DOCKET NO. 70-820

UNITED NUCLEAR CORPORATION
COMMERCIAL PRODUCTS DIVISION / ROUTE 21A, HEMATITE, MISSOURI 63047

Regulatory File Cy.

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February 27, 1970

In Reply Refer to NIS:LJS-70-673

Wood River

Mr. Donald A. Nussbaumer, Chief
Source & Special Nuclear Materials Branch
Division of Material Licensing
4915 St. Elmo Place
Bethesda, Maryland 30014

Subject: Renewal of Special Nuclear Material License SNM-777, Docket: 70-820

Reference:
1) Application for Renewal dated October 31, 1968, NIS:CP:LJS-22
2) Supplemental Information forwarded by Letter July 29, 1969 NIS:LJS-69-427
3) Your Letter dated October 28, 1969, DML:RLL 70-820; 70-36

Gentlemen:

Enclosed are seven copies each of revised pages consisting of Revision 1 of the manual, "General Information and Procedures Applicable to the Handling of Special Nuclear Material" dated October 31, 1968. These revisions were made to incorporate the additional information provided by Reference 2 and discussions in your office on November 13 and 14, 1969.

It is requested that report NDEO-1501 be withheld from Public Documentation and treated as proprietary to United Nuclear Corporation. The disclosure of this information within the public domain would permit its use by competitors without the investment funds as required by United Nuclear Corporation to develop this information. This report has been submitted as support information and does not materially affect the safety evaluation. The information which is disclosed to the public does enable an interested person to assure himself of the safety of the operation.

PDR (except NDEO-1501) 4/27/70 - RRR

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 4
FOIA-2004-0234

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ACKNOWLEDGED

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Special Nuclear Material License SNM-777 expires March 30, 1970. It is requested that this license be extended as necessary to allow your review of the material forwarded herewith and issuance of a renewed license.

Respectfully yours,

L. J. Swallow, Manager
Nuclear & Industrial Safety

LJS/gm

Enc.

AEC (7)

General Information and Procedures Applicable
to the
Handling of Special Nuclear Material

Page Changes, Additions, Deletions, Constituting Revision 1, February 6, 1970

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102. Location and Facilities (continued)

2. Fuel Fabrication Operations

The Fuel Fabrications Operations are performed at the United Nuclear Corporation fabrication plant located in New Haven, Connecticut and the assembly plant located at Montville, Connecticut.

The Naval Products Division has control over Buildings 3H, 6H, 7H, 8H, 9H basement, 10H basement, 11H basement, 14H 41H and 24D of the New Haven Facilities and the Montville Assembly Plant. Commercial
* Products Division controls the first floor of building 9H, the first floor of building 10H except the chem sample storage area, the first floor of building 11H except the experimental welding area, the first floor of building 19H, the 19H basement waste disposal area and all of building 50H (except Navy*). The Naval Products Division and the Commercial Products Division are responsible for the nuclear criticality safety and health physics in their buildings.

2.1 New Haven Operations

Figure 102-II is a map of the Corporation in New Haven. This property is located in the area generally bounded by Division Street on the north, Munson and Henry Streets on the east. The buildings indicated by the letter "H" on the map, are known as "D" Tract. The remaining buildings on the map marked as "E" Tracts (buildings 5E and 6E) do not comprise any manufacturing operations, but only office, administration, and engineering activities.

Further details on the "H" Tract arrangement and building 9H, 10H, 11H, 19H and 50H are shown in following listed drawings:

| | |
|---|---|
| B-304 238 | Equipment layout 9H, 10H, 11H |
| B-304 236 | Equipment layout 19H |
| B-304 237 | Equipment layout 50H |
| B-304 23A | Change Room layout 19H |
| B-304 252 | H Tract Building layout |

*An area approximately 50 feet x 65 feet on the north side, approximately 100 feet from the entrance.

*Indicates Change

## 103. Summary of Activities

SNM bearing materials are received, handled, stored, processed and shipped in accordance with Regulations of the Atomic Energy Commission or as provided by this license.

The maximum quantity of slightly irradiated material+ at any one time will be less than 2% of the total allowable SNM. The maximum radiation levels acceptable will be 10 mr/hr at one foot.

### 1. Recovery Operations

These materials may be in the form of uranium metal or its alloys, compounds and solutions. This includes those materials in fabricated reactor component shapes. The U-235 isotopic content of the uranium will be up to and including full enrichment.

### 2. Fuel Fabrication Operations

Operations under this license are primarily the fabrication of uranium bearing materials into specified shapes, cladding these with corrosion resistant materials such as: zirconium, stainless steel or aluminum, and assembling these into larger components or into cores for reactors.

* These materials may be in the form of uranium metal or its alloys, compounds and solutions and plutonium oxide-uranium oxide mixtures. The U-235 isotopic content of the uranium will be up to and including full enrichment. Plutonium oxide-uranium oxide mixtures will be in the form of sintered pellets in sealed rods.

+10CFR 50.2.3.ii ----- the material to be processed contains not more than $10^{-6}$ grams of plutonium per gram of U-235 and has fission product activity not in excess of 0.25 millicuries of fission products per gram of U-235. This material is in the form of fuel rods and assemblies and is only non-destructively modified and inspected.

*Indicates Change

* 104. Source and Special Nuclear Materials Possession Limits

1. Recovery Operations

* The maximum quantity on hand at any one time is limited to 4000 kgs of contained U-235.

2. Fuel Fabrication Operations

At any one time, the maximum quantity expected to be on hand is
* limited to 4000 kgs of contained U-235 and 10 kgs contained Plutonium.

*Indicates Change

202. Nuclear and Industrial Safety Department (continued)

procedures and general plant conditions for the benefit of both the Nuclear and Industrial Safety Department and Operating Personnel. These audits may serve as a management tool for joint action to correct any deficiencies noted.

2.3 Nuclear Safety Specialist

The Nuclear Safety Specialist assists the Department Manager in providing a sound program in compliance with Division Policy, Federal and State regulations. He performs Nuclear Criticality Safety evaluations of processes and equipment, plant inspections and follow up with responsible operating management. He also provides competent technical support to other Divisions on a consulting basis.

2.4 Health Physics Specialist

The Health Physics Specialist assists the Department Manager in providing a sound program in compliance with Division Policy, Federal and State regulations. He performs evaluations of radiological safety and plant inspections, and follows up with responsible operating management. He also provides competent technical support to other Divisions on a consulting basis.

2.5 Consultants

Consultants to the Nuclear and Industrial Safety Department, assist the Department Manager through reviews, technical evaluations, etc., within the area of their specialty. Such assistance is at the request of the Department Manager.

* 3. Personnel Qualifications

The Nuclear and Industrial Safety Department Manager shall hold a degree in science or engineering and have at least ten years experience in a responsible position in the nuclear industry at least three years of which have been in an activity in which he has performed nuclear criticality safety assessments and has developed an understanding of health physics and industrial safety problems and controls.

Nuclear and Industrial Safety Department Specialist shall have a B.S. Degree in Science or Engineering and not less than three years experience in a responsible nuclear engineering or physics position and not less than one year of experience in the area of their speciality. In addition, the Nuclear Safety Specialist will be required to have at least one year experience in performing nuclear safety assessments.

Nuclear and Industrial Safety Department Representatives shall have a high school diploma and not less than three years experience in the

*Indicates Change

202. Nuclear and Industrial Safety Department (continued)

nuclear industry including specific training in nuclear criticality safety, health physics, and industrial safety, and fire prevention.

Nuclear and Industrial Safety Department Consultants shall meet the same requirements as listed for the Specialist.

Resumes of the qualifications of the personnel performing these functions are included as back up information.

4. Combined Activities

* Routine health physics control services for the Commercial Products Division Fuel Fabrication Operations at New Haven will be contracted from Naval Products Division NIS personnel. These services include contamination control monitoring, sampling of ariborne effluents, monitoring and handling of liquid radioactive effluents, and health physics daily checks, etc. This service includes the evaluation of the data taken during the performance of these tasks. Results of these services, including the evaluations, will be furnished to the Commercial Products NIS Representative for record and follow-up action if required.

However, Naval Products NIS personnel have the authority to suspend unsafe operations in the Commercial Products activities.

*Indicates Change

## 204. Production Organization

### 1. General Description

1.1 United Nuclear Corporation's Commercial Products Division is organized for the specific purpose of chemical and ceramic processing SNM for further fabrication of fuel elements, actual fabrication of fuel elements and the recovery of the SNM from unirradiated scrap material. The Organization chart, Figure 201-I shows how this organization is set up to carry out and control the various functions essential for this type of manufacturing operating. The General Manager of the Commercial Products Division reports directly to the President of the Corporation.

* 1.2 Each department is headed by a Department Manager who has full responsibility and authority to carry out the functions of that department in conjunction with contributions of other departments to achieve the overall objectives of the Commercial Products Division. Each Department Manager is directly responsible to the Division General Manager for the conduct of his departmental affairs including implementation of disciplinary action against personnel failing to follow instructions. Further, each Department Manager has line responsibility to the members of his department.

### 2. Manufacturing Departments

* 2.1 The manufacturing covered by this license will be carried out under the direct responsibility of the respective Manufacturing Department Manager or Superintendent. These departments have the responsibility for manufacture, engineering and shipment applicable to the production of products described in this application.

2.2 Specific procedures are set up to insure that the proper quantities of uranium are present in the various products produced. Processing procedures are set up within the responsible department.

Management channels are established as the need for delegation of work arises. Changes at levels below the first line management level reporting to the head of a department are a management prerogative, and therefore, a detailed listing of the present supervisory levels is not provided except for the Nuclear and Industrial Safety Department which is described in a separate subsection.

*Indicates Change

## 205. Process Control

* 1. General

Commercial Products Division policy requires that supervision at all levels assure themselves that all handling, processing, storing and shipping of SNM, is given prior review and approval by the Nuclear and Industrial Safety Department, that suitable control measures are prescribed, and that all pertinent regulations, controls procedures relative to nuclear criticality safety or radiological safety, are followed by supervision and all operating personnel.

Approval by the Nuclear and Industrial Safety Department shall be in accordnace with criteria established by the license. The mechanism of such approval is described in more detail in Subsection 206.

2. Recovery Operations

Control of the process is maintained through a system of standard procedures and parameters, and provisions for reporting and correcting abnormal occurences.

2.1 Operating Procedures

Issued by the Process Engineer, provide detailed instructions for equipment operation and material handling, including specific safety requirements. S.O.P.'s are the basic control document; before issuance or revision they require Engineering, Production, Nuclear and Industrial Safety and SNM Management approval by signature.

2.2 Process Parameter Sheets

Issued by Process Engineer, establish and communicate detailed parameters for each job, within limits established by S.O. P.

2.3 Operating Reports

Operating reports are used and prepared as required for production planning and control.

3. Fuel Fabrication Operations

Control of the process is maintained through a system of written operating procedures and provisions for reporting and correcting abnormal occurrences. Operations involving SNM require prior written approval by the Nuclear and Industrial Safety Department. This is accomplished by the posting of signs with nuclear criticality safety and health physics control limits. These signs will be prepared and issued in accordance with Sections 300 and 400.

*Indicates Change

## 206. Nuclear and Industrial Safety Controls

### 1. Responsibility

On-site nuclear and industrial safety control is exercised by Operating Supervision with overchecks performed by Process Engineers and the Nuclear and Industrial Safety Representative. Operating Supervision must assure that nuclear criticality safety and health physics control procedures are followed as defined by approved operating procedures or posted control limits.

* 2. Nuclear and Industrial Safety Department Approval

NIS Department approval on equipment and operating procedures is identified by signature of the NIS Department Representative on the operating procedures and/or criticality signs. This approval shall only be granted when:

2.1 Deleted

2.2 (a) Nuclear criticality safety evaluation has been performed by the NIS Representative based on the criteria and standards of Section 300.

(b) The NIS Representative's evaluation has been reviewed by the Nuclear Safety Specialist or Nuclear and Industrial Safety Department Manager. This review is based on the criteria and standards of Section 300 and includes verification of each of the following:

1. assumptions
2. correct application of criteria of Section 300
3. completeness and accuracy of the evaluation
4. familiarity of the installation.

(c) Health Physics criteria of Section 400 have been fulfilled.

* 3. Records

Records of NIS evaluations and approvals will be maintained for a period of at least six (6) months after use of the operation has been terminated.

* 4. Suspension of Operations

Primary responsibility and authority to suspend unsafe operations is placed with Operating Supervision. Within their respective responsibilities the members of the Nuclear and Industrial Safety Department also have authority to suspend operations not being performed in accordance with approved procedures.

*Indicates Change

## 207. Inspections and Audits

### 1. General

A continuous re-appraisal of the safety program is provided through a system of daily checks, regular inspections, and audits. Health physics personnel, thoroughly familiar with regular operations, make daily checks to determine that there has been no change in the parameters or conditions of operations, that may affect the safety of these operations. A planned schedule of regular inspections is established by the Department Manager. Infractions and violations are corrected on the spot with the concurrence of the cognizant Specialist and/or Manager of Industrial Safety. Results of inspections and audits are included in the department monthly report.

### 2. Daily Checks

Daily checks and visits are observations made routinely by Health Physics Technicians who observe, note, and make general observations in addition to their radiation survey functions.

### * 3. Inspections

Inspections are performed by NIS Department Representative (non-resident), Specialist or NIS Manager. An inspection includes a review of checks to determine the area or areas requiring more detailed observation. Generally, a specific area will be observed for a sufficient time to indicate corrective action if needed. Inspections are documented and maintained as a record for at least one year. These inspections will be performed as follows:

| Function | Minimum Frequency# |
|---|---|
| Health Physics | 2 months |
| Nuclear Criticality Safety | 2 months |

#The minimum frequency is increased when new operations are in the startup phase.

*Indicates Change

207. Inspections and Audits (continued)

4. Audits

An audit is a more thorough inspection in which the results of previous checks, inspections or audits are also reviewed, as an evaluation on the effectiveness of the program. All aspects of the activities involved, including the equipment, facilities and operator's knowledge are covered. A review of the follow-up action taken on previous audits and inspections, the recommended corrective action, and a schedule date which such action will be accomplished are also covered. These audits may also involve a detailed review of non-safety documents such as operating procedures, route cards, etc. It may also include a review of the nuclear materials management program. These audits are documented by a formal report to the Division General Manager. Records of audits are maintained for at least one year. An audit is performed once a year by a team appointed by the General Manager.

*Indicates Change

## 302. General Requirements

### 1. Purpose

1.1 These standards provide basic nuclear criticality safety criteria for:

1.1.1 Design of new plant facilities and equipment, or

1.1.2 Modification of existing facilities and equipment, or

1.1.3 UNC Commercial Products Division internal evaluations, reviews and authorizations in lieu of obtaining formal AEC license approval.

* 1.2 The type of activities for which these standards are applicable are those described in Sections 700 and 900.

### 2. Basic Principles

2.1 All new and revised plant facilities, and equipment intended for processing, handling or storing special nuclear material shall be designed for nuclear criticality safety. The nuclear criticality safety evaluation shall be based on principle that at least two unlikely and unrelated events must occur before accidental criticality can result.

2.2 All evaluations shall assume complete water reflection and optimum moderation unless mechanical or physical controls are present
* which prevent these conditions under normal and abnormal conditions.

2.3 In cases in which different SNM are present in an area at the same time, the limits for the more reactive SNM will apply to all SNM in the area.

*Indicates Change

## 303. Evaluations

### 1. Purpose

Evaluations will be performed considering factors which may affect the criticality of the system. These include:

1.1 Enrichment
1.2 Geometry
1.3 Moderation
1.4 Reflection
1.5 Mass
1.6 Volume
1.7 Concentration
1.8 Interaction
1.9 Structural Integrity
1.10 Poisons (if applicable)
1.11 Homogeneity and Heterogeneity of the System

### 2. Determination of Safe Values

#### 2.1 Individual Units

The tables and graphs in Subsection 309 contain the basic limits which are used to obtain operating criticality safety limits. These graphs and tables have safety factors incorporated into them.

#### 2.2 Interaction

When evaluating interaction between units of an array or group of arrays, the following techniques will be applied:

2.2.1 Solid Angle Method - Solid angle of the most reactive units shall be calculated in accordance with the following:

2.2.1.1 The criteria set forth in TID-7016, Rev. 1.
2.2.1.2 Keff values listed in Table XVII and footnote 6 on page 30, K-1019, Rev. 5, will be utilized
* except for specific values used and explicitly quoted in Section 800.
2.2.1.3 Solid angles equal to or less than 0.005 steradians will be neglected.

2.2.2 K$\Omega$ Method - Solid angles shall be calculated using the figures in Y-1272.

*Indicates Change

303. Evaluations (continued)

2.2.3 Criticality Zones

The interaction between criticality zones may not be specifically evaluated if the following are met:

a) The SNM in each criticality zone is separated from SNM in other criticality zones by at least one (1) foot.

b) The plant average surface density of SNM does not exceed 175 grams U-235 per square foot of aspect area. The aspect area applies to plant areas where SNM is processed and handled. This value may be used if the SNM in the criticality zone has a fraction critical which does not exceed 0.3. The maximum size units in each zone shall be evaluated as illustrated in attached Nuclear Safety Evaluation.

* Storage devices are not considered criticality zones. Interaction between storage devices and criticality zones is considered only when adjacent criticality zones are not isolated in accordance with the criteria of Subpart 303.2.2.4.

2.2.4 Isolation

Individual units or arrays are considered isolated from neutron interaction when separated by one of the following:

* a) Eight (8) inches of solid concrete with a density of 140 pounds per cubic foot. Less dense concrete may be used, provided the thickness is increased in inverse proportion to the concrete density. This is applicable only to units no more reactive than those of Table 309-II (refer to attached Nuclear Safety Evaluation, Concrete Isolation).

b) Twelve (12) feet or the greatest distance across the orthographic projection of the largest unit or array on a plane perpendicular to a line joining the center of that unit or array to other units or arrays, whichever is greater.

*Indicates Change

Nuclear Safety Evaluation - Safe Surface Density

## I. Determination of Limiting Value

A safe surface density value of 175 grams U-235 per square foot of planar area was obtained by replotting the data on Figure 11, TID-7028, in terms of U-235 content per square foot of planar area versus U-235 density. The curve for a fully reflected uniform slab has a minimum of about 350 grams U-235 per square foot. A safety factor of 2 was applied to determine the safe surface density value. Also, the 175 gram U-235/square foot is consistent with the safe 200 gram U-235 per square foot reported by R. S. Stevenson and R. H. Odgaarden in "Studies of Surface Density Spacing Criteria using KENO Calculations" for units of a maximum fraction critical of 0.3.

## II. Effect of Individual Units in Criticality Zones

The SNM in criticality zones is usually in bottles, containers or reactor component shapes. The effects of these units on the safe surface density approach was evaluated. The safe surface density approach used to evaluate thse units is described by H. C. Paxton in LA-3366 and R. S. Stevenson and R. H. Odgaarden in "Studies of Surface Density Spacing Criteria using KENO Calculations". These reports discuss the method in some detail.

## III. Calculations

### A. Safe Limit (10 kg U-235)

Assuming a spherical shape for the 10 kg unit at full metal density, the actual mass is

$$M_a = 10 \text{ kg U-235}$$

From Figure 8, TID-7028, the minimum bare critical mass for full density uranium with H/U-235 $\leq 2$ is

$$M_{c,b} = 48 \text{ kg U-235}$$

Then, the fraction critical is

$$f = \frac{M_a}{M_{c,b}} = \frac{10 \text{ kg U-235}}{48 \text{ kg U-235}} = .208$$

### B. Safe Wet Zone Limit (350 gm U-235)

Assuming a spherical shape for the 350 gm unit, the actual mass is

$$M_a = 350 \text{ gm U-235}$$

| | |
|---|---|
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| SECTION: 303 | SUBPART: 303.2.2. |
| Nuclear Safety Evaluation - Safe Surface Density | |
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| SUPERSEDES: 10/31/68 | |
| APPROVED: | |
| AMENDMENT NO.: | |

Safe Wet Zone Limit (continued)

From Figure 8, TID-7028, the minimum bare critical mass for a moderated system occurs at a H/U-235 of approximately 400 and is

$M_{c,b}$ = 1400 gm U-235

Then, the fraction critical is

$$f = \frac{M_a}{M_{c,b}} = \frac{350 \text{ gm U-235}}{1400 \text{ gm U-235}} = .250$$

C. <u>Safe U-Al Fuel Element Wet Zone Limit (3 Elements)</u>

The highest loaded fuel element processed is the BAWTR element which is described in Subpart 822 (for SNM-777). Each fuel element has the following dimensions and characteristics:

Fuel envelop dimensions = 2.4" x 2.9" x 30"

U-235 mass per element = 592.4 gm

H/U-235 = 102

The volume of the fuel envelop is

$V_{fe}$ = 2.4" x 2.9" x 30" x 16.38 $cm^3/in^3$ = 3.42 l

The density per fuel element, which is the system density, is

$$\rho_{fe} = \frac{.5924 \text{kg}}{3.42 \text{ l}} = 0.173 \text{ kg/l}$$

The bare buckling data in the "Nuclear Safety Evaluations-BAWTR Fuel Elements", Subpart 822, has been plotted on Figure 1. The bare buckling for 3 fuel elements is

$B_g^2 = 0.058 \text{ cm}^{-2}$ where $\delta = 3$ cm

Using equation (6) page 6, NDEO-1050, and rearranging terms, the radius of an equivalent sphere is

$$r = \sqrt{\frac{\pi^2}{B_g^2}} - \delta$$

$$= \sqrt{\frac{9.87}{.058}} - 3.0$$

$$= \sqrt{170.17} - 3.0$$

$$= 10.04 \text{ cm}$$

Safe U-Al Fuel Element Wet Zone Limit (continued)

The volume of the sphere with this radius is

$$V = \frac{4}{3}\pi r^3 = 4240 \text{ cc}$$

At a density of 0.173 gm/cc, the equivalent mass is

$$M_e = \rho_{fg} \times V = .173 \text{ gm/cc} \times 4240 \text{ cc} = 734 \text{ gm}$$

From Figure 8, TID-7028, the minimum bare critical mass for a system with a density of 0.173 gm/cc is

$$M_{c,b} = 2650 \text{ gm U-235}$$

Then, the fraction critical is

$$f = \frac{M_e}{M_{c,b}} = \frac{734 \text{ gm U-235}}{2650 \text{ gm U-235}} = .276$$

D. Safe $UO_2$ Pellet Type Assembly Wet Zone Limit (1 element)

The YANKEE fuel assembly is the most reactive unit processed and is described in Subpart 823 (of SNM-777). Each fuel assembly has the following dimensions and characteristics:

Fuel envelop dimensions = 7.615" x 7.615" x 91"

U-235 mass per element = 238 rods x 36.49 gm U-235/rod = 8.685 kg U-235

The volume of the fuel envelop is

$$V_{fe} = 7.615" \times 7.615" \times 91" \times 16.38 \text{ cm}^3/\text{in}^3 = 86.436 \text{ liters}$$

The density per fuel element, which is the system density, is

$$\rho_{fe} = \frac{8.685 \text{ Kg U-235}}{86.436 \text{ l}} = 0.100 \text{ kg U-235/l}$$

The bare buckling data in the "Nuclear Safety Evaluation-Yankee Fuel Elements (PWR)", Subpart 823, calculated a bare buckling for one fuel assembly of

$$B^2_g = 0.0333 \text{ cm}^{-2} \qquad \text{where } \delta = 2.5$$

Using equation (6) page 6, NDEO-1050, and rearranging terms, the radius of an equivalent sphere is,

$$r = \sqrt{\frac{\pi^2}{B_g^2}} - \delta$$

$$= \sqrt{\frac{9.87}{.0333}} - 2.5$$

$$= \sqrt{296.39} - 2.5$$

$$= 14.71 \text{ cm}$$

Safe $UO_2$ Pellet Type Assembly Wet Zone Limit (continued)

The volume of a sphere with this radius is,

$$V = \frac{4}{3}\pi r^3 = 13.337 \text{ liters}$$

At a density of 0.100 kg U-235/1, the equivalent mass is

$$M_e = \rho_{fe} \times V = 0.100 \text{ kg U-235/1} \times 13.337 \text{ 1} = 1.334 \text{ kg U-235}$$

As listed in NDEO-1134 which is included with the "Nuclear Safety Evaluation-Yankee Fuel Elements (PWR)", Subpart 823,

$$k_\infty = 1.405, \quad M^2 = 40.9 \text{ cm}^2$$

Using the standard equation for effective multiplication factor, and rearranging terms, the critical buckling is

$$B_c^2 = \frac{K_\infty - Keff}{M^2} \qquad \text{where Keff} = 1$$

$$= \frac{.405}{40.9}$$

$$= 0.0099 \text{ cm}^{-2} \qquad \text{where } \delta = 25 \text{ cm}$$

Using equation (6) page 6, NDEO-1050, and rearranging terms, the radius of an equivalent sphere is,

$$r = \sqrt{\frac{\pi^2}{B_c^2}} - \delta$$

$$= \sqrt{\frac{9.87}{.0099}} - 2.5$$

$$= \sqrt{996.96} - 2.5$$

$$= 29.1 \text{ cm}$$

The volume of a sphere with this radius is

$$V_c = \frac{4}{3}\pi r^3 = 103.25 \text{ liters}$$

At a density of 0.100 kg U-235/1, the critical mass is,

$$M_{c,b} = \rho_{fe} \times V_c = 0.100 \text{ kg U-235/liter} \times 103.25 \text{ 1} = 10.325 \text{ kg U-235}$$

Then the fraction critical is

$$f = \frac{M_e}{M_{c,b}} = \frac{1.334 \text{ kg U-235}}{10.325 \text{ kg U-235}} = .129$$

IV. Conclusions

The individual units meet the criteria

$$f \leq 0.3$$

| | |
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| SECTION: 300 | SUBPART: 303.3.3.2 |
| Nuclear Safety Evaluation-Safe Surface Density | |
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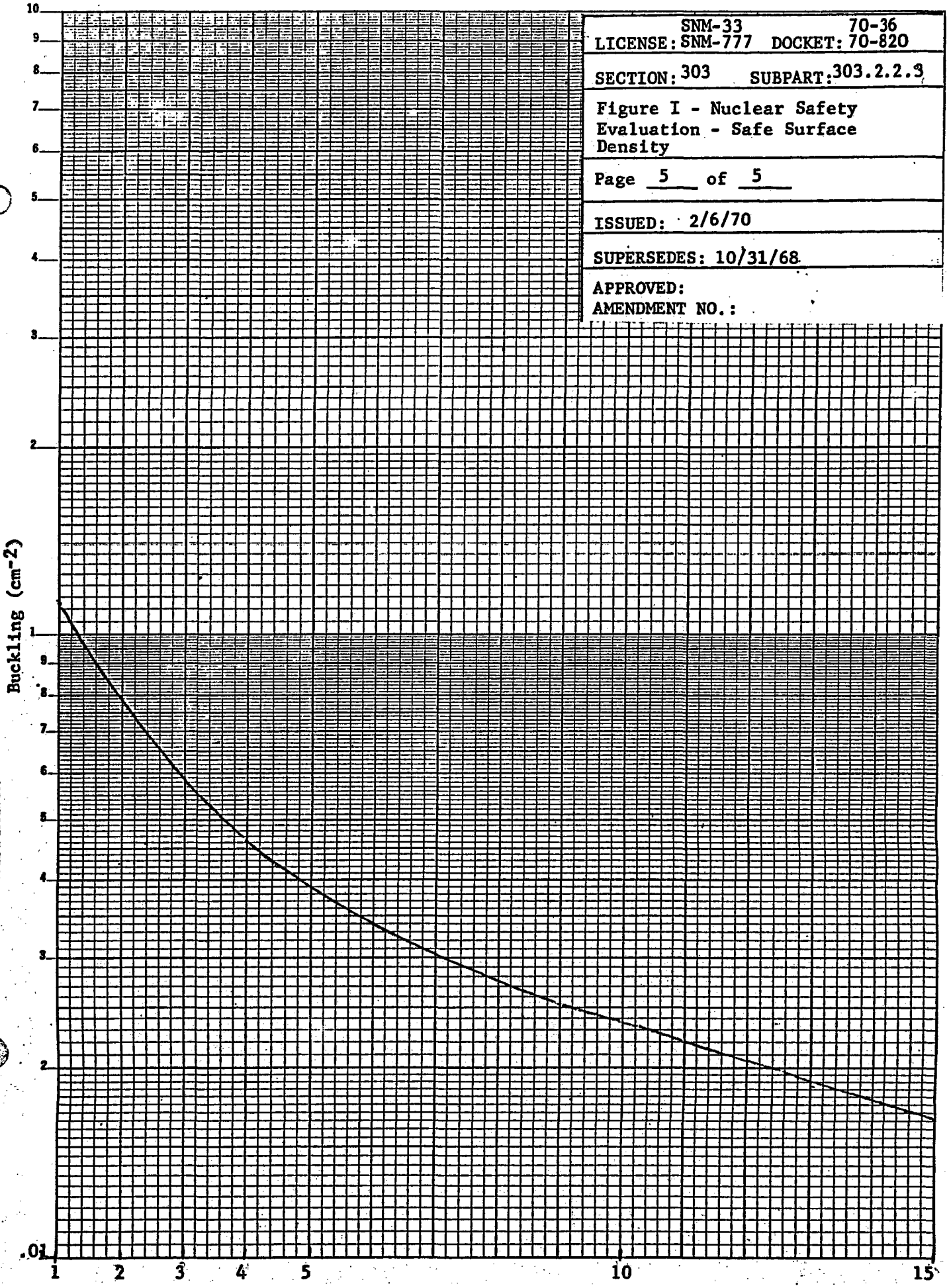
| APPROVED: AMENDMENT NO.: | |

FIGURE I

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| LICENSE: SNM-33 SNM-777 | DOCKET: 70-36 70-820 |
| SECTION: 303 | SUBPART: 303.2.2.3 |
| Figure I - Nuclear Safety Evaluation - Safe Surface Density | |
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Subpart 303.2.2.4

Nuclear Safety Evaluation, Concrete Isolation

The use of 8" of concrete as an effective neutron isolator has been evaluated by J. D. White and C. R. Richey, "Neutron Interaction between Multiplying Media Separated by Various Materials", pages 57-67, BNL-193. These experiments were performed for $PuO_2$-polystyrene plus plexiglas critical arrays. These materials had H/Pu = 15 in the fuel compact and H/Pu = 35.6 in the 3 dimensional checkerboard heterogeneous array. This resulted in a somewhat thermalized spectrum; Figure 5, TID-7016, Rev. 1 indicates the H/Pu atomic ratios greater than 20 are considered "solutions" and therefore thermal systems. Although the experiments were performed for moderated Pu, the results should be acceptable for moderated U systems since neutrons from Pu and U thermal systems have similar behavior. Figure 7 indicates that 8" (20.32 cm) of concrete acts as an effective isolator for thermalized systems. It should be noted that the fuel core had cross sectional dimensions of 30.632 x 30.886 cm so that the results should be valid for small slabs as well as cylinders or spheres. Based on this data, the use of 8" of concrete to isolate solution units is considered valid.

Regarding other than solution systems (i.e., metal-water, dry metal, dry oxide, etc) page 41, LA-2063 states "Two arrays are effectively isolated from one another if the arrays are completely separated by concrete at least 8" thick." Again on page 45, "A complete concrete wall at lease 8" thick effectively isolates one process area from another." To use this criteria, individual units must meet the maximum unit quantities listed in Table V, page 39, LA-2063.

The units listed in Table 309-II meet these requirements since Table 309-II was developed from Table IV of TID-7016, Rev. 1. Table IV of TID-7016, Rev. 1 is similar and somewhat consistent with Table V of LA-2063.

## 303. Evaluations

### 303.2.3 $N B_N^2$ Method for Interaction Evaluation

General Description of Method:

The $N B_N^2$ method extends simple, well-known concepts applicable to individual critical units to critical arrays of subcritical components. The approach is based on the following postulates:

1. The fraction of neutrons that leak from different critical arrays of the same units is a constant, and

2. The non-leakage fraction of a critical array of N units is suitably represented by the expression $(1 + N B_N^2)^{-1}$ where $B_N^2$ is the goemetrical buckling of the system.

Therefore, for two systems of identical units, one with N and the other with N' units

$$N B_N^2 = N' B_{N'}^2$$

and for cubical arrays the buckling is:

$$B_N^2 = \sum_{i=1}^{3} \frac{\left(\pi\right)^2}{(a_N + 2T)_i} = \sum_{i=1}^{3} \frac{\pi^2}{[n_i (d_i + \delta_i) + 2T]^2}$$

where: $n_i = \sqrt[3]{N}$

$a_N$ = characteristic length of the cubical array (vault length)
$\delta_i$ = surface to surface separation distance of units
$d_i$ = unit dimensions (diameter of sphere and cylinder, side length of cube or rectangle)
$T$ = array reflector savings
$d+\delta$ = unit cell length.

303.2.3 (continued)

Application of the method consists of determining N $B_N^2$ of a known (reference) critical array, holding this constant and by varying either or both the number of units or array density establish parameters for an equivalent critical array. Further, effect of:

1. Array reflector must be included. The significance of the reflector is reduced for arrays of a large number of units. The data against which the method is correlated is based on a 6" thick paraffin reflector. Our applications involve 8"-12" concrete walls as the array reflector. Because the number of units in our arrays is large the difference in these reflector materials is therefore not significant.

2. The units in our array must be equivalent to the units in the critical array. This is accomplished when the units are such that the density of the equivalent critical array is the same as the density of the reference array. (Reference 3)

3. Unit cell shape must be considered. Variations of unit cell shape have a minimum effect provided the total volume of the unit cell remains constant. The same is true for deviation of the array shape from optimum (cubical). (Reference 3 and 5)

4. Enrichment must be considered. The method has been derived from array experiment data for 93% enrichments. In applying it to 97% enriched arrays, the number of units is conservatively reduced by a factor of 2. (Reference 5)

Application of Method

The referenced critical arrays are those defined by Figure 4 of "Remarks on Array Criticality Techniques", J. T. Thomas, CONFR 680909. This is reproduced and attached herewith.

303.2.3 N $B_N^2$ Method for Interaction Evaluation (continued)

Calculate the number of units, n, on each side of a cubic array from :

$$(CTC)_N = \sqrt{\frac{3\,\pi^2\,n}{(N\,B_N^2)^1}} - \frac{2\,T}{n}$$ equation 4; reference 3

where,

$(CTC)_N$ = center to center distance of units is equilateral cubic array, i.e., length of side of cubic unit cell.

T = array reflector savings
n = number of units on each side of a cubic array
$N = n^3$ = total number of units in the array.

$(NB_N{}^2)^1$ = determined from referenced array of total of $N^1$ units and $(B_N{}^2)^1$ buckling.

To determine $(NB_n{}^2)^1$, enter Figure 4, based on the unit cell density determined from the actual unit mass (in array to be evaluated) and the cell volume 28.04 liters for which Figure 4 is developed. From this, the corresponding number of units in a critical array can be determined. The buckling of the critical array is then calculated from the $B_N{}^2$ equation above using (d x δ) = 30.4 cm.

Unit cell volume is:

a) For equilateral spaced units in a cubic array--the volume calculated by the cube of the center to center spacing.

b) For non-equilateral spaced units--the average array volume per unit; or, one-half the volume determined by the product of the actual center to center spacing of the units.

For a safe array:

A margin of safety is established such that $K_{eff}$ of the array is .9. This is established as follows:

$$\frac{\Delta k}{K} = \frac{\Delta \bar{\rho}}{\bar{\rho}} \quad \text{(Eq. 11, Ref. 2)} \qquad [1]$$

303.2.3 N $B_N^2$ Method for Interaction Evaluation (continued)

Since the density of the array is the unit cell density and holding the mass per unit cell constant, eqn. [1] becomes a function of the array volume ($V_{ar}$) and may be re-written

$$\frac{\Delta k}{k} = \frac{\Delta V_{ar}}{V_{ar}} \qquad [2]$$

The critical array volume is determined by the product of the number of units critical (Nc) and the unit cell volume ($V_{uc}$). Thus:

$$V_{ar} = N_c \times V_{uc} \qquad [3]$$

Using eqn. [3], the change in critical array volume ($\Delta V_{ar}$) becomes:

$$\Delta V_{ar} = \Delta (N_c \times V_{uc}) \qquad [4]$$

Since the volume of the unit cell is constant, the number of units which comprise the array is the only variable quantity so eqn [4] becomes:

$$\Delta V_{ar} = \Delta N_c \times V_{uc} \qquad [5]$$

Substituting eqns. [3] and [5] into eqn [2]

$$\frac{\Delta k}{k} \quad \frac{\Delta N_c}{N_c} \qquad [6]$$

Rearranging eqn. [6] to determine the change in number of units required to reduce the array reactivity:

$$\Delta N_c = N_c \frac{(\Delta k)}{(k)} \qquad [7]$$

Setting the safe array $K_{eff} = .9$:

$$\Delta K = K - .9 = 1.0 - .9 = .1 \qquad [8]$$

And:

$$\frac{\Delta k}{k} = \frac{.1}{1} = .1 \qquad [9]$$

303.2.3 N $B_N^2$ Method for Interaction Evaluation (continued)

Substituting eqn. [9] into eqn. [7]

$$\Delta N_c = .1 N_c \qquad [10]$$

The safe number of units ($N_s$) becomes

$$N_s = N_c - \Delta N_c = N_c - .1 N_c = .9 N_c \qquad [11]$$

To allow for enrichments from 93.5% to 97%, the allowable numbers of units (Na) will be safe number of units reduced by a factor of two (2). Applying this to eqn. [11]

$$N_a = \frac{N_s}{2} = \frac{.9N_c}{2} = .45 N_c \qquad [12]$$

The arrays of each type of material must meet the criteria that the actual number of units in the array ($N_{act}$) does not exceed the allowable number of units

$$N_{act} \leq N_a$$

References:

1. ORNL-CDC-1, "Criticality of Large Systems of Subcritical U(93) Components", J. T. Thomas (Aug. 1967).

2. ORNL-CDC-4, "The Effect of Unit Shape on the Criticality of Arrays", J. T. Thomas (Oct. 1967).

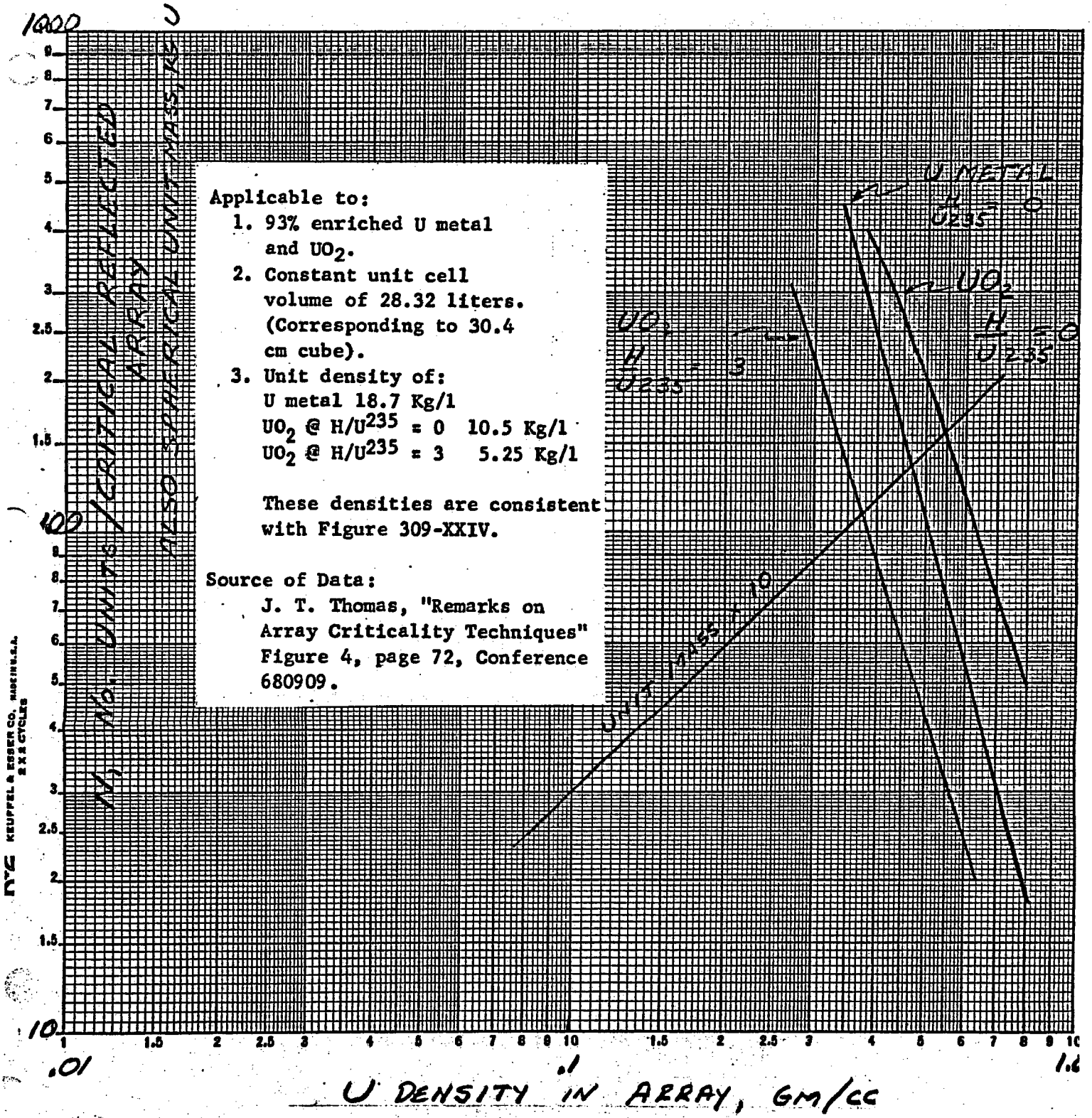
3. CONF-680909, "Remarks on Array Criticality Techniques", J. T. Thomas (Sept. 1968).

4. CONF-680909, "Dendity Analog Techniques", H. C. Paxton (Sept. 1968).

5. Personnel Communication with J. T. Thomas, Dec. 24, 1969.

6. NED 393, UNC Internal Communication, 1/2/70, Tomonto to Swallow.

# Variation of Spherical Unit Mass and Number of Units in Reflected Critical Array




Applicable to:

1. 93% enriched U metal and $UO_2$.
2. Constant unit cell volume of 28.32 liters. (Corresponding to 30.4 cm cube).
3. Unit density of:
   U metal 18.7 Kg/1
   $UO_2$ @ $H/U^{235}$ = 0 10.5 Kg/1
   $UO_2$ @ $H/U^{235}$ = 3 5.25 Kg/1

   These densities are consistent with Figure 309-XXIV.

Source of Data:

J. T. Thomas, "Remarks on Array Criticality Techniques" Figure 4, page 72, Conference 680909.

## 305. Nuclear Poisons

### 305.1 Use of Nuclear Poisons

Nuclear poisons are used only as secondary nuclear criticality control. The following types may be used.

#### 1.1 Boro-silicate Glass Raschig Rings

Boro-silicate glass raschig rings may be used in vessels containing solutions when primary nuclear criticality safety is maintained by concentration control. Such use shall be as described herein.

1.1.1. Boro-silicate glass raschig rings shall be constructed of a low expansion corrosion resistant type glass compatible with the chemical and physical enviornment.

1.1.2. Raschig rings shall be Corning Glass 7740 type or equivalent containing nominal four (4) w/o natural Boron with a range to 11.2 to 13.8 w/o $B_2O_3$.

1.1.3. Raschig rings are small hollow cylinders with length and diameter approximately equal. Wall thickness is a maximum of 1/4".

1.1.4. The raschig rings will be uniformly distributed in the vessel and will occupy at least 22% of the volume.

1.1.5. Maximum solution concentration is 10 grams U-235/liter.

* 1.1.6. Deleted

#### 1.2 Inspections

1.3.1 Samples of the raschig rings from the bottom of the vessels or from a selected sample point typical of the vessel contents will be evaluated at least once each year.

Raschig rings shall be replaced when the material is damaged or the Boron content of the glass is reduced to less than 3 w/o as determined by analysis.

1.3.2 Vessels will be checked monthly to insure that the tank contains the required amount of raschig rings and that the material is intact and in a planned location.

*Indicates Change

## 306. Criticality Zones

### 306.1 General

Nuclear Criticality Safety control may be achieved by subdividing the manufacturing or process area into logical work stations or process regions called criticality zones.

### 306.2 Criticality Zone Specifications

#### 2.1 Boundary

Criticality zone boundaries are established such that SNM within the zone is a minimum of one foot from the SNM in an adjacent zone except when transferring in or out of the zone.

#### 2.2 Zone Control Limits

Nuclear criticality safety control limits within a zone are established as described in Subsection 303.

#### 2.3 Interaction between Zones

Interaction between zones is controlled as described in Subsection 303.

#### 2.4 Type of Criticality Zones

##### 2.4.1 Wet Zones

Wet zones are established when there are no controls against introduction of moderating materials. In general, wet zones are applicable to chemical and ceramic processing, and chemical laboratories, and pickling, rinsing and degreasing operations in fuel element machining and fabrication areas.

* 2.4.2 Special Zones

Special zones are established when there are specific controls against introduction and/or use of moderating materials. These controls include:

a) Provision for free drainage of the zone or exclusion of liquids, no water lines connected to the equipment, specific limits and controls on the quantity of water or other moderating materials (such as plastics, wood, paper).

*Indicates Change

* 2.4.2 Special Zones (continued)

b) The effect of moderating materials permitted is included in the nuclear criticality safety evaluation performed for the activities within the zone.

In general, special zones are applicable to Fuel Element Fabrication Operations and dry box operations in the Chemical and Ceramic process operations.

*Indicates Change

## 307. Marking and Labeling

### 1. Criticality Limits

Signs listing approved nuclear criticality safety limits shall be posted so that information is readily discernible to employees. This posting may be for individual pieces of equipment or groups of equipment, depending on the nature of the operations covered.

1.1 Signs are prepared and issued by the NIS Representative.

1.2 Signs must be posted prior to use of SNM in the equipment or at the work station.

* 1.3 Criticality limit signs are signed in approval by the Nuclear and Industrial Safety Representative and the Production Manager.

### 2. Process Containers

Empty containers used for SNM shall be identified or marked as empty.

Process containers will have information readily available to allow identification of their contents.

*Indicates Change

* 309. Tables and Graphs

1. Purpose

The tables and graphs of this Subsection 309 are used in the nuclear criticality safety evaluation as described in Subsection 303.

2. Safety Factors

As used on the curves of this Subsection, the safety factor is the ratio of the safe unit to the critical unit.

In establishing a particular safety factor, consideration must be given to:

a) Accuracy of the data used to establish the critical unit.
b) Operating controls applicable, i.e., degree of administrative control required versus geometry control. In general, a larger safety factor is required when safety is primarily dependent on administrative controls as in the case of mass or batch control.

The safety factors applicable to the safe standards of this Subsection are the same as those in common use in the industry (reference TID-7016, Rev. 1). The critical data from which the standard safe data is developed has been reviewed and determined to be sufficiently accurate as not to warrant a further increase.

*Indicates Change

## Safe Limits for Individual Units as Metal, Compound and Solution Systems

| Safe Control Parameter | Safe Uranium Limits* | | |
|---|---|---|---|
| | Metal Systems | Compound Systems | Solution Systems |
| Mass | 10 kgs. U-235 | 350 gms. U-235 | 350 gms. U-235 |
| Cylinder Diameter | 2.7 inches | 5.0 inches | 5.0 inches |
| Cross Sectional Area | 5.725 sq. inches | 19.64 sq. inches | 19.64 sq. inches |
| Volume | 1.0 liter | 4.8 liters | 4.8 liters |
| Slab Thickness | 0.5 inches | 1.5 inches | 1.5 inches |

*Applicable to:

1. Any U-235 enrichment.
2. Full water reflection.

*Specific Conditions to be Maintained:

1. For metal,
   a) Solid metal pieces with no re-entrant holes.
   b) Smallest individual piece is 4 kg U.
   c) Densities up to and including full density.
2. For compounds,
   a) Total U density vs. H/U ratio is not greater than that of $UO_2$ as per Figure 309-XXIV.
   b) Bulk density up to and including 4 kg U/liter.
3. For solutions,
   a) Total U density vs. H/U ratio is not greater than that of $UO_2F_2$ as per Figure 309-XXIV.

Source of Data:

a) Metal - Figures 1-4, TID-7016, Rev. 1
b) Compounds - Figures 309-XXV thru XXVIII and Nuclear Safety Evaluation, 4.8 liter sphere.
c) Solutions - Figures 1-4, TID-7016, Rev.1.

| | |
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| SECTION: 300 | SUBPART: 309 |
| Table 309-I, Safe Limits for Metal, $UO_2$ and Solution Systems | |
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## I. DESCRIPTION

A. The maximum sized container for U compound handling and storage will be 4.8 liters

B. The maximum density (bulk) of material to be placed in these containers will not exceed 4 Kg U/liter.

## II. ASSUMPTIONS

A. The maximum "crystal" density of material to be placed in these containers will not exceed that of $UO_2$ (9.66 Kg U/liter).

B. The $UO_2$ - Water data of LA-3612 is applicable.

C. The calculational method set forth in Section 2.2, NDEO-1050, will be used to determine reflector savings(δ) and keff values.

D. The density vs. H/U ratio relationship will be per Fig. 309-XXIV.

## III. CALCULATIONS

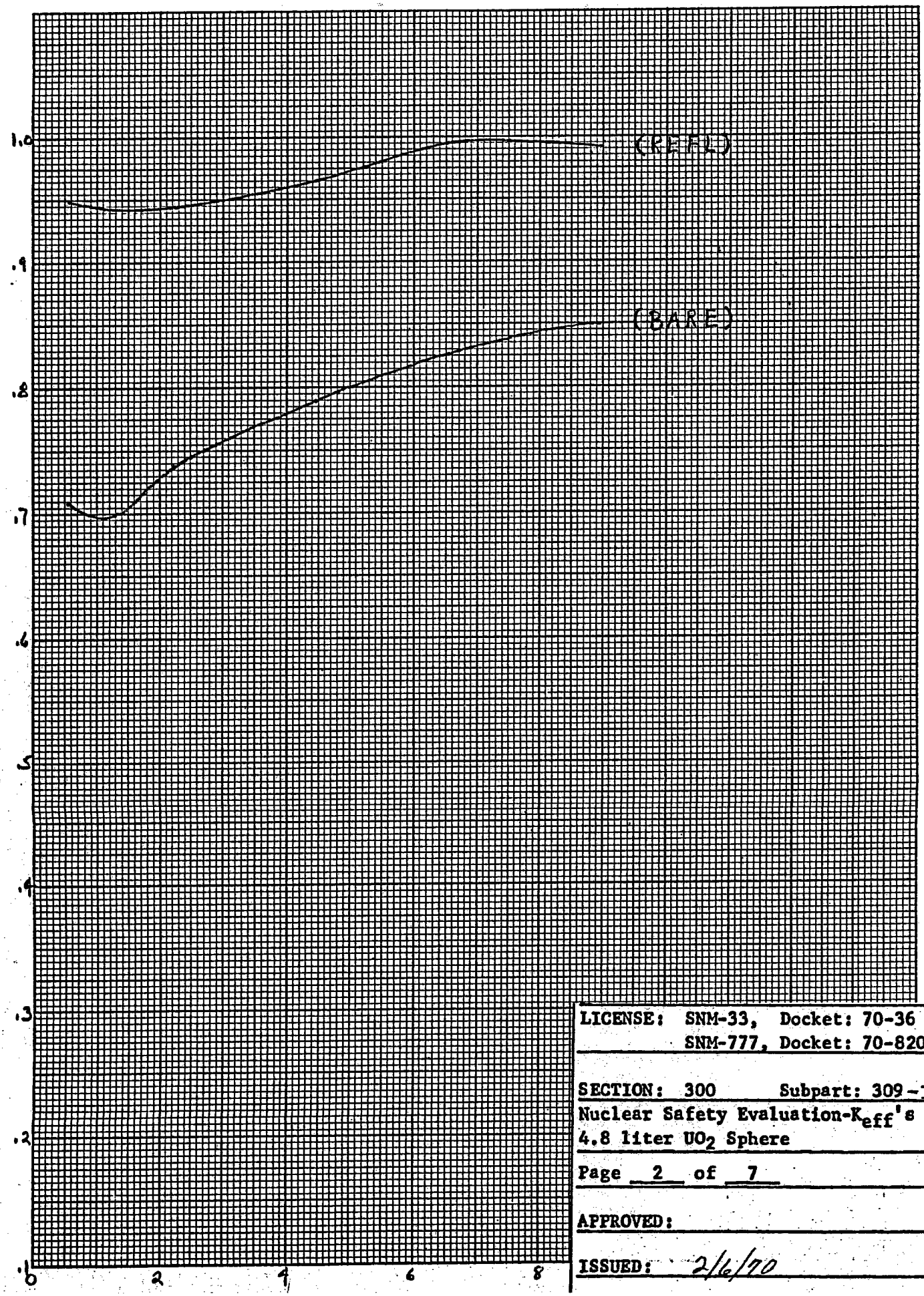
See attached tables

## IV. CONCLUSIONS

This volume is sub critical when moderated and reflected for materials with densities not exceeding 4 Kg U/liter. The keff values at a density of 4 Kg U/liter are:

keff (reflected) = .960

keff (bare) = .780

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SECTION: 300 Subpart: 309-T

Nuclear Safety Evaluation-$K_{eff}$'s of 4.8 liter $UO_2$ Sphere

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# CRITICAL BUCKLINGS

## REFLECTED

At $H/u\text{-}235 = 0$, $\rho = 8.930\ Kg/\ell$

$$B^2 = \frac{9.87}{(10.63+7.3)^2} = \frac{9.87}{(17.93)^2} = \frac{9.87}{321.48} = .0307$$

At $H/u\text{-}235 = .98$, $\rho = 6.700\ Kg/\ell$

$$B^2 = \frac{9.87}{(10.49+7.26)^2} = \frac{9.87}{(17.75)^2} = \frac{9.87}{315.06} = .0313$$

At $H/u\text{-}235 = 2.94$, $\rho = 4.470\ Kg/\ell$

$$B^2 = \frac{9.87}{(11.14+7.18)^2} = \frac{9.87}{(18.32)^2} = \frac{9.87}{335.62} = .0294$$

At $H/u\text{-}235 = 8.96$, $\rho = 2.230\ Kg/\ell$

$$B^2 = \frac{9.87}{(11.75+6.74)^2} = \frac{9.87}{(18.49)^2} = \frac{9.87}{342} = .0288$$

At $H/u\text{-}235 = 20.60$, $\rho = 1.120\ Kg/\ell$

$$B^2 = \frac{9.87}{(11.53+6.5)^2} = \frac{9.87}{(18.03)^2} = \frac{9.87}{325.08} = .0304$$

At $H/u\text{-}235 = 43.90$, $\rho = 0.558\ Kg/\ell$

$$B^2 = \frac{9.87}{(11.39+6.5)^2} = \frac{9.87}{(17.89)^2} = \frac{9.87}{320.05} = .0308$$

## BARE

At $H/u\text{-}235 = 0$, $\rho = 8.930\ Kg/\ell$

$$B^2 = \frac{9.87}{(13.43+5.46)^2} = \frac{9.87}{(18.89)^2} = \frac{9.87}{356.93} = .0277$$

At $H/u\text{-}235 = .98$, $\rho = 6.700\ Kg/\ell$

$$B^2 = \frac{9.87}{(13.90+5.32)^2} = \frac{9.87}{(19.22)^2} = \frac{9.87}{369.49} = .0267$$

At $H/u\text{-}235 = 2.94$, $\rho = 4.470\ Kg/\ell$

$$B^2 = \frac{9.87}{(14.87+5.03)^2} = \frac{9.87}{(19.90)^2} = \frac{9.87}{396.01} = .0249$$

At $H/u\text{-}235 = 8.96$, $\rho = 2.230\ Kg/\ell$

$$B^2 = \frac{9.87}{(15.87+4.13)^2} = \frac{9.87}{(19.98)^2} = \frac{9.87}{399.20} = .0247$$

At $H/u\text{-}235 = 20.60$, $\rho = 1.120\ Kg/\ell$

$$B^2 = \frac{9.87}{(15.6+2.5)^2} = \frac{9.87}{(18.10)^2} = \frac{9.87}{327.61} = .0301$$

At $H/u\text{-}235 = 43.90$, $\rho = 0.558\ Kg/\ell$

$$B^2 = \frac{9.87}{(15.27+2.5)^2} = \frac{9.87}{(17.77)^2} = \frac{9.87}{315.77} = .0313$$

# KEFF VALUES FOR 4.8 LITER SPHE :AL VOLUME $U(93.5)O_2$-WATER MIXTURES

| H/U-235 | $\rho$ (Kg/ℓ) | $r_{total}$ (cm) | $\delta_R$ (cm) | $B^2_{s,R}$ (1) ($cm^{-2}$) | $\tau$ (2) ($cm^2$) | keff (3) | $\delta_B$ (cm) | $B^2_{s,B}$ (1) ($cm^{-2}$) | $\tau$ (2) ($cm^2$) | keff (3) |
|---|---|---|---|---|---|---|---|---|---|---|
| 0 | 8.930 | 10.45 | 7.30 | .0313 | 28.0 | .991 | 5.46 | .0390 | 28.0 | .849 |
| .98 | 6.700 | ↓ | 7.26 | .0315 | ↓ | .997 | 5.32 | .0397 | ↓ | .828 |
| 2.94 | 4.470 | ↓ | 7.18 | .0318 | ↓ | .965 | 5.03 | .0412 | ↓ | .789 |
| 8.96 | 2.230 | ↓ | 6.94 | .0326 | ↓ | .944 | 4.13 | .0462 | ↓ | .738 |
| 20.60 | 1.120 | ↓ | 6.50 | .0344 | ↓ | .943 | 2.50 | .0589 | ↓ | .696 |
| 43.90 | 0.558 | ↓ | 6.50 | .0344 | ↓ | .949 | 2.50 | .0589 | ↓ | .708 |

NOTES: (1) $B^2 = \frac{\pi^2}{(r_s + \delta)^2}$

(2) $\tau$ = 28 cm from Section 2.2, NDEO-1050

(3) $k_{eff} = \frac{1 + \tau B_0^2}{1 + \tau B_s^2}$

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# 4.8 LITER SPHERE BUCKLING

## REFLECTED

At $H/U\text{-}235 = 0$, $\rho = 8.930\ Kg/\ell$

$$B^2 = \frac{9.87}{(10.45+7.3)^2} = \frac{9.87}{(17.75)^2} = \frac{9.87}{315.06} = .0313$$

At $H/U\text{-}235 = .98$, $\rho = 6.700\ Kg/\ell$

$$B^2 = \frac{9.87}{(10.45+7.26)^2} = \frac{9.87}{(17.71)^2} = \frac{9.87}{313.64} = .0315$$

At $H/U\text{-}235 = 2.94$, $\rho = 4.470\ Kg/\ell$

$$B^2 = \frac{9.87}{(10.45+7.18)^2} = \frac{9.87}{(17.63)^2} = \frac{9.87}{310.82} = .0318$$

At $H/U\text{-}235 = 8.96$, $\rho = 2.230\ Kg/\ell$

$$B^2 = \frac{9.87}{(10.45+6.94)^2} = \frac{9.87}{(17.39)^2} = \frac{9.87}{302.41} = .0326$$

At $H/U\text{-}235 = 20.60$, $\rho = 1.120\ Kg/\ell$

$$B^2 = \frac{9.87}{(10.45+6.5)^2} = \frac{9.87}{(16.95)^2} = \frac{9.87}{287.30} = .0344$$

At $H/U\text{-}235 = 43.90$, $\rho = 0.558\ Kg/\ell$

$B^2 = .0344$ (same as for $\rho = 1.120$)

## BARE

At $H/U\text{-}235 = 0$, $\rho = 8.930\ Kg/\ell$

$$B^2 = \frac{9.87}{(10.45+5.46)^2} = \frac{9.87}{(15.91)^2} = \frac{9.87}{253.13} = .0390$$

At $H/U\text{-}235 = .98$, $\rho = 6.700\ Kg/\ell$

$$B^2 = \frac{9.87}{(10.45+5.32)^2} = \frac{9.87}{(15.77)^2} = \frac{9.87}{248.69} = .0397$$

At $H/U\text{-}235 = 2.94$, $\rho = 4.470\ Kg/\ell$

$$B^2 = \frac{9.87}{(10.45+5.03)^2} = \frac{9.87}{(15.48)^2} = \frac{9.87}{239.63} = .0412$$

At $H/U\text{-}235 = 8.96$, $\rho = 2.230\ Kg/\ell$

$$B^2 = \frac{9.87}{(10.45+4.13)^2} = \frac{9.87}{(14.58)^2} = \frac{9.87}{212.58} = .0462$$

At $H/U\text{-}235 = 20.60$, $\rho = 1.120\ Kg/\ell$

$$B^2 = \frac{9.87}{(10.45+2.5)^2} = \frac{9.87}{(12.95)^2} = \frac{9.87}{167.70} = .0589$$

At $H/U\text{-}235 = 43.90$, $\rho = 0.558\ Kg/\ell$

$B^2 = .0589$ (same as for $\rho = 1.120$)

# 4.8 LITER SPHERE KEFF'S

## REFLECTED

At H/U-235 = 0, $\rho$ = 8.93 Kg/l

$$k_{eff} = \frac{1+(28 \times .0307)}{1+(28 \times .0313)} = \frac{1.860}{1.876} = .991$$

At H/U-235 = .98, $\rho$ = 6.700 Kg/l

$$k_{eff} = \frac{1+(28 \times .0313)}{1+(28 \times .0315)} = \frac{1.876}{1.882} = .997$$

At H/U-235 = 2.94, $\rho$ = 4.470 Kg/l

$$k_{eff} = \frac{1+(28 \times .0294)}{1+(28 \times .0318)} = \frac{1.823}{1.890} = .965$$

At H/U-235 = 8.96, $\rho$ = 2.230 Kg/l

$$k_{eff} = \frac{1+(28 \times .0288)}{1+(28 \times .0326)} = \frac{1.806}{1.913} = .944$$

H/U-235 = 20.60, $\rho$ = 1.120 Kg/l

$$k_{eff} = \frac{1+(28 \times .0304)}{1+(28 \times .0344)} = \frac{1.851}{1.963} = .943$$

At H/U-235 = 43.98 $\rho$ = 0.558 Kg/l

$$k_{eff} = \frac{1+(28 \times .0308)}{1+(28 \times .0344)} = \frac{1.862}{1.963} = .949$$

## BARE

At H/U-235 = 0, $\rho$ = 8.93 Kg/l

$$k_{eff} = \frac{1+(28 \times .0277)}{1+(28 \times .0340)} = \frac{1.776}{2.092} = .849$$

At H/U-235 = .98, $\rho$ = 6.700 Kg/l

$$k_{eff} = \frac{1+(28 \times .0267)}{1+(28 \times .0397)} = \frac{1.748}{2.112} = .828$$

At H/U-235 = 2.94, $\rho$ = 4.470 Kg/l

$$k_{eff} = \frac{1+(28 \times .0249)}{1+(28 \times .0412)} = \frac{1.697}{2.154} = .789$$

At H/U-235 = 8.96, $\rho$ = 2.230 Kg/l

$$k_{eff} = \frac{1+(28 \times .0247)}{1+(28 \times .0462)} = \frac{1.692}{2.294} = .738$$

At H/U-235 = 20.60, $\rho$ = 1.120 Kg/l

$$k_{eff} = \frac{1+(28 \times .0301)}{1+(28 \times .0589)} = \frac{1.843}{2.649} = .696$$

At H/U-235 = 43.90, $\rho$ = 0.558 Kg/l

$$k_{eff} = \frac{1+(28 \times .0313)}{1+(28 \times .0589)} = \frac{1.876}{2.649} = .708$$

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## Safe Limits for Individual Units of Metal and Compounds

| Degree of Moderation H/U-235 More Than | Not More Than | Mass Limits* (Kgs. U-235) | Keff Reflected | Keff Bare |
|---|---|---|---|---|
| - | 2 | 10.0 | .896 | .708 |
| 2 | 3 | 9.0 | .899 | .715 |
| 3 | 5 | 7.3 | .897 | .708 |
| 5 | 10 | 5.2 | .904 | .697 |
| 10 | 20 | 3.6 | .913 | .675 |
| | | Volume Limits (Liters) | | |
| 20 | - | 3.6 | .947 | .721 |

*Applicable to:

1. Any U-235 enrichment.
2. Full water reflection.

*Specific Conditions to be Maintained:

1. Interspersed moderation does not exceed listed H/U atom ratio.
2. Total U density vs. H/U atom ratio is not greater than that of metal or compound per Figure 309-XXIV.

Source of Data:

1. TID-7016, Rev. 1., Table IV (modified)
2. Nuclear Safety Evaluation, Table 309-II.

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## I. DESCRIPTION

A. Units will be as described in Table 309-II (taken from Table IV modified, TID-7016, Rev. 1)

## II. ASSUMPTIONS

A. Critical values will be determined from data in TID-7028
B. Reflector savings and keff calculations will be obtained using the technique in Section 2.2, NDEO-1050.
C. A $\tau$ value of 28 cm will be used.
D. Formulas used are

1. $\delta_R = 7.3 - .04\,(H/x)$

2. $\delta_B = 5.46 - .148\,(H/x)$

3. $B^2 = \left(\frac{\pi}{r+\delta}\right)^2$

4. $k_{eff} = \frac{1+\tau B_c^2}{1+\tau B_A^2}$ where $B_c^2$ = critical buckling, $B_A^2$ = actual buckling

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For U Metal - Water Mix ture @ $H/X \simeq 2$

$V_{c,R} = 2.4\,\ell$, $V_{c,B} = 5.4\,\ell$, $\rho_c = 7.48\,Kg/\ell$ from Fig. 9, T10-7028

$$r_{c,R} = \sqrt[3]{\frac{3V_{c,R}}{4\pi}} = \sqrt[3]{\frac{2400}{4.19}} = \sqrt[3]{573} = 8.32 \text{ cm}$$

$$r_{c,B} = \sqrt[3]{\frac{3V_{c,B}}{4\pi}} = \sqrt[3]{\frac{5400}{4.19}} = \sqrt[3]{1290} = 10.88 \text{ cm}$$

$$\delta_R = 7.3 - .04\,(H/X) = 7.3 - .08 = 7.26 \text{ cm}$$

$$\delta_B = 5.46 - .148\,(H/X) = 5.46 - .396 = 5.064 \text{ cm}$$

$$B^2_{c,R} = \frac{\pi^2}{(r_{c,R}+\delta_R)^2} = \frac{9.87}{(8.32+7.26)^2} = \frac{9.87}{(15.58)^2} = \frac{9.87}{242.74} = .0407 \text{ cm}^{-2}$$

$$B^2_{c,B} = \frac{\pi^2}{(r_{c,B}+\delta_B)^2} = \frac{9.87}{(10.88+5.064)^2} = \frac{9.87}{(15.944)^2} = \frac{9.87}{254.21} = .0388 \text{ cm}^{-2}$$

With a 10 Kg $U^{235}$ limit

$$V_A = \frac{10\,Kg}{7.48\,Kg/\ell} = 1.34\,\ell$$

$$r_a = \sqrt[3]{\frac{3V_a}{4\pi}} = \sqrt[3]{\frac{1340}{4.19}} = \sqrt[3]{320} = 6.85 \text{ cm}$$

$$B^2_{A,R} = \frac{\pi^2}{(r_A+\delta_R)^2} = \frac{9.87}{(6.85+7.26)^2} = \frac{9.87}{(14.11)^2} = \frac{9.87}{199.09} = .0496 \text{ cm}^{-2}$$

$$B^2_{A,B} = \frac{\pi^2}{(r_A+\delta_B)^2} = \frac{9.87}{(6.85+5.064)^2} = \frac{9.87}{(11.914)^2} = \frac{9.87}{141.94} = .0695 \text{ cm}^{-2}$$

The keff's are calculated using

$$k_{eff} = \frac{1+\tau B^2}{1+\tau B^2}$$ where $\tau = 28 \text{ cm}^2$ from Pg. 24, Sect. 22, NDEO-

$$k_{eff,R} = \frac{1+(28 \times .0407)}{1+(28 \times .0496)} = \frac{2.1396}{2.3888} = .896$$

$$k_{eff,B} = \frac{1+(28 \times .0388)}{1+(28 \times .0695)} = \frac{2.0864}{2.946} = .708$$

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For U Metal-Water Mixture @ $2 \leq H/x \leq 3$

$V_{c,R} = 2.7\ \ell$, $V_{c,B} = 6.0\ \ell$, $\rho_c = 5.18\ Kg/\ell$ from Fig. 9, TID-7028

$$r_{c,R} = \sqrt[3]{\frac{3V_{c,R}}{4\pi}} = \sqrt[3]{\frac{2700}{4.19}} = \sqrt[3]{645} = 8.65\ cm.$$

$$r_{c,B} = \sqrt[3]{\frac{3V_{c,R}}{4\pi}} = \sqrt[3]{\frac{6000}{4.19}} = \sqrt[3]{1430} = 11.28\ cm.$$

$$\delta_R = 7.3 - .04(H/x) = 7.3 - .12 = 7.18\ cm$$

$$\delta_B = 5.46 - .148(H/x) = 5.46 - .444 = 5.016\ cm$$

$$B^2_{c,R} = \frac{\pi^2}{(r_{c,c}+\delta_R)^2} = \frac{9.87}{(8.65+7.18)^2} = \frac{9.87}{(15.83)^2} = \frac{9.87}{250.54} = .0394\ cm^{-2}$$

$$B^2_{c,B} = \frac{\pi^2}{(r_{c,B}+\delta_B)^2} = \frac{9.87}{(11.28+5.016)^2} = \frac{9.87}{(16.296)^2} = \frac{9.87}{265.56} = .0372\ cm^{-2}$$

With a 9 Kg $U^{235}$ limit

$$V_A = \frac{9\ Kg}{5.18\ Kg/\ell} = 1.55\ \ell$$

$$r_A = \sqrt[3]{\frac{3V_A}{4\pi}} = \sqrt[3]{\frac{1550}{4.19}} = \sqrt[3]{370} = 7.19\ cm$$

$$B^2_{A,R} = \frac{\pi^2}{(r_A+\delta_R)^2} = \frac{9.87}{(7.19+7.18)^2} = \frac{9.87}{(14.37)^2} = \frac{9.87}{206.50} = .0478\ cm^{-2}$$

$$B^2_{A,B} = \frac{\pi^2}{(r_A+\delta_B)^2} = \frac{9.87}{(7.19+5.016)^2} = \frac{9.87}{(12.206)^2} = \frac{9.87}{148.99} = .0662\ cm^{-2}$$

The keff's are calculated using

$$k_{eff} = \frac{1+\tau B^2}{1+\tau B^2}$$ where $\tau = 28\ cm^2$ from Pg. 24, Sect. 2.2, NDCO-1050

$$k_{eff,R} = \frac{1+(28 \times .0394)}{1+(28 \times .0478)} = \frac{2.1032}{2.3384} = .899$$

$$k_{eff,B} = \frac{1+(28 \times .0394)}{1+(28 \times .0662)} = \frac{2.0416}{2.8536} = .715$$

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For U Metal - Water Mixture @ $3 \leq H/X \leq 5$

$V_{c,R} = 3.2\,\ell$, $V_{c,B} = 7\,\ell$, $\rho_c = 4\ Kg/\ell$ from Fig. 9, TID-7028

$$r_{c,R} = \sqrt[3]{\frac{3200}{4.19}} = \sqrt[3]{765} = 9.16\ cm$$

$$r_{c,B} = \sqrt[3]{\frac{7000}{4.19}} = \sqrt[3]{1675} = 11.9\ cm$$

$$\delta_R = 7.3 - .04(H/X) = 7.3 - .2 = 7.1\ cm$$

$$\delta_B = 5.46 - .148(H/X) = 5.46 - .74 = 4.72\ cm$$

$$B^2_{c,R} = \frac{9.87}{(9.16+7.1)^2} = \frac{9.87}{(16.26)^2} = \frac{9.87}{264.39} = .0373\ cm^{-2}$$

$$B^2_{c,B} = \frac{9.87}{(11.9+4.72)^2} = \frac{9.87}{(16.62)^2} = \frac{9.87}{276.22} = .0357\ cm^{-2}$$

With a 7.3 Kg $U^{235}$ limit

$$V_A = \frac{7.3\ Kg}{4\ Kg/\ell} = 1.825\,\ell$$

$$r_A = \sqrt[3]{\frac{1825}{4.19}} = \sqrt[3]{436} = 7.59\ cm$$

$$B^2_{A,R} = \frac{9.87}{(7.59+7.1)^2} = \frac{9.87}{(14.69)^2} = \frac{9.87}{215.80} = .0457\ cm^{-2}$$

$$B^2_{A,B} = \frac{9.87}{(7.59+4.72)^2} = \frac{9.87}{(12.31)^2} = \frac{9.87}{151.54} = .0651\ cm^{-2}$$

The Keff's are

$$K_{eff,R} = \frac{1+(28 \times .0373)}{1+(28 \times .0457)} = \frac{2.0444}{2.2796} = .897$$

$$K_{eff,B} = \frac{1+(28 \times .0357)}{1+(28 \times .0651)} = \frac{1.9996}{2.8228} = .708$$

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For U-Metal-Water Mixture @ $5 \leq H/X \leq 10$

$V_{c,A} = 3.9\ \ell$, $V_{c,B} = 8.6\ \ell$, $\rho_c = 2.25\ Kg/\ell$ From Fig. 9, TID-2026

$$r_{c,A} = \sqrt[3]{\frac{3V_{c,A}}{4\pi}} = \sqrt[3]{\frac{3900}{4.19}} = \sqrt[3]{931} = 9.76\ cm$$

$$r_{c,B} = \sqrt[3]{\frac{3V_{c,B}}{4\pi}} = \sqrt[3]{\frac{8600}{4.19}} = \sqrt[3]{2055} = 12.72\ cm$$

$$\delta_A = 7.3 - .04(H/X) = 7.3 - .4 = 6.9\ cm$$

$$\delta_B = 5.46 - .148(H/X) = 5.46 - 1.48 = 3.98\ cm$$

$$B^2_{c,A} = \frac{9.87}{(9.76+6.9)^2} = \frac{9.87}{(16.66)^2} = \frac{9.87}{277.52} = .0356\ cm^{-2}$$

$$B^2_{c,B} = \frac{9.87}{(12.72+3.98)^2} = \frac{9.87}{(16.70)^2} = \frac{9.87}{278.89} = .0354\ cm^{-2}$$

With a 5.2 Kg $U^{235}$ limit

$$V_A = \frac{5.2\ Kg}{2.25\ Kg/\ell} = 2.31\ \ell$$

$$r_A = \sqrt[3]{\frac{2310}{4.19}} = \sqrt[3]{552} = 8.22\ cm$$

$$B^2_{A,A} = \frac{9.87}{(8.22+6.9)^2} = \frac{9.87}{(15.12)^2} = \frac{9.87}{228.61} = .0432\ cm^{-2}$$

$$B^2_{A,B} = \frac{9.87}{(8.22+3.98)^2} = \frac{9.87}{(12.20)^2} = \frac{9.87}{148.84} = .0663\ cm^{-2}$$

The keff's are

$$k_{eff,A} = \frac{1+(28 \times .0356)}{1+(28 \times .0432)} = \frac{1.9968}{2.2096} = .904$$

$$k_{eff,B} = \frac{1+(28 \times .0354)}{1+(28 \times .0663)} = \frac{1.9912}{2.8564} = .697$$

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For U Metal - Water Mixture @ $10 \leq H/X \leq 20$

$V_{c,R} = 4.7\,\ell$, $V_{c,B} = 10\,\ell$, $\rho_c = 1.2\ Kg/\ell$ From Fig. 9, TID-7028

$$r_{c,R} = \sqrt[3]{\frac{3V_{c,R}}{4\pi}} = \sqrt[3]{\frac{4700}{4.19}} = \sqrt[3]{1120} = 10.4\,cm$$

$$r_{c,B} = \sqrt[3]{\frac{3V_{c,B}}{4\pi}} = \sqrt[3]{\frac{10000}{4.19}} = \sqrt[3]{2390} = 13.35\,cm$$

$$\delta_R = 7.3 - .04(H/X) = 6.5\,cm$$

$$\delta_B = 5.46 - .148(H/X) = 2.5\,cm$$

$$B^2_{c,R} = \frac{9.87}{(10.4+6.5)^2} = \frac{9.87}{(16.9)^2} = \frac{9.87}{285.61} = .0346\ cm^{-2}$$

$$B^2_{c,B} = \frac{9.87}{(13.35+2.5)^2} = \frac{9.87}{(15.85)^2} = \frac{9.87}{251.22} = .0393\ cm^{-2}$$

With a 3.6 Kg $U^{235}$ limit

$$V_A = \frac{3.6\ Kg}{1.2\ Kg/\ell} = 3\,\ell$$

$$r_A = \sqrt[3]{\frac{3000}{4.19}} = \sqrt[3]{718} = 8.95\,cm$$

$$B^2_{A,R} = \frac{9.87}{(8.95+6.5)^2} = \frac{9.87}{(15.45)^2} = \frac{9.87}{238.70} = .0413\ cm^{-2}$$

$$B^2_{A,B} = \frac{9.87}{(8.95+2.5)^2} = \frac{9.87}{(11.45)^2} = \frac{9.87}{131.10} = .0753\ cm^{-2}$$

The keffs are

$$keff_{,R} = \frac{1+(28 \times .0346)}{1+(28 \times .0413)} = \frac{1.9688}{2.1564} = .913$$

$$keff_{,B} = \frac{1+(28 \times .0393)}{1+(28 \times .0753)} = \frac{2.1004}{3.1084} = .675$$

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For a 3.6 liter volume @ H/D = 20

$V_A = 3.6\ \ell$

$$r_A = \sqrt[3]{\frac{3600}{4.19}} = \sqrt[3]{860} = 9.52\ cm$$

$$B^2_{A,R} = \frac{9.87}{(9.52+6.5)^2} = \frac{9.87}{(16.02)^2} = \frac{9.87}{256.64} = .0385\ cm^{-2}$$

$$B^2_{A,B} = \frac{9.87}{(9.52+2.5)^2} = \frac{9.87}{(12.02)^2} = \frac{9.87}{144.48} = .0683\ cm^{-2}.$$

The keff's are

$$k_{eff,R} = \frac{1+(28 \times .0346)}{1+(28 \times .0385)} = \frac{1.9688}{2.078} = .947$$

$$k_{eff,B} = \frac{1+(28 \times .0343)}{1+(28 \times .0683)} = \frac{2.1004}{2.9124} = .721$$

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| SECTION: 300, SUBPART: 309-II |
| NUCLEAR SAFETY EVALUATION — |
| TABLE 309-II BUCKLINGS & KEFF: |
| PAGE 7 OF 7 |
| APPROVED |
| ISSUED 2/6/70 |

## LIMITS FOR WET ZONES

| CONTROL PARAMETER | LIMIT |
|---|---|
| Safe Linear Density* | 20 grams U-235 per linear inch* |
| Safe Mass | 350 grams U-235 |

## LIMITS FOR DRY ZONES — Special Limits? § 306 1&2

| CONTROL PARAMETER | LIMIT |
|---|---|
| Safe Mass | 700 grams U-235 |
| Safe Linear Density* | 40 grams U-235 per linear inch* |

*NOTE: The safe linear density limits will be maintained irrespective of the arrangement of fuel pieces.

Applicable to:
1. Fuel fabrication operations for purpose of establishing safe piece count.
2. Any U-235 enrichment.
3. Full water reflection.
4. Wet zone limits for:
   Uranium solutions; alloys of U-Al& U-SS, up to 50 weight % U-235; and U-Zr up to 25 weight % U-235.
5. Dry zone limits for:
   Alloys of: U-Al and U-SS up to 50 weight % U-235; U-Zr up to 25 weight % U-235.

Specific conditions to be maintained:
1. Wet and dry zone controls specified in Subsection 306.2.4.1.

Source of Data:
1. Wet Zone - Safe Mass for Table 309-I
   Safe Linear Density NDEO-1050.
2. Dry Zone - Twice wet zone limits.

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| SECTION: 300 | SUBPART: 309 |
| Table III, Special Limits | |
| Page 1 of 1 | |
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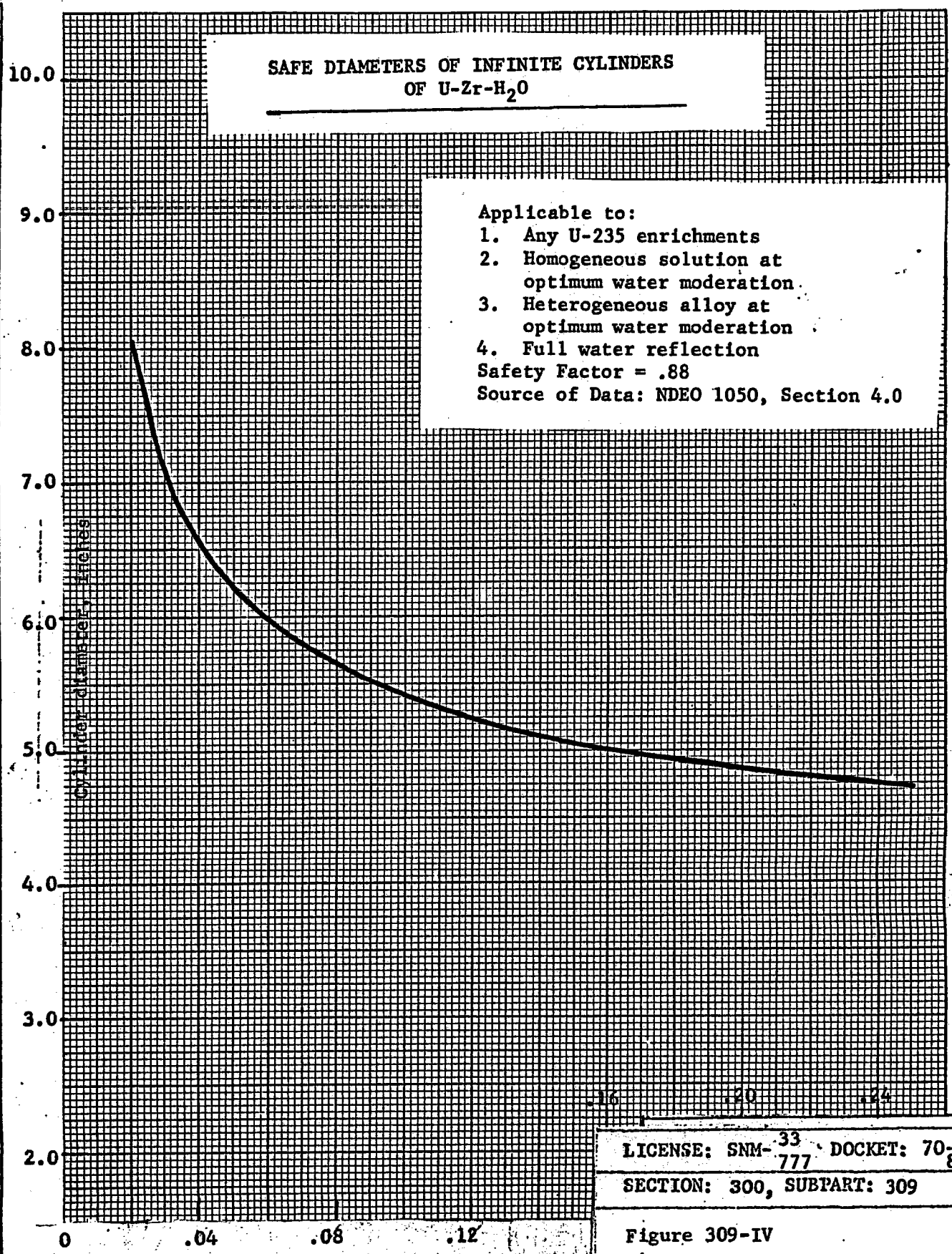


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SECTION: 300, SUBPART: 309

Figure 309-IV

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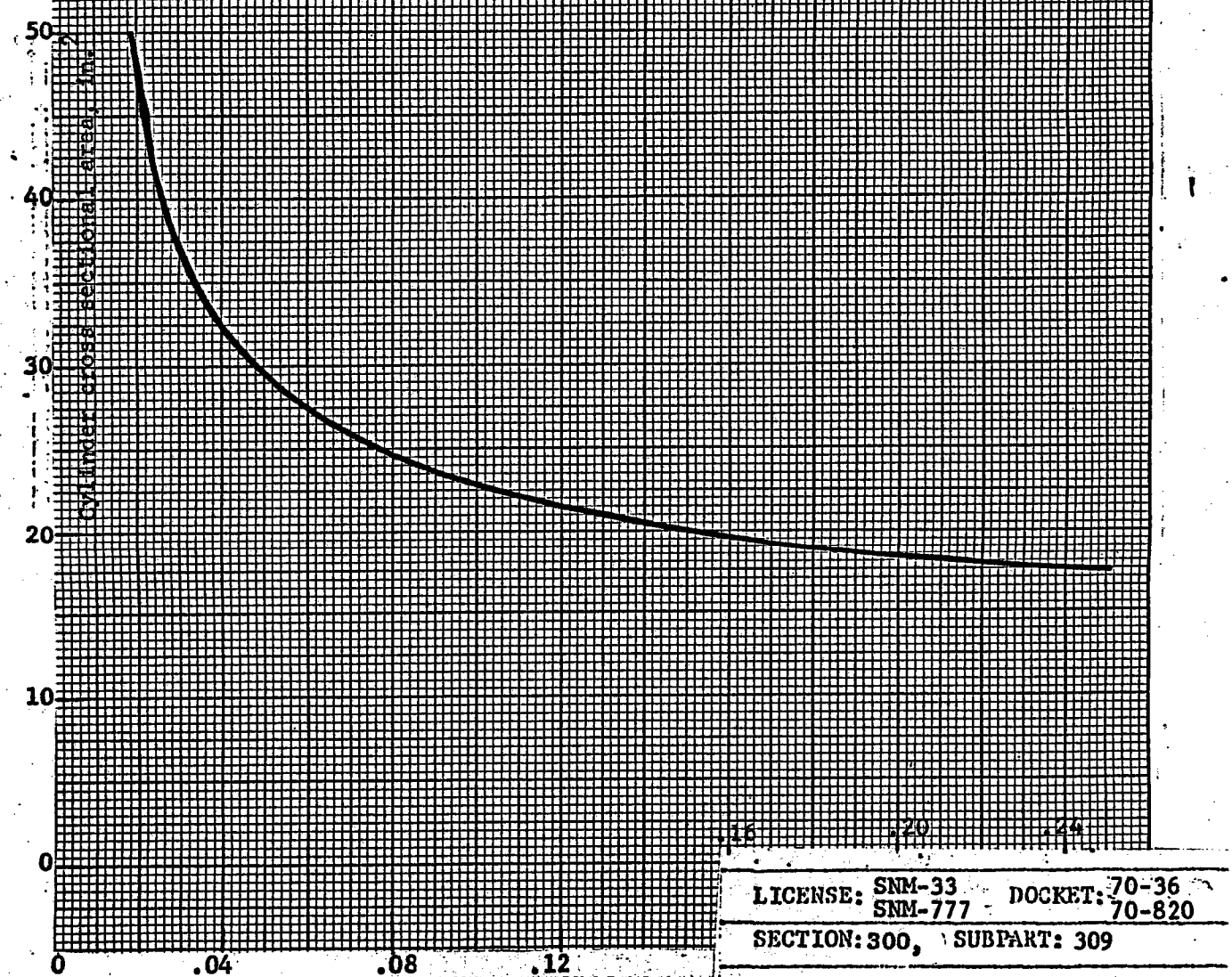
ISSUED: 2/6/70

## SAFE CROSS-SECTIONAL AREAS OF INFINITE CYLINDERS OF U-ZR-$H_2O$

Applicable to:
1. Any U-235 enrichments
2. Homogeneous solution at optimum water moderation
3. Heterogeneous alloy at optimum water moderation
4. Full water reflection

Safety Factor = .77

Source of Data: NDEO 1050, Section 4.0

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| Figure 309-V | |
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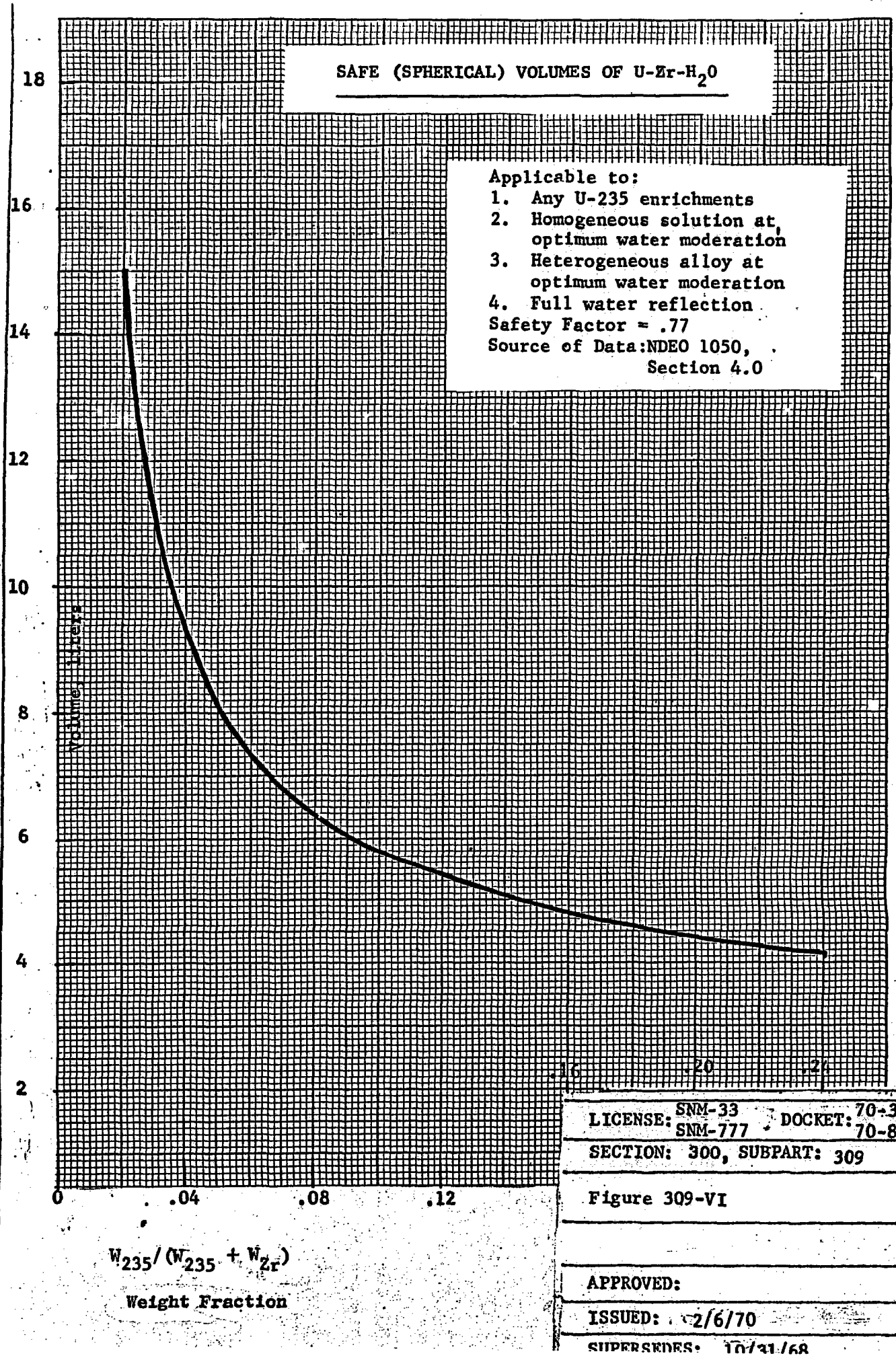


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| SECTION: 300, SUBPART: 309 | |
| Figure 309-VI | |
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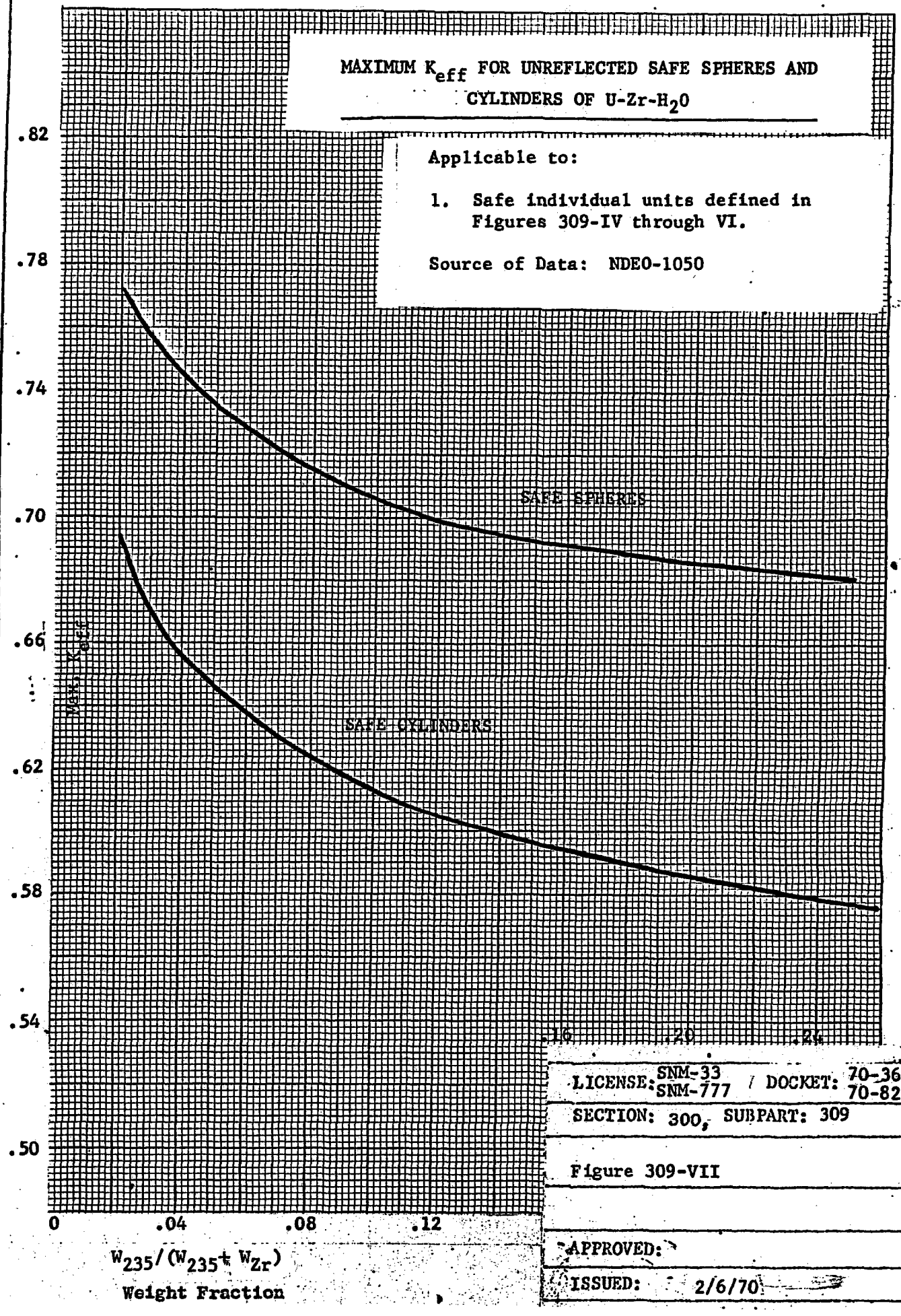
| SUPERSEDES: 10/31/68 | |

## MAXIMUM $K_{eff}$ FOR UNREFLECTED SAFE SPHERES AND CYLINDERS OF U-Zr-$H_2O$

Applicable to:

1. Safe individual units defined in Figures 309-IV through VI.

Source of Data: NDEO-1050

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SECTION: 300, SUBPART: 309

Figure 309-VII

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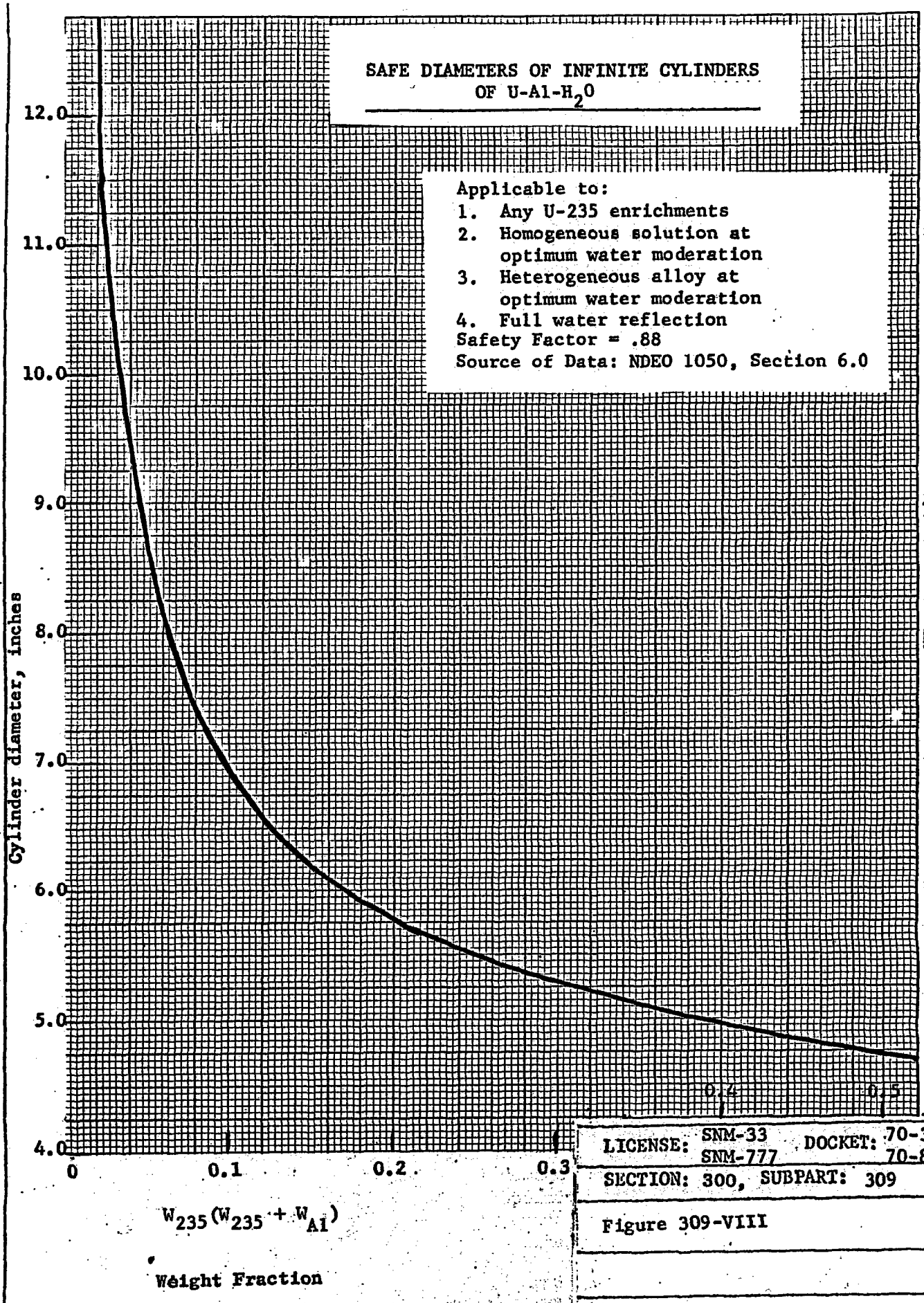


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| SECTION: 300, | SUBPART: 309 |
| Figure 309-VIII | |
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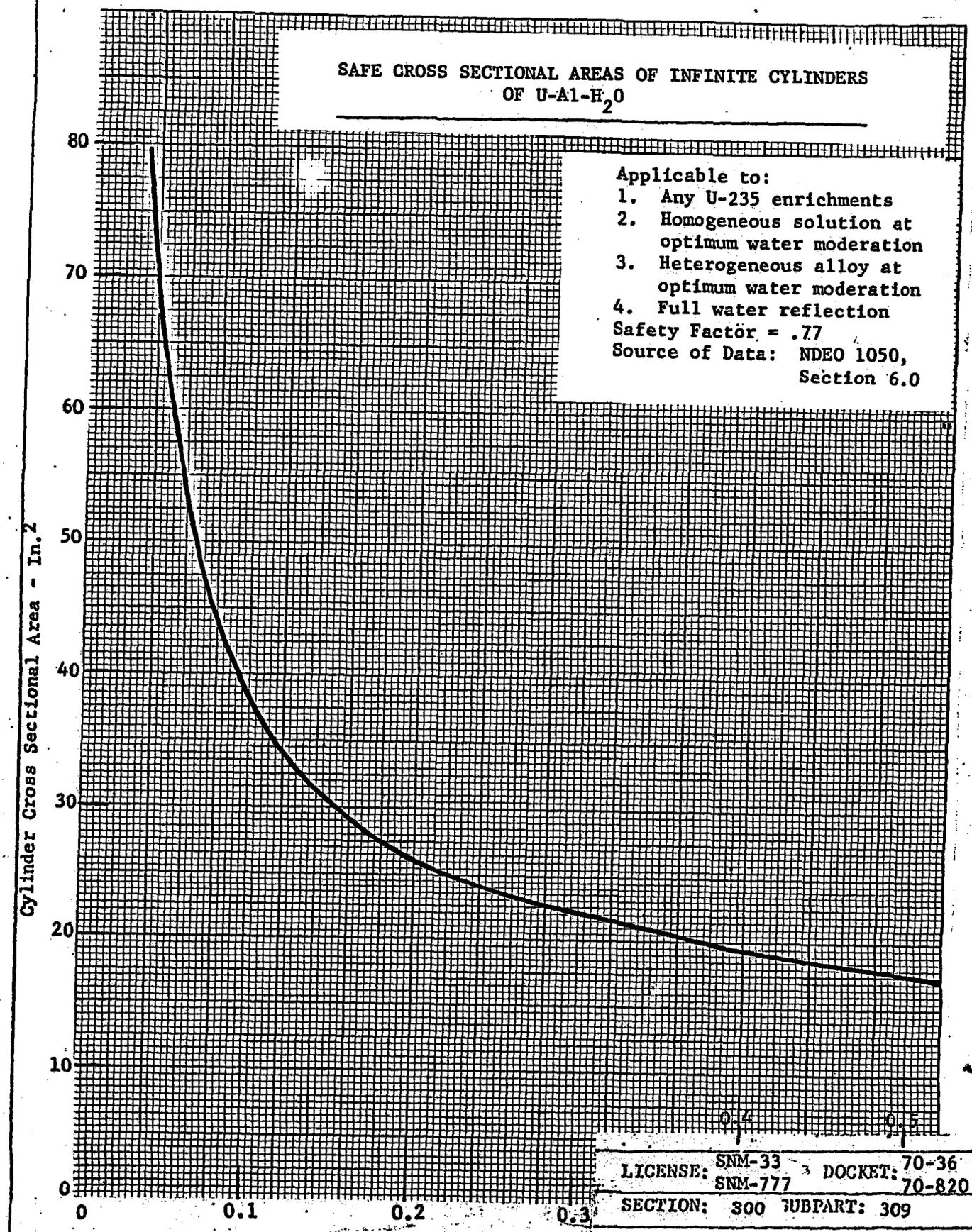


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SECTION: 300 SUBPART: 309

Figure 309-IX

APPROVED:

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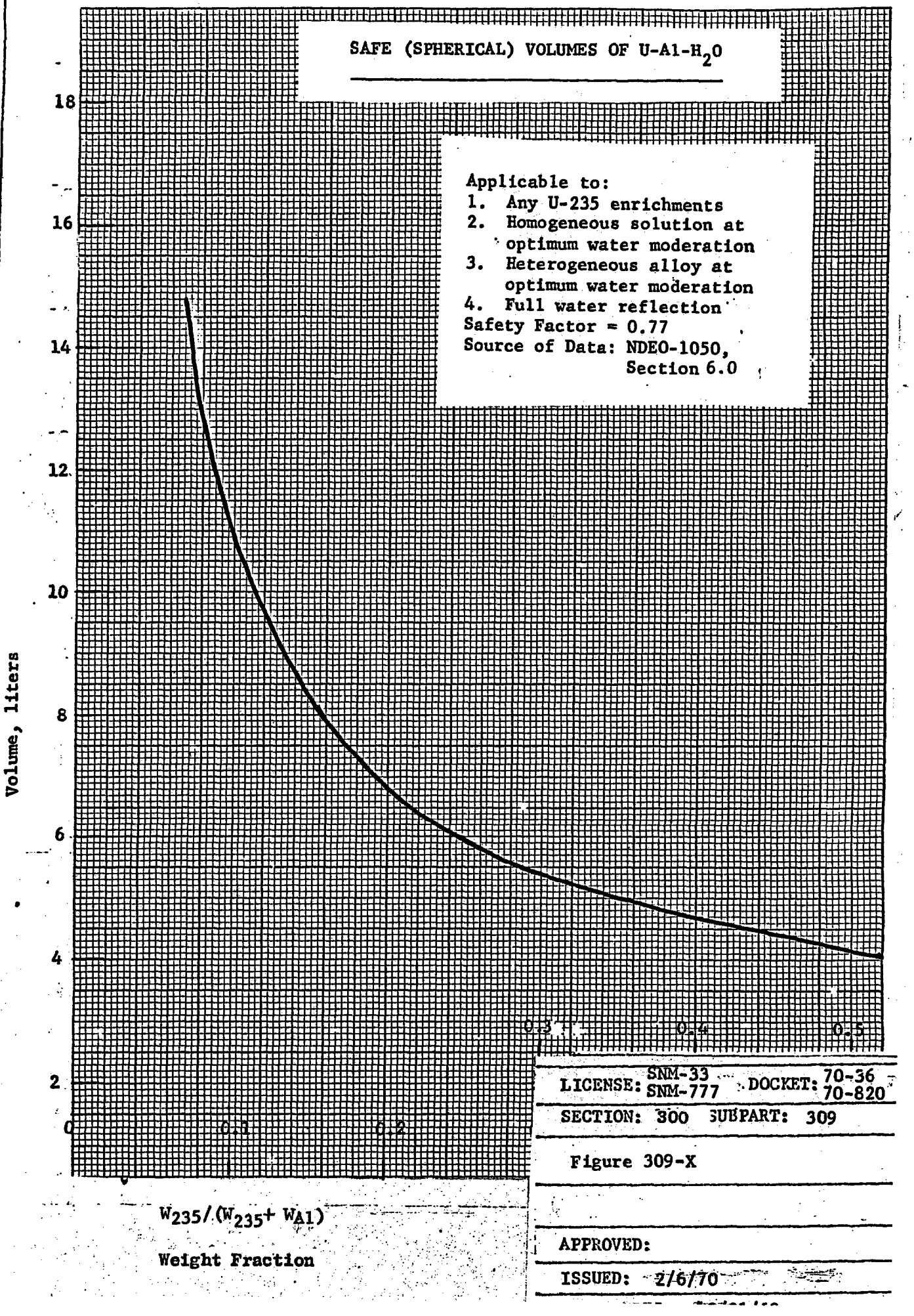


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SECTION: 300 SUBPART: 309

Figure 309-X

APPROVED:

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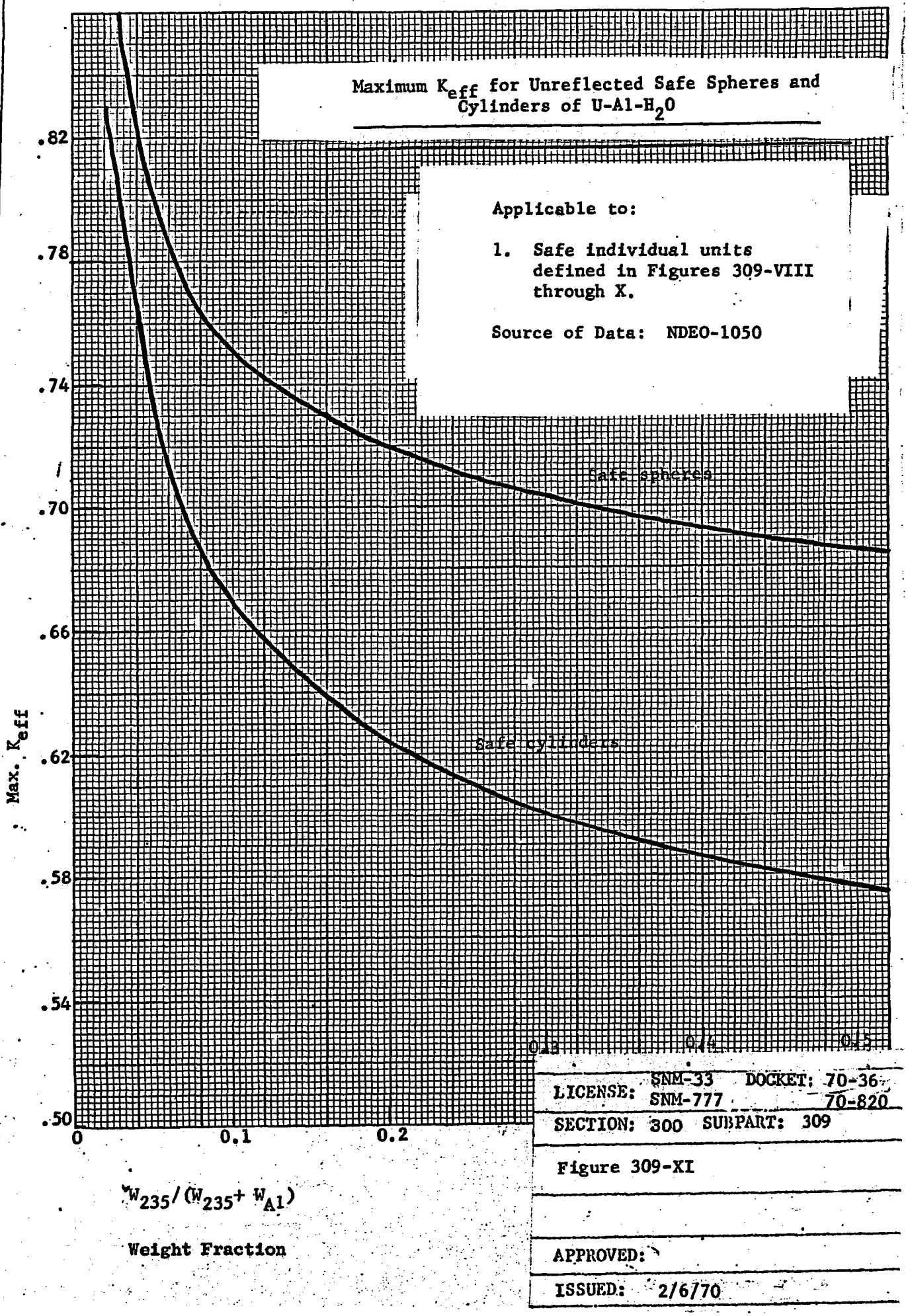


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| SECTION: 300 | SUBPART: 309 |
| Figure 309-XI | |
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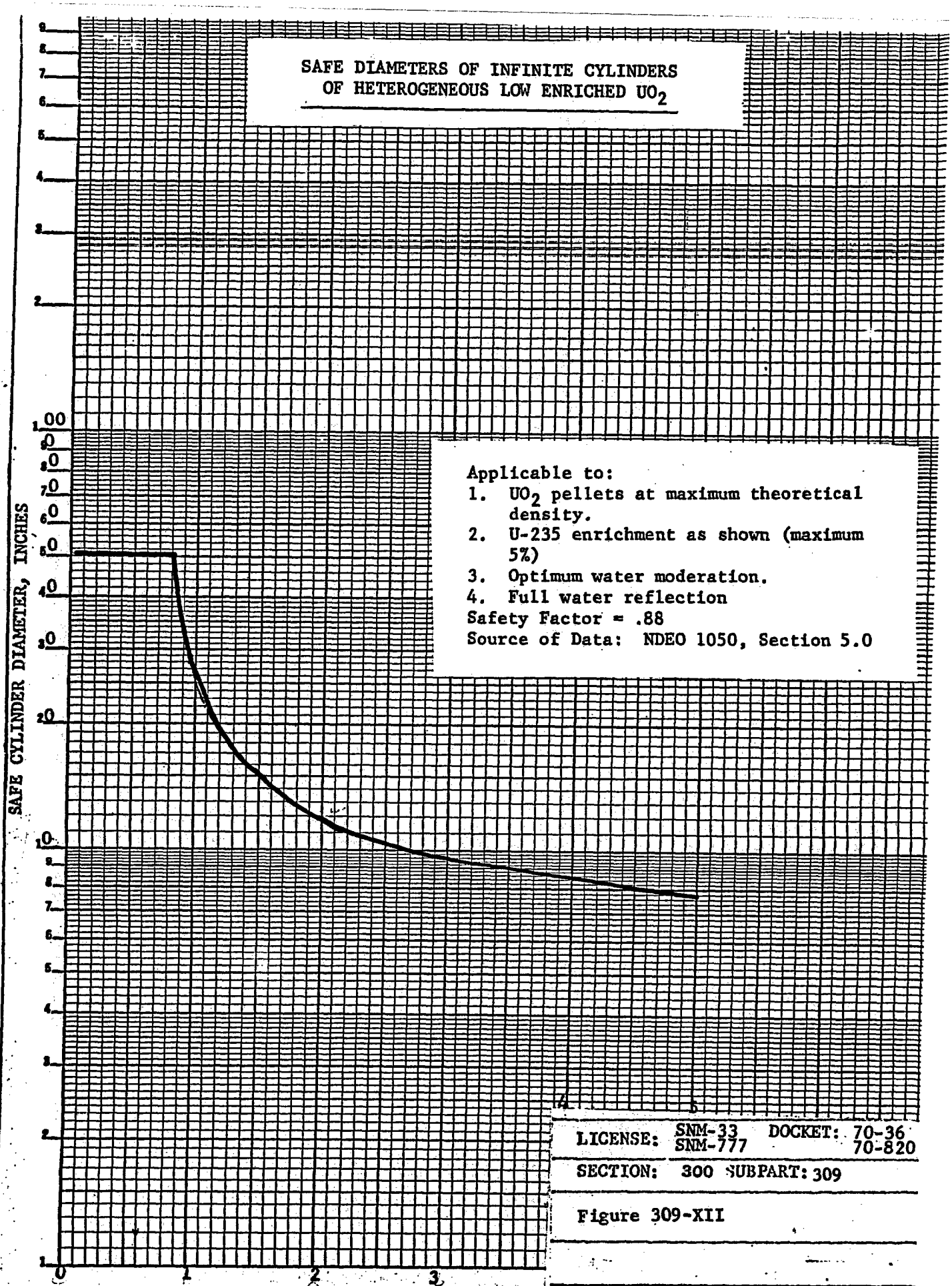
## SAFE DIAMETERS OF INFINITE CYLINDERS OF HETEROGENEOUS LOW ENRICHED $UO_2$




Applicable to:

1. $UO_2$ pellets at maximum theoretical density.
2. U-235 enrichment as shown (maximum 5%)
3. Optimum water moderation.
4. Full water reflection

Safety Factor = .88

Source of Data: NDEO 1050, Section 5.0

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| Figure 309-XII | |
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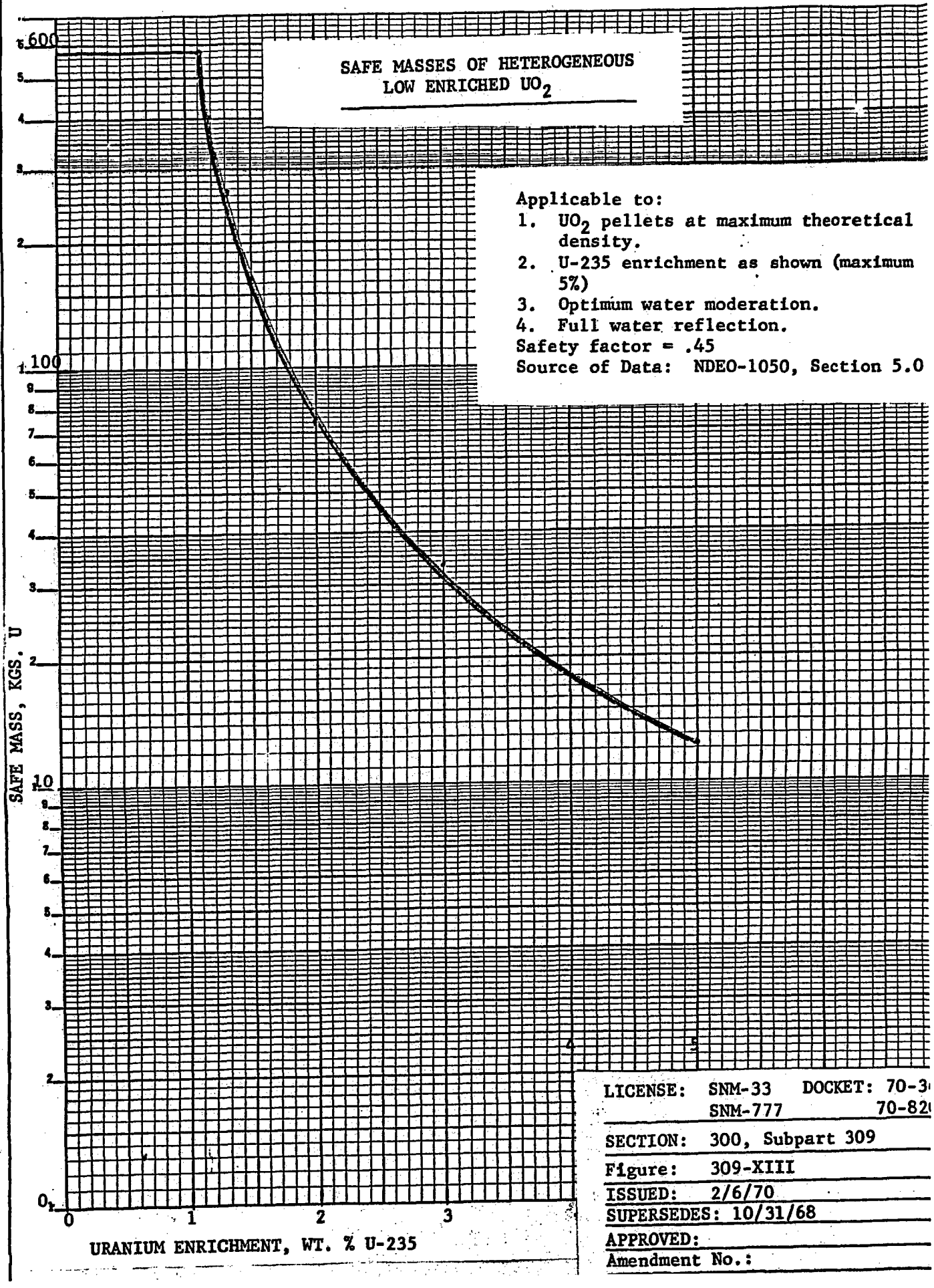


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SECTION: 300, Subpart 309

Figure: 309-XIII

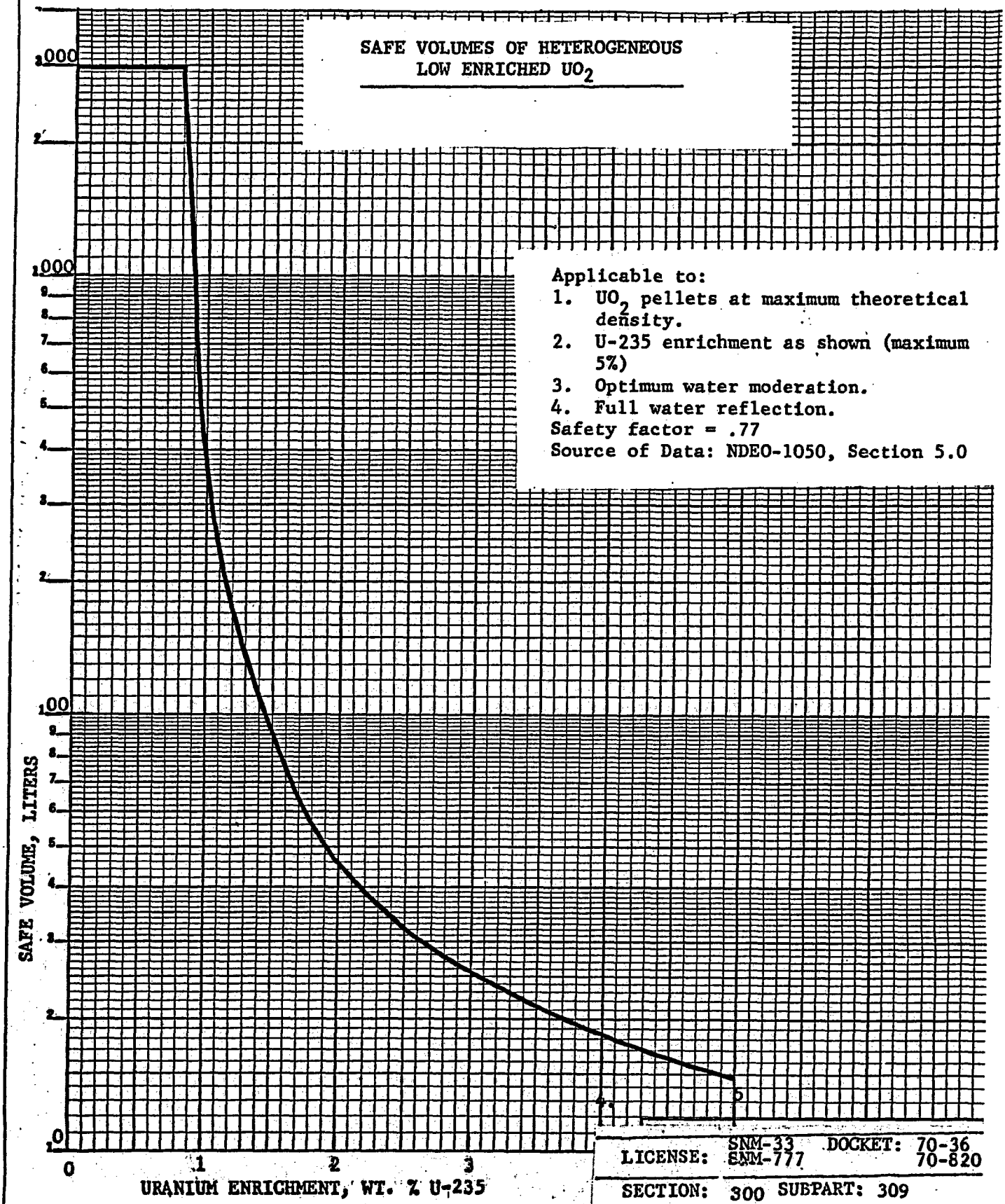
ISSUED: 2/6/70

SUPERSEDES: 10/31/68

APPROVED:

Amendment No.:

## SAFE VOLUMES OF HETEROGENEOUS LOW ENRICHED $UO_2$




Applicable to:

1. $UO_2$ pellets at maximum theoretical density.
2. U-235 enrichment as shown (maximum 5%)
3. Optimum water moderation.
4. Full water reflection.

Safety factor = .77

Source of Data: NDEO-1050, Section 5.0

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| Figure 309-XIV | |
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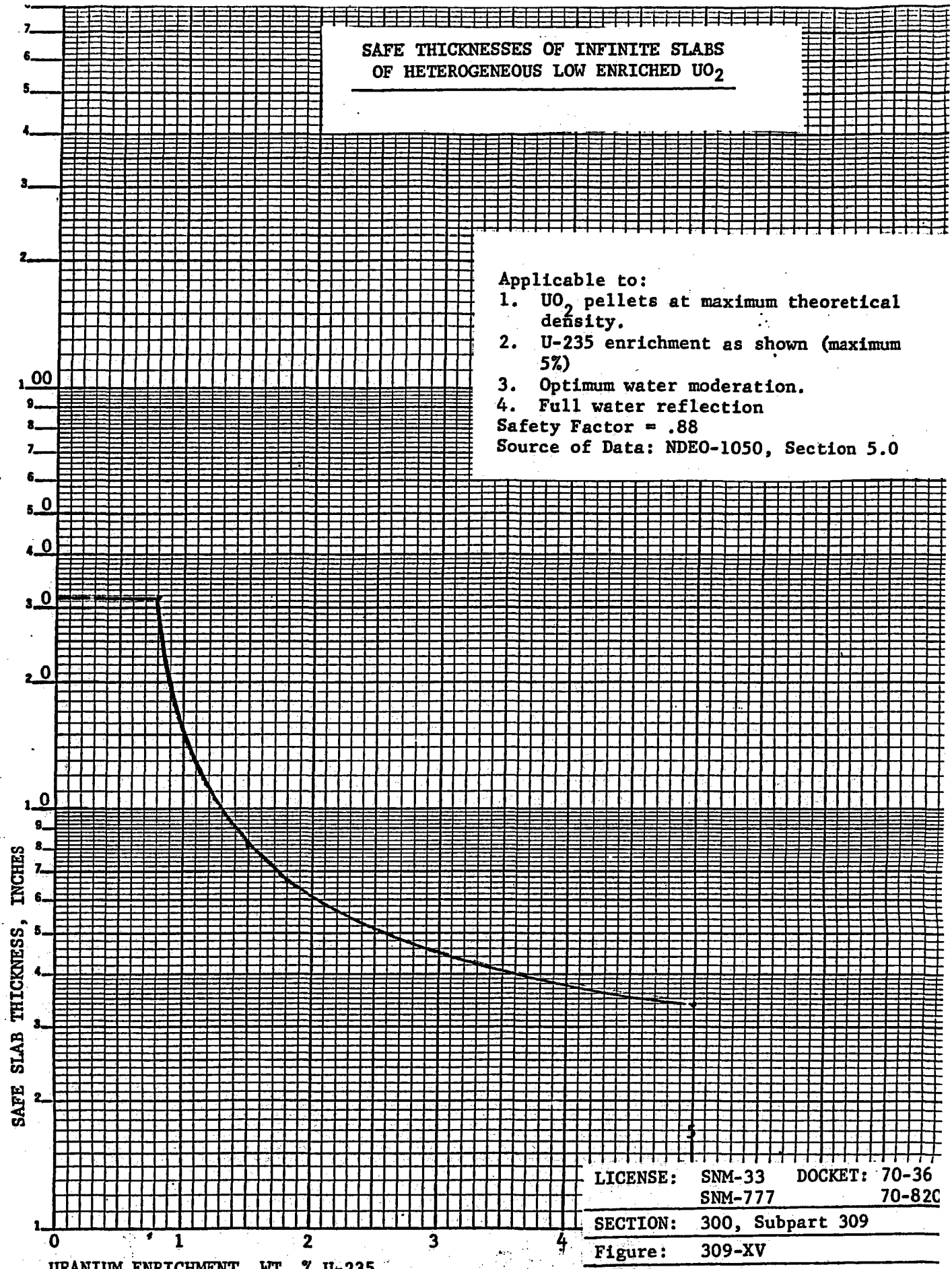
## SAFE THICKNESSES OF INFINITE SLABS OF HETEROGENEOUS LOW ENRICHED $UO_2$

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| Figure: | 309-XV | | |
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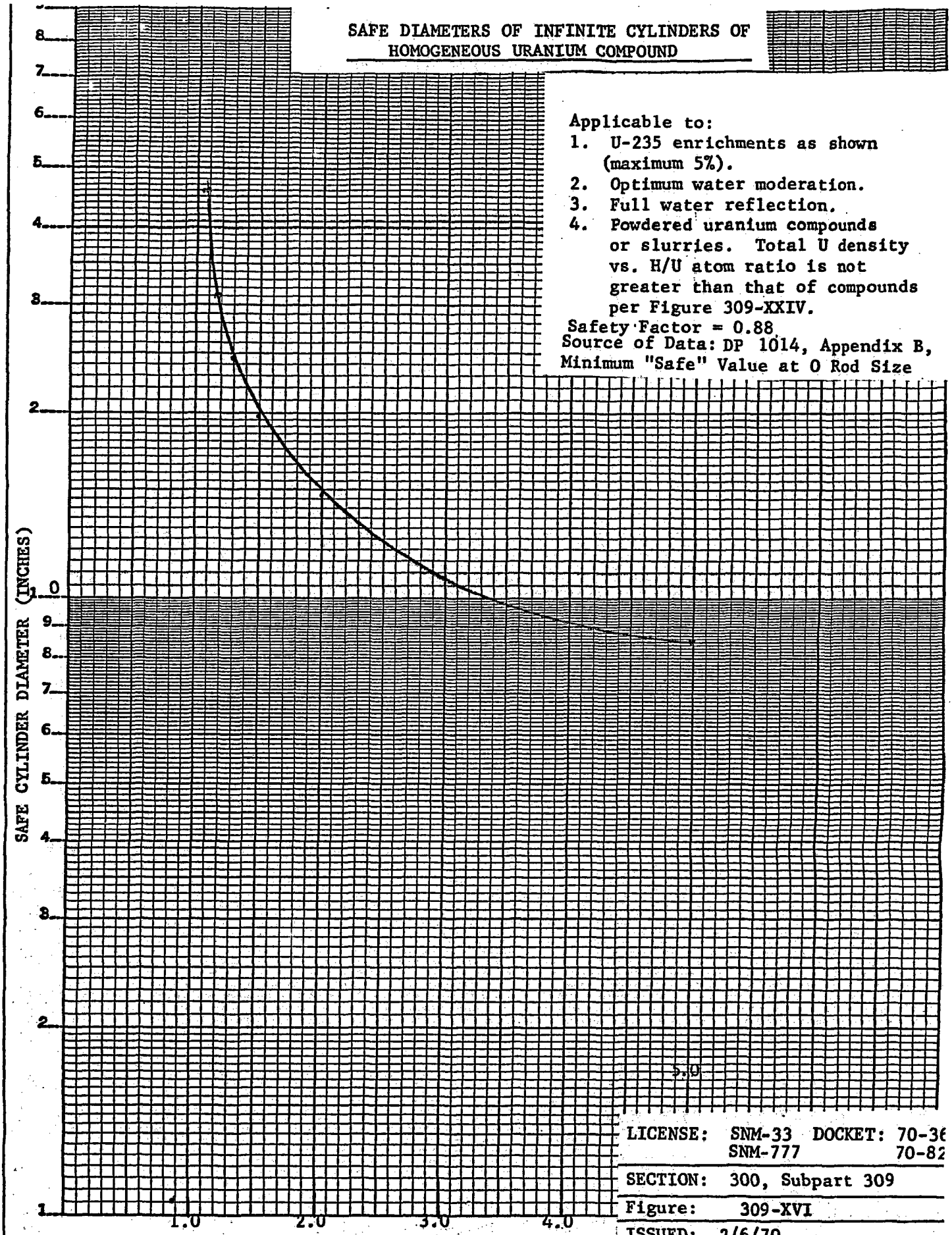
## SAFE DIAMETERS OF INFINITE CYLINDERS OF HOMOGENEOUS URANIUM COMPOUND




Applicable to:

1. U-235 enrichments as shown (maximum 5%).
2. Optimum water moderation.
3. Full water reflection.
4. Powdered uranium compounds or slurries. Total U density vs. H/U atom ratio is not greater than that of compounds per Figure 309-XXIV.

Safety Factor = 0.88

Source of Data: DP 1014, Appendix B, Minimum "Safe" Value at 0 Rod Size

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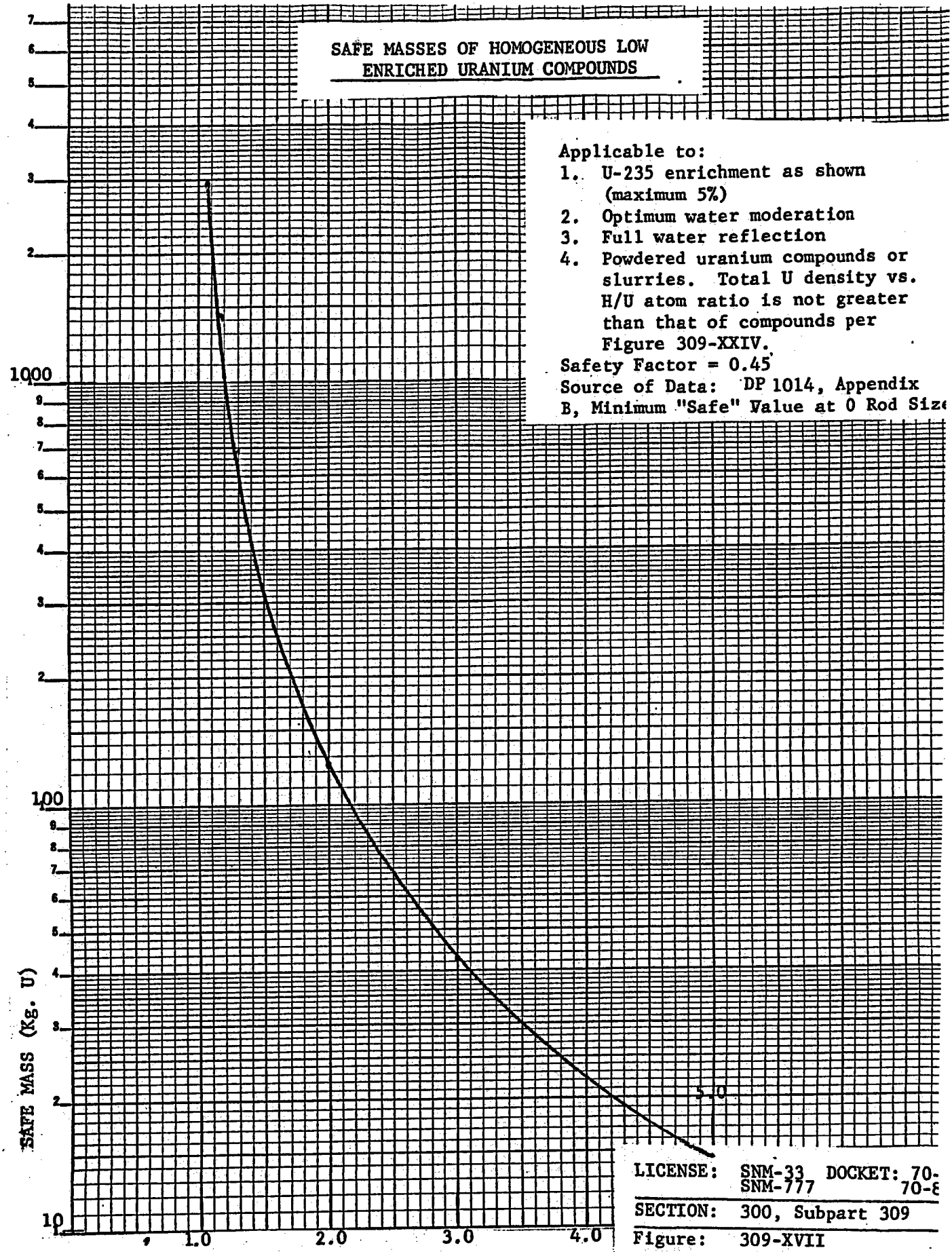
## SAFE MASSES OF HOMOGENEOUS LOW ENRICHED URANIUM COMPOUNDS

Applicable to:

1. U-235 enrichment as shown (maximum 5%)
2. Optimum water moderation
3. Full water reflection
4. Powdered uranium compounds or slurries. Total U density vs. H/U atom ratio is not greater than that of compounds per Figure 309-XXIV.

Safety Factor = 0.45

Source of Data: DP 1014, Appendix B, Minimum "Safe" Value at 0 Rod Size

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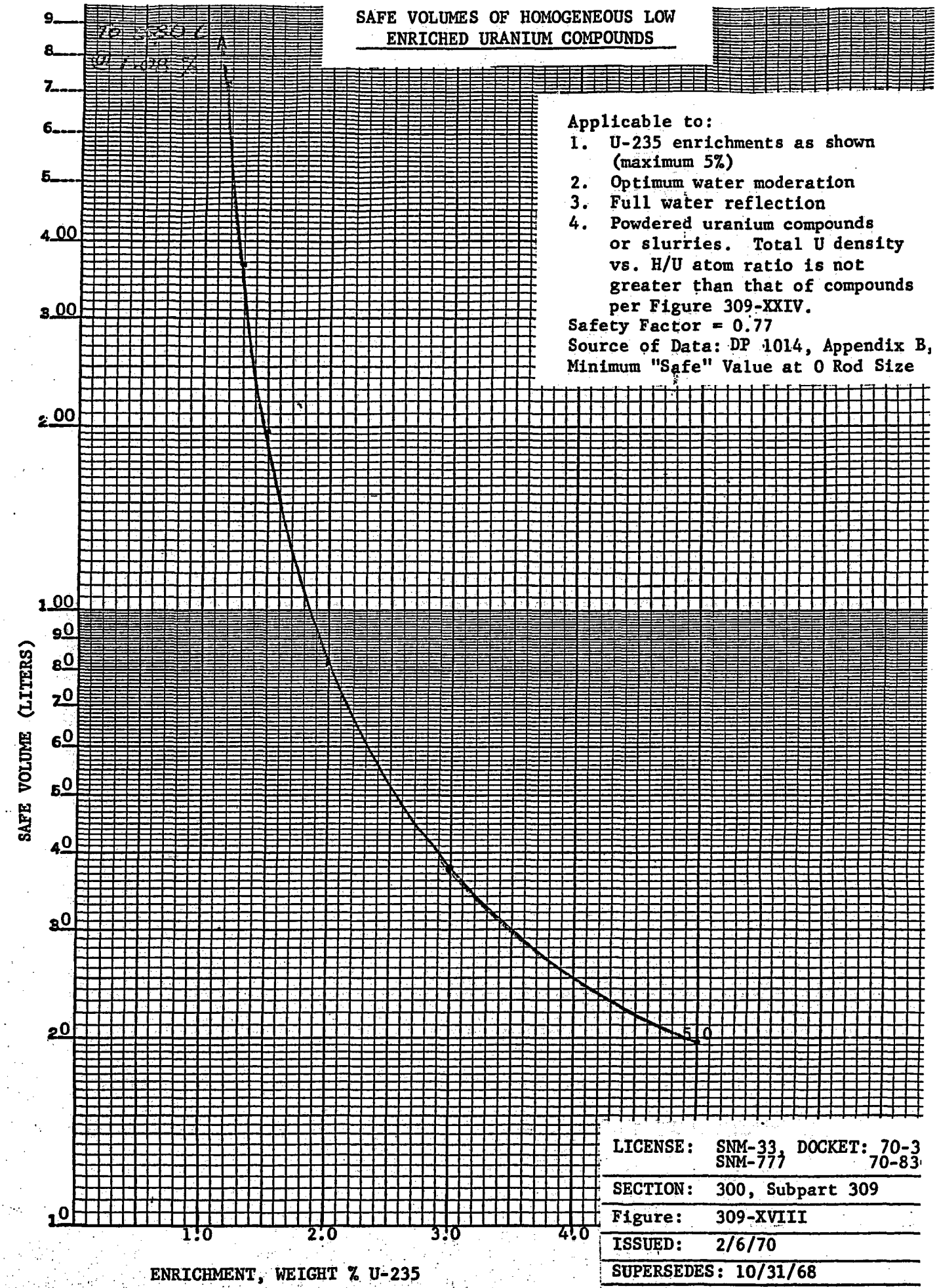
# SAFE VOLUMES OF HOMOGENEOUS LOW ENRICHED URANIUM COMPOUNDS




Applicable to:

1. U-235 enrichments as shown (maximum 5%)
2. Optimum water moderation
3. Full water reflection
4. Powdered uranium compounds or slurries. Total U density vs. H/U atom ratio is not greater than that of compounds per Figure 309-XXIV.

Safety Factor = 0.77

Source of Data: DP 1014, Appendix B, Minimum "Safe" Value at 0 Rod Size

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| Figure: | 309-XVIII |
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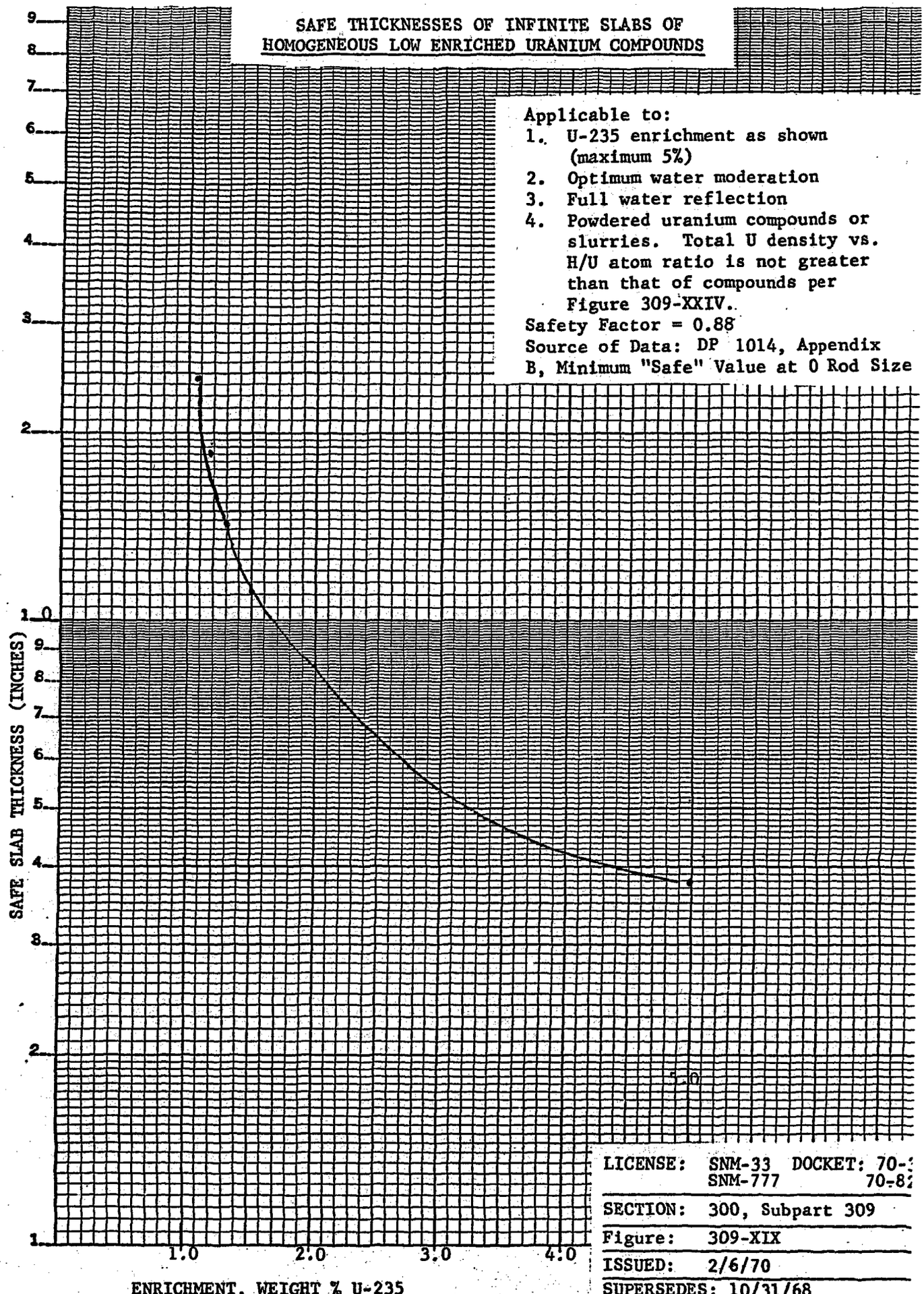
## SAFE THICKNESSES OF INFINITE SLABS OF HOMOGENEOUS LOW ENRICHED URANIUM COMPOUNDS

Applicable to:

1. U-235 enrichment as shown (maximum 5%)
2. Optimum water moderation
3. Full water reflection
4. Powdered uranium compounds or slurries. Total U density vs. H/U atom ratio is not greater than that of compounds per Figure 309-XXIV.

Safety Factor = 0.88

Source of Data: DP 1014, Appendix B, Minimum "Safe" Value at 0 Rod Size

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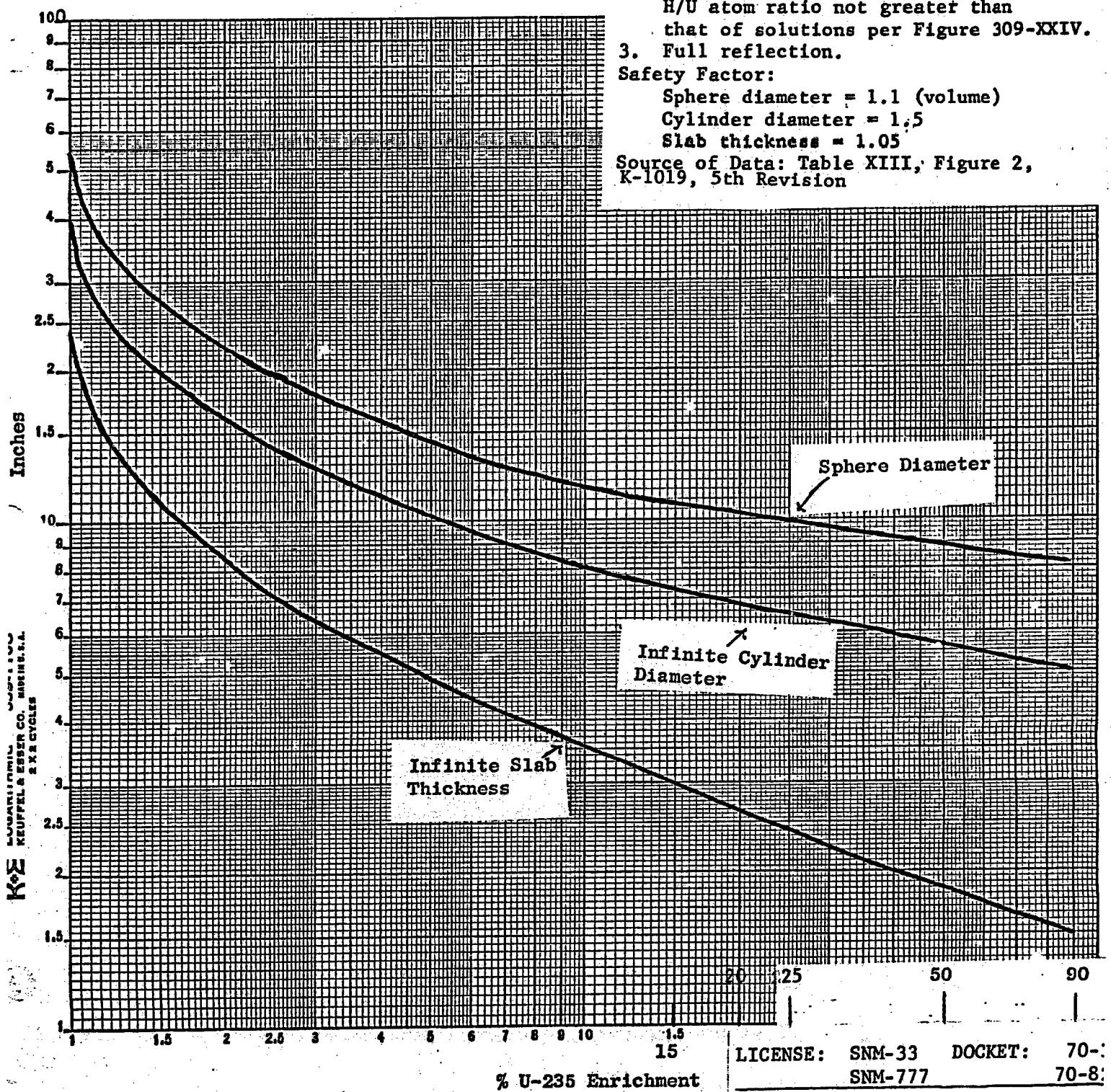
SAFE GEOMETRIC VARIABLES

FOR SPECIFIED U-235 ENRICHMENTS

Applicable to:
1. All U-235 enrichments.
2. Solutions with total U density vs. H/U atom ratio not greater than that of solutions per Figure 309-XXIV.
3. Full reflection.

Safety Factor:
Sphere diameter = 1.1 (volume)
Cylinder diameter = 1.5
Slab thickness = 1.05

Source of Data: Table XIII, Figure 2, K-1019, 5th Revision

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| Figure: | 309-XX | | |
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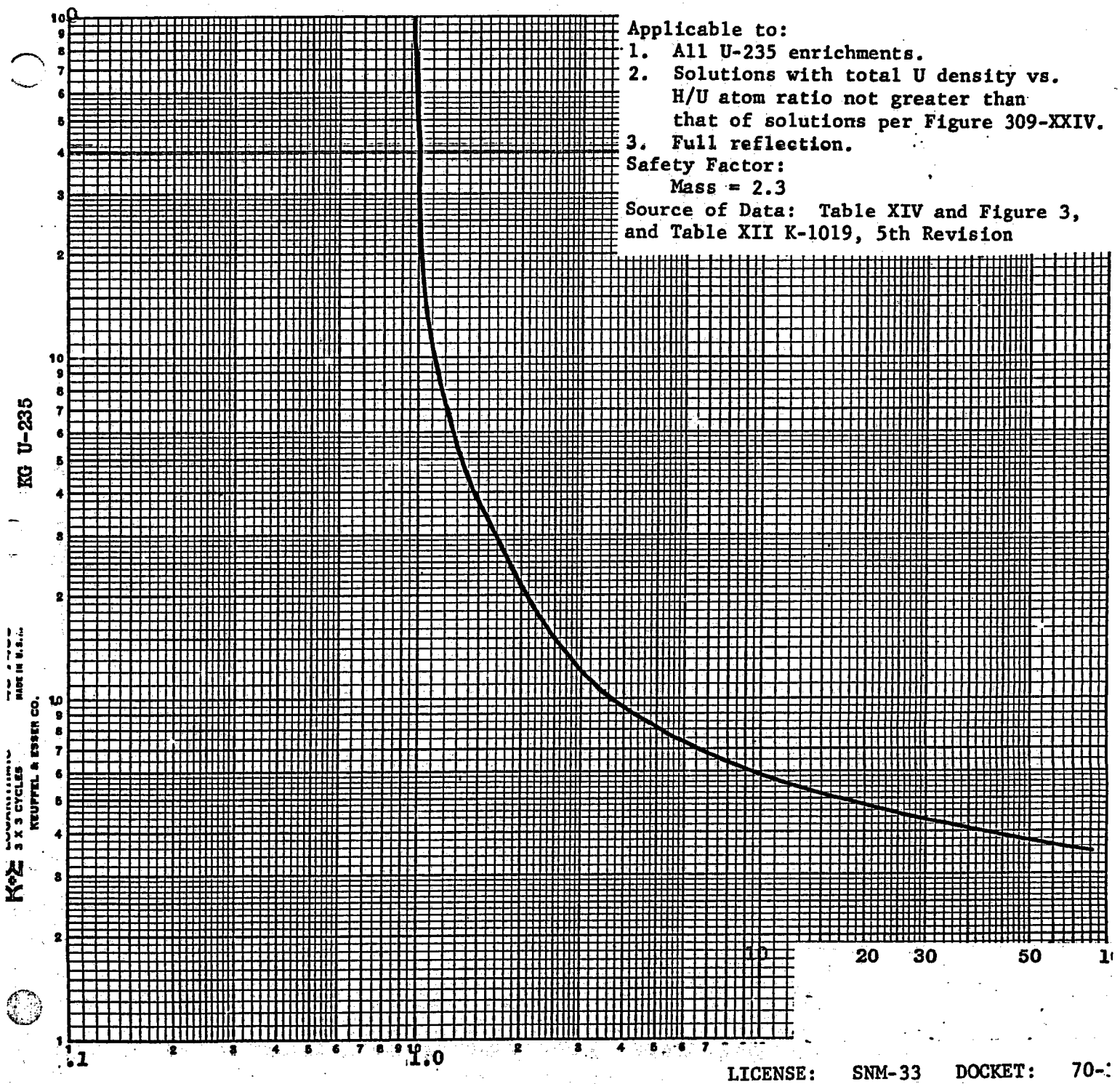
SAFE MASSES FOR

SPECIFIED U-235 ENRICHMENT




Applicable to:

1. All U-235 enrichments.
2. Solutions with total U density vs. H/U atom ratio not greater than that of solutions per Figure 309-XXIV.
3. Full reflection.

Safety Factor:

Mass = 2.3

Source of Data: Table XIV and Figure 3, and Table XII K-1019, 5th Revision

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| Figure: | 309-XXI | | |
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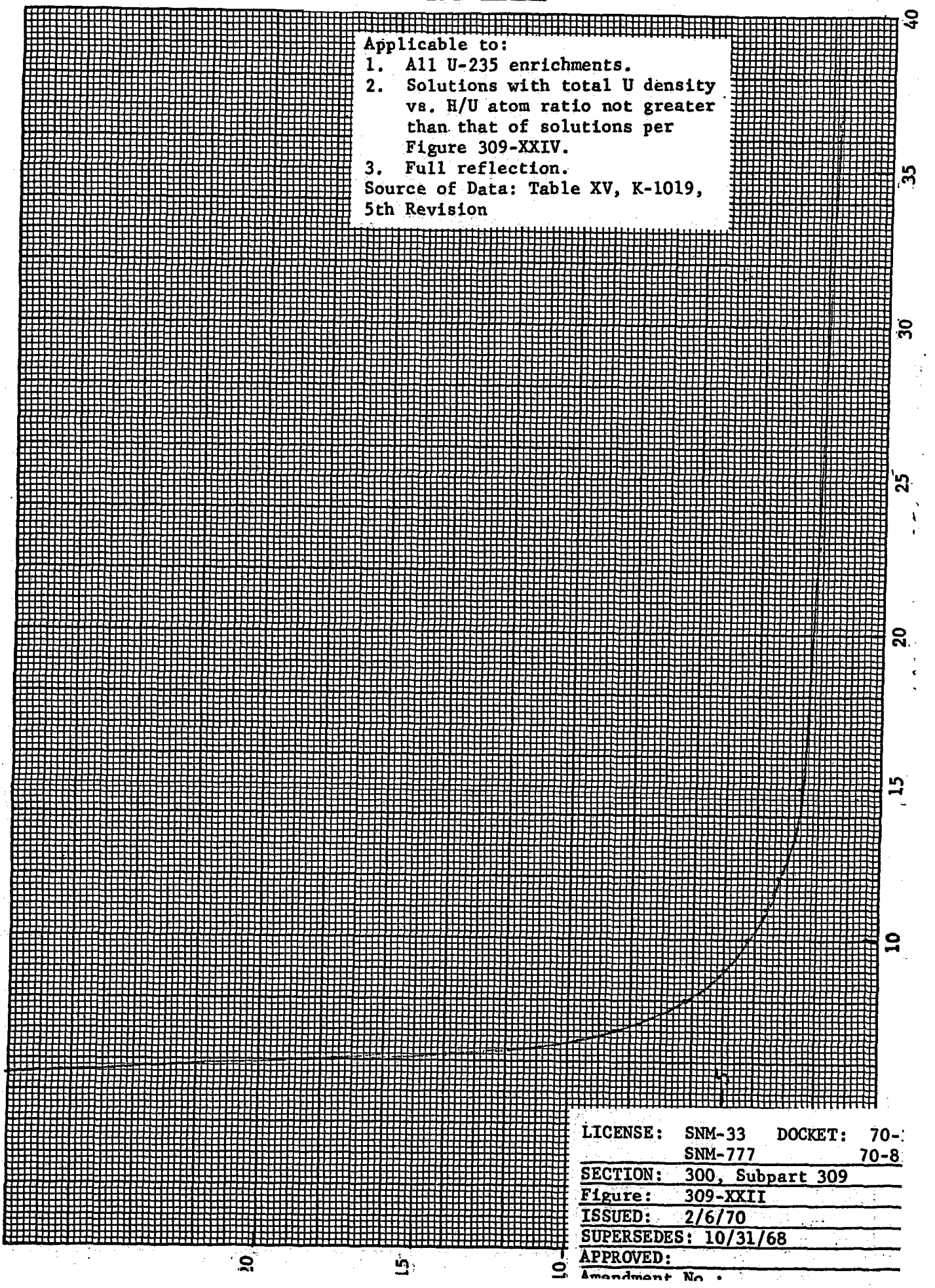
SAFE CYLINDERS

ANY U-235 ENRICHMENT

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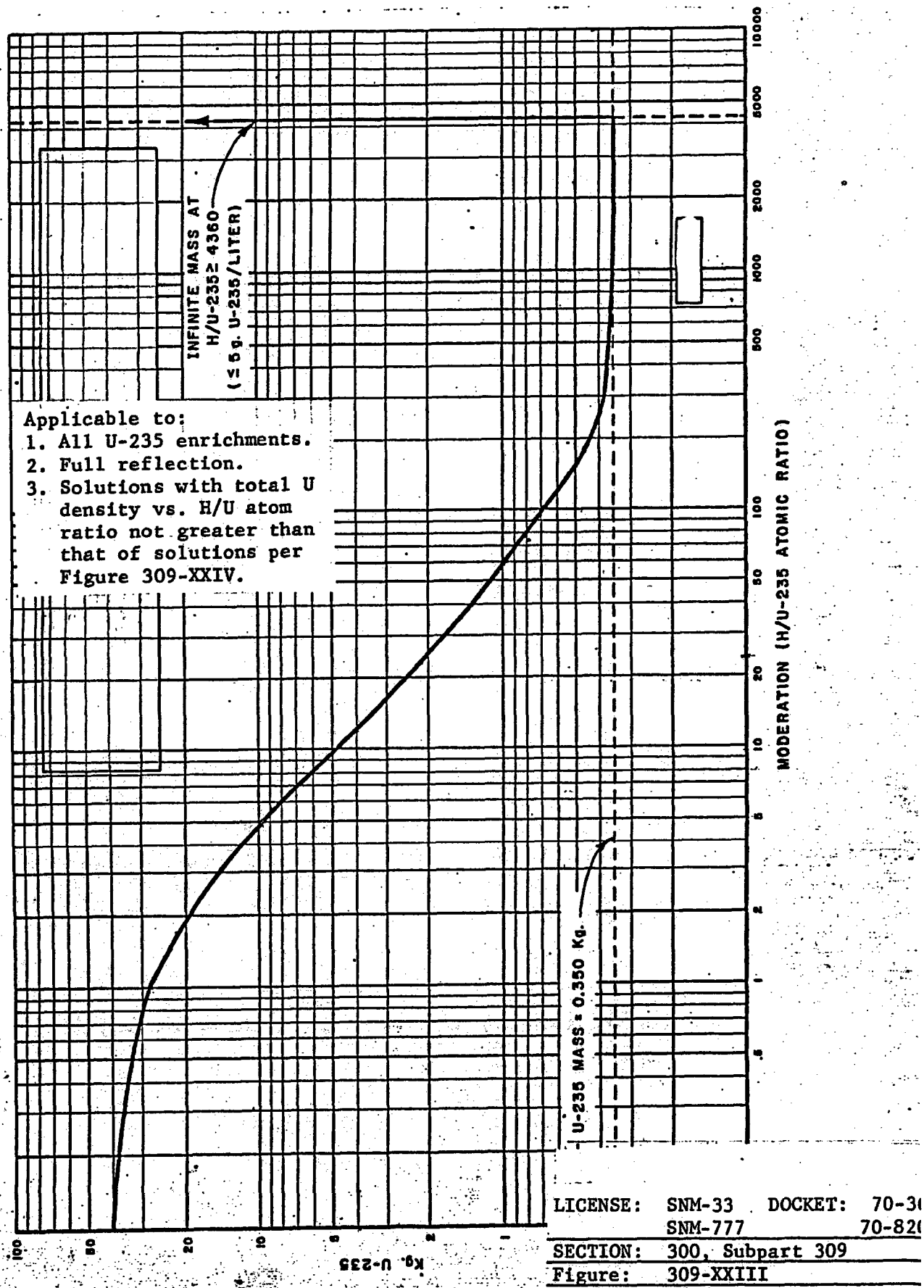
K+E 10 X 10 TO ½ INCH 46 1323
7 X 10 INCHES MADE IN U.S.A.
KEUFFEL & ESSER CO.

NUCLEARLY SAFE MASSES FOR SPECIFIED MODERATION

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| Figure: | 309-XXIII | | |
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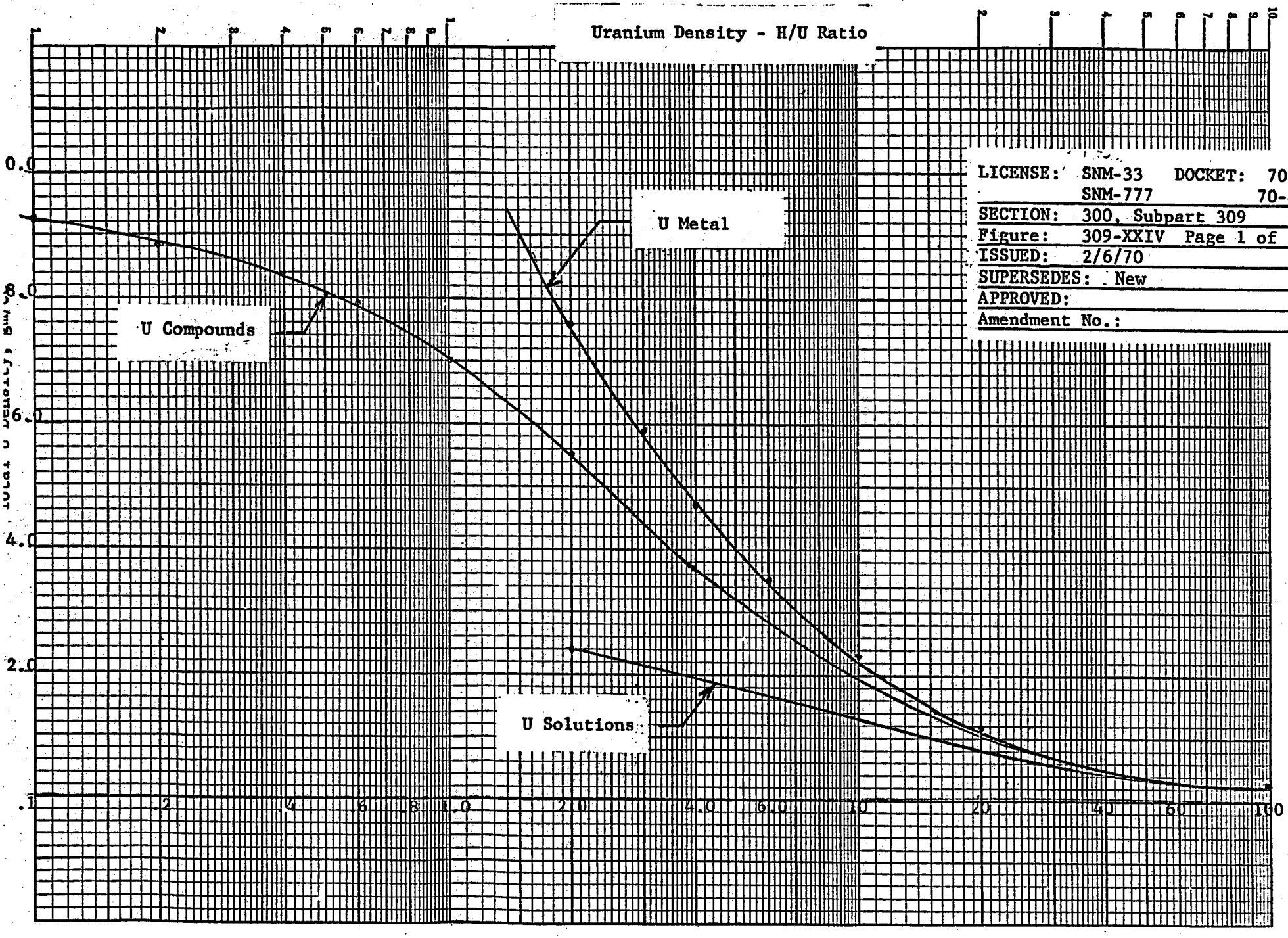
KEUFFEL & ESSER CO. MADE IN U.S.A.
3 CYCLES X 70 DIVISIONS

# Uranium Density - H/U Ratio




H/U Atom Ratio

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| Figure: | 309-XXIV Page 1 of 2 | | |
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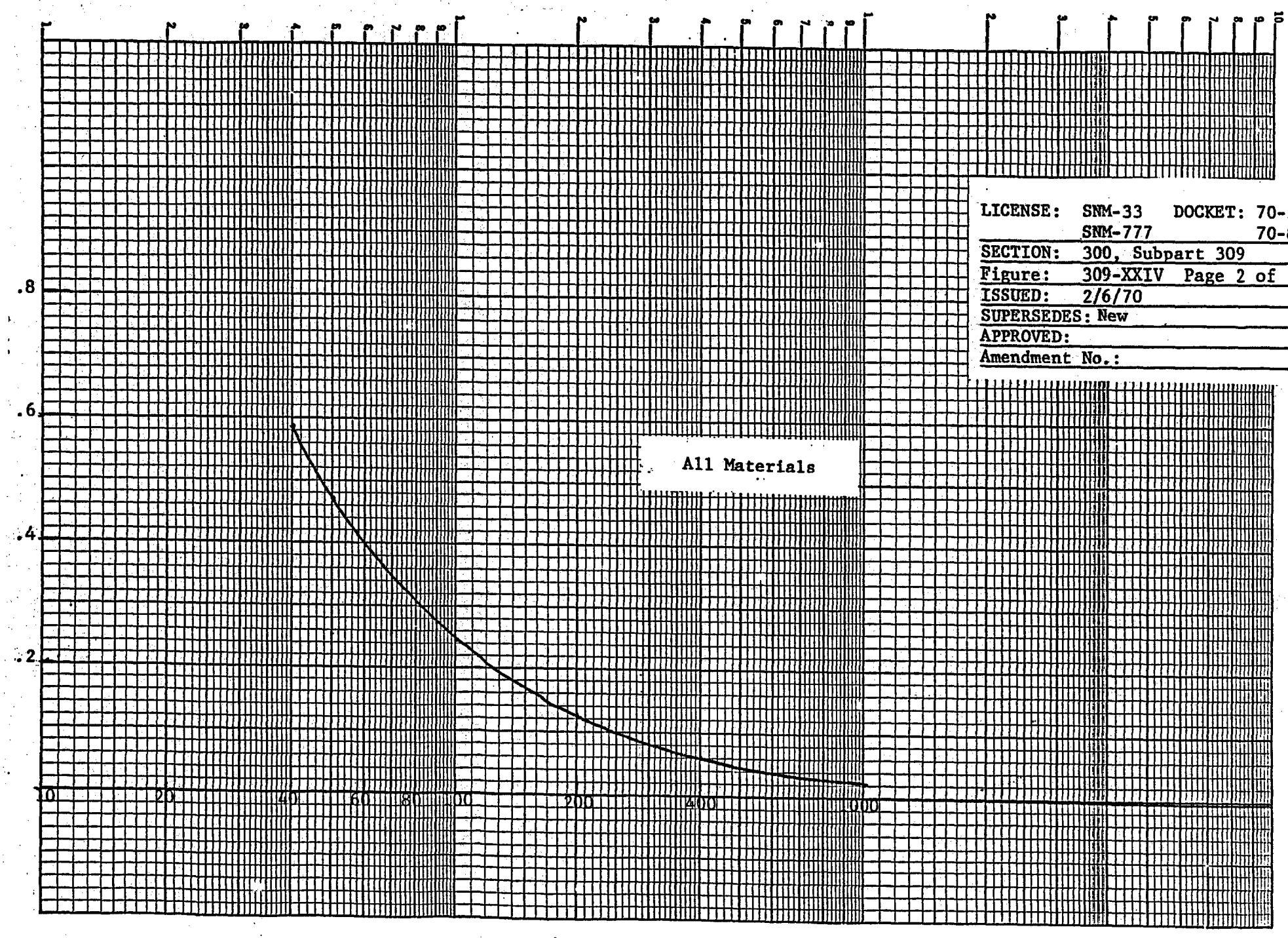


H/U Atom Ratio

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| Figure: | 309-XXIV Page 2 of 2 | | |
| ISSUED: | 2/6/70 | | |
| SUPERSEDES: | New | | |
| APPROVED: | | | |
| Amendment No.: | | | |

## Uranium Density vs. H/U Ratio

### I. General

Nuclear criticality safety parameters are a function of uranium density and the degree of moderation of neutrons within the uranium system. Unless specifically stated otherwise, hydrogen in water is the most efficient moderator for which nuclear safety parameters have been established. The degree of moderation is measured by the ratio of the hydrogen to uranium atoms ($\frac{H}{U}$) of the system considered. Generally throughout the literature, the U-235 isotope is used as the reference for defining the density and moderation effects; however individual publications are usually confined to one enrichment level and use of U-235 as the reference is convenient. At the Commercial Products Division plants, uranium of all enrichments is processed. To standardize nuclear safety parameters, it is more convenient to work in terms of total uranium as it applies to density and degree of moderation; therefore, unless specifically stated otherwise, total uranium will be used in all discussions and data involved in the relationship between density and H/U ratio.

### II. Material Types

The types of material processed at Commercial Products Division facilities falls into four physical categories:

Solutions
U Metal and compound water mixtures
U Metal
U alloys with aluminum, zirconium and stainless steel

The relationship of uranium density and H/U ratio for water mixtures of metal and compounds and solution is shown on Figure 1. Based on this figure, there are three standard curves for uranium density and H/U ratio. These are for:

Uranium Metal
Uranium Compounds
Uranium Solutions

#### A. Uranium Compounds

The density vs. H/U relationship of $UO_2$ is the maximum for uranium compounds processed by CPD.

Uranium Compounds (continued)

Water mixtures of uranium compounds are subject to the most variation of the density and H/U relationship which is caused by the variation of the maximum theoretical density of the individual compounds and weight fraction of uranium in the compound. The typical compounds processed by CPD are $UO_2$, $U_3O_8$, $UF_4$, $UO_2F_2$, $UO_4$, and ADU (ammonium diuranate). Of these $UO_2$ has the maximum density for a given H/U level. This is illustrated on Figure 1 and Table 1. For purposes of establishing standard nuclear criticality safety parameters for compounds, the total uranium density vs. H/U relationship for $UO_2$ is established as the upper limit for which such standards are applicable.

The maximum theoretical density has been used to develop the data of Table 1 and Figure 1. These maximum densities can be achieved only by special ceramic processing such as the process of making $UO_2$ pellets from $UO_2$ powder. These densities are not possible from the chemical process of converting $UF_6$ to $UO_2$ or recovery of uranium from scrap and residues; these processes result in a $UO_2$ powder having a maximum bulk density of 4 kg U/liter corresponding to an H/U of 4. Accordingly, standards for compounds in homogeneous form are applicable to this level. Standards for compounds having bulk densities in excess of 4 kg U/liter are applicalbe to densities up to the theoretical maximum.

Experiments to determine effect on density when water is added to dry $UO_2$ powder were unsuccessful. The water would not "wet" the powder without special agitation causing a reduction of the density.

This is typical of plant applications. The oxide does not achieve the upper density levels except when it has been dried. The maximum density limit is just that, i.e., dry powder introduced into a wet process is either added to the liquid in incremental quantities (which action reduces its density) or the liquid is added to the powder (which action requires mechanical mixing) also reducing the density.

B. <u>Uranium Solutions</u>

The density vs. H/U relationship for $UO_2F_2$ solutions is the maximum for uranium solutions processed at CPD.

Historically, $UO_2F_2$ solutions have been used in experimental measurements of critical parameters because it permitted the highest concentration and lowest non-fissioning absorption cross section of solutions generally processed. A review of solutions processed at CPD confirms this; the most typical solution being uranyl nitrate. These data are also illustrated in Figure 1 and Table 1.

For purposes of establishing standard nuclear criticality safety parameters for solutions the density vs. H/U relationship of $UO_2F_2$ solutions is established as the upper limits for which the standards will be applicable.

C. Uranium Metal

The relationship of metal density versus H/U shown on Figure 1 is that obtained from Table 1, TID 7028. This is the maximum possible density for any form of uranium. Safe parameter standards for uranium metal are based on this relationship.

D. Alloys

The effect of the relationship of uranium density to H/U ratio for alloys has been included in the special calculations (NDEO 1050) performed for the alloys. Safe parameters are reported as a function of the U-235 content of the alloy.

III. Development of the Density vs. H/U Ratio

A. Compounds and Water Mixtures

The relationship for uranium compounds and water mixture is based on the maximum theoretical density of the compound and the volume additive mixtures of the compound with water. Specifically H/U ratios were calculated from the formula:

$$\frac{H}{U} = \frac{\text{Weight } H_2O \times 2 \frac{\text{atom H}}{\text{mole } H_2O} \times 238 \frac{\text{weight U}}{\text{mole U}}}{18 \frac{\text{weight } H_2O}{\text{moles } H_2O} \times \text{weight U} \times 1 \frac{\text{atom U}}{\text{mole U}}}$$

$$= 26.45 \left( \frac{\text{weight } H_2O}{\text{weight U}} \right)$$

The data plotted on Figure 1 and calculated by the above formula is tabulated on Table 1.

B. Metal Water Mixtures

The relationship for uranium metal and water mixtures has been obtained from the data of Figures 1 through 4, TID-7016, Rev. 1.

C. Solutions

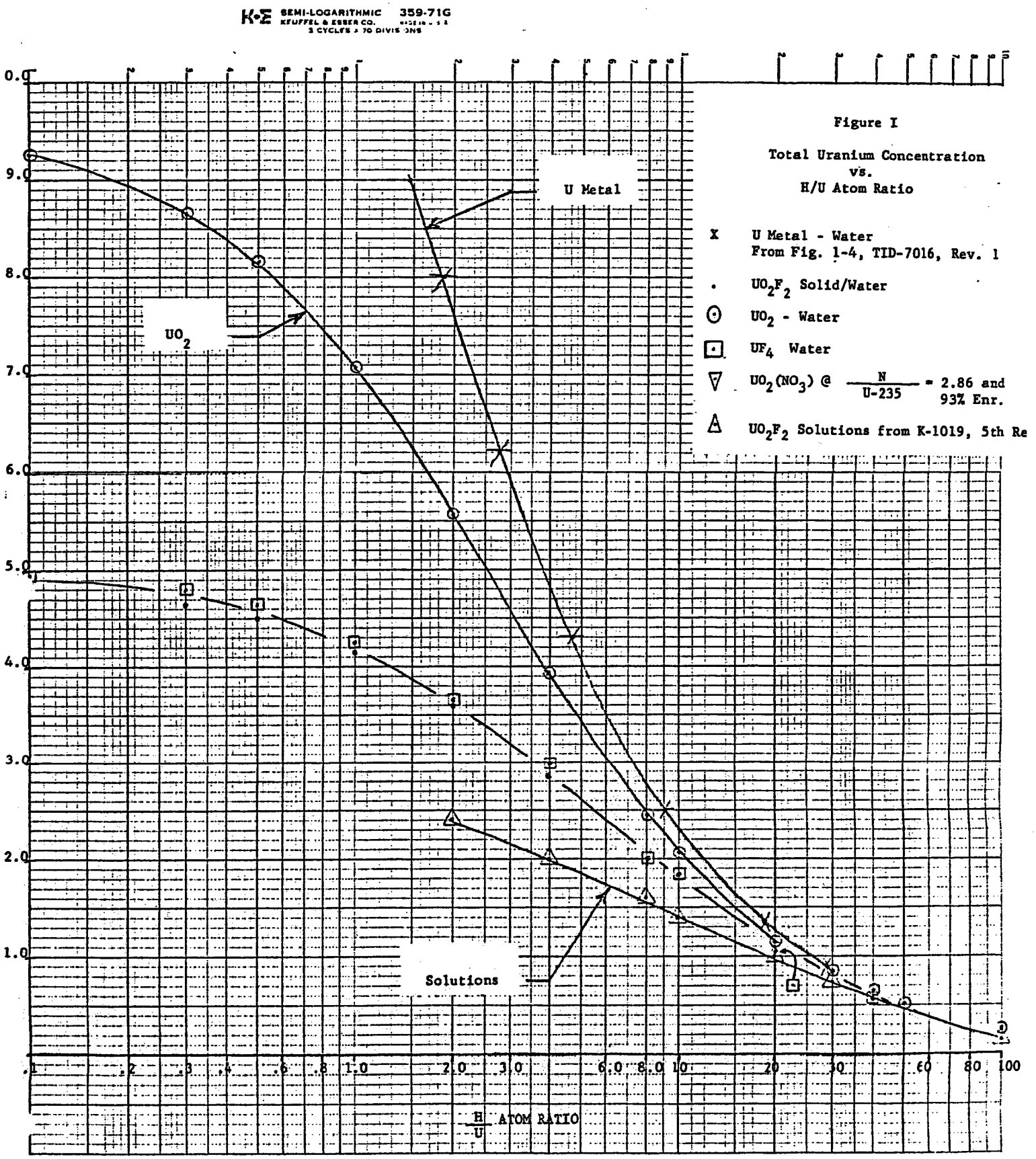
The data for solutions has been obtained from Figure 1, K-1019, 5th Revision.

In examining Figure 1 it can be seen that all of the curves converge into a single curve at an H/U ratio of approximately 20 and a density of approximately 1. This single curve follows that of the solution curve for H/U ratios in excess of 20. Unless specifically stated otherwise, safe parameters for all systems with H/U ratios greater than 20 will therefore follow those of solutions.

TABLE 1

| H/U | Density, grams U/cc | | | | | |
|---|---|---|---|---|---|---|
| | | | | | $(UO_2)$ $(NO_3)$ @ N/U-235 = 2.86 93% Enriched * | |
| | $UO_2$ | $UF_4$ | $UO_2F_2$ Solid | $UO_2F_2$ Solution(2) | H/U | Density |
| 0 | 9.66(1) | 5.07(1) | 4.93(1) | | 58 | .387 |
| .1 | 9.24 | 4.99 | 4.84 | | 93 | .247 |
| .3 | 8.69 | 4.80 | 4.64 | | 186 | .135 |
| .5 | 8.16 | 4.64 | 4.50 | | 279 | .092 |
| 1 | 7.06 | 4.26 | 4.15 | | | |
| 2 | 5.59 | 3.67 | 3.6 | 2.4 | | |
| 4 | 3.92 | 2.99 | 2.82 | 2.0 | | |
| 8 | 2.46 | 2.00 | 1.98 | 1.6 | | |
| 10 | 2.08 | 1.74 | 1.72 | 1.4 | | |
| 20 | 1.16 | 1.04 | 1.04 | 1.0 | | |
| 30 | .81 | | .748 | .75 | | |
| 40 | .66 | .585 | .582 | | | |
| 50 | .50 | .479 | .477 | | | |
| 100 | .258 | .251 | .25 | .25 | | |
| 200 | | | | .12 | | |
| 600 | | | | .02 | | |
| 1000 | | | | .01 | | |

(1) Based on maximum theoretical density of compound reported by Katz & Rabinowitch, The Chemistry of Uranium, First Edition.
(2) Figure 1, K-1019, 5th Rev.
* Figure 2.5 NDEO-1050


K+E SEMI-LOGARITHMIC 359-71G
KEUFFEL & ESSER CO.
3 CYCLES X 70 DIVISIONS
Figure I
Total Uranium Concentration
vs.
H/U Atom Ratio
X U Metal - Water
From Fig. 1-4, TID-7016, Rev. 1
• $UO_2F_2$ Solid/Water
$UO_2$ - Water
$UF_4$ Water
$UO_2(NO_3)$ @ $\frac{N}{U\text{-}235}$ = 2.86 and 93% Enr.
$UO_2F_2$ Solutions from K-1019, 5th Re
U Metal
$UO_2$
Solutions
0.0
9.0
8.0
7.0
6.0
5.0
4.0
3.0
2.0
1.0
.1
.2
.3
.4
.6
.8
1.0
2.0
3.0
6.0
8.0
10
20
30
60
80
100
$\frac{H}{U}$ ATOM RATIO

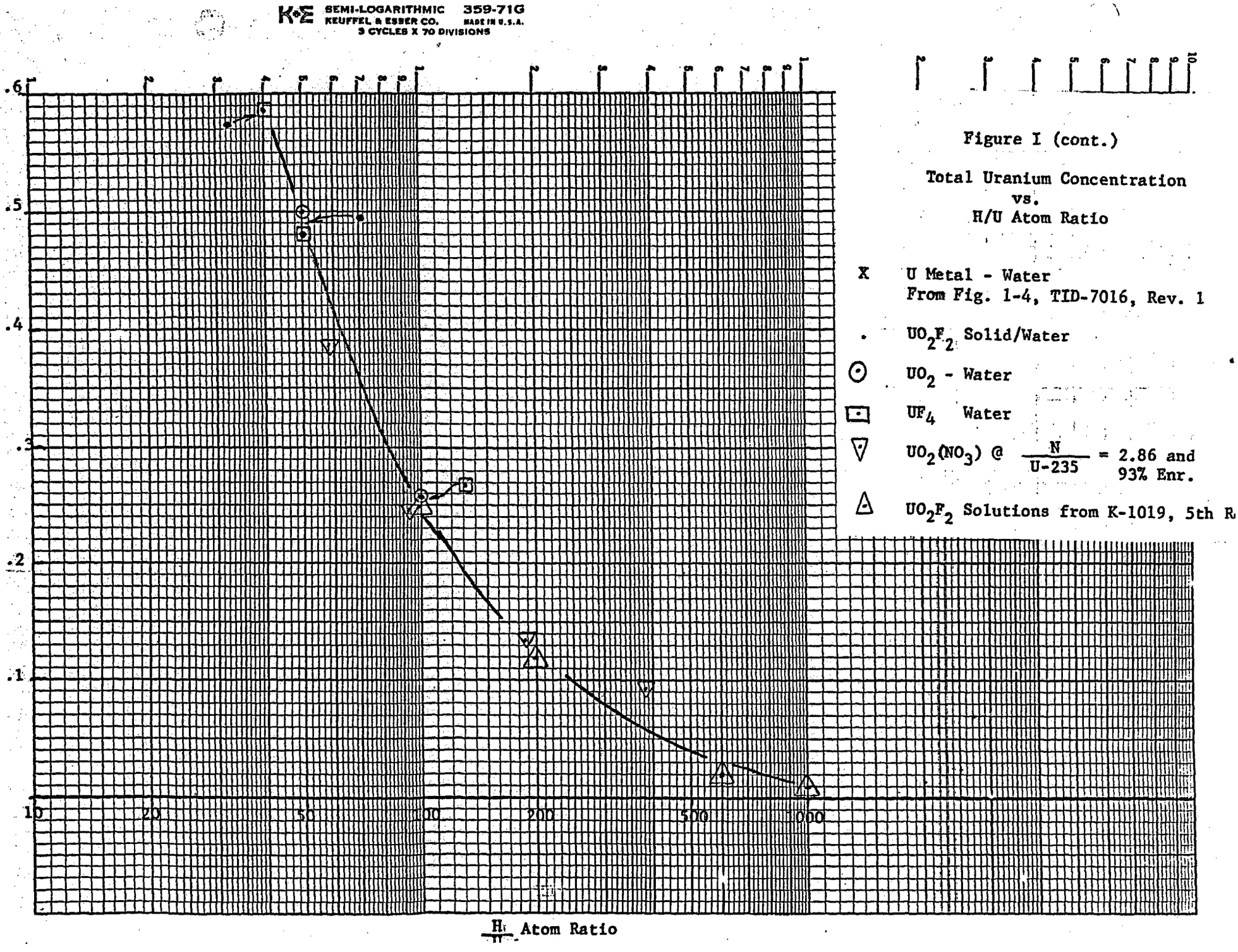
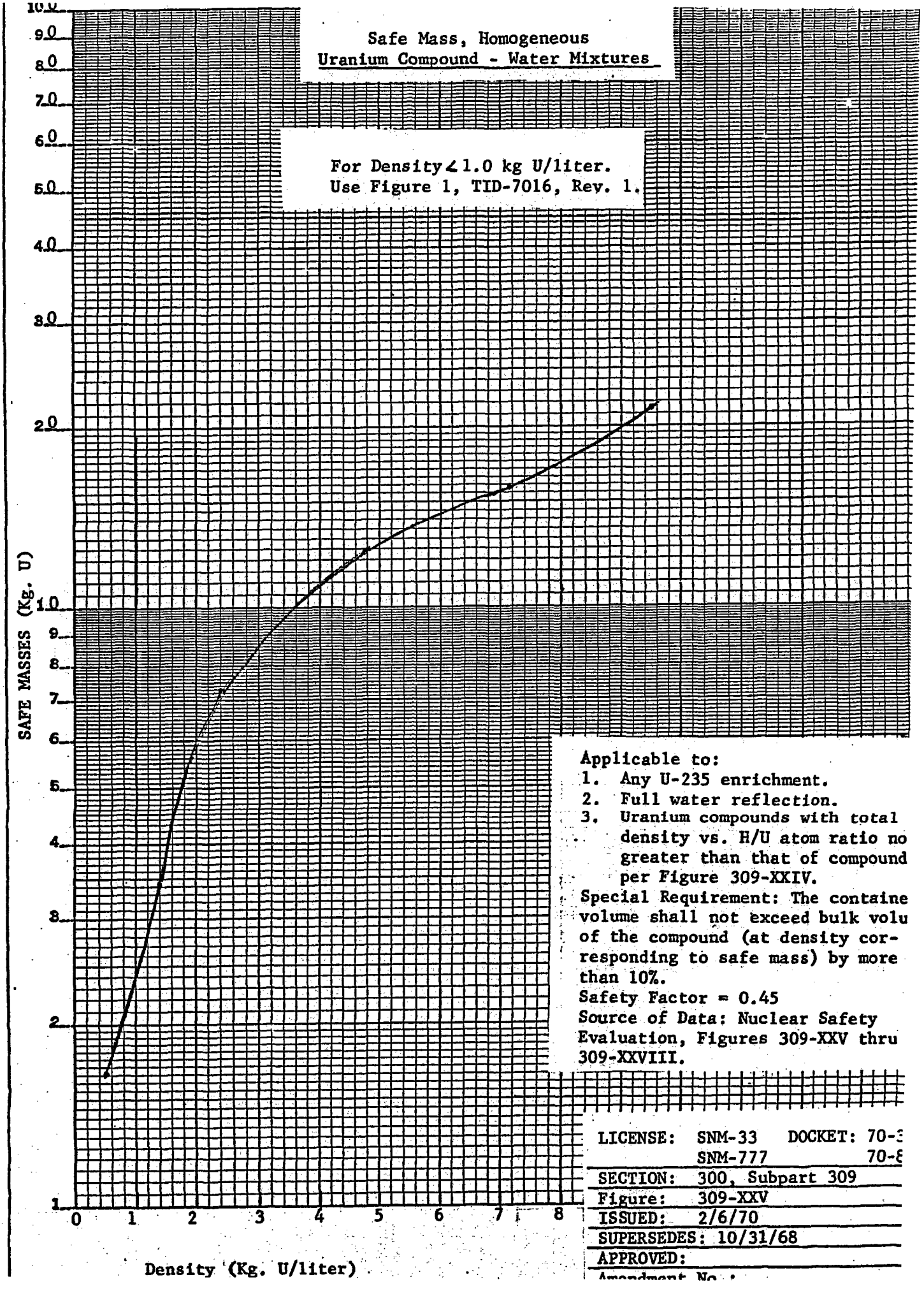



Figure I (cont.)

Total Uranium Concentration
vs.
H/U Atom Ratio


Safe Mass, Homogeneous
Uranium Compound - Water Mixtures
For Density < 1.0 kg U/liter.
Use Figure 1, TID-7016, Rev. 1.
SAFE MASSES (Kg. U)
10.0
9.0
8.0
7.0
6.0
5.0
4.0
3.0
2.0
1.0
9
8
7
6
5
4
3
2
1
0
1
2
3
4
5
6
7
8
Density (Kg. U/liter)
Applicable to:
1. Any U-235 enrichment.
2. Full water reflection.
3. Uranium compounds with total density vs. H/U atom ratio no greater than that of compound per Figure 309-XXIV.
Special Requirement: The containe volume shall not exceed bulk volu of the compound (at density corresponding to safe mass) by more than 10%.
Safety Factor = 0.45
Source of Data: Nuclear Safety Evaluation, Figures 309-XXV thru 309-XXVIII.
LICENSE: SNM-33 DOCKET: 70-3
SNM-777 70-8
SECTION: 300, Subpart 309
Figure: 309-XXV
ISSUED: 2/6/70
SUPERSEDES: 10/31/68
APPROVED:
Amendment No.

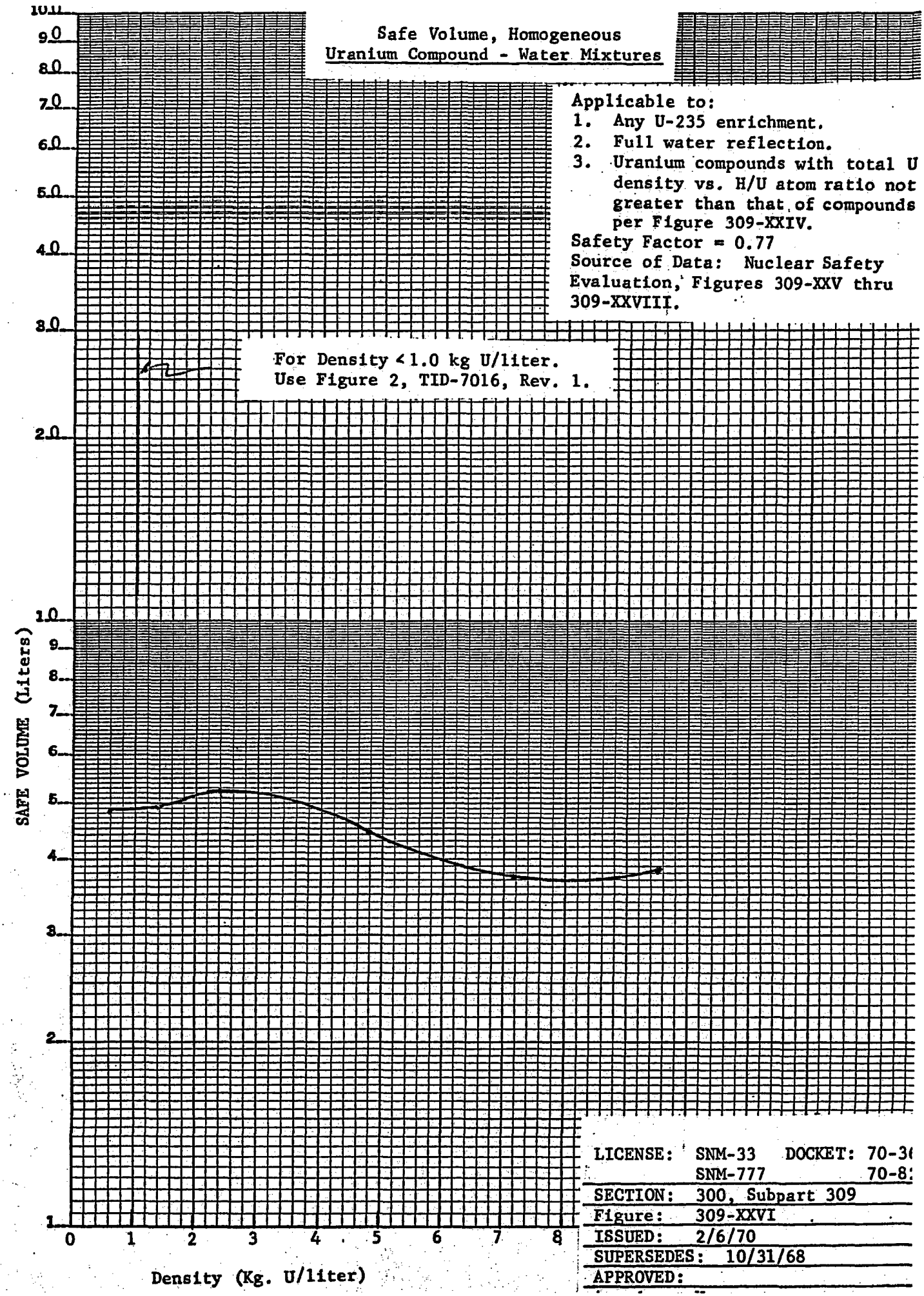
## Safe Volume, Homogeneous Uranium Compound - Water Mixtures

Applicable to:
1. Any U-235 enrichment.
2. Full water reflection.
3. Uranium compounds with total U density vs. H/U atom ratio not greater than that of compounds per Figure 309-XXIV.

Safety Factor = 0.77

Source of Data: Nuclear Safety Evaluation, Figures 309-XXV thru 309-XXVIII.

For Density < 1.0 kg U/liter.
Use Figure 2, TID-7016, Rev. 1.

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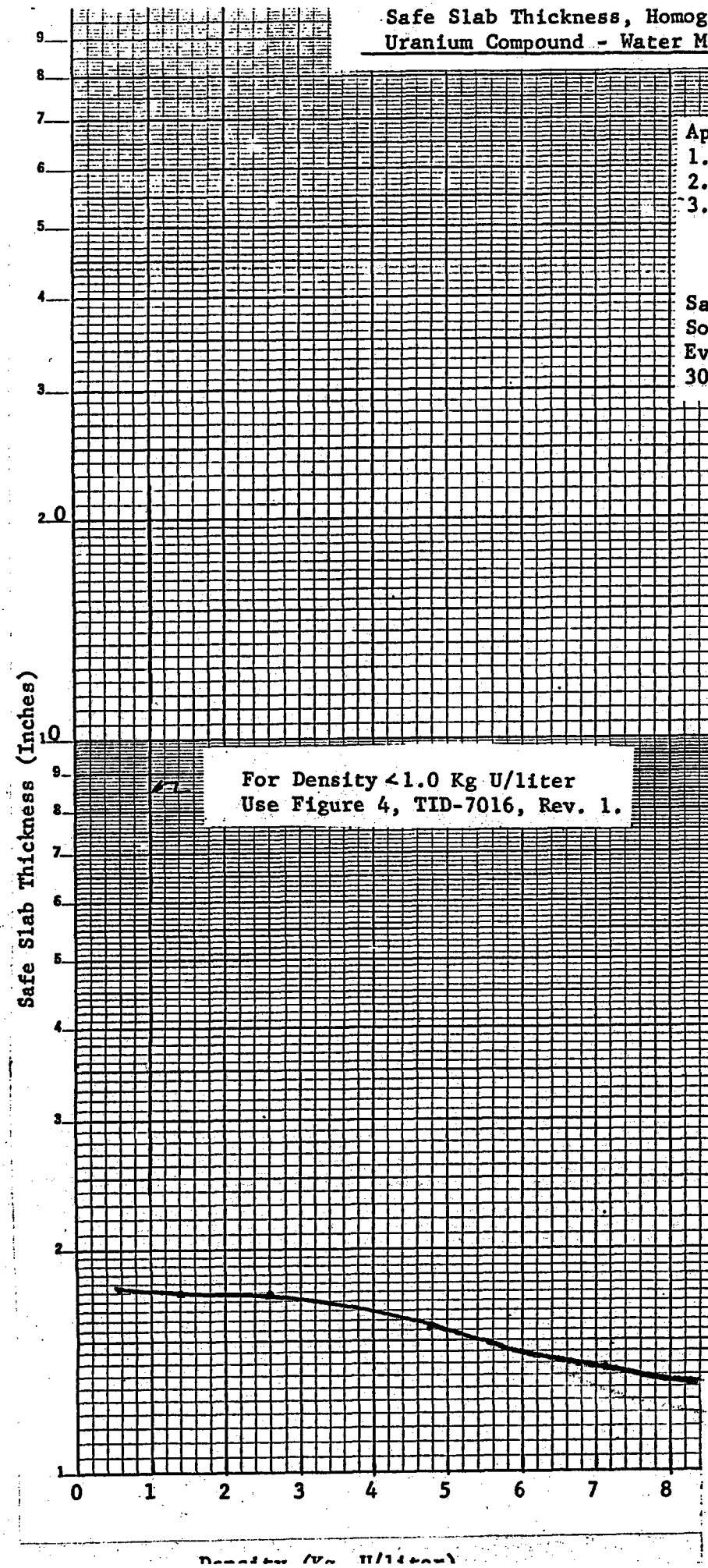
| SECTION: | 300, Subpart 309 | | |
| Figure: | 309-XXVI | | |
| ISSUED: | 2/6/70 | | |
| SUPERSEDES: | 10/31/68 | | |
| APPROVED: | | | |

## Safe Slab Thickness, Homogeneous Uranium Compound - Water Mixtures




Applicable to:

1. Any U-235 enrichment.
2. Full water reflection.
3. Uranium compounds with total U density vs. H/U atom ratio not greater than that of compounds per Figure 309-XXIV.

Safety Factor = 0.88

Source of Data: Nuclear Safety Evaluation, Figures 309-XXV thru 309-XXVIII.

| | | | |
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| | SNM-777 | | 70-82 |
| SECTION: | 300, Subpart 309 | | |
| Figure: | 309-XXVII | | |
| ISSUED: | 2/6/70 | | |
| SUPERSEDES: | 10/31/68 | | |
| APPROVED: | | | |

## Safe Cylinder Diameter, Homogeneous Uranium Compound - Water Mixtures

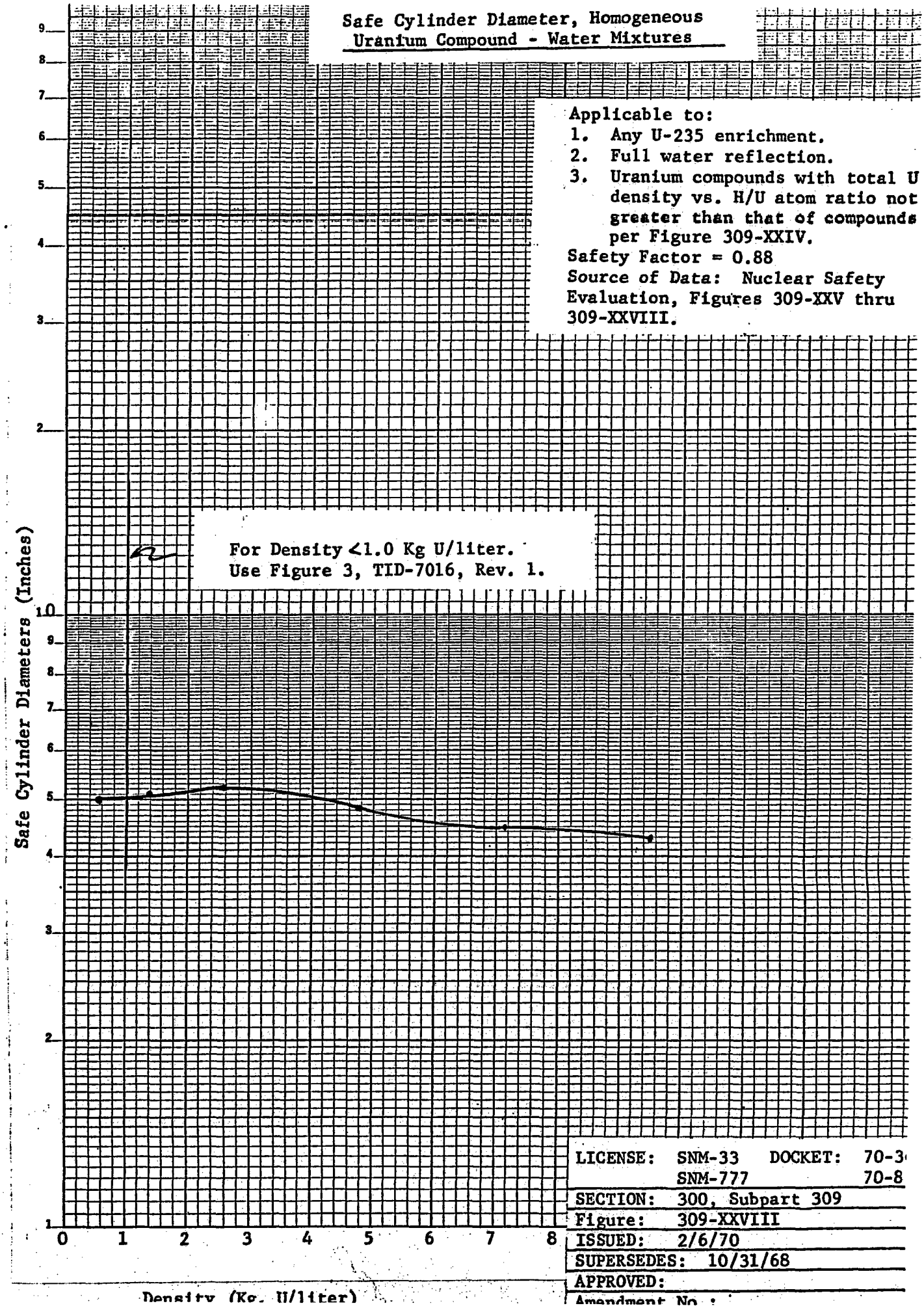
Applicable to:

1. Any U-235 enrichment.
2. Full water reflection.
3. Uranium compounds with total U density vs. H/U atom ratio not greater than that of compounds per Figure 309-XXIV.

Safety Factor = 0.88

Source of Data: Nuclear Safety Evaluation, Figures 309-XXV thru 309-XXVIII.

For Density <1.0 Kg U/liter.
Use Figure 3, TID-7016, Rev. 1.

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| | SNM-777 | | 70-8 |
| SECTION: | 300, Subpart 309 | | |
| Figure: | 309-XXVIII | | |
| ISSUED: | 2/6/70 | | |
| SUPERSEDES: | 10/31/68 | | |
| APPROVED: | | | |
| Amendment No. : | | | |

## Critical and Safe Homogeneous Uranium Compound - Water
## Cylinder Diameters and Slab Thicknesses at all Enrichments

| H/U-235 | Density U-235 (Kg./L) | Density Total U Kg. U/L | Cylinder $R_{c,r}$ (cm) | Cylinder $D_{s,r}$ (1) (in.) | Cylinder $R_{c,b}$ (cm) | Cylinder $D_{s,b}$ (1) (in.) | Slab $T_{c,r}$ (cm) | Slab $T_{s,r}$ (2) (in.) | Slab $T_{c,b}$ (cm) | Slab $T_{s,b}$ (2) (in.) |
|---|---|---|---|---|---|---|---|---|---|---|
| 0 | 8.930 | 9.55 | 6.21 | 4.31 | 9.69 | 6.71 | 3.748 | 1.29 | 11.35 | 3.92 |
| 0.98 | 6.700 | 7.16 | 6.49 | 4.5 | 10.00 | 6.96 | 3.990 | 1.38 | 11.86 | 4.09 |
| 2.94 | 4.470 | 4.78 | 7.00 | 4.85 | 10.78 | 7.47 | 4.549 | 1.57 | 12.78 | 4.41 |
| 8.96 | 2.230 | 2.38 | 7.51 | 5.21 | 11.53 | 8.0 | 5.114 | 1.76 | 13.74 | 4.74 |
| 20.60 | 1.120 | 1.20 | 7.34 | 5.09 | 11.34 | 7.85 | 5.098 | 1.76 | 13.55 | 4.67 |
| 43.90 | 0.558 | .596 | 7.22 | 5.0 | 11.13 | 7.71 | 5.197 | 1.79 | 13.32 | 4.60 |

NOTE:

NOTE: (1) $D_s = \dfrac{2\ R_c\,(cm)}{2.54\ (cm/in.) \times 1.13\ s.f.} = .693\ R_c$, where $R_c$ values are obtained from Table V-K, LA-3612

(2) $T_s = \dfrac{T_c\,(cm)}{2.54\ (cm/in.) \times 1.13\ s.f.} = .346\ T_c$, where $T_c$ values are obtained from Table V-K, LA-3612

Subscripts:
- c = critical
- s = safe
- b = bare
- r = full water reflector
- s.f. = safety factor

| | |
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| LICENSE: | SNM-33 Docket: 70-36<br>SNM-777 70-820 |
| SECTION: | 300, Subpart 309 |
| Nuclear Safety Evaluation-Data fo | |
| Figures 309-XXV thru XXVIII | |
| Page 1 of 2 | |
| ISSUED: | 2/6/70 |
| SUPERSEDES: | New |
| APPROVED: | |
| Amendment No: | |

## Critical and Safe Homogeneous Uranium Compounds - Water Volumes and Masses at all Enrichments

| /U-235 | Density U-235 (Kg./L) | Density Total U Kg U/L | Sphere $V_{c,r}$ (1) (Liters) | Sphere $V_{s,r}$ (2) (Liters) | Sphere $V_{c,b}$ (1) (Liters) | Sphere $V_{s,b}$ (2) (Liters) | Mass $M_{c,r}$ (3) (Kg. U-235) | Mass $M_{s,r}$ (4) (Kg. U) | Mass $M_{c,b}$ (3) (Kg. U-235) | Mass $M_{s,b}$ (4) (Kg. U) |
|---|---|---|---|---|---|---|---|---|---|---|
| | 8.930 | 9.55 | 5.03 | 3.87 | 10.15 | 7.81 | 44.93 | 21.6 | 90.63 | 43.6 |
| .98 | 6.700 | 7.16 | 4.84 | 3.73 | 11.25 | 8.65 | 32.41 | 15.58 | 75.39 | 36.2 |
| .94 | 4.470 | 4.78 | 5.79 | 4.45 | 13.78 | 10.60 | 25.89 | 12.45 | 61.58 | 29.6 |
| .96 | 2.230 | 2.38 | 6.80 | 5.24 | 16.68 | 12.81 | 15.16 | 7.30 | 37.21 | 17.9 |
| .60 | 1.120 | 1.20 | 6.42 | 4.94 | 15.91 | 12.22 | 7.19 | 3.46 | 17.82 | 8.59 |
| .90 | 0.558 | .596 | 6.19 | 4.76 | 14.92 | 11.50 | 3.45 | 1.66 | 8.32 | 4.0 |

NOTE: (1) $V_c = 4/3\pi R_c^3$ where $R_c$ values are obtained from Table V-K, LA-3612

(2) $V_s = .77\ V_c$

(3) $M_c = V_c \rho$

(4) $M_s = \frac{.45\ M_c}{.935}$

Subscripts: c = critical
s = safe
b = bare
r = full water reflector

| | |
|---|---|
| LICENSE: | SNM-33 Docket: 70-36<br>SNM-777 70-820 |
| SECTION: | 300, Subpart 309 |
| Nuclear Safety Evaluation-Data fo<br>Figures 309-XXV thru XXVIII | |
| Page 2 of 2 | |
| ISSUED: | 2/6/70 |
| SUPERSEDES: | New |
| APPROVED: | |

## 603. Emergency Plans

Because of the basic difference in required response, emergency plans are broken into four categories and detailed procedures or check lists are maintained in up-to-date form to implement each category. The categories are:

A. Nuclear Alarm - False and Accidental Criticality
B. Fire Alarm
C. Release of Radioactive Materials
D. Other non-nuclear emergency which may result in B. or C.

Procedures or check lists to implement the plans for each of these emergencies, will:

A. Be brief to permit reading during the emergency.
B. Be posted where they can be used.
C. Establish limitations of authority where needed.
D. Provide guidelines for emergency judgments.
E. Insure that all needed outside assistance is summoned in timely manner.
F. Provide for relocation in event of emergency assembly area becomes untenable.
* G. Include criteria for first-aid treatment and decontamination of personnel.

### 603.1 Nuclear Alarm

603.1.1 Detailed Procedure Requirements
The detailed procedures will establish a sequence of required actions. This includes:

A. Immediate plant evacuation by all personnel upon sounding of the nuclear alarm.
B. Check list of instructions for the Emergency Director.

This will include but not be limited to:

a) Immediate accounting for all personnel
b) Identification of exposed personnel and initiating emergency treatment.
c) Preliminary radiation survey of emergency assembly building to insure safe occupation.
d) Implementation of Emergency Procedures.

C. A list of persons authorized to function as Emergency Director.

D. A check list of instructions to the plant guard. This includes establishing emergency road blocks.

E. A list of persons certified by the Nuclear and Industrial Safety Manager to function as a plant re-entry team. The Emergency Director will not normally be part of this team.

*Indicates Change

## 701. Introduction

This section 700 describes the packages, handling and administrative procedures applicable to the shipment of Special Nuclear Material.

* The Manufacturing Departments are responsible (as described in Subsection 204) for administration of the procedures described in this section 700.

*Indicates Change

## 812. High Enrichment Uranium Metal Storage

### 1. Description

These racks will be used primarily for the storage of high enrichment uranium metal after unloading from the as received shipping container and prior to processing. Miscellaneous process scrap, residues and other such material in bottles or cans may also be stored on this type of arrangement. Racks are formed by individual holders placed on an existing concrete block wall. Other walls of this type may be provided at a later date and these other walls willbe constructed of mortored 8" high density concrete blocks, or equivalent, to insure effective isolation. See Sketch 812.1-I.

Uranium metal will be stored in as received containers. These containers will be 5½"ID x 4-3/4" high (1.85 liters) maximum and will be limited to 10 kgs U-235. Process scrap, residues and other such material will be in 6" ID x 10" high (1 gallon capacity) plastic or metal bottles, jars or cans.

Racks will be constructed with individual holders fastened to a concrete block wall. These holders will be 7"x7"x7"-20 gage or heavier welded steel with two or more powder actuated bolts holding them to the wall. Individual holders will be arranged in a 4x12 arrangement with 16.8" center to center side separation and 24" center to center top to bottom separation.

Details of this arrangement are shown on Sketch 812.2-I.

### 2. Nuclear Safety

Cans for metal provide a safe volume when limited to 10 kgs U-235. The 1 gallon bottle is safe for densities up to and including 3.2
* grams U-235 per cubic centimeter. Containers meet the maximum unit quantities listed in Table V, LA-2063, Groups of containers on each wall form planar arrays which are separated by 8" to 12" of high density concrete. As stated on page 41, LA-2063, "two arrays are effectively isolated from one another if the arrays are completely separated by concrete at least 8" thick".

*Indicates Change

## 812.3 HIGH ENRICHMENT ALLOY STORAGE

### 1. Description

These racks will be used to store U-AL alloy. The material will be in the form of pieces, melting splatter, residues left from core punching, etc. or cast ingots or rolled slabs. Two racks, maximum, will be placed end-to-end forming individual arrays. There will be two such arrays side-by-side separated by 10 feet in this Storage Area. See Sketch 812.1-I for the rack arrangement.

The small pieces will be held in metal tote boxes. These boxes are constructed of 1/16" aluminum with 3"x6"x16" inner dimensions. Large pieces (ingots and slabs) will be limited so that the thickness and width do not exceed 18 sq. inches (3"x6").

Racks will be constructed of 3/16" slotted angle bolted together. The outside will be covered with a thin metal (24 gage or heavier) to provide 14" edge-to-edge and 12-1/2" top to bottom separation between ports. Ports are long troughs running the entire length of the rack with a maximum 3-3/4"x7-1/2" opening. Steel hasps with 16 gage steel plates welded to them are welded along the length of each opening to provide closures which ensure retention of the tote boxes or ingots or slabs. Racks are fastened to the floor by bolting with powder actuated bolts.

Details of construction are shown on Sketch 812.3-I.

### 2. Nuclear Safety

* The tote boxes or the ingot or slab geometrics provide a safe cross section for the material to be stored. The nuclear safety of this storage arrangement is evaluated in the attached Nuclear Safety Evaluation.

*Indicates Change

## DESCRIPTION

The material to be stored will be U-Al alloy not exceeding 50 w/o. Large pieces such as ingots will be stored in their process shapes. Small pieces such as "cores" or fillers, melting splatter and other process residues will be stored in 3"x6" cross sectional containers and placed in the racks forming rectangular parallelepipeds not exceeding 156" in length. The volume of material on a shelf will not exceed

$$V_{SHELF} = 3'' \times 6'' \times 156'' = 7.62\ cm \times 15.24\ cm \times 396.24\ cm = 46.05\ liters$$

The maximum number of shelves is 24

## MODEL

The storage arrangement will be evaluated using the Density Analog model as developed by J.T. Thomas in CONF-680909. Values for unmoderated, bare systems were obtained from Fig. 28, page 215, DP-532. Values for undermoderated and moderated, bare systems were calculated using the CRITICAL Code which is a computer code based on the analytical method set forth in NDEO-1050. These data are summarized in Table A.

The number of units in a critical array will be obtained using Equation 5a, Page 9, CONF-680909

$$N = (0.012/f)(\rho_o/\bar{\rho})^{1.8} \quad , f \leq 0.4$$

Where, $f$ = fraction critical of a unit cell = $\frac{\text{mass of equivalent sphere}}{\text{mass of critical bare sphere}} = \frac{M_E}{M_{CRB}}$

$\bar{\rho}$ = average density of fissile material (includes spacing variations)

$\rho_o$ = density of material at full density

## CALCULATIONS

The maximum loading per shelf when the material is dry is approximately 109 Kg U. At 50 w/o,

$$109\ Kg\ U: V_U = \frac{109\ Kg}{19.1\ Kg/l} = 5.83\ l$$

$$109\ Kg\ AL: V_{AL} = \frac{109\ Kg}{2.7\ Kg/l} = 40.40\ l$$

$$\Sigma = 46.23\ l \approx V_{SHELF}$$

| | |
|---|---|
| LICENSE: SNM-777 DOCKET: 70-820 | |
| SECTION: 800 | SUBPART: 812.3 |
| NUCLEAR SAFETY EVALUATION- HIGH ENRICHMENT ALLOY STORAGE | |
| Page 1 of 6 | |
| ISSUED: 2/6/70 | |
| SUPERSEDES: 10/31/68 | |
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The volume fraction of uranium is

$$V_F(U) = \frac{V_U}{V_{SHELF}} = \frac{5.83\,l}{46.05\,l} = .127$$

From Fig. 2.8, Page 215, DP-532, the critical mass of a bare sphere of U-AL with a volume fraction (uranium) of .127 is

$$M_{c,B} = 870 \text{ Kg } U^{235}$$

The density of the uranium on the shelf is

$$\rho_o = \frac{M_U}{V_{SHELF}} = \frac{109 \text{ Kg}}{46.05\,l} = 2.367 \text{ Kg}/l$$

Since the material is dry

$$H/U^{235} = 0$$

Using Equation 17, page 25, NDEO-1050, the reflector savings is

$$\delta = 5.46 - 0.145\,(H/U^{235}) = 5.46 \text{ cm.}$$

The buckling of the material on the shelf is obtained using Equation 8, page 6, NDEO-1050

$$B_g^2 = \frac{\pi^2}{(\text{length}+2\delta)^2} + \frac{\pi^2}{(\text{width}+2\delta)^2} + \frac{\pi^2}{(\text{thickness}+2\delta)^2} = \frac{9.87}{(396+10.92)^2} + \frac{9.87}{(15.24+10.92)^2} + \frac{9.87}{(7.62+10.92)^2}$$

$$= .0432 \text{ cm}^{-2}$$

Using Equation 6, page 6, NDEO-1050 and rearranging terms, the radius of an equivalent sphere is

$$r_E = \sqrt{\frac{\pi^2}{B_g^2}} - \delta = \sqrt{\frac{9.87}{.0432}} - 5.46 = 9.66 \text{ cm}$$

The volume of an equivalent sphere is

$$V_E = \frac{4}{3}\pi r^3 = 3.777 \text{ liters}$$

The mass of an equivalent sphere is

$$M_E = \rho_o V_E = 2.367 \text{ Kg}/l \times 3.777\,l = 8.940 \text{ Kg}$$

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|---|
| LICENSE: SNM-777 DOCKET: 70-820 |
| SECTION: 800 SUBPART: 812.3 |
| NUCLEAR SAFETY EVALUATION - HIGH ENRICHMENT ALLOY STORAGE |
| Page 2 of 6 |
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| APPROVED: AMENDMENT NO.: |

The fraction critical is

$$f = \frac{M_S}{M_{CJB}} = \frac{8.940\ Kg}{870\ Kg} = .0103$$

The volume of the unit cell will be calculated assuming the dimensions of one shelf in a rack plus separation from adjacent shelves. The other rack will be assumed to be located one foot away giving at least one foot edge-to-edge spacing between shelves in any direction. This is a conservative approach since the other rack is actually located ten feet away. However, this approach will result in a more cubic arrangement of the unit cell. Using the dimensions shown on Fig. 812.3-1

thickness = 3.75" + 12.25" = 16"
width = 7.5" + 14" = 21.5"
length = 156" + 12" = 168."

The volume of the unit cell will be

$$V_{uc} = 16" \times 21.5" \times 168" = 947.79 \text{ liters}$$

The density of the unit cell is

$$\rho = \frac{M_u}{V_{uc}} = \frac{109\ Kg}{947.79\ l} = 0.115\ Kg/l$$

The number of critical units will be

$$N_c = \left(\frac{.012}{.0103}\right)\left(\frac{2.367}{0.115}\right)^{1.8} = 1.165(20.58)^{1.8} = 1.165 \times 232 = 270 \text{ units (shelves)}$$

To check the effect of some moderation, the units will be evaluated assuming 50% water. At $H/U^{235} = 5.23$, $M/W = 1.8519$, the maximum loading per shelf is approximately 72 Kg U. At 50%,

$$72\ Kg\ U:\ V_U = \frac{72\ Kg}{18.7\ Kg/l} = 3.85\ l$$
$$72\ Kg\ AL:\ V_{AL} = \frac{72\ Kg}{2.7\ Kg/l} = 26.65\ l$$
$$\Sigma = 30.50\ l$$

The volume of water is

$$V_{H_2O} = 46.05 - 30.50 = 15.55\ l$$

And the weight of water is

$$M_{H_2O} = 15.55\ Kg$$

| |
|---|
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| NUCLEAR SAFETY EVALUATION - HIGH ENRICHMENT STORAGE |
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Using the relationship shown on page 14, NDEO-1050 as an over check

$$H/U^{235} = 26.1\left(\frac{N_{H_2O}}{N_u}\right) = 26.10\left(\frac{15.55}{72}\right) = 5.64$$

From Table A, the critical mass of a bare sphere at 50 N/o and a $H/U^{235} = 5.22$ is

$$M_{c,o} = 38.7\ Kg\ U^{235}$$

The critical density is

$$\rho_o = \frac{M_{c,o}}{V_{c,R}} = \frac{25.129\ Kg}{15.676\ \ell} = 1.603\ Kg/\ell$$

Using Equation 17, page 25, NDEO-1050, the reflector savings is

$$\delta = 5.46 - 0.148(H/U^{235}) = 5.46 - (0.148 \times 5.22) = 4.69\ cm$$

The buckling of the material on the shelf is determined using Equation 8, page 6, NDEO-1050

$$B_g^2 = \frac{\pi^2}{(length+2\delta)^2} + \frac{\pi^2}{(width+2\delta)^2} + \frac{\pi^2}{(thickness+2\delta)^2} = \frac{9.87}{(3.96+9.38)^2} + \frac{9.87}{(15.24+9.38)^2} + \frac{9.87}{(7.62+9.38)^2}$$

$$= .0506\ cm^{-2}$$

Using Equation 6, page 6, NDEO-1050 and rearranging terms, the radius of an equivalent sphere is

$$r_E = \sqrt{\frac{\pi^2}{B^2}} - \delta = \sqrt{\frac{9.87}{.0506}} - 4.69 = 9.28\ cm$$

The volume of an equivalent sphere is

$$V_E = \frac{4}{3}\pi r^3 = 3.349\ liters$$

The mass of an equivalent sphere is

$$M_E = \rho_o V_E = 1.603\ Kg/\ell \times 3.349\ \ell = 5.368\ Kg$$

The fraction critical is

$$f = \frac{M_E}{M_{c,o}} = \frac{5.368}{38.7} = .1387$$

| |
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As before, the volume of the unit cell will be

$$V_{uc} = 947.79\ \ell$$

The density of the unit cell will be

$$\bar{\rho} = \frac{M_u}{V_{uc}} = \frac{72\ Kg}{947.79\ \ell} = 0.076\ Kg/\ell$$

The number of critical units when partially moderated ($H/U^{235} = 5.22$) will be.

$$N_c = \left(\frac{.1012}{.1587}\right)\left(\frac{1.663}{.076}\right)^{1.8} = .087\,(21.09)^{1.8} = .087 \times 242 = 21.054$$

If accidentially flooded, the storage arrangement also will be safe. The uranium on each individual shelf is restricted to an 18 square inch cross section (3"x6"). Under conditions of optimum water moderation, this is a safe cross section for weight percents up to approximately 50. This is shown on Figure 309-~~IX~~.

Individual shelves would be separated by at least 12 inches of water under these conditions. As shown in Section 300, this meets the isolation criteria.

CONCLUSIONS

Based on the calculations, approximately 15 liters or 15 Kgs (38 lbs) of water would have to be added to 142 Kgs of U-Al in each shelf before criticality could occur. The material is usually stored dry with no hydrogeneous material stored with it. Occasionally, it may be necessary to keep process documentation with specific enrichments, however this amount of paper and/or polyethylene protective wrapping will never total more than a few grams or ounces per shelf.

Therefore, this storage arrangement is nuclearly safe.

| | |
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# 50 w/o U-AL-$H_2O$ CRITICAL SYSTEM

| $H/U^{235}$ | $k_\infty$ | $M^2$ ($cm^2$) | $B_M^2$ (critical) ($cm^{-2}$) | $M_{c,R}$ ($Kg\ U^{235}$) | $V_{c,R}$ (liters) | $M_{c,B}$ ($Kg\ U^{235}$) | $M/W$ * |
|---|---|---|---|---|---|---|---|
| 0 | – | – | – | ~300,000 | – | ~870,000 | ∞ |
| 5.22 | 2.400 | 72.560 | 0.13890 | 25.129 | 15.676 | 38.700 | 1.8519 |
| 15.66 | 2.055 | 44.604 | 0.15383 | 11.042 | 11.306 | 23.555 | .6173 |
| 26.10 | 2.045 | 37.279 | 0.16741 | 5.429 | 7.730 | 12.660 | .3704 |
| 36.54 | 2.034 | 34.306 | 0.17363 | 3.579 | 6.527 | 8.708 | .2646 |
| 52.20 | 2.019 | 32.140 | 0.17863 | 2.393 | 5.801 | 6.005 | .1852 |
| 156.60 | 1.920 | 29.076 | 0.17744 | .905 | 5.815 | 2.270 | .0617 |
| 261.00 | 1.831 | 28.720 | 0.17014 | .688 | 7.175 | 1.635 | .0370 |
| 391.50 | 1.731 | 28.758 | 0.15943 | .625 | 9.645 | 1.383 | .0247 |
| 522.00 | 1.641 | 28.931 | 0.14885 | .639 | 13.050 | 1.322 | .0185 |
| 625.50 | 1.560 | 29.138 | 0.13863 | .696 | 17.684 | 1.353 | .0148 |
| 783.00 | 1.487 | 29.351 | 0.12875 | .789 | 24.027 | 1.448 | .0123 |
| 1044.00 | 1.359 | 29.756 | 0.10978 | 1.121 | 45.313 | 1.843 | .0093 |

* VOLUME RATIO, METAL TO WATER

TABLE A

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NUCLEAR SAFETY EVALUATION – HIGH ENRICHMENT ALLOY STORAGE



ISSUED: 2/6/70

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## SECTION 800 - FUEL FABRICATION OPERATION

## SUBSECTION 820 - PROCESSING

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SUBPART 821 INTRODUCTION

SUBPART 822 FABRICATION OF ALUMINUM PLATE TYPE ELEMENT ASSEMBLIES
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822.2 Process Description

SUBPART 823 FABRICATION OF STAINLESS STEEL, ALUMINUM, OR ZIRCALOY CLAD TUBULAR TYPE $UO_2$ PELLET ASSEMBLIES
823.1 General Considerations
823.2 Process Description

SUBPART 824 FABRICATION OF ENRICHED URANIUM METAL
824.1 General Considerations
824.2 Process Description

SUBPART 825 FABRICATION OF URANIUM ALUMINUM INTER METALLIC CORES
825.1 General Considerations
825.2 Process Description
825.3 Support or Auxiliary Operation

## 822.1 GENERAL CONSIDERATIONS

1. This Subpart covers the preparation and processing of up to and including fully enriched uranium-aluminum fuel material containing up to and 50% $U^{235}$ by weight and its fabrication into plate type fuel element assemblies. Individual operations which may be performed are shown on Diagram 822-I and outlined in their expected sequence.

2. Operations on unclad material are performed in Building 19H-1, Alloy Shop. Plate and element assembly operations are noramlly performed in Buildings 50H.

3. BAWTR fuel elements are the most reactive component fabricated according to this subsection. Therefore, the Nuclear Criticality and Safety limits for this type of fuel element will be the most
* restrictive. Two BAWTR fuel elements meet the criteria of Section 300, and requirements of Subpart 822.2, that $K_{eff}$ (bare) $\leq 0.65$ and $K_{eff}$ (reflected) $\leq 0.9$. Refer to Nuclear Safety Evaluation, BAWTR Fuel Elements, Subpart 822.

4. The x-ray or fluoroscoping of unclad material is sometimes required. (Reference Subparts 822.2.10 and 16). These operations are normally performed in Building 50H. Ingots, rolled ingots, and cores are wrapped in clean polyethylene wrappers prior to removing them from Building 19H. After x-raying or flouroscoping, the wrapped, unclad pieces are removed from Building 50H and returned to Building 19H for further processing or storage.

5. Unless stated otherwise, safe values referenced in the following Subparts will be obtained using the Figures referenced in the following table:

| | Safe Value | Figure |
|---|---|---|
| 6.1 | Safe Diameter | 309-VIII |
| 6.2 | Safe Cross Sectional Area | 309-IX |
| 6.3 | Safe Volume | 309-X |

*Indicates Change

## 822.2 PROCESS DESCRIPTION

1. Receive Enriched Uranium

Enriched uranium will be received and handled in accordance with Subsection 810.

2. Pickle Scrap Plates to Remove Clad

This operation is not always required. Up to 350 grams of U-235 is allowed at this operation. Plates are pickled to within 0.005 inches of the SNM, then removed. Upon removal from the acid, the plates will be measured for thickness. If the thickness value is exceeded, the spent pickle acid is analyzed for U-235 content either
* chemically or by gamma counting and emptied into drums if the concentration is less than 5 grams/liter and 700 grams U-235 per drum.

3. Shear to Size

This is normally dry and limited to 700 grams U-235. Croppings or sheared pieces may be collected in a safe geometry container which will be physically spaced one foot or more from the material being sheared.

4. Degrease Recycle Material

Up to 350 grams U-235 may be processed at one time.

5. Charge Preparation

The required quantity of SNM for a single ingot is weighed as either one batch in a safe volume container or increments of 700 grams of U-235 which is then consolidated into a safe volume container. Safe volume containers are normally dry (H/X < 2) and limited to 10 kgs U-235 each. The safe volume container will be spaced one foot or more from the material being handled.

6. Load and Evacuate Furnace

The furnace used for this operation is a vacuum induction furnace with a chamber approximately 5' in diameter and 8' long. The maximum size graphite or aluminum crucible which is used for U-Al material is 7-3/8" dia. x 12" high. The graphite molds used for ingot casting are 1" x 6" x 24" or 1-1/8" x 5" x 31". Details of the furnace are shown on Sketch 822-II.

One safe volume container or safe cross section tote box of alloy or one 1 liter container of unalloyed U-235 at a time is brought into the furnace and its contents loaded into the crucible.

*Indicates Change

42. Assemby Stacking

This is performed in a special zone and loose plates are limited to 40 grams per linear inch or placed in a safe cross-section container.

43. Pack in Plastic Bags

Same considerations as in Subpart 822.2.42.

44. Store Plate

Plates will be stored in accordance with Subsection 810.

45. Procure Plates from Storage

Same considerations as Subpart 822.2.26.

46. Roll Swage

One elements worth of plates will be roll swaged into place to form assemblies. Loose plates will be handled to the limits set forth in Subpart 822.2.26. SNM bearing plates will be placed in safe cross-section fixtures for swaging. After swaging, one element will be handled. The one element will have $K_{eff}$ (bare) <0.65 and $K_{eff}$ (reflected) <0.9.

47. Clean

The elements will then be cleaned in hot water. Normally, only one element will be cleaned at a time. The number of elements will be limited such that k eff (bare) < 0.65 and k eff (reflected) < 0.9 or the elements will have a safe cross-sectional area or will be limited to 20 grams per linear inch.

48. Inspect Assembly

The number of elements will be limited so that k eff (bare) < 0.65 and k eff (reflected) < 0.9.

49. Attach Combs

If required, combs will be attached to the assemblies. The same limits used in Subpart 822.2.47 will apply.

50. Weld End Fittings

End fittings will be welded to the assemblies at this operation, if required. The same limits used in Subpart 822.2.47 will apply.

51. Hand Finish Welds

If required, welds will be hand finished. The limits used in Subpart 822.2.47 apply.

52. Final Machine

A final machining operation will be performed if required. The same limits used in Subpart 822.2.47 will again apply.

| 2. Type Rod | No Required Per Type I Element | No Required Per Type II Element | No Required Per Instrument Element |
|---|---|---|---|
| Free End | 20 | 20 | 20 |
| Fixed Type A | 8 | 8 | 8 |
| Remov. Type 1 | 1 | - | - |
| Remov. Type 2 | - | 1 | - |
| PPC Type | 6 | 6 | 6 |
| Segmented | 1 | 1 | 1 |
| Instrument | - | - | 1 |

D. General Information

1. Enrichment

a. Normal Fuel 2.34 ± 0.05 w/o

b. Segmented Fuel 2.34 ± 0.05 w/o

c. PPC Fuel 1.77 ± 0.05 w/o

d. Gadolinia Content 1.74 ± 0.05 w/o (Remov. Type 2 Pellets Only With 2.34 ± 0.05 w/o U-235)

Note: Free End, Type A, Remov. No. 1 and Remov. No. 2 Know as "Normal"

2. Pellet Density: 93.5% Theoretical ± 2.0%

II. ASSUMPTIONS

A. The design reactivity information contained in NDEO-1033 is applicable and will be used to determine effective multiplication factors ($K\infty = 1.21$ and $M^2 = 39\ cm^2$)

B. Calculations will be performed using the following equations:

$$B^2 = \frac{\pi^2}{(\text{length} + 2\delta)} + \frac{\pi^2}{(\text{width} + 2\delta)} + \frac{\pi^2}{(\text{thickness} + 2\delta)^2} \quad \text{and}$$

$$K_{eff} = \frac{k\infty}{1 + M^2B^2}$$

C. Using Fig. 4-27 of ANL-5800, 2nd Edition, a reflector savings ($\delta$) of 8 cm. for a full water reflector was selected.

* D. From Fig. 3 and 4 of TID-7028, an extrapolation length (also designated $\delta$) of 2.5 cm. was selected for bare, moderated systems. This value is based on highly enriched uranium experimental data; however, it is consistent with the calculated results shown on Fig. 2.7 of DP-532. At lower enrichment DP-532 indicates a higher extrapolation length (approximately 4 cm.) which is probably due to calculated or experimental error or perhaps a real effect due to increasing system radii. This variation of extrapolation length from 2.5 to 4 cm. will yield minor reactivity changes, probably 10% or less in K eff.

*Indicates Change

E. Sixteen Elements (continued)

2. Bare Case

$$B^2_{Bare} = \frac{9.87}{(50.25 + 5)^2} + \frac{9.87}{(50.25 + 5)^2} + \frac{9.87}{(275 + 5)^2} = \frac{9.87}{(55.25)^2} + \frac{9.87}{(55.25)^2}$$

$$+ 0.0001 = \frac{9.87}{3052.56} + \frac{9.87}{3052.56} + 0.0001 = 0.00323 + 0.00323 + 0.0001 =$$

$$0.00656$$

$$K_{eff} = \frac{1.21}{1 + (39 \times 0.00656)} = \frac{1.21}{1 + 0.2558} + \frac{1.21}{1.2558} = \boxed{0.964}$$

IV. CONCLUSIONS

* A. These calculations indicate that 16 elements in a 4 x 4 array would be just critical. Calculations in NDEO-1033, -1164, and 1501, indicate that 17 nominally fabricated elements would be required for criticality. Therefore, these calculations are representative.

*Indicates Change

Supplement to
NUCLEAR SAFETY EVALUATION
DRESDEN FUEL ELEMENT (BWR)

## I. Material Description

$PuO_2$ - $UO_2$ Fuel Rods and Elements.

### A. General

This supplement provides additional information applicable to the shipment of Dresden $PuO_2$ - $UO_2$ fuel rods and elements. The orginial nuclear safety evaluation remains unchanged.

The rods and elements are the same type and with the same dimensions and arrangement as the previously described Dresden BWR elements.

### B. Pellets

Pellets are sintered (high fired) ceramic of two types:

a) $UO_2$ 2.34% $U^{235}$ enrichment.

b) Mixed oxide $PuO_2$ - $UO_2$

Nominal composition:

97.7 ± w/o $UO_2$ natural enrichment
2.3 ± w/o $PuO_2$
100.0

Nominal isotopic composition of Pu is:

| | |
|---|---|
| Pu 238 | 0.37% |
| Pu 239 | 71.34% |
| Pu 240 | 20.63% |
| Pu 241 | 6.09% |
| Pu 242 | 1.57% |
| | 100.00% |

### C. Rods

Two types

a) $UO_2$ pellets

Dimensions, weight and enrichment as described in Nuclear Safety Evaluation--Dresden I Fuel Elements (BSR) dated 9/15/67.

b) $PuO_2$ - $UO_2$ pellet

| | | |
|---|---|---|
| Net weight | $PuO_2$ - $UO_2$ | 3254 gms/rod |
| | $UO_2$ | 3179 gms/rod |
| | $PuO_2$ | 75 gms/rod |
| | U-235 | 19.9 gms/rod |
| | Pu | 66 gms/rod |

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SECTION: 800 SUBPART: 823

Nuclear Safety Evaluation
Dresden Fuel Element (BWR)



ISSUED: 2/6/70

SUPERSEDES: New

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D. Element

| Type Rod | Number | Net Weight Per Rod, Grams | Net Weight Per Element, Grams |
|---|---|---|---|
| $UO_2$ Pellet, Free End and Fixed End | 19 | 3251 | 61769 |
| $UO_2$ Pellet, PPC | 6 | 3251 | 19506 |
| $UO_2$ Pellet, | 1 | 3013 | 3013 |
| $UO_2$ -$Gd_2O_3$ Pellet | 1 | 3251 | 3251 |
| $PuO_2$ - $UO_2$ Pellet, Free end and fixed end | 9 | 3254] | 29286 |
| | 36 | | 116,825 |

II. Nuclear Criticality Safety

A. A. Elements

The $PuO_2$ - $UO_2$ elements are neutronically equivalent to standard $UO_2$ Dresden element. (Ref. NDEO 1501). The Nuclear Safety Evaluation for assembly and handling Dresden elements remains unchanged and is applicable to these elements.

B. Rods

The standard batch size will be 18 rods. As shown in NDEO-1771, 90 rods at the optimum moderating condition and fully reflected will have a $K_{eff}$ not exceeding 0.92. Therefore, 18 rods will be subcritical by a larger margin and are nuclearly safe.

## I. BASIC INFORMATION

A. Pellet Description: 0.3145" Ø x 0.6" → 1.0" long

B. Rod Description:

| Type Rod | Fuel Stack Length | Tube Length | Weight | Fuel Loading/Rod |
|---|---|---|---|---|
| 1 | 91.00 ± .25 | 94.57" | 3.3 lbs. | 1182.8 g $UO_2$ (36.49 g U-235) |

2. Zircaloy-2 Tubing
Std. Tube: 0.020" W.T. x 0.368" ± 0.002" OD x 0.3210" ± 0.001" ID

C. Element Description

7.615" x 7.615" x
- Length Over Active Fuel - 91.00"
- Length Between Grids - 95.44"
- Length Overall - 111.9"
- 238 Fuel Rods Per Element
- Pitch - 0.468"

D. General Information

1. Enrichment: 3.5 w/o
2. Pellet Density: 94% Theoretical

## II. ASSUMPTIONS

A. The design reactivity information contained in NDEO-1134 is applicable and will be used to determine effective multiplication factors ($K_\infty$ = 1.405 and $M^2$ = 40.9 $cm^2$ )

B. Calculations will be performed using the following equations

$$B^2 = \frac{\pi^2}{(\text{length} + 2\delta)} + \frac{\pi^2}{(\text{width} + 2\delta)} + \frac{\pi^2}{(\text{thickness} + 2\delta)} \quad \text{and}$$

$$K\ eff = \frac{K_\infty}{1 + M^2 B^2}$$

C. Using Fig. 4-27 of ANL-5800, 2nd Edition, a reflector savings ($\delta$) of 8 cm for a full water reflector was selected.

* D. From Fig. 3 and 4 of TID-7028, an extrapolation length (also designated $\delta$ ) of 2.5 cm. was selected for bare, moderated systems. This value is based on highly enriched uranium experimental data; however, it is consistent with the calculated results shown on Fig. 2.7 of DP-532. At lower enrichments, DP-532 indicates a higher extrapolation length (approximately 4 cm.), which is probably due to calculated or experimental error or perhaps a real factor due to increasing system radii. This variation of extrapolation length from 2.5 to 4 cm. will yield minor reactivity changes, probably 10% or less in k eff.

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Dresden 2.34 w/o $PuO_2$ - $UO_2$

Fuel Rod and Assembly Fabrication

1. Receive Rods & Assemblies
2. Open Containers & Check Packaging for Damage
3. Monitor & Verify Previously Determined Health Physics Limits
4. Remove Components from Containers. Load Elements on to carts & Place Rods in Modules
5. Visually Check Rods & Elements
6. Helium Leak Check Rods & Elements
7. Visually & Dimensionally Inspect Rods & Elements
8. Assemble $PuO_2$ - $UO_2$ & $UO_2$ Rods into Required Elements
9. Store

| | |
|---|---|
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| Flow Diagram 823-I A | |
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7. Heat Treat Ingot (continued)
The heat treatment step requires very careful controls and immediate quenching in order to meet process specifications. The operator is taught the operation under the cognizance of a supervising engineer before being permitted to perform this step on his own. Strict adherence to written instructions (route card, process outline and manufacturing operating procedure) is required. Any delay in transfer is considered just cause to halt the operation before the quenching step. As weight loss figures have shown that there is less than one-half of 1% lost during the entire heat treating cycle (including the salt bath), a many-fold increase, entirely in the water quench tank, would be necessary before the step could approach an unsafe condition.

Cleaning and changing of this quench tank will be performed so that no appreciable amount of sludge would be present, but in no case would the tank be allowed to accumulate more than 350 gm of U-235 either as sludge or solution. A running inventory by means of weight loss methods will be maintained for accounting and control purposes.

8. Machine Ingot (if required)
Ingots will be machined one at a time. Chips and turnings are
* transferred to one liter container after each machining pass. Each machining pass removes less than 350 grams $U^{235}$.

9. Inspect Ingot
One ingot at a time will be inspected.

10. Warm in Oil Bath (if required)
One ingot at a time will be warmed.

11. Hot Roll Ingot
Same considerations at Subpart 822.2.9.

12. Wash Ingot (if required)
One ingot at one time will be washed in hot water, rinsed and dried.

13. Radiograph
Same considerations as Subpart 822.2.10.

14. Shear into Slabs and Cold Roll
Same considerations as Subpart 822.2.11.

*Indicates Change

## 825.1 General Considerations

1. This Subpart covers the fabrication of the $UAL_X$ ceramic cores. $UAL_X$ is a fine mesh uranium-aluminum powder with a bulk density of 2.6 grams per cubic centimeter and a U-235 weight fraction up to and including .70. Compacts contain $UAL_X$, aluminum and sometimes nuclear poisons such as $B_4C$ powders, with a theoretical density of 6.03 grams per cubic centimeter and a U-235 weight fraction up to and including .50. The process is shown on flow diagram 825-I.

* 2. These operations will be performed in Building 19H. The processing locations and a general equipment layout are shown on Drawing B-304236. A more detailed layout of the $UAL_X$ process area is shown on Sketch 825.II.

3. Except for the specific size containers described in Subpart 825.2, all containers will have a capacity of 1 gallon or less.

*Indicates Change

Part II

# GENERAL INFORMATION AND PROCEDURES APPLICABLE TO THE HANDLING OF SPECIAL NUCLEAR MATERIAL

Applicable to

UNITED NUCLEAR CORPORATION

Commercial Products Division

Fuel Fabrication Operations
New Haven, Connecticut

Fuel Recovery Operations
Rhode Island

Date Issued October 31, 1968

Revision 1: February 6, 1970

## SECTION 900 - RECOVERY OPERATION

### Subsection 910 - STORAGE (Formerly 500)

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915.1 11 Liter Bottle Carts (Formerly 504.2.1)
915.2 Calciner Tray Carts (Formerly 504.2.5)
915.3 Bottle, Jar, Can and Bag Carts (New)

Subpart 916 COMPONENT STORAGE

916.1 Element Storage Racks (New)

Subpart 917 INPROCESS STORAGE

917.1 Sample Storage

## 913.1 Storage Area Description

There are five 8' high, 12" thick solid concrete block walls providing separation and support for storage. These are located as shown in Bay XII. The shelves are used for storage of any combinations of containers of metal, solutions, and wet or dry compounds. The following is a schedule of containers used:

| Type Material | Enrichment | Maximum Container Size | Maximum Amount in Container |
|---|---|---|---|
| Dry metal or alloy up to and including full density | Up to and including fully enriched | 5 1/2" I.D. x 4 3/4" high (1.85 L) | 10 kg U-235 |
| Dry U compounds up to and including density of 4.0 gm U/cc. | Up to and including fully enriched | 6" I.D. x 10" high (4.0 L) | 10 kg U-235 |
| Dry U compounds density >4.0 gm U-235 /cc up to and including 5.3 gm U/cc | Up to and including fully enriched | 5" I.D. x 7" high (2.2 l.) OR 3" I.D. x 6.5" high+ (0.75 liters) | 10 kg U-235<br>3.33 kg U-235+ |
| U scrap, residue, solutions with densities up to and including 1.0 gm U/cc. | Up to and including fully en- | 6" I.D. x 10" high (4.0 L) | 4 kg U-235 |

+Three such cans may be grouped in a triangular array in each storage position. Each storage position will contain no more than 10 kgs U-235.

The shelves are as shown on drawing A-602. Storage is on individual shelves for the containers arranged to give 1'-4" horizontal center-to-center spacing, and 1'-8" vertical center-to-center spacing.

Containers meet the maximum unit quantities listed in Table V, LA-2063. Groups of containers on each wall form planar arrays which are separated by 12" of high density concrete. As stated on page 41, LA-2063, "Two arrays are effectively isolated from one another if the arrays are completely separated by concrete at least 8" thick."

913.1 Storage Area Description (continued)

Storage of in process control or accountability samples is provided by an expanded metal shelf, mounted on the east window ledge of the Foreman's office. This rack, the containers used and the nuclear safety are described in Subpart 917.

## 917. In-Process Storage

### 917.1 Sample Storage

Racks for the storage of sample bottles will be placed throughout the entire plant. These racks form troughs with 2-1/2" thick x 4" high openings. Racks are constructed of angle iron, slotted metal or sheet metal. The rows are in line forming a plane 2-1/2" thick.

Locations of these type of racks are:

1) Mounted on the east window of the Foreman's office.

2) The laboratory.

Refer to Nuclear Safety Evaluation, Sample Racks.

## I. DESCRIPTION

The containers to be stored are described on Table A. The storage arrangement is constructed of metal troughs fastened to concrete (blo walls. Troughs are separated 8" on centers. Containers are placed side-by-side in the troughs forming rectangular parallelepipeds with cross sectional dimensions up to 2½" x 4".

## II. BASE AREA

The base area ($A_B$) will be determined using the diameters or thickness of the various containers and the separation distance. Base areas are listed below

| Material | $A_B$ |
|---|---|
| U Metal, $UO_2$ Powder, U Solutions | 102.66 $cm^2$ |
| Item, $UO_2$ Pellets | 128.32 $cm^2$ |

## III. MODEL

The storage arrangement will be evaluated using the Surface Density Model as described by Paxton in LA-3366 and Stevenson and Odegaarden in "Studies of Surface Density Spacing Criteria Using KENO Calculations" These reports discuss the method in some detail.

Data for $UO_2$ Systems were obtained from Table V k, LA-3612. This data is relisted as Table B and the critical mass data plotted on Figure 1. U Metal and U Metal-Water data was obtained from TID-7028

## IV. CALCULATIONS

### A. Metal Storage

1. Fraction Critical

From Fig. 2.4, NDEO-1050, at a density of 0.8t gm U/cc

$H/U = 30$ ($H/U^{235} = 31$)

| | |
|---|---|
| LICENSE: SNM-33 / SNM-777 | DOCKET: 70-36 / 70-932 |
| SECTION: 800 / 900 | SUBPART: 812.3 / 912.1 |
| NUCLEAR SAFETY EVALUATION SAMPLE RACKS | |
| Page 1 of 12 | |
| ISSUED: 2-6-70 | |
| SUPERSEDES: NEW | |
| APPROVED: | |
| AMENDMENT NO.: | |

As stated on page 4, NDEO-1050, the system is assumed to be thermal at this moderation ratio. Thus for a bare thermal (moderated) system is

$$\delta = 2.5 \text{ cm}$$

Using Equation (8), page 6, NDEO-1050, the buckling of a 1½" (3.81 cm) x 4' (10.16 cm) x infinite length rectangular parallelepiped is,

$$B_g^2 = \left(\frac{\pi}{a+2\delta}\right)^2 + \left(\frac{\pi}{b+2\delta}\right)^2 + \left(\frac{\pi}{c+2\delta}\right)^2 = \frac{9.87}{(3.81+5)^2} + \frac{9.87}{(10.16+5)^2} + \frac{9.87}{\infty}$$

$$= .1701 \text{ cm}^{-2}$$

Using Equation (6), page 6, NDEO-1050, and rearranging the terms, the radius of an equivalent sphere is

$$r = \sqrt{\frac{\pi^2}{B_g^2}} - \delta = \sqrt{\frac{9.87}{.1701}} - 2.5 = 5.12 \text{ cm}$$

The volume of a sphere with this radius is

$$V = \frac{4}{3}\pi r^3 = 562.37 \text{ cm}^3$$

At a density of 0.815 gm $U^{235}$/cc, the equivalent mass ($M_E$) is

$$M_E = \rho V = 0.815 \text{ gm } U^{235}/\text{cc} \times 562.37 \text{ cm}^3 = 0.458 \text{ Kg } U^{235}$$

From Fig. 8, TID-7028, the minimum critical mass for a bare sphere with a density of 0.815 gm $U^{235}$/cc is

$$M_{c,0} = 9.4 \text{ Kg } U^{235}$$

The fraction critical is

$$f = \frac{M_E}{M_{c,0}} = \frac{0.458 \text{ Kg } U^{235}}{9.4 \text{ Kg } U^{235}} = .049$$

2. Thickness Ratio

At full metal density of 18.7 gm $U^{235}$/cc, the 100 gm $U^{235}$/container will occupy an actual volume of

| | |
|---|---|
| LICENSE: SNM-777 (SNM-33) | DOCKET: 70-825 (70-33) |
| SECTION: 900 (800) | SUBPART: 917.1 (512.6) |
| NUCLEAR SAFETY EVALUATIO SAMPLE RACKS | |
| Page 2 of 12 | |
| ISSUED: 2-6-70 | |
| SUPERSEDES: New | |
| APPROVED: AMENDMENT NO.: | |

$$V_A = \frac{100 \text{ gm } U^{235}}{18.7 \text{ gm } U^{235}/cc}(.97) = 5.19 \text{ cm}^3$$

Then the actual smeared thickness is

$$t_s = \frac{V_A}{A_b} = \frac{5.19 \text{ cm}^3}{103 \text{ cm}^2} = .0504 \text{ cm}$$

From Fig. 11, TID-7028, the minimum, reflected critical thickness at full metal density is

$$t_c = 0.6 \text{ cm} = 1.52 \text{ cm}$$

The ratio is

$$t_s/t_c = \frac{.0504 \text{ cm}}{1.52 \text{ cm}} = .033$$

B. <u>ITEM Storage</u>

1. <u>Fraction Critical</u>

From Subpart 309, at a density of 5.3 gm U/cc

$$H/U = 2.2 \quad (H/U^{235} = 2.27)$$

Using Equation (17), page 25, NDEO-1050, the reflector savings for a bare system is

$$\delta = 5.46 - 0.148(H/U^{235}) = 5.46 - (.148 \times 2.27) = 5.134 \text{ cm}$$

Using Equation (8), page 6, NDEO-1050, the buckling of a 2½" (6.35 cm) x 1¾" (4.445 cm) high and infinite length rectangular parallelepiped is

$$B_g^2 = \left(\frac{\pi}{a+2\delta}\right)^2 + \left(\frac{\pi}{b+2\delta}\right)^2 + \left(\frac{\pi}{c+2\delta}\right)^2$$

$$= \frac{9.87}{(6.35+10.268)^2} + \frac{9.87}{(4.445+10.268)^2} + \frac{9.87}{\infty}$$

$$= .0812 \text{ cm}^{-2}$$

Using Equation (6), page 6, NDEO-1050, and rearranging terms, the radius of an equivalent sphere is

| | |
|---|---|
| LICENSE: SNM-777 (SNM-33) DOCKET: 70-820 (70-36) | |
| SECTION: 900 (800) SUBPART: 617.1 (812.6) | |
| NUCLEAR SAFETY EVALUATION-SAMPLE RACKS | |
| Page 3 of 12 | |
| ISSUED: 2-6-70 | |
| SUPERSEDES: New | |
| APPROVED: AMENDMENT NO.: | |

$$r = \sqrt{\frac{\pi^2}{B_g^2}} - \delta = \sqrt{\frac{9.87}{.0812}} - 5.14 = 5.856 \text{ cm}$$

The volume of a sphere with this radius is

$$V = \frac{4}{3}\pi r^3 = 841.43 \text{ cm}^3$$

At a density of 5.1 gm $U^{235}$/cc, the equivalent mass ($M_E$) is

$$M_E = \rho V = 5.1 \text{ gm } U^{235}/\text{cc} \times 841.43 \text{ cm}^3 = 4.290 \text{ Kg } U^{235}$$

From Figure 1 attached, the minimum critical mass for a bare sphere with a density of 5.1 gm $U^{235}$/cc is

$$M_{c,B} = 66 \text{ Kg } U^{235}$$

The fraction critical is

$$f = \frac{M_E}{M_{c,B}} = \frac{4.290 \text{ Kg } U^{235}}{66 \text{ Kg } U^{235}} = .065$$

2. <u>Thickness Ratio</u>

The 715 gm/container will occupy the full volume of the container so the actual volume occupied by the $U^{235}$ is

$$V_A = 140 \text{ cc } (.97) = 136 \text{ cm}^3$$

Then the actual smeared thickness is

$$t_S = \frac{V_A}{A_B} = \frac{136 \text{ cm}^3}{128.32 \text{ cm}^2} = 1.06 \text{ cm}$$

From Table B attached, the minimum reflected critical thickness at the system density of 5.1 gm $U^{235}$/cc is

$$t_c \approx 4.2 \text{ cm}$$

The ratio is

$$t_S/t_c = \frac{1.06 \text{ cm}}{4.2 \text{ cm}} = .252$$

| | |
|---|---|
| LICENSE: SNM-777 (SNM-33) | DOCKET: 70-826 (70-36) |
| SECTION: 900 (200) | SUBPART: 917.1 (1120) |
| NUCLEAR SAFETY EVALUATION SAMPLE RACKS | |
| Page 4 of 12 | |
| ISSUED: 2-6-70 | |
| SUPERSEDES: None | |
| APPROVED: | |
| AMENDMENT NO.: | |

c. $UO_2$ Pellet Storage

1. Fraction Critical

From Subpart 309, at a density of 2.5 gm U/cc

$$H/U = 5.5 \quad (H/U^{235} = 147 \text{ at } 3.75\% \text{ enrichment})$$

As stated on page 4, NDEO-1050, the system is assumed to be thermal at this moderation ratio. Thus for a bare thermal (moderated) system,

$$\delta = 2.5 \text{ cm}$$

Using Equation (8), page 6, NDEO-1050, the buckling of a 2½" (6.35 cm) x 4" (10.16 cm) x infinite length rectangular parallelepiped is

$$B_g^2 = \left(\frac{\pi}{a+2\delta}\right)^2 + \left(\frac{\pi}{b+2\delta}\right)^2 + \left(\frac{\pi}{c+2\delta}\right)^2 = \frac{9.87}{(6.35+5)^2} + \frac{9.87}{(10.16+5)^2} + \frac{9.87}{\infty}$$

$$= .1722 \text{ cm}^{-2}$$

Using Equation (6), page 6, NDEO-1050 and rearranging the terms, the radius of an equivalent sphere is

$$r = \sqrt{\frac{\pi^2}{B_g^2}} - \delta = \sqrt{\frac{9.87}{.1722}} - 2.5 = 5.07 \text{ cm}$$

The volume of a sphere with this radius is

$$V = \frac{4}{3}\pi r^3 = 546 \text{ cm}^3$$

At a density of 2.5 gm U/cc (.935 gm $U^{235}$/cc), the equivalent mass is

$$M_E = \rho V = .935 \text{ gm } U^{235}/\text{cc} \times 546 \text{ cm}^3$$

$$= 510.56 \text{ gm } U^{235}$$

From Fig. 8, TID-7028, the minimum critical mass for a bare sphere with a density of .935 gm $U^{235}$/cc is

| | |
|---|---|
| LICENSE: SNM-777 (SNM-33) | DOCKET: 70-820 (70-36) |
| SECTION: 900 (800) | SUBPART: 917.1 (916) |
| NUCLEAR SAFETY EVALUATION-SAMPLE RACKS | |
| Page 5 of 12 | |
| ISSUED: 2-6-70 | |
| SUPERSEDES: New | |
| APPROVED: | |
| AMENDMENT NO.: | |

$$M_{c,B} = 11\ K_g\ U^{235}$$

The fraction critical is

$$f = \frac{M_c}{M_{c,B}} = \frac{.511\ K_g\ U^{235}}{11\ K_g\ U^{235}} = .046$$

2. Thickness Ratio

The 1000 gm of pellets/container will occupy the full volume of the container so the actual volume occupied by the $U^{235}$ is

$$V_A = 350\ cc\ (.0375) = 13.13\ cm^3$$

Then the actual smeared thickness is

$$t_s = \frac{V_A}{A_B} = \frac{13.13\ cm^3}{128.32\ cm^2} = .1023\ cm$$

From Fig. 11, TID-7028, the minimum reflected critical thickness at the system density of 2.5 gm/cc (.935 gm $U^{235}$/cc) is

$$t_c = 1.55\ in = 3.937\ cm$$

The ratio is

$$t_s / t_c = \frac{.1023\ cm}{3.937\ cm} = .026$$

D. $UO_2$ Powder Storage

1. Fraction Critical

From Subpart 309, at a density of 4.0 gm $U^{235}$/cc

$$H/U = 3.5\ (H/U^{235} = 3.61)$$

Using Equation (17), page 25, NDEO-1030 the reflector for a bare system is

$$\delta = 5.46 - 0.148\ (H/U^{235})$$

| | |
|---|---|
| LICENSE: SNM-777 (SNM-33) | DOCKET: 70-220 (70-26) |
| SECTION: 900 (300) | SUBPART: 917.1 (F.E.6) |
| NUCLEAR SAFETY EVALUATION SAMPLE RACKS | |
| Page 6 of 12 | |
| ISSUED: 2-6-70 | |
| SUPERSEDES: None | |
| APPROVED: AMENDMENT NO.: | |

$$= 5.46 - (0.148 \times 3.61) = 4.916 \text{ cm}$$

Using Equation (8), page 6, NDEO-1050, the buckling of a 1½"(3.81 cm) x 4"(10.16 cm) x infinite length rectangular parallelepiped is

$$B_g^2 = \left(\frac{\pi}{a+2\delta}\right)^2 + \left(\frac{\pi}{b+2\delta}\right)^2 + \left(\frac{\pi}{c+2\delta}\right)^2 = \frac{9.87}{(3.81+9.832)^2} + \frac{9.87}{(10.16+9.832)^2} + \frac{9.87}{\infty}$$

$$= .0769 \text{ cm}^{-2}$$

Using Equation (6), page 6, NDEO-1050, and rearranging the terms, the radius of an equivalent sphere is

$$r = \sqrt{\frac{\pi^2}{B_g^2}} - \delta = \sqrt{\frac{9.87}{.0769}} - 4.916 = 6.404 \text{ cm}$$

The volume of a sphere with this radius is

$$V = \frac{4}{3}\pi r^3 = 1100 \text{ cm}^3$$

At a density of 4.0 gm $U^{235}$/cc, the equivalent mass is

$$M_E = \rho V = 4.0 \text{ gm } U^{235}/\text{cc} \times 1100 \text{ cc} = 4.40 \text{ Kg } U^{235}$$

From Fig. 1 attached, the minimum critical mass for a bare sphere with a density of 4.0 gm $U^{235}$/cc is

$$M_{c,0} = 57 \text{ Kg } U^{235}$$

The fraction critical is

$$f = \frac{M_E}{M_{c,0}} = \frac{4.40 \text{ Kg } U^{235}}{57 \text{ Kg } U^{235}} = .077$$

2. <u>Thickness Ratio</u>

The 560 gm U /container will occupy the full volume of the bottle so the actual volume is

$$V_A = 125 \text{ cc } (.97) = 121 \text{ cm}^3$$

The actual smeared thickness is

$$t = \frac{V_A}{A} = \frac{121 \text{ cm}^3}{[illegible]} = 1.18 \ldots$$

| | |
|---|---|
| LICENSE: SNM-33 SNM-777 | DOCKET: 70-820 70-530 |
| SECTION: 900 | SUBPART: 900 917.1 |
| NUCLEAR SAFETY EVALUATION - SAMPLE RACKS | |
| Page 7 of 12 | |
| ISSUED: 2-6-70 | |
| SUPERSEDES: New | |
| APPROVED: | |
| AMENDMENT NO.: | |

From Table A attached, the minimum, reflected critical thickness at the system density of 4.0 gm $U^{235}$/cc is

$$t_c \approx 4.6 \text{ cm}$$

The ratio is

$$t_s/t_c = \frac{1.18 \text{ cm}}{4.6 \text{ cm}} = .256$$

E. U Solution Storage

1. Fraction Critical

For solutions, the moderation ratio is

$$H/U^{235} \geq 20$$

Therefore, the buckling, sphere reflins and spherical volume calculations performed in IV.A.1 apply. Since the density of the solutions varries up to but not exceeding 1 gm $U^{235}$/cc, the fraction critical was calculated for various system densities. These calculations are summarized on Table C attached. The maximum fraction critical occurs at a density of 1.0 gm $U^{235}$/cc and is

$$f = .035$$

2. Thickness Ratio

The maximum thickness ratio occurs at a density of 1.0 gm $U^{235}$/cc and is

$$t_s/t_c = .021$$

V. CONCLUSIONS

All fraction critical values are less than 0.3 and all thickness ratios are less than 0.4. These values meet the requirements for safe surface densities and the arrangement is considered nuclearly safe.

| | |
|---|---|
| LICENSE: SNM-777 (SNM-33) | DOCKET: 70-12 (70-36) |
| SECTION: 900 (800) | SUBPART: 777 (700) |
| NUCLEAR SAFETY EVALUATION SAMPLE RACK | |
| Page 8 of 12 | |
| ISSUED: 2-6-70 | |
| SUPERSEDES: None | |
| APPROVED: | |
| AMENDMENT NO.: | |

## CONTAINER AND MATERIAL DESCRIPTION

| ATERIAL TYPE | MAXIMUM PER CONTAINER (GMS) Total | MAXIMUM PER CONTAINER (GMS) $U^{235}$ | MATERIAL DENSITY (GM/cc) | URANIUM DENSITY (GM U/cc) | CONTAINER DIMENSIONS (NOM. IN) | CONTAINER VOLUME (cc) | AVERAGE DENSITY (GM U/cc) | AVERAGE DENSITY ($GM\ U^{235}/cc$) |
|---|---|---|---|---|---|---|---|---|
| l Metal | 103 (2) | 100 | 18.7 | 18.7 | 1½×2×4 | 125 | .84 | .815 |
| . tem | 840 | 715 | 6.0 (Bulk) | 5.3 | 2½ ϕ×1¾ | 140 | 5.30 | 5.100 |
| $10_2$ Pellets (1) | 1000 (2) | 327 | 10.5 | 9.25 | 2½ ϕ×4 | 350 | 2.50 | .935 |
| $10_2$ Powder | 560 | 218 | 4.55 (Bulk) | 4.0 | 1½×2×4 | 125 | 4.00 | 3.900 |
| olution | 125 | 121 | – | 1.0 | 1½×2×4 | 125 | 1.00 | .970 |

NOTES: (1) Pellets have maximum enrichment of 3.75%. For pellets with higher enrichments, U Metal limits apply.

(2) Administrative limits; other limits are based on container capacity

TABLE A

| | |
|---|---|
| LICENSE: SNM-33 SNM 777 | DOCKET: 70-3-70-620 |
| SECTION: 700 800 | SUBPART: 712.1 |
| NUCLEAR SAFETY EVALUATION - SAMPLE RACK | |
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| ISSUED: 2-6-70 | |
| SUPERSEDES: None | |
| APPROVED: | |
| AMENDMENT NO.: | |

# CRITICAL $U(93.5)O_2$ - WATER MIXTURES

## TABLE V K, LA-3612

| | | Unreflected | | | | | | | Reflected | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Slab Thickness | | Cylinder Radius | | Sp. Rad | Vol. | Mass | Slab Thickness | | Cylinder Radius | | Sp. Rad | Vol | Mass |
| $/U^{235}$ | Density | cm | in | cm | in | cm | liters | kgs | cm | in | cm | in | cm | liters | kgs |
| 0 | 8.93 | 11.35 | 4.47 | 6.69 | 2.63 | 13.43 | 10.15 | 90.63 | 3.748 | 1.48 | 6.21 | 2.44 | 10.63 | 5.03 | 44.9 |
| .98 | 6.70 | 11.86 | 4.67 | 10.06 | 3.96 | 13.90 | 11.25 | 75.39 | 3.990 | 1.57 | 6.49 | 2.56 | 10.49 | 4.84 | 32.4 |
| .94 | 4.47 | 12.78 | 5.03 | 10.78 | 4.24 | 14.87 | 13.78 | 61.58 | 4.549 | 1.79 | 7.00 | 2.76 | 11.14 | 5.79 | 25.8 |
| .96 | 2.23 | 13.74 | 5.41 | 11.53 | 4.54 | 15.85 | 16.68 | 37.21 | 5.114 | 2.01 | 7.51 | 2.96 | 11.75 | 6.80 | 15.1 |
| .6 | 1.12 | 13.55 | 5.33 | 11.34 | 4.46 | 15.60 | 15.91 | 17.82 | 5.098 | 2.00 | 7.34 | 2.89 | 11.53 | 6.42 | 7.1 |
| 3.9 | 0.588 | 13.32 | 5.24 | 11.13 | 4.38 | 15.27 | 14.92 | 8.77 | 5.197 | 2.05 | 7.22 | 2.84 | 11.39 | 6.19 | 3.6 |

Table B

LICENSE: SNM-38 DOCKET: 70-36
SECTION: 900, SUBPART: 917.1
NUCLEAR SAFETY EVALUATION - SAMPLE RACKS
PAGE 10 OF 12
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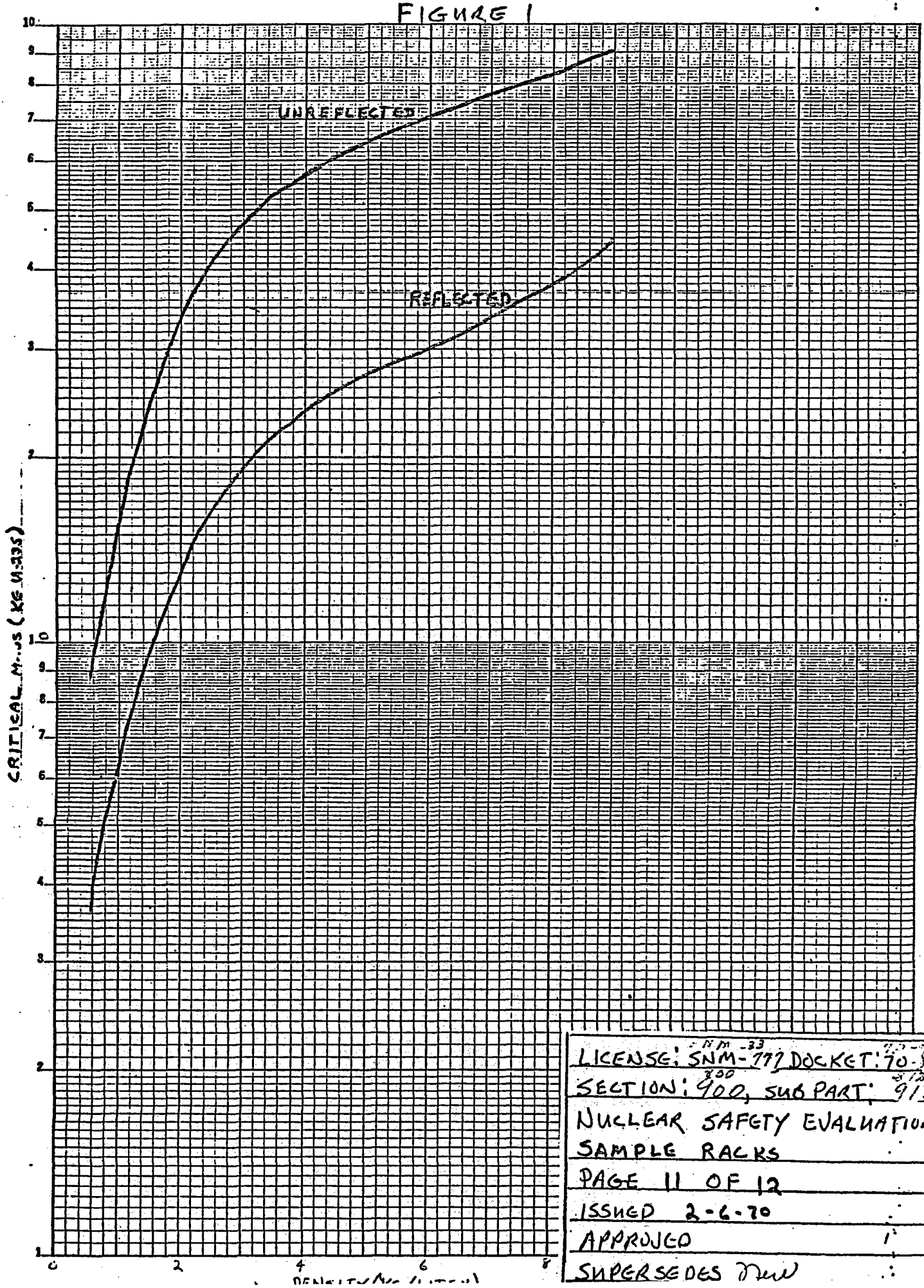
SUPERSEDES

FIGURE 1




LICENSE: SNM-777 DOCKET: 70-820
SECTION: 900, SUB PART: 917.1
NUCLEAR SAFETY EVALUATION-
SAMPLE RACKS
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ISSUED 2-6-70
APPROVED
SUPERSEDES New

Sample Solution: age f's mal $t_s/t_c$'s

| | Metal | | | | | | | $O_2$ | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| ystem m/cc | $M_A$ $Kg\,U^{235}$ | $M_{c,B}$ $(Kg\,U^{235})$ | f | $V_F$ | $t_s$ (cm) | $t_c$ (cm) | $t_s/t_c$ | $M_A$ $(Kg\,U^{235})$ | $M_{c,B}$ $(Kg\,U^{235})$ | f | $V_F$ | $t_s$ (cm) | $t_c$ (cm) | $t_s/$ |
| .2 | 1,800 | >32 | .058 | .1711 | .201 | 3.18 | .063 | .400 | ~50 | .008 | .2936 | .345 | ~5.10 | .0 |
| 1.0 | .562 | ~16 | .035 | .0535 | .063 | 4.45 | .014 | .125 | ~1.8 | .007 | .0916 | .108 | 5.10 | .0 |
| .8 | .450 | 13 | .035 | .0428 | .050 | 4.45 | .011 | .100 | ~13 | .008 | .0734 | .086 | 5.20 | .01 |
| .5 | .281 | 7 | .040 | .0268 | .031 | 4.45 | .007 | .062 | 7 | .009 | .0458 | .054 | 5.00 | .0 |
| .3 | .169 | 4.2 | .040 | .0160 | .019 | 4.62 | .004 | .037 | 4.2 | .009 | .0275 | .032 | 4.62 | .0 |
| .1 | .056 | 1.8 | .031 | .0054 | .006 | 5.84 | .001 | .013 | 1.8 | .007 | .0042 | .011 | 5.84 | .0 |

NOTES: (1) Actual Mass ($M_A$) = System Density ($\rho_{system}$) × 562.37 cc

(2) Fraction Critical (f) = $\frac{M_A}{M_{c,B}}$

(3) Bare Critical Mass ($M_{c,B}$) = $M_c$ from Fig. 8, TID-7028 for Metal OR from Fig. 1 attached for $O_2$

(4) Volume Fraction ($V_F$) = $\frac{\rho_{system}}{18.7\ gm/cc}$ (for metal) OR $\frac{\rho_{system}}{10.9\ gm/cc}$ (for $O_2$)

(5) Smeared Thickness ($t_s$) = $\frac{\text{Unit Vol.} \times \text{Enrichment} \times V_F}{A_B} = \frac{125\ cm^3 \times .97 \times V_F}{103\ cm^2}$ = 1.175 $V_F$ (cm)

(6) Critical Thickness ($t_c$) = $t_c$ from Fig. 11, TID-7028 × 2.54 $\frac{cm}{in}$ (for metal) OR from Table A attached

TABLE C

LICENSE: SNM-777 DOCKET: 70-826
SECTION: 900, SUBPART: 917.1
NUCLEAR SAFETY EVALUATION - SAMPLE RACKS
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| NO. | DESCRIPTION | | ASSOCIATED EQUIPMENT | BAY | CAPACITY |
|---|---|---|---|---|---|
| DISSOLVERS | | | | | |
| 1-J-1 | Teflon Lined Dissolver | | | XIV | |
| 1-J-4 | Unlined Dissolver | | | XIV | |
| -J-5 A, B | Tray Dissolver | | 1-L-3 A, B, C | XIV | |
| PULVERIZERS - MILLS | | | | | |
| 1-K-1 | Ball Mill | | 1-L-9A | XVII | 1/2 pint |
| 1-K-2 | Pulverizer | | 1-L-8 | XV | 50-100 lbs/hr |
| 1-K-3 | Hammer Mill | | 1-L-9B | XVII | 50-100 lbs/hr |
| 1-K-4 | Ball Mill | | 1-L-14 | XIV | 1/2 pint |
| HOODS-GLOVE BOXES | | | | | |
| 1-L-1 | Glove Box | 1-W-1 | Counting Hood | XIV | |
| 1-L-3 A-C | Hood | | Tray Dissolver | XIV | |
| 1-L-4 | Glove Box | | S.S. Dissolver | XIV | |
| 1-L-6 | Glove Box | 1-X-14 | Sample Blender | XV | |
| 1-L-7 | Hood Precipitator | 1-D-19 | Precipitator | XXI | |
| 1-L-8 | Glove Box | | Reactor Unload | XV | |
| 1-L-9 A | Hood | | Blending | XVI | |
| 1-L-9 B | Glove Box | 1-K-3 | Milling | XVI | |
| 1-L-11 | Hood | 1-H-1 | Filter Drying | XXI | |
| 1-L-13 | Glove Box | | Teflon Dissolver | XIV | |
| 1-L-14 | Hood | 1-K-4 | Milling Drying | XIV | |
| 1-L-16 A | Hood | 1-H-2 | Drying | XXI | |
| 1-L-16 B | Glove Box | 1-H-2 | Loading | XXI | |
| 1-L-17 | Hood | | Declading | XVII | |
| 1-L-18 | Hood | | All Purpose | XVII | |
| 1-L-20 | Glove Box | 1-H-1 | Reactor Loading | XXI | |
| PUMPS | | | | | |
| 1-P-1 | $NH_4OH$ Feed Pump | | Outside | | 15 GPM |
| 1-P-2 A, B | Extractor Feed Pumps | | 1-D-41 | XVI | 0-1 |
| 1-P-3 | Process Waste Transfer Pump | | 1-D-14 | XIX | 25 |
| 1-P-4 | Raffinate Pump | | | XX | 1 |
| 1-P-5 | Filter Feed Pump | | 1-D-20 | XXI | 5 |
| 1-P-6 | Adj. Liquor Transfer Pump | | 1-D-12 | XIX | 10 |
| 1-P-8 | Dissolver Pump, Teflon | | 1-J-1 | XVII | 10 |
| 1-P-17 | Filtrate Recovery Pump | | 1-D-24 | XVIII | 10 |
| 1-P-20 | $HNO_3$ Pump | | Outside | | 15 |
| 1-P-21 | Slop Pump | | Outside | | 10 |
| 1-P-23 | Conc. Liquor Feed Pump | | | XXI | 0-2 |
| 1-P-28 | Raffinate Pump | | | XX | |

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922.2 Glove Box 1-L-1

One as received container and one process container will be in this box at a time. The scrap will be transferred into the process container in the required process batch quantity. This will be weighed on scale 1-W-1. The maximum size process container will be one gallon. Material having a density greater than 3.2 grams of U-235 per cubic centimeter will be handled in containers where the total volume will be restricted to 3.6 liters and a mass content of 3.6 kgs U-235.

Access to the glove box is through a double door airlock. The size of this airlock restricts the glove box to containers of a size only comparable to the one gallon process container.

Nuclear safety is assured since the "as received" container was subcritical during shipment, subdividing its contents into the process container will serve to further reduce the reactivity of the material.

Other materials for calcining or further processing may be loaded into trays or into safe geometry containers, for example, high density oxide to be reduced in density by calcination. This is done by loading into trays that do not exceed 2" depth, 8" width for insertion into the calcining furnace. The cross sectional array of a calciner tray (2" x 8") is less than that of a 5" diameter cylinder.

## INTERACTION SUMMARY 1-L-18 HOOD

| Item | Max. Keff | Reference | Allowable Solid Angle |
|---|---|---|---|
| 4" Columns | .425 | Figure 4 K-1317 | 4.75 |
| Funnel 10.5" dia. x 4" high | .623 | Figure 4 K-1317 ($\lambda = \frac{4}{10.5} = 0.381$) | 2.77 |
| 9" dia. x 4" high | .572 | | 3.28 |
| 4-liter flask | .63 | Assume equivalent to 7" ID cyl. 6.34" high ($\lambda = 0.906$) conservative since 8" ID cyl. or 6" ID cyl. of same volume have Keff = .605 and .59 respectively | 2.7 |
| Filter Press | .670 | Equiv. to 8.18" dia. cyl. 6" high | 2.3 |
| 5" dia. 11-liter bottle in cart | .58 | | 3.2 |
| 1 gal. bottle | .596 | | 3.04 |
| Standpipe (replaces funnel) 5" dia. x 15" | $\angle$.58 | | 3.2 |

If the central column is the centermost unit, the total interaction is 2.437 steradians. Similarly, treating the filter press as the centermost unit, the total interaction is 1.677 steradians.

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## INTERACTION CALCULATIONS FOR 1-L-18 HOOD

$\Omega$ at central glass column

$\Omega = \frac{2d}{h} \sin \theta$

$d = 4''$
$h = 29-2 = 27''$
$L = 48; L/2 = 24''$

$\tan \theta = .8889$

$\sin \theta = .6644$

$\Omega = \frac{2\ (4)}{27} (.6644) = 0.197$ steradians/column

2 columns, $\Omega = 0.394$ steradians

However, representing each 4" column with associated filter flasks and funnels as 9" diam cylinders with a height equal to that of the hood = 77"

then $d = 9''$
$h = 29-4.5 = 24.5''$
$L/2 = 38.5$ in

$\tan \theta = \frac{38.5}{24.5} = 1.571 \quad \sin \theta = 0.8436$

$\Omega = \frac{2\ (9)}{24.5} (0.8436) = 0.620$ steradians/column

$\Omega$ due to adjacent column = 1.24 steradians max

Contribution from assay tanks

assume $\infty$ long $\quad d = 5''$
$h = 15.5\ (9) + 15.5-25 = 152.5$ in

$\Omega = \frac{2d}{1} (1) = .066$ steradians/tank

$\Omega$ due to 3 tanks $\approx 0.198$ steradians

Contribution from an 11 liter bottle in a cart placed directly in front of hood. Cart frame insures 13.5" from edge of frame to surface of bottle.

$d = 5'' \quad h = 15.5 + 13.5 = 29''$
$L = 48'' \quad L/2 = 24''$

$\tan \theta = \frac{24}{29} = 0.8276 \quad \sin \theta = 0.6376$

$\Omega = \frac{2(5)}{29} (0.6376) = 0.220$; 2 bottles in 2 carts = 0.44 steradians

Contribution from funnel or standpipe outside hood funnel.

Funnel:

$d = 10'', L = 4'', L/2 = 2'' \quad h = 11''$

$\tan \theta = \frac{L/2}{h} = \frac{2''}{11''} = .182 \quad \sin \theta = .179$

$\Omega = \frac{20''}{11''} (.179) = .326$ steradians

Standpipe:

$d = 5'', L = 16'' \ L/2'' = 9'', \quad h = 11''$

$\tan \theta = \frac{L/2}{h} = \frac{8''}{11''} = .73 \quad \sin \theta = .587$

$\Omega = \frac{10''}{11''} (.587) = .534$ steradians

The standpipe contributes the most.

Total $\Omega = 1.24 + .198 + .440 + .534 = 2.412$ steradians

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Ertel Filter

$\Omega$ at Ertel Filter:

Assume filter is a squat cylinder for solid angle calculations

$d = 7.25\sqrt{2} = 10.25''$ $L = 6''$ $L/2 = 3''$

36" distance (h) to left hand column (representing 4 liter beaker and flask 6" dia. x 78" tall)

33" distance (h) to right hand column representing 4" column and associated funnel and flask (9" dia. x 78" tall)

$\Omega$ from LH $= \frac{2\ (6)\ (.735)}{36} = 0.245$

$\Omega$ from RH $= \frac{2\ (9)\ (.768)}{33} = 0.417$

$\Omega$ from possible 2 11-liter bottles in carts = .44

$\Omega$ from standpipe outside hood $\leq$ .534

Total $\Omega$ = .245 + .417 + .440 + .534 = 1.636 steradians

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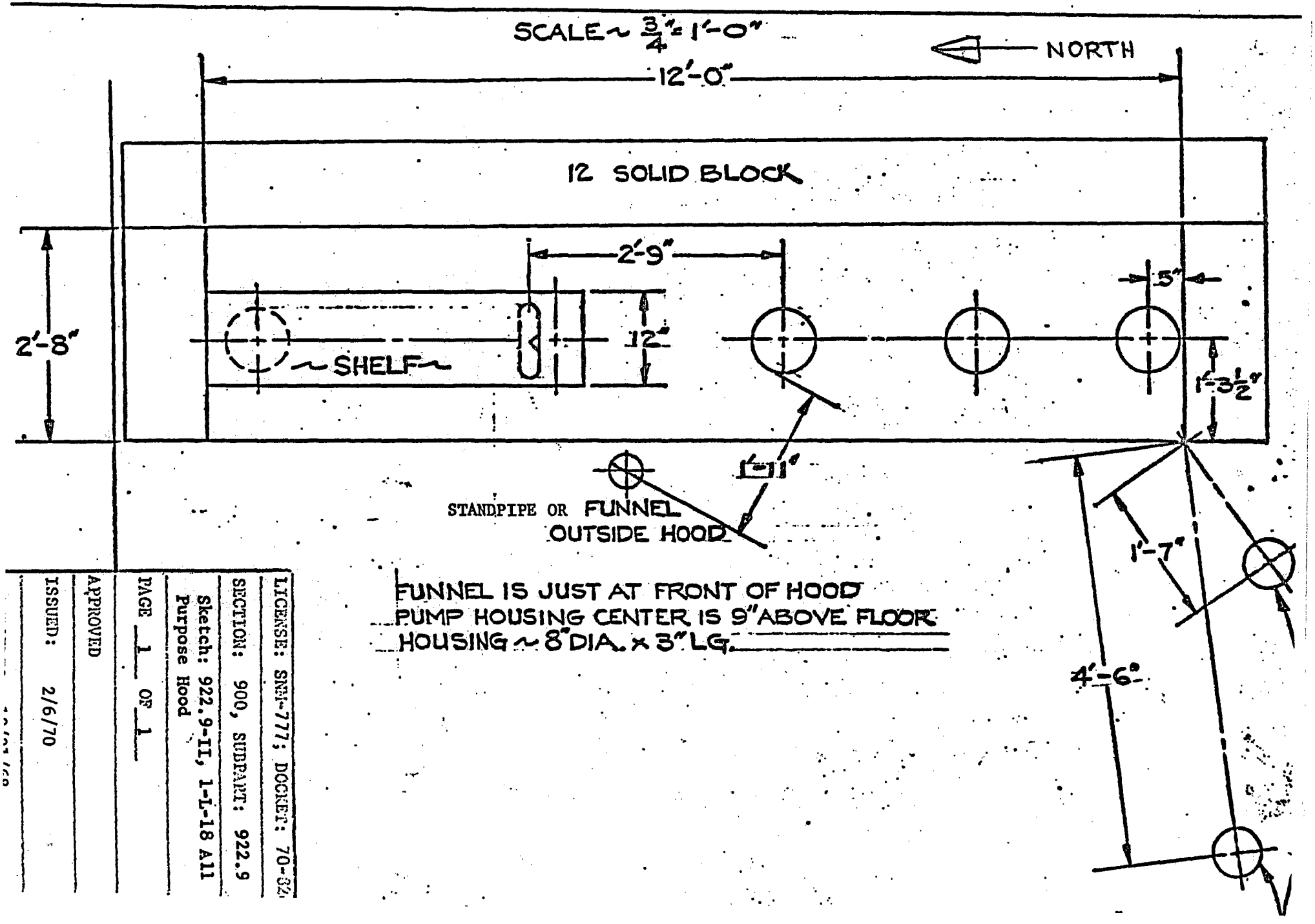


FUNNEL IS JUST AT FRONT OF HOOD
PUMP HOUSING CENTER IS 9" ABOVE FLOOR
HOUSING ~ 8" DIA. x 3" LG.

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| Sketch: 922.9-II, 1-L-18 All Purpose Hood | |
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## 922.14 Boat Calciner System

### 1. General

The drying and/or calcining of liquid and solid residues containing uranium will be done in the equipment shown on Sketch 922.14-IL. No known material with explosive or pyrophoric characteristics will be processed in this system. This system is located in Bays X, XIII and XVI as shown on Figure 922.14-III.

The equipment consists of an inlet and exit glove box, two tube furnaces and a return tube for empty furnace boats. The equipment is built as an enclosed loop, the single return tube services both furnaces. This is shown on Figure 922.14-IL.

### 2. Process

The process consists of loading individual boats in glove box 2-L-4, manually pulling the boats through the furnace tube into glove box 2-L-5, unloading the ash into a standard container, and returning the empty boat to glove box 2-L-4 by manually pulling it through the return tube.

Containers used for introducing and unloading the material are standard one (1) gallon bottles (actual capacity 4.0 liters) or the "as-received" container in which the raw scrap was received.

Combustion gases are filtered through a porous metal filter, cooled by air or nitrogen dilution, filtered through a high efficiency filter and exhausted to the atmosphere.

### 3. Equipment and Nuclear Safety

#### 3.1 Inlet Glove Box 2-L-4

A single safe volume, safe goemetry, or as received container of feed is introduced into the glove box for loading into furnace boats. Due to the method of operation, no more than one boat will be in the glove box at one time. Since the operation does not require (nor is it practicable) the lifting of the boats off the floor of the glove box, the possibility of stacking or spilling boats has been neglected.

The safety of the hood is assured since only one feed container and one furnace boat (cross sectional area less than an equivalent 5" diameter cylinder) are allowed in the glove box.

3.2 Furnace System

Dimensions of the furnace tube, boat return tube and boats are set forth in Table 922.14-I. All have cross sectional areas less than that of a 5" diameter schedule 40 pipe.

3.3 Exit Glove Box 2-L-5

The discussion of Subpart 922.14.3.1 above, is applicable here.

In addition, the material entering this glove box will have been through the furnace and is therefore unmoderated.

3.4 Product Collection and Storage Bottle

The calcined residue will be scraped from the boats and transferred to a safe volume bottle, not exceeding 4.0 liter capacity, through a funnel in the floor of the Outlet Glove Box. This is shown on Sketch 922.14-II.

3.5 Balance Table

A balance for weighing the resultant product has been located approximately 3 feet from glove box 2-L-5. The filled product bottle is weighed on this scale prior to transfer to storage. As such only one bottle is at the scale or in the filling position.

3.6 Combustion Gas Exhaust System

The gases will exit from the 2-L-4 glove box end of the furnaces through 2" diameter pipe, a porous metal filter, to a 12 x 12 x 12 absolute filter thence to the exhaust fan on the roof and be released to the atmosphere. This is shown on Figure 922.14-II.

The glove boxes 2-L-4 and 2-L-5 are maintained at a negative pressure relative to the furnaces and as such are exhausted through the furnace off gas system.

3.7 Nuclear Safety

Refer to attached Nuclear Safety Evaluation: Boat Calciner System.

I. Description

A. The Furnace Tubes and Return Tube are 5" dia x 18' long on 18½" centers.

B. Furnace Boats are 3 3/8" wide x 3¼" high x 24" long with a 4.25 liter volume

C. Containers will be 6" diameter--4 liter volume.

II. Assumptions

A. Material to be processed and in process will have densities not exceeding 4 Kg $U^{235}$ per liter.

B. Interaction will be evaluated assuming both the North Furnace Tube and the South Furnace Tube as the centermost unit.

C. Interaction will be calculated using the solid angle method.

III. Individual Units

A. 5" schedule 40 pipe.

1. This is a .05" larger than standard safe diameter for U (93.5) oxide-water systems as listed on Table 309-I. Since material is contained in boats with equivalent cross section area less than 5" diameter and furnace is dry with no water connections, an adequate margin of safety is maintained.

B. 6" diameter--4 liter volume

1. This is a safe volume for U (93.5) oxide-water systems as listed on Table 309-I.

C. $K_{eff}$ bare of most reactive unit

1. The most reactive unit is the 6" dia.--4 liter container. From the "Nuclear Safety Evaluation--$K_{eff}$'s for 6" Ø - 4 liter $UO_2$ cylinder", attached,

$$K_{eff} \text{ (bare)} = .73$$

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## IV. Interaction Calculations

A. North Furnace Tube

1. Contribution From South Furnace Tube

$d = 5''$, $L = 18' = 216''$, $L/2 = 108''$, $h = 34''$.

$$\Omega_1 = \frac{2d}{h} \sin \theta \qquad \text{where } \tan \theta = \frac{L/2}{h} = \frac{108''}{34''} = 3.18$$

$$= \frac{10''}{34''} (.954) \qquad \sin \theta = .954$$

$$= .281$$

$\Omega_1$(Total) = .281 steradians

2. Contribution from boat in hoods 2-L-4 and 2-L-5

$d$ = height of boat = 3.25", $L$ = length of boat = 24", $L/2$ = 12"
$h \approx 9' = 108''$.

$$\Omega_2 = \frac{2d}{h} \sin \theta \qquad \text{where } \tan \theta = \frac{L/2}{h} = .058$$

$$\sin \theta = .111$$

$= .007$ steradians

For boat in each hood 2 $\Omega_2 = .014$.

3. Contribution from containers on hoods 2-L-5

$d = 6''$, $L = 10''$, $L/2 = 5''$, $h \approx 9' = 108''$

$$\Omega_3 = \frac{2d}{h} \sin \theta \qquad \text{where } \tan \theta = .046$$

$$\sin \theta = .046$$

$= .005$ steradians

4. Contribution from porous metal filter

$d = 5''$, $L = 3' = 36''$, $L/2 = 18''$, $h = \sqrt{(108)^2 + (38)^2} = 114$

$$\Omega_4 = \frac{2d}{h} \sin \theta \qquad \text{where } \tan \theta = .158$$

$$\sin \theta = .156$$

$= .014$ steradians

5. Contribution from absolute filters

Neglect, since any dust carry over in exhaust is removed by porous metal filter.

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6. Contribution from container in the 1-L-14, 2-L-2, 1 L 1, 2-L-5 hoods.

Assume 6" dia., 4 liter bottle in each.

$d = 6''$, $L = 10''$, $L/2 = 5''$, $h = 4.5' = 54''$
for 1 L 14:

$$\Omega_6 = \frac{2d}{h} \sin\theta \qquad \text{where } \tan\theta = \frac{L/2}{h} = .093$$
$$= .021 \text{ steradians} \qquad \sin\theta = .093$$

Assume equal contribution by 2 L 2 & 1 L 1

$\Omega_6$ (Total) $= 4 \times \Omega_6 = .08$ steradians

7. Contribution from DDU Retort 2-R-1

$a = 21''$, $b = 40''$, $h = 10' = 120''$, $h^2 = 14{,}400$

$a/2 = 11.5''$, $(a/2)^2 \approx 132$, $b/2 = 20''$, $(b/2)^2 = 400$

$$\Omega_9 = 4 \sin^{-1} \frac{(a/2)\ (b/2)}{\sqrt{(a/2)^2 + h^2}\ \sqrt{(b/2)^2 + h^2}}$$

$= .064$ steradians

$\Omega_9$ (Total) $= 1 \times \Omega_9 = .064$ steradians

8. Contribution by 1 D 41

Neglect since U concentration is 1 D 41 is less than 5 grams per liter.

9. Contribution form overhead storage tanks, 1-D-9 (6 tanks).

$d = 5''$, $L = 45'$, $L/2 = 22.5'$, $h = 10' = 120''$

$$\Omega_{11} = \frac{24}{h} \cdot \sin\theta \qquad \text{where } \tan\theta \approx \infty$$
$$= \frac{10''}{120''} = .083 \qquad \sin\theta = 1$$

$\Omega_{11}$ (Total) $= 6 \times \Omega_{11} = 6 \times .083 = .498$ steradians

10. Based on above calculations, other equipment items are sufficiently separated that this contribution can be neglected

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11. Total Interaction

$\Omega_T = \Omega_1$ (Total) $+ \ldots + \Omega_8$ (Total) = .96 steradians

This total represents interaction at one of the furnace tubes; it is larger than solid angle at the 4 liter bottle, located in any of the near by hoods. The 4 liter bottle is the most reactive (highest $K_{eff}$), therefore, it is conservative to compare this total solid angle (.96 steradians) to the allowable for the 4 liter volume 1.7 steradians;

Allowable = 9 - 10 k = 1.7 steradians.

A. This container is referenced to as the "one gallon container" or "one gallon bottle" throughout the licenses.
B. This container has a measured volume of 4 liters and a measured diameter of 6" (r=7.62cm).
C. The maximum density (bulk) of material to be placed in these containers will not exceed .4 Kg/U liter.

## I. ASSUMPTIONS

A. The maximum "crystal" density of material to be placed in these containers will not exceed that of $UO_2$ (9.66 Kg U/liter).
B. The $UO_2$-Water data of LA-3612 is applicable.
C. The calculational method set forth in Section 2.2, NDEO-1050, will be used to determine reflector savings (δ) and keff values.
D. The density vs. H/U-235 ratio relationship will be per Fig. 309-XXIV.
E. The effective height is:

$$h = \frac{V}{\pi r^2} = \frac{4000\,cc}{\pi(7.62cm)^2} = 21.93\ cm$$

## II. CALCULATIONS

See attached tables

## IV. CONCLUSIONS

This container is subcritical when moderated and reflected for materials with densities not exceeding 4 Kg U/liter. The keff values at a density of 4 Kg U/liter are:

keff (reflected) = .925

keff (bare) = .730

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| NUCLEAR SAFETY EVALUATION | |
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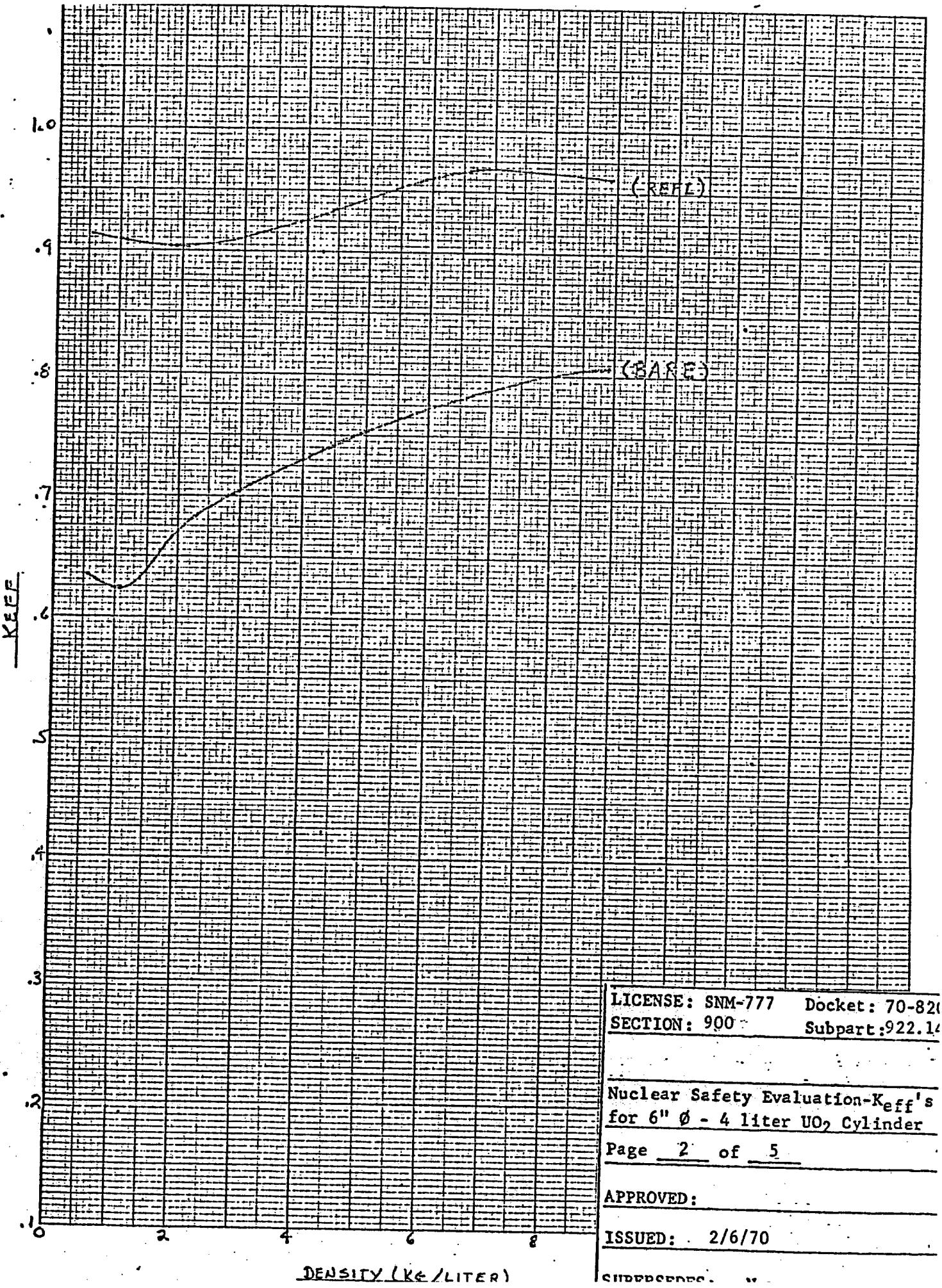


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Nuclear Safety Evaluation-$K_{eff}$'s for 6" Ø - 4 liter $UO_2$ Cylinder

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# KEFF VALUES FOR 6" DIAMETER - 4 LITER CYLINDRICAL VOLUME $U(93.5)O_2$ - WATER MIXTURES

| H/U-235 | $\rho$ (Kg/l) | $\delta$ (1) (cm) | $B^2_{S,A}$ (2) ($cm^{-2}$) | $\tau$ (3) ($cm^2$) | keff (4) | $\delta$ (5) (cm) | $B^2_{S,B}$ (2) ($cm^{-2}$) | $\tau$ (3) ($cm^2$) | keff (4) |
|---|---|---|---|---|---|---|---|---|---|
| 0 | 8.930 | 7.30 | .0334 | 28.0 | .961 | 5.46 | .0429 | 28.0 | .807 |
| 0.98 | 6.700 | 7.26 | .0335 | ↓ | .968 | 5.32 | .0438 | ↓ | .785 |
| 2.94 | 4.470 | 7.18 | .0339 | ↓ | .935 | 5.03 | .0457 | ↓ | .744 |
| 8.96 | 2.230 | 6.94 | .0350 | ↓ | .905 | 4.13 | .0527 | ↓ | .683 |
| 20.60 | 1.120 | 6.50 | .0371 | ↓ | .908 | 2.50 | .0700 | ↓ | .623 |
| 43.90 | 0.558 | 6.50 | .0371 | ↓ | .913 | 2.50 | .0700 | ↓ | .634 |

NOTES: (1) $\delta = 7.3 - .040(H/U\text{-}235)$, cm; for thick water reflector at $H/U\text{-}235 \leq 20$; $\delta = 6.5$ cm at $H/U\text{-}235 > 20$

(2) $B^2 = \frac{J_0^2}{(r+\delta)^2} + \frac{\pi^2}{(h+2\delta)^2}$ where $r = 3'' = 7.62$ cm, $h = 21.43$ cm

(3) $\tau = 28$ cm from Section 2.2, NDEO-1050

(4) $k_{eff} = \frac{1+\tau B_c^2}{1+\tau B_s^2}$ where $B_c^2$ are listed in the Nuclear Safety Evaluation, Keff's of 4.8 Liter $UO_2$ Sphere, Subpart 309.

(5) $\delta = 5.46 - .148(H/U\text{-}235)$, cm; for bare systems at $H/U\text{-}235 \leq 20$; $\delta = 2.5$ cm at $H/U\text{-}235 > 20$

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NUCLEAR SAFETY EVALUATION - KEFF'S FOR 6" Φ - 4 LITER $UO_2$ CYL



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## REFLECTED

At H/U-235 = 0, $\ell$ = 8.930 Kg/ℓ

$$B^2 = \frac{5.78}{(7.62+7.3)^2} + \frac{9.87}{(21.93+14.6)^2} = \frac{5.78}{(14.92)^2} + \frac{9.87}{(36.53)^2}$$

$$= \frac{5.78}{222.61} + \frac{9.87}{1334} = .0260 + .0074 = .0334$$

At H/U-235 = .98, $\ell$ = 6.700 Kg/ℓ

$$B^2 = \frac{5.78}{(7.62+7.26)^2} + \frac{9.87}{(21.93+14.52)^2} = \frac{5.78}{(14.88)^2} + \frac{9.87}{(36.45)^2}$$

$$= \frac{5.78}{221.41} + \frac{9.87}{1329} = .0261 + .0074 = .0335$$

At H/U-235 = 2.94, $\ell$ = 4.470 Kg/ℓ

$$B^2 = \frac{5.78}{(7.62+7.18)^2} + \frac{9.87}{(21.93+14.36)^2} = \frac{5.78}{(14.80)^2} + \frac{9.87}{(36.29)^2}$$

$$= \frac{5.78}{219.04} + \frac{9.87}{1317} = .0264 + .0075 = .0339$$

At H/U-235 = 8.96, $\ell$ = 2.230 Kg/ℓ

$$B^2 = \frac{5.78}{(7.62+6.94)^2} + \frac{9.87}{(21.93+13.88)^2} = \frac{5.78}{(14.56)^2} + \frac{9.87}{(35.81)^2}$$

$$= \frac{5.78}{211.99} + \frac{9.87}{1282} = .0273 + .0077 = .0350$$

At H/U-235 = 20.60, $\ell$ = 1.120 Kg/ℓ

$$B^2 = \frac{5.78}{(7.62+6.5)^2} + \frac{9.87}{(21.93+13)^2} = \frac{5.78}{(14.12)^2} + \frac{9.87}{(34.93)^2}$$

$$= \frac{5.78}{199.37} + \frac{9.87}{1220} = .0290 + .0081 = .0371$$

At H/U-235 = 43.90, $\ell$ = 0.558 Kg/ℓ

$B^2$ = .0371 (same as for $\ell$ = 1.120)

At H/U-235 = 43.90, $\ell$ = 0.558 Kg/ℓ

$B^2$ = .0700 (same as for $\ell$ = 1.120)

## BARE

At H/U-235 = 0, $\ell$ = 8.920 Kg/ℓ

$$B^2 = \frac{5.78}{(7.62+5.46)^2} + \frac{9.87}{(21.93+10.92)^2} = \frac{5.78}{(13.08)^2} + \frac{9.8}{(32.8}$$

$$= \frac{5.78}{171.09} + \frac{9.87}{1079} = .0338 + .0091 = .04$$

At H/U-235 = .98, $\ell$ = 6.700 Kg/ℓ

$$B^2 = \frac{5.78}{(7.62+5.32)^2} + \frac{9.87}{(21.93+10.64)^2} = \frac{5.78}{(12.94)^2} + \frac{9.8}{(32.5}$$

$$= \frac{5.78}{167.44} + \frac{9.87}{1061} = .0345 + .0093 = .043$$

At H/U-235 = 2.94, $\ell$ = 4.470 Kg/ℓ

$$B^2 = \frac{5.78}{(7.62+5.03)^2} + \frac{9.87}{(21.93+10.06)^2} = \frac{5.78}{(12.65)^2} + \frac{9.8}{(31.9}$$

$$= \frac{5.78}{160.02} + \frac{9.87}{1023} = .0361 + .0096 = .04$$

At H/U-235 = 8.96, $\ell$ = 2.230 Kg/ℓ

$$B^2 = \frac{5.78}{(7.62+4.13)^2} + \frac{9.87}{(21.93+8.26)^2} = \frac{5.78}{(11.75)^2} + \frac{9.8}{(30.1}$$

$$= \frac{5.78}{138.06} + \frac{9.87}{911} = .0419 + .0108 = .05$$

At H/U-235 = 20.60, $\ell$ = 1.120 Kg/ℓ

$$B^2 = \frac{5.78}{(7.62+2.5)^2} + \frac{9.87}{(21.93+5)^2} = \frac{5.78}{(10.12)^2} + \frac{9.8}{(26.9}$$

$$= \frac{5.78}{102.41} + \frac{9.87}{725} = .0564 + .0136 = .07$$

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NUCLEAR SAFETY EVALUATION - KEFF'S FOR 6"ϕ - 4 LITER $UO_2$ CYL

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APPROVED:

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REFLECTED

At H/u-235 = 0, $\rho$ = 8.93 Kg/l

$$k_{eff} = \frac{1+(28X.0367)}{1+(28X.0334)} = \frac{1.860}{1.935} = .961$$

At H/u-235 = .98, $\rho$ = 6.700 Kg/l

$$k_{eff} = \frac{1+(28X.0313)}{1+(28X.0335)} = \frac{1.876}{1.938} = .968$$

At H/u-235 = 2.94, $\rho$ = 4.470 Kg/l

$$k_{eff} = \frac{1+(28X.0294)}{1+(28X.0339)} = \frac{1.823}{1.949} = .935$$

At H/u-235 = 8.96, $\rho$ = 2.230 Kg/l

$$k_{eff} = \frac{1+(28X.0283)}{1+(28X.0350)} = \frac{1.792}{1.980} = .905$$

At H/u-235 = 20.60, $\rho$ = 1.120 Kg/l

$$k_{eff} = \frac{1+(28X.0304)}{1+(28X.0371)} = \frac{1.851}{2.039} = .908$$

At H/u-235 = 43.90, $\rho$ = 0.558 Kg/l

$$k_{eff} = \frac{1+(28X.0305)}{1+(28X.0371)} = \frac{1.862}{2.039} = .913$$

BARE

At H/u-235 = 0, $\rho$ = 8.93 Kg/l

$$k_{eff} = \frac{1+(28X.0277)}{1+(28X.0422)} = \frac{1.776}{2.201} = .807$$

At H/u-235 = .98, $\rho$ = 6.700 Kg/l

$$k_{eff} = \frac{1+(28X.0267)}{1+(28X.0438)} = \frac{1.748}{2.226} = .785$$

At H/u-235 = 2.94, $\rho$ = 4.470 Kg/l

$$k_{eff} = \frac{1+(28X.0249)}{1+(28X.0457)} = \frac{1.697}{2.280} = .744$$

At H/u-235 = 8.96, $\rho$ = 2.230 Kg/l

$$k_{eff} = \frac{1+(28X.0247)}{1+(28X.0527)} = \frac{1.692}{2.476} = .683$$

At H/u-235 = 20.60, $\rho$ = 1.120 Kg/l

$$k_{eff} \frac{1+(28X.0261)}{1+(28X.0700)} = \frac{1.843}{2.960} = .623$$

At H/u-235 = 43.90, $\rho$ = .0558

$$k_{eff} = \frac{1+(28X.0313)}{1+(28X.0700)} = \frac{1.876}{2.960} = .634$$

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NUCLEAR SAFETY EVALUATION-
KEFF'S FOR 6"-4 LITER U.O. CY
PAGE 5 OF 5

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GEOMETRY OF SUBCRITS

| Subcrit | Configuration (Cross Section) | Volume | Cross Section Area (Inside) | Equiv. Dia. |
|---|---|---|---|---|
| Furnace Tube | 5" Diameter schedule 40 pipe | | 18.2 sq. in. | 5.05" |
| Boat Return Tube | 4½" wide x 4¼" high | | 19.1 sq. in. | 4-9/16" |
| Furnace Boats | 3-3/8" wide x 3¼" high x 24" long | 4.25 liters | 10.95 sq. in. | 3-3/4 |
| Furnace and Hood Filters | 12 x 12 x 12 | | | |
| Unloading Hood Discharge Chute | Conical - 8" I.D. at top x 2" I.D. at bottom x 8¼" high | 3.04 liters | | |
| Containers | 6" O.D. x 11-1/8" overall height | 4.0 liters | | |

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Table 922.14-I

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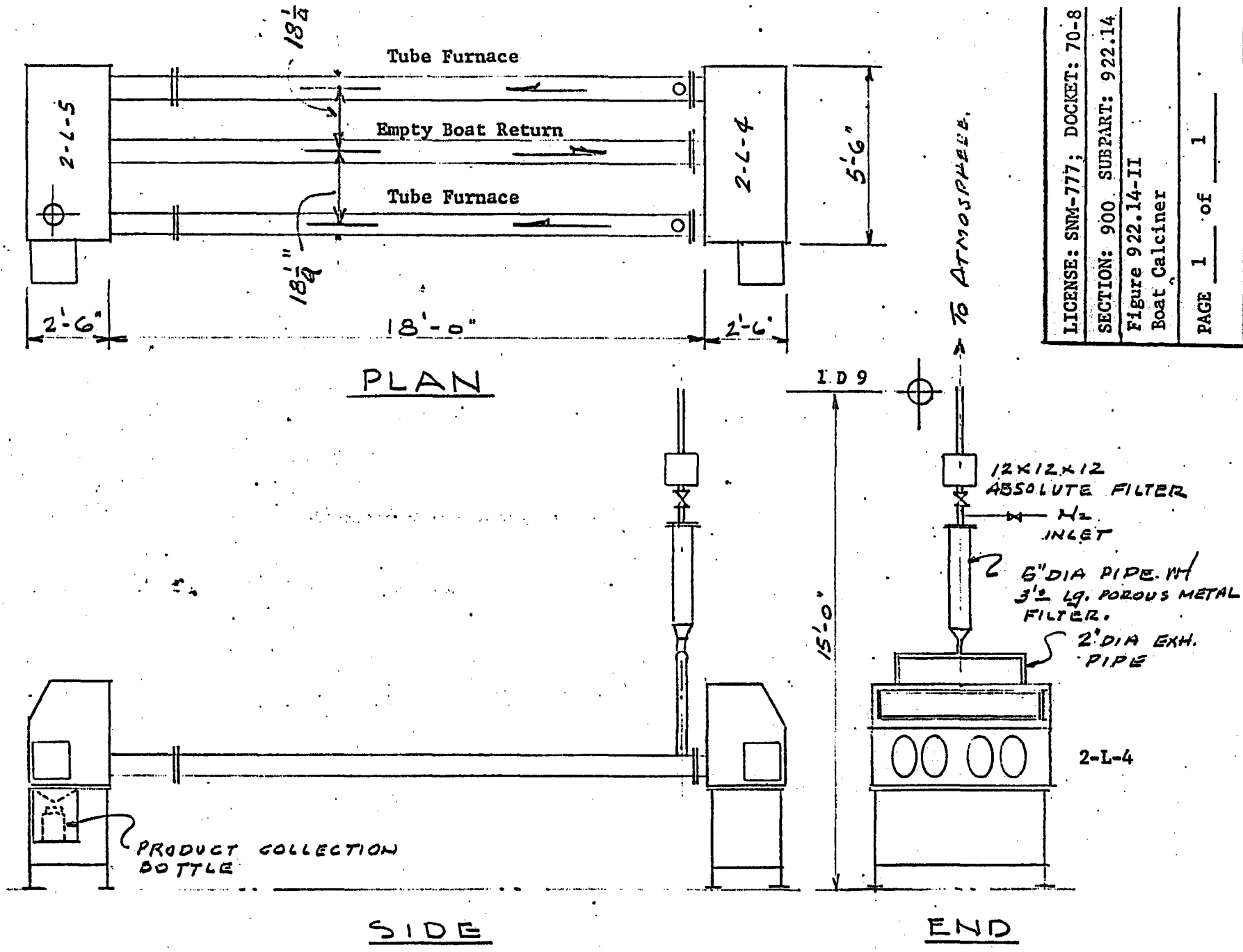


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| Figure 922.14-II Boat Calciner | |
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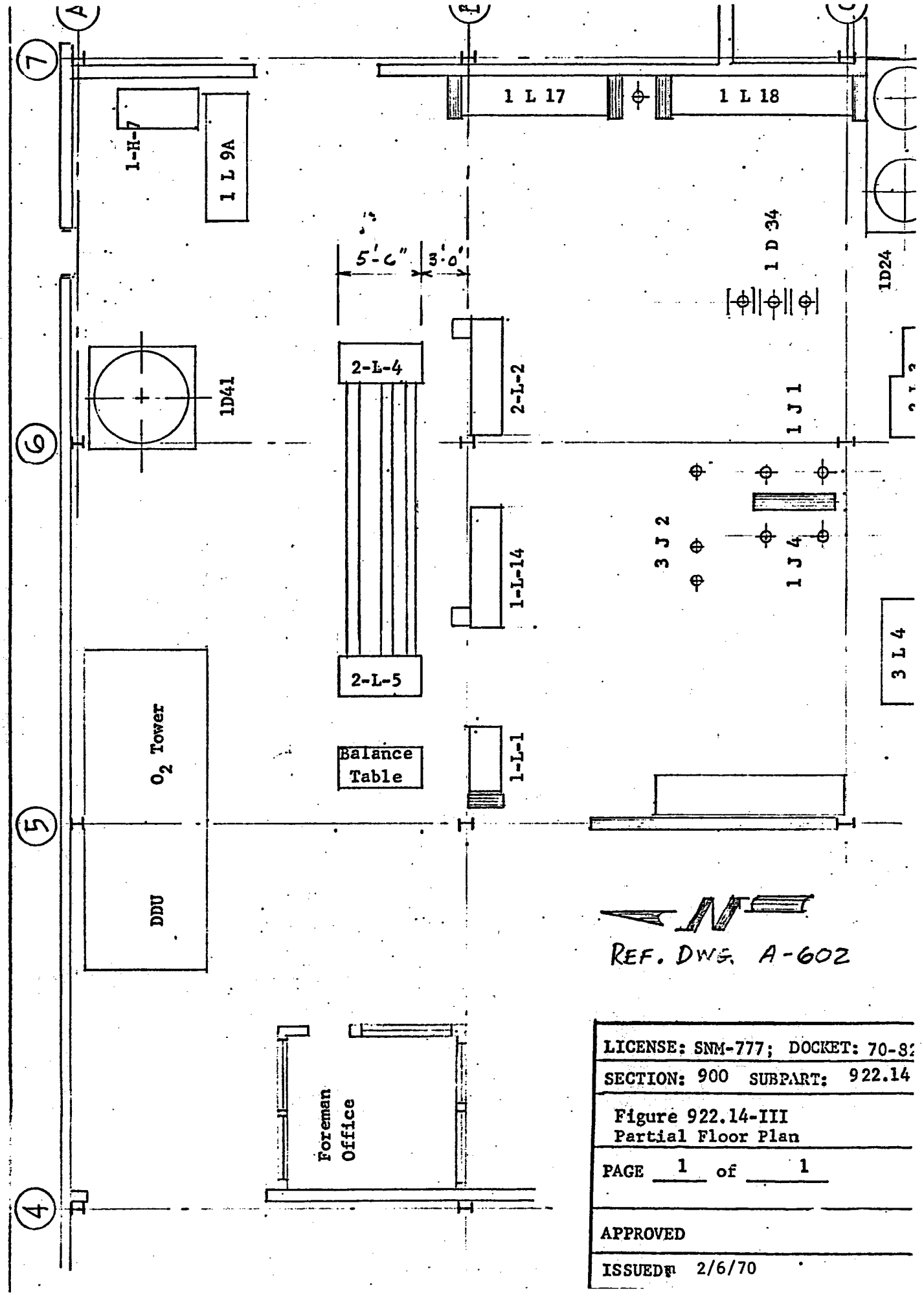


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Figure 922.14-III
Partial Floor Plan

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## 922.16 Wet Cut-Off Saw

### 1. General

* A modified power hack saw located in the Maintenance Shop may be used to disassemble a fuel element (unit). The saw is modified by the addition of a sump drain, settling bulb (capacity one quart) and a 5 gallon reservoir tank for coolant liquid. The coolant tank will be packed with Raschig Rings. A recirculating pump will provide coolant flow from the coolant tank to the part in the hack saw. The coolant flows by gravity from the saw table through the settling bulb back to the reservoir.

### 2. Nuclear Safety

Only one unit at a time will be processed. Disassembled pieces will be immediately removed, cleaned, wiped dry and transferred to storage or introduced to the process line.

The coolant will be limited to not more than 350 grams U-235. This will be accomplished by checking the settling bulb for building of solids and subsequent clean out. This check and cleanup will be performed at least daily when the equipment is in use. However, it is expected that most of the solids will be retained on the saw table which will be cleaned off after each individual use. The coolant tank will be checked with a Gamma Portable Monitor to note material buildup.

*Indicates Change

LICENSE: SNM-777, Docket: 70-820
SECTION: 900 - RECOVERY OPERATION
Subsection: 920 - Processing
Subpart: 922 - Preparation of Solid Scraps
922.17 - Milling Machine

Approved

ISSUED 2/6/70

SUPERSEDES 10/31/68

## 922.17 Milling Machine

### 1. General

* A wet milling machine located in the Maintenance Shop may be used to disassemble a fuel element (unit), remove excess cladding, etc. This mill is modified to use the Wet Cut Off Saw Coolant System as described in Subpart 922.15. Also, a coolant system consisting of 4 - 5" diameter pipes, 36" long, spaced a minimum of 18" center-to-center apart may be used. This array of pipes will be located a minimum of 4' away from the milling machine. This alternate coolant system is shown in Sketch 922.17-I.

### 2. Nuclear Safety

Same as Subpart 922.15. The pipe coolant system will have a solid angle of .88 steradians. From Table XVII, K-1019, Rev. 5, a 5" cylinder has an allowable interaction of 3.2 steradians.

Contribution From Nearest Pipes

d=5", L=36", L/2=18", h=18" + 2.5" = 20.5"

$$\Omega = \frac{2d}{h} \sin \theta \qquad \text{where } \tan \theta = \frac{L/2}{h} = \frac{18''}{20.5''} = .88$$

$$= \frac{10''}{20.5''}(.67) = .49 \times .67 \qquad \sin \theta = .67$$

$$= .326$$

Contribution From Furthest Pipes

d=5", L=36", L/2=18", $h = \sqrt{18''+18''} = \sqrt{648} = 25.4''$

$$\Omega = \frac{2d}{h} \sin \theta \qquad \text{where } \tan \theta = \frac{L/2}{h} = \frac{18''}{25.4} = .71$$

$$= \frac{10''}{25.4''}(.58) = .394 \times .58 \qquad \sin \theta = .58$$

$$= .228$$

Total Interaction

$$\Sigma \Omega = .326 + .326 + .228 = .880 \text{ steradians}$$

*Indicates Change

## 922.18 PEXTO Splitter

### 1. General

* This splitter is used to remove fuel plates from a fuel element (unit) which is roll swaged together, etc. It is also located in the Maintenance Shop.

### 2. Nuclear Safety

Only one unit will be processed at a time. Disassembled pieces will be immediately transferred to storage or introduced to the process line.

*Indicates Change

## 922.19 Alligator Shear

### 1. General

* This shear is used to cut fuel plates or elements. It is located in the Maintenance Shop.

### 2. Nuclear Safety

Only one unit or plate will be processed at a time. The cut pieces will be immediately transferred to storage or introduced to the process area.

*Indicates Change

## 923.2 Tray Dissolvers System

### 1. General

The tray dissolvers are typically used for the dissolution of solid uranium bearing scrap. A batch is weighed, measured and packaged in Glove Box 1-L-1 or Hoods 1-L-14, 1-L-9A, 2-L-2, or 2-L-3, (ref: Subpart 922.2, 922.3, 922.4 and 922.10 respectively).

The batch is transferred by hand to the tray dissolver where it is spread uniformly over the surface of the tray. A measured quantity of acid or other solvent is sprayed onto the material filling the tray. When dissolution is complete, it is adjusted as necessary for extraction and then drained by gravity to Filter 1-F-6 and from the filter by vacuum to Holding Tank 1-D-36. When Tank 1-D-36 is filled,
* the solution is transferred to the Assay Tanks 1-D-36A thru G or 3-D-12

### 2. Equipment and Nuclear Safety

#### 2.1 Tray Dissolvers 1-J-5 A and B

The trays are 4 inches high by 26-1/4" wide outside dimensions. A drain pipe is located near one corner of the tray for solution and solid removal.

The dissolution batch will be limited to 0.887 kg U-235.

All solids will be removed from the tray before recharging. Solution and solids are drained from the tray through a one inch pipe.

The maximum solid angle calculated for the tray dissolver (1-J-5 A and B) is less than 0.5 steradians.

The nuclear safety of these trays is demonstrated in the Nuclear Safety Evaluation for the Tray Dissolvers

#### 2.2 Filter 1-F-6

This filter is similar to the dissolvers, being 21" square and 2" deep. It is located in Hood 1-L-3B. Nuclear safety is assured by the safe slab height. The only solids collected on the filter will be the insolubles from the dissolution step. These will be removed as necessary, insolubles not to exceed one inch depth, packaged in a one gallon polyethylene bottle and transferred to storage. The filtrate is drained through a one inch pipe to 1-D-36.

The solid angle calculated at the filter 1-F-6 is less than 0.6 steradians.

2. Equipment and Nuclear Safety (continued)

2.3 Tank 1-D-36

This tank is 5 inches in diameter and hence nuclearly safe. The tank is filled by means of vacuum transfer from the pan filter and is emptied by gravity or low pressure air. A moisture alarm is installed in the vacuum line to detect any liquid entrained to the vacuum system.

2.4 Fume Scrubber 2-X-1 (See Sketch 923.2-II)

Vapors, dust and mist from the 1-L-3 hoods are washed in a 14" diameter vertical venturi type scrubber with internal spray nozzles. A horizontal pan 45" long by 20" wide and 6" sloping to 12" in depth serves as a plenum for separating the liquid from the scrubbed gas. This pan drains to a 5" diameter vertical sump approximately 5' long. The liquid level in this sump is assured by a 1-1/2" diameter overflow pipe return to the tray 1-J-5. A vertical demister section in the exhaust duct from the plenum removes entrained droplets from the gas.

The sump is 5" in diameter. The overflow position prevents buildup of solution into the plenum chamber. Frequent inspection and/or cleaning of the plenum chamber will prevent any accumulation of material. The sump will be periodically drained and recharged with
* fresh solution to insure a concentration of less than 5 g U/l. The intersection of the sump and overflow pipe has an effective diameter of 3.6" determined by the technique of page 20, TID-7016, Rev. 1 and is safe as per Table III, TID-7016, Rev. 1.

2.5 Hood 1-L-3 A, B, C

These hoods contain the Tray Dissolvers 1-J-5 A and B. Fumes from these hoods are washed in the Fume Scrubber 2-X-1.

2.6 Physical Air Breaks

Chemical reagent tanks of unsafe geometry will be separated from the uranium process stream by a physical air break.

2.7 Beaker or Bottle Storage

A fixed beaker or bottle storage rack will be installed under the Hoods 1-L-3 A, B, and C. The rack will be approximately 30" from the bottom of the hoods, 6" off the floor, 12" from the wall and 12" from the front of the hoods.

The rack will accommodate up to eight covered containers of 1 gallon volume or less. Bottle or beaker stations will be located on 16" centers.

The containers are a safe volume and the centermost bottle subtends a solid angle of 1.96 steradians and has an allowable solid angle of 2.36 steradians. Details of the rack location are shown on Sketch 923.2-I.

*Indicates Change

$= \frac{10''}{36''}(.316) = .278 \times .316$ $\sin\theta = .316$

$= .088$

4. Contribution from Hold Tank 3-D-10

d=5", L=15' (assume L=∞ and sin θ = 1), h = 50"

$\Omega_4 = \frac{2d}{h}\sin\theta = \frac{10''}{50''} = .200$

5. Contribution from 11 liter Over-Flow Bottle

d=5", L=48", L/2=24", h=36"

$\Omega_5 = \frac{2d}{h}\sin\theta$ where $\tan\theta = \frac{L/2}{h}$

$= \frac{10''}{36''}(.56)$ $= \frac{24''}{36''}$

$= .276 \times .56$ $= .67$

$= .156$ $\sin\theta = .56$

6. Total Interaction

$\Omega$ (Total) = $\Omega_1$ (Total) +...+ $\Omega_5$ = 1.075 steradians

7. Allowable Interaction

Under normal conditions, the most reactive case occurs with a 16" ϕ x 4" high cylinder (see Nuclear Safety Evaluation - Tray Dissolver, Subpart 923.2). The actual volume of a 16" ϕ x 4" high cylinder is:

V = 13.2 liters

The charge of SNM is 1 kg $U^{235}$, so the actual density is:

$\rho A = \frac{1 \text{ kg } U^{235}}{13.2 \text{ liters}} = .0256 \text{ kg } U^{235}/\text{liter}$

The effective multiplication factor will be calculated using the technique set forth in K-1380, Part I.

The corresponding moderation ratio is
$H/U^{235} = 350$ Fig. I-4

Then
$\eta f = 1.785$ Fig. I-5

$B^2$(fast)=.043 +.011=.054cm$^2$ Fig.I-6

$U_f = .37$ Fig. I-7

$B^2$(thermal)=.083+.0135=.0965 cm$^2$ Fig.I-8

$U_t = .897$ Fig. I-9

$Keff = \eta f U_f U_t = .592$

The allowable interaction is

$\Omega A$ = 9-10K - 3.08 steradians

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## 924.2 Pickle Liquor System

### 1. Process

Solution that has been received and accepted for processing is transported from the outside storage area set aside for pickle liquor storage to inside the facility, Bay XIX. These solutions are received in 55 gallon drums. The uranium concentration is normally a maximus of 2 grams per liter. The contents of the drum are transferred to the Raw Liquor Tank 1-D-12, (90 gallon capacity) by pouring with the aid of the drum tilter attachment of the monorail hoist, or pumping or syphoning from the drum.

The pickle liquor is treated by mixing with aluminum compounds if HF is present. It is then neutralized by addition of ammonium hydroxide to yield a solution approximately 3 molar with respect to nitric acid.

If required ammonium hydroxide is transferred to the adjustment tank from a drum located outside the building.

The treated pickle liquor is transferred to the Extractor Feed Tank 1-D-41, (1500 gallon capacity, 1200 gallons net capacity due to Boron Pyrex Rasching Rangs) through Filter 1-F-1 which is 4" diameter by 13-1/8" long.

### 2. Equipment and Nuclear Safety, Pickle Liquor System

The nuclear safety of the start of the pickle liquor process is assured by limiting the U-235 concentrations to less than 5 grams per liter (normally less than 2 grams per liter for pickle liquor) and limiting each batch in the Treatment Tank 1-D-12 to 350 grams U-235.

Controls against double batching Treatment Tank 1-D-12 exist since the capacity of the tank is 90 gallons. The 55 gallon pickle liquor batch plus the required neutralizing volume of chemicals will exceed half the volume of the tank.

The quantity of ammonium hydroxide used is determined for neutralizing but not to cause precipitation. Should precipitation occur, the 350 gram batch will assure safety.

Extractor Feed Tank 1-D-41 safety is assured by the less than 5 gram per liter concentration. In addition, the tank is filled with Boron Rasching Rings as backup safety. The Filter 1-F-1 is safe geometry and will filter any insolubles and precipitate that may inadvertently occur in Treatment Tank 1-D-12.

Since the solutions processed in Tanks 1-D-12 and 1-D-41 are less than 5 grams U-235 per liter interaction effects on and by these tanks can be neglected.

## 926.4 ADU Precipitators 1-D-19 A,B &C and 1-D-20 A,B,C & D

Feed for the precipitators is transferred from the OK Liquor Storage Tanks 1-D-10 A,B and C, and the 1-D-16 as concentrated solution. A pre-determined amount of solution is fed to the precipitators, three separate tanks in this system, and anhydrous ammonia is bubbled through the solution at a controlled rate. Agitation is induced by the addition of air.

Precipitation being complete, the slurry is filtered through a Buchner Filter. The filter cake is washed with distilled water which has been added to the precipitator and treated with anhydrous ammonia. The filtrate
* is directed through a 4" diameter x 13-1/8" polishing filter 1-F-15 to the filtrate tanks 1-D-24 A&B or is collected in 4 liter flasks and transferred to eleven liter bottles.

These precipitators have 5" diameters and the 1-D-19's are 5' long and the 1-D-20's are 8' long.

Nuclear Safety of the precipitators is assured by the 5" diameter tanks. The maximum solid angle subtended in 1-D-20 B is less than one steradian.

*Indicates Change

## 924.6 Aluminum Nitrate System

### 1. General

Aluminum nitrate is normally stored outside in the 1-D-1 Aluminum Nitrate Tank. As an alternate method, aluminum nitrate will be dissolved or generated in the 1-D-28, 500 gallon Aluminum Nitrate Make-Up Tank. No uranium bearing material will be permitted in these tanks.

* The aluminum nitrate solution is transferred by Pump 3-P-4 to the 1-D-6A, 400 gallon Aluminum Nitrate Storage Tank on the third floor of Bay XX. This 1-D-6A was originally a strip acid tank.

The aluminum nitrate storage feeds by gravity from the 1-D-6A Aluminum Nitrate Storage Tank to all the aluminum nitrate headers throughout the plant. Air gaps are incorporated into the piping between all aluminum nitrate headers, and the equipment they serve to prevent any nuclear solutions from entering these headers.

*Indicates Change

## 926.13 ADU Drying and Unloading (Hoods 1-L-16 A,B&C & Drying Ovens 1-H-1,2&8)

1. General

After filtration, the ADU cake (still on the Buchner Filter Funnel) is dried. This drying, and the subsequent unloading of the filter funnels, is performed in a work station formed by the 1-L-16 A,B &C hoods with the 1-H-1, 1-H-2 and 1-H-8 Drying ovens attached to the backs of these hoods. The Buchner Filter Funnels are introduced into the drying ovens through the open face front of the 1-L-15 A or C hoods. Dried ADU is scraped or placed in a funnel at the bottom of the glove box. This funnel directs the dried ADU into a 1 gallon bottle attached to the bottom of the glove box. Filled bottles of ADU are transferred to the Reactor Loading Glove Box 1-L-20 or to storage. Sketch 926.13-I shows the arrangement of the work station.

* 2. Nuclear Safety

Nuclear safety is provided by having the SNM in a bottle or funnel with a volume not exceeding one gallon. This is a safe volume as shown on Table 390-I. Only one funnel may be in transit at any one time. Therefore, there will be a maximum of one funnel in each of the drying ovens, one funnel in transit and one funnel and one bottle in the unloading hood. Individual funnels will be on 16" centers and the funnel and bottle in the 1-L-16B unloading hood are on 10" centers. The solid angle subtended by the center-most funnel in the 1-H-8 drying oven is approximately 1 steradian.

*Indicates Change

I. Description

A. The funnel in the 1-H-8 oven is the centermost unit.
B. Each station will have one 10" diameter x 4" high filter funnel or one six inch diameter by 10" high one gallon bottle in it.
C. Material will be highly enriched ADU.

II. Assumptions

A. Individual filter funnels are safe as per Table XV, K-1019, Rev. 5.
B. One gallon bottles are safe as per Table 309-I.
C. Keff will be determined using Figure 4, K-1317.

III. Keff Value

The Keff of a filter funnel in the 1-H-8 oven will be approximately 0.6.

diameter = 10", height = 4"

$$\lambda = \frac{\text{height}}{\text{diameter}} = 0.4$$

From Figure 4, K-1317

Keff = 0.6

IV. Interaction Calculations

A. Contribution from funnel in 1-H-1

diameter = 10", lenght = 4", L/2 = 2", height = 21"

$$\tan\theta = \frac{L/2}{h} = .095$$
$$\sin\theta = .094$$

$$\Omega_1 = \frac{2\,d}{h}\sin\theta = .087 \text{ steradians}$$

B. Contribution from funnel in 1-H-2

Same as for 1-H-1

$\Omega_2$ = .087 steradians

C. Contribution from funnel in 1-L-16B

d = 10", L/2 = 2", h = 54"

$$\tan\theta = \frac{L/2}{h} = .037$$
$$\sin\theta = .036$$

$$\Omega_3 = \frac{2\,d}{h}\sin\theta = .013 \text{ steradians}$$

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IV. Interaction Calculations (continued)

D. Contribution from one gallon bottle in bottom of 1-L-16B

$d = 6''$, $L = 10''$, $L/2 = 5''$, $h = 66''$

$$\tan\theta = \frac{L/2}{h} = 0.76$$
$$\sin\theta = .075$$
$$\Omega_4 = \frac{2d}{h} \sin\theta = .013 \text{ steradians}$$

E. Contribution from funnel in 1-L-16A

$d = 10''$, $L/2 = 2''$, $h = 32''$

$$\tan\theta = \frac{L/2}{h} = .063$$
$$\sin\theta = .062$$
$$\Omega_5 = \frac{2d}{h} \sin\theta = .039 \text{ steradians}$$

F. Contribution from funnel in 1-L-16C

$d = 10''$, $L/2 = 2''$, $h = 11''$

$$\tan\theta = \frac{L/2}{h} = .182$$
$$\sin\theta = .179$$
$$\Omega_6 = \frac{2d}{h} \sin\theta = .326$$

G. Contribution from 1-D-19's

$d = 5''$, $L = 8' = 96''$, $L/2 = 48'$, $h = 8.5' = 102''$

$$\tan\theta = \frac{L/2}{h} = .47$$
$$\sin\theta = .425$$
$$\Omega(1 \text{ unit}) = \frac{2d}{h} \sin\theta = .0416$$

Since 3 precipitators

$$\Omega_7 = 3 \times \Omega(1 \text{ unit}) = .125 \text{ steradians}$$

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IV. Interaction Calculations (continued)

H. Contribution from Overhead Tanks

$d = 5''$, $L = 60' = 720''$, $L/2 = 360''$, $h \approx 15' = 180''$

$$\tan\theta = \frac{L/2}{h} = 2$$

$$\sin\theta = .895$$

$$\Omega\,(1\text{ tank}) = \frac{2\,d}{h}\sin\theta = .05$$

Since 6 tanks

$$\Omega_8 = 6 \times \Omega(1\text{ tank}) = .3 \text{ steradians}$$

I. Contribution from other surrounding equipment.

All other equipment is greater than 15 feet away and are either one-gallon bottles, 11 liter bottles or similar small shapes or volumes. Therefore, their interaction effects are considered negligible and are neglected.

J. Total Interaction

$$\Omega_t = \Sigma\Omega_1 + \cdots + \Omega_8 \approx 1.0 \text{ steradians}$$

K. Allowable Interaction

$$\Omega_A = 9\text{-}10K = 9\text{-}6.0 = 3 \text{ steradians}$$

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| Nuclear Safety Evaluation |
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## I. Description

A. No more than 3 trays per muffle box.

B. Each muffle furnace can hold one muffle box.

C. Muffle trays are 8¼" x 15 5/8" x 1¼".

D. The maximum density (bulk) of material in these trays will not exceed 4 kg U/liter (3.29 kg U-235/liter) and does not exceed the density-H/U ratio or the compound of Figure 309 XXIV.

## II. Assumptions

A. The $UO_2$ water data of LA-3612 is applicable.

B. The calculation method set forth in Section 2.2, NDEO-1050 will be used to determine reflector savings ($\delta$) and Keff values.

C. The effective thickness of SNM will be determined assuming the three trays are stacked one on top of the other. So,

width = 8¼" = 20.96 cm, length = 15 5/8" = 39.69 cm,
thickness = 3 x 1¼" = 9.53 cm

## III. Calculations

See attached tables.

## IV. Conclusions

Three muffle trays are subcritical when moderated and reflected for materials not exceeding 4 kg U/liter. The Keff values at a density of 4 kg U/liter are:

Keff (reflected) = .987

Keff (bare) = .79

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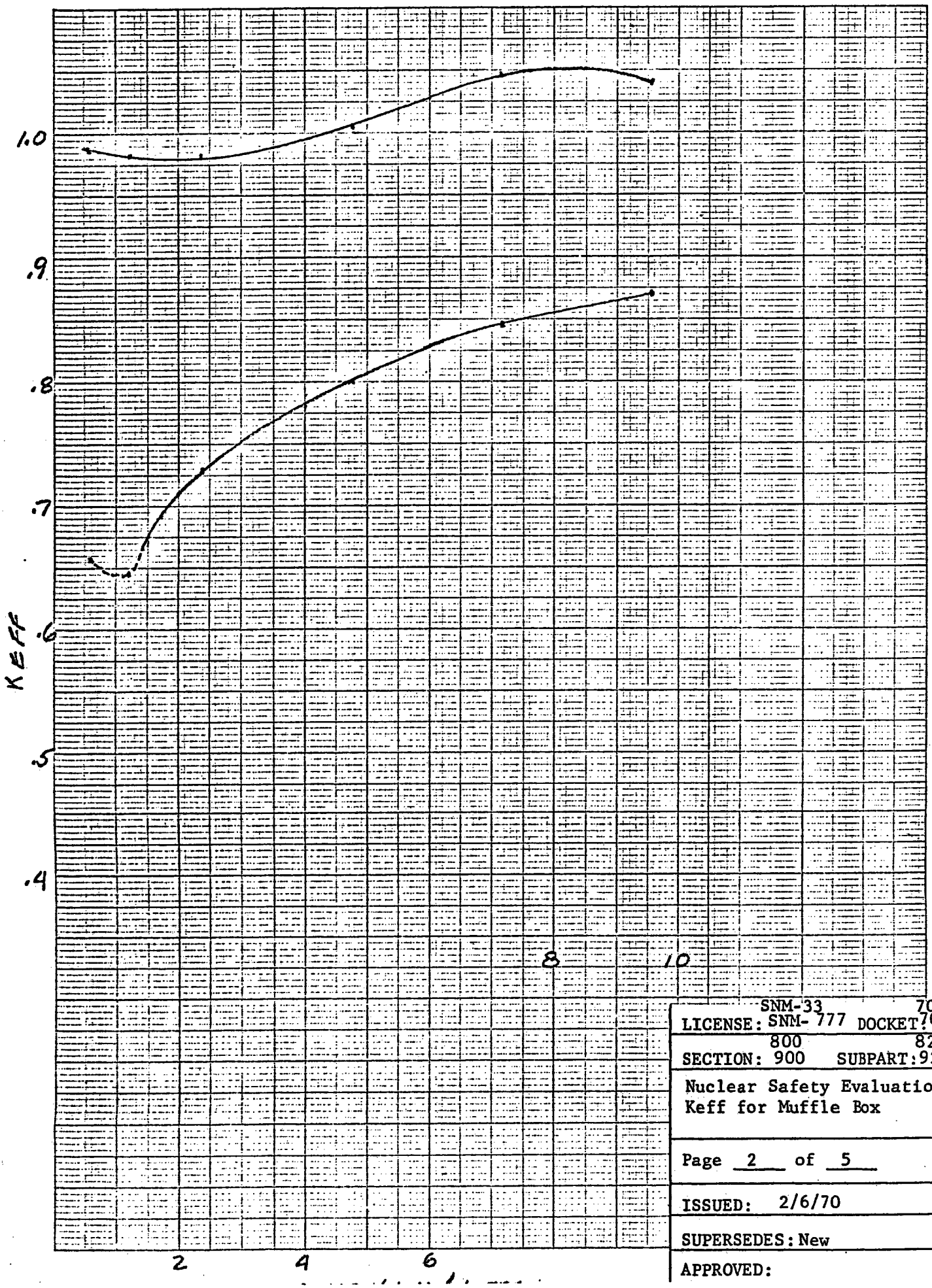


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Keff Values for ' -ee Muffle Trays
(20.96 cm x 39.69 cm x 9.53 cm)
with $U(93.5)O_2$ - Water

| H/U-235 | Density U-235 (Kg./L) | Density Total U Kg. U/L | $\delta_r$ (1) (cm) | $B^2_{s,r}$ (2) ($cm^{-2}$) | T (3) ($cm^2$) | Keff (4) | $\delta_b$ (5) (cm) | $B^2_{s,b}$ (2) ($cm^{-2}$) | T (3) ($cm^2$) | Keff (4) |
|---|---|---|---|---|---|---|---|---|---|---|
| 0 | 8.930 | 9.55 | 7.30 | 0.0281 | 28.0 | 1.041 | 5.46 | 0.0312 | 28.0 | .870 |
| 0.98 | 6.700 | 7.16 | 7.26 | 0.0283 | 28.0 | 1.047 | 5.32 | 0.0381 | 28.0 | .846 |
| 2.94 | 4.470 | 4.79 | 7.18 | 0.0290 | 28.0 | 1.006 | 5.03 | 0.0400 | 28.0 | .800 |
| 8.96 | 2.230 | 2.39 | 6.94 | 0.0295 | 28.0 | .981 | 4.13 | 0.0471 | 28.0 | .730 |
| 20.64 | 1.120 | 1.2 | 6.50 | 0.0316 | 28.0 | .982 | 2.50 | 0.0663 | 28.0 | .645 |
| 43.90 | 0.558 | ..596 | 6.50 | 0.0316 | 28.0 | .988 | 2.50 | 0.0663 | 28.0 | .657 |

NOTES: (1) $\delta_r = 73 - .040$ (H/U-235), cm; for thick water reflector at H/U-235 $\leq 20$; $\delta_r$ 6.5 cm at H/U-235 $> 20$

(2) $B_s^2 = \frac{\pi^2}{(\text{width} + 2\ )^2} + \frac{\pi^2}{(\text{thick} + 2\ )^2} + \frac{\pi^2}{(\text{length} + 2\ )^2}$ where width = 20.96 cm, length = 39.69 cm, thick = 9.53 cm

(3) T = 28 cm from Section 2.2 NDEO-1050

(4) $\text{Keff} = \frac{1 + T B_c^2}{1 + T B_s^2}$ where $B_c^2$ (critical bucklings) are listed in the Nuclear Safety Evaluation, Keff's of 4.8 liter $UO_2$ Sphere, Subpart 309

(5) $\delta_b = 5.46 - .148$ (H/U-235), cm; for bare systems at H/U-235 $\leq 20$; $\delta_r = 2.5$ cm at H/U-235 $> 20$

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# MUFFLE TRAY BUCKLINGS

## EFLECTED

At $H/U\text{-}235 = 0$, $\rho = 8.930\ K_g/\ell$

$$B^2 = \frac{9.87}{(20.96+14.6)^2} + \frac{9.87}{(39.69+14.6)^2} + \frac{9.87}{(9.53+14.6)^2} = \frac{9.87}{(35.56)^2} + \frac{9.87}{(57.29)^2} + \frac{9.87}{(24.13)^2}$$

$$= \frac{9.87}{1264.51} + \frac{9.87}{2947.40} + \frac{9.87}{582.26} = .0078 + .0033 + .0170 = .0281$$

At $H/U\text{-}235 = .98$, $\rho = 6.700\ K_g/\ell$

$$B^2 = \frac{9.87}{(20.96+14.52)^2} + \frac{9.87}{(39.69+14.52)^2} + \frac{9.87}{(9.53+14.52)^2} = \frac{9.87}{(35.48)^2} + \frac{9.87}{(54.21)^2} + \frac{9.87}{(24.05)^2}$$

$$= \frac{9.87}{1258.83} + \frac{9.87}{2938.72} + \frac{9.87}{578.40} = .0078 + .0034 + .0171 = .0283$$

At $H/U\text{-}235 = 2.94$, $\rho = 4.470\ K_g/\ell$

$$B^2 = \frac{9.87}{(20.96+14.16)^2} + \frac{9.87}{(39.69+14.16)^2} + \frac{9.87}{(9.53+14.16)^2} = \frac{9.87}{(35.12)^2} + \frac{9.87}{(53.85)^2} + \frac{9.87}{(23.69)^2}$$

$$= \frac{9.87}{1233.41} + \frac{9.87}{2899.82} + \frac{9.87}{561.22} = .0080 + .0034 + .0176 = .0290$$

At $H/U\text{-}235 = 8.96$, $\rho = 2.230\ K_g/\ell$

$$B^2 = \frac{9.87}{(20.96+13.88)^2} + \frac{9.87}{(39.69+13.88)^2} + \frac{9.87}{(9.53+13.88)^2} = \frac{9.87}{(34.84)^2} + \frac{9.87}{(53.57)^2} + \frac{9.87}{(23.41)^2}$$

$$= \frac{9.87}{1213.83} + \frac{9.87}{2869.74} + \frac{9.87}{548.03} = .0081 + .0034 + .0180 = .0295$$

At $H/U\text{-}235 = 20.60$, $\rho = 1.120\ K_g/\ell$

$$B^2 = \frac{9.87}{(20.96+13)^2} + \frac{9.87}{(39.69+13)^2} + \frac{9.87}{(9.53+13)^2} = \frac{9.87}{(33.96)^2} + \frac{9.87}{(52.69)^2} + \frac{9.87}{(22.53)^2}$$

$$= \frac{9.87}{1153.28} + \frac{9.87}{2776.24} + \frac{9.87}{507.60} = .0086 + .0036 + .0194 = .0316$$

At $H/U\text{-}235 = 43.90$, $\rho = 0.558\ K_g/\ell$

$B^2 = .0316$ (same as for $\rho = 1.120$)

---

At $H/U\text{-}235 = 43.90$, $\rho = 0.558\ K_g/\ell$

## BARE

At $H/U\text{-}235 = 0$, $\rho = 8.93\ K_g/\ell$

$$B^2 = \frac{9.87}{(20.96+10.92)^2} + \frac{9.87}{(39.69+10.92)^2} + \frac{9.87}{(9.53+10.92)^2} = \frac{9.87}{(31.88)^2} + \frac{9.87}{(50.61)^2} + \frac{9.8}{(20.}$$

$$= \frac{9.87}{1016.33} + \frac{9.87}{2561.37} + \frac{9.87}{418.20} = .0097 + .0039 + .0236 = .03$$

At $H/U\text{-}235 = .98$, $\rho = 6.700\ K_g/\ell$

$$B^2 = \frac{9.87}{(20.96+10.64)^2} + \frac{9.87}{(39.69+10.64)^2} + \frac{9.87}{(9.53+10.64)^2} = \frac{9.87}{(31.60)^2} + \frac{9.87}{(50.33)^2} + \frac{9.8}{(20.1}$$

$$= \frac{9.87}{998.56} + \frac{9.87}{2533.11} + \frac{9.87}{406.83} = .0099 + .0039 + .0243 = .03$$

At $H/U\text{-}235 = 2.94$, $\rho = 4.470\ K_g/K$

$$B^2 = \frac{9.87}{(20.96+10.06)^2} + \frac{9.87}{(39.69+10.06)^2} + \frac{9.87}{(9.53+10.06)^2} = \frac{9.87}{(31.02)^2} + \frac{9.87}{(49.75)^2} + \frac{9.37}{(19.59}$$

$$= \frac{9.87}{962.24} + \frac{9.87}{2475.06} + \frac{9.87}{383.77} = .0103 + .0040 + .0257 = .04$$

At $H/U\text{-}235 = 8.96$, $\rho = 2.230\ K_g/\ell$

$$B^2 = \frac{9.87}{(20.96+8.26)^2} + \frac{9.87}{(39.69+8.26)^2} + \frac{9.87}{(9.53+8.26)^2} = \frac{9.87}{(29.22)^2} + \frac{9.87}{(47.95)^2} + \frac{9.87}{(17.79}$$

$$= \frac{9.87}{853.81} + \frac{9.87}{2299.20} + \frac{9.87}{316.48} = .0116 + .0043 + .0312 = .04$$

At $H/U\text{-}235 = 20.60$, $\rho = 1.120\ K_g/\ell$

$$B^2 = \frac{9.87}{(20.96+5)^2} + \frac{9.87}{(39.69+5)^2} + \frac{9.87}{(9.53+5)^2} = \frac{9.87}{(25.96)^2} + \frac{9.87}{(44.69)^2} + \frac{9.87}{(14.5}$$

$$= \frac{9.87}{673.92} + \frac{9.87}{1997.20} + \frac{9.87}{211.12} = .0146 + .0049 + .0468 = .06$$

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# MUFFLE TRAY K-EFF'S

**FLECTED**

U-235 = 0, $\rho$ = 8.930 Kg/l

$k_{eff} = \frac{1+(28 \times .0307)}{1+(28 \times .0281)} = \frac{1.860}{1.787} = 1.041$

H/U-235 = .98, $\rho$ = 6.700 Kg/l

$k_{eff} = \frac{1+(28 \times .0313)}{1+(28 \times .0283)} = \frac{1.876}{1.792} = 1.047$

H/U-235 = 2.94, $\rho$ = 4.470 Kg/l

$k_{eff} = \frac{1+(28 \times .0294)}{1+(28 \times .0290)} = \frac{1.823}{1.812} = 1.006$

H/U-235 = 8.96, $\rho$ = 2.230 Kg/l

$k_{eff} = \frac{1+(28 \times .0283)}{1+(28 \times .0295} = \frac{1.792}{1.826} = 0.981$

H/U-235 = 20.60, $\rho$ = 1.120 Kg/l

$k_{eff} = \frac{1+(28 \times .0304)}{1+(28 \times .0316)} = \frac{1.851}{1.885} = 0.982$

H/U-235 = 43.90, $\rho$ = 0.558 Kg/l

$k_{eff} = \frac{1+(28 \times .0308)}{1+(28 \times .0316)} = \frac{1.862}{1.885} = 0.988$

**BARE**

At H/U-235 = 0, $\rho$ = 8.930 Kg/l

$k_{eff} = \frac{1+(28 \times .0277)}{1+(28 \times .0372)} = \frac{1.776}{2.042} = 0.870$

At H/U-235 = .98, $\rho$ = 6.700 Kg/l

$k_{eff} = \frac{1+(28 \times .0267)}{1+(28 \times .0381)} = \frac{1.748}{2.067} = 0.846$

At H/U-235 = 2.94, $\rho$ = 4.470 Kg/l

$k_{eff} = \frac{1+(28 \times .0249)}{1+(28 \times .0400)} = \frac{1.697}{2.120} = 0.800$

At H/U-235 = 8.96, $\rho$ = 2.230 Kg/l

$k_{eff} = \frac{1+(28 \times .0247)}{1+(28 \times .0471)} = \frac{1.692}{2.319} = 0.730$

At H/U-235 = 20.60, $\rho$ = 1.120 Kg/l

$k_{eff} = \frac{1+(28 \times .0301)}{1+(28 \times .0663)} = \frac{1.843}{2.856} = 0.645$

At H/U-235 = 43.90, $\rho$ = 0.558 Kg/l

$k_{eff} = \frac{1+(28 \times .0313)}{1+(28 \times .0663)} = \frac{1.876}{2.856} = 0.657$

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## 926.14 Muffle Boxes

The maximum dimensions of the muffle boxes are 9" x 9" x 20" long or 9" diameter x 20" long. No more than two trays, each no larger than 8-1/4" by 15-5/8" by 1-1/4", will be allowed in the boxes, one above the other.

* The nuclear safety of the muffle box is assured by:

Keff calculations as a function of density show the box is subcritical at all densities up to the maximum of 4 kg U/liter. These calculations are conservative in that they assumed:

a) The trays are full at maximum density. Process design requirements limit the box to a maximum of 10 kg U. Therefore, at maximum density, the trays are only one-third full.

b) The poisoning effect of the Inconel trays (high nickel alloy) was not included.

c) Moderation to limits corresponding to Figure 309-XXIV. The boxes are operated at high temperature. Actual moderation is less than assumed by the calculation.

d) Full water reflection. The furnaces have insulation; however, this is not as efficient a reflector as water.

These calculations were performed assuming three trays per box. Since only two trays are placed in a box, this would further reduce the Keff under actual conditions.

*Indicates Change

926.15 Muffle Box Furnace 1-H-4, A and B

The muffle box furnaces are high temperature furnaces used for pyrohydrolysis of uranium bearing materials to a compound compatible with the solutions used in processing uranium. Also, the furnaces can be used to reduce the density of materials that have densities greater than 3.2 grams U-235/cc. A furnace will hold only one muffle box. The furnace arrangement is shown on Sketch 926.15-I.

* The solid angle subtended at the muffle box in furnace 1-H-4 B is 1.06 steradians.

*Indicates Change

I. Description

A. Both the 1-H-4A and the 1-H-4B will be evaluated as the centermost unit.
B. Each furnace will hold one muffle box.
C. Muffle boxes are 9" x 9" x 20" long or 9" diameter x 20" long.
D. Each muffle box will hold two trays, one above the other.
E. Trays will be no longer than 8 1/4" x 15 5/8" x 1 1/4".

II. $K_{eff}$ Value

The $K_{eff}$ of the two trays will not exceed 0.79 as shown in the Nuclear Safety Evaluation--$K_{eff}$ for Muffle Box, Subpart 926.14.

III. Interaction Calculations

A. 1-H-4A Centermost Unit

1. Contribution from 1-H-4B.

$d = 6''$, $L \approx 18''$, $L/2 = 9''$, $h = 16'' + 3'' = 19''$

$$\tan \theta = \frac{L/2}{h} = \frac{9''}{18''} = .474$$

$$\sin \theta = .431$$

$$\Omega A1 = \frac{2d}{h} \sin\theta = \frac{12''}{19''} = .3 \text{ steradians}$$

2. Contribution from 1-J-1

$d = 6''$, $L \approx 116''$, $L/2 = 58''$, $h = 58''$

$$\tan \theta = \frac{L/2}{h} = \frac{58''}{58''} = 1.0$$

$$\sin \theta = .707$$

$$\Omega A2 = \frac{2d}{h} \sin \theta = \frac{12''}{58''} (.707) = .146 \text{ steradians}$$

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Interaction Calculations (continued)

3. Contribution from 1-D-34 A, B and C

$d = 5''$, $L = 132''$, $L/2 = 66''$, $h = 71.5''$

$$\tan\theta = \frac{L/2}{h} = .922$$
$$\sin\theta = .678$$
$$\Omega_{A3} = \frac{2\,d}{h}\ \sin\theta = .095$$

Since 3 tanks,

$$\Omega_{A3}\ (\text{total}) = 3 \times \Omega_{A3} = 3 \times .095 = .284 \text{ steradians}$$

4. Contribution from other surrounding equipment.

All other equipment is greater than 10 feet away and either one-gallon bottles, 11 liter bottles or similar small shapes or volumes. Therefore, their interaction effects are considered negligible and neglected.

5. Total Interaction

$$\Omega_A\ (\text{total}) = \sum \Omega_{A1} + \Omega_{A2} + \Omega_{A3} = 1.03 \text{ steradians}$$

B. 1-H-4B Centermost Unit

1. Contribution from 1-H-4A

Same as IV.A.1

$$\Omega_{B1} = \Omega_{A1} = .3 \text{ steradians}$$

2. Contribution from 1-J-1

$d = 6''$, $L = 108''$, $L/2 = 54''$, $h = 100''$

$$\tan\theta = \frac{L/2}{h} = .54$$
$$\sin\theta = .495$$
$$\Omega_{B2} = \frac{2\,d}{h}\ \sin\theta = .0594 \text{ steradians}$$

3. Contribution from 1-D-34 A, B and C

$d = 5''$, $L = 132''$, $L/2 = 66''$, $h = 37.5''$

$$\tan\theta = \frac{L/2}{h} = 1.76$$
$$\sin\theta = .8695$$
$$\Omega_{B3} = \frac{2\,d}{h}\ \sin\theta = .696 \text{ steradians}$$

4. Contribution from other surrounding equipment.

Same as IV.A.4

5. Total Interaction

$$\Omega_B(\text{total}) = \Sigma\ \Omega_{B1} + \Omega_{B2} + \Omega_{B3} = 1.055 \text{ steradians}$$

C. Allowable Interaction

$$\Omega_{\text{allowable}} = 9\text{-}10K = 9 - 7.9 = 1.1 \text{ steradians}$$

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932.2 Equipment and Nuclear Safety

Samples will be received in four ounce size containers and placed in the safe geometry storage rack. One gallon bottles are used to consolidate discard solutions.

Criticality control away from the rack will be achieved by limiting (through logging control) the amount being worked on in the lab facilities to one limited safe mass amount (350 grams $U^{235}$) which is distributed throughout the laboratory.

The laboratory scale may be used for final product weighing. During this product weighing, one gallon bottles will be brought to the laboratory, weighed and returned to the process area or the receiving area one at a time.

* The storage rack consists of shelves on both sides of a 12" thick concrete block wall. The shelf construction is such that it will hold sample containers as described in Subpart 917. The 12" thick solid concrete block wall provides adequate protection from interaction of the two slab configurations. The nuclear safety of this storage is set forth in Subpart 917.

There is an aisle on each side of this wall, providing 5' spacing from the nearest additional uranium which would be samples being worked on in laboratory equipment across the aisles.

Fixed bottle storage racks to accomodate up to 12 containers of 1 gallon size or less as shown on the attached Sketch 432.3-I. Bottles will be in metal holders on 16 inch centers, 48 inches apart.

*Indicates Change