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SRT-CMA-940005

SRTC CRITICALITY SAFETY TECHNICAL REVIEW: PHASE I CRITICALITY ANALYSIS FOR THE 9972-9975 FAMILY OF SHIPPING CASKS (U); (SRT-CMA-940003)

by

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SAVANNAH RIVER TECHNOLOGY CENTER
Applied Technology Section
Applied Physics Group

SRT-CMA-940005

SRTC CRITICALITY SAFETY TECHNICAL REVIEW:

PHASE I CRITICALITY ANALYSIS FOR THE 9972-9975 FAMILY OF SHIPPING CASKS (U); (SRT-CMA-940003)

March 2, 1994

Reviewer: R. W. Rathbun
Roy Rathbun, ext. 58232
Applied Physics Group/Criticality Analysis

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Reviewing Official: C E Apperson Jr. — C. E. Apperson, Manager, APG
Date: 3-3-94

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SUMMARY

Review of SRT-CMA-940003, "Phase I Criticality Analysis For The 9972-9975 Family Of Shipping Casks (U); (SRT-CMA-940003)," January 22,1994, has been performed by the SRTC Applied Physics Group. The NCSE is a criticality assessment of the 9972-9975 family of shipping casks. This work is a follow-on of a previous criticality safety evaluation, with the differences between this and the previous evaluation are that now wall tolerances are modeled and more sophisticated analytical methods are applied.

The NCSE under review concludes that, with one exception, the previously specified plutonium and uranium mass limits for 9972-9975 family of shipping casks do ensure that WSRC Nuclear Criticality Safety Manual requirements (ref. 1) are satisfied. The one exception is that the plutonium mass limit for the 9974 cask had to be reduced from 4.4 to 4.3 kg. In contrast, the 7.5 kg uranium mass limit for the 9974 cask was raised to 14.5 kg, making the uranium mass limit identical for all casks in this family.

After a thorough review of the NCSE, this reviewer agrees with all conclusions stated therein.

SCOPE OF TECHNICAL REVIEW

This technical review consisted of:

- an independent check of the methods and models employed,
- application of ANSI/ANS 8.1 and 8.15,
- verification of WSRC Nuclear Criticality Safety Manual(1) procedures.

DOCUMENTATION

Issuance of this memorandum transmits this technical review as critical data.

METHOD AND MODEL REVIEW

Method:

Cross-Sections

Hansen-Roach 16-Group Cross-Section Data Library

This library is an extensively utilized database used for criticality safety analysis that is installed and verified on the SRS IBM mainframe computer.

SCALE Library

This library is an extensively utilized database for criticality safety analysis developed at ORNL and is installed and verified on the SRS Cray mainframe computer.

Computer Codes

Three sets of computing codes were utilized for the NCSE under review, and are listed here;

HRXN/TWOTRAN

Cross-section preparation and processing for this set of tools was performed with the Joshua 70 version of HRXN. The 16-group Hansen-Roach library is processed. This is a validated code developed at SRS.

TWOTRAN is a 2-dimensional, neutron transport theory criticality code. The validated J70 version was employed.

DORT

This is a neutron transport theory criticality code similar to TWOTRAN. This code developed at ORNL and is installed and verified on the SRS Cray mainframe computer. The SCALE cross section data is employed with DORT.

HRXN/JSWL/KENO-V.a

Cross-section preparation and processing was performed with the Joshua 70 versions of HRXN and JSWL. These are validated codes developed at SRS. The system keff is predicted with KENO-V.a, an SRS validated Monte Carlo criticality calculational code developed at Oak Ridge National Laboratory as part of the SCALE package.

These codes are widely used throughout the industry for performing criticality safety calculations.

Model:

The materials and dimensions used in all calculations were provided to this reviewer in microfiche form. The drawings upon which all this data was generated were also provided. These sources were thoroughly checked and were seen to be consistent, with only one exception. This exception was corrected, re-calculated and shown to have an insignificant impact on the originally calculated keff.

EVALUATION

Data to perform this evaluation were derived from the NCSE under review, from references listed in the NCSE, and from private communications with the author.

ꝏꝏꝏ **NCSE CONTENT EVALUATION** ꝏꝏꝏ

Bias Applied, Subcritical Margin and K-Safe:

For uranium limit calculations;

Standard Bias: The bias applied is 0.03, the maximum for uranium systems predicted with the HRXN/ANISN criticality method, which are deemed to apply to HRXN/TWOTRAN and HRXN/KENO-V.a calculations since these code package methodologies are nearly equivalent.

Non-uniformity Bias: The NCSE assumes full-flooding and unguaranteed fissile material distribution. Since the calculations are performed for uniform distribution, additional margin is added to account for non-uniform distribution of materials that can potentially yield higher keff's. The bias applied is 0.02, deemed to be conservative. This margin is applied in lieu of performing actual non-uniform calculations. For the solid metal cases, this additional margin does not apply, since fissile materials are at their maximum density throughout the unit in question.

Subcritical Margin: The subcritical margin imposed for both normal and accident conditions is 0.05 Δk, which is a commonly used value for criticality evaluations.

K-safe: The K-safe used for the NCSE is 1.000-0.030(standard bias)-0.020(non-uniform bias)-0.050(margin) = 0.900 for solutions and 0.920 for the solid metal cases. Thus, the k-eff predicted for acceptable configurations will be less than or equal to this value.

For plutonium limit calculations;

Standard Bias: The bias applied is 0.01, the maximum for plutonium systems predicted with the HRXN/ANISN criticality method, which are deemed to apply to HRXN/TWOTRAN and HRXN/KENO-V.a calculations since these code package methodologies are nearly equivalent.

Non-uniformity Bias: Same as uranium, see above.

Subcritical Margin: Same as uranium, see above.

K-safe: The K-safe used for the NCSE is 1.000-0.010(standard bias)-0.020(non-uniform bias)-0.050(margin) = 0.920 for solutions and 0.940 for the solid metal cases. Thus, the k-eff predicted for acceptable configurations will be less than or equal to this value.

It should be noted that for the KENO calculations, 3 standard deviations (σ) are always added to the nominal keff.

Application of ANSI/ANS Standards:

The NCSE uses equivalencies for isotopes other than the dominant ones, U-235 and Pu-239.

The substitutions used are:

U-235 in place of U-234, U-232 or U-236;
U-238 not included;
Pu-239 is substituted for Pu-240 and Pu-241, provided that Pu-240 is greater than Pu-241.

These substitutions are in accord with ANSI/ANS Standards 8.1 and 8.15.

Review of NCSE Conclusions:

The NCSE under review concludes that, with one exception, the previously specified plutonium and uranium mass limits for 9972-9975 family of shipping casks do ensure that WSRC Nuclear Criticality Safety Manual requirements (ref. 1) are satisfied. The one exception is that the plutonium mass limit for the 9974 cask had to be reduced from 4.4 to 4.3 kg. In contrast, the 7.5 kg uranium limit mass limit for the 9974 cask was raised to 14.5 kg, making the uranium mass limit identical for all casks in this family. The following is an assessment of how that conclusion was reached.

<u>Normal conditions:</u>

All calculations were performed under accident conditions.

<u>Accident conditions:</u>

All calculations were performed under fully flooded conditions, i. e., water inside the casks and at least a one foot water reflector in all directions. It is assumed that there are no controls to prevent flooding, and hence, flooding could happen at any time the cask is loaded and in use. Additionally, a full range of cases were executed to ensure that optimally moderated configurations were covered.

Double contingency is addressed by stating that two independent administration checks must be placed on the fissile mass limit before a cask is loaded. This assumes that the uncertainty in scale measurement is taken into account. That being the case, it is fully demonstrated that while the fissile mass limit is adhered to, predictions for fully flooded casks do not exceed K-safe. This reviewer agrees with that conclusion.

∞∞∞ INDEPENDENT CALCULATIONS ∞∞∞

Due to the extensive treatment by the NCSE author and the conservatism built into the accident scenarios, it was not felt that independent calculations were necessary.

SAFETY MANUAL FORMAT AND PROCEDURES

The WSRC Nuclear Criticality Safety Manual (ref. 1) describes certain requirements that are to be included in a specifically formatted NCSE. This section reviews the compliance with that document.

SECTION	REMARKS
1.0 Introduction:	Included with appropriate contents
2.0 Description:	Included with appropriate contents
3.0 Requirements Documentation	Included with appropriate contents
4.0 Methodology	Included with appropriate contents
5.0 Discussion of Contingencies	Included with appropriate contents
6.0 Evaluation of Results	Included with appropriate contents
7.0 Administratively Controlled Limits and Requirements	Included with appropriate contents
8.0 Summary and Conclusions	Included with appropriate contents
9.0 References	Included with appropriate contents

REFERENCES

1. WSRC Nuclear Criticality Safety Manual (U), WSRC-IM-93-13, Rev.1, 7/1/93.

WESTINGHOUSE SAVANNAH RIVER COMPANY
SAFETY TECHNOLOGY DEPARTMENT

INTER-OFFICE MEMORANDUM

SRT-CMA-940003

Phase I Criticality Analysis for the 9972-9975 Family of Shipping Casks

Date: January 22, 1994

Author:

R.L. Frost

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To: R.S. Maurer

From: R.L. Frost

1. INTRODUCTION AND SUMMARY

A criticality analysis of the 9972-9975 shipping casks has been performed that accounts for tolerances on pipe diameters and wall thicknesses as allowed in the ASTM standards. The previous criticality analysis for these casks was performed using a maximum inner diameter for the Primary Containment Vessel (PCV) of 5.10 inches; an evaluation of the ASTM standards indicates a maximum inner diameter of 5.174 inches is appropriate. The goal of the new analysis was to confirm that the mass of fissile material permitted by the previous work (Ref. 1) does indeed meet criticality safety requirements when the larger inner diameter of the PCV is taken into consideration. This required a re-analysis of the single flooded unit scenario. With one exception, it was found that the previous mass limits do indeed ensure criticality safety limits are not exceeded for the 9972-9975 series of shipping casks. The exception was for the 9974 cask with plutonium. The plutonium mass limit for this cask had to be reduced to 4.3 kg (from 4.4 kg) to ensure criticality safety. In contrast, the uranium mass limit for this same cask (9974) has been increased from 7.5 kg to 14.5 kg. The previous uranium limit was based on a physically unrealizable geometry.

The previous analysis of the damaged array scenario remains valid. The revised SARP will reference the current work for the single flooded unit scenario and Ref. 1 for the damaged array scenario. Note that both the current work and the previous analysis apply to dry uranium compounds that are predominantly U-235 and dry plutonium compounds that are predominantly Pu-239. In particular, materials containing more than one fissile element are not covered by the current analysis; they will, however, be considered in phase II of this task. This work was completed under the guidance of Task 93-006-H-W-1.

This document is not intended to be a new SARP Chapter, and was not written to NRC Regulatory Guide 7.9 requirements. This memorandum was written to transfer intermediate results. A new SARP chapter will be written after completion of phase II of this work. That document will conform to the NRC requirements, and will include the material in this memorandum as well as the phase II results.

2. DESCRIPTION

Shipping Cask Models

The objective of this work was to show that the criticality safety limits discussed above could not be exceeded with the currently allowed fissile masses in the 9972-9975 series of shipping casks. The fissile mass limits derived from the original analysis are given in Table 1. The first step in the current work was to prepare computer models for each of the shipping casks. The J70 version of TWOTRAN (Ref. 2) was used to calculate k-eff for each configuration considered in this work. The validation of TWOTRAN for these calculations is based on that for its 1-d counterpart, ANISN, and is reported in Refs. 3 and 4. Hansen-Roach cross sections were processed by HRXN (Ref. 5) and used by TWOTRAN. Figures 1-8 show the TWOTRAN models for the 9972-9975 shipping casks used in this analysis. For each cask, a scale drawing of the model is given, to orient the reader, and then a detailed TWOTRAN model is shown. The following points should be noted:

1. The Primary Containment Vessel (PCV) is made from 5 inch schedule 40 pipe. In order to maximize its volume (which maximizes the volume of fissile material), the maximum inner diameter permissible by ASTM standards (Ref. 6) was used in the TWOTRAN model. This maximum is found by applying an allowed tolerance of +1/16 inch onto the nominal outer diameter, and then decreasing the nominal wall thickness by the allowable 12.5%. Note this procedure also minimizes the PCV wall thickness, which reduces neutron loss in the stainless steel (a further conservatism).

2. The Secondary Containment Vessel (SCV) is constructed of 6 inch schedule 40 pipe. Also, small lengths of 4 inch and 5 inch schedule 40 pipe are welded to the bottom of the PCV and SCV, respectively, to mate with the anti-rotation plate. Since none of these components restricts fissile volume, nominal dimensions were used.

3. The celotex insulation was assumed capable of absorbing enough water to become equal to water as a reflector. Thus, the celotex region is modeled as water.

4. The drum wall is not included in the models. This omission is made because their is such a large distance of water reflector between the PCV and the drum wall that its inclusion or omission will make no difference in the results. Since the only difference between the 9972-9975 series of shipping casks and the 9965-9968 series is in the drum wall material, omission of this detail also means that the results of this study are applicable to both series of shipping casks.

5. The modeling of the PCV and SCV was simplified to conform to the cylindrical geometry limitations of TWOTRAN. Where approximations would have a noticeable effect on the results, a conservative approach was taken. For example, the bottom of the PCV is a concave cap. Modeling it as flat (making the PCV a perfect right cylinder) increases the fissile volume and thus k-eff.

6. The bottom of the PCV contains an aluminum honeycomb spacer. This spacer was conservatively ignored (this increases the volume available for fissile solution), although the honeycomb spacers in the SCV were modeled.

7. No credit is taken for any type of packaging material inside the PCV.

8. The materials used in the analysis are listed in Table 2. The densities of natural uranium, Pu-239, and aluminum honeycomb are the same as those used in the original work. HRXN was set up to adjust the natural uranium density to that of pure U-235. HRXN also calculated the isotopics for the uranium/water and plutonium/water homogeneous mixtures, and then produced the appropriate cross sections.

3. REQUIREMENTS DOCUMENTATION

The current work, along with that of the current SARP (ref. 1), meet the requirements of 10-CFR-71, NRC Reg. Guide 7.9, and DOE Order 5480.3.

4. METHODOLOGY

Criticality Safety Limits

The analysis considers U-235 systems with concentrations ranging from pure metal to 1 g U/L. Criticality safety limits for U-235 systems using HRXN-ANISN were reported in Ref. 3. The current calculations used HRXN-TWOTRAN; TWOTRAN is a 2-d counterpart to the 1-d ANISN, and it is assumed the HRXN-ANISN biases apply to HRXN-TWOTRAN. Ref. 3 provides a graph from which one can determine the required bias versus H/U-235 ratio for HRXN-ANISN calculations. For the current work, it is necessary to take the largest bias obtained over the H/U-235 range. In the previous work (Ref. 1), this value was reported to be 0.025. However, that value corresponds to an "eyeball fit" line through the data; considering the actual data points, a value of 0.03 is appropriate. Reference 4 contains a similar graph for plutonium systems. This graph indicates a bias of 0.01 as being maximum across the H/fissile Pu range, and this reading is in agreement with the value reported in the previous SARP. In the previous SARP, a sub-critical margin of 0.05 was added to the above biases to find the maximum allowable k-eff. This value was confirmed as being adequate in Ref. 7, and is the value used in the current work.

The maximum allowable values of k-eff can be found by subtracting the bias and bias correction from 1.0. This results in an allowed k-eff for the U-235 systems of 0.92, and 0.94 for the plutonium systems.

Equivalences

The work performed for this task considered two different materials: dry uranium material consisting primarily of U-235 and dry plutonium material consisting primarily of Pu-239. According to Ref. 7, small amounts of U-232, U-234, and U-236 could be counted as equivalent to U-235 on a gram for gram basis in meeting the U-235 mass limit. U-238 does not count toward the mass limit. U-233 is not covered by the current work. For the plutonium material, small amounts of the other plutonium isotopes (non Pu-239) can be included in the Pu-239 mass limit if the following rules are adhered to:

1. The mass of Pu-240 and Pu-241 can be counted as equivalent on a gram for gram basis to Pu-239 if the Pu-240 mass exceeds the Pu-241 mass and if their sum is less than the Pu-239 mass.

2. Pu-242 can be treated equivalent to Pu-239 on a gram for gram basis if the Pu-241 mass exceeds the Pu-242 and if the sum of the Pu-240, Pu-241, and Pu-242 mass is less than the Pu-239 mass.

3. No equivalencies for Pu-238 or Pu-236 can be made.

These guidelines are outlined in References 4 and 7.

Analysis

The criticality analysis performed for the current work differs somewhat from the previous work. The latter used the ANISN code, a 1-d discrete ordinates code, to find the fissile mass limits. This was accomplished by first finding the critical radius, buckling, and extrapolation length of an infinite cylinder using HRXN-ANISN, and then calculating the critical height from the definition of geometric buckling. The appropriate extrapolation length was then applied to both the diameter and height. This set of calculations was repeated over a wide concentration range of fissile solutions, to ensure that the optimal moderating conditions were found. The calculation performed in the first step used the proper wall thickness for the PCV, SCV, and shielding material (if any), but the volume of the PCV itself was varied as needed to contain the critical mass. The inside diameter was, however, limited to no greater than 5.10 inches. There are three points of interest here:

1. While the PCV diameter was limited, the height was not. Therefore, critical volumes greater than the PCV volume were achieved in some cases. As will be shown later, this resulted in a much more stringent mass limit than is needed for one of the cases considered.

2. The real geometry of the shipping casks was not maintained (radially or axially).

3. The analysis assumes separability of the axial and radial flux. This assumption may not be valid for small fissile volumes.

4. Heterogeneous solutions were not considered.

In addition, the inner diameter limit on the PCV should be 5.174 inches as opposed to 5.10 inches, in order to account for pipe tolerances. This was discussed in the section above.

The current analysis sought to address these issues. Two-dimensional discrete ordinates calculations were performed using TWOTRAN. The geometry of the shipping cask was always the same, with only the fissile solution geometry inside the PCV changing with concentration. Three types of TWOTRAN calculations had to be performed. Since the mass of fissile material was fixed (see Table 1), the volume required for a fissile solution of a known concentration can be easily calculated. In cases where this volume exceeds the PCV volume, the PCV was filled with the fissile solution. In these cases, the fissile mass is less than the permitted limit. In cases where the fissile concentration is high enough that the fissile solution volume is less than the PCV volume, a cylinder of fissile solution with appropriate volume is centered within the PCV, and surrounded by water. Tables 3-7 are worksheets that break up the concentration range into three phases. The first range corresponds to complete filling of the PCV volume with fissile solution. In the second range, the fissile cylinder has a radius equal to the PCV but a height less than the PCV height. In phase 3, the fissile cylinder is centered radially and axially inside the PCV. These three phases are illustrated in Figure 9.

5. DISCUSSION OF CONTINGENCIES

The current analysis, combined with that of Ref. 1, considered two abnormal events: (1) the cask, and in particular, the primary containment vessel, under flooded conditions (the single unit flooded analysis), and (2) an infinite array of damaged casks (damage is assumed to reduce the separation between fissile units). The analysis also investigated the casks under normal (undamaged) conditions. Because the fissile contents are in solid form, it is not necessary to consider the fissile material leaking out of the PCV. The cases envelope

those required by 10-CFR-71 for fissile Class I packages, and are sufficient to ensure criticality safety under all conditions.

6. EVALUATION AND RESULTS

The results of the TWOTRAN runs are shown in Figures 10 and 11. As can be seen, the calculated k-eff is less than the critical limit for all of the casks for both the U-235 and Pu-239 cases. In all cases the curve of k-eff versus concentration has two maxima. The first maxima occurs when the fissile volume is exactly equal to the PCV volume (*i.e.*, the concentration is equal to the mass limit divided by PCV volume). Any decrease in concentration from this point results in less fissile material inside the PCV, and thus k-eff decreases to the left of this maxima. Increasing the concentration beyond this point results in a decrease in fissile volume and thus increase in neutron leakage. At first, the increase in neutron leakage is more important than the increase in fissile concentration, and the k-eff decreases. A minimum is achieved at about 10 Kg/L for U-235 and about 4 kg/L for Pu-239, and then k-eff increases as concentration increases. In this region, the increase in fissile concentration overshadows the increase in neutron leakage. This effect is more pronounced due to the large amount of reflection present in these systems. The second maxima is reached at the pure metal state. For the U-235 cases, the global maxima corresponded to the first peak for the 9972, 9974 and 9975 casks, and to the second peak for the 9973 cask. The global maxima was the second peak for all of the plutonium cases.

Note that the k-eff versus concentration curve for the 9974 cask with U-235 is very much lower than that for the other casks. This is due to the much lower mass limit placed on the U-235 in this cask (7.5 kg versus 14.5 kg for the other casks). This limit was imposed due to the very low critical mass of this system at very small concentrations. Recall that the methodology used in Ref. 1 for finding the mass limit placed no limit on the fissile volume. It turns out that the 7.5 kg mass limit was based on a point on the curve of fissile mass versus concentration that results in a fissile volume far greater than the PCV volume (larger than the PCV volume by about a factor of 3). Since the current work restricts the fissile volume to the actual geometry, this peak is not seen.

Verifying Calculations

The J70 version of TWOTRAN has a maximum mesh capability of 8000 cells. For casks as large as those modeled here, this results in a rather coarse mesh. In addition, since it is an old code without the modern acceleration routines, execution is very time consuming. The result is that the TWOTRAN models used had a fairly coarse mesh and an S_4 quadrature order. It is desirable to run with a much finer mesh and a higher quadrature order to confirm the calculational results. This is not possible with J70 TWOTRAN; however, it is quite easy to do with DORT. DORT is a modern 2-D discrete ordinates code that has recently been obtained from the Radiation Shielding and Information Center. It has been installed and certified on the SRS Cray (Ref. 8). DORT was first set up to duplicate the TWOTRAN mesh and quadrature at each of the two maxima for all casks, to ensure that good agreement with the TWOTRAN result was obtained. The results shown in Table 8 indicate reasonable agreement between the TWOTRAN and DORT coarse mesh S_4 results. For both codes, the eigenvalue convergence criteria was 1.0×10^{-4}. It is apparent the codes do not calculate the same eigenvalue to within convergence criteria. This is due to differences in the way the two codes perform negative flux fixup, rebalance acceleration, and the way they choose a starting direction. The minor discrepancies that arise from these differences have been noted by other authors (Ref. 9).

Table 8 indicates two cases in which the differences between coarse mesh TWOTRAN and coarse mesh DORT are much larger than the others (9974 cask with uranium at 2.0 and 19.05 kg/L). Comparison to the other results, and to the DORT fine mesh calculations (discussed below) indicate the DORT results are more accurate. A review of the TWOTRAN input turned up no errors. The author suspects that the TWOTRAN algorithm simply underpredicted k-eff in this case due to the very coarse mesh and the large amount of lead in the system.

After verifying that DORT gives similar results to TWOTRAN when the same mesh and quadrature are used, a very fine mesh DORT calculation with S_{16} quadrature was performed. The mesh spacing was calculated using the formulae suggested in the TWOTRAN section in Ref. 5. The total number of mesh cells in these DORT calculations was between

100,000 and 150,000. The results of these calculations are also tabulated in Table 8. Note that increasing the quadrature order and number of mesh cells, *i.e.*, doing a more accurate calculation, results in a decrease in k-eff as compared to the coarse mesh results in all cases, indicating the coarse mesh results are conservative. Also note that the fine mesh DORT values are all well below the fissile limits.

One final set of calculations was performed. The next to last column in Tables 3-7 gives the diameter of a sphere containing the fissile volume. For higher concentrations of fissile solution, the sphere diameter is small enough to fit inside the PCV. However, it is not possible to model a sphere inside a cylinder with the discrete ordinates codes currently available at SRS. In spite of this, it is important to consider the sphere case, since a sphere will have a larger k-eff than a cylinder of equal volume. Monte Carlo codes do not have the geometry limitations of discrete ordinates codes, and thus provide a means of analyzing this situation. KENO V.a models were set up for the pure metal cases, modeling first the fissile cylinder, and then the fissile sphere. The results are shown in Table 9. In all of the KENO V.a cases, 1200 batches with 1200 histories/batch were used, for a total of 1,440,000 histories. The standard deviations for the k-eff values reported in Table 9 are between .0065 and .0084. The KENO V.a results for the fissile cylinder are in excellent agreement with the fine mesh-high quadrature order DORT calculations, as indicated in the table. The KENO V.a sphere results are about .02 higher in k-eff than the cylinder results in all cases. However, the results are still somewhat below the critical limits in all cases. It should be noted that the cylinders used in the analysis are not of optimal shape. DORT fine mesh calculations of optimal metal cylinders resulted in k-eff values that are about .006-.01 smaller than the sphere values. It was not necessary to consider optimal cylinders, however, since spheres were analyzed, and spheres are always more reactive than cylinders.

Consideration of a Heterogeneous Geometry

The calculations discussed thus far have assumed the fissile material to be evenly distributed throughout the fissile volume; i.e., a homogeneous mixture. It has been demonstrated through both experiment and calculation that the maximum k-eff of a solution occurs when the fissile material is heterogeneously distributed in a manner that results in a flat neutron flux (Ref. 10). Using Table VII of Ref. 11, one can determine that adding a value of .02 to the k-eff found with a homogeneous distribution of a material will ensure that the maximum k-eff has been bounded, regardless of fissile material distribution. For the current work, it is appropriate to consider this addition at the point of the first maximum in Figures 10 and 11. At the second maximum, the material is pure metal; redistributing it would only recreate the situation achieved at the first maximum. From Table 8, adding .02 to the TWOTRAN values achieved at the first maximum results in a k-eff that is greater than the critical limit. However, as was shown above, the TWOTRAN values are overly conservative due to the coarse mesh and low quadrature order. If one considers the fine-mesh, high quadrature order DORT calculations, adding .02 to the values at the first maximum results in values that are still below the critical limit.

New Mass Limits

The value of k-eff at each of the two peaks must be below the critical limit to ensure criticality safety. As discussed above, this is true for all cases at the first maxima, even with the .02 factor added to account for heterogeneity. At the second maximum, the most reactive case is that of the fissile metal sphere. Since this case was analyzed with KENO V.a, it is necessary to add 3σ, where σ is the standard deviation, to the value of k-eff; the result must still be below the critical limit. As can be seen from the values in Table 9, this is not the case for the 9974 with plutonium, where the result is 0.9395 and σ=0.0008 (at $+3\sigma$, k=0.9419 which is greater than the critical limit of 0.94). In order to ensure criticality safety for this cask, it was necessary to reduce the mass limit to 4.3 kg (from 4.4 kg), which results in a k-eff of 0.9365 and σ=0009. This value is still below the critical limit at $+3\sigma$.

It is also desirable to increase the uranium mass limit for the 9974 cask. As was discussed previously, the original value was based on a physically unrealizable geometry and fissile volume. The DORT fine mesh calculation was repeated at the first maximum using a 14.5 kg mass limit (the old limit was 7.5 kg), and the uranium metal sphere calculation was repeated at the other maximum. In both cases, the results were below the critical limit.

Table 10 lists the maximum values of k-eff obtained for each of the casks with U-235 and Pu-239. The maximum k-eff values reported in this table are the larger of the two k-eff from the following cases:

1. The k-eff produced by DORT at the first maximum, plus .02.

2. The k-eff produced by KENO V.a for the spherical metal fissile mass, at the $+3\sigma$ limit.

The values for the 9974 cask correspond to the new mass limits for that cask. It turns out that for the U-235 cases, the larger value occurs with case 1 above, while with Pu-239, it occurs with case 2. This is true for all four of the casks.

Note that in all cases, the maximum k-eff is below the critical limit.

The new mass limits are presented in Table 11.

7. ADMINISTRATIVELY CONTROLLED LIMITS AND REQUIREMENTS

As a requirement of this analysis, the amount of fissile material in a single shipping cask must not exceed the limits given in Table 11. The controls on this limit should follow the guidance of the double contingency principle, with a minimum of two independent administrative checks of the fissile mass in a cask.

The uranium isotopes U-232, U-234, and U-236 can be counted as equivalent to U-235 on a gram for gram basis in meeting the uranium mass limit. U-238 does not count toward the uranium mass limit. *U-233 in not covered by the current analysis.*

Small amounts of Pu-240, Pu-241, and Pu-242 can be counted as equivalent to Pu-239 on a gram for gram basis in meeting the plutonium mass limit if the following rules are adhered to:

1. The mass of Pu-240 and Pu-241 can be counted as equivalent on a gram for gram basis to Pu-239 if the Pu-240 mass exceeds the Pu-241 mass and if their sum is less than the Pu-239 mass.

2. Pu-242 can be treated equivalent to Pu-239 on a gram for gram basis if the Pu-241 mass exceeds the Pu-242 and if the sum of the Pu-240, Pu-241, and Pu-242 mass is less than the Pu-239 mass.

Pu-236 and Pu-238 are not covered by the current analysis.

8. SUMMARY AND CONCLUSIONS

A thorough single flooded unit analysis of the 9972-9975 shipping casks for dry uranium material consisting primarily of U-235 and for dry plutonium materials consisting primarily of Pu-239 has demonstrated that the mass limitations in Table 11 are sufficient to maintain criticality safety. Both the uranium and plutonium mass limits for the 9974 cask have changed from the original values, while the limits for the remaining casks have not changed.

9. REFERENCES

1. J.H. Dickson, *NCSA 83-1: Safe Loadings for the 9965, 9966, 9967, 9968 Shipping Packages*, DPSPU-83-272-66, Apr. 28, 1983. See also Rev. 1, Sept. 22, 1983.

2. A.N. Mallen, TWOTRAN2: *A Two-Dimensional Transport Theory Code for JOSHUA*, DPST-75-203, Jan. 7, 1975.

3. H.K. Clark, *Subcritical Limits for Uranium-235 Systems*, Nucl. Sci. & Eng. 81, 351 (1982)

4. H.K. Clark, *Subcritical Limits for Plutonium Systems*, Nucl. Sci. & Eng. 79, 65 (1981).

5. H.K. Clark, *JOSHUA Nuclear Criticality Safety Modules*, DPSTM-86-700-3, Mar. 1987.

6. ASTM A 530/A 530M-91a, *Standard Specification for General Requirements for Specialized Carbon and Alloy Steel Pipe*, approved Jul. 15, 1991.

7. J.F. Mincey, *Chapter 6 of the SARP for the 9965, 9966, 9967 and 9968 Containers (U)*, SRT-CMA-930029, May 28, 1993.

8. R.L. Frost, DORT Certification Package, WSRC-TR-94-022, Jan. 1994.

9. W.F. Miller, Jr., R.E. Alcoufee, G.E. Bosler, F.W. Brinkley, Jr., and R.D. O'dell, *A Timing Comparison of Two-Dimensional Discrete-Ordinates Codes for Criticality Calculations*, Proceedings of the Topical Meeting on Computational Methods in Nuclear Engineering, April 23-25, 1979, Williamsburg, VA.

10. G. Goertzel, *Minimum Critical Mass and Flat Flux*, J. Nucl. Energy 2, 193-201 (1956).

11. H.K. Clark, *Effect of Distribution of Fissile Material on Critical Mass*, Nucl. Sci. & Eng. 24, 133-141 (1966).

TABLE 1

Previously Determined Shipping Cask Mass Limits* From Ref. 1

Cask	Maximum Uranium Mass (kg)	Maximum Plutonium Mass (kg)
9972 (9965)	14.5	4.4
9973 (9966)	14.5	4.4
9974 (9967)	7.5	4.4
9975 (9968)	14.5	4.4

*Note: The limits have changed as a result of the current analysis. See Table 11.

TABLE 2

Materials Used in the Criticality Analysis

Material	Description
H2O at 20°C	HRXN Standard Material #1
SS-304	HRXN STandard Material #5
U/H2O Homogeneous Solution	U density = 19.05 g/cc
Pu/H2O Homogeneous Solution	Pu density = 19.84 g/cc
Al-1100	Used pure Al at 2.70 g/cc
Aluminum Honeycomb	Al effective density = 0.2781 g/cc
Lead	Pb density = 11.29 g/cc

TABLE 3

Calculation Worksheet for the 9972 and 9973 Casks with U-235 Aqueous Solution

Conc U-235 (g/L)	Fissile Volume (cc)	Cylinder Radius	Sphere Radius	Type
1000	14500	6.57	15.13	I
1500	9667	6.57	13.21	
2000	7250	6.57	12.01	
3000	4833	6.57	10.49	
3860	3756	6.57	9.64	
4000	3625	6.57	9.53	II
5000	2900	6.57	8.85	
6000	2417	6.44	8.32	III
7000	2071	6.06	7.91	
8000	1813	5.76	7.56	
9000	1611	5.53	7.27	
10000	1450	5.32	7.02	
11000	1318	5.13	6.80	
12000	1208	4.99	6.61	
13000	1115	4.85	6.43	
14000	1036	4.73	6.28	
15000	967	4.63	6.13	
16000	906	4.52	6.00	
17000	853	4.43	5.88	
18000	806	4.35	5.77	
19000	763	4.29	5.67	

Note: Approximate volume of the 9972, 9973, and 9974 PCV as modeled is 3761 cm^3; the uranium mass limit for the 9972 and 9973 is 14.5 kg.

Attachment 1:SRT-CMA-940003

TABLE 4

Calculation Worksheet for the 9974 Cask with U-235 Aqueous Solution

Conc U-235 (g/L)	Fissile Volume	Cylinder Radius	Sphere Radius	Type
1000	7500	6.57	12.14	I
1500	5000	6.57	10.61	
2000	3750	6.57	9.64	II
3000	2500	6.57	8.42	
4000	1875	5.97	7.65	III
5000	1500	5.49	7.10	
6000	1250	5.17	6.68	
7000	1071	4.87	6.35	
8000	938	4.62	6.07	
9000	833	4.44	5.84	
10000	750	4.27	5.64	
11000	682	4.12	5.46	
12000	625	4.00	5.30	
13000	577	3.89	5.16	
14000	536	3.79	5.04	
15000	500	3.72	4.92	
16000	469	3.63	4.82	
17000	441	3.56	4.72	
18000	417	3.49	4.63	
19000	395	3.44	4.55	

Note: Approximate volume of the 9972, 9973, and 9974 PCV as modeled is 3761 cm^3; the uranium mass limit for the 9974 is 7.5 kg.

TABLE 5

Calculation Worksheet for the 9975 Cask with U-235 Aqueous Solution

Conc U-235 (g/L)	Fissile Volume	Cylinder Radius	Sphere Radius	Type
1000	14500	6.57	15.13	I
1500	9667	6.57	13.21	
2000	7250	6.57	12.01	
2690	5390	6.57	10.88	
3000	4833	6.57	10.49	II
4000	3625	6.57	9.53	
5000	2900	6.57	8.85	
6000	2417	6.44	8.32	III
7000	2071	6.06	7.91	
8000	1813	5.76	7.56	
9000	1611	5.53	7.27	
10000	1450	5.32	7.02	
11000	1318	5.13	6.80	
12000	1208	4.99	6.61	
13000	1115	4.85	6.43	
14000	1036	4.73	6.28	
15000	967	4.63	6.13	
16000	906	4.52	6.00	
17000	853	4.43	5.88	
18000	806	4.35	5.77	
19000	763	4.29	5.67	

Note: Approximate volume of the 9975 PCV as modeled is 5385 cm^3; the uranium mass limit for the 9975 is 14.5 kg.

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TABLE 6

Calculation Worksheet for the 9972, 9973, and 9974 Casks with Pu-239 Aqueous Solution

Conc U-235 (g/L)	Fissile Volume	Cylinder Radius	Sphere Radius	Type
1000	4400	6.57	10.17	I
1170	3761	6.57	9.65	
1500	2933	6.57	8.88	II
2000	2200	6.30	8.07	III
3000	1467	5.50	7.05	
4000	1100	5.00	6.40	
5000	880	4.60	5.94	
6000	733	4.33	5.59	
7000	629	4.07	5.31	
8000	550	3.87	5.08	
9000	489	3.71	4.89	
10000	440	3.58	4.72	
11000	400	3.45	4.57	
12000	367	3.35	4.44	
13000	338	3.26	4.32	
14000	314	3.18	4.22	
15000	293	3.11	4.12	
16000	275	3.04	4.03	
17000	259	2.98	3.93	
18000	244	2.92	3.88	
19840	222	2.84	3.75	

Note: Approximate volume of the 9972, 9973 and 9974 PCV as modeled is 3761 cm^3; the plutonium mass limit for these casks is 4.4 kg.

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TABLE 7

Calculation Worksheet for the 9975 Cask with Pu-239 Aqueous Solution

Conc U-235 (g/L)	Fissile Volume	Cylinder Radius	Sphere Radius	Type
818	5379	6.57	10.87	I
1000	4400	6.57	10.17	II
1500	2933	6.57	8.88	
2000	2200	6.30	8.07	III
3000	1467	5.50	7.05	
4000	1100	5.00	6.40	
5000	880	4.60	5.94	
6000	733	4.33	5.59	
7000	629	4.07	5.31	
8000	550	3.87	5.08	
9000	489	3.71	4.89	
10000	440	3.58	4.72	
11000	400	3.45	4.57	
12000	367	3.35	4.44	
13000	338	3.26	4.32	
14000	314	3.18	4.22	
15000	293	3.11	4.12	
16000	275	3.04	4.03	
17000	259	2.98	3.95	
18000	244	2.92	3.88	
19840	222	2.84	3.75	

Note: Approximate volume of the 9975 PCV as modeled is 5385 cm^3; the plutonium mass limit for these casks is 4.4 kg.

TABLE 8

DORT Verifying Calculations

Cask	Case	Coarse Mesh Calculations			Fine Mesh
		TWOTRAN	DORT	Difference	DORT
9972	3.86 kg U/L	0.9030	0.9085	-0.0055	0.8956
	19.05 kg U/L	0.8903	0.8889	0.0014	0.8778
	1.17 kg Pu/L	0.8747	0.8737	0.0010	0.8676
	19.85 kg Pu/L	0.9351	0.9322	0.0029	0.9149
9973	3.86 kg U/L	0.8786	0.8767	0.0019	0.8695
	19.05 kg U/L	0.8911	0.8891	0.0020	0.8783
	1.17 kg Pu/L	0.8497	0.8477	0.0020	0.8406
	19.85 kg Pu/L	0.9363	0.9335	0.0028	0.9167
9974	2.00 kg U/L	0.7917	0.8497	-0.0580	0.8432
	19.05 kg U/L	0.7469	0.7692	-0.0223	0.7507
	1.17 kg Pu/L	0.8756	0.8738	0.0018	0.8669
	19.85 kg Pu/L	0.9326	0.9391	-0.0065	0.9219
9975	2.69 kg U/L	0.8898	0.8932	-0.0034	0.8849
	19.05 kg U/L	0.8864	0.8937	-0.0073	0.8832
	0.818 kg Pu/L	0.8712	0.8690	0.0022	0.8605
	19.85 kg Pu/L	0.9341	0.9355	-0.0014	0.9183

TABLE 9

KENO V.a Sphere Calculations

Cask	Case	DORT Fine Mesh	KENO V.a	Difference
9972	19.05 kg U/L - Cylinder	0.8778	0.8782	-0.0004
	19.05 kg U/L - Sphere	---	0.8846	--
	19.84 kg Pu/L - Cylinder	0.9149	0.9155	-0.0006
	19.84 kg Pu/L - Sphere	---	0.9339	--
9973	19.05 kg U/L - Cylinder	0.8783	0.8777	0.0006
	19.05 kg U/L - Sphere	---	0.8889	--
	19.84 kg Pu/L - Cylinder	0.9167	0.9176	-0.0009
	19.84 kg Pu/L - Sphere	---	0.9357	--
9974	19.05 kg U/L - Cylinder	0.7507	0.7506	0.0001
	19.05 kg U/L - Sphere	---	0.7585	--
	19.84 kg Pu/L - Cylinder	0.9219	0.9238	-0.0019
	19.84 kg Pu/L - Sphere	---	0.9395	--
9975	19.05 kg U/L - Cylinder	0.8832	0.8818	0.0014
	19.05 kg U/L - Sphere	---	0.8940	--
	19.84 kg Pu/L - Cylinder	0.9183	0.9186	-0.0003
	19.84 kg Pu/L - Sphere	---	0.9367	--

TABLE 10

Maximum k-eff Values Compared to Critical Limits

	U-235		Pu-239	
Cask	Max k-eff	Critical Limit	Max k-eff	Critical Limit
9972	0.9156	0.9200	0.9364	0.9400
9973	0.8895	0.9200	0.9381	0.9400
9974	0.9143	0.9200	0.9391	0.9400
9975	0.9049	0.9200	0.9399	0.9400

TABLE 11

New Mass Limits for the 9972-9975 Casks

Cask	Maximum Uranium Mass (kg)	Maximum Plutonium Mass (kg)
9972 (9965)	14.5	4.4
9973 (9966)	14.5	4.4
9974 (9967)	14.5	4.3
9975 (9968)	14.5	4.4

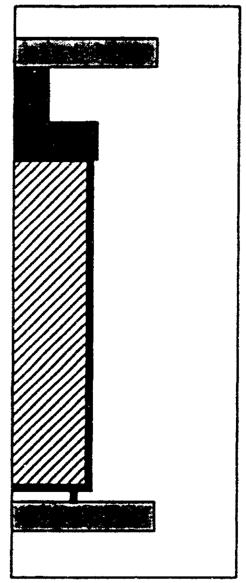

Figure 1. Model of the 9972 cask. The unshaded areas represent water. The light shading indicates aluminum, the dark shading SS-304. The composition and geometry of the interior of the PCV depends on the case being modeled (see Figure 5). Everything out to the top and bottom plate is to scale.

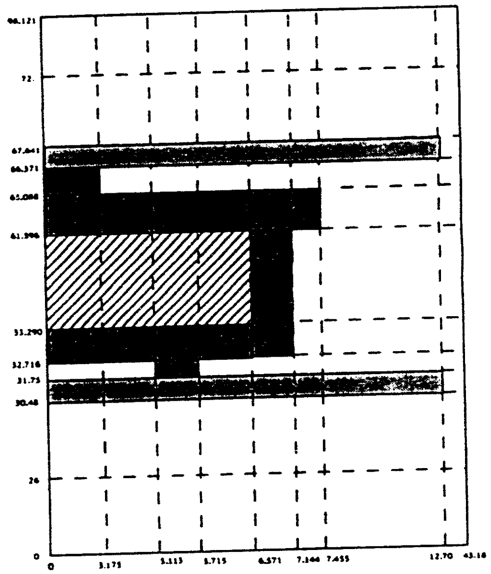

Figure 2. TWOTRAN model of the 9972 shipping cask.

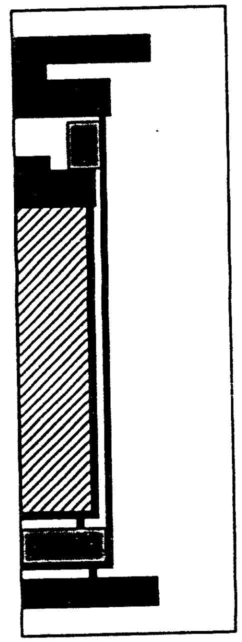

Figure 3. Model of the 9973 cask. The unshaded areas represent water. The lightest shading indicates aluminum honeycomb, the darker shading aluminum, and the darkest shading SS-304. The composition and geometry of the interior of the PCV depends on the case being modeled (see Figure 5). Everything out to the top and bottom plates is to scale.

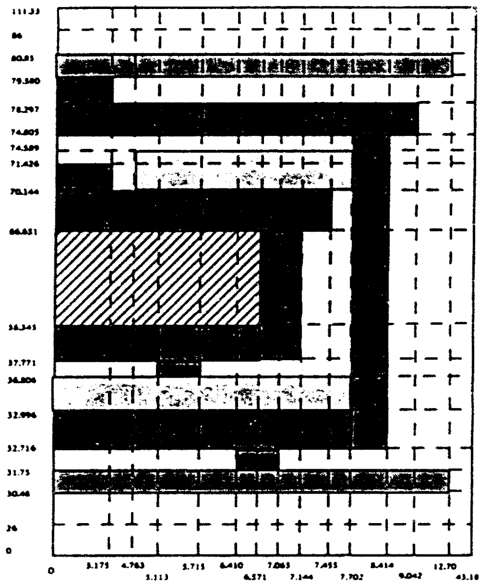

Figure 4. TWOTRAN model of the 9973 shipping cask.

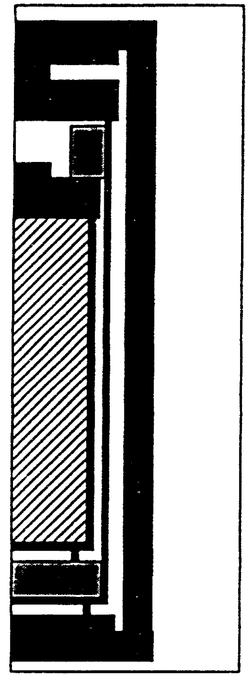

Figure 5. Model of the 9974 cask. The unshaded areas represent water. The lightest shading indicates aluminum honeycomb, the darker shading SS-304, and the black lead. The composition and geometry of the interior of the PCV depends on the case being modeled (see Figure 5). Everything out to the top and bottom plate is to scale.

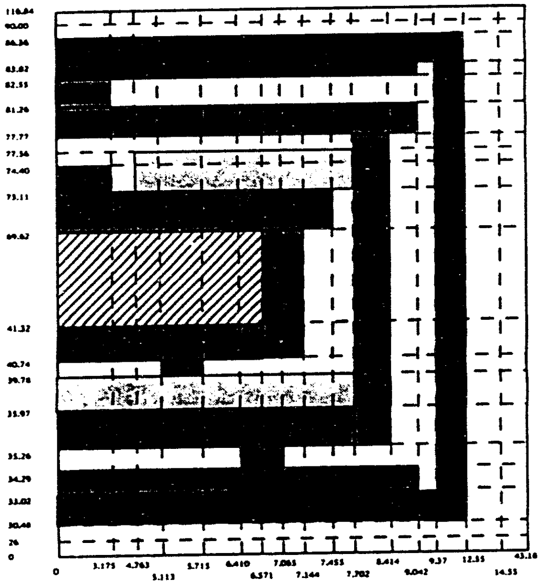

Figure 6. TWOTRAN model of the 9974 shipping cask.

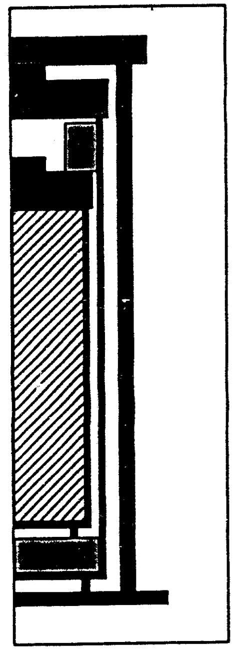

Figure 7. Model of the 9975 cask. The unshaded areas represent water. The lightest shading indicates aluminum honeycomb, the next darker shading aluminum, followed by SS-304. Solid black indicates lead. The composition and geometry of the interior of the PCV depends on the case being modeled (see Figure 5). Everything out to the top and bottom plate is to scale.

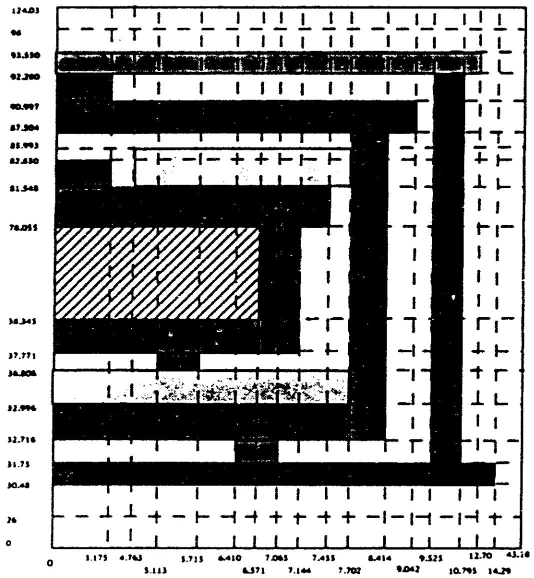

Figure 8. TWOTRAN model of the 9975 cask.

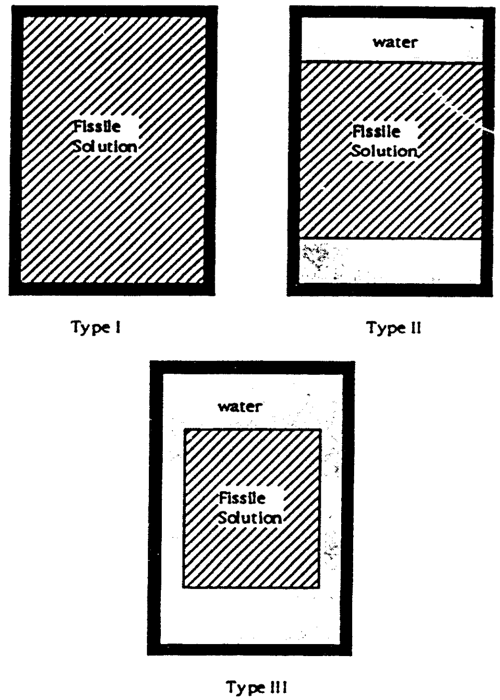

Figure 9. Illustration of the different calculation types required. The solid black region represents the PCV wall.

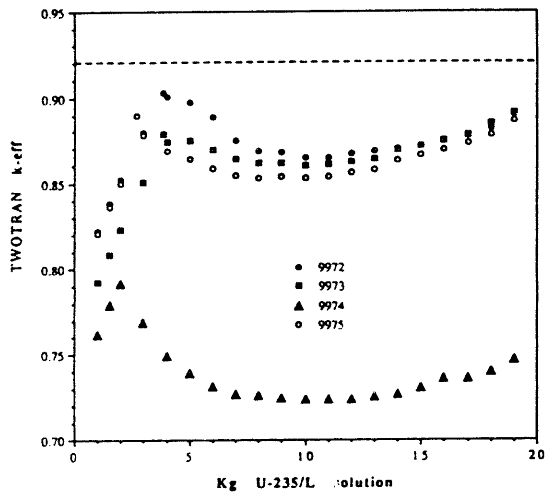

Figure 10. Results of the TWOTRAN analysis for U-235.

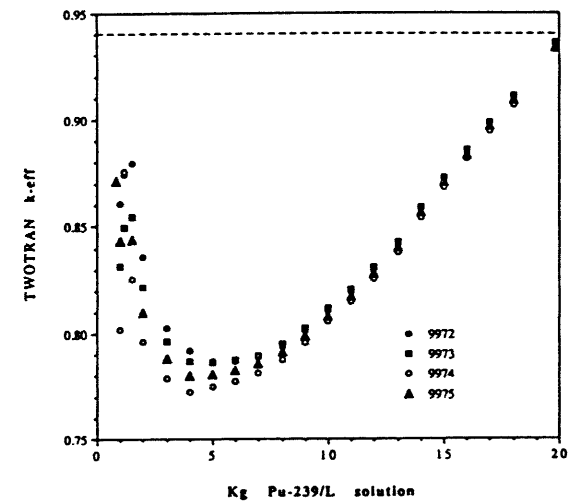

Figure 11. Results of the TWOTRAN analysis for Pu-239.

Attachment B
Review Sheet Form

OSR 24-K7

Task title Technical Review of SCS-CMA-940003 **Task number** N/A
Item reviewed SCS-CMA-940003 **Page** 1 **of** 1
(Attach additional pages as necessary; marked-up pages are acceptable.)

1. Areas reviewed *(identify clearly each area reviewed).*

See scope on page 2 of 6

2. Approaches used to perform the review.

See scope on page 2 of 6

3. Questions, comments to be resolved.

None

✓ I agree with the technical content. ___ I disagree with the technical content.

✓ I accept the conclusions and recommendations.
___ I do not accept the conclusions and recommendations for the following reasons:

Reviewer signature Roy Rathlun **Date** 3/1/94

Refer to NRTSC procedure II-14, "Technical Review," for additional information.

DATE
FILMED
6/17/94

END