REGULATORY FILE CY

GULF NUCLEAR FUELS COMPANY

A DIVISION OF GULF OIL CORPORATION

GRASSLANDS ROAD
ELMSFORD, NEW YORK 10523
914-592-9000

April 15, 1974
In reply, refer to MRA-74-67

RECEIVED APR 22 1974 Mail Section

Mr. Leland C. Rouse, Chief
Fuel Fabrication & Reprocessing Branch
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Rouse:

Gulf Nuclear Fuels Company hereby requests that its activities at New Haven, Connecticut authorized by AEC License No. SNM-33 henceforth be authorized by AEC License No. SNM-871, for a period of one year. This request is made to separate the authorization of activities conducted at Gulf Nuclear's locations at Hematite, Missouri and New Haven for future AEC action.

To facilitate the transfer from SNM-33 to SNM-871, two definitive steps have been taken. First, all of the activities presently authorized by SNM-871 have ceased and a request made to the Directorate of Licensing to delete the locations of use, Pawling and Eastview, New York. Second, the renewal application for SNM-33 has been revised by deleting portions of the application which pertain only to Hematite. It is this application for license which, if renewed as SNM-871, would apply to New Haven. Eight copies of the application are attached for your use.

Other than deletion of portions which pertain only to Hematite, no substantive changes were made in the application beyond the revisions dated April 21, 1972. Minor changes were made to reflect changes in organization and personnel, to correct errors, to incorporate the AEC standard emergency control plan by reference, to delete the subsection on fabrication of high enrichment rods, and to delete reference to Building 11-H. A revised list of effective pages is included with the application to assist your review of the up-dated pages.

In the interest of time, application pages bearing the predecessor company names, United Nuclear Corporation -CPD and Gulf United Nuclear Fuels Corporation were not revised; nor were those bearing AEC License No. SNM-777. Also, some pages in and references to Section 900 were left in their former 800 section designation. They should be regarded as Section 900, however.

~~Ignore Prop. markings per ltr. 4-22-74~~

DOCKETED USAEC MAY 9 1974 REGULATORY MAIL SECTION DOCKET CLERK

.... Cont'd ...

1060

In addition to the application, Gulf Nuclear Fuels Company agrees to incorporation of the license condition on use and maintenance of effluent treatment systems, safeguards amendment SG-4 and the Fabrication Operations Physical Protection Plan.

Please let me know promptly if you have any questions or problems with this request, for we are anxious for the appropriate actions to be completed as quickly as possible.

Sincerely,

Peter Loysen

Peter Loysen, Manager
Regulatory Administration

PL:am
Attachments -
(8 copies-Application for Renewal of AEC License No. SNM-33)

bc: G. O. Amy w/attachment
H. E. Clow w/attachment
D. T. Farney w/attachment
F. G. Stengel
M. P. Wittner

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

GULF NUCLEAR FUELS COMPANY

Fabrication Operations
New Haven, Connecticut

Date issued: April 15, 1974

AEC

SPECIAL NUCLEAR MATERIALS LICENSE

SNM - 33

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		4 of 6	10/31/68
		5 of 6	2/6/70
		6 of 6	10/31/69
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	Evaluation - Dresden Fuel	2 of 2	2/6/70
	Elements (BWR)		
	Nuclear Safety Evaluation -	1 of 4	2/6/70
	Yankee Fuel Elements (PWR)	2 of 4	10/31/68
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	Nuclear Safety Evaluation -	1 of 5	3/31/71
	Yankee Reload Fuel (PWR)	2 of 5	3/31/71
		3 of 5	3/31/71
		4 of 5	3/31/71
		5 of 5	3/31/71
	Nuclear Safety Evaluation	1 of 2	6/14/71
	Pickle-Corrosion Fixture	2 of 2	3/19/71
	Nuclear Safety Evaluation	1 of 3	11/28/72
	Connecticut Yankee Fuel (PWR)	2 of 3	11/28/72
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5/1/73

DATE: 4/15/74 SUPERSEDES: 5/1/73

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	Figure 926-I	1 of 1	3/6/72
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NATIONAL
23-681
23-685

NATIONAL INSERTABLE-TAB INDEXES ENABLE YOU TO MAKE YOUR OWN SUBJECT ARRANGEMENT, USING PLAIN INSERTS ON WHICH TO WRITE YOUR OWN CAPTIONS.

The Beaded edge on tab makes it easy to insert captions

Made in U. S. A.

SECTION 100

101. General

101.1 Introduction

This manual has been prepared to provide the information required by the Atomic Energy Commission regulations 10 CFR 20, 10 CFR 70, and 10 CFR 71 for renewal of special nuclear material license SNM-33. The manual outlines the Gulf Nuclear Fuels Company practices, guides, procedures and controls applied to insure the safe handling of special nuclear materials at its facilities at New Haven, Connecticut.

101.2 Corporate Information

Gulf Nuclear Fuels Company maintains headquarters at Elmsford, New York.
The Company is a wholy owned subsidiary of Gulf Oil Corporation for the purpose of designing, manufacturing and selling nuclear fuel for light water nuclear power reactors and certain types of research reactors. Facilities include corporate Offices and Engineering Operation at Elmsford, New York; and Fabrication Operations at New Haven.

101.3 Officers

Corporate Officers are:

President	Arnold R. Fritsch 312 Ocean Drive West Stamford, Connecticut 06902
Vice President, Utility Fuel Engineering	Richard A. Dean Revere Drive Riverside, Connecticut 06878
Vice President, Manufacturing	Fred G. Stengel 95 Pond Road Wilton, Connecticut 06460

101. General (Continued)

101.3 Officers

Vice President, Treasurer & Secretary — Kenneth L. Wiley
16 Surrey Drive
Riverside, Conn. 06878

All of the officers are United States citizens. There will be no known control or ownership exercised over Gulf Nuclear Fuels Company by any alien, foreign corporation or foreign government.

102. Location and Facilities (continued)

102.1 Fabrication Operation

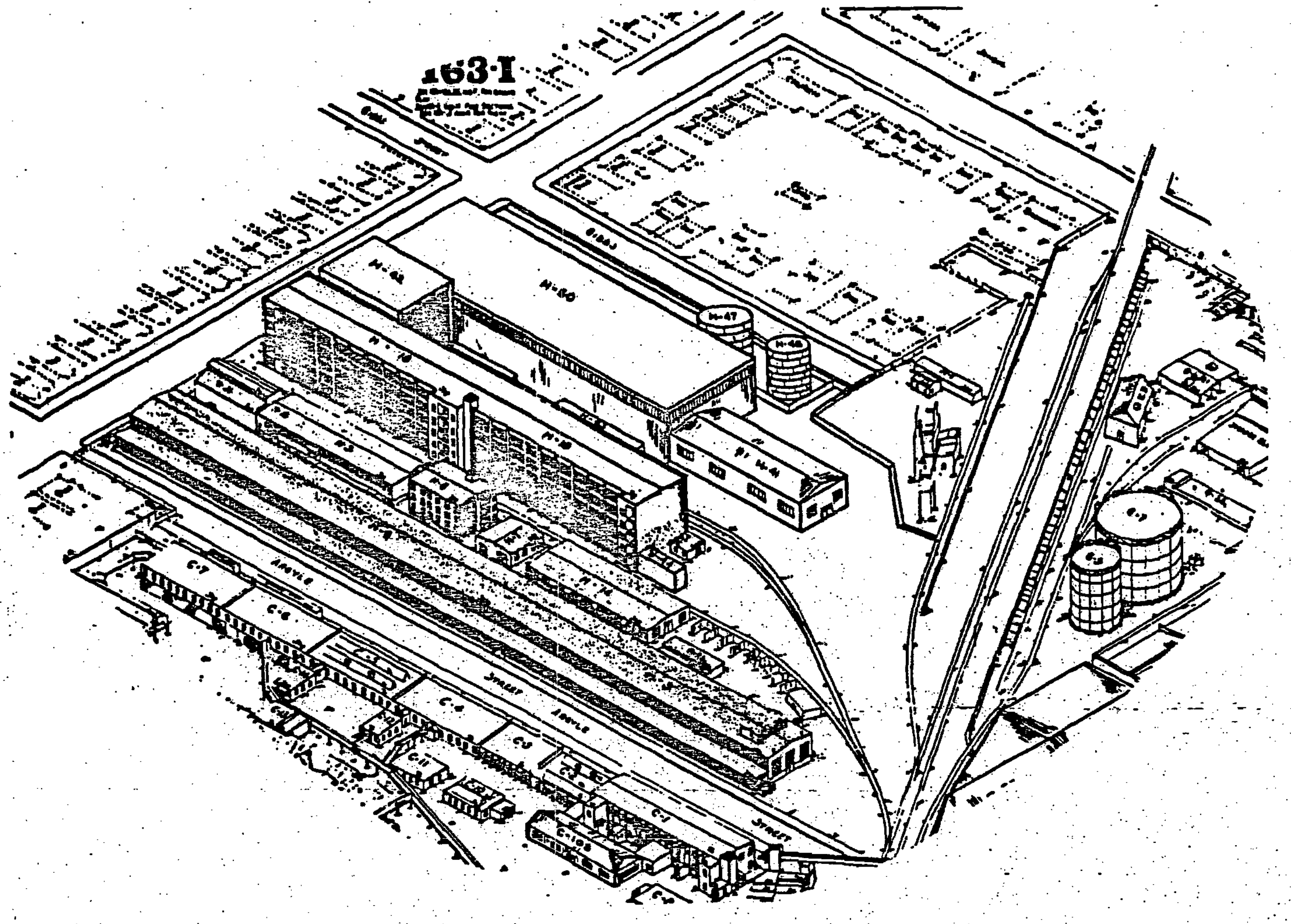
Figure 102-I shows the general arrangement of the plant in New Haven. This property is located in the area generally bounded by Division Street on the north, Munson and Henry Streets on the east.

Fabrication Operation has control over part of the first floor of building 19 H, part of the third floor of building 18 H (offices), the 19 H basement waste disposal area, all of building 41 H and all of building 50 H except for an area approximately 50 feet x 65 feet on the north side approximately 100 feet from the entrance which is controlled by United Nuclear Corporation, Naval Products Division. The 18 H building is shared with Olin Corporation.

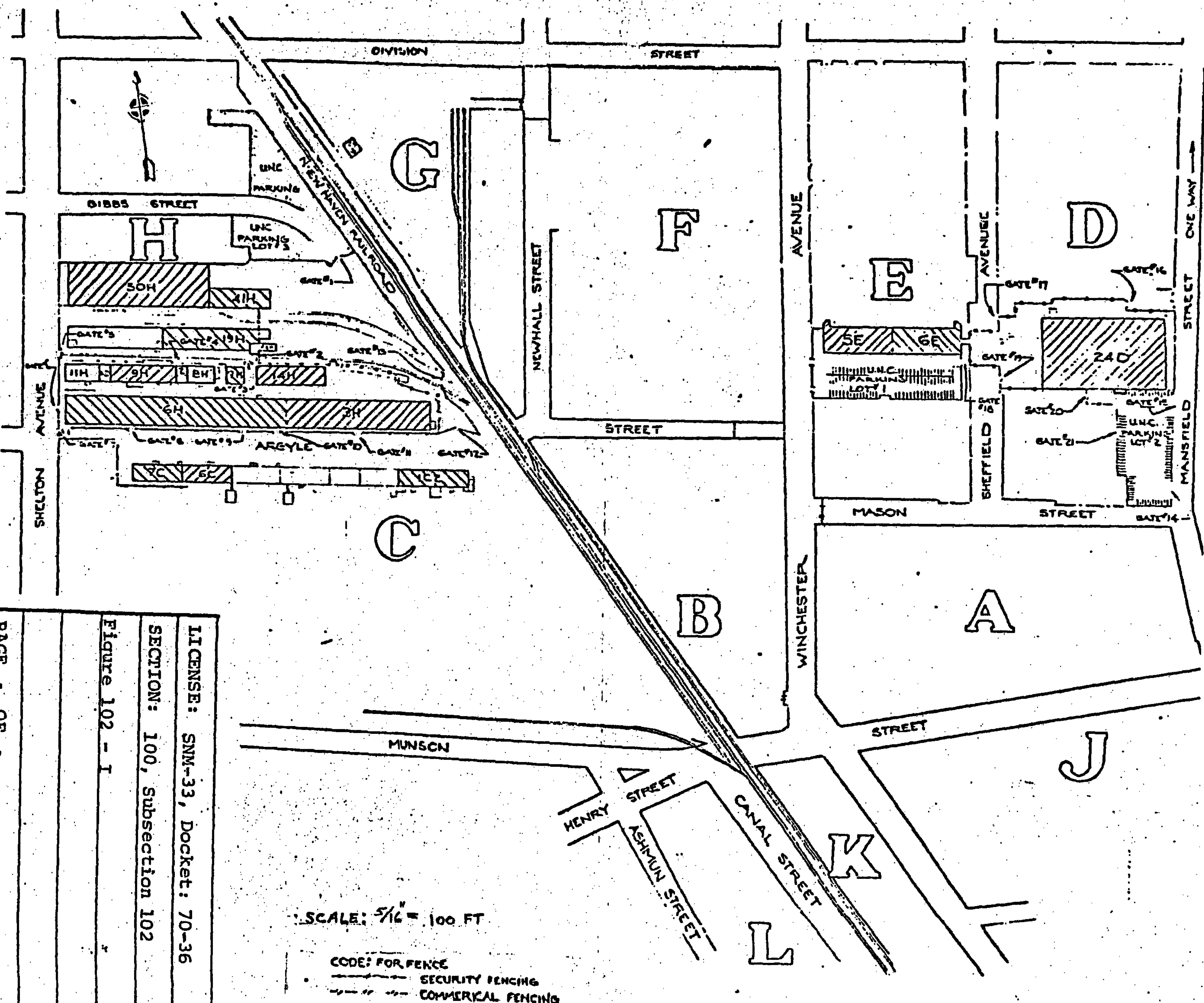
Further details on the "H" Tract arrangement and buildings 19 H, 41 H and 50 H are shown on the following listed drawings:

B-304 236	Equipment layout 19H
DD-394	Equipment layout 50H
B-304 795	Equipment layout 41H

The third floor of building 18 H houses the administrative offices of the Fabrication operation and thus a layout drawing is not provided.



Gulf United Manufacturing Facilities, New Haven – Aerial View



LICENSE: SNM-33, Docket: 70-36

SECTION: 100, Subsection 102

Figure 102 - I

PAGE 1 OF 1

APPROVED:

ISSUED: 4/21/72

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.

103.1 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.

104. Source and Special Nuclear Materials Possession Limits

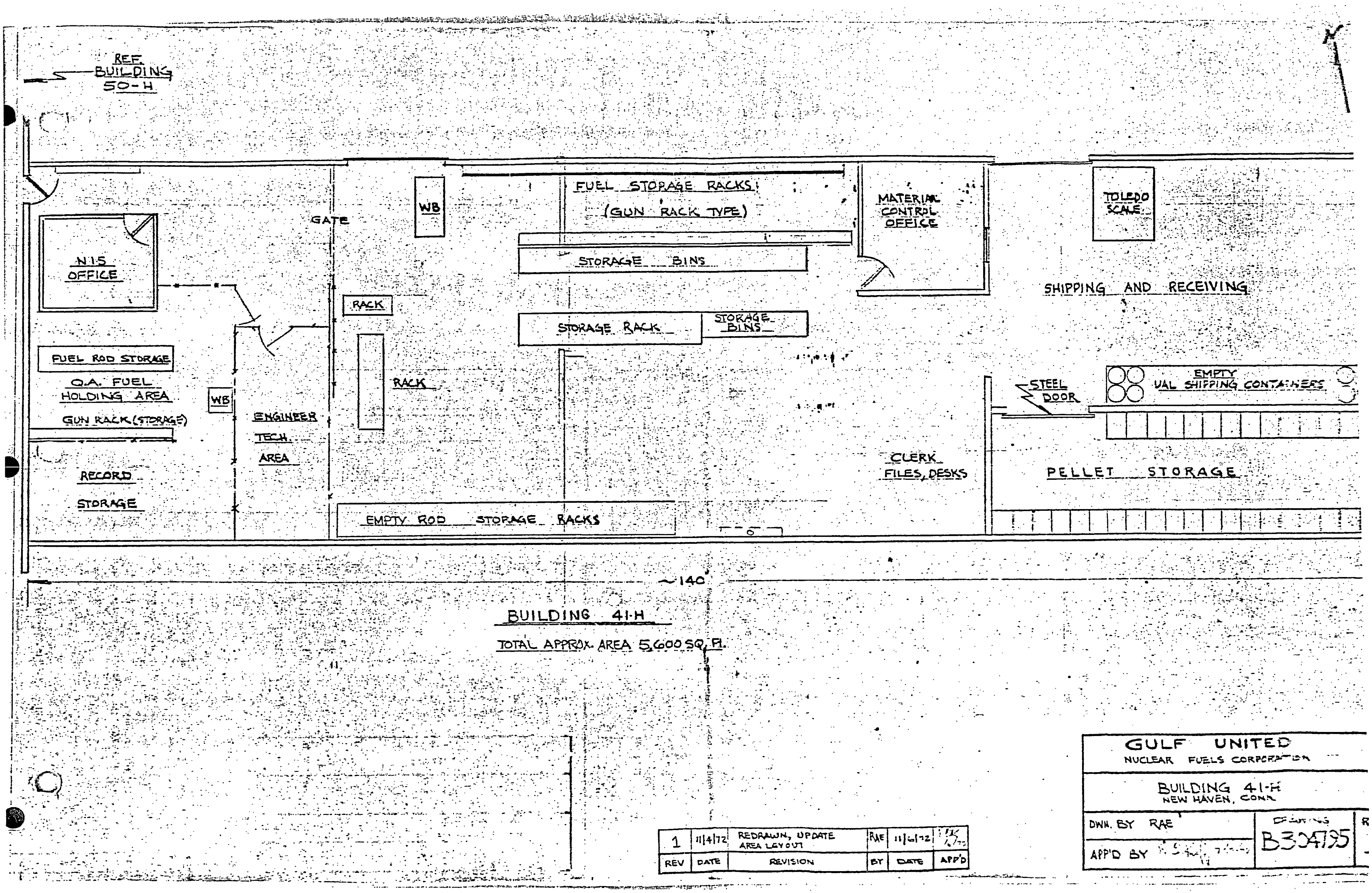
104.1 Fabrication Operation

1. Special Nuclear Materials

   At any one time, the maximum quantity expected to be on hand is limited to 1,400 kgs. of contained U-235 and 10 kgs contained Plutonium.

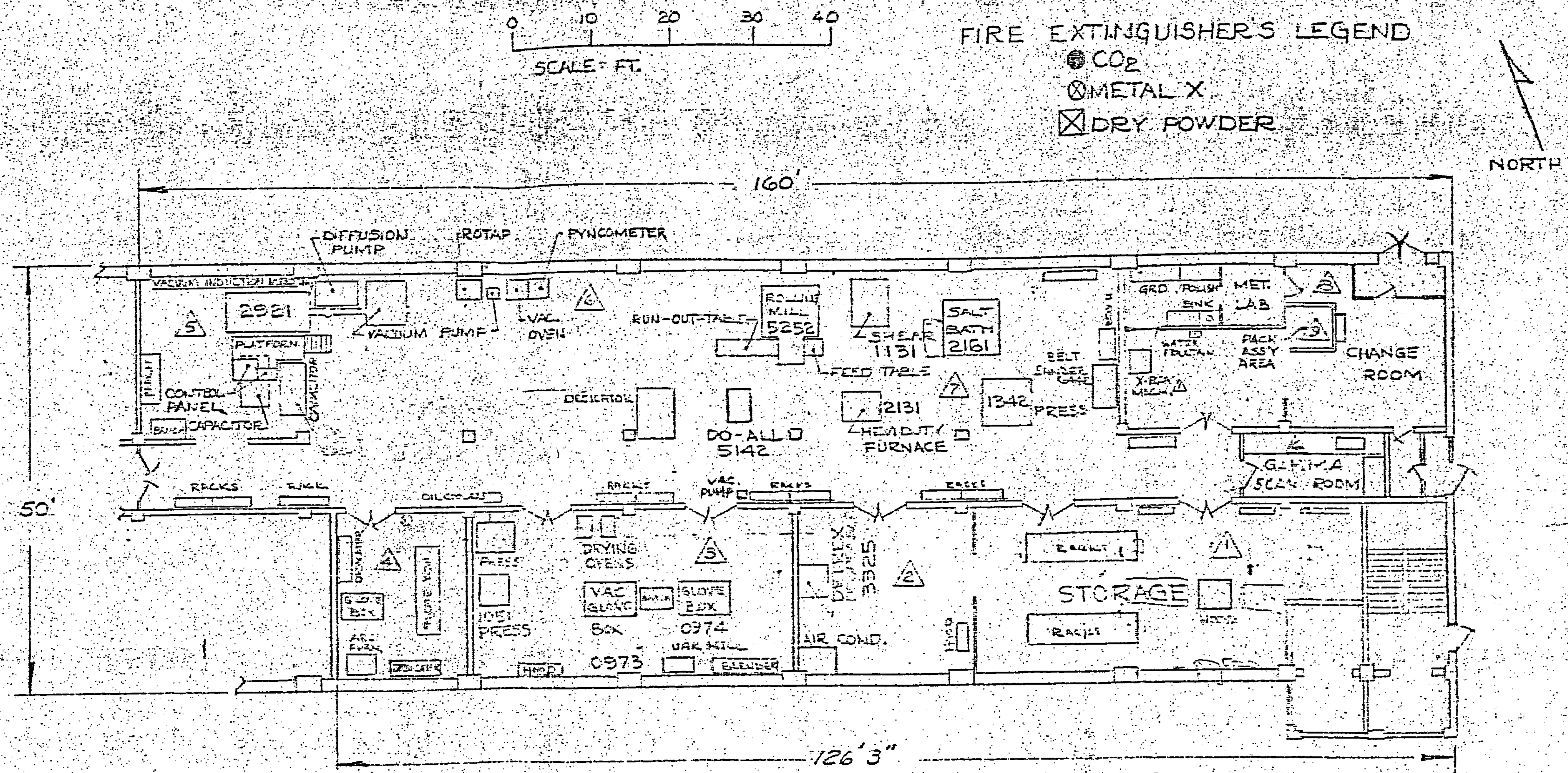
2. Source Materials

   The maximum quantity on hand at any one time is limited to 20,000 kgs. of uranium and thorium.


REF BUILDING 50-H
N.I.S. OFFICE
GATE
WB
FUEL STORAGE RACKS (GUN RACK TYPE)
MATERIAL CONTROL OFFICE
TOLEDO SCALE
STORAGE BINS
RACK
SHIPPING AND RECEIVING
STORAGE RACK
STORAGE BINS
FUEL ROD STORAGE
Q.A. FUEL HOLDING AREA
GUN RACK (STORAGE)
WB
ENGINEER TECH. AREA
RACK
STEEL DOOR
EMPTY UAL SHIPPING CONTAINERS
CLERK FILES, DESKS
PELLET STORAGE
RECORD STORAGE
EMPTY ROD STORAGE RACKS
~140'
BUILDING 41-H
TOTAL APPROX. AREA 5,600 SQ. FT.
GULF UNITED NUCLEAR FUELS CORPORATION
BUILDING 41-H NEW HAVEN, CONN.
DWN. BY RAE
APP'D BY
B304795
1 11/4/72 REDRAWN, UPDATE AREA LAYOUT RAE 11/6/72
REV DATE REVISION BY DATE APP'D

~~PROPRIETARY INFORMATION~~

~~THIS DATA SHALL NOT BE DISCLOSED TO OTHER MANUFACTURERS OR USED FOR OTHER THAN THE EXPRESS PURPOSE FOR WHICH LOANED WITHOUT WRITTEN CONSENT OF GULF UNITED NUCLEAR FUELS CORPORATION.~~



NOTE: TOTAL AREA APPROX. 7600 SQ. FT.

LEGEND

1. FUEL STORAGE AREA.
2. CHARGE PREPARATION AREA.
3. UAl$_x$ PROCESSING AREA.
4. UAl$_x$ POWDER PROCESSING AREA.
5. MELTING AREA.
6. INSPECTION AREA.
7. ROLLING, [illegible], SHEARING, & [illegible] AREA.
8. METALLOGRAPHIC LABORATORY.
9. [illegible] ASSEMBLY AREA.

REV	DATE	REVISION	APP'D
4	7-1-72	ADD AREA LEGEND & DIM.	RBE KRK
3	6-7-72	UPDATED DWG WITH MET LAB & POWDER [illegible]	RBE EP
2	11-[illegible]	[illegible] UPDATE DWG	RM
1	[illegible]	UPDATE DWG	RM

19H-1

L.C.W. 2-25-64
G.A.A. 8-8-67
ALM 9/12/69

B-2098

B-504236-1

REV 4

~~PROPRIETARY INFORMATION~~

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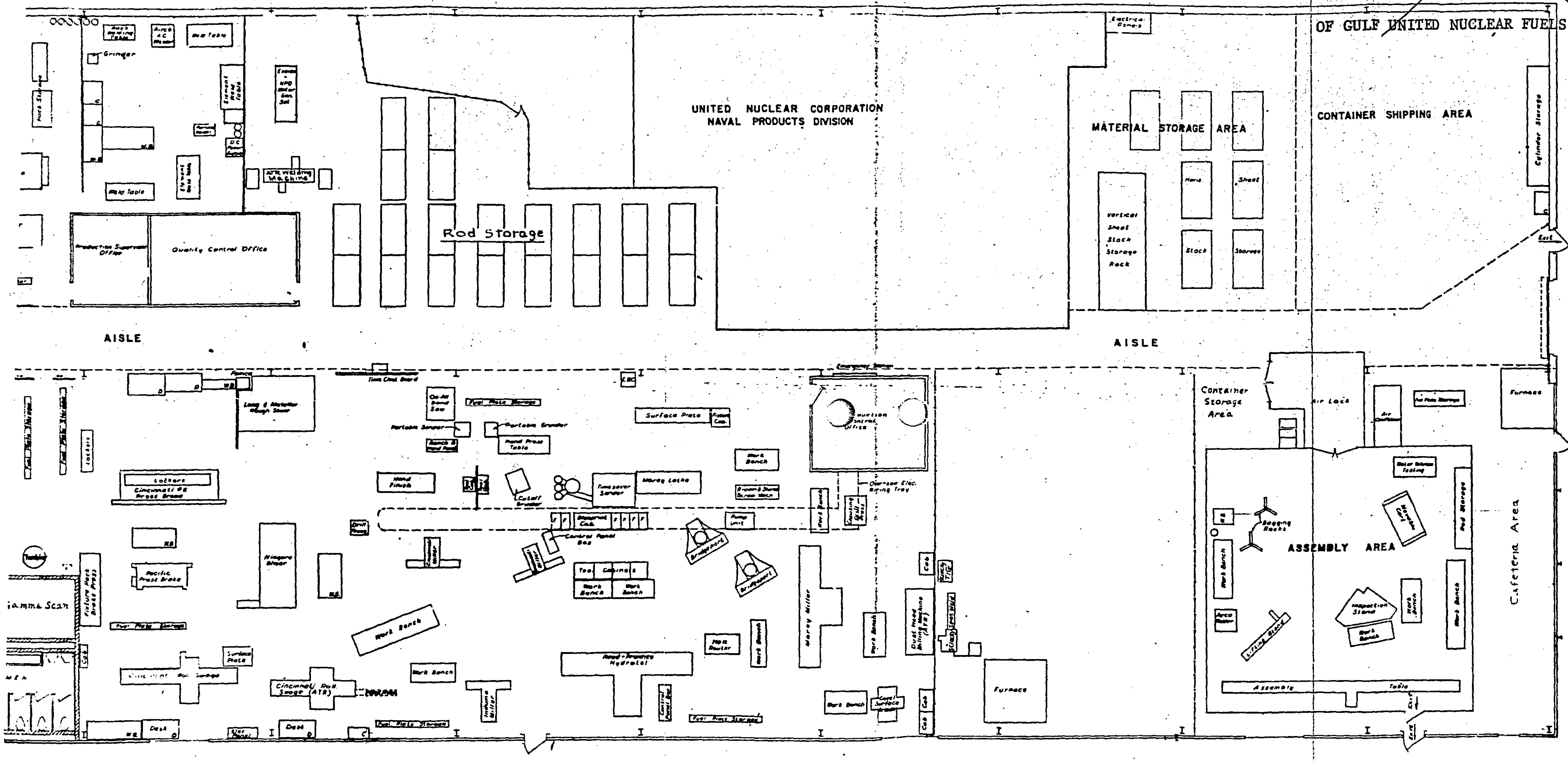


BUILDING 50H

1		5/14/72	Changed Cleaning Area. Added Rod Loading Details	VRR	SEJ
NO		DATE	REVISIONS	DWN	APPD

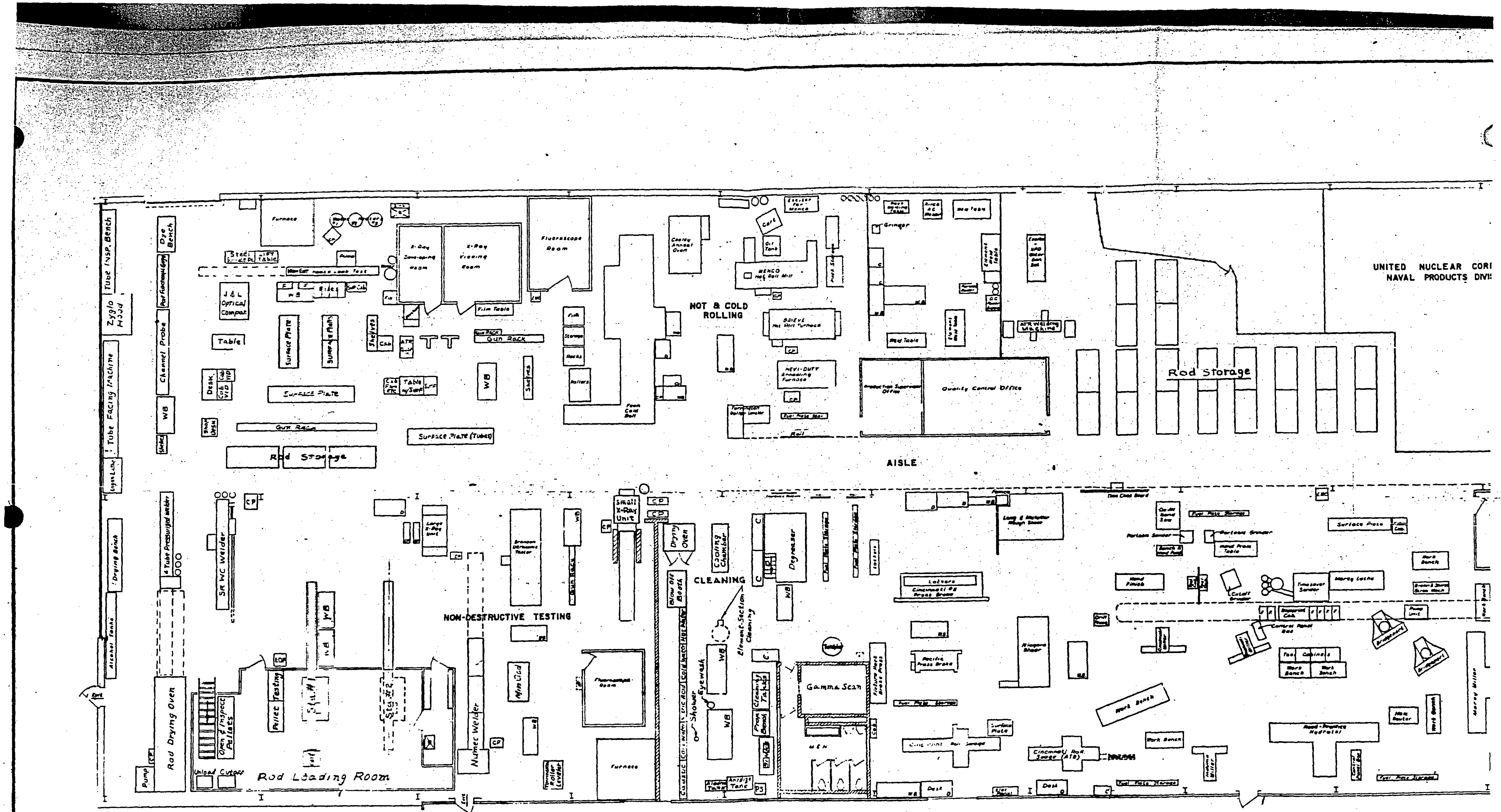
GULF UNITED NUCLEAR FUELS CORPORATION

NEW ARRANGEMENT
BUILDING 50H

PROJECT MANAGER: S E Johnson
5168
V. RUTIGLIANO 5/1/72

SIZE: X | DWG NO: DD-394 | REV: 1

GULF UNITED NUCLEAR FUELS CORPORATION — REHABILITATION OF ... 50H — ANDERSON-NICHOLS



BUILDING 50H

LEGEND

WB WORKBENCH
D DESK
C CABINET

GULF UNITED
NUCLEAR FUELS CORPORATION

SECTION 200

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		Issued 4/21/72
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SUBSECTION:	201 - Corporate Organization	Issued 4/15/74
		Supersedes 4/21/72

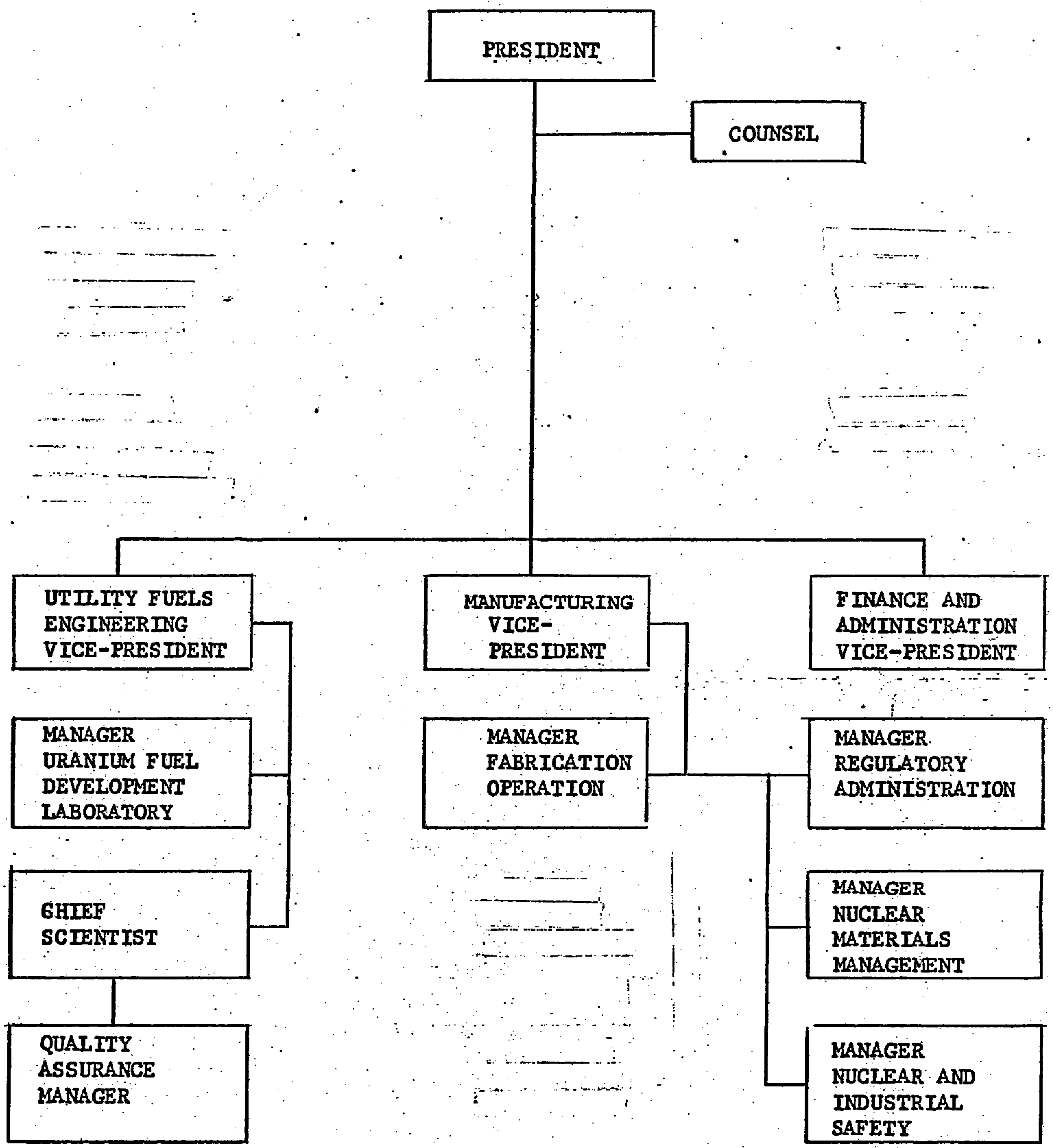
201. Corporate Organization

Gulf United Nuclear Fuels Company is organized for efficient management and administration of individual operations at different locations in separated plants or facilities. The Vice President, Manufacturing has the overall responsibility for Fabrication Operations and reports to the President of Gulf Nuclear Fuels Company.
This is shown on Figure 201-I.

The organization of the Corporation provides for administration, production, technical support; nuclear and industrial safety and nuclear materials management on a corporate-wide as well as Operation-wide basis.

Personnel having health and safety responsibilities will be selected by management based on their qualifications. The selection of these personnel requires approval of two levels of management above the position being filled.

GULF UNITED
NUCLEAR FUELS CORPORATION



License: SNM-33 Docket: 70-36 Section: 200 Subsection/Subpart: 201

Subject: Figure 201-I -- Corporate Organization

Issued: 4/15/74 Supersedes: 5/1/73 Approved: ____________ Page 1 of 1

GU-36

202. Nuclear and Industrial Safety Organization

1. Organization

Nuclear and industrial safety is organized to provide a complete program for nuclear criticality safety, health physics, industrial safety and fire prevention, and medical services for the Corporation. This is accomplished by providing operating and staff nuclear and industrial safety groups.

On site nuclear criticality safety, health physics, industrial safety and fire prevention, and medical services control functions are provided by operating nuclear and industrial safety personnel. Figures 204-I and -II show the arrangements of these operating nuclear and industrial safety groups.

Staff nuclear and industrial safety control is provided by the Nuclear and Industrial Safety Department. The Nuclear and Industrial Safety Manager is responsible for the following activities:

1.1 Establishment of corporate nuclear and industrial safety policy.

1.2 Preparation of Regulatory Agency licenses.

1.3 Technical support services, as related to nuclear criticality safety and health physics for review of proposed additions to or modifications of proposed equipment.

1.4 Systematic auditing of plant operations.

The Nuclear and Industrial Safety Department organization is shown in Figure 202-I.

GULF UNITED
NUCLEAR FUELS CORPORATION

202. Nuclear and Industrial Safety Organization

2. Basic Responsibilities

2.1 Nuclear and Industrial Safety Department Manager

The NIS Department Manager is responsible to ensure effective and timely administration of the nuclear and industrial safety control and audit function. He assists in establishing sound programs in compliance with Corporate Policy and appropriate Federal and State Regulations and ensures continued compliance of these official programs through regular audits and follow-up with responsible Corporate Management. He must provide competent technical support services to the Division from either in-house specialists or from specialists outside of the Corporation on a consulting basis.

2.2 Nuclear and Industrial Safety Representative

The Nuclear and Industrial Safety Representative is responsible for daily surveillance of nuclear criticality/industrial safety and health physics at his assigned plant. He initiates NIS Department Nuclear criticality safety and health physics evaluations of proposed modifications to processes and equipment. He may perform preliminary nuclear criticality safety evaluations of these proposed changes. He performs inspections of operating 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 Corporate Policy, Federal and State regulations. He performs nuclear criticality safety evaluations of processes and equipment, plant inspections and follow-up with responsible operating management.

GULF UNITED
NUCLEAR FUELS CORPORATION

202. Nuclear and Industrial Safety Organization

2. Basic Responsibilities (continued)

2.4 Health Physics Specialist

The Health Physics Specialist assists the Department Manager in providing a sound program in compliance with Corporate Policy, Federal and State regulations. He performs evaluations of radiological safety and plant inspections, and follows up with responsible operating management.

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

3.1 Nuclear and Industrial Safety Department Manager

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.

3.2 Nuclear and Industrial Safety Department Specialists

Nuclear and Industrial Safety Department Specialists 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 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.

3.3 Nuclear and Industrial Safety Department Representatives

Nuclear and Industrial Safety Department Representatives shall have a college degree or its equivalent experience in the nuclear industry including formal training in Nuclear criticality safety, health physics, and industrial safety, and fire prevention.

3.4 Nuclear and Industrial Safety Department Consultants

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

3.5 Equivalent Experience

Two years responsible and appropriate experience may be considered equivalent to each year of college work. This experience need not cover all phases of the discipline but must contribute to the general field of the discipline.

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

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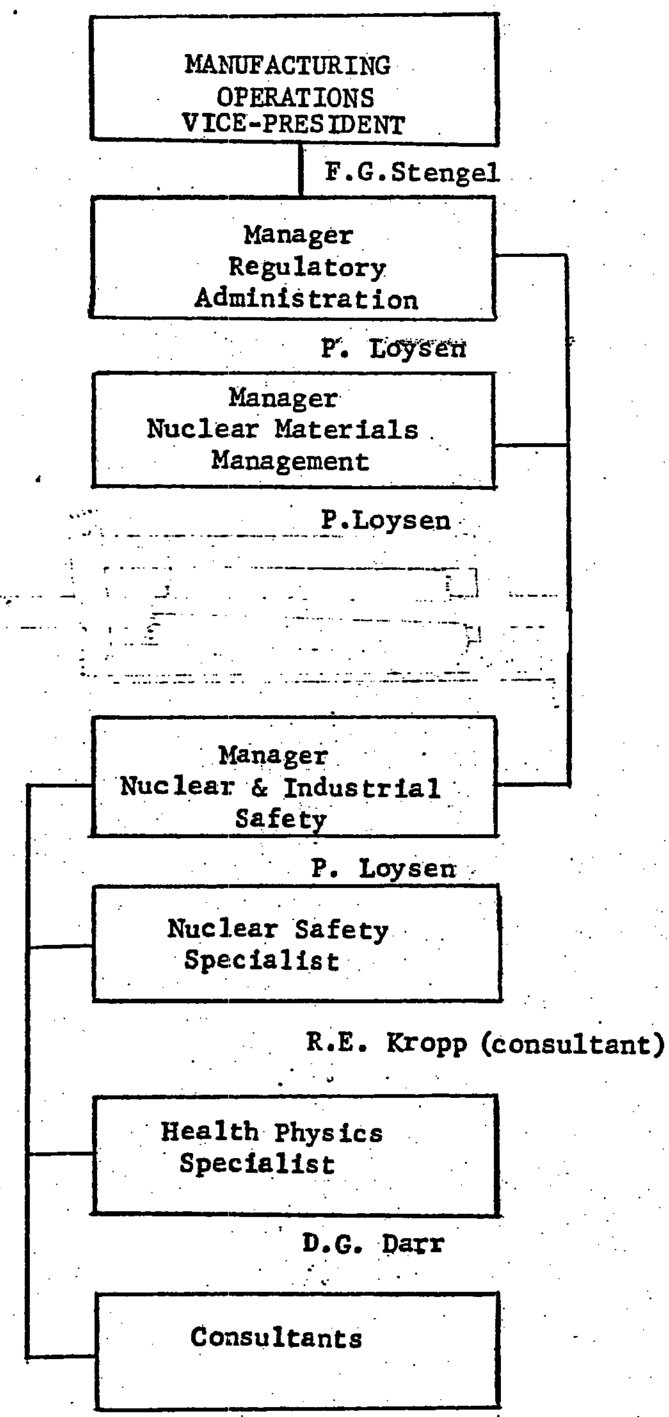

License: SNM-33 Docket: 70-36 Section: 200 Subsection/Subpart: 202

Subject: Figure 202-I -- Organization Chart, Nuclear & Industrial Safety Dept.

Issued: 4/15/74 Supersedes: 11/10/72 Approved: ______ Page 1 of 1

GULF UNITED
NUCLEAR FUELS CORPORATION

LICENSE: SNM-33, Docket: 70-36 SECTION: 200 - ORGANIZATION, PERSONNEL AND ADMINISTRATION SUBSECTION: 203 - Nuclear Materials Management	 Approved Issued 4/21/72 Supersedes 2/6/70

203. Nuclear Materials Management

The Nuclear Materials Management responsibilities, controls and operations are described in Section 500.

204. Manufacturing Organization

1. General Description

1.1 Gulf United Nuclear Fuels Company's manufacturing operations are organized for the specific purpose of processing SNM for further fabrication of fuel assemblies and actual fabrication of fuel assemblies. The organization chart, Figure 201-I, shows the subdivision of the manufacturing operations.

1.2 The operation is headed by an Operation Manager who has full responsibility and authority to carry out the functions of that operation in conjunction with contributions of other departments or groups to achieve the overall objectives of the Company. Each Department Manager is directly responsible to his immediate superior 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 Operation Manager. The manager has the responsibility for manufacture, engineering and shipment applicable to the production of products described in this application. The organization chart, Figure 204-II shows functions essential for the operation.

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 operating Nuclear and Industrial Safety group.

204. Manufacturing Organization

2. Manufacturing Departments (continued)

2.3 The Fabrication Operation is organized for the purpose of fabricating SNM bearing components for test and power reactors. Figure 204-II shows the organization established to perform these functions.

3. Basic Responsibilities

3.1 Operation Manager

The Operation Manager is responsible for the safe efficient operation and maintenance of the plant in conformance with established policies and procedures for required administrative and process development work.

3.2 First Line Management

First Line Management reporting to the Plant Manager are responsible for the safe efficient operation of their assigned portions of the facilities. This includes the supervision of any activities assigned to them.

4. Personnel Qualifications

The minimum qualifications of the Operation Manager, and first line management shall be a B.S. degree in a technical field with two years experience in Nuclear plants and laboratories, or high school with ten years nuclear industry experience.

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

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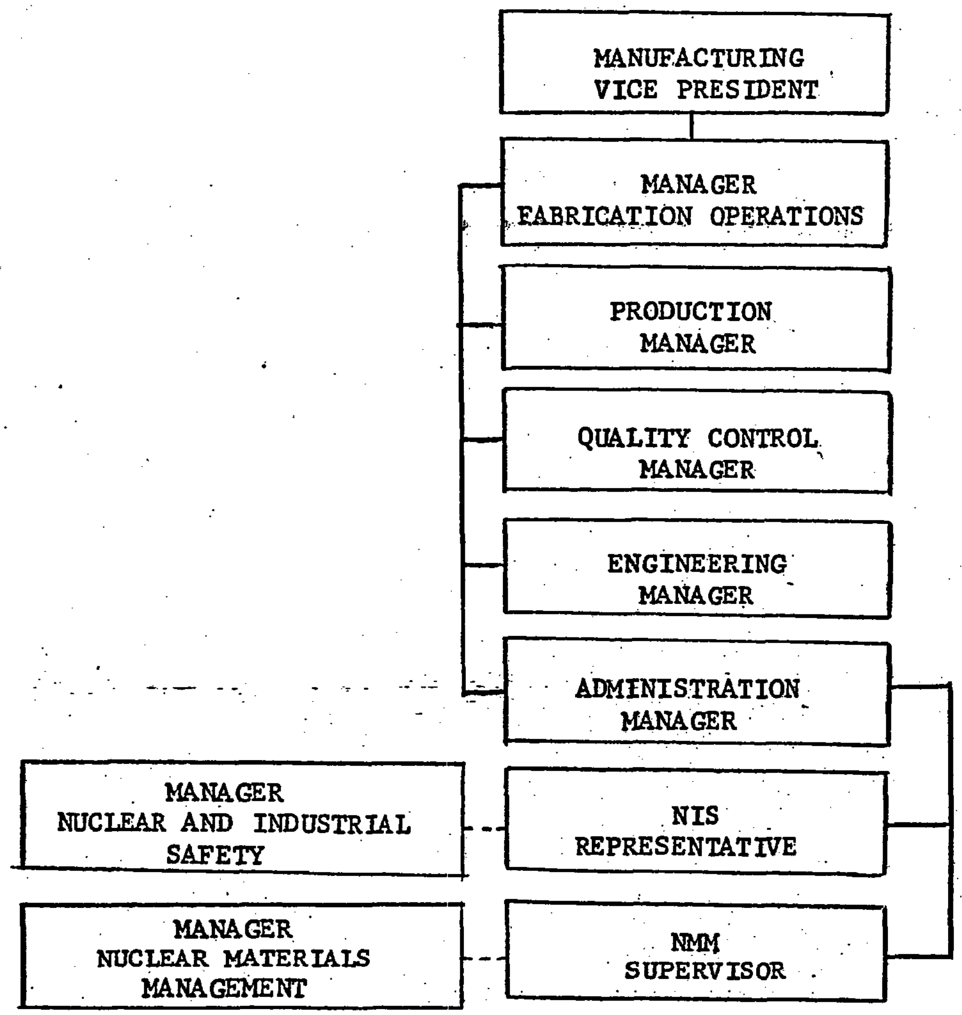
NUCLEAR FUELS CORPORATION



License: SNM-33 Docket: 70-36 Section: 200 Subsection/Subpart: 204

Subject: Figure 204-II--Organization Chart, Fabrication Operation

Issued: 11/10/72 Supersedes: 8/17/72 Approved: ______ Page 1 of 1

GULF UNITED
NUCLEAR FUELS CORPORATION

LICENSE: SNM-33, Docket: 70-36 SECTION: 200 - ORGANIZATION, PERSONNEL AND ADMINISTRATION SUBSECTION: 205 - Process Control	 Approved Issued 4/15/74 Supersedes 4/21/72

205. Process Control

1. General

Corporate 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, control 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 accordance with criteria established by the license. The mechanism of such approval is described in more detail in Subsection 206.

2. Fabrication Operation

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.

LICENSE:	SNM-33, Docket: 70-36	Page 1 of 2
SECTION:	200 - ORGANIZATION, PERSONNEL AND ADMINISTRATION	Approved
SUBSECTION:	206 - Nuclear and Industrial Safety Control	Issued 4/21/72
		Supersedes 3/19/71

206. Nuclear and Industrial Safety Control

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 Nuclear criticality safety evaluation has been performed by NIS Representative based on the criteria and standards of Sections 300 and 400.

2.2 The NIS Representative's evaluation has been reviewed by two Nuclear Safety Specialists. 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

2.3 The NIS Representative's evaluation has been reviewed by a Health Physics Specialist. This review is based on the criteria and standards of Section 400.

2.4 Review and verification shall include written approval by the reviewers.

2.5 All evaluations, reviews and verifications have been overchecked by the Nuclear and Industrial Safety Manager. This overcheck will be indicated by written approval.

206. Nuclear and Industrial Safety Control (continued)

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.

GULF UNITED
NUCLEAR FUELS CORPORATION

LICENSE:	SNM-33, Docket: 70-36	
SECTION:	200 - ORGANIZATION, PERSONNEL AND ADMINISTRATION	Approved
SUBSECTION:	207 - Inspections and Audits	Issued 5/8/72
		Supersedes 4/21/72

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 (nonresident), Specialist or NIS Manager. An inspection includes a review of checks to determine the area or areas requireing 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.

GULF UNITED
NUCLEAR FUELS CORPORATION

LICENSE:	SNM-33, Docket: 70-36	Page 1 of 1
SECTION:	200 - ORGANIZATION, PERSONNEL AND ADMINISTRATION	Approved
SUBSECTION:	208 - Training	Issued 4/21/72
		Supersedes 10/31/68

208. Training

1. Purpose

The purpose of the training program is to inform and instruct all employees in the policy and programs of the company as they relate to nuclear criticality safety, health physics and industrial safety, and proper and safe performance of their assignments.

2. New Employees

The indoctrination of new employees in the safety aspects of the facility is conducted by, or under the supervision of specialist in the various topics. The indoctrination topics include but are not limited to:

a) Fundamentals of nuclear criticality safety and controls.

b) Fundamentals of the health physics program and controls.

c) Emergency alarms and actions required.

d) A review of the facility operations.

e) On the job training, under direct line supervision and/or by experienced personnel.

3. Continued Training

The training and personnel safety program is continued with on the job training supplemented by regularly scheduled meetings conducted by line supervision and specialist in the subjects covered. Included are personnel protection equipment, industrial safety and accident prevention and other topics applicable to the facility operations.

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION President

PERSON Arnold R. Fritsch

EXPERIENCE

Dr. Fritsch has over fifteen years experience throughout the nuclear industry. Assignments have included policy formulation and consultation to the Chairman of the U.S. Atomic Energy Commission; coordination of all fuel reprocessing and materials control and management; and establishing, staffing and planning of a spent fuel reprocessing company. He has also had extensive experience in long range studies of reactor programs.

FORMER POSITIONS

1970 - 1971	Coordinator - Allied Gulf Nuclear Services, Gulf Energy and Environmental Systems
1968 - 1970	Manager, Program Evaluation and Sr. Advisor to Group Vice Presidents, Gulf Energy and Environmental Systems
1961 - 1968	Technical Assistant and Special Assistant to the Chairman, U.S. Atomic Energy Commission
1959 - 1961	Division of International Affairs, U.S. Atomic Energy Commission
1956 - 1959	Senior Engineer, Nuclear Physics, Westinghouse Atomic Power Division

EDUCATION

B.S.	Physical Chemistry, University of Rochester, [redacted] Ex.6
Ph.D	Physical Chemistry, University of California at Berkeley, 1956

Issued: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Vice President, Finance and Administration

PERSON Kenneth L. Wiley

EXPERIENCE

Mr. Wiley has over twenty-two years experience in a variety of administrative positions throughout a major energy company. Assignments have included all levels of accounting and financial control for production and mining operations. He also has had extensive administrative and managerial experience within various operating divisions.

FORMER POSITIONS

1969 - 1971	Manager, Finance and Services, Gulf Mineral Resources Division, Gulf Oil Corporation
1967 - 1969	Division Accountant, Gulf Mineral Resources Division, Gulf Oil Corporation
1964 - 1967	Area Accounting Supervisor, Gulf Oil Corporation
1961 - 1967	Senior Unit Accounting Supervisor, Gulf Oil Corporation
1956 - 1960	Unit Accounting Supervisor, Gulf Oil Corporation
1954 - 1956	Group Accounting Supervisor, Gulf Oil Corporation
1953 - 1954	Assistant Area Accounting Supervisor, Gulf Oil Corporation
1948 - 1953	Various Accounting Positions, Gulf Oil Corporation

EDUCATION

B.S.	Business Management, Oklahoma State University, [redacted] Ex. 6
	Graduate Studies Leading to MBA, Finance, University of Colorado

Issued: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Vice President, Manufacturing Operations

PERSON Frederick G. Stengel

EXPERIENCE

Mr. Stengel has over eighteen years experience in development and fabrication of nuclear core fuel materials, fuel elements, and assemblies. His developmental experience includes several prototype naval reactors, PWR and HTGR. He has extensive background in the development of uranium alloy, carbide, and in management of related technical programs.

FORMER POSITIONS

1967 - 1971	General Manager, United Nuclear Corporation, Commercial Products Division
1965 - 1967	Chemical Operations Manager, United Nuclear Corporation, Fuels Division
1963 - 1965	Manager, Process Engineering, General Atomic, Fuel Operations Division
1958 - 1963	Supervisor, Westinghouse Electric Corporation, Bettis Atomic Power Division
1952 - 1958	Junior to Senior Engineer, Westinghouse Electric Corporation, Bettis Atomic Power Division

EDUCATION

B.S.	Metallurgical Engineering, Massachusetts Institute of Technology, [redacted]
B.A.	University of Pittsburgh, 1959

Ex. 6

Issued: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Vice President, Utility Fuel Engineering

PERSON Richard A. Dean

EXPERIENCE

Dr. Dean has over 13 years experience in the design and development of fuel rods, fuel assemblings and core internals, including the thermal and hydraulic design of commercial LWR's and most recently, total responsibility for a company sponsored utility fuel development program.

FORMER POSITIONS

1970 - 1971	Technical Director, LWR Fuel Division, Gulf General Atomics Corp., LaJolla, California
1966 - 1970	Nuclear Energy Systems Manager, Westinghouse Electric Corp.
1959 - 1966	Junior to Senior Engineer, Westinghouse Atomic Power Division, Westinghouse Electric Corp.

EDUCATION

B.S.	Mechanical Engineering, Georgia Institute of Technology, [redacted] Ex. 6
M.S.	Mechanical Engineering, University of Pennsylvania, 1963
PhD	Mechanical Engineering, University of Pennsylvania, 1970

Issued: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Projects Manager

PERSON William J. Compas

EXPERIENCE

Mr. Compas has had over twenty years experience in technology of nuclear reactor cores and nuclear material fabrication and testing, with particular emphasis on uranium alloys, zircaloy and hafnium. He has extensive background in quality control and process development and improvement.

FORMER POSITIONS

1971 - 1973	Fabrication Operations Plant Manager, Gulf United Nuclear Fuels Corporation.
1968 - 1971	Fabrication Operations Plant Manager, United Nuclear Corporation, Commercial Products Division.
1965 - 1967	Quality Control Manager, Naval Products Department, Fuels Division, United Nuclear Corporation.
1962 - 1965	Manager, Manufacturing Engineering, Fuels Division, United Nuclear Corporation.
1961 - 1962	Superintendent of Engineering, Fuels Division, United Nuclear Corporation.
1958 - 1961	Technical Project Supervisor, Nuclear Fuel Operation, Olin Mathieson Chemical Corporation.
1957 - 1958	Senior Process Engineer, Nuclear Fuel Operation, Olin Mathieson Chemical Corporation.
1956 - 1957	Process Engineer, Nuclear Fuel Operation, Olin Mathieson Chemical Corporation.
1954 - 1956	Supervisor, Process Development in Establishing Roll Bond Tube in Sheet Technology, Olin Mathieson Chemical Corporation.
1952 - 1954	Research Metallurgist, Ames Laboratory (AEC) Iowa State University.

EDUCATION

B.S. St. Louis University, [redacted] Ex. 6

Graduate Studies at Iowa State University, 1952 - 1954

Issued: 4/21/72
Re-issued: 5/1/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Regulatory Administration Manager

PERSON Peter Loysen

EXPERIENCE

Mr. Loysen has over eighteen years of comprehensive experience in radiation protection. His experience includes six years of Health and Safety Program direction in nuclear fuel and large isotopic source operations and six years of occupational and environmental health surveillance of AEC contractor facilities. In addition, he has four years of experience preparing and reviewing criticality evaluations and auditing nuclear safety programs. Mr. Loysen served on the committee that prepared the American Standard, Radiation Protection in Nuclear Fuel Fabrication Plants.

FORMER POSITIONS

1971 - 1972	Nuclear & Industrial Safety Manager; Gulf United Nuclear Funds Corporation
1970 - 1971	Nuclear & Industrial Safety Manager, United Nuclear Corporation, Commercial Products Division
1968 - 1970	Radiation Counsel, Radiation Machinery Corporation
1961 - 1968	Assistant to the Director, Health Protection Engineering Division, U.S. Atomic Energy Commission
1956 - 1961	Health and Safety Director, Metals and Controls Division, Texas Instruments, Incorporated
1952 - 1956	Industrial Hygienist, U.S. Atomic Energy Commission

EDUCATION

B.ChE.	Rensselaer Polytechnic Institute, Troy, New York, [redacted] Ex. 6
	Harvard University School of Public Health, Industrial Air Analysis, 1954
	Oak Ridge National Laboratory, Nuclear Safety Training Course, 1957
	Commonwealth of Massachusetts, Civil Defense Training Course, 1960

LICENSE

Certified Health Physicist, No. 62-97

Issued: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION — Consultant

PERSON — Robert E. Kropp

EXPERIENCE

Mr. Kropp has over nineteen years experience in nuclear safety, reactor hazards analysis, reactor physics and reactor design. His experience covers the design, fabrication and operation of Swimming Pool, Pressurized Water and High Temperature Gas Cooled Reactors. He has extensive background in directing nuclear safety programs, preparing manuals and procedures for criticality control and reactor operations and preparing computer programs for health and safety analysis. Mr. Kropp serves on the American National Standards Institute committee on Transportation of Radioactive Materials.

FORMER POSITIONS

1971-1972	Nuclear Safety Specialist, Gulf United Nuclear Fuels Corporation
1968-1971	Nuclear Safety Specialist, United Nuclear Corporation, Commercial Products Division
1965-1968	Nuclear Safety Specialist, Fuels Division, United Nuclear Corporation
1963-1965	Staff Associate, Nuclear Analysis and REactor Physics Department and Member, Criticality Safeguard Committee, John J. Hopkins Laboratory for Pure and Applied Sciences, General Atomic
1958-1963	Lead Engineer, Criticality Control Standards, Bettis Atomic Power Laboratory, Westinghouse Electric Corporation
1956-1957	Test Engineer, Reactor Operations and Hazards Group, Convair, Fort Worth Division
1952-1956	Aerological and Research Officer, U.S. Navy
1951-1952	Meteorological Aid, U.S. Weather Bureau

EDUCATION

B.S.	Meteorology, Florida State University, [redacted] Ex. 6
	Graduate Physics, University of California, 1954-1955
	Graduate Physics, Texas Christian University, 1956-1957
	Bettis School of Reactor Engineering, 1958-1961
M.B.A.	General Management, University of New Haven, 1971

LICENSE

Certified Safety Professional, No. 2683
U. S. Atomic Energy Commission Operator License, OP-1800, 1964 (Not active)

Issued: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Health Physics Specialist

PERSON David G. Darr

EXPERIENCE

Mr. Darr has over eighteen years experience in Health Physics involving source and special nuclear materials, by-product materials, x-ray equipment and particle accelerators. His background includes planning and administering Health Physics programs, preparing manuals and procedures for Health Physics controls, equipment and facility design, and environmental sampling and evaluation.

He has four years experience in nuclear safety involving audit functions for both reactor component fabrication and special nuclear materials processing. One year of this experience included performing basic nuclear safety evaluations and reviewing and preparing manuals and procedures for criticality control. Mr. Darr has completed the Gulf United Nuclear Criticality Safety Training Program.

FORMER POSITIONS

1968 - 1971	Health Physics Specialist, United Nuclear Corporation, Commerical Products Division
1967 - 1968	Nuclear Licensing and Safety Specialist, United Nuclear Corporation, Fuels Division
1964 - 1967	Health Physics and Safety Supervisor, United Nuclear Corporation, Fuels Division
1961 - 1964	Health Physics Officer, Member of Isotope Committee and Consulting Industrial Physicist, Nuclear Consultants Corporation
1957 - 1961	Health Physics Supervisor, Nuclear Fuels Operation, Olin Mathieson Chemical Corporation
1956 - 1957	Technician, Health Physics Department, Uranium Division, Mallinckrodt Chemical Works
1954 - 1956	NCOIC, Operations Group, 1st Radiological Safety Support Unit, U.S. Army, Nevada Test Site and Ft. McClellan, Alabama

EDUCATION

Engineering	Central Missouri State College, [redacted] Ex. 6
Engineering	Washington University, 1957
Physics	New Haven College, 1960

Issued: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Consultant, Health Physics

PERSON Percy E. Clemons

EXPERIENCE

Mr. Clemons has over twelve years experience in all phases of a Health and safety program associated with Plutonium and uranium facilities. This includes laboratory, metal working and critical facility operations. His background includes responsibility for personnel monitoring, bioassay program, environmental monitoring, handling of SNM and other hazardeous materials, waste disposal, transportation of radioactive materials and licensing.

FORMER POSITIONS

1960 - 1971	Director of Health and Safety, United Nuclear Corporation, Research and Engineering Center
1958 - 1960	Research Physist, Curtiss-Wright Corporation, Research Division
1957 - 1958	Health Physics Trainee, Brookhaven National Laboratory

EDUCATION

B.S.	Physics, Hampton Institute, [redacted] Ex.6
Graduate Study	Health Physics, University of Rochester, 1957 - 1958

Issued: 4/21/72

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Consultant, Nuclear Criticality Safety

PERSON Edward Fass

EXPERIENCE

Mr. Fass has over four years experience in reactor design and analysis, and heat transfer. His background includes development and testing of advanced analytical methods capable of predicting behavior of light water power reactors. In addition he has over three years experience in performing nuclear criticality safety calculations using reactor design codes and the KENO criticality safety code.

FORMER POSITIONS

1969 - 1971	Consultant, Nuclear Criticality Safety, Commercial Products Division and Nuclear Engineer, Research and Engineering Center, United Nuclear Corporation
1968 - 1969	Graduate Assistant and Radiation Safety Officer, New York University, Nuclear Engineering Department

EDUCATION

B.S.	Engineering Physics, New York University, [redacted] Ex.6
M.E.	Nuclear Engineering, New York University, 1968

Issued: 4/21/72

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Consultant, Nuclear Criticality Safety (Physicist, Engineering Operation)

PERSON James H. Ray

EXPERIENCE

Mr. Ray has over twenty-two years experience in reactor analysis, nuclear data evaluation and compilation, radiation shielding, and criticality safety. His background also includes heat transfer, fluid flow measurement, temperature measurement, high pressure measurement, combustion equilibrium calculation, and radio-frequency noise measurement and reduction.

FORMER POSITIONS

1968 - 1971	Physicist, Research and Engineering Center United Nuclear Corporation
1962 - 1968	Physicist, Physics and Mathematics Department, United Nuclear Corporation
1957 - 1962	Physicist, Physics and Mathematics Department, Nuclear Development Corporation of America
1955 - 1957	Physicist, Physics and Mathematics Department, Nuclear Development Associates
1954 - 1955	Physicist, Philips Laboratories Division, North American Philips Company
1952 - 1954	Engineer, Electro-Search Company (Summers only)
1947 - 1951	Physicist, Reaction Motors, Inc.

EDUCATION

A.B.	Physics, Harvard University, [redacted] Ex.6
M.S.	Physics, University of Pennsylvania, 1954

Issued: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Nuclear and Industrial Safety Representative, FabricationOperations

PERSON Harold E. Clow, Jr.

EXPERIENCE

Mr. Clow has over sixteen years experience in health physics, industrial hygiene and safety, nuclear criticality safety and nuclear materials management. He has had practical experience in fuel production facilities, research and development laboratories and operating reactors.

This background covers preparing operating and auditing manuals and procedures for all phases of health and safety control as well as responsibilities for supervision of health physics for a major site including four operating reactors, a Hot Lab and a major waste disposal facility. Assignments have included membership on a radiological emergency re-entry team, field assignment to an operating reactor and auditing for health physics, nuclear criticality safety and nuclear materials management.

FORMER POSITIONS

1971 - Present	Associate Health Physist - Nuclear Materials Management, Atomics International
1967 - 1971	Associate Health Physist - Criticality Safeguards Staff, Atomics International
1959 - 1967	Associate Health Physist - Health, Safety and Radiation Services, Atomics International
1957 - 1959	Junior Health Physics Engineer - Bettis Atomic Power Laboratory
1956 - 1957	Health Physics Technician - Bettis Atomic Power Laboratory

EDUCATION

Science and Engineering	Pittsburgh Technical Institute (Junior College) 1955-1956.

ISSUED: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Manager, Uranium Fuel Development Laboratory

PERSON Nathan Fuhrman

EXPERIENCE

Dr. Fuhrman has twenty years experience in materials technology of which the past fourteen years have been in the area of nuclear applications. Responsibilities have included the research and development of high loaded refractory metal matrix cermet fuel, partical coating of uranium fuels, and low temperature sintering of uranium fuels. Other assignments have included evaluation of process specifications for various nuclear fuel materials.

FORMER POSITIONS

1959 - Present	Manager, Chemistry and Ceramics, Gulf United (formerly United Nuclear, REC)
1957 - 1959	Supervised the development of a vapor deposition process for the production of silicon from silan, Lansdale Tube Company, Division of Philco Corporation
1954 - 1957	Development, Sylvania Corning Nuclear Corporation
1953 - 1954	Research, Polytechnic Institute of Brooklyn

EDUCATION

B.Ch.E.	Rensselaer Polytechnic Institute, [redacted] Ex.6
Ph.D.	Polytechnic Institute of Brooklyn, 1953 (Physical Chemistry)

ISSUED: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Manager, Materials Technology Laboratory

PERSON George Stern

EXPERIENCE

Mr. Stern has over thirty years total experience in materials research and metallurgy of which the past fifteen years have been in the area of nuclear fuels research and development. Responsibilities have included fuel and materials selection criteria for reactor cores, liquid metal, analytical techniques and process specifications for heat treating, welding, brazing, and corrosion testing.

FORMER POSITIONS

1958 - Present	Manager, Materials Technology Department, Gulf United (formerly United Nuclear, REC)
1955 - 1958	Technical Director, American Sinteel Corporation
1944 - 1955	Technical Director, American Electro Metal Corporation
1941 - 1944	Associate Metallurgist, Armour Research Foundation
1940 - 1941	Metallurgist, American Electro Metal Corporation
1939 - 1940	Metallurgist, New York Testing Laboratories

EDUCATION

B.Ch.E.	College of the City of New York, [redacted] Ex.6
M.S.	University of Michigan, 1939 (Metallurgical and Chemical Engineering)

ISSUED: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Chief Scientist

PERSON Robert B. Holden

EXPERIENCE

Dr. Holden has over twenty-six years of comprehensive experience in the management and conduct of nuclear research and development, ~~including: nuclear materials and transport properties~~, high-temperature thermodynamics processing and consolidation of ceramics and metals, high-temperature behavior of nuclear fuels and of impurities in liquid metal systems. He serves as principal advisor on all scientific matters in which the company may be involved.

Dr. Holden has authored a large number of publications in the scientific literature and is a member or fellow of several professional societies.

FORMER POSITIONS

1969-1971	Manager, Research Operations, United Nuclear Corporation, Research and Engineering Center
1961-1969	Manager, Chemistry and Ceramics Department, United Nuclear Corporation, Research and Engineering Center
1957-1961	Chief, Chemistry and Ceramics Section, Nuclear Fuel Research Laboratory, Olin Mathieson Chemical Corporation
1950-1957	Senior Research Engineer and Head, Chemistry Section and Ceramics Section, Sylvania Electric Products, Inc., Atomic Energy Division
1947-1949	Research Fellow, Ohio State University
1946-1947	Research Assistant Chemist, Monsanto Chemical Corporation (Manhattan Project)

Ex. 6

EDUCATION

A. B.	University of Missouri, Mathematics and Chemistry, [redacted]
	Cornell University, 1944-1945
Ph.D.	Ohio State University, Physical Chemistry, 1950

Issued: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Advisory Scientist

PERSON Walter L. Brooks

EXPERIENCE

Dr. Brooks has over twenty-six years of experience in reactor experimentation, operation and control. He has developed unique instrumentation and techniques for experimental reactor and other programs. Dr. Brooks is a licensed senior reactor operator and superivsor of the company's two reactors.

FORMER POSITIONS

1953-1971	Advisory Scientist, United Nuclear Corporation, Research and Engineering Center
1948-1953	Physics Researcher, New York University, Millimeter Wave Project
1946-1947	Instructor in Physics, Lincoln Memorial University

EDUCATION

B. A.	Lincoln Memorial University, Mathematics, [redacted] Ex. 6
M. S.	New York University, Physics, 1950
Ph.D.	New York University, Physics, 1953

Issued: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Consultant Engineer, Nuclear Design Department

PERSON Peter Buck

EXPERIENCE

Dr. Buck has over fifteen years of experience in both experimental and theoretical determinations or nuclear characteristics of power reactors and critical assemblies. His experience also includes fuel management for reload cores.

FORMER POSITIONS

1969-1971	Consultant Engineer, United Nuclear Corporation, Research and Engineering Center
1965-1969	Manager, Utility Core Analysis Section, United Nuclear Corporation, Research and Engineering Center
1957-1965	General Electric Company, Knolls Atomic Power Laboratory

EDUCATION

B.A.	Bowdoin College, Physics, [[redacted]] Ex. 6
Ph.D.	Columbia University, Physics, 1958

Issued: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION: Manager, Physics and Mathematics Department

PERSON: Robert D. Schamberger

EXPERIENCE

Dr. Schamberger has more than twenty-one years of broad experience in the direction and performance of nuclear physics analyses, and in radiation transport phenomena and nuclear reactor shielding. He is the anuthor of a number of publication is the scientific literature dealing with these subjects.

FORMER POSITIONS

1958 - 1971	Manager, Physics and Mathematics Department, United Nuclear Corporation, Research and Engineering Center
1956 - 1958	Supervisory Physicist, Wright Air Development Center, Aeronautical Laboratory
1951 - 1956	Associate Physicist, Bookhaven National Laboratory

EDUCATION

B.S.	Union College, Physics, [redacted] Ex.6
Ph.D.	University of Rochester, Physics, 1951

Issued: 1/19/73

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Fabrication Operations Plant Manager

PERSON Glenn O. Amy

EXPERIENCE

Mr. Amy has over twenty-three years of professional management and technical experience in the startup, operation, and maintenance of nuclear reactors and associated facilities, and in the manufacture of fuel and target elements and special reactor products for reactor use.

FORMER POSITIONS

1971 - 1973	Chemical Operations Plant Manager, Gulf United Nuclear Fuels Corporation	
1970 - 1971	Chemical Operations Plant Manager, United Nuclear Corporation, Commercial Products Division	
1967 - 1970	Douglas United Nuclear, Inc., Richland, Washington:	
	1969 - 1970	Manager, Manufacturing Engineering
	1968 - 1969	Manager, Fuels Manufacturing
	1967 - 1968	Manager, Shop Operations
1949 - 1967	General Electric Company, Richland, Washington:	
	1967 -	Acting Manager, N Reactor Plant
	1965 - 1967	Manager, N Reactor Plant Maintenance
	1962 - 1965	Manager, N Reactor Operations
	1960 - 1962	Analyst, Reactor Operations
	1956 - 1960	Supervisor, Plant Engineering
	1955 - 1956	Group Leader, Reactor Operation Engineering
	1955 -	Supervisor, Drafting and Design
	1951 - 1955	Process Engineer
	1949 - 1951	Radiation Engineer

EDUCATION

B.S. Mechanical Engineering, University of Denver, [redacted] Ex. 6

Issued 5/1/73
Replaces: 4/21/72

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Engineering Manager, Fabrication Operations

PERSON Eugene Krinick

EXPERIENCE

Mr. Krinick has over eighteen years experience in the fabrication of nuclear fuel elements. Work in this field includes development and production on aluminum-uranium alloys, stainless steel UO_2 dispersion plates, rolling, machining and casting of uranium and critical experiments; fabrication of plate type and tubular aluminum fuel elements for research and test reactor use, and the fabrication of UO_2 pellet - Zircaloy tubular clad fuel elements for many utility power reactors.

FORMER POSITION

1968 - 1971	Engineering Manager, United Nuclear Corporation, Commercial Products Division, Fabrication Operations
1967 - 1968	Engineering Manager, United Nuclear Corporation, Fuels Division
1965 - 1967	Supervisor of Mechanical Design, United Nuclear Corporation, Fuels Division
1961 - 1965	Engineering Specialist, United Nuclear Corporation, Fuels Division
1959 - 1961	Engineering Specialist, Olin Mathieson Chemical Corporation, Nuclear Fuel Operation
1955 - 1959	Supervisor, Mechanical Engineering, Sylvania-Corning Nuclear Corporation
1954 - 1955	Project Engineer, Sylvania-Corning Nuclear Corporation
1953 - 1954	Rocket Design Engineer, U.S. Army
1952 - 1953	Mechanical Engineer, Atomic Energy Division, Sylvania Electric Products, Inc.
1951 - 1953	Mechanical Engineer, Norfolk Naval Shipyard

EDUCATION

B.S.	Mechanical Engineering, Brooklyn Polytechnic Institute, [[redacted]] Ex. 6
M.S.	Industrial Engineering, New York University, 1958

Issued: 4/21/72

GULF UNITED
NUCLEAR FUELS CORPORATION

<u>POSITION</u> Manager, Administrative Services, Fabrication Operatio

<u>PERSON</u> Miles P. Wittner

<u>EXPERIENCE</u>

Mr. Wittner has over ten years experience in the nuclear and defense industries. His background includes assignments in industrial engineering, project management, quality assurance and nuclear fuel manufacturing. Industrial engineering experience includes establishing and integrating financial controls and reporting including budget preparation, performance reviews, cost estimates for capital programs and customer projects, establishment of financial models for facilities and planning and performance of make or buy studies on broad manufacturing scopes. Project management experience includes planning and implementing production, manufacturing engineering, purchasing, quality and production control. Quality assurance experience includes program planning, customer interface administration, analysis and implementation of contractural requirements, design reviews, etc. Nuclear fuel manufacturing experiences covers Naval Reactors, EBR, and utility fuel fabrication.

<u>FORMER POSITIONS</u>

1971 - 1973	Staff Industrial Engineer, Manufacturing Operations Gulf United Nuclear Fuels Corporation
1970 - 1971	Industrial Engineer, Chemical Operations Gulf United Nuclear Fuels Corporation
1968 - 1970	Manufacturing Project Manager, Hamilton Standard
1965 - 1968	Senior Quality Assurance Engineer, Hamilton Standard
1963 - 1965	Quality Engineer, Hamilton Standard
1961 - 1963	Manufacturing Management Program, General Electric

<u>EDUCATION</u>

B.S.I.E.	Industrial Management and Mechanical Engineering, Pratt Institute, [redacted] Ex. 6
M.B.A.	Finance and Management, University of Hartford, 1970

Issued: 5/31/73
Replaces: 8/11/72

GULF UNITED
NUCLEAR FUELS CORPORATION

POSITION Production Manager, Gulf United Nuclear Fuels Corporation

PERSON Nicholas C. Kazanas

EXPERIENCE

Mr. Kazanas has six years experience in the nuclear fuel industry. His experience includes quality control and process development for nuclear materials. This experience includes an extensive background in powder metallurgy, alloy shop production, forming and roll bonding. He has participated in materials research and product application directed at the fabrication of research reactor and utility reactor type fuel elements. His background also includes six years experience in the aerospace industry. This experience includes materials research and product development. Mr. Kazanas presently serves as an adjunct Assistant Professor of Statistics and Mathematics at Quinnipiac College.

FORMER POSITIONS

1972 - Present	Production Manager, Fabrication Operations, Gulf United Nuclear Fuels Corporation, New Haven, Connecticut
1970 - 1972	Quality Control Superintendent, Fabrication Operations, Gulf United Nuclear Fuels Corporation, New Haven, Connecticut
1969 - 1970	Engineer in charge of Metallurgy, Fuels Division, United Nuclear Corporation, New Haven, Connecticut
1967 - 1969	Engineering Specialist, Fuels Division, United Nuclear Corporation, New Haven, Connecticut
1964 - 1967	Materials Engineer, Space and Life Systems Division, Hamilton Standard Division, United Aircraft, Windsor Locks, Connecticut
1962 - 1964	Research and Development Engineer, Aerospace Products, Hamilton Standard Division, United Aircraft, Windsor Locks, Connecticut

EDUCATION

B.S.	Metallurgical Engineering, Lafayette College, [redacted] Ex. 6
M.B.A.	Production, Hartford University, 1969

Issued: 11/10/72

SECTION 300

UNITED NUCLEAR
C O R P O R A T I O N

PAGE 1 OF 1

LICENSE: SNM-33, Docket: 70-36

SECTION: 300 - NUCLEAR CRITICALITY SAFETY STANDARDS

Approved

ISSUED October 31, 1968

SUPERSEDES New

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SECTION: 300 - NUCLEAR CRITICALITY SAFETY STANDARDS	ISSUED October 31, 1968
Subsection: 301 - Statement of Policy	SUPERSEDES New

301. Statement of Policy

It is the policy of United Nuclear Corporation to establish management controls, plant facilities and equipment designs, and operating procedures to reduce the possibility of a nuclear criticality accident to a minimum. The standards contained in this section 300, describe the controls and criteria applicable to the implementation of this policy.

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 KΩ Method - Solid angles shall be calculated using the figures in Y-1272.

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. Individual units meeting the fraction critical limit must be spaced to avoid possible critical subarrays of density greater than 175 grams per square foot in addition to requiring that the overall average of the aspect area be below 175 grams per square foot.

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).

2.2.4 Isolation (continued)

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.

Evaluations of isolation must consider unit geometries as well as keff values.

3. <u>Consideration of Fire Hazards</u>

Proposed changes in facilities, equipment or operating procedures will include consideration of fire hazards. Evaluations will consider construction materials, fire detection and fighting equipment, and handling of pyrophoric or highly combustible materials.

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.

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 ≤ 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

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

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

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

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. Safe U-Al Fuel Element Wet Zone Limit (3 Elements)

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

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

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

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

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

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

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

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

$$= 10.04 \text{ cm}$$

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,

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

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

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

$$= 14.71 \text{ cm}$$

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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/l, the equivalent mass is

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

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

$$B^2_c = \frac{K_\infty - K_{eff}}{M^2} \qquad \text{where } K_{eff} = 1$$

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

$$= 0.0099 \text{ cm}^{-2} \qquad \text{where } \delta = 2.5 \text{ cm}$$

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/l, the critical mass is,

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

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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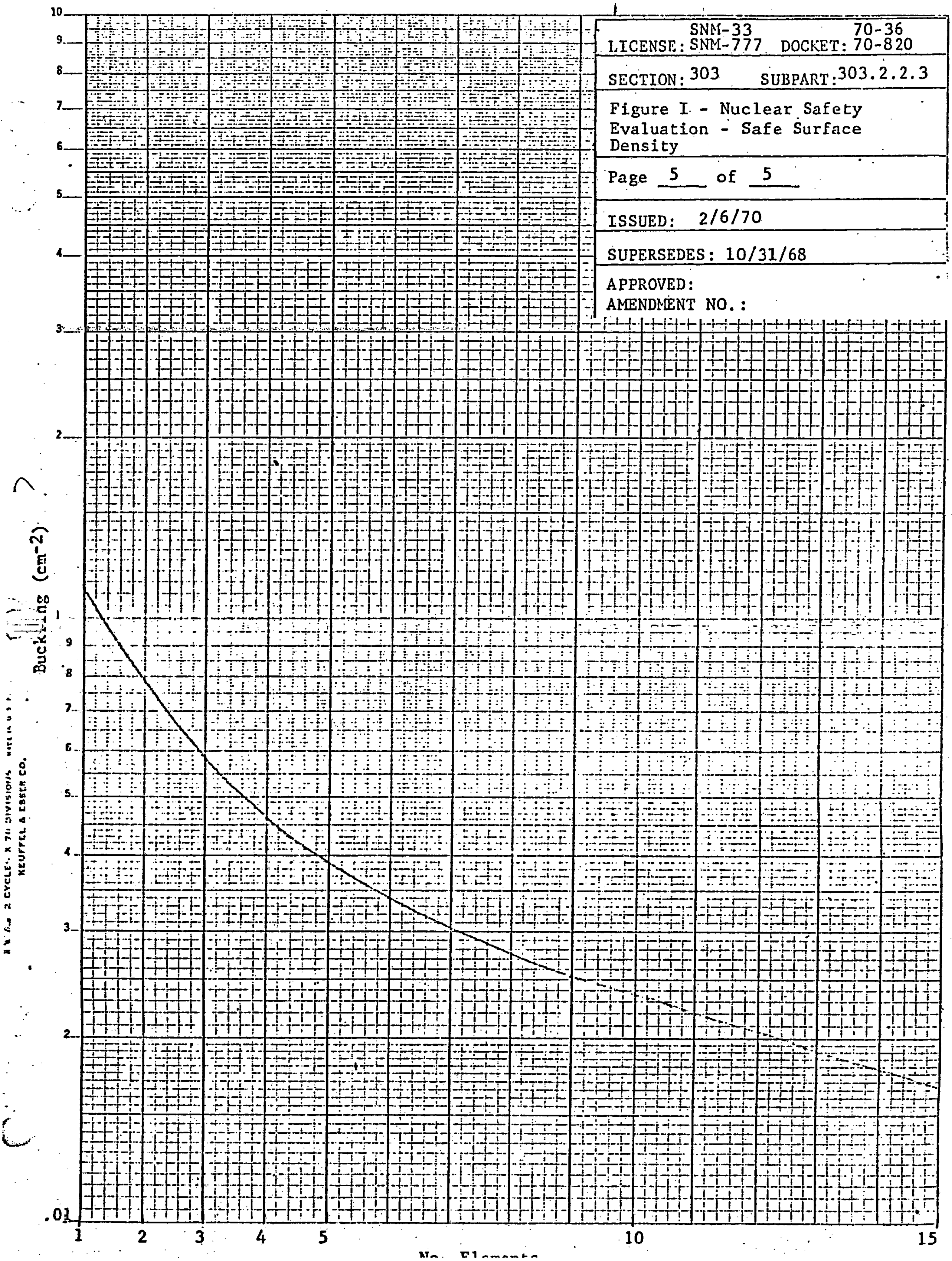
Figure I - Nuclear Safety Evaluation - Safe Surface Density

ISSUED: 2/6/70

SUPERSEDES: 10/31/68

APPROVED:
AMENDMENT NO.:



UNITED NUCLEAR
C O R P O R A T I O N

LICENSE: SNM- 33, Docket: 70-36
SECTION: 300 - NUCLEAR CRITICALITY SAFETY STANDARDS
Subsection: 304 - Structual Integrity

Approved-

ISSUED October 31, 1968

SUPERSEDES New

304. Structural Integrity

1. Purpose

Whenever nuclear criticality safety is directly dependent on the integrity of a fixture, container, storage rack or isolation structure, the fixture, container, storage rack or isolation sturcture shall be designed in accordance with the following criteria.

2. Specifications

2.1 Materials shall be selected to be fire and corrosion resistant.

2.2 The safety factor is at least five (5) (applicable to ultimate strength of material at design conditions). Assurance that the conditions of this Section are met will be accomplished by test or design by an engineer knowledgeable in material properties and design.

2.3 Records of test results and design calculations will be maintained as provided in Subsection 206.

3. Inspections

Fixtures, containers, storage racks, or isolation structures which maintain a safe geometry or spacing will be inspected to assure the continued reliability of such devices.

3.1 Fixtures Exposed to Corrosive Enviornments

3.1.1 Fixtures such as pickling fixtures will be visually inspected for defects such as cracks at least monthly. These checks will be performed by operating supervision who will maintain a record of these checks.

3.1.2 Defective fixtures will be withdrawn from service and repaired. Fixtures shall be inspected by operating supervision to insure that original design conditions have been restored.

3.2 Other Devices

3.2.1 Devices such as storage racks and containers shall be checked by NIS personnel during inspections and audits.

3.2.2 Devices requiring repair shall be identified and repaired. Repaired devices shall be inspected by NIS personnel or operating supervision to insure that original design conditions have been restored.

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 Boro 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

305. Nuclear Poisons (continued)

2.1 Soluble Salts

Soluble boron or cadmium salts may be added to drums of solutions such as pickle liquids, analytical laaboratory residues or other solutions when primary nuclear criticality safety is based on concentration control. Such use shall be as described herein.

2.1.1 Specifications

Maximum solution concentrations are 10 grams U235 per liter. Quantity of salt shall be established to maintain equal molal quantities of U235 and boron or cadmium . (Reference: TID 7016, Rev. 1, page 32, Soluble Poisons)

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

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LICENSE: SNM-33 & SNM-777, Docket: 70-36 & 70-820 SECTION: 300 - NUCLEAR CRITICALITY SAFETY STANDARDS Subsection: 307 - Marking and Labeling	Supersedes 10/31/68
	Approved
	Amendment No.

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

Approved:

SUBJECT: LICENSE: SNM-33, Docket: 70-36
SECTION: 300 - NUCLEAR CRITICALITY SAFETY STANDARDS

ISSUED October 31, 1968

Subsection: 308 - Equipment Design

SUPERSEDES

308. Equipment Design

1. Purpose

Certain criteria apply to the all phases of equipment design. These criteria are considered in the design of all equipment used in the processing of SNM.

2. Specifications

2.1 Vessels of unsafe geometry shall be separated by air breaks or other positive method from safe geometry vessels used for SNM bearing solutions to prevent siphoning the SNM bearing solutions into the unsafe geometry vessels.

2.2 Catch pans which are located under some of the equipment are for the purpose of controlling any minor leaks or drips. This improves housekeeping, reduces the spread of contamination, and reduces SNM losses. The depth or volume of the pans shall not exceed the safe slab thickness or volume established for the area.

2.3 The diameter or volume of overflow and vent bottles shall not exceed the safe diameter or volume established for the process area.

2.4 Insulation on pipes and equipment which contain SNM bearing solutions will be made of impervious materials (e.g., foam glass) and provided with weep holes to prevent the possibility of obtaining greater dimensions (volume, diameter, etc.).

LICENSE: SNM-33, Docket: 70-36
SECTION: 300 - NUCLEAR CRITICALITY SAFETY STANDARDS
Subsection: 309 - Tables and Graphs
Subpart:

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.

3. Density-Moderator Relationship

Provisions will be made to insure the continued validity of the maximum density - moderator relationship of the uranium oxide as expressed in Figure 309-XXIV. As process conditions are varied which could affect the oxide bulk density, such as reaction temperatures or drying temperatures, additional tests or experiments will be conducted to determine the oxide properties.

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.

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 container 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 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) = .960

keff (bare) = .780

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LICENSE: SNM-871, DOCKET: 70-9[illegible]
SECTION: 300, SUB PART: 309-
NUCLEAR SAFETY EVALUATION - KEFF'S OF 4.8 LITER UO_2 SPHE[illegible]
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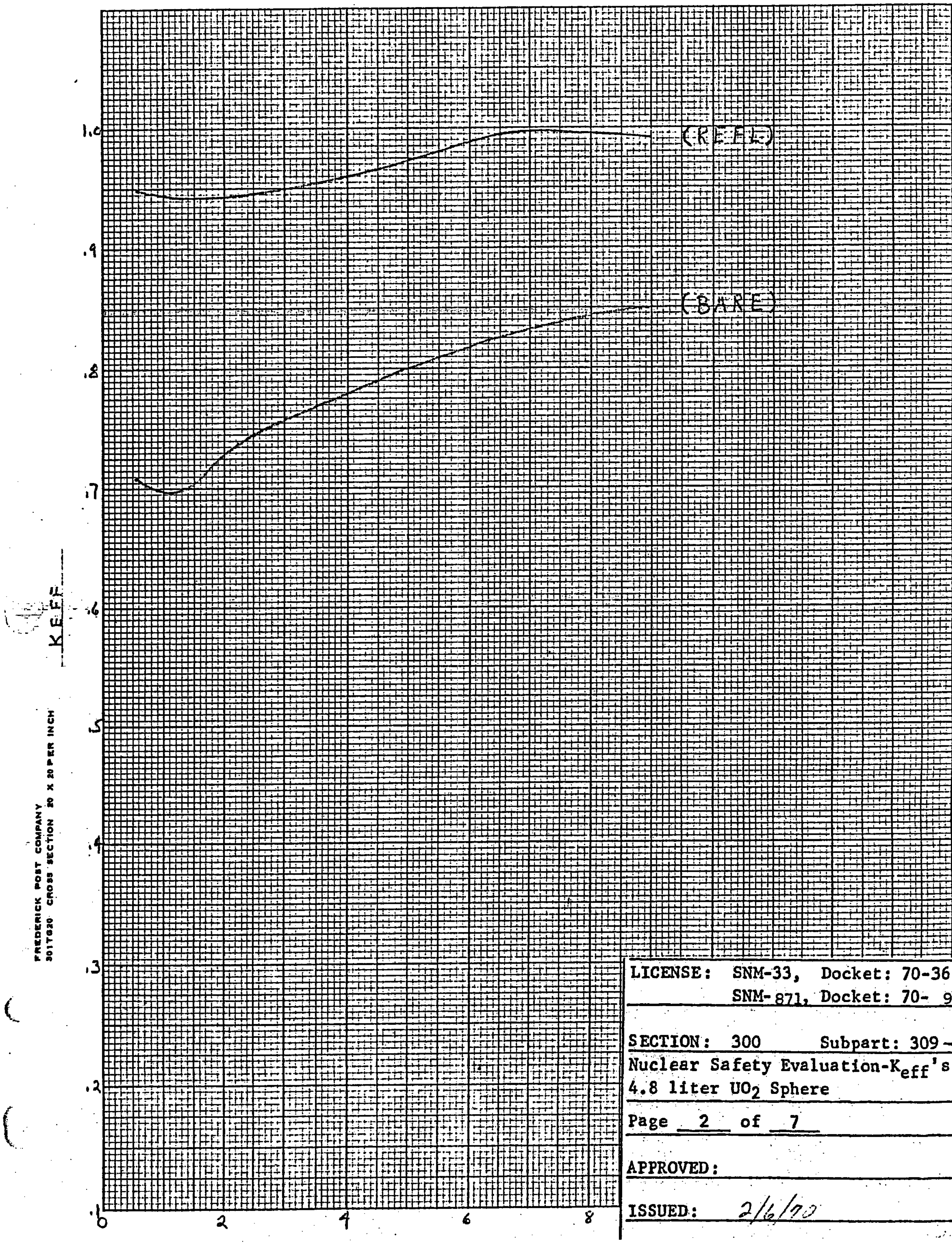

LICENSE: SNM-33, Docket: 70-36
SNM-871, Docket: 70- 903

SECTION: 300 Subpart: 309-T

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

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CRITICAL (U93.5) [illegible] WATER VOLUMES

H/U-235	ρ (Kg/l)	$r_{c,R}$ [1] (cm)	$V_{c,R}$ [2] (liters)	δ [3] (cm)	$B_{c,R}(s)$ [4] (cm^{-2})	$r_{c,B}$ [1] (cm)	$V_{c,B}$ [2] (liters)	δ [5] (cm)	$B_{c,B}(s)$ [6] (cm^{-2})
0	8.930	10.63	5.03	7.30	.0307	13.43	10.15	5.46	.0277
.98	6.700	10.49	4.84	7.26	.0313	13.90	11.25	5.32	.0267
2.94	4.470	11.14	5.79	7.18	.0294	14.87	13.78	5.03	.0249
8.96	2.230	11.75	6.80	6.94	.0288	15.85	16.68	4.13	.0247
20.60	1.120	11.53	6.42	6.50	.0304	15.60	15.91	2.50	.0301
43.90	0.558	11.39	6.19	6.50	.0308	15.27	14.92	2.50	.0313

NOTES: (1) Data from Table V K, LA-3612

(2) $V = 4.19\ r^3$

(3) $\delta = 7.3 - .040\ (H/U\text{-}235)$, cm; for thick water reflector at $H/U\text{-}235 \leq 20$; $\delta = 6.5$ cm at $H/U\text{-}235 > 20$

(4) $B_{c,R}(s) = \left(\frac{\pi}{r_{c,R}(s) + \delta}\right)^2$

(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 $H/U\text{-}235 > 20$

(6) $B_{c,B}(s) = \left(\frac{\pi}{r_{c,B}(s) + \delta}\right)^2$

CRITICAL BUCKLINGS

REFLECTED

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

$$B^2 = \frac{9.87}{(10.63+1.3)^2} = \frac{9.87}{(17.93)^2} = \frac{9.87}{321.48} = .0307$$

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

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

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

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

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

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

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

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

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

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

BARE

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

$$B^2 = \frac{9.87}{(13.43+5.46)^2} = \frac{9.87}{(18.89)^2} = \frac{9.87}{356.83} = .027$$

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

$$B^2 = \frac{9.87}{(13.90+5.32)^2} = \frac{9.87}{(19.22)^2} = \frac{9.87}{369.44} = .0267$$

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

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

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

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

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

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

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

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

KEFF VALUES FOR 4.8 LITER SPHERICAL VOLUME U(93.5)O_2-WATER MIXTURE

H/U-235	ρ (Kg/l)	$r_{4.8\,l\,S}$ (cm)	δ_R (cm)	$B^2_{S,R}$ (1) (cm^{-2})	τ (2) (cm^2)	Keff (3)	δ_B (cm)	$B^2_{S,B}$ (1) (cm^{-2})	τ (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) τ = 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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SUPERSEDES: NEW

4.8 LITER SPHERE BUCKLING

REFLECTED

At H/U-235 = 0, ρ = 8.930 Kg/ℓ

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

At H/U-235 = .98, ρ = 6.700 Kg/ℓ

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

At H/U-235 = 2.94, ρ = 4.470 Kg/ℓ

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

At H/U-235 = 8.96, ρ = 2.230 Kg/ℓ

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

At H/U-235 = 20.60, ρ = 1.120 Kg/ℓ

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

At H/U-235 = 43.90, ρ = 0.558 Kg/ℓ

B^2 = .0344 (same as for ρ = 1.120)

BARE

At H/U-235 = 0, ρ = 8.930 Kg/ℓ

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

At H/U-235 = .98, ρ = 6.700 Kg/ℓ

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

At H/U-235 = 2.94, ρ = 4.470 Kg/ℓ

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

At H/U-235 = 8.96, ρ = 2.230 Kg/ℓ

$$B^2 = \frac{9.87}{(10.45+4.13)^2} = \frac{9.87}{(14.58)^2} = \frac{9.87}{213.58} = .0462$$

At H/U-235 = 20.60, ρ = 1.120 Kg/ℓ

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

At H/U-235 = 43.90, ρ = 0.558 Kg/ℓ

B^2 = .0589 (same as for ρ = 1.120)

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KEFF'S OF 4.8 LITER UO_2 SPH
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4.8 LITER SPHERE KEFF'S

REFLECTED

H/U-235 = 0, ρ = 8.93 Kg/l

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

At H/U-235 = .98, ρ = 6.700 Kg/l

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

At H/U-235 = 2.94, ρ = 4.470 Kg/l

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

At H/U-235 = 8.96, ρ = 2.230 Kg/l

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

H/U-235 = 20.60, ρ = 1.120 Kg/l

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

At H/U-235 = 43.93 ρ = 0.558 Kg/l

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

BARE

At H/U-235 = 0, ρ = 8.93 Kg/l

$$k_{eff} = \frac{1+(28X.0277)}{1+(28X.0390)} = \frac{1.776}{2.092} = .849$$

At H/U-235 = .98, ρ = 6.700 Kg/l

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

At H/U-235 = 2.94, ρ = 4.470 Kg/l

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

At H/U-235 = 8.96, ρ = 2.230 Kg/l

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

At H/U-235 = 20.60, ρ = 1.120 Kg/l

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

At H/U-235 = 43.90, ρ = 0.558 Kg/l

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

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NUCLEAR SAFETY EVALUATION—
KEFF'S OF 4.8 LITER UO_2 SPHERE

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SUPERSEDES: NEW

Safe Limits for Individual Units of Metal and Compounds

Degree of Moderation H/U-235		Mass Limits* (Kgs. U-235)	Keff Reflected	Keff Bare
More Than	Not More Than			
-	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.

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, NDCO-1050.
C. A τ 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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NUCLEAR SAFETY EVALUATION - TABLE 309-II BUCKLINGS & KEFF'S	
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For U Metal - Water Mixture @ H/X ≐ 2

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

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

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

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

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

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

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

With a 10 Kg U^{235} limit

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

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

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

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

The keff's are calculated using

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

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

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

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

$V_{c,R} = 2.7\,l$, $V_{c,B} = 6.0\,l$, $\rho_c = 5.8\ Kg/l$ from Fig. 9, TID-7028

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

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

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

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

$$B^2_{c,R} = \frac{\pi^2}{(r_{c,R}+\delta_R)^2} = \frac{9.87}{(8.65+7.18)^2} = \frac{9.87}{(15.83)^2} = \frac{9.87}{250.59} = .0394\ cm^{-2}$$

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

With a 9 Kg U^{235} limit

$$V_A = \frac{9\ Kg}{5.8\ Kg/l} = 1.55\ l$$

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

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

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

The k_{eff}'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, NDGO-1050

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

$$k_{eff,B} = \frac{1+(28 \times .0372)}{1+(28 \times .0662)} = \frac{2.0416}{2.8536} = .715$$

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NUCLEAR SAFETY EVALUATION - TABLE 309-II BUCKLINGS & KEFF'S
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For U Metal - Water Mixture @ $3 \le H/X \le 5$

$V_{c,R} = 3.2\ \ell$, $V_{c,B} = 7\ \ell$, $\rho_c = 4\ Kg/\ell$ 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

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

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

The keff's are

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

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

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NUCLEAR SAFETY EVALUATION - TABLE 309-II BUCKLINGS & KEFF'S
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For U-Metal-Water Mixture @ $5 \le H/x \le 10$

$V_{c,R} = 3.9\ \ell$, $V_{c,B} = 8.6\ \ell$, $\rho_c = 2.25\ Kg/\ell$ From Fig. 9, TID-7028

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

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

$$\delta_R = 7.3 - .04(H/x) = 7.3 - .4 = 6.9\ cm$$

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

$$B^2_{c,R} = \frac{9.87}{(9.76+6.9)^2} = \frac{9.87}{(16.66)^2} = \frac{9.87}{277.56} = .0356\ cm^{-2}$$

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

With a 5.2 Kg U^{235} limit

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

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

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

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

The keff' are

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

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

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NUCLEAR SAFETY EVALUATION - TABLE 309-II BUCKLINGS & KEFF
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For U Metal - Water Mixture @ $10 \le H/X \le 20$

$V_{c,R} = 4.7\ell$, $V_{c,B} = 10\ell$, $\rho_c = 1.2\ Kg/\ell$ 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

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

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

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For a 3.6 liter volume @ H/≙ 20

$V_A = 3.6\ l$

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

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

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

The keff's are

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

$$k_{eff,B} = \frac{1+(28 \times .0393)}{1+(28 \times .0683)} = \frac{2.1004}{2.9124} = .721$$

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

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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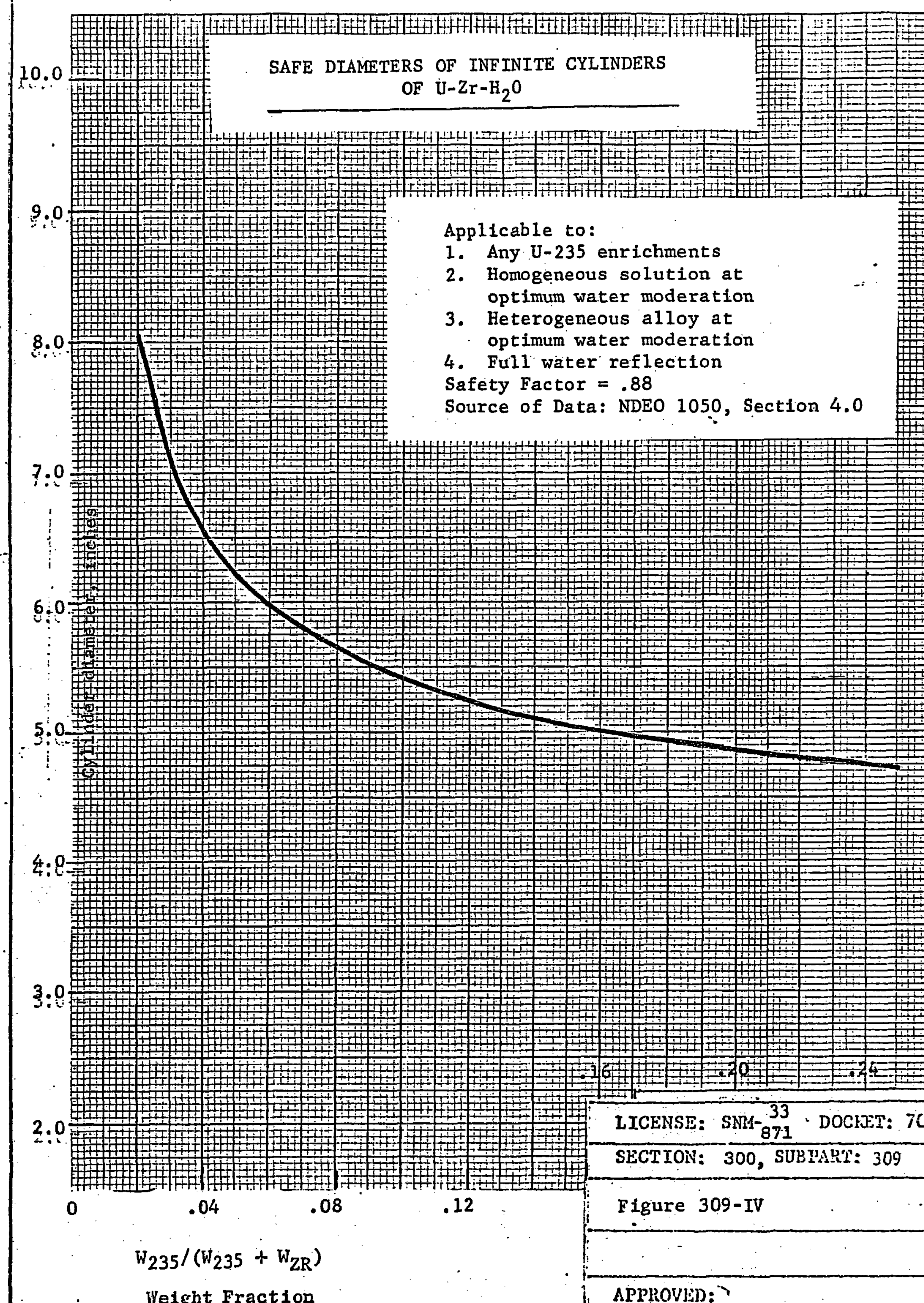
SUPERSEDES: 10/31/68	
APPROVED:	
AMENDMENT NO.:	



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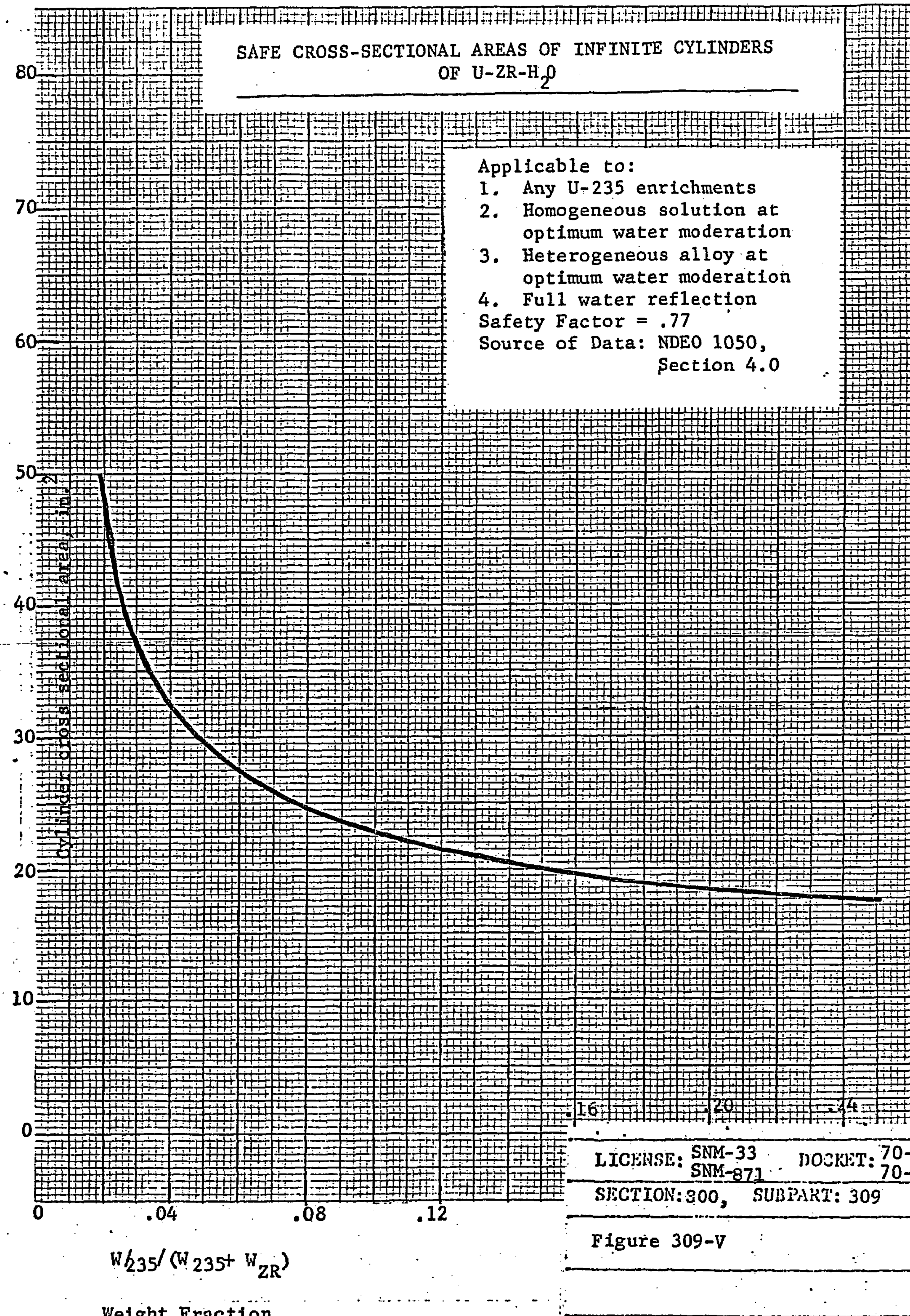
SECTION: 300, SUBPART: 309

Figure 309-IV

APPROVED:

ISSUED: 2/6/70

SUPERSEDES: 10/31/68



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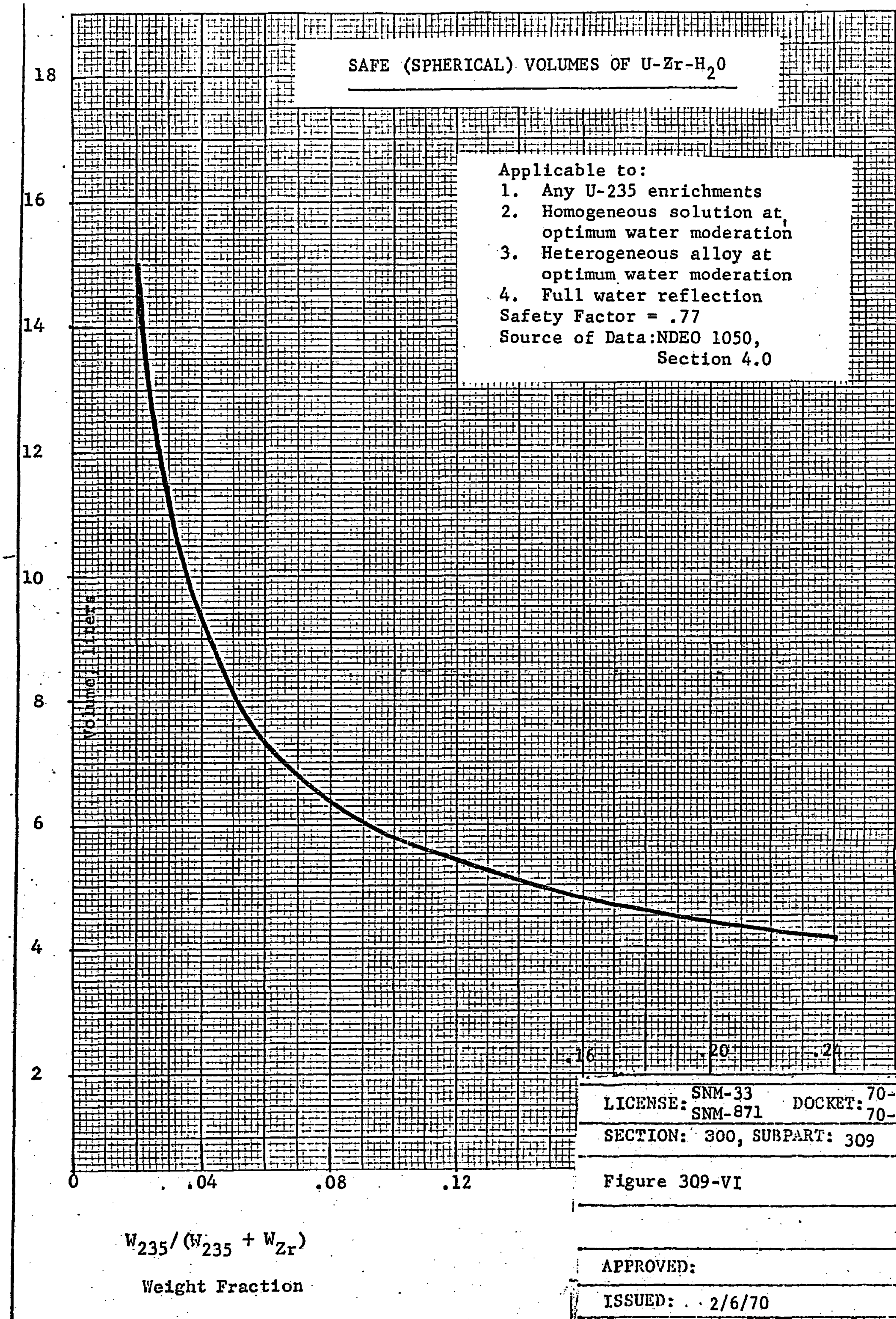
SECTION: 300,	SUBPART: 309

Figure 309-V

APPROVED:

ISSUED: 2/6/70

SUPERSEDES: 10/21/6[illegible]



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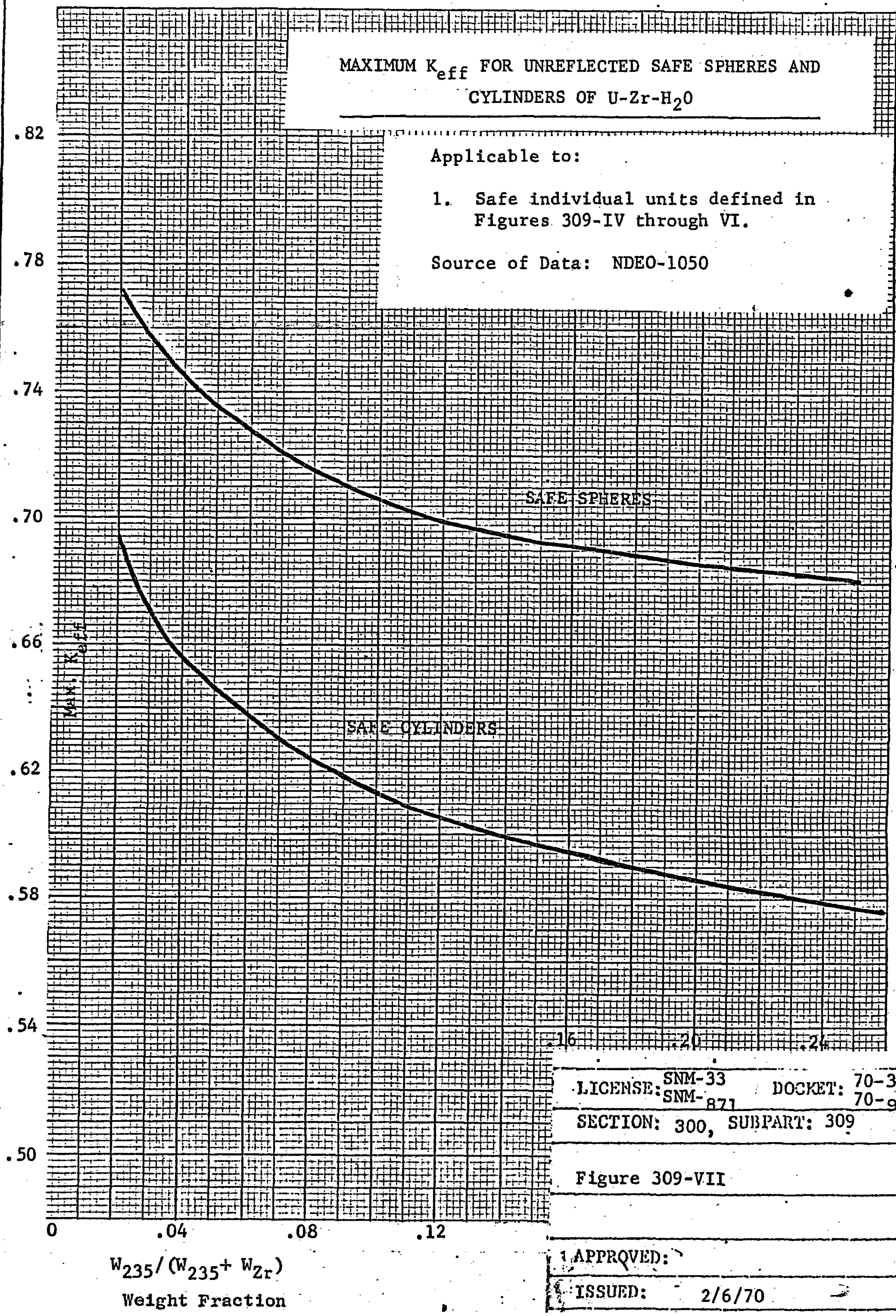
SECTION: 300, SUBPART: 309	
Figure 309-VI	
APPROVED:	
ISSUED: 2/6/70	
SUPERSEDES: 10/21/68	

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

APPROVED:

ISSUED: 2/6/70

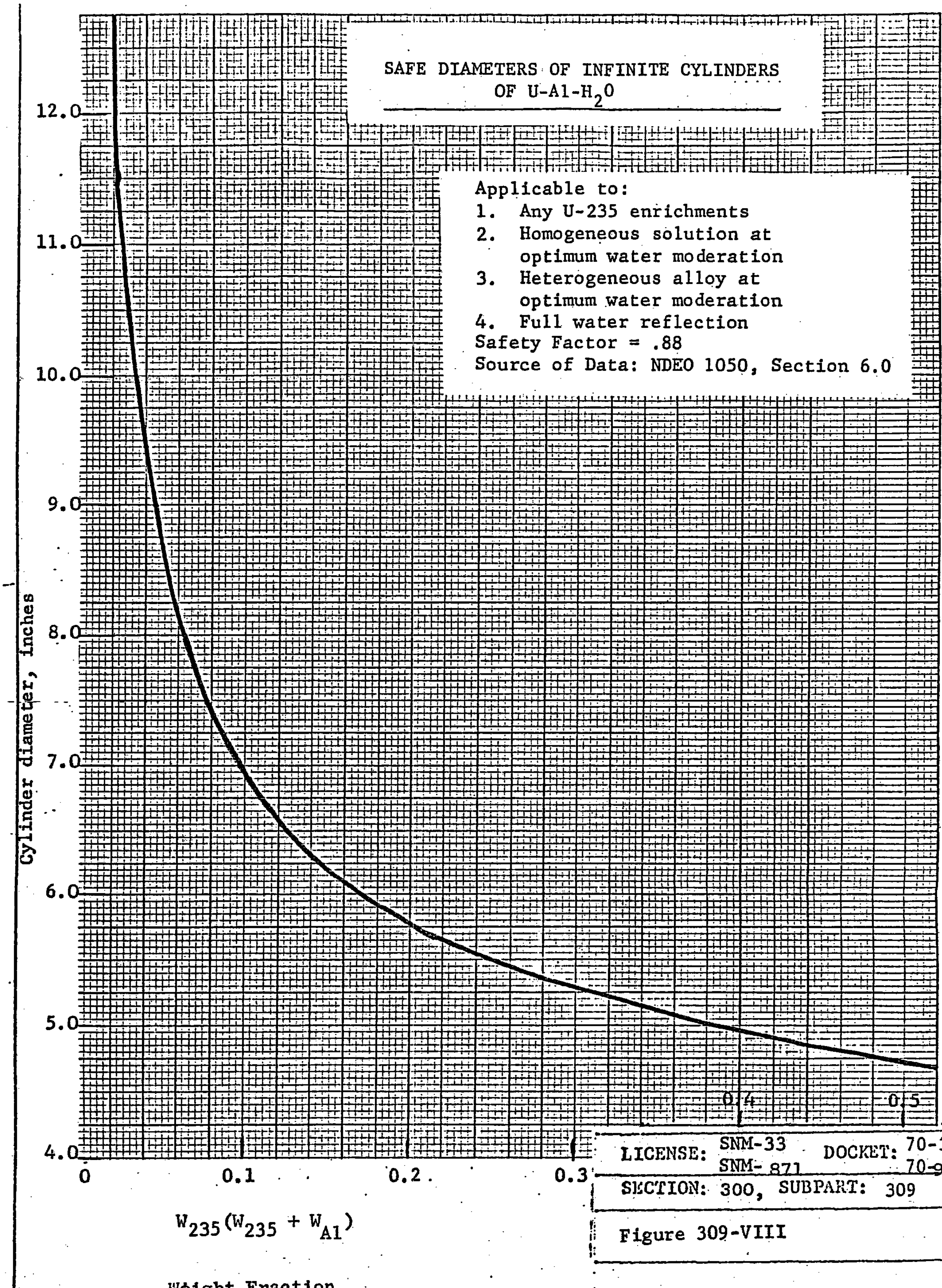

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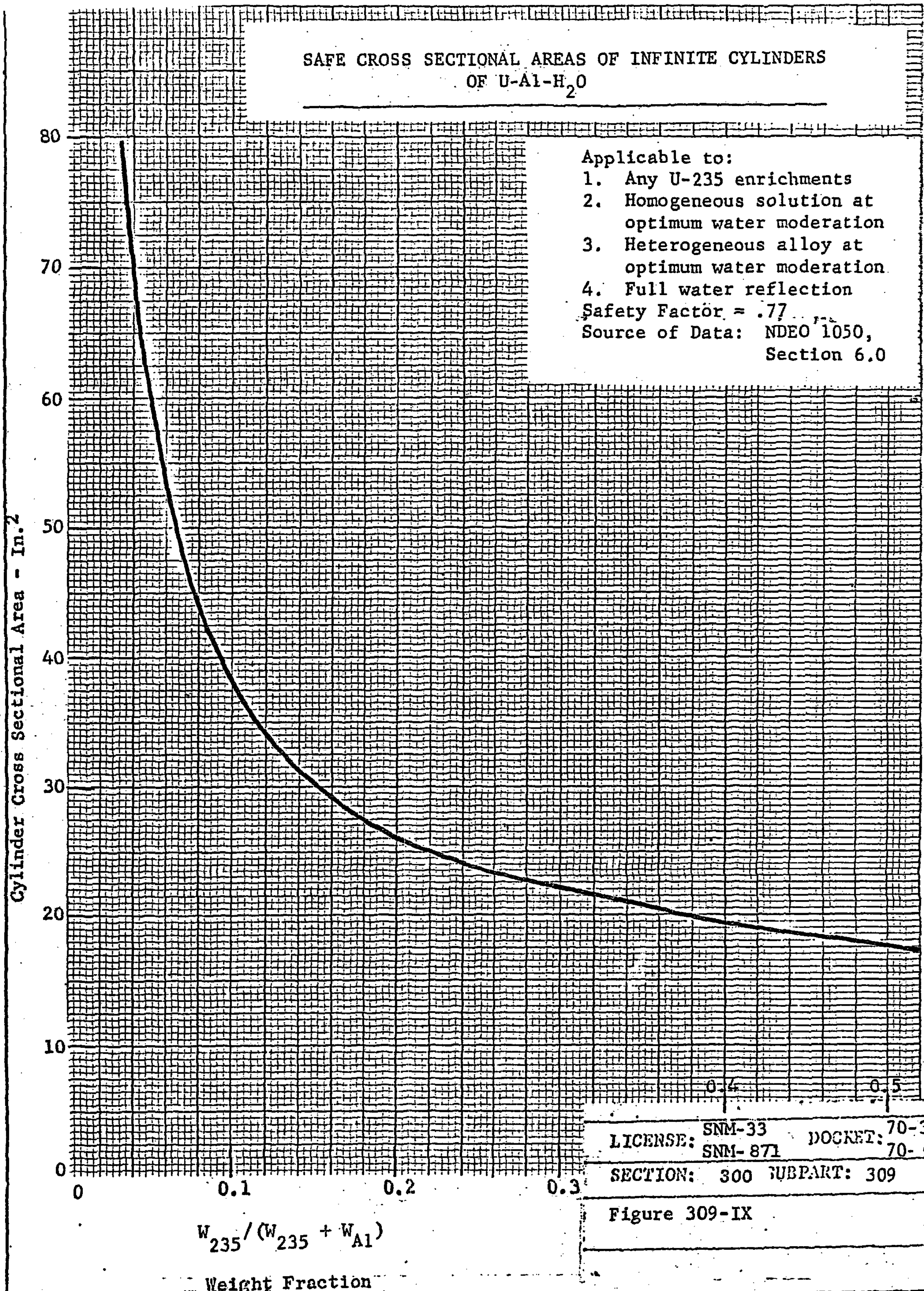
Figure 309-VIII

APPROVED:

ISSUED: 2-6-70



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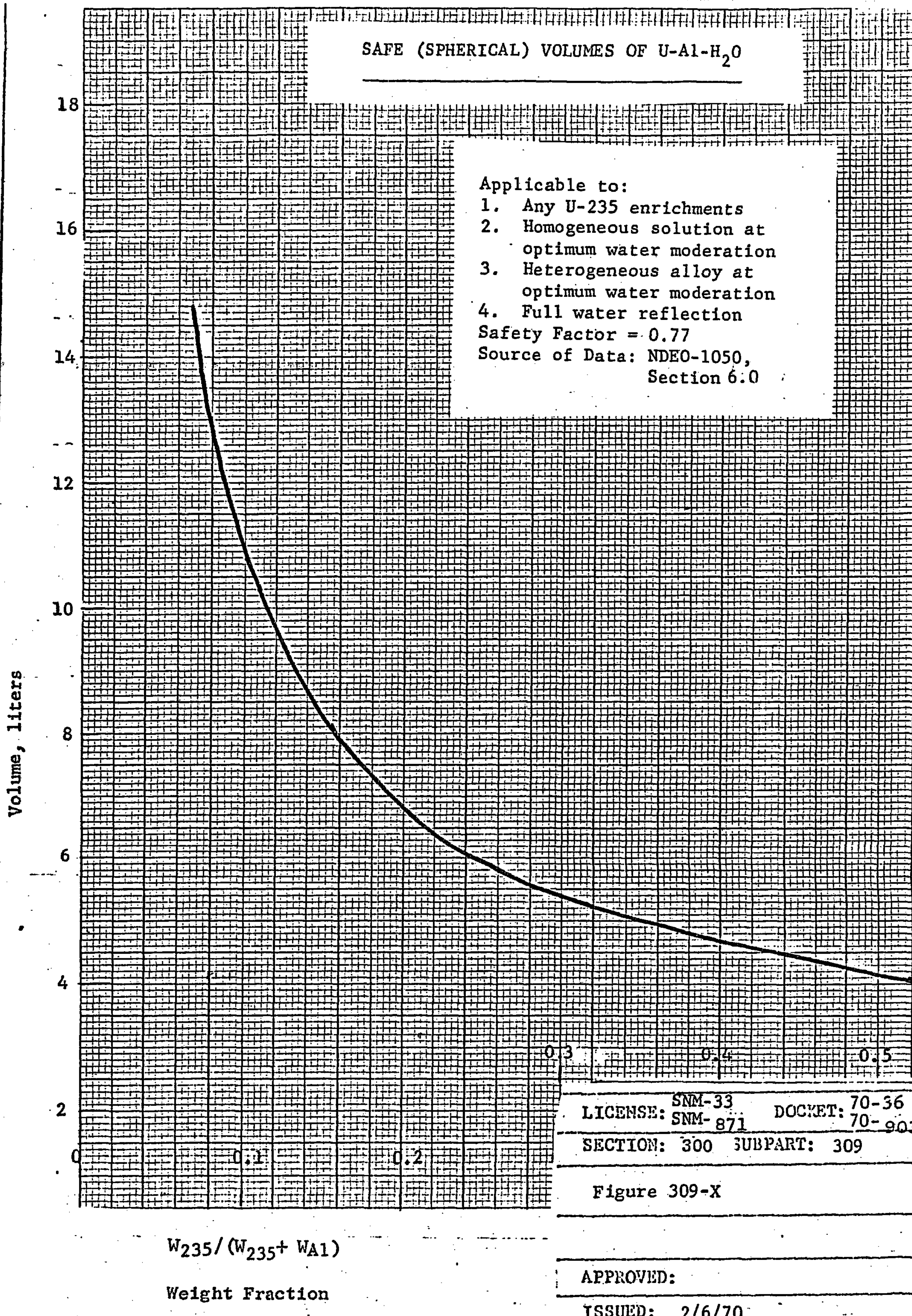
SECTION:	300	SUBPART:	309

Figure 309-IX

APPROVED:

ISSUED: 2/6/70

SUPERSEDES: 10/21/68



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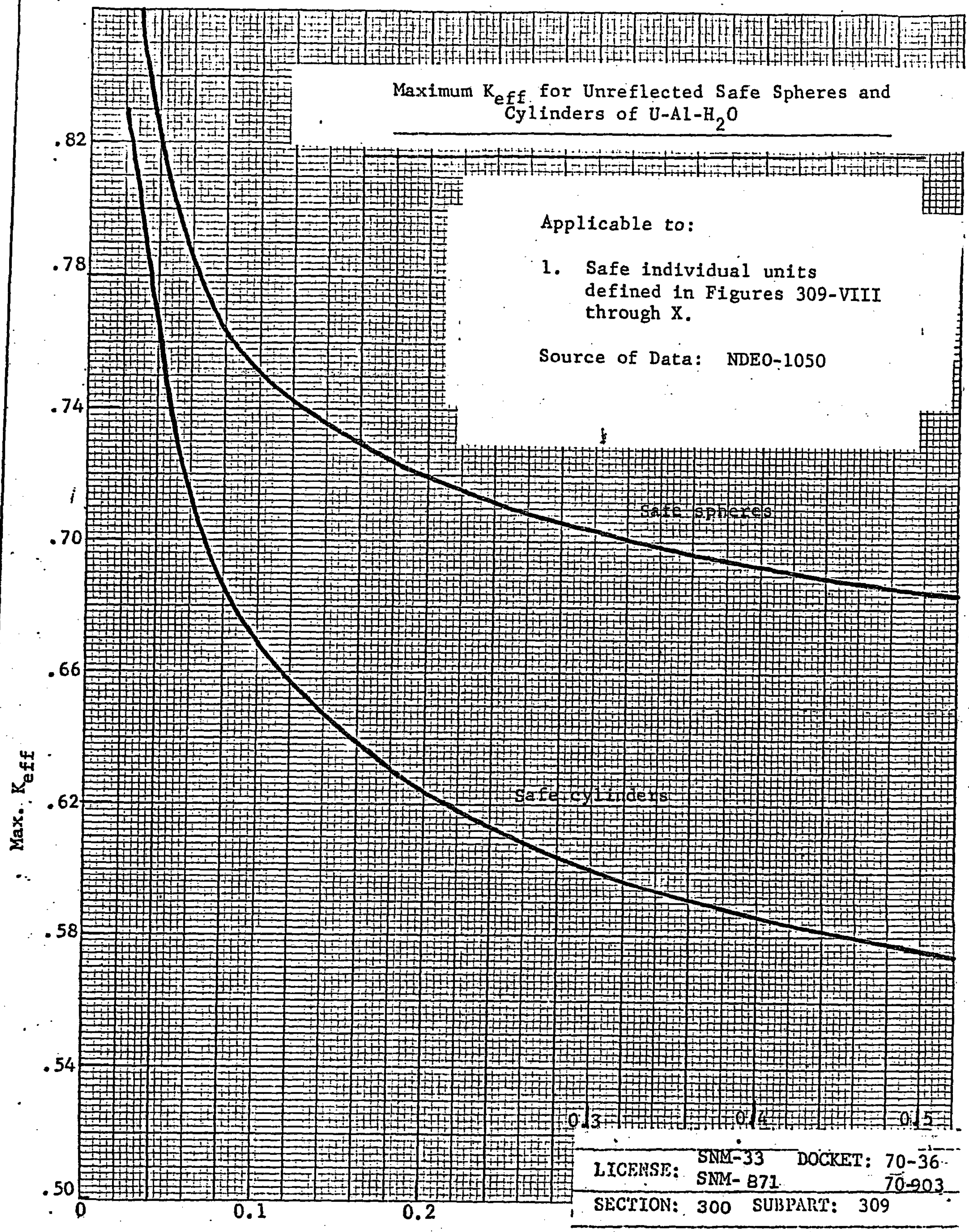
SECTION: 300	SUBPART: 309

Figure 309-X

APPROVED:

ISSUED: 2/6/70

SUPERSEDES: 10/31/68



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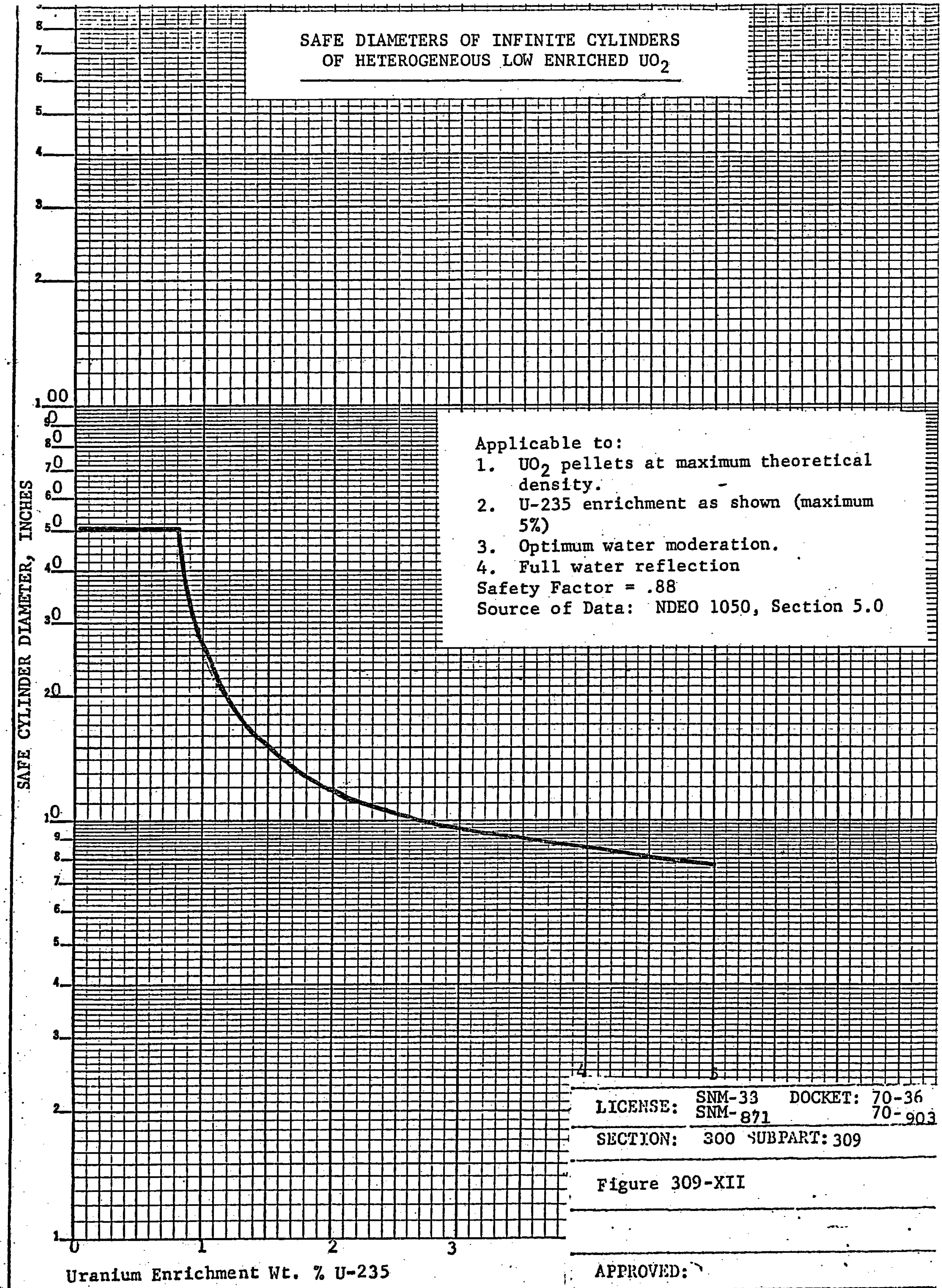
SECTION:	300	SUBPART:	309

Figure 309-XI

APPROVED:

ISSUED: 2/6/70

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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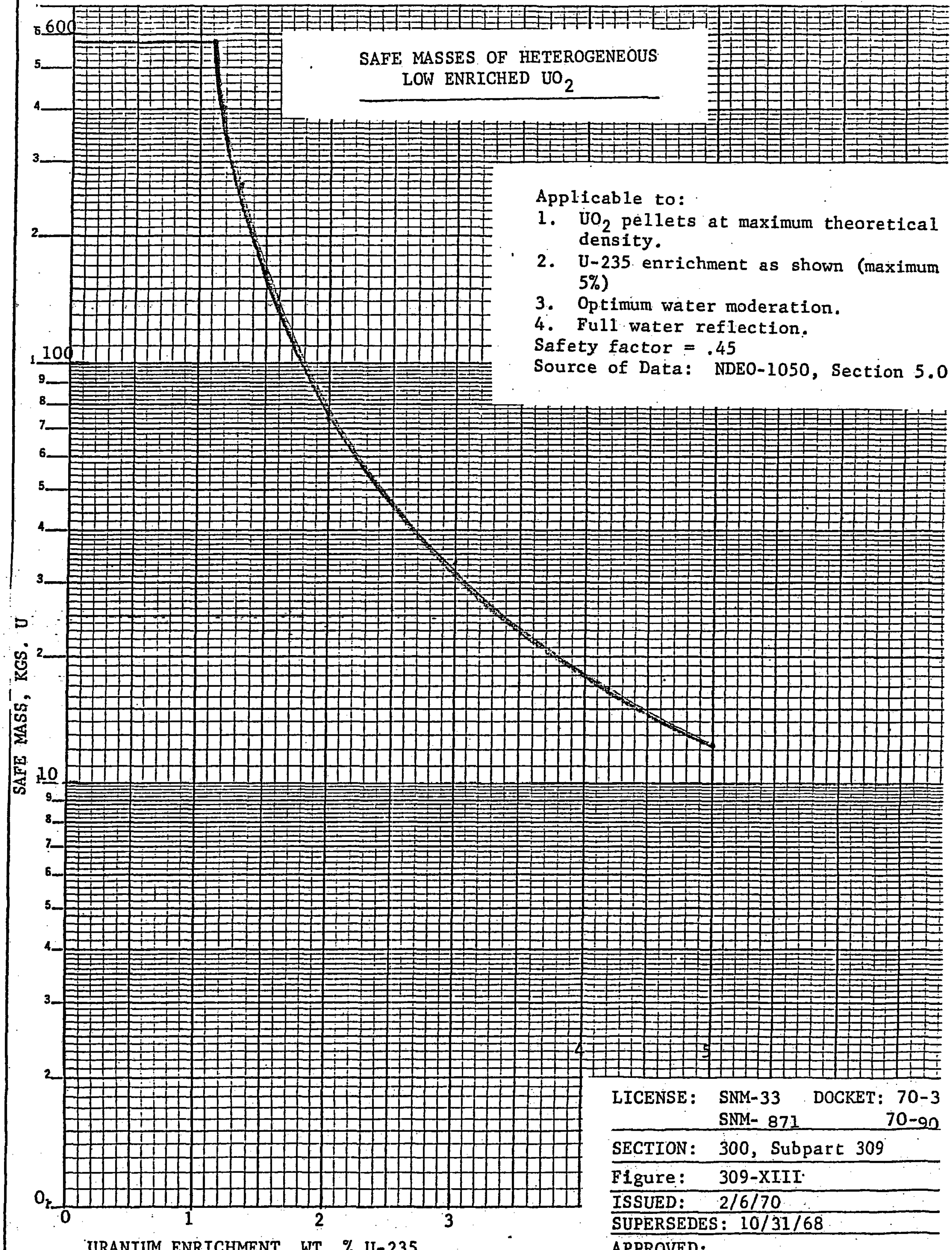
Figure 309-XII

APPROVED:

ISSUED: 2/6/70



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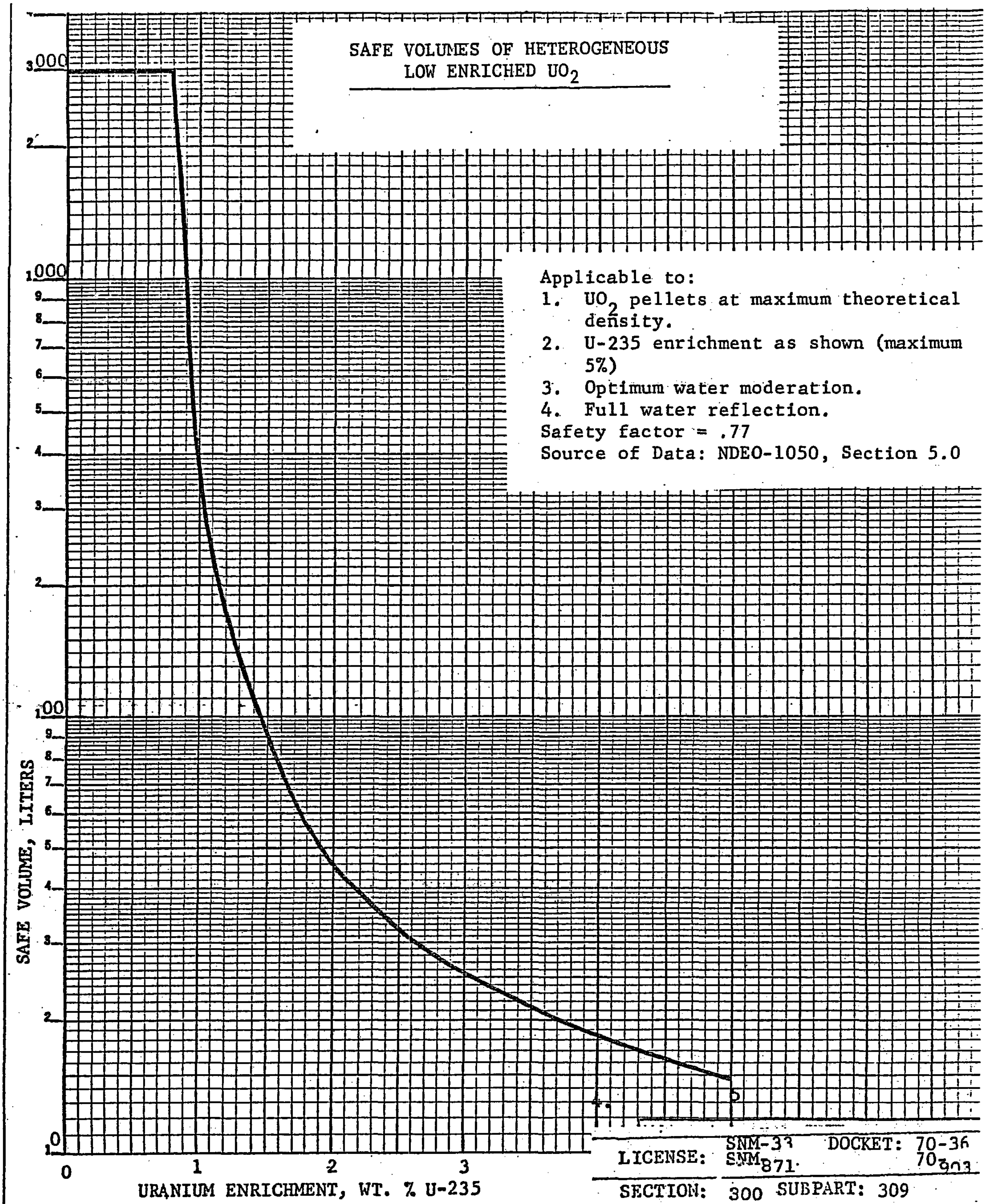
SECTION:	300, Subpart 309		
Figure:	309-XIII		
ISSUED:	2/6/70		
SUPERSEDES:	10/31/68		
APPROVED:			
Amendment No.:			



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SECTION: 300 SUBPART: 309

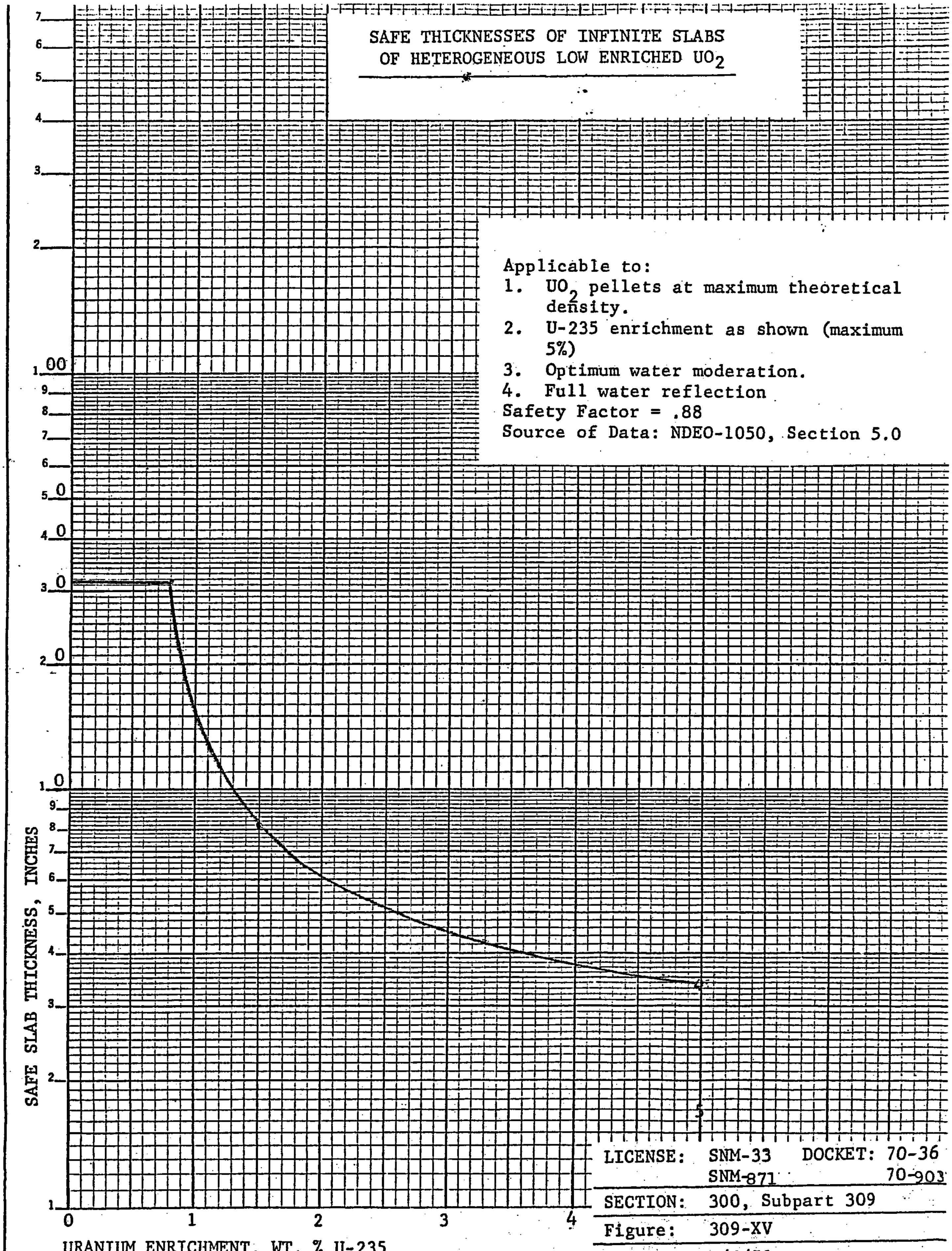
Figure 309-XIV

APPROVED:

ISSUED: 2/6/70

SUPERSEDES: 10/31/68

SAFE THICKNESSES OF INFINITE SLABS 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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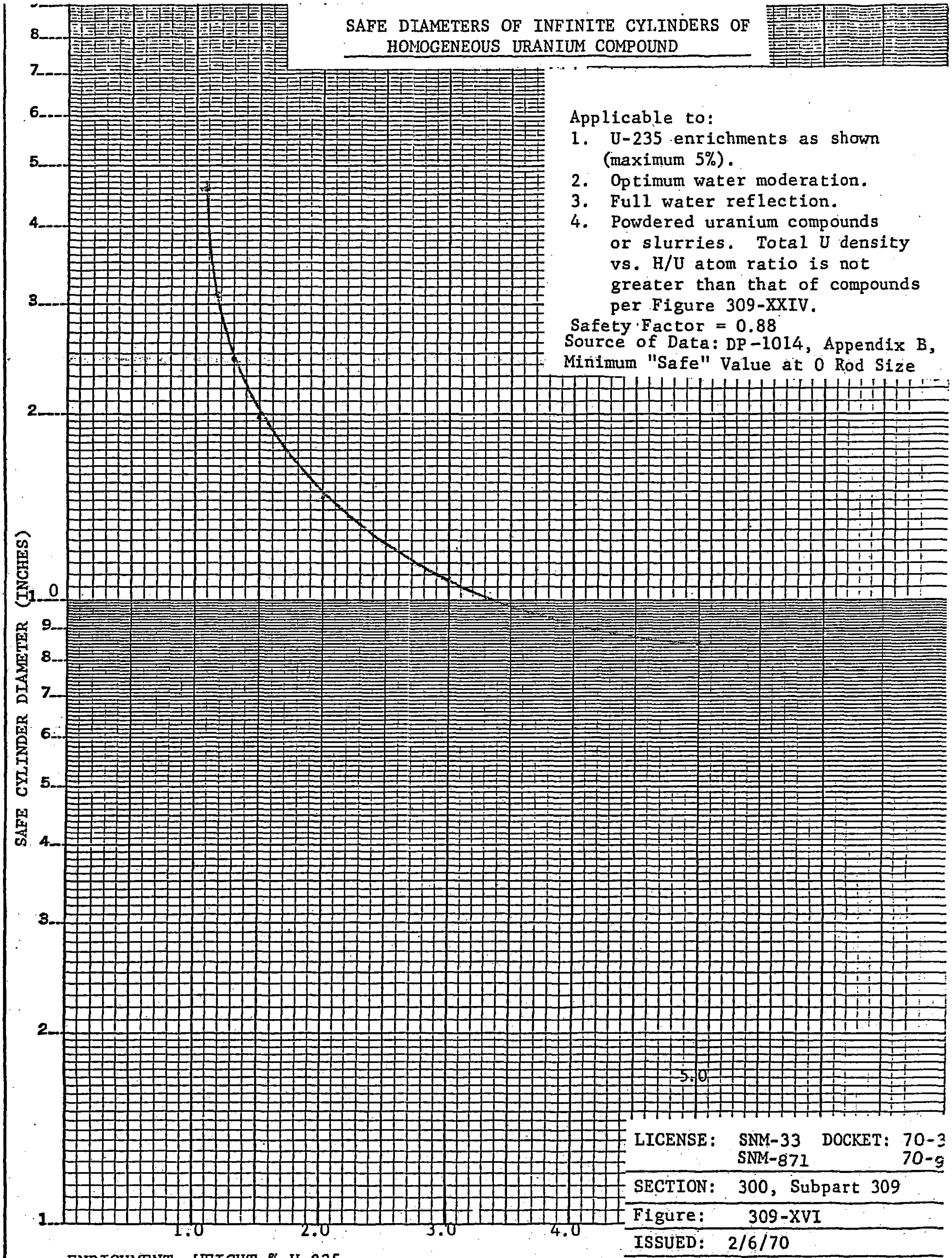
SECTION:	300, Subpart 309		
Figure:	309-XV		
ISSUED:	2/6/70		
SUPERSEDES:	10/31/68		
APPROVED:			
Amendment No.:			



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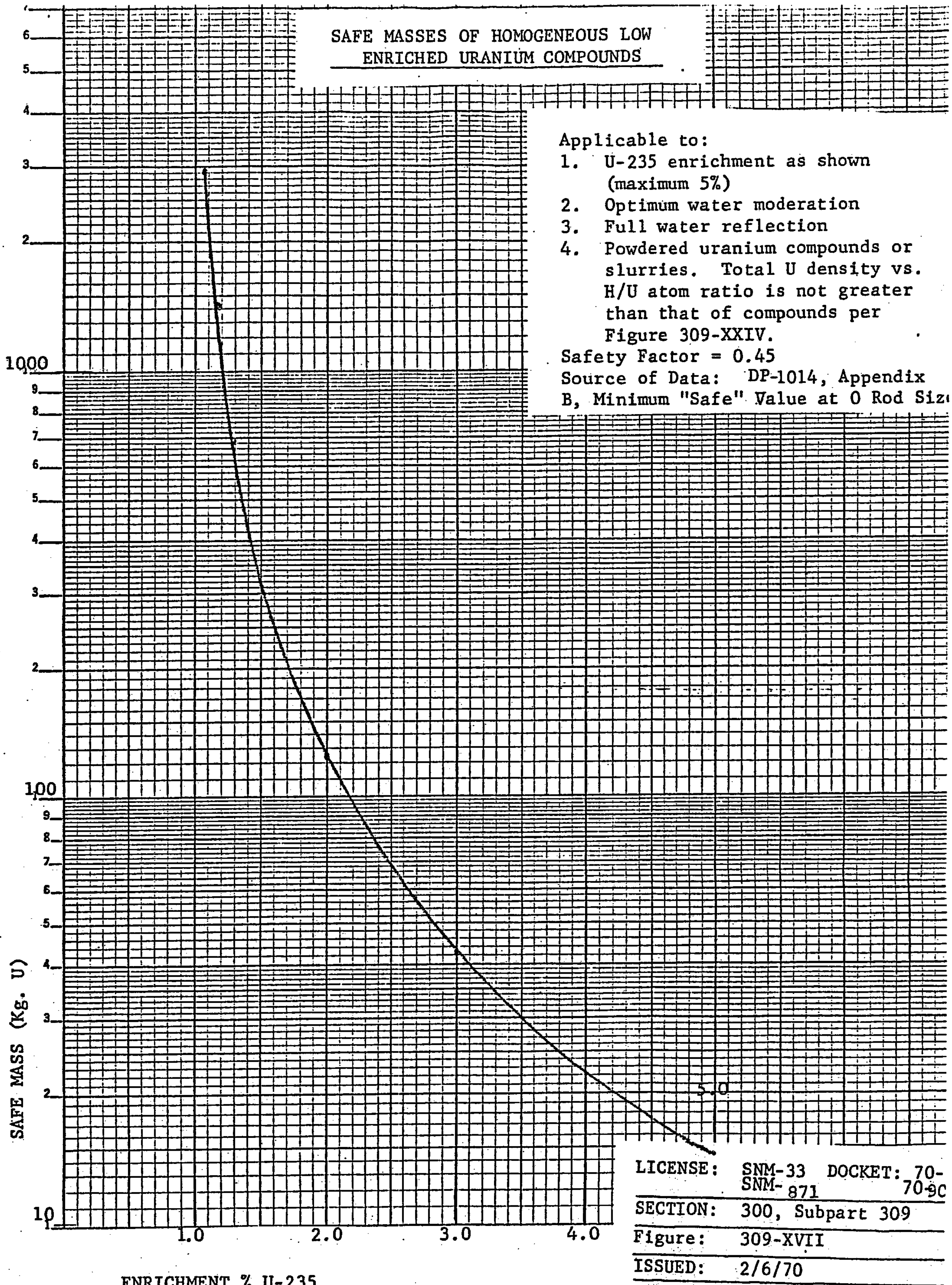
SECTION: 300, Subpart 309	
Figure: 309-XVI	
ISSUED: 2/6/70	
SUPERSEDES: 10/31/68	
APPROVED:	
Amendment No.:	



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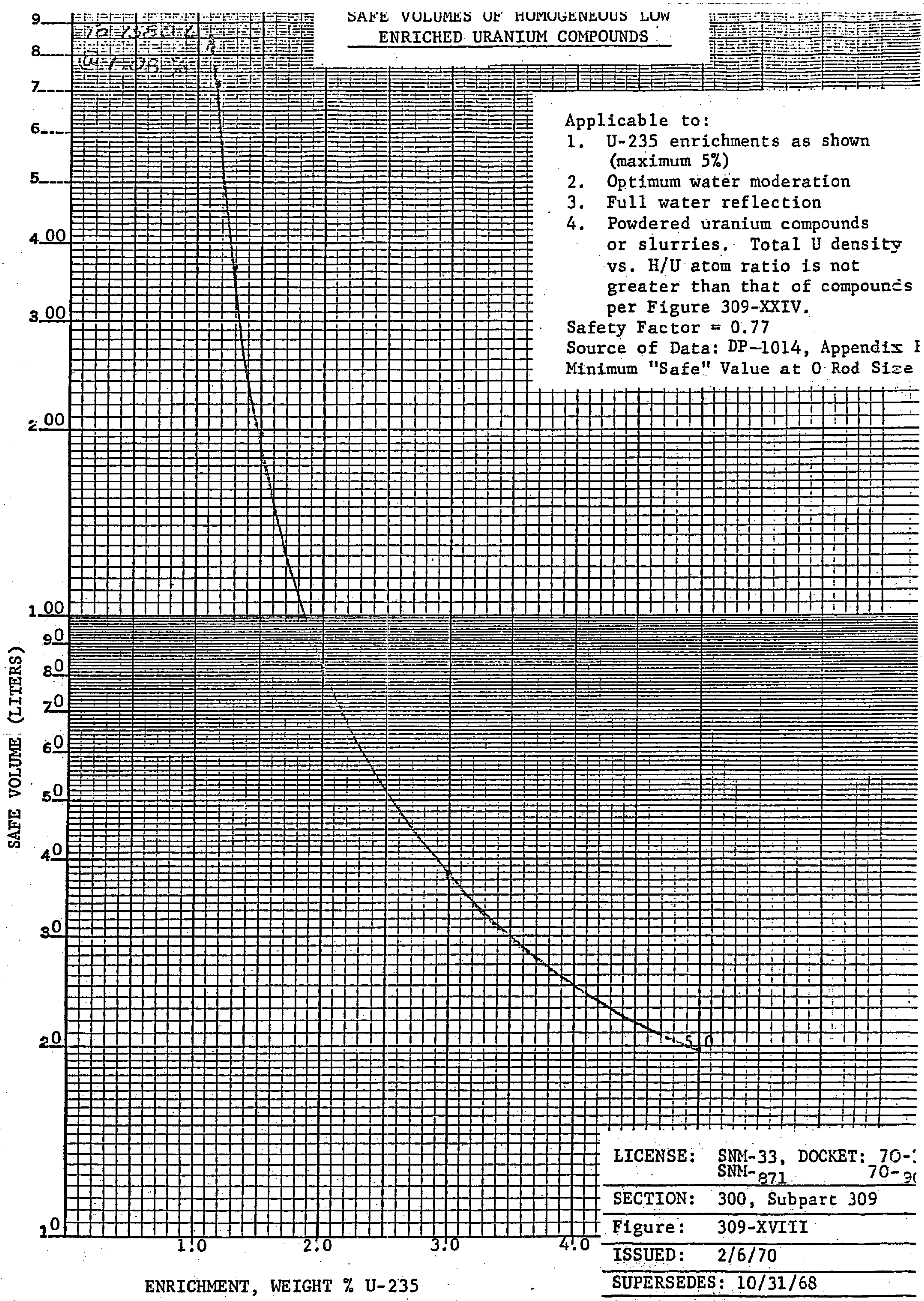
SECTION:	300, Subpart 309
Figure:	309-XVII
ISSUED:	2/6/70
SUPERSEDES:	10/31/68
APPROVED:	
Amendment No.:	



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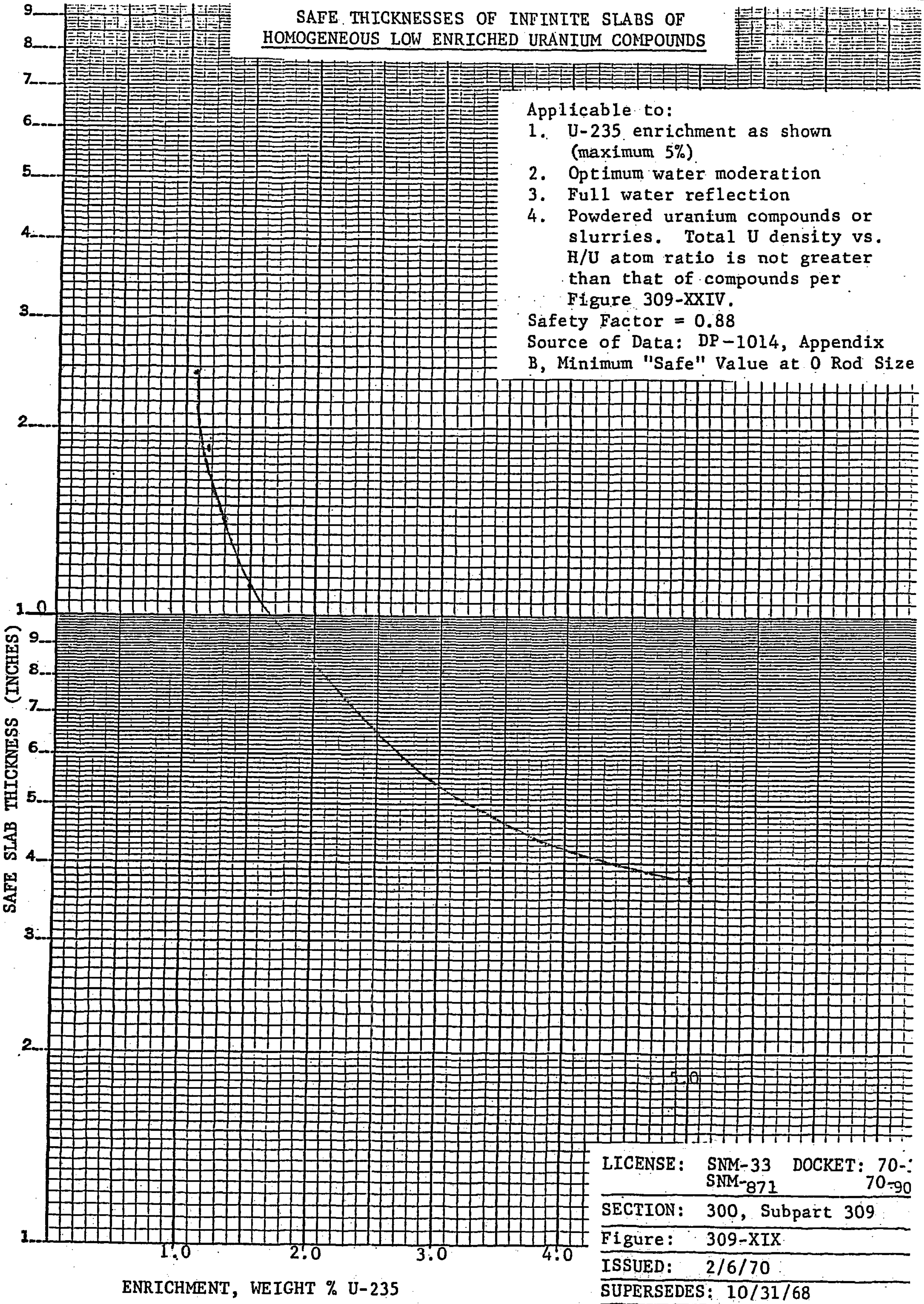
SECTION:	300, Subpart 309
Figure:	309-XVIII
ISSUED:	2/6/70
SUPERSEDES:	10/31/68
APPROVED:	

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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SECTION:	300, Subpart 309
Figure:	309-XIX
ISSUED:	2/6/70
SUPERSEDES:	10/31/68
APPROVED:	
Amendment No.:	

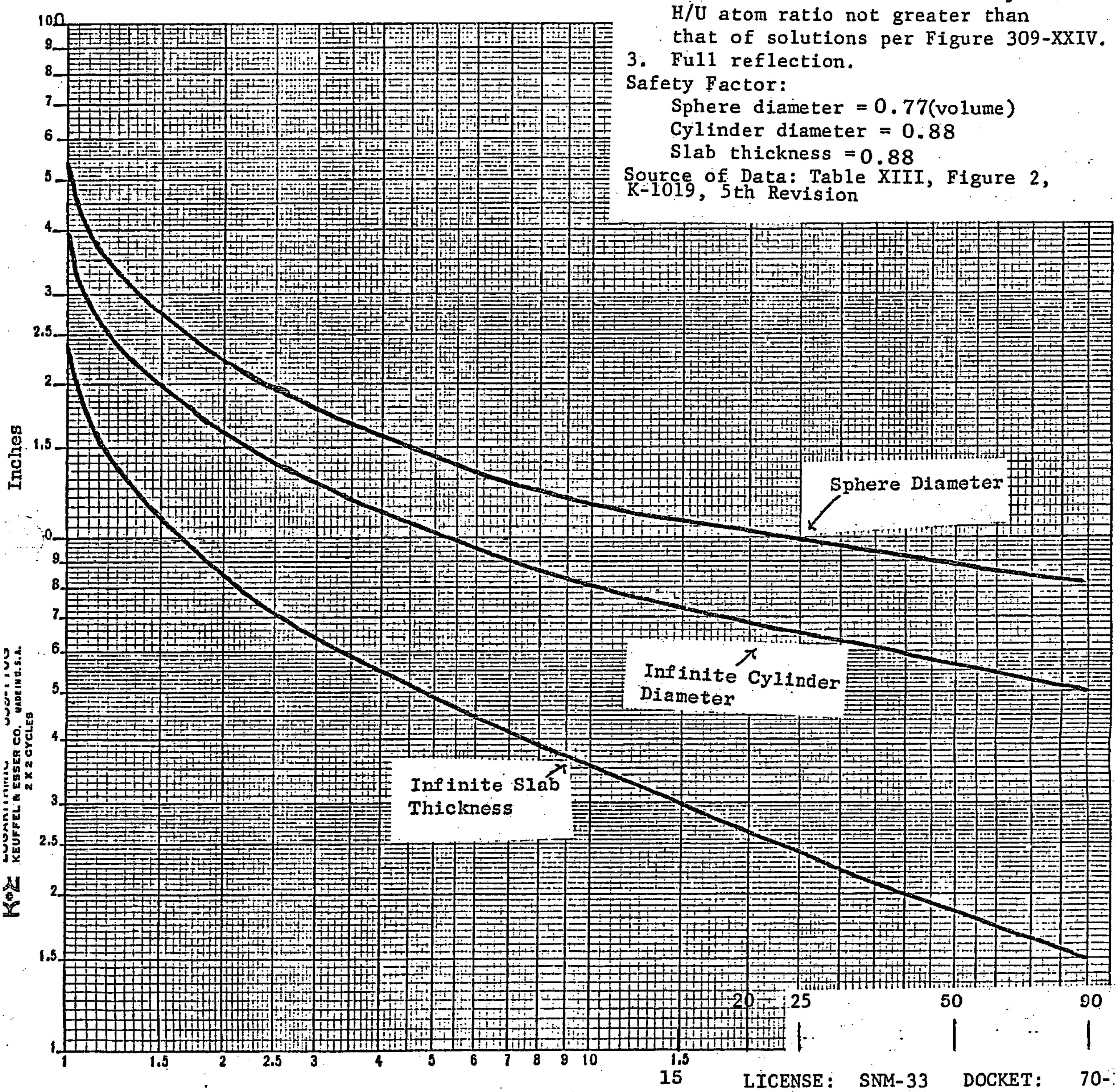
SAFE GEOMETRIC VARIABLES

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 = 0.77(volume)
Cylinder diameter = 0.88
Slab thickness = 0.88

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



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SNM-871 70-903

SECTION: 300, Subpart 309

Figure: 309-XX

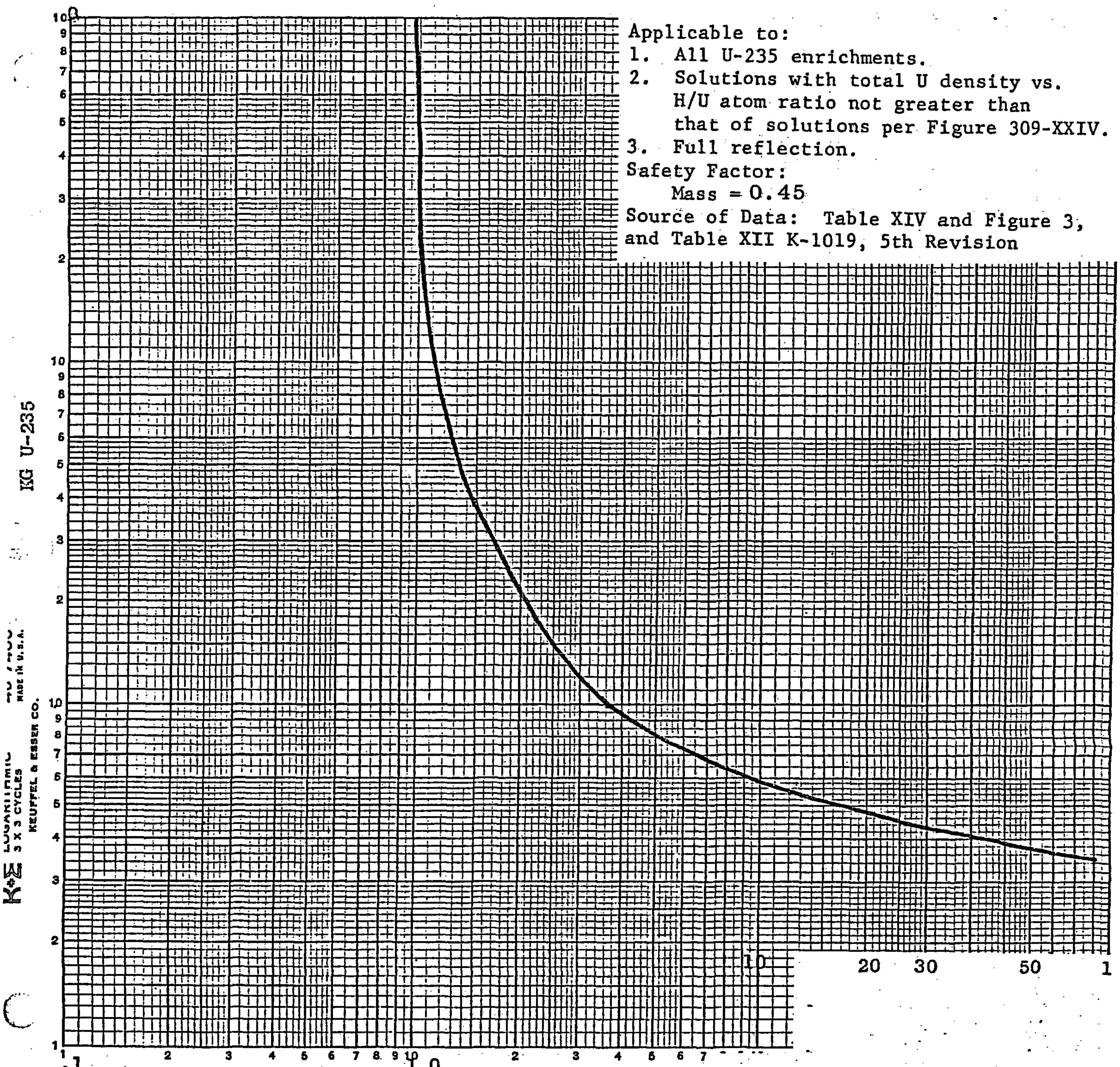
ISSUED: 4/21/72

SUPERSEDES: 3/19/71

APPROVED:

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 = 0.45

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

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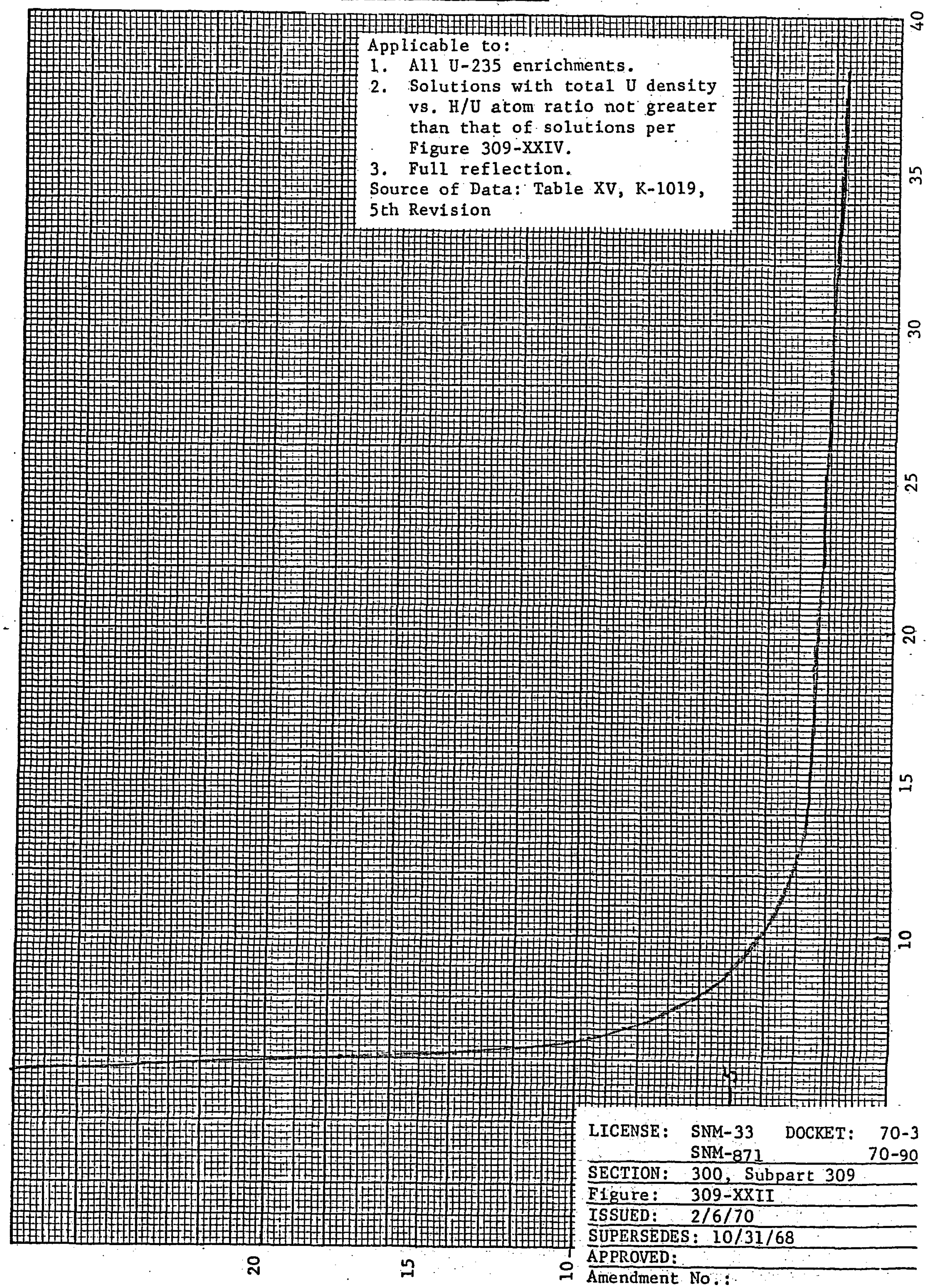
SECTION:	300, Subpart 309		
Figure:	309-XXI		
ISSUED:	4/21/72		
SUPERSEDES:	2/6/70		
APPROVED:			
Amendment No.:			

ANY U-235 ENRICHMENT



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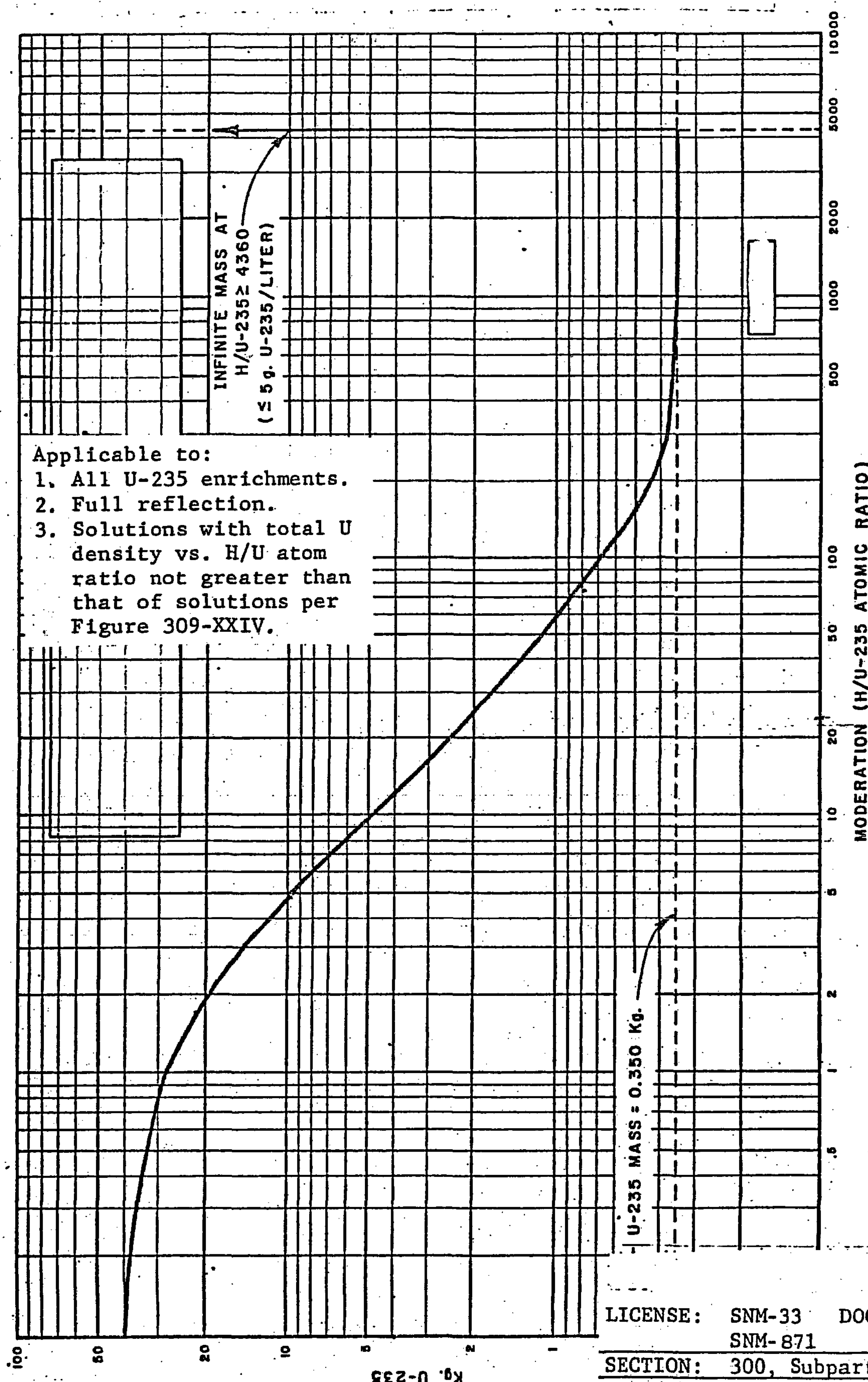
SECTION: 300, Subpart 309
Figure: 309-XXII
ISSUED: 2/6/70
SUPERSEDES: 10/31/68
APPROVED:
Amendment No.:

NUCLEARLY SAFE MASSES FOR SPECIFIED MODERATION



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SECTION:	300, Subpart 309		
Figure:	309-XXIII		
ISSUED:	2/6/70		
SUPERSEDES:	10/31/68		
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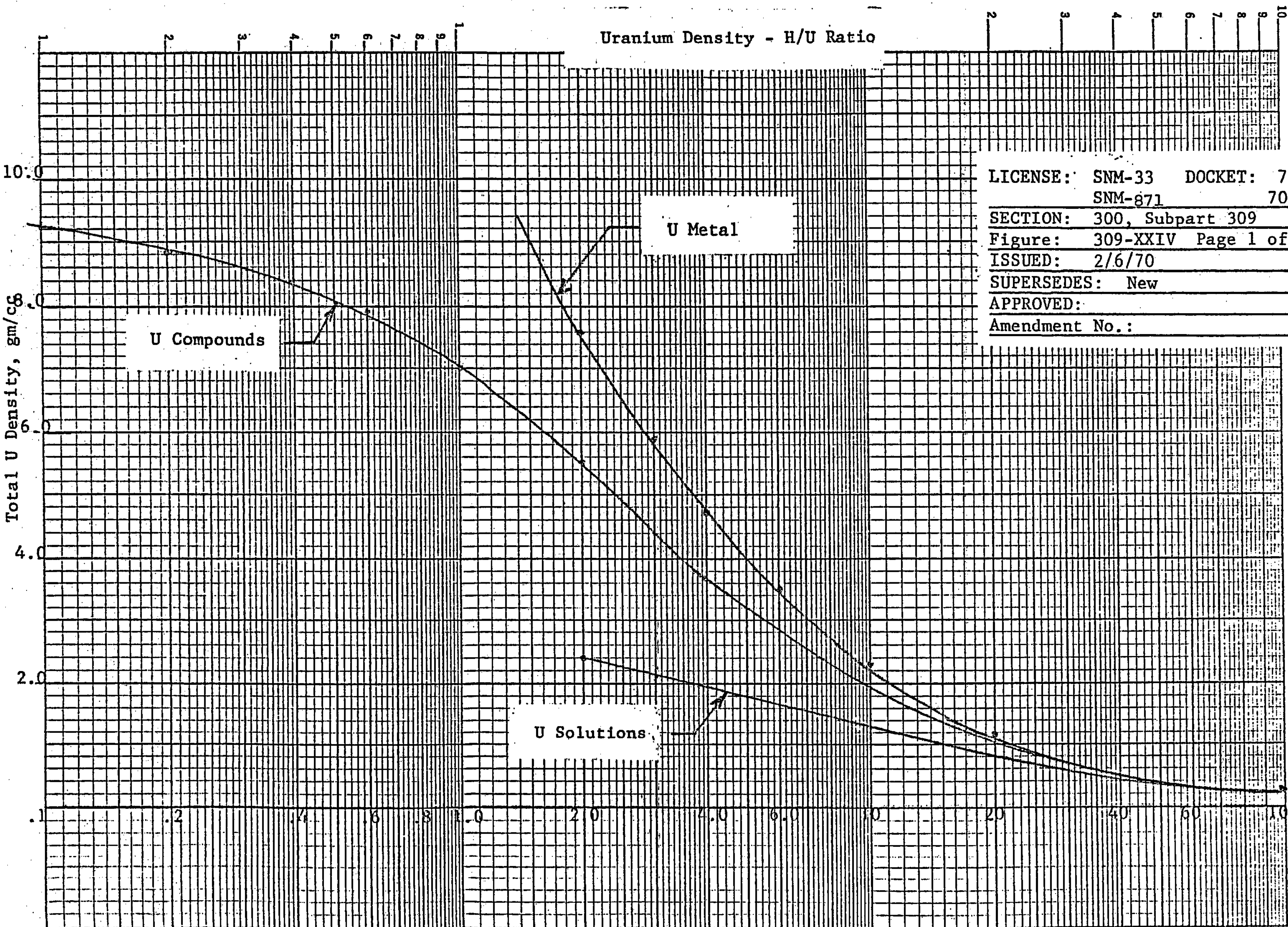
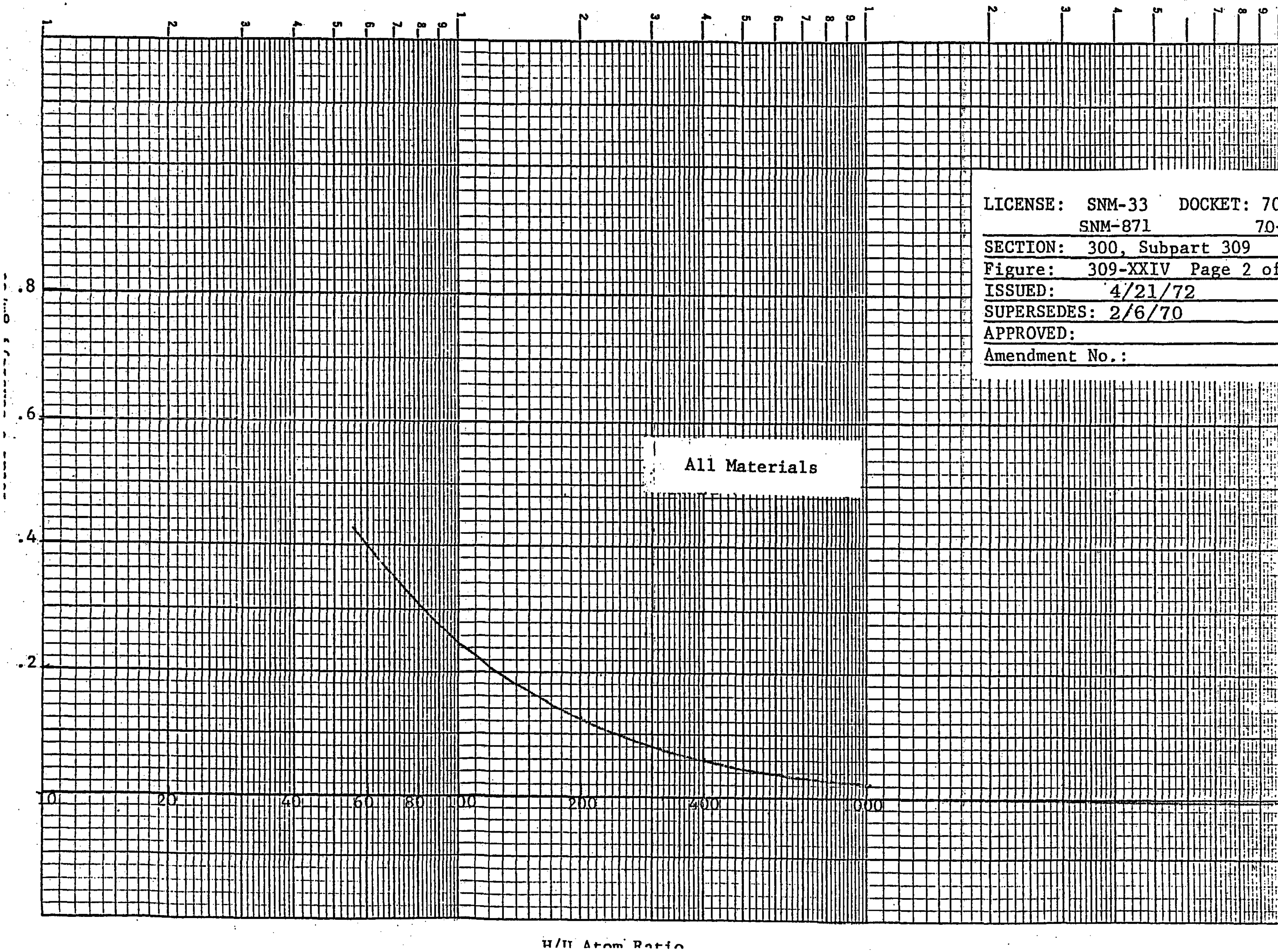
SECTION:	300, Subpart 309		
Figure:	309-XXIV Page 1 of 2		
ISSUED:	2/6/70		
SUPERSEDES:	New		
APPROVED:			
Amendment No.:			



LICENSE:	SNM-33	DOCKET:	70-36
	SNM-871		70-903
SECTION:	300, Subpart 309		
Figure:	309-XXIV	Page 2 of 2	
ISSUED:	4/21/72		
SUPERSEDES:	2/6/70		
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.

LICENSE:	SNM-777 Docket 70-820 SNM-871 Docket 70-903
SECTION:	300, Figure 309-XXIV
Nuclear Safety Evaluation, Density vs. H/U Ratio	
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SUPERSEDES:	New

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. Uranium Solutions

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:

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

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

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

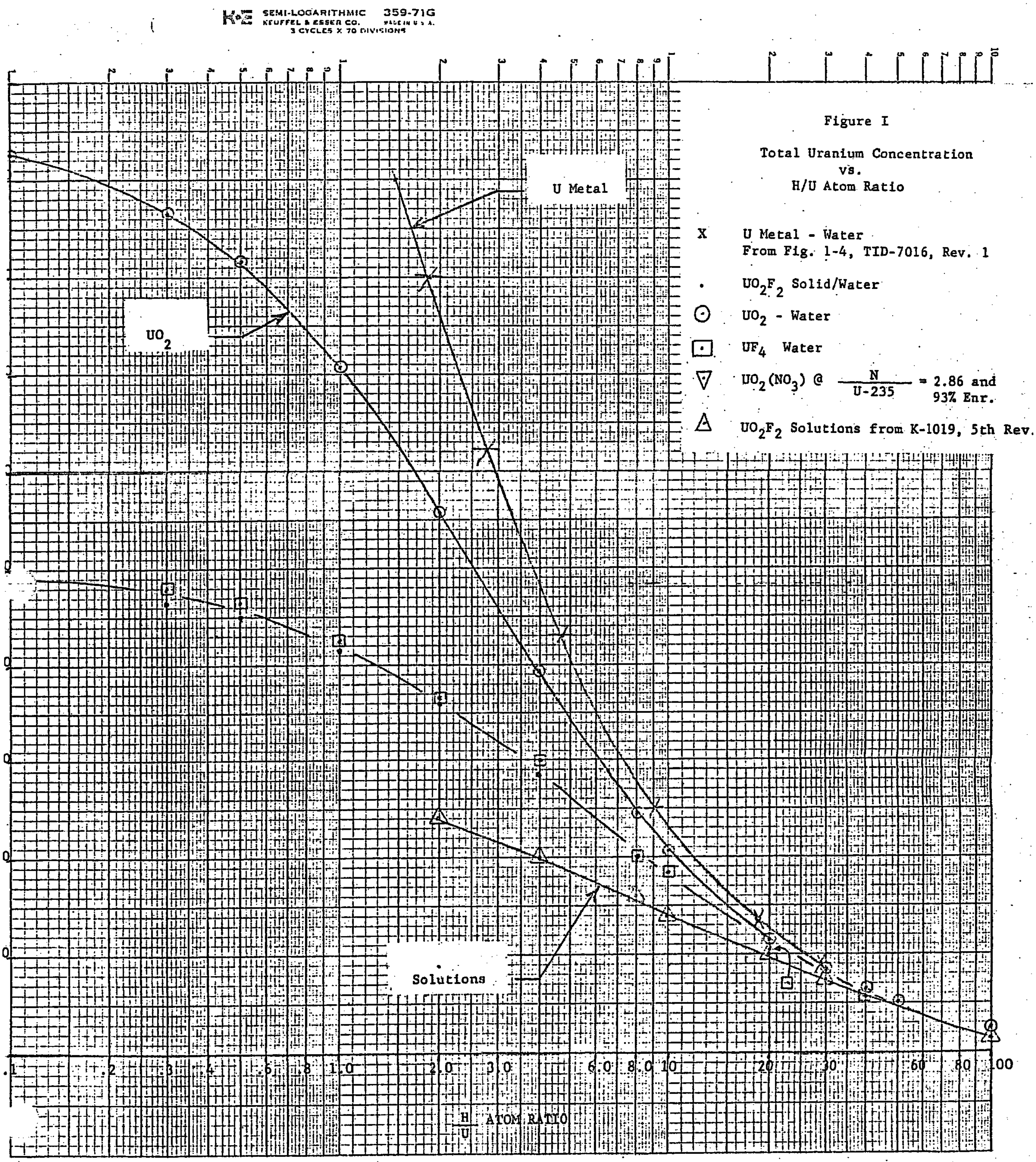
The data for solutions has been obtained from Figure 1, K-1019, 5th Revision.

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 55 to 60 and a density of approximately 0.42. This single curve follows that of the solution curve for H/U ratios in excess of 56. Unless specifically stated otherwise, safe parameters for all systems with H/U ratios greater than 56 will therefore follow those of solutions.

TABLE 1

$\frac{H}{U}$	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. MADE IN U.S.A.
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 Rev.
U Metal
UO_2
Solutions
.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

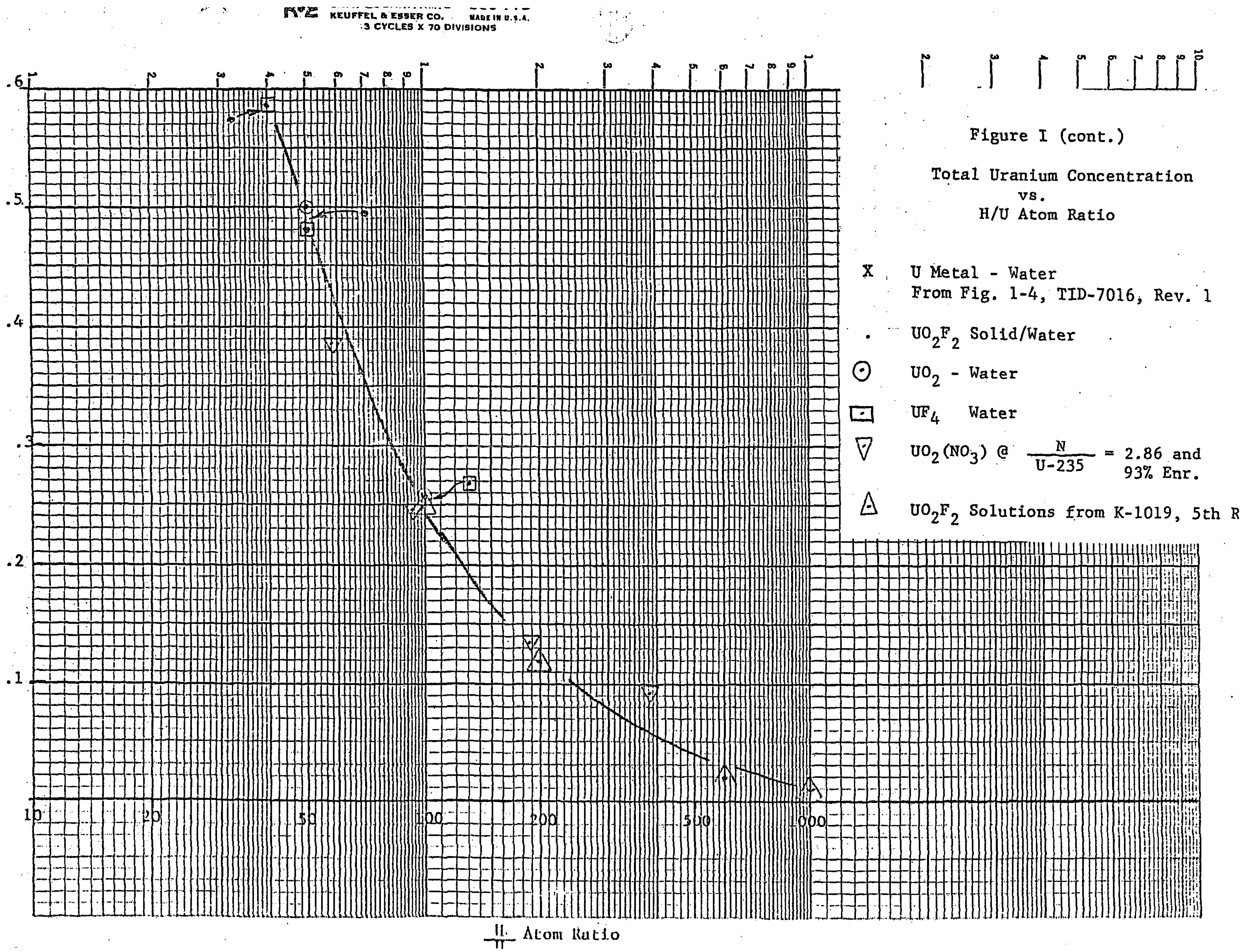
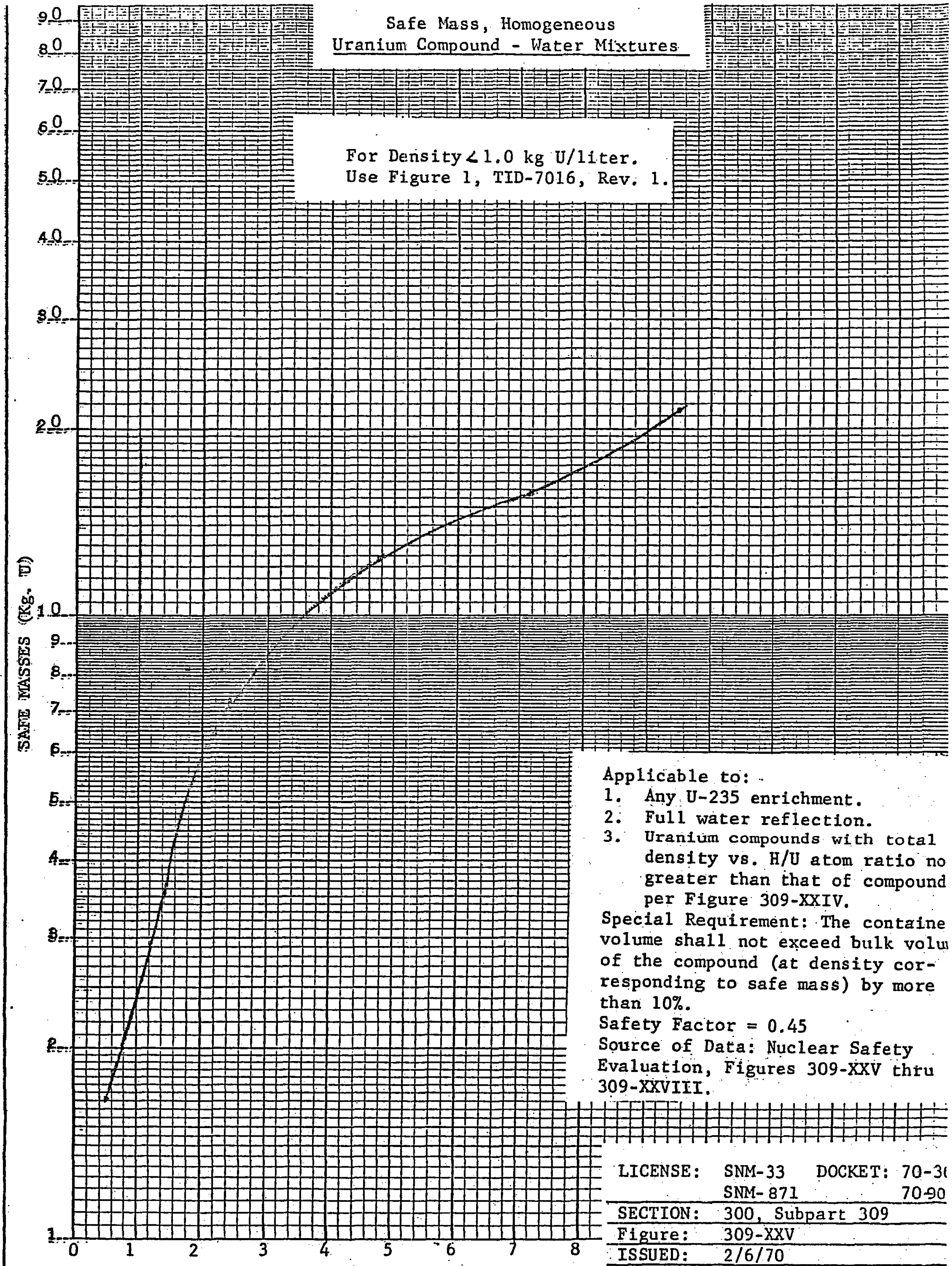


Figure I (cont.)

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.



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 SNM-871	DOCKET:	70-3(70-90
SECTION:	300, Subpart 309		
Figure:	309-XXV		
ISSUED:	2/6/70		
SUPERSEDES:	10/31/68		
APPROVED:			
Amendment No.:			

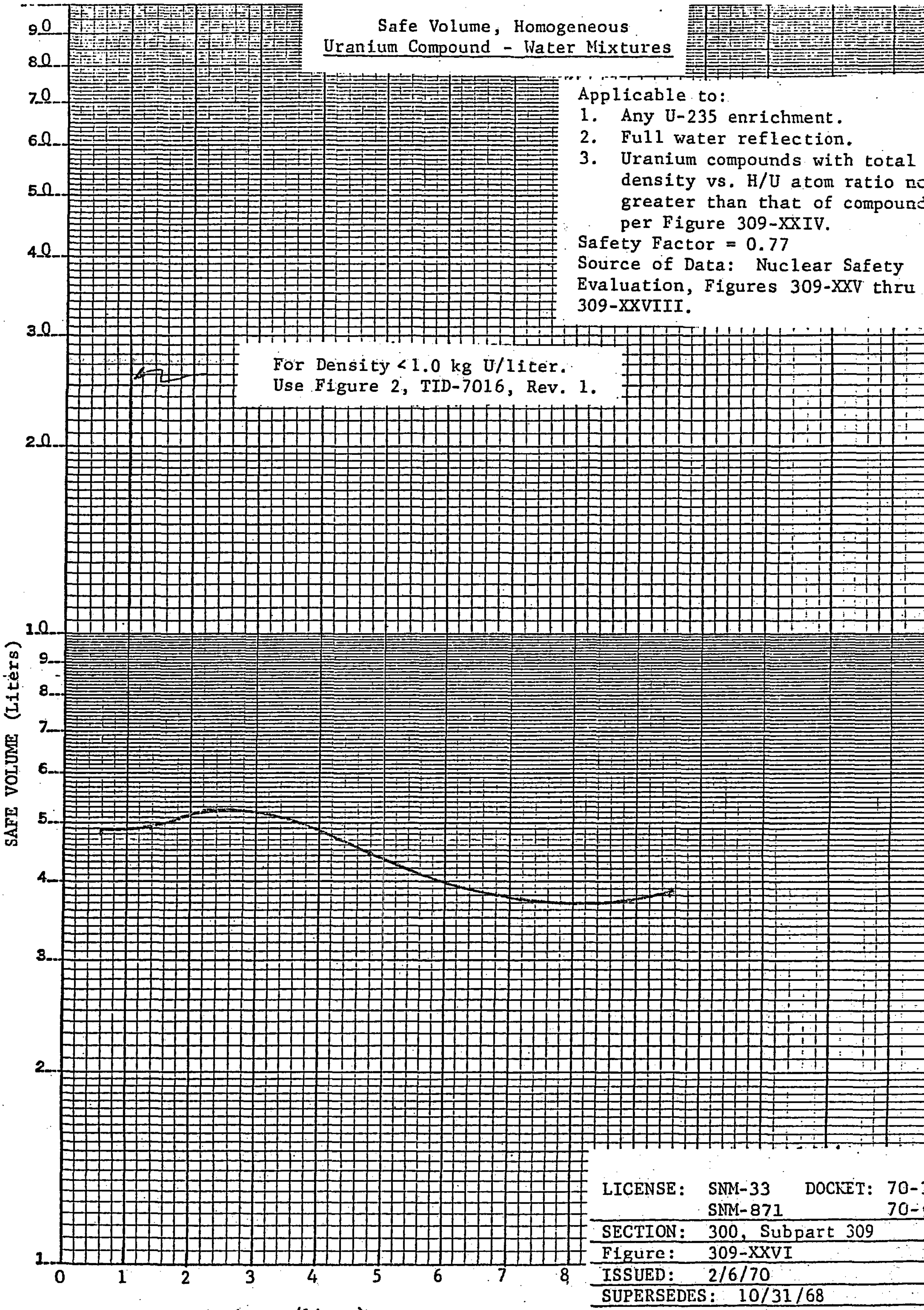
Safe Volume, Homogeneous Uranium Compound - Water Mixtures

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.

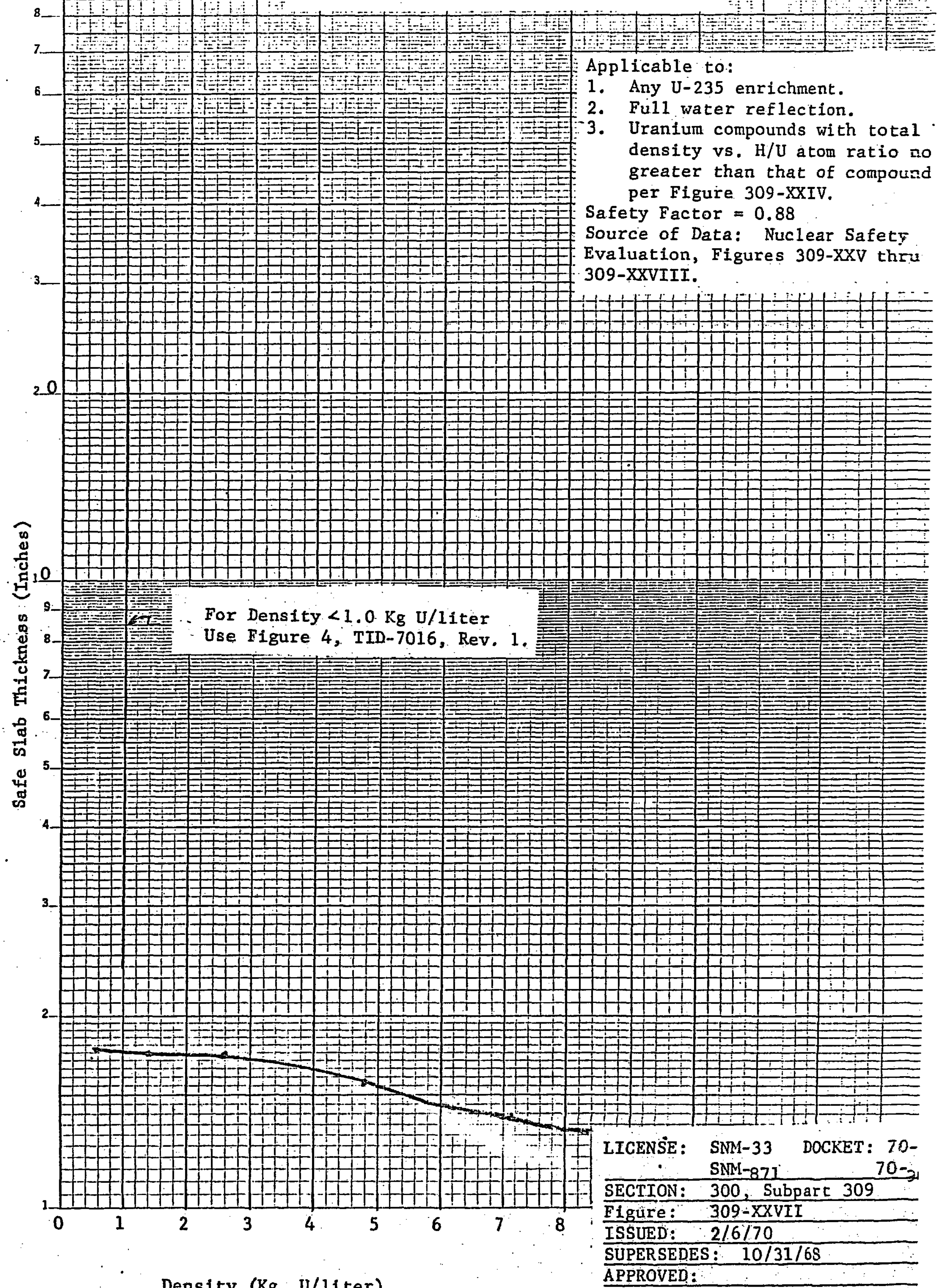
Safety Factor = 0.77

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



LICENSE:	SNM-33 DOCKET: 70-3
	SNM-871 70-9
SECTION:	300, Subpart 309
Figure:	309-XXVI
ISSUED:	2/6/70
SUPERSEDES:	10/31/68
APPROVED:	
Amendment No.:	

Uranium Compound - Water Mixtures



LICENSE:	SNM-33 DOCKET: 70-
	SNM-871 70-
SECTION:	300, Subpart 309
Figure:	309-XXVII
ISSUED:	2/6/70
SUPERSEDES:	10/31/68
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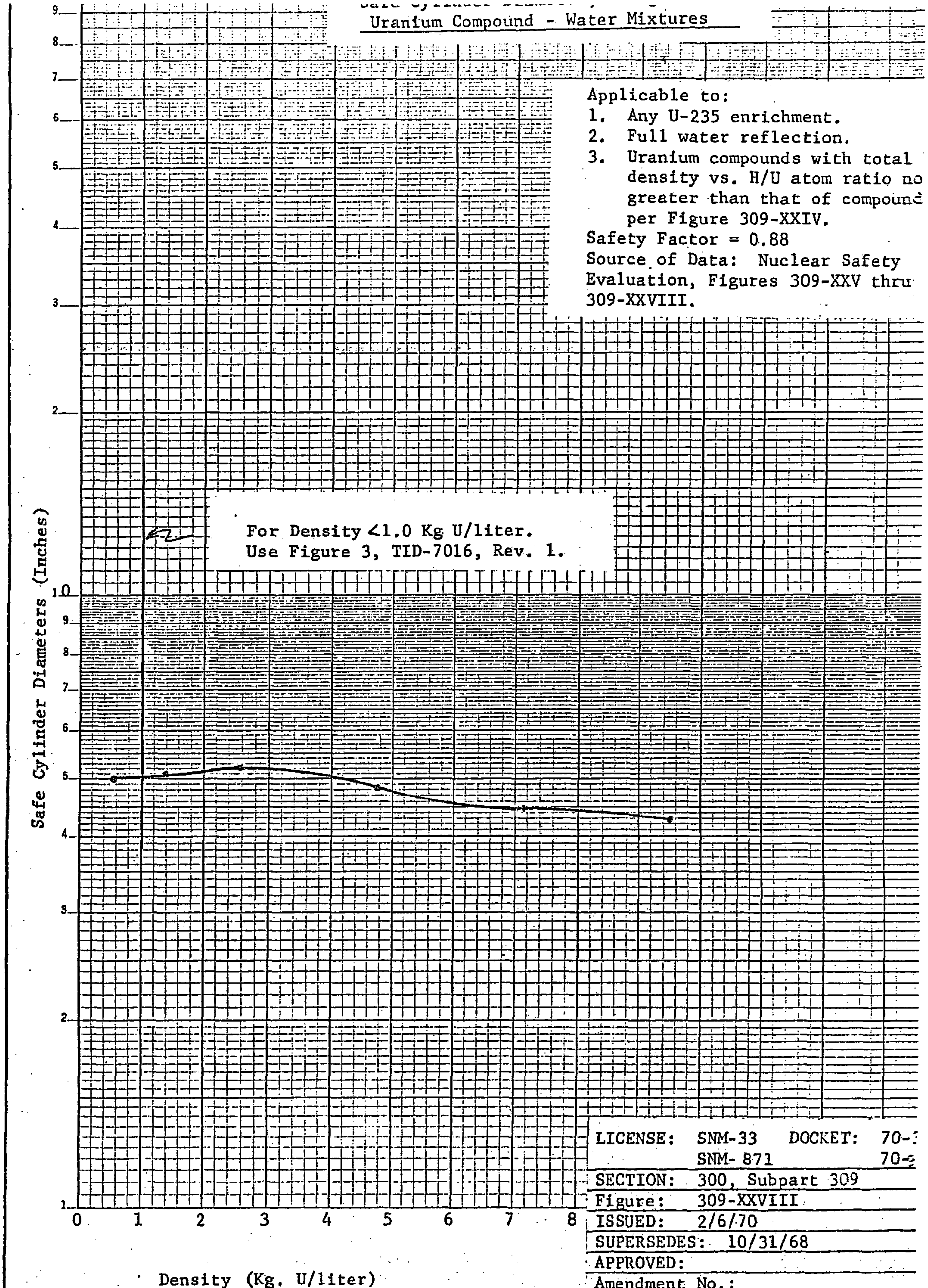
Amendment No.:	

Uranium Compound - Water Mixtures



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	SNM-871		70-[illegible]
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 (3)

	Density		Cylinder				Slab			
H/U-235	U-235 (Kg./L)	Total U Kg. U/L	$R_{c,r}$ (cm)	$D_{s,r}$ (1) (in.)	$R_{c,b}$ (cm)	$D_{s,b}$ (1) (in.)	$T_{c,r}$ (cm)	$T_{s,r}$ (2) (in.)	$T_{c,b}$ (cm)	$T_{s,b}$ (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: (1) $D_s = \dfrac{2\ R_c\,(\text{cm})}{2.54\ (\text{cm/in.}) \times 1.13\ \text{s.f.}} = .693\ R_c$, where R_c values are obtained from Table V-K, LA-3612

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

(3) Safe bare values are to be used in safety analyses of interacting units. Plant conditions and vessel walls may cause sufficient reflection of neutrons so that even isolated units may have the equivalent of somewhat less than a 1-inch water reflector.

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

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

	Density		Sphere				Mass			
H/U-235	U-235 (Kg./L)	Total U Kg U/L	$V_{c,r}$ (1) (Liters)	$V_{s,r}$ (2) (Liters)	$V_{c,b}$ (1) (Liters)	$V_{s,b}$ (2) (Liters)	$M_{c,r}$ (3) (Kg. U-235)	$M_{s,r}$ (4) (Kg. U)	$M_{c,b}$ (3) (Kg. U-235)	$M_{s,b}$ (4) (Kg. U)
0	8.930	9.55	5.03	3.87	10.15	7.81	44.93	21.6	90.63	43.6
0.98	6.700	7.16	4.84	3.73	11.25	8.65	32.41	15.58	75.39	36.2
2.94	4.470	4.78	5.79	4.45	13.78	10.60	25.89	12.45	61.58	29.6
8.96	2.230	2.38	6.80	5.24	16.68	12.81	15.16	7.30	37.21	17.9
20.60	1.120	1.20	6.42	4.94	15.91	12.22	7.19	3.46	17.82	8.59
43.90	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
	SNM-871 70- 9(
SECTION:	300, Subpart 309
Nuclear Safety Evaluation-Data Figures 309-XXV thru XXVIII	
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ISSUED:	2/6/70
SUPERSEDES:	New
APPROVED:	

SECTION 400

SECTION 400 - HEALTH STANDARDS

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401. General Health Physics Requirements

The Radiation Protection Program shall comply with the standards established in Title 10, Code of Federal Regulations, Part 20, the Standards of this Subsection and the requirements of other regulatory agencies. Every reasonable effort will be made to maintain radiation exposure of employees and releases of radioactive materials in effluents to unrestricted areas as far below these standards as practicable.

Internal procedures and/or data forms are used in performing and documenting the Health Physics functions in accordance with this section. Changes to these procedures shall be reviewed by the Health Physics Specialist or Health Physics Consultant prior to approval by the Manager of the Nuclear and Industrial Safety Department.

1. Surface Contamination*

1.1 Restricted areas (As defined in 10 CFR 20)

Action	Contamination Action Level (Excluding Process Equipment)
Immediate Cleanup	10,000 alpha dpm/100 cm^2 removable (smear) 100,000 beta dpm/100 cm^2 removable (smear)
End of Shift Cleanup	5,000 alpha dpm/100 cm^2 removable (smear) 50,000 beta dpm/100 cm^2 removable (smear)

Material on processing equipment or fixed on surfaces shall be limited as required to control airborne radioactivity and external radiation exposures.

1.2 Unrestricted Areas (Release of Materials and equipment but does not include the abandonment of buildings)

1.2.1 The maximum amount of fixed alpha radioactivity in disintegrations per minute per 100 square centimeters shall not exceed 25,000.

*Apply to uranium, natural thorium and mixed fission and activation products.

1.2.2 The average amount of fixed alpha radioactivity in disintegrations per minute per 100 square centimeters shall not exceed 5,000.

1.2.3 The maximum amount of removable (capable of being removed by wiping the surface with a filter paper or soft absorbent paper) radioactivity (alpha or beta) in disintegrations per minute per 100 square centimeters shall not exceed 1,000.

1.2.4 The maximum level at one centimeter from the most highly contaminated surface measured with an open-window beta-gamma survey meter through a tissue equivalent absorber of not more than seven milligrams per square centimeter shall not exceed one millirad per hour.

The average radiation level at one centimeter from the contaminated surface measured in the same manner shall not exceed 0.2 millirad per hour.

1.2.5 A reasonable effort shall be made to eliminate residual contamination.

1.2.6 Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering materials unless contamination levels, as determined by a survey and documented, are below the limits specified above prior to applying the covering. A reasonable effort must be made to minimize the contamination prior to the use of any covering.

2. Air and Gaseous Effluents

The radioactivity concentration limits of 10 CFR 20 will be followed.

3. Records

Records of Personnel Monitoring, Monitoring Surveys, Respiratory Protection Program Personnel Instructions and Instrument Mainenance and Calibration shall be maintained by Health Physics.

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402. Personnel Monitoring

Personnel monitoring shall be supplied to each individual who is likely to receive a dose in excess of 25% of the applicable limits in 10 CFR 20 and those personnel who routinely work in the process area.

1. Dosimetry

The personnel dosimeters shall be sensitive to an exposure of 25 millirem. Hand exposures will be determined by surveys. Exposures in excess of 25% of the applicable limits shall be investigated.

2. Bioassay

The urine analyses shall be sensitive to concentrations of 10 alpha dpm/liter. Insofar as possible, samples are collected after two days off the job as follows:

2.1 At the start and termination of employment.

2.2 On a routine schedule consistent with the degree of exposure and results of past samples but at least as follows:

Minimum Bioassay Frequency*

Area	Frequency
Clean or clear areas	Employment and termination
Intermediate or limited contaminated areas	6 months
Contaminated or restricted areas	3 months
If respiratory protection is required, or if the radionuclides are in a soluble form	Monthly

*These frequencies apply to personnel spending 10% or more of their assigned work schedule in the area listed.

2.3 Following a suspected potential overexposure or ingestion of contaminated material.

2.4 The investigation and action levels follow:

2.4.1 All samples above 50 dpm/liter shall be investigated.

2.4.2 A sample in excess of 100 dpm/liter will require immediate restriction of the individual to jobs where airborne radioactivity levels are not expected to exceed 25% MPC. The individual will remain on restricted jobs until two consecutive samples less than 50 dpm/liter are obtained.

2.4.3 More intensive investigation shall be performed on the circumstances of exposure for persons who remain on restriction for three (3) or more resamplings.

3. <u>In-Vivo Counting</u>

In-vivo counting is performed on individuals for whom personnel monitoring is required. Persons will be scheduled for counting in accordance with the following criteria:

3.1 All individuals for whom in-vivo counting is required shall be counted at least bi-annually.

3.2 Counting will be performed more frequently than bi-annually when:

3.2.1 A person is known or suspected to have been subject to an overexposure.

3.2.2 An individual is assigned to sensitive operations where the potential for significant ingestion exists.

3.2.3 A person has been found to have an elevated lung burden by past in-vivo counts.

3.3 The counting unit is available on an approximate six month schedule. On each placement, persons counted shall consist of:

3.3.1 Those persons for whom counting is predicated by subparagraph 3.2.

3.3.2 Routine scheduling as required to meet the minimum frequency criteria of subparagraph 3.1.

4. Exposure Evaluations

In-vivo counting is used and fecal sampling may be performed to augment exposure evaluations.

403. Respiratory Protection Program

1. In circumstances in which adequate limitation of the inhalation of radioactive materials by use of process or other engineering controls is impractical, United Nuclear Corporation may permit an individual in a restricted area to be exposed to average concentrations of airborne radioactive materials in excess of the limits specified in Appendix B, Table I, Colum 1 of 10 CFR 20 provided:

1.1 The individual uses respiratory or other appropriate protective equipment such that the total intake, in any period of seven consecutive days by inhalation, ingestion or absorption, would not exceed that intake which would result from breathing the concentrations specified in Appendix B, Table I, Column 1 of 10 CFR 20 for a period of 40 hours.

1.2 UNC shall advise each respirator user that he may leave the area for relief from respirator use in case of equipment malfunction, physical or psychological discomfort, or any other condition that might cause reduction in the protection afforded the wearer.

1.3 UNC shall maintain a respiratory protection program adequate to assure that the objectives of 1.1 above is met. Such program shall include:

1.3.1 Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposure, and to permit proper selection of the respiratory protective equipment;

1.3.2 Procedures to assure proper selection, supervision and adequate training of personnel using such protective equipment;

1.3.3 Procedures to assure the adequate fitting of respirators and the testing of equipment for operability;

1.3.4 Procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair and storage.

1.3.5 Bio-assays of individuals and other surveys as may be appropriate to evaluate individual exposures and to assess protection actually provided; and

1.3.6 Records sufficient to permit periodic evaluation of the adequacy of the respiratory protective program.

403. Respiratory Protection Program (continued)

1.4 UNC has evaluated the protective equipment (1) and has determined that, when used to protect against radioactive material under the conditions of use to be encountered, such equipment is capable of providing a degree of protection at least equal to the protection factors listed in Table 403-I.(2)

2. United Nuclear Corporation shall not assign protection factors in excess of those given in Table 403-I, in selecting equipment.

(1) In evaluating respiratory protective equipment for use against radioactive materials to assure that the equipment provides the protection factors listed in Table 403-I, UNC may accept equipment approved under appropriate test schedules of the U. S. Bureau of Mines to the extent pertinent.

(2) The factors listed apply only to protection against radioactive materials. Additional precautions may have to be taken to protect against concurrent non-radiation hazards.

Description	Modes 1/	PROTECTION FACTORS 2/ Particulates and Vapors and Gases Except Tritium Oxide 3/	Tritium Oxide
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask		10	1
Facepiece, full		100	1
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Air-line respirator			
Facepiece, half-mask	CF	100	2
Facepiece, half-mask	D	100	2
Facepiece, full	CF	1000	2
Facepiece, full	D	500	2
Facepiece, full	PD	1000	2
Hood	CF	1000	2
Suit	CF	[illegible]4/	4/
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full	D	500	2
Facepiece, full	PD	1000	2
Facepiece, full	R	1000	2
3. Combination respirator — Any combination of air-purifying and atmosphere supplying respirator.		Protection factor for type and mode of operation as listed above.	

1/ CF: continuous flow
D : demand
PD: pressure demand (i.e., always positive pressure)
R : recirculating

2/ (a) For purposes of this authorization the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the facepiece) under the conditions of use. It is applied to the airborne concentration to determine the concentration inhaled by the wearer, according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Airborne Concentration}}{\text{Protection Factor}}$$

LICENSE: SNM- 33	Docket: 30-3
SECTION: 400, Subsection: 403	
Table 403-I	
APPROVED	
ISSUED October 31, 1968	
SUPERSEDES: New	
Page 1 of 2	

2/ (b) The protection factors apply:

(i) only for individually fitted respirators worn by trained individuals and used and maintained under supervision in a well-planned respiratory protection program.

(ii) for air purifying respirators only when high efficiency particulate filters and/or sorbents appropriate to the hazard are used.

(iii) for atmosphere supplying respirators only when supplied with adequate respirable air.

3/ Excluding radioactive contaminants that present an absorption or submersion hazard.

4/ Appropriate protection factors must be determined taking account of the permeability of the suit to the contaminant under conditions of use. No protection factor greater than 1000 shall be used except as authorized by the Commission.

NOTE 1: Protection factors for respirators as may be approved in the future by the U. S. Bureau of Mines according to approval schedules for respirators to protect against airborne radionuclides may be used in lieu of the protection factors listed in this Table. Where additional respiratory hazards other than radioactive ones are present, especially those immediately dangerous to life, the selection and use of respirators shall also be governed by the approvals of the U. S. Bureau of Mines in accordance with their applicable schedules.

404. Facility and Equipment Requirements

1. Zoning

The facility shall be zoned to define contamination areas, limited contamination areas and clear areas. Protective clothing or special clothing, shower and change facilities shall be provided for use in the contamination area. A sink and alpha survey meter or hand monitor shall be provided at the exit from the contamination area.

2. Ventilation

Air flow shall be from areas of lower to areas of higher contamination. Hoods, glove boxes, or local exhaust will be used to control contamination and airborne concentrations.

Ventilation equipment will be installed that is suited to the operations being performed, including in such considerations the specific activity of the uranium and its relative dispersability. General guidelines for ventilation and air movement requirements are:

2.1 Class A Ventilation
Fume Hoods: Minimum 100 ft/min face velocity.
Used for control of fumes where modest quantities of airborne uranium are generated. Typically these hoods might handle uranium primarily in solution form.

2.2 Class B Ventilation
Local Exhaust: Minium 150 ft/min spot velocity.
Spot ventilation used at locations such as sampling ports and product take-off parts where the need for air control is momemtary or of low frequency and generation of airborne material is limited by equipment design.

2.3 Class C Ventilation
General Purpose Hoods: Minimum 150 ft/min face velocity.
Used for material transfer and similar operations where generation of airborne material is limited.

404. Facility and Equipment Requirements (continued)

2.4 Class D Ventilation
Restricted Access Hoods: Minimum 150 ft/min face velocity
Hoods designed for mixed use in which operations are performed that would require Class C ventilation but for which total containment is desirable during certain process steps involving active or potentially active generation of airborne contamination. Such hoods will frequently include gloveports, and will normally be closed during production operations.

2.5 Class E Ventilation
Glove Boxes: Enclosed hood under negative pressure. Operations where airborne material is actively generated, or where large quantities of material are handled such that Classes C or D hoods would not be adequate will be contained in glove boxes. Airlock entry ports will be provided.

Fire prevention, the potential for generating explosive atmospheres, and the presence of pyrophoric materials will be considered in ventilation design. Disaster control techniques such as air dilution, quenching or inert atmospheres and fire detection or automatic extinguishing systems will be utilized as appropriate for such hazards.

3. Exhaust Air Cleaning

Air effluents from processs areas and process equipment involving uranium in a dispersable form will be subject to air cleaning. Exhaust air cleaning will include use of high efficiency filters except where the effluents, evaluated individually, do not contribute significantly to the total emission.

Air cleaning equipment that may be used is:

3.1 Type 1
Cyclone Collectors.
Used to remove particulates from exhaust streams that are heavily loaded.

3.2 Type 2
High Efficiency Particulate Air Filters
Used in the majority of cases for highest efficiency air cleaning, normally in conjunction with roughing filters to extend useful life and improve reliability.

404. Facility and Equipment Requirements (continued)

3.3 Type 3.
Wet Scrubbers.
Used to clean heavily loaded air streams that are not suited, due to air quality or temperature, to other cleaning methods.

3.4 Type 4.
Dry Scrubbers.
Used primarily for cleaning air streams containing corrosive agents that render wet scrubbing impractical.

3.5 Type 5.
Fabric Filters.
Normally used in systems where material impinging on them can be returned to the process using reverse jet, pulsed air or other dislodging methods.

3.6 Type 6.
Special Filters
Ceramic or metallic frit filters, usually an integral part of process equipment, may be used for special air cleaning requirements.

4. Liquid Effluents

Process waste and laundry water is transferred to a lagoon or liquid handling system prior to discharge. Where particulate contaminants constitute a significant radioactive component of the liquid, filtration may be required before discharge. The contamination level of these effluents is monitored.

Chemical processing of liquid wastes may be performed. Such treatment might include precipitation, co-precipitation, flocculation, sedimentation, or other appropriate removal techniques.

Untreated liquid effluents may originate from the following sources: storm drains, showers, change room floor drains, and lab sink drains. Disposal of lab analytical residues to the sink drains is not practiced, but instead they are recycled for recovery.

LICENSE: SNM-33 Docket: 70-36
SECTION: 400 - HEALTH PHYSICS STANDARDS
Subsection: 405 - Instrumentation

405. Instrumentation

The minimum instrumentation required for operational surveillance is listed below. All instruments are calibrated quarterly or in accordance with the manufacturer's recommendations. The manufacturer's calibration of flowmeters, velometers, rotameters and orifices are used.

1. Nuclear Alarm System

The nuclear alarm system consists of gamma sensitive detectors, audible alarms and remote indicator panel at or near the guard station. The requirements for this alarm system follow:

1.1 Detector units shall have a pre-set alarm level of not less than 5 MR/hr or greater than 20 MR/hr.

1.2 Detector units shall also have a response time no greater than 3 seconds at a radiation level of 20 MR/hr.

1.3 Detectors shall be located so as to be capable of detecting and operating the alarm from an incident of the magnitude that would result in a gamma flux of 3×10^5 mrem/hr one (1) foot from the source of radiation.

1.4 Detectors shall be installed within 120 feet of every location where 500 grams or more of Special Nuclear Material is handled, used, or stored.

1.5 Whenever possible, the location and spacing of the detectors is chosen to avoid the effect of shielding by massive equipment or materials. Low density materials of construction such as 2 x 4 stud construction walls, plaster or metal corrugated panels, asbestos panels, doors, panel walls and steel office partitions are disregarded in determining the spacing. The spacing is reduced where high density building materials such as brick, concrete, concrete or cinder blocks, or lead-lined x-ray rooms, shield a potential accident area from the detector.

Calculations to determine adequate coverage through significant shielding materials is performed using the following formula:

$$I = \frac{I_o\ (e^{-\mu t})}{d^2}$$

where I = gamma intensity at the detector (minimum for calculations will be 20 mrem/hr)

405. Instrumentation (continued)

I_o = Unattenuated gamma intensity one (1) foot from the flux source

μ = mass absorption cross section of the shielding material x density of the shield

t = thickness of shield in centimeters. Where angle of incidence Θ is not 90° to the plane of the barrier, t will assume the dimension, csc Θ t.

d = distance from source in feet.

Such calculations will not include the effect of broad beam attenuation. An example of the calculational technique is shown in the health physics evaluation for this subsection.

1.6 The detector and alarm circuits shall be equipped with an auxilliary self starting diesel generator which will automatically supply power to the system in the event of disruption of primary power. This backup power system will be checked at least quarterly.

1.7 The system will be tested by sounding the alarm at least monthly and at the time of each practice evacuation drill.

1.8 Automatic monitors shall give warning in case of any malfunction which renders the system inoperable.

1.9 The alarm shall be clearly audible in all portions of areas in which Special Nuclear Materials are handled, used, or stored and in all adjacent areas where significant exposure to radiation may result from an incident.

2. Alpha Counting System

Minimum detectability - 10 DPM

3. Alpha Survey Meter

Minimum counting efficiency - ~ 30% (calibrated to read 2π)
Minimum Range - 0 - 100,000 counts per minute

4. Air Sampling Equipment

Routine - Nominal 20 liters per minute sampling rate

405. Instrumentation (con't)

5. Beta-Gamma Survey Meter

GM type with maximum window thickness of not more than thirty milligrams per square centimeter.

Minimum range - 0 - 60,000 counts per minute

0 - 20 mr/hr

6. Beta Gamma Counting System

Minimum detectability - 200 dpm

Emergency instrumentation is listed in Section 600.

NUCLEAR ALARM SYSTEM COVERAGE CALCULATIONS

I. Following are calculations of the effectiveness of nuclear alarm system coverage through significant barriers encountered in facility construction.

Using the Formula $$I = \frac{I_o\,(e^{-\mu t})}{d^2}$$

A. Barrier: 8" Concrete Block

Thickness, t = 7.625 cm/in = 19.37 cm.

$\mu = 0.0317\ cm^2 \times 1.22\ gm/cm^3 = 0.0387\ cm^{-1}$

$$d^2 = \frac{3 \times 10^5\ mr/hr\ (e^{-0.750})}{20\ mr/hr}$$

$d^2 = 7.10 \times 10^3$

d = 84 ft. maximum permitted distance of source from detector to provide coverage.

B. Barrier: 12" Concrete Block

Thickness t = 11.625" x 2.54 cm/in = 29.6cm

$\mu = 0.0317\ cm^2 \times 1.17\ gm/cm^3 = 0.0371\ cm^{-1}$

$$d^2 = \frac{3 \times 10^5\ mr/hr\ (e^{-1.10})}{20\ mr/hr}$$

$d^2 = 5 \times 10^4$

d = 70.7 ft. maximum permitted distance of source from detector to provide coverage.

C. Barrier: 8" Poured Concrete Wall

Thickness t = 8" x 2.54 cm/in = 20.3 cm.

$\mu = 0.317\ cm^2 \times 2.3\ gm/cm^3 = 0.735\ cm^{-1}$

$$d^2 = \frac{3 \times 10^5\ mr/hr\ (e^{-1.49})}{20\ mr/hr}$$

$d^2 = 3.375 \times 10^3$

d = 58.1 ft. maximum permitted distance of source from detector to provide coverage.

II. The effectiveness of the nuclear alarm system when a barrier is interposed between the source and the detector at an angle which is not normal to the line between them is calculated using the following formula:

$$I = \frac{I_o\,[e^{-\mu t(\csc\theta)}]}{d^2}$$

where θ is the angle of incidence of the line between the source and detector with respect to the plane of the barrier.

In this manner, increased attenuation caused by angular incidence of the beam on a barrier of given thickness is calculated.

LICENSE: SNM-33	Docket 70-3[illegible]
SECTION: 400, Subsection: 405.1	
Health Physics Evaluation Nuclear Alarm System Coverage	
APPROVED:	
ISSUED: October 31, 1968	
SUPERSEDES: New	
Page 1 of 1	

406. Surveillance

1. Special Surveys

All new, non-routine, and spill cleanup operations shall be performed under the congizance of Health Physics.

With the exception of incidents requiring immediate evacuation, major spills or other accidental releases shall be cleaned up immediately. Criticality restrictions on the use of containers and water shall be followed at all times. The Foreman and Health Physics must be notified immediately of such incidents. Appropriate precautions such as use of respirators shall be observed.

2. Routine Surveillance

Surveys shall be conducted on a regularly scheduled basis consistent with plant operation and survey results. The frequency of survey depends upon the contamination levels common to the area, the extent to which the area is occupied, and the probability of personnel exposures.

3. Surface Contamination

Corrective action and/or cleanup is initiated when contamination exceeds the action levels.

4. Airborne Concentrations in Restricted Areas

4.1 Airborne levels in excess of 25% of the maximum permissible concentration require posting in accordance with 10 CFR 20 and an investigation of the causes.

4.2 Airborne levels in excess of the maximum permissible concentration require exposure evaluation. Controls to restrict the personnel to 40 MPC hours per week shall be required.

4.3 Air sampling will be performed using fixed sample stations, breathing zone samples, and lapel type samplers.

406. Surveillance (continued)

The type of air sample collected at a specific operation or location will depend on the type, frequency, and duration of operations being performed. One or more of these sample methods will be employed at intervals prescribed by the NIS Representative. General criteria for sampling are:

4.3.1 Fixed sample stations will be used where uranium handling operations are pursued for extended periods of time, or where short term operations occur frequently. These samples will be located as near as practical to the breathing zone of the person performing the operations.

4.3.2 Breathing zone samples will be collected where operations are performed at a location for limited periods of time and may include consecutive short term operations performed at the same location. These samples may be attached to the worker or hand held or fastened located so as to sample air that is representative of the air he is breathing.

4.3.3 Lapel samplers may be used to supplement sample information collected by fixed samples or breathing zone samples. Since they integrate exposure over the time they are worn and used, they might not provide concentration information for specific operations permitting corrective action. The sample head shall be attached to the lapel or upper torso as close to the breathing zone as practical.

4.3.4 Emphasis will be placed on sampling new operations or processes until adequate, effective, control of airborne contamination is assured.

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LICENSE: SNM-33, Docket: 70-36
SECTION: 400 - HEALTH PHYSICS STANDARDS
SUBSECTION: 406 - Surveillance

Approved

Issued 4/21/72

Supersedes 3/19/71

5. Air and Gaseous Effluents

Air effluents from process areas and process equipment will be sampled continuously during operations prior to discharge. These samples will be changed at intervals not exceeding seven days. All samples shall be counted after suitable delay for decay of radon daughters, and the results evaluated.

For new process exhausts, the samples will be changed at more frequent intervals to assure proper and adequate function of air cleaning equipment.

406. Surveillance (continued)

Effective air control by ventilation systems will be assured by face velocity checks performed at least weekly. These checks may be supplemented by pressure drop measurements across air cleaning devices or inspection of such devices for continued integrity or loading that would impair their effectiveness. When ventilation control suffers or effluent concentrations rise, cleaning devices will be cleaned or replaced.

6. Liquid Effluent Sampling

Levels of contamination in liquid effluents are measured by representative grab sampling of batch discards, by proportional sampling of continuous discharges, or both. Samples will be collected at or prior to the point of discharge from the waste handling system.

Samples will be analyzed for alpha and beta activity. Where liquid wastes are discharged into a river or stream, a grab sample shall be collected at least monthly from above and below the plant outfall.

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Issued	2/6/70
Supersedes	10/31/68
Approved	
Amendment No.	

LICENSE: SNM-33 Docket 70-36: SNM-777 Docket 70-820
SECTION: 500 - NUCLEAR MATERIAL MANAGEMENT
Subsection:
Subpart:

500. Nuclear Material Management

This section contains the "Fundamental Material Control" (FMC) manual forwarded under separate cover.

NATIONAL INSERTABLE-TAB INDEXES ENABLE YOU TO MAKE YOUR OWN SUBJECT ARRANGEMENT, USING PLAIN INSERTS ON WHICH TO WRITE YOUR OWN CAPTIONS.

The Beaded edge on tab makes it easy to insert captions

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SECTION 600

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601. Content of Emergency Control Plan

The emergency control plan applicable to Fabrication Operations will meet the AEC's Annex B, Minimum Requirements for Licensee's Plans for Coping with Radiation Emergencies.

602. Administration of Emergency Control Plan

The emergency control plan for Fabrication Operations is that plan applicable to the Naval Products Division of United Nuclear Corporation, New Haven, Connecticut. (SNM-368, Docket 70-731).

United Nuclear Corporation has prime responsibility for establishing and maintaining emergency plans and procedures, although Gulf Nuclear Fuels Company will assure that those plans and procedures meet the requirements stated in Subsection 601.

SECTION 700

SECTION 700 - TRANSPORTATION

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LICENSE:	SNM-33 Docket: Docket: 70-36
SECTION:	700
ISSUED:	4/15/74
SUPERSERED:	10/31/68

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	Issued 2/6/70
LICENSE: SNM-33 & SNM-777, Docket: 70-36 & 70-820 SECTION: 700 - TRANSPORTATION	Supersedes 10/31/68
Subsection: 701 - Introduction	Approved
	Amendment No.

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

LICENSE: SNM-33 Docket 70-36	Approved
SECTION: 700 - Transportation Sub-section: 702 - Shipping standards	ISSUED October 31, 1968
	SUPERSEDES New

702. Shipping Standards

1. Purpose

1.1 To assure compliance with all UNC, local, State and Federal criteria, restrictions or regulations concerning the shipment of SNM.

1.2 To outline periodic inspection criteria to insure that shipping containers meet approved standards.

1.3 To list records and reports required.

2. Handling of Material of Unknown Enrichment

2.1 The material is treated as fully enriched unless a lower enrichment value has been verified.

3. Container Inspection

3.1 Prior to each use of any container, the container is inspected to insure that:

3.1.1 It has not been significantly damaged.

3.1.2 Original design conditions approved by AEC and DOT are maintained.

3.1.3 Marking and labeling is correct as required by the AEC and DOT approvals.

The Shipping Department is responsible for this inspection. The NIS Department overchecks as part of its audit function.

4. Records

4.1 A record of each shipment will be maintained for a period of 2 years.

4.1.1 The record will include (for unirradiated SNM only);

4.1.1.1 Identification of the container used by model number.

4.1.1.2 Details of any significant defects in the container, including the means used to repair the defects and prevent their recurrence.

4.1.1.3 Volume and identification of coolant (where applicable).

4.1.1.4 Type and quantity of SNM in each package.

LICENSE: SNM-33 Docket 70-36	Approved
SECTION: 700 - TRANSPORTATION Subsection: 702 - Shipping Standards	ISSUED October 31, 1968
	SUPERSEDES New

702. Shipping Standards (continued)

4.1.1.5 Total quantity of SNM in each shipment.

4.1.1.6 Date of Shipment.

4.1.1.7 For Fissile Class III, any special controls exercised.

4.1.1.8 Name and address of the transferee.

4.1.1.9 Address to which shipment was made.

4.1.1.10 Results of inspection described in Subpart 702.3 above.

UNITED NUCLEAR CORPORATION

Approved

LICENSE:	SNM-33, Docket: 70-36
SECTION:	700 - TRANSPORTATION
Subsection:	703 - Shipping Containers

ISSUED 4/15/74

SUPERSEDES 10/31/68

703. Shipping Containers

The following shipping containers will be used for the transportation of SNM:

Shipping Container Model Number	AEC Amendment Number
1352	71-27
1484	71-25
1634	71-20 (SNM-777)
2400	71-7
2600	71-3
2700	71-14
2800	71-31
2900	71-24
2901	71-30
3000	71-13
OR-30	71-33
30A, 30B	*

*Approved by Oak Ridge Operations Office

The use of these containers is subject to the conditions specified in the above listed AEC amendments and to the conditions specified in this license.

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LICENSE: SNM-33, Docket: 70-36 SECTION: 700 - TRANSPORTATION SUBSECTION: 704 - Handling of Incoming & Outgoing Shipments	 Approved Issued 4/21/72 Supersedes 10/31/68

704. Handling of Incoming and Outgoing Shipments

1. Storage of Undamaged Containers

Storage of as-received containers or containers awaiting shipment may be stored anywhere within the fenced area.

The specific location shall be covered by the nuclear criticality monitoring and alarm system. The storage arrangement shall be one of the methods listed below:

1.1 The same arrangement that they occupied on the transportation vehicle.

1.2 Under controlled storage conditions, the total number of containers meet a "100 unit" rule (i.e., no more than that number of containers whose assigned radiation units can be summed to a maximum value of 100.0). Controlled storage is here defined as an area where a positive safeguard is provided against the inadvertant addition of a moderating media (i.e., in a roofed warehouse or covered area including use of a water-proof tarpulin).

When the storage conditions do not meet the definition for controlled storage, the total number of containers meet a "75 unit" rule.

1.3 Separation of arrays described in 1.1 and 1.2 above is maintained in accordance with criteria of Subsection 303.

2. Loading, Unloading and Handling of Contents

1.1 During unloading and the subsequent handling, incidental to receipt and storage, the contents of received shipping containers will be handled and stored to the enrichment as certified by the shipper. If the enrichment is unknown or doubtful, the material will be handled and stored as fully enriched until enrichment has been determined by analyses.

1.2 Containers will be closed and will not be opened in outside areas.

704. Handling of Incoming and Outgoing Shipments

2. Loading, Unloading and Handling of Contents (continued)

1.3 Containers will identified so that the enrichment and quantity of SNM can be readily determined. This identification will be able to withstand exposure to adverse weather conditions.

1.4 When storing acids or other liquid wastes, provisions will be made to prevent loss of containment by corrosion, freezing, or other means.

1.5 Containers will be elevated from the ground in a fixture or device which will not retain water at the base of the container.

1.6 Provisions will be made for protection against adverse weather conditions including high winds.

1.7 If relevant to nuclear criticality safety control, incoming shipments will be sampled prior to processing to verify enrichment and moisture content. Sampling and verification techniques will be in accordance with Section 500.

1.8 Checks will be performed to insure that the requirements of this Subsection are met. These checks will be performed in accordance with Subsection 207.

3. Damaged Containers

Containers received in a damaged condition will be held separate from other SNM (in accordance with criteria of Subsection 303).

Prior to further handling or unloading, the extent of damage will be evaluated to establish proper action. Results of the evaluation and action to be taken shall be reviewed and approved by the NIS Representative.

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SECTION 900 - FUEL FABRICATION OPERATION

SUBSECTION 910 - STORAGE

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LICENSE:	SNM-33, Docket: 70-36	Page 1 of 1
SECTION:	900 - Fabrication Operation	Approved
SUBSECTION:	910 - Storage	Issued 4/21/72
SUBPART:	911 - General Considerations 911.1 - Outside Storage	Supersedes 10/21/68

911.1 Outside Storage

SNM bearing materials may be stored outside the buildings of the Fabrication Operation within the fenced-in area if the SNM is in a shipping container. Arrays of containers will be stored as described in Subsection 704. Specifically, the following limitations will be employed:

1.1 Containers will be sealed and not opened in outside areas. Prior to opening, containers will be taken into a process, shipping-receiving or storage area.

1.2 Containers will be labeled such that their contents can be readily determined. These labels will be weather proof and will either list or reference appropriate internal documents which list the enrichment, quantity of SNM and other pertinent process and health and safety information.

1.3 SNM bearing liquids will not be stored outside. Acids and corrosive materials will be stored in containers which also meet the criteria of Title 49 CFR.

1.4 Containers will be stored on pallets to provide elevation off the ground.

1.5 During periods of high winds, containers will be either moved inside or secured by rope, chain, etc. to the ground or to building walls.

1.6 Containers and outside storage arrangements will be checked during routine inspections as per Subsection 207.

811.2 INSIDE STORAGE

SNM may be stored in buildings in specified locations, in shipping containers. Arrays of containers will be stored as described in Subsection 704.

After unloading from shipping containers, SNM will be stored in storage areas or devices described in this Subsection.

In-process storage devices are placed throughout the buildings to retain SNM during processing or between process steps. These devices are metal racks or concrete bunkers which provide spacing between safe cross section metal boxes or ports, or safe piece count batches.

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Approved

ISSUED October 31, 1968

SUPERSEDES NEW

LICENSE: SNM-777, Docket: 70-820
SECTION: 800 - FUEL FABRICATION OPERATION
Subsection: 810 - Storage
Subpart: 812 - 19H Fuel Storage Area
812.1 - General

812.1 GENERAL

A locked, controlled storage area is located in Building 19H which is used primarily for the storage of incoming SNM. Uranium metal, uranium compounds such as UO_2, UALX, uranium-aluminum alloy recycle material and various uranium bearing scraps are currently stored in this area. The various storage devices in this area are described in further detail in the remainder of this subpart. The layout of this area is shown on Sketch 812.1-I.

A small hood and a work table or desk are provided in this area for routine nuclear materials management and process work. Examples of types of operations to be performed in this hood and work table or desk are:

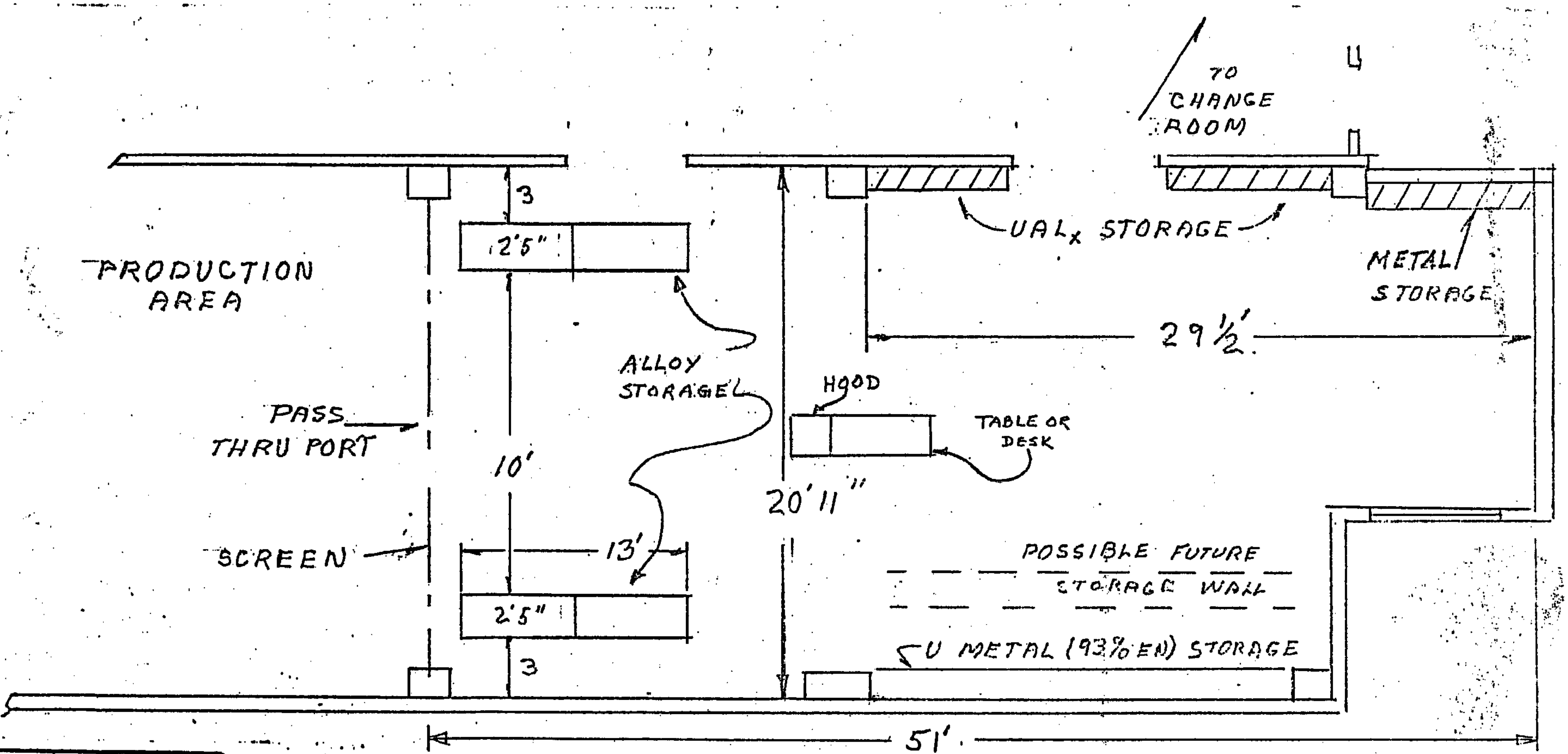
1. Weighing as received shipping containers.

2. Transfer of material from one container to another.

3. U-AL alloy change preparation.

Work involving opening containers, transfers of powders (except UALX) etc. will be performed only in the hood. The hood and table or desk will be centrally located in the area at least 5 feet from any other SNM in racks.

The hood will be limited to two safe geometry containers plus 700 grams of U-235 as loose pieces, such as cores, etc. Each container and the loose pieces will be separated 12 inches using administrative control. Safe geometrics will be obtained using the Figs. in Subsection 309. The same safety requirements apply to the work table or desk.



FUEL STORAGE - 19H

LICENSE: SNM-777; DOCKET: 70-3

SECTION: 800, SUBPART: 812.1

SKETCH 812.1 - I

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APPROVED

ISSUED: OCTOBER 31, 1968

UNITED NUCLEAR
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LICENSE: SNM-33 & SNM-777, Docket: 70-36 & 70-820	Issued 2/6/70
SECTION: 800 - FUEL FABRICATION OPERATION	Supersedes 10/31/68
Subsection: 810 - Storage	Approved
Subpart: 812 - 19H Fuel Storage Area; 812.2 - High Enrichment Uranium Metal Storage	Amendment No.

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.

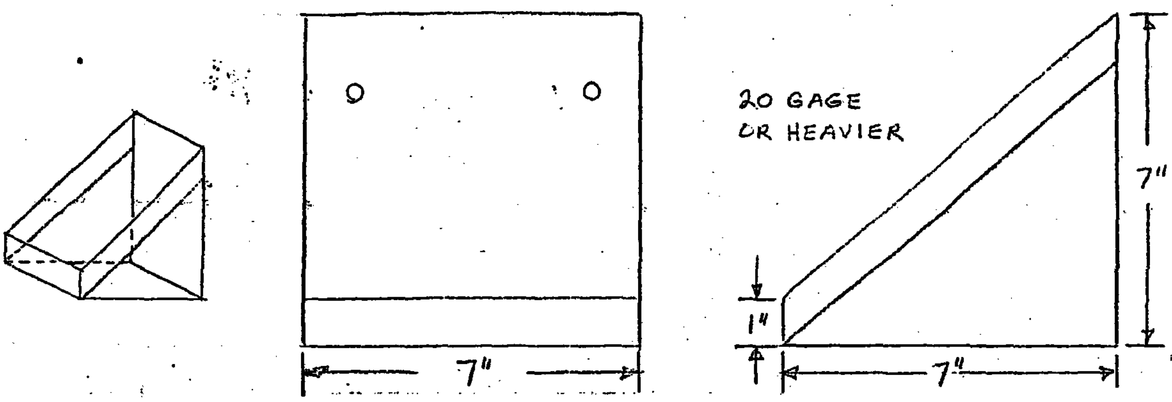
Racks will be constructed with individual holders fastened to a concrete block wall. These holders will be 7"x7"x7"-20 gage or heavier welded steel with two or more powder actuated bolts holding them to the wall. Individual holders will be arranged in a 4x12 arrangement with 16.8" center to center side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

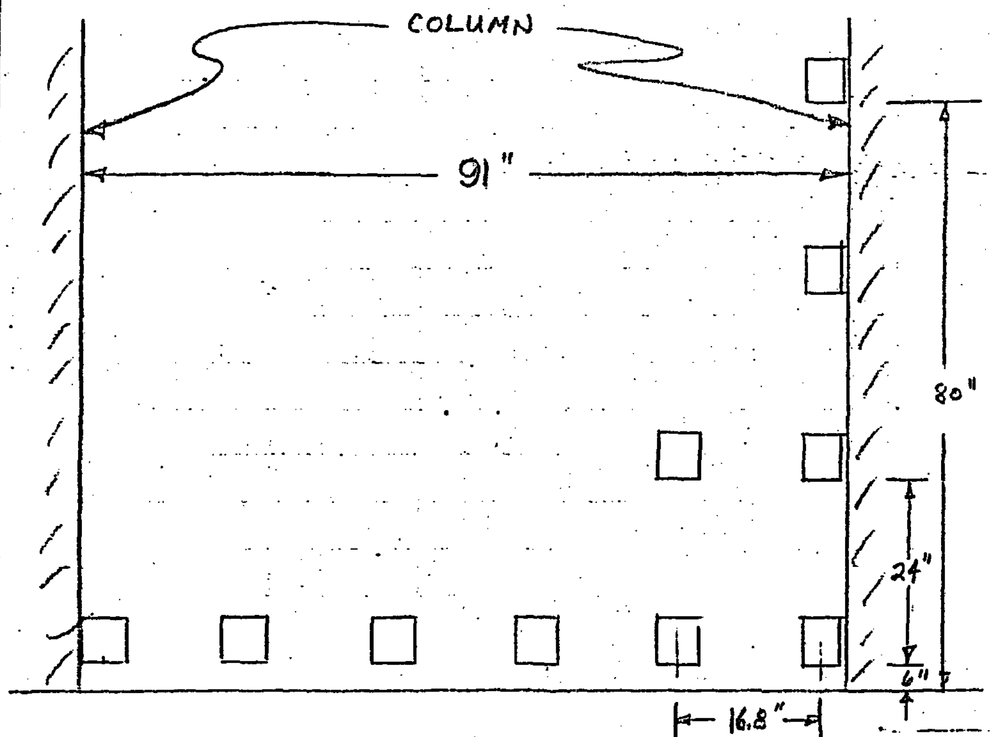
Details of this arrangement are shown on Sketch 812.2-I.

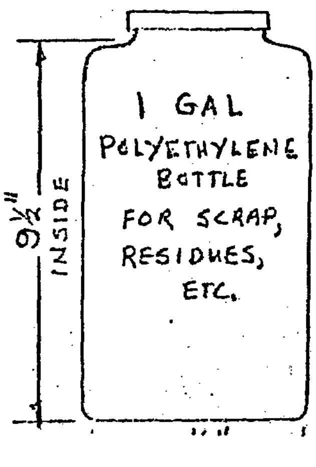
2. Nuclear Safety

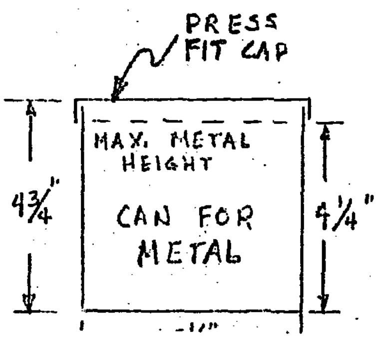
Cans for metal provide a safe volume when limited to 10 kgs U-235. The 1 gallon bottle is safe for densities up to and including 3.2
* grams U-235 per cubic centimeter. Containers meet the maximum unit quantities listed in Table V, LA-2063. Groups of containers on each wall form planar arrays which are 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









LICENSE: SNM-777; DOCKET: 70-8

SECTION: 800, SUBPART: 812.2

SKETCH 812.2 - I

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APPROVED

ISSUED: OCTOBER 31, 1968

I. DESCRIPTION

1. The maximum size container will be 1 gallon (6" OD x 10").

2. Holders (ports) are spaced 16.8" side-by-side and 24" top to bottom.

3. Arrays will be isolated in accordance with the criteria of Subpart 303.2.3.

II. NUCLEAR SAFETY OF INDIVIDUAL UNITS

Metal will be stored in 1.85 liter cans limited to 10 Kgs U-235 each. A 1.85 liter volume is subcritical for U-235 densities not exceed in 10 Kgs U-235 per liter as shown on Fig. 9, TID-7028. A mass of 10 Kgs U-235 is subcritical for densities greater than 225 Kgs U-235 per liter as shown on Fig. 8, TID-7028. Therefore, 10 Kgs U-235 in a 1.85 liter volume is subcritical. A one (1) gallon volume is nuclarly safe for material with a density not exceeding 3.2 kg per liter as shown in Table 309-I.

III. INTERACTION CALCULATIONS

The nuclear safety of a planar array of units on a wall will be evaluated using the solid angle method. The most reactive unit which will be stored in this arrangement will be a one gallon bottle.

1. Contribution from Units Above and Below Centermost Unit (#1 Units)

$r = 3"$, $h = 19"$

$$\Omega_1 = 2\pi(1-\cos\theta) \qquad \text{where } \tan\theta = \frac{r}{h} = \frac{3"}{19"} = .158$$

$$= 6.28\,(1-.988) = 6.28\,(.012) \qquad \cos\theta = .988$$

$$= .075$$

$$\Omega_1\,(\text{Total}) = 2 \times \Omega_1 = 2 \times .075 = .15 \text{ steradians}$$

2. Contribution from Units on Each Side (#2 Units)

$d = 6''$, $L = 10''$, $L/2 = 5''$, $h = 14''$

$\Omega_2 = \frac{2d}{h} \sin \theta$ where $\tan \theta = \frac{L/2}{h} = \frac{5''}{14''} = .375$

$= \frac{12''}{14''} (.336)$ $\sin \theta = .336$

$= .288$

Ω_2 (Total) $= 2 \times \Omega_2 = 2 \times .288 = .576$ Steradians

3. Contribution from Nearest Units in Next Row (#3 Units)

a = diagonal $= \sqrt{6^2+10^2} = \sqrt{136} = 11.7''$, b = diameter $= 6''$, $q = 23.5''$

$h = 19''$, $r = 14''$

$\Omega_3 = \frac{(ab)}{(q^2)} \cos \theta$ where $\tan \theta = \frac{r}{h} = \frac{14''}{19''} = .737$

$= \frac{(11.7 \times 6)}{(23.5)^2} (.805) = \frac{70.2}{552} (.805)$ $\cos \theta = .805$

$= .127 \times .805 = .102$

Ω_3 (Total) $= 4 \times \Omega_3 = 4 \times .102 = .408$ steradians

4. Contribution from Next Nearest Units in Next Row (#4 Units)

a = diagonal $= 11.7''$, b = diameter $= 6''$, $q = 45''$, $h = 43''$, $r = 14''$

$\Omega_4 = \frac{(ab)}{(q^2)} \cos \theta$ where $\tan \theta = \frac{r}{h} = \frac{14''}{43''} = .326$

$= \frac{(11.7 \times 6)}{(45)^2} (.950) = \frac{70.2}{2025} (.950)$ $\cos \theta = .950$

$= .0347 \times .95 = .033$

Ω_4 (Total) $= 2 \times \Omega_4 = 2 \times .033 = .066$ steradians

5. Contribution from Nearest Units in Second Row (#5 Units)

a = diagonal $= 11.7''$, b = diameter $= 6''$, $q = 36.5''$, $h = 31''$, $r = 14''$

$\Omega_5 = \frac{(ab)}{(q^2)} \cos \theta$ where $\tan \theta = \frac{r}{h} = \frac{14''}{31''} = .452$

$= \frac{(11.7 \times 6)}{(36.5)^2} (.911) = \frac{70.2}{1330} (.911)$ $\cos \theta = .911$

$= .053 \times .911 = .048$

Ω_5 (Total) $= 4 \times \Omega_5 = 4 \times .048$

$= .192$ steradians

6. Contribution from Next Nearest Units in Second Row (#6 Units)

a = diagonal of unshielded portion = 8.5", b = diameter of unshielded portion = 3"

q = 51.5", h = 48", r = 14"

$$\Omega_6 = \frac{(ab)}{(q^2)} = \cos\theta \qquad \text{where } \tan\theta = \frac{r}{h} = \frac{14''}{48''} = .292$$

$$= \frac{(8.5\text{x}3)}{(51.5)^2}(.860) = \frac{25.5}{2650}(.860) \qquad \cos\theta = .960$$

$$= .00964 \text{ x } .96 = .00925$$

Ω_6 (Total) = 4 x Ω_6 = 4 x .00925 = .037 steradians

7. Contribution from All Other Units in Array

All other units in the array are shielded.

8. Total Interaction

Total $\Omega = \Sigma\, \Omega_1$ (Total) + ... + Ω_6 (Total) = 1.43 steradians

9. Allowable Interaction

From Figure XVII, K-1019, Rev. 5, the allowable interaction for a 4.8 liter (approximately 1 gallon) volume is 1.9 steradians

IV. CONCLUSIONS

The storage arrangement is nuclearly safe.

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").

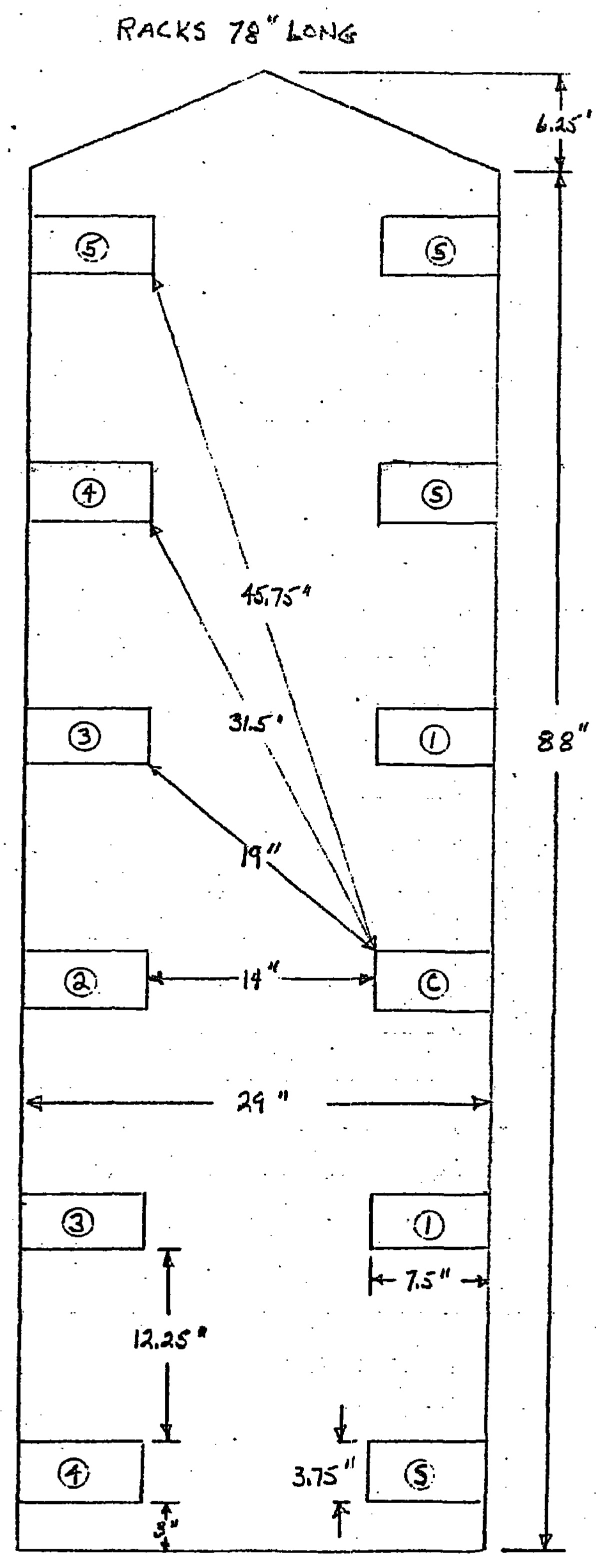
Racks will be constructed of 3/16" slotted angle bolted together. The outside will be covered with a thin metal (24 gage or heavier) to provide 14" edge-to-edge and 12-1/2" top to bottom separation between ports. Ports are long troughs running the entire length of the rack with a maximum 3-3/4"x7-1/2" 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



LICENSE: SNM-777; DOCKET: 70-[illegible]

SECTION: 800, SUBPART: 812.3

SKETCH 812.3 - I

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APPROVED

ISSUED: OCTOBER 31, 1968

I. 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 18 square inch (3" X 6") cross section containers and placed in racks forming rectangular parallelepipeds not exceeding 156" in length.

II. CALCULATION

Looking at the plane formed parallel to a wall, the fuel occupies a 468 square inch (3" X 156") cross sectional area in any one shelf. Shelves are separated by 12.25" therefore forming an area of

$$\text{Area} = \frac{(3"+12.25") \times 156"}{144\ in^2/ft^2} = 16.5\ ft^2/\text{shelf}$$

Using the Surface-Density Rule Limits set forth on page 10, "Density-Analog Techniques", H. C. Payten, Livermore Array Symposium, CONF 680909.

U(93) metal "surface density" limit = 15 kgsU/ft^2

Therefore, each shelf would be limited to

Shelf Limit = 15 kgsU/ft^2 X 16.5 ft^2/shelf = 248 kgsU/she

However, the two racks have 4 shelves so that

$$\text{Limit per shelf} = \frac{248\ \text{kgsU/shelf}}{4} = 62\ \text{kgsU/shelf}$$

Since there are 24 shelves total

Limit per Racks = 24 shelves X 62 kgsU/shelf = 1490 kgsU

Assuming each shelf in an individual limit, the actual mass is

M_A = 62 kgsU

From DP-532, the minium critical mass of a sphere of U-Al with a volume fraction of 0.127 is

$M_{C,B}$ = 870 kgsU

LICENSE: SNM-33, Docket: 70-36	
SECTION: 900, Subpart: 912.3	
Nuclear Safety Evaluation -	
High Enrichment Alloy Storage	
PAGE 1 OF 2	
APPROVED:	
ISSUED: 4/21/72	
SUPERSEDES 9/6/70	

Therefore, the fraction critical is

$$f = \frac{MA}{M_{C,B}} = \frac{62 \text{ kgsU}}{870 \text{ kgsU}} = 0.071$$

This meets the criteria in LA-3316 and CONF 680909 that

$$f \leq 0.3$$

III. CONCLUSION

The storage racks are safe but must be limited to 1490 kgU unless restricted to lessen loadings for structural reasons. The structural requirements of Subsection 304 will be met.

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ISSUED OCTOBER 31, 1968

SUPERSEDES NEW

LICENSE: SNM-777, Docket: 70-820
SECTION: 800 - FUEL FABRICATION OPERATION
Subsection: 810 - Storage
Subpart: 812 - 19 H Fuel Storage Area
812.4 - High Enrichment UALX Storage

812.4 HIGH ENRICHMENT UALX STORAGE

1. Description

These racks will be used for the storage of high enrichment UALX Powder after unloading from a shipping container.. Two racks, lined up end to end, are currently used. They will be arranged as shown on Sketch 812.1-I.

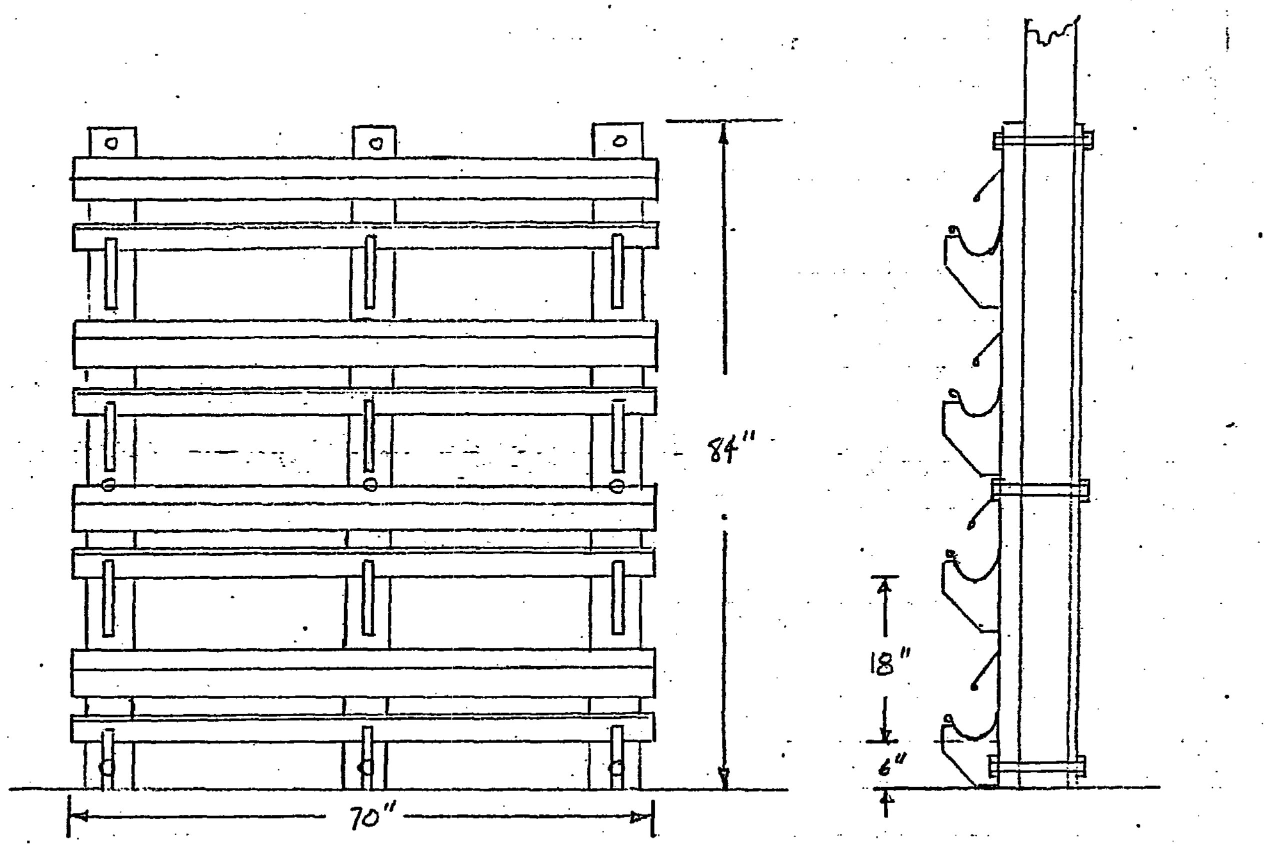
UALX will be stored in as received containers. Sealed "coffee can" type containers with 5" OD x 6" OH are used. Containers will be limited to 4.5 Kgs UALX (approximately 3.15 Kgs U-235).

Racks are 18 gage steel half cylinder troughs welded to 18 gage support gussets and to 5" x 1-3/4" channel iron. 18 gage non-stacking devices are placed across the openings to prohibit stacking cans on top of each other. Racks are anchored to the wall by 9- 1/2" bolts through the wall. This arrangement forms four 5-1/2" dia. x 70" long troughs spaced on 18" centers.

Details of construction are shown on Sketch 812.4-I.

2. Nuclear Safety

The 5" diameter cans are nuclearly safe for materials with densities up to and including 3.2 grams U-235 per cubic centimeter. The solid angle subtended by a centermost trough is 1.616 steradians.



LICENSE: SNM-777; DOCKET: 70-8

SECTION: 800, SUBPART: 812.4

SKETCH 812.4 - I

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APPROVED

ISSUED: OCTOBER 31, 1968

I. DESCRIPTION

1. UALX is contained in 5" OD x 6" OH "coffee cans" placed end to end in a trough forming a 5" dia. x 70" long cylinder.

2. Troughs are spaced 18" on centers forming a planar array.

3. Two racks aligned end to end will allow 140" array length.

II. NUCLEAR SAFETY OF INDIVIDUAL UNITS OR TROUGHS OF UNITS

Individual cans and cans placed in a trough have a 5" diameter maximum. This is a safe infinite cylinder diameter for materials with a density not exceeding 3.2 grams U-235 per cubic centimeter as listed on Table XII, K-1019, 5th Rev.

III. INTERACTION CALCULATIONS

The nuclear safety of these arrays will be evaluated using the solid angle method. Assuming an infinite length,

$$d = 5'', \sin \theta = 1, h = 18'' - \frac{5''}{2} = 15.5''$$

$$\Omega_{\text{trough}} = \frac{2d}{h} \sin \theta = \frac{2d}{h} = \frac{10''}{15.5''} = .645$$

Since the centermost trough sees only the troughs above and below

$$\Omega_{\text{trough (Total)}} = 2 \times \Omega_{\text{trough}} = 1.29 \text{ steradians.}$$

Assuming one can in transit, one foot from the centermost trough

$$d = 5'', L = 6'', L/2 = 3'', h = 12'' - \frac{5''}{2} = 9.5''$$

$$\Omega_{\text{can}} = \frac{2d}{h} \sin \theta \qquad \text{where } \tan \theta = \frac{L/2}{h} = \frac{3''}{9.5''} = .316$$

$$= \frac{10''}{9.5''} (.301) = 1.05 \times .301 \qquad \sin \theta = .301$$

$$= .317 \text{ Steradians}$$

The total interaction would be

$$\text{Total } \Omega = 1.29 + .317 = 1.61 \text{ steradians}$$

From Figure XVII, K-1019, Rev. 5, the allowable interaction for a 5" dia. infinite cylinder is 3.2 steradians.

IV CONCLUSIONS

The storage arrangement is nuclearly safe.

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LICENSE: SNM-777, Docket: 70-820
SECTION: 800 - FUEL FABRICATION OPERATION
Subsection: 810 - Storage
Subpart: 813 - 19H Inprocess Storage
813.1 - High Enrichment Alloy Inprocess Storage

Approved

ISSUED OCTOBER 31, 1968

SUPERSEDES NEW

813.1 HIGH ENRICHMENT ALLOY INPROCESS STORAGE

1. Description

These racks will be used to store U-AL alloy during processing. The material usually will be in the form of cores, met mounts, etc. or cast ingots or rolled slabs. Racks may be placed indivdually or in larger arrays formed by placing them end to end.

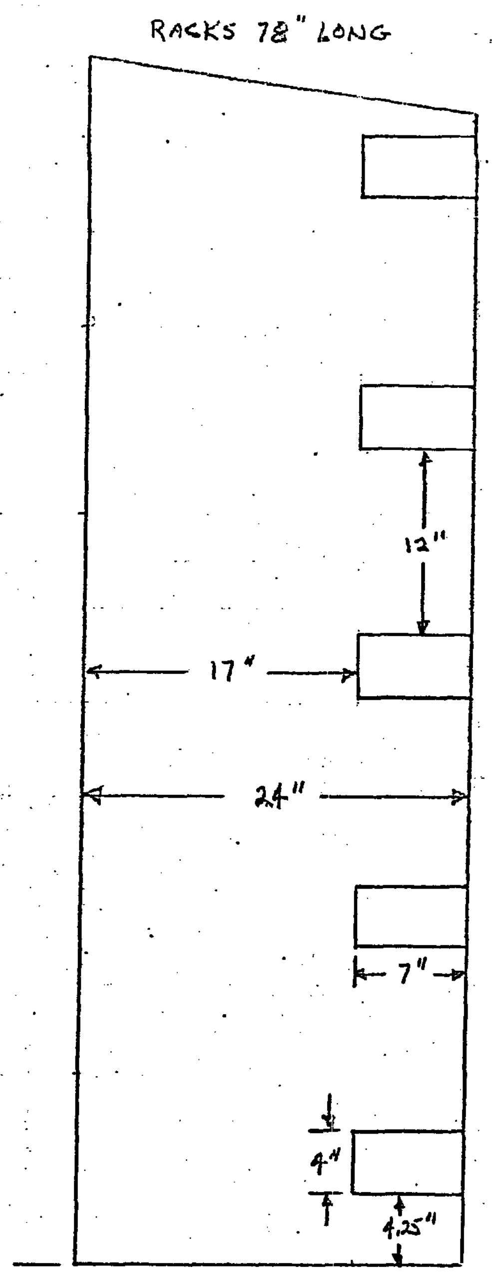
The small pieces will be held in metal tote boxes. These boxes are constructed of 1/16" aluminum with 3"x6"x16" inner dimensions. Large pieces (ingots or slabs) will be limited so that the thickness and width do not exceed 18 sq. inches (3"x6").

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 12" top to bottom separation between ports. Ports are long troughs running the entire length of the rack with a maximum 4" x 7" opening. Steel hasps with 16 gage steel plates welded to them are welded along the entire length of each opening to provide closure which ensure retention of the tote boxes or ingots or slabs. Racks are fastened to the flooring wall by bolting to powder actuated bolts.

Details of construction are shown on Sketch 813.1-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 solid angle subtended by the centermost port in an array does not exceed 2.0 steradians.



LICENSE: SNM-777; DOCKET: 70-8
SECTION: 800, SUBPART: 813.1
SKETCH 813.1 - I
PAGE 1 of 1
APPROVED
ISSUED: OCTOBER 31, 1968

I. DESCRIPTION

1. SNM stored in these racks will be small pieces in 3" x 6" x 16" tote boxes or will be ingots or slabs with 3" thickness and a 6" width.
2. Ports are spaced 12" vertically.
3. Racks contain 5 ports in a 1x5 arrangement.
4. Racks may be placed end-to-end to form long planar arrays.

II. NUCLEAR SAFETY OF INDIVIDUAL UNITS OR PORTS

Individual units are restricted to an 18 square inch cross sectional area. From Figure 309-IX, the maximum weight fraction for an 18 square inch cross sectional area is .48. Therefore, this is a safe cross section for materials up to and including a weight fraction of .48.

III. INTERACTION CALCULATIONS

The interaction of a long planar array formed by placing these racks end to end will be evaluated using the solid angle method. Since these racks may be placed end to end, assume an infinite length and thence sin e = 1.

$$d = \text{width of SNM} = 6'', \quad h = 12'' + \frac{3''(\text{thickness})}{2} = 13.5''$$

$$\Omega = \frac{2d}{h} \sin e = \frac{2d}{h} = \frac{12''}{13.5''} = .89$$

Since the centermost port sees one port above and one port below.

$$\Omega\ (\text{Total}) = 2 \times \Omega = 1.78 \text{ steradians}$$

From Fig. 309-XI, the maximum k eff for an unreflected safe cylinder of U-Al with a weight fraction of .48 is .578. Therefore

$$\text{Allowable } \Omega = 9-10\ k = 3.22 \text{ steradians}$$

IV. CONCLUSIONS

These racks are nuclearly safe.

LICENSE: SNM-777; DOCKET: 70-[illegible]
SECTION: 800, SUBPART: 813.1
Nuclear Safety Evaluation - High Enrichment Alloy Inprocess Storage
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ISSUED OCTOBER 31, 1968

SUPERSEDES NEW

LICENSE: SNM-777, Docket: 70-820
SECTION: 800 - FUEL FABRICATION OPERATION
Subsection: 810- Storage
Subpart: 814 - 50 H Storage
814.1 - U-AL Plate and Element Improcess Storage

814.1 U-AL PLATE AND ELEMENT INPROCESS STORAGE

1. Description

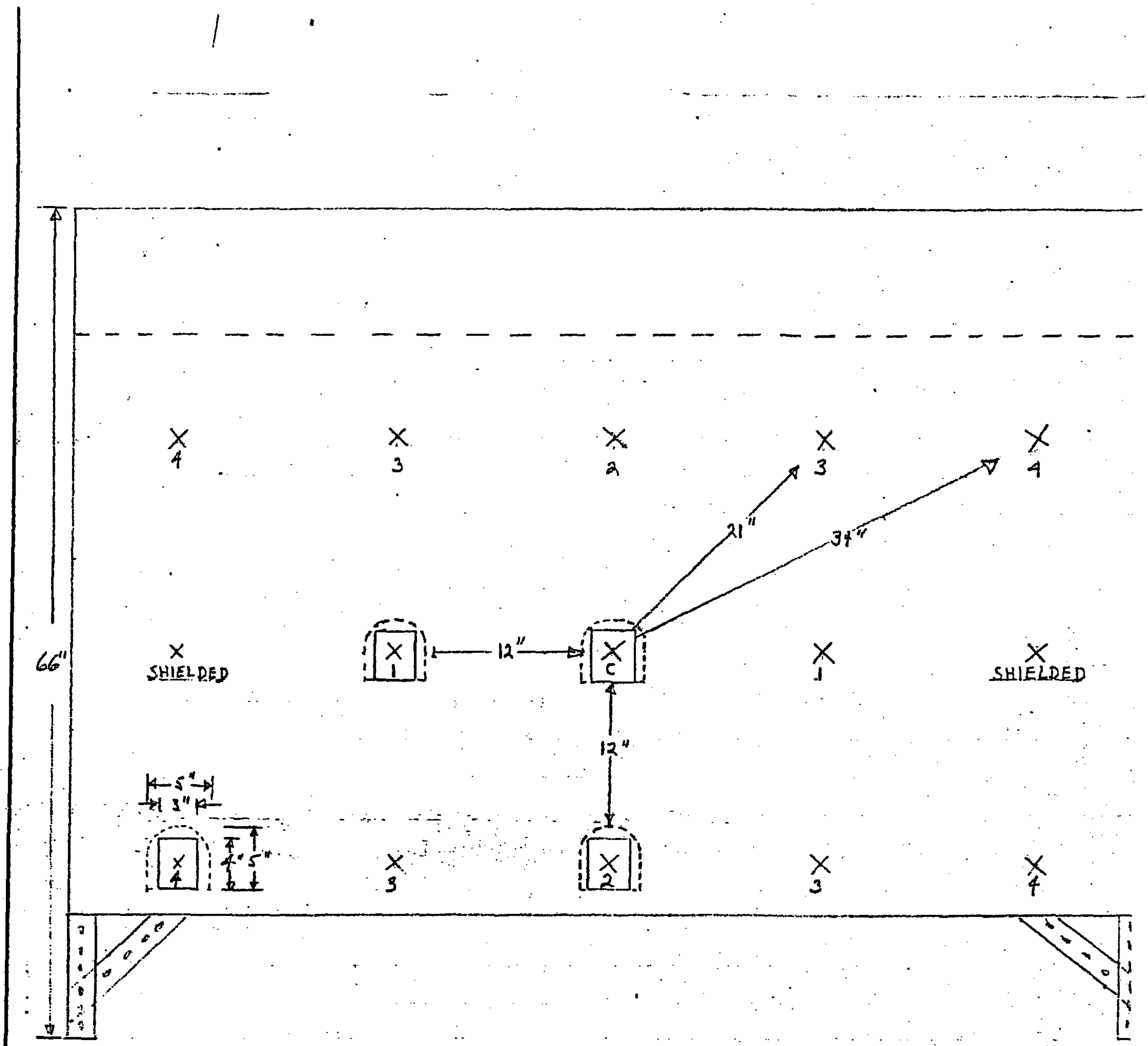
These racks will be used to store U-AL plates and elements during processing. These plates and elements may be in plastic wrappers. They will be placed stragetically throughout the facilities near work stations. Racks will be located individually or in groups placed side by side but will never be back to back.

Racks will be constructed of 3/16" slotted angle bolted together. The outside will be obvered with a thin metal (24 gage or heavier) with ports or openings cut out. Each port is constructed with a metal channel and has a restrictive opening in the rack shell such that the SNM is limited to a 3" x 4" stack of plates or 1 fuel element (3" x 3").

Details of construction are shown on Sketches 814.1-I and -II.

2. Nuclear Safety

These racks provide a safe cross section for the materials stored in each port. The solid angle subtended by the centermost port of an array of these racks placed side by side is less than 4.0 steradians.



LICENSE: SNM-777; DOCKET: 70-8

SECTION: 800, SUBPART: 814.1

U-Al PLATE & ELEMENT IN PROCESS STORAGE RACKS - FRONT VIEW
SKETCH 814.1 - I

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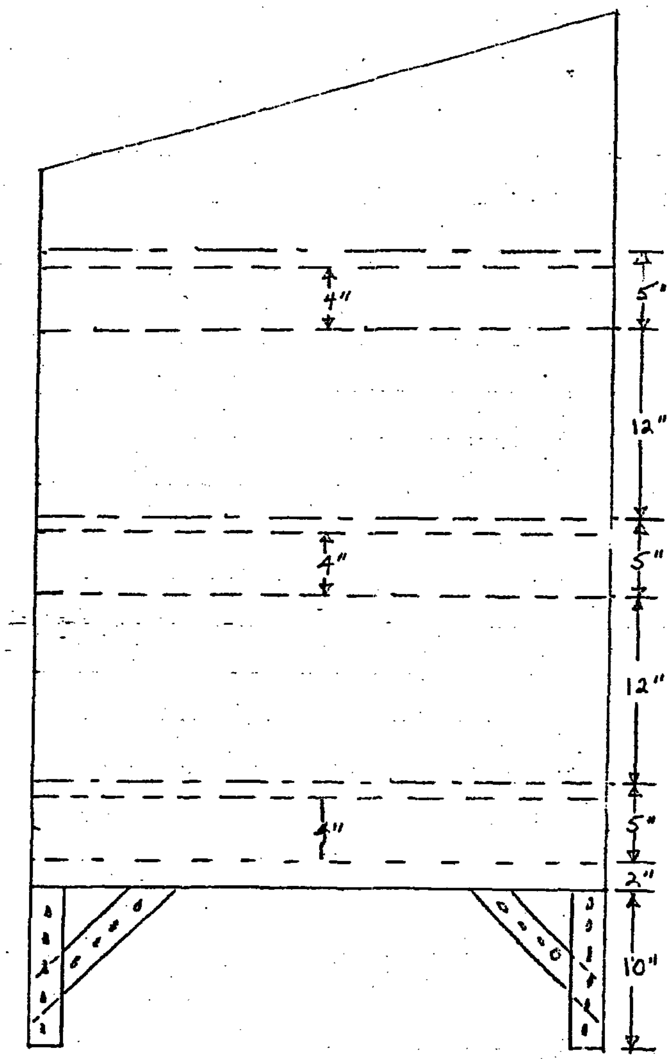
ISSUED: OCTOBER 31, 1968



LICENSE: SNM-777; DOCKET: 70-8
SECTION: 800, SUBPART: 814.1
U-Al PLATE & ELEMENT IN PROCESS STORAGE RACKS - SIDE VIEW SKETCH 814.1 - II
PAGE 1 of 1
APPROVED
ISSUED: OCTOBER 31, 1968

I. DESCRIPTION

1. SNM stored in these racks will be clad U-Al plates and elements.
2. Plates and elements may be in plastic wrappers. Therefore, optimum water moderation is assumed.
3. Ports (openings) will restrict the SNM to a 3"x4"x36" rectangle.
4. Ports are spaced 12" edge-to-edge.
5. Racks contain 15 ports in a 3 high x 5 wide arrangement.
6. Racks may be placed side by side but will never be back to back.

II. NUCLEAR SAFETY OF INDIVIDUAL UNITS OR PORTS

Individual ports have a maximum cross sectional opening of 12 square inches. From Figure 309-IX, the maximum weight fraction for a 12 square inch cross sectional area greater than .5. Therefore this is a safe cross section for materials up to and including a weight fraction of .5.

III. INTERACTION CALCULATIONS

The nuclear safety of the array formed by a group of these racks placed side by side will be evaluated using the k-Ω method. Ports in the same row as the centermost port will be considered shielded after considering the port in that row which is next to the centermost port.

1. Contribution from ports on each side of centermost port (Ports $\neq$1)

h = 36", d= height of SNM = 4", e-e = 12"

$$\lambda = \frac{h}{d} = \frac{36''}{4''} = 9, \qquad = \frac{e\text{-}e}{d} = \frac{12''}{4''} = 3$$

Ω_f = .03 from Fig. F-1.1, Y - 1272

$\Omega_7 = 4\pi\Omega_f$ x no. ports = 12.56 x .03 x 2 = .75 steradians

LICENSE: SNM-777; DOCKET: 70-8:
SECTION: 800, SUBPART: 814.1
Nuclear Safety Evaluation - U-Al Plate + Element Inprocess Storage Racks
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APPROVED
ISSUED: OCTOBER 31, 1968

2. Contribution from Ports Above and Below Centermost Port (Ports #2)

$h = 36''$, d = width of SNM = $3''$, e-e = $12''$

$$\lambda = \frac{h}{d} = \frac{36''}{3''} = 12, \quad \sigma = \frac{e\text{-}e}{d} = \frac{12''}{3''} = 4$$

$\bar{\Omega}_f = .023$ from Fig. F-1.1, Y-1272

$\Omega_2 = 4\pi \bar{\Omega}_f$ x no. ports = 12.56 x .023 x 2 = .58 steradians

3. Contribution from Next Nearest Ports (Ports #3)

$h = 36''$, d = diagonal = $\sqrt{3^2 + 4^2} = 5''$, e-e = $21''$

$$\lambda = \frac{h}{d} = \frac{36''}{5''} = 7.2, \quad \sigma = \frac{e\text{-}e}{d} = \frac{21''}{5''} = 4.2$$

$\Omega_c = .019$ from Fig. F-1.1, Y-1272

$\Omega_3 = 4\pi \bar{\Omega}_f$ x No. ports = 12.56 x .019 x 4 = .95 steradians

4. Contribution from Next Nearest Ports (Ports #4)

$h = 36''$, d = diagonal = $5''$, e-e = $3\,5''$

$$\lambda = \frac{h}{d} = \frac{36''}{5''} = 7.2, \quad \sigma = \frac{e\text{-}e}{d} = \frac{35''}{5''} = 7$$

f = .0082 from Fig. F-1.1, Y - 1272

$\Omega_4 = 4\pi \bar{\Omega}_f$ x no. ports = 12.56 x .0082 x 4 = .41 steradians

5. Contribution from the Closest Ports in the Next Racks (Ports #5)

$h = 36''$, d = diagonal = $5''$, e-e = $51''$

$$\lambda = \frac{h}{d} = \frac{36''}{5''} = 7.2, \quad \sigma = \frac{e\text{-}e}{d} = \frac{51''}{5''} = 10.2$$

$\bar{\Omega}_f = .0045$ from Fig. F-1.1, Y-1272

$\Omega_5 = 4\pi \bar{\Omega}_f$ x no. Ports

= 12.56 x .0045 x 4 = .23 steradians

6. Contribution fron Next Nearest Ports in Next Racks (Ports #6)

$h = 36''$, d = diagonal = $5''$, e-e = $67''$

$$\lambda = \frac{h}{d} = \frac{36''}{5''} = 7.2, \quad \sigma = \frac{e\text{-}e}{d} = \frac{67''}{5} = 13.4$$

$\bar{\Omega}_f$ = .003 from Fig. F-1.1, Y-1272

$\Omega_6 = 4\pi \bar{\Omega}_f$ x no. ports = 12.56 x .003 x 4 = .15 steradians

7. Contribution from Next Nearest Ports in Next Racks (Ports #7)

$h = 36''$, d = diagonal = $5''$, e-e = $84''$

$$\lambda = \frac{h}{d} = \frac{36''}{5} = 7.2, \quad \sigma = \frac{e\text{-}e}{d} = \frac{84''}{5''} = 16.8$$

$\bar{\Omega}_f$ = .002 from Fig. F-1.1, Y-1272

$\Omega_7 = 4\pi \bar{\Omega}_f$ x no. ports = 12.56 x .002 x 4 = .10 steradians

8. Contribution from Next Nearest Ports in Next Racks (Ports #8)

$h = 36''$, d = diagonal = $5''$, e-e = $100.5''$

$$\lambda = \frac{h}{d} = \frac{36''}{5''} = 7.2, \quad \sigma = \frac{e\text{-}e}{d} = \frac{100.5''}{5''} = 20.1$$

$\bar{\Omega}_f$ = .0014 from Fig. F-1.1, Y-1272

$\Omega_8 = 4\pi \bar{\Omega}_f$ x no. ports = 12.56 x .0014 x 4 = .07 steradians

9. Contribution from Next Nearest Ports in Next Racks (Ports #9)

$h = 36''$, d = diagonal = $5''$, e-e = $117.5''$

$$\lambda = \frac{h}{d} = \frac{36''}{5''} = 7.2, \quad \sigma = \frac{e\text{-}e}{d} = \frac{117.5''}{5''} = 23.5$$

$\bar{\Omega}_f$ = .0012 from Fig. F-1.1, Y-1272

$\Omega_9 = 4\pi \bar{\Omega}_f$ x no. ports

= 12.56 x .0012 x 4 = .06 steradians

10. Contribution from other Units

It is assumed that all other ports in any racks aligned with this array or other surrounding equipment which is greater than 4 feet away from the rack with the centermost unit will contribute less than 0.7 steradians. So

$$\Omega_{other} = 0.7 \text{ steradians}$$

11. Total Interaction

$$\Omega_T = \Omega_1 + \Omega_2 + \Omega_3 + \Omega_4 + \Omega_5 + \Omega_6 + \Omega_7 + \Omega_8 + \Omega_9 + \Omega_{other}$$

$$\approx 4.0 \text{ steradians}$$

12. Allowable Interaction

From Fig. 309-XI, the maximum k eff for an unreflected cylinder of U-Al with a weight fraction of approximately .5 is .574. From the nuclear safety study for the highest loaded fuel elements (a BAWTR fuel element with an active fuel cross sectional area not exceeding 7.1 square inches), the k eff for a single unreflected fuel element will be approximately .307 (Ref. Nuclear Safety Evaluation Subpart 822). Therefore, it is conservative to estimate that the maximum k eff would be .5. Therefore,

$$\text{Allowable } \Omega = 9-10 \text{ K} = 9-5.0 = \text{steradians} = 4 \text{ steradians.}$$

IV. CONCLUSIONS

These racks are nuclearly safe.

LICENSE: SNM-777, Docket: 70-820
SECTION: 800- FUEL FABRICATION OPERATION
Subsection: 810 - Storage
Subpart: 814 - 50 H Storage
814.2 - U-AL Plate and Element Storage Racks

Approvad

ISSUED October 31, 1968

SUPERSEDES New

814.2 U-AL Plate and Element Storage Racks

1. Description

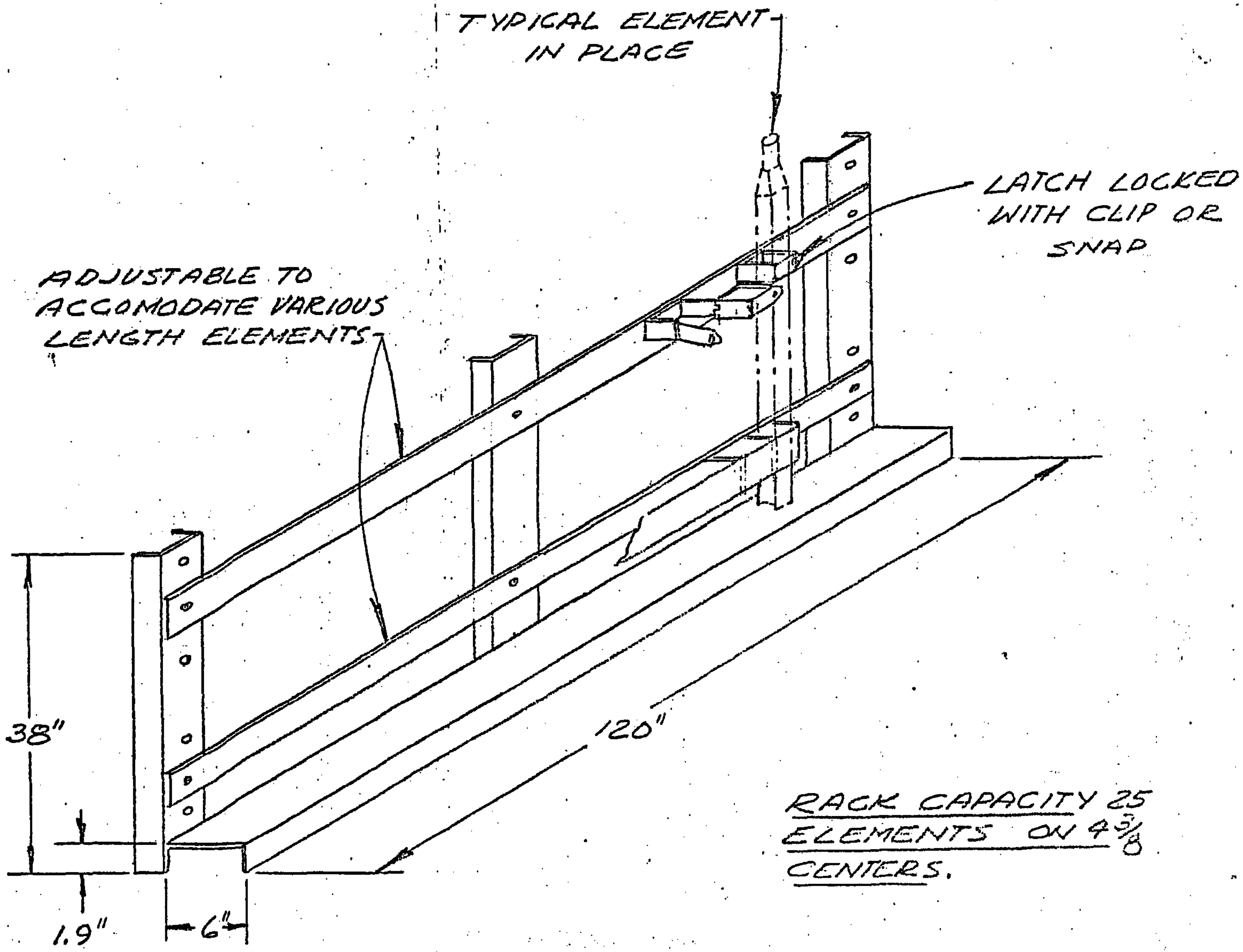
These racks will be used for the storage of U-AL plates and fuel elements usually after the completion of their processing. These racks will be located in the Finished Plate and Element Storage Area. Other racks of this type may be used as inprocess storage. They will be arranged so that they are separated at least 3 feet from any equipment with SNM in or on it. Racks may be placed end to end to form long in line arrays of fuel elements. Racks also may be placed side by side if isolated as per Subpart 303.

Racks will be constructed to form 4"x4" brackets, each fuel element being held by an upper and lower bracket. Fuel elements are spaced on 4-3/8" centers forming an in line array: Brackets are 3/32" x 1-1/2" steel. The upper brackets use a steel hasp in the front to allow easy insertion and removal of fuel elements. Brackets will be plug welded to 4" x .320" x 1.720" x 38" long steel channels which are welded to 6" x .200" x 1.920" x 120" long inverted steel channels which form the base.

Details of construction are shown in Sketch 814.2-I.

2. Nuclear Safety

Under accident conditions of optimum water moderation and complete water reflection, one of these racks will have a maximum effective multiplication factor of 0.919. In line arrays will be isolated as per Subpart 303.



RACK CAPACITY 25 ELEMENTS ON 4 5/8 CENTERS.

LICENSE: SNM-777; DOCKET: 70-3

SECTION: 800, SUBPART: 814.2

SKETCH 814.2-1

PAGE 1 of 1

APPROVAL

ISSUED: OCTOBER 31, 1968

I. DESCRIPTION

1. Only U-AL fuel plates and elements will be stored in these racks.
2. BAWTR fuel elements will be the most reactive material stored.
3. Racks form an inline, planar array infinitely long. So, $B_y^2 = 0$.
4. Fuel elements will be nominally 2.4" x 2.9" x 30" (active fuel length) = 6.1 cm x 7.4 cm x 76 cm.
5. Use the B & W reactor design parameters contained in Subpart 822.

$$K_\infty = 1.842 \quad ; \quad M^2 = 40.28 \text{ cm}^2$$

II. NUCLEAR SAFETY

Calculating the geometric buckling of the array

$$B_g^2 = B_x^2 + B_y^2 + B_z^2 = \frac{\pi^2}{(\text{element width} + 2\delta)^2} + 0 + \frac{\pi^2}{(\text{active fuel length} + 2\text{:}}$$

Where δ = 6.5 cm from Fig. 3, TID-7028

$$B_g^2 = \frac{9.87}{(7.4 + 13)^2} + 0 + \frac{9.87}{(76 + 13)^2} = \frac{9.87}{(20.4)^2} + \frac{9.87}{(84)^2} = \frac{9.87}{416} + \frac{9.87}{7921}$$

$$= .0237 + .0012 = .0249 \text{ cm}^{-2}$$

Calculating the effective multiplication factor of the array

$$K_{eff} = \frac{K_\infty}{1 + M^2B^2} = \frac{1.84}{1 + (40.28 \times .0249)} = \frac{1.84}{2.003} = .919$$

III. CONCLUSIONS

The storage arrangement will be subcritical even after accidental optimum water moderation and complete water reflection. If lesser loaded (less reactive) fuel elements are stored, the effective multiplication factor will decrease and the storage arrangement will be safer.

LICENSE: SNM-777; DOCKET: 70-8:
SECTION: 800, SUBPART: 814.2
Nuclear Safety Evaluation - U-AL Finished Element Storage
PAGE 1 of 1
APPROVED
ISSUED: OCTOBER 31, 1968

LICENSE: SNM-777, Docket: 70-820
SECTION: 800 - FUEL FABRICATION OPERATION
Subsection: 810 - Storage
Subpart: 814 - 50 H Storage
814.3 - Low Enrichment UO_2 Rod Inprocess Storage

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ISSUED October 31, 1968

SUPERSEDES New

814.3 Low Enrichment UO_2 Rod Inprocess Storage

I. Description

These racks will be used to store UO_2 rods not exceeding 5% enriched during processing. They will be placed individually or in groups formed by placing racks end-toend but never side by side.

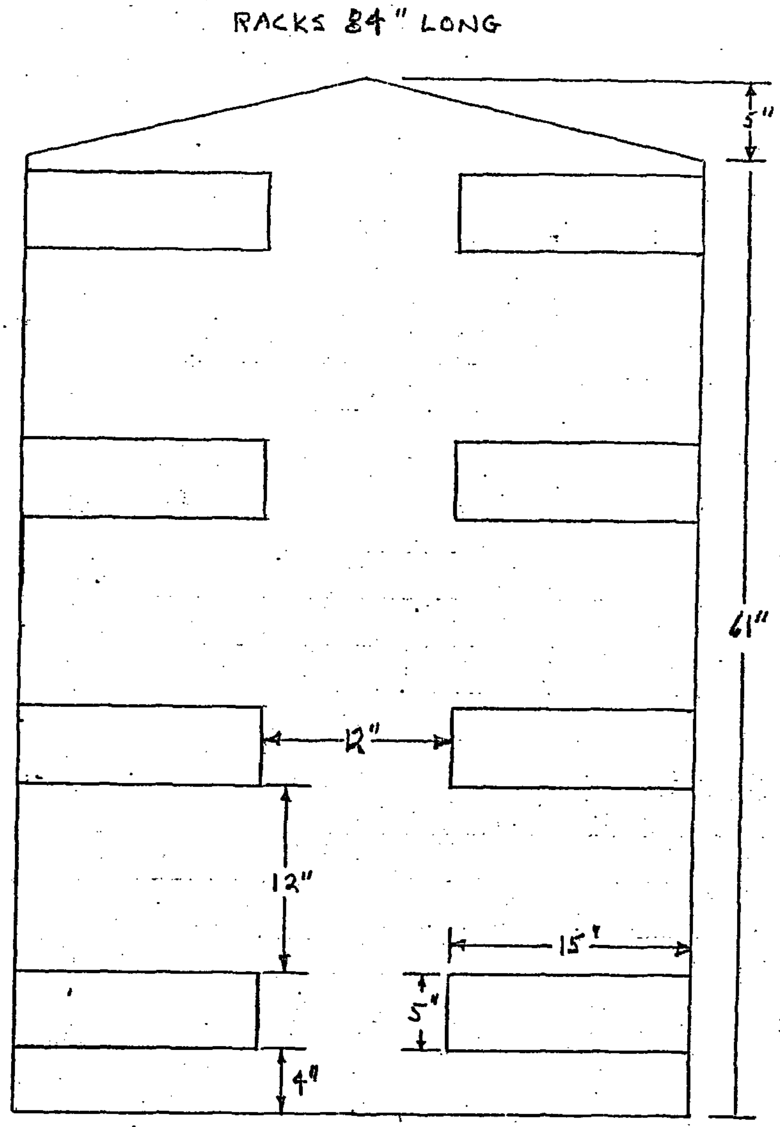
Rods are contained in modules for ease in handling. These modules are open end troughs or channels made of 1/16" or 16 gage stainless steel. 3/8" diameter drain holes are placed 4" apart approximately 9/16" above the module bottom along the entire length of the module. 16 gage safety bars or lids may be placed over the top of the module for product protection. Modules have 1-1/2"x 2-3/4" x various length inside dimensions.

Racks are constructed of 1" x 1" x 1/8" frame steel angle welded together. The outside is covered with at least 20 gage steel to provide 12" edge-to-edge and top to bottom separation between ports. Ports are long troughs running the entire length of the rack with a maximum 5" x 15" 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 modules and tubes. These racks may form one or two planer arrays.

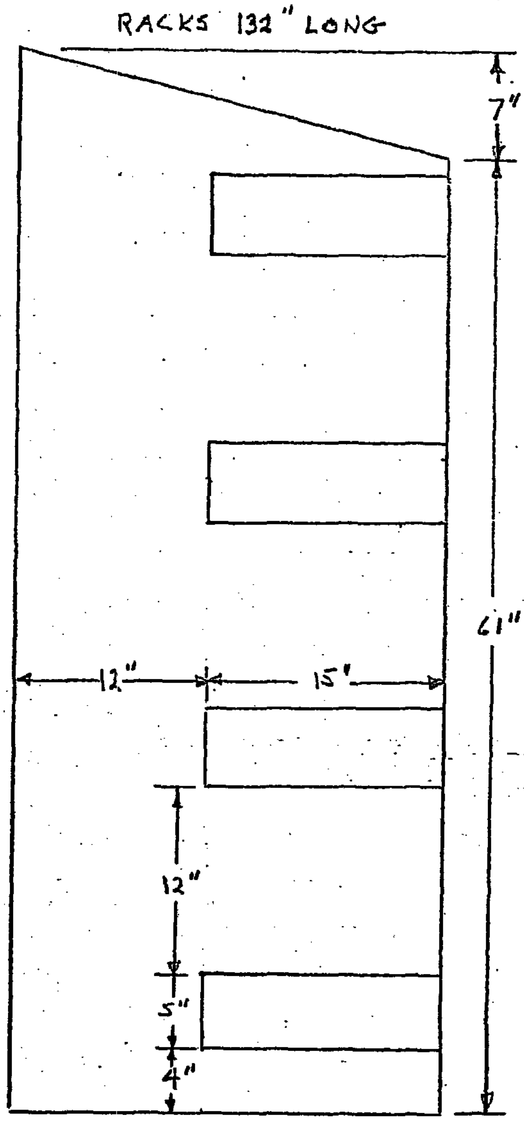
Details of construction are shown on Sketch 814.3 - I and - II.

2. Nuclear Safety

Under normal conditions, these racks are unmoderated and unreflected and are nuclearly safe. They are also safe under accident conditions of optimum water moderation and complete water reflection. The ports or openings are safe when filled with uranium not exceeding 2.8% enrichment. The storage of higher enrichments require reduced opening sizes so the material is safe under accident conditions. Details of their nuclear safety are set forth in Nuclear Safety Evaluation 814.3.



LICENSE: SNM-777; DOCKET: 70-8
SECTION: 800, SUBPART: 814.3
UO_2 ROD IN PROCESS STORAGE RAC DOUBLE RACKS - SKETCH 814.3-I
PAGE 1 of 1
APPROVED
ISSUED: OCTOBER 31, 196?



LICENSE: SNM-777; DOCKET: 70-8

SECTION: 800, SUBPART: 814.3

UO_2 ROD IN PROCESS STORAGE RAC

SKETCH: 814.3-II

PAGE 1 of 1

APPROVED

ISSUED: OCTOBER 31, 1968

I. DESCRIPTION

1. Rods will be stored in metal modules.
2. Up to 15 modules (3x5) may be stored in each port opening.
3. Ports are separated 12 inches edge-to-ege.
4. There will be <u>no</u> hydrogeneous material (e.g., paper, polyethylene, etc.) with the rods or modules in each port.
5. SNM will be restricted to UO_2 pellets with enrichments not exceeding 5.0% which are encased in metal (usually zircaloy or stainless steel) to form rods.
6. Racks may be placed end-to-end to form planar arrays.
7. Individual racks or planar arrays formed by racks placed end-to-end will be separated at least 3 feet side-by-side.

II. NUCLEAR SAFETY OF INDIVIDUAL PORTS

Under normal conditions, there will be no moderating material in the ports with the SNM. As indicated on page 10, TID-7028, "unmoderated uranium cannot become critical if the U-235 content is below 5 or 6 wt. %".

Each port opening has a cross sectional area of

$$A = 5'' \times 15'' = 75 \text{ sq. in.}$$

This cross section corresponds to a cylinder diameter of 9.78 inches.

$$A = 75 \text{ in}^2 = \frac{\pi d^2}{4}, \quad d = \sqrt{\frac{4A}{\pi}} = \sqrt{\frac{300}{\pi}} = \sqrt{95.5} = 9.78''$$

This is a safe diameter for enrichments up to and including 2.8% as shown in Figure 309-XII. Therefore, the SNM in a port would be safe even if accidentally flooded to cause optimum water moderation and complete water reflection. If higher enrichments are to be stored, the equivalent diameter will be reduced using Figure 309-XII.

III. INTERACTION CALCULATIONS

Under normal conditions, there will be no moderating material in the ports with the SNM or between the ports. As stated in II above, unmoderated material with enrichments less than 5% cannot become critical.

If accidental flooding should occur, individual ports will be separated by 12 inches of water. Twelve inches of water gives effective neutron isolation as indicated in TID-7016, Rev. 1. Since, the modules, ports and racks have many openings, there is no place for water retention. So, all water would "run out" of the modules, ports, and racks as the flooding subsided. Also, the rack construction is such that in the event of a water spray (eg., sprinkler release, water pipe rupture, etc.), it is considered unlikely that water would enter the modules, ports or racks. Therefore, these racks are safe under accident conditions.

IV. CONCLUSIONS

These racks are safe under normal or accident conditions.

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LICENSE: SNM-777, Docket: 70-820 SECTION: 800 - FUEL FABRICATION OPERATION Subsection: 810 - Storage Subpart: 814 - Clad SNM Storage 814.4 - Low Enrichment UO_2 Finished Component Storage	 Approved ISSUED OCTOBER 31, 1968 SUPERSEDES NEW

814.4 Low Enrichment UO_2 Finished Component Storage

1. Description

Finished UO_2 components will be stored along the north wall of Building 50H. The rack contains spaces for 16 components on 18" centers. This arrangement consists of hanging components from 2" x 1-1/4" steel bars which are welded on 18" centers to an 8" x 8" wide flange H beam. The H beam is supported from the floor by six 5" Schedule 40 steel pipes and secured through the cinder block wall in six locations. An adjustable clamping bar contains each component at its bottom fitting to prevent it from coming into contact with adjacent fuel elements.

Structural calculations indicate that the storage arrangement would support a total of 13,600 lbs. or 16 components weighing 850 lbs.

Details of this storage arrangement are shown on Sketch 814.4-I.

2. Nuclear Safety

Individual components will be nuclearly safe by themselves. The solid angle subtended by the centermost component in the array when one other component is in transit is

Type of Component	Solid Angle
Dresden	1.35 steradians
Yankee	2.4 steradians

No more than one (1) component will be permitted to be in transit at any one time. Other SNM bearing material in the general vicinity of this rack will be spaced at least four (4) feet from the rack.

LICENSE: SNM-777; DOCKET: 70-3

SECTION: 800, SUBPART: 813.4

UO_2 FINISHED COMPONENT STORAGE

SKETCH 814.4-I

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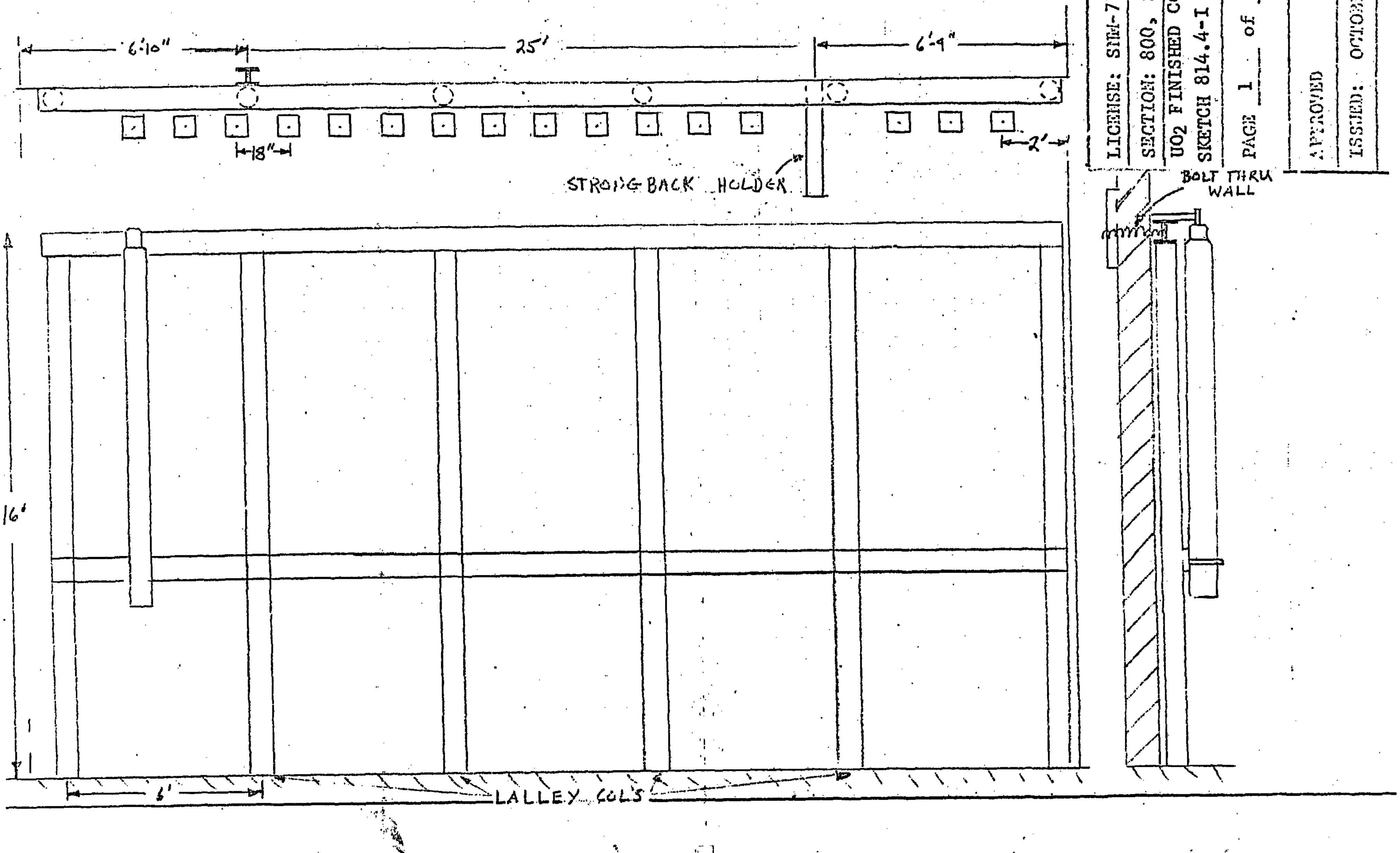
APPROVED

ISSUED: OCTOBER 31, 1968



I. DESCRIPTION

1. Components will be stored on 18" centers.

2. No more than 1 component will be permitted to be in transit at any one time.

3. Other SNM bearing material in the general vicinity of the rack will be spaced at least 4 feet from the rack.

II. NUCLEAR SAFETY OF INDIVIDUAL UNITS OR COMPONENTS

Individual components will be subscritical with optimum possible water moderation and complete water reflection. A summary of components and their effective multiplication factors is listed below:

Component Type	Keff	Reference
Dresden	.591	Nuclear Safety Eval.-Dresden I.F.E.-Subsection 823
Yankee	.852	Nuclear Safety Eval.-Yankee F.E.-Subsection 823

III. INTERACTION CALCULATIONS

The nuclear safety of planar array formed by the 16 components in a line will be evaluated using the K - Ω method.

1. Dresden Fuel Elements

h = active fuel length = 108.25", d = side dimension = 4.38", e-e = 18" - 4.38" = 13.62"

$$\lambda = \frac{h}{d} = \frac{108.25''}{4.38''} = 24.7, \quad \sigma = \frac{e\text{-}e}{d} = \frac{13.62''}{4.38''} = 3.1$$

$$\bar{\Omega}_f = .035 \text{ from Fig. F-1.1, } Y\text{-1272}$$

Since the rack forms a planar array, the centermost element sees only two other elements in the array--the ones on each side of it. Assuming one element in transit at the same spacing, the centermost unit sees three elements.

$$\Omega_T = 4\pi\bar{\Omega}_f \times \text{number elements}$$

$$= 12.56 \times .035 \times 3 = 1.35 \text{ steradians}$$

III. INTERACTION CALCULATIONS

1. Dresden Fuel Elements (Continued)

From the Nuclear Safety Evaluation - Dresden I Fuel Elements (Subsection 823), the effective multiplication factor for one water moderated, unreflected element is

$K_{eff} = 0.306$

Therefore,

Allowable $\Omega = 9-10K = 9-3.06 = 5.94$ steradians

This rack is safe for these components.

2. Yankee Fuel Elements

h = active fuel length - 91", d = side dimension = 7.615", e-e = 18" - 7.615" = 10.385"

$$\lambda = \frac{h}{d} = \frac{91''}{7.615''} = 12, \quad \sigma = \frac{e-e}{d} = \frac{10.385''}{7.615''} = 1.4$$

$\bar{\Omega}_f = .063$ from Fig. F-1.1, γ-1272

Since the rack forms a planar array, the centermost element sees only two other elements in the array--the ones on each side of it. Assuming one element in transit at the same spacing, the centermost unit sees three elements.

$\Omega_T = 4\pi\ \bar{\Omega}_f \times$ no. units $= 12.56 \times .063 \times 3 = 2.4$ steradians

From the Nuclear Safety Evaluation - Yankee Fuel Elements (subsection 823) the effective multiplication factor for one water moderated, unreflected element is

$K_{eff} = 0.595$

Therefore,

Allowable $\Omega = 9-10K = 9-5.95 = 3.05$ steradians

This rack is safe for these components.

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815.1 Low Enrichment UO_2 Pellet Storage

1. Description

UO_2 pellets of enrichment up to 4.1% will be stored in packages as prescribed for shipping containers in single layers on wooden pallets. The pallets will be stored in single layers on steel shelves in the Building 41-H storage area.

2. Nuclear Safety

The analyses described in NED-1852 and NED-2031 show the fully-loaded storage array to be safe under conditions of optimum moderation (maximum k_{eff} = 0.803). These reports also depict the storage geometry.

GULF UNITED
NUCLEAR FUELS CORPORATION

INTER-OFFICE MEMO

NED-2031

TO R. Kropp AT

DATE October 13, 1972

FROM J. H. Ray AT

COPY TO J. R. Tomonto
S. Johnson
P. Loysen

SUBJECT Revised Criticality Safety Analysis Of the Fabrication Operation 41-H Storage Area (5168-1311)

The original analysis to determine the criticality safety of the Fabrication Operation 41-H storage area when loaded with 4.1 w/o U^{235} pellet packages was reported in NED-1852. At the time that analysis was undertaken, the thickness of the steel storage shelves was not known, and the thickness of 0.375 inch was assumed. Since then, a thickness of 0.108 inch has been established for the shelves.

In the revised analysis, the most reactive configuration, that with six shelves, was investigated as a function of the amount of airborne water which might result from operation of the automatic sprinkler system. The configuration is shown in Figure 1. In all respects except the steel shelf thickness, the analysis is identical to that described in NED-1852.

The results of the revised analysis are shown in Table 1 and Figure 2 along with the results of the original analysis. The maximum calculated k_{eff} of the area with six axial storage locations was 0.803 ± 0.012 at the 95% confidence level. The k_{eff} values for the thin steel shelves are all higher than those calculated for the thick shelves, and the difference is roughly constant for airborne water densities from 0.05 to 0.15 gm/cm^3. This effect is probably due to the larger low-energy neutron absorption of the thick steel shelves. The smaller difference in k_{eff} with zero airborne water is also not entirely unexpected since steel is almost transparent to middle-energy neutrons and, because of inelastic scattering, is something of a moderator for high-energy neutrons.

James H. Ray
J. H. Ray

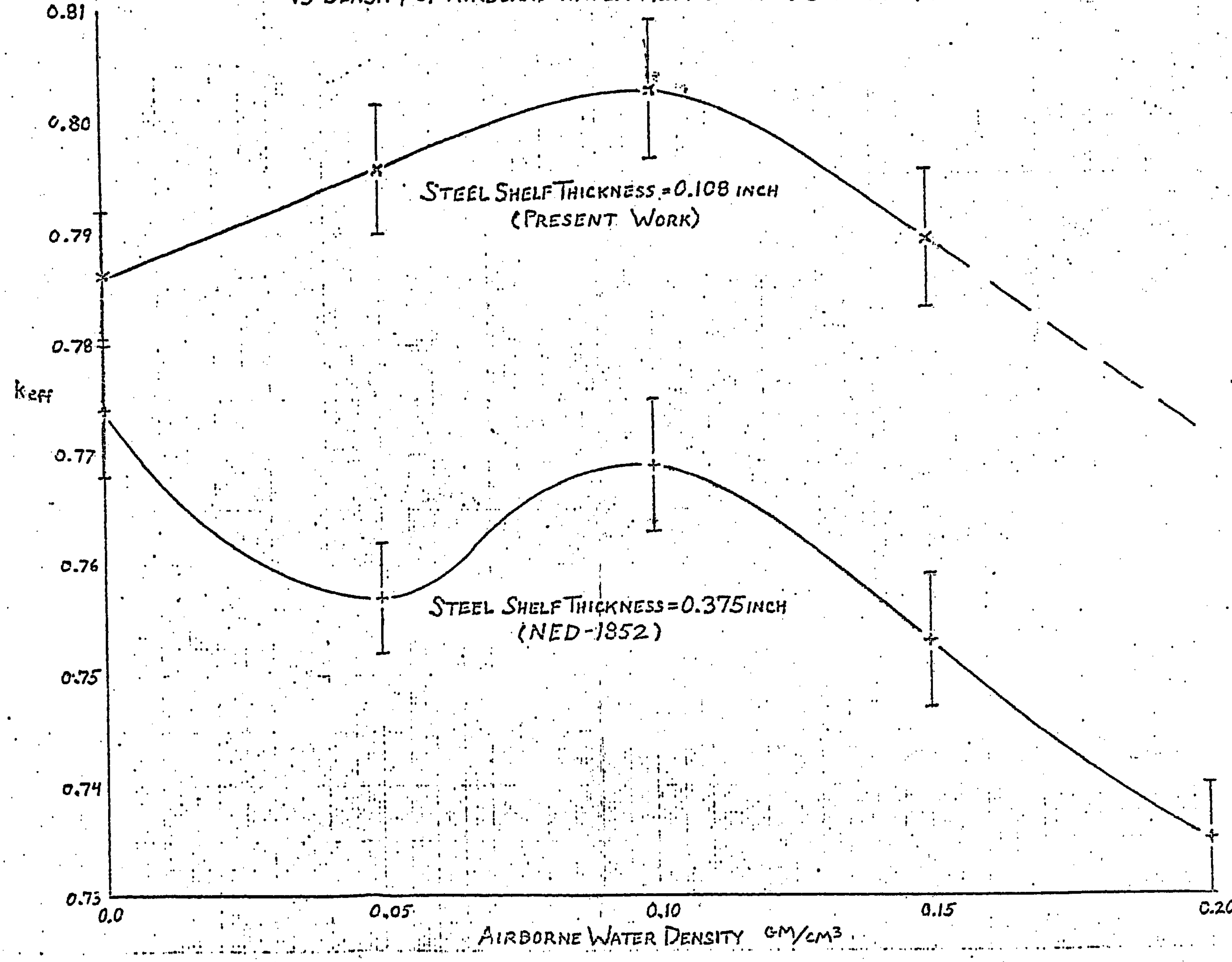
JHR/cc
Attachments

TABLE 1

KENO Calculated K_{eff} For The New Haven 41-H
Storage Area With 4.1 w/o U^{235} Fuel Pellet Packages
In Six Axial Storage Locations

Airborne Water Density gm/cm^3	Calculated K_{eff} ± 1 Standard Deviation NED-1852 0.375" Shelves	Present Work 0.108" Shelves
0.0	0.774± 0.006	0.786± 0.006
0.05	0.757± 0.005	0.796± 0.006
0.10	0.769± 0.006	0.803± 0.006
0.15	0.753± 0.006	0.789± 0.006
0.20	0.735± 0.005	

FIGURE 2. KENO CALCULATED k_{eff} OF 41-H STORAGE AREA WITH 6 AXIAL SHELVES VS DENSITY OF AIRBORNE WATER FROM OVERHEAD SPRINKLER SYSTEM



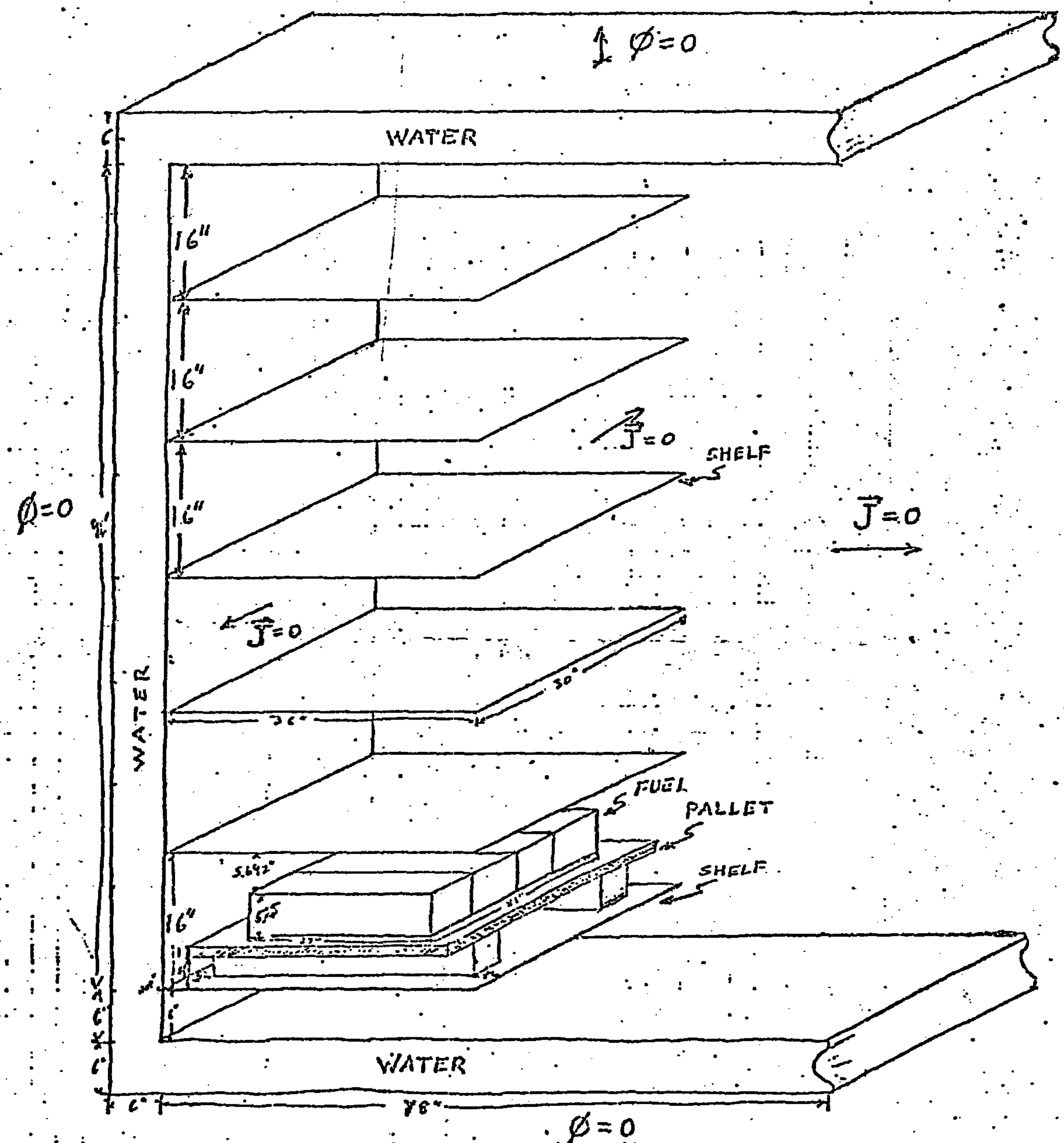


FIGURE I. KENO CALCULATION MODEL OF 41-H STORAGE AREA WITH SIX SHELVES. NEUTRON BOUNDARY CONDITIONS ARE INDICATED.

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INTER-OFFICE MEMO

NED-1852

TO R. Kropp AT DATE May 22, 1972

FROM E. Fass AT COPY TO J. R. Tomonto
R. Carlson
Miles Wittner

SUBJECT Criticality Safety Analysis of the New Haven 41-H Storage Area (5168-6053)

1.0 Summary and Conclusions

An analysis has been performed to determine the criticality safety of the New Haven 41-H storage area when loaded with 4.1 w/o U^{235} Yankee fuel pellet packages. The study considered the criticality condition of the area as a function of:

1. The number of shelves of fuel packages which can be stacked against the walls,
2. The amount of airborne water which might result from operation of the overhead automatic sprinkler system.

A maximum of six shelves can be constructed and still leave room for the automatic handling equipment to move pallets containing fuel packages on and off the shelving. The maximum calculated k_{eff} of the area for the condition of six storage locations in the vertical direction was 0.774 ± 0.013, at the 95% confidence level.

2.0 Description of the 41-H Storage Area

The storage area arrangement is shown on Figure 1. It is composed of two concrete walls whose opposing faces each support a number of storage shelves. The steel shelves run the full length of the room on each wall but may be varied from four to six up the height of the wall.

At the time this analysis was undertaken, the thickness of the steel shelves had not been established so a 3/8" thickness was assumed in the calculations. The actual dimensions of the shipping pallet upon which the pellet packages would be placed were also not fully established. The width and breadth of the pallet are each 30" but the height was assumed to be 5" based on the height of a standard pallet. The equivalent thickness of the standard pallet is 1" of wood and this was used in the present analysis.

The packages of fuel pellets are those which are designed for shipment in the Gulf United Type 2901 shipping container(1). To make up a pellet package the UO_2 pellets are placed into the grooves of corrogated, polyethylene coated, steel trays. The trays are stacked one one top of the other until 60 pounds of the oxide fuel is contained in the stack. The stack of trays is wrapped with a sheet of gum rubber, sandwiched between two 1/2" hard maple boads and taped together. The overall dimensions of the pellet package are 5'/4" x 5'/4" x 26" and four such packages are placed side by side on a pallet. As such, each pallet contains 240 pounds of oxide in a 5'/4" x 21" x 26" configuration. The materials of the pellet package provide an H/U^{235} ratio of approximately 108.

3.0 Method of Analysis

Reactivity levels for the storage area were calculated with the KENO multigroup Monte Carlo criticality code(2). The 16 group Hansen-Roach cross sections(3) were used with P_1 linear anisotropic hydrogen scattering.

The calculational model for KENO assumed a homogeneous region for the fuel pellet package. However, the reactivity of the package is dependent on the moderating ratio and pellet dimensions because of the U^{238} resonance captures. Therefore, the homogenization procedure for the package must take account for this effect by an appropriate choice of the U^{238} cross section set. A previous analysis of the reactivity levels of Yankee fuel rods in water(4) has yielded the effective resonance integral and resonance escape probability as a function of the moderation ratio. For H/U^{235}= 108 the resonance data yields an effective potential scattering cross section per resonance absorber atom, i.e. σ_p, of 104 barns. The U^{238} cross section set used in the KENO analysis was conservatively selected as σ_p = 100 barns.

The KENO program has been widely used to determine the reactivity levels of similar types of systems, see for example references 5 and 6.

4.0 Nuclear Safety Evaluation of the Storage Area

The input description to the KENO program included several conservative assumptions regarding the composition of materials and geometry of the 41-H storage area. First, it was assumed that the shelves and walls were linearly infinite in extent; this makes it possible to define unit cell boundaries in the x-y plane with neutron mirror reflection boundary conditions. Second, the concrete walls and concrete floor were replaced by 15 cm thick water slabs. Since the nature of any proposed roofing material over the storage area was still uncertain, the third assumption involved the placing of a 15 cm thick water slab over the top of the storage area. Figure 2 illustrates the 41-H storage area as modeled in KENO.

The first KENO problem considered the case of four shelves up the side of each wall. The program was run for 14,100 histories and calculated a $k_{eff} = 0.636 \pm 0.012$. Since there is an economic incentive to increase the number of storage locations, and since this k_{eff} is so low, the number of shelves up each wall was increased. The calculated k_{eff}'s for five and six axial locations were 0.712 ± 0.011 and 0.774 ± 0.013, respectively. Six shelves high is the maximum which can be build without encountering handling problems in getting the pallets on and off the shelves; therefore, the calculations were discontinued at this point. Table 1 summarizes the KENO calculations.

The storage area, with six axial storage shelves, was examined as a function of the density of airborne water which might result from the operation of the overhead automatic sprinkler system. Since the exact density of the water moderation could not be determined for this system, a series of KENO calculations were performed as a function of airborne water density to see if an optimum existed. The water density was increased in 0.05 gm/cm^3 steps up to a density of 0.20 gm/cm^3. The results of these calculations are shown in Table 1 and Figure 3. It is seen that the maximum array reactivity occurs with no airborne water, and that, although there is an increase in the array reactivity after an initial decrease, it it does not exceed the value at zero water density.

E. Fass

E. Fass

EF/ah
Attachments

Table 1

Results of KENO Calculated k_{eff} for the New Haven 41-H Storage Area with 4.1 w/o U^{235} Yankee Fuel Pellet Packages

Number of Axial Storage Locations	Airborne Water Density, gm/cm^3	Calculated k_{eff} ± 1 Standard Deviation
4	0.0	0.636 ± 0.006
5	0.0	0.712 ± 0.006
6	0.0	0.774 ± 0.006
6	0.05	0.757 ± 0.005
6	0.10	0.769 ± 0.006
6	0.15	0.753 ± 0.006
6	0.20	0.735 ± 0.005

References

1. Gulf United Nuclear Fuel Corporation, Drawing Number 5008-8192, Revision 1.

2. Whitesides, G.E., and Cross N.F., "KENO-A Multigroup Monte Carlo Criticality Program," CTC-5, Union Carbide Corporation, Nuclear Division, Computing Technology Center, Oak Ridge, Tennessee, (September 1969).

3. Hansen, G.E., and Roach W.H., "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," LAMS-2543, Los Alamos Scientific Laboratory, (December 1960).

4. Fass, E., "Criticality of Yankee Fuel Rods During Pickling and Corrosion," Gulf United Nuclear Fuel Corporation, GUNFC Memo NED-1083, (February 1971).

5. Crume, E.C., "Some Considerations in Regard to the Development and Use of the KENO Program," CONF-680909, <u>Proceeding of the Livermore Array Symposium</u>, Lawrence Radiation Laboratory, pp 18-22 (September 1968).

6. Handley, G. R., "Effect of Water Sprinklers on Array Criticality Safety Analysis," Transactions of the American Nuclear Society, Volume 13 Number 2, pp 665-666 (November 1970).

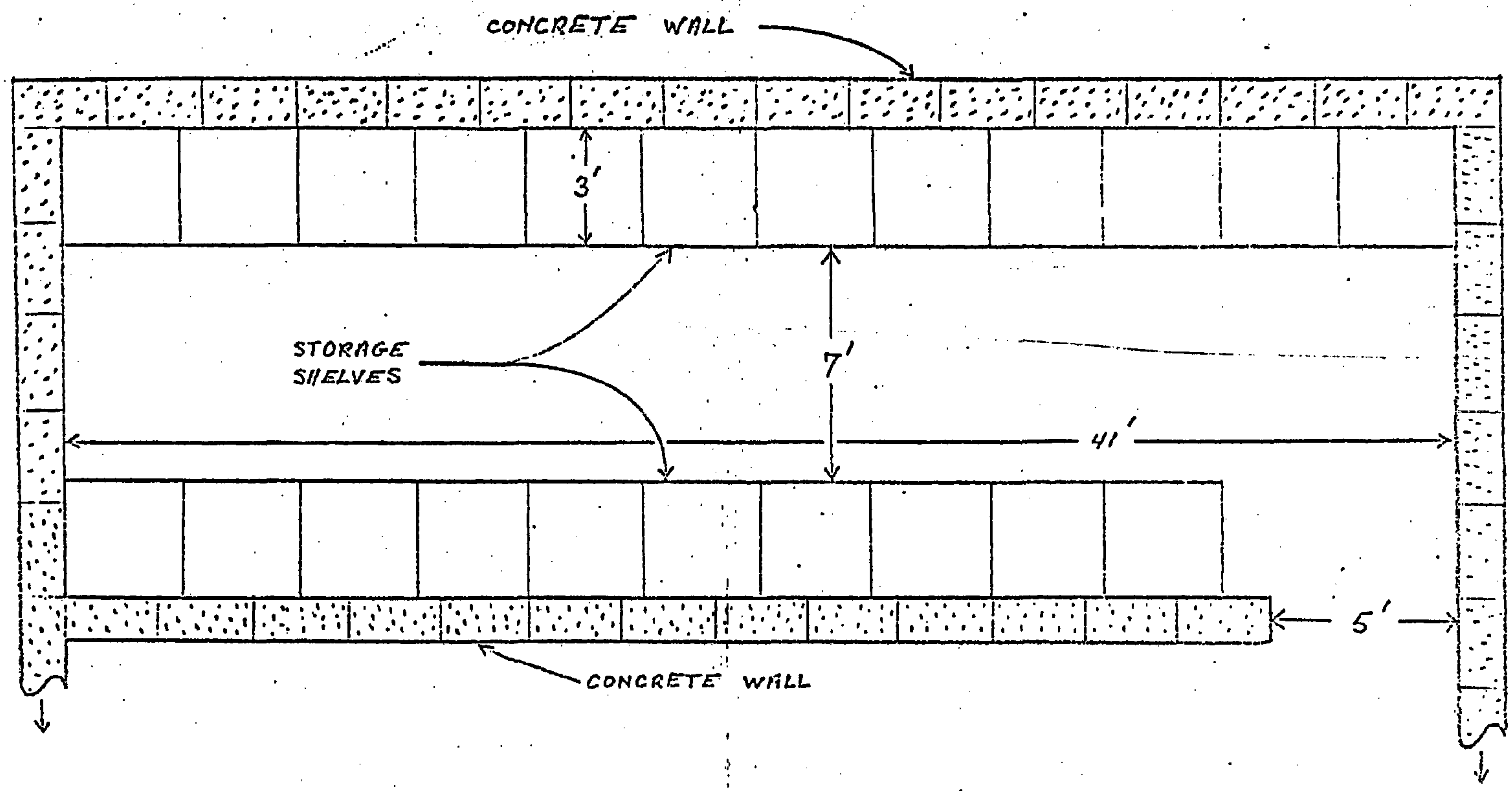


Figure 1
Overhead Plan of 41-H Storage Area

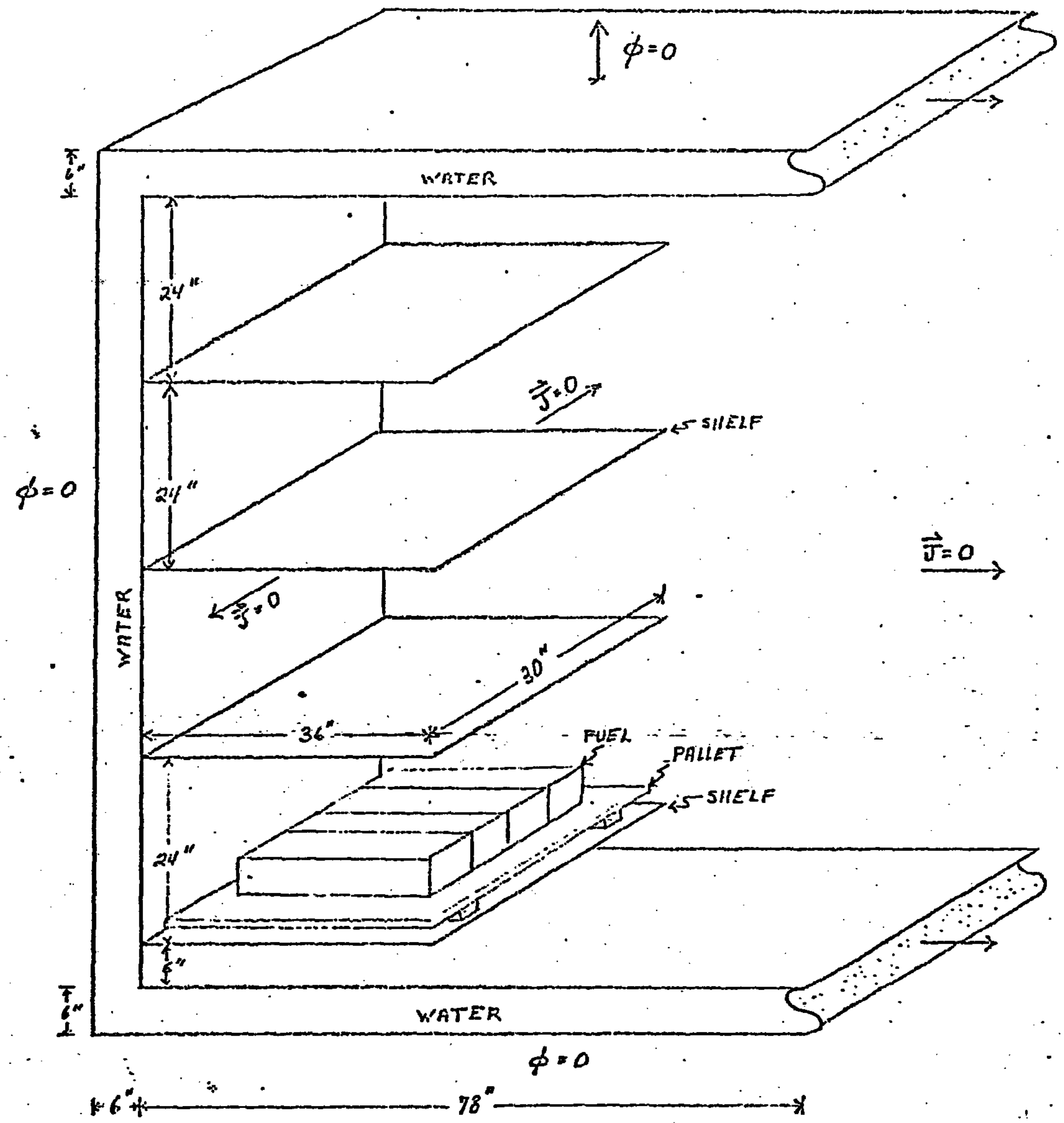
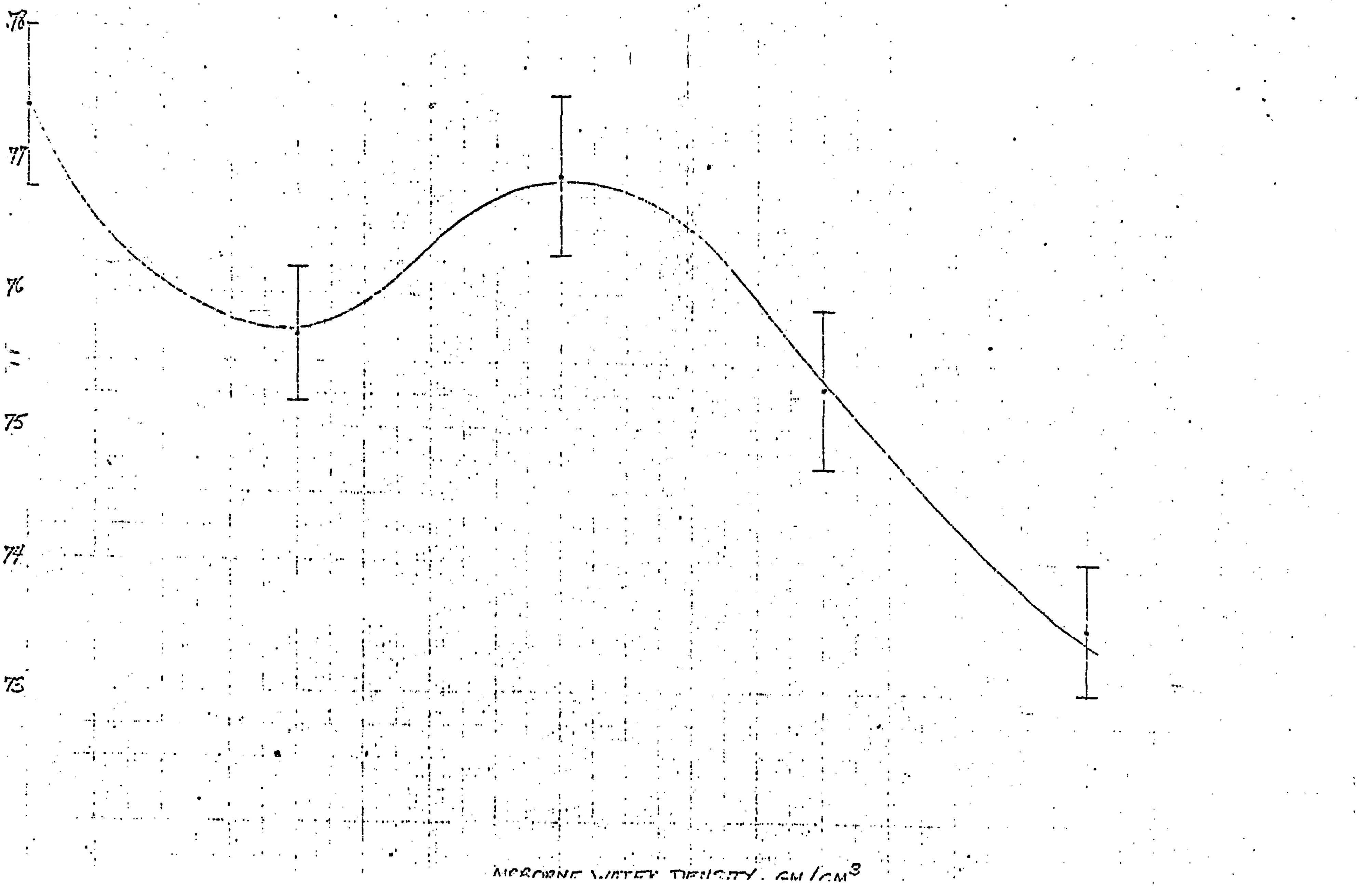


Figure 2

41-H Storage Area Model For KENO Calculations for 4 Shelves
The Neutron Boundary Conditions Used are Indicated

Figure 3

Keno Calculated keff of 41-H Storage Area with 6 Axial Shelves as a Function of the Amount of Airborne Water From the Overhead Sprinkler System



LICENSE: SNM-777, Docket: 70-820
SECTION: 800 - FUEL FABRICATION OPERATION
Subsection: 810 - Storage
Subpart: 816 - Coat Storage
816.1 - Low Enriched UO_2 Rod Mobile Work Tables

816.1 LOW ENRICHED UO_2 ROD MOBILE WORK TABLES

1. Description

These carts will be used as mobile work tables during the processing of UO_2 Rods. They also may serve as temporary storage for these rods while they are awaiting further processing. Rods will normally be kept in modules for ease in handling. These modules are described in Subpart 814.3.

Carts are constructed of angle iron, usually 1/2" x 1/2" x 1/8", which is covered with a thin metal skin, usually 24 gage or heavier. The cart assembly is welded construction. These carts are designed with a horizontal trough down the middle for holding loose rods or rods in modules. Non-stacking devices will be placed in each side of the trough. Each cart is designed so that the cart width and length or added bumpers provide at least 12" separation from other SNM.

2. Nuclear Safety

The troughs will be restricted to a safe slab thickness, depending on the enrichment of the SNM being processed. Safe slab thickness valves will be obtained using Figure 309-XV. The interaction effects of Subpart 814.3 also apply to these tables.

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SECTION 900 - FABRICATION OPERATION

SUBSECTION 920 - PROCESSING

TABLE OF CONTENTS

I. DETERMINATION OF SAFE SURFACE DENSITY VALUE

A safe surface density value of 175 gms U-235 per sq.ft. of floor area was used. Reference subsection 300.

II. PROCESS AREA LIMIT - BUILDING 41H

The total floor area is approximately 5600 sq.ft.
Assuming that the inprocess storage devices occupy approximately 1600 sq.ft. there is approximately 4000 sq.ft. as process area.

Process Area Limit $=4000 \text{ ft.}^2 \times 175 \text{ gm U-235/ft.}^2 = 700$ Kg U-235.

III. PROCESS AREA LIMIT - BUILDING 19H

The total floor area is approximately 6600 sq.ft. Assuming that inprocess storage devices occupy approximately 600 sq.ft., there is approximately 6000 sq.ft. as process area.

Process Area Limit $= 6000 \text{ ft.}^2 \times 175 \text{ gm U-235/ft.}^2 = 1050$ Kg U-235

IV. PROCESS AREA LIMIT - BUILDING 50H

The total floor area is approximately 32,175 sq.ft. Assuming that inprocess storage devices occupy approximately 3,175 sq.ft., there is approximately 29,000 sq.ft. as process area.

Process Area Limit $= 29{,}000 \text{ ft.}^2 \times 175 \text{ gm U-235/ft.}^2 = 5{,}075$ Kg U-235.

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LICENSE:	SNM-777, Docket 70-820
SECTION:	800 - FUEL FABRICATION OPERATION
Subsection:	820 - Processing
Subpart:	821 - Introduction

Approved

ISSUED 4/15/74

SUPERSEDES 10/31/68

821 INTRODUCTION

1. The subsections which follow describe the handling, processing steps, and procedures which will be used in the fabrication of finished SNM bearing components. Operations, equipment items, safety limitations and considerations, and processing information are discussed and described in the appropriate subsections.

2. The use of up to and including 350 grams of U-235 at any enrichment is authorized for research, development, and pilot operations under the following conditions:

   2.1 Each 350 gram batch may contain any number of smaller units.

   2.2 The maximum amount of material to be so regulated is 10 kgs U-235 at any enrichment per room.

   2.3 Each work station, operation, zone, or piece of equipment will be separated a minimum of four feet from each 350 gram U-235 batch or any other SNM.

3. The interaction between process operations is not calculated. Generally, each process operation is separated by three feet and identified as a criticality zone. This separation is exclusive of ports or shelves in storage devices. In lieu of determining the interaction between these process operations, a process area limit has been established using a safe surface density of 175 gm U-235 per square ft. The Process Area Limits are:

Building	Process Area Limit
41H	700 Kg U-235
19H	1050 Kg U-235
50H	5075 Kg U-235

Details concerning these limits are set forth in Nuclear Safety Evaluation 821. The Possession Limits set forth in Section 100 will not be exceeded regardless of the Process Area Limit.

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SECTION:	800 - FUEL FABRICATION OPERATION	
Subsection:	820 - Processing	Approved 6/25/71
Subpart:	822 - Fabrication of Aluminum Plate Type Element Assemblies	Issued 2/6/70
	822.1 - General Considerations	Supersedes

822.1 GENERAL CONSIDERATIONS

1. This Subpart covers the preparation and processing of up to and including fully enriched uranium-aluminum fuel material containing not more than 51% 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 normally performed in Building 50H.

3. ATR 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 ATR 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, ATR 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 sometimes 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

21 pages
withheld in entirety
Ex. 4

823.1 GENERAL CONSIDERATIONS

1. This subsection covers preparation and processing of up to and including 5% enriched UO_2 fuel material as pellets and its fabrication into tubular type fuel rod assemblies or clusters with the pellets clad in Stainless Steel, Zircaloy, or Aluminum. Individual operations which may be performed are shown on Diagram 823-I and are listed in their expected sequence.

2. Operations will be performed in Building 50H.

3. Fuel assemblies which may be fabricated under the provisions of this Subsection will be limited to the following types of components:

   3.1 Dresden (BWR)
   3.2 Yankee (PWR)
   3.3 Yankee Reload (PWR)
   3.4 Connecticut Yankee (PWR)

4. Safe cross section values used in Subpart 823.2 will be obtained using the safe diameter values shown on Fig. 309-XII. The cross section will be calculated using:

$$\text{Safe Cross Section} = \frac{\pi \times (\text{safe diameter})^2}{4}$$

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

	Safe Value	Figure
5.1	Heterogeneous Diameter	309-XII
5.2	Heterogeneous Mass	309-XIII
5.3	Heterogeneous Volume	309-XIV
5.4	Heterogeneous Thickness	309-XV
5.5	Homogeneous Diameter	309-XVI
5.6	Homogeneous Mass	309-XVII
5.7	Homogeneous Volume	309-XVIII
5.8	Homogeneous Thickness	309-XIX
5.9	Geometric Variables for Specified U235 Enrichments	309-XX

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	Safe Value	Figure
5.10	Masses for Specified U235 Enrichment	309-XXI
5.11	Safe Cylinders for any U235 Enrichment	309-XXII

6. K_{eff} values will be determined by individual Nuclear Safety evaluations included in this Subpart (923).

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SECTION:	800 - FUEL FABRICATION OPERATION	ISSUED OCTOBER 31, 1968
Subsection:	820 - Processing 823 - Fabrication of Stainless Steel, Aluminum, or Zircaley Clad Tubular Type UO_2 Pellet Assemblies	SUPERSEDES New
Subpart:	823.2 - Process Description	

823.2 PROCESS DESCRIPTION

1. Receive Pellets

Pellets will be received and handled in accordance with Subsection 810.

2. Measure and Weigh Stack

This is a dry operation and will be limited to a safe heterogeneous mass. If pellets are damaged, they will be placed in safe heterogeneous diameter or volume or cross section containers or they will be placed in a container and limited to a safe heterogeneous mass If the pellets are sufficiently damaged so that UO_2 powder is found, the UO_2 powder will be placed in safe homogeneous diameter or volume or cross section containers or placed in a container and limited to a safe homogeneous mass.

3. Pellet Dryness Test

A few representative pellets are removed from each batch or lot and tested for dryness. This operation will be limited to a safe heterogeneous mass.

4. Load and Weld

Pellets are loaded into tubes and the end plugs added and welded to the tubes during this operation. It will be limited to a safe heterogeneous mass or a fixture may be used which provides a safe heterogeneous cross-section.

5. Machine or Hand Finish Welds

This operation is usually performed on individual loaded rods one at a time. It is limited to a safe heterogeneous mass.

6. Inspect

This operation will be limited to a safe heterogeneous mass.

7. Zyglo Test

Dye penetrantis brushed on the welded end caps and the welds checked. Test materials are removed by dipping the ends of the rods in a liquid. Since only the ends are placed in a liquid (approximately 1 to 2 inches), this operation is limited to a safe heterogeneous mass.

8. Leak Test by Alcohol Immersion

Prior to leak testing, rods are immersed in alcohol. This is a wet operation and is limited to a safe hetcorogeneous mass or the rods are held by a fixture or the equipment in a safe heterogeneous cross-section.

9. Helium Leak Test

This operation is dry and performed in leak tight vessels. Each vessel is limited to a safe heterogeneous mass or the rods are retained in a safe heterogeneous cross-section.

10. Clean and Pickle

Typically, the sequence of operations are a detergent clean, a water rinse, an acid dip, a cold water and/or a hot water rinse, and a warm water soak. Tanks are spaced close together to allow quick transfer and each is roughly 1-1/2 ft. square and approximately 10 ft. deep. All SNM material is clad and fixtured. Fixtures are moved from tank to tank via an overhead crane one at a time. Only one fixture at a time is allowed in movement and one other fixture is allowed in one of the other tanks at any one time. Fixtures are designed to hold only a safe mass of SNM, or the SNM is retained in a safe cross-section, or the fixture has a k eff (bare) ≤ 0.65 and k eff (reflected) < 0.9. The safe geometry for the fixture will be determined neglecting the affects of the cladding.

Fixtures are further designed to preclude accidental insertion of two fixtures into one tank. An acid tank is assumed to have enough acid to dissolve one batch completely. Acid tanks are continuously agitated, usually by air bubbling, which insures uniform mixing and pickling. No uranium is allowed in the solutions. Pickling solutions are checked, usually by sampling, before the start of each shift to determine how spent they are. If there is reason to believe that the solution contains SNM, the solution is sampled and analyzed for uranium content prior to processing additional SNM. Material is inspected after pickling. Any over-pickling or penetration to SNM causes rejection of the material and sampling of the pickle solution. In addition, the pickle lines are equipped with an automatic withdrawal system. The system will be designed to insure automatic withdrawal of the SNM material or dumping of the acid even in the event of loss of primary power. Therefore, the insertion of a fixture containing SNM into a solution containing SNM or the dissolution of the SNM is considered unlikely. In addition, acid tanks are equipped with an automatic dump system to preclude SNM being in the tanks for extended periods due to electrical or mechanical failure. The nuclear safety of the fixtures is shown on the Nuc.Saf.Eval.-Pickle-Corrosion Fixture.

11. Load Autoclave and Corrosion Test

After pickling, the fixture containing SNM may undergo a water soak (if required), then is loaded into an autoclave and corrosion tested. Again, individual fixtures will hold only a safe mass of SNM, or the SNM is retained in a safe cross-section, or the fixture has a k eff (bare) ≤ 0.65 and k eff (reflected) < 0.9. The interaction between autoclaves is approximately 1.82 steradians with 2.5 steradians allowed.

12. Visual Inspection

This operation is limited to a safe heterogeneous mass.

13. Storage/Issue for Assembly

Storage of rods will be in accordance with Subsection 810. During issue for assembly, loose rods will be retained in a safe heterogeneous cross-section container or limited to a safe heterogeneous

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SECTION: 800 - FUEL FABRICATION OPERATION
Subsection: 820 - Processing
823 - Fabrication of Stainless Steel, Aluminum, or Zircaley Clad Tubular Type UO_2 Pellet Assemblies
Subpart: 823.2 - Process Description

14. Alpha Count

Prior to assembly, rods may be alpha counted for surface contamination. This operation is limited to a safe heterogeneous mass.

15. Assemble Elements

Rods are assembled with non-SNM components and welded (if required) into assemblies. This is a dry operation and limited to a safe mass or the assemblies will have a k eff (bare) ≤ 0.65 and k eff (reflected) ≤ 0.9.

16. Machine or Hand Finish

If required, excess weld material may be machined off or the assemblies hand finished. This operation is limited to that number of assemblies with a k eff (bare) ≤ 0.65 and k eff (reflected) ≤ 0.9.

17. Rivet Grids, Assemble End Fittings, etc.

This operation is limited to that number of assemblies with k eff (bare) ≤ 0.65 and k eff (reflected) ≤ 0.9.

18. Dimensional and Visual Inspect

Same considerations as Subpart 823.2.17.

19. Radiograph

Same considerations as Subpart 823.2.17.

20. Degrease or Water Rinse, etc.

A final cleaning operation is performed. The number of assemblies will be limited such that k eff (bare) ≤ 0.65 and k eff (reflected) ≤ 0.9.

21. Alpha Count

Same considerations as Subpart 823.2.17.

22. Store

Assemblies will be stored in accordance with Subsection 810.

ROD FABRICATION

1 Receive Pellets

2 Measure & Weigh Stack

3 Pellet Dryness Test

4 Load and Weld

5 Machine or Hand Finish Welds

6 Inspect

7 Zyglo Test

8 Leak Test by Alcohol Immersion

9 Helium Leak Check

10 Clean and Pickle

11 Load Autoclave and Corrosion Test

12 Visual Inspection

13 Storage/Issue for Assembly

14 Alpha Count

ELEMENT FABRICATION

15 Assemble Elements

16 Machine or Hand Finish

17 Rivet Grids, Assemble End Fittings, Etc.

18 Dimensional & Visual Inspect

19 Radiograph

20 Degrease or Water Rinse, Etc.

21 Alpha Count

22 Store

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Flow Diagram 823.-I

APPROVED

ISSUED: October 31, 1968

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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ISSUED: 2/6/70

SUPERSEDES: New

APPROVED:

AMENDMENT NO.:

I. BASIC INFORMATION

A. Pellet Description: 0.482" Ø x 0.6"   1.0" long

B. Rod Description:

1.

Type Rod	Fuel Stack Length	Tube Length	Wght.	Fuel Loading/Rod
Free End	108.25 ± .50	114.70"Length, Spring, Disc	8#	3251 ± 98 g (67.0 g U-235)
Fixed Type	108.25 ± .50	114.70"Length, Spring, Disc	8#	3251 ± 98 g (67.0 g U-235)
Remov. Type 1	108.25 ± .50	114.70"Length, Spring, Disc	8#	3251 ± 98 g (67.0 g U-235)
*Remov. Type 2	108.25 ± .50	114.70"Length, Spring, Disc	8#	3175 ± 95 g
PPC Type	108.25 ± .50	114.70"Length, Spring, Disc	8#	3251 ± 98 g (50.7 g U-235)
Segmented	17.54 ± .05 5 per Rod 12.63 ± .050 1 per Rod	114.70"Length, Spring, Disc	10#	3013 ± 90 g (62.1 U-235 Total)
Instrumented	None	114.70"Length, No Springs, Disc	3#	None

*Uranium Dioxide-Gadolinia (64.1 g U-235) (55.0 g Gd_2O_3)

2. Zircaloy-2 Tubing

Std Tube: 0.0370" ± .0025" W.T. x 0.4925" ± .0020" I.D.

Instru. Tube: 0.0293" ± .0025" W.T. x 0.540" ± .002" I.D.

C. Element Description

4.38" x 4.38" x {
Length Over Active Fuel - 108.25"
Length Between Grids - 115.33"
Length Overall - 134.35"
36 Rods Per Element

1.

Type Element	Required/Batch No 6
Type 1	53
Type 2	30
Instrumented	13

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Nuclear Safety Evaluation - DRESDEN-1 Fuel Elements (BWR)

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2. Type Rod

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 (δ) of 8 cm. for a full water reflector was selected.

* D. From Fig. 3 and 4 of TID-7028, an extrapolation length (also designated δ) 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. When multiple elements are evaluated, a 0.75" (1.91 cm.) gap between elements is assumed. Under moderated conditions, this will yield higher k eff values than would occur if the elements were closely packed.

III. CALCULATIONS

A. One Element - envelope = 11.13 cm. x 11.13 cm. x 275 cm.

1. Reflected Case

$$B^2_{Refl.} = \frac{9.87}{(11.13+16)^2} + \frac{9.87}{(11.13+16)^2} + \frac{9.87}{(275+16)^2} = \frac{9.87}{(27.13)^2} + \frac{9.87}{(27.13)^2} + \frac{9.87}{(291)^2} = \frac{9.87}{736.04} + \frac{9.87}{736.04} + \frac{9.87}{84,861} = 0.0134 + 0.0134 + 0.0001 = 0.0269$$

$$K_{eff} = \frac{1.21}{1 + (39 \times 0.0269)} = \frac{1.21}{1 + 1.049} = \frac{1.21}{2.049} = \boxed{0.591}$$

2. Bare Case

$$B^2_{Bare} = \frac{9.87}{(11.13+5)^2} + \frac{9.87}{(11.13+5)^2} + \frac{9.87}{(275+5)^2} = \frac{9.87}{(16.13)^2} + \frac{9.87}{(16.13)^2} + \frac{9.87}{(280)^2} = \frac{9.87}{260.18} + \frac{9.87}{260.18} + \frac{9.87}{78,400} = 0.0379 + 0.0379 + 0.0001 = 0.0759$$

$$K_{eff} = \frac{1.21}{1 + (39 \times 0.0759)} = \frac{1.21}{1 + 2.9601} = \frac{1.21}{3.9601} = \boxed{0.306}$$

B. Two Elements - envelope = (11.13 + 1.91 + 11.13 = 24.17) = 24.17 cm x 11.13 cm x 275 cm.

1. Reflected Case

$$B^2_{Refl.} = \frac{9.87}{(24.17+16)^2} + \frac{9.87}{(11.13+16)^2} + \frac{9.87}{(275+16)^2} = \frac{9.87}{(40.17)^2} + 0.0134 + 0.0001 = \frac{9.87}{1613.63} + 0.0134 + 0.0001 = 0.0061 + 0.0134 + 0.0001 = 0.0196$$

$$K_{eff} = \frac{1.21}{1 + (39 \times 0.0196)} = \frac{1.21}{1 + 0.764} = \frac{1.21}{1.764} = \boxed{0.686}$$

2. Bare Case

$$B^2_{Bare} = \frac{9.87}{(24.17+5)^2} + \frac{9.87}{(11.13+5)^2} + \frac{9.87}{(275+5)^2} = \frac{9.87}{(29.17)^2} + 0.0379 + 0.0001 = \frac{9.87}{850.89} + 0.0379 + 0.0001 = 0.0116 + 0.0379 + 0.0001 = 0.0496$$

$$K_{eff} = \frac{1.21}{1 + (39 \times 0.0496)} = \frac{1.21}{1 + 1.934} = \frac{1.21}{2.934} = \boxed{0.412}$$

C. <u>Four Elements</u> (2 x 2) - envelope = 24.17 cm. x 24.17 cm. x 275 cm.

1. <u>Reflected Case</u>

$B^2_{Refl.} = 0.0061 + 0.0061 + 0.0001 = 0.0123$

$$K_{eff} = \frac{1.21}{1 + (39 \times 0.0123)} = \frac{1.21}{1 + 0.480} = \frac{1.21}{1.48} = \boxed{0.819}$$

2. <u>Bare Case</u>

$B^2_{Bare} = 0.0116 + 0.0116 + 0.0001 = 0.0233$

$$K_{eff} = \frac{1.21}{1 + (39 \times 0.0233)} = \frac{1.21}{1 + 0.909} = \frac{1.21}{1.909} = \boxed{0.634}$$

D. <u>Nine Elements</u> (3 x 3) - envelope (11.13 + 1.91 + 11.13 + 1.91 + 11.13 = 37.21 cm.) = 37.21 cm. x 37.21 cm. x 275 cm.

1. <u>Reflected Case</u>

$$B^2_{Refl.} = \frac{9.87}{(37.21 + 16)^2} + \frac{9.87}{(37.21 + 16)^2} + \frac{9.87}{(275 + 16)^2} = \frac{9.87}{(53.21)^2} + \frac{9.87}{(53.21)^2} + 0.0001 = \frac{9.87}{2831.30} + \frac{9.87}{2831.30} + 0.0001 = 0.0035 + 0.0035 + 0.0001 = 0.0071$$

$$K_{eff} = \frac{1.21}{1 + (39 \times 0.0071)} = \frac{1.21}{1 + 0.2769} = \frac{1.21}{1.2769} = \boxed{0.948}$$

2. <u>Bare Case</u>

$$B^2_{Bare} = \frac{9.87}{(37.21 + 5)^2} + \frac{9.87}{(37.21 + 5)^2} + \frac{9.87}{(275 + 16)^2} = \frac{9.87}{(42.21)^2} + \frac{9.87}{(42.21)^2} + 0.0001 = \frac{9.87}{1781.68} + \frac{9.87}{1781.68} + 0.0001 = 0.0055 + 0.0055 + 0.0001 = 0.0111$$

$$K_{eff} = \frac{1.21}{1 + (39 \times 0.0111)} = \frac{1.21}{1 + 0.433} = \frac{1.21}{1.433} = \boxed{0.844}$$

E. <u>Sixteen Elements</u> (4 x 4) - envelope = (11.13 + 1.91 + 11.13 + 1.91 + 11.13 + 1.91 + 11.13 = 50.25 cm) 50.25 cm x 50.25 cm x 275 cm.

1. <u>Reflected Case</u>

$$B^2_{Refl.} = \frac{9.87}{(50.25 + 16)^2} + \frac{9.87}{(50.25 + 16)^2} + \frac{9.87}{(275 + 16)^2} = \frac{9.87}{(66.25)^2} + \frac{9.87}{(66.25} + 0.0001 = \frac{9.87}{4389.1} + \frac{9.87}{4389.1} + 0.0001 = 0.00224 + 0.00224 + 0.0001 = 0.00458$$

$$K_{eff} = \frac{1.21}{1 + (39 \times 0.00455)} = \frac{1.21}{1 + 0.1786} = \frac{1.21}{1.1786} = \boxed{1.027}$$

E. Sixteen Elements (continued)

2. Bare Case

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

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

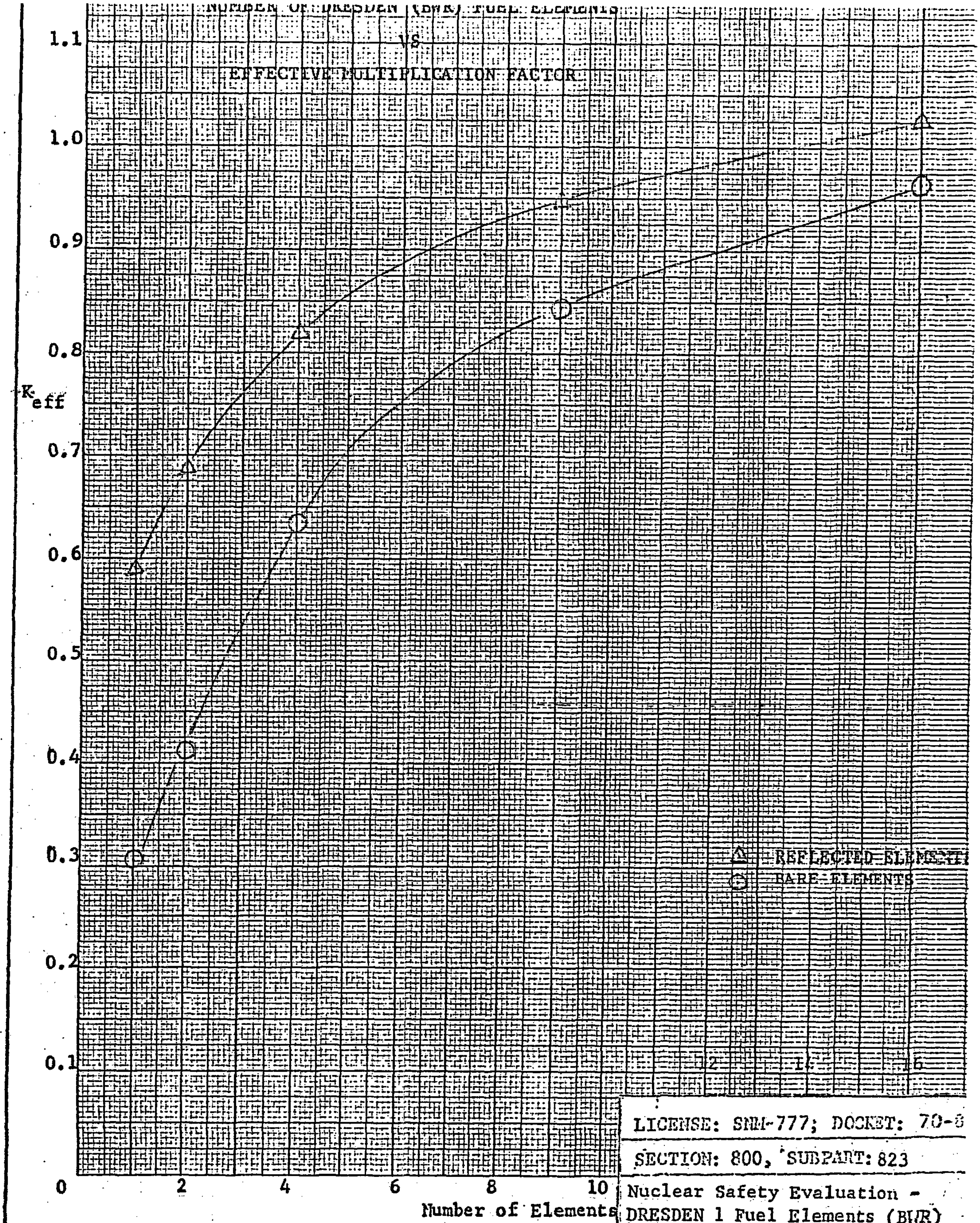
$$0.00656$$

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

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



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Nuclear Safety Evaluation - DRESDEN 1 Fuel Elements (BWR)

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APPROVED

ISSUED: OCTOBER 31, 1968

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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APPROVED: AMENDMENT NO.:

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:

1.

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

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

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

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

* D. From Fig. 3 and 4 of TID-7028, an extrapolation length (also designated δ) 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.

*Indicates Change

E. When multiple elements are evaluated, a 0.75" (1.91 cm.) gap between elements is assumed. Under moderated conditions, this will yield higher k eff values than would occur if the elements were closely packed.

III. CALCULATIONS

A. One Element - envelope = 19.34 cm x 19.34 cm x 231 cm

1. Reflected Case

$$B^2_{Refl.} = \frac{9.87}{(19.34 + 16)^2} + \frac{9.87}{(19.34 + 16)^2} + \frac{9.87}{(231 + 16)^2} = \frac{9.87}{(35.34)^2} +$$

$$\frac{9.87}{(35.34)^2} + \frac{9.87}{(247)^2} = \frac{9.87}{1249} + \frac{9.87}{1249} + \frac{9.87}{61,009} = 0.0079 + 0.0079 + 0.0001 =$$

$$0.0159$$

$$K_{eff} = \frac{1.405}{1 + (40.9 \times 0.0159)} = \frac{1.405}{1 + 0.650} = \frac{1.405}{1.650} = \boxed{0.852}$$

2. Bare Case

$$B^2_{bare} = \frac{9.87}{(19.34 + 5)^2} + \frac{9.87}{(19.34 + 5)^2} + \frac{9.87}{(231 + 5)^2} = \frac{9.87}{(24.34)^2} +$$

$$\frac{9.87}{(24.34)^2} + \frac{9.87}{(236)^2} = \frac{9.87}{592} + \frac{9.87}{592} + \frac{9.87}{55,696} = 0.0166 + 0.0166 + 0.0001$$

$$0.0333$$

$$K_{eff} = \frac{1.405}{1 + (40.9 \times 0.0333)} = \frac{1.405}{1 + 1.362} = \frac{1.405}{2.362} = \boxed{0.595}$$

B. Two Elements - envelope=(19.34 + 1.91 + 19.34 = 30.59)=30.59 cm x 19.34 cm x 231 cm.

1. Reflected Case

$$B^2_{Refl.} = \frac{9.87}{(30.59 + 16)^2} + \frac{9.87}{(19.34 + 16)^2} + \frac{9.87}{(231 + 16)^2} = \frac{9.87}{(46.59)^2} + 0.007$$

$$+ 0.001 = \frac{9.87}{2171} + 0.0079 + 0.001 = 0.0045 + 0.0079 + 0.001 = 0.0125$$

$$K_{eff} = 1 + \frac{1.405}{(40.9 \times 0.0125)} = \frac{1.405}{1 + 0.511} = \frac{1.405}{1.511} = \boxed{0.930}$$

B. Two elements (continued)

2. Bare Case

$$B^2_{Bare} = \frac{9.87}{(30.59 + 5)^2} + \frac{9.87}{(19.34 + 5)^2} + \frac{9.87}{(231 + 5)^2} = \frac{9.87}{(35.59)^2} + 0.0166 +$$

$$0.001 = \frac{9.87}{1267} + 0.0166 + 0.0001 = 0.0077 + 0.0166 + 0.0001 = 0.0244$$

$$K_{eff} = \frac{1.405}{1 + (40.9 \times 0.0244)} = \frac{1.405}{1 + 0.998} = \frac{1.405}{1.998} = \boxed{0.703}$$

C. Four Elements (2 x 2) envelope = 30.59 cm. x 30.59 cm.x 231 cm.

1. Reflected Case

$$B^2_{Refl.} = 0.0045 + 0.0045 + 0.0001 = 0.0091$$

$$K_{eff} = \frac{1.405}{1 + (40.9 \times 0.0091)} = \frac{1.405}{1 + 0.372} = \frac{1.405}{1.372} = \boxed{1.024}$$

2. Bare Case

$$B^2_{Bare} = 0.0077 + 0.0077 + 0.0001 = 0.0155$$

$$K_{eff} = \frac{1.405}{1 + (40.9 \times 0.0155)} = \frac{1.405}{1 + 0.634} = \frac{1.405}{1.634} = \boxed{0.860}$$

D. Nine Elements (3 x 3) - envelope (19.34 + 1.91 + 19.34 + 1.91 + 19.34 = 61.84 cm) = 61.84 cm x 61.84 cm x 231 cm.

1. Reflected Case - Critical

2. Bare Case

$$B^2_{Bare} = \frac{9.87}{(61.84 + 5)^2} + \frac{9.87}{(61.84 + 5)^2} + \frac{9.87}{(231 + 5)^2} = \frac{9.87}{(66.84)^2} +$$

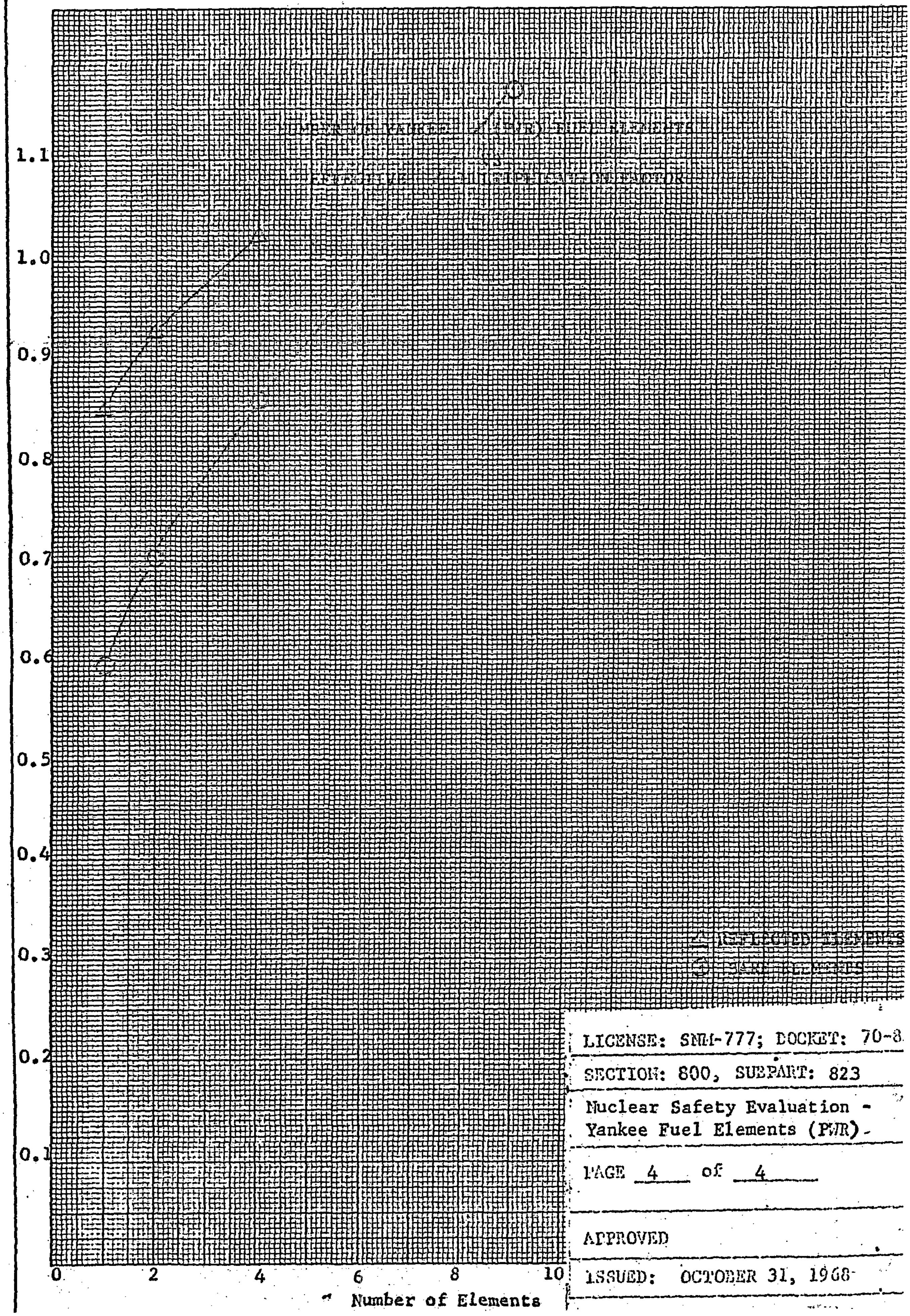
$$\frac{9.87}{(66.84)^2} + 0.0001 = \frac{9.87}{4468} + \frac{9.87}{4468} + 0.0001 = 0.0022 + 0.0022 + 0.0001 =$$

$$0.0045$$

$$K_{eff} = \frac{1.405}{1 + (40.9 \times 0.0045)} = \frac{1.405}{1 + 0.184} = \frac{1.405}{1.184} = \boxed{1.187}$$

IV. CONCLUSIONS

A. These calculations indicate that just greater than 3 elements would be required for criticality. NDEO-7034-1164 indicate that greater than 3 elements would be required for criticality. Therefore, these calculations are representative.



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Nuclear Safety Evaluation - Yankee Fuel Elements (PWR)

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APPROVED

ISSUED: OCTOBER 31, 1968

I. BASIC INFORMATION

A. Pellet Description: 0.3105 Ø x 0.6" → 1.0" long

B. Rod Description:

1.

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. Zirc-4 Tubing
Std. Tube: 0.024" W.T. x 0.365 ± 0.002" OD x 0.317 ± 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: 4.1 w/o

2. Pellet Density: 94% Theoretical

II. ASSUMPTIONS

A. The design reactivity information contained in NDEO-1164 and NEO-1083 is applicable and will be used to determine effective multiplication factors
(H/U = 121)
(KOO = 1.441)
($M^2 \approx 39\ cm^2$)

B. Calculations will be performed using the following equations

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

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

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

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D. From Fig. 3 and 4 of TID-7028, an extrapolation length (also designated δ) of 3.0 cm. was selected for bare, moderated systems. This value 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 3 to 4 cm. will yield minor reactivity changes, probably 10% or less in Keff.

E. When multiple elements are evaluated, a 0.75" (1.91 cm.) gap between elements is assumed. Under moderated conditions, this will yield higher K_{eff} values than would occur if the elements were closely packed.

III. CALCULATIONS

A. One Element - envelope = 19.34 cm x 19.34 cm x 231 cm

1. Reflected Case:

$$B^2_{\text{Refl.}} = \frac{9.87}{(19.34 + 16)^2} + \frac{9.87}{(19.34 + 16)^2} + \frac{9.87}{(231 + 16)^2} = \frac{9.87}{(35.34)^2} +$$

$$\frac{9.87}{(35.34)^2} + \frac{9.87}{(247)^2} = \frac{9.87}{1249} + \frac{9.87}{1249} + \frac{9.87}{61,009} = 0.0079 + 0.0079 + 0.0001 =$$

$$0.0159 \text{ cm}^{-2}$$

$$K_{eff} = \frac{1.441}{1 + (39 \times 0.0159)} = \frac{1.441}{1 + 0.620} = \frac{1.441}{1.620} = 0.890$$

2. Bare Case:

$$B^2_{\text{bare}} = \frac{9.87}{(19.34 + 6)} + \frac{9.87}{(19.34 + 6)^2} + \frac{9.87}{(231 + 6)^2} = \frac{9.87}{(25.24)^2} +$$

$$\frac{9.87}{(25.34)^2} + \frac{9.87}{(237)^2} = \frac{9.87}{637} + \frac{9.87}{637} + \frac{9.87}{56,169} = 0.0154 + 0.0154 + 0.0001 =$$

$$0.0308 \text{ cm}^{-2}$$

$$K_{eff} = \frac{1.441}{1 + (39 \times 0.0308)} =$$

$$\frac{1.441}{1 + 1.201} = \frac{1.441}{2.201} = 0.655$$

B. Two Elements - envelope = (19.34 + 1.91 + 19.34 = 30.59) = 30.59 cm x 19.34 cm x 231 cm.

1. Reflected Case:

$$B^2_{Refl.} = \frac{9.87}{(30.59+16)^2} + \frac{9.87}{(19.34+16)^2} + \frac{9.87}{(231+16)^2} = \frac{9.87}{(46.59)^2} +$$

$$+ 0.0079 + 0.0001 = \frac{9.87}{2171} + 0.079 + 0.0001 = 0.0045 + 0.0079 + 0.0001 = 0.0125\ cm^{-2}$$

$$K_{eff} = 1 + \frac{1.441}{(39 \times 0.0125)} = \frac{1.441}{1+0.488} = \frac{1.441}{1.488} = 0.968$$

2. Bare Case:

$$B^2_{Bare} = \frac{9.87}{(30.59+6)^2} + \frac{9.87}{(19.34+6)^2} + \frac{9.87}{(231+6)^2} = \frac{9.87}{(36.59)^2} + 0.0154 +$$

$$0.0001 = \frac{9.87}{1339} + 0.0154 + 0.0001 = 0.0073 + 0.0154 + 0.0001 = 0.0227\ cm^{-2}$$

$$K_{eff} = \frac{1.441}{1+(39 \times 0.0227)} = \frac{1.441}{1+0.885} = \frac{1.441}{1.885} = 0.764$$

C. Four Elements (2 x 2) envelope = 30.59 cm. x 30.59 cm. x 231 cm.

1. Reflected Case:

$$B^2_{Refl.} = 0.0045 + 0.0045 + 0.0001 = 0.0091\ cm^{-2}$$

$$K_{eff} = \frac{1.441}{1+(39 \times 0.0091)} = \frac{1.441}{1+0.355} \quad \frac{1.441}{1.355} = 1.063$$

2. Bare Case:

$$B^2_{Bare} = 0.0073 + 0.0073 + 0.0001 = 0.0147\ cm^{-2}$$

$$K_{eff} = \frac{1.441}{1+(39 \times 0.0147)} = \frac{1.441}{1+0.573} =$$

$$\frac{1.441}{1.573} = 0.916$$

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D. Nine Elements (3 x 3) = envelope (19.34 + 1.91 + 19.34 + 1.91 + 19.34 = 61.84 cm) = 61.84 cm x 61.84 cm x 231 cm.

1. Reflected Case:- Critical

2. Bare Case:

$$\bar{B}^2_{Bare} = \frac{9.87}{(61.84+6)^2} + \frac{9.87}{(61.84+6)^2} + \frac{9.87}{(231+6)^2} = \frac{9.87}{(67.84)^2} +$$

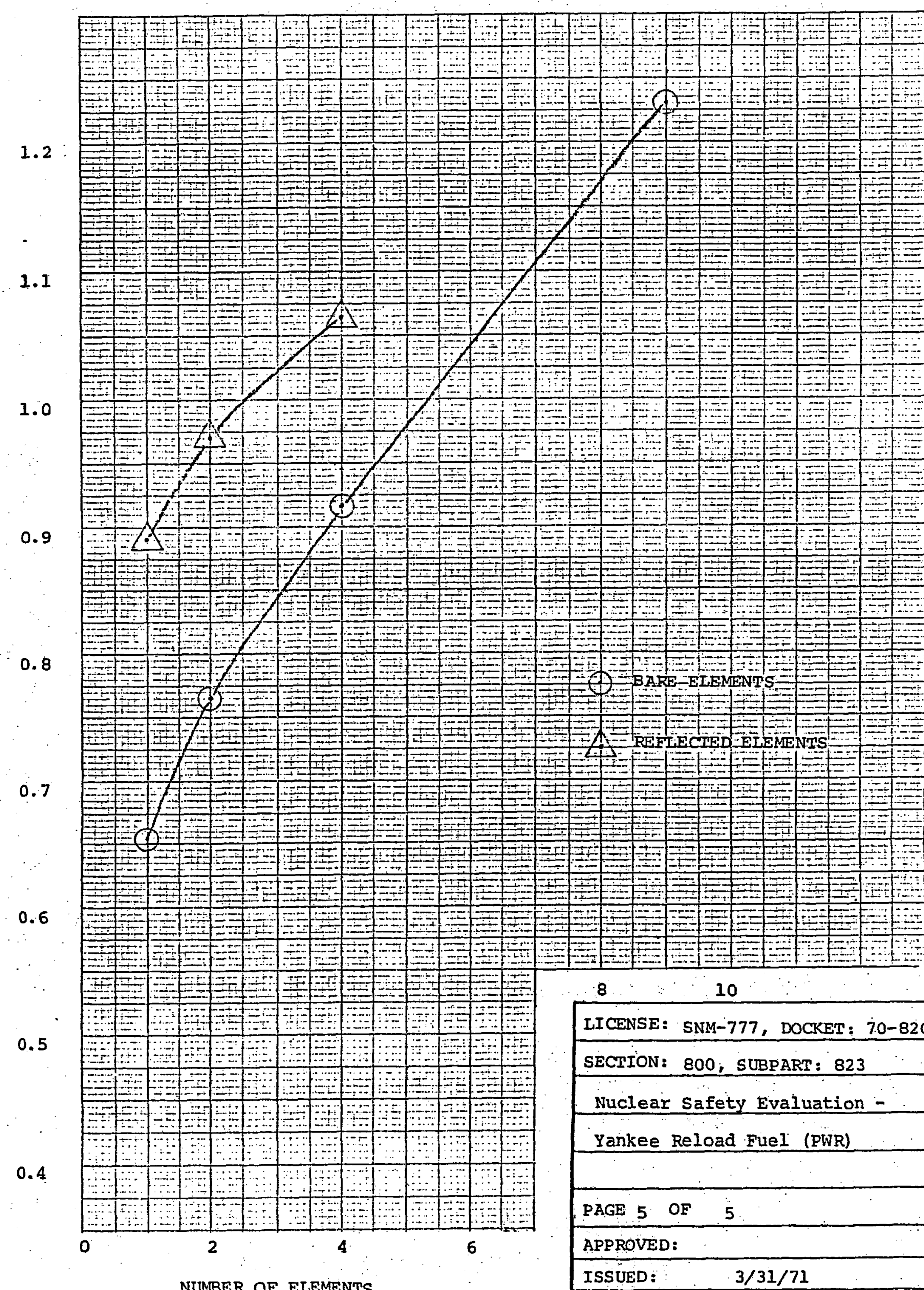
$$\frac{9.87}{(67.84)^2} + 0.0001 = \frac{9.87}{4602} + \frac{9.87}{4602} + 0.0001 = 0.0021 + 0.0021 + 0.0001 =$$

$$0.0043 \text{ cm}^{-2}$$

$$K_{eff} = \frac{1.441}{1 + (39 \times 0.0043)} = \frac{1.441}{1 + 0.168} = \frac{1.441}{1.168} = 1.234$$

IV. CONCLUSIONS

A. These calculations indicate that just greater than 2 elements would be required for criticality.



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I. DESCRIPTION

A stainless steel fixture with different circular plate holders for various core types held in place by a standard hollow center rod is used to pickle and corrosion test rods filled with low enrichment UO_2 pellets. The fuel rods are arranged in concentric rings formed by holes in the top and bottom plates. Details of construction are shown on the Referenced drawings listed on Table I.

II. NUCLEAR SAFETY OF INDIVIDUAL FIXTURES

The criticality of the fixtures has been analyzed using UNC reactor design codes. These analyses are referenced and K_{eff} values listed on Table I. It should be noted that the fixture for Yankee Reload Fuel was modified by removing the inner ring so that there are only six rings as was recommended in NED-1083. The reactivity increase due to loss of cladding was considered in all cases and this contribution is also listed on Table I.

Although the K_{eff} value for Yankee Reload Fuel slightly exceeds the limit stated in Subpart 823.2.10 (K_{eff} reflected $\leq$ 0.9), the fixture is considered to be subcritical by a sufficient amount.

III. INTERACTION

Based on the analysis set forth on page 4, NED-21, the interaction effects for fixtures in pickle tanks, rinse tanks and corrosion test vessels is considered negligible.

IV. STRUCTURAL INTEGRITY

A structural evaluation was performed to determine that the fixture would contain the rods in the proper arrangement under accident conditions of a dropped full fixture. The results of this structural evaluation are shown on ENH- 71-263 and its Addendum. The fixture for 252 Dresden Rods has not been evaluated and will not be used at this time. The inspection criteria of Subsection 304 will be applied during use for processing.

V. CONCLUSION

Fixtures D-302863-2 and D-302863-4 are nuclearly safe.

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TABLE I

Reactor Core Type	Number of Rods In Fixture	K_{eff} Value	Reactivity Change Due To Loss of Cladding	Criticality Analysis Reference(s)	Reference Drawing
Dresden	158	0.854	+ 0.010 $\Delta k/k$	NDEO-1077 & - 1359	D-302863-2
Dresden (1)	252	0.910	+ 0.013 $\Delta k/k$	NED-21	D-302863-3
Yankee Reload	285	0.923	+ 0.011547 $\Delta k/k$	NED-1083	D-302863-4

Note: (1) This fixture not approved for use.

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UNITED NUCLEAR CORPORATION

INTER-OFFICE MEMO

NED-21

TO	L. Swallow	AT Hematite	DATE May 6, 1969
FROM	R. Wendlandt, J. Tomonto	AT Elmsford	COPY TO G. Sofer, J. O'Toole, R. Kropp
SUBJECT	CRITICALITY OF DRESDEN FUEL RODS LOCATED IN A 12.25" OD RACK DURING CORROSION AND PICKLING OPERATIONS		

1.0 SUMMARY

An analysis has been performed to determine criticality safety limits for Dresden 2.34 w/o UO_2 fuel rods during corrosion and pickling in a 12.25" OD fixture. This study is an extension of previous work(1) for a larger fixture accommodating 252 rods vs. 158 rods. The Dresden regular fuel was located in a maximum of seven concentric rings around a central stainless steel tube (1.900 in. OD x 1.500 in. ID). A 1 in. schedule 40 stainless steel tube containing B_4C powder was located within the central stainless steel tube in the reference drawing. The outer ring of rods is located on a 12.25 in. diameter bolt circle. This pickling rack can be located in either the 14 in. or the 20 in. NPD autoclaves.

The moderating ratio for the fuel rods in a fully loaded fixture is less than optimum. However, since there is a possibility that the fixture may be loaded with less than its full complement of rods, all calculations were performed for fuel rods at optimum moderation and at room temperature. Operation of these units at elevated temperatures will result in lower reactivity levels because of the presence of negative Doppler and moderator temperature coefficients.

The results of this analysis indicate:

(1) The maximum k_∞ for a repeating array of 2.34 w/o Dresden UO_2 fuel rods in water ($T = 70^oF$) is 1.330 and occurs at a water to fuel volume ratio of 2.20.

(2) The maximum k_{eff} for the design fixture(2) located in a fully reflected 20 in. ID autoclave is 0.9095. The k_{eff} of this

system in which the central B_4C tube is replaced by water was calculated to be 0.912. It is concluded that the B_4C tube located in the center of the test fixture is relatively ineffective in reducing reactivity. This occurs because the central region of the fixture is already overmoderated and the addition of either water or B_4C results in additional absorptions.

(3) The maximum k_{eff} for the design fixture(2) located in a fully reflected 14 in. ID autoclave is 0.9000. A slight decrease in reactivity compared to the 20 in. ID autoclave results from additional absorptions as the 2 in. thick autoclave wall is brought closer to the fuel rods.

(4) Removing the center or outermost ring of Dresden fuel rods from the pickling fixture reduced k_{eff} to 0.843 for the fully reflected 14 in. ID autoclave at optimum internal moderation (water/fuel volume ratio 2.20).

(5) In evaluating interaction between autoclaves, the k_{eff} of the bare system is important. The calculated k_{eff} for systems with a void boundary condition at the outside of the 14 in. ID autoclave, at the inner wall of the 14 in. ID autoclave and at the edge of the pickling fixture were 0.8861, 0.7957 and 0.7529 respectively. The calculated k_{eff} for the design system(2) with the void boundary condition at the inner wall of the 20 in. ID autoclave was 0.9004. It was concluded that the maximum leakage probability from a bare autoclave is 0.9845, i.e., 0.8861/0.9000.

2.0 METHOD OF ANALYSIS

Calculations of the fueled regions assumed a regular array of regular enrichment Dresden 1 reload fuel rods in water ($T = 70^oF$). The Dresden regular fuel rods were selected because they are the most reactive of the three types of UO_2 rods used in Type 7 Dresden reload assemblies The fuel is 2.34 w/o enriched UO_2 pellets, (OD = 0.482") density 10.34 gm UO_2/cm^3, in Zircaloy tubing (ID = 0.4925", OD = 0.5625).

Calculations of k_∞ were performed using the LASER[3] code. This code is a one dimensional (cylindrical), multi-energy (85 groups) lattice-cell program which is based on the MUFT and THERMOS codes. Fast group parameters ($E > 1.85$ ev) are averaged over a semi-infinite medium spectrum calculated by the MUFT method. A correlation of the U^{238} resonance integral data is incorporated in the code to correct for resonance spatial and energy self-shielding effects. Thermal group constants were determined for a one dimensional THERMOS type calculation using the NELKIN scattering kernel. All basic microscopic cross section data are contained within the LASER library, which was compiled by WAPD based on recently reported experimental data. The LASER cross sections and calculational method have been extensively tested with experimental data and do not require any arbitrary adjustments to improve agreement with critical experiments[4].

Calculations of the pickling rack with internal and external water regions were performed using the AIM-6 one dimensional diffusion theory code with two neutron energy groups. Cross sections for the reflectors and autoclave wall were calculated with the FORM-TEMPEST codes.

3.0 RESULTS

The calculated variation of k_∞ for a repeating array of Dresden 4 regular fuel rods is shown in Figure 1. A point noted on this figure corresponds to the calculated k_∞ for a completed Type 6P or 6I assembly[1] (having an average enrichment of 2.24). The maximum k_∞ of 1.33 occurs at a water to fuel ratio of 2.2.

Calculations were performed to determine the loss of reactivity when the inner or outermost ring of fuel rods were removed from the pickling rack. The calculated k_{eff} for both conditions was 0.843 for a water to fuel ratio of 2.2 in the fuel region and with a thick water reflector surrounding the pickling rack.

The worth of removing the zirconium cladding was calculated as +1.3% $\Delta k_\infty / k_\infty$ for an array of Dresden UO_2 fuel rods at optimum moderation. This is the maximum reactivity increase that could result from dissolving the zirconium cladding material.

The formation of steam in the center of the test fixture would result in a positive change of reactivity for the system. In the present design, utilizing a central B_4C rod, there is little extra water in this region. If the B_4C rod were removed, this effect could be a potential problem (see Ref. 1).

Results of other calculations are summarized in Section 1.0.

4.0 INTERACTION BETWEEN AUTOCLAVE UNITS

In analyzing the interaction between adjacent fully loaded autoclaves, at optimum internal moderation, the following conditions are applicable:

(1) Full Water Flooding Between Assemblies

In this situation, the presence of water essentially isolates each autoclave and the maximum k_{eff} is that calculated for a fully reflected unit (e.g., 0.9095 for a 20 in. ID autoclave).

(2) No Moderator Between Autoclaves

This situation may be evaluated by either of two methods:

a. Solid Angle Method - In this method, the standard solid angle formulae is presented in Ref. 6. In this method, the k_{eff} of the unit is that calculated for a bare test fixture with optimum internal moderation, e.g., 0.7529. This method is highly conservative since it assumes no material between interacting bare units. Recent calculations[7] indicate that as little as 0.126 in. of steel surrounding individual units of a reflected array is sufficient to maintain subcriticality independent of the amount and distribution of interspaced hydrogeneous moderator present. The autoclave wall in this application is 2 in. of stainless steel.

b. General Interaction Formulae - A generalized method for evaluating interaction between subcritical units is derived in Appendix A. This method can be applied to determine k_{eff} of a line of autoclaves. In evaluating equation 4 of Appendix A, the leakage probability is determined from

$$P_i = 1 - P^i_{NL}$$

where: P_{NL}^{i} = the non-leakage probability of a bare autoclave

The term $P_{NL}^{i} = k_{eff}^{BARE} / k_{eff}^{FULLY\ REFLECTED}$ as determined from the previous calculations.

5.0 REFERENCES

1. NDEO-1077, "Criticality of Dresden Fuel Rods During Corrosion and Pickling Operations", J. R. Tomonto (June 21, 1967).

2. UNC Commercial Products Division Drawing D-302863-3 (March 15, 1969).

3. WCAP-6073, "LASER - A Depletion Program for Lattice Calculations Based on MUFT and THERMOS", C. G. Poncelet, April, 1966.

4. Eich, W. J., "Analysis of PuO_2-UO_2 Critical Experiments", Trans. American Nuclear Society, Vol. 10-1 (June, 1967).

5. NDEO-1033, "Nuclear Information Required for Dresden Shipping Container Design", Letter to R. E. Kropp from L. Goldstein, May 19, 1967.

6. TID-7016, Rev. 1, Nuclear Safety Guide, Page 34.

7. CDC-6, "Some Effects of Interspaced Moderation on Array Criticality", J. T. Thomas (March 14, 1969).

Robert V. Wendlandt

R. Wendlandt

James R. Tomonto

J. R. Tomonto

RW:JRT/jk

APPENDIX A

Interaction Between Subcritical Units

The criticality of a system of individual subcritical units can be determined through a consideration of the multiplication factor of each unit (M_i), the leakage probability of each unit (P_i) and the fractional solid angle subtended by unit j from unit i (I_{ij}). The term multiplication factor, as commonly used in reactor experiments, is the number of neutrons created per neutron introduced into a system (e.g., loading experiments by the inverse multiplication method). Therefore, the approach is to determine the multiplication of the composit system resulting from the introduction of a neutron into one of the subcritical units.

Consider the configuration shown in Figure 2 in which one neutron is introduced into unit 1. The total number of neutrons resulting is

$$M_1 \times 1 \tag{1}$$

Of these neutrons, $P_1\ M_1\ I_{12}$ neutrons leave unit 1 and enter unit 2 resulting in

$$M_2\ (P_1\ M_1\ I_{12}) \tag{2}$$

source neutrons. Of these neutrons $(P_2\ M_2\ I_{21})\ (P_1\ M_1\ I_{12})$ neutrons return to unit 1 causing

$$M_1\ (P_2\ M_2\ I_{21})\ (P_1\ M_1\ I_{12}) \tag{3}$$

source neutrons. The multiplication of the system is determined by the sum of equations 1 to 3 plus further contributions, i.e.:

$$M_S = \sum_{i=0}^{\infty} \left\{ M_1 (M_1 P_1 I_{12})^i (M_2 P_2 I_{21})^i + M_2 (M_1 P_1 I_{12})^{i+1} (M_2 P_2 I_{21})^i \right\} \tag{4}$$

where $M_S = k_S/(1-k_S)$ (5)

$M_j = k_j/(1-k_j)$ (6)

Fortunately, contributions to the sum become very small for $i > 3$ and equation 4 is easily evaluated.

The interaction expression derived above is conservative in that neutrons leaking from one unit to another are assigned the same multiplication factor and leakage probability as a neutron born uniformly in the system.

Equation 4 also applies to the situation where there are multiple units of one type shielded from each other so that there is no direct interaction between units of the same type, e.g., I_{11}. In this case, the fractional interaction angle is the sum of the contribution from all sources.

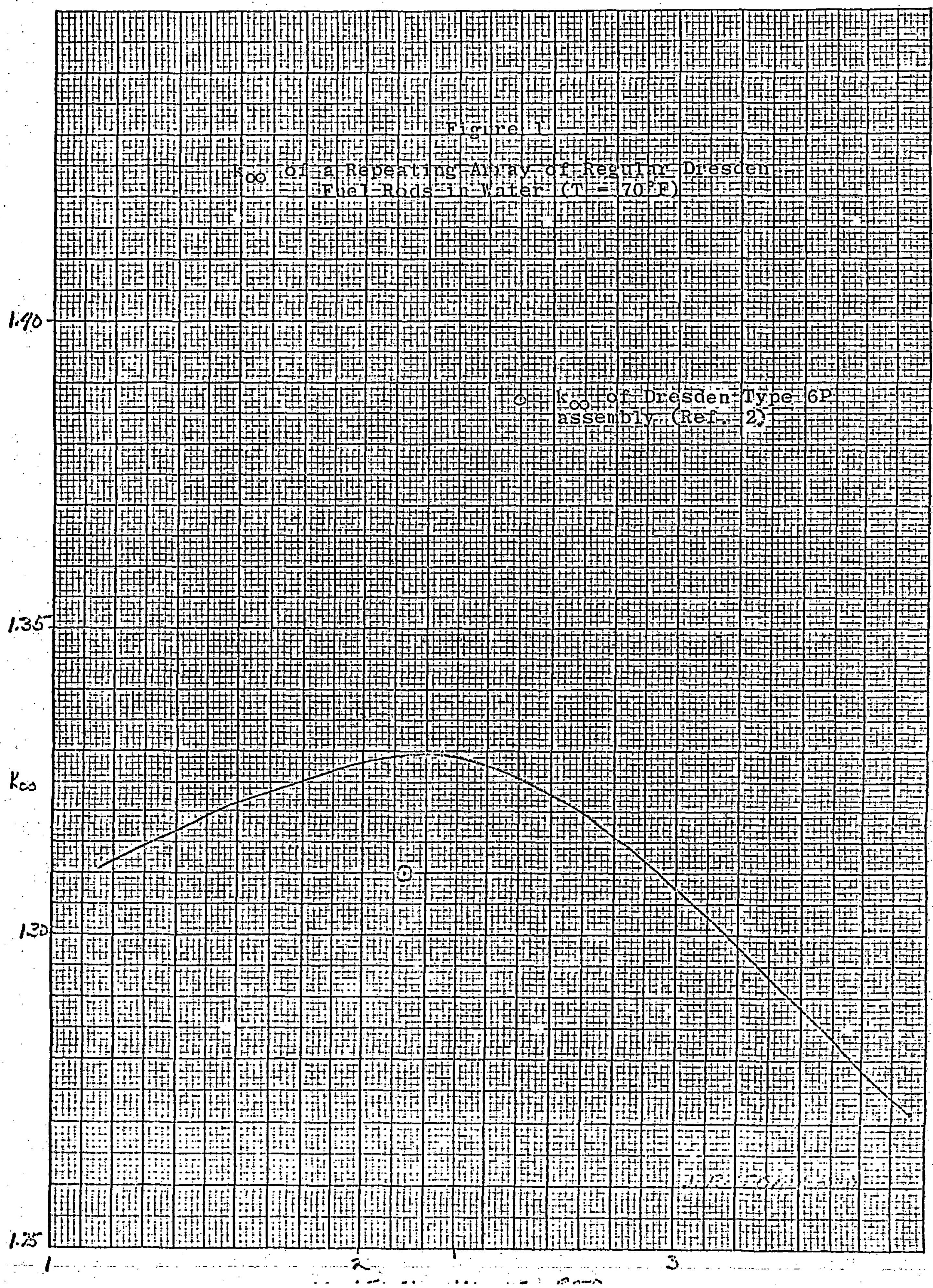


Figure 1
k_∞ of a Repeating Array of Regular Dresden
Fuel Rods in Water (T = 70°F)

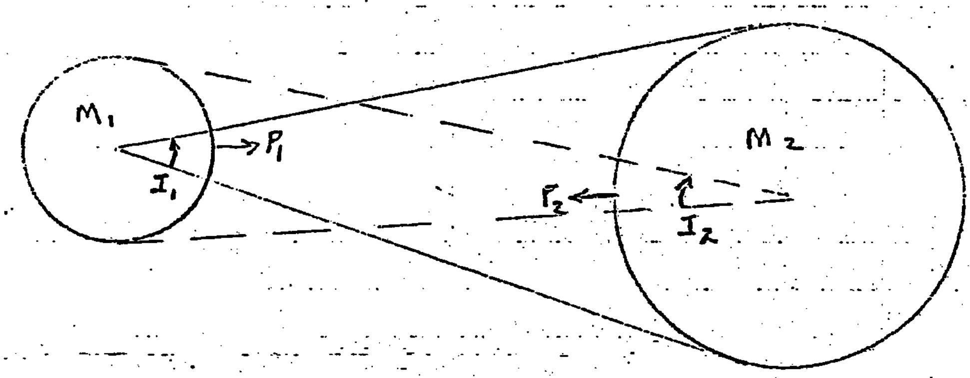


FIGURE 2

Interaction Between Subcritical Units

UNITED NUCLEAR CORPORATION

INTER-OFFICE MEMO

NED-1083

TO	R. Kropp	AT	DATE February 23, 1971
FROM	E. Fass	AT	COPY TO J. R. Tomonto
SUBJECT	Criticality of Yankee Fuel Rods During Pickling and Corrosion		

FEB 24 1971

1.0 Summary

An analysis has been performed to determine reactivity levels of Yankee 4.1w/o UO_2 fuel rods during pickling and corrosion operations. The Yankee fuel rods analyzed are located in a maximum of seven concentric rings around a central water region (OD-3.125") as shown in UNC Dwg. D-302863-4. The outer ring of rods is located in a 10.125" diameter bolt circle.

The results of this analysis indicate:

1. The maximum k_∞ for a repeating triangular pitched array of Yankee fuel rods in water ($T=68^oF$) is 1.475 and occurs at an H/U^{235} ratio of 200. The maximum critical buckling is 0.01370 cm^{-2} and occurs at a slightly higher H/U^{235} ratio of 240.

   The H/U^{235} of the Yankee fuel rods in the pickling tank when loaded with six or seven rings of fuel is about 175. If some rods were removed from random locations in the rack the system could move toward a higher level of reactivity.

2. The maximum k_{eff} for the fully reflected rack loaded with Yankee fuel rods is 0.969. This value of k_{eff} does not represent an adequate margin of safety from criticality. The calculational model was constructed conservatively. It was assumed that acid was not present in the moderator and the steel plates of the pickling rack and steel walls of the pickling tank were neglected.

   For the rack loaded into a 20" ID tank, the reflector region between the rack and inside wall is essentially infinite and the value of k_{eff} of the tank bare and reflected will be the same and determined by the k_{eff} of the fully reflected rack. Even for the 14" I.D. tank the distance

between the fueled zone and inside tank walls is about 1.5 diffusion lengths which makes the reflector savings 90% that of an infinite reflector. With this thickness of water and the 2" thick tank walls, the k_{eff} of the 14" tank reflected is approximately determined by the value above. The k_{eff} bare will be slightly less. This behaviour of the pickling tanks has been demonstrated previously for calculations of Dresden fuel.[1]

3. The k_{eff} for the fully reflected rack with the outermost ring of rods removed is 0.923. Reference 1 indicates removal of the inner ring produces the same reduction in reactivity.

4. For the fully loaded rack, the maximum $\Delta k/k$ associated with the dissolution of the zirconium clad into the acid is +1.15%.

1.1 Conclusions

It is concluded that the pickling rack fully loaded with 4.1 w/o Yankee fuel rods submerged in room temperature water does not have an adequate margin of safety from criticality. An adequate margin could be gained by removal of either the outer or inner ring of fuel rods.

2.0 Method of Analysis

Calculations of the fueled regions assumed a triangular pitched array of Yankee fuel rods of 4.1 w/o U^{235}. Although the nominal enrichment of these rods is 4.0 w/o, a variation of $\pm$ 0.1 w/o was taken into the calculations in the interest of conservatism. A description of the Yankee fuel rods is given in Table 1.

Cross sections, migration area, and k_∞ for a unit cell of a Yankee fuel rod in water were calculated with the UNC code LOCALUX-2[2]. LOCALUX is an improved version of the LASER[3] code. It is a one dimensional (cylindrical), multienergy (85 groups) lattice cell program which is based on the MUFT[4] and THERMOS[5] codes. Fast group parameters ($E > 1.855$ eV) are averaged over a semi-infinite medium spectrum calculated by MUFT. A correlation of the U^{238} resonance integral data is incorporated in the code to correct for resonance spatial and energy self-shielding effects. Thermal group constants were determined for a one dimensional THERMOS type calculation using the Nelkim scattering kernel.

Since the fuel rods in the pickling rack are not spaced on a true triangular pitch, the area of moderator associated with each pin was determined by dividing the area of the annular region in which the rods are grouped by the total number of rods. The I.D. of this annulus was assumed to be one inch less than the diameter of the circle lying on the centers of the first ring of fuel rods. The O.D. was assumed to be one inch greater than the diameter of the circle lying on the centers of the outer ring of fuel rods.

In order to determine the reactivity increase which might occur should the Zr clad be totally dissolved, a LOCALUX-2 was run replacing the clad and gas gap with water. The moderator thickness of the unit cell in this calculation was that determined by the fully loaded pickling rack.

Calculations of the pickling rack with internal and external water regions were performed using the AIM-6[6] one dimensional diffusion theory code with two neutron groups. Cross sections for the water were calculated with the FORM [7] and TEMPEST [8] codes. The external reflector was taken to be 30 cm. thick.

3.0 Results

The LOCALUX calculated k_∞ of Yankee fuel rods as a function of the thickness of the moderator region are tabulated in Table 2 and shown plotted in Figure 1. The curve is seen to peak at k_∞ = 1.475 (H/U^{235} ratio of 200). As the H/U^{235} ratio increases the neutron migration area, M^2, decreases. The result is that the maximum critical buckling, B^2_{CRIT}, determined by

$$B^2_{CRIT} = \frac{k_\infty - 1}{M^2}$$

is 0.01370 cm^{-2} at an H/U^{235} ratio of 240. Tabulated values and plots of M^2 and B^2_{CRIT} are given in Table 2 and Figure 2.

AIM-6 was run for two different loadings of the pickling rack. The first was fully loaded, i.e. 7 rows of fuel rods. The second was with the outer row of rods removed. As discussed above, the method of determining the amount of moderator associated with the fuel depends on the calculated area of the fueled annulus. For the case containing seven fueled rows and six fueled rows the H/U^{235} ratios were 173 and 178 respectively. As seen from Figure 1 both of these systems are somewhat undermoderated. The effective multiplication values are given below.

Number of Rows	AIM-6 k_{eff}
Seven	0.969
Six	0.923

The increase in reactivity associated with the dissolving of the zirconicum clad as discussed in section 2.0 is:

$$\Delta K/K = \frac{k_\infty \text{ (unclad)} - k_\infty \text{ (clad)}}{k_\infty \text{ (unclad)}}$$

$$= \frac{1.4896 - 1.4724}{1.4896} = 0.011547$$

The maximum increase in reactivity should the clad be removed is 1.15%.

References

1. Tomonto, J., Wendlandt, R., "Criticality of Dresden Fuel Rods Located in a 12.25" OD Rack During Corrosion and Pickling Operations", NED-21, UNC (May 1969).

2. Fiscella, J.M., "LOCALUX-2 Program Description Revision 3 of Phys/Math 5201," NED-725, UNC (July 1970).

3. Poncelet, C. G., "LASER - A Depletion Program for Lattice Calculations Based on Muft and Thermos," SCAP-6073 (April 1966).

4. Bohl, H., Gelbard, E.M., and Ryan, G.M., "MUFT-4 Fast Neutron Spectrum Code for the IBM-704," WAPD-TM-72 (1957).

5. Honeck, H.C. "THERMOS - A Thermalization Transport Theory Code for Reactor Lattice Calculations," BNL-5826 (1961).

6. Flatt, H.P., Baller, D.C., "AIM-5 a Mutligroup One Dimensional Diffusion Theory Code", NAA-SR-4694.

7. Mc Goff, D.J., "FORM-A Fourier Transform Fast Spectrum Code for the IBM-709," NAA-SR-Memo 5766 (September 1960)

8. Shuddle, R.H. and Dyer, J. "TEMPEST-II - A neutron Thermalization Code", AMTD-III (June 1962).

Ed Fass

E. Fass

EF:ah

Table 1
Description of UNC Yankee Reload Fuel

Enrichment w/o	4.0
O. D. rod, in.	0.365
I. D. rod, in.	0.317
Clad material	Zr - 4
Clad thickness, in.	0.024
Gap thickness, in.	0.0065
O. D. pellet, in.	0.3105
Active fuel length, in.	91.0

Table 2
LOCALUX-2 Results on Yankee
Fuel Pins (4.1 w/o U^{235})

H/U^{235}	k_∞	M^2, cm^2	B^2_{CRIT}, cm^{-2}
103	1.4187	40.28	0.01040
124	1.4449	38.56	0.01154
168	1.4712	36.25	0.01300
188	1.4748	35.56	0.01335
199	1.4752	35.26	0.01348
209	1.4749	34.99	0.01357
227	1.4730	34.60	0.01367
242	1.4699	34.31	0.01370
283	1.4576	33.70	0.01358

KEUFFEL & ESSER CO.

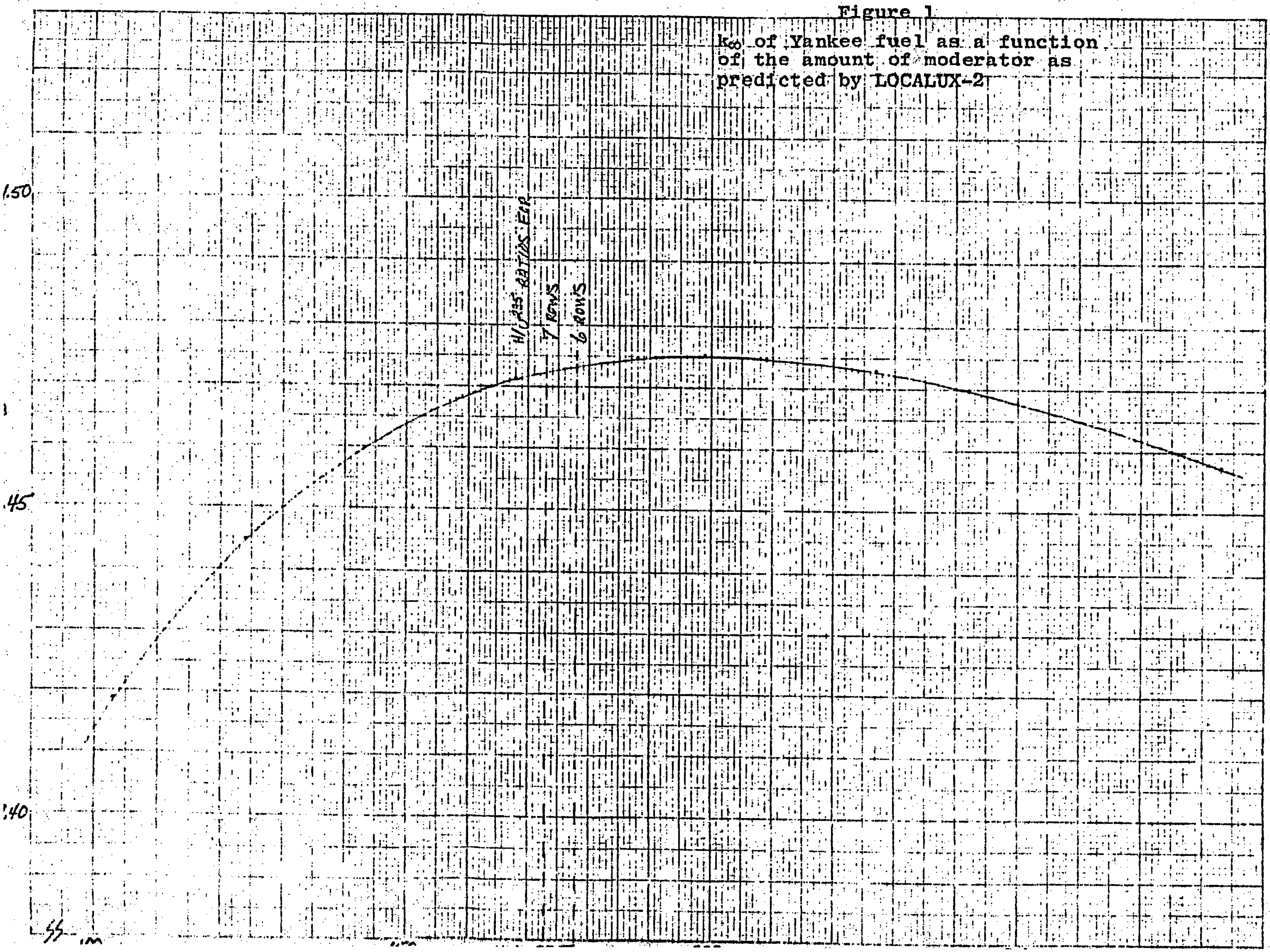


Figure 1

k_∞ of Yankee fuel as a function of the amount of moderator as predicted by LOCALUX-2

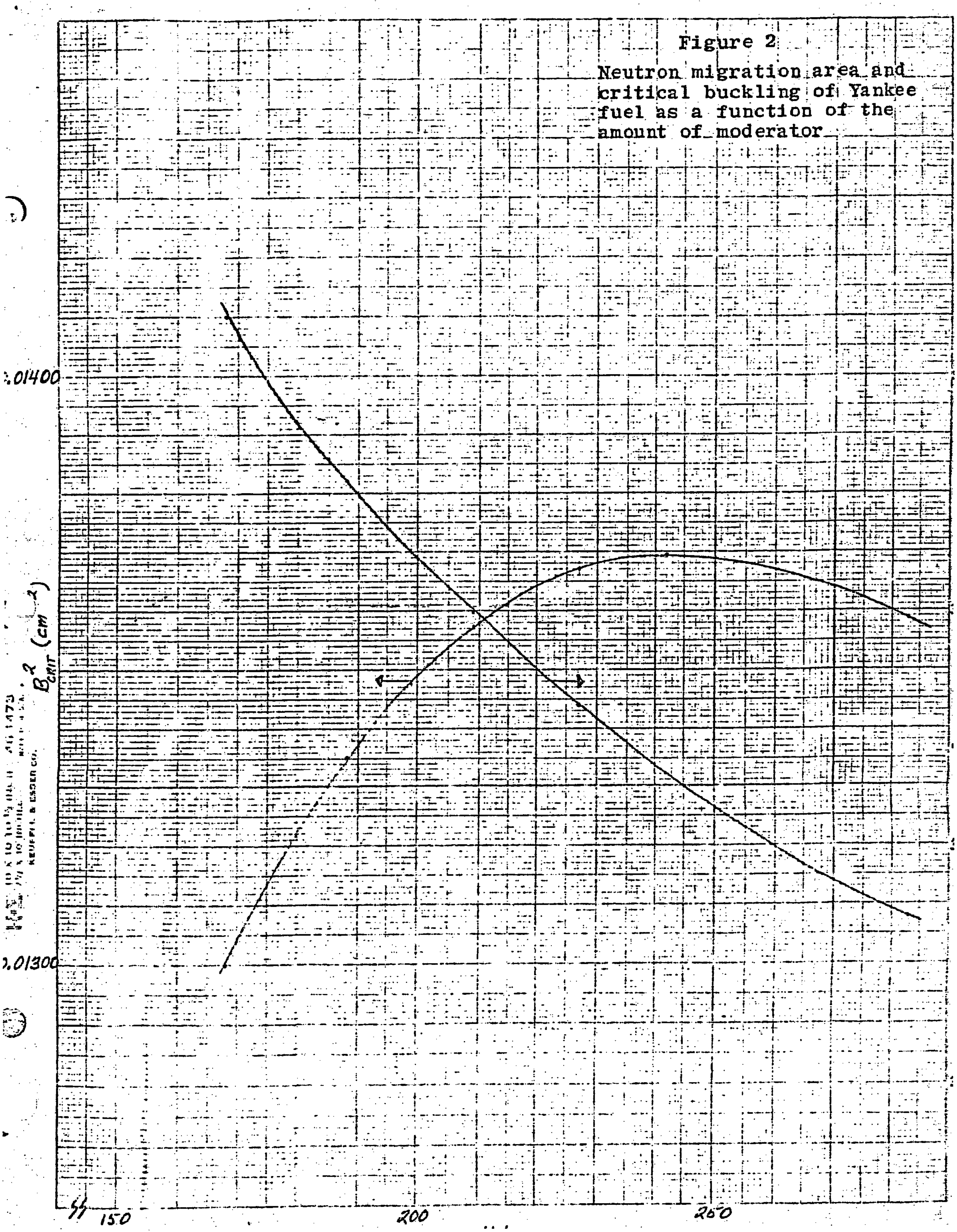


Figure 2
Neutron migration area and critical buckling of Yankee fuel as a function of the amount of moderator
.01400
.01300
B_{CRIT}^2 (cm^{-2})
150
200
250

ENH-71-263
6/2/71

STRUCTURAL ANALYSIS OF DRESDEN CORROSION RACK E-302863

Requirements

Corrosion rack must retain fuel rods within original safe geometry in the event the rack is dropped from any crane in building 24-D. The maximum height of drop would be 20 feet and could only occur within an autoclave, which would guide the rack so that it could only land on its bottom supporting legs. The autoclave would also prevent the possibility of the rack falling on its side after impact.

Assumptions

Fuel rods would remain within safe geometry providing holddown plate would remain in position after drop. Holddown plate is secured in main structure by means of a 3/8 inch diameter stainless steel pin. Criteria of failure is based on shearing of pin (maximum allowable shear stress = $.75\ S_t$ maximum = 67,500 psi) as a result of force developed in drop. Calculations are based on rack and legs stopping simultaneously while upper portion of rack continues downward with fuel rods, absorbing energy through elasticity of fuel rods and center pipe only. The cushioning effect of the fluid in the autoclave is neglected.

Comments

The shearing load applied to the pin was chosen as the criterion for failure for the following reasons.

1. A deflection of the legs and/or bottom plate would tend to cushion the impact.

2. The shear area of the pin permits higher stresses to be applied at the pin than would occur elsewhere in the fixture. A failure of this pin might conceivably allow the fuel rods to rebound out of the fixture following impact.

References

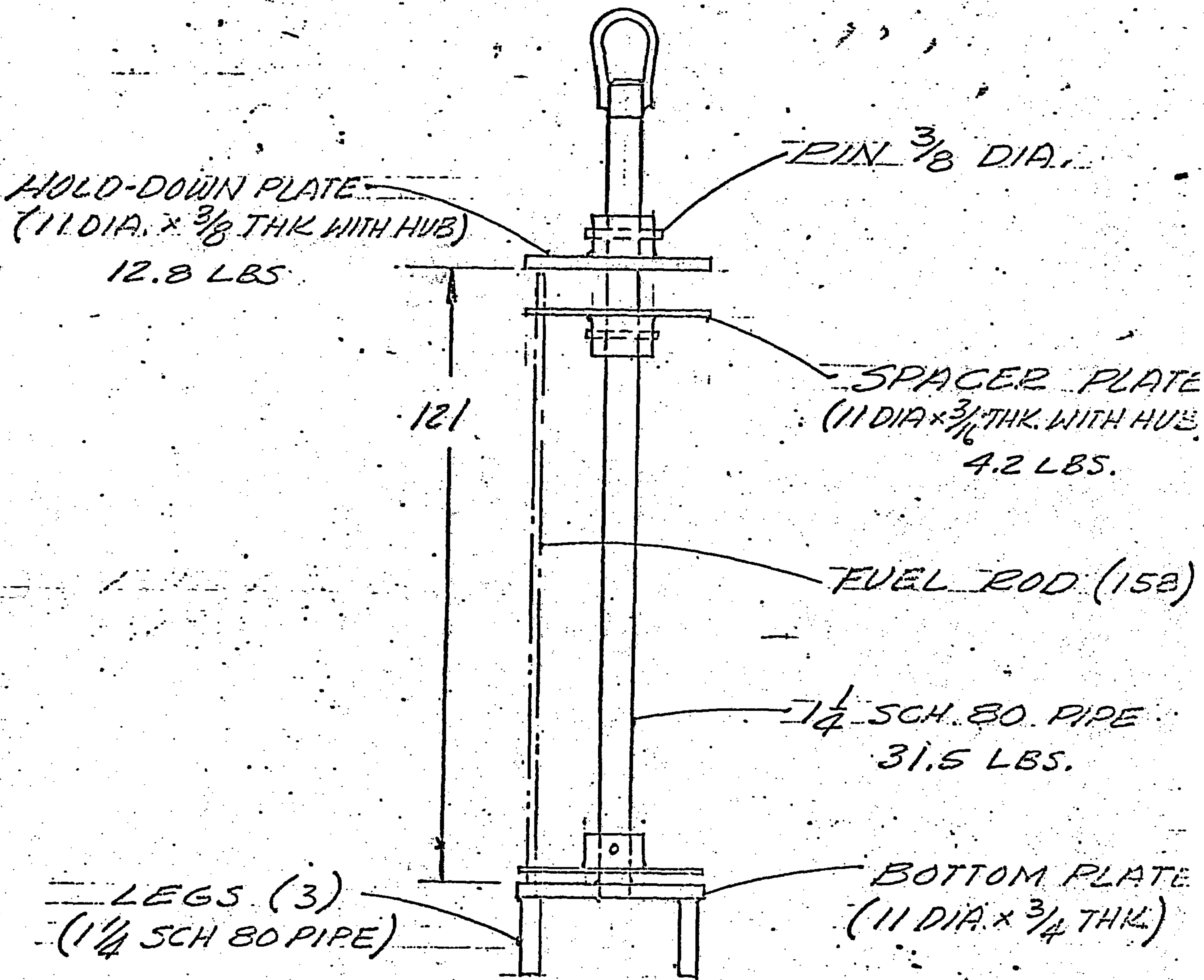
Mark's Handbook for Mechanical Engineers, 7th edition, Pg. 5-62, 5-27 (Figure 22)
Ryerson Catalog #118
ASM Metal Progress Databook

Data

Capacity of fixture - 158 Dresden rods
Net weight of rack - 91 lbs.
Gross weight (158 fuel rods) - 1,434 lbs.
Weight of holddown plate - 12.8 lbs.
Weight of spacer plate - 4.2 lbs.
Weight of center pipe - 31.5 lbs.
Pin 3/8 diameter - tensile strength: 90,000 to 95,000 psi

Conclusion

We find the corrosion rack to be structurally safe based on the following calculations:



DRESDEN CORROSION
RACK - E-302863

ALL STAINLESS STEEL CONST.

CALCULATIONS

POSSIBLE FAILURE DUE TO WEIGHT OF FIXTURE

MAX. S_f OF PIN = 90,000 P.S.I.
E PIPE = 28,000,000 P.S.I.

$$Sd_F = S_S\left(1+\sqrt{1+\frac{2h}{e}}\right)$$ WHERE

Sd_F = DYNAMIC STRESS AT PIN (SHEAR) DUE TO FIXTURE WEIGHT
S_S = STATIC STRESS AT PIN = $\left(\frac{P}{A}\right)$ PIN
h = HEIGHT OF DROP IN INCHES $(12 \times 20) = 240$
e = ELONGATION OF PIPE WITH STATIC LOAD = $\left(\frac{SL}{E}\right)$ PIPE
S PIPE = $\frac{P'}{A'} = \frac{52.7}{.600} = 88$ P.S.I.
P' = WEIGHT OF HOLD DOWN PLATE, SPACER AND PIPE (48.5 LBS)
A' = AREA OF PIPE (.600) SQ IN.
P = WEIGHT OF HOLD DOWN PLATE (12.8 LBS)
A = SHEAR AREA OF PIN (.221) SQ IN.
L PIPE = 121 INCHES
E PIPE = 28×10^6 PSI

$$Sd_F = \frac{12.8}{.221}\left[1+\sqrt{1+\frac{(2)(240)}{\frac{(88)(121)}{(28)(10^6)}}}\right]$$

$$Sd_F = 57.8\left[1+\sqrt{1+\frac{(480)(28)(10^6)}{(81)(121)}}\right.$$

$$Sd_F = 57.8\left(1+\sqrt{1+1.372(10^6)}\right)$$

$$Sd_F = 57.8\left(1+1000\sqrt{1.372}\right)$$

$$Sd_F = (57.8)(1{,}172)$$

$$Sd_F = \underline{\underline{67{,}700 \text{ P.S.I.}}}$$

POSSIBLE FAILURE CAUSED BY FUEL RODS REBOUNDING AFTER IMPACT, ASSUMING n RODS STRIKE TOP PLATE AT SAME INSTANT.

A DROP TEST WAS PERFORMED TO DETERMINE THE MAX. REBOUND HEIGHT, h'. THE VALUE WAS FOUND TO BE 11/16 INCHES REBOUND FROM A 20 FT. DROP. ADDING 1/32 INCHES TO COMPENSATE FOR A POSSIBLE MEASUREMENT ERROR, THE VALUE OF h' BECOMES .72 INCHES.

$$S_d \text{ DUE TO ONE ROD} = S_{dr} = S_{se}\left(1+\sqrt{1+\frac{2h'}{e}}\right)$$

$$S_{dr} = S_{se}\left(1+\sqrt{1+\frac{1.44}{e}}\right)$$

$$S_{se} = \frac{\text{ROD WT.}}{\text{AREA PIN}} = \frac{8.5}{.221} = 38.46 \text{ P.S.I.}$$

$$e = \left(\frac{PL}{AE}\right)_{ROD}$$

A ROD = .052 IN.^{2}

L ROD = 116 INCHES

E ROD = $(11)(10)^6$ PSI

$$e = \frac{(8.5)}{(.0521)}\frac{(.116)}{(11)(10)^6} = (1721)(10)^{-6} \text{ INCHES}$$

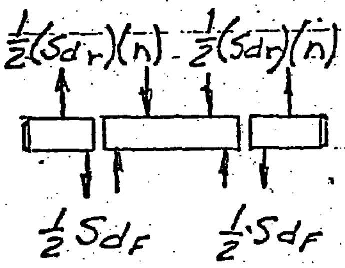
$$S_{dr} = 38.46\left(1+\sqrt{1+\frac{1.44}{(1721)(10)^{-6}}}\right)$$

$$S_{dr} = 38.46\left(1+\sqrt{1+(.0008365)(10)^6}\right)$$

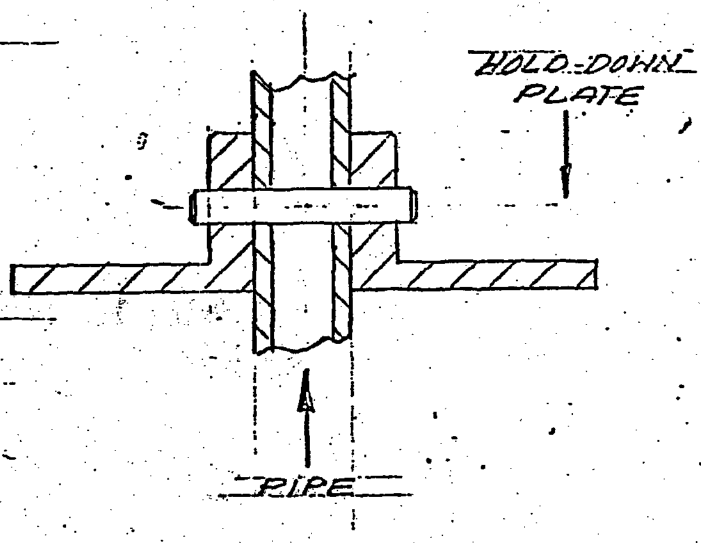
$$S_{dr} = 38.46\left(1+\sqrt{837}\right)$$

$$S_{dr} = 38.46(29.9) = 1150 \text{ P.S.I.}$$

COMBINED DYNAMIC MAX, LOAD, Sdc



SHEAR STRESSES ON PIN



TOTAL FORCES ON HOLD DOWN PLATE

$Sdf = 67{,}700 \uparrow\downarrow$; $Sdr = 1150 \downarrow\uparrow$

TOTAL DYNAMIC STRESS, $Sd_c = Sdf + Sdr(n)$ (WHERE n = No. OF RODS REBOUNDING SIMULTAN.

MAX. ALLOWED $Sde = 67{,}500$ P.S.I. $= .75(90{,}000)$

$$67{,}500 = -67{,}700 + (1150)(n)$$
$$135{,}200 = 1150\, n$$
$$117 = n$$

THE PROBABILITY OF 117 RODS REBOUNDING TO STRIKE THE HOLD DOWN PLATE SIMULTANEOUSLY IS EXTREMELY SMALL.

Addendum I to ENH-71-263

Subject: Structural Analysis of Dresden Corrosion Rack E-302863

E-302863-4 Modification for Yankee Fuel Rods

The rack, as modified, will hold 285 Yankee rods at 3.16 lbs. each. The weight of the rods will be 901 lbs. compared with 1,434 lbs. for 158 Dresden rods.

Since the rack has been shown to be structurally safe for the 158 Dresden rods, it will be structurally safe when loaded with 285 Yankee rods.

CPD recommends structural approval for 285 Yankee rod loading for corrosion rack E-302863-4.

Prepared G. Halla
G. Halla, Process Eng.

Reviewed L. Studwell
L. Studwell, Eng.-In-Charge/Mech. Section

ADDENDUM II

STRUCTURAL ANALYSIS OF DRESDEN CORROSION RACK

DROP TESTING OF YANKEE RELOAD FUEL RODS

I. Description of Test

The bending and bowing effects caused by dropping Yankee-Reload fuel rods were checked by actual drop test. Three different tests were performed on individual dummy rods. The dummy rods were constructed to rod design dimensions but were loaded with stainless steel pellets.

Rods were dropped by hand such that they fell 22 to 25 feet striking a concrete driveway. During each test, the rod was dropped in such a manner to cause impact to occur at an increasing angle.

II. Test Results

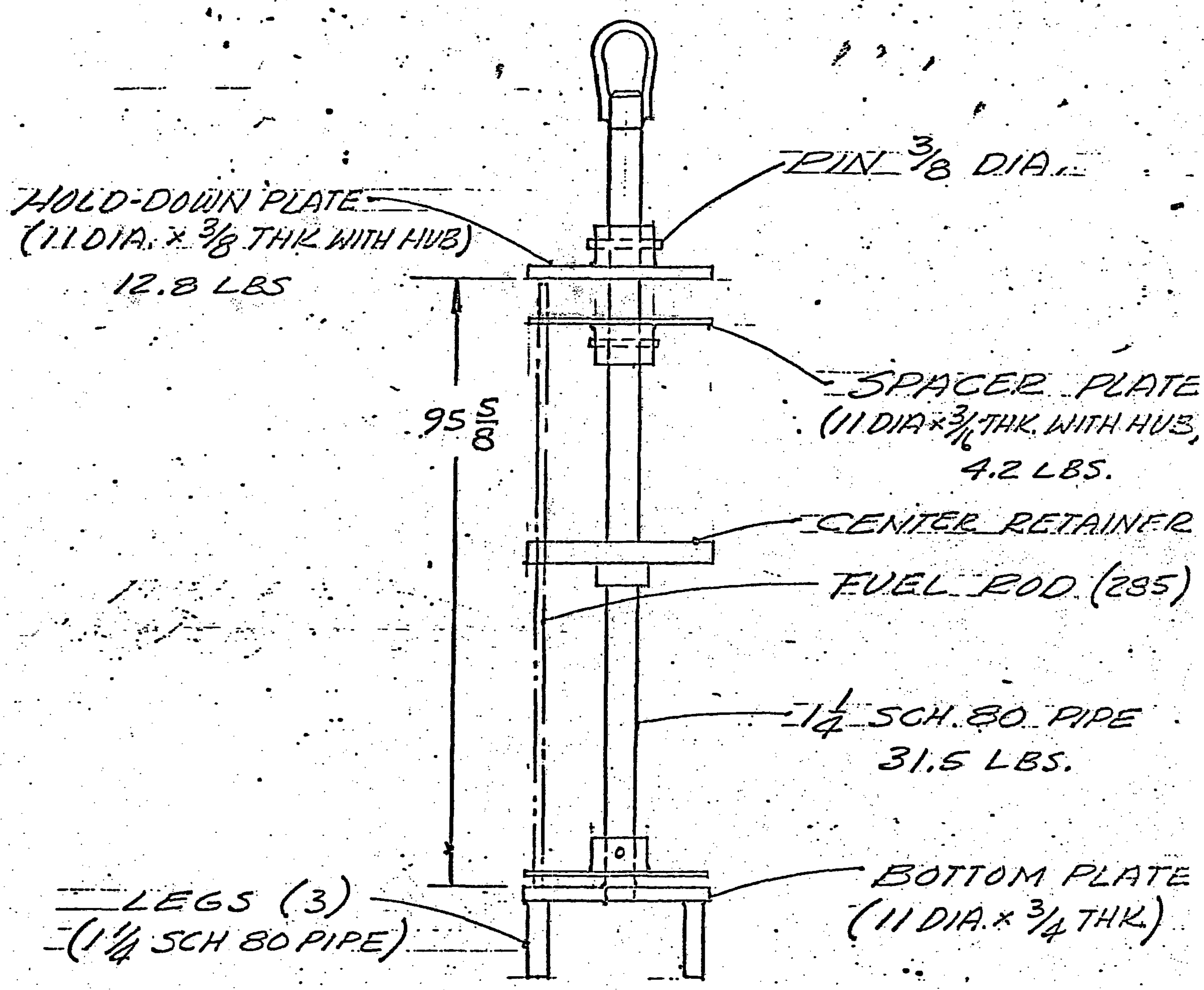
Test No. 1 was a straight drop. The rod struck the impact surface essentially vertically and rebounded directly upward approximately 6 inches. During the rebound the rod was noted to oscillate with maximum deflection of approximately 1/2 inch. Rod damage was confined to some crushing of the bottom end plug.

Test No. 2 was performed with the rod at an approximate 5° angle. After striking the impact surface, the rod rebound upward and at an angle in the direction of its axis. Rebound distance was approximately 2 to 3 inches vertically. During rebound the rod oscillated approximately 1 to 1 1/2 inches. Rod damage noted was again some crushing of the bottom end plug plus a permanent bend of approximately 1/4 inch starting about 5 inches from the point of impact.

Test No. 3 was performed with the rod at an approximately 30° angle. After striking the impact surface, the impact end deflected upward violently and the upper end struck the impact surface rather heavily. After both impacts, the rod oscillated violently approximately 4 to 6 inches. Rod damage was extensive with top and bottom bows of approximately 9 inches. These bends were located approximately 1/3 of the distance from the top and the bottom.

III. Conclusions

The most likely accidental drop of these rods held in the Pickle-Corrosion Test Fixture would be vertical. This is assumed, since the fixture weight results in nearly vertical orientation during movement plus the openings of corrosion vessels, pickle tanks and rinse tanks are sufficiently small to preclude angular dropping of a loaded fixture. Therefore, a drop of the fixture would result in the vertical displacement of the rods upon impact. This is further assured since the rods are not attached but rest on the fixture bottom plate and the rods are constrained by individual openings in the top and bottom spacer plates of the fixture plus a centra retainer which restricts both outward and inward deflection. Thus, the results of Test No. 1 best describes the expected effects of an accidental fixture drop on the individual rods contained therein.



DRESDEN CORROSION RACK - AS REVISED FOR USE WITH YANKEE RODS E-302863-4

ALL STAINLESS STEEL CONST.

4 pages
Withheld in Entirety
Ex. 4

UNITED NUCLEAR • INTER-OFFICE MEMO
CORPORATION

NDEO-1033

TO	R. E. Kropp	AT	New Haven	DATE	May 19, 1967
FROM	L. Goldstein	AT	Elmsford	COPY TO	W. Compas-New Ha D. Cronin " " E. Krinick " " G. Sofer B. Teer P. Buck R. Tomonto
SUBJECT	Nuclear Information Required for Dresden Shipping Container Design				

As per your request, nuclear information for Dresden Type 6 fuel elements is enclosed. Basic physics data and their calculational backup are presented first, followed by criticality safety data.

Physics Data and Backup

The basic physics data for the three types of United Nuclear fuel elements are given in Table 1 (as per DRC-2 dated 5/10/67). All evaluations were made for clean assemblies at 68°F.

Table 1 -

Fuel Element Type	Average Enrichment w/o U-235	k_{∞}	M^2
6C	2.24	1.210	39.0
6P	2.24	1.310	41.0
6I	2.24	1.310	41.0
3-F (G.E.)	2.24	1.210	39.0

The above data were computed using the LEOPARD[1] zero dimensional cross section calculation and depletion program. The code determines fast and thermal spectra using only basic fuel assembly geometry and temperature data, based on a modified FORM-TEMPEST slowing down-thermalization model. A correlation of the U^{238} resonance integral data is incorporated in the code to correct for resonance spatial and energy self-shielding effects. The multigroup Amouyal-Benoist method is used to calculate thermal self-shielding. The code computes fuel depletion effects for an infinite homogeneous reactor and recomputes the spectra and corresponding four-group cross sections before each discrete burnup step. All basic microscopic cross-section data are contained within the LEOPARD library, which was compiled by WAPD based on reported data. The LEOPARD cross sections do not contain any arbitrary adjustments to improve agreement with critical experiments.

The LEOPARD program is used routinely at UNC-REC as a basic tool for the generation of depletion dependent reactor constants for both PWRs and BWRs. The code has been checked extensively at REC against PWR and BWR experimental data, as well as against Monte Carlo calculations. Comparison with more than fifty oxide fueled critical and exponential experiments was performed by WAPD[2]. Both REC and WAPD evaluations of the LEOPARD program showed excellent agreement ($\pm 0.4\%$ in k_{eff}) with experiments in all cases.

Furthermore, when LEOPARD data are used to develop depletion dependent constants for use in the UNC-TRILUX fuel management program, excellent agreement is obtained with operational fuel cycle lengths and power distributions for both BWRs and PWRs. Fuel cycle lengths calculated by TRILUX with LEOPARD derived constants are within 5% of those actually achieved in Cycles 1 and 4 of Dresden 1, and also within 5% of the cycle length expected for Cycle 1 of the Trino Vercellese PWR. Calculated element-wise radial power distributions for Cycle 1 of the Trino PWR at about 4000 MWD/MTU deviate by less than 2% relative to experimental power distributions derived from Mn^{56} activations.

Criticality Safety Data

UNC Type 6C fuel is neutronically equivalent to the G.E. Type 3-F assembly. Furthermore, the nuclear properties of UNC Type 6I and 6P are neutronically equivalent to the G.E. Type 3-F with its gadolinia poison rod removed. Commonwealth Edison Company in Addenda (dated March 2, 1965) to Exhibit I in Submittal, dated December 24, 1964, for Amendment of Appendix A of License DPR-2 to Permit Operation with Type 3-F Fuel-Change Number 10; Docket 50-10; addendum Section V Safety Evaluation, reports minimum critical measurement data for Type 3-F fuel performed at Vallecitos by G.E. These data are given in Table 2.

Table 2 -

G.E. Fuel Type	Corresponding UNC Fuel Type	Minimum Number of Elements to Reach Criticality
3-F	6C	17
3-F Gd poison removed	6I, 6P	14

L. Goldstein

L. Goldstein

References:

1. WCAP-3269-26, LEOPARD-A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094, R. F. Barry, Sept. 1963.

2. WCAP-3269-25, "Calculation of Lattice Parameters and Criticality for Uniform Water Moderated Lattices", L. E. Strawbridge, September 1963.

12/9/67

UNITED NUCLEAR CORPORATION · INTER-OFFICE MEMO

NDEO-1077

TO F. Cronin ✓ AT New Haven DATE June 21, 1967

FROM J. R. Tomonto AT Elmsford COPY TO G. Sofer
J. O'Toole

SUBJECT Criticality of Dresden Fuel Rods During Corrosion and Pickling Operations

Summary

An analysis has been performed to determine the criticality safety limits for the Dresden regular enrichment UO_2 fuel rods during corrosion and pickling operations. The Dresden regular enrichment fuel rods analyzed were located in a maximum of five concentric rings around a central water region (OD = 3.88 in.) as shown in Dwg. D-302863-2. The outer ring of rods is located in a 10" diameter bolt circle. This pickling rack can be located in several types of autoclaves (ID from 11" to 20"). Therefore the approach was to determine that individual autoclaves would be subcritical when fully reflected for all possible conditions of partial to full loading of the pickling rack fuel rod locations and for variations in water density caused by heating and possible boiling.

The results of this analysis indicate:

1. The maximum k_∞ for a repeating array of Dresden regular enrichment fuel rods in water (T = 70°F) is 1.330 and occurs at a water to fuel volume ratio of 2.20.

2. The maximum k_{eff} for the fully reflected rack loaded with Dresden regular fuel rods is 0.854 and occurs at a water to fuel volume ratio of 2.2.

3. Removing either the center or outermost ring of Dresden regular fuel rods from the fully reflected pickling rack reduces k_{eff} to 0.765 (water/fuel volume ratio 2.20).

4. The k_{eff} of the bare but fully reflected and moderated pickling rack is 0.648 and occurs at a water/fuel volume ratio of 2.5.

Method of Analysis

Calculations of the fueled regions assumed a regular array of regular enrichment Dresden 1 reload fuel rods in water (T = 70°F). The Dresden regular fuel rods were selected because they are the most reactive of the three types of rods used in Type 6 Dresden reload assemblies. The fuel is 2.34 w/o enriched UO_2 pellets, (OD = 0.482") density 10.34 gm UO_2/cm^3, in Zircaloy tubing (ID = 0.4925", OD = 0.5625).

Calculations of k_∞ were performed using the LASER[1] code. This code is a one dimensional (cylindrical), multi-energy (85 groups) lattice-cell program which is based on the MUFT and THERMOS codes. Fast group parameters (E > 1.85 ev) are averaged over a semi-infinite medium spectrum calculated by the MUFT method. A correlation of the U^{238} resonance integral data is incorporated in the code to correct for resonance spatial and energy self-shielding effects. Thermal group constants were determined for a one dimensional THERMOS type calculation using the NELKIN scattering kernel. All basic microscopic cross section data are contained within the LASER library, which was compiled by WAPD based on recently reported experimental data. The LASER cross sections and calculational method have been extensively tested with experimental data and do not require any arbitrary adjustments to improve agreement with critical experiments[3].

Calculations of the pickling rack with internal and external water regions were performed using the AIM-6 one dimensional diffusion theory code with two neutron energy groups. Cross sections for the reflectors and autoclave wall were calculated with the FORM-TEMPEST codes.

Results

The calculated variation of k_∞ for a repeating array of Dresden 1 regular fuel rods is shown in Figure 1. A point noted on this figure corresponds to the calculated k_∞ for a completed Type 6P or 6I assembly[2] (having an average enrichment of 2.24). The maximum k_∞ of 1.33 occurs at a water to fuel ratio of 2.2.

Calculations were performed to determine k_{eff} of pickling racks when loaded into autoclaves and fully reflected. Because of the possibility of using several sizes of autoclaves, the calculations conservatively neglected the effects of the autoclave wall (t ≈ 2"). The calculations were performed for an infinite cylinder containing the pickling rack and surrounded by a thick water reflector. The maximum calculated k_{eff} of 0.854 was found at a water to fuel volume ratio of 2.2 as shown in Figure 2.

In order to assess the potential positive reactivity associated with changes in the moderating properties of the central water region, a calculation was performed in which the absorption in water of the central region was neglected. The k_{eff}

for this condition was 0.919 for a water to fuel ratio of 2.2 in the fuel region and a thick outer water reflector. It is concluded that the maximum reactivity change that can be associated with loss of water from the central region is +7.5% Δk_{eff}. This is an upper limit on the worth of water in the central region.

Calculations were performed to determine the loss of reactivity when the inner or outermost ring of fuel rods were removed from the pickling rack. The calculated k_{eff} for both conditions was 0.765 for a water to fuel ratio of 2.2 in the fuel region and with a thick water reflector surrounding the pickling rack.

In analyzing the interaction between adjacent fully loaded autoclaves, at optimum internal moderation, the following situations are applicable:

1. Full water flooding between autoclaves - In this situation the presence of water essentially isolates each autoclave and the maximum k_{eff} is that calculated for the fully reflected pickling rack (e.g., from Figure 2, $k_{eff} = 0.854$).

2. No moderator between autoclaves - If the pickling rack is located in a 20" ID autoclave, it will be essentially fully reflected and therefore isolated from the other units ($k_{eff} \sim 0.86$). If the rack is located in a 14" ID autoclave, the calculated k_{eff} of an individual unit is 0.804. If the rack is located in an 11" ID autoclave, k_{eff} will be less than 0.8 because of the lack of reflector. In the latter two cases, there is a possibility of interaction between adjacent units even though the neutron flux is attenuated by the autoclave wall ($\sim$2" of iron).

The maximum k_{eff} of the pickling rack internally moderated but unreflected was calculated as 0.648.

References

1. WCAP-6073, "LASER - A Depletion Program for Lattice Calculations Based on MUFT and THERMOS", by C. G. Poncelet, April 1966.

2. Letter to R. E. Kropp from L. Goldstein, "Nuclear Information Required for Dresden Shipping Container Design", NDEO-1033, May 19, 1967.

3. Eich, W. J., "Analysis of PuO_2-UO_2 Critical Experiments", Trans. American Nuclear Society, Vol. 10-1 (June 1967).

James R. Tomonto
J. R. Tomonto

JRT:jk

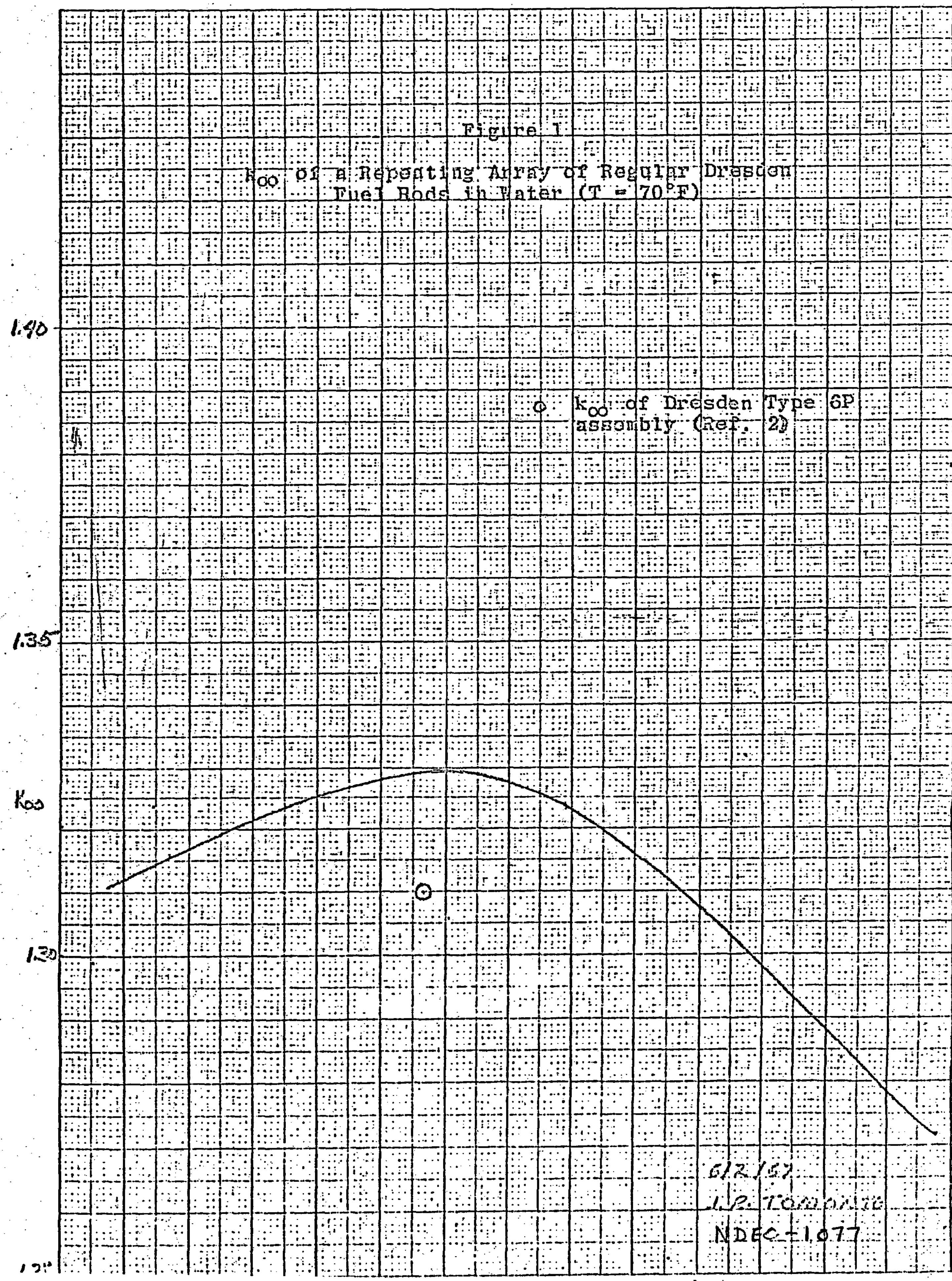


Figure 1
k_{∞} of a Repeating Array of Regular Dresden Fuel Rods in Water (T = 70°F)
1.40
1.35
k_{∞}
1.30
⊙ k_{∞} of Dresden Type 6P assembly (Ref. 2)
6/2/67
NDEC-1077

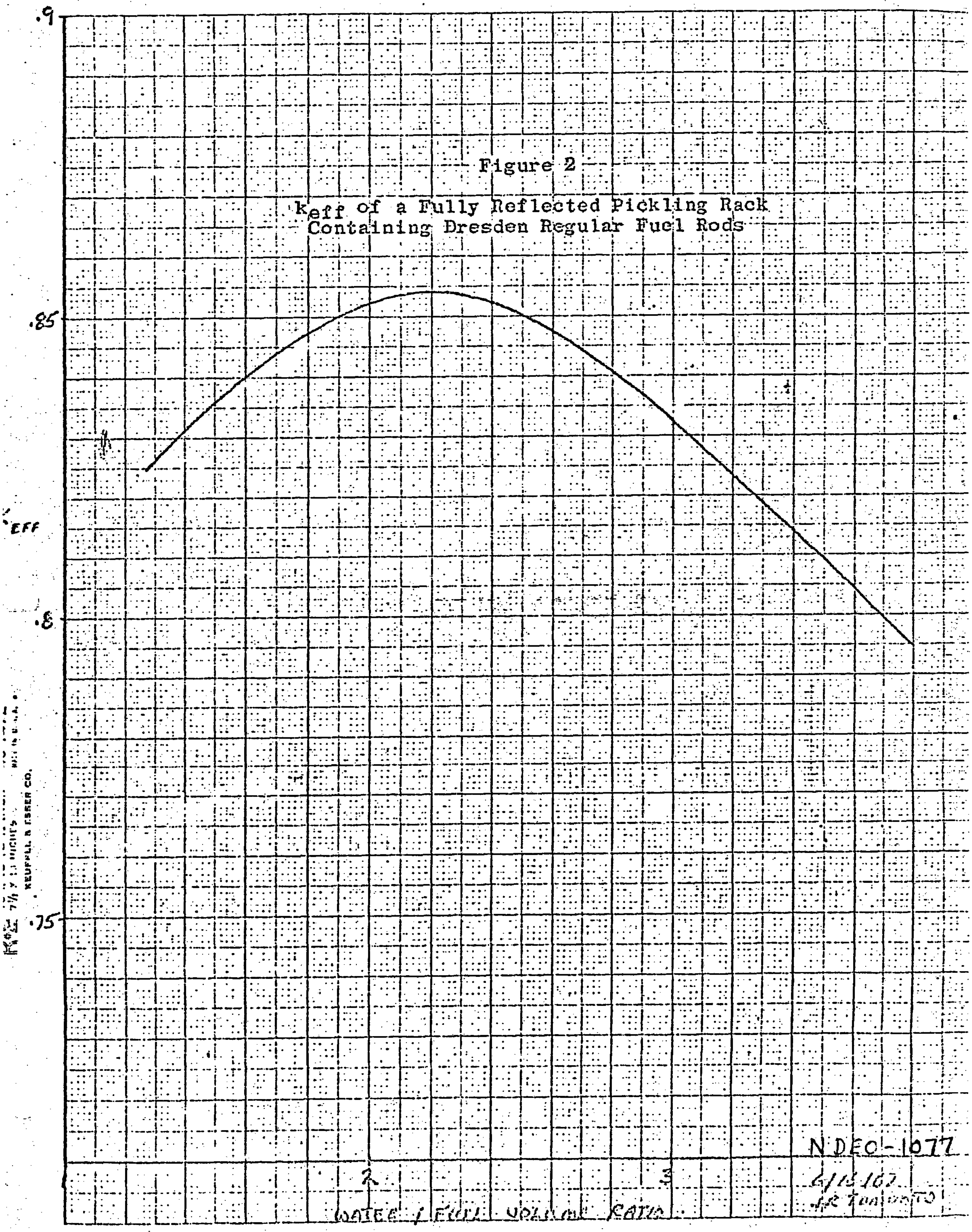


Figure 2

k_{eff} of a Fully Reflected Pickling Rack Containing Dresden Regular Fuel Rods

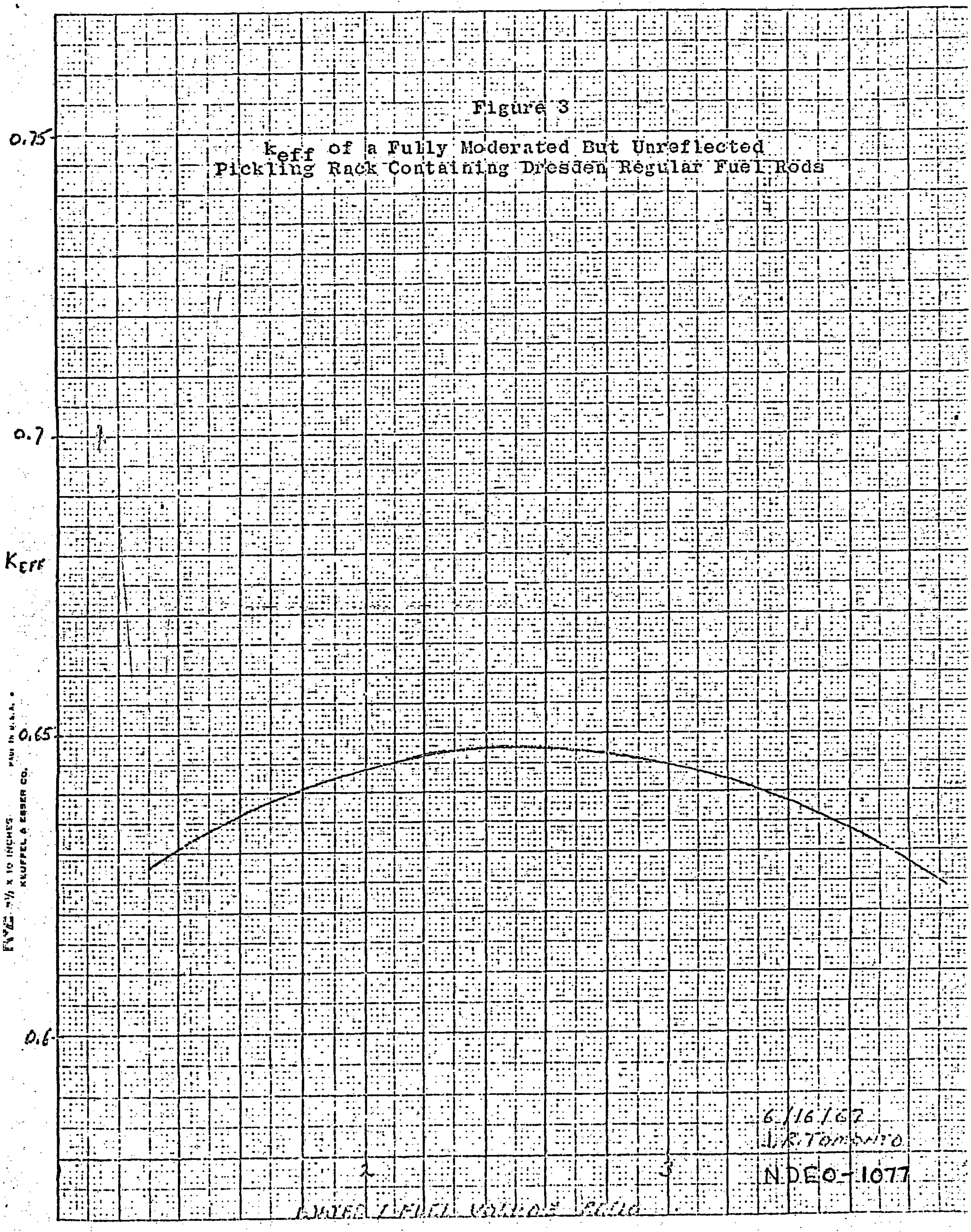


Figure 3

k_{eff} of a Fully Moderated But Unreflected Pickling Rack Containing Dresden Regular Fuel Rods

UNITED NUCLEAR
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NDEO-1134

TO R. E. Kropp ✓ AT New Haven DATE August 15, 1967

FROM L. Goldstein AT Elmsford COPY TO W. Compas
D. Cronin
E. Krinik
G. Sofer
M. Labar
R. Tomonto
P. Buck

SUBJECT Nuclear Information Required For Shipping Yankee Fuel Bearing Components.

As per your request, nuclear information for Yankee fuel elements are given below. The k_∞ and M^2 data were obtained by the methods described in NDEO-1033.* The number of assemblies required to reach criticality was determined from the basic physics data (k_∞&M^2) by calculating the critical buckling from which the dimensions of a fully reflected system were evaluated.

The required information for the Yankee fuel elements at 68°F is as follows:

Average Enrichment w/oU-235	k_∞	M^2	Minimun Number of Elements to Reach Criticality:
3.50	1.405	40.9	>3

For these elements, the k_{eff} of a fully reflected 2 element array is 0.921.

L. Goldstein (LH)
L. Goldstein

LG/lh

*NDEO-1033, Nuclear Information Required for Dresden Shipping Container Design, L. Goldstein, May 19, 1967.

UNITED NUCLEAR
C O R P O R A T I O N

• INTER-OFFICE MEMO

NDEO-1164

TO R. E. Kropp AT New Haven DATE Sept. 11, 1967

FROM L. Goldstein AT Elmsford COPY TO Distribution

SUBJECT Additional Information Required for Shipping Dresden I and Yankee Rowe Fuel Elements

Reference: NDEO-1033 and NDEO-1134

This memo is to confirm the numbers given by L. Goldstein to R. E. Kropp in a telephone conversation of 9/11/67 concerning H/U^{235} ratios inside the shroud for Dresden I and Yankee Rowe fuel elements. The data are as follows:

Fuel Element Type	H/U^{235}
Dresden I	165
Yankee Rowe	121

Data are for cold, clean, 0 void condition.

L. Goldstein

L. Goldstein

LG:JK

Distribution

W. Compas
D. Cronin
E. Krinik
G. Sofer
M. LaBar
J. R. Tomonto
P. Buck

INTER-OFFICE MEMO

UNITED NUCLEAR CORPORATION

NDEO-1359

TO F. Cronin AT DATE March 22, 1968

FROM J. R. Tomonto AT COPY TO

SUBJECT REACTIVITY CONTRIBUTION OF CLADDING IN DRESDEN 6P AND PATHFINDER FUEL RODS

References 1 and 2 present the results of a criticality safety evaluation of Dresden and Pathfinder fuel rods during corrosion and pickling operations. These evaluations were based on the most reactive arrangement of Zr clad fuel rods. The AEC has requested further information relating to the effort of cladding desolution during pickling operations. The following items apply:

(1) If some of the Zr clad is disolved, it would most likely remain in solution in the vicinity of the fuel rods and there would be a slight decrease in k_∞ of the lattice because of the reduced Zr thermal self-shielding factor.

(2) If the Zr clad is completely disolved, the fuel rods will collapse and fall to the bottom of the autoclave. This configuration would be less reactive than the configurations analyzed.

(3) The reactivity controlled by Zr absorptions is $+0.010\ \Delta k_\infty/k_\infty$ for the Dresden rods and $+.012\ \Delta k_\infty/k_\infty$ for the Pathfinder rods at optimum moderation.

James R. Tomonto
J. R. Tomonto

JRT:jk

NDEO-1359

References

1. NDEO-1077, "Criticality of Dresden Fuel Rods During Corrosion and Pickling Operations", by J. R. Tomonto, 6/21/67.

2. NDEO-834, "Criticality of Pathfinder Fuel Rods During Corrosion and Pickling Operations", by J. R. Tomonto, 11/11/66.

GULF UNITED
NUCLEAR FUELS CORPORATION

I. BASIC INFORMATION

A. Pellet Description: 0.3835" ϕ x 0.37" - 0.47" long

B. Rod Description: (nom. values)

1.

Type Rod	Fuel Stack Length	Tube Length	Fuel Loading/Rod
Standard	121.5"	126.7"	2295 g. UO_2 (80.9 g. U-235)

2. Stainless Steel Tubing (nom. values)

Std. Tube: 0.0165" W.T. x 0.422" O.D. x 0.389" I.D. x 125.9" long

C. Element Description: (nom. values)

1. Envelope - 8.449" x 8.449" x 137.7"
2. Length Between Nozzles - 127.07"
3. Active Fuel Length - 121.5"
4. Fuel Rods/Element - 204
5. Array - 15 x 15
6. Pitch - 0.563"
7. U-235/Element - 16.404 kg. (17.225 max.)

D. General Information

1. Enrichment - 4.0 w/o
2. Pellet Density - 93% theoretical

II. ASSUMPTIONS

A. The design reactivity information contained in NED-2053 is applicable and will be used to determine effective multiplication factors.

(H/U-235 - 133)
(K_{∞} = 1.3597)
($M^2 = 35.6649\ cm^2$)

B. Calculations will be performed using the following equations:

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

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

License: SNM-33 Docket: 70-36 Section: 900 Subsection/Subpart: 923

Subject: Nuclear Safety Evaluation - Connecticut Yankee Fuel (PWR)

Issued: 11/28/72 Supersedes: New Approved: ________

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

D. From Fig. 3 and 4 of TID-7028, an extrapolation length (also designated δ) of 3.0 cm. was selected for bare, moderated systems. This value 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 3 to 4 cm. will yield minor reactivity changes, probably 10% or less in K_{eff}.

E. When multiple elements are evaluated, a 0.75" (1.91 cm.) gap between elements is assumed. Under moderated conditions, this will yield higher K_{eff} values than would occur if the elements were closely packed.

III. CALCULATIONS

A. One Element - envelope = 21.46 cm. x 21.46 cm. x 308.61 cm

1. Reflected Case:

$$B_R^2 = \frac{9.87}{(21.46 + 16)^2} + \frac{9.87}{(21.46+16)^2} + \frac{9.87}{(308.61+16)^2} = \frac{9.87}{(37.46)^2} + \frac{9.87}{(37.46)^2} + \frac{9.8}{(324.6}$$

$$= \frac{9.87}{1403} + \frac{9.87}{1403} + \frac{9.87}{105372} = 0.0070 + 0.0070 + 0.0001 = 0.0141 \text{ cm.}^{-2}$$

$$K_{eff} = \frac{1.3597}{1+(35.66\text{x}0.0141)} = \frac{1.3597}{1+0.5028} = \frac{1.3597}{1.5028} = 0.905$$

2. Bare Case:

$$B_B^2 = \frac{9.87}{(21.46+6)^2} + \frac{9.87}{(21.46+6)^2} + \frac{9.87}{(308.61+6)^2} = \frac{9.87}{(27.46)^2} + \frac{9.87}{(27.46)^2} + \frac{9.87}{(314.61)^2}$$

$$= \frac{9.87}{754} + \frac{9.87}{754} + \frac{9.87}{98979} = 0.0131 + 0.0131 + 0.0001 = 0.0263 \text{ cm.}^{-2}$$

$$K_{eff} = \frac{1.3597}{1+(35.66\text{x}0.0263)} = \frac{1.3597}{1+0.9379} = \frac{1.3597}{1.9379} = 0.702$$

B. Two Elements - envelope = (21.46 + 1.91 + 21.46 = 44.83) = 44.83 CM x 21.46 CM x 308.61 cm)

1. Reflected Case:

$$B_R^2 = \frac{9.87}{(44.83+16)^2} + \frac{9.87}{(21.46+16)^2} + \frac{9.87}{(308.61+16)^2} = \frac{9.87}{(60.83)^2} + 0.0070+0.0001$$

$$= \frac{9.87}{3700} + 0.0070 + 0.0001 = 0.0027+0.0070+0.0001 = 0.0098 \text{ CM}^{-2}$$

$$k_{eff} = \frac{1.3597}{1+(35.66 \times 0.0098)} \quad \frac{1.3597}{1+0.349.5} \quad \frac{1.3597}{1.3495} = 1.008$$

2. Bare Case:

$$B_B^2 \doteq \frac{9.87}{(44.83+6)^2} + \frac{9.87}{(21.46+6)^2} + \frac{9.87}{(308.61+6)^2} = \frac{9.87}{(50.83)2} + 0.0131+0.0001$$

$$= \frac{9.87}{2584} + 0.0131+0.001 = 0.0038+0.0131+0.0001 = 0.0170 \text{ CM}^{-2}$$

$$k_{eff} = \frac{1.3597}{1+(35.66 \times 0.0170)} = \frac{1.3597}{1.+0.6062} = \frac{1.3597}{1.6062} = 0.847$$

IV. Conclusions

A. These calculations indicate that 2 elements would be close to critical.

B. Although 1 element exceeds a k_{eff} (bare) $\leq$ 0.65, a criticality limit of 1 element may be used.

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INTER-OFFICE MEMO

NED-2053

TO R. Kropp AT DATE October 19, 1972

FROM R. Weader AT COPY TO G. Hamilton
J. R. Tomonto

SUBJECT Nuclear Properties of Connecticut Yankee 4.0 w/o Stainless Steel Clad Fuel

A LL-2 problem was run to represent Connecticut Yankee 4.0 w/o stainless steel clad fuel at 68 °F in unborated water. The following results were obtained:

$$K_\infty = 1.3597$$

$$M^2 = 35.6649$$

Richard R. Weader
R. Weader

RW/cc

UNITED NUCLEAR
CORPORATION

Approved

ISSUED October 31, 1968

SUPERSEDES NEW

LICENSE: SNM-777, Docket: 70-820
SECTION: 800 - FUEL FABRICATION OPERATION
Subsection: 820 - Processing
Subpart: 824 - Fabrication of Enriched Uranium Metal
824.1 - General Considerations

824.1 General Considerations

1. This subsection covers the preparation and processing of up to and including fully enriched uranium metal from feed material through chopped stock (approximately 1/8" cubes). Individual operations which may be performed are shown on Diagram 824.I and are listed in their expected sequence.

2. All operations normally will be performed in Building 19H unless otherwise noted.

3. Unless stated otherwise, safe values referenced in the following Subpart will be obtained using the limits listed in Table 309-I.

4. For this undiluted uranium metal in which the U235 content is less than 93%, the U235 mass limit may be increased by the appropriate factor from Figure 20, TID-7016, Rev. 1.

LICENSE:	SNM-777, Docket: 70-820	Approved
SECTION:	800 - FUEL FABRICATION OPERATION	ISSUED October 31, 1968
Subsection:	820 - Processing	
Subpart:	824 - Fabrication of Enriched Uranium Metal	SUPERSEDES NEW
	824.2 - Process Description	

824.2 Process Description

1. Receive Enriched Uranium
Enriched uranium will be received and hendled in accordance with Subsection 810.

2. Enrichment Check
If required, an enrichment verification will be performed, usually by auto gamma spectrometry. The limits for this operation will be 10 kg. U-235 in one piece or 350 grams U-235 in small pieces.

3. Charge Preparation
Same considerations as Subpart 822.2.6.

4. Load and Evacuate Furnace
Same considerations at Subpart 822.2.6.

5. Melt and Cast Ingot
Same considerations as Subpart 822.2.6 and 7.

6. Remove Flash and Weigh Ingot
Same considerations as Subpart 822.2.8.

7. Heat Treat Ingot
The cropped casting will be heat treated by alternately immersing it in a molten chloride salt bath and a water quench. Only a single casting (approx. 6.95 kg. of U-235) will be present in the heat treatment area at any time. The casting will be dried and weighed at the end of the heat treatment and the weight loss noted so as to provide a record of the uranium content of the heat treatment baths. The total uranium content of billet and uranium in the solution of the heat treatment chloride bath will not be permitted to exceed 10kg. of U-235. This is a rigidly controlled operation which requires quenching of the piece as it is removed from the heat treatment bath. Quenching will be done in a water tank located approximately two feet from the edge of the salt bath. There will be a sheet metal hood over the quench tank with a flexible hose connection to the main exhaust system. The tank volume in relation to the size of the piece is such that there will be a minimum of vaporization or "plume" effect associated with this operation.

LICENSE: SNM-777, Docket: 70-820
SECTION: 800 - FUEL FABRICATION OPERATION
Subsection: 820- Processing
Subpart: 824 - Fabrication of Enriched Uranium Metal
824.2 - Process Description

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

LICENSE: SNM-777, Docket: 70-820
SECTION: 800 - FUEL FABRICATION OPERATION
Subsection: 820 - Processing
Subpart: 824 - Fabrication of Enriched Uranium Metal
824.2 - Process Description

Approvad

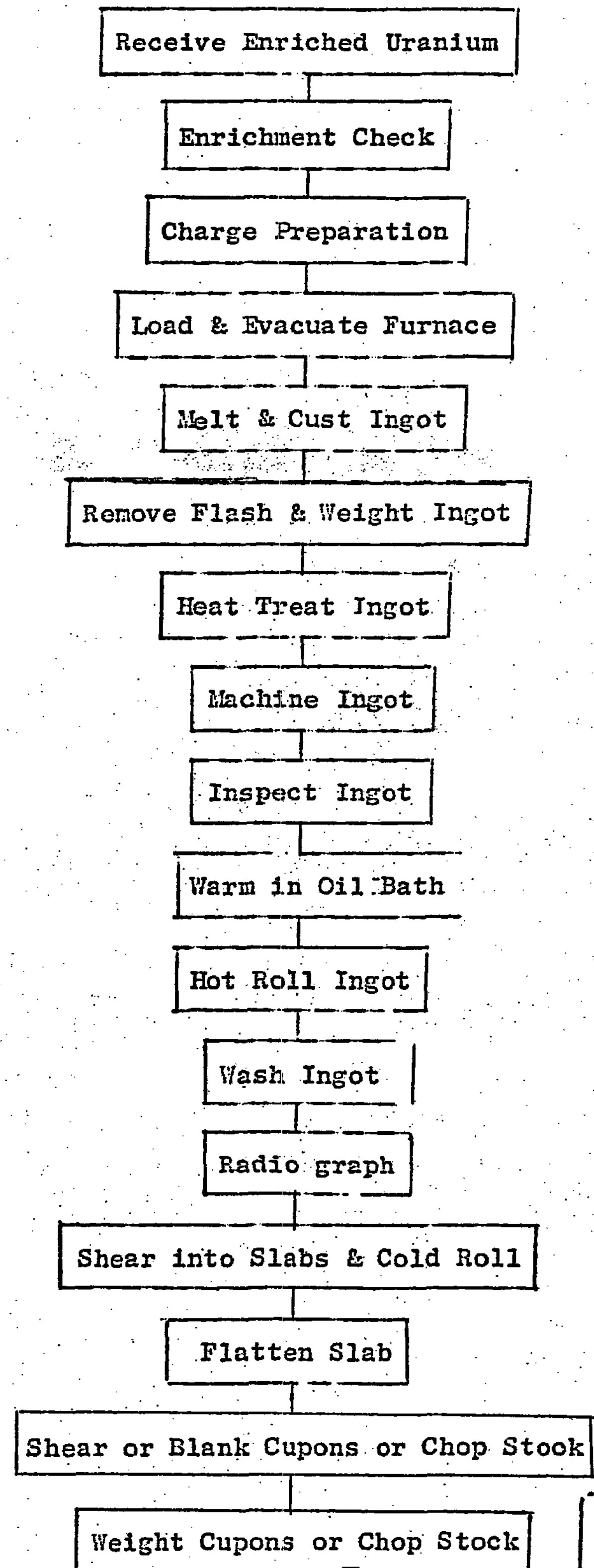
ISSUED October 31, 1968

SUPERSEDES NEW

15. Platten Slab
Same considerations as Subparts 822.2.11 and 12.

16. Shear or Blank Coupons or Chop Stock
Slabs will be sheared or blanked one at a time. The coupons or the chop stock will be collected in a safe volume container which will be located at least one foot from the slab being processed.

17. Weigh Cupons or Chop Stock
One safe volume container will be weighed at a time. If cupons require weighing, they will be weighed individually and returned to their safe volume container.



LICENSE: SNM-777, Docket: 70-82 SECTION: 800, Subpart 824
U Metal Process Flow
Diagram 824-I
APPROVED:
ISSUED: October 31, 1968
SUPERSEDES: New

UNITED NUCLEAR CORPORATION	PAGE 1 OF 1
LICENSE: SNM-777; DOCKET: 70-820 SECTION: 800 - FUEL FABRICATION OPERATION Subsection: 820 - Processing Subpart: 825 - Fabrication of Uranium Aluminum Inter Metallic Cores 825.1 - General Considerations	APPROVED ISSUED 6/25/71 SUPERSEDES New

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 .51. 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-2098. 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.

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ISSUED October 31, 1968

SUPERSEDES New

LICENSE: SNM-777; Docket: 70-820
SECTION: 800 - FUEL FABRICATION OPERATION
Subsection: 820 - Processing
Subpart: 825 - Fabrication of Uranium Aluminum Inter Metallic Cores
825.2 - Process Description

825.2 Process Description

1. Receive UAL_x Powders

   UAL_x will be recieved and stored in accordance with Subsection 810.

2. Prepare UAL_x Powders

   One can or bottle of UAL_x will be removed from storage and placed in a low oxygen atmosphere dry box. Contents of this can or bottle plus residues left in a one liter maximum volume tray from previous screening are then emptied into the screen and sieved. While still in this dry box, the sieved material will then be split into four equal portions by use of a sample splitter. Each portion is then loaded into an individual quart capacity V-Blender and the V-Blender is colsed for subsequent blending.

2.2 Nuclear Safety

These process steps are performed in a dry box with a controlled atmosphere. There is no water or other moderating material connected to or used in the dry box. Therefore, the H/X will be less than 2 and the operation is considered dry. The dry box will be limited to 10 kgs U-235. From Fig. 1, TID-7016, Rev. 1, 10 kgs U-235 with an H/X less than 2 is safe. In addition, each piece of process equipment in the dry box (e.g., UAL_X container, screens plus receiver, splitter, trays, etc.) will have either a 5" dia. or 1 gallon capacity maximum, and will be limited to 5 kgs U-235. From Figures 2 and 3, TID-7016, Rev. 1, a 1 gallon capacity and a 5" diameter are safe for uranium mixtures with U-235 densities not exceeding 2 kgs/liter. From Figure 1, TID-7016, Rev. 1, a 5 kgs U-235 mass limit is safe for uranium mixtures greater than 2 kgs/liter.

3. Blend UAL_X Powders

3.1 Equipment

Two closed V-Blenders are removed from the dry box, placed on a blending yoke and blended. A small rack is provided on the table holding the yoke for holding two additional sets of two blenders.

3.2 Nuclear Safety

V-Blenders have a volume of 1.9 liters. V-Blenders will be handled and stored in groups of two. Therefore, each group of two blenders will have a 1 gallon capacity. From Fig. 2, TID-7016, Rev. 1, a 1 gallon capacity is safe for uranium mixtures with U-235 densities not exceeding 2 kg/liter. Each group of two V-Blenders will be separated from other UAL_X materials by at least 12". The interaction between the three groups of two blenders will be .368 steradians.

Assume the one gallon capacity is in a cylindrical shape with L/D = 1,

$$V = 1 \text{ gal.} = 3.785 \text{ liters} = \frac{\pi d^2 h}{4} = \frac{\pi d^3}{4}$$

$$d = \sqrt[3]{\frac{4V}{\pi}} = \sqrt[3]{\frac{4 \times 3785 \text{ cc}}{\pi}} = \sqrt[3]{4819 \text{ cc}} = 16.9 \text{ cm} = 6.65''$$

$$h = 12'' + \frac{6.65''}{2} = 15.325'', \quad L = 6.65'', \quad L/2 = 3.325''$$

$$\Omega_3 = \frac{2d}{h} \sin \theta \qquad \text{where } \tan \theta = \frac{L/2}{h} = \frac{3.325''}{15.325''} = .217$$

$$= \frac{13.30''}{15.325''} (.212) \qquad \sin \theta = .212$$

$$= .184$$

Since the centermost group sees two other groups of blenders

Ω_3 (Total) = $2 \times \Omega_3 = 2 \times .184$ - .368 steradians

From Table XVII, K-1019, Rev. 5, a 1 gallon capacity has an allowable interaction of 1.9 steradians.

4. Prepare Changes

4.1. Equipment

When all the UAL_X material has been blended, four V-Blenders are placed in a second low oxygen atmosphere dry box. Blenders are again moved two at a time. The contents of each V-Blender is emptied into a tray. Small amounts of material are removed from the tray and placed in charge containers. Charge containers are then weighed, identified and have their loading verified. These charge containers are 2½" OD x 2-3/4" OH (4 oz = 118.28 cc volume) and already contain aluminum and B_4C powders if required. Loaded charge containers are then placed in an aluminum carrier. This carrier is 2" x 7½" x 17" long and holds ten charges in a 2 x 5 arrangement. Individual charge containers are located on 3½" centers in the same row and 4" centers between rows. Details of the charge carrier are shown on sketch 825-III.

4.2 Nuclear Safety

Same as B.2 above. The UAL_X in the charge containers form an 18.2 square inch (2-3/4" x 6-5/8") cross section while in the carrier. An 18.2 square inch cross section is equivalent to the cross section of a cylinder with a 4.83" diameter. Also, the carrier will be limited to 4 kgs U-235. Therefore, the safety justification of Subpart 825.2.2 also applies to the carrier.

5. Blend Charges

5.1 Equipment

The carrier is then removed from the dry box and taken to a work table for loading the blender. Approximately ten charge containers are loaded into an oblique can blender usually in two layers of five, and the remaining void filled with non-absorbent packing material. Loaded blenders are then moved to and placed on ball mill racks. These racks have two sets of rollers and no more than four blenders will be on the racks at any one time. Details of the oblique can blender, including the arrangement of the charge containers, is shown on sketch 825-IV.

5.2 Nuclear Safety

Although the oblique blender can has a total void volume of 5.58 liters, each blender is restricted to 3.6 kgs U-235 contained in 10 charge containers having a total volume not exceeding 3.6 liters (1.18 liters actual). The volume occupied by the UAL_X in the 10 charge containers in the blender, plus the mass limit, is safe as shown on Table IV, TID-7016, Rev. 1. The ball mill racks are arranged so that blenders are separated at least 12" edge-to-edge. This arrangement of the 4 blenders on the racks, as described above, is also safe as shown on Table IV and Fig. 22, TID-7016, Rev. 1.

6. Store Charges

6.1 Equipment

After blending, individual blenders are moved to a work bench, unloaded and placed in a carrier or in a storage rack attached to the back of the bench. The storage rack forms 3 shelves, with 12 holes per shelf. The bottom shelf is 3" off the surface of the bench. Each shelf is separated 8" edge-to-edge. This storage rack is constructed of sheet and angle aluminum bolted together. Details of construction are shown on sketch 825-V. During unloading, the blender will be kept 12" away from the shelf by administrative control.

6.2 Nuclear Safety

Each shelf forms a rectangle of material with a maximum cross section of 6.87 sq. in. (2½" x 2-3/4"). This is a safe cross section for weight fractions not exceeding .52, as shown on Fig. 309-IX. The interaction between troughs and a blender on the table will be 1.14 steradians.

d = diameter of charge container = 2.5", L = 36", L/2 = 18",
$h = 8'' + \frac{2.5''}{2} = 9.25''$

$$\Omega_{6-1} = \frac{2d}{h} \sin \theta \qquad \text{where } \tan\theta\ \frac{L/2}{h} = \frac{18''}{9.25''} = 1.95$$

$$\sin\theta = .890$$

$$= \frac{5''}{9.25''} \quad (.890)$$

$$= .48$$

The effects of the blender on the centermost shelf will be

$V = \text{(blender)} = 3600 \text{ cc} = \frac{\pi d^2 h}{4}$ where $h/d = 1$

$d = \sqrt[3]{\frac{4 \times 3600}{\pi}} = \sqrt[3]{4584} = 16.6 \text{ cm} = 6.54''$

$\Omega_{6-2} = \frac{2d}{h} \sin \theta$ where $\tan \theta = \frac{L/2}{h} = \frac{3.27''}{15.27''} = .214$

$= \frac{13.08''}{15.27''} (.209)$

$\sin \theta = .209$

$= .179$

$\Omega_6 \text{(total)} = (2 \times \Omega_{6-1}) + \Omega_{6-2} = (2 \times .48) + .179 = .96 + .179$
$= 1.139$ Steradians

For safe cross sections with UAL_X weight fractions not exceeding .52", the maximum effective multiplication will not exceed 0.58 as shown in Fig. 309-XI. The allowable interaction is

$\Omega \text{(Allowable)} = 9 - 10K = 9.5.8 = 3.2$ steradians

7. Press Compacts

7.1. Equipment

Not more than 10 charge containers will then be moved to the press. This move will be accomplished by either using a carrier or transferring two charge containers at a time by hand. The press will then be charged by emplying a container into the die and pressing the compact. When ten compacts have been pressed, they are then removed from the die, identified, and placed in a safe cross section (3" x 6") tote tray. Residues will be collected in a bottle or covered tray, with a volume not exceeding 1 liter, and kept at least 12" edge-to-edge from other UAL_X material.

7.2. Nuclear Safety

This operation will be restricted to 700 grams U-235 as loose compacts plus one (1) charge carrier, or 10 charge cans grouped together, plus 1 safe cross section tote tray, plus 1 residue container. The 1 liter volume of the residue container is safe as shown in Fig. 309-X. The 18 sq. in. cross sectional area is safe as shown on Fig. 309-IX. Loose compacts, the charge containers or the charge carrier, the tote box and the residue container will all be separated from each other by 12". The interaction at the 700 grams of loose compacts will be .783 steradians.

Contribution from tote tray (length facing)

$a = 3"$, $b = 16"$, $h = 12"$ and $\cos e = 1$

$$\Omega_{7-1} = \frac{ab\cos e}{h^2} = \frac{3" \times 16"}{(12")^2} = \frac{48}{144} = .333$$

Contribution from Charge Carrier (length facing)

$a = 2.75"$, $b = 17"$, $h = 12"$ and $\cos e =$

$$\Omega_{7-2} = \frac{ab\cos e}{h^2} = \frac{2.75" \times 17"}{(12")^2} = \frac{46.7}{144} = .325$$

Contribution from residue container (assume cylinder with L = D)

$$V = 1000 \text{ cc} = \frac{\pi d^3}{4}, \quad d = \sqrt[3]{\frac{4000}{\pi}} = 10.8 \text{ cm} = 4.25"$$

$a = 4.25"$, $b = 4.25"$, $h = 12"$ and $\cos e = 1$

$$\Omega_{7-3} = \frac{ab\cos e}{h^2} = \frac{4.25" \times 4.25"}{(12")^2} = \frac{18.1}{144} = .125$$

$$\Omega_7 \text{ (Total)} = .333 + .325 + .125 = .783 \text{ steradians}$$

8. Inspect Compacts

8.1 Process

After identification, the tray is carried to the inspection table and transferred to a tote box. Compacts will be inspected on a work bench. During this operation, ten compacts may be out on the bench top. They are then replaced in a safe cross section tote box.

8.2 Nuclear Safety

No more than 700 grams U-235 as loose compacts will be allowed out of the tote box. This is the standard dry operation limit. The tote box will have a cross section of 18 sq. in. (3" x 6"). This is a safe cross section for weight fractions not exceeding .48 as shown on Fig. 309-IX.

9. Subsequent Process Operations

Subsequent operations are the same as operations numbered 15 through 56 in Subpart 822.2.

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ISSUED October 31, 1968

SUPERSEDES NEW

LICENSE: SNM-777; DOCKET: 70-820
SECTION: 800 - FUEL FABRICATION OPERATION
Subsection: 820 - Processing
Subpart: 825 - Fabrication of Uranium Aluminum Intermetallic Cores
825.3 - Support or Auxiliary Operations

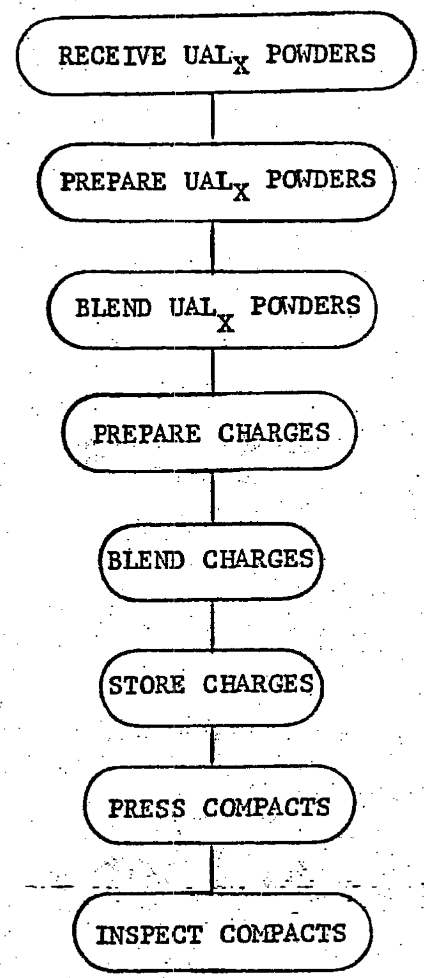
825.3 SUPPORT OR AUXILIARY OPERATIONS

1. Process

Other work benches, tables or hoods may be used for support or auxiliary operations. Generally, these benches or tables will be used to place containers, blenders, tote boxes, etc. on them during transfers between normal process steps. No loose powder will be allowed out of containers.

2. Nuclear Safety

One container plus 700 grams U-235 as a group of small containers (e.g. change bottles, etc) or pieces (e.g. cores, etc.) will be allowed at each work station or critical zone. Containers will have either a safe volume or safe cross section as described in Subpart 825.2.

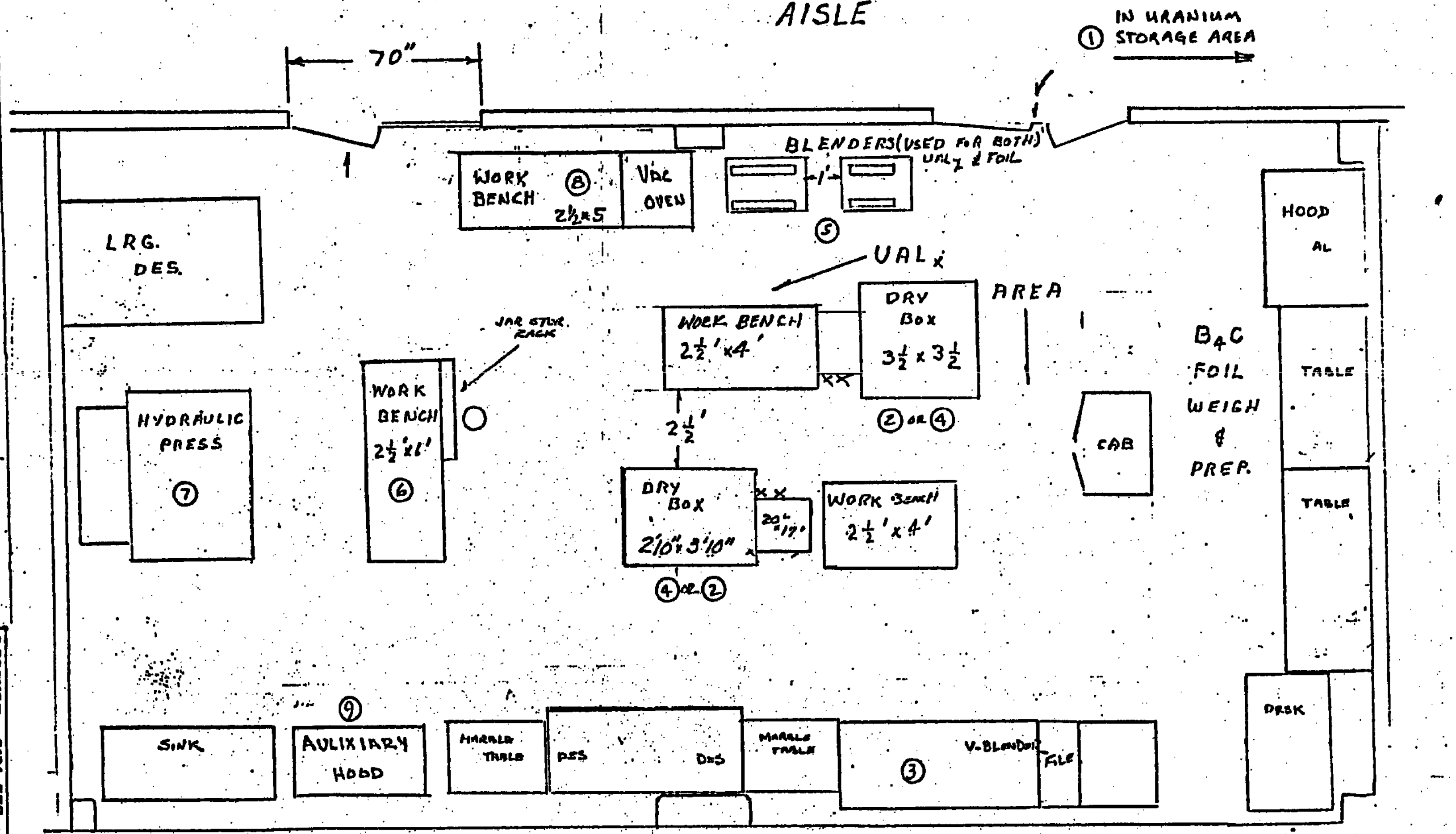


LICENSE: SNM-777; DOCKET: 70-	
SECTION: 300, Subpart: 825	
UAL_X- PROCESS FLOW	
DIAGRAM 825-I	
Page 1 of 1	
APPROVED:	
ISSUED: October 31, 1968	

Stamp proprietary!!!

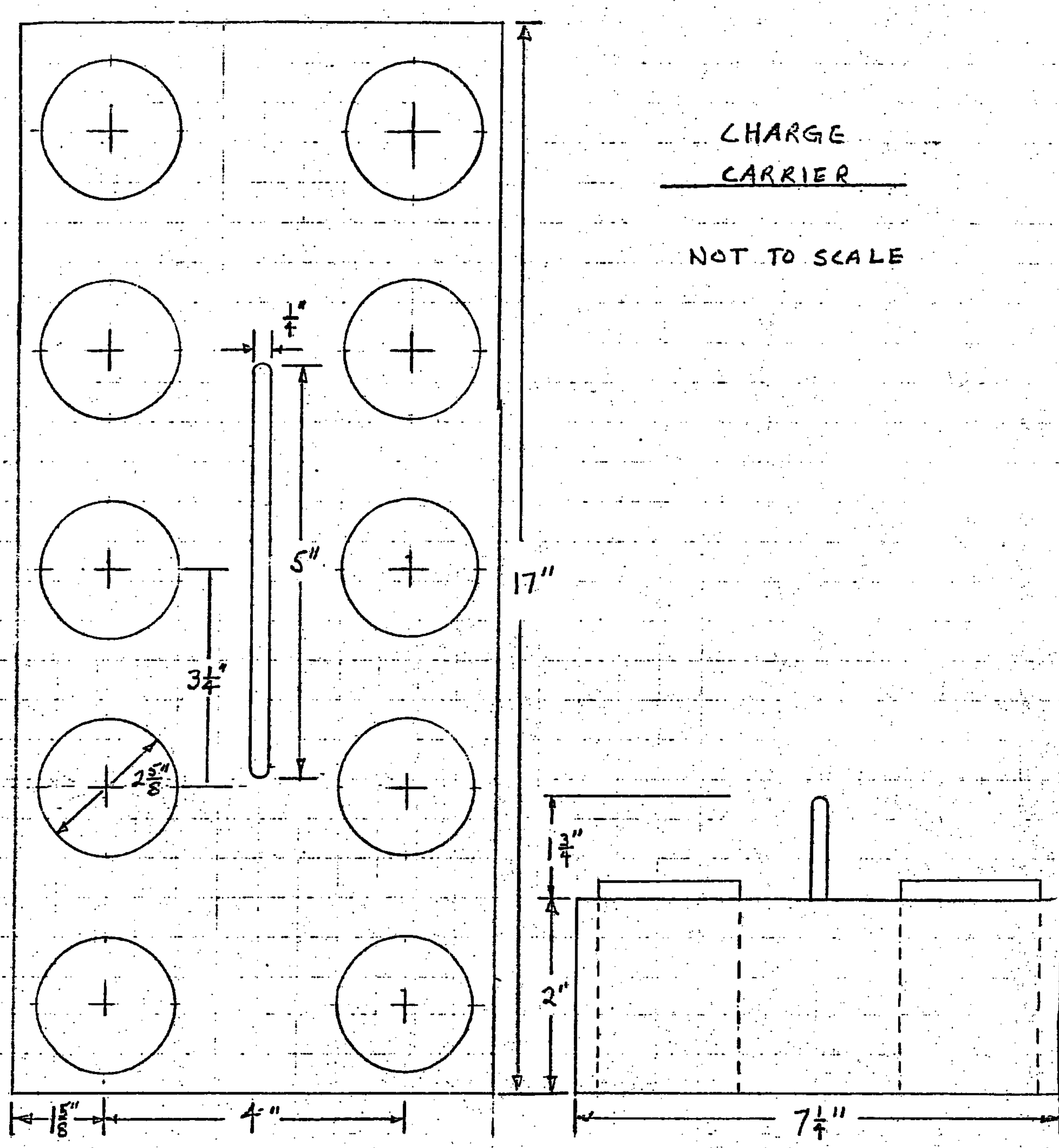
19-H POWDER FACILITY

ROOM DIMENSIONS 21'x 39½'



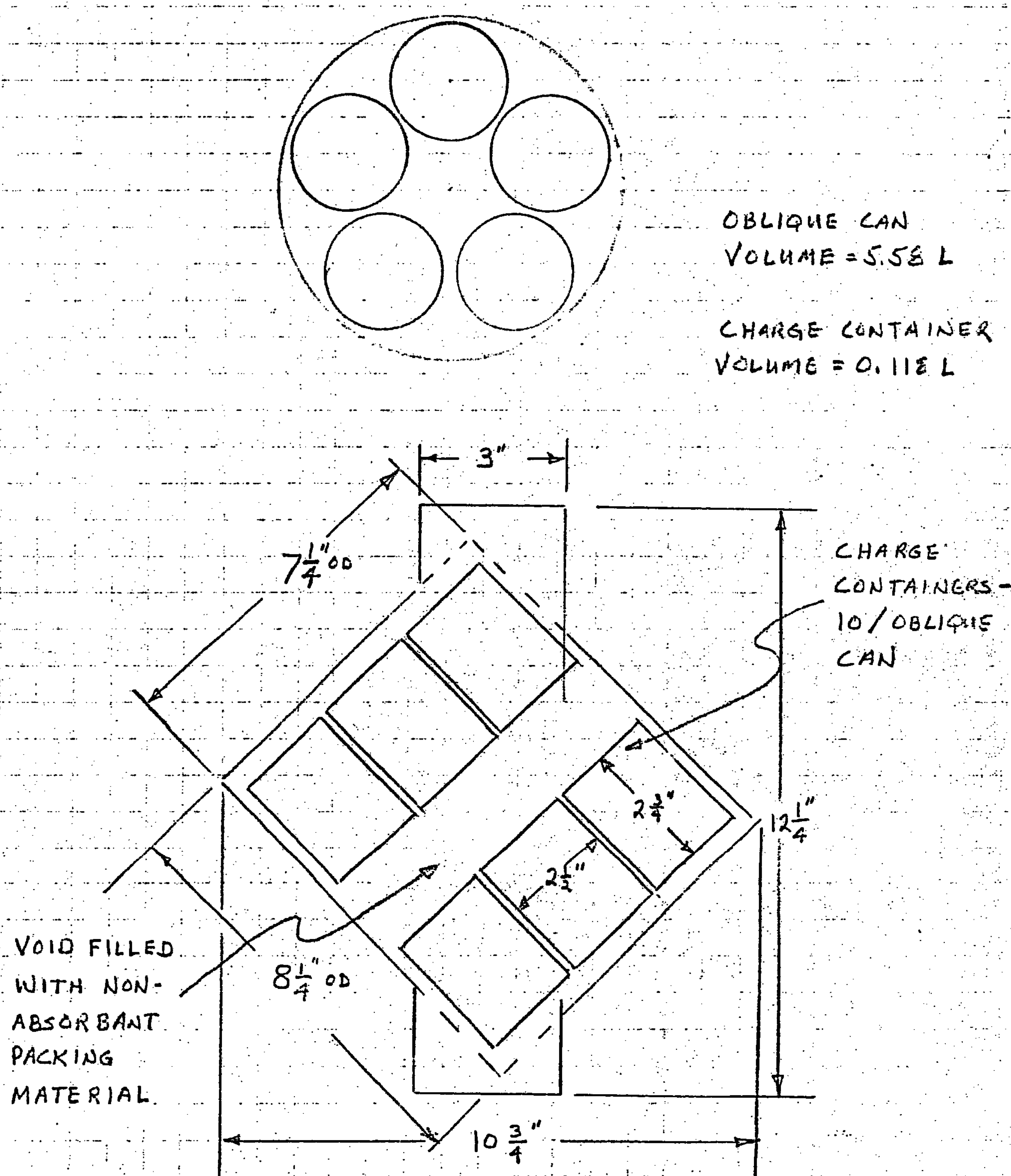
① - RECEIVE UALx POWDER
② - PREPARE UALx POWDER
③ - BLEND UALx POWDER
④ - PREPARE CHARGE
⑤ - BLEND CHARGE
⑥ - STORE CHARGE
⑦ - PRESS CHARGE
⑧ - INSPECT COMPACTS
⑨ - SUPPORT OR AUXILIARY OPERATIONS

LICENSE: SNM-777, Docket 70-820
SECTION: 800, Subpart 825
Sketch 825-II- UALx Area
APPROVED:
ISSUED: October 31, 1968
SUPERSEDES: New



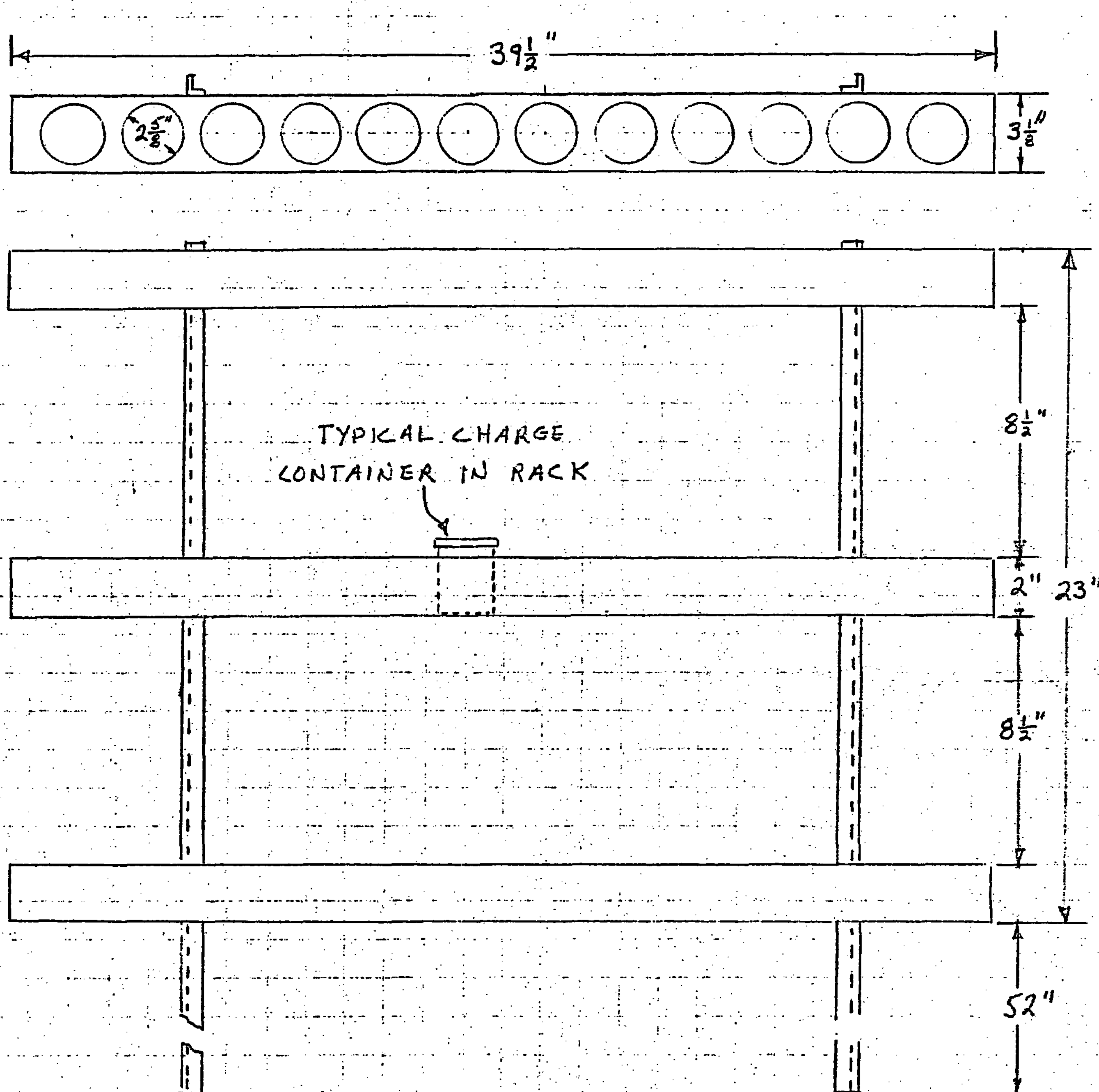
LICENSE: SNM-777, Docket 70-820	
SECTION: 800, Subpart 825	
Sketch 825-III-Charge Carrier	
Page 1 of 1	
APPROVED:	
ISSUED: October 31, 1968	
SUPERSEDES: New	

UALX-AL-B4C POWDER BLENDING LAYOUT



LICENSE: SNM-777, Docket 70-820
SECTION: 800, Subpart 825
Sketch 825-IV-Oblique Blender
Page 1 of 1
APPROVED:
ISSUED: October 31, 1968
SUPERSEDES: New

CHARGE STORAGE RACK



LICENSE: SNM-777, Docket 70-820	
SECTION: 800, Subpart 825	
Sketch 825-V-Charge Storage Rack	
Page 1 of 1	
APPROVED:	
ISSUED: October 31, 1968	
SUPERSEDES: New	

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LICENSE: SNM-33, Docket: 70-36 SECTION: 900 - FABRICATION OPERATION SUBSECTION: 920 - PROCESSING SUBPART: 926 - FABRICATION OF URANIUM ALUMINUM POWDERS 926.1 - GENERAL CONSIDERATIONS	Page 1 of 1 Approved Issued 3/6/72 Supersedes 1/20/72

926.1 GENERAL CONSIDERATIONS

1. This subpart covers the fabrication of uranium aluminum powders. Final powders are fine mesh with bulk densities of approximately 6.6 grams per cubic centimeter and U-235 weight fractions up to and including 0.70.

2. These operations will be performed in Building 19H. The processing location and general layout is shown on Drawing B-304236. A more detailed layout of the process area is shown on Sketch 926-I.

3. Equipment design and location was determined with National Fire Protection Association (NFPA) Standards being considered.

4. Upon completion of fabrication of uranium aluminum powders, further processing will be as described in Subpart 925.

5. The health physics criteria of Section 400 will be utilized in the equipment design and location.

926.2 PROCESS DESCRIPTION

1. Receive Enriched Uranium Metal

Enriched Uranium Metal will be received and stored in accordance with subsection 910.

2. Enrichment Check

If required, an enrichment verification will be performed, usually by auto gamma spectrometry. The limits for this operation will be 10 Kg U-235 as one piece or 350 grams U-235 in small pieces as listed on Table 309-I.

3. U-Metal Preparation

If required, the uranium metal will be cleaned by pickling to remove oxide and lose surface contamination. The pickling solution will be contained in three 4-liter containers separated at least 12 inches edge-to-edge from each other. These containers will be placed in a hood and the entire operation will be limited to 2.1 Kg U-235. Pickeled U-metal will be reweighed and the differences (losses) will be recorded on process documents and a tally sheet at the pickling station. The maximum amount of material allowed to accumulate in the pickle solution will not exceed 200 grams U-235 in each container.

When the maximum 200 grams U-235 has been accumulated or the pickle solution is spent, it will be emptied into 4-liter or smaller bottles and transferred to storage. The operation will be performed in either the 19-H Fuel Storage Area, the charge preparation area or the UAL_x Fabrication Area.

Nuclear Safety of this operation is provided by limiting the mass in any 4-liter container to not exceed 2300 grams U-235. As shown on Figure 926-II, this is a safe mass for volumes not exceeding 4-liters. Figure 926-II is a plot of the maximum safe batch size (45% of critical) as a function of spherical container volume for mixtures of highly enriched uranium and water. This is independent of the degree of water moderation and fuel piece size. These safe masses are based on experimental data presented in TID-7028. The hydrided data presented in TID-7028 have been used to be sure that the most reactive systems of uranium and hydrogen moderation have been considered. As the volume increases, the maximum safe batch size approaches 0.35Kg contained U-235; as it becomes smaller, it approaches 10.0 KG contained U-235 (at a volume no greater than 2.2 liters).*

*It is recognized that these non-uniform systems are unique. These non-uniform systems might have masses less than those for uniformly distributed systems. However, this is not a problem for the proposed small volumes and masses shown above.

4. Charge Preparation

The required quantity of material for a single melting operation is weighed as either a single batch in a safe volume container or in increments of 358 grams of uranium which is then consolidated in a safe volume container. Safe volume containers are normally dry ($H/X \leq 2$) and limited to 10 Kgs U-235 each. The safe volume container will be spaced 12-inches or more from the material being handled. Safe volume containers will not exceed 2.2 liters.

5. Melting

The Uranium-Aluminum powder is prepared by melting the uranium and aluminum in the proper proportions in a graphite hearth containing six cavities. A furnace contains the graphite hearth which rests upon a water cooled plate. The six cavities are in a single plane array, equally spaced around a 9 ¼ inch diameter bolt circle. Each cavity is a spherical cup, 4-3/8 inches inside diameter by 1¼ inches deep. The total charge is 500 grams and shall not exceed 358 grams U per cavity or 2.148 Kg U per furnace loading. Because of having a single electrode in the furnace, there will never be more than a part of one button at a time in molten state. Each button will be melted four times, flipping over between each melt. This multiple melting is required to effect the required compositional uniformity. Upon completion of melting, buttons will be removed from their melt positions one at a time and placed in a safe volume container for transfer to further processing or storage.

The safety factor for this process step is very large as can be seen from Figure 926-II. The total volume of the array of furnace hearths (approximately 12-inch diameter X 1¼ inches deep) is approximately 2.32 liters. This includes all the graphite. A volume at least twice as great (2.5 inch depth) and in a spherical geometry would be safe for the planned charge. Any splattering resulting from the melting operations would be cleaned out of the furnace shall before adding new charges to the hearths.

6. Button Storage

If required, buttons will be stored in accordance with subsection 910.

7. Pulverizing and Sieving

The pulverizing and sieving glove box is a negative pressure inert atmosphere box containing a jaw crusher, sieve shaker, weighing balance and an eight inch diameter x 12 inch long blender. Operations to be performed include crushing of buttons and screening to the correct particle size, addition of recycle material (if necessary), composition blending and sample collection for chemical analysis. Nuclear Safety is provided by processing this material in a glove box with a controlled atmosphere. There is no water or other moderating material connected to or used in the glove box. Therefore, the H/X will be less than 2 and the operation is considered dry. The glove box will be limited to 10 Kgs U-235. From Fig. 1, TID-7016, Rev 1, 10 Kgs U-235 with an H/X less than 2 is safe. The crushing and pulverizing of buttons is conducted in a small laboratory hammermill. The batch size will be determined using Figure 926-II. The pulverized material from the hammermill will be collected in a set of stacked 8 inch diameter screens plus receiver. The height of the stack will be determined using the mass - volume restrictions of Figure 926-II. Blending will be done to insure a composite blend of particle size. Although having a large volume (approximately 16 liters) the blending operation is dry and will never exceed the glove box limit of 10 Kgs U-235. Weighing operations will be done in individual safe volume containers.

8. <u>Desiccator Storage</u>

Special desiccator racks are provided for storage. These racks are constructed to accomodate six or nine desiccator jars. The racks have two or three shelves with a separation of 21 inches between shelves. Each shelf will contain three desiccator jars with a separation of 12 inches edge-to-edge from each other.

Nuclear safety of each desiccator jar will be maintained by mass control in each jar. One of the following limits apply to each jar:

1. 350 grams U-235 in any size container.
2. 24 compacts per 4-liter container (1.83 Kg U-235).
3. 4.0 Kg U-235 per 4-liter or smaller container.
4. 5.9 Kg. uranium-aluminum powder (4.0 Kg U-235) per 4-liter or smaller container.
5. 24 compacts in frames (1.83 Kg U-235) in desiccator only.

Desiccator jars are approximately 10 inches in diameter and 12 inches high. They are constructed of approximately 1/8 to 3/16 inch aluminum.

9. <u>Storage</u>

Storage will be in accordance with subsection 910.

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LICENSE: SNM-33, Docket 70-36	Page 1 of 1
SECTION: 900 - Fabrication Operation	Approved
SUBSECTION: 920 - Processing	Issued 3/6/72
SUBPART: 926.3 - Health Physics and Industrial Safety Considerations	Supersedes 1/20/72

926.3 <u>HEALTH PHYSICS AND INDUSTRIAL SAFETY CONSIDERATIONS</u>

1. <u>Pulverizing & Sieving Glove Box</u>

The atmosphere in The Pulverizing and Sieving Glove Box will be innert gas (once through) maintained negative relative to room pressure. The pressure will be maintained by automatic differential pressure switches and solenoid valves located in the box exhaust system and the innert gas inlet lines.

Under normal operation, the innert atmosphere in the glove box will be exhausted through high efficiency filters located in the exhaust port. Under emergency conditions (large pressure increase in the glove box, such as caused by opening the box to room atmosphere) an 8 inch pneumatically operated valve opens and provides sufficient exhaust to yield a 100 foot/min. lineal air velocity across the face of an opening in the box. This emergency exhaust condition does not pass through a high efficiency filter at the box but is exhausted through high efficiency filters in the facility exhaust system. Also this glove box is equipped with an oxygen analyzer to continually monitor and control the atmosphere in the box. This box is equipped with air locks and/or transfer ports for all transfers.

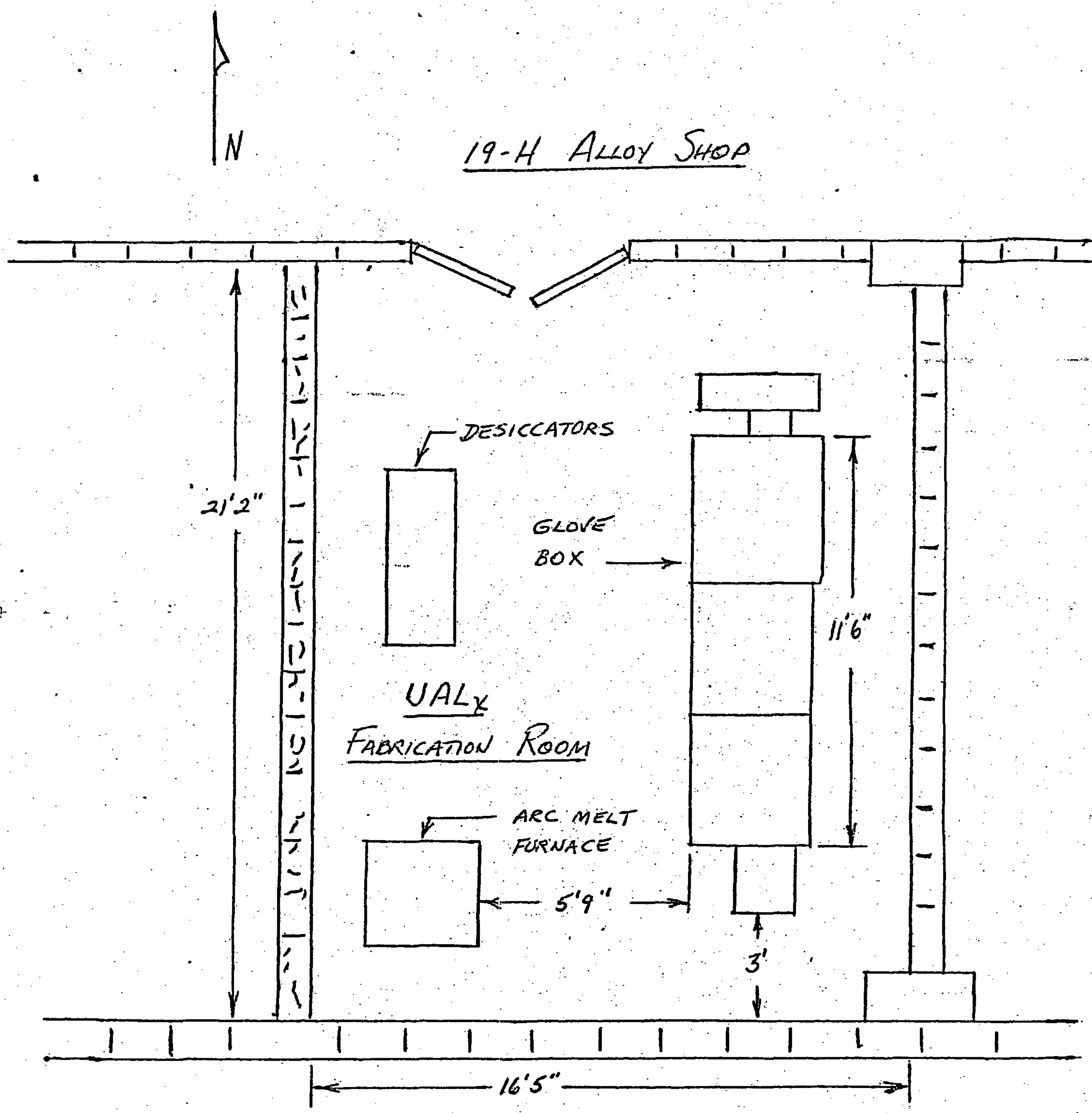

Figure 926-I
Layout - UALx Fabrication Area

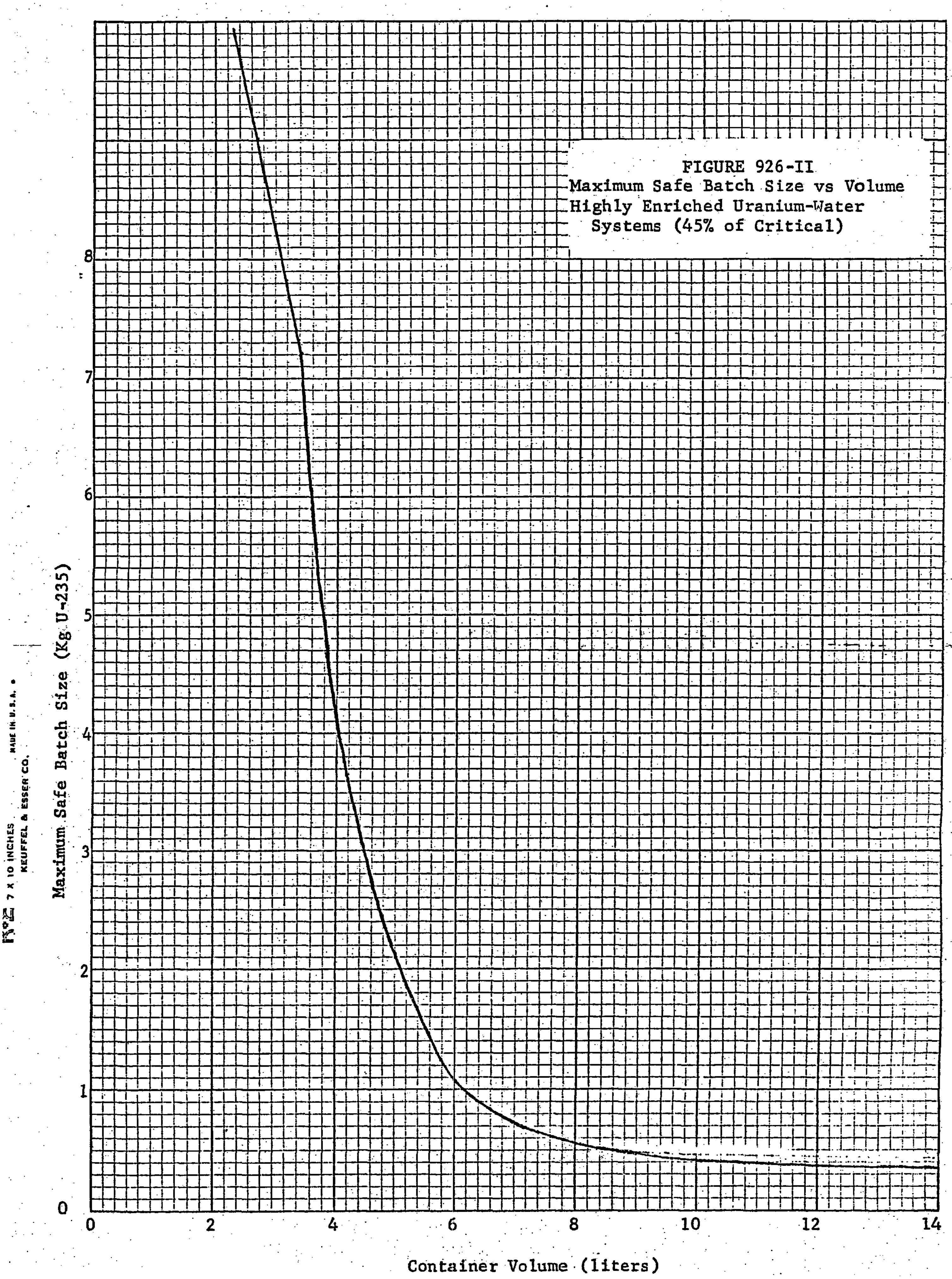

FIGURE 926-II
Maximum Safe Batch Size vs Volume
Highly Enriched Uranium-Water
Systems (45% of Critical)
Maximum Safe Batch Size (Kg U-235)
Container Volume (liters)
0
1
2
3
4
5
6
7
8
0
2
4
6
8
10
12
14
7 X 10 INCHES
KEUFFEL & ESSER CO.
MADE IN U.S.A.

TABLE 926 - II

MAXIMUM SAFE BATCH SIZE vs VOLUME
HIGHLY ENRICHED URANIUM-WATER SYSTEMS
(45% of Critical)

Density, (Kg U-235/liter)	Critical Mass[1] (Kg U-235)	Critical Vol.[2] (Liters)	Safe Batch[3] (Kg U-235)
.06	.82	14.0	.37
.07	.85	12.0	.38
.08	.88	11.0	.40
.09	.92	10.2	.41
.10	.95	9.7	.43
.15	1.20	8.2	.54
.20	1.50	7.4	.68
.40	2.40	6.0	1.08
.60	3.30	5.6	1.49
.80	4.20	5.2	1.89
1.00	5.00	4.9	2.25
1.50	7.00	4.4	3.15
2.00	9.00	4.0	4.05
3.00	13.00	3.6	5.85
4.00	16.00	3.4	7.20
6.00	20.00	2.7	9.00
8.00	22.00	2.3	9.90

(1) Figure 8, TID-7028

(2) Figure 9, TID-7028

(3) 45% of critical mass

LICENSE:	SNM-33, Docket 70-36
SECTION:	900, Subpart 926.2
	Table 926-II
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SUPERSEDES:	1/20/72

The most reactive unit will be the 4-liter container with not more than 4.0 Kgs U-235. The dimensions of this container will be 6-inches diameter by 10-inches high.

The Desiccator Rack is constructed to accomodate not more than 9 desiccator jars forming a 3-by-3 planar array. Each unit will be separated 21-inches center-to-center vertically and 12-inches edge-to-edge horizontally. Thus, each unit will occupy a surface area of:

$$A = 18'' \times 21'' = 2.62 \text{ ft}^2$$

The resultant surface density is:

$$\bar{\sigma} = \frac{4.0 \text{ Kg U-235}}{2.62 \text{ ft}^2} = 1.53 \text{ Kg U-235/ft}^2$$

This value is less than the limit for U(93) metal set forth on the table on page 10, "Density-Anolog Techniques", Livermore Array Symposium, CONF 680909.

Since this material is in desiccated storage, it is dry and will be kept dry. Thus, H/U-235 $\leq$ 2. For dry U(93) metal, reflected, the critical mass set forth on Fig. 8, TID-7028, is:

$$M_c = 22 \text{ Kg U-235}$$

The actual mass as listed above is:

$$M_A = 4 \text{ Kgs U-235}$$

Then, the fraction critical is:

$$f = \frac{M_A}{M_c} = \frac{4 \text{ Kg U-235}}{22 \text{ Kg U-235}} = 0.182$$

This value meets the individual unit criteria for fraction critical which requires:

$$f \leq 0.3$$

Therefore, the array is safe.

LICENSE:	SNM-33, Docket 70-36
SECTION:	900, Subpart 926.2
Nuclear Safety Evaluation -	
Desiccator Jar Rack Interaction.	
PAGE 1 OF 1	
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LICENSE: SNM-777, Docket: 70-820
SECTION: 800 - FUEL FABRICATION OPERATION
Subsection: 830 - Support Activities

APPROVED

ISSUED October 31, 1968

SUPERSEDES New

830. Support Activities

Support activities such as those performed in a chemistry laboratory, met laboratory, etc., are preformed for the Fuel Fabrication operation by an outside, licensed contractor. These services are usually supplied by the United Nuclear Corporation, Naval Products Division and are described in SNM-368, Docket: 70-371.

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LICENSE: SNM-777, Docket 70-820
SECTION: 800 - Fuel Fabrication Operation
Subsection: 840 - Nuclear Criticality Alarm System

Approved

ISSUED October 31, 1968

SUPERSEDES New

840. Nuclear Criticality Alarm System

The Fuel Fabrication Operation facilities at New Haven are covered by the Nuclear Criticality Alarm System established and mainatined by the Naval Products Division which has prime responsibility for this system. Details of the system are described in license SNM-368, Docket 70-731.

NDEO-1050

CRITICALITY SAFETY CALCULATION METHOD FOR ENRICHED URANIUM-WATER SYSTEMS

UNITED NUCLEAR CORPORATION
Research and Engineering Center
Elmsford, N.Y.

by

M. Raber

J. R. Tomonto

August 22, 1967

Table of Contents

1.0 INTRODUCTION

The following sections outline a simplified procedure for calculating criticality safety parameters for unpoisoned, light water moderated, enriched uranium systems. The method is presented as a basic standard to be applied in determining mass and volume limits for uranium fuel manufacturing processes involving:

uranium solutions,
heterogeneous uranium-zirconium alloy, uranium-steel, and uranium-aluminum mixtures in light water

The calculational method (Section 2) is based on a conservative four factor formulation for k_{eff} with leakage treated by the standard Fermi age approximation. Values for the age to thermal energies (0.625 ev) are obtained from a correlation of experimental and calculated data. The calculation method is divided into three approaches depending on the degree of moderation in the systems. Separate sets of assumptions and cross section values are made depending on whether the system is thermal (Section 2.1), undermoderated (Section 2.2) or unmoderated (Section 2.3). The method has been checked with a large number of experiments and found to be conservative in all instances (Section 3). Section 4 presents calculated parameters for uranium-zirconium alloy systems.

The assumptions made in Section 2 for the calculational method are overly conservative when applied to slightly enriched uranium oxide systems. Therefore, the results of detailed calculations[1] have been used to set mass and volume standard limits for water moderated, uranium oxide systems ($E \leq 5$ w/o U^{235}) as shown in Section 5.

2.0 BASIC CALCULATION METHOD FOR URANIUM-WATER SYSTEMS

The calculation method is divided into three approaches depending on the degree of moderation in the system. The system is assumed thermal if:

$\frac{H}{U^{235}} >$ 20 for uranium solutions and uranium metal-water mixtures

$\frac{V_{alloy}}{V_{water}} <$ 1.70 for uranium-alloy-water mixtures

If the above limits are satisfied, the calculation method presented in Section 2.1 should be used. If the system is under-moderated (reverse the above limits), the calculation method presented in Section 2.2 is recommended. If the system does not contain moderator, the method shown in Section 2.3 should be used.

2.1 Calculation Method for Thermal, Enriched Uranium-Water Systems

If the atom ratios for uranium solutions, uranium-metal-water mixtures or uranium-alloy-water mixtures exceed the limits defined in Section 2.0, the system is assumed to be thermal to some degree. The following method is recommended.

The infinite multiplication factor is defined with the standard four factor formula, i.e.;

$$k_{\infty} = \epsilon p \eta f \qquad (1)$$

In a thermal system the product of the fast fission factor (ϵ) and the resonance escape probability are conservatively assumed to be unity. The non-leakage probability is determined from the Fermi age approximation:

$$P_{non\text{-}leakage} = \frac{1}{1+M^2 B^2} \qquad (2)$$

Therefore the expression used to determine the effective multiplication factor is

$$k_{eff} = \frac{\eta f}{1+B^2M^2} \qquad (3)$$

where
f = thermal utilization of U^{235}
M^2 = $L^2 + \tau$
L^2 = thermal diffusion area, cm^2
τ = age to thermal energy, cm^2
B_g^2 = geometric buckling, cm^{-2}

The following expression is used to calculate f:

$$f^{-1} = 1 + \sum_i C_i \left(\frac{W_i}{W_{25}}\right) \frac{1}{F_i} \qquad (4)$$

where subscript i denotes any component other than U^{235}.

$$C_i = \frac{A_{25}\, \bar{\sigma}_{ai}}{A_i\, \bar{\sigma}_{a25}} \qquad \text{(see Table 2.1)}$$

$\bar{\sigma}_a$ = average thermal absorption cross section, barns

A = atomic or molecular weight, gm/gm-atom or gm/gm-mole

$$F_i = \frac{\text{avg. thermal neutron flux in } U^{235} \text{ region}}{\text{avg. thermal neutron flux in region containing i}}$$

$\frac{W_i}{W_{25}}$ = mass ratio of component i to U^{235}

For mixtures of water and fuel materials, L^2 is computed from

$$L^2 = 8(1-f) \qquad (5)$$

This corresponds to a value of 8 cm^2 for L^2 of pure water at room temperature.

$B_g{}^2$ depends on the system geometry, and is computed from one of the following formulas.

Sphere: $$B_g{}^2 = \left[\frac{\pi}{R + \delta}\right]^2 \tag{6}$$

Cylinder: $$B_g{}^2 = \left[\frac{2.405}{R + \delta}\right]^2 + \left[\frac{\pi}{H + 2\delta}\right]^2 \tag{7}$$

Rectangular Parallelepiped:

$$B_g{}^2 = \left[\frac{\pi}{a + 2\delta}\right]^2 + \left[\frac{\pi}{b + 2\delta}\right]^2 + \left[\frac{\pi}{c + 2\delta}\right]^2 \tag{8}$$

where R = sphere or cylinder radius, cm.
H = cylinder height, cm.
a,b,c = side dimensions of rectangular parallelepiped, cm.

δ is the extrapolation length for bare systems or reflector savings plus extrapolation length for reflected systems.

δ = 2.5 cm for bare systems

δ = 6.5 cm for systems surrounded by a thick water reflector (15 cm or more thick). (9)

Cross Section Data

The cross sections were obtained from Reference 2. For the calculation of thermal utilization and self-shielding factors, absorption cross sections averaged over a room temperature Maxwellian spectrum are used. In addition, a correction factor of 0.98 is applied to the U^{235} absorption cross section to take into account the non-1/v behavior of this cross section. Table 2.1 presents a compilation of the nuclear data used in this calculation method.

Table 2.1

Nuclear Data

Material	σ_a(2200 m/sec)(2) barns	$\bar{\sigma}_a = 0.8862\bar{\sigma}_{a\ 2200}$	σ_s, barns	A(3)	C
U^{235}	678.2	589*	10	235.1	1.0
U^{238}	2.73	2.42	8.3	238.1	0.00406
H_2O	0.670	0.594	-	18.016	0.01316
Zr	0.188	0.167	8	91.22	0.00073
Zircaloy-2	0.201	0.178	8	91.51	0.00078*
Al	0.235	0.208	1.4	26.98	0.00308
Fe	2.55	2.260	11	55.85	0.01615
Stainless Steel (Type 347)	2.98	2.64	11	55.38	0.01903
B(natural)**	759	672.62	4	10.82	24.813
C	0.0034	0.0030	4.8	12.01	0.0010
Nb	1.15	1.02	5	92.91	0.00438

*678 x 0.8862 x 0.98 = 589

**19.81 a/o B^{10}

For U^{235}, the following constants are used[2]:

$$\sigma_a \text{ (2200 meters/sec)} = 678.2 \text{ barns}$$

$$\eta = 2.078$$

For highly enriched uranium systems, the value of η used in Eq. 3 is 2.078. For lower enrichment uranium systems, η is calculaded from

$$\eta = \frac{2.078}{1 + \frac{W^{28}C^{28}}{W^{25}}} \tag{10}$$

τ Data

For highly enriched uranium-water mixtures, the calculated τ data given in Reference 3 are used. τ is shown as a function of H/U^{235} atom ratio in Figure 2.1.

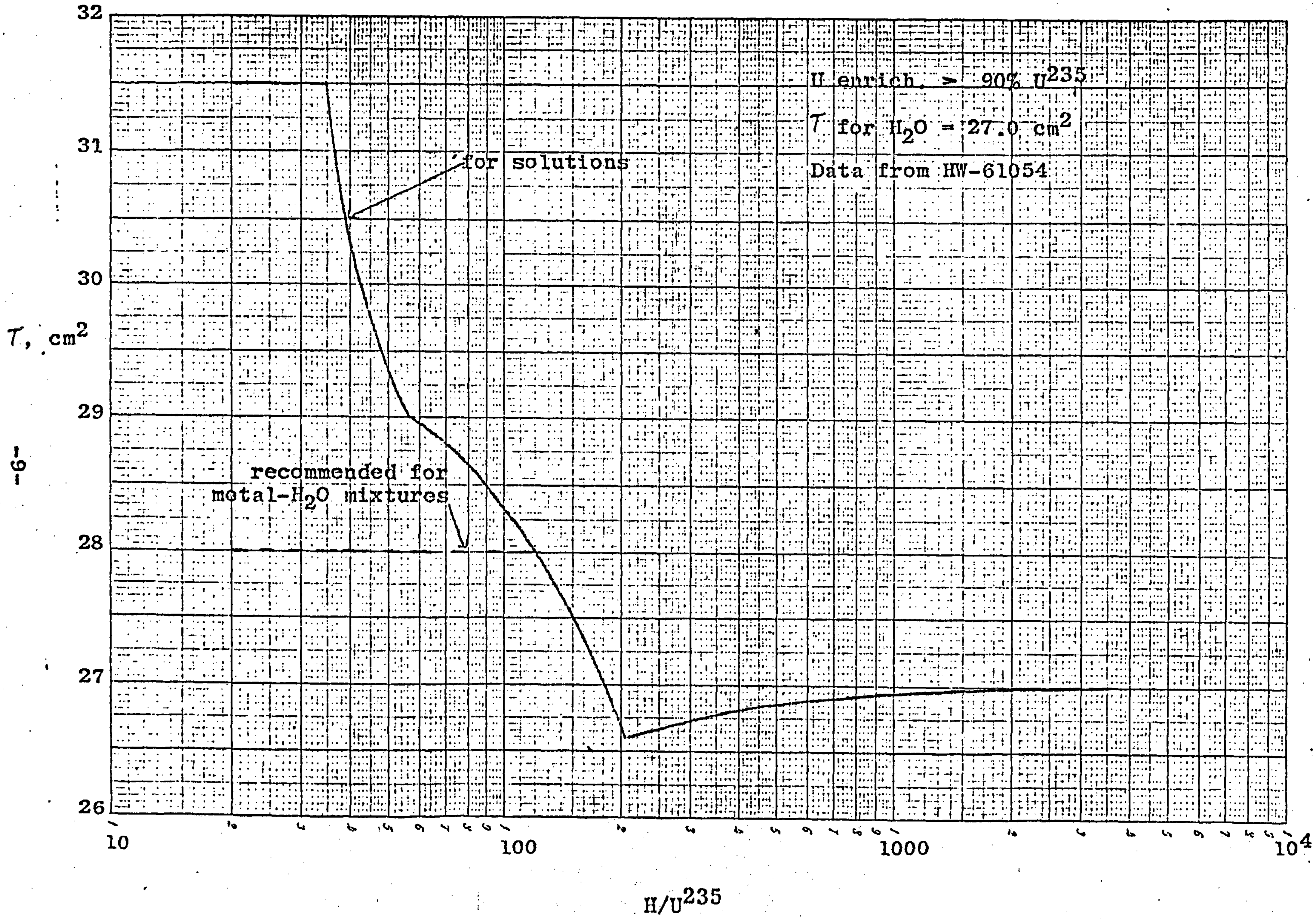
For mixtures of U-Zr alloy and water, U-Fe and water and U-Al and water, the τ data for metal-water mixtures presented in Figure 2.2 are used. These data are based on the latest measurements of the age to the indium resonance (1.44 ev) and in metal water mixtures[4,5,6]. The measured values of age to the indium resonance were corrected to thermal energy (0.625 ev) using the FORM code[7]. This correction amounted to only $\sim$2% of the measured age values, therefore they may be applied without risk of seriously affecting the reliability of the measured data. Values of the age to thermal energy calculated[8,9] with the FORM code are also shown in Figure 2.2. The effect of uranium in the metal-water mixtures on the age to thermal energy is neglected in Figure 2.2.

Self-Shielding Factors

For slab systems, the integral transport theory formulations of Michel Theys (Reference 10) are used to calculate self-shielding factors. For an array of parallel slabs immersed in a water moderator, the ratio of the neutron flux at the surface of a slab to the average flux in the slab is given by:

Figure 2.1

τ for UO_2F_2 Solutions (in water)



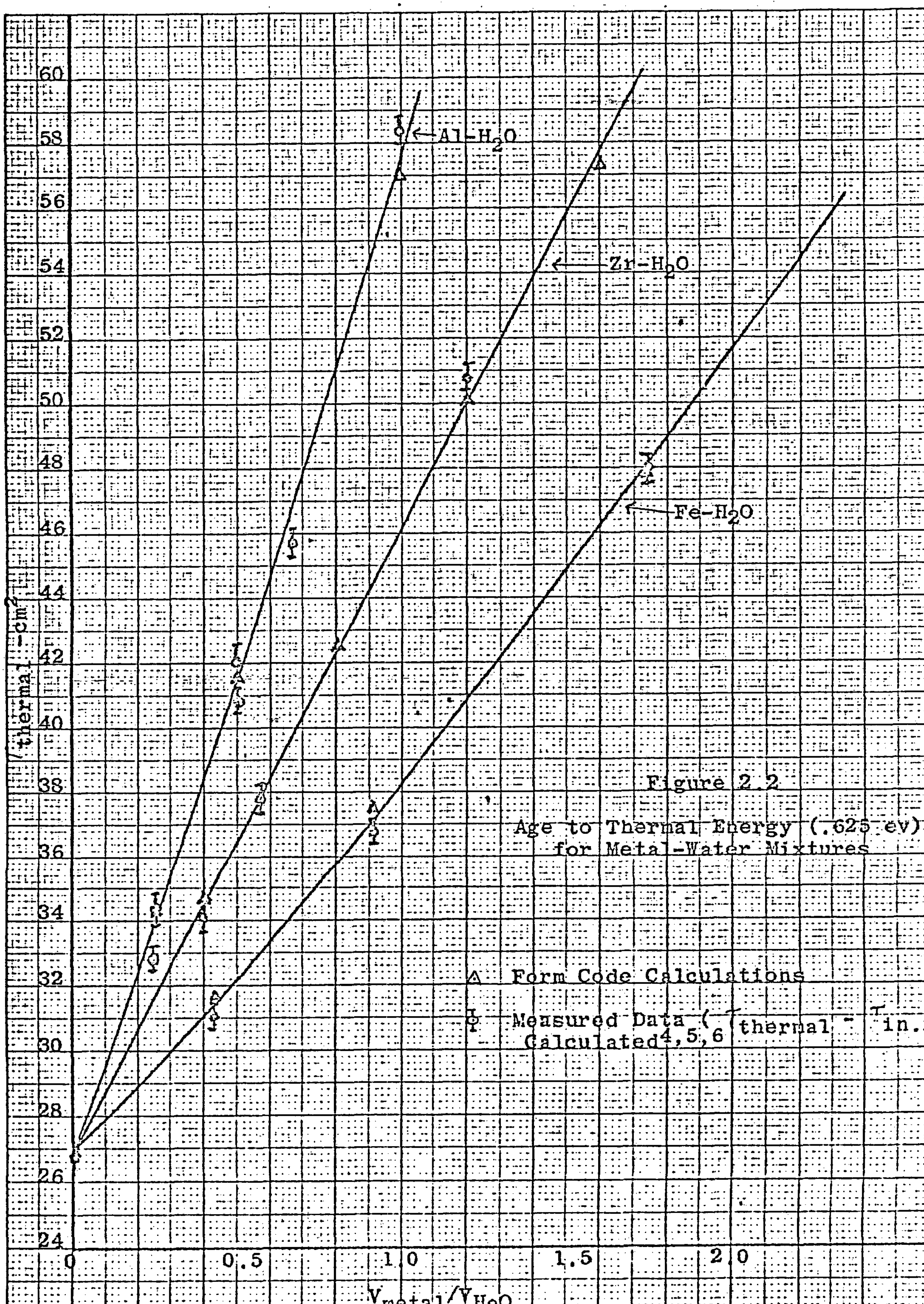


Figure 2.2

Age to Thermal Energy (.625 ev) for Metal-Water Mixtures

$$G = 1 + \frac{\Sigma_a^o}{\Sigma_t^o} (A^*) \left[1 + \alpha^* \frac{\Sigma_s^o}{\Sigma_t^o} + \beta^* \left(\frac{\Sigma_s^o}{\Sigma_t^o} \right)^2 \right]$$

where Σ_a^o = absorption cross section of the slab, cm^{-1}. This is computed using the $\bar{\sigma}_a$ values given in Section 2.0.

Σ_s^o = scattering cross section of the slab, cm^{-1}

Σ_t^o = total average cross section of the slab, cm^{-1} = $\Sigma_a^o + \Sigma_s^o$

a = half-thickness of the fuel plates - cm

A* is tabulated as a function of (a) Σ_t^o in Table 2.2[10].

α^* and β^* are plotted as a function of $(a\Sigma_t^o)$ in Fig. 2.3[10].

For clad slabs, if the calculation of G is based on the cross sections and thickness of the fueled meat only, the self-shielding factor of the cladding can conservatively be taken as G^{-1}.

The ratio of the average flux in the water moderator to the average flux in the slab is given by:

$$\frac{\bar{\varphi}_1}{\bar{\varphi}_0} = G + a \Sigma_a^o \left[\Sigma_{tr}^1 (b-a) + 0.13 \right]$$

where (b-a) = half-thickness of water region between parallel slabs, cm

$a\Sigma_{t,U}$	A^*	$a\Sigma_{t,U}$	A^*	$a\Sigma_{t,U}$	A^*	$a\Sigma_{t,U}$	A^*	$a\Sigma_{t,U}$	A^*
0.00	0.0	0.46	0.5161	1.34	1.8236	2.26	3.5462	3.18	5.3624
0.01	0.0309	0.47	0.5277	1.36	1.8604	2.28	3.5891	3.20	5.4041
0.02	0.0506	0.48	0.5388	1.38	1.8961	2.30	3.6251	3.22	5.4466
0.03	0.0670	0.49	0.5504	1.40	1.9307	2.32	3.6623	3.24	5.4902
0.04	0.0814	0.50	0.5621	1.42	1.9672	2.34	3.7009	3.26	5.5178
0.05	0.0946	0.52	0.5861	1.44	2.0024	2.36	3.7407	3.28	5.5634
0.06	0.1068	0.54	0.6096	1.46	2.0359	2.38	3.7820	3.30	5.6100
0.07	0.1185	0.56	0.6341	1.48	2.0714	2.40	3.8246	3.32	5.6402
0.08	0.1297	0.58	0.6593	1.50	2.1087	2.42	3.8593	3.34	5.6890
0.09	0.1405	0.60	0.6843	1.52	2.1443	2.44	3.8952	3.36	5.7209
0.10	0.1508	0.62	0.7099	1.54	2.1817	2.46	3.9419	3.38	5.7718
0.11	0.1610	0.64	0.7357	1.56	2.2171	2.48	3.9804	3.40	5.8054
0.12	0.1710	0.66	0.7617	1.58	2.2544	2.50	4.0200	3.42	5.8399
0.13	0.1809	0.68	0.7886	1.60	2.2935	2.52	4.0610	3.44	5.8752
0.14	0.1906	0.70	0.8154	1.62	2.3303	2.54	4.0929	3.46	5.9304
0.15	0.2004	0.72	0.8430	1.64	2.3645	2.56	4.1364	3.48	5.9676
0.16	0.2100	0.74	0.8701	1.66	2.4004	2.58	4.1706	3.50	6.0056
0.17	0.2196	0.76	0.8976	1.68	2.4381	2.60	4.2167	3.55	6.1045
0.18	0.2291	0.78	0.9268	1.70	2.4776	2.62	4.2532	3.60	6.2092
0.19	0.2384	0.80	0.9549	1.72	2.5141	2.64	4.2908	3.65	6.2985
0.20	0.2481	0.82	0.9832	1.74	2.5523	2.66	4.3296	3.70	6.3929
0.21	0.2576	0.84	1.0128	1.76	2.5873	2.68	4.3696	3.75	6.4925
0.22	0.2671	0.86	1.0422	1.78	2.6239	2.70	4.4108	3.80	6.5976
0.23	0.2767	0.88	1.0714	1.80	2.6622	2.72	4.4532	3.85	6.7083
0.24	0.2864	0.90	1.1017	1.82	2.7021	2.74	4.4969	3.90	6.8006
0.25	0.2960	0.92	1.1330	1.84	2.7384	2.76	4.5297	3.95	6.8978
0.26	0.3058	0.94	1.1637	1.86	2.7763	2.78	4.5759	4.00	7.0000
0.27	0.3156	0.96	1.1935	1.88	2.8158	2.80	4.6108	4.05	7.1075
0.28	0.3254	0.98	1.2259	1.90	2.8511	2.82	4.6467	4.10	7.1934
0.29	0.3353	1.00	1.2571	1.92	2.8940	2.84	4.6836	4.15	7.3113
0.30	0.3451	1.02	1.2890	1.94	2.9325	2.86	4.7348	4.20	7.4067
0.31	0.3551	1.04	1.3195	1.96	2.9664	2.88	4.7740	4.25	7.5068
0.32	0.3652	1.06	1.3523	1.98	3.0081	2.90	4.8144	4.30	7.6117
0.33	0.3754	1.08	1.3855	2.00	3.0451	2.92	4.8422	4.35	7.6913
0.34	0.3857	1.10	1.4168	2.02	3.0835	2.94	4.8847	4.40	7.8056
0.35	0.3960	1.12	1.4504	2.04	3.1234	2.96	4.9283	4.45	7.8936
0.36	0.4068	1.14	1.4842	2.06	3.1581	2.98	4.9732	4.50	7.9856
0.37	0.4170	1.16	1.5156	2.08	3.2012	3.00	5.0048	4.55	8.1149
0.38	0.4277	1.18	1.5493	2.10	3.2388	3.02	5.0519	4.60	8.2162
0.39	0.4383	1.20	1.5826	2.12	3.2779	3.04	5.0855	4.65	8.2874
0.40	0.4492	1.22	1.6182	2.14	3.3111	3.06	5.1199	4.70	8.3970
0.41	0.4600	1.24	1.6508	2.16	3.3530	3.08	5.1705	4.75	8.5114
0.42	0.4711	1.26	1.6856	2.18	3.3889	3.10	5.2069	4.80	8.5939
0.43	0.4819	1.28	1.7198	2.20	3.4339	3.12	5.2444	4.85	8.7175
0.44	0.4930	1.30	1.7533	2.22	3.4727	3.14	5.2828	4.90	8.8078
0.45	0.5044	1.32	1.7889	2.24	3.5047	3.16	5.3222	4.95	8.9020
								5.00	9.0000

Table 2.2

Table of $A^* = [P_c/(1-P_c)] - 2a\Sigma_{t,U}$ for Slabs of Half-Thickness a

Data from Ref. 10

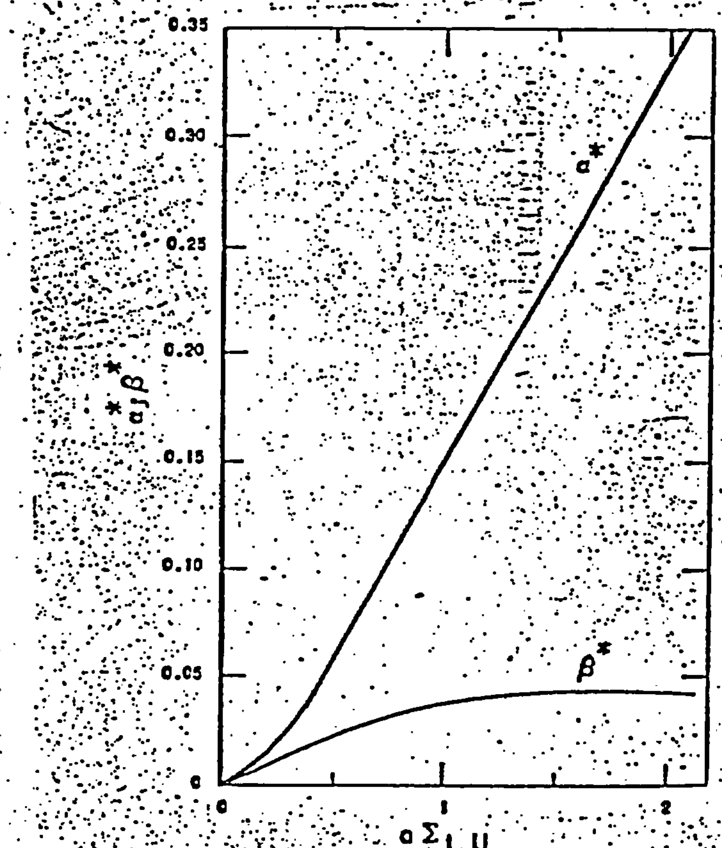


Figure 2.3

α^* and β^* as a function of $a\Sigma_{t,U}$

Reproduced from Ref. 10

$$\Sigma^{1}_{tr} = \text{transport cross section of water, cm}^{-1}$$

For room temperature water, Σ_{tr} is taken to be 2.15 cm^{-1}

This is based on the formula

$$\Sigma_{tr} = \frac{1}{3 \ \Sigma_a \ L^2}$$

A summary of the results of recent L measurements for room temperature water is given in Reference 11. The average value of L at 22°C is 2.787 cm. This value of L, plus a value of Σ_a for H_2O of 0.01984 cm^{-1} was used to obtain Σ_{tr}. This Σ_a corresponds to a water density of 1.0 gm/cm^3 and $\bar{\sigma}_a = 0.594$ (Table 2.1).

The self-shielding factor for water is

$$\left[\frac{\bar{\varphi}_1}{\bar{\varphi}_0}\right]^{-1}$$

Density and Atom Ratios for Uranium Solutions

The following expressions relate the weight ratio of water to U^{235} to the hydrogen to U^{235} atom ratio.

$$\frac{W_{H_2O}}{W_{25}} = \left[\left(\frac{H}{25}\right) \frac{\text{moles H}}{\text{mole } U^{235}}\right] \left(\frac{1 \text{ mole } U^{235}}{235.1 \text{ gm } U^{235}}\right) \left(\frac{18.02 \text{ gm } H_2O}{2 \text{ moles H}}\right)$$

$$\frac{W_{H_2O}}{W_{25}} = 0.03832 \ (H/25)$$

$$(H/25) = 26.10 \ (W_{H_2O}/W_{25})$$

where (H/25) denotes atom (or mole) ratio of hydrogen to U^{235}. A relationship between (H/25) and U^{235} density in UO_2F_2

solutions and in metal-water mixtures is shown in Figure 2.4. Similar data for $UO_2(NO_3)_2$ solutions are shown in Figure 2.5. The data shown in these figures were obtained from References 12 and 13 respectively.

Alloy Density for U-Zr Systems

Density of U-Zr alloy is shown as a function of weight percent U in the alloy in Figure 2.6.

Metal to Water Volume Ratios

For a system containing fuel alloyed and also clad with either zirconium, iron or aluminum, the volume fraction of metal to water is required in the calculation method. This volume ratio can be obtained from the following equations. The derivation is based on a uranium-zirconium alloy but the results apply equally to uranium-aluminum and uranium-iron with appropriate substitutions.

$$\frac{V_{metal}}{V_{H_2O}} = \frac{V_{alloy}}{V_{H_2O}} + \frac{V_{clad}}{V_{H_2O}} \qquad (11)$$

$$\frac{V_{alloy}}{V_{H_2O}} = \left[\left(\frac{W_{H_2O}}{W_{25}}\right)\left(\frac{1\ \text{cm}^3\ H_2O}{1\ \text{gm}\ H_2O}\right)\left(\frac{E\ Z\ \text{gm}\ U^{235}}{\text{gm alloy}}\right)\left(\rho_{alloy}\ \frac{\text{gm alloy}}{\text{cm}^3\ \text{alloy}}\right)\right]^{-1}$$

where E = uranium enrichment, gm U^{235}/gm U
Z = gm U/gm alloy
ρ_{alloy} = density of U-Zr alloy gm/cm^3

$$\frac{V_{alloy}}{V_{H_2O}} = \left[\frac{W_{H_2O}}{W_{25}}\ E\ Z\ \rho_{alloy}\right]^{-1} \qquad (12)$$

$$\left(\frac{W_{Zr}}{W_{25}}\right)_{alloy} = \frac{1 - Z}{E\ Z} \qquad (13)$$

$$\frac{V_{alloy}}{V_{H_2O}} = \frac{(W_{25}/W_{25})_{alloy}}{[\rho_{alloy}](1-Z)(W_{H_2O}/W_{25})} \tag{14}$$

$$\frac{V_{clad}}{V_{H_2O}} = \left[\left(\frac{W_{Zr}}{W_{25}}\right)_{clad} \frac{\text{gm Zr}}{\text{gm 25}}\right]\left[\frac{1}{\rho_{Zr}} \frac{\text{cm}^3\text{ Zr}}{\text{gm Zr}}\right]\left[\frac{1}{\frac{W_{H_2O}}{W_{25}} \frac{\text{cm}^3 H_2O}{\text{gm 25}}}\right]$$

$$\frac{V_{clad}}{V_{H_2O}} = \frac{(W_{Zr}/W_{25})_{clad}}{(\rho_{Zr})(W_{H_2O}/W_{25})}$$

$$\frac{V_{metal}}{V_{H_2O}} = \frac{1}{\left(\frac{W_{H_2O}}{W_{25}}\right)}\left[\frac{1}{(\rho_{alloy})(1-Z)}\left(\frac{W_{Zr}}{W_{25}}\right)_{alloy} + \left(\frac{1}{\rho_{Zr}}\right)\left(\frac{W_{Zr}}{W_{25}}\right)_{clad}\right] \tag{15}$$

where ρ_{Zr} = density of Zr or Zr alloy cladding material, gm/cm^3

In using Figure 2.2 to obtain τ for U-Zr alloys in water, V_{Zr}/V_{H_2O} is taken to be V_{metal}/V_{H_2O} as computed from Eq. (15) Equation 12 may be used if all metal present is fuel alloy.

Sample Calculation of the Maximum k_{eff} for a Thermal, Unreflected Cylinder Containing 20 gm U^{235} per Inch of Length

A standard problem arising in calculating effects of interaction between fueled units is to determine the effective multiplication factor for each body. The method previously described can be used for this type of calculation as shown below.

Assumptions

1. Cylinder consists of UO_2F_2 solution.
2. Uranium enrichment is 93.4% U^{235}, but absorptions in U^{238} are neglected.

Figure 2.4

H/U^{235} Atom Ratio vs. U^{235} Density

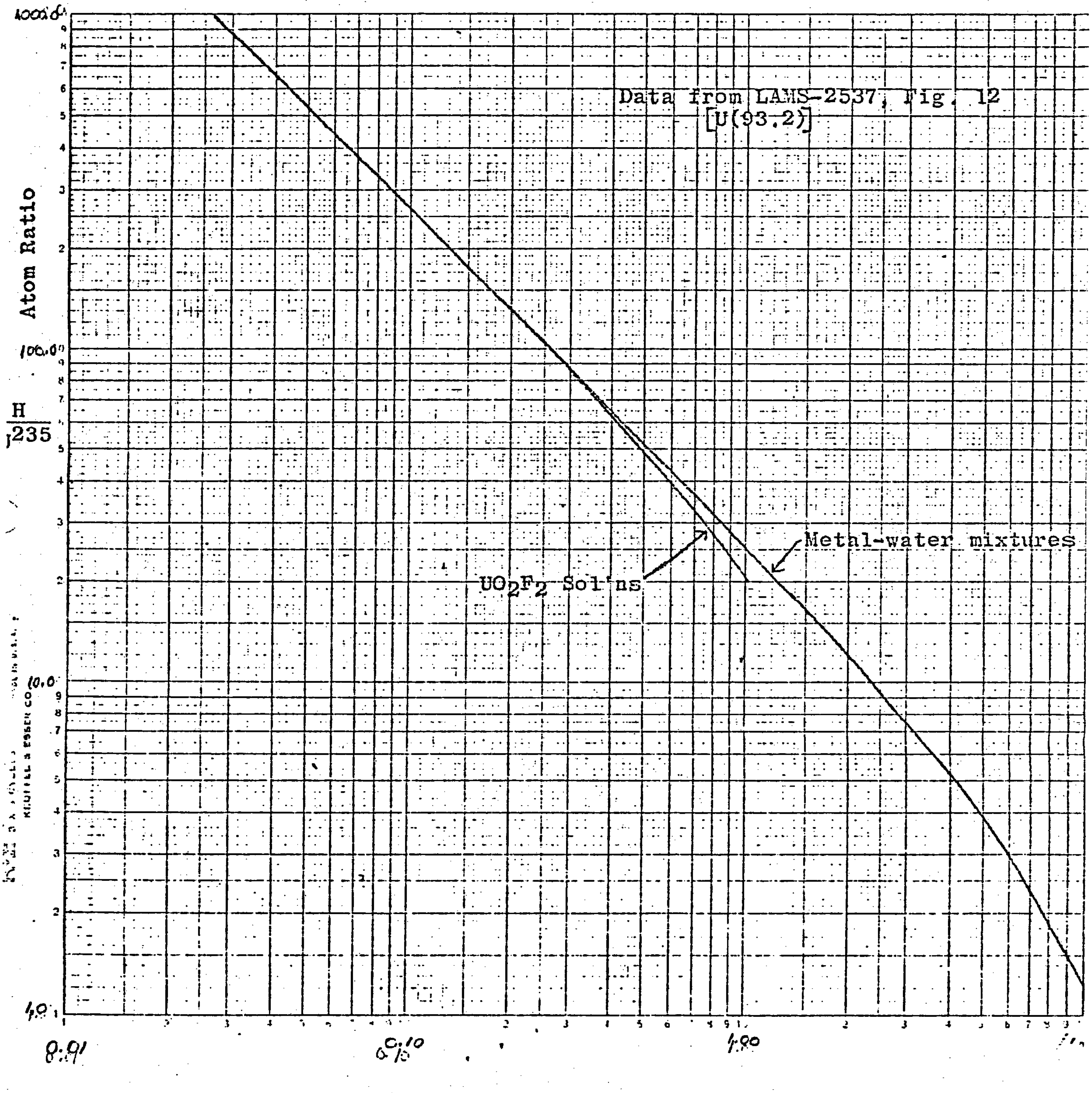
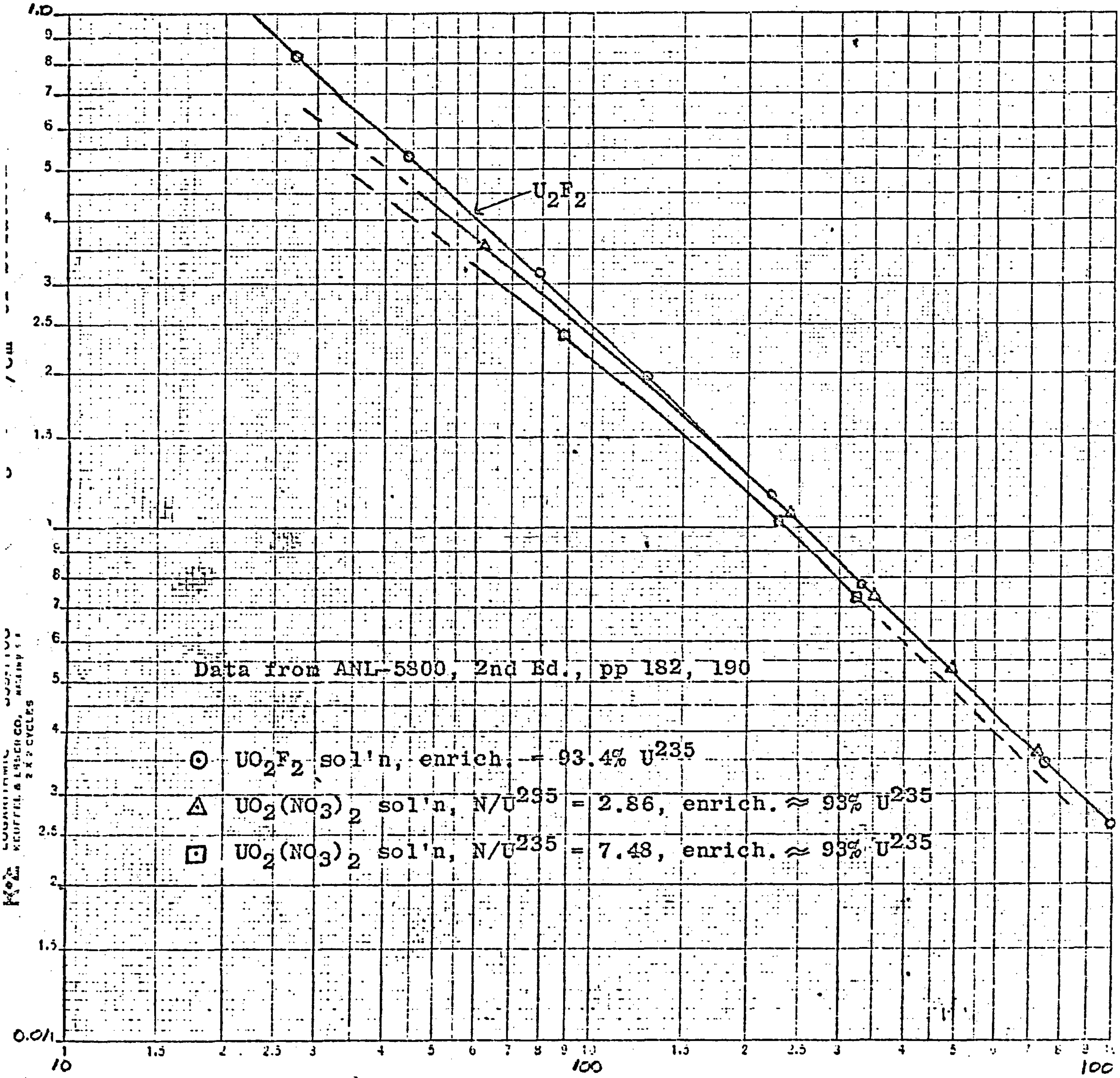


Figure 2.5

U^{235} Density in Solutions



H/U^{235} Atom Ratio

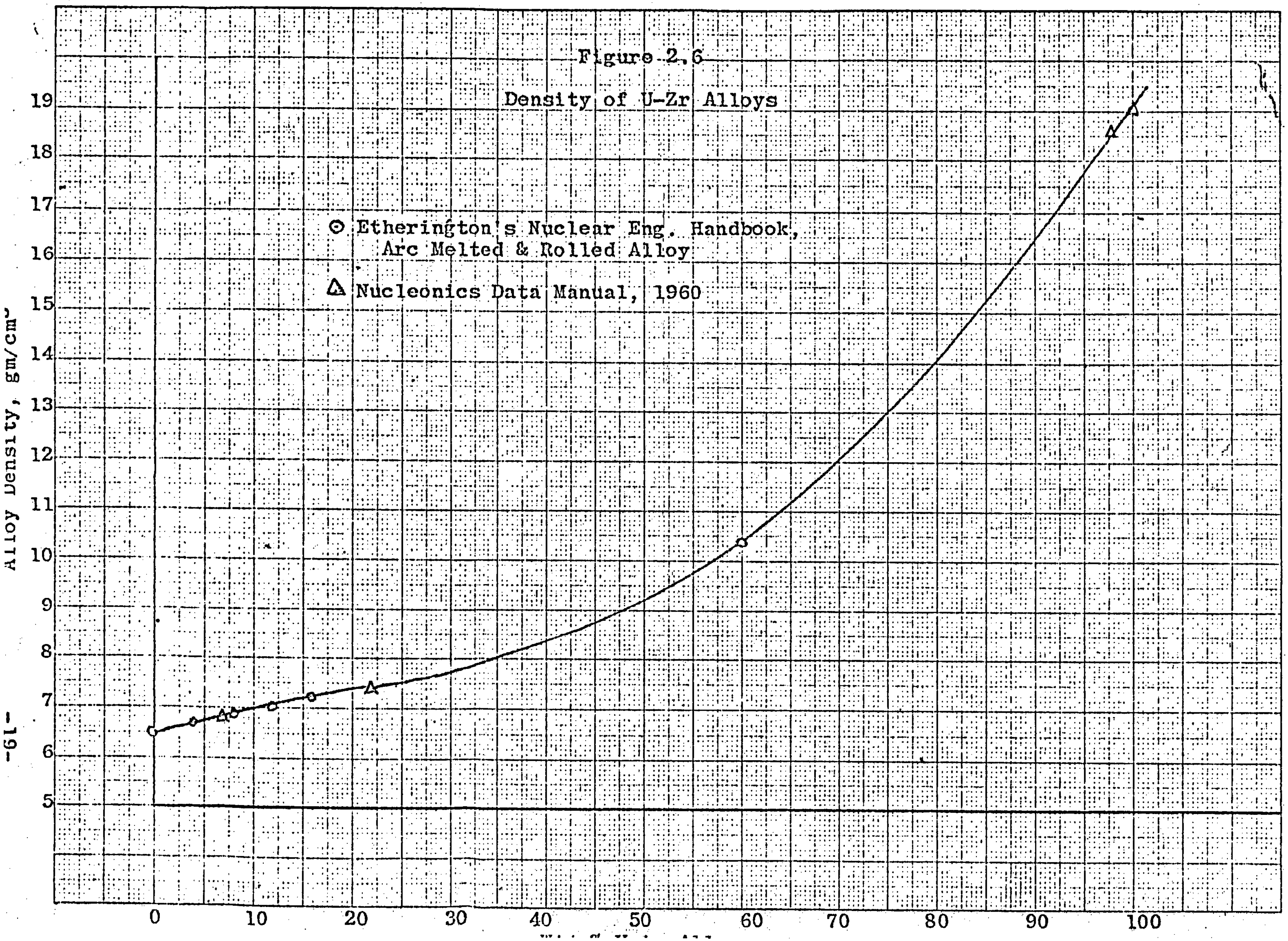


Figure 2.6

Density of U-Zr Alloys

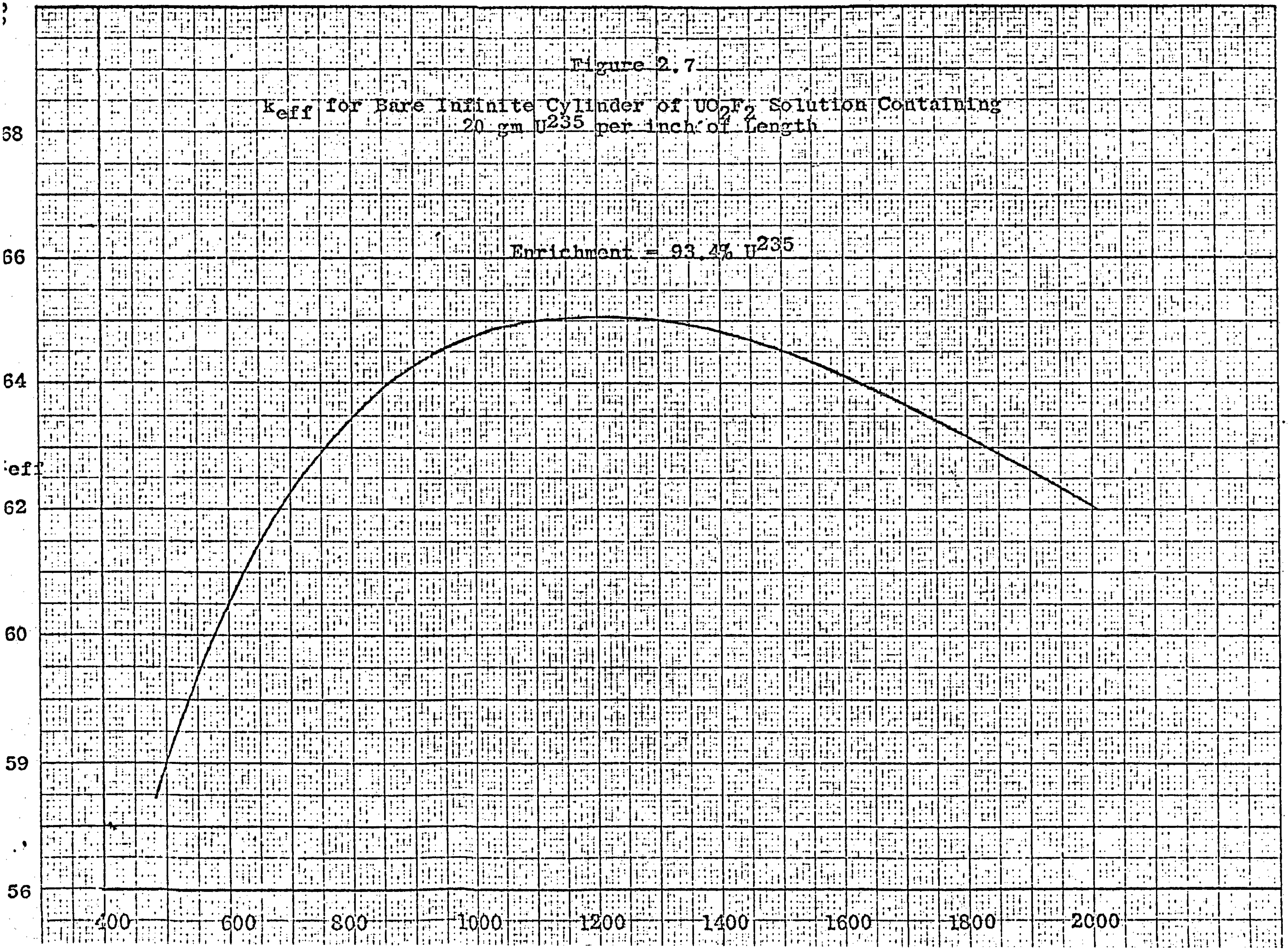


Figure 2.7
k_{eff} for Bare Infinite Cylinder of UO_2F_2 Solution Containing
20 gm U^{235} per inch of Length
Enrichment = 93.4% U^{235}
68
66
64
eff
62
60
59
56
400
600
800
1000
1200
1400
1600
1800
2000

The reactivity recipes described in Section 2.1 were used for this calculation. The following numerical example is presented as an illustration of the calculation technique used:

Choose $H/U^{235} = 1000.$

$$\frac{W_{H_2O}}{W_{25}} = 0.03832\ (1000) = 38.32$$

$$f^{-1} = 1 + 0.01316\ (38.32) = 1.5043$$

$$f = 0.648$$

$$L^2 = 8\ (1 - 0.6648) = 2.68\ \text{cm}^2$$

$$\tau = 26.95\ \text{cm}^2\ \text{(Fig. 2.1)}$$

$$M^2 = 29.63\ \text{cm}^2$$

For a cylinder containing 20 gm U^{235}/inch = 7.874 gm U^{235}/cm,

$$\pi R^2 = \frac{7.874\ \text{gm}\ U^{235}/\text{cm}}{C\ \ \text{gm}\ U^{235}/\text{cm}^3\ \text{solution}}$$

C was obtained from Figure 2.5 of Section 2.1. For

$$H/25 = 1000, \qquad C = 0.0260\ \text{gm}\ U^{235}/\text{cm}^3$$

$$R^2 = \frac{7.874}{\pi(.0260)} = 96.4\ \text{cm}^2$$

$$\therefore \quad R = 9.82\ \text{cm}$$

$$B_g^{\,2} = \left[\frac{2.405}{9.82 + 2.5}\right]^2 = 0.03811\ \text{cm}^{-2}$$

$$k_{eff} = \frac{2.07\ (0.666)}{1 + 29.62\ (0.03811)} = 0.648$$

This procedure was repeated for several other values of (H/25). For values of H/25 greater than 1000, the relationship between (H/25) and U^{235} concentration was obtained from the data on page 11 of TID-7028. K_{eff} is shown as a function of (H/25) in Figure 2.7. The maximum k_{eff} is 0.652.

2.2 Calculation Method for Undermoderated, Enriched Uranium Water Systems

If the H/U^{235} atom ratio is less than 20 for uranium solutions and uranium metal-water mixtures or the metal to water volume ratio is greater than 1.70 for uranium-alloy-* water mixtures, the system is considered undermoderated and the following calculation procedure is recommended. This method is based on the standard four factor formulae with the Fermi age non-leakage probability to determine k_{eff}, i.e.:

$$k_{eff} = \frac{\eta \epsilon pf}{1+B^2M^2} \qquad (16)$$

However, the method of determining values for use in Eq. 16 are modified to account for neutron events occurring at epithermal and fast energies.

η

η, defined as $\nu\sigma_f/\sigma_a$ for U^{235}, which is the only fissile nuclide considered in the calculative method, is a function of the neutron spectrum in the multiplying medium, and is therefore a function of the degree of moderation. For well-moderated systems, a value of 2.07 is used corresponding to thermal neutron energies. Data on the variation of η with neutron energy are presented in Table 2.3[14]. At energies above thermal, η is generally less than 2.07 except for certain narrow energy bands in which η may be as high as 2.2-2.3. Above 0.25 Mev, η is consistently above 2.07 and, for fission neutron energies (1-2 Mev), η increases to about 2.5.

The upper limit for the neutron energy distribution in an undermoderated system is obtained by assuming that the energy distribution function is identical to the prompt neutron spectrum from U^{235} fission. The average energy of prompt neutrons found by integrating over the prompt neutron spectrum is 1.98 Mev while the most probable energy is 0.85 Mev[15]. The average neutron energy would most certainly be lower than the average from the prompt neutron spectrum even for a 100% uranium system because of inelastic scattering in the uranium. Therefore, the energy averaged value of η = 2.4 is recommended for undermoderated systems. This value of η

*Alloy refers to uranium metal alloyed or clad with zirconium, iron or aluminum.

Table 2.3

U^{235} Cross Sections

E(MeV)	Total	Scattering	Absorption	Fission	Capture	α	η	σ_{non}	σ_{in}	$\bar{\nu}$
0.6	7.22	4.67	1.342	1.17	0.178	0.152	2.161	2.55	1.205	2.49
0.7	6.94	4.34	1.321	1.16	0.161	0.139	2.195	2.60	1.279	2.50
0.8	6.78	4.13	1.301	1.17	0.131	0.112	2.257	2.65	1.349	2.51
0.9	6.63	3.93	1.320	1.20	0.120	0.100	2.295	2.70	1.380	2.525
1.0	6.55	3.81	1.325	1.22	0.105	1.086	2.330	2.74	1.415	2.53
1.2	6.54	3.73	1.320	1.23	0.090	0.073	2.377	2.81	1.490	2.55
1.4	6.59	3.67	1.316	1.24	0.076	0.061	2.422	2.92	1.604	2.57
1.6	6.75	3.75	1.348	1.28	0.065	0.056	2.453	3.00	1.652	2.59
1.8	6.89	3.88	1.369	1.31	0.059	0.045	2.497	3.01	1.651	2.61
2.0	7.10	4.01	1.374	1.32	0.054	0.041	2.526	3.09	1.716	2.63
2.2	7.35	4.25	1.370	1.32	0.050	0.038	2.563	3.10	1.730	2.66
2.4	7.50	4.40	1.364	1.32	0.044	0.033	2.604	3.10	1.736	2.69
2.6	7.60	4.50	1.351	1.31	0.04[illegible]	0.031	2.638	3.10	1.749	2.72
2.8	7.75	4.66	1.337	1.30	0.037	0.028	2.675	3.09	1.753	2.75
3.0	7.80	4.73	1.305	1.27	0.035	0.0275	2.706	3.07	1.765	2.78
3.5	7.80	4.75	1.260	1.23	0.030	0.024	2.783	3.05	1.790	2.85
4.0	7.80	4.76	1.206	1.18	0.026	0.022	2.867	3.04	1.834	2.93
4.5	7.60	4.57	1.174	1.15	0.024	0.021	[illegible]	3.03	1.856	3.01
5.0	7.35	4.33	1.160	1.14	0.020	0.0175	3.027	3.02	1.860	3.08
5.5	7.15	4.14	1.158	1.14	0.018	0.0158	3.111	3.01	1.802	3.16
6.0	6.90	3.91	1.186	1.17	0.016	0.0136	3.188	2.99	1.684	3.23
6.5	6.90	3.92	1.375	1.36	0.015	0.0110	3.269	2.98	1.405	3.305
7.0	7.00	4.03	1.534	1.52	0.014	0.009	3.350	2.97	1.156	3.38
7.5	6.90	3.94	1.683	1.67	0.013	0.007	3.421	2.96	0.867	3.455
8.0	6.90	3.96	1.742	1.73	0.012	0.007	3.505	2.94	0.678	3.53
8.5	6.85	3.92	1.811	1.80	0.011	0.006	3.583	2.93	0.539	3.605
9.0	6.83	3.91	1.810	1.80	0.010	0.006	3.658	2.92	0.510	3.68
9.5	6.83	3.92	1.809	1.80	0.009	0.005	3.736	2.91	0.491	3.755
10.0	6.80	3.92	1.799	1.79	0.009	0.005	3.811	2.88	0.421	3.83
10.5	6.79	3.94	1.749	1.74	0.009	0.005	3.886	2.85	0.411	3.905
11.0	6.68	3.85	1.728	1.72	0.008	0.005	3.960	2.83	0.402	3.98
11.5	6.62	3.81	1.768	1.76	0.008	0.004	4.039	2.81	0.392	4.055
12.0	6.47	3.68	1.818	1.81	0.008	0.004	4.114	2.79	0.372	4.13
12.5	6.24	3.47	1.877	1.87	0.007	0.003	4.192	2.77	0.353	4.205
13.0	6.17	3.43	1.967	1.96	0.007	0.003	4.267	2.74	0.343	4.280
13.5	6.21	3.49	2.077	2.07	0.007	0.003	4.342	2.72	0.333	4.355
14.0	6.24	3.54	2.186	2.18	0.006	0.003	4.416	2.70	0.324	4.43
14.5	6.26	3.57	2.216	2.21	0.005	0.003	4.492	2.69	0.314	4.505
15.0	6.29	3.62	2.246	2.24	0.006	0.003	4.513	2.68	0.304	4.565

E(keV)	Total	Scattering	Absorption	Fission	Capture	α	η	σ_{non}	σ_{in}	$\bar{\nu}$
5.572	16.82	11.28	5.54	3.97	1.57	0.396	1.741	σ_{abs}	0.000	2.43
5.686	16.99	11.27	5.72	4.10	1.62	0.395	1.742			
5.804	19.51		8.24	5.91	2.33	0.394	1.743			
5.926	17.03	11.26	5.77	4.14	1.63	0.393	1.744			
6.051	16.53		5.27	3.78	1.49	0.393	1.745			
6.181	16.00	11.25	4.75	3.41	1.34	0.392	1.746			
6.314	16.54	11.24	5.30	3.81	1.49	0.392	1.746			
6.452	16.02	11.23	4.79	3.44	1.35	0.391	1.747			
6.595	16.12		4.89	3.52	1.37	0.390	1.748			
6.742	18.01		6.78	4.88	1.90	0.389	1.749			
6.895	16.31	11.22	5.09	3.67	1.42	0.388	1.751			
7.052	16.06		4.84	3.49	1.35	0.388	1.751			
7.215	16.30	11.21	5.09	3.67	1.42	0.387	1.752			
7.384	16.55		5.34	3.85	1.49	0.386	1.753			
7.559	15.92	11.20	4.72	3.41	1.31	0.385	1.755			
7.740	16.15		4.95	3.58	1.37	0.384	1.756			
7.927	15.74	11.19	4.55	3.29	1.26	0.383	1.757			
8.122	16.72	11.18	5.54	4.01	1.53	0.382	1.758			
8.324	15.85	11.17	4.68	3.39	1.29	0.381	1.760			
8.533	15.88	11.17	4.71	3.41	1.30	0.380	1.761			
8.751	15.64	11.16	4.48	3.25	1.23	0.379	1.762			
8.976	16.06	11.15	4.91	3.56	1.35	0.379	1.762			
9.211	16.06	11.15	4.91	3.56	1.35	0.378	1.763			
9.455	15.57	11.14	4.43	3.22	1.21	0.377	1.765			
9.709	15.77	11.13	4.64	3.37	1.27	0.376	1.766			
9.973	15.97	11.14	4.83	3.51	1.32	0.375	1.767			
0.01 MeV	15.1	10.70	4.40	3.20	1.20	0.375	1.767	4.40		
0.015	14.75	10.94	3.81	2.80	1.01	0.359	1.788	3.81	–	
0.02	14.42	10.93	3.461	2.56	0.901	0.352	1.797	3.486	0.025	
0.025	14.22	10.88	3.305	2.45	0.855	0.349	1.801	3.34	0.035	
0.03	13.91	10.58	3.192	2.37	0.822	0.347	1.804	[illegible]	0.050	
0.04	13.31	10.44	2.981	2.23	0.751	0.337	1.818	3.071	0.090	
0.05	13.21	10.27	2.815	2.12	0.695	0.328	1.830	2.94	0.125	
0.06	12.98	10.13	2.681	2.03	0.651	0.321	1.840	2.85	0.169	
0.07	12.70	9.95	2.545	1.94	0.605	0.312	1.852	2.75	0.205	
0.08	12.36	9.67	2.432	1.87	0.562	0.301	1.868	2.69	0.258	
0.09	12.13	9.50	2.335	1.80	0.535	0.297	1.874	2.63	0.295	
0.10	11.88	9.26	2.247	1.75	0.497	0.284	1.900	2.62	0.371	2.44
0.15	11.14	8.66	1.936	1.55	0.386	0.249	1.962	2.48	0.544	2.45
0.20	10.43	8.00	1.749	1.43	0.319	0.223	2.003	2.43	0.681	2.45
0.25	9.65	7.24	1.632	1.36	0.272	0.200	2.050	2.41	0.778	2.46
0.3	9.15	6.73	1.583	1.340	0.243	0.181	2.093	2.42	0.837	2.46
0.4	8.26	5.76	1.495	1.29	0.205	0.159	2.131	2.50	1.005	2.47
0.5	7.6	5.06	1.415	1.23	0.185	0.150	2.157	2.54	1.125	2.48

corresponds to an average neutron energy of 1.3 Mev. The average neutron energy in the EBR-II uranium metal fast critical configuration was calculated to be only 0.5 Mev[16] with a corresponding value of $\eta^{255} = 2.157$ from Table 2.3.

Thermal Utilization (f), Fast Fission Factor (ϵ), Resonance Escape Probability (p)

Parasitic absorptions by materials such as H, Zr, Al and Fe are relatively small in highly undermoderated systems. In the absence of strong resonance absorbers (e.g., hafnium) or other poisons, the value of the thermal utilization approaches unity. The conservative assumption is made in this model that

$$f \, p \, \epsilon = 1.0$$

L^2

Since there will be relatively few thermal neutrons in small, highly undermoderated systems, a value of zero for L^2 is recommended. This is conservative.

τ

τ data for UO_2F_2 solutions and for Zr-water mixtures are shown in Section 2.1. In both cases, τ is seen to increase as hydrogen content is reduced. The following limiting values of τ are recommended for undermoderated systems:

31.5 cm^2 for U solutions with $H/U^{235} \leq 34$
28.0 cm^2 for U metal-water mixtures with $H/U^{235} \leq 120$

60 cm^2 for U Zr alloy-water mixtures with $V_m/V_{H_2O} \geq 1.70$
73 cm^2 for U aluminum-water mixtures with $V_m/V_{H_2O} \geq 1.50$
48 cm^2 for U iron-water mixtures with $V_m/V_{H_2O} \geq 1.74$

These limits are based on the data shown in Figures 2.1 and 2.2 respectively of Section 2.1. The limits have been conservatively established to avoid those areas where the age to thermal energies becomes very large at low hydrogen atom densities.

${B_g}^2$

${B_g}^2$ is computed with the usual formulas for spheres, cylinders and slabs, but requires a knowledge of extrapolation length for bare systems and reflector savings for reflected systems. These quantities depend on neutron spectrum, reflector properties, and the curvature of the fuel region boundary. The following expressions yield conservative results compared with data presented in Reference 17.

For bare systems

$$\delta = 5.46 - 0.1480\ (H/U^{235}),\ \text{cm} \qquad H/U^{235} \leq 20 \qquad (17)$$

For systems surrounded by a thick water reflector

$$\delta = 7.30 - 0.040\ (H/U^{235}),\ \text{cm} \qquad H/U^{235} \leq 20 \qquad (18)$$

$\underline{k_{eff}}$

For undermoderated systems, with $H/U^{235} \leq 20$ for solutions or U metal-water mixtures, or with $U_{metal}/V_{H_2O} > 1.70$ for U alloy*-water mixtures, the expression for k_{eff} becomes

$$k_{eff} = \frac{2.40}{1 + \tau B_g^2} \qquad (19)$$

2.3 Calculation Method for Unmoderated and Unreflected, Highly Enriched Uranium Systems

The calculation procedure presented in Section 2.2 is valid for undermoderated uranium systems with H/U^{235} ratios between 20 and 0. The results of the method presented in Section 2.2 is overly conservative when applied to unreflected systems containing no moderator (see Section 3.4, Table 3). As an alternate procedure for unmoderated and unreflected systems, k_{eff} can be calculated from:

$$k_{eff} = \frac{k_{\infty}}{1+M^2\beta^2} \qquad (20)$$

*Alloy refers to uranium metal alloyed or uniformly mixed and/or clad with zirconium, aluminum, iron or materials of similar nuclear characteristics.

where: k_∞ = 2.17

M^2 = 16.97 cm^2

δ = 3.24 cm

These constants are recommended in Reference 18.

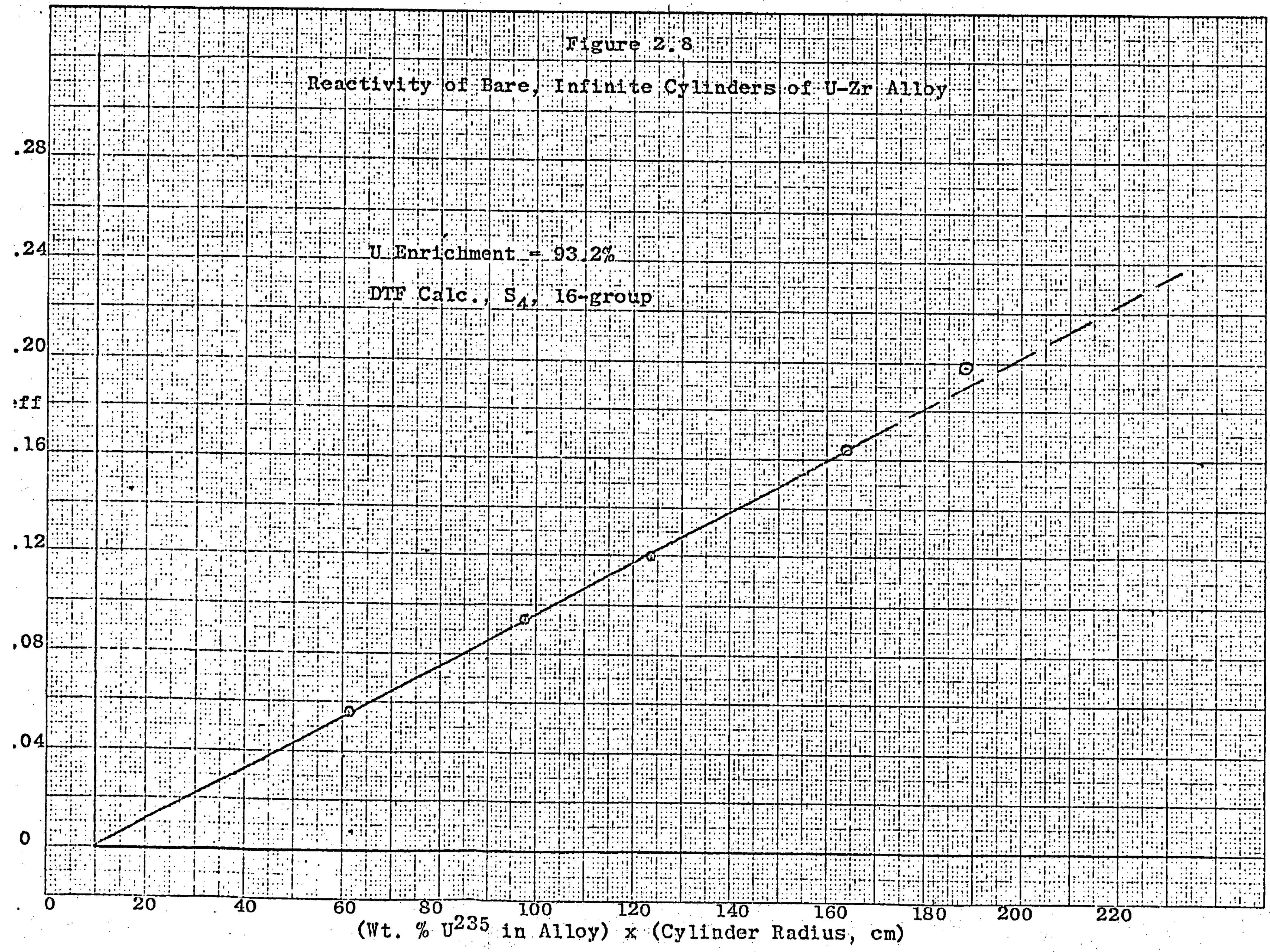
These sets of constants have been checked against DTF[19] calculations of bare U metal spheres having a density of 18.70 gm/cm^3 and on enrichment of 93.2% U^{235}. All DTF calculations used the Hansen-Roach[20] 16-group cross section library and the S_4 approximation. The following results were obtained.

	k_{eff}		
kg U^{235}	Sphere radius, cm	Calculated from Eq. 20	DTF
46	8,573	0.986	0.984
30	7.434	0.879	0.872
18.5	6.327	0.767	0.757
13	5.626	0.693	0.679
6.5	4.465	0.568	-
18.5 @ 60% of above density		-	0.549

The results of the DTF calculations of k_{eff} for unreflected, unmoderated U-Zr alloy cylinders are shown in Figure 2.8.

Figure 2.8

Reactivity of Bare, Infinite Cylinders of U-Zr Alloy



3.0 COMPARISON OF RECOMMENDED METHODS WITH EXPERIMENT

In order to test the recommended calculation method, a number of critical experiments have been calculated. The experiments represent the range of variables for thermal and undermoderated systems. In all instances, the results obtained with the calculation method are conservative.

3.1 Calculations of Thermal UO_2F_2 Solution Critical Experiments

In order to check the accuracy and degree of conservativeness of the calculation method described in the preceding sections, the method was applied to a number of UO_2F_2 solution critical experiments. A selection of these experiments, including both bare and reflected systems, and systems having spherical, cylindrical and rectangular parallelepiped configurations were calculated. The results are presented in Table 3.1.

In general, the calculated values of k_{eff} are three to six percent above the measured value of unity except at values of H/U^{235} of 1000 or higher. At high H/U^{235} ratios, k_{eff} is overestimated by 2% or less, but is never underestimated. These results show that this calculation method is reasonably accurate and suitably conservative for homogeneous mixtures of fully enriched uranium and water over a H/U^{235} atom ratio range from about 27 to about 1400.

Reference 12 presents a compilation of calculated and experimental critical data on homogeneous mixtures and solutions of highly enriched uranium and water. The figures presented in this reference are frequently used in criticality safety studies. The thermal calculation method described here was compared with the curves of LAMS-2537(12) for bare and reflected spheres and cylinders. The results of this comparison are presented in Table 3.2.

The calculation method presented in Section 2.1 has been tested against low enrichment UO_2F_2 aqueous solution critical data(26) as shown in Table 3.3. The procedure is the same as shown in Section 2.1 and detailed in the previous sample calculations with the exception that the value of η is determined from Eq. 10 . The age to thermal energies was determined from Figure 2.2.

3.2 Calculation of Thermal Homogeneous U^{235}-Zr-H Critical Experiments

SFR Critical Experiment

Reference 21 describes a critical experiment performed with a homogeneous core in the shape of a rectangular paralle-

Table 3.1

Calculation of UO_2F_2 Critical Experiments[13]

Geometry	Reflector*	H/25	Diameter, cm	Height, cm	Calculated k_{eff}**
Sphere	H_2O	76.1	23.0	-	1.058
	H_2O	126.5	23.6	-	1.060
	none	1112	55.8	-	1.009
	H_2O	1270	55.8	-	1.012
	none	1393	69.2	-	1.003
Cylinder	H_2O	27.1	15.2	89.3	1.047
	H_2O	44.3	16.5	38.7	1.061
	H_2O	290	20.3	40.1	1.079
	H_2O	499	25.4	35.2	1.057
	H_2O	755	38.1	27.10	1.046
	H_2O	999	38.1	44.30	1.022
Cylinder	none	27.1	25.4	38.9	1.045
	none	83.1	25.4	34.4	1.041
	none	331	38.1	22.9	1.044
	none	755	38.1	43.6	1.013
Slab, 50.8 cm x 50.8 cm in cross section					
	H_2O	27.1	-	6.3	1.026
	H_2O	72.4	-	6.2	1.041
	none	27.1	-	15	1.031
	none	331	-	17.9	1.044

* When present, H_2O reflector was effectively infinite on all sides.
**Measured k_{eff} = 1.0 in all cases.

Table 3.2

Calculation of Critical Data in LAMS-2537[12]

Geometry	Reflector*	H/25	Critical Diameter, in.	Critical Vol., liters	Calc. k_{eff}
Infinite Cylinder	H_2O	20(mix.)	5.07	-	1.040
	H_2O	20(sol'n)	5.9	-	1.060
	H_2O	100	5.7	-	1.031
	H_2O	500	8.0	-	1.052
	H_2O	1000	11.8	-	1.007
Infinite Cylinder	none	20(sol'n)	9.5	-	1.101
	none	100	8.6	-	1.007
	none	500	10.9	-	1.037
	none	1000	14.9	-	1.005
Sphere	H_2O	20(sol'n)	-	7.4	1.121
	H_2O	100	-	7.0	1.054
	H_2O	500	-	16.0	1.065
	H_2O	1000	-	46.0	1.019
Sphere	none	20(sol'n)	-	16.5	1.061
	none	100	-	14.0	1.003
	none	500	-	27.0	1.028
	none	1000	-	69.0	1.080

* When present, reflector is effectively infinite.
Sphere data from Fig. 10 of LAMS-2537.
Cylinder data from Fig. 13 of LAMS-2537.

Table 3.3

Calculated Results for Critical $U(4.9)O_2F_2$ Aqueous Solutions[26]

A. Reflected Assemblies

H/235	Geometry	Diameter (cm)	Height (cm)	Calculated k_{eff}
495	Cylinder	38.1	41.7	1.0747
495	Cylinder	30.7	173.2	1.0694
524	Cylinder	38.1	44.8	1.0783
643	Cylinder	76.2	23.9	1.0602
735	Cylinder	76.2	24.2	1.0275
735	Cylinder	38.1	153.0	1.0475
994	Cylinder	76.2	37.9	1.0166
1099	Sphere	69.3	-	1.0051

B. Bare Assemblies

H/235	Geometry	Diameter (cm)	Height (cm)	Calculated k_{eff}
524	Cylinder	50.8	38.7	1.0525
643	Cylinder	76.2	28.9	1.0159
735	Cylinder	76.2	31.5	1.0075
994	Cylinder	76.1	44.8	.9982
1002	Sphere	69.3	-	1.0033

lepiped 6 in. x 30 in. x 40.5 in. All sides were reflected by 20 inches of water. The core had the following composition:

Material	atoms/cm^3 x 10^{-24}
U^{235}	0.0001717
U^{238}	0.0000120
Zr	0.02132
H	0.03368
O	0.01233
C	0.005020

The composition of this core closely resembles that of a homogeneous mixture of uranium and zirconium metal in water at room temperature, with a V_{Zr}/V_{H_2O} ratio of 0.986, based on 1.0 gm/cm^3 and 6.5 gm/cm^3 for water and zirconium respectively. The corresponding value of τ is 46.0 cm^2 (see Figure 2.2). The calculated value of k_{eff} was 1.092, substantially higher than the measured value of unity.

SHA Critical Experiments

Critical experiments have been performed at KAPL on homogeneous U^{235}-Zr-H systems. These data, which are reported in Reference 22, were obtained in the Solid Homogeneous Critical Assembly facility.

The cores of these critical experiments were made up of blocks having the following composition:

Material	Ni, atoms/cm^3 x 10^{-24}
H	0.02555
C	0.01235
Zr	0.01565
O	0.03156
Al	0.0002481
U^{235}	0.0001116
U^{238}	0.00000834

These blocks were assembled to form rectangular parallelepiped cores whose critical dimensions depended on the reflector material and thickness. The use of a split-bed facility eliminated the need for extraneous structural material and control rods.

This core composition does not closely correspond to a mixture of Zr metal and water at normal room temperature densities, but corresponds to a reduced density mixture τ for the core was computed in the following manner: The τ data presented in Section 2.1 can be analyzed in terms of N_{Zr}/N_H rather than V_{Zr}/V_{H_2O} based on the room temperature densities of 1.0 gm/cm^3 and 6.5 gm/cm^3 for water and Zr respectively.

$$\frac{N_{Zr}}{N_H} = \left[\left(\frac{V}{V_{H_2O}}\right) \frac{cm^3\ Zr}{cm^3\ H_2O}\right] \left(\frac{6.5\ gm\ Zr}{cm^3\ Zr}\right)\left(\frac{1\ mole\ Zr}{91.22\ gm\ Zr}\right) \times$$

$$\left(\frac{1\ cm^3\ H_2O}{1\ gm\ H_2O}\right) \left(\frac{18.02\ gm\ H_2O}{2\ moles\ H}\right)$$

$$\frac{N_{Zr}}{N_H} = 0.6420 \left(\frac{V_{Zr}}{V_{H_2O}}\right)$$

or, equivalent $\frac{V_{Zr}}{V_{H_2O}} = 1.558\left(\frac{N_{Zr}}{N_H}\right) = 1.558 \left[\frac{0.01565}{0.02555}\right] = 0.954$

$\tau = 45.4\ cm^2$ from Figure 2.2.

For a mixture of Zr metal and water having $V_{Zr}/V_{H_2O} = 0.954$, which is equivalent to a water volume fraction of 0.5118,

$$N_H = \left(\frac{1\ gm\ H_2O/cm^3\ H_2O}{18.02\ gm\ H_2O/mole\ H_2O}\right) \left(0.6023 \times 10^{24}\ \frac{molecules}{mole}\right) \times$$

$$\left(\frac{2\ atoms\ H}{molecule}\right) \left(0.5118\right)$$

$$N_H = 0.03421 \times 10^{24}\ atoms/cm^3$$

The actual value of N_H is 0.02555×10^{24} $atoms/cm^3$. Therefore, the core material corresponds to a reduced density mixture of Zr and water. It was assumed that τ is inversely proportional to the square of the core material density.

$$\tau = 45.4 \left[\frac{0.03421}{0.02555}\right]^2 = 81.4 \text{ cm}^2$$

Similarly, L^2 was computed from

$$L^2 = 8(1-f)\left[\frac{0.03421}{0.02555}\right]^2 = 0.56 \text{ cm}^2$$

This latter correction has a negligible effect on reactivity because L^2 is very small compared with τ.

Thermal utilization was computed to be 0.961 in the usual manner for a homogeneous medium.

$$k_{eff} = \frac{2.07(0.961)}{1 + (82.0)(Bg^2)}$$

Core #5 — No Reflector

The dimensions of this core are 18.1 in. x 20 in. x 20 in. and the corresponding Bg^2 was computed to be 0.01014 cm^{-2}. The computed value of k_{eff} is 1.086, which is about 8% higher than the measured value of 1.002.

Core #1 - 4-Inch Polyethylene Reflector All Around

The critical core dimensions are 18.1 in. x 20 in. x 16 in. The core was completely surrounded by a polyethylene reflector four inches thick.

The reflector savings due to this polyethylene reflector is uncertain. The hydrogen density of the polyethylene used is given as 0.07861 x 10^{24} atoms/cm^3, which is about 17.6% greater than that of water. About 6 inches of liquid water constitute an effectively infinite reflector. This corresponds to about 5.1 inches of polyethylene, based on an equivalent surface density of hydrogen. The 4-inch reflector should, therefore, constitute something less than an infinite reflector and the corresponding reflector savings should be somewhat less than the 6.5 cm used for infinite water reflectors. k_{eff} was computed as a function of reflector savings. The following results were obtained:

Reflector Savings, cm (δ)	$Bg^2 cm^{-2}$	k_{eff}
7.5	0.008121	1.194
6.5	0.008692	1.161
5.5	0.009324	1.127
4.5	0.01003	1.092
2.5	0.01170	1.015

The measured value of k_{eff} is 1.003. These results show that k_{eff} is predicted conservatively even if the reflector savings is taken to be 2.5 cm which corresponds to the extrapolation length for a bare system. If the reflector savings is assumed to be 4.5 cm, then the predicted k_{eff} is about 9% conservative.

3.3 Calculation of Thermal Heterogeneous U-Zr-H_2O Critical Experiments

1. Bettis U-Zr Alloy Plate Core Critical Experiment Performed in the HTTR

Reference 24 describes critical experiments performed on rectangular parallelepiped cores made up of rectangular fuel assemblies of U-Zr alloy plates. These fuel assemblies are illustrated in Figure 3.1 (reproduced from Ref. 23).

One core contained 44 fuel assemblies, no poison septum, and had dimensions of 6.302 in. x 17.320 in. x 30.0 in. The fuel was 9 wt % highly enriched U-Zr alloy clad with Zr-2. All structural material was also Zr-2. The fueled meat thickness was 0.060 in. and the clad thickness was 0.015 in. Water gaps between fuel plates were about 0.090 in. thick and the metal to water volume ratio for the core was given as 1.2. There was very little water between fuel assemblies. A thick water reflector completely surrounded the core.

Self-shielding factors for the fuel plates and the water in channels were computed using the method described in Section 2.1. The self-shielding factor for the Zr structural material was taken to be the same as that computed for the water channels. The core composition and computed self-shielding factors are given in the following table.

Material	Mass of Mat'l in Core, gm	Disadvantage Factor (calculated)
U^{235}	8,885	1.0
Zr in meat	91,000	1.0
Zr in clad and frame	56,700	0.905
Zr structure	46,844	0.885 (assumed)
H_2O	24,990	0.885

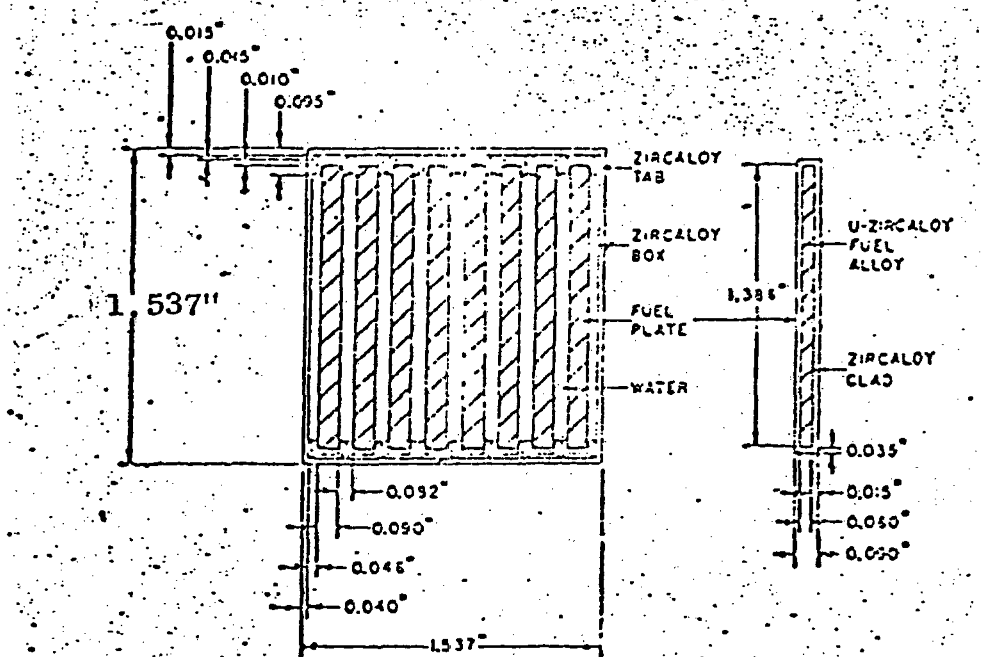


FIGURE 3.1 Schematic Arrangement of Fuel Module and Fuel Element for Slab Experiments in the HTTF.

Measured value of k_{eff} = 1.013
Calculated value of k_{eff} = 1.078

2. Other Check Calculations on Heterogeneous U^{235}-Zr-H_2O Critical Experiments

A series of clean critical experiments performed with thin U-Zr alloy fuel strips and Zr structural material in water is described in Reference 24, page 1150-1174. The cores were made up of arrays of square zirconium module boxes arranged to form rectangular parallelepipeds. The cores were flooded and completely reflected by water. The core arrangement is shown in Figure 3.2 (reproduced from Ref. 24).

Each module box was 1.109 inches square. Three or four Zircaloy strips were inserted into each module box. U-Zr alloy fuel shims (thin strips) were fastened to each Zircaloy strip with two aluminum clips. A typical module box arrangement is shown in Figure 3.3 (reproduced from Ref. 24). Zircaloy strips of two thicknesses, 110 mils and 70 mils, were used in these experiments. Fuel loading was varied in the series of experiments by sometimes placing a fuel shim on each side of each Zircaloy strip, and sometimes placing a fuel shim on only one side of each Zircaloy strip. In one case, Core #75, two fuel shims were placed on each side of each Zircaloy strip.

Each fuel shim was 0.020 inches thick and consisted of U-Zr alloy containing 19.8 wt % U. The uranium was enriched to 93.14% U^{235}.

Reference 24 presents tabulations of metal cross sectional area for each component of a module, and fuel density, metal to water volume ratio, core and module configuration, and measured k_{eff} for each experiment. These are summarized in Table 3.4. The measurements were made in such a way that these k_{eff} values correspond to a clean uniform core with no control poisons present. In these experiments, there were no water gaps between adjacent modules. In all cases, the fueled core height was 36 inches. The axial reflectors contained Zr metal extensions of the module boxes as well as water.

The following assumptions were made in calculating these critical experiments with this hand-calculation recipe.

1. The Zircaloy strips were assumed to be pure Zr and the 52S aluminum clips were assumed to be pure Al.

2. A reflector savings of 6.5 cm was assumed for all directions. The effects of metal in the axial reflectors were neglected. These effects

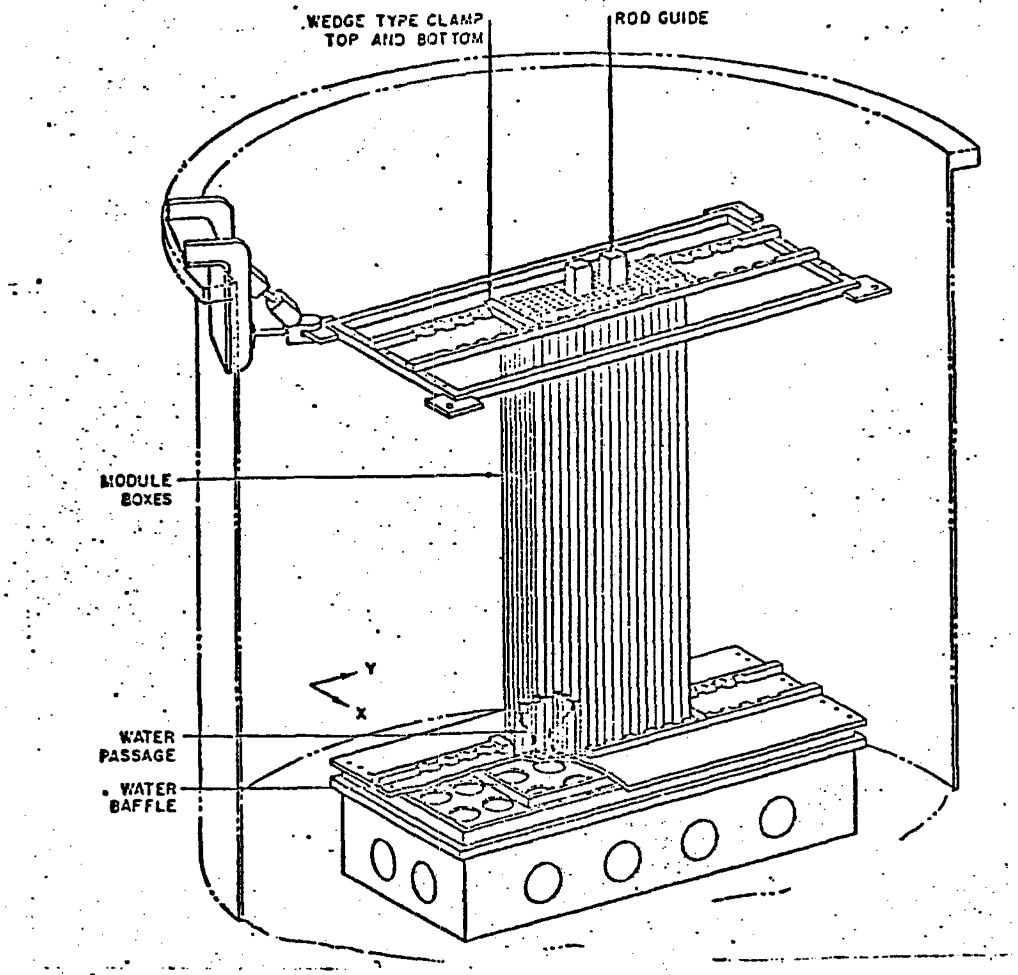


Figure 3.2

Perspective View of Metal Slab Assembly of Small Module Box Construction Showing the Highly Flexible Nature of the Construction

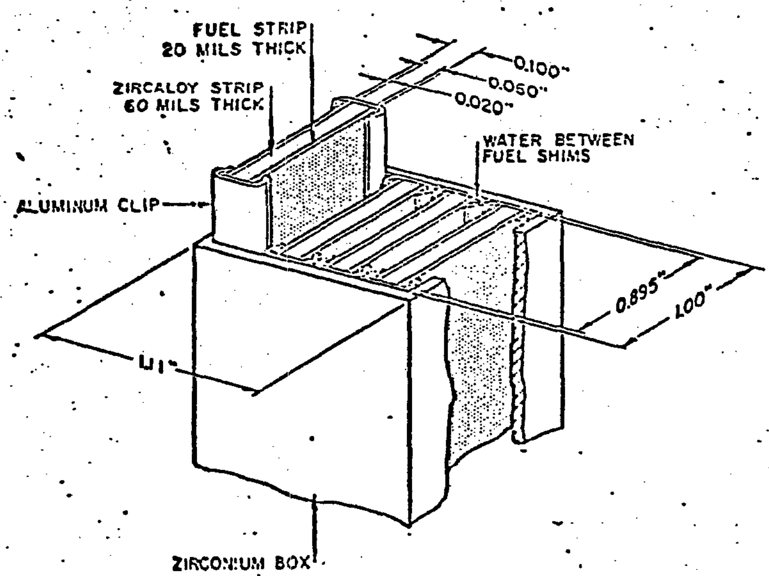


Figure 3.3

Detail of Small Module Box Construction Showing a 4 x 8 Array of Fuel Elements

Table 3.4

Description of Heterogeneous U-Zr-H_2O Critical Experiment Data (24)

Core Number	No. Zr-2 Strips per Module	No. Fuel Shims per Module	Fuel Density Gm U^{235}/liter of core	Metal to Water Vol. Ratio	Core Width, cm	Core Length, cm
1	3	3	58.1	1.17	19.7	56.3
11	4	4	77.5	1.75	19.7	64.8
15	3	5	96.8	1.32	16.9	53.5
24	3	3	52.2	0.95	16.9	73.2
26	3	6	106	1.13	14.1	62.0
30	3	6	116	1.40	16.9	53.5
32	4	8	155	1.38	14.1	76.1
75	3	12	232	1.41	14.1	64.8

Table 3.5

Comparison of Calculated and Measured Values of k_{eff}

Core No.	Computed Self-Shielding Factor for Water*	Measured k_{eff} [24]	Calculated k_{eff}
1	0.885	1.005	1.097
11	0.894	0.9996	1.078
15	0.887	0.9999	1.078
24	0.885	1.003	1.071
26	0.887	1.003	1.055
30	0.887	1.005	1.081
32	0.892	1.003	1.059
75	0.822	1.002	1.061

*Avg. thermal neutron flux in fuel shim/avg. thermal neutron flux in water.

should be negligible because the core height is substantially larger than the core width and length in each experiment. The core width is the dominant factor in B_g^2 for these systems.

3. τ was determined using values of metal to water volume ratio that included the small volume of aluminum in the clips. This Al made up less than 9% of the total metal volume in all cases.

4. The self-shielding factors for the Al clips and Zr module boxes were taken to be equal to that computed for the water in each experiment.

5. The self-shielding factors for the Zircaloy strips were also taken to be equal to those computed for the water in each experiment. These self-shielding factors (reciprocals of thermal neutron flux disadvantage factors) are expected to be too low for these Zircaloy strips, which, in some cases, are sandwiched between fuel shims. This assumption is conservative when computing values of k_{eff} to compare with measured values in that the use of a too low self-shielding factor for structure material leads to a low calculated value of k_{eff}. These check calculations were performed to show that calculated values of k_{eff} are greater than measured values; hence, this assumption is conservative.

6. In all cores except #75, the temperature (given in Ref. 24) is close to 75°F. The water content of these cores was computed using a water density of 0.9969 gm/cm^3, which corresponds to this temperature. For Core #75, a water density of 0.9907 gm/cm^3, which corresponds to the given temperature of 111°F, was used.

Self-shielding factors were computed using the formulas described in Section 2.1. For a single fuel shim, the self-shielding factor (ratio of the average thermal neutron flux in the shim to the flux at the surface of the shim) was computed to be 0.914. For two adjacent fuel shims (Core #75), the self-shielding factor was computed to be 0.869. The self-shielding factors computed for water are given in Table 3.5 along with the calculated values of k_{eff}. These results show that calculated values of k_{eff} are conservative (high) by 5% to 9% even after self-shielding is taken into account.

3.4 Special Thermal, Heterogeneous Uranium-Zirconium Fueled Assemblies

This section consists of internal UNC Memo NDEO-403, which describes a check calculation performed for a special fuel assembly whose reactivity has been estimated by KAPL from critical experiment and measured reactivity coefficient data. This memo is Confidential and has been deleted from unclassified documents in which this reactivity calculation method is described.

3.5 Undermoderated Uranium Metal-Water Systems

Reference 17 (LAMS-2557) presents estimated critical cylinder diameters, estimated critical slab thickness, and estimated critical volumes for undermoderated U metal-water mixtures. The calculation method presented in Section 2.2 has been compared with the dashed lines of Figures 10, 13 and 14 of Reference 17. For a critical system, $k_{eff} = 1$, Eq. 19 yields:

$$B_g^2 = \frac{2.40-1}{\tau} \tag{21}$$

$$B_g^2 = \frac{1.40}{\tau} = \frac{1.40}{28 \text{ cm}^2} = .050 \text{ cm}^{-2} \tag{22}$$

$$B_g = 0.2236 \text{ cm}^{-1} \tag{23}$$

where the value of $\tau = 28 \text{ cm}^2$ was obtained from Section 2.2 for uranium water mixtures with H/U^{235} atom ratios less than 120. Equation 19 is independent of H/U^{235} ratio and enrichment according to the calculation method.

For an infinite cylinder Eq.23 is

$$B_g = 0.2236 \text{ cm}^{-1} = \frac{2.405}{R+\delta} \tag{24}$$

or $$R + \delta = 10.76 \text{ cm}$$

where: R = the initial cylinder radius

For spheres Eq.23 is:

$$Bg = 0.2236\ cm^{-1} = \frac{\pi}{R+\delta} \tag{25}$$

$$R + \delta = 14.05\ cm$$

$$V_{critical} = \frac{4}{3}\ \pi\ R^3$$

For slabs Eq.23 is:

$$Bg = 0.2236\ cm^{-1} = \frac{\pi}{H+2\delta} \tag{26}$$

Using equations 17 and 18 for values of reflector saving in equations 24-26 values of critical dimensions have been calculated for a range of H/U^{235} atom ratios between 0 and 20. The results compared in Table 3.6 to experimental data[17] are conservative for all H/U^{235} atom ratios.

3.6 Calculations of Unreflected and Unmoderated, Highly Enriched Uranium Systems

The calculation method presentation Section 2.3 for bare, unmoderated, highly enriched uranium systems has been tested against DTF[18] calculations of bare U metal spheres having a density of 18.7 gm/cm^3 and an enrichment of 93.2% U^{235}. The results of this comparison, shown in Section 2.3, indicates good agreement with the DTF results. This method can be used to determine k_{eff} for interaction calculations.

Table 3.6

Comparison of Experimental[17] and Calculated Critical Parameters for Undermoderated Uranium Water Mixtures

A. Infinite Cylinders

	Calculated Reflector Savings (δ cm)		Critical Cylinder Radius, cm			
					Experimental Data (LAMS-2537)	
H/U^{235}	Bare	Reflected	Calc. Bare	Calc. Reflected	Bare	Reflected
20	2.50	6.50	8.26	4.26	9.41	6.48
5	4.72	7.10	6.04	3.66	8.76	5.84
1	5.31	7.26	5.45	3.50	8.26	4.76
0	5.46	7.30	5.38	3.46	5.78	3.94

B. Spheres

	Calc. Critical, Radius, cm		Calc. Critical vol., liters		Crit. Vol. from LAMS-2537, liters	
H/U^{235}	Bare	Reflected	Bare	Reflected	Bare	Reflected
20	11.55	7.55	6.45	1.80	10.0	4.7
5	9.33	6.95	3.40	1.41	7.1	3.2
1	8.74	6.79	2.80	1.31	4.4	1.97
0	8.59	6.75	2.65	1.29	2.65	1.30

Table 3.6 (con't)

C. Slabs

H/U^{235}	Calculated H, cm: Bare	Calculated H, cm: Reflected	H from LAMS-2537, cm: Bare	H from LAMS-2537, cm: Reflected
20	9.05	1.05	10.29	3.78
5	4.61	0	8.92	3.02
1	3.43	0	6.93	2.13
0	3.13	0	5.59	1.52

3.7 Calculation of Thermal, Heterogeneous U-Al-H_2O Critical Experiments

1) ORNL 5.4 w/o Uranium-Aluminum Alloy Rods

Reference 31 describes critical experiments performed with cylindrical pieces of 93% U^{235} enriched uranium and aluminum to establish safe practices in fabrication and shipment. The uranium content of the alloy was 5.4 w/o U. The unclad rods were 1.015" in diameter and 12" long.

The number of U-Al rods required for criticality was measured as a function of pitch between rods. The arrays of fuel rods were loaded in both square and rounded outlines (see Table 3.7). However, as noted in Table 3.7, the number of rods required for criticality did not result in perfectly square or round lattices. A second difficulty associated with the analysis of this type of data is assigning a dimension to the fueled portion of the core when calculating the buckling. Generally the parameters entering into k_∞ are evaluated for a unit cell consisting of a fuel rod and its associated moderator. The dimensions of the fueled portion of the core are determined by the volume of a number of unit cells. This procedure tends to overestimate the core size since water near the outer ring of fuel rods can act as both reflector and fuel-rod moderator. This method of homogenization is always conservative and becomes more conservative as the lattice pitch is increased.

Self-shielding factors for the U-Al rods were computed using the method described in Section 2.1. The calculated disadvantage factors and eigenvalues are shown in Table 3.7. The results of this analysis are conservative for all cases studied.

2) ORNL SPERT-D Fuel Elements in Water

Recently, an experimental study has been made at ORNL of the critical numbers of SPERT-D Fuel Elements in water having various water gaps between elements[32]. The experiments were performed in order to extend basic criticality data in this area. Critical parameters were obtained for uniformly spaced rectangular arrays as a function of water-gap width and on a series of three- and four-slab arrays in which the water gaps between each varied. Those arrays with no separation between slabs were analyzed using the calculation method presented in Section 2.1.

Each 3" square fuel assembly contained 306.5 gms U^{235} as highly enriched U in aluminum plates clad with 0.020" of aluminum. Two configurations were analyzed corresponding to 24" and 72" long fuel assemblies. The calculated k_∞ for these critical configurations were 1.0212 and 1.0267 respectively.

Table 3.7

Critical Parameters of Water-Moderated and Reflected 5.4 w/o U-Al Slugs[31]

Spacing Center-to-Center (in.)	Array Outline	Minimum Critical Number of Rods	Calculated Disadvantage Factor	k_{eff}
1.25	Square	170	1.2618	1.007
	Rounded	163	1.2618	1.020
1.37	Square	146	1.3007	1.042
	Rounded	140	1.3007	1.074
1.50	Square	143	1.3419	1.066
	Rounded	138	1.3419	1.092
1.63	Square	164	1.3819	1.059
1.75	Square	228	1.4182	1.087

4.0 CALCULATED CRITICALITY LIMITS FOR THERMAL, U-Zr ALLOY-H_2O CYLINDERS, SLABS AND SPHERES

The calculation method described in Section 2.1 was used to compute criticality limits for thermal U-Zr alloy-H_2O systems. A similar procedure can be used to determine critical parameters for U-aluminum and U-stainless steel mixtures. A range of uranium weight fraction in the alloy up to 0.25 gm U^{235} per gm alloy was considered. At any particular weight fraction of uranium, the quantities of primary interest in establishing the nuclear safety of various operations involving this fuel are the values of the critical parameters (e.g., critical mass, critical volume, critical cylinder diameter, etc.) under conditions of optimum moderation and full reflection by water. Under these conditions, the critical system dimensions reach their minimum values. If safe-system dimensions are selected to be less than these minimum critical dimensions, then the safety of the systems is established for all moderator-to-fuel ratios and degrees of reflection by water for the particular fuel alloy composition.

4.1 Assumptions

The following conservative assumptions were made in these calculations:

1. All systems were assumed to be homogeneous. Self-shielding was neglected.

2. Uranium enrichment was assumed to be 100% U^{235}.

4.2 Calculations and Results

For each of several values of weight % U^{235} in metal, critical infinite cylinder diameter, critical volume, etc., were computed as a function of moderator to fuel ratio for fully reflected systems. The minimum critical dimension, which corresponds to optimum moderation, was determined. Plots were made of minimum critical dimensions as a function of gm U^{235}/gm (U^{235} + Zr), are shown as follows:

1. Minimum critical diameter and safe diameters for infinite cylinders (Fig. 4.1).

2. Minimum critical cross sectional areas and safe cross sectional areas for infinite cylinders (Fig. 4.2).

3. Minimum critical heights and safe heights for infinite slabs (Fig. 4.3).

4. Minimum critical volumes and safe volumes (spherical geometry)(Fig. 4.4).

In order to establish the safety of arrays of units containing fuel, the interaction criteria of TID-7016, Rev. 1[25], are frequently used. In order to apply these criteria, the k_{eff} for each unit under unreflected conditions must be known. Maximum k_{eff} was computed for unreflected safe cylinders and spheres (safe under conditions of full reflection). These values correspond to optimum moderation for the bare condition, which in general occurs at a different water-to-fuel ratio than the optimum moderation for a similar reflected system. Maximum k_{eff} is shown as a function of gm U^{235}/gm (U^{235} + Zr) in Figure 4.5.

Detailed sample calculations of minimum critical dimensions under reflected conditions and maximum k_{eff} for unreflected safe systems are shown in Section 4.5.

4.3 Safety Factors

The following safety factors or relationships between minimum critical dimensions and safe dimensions were used. These safety factors are consistent with those used in TID-7016, Rev. 1[25].

1. For infinite cylinders,

$$\frac{\text{safe diameter}}{\text{min. critical diameter}} = 0.88 \qquad (27)$$

$$\frac{\text{safe area}}{\text{min. critical area}} = (0.88)^2 = 0.7744$$

2. For infinite slabs,

$$\frac{\text{safe height}}{\text{min. critical height}} = 0.84 \qquad (28)$$

3. For spherical volumes,

$$\frac{\text{safe volume}}{\text{min. critical volume}} = 1/1.3 = 0.769 \qquad (29)$$

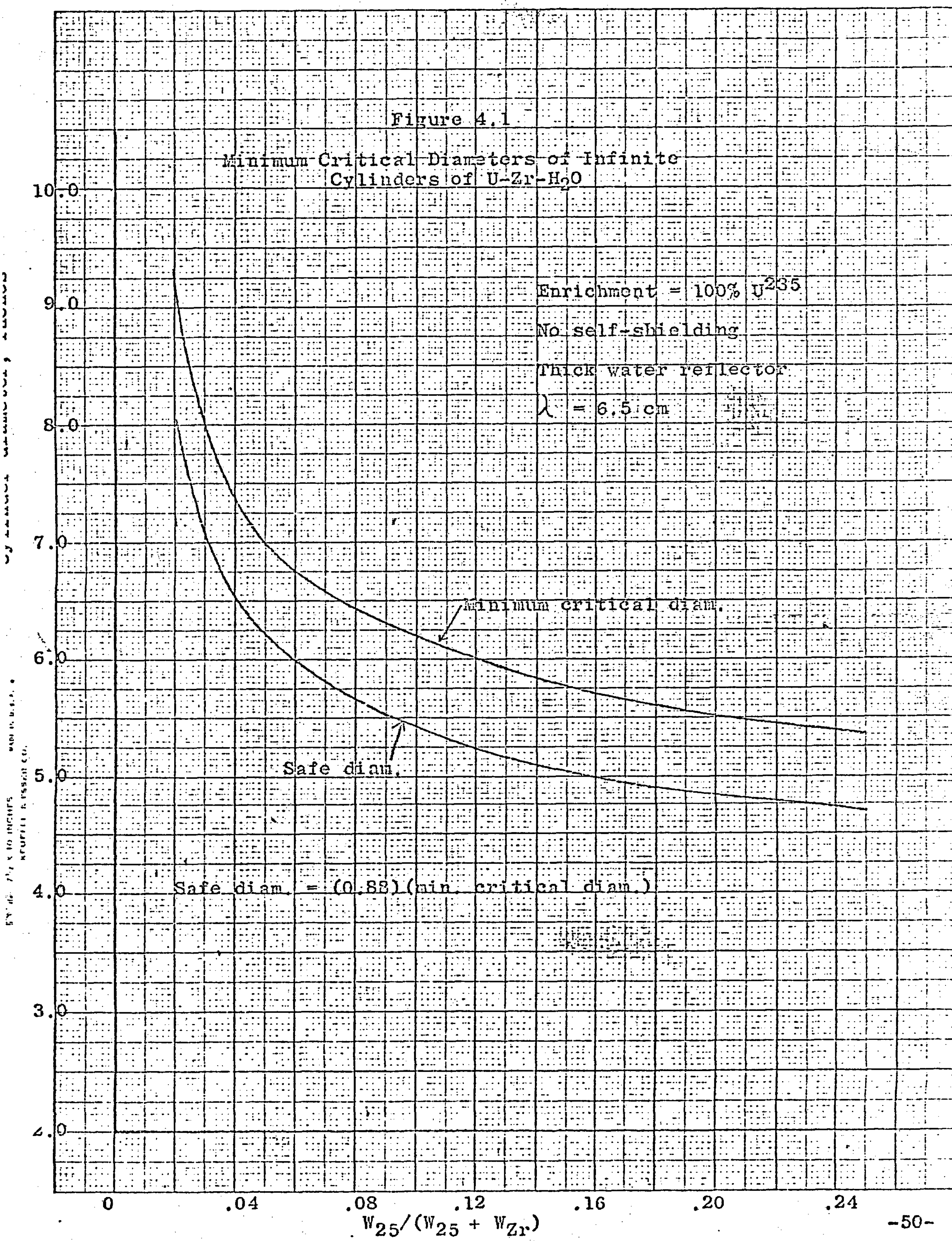


Figure 4.1
Minimum Critical Diameters of Infinite Cylinders of U-Zr-H_2O
Enrichment = 100% U^{235}
No self-shielding
Thick water reflector
λ = 6.5 cm
Minimum critical diam.
Safe diam.
Safe diam. = (0.88)(min. critical diam.)
10.0
9.0
8.0
7.0
6.0
5.0
4.0
3.0
2.0
0
.04
.08
.12
.16
.20
.24
$W_{25}/(W_{25} + W_{Zr})$

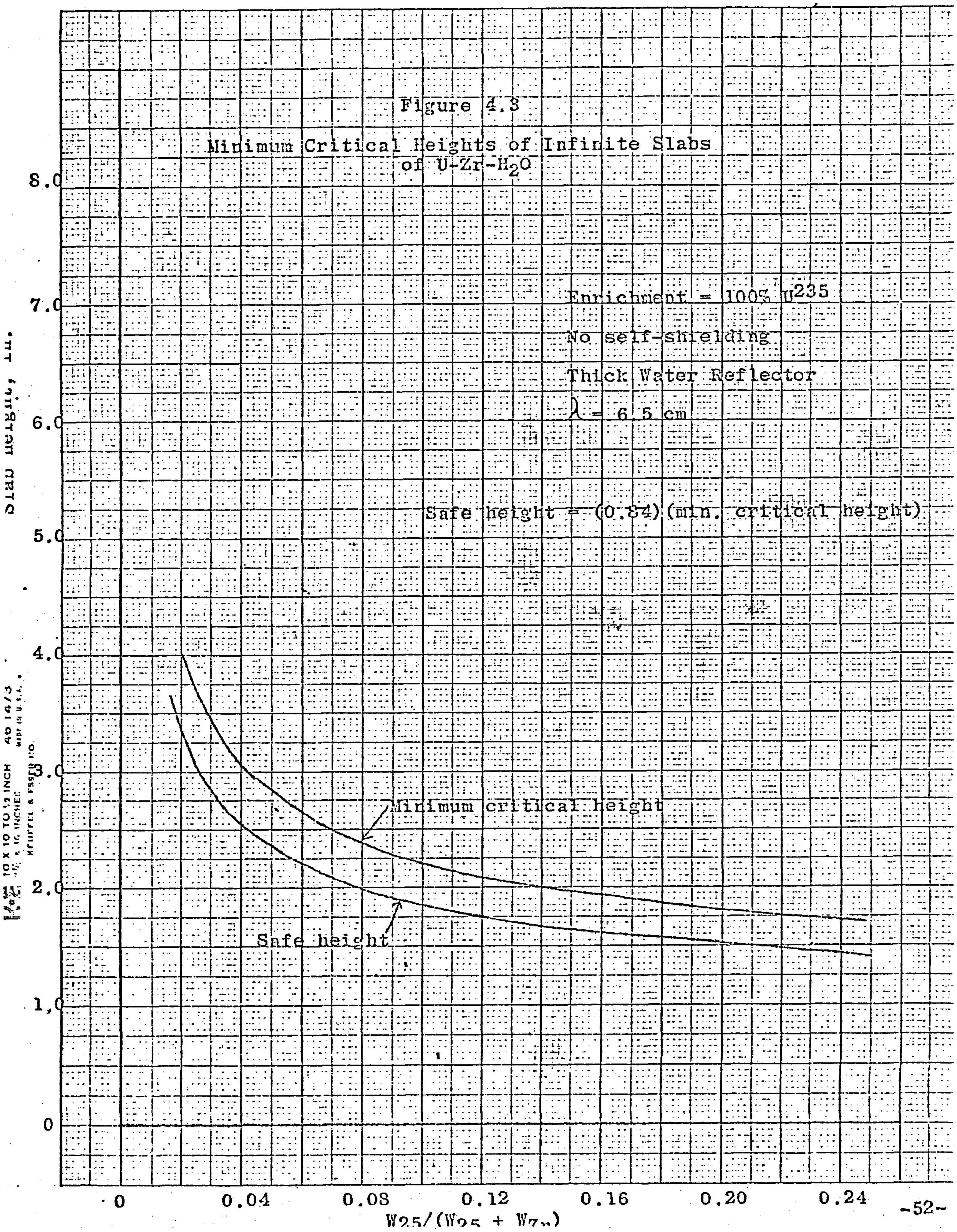


Figure 4.3
Minimum Critical Heights of Infinite Slabs of U-Zr-H_2O
Enrichment = 100% U^{235}
No self-shielding
Thick Water Reflector
λ = 6.5 cm
Safe height = (0.84)(min. critical height)
Minimum critical height
Safe height
Slab height, in.
8.0
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0
0
0.04
0.08
0.12
0.16
0.20
0.24
$W_{25}/(W_{25} + W_{Zr})$
10 X 10 TO 1/2 INCH 46 1473
KEUFFEL & ESSER CO.
MADE IN U.S.A.

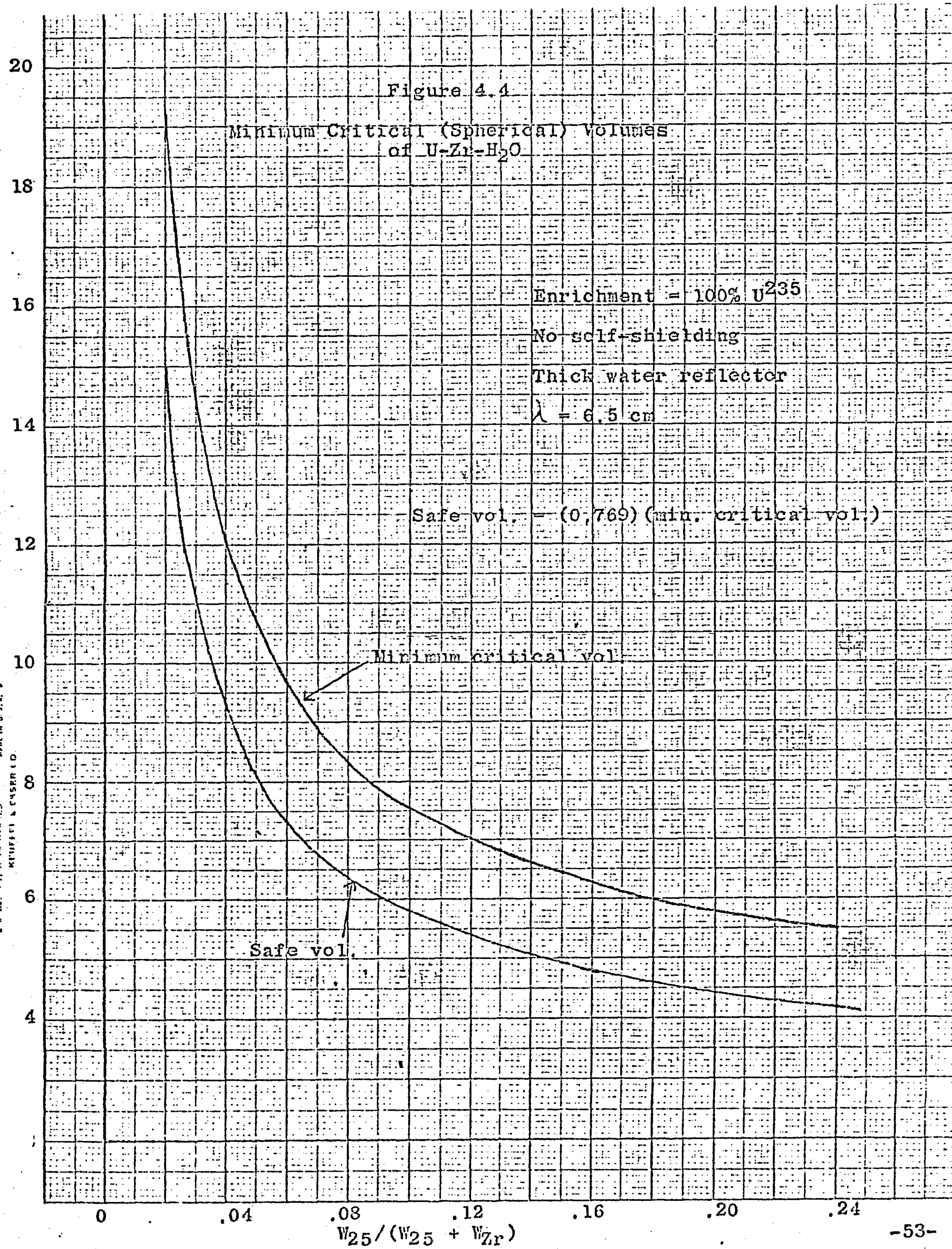


Figure 4.4
Minimum Critical (Spherical) Volumes
of U-Zr-H_2O
Enrichment = 100% U^{235}
No self-shielding
Thick water reflector
λ = 6.5 cm
Safe vol. = (0.769)(min. critical vol.)
Minimum critical vol.
Safe vol.
20
18
16
14
12
10
8
6
4
0
.04
.08
.12
.16
.20
.24
$W_{25}/(W_{25} + W_{Zr})$

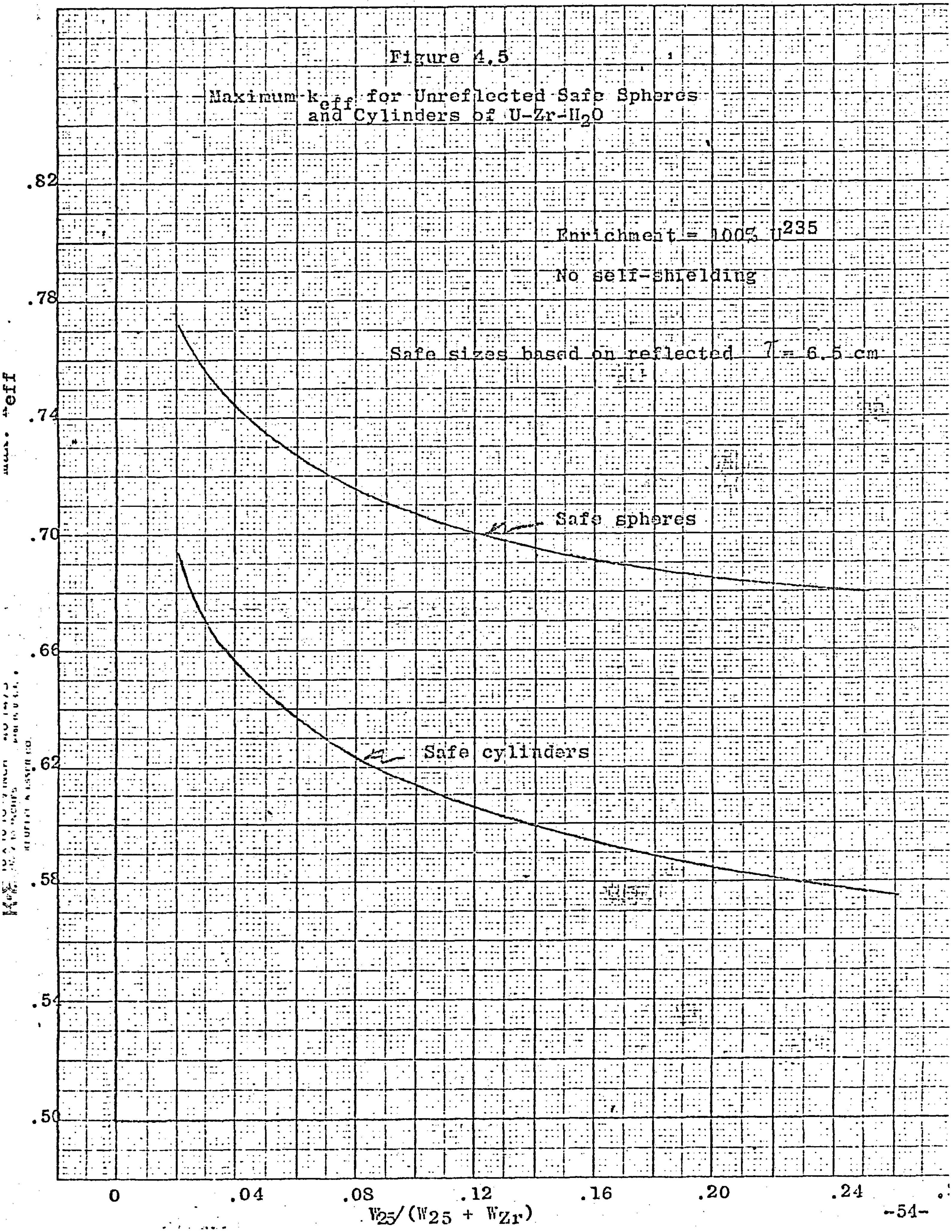


Figure 4.5
Maximum k_{eff} for Unreflected Safe Spheres and Cylinders of U-Zr-H_2O
Enrichment = 100% U^{235}
No self-shielding
Safe sizes based on reflected τ = 6.5 cm
Safe spheres
Safe cylinders
Max. k_{eff}
.82
.78
.74
.70
.66
.62
.58
.54
.50
0
.04
.08
.12
.16
.20
.24
$W_{25}/(W_{25} + W_{Zr})$

4.4 Use of the Curves

Fuel fabrication operations at New Haven involve the handling of U-Zr alloys clad with Zr as well as unclad alloy material. It would be desirable to be able to utilize the attached curves for both materials and take credit for the fact that the presence of cladding reduced reactivity and permits increases in the dimensions of safe systems. The calculations have been performed in such a way as to permit this.

The presence of Zr cladding affects two parameters, thermal utilization and τ. (There is also a small effect on L^2 but this is negligible in terms of effect on reactivity). Since self-shielding has been neglected in these calculations, thermal utilization depends only on the total quantity of Zr present (i.e., on W_{Zr}/W_{25}) and not on the distribution of the Zr between alloy meat and cladding. However, since the density of U-Zr alloys is different from that of Zr, the total metal-to-water volume ratio, V_{metal}/V_{H_2O}, does depend on this distribution and therefore τ depends to some extent on this distribution.

τ increases linearly with increasing V_{metal}/V_{H_2O}. For a given total Zr to U^{235} weight ratio, the Zr may be distributed between cladding and fuel alloy in many ways. However, it is shown in Section 4.6 that, for a particular value of W_{Zr}/W_{25} or weight fraction U^{255}, e.g., $\left[W_{25}/(W_{25} + W_{Zr})\right]$ total, V_{metal}/V_{H_2O} is smallest when all Zr is in the form of cladding. The corresponding formula for V_{metal}/V_{H_2O} is:

$$\left[\frac{V_{metal}}{V_{H_2O}}\right] \text{min.} = \frac{(W_{Zr}/W_{25})\,\text{total}}{\rho_{Zr}\;(W_{H_2O}/W_{25})} \qquad (30)$$

where ρ_{Zr} = density of Zr, 6.5 gm/cm^3

W_{Zr} = weight of Zr

W_{25} = weight of U^{235}

W_{H_2O} = weight of water

This formula was used in the present calculations. For a given value of $W_{25}/(W_{25} + W_{Zr})$, the curves would apply rigorously if none of the Zr is alloyed with uranium. When some or all of the Zr is present as U-Zr alloy, critical parameters obtained from the curves at the particular value of $\left[W_{25}/(W_{25} + W_{Zr})\right]$total (which includes the weights of both alloyed and unalloyed Zr) are slightly conservative because the value of V_{metal}/V_{H_2O} used to obtain τ was slightly smaller than the actual value. This resulted in the use of a conservative (low) value of τ which leads to conservative values of reactivity and critical dimensions.

4.5 Sample Calculation of Minimum Critical Dimensions

As an illustrative example of the calculation method used to obtain the curves presented in Section 4.4, the minimum critical dimensions for 7 wt. % U in U-Zr alloy are calculated. A uranium enrichment of 100% U^{235} is assumed and all self-shielding factors are taken to be 1.0. For 0.07 gm, U^{235}/gm (U^{235} + Zr),

$$\frac{W_{Zr}}{W_{25}} = \frac{1-0.07}{0.07} = 13.29$$

In order to determine the optimum water-to-fuel ratio for a particular alloy, k_{eff} is determined for various water fractions and the results plotted to determine the maximum k_{eff}. Therefore, assume

$$\frac{W_{H_2O}}{W_{25}} = 6.0$$

$$f^{-1} = 1 + 0.00073 \quad (13.29) + 0.01316 \ (6.0)$$

$$f^{-1} = 1.0887$$

$$f \quad = 0.919$$

$$L^2 = 8(1-0.919) = 0.65 \text{ cm}^2$$

τ is shown as a function of V_{Zr}/V_{H_2O} in Figure 2.2 of Section 2.1. For mixtures of U-Zr alloy and Zr metal, V_{Zr}/V_{H_2O} is taken to be V_{metal}/V_{H_2O}. The following expression was used to compute V_{metal}/V_{H_2O}.

$$\frac{V_{metal}}{V_{H_2O}} = \frac{1}{\rho_{Zr}} \frac{(W_{Zr}/W_{25})}{(W_{H_2O}/W_{25})}$$

where $\rho_{Zr} = 6.5$ gm/cm^3

It is shown in Section 4.6 that this expression leads to the lowest value of V_{metal}/V_{H_2O} for a given value of W_{Zr}/W_{25}, regardless of how the Zr is distributed between alloy meat and cladding. Since τ increases linearly with increasing V_{metal}/V_{H_2O}, this expression leads to the lowest value of τ for the particular W_{Zr}/W_{25}, and hence leads to conservative values of reactivity and critical dimensions.

$$\frac{V_{metal}}{V_{H_2O}} = \frac{1}{6.5} \frac{(13.29)}{(6.0)} = 0.341$$

$\tau = 33.5 \text{ cm}^2$ (from Fig. 2.2)

$M^2 = 33.5 + 0.65 = 34.15 \text{ cm}^2$

At criticality, $k_{eff} = 1$ and $Bm^2 = Bg^2$

$$Bm^2 = Bg^2 = \frac{2.07(0.919) - 1}{34.15} = .02643 \text{ cm}^{-2}$$

These calculations are repeated for several other values of W_{H_2O}/W_{25}, and Bm^2 is plotted as a function of W_{H_2O}/W_{25}. This plot is shown in Figure 4.6. Minimum critical dimensions correspond to maximum Bm^2, which is 0.02647 cm^{-2} from Figure 4.6.

$$Bg = Bm = \sqrt{0.02646} = 0.1627 \text{ cm}^{-1}$$

For an effectively infinite water reflector, the reflector savings is taken to be 6.5 cm. The minimum critical cylinder radius is (for an infinitely long cylinder):

$$R = \frac{2.405}{0.1627} - 6.5 = 8.28 \text{ cm}$$

Min. critical diameter = 6.52 inches

Min. critical cross sectional area = $\frac{\pi}{4}(6.52)^2 = 33.4 \text{ in.}^2$

The minimum critical slab height (for an infinite slab) is:

$$H = \frac{\pi}{0.1627} - 2(6.5) = 6.31 \text{ cm} = 2.48 \text{ inches}$$

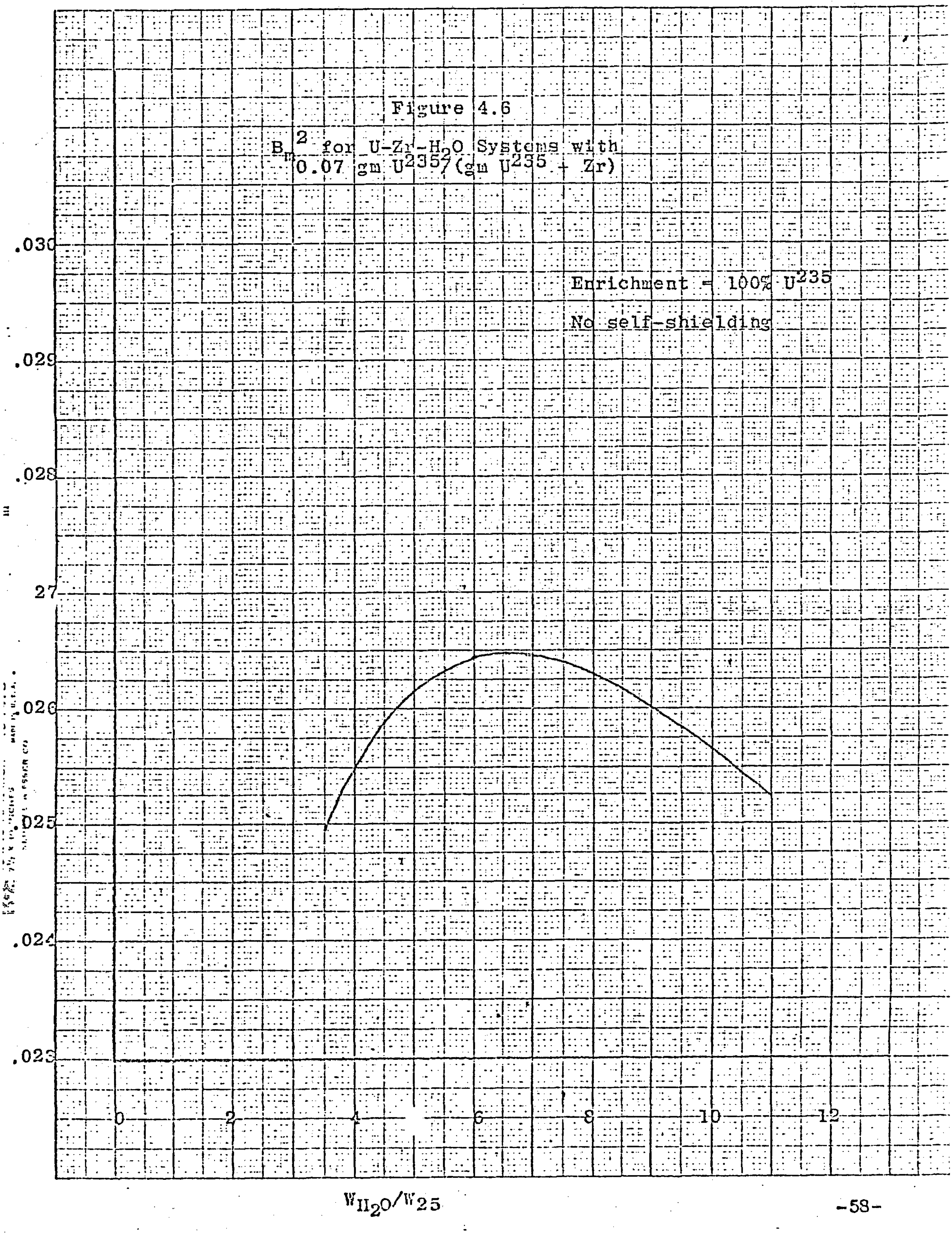


Figure 4.6
B_m^2 for U-Zr-H_2O Systems with 0.07 gm U^{235}/(gm U^{235} + Zr)
Enrichment = 100% U^{235}
No self-shielding
.030
.029
.028
27
.026
.025
.024
.023
0
2
4
6
8
10
12
W_{H_2O}/W_{25}

The minimum critical volume is obtained with a spherical configuration. The radius of the minimum critical sphere is:

$$R = \frac{\pi}{0.1627} - 6.5 = 12.81 \text{ cm}$$

$$\text{Vol.} = \frac{4}{3}\pi (12.81)^3 \, 10^{-3} \frac{\text{liters}}{\text{cm}^3} = 8.81 \text{ liters}$$

This entire procedure was repeated for several other values of weight fraction U^{235} in the metal. The minimum dimensions obtained for each value are plotted in the figures presented in Section 4.4.

Since the optimum water-to-fuel volume ratio is generally different for reflected and unreflected systems, the following procedure was used to compute the maximum k_{eff} for unreflected safe cylinders and spheres. Maximum k_{eff} (bare) is needed when the interaction criteria of TID-7016, Rev. 1[25] are applied. For 0.07 gm U^{235} per gm (U^{235} + Zr),

Safe cyclinder diameter = 0.88 (min. critical diameter)

R for safe cyclinder = 0.88 (8.28 cm) = 7.285 cm

$$B_g^2 \text{ for unreflected safe cylinder} = \left[\frac{2.405}{7.285 + 2.5}\right]^2 =$$

$$0.06041 \text{ cm}^{-2}$$

$$\text{Since } \frac{W_{H_2O}}{W_{25}} = 6.0$$

$$k_{eff} = \frac{2.07\ (0.919)}{1 + 34.15\ (0.06041)} = 0.621$$

This procedure is repeated for several other values of W_{H_2O}/W_{25} and k_{eff} is plotted as a function of W_{H_2O}/W_{25}. This plot was shown in Figure 4.7. The maximum value of k_{eff} is 0.626.

This procedure was repeated for several other values of wt. % U^{235} in metal, and the maximum values of k_{eff} are shown in Figure 4.5. An identical procedure was followed to obtain the maximum k_{eff} values for safe-volume spheres that are shown in Figure 4.5.

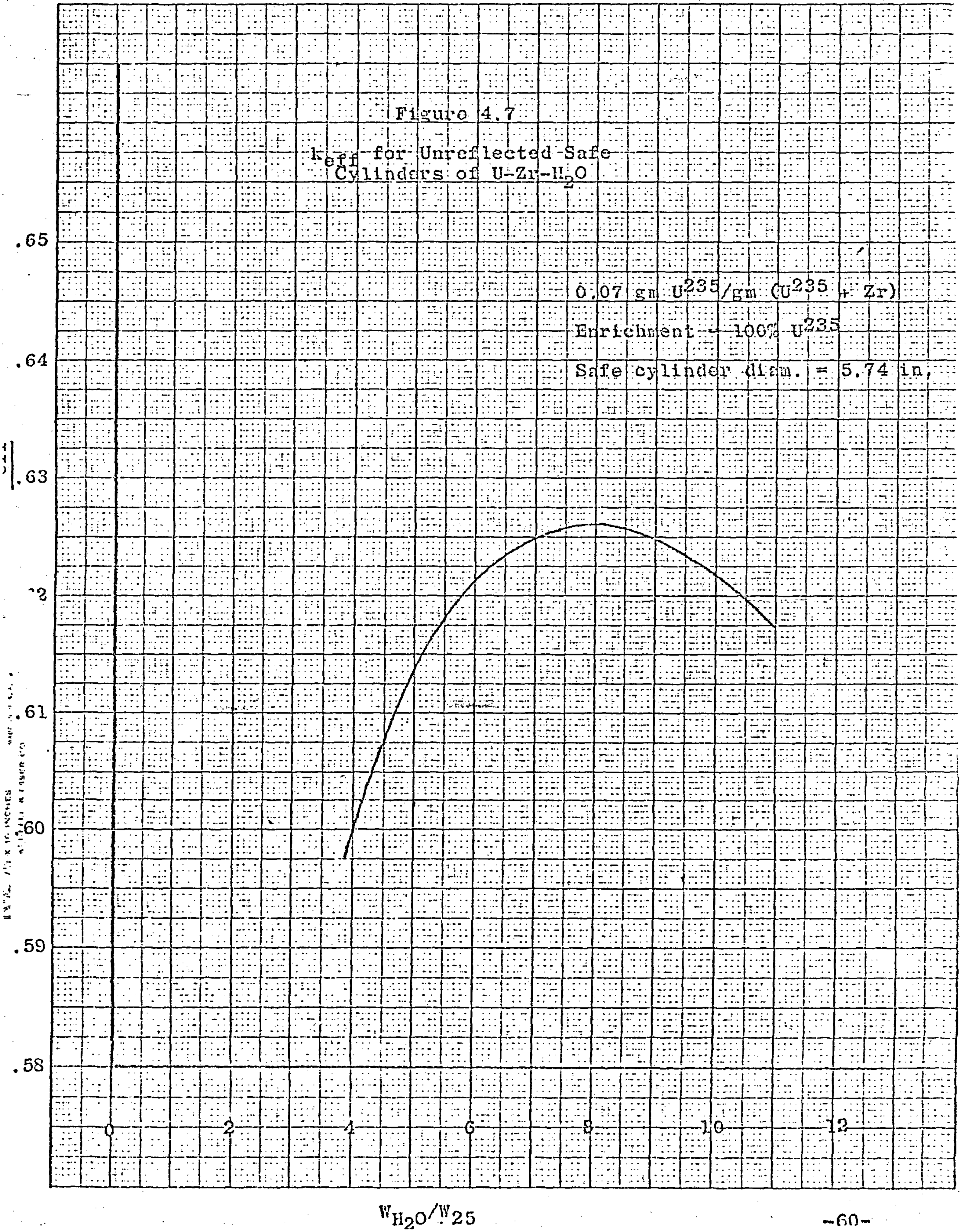


Figure 4.7
k_{eff} for Unreflected Safe Cylinders of U-Zr-H_2O
0.07 gm U^{235}/gm (U^{235} + Zr)
Enrichment - 100% U^{235}
Safe cylinder diam. = 5.74 in.
.65
.64
.63
.62
.61
.60
.59
.58
0
2
4
6
8
10
12
W_{H_2O}/W_{25}

4.6 Derivation of an Expression for the Minimum Metal to Water Volume Ratio for Clad Fuel Elements

The criticality limit curves of section have been prepared as a function only of the weight fraction of U^{235} in zirconium alloy. This data has been prepared to be used for both clad and unclad fuel elements. The basis for this approach is derived by determining the most conservative distribution of zirconium in the Zr-alloy.

The purpose of Zr cladding affects two parameters, thermal utilization and τ . (There is also a small effect on L^2 but this is neglected). Since self-shielding is neglected in the derivation of the U-Zr alloy criticality limits, the thermal utilization depends only on the total Zr present (i.e., on W_{Zr}/W_{25}) and not on the distribution of the Zr between alloy meat and cladding. However, since the density of U-Zr alloys is different from that of pure Zr, the total metal-to-water volume ratio, V_{metal}/V_{H_2O}, does depend on this distribution and therefore τ depends to some extent on the distribution of the Zr in the fuel alloy mixture.

From Eq.15 in Section 2.1:

$$\frac{V_{metal}}{V_{H2O}} = \frac{1}{\left(\frac{W_{H2O}}{W_{25}}\right)} \left[\frac{1}{(\rho alloy)(1 - Z)} \left(\frac{W_{Zr}}{W_{25}}\right)_{alloy} + \left(\frac{1}{\rho Zr}\right)\left(\frac{W_{Zr}}{W_{25}}\right)_{clad} \right] \quad (31)$$

where this expression was derived for a system containing both U-Zr alloy and non-fuel-bearing Zr in water.

$$\text{Let } \left(\frac{W_{Zr}}{W_{25}}\right)_{tot} = \left(\frac{W_{Zr}}{W_{25}}\right)_{alloy} + \left(\frac{W_{Zr}}{W_{25}}\right)_{clad} \quad (32)$$

For a given value of $(W_{Zr}/W_{25})_{tot}$, V_{metal}/V_{H_2O} will depend on the distribution of Zr between U-Zr alloy and clad material. It is desired to find an expression that gives the lowest value of V_{metal}/V_{H_2O} for a particular $(W_{Zr}/W_{25})_{tot}$.

$$\text{Let } R = \frac{(W_{Zr}/W_{25})_{clad}}{(W_{Zr}/W_{25})_{tot}} \tag{33}$$

$$1-R = \frac{(W_{Zr}/W_{25})_{alloy}}{(W_{Zr}/W_{25})_{tot}} \tag{34}$$

$$\frac{V_{metal}}{V_{H_2O}} = \frac{1}{\left(\dfrac{W_{H_2O}}{W_{25}}\right)} \left[\frac{1}{\rho_{alloy}\ (1-Z)} (1-R)\left(\frac{W_{Zr}}{W_{25}}\right)_{tot} + \frac{R}{Zr}\left(\frac{W_{Zr}}{W_{25}}\right)_{tot} \right] \quad ($$

$$\frac{V_{metal}}{V_{H_2O}} = \frac{(W_{Zr}/W_{25})_{tot}}{(W_{H_2O}/W_{25})} \left[\frac{1}{(\rho_{alloy})\,(1-Z)} - R\left\{ \frac{1}{alloy\ (1-Z)} - \frac{1}{\rho_{Zr}} \right\} \right] \quad (3$$

From Figure 2.6 of Section 2.1, the following densities are obtained:

Z gm U/gm alloy	ρ_{alloy}, gm/cm^3	ρ_{alloy} (1-Z)
0	6.50	6.50 $= \rho_{Zr}$
0.07	6.86	6.38
0.14	7.20	6.19
0.21	7.43	5.87
0.28	7.65	5.51

These data show that $1/[\rho_{alloy}\ (1-Z)]$ will always be greater than or equal to $1/\rho_{Zr}$. Therefore, in Eq. 36, the term by which R is multiplied is always ≥ 0. Examination of Eq.36 shows that, for a particular value of $(W_{Zr}/W_{25})_{tot}$, V_{metal}/V_{H_2O} is smallest when R = 1.0.

$$\left(\frac{V_{metal}}{V_{H_2O}}\right)_{min.} = \frac{(W_{Zr}/W_{25})_{tot}}{\rho_{Zr}(W_{H_2O}/W_{25})} \qquad (37)$$

For a particular value of $(W_{Zr}/W_{25})_{tot}$, Eq. 37 yields the smallest value of V_{metal}/V_{H_2O}. Equation 37 corresponds to the situation where all Zr is present as cladding. When some of the Zr is present as alloy, the actual V_{metal}/V_{H_2O} will be larger than the predicted by Eq. 37 .

Equation 37leads to the minimum value of τ for the particular W_{H_2O}/W_{25} because τ decreases linearly with decreasing V_{metal}/V_{H_2O}. Low values of τ are conservative in that they lead to high values of reactivity. The use of $(V_{metal}/V_{H_2O})_{min.}$ as computed from Eq.37 is therefore conservative for U-Zr alloy or mixtures of U-Zr alloy and Zr metal in water.

5.0 CALCULATED CRITICALITY LIMITS FOR THERMAL, LOW ENRICHMENT UO_2 FUEL RODS IN WATER

The calculational method presented in Section 2.1 for thermal systems is applicable to low enrichment UO_2 fuel rods as well. However, the check calculations of these systems (see Section 3.1) indicate that the method is overly conservative when applied to systems with H/U^{235} ratios less than 600. This arises because the calculation method of Section 2.1 neglects the decrease in k_{eff} caused by U^{238} resonance capture. To supplement the calculation method for UO_2 fuel rods with U^{235} enrichment less than 5% the data reported in Ref. 1 is recommended.

The method of analysis used in Ref. 1 was as follows: the material buckling was determined by an asymptotic multigroup transport code calculation and the extrapolation distances by a two-group diffusion theory code from parameters generated in the multigroup calculation.

The material buckling code calculation used twelve energy groups. The top ten groups correspond to Loewensteins' and Okrents'(27) top ten groups. Cross sections for the eleventh (resonance) group extending from 9.1 kev down to 0.625 were derived from a zero dimensional multigroup calculation in which a lethargy width of 0.1 was used and in which allowance was made for anisotropic scattering. The cross sections in the thermal group are taken from Amsters compendium(28).

Hellstrands(29) resonance integrals for UO_2 were used in the resonance group, together with effective surface-to-volume ratios calculated on the assumptions of black fuel and uniform source distributions within the moderator. In the thermal group a P-3 calculation was used to provide disadvantage factors.

The method of analysis used for criticality determination of the slightly enriched UO_2 rods in water was checked against some 52 critical experiment(30) involving enrichments between 1.3 and 5 w/o U^{235} and water to uranium volume ratios between 2.9 and 10. In all cases the calculated eigenvalues agreed with unity to within $\pm .7\%$ $\Delta k/k$. In general the predominant trend is for the model to underpredict k_{eff} very slightly.

Because the calculation method presented in Ref. 1 was found to underpredict k_{eff} for several systems, a conservative approach has been taken to use the "safe" tabulated values of Ref. 1 rather than the critical values. The "safe" values include a safety factor of 0.02 Δk.

Criticality limits for UO_2 rods were obtained from the data of Ref. 1 for optimum moderation and no cladding on the fuel rod. These were obtained for each w/o U^{235} in UO_2 by

Table 5.1

Nuclear Criticality Safety Standards[1]

Low Enrichment Uranium Oxide Rods in Water (UO^2 Density 10.9 g/cc)

Enrichment	Critical Diameter	Safe Diameter	Critical Mass	Safe Mass	Critical Thickness	Safe Thickness	Critical Volume	Safe Volume
1.08	63.6 cm	22.01 in.	13.9 kg U^{235}	6.05 kg U^{235}	36.9 cm	12.77 in.	348 ℓ	264.48 ℓ
1.5	43.4 cm	15.02 in.	5.36 kg U^{235}	2.33 kg U^{235}	23.6 cm	8.17 in.	118 ℓ	89.68 ℓ
2.0	34.0 cm	11.76 in.	3.37 kg U^{235}	1.47 kg U^{235}	17.4 cm	6.02 in.	60.3 ℓ	45.83 ℓ
3.0	27.4 cm	9.48 in.	2.12 kg U^{235}	0.92 kg U^{235}	13.0 cm	4.50 in.	33.4 ℓ	25.38 ℓ
5.0	22.3 cm	7.72 in.	1.36 kg U^{235}	0.59 kg U^{235}	9.79 cm	3.39 in.	19.5 ℓ	14.82 ℓ

Safe diameter = critical diameter (cm) ÷ 2.54 (cm/in.) x 0.88 SF = critical diameter x 0.346

Safe mass = critical mass (kg U^{235}) x 1/2.3 SF = critical mass x 0.435

Safe thickness = critical thickness (cm) ÷ 2.54 (cm/in.) x 0.88 SF = critical thickness x 0.346

Safe volume = critical volume (liters) x 0.76 SF computing the infinite cylinder diameter, critical volume, etc., as a function of moderator to fuel volume and the fuel rod diameter. The minimum critical parameters obtained for these systems are shown in Table 5.1. Plots of these are presented as follows:

1. Minimum safe diameters for infinite cylinders (Figure 5.1).
2. Minimum safe thickness for infinite slabs (Figure 5.2).
3. Safe masses and volumes for spheres (Figure 5.3, 5.4).

In order to establish the safety of arrays of units containing UO_2 fuel rods in water, the interaction criteria of Ref. 23 should be used. In order to apply these criteria, the k_{eff} for each unit under unreflected conditions must be known. This may be obtained from the data tabulated in Appendix C of Ref. 1 by determining k_{∞} for the reflected system and calculating the non-leakage probability,

$$P_{non-leakage} = \frac{1}{1 + B^2M^2}$$

on the basis of a bare system (reflector savings = 2.5 cm).

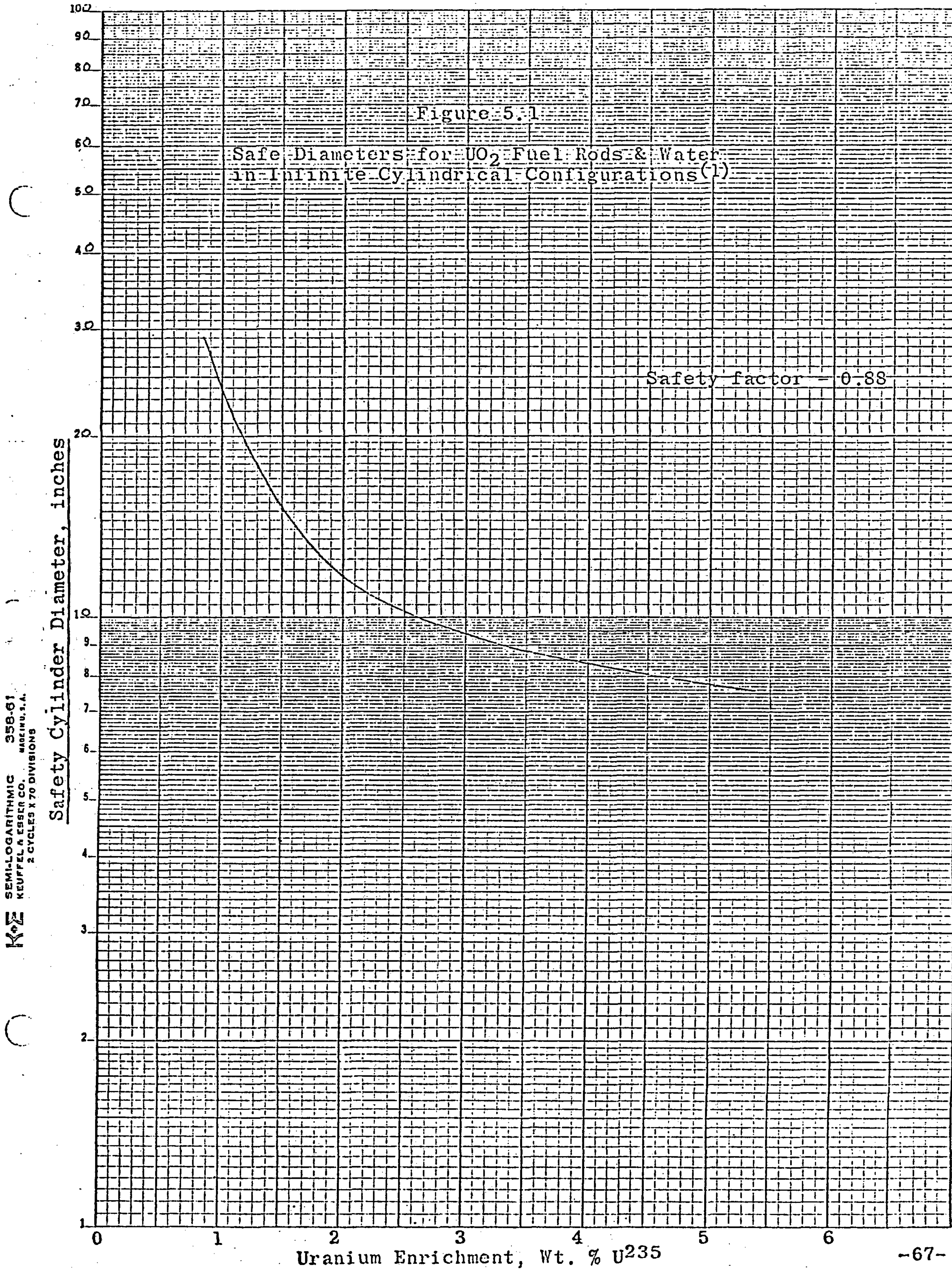


Figure 5.1
Safe Diameters for UO_2 Fuel Rods & Water in Infinite Cylindrical Configurations(1)
Safety factor = 0.88
Safety Cylinder Diameter, inches
Uranium Enrichment, Wt. % U^{235}
100
90
80
70
60
50
40
30
20
10
9
8
7
6
5
4
3
2
1
0
1
2
3
4
5
6
SEMI-LOGARITHMIC 358-61
KEUFFEL & ESSER CO. MADE IN U.S.A.
2 CYCLES X 70 DIVISIONS

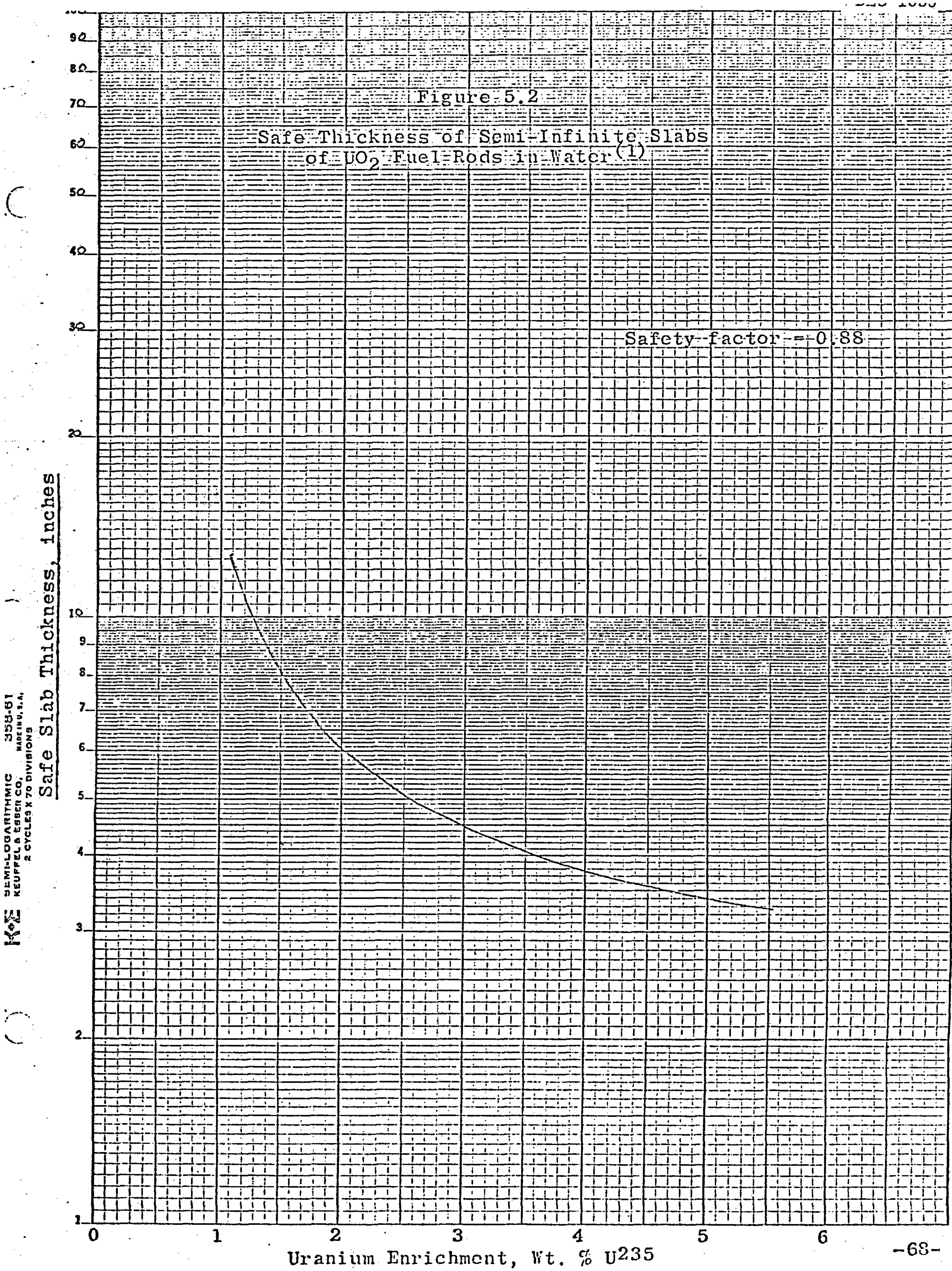


Figure 5.2
Safe Thickness of Semi-Infinite Slabs
of UO_2 Fuel Rods in Water(1)
Safety factor = 0.88
Safe Slab Thickness, inches
Uranium Enrichment, Wt. % U^{235}
90
80
70
60
50
40
30
20
10
9
8
7
6
5
4
3
2
1
0
1
2
3
4
5
6
K+E SEMI-LOGARITHMIC 358-61
KEUFFEL & ESSER CO. MADE IN U.S.A.
2 CYCLES X 70 DIVISIONS

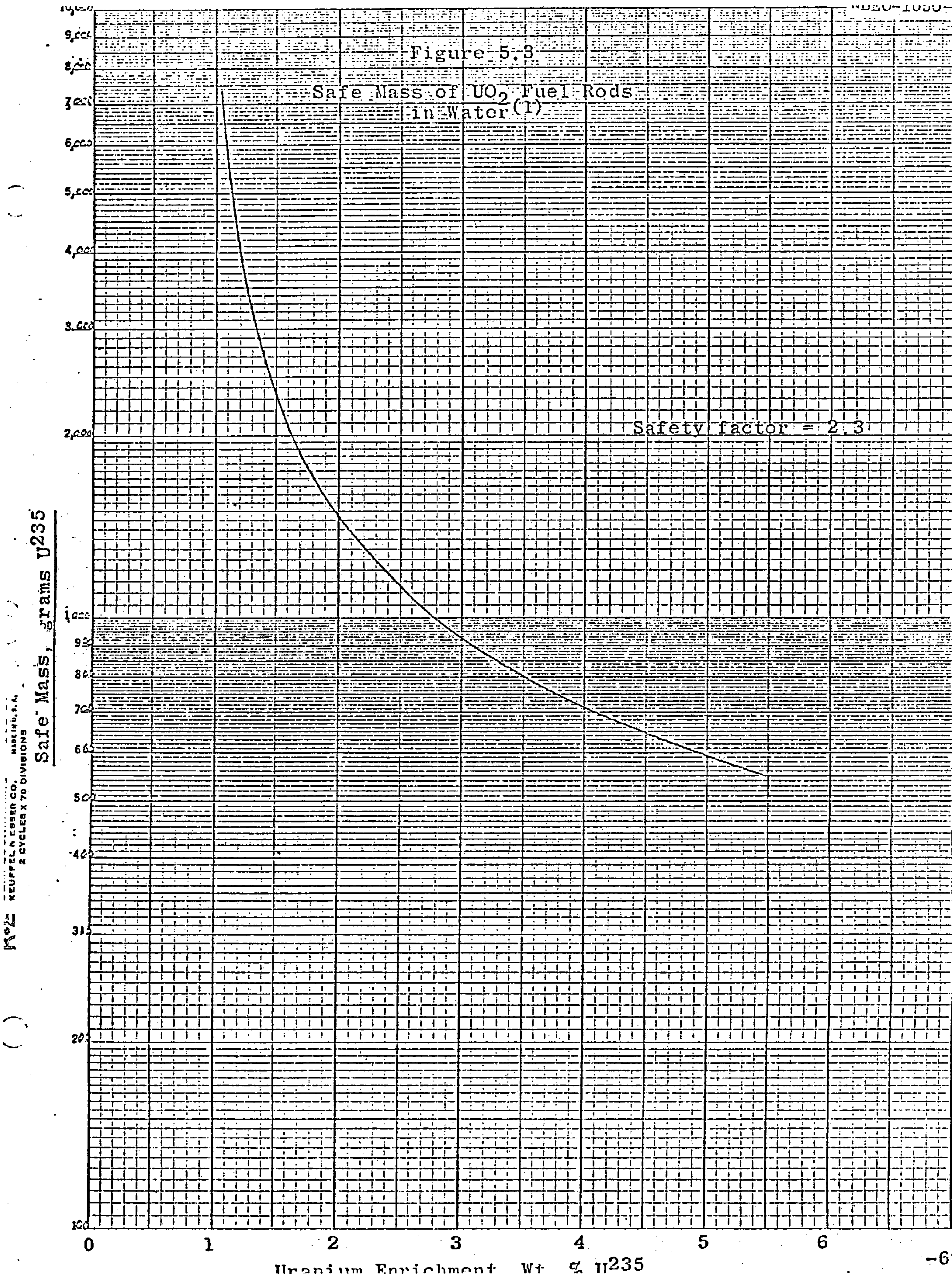


Figure 5.3
Safe Mass of UO_2 Fuel Rods in Water(1)
Safety factor = 2.3
Safe Mass, grams U^{235}
Uranium Enrichment Wt % U^{235}
10,000
9,000
8,000
7,000
6,000
5,000
4,000
3,000
2,000
1000
900
800
700
600
500
400
300
200
100
0
1
2
3
4
5
6
KEUFFEL & ESSER CO. MADE IN U.S.A.
2 CYCLES X 70 DIVISIONS

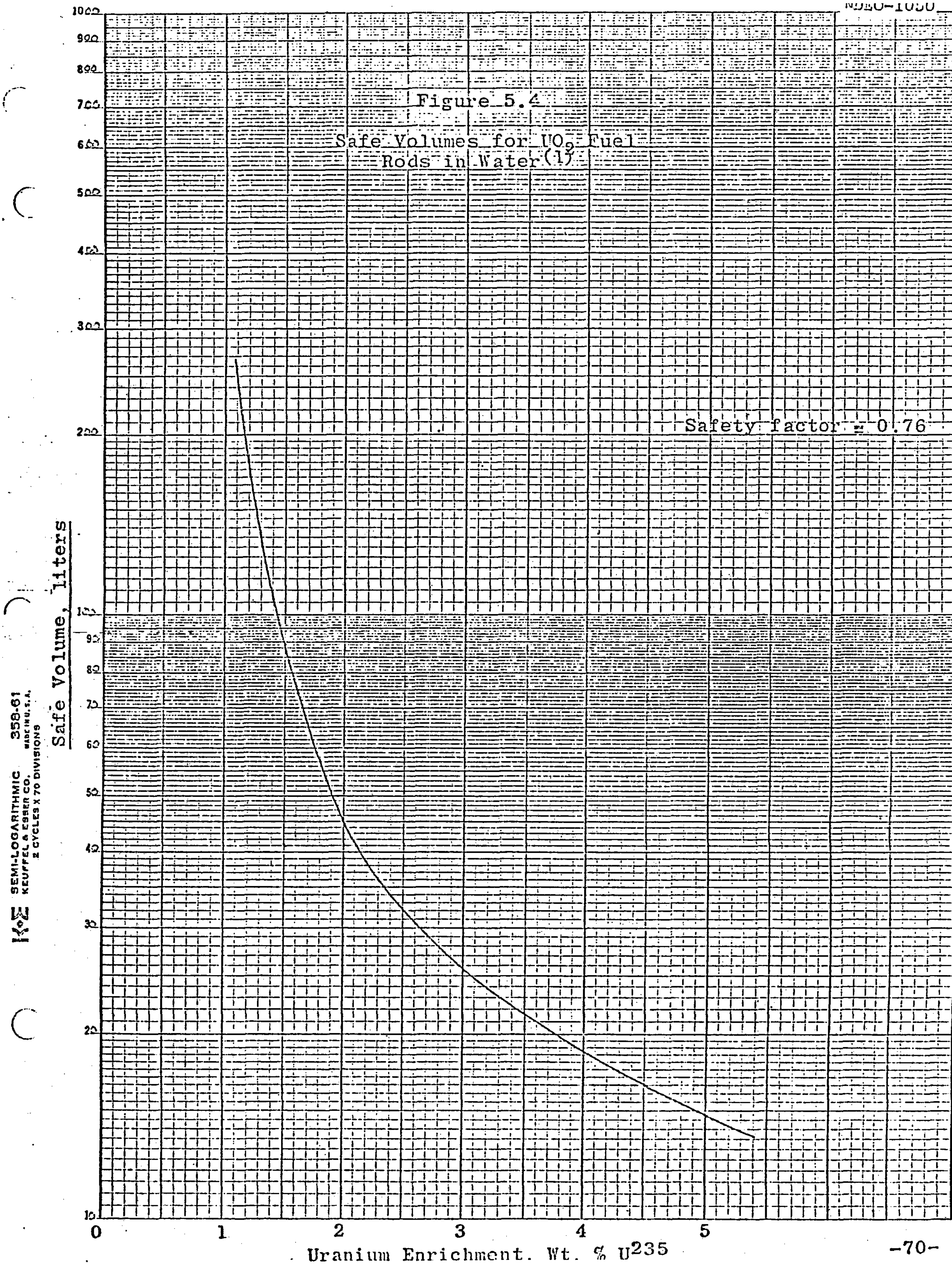


Figure 5.4
Safe Volumes for UO_2 Fuel Rods in Water[1]
Safety factor = 0.76
Safe Volume, liters
1000
900
800
700
600
500
400
300
200
100
90
80
70
60
50
40
30
20
10
0
1
2
3
4
5
Uranium Enrichment. Wt. % U^{235}
SEMI-LOGARITHMIC 358-61
KEUFFEL & ESSER CO.
2 CYCLES X 70 DIVISIONS

6.0 CALCULATED CRITICALITY LIMITS FOR THERMAL, U-ALUMINUM ALLOY-WATER CYLINDERS, SLABS AND SPHERES

The calculation method described in Section 2.1 was used to compute criticality limits for thermal U-Aluminum alloy water mixtures. A range of weight fractions in the alloy up to 0.50 gm U^{235} per gm alloy was considered. At any particular weight fraction of uranium, the quantities of primary interest in establishing the nuclear safety of various operations involving this fuel are the values of the critical parameters (e.g., critical mass, critical volume, critical cylinder diameter, etc.) under conditions of optimum moderation and full reflection by water. Under these conditions, the critical system dimensions reach their minimum values. If safe-system dimensions are selected to be less than these minimum critical dimensions, then the safety of the systems is established for all moderator-to-fuel ratios and degrees of reflection by water for the particular fuel alloy composition.

6.1 Assumptions

The following conservative assumptions were made in these calculations:

1. All systems were assumed to be homogeneous. Self-shielding was neglected.
2. Uranium enrichment was assumed to be 100% U^{235}.

6.2 Calculations and Results

For each of several values of weight % U^{235} in metal, critical infinite cylinder diameter, critical volume, etc., were computed as a function of moderator to fuel ratio for fully reflected systems. The minimum critical dimension, which corresponds to optimum moderation, was determined. Plots were made of minimum critical dimensions as a function of gm U^{235}/gm (U^{235} + Al), are shown as follows:

1. Minimum critical diameter and safe diameters for infinite cylinders (Fig. 6.1).
2. Minimum critical cross sectional areas and safe cross sectional areas for infinite cylinders (Fig. 6.2).
3. Minimum critical heights and safe heights for infinite slabs (Fig. 6.3).

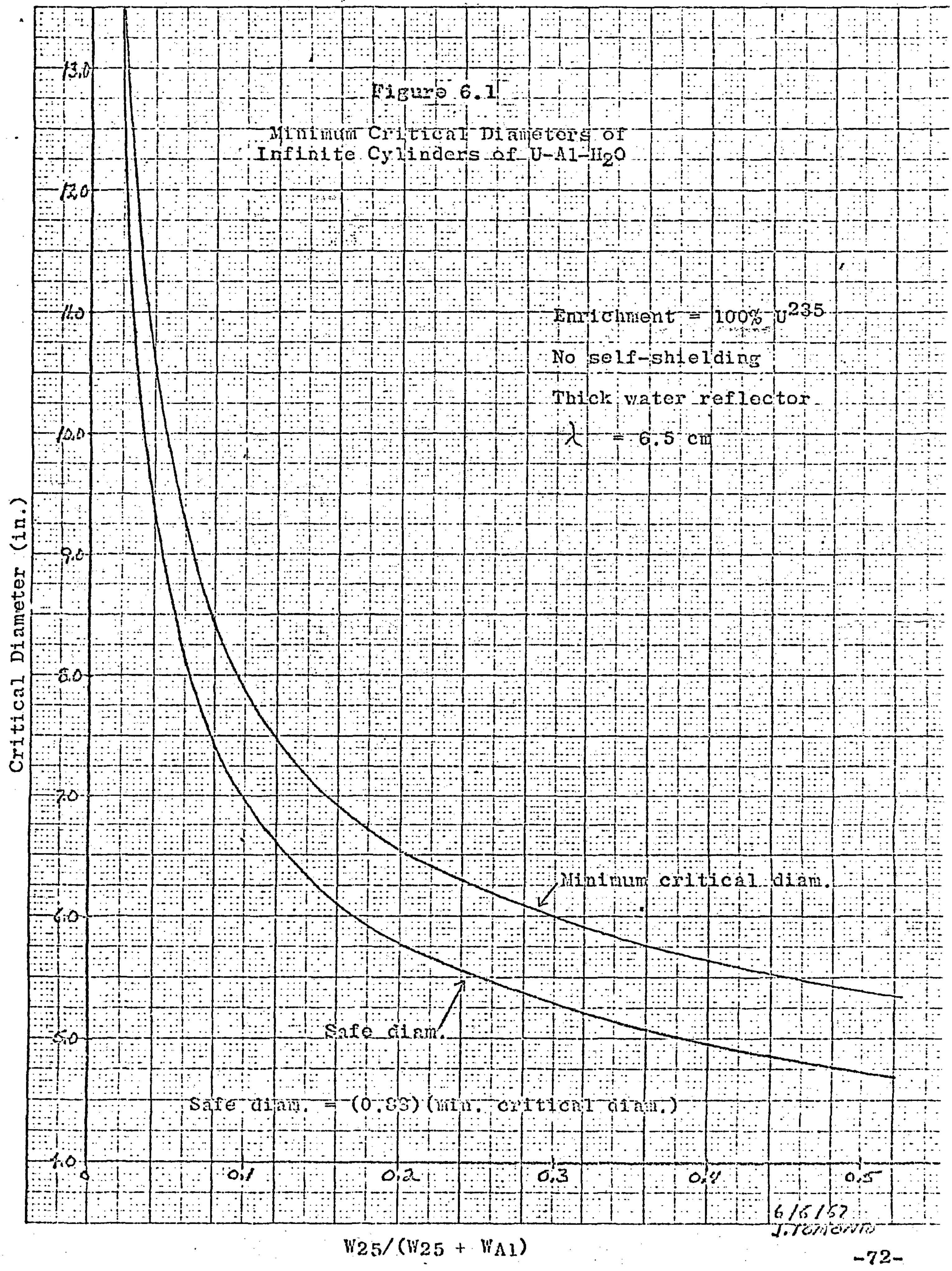


Figure 6.1
Minimum Critical Diameters of
Infinite Cylinders of U-Al-H_2O
Enrichment = 100% U^{235}
No self-shielding
Thick water reflector
λ = 6.5 cm
Minimum critical diam.
Safe diam.
Safe diam. = (0.83)(Min. critical diam.)
Critical Diameter (in.)
13.0
12.0
11.0
10.0
9.0
8.0
7.0
6.0
5.0
4.0
0
0.1
0.2
0.3
0.4
0.5
6/6/57
$W_{25}/(W_{25} + W_{Al})$

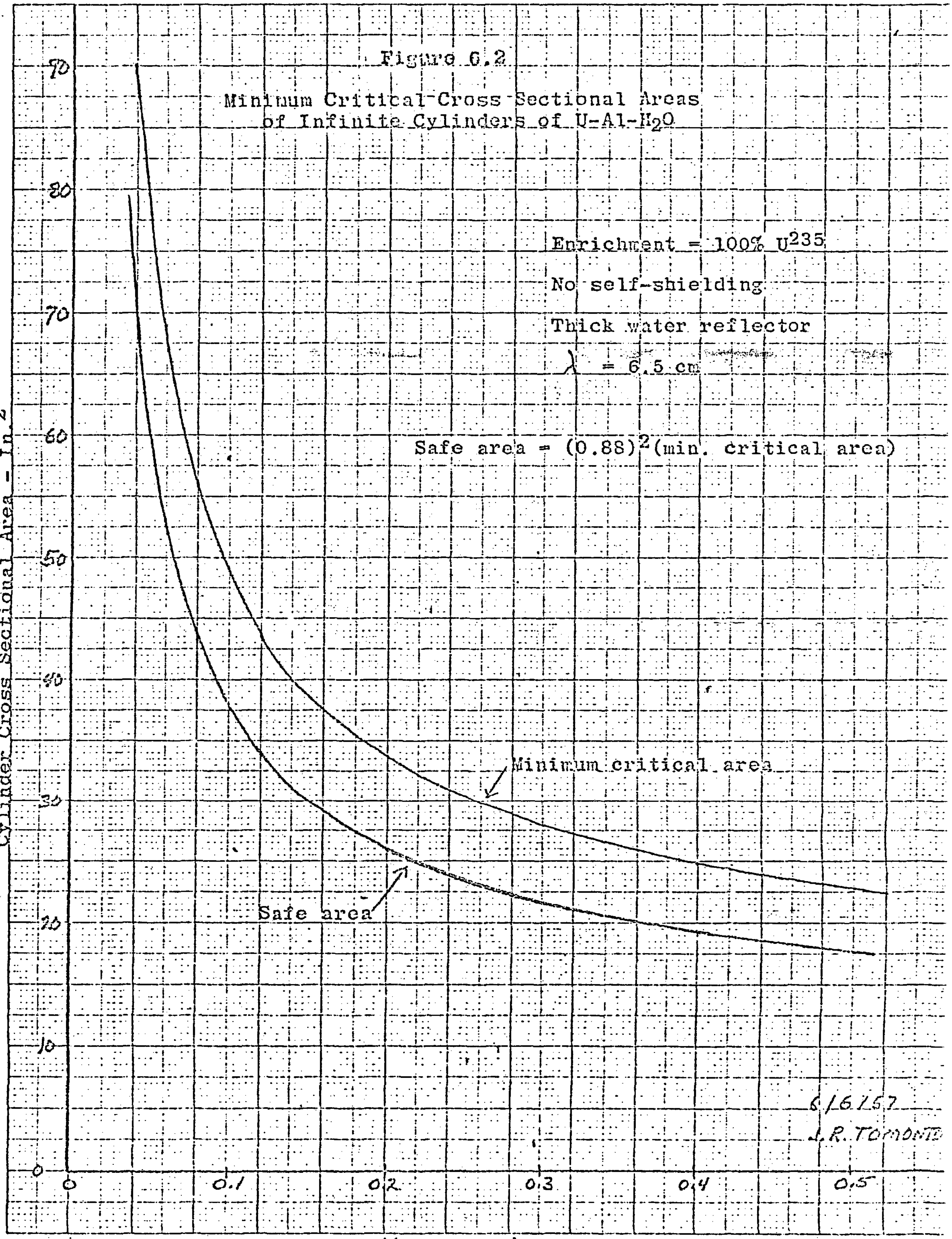


Figure 6.2
Minimum Critical Cross Sectional Areas
of Infinite Cylinders of U-Al-H_2O
Enrichment = 100% U^{235}
No self-shielding
Thick water reflector
λ = 6.5 cm
Safe area = $(0.88)^2$(min. critical area)
Minimum critical area
Safe area
Cylinder Cross Sectional Area - In.^{2}
$W_{25}/(W_{25} + W_{Al})$
0
0.1
0.2
0.3
0.4
0.5
10
20
30
40
50
60
70
80
90
6/6/57
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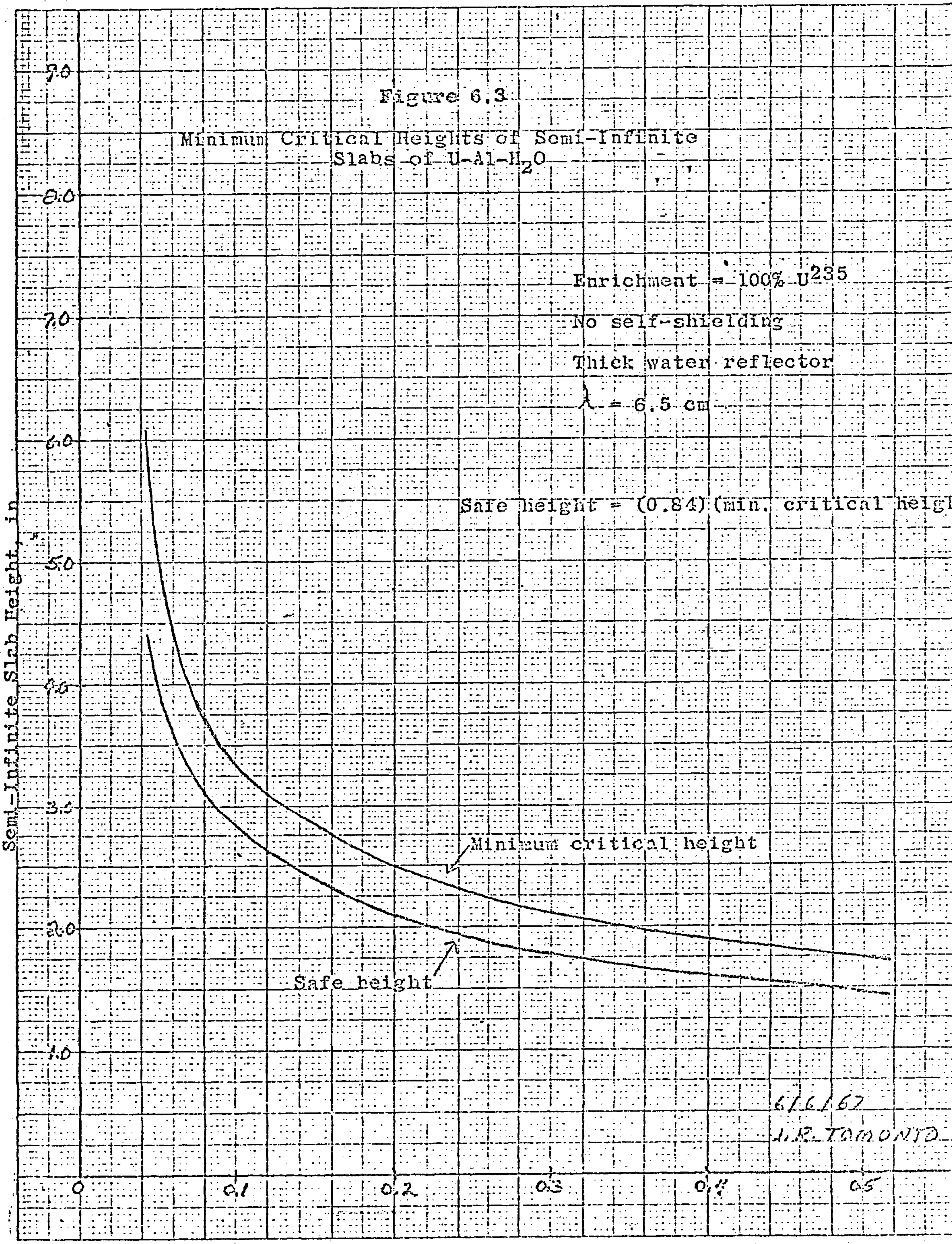


Figure 6.3

Minimum Critical Heights of Semi-Infinite Slabs of U-Al-H_2O

$W_{25}/(W_{25} + W_{Al})$

4. Minimum critical volumes and safe volumes (spherical geometry) (Fig. 6.4).

In order to establish the safety of arrays of units containing fuel, the interaction criteria of TID-7016, Rev. 1[25], are frequently used. In order to apply these criteria, the k_{eff} for each unit under unreflected conditions must be known. Maximum k_{eff} was computed for unreflected safe cylinders and spheres (safe under conditions of full reflection). These values correspond to optimum moderation for the bare condition, which in general occurs at a different water-to-fuel ratio than the optimum moderation for a similar reflected system. Maximum k_{eff} is shown as a function of gm U^{235}/gm (U^{235} + Al) in Figure 6.5.

Another parameter frequently of interest is the minimum critical linear density for infinite, reflected cylinders of the material at optimum moderation. This quantity is shown in Figure 6.6 as a function of w/o U^{235}. The minimum critical, linear density must be used with appropriate safety factors to provide criticality limits.

6.3 Safety Factors

The safety factors or relationships between minimum critical dimensions and safe dimensions used in this analysis are shown in Section 4.3. These safety factors are consistent with those used in TID-7016, Rev. 1[25].

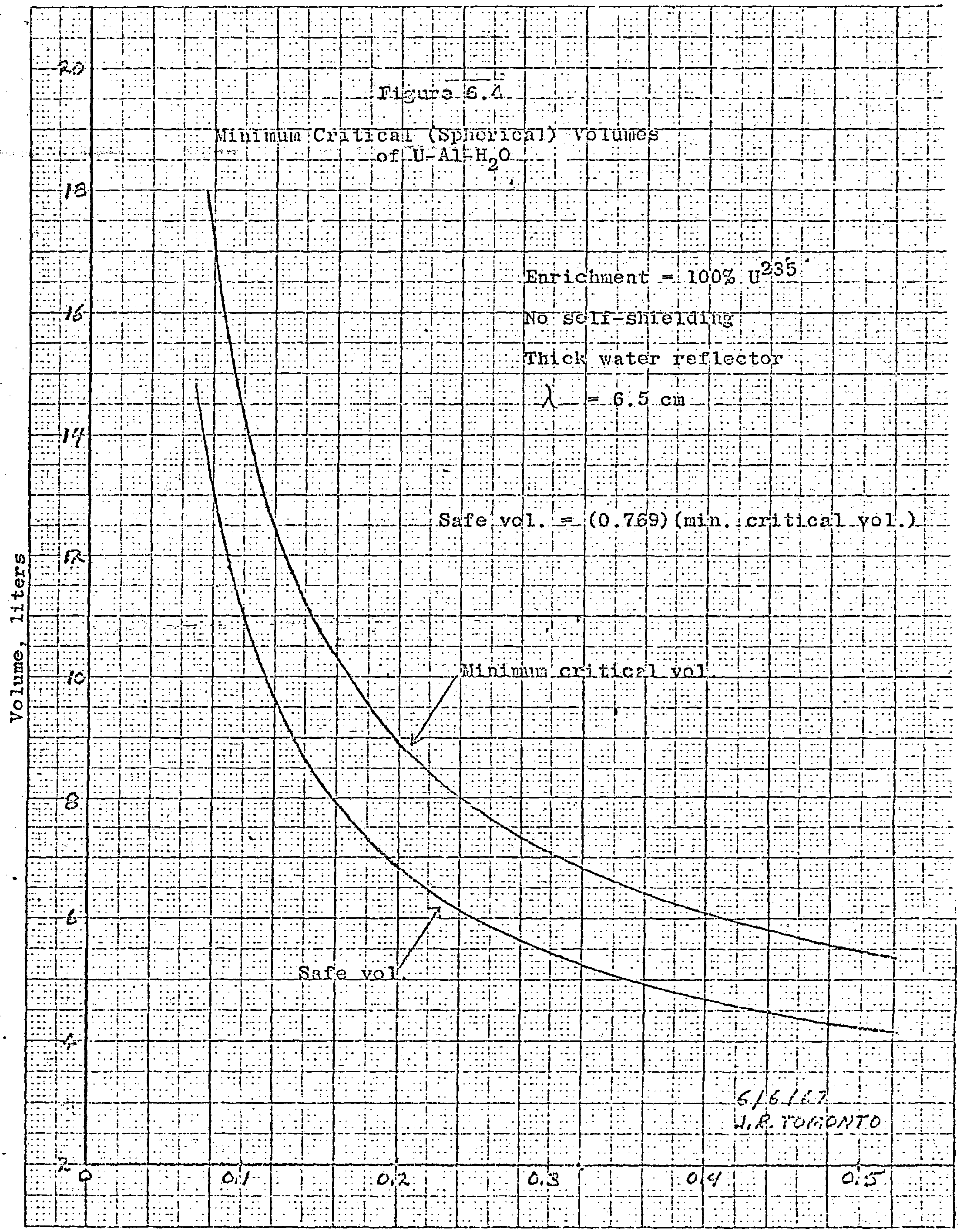


Figure 6.4
Minimum Critical (Spherical) Volumes
of U-Al-H_2O
Enrichment = 100% U^{235}
No self-shielding
Thick water reflector
λ = 6.5 cm
Safe vol. = (0.769)(min. critical vol.)
Minimum critical vol.
Safe vol.
Volume, liters
20
18
16
14
12
10
8
6
4
2
0
0.1
0.2
0.3
0.4
0.5
6/6/67
J.R. TOMONTO

$W_{25}/(W_{25} + W_{Al})$

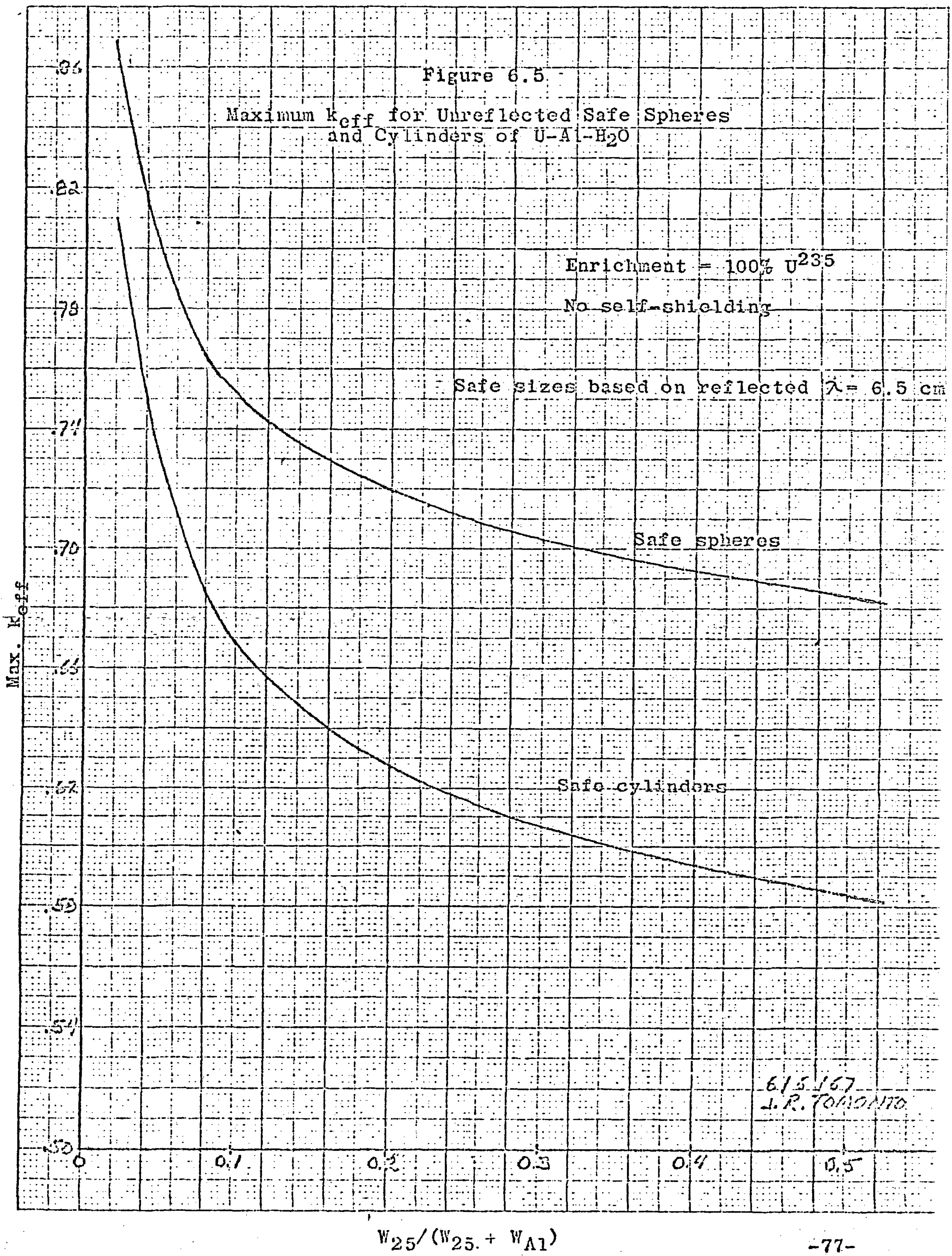
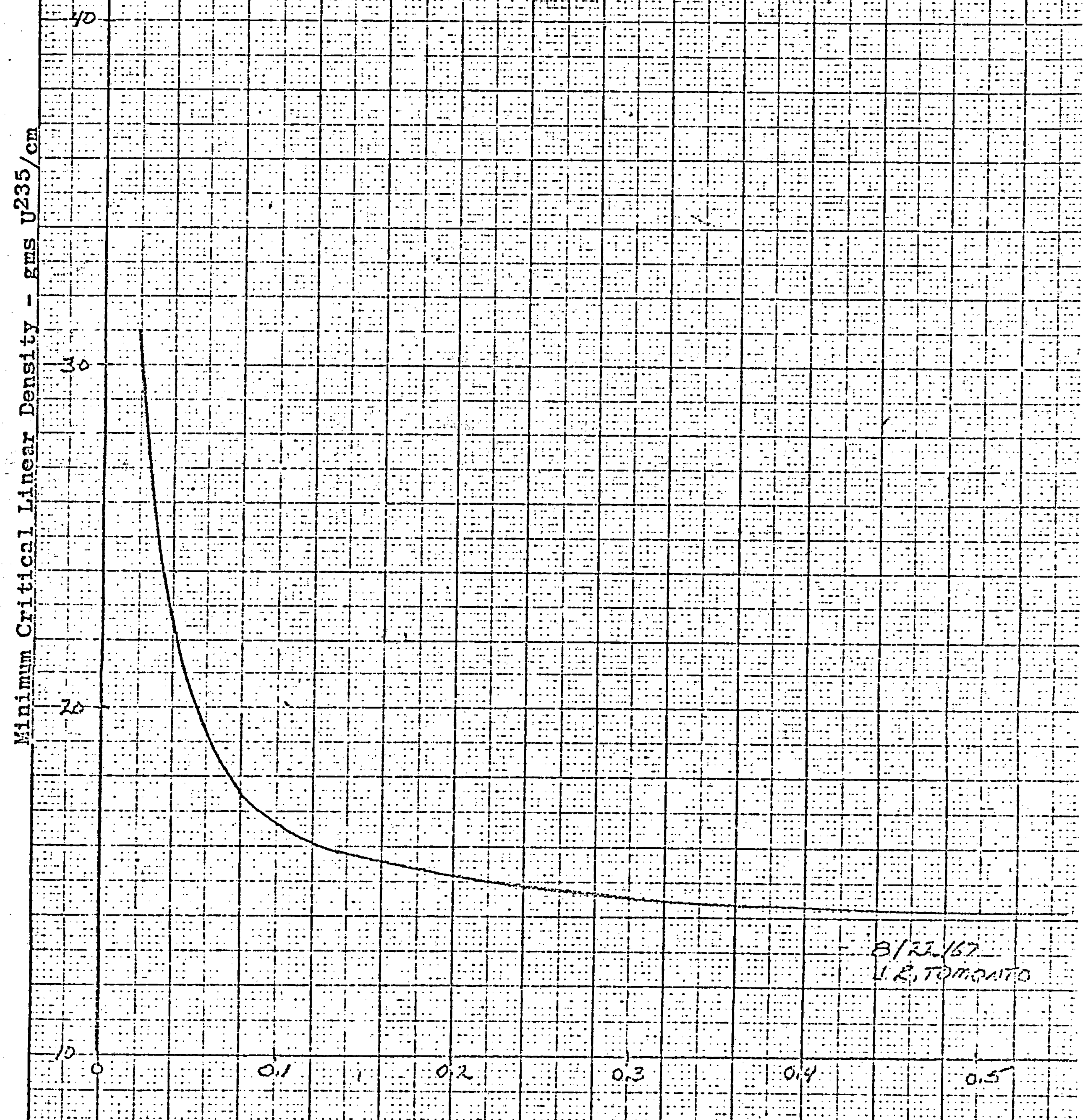


Figure 6.5
Maximum k_{eff} for Unreflected Safe Spheres and Cylinders of U-Al-H_2O
Enrichment = 100% U^{235}
No self-shielding
Safe sizes based on reflected λ = 6.5 cm
Safe spheres
Safe cylinders
Max. k_{eff}
.86
.82
.78
.74
.70
.66
.62
.58
.54
.50
0
0.1
0.2
0.3
0.4
0.5
6/5/67
J.R. Tomonto
$W_{25}/(W_{25} + W_{Al})$

Figure 6.6

Minimum Critical Linear Density for Reflected, Infinite Cylinders Containing U-Aluminum Alloy-Water Mixtures



$W_{25}/(W_{25}+W_{SS})$

7.0 CALCULATED CRITICALITY LIMITS FOR THERMAL, U-STAINLESS STEEL ALLOY-WATER CYLINDERS, SLABS AND SPHERES

The calculation method described in Section 2.1 was used to compute criticality limits for thermal U-SS alloy-water mixtures. A range of weight fractions in the alloy up to 0.50 gm U^{235} per gm alloy was considered. At any particular weight fraction of uranium, the quantities of primary interest in establishing the nuclear safety of various operations involving this fuel are the values of the critical parameters (e.g., critical mass, critical volume, critical cylinder diameter, etc.) under conditions of optimum moderation and full reflection by water. Under these conditions, the critical system dimensions reach their minimum values. If safe-system dimensions are selected to be less than these minimum critical dimensions, then the safety of the systems is established for all moderator-to-fuel ratios and degrees of reflection by water for the particular fuel alloy composition.

7.1 Assumptions

The following conservative assumptions were made in these calculations:

1. All systems were assumed to be homogeneous. Self-shielding was neglected.

2. Uranium enrichment was assumed to be 100% U^{235}.

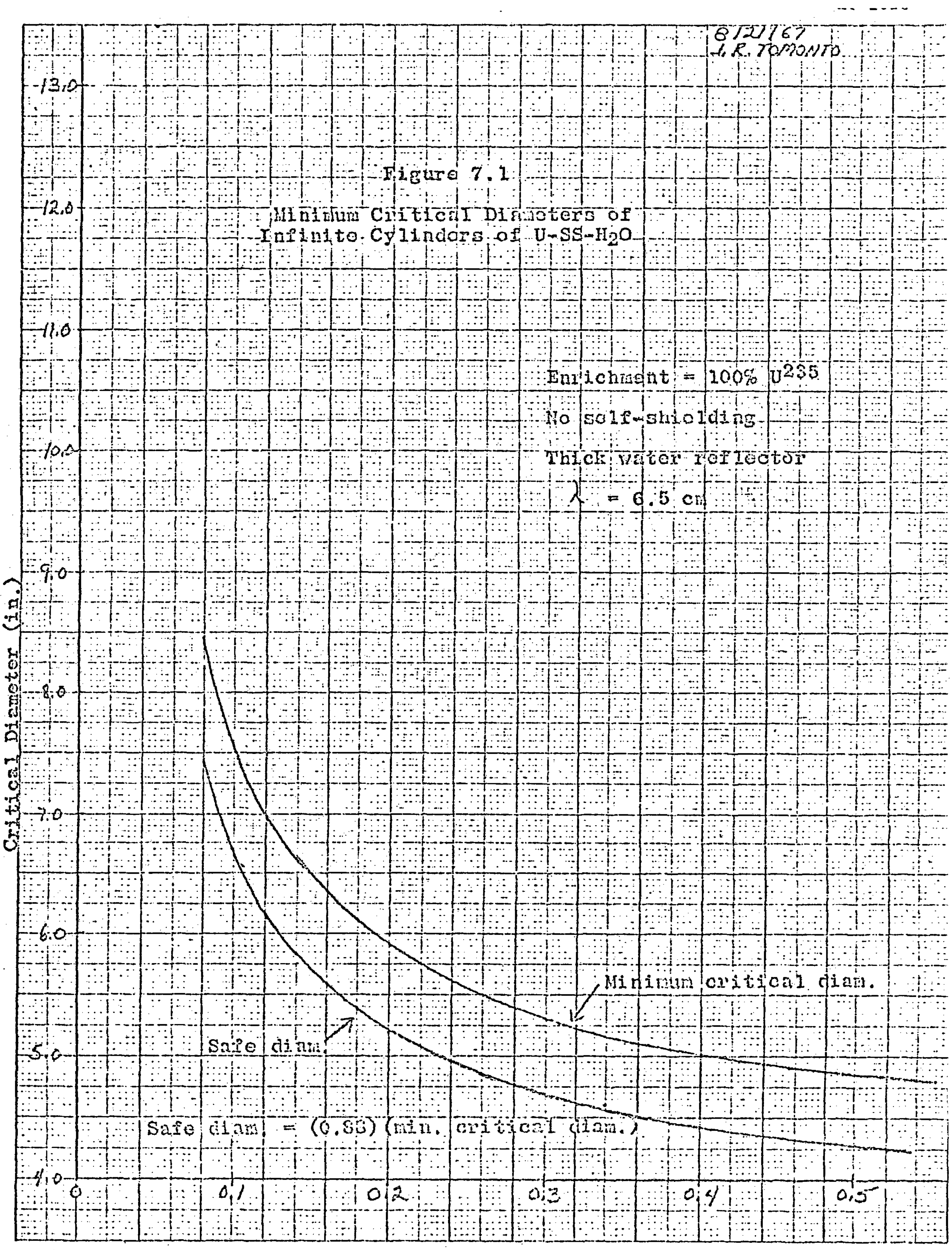
7.2 Calculations and Results

For each of several values of weight % U^{235} in metal, critical infinite cylinder diameter, critical volume, etc., were computed as a function of moderator to fuel ratio for fully reflected systems. The minimum critical dimension, which corresponds to optimum moderation, was determined. Plots were made of minimum critical dimensions as a function of gm U^{235}/gm (U^{235} + SS), are shown as follows:

1. Minimum critical diameter and safe diameters for infinite cylinders (Fig. 7.1).

2. Minimum critical cross sectional areas and safe cross sectional areas for infinite cylinders (Fig. 7.2).

3. Minimum critical heights and safe heights for infinite slabs (Fig. 7.3).



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J. R. TOMONTO
Figure 7.1
Minimum Critical Diameters of
Infinite Cylinders of U-SS-H2O
Enrichment = 100% U235
No self-shielding
Thick water reflector
λ = 6.5 cm
Critical Diameter (in.)
13.0
12.0
11.0
10.0
9.0
8.0
7.0
6.0
5.0
4.0
Minimum critical diam.
Safe diam.
Safe diam = (0.88)(min. critical diam.)
0
0.1
0.2
0.3
0.4
0.5

$W_{25}/(W_{25} + W_{SS})$

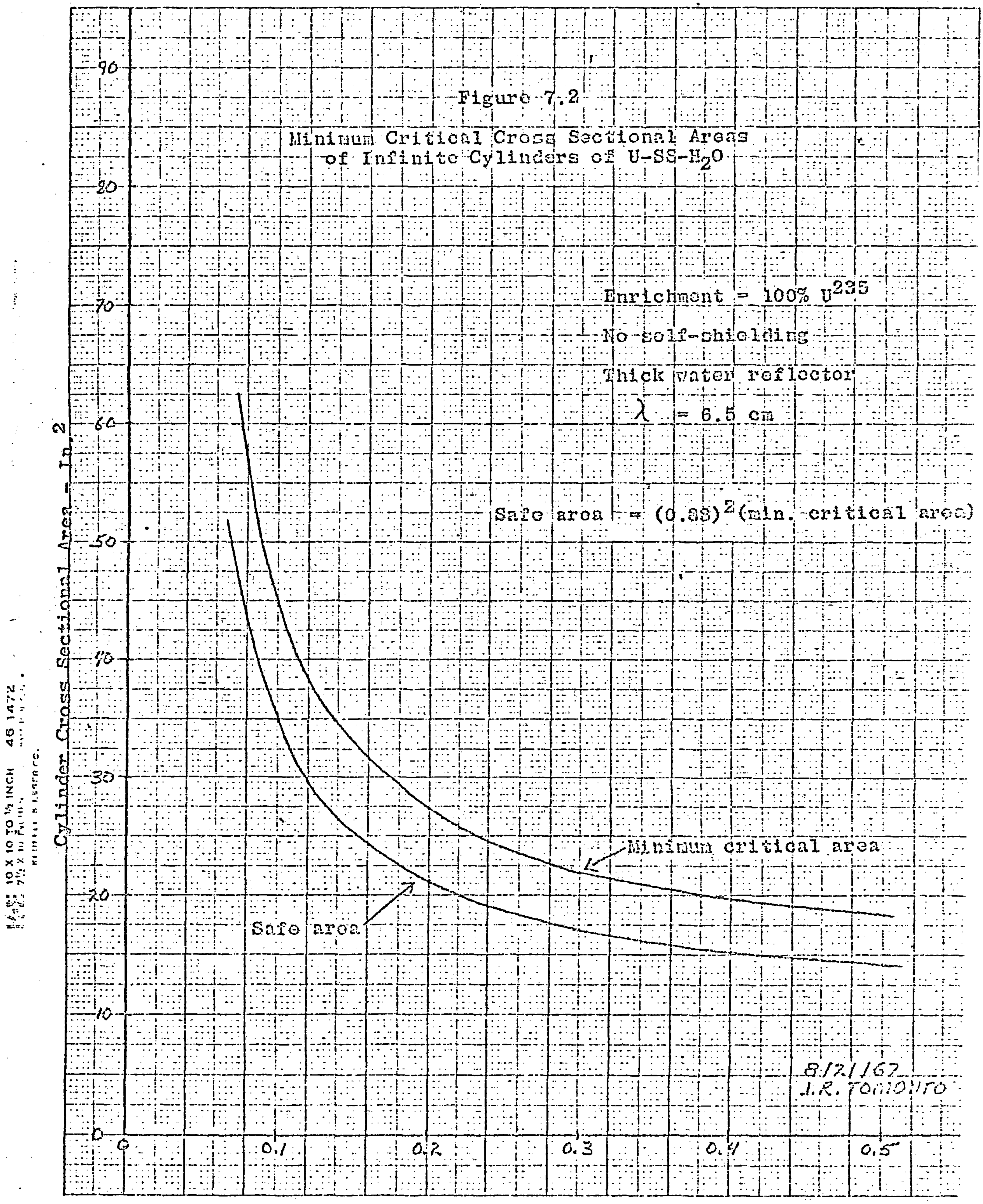
Figure 7.2
Minimum Critical Cross Sectional Areas of Infinite Cylinders of U-SS-H_2O
Enrichment = 100% U^{235}
No self-shielding
Thick water reflector
λ = 6.5 cm
Safe area = $(0.88)^2$(min. critical area)
Minimum critical area
Safe area
Cylinder Cross Sectional Area - In^2
90
80
70
60
50
40
30
20
10
0
0
0.1
0.2
0.3
0.4
0.5
8/21/67
J.R. TOMONTO
10 X 10 TO ½ INCH 46 1472

$W_{25}/(W_{25} + W_{SS})$

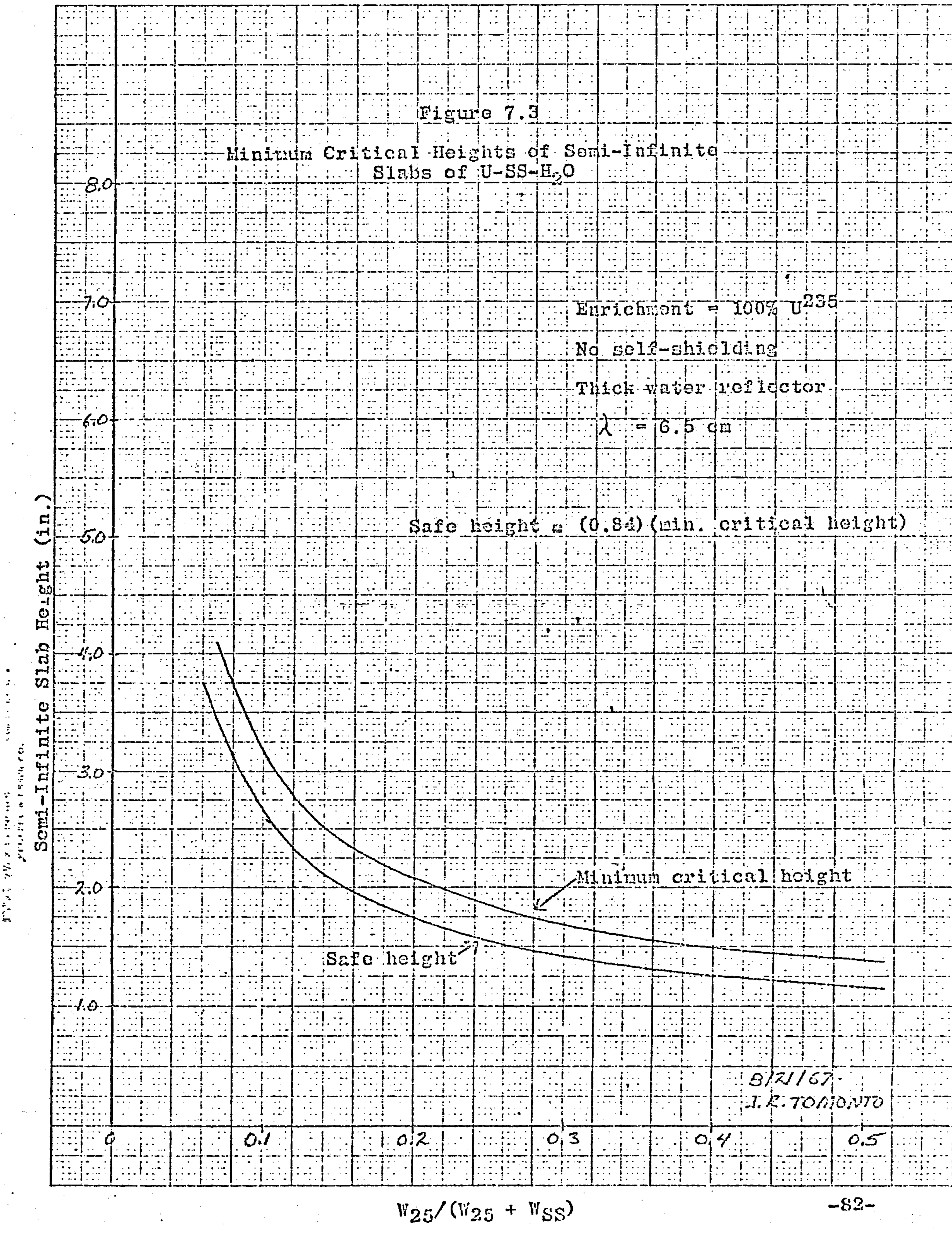

Figure 7.3
Minimum Critical Heights of Semi-Infinite Slabs of U-SS-H_2O
Enrichment = 100% U^{235}
No self-shielding
Thick water reflector
λ = 6.5 cm
Safe height = (0.84)(min. critical height)
Minimum critical height
Safe height
8/21/67
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8.0
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0
0.1
0.2
0.3
0.4
0.5
Semi-Infinite Slab Height (in.)
$W_{25}/(W_{25} + W_{SS})$

4. Minimum critical volumes and safe volumes (spherical geometry) (Fig. 7.4).

In order to establish the safety of arrays of units containing fuel, the interaction criteria of TID-7016, Rev. 1[25], are frequently used. In order to apply these criteria, the k_{eff} for each unit under unreflected conditions must be known. Maximum k_{eff} was computed for unreflected safe cylinders and spheres (safe under conditions of full reflection). These values correspond to optimum moderation for the bare condition, which in general occurs at a different water-to-fuel ratio than the optimum moderation for a similar reflected system. Maximum k_{eff} is shown as a function of gm U^{235}/gm (U^{235} + SS) in Figure 7.5.

Another parameter frequently of interest is the minimum critical linear density for infinite, reflected cylinders of the material at optimum moderation. This quantity is shown in Figure 7.6 as a function of w/o U^{235}. The minimum critical, linear density must be used with appropriate safety factors to provide criticality limits.

7.3 Safety Factors

The safety factors or relationships between minimum critical dimensions and safe dimensions used in this analysis are shown in Section 4.3. These safety factors are consistent with those used in TID-7016, Rev. 1[25].

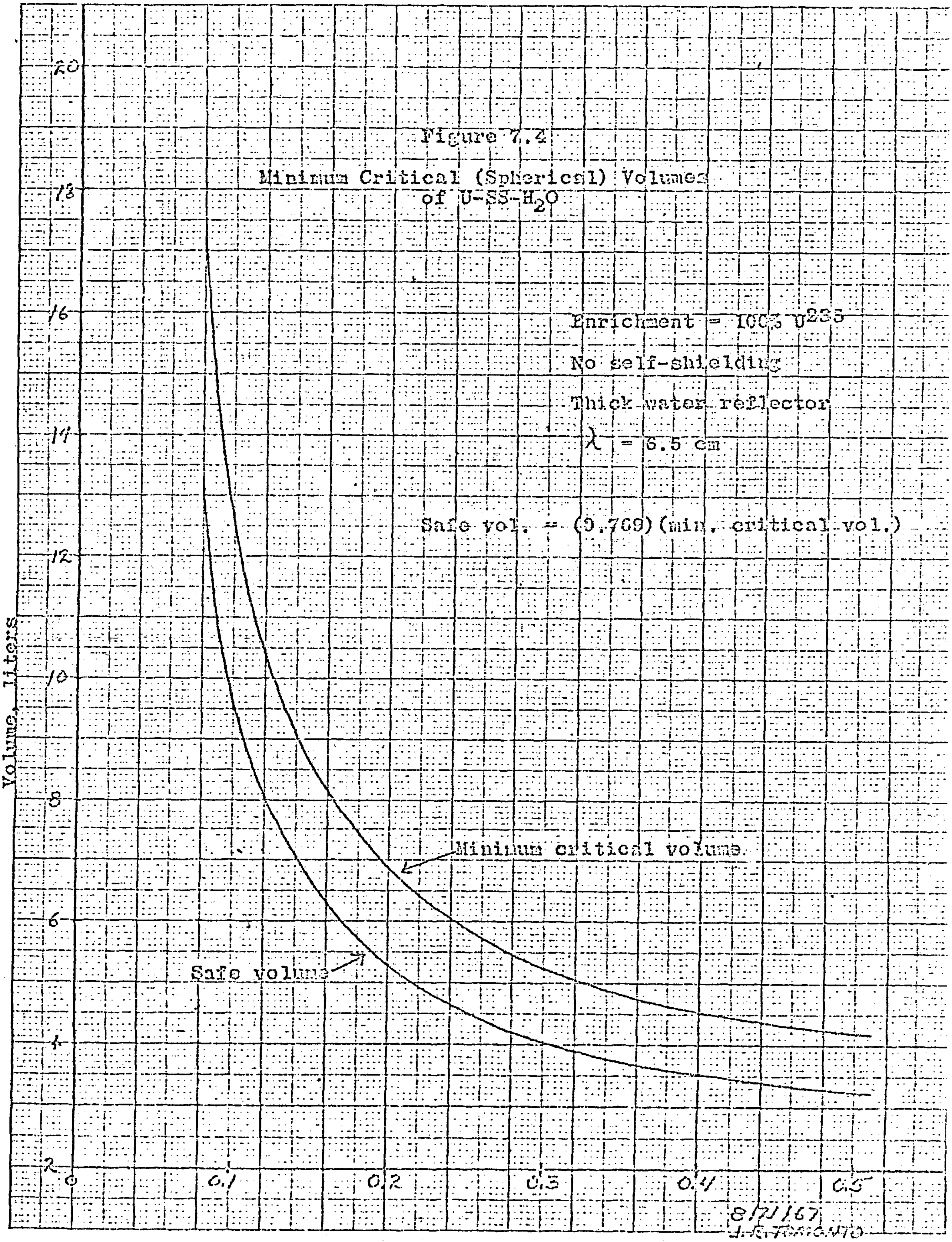

Figure 7.4
Minimum Critical (Spherical) Volumes of U-SS-H_2O
Enrichment = 100% U^{235}
No self-shielding
Thick water reflector
λ = 6.5 cm
Safe vol. = (0.769)(min. critical vol.)
Minimum critical volume
Safe volume
Volume, liters
20
18
16
14
12
10
8
6
4
2
0
0.1
0.2
0.3
0.4
0.5
8/21/67

$W_{25}/(W_{25} + W_{SS})$

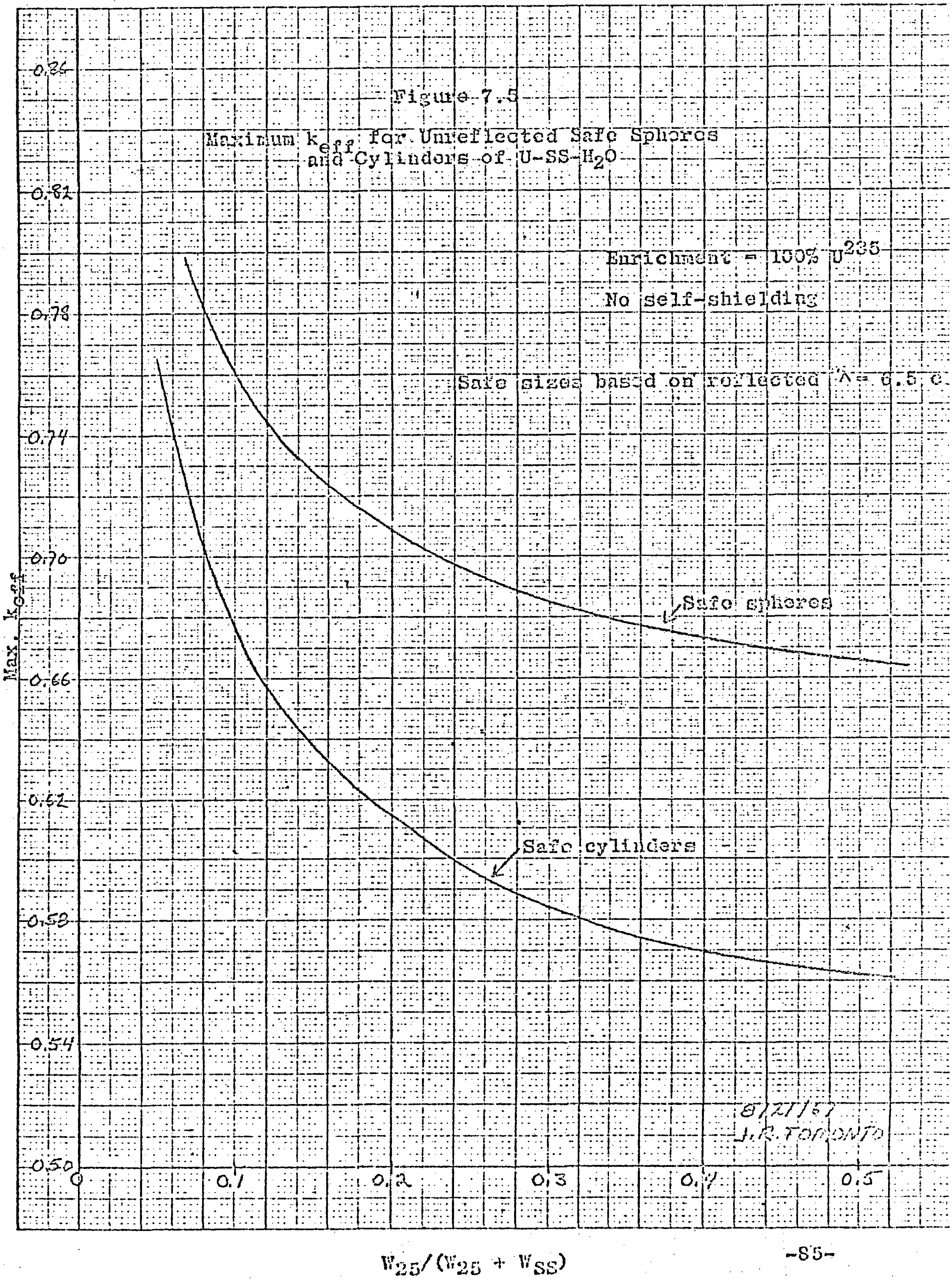
Figure 7.5
Maximum k_{eff} for Unreflected Safe Spheres and Cylinders of U-SS-H_2O
Enrichment = 100% U^{235}
No self-shielding
Safe sizes based on reflected λ = 6.5 c
Safe spheres
Safe cylinders
Max. k_{eff}
0.86
0.82
0.78
0.74
0.70
0.66
0.62
0.58
0.54
0.50
0
0.1
0.2
0.3
0.4
0.5
8/27/57
J.R. TORONTO
$W_{25}/(W_{25} + W_{SS})$

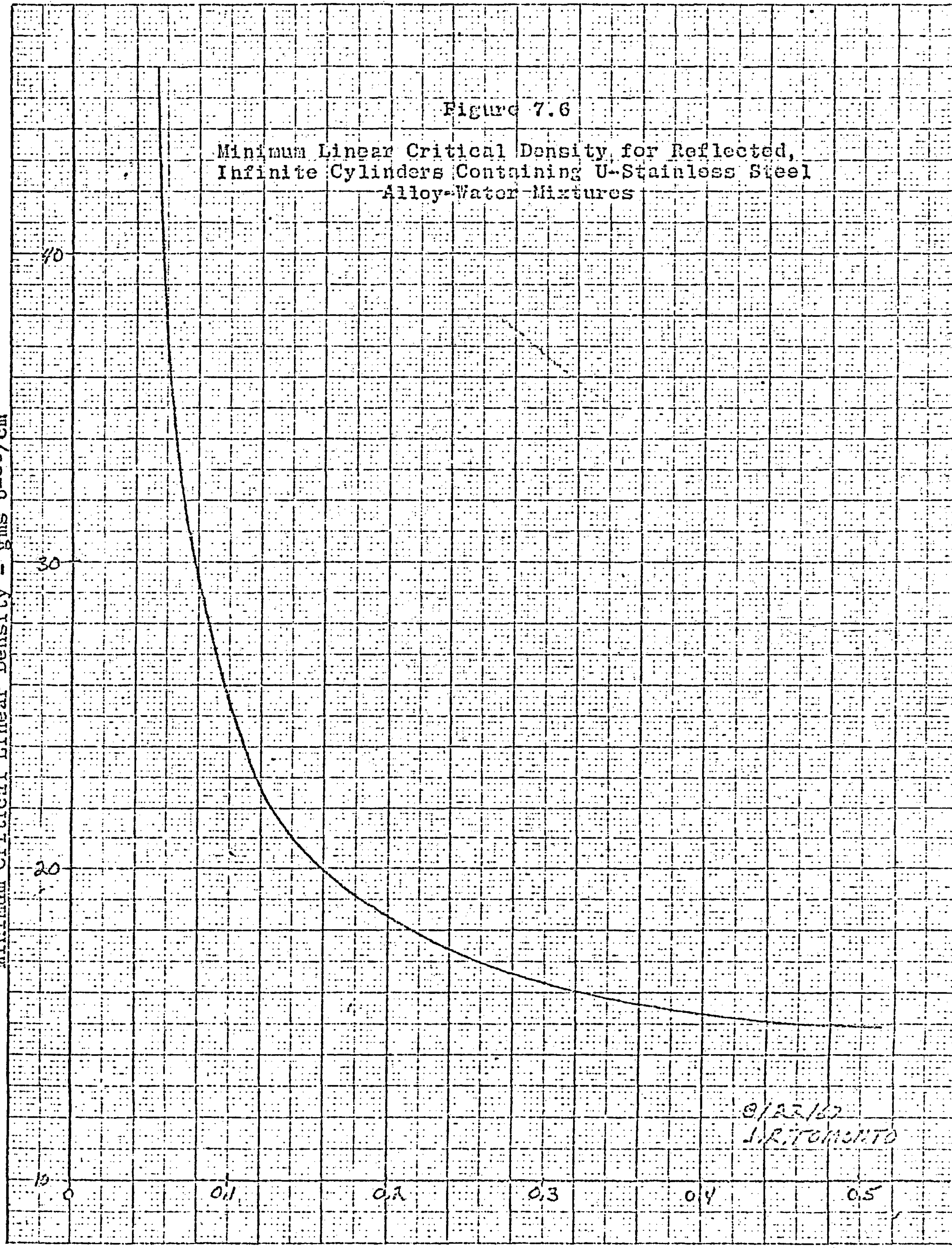
Figure 7.6
Minimum Linear Critical Density for Reflected, Infinite Cylinders Containing U-Stainless Steel Alloy-Water Mixtures
Minimum Critical Linear Density - gms U235/cm
40
30
20
10
0
0.1
0.2
0.3
0.4
0.5
J.R.TOMONTO

$W_{25}/(W_{25} + W_{SS})$

8.0 REFERENCES

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Site and Location

The Fabrication Operation is part of the "H" Tract industrial area of approximately 2,400 x 1,380 feet which is shared with Olin-Mathieson Corporation and United Nuclear Corporation. The area is shown on Figure B.1. This tract is situated in the area generally bounded by Gibbs Street on the north, the Penn Central Railroad (e.g. New Haven Railroad) on the east, Shelton Avenue on the west, and Munson Street over one block south. These facilities generally located in the north central portion of the city of New Haven, which is in the southern part of the State of Connecticut. New Haven, Connecticut, is located on Long Island Sound, 17 miles east of the city of Bridgeport and 34 miles south of Hartford. It is 79 miles northeast of New York and 140 miles southwest of Boston.(1)

Land Use

New Haven is a primary industrial city and rather extensively developed with over 400 manufacturing plants ranging from one-man operations to plants employing several thousand people. The products manufactured range from arms and ammunition to wire and wire goods with such outstanding products as paperboard and boxes, rubber goods, builders hardware, aircraft parts, toys, machinery and machine tools, clothing boilers and razor blades.(1)

Population

Population figures have been obtained from the New Haven Standard Metropolitan Statistical Area (SMSA), New Haven proper and the immediate plant area. These figures are shown on Figure B.2.

Meteorology

New Haven's climate is typical of coastal areas of southern New England. It is vigorous without being overly severe. New Haven is located at the widest part of Long Island Sound and the tempering effect of the water is most pronounced in this vicinity. During the summer season, the sea breeze holds temperatures 5 to 15 degrees lower in the afternoon and during the winter season minimum temperatures are usually 5 to 10 degrees higher then those reported from northern sections of the City.

Precipitation is quite evenly distributed throughout the year and is adequate for agricultural pursuits. The elevation of the land increases northward and results in somewhat higher amounts of precipitation in the northern suburbs as well as a few more thunderstorms each year. During the winter months, a variety of precipitation is found in most storms. It is common to have rain along the shore, freezing rain and sleet a short distance inland and snow in the northern parts of the City.

(1) New Haven Market Report, 1970, Southern New England Telephone Company, April, 1970

Heavy snow is rather uncommon in the immediate coastal area and usually melts in a few days. Further inland, the snow becomes progressively heavier and a layer of snow covers the ground most of the winter.

Prevailing wind direction varies with the seasons. From late spring, until fall, winds are predominantly south to southwest due to the effect of the sea breeze. During the winter months, the prevailing winds are northerly. Strong southeast winds cause unusually high tides and some local flooding in low-lying coastal areas two or three times a year. [(1)] Seasonally averaged data is summarized below:

Season	Prevailing Wind Direction	Mean Speed (miles/hr)	Mean Temperature (^oF)	Mean Precipitation (inch.)
Winter	NW	8.5	32.0	3.92
Spring	NNW	7.5	56.8	3.71
Summer	WNW	6.6	68.8	3.87
Fall	NW	7.9	43.3	3.84

Hydrology

Liquid effluents are discharged to the City sewage system. Since these effluents are not discharged to a stream or river, mean seasonal flow rates are not included in this evaluation.

Geology

New Haven lies in a belt of country underlain by peculiar rocks that were formed in the Triassicperiod of the earth's history. This Triassic area of southern New England extends from the city of New Haven on Long Island Sound for about 100 miles to the northern part of Massachusetts. Its width varies greatly but averages about 16 miles; its area some 1600 square miles. The area is characterized by its unity of character throughout its entire extent.

Topographically, the area is characterized by many hills and ridges, usually several hundred feet high. The dark colored heavy rock in these ridges is igneous in origin, and is often called trap. The red sandstone, the prevailing sedimentary rock of the region, does not ordinarily form prominent hills, except as protected by the harder trap. Although locally it is firm rock, the sandstone is usually too weak to have withstood the age-long attacks of weathering and erosion. Hence, the hills of sandstone are low, rounded, and generally covered with glacial debris left by the ice sheet which once burried New England. Soft shales or harder sandstone ledges are burried beneath those hills. The soil is commonly of a striking red color from disintegration of the sandstone and shale. The sandstone and trap which underlie most of the region are conservatively estimated to be at

[(1)] New Haven, Conn, Local Climatelogical Data, U.S. Department of Commerce, 1968.

least 150 million years old. All the sedimentary layers contain an abundance of mica plus feldspar and quartz. Some pebbles and sand grains are rounded and smooth and some are angular and rough. Toward the west, several miles, metamorphic rock is encountered with many schists which represents the deep core of an old mountain range which has been eroded until only the roots remain.

In conclusion, near New Haven, the Triassic sediments are made up of waste derived by erosion of older rocks in adjoining areas. The belts of crumpled metamorphic schists are remnants of mountains under which great bodies of molten rock formed and cooled. These mountaihs have been completely eroded and their granite cones are exposed to view in a country of low relief. Probably, the Triassic sediments in Connecticut and Massachusetts represent only a small fraction of the enormous volume of rock waste resulting from the prolonged denudation. The rest was distributed widely and is now consealed beneath the sea.[1]

Seismology

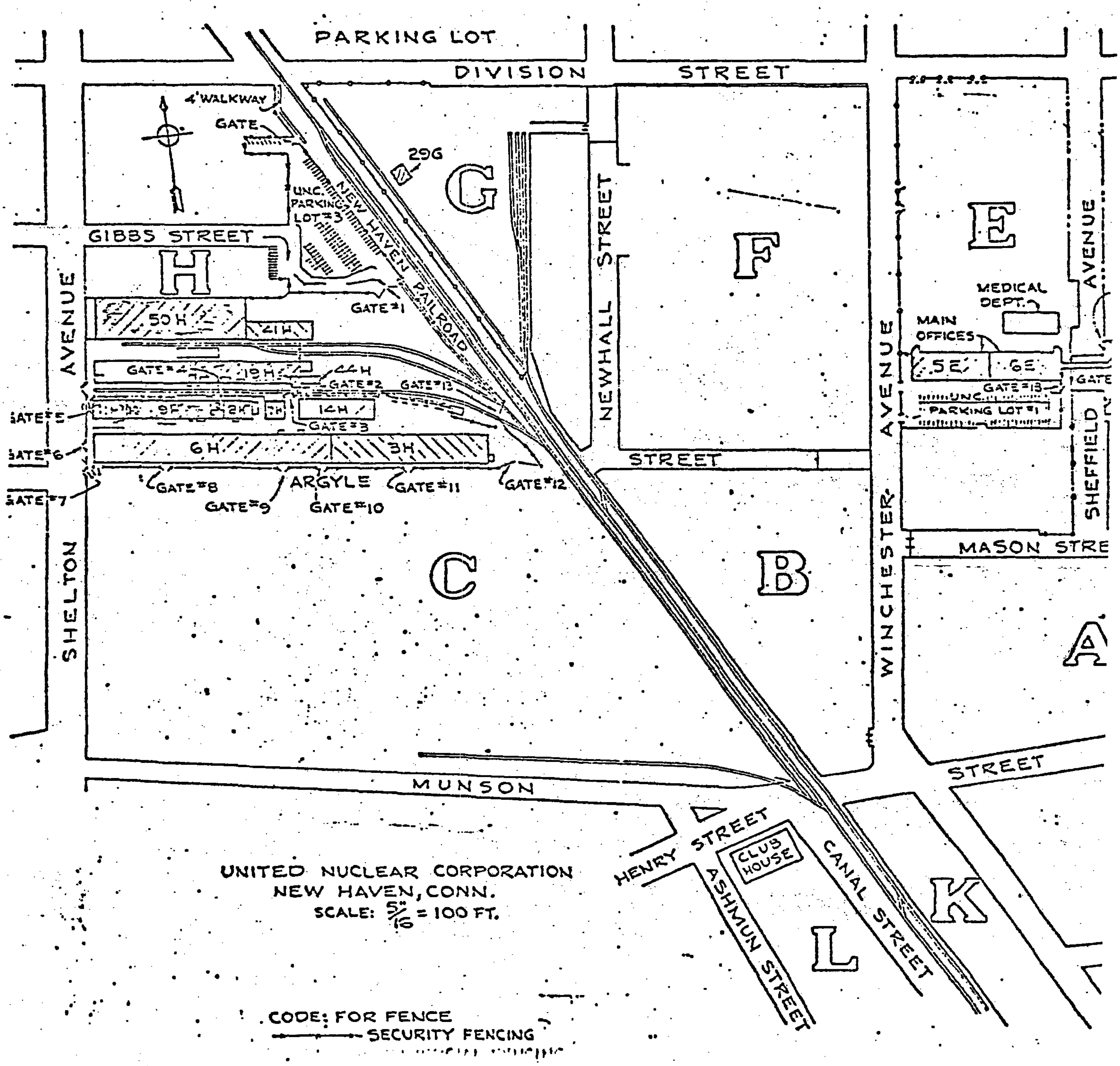
The New England area is relatively inactive seismically. The largest earthquake to occur in the northeastern region of the United States and adjacent Canada was of Epicentral Intensity X (M.M.) and took place on February 5, 1663, in the St. Lawrence River area. The largest earthquake in New England was of Epicentral Intensity VIII (M.M.) and took place on November 18, 1755, east of Cape Ann, Massachusetts, approximately 200 miles northeast of New Haven. The closest earthquakes to New Haven which have been strong enough to cause some minor damage (e.g., M.M. Intensity VI or greater) have occurred in the vicinity of Haddam, Connecticut, approximately 50 miles northeast of New Haven. Two of these earthquakes occurred in 1791 and one in 1925.[2]

(1) Longwell, C. R. and Dana, E. S., Walks and Rides in Central Connecticut and Massachusetts, the Tuttle, Morehouse and Taylor Co., 1932

(2) Eppley, R. A., Earthquake History of the United States, Part I, No. 41-1 Revised Edition, U.S. Department of Commerce, 1965

GULF UNITED
NUCLEAR FUELS CORPORATION

FIGURE B.1



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...ct:	Appendix B - Fabrication Operation Site Evaluation						
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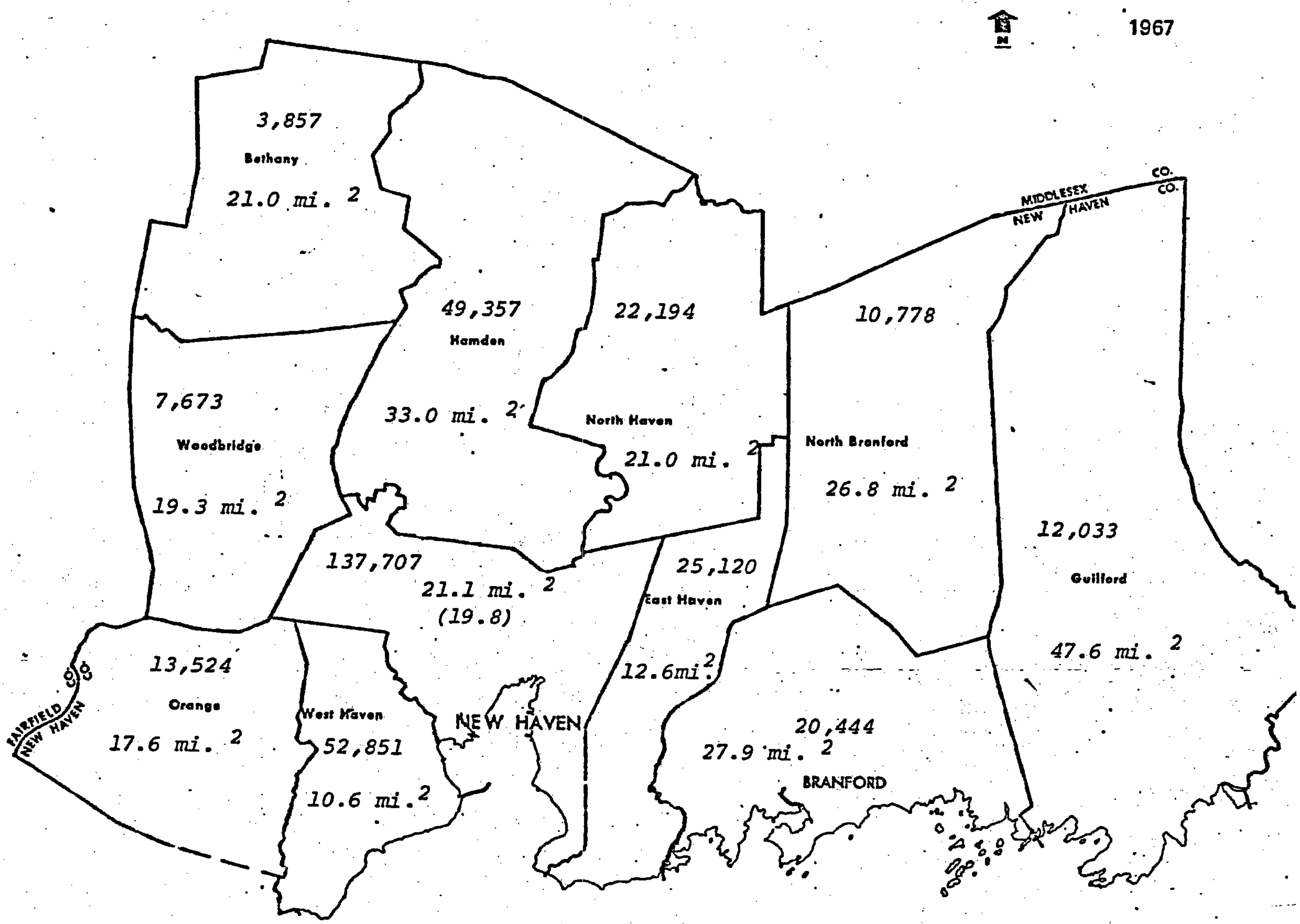
GULF UNITED
NUCLEAR FUELS CORPORATION

FIGURE B.2



New Haven Proper:

Area: 19.8 sq. mi.
Pop: 137,707
Pop Dens: 6,858

Immediate Plant Area:

Area: 1.06 sq. mi.
Pop: 13,863
Pop Dens: 13,080

NH SMSA:

Area: 431 sq. mi.
Pop: 355,538
Pop Dens: 825 persons/sq. mi

NEW HAVEN, CONN., STANDARD METROPOLITAN STATISTICAL AREA

CONNECTICUT STATE REGISTER AND MANUAL 1970.