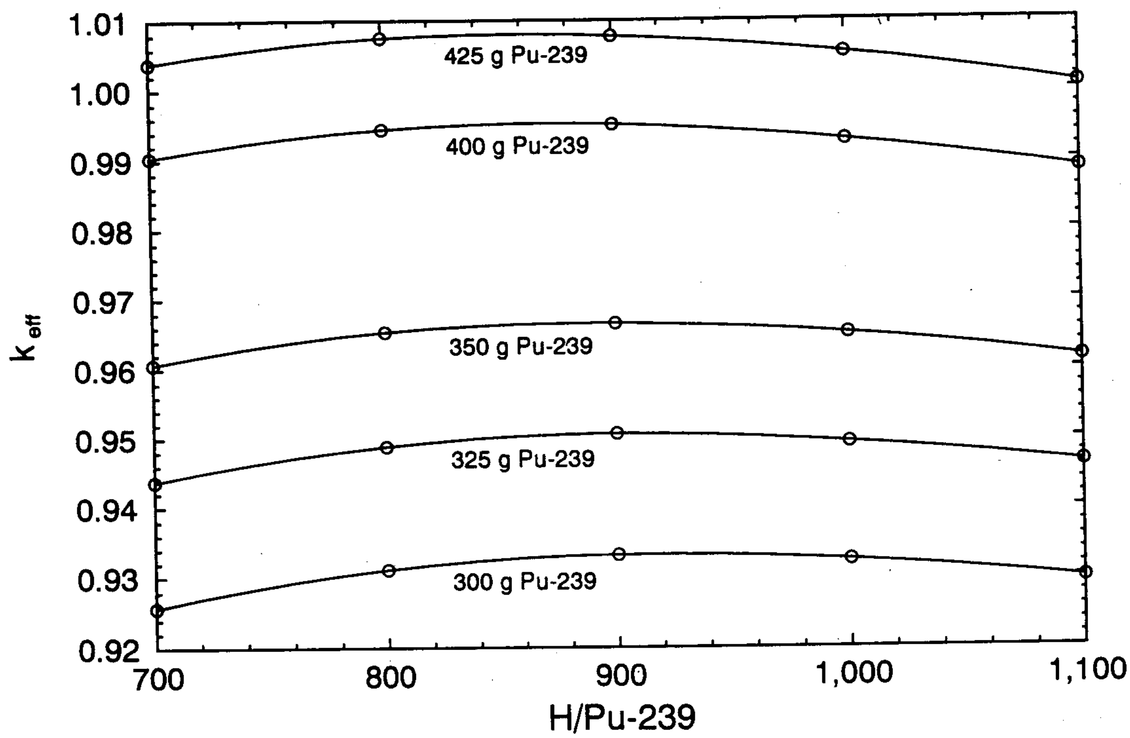

Figure 1. Calculated k-effective versus H/Pu-239 ratio for polyethylene (60 vol.%) - water (40 vol.%) moderated and reflected spheres.

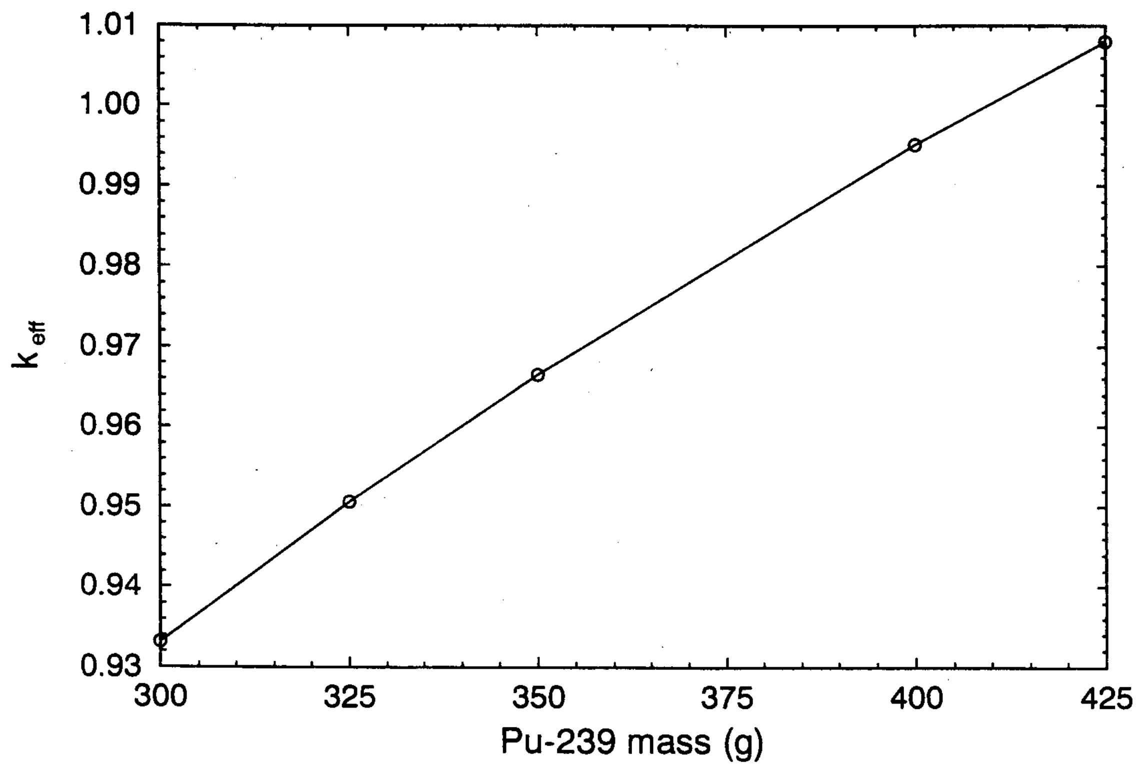

Figure 2. Maximum calculated k-effective versus Pu-239 mass for polyethylene (60 vol.%) - water (40 vol.%) moderated and reflected spheres.

7.0 DESIGN FEATURES (PASSIVE AND ACTIVE) AND ADMINISTRATIVELY CONTROLLED LIMITS AND REQUIREMENTS

There are no engineered safety features associated with these shipments that are required to maintain criticality safety margins. The following controls shall be implemented prior to beginning shipments of radioactive waste in generic casks that have fissile material limits based on the data presented in this evaluation.

1. Prior to loading a cask, an Operation Specialist shall verify that the intended contents contain less than 160 Fissile Gram Equivalents (FGE) Pu-239.
2. Only one cask shall be in the loading area at a time. The cask shall be separated from other fissile material by at least 12 feet.
3. No more than trace amounts of superior reflector/moderator materials such as beryllium shall be loaded into the casks without additional review and approval.

The means of verification are yet to be determined.

[Note: The intent of this evaluation is not to obtain an NRC license, but to demonstrate, from a criticality safety standpoint, that NRC requirements can be satisfied. Additional transportation requirements may be required if an NRC license is sought for a particular cask.]

8.0 SUMMARY AND CONCLUSIONS

The analytical techniques applied to the evaluation of radioactive waste shipments at the INEL are considered to be very conservative and postulated accident scenarios are considered to be incredible. Based on the results of the calculations presented in this evaluation, adequate safety margins exist for the shipment of a single cask containing no more than 160 FGE of Pu-239.

The following limits have been established:

Failure Limit (k = 1.0)	405 FGE Pu-239
Operating Limit (<45% of failure limit)	160 FGE Pu-239

[Note: The term "Failure Limit" is local to the INEL]

Compliance with the following DOE-ID and CFR requirements has been demonstrated:

1. <u>DOE-ID 5480.5A</u>

   a. Double Contingency Principle (see Section 6.0)

   b. Mass Control (operating limit $\leq$ 45% of critical mass)

2. <u>10 CFR Part 71 Sections 71.55 and 71.61</u>

   a. Optimum moderation, reflection, and geometry was analyzed (71.55).

   b. Twice the number of allowed packages (undamaged or damaged) is subcritical when optimally configured, moderated, and reflected (i.e., twice the operating limit of 160 FGE of Pu-239 or 320 g Pu-239 is less than the minimum mass of Pu-239 required to obtain a k-effective value of 0.95 in a polyethylene-water environment under optimum conditions (71.61)).

From a criticality safety view point, shipments of radioactive waste that are limited to a single cask containing no more than 160 FGE of Pu-239 may be safely made within the boundaries of the INEL. Specific limits for casks that have their own approved criticality safety evaluation supersede the limits established by this analysis.

9.0 REFERENCES

1. Safety of Nuclear Facilities, U.S. Department of Energy Idaho Operations Office, DOE-ID 5480.5A, August 2, 1990.

2. Packaging and Transportation of Radioactive Materials, Code of Federal Regulations, 10 CFR 71, August 24, 1983.

3. Nuclear Packaging Safety Analysis Report for the TRUPACT-II Shipping Package, NRC Docket 71-9218, U.S. Nuclear Regulatory Commission, February 1989.

4. Nuclear Engineering Standard Practice, Criticality Safety Analysis, NE-SP-2 Rev. 2, October 14, 1991.

5. D. K. Parsons, ANISN/PC Manual, EGG-2500, EG&G Idaho, Inc., April 1987.

6. Norman L. Pruvost, Melvin L. Prueitt, The Hansen-Roach Cross Sections, A Graphical Representation, LALP-88-20, October 1988.

7. ENDF-202 Cross-Section Evaluation Working Group Benchmark Specifications, BNL 19302, Brookhaven National Laboratory, November 1974, Rev. September 1982.

8. ANISN/PC Version 3.2, Software Configuration Control Package, A00004, January 2, 1991. (EG&G Idaho internal documentation)

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APPENDIX A

Materials and Compositions

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Table A-1. Summary of materials and compositions.

Description[a,b]	Nuclide or Material	KENO Lib. ID.	Atoms/barn-cm or (Volume Fraction)
Pu-239, polyethylene, & water fuel material with H/Pu = 700	Pu-239	94912 94913	4.39875E-05 6.22368E-05
Pu-239, polyethylene, & water fuel material with H/Pu = 800	Pu-239	94912 94913	1.74460E-05 7.55002E-05
Pu-239, polyethylene, & water fuel material with H/Pu = 900	Pu-239	94913 94914	8.10202E-05 1.59868E-06
Pu-239, polyethylene, & water fuel material with H/Pu = 1000	Pu-239	94913 94914	6.45010E-05 9.85602E-06
Pu-239, polyethylene, & water fuel material with H/Pu = 1100	Pu-239	94913 94914	5.09852E-05 1.66120E-05
Polyethylene-water reflector H = 7.43570-2 C = 2.38296-2 O = 1.33488-2	CH_2 H_2O	402 502	(0.60) (0.40)

a. The masses of Pu-239, when represented in ANISN calculations, depends on the radii of the spheres that are tabulated in Table 3.

b. Unless specifically stated otherwise, all plutonium bearing compositions were moderated with 60 vol.% polyethylene (ID = 402) and 40 vol.% water (ID = 502). A moderator volume fraction of 1.0 was used in all cases.

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APPENDIX B

Typical ANISN/PC Input Listings

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Listing 1. ANISN input listing for Case 2 of Table 3; H/Pu – 900.

```
PU IN CH2-H2O SPHERE: H/X=900 M=325 R=13.32464

15$$               0                 0                 1                 8
                   3                 1                 0                 2
                 200                 1                16                 3
                   4                 9                16                 0
                  52                56                 0                 0
                   0                 0                 0                50
                   0                 1                 0                 0
                  50                 0                 0                 0
                   0                 0                 1                 0
16**   +1.00000E+00      +0.00000E-01      +1.00000E-04   4R+0.00000E-01
       +1.00000E+00      +0.00000E-01      +5.00000E-01     +2.00000E-04
     3R+0.00000E-01
  T
13$$               1                 2                 3                 4
                   5                 6                 7                 8
                   9                10                11                12
                  13                14                15                16
                  17                18                19                20
                  21                22                23                24
                  25                26                27                28
                  29                30                31                32
                  33                34                35                36
                  37                38                39                40
                  41                42                43                44
                  45                46                47                48
                  49                50                51                52
  T
2**    F+1.00000E+00
  T
1**    +2.25000E-01      +3.47000E-01      +1.61000E-01      +1.70000E-01
       +8.40000E-02      +1.30000E-02    10R+0.00000E-01
4**    +0.00000E-01      +1.33246E-01      +2.66493E-01      +3.99739E-01
       +5.32986E-01      +6.66232E-01      +7.99478E-01      +9.32725E-01
       +1.06597E+00      +1.19922E+00      +1.33246E+00      +1.46571E+00
       +1.59896E+00      +1.73220E+00      +1.86545E+00      +1.99870E+00
       +2.13194E+00      +2.26519E+00      +2.39844E+00      +2.53168E+00
       +2.66493E+00      +2.79817E+00      +2.93142E+00      +3.06467E+00
       +3.19791E+00      +3.33116E+00      +3.46441E+00      +3.59765E+00
       +3.73090E+00      +3.86415E+00      +3.99739E+00      +4.13064E+00
       +4.26389E+00      +4.39713E+00      +4.53038E+00      +4.66362E+00
       +4.79687E+00      +4.93012E+00      +5.06336E+00      +5.19661E+00
       +5.32986E+00      +5.46310E+00      +5.59635E+00      +5.72960E+00
       +5.86284E+00      +5.99609E+00      +6.12933E+00      +6.26258E+00
       +6.39583E+00      +6.52907E+00      +6.66232E+00      +6.79557E+00
       +6.92881E+00      +7.06206E+00      +7.19531E+00      +7.32855E+00
       +7.46180E+00      +7.59505E+00      +7.72829E+00      +7.86154E+00
       +7.99478E+00      +8.12803E+00      +8.26128E+00      +8.39452E+00
       +8.52777E+00      +8.66102E+00      +8.79426E+00      +8.92751E+00
       +9.06076E+00      +9.19400E+00      +9.32725E+00      +9.46049E+00
```

Listing 1. Continued.

	+9.59374E+00	+9.72699E+00	+9.86023E+00	+9.99348E+00
	+1.01267E+01	+1.02600E+01	+1.03932E+01	+1.05265E+01
	+1.06597E+01	+1.07930E+01	+1.09262E+01	+1.10595E+01
	+1.11927E+01	+1.13259E+01	+1.14592E+01	+1.15924E+01
	+1.17257E+01	+1.18589E+01	+1.19922E+01	+1.21254E+01
	+1.22587E+01	+1.23919E+01	+1.25252E+01	+1.26584E+01
	+1.27917E+01	+1.29249E+01	+1.30581E+01	+1.31914E+01
	+1.33246E+01	+1.36246E+01	+1.39246E+01	+1.42246E+01
	+1.45246E+01	+1.48246E+01	+1.51246E+01	+1.54246E+01
	+1.57246E+01	+1.60246E+01	+1.63246E+01	+1.66246E+01
	+1.69246E+01	+1.72246E+01	+1.75246E+01	+1.78246E+01
	+1.81246E+01	+1.84246E+01	+1.87246E+01	+1.90246E+01
	+1.93246E+01	+1.96246E+01	+1.99246E+01	+2.02246E+01
	+2.05246E+01	+2.08246E+01	+2.11246E+01	+2.14246E+01
	+2.17246E+01	+2.20246E+01	+2.23246E+01	+2.26246E+01
	+2.29246E+01	+2.32246E+01	+2.35246E+01	+2.38246E+01
	+2.41246E+01	+2.44246E+01	+2.47246E+01	+2.50246E+01
	+2.53246E+01	+2.56246E+01	+2.59246E+01	+2.62246E+01
	+2.65246E+01	+2.68246E+01	+2.71246E+01	+2.74246E+01
	+2.77246E+01	+2.80246E+01	+2.83246E+01	+2.86246E+01
	+2.89246E+01	+2.92246E+01	+2.95246E+01	+2.98246E+01
	+3.01246E+01	+3.04246E+01	+3.07246E+01	+3.10246E+01
	+3.13246E+01	+3.16246E+01	+3.19246E+01	+3.22246E+01
	+3.25246E+01	+3.28246E+01	+3.31246E+01	+3.34246E+01
	+3.37246E+01	+3.40246E+01	+3.43246E+01	+3.46246E+01
	+3.49246E+01	+3.52246E+01	+3.55246E+01	+3.58246E+01
	+3.61246E+01	+3.64246E+01	+3.67246E+01	+3.70246E+01
	+3.73246E+01	+3.76246E+01	+3.79246E+01	+3.82246E+01
	+3.85246E+01	+3.88246E+01	+3.91246E+01	+3.94246E+01
	+3.97246E+01	+4.00246E+01	+4.03246E+01	+4.06246E+01
	+4.09246E+01	+4.12246E+01	+4.15246E+01	+4.18246E+01
	+4.21246E+01	+4.24246E+01	+4.27246E+01	+4.30246E+01
	+4.33246E+01			
5**	F+1.00000E+00			
6**	+0.00000E-01	+6.04938E-02	+9.07407E-02	+1.37037E-01
	2R+2.11728E-01	+1.37037E-01	+9.07407E-02	+6.04938E-02
7**	-9.75900E-01	-9.51190E-01	-7.86796E-01	-5.77350E-01
	-2.18218E-01	+2.18218E-01	+5.77350E-01	+7.86796E-01
	+9.51190E-01			
8$$	1	1	1	1
	1	1	1	1
	1	1	1	1
	1	1	1	1
	1	1	1	1
	1	1	1	1
	1	1	1	1
	1	1	1	1
	1	1	1	1
	1	1	1	1
	1	1	1	1
	1	1	1	1

Listing 1. Continued.

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                    1                   1                   1                   1
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                    2                   2                   2                   2
                    2                   2                   2                   2
9$$                53                  55
10$$               53                  54                  53                  54
                   53                  54                  53                  54
                   53                  54                  55                  56
                   55                  56                  55                  56
11$$                1                   2                   3                   4
                    7                   8                  45                  46
                   47                  48                   1                   2
                    3                   4                   7                   8
12**  2R+7.43570E-02      2R+2.38296E-02      2R+1.33488E-02      2R+8.10202E-05
      2R+1.59868E-06      2R+7.43570E-02      2R+2.38296E-02      2R+1.33488E-02
19$$                1                   1                   1
  T   T
```

Listing 2. ANISN input listing for Benchmark PNL-7.

PNL-7 BENCHMARK CALCULATION; HR XSEC; REFLECTED SPHERE; H/X=980

```
15$$             0             0             1             8
                 3             1             0             3
               210             1            16             3
                 4             9            26             0
                52            58             0             0
                 0             0             0            50
                 0             1             0             0
                50             0             0             0
                 0             0             1             0
16**  +1.00000E+00  +0.00000E-01  +1.00000E-04  4R+0.00000E-01
      +1.00000E+00  +0.00000E-01  +5.00000E-01  +2.00000E-04
    3R+0.00000E-01
  T
13$$             1             2             3             4
                 5             6             7             8
                 9            10            11            12
                13            14            15            16
                17            18            19            20
                21            22            23            24
                25            26            27            28
                29            30            31            32
                33            34            35            36
                37            38            39            40
                41            42            43            44
                45            46            47            48
                49            50            51            52
  T
2**F+1.00000E+00
T
1**   +2.25000E-01  +3.47000E-01  +1.61000E-01  +1.70000E-01
      +8.40000E-02  +1.30000E-02 10R+0.00000E-01
4**   +0.00000E-01  +1.77800E-01  +3.55600E-01  +5.33400E-01
      +7.11200E-01  +8.89000E-01  +1.06680E+00  +1.24460E+00
      +1.42240E+00  +1.60020E+00  +1.77800E+00  +1.95580E+00
      +2.13360E+00  +2.31140E+00  +2.48920E+00  +2.66700E+00
      +2.84480E+00  +3.02260E+00  +3.20040E+00  +3.37820E+00
      +3.55600E+00  +3.73380E+00  +3.91160E+00  +4.08940E+00
      +4.26720E+00  +4.44500E+00  +4.62280E+00  +4.80060E+00
      +4.97840E+00  +5.15620E+00  +5.33400E+00  +5.51180E+00
      +5.68960E+00  +5.86740E+00  +6.04520E+00  +6.22300E+00
      +6.40080E+00  +6.57860E+00  +6.75640E+00  +6.93420E+00
      +7.11200E+00  +7.28980E+00  +7.46760E+00  +7.64540E+00
      +7.82320E+00  +8.00100E+00  +8.17880E+00  +8.35660E+00
      +8.53440E+00  +8.71220E+00  +8.89000E+00  +9.06780E+00
      +9.24560E+00  +9.42340E+00  +9.60120E+00  +9.77901E+00
      +9.95681E+00  +1.01346E+01  +1.03124E+01  +1.04902E+01
      +1.06680E+01  +1.08458E+01  +1.10236E+01  +1.12014E+01
      +1.13792E+01  +1.15570E+01  +1.17348E+01  +1.19126E+01
      +1.20904E+01  +1.22682E+01  +1.24460E+01  +1.26238E+01
```

Listing 2. Continued.

	+1.28016E+01	+1.29794E+01	+1.31572E+01	+1.33350E+01
	+1.35128E+01	+1.36906E+01	+1.38684E+01	+1.40462E+01
	+1.42240E+01	+1.44018E+01	+1.45796E+01	+1.47574E+01
	+1.49352E+01	+1.51130E+01	+1.52908E+01	+1.54686E+01
	+1.56464E+01	+1.58242E+01	+1.60020E+01	+1.61798E+01
	+1.63576E+01	+1.65354E+01	+1.67132E+01	+1.68910E+01
	+1.70688E+01	+1.72466E+01	+1.74244E+01	+1.76022E+01
	+1.77800E+01	+1.77927E+01	+1.78054E+01	+1.78181E+01
	+1.78308E+01	+1.78435E+01	+1.78562E+01	+1.78689E+01
	+1.78816E+01	+1.78943E+01	+1.79070E+01	+1.82070E+01
	+1.85070E+01	+1.88070E+01	+1.91070E+01	+1.94070E+01
	+1.97070E+01	+2.00070E+01	+2.03070E+01	+2.06070E+01
	+2.09070E+01	+2.12070E+01	+2.15070E+01	+2.18070E+01
	+2.21070E+01	+2.24070E+01	+2.27070E+01	+2.30070E+01
	+2.33070E+01	+2.36070E+01	+2.39070E+01	+2.42070E+01
	+2.45070E+01	+2.48070E+01	+2.51070E+01	+2.54070E+01
	+2.57070E+01	+2.60070E+01	+2.63070E+01	+2.66070E+01
	+2.69070E+01	+2.72070E+01	+2.75070E+01	+2.78070E+01
	+2.81070E+01	+2.84070E+01	+2.87070E+01	+2.90070E+01
	+2.93070E+01	+2.96070E+01	+2.99070E+01	+3.02070E+01
	+3.05070E+01	+3.08070E+01	+3.11070E+01	+3.14070E+01
	+3.17070E+01	+3.20070E+01	+3.23070E+01	+3.26070E+01
	+3.29070E+01	+3.32070E+01	+3.35070E+01	+3.38070E+01
	+3.41070E+01	+3.44070E+01	+3.47070E+01	+3.50070E+01
	+3.53069E+01	+3.56069E+01	+3.59069E+01	+3.62069E+01
	+3.65069E+01	+3.68069E+01	+3.71069E+01	+3.74069E+01
	+3.77069E+01	+3.80069E+01	+3.83069E+01	+3.86069E+01
	+3.89069E+01	+3.92069E+01	+3.95069E+01	+3.98069E+01
	+4.01069E+01	+4.04069E+01	+4.07069E+01	+4.10069E+01
	+4.13069E+01	+4.16069E+01	+4.19069E+01	+4.22069E+01
	+4.25069E+01	+4.28069E+01	+4.31069E+01	+4.34069E+01
	+4.37069E+01	+4.40069E+01	+4.43069E+01	+4.46069E+01
	+4.49069E+01	+4.52069E+01	+4.55069E+01	+4.58069E+01
	+4.61069E+01	+4.64069E+01	+4.67069E+01	+4.70069E+01
	+4.73069E+01	+4.76069E+01	+4.79069E+01	
5**	F+1.00000E+00			
6**	+0.00000E-01	+6.04938E-02	+9.07407E-02	+1.37037E-01
	2R+2.11728E-01	+1.37037E-01	+9.07407E-02	+6.04938E-02
7**	-9.75900E-01	-9.51190E-01	-7.86796E-01	-5.77350E-01
	-2.18218E-01	+2.18218E-01	+5.77350E-01	+7.86796E-01
	+9.51190E-01			
8$$	1	1	1	1
	1	1	1	1
	1	1	1	1
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Listing 2. Continued.

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	3	3		
9$$	53	55	57	
10$$	53	54	53	54
	53	54	53	54
	53	54	53	54
	53	54	53	54
	55	56	55	56
	55	56	57	58
	57	58		

Listing 2. Continued.

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11$$           1             2             5             6
               7             8            11            12
              25            26            27            28
              45            46            47            48
               9            10            11            12
              13            14             1             2
               7             8
12**  2R+6.50060E-02  2R+7.51140E-04  2R+3.45140E-02  2R+1.54570E-06
      2R+0.00000E-01  2R+3.56890E-07  2R+5.86600E-05  2R+7.34200E-06
      2R+1.65800E-02  2R+6.17500E-02  2R+8.15800E-03  2R+6.66300E-02
      2R+3.33200E-02
19$$           1             1             1
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EXAMPLE 5

Provided By

Rocky Flats Environmental Technology Site

Building 371 Caustic Waste Treatment System

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1.0 INTRODUCTION

This evaluation has been performed for the Caustic Waste Treatment System (CWTS) in Building 371 and is intended for any liquid wastes which currently exist or could be generated within the facility. These wastes include liquids in bottles and tanks or from spills, maintenance, etc. Activities within the scope of this evaluation include direct draining of tanks into the CWTS, operation of the CWTS, and subsequent shipment of effluents to Building 374. The vacuum source for the CWTS is the PROVE Vacuum System which was evaluated in NMSL-940110 and thus will not be further addressed in this evaluation. Tasks such as material handling, drum loading, spill response, etc., are conducted per building general comments and miscellaneous limits. This evaluation was originally performed in accordance with procedure 3-B69-NSPM-5B-01 and revised in accord with SSOC-NCSI-040.

In this revision, controls were added to allow caustic liquids to be received and filtered in the CWTS.

2.0 DESCRIPTION

The CWTS is a RCRA permitted treatment process which is designed to reduce fissionable content of waste liquids to within discard limit for transfer to Building 374 liquid waste treatment processes. Reference Appendix F for a flow diagram and room layouts of the CWTS. Waste streams are fed into a manifold in glovebox 2401 via hardpiping, tygon tubing, or from other sources, such as a “portable fill station” or four-liter bottle, and vacuum transferred to shielded annular tanks D-2401 A/B/C/D where they are mixed, sampled, adjusted to 25 g Pu/l or less, and transferred to clarifier columns T2411 A/B in glovebox 18 by use of pumps P2401 A/B in glovebox 2402. A caustic in powder form, such as Magnesium Hydroxide, is added to the liquids in these columns to neutralize the acidity and precipitate Pu out of the solutions. The contents of the clarifier columns are then filtered through R6 filter FL2431 in glovebox 18 and the supernate solution is collected in decant columns T2411 C/D, also in glovebox 18.

Plutonium Hydroxide precipitate from FL2431 is placed into nominal 2” high pans and passed to glovebox 23 for drying on hotplate W-2. Dried material is placed into 8801 type cans, bagged out, and managed in accordance with the appropriate limits, i.e. approved shipping container, cart, etc. Used R6 filter media (pad) from FL2431 is dried in glovebox 18 and/or glovebox 23. The dry filter media is then bagged out and managed in accordance with appropriate limits.

Supernate liquids in decant columns T2411 C/D are pumped by pump P2411 in glovebox 18 across either of two banks of 1 micron bag filters, i.e., FL2416-1,2,3 A/B in glovebox 2404, into shielded annular tanks D-2402 A/B. Contents of these tanks are sampled and vacuum transferred into shielded annular tank D-2403 if less than the Building 374 waste transfer limit of 4x10-3 g/l fissionable material. If solutions do not meet the waste transfer limit, they are vacuum transferred to tanks D-2401 A/B/C/D and reprocessed in the CWTS.

Liquids in tank D-2403 are sampled and, if less than the waste transfer limit, pumped by pumps P2403 A/B in glovebox 2403 into Building 374 receiving tanks D811 A/B. Liquids which do not meet the waste transfer limit are either further filtered or vacuum transferred into tanks D-2401 A/B/C/D and reprocessed in the CWTS.

The system has only been in operation for a relatively short amount of time, such that any hold-up of fissionable material would be contamination only. Also, the CWTS is in locations with appropriate criticality detector coverage.

For further details of this system please reference procedures 4-U84-C0-6090, "Building 371 Caustic Waste Treatment System" and 4-R76-WO-5017, "Caustic Waste Transfer from Building 371".

3.0 REQUIREMENTS DOCUMENTATION

There are no unique documentation requirements for this evaluation.

4.0 METHODOLOGY

4.1 The CWTS was designed with geometrically favorable tanks and equipment. Thus, evaluation methodology consists primarily of identification of components which meet established standards and criteria for geometrically favorable equipment. In addition, this evaluation discusses tank draining in Building 371 and waste transfer to Building 374.

4.2 The following assumptions were used in this evaluation.

4.2.1 No fissionable solution ever overflowed into the building process vacuum system. This was substantiated during conversation with Jack D. Weaver on 11/16/95. Mr. Weaver was one of the supervisors of the building processes during facility startup and the duration in which the processes were operated. Also, as the criticality engineer for Building 371 from 1983 through 1988, I did not respond to or observe any fissionable solution overflows into the building process vacuum system.

5.0 DISCUSSION OF CONTINGENCIES

Tanks, equipment in gloveboxes, and the activities associated with waste transfer to Building 374 and draining of Building 371 tanks are individually evaluated in section 6.0. Unlikely contingencies are summarized as follows. Note that there are no credible criticality scenarios associated with glovebox 18, since columns and filters are critically safe by dimension. In addition, seismic events would only cause glass columns to break in glovebox 18 and that liquid would be restricted to a critically safe 2" depth in the glovebox due to the criticality drain.

Equipment	Contingencv	Barrier
Shielded Annular Tanks	1. Misconstruction a. Annulus >3” b. Insufficient Shielding	1. Physical Verification a. Verified as < 3” b. Verified by TES to meet design criteria.
	2. Annulus Bulges	2. Significant bulging not credible since tanks are vented and certified to meet pressurization from vacuum transfers and low pressure diaphragm pump. Reference Appendix O for tank certification.
	3. Loss of Shielding	3. Materials encased in stainless steel, which is observed to be intact with no corrosion.
	4. Seismic Activity	4. Tanks are neutronically isolated.
	5. Overbatching to 300 g/l.	5. Tanks are limited to 150 g/l. Liquids required to be characterized as less than 150 g/l before introduced into tanks D-240 1 A/B/C/D and the CWTS process.
	6. Precipitation in tanks from addition of caustic.	6a. Caustic liquid not accepted unless receiving tank is empty and isolated from other tanks. 6b. Inlet valve to tanks D-2401 A/B/CID to be closed when greater than four liters of caustic is collected or stored in any glovebox or room which drains to CDS “A”. Liquid may be drained into CDS “A” per 6a.
GB-2401/GB-2402 (equipment) GB-23	1. Fissionable Solutions in excess of 150 g/l.	1. Equipment is subcritical with no spacing and nominal water reflection for concentrations up to 300 g/l.
	2. Full water reflection	2. Criticality drain.

Equipment	Contingencv	Barrier
	1. Too many containers/material pileup from seismic activity, etc.	1. Limited to three 2” deep pans, two 1-liter containers, and three used filters. Operations are also only performed by specially trained process specialists, i.e., Core team.
	2. Fissionable material overbatch	2. It would be unlikely for the following barriers to simultaneously fail. a. Only precipitate from one “run” per pan. b. Clarifier columns limited to a maximum of 25 g/l fissionable liquids. c. Fissionable liquid concentrations known before blending in tanks D-2401 A/B/C/D. d. l-liter containers limited to a maximum of 1000 grams net weight. e. Process design results in clarifier column operating volume less than 20 liters.
	3. Reflection	3. Criticality drain.

6.0 EVALUATION RESULTS

Tanks, equipment, and waste transfer activities involved with the CWTS are shown double contingent as follows.

6.1.1 SAT Design: SATs were designed at RFETS for the critically safe storage of large volumes of high-level fissionable solutions. The design consists of a nominal 3”annular tank surrounded on both sides by shielding composed of a 2.5” annulus of polyethylene sandwiched between two 1/8” annual nuclear poison regions. This shielding is fully encased by stainless steel to assure its integrity and as a fire barrier. Reference figure 1 for a cross section view of a SAT. Note that the robust nature of the cladded shielding precludes loss of the poison/moderator regions, such that further verification of those regions is not required. Note also that no corrosion of the shielding is observed.

Various monte-carlo computer calculations (KENO analyses, reference Appendix G), were performed in support of SAT design. The shielding uses a polyethylene moderator to thermalize neutrons and optimize the effect of neutron absorbers located on either side of the moderator. This configuration results in neutronic isolation of the SAT tank, thus there are no spacing requirements. Note that this isolation also precludes seismic contingencies from tank shifting. These analyses, contained in evaluations JNM-25 and JNM-28, became the basis for the SAT design criteria, which is included in Appendix H.

Validation of KENO analyses in JNM-25 and JNM-28 was completed by Critical Mass Laboratory (CML) experimentation as reported in Appendix I. Though initial design analyses used KENOIV and CML validation used KENOV, there is no significant difference in code usage as reported in Appendix J. Note that CML experimentation was specifically conducted to validate poison and moderator application in the Shielded Annular Tanks, thus U235 solution was used in the critical mass experiments. This is appropriate as cross sections for Pu and U235 solutions in annular tanks have been previously validated in CML experiments at RFETS and within the complex.

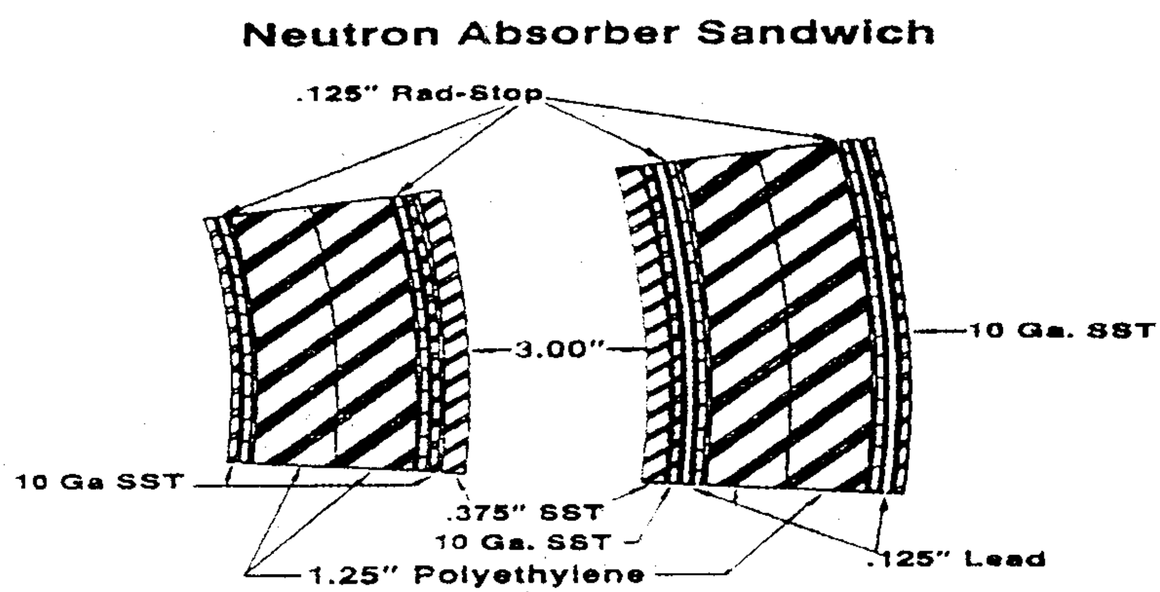

Figure 1

6.1.2 SATs D2401 A/B/C/D, D-2402 A/B, D-2403: The CWTS exclusively uses these seven tanks for storage of fissionable solutions. Each tank can contain approximately 725 liters of liquid and is nominally 64.24" OD by 87.25" H. These tanks meet design criteria as follows. Accordingly, CWTS SATs are subcritical for any contingency, including overbatching to 1000 g/l fissionable solution. (reference Appendix G).

6.1.2.1 All CWTS SATs met the criteria for solution annular thickness of 2.88" +/- 1/8". Verification using a digital ultrascope at six different heights for each of eight radial positions can be found in Appendix K.

6.1.2.2 Pure polyethylene with a density of 0.92 g/cc was used in the CWTS SATs. A certification letter to this effect is in Appendix K. In addition, the Tank Evaluation System (TES) was used to verify the presence of required polyethylene in these tanks. Reference Appendix K for TES results.

6.1.2.3 Type 238Z Flex/Boron (silicone based rubber mixed uniformly with Boron Carbide) manufactured by Reactor Experiments, Inc. was used as the nuclear poison in the CWTS SATs. An elemental analysis of the Flex/Boron is in Appendix K. In addition, TES was used to verify the presence-of required nuclear poisons in these tanks. Reference Appendix K for TES results.

6.1.2.4 Since tanks which could be drained to the CWTS were operated at less than 150 g/l Pu and the highest fissionable solution currently in Building 371 tanks is less than 10 g/l Pu, the maximum credible solution concentration is 150 g Pu/l. In addition, liquids from other facilities are to be sampled to contain less than 150 g/l fissionable material, i.e., Pu + U235, before accepted into Building 371. Thus the 150 g/l fissionable material limit for tanks D-2401 A/B/C/D can be maintained during collection of liquids. In addition, liquids received by tanks D-2402 A/B have undergone precipitation to remove fissionable materials and supernate liquid has been filtered through an R6 filter and three 1-micron bag filters. Caustic liquids are only filtered since materials are suspended and not dissolved.

6.1.2.5 Overbatching of collection tanks D-2401 A/B/C/D as the result of precipitation due to the introduction of caustic liquids would not exceed 1000 g Pu/l, thus tanks D-2401 A/B/C/D would remain subcritical. Reference correspondence in Appendix R. Precipitation in tanks is prevented by ensuring the receiving tank is empty and isolated from other tanks prior to introduction of caustic liquids. Also, inlet valves to these tanks are required to be closed when greater than four liters of caustic is present in any glovebox or room which drains to Criticality Drain System "A". In addition, caustic liquids are not allowed to be used in decontamination activities in rooms with drains to Criticality Drain System "A" or disposed in it. Reference the criticality safety evaluation for Criticality Drain System "A".

6.1.2.6 Overbatching of tanks D-2401 A/B/C/D as the result of plutonium polymer formation (water added to tanks containing high concentration plutonium nitrate solution) would not exceed 1000 g Pu/l, thus tanks D-2401 A/B/C/D would remain subcritical. Reference correspondence in Appendix R. This is also judged to bound density increases due to freezing of tank contents.

6.1.2.7 Tank D-2403 is limited to a total of 200 grams Pu and U235 in support of Building 374 waste transfer contingencies. Reference section 6.8 for this discussion.

6.2 Glovebox 2401 (GB-2401): GB-2401 is located in room 1105. It is constructed of stainless steel with 0.25” lead and is nominally 17’ long x 2.33’ wide x 4’ high. This glovebox contains a pipe manifold which has fifteen nominal ¾” ID inlet feed pipes (tie-points) from specific building tanks joined to a nominal 2” ID header pipe, which in turn, is connected to a nominal ¾” ID outlet pipe to tanks D-2401 A/B/C/D and a separate nominal ¾” return pipe from pumps P2401 A/B. Currently only tie point 45 (Process Support Tanks), tie point 46 (PROVE Vacuum Tanks), Tygon-tubing tie point, and the outlet pipe to tanks D-2401 A/B/C/D are connected (reference CWTS Solution Collection Process sketch in Appendix F). A vacuum pickup wand connected to fulflo filter Fl-2403 (filter housing is nominally 3.75” ID x 10” H) for spill pickup and transfer to tanks D-2401 A/B/C/D is also in this glovebox. Since the maximum credible concentration in Building 371 is 150 g/l, thus meeting the concentration limit on the D-2401 A/B/C/D tanks, the vacuum pickup wand may be used for spills both internal and external to GB2401. A standard SG-508 criticality drain is installed in GB-2401, thus flooding above a two inch depth is precluded.

In Appendix M it was shown that the cross sectional area of the equipment in GB-2401 is equivalent to that of 7.82” diameter cylinder, which per TID-7016 is subcritical with nominal 1” water reflection up to 300 g/l or twice the maximum credible concentration. Note that full reflection of this cylinder would be at least double contingent since it would require the criticality drain to clog and a source of liquid to fill the glovebox. Note that 3 used ful-flo filters were included in the calculation to address the removal of ful-flo filters, as was a 3/4” ID tygon tubing for direct draining of Building 371 tanks.

In reality, the fixed equipment in GB-2401 is well spread out and at least 12” off of the glovebox floor. Thus it would contribute only negligible reactivity to items on the glovebox floor. Accordingly, these items can be grouped together and evaluated separately. The total volume of two 4.4 liter containers and 6 used ful-flo filters (each approximately 800 ml) was estimated at 13.6 liters. The total volume increases to 15.8 liters with the addition of a 2.2 liter container of kimwipes used to collect spills. This volume is subcritical for the maximum credible concentration of 150 g/l in an equivalent hemispherical volume with 1” of water reflection. As calculated in Appendix O, a nominally reflected hemispherical volume of 20.2 liters is equivalent to a 10.5 liter nominally reflected sphere, which is subcritical for a concentration up to at least 1000 g Pu/l. Note that the hemispherical volume calculation was selected to more realistically envelope available containers in the glovebox. Note that 6 used ful-flo filters were considered, as stated above, with the solution containers to account for filters stored on the glovebox floor. It can, thus, be concluded that one 4-liter container in GB-2401 is double contingent, as unlikely contingencies of overbatch and extra container, together, are subcritical. The double contingency argument for the 4-liter container clearly bounds contingencies of overbatch and extra container for the 2-liter container of kimwipes due to its smaller volume.

A large bag-in-bag, connected from a gloveport to a 55-gallon drum, could be used to move one bottle of fissionable liquid at a time into the glovebox with existing limits, as it is considered part of the glovebox. Clearly, if the liquid were to leak from the bottle into the bag, it would still only be a 4 liter volume already bounded by the previous paragraph. Any leak from the bottle into the drum must be considered in the criticality safety evaluation for the drum. Note that the bag-in bag itself, per bag-in procedures, must be secured such that it does not collect fissionable liquids when unattended.

6.3 Glovebox 2402 (GB-2402): GB-2402 is located in room 1103. It is constructed of -stainless steel with 0.25" lead and is nominally 12' long x 2.5' wide x 4.74' high. This glovebox contains two maximum-size l -liter pumps (P2401 A/B), one vacuum pick-up wand with fulflo filter FL2406, and additional fulflo filter FL2407 installed on the return line from pumps P2401 A/B to tanks D-2401 A/B/C/D. Process sampler, ME-2403, is used to sample tanks D-2401 A/B/C/D. Samples are collected in three 20 ml vials and removed via sample drop. A standard SG-508 criticality drain is installed in GB-2402, thus flooding above a two-inch depth is precluded. Since the maximum credible concentration in Building 371 is 150 g/l, thus meeting the concentration limit on the D-2401 A/B/C/D tanks, the vacuum pickup wand may be used for spills both internal and external to GB-2402.

In Appendix M the total volume of two pumps, two filter housings, three used fulflo filters, and 200 ml of samples in GB-2402 was estimated at 8.2 liters. Increasing the sample volume to 1000 ml will result in a total volume of 9 liters. The total volume further increases to 11.2 liters with the addition of a 2.2 liter container-of kimwipes used to collect spills. This is clearly subcritical, even in a spherical volume, for concentrations up to at least 250 g/l, as referenced in figure 32 of LA-10860. Note that full reflection of this sphere would be at least double contingent since it would require the criticality drain to clog (unlikely) and a large source of liquid to fill the glovebox, such as a major pipe leak or fire suppression liquid (either also unlikely). While this curve from figure 32 is for unreflected spherical volumes, the conservatism of using a combined spherical volume as opposed to individual containers in a on-layer planar array, as well as, the assumption of the 2 liter container of kimwipes being 150 g/l Pu solution, is judged to well bound the nominal reflection of operator hands. Further note that at 250 g/l, a spherical volume of over 15 liters is required before criticality is possible.

6.4 Glovebox 18: GB- 18 is located in room 1115. It is constructed of stainless steel, 1/8" lead, and 2" waterwall, and is nominally 11' long x 3' wide x 10' high. Reference Appendix F for a layout of this glovebox. Solution is received into two nominally 6" ID x 66" high clarifier columns spaced approximately 35" center-to-center from each other. A caustic, such as Mg(OH)2, is added and the liquid filtered through an R6 filter (nominally 18" ID x 2" H) located over two feet away at the other end of the glovebox. Supernate liquid is collected in two nominal 6" ID x 66" high decant columns spaced on approximate 32" centers from each other. Nearest spacing between decant columns and clarifier columns is approximately 17" center-to-center from each other. A vacuum pickup wand and fulflo filter FL-2434 is approximately 15" edge-to-edge above and approximately 14" center-to-center horizontally from the R6 filter. A maximum-size 1-liter pump (P-2411) is also located in this box. A standard SG-508 criticality drain is installed in

GB-18, thus flooding above a two-inch depth is precluded. The following conditions are thus evaluated.

6.4.1 Normal Condition: There is clearly no criticality safety issue, due to favorable geometry, for the columns or equipment containing supernate liquid from the R6 filter. Due to the large spacings between these components, each column can be evaluated individually. Accordingly, each 6” column is subcritical for concentrations up to about 2500 grams Pu/l, with one inch water reflection, as referenced in figure 3.7 of TID-7016. Even with an additional 3 centimeters to account for tolerances in diameter, each column is still subcritical for concentrations up to 1000 grams Pu/l. Since the maximum upset mass from section 6.5 is 6000 grams, concentrations greater than 1000 grams Pu/l in each column are still subcritical, since the minimum subcritical mass for concentrations greater than 1000 grams Pu/l is about 10.1 kg in a one inch water reflected sphere, reference figure 3.5 of TID-7016. The R6 filter assembly is over two feet away from the clarifier columns, thus it is also evaluated individually with a one-layer planar array of three 2” pans loaded with precipitate from the R6. The 2” height of the R6 filter assembly, pans, and used R6 filters is bound by an infinite 2” slab, which is, per figure 3.8 of TID-7016, subcritical for all concentrations up to at least 5 kg Pu/l, even with one inch of water reflection. The 2” slab also bounds any precipitate or materials which could be spilled onto the glovebox floor. The used R6 filter media is very thin, i.e., <1/4” thick, and six of them would, together, be less than two inches thick, even if folded up. Six used R6 filter media were evaluated, in the unlikely event that two additional filters were present. Note that precipitate in pans is restricted to the approximate height of each pan, though it is reasonable for small amounts to be slightly above the pan height as a result of pan loading or handling. Fulflo filter FL-2434 with two used filter cartridges in contact would have a cross sectional area equal to a 5.16 inch D cylinder, which is critically safe, as referenced in TID-7016, for the small amount of fissionable material in the filters. Finally, the 1 - liter pump is critically safe as referenced in LA- 10860, assuming a spherical volume.

It should be noted that a significant margin exists between the critical concentration for a 6” diameter cylinder (reference TID-7016) and the maximum credible concentration available to columns in the CWTS, i.e., > 1000 g Pu/l versus 150 g Pu/l. Note that concentrations in excess of 150 Pu/l, though evaluated as an upset condition in paragraph 6.1.2.5 and 6.1.2.6, would notbe credible in the columns, since liquids are verified as less than 25 g/l from the D-2401 A/B/C/D tanks, thus any upset concentration would be detected. These significant margins are supportive of the approach taken with interaction in the preceding paragraph.

Further review of criticality safety evaluation HD-20, “Precipitation and Calcination”, also substantiates the subcriticality of glovebox 18 components. In HD-20, KENO calculations for six 6” diameter by 72” precipitators, each directly above a stacked 3” filter boat and 1-3/8” catch pan, arrayed circularly on 32” centers resulted in a k-effective of 0.8346 +/0.0062. In this case 200 g Pu/l solution was in each precipitator, 2.101 kg Pu/l was in each filter boat, and 4.708 kg Pu/l was in each pan. This arrangement is clearly more reactive than that in glovebox 18, as there was no spacing between precipitators and filter boat/pan combinations as there is between the columns and R6 filter/pans in glovebox 18. Also, spacing between precipitators in HD-20 is, overall, closer than that between columns in glovebox 18.

6.4.2 Seismic: In a seismic event, it is judged that the glass columns would break and up to 2" of liquid could fill the floor. This is subcritical, as can be seen in TID-7016.

6.4.3 Samples: 1-liter of samples may be taken and handled in this glovebox, since the relatively small volume of liquid would be bounded by the unrealistically high concentrations for which equipment in this glovebox has been evaluated. In effect, the most reactive configuration of the samples would be either next to the R6 filter assembly, evaluated unrealistically to 5000 g/l, or any of the columns, unrealistically evaluated to 1000 g/l, thus clearly bounding the maximum credible concentration of 150 g/l in the 1 liter of samples.

6.5 Glovebox 23: GB-23 is located in room 1115. It is constructed of stainless steel with ¼" lead and 2" waterwall. Precipitate in nominal 2" deep pans from glovebox 18 is dried on a hotplate in this glovebox. Dried material is then collected into 1-liter containers and bagged out. Used bag filters and cartridge filters from the CWTS are also dried in this box. This glovebox has an approved criticality drain and is also open to GB18, which has a standard criticality drain, thus flooding above a two-inch depth is precluded.

6.5.1. Normal Condition: Normally, a maximum fissionable concentration of 25 g/l is processed in approximately 20 liter batches in each clarifier column in glovebox 18. This amount of liquid is measured by a timed metering pump. Before adding Magnesium Oxide for precipitation, each column is adjusted to remove any liquid in excess of 20 liters. Note that about 7 liters of up to 150 g/l fissionable liquid could be introduced into the columns, instead of 7 liters of 25 g/l fissionable liquid. This is due to a small amount of liquid which could be left in common tank transfer lines after blending and sampling. Reference Appendix O. Accordingly, precipitate from the two columns or processing "run", would have a maximum fissionable mass of 1875 grams (33 liters x 25 g/l + 7 liters x 150 g/l). As only one "run" is allowed in each pan, the three pans allowed in glovebox 23 would contain a combined fissionable mass of 5625 grams. Since material is bulk weighed into the two maximum-size one-liter cans and assuming the material is Plutonium Oxide (Pu atomic weight of 239/Plutonium Oxide atomic weight of 271=88 wt%), the contribution of fissionable material from the two cans is 1760 grams. The addition of fissionable material in the three filters is expected to provide no more than 150 grams, particularly as operators are trained to efficiently scrape precipitate from the R6 filter media, as bag filters receive only low level filtrate, and as ful-flo filters accumulate less than 50 grams fissionable material. Reference Appendix O. Thus, the normal fissionable mass loading in glovebox 23 is 7535 grams. This amount of fissionable material is bounded by the mass overbatch condition.

6.5.2 Mass Overbatch: There are four credible means of mass overbatch in glovebox 23. The first means is by processing the maximum credible solution of 150 g/l in 20 liter batches in each clarifier column. This overbatch would result in one pan containing 6000 grams of fissionable material. The second means is by loading two "runs" into one pan for a fissionable mass of 3750 grams. The third overbatch would involve completely filling each clarifier column to its full 30.6 liter capacity with 25 g/l fissionable solution, such that the fissionable mass available to one pan would be 2405 grams (includes 7 liters at 150 g/l). The fourth overbatch results in the unlikely event that two pans or 3750 grams of fissionable mass were put into one can before the error was identified. The first overbatch was evaluated since its higher fissionable material mass bounds the other three. Accordingly, the total fissionable mass in glovebox 23 for this condition would be 4125 grams more than the normal condition or 11660 grams. In three approximate 1.9 liter pans (6"x 9.5" x 2"), two 1.1 liter cans (10% variance added), and three approximate 800 ml filters or a total volume of 10.3 liters, this total fissionable material would have a density of about 1132 g/l, which is subcritical, per figure 3.6 of TID-7016, when reflected with one inch of water in a spherical volume. Thus, the mass overbatch condition is subcritical. Note also that this bounds interaction with items such as handcarry and carts.

Extra container: The unlikely addition of two pans, three cans, or four filters (or an equivalent volume combination), provides an additional 3.8 liters to the allowed 10.3 liters of containers in glovebox 23. The bounding case for this contingency would be two additional pans, each with 1875 grams Pu, brought in from glovebox 18. The total available fissionable mass (11285 grams) in 14.1 liters would have a density of about 800 g/l, which is subcritical for an equivalent hemispherical volume with 1" of water reflection. As calculated in Appendix O, a nominally reflected hemispherical volume of 20.2 liters is equivalent to a 10.5 liter nominally reflected sphere, which is subcritical for a concentration up to at least 1000 g Pu/l. Note that the hemispherical volume calculation was selected to more realistically envelope available containers in the glovebox. Also note that a concentration of 1000 g/l accounts for varying concentrations of fissionable material, particularly in the one liter cans. Thus the extra container condition is subcritical.

6.6 Glovebox 2404:

6.6.1 GB-2404 is located in room 1115. It is constructed of 1/8" lead shielding sandwiched between two 1/8" stainless steel layers and is nominally 11' 8" long x 3' ¾" wide x 4' 6" high. Additional shielding by 2" water walls is on the front and back sides of the box. This glovebox contains two parallel banks with three 1-micron bag filters connected in series per bank. Each bag filter canister FL2416-1, 2, 3 (A/B) is nominally 4.5" ID by 18" in height and is spaced as shown in figure 2. This glovebox also has a vacuum pickup wand connected to fulflo filter Fl-2417 (filter housing is nominally 3.75" ID x 8" H) for spill pickup and transfer to tanks D-2401 A/B/C/D. Two standard SG-508 criticality drains are installed in GB-2404, thus flooding above a two inch depth is precluded.

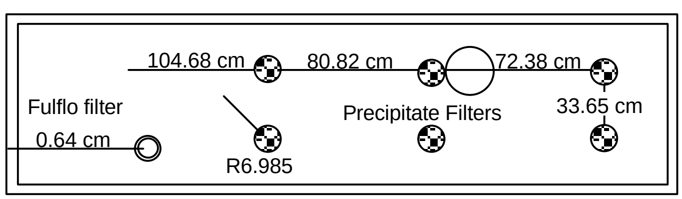

Figure 2 GB2404 Filter

6.6.2 Supernate liquid from the R6 filter in glovebox 18 is further filtered in this glovebox. Fissionable materials in this liquid have previously removed via precipitation in clarifier columns T2411 A/B and filtration through the R6 filter in glovebox 18. Testing of this process, reference correspondence from L. L. Martella to B. S. Mo, "Filtrate Concentration OffR-6 Filter", dated December 13, 1995, in Appendix O, has shown that this liquid is expected to be less than gram value, i.e., typically 10-2 g/l Pu. However, in the event that upset conditions result in gram value concentrations of fissionable material, i.e., greater than 7 g/l fissionable solution introduced into the clarifier columns, failure to precipitate in the clarifier columns, and failure to use the R6 filter, equipment in glovebox 2404 has been configured with geometrically favorable equipment and large spacings such that subcriticality is maintained. Individual equipment is discussed as follows.

6.6.3 Bag-filters FL2416-1,-2,-3 (A/B): Each bag-filter canister is nominally 4.5” ID by 18” H and is configured as shown in figure 2. Since minimum center-to-center spacing between filters nominally ranges from 33 to 83 centimeters, bag-filter canisters are individually evaluated as follows. The worst case physical configuration for an individual bag-filter canister is in contact with four 4” D bag-filters (an extra is included in consideration of a limit violation). Per section 6.5.1, each bag filter contains at most 50 grams fissionable material. Accordingly, four bag filters next to the bag filter canister would have a maximum fissionable mass of 250 grams which is less than the minimum subcritical mass limit of 450 grams as referenced in TID-7016. 200 ml of samples may be taken and handled as the small volume would not significantly affect reactivity.

6.7 Glovebox 2403: GB-2403 is located in room 1103. It is constructed of stainless steel with 0.25” lead and is nominally 15’ long x 2’ 5” wide x 4’ 9” high. This glovebox contains two maximum-size 1-liter pumps (P2403 A/B), one vacuum pick-up wand with fulflo filter FL2410, and fulflo filter FL2411 installed on the return line from pumps P2403 A/B to tanks D-2401 A/B/C/D. Process sampler, ME-2404, is used to sample tanks D-2402 A/B. Samples are collected and removed via sample drop. Bag filter FL-2422 does not require a filter from a criticality standpoint, since wastes transferred to Building 374 have already been repeatedly filtered. A standard SG-508 criticality drain is installed in GB-2403, thus flooding above a two-inch depth is precluded. The vacuum pickup wand may be used for spills both internal and external to GB-2403. Note that external spills may only be picked up with the wand if sample analysis shows liquids to be less than 4x10-3 g/l Pu and U-235. These requirements are in place to support double contingency of liquids transferred to Building 374.

In Appendix M the total volume of two pumps, two filter housings, three used fulflo filters, and 200 ml of samples in GB-2403 was estimated at 8.2 liters. Increasing the sample volume to 1000 ml will result in a total volume of 9 liters. This is clearly subcritical, even in a spherical volume, for concentrations up to at least 250 g/l, as referenced in figure 32 of LA- 10860. This is a reasonable upper bound since the highest concentration of Pu is less than 10 g/l in tank D-55B; process tanks operated at a limit of 150 g/l; high-level Pu solutions are no longer generated in Building 371, liquids are sampled as less than 150 g/l fissionable material before introduced into the CWTS process, and solutions handled in this glovebox have been filtered through an R6 filter and three 1-micron bag filters. Note also that full water reflection is only possible in the unlikely event that the glovebox criticality drain becomes blocked. Thus it is asserted that equipment in GB-2403 is subcritical for any credible contingency.

6.8 Transfer of waste liquid to Building 374:

6.8.1 Waste liquids which are transferred to tanks D-811 A/B must contain less than the Building 374 waste transfer limit of 4 x 10-3 g/l Pu + U-235 since these tanks are critically unsafe. Though effluent liquids from the CWTS have undergone precipitation to remove Plutonium, and supernate liquids from glovebox 18 have been filtered through an R6 filter and three l-micron bag filters, rigorous controls have been instituted to preclude any likely contingency in the CWTS which could cause concentrations exceeding the waste transfer limit to be inadvertently sent to tanks D-811 A/B. Note that caustic liquids are only filtered since particles are suspended and not dissolved. Valve MV-24425 is closed and locked out, and valves MV-24401 and MV-24527 are verified as closed when solution is pumped into clarifier columns T-2411 A/B to preclude bypass of the precipitation process in glovebox 18. Note that residual liquid between closed valves would not pose a criticality safety issue as it would be of small volume. Also note that correspondence from T.E. Boyd to L. Martella, “Precipitation of Plutonium with Magnesium Hydroxide: TES 008.97”, October 10, 1997, shows that supernate liquid from precipitation of 25 g/l fissionable liquids would be less than 7 g Pu/l. Reference Appendix O.

6.8.2 Supernate liquid from clarifier columns T2411 C/D are pumped through either of the banks of three 1 -micron bag filters FL2416-1, 2, 3 (A/B) into either of tanks D-2402 A/B. It would be unlikely for this liquid to be in excess of 7 g/l Pu since it would require a combination of upset conditions, including failure to precipitate, failure to use the R6, and/or failure to use the one or more of the l -micron bag filters. This forms the first contingency.

6.8.3 Waste liquid in tank D-2402 A or B is mixed via vacuum sparge and sampled by the Building 371 operating group prior to transfer to tank D-2403. If lab results show the liquid to be less than the waste transfer limit, valve AOV-213, which is normally locked out in the closed position, is opened and the tank contents transferred to tank D-2403. Liquid above the waste transfer limit is recycled back to tanks D-2401 A/B/C/D and reprocessed in the CWTS. Waste liquid in tank D-2403 is then mixed via vacuum sparge and sampled twice by the Liquid Waste Treatment organization from Building 374, with additional mixing between samples. If both of these samples are less than the waste transfer limit, valves AOV-201, AOV-206, and MV-0884, all normally locked in the closed position, are opened and the waste liquid is pumped to tanks D-811 A/B in Building 374. In addition, tank D-2403 is limited to a maximum of 200 grams total fissionable material (39% of the minimum critical mass of Pu-239) and assayed on a bi-monthly basis as additional barriers. Since two independent organization have multiple opportunities to verify waste liquid concentration, numerous independent samples would have to be misanalyzed, and several normally locked closed valves would have to fail or be left open, it is judged that failure to transfer only waste liquids below the waste transfer limit from tank D-2403 to tanks D-811 A/B is unlikely. This forms the second contingency.

6.9 Direct draining of Building 371 tanks to tanks D-2401 A/B/C/D:

One method of draining Building 371 tanks involves the installation of a tap in the outlet of a tank, which is then directly connected to tanks D-2401 A/B/C/D via nominal ¾” tygon tubing into the manifold in GB-2401. The contents of the target tank are then drained into the CWTS using gravity assisted by the vacuum (PROVE Vacuum System) on tanks D-2401 A/B/C/D. This activity can be conducted with Building 371 General Comments and Miscellaneous NMSL/CSOLs. Since Building 371 tanks were pumped systems and no fissionable solution was introduced into the vacuum system by overfilling tanks, no specific criticality controls are required. Note that tanks currently connected via hard piping into GB-2401, reference section 6.2, may be drained into the CWTS. Also note that mixing of tanks prior to draining is not addressed as there is no operational requirement for it. Finally, this evaluation does not authorize the violation of USQDs which are in effect.

6.10 Processing of liquids from Tank D-452, Bldg 771: Liquids drained from tank D452, Building 771 have been well characterized as less than 6 g/l fissionable material. Reference Appendix P. Therefore, the existing safety basis and controls established for a maximum credible concentration of 150 g/l fissionable material well bound this evolution.

6.11 Processing of liquids from other facilities: Liquids from other facilities which are characterized as less than 150 g/l fissionable material are bounded by the safety basis and controls established in this evaluation for a maximum credible concentration of 150 g/l fissionable material.

7.0 DESIGN FEATURES AND ADMINISTRATIVELY CONTROLLED LIMITS AND REQUIREMENTS

7.1 Design Features: The technical basis for each engineered safety feature is addressed in sections 5.0 and 6.0.

7.1.1 GB-2401: Piping manifold per drawing D-25321-001, floor pickup wand with fulflo filter configured to return liquid to tanks D-2401 A/B/C/D, and standard criticality drain.

7.1.2 GB-2402: Maximum 1 -liter pumps (P2401 A/B), floor pickup wand with fulflo filter configured to return liquid to tanks D-2401 A/B/C/D, and standard criticality drain.

7.1.3 GB-2403: Maximum 1-liter pumps (P2403 A/B), floor pickup wand configured to return liquid to tanks D-2402 A/B, and standard criticality drain.

7.1.4 GB-2404: Each bag filter canister, FL-2416-1,2,3 (A&B), is nominally 4.5” ID by 18” H and is configured per drawing D-50407-451. Two standard criticality drains are installed on this box. This box also includes a floor pickup wand with fulflo filter configured to return liquid to tanks D-2401 A/B/C/D. Bag filters not to exceed 1 micron.

7.1.5 GB-18: Each column,T2411 A/B/C/D, is nominally 6” ID by 66” H. The R6 filter assembly is a maximum of 2 inches deep. Equipment is configured per drawings D-50407-0462 through -0467. The pump is timed to deliver no more than about 20 liters of liquid to each clarifier column. A standard criticality drain is installed on this box.

7.1.6 D-2401 A/B/C/D; D-2402 A/B; D-2403: Each shielded annular tank is constructed per drawings D41005-501 through -505.

7.2 Administrative Controls (Posted Limits): The technical basis for each of these controls is discussed in sections 5.0 and 6.0.

7.2.1 GB-2401: Liquids introduced into the manifold are limited to a maximum of 150 g/l fissionable material. Six used fulflo filters may be present outside the cartridge filter housing. One maximum-size 4-liter container of maximum 150 g/l fissionable material may also be in GB-2401. Floor pickup wand may be used to clean up spills inside and outside the glovebox. One maximum-size 2-liter container of kimwipes used for spill collection (free liquid is allowed) may also be present. 200 ml of samples may be taken and handled in this glovebox.

7.2.2 GB-2402: Three used fulflo filters may be present outside cartridge filter housings. 1000 ml of samples may be taken and handled in this glovebox. Floor pickup wand may be used to clean up spills inside and outside the glovebox. One maximum-size 2-liter container of kimwipes (free liquid is allowed) used for spill collection may also be present.

7.2.3 GB-2403: Three used fulflo filters may be present outside cartridge filter housings. 1000 ml of samples may be taken and handled in this glovebox. Floor pickup wand may be used to clean up spills inside and outside the glovebox. Samples from liquid outside the glovebox must show liquid to be less than 4 x 10-3 g/l Pu and U-235 prior to spill clean up.

7.2.4 GB-2404: Three used filters may be present outside filter housings. 200 ml of samples may be taken and handled in this glovebox.

7.2.5 GB-18: Tank contents must be characterized as less than 25 g/l Pu before transferred to the clarifier columns (T2411 A/B). Material (precipitate) from the R6 filter may be placed into a one-layer planar array of three maximum dimension 6"x 9.5"x 2" pans. Only precipitate from one processing "run" (2 columns) is allowed into any single pan. Precipitate is restricted to the approximate height of each pan, however, it is acceptable for small amounts to heap slightly above the pan height during loading or as a result of pan handling. Four used R6 filters may be present in addition to the one in the R6 filter housing. Two used fulflo filters may be present outside of the cartridge filter housing. 1-liter of samples (any number of containers less than or equal to l-liter combined volume) may be taken and handled in this glovebox. No stacking.

7.2.6 GB-23: Limited to two maximum-size 1-liter containers, each limited to a maximum of 1000 grams net weight, and three maximum dimension 6"x 9.5" x 2" pans of precipitate from glovebox 18. Precipitate is restricted to the approximate height of each pan, however, it is acceptable for small amounts to heap slightly above the pan height during loading or as a result of pan handling. In addition, three used R6, fulflo, and/or bag filters from the CWTS may be present in this box. No stacking of pans allowed.

7.2.7 Tank D-2403: Limited to a maximum of 200 g Pu + U235. Tank must be assayed every 2 months.

7.3 Administrative Controls (Procedural):

7.3.1 Sampling and valve-positions for waste transfers to Building 374, as required in section 6.8.3, are controlled by procedure 4-R76-WO-5017.

7.3.2 Precipitation in GB-18 is controlled by procedure 4-U84-CO-6090.

7.3.3 Manual valves MV-24401 and MV-24527 are closed when clarifier columns T2411 A/B are filled, per section 7.1 of procedure 4-U84-C0-6090.

7.3.4 Operations order or procedure requires fissionable liquids to be characterized as less than 150 g/l Pu before acceptance by Bldg 371 Operations for processing.

7.3.5 Bag-in procedures require bag-in bags to be secured such that fissionable liquids cannot accumulate within them when unattended.

7.3.6 Procedure 4-U84-CO-6090 is to provide instructions that precipitate from only one processing "run" is to be placed into any single pan.

7.3.7 Procedure 4-U84-CO-6090 is to provide instructions that each column is to be adjusted to a volume of 20 liters or less before Magnesium Oxide is added for precipitation.

7.3.8 Procedure 4-U84-CO-6090 does not allow bottles of caustic liquid from other facilities without approval from Building 371 Operations and can only be introduced into the system if the receiving tank is empty.

7.3.9 Procedure 4-U84-C0-6090 requires inlet to tanks D-2401 AIB/C/D to be valved closed if greater than four liters of caustic liquid is in any glovebox or room which drains to CDS "A". A tank may only receive these liquids if it is empty and isolated from the other tanks.

7.4 Other Administrative Controls

7.4.1 Valve MV-24425 is closed and locked out per HSP-2.08.

8.0 SUMMARY AND CONCLUSIONS

The Caustic Waste Treatment System (CWTS) is comprised of tanks and equipment of geometrically favorable dimension which have been demonstrated subcritical for fissionable solutions well in excess of historical and future generated process liquids in Building 371. In addition, the processing of liquids from other facilities characterized as less than 150 g/l Pu + U235 are bounded by the controls and safety basis in this evaluation. Waste transfers to Building 374 are conducted using multiple samplings by several independent organizations and interim tank transfers. The activities associated with direct draining of tanks, CWTS operations, and waste transfer to Building 374 are accordingly double contingent.

9.0 REFERENCES

9.1 WDW-01.3
9.2 LA-10860
9.3 TID-7016
9.4 Procedure 4-U84-C0-6090, “Building 371 Caustic Waste Treatment System”
9.5 Procedure 4-R76-WO-5017, “Caustic Waste Transfer from Building 371”
9.6 LA-12808
9.7 GSR-043
9.8 HD-20

Intentionally Blank

EXAMPLE 6

Provided By

Savannah River Site

Nuclear Criticality Safety Evaluation: Experimental Breeder Reactor (EBR)-II & Taiwan Research Reactor (TRR) Fuel Dissolving in F-Canyon

Intentionally Blank

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Intentionally Blank

1.0 Introduction

F-canyon personnel are planning to dissolve Experimental Breeder Reactor II (EBR-II) bundle DU006 in the annular dissolver in F-canyon. This bundle contains a failed bundle of 41 rods, each containing fuel slugs exposed in the blanket of EBR-II. The Nuclear Materials Stabilization Program Division has decided to dissolve this bundle, along with 81 failed Taiwan Research Reactor (TRR) fuel rods that are also targeted for processing, to eliminate the possibility of future leakage problems and eliminate the vulnerabilities presented by these materials as identified by the DNFSB in the 94-1 decision.

Though the EBR-II bundle contains depleted uranium, and the TRR fuel rods contain natural uranium, there is still a criticality concern because of the bred plutonium that they contain. Previous studies (Ref. 1 and 2) have proven that natural uranium TRR rods containing plutonium can be dissolved in a critically safe manner. The difference between this current study, and those previous studies, is the presence of the EBR-II bundle. Due to its size, the EBR-II bundle must be inserted into the dissolver without using an insert to hold it. Therefore, it must be assumed that the EBR-II bundle is able to fall over in the dissolver. This causes additional criticality concerns that are not present when only TRR rods are in the dissolver, since they have a small enough diameter to fit into the inserts. But the BR-II fuel in bundle DU006 has a similar equivalent U-235 enrichment to the fuel in the TRR bundles (0.77 % vs. 0.79 %), so that including the EBR-II bundle in the dissolution is a relatively small perturbation on what has been dissolved in the past.

This document records the results of the nuclear criticality safety evaluation (NCSE), including consideration of contingencies, for the dissolution of BR-II bundle DU006 along with 81 TRR bundles in the F-canyon annular dissolver in four dissolver batches. It was found that the fissile enrichment, including U-235 and Pu-239, is low enough to assure the criticality safety for this EBR/TRR dissolution.

2.0 Description

2.1 Dissolver

The EBR-II bundle DU006 and TRR bundles are to be dissolved in one of the two annular dissolvers in building 221-F (F-canyon). This NCSE applies to either dissolver tank 6.1 or 6.4 since they both have the same size and geometry. Any difference between these two tanks is not relevant to their criticality safety. Each of these dissolver tanks, shown in Figure 1, is constructed of stainless steel and concrete consisting of an outer annulus section (approximately 10 feet diameter by 13 feet high), an inner annulus section (8 by 20 feet), and a hollow center section (6 by 20 feet). The lower portion of the vessel contains an 8-inch concrete neutron shield (5.5 by 8 feet). The outer annulus contains heating and cooling coils in addition to instrumentation tubing that is used to detect liquid level, specific gravity, and differential pressure. The inner annulus contains a fuel crib where the fuel sits during dissolution. The bottom of the dissolver contains an air sparge ring which provides agitation.

2.2 Port Inserts

There are four access ports, 90° apart, around the top of the dissolver (Figure 2) which are used to place material into the inner annulus of the dissolver. Mark-42 inserts (each containing two TRR inserts as shown in Figure 3) and a 3-well insert (Figure 4) will be placed into the dissolver ports to hold TRR bundles, with each TRR bundle holding one TRR rod. Each Mark-42/TRR insert has locations to hold six TRR bundles. The 3-well insert contains three cylindrical tubes (wells) which can each hold a TRR bundle. The inside diameter-of each well in the 3-well insert is 5.487”. It is planned to place the EBR-II bundle into the dissolver without using an insert in the port, since the EBR-II bundle is too large to fit into either of these inserts. The EBR-II bundle will simply be lowered by the crane through one of the ports to sit on the bottom of the dissolver.

2.3 Materials to be Dissolved

Each assembly used in the blanket of EBR-II contained 19 rods, each containing 5 depleted uranium (0.22% U-235) slugs, 0.433 inch OD x 11 inches long, sodium bonded to 304 stainless steel cladding. As a result of irradiation in EBR-II, these slugs bred varying amounts of plutonium, depending on their location and residence time in the blanket of EBR-II. At Rocketdyne in the mid 1980s, these EBR-II blanket assemblies were processed by removing the steel cladding and sodium from the blanket slugs, and loading them into thin-walled aluminum tubes (up to 14 slugs per tube). These aluminum tubes containing the slugs were bundled together, with each bundle containing 41 tubes. Each bundle was then placed into a welded aluminum can. Each can was 14 ft. long, with a diameter small enough to fit into the wells of the 3-well insert (< 5.375 in.). Each bundle in a can could hold the slugs from up to 6 blanket assemblies (6 x 19 x 5 = 570 slugs). EBR-II can DU006 was produced by this process, and was originally received at SRS in 10-86. In June 1992, can DU006 was found to have failed due to a lid weld defect. The can was then encapsulated in a vented overpack bundle, and isolated on 11-17-92 in row 15 of Bundle and Test Tube Storage at the Receipt Basin for Offsite Fuels (RBOF). The OD of the overpack bundle is 6.469 inches. This dimension is too large to fit into a dissolver insert well, so that DU006 will have to be placed into the dissolver without using an insert in the port. The heavy metal weights in DU006 are given in Table 1 as 285 kg of uranium, 0.628 kg U-235, and 0.789 kg of plutonium. The plutonium composition should be greater than 98% Pu-239. It is planned to dissolve this bundle during the dissolving of failed Taiwan Research Reactor (TRR) fuel that is also targeted for processing.

TRR fuel is natural uranium metal rods (1.36” OD and 120” long) in aluminum cladding (1.52” OD), and has historically had nominal weights of 1.45 kg Al, 0.065 kg Pu, and 53.6 kg U per rod (Ref. 3). The TRR fuel to be processed consists of 81 failed fuel rods currently stored in RBOF. The condition of these rods ranges from those which are still intact but have an indication of breached cladding, to those that are fully oxidized, to bits and pieces mixed with oxide. These 81 failed rods are stored in aluminum bundles (each “bundle” is a cylindrical can which is 131” long) with one rod per bundle. Of the 81 bundles, 54 bundles have an inside diameter (ID) of 2.75 inches, and 27 bundles have an ID of 4.00 inches. Though it is possible for a 4.00 inch ID bundle to contain up to four TRR rods (Ref. 14), there is only one TRR rod in each of the 81 TRR bundles that will be in this dissolution. The normal 5.00 inch bundle, which can contain up to seven TRR rods, will not be used in this dissolution. Both the 2.75 inch ID bundles, and the 4.00 inch ID bundles can fit into either the Mark-42/TRR insert or the 3-well insert. The total heavy metal weights of U, U-235, and Pu in the 54 bundles and the 27 bundles are given in Table 1, based on data in Ref. 15.

It is planned to dissolve the EBRn bundle DU006 and the 81 TRR bundles in four batches, as given in Table 2. In all four batches, six TRR bundles will be placed into each of three Mark-42/TRR inserts (Figure 3), which are to be located in three of the four loading ports. In batches one, two, and four, three TRR bundles will also be placed into a three well insert (Figure 4), which is to be located in the remaining port. The EBR-II bundle DU006 will be placed by itself into the dissolver port in batch 3 without the use of a loading port insert.

The data in Tables 1 and 2 can be used to calculate the expected average loadings for each insert and for each batch. These are given in Tables 3 and 4. In Table 3, for example, the average amount of uranium that would be expected in a Mark-42/TRR insert was calculated from data in Table 1 to be (2758 + 1180) x 6 / (54 + 27) = 291.7 kg, and the average amount of U-235 is (17.02 + 7.299) x 6 / (54 + 27) = 1.802 kg. In Tables 1, 3, and 4, a conservative value of 2.0 is used for the Plutonium Equivalency Factor (see Section 6.1) to calculate the equivalent U-235 loading or enrichment. For example, the average value for the equivalent U-235 loading in the Mark-42URR insert, as listed in Table 3, is (1.802 + 2.0 x 0.246) = 2.29 kg, using the average U-235 loading (1.802 kg) and the average plutonium loading (0.246 kg) also listed in Table 3. The equivalent U-235 enrichment is then 2.29 / (291.7 + 0.246) = 0.79 %. In Table 4, the expected uranium loading in the third batch can be calculated from data in Table 3 to be (3 x 291.7) + 285 = 1160.1 kg.

As listed in Table 3, the maximum fissile content of the TRR rods must also be considered in this NCSE. Previous data (Ref. 1 and 14) have indicated that the highest plutonium content in a TRR rod was 79 grams at 0.1475 wt. % Pu. For this rod, the heavy metal weight must have been 79 /.001475 = 53560 grams, which would have had a U-235 weight of (53560 - 79) x 0.0072 = 385 g. If it is assumed that all of the 79 grams plutonium is Pu-239, then the equivalent U-235 enrichment for this rod would be (385 + 2.0 x 79) / 53560 = 1.01 %. If six of these maximum plutonium TRR rods were loaded into a Mark-42 / TRR insert, then the total uranium loading would be 6 x (53560 - 79) = 320.9 kg, and the Pu-239 loading would be 6 x 0.079 kg = 0.474 kg. These values are listed in Table 3 for the “Max. Pu” case for the Mark-42/TRR insert.

Table 1. Materials to be Dissolved (Based on Ref. 15)

Fuel	Items	Length	Diameter	Weights in Kg Total U		Total Pu	Equivalent U-235 Enrichment
EBR-II	1 bundle	14' 2.25"	6.469" OD	285	0.628	0.789	0.77 %
TRR	54 bundles	131"	2.75" ID	2758	17.02	2.341	0.79 %
TRR	27 bundles	131"	4.00" ID	1180	7.299	0.985	0.78 %
Total:	82 bundles			4223	24.95	4.115	0.78 %

Table 2. Number of Bundles Loaded During Each Batch

	First Port		Second Port		Third Port		Fourth Port		Entire Dissolver	
Batch	TRR	EBR-II	TRR	EBR-II	TRR	EBR-II	TRR	EBR-II	TRR	EBR-II
1	6	0	6	0	6	0	3	0	21	0
2	6	0	6	0	6	0	3	0	21	0
3	6	0	6	0	6	0	0	1	18	1
4	6	0	6	0	6	0	3	0	21	0
Total									81	1

Table 3. Heavy Metal Loadings Per Port

Type of Insert	Batches That Insert Are Present	Fuel in Insert	Fuel Type Assumed	Loadings Per Port (kg) Total U		Pu-239	Equivalent U-235 Loading	Equivalent U-235 Enrichment
Mark-42	1, 2, 3, 4	TRR	Average	291.7	1.802	0.246	2.29	0.79 %
			Max. Pu	320.9	2.310	0.474	3.26	1.01 %
3-Well	1, 2, 4	TRR	Average	145.9	0.901	0.123	1.15	0.79 %
			Max. Pu	160.5	1.155	0.237	1.63	1.01 %
No Insert	3	EBR-II	Total	285	0.628	0.789	2.21	0.77 %

Table 4. Average Loadings Per Batch

	Average Loadings Per Batch (kg)				
Batch	Total U	U-235	Pu-239	Equivalent U-235 Loading (kg)	Equivalent U-235 Enrichment
1	1021.0	6.305	0.862	8.03	0.79 %
2	1021.0	6.305	0.862	8.03	0.79 %
3	1160.0	6.033	1.528	9.09	0.78 %
4	1021.0	6.305	0.862	8.03	0.79 %
Total	4223.0	24.95	4.115	33.18	0.78 %

2.4 Dissolution Process

In the first batch, the aluminum bundle and cladding will be dissolved using 50% sodium hydroxide (NaOH), and the aluminum waste products will be transferred to waste. The U-Pu metal will then be dissolved with nitric acid (HNO_3), according to the following chemical reactions:

$$U + 4(HNO_3) \ ? \quad UO_2(NO_3)_2 + 2(NO) + 2(H_2O)$$

$$U + 8(HNO_3) \ ? \quad UO_2(NO_3)_2 + 6(NO_2) + 4(H_2O)$$

$$Pu + 4(HNO_3) \ ? \quad Pu(NO_3)_4 + 2H_2$$

so that the uranium and plutonium, when they are dissolved, will be in solution as either $UO_2(NO_3)_2$ or $Pu(NO_3)_4$. The nitric acid solution present in the dissolver at the end of the first batch will remain in the dissolver, and additional nitric acid and mercury as a catalyst will be added to dissolve the materials in the second batch. After dissolution of the second batch is completed, the product solution will be processed as for TRR dissolution solutions in the past. For the third batch, the three well insert will be removed from the dissolver port, and the EBR-II bundle DU006 and the 18 TRR bundles will be loaded into the dissolver. NaOH will again be used to dissolve the aluminum bundles and cladding, and the aluminum waste products will be transferred to waste. It should be noted that the individual fuel slugs in the EBR-II bundle will fall to the bottom of the dissolver as the aluminum in the bundle and cladding is dissolved. The 0.5 inch spacing of the slats at the outer bottom of the charging crib will restrict pieces of the EBR-II fuel from escaping the inner annulus of the dissolver until most of the fuel has been dissolved. After the aluminum solution has been removed from this third batch, nitric acid will again be used to dissolve the U-Pu metal. At the start of batch 4, the three well insert is replaced into the dissolver port, and the 21 TRR bundles are loaded into the four ports. Nitric acid and mercury as a catalyst will be added to the solution remaining from the third batch to dissolve all the materials of the fourth batch. When these materials are all dissolved, the product solution will be processed as for TRR dissolution solutions in the past.

During each batch discussed above, nitric acid in excess of that required to dissolve the uranium and estimated aluminum will be added to the dissolver to minimize the possibility of acid deficient dissolver solution causing plutonium polymerization. After the nitric acid is added to the dissolver, the solution is heated to its specified temperature range and maintained there until dissolution is complete. During the dissolution, the specific gravity is monitored, and at the end point, sampling is used to calculate the concentration of the uranium and plutonium that is present. A material balance is then done to confirm the completeness of dissolution before subsequent charges are made. In the event that a significant discrepancy (> 10%) exists between the known mass of the material charged and the measured mass in the solution, the dissolver will be probed using the probing device, and an additional dissolution cycle (without additional fissile charges) will be made.

3.0 Requirements Documentation

This evaluation was completed according to procedure manual WSRC-SCD-3 "WSRC Criticality Safety Manual" (Ref. 4). Inherent in the requirements of SCD-3 are applicable ANS standards and DOE Order 5480.24 (Ref. 5).

4.0 Methodology

Computer code calculations were not needed for this NCSE due to the low fissile enrichment of the EBR-II and TRR fuel rods being dissolved. Since these rods contain both U-235 and plutonium, an equivalent U-235 content and enrichment was calculated for various configurations of these rods, which were then compared to "safe" subcritical limits (13.1 kg U-235 for $\leq$ 1.0 % enriched, and 10.0 kg for 1.01 % enriched) published in national standards and other published criticality data. These limits of 13.1 and 10.0 kg U-235 were obtained from an optimally moderated and reflected lattice in a spherical configurations. In general, a critically "safe" configuration means that the calculated K = 0.95. However, for this NCSE, a critically "safe" K = 0.98 was accepted for data coming from Ref. 6 and 7 because of their wide acceptance and long use as sources of valid nuclear criticality data. A critically safe limit of 19 MT of uranium was also used based on KENO calculations (Ref. 1) for the actual annular dissolver geometry and steel material in the dissolver walls.

Since the dissolution of the EBR-II bundle is the main difference from previous campaigns where TRR rods have been dissolved, the previous double contingency analysis (DCA) for TRR rods (Ref. 2) was used for initiating events that are common between this EBR/TRR dissolution and previous TRR dissolution. Possible initiating events that were altered by the presence of the EBR-II bundle in this EBR/TRR dissolution, compared to the previous TRR dissolution, include the EBR-II bundle falling over in the dissolver, a missing insert, and double batching. The criticality safety of these events was considered, and it was found that these events can not cause a criticality. The only new initiating event that could be found that might cause a criticality was starting the EBR/TRR dissolution with a significant amount of fissile material already in the dissolver solution from the previous dissolver campaign. A criticality safety limit, and criticality safety controls, were specified to preclude this initiating event from occurring.

5.0 Discussion of Contingencies

DOE Order 5480.24 Ref. 5) and WSRC Nuclear Criticality Safety Manual (Ref. 4) require sufficient factors of safety so that at least two unlikely, independent, and concurrent changes in process conditions shall occur before a criticality is possible. The protection factors shall provide either two independent process parameter controls or multiple controls (defenses) of a single controlled parameter.

The envelope of criticality safety for dissolution of the EBR-II bundle and the TRR bundles is summarized in Table 5, based on consideration of the nine criticality control parameters (fissile mass, moderation, geometry, spacing, concentration / fissile density, neutron absorption, enrichment, reflection, and temperature). Table 5 lists the bounding assumptions (BA) and criticality safety limits (CSL) for the nine criticality control parameters. Since the average equivalent enrichments of the EBR-II and TRR materials are both so low (0.77 % and 0.79 %; respectively), the restrictions that had to be assumed in the criticality safety evaluation (Section 6) are minimal: no limits were assumed on the moderation, spacing, neutron absorbers, reflection, or temperature. The fissile mass and the enrichment are controlled only in the sense that the items allowed to be loaded into the dissolver are restricted to the specified EBR-II bundle and TRR bundles. The concentration / fissile density is controlled two ways: (1) The beginning of the first batch will contain 5 0.1 gram fissile per liter in the jet heel after the last rinse just prior to the first batch. This limit only applies if the enrichment of the materials in the previous dissolution campaign > 0.9635 %.

Table 5

Summary of Criticality Safety Envelope for dissolution Of EBR-II bundle and TRR Bundles

Criticality Control Parameter	Bounding Assumption (BA)	Criticality Safety Limit (CSL)
Fissile Mass	Unrestricted, except that only the TRR bundles and EBR-II bundle may be loaded into the dissolver.	
Moderation	Unrestricted	
Geometry	Fuel rods are unrestricted. Annular dissolver geometry is assumed.	
Spacing	Unrestricted	
Concentration, Fissile Density	Fuel rods are unrestricted.	Nitric acid concentration is high enough in the dissolver to prevent Pu polymerization. Must be $\leq$ 0.1 g fissile/liter in jet heel after last rinse prior to first batch.
Absorbers	Unrestricted	
Enrichment	Uranium is depleted (EBR-II bundle) or is natural (TRR bundles), since only the TRR bundles and EBR-II bundle may be loaded into the dissolver.	
Reflection	Unrestricted	
Temperature	Unrestricted	

And, (2) the nitric acid concentration in the dissolver will be maintained high enough to prevent plutonium in solution from polymerizing. The geometry is assumed to be that of the annular dissolver. Since the EBR-II bundle will not be placed into an insert, there is no geometry control to prevent it from falling over in the dissolver. This is an allowed condition, and is shown to be critically safe in Sections 6.2 and 6.3. Process changes (initiating events) which can challenge the criticality safety of the EBR/TRR dissolution can only come from exceeding or violating the assumptions / limits in Table 5. Based on these assumptions and limits, only three process changes can challenge criticality safety. These possible initiating events are:

IE #1. Other than the intended EBR-II and TRR material could, by mistake, be loaded into the dissolver.

IE #2. The dissolver solution may not be maintained sufficiently acidic. This could cause a local concentration of plutonium to build up due to polymerization of the plutonium in the dissolver solution.

Initiating events IE #1 and IE #2 are common to the previous campaign which dissolved only TRR rods, i.e. the presence of one EBR-II bundle does not make these initiating events more severe than they were in the TRR campaign. This means that the double contingency analysis (DCA) for these initiating events for the EBR/TRR dissolution is already covered by the DCA for dissolution of only TRR rods, which is given in Ref. 2. By adopting the administrative controls from the TRR dissolution for the EBR/TRR dissolution, the results of a DCA for the above two initiating events will automatically be incorporated into the administrative controls for the EBR/TRR dissolution. Therefore, there is no need to repeat that documentation here.

IE #3. The dissolver solution at the start of the first EBR/TRR batch could already contain significant (defined below) fissile material, at a significant fissile enrichment (> 0.9635 %), from the previous dissolver campaign. This initiating event only applies if the enrichment of the materials in the previous dissolution campaign > 0.9635 %, i.e. was plutonium without uranium.

As shown in Section 6.3, initial dissolver solutions containing fissile enrichments $\leq$ 0.9635 % can only decrease the dissolver K-eff, so that there are no additional criticality safety requirements for these low enriched (E $\leq$ 0.9635 %) solutions. But it is also possible for the fissile enrichment of the initial dissolver solution to be > 0.9635 %. For example, this may be due to the previous dissolution of the sand, slag, and crucible (SS&C) material that contains plutonium but no uranium. If the fissile enrichment is above 0.9635 % in the dissolver solution just prior to the first batch, then sampling must be done to prove that the amount of fissile material in solution is insignificant. Since an average equivalent U-235 loading for the first batch will be approximately 8.03 kg (Table 4), a value of 0.01 kg = 10 grams fissile in solution will have a very insignificant effect on the dissolver K-eff. 10 grams of fissile material in solution thus qualifies as an “insignificant amount”. Therefore, if E > 0.9635 %, then during the last rinse after the previous dissolution campaign, and before loading the first batch for the EBR/TRR dissolution, the rinse solution must be sampled to assure that it contains $\leq$ 0.1 grams fissile (U-235 + Pu) per liter. When the pump removes the rinse from the dissolver, it will leave approximately 100 liters of the solution, called the jet heel, in the bottom of the dissolver tank. So if the fissile concentration is measured to be $\leq$ 0.1 g/liter, then $\leq$ 10 grams of fissile material (0.1 g/liter x 100 liters) will be present in the dissolver just before the EBR/TRR material is loaded for the first batch. This value of $\leq$ 0.1 gram fissile/liter is the criticality safety limit (CSL) for IE #3 to assure criticality safety.

If the fissile material in the previous dissolution campaign has an enrichment > 0.9635 %, then the process operator must have dual independent sampling and chemical analysis (blue label) to determine the fissile concentration of the last rinse prior to the EBR/TRR material loading. The process operator must be assured that both values of the chemical analysis of the fissile concentration are $\leq$ 0.1 grams per liter in the last rinse of the previous dissolution solution, and that this last rinse is transferred out of the dissolver (this would leave a jet heel of approximately 100 liters) before the new liquids for the EBR/TRR dissolution are brought into the dissolver.

Second Defense: The second sampling and analysis acts as a defense against the first sampling and analysis producing a wrong value for the fissile concentration in the solution. As a defense against operator error, a second individual such as a supervisor must independently verify whether dual independent sampling and analysis was needed or not (based on the enrichment of the material in the previous dissolution), and if it was needed, whether dual independent sampling was done during the last rinse of the previous dissolution campaign, whether the dual independent analysis values were both $\leq$ 0.1 grams fissile per liter, and whether the last rinse was transferred out of the dissolver. Only after the completion of these items (as needed) shall loading of the EBR/TRR material be approved.

Common Mode Failure (CMF) Potential: For both individuals to independently be mistaken about the need for dual independent sampling and analysis is very unlikely, since they would both know, or have access to information regarding the material that was dissolved in the previous campaign. There are no other obvious CMF paths.

6.0 Evaluation and Results

6.1 Plutonium Equivalency Factor

Since the TRR and EBR-II fuel being dissolved contain both U-235 and plutonium, and Pu-239 is more reactive than U-235, the determination of the nuclear criticality safety of the materials was based on consideration of the equivalent U-235 enrichment This is defined as (U-235 + F x Pu) / (U + Pu) wt. $\cong$ % U-235 + F x wt. % Pu-239, where F = Plutonium Equivalency Factor is the effectiveness of an atom of Pu-239 in increasing the system K-eff compared to the effectiveness of an atom of U-235.

In the past (Ref. 8), a value of F = 2.0 was used based on calculations (Ref. 9) of the limiting concentration of PuO_2 mixed homogeneously with natural UO_2 and H_2O. And a value of F = 1.6 is used in the 221-F Building Technical Standards (Ref. 17). For a heterogeneous system of EBR-II blanket rods, a "best" value of F = 1.71 was calculated (Ref. 10). To assure that a conservative value of the equivalent U-235 enrichment would be used for this NCSE, the most conservative value of F = 2.0 was assumed. Using 2.0 for the Plutonium Equivalency Factor gives average equivalent U-235 enrichments of 0.77% for the EBR-II blanket rods in bundle DU006, and 0.79 % for the TRR rods in the 54 and 27 TRR bundles, as shown in Table 1. These values of the enrichment can then be used with published curves and tables to assure the criticality safety of possible configurations.

6.2 Analysis for No Uranium in Solution

For the case where there is initially no fissile material in solution, then standard criticality figures and tables for lattices in water can be used. The curve for uranium metal lattices in water, given in Figure 5 (taken from Figure 1 of Ref. 6), indicates that optimally configured fuel rods with a U-235 enrichment of 1.0 wt. % will be critically safe with up to approximately 12 or 13 kg of U-235. The uncertainty about this value is a result of the difficulty in interpolating on the logarithmic vertical scale in Figure 5. Since this figure is based on the data in Appendix B of Ref. 7, the exact value for 1.0 wt. % uranium metal lattices can be obtained from Table 6 of Ref. 7. This value is 13.1 Kg U-235 to produce a safe K-eff of 0.98 with any rod diameter, any rod spacing, and optimal water moderation in a spherical configuration with full water reflection. t Since the equivalent U-235 enrichment of EBR-II bundle DU006 is less than 1.0 %, the rods in DU006 will be critically safe ($K \leq 0.98$) in any configuration with full flooding as long as there is less than 13.1 kg of U-235, or its equivalent. The equivalent U-235 weight in the EBR-II bundle DU006 can be calculated to be 0.628 + (2.0 x 0.789) = 2.21 kg (Table 3), which is much less than 13.1 kg U-235. Therefore, if the rods in DU006 are isolated from other fissile material, they will always be critically safe. This will be true whether they are intact in the bundle, or whether the aluminum bundle and cladding have been dissolved and the EBR-II fuel slugs have fallen to the bottom of the inner annulus in the dissolver.

By the same reasoning, six average TRR bundles that have been loaded into a Mark42/TRR insert can not go critical. With an effective U-235 enrichment of 0.79 °/e, the limit is again 13.1 kg U-235. The effective U-235 content in a Mark-42/TRR insert, according to Table 3, for average TRR rods is 2.29 kg. This value is far below the 13.1 kg limit for criticality safety.

Variations in the fissile content of the TRR rods must also be considered. According to Table 3, a Mark-42/TRR insert could contain an equivalent U-235 loading of up to 3.26 kg if all of the TRR rods in it had a maximum plutonium content. The equivalent U-235 enrichment of this material would be 1.01 %. From Figure 5, for a lattice of optimally configured and flooded uranium metal rods at 1.01 % U-235, the critically safe mass limit for U-235 is at least 10 kg. This 10 Kg limit is far larger than the 3.26 kg of U235 that could be present in a Mark-42/TRR insert. Therefore, maximum plutonium TRR rods in a Mark-42/TRR insert will be critically safe. Maximum plutonium TRR rods placed into the 3-well insert will also be critically safe for the same reason.

There are three events related to the EBR/TRR dissolution which require special consideration for criticality safety. These events are: 1) the EBR-II bundle falling over in the dissolver, 2) a missing insert, and 3) double batching. It is shown below that none of these events can lead to a criticality.

Since the EBR-II bundle will not be placed into an insert in the inner annulus of the dissolver, there is no control to prevent it from falling over in the dissolver toward one of the Mark-42/TRR inserts containing TRR fuel, especially during dissolution of the bundle aluminum during the NaOH strike. Though the flat bottom of the EBR-II bundle could make it unlikely that the bundle would fall over, the annular geometry of the dissolver and the diameter of the EBR-II bundle would make it possible for the top of the EBR-II bundle to fall against one of the Mark 12/TRR inserts. To bracket this possibility, let us assume that the EBR-II bundle DU006 were, in some unspecified way, to come into a close parallel orientation to the Mark-42/TRR insert containing six TRR rods, each containing the maximum amount of Pu-239. According to Table 3, the equivalent U-235 content of the combined EBR-II bundle and six TRR rods would be 2.21 kg + 3.26 kg = 5.47 kg. The equivalent U-235 enrichment would be 5.47 / (320.9 + 285) = 0.90 %. The 5.47 kg equivalent U-235 is much below the appropriate 13.1 kg limit for $\leq$ 1.0 % enrichment, so that criticality can not occur under any condition with this low fissile content and low enrichment.

An insert may not be present in one of the dissolver ports when it is required. It should be noted that if the insert is not present in the dissolver port, then it would be very difficult for the crane operator to lower six TRR bundles into the port without the inserts to guide them. But if six TRR bundles could be inserted into the port even though the Mark-42/TRR insert is not present, they could fall over toward one of the other inserts containing TRR rods. If the EBR-II bundle also falls (there are no controls against this) toward the same insert, then according to Table 3, the total equivalent U-235 content (for the maximum plutonium case) could be up to 3.26 + 3.26 + 2.29 = 8.81 kg, which is less than the 10 kg limit for < 1.01 % enriched material. This condition will therefore be critically safe.

Double batching of materials into the same port may take place. The worst hypothetical double batching event is to place 12 TRR rods into a Mark-42/TRR insert, instead of 6 TRR rods. Though this may not be possible due to spacing constraints, it is simply assumed to occur for this argument. According to Table 3, the equivalent U-235 loading of this insert (for the maximum plutonium case) would be 3.26 + 3.26 = 6.52 kg. If the EBR-II bundle were to also fall over toward this double-batched Mark-42/TRR insert, the total loading of the combination would be 6.52 + 2.29 = 8.81 kg, the same value as for the above case. Therefore, this condition will also be critically safe for the same reason.

Even if there is a missing insert and a double batching at the same time, there will still not be a criticality. For example, if 12 TRR bundles (doubling batching) were loaded into a port with the insert missing, then they could fall toward an insert containing 6 TRR rods. If the EBR-II bundle also fell toward the same insert, then the EBR-II bundle could be assumed to be next to 18 TRR rods. If these 18 TRR rods are assumed to contain an average amount of plutonium, then the total equivalent U-235 loading in this configuration, based on the values in Table 3, would be (3 x

2.29) + 2.21 = 9.09 kg. Since all of this material is < 1.0 % enriched, the limiting value (from Section 6.2) would be 13.1 kg of U-235. Therefore, if the EBR-II bundle were surrounded by 18 average TRR rods, the configuration could not go critical, since 9.09 kg < 13.1 kg. If six of the TRR rods were assumed to contain the maximum plutonium possible (79 grams each), then the total equivalent U-235 loading, based on the values in Table 3, would be (2 x 2.29) + 3.26 + 2.21 = 10.0 kg. This is less than the 13.1 kg limit for the average enrichment of < 1.0 %, and equals the 10.0 kg limit (from Section 6.2) for the maximum enrichment of 1.01 %.

The above 10.0 and 13.1 limits assume a spherical configuration of optimally moderated and reflected fissile material, without neutron absorption in structural materials. KENO calculations (Ref. 1) which account for the annular geometry of the dissolver, as well as the neutron absorption in the steel walls of the dissolver, have obtained a maximum K = 0.94211 (Table 6, case 5 of Ref. 1) for 19 metric tons (19,000 kg) of uranium in the annular dissolver with an equivalent U-235 enrichment of 0.9635 % (Table 1 of Ref. 1). This calculation was conservative, since it did not restrict the fissile material to the inserts, ignored the neutron absorption in the steel material of the inserts, and ignored neutron absorption in the nitrogen in the nitric acid. The total uranium loading of all the EBR/TRR material in this dissolution is only 4223 kg (Table 1), with an average equivalent*U-235 enrichment of 0.78 %. Therefore, when the actual dissolver geometry and wall materials are accounted for, the following conclusions can be made:

1. Even if all of the EBR/TRR material (4224 kg) were put into the dissolver at once, it could not go critical. Therefore, double batching of any of the planned four batches is critically safe.

2. The neutronic isolation of the inserts does not have to be assumed to maintain the required criticality safety margin.

3. The TRR material does not have to be placed into inserts to maintain the required criticality safety margin.

6.3 Analysis With Uranium in Solution

Even if, as assumed in the above section, the dissolver solution initially contains no fissile material at the start of the batch, the uranium and plutonium in the fuel rods will be going into solution as the dissolution progresses. As they dissolve, the fuel rods will be decreasing in diameter, which will cause the U-238 absorption resonances to be less spatially self shielded, which will cause the K-eff to decrease. This decrease in K-eff does not depend on the dissolved uranium and plutonium being swept away from the fuel rods, for it will even occur for an infinite lattice as it dissolves into a homogeneous mixture. This is shown by Figure 6, which is taken from Figure 22 of Ref. 11, where the curve for the homogeneous water reflected case lies entirely to the right of the curve for the heterogeneous water reflected case. This is also implied by the results in Table 5 of Ref. 1. But in making this argument, two qualifications need to be made:

1. The monotonically decreasing K-eff that results from a heterogeneous fuel geometry dissolving into a homogeneous fuel geometry is only claimed for fairly low enrichments,

where the effect of spatial shelf shielding of the U-238 absorption resonances dominates over the shelf shielding of the low energy U-235 fission cross section. This effect is shown in Figure 6 between a U-235 enrichment of 0.7% and 5.0%

2. The heterogeneous fuel geometry is assumed to have an optimum fuel rod diameter, optimum spacing, and optimum water moderation and reflection, so that the heterogeneous K-eff starts as high as possible. This is true for the heterogeneous case in Figure 6.

It also needs to be recognized that uranium and plutonium that is dissolved will be swept away from the fuel rods, and will be diluted in the solution in the entire dissolver, i.e. the mixing rate will be much greater than the dissolution rate of the uranium and plutonium from the EBR-II and TRR fuel rods. This will result from the convective currents within the dissolver, as well as from sparging from the bottom of the dissolver. Therefore, the fuel rods from the EBR-II bundle and the TRR bundles will be critically safe during any stage of dissolution.

The criticality safety of TRR rods in the dissolver was shown in Ref. 1. All cases that were considered obtained $K \leq 0.95$, except for the case listed in Table 2 of Ref. 1, where it was assumed that the entire inner annulus was filled with TRR rods with optimized diameters and spacing. Since this condition was unrealistically conservative, Ref. 1 was accepted as proving the criticality safety of TRR dissolution. Of particular interest is that the K-eff decreased by 0.05 (Ref. 1, Table 2 vs. Table 4) when 400 grams of natural uranium containing 0.129 wt. % Pu-239 was dissolved per liter of solution in the dissolver. It would have decreased by even more if the neutron absorption in the nitrogen in the $UO_2(NO_3)_2$ in solution were also included in the calculation. The equivalent U-235 enrichment for this calculation (from Table 1 of Ref. 1) was $0.705 + (2.0 \times 0.129) = 0.9635$ %. Therefore, the effect on K-eff of ignoring old fissile material, with an equivalent enrichment < 0.9635 %, in the dissolver solution is conservative. Since, from Table 1, the equivalent U-235 enrichment for the EBR-II rods and the TRR rods that are being considered in this NCSE is only 0.77 % and 0.79 %, respectively, dissolved material from these rods will also cause the K-eff of the EBR/TRR rod lattice to decrease. This is the result of the low fissile enrichment of the EBR/TRR material that is being dissolved. Therefore, it is conservative to ignore the effect of the dissolved EBR/TRR material on the criticality safety of the system. This is true for EBR/TRR rods as they dissolve, and for a new batch of rods that are placed into old solution, provided that the enrichment of the old fissile material in solution is < 0.9635 %. Fissile enrichments > 0.9635 % in solution are considered in Section 5.

Another very significant result in Ref. l (Table 6, case 7) is the case for TRR rods in inserts in the dissolver, including the effect of absorption in the insert wall material. This case calculated K = 0.488, which shows the criticality safety margin that actually exists for TRR dissolution. Since, as shown in Table 1, the equivalent U-235 enrichments for the EBR-II rods and for the TRR rods are both lower than for previous TRR calculations in Ref. 1 (0.9635 % enriched), criticality safety will be assured for loading and dissolving the EBR/TRR bundles in the dissolver.

According to Table 2 of ANSI-8.1 (Ref. 6), an infinite homogeneous solution of uranium nitrate [$UO_2(NO_3)_2$] in water, without a fuel rod lattice, will be critically safe as long as the U-235 enrichment is $\leq$ 1.96 wt. %. The limiting U-235 enrichment is as high as this value (1.96 wt. %)

because of neutron absorption in the nitrogen in the nitrate compound. When the uranium and plutonium in the EBR-II and TRR rods are dissolved in nitric acid in the dissolver, they will be in solution as $UO_2(NO_3)_2$ and $Pu(NO_3)_4$ respectively. Since the equivalent U-235 enrichment of the EBR-II bundle is 0.77 %, and the equivalent U-235 enrichment of the TRR rod with the highest plutonium content (79 grams) is 1.01 °/e, fully dissolved uranium and plutonium from the EBR-II and TRR rods can not go critical (0.77 % and 1.01 % are both less than 1.96 %) regardless of the quantity that has been dissolved.

6.4 Conservative Assumptions

The above criticality safety analysis includes the following conservative assumptions:

1. Use of data from Ref. 6 and 7 effectively assumed a fuel rod lattice with optimum diameter fuel rods, optimum fuel rod spacing, and optimum water moderation and reflection.

2. Use of data from Ref. 6 and 7 effectively ignored neutron leakage from the fueled ports in the inner annulus of the dissolver. An infinite system is thus assumed.

3. The highest plutonium content (79 g) is assumed for many of the TRR rod configurations.

4. All plutonium is assumed to be Pu-239. This assumption ignores the absorption in Pu-240. Historically, the nominal composition of plutonium in TRR blanket rods received at SRS has been listed (Ref. 1) as 90.45 % Pu-239, 8.92 % Pu-240, and 0.64 % Pu-241, but the plutonium composition can vary significantly depending on the location and duration of the fuel irradiation.

5. Except for the use of data from Ref. 1, neutron absorption is ignored in the stainless steel used for the dissolver walls and inserts, and in concrete used in the dissolver and the floor.

6. Neutron absorption is ignored in the fission products, in aluminum components and aluminum dissolved in solution, and in nitrogen in the nitric acid.

7. A conservative value of 2.0 was used for the Plutonium Equivalency Factor, to calculate the effective U-235 enrichment.

The effect of these conservative assumptions should produce significantly greater criticality safety margins than discussed above.

6.5 Transportation

The TRR bundles will be shipped from RBOF to the F-canyon dissolver according to existing administrative controls. These procedures (Ref. 16) allow the shipping cask to hold up to 42 TRR rods. Based on data in Table 1, an average TRR rod would contain approximately

2758 / 54 = 51.1 kg of uranium. Since the EBR-II bundle contains 285 kg of uranium, and the enrichment is essentially the same as for the TRR rods, shipping the EBR-II bundle is equivalent to shipping 5.6 average TRR rods. So, from a criticality standpoint, the EBR-II bundle could be shipped along with up to 42 - 6 = 36 TRR rods in the shipping cask. However, since the EBR-II bundle has a larger diameter and length than the TRR bundle, the EBR-II bundle may not fit into the shipping cask together with TRR bundles, so that it may be necessary to ship the EBR-II bundle by itself. This would, of course, also be critically safe.

7.0 Design Features (Passive and Active) and Administratively Controlled Limits & Requirements

The design of the dissolver provides for the use of inserts in the four loading ports, which are separated by 90° around the top of the inner annulus. Previous studies have shown that this separation of the loading ports assures that fissile material loaded into any one insert will be neutronically separated from fissile material in the other inserts. Previous studies (Ref. 12 and 13) have also shown that the design of the dissolver and the dissolution process assures that the plutonium mixes with the uranium throughout the dissolution process. This means that the plutonium dissolves as the uranium dissolves, and that the mixing within the dissolver is sufficient to preclude regions of high concentration solution forming in the vicinity of undissolved material. These things occur due to convective mixing, as well as air sparging. Also, the annular design of the dissolver, and neutron absorption in the steel walls of the dissolver, significantly increase the fissile material limit in the dissolver, as shown in Ref. 1.

Based on the contingency considerations in Section 5, criticality safety controls (CSCs) are required to assure a critically "safe" dissolution process. To assure the non-occurrence of initiating events IE #1 (other than intended EBR/TRR material being loaded into the dissolver) and IE #2 (insufficiently acidic dissolver solution), the administrative controls for this EBR/TRR dissolution must be based on the administrative controls for previous TRR dissolution. This will satisfy the CSCs arising from the DCA for TRR dissolution (Ref. 2). To assure the non-occurrence of E #3 (significant fissile in initial dissolver solution), the following CSCs are required:

IE#	CSC#	DESCRIPTION OF CONTROL REOUIREMENT
3	3A	The process operator must determine whether the enrichment of the previous dissolution campaign > 0.9635 %. If it is, then CSC 3B to 3D must be performed. If it is not, then CSC 3B to 3D can be omitted. CSC 3E must be done in either case.
3	3B	During the last rinse prior to EBR/TRR material loading, dual independent sampling and chemical analysis (blue label) of the fissile concentration must be performed.

3	3C	Both chemical analysis values must be $\leq$ 0.1 gram fissile/liter, before permitting the loading of EBR/TRR material into the dissolver.
3	3D	The last rinse must be transferred out, before permitting the loading of the EBR/TRR dissolution materials into the dissolver.
3	3E	The performance of the above items (3A, and 3B to 3D if necessary) by the process operator must be independently verified by a second individual, such as a supervisor.

8.0 Summary and Conclusions

A nuclear criticality safety evaluation (NCSE), including a double contingency analysis (DCA), was conducted regarding the dissolution of EBR-II bundle DU006, and 81 TRR bundles, in the Feanyon annular dissolver in four dissolver batches. An NCSE and DCA (Ref. 1 and 2) were previously performed for dissolution of TRR rods in the F-canyon dissolver. The difference in this NCSE, relative to Ref. 1 and 2, is that one EBR-II bundle (DU006) is also to be dissolved, and that it will be placed by itself into a dissolver port without using an insert.

The EBR-II fuel in bundle DU006 is comparable to the fuel in the TRR bundles (Tables 1 to 4), so that including the EBR-II bundle in the dissolution is a relatively small perturbation on what has been dissolved in the past. For example, the average equivalent U-235 enrichment (including a multiplier of 2.0 for plutonium) of the EBR-II bundle is 0.77 %, whereas the average equivalent U-235 enrichment of the TRR bundles is 0.79 %. The equivalent U-235 loading of the EBR-II.bundle, which would be loaded by itself into a port, is 2.21 kg, whereas the average equivalent U-235 loading of six TRR bundles, which would be loaded into a Mark-42/TRR insert, is 2.29 kg.

Based on safe limits (13.1 kg U-235 at 1.0 % enrichment, and 10.0 kg U-235 at 1.0 % enrichment) the various allowed configurations in the EBR/TRR dissolution were shown to be critically safe: the EBR-II bundle DU006 placed in an isolated port (2.21 kg U-235 < 13.1 Kg U-235 safe limit), six maximum plutonium TRR bundles in an isolated Mark-42/TRR insert (3.26 kg U-235 < 10 kg U-235 safe limit), and the EBR-II bundle DU006 in close proximity to six maximum plutonium TRR bundles (5.47 kg U-235 < 13.1 kg U-235 safe limit). Allowing for a double vidation of administrative controls, the most severe credible configuration would result from grouping 18 TRR rods together with the EBR-II rods. This could result from double batching and a missing Mark-42/TRR insert. This configuration was also found to be critically safe (9.09 kg U-235 < 13.1 kg U-235 safe limit).

The DCA initiating events that result from the presence of the TRR bundles in this EBR/TRR dissolution were dealt with in Ref. 2. The administrative controls used for the previous TRR dissolution should therefore be adopted for this EBR/TRR dissolution, with changes to handle the EBR-II bundle, of course. The only new initiating event (IE #3) is the possibility of significant fissile material being in the dissolver solution before the EBR/TRR materials are loaded into the dissolver. A criticality safety limit (0.1 g fissile/liter) and criticality safety controls (CSCs) were developed to preclude E #3 from occurring. With these CSCs, the EBR-II bundle DU006 can be dissolved with the 81 TRR bundles in four batches in the F-canyon dissolver in a critically safe manner.

9.0 References

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17. DPSTS-221-FC-200, Rev. 2, Revised 7/94, from WSRC-TN-45, “221-F Building Technical Standards”

Figure 1. Annular Dissolver Tank 6.1 or 6.4 (Drawing S5-2-136710)

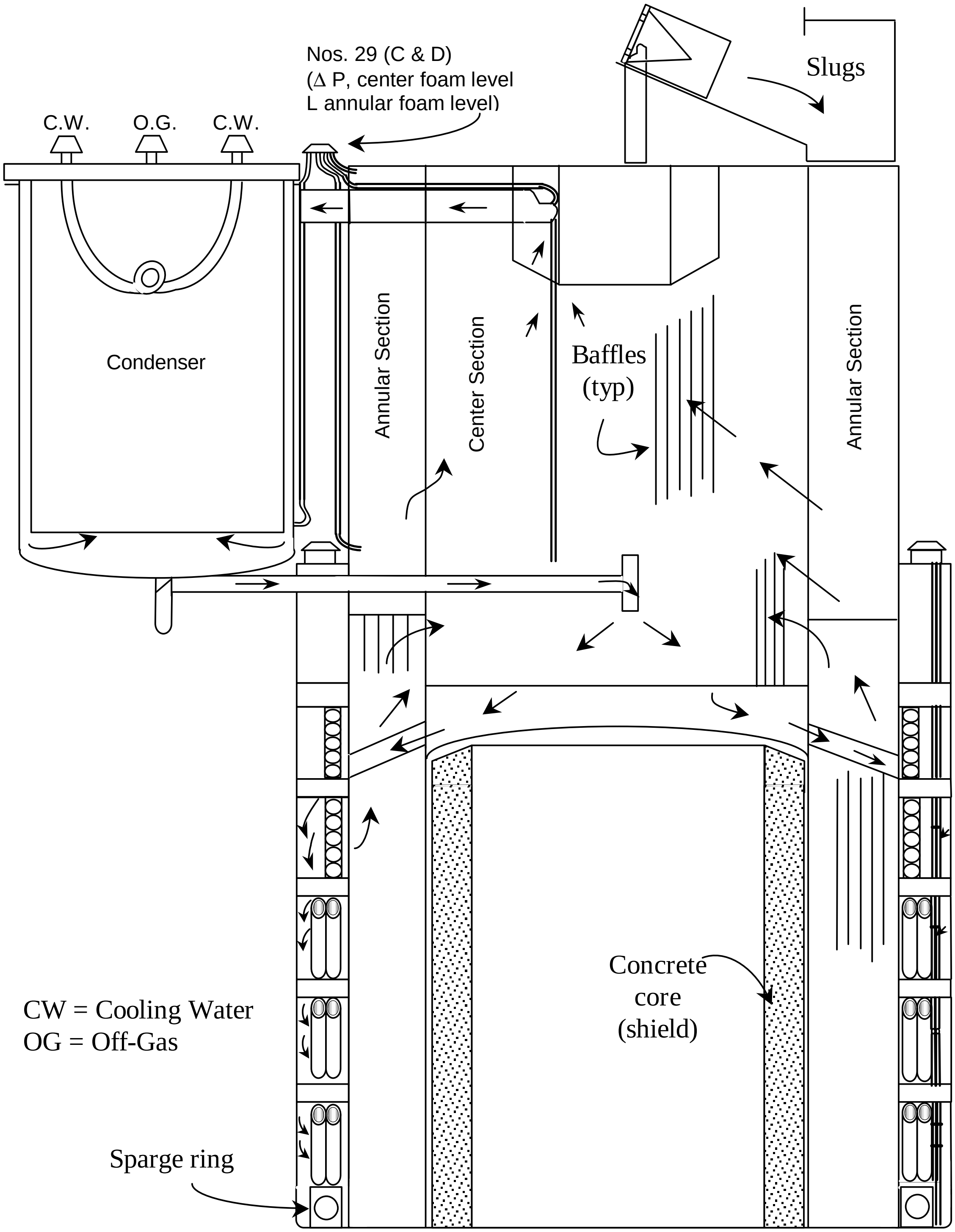


Figure 2. Top View of Annular Dissolver With Port Spacing

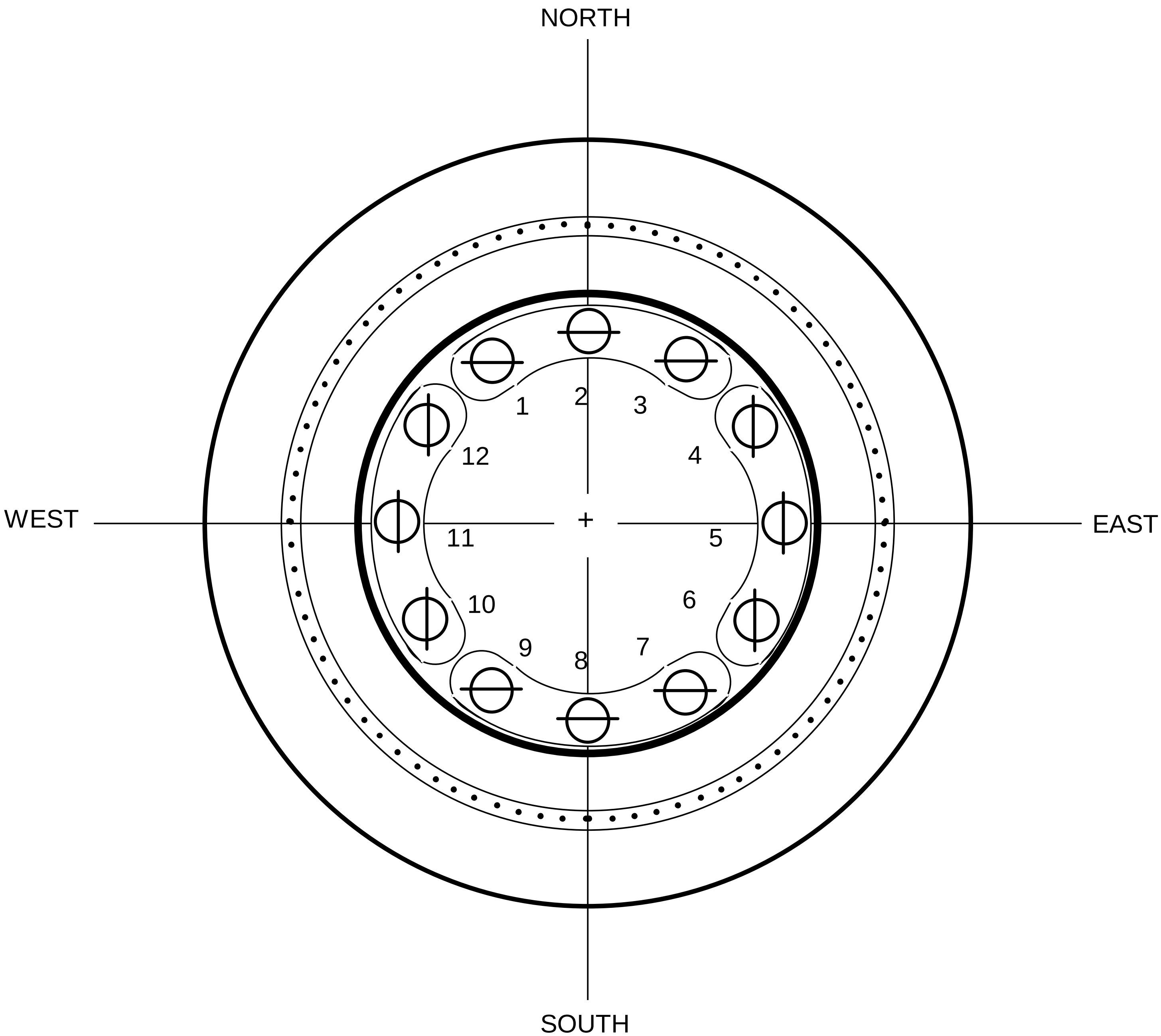


Figure 3. Mark-42 Insert With "TRR" Insert (6 Well Insert)

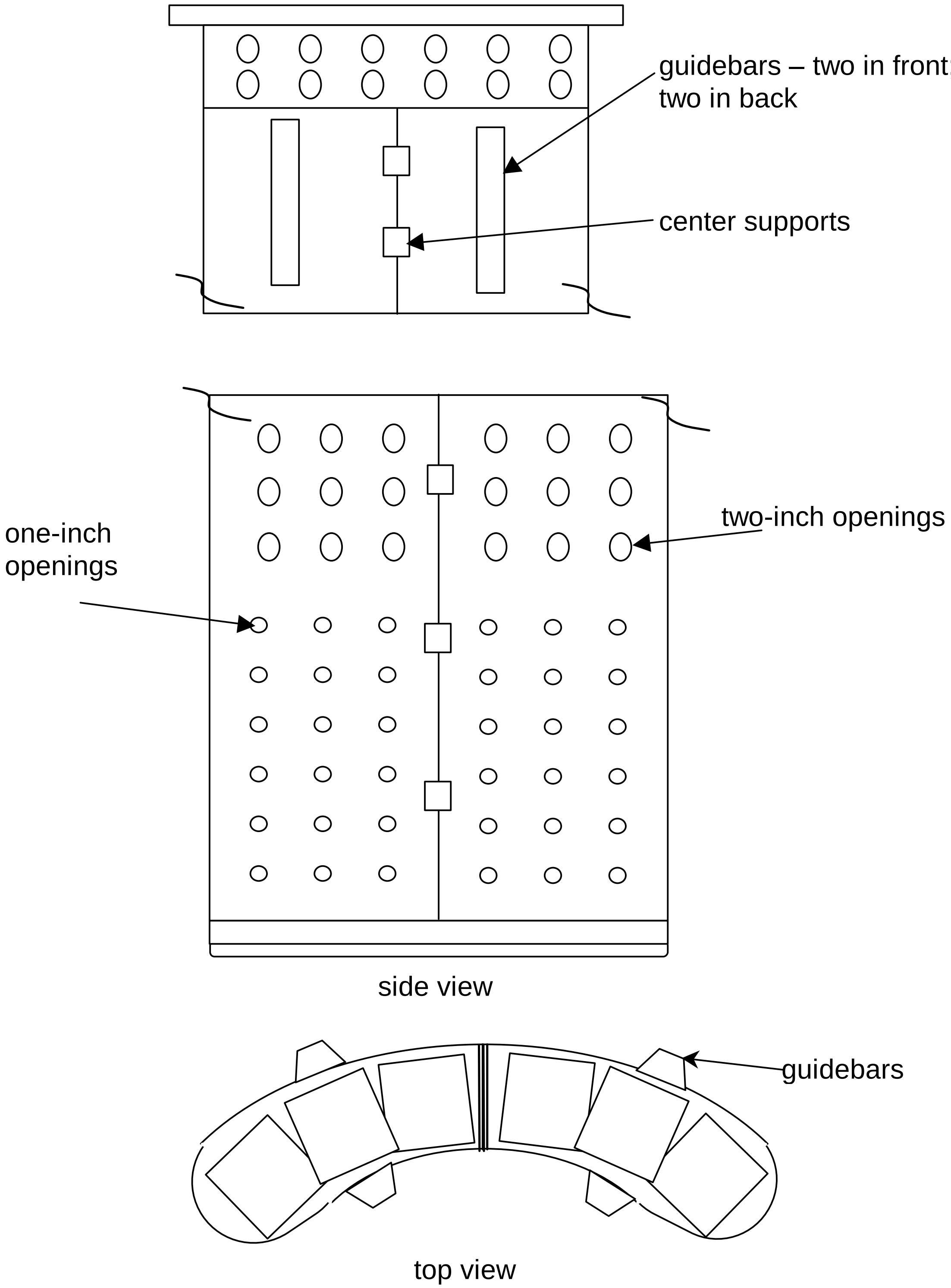


Figure 4. Three Well Insert

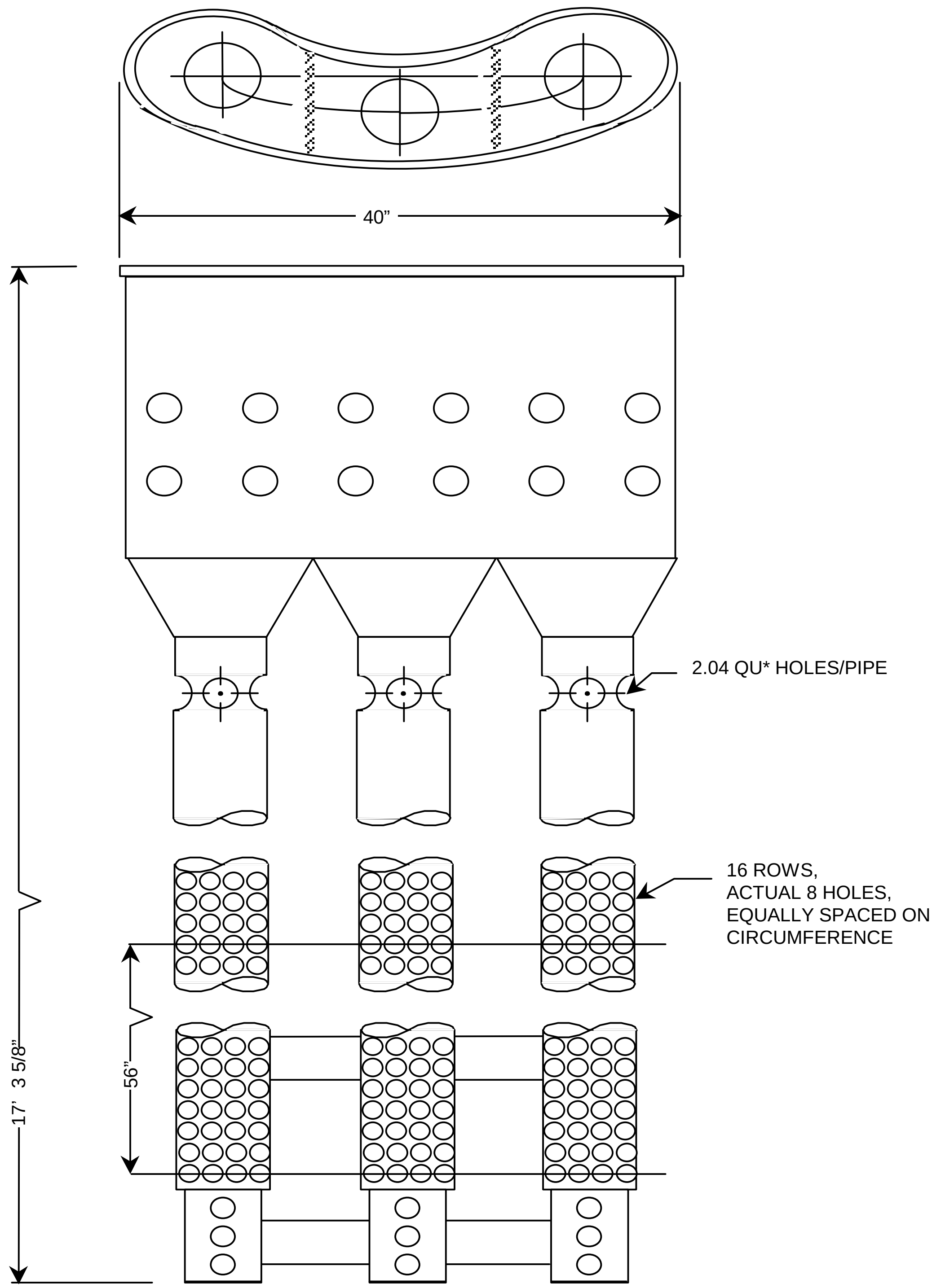


Figure 5. Critically Safe Concentrations for Lattices of Uranium Metal in Water

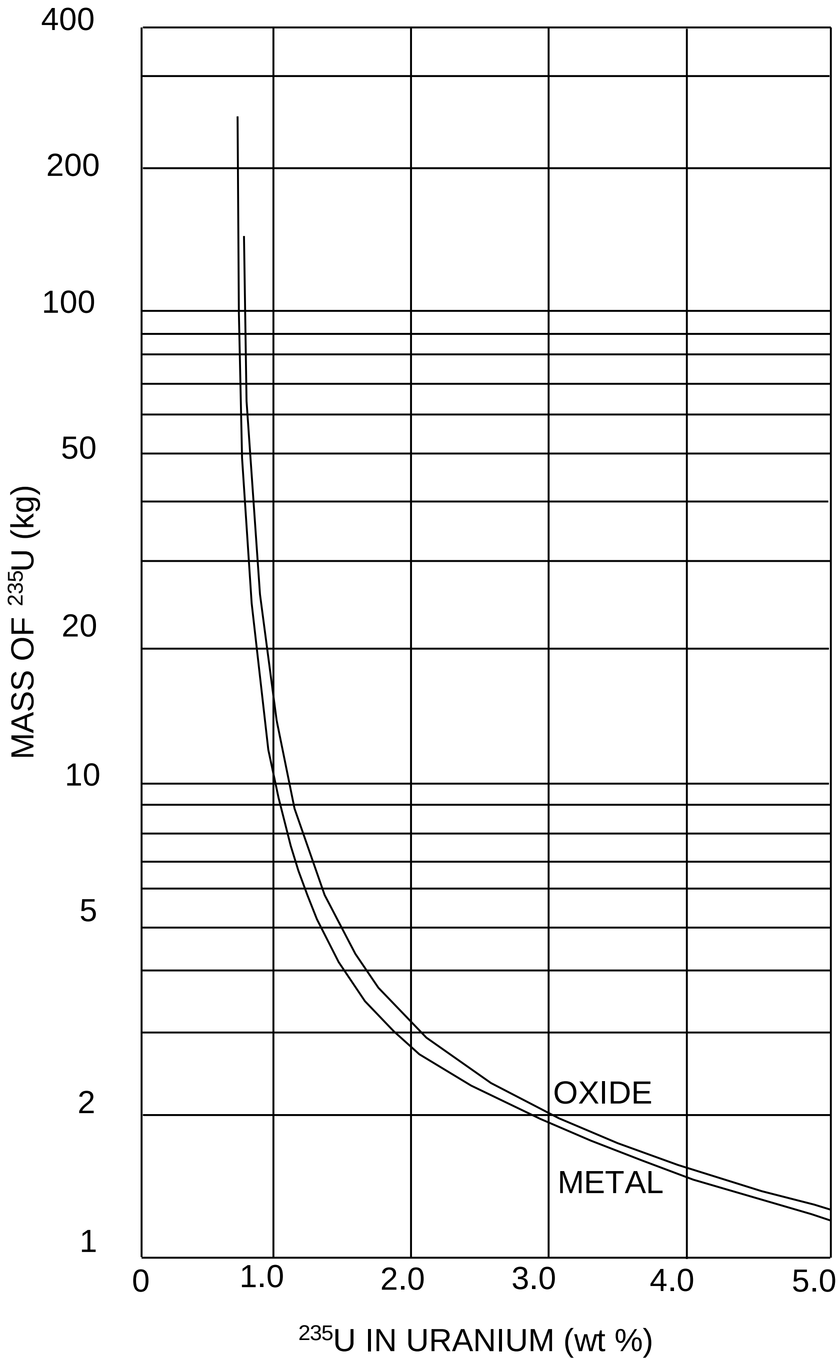
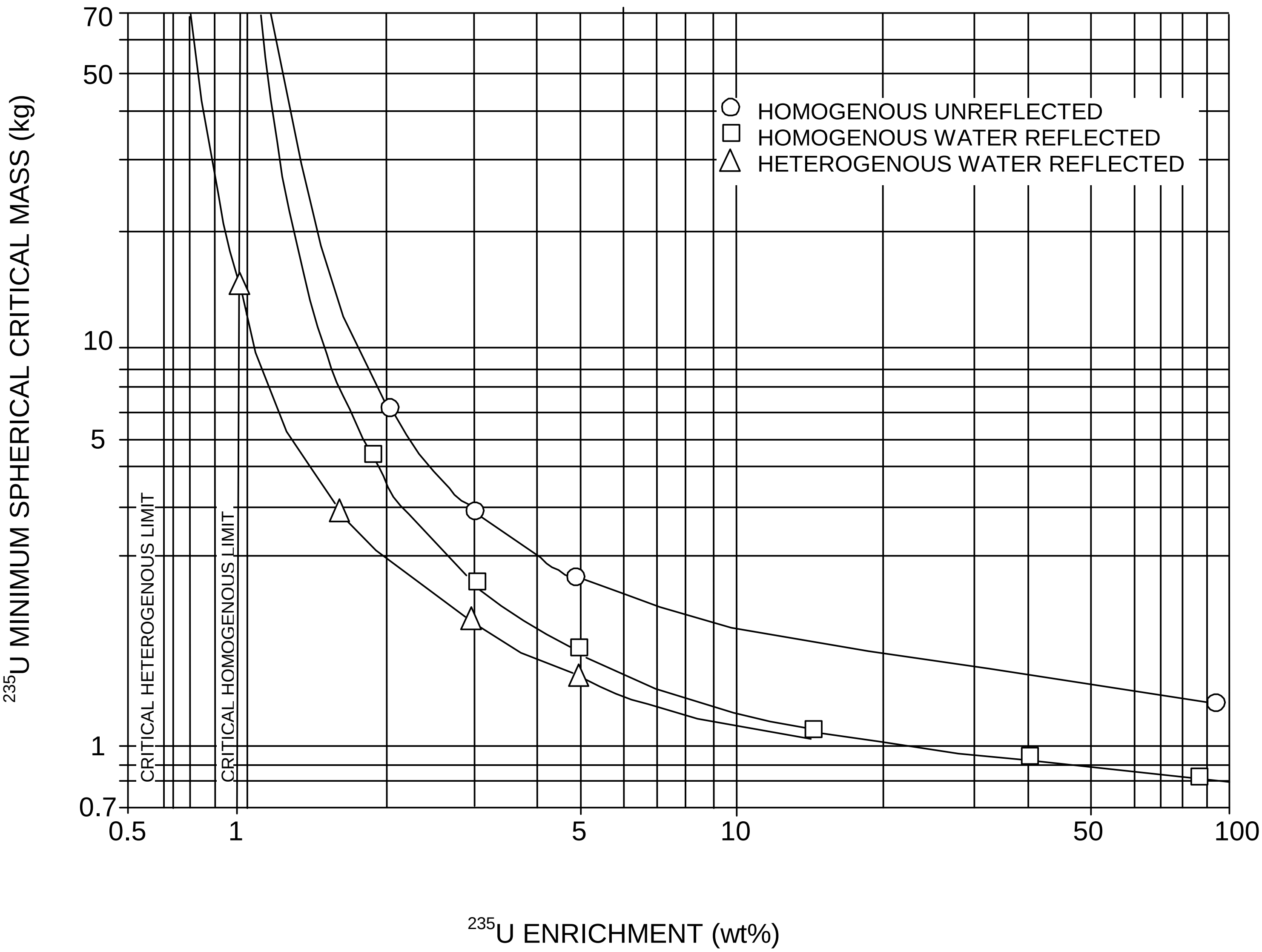


Figure 6. Critical Mass of U-235 for a Sphere



CONCLUDING MATERIAL

Review Activity:		**Preparing Activity:**
<u>DOE</u>	<u>Field Offices</u>	DOE EH-34
DP	AAO	
EH	AL	
EM	CH	
ER	ID	**Project Number:**
NE	OAK	SAFT-0011
	OR	
	RF	
	RL	
	SR	
	Y-12	
	<u>National Laboratories</u>	
	ANL	
	INEL	
	LANL	
	LLNL	
	ORNL	
	PNL	
	SNLA	