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**DOE-STD-3007-93
November 1993**

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**

AREA SAFT

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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 Order 5480.24 that pertains 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."	

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.

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, 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 5480.24, 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		Barriers
No.	Description	
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 allows 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).

Control Parameter	Controls	
	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 FEATURE'S (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.

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

- 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.

Control Parameter	Controls ^a	
	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"
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" (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.	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

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

Operation

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 data¹, 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-7016². 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 overflow 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

Overflow 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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**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.

Note: CPP-764 (location of VES-SFE-126) underground and SE of CPP-642.

Figure 1. CPP 603 Fuel Storage Facility Layout.

Figure 2. North/Middle Basin Concrete Dividers.

Figure 3. Example of Monorail Hanger - 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 modelled 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 modelled 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 $k_{\text{eff}} \pm 1\sigma = 1.0032 \pm 0.0034$. This array was modelled 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 modelling 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 modelled for the purposes of validation. The first of the arrays modelled 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 $k_{\text{eff}} \pm 1\sigma = 1.0016 \pm 0.0033$.

The second of the arrays modelled 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 $k_{eff} \pm 1\sigma = 1.0055 \pm 0.0031$.

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 modelled for the purposes of validation and applicability were similar to the modelled 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 modelled 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 modelled in this evaluation.

Figure 4. BORAX V Superheater Fuel Assembly.

Figure 5. 4 x 4 Array of BORAX V Superheater Fuel Assemblies³.

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 2¹, 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 modelled 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 $3 \times 3 \times 1$ array of fuel assemblies exceeded the $k_{eff} + 2\sigma > 0.95$ safety limit criteria and a $3 \times 4 \times 1$ array is shown to be critical with $k_{eff} + 2\sigma > 1.0$.

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 concrete¹² channel walls and basin floors were modelled, various arrangements of fuel assemblies were evaluated. These arrangements are shown in Figure 9. As shown in Figure 9, a $2 \times \text{infinite} \times 1$ array of fuel elements placed in a water filled concrete storage channel gives a $k_{eff} + 2\sigma = 1.0386$. 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 $k_{\text{eff}} + 2\sigma = 0.9340$, which is within the ICCF 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 modelled. 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.

<u>Distance Between Fuel Assemblies</u>	<u>$k_{eff} + 2\sigma$</u>
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

<u>Nuclide</u>	<u>Atom Density</u>	<u>Nuclide I.D.</u>
U-235	5.8990 x 10-3	92235
U-238	4.3840 x 10-4	92238
O	1.2967 x 10-2	8016
Cr	1.2918 x 10-3	24304
Mn	4.4162 x 10-2	25055
Fe	5.7442 x 10-3	26304
Ni	1.2670 x 10-2	28304

UO2 - Stainless Steel Fuel Matrix - Half Peripheral Fuel Plates

<u>Nuclide</u>	<u>Atom Density</u>	<u>Nuclide I.D.</u>
U-235	3.1430 x 10-3	92235
U-238	2.3360 x 10-4	92238
O	1.5036 x 10-2	8016
Cr	1.4979 x 10-3	24304
Mn	5.1208 x 10-2	25055
Fe	6.6607 x 10-3	26304
Ni	6.7530 x 10-2	28304

304 Stainless Steel (Density = 7.92 g/cc)

<u>Nuclide</u>	<u>Atom Density</u>	<u>Nuclide I.D.</u>
Cr	1.7428 x 10-2	24304
Mn	1.7363 x 10-3	25055
Fe	5.9358 x 10-2	26304
Ni	7.7207 x 10-3	28304

Water (Density = 0.9982 g/cc)

<u>Nuclide</u>	<u>Atom Density</u>	<u>Nuclide I.D.</u>
H	6.6751 x 10-2	1001
O	3.3375 x 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 Pre
=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
  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

```



```

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

```

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

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-R8¹ 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-MS², 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-MS².

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 procedures⁴ 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.

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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**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 5480.5A¹ and 10 CFR Part 71², 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-239³. 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 Pu-239 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-2⁴, 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, ANISN⁵. Sixteen-energy-group Hansen-Roach⁶ 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 DOE¹, "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 optimumly 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 optimumly 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-II⁴ 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. DOE-ID 5480.5A
 - a. Double Contingency Principle (see Section 6.0)
 - b. Mass Control (operating limit $\leq 45\%$ of critical mass)
2. 10 CFR Part 71 Sections 71.55 and 71.61
 - a. Optimum moderation, reflection, and geometry was analyzed (71.55).
 - b. Twice the number of allowed packages (undamaged or damaged) is subcritical when optimumly 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)

APPENDIX A
Materials and Compositions

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	4.39875E-05
		94913	6.22368E-05
Pu-239, polyethylene, & water fuel material with H/Pu = 800	Pu-239	94912	1.74460E-05
		94913	7.55002E-05
Pu-239, polyethylene, & water fuel material with H/Pu = 900	Pu-239	94913	8.10202E-05
		94914	1.59868E-06
Pu-239, polyethylene, & water fuel material with H/Pu = 1000	Pu-239	94913	6.45010E-05
		94914	9.85602E-06
Pu-239, polyethylene, & water fuel material with H/Pu = 1100	Pu-239	94913	5.09852E-05
		94914	1.66120E-05
Polyethylene-water reflector H = 7.43570-2 C = 2.38296-2 O = 1.33488-2	CH ₂	402	(0.60)
	H ₂ O	502	(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.

APPENDIX B

**Typical ANISN/PC
Input Listings**

2 of 2

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
	+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

Listing 1. Continued.

Iteration	Iteration	Iteration	Iteration
1	1	1	1
2	2	2	2
3	3	3	3
4	4	4	4
5	5	5	5
6	6	6	6
7	7	7	7
8	8	8	8
9	9	9	9
10	10	10	10
11	11	11	11
12	12	12	12
13	13	13	13
14	14	14	14
15	15	15	15
16	16	16	16
17	17	17	17
18	18	18	18
19	19	19	19
20	20	20	20
21	21	21	21
22	22	22	22
23	23	23	23
24	24	24	24
25	25	25	25
26	26	26	26
27	27	27	27
28	28	28	28
29	29	29	29
30	30	30	30
31	31	31	31
32	32	32	32
33	33	33	33
34	34	34	34
35	35	35	35
36	36	36	36
37	37	37	37
38	38	38	38
39	39	39	39
40	40	40	40
41	41	41	41
42	42	42	42
43	43	43	43
44	44	44	44
45	45	45	45
46	46	46	46
47	47	47	47
48	48	48	48
49	49	49	49
50	50	50	50
51	51	51	51
52	52	52	52
53	53	53	53
54	54	54	54
55	55	55	55
56	56	56	56
57	57	57	57
58	58	58	58
59	59	59	59
60	60	60	60
61	61	61	61
62	62	62	62
63	63	63	63
64	64	64	64
65	65	65	65
66	66	66	66
67	67	67	67
68	68	68	68
69	69	69	69
70	70	70	70
71	71	71	71
72	72	72	72
73	73	73	73
74	74	74	74
75	75	75	75
76	76	76	76
77	77	77	77
78	78	78	78
79	79	79	79
80	80	80	80
81	81	81	81
82	82	82	82
83	83	83	83
84	84	84	84
85	85	85	85
86	86	86	86
87	87	87	87
88	88	88	88
89	89	89	89
90	90	90	90
91	91	91	91
92	92	92	92
93	93	93	93
94	94	94	94
95	95	95	95
96	96	96	96
97	97	97	97
98	98	98	98
99	99	99	99
100	100	100	100

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
	+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

5**
6**
7**
8\$\$\$

CONCLUDING MATERIAL

Preparing Activity:

DOE-DP-62

Review Activity:

Project Number:

DOE

Field Offices

SAFT-0011

DP

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